SAFETY EVALUATION REPORT
FOR
DECOMMISSIONING OF FORT ST. VRAIN

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1 SUMMARY OF PLAN

1.1 Description of Decommissioning Plan and Decommissioning Alternative

1.1.1 Introduction

On December 5, 1988, the Public Service Company of Colorado (PSC) notified the Nuclear Regulatory Commission (NRC) that it had elected to terminate Fort St. Vrain operations early because of economic considerations associated with the ongoing operating costs at the plant.

PSC submitted its proposed decommissioning plan (PDP) in accordance with Title 10 of the Code of Federal Regulations (10 CFR) 50.82(a), which requires the PDP be submitted "within two years following permanent cessation of operations." The PDP was submitted on November 5, 1990, with the DECON option as the selected decommissioning alternative. The NRC staff submitted several requests for additional information (RAIs) to PSC. In response to the staff's RAIs, PSC revised the PDP on December 17 and 21, 1990; January 14, April 15 and 26, May 15, June 6 and 17, July 1, August 28 and 30, November 15, and December 6, 1991; and January 9, March 19, and April 17, 1992. PSC submitted an Environmental Report Supplement on July 10, 1991, and revisions on March 20 and April 30, 1992. Subsequent PSC submittals have addressed all outstanding NRC RAIs.

The NRC staff evaluated the adequacy of the licensee's proposal on the basis of applicable NRC regulations and regulatory guidance and in accordance with applicable sections of the Standard Review Plan (SRP, NUREG-0800). The results of its evaluation are provided below.

1.1.2 Background

Fort St. Vrain (FSV) was shut down on August 18, 1989. On August 29, 1989, the PSC Board of Directors confirmed the decision that FSV would not be restarted and that PSC would pursue the decommissioning.

PSC identified problems with the control rod drive assemblies and the steam generator steam ring headers that presented significant technical obstacles that could be overcome, but at significant cost and time to PSC. Additionally, the uniqueness of the one-of-a-kind high-temperature gas-cooled reactor fuel cycle made the cost to purchase new fuel prohibitive. This, in conjunction with low plant availability and correspondingly high operating costs, was the basis for the PSC's decision to discontinue operation of FSV.

1.1.3 Proposed Action

The PSC selected the DECON option as the decommissioning alternative and intends to decontaminate and dismantle the prestressed concrete reactor vessel (PCRV) and supporting systems to the extent necessary to ensure removal of radioactive materials and to allow release of the facility and site for unrestricted use.

The contamination and activation levels are low at FSV because the plant had a relatively short operating history of approximately 447 full-power days since 1979 when commercial operation was initiated. The licensee elected the DECON alternative (1) to allow maximum flexibility in use of the site and facility; (2) to decommission the facility without significant radiation exposure; (3) to eliminate the need for long-term monitoring, surveillance, and maintenance; (4) to avoid any significant effects to the environment; and (5) to support the agreement between the Colorado Public Utilities Commission and PSC regarding funding for the DECON alternative.

The proposed decommissioning is necessary to terminate the FSV license in accordance with the requirements of 10 CFR 50.82. Dismantlement and decontamination of the plant systems and the PCRV to conditions suitable for unrestricted release are the required results of the decommissioning action that the licensee will be undertaking. The licensee's selected decommissioning alternative meets the requirements of 10 CFR 50.82(b)(1) and is acceptable because the decommissioning will be completed within 60 years, adequate procedures and controls to protect occupational and public health have been developed, the licensee has provided an adequate description of the final radiation survey, and developed an updated cost estimate for the DECON alternative.

1.2 Major Tasks, Schedules, and Activities

1.2.1 Description of Major Activities

The major dismantlement and decontamination activities to be performed during decommissioning are divided into three major work areas: decontamination and dismantlement of the PCRV, decontamination and dismantlement of the contaminated balance-of-plant (BOP) systems, and site cleanup and final site radiation survey.

(1) Decontamination and Dismantlement of the PCRV

The major decommissioning task is the dismantlement and decontamination of the reactor internal components and the radioactive portions of the PCRV. Activities to dismantle the PCRV will begin after all irradiated fuel has been removed from the reactor building and transferred to the independent spent fuel storage installation (ISFSI) or to the Idaho National Engineering Laboratory. Section 2.3 of the PDP provides a detailed description of the steps necessary to dismantle and decontaminate the PCRV. All dismantlement and decontamination activities must be accomplished in accordance with Technical Specification (TS) 5.7, "Radiation Protection Program."

After it evaluated several technical options for dismantling radioactive portions of the PCRV, PSC decided to flood the PCRV so that a majority of dismantlement activities could be performed under water. This approach allows direct access to highly radioactive portions of the PCRV, while affording the maximum shielding benefit, which provides significantly lower reduced estimated worker exposure than other approaches. However, this approach has raised concern regarding the sealing of the PCRV penetrations and about the possible releases of large amounts of tritium from the graphite blocks.

The PCRV will be dismantled using a diamond wire cutting technique. This is a standard construction method for cutting large volumes of concrete. In summary, the diamond wire cutting system consists of a wire with collars containing a diamond-matrix, made to length for each individual cut, and a hydraulic pulley to drive the system to circulate the wire. The diamond wire is routed to envelop the cut area. Chapter 2 of the PDP provides a detailed discussion on the use of the diamond wire cutting to dismantle the PCRV. The PCRV top head will be cut in several sections using the diamond wire and removed. The activated concrete in the PCRV walls will be cut with the diamond wires. This will be accomplished by removing vertical and circumferential tendons for access for the diamond wires. In cases were the tendon tubes are not useable, new vertical holes will be core drilled to allow a complete cut. After the concrete has been removed by the diamond wire method, additional decontamination by scabbling, vacuum sand blast, or wiping may be required in some areas to met the release criteria. The licensee has demonstrated that the use of these methods to dismantle the PCRV provides adequate protection of the worker and maintains occupational doses ALARA.

The sealing of the PCRV, and the possible release of large amounts of tritium are addressed in Chapters 2 and 3 of the PDP respectively, and in Chapter 4 of the Environmental Report Supplement. TS 5.7 provides adequate radiation protection requirements with regard to tritium in the PCRV. The licensee has committed to the application of Regulatory Guide 1.143 for activities related to flooding the PCRV. Before flooding the PCRV, all penetrations through the PCRV will be sealed. The penetrations will be sealed by either cutting and capping the penetration outside the PCRV or by installing bolted and gasketed flanges. In addition, where welding is required, all welds will be nondestructively tested in accordance with applicable codes. All leakage resulting from flooding the PCRV will be treated by means of the disposal demineralization and filtration system that is part of the PCRV water cleanup and clarification system. The leakage will be detected by visual inspection. Section 2.2 of this report provides a detailed discussion regarding the dismantlement of the PCRV, and Section 3.3.2 of this report provides a detailed analysis and evaluation of this concern.

(2) Decontamination and Dismantlement of Contaminated BOP Systems

For the purposes of the PDP, BOP systems refers to those contaminated plant systems outside the PCRV. PSC will decontaminate and dismantle contaminated BOP systems as described in Chapter 2 of the PDP. The BOP system are listed below. In summary, the BOP systems will either be decontaminated in place by conventional methods or removed and disposed of as low-level radioactive waste. The licensee will use conventional methods such as shears, scabbling, mechanical cutting and flame cutting to remove the BOP. These methods minimize worker exposure and maintain occupational doses ALARA.

BOP SYSTEMS

System 13 - Fuel Handling Equipment
System 14 - Fuel Storage Facility
System 16 - Auxiliary Equipment
System 21 - Helium Circulatory Auxiliary Equipment
System 23 - Helium Purification System
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 & Controls

(3) Final Radiation Survey Plan and Site Cleanup

Chapter 4 of the PDP specifies the release criteria that PSC will use in decommissioning FSV and provides a detailed description of the final radiation survey. The PDP release criteria is consistent with criteria provided to PSC in NRC letter dated October 4, 1989, and SECY-92-106 and confirmed by NRC letter dated April 27, 1992. Therefore, decontamination of FSV to levels that meet the release criteria specified in the PDP will allow termination of License DPR-34.

The proposed final radiation survey must demonstrate the effectiveness of the decommissioning and provide documentation that all contaminated materials, structures, areas, and components have been successfully removed or decontaminated to acceptable levels to permit release for unrestricted use. This final radiation survey to release the FSV site, facilities, and installed equipment for unrestricted use will be performed following the completion of the decontamination and dismantlement activities. The proposed survey plan is acceptable for proceeding with decommissioning, but the staff will reevaluate the adequacy of the survey and the survey results when FSV decommissioning is complete. An independent survey by the NRC or an independent contractor will be used to confirm that the residual radioactivity at FSV meets the NRC criteria discussed above. Section 4 of the SER evaluates the final radiation survey methodology and criteria and concludes that the final survey plan meets the requirements of 10 CFR 50.82(b)(3).

1.2.2 Schedule for Decommissioning Activities

PSC has completed the decommissioning planning phase. It consisted of preparation of work scope planning, work specifications and procedures, and equipment and material staging. PSC estimated that decontamination and dismantlement (i.e., actual dismantlement, decontamination, and physical decommissioning activities) will take about 39 months. Section 2.3.5 of the PDP provides a detailed schedule of FSV decommissioning and decontamination activities. Figure 1 of this report provides a time line of FSV decommissioning. The NRC staff concludes that PSC has addressed all major activities and the schedule for completion of decontamination of FSV is reasonable on the basis of NUREG/CR-0130 as well as a comparison with other facilities.

1.3 Cost Estimate and Availability of Funds

1.3.1 Decommissioning Cost

PSC estimated a total decommissioning cost of $157,472,700 for FSV. Assumptions used as the basis for these costs are identified in the PDP. Section 5 of this report provides a detailed evaluation of the cost for decommissioning FSV. The NRC staff concludes this cost estimate is reasonable and satisfies the requirement of 10 CFR 50.82(b)(4).

1.3.2 Decommissioning Funding Plan

The First Interest Bank of Denver, N.A., entered into a standby trust agreement with PSC for the purpose of receiving payment under an irrevocable letter of credit issued to the PSC account. The letter of credit provides financial assurance for the decommissioning of FSV.

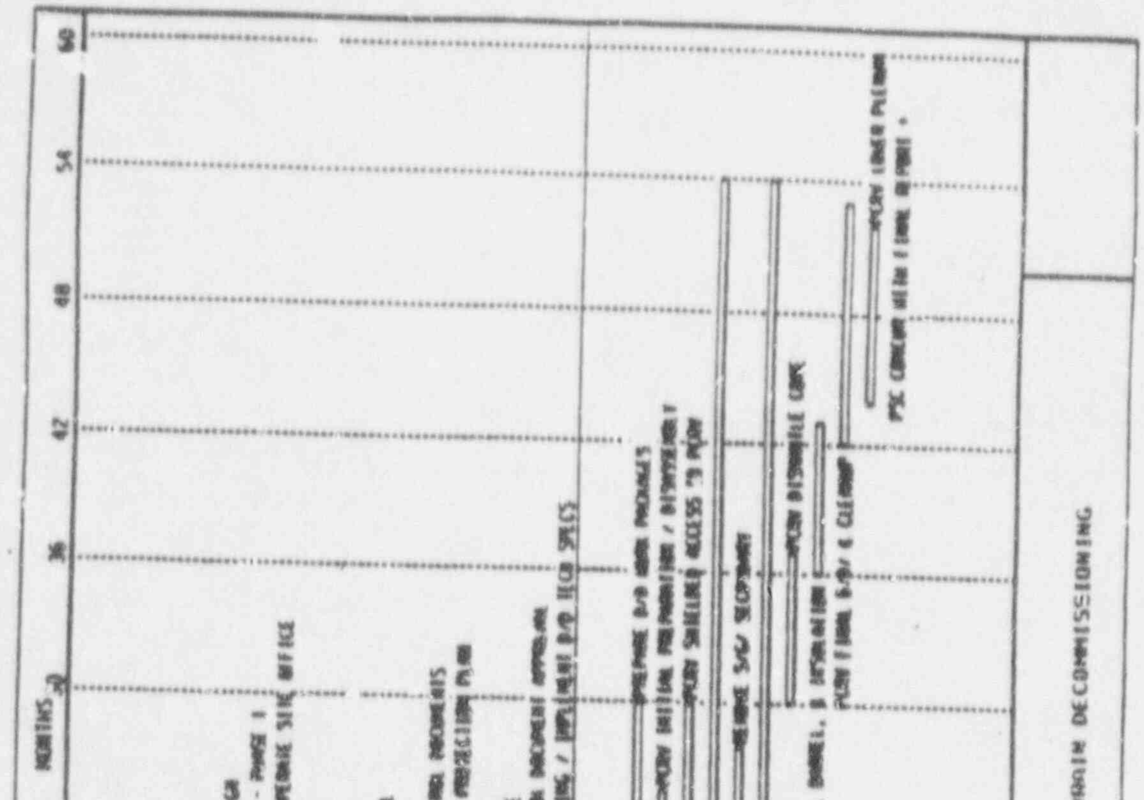

Figure 1 Summary Schedule of Major Decommissioning Tasks

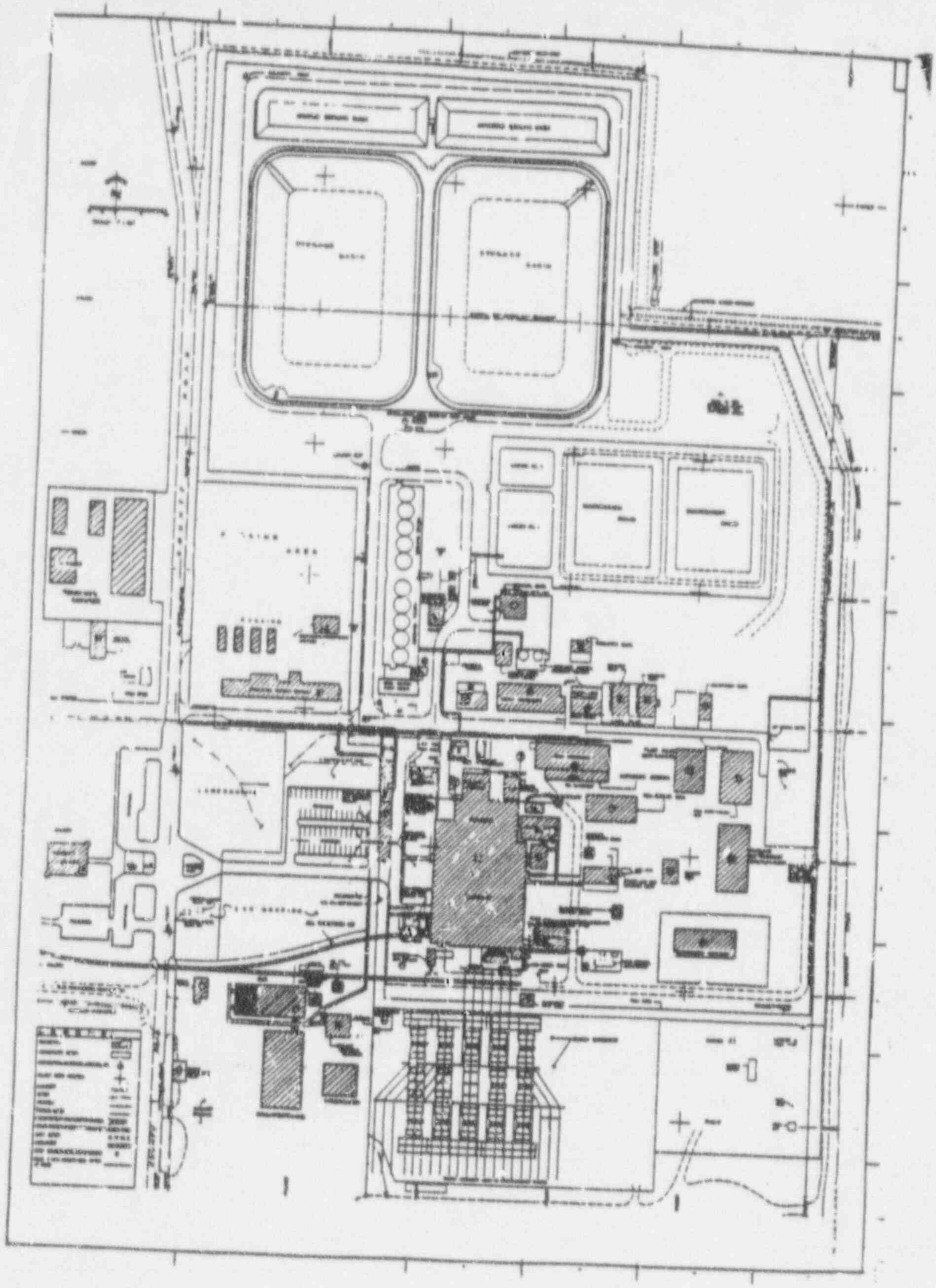

Figure 2 Fort St. Vrain Plot Plan

The term of the standby trust agreement commences when the letter of credit is issued and expires when the decommissioning activities at the facility are completed, or as otherwise provided in the standby trust agreement. The bank acts as trustee and administers any funds received to fund costs for decommissioning in accordance with the terms of the standby trust agreement. PSC proposes to use a $28 million external trust fund plus a $125 million letter of credit that will decline as DECON operations are completed. The combined total of $153 million exceeds the estimated decommissioning cost of $157.5 million, less expenses to date of $10.5 million. The NRC staff concludes that PSC's proposed decommissioning funding assurance mechanism is acceptable and in compliance with 10 CFR 50.75(e).

2 GENERAL DESCRIPTION OF FORT ST. VRAIN

FSV is a high-temperature gas-cooled reactor (HTGR) owned and operated by PSC. FSV is located approximately 35 miles north of Denver and 3.5 miles northwest of the town of Platteville in Weld County, Colorado. FSV had a capacity of 330 MWe. Figure 2 of this report shows the completed facility.

The PSC-owned site consists of 2798 acres. Approximately 1 mi^2 within the site area is designated as the exclusion area; PSC maintains complete control over this area. The closest distance from the reactor building to the nearest exclusion area boundary is about 1935 feet, but the reactor building is about 3500 feet from the nearest site boundary.

The Atomic Energy Commission (AEC) issued a provisional construction permit to PSC on September 17, 1968 (AEC Docket No. 50-267). Fort St. Vrain was initially scheduled for commercial operation in 1972. Although PSC received a full-power operating license in 1973, NRC mandated extensive pre-operational testing and the resulting engineering modifications delayed commercial operation of the test until 1979.

Chapter 2 of the PDP provides a complete description of the Fort St. Vrain facility. The major events and milestones that occurred at FSV are listed below.

1973	December	Plant construction completed, facility Operating License DPR-34 issued to PSC
1974	January	Initial criticality achieved, startup testing, low-power operation, and required plant modifications implemented (1974-1979)
1979	July	Commercial operation began
1981	November	100-percent full-power operation achieved
1984	June	Six control rod drives (CRDs) failed to automatically scram causing shutdown
1986	February	Plant restarted following CRD refurbishment outage
	May	Environmental qualification outage
	September	FSV removed from the rate base, initial decommissioning started

	October	Safe shutdown cooling reanalysis performed, reducing maximum power level to 82% of rated power (270 of 330 Mwe)
1987	July	Plant shut down following helium circulator bolt failure
	October	Hydraulic fire during plant restart
1988	June	Plant record achieved for MWe generated for 1-month period
	July	Plant shut down to refurbish helium circulators
	December	Decision approved by PSC Board of Directors to shut down and decommission (Operations will cease on or before June 30, 1990.)
1989	August	Plant permanently shut down because of control rod failures and subsequent discoveries of failure of steam generator ring headers
1990	November	Decommissioning plan (DECON option) submitted
1991	May	Possession-only license issued
	December	Movement of the fuel to ISFSI initiated
1992	May	Core defueling completed

2.1 Decommissioning Activities, Planning, and Exposure Estimates

Decommissioning of FSV includes the dismantlement, decontamination, and disposal of radioactively contaminated material and components within the PCRV, of contaminated balance-of-plant systems, and of contamination on the remaining site, followed by the final radiation survey. The activated and contaminated portions of FSV will be decontaminated, dismantled, and removed during the decommissioning process in compliance with TS 5.7 and TS 5.8, "High Radiation Area," of Appendix A to Docket 50-267. The activities are divided into decontamination of the PCRV and decontamination and dismantlement of the contaminated BOP. The licensee has provided a detailed analysis in Section 3.1.2 of the PDP regarding the radiation levels at the facility. The radiation levels were determined on the basis of an August 1990 survey and supporting activation analysis.

Table 1 of this report provides a summary of the estimated exposures for decommissioning the facility. While FSV is an HTGR and is different from the typical boiling-water and pressurized-water reactors, many activities, and procedures are similar. A comparison of FSV estimated exposure rates to those of Pathfinder, Shoreham, and the generic estimates of NUREG/CR-0130 show that the exposure rates are reasonable and the estimated exposure for decommissioning FSV is considerably less than the generic estimates. NUREG/CR-0130 provides total exposure estimates of 1400 man-rem compared to 433 man-rem for FSV. A detailed evaluation of these exposure estimates is provided in Chapter 3 of this report. The staff concludes that although personnel conducting the dismantling activities will be exposed to radiation during the dismantling and decontamination, PSC has developed activities and procedures to limit exposure and control radioactive material in order to maintain occupational doses as low as is reasonably achievable (ALARA). Exposure estimates to accomplish the individual tasks and overall project are reasonable.

Table 1 Projected Person-Rem Exposure for the Fort St. Vrain Decommissioning Project

Work Activity	Estimated Person Hours*	Estimated Person Exposure**
PCRV Dismantlement and Decontamination (D/D):		
Initial preparation/ disassembly	23,733	7.4
Remove PCRV concrete top head	20,578	20.4
Dismantle PCRV core and core barrel	49,368	157.3
Remove core support floor, barrel, and insulation	9,213	103.4
D/D PCRV Lower plenum	16,103	59.9
Final PCRV dismantlement, decontamination and cleanup	15,047	17.7
Subtotal:	134,042	366.1
Contaminated BOP D/D and Waste Packaging:		
Initial preparation/characterization	7,279	0.25
Dismantle/decon operations	58,684	1.4
Subtotal:	65,963	1.65
Waste Preparation, Packaging, Shipping, and Disposal	33,055	65.4
Total:	233,060	433.15

*Person-hours only for those tasks where the potential for measuring radiation exposures exists.

**Exposure time (worker efficiency) is estimated to be 50% of scheduled work time for PCRV tasks where the potential for radiation exposure exits.

To accomplish decommissioning, substantial portions of the existing plant will be dismantled and removed. However, the reactor and turbine buildings and structures that are not radioactive above limits suitable for unrestricted use will remain. The radiation program provides adequate requirements for radiation protection of workers and the public. Flooding of the PCRV with water to provide shielding will be the principal reason for the reduced worker exposure. This reduced exposure is discussed in detail in the Environmental Report Supplement to the PDP.

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 of this report.

2.2 Decommissioning Organization and Responsibilities

In Section 2.4 of the PDP and in TS 5.0, "Administrative Control," PSC identified the key positions in the decommissioning organization and described their functions. The lines of authority to the corporate level are indicated in Figure 2.4.1 of the PDP. The education, training, and experience requirements are described for all positions important to decommissioning safety. The person with ultimate onsite authority for various functional areas is the PSC Decommissioning Program Director who has overall responsibility for all decommissioning activities conducted by PSC and contractors. The decommissioning organization also includes a Decommissioning Safety Review Committee (DSRC) to monitor the decommissioning operation to ensure that it is being performed safely. The DSRC will review and audit major decommissioning operations dealing with radioactive material, radiological controls, review procedures, records, reportable occurrences under 10 CFR Parts 20 and 50, and changes made in accordance with 10 CFR 50.59. The responsibilities and function of the DSRC are defined in TS 5.3, "Decommissioning Safety Review Committee." The committee reports to the Vice President Nuclear Operations.

The staff concludes that the licensee's proposed decommissioning organizational structure is acceptable and is accordance with the provisions of NUREG-0800 Sections 13.1.1, "Management and Technical Support," and 13.1.2 and 13.1.3, "Operating Organization."

2.3 Training Program

The licensee training program is described in Section 2.6 of the PDP and provides general employee training for all decommissioning personnel. The radiation worker training will incorporate the requirements of 10 CFR 20.103 and the guidance of Regulatory Guide 8.15. The training and qualifications of the health physics technicians and supervisors will be conducted in accordance with American Nuclear Society/American National Standards Institute (ANS/ANSI) Standard 3.1-1981. PSC stated that specific job training will be provided for decommissioning personnel on the basis of specific job requirements. Records of all training will be maintained. Because the training program for the decommissioning personnel is in accordance with the provisions of NUREG-0800 Section 13.2, "Training," it is acceptable and ensures that the licensee will be able to maintain ALARA.

2.4 Contractor Assistance

PSC will retain overall responsibility for the decommissioning of FSV. PSC selected Westinghouse and its support contractors to perform the decommissioning of FSV. Westinghouse is the prime contractor and will provide engineering and licensing support to PSC. M. K. Ferguson will provide site labor, labor management, and support to Westinghouse.

Chapter 2.5 of the PDP describes the scope of work to be accomplished, the administrative controls to be used to ensure adequate health and safety protection, and the qualifications and experience of the contractors. The staff concludes that PSC provided adequate information on its contractors and is capable of retaining overall responsibility for decommissioning.

3 PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

3.1 Facility Radiological Status

The staff reviewed the operating history and radiological conditions in the plant and evaluated the activities and tasks to be carried out in contaminated areas. The staff relied on Regulatory Guide DG-1005, "Standard Format and Content for Decommissioning Plans for Nuclear Reactor," and applicable sections of 10 CFR Parts 20 and 50 for review guidance.

3.1.1 Facility Operating History

In Section 3.1.1 of the PDP, PSC addressed conditions in the plant that could affect decommissioning, such as radioactive spills, potential contamination in inaccessible areas, and operating events that had the potential to spread radiation. During the operational history of the plant there have been no spills or releases of radioactive effluents resulting in significant residual radioactive contamination either on site or off site. However, there have been a few routine plant operations that may have resulted in residual radioactive contamination in areas that are inaccessible.

Specifically, the fuel storage wells and equipment storage wells on the refueling floor were used to store spent fuel and highly radioactive components. Over the years of transferring various components and spent fuel, it is anticipated that high levels (e.g., 5,000,000 dpm/100 cm^2) of loose surface contamination have accumulated on horizontal surfaces. The lower portions of these wells are inaccessible. At various times throughout plant history, the hot service facility also has had levels of loose surface contamination measuring greater than 5,000,000 dpm/100 cm^2. Periodic decontamination was typically performed using water; as a result, crud traps may have been created in inaccessible areas. To date, no crud traps have been identified in accessible areas containing drain piping from the hot surface facility.

The results of the August 1990 radiation survey demonstrated that greater than 95 percent of the plant areas have radiation levels corresponding to background. Table 3.1-1 of the PDP identifies those areas with radiation levels above background.

The staff finds that PSC provided sufficient information and that the information is acceptable and meets the requirements of 10 CFR 50.75(g)(1).

3.1.2 Radiological Status of Plant

The radiological conditions of the plant are described in Section 3.1.2 of the PDP. The staff evaluated the radiation hazards at the plant on the basis of the activation analyses performed and the August 1990 radiation and contamination survey performed on the reactor building and turbine building. Table 3.1-1 of the PDP provides a summary of the contamination levels at the facility with a description of major contributors.

General area radiation levels throughout the turbine building are primarily due to natural background. Contamination levels both fixed and loose are less than 1000 dpm/100 cm^2 at all locations and generally less than 100 dpm/100 cm^2. External radiation levels outside both the turbine building and reactor buildings are typically less than 2 mrem/hr.

The PCRV served as containment of the nuclear steam supply system and all the internal PCRV components will either require decontamination or will be removed and disposed of as radioactive waste.

PSC conducted activation analyses on the PCRV. Table 3-1.2 of the PDP provides a summary of the radiation levels for the PCRV and its components. PSC has identified several additional surveys and samplings to be conducted once the fuel is removed to confirm the radiation levels.

In addition, the radiological status of the site and surrounding areas has been monitored during the entire life of the facility through the radiological environmental monitoring program. The results are included in the PDP.

The staff concludes that PSC has provided sufficient information on the radiological status of the plant to meet the requirements of 10 CFR 20.201 to survey the facility for radiological hazards.

3.2 Radiation Protection

The staff reviewed the licensee's radiation protection program and the licensee's commitment to the protection of the workers and public during decommissioning and evaluated the task and activities that would be required to support decontamination. The staff relied on Regulatory Guide DG-1005 and applicable sections of 10 CFR Parts 20 and 50 for review guidance.

Section 3.2 of the PDP was prepared to NUREG-0761, which provides guidance for the content of a radiation protection plan. It also incorporates the guidance contained in Regulatory Guides 8.8 and 8.10. Section 3.2 of the PDP also stated that the radiation protection program for FSV will incorporate the requirements of the 1991 revision to 10 CFR Part 20 no later than January 1, 1994, in accordance with the current schedule for implementation. However, depending on the decommissioning status at that time, PSC may apply for an exemption to complete decommissioning under the current Part 20.

PSC's radiation exposure estimates are discussed in Section 2.1 of this report. In support of the ALARA goals, PSC will develop and implement the respiratory protection program in accordance with 10 CFR Part 20, Regulatory Guide 8.15, and NUREG-0041. In addition, the Radiation Protection Managers (RPMs) and radiation protection staff will be qualified in accordance with Regulatory Guide 1.8 and ANS/ANSI 3.1.

The administrative organization and functional responsibilities for implementation of the radiation protection program are described in Section 3.2.3 of the PDP and in TS 5.0. The PSC RPM is assigned primary responsibility for implementation of the program with administration of the Westinghouse team radiation protection activities under the Westinghouse Project Radiation Protection Manager (PRPM). The PSC RPM will have direct communication interface with the PRPM and maintain overall responsibility for the radiation protection program. Adequate radiation protection staffing will be maintained consistent with the decommissioning activities in progress. The radiation protection staff and the RPMs will be qualified and trained in accordance with Regulatory Guide 1.8 and ANS/ANSI 3.1.

Section 3.2.4 of the PDP describes the radiation protection initial training, qualification, and retraining program. Appropriate training will be provided for non-radiation workers, radiation workers, and radiation protection personnel in accordance with 10 CFR Part 19 and applicable guidance contained in Regulatory Guides 8.27, 8.13, and 8.29. The content of radiation worker training also will be consistent with Appendix A of NUREG-0761.

Section 3.2.3 of the PDP describes the radiation dose control elements to be incorporated in decommissioning radiation protection procedures. These include controlling sources of radiation, controlling access to areas containing radioactive materials, using radiation work permits for administrative control of personnel entering and working in radiological areas, measuring radiation exposures of workers, controlling and monitoring internal doses, and administering a program to maintain occupational doses ALARA. Appropriate caution signs and labels will be provided in accordance with 10 CFR 20.203 and 20.204. All project workers entering radiologically controlled areas will be required to wear external radiation monitoring devices consisting of thermoluminescent dosimeters (TLDs) and self-reading or digital alarming dosimeters as described in Section 3.2.5.6 of the PDP. The TLDs will be processed at an appropriate frequency by an outside vendor accredited by the National Voluntary Laboratory Accreditation Program. Whole-body counts of all radiation workers will be conducted on a scheduled basis and indirect bioassay measurements will be made as necessary to assess the intake of radioactive materials in accordance with 10 CFR 20.103.

The health physics instruments and equipment used in the radiation protection program are described in Chapter 3 of the PDP and include portable radiation survey instruments, personnel monitoring equipment, air samplers, respiratory protection equipment, and protective clothing. Table 3.2-2 of the PDP provides a summary of the types and models of equipment used in the radiation protection program. Section 3.2.8.2 of the PDP identifies the procedures for calibration and response checks of the radiation monitoring equipment and air sampling equipment. Radiological surveys will be conducted with appropriate instruments in accordance with 10 CFR Part 20.

All radioactive material entering or leaving the radiologically restricted areas will be controlled as described in Section 3.2.6 of the PDP. Interim storage of radioactive materials and processing of liquids containing radioactive materials will require a safety evaluation and will be in compliance with NRC Generic Letter 81-38. Materials and equipment released from radiologically controlled areas for unrestricted use will not contain detectable amounts of radioactive material as determined in accordance with the guidance of NRC Circular 81-07 and NRC Information Notice 85-92.

Radioactive liquid and gaseous effluent releases will be monitored and controlled using installed plant equipment in accordance with TS 5.4.4 and the methodology contained in the Offsite Dose Calculation Manual (ODCM), in conformance with 10 CFR Part 20, Appendix A and Appendix I to 10 CFR Part 50, and 40 CFR Part 190. The ODCM will be used to establish set points for FSV effluent radiation monitors so that the concentration limits at the site boundary are within limits of 10 CFR Part 20. The ODCM also will be used to establish the methods for periodic assessment of doses to individuals from routine gaseous and liquid effluents to demonstrate compliance with Appendix I to 10 CFR Part 50. This will limit the average concentrations of radioactive materials released in effluents to unrestricted areas to a small fraction of the limits in Appendix B, Table II, Columns 1 and 2 to 10 CFR Part 20. The resulting individual doses per year for gaseous effluents will not exceed 5 mrem to the total body and 15 mrem to the skin or 15 mrem to any organ from particulate radioactivity. For liquid effluents, including tritium, the resulting doses per year will not exceed 3 mrem to the total body or 10 mrem to any organ. Compliance with Appendix I to 10 CFR Part 50 will ensure that the dose from tritium in the drinking water pathway will not exceed the standard in 40 CFR 141. The effect on radiological conditions in the environment as a result of decommissioning activities will be determined by continuation of specific parts of the existing radiological environmental monitoring program that will monitor area radiation, water samples, air samples, and vegetation samples.

The staff concludes that the radiation protection program provides sufficient control of radioactive materials during decommissioning and meets the requirements of 10 CFR 50.82(b)(2) regarding the description of the controls and limits on procedures and equipment to protect occupational and public health and safety.

3.3 Radioactive Waste Management

Section 3.3 of the PDP provides detailed information on the technologies, equipment, and procedures to be implemented for the management of radioactive waste during the decommissioning of FSV.

3.3.1 Spent Fuel Disposal

Although not directly related to proposed decommissioning plans, the following information is provided on the disposition of the FSV spent fuel. PSC's preferred plan to manage spent fuel is to ship the spent fuel to the DOE facility in Idaho. A three-party agreement between PSC, General Atomic, and the Atomic Energy Commission provided storage for eight segments of FSV spent fuel at the Idaho National Engineering Laboratory (INEL) facility. To date, PSC has shipped three segments of spent fuel to INEL as a result of three previous refuelings.

Because of the uncertain schedule for shipping of spent fuel to Idaho or other DOE facilities, PSC pursued an alternate plan and licensed, constructed, and is operating an ISFSI that is separately licensed under 10 CFR 72. The ISFSI facility is located immediately adjacent to the current site and the location is outside the plant's existing protected area and is approximately 1500 feet northeast of the reactor building. The ISFSI, using the modular vault dry store system, is designed to store all of the remaining FSV spent fuel, up to 37 metal-clad reflector blocks (MCRBs) and up to 6 neutron sources. FSV is currently utilizing the ISFSI alternative and all remaining spent fuel has been transferred to the ISFSI.

3.3.2 Radioactive Waste Processing

During the FSV decommissioning project, the PCRV cavity will be flooded with water to provide shielding and contamination control. Flooding the PCRV will result in the release of radionuclides (that exist in the PCRV as a result of activation and plateout) into the water. The radionuclide of primary concern is tritium. Part of the tritium inventory is expected to leach out of the graphite blocks into the water and tritium cannot be removed by conventional processing means employed by the PCRV shield water system. The amount of tritium to be handled by the PCRV shield water system and potential exposure to personnel will depend on both the total amount of tritium present in the graphite and other components inside the PCRV and the fraction that is released to the water. Chapter 4 of the Environmental Report Supplement provides a detailed discussion of the effect resulting from release of tritium. Because the tritium concentrations in the PCRV graphite components and the rate at which the tritium leaches into the water from the graphite cannot be easily measured, the amount of tritium that enters the PCRV water has been estimated, based on a conservative calculation of the total amount of tritium produced during power operation (i.e., 100,000 curies [Ci]) and actual measurements of tritium leach rates from British Magnox reactor graphite. PSC estimates that approximately 500 Ci (or 0.5% of the total tritium inventory) will enter the water. The PCRV shield water system will process this tritium inventory for discharge using the existing liquid effluent discharge path and dilution.

The maximum tritium inventory in the graphite that could exist in the PCRV when it is flooded is:

Source	Curies
Large permanent side reflectors	82,588
Boronated side spacer blocks	11,532
Removable hexagonal reflector blocks	3,500
Core support blocks and bottom reflectors with hastelloy cans	48
Total:	97,638

For the purposes of estimating the amount of tritium in the graphite, a tritium inventory of 100,000 Ci is assumed. The 100,000 Ci inventory was based on the March 1992 Actuation Analysis, EE-DEC-0010, for FSV.

Data on tritium leaching from graphite obtained by the British was the basis for the estimate of the fraction of the tritium inventory likely to be leached from the FSV graphite after the PCRV is flooded. These British measurements were made in support of decommissioning of the Magnox plants by P. B. Woolam and I. G. Pugh. For the two samples of British graphite that were tested in demineralized water, the leach rate of the tritium was measured to decrease with time starting at about 0.1 percent per day and declining to below 0.0001 percent per day after several months. Applying these values to FSV, a curve of tritium release rate versus time was prepared with a cumulative tritium release rate of 0.5 percent of the tritium inventory in the graphite released in about the first month after flooding the PCRV. Use of this release rate results in a release of 500 Ci from the graphite and absorbed by the water, based on an assumed initial tritium inventory of 100,000 Ci in the core graphite.

The large side reflector blocks and the boronated side reflector blocks at FSV are made of commercial grade HLM graphite and have about 50 times as much lithium per unit volume as the British reactor grade graphite. Therefore, the production rate for tritium during reactor operations at FSV was also 50 times as much. The HLM graphite at FSV differs from the reactor grade British graphite in other aspects also and no confirming tritium release tests have been done on the FSV HLM graphite or any other similar, commercial grade graphite. The entire analysis is based on the assumption that the tritium leaching properties of the FSV HLM graphite are expected to result in a more conservative behavior than the British Magnox test samples. HLM surface-to-volume ratios are significantly lower, indicating that HLM graphite water ingress will not occur as rapidly and tritium migration to the graphite surface will take significantly longer. The densities of the irradiated HLM are greater than the British graphite samples, indicating lower porosity and a lower leach rate in the HLM graphite as a result of density. In addition, effect on increased porosity should be greater in the British samples than in the HLM because the effects of reactor power history favor the HLM graphite. Therefore, with no applicable confirming tests for the 0.5 percent (500 Ci) release, PSC has established additional administrative limits on the amount and the rate that tritium could be released through dilution of the shield water.

PSC established administrative controls to limit the total tritium release to 8000 Ci and to restrict the tritium release rate, following dilution, to 20,000 picoCi per liter in the South Platte River (the U.S. Environmental Protection Agency's [EPA's] average annual concentration limit for tritium in drinking water, 40 CFR 141.16). PSC evaluated the radiological consequences of a release of tritium to the environment and determined that the maximum exposure to any member of the public would be less than 4.0 mrem per year through any pathway with the above administrative limits maintained. If more than 8000 Ci of tritium are released to the shield water, PSC has stated that it would solidify the water and dispose of it as a solid waste.

PSC estimated that filling the PCRV will require approximately 325,000 gallons of water. Filling of the PCRV will be stopped at predetermined levels (1/4 core increments) to allow tritium sampling and analysis. 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/cc. Limits have been established in the decommissioning technical specifications to ensure that tritium activity concentrations in the PCRV shield water system will not exceed those postulated in the decommissioning accident analyses. TS 3.4, "PCRV Shielding Water Tritium Concentration," established specific requirements regarding tritium concentration and frequency of the analysis of the PCRV shield water.

Because the entire estimate of the release is based on theoretical analysis, PSC assessed what the effects might be if the maximum theoretical amount of tritium (100,000 Ci) is released into the PCRV shield water, including effects on air handling, tritiated water disposal, contamination, and personnel protection. If the 100,000 Ci is released, the licensee has allowed sufficient funding to solidify the tritiated water, and ship it to a low-level waste disposal site. Allowing for this case, decommissioning can proceed and will be accomplished within the decommissioning cost estimate previously submitted to the NRC. In addition, with considerations for the worst credible accident and this extreme case, the staff finds that decommissioning can be accomplished without undue risk to the safety of the public.

In conclusion the assumptions regarding the amount of tritium released into the water are reasonable, and the worst-case scenario was analyzed for the entire

100,000 Ci of tritium released into the PCRV. The staff concludes the requirements of 10 CFR 50.75(f)(2) have been adequately addressed. The environmental effects of the volume of tritium are addressed in detail in the Evironmental Report Supplement.

3.3.3 Radioactive Waste Disposal

PSC initially estimated the processed and volume-reduced radioactively contaminated waste for disposal as 100,072 ft^3, with 99,219 ft^3 from the PCRV and associated operations, and 853 ft^3 from BOP. PSC stated in Section 3.3.3 of the PDP that it is negotiating a contract between the Rocky Mountain Compact (RMC) Board and the Northwest Compact Board to allow access for the waste generated from RMC States to the existing Northwest Compact disposal facility beginning in January 1993. In support of this effort, PSC has added an additional $12,441,000 to the standby trust agreement to cover the additional cost of disposal. The waste from the PCRV consists of activated concrete, graphite blocks, other activated components, miscellaneous equipment and piping, and concrete rubble. The PCRV waste is contaminated principally with Fe-55, tritium, and Co-60. The waste from the BOP consists of tanks; pumps; heating, ventilation, and air conditioning (HVAC) filters; and miscellaneous equipment and piping. There also may be radioactively contaminated asbestos. After processing and volume reduction, PSC estimated that the volume of radioactive waste will be segregated into the following categories:

Class	Volume (cubic feet)
A	84,000
B	15,000
C	1,000

PSC stated that, because of the uncertainties in the analysis, as much as 400 ft^3 of Class C wastes may be reclassified as greater than Class C (GTCC). The PDP has stated that waste volume estimates may change as decommissioning operations proceed. Tables 2 and 3 of this report provide summaries of the estimated volume of wastes and the classification, number, and type of containers necessary for shipping and disposal. PSC also stated that, if mixed wastes are generated, they will be managed according to Subtitle C of the Resource Conservation and Recovery Act (RCRA). PSC also stated that it did not intend to petition the EPA to delist any mixed waste.

Section 3.3.3.6 of the PDP addresses the storage of the waste at FSV. The wastes will be stored at various plant locations depending on the classification of the wastes. As an example, the ISFSI will be used for storage of greater than Class C waste. In addition the fuel storage building, compactor building, the reactor building, as well as additional areas, will be available for storage. The waste storage will be based on guidelines in NRC Generic Letter 81-38 and Appendix 11.4-A to NUREG-0800.

The staff finds that PSC's analyses and estimates of the volumes of waste generated during decommissioning as well as the waste classification of the proposed waste and practices and methods for meeting the transportation requirements are reasonable and consistent with the applicable requirements of 10 CFR Parts 20, 61, 71, and the requirements of 10 CFR 50.82(b)(1)(iii).

Table 2 FSV PCRV Waste Volume Estimates

Item/System	Class	LSA	Number	Volume (Ft^3)*	Con-tainers
Region constraint device and pin	C	No	84	200	2
Metal control rod reflectors	C	No	37	400	3
Metal block, non control rod	B	No	276	2000	13
Defueling blocks	A	Yes	1482	7200	75
Top reflector graphite blocks	A	No	1215	1500	8
Bottom reflector graphite blocks	A	No	1215	1400	8
Radial reflector (perm. and rmvble)	A	No	480	1900	9
Large reflector blocks	B	No	312	12600	50
Half-size reflector blocks	A	NO	312	2100	8
Upper reflector keys (carbon steel)	A	No	24	200	2
Side spacer blocks with					
boron rods	B	No	197	[illegible]	66
rods removed	A	No	1152	2400	25
Bottom reflector blocks with					
cans (Hastelloy)	C	No	20061	375	50
cans removed	A		276	800	8
Lower reflector keys (Hastelloy)	B	No	24	200	1
Core support blocks	A	Yes	61	1500	15
Core support posts	A	Yes	183	200	2
Core support floor columns	A	Yes	12	600	7
Misc steel from beneath CSF	A	Yes		1000	10
Metal on large side reflector	A	Yes	24	100	1
Core barrel	A	Yes	1	1400	31
Lower plenum insulation	A	Yes		900	10
Silica blocks (25,000 lbs)	A	Yes		500	12
Concrete - top	A	Yes		3700	9
Concrete - CSF	A	Yes		6200	15
Concrete - side	A	Yes		19000	45
Concrete rubble - jackhammer	A	Yes		700	16
Misc Inconel parts on CSF	A	No		400	5
Concrete cutting debris - top	A	Yes		200	
Concrete cutting debris - CSF	A	Yes		200	8
Concrete cutting debris - side	A	Yes		300	
Helium purifiers in PCRV head	A	Yes	10	500	5
Helium diffusers	A	Yes	4	1750	4
Helium circ shutoff valve assembly	A	Yes	4	200	2
Helium bellows	A	Yes	12	1600	12
Steam generators	A	Yes	12	21000	12
Themocouples and guide tubes	B	No		100	1
Lower floor/appurtenances	A	Yes		1200	42
Platform/handling tools/jib cranes	A	Yes	576	10	
Crane cable/drum/3 bucket inverters	A	Yes		500	5
Misc containers	A	Yes		300	3
PCRV water system	A	Yes		2100	2
Resins - solidify, ship, bury	A**	No	20	2700	20
Misc soft waste	A	Yes		13000	125
PCRV Totals:				114,500	740

* Estimated pre-volume reduced quantity.

** Estimated Burial Class - Specific burial class identification may require additional analysis with 10 CFR 61.

Table 3 FSV BOP Waste Volume Estimates

Item/System	Class	LSA	Number	Volume (FT^3)**
Reaction isolation valves	A	Yes	5	1000
Refueling sleeves	A	Yes	2	200
Sand from fuel storage wells	A	Yes		800
Sand from equipment storage wells	A	Yes		200
Sand from helium regeneration pit	A	Yes		100
Auxiliary transfer cask sand	A	Yes		100
Hot cell facility	A	Yes		400
Sand from hot cell facility	A	Yes		500
Core support vent filters	A	Yes		10
Gaseous waste surge tanks	A	Yes	1	1000
Gaseous waste compressors	A	Yes	2	2100
Liquid waste monitor tank	A	Yes	1	600
Liquid waste demineralizers	A	Yes	2	200
Liquid waste receivers	A	Yes	2	1100
Liquid waste sump (sand)	A	Yes		20
Liquid waste transfer pumps	A	Yes	2	100
Liquid waste sump pumps	A	Yes	2	10
Liquid waste filters	A	Yes	2	10
Decon solution tank	A	Yes	1	400
Decon recycle pump	A	Yes	1	2
Decon chem supply pump	A	Yes	1	2
Purified helium filters	A	Yes	2	10
Helium removal filter	A	Yes	1	100
Helium getter units	A	Yes	2	10
HVAC filters	A	Yes		1000
Fuel handling machine	A	Yes		200
Fuel handling machine components	A	Yes		400
Small and large bore piping	A	Yes		600
Reactor bldg drain system	A	Yes		100
Instrumentation and controls	A	Yes		200
BOP Totals				11,500

* Estimated pre-volume reduced quantities.

3.4 Accident Analysis

In Section 3.4 of the PDP, PSC evaluated the effect of potential decommissioning accidents at FSV on the health and safety of the public. The activities, equipment, and circumstances associated with decommissioning are different from those evaluated in the FSV Final Safety Analysis Report for power operations and refueling.

The risk of accidents resulting in a radiological release during decommissioning activities was considerably less than during plant operation because all spent fuel will be removed from the reactor building. Therefore, only non-operations accident scenarios will be evaluated in this section.

The type of postulated accident and the resultant doses to an individual at the emergency planning zone (EPZ, 100 Meter Minimum) are given below.

Accident	2-Hour Dose (mRem) Whole-Body	Organ
Dropping of concrete rubble	4.92	58.0 (bone)
Heavy load drop	7.10	202 (lung)
Fire	121	215 (lung)
Loss of PCRV shielding water	34.8	34.8 (lung)
Loss of power	1.54	40.0 (lung)
Natural disaster (tornado)	0.58	16.8 (lung)
Dropping of steam generator primary module	8.3	90.7 (lung)

The results of the accident scenarios postulated for FSV decommissioning indicate radiation exposures to the general public are very low. The resulting analysis show that the radiological consequences at the EPZ are within the 10 CFR Part 100 guidelines and are only a small fraction of the EPA Protection Action Guidelines (EPA-520/1-75-001-A) and would therefore require no offsite response to the accident.

The staff compared the accident scenarios and releases to accidents in NUREG/CR-0130 and concludes the scenarios analyzed are representative of accidents that could occur at FSV during decommissioning and that none of the accidents has potential consequences (radiation doses) in excess of the EPA Protection Action Guidelines.

3.5 Industrial Safety

The proposed decommissioning activities involve a number of routine industrial safety hazards that are subject to regulation by other Federal agencies. In these areas, the NRC staff has not reviewed the licensee's decommissioning plan for regulatory compliance, limiting its review to radiological aspects only. Nevertheless, the staff has noted the presence of these hazards.

3.6 Asbestos

PSC's asbestos removal procedures will follow procedures for the safe removal and disposal of materials containing asbestos required by EPA regulations promulgated as "National Emission Standards for Hazardous Air Pollutants" (40 CFR Part 61) and Occupational Safety and Health Administration (OSHA) safe work practices required under 29 CFR 1926.58. PSC will provide respiratory protection in accordance with OSHA regulation 29 CFR 1910.134.

The BOP systems were surveyed and material containing asbestos was found on some piping in the helium purification system and radioactive waste gas system. Approximately 1500 linear feet of metal jacketed material will be required to be removed, packaged, and disposed. Two industrial hygienists and necessary industrial services will be on site to support this operation. The asbestos removal is addressed in detail in Section 3.5 of the Environmental Report Supplement and in the FSV decommissioning cost estimate (WBS 2.4.1). The staff concludes that the removal of asbestos is adequately addressed and follows the procedures required by 29 CFR 1926.58.

4 FINAL RADIATION SURVEY PLAN

Chapter 4 of the PDP describes the methodology and criteria that will be used in performing the final surveys at FSV. It included a definition of the residual radioactivity limits, radiation survey methods, materials release criteria, and the site release criteria. The final radiation survey plan is based on the guidance provided on NUREG/CR-2082, in addition to the criteria discussed for unrestricted release.

PSC will follow the guidance in Regulatory Guide 1.86 for both loose and fixed-surface contamination, adopting NRC's guidance of 5 microR/hr above background. In addition, equipment and materials will be released according to NRC Circular 81-07 and NRC Information Notice 85-92. PSC stated that the effective dose equivalent for an individual will be less that 10 mrem/yr for residual contamination in groundwater and soil. The staff considers these criteria to be reasonable and acceptable.

The staff concludes that the final survey plan meets the requirements of 10 CFR 50.82 (b)(3) and is reasonable and acceptable.

5 UPDATED COST ESTIMATE FOR DECOMMISSIONING

It is the responsibility of the NRC to determine if the cost estimate provided in the PDP provides a reasonable basis for sufficient funding to complete decommisioning of the facility. The review of the cost estimate for decommissioning the FSV facility was based on independent estimates and comparison of several cost activities to be conducted at this facility to similar activities conducted at other facilities. The review included an evaluation of the cost assumptions used, major decommissioning activities and tasks, dismantlement and decontamination costs, volumes of waste to be removed, disposal costs, transportation costs, equipment costs, and labor rates. The basis for the evaluation was similar information provided in the Pathfinder decommissioning cost estimate, the Shoreham decommissioning cost estimate, the "1992 Means Building Construction Cost Data," the "Dodge Manual for Building Construction Cost Data 1984," and in NUREG/CR-0130. All cost information was escalated to 1991 dollars using an inflation rate of 5 percent. The estimated cost of $157,472,700 represents a reasonable estimate of decommissioning the FSV facility.

While FSV is an HTGR, many activities that will be conducted to decontaminate and dismantle this facility are similar to activities conducted at other reactor facilities that are or have been decommissioned. In addition, several activities that support decommissioning are standard construction practices.

The staff reviewed several areas to ensure the estimated cost to DECON the FSV facility are reasonable. For example, the cost of removal of contaminated pumps (1,000-10,000 lbs) was compared to similar activity that was conducted at the Pathfinder facility. The removal of similiar pumps (1,000-1,000 lbs) at Pathfinder, cost approximately $1900.00. The removal of similar pumps for the FSV liquid waste system is estimated to be $3065.00. Even after adjustments for regional differences and inflation, the FSV costs were greater than the estimated cost at Pathfinder. To date, the actual costs for decommissioning at Pathfinder have been consistent with the initial estimate and, therefore, represents an example cost for comparison.

The staff compared the labor rates summarized in Table 3.1-1 of the PDP cost estimate to the labor rates for Pathfinder, which were escalated at 5 percent per year to 1991 dollars, and the city cost indexes in "1992 Means Building Construction Cost Data" and found them reasonable. In addition, the staff compared labor rates for FSV to Shoreham, using the city cost indexes listed in the "1992 Means Building Construction Cost Data" and found them reasonable.

The staff compared PSC's estimated equipment rental costs to the cost for equipment rentals listed in "1992 Means Building Construction Cost Data" and adjusted the costs using the city cost index. It examined the rental costs of many different types and sizes of equipment ranging from small air compressors to 50-70 ton cranes. For example, the cost to rent and run a crane (RT 15-24 T) for PCRV work was estimated to be about $30 per hour for FSV compared to the industry estimate of $100 per hour. After adjustment for regional differences, the FSV cost was considerably less. The estimated cost for rental of a 750-cfm compressor used for blasting and for running tools was $100 for 40 hours of use. Typical industry rate for a similar compressor for a weeks rental (40 hours) was estimated to be $865. Even after adjustment for regional differences, the FSV estimate was less. M. K. Fergerson/Westinghouse stated that equipment costs used in the cost estimate included depreciation costs on company-owned equipment and these costs were considerably less than actual rental costs. If the company is required to rent equipment because they do not own a particular piece of equipment, the additional costs will be taken from the $23 million contingency included in the cost estimate. Estimated operating costs were consistent with industry standards.

The estimated costs of removing many of the BOP systems at FSV were compared to those at Pathfinder. For example, the cost to remove piping up to 5 inches in diameter for the helium purification system is $74.83 per foot compared to the cost or removing 2- to 8-inch piping at Pathfinder of $30.54 per foot. The estimated cost for removing the FSV cooling water system piping, which consisted of over 4 ,000 feet of piping ranging in diameter from 0.5 to 20 inches, averaged $108.79 per foot. The estimated cost to remove piping greater than 8 inches in diameter at Pathfinder was $60.36 per foot. After adjustments for inflation and regional differences, the FSV estimates were considered conservative.

In addition, the staff compared the estimated cost of removing contaminated concrete from the PCRV at FSV to the actual cost of removing contaminated reinforced concrete at Pathfinder. Although the methods of removing the concrete were different, the cost of removal should be similar. PSC estimated the cost for cutting the core support floor at approximately $1120.00 per cubic yard compared to the cost of removing contaminated concrete at Pathfinder of $650.00 per cubic yard. Therefore, the estimate to remove the contaminated concrete at FSV is conservative. The staff also reviewed the cost for disposal of the 100,000 ft^3 of radioactive materials and finds it reasonable on the basis of the proposed contractual agreement with Richland Site, and current disposal costs.

The staff concludes that this cost estimate for decommissioning the FSV facility meets the requirement of 10 CFR 50.82(b)(4).

6 TECHNICAL SPECIFICATIONS IN PLACE DURING DECOMMISSIONING

The staff reviewed PSC's summary of the technical specifications that will be in place for decommissioning. The staff has approved these technical specifications, which are incorporated in Appendix A to the license. These technical specifications address the radiation protection program, as well as many other activities conducted during the decommissioning, such as flooding the PCRV (TS 3.4). The technical specifications address all the activities necessary during decommissioning and meet the requirement of 10 CFR 50.82(b)(5).

7 QUALITY ASSURANCE

PSC's quality assurance program (QAP) is described in Chapter 7 of the PDP and is designed to meet the requirements of Appendix B to 10 CFR Part 50.

The Corporate Vice-President, Nuclear Operations of PSC is the corporate officer responsible for implementation of the QAP. The Vice President has direct access to the President of PSC. The Project Quality Assurance Manager reports directly to the Vice-President. The staff concludes that the Quality Assurance Plan is adequate and meets the requirements of 10 CFR 50.82(b)(5) and the guidelines of SRP Section 17.2.

8 DECOMMISSIONING ACCESS CONTROL PLAN

PSC's access control plan is described in Chapter 8 of the PDP. It is designed to meet the requirements of 10 CFR 20.105 and follows the guidance in NRC Regulatory Guide 1.86. The staff concludes this access control plan is reasonable and acceptable.

9 DECOMMISSIONING EMERGENCY RESPONSE PLAN

PSC's emergency response plan has been reviewed and approved by the NRC staff on March 3, 1992, separately from the PDP.

10 DECOMMISSIONING FIRE PROTECTION PLAN

PSC's fire protection program has been revised and revisions related to decommissioning were approved on March 28, 1992 and June 5, 1992.

11 REFERENCES

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