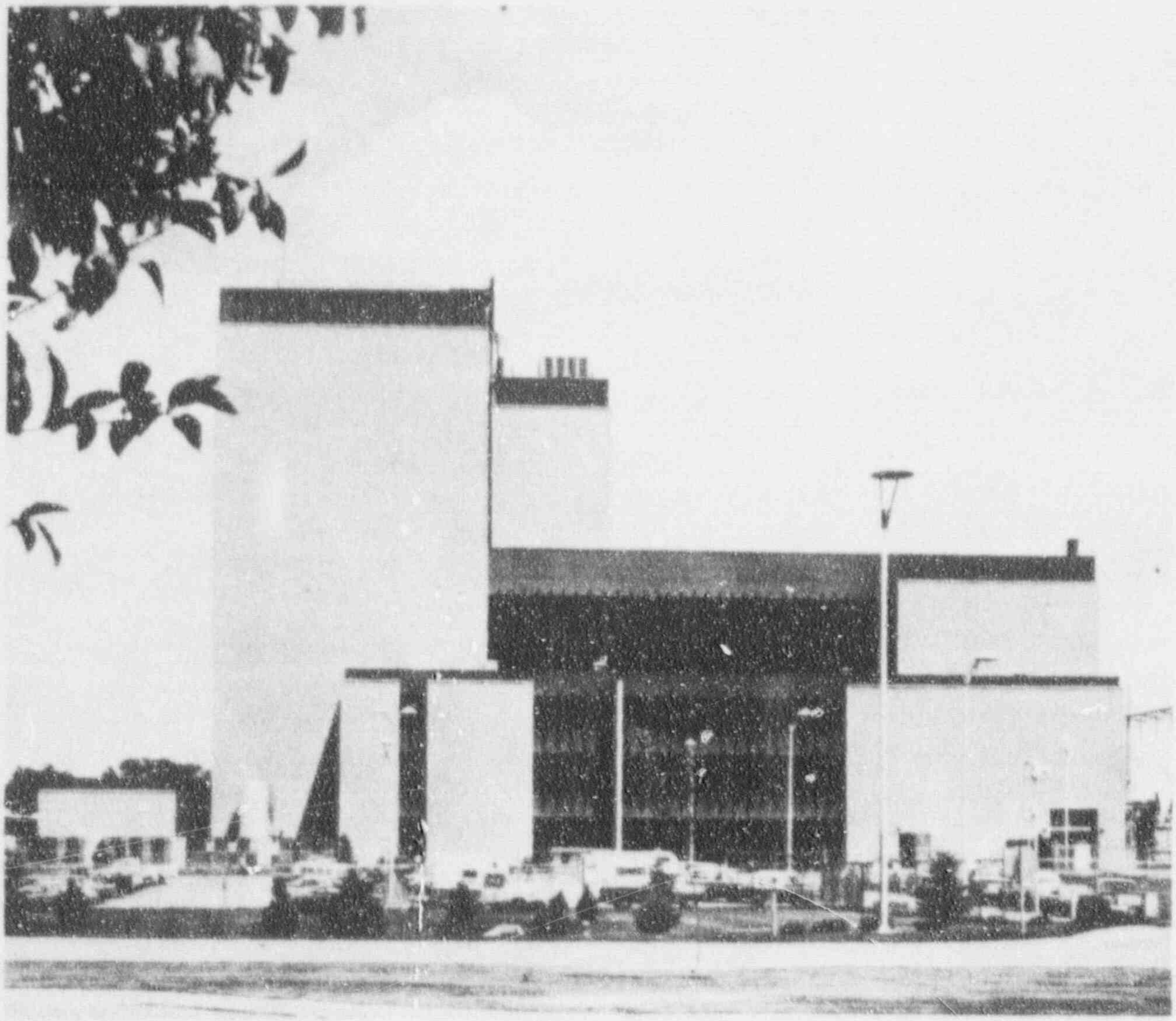

Supplement to Applicant's
Environmental Report - Post Operating
License Stage - Decommissioning

Fort St. Vrain Nuclear Generating Station

Public Service Company of Colorado

Supplement to Applicant's Environmental Report Post Operating License Stage For Proposed Decommissioning of the Fort St. Vrain Nuclear Generating Station

Public Service Company of Colorado

April, 1992

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TABLE OF CONTENTS

TABLE OF CONTENTS (Cont.)

LIST OF TABLES

LIST OF FIGURES

LIST OF FIGURES (Cont.)

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1.0 INTRODUCTION

1.1 Purpose

The Fort St. Vrain (FSV) Nuclear Generating Station is a High Temperature Gas Cooled Reactor facility that was operated by Public Service Company of Colorado (PSC) from 1977 through 1989. Because of technical problems associated with failure of the control rod drives and degradation of the steam generator ring headers, FSV was permanently shutdown on August 18, 1989. Subsequently, a Confirmatory Order was issued by the NRC in May 1990 confirming the permanent shutdown of FSV. A Possession Only License was issued in May 1991. PSC is proceeding with defueling activities, plant closure activities, and planning activities in preparation for decommissioning the facility. Spent fuel shipments were initiated to the DOE graphite fuel storage facility in Idaho, in keeping with PSC's original plans and contract agreements. However, legal challenges resulted in a decision to store spent fuel on-site, in an Independent Spent Fuel Storage Installation (ISFSI) that was licensed, constructed and is being operated in accordance with 10 CFR 72. Interim storage of fuel, top reflector elements, and neutron source elements can be provided at the ISFSI. The ISFSI is located on PSC owner-controlled property, within the historical FSV exclusion area boundary. The ISFSI Environmental Report describes the environmental effects associated with all aspects of the construction and operation of this alternative fuel storage facility (Reference 1).

In preparation for decommissioning, a Proposed Decommissioning Plan (Reference 2) has been developed and submitted to the NRC pursuant to 10 CFR 50.82. PSC is proposing the immediate dismantlement and decommissioning of the nuclear portion of the plant, after the removal of all irradiated fuel from the Reactor Building to a DOE facility or to the ISFSI. This Environmental Report Supplement addresses all actual or potential environmental impacts associated with the proposed decommissioning activities and is responsive to 10 CFR 51.35(b). The level of detail in this Environmental Report Supplement is proportional to the significance of the associated impact.

Decommissioning of Fort St. Vrain is not expected to cause any significant impact to the general public or environment. Decommissioning of Fort St. Vrain has overall positive environmental impacts. The result of decommissioning activities will be termination of the 10 CFR 50 license. Commitment of resources, compared to plant operational resources, is small. The major environmental impact of decommissioning is the commitment of small amounts of land for waste burial at authorized disposal sites in exchange for reuse of the facility and site for other purposes.

The overall environmental impact of decommissioning activities will be small when compared to continued operation of the Fort St. Vrain plant. The following benefits will result from the completion of decommissioning:

- Reduced local traffic (fewer employees, contractors and materials shipments than required to support an operating nuclear power plant).

- Elimination of the radiological sources that create the potential for radiation exposure to site workers and the general public.

- Return of the site to unrestricted use.

1.2 Decommissioning Plan Description

Plans to decommission the nuclear facilities at the Fort St. Vrain plant are as follows:

- Decontamination and dismantlement of the Prestressed Concrete Reactor Vessel (PCRV).

- Decontamination and dismantlement of the contaminated balance of plant systems.

- Site cleanup and final site radiation survey.

As discussed in Section 4.2 of the proposed Decommissioning Plan, PSC has committed to comply with Reg. Guide 1.86 (Reference 3), NUREG 0586 (Reference 4), and interim NRC guidance (Reference 5) when decontaminating the Fort St. Vrain site, to allow release of the site for unrestricted use and eventual termination of the 10 CFR 50 license. All decommissioning activities and schedules are based on decontamination to the above limits.

It is currently planned to use the remainder of the Fort St. Vrain power plant as a natural gas-fired power plant. That action will be the subject of a separate environmental evaluation. There are no plans to again use the Fort St. Vrain site for nuclear power plant operations.

1.3 Radioactive Waste Disposal

Radioactive waste disposal will follow the requirements established in 10 CFR 20 and 10 CFR 61, the disposal site criteria, and other applicable Federal and State regulations. The waste will be classified, packaged according to approved procedures, manifested, and shipped to a disposal facility under an approved QA/QC program.

As radioactive waste is generated, it will be processed, packaged, and stored onsite until a convenient or efficient transportation limit (i.e., weight, volume, curie content) is reached.

The Classes A, B, and C waste will be packaged, stabilized if required, and shipped in accordance with relevant State and Federal regulations and burial site criteria. The Greater than Class C (GTCC) waste, if any, will be packaged for interim storage and retained onsite until such time as it can be transported to a DOE facility, such as Hanford, or as mandated by the amended license.

The steam generator primary assemblies are planned to be shipped in special shielded shipping containers by rail to a licensed disposal facility for burial. All other radioactive waste shipments will be by truck.

1.4 Regulatory Considerations

Decommissioning of nuclear facilities is a regulated process whereby equipment, structures and portions of the facility that contain radioactive material are removed and NRC licenses are terminated. The voluntary termination of an operating license and the subsequent dismantlement of the facility requires NRC approval as specified in 10 CFR 50.82. Prior to the initiation of the dismantlement of radioactive components of the facility, a Decommissioning Plan must be submitted and approved. Pursuant to 10 CFR 50.82, PSC has prepared and submitted a Proposed Decommissioning Plan (PDP, Reference 2) to the NRC for review and approval. The Decommissioning Plan describes an organized means for removing all radioactive components and all radioactivity within the reactor facility. The PDP also describes how PSC will continue to protect the health and safety of the public and environment during dismantlement activities.

The current requirements with regard to occupational or public doses or effluents to the environment continue to apply throughout the decommissioning period until the part 50 license is terminated by the Commission. The decommissioning planning requirements are considered appropriate means of assuring that radiation exposures to both plant personnel and the public will be maintained as far below the 10 CFR Part 20 limits as is reasonably achievable. The transportation of decommissioning wastes will not involve any additional technical considerations beyond those for transportation of existing radioactive material. The shipment and disposal of all radioactive wastes will be governed by the existing NRC regulations in 10 CFR parts 20, 61, 71 and 73 and appropriate Department of Transportation regulations in 49 CFR. A Quality Assurance program will be applied to decommissioning activities to ensure compliance with all applicable regulations and the approved Decommissioning Plan.

All activities involving asbestos will be conducted in accordance with Occupational Safety and Health Administration regulations 29 CFR Parts 1910 and 1926 and Environmental Protection Agency regulations 40 CFR Part 61, Subpart M.

1.5 Environmental Technical Specifications

The requirements of the Radiological Effluent Technical Specifications (RETS) and the Radiological Environmental Monitoring Program (REMP) have historically been provided in the FSV Technical Specifications. For decommissioning activities, PSC submitted proposed Decommissioning Technical Specifications (Reference 6) that will totally supersede the historical FSV Technical Specifications. Consistent with the guidance provided in Generic Letter 89-01 (Reference 7), the FSV Decommissioning Technical Specifications will contain the programmatic controls and reporting requirements for the RETS and REMP. Associated procedural details, such as the methodology for calculating offsite doses due to radioactive gaseous and liquid effluents and monitoring instrumentation operability and surveillance requirements, will be included in the Offsite Dose Calculation Manual (ODCM). Procedural details regarding the processing, packaging and disposal of solid radioactive waste will be included in the Process Control Program (PCP).

1.6 Equipment Salvage

Each piece of equipment or material removed during the decommissioning process will be assumed to be contaminated until determined otherwise. Equipment found by radiation surveys to be suitable for release for unrestricted use will be released (References 8 and 9). All material contaminated with radioactivity will be disposed of in accordance with NRC regulatory requirements.

1.7 Need for Proposed Action

Decommissioning of the Fort St. Vrain Nuclear Generating Station is a regulatory requirement under 10 CFR 50.82, which allows a licensee to choose one of the three available decommissioning alternatives: DECON, SAFSTOR or ENTOMB. Of these alternatives, PSC has selected DECON (immediate dismantlement) as the basis for its Proposed Decommission Plan. Further discussion of the decommissioning alternatives and basis for selection of the DECON alternative is provided in Sections 6 and 7 of this report.

1.8 References

1. Independent Spent Fuel Storage Installation (ISFSI) Environmental Report, Rev. 0, Public Service of Colorado, 1990.

2. Proposed Decommissioning Plan for the Fort St. Vrain Nuclear Generating Station, Public Service of Colorado, April 17, 1992.

3. USAEC Regulatory Guide 1.86, "Termination of Operating License for Nuclear Reactors," June 1974.

4. Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, NUREG 0586, August 1988.

5. Residual Radioactive Contamination from Decommissioning NUREG/CR-5512 Draft Report, January 1990.

6. PSC letter, Crawford to Weiss, dated March 19, 1992, (P-92115); Subject: Decommissioning Technical Specifications.

7. "Implementation of Programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or the Process Control Program," NRC Generic Letter 89-01, January 21, 1989.

8. NRC Circular 81-07, "Control of Radioactively Contaminated Materials."

9. NRC Information Notice 85-92, December 1985, "Surveys of Wastes Before Disposal From Nuclear Reactor Facilities."

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2.0 HISTORY AND CURRENT STATUS OF FACILITY

2.1 Nuclear Operating History

Fort St. Vrain is a High Temperature Gas Cooled Reactor (HTGR) facility owned and operated by Public Service Company of Colorado (PSC). Fort St. Vrain's location is approximately 35 miles north of Denver and three and one half miles northwest of the center of the town of Platteville in Weld County, Colorado. The site consists of 2798 acres owned by the Licensee.

Construction of Fort St. Vrain was authorized by the Atomic Energy Commission (AEC) by issuance to PSC of a provisional construction permit on September 17, 1968, in AEC Docket No. 50-267. Table 2.1-1 provides a brief summary of plant milestones and major events.

In 1968, the Colorado Public Utilities Commission (CPUC) issued a certificate of public convenience and necessity to build Fort St. Vrain. However, in its order, the CPUC stated that the authority to build a nuclear plant rather than a fossil-fueled plant was "subject to the condition that the Commission may disallow portions of investment and operating expenses which are due to the fact that the plant is a nuclear powered plant rather than a fossil fuel powered plant, if the allowance of such portions of investment and operating expenses would adversely affect the ratepayer."

Fort St. Vrain was initially scheduled for commercial operation in 1972. Although PSC received a full power operating license in 1973, extensive pre-operational testing mandated by the NRC and resulting engineering modifications delayed the commercial operation of the plant until 1979. General Atomics (GA), the prime contractor for Fort St. Vrain, reimbursed PSC for all increases in electric operating expenses incurred by PSC due to this delay.

On June 27, 1979, PSC and GA settled all contracts and claims between them relating to Fort St. Vrain by entering into a Settlement Agreement and associated agreements. Pursuant to the GA Settlement Agreement, PSC accepted Fort St. Vrain for commercial operation at a reduced capacity of 200 Mwe (at 60% capacity factor) instead of 330 Mwe (at 80% capacity factor), as originally designed. GA paid PSC $60 million as an adjustment to the cost of the plant to reflect the 130 Mwe reduction in capacity. In the period 1980 through 1984, GA paid PSC approximately $97.1 million to compensate PSC for the cost of replacing the 130 Mwe reduction in capacity with other generating facilities.

Fort St. Vrain was first included in PSC's rate base in a general rate case in December 1980. In that rate decision, the CPUC allowed PSC to collect approximately $39 million of revenues (an amount subsequently increased to $46 million) to cover operating expenses and provide a return on investment.

As illustrated in Figure 2.1-1, Fort St. Vrain's record of operation after initial criticality has been inconsistent, with a historical capacity factor of less than 15%. This inconsistent record of operation was the result of technical problems and implementation of regulatory requirements, including:

- core thermal and neutron flux oscillations
- reactor vessel moisture ingress problems
- multiple control rod drive failures to automatically scram
- implementation of an environmental qualification program
- helium circulator material failures
- major turbine building fire damage
- inadequate original design analyses (which limited maximum capacity to 82% power)
- recent technical problems related to control rod drive material failures and cracking of steam generator steam outlet piping

In response to Fort St. Vrain's historically reduced levels of generation, the Colorado Public Utilities Commission (CPUC) instituted penalties against PSC to reduce the revenues recovered from its customers. The Office of Consumer Counsel (OCC) filed a complaint with the CPUC against PSC alleging that in light of its operating history, Fort St. Vrain was not "used and useful" in rendering a utility service. In view of the various legal and administrative proceedings regarding Fort St. Vrain, PSC entered into a Stipulation and Settlements Agreement in September 1986 with the CPUC, OCC and other parties. Significant provisions of the 1986 Settlement Agreement included: (1) removal of Fort St. Vrain from the rate base; (2) a provision for the sale of future energy produced at Fort St. Vrain to PSC customers at a rate of 4.8 cents per Kwh; and (3) recovery over 5 years of $11.5 million of decommissioning costs. This effectively made Fort St. Vrain an independent power producer, with the associated risks of operation assumed solely by the PSC shareholders.

As a result of its unfavorable plant operating performance, Fort St. Vrain did not produce revenues adequate to offset expenses during 1987 - 1989. Shortfalls of approximately $24.5 million (1987), $35.6 million (1988) and $30.1 million (1989) were recorded in unrecoverable operating and capital expenditures.

The latest annual budget while operating (1989) was $77.9 million, which included all O&M costs, as well as capital improvement expenses required to meet regulatory requirements and NRC commitments. Assuming a maximum power limit of 82% (based on safe shutdown reanalyses), Fort St. Vrain would have to operate at a capacity factor of 68% to break even based on the 4.8 cents per Kwh allowed by the 1986 Settlement Agreement. In order to purchase new fuel (annual expense - $26 million) for continued future operations, a capacity factor of greater than 90% was required. To cover eventual defueling and decommissioning expenses, the required capacity factor exceeded 100%. Reanalyses of the safe shutdown limits to allow full power operation (330 Mwe) would still require a capacity factor in excess of 85% in order to break even. Without third-party support, Fort St. Vrain was unable to generate enough electricity to support annual expenses. Even if the known technical problems were resolved, the benefits of plant operation do not justify this use of resources.

In August 1989, the plant was shutdown due to control rod problems. During subsequent inspections, significant cracking was discovered in the steam generator main steam outlet piping assemblies. Due to these problems, as well as additional mechanical and financial concerns, the PSC Board of Directors decided not to restart the plant. This announcement was made August 29, 1989.

On May 21, 1991, the NRC formally acknowledged the permanent shutdown condition of the plant by issuing a Possession Only License, Amendment No. 82 to Fort St. Vrain's Facility Operating License.

2.2 Radionuclide Inventory

2.2.1 Activated Components Within the PCRV

An activation analysis was performed for the PCRV and associated internal components. Calculational details of the activation analysis are included in Appendix II of Reference 1 (the activation analysis is also identified as Reference 2), and in Reference 3. The analysis was performed to estimate the isotopic composition, magnitude and extent of residual radioactivity which could be present in the PCRV after the end of operations. The actual operating history of the plant was used in the analysis by considering the total effective full power days (890 EFPD) generated by the plant until its shutdown in August 1989. The analysis consisted of three sections: (1) neutron flux estimates in the PCRV; (2) activation analysis of the PCRV and internal components; and (3) calculation of gamma dose rates (in air) inside the PCRV due to non-removable (fixed) components.

2.2.1.1 Computer Codes

The activation analysis required the use of several computer codes and various input data libraries. The ANISN code (Reference 4) was used to determine the neutron flux throughout the reactor core and outward through the reflectors, helium flow paths, insulation, PCRV liner and PCRV concrete. The activation of selected components within the PCRV was then determined using the REBATE computer code (Reference 5). Finally, gamma doses (in air) within the PCRV were calculated using the REBATE, ANISN and other data manipulation codes.

2.2.1.2 Material Compositions

Material compositions of components were determined from a variety of sources. In most cases, material compositions were identified from component drawings which referenced standard material specifications. Assumptions for the number densities of trace elements, such as europium (Eu), cobalt (Co), and niobium (Nb), were based on design manuals, previous analytical investigations and NUREG/CR-3474, "Long Lived Activation Products in Reactor Materials" (Reference 6). In general, the reactor internals are made of carbon steel, graphite or concrete, with a few major components made from stainless steel or Inconel. Details of the actual compositions and trace element assumptions are found in Reference 2.

2.2.1.3 Activation Analysis Results

The results of the activation analysis are summarized in Table 2.2-1. The nuclides of importance as well as the total estimated radionuclide inventory for activated components inside the PCR are listed in this table. Detailed results can be found in References 2 and 3.

The dominant nuclides for metallic components are Fe-55, Co-60, Ni-63 and Mn-54. The dominant gamma emitter in the stainless steel components was determined to be Co-60, although Nb-94 is also present. Due to the high concentrations of Co-60 in the boronated spacer blocks, these components are the primary dose contributors inside the PCRV.

The activity in graphite components is dominated by tritium (H-3) and Fe-55, which were generated due to impurities in the graphite. Due to the large volume of graphite and the curie content of H-3 and Fe-55, these components are the largest contributors to the overall radionuclide inventory. No credit was taken for the migration of H-3 out of the graphite.

The Kaowool insulation and silica blocks were determined to have fairly low activities. The carbon steel cover plate contains almost all the activity in the Kaowool/cover plate assemblies. The silica block activity is dominated by Fe-55.

The PCRV concrete/rebar matrix contains many activation products due to the presence of trace elements. In the short term, Co-60 is the dominant gamma emitter, while Eu-152 and Eu-154 are the dominant long term gamma emitters. The nuclide contributing most to the total activity is Fe-55. Other nuclides present in lower activities were: Cs-134, Ca-45, Ag-110m, H-3, C-14, Fe-59, Ni-59, Ni-63, Nb-94, Mn-54, and Ca-41.

As indicated in Table 2.2-1, the majority of the activity in the concrete is contained in the inner first 1.5 feet in all directions. Table 2.2-2 indicates the estimated required amount of concrete which must be removed to achieve the recommended release limit for unrestricted use (5 μR/hr above background). Table 2.2-2 lists dose rate estimates for each direction within the PCRV. The total dose inside the center of the PCRV (in air) is the sum of the dose rate for all three directions. Table 2.2-2 also indicates the estimated dose rate contribution (in air) for various stages of component removal for an individual located at the center of the PCRV.

2.2.1.4 Activation Analysis Verification

The thermal neutron flux and trace element abundances are considered to be the two most important potential sources of uncertainty in the activation analysis. A verification program was performed to validate the assumptions in the activation analysis and to provide additional confidence that the activation analysis results provide reasonable predictions of component activities and resulting dose rates. The results of these verification efforts are discussed below.

Neutron Flux Verification

The validation of the neutron flux was assessed by comparison of predicted specific activities to measured activities for specimens of known composition. Wire and Charpy specimens were removed from the PCRV top head to verify thermal neutron flux predictions from the ANISN (Reference 4) calculations and to assess the conservatisms of cobalt impurity assumptions used in carbon steel. PCRV tendon wires from a vertical PCRV tendon located 8 inches beyond the predicted concrete removal depth were also removed, analyzed, and the results compared to analytical activity predictions.

The activities of wire specimens in the top head agreed within 3% of their predicted activities, indicating that the estimates of the thermal flux in the PCRV top head area are quite reasonable. Activities of the Charpy specimens were approximately half the value predicted, again indicating that the flux predictions are reasonable and the assumptions of cobalt levels were conservative.

The results of tendon wire analysis indicate that the thermal flux prediction (based on the Co-60 activity at core midplane) at the tendon location was low by a factor of about 2.8. However, this is considered to be in reasonable agreement, considering the distance from the core and the number of mean free paths (about 12) traveled by the neutrons through the concrete before reaching the tendon wire.

Material Composition Verification

To validate the trace element assumptions in the activation analysis, surface samples of the PCRV radial concrete were taken and analyzed to determine the actual trace element abundances in the PCRV concrete. Six samples were taken at the same elevation as the active core. Analysis indicates that there is no significant variation of the trace element abundances among the samples and provides support for the assumption that PCRV concrete trace element abundances do not vary widely throughout the different mixes of concrete. The trace element abundances in the PCRV concrete all fall within or below the ranges listed in NUREG/CR-3474 (Reference 6). It is therefore reasonable to assume that the results of the surface sample analysis, when combined with the sensitivity analysis and NUREG/CR-3474 data, provide a high level of confidence in the activation analysis results.

Conclusion

The data obtained from the verification efforts and the results of the recalculated activation analysis provide strong evidence that the predictions from the original activation analysis are reasonable. Additionally, a comparison with the results from the NRC's radionuclide characterization of decommissioning wastes in NUREG/CR-5343 (Reference 7) indicates that the accuracy of the results are acceptable. Based upon the measured data and comparisons with analytical predictions, the calculational methods and material composition assumptions for the concrete provides reliable estimates of the activation products within the PCRV. Additional details of the verification effort can be found in Reference 8.

2.2.2 Plateout Analysis for PCRV Internal Components

2.2.2.1 Basis of Computer Code Analysis

A plateout distribution analysis of radioactive nuclides produced in the reactor core was performed for the PCRV and internal components (Reference 9). The purpose of this analysis was to estimate the plateout concentrations and distributions in the primary coolant circuit. Analyses were conservatively performed from the Beginning Of Cycle (BOC) 1 to the end of operations after 890 EFPD. The axial and radial core power distributions were calculated and used with flux distribution data as input to fission product release codes. Full-core fuel and graphite temperature distributions, fuel failure and release of key fission gases and metals were then calculated. Based on the full-core analysis for key fission gases and metals, the total plateout and helium purification system inventories of radioactive nuclides were estimated.

Plateout distributions were calculated using the PADLOC computer code (Reference 10). The PADLOC code performs a mass transfer calculation using mass transfer correlations and sorption isotherms to determine the partitioning of condensable radionuclides between the flowing coolant and the fixed surface in a recirculation loop. The plateout model in PADLOC is limited to one-dimensional cylindrical geometry, such that all components of the primary circuit must be modeled as an equivalent series of coupled sections of parallel banks of cylindrical tubes. Reference sorption isotherms were used to describe the sorptive capacity of the primary circuit materials for the radionuclides of concern.

2.2.2.2 Plateout Analysis Methodology

Typically, the two dominant sources of fission products released from the core are heavy metal contamination (heavy metal outside the coated fuel particles) and fuel particles whose coatings fail in service. In addition, the volatile metals (Cs and Sr) can, at sufficiently high temperatures and over long periods of time, diffuse through the silicon carbide (SiC) coatings and be released from the intact fuel particles.

Calculations were performed to predict plateout distributions in the primary coolant circuit for the following key nuclides: Sr-90, I-129, I-131, Cs-137, Cs-134 and Te-127m. The source terms for fission product plateout analysis include both a direct release contribution and, where applicable, a precursor contribution. In the case of the cesium isotopes, there is a direct release of both Cs-137 and Cs-134 metal from the core. Cs-137 plateout also results from the release and subsequent decay of its precursor contributor, Xe-137. Cs-134 has no gaseous precursor. Similarly for Sr-90, there is a direct Sr-90 metal release as well as the contribution from its Kr-90 precursor. Only direct release contributions are considered for I-129, I-131 and Te-127m. The plateout analysis included a comparison of the calculated and measured plateout inventories for Cs-134, Cs-137 and Sr-90 based on measurements on the plateout probe that was removed from the PCRV after final reactor shutdown.

2.2.2.3 Plateout Analysis Results

It is anticipated that any internal PCRV component that had come in contact with primary coolant is contaminated and will be removed for disposal as radioactive waste. This includes not only the core graphite and structural components (which are also activated), but also the steam generator modules, helium circulators and Kaowool insulation. The preliminary results of the plateout analysis are shown in Tables 2.2-3 and 2.2-4. Table 2.2-3 lists the plateout concentration (Ci/cm^2) on primary circuit components for the key nuclides, Cs-137 and Sr-90. Table 2.2-4 identifies the integrated plateout (Ci) of primary circuit components for the following nuclides: Cs-134, Cs-137, I-131, I-129, Sr-90 and Te-127m. Additional information on the analysis results, analytical models and comparisons with measured data are located in Reference 7.

2.2.3 <u>Contaminated Systems, Structures and Components</u>

An engineering analysis of the total curie inventory at Fort St. Vrain was completed in June 1989 and the results of this analysis have been summarized in Table 2.2-5. This analysis is based upon past survey results, activation analysis, plateout analysis and general estimation of contamination levels occurring in the various systems. The survey results and estimation of contamination levels were then applied over the estimated surface area of the associated system. This analysis accounts for all expected radioactivity at Fort St. Vrain with the exception of fuel.

Section 2.3 contains a detailed summary of the radiation survey results. These surveys were performed to identify general radiation and contamination levels in frequently accessed areas of the facility. More detailed surveys of individual areas will be required when determining specific work plans during actual decommissioning.

2.2.4 Tritium Analysis

During the Fort St. Vrain 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. One of the radionuclides of primary concern is tritium, since a fraction of the tritium inventory is expected to leach out of the graphite blocks into the water and the 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 depends 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.

Measured data on the actual tritium concentrations that are in the Fort St. Vrain PCRV graphite components and the rate at which the tritium leaches into the water from the Fort St. Vrain graphite do not exist. Therefore, the amount of tritium that enters the PCRV water has been estimated, based on (1) a conservative calculation of the total amount of tritium produced during power operation (i.e. 100,000 curies) and (2) actual measurements of tritium leach rates from British Magnox reactor graphite. It is estimated that approximately 500 Curies (or 0.5% of the total tritium inventory) will enter the water. The PCRV Shield Water System is being designed to discharge this tritium inventory using the existing liquid effluent discharge path.

An assessment has also been made of the impact if the maximum theoretical amount of tritium (100,000 curies) is released into the PCRV shield water. Included are impacts on air handling, tritiated water disposal, contamination, and personnel protection. It was found that these impacts, although significant, can be managed without undue safety hazards and within reasonable costs. Allowing for this extreme case, decommissioning can 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 decommissioning will also be accomplished without undue risk to the health and safety of the public.

2.2.4.1 Expected Tritium Release Into the PCRV Shield Water System

Data on tritium leaching from graphite obtained by the British (Reference 11) is considered to be directly relevant to determine the fraction of the tritium inventory likely to be leached from the Fort St. Vrain graphite after the PCRV is flooded. These British measurements were made in support of decommissioning of the Magnox and AGR plants, and form the basis for disposal planning for irradiated graphite for the European Community. The graphite in the British tests is typical of that used in the Magnox and AGR reactors. Key parameters for the British graphite test samples are included in Table 2.2-6 for comparison with data for Fort St. Vrain graphites.

The leach rate of the tritium in the British test was measured to decrease with time starting at about 0.1% per day and declining to below 0.0001% per day after several months. Applying these values to Fort St. Vrain, a curve of tritium release rate versus time was prepared with an initial tritium release rate of 0.5% 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 Curies from the graphite to the water, based on an assumed initial tritium inventory of 100,000 Curies in the core graphites. Thereafter, the tritium release rate from the graphite is assumed to continue to decrease, falling to a release rate of less than 0.0001% per day within several months, consistent with the results of the British test.

2.2.4.2 Conclusions Regarding Expected Tritium Leaching

Based on a comparison of key properties and operational history between the British graphite test samples and the HLM-type graphite used at Fort St. Vrain, it is concluded that the British Magnox test results are conservative with respect to tritium leach rate and the fractional release of tritium. HLM surface-to-volume ratios are significantly lower, indicating that water ingress will not occur as rapidly and tritium migration to the graphite surface will take significantly longer. Irradiated densities of the HLM are greater than the British graphite samples, indicating lower porosity and a lower leach rate in the HLM graphite due to density. Effects of both reactor power history and primary coolant favor the HLM graphite, since the effect on increased porosity should be greater in the British samples than in the HLM.

Therefore, the leach rate for the HLM graphite is not expected to be greater than that determined for the British Magnox graphite samples, and use of the leach rate determined by the British test in demineralized water should represent a conservative upper bound on the leach rate that should be experienced when the PCRV is flooded and the HLM graphite is immersed in water.

2.2.5 Initial Site Characterization Program

The initial site radiological characterization program was performed to determine the radiological status of Fort St. Vrain balance of plant systems, auxiliary systems, buildings and site. Radiological measurements for direct radiation and residual contamination (fixed and removable) were conducted and recorded. Biased and unbiased samples (i.e., sample locations that are or are not influenced by previous data or historical information) were taken, analyzed and recorded.

The results of the Fort St. Vrain Initial Site Characterization Program were reported to the NRC in Reference 12. The purpose of this program was to collect and analyze radiological survey data to determine the extent of decontamination and dismantlement activities. In accordance with this program, more than twenty thousand measurements were made on more than five thousand survey locations. Reference 12 identifies the areas examined and summarizes the results of these examinations.

2.2.5.1 Results of Site Characterization Surveys

The Initial Site Characterization Program divided the Fort St. Vrain facility into four major elements, identified below, with the results of the characterization in each area.

1. Structural Characterization

The results of the structural characterization indicate that the only structures with contamination levels above release limits are the Reactor Building and the Radwaste Compactor Building. Structures with contamination levels above background were the Reactor Building, Turbine Building, Helium Storage Building, New Fuel Storage Building, Radiochemistry Laboratory, and Radwaste Compactor Building. The loose contamination was composed primarily of Co-60. Alpha contamination was not detected in any structures.

2. BOP and Equipment Characterization

The systems characterization determined that twenty plant systems had no radioactivity above background and 17 systems contained various levels of contamination. System 79 (Radiochemistry Laboratory Ventilation) was not verified as contaminated at the time of the characterization survey. However, it is frequently used to vent radioactive materials and is treated as a contaminated system when conducting filter changes and maintenance.

3. PCRV and Internals

The PCRV was inaccessible for characterization. Historical data, calculations, and limited measurements support the general dose rates provided in the activation analysis results. Additional measurements will be made during actual decommissioning.

4. Environmental

The characterization used historical data from the Radiological Environmental Monitoring Program, and assessed soil and water samples and direct measurements.

2.2.5.2 Conclusions of Site Characterization Program

Some of the more noteworthy features of Reference 12 are the following:

- The only structures with contamination levels that will require decontamination are the Reactor Building and the Radioactive Waste Compactor Building.
- Only a few of the PCRV internal components were available for sampling, since this area is generally inaccessible. The measured results from this limited sampling generally agree with the activation analysis provided in Reference 2. Based on these measurements, PCRV internal dose rates are not expected to differ substantially from those previously estimated.
- The analysis of plant systems showed general agreement with the extent of contamination previously expected, although minor contamination was found in several systems that were previously thought to be less than the release criteria. Any additional decontamination or disposal activities can be performed within the cost estimate contingency funds.

The Fort St. Vrain Radiological Site Characterization Program will be an ongoing program throughout the dismantlement process. The results of the site characterization will assist in the determination of final survey plans, frequency of surveys and instrumentation to be used. It will also be used as a general performance indicator to assess the effectiveness of the overall site decontamination. The data will be utilized for radioactive waste management, assessing potential hazards during the decontamination and decommissioning work, for determining safety controls, and accurately scheduling the decommissioning activities.

2.3 Radiation Survey Results

In August 1990, radiation and contamination surveys were performed in the Reactor and Turbine Buildings. These surveys focused on identifying the major contributors to radiation levels above background and areas containing both fixed and loose surface contamination.

Historical radiological surveys have shown that alpha contamination (both fixed and loose surface) is not present above natural background levels at Fort St. Vrain. Surveys for alpha contamination are performed on a routine basis to confirm this.

Generally, the results of these surveys demonstrated that greater than 95% of the plant areas have radiation levels corresponding to natural background (in the 0.004 to 0.032 mrem/hr range as determined from historical surveys). In the results summarized in Table 2.3-1, only those areas with radiation levels above background are noted.

Additionally, fixed contamination levels are generally less than 1000 dpm/15 cm^2 and loose surface contamination levels less than 1000 dpm/100 cm^2. Most loose survey results are less than 100 dpm/100 cm_2. In some locations, tritium may be present as fixed contamination. Due to the low energy beta activity emitted by tritium (Eavg = 0.005 MeV), normal survey methods will not detect the tritium and therefore actual tritium levels were not considered in this survey. Fixed contamination is typically imbedded within the first few centimeters of concrete surfaces.

Figures 2.3-1 to 2.3-19 provide specific results of these area radiation surveys. Table 2.3-1 provides a summary of the survey results with a description of the major contributors to the radiation levels. Reactor and Turbine Building elevations are shown in Figure 2.3-20. Where results are not listed, contamination and/or radiation levels are not greater than background levels. Systems which are potentially contaminated are identified in Table 2.3-1 by system number for each elevation on which they are located.

2.3.1 Turbine Building Survey Results

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 in all locations and generally less than 100 dpm/100 cm^2. Piping from the potentially internally contaminated Systems 11 (PCRV and Internal Components) and 73 (Reactor Building Ventilation) extends from Level 7 (El. 4829') to the roof of the Reactor Building.

2.3.2 Radiation Sources Outside the Reactor and Turbine Buildings

Radioactive materials are stored on a temporary basis inside Sea-Vans, cargo trailers and designated radwaste areas. Varying amounts of radioactive materials may be stored in these trailers, but external radiation levels are typically less than 0.2 mrem/hr.

The only contaminated area outside the Reactor and Turbine Buildings is the Compactor Building directly east of the main cooling tower (see Figure 2.3-1). General area radiation levels vary from 0.2 to 0.5 mrem/hr primarily due to residual contamination inside a radioactive waste compactor. Loose surface contamination levels are generally less than 100 dpm/100 cm^2. The compactor contains loose surface contamination of 50,000 dpm/100 cm^2 and fixed contamination levels of 50,000 dpm/15 cm^2. There are two concrete bunkers in the Compactor Building which have loose surface contamination levels of 5,000 dpm/100 cm^2 and fixed contamination levels in the first few centimeters of the concrete averaging approximately 20,000 dpm/100 cm^2. The presence of tritium is also suspected in the fixed contamination of the bunkers. This building is also used for staging of radioactive wastes and materials.

Piping associated with the Radioactive Liquid Waste System (System 62) also runs underground from the exit point from the Reactor Building to the main cooling tower blowdown line. Sample results of oil collected in an associated oil separator have occasionally shown trace amounts of tritium, Co-60, Cs-137 and Cs-134.

Routine surveys do not indicate any radiation or contamination levels above background in the Radiochemistry Laboratory located in the Technical Support Building (See Figure 2.3-2), although small amounts of radioactivity are expected in drain piping from this facility to the Radioactive Liquid Waste System (System 62).

2.3.3 Current Environmental Radiological Status

2.3.3.1 Beta-Gamma Radiation in Surrounding Environs

The environmental radiological status of the site and surrounding areas has been monitored during the entire pre-operational, operational, and post-operational phases of the plant through the Radiological Environmental Monitoring Program (REMP). This program includes surveillances in surrounding areas to gather environmental data in the following areas: external gamma activity levels, air sampling data, water sampling data, milk data, aquatic pathways, and food products. Sample locations are situated near the site boundaries and in outlying areas. Details of the results of these surveillances can be found in Reference 13 and in past REMP reports, which are provided annually to the NRC.

During the spring and summer of 1990, additional data were taken to further characterize the site. Soil samples were taken inside and outside the protected area, gamma radiation surveys were performed inside the protected area, and downwind air samples were taken with respect to the predominant wind direction (from the NE). Environmental radiation surveillance data from all past REMP reports and the recent characterization data indicate that the predominant source terms found above natural background levels are due to Chernobyl and past nuclear weapons test fallout. External radiation sources to area residents are due to naturally-occurring background radiation and atmospheric fallout.

The recent characterization data included the exposure rate from gamma-ray emitting radionuclides and were measured using thermoluminescent dosimeters (TLD). The TLD stations were constructed at 72 different locations inside and outside the controlled area boundary. Each station contained packets with two chips of CaF_2(Dy), which are identical to those used in the REMP. The measurement period for the TLDs was 92 days. The mean of the two chips in each station was used to determine the mean exposure rate. The overall mean exposure rate from the TLD packages was 0.32 mrem/day. This value is not statistically different from the mean value found in the 1989 REMP report (Reference 13) of 0.38 mrem/day for the Fort St. Vrain facility area. Reference 13 indicated that since the inception of power production by the reactor, there has been no detectable increase in the external exposure rate due to planned or unplanned reactor releases.

The concentrations of gross beta activity due to the combination of naturally occurring radionuclides and fission product radionuclides were determined from air samples at two locations downwind from the predominant wind direction. A particulate filter for gross beta analysis and an activated charcoal cartridge for I-131 or noble gas radionuclide analyses were in the sample line. Tritium in atmospheric water vapor was collected passively by silica gel at each of these locations. Sampling methodology was identical to that utilized in the REMP. Fort St. Vrain operational Technical Specifications no longer require measurement of gross alpha activity. Gross beta activity measured in air particulates was principally due to naturally occurring radionuclides or from soil resuspension. The mean weekly activity air concentrations measured at the northern and southern monitors were 16 femtoCuries (16 E-15 Curies)/m^3. These concentrations are comparable to those found in the REMP program. Past REMP data has shown that there has never been a significant difference observed between facility and reference sites (Reference 13). It is concluded, therefore, that based on the current radiological data and past REMP data, the reactor air effluents of particulate fission products or activation products are not a source of dose commitment for the Fort St. Vrain environs population.

2.3.3.2 Soil Samples

Soil samples were taken at 124 locations inside and outside the controlled area. Samples were taken at each location from a depth of ten centimeters and an area of 95 square centimeters. Two samples were taken at each location to produce a sample size sufficient to fill a one quart volume. Samples were dried, ground to a constant density, and sealed in the quart container. After a three week period, each container was counted using Ge(Li) gamma-ray spectroscopy to determine the activity concentration of important fission products, activation products and naturally occurring radionuclides. No beta analysis was performed.

Deep core samples (taken at approximately 12 percent of the soil sample locations) were taken to approximately 150 centimeters in depth. The core samples were collected in polyethylene tubes, which were frozen and sectioned to obtain samples at various depths. The deep core samples were analyzed using the same techniques as the soil samples.

Results of the soil samples indicate the presence of statistically significant Cs-137 concentrations. These concentrations are due to world-wide fallout remaining from the United States, USSR and Chinese nuclear weapons tests, and the Chernobyl accident. This is supported by the fact that the Cs-137 concentrations are the same in the entire front range of Colorado, and other reactor-generated fission products or activation products were not present in the samples.

2.3.3.3 Results of REMP Surveillances

Tritium is the only radionuclide that was detected in concentrations above background in any effluent pathways that could be attributed to reactor operation. Since tritium is released as tritiated water, the dilution by the surrounding hydrosphere is significant. Elevated levels of tritiated water (References 13, 14) were detected in downstream surface water samples on occasion, but the yearly mean values of downstream surface water were not statistically greater than upstream concentrations. Tritium concentrations measured in milk were all less than the lower limit of detection (LLD). However, slight increases in the downstream tritium levels, which were discussed in the 1986 Annual Radiological Environmental Monitoring Program (REMP) report, showed that the radiation dose commitment that can be calculated as a result of the increases was found to be negligible as compared to natural background radiation dose rates.

The REMP program over the years has been shown to be of adequate scope and sensitivity to detect any accidental releases from Fort St. Vrain operation. It is concluded that the dose commitments calculated for the closest inhabitants or other parts of the nearby ecosystems due to reactor operations are negligible. In addition to the REMP data, the most recent characterization data both inside and outside the controlled area boundary supports this conclusion. The negligible release of radioactivity from Fort St. Vrain is due to its unique gas cooled design.

2.4 Current Facility Status

The Fort St. Vrain physical facility is described in Section 2.1 of the Proposed Decommissioning Plan (Reference 1). The reactor core is in the process of being defueled, and all nuclear fuel is expected to be removed from the Fort St. Vrain Protected Area by mid-summer 1992.

The Proposed Decommissioning Plan (Reference 1) was submitted to the NRC for approval on November 5, 1990. PSC subsequently addressed several requests for additional information from the NRC. The Proposed Decommissioning Plan was updated to incorporate this additional information, and was submitted to the NRC for approval on April 17, 1992. PSC has also submitted and received NRC approval to reduce fire protection, emergency preparedness and physical security programs consistent with plant requirements during defueling. Other non-nuclear facilities on the FSV site are maintained and operated by PSC as needed to support PSC electric power demands.

The reactor building is the only nuclear facility to be decommissioned. An access control plan, included as part of the Proposed Decommissioning Plan, will provide suitable controls to prevent unauthorized or uncontrolled access to either the decommissioning site, or to radiologically controlled areas within the decommissioning site.

Building services, including electric power, cooling and domestic water, HVAC, lighting and other services are available and will continue to be provided throughout the decommissioning effort. These functional services and systems are identified in Section 2.2 of the Proposed Decommissioning Plan.

2.5 References

1. Proposed Decommissioning Plan for the Fort St. Vrain Nuclear Generating Station, Public Service Company of Colorado.

2. Engineering Evaluation, "Fort St. Vrain Activation Analysis," EE-DEC-0010, Revision D, March 1992.

3. PSC Letter, Crawford to Weiss dated April 26, 1991, (P-91118); Subject: "PSC Response to NRC Request for Additional Information on the Fort St. Vrain Proposed Decommissioning Plan".

4. Engle, W.W., "ANISN-P-Multigroup One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," Radiation Shielding Information Center, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

5. Ebasco Services Incorporated, "REBATE - A Computer Program for Calculation of Decay Gamma Source Strength For One or Two Dimensional Gamma Transport Analysis," Ebasco Services Incorporated, New York, New York.

6. NUREG/CR-3474, "Long Lived Activation Products in Reactor Materials," August 1984.

7. NUREG/CR-5343, "Radionuclide Characterization of Reactor Decommissioning Waste and Spent Fuel Assembly Hardware," January 1991.

8. PSC Letter, Crawford to Weiss, dated August 28, 1991 (P-91277); Subject: "Verification of FSV Activation Analysis".

9. GA International Services Corporation, Report 909658, "Fort St. Vrain Plateout Analysis for Decommissioning Study," Issue B, February 28, 1992 (GP-3538).

10. Hudritsch, W.W., "PADLOC, A One-Dimensional Computer Program for Calculating Coolant and Plateout Fission Product Concentrations," General Atomic Report GA-A14401, September 1981.

11. I. F. White, et.al., "Assessment of Management Modes for Graphite from Reactor Decommissioning", EUR 9232 en, Commission of the European Communities.

12. PSC Letter, Warembourg to Weiss, dated February 5, 1992 (P-92036); Subject: "Initial Site Characterization Program Results".

13. PSC Letter, Crawford to NRC, dated April 23, 1990, (P-90139); Subject: "Fort St. Vrain Nuclear Generating Station Radiological Environmental Monitoring Program Annual Summary Report for 1989".

14. PSC Letter, Warembourg to NRC, dated April 24, 1992 (P-92171); Subject: "Annual Rad gical Environmental Monitoring Report" for 1991.

TABLE 2.1-1
FORT ST. VRAIN MILESTONES AND MAJOR EVENTS

December	1973	Plant construction completed.
December 21	1973	Facility Operating License No. DPR-34 issued to PSC.
December 26	1973	Initial fuel loading.
January 31	1974	Initial nuclear criticality.
	1974-1979	Startup testing, low power operation, and required plant modifications.
July	1979	FSV committed for commercial operation.
May	1986	NRC mandated Environmental Qualification (EQ) outage.
September	1986	Stipulation and Settlement Agreement removing FSV from the rate base.
October	1986	Steam generator reanalyses performed which reduced maximum attainable power level to 82% of rated power (270 Mwe).
May	1987	Plant restart following EQ outage.
July	1987	Shutdown following helium circulator failure.
October	1987	Hydraulic fire during plant restart.
December 5	1988	PSC informs NRC that FSV will be permanently shutdown not later than June 30, 1990.
June 30	1989	PSC submitted the FSV Preliminary Decommissioning Plan.
July	1989	Plant record for Kw generated for one month period.
August 29	1989	PSC Board of Directors announce decision to terminate operations at FSV effective that date.
May 1	1990	NRC issues Confirmatory Shutdown Order for FSV.
July 2	1990	The Westinghouse Team selected as the FSV decommissioning contractor.
November 5	1990	PSC submitted the FSV Proposed Decommissioning Plan.
December 21	1990	PSC submitted the Decommissioning Technical Specifications.
May 21	1991	Possession Only License issued.

TABLE 2.2-1
ACTIVATION ANALYSIS RESULTS
(Estimated Total Curies Three Years After Shutdown)

Fixed Components	H-3	C-14	Ca-41	Ca-45	Mn-54	Fe-55	Fe-59	Co-60	Ni-59	Ni-63	Nb-94	Ag-110	Eu-152	Eu-154	Others	Total Curies
1. Core Barrel	-	-	-	-	0.11	7.28	<0.01	1.03	-	-	<0.0:	-	-	-	0.01	8.43
2. CSF Liner	-	-	-	-	<0.01	139.98	<0.01	2.27	<0.01	0.04	<0.01	-	-	-	<0.01	142.29
3. PCRV Liner	-	-	-	-	0.08	110.34	<0.01	4.34	<0.01	0.03	<0.01	-	-	-	<0.01	114.79
4. CSF/Kaowool Insulation and Cover Plates	-	<0.01	<0.01	-	<0.01	87.20	<0.01	2.24	<0.0:	0.08	-	-	-	-	<0.01	89.52
5. CSF Silica Blocks	-	<0.01	<0.01	-	<0.01	246.54	<0.01	6.33	<0.01	0.22	-	-	-	-	<0.01	253.08
6. PCRV Kaowool Insulation and Cover Plates	-	<0.01	<0.01	-	0.01	5.57	<0.01	0.44	-	-	-	-	-	-	0.01	6.03
7. Metal Shell-Large Side Reflector	-	-	-	-	<0.01	0.01	<0.01	<0.01	-	-	-	-	-	-	<0.01	0.01
8. Large Side Reflector and Permanent Hexagonal Blocks	82557.70	-	20.11	77.44	0.30	441114.00	<0.01	3446.24		-	-	-	-	-	0.21	527216.00
9. Core Support Blocks	47.19	-	<0.01	0.01	<0.01	69.15	-	0.54	-	-	-	-	-	-	<0.01	116.89
10. Reflector Keys	-	-	-	-	<0.01	0.01	<0.01	<0.01	-	-	-	-	-	-	<0.01	0.01
11. Boronated Spacer Blocks	11531.50	1.02	0.81	3.12	<0.01	47208.60	<0.01	7097.13	2.81	392.45	-	-	-	-	0.26	66237.70
TOTAL																594,184.76
Removable Components:																
1. Metal Clad Block - CR	-	-	-	-	0.06	18451.70	<0.01	4342.78	2.08	290.39	0.18	-	-	-	0.01	23087.20
2. Metal Clad Block-NCR	-	-	-	-	0.52	169668.20	<0.01	2786.29	-	-	<0.01	-	-	-	<0.01	172455.01
3. Region Constraint Device	-	-	-	<0.01	0.07	71.85	<0.01	48.93	0.01	1.27	0.01	-	-	-	<0.01	122.14
4. Orifice Valve	-	-	-	-	0.17	299.25	<0.01	115.56	-	-	-	-	-	-	<0.01	414.98
5. Reflector Block with Hastelloy Cans		-	-	<0.01	0.07	300.94	<0.01	3407.45	0.67	88.35	<0.01	-	-	-	<0.01	3797.48
6. Removable Graphite Reflector	3208.53	-	4.34	16.99	0.13	6034.36	<0.01	2483.61	-	-	-	-	-	-	<0.01	11747.96
TOTAL																211,624.77

TABLE 2.2-1 (Continued)

Component/Isotope	H-3	C-14	Ca-41	Ca-45	Mn-54	Fe-55	Fe-59	Co-60	Ni-59	Ni-63	Nb-94	Ag-110	Eu-152	Eu-154	Others	Total
Concrete Radial:																
1st Six Inches	0.26	<0.01	<0.01	0.02	<0.01	7.93	<0.01	0.31	<0.001	<0.01	<0.01	<0.01	0.31	0.31	0.03	8.89
2nd Six Inches	0.09	<0.01	<0.01	<0.01	<0.01	2.81	<0.01	0.10	<0.001	<0.01	<0.01	<0.01	0.11	<0.01	0.02	3.13
3rd Six Inches	0.01	<0.01	<0.01	<0.01	<0.01	0.33	<0.01	0.01	<0.001	<0.01	<0.01	<0.01	0.01	<0.01	0.01	0.37
4th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	0.04	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	0.04
5th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	0.01
6th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
7th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
8th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
TOTAL																12.44
Concrete - Top Head:																
1st Six Inches	2.84	<0.01	0.04	0.18	0.01	87.79	<0.01	3.37	<0.001	<0.01	<0.01	0.01	3.45	0.35	0.28	98.32
2nd Six Inches	0.73	<0.01	0.01	0.05	<0.01	22.95	<0.01	0.79	<0.001	<0.01	<0.01	<0.01	0.90	<0.01	0.04	25.55
3rd Six Inches	0.07	<0.01	<0.01	<0.01	<0.01	2.26	<0.01	0.08	<0.001	<0.01	<0.01	<0.01	0.09	<0.01	0.01	2.52
4th Six Inches	0.01	<0.01	<0.01	<0.01	<0.01	0.24	<0.01	0.01	<0.001	<0.01	<0.01	<0.01	0.01	<0.01	<0.01	0.27
5th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	0.03	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	0.04
6th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	0.01
7th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
8th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
TOTAL																126.71
Concrete - CSF:																
1st Six Inches	0.16	<0.01	<0.01	0.01	<0.01	5.12	<0.01	0.17	<0.001	<0.01	<0.01	<0.01	0.20	0.02	0.01	5.69
2nd Six Inches	0.01	<0.01	<0.01	<0.01	<0.01	0.34	<0.01	0.01	<0.001	<0.01	<0.01	<0.01	0.01	<0.01	0.01	0.38
3rd Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	0.03	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	0.03
4th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
5th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
6th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
7th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
8th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
9th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
10th Six Inches	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.001	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01	<0.01
TOTAL																6.10

TABLE 2.2-2

PCRV DOSE RATES IN AIR
AT 5 YEARS AFTER SHUTDOWN

RADIAL	GAMMA DOSE RATE R/Hr
All components (from large side reflector to PCRV concrete)	9.7E + 01
Large side reflectors removed (from spacers to PCRV concrete)	2.3E + 02
From core barrel to PCRV concrete	2.1E - 02
PCRV liner and concrete only	8.8E - 03
PCRV concrete only	4.5E - 03
22" PCRV concrete removed	6.3E - 06
24" PCRV concrete removed	3.4E - 06
AXIAL UP	
All components (from Kaowool insulation to PCRV concrete)	1.7E - 01
PCRV liner and concrete only	4.4E - 01
PCRV concrete only	1.7E - 01
32" PCRV concrete removed	7.6E - 06
34" PCRV concrete removed	4.4E - 06
36" PCRV concrete removed	2.6E - 06
AXIAL DOWN	
All Components (from core support blocks to core support floor)	6.1 - 02
PCRV liner and concrete only	2.5E - 01
PCRV concrete only	1.8E - 02
20" PCRV concrete removed	5.3E - 06
22" PCRV concrete removed	2.7E - 06

TABLE 2.2-3

ESTIMATED PLATEOUT CONCENTRATION ON MAJOR PRIMARY CIRCUIT COMPONENTS AT END OF REACTOR LIFE (890 EFPD)

COMPONENT	PLATEOUT CONCENTRATION			
	Sr-90		Cs-137	
	dpm/100 cm²	Ci/cm²	dpm/100 cm²	Ci/cm²
Lower Reflectors	5.98 E6	2.69 E-8	7.85 E7	3.54 E-7
Steam Generator				
Reheater Section[1]	1.11 E7	5.02 E-8	2.54 E7	1.15 E-7
Economizer Section	2.17 E5	9.78 E-10	4.85 E7	2.18 E-7
Circulator	2.88 E6	1.30 E-8	6.10 E6	2.75 E-8
Circulator Outlet	1.24 E5	5.60 E-10	9.56 E4	4.31 E-10
Core Barrel Annulus	1.53 E6	7.33 E-9	9.27 E5	4.18 E-9
Upper Reflectors	1.10 E7	4.95 E-8	5.31 E6	2.39 E-8

[1] Steam generator component with the highest estimated plateout concentration

TABLE 2.2-4

INTEGRATED PLATEOUT ON EACH PRIMARY CIRCUIT COMPONENT AT END OF LIFE (890 EFPD)

BRANCH NAME	Cs-134[1] (Curies)	Cs-137[1] (Curies)	I-131[2] (Curies)	I-129[2] (Curies)	Sr-90[2] (Curies)	Te-127m[2] (Curies)
Lower Reflector	2.729E+00	5.752E+00	6.490E-03	5.662E-09	4.380E-01	3.412E+01
Core Support Blocks	5.869E-02	1.319E-01	3.122E-04	2.723E-10	7.671E-03	5.964E-01
Core Exit Plenum	1.198E-02	2.822E-02	2.166E-04	1.890E-10	1.449E-03	1.052E-01
Steam Generator Inlet	4.951E-04	7.086E-04	1.027E-02	5.563E-09	1.331E-02	8.466E-01
Steam Generator Reheater	2.000E-01	7.807E-01	7.059E-01	3.295E-08	3.420E-01	2.031E+01
Superheater	5.331E-01	1.988E+00	1.664E+00	8.942E-08	1.098E-01	6.648E+00
Economizer	1.291E+00	3.378E+00	3.134E+01	8.284E-08	1.514E-02	7.719E-01
Evaporator	7.847E-01	7.063E-01	1.186E+03	2.725E-07	5.726E-03	1.296E-01
Steam Gen. Outlet Plenum	2.064E-03	1.994E-03	1.199E+01	1.119E-08	5.129E-04	3.388E-04
Circulators	1.383E-02	1.443E-02	4.744E+00	3.312E-09	6.801E-03	2.270E-03
Circulator Outlet Plenum	4.851E-04	7.594E-0[illegible]	6.790E+00	1.128E-08	9.875E-04	7.962E-05
Core Barrel/Liner Annulus	1.076E-02	2.392E-02	4.363E+01	3.614E-08	4.197E-02	1.767E-03
Core Inlet Plenum	1.267E-03	4.949E-03	1.901E+01	3.164E-08	1.004E-02	2.083E-04
Upper Reflectors	7.228E-02	3.893E-01	2.641E-01	3.326E-07	8.043E-01	1.186E-02
Side Reflectors	1.041E-04	1.037E-03	1.669E-01	2.858E-07	1.443E-03	1.707E-05
Purification System	9.754E-05	3.984E-04	2.554E+00	4.213E-04	8.725E-04	1.610E-05
TOTAL	5.712E+00	1.32E+01	1.309E+03	4.225E-04	1.800E+00	6.355E+01
TOTAL (3 YEARS		2.081E+00	1.23E+01	0.00E+00	4.224E-04	1.675E+005

(1) Plateout distribution based upon sorption isotherms for unoxidized alloy steel surfaces.
(2) Based on the source rate calculated from the xenon data using the square root of half-life dependence.

TABLE 2.2-5

ESTIMATED CURIE TOTAL AT FSV
(Three Years After Shutdown)

NOTE: The systems listed below are those systems which are known to be contaminated, or experiencing on-going maintenance, defueling and component removal which may transfer contamination to other systems and/or locations.

System No.	System	Total Curies	
		From Activation	From Loose Contamination (1)
11	PCRV and Internal Components	7.94 E+05	2.54 E+02
12	Controls Rods and Drives	1.84 E+04	N/A
13	Fuel Handling Equipment	N/A	8.95 E-03
14	Fuel Storage Facility	N/A	2.08 E-02
16	Auxiliary Equipment	N/A	9.05 E-03
17	Reactor Removable Reflector	4.82 E+05	N/A
21	Primary Coolant	N/A	6.01 E+01
22	Secondary Coolant	N/A	5.68 E+03
23	Helium Purification	N/A	9.33 E-01
61	Decontamination Systems	N/A	1.06 E-05
62	Radioactive Liquid Waste	N/A	4.06 E-05
63	Radioactive Gas Waste	N/A	8.15 E-05

(1) Includes an estimate of loose surface contamination due to activated corrosion products.

TABLE 2.2-6
GRAPHITE PROPERTIES COMPARISON TABLE

PARAMETER	Large Side Reflector	Boronated Side Spacer Blocks	Removable Reflector	Core Support Blocks	British Test Sample	Remarks
Type of Graphite	HLM	HLM	H-451/H-327	PGX	(Reactor Grade)	2 Samples from Magnox Reactor
Density						
Unirradiated	1.8	1.8	1.72 - 1.77	1.76	1.82	
Irradiated	1.8	1.8	1.72 - 1.77	1.76	1.7	
Surface to Volume Ratio (cm^{-1})	0.08	0.75	0.12 - 0.53	**	1.5	** - A/V ratios not significant due to very small tritium curie content
Total Mass (g)	1.83 E8	6.11 E7	1.5 E8	8 E7	680	
Total Volume (cc)	1.015 E8	3.40 E7	8 E7	4.5 E7	376*	* - Actual sample size, 2 samples tested
Total Tritium Content (Ci)	82,558	11,532	3500	47		
Tritium Concentration (μCi/cc)	813	340	<0.01	0.6	10.7*	* - Measured value 2.2 E5 Bq/g
Major Impurities (ppm)						
Li	<2	<2	<0.1	<2	<0.05	
Fe	2000	2000	<20	1900	10	
Co	0.2	0.2	<0.01	0.2	0.02	
Flux History (EFPD)	890	890	≤890	890	≈3550	
Thermal Flux (n/cm²/sec)	3.8 E13	<3.8 E13	<3.8 E13	<3.8 E13	3.4 E13	
Maximum Temperature (°C)	300 - 500	300 - 500	400 - 700	700		
Primary Coolant	Helium	Helium	Helium	Helium	CO_2	

TABLE 2.3-1
RADIOLOGICAL SURVEY SUMMARY

Level	Elevation	Internally Radioactive Systems	Source of Radiation/ Contamination	Radiation Levels (mR/hr)	Contamination Levels Loose (DPM/100cm²)	Fixed (DPM/15 cm²)
Level 13	4916'-8" 4921'-0"	46, 47				
Level 12	4904'-0" 4906'-8"	46, 47				
Level 11	4881'-0"	11, 12, 13, 14, 15, 16, 21, 23, 46, 47, 72, 93	Shine through FSW's and ESW's	Gen. Area - 0.044		
Level 10	4864'-0"	11, 13, 14, 16, 21, 23, 46, 47, 72 93	New Fuel Loading Port	Gen. Area - 0.8 Contact - 6.0	30,000	5,000-10,000
			Hot Service Facility		(See results from Level 9 below)	
			Purge Vacuum Pumps	Gen. Area - 0.032		
Level 9	4849'-0"	11, 14, 16, 21, 23, 46, 47, 61, 62, 63, 72, 93	Regeneration System	Gen. Area 0.15		
			Hot Service Facility	Gen. Area - 0.5	100,000	10,000 - 50,000
Level 8	4839'-0" 4846'-0"	11, 14, 16, 23, 46, 47, 61, 62, 63, 72, 93	Access to Hot Service Facility	Gen. Area - 0.5 Contact - 200	50,000	
			Hot Service Facility sump	Gen. Area - 50	100,000	10,000 - 50,000
Level 7	4829'-0"	11, 14, 16, 46, 61, 62, 63, 72, 93	Irradiated Thermocouples	Gen. Area - 2.8 Contact - 4.0		
Level 6	4811'-0" 4816'-0"	11, 46, 61, 62, 63, 72, 93	Gas/Liquid Waste System Piping Turbine Building	Gen. Area - 0.02 Contact - 1.2		

TABLE 2.3-1
RADIOLOGICAL SURVEY SUMMARY (Continued)

Level	Elevation	Internally Radioactive Systems	Source of Radiation/ Contamination	Radiation Levels (mR/hr)	Contamination Levels Loose (DPM/100cm^2) /100cm^2	Fixed (DPM/15 cm^2)
Level 5	4791'-0"	11, 46, 61, 62, 63, 72, 93	Gas/Liquid Waste System Piping Compactor Bldg.	Contact - 0.75		5,000 - 10,000
Level 4	4781'-0"	46, 47, 61, 62, 63, 72, 93	Decontamination System	Gen. Area - 0.5 Contact - 3.0	600	500 - 1,000
Level 3	4771'-0"	46, 47, 61, 62, 63, 72, 93	Decontamination Laundry	Gen. Area - 0.4 Contact 2.2	1,400	100 - 500
			Floor of Vault Containing T-6101			500 - 1,000
Level 2	4756'-0"	46, 47, 61 62, 63, 72, 93				
Level 1	4740'-6"	21, 46, 47, 61, 62, 63, 72, 93	Gas Waste Compressor Drains (3)			100 - 500
			Liquid Waste Sump		5,600	30,000
Level 1	Below Floor Level	72	Reactor Building Sump			100 - 500

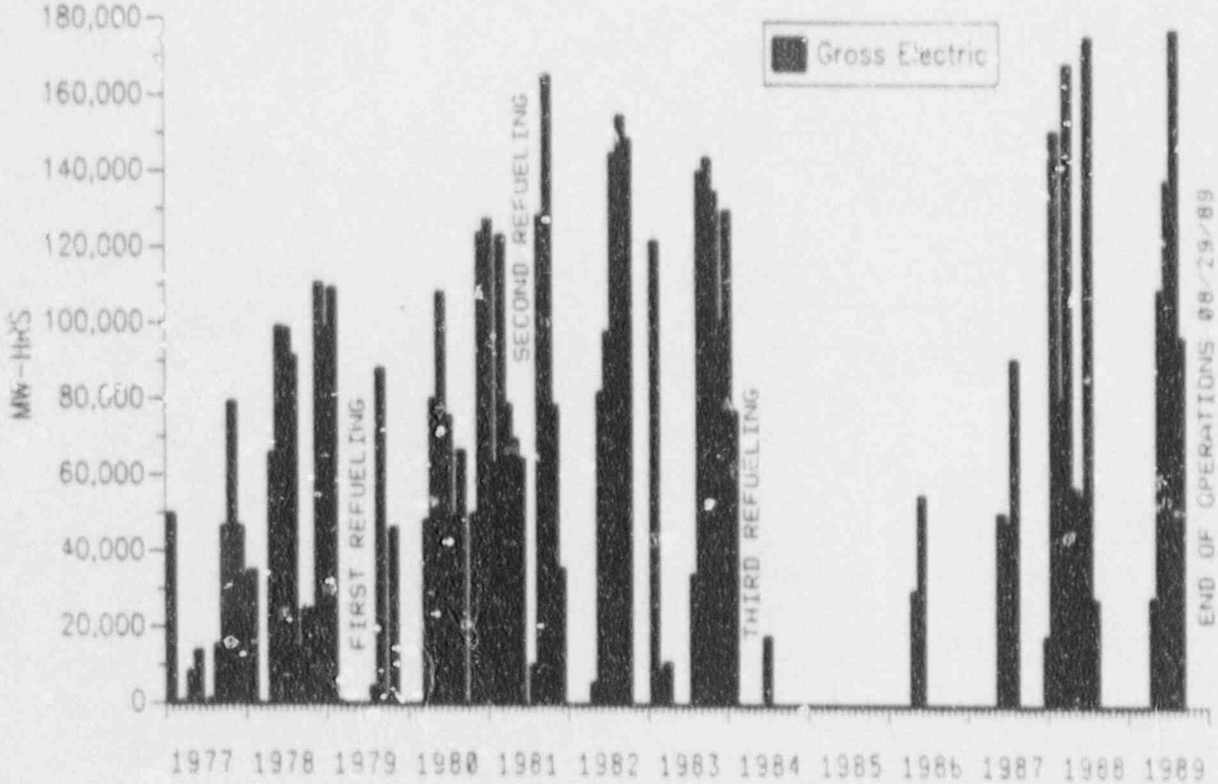


Figure 2.1-1 Fort St. Vrain Power Generation

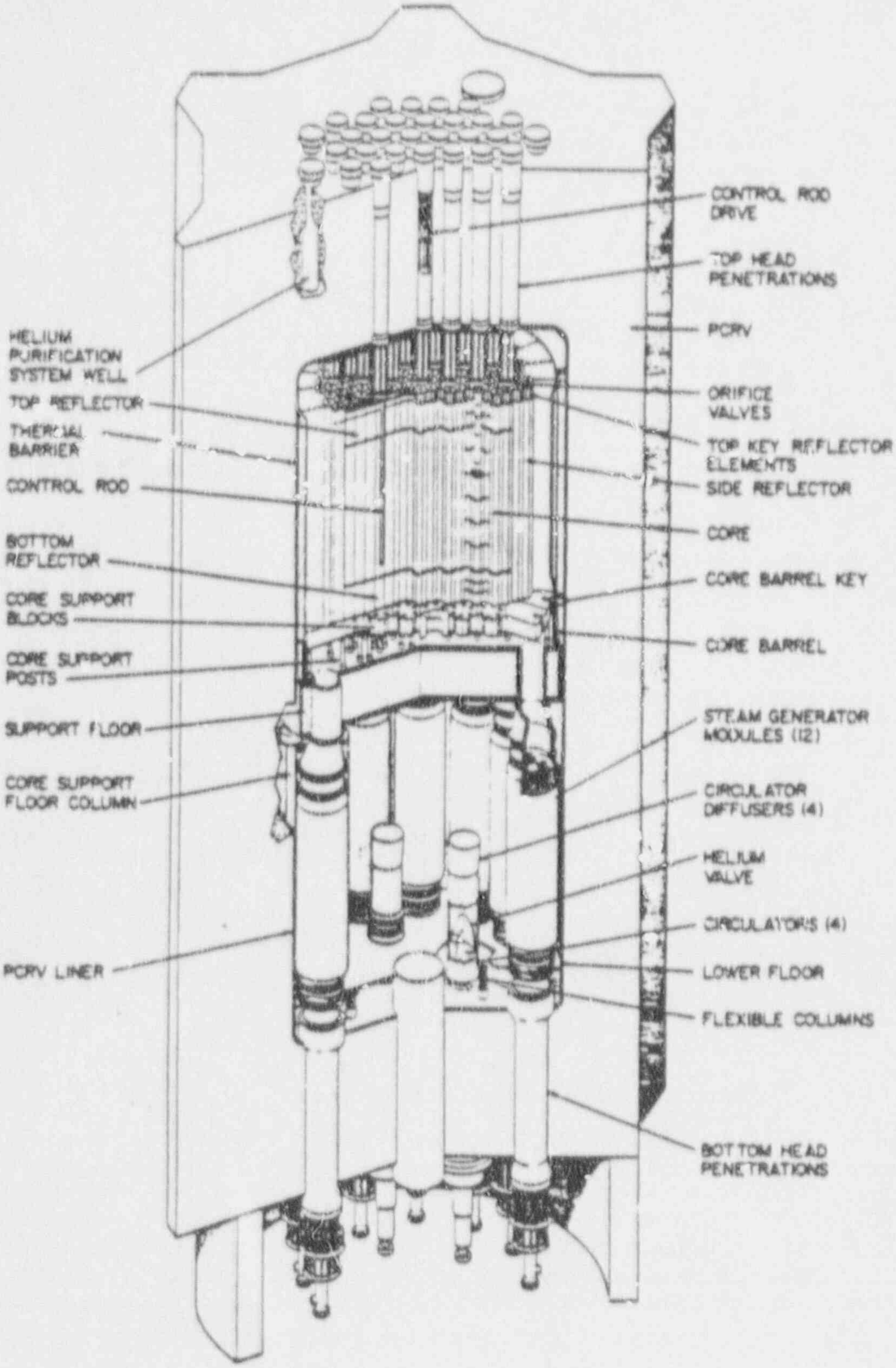
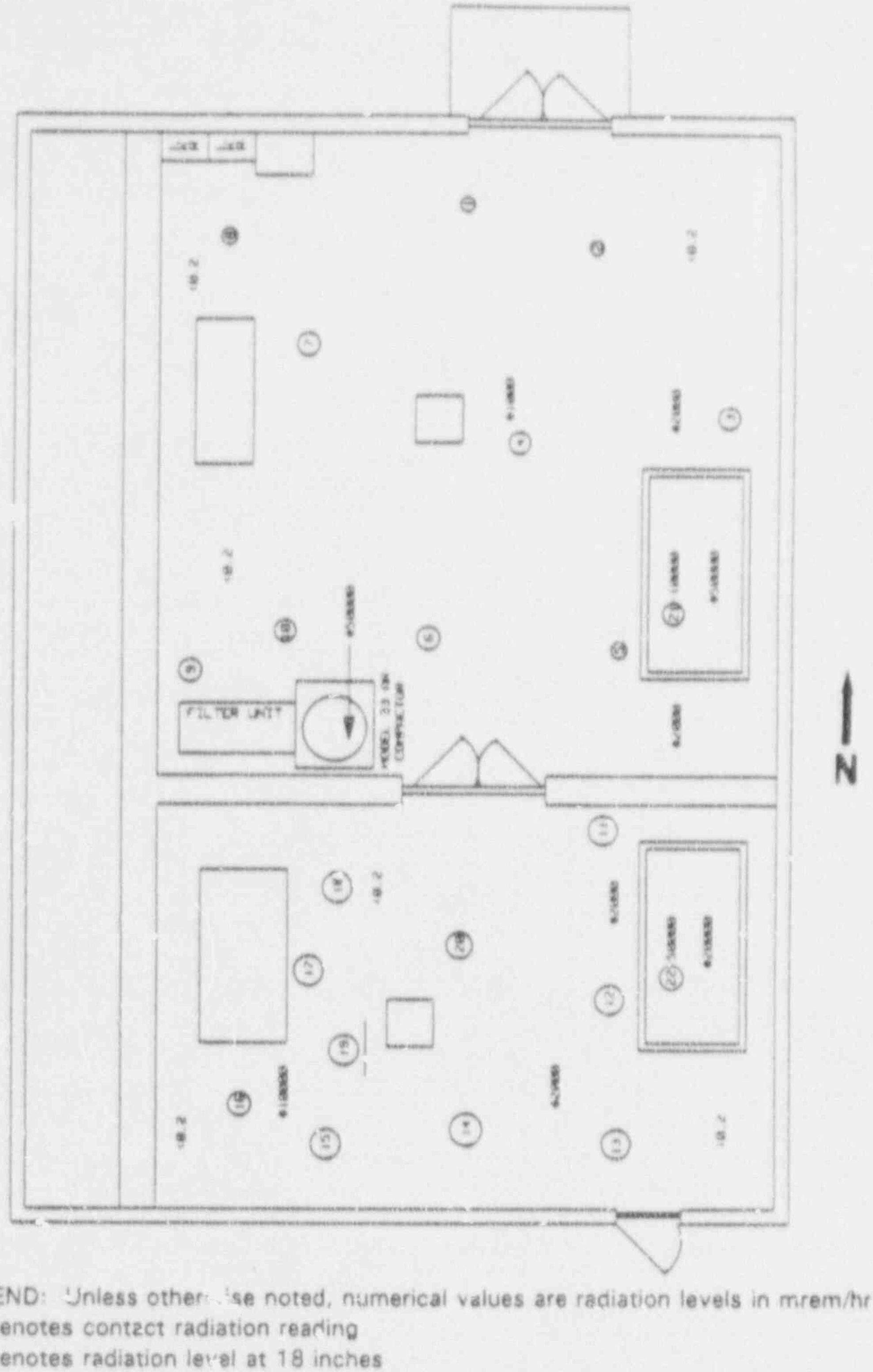
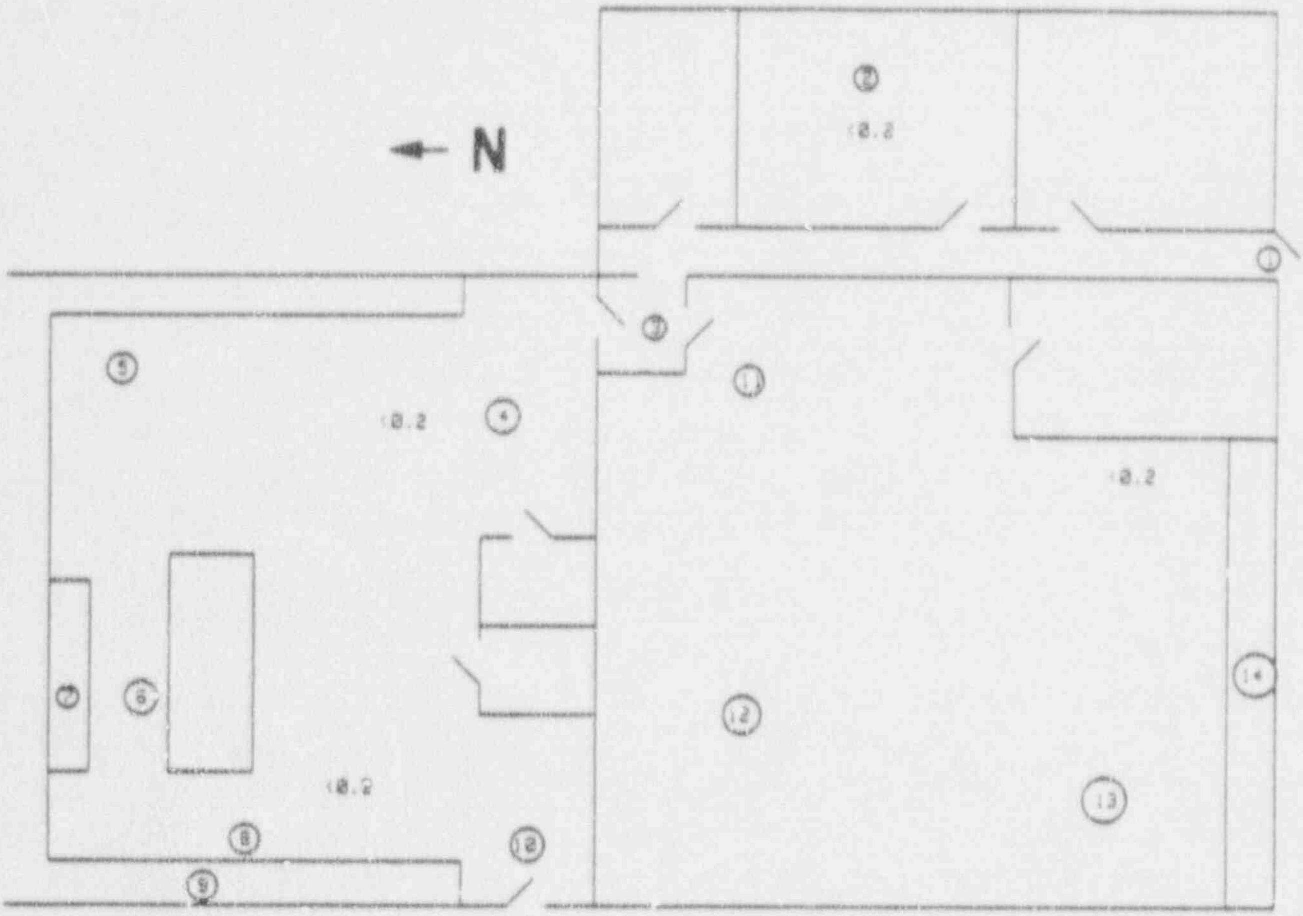


Figure 2.2-1 PCRV and Internal Components



LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-1 Compactor Building Radiation Survey



LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
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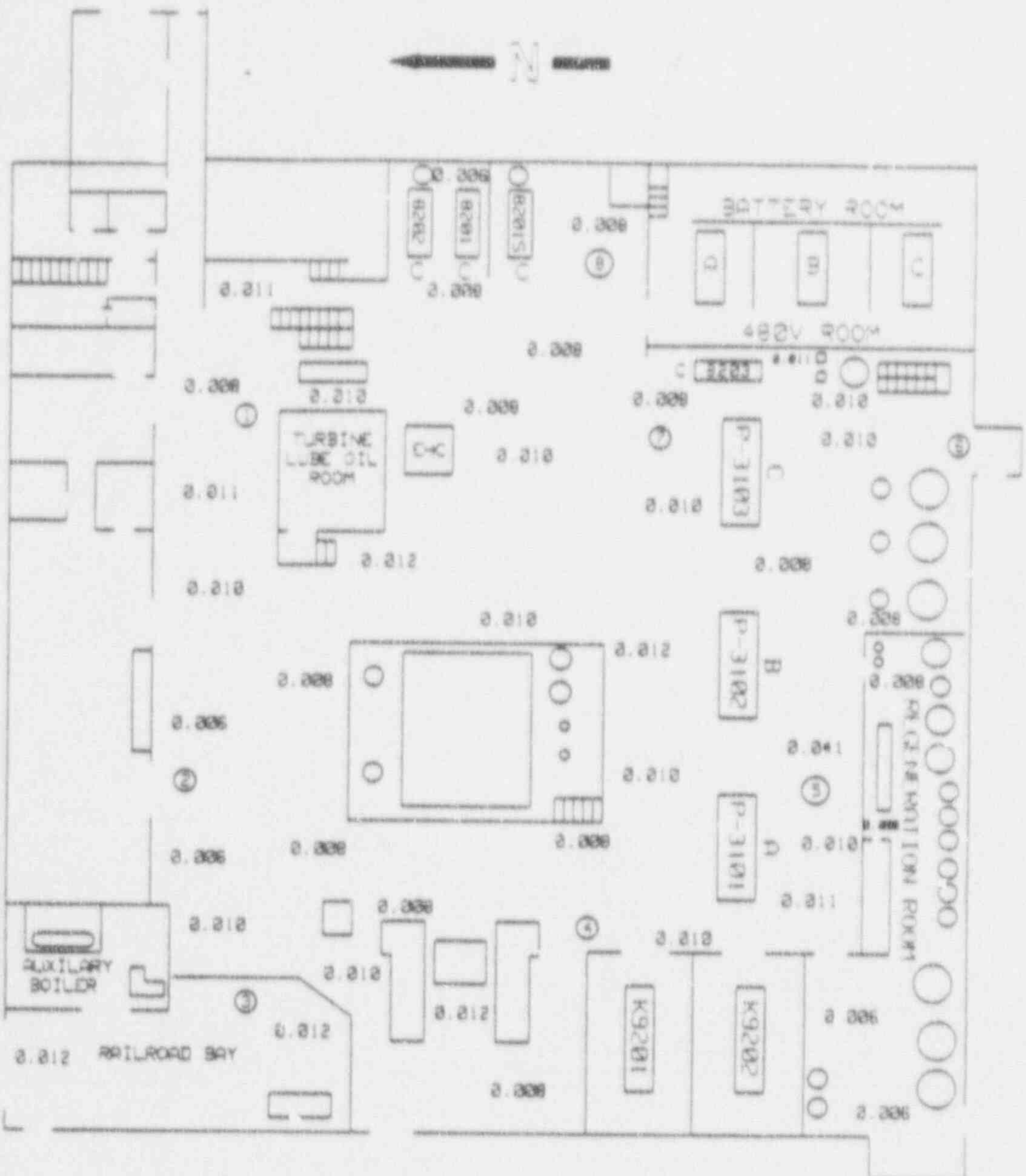
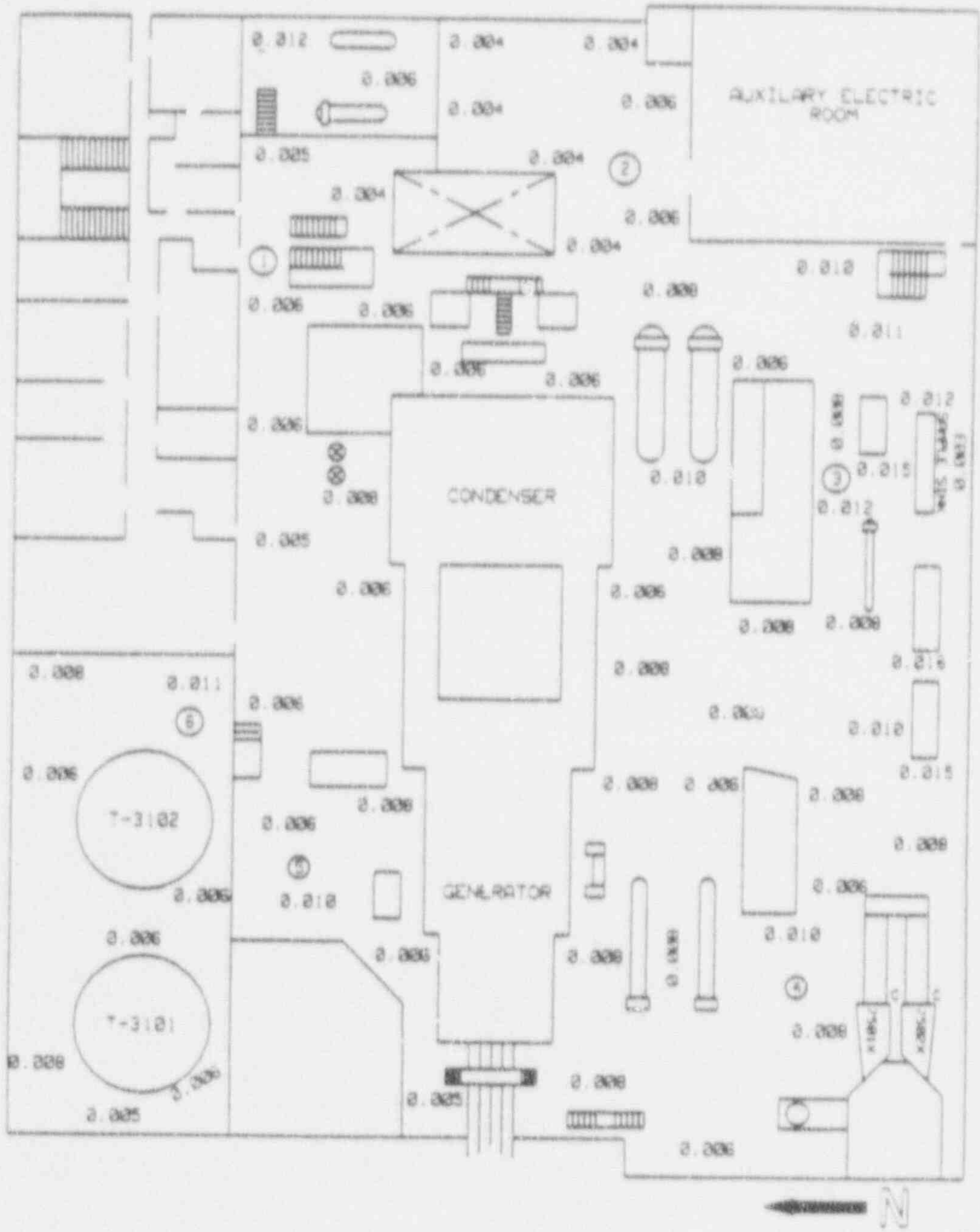
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-2 Radiochemistry Laboratory Radiation Survey



LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-3 Turbine Building Radiation Survey - Level 5 (El. 4791')



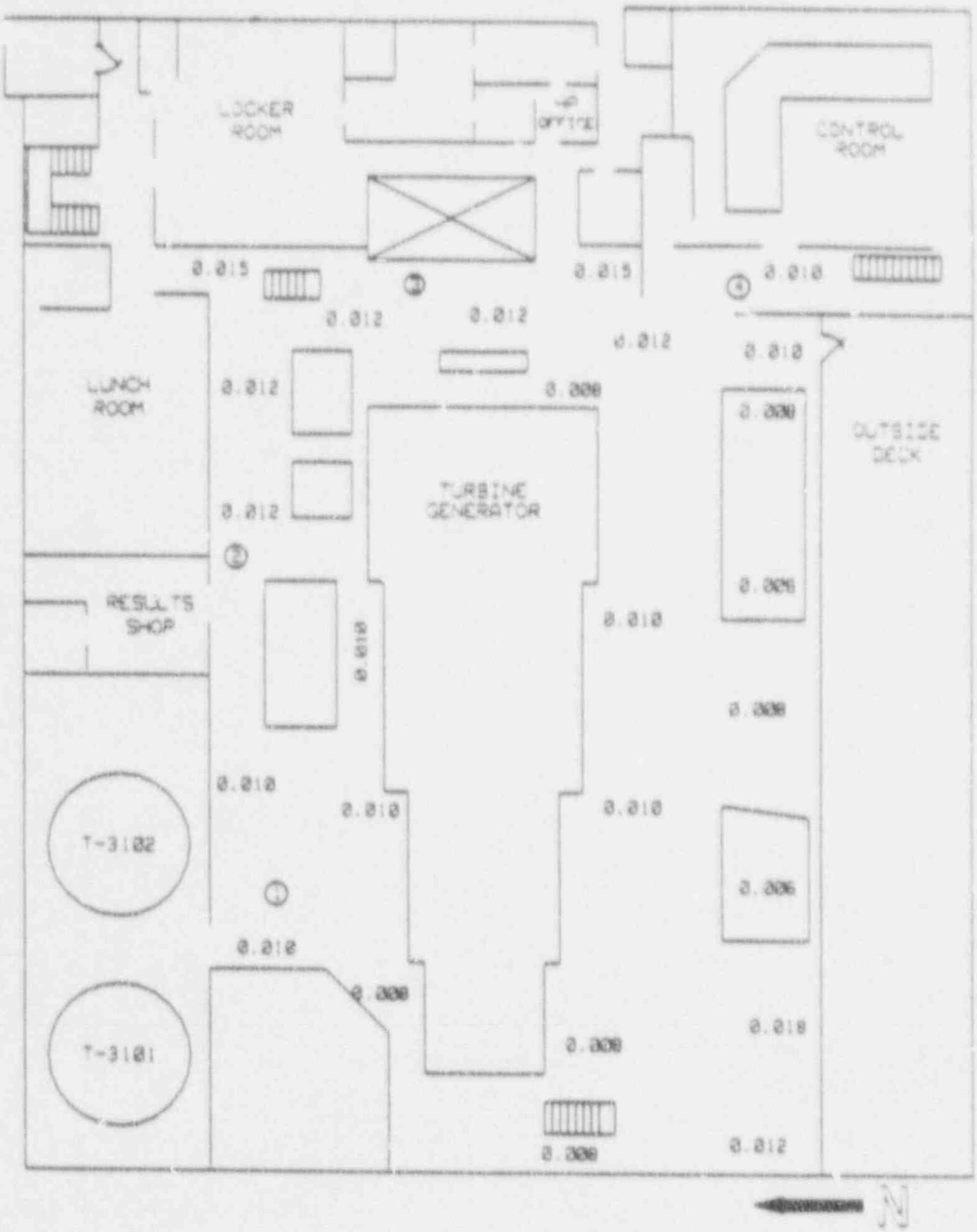
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr

* denotes contact radiation reading

( ) denotes radiation level at 18 inches

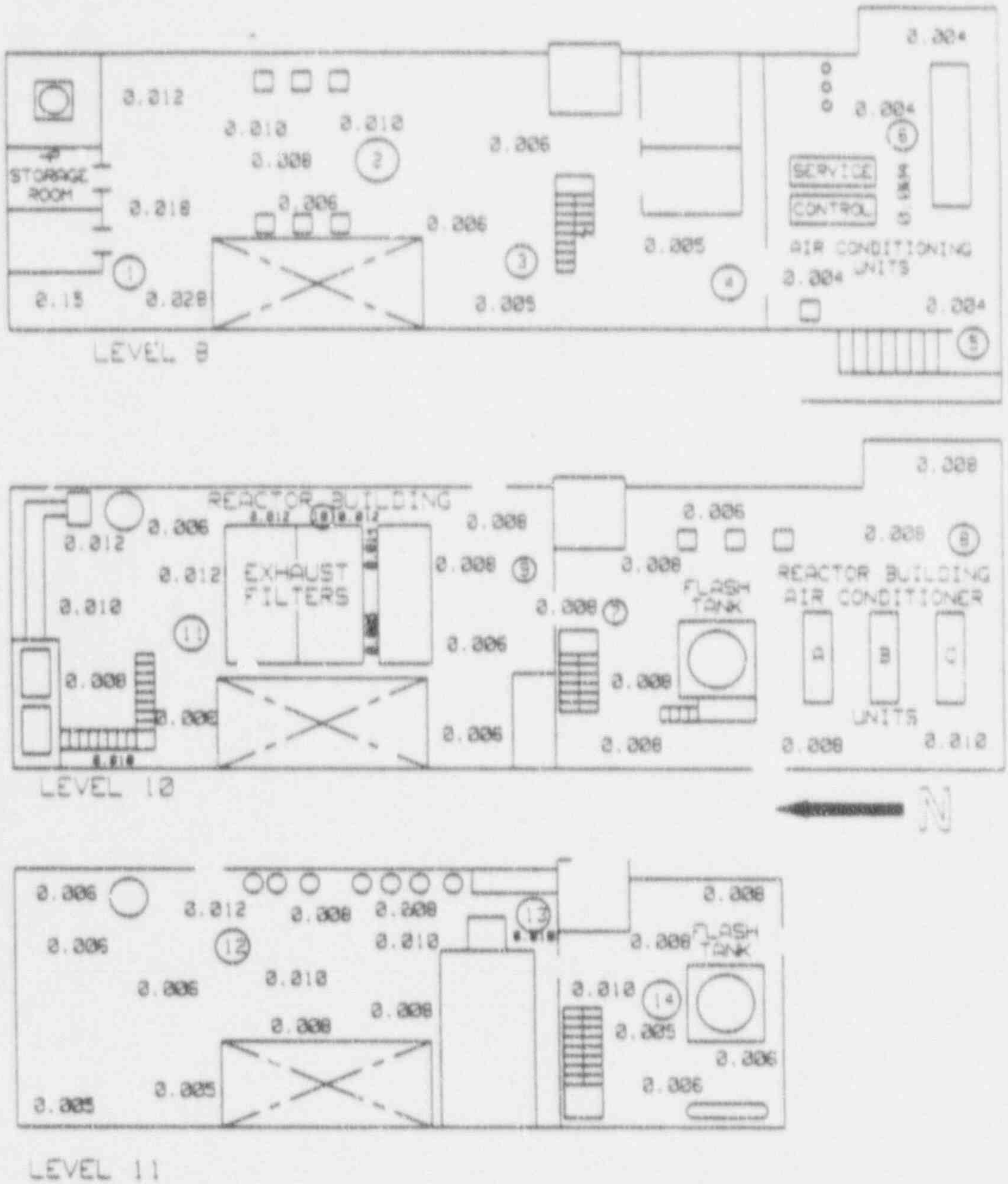
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-4 Turbine Building Radiation Survey - Level 6 (El. 4811')



LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-5 Turbine Building Radiation Survey - Level 7 (El. 4829')



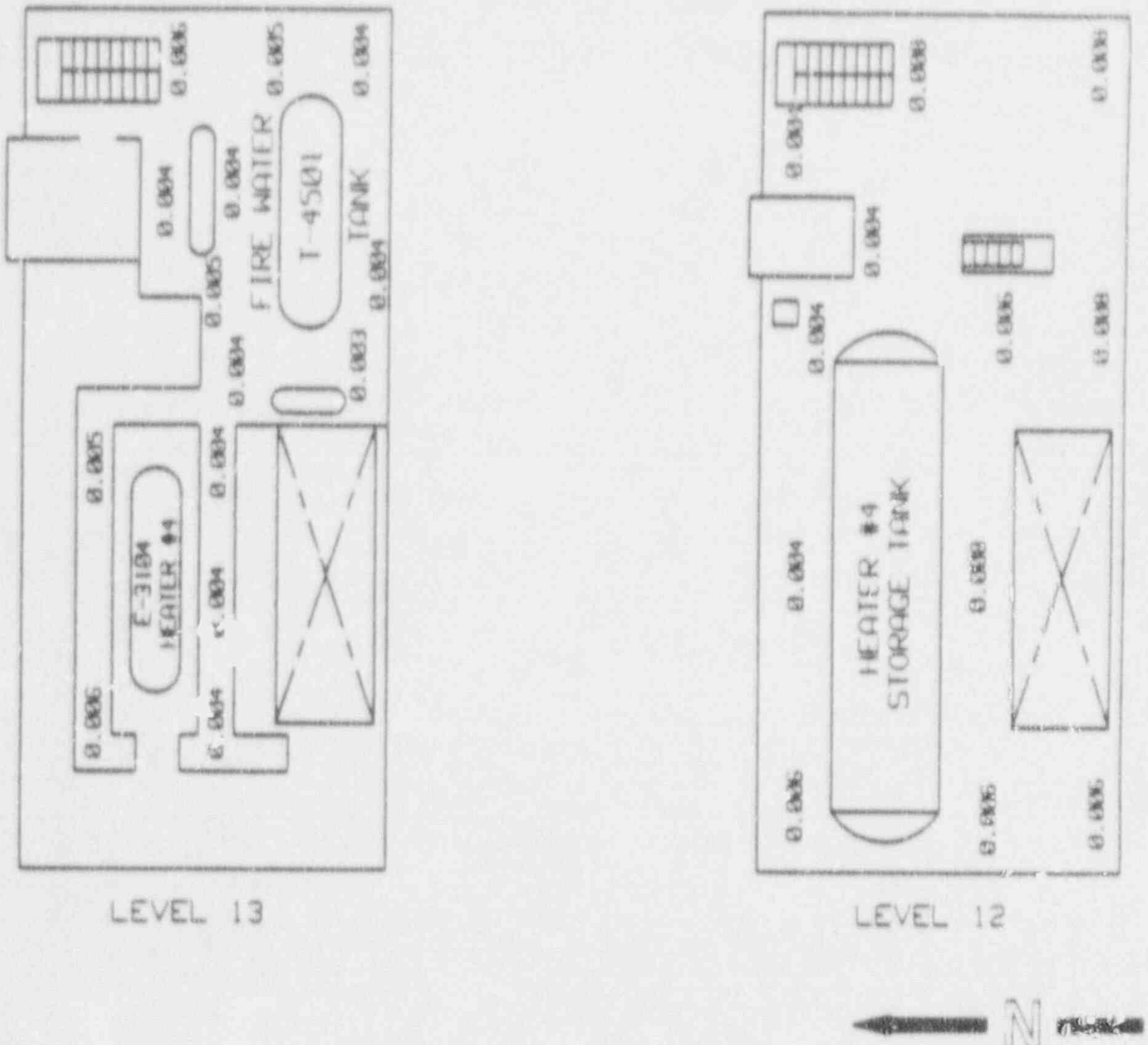
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr

* denotes contact radiation reading

( ) denotes radiation level at 18 inches

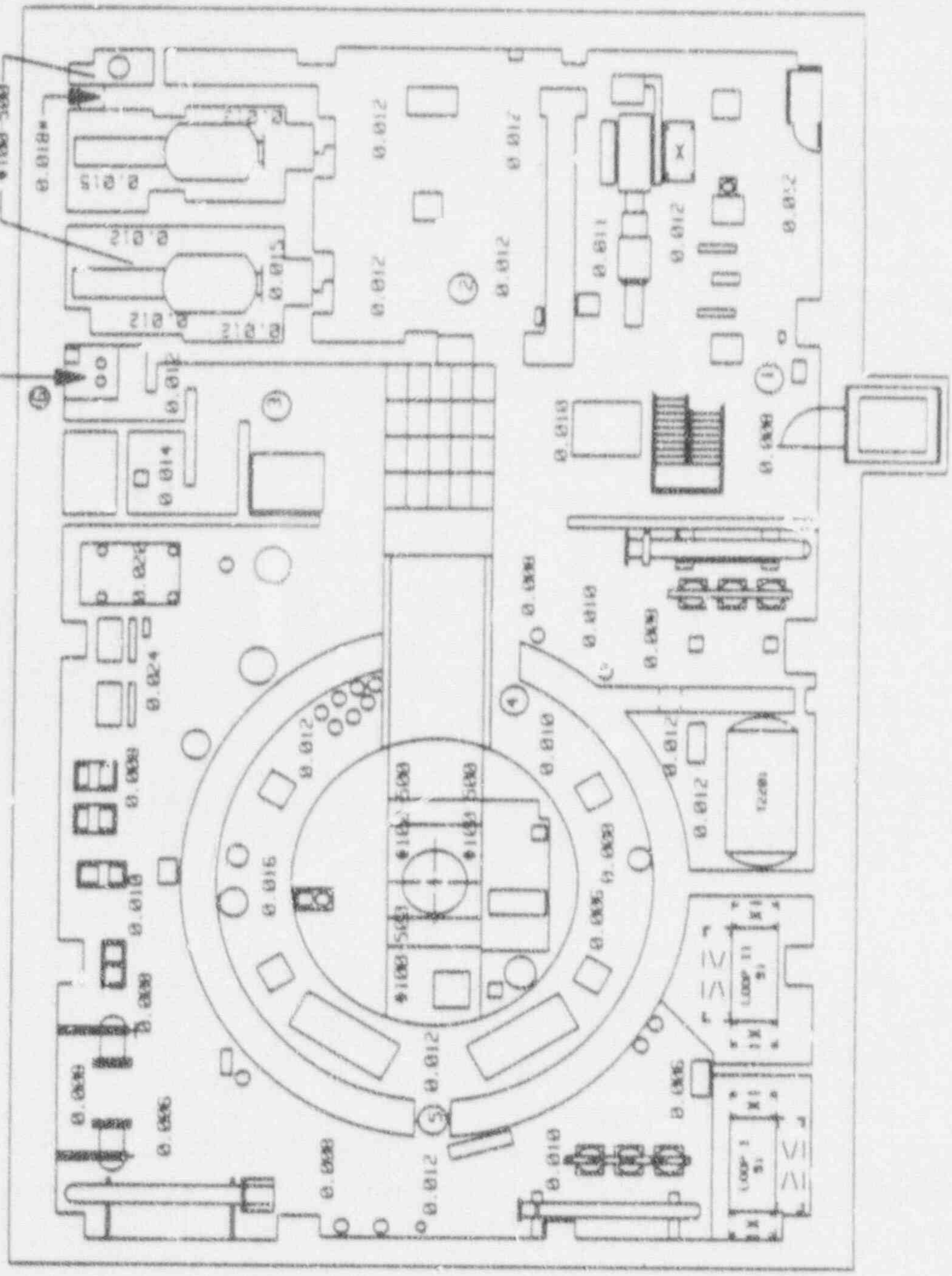
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3.6 Turbine Building Radiation Survey - Levels 8, 10 & 11 (El. 4846', 4864', 4884')



LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

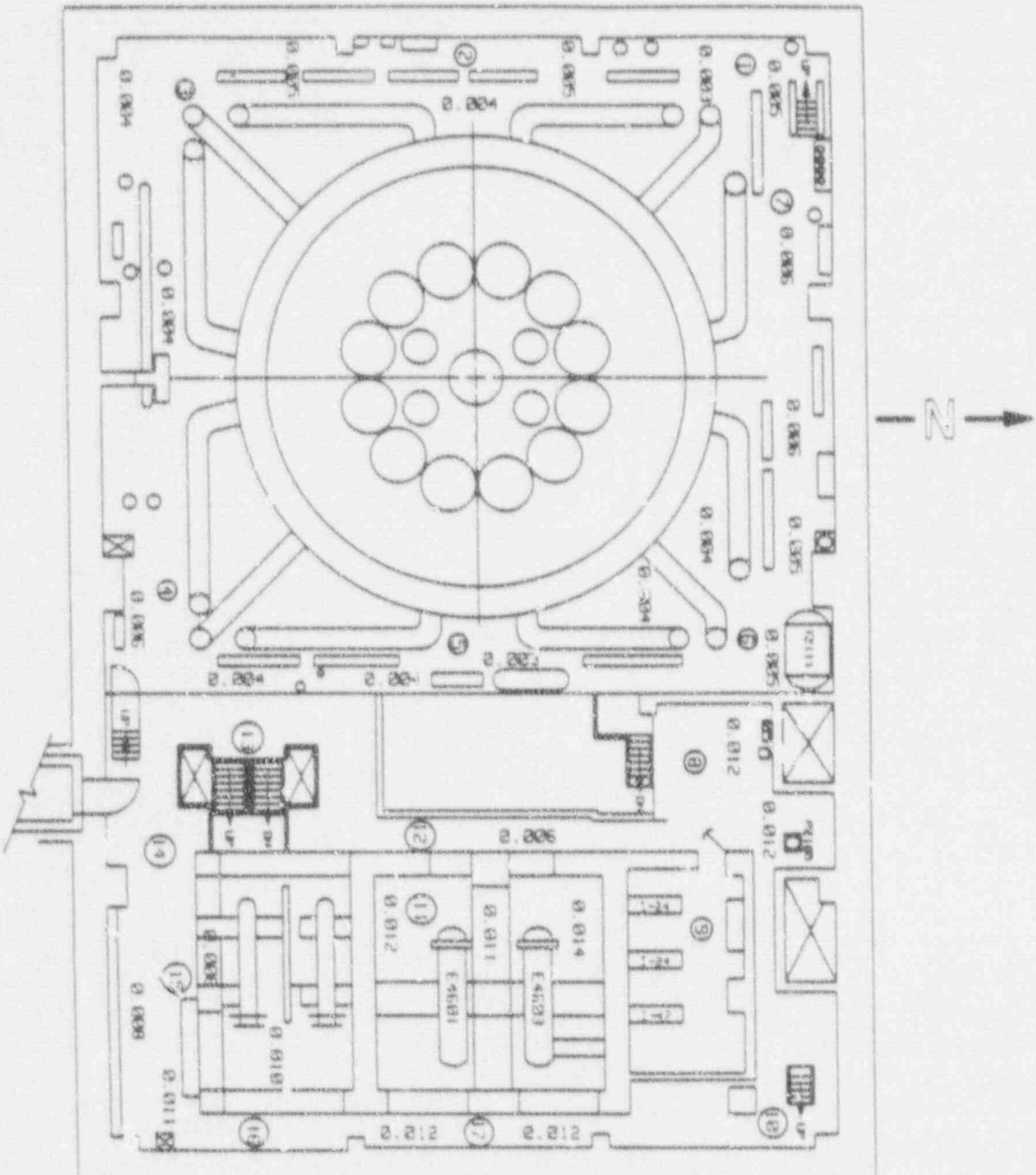
Figure 2.?-7 Turbine Building Radiation Survey - Levels 12 & 13 (eL. 4904', 4921')



LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr

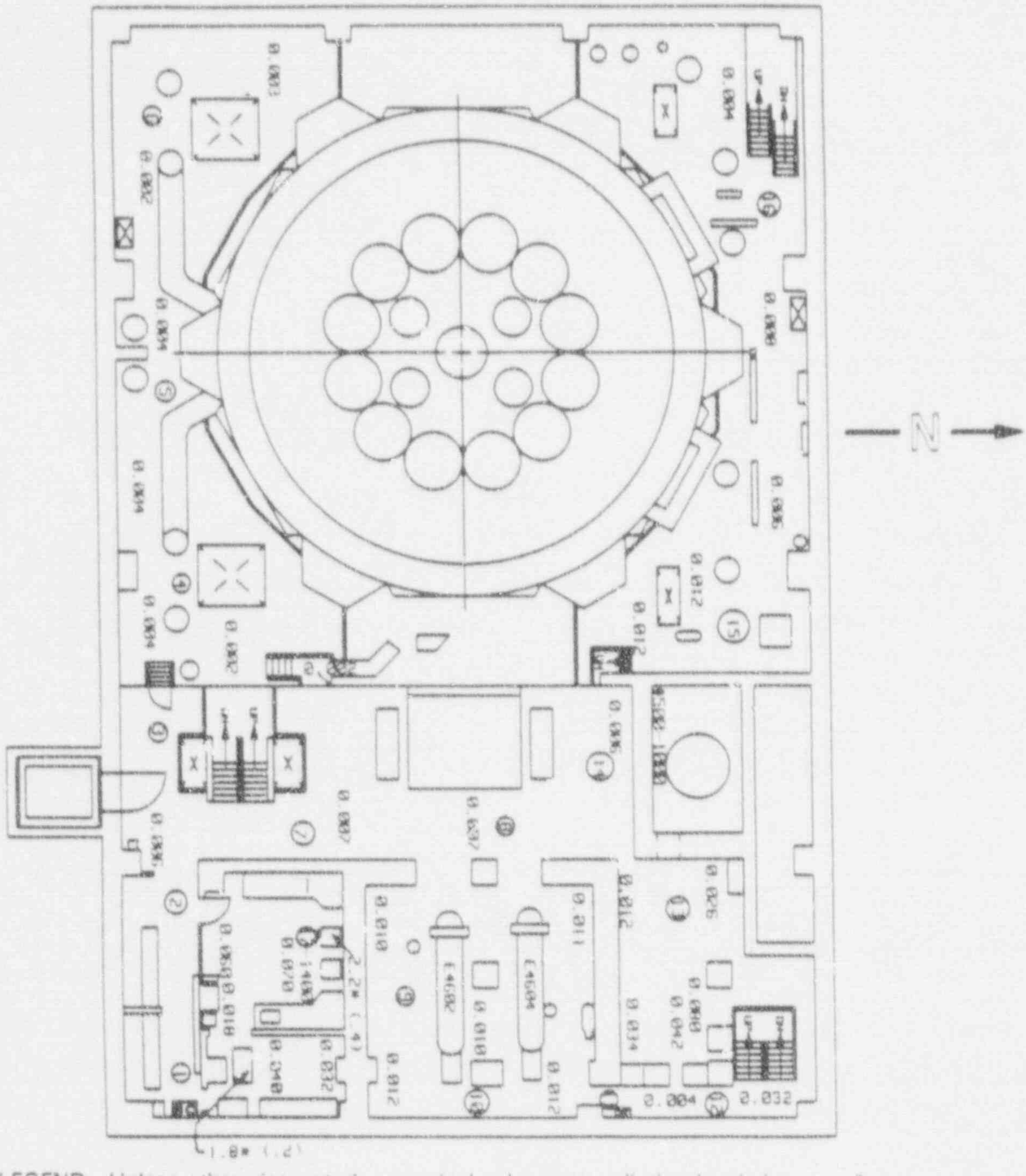
- * denotes contact radiation reading
- ( ) denotes radiation level at 18 inches
- ○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-8 Reactor Building Radiation Survey - Level 1 (El. 4740')



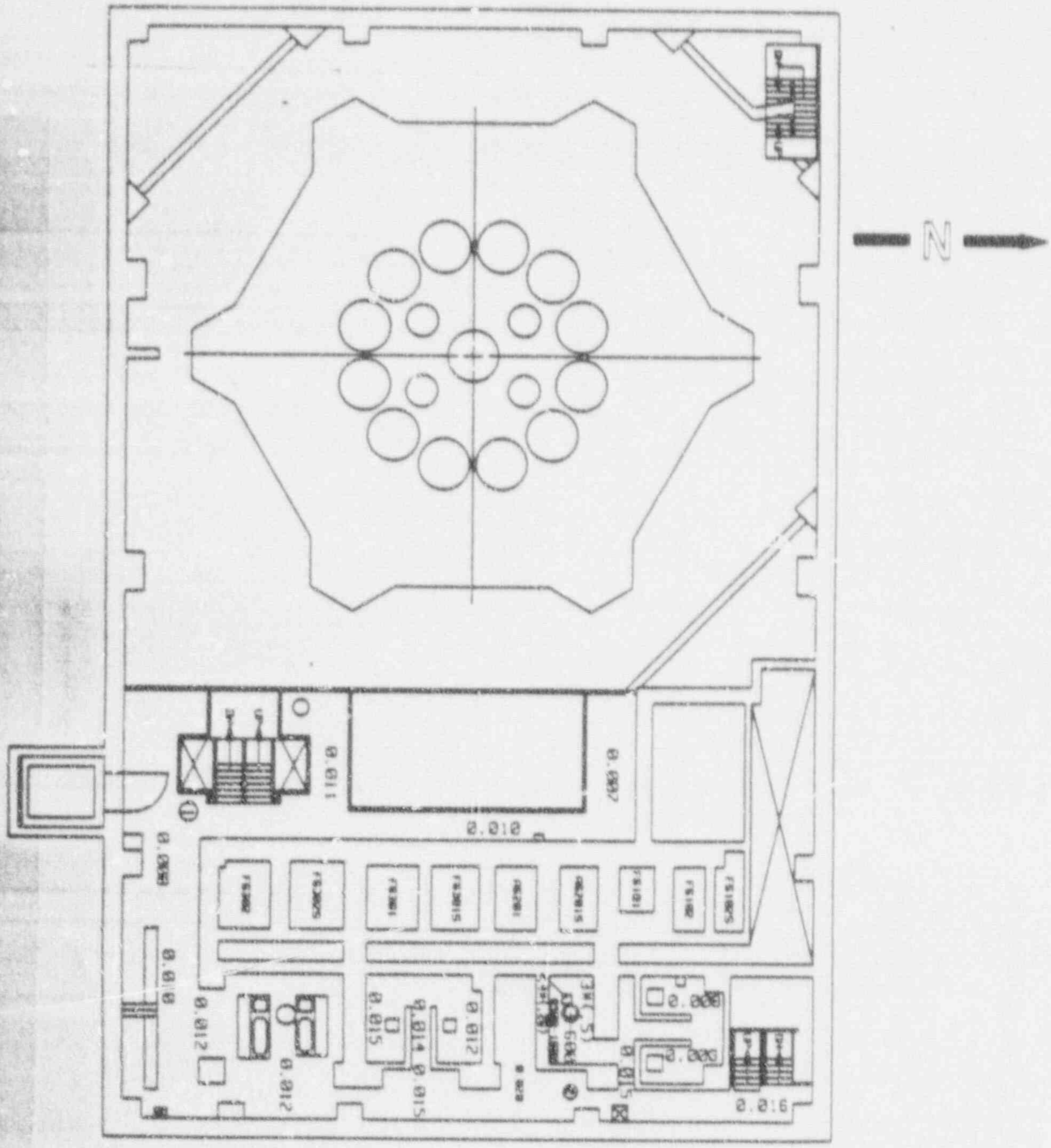
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-9 Reactor Building Radiation Survey - Level 2 (El. 4756')



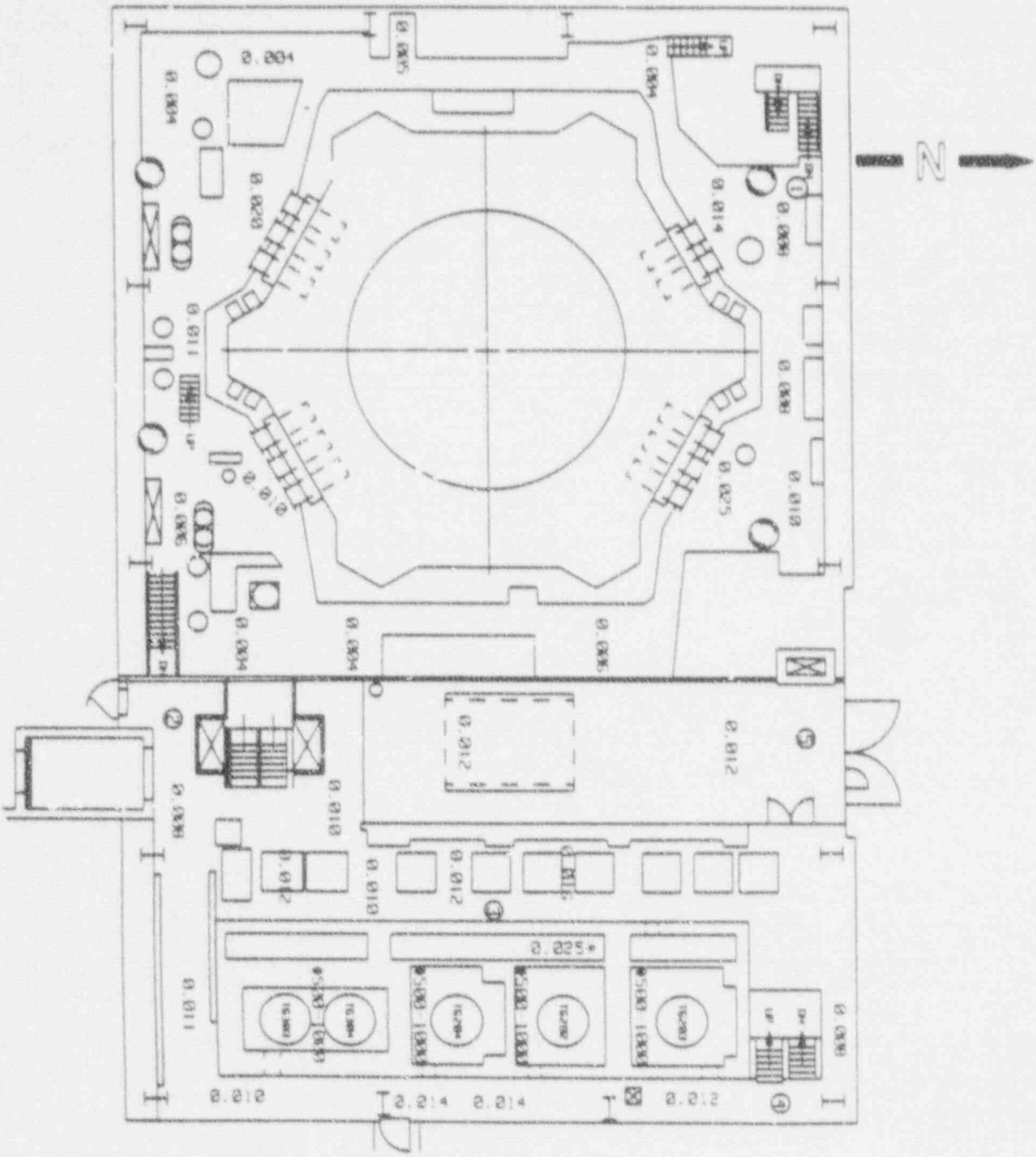
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
- * denotes contact radiation reading
- ( ) denotes radiation level at 18 inches
- ○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-10 Reactor Building Radiation Survey - Level 3 (El. 4771')



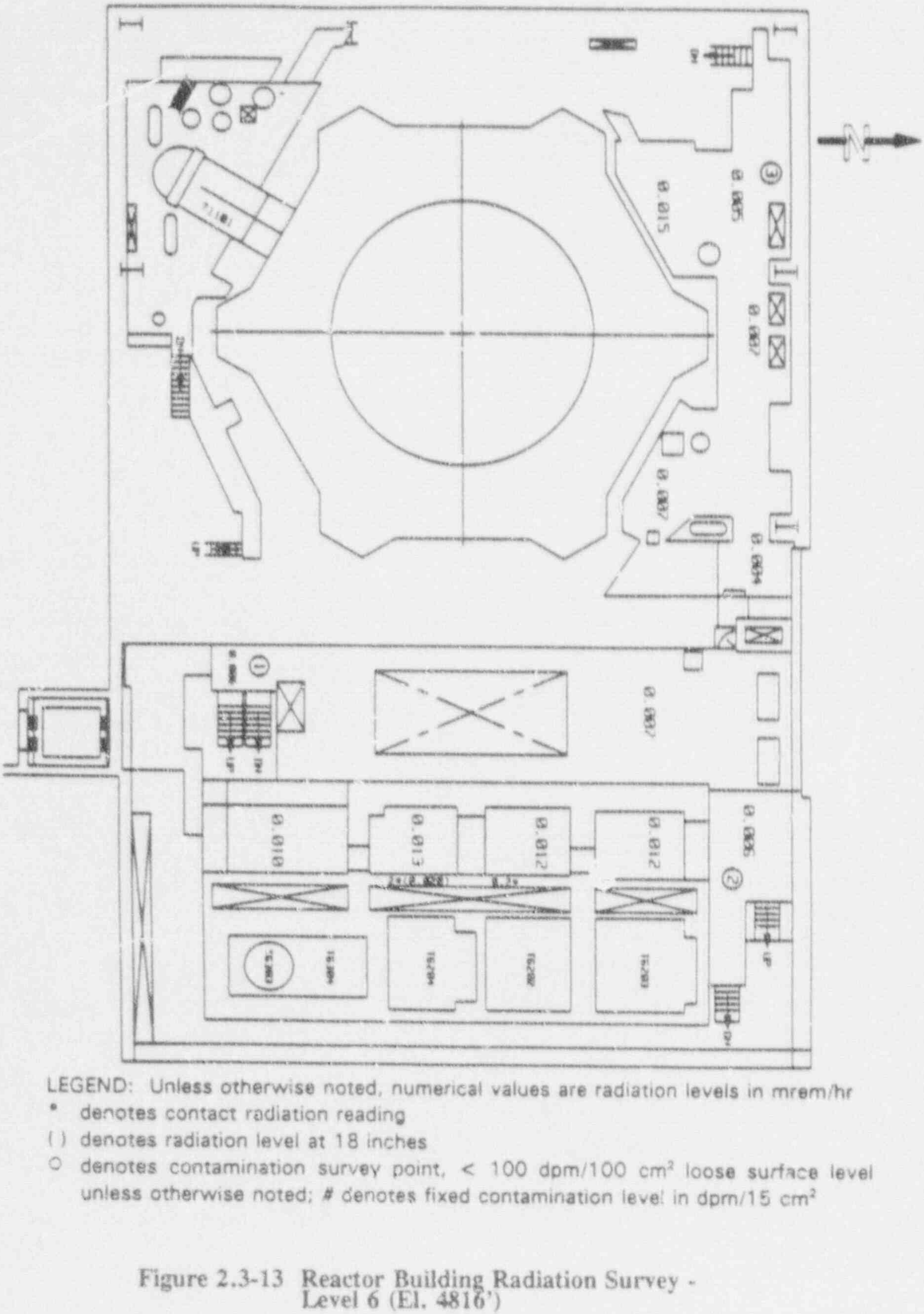
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-11 Reactor Building Radiation Survey - Level 4 (El. 4781')



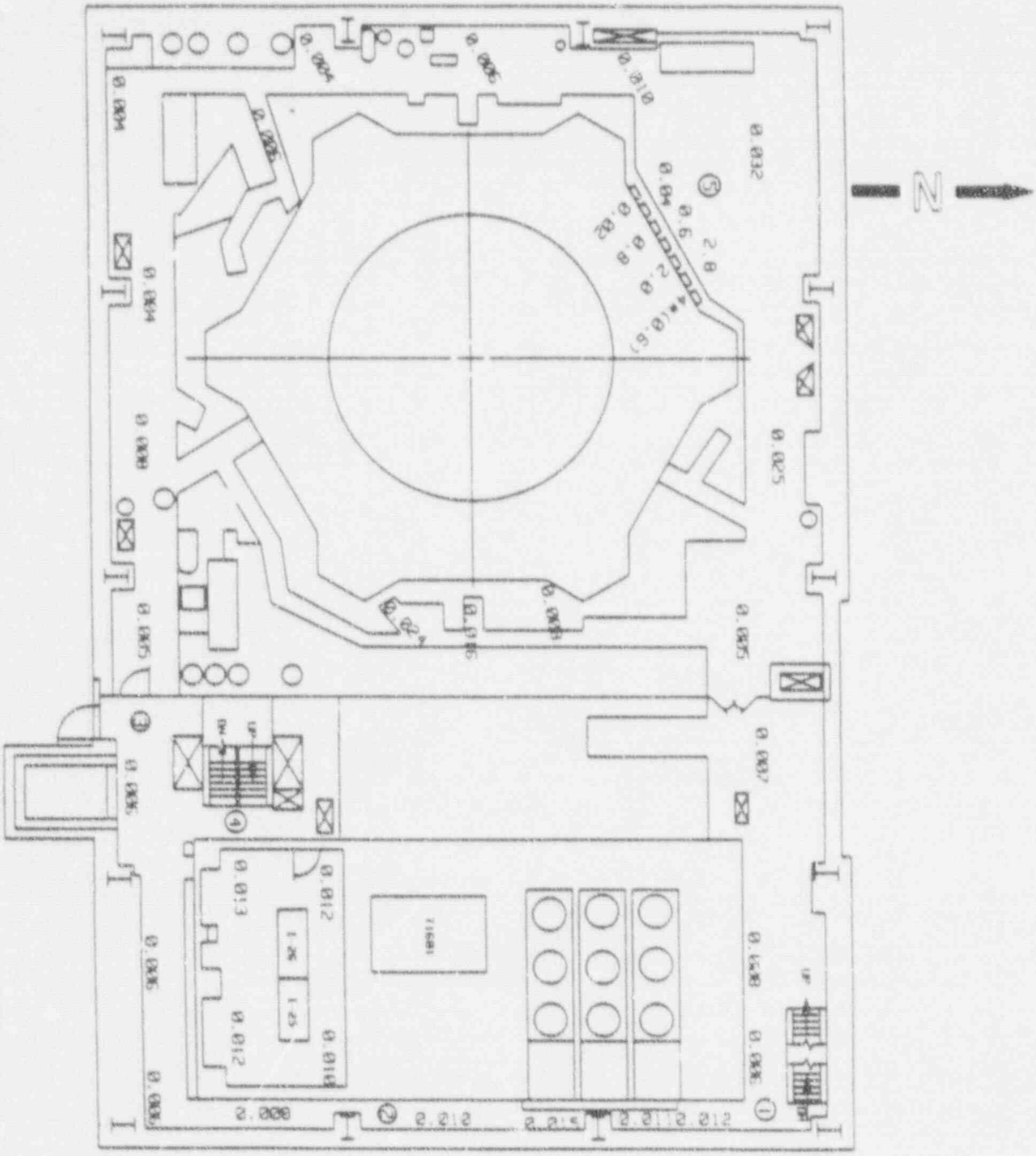
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
- * denotes contact radiation reading
- ( ) denotes radiation level at 18 inches
- ○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-12 Reactor Building Radiation Survey - Level 5 (El. 4791')



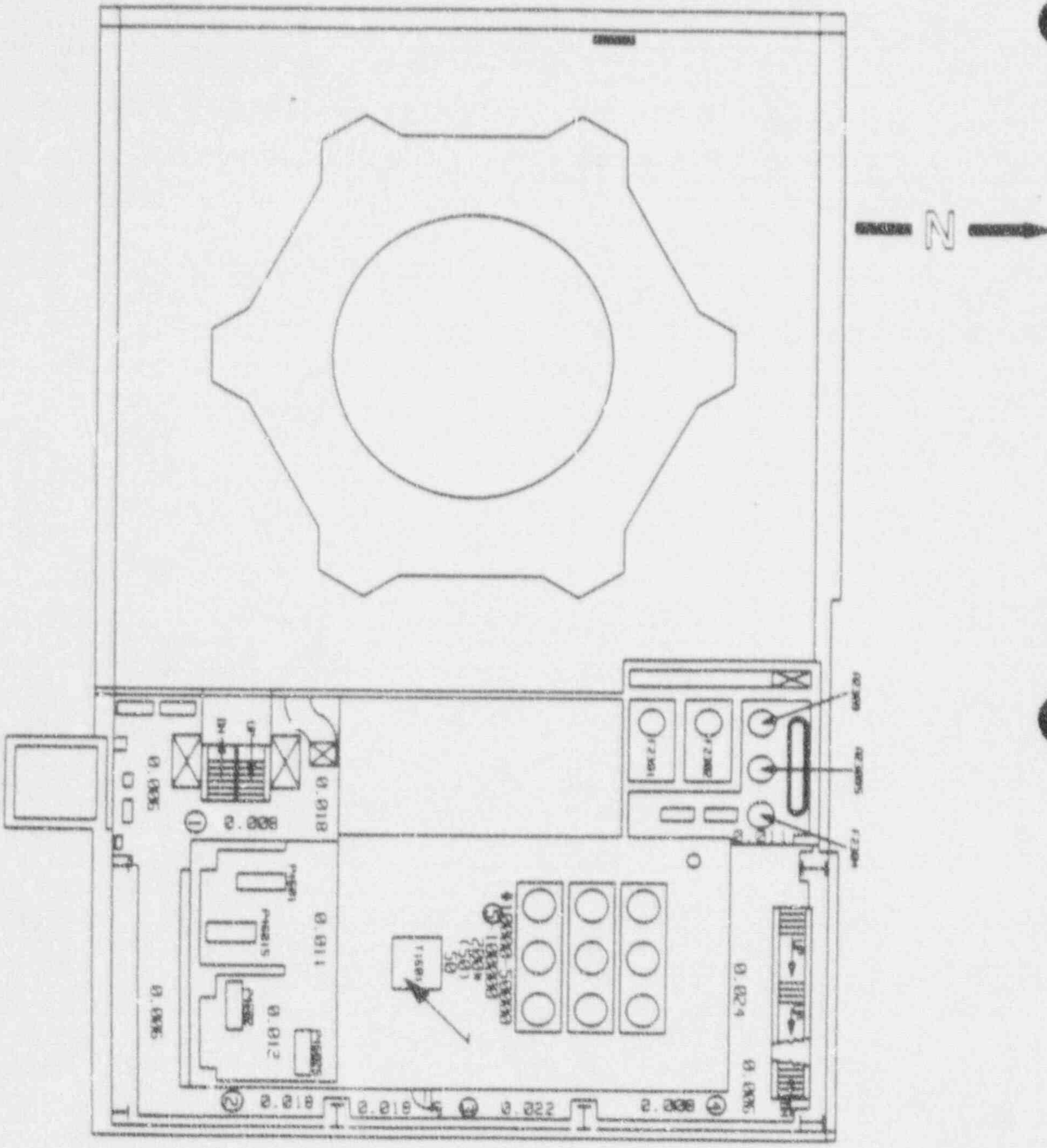
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
- * denotes contact radiation reading
- ( ) denotes radiation level at 18 inches
- ○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-13 Reactor Building Radiation Survey - Level 6 (El. 4816')



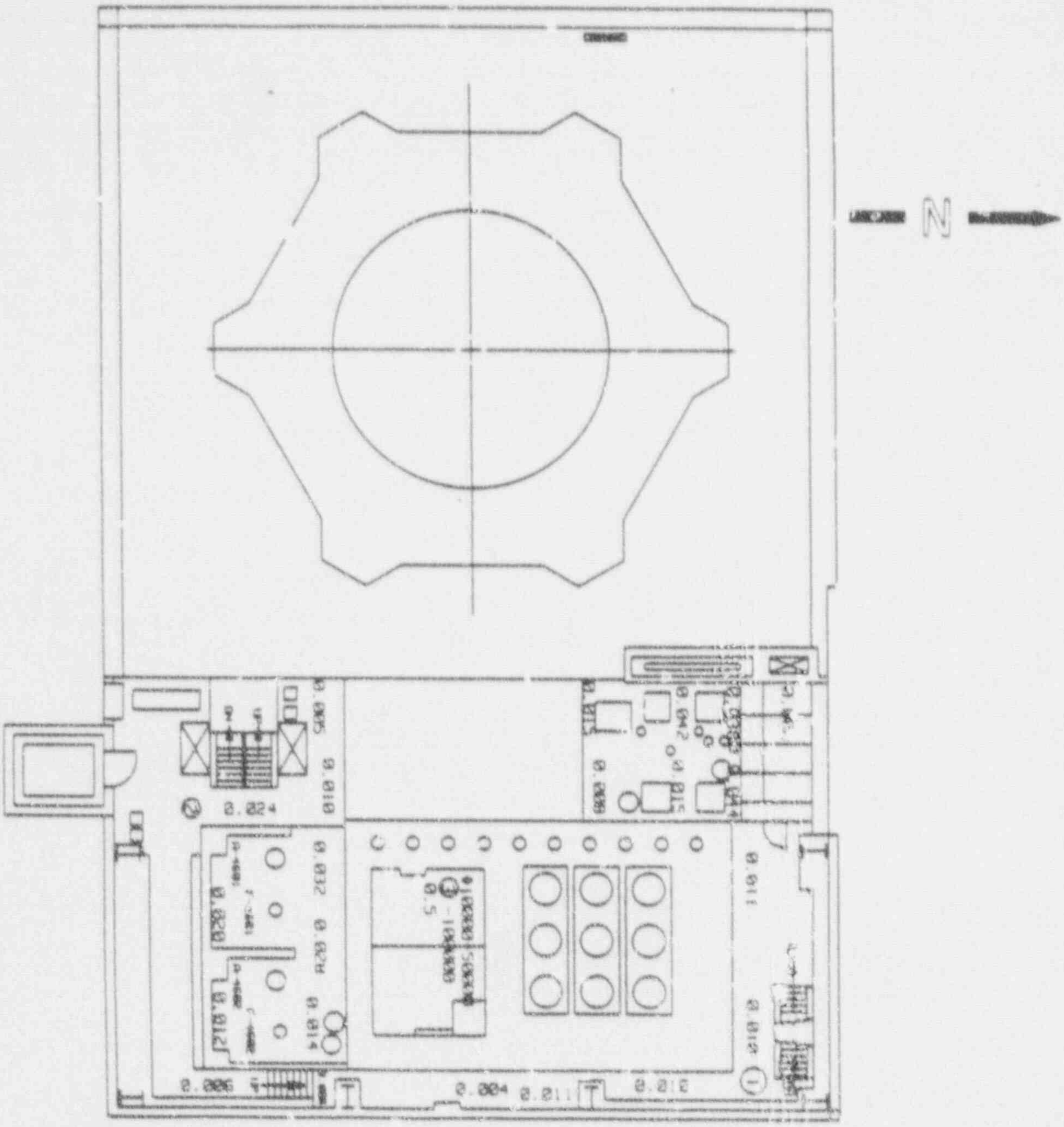
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm^2 loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm^2

Figure 2.3-14 Reactor Building Radiation Survey - Level 7 (El. 4829')



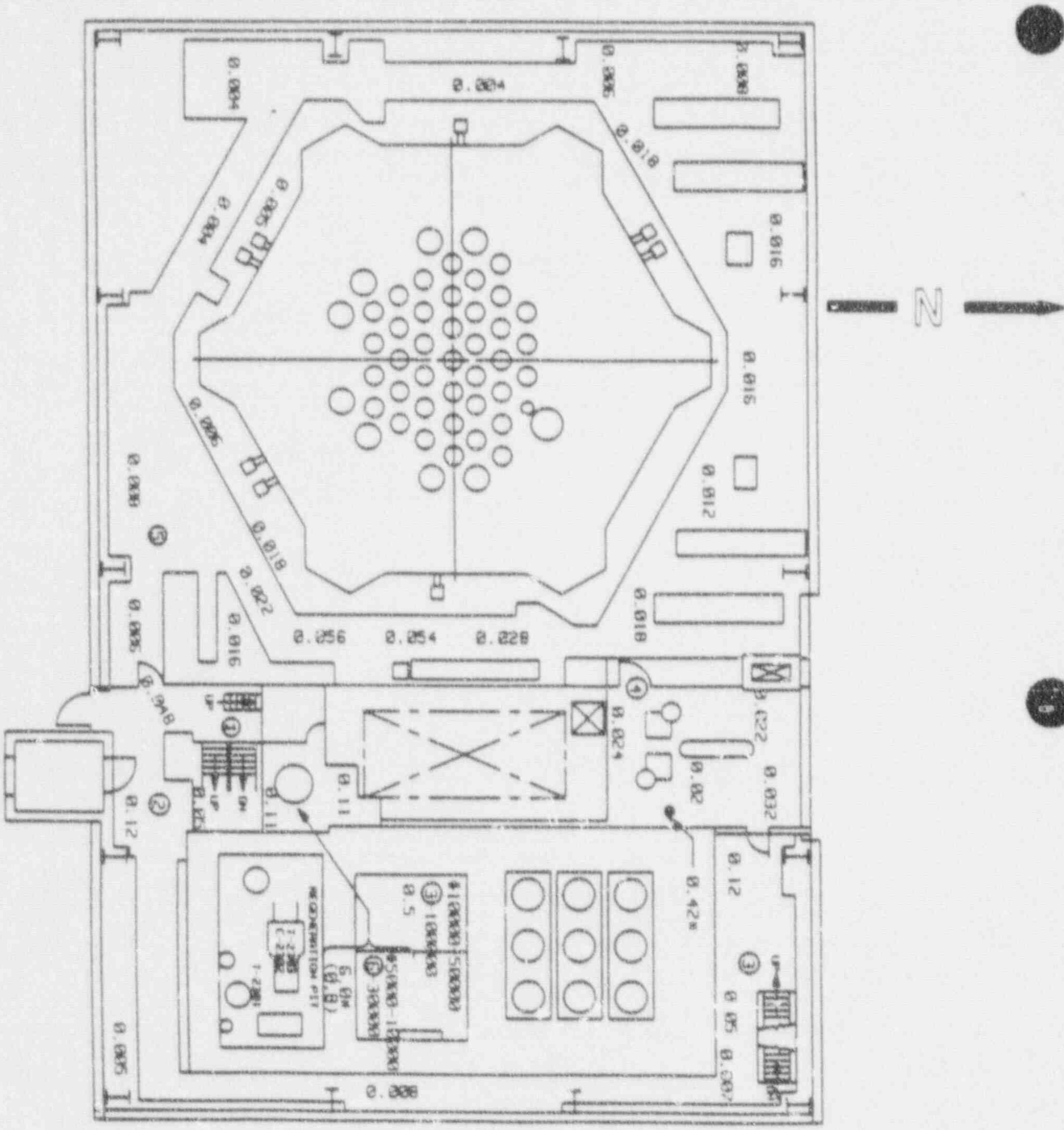
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-15 Reactor Building Radiation Survey - Level 8 (El. 4839')



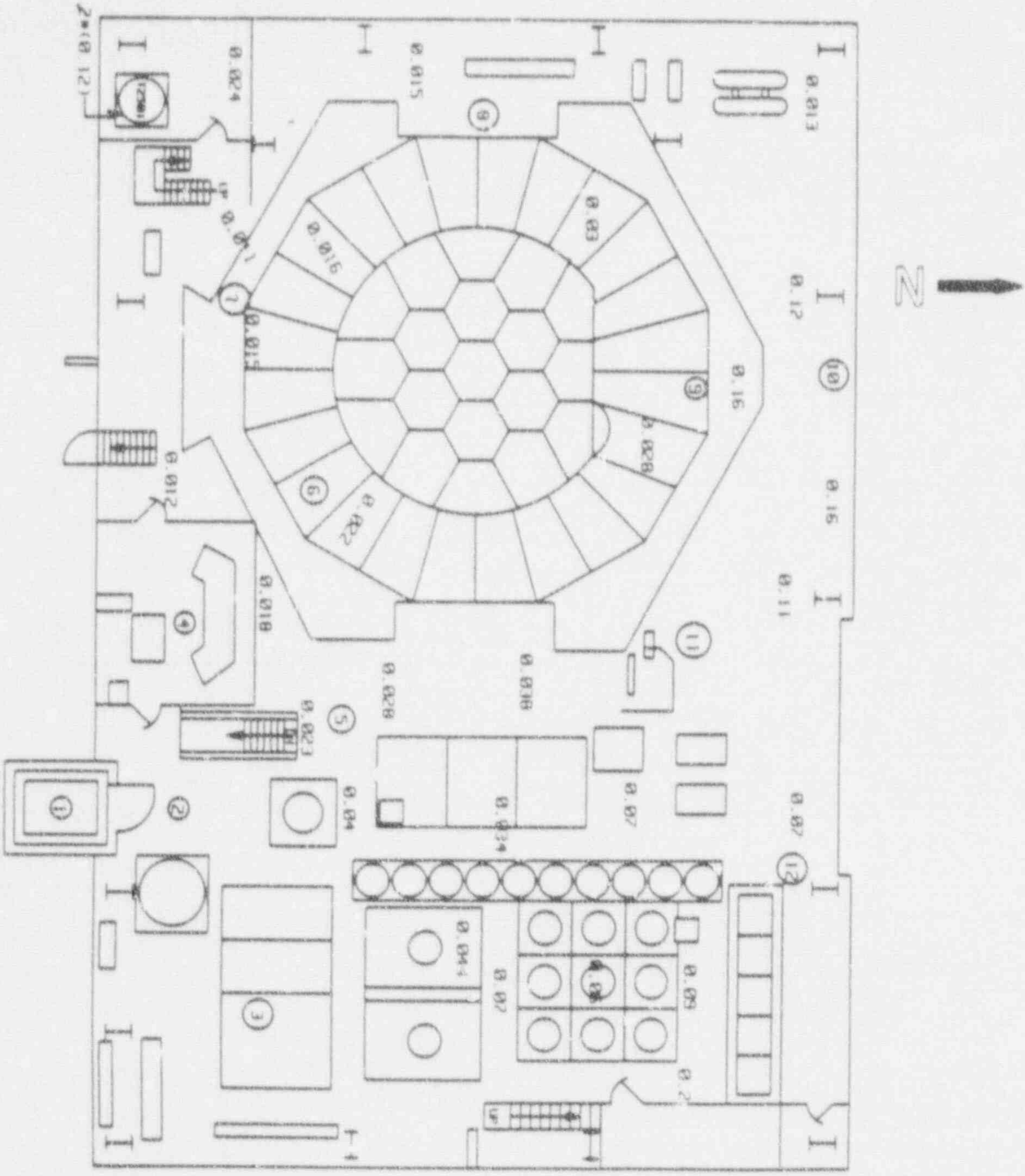
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-16 Reactor Building Radiation Survey - Level 9 (El. 4849')



LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-17 Reactor Building Radiation Survey - Level 10 (El. 4864')



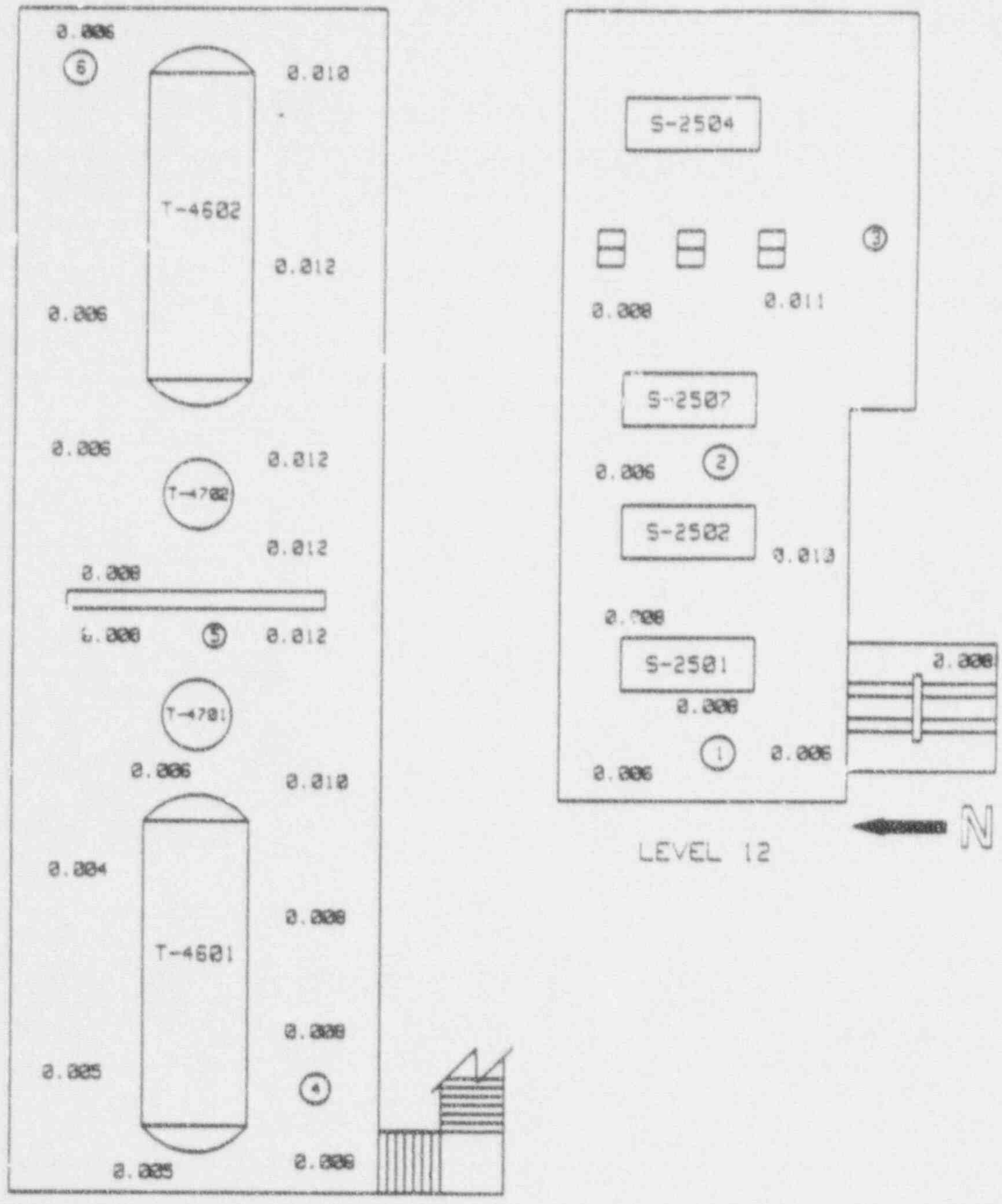
LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr

* denotes contact radiation reading

( ) denotes radiation level at 18 inches

○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-18 Reactor Building Radiation Survey - Level 11 (El. 4881') Refueling Floor



LEGEND: Unless otherwise noted, numerical values are radiation levels in mrem/hr
* denotes contact radiation reading
( ) denotes radiation level at 18 inches
○ denotes contamination survey point, < 100 dpm/100 cm² loose surface level unless otherwise noted; # denotes fixed contamination level in dpm/15 cm²

Figure 2.3-19 Reactor Building Radiation Survey - Levels 12 & 13 (El. 4904' & 4921')

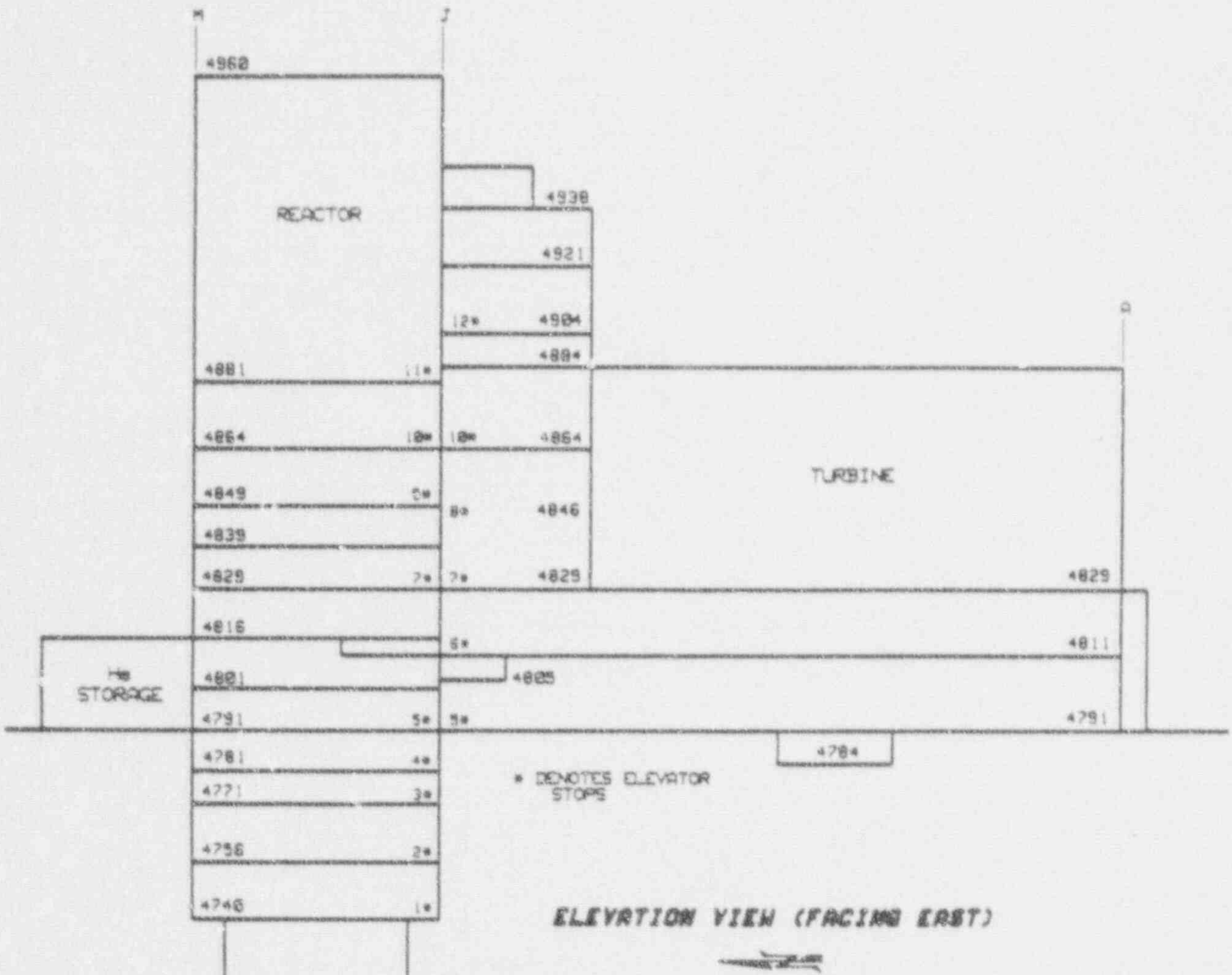


Figure 2.3-20 Turbine and Reactor Building Elevations

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3.0 PROPOSED ACTION

3.1 Introduction

Pursuant to 10 CFR 50.82, PSC has submitted a Proposed Decommissioning Plan (PDP, Reference 1) to the NRC for review and approval. PSC has selected the DECON option for decommissioning Fort St. Vrain and is proposing the immediate dismantlement and decontamination of the activated and contaminated portions of the Fort St. Vrain Nuclear Generating Station. The decommissioning of nuclear facilities is a regulated process, whereby radioactive material will be removed from the plant site, the site will be decontaminated to limits established by the NRC to allow release of the site for unrestricted use and the NRC license will be terminated.

To accomplish the decommissioning of Fort St. Vrain, substantial portions of the existing plant will be decontaminated, dismantled and/or removed to release all site areas for unrestricted use. However, the site will not be completely torn down and restored to agricultural use or a natural environment. Components and structures which are not activated or radioactively contaminated or which have been decontaminated will remain and may be utilized in the conversion of the balance of plant (BOP) systems to a natural gas-fired power plant.

An overview of the decommissioning work scope is:

1. Remove the Prestressed Concrete Reactor Vessel (PCRV) internal radioactive components remaining after the defueling of the reactor.

2. Dismantle those portions of the PCRV structure and radioactive balance-of-plant systems which exceed limits for release of residual radioactive materials. Decontaminate those structures or systems where only low level surface contamination exists.

3. Ship all radioactive waste offsite for disposal.

4. Perform a final site radiation survey to confirm that all site areas can be released for unrestricted use.

5. Terminate the 10 CFR 50 operating license.

The detailed dismantling activities that will be performed during decommissioning are described in Section 2.3 of the PDP (Reference 1).

3.2 Site Cleanup, Final Site Survey and Site Release for Unrestricted Use

Upon completion of all PCRV dismantling operations and removal of the radioactive wastes from the site, the remaining portions of the site will be cleaned-up by performing the following major activities:

- Contaminated tools and equipment will be cleaned for reuse or packaged for disposal.
- All Surface Contamination will be removed to the levels required for release for unrestricted use.
- Radioactive liquid, gas and solid waste systems will be removed and disposed.
- A final radiation survey of the site and environs, to verify that all radioactive material has been removed, will be performed.
- A final decommissioning program report and request for termination of the Part 50 License will be prepared and submitted to the NRC.

As discussed in Section 3.3, materials and equipment released from the site for unrestricted use will be released in accordance with regulatory requirements. If contamination of materials and/or equipment is found below the surface, the material will be disposed of as radioactive waste unless it is determined to be cost effective to decontaminate to unrestricted use criteria.

All temporary structures and modifications to permanent site structures will be removed or returned to their original condition, as required. The potentially contaminated structures inside the controlled areas, such as platform scaffolding will be decontaminated to release limits for unrestricted use or shipped offsite as low-level waste. Temporary structures such as the laundry and respirator cleaning trailers will be decontaminated as required and returned to their licensed vendor. The contaminated materials will be packaged and shipped to a volume reduction facility, or to a burial site for final disposition. The trailers used for personnel occupancy will be surveyed and decontaminated, if necessary, for release and shipment offsite.

After all activated systems have been removed from the site, the remaining structures will be surveyed. All contaminated surfaces such as floors and sumps will be decontaminated to remove residual contamination. The structures will be surveyed to ensure all activated and contaminated material has been removed.

The provisions of Regulatory Guide 1.86 (Reference 2) will be used for residual fixed and removable contamination on the surface of materials, equipment and facilities left at the site. In addition, radiation levels of 5 μR/hr above background measured at a distance of 1 meter will be used as the upper limit for external radiation levels. The criteria for total concentrations of radioactive materials above background in soil will be based upon those established in NUREG/CR-5512 (Reference 3). The use of this methodology will ensure an average total effective dose equivalent of less than 10 mrem/yr to an individual in a given population group.

Pursuant to Regulatory Guide 1.86 (e), a comprehensive final radiation survey will be conducted at the facility upon completion of all activities. The goal of the survey is to ensure that ambient radiation levels and surface contamination levels are below the limits specified in Regulatory Guide 1.86 and to verify that the site can be released for unrestricted use by meeting the acceptance criteria of 5 μR/hr above background at 1 meter from the surface.

The survey will cover all pertinent structures, surfaces, systems and components, focusing on those items identified as potentially troublesome during the pre-decommissioning and during the decontamination/dismantling phases. The survey will include:

- PSC property, soil, stream, and pond sediment, and water sampling outside the fenced area for radioisotopic analysis

- PSC property soil, lagoon sediment, and water sampling inside the fenced area for radioisotopic analysis

- Radiological surveys for the PCRV and reactor building

NUREG/CR-2082 (Reference 4), "Monitoring for Compliance with Decommissioning Termination Survey Criteria" will be used as a guide in developing the comprehensive survey program for quantifying residual radioactivity and radiation levels.

The current Radiological Environmental Monitoring Program (REMP) will be continued in part specifically tailored to accurately monitor the environmental radiation and radioactivity levels, and to determine the effect on the radiological conditions of the environment due to decommissioning activities. The current REMP contains provisions for sampling and analyzing airborne, waterborne, ingestion, and direction radiation exposure pathways. The results of the REMP will be included in the final radiological survey report. In addition, the Offsite Dose Calculation Manual (ODCM) will provide the methodologies to assure compliance with the 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 and Appendix I, and 40 CFR 190.

At the completion of the decommissioning effort, PSC will prepare a final project summary report. The report will document the decommissioning activities that were performed and will be submitted with the application for termination of the Part 50 License and along with the final radiological survey report, will provide the basis for the approval to terminate the license.

3.3 Criteria for Release of Equipment and Non-Radioactive Waste

The guidance provided in NRC Circular 81-07, "Control of Radioactively Contaminated Materials" (Reference 5) will be used to ensure appropriate survey methods are employed for the release of decontaminated items (e.g., tools and equipment) and scrap materials. The guidance provided in NRC IEN 85-92, "Surveys of Wastes before Disposal from Nuclear Reactor Facilities" (Reference 6) will be used to ensure appropriate survey methods are employed for the monitoring of segregated waste prior to disposal in a sanitary landfill. Procedures that have been prepared with reference to NRC Circular 81-07 and NRC IEN 85-92 will be used for surveying material to be released off-site.

3.4 Radioactive Waste

During the decommissioning activities radioactive materials (radwaste) in liquid, solid and gaseous forms are expected to be generated. Management of these wastes is an integral part of the decommissioning plan and includes provisions for minimizing the quantity of waste generated, waste collection treatment, volume reduction, packaging and off-site shipment for disposal.

Some solid materials may be volume reduced onsite using the existing Fort St. Vrain compactor, a mobile compactor, or other equipment as needed and as supplied by the decommissioning contractor (Westinghouse). If a onsite compactor is used, its effluent discharge will be filtered by HEPA ventilation and airborne contamination surveys will be conducted during operation. Compactor operation will be controlled by a specific radioactive waste procedure to ensure safe operation. Compacted waste will be placed in appropriate containers such as 52- or 55-gallon drums, compacted, overpacked if required, and shipped offsite for disposal or additional volume reduction. In order to reduce disposal quantities, some solid materials may also be shipped offsite for decontamination. All activities will be performed in accordance with applicable radiation protection and radioactive waste procedures.

Supercompaction, incineration, and melting for volume reduction will be accomplished offsite at a facility licensed for these specific waste volume reduction techniques.

3.4.1 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.

Materials will be categorized as follows:

1. Potentially contaminated or requiring minor spot decontamination: Materials that, on a gross basis, 1) appear to be uncontaminated, 2) have geometries with all surfaces easily accessible, and 3) have a small surface area-to-weight ratio will be surveyed to determine if the material can be released 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 that have good geometries for purposes of surveying and decontamination, and that possess a low surface area-to-weight ratio, may be shipped directly to a licensed facility for disposal or decontamination of the surfaces and subsequent release for unrestricted use or decontaminated on site to the criteria for release for unrestricted use.

3. General contamination/inaccessible surfaces/high surface-area-to weight ratio: Smaller metallic-scrap or metals that have poor geometries 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 or other non-recoverable materials: activated materials and high specific activity materials will either be packaged and shipped directly to a disposal facility or to a licensed facility for further processing and volume reduction.

Radioactive materials identified above will be evaluated to determine the optimum method for release, decontamination, or shipment off-site for further processing or for burial, generally as described in Tables 3.4-1 and 3.4-2. Items not considered for decontamination or items that, following decontamination, are considered to have too high a specific activity for off-site volume reduction, will be packaged and shipped directly for disposal at a disposal facility. Greater than Class C (GTCC) wastes, if any, will be packaged and stored in the adjacent Independent Spent Fuel Storage Installation (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.

3.4.2 Projected Waste Generation

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) Systems. 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.

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	70,768
B	28,293
C	1,011

Due to uncertainties in the analysis, as much as 400 cubic feet of Class C wastes may be reclassified as GTCC. Waste class and volume estimates may change as ongoing planning and decommissioning operations proceed. Tables 3.4-1 and 3.4-2 identify the radioactive wastes that may be shipped for further processing. The pre-volume reduction totals and number of waste containers are delineated on Tables 3.4-3 and 3.4-4.

There will be approximately 20,000 cubic feet of uncontaminated concrete and metallic scrap. This will be transported by truck to local landfills for disposal or to commercial scrap facilities for re-use as scrap materials.

3.4.3 Waste Treatment

3.4.3.1 PCRV and Internal Components

The major decommissioning task is the dismantlement and decontamination of the radioactive portions of the PCRV. This activity is described in detail in Section 2.3 of the PDP. In general, it involves removing a plug of concrete from the top of the PCRV to gain access to the PCRV cavity, and then removing the core components and other equipment and structural materials located inside the PCRV. Upon removal from the PCRV, concrete sections and large equipment items will be moved to a waste processing area for further sectioning, segregation, and preparation for disposal. Radioactive waste packaging 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 in Reactor Building
- Fuel Storage Building
- Temporary facilities designed for waste packaging

Off-site facilities will be utilized when necessary and practical for waste processing and final packaging.

3.4.3.2 Steam Generators

Because of the high contact dose rates that are anticipated in the steam generator primary assembly (economizer, evaporator, and superheater sections), a special shielded shipping container will be required. The following methodology will be employed to remove and ship the steam generator primary assembly.

As the primary assemblies are lifted from the PCRV by the Reactor Building crane, the outer shroud and tube 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.

The steam generator primary assemblies will then be placed in a shipping container, located in the truck bay. The shipping container will consist of a metal culvert section seven foot in diameter by 27 feet long. The culvert section will be cut in half lengthwise to provide a hollow half-cylinder. Structural supports will be welded to the half section of culvert to provide structural support and allow the culvert to be upended with the primary module inside.

Support saddles will be mounted inside the culvert and serve a dual purpose. First, the saddles provide a means of attaching the steam generator primary assembly to the culvert and transmitting the load to the structural supports on the outside of the culvert. Second, the saddles will keep the steam generator primary assembly centered in the culvert with an annular space of about 8 inches between the inside diameter of the culvert and the outside diameter of the steam generator primary assembly.

If required, the annular portion of the steam generator between the shroud and the cold reheat piping will be filled with grout which will encapsulate the tube bundle of the steam generator. In addition, grout may be pumped into the feedwater and steam tubes of the primary assembly. If necessary due to the high contamination levels, the 8 inch annular region between the outside of the steam generator shroud and the inside of the culvert will be filled with grout for shielding.

The combined weight of the shipping container, steam generator assembly, and grout will be approximately 195,000 pounds. If actual contamination levels in the steam generator primary assemblies are lower than expected, the shielding grout in the annular space between the steam generator shroud and the container may be omitted with a weight savings of about 56,000 pounds. A final radiological survey will be performed to ensure DOT requirements are met before the steam generators are released for shipment. The steam generators will be shipped by rail for disposal at a licensed burial site.

3.4.3.3 On-site Processing of Liquid Wastes

Radioactive liquid wastes which exceed 10 CFR 20 limits for the general public will not be released to the environment. To the maximum extent practicable, prior to discharge to the environment, the liquid waste will be processed to decrease the concentrations of radioactive material to levels below the limits established in 10 CFR 20. Processing will include filtering, demineralization and/or dilution. The liquid effluent will be monitored to assure that the liquid effluent released to the environment does not exceed the 10 CFR 20 limits. The method for processing on-site liquid wastes is described in Section 4.2. This method provides for proper dilution while also allowing a liquid effluent release rate that is greater than the 10 gpm value previously described in the Final Environmental Statement (Reference 7).

Liquid wastes will be processed and disposed of in accordance with written approved procedures. Typical liquid waste expected includes oils from plant systems, water from PCRV bleed and feed operations, and sludges from diamond wire cutting and sump clean-outs. Disposal of contaminated oils is expected to be accomplished by transfer of the oil to a licensed vendor for incineration. The chemical/hazardous nature of oils will be known prior to transfer for incineration. Water from the PCRV will be processed through the PCRV Shield Water System and discharged through normal plant effluent systems.

| 3.4.3.4 Hazardous Materials and Mixed Wastes

Chemicals used for decontamination will be evaluated for hazardous constituents using 40 CFR and the Material Safety Data Sheets (MSDS). Decontamination chemical wastes expected 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 will 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 wipers. The wiping material will be dried prior to packaging for disposal or volume reduction.

All hazardous chemicals and materials will be subjected to a chemical control review to determine if a non-hazardous or a less toxic chemical can be substituted to prevent generation of mixed wastes. In the event that hazardous chemicals or materials must be used, procedures will ensure that all waste minimization techniques will be applied during usage. Steps will be taken to ensure that if hazardous materials must be used, the necessary controls will be in place so these materials will not inadvertently become radioactively contaminated. If some hazardous material does inadvertently become radioactively contaminated, it will be considered as mixed waste and subject to applicable regulations.

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 de-list any mixed waste. However, if PSC determines it is necessary to de-list any mixed waste, the procedures outlined in 40 CFR 260.20 and 260.22 will be used to exclude that waste form from regulations.

| 3.4.3.5 Airborne Contamination/Gaseous Effluent

Some low-level airborne wastes may be generated during decommissioning. Disposal of airborne radioactive wastes will be accomplished by filtration (of particulates) and disposal of the filter as solid waste

| Tritiated water vapor will be created due to evaporation of PCRV shielding water and
| will be released as gaseous effluent. Much of this tritiated water vapor will be
| drawn into the Reactor Building ventilation system from under the PCRV work
| platform. Air will be drawn down through the PCRV work platform access
| openings, over the surface of the tritiated shield water. The water vapor that

becomes entrained in the ventilation air will then be exhausted out the Reactor Building ventilation exhaust stack.

Any gaseous effluent in the Reactor Building flows through the existing ventilation exhaust system. 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 system consists of three trains, one of which is normally in continuous operation. The design flow rate for each train is 19,000 cfm. Allowing 10% for degradation, the minimum flow rate is 17,100 cfm. One train is sufficient to maintain the Reactor Building subatmospheric and thereby minimize unfiltered fission product release from the building. With only one exhaust fan operating, the ventilation system controls will throttle fresh air supply to the air handler in order to maintain the building pressure subatmospheric. The ventilation exhaust system is equipped with roughing filters, to capture large size particles, and with high efficiency particulate air filters (HEPAs) with a specified capacity to remove 99.9% of particulates greater than 3 microns in diameter. HEPA filters will be changed upon high radiation levels reading/alarm in the exhaust duct, or based on maximum pressure differential readings indicating that the filters are filled with dust. Radiation monitoring is provided for the exhaust air to atmosphere and readings above prescribed limits are alarmed.

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.

Any ventilation units serving localized confined enclosures will be mobile type having in-line HEPA filters and connected with flexible ducts to the enclosure and to the exhaust duct.

3.4.3.6 Piping

Contaminated piping will be packaged as LSA material in strong tight containers. The piping will be sectioned to fit into commercially available steel LSA boxes. Within the boxes, when feasible, smaller bore pipe will be nested within larger bore pipe to enhance packing efficiency. In addition, when practical the boxes will be topped with lighter dry active waste such as paper, plastic and clothing to maximize the box weight limits.

3.4.3.7 Dry Active and Solid Waste

Material such as barrier plastic sheets, rags, paper and protective clothing will probably be compacted and packaged.

Sludges will be dewatered and dried, or solidified/absorbed using disposal site approved solidification/absorbent media. Solidification, if required, will be controlled by an approved Process Control Program (PCP).

Solid wastes will be processed in accordance with written procedures. A general plan for solid waste processing is as follows:

1. The waste will be initially identified at the point of generation as to the type of material and exposure rate.

2. The material will be segregated to allow for decontamination on-site, shipped to an off-site vendor for volume reduction or packaged for disposal at an approved disposal site.

3.4.3.8 Equipment and Tools

This includes such items as saws, jack hammers, monorail, forklift, shovels, pumps, tanks, ventilation system components, filters, piping, etc. Only some, and parts of some of this equipment are expected to be discarded as radwaste. A determination of volumes of solid radwaste generated from this category will be possible only during the clean-up task when final measurement of contamination level and evaluation for decontamination will be made.

3.4.4 <u>Radioactive Waste Disposal</u>

The radioactive waste disposal program will follow the regulations established in 10 CFR 20 and 10 CFR 61, the disposal site criteria, and other applicable Federal and State regulations. Most packaged waste will be put in strong tight metal containers suitable for Class A Low Level Waste (ALLW). Examples of the waste containers that may be used are drums (52-gallon, 55-gallon), boxes (2'x4'x6', 4'x4'x6'), liners, HIC's, sea/land containers, casks, and specialty containers. The capacity and weight limitations are governed by the activity levels and classification of the enclosed materials as specified in 49 CFR and 10 CFR 71. The waste container will be determined by the size, weight, classification, and activity level of the different types of materials. Guidance for this activity will be contained in radioactive waste procedures.

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 will be 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. The use of a super compactor may also be an 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.

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.

In general, radioactive waste will be packaged and shipped off-site for disposal on a continuous basis. Long-term storage of waste is not anticipated. Packaged waste ready for shipment will be temporarily stored onsite so that shipping loads can be maximized on a practical basis.

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 IACM Building (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 the accident analysis, Section 3.4 of the PDP (Reference 1).

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 Generic Letter 81-38 and the Standard Review Plan (NUREG-0800), Appendix 11.4-A. The Fort St. Vrain Possession Only License and 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 Generic Letter 81-38. 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 will also be used to store packaged dry and solidified low-level radwaste. The Fuel Storage Building is a single level concrete structure which is designed to withstand a 202 mph tornado wind and can withstand the design basis tornado missile. 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, fire detection systems 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 protected 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.

All interim low-level radwaste storage locations described above exist within the plant's protected area. Radiation protection procedures will be implemented that will specify requirements for processing and packaging radwaste. Procedures will also be implemented to provide suitable instructions to establish radiologically controlled areas to ensure occupational and public exposure are kept within the requirements of 10 CFR 20, 10 CFR 100, and 40 CFR. Procedures will be implemented to monitor storage areas periodically and to check waste container integrity.

3.4.5 Transportation Plan

Before shipment from Fort St. Vrain, each package will be inspected to determine its adequacy for retaining the radioactive contents and its proper condition for shipment. Bar codes capable of being read by computerized scanners will be affixed to the container and the corresponding lid.

Waste will be packaged into waste packages that meet 49 CFR and 10 CFR 71 requirements. Certain wastes may require use of an approved shielded shipping cask due to radiation levels or limits for quantities of radioactivity. Trucks will be the primary mode of transportation during this decommissioning project. It is anticipated that the shipping containers with the steam generators and grout will be shipped by rail for disposal.

Transportation surveys and documents will be prepared prior to any shipment offsite. Analyses will be performed to determine isotopic inventory and concentration for classification.

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 regulations. Trained personnel will inspect and oversee shipping, in accordance with DOT and NRC regulations.

During decommissioning operations, PSC does not expect that truck shipments will exceed normal highway axle load limits. It is planned that the large concrete blocks being removed from the PCRV will be further cut and reduced in size, first so that only the radioactive portion of the concrete is disposed of as radioactive waste, and secondly so that the concrete blocks will be of a small enough size for truck haulage within normal axle load limits. Trucks meeting the highway axle load limit requirements will be used to carry heavy loads such as concrete blocks or steam generator packages.

The large steam generator packages are planned to be shipped by rail, although special heavy hauling equipment may also be used. If the steam generators are shipped by rail, the rail spur near the plant will be refurbished.

The onsite road within the Fort St. Vrain protected area (that leads to the Reactor Building Truck Bay) is a paved road and is in good physical condition. No additional improvements to this onsite road are necessary. The present access roads from the Fort St. Vrain site to the main interstate highway (I-25) are paved roads, in good physical condition, and are well maintained by Weld County and the State of Colorado. These access roads have been in continuous use during the life of the Fort St. Vrain facility and have seen considerable truck traffic over the routes.

Transportation of decommissioning radioactive waste will not involve any additional considerations beyond those for transportation of existing radioactive waste material. Existing regulations for transporting radioactive waste material are covered under NRC regulations in 10 CFR 20, 71 and 73 as well as DOT regulations in 49 CFR 170-189.

In accordance with 10 CFR 71.5(a) all off-site waste shipments will meet the applicable DOT requirements of 49 CFR 170 - 189. This includes:

- Shipments will be loaded by the consignor (licensed shipper) and unloaded by the consignee,

- Licensee-shipper will be responsible for assuring that the shipment is properly loaded and secured.

3.5 Asbestos

The Fort St. Vrain Plant was completed in the early 1970s and continuing maintenance and modification have been going on since that time. As a result, there is asbestos containing material in the plant. Thus, all decommissioning activities must recognize this potential and appropriate precautions taken to assure the health and safety of the workers and the public.

As part of the preparations for the initiation of actual decommissioning activities, a characterization survey was performed of those systems and areas that will be involved with the dismantlement and decontamination of the plant. That characterization found that 60 out of 155 samples taken contained asbestos. Asbestos is associated almost entirely with steam generator systems, helium circulator systems, the helium purification system, and the radioactive gas waste system. None of the samples were radiologically contaminated. Thus, asbestos is not expected to present a major problem for decommissioning.

During decommissioning, only that asbestos that is associated with the decontamination, dismantlement, and removal of radioactive systems and components will be disturbed and removed.

All activities involving asbestos will be conducted in accordance with federal and state regulations (OSHA 29 CFR 1910 and 1926, EPA 40 CFR 61 Subpart M). An Asbestos Removal Specification will be developed for the site consistent with the National Institute of Building Sciences Format. All asbestos removal work will be performed by a competent asbestos removal contractor, with appropriately trained and certified personnel. All asbestos will be packaged for shipment and disposed of at an authorized disposal site.

It is not expected that any of the asbestos will be contaminated with radioactivity. However, if any is found that has radioactive contamination, it will be disposed of at a commercial radioactive waste disposal facility. At the present time, the operators of the low-level radwaste disposal facilities near Beatty, Nevada and Richland, Washington accept asbestos for disposal.

3.6 Manpower Levels and Schedule

The individual tasks making up the decommissioning effort have been delineated using a work breakdown structure (WBS) approach. Figure 3.6-1 is a schedule of the major decommissioning tasks which include PCRV dismantling and decontamination, and balance of plant (BOP) 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. The planning phase occurs over an 18 month period and the actual dismantling and decommissioning activities at the site are planned for a 39 month period. The workforce will consist of a combination of PSC and Westinghouse Team personnel and is estimated at 300 people. This manpower level does not include personnel involved in conversion activities.

3.7 References

1. Proposed Decommissioning Plan for the Fort St. Vrain Nuclear Generating Station, Public Service of Colorado.

2. USAEC Regulatory Guide 1.86 "Termination of Operating Licenses for Nuclear Reactors,"June 1974.

3. NUREG/CR-5512 "Residual Radioactive Contamination from Decommissioning." January 1990. Draft Report for comment.

4. NUREG/CR-2082, "Monitoring for Compliance with Decommissioning Termination Survey Criteria." Draft Report, January 1990.

5. NRC Circular 81-07, "Control of Radioactively Contaminated Materials." May 1981.

6. NRC Information Notice 85-92, "Surveys of Wastes Before Disposal from Nuclear Reactor Facilities." December 1985.

7. Final Environmental Statement Related to Operation of the Fort St. Vrain Nuclear Generating Station of Public Service Company of Colorado, August 1972.

TABLE 3.4-1
PCRV WASTE CLASSIFICATION AND VOLUME REDUCTION

COMPONENT	A	B	C (1)	STABILIZE	INCINERATE	COMPACT	MELT	OVERFILL
Region constraint devices	X					X		
RCD Pins			X	X				
Metal Control Rod Reflectors			X	X				
Metal Block - Non-control rod		B		X				
Defueling Elements	X				X	Y		
Top reflector graphite blocks	X							
Bottom reflector graphite blocks	X							
[illegible] reflector graphite blocks	X							
[illegible] side reflector blocks		X		X				
[illegible] reflector blocks		X		X				
Upper reflector keys	X					X	X	
Slot spacer blocks w/boron pins								
Boron Pins		X		X				
Blocks with pins removed	X					X		
Bottom ref. blocks w/Hastelloy cans								
Hastelloy cans			X	X				
Blocks without Hastelloy cans	X							
Lower reflector keys		X		X				
Core support blocks	X					X		
Core support posts	X				X	X		
CSF support columns	X					X	X	
Metal on large side reflector	X					X	X	
Misc steel beneath CSF	X					X	X	
Core barrel	X						X	
Lower plenum insulation	X					X		
Silica Blocks	X					X		
Concrete - Top	X							
- CSF	X							
- side	X							
- Rubble	X							X
Misc. Inconel parts (CSF)	X					X		
Concrete cutting debris	X					X		
Helium purifiers (PCRV head)	X					X		
Helium diffusers	X					X		X
Helium circulators	X					X		
He Circ Shutoff valve assembly	X					X		
Helium bellows	X					X		
Steam Generators	X							
Lower Floor/appurtenances	X					X	X	
Platform/ tools /jib cranes	X					X	X	
Crane cable/drum/3 bucket inverters	X					X		
Resins	X			X				
Miscellaneous soft waste	X					X		
Reactor Isolation valve	X					X	X	

(1) Due to uncertainties in the analysis, some Class C wastes may be reclassified as Greater Than Class C wastes.

TABLE 3.4-2
CONTAMINATED BOP WASTE CLASSIFICATION AND VOLUME REDUCTION

COMPONENT	A	B	C	STABILIZE	INCINERAT	COMPACT	MELT	OVERFILL	DECON @ SITE	DECON @ WSEC
Miscellaneous soft waste	X					X				
Reactor Isolation Valves	X					X	X			
Refueling sleeves	X					X	X			
Refueling sleeve sand	X							O/F		
Sand from FSWs	X							O/F		
ATC	X								X	
ATC sand	X							O/F		
ESW sand	X							O/F		
Hot Service Facility	X								X	
HSF sand	X							STAB.		
Core Support Vent filters	X					X				
Gaseous Waste surge tanks	X									X
Gaseous waste surge tank sand	X							O/F		
Liquid drain tank	X									X
Gas waste vacuum tank	X									X
Gas waste vacuum tank sand	X							O/F		
Gas waste compressors	X					X				
Gas waste compressor sand	X							O/F		
Liquid monitor tank	X									X
Liquid waste monitor tank	X							O/F		
Liquid waste demineralizers	X					X				
Liquid waste receivers	X									X
Liquid waste receivers sand	X							O/F		
Liquid waste sump sand	X							O/F		
Liquid transfer pumps	X					X				
Liquid waste sump pumps	X					X				
Liquid waste resins	X							O/F		
Liquid waste filters	X			X						
Decon solution tank	X									X
Decon solution tank sand	X							O/F		
Decon recycle pump	X					X				
Decon chemical supply pump	X					X				
Purified helium filters	X				X					
Helium removal filter	X					X				
Helium getter units	X					X				
HVAC filters	X					X				
Small & large bore piping	X					X	X			
FHM	X								X	
FHM components	X					X				
FHM sand	X							O/F		
Solid waste compactor	X					X				
System 21 Components	X					X	X			
System 24 Components	X					X	X			
System 46 Components	X					X	X			
System 47 Components	X					X	X			
System 73 Components	X					X	X			

TABLE 3.4-3
PCRV WASTE VOLUME ESTIMATES

COMPONENT	CLASS	R/hr CONTACT	LSA	NUMBER	VOLUME (FT³)**	NO. OF CONTAINERS
Region constraint devices	C	70	No	84	235	2
RCD Pins						
Metal Control Rod Reflectors	C	300	No	37	401	3
Metal Block - Non-control rod	B	300	No	276	2025	13
Defueling Elements	A	Contam.	Yes	1482	7200	75
Top reflector graphite blocks	A	0.5	No	1215	1515	8
Bottom reflector graphite blocks	A	0.5	No	1215	1415	8
Radial reflector graphite blocks	A	0.5	No	480	1903	9
Large side reflector blocks	B	≤30	No	312	12600	50
Half-size reflector blocks	B	<30	No	312	2160	8
Upper reflector keys	A	0.1	No	24	192	2
Side spacer blocks w/boron pins		30		1152		
Boron Pins	B	60	No	309792		
Blocks with pins removed	A	<3	No	1152	2393	10
Bottom ref. blocks w/Hastelloy cans		300		276		
Hastelloy cans	C	10000	No	20061		
Blocks without Hastelloy cans	A	0.5		276	816	14
Lower reflector keys	B	1000	No	24	180	1
Core support blocks	A	<0.1	Yes	61	1468	15
Core support posts	A	<0.1	Yes	183	174	2
CSF support columns	A	<0.1	Yes	12	636	7
Metal on large side reflector	A	<0.1	Yes	24	96	1
Misc steel beneath CSF	A	<0.1	Yes		960	10
Core barrel	A	0.02	Yes	1	1400	31
Lower plenum insulation	A	<0.1	Yes		940	10
Silica Blocks	A	0.5	Yes		503	12
Concrete - Top	A	<0.2	Yes		3744	9
- CSF	A	<0.015	Yes		6240	15
- side	A	<0.01	Yes		18720	45
- Rubble	A	<0.015	Yes		706	16
Misc. Inconel parts (CSF)	A	0.4	No		415	5
Concrete cutting debris - top	A	<0.2	Yes		210	
- CSF	A	<0.015	Yes		200	8
- side	A	<0.01	Yes		325	
Helium purifiers (PCRV head)	A	0.15	Yes	10	480	5
Helium diffusers	A	<0.1	Yes	4	1752	4
Helium circulators						
He Circ Shutoff valve assembly	A	<0.1	Yes	4	192	2
Helium bellows	A	<0.1	Yes	12	1560	12
Steam Generators	A	2	Yes	12	20736	12
Thermocouples & guide tubes	B	50	No		105	1
Lower Floor/appurtenances	A	<0.01	Yes		1200	42
Platform/ tools /jib cranes	A	<0.01	Yes		576	6
Crane cable/drum/3 bucket inverters	A	<0.01	Yes		512	5
Miscellaneous Containers	A	<0.01	Yes		288	3
PCRV Water System	A	<0.01	Yes		2080	2
Resins - solidify, ship, bury	A*	15	No	20	2720	20
Miscellaneous soft waste	A	<0.01	Yes		12000	125
PCRV TOTALS					113,972	628

TABLE 3.4-4
BOP WASTE VOLUME ESTIMATES

COMPONENT	CLASS	R/hr CONTACT	LSA	NUMBER	VOLUME (FT³)**	NO. OF CONTAINERS
Reactor isolation valves	A	<0.01	Yes	5	960	10
Refueling sleeves	A	<0.01	Yes	2	192	2
Sand from FSWs	A	<0.01	Yes		750	Note 1
ESW Sand	A	<0.01	Yes		225	Note 1
Helium regeneration pit sand	A	<0.01	Yes		135	Note 1
ATC sand	A	<0.01	Yes		15	Note 1
HSF	A	<0.01	Yes		384	4
HSF sand	A	<0.01	Yes		500	Note 1
Core support vent filters	A	<0.01	Yes		15	2
Gas waste surge tanks	A	<0.01	Yes		2646	2
Gas waste vacuum tank	A	<0.01	Yes	1	980	1
Gas waste compressors	A	<0.01	Yes	2	2058	2
Liquid drain tank	A	<0.01	Yes	1	20	1
Liquid waste monitor tank	A	<0.01	Yes	1	576	1
Liquid waste demineralizers	A	<0.01	Yes	2	192	2
Liquid waste receivers	A	<0.01	Yes	2	1152	2
Liquid waste sump (sand)	A	<0.01	Yes		23	Note 1
Liquid transfer pumps	A	<0.01	Yes	2	96	1
Liquid waste sump pumps	A	<0.01	Yes	2	5	Note 2
Liquid waste filters	A	<0.01	Yes	2	15	2
Decon solution tank	A	<0.01	Yes	1	366	1
Decon recycle pump	A	<0.01	Yes	1	2	Note 3
Decon chemical supply pump	A	<0.01	Yes	1	2	Note 3
Purified Helium filters	A	<0.01	Yes	2	14	Note 3
Helium removal filter	A	<0.01	Yes	1	96	1
Helium getter units	A	<0.01	Yes	2	4	Note 4
HVAC filters	A	<0.01	Yes		1030	1
FHM	A	<0.01	Yes		192	2
FHM sand	A	<0.01	Yes		420	Note 1
Small and large bore piping	A	<0.01	Yes		576	6
Reactor Building Drain system	A	<0.01	Yes		125	1
Instrumentation & controls	A	<0.01	Yes		225	2
System 21 Components	A	<0.01	Yes		1,080	2
System 24 Components	A	<0.01	Yes		13,000	3
System 46 Components	A	<0.01	Yes		9,000	10
System 47 Components	A	<0.01	Yes		200	2
System 73 Components	A	<0.01	Yes		1,600	2
BOP TOTALS					38,871	6[illegible]

* - Pre-volume reduced quantities

Notes:
1. Will be used as overfill
2. Will be packaged with liquid transfer pumps
3. Will be packaged with Decon solution tank
4. Will be packaged with helium removal filter

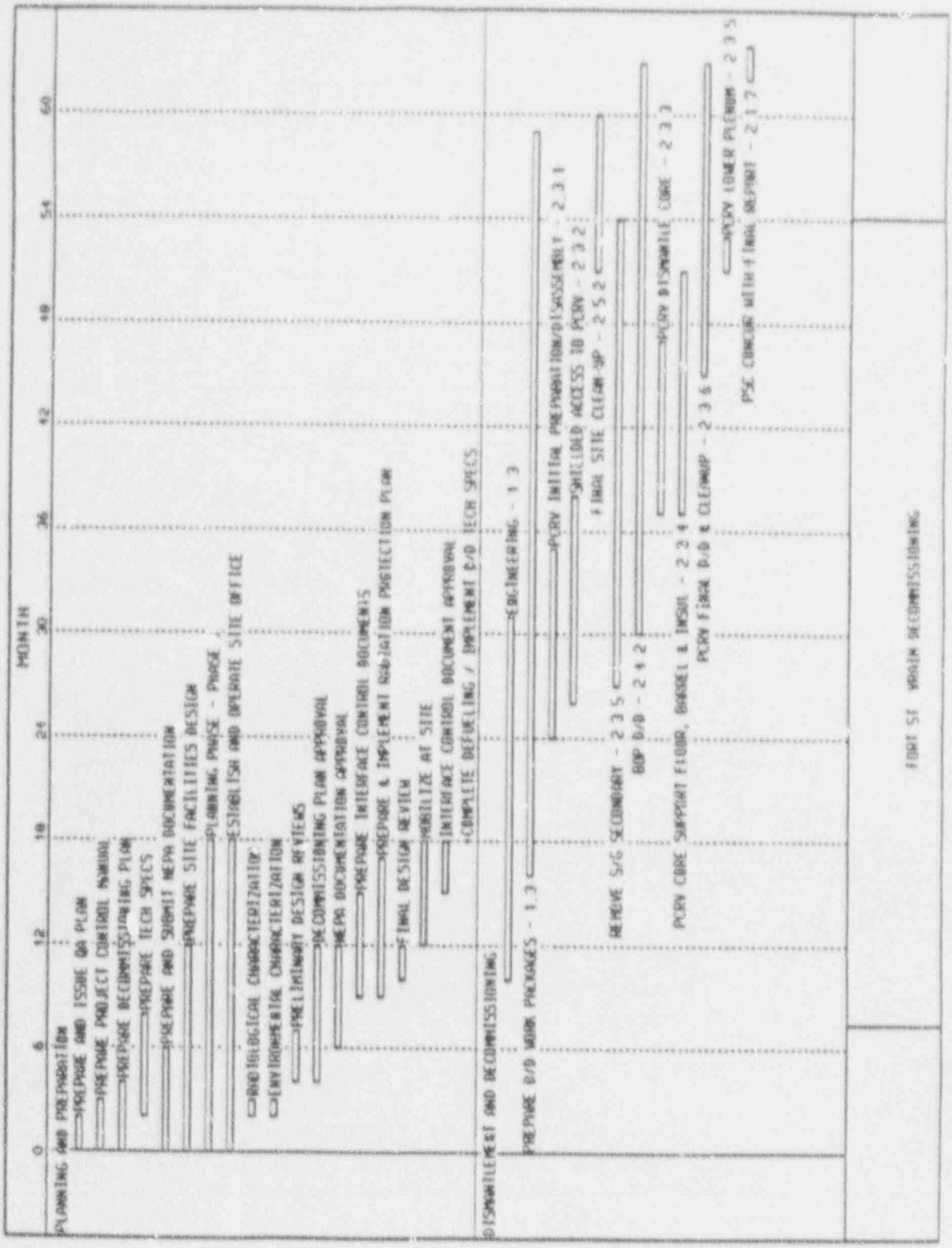

Figure 3.6-1 Fort St. Vrain Decommissioning Schedule (Sh. 1 of 6)

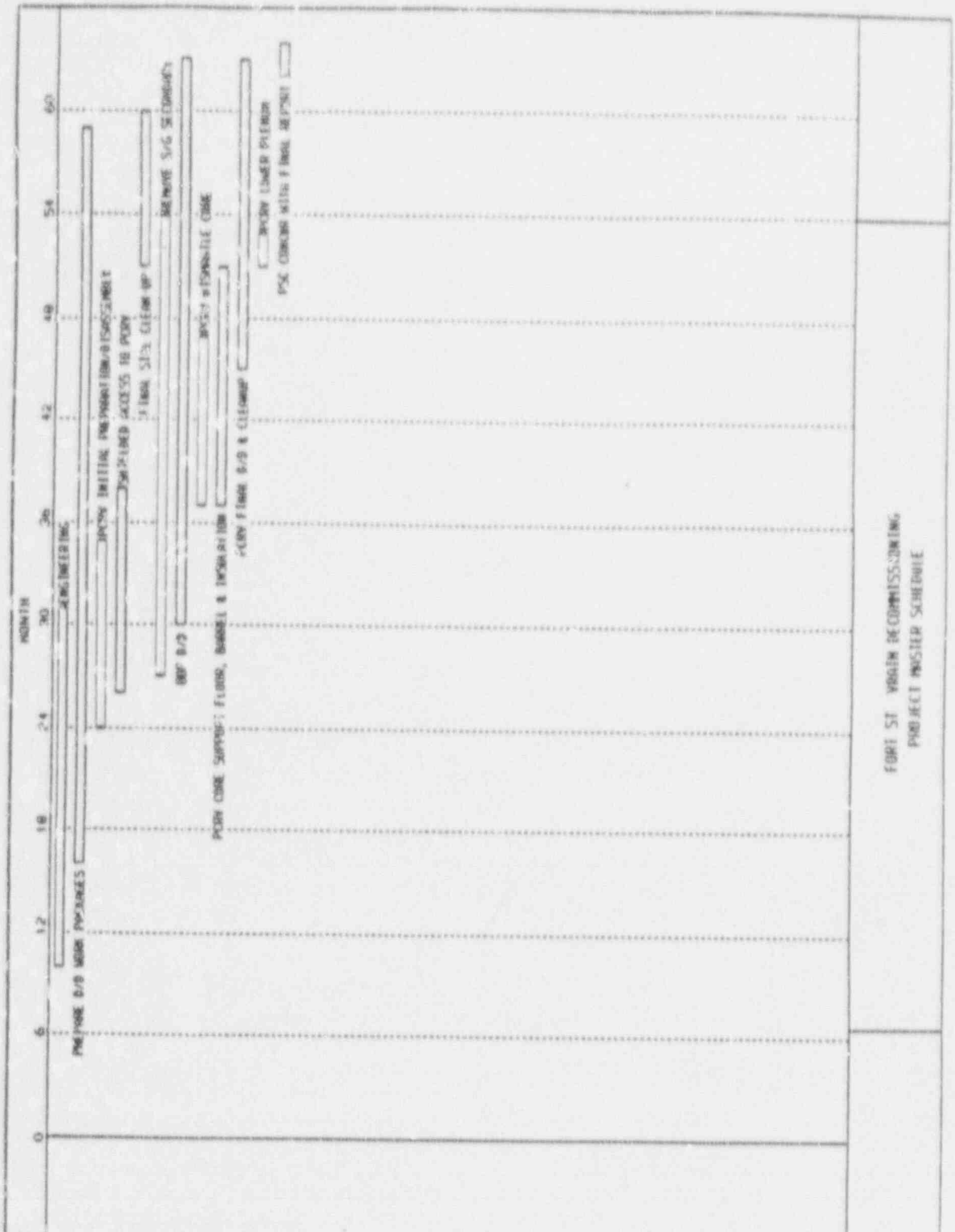

Figure 3.6-1 Fort St. Vrain Decommissioning Schedule (Sh. 2 of 6)

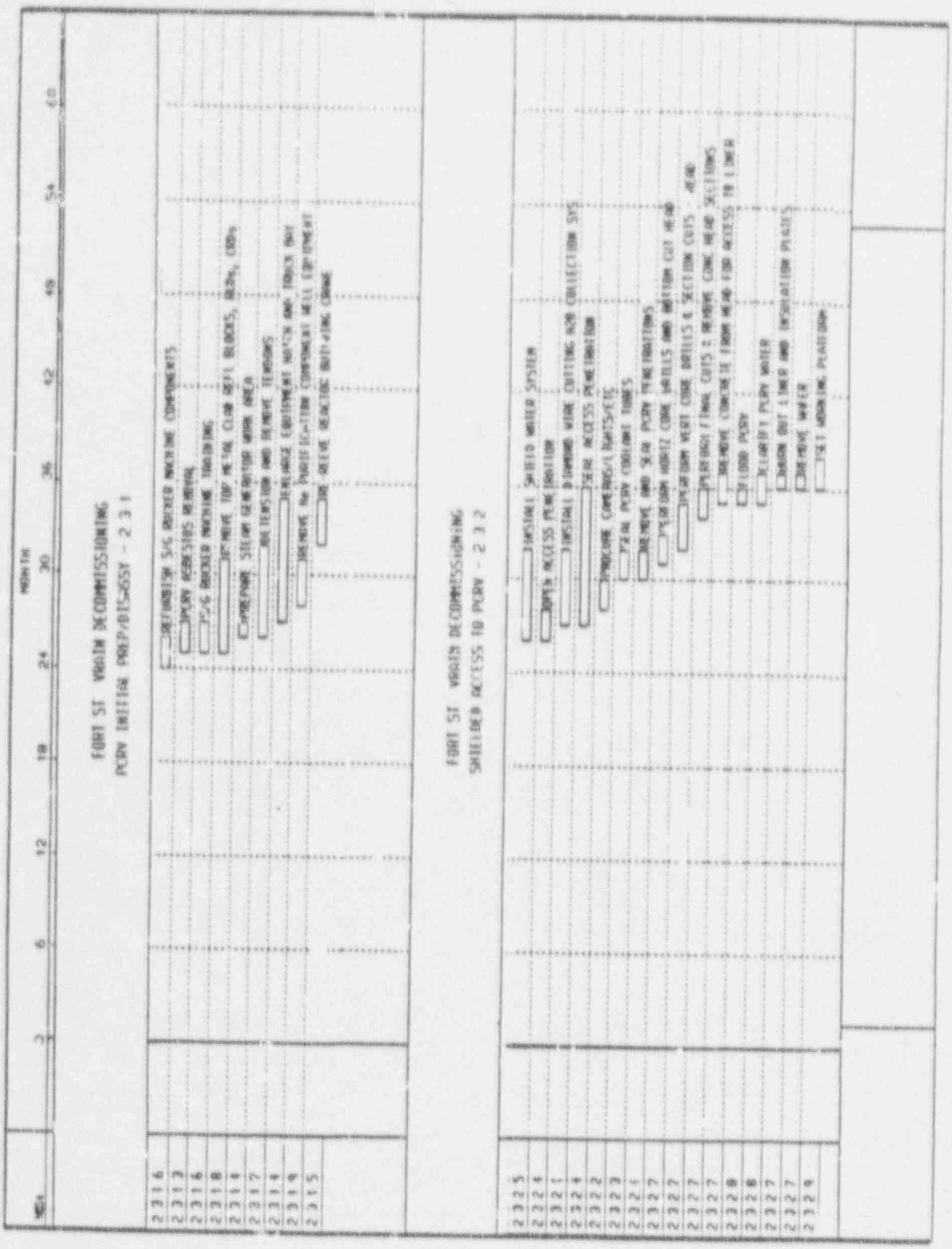

Figure 3.6-1 Fort St. Vrain Decommissioning Schedule (Sh. 3 of 6)

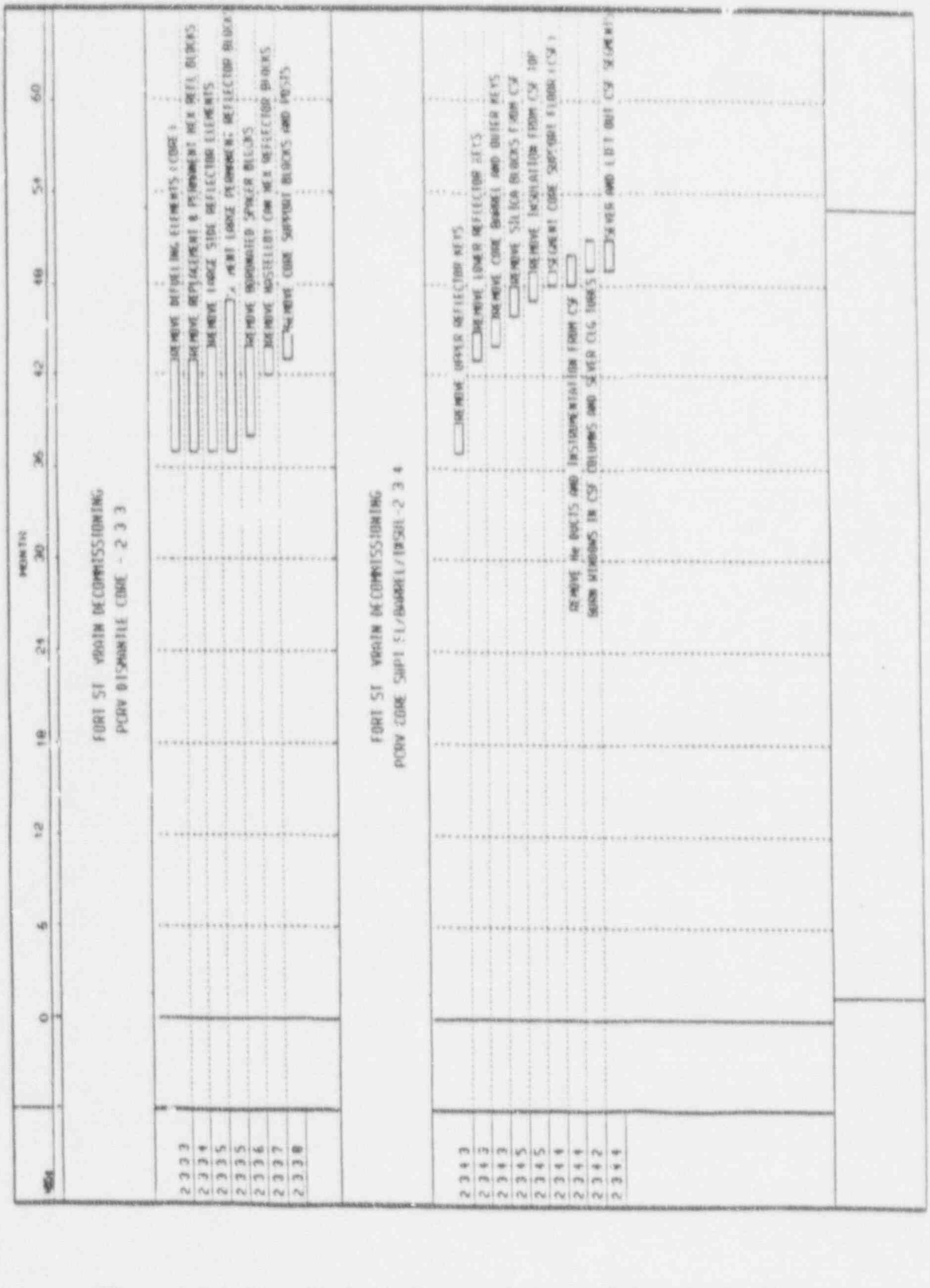

Figure 3.6-1 Fort St. Vrain Decommissioning Schedule (Sh. 4 of 6)

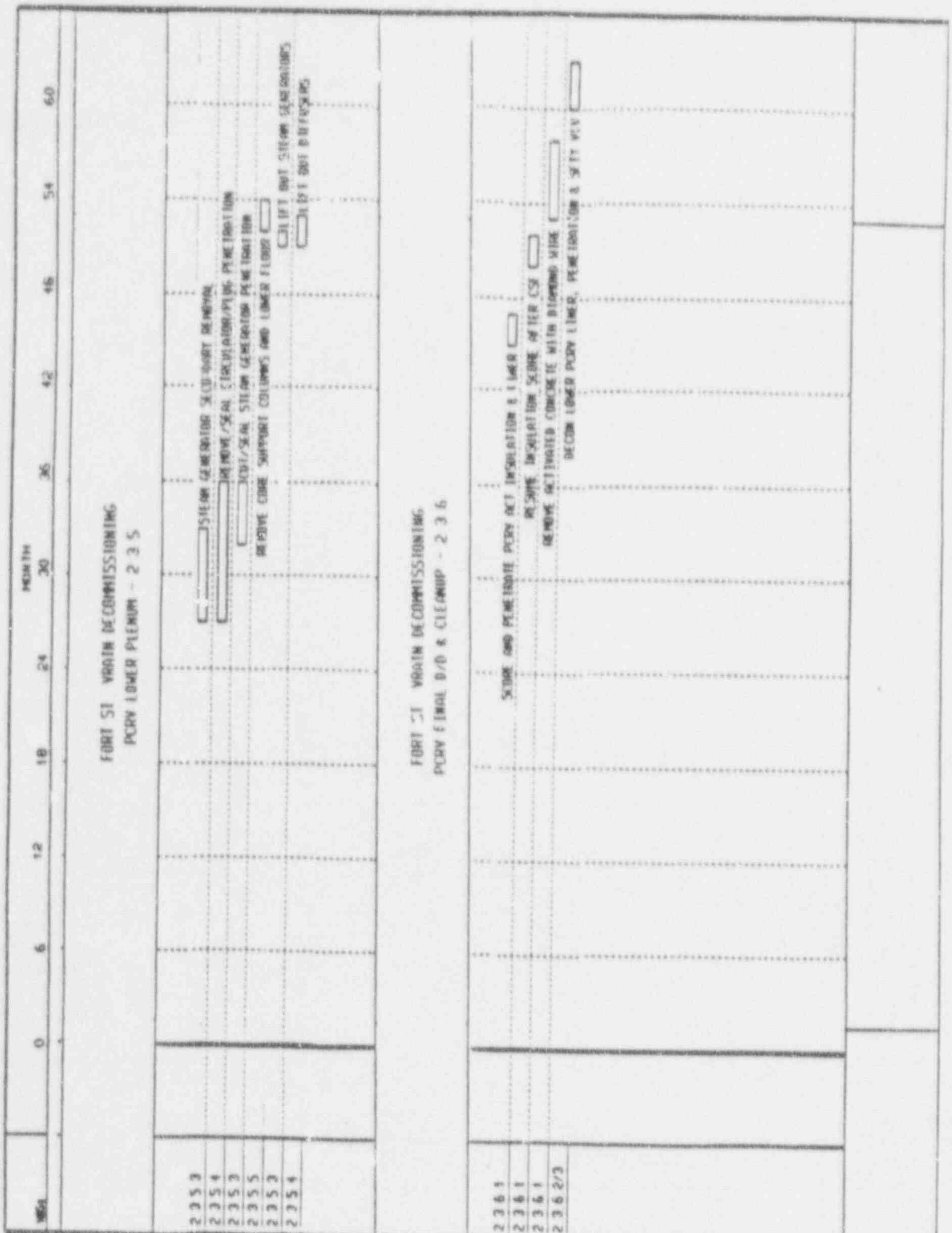

Figure 3.6-1 Fort St. Vrain Decommissioning Schedule (Sh. 5 of 6)

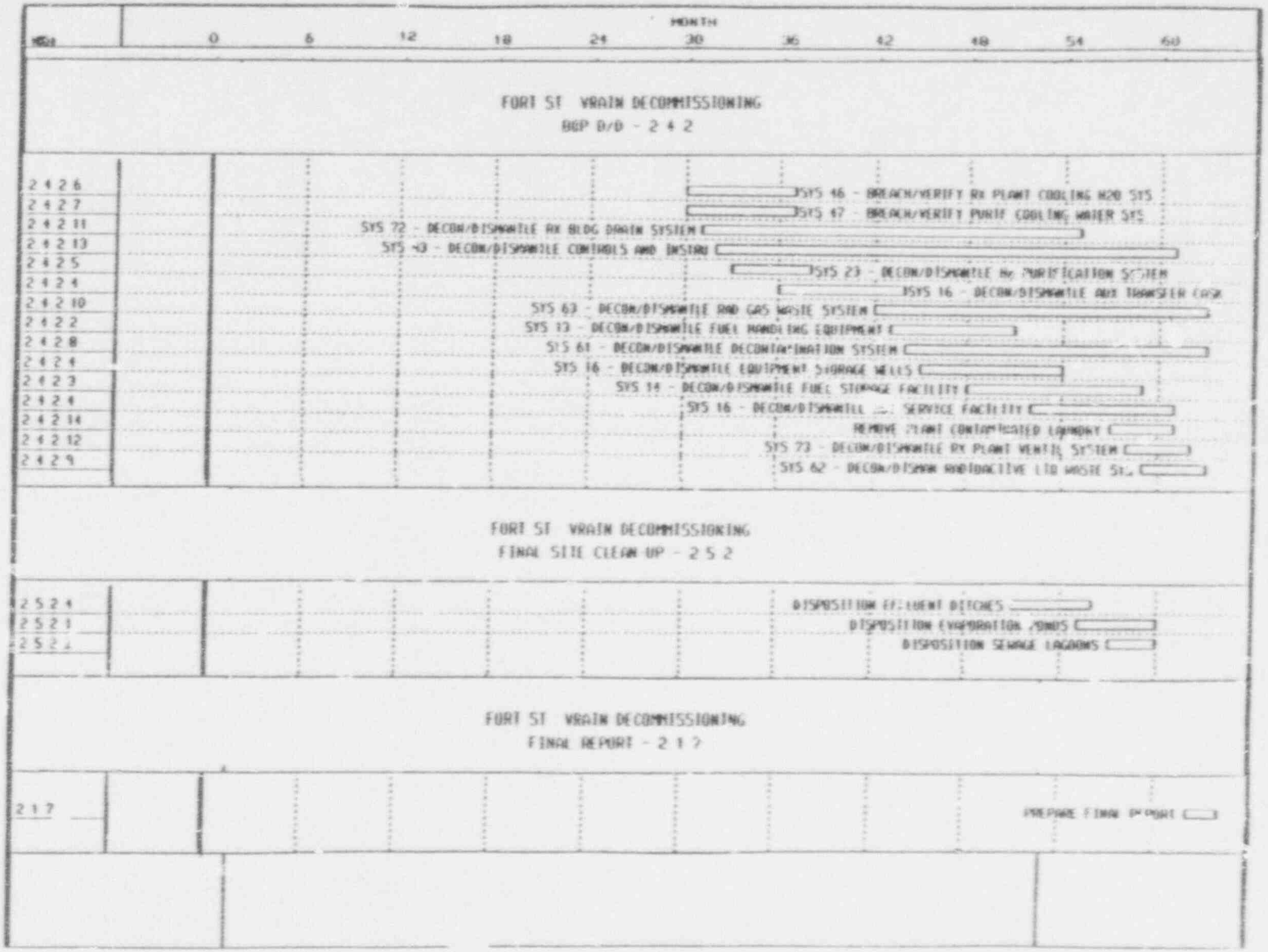

Figure 3.6-1 Fort St. Vrain Decommissioning Schedule (Sh. 6 of 6)

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4.0 ENVIRONMENTAL EFFECTS OF DECOMMISSIONING ACTIVITIES

4.1 Affected Environment

4.1.1 Overall Affected Environments

The following areas, not restricted to the immediate area of the plant, will be directly affected by the decommissioning of Fort St. Vrain:

1) Fort St. Vrain Site
Same areas as disturbed during plant construction and operation.

2) Replacement power sources
Same power sources as utilized during plant operation.

3) Radioactive Low Level Waste (LLW) site
Same sites as have been utilized during plant operation.

4) DOE Idaho Graphite Storage Facility (GSF)
Same facility as utilized during previous refuelings.

4.1.2 Demography

The population density in the rural areas surrounding Fort St. Vrain is relatively low. The nearest resident is located approximately one-half mile north of the Reactor Building, with the nearest town of Platteville located approximately 3-1/2 miles southeast. This is well outside the Exclusion Area Boundary (EAB) and the Emergency Planning Zone (EPZ). The population of Platteville, based on preliminary 1990 census figures, is 1515. 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.

4.1.3 Geography and Land Use

The site is located in Weld County, Colorado. The area surrounding the site is shown in Figure 4.1-1 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 Long's 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.

4.1.4 Geology and Seismology

The geologic structure of the general area in which the site is located is shown in Figure 4.1-2. 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 45 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 4.1-3 and 4.1-4. 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.

4.1.5 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 4.1-4). 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.

4.1.5.1 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 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 4.1-5.

4.1.5.2 Plant Effluent

The same liquid effluent release path will be used during decommissioning as was used during normal plant operations. As shown in Figure 4.1-6, diluted liquid effluent will normally be released from the Fort St. Vrain protected 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. Dilution flow rates will typically range from 1100 gpm to 2650 gpm during decommissioning. Availability of this 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 where effluents are normally 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 natural drainage slough. Similarly, the reactor plant drains flow to a diversion box from which the flow is normally directed to the South Platte River via the continuation of the Goosequill ditch, or alternatively to the St. Vrain Creek via the slough. Discharge to the slough is not a normal flow path and will be used on a limited basis for brief periods of time.

Further downstream from the plant, the Goosequill ditch flows into the Jay Thomas ditch and the combined stream flows into a 25 acre farm pond. The overflow from the farm pond flows into the South Platte River close to its confluence with the St. Vrain Creek. The drainage path via the Goosequill ditch and the pond is normally used.

Three lined evaporation ponds (total surface area of 3.6 acres) are present and are utilized to receive chemically treated effluent (primarily produced by periodic regeneration of plant demineralizers). 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 4.1-7. 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 waterways. 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 4.1-1. 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.

4.1.6 Meteorology

4.1.6.1 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 (Reference 1).

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.

4.1.6.2 Severe Weather

Tabulated below are temperature and precipitation records for three cities within 20 miles of Fort St. Vrain (see Figure 4.1-1). The recording periods were 1977-1988 (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 4.1-2 and 4.1-3, 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 5 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.

4.2 Radiological Impact from Routine Operations

The current Radiological Environmental Monitoring Program (REMP) will be continued in part specifically tailored to accurately monitor the environmental radiation and radioactivity levels, and to determine the effect on the radiological conditions of the environment due to decommissioning activities. The results of the REMP will be included in the final radiological survey report.

4.2.1 Liquid Waste Release Plans

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 decontamination operations. Flooding the PCRV will put into solution radionuclides that exist in the PCRV as a result of activation and platcout. 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 Section 3.3.2.3 of the PDP. Tritium processing, release and disposal options are discussed in the following sections.

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. The concentration of Co-60, Cs-137, and Fe-55 will be reduced by the demineralizers to an average concentration less than approximately 1.0 % of 10 CFR 20 MPC values. As discussed in Section 2.2.4, it is estimated that approximately 500 Curies of tritium will be released into the PCRV Shield Water. Tritium will either be diluted to releasable levels or processed for disposal as radioactive waste. Dilution and discharge as liquid effluent has been selected as the best alternative for tritiated water release.

Discharge of approximately 500 Curies of tritium as liquid effluent is not inconsistent with historical tritium releases made during operation of Fort St. Vrain. Over 2100 Curies of tritium have been released as liquid effluents since 1978, with the greatest annual release of 370 Curies occurring in 1983. If normal plant operations had continued for the entire 35 year period of the facility operating license, and if an average of 200 Curies of tritium were released each year, a total of 7000 Curies of tritium would have been released into the downstream surface waters.

The discharge of approximately 500 Curies of tritium is also not inconsistent with typical annual operating releases from a number of Light Water Reactors. Based on Semiannual Radioactive Effluent Release Report data (Reference 2), over a dozen single unit Pressurized Water Reactor plants commonly release over 500 Curies of tritium via liquid effluents, and there are two facilities that typically release several thousand Curies of tritium each year.

If the actual amount of tritium released into the PCRV Shield Water System is greater than 500 Curies, PSC has established an upper limit for liquid effluent releases of 8000 Curies of tritium, to be released over a period of time greater than one year, and most likely within three years. The discharge of 8000 Curies of tritium over a time period between one and three years is less than the 8000 Curie annual tritium discharges previously evaluated and accepted at the Haddam Neck Nuclear Generating Station (Reference 14).

Discharge of 500 Curies of tritium (up to a maximum of 8000 Curies) in liquid effluent will be controlled within the following limits:

1. EPA Safe Drinking Water Standards in 40 CFR 141 (20,000 pCi/l average concentration), in the South Platte River, 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), and

3. 10 CFR 20 MPC limits on concentrations in effluents released to unrestricted areas (e.g., 0.003 μCi/ml for tritium).

Administrative controls will be implemented to ensure that the above limits are not exceeded. The above standards also ensure compliance with the EPA public dose limits in 40 CFR 190 of 25 mrem per year.

4.2.2 <u>On-Site Processing of Liquid Wastes</u>

4.2.2.1 PCRV Shield Water System

The PCRV Shield Water System will be installed to maintain water clarity and control radioactive material concentrations in the PCRV. A description of this system is provided in PDP Section 2.3.3.6. The system will consist of two parallel trains. Each train will consist of a coarse strainer designed to remove gross debris and to protect downstream equipment. A standard dimension process pump will provide the driving head for the purification flowrate through two banks of filters located downstream of the pump. A prefilter will remove larger suspended solids, and a final filter will provide the degree of filtration necessary to ensure acceptable water clarity. Suitable valving and cross connection between trains will enhance system flexibility and availability. Each train of the PCRV Shield Water System will have a recirculation ("full flow filtration") capacity of approximately 500 gpm. In addition to the capability for full-flow filtration of the PCRV water inventory, the system design will also include partial (side stream) demineralization for controlling dissolved solids. Connections for adding clean makeup water and for removing contaminated water will also be provided to control the tritium concentration.

Chemical addition tanks are included for chemistry and pH control, and to suppress biological growth. The system design will also include instrumentation, controls, and sampling points. These will enable proper operation in monitoring the system and effectiveness of its components. The purified water will return to the top of the PCRV cavity by means of a distribution header designed to minimize local velocities and turbulence to maintain underwater visibility.

The equipment will be appropriate for the radioactive nature of the process fluid. Equipment that can generate a high radiation field, such as filters and demineralizers, will be shielded and provided with remote handling capability. Equipment fluid drains and leakoffs will be collected, treated and disposed of as discussed.

Effluents from the PCRV Shield Water System will be diluted by the dilution flow from the circulating water makeup pumps to the main cooling tower blowdown line prior to release to the surrounding surface waters. The rate of effluent release from the PCRV, along with the dilution flow, will be controlled to assure that the radionuclide concentrations following dilution do not exceed the limits specified in the Offsite Dose Calculation Manual (ODCM). The same methods for controlling release of radioactive liquids have been, and will continue to be, used for releases from the Reactor Building sump and the radioactive liquid waste system.

The effluent release flow rate from the PCRV is adjustable, with a controlled maximum flow rate of approximately 100 gpm. Likewise, the dilution flow rate from the circulating water makeup pumps to the main cooling tower blowdown line is adjustable, with a typical dilution flow rate of approximately 1500 to 2000 gpm. Flow rates will be controlled to ensure that discharge concentrations do not exceed the limits specified in the ODCM.

Since the PCRV Shield Water System discharge outlet will connect to the outlet piping of the reactor building sump pumps, the automatic monitoring and protection features which currently govern releases from the Reactor Building sump and radioactive liquid waste system will also govern releases from the PCRV (Figure 4.5-1). These features include two redundant activity monitors in the radioactive liquid waste discharge line, which monitor gross gamma activity in the line, although these monitors will not be useful for detecting tritium concentrations. The signals from these activity monitors are arranged in one-out-of-two logic, so that if either monitor detects high concentrations of gross gamma activity, the monitor will alarm and automatically close the block valves in the radioactive liquid waste discharge line. The block valves are also interlocked with main cooling tower blowdown flow and will automatically close when the flow drops below a predetermined setpoint.

The PCRV Shield Water System discharge flow rate, low dilution water flow switch setting, and the setting of the radioactivity monitors will be established in accordance with the Offsite Dose Calculation Manual (ODCM) prior to radioactive liquid release.

Prior to release and periodically during liquid effluent releases, representative samples obtained from the PCRV Shield Water System will be analyzed for gross alpha and beta activity, principal gamma emitters, tritium and other radioisotopes of concern to determine the dilution factor required to assure that the concentration in the cooling tower blowdown flow will not exceed the values specified in 10 CFR 20. Once the liquid waste release rate, blowdown flow, and activity monitors have been set, the liquid waste effluent will be released at a controlled and monitored flow rate to the cooling tower blowdown line for dilution and release to the environment.

4.2.2.2 Radioactive Liquid Waste System

Initially, water to be released will be transferred to a liquid waste holdup tank in the existing Radioactive Liquid Waste System (System 62) for sampling and analysis. The liquid waste system consists of two 3000 gallon receivers, a 3000 gallon monitoring (or holdup) tank, and associated filters and demineralizers. The holdup tank will be sampled and analyzed for tritium and other principal radionuclides. Based on sample results and the limits in the FSV Offsite Dose Calculation Manual (ODCM), an allowable release rate will be determined to ensure that the established limits will not be exceeded.

Initial tritium concentrations are expected to allow liquid waste in the holdup tank to be discharged at a rate of 1.4 to 10 gpm. Liquid effluent will be diluted by the cooling tower blowdown flow prior to release to the Goosequill Ditch. A minimum cooling tower blowdown flow of 1100 gpm is defined in the ODCM, which provides a dilution factor of at least 110. PSC is planning to release tritiated liquid effluent at a total flow rate of approximately 2000 gpm, which is within the 2650 gpm maximum blowdown flow identified in the FSV Final Environmental Statement (FES, Reference 3).

After 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 after tritium has been reduced to less than the 10 CFR 20 MPC, the water will be sent directly to the discharge line, where it will be diluted with blowdown flow and released. The liquid waste holdup tank will not be used. 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 and release them into the PCRV Shield Water System.

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.

4.2.3 Other Tritium Release Alternatives

Two other options were evaluated for releasing tritium, including (1) construction of a large shallow water impoundment for solar evaporation of tritium and (2) installation of a series of mechanical evaporators for forced evaporation of tritium. Both of these alternatives were determined to be less desirable than releasing tritiated water via the normal liquid effluent release path.

Solar evaporation from a shallow water impoundment (i.e., pond) was determined to be less desirable at this time for the following reasons:

1. The evaporation rate cannot be easily controlled.
2. Rainfall or snow could add to the amount of tritiated liquid to be evaporated.
3. Access fencing and an impoundment liner would be required.
4. It would be difficult to prevent migrating birds and small animals from entering the area.
5. Additional occupational radiation exposure to a decommissioning worker, standing 100 m from the pond for 8 hr per day for 200 days (e.g., a security guard) is conservatively calculated to be 342 mrem. This is based on a 10 gpm release of 500 Curies of tritium in liquid effluent, and it includes Co-60, Cs-137, and Fe-55.
6. The impoundment presents an additional accident source term, specifically in the event of rupture of the impoundment lining.
7. Additional costs are associated with lining the impoundment.
8. Disposal of sludge and ultimate decontamination of the impoundment area must be performed, whereas no additional remediation to the Goosequill ditch areas is expected.
9. The inhalation dose to an adult member of the public standing 100 m from the pond is conservatively calculated to be 0.43 mrem during a 2 hr exposure.

Mechanical evaporators were also determined less desirable at this time for the following reasons:

1. A large throughput mechanical evaporator is expensive and requires constant attention.

2. A number of evaporators are required for the necessary throughput and operational flexibility in the event of an evaporator breakdown.
3. The additional occupational radiation exposure to a decommissioning worker standing 100 m from the evaporator for 8 hr per day for 200 days is conservatively calculated to be 342 mrem. This is based on a 10 gpm release of 500 Curies of tritium in liquid effluent and it includes Co-60, Cs-137, and Fe-55.
4. Eventual decontamination and disposal of evaporators is required.
5. The inhalation dose to an adult member of the public standing 100 m from the evaporator is conservatively calculated to be 0.43 mrem during a 2 hr exposure.

4.2.4 Processing 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 process it for disposal. An acceptable processing method would be to use a product such as Aquaset, which can absorb approximately 45 gallons in a 55 gallon drum and sets to a chalk-like consistency. This process requires about 1 hour per drum.

Processing of tritiated water in this manner would be accomplished in accordance with the Process Control Program (PCP). Appropriate radiological controls would be implemented 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.

4.2.5 Surface Water Dilution

Dilution of liquid effluent in downstream surface waters depends to a large extent on river flow rates. PSC has reviewed historical river flow rates and has evaluated the dilution that can be expected during the FSV tritium releases during decommissioning. Historical river concentrations from the FSV REMP confirm that sufficient dilution exists to maintain river tritium concentrations below the EPA Safe Drinking Water Standards.

Downstream surface water tritium concentration will vary due to many considerations, and is particularly affected by river flow rates. South Platte River flow rates have stabilized since the Chatfield Reservoir was installed over 40 miles upstream in the early 1970s. FSV FSAR Section 2.5 lists average monthly flow rates for the nine year period from 1971 through 1979. Seasonal flow variations during this time period are as follows:

Average ('75-'79 data)	353 cfs	(158,000 gpm)
Peak flow (May)	1077 cfs	(483,000 gpm)
Minimum flow (December)	249 cfs	(112,000 gpm)

These flow rates were taken at the upstream gauging station in Henderson, and are somewhat reduced by irrigation ditches that take water before it gets to Fort St. Vrain. However, much of this irrigation water is returned to the rivers as groundwater, and in addition, the St. Vrain Creek joins the South Platte River approximately 100 yards downstream of the farm pond outlet and adds an annual average flow of 203 cfs. Based on a study of local groundwater flows (Reference 5), nominal river flow at the convergence of the two rivers is estimated to be 632 cfs (283,000 gpm).

With a 2000 gpm discharge flow into and out of the farm pond, this results in a nominal dilution factor of 140. If the discharge concentration is 0.003 μCi/ml (the 10 CFR 20 MPC), the resulting river concentration would be slightly over the EPA Safe Drinking Water Standard of 20,000 pCi/l. However, this calculation shows that the 20,000 pCi/l limit can be met by slightly reducing the release rate, slightly reducing the discharge concentration, or by additional dilution of the discharge flow (with water from the Jay Thomas ditch, for example).

For ALARA purposes, and to the extent possible considering ir[illegible] needs, water from the South Platte River will be supplied to the Jay Thomas ditch (See Figure 4.1-6) during liquid effluent releases when tritium concentrations are near the 10 CFR 20 MPC limits. This flow will mix with liquid effluent in the Goosequill Ditch and provide additional dilution of tritium concentration in the ditch and farm pond.

4.2.6 Radiological Consequences of Tritiated Water Releases

The radiological consequences of releasing tritium into the environs downstream of Fort St. Vrain have been thoroughly evaluated and documented via the FSV Environmental Radiation Surveillance Program (ERSP, 1983 and before) and the Radiological Environmental Monitoring Program (REMP, 1984 and after). The relationships (or independence) between releases of tritium from the plant and concentrations of tritium in the environment have been observed through these programs. These field observations are of significant value in assessing the impact of the planned 500 Curie tritium release during decommissioning.

As during plant operations, liquid effluents during decommissioning are controlled to ensure compliance with the public dose limits of 10 CFR 20 and 10 CFR 50

Appendix I. The most limiting requirements are in Appendix I, which PSC has committed to in the proposed Decommissioning Technical Specifications. Appendix I limits dose or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas to the following:

a. During any calendar quarter to less than or equal to 1.5 millirem to the whole body and to less than or equal to 5 millirems to any organ, and

b. During any calendar year to less than or equal to 3 millirems to the total body and to less than or equal to 10 millirems to any organ.

Radionuclides Other Than Tritium

As noted previously, the PCRV Shield Water System will include demineralizers that have been designed to ensure that radionuclides such as Co-60, Cs-137, and Fe-55 are reduced to an average concentration less than approximately 1.0 % of their respective 10 CFR 20 MPCs prior to dilution and release. These radionuclides are further reduced by an ion exchange phenomenon with the sediment in the bottom of the ditch. From the junction of the Goosequill ditch and the Jay Thomas ditch, the ditch bottom is sediment and radionuclide absorption will occur on the clay mineral sediments.

The parameter describing this reduction is called the distribution coefficient, and has been studied extensively in agronomy and radioecological literature. A conservative value of the distribution coefficient is 1000 for Co-60 and Fe-55, and 200 for Cs-137 (Reference 4). Therefore the concentration of these radionuclides will reach an equilibrium value 200 to 1000 times less than the concentration entering the system before any aquatic or biological uptake occurs. Taking a conservative case where humans were to directly ingest waterfowl or some other food source with water contents equivalent to consuming 2 liters per day of this water, a resultant 50 year Committed Effective Dose Equivalent (CEDE), using ICRP-30 dose commitment parameters, would be 4.4 E(-4) mrem for Co-60, which is negligible. The CEDE is a weighted sum of all organ and tissue doses and should be compared with whole body dose limits. CEDEs for Cs-137 and Fe-55 would be comparable and therefore, further pathway analysis for these radionuclides is not warranted.

4.2.6.1 Pathways

As tritium in liquid effluent progresses into the environment, it is potentially available to various sources of public consumption in three basic concentrations:

1. At approximately the 10 CFR 20 MPC limit of 0.003 μCi/ml in the Goosequill Ditch and the farm pond,

2. At the EPA Safe Drinking Water Standard limit of 20,000 pCi/l in the downstream surface water, and
3. At a more diluted level in the groundwater that reaches the Gilcrest town well, which is used for public drinking water.

Cattle and waterfowl have access to water in the Goosequill Ditch. For various reasons that are discussed later, ingestion of these food sources is not considered reasonable; however, doses have been calculated separately.

The most reasonably conservative pathways are considered to be the following:

1. Consumption of drinking water from the Gilcrest town well,
2. Consumption of milk from a family cow pastured on fields irrigated from downstream surface water from the South Platte River,
3. Consumption of vegetables from a local garden watered from downstream water from the South Platte River,
4. Consumption of beef pastured on fields irrigated from downstream surface water from the South Platte River,
5. Consumption of waterfowl from downstream surface water in the South Platte River basin, and
6. Consumption of fish from downstream surface water in the South Platte River basin.

The above pathways are evaluated as follows:

4.2.6.2 Consumption of Gilcrest Well Water

The closest downstream public water supply is the Gilcrest town water well. This well is shallow, as evidenced by high nitrate concentrations from fertilizers used by area farmers. During previous tritium releases, PSC has sampled Gilcrest well water and has not been able to establish a definitive correlation with tritium concentrations in FSV effluent. A site specific groundwater study concluded that a possible groundwater flowpath from the farm pond to the Gilcrest well does exist (Reference 5), and in the course of PSC's environmental sampling efforts, the Gilcrest well water has on occasion had detectable tritium concentrations (which have been well below EPA standards for safe drinking water). Based on REMP observations that tritium has been detected in the Gilcrest water well, this pathway is considered the most probable route of tritium to local inhabitants.

Based on a review of tritium release data and the corresponding tritium concentrations measured at the Gilcrest town well since 1984, the nominal effective dilution volume was determined for the hydrological environment between the plant release point and the Gilcrest town well (Reference 6). This effective dilution volume was then applied to the release of approximately 500 Curies of tritium, and the resultant drinking water concentration was calculated to be 1000 pCi/l. The 50 year Committed Effective Dose Equivalent (CEDE) to an adult member of the general public consuming only Gilcrest well water during the 200 day estimated release period is conservatively calculated to be 0.026 mrem. As discussed previously, the CEDE should be compared to whole body dose limits.

If the actual amount of tritium in the PCRV Shield Water is 8000 Curies, as discussed previously, the release duration will be extended beyond the 200 days evaluated above. In this case, the assumed tritium concentrations could be sustained for between one and two years. The annual 50 year CEDE from drinking Gilcrest well water for an entire year is calculated to be 0.047 mrem.

Consumption of Gilcrest well water is the only reasonable pathway identified above that involves groundwater. Both the 200 day CEDE of 0.026 mrem and the annual CEDE of 0.047 mrem identified above are well within the EPA drinking water standard of 4 mrem per year, as addressed in Reference 15.

There is another well located closer to the farm pond than the Gilcrest well, but it is not used for drinking water. This well is on the Russell farm and is located south of the farm pond. Domestic water used on the Russell farm is supplied by a pipeline from the Central Weld County Water District.

4.2.6.3 Milk Consumption

As described in previous REMP reports and as recently confirmed in the 1990 Land Use Census (Reference 7), there are no dairies or personal milk cows within a 1 mile (1.6 km) radius of the plant. Six dairies within 5 miles (8 km) of the plant have been extensively sampled in the REMP programs. The REMP report for 1988 stated that elevated tritium concentrations due to reactor effluents have never been observed during the operational period of the reactor. This implies the tritium from reactor effluents is not contributing any radiation dose to humans via the milk pathway. Tritium concentrations in milk should respond rapidly to changes in tritium concentrations of the forage water intake or drinking water intake to the cow. This is due to the short biological half life for water in the cow (about three days for the lactating cow).

In 1988, a survey of the dairies in the REMP program determined that herd management practices are similar at all dairy locations. The cows in the milking herd are never on pasture that could be irrigated by downstream water but instead are under dry-lot management typical of Eastern Colorado. (Reference 8).

The 1990 Land Use Census confirmed that there are no milk animals in areas with direct access to the Goosequill ditch or the farm pond. A few residents up to a distance of 5 miles (8 km) from the plant have cows or goats that could be used for personal milk consumption. However, from direct discussions held between CSU surveyors and local residents, personal consumption is not a common practice and all cow milk produced is transported to commercial processors. The milk produced locally is diluted, processed, and distributed over a large area for consumption. CSU concluded in the 1990 REMP that elevated radionuclides (I-131 from upstream medical uses was the radionuclide of concern at that time) from the closest dairy farm would never be detected in the composite milk supply.

Notwithstanding the above, a conservative calculation was performed of doses resulting from drinking milk from a family cow that had been watered from and had grazed on pasture land irrigated from South Platte River water containing tritium at 20,000 pCi/l. The resultant 50 year CEDE dose from drinking only this milk for the entire 200 day portion of an assumed 400 liter annual consumption is calculated to be 0.281 mrem.

If the actual amount of tritium in the PCRV Shield Water is 8000 Curies, as discussed previously, the release duration will be extended beyond the 200 days evaluated above. In this case, the assumed tritium concentrations could be sustained for between one and two years. The annual 50 year CEDE from drinking milk from the family cow described above for an entire year is calculated to be 0.512 mrem.

4.2.6.4 Vegetable Consumption

The 1990 Land Use Census in the REMP (Reference 7) confirmed that there are no gardens in the area with access to the Goosequill ditch or the farm pond. Thus, the downstream water that could be used to irrigate crops and vegetables for human consumption will be diluted by the downstream rivers.

The intake of tritiated water from crops and vegetables irrigated by downstream water is a possible dose pathway. The FSV REMP has sampled food products and analyzed for principle gamma emitters. No radionuclides have been found that were attributed to FSV effluent releases.

The REMP has typically monitored six to twelve locations from areas possibly irrigated by surface water downstream of the FSV discharge point or by well water from the aquifer most likely to be contaminated by seepage from the farm pond. One sample of each principal class of food product was collected at each location, and typically included items such as corn, melons, tomatoes, beans, onions, zucchini, broccoli, potatoes, beets, beet tops, squash, turnips, cucumbers, and cabbage.

Based on the fact that no gamma emitting radionuclides attributed to FSV activities have been detected in crops and vegetables during previous monitoring, this pathway is not considered a likely source of tritium to local inhabitants. Nevertheless, if an individual were to consume a 200 day portion of an assumed annual consumption of

340 kg of vegetables from a local garden watered with South Platte River water at 20,000 pCi/l, the resultant 50 year CEDE would be 0.179 mrem.

If the actual amount of tritium in the PCRV Shield Water is 8000 Curies, as discussed previously, the release duration will be extended beyond the 200 days evaluated above. In this case, the assumed tritium concentrations could be sustained for between one and two years. The annual 50 year CEDE from consuming the vegetables described above for an entire year is calculated to be 0.326 mrem.

4.2.6.5 Consumption of Beef and Waterfowl

Beef cattle are not a likely exposure pathway. Typical cattle sent to slaughter are removed to a dry feed lot for approximately 90 days before slaughter. Assuming the biological half-life of water in the beef cow is approximately eight days, this produces at least ten biological half-lives before slaughter. Notwithstanding these common practices, it is conceivable that beef cattle grazing in pastures irrigated from the South Platte River were slaughtered for private consumption, bypassing the feed lot process.

The area around the downstream surface waters has a plentiful supply of ducks and geese. Hunting is permitted in season, and it is possible that waterfowl that have been drinking surface water could be consumed.

If an individual consumed a 200 day portion of an assumed annual combined beef and waterfowl consumption of 110 kg, that had been drinking or pasturing on land irrigated from the South Platte River at 20,000 pCi/l, the resultant 50 year CEDE dose would be 0.046 mrem.

If the actual amount of tritium in the PCRV Shield Water is 8000 Curies, as discussed previously, the release duration will be extended beyond the 200 days evaluated above. In this case, the assumed tritium concentrations could be sustained for between one and two years. The annual 50 year CEDE from consuming the beef and waterfowl described above for an entire year is calculated to be 0.084 mrem.

4.2.6.6 Consumption of Fish

There are few if any sport fish in the river near the farm pond outlet. Carp are present in these waters, but they are generally not consumed. Nevertheless, if an individual were to consume a 200 day portion of an assumed annual consumption of 21 kg, 90% of which is water at 20,000 pCi/l, the resultant 50 year CEDE would be 0.013 mrem.

If the actual amount of tritium in the PCRV Shield Water is 8000 Curies, as discussed previously, the release duration will be extended beyond the 200 days evaluated above. In this case, the assumed tritium concentrations could be sustained for between one and two years. The annual 50 year CEDE from consuming the fish described above for an entire year is calculated to be 0.024 mrem.

4.2.6.7 Total Dose Due to Above Pathways

The total 50 year CEDE to an individual consuming each of the above food sources over the 200 day period of the 500 Curie tritium release is as follows:

Drinking Gilcrest Well Water	0.026 mrem
Milk Consumption	0.281 mrem
Vegetable Consumption	0.179 mrem
Beef and Waterfowl Consumption	0.046 mrem
Fish Consumption	0.013 mrem
Total	0.545 mrem

This total dose is within 10 CFR 50, Appendix I limits. This estimate is conservative in that it assumes the initial concentration of tritium in the water source remains constant for the entire 200 day period, whereas in reality, tritium concentrations are expected to gradually decrease as the tritium in the PCRV Shield Water is released.

If the actual amount of tritium in the PCRV Shield Water is 8000 Curies, as discussed previously, the release duration will be extended beyond 200 days. In this case, the assumed tritium concentrations could be sustained for a period of time between one and two years. The annual 50 year CEDE dose from the conservative pathways described above, based on a release of 8000 Curies of tritium over more than one year is calculated to be as follows:

Drinking Gilcrest Well Water	0.047 mrem
Milk Consumption	0.512 mrem
Vegetable Consumption	0.326 mrem
Beef and Waterfowl Consumption	0.084 mrem
Fish Consumption	0.024 mrem
Total	0.993 mrem

This total dose is also within 10 CFR 50, Appendix I limits.

4.2.6.8 Beef Consumption - Goosequill Water Supply

Beef cattle have access to liquid effluent at several locations along the Goosequill Ditch and at the farm pond. In addition, these beef cattle have access to pasture land that may be irrigated from the Goosequill Ditch. This is not considered a realistic dose pathway, however, due to local agricultural practices. Dr. Gerald Ward, Professor of Animal Sciences at CSU, was consulted about beef cattle practices. Based on his discussions with Ben Houston, the local rancher whose cattle are pastured in the area near the plant, he determined that the local cattle are registered purebred Aberdeen Angus cattle and nearly all are sold for breeding purposes. Cattle that are sent to slaughter are removed to a dry feed lot for approximately 90 days before slaughter. Assuming the biological half-life of water in the beef cow is approximately eight days, this produces at least ten biclogical half-lives before slaughter. Dr. Ward also confirmed that no beef cattle are slaughtered for personal use or local consumption directly out of the pastures, without having gone through the dry feed lot process.

Notwithstanding the above, PSC determined the resultant doses from consuming beef from one of the local cows that consumed Goosequill water or that grazed on land irrigated from Goosequill water. Since some of the water provided to the cattle for drinking water comes from the Carter Lake/ Central Weld County domestic water supply, it was assumed that 50% of the drinking water was FSV liquid effluent, at its release limit of 0.003 μCi/ml of tritium. It was also assumed that the animal is slaughtered after tritium concentration has reached a maximum steady state level in its tissues, and 60% of the tissue weight is water (this assumes that 20 % is fat and that 73 % of the remaining fat-free weight is water). An individual who consumed 40 kg of this animal would receive a 50 year CEDE dose of 2.3 mrem, using a CEDE dose commitment factor for tritium of 6.4 E-8 mrem/pCi.

This CEDE is calculated by ICRP-30 methodology, using a dose commitment factor from EPA-520/1-88-020. Consumption of 40 kg of beef would likely occur over a period of time greater than 3 months, since the assumed consumption of beef and waterfowl combined is 110 kg for an entire year. Therefore, it is reasonable to assume that this CEDE should be compared to the annual dose limit. While substantial, this 2.3 mrem dose due to the hypothetical consumption of this animal would be within the 10 CFR 50 Appendix I limits.

4.2.6.9 Waterfowl Consumption - Goosequill Water Supply

The area around the farm pond has a plentiful supply of ducks and geese, the majority of which are considered migratory and transient. There is a sizable resident waterfowl population and hunting has been a practice in the past. To eliminate this pathway during the tritium release period during decommissioning, PSC will post the farm pond area and hunting will not be permitted.

Notwithstanding the no hunting policy, the dose due to consuming waterfowl from the farm pond is calculated. It is reasonable to assume that an individual could consume 5 waterfowl (ducks and/or geese) per year. The 50 year CEDE dose from consuming 5 waterfowl, weighing an average of 4 kg each, where 60% of the weight is assumed to be water (half of which came from the farm pond at the release concentration of 0.003 μCi/ml of tritium and half of which is from time spent in the adjacent rivers and is diluted to less than 20,000 pCi/l), would be 1.2 mrem. This dose is within the 10 CFR 50 Appendix I limits.

4.2.6.10 Fish Consumption - Goosequill Water Supply

There are no sport fish in the flow path to the farm pond or in the farm pond itself. Carp are present in these waters, but they are generally not consumed. During the tritium release period for decommissioning, PSC will post the farm pond and fishing will not be permitted.

Notwithstanding the no fishing policy, the dose due to consumption of fish from the farm pond is calculated. An individual who consumed 11.5 kg of fish (200 days portion of a 21 kg annual fish consumption), 90% of which is water at 0.003 μCi/ml, would receive a 50 year CEDE dose of 2.0 mrem. This dose would likely be received over more than three months and would be within 10 CFR 50 Appendix I limits.

Due to the fact that the consumption of fish, waterfowl, and beef cattle that have been in contact with Goosequill water is highly hypothetical, PSC considers that the above CEDE doses need not be considered in combination with each other. Thus it is shown that even these unlikely pathways result in doses to the public that are within regulatory limits.

4.2.6.11 Inhalation

The REMP program collects atmospheric water vapor samples continuously by passive absorption on silica gel at two locations near the Goosequill ditch. Although elevated tritium levels have been observed, the 1988 REMP report (Reference 8) indicates that the tritium levels have always been below the limit of regulatory concern and that inhalation is not a significant pathway for dose to humans.

While the work platform is in use over the PCRV Shield Water, the Reactor Building ventilation system will be used to draw air down through the work platform access openings, over the surface of the shield water, and out the Reactor Building ventilation exhaust stack. Tritiated water vapor that evaporates from the surface of the shield water will thus be released as gaseous effluent. Potential doses will result to members of the public due to inhalation and due to ingestion of milk from cows and goats that inhale this effluent. This release will result in an atmospheric tritium concentration at the fenced boundary of the Emergency Planning Zone that is less than 1 % of the 10 CFR 20 MPC (2 E(-7) μCi/ml). Using the dose calculation methodology of ICRP-30, the resultant annual inhalation doses have been calculated in terms of a 50 year CEDE as follows:

Individual	Dose
Member of public at 10 mile Low Population Zone	1.47 E(-4) mrem
Member of public at 1/2 mile (Nearest Residence)	0.0147 mrem

These doses are well within the 5 millirem whole body annual dose limit of 10 CFR 50 Appendix I for individuals in unrestricted areas, due to gaseous effluents. As noted previously, the CEDE is a weighted sum of all organ and tissue doses and should be compared with whole body dose limits. All gaseous releases will be made in accordance with controls to be included in the Offsite Dose Calculation Manual (ODCM).

The area around the Goosequill ditch is unrestricted but it is not an area where members of the public or decommissioning workers would spend time. Occasionally brush or other trash must be removed from the ditch but this is not a time consuming task. There are no buildings or work stations located adjacent to the ditch, and occupational activity is very limited. This is not considered to be a significant pathway for tritium during decommissioning, either through inhalation or skin absorption.

4.2.6.12 Other Aquatic Pathways

Since tritium is a constituent of the water being released, diluted, and carried downstream, it will not be deposited along the shoreline or in sediment. Due to the distribution coefficient ion exchange discussed previously, other radionuclides are not expected to be released to the surface water. No other aquatic pathways are considered reasonable.

4.2.7 Enhanced Monitoring Program

At several times during plant operations, enhanced monitoring programs were implemented to better understand the impact of plant activities on the environment. In a similar manner, PSC will implement an enhanced monitoring program during the release of tritiated PCRV Shield Water.

This program will include weekly sampling of the farm pond outlet (from the continuous sampler), of upstream and downstream river locations, of the nearest farm wells, and of the Gilcrest well.

Adequate river flow is necessary for effluent dilution, however flow cannot be monitored at FSV because there is no river flow gauging station near the plant. To ensure adequate dilution, during the first three months of releases of PCRV Shield Water (when tritium concentrations are expected to be at their highest levels), PSC will enhance the environmental monitoring program by performing daily downstream sampling and analysis for tritium concentration. After the first three months, weekly sampling will be performed.

This enhanced monitoring program will allow PSC to take appropriate actions (e.g., adjust discharge flow, adjust dilution flows from the plant or the Jay Thomas Ditch) to ensure that tritium concentrations downstream of the convergence of the South Platte River and the St. Vrain Creek do not exceed the EPA Safe Drinking Water Standard (i.e., 20,000 pCi/l average).

The results of this enhanced monitoring program will be reported in the annual REMP reports that will be prepared throughout the decommissioning period.

PSC will also implement an enhanced surveillance of the Goosequill ditch to clean out debris in a timely manner. This will minimize opportunities for ditch water to back up and possibly overflow into pasture areas.

4.3 Radiation Protection Program and Occupational Radiation Exposure

4.3.1 Introduction

The primary objective of the Fort St. Vrain radiation protection program is to protect 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 Achievable (ALARA). The ALARA philosophy will be incorporated into all decommissioning activities and will have full management support.

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 9 and 10). The ALARA program will incorporate current technology and sound radiation protection practices to maintain exposure to ionizing radiation ALARA.

The decommissioning of Fort St. Vrain is scheduled to occur during the transition to the revised 10 CFR 20 regulations. PSC intends to implement the revised 10 CFR

20 no later than January 1, 1994, in accordance with the current schedule for providing sufficient regulatory guidance to allow for effective implementation. Depending on the projected status of the Fort St. Vrain decommissioning project as of January 1, 1994, PSC may apply for an exemption of the implementation of the requirement of the revised 10 CFR 20 for the remainder of the decommissioning project. Following implementation, the Radiation Protection Program will be periodically assessed against any newly issued regulatory guidance and modified, if necessary.

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.

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.

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

4.3.2 Occupational Exposure Estimates

The total cumulative on-site 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 4.3-1. 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. 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 (DSRC). The DSRC is an Administrative Control required by the Decommissioning Technical Specifications.

4.3.3 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. 11). 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.

4.3.4 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).

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

4.3.5 Contamination Control

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.

4.4 Ambient Air Quality

4.4.1 Fugitive Dust

During various demolition and dismantling operations, some fugitive dust will be generated. It is planned that the nonradioactive portions of the PCRV tophead concrete will be cut using diamond wire techniques. This technique uses water as a lubricant which minimizes the generation of fugitive dust. Any dust that is generated from this operation or similar operations involving concrete will normally be filtered by the existing reactor building ventilation system. This system consists of three trains, one of which is normally in continuous operation. This system contains exhaust filters composed of banks of moisture separators, HEPA filters and charcoal absorbers. Each bank contains 16 individual HEPA elements. In addition, portable air handling units with HEPA filters will be used to ventilate localized work areas, such as when cutting or sectioning operations are conducted in isolated areas. Finally, general dust control will be maintained by the use of HEPA-filtered vacuum cleaners and the use of scabblers equipped with shrouds connected to HEPA filtered vacuums as needed. Therefore, it is not anticipated that fugitive dust generated from decommissioning activities will have an adverse affect on ambient air quality.

4.4.2 Exhaust Emissions

Exhaust emissions from diesel-powered equipment and vehicles may have a slight impact on air quality. It is estimated that approximately 350 truck shipments of radioactive wastes will be made. In addition, several hundred shipments of nonradioactive wastes will be made to local landfills. However, the total number of vehicles at the site has been significantly reduced from the levels required to support continued plant operation for the duration of operating license.

The primary source of plant emissions during decommissioning activities will be the auxiliary boiler. While the plant was in operation, the auxiliary boilers, rated at 160,000 lbm/hr (total), were relied upon during shutdown conditions and start up operations to provide the steam motive power for the main feedpumps and helium circulators. The auxiliary boilers were also relied upon to provide steam heating for plant buildings and the PCRV. However, during decommissioning, one auxiliary boiler has been modified and derated to maximum capacity of 15,000 lbm/hr and will only be used for building heating. Consequently, the exhaust emissions during decommissioning activities will be significantly less than those emitted during plant operation. Therefore, with respect to exhaust emissions, decommissioning will have a positive environmental impact.

4.4.3 Asbestos

In the areas where it has been determined that asbestos must be removed or disturbed for decommissioning activities, removal of asbestos will be performed within control ventilated areas equipped with exhaust filtration to minimize the release of asbestos.

The public will be prohibited from entering work areas where asbestos removal and packing operations are being performed. The concentrations of asbestos fiber will be maintained within allowable levels. It is expected that no public exposure to asbestos will result from asbestos dismantling and packaging operations. All asbestos that is removed from non-radiologically controlled areas will be trucked to an approved landfill. All requirements of 29 CFR Parts 1910 and 1926 concerning asbestos removal, handling and packaging will be met.

4.5 Effects of Chemical and Biocide Discharges

The decommissioning process at Fort St. Vrain does not use chemical decontamination to any significant extent. It is planned to add sodium hydroxide to the PCRV Shield Water for pH control and hydrogen peroxide for a biocide. Approximately 600 lb. of 50 %(wt) sodium hydroxide and 550 gallons of 30 %(wt) hydrogen peroxide are planned to be used in total during decommissioning. In addition, sodium sulfite is used in the condensate water system that may be used as a makeup water supply. The only other effluent of a chemical or biocide nature expected is water containing the normal amount of cleaning fluids (detergent) that might be used to decontaminate walls and floors in the facility. These and any other chemicals used on-site will be released or disposed of in accordance with applicable NPDES permit requirements.

During plant operation, waste water was produced by demineralizer regeneration, blowdown from the main cooling tower and blowdown from the service water cooling tower. However, during decommissioning activities the primary source of discharge, the main cooling tower, will be out of service. Demineralizer regenerations as well as waste water discharged from the service water tower will be greatly reduced. It is therefore concluded that there will be greatly reduced consumption of chemical/biocides and resulting effluents during decommissioning. Waste water produced during decommissioning activities will continue to be discharged in accordance with the site NPDES permit. Figure 4.5-1 schematically shows the origins and pathways for various liquid effluents.

4.6 Effects of Sanitary Waste Discharge

Sanitary and sink drains are combined and treated in an activated sludge process which consists of a pre-aeration chamber, two aeration lagoons and a polishing pond. The final stage of treatment involves chlorination by the addition of calcium hypochlorite Ca(OCL) 2. The effluent from this pond (3 gpm, average) will be combined with the blowdown from the service water cooling tower and discharged from the Station (Figure 4.5-1) via the Goosequill Ditch to the South Platte River as it was done during plant operation.

Sanitary wastes from approximately 320 PSC workers and contractors on a daily basis have been previously treated by the site sewage aeration system as described above. The number of people required on site during decommissioning is expected to be a maximum of approximately 300 per day. With this decrease in personnel from that of the normal plant operations staff, the actual sanitary wastes to be treated and discharged will also be reduced. Therefore there will be less environmental impact from sanitary wastes discharges.

4.7 Endangered Species

Based on References 3 and 12, no significant impact on the flora and fauna of the site or its surrounding area has been experienced as a result of construction or operation of FSV. Accordingly, the impact on flora and fauna due to the decommissioning of FSV is expected to be negligible.

No natural habitat exists on land areas directly involved with decommissioning activities which harbors sensitive species. Noise and other activity associated with decommissioning activities is not expected to have significant impacts on wildlife species in the surrounding non-developed areas.

PSC has obtained two separate reports from the Colorado Department of Wildlife, Northeast Region (Reference 13). The area reported covers 3500 square miles and is bounded (approximately) on the north by the Colorado-Wyoming border, on the south by the city of Brighton, on the east by the town of Goodrich and on the west by Colorado Interstate Highway 25. Actual boundaries are south - Latitude 40 degrees; north - Latitude 41 degrees; east - Longitude 104 degrees; and west - Longitude 105 degrees. For this entire area, the following is the listing of threatened or endangered wildlife including birds, mammals, reptiles, and amphibians:

Bald Eagle	Endangered (Federal and State)
Peregrine Falcon	Endangered (Federal and State)
Whooping Crane	Endangered (Federal and State)

In the second listing provided to PSC, the Department of Wildlife prepared a separate list which included all species which occur in habitat types similar to those in the Fort St. Vrain area. The list is as follows:

Bald Eagle	Endangered (Federal and State)
Peregrine Falcon	Endangered (Federal and State)

The Department of Interior, Fish and Wildlife Service prepared a list of Federal threatened and endangered species which may occur within the area of influence of Fort St. Vrain and the site to be decommissioned. The list is as follows:

Bald Eagle
Black-footed Ferret

References 3 and 12 were prepared for and approved by the Atomic Energy Commission in 1972. Both the bald eagle and the peregrine falcon were identified in these documents as endangered. The Atomic Energy Commission granted PSC a license to construct and operate FSV based on their findings. The whooping crane does not occur in habitat types similar to those in the FSV area. Regarding the black-footed ferret, the Fish and Wildlife Service states "the standard that is used by the Service for determining possible project effects to black-footed ferrets is the disturbance of currently occupied prairie dog habitat." There is no such habitat on the Fort St. Vrain site.

4.8 Other Effects

4.8.1 Noise

There will be very little noise impacts as a result of the proposed dismantling project. In general, demolition activities will normally be conducted by one shift of up to 300 workers during the day. The exceptions to one shift operation are the diamond wire cutting of PCRV concrete and core component removal which will be on a two or three shift basis. This cutting method is very quiet and is not expected to be heard offsite. There will be no significant noise impacts during the evening hours.

The areas in the vicinity of the plant site are sparsely populated, and the nearest population center, Platteville, is approximately 3 1/2 miles Southeast of the site.

4.8.2 Effects of Runoff from Decommissioning

During the decommissioning activities there are no additional buildings constructed and no extra parking needed for workers or equipment. Decontamination work is planned to take place indoors so as to limit the amount of dust released to the environment (see Section 4.4.1). The precipitation runoff resulting from decommissioning is not expected to be greater than the runoff from Fort St. Vrain during normal plant operations.

4.8.3 Socioeconomic Impacts

Decommissioning is expected to be completed over a 57 month period and to employ a peak work force of approximately 300 (this includes PSC management, support staff and contractor personnel, but does not include any personnel involved with potential conversion activities). The temporary nature of the project and the limited work force support the conclusion that no significant demographic shifts or significant effects on the regional economy will result from the decommissioning.

The decommissioning work force will include administrative and clerical employees, technicians and engineers, security personnel and construction workers, including laborers, craftsmen, equipment operators, etc.

PSC plans to obtain all of its clerical, technical, and engineering staff from the existing work force. Most of the contractor technical and support personnel will be relocated to the Fort St. Vrain area. It is expected that most of the decommissioning construction workers will be drawn from the Denver regional labor pools.

Decommissioning is not expected to have a significant impact on regional or local employment and unemployment rates, whether the work force is drawn from the Denver and Front Range population or relocated to the area. Due to the specialized nature of the task, and the small labor force required, no significant impacts on local or regional labor markets or demand for services are anticipated.

4.9 References

1. "Evaporation Atlas for the 48 Contiguous United States," NOAA Technical Report NWS-33, Department of Commerce, 1982.

2. NUREG/CR-2907, BNL-NUREG-51581, Radioactive Materials Released from Nuclear Power Plants, Brookhaven National Laboratory, 1988.

3. Final Environmental Statement Related to Operation of the Fort St. Vrain Nuclear Generating Station of Public Service Company of Colorado, Docket No. 50-267, August 1972.

4. Transport in Surface Waters, Onishi et al, 1981.

5. Potential Groundwater Contamination by Tritium from Fort St. Vrain Nuclear Generating Station, Colorado State University Master's Thesis, C. A. Dacey, 1985.

6. CSU Letter, Dr. J. Johnson to S. Chesnutt, dated January 31, 1992 (G-92069); Subject: "Analysis of Possible Radiation Dose Commitments from the Release of Tritiated Water During Reactor Decommissioning."

7. Radiological Environmental Monitoring Program Summary Report for 1990, Colorado State University, submitted via PSC Letter, Crawford to NRC, dated April 12, 1991 (P-91133).

8. Radiological Environmental Monitoring Program Summary Report for 1988, Colorado State University, submitted via PSC Letter, Williams to NRC, dated April 20, 1989 (P-89151).

9. Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Reasonably Achievable," Revision 3, June 1978.

10. Regulatory Guide 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable," Revision 1R, May 1977.

11. NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactivity Materials," October 1976.

12. Public Service Company of Colorado, "Fort St. Vrain Nuclear Generating Station, Applicant's Environmental Report Operating License Stage," dated December 1970.

13. United States Department of the Interior, Fish and Wildlife Service, "Endangered Species List - Fort St. Vrain Power Station," dated November 21, 1990.

14. Final Environmental Statement, Haddam Neck Nuclear Generating Station, October 1973, Docket No. 50-213.

15. NRC Letter, Weiss to PSC, dated April 27, 1992 (G-92112); Subject: "Summary of 4/13/92 Meeting with PSC to Discuss Site Release Process for Fort St. Vrain."

TABLE 4.1-1
SOURCES OF PUBLIC WATER SUPPLY

DIST. MILES	MUNICIPALITY	POP. SERVED	TYPE OF SUPPLY	SOURCE OF SUPPLY
	DOWNSTREAM IN THE PLATTE RIVER VALLEY			
5-10	Gilcrest	382	Wells	Alluvium of South Platte River
10-15	LaSalle	1,300	Wells	Alluvium of South Platte River
10-20	Greeley	39,000	Surface	Cache la Poudre River; Colorado-Big Thompson Project; Nunn and Deadman Creeks
	OTHER MUNICIPALITIES			
3- 4	Platteville	950	Wells	Alluvium
5-10	Mead	900	Surface	Big Thompson River and St. Vrain Creek
10-15	Johnstown	1,200	Surface	Big Thompson River
10-15	Fort Lupton	4,000	Wells	Alluvium of South Platte River
10-15	Frederick	1,500	Surface	Boulder Creek
10-15	Longmont	50,000	Surface	North and South St. Vrain Creek
10-20	Loveland	30,000	Surface	Big Thompson River
10-15	Berthoud	3,200	Surface	Big Thompson River
15-20	Hudson	540	Wells	Alluvium
15-20	Brighton	13,000	Wells	Alluvium of South Platte River
15-20	Erie	1,375	Surface	South Boulder Creek
15-20	Windsor	1,500	Surface	Greeley
20-25	Eaton	1,500	Wells	Alluvium
20-25	Keenesburg	475	Wells	Laramie and Fox Hills formations
20-25	Broomfield	20,000	Wells	Fox Hills sandstone
20-25	Lafayette	10,000	Surface	South Boulder Creek, Woneka Reservoir
20-25	Louisville	6,000	Surface	South Boulder Creek
20-25	Lyons	1,340	Surface	North St. Vrain Creek
20-25	Timnath	150	Surface	Greeley
20-30	Fort Collins	80,000	Surface	Cache la Poudre River
25-30	South Adams Water and Sanitary Dist. (Commerce City)	25,000	Wells	Alluvium of South Platte River
25-30	Lochbuie	1,000	Wells	Alluvium of South Platte River
25-30	North Huron Water Dist (near Broomfield)	80	Wells	Fox Hills sandstone
25-30	Northwest Utilities Company	15,000	Wells	Arapahoe and Fox Hills formations; alluvium of South Platte River
25-30	Federal Heights	8,000	Wells	Arapahoe and Fox Hills formations
25-30	Westminster	60,000	Surface	Clear Creek Wells Arapahoe and Fox Hills formations
25-30	Boulder	96,000	Surface	North Boulder Creek
25-30	Jamestown	230	Ground	Alluvium

TABLE 4.1-2
FORT ST. VRAIN WIND SPEED AND DURATION

PERIOD OF RECORD: 1986-1989
STABILITY CLASS: All Classes

Wind Speed (mph) at 10m Level

WIND DIRECTION	1-3	4 - 7	8 - 12	13 - 18	19 - 24	>24	TOTAL
N	545.[illegible]	579.12	355.24	272.14	76.43	21.50	1849.56
NNE	738.41	729.08	420.54	217.72	72.83	25.22	2202.80
NE	803.04	964.27	353.48	101.27	19.11	3.53	2244.70
ENE	820.38	105[illegible].09	303.93	38.20	4.26	1.26	2219.12
E	597.95	845.41	227.74	27.21	2.77	0.76	[illegible]01.84
ESE	570.52	748.32	256.38	41.60	4.52	1.51	1622.85
SE	526.77	584.33	231.54	61.27	6.04	2.77	1412.72
SSE	637.06	666.42	265.01	68.02	23.41	9.85	1669.77
S	872.38	805.30	228.23	56.06	19.41	7.31	1988.69
SSW	1072.95	937.43	120.03	23.18	2.92	2.36	2158.87
SW	1204.10	1537.78	157.65	24.11	5.54	2.27	2931.45
WSW	867.01	1113.02	166.28	62.69	11.57	6.03	2226.60
W	369.11	263.26	75.50	50.46	26.98	10.84	796.15
WNW	205.06	169.15	84.86	90.29	29.78	20.91	600.05
NW	278.20	299.73	160.76	87.95	29.96	8.83	865.43
NNW	388.84	380.16	221.87	129.06	36.58	4.28	1160.79
VARIABLE	0.00	0.00	0.00	0.00	0.00	0.00	0.00
TOTAL	10496.91	11673.87	3629.04	1351.23	371.11	129.23	27651.39

Periods of calm (hours): 1241.77
Hours of missing data: 1728.56

TABLE 4.1-3
FORT ST. VRAIN WIND FREQUENCY DISTRIBUTION

PERIOD OF RECORD: 1986-1989
STABILITY CLASS: All Classes

Wind Speed (mph) at 10m Level

WIND DIRECTION	1 - 3	4 - 7	8 - 12	13 - 18	19 - 24	>24	TOTAL
N	0.020	0.021	0.013	0.010	0.003	0.001	0.067
NNE	0.027	0.026	0.015	0.008	0.003	0.001	0.080
NE	0.029	0.035	0.013	0.004	0.001	0.000	0.081
ENE	0.030	0.038	0.011	0.001	0.000	0.000	0.080
E	0.022	0.031	0.008	0.001	0.000	0.000	0.062
ESE	0.021	0.027	0.009	0.002	0.000	0.000	0.059
SE	0.019	0.021	0.008	0.002	0.000	0.000	0.051
SSE	0.023	0.024	0.010	0.002	0.001	0.000	0.050
S	0.032	0.029	0.008	0.002	0.001	0.000	0.072
SSW	0.039	0.034	0.004	0.001	0.000	0.000	0.078
SW	0.044	0.056	0.006	0.001	0.000	0.000	0.106
WSW	0.031	0.040	0.006	0.002	0.000	0.000	0.081
W	0.013	0.010	0.003	0.002	0.001	0.000	0.029
WNW	0.007	0.006	0.003	0.003	0.003	0.001	0.022
NW	0.010	0.011	0.006	0.003	0.001	0.000	0.031
NNW	0.014	0.014	0.008	0.005	0.001	0.000	0.042
VARIABLE	0.000	0.000	0.000	0.000	0.000	0.000	0.000
TOTAL	0.380	0.422	0.131	0.049	0.013	0.005	1.000

Periods of calm fraction: 0.045
Fraction of missing data: 0.063

TABLE 4.3-1
OCCUPATIONAL RADIATION EXPOSURE ESTIMATES

(PCRV DISMANTLEMENT ACTIVITIES)

DESCRIPTION OF WORK ACTIVITY	SCHEDULED WORK TIME (Per-Hrs)	WORKER EXPOSURE TIME[(1)] (Per-Hrs)	CREW AVG. RADIATION FIELDS (Per-Hrs)	WORKER EXPOSURE (Per-Hrs)
PCRV INITIAL PREPARATION/DISASSEMBLY				
Modify Main Crane	682	341	0.1	0.03
Detension PCRV Tendons	25,590	12,795	0.1	1.28
Remove Core Elements with FHM				
PCRV Region Constraint Devices	2,208	1,104	1.7	1.88
Remove Metal Clad & CRD Blocks	12,267	6,134	0.4	2.45
Helium Purification Component Wells	2,013	1,007	1.0	1.00
SUBTOTAL		21,381		6.64
SHIELDED ACCESS TO PCRV				
Seal PCRV Cooling Tubes & Tendon Conduits	4,110	2,055	1.0	2.06
Center Access Penetration	1,190	595	1.1	0.65
PCRV Shield Water System	4,980	2,490	1.0	2.49
Airborne Contamination Control System	3,633	1,816	0.3	0.54
Cut PCRV Top Head	21,660	10,830	1.1	11.91
Flood PCRV	180	90	0.6	0.05
PCRV Cavity Shielded Work Platform	1,325	663	1.0	0.66
SUBTOTAL		18,539		18.36
DISMANTLE PCRV CORE COMPONENTS				
Defueling Elements	16,683	8,342	1.7	14.18
Replaceable and Permanent Hex Reflector Blocks	16,683	8,341	3.7	30.86
Large Side Reflector Blocks	27,782	13,891	3.6	50.00
Boronated Spacer elements	16,683	8,342	1.9	15.85
Hastelloy Can Hex Reflector Blocks	8,341	4,170	6.8	28.36
Core Support Blocks and Posts	2,780	1,390	1.8	2.50
SUBTOTAL		44,476		141.75

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

TABLE 4.3-1
Occupational Radiation Exposure Estimates

(PCRV DISMANTLEMENT ACTIVITIES)

DESCRIPTION OF WORK ACTIVITY	SCHEDULED WORK TIME (Per-Hrs)	WORKER EXPOSURE TIME(1) (Per-Hrs)	CREW AVG. RADIATION FIELDS (Per-Hrs)	WORKER EXPOSURE (Per-Hrs)
CORE BARREL, INSULATION & CSF D/D				
Core Barrel and 24 Outer Keys	4,913	2,456	9.2	22.59
24 Core Barrel to Graphite Lower Key Removal	952	476	13.3	6.33
24 Core Barrel to Graphite Upper Key Removal	1,624	812	10.0	8.12
Core Support Floor	7,762	3,881	12.5	48.51
Top CSF Insulation	1,350	675	11.2	7.56
SUBTOTAL		8,300		93.11
PCRV LOWER PLENUM D/D				
12 Steam Generator Primary Modules	14,688	7,344	0.8	5.88
12 Steam Generator Secondary Modules	7,594	3,797	4.7	17.85
4 Helium Circulator Primary Modules	1,056	528	11.1	5.86
4 Helium Circulator Secondary Modules	1,144	572	3.5	2.00
CSF Columns, Lower Plenum Floor, and Supports	1,144	572	22.2	12.70
PCRV Inside Top, Bot. & Side Insulation/Plates	3,388	1,694	5.7	9.66
SUBTOTAL		14,507		53.95
FINAL PCRV DISMANTLE, DECON & CLEANUP				
Remove Beltline Activated Concrete	17,284	8,642	1.7	14.52
Decon Lower PCRV Liner	984	492	0.8	0.39
PCRV Wall & Liner Penetrations;	7,404	3,702	0.2	0.79
PCRV Safety Valve Instrumentation & Piping				
Demobilize and Cleanup Area	1,440	720	0.3	0.22
Decon PCRV for Final Release Survey	1,440	0	0.0	0.00
SUBTOTAL		13,556		15.92
HP & QA COVERAGE (11%)		13,283		36.26
GRAND TOTAL - PCRV ORE		134,042		366.00

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

TABLE 4.3-1
Occupational Radiation Exposure Estimates

(BOP DISMANTLEMENT ACTIVITIES)

DESCRIPTION OF WORK ACTIVITY	SCHEDULED WORK TIME (Per-Hrs)	WORKER EXPOSURE TIME[1] (Per-Hrs)	CREW AVG. RADIATION FIELDS (Per-Hrs)	WORKER EXPOSURE (Per-Hrs)
RADIOLOGICAL CHARACTERIZATION	7,279	7,279	<1	0.25
BOP DISMANTLEMENT OPERATION				
System 13 Fuel Handling System	4,648	4,648	<1	0.22
System 14 Fuel Storage Wells	3,742	3,742	<1	0.10
System 16 HSF, ATC and ESWs	4,477	4,477	<1	0.10
System 23 Helium Purification System	5,448	5,448	<1	0.23
System 46 Reactor Plant Cooling Water	1,500	1,500	<1	0.06
System 47 Purification Cooling Water	250	250	<1	0.02
System 61 Decontamination System	3,493	3,493	<1	0.10
System 62 Liquid Waste System	10,610	10,610	<1	0.25
System 63 Gas Waste System	9,948	9,949	<1	0.13
System 72 Reactor Building Drain	4,577	4,577	<1	0.02
System 73 Reactor Plant Ventilation	1,694	1,694	<1	0.02
System 93 I & C Piping, Instrumentation, & Structure External to PCRV	1,370	1,370	<1	0.02
Contaminated Laundry Facility & Radwaste Compactor	930	930	<1	0.01
HP coverage (10%)		5,997		0.15
SUBTOTAL BOP		65,963		1.68
RADWASTE PROCESSING AND SHIPPING	30,050	30,050	VARIOUS	59.43
HP Coverage (10%)		3,005		5.94
SUBTOTAL		33,055		65.37
GRAND TOTAL - PCRV, BOP, RADWASTE PKG		233,060		433.06

[1] - Exposure work time (worker efficiency) is estimated to be 100% of scheduled work time for BOP and Radwaste tasks where the potential for radiation exposure exists.

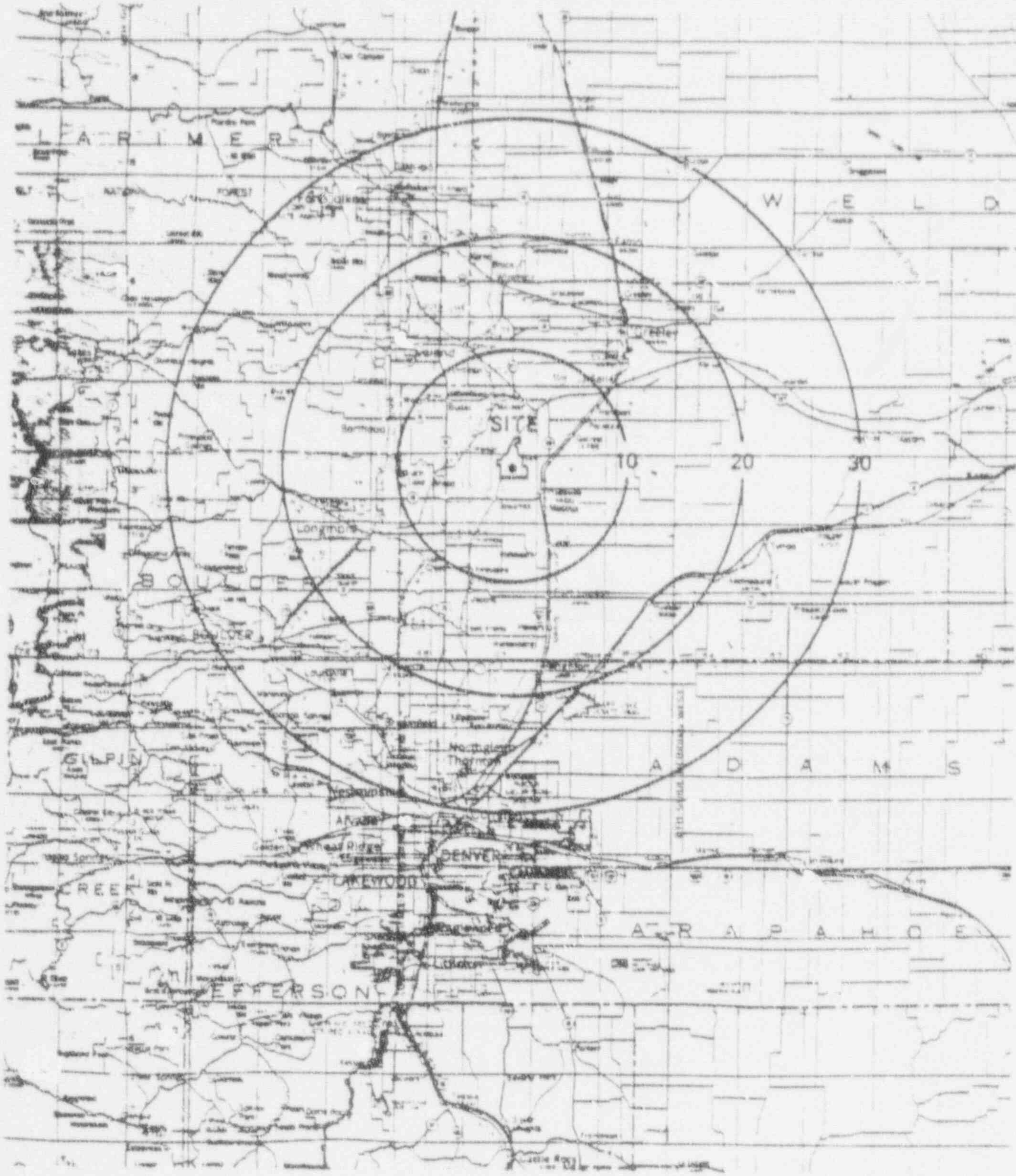

Figure 4.1-1
Area Surrounding Fort St. Vrain Site

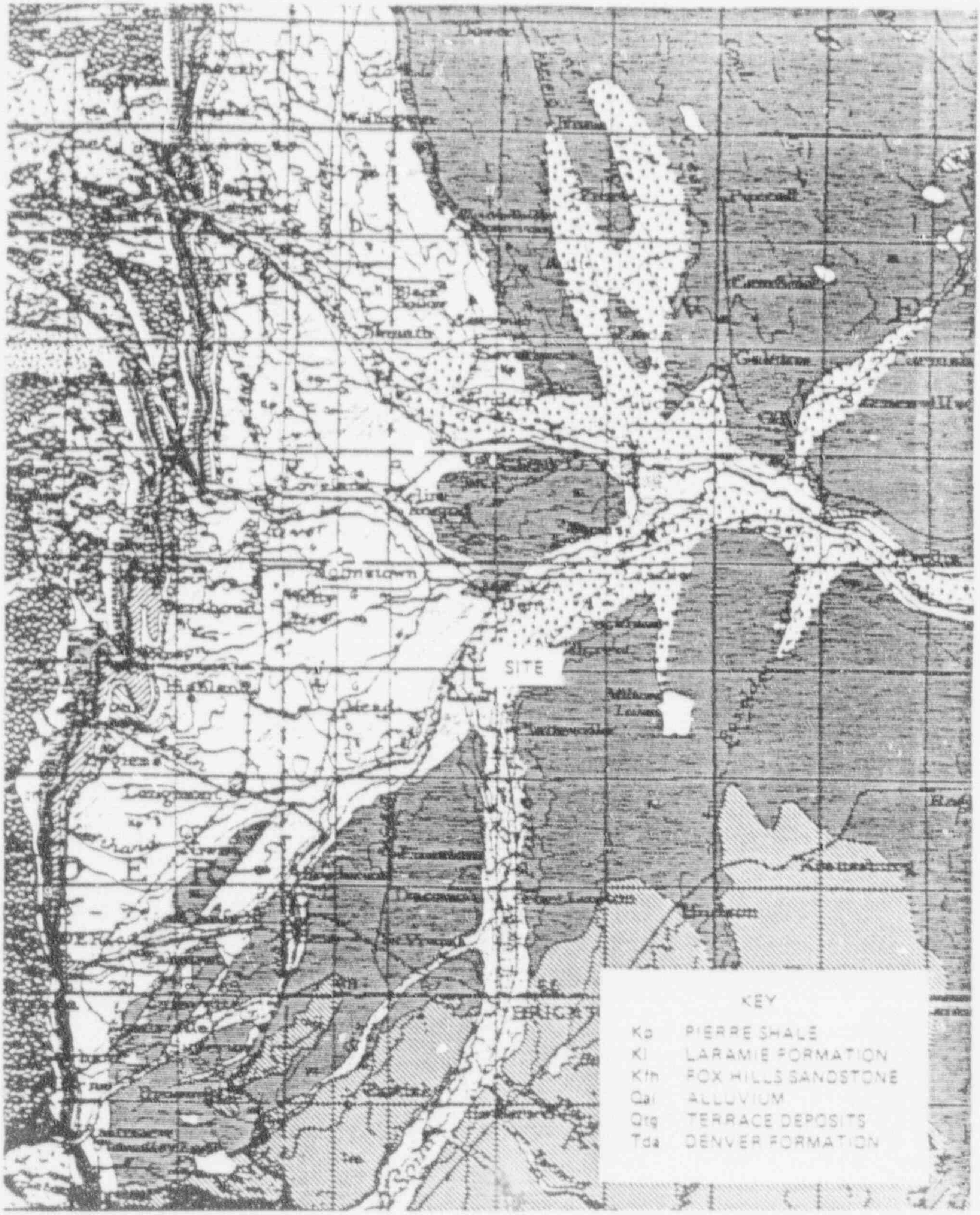

Figure 4.1-2
Geological Structure of General Area

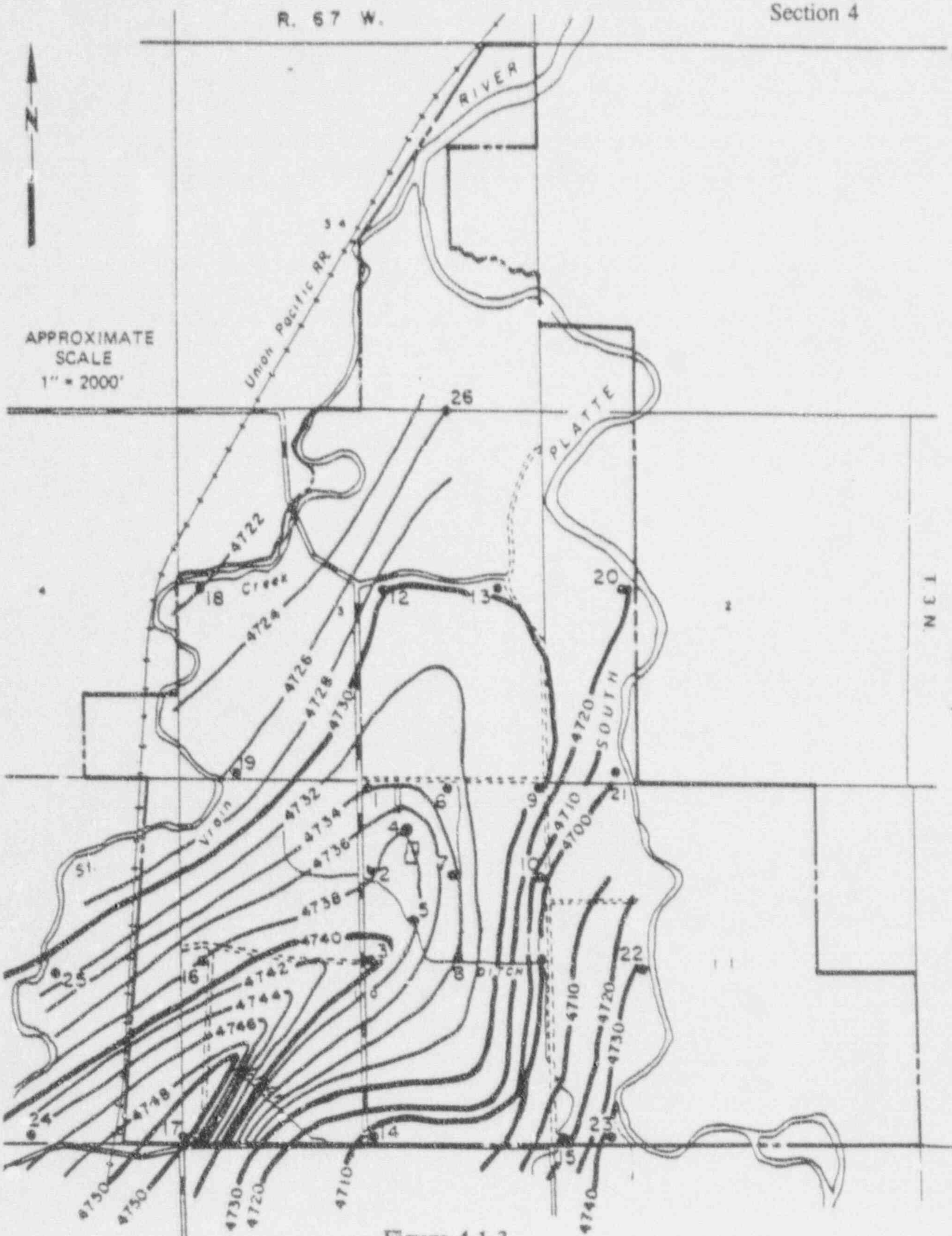

Figure 4.1-3
Contours of Bedrock

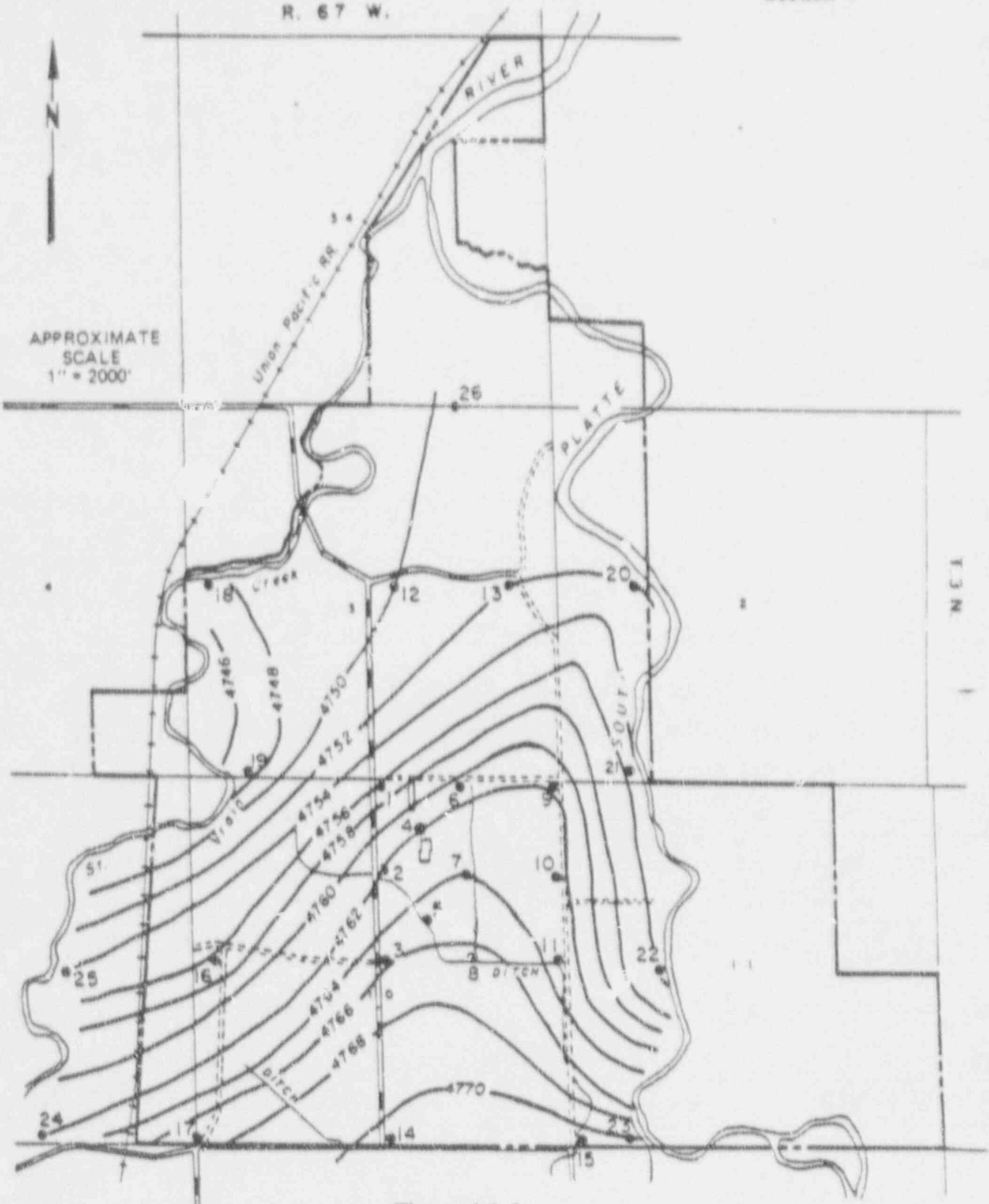

Figure 4.1-4
Estimated Water Table Contours
Fort St. Vrain Nuclear Generation Station

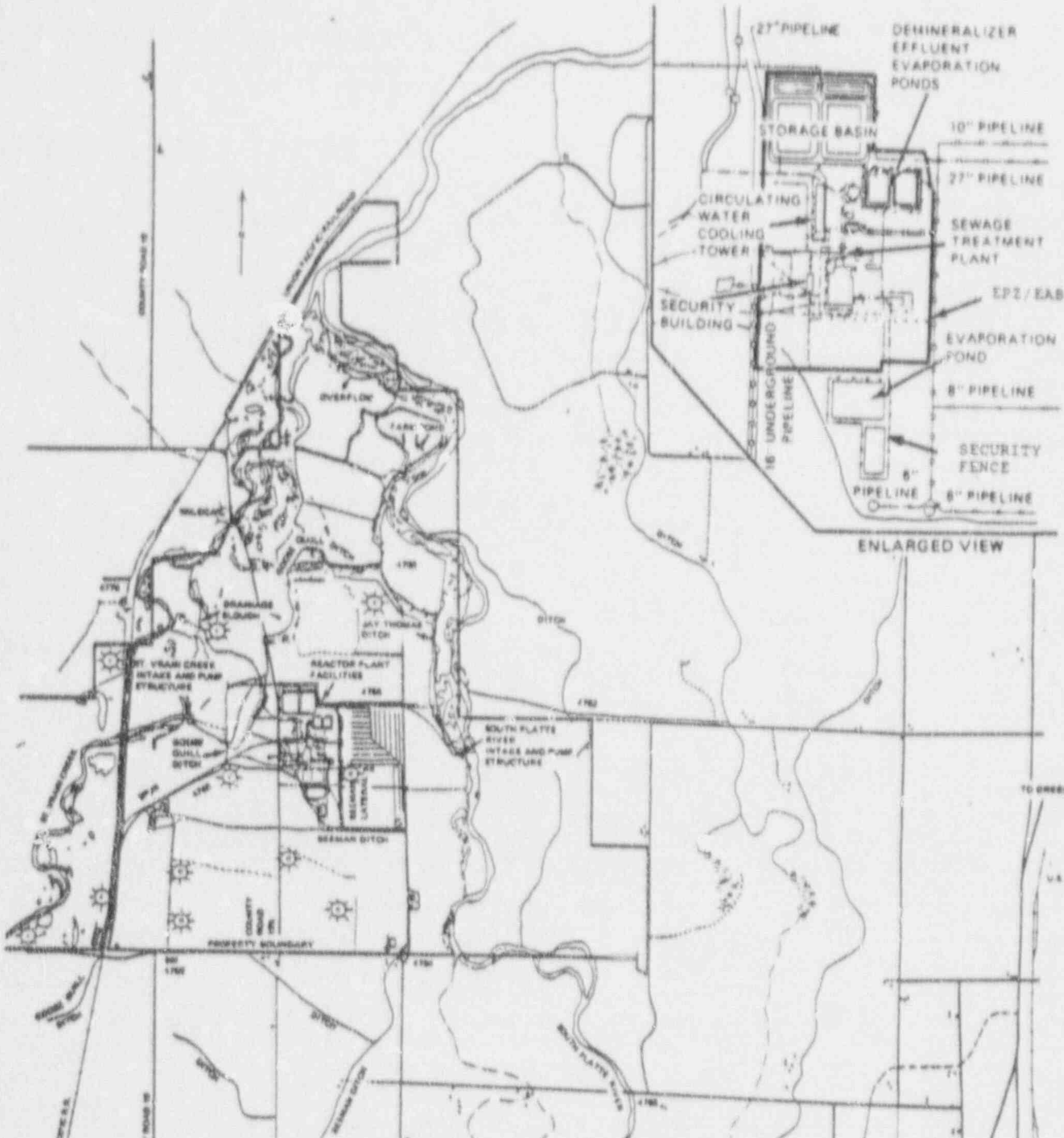

Figure 4.1-5
Plant Water Supply Systems
and Effluent Drainage Paths

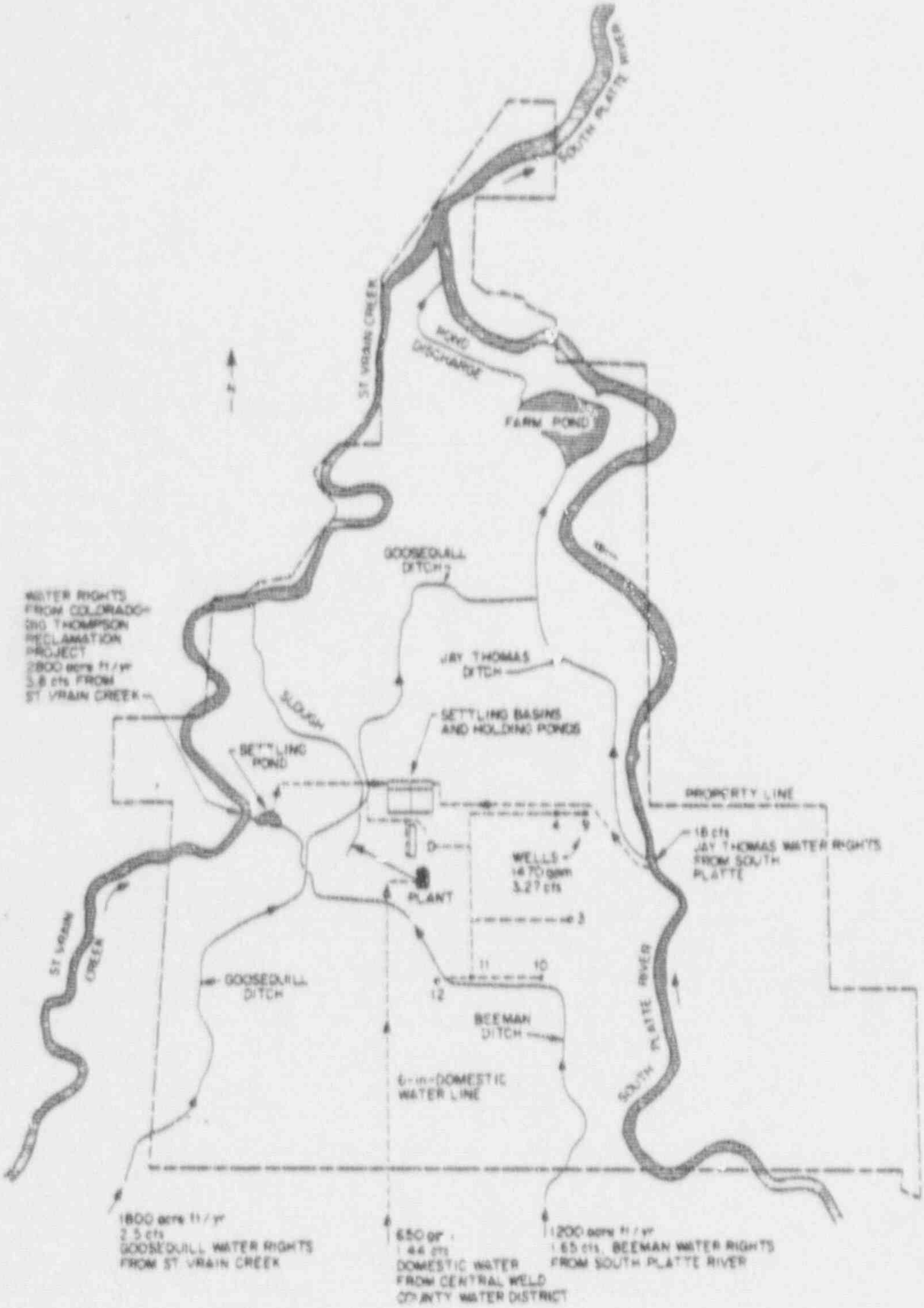

Figure 4.1-6
Irrigation Ditches Around Fort St. Vrain

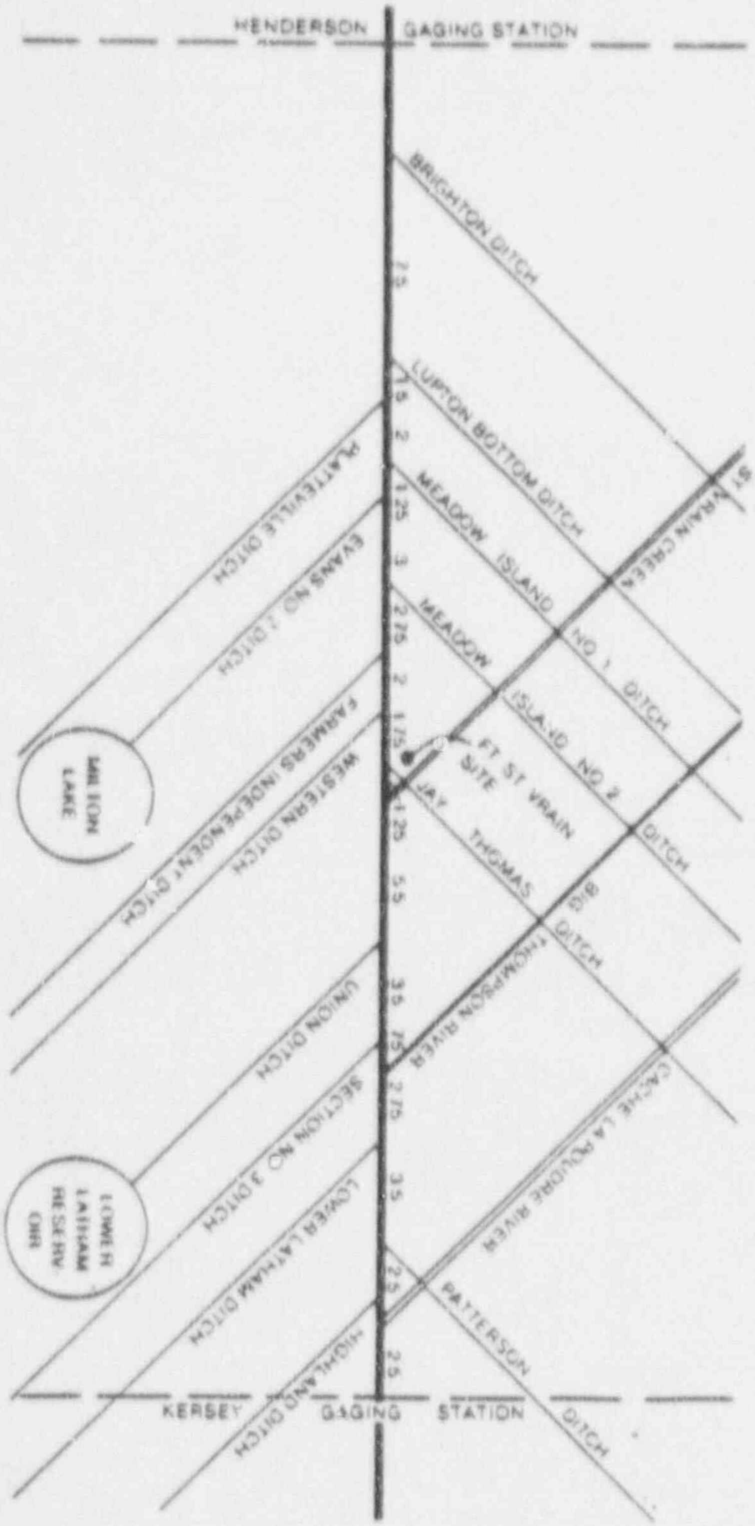

Figure 4.1-7
Major Tributaries and Irrigation Ditches, South Platte River Between Henderson and Kersey, Colorado

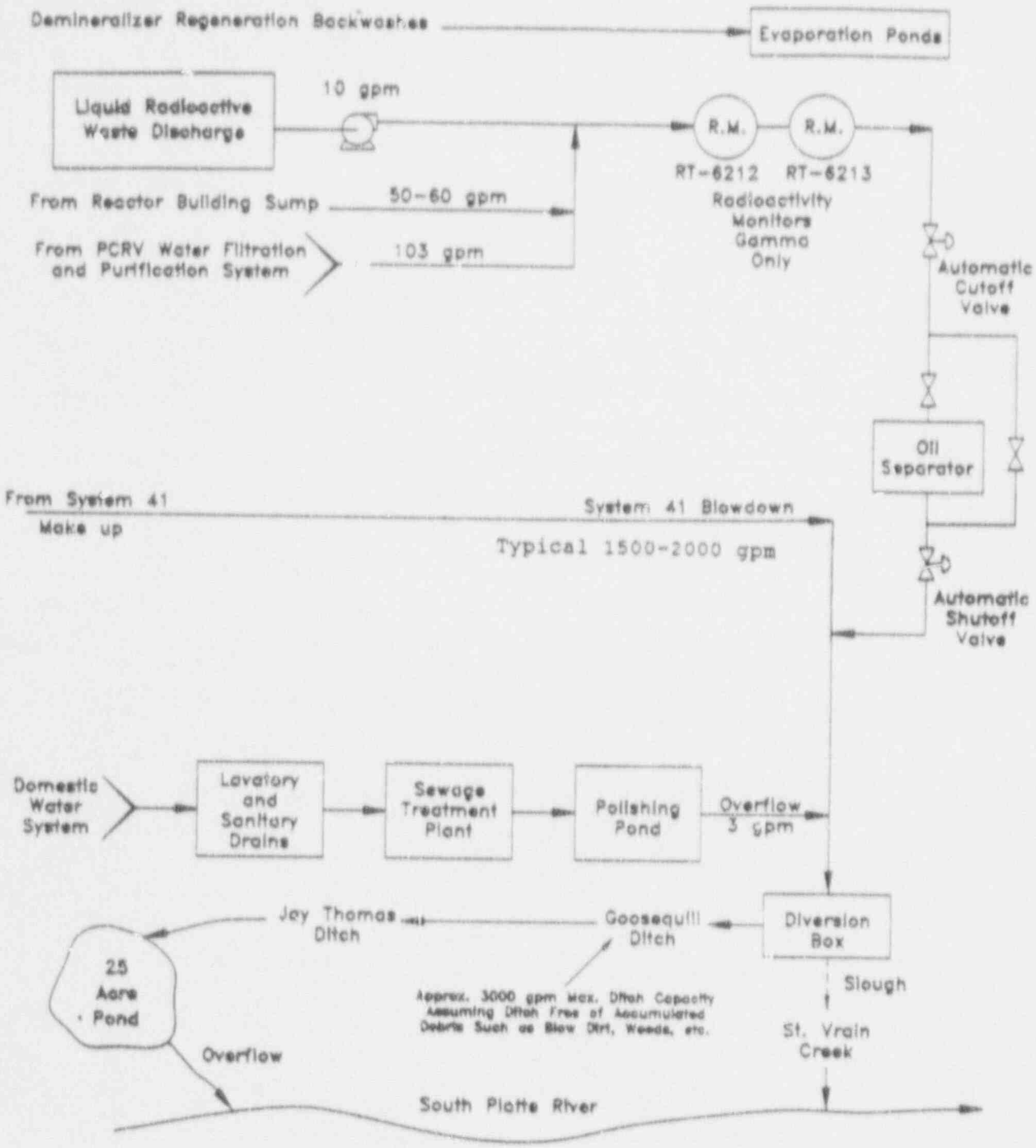

Figure 4.5-1
Liquid Waste Discharges from Fort St. Vrain

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5.0 ENVIRONMENTAL IMPACTS OF ACCIDENTS

5.1 Facility Accidents Involving Radioactivity

5.1.1 Introduction

The radiological impact on the general public within 30 miles of Fort St. Vrain for seven accident scenarios (Reference 1) were analyzed using the AIRDOSE-EPA computer code (Reference 2).

The AIRDOSE-EPA computer code is a methodology that estimates radionuclide concentrations in air; rates of deposition on ground surfaces; ground surface concentrations; intake rates via inhalation of air and ingestion of meat, milk, and fresh vegetables; and radiation doses to man from airborne releases of radionuclides. These doses are presented in terms of one year committed effective dose equivalents.

A modified Gaussian plume equation is used to estimate both horizontal and vertical dispersion of as many as 36 radionuclides released from one to six stacks or area sources. Radionuclide concentrations in meat, milk and fresh produce consumed by man are estimated by coupling the output of the atmospheric transport models with the Regulatory Guide 1.109 (Reference 3) terrestrial food chain models. Dose conversion factors are input to the code, and dose to man at each distance and direction specified are estimated for red bone marrow, lungs, endosteal bone tissue cells, breast, thyroid, and gonads through the following exposure modes:

- immersion in air containing radionuclides,
- exposure to ground surfaces contaminated by deposited radionuclides,
- immersion in contaminated water,
- inhalation of radionuclides in air, and
- ingestion of feed produced in the area.

Meteorology data used in the analysis was obtained from the Fort St. Vrain meteorological station. Population data was based on the 1980 United States census. Estimates of significant water areas were obtained from map data supplied by the United States Geological Survey (USGS).

The accident scenarios and the corresponding radiological source terms are taken from Section 3.4 of the Proposed Decommissioning Plan. However, the dose analysis to a maximally exposed individual has been performed with the AIRDOSE-EPA code.

The area shown in Figure 5.1-1 is a circular area of 7319.2 square kilometers (2826 square miles) with the Fort St. Vrain plant at the center. The population within this square was estimated at 1,531,600. The sectors making up this area are 4828 m. (5280 yards) on a side. The dose output from the AIRDOSE-EPA computer code is keyed to a radial distance from the plant as well as a compass direction for the radionuclides mentioned.

The risk of accidents resulting in a radiological release during decommissioning activities is considerably less than during plant operation, due to the removal of irradiated fuel from the Reactor Building. Since the reactor will be defueled prior to the commencement of decommissioning operations and all fuel will be removed from the Reactor Building, only non-reactor accident scenarios are evaluated in this section. The focus of these decommissioning accident analyses will be on public health and safety.

The following postulated accident scenarios have been analyzed, considering activation levels and isotopic composition of components to be processed, and the anticipated dismantling activities:

1. Dropping of contaminated concrete rubble
2. Conversion construction near PCRV dismantlement
3. Heavy load drop
4. Fire
5. Loss of PCRV shielding water
6. Loss of Power
7. Natural disasters
8. Dropping of a steam generator primary module

The components with the highest activation levels were used in the accident analyses. Therefore, accidents that were analyzed bound the radiological consequences from other postulated accident scenarios. In evaluating the postulated accidents, conservative assumptions were made when data or knowledge to support more realistic analyses were lacking. Conservatism in this context is defined to mean that the radiological consequences from the postulated accidents will be overestimated rather than underestimated.

A capsule summary of the accident scenarios is given in Table 5.1-1. The short-term (0-2 hr) doses from these accidents are discussed in Chapter 3 of the Proposed Decommissioning Plan. The doses to a maximally exposed off-site individual from the postulated accidents, are presented in Table 5.1-2. From this table, the limiting accident is a tornado-induced missile resulting in a red bone marrow dose of 7.2 E(-5) mrem and a dose of 2.0 E(-4) mrem to the lungs. These doses are minuscule and well within the 25 rem whole body dose and 300 rem to any specific organ guidelines to a maximum individual established by 10 CFR 100.

These doses are also a small fraction of the 1 rem whole body dose and 5 rem to any specific organ dose guidelines cited in EPA Protective Action Guidelines (Reference 4).

The following natural disasters were considered in the accident analyses:

External Event	Mitigating Feature	Radiological Consequences
Earthquake	Low Probability of Occurrence	Not postulated; See Section 5.1.9
High Winds, Hail	Bounded by Tornado	See analysis in Section 5.1.9
Rainfall, Flood	Site Location	No release
Range Fire	Plant buffer	No release

The activity concentrations of the various components used in the following accident analyses were derived from the detailed neutron activation analysis (Reference 5). Where chemical impurities were involved in neutron activation reactions, the maximum impurity levels permitted by the pertinent specifications were conservatively assumed to exist. For impurities not governed by specifications, nuclides of interest were assumed to be present in the graphite in concentrations that were considered conservative. With the exception of tritium concentrations, the radioisotope concentrations of interest used in the accident analyses have been taken directly from the activation analysis. Tritium concentrations predicted by the activation analysis were considered extremely unrealistic for the following reasons:

1. In the activation analysis, the dominant source of tritium was from activation of lithium impurities. The activation analysis assumed that no tritium formed by lithium activation migrated out of the graphite into the primary coolant. The lithium concentration assumed to be present prior to irradiation in the graphite blocks was based on the maximum concentration permitted by the specifications. In actuality, lithium is relatively volatile and tends to migrate out of the graphite during the high temperature graphitization process. Therefore, it is considered probable that the lithium impurity concentrations in the graphite used to form the large side reflectors and side spacer blocks were an order of magnitude lower than the maximum specification limit.

2. The large graphite side reflectors and side spacer blocks were exposed to relatively low temperatures (300-500 degrees C) during reactor operations. These low temperatures preclude a significant amount of tritium from being chemically absorbed in the graphite and retained.

Since tritium has a small atomic radius, it is likely that tritium formed by activation of lithium (Li-6 and Li-7) will migrate out of the graphite. Due to this temperature dependence of chemical absorption, it is considered that tritium concentrations are two or three orders of magnitude below those predicted by the activation analysis.

3. In the presence of moisture, hydrogen atoms from water molecules compete with and replace tritium atoms at active carbon sites in the graphite matrix, releasing tritium from the graphite. During FSV reactor operation, a number of moisture ingress events occurred, in which measured tritium levels in the primary coolant increased significantly. This was probably caused by exchange of hydrogen in the water with tritium in the graphite. Before the graphite blocks are removed from the PCRV, they will be submerged under water when the PCRV is flooded. Exposure of the irradiated graphite to moisture, both during reactor operation and after core defueling, is expected to result in the release of a small but noticeable portion of the total tritium.

Based on the effects noted above, it is considered that a value of 10 μCi/g of tritium represents a conservative estimate of tritium concentration in the large side reflector and side spacer blocks (Reference 6). While this concentration is a factor of approximately 40 below that projected in the activation analysis for these blocks, it provides a more realistic representation of the tritium concentration of the graphite blocks after they are removed from the PCRV. Therefore, a tritium concentration of 10 μCi/g in the large side reflector and side spacer blocks is assumed for the postulated decommissioning accident scenarios, with the exception of the loss of PCRV shielding water accident. The loss of PCRV shielding water accident analysis conservatively assumed the graphite blocks contained the levels of tritium predicted by the activation analysis, and that all of this tritium (nearly 100,000 Curies) migrated out of the graphite into the PCRV shield water.

5.1.2 Assumptions

The following are the major assumptions used in the analysis of postulated accidents which may occur during the dismantling activities:

1. The reactor is defueled and all irradiated fuel is removed from the Reactor Building.

2. Since all fuel is removed from the reactor, there will be no need for shutdown/cooldown systems such as decay heat removal.

3. The Reactor Building ventilation system will remain operable, providing filtration of effluents to the environment, while the potential exists for drop of a large activated graphite block.

4. The analyses for some of the accidents conservatively assume a Curie content that exceeds allowable Curie contents for a Low Specific Activity (LSA) Type A-2 waste container, as specified in Table A-1 of 10 CFR 71.

5.1.3 Dropping of Contaminated Concrete Rubble Accident

5.1.3.1 Identification of Cause

After the majority of the PCRV top head concrete is removed in large pieces by diamond wire cutting, a thin wafer of concrete (two to three inches of concrete just above the PCRV top head liner) will be removed by utilizing a mechanical breaker to break up the concrete around the perimeter of the PCRV top head liner, enabling the removal of the remaining concrete wafer in sections. This accident scenario assumes that radioactivity is released from the drop of the rubblized concrete produced from this activity and prepared for shipment in a transport container. The drop of the transport container occurs due to a faulty crane or operator error.

5.1.3.2 Accident Description

An activation analysis performed for Fort St. Vrain (Reference 5) shows that the highest concentration of radioactivity in the PCRV concrete is in the six inch increment of the PCRV top head immediately above the top head liner as shown in Table 5.1-3. The values in Table 5.1-3 are based on three years decay, the approximate time frame in which the dismantling work is expected to take place. The percentage contribution of activation products within this concrete is given in Table 5.1-4. As shown, nearly 100 % of the total activity is accounted for by the nuclides listed.

Only two to three inches of concrete will remain adjacent to the PCRV top head liner when it is being removed. However, six inches of concrete were conservatively assumed to remain for ease of determining concentration of activation products, since the activation analysis evaluated concrete in six inch increments. A portion of this concrete will be rubblized to enable access to the PCRV liner for thermal cutting.

Iron and cobalt are trace constituents in cements and aggregates. Therefore, the activation products Fe-55 and Co-60 will occur in the concrete of the PCRV, although in much smaller concentrations than in the steel rebar. However, none of the activation products in the embedded rebar are assumed released in this postulated accident since the rebar, unlike the concrete, would remain intact upon impact. Of the entire PCRV 5.8% by weight (2% of volume) is rebar. Based on the Activation Analysis (Reference 5), a total of 98.3 Curies are contained in the 6 inches of concrete adjacent to the top head liner. Of this total, 32.83 Curies of Fe-55 and 1.43 Curies of Co-60 are contained within the concrete. It was conservatively assumed that the remaining activity is 9.83 Curies (10% of the total), and that 60% of this remaining activity is Eu-154 and 40% tritium, since tritium comprises approximately 40% of the activity and Eu-154 has the highest dose conversion factor of all the other isotopes involved.

It is conservatively assumed that 10 % (approximately 7,500 lbs.) of the concrete on top of the PCRV top head liner is rubblized and packaged in a transport container that is then dropped in the Reactor Building. The resulting impact is assumed to cause 1% of the activity in the concrete to be released to the Reactor Building atmosphere.

The airborne activity was calculated to be 32.8 millicuries of Fe-55, 1.43 millicuries of Co-60, 3.93 millicuries of tritium and 5.90 milliCuries of Eu-154. No credit was taken for particulate filtration by the Reactor Building ventilation system.

5.1.3.3 Analysis of Effects and Consequences

This scenario was modeled as an elevated stack release using the AIRDOSE-EPA computer code. The maximum individual dose is 7.1 E(-7) mrem red bone marrow and 1.8 E(-6) mrem to the endosteal bone tissue.

5.1.4 Conversion Construction Accident Near PCRV Dismantlement

5.1.4.1 Identification of Causes

1. Crane Failure

An evaluation was performed on the potential impact of a construction crane toppling which would impact the Reactor Building. Due to the proximity of the planned new boiler building to the Reactor Building, it will be possible for a crane boom to strike the Reactor Building above the refueling floor level.

A crane boom is relatively light and fragile. An impact with the Reactor Building is not expected to cause structural damage to the building. At worst, the crane boom could drape over the Reactor Building siding. No radiological impact is expected from such an accident. LSA containers located outside the Reactor Building will be protected if they are stored within the fall radius of the construction cranes. This accident is bounded by the heavy load drop (Section 5.1.5) and tornado (Section 5.1.9).

2. Explosion/Fire Due to Natural Gas Line Leak:

There are plans to repower Fort St. Vrain with a natural gas-fired boiler. Accidents can be postulated during decommissioning activities involving a natural gas line leak resulting in an unconfined vapor explosion or fire, or an explosion of the gas-fired boiler itself. The decommissioning and repowering schedules have been reviewed.

There are over three years between completion of the removal of highly radioactive components (graphite blocks) from the PCRV and introduction of natural gas on site to supply the gas boiler. In the event of a slippage in the dismantling schedule, administrative controls will be implemented to prevent charging the gas-fired boiler natural gas line on site concurrent with handling of the activated graphite blocks from the PCRV. Therefore, given the actual schedule and administrative controls, an explosion or fire due to a natural gas line leak associated with the Fort St. Vrain conversion project is not credible during the decommissioning process.

Natural gas wells and collection pipeline facilities are located in the vicinity of Fort St. Vrain. In 1991, extensive analyses were performed to determine potential effects of accidents involving natural gas releases on the Reactor Building. These analyses concluded that worst case postulated detonations of unconfined natural gas vapor clouds resulting from pipeline ruptures would not affect the structural integrity of the Reactor Building, and the Reactor Building blowoff panels would remain in place. The NRC also arrived at this conclusion, as documented in the NRC safety evaluation contained in Reference 7.

As a result of the concerns associated with natural gas facilities in the vicinity of Fort St. Vrain, Decommissioning Technical Specifications contain a restriction on the introduction of new natural gas sources into the vicinity of Fort St. Vrain. This restriction precludes the introduction of natural gas to supply the conversion boiler, or any other new natural gas source, within 0.5 miles of the location where activated graphite blocks are stored without prior NRC approval. PSC shall submit an analysis of any proposed new natural gas source demonstrating that the new source will not present an unacceptable hazard to the activated graphite blocks or to the equipment or systems needed to protect the activated graphite blocks.

Accidental release of activity caused by a postulated explosion of a container of flammable gas, such as those used to support decommissioning (e.g., propane or acetylene tank or bottle), was taken into consideration. Flammable liquids and gases will be administratively controlled during decommissioning and conversion to prevent use or storage of substantial quantities of flammable liquids or gas near areas containing highly activated wastes. However, even if it were postulated that an explosion did occur near radioactive waste containers, this event would not produce consequences exceeding those analyzed in this section for a heavy load drop, tornado or fire. This conclusion is based on the relatively small size of the missiles resulting from such a postulated explosion, and the relatively large amounts of activity postulated to be released in the above mentioned accidents.

5.1.5 <u>Heavy Load Drop Accident</u>

The dismantling of the PCRV will be accomplished with the aid of three types of hoist systems. These systems include the main Reactor Building bridge crane, the auxiliary 17-1/2 ton hoist on the bridge crane, and three 1-1/2 ton jib cranes on the refueling floor level. The Reactor Building crane will be re-reeved to allow the 170 ton main hook to travel from the refueling floor to ground level. An elevat view of the PCRV work area is shown in Figure 5.1-2. There will be many heavy loads removed during the dismantling process. These loads include:

1. Large side reflector blocks
2. Large concrete sections
3. Steam generators
4. Helium diffusers
5. Concrete Core Support Floor or CSF sections

The accident scenarios developed for heavy load drops in nuclear power plants consider the dropping of a heavy load (e.g., fuel shipping cask) on a very large radionuclide inventory such as fuel or spent fuel (Reference 8). In the case of Fort St. Vrain, all fuel will have been removed from the Reactor Building prior to commencement of dismantling operations. Therefore, the full spectrum of heavy load drop accidents is much less severe than in an operational nuclear power plant.

The most severe heavy load drop accident is postulated to consist of dropping the component containing the largest inventory of dispersible radioactive material. Table 5.1-5 has been compiled to show the various components and their respective radioactive inventories. Sampling will be performed prior to waste movement to determine and verify the radionuclide composition and total Curie content. Review of this table indicates that the large side reflector blocks contain the largest dispersible radioactive inventory. The use of an entire large side reflector for this accident analysis is conservative since the predicted activity inventory exceeds the LSA Curie limit specified in 10 CFR 71, Table A-1, for Type A-2 waste containers.

The drop of a heavy load onto a highly radioactive component was evaluated and determined not to represent the worst case scenario. For instance, the dropping of one of the 312 large side reflector blocks back into the PCRV might crush portions of adjacent reflector blocks. However, since all highly radioactive components are kept under water unless they are being removed, the debris and its attendant activity would remain in the water. This activity would be cleaned up in the PCRV Shield Water System. Any "slosh" created by the block drop would drain back to the PCRV cavity or drain down inside the Reactor Building, eventually to the Reactor Building sump and keyway, which have a capacity of approximately 350,000 gallons. These accident scenarios are bounded by the Loss of PCRV Shielding Water accident described in Section 5.1.7.

The 270-ton concrete CSF will be removed from the PCRV by raising the entire CSF to the top of the PCRV with specially installed high capacity jacks. Since the activated graphite blocks would have been removed from the PCRV prior to removal of the CSF, and since the CSF concrete is predicted to contain only 6 Curies of activity, a heavy load drop during this operation does not have the potential for release of significant quantities of radioactivity. If the entire CSF is raised by high capacity jacks, drop of the CSF is not considered credible since such an accident would require multiple jack failures.

5.1.5.1 Identification of Cause

A heavy load drop accident is a relatively low probability event. A failure of the hoisting cable could cause a drop of the load. In accordance with Reference 8, the probability of this event is on the order of 1.0E(-5) to 1.0E(-6) per demand (hoist lift). The loss of the crane brakes could be due to mechanical failure, operator error, or an incorrect maintenance operation. Since the Fort St. Vrain Reactor Building bridge crane does not qualify as a Single-Failure-Proof crane in accordance with NUREG-0554 (Reference 9) guidelines, the loss of crane brakes is postulated as a credible failure mode.

5.1.5.2 Accident Description

For this accident it is postulated that the Reactor Building bridge crane is hoisting one of the 312 large side reflector blocks. These blocks vary in size from 522 lbs. to 2030 lbs., and the largest (2030 lbs.) was assumed to be involved in this accident. It is currently planned to section these reflector blocks into smaller pieces for packaging in LSA shipping containers. Moreover, it is conservative to assume that a single reflector block may be transported intact in its own shipping container.

After appropriate radiation surveys and removal of surface contamination, the container with the single unsectioned side reflector block is lowered down the enlarged equipment hatch. Failure of the crane is postulated at this point. This results in the side reflector block container falling approximately 100 feet to the level of the truck loading bay. The shipping container ruptures, spilling its contents on the truck loading bay floor.

Administrative controls will be in place that will prevent the tractor of the tractor trailer from being in the loading bay during lowering of the container, and will ensure that all the truck loading bay doors are closed. It is conservatively assumed that one percent of the activity of the largest reflector block is dispersed from the drop. The dust is postulated to remain airborne and will escape the immediate area through the Reactor Building ventilation exhaust. Credit is taken for cleanup afforded by the Reactor Building ventilation system.

The Fort St. Vrain activation analysis (Reference 5) indicates that the major contributors to the activity in these large side reflector blocks are Fe-55, tritium, and Co-60. The total activity in an average large side reflector block has been calculated to be 1477 Curies, with the largest block containing 2250 Curies. A one percent release for this scenario results in 22.5 Curies becoming airborne in the Reactor Building. Of this amount, 22.24 Curies are Fe-55, 0.09-Curies are tritium and 0.17

Curies are Co-60. These activities are based on a three year decay period. Credit is taken for a 95 percent filter efficiency for Fe-55 and Co-60. Tritium is released unfiltered.

5.1.5.3 Analysis of Effects and Consequences

This scenario was modeled as an elevated stack release using the AIRDOSE-EPA computer code. The maximum individual dose is 1.1 E(-6) mrem red bone marrow dose and 2.7 E(-6) mrem to the lungs.

5.1.6 Fire

5.1.6.1 Identification of Cause

During decommissioning and repowering activities, there are many possible fire initiators that could result in a release of radioactive materials. These possible fire initiators include:

1. Fires started from cutting torches.
2. Fires associated with component processing activities on the refueling level.
3. Electrical fires.

The most likely initiator has been determined to be a cable tray fire started from a spark during PCRV random cutting operations. The fire would be quickly extinguished by the fire watch on duty for the cutting operations. The radiological consequence of this accident would be negligible since the cable trays contain virtually no radioactivity contamination.

The release of activity from a fire involving a contamination control tent was considered. However, potential consequences from this accident would be much less than those associated with a fire involving activated graphite blocks. The materials that will be used for these tents (typically "Herculite") will be fire retardant and not propagate flames, and the activity collected on the surface of a contamination control tent would not be expected to approach the activity inventory of activated graphite blocks, discussed below.

The postulated fire accident involves a fire enveloping LSA waste containers. The greatest exposure for a fire accident to occur is during the approximate six month period when the highly radioactive large side reflector blocks and side spacer blocks are being removed from the PCRV.

Controls will be implemented prior to the storage of the LSA containers. LSA containers will be limited to groupings with activity levels limited such that the dose consequences resulting from a fire involving a single group of LSA containers will consequences resulting from a fire involving a single group of LSA containers will not exceed those identified below, given the same assumptions. Sufficient spatial separation will be imposed to preclude fire propagation to an adjacent group of LSA containers. The packaging of these boxes and/or drums is planned to be completed inside the Reactor Building. Temporary storage or staging of these containers prior to shipment is also expected. It is assumed that interim radioactive material storage will be available for up to 15 LSA boxes and 200 drums in the former new Fuel Storage Building.

Fire detection capability will be installed in the LSA container storage area prior to the storage of the LSA containers. There will be no uncontrolled combustible materials in this building. The controls defined above will be implemented prior to the storage of the containers to limit the grouping of LSA packages containing combustible materials. These controls will ensure sufficient spatial separation is available to preclude fire propagation to an adjacent group of LSA containers and precludes the possibility of a fire with consequences greater than that which is analyzed.

5.1.6.2 Accident Description

For the fire accident it is postulated that a tractor trailer begins to transport packaged waste from the Reactor Building truck loading bay to an off-site burial ground/processing facility. The shipment is conservatively postulated to consist of 230 side spacer blocks with their boron pins removed. There are 1152 side spacer blocks to be removed during the decommissioning process.

Except for waste shipments in shielded casks, this 3706 Curie source term is the largest that will be contained in a transport truck destined for a burial ground.

It is postulated that an engine fire develops on the transport tractor and the fire spreads to the tractor's diesel fuel tanks. Based on work at the Waste Isolation Pilot Plant, the frequency of an unsuppressed truck fire is in the range of 1.0 E(-4) to 1.0 E(-5) per year (References 10 and 11). The tractor diesel fuel tanks may contain a combined capacity of up to 300 gallons of fuel. The fuel tanks are postulated to rupture from the heat and engulf the entire tractor trailer and the LSA containers in a diesel fuel pool fire. It is conservatively assumed that graphite side spacer blocks are enveloped by the diesel fuel fire.

A fire involving 300 gallons of diesel fuel spilled onto a relatively flat surface will burn out within thirty minutes. The resultant fire temperature will be bounded by the ASTM-E119 (Reference 12) standard fire curve. Most of the graphite will be exposed to temperatures well below the fire temperatures due to insulation provided by adjacent graphite blocks and some protection afforded by the shipping containers.

Under these conditions, it is conservative to assume that 50 percent of the graphite inventory on a shipping trailer is oxidized during the 30-minute fire. It is assumed that all of the tritium in the oxidized fraction (50 percent of the total tritium inventory), is released. In addition to tritium release, it is assumed that 0.015 percent of the balance of the radionuclide inventory is released in the form of particulates (Reference 13). The accident is assumed to occur at ground level immediately outside of the Reactor Building truck loading bay. The radioactive inventory for the 230 graphite side spacer blocks is calculated to be 3,706 total Curies. This total inventory consists of 3,556 Curies of Fe-55, 122 Curies of tritium and 28 Curies of Co-60. Fifty percent of the tritium is assumed to be released (approximately 61 Curies). The additional release of the remaining radionuclides will be 0.534 Curies of Fe-55 and 0.0042 Curies of Co-60.

5.1.6.3 Analysis of Effects and Consequences

This scenario was modeled as a surface area release 14.6 m. in diameter using the AIRDOSE-EPA computer code. The maximum individual dose is 2.1 E(-5) mrem red bone marrow dose and 3.1 E(-5) mrem to the lungs.

5.1.7 Loss of PCRV Shielding Water Accident

5.1.7.1 Identification of Causes

During a portion of the Fort St. Vrain decommissioning, the PCRV cavity will be flooded with water. This water will be circulated and purified by the PCRV Shield Water System to gradually decrease the radioactivity in the water. This system is expected to be in operation during the period when the PCRV internals are being removed.

This accident scenario assumes that there is a leak or rupture of the PCRV Shield Water System piping resulting in a liquid release due to a mechanical impact or a mechanical failure of a weld or flange.

5.1.7.2 Accident Description

This accident scenario assumes that a mechanical failure of the PCRV Shield Water System piping to the PCRV cavity occurs, resulting in a pipe rupture. Other leaks/breaks can be postulated (i.e., seal failures). However, the results are the same. Tritiated water with dissolved cesium, iron and cobalt would be spilled into the Reactor Building sump and keyway. Assuming the worst case (complete emptying of the PCRV), calculations indicate that 423,500 gallons could fill the Reactor Building sump/keyway, and flood the basement floor to a height of two feet. This water would be 49 feet below grade and would be contained by the Reactor Building sump/keyway and walls. No credit is taken for the Reactor Building ventilation system for this accident scenario.

In order to conservatively maximize the consequences of this accident, an extremely large amount of water was assumed for the PCRV inventory, reflecting the PCRV full of water and empty of most of its contents, including the core. This assumption results in overflow of the Reactor Building sump/keyway to the Reactor Building basement, thereby increasing the surface area of water available for evaporation by approximately a factor of five.

Since the non-gaseous activities will be retained in the spilled water, tritium (released through evaporation) is the only significant activity available. This will be evaporated from the surface area of the spilled water in the Reactor Building basement.

The PCRV liquid release will not seep through the sump concrete seams as the water table is well above the 49 foot below grade level.

The Reactor Building is approximately 120 feet long and 76 feet wide which conservatively provides (neglecting equipment) a surface area for the spilled water of 9120 square feet (848 square meters). From Westinghouse Report WCAP-11002 (Reference 14), the best fit evaporation rate at 70 percent relative humidity and an air circulation speed of 1 m/sec is 0.046 g/m^2-sec or 0.046 cc/m^2-sec (assuming 1 gram = 1 cc of water). It is predicted that tritium levels in the PCRV water will be less than approximately 535 Curies. However, for this analysis, it is conservatively assumed that the theoretical maximum amount of tritium is transferred to the PCRV shielding water from the graphite blocks, which is approximately 1 E(5) Curies. Therefore, the tritium concentration in the spilled water is calculated to be 62.4 μCi/cc.

With an evaporation rate of 0.046 cc/m^2-sec and a tritium concentration of 62.4 μCi/cc, the tritium release rate is about 2.5 mCi/sec over the 848 square meters of surface area. Over a two hour period, 18 Curies would be released to the atmosphere.

5.1.7.3 Analysis of Effects and Consequences

This scenario was modeled as a surface area release 29.1 m. in diameter using the AIRDOSE-EPA computer code. The maximum individual dose is 4.6 E(-6) mrem red bone marrow dose and 4.6 E(-6) mrem to the lungs.

5.1.8 Loss of Power

During the plant decommissioning, power will be normally supplied by off-site sources. No backup power is assumed available during a loss of power. The primary machinery using power during the decommissioning will be:

Pumps:	Demolition Tools:
Deionized Water System	Plasma Arc Torch
Fire Water Pumps	Diamond-Wire Cutter
Service Water Pumps	Water Jet Cutter
Water Treatment	Drills
PCRV Shield Water Pump	Mobile Laundry
Cranes	PCRV Work Platform
Lighting:	HVAC:
Underwater Lighting	Ventilation Fans
Building	HEPA Filters/Fans
Plant Area	HEPA Vacuums/ Portable Cleaners

5.1.8.1 Identification of Causes

This accident postulates the loss of off-site power due to weather related events. Such events could include downed power lines due to strong winds or heavy icing conditions. The likelihood of this occurrence is remote since off-site power can be supplied to the site through six separate lines.

5.1.8.2 Accident Description

Loss of power would result in the loss of plant ventilation (HVAC) systems, lighting, plant water systems, and demolition power. Decommissioning activities would cease until power is restored.

Loss of power to the PCRV Shield Water System pumps will not result in a radioactive release since the flow of bleed water to the evaporation ponds will be stopped. While loss of ventilation will force personnel from radiological control areas, no off-site consequences are anticipated.

The postulated accident scenario is the loss of power of the HVAC while a large side reflector block has been removed from the PCRV for cutting. These graphite blocks will be grappled and hoisted by a jib crane to the HSF or a refueling floor work station where the blocks will be cut into sections in preparation for packaging into LSA containers. The loss of power is assumed to occur after the cutting/cleaving operation.

It is assumed that these processing operations (kerfing debris) are performed in a local containment on the refueling floor, and will release 1.5 percent of the total activity of a single large side reflector block. It is conservatively judged that the combination of radiological controls in place at the work station (e.g., confinement through tenting) and the confinement function provided by the Reactor Building itself will result in retention of 99 percent of the Fe-55 and Co-60 kerfing debris in the Reactor Building. It is therefore assumed that one percent of the Fe-55 and Co-60 in the kerfing debris and 100 percent of the tritium in the kerfing debris are released at ground level from the Reactor Building. No credit is taken for the Reactor Building ventilation system.

The total activity in an average large side reflector blocks has been calculated to be 1477 Curies, as shown in Table 5.1-5. A release of 1.5 percent of the radioactive material is assumed from the kerfing debris in a single block. Of that amount, one percent of the Fe-55 and Co-60, and 100 percent of the tritium is released, resulting in a total of 0.31 Curies released to the environment. This total release consists of 0.219 Curies of Fe-55, 0.091 Curies of tritium and 0.0017 Curies of Co-60. These activities are based on a three year decay period.

5.1.8.3 Analysis of Effects and Consequences

This scenario was modeled as a surface area release with an effluent velocity of zero using the AIRDOSE-EPA computer code. The maximum individual dose is 2.4 E(-7) mrem red bone marrow dose and 5.8 E(-7) mrem to the lungs.

5.1.9 Natural Disasters

For the effects of natural disasters, the following external initiating events were considered:

1. Earthquake

   The Reactor Building is designed to withstand the Design Basis Earthquake of 0.10 g horizontal ground acceleration at the site without unsafe damage or failure to function. For decommissioning, it is required that the Reactor Building continue to perform its confinement function following a seismic event. The seismic qualification of the Reactor Building will be maintained during decommissioning. No other new or existing systems or equipment are required to function during or following an earthquake.

   The most severe event which could result from a large earthquake is considered to be a drop of a radioactive waste container holding a highly activated graphite block (see heavy load drop accident). However, the simultaneous occurrence of an earthquake and the hoisting of a heavy load is not considered credible. The consequences of this simultaneous earthquake and heavy load drop scenario were not analyzed due to the low probability of such an event.

2. Tornado and Wind Effects

   From Reference 15, Section 14.1.2, the basic design wind velocity for the plant is 90 mph. The equipment and structures exposed to wind load are designed to support design wind load combined with functional loads within the specified allowable stresses.

The tornado danger at the plant site is extremely remote. However, the Reactor Building was designed to withstand wind loadings developed by a tornado of 202 mph (total horizontal wind velocity) without exceeding yield stresses in the basic building structure. The Reactor Building was also designed to withstand a maximum tornado of 300 mph (total horizontal wind velocity) acting on the full area of all structures and a drop in atmospheric pressure of 3 psi within a period of 3 seconds, without exceeding ultimate stress levels in the main structural members. At the 300 mph wind speed, the siding on the Turbine and Reactor Buildings above the turbine deck and refueling floor levels may be carried away, but the basic building structure will not collapse.

3. Floods

From Reference 15, Section 14.1.3, the plant site is protected from excessive runoff and flood by design of the yard drainage system. Grade level is approximately 17 feet above the highest observed flood level, and from 10 to 13 feet above the maximum probable flood level. The walls of the structures extending below grade level are watertight, and buoyancy effects were taken into account in their construction. Therefore, there will be no further consideration of accidents due to flooding during decommissioning activities.

4. Range Fire

The Fort St. Vrain site is located in an area of Weld county devoted to agriculture. The site itself is surrounded by pasture land and irrigated fields. Within the plant exclusion area is a fire buffer area consisting of maintained grass and ornamental landscaping. A 20 foot wide concrete pad rings the site. Therefore, a brush or range fire is not a credible accident during decommissioning activities.

5.1.9.1 Identification of Causes

The risks from a tornado at Fort St. Vrain during decommissioning are quite low for two reasons. First, the probability that a tornado will strike the site is diminishingly small. Second, the plant specific vulnerability to a tornado and its consequences are also small. Unlike an operating nuclear power plant with active safety systems to contain large quantities of radioactive materials at high energy levels, all spent fuel will be removed from Fort St. Vrain and the PCRV will essentially be a passive container of radioactive material. Possible loss of power, which could be caused by a tornado, is specifically analyzed in Section 5.1.8.

The Reactor Building roof and siding above the refueling floor are designed to withstand a tornado with a wind speed up to 202 mph. The probability of experiencing a tornado with wind speeds above 202 mph during decommissioning is extremely low based upon information and methodology provided in the draft Individual Plant Examination of External Events (IPEEE), NUREG-1407 (Reference 16).

Based on the work of Abbey and Fujita (Reference 17), the continental United States was broken down into 20 distinct tornado hazard regions. These regions were generalized into 4 broad areas shown in Figure 5.1-3, ranging from a highest risk in region A to the lowest risk in Region D. The Fort St. Vrain site if classified into Region C.

Reference 18 is used to establish the occurrence rate for different classifications of tornadoes. The National Severe Storms Forecast Center (NSSFC) national database for the years 1950 - 1978 was used as the basis for the occurrence rate analysis. The NSSFC data are categorized by Fujita intensity scales (F-scales). To predict the probability that a tornado with maximum windspeed will strike a nuclear power plant requires adjusting the F-scales for: tornado reporting trends, F-scale classification errors, path length intensity variation, and occurrence rates and windspeed relationships adjusted for intensity variation. The adjusted, or updated, tornado scales are denoted by "F". Tornado wind velocities for the F- and F'- scales are compared as follows:

F-Scale	Maximum Windspeed Interval (mph)	F'-Scale	Maximum Windspeed Interval (mph)
F0	40 - 72	F'0	40 - 73
F1	73 - 112	F'1	73 - 103
F2	113 - 157	F'2	103 - 135
F3	158 - 206	F'3	135 - 168
F4	207 - 260	F'4	168 - 209
F5	261 - 318	F'5	209 - 277

The following evaluation demonstrates the low probability of occurrence of a tornado with wind velocity exceeding 202 mph at Fort St. Vrain, by comparing the frequency of occurrence of tornadoes in Weld County with the NSSFC data. The occurrence rate of a F4 tornado is 3.4 E(-6)/square mile/yr (Reference 18). According to the National Weather Bureau's historical data for Weld County from 1950 through 1987, there was only one tornado in the F3 range. That single F3 tornado is the only tornado in the vicinity of Fort St. Vrain of the 256 tornadoes recorded by NSSFC for all of Region C that had estimated windspeeds greater than 158 mph. Based on this sample from the population, it can be inferred that the probability of a tornado at Fort St. Vrain in the F3 range is much less than 3.4 E(-6)/square mile/yr.

The occurrence rate for a F5 tornado in Region C is 3.5 E(-7)/square mile/yr (Reference 18). The National Weather Bureau's Weld County data show no tornado occurrence with intensity of F4 or greater. Thus, the 56 F4 and nine F5 tornadoes recorded by NSSFC all occurred outside the Fort St. Vrain area.

From this data, it can be concluded that the probability of occurrence of an F4 or greater tornado is less than 3.5 E(-7)/square mile/yr. According to the draft IPEEE (Reference 16) "Plants Designed Against NRC Current Criteria", these events pose no significant threat of a severe accident because the current design criteria for wind are dominated by tornadoes having a frequency of exceedance of about 1 E(-7). The following section contains a specific accident analysis for a postulated tornado with winds less than 202 mph.

5.1.9.2 Accident Description

Temporary storage or staging of radioactive waste containers prior to shipment is expected. It is assumed that interim radioactive material storage will be available for 15 LSA boxes and 200 drums in the Fort St. Vrain Fuel Storage Building. Calculations demonstrate that neither forces generated by 202 mph wind loading, nor the impact from the tornado-driven design basis missile, will result in breach of the walls or roof of this building.

In this scenario, it is assumed that a 202 mph tornado strikes the Fort St. Vrain site. At this lower wind level, the walls of the Reactor Building enclosing the PCRV will remain intact.

The tornado-driven design basis missile is a 12 foot x 12 inch x 4 inch thick fir plank, weighing 105 pounds, which impacts and penetrates the Reactor Building above the refueling floor level. It is assumed that this missile strikes and ruptures a container with 46 graphite side spacer blocks. It is conservatively assumed that one percent of the activity in the container is dispersed and released to the environment. No filtration credit is assumed.

The total radioactivity inventory for the 46 side spacer blocks is approximately 741 Curies. This total inventory is comprised of 711 Curies of Fe-55, 24.4 Curies of tritium and about 5.5 Curies of Co-60. Assuming a one percent release results in 7.41 Curies released to the environment. These activities are based on a three year decay period. The major exposure path was assumed to be air inhalation to an adult standing at the Exclusion Area Boundary (EAB).

5.1.9.3 Analysis of Effects and Consequences

This scenario was modeled as a surface area release of 0.18 m in diameter using the AIRDOSE-EPA computer code. The maximum individual dose is 7.2 E(-5) mrem red bone marrow dose and 2.0 E(-4) mrem to the lungs.

5.1.10 Dropping of a Steam Generator Primary Module

5.1.10.1 Identification of Cause

As stated in Section 5.1.5.1, a heavy load drop is a relatively low probability event. A failure of the hoisting cable could cause a drop of the load. In accordance with Reference 8, the probability of this event is on the order of 1.0 E(-6) per demand (hoist lift). The loss of the crane brakes could be due to mechanical failure, operator error, or an incorrect maintenance operation. Since the Fort St. Vrain Reactor Building crane does not qualify as a Single-Failure-Proof crane in accordance with NUREG-0554 (Reference 9) guidelines, the loss of crane brakes is postulated as a credible failure mode. The dropping of the component with the largest inventory of dispersible radioactive material, the large side reflector blocks, was postulated and analyzed in Section 5.1.5. Credit is taken for decontamination of the particulate afforded by the Reactor Building Ventilation Exhaust System. However, no credit has been taken for the Reactor Building ventilation exhaust system in a postulated steam generator primary module drop accident. Therefore, this accident scenario assumes that radioactivity is released from the drop of a steam generator primary module due to the loss of the crane brakes. While the other postulated decommissioning accidents involved the release of activation products, the release from the drop of a primary steam generator module is expected to be plateout.

5.1.10.2 Accident Description

It is postulated that the Reactor Building crane is hoisting one of the steam generator primary modules. After appropriate radiation surveys, the steam generator primary module is lowered into the enlarged equipment hatch. Failure of the crane is postulated at this point. This failure results in the steam generator primary module falling approximately 100 feet to the level of the truck loading bay, and possibly into the Reactor Building basement, should it break through the truck loading bay floor slab. It is conservatively assumed that one percent of the activity plated out on the surface of a single steam generator primary module is dispersed from the drop. The activity is postulated to remain airborne and escape the Reactor Building unfiltered.

In order to conservatively estimate the activity levels of radionuclides that are present on the surfaces of the steam generators, the plateout levels predicted by the revised FSV Plateout Analysis for Decommissioning Study (Reference 19) were used. The projected plateout levels on the reheater, superheater, economizer and evaporator tube bundles of all 12 steam generator modules are presented in Table 2.2-4.

Table 2.2-4 identifies projected plateout levels of six key radionuclides that are either fission products or come from fission products (Sr-90, Te-127m, I-129, I-131, Cs-134 and Cs-137). In addition to the six key radionuclides associated with fission products discussed above, it is also necessary to consider activation products which may have accumulated on the steam generator surfaces, such as Mn-54, Fe-55 Co-60, Ni-63, etc. While it is not presently feasible to obtain samples from the surfaces of the steam generator tube bundles, surface concentrations of activation products can be estimated based on samples taken from the surfaces of helium circulators, which are in the primary coolant flow stream.

Although measured surface concentrations of Co-60 were a factor of 15 lower than those measured for Cs-137 on a helium circulator, it was conservatively assumed that the Co-60 inventory on a steam generator module was equal to the Cs-137 inventory that is predicted by the revised FSV Plateout Analysis for Decommissioning Study to be on a steam generator module. The ratio of the surface concentrations of remaining activation products to Co-60 was assumed to be identical to those previously measured on the surfaces of a helium circulator during radwaste characterization (Reference 20). Based on these assumptions, the inventory of various radionuclides plated out on the surfaces of a steam generator module were derived and are shown in Table 5.1-6.

It was then conservatively assumed that none of the activity on the surfaces of the steam generator modules is removed while the steam generators are underwater. It was also assumed that 1% of the total surface activity inventory of one steam generator module becomes airborne as a result of the postulated drop accident. Furthermore, a three year decay period was assumed, although the steam generators are not scheduled for removal until about two years into decommissioning, or about 5 years after reactor shutdown. The radionuclides assumed to be released as the result of a postulated drop accident are thus 1% of the values presented in the 3 year decay column of Table 5.1-6. No credit is taken for particulate filtration by the Reactor Building ventilation system.

5.1.10.3 Analysis of Effects and Consequences

This scenario was modeled as an elevated stack release using the AIRDOSE-EPA computer code. The maximum individual organ dose is 2.3 E(-6) mrem to the lungs and the red bone marrow dose is 1.1 E(-6) mrem.

5.1.11 Summary

The results of the preceding accident scenarios, postulated for Fort St. Vrain decommissioning activities, indicate that the radiation exposures to the general public will be very low. These evaluations have determined that, in all cases, the radiological consequences are well within the 10 CFR 100 guidelines of 25 rem whole body dose and 300 rem to any specific organ of a maximally exposed individual. These doses are also a small fraction of the 1 rem whole body dose and 5 rem to any specific organ dose guidelines cited in the EPA Protective Action Guidelines (Reference 4).

These scenarios are considered to have a low probability of occurrence and their radiological consequences bound other less severe accident scenarios. Therefore, it is concluded that the Fort St. Vrain decommissioning activities do not pose any undue risk to the health and safety of the general public.

5.2 Transportation Accidents Involving Radioactivity

All shipments of waste from Fort St. Vrain are expected to be transported by truck except the upper portions of the steam generators.

The potential exists for truck accidents which could lead to radiation exposure to transportation personnel and the general public. Truck accidents also could result in non radiological injuries. Typical truck transportation of waste generated by Fort St. Vrain decommissioning activities is not expected to result in radiation doses above background exposure to transportation personnel or to the general public.

A worst case on-site truck accident was analyzed in Section 3.4.6 of the Proposed Decommissioning Plan. (This was for a tractor trailer loaded with 230 graphite side spacer blocks.) The results of this conservative analysis for a maximally exposed offsite individual (at the EAB) was a whole body dose (0-2 hr.) of 121 mrem and a lung dose of 215 mrem.

The population dose from this accident scenario have also been calculated using the AIRDOSE-EPA code. The results are documented in Section 5.1.6.3.

The on-site truck fire can also be postulated to occur off-site. Application of a conservative air dispersion factor of 3.0 E(-2) s/m^3 (Reference 21) results in a projected dose to an onlooker at 100 meters of 103 mrem to the whole body and 183 mrem to the lung. Potential doses to a transportation worker or a firefighter are estimated to be 3 times higher.

These doses are based on the very conservative assumption that the diesel truck fire is not extinguished but burns itself out after consuming all the truck's diesel fuel. It is unlikely that such a fire would not be contained and extinguished within 30 minutes.

While these consequences are not trivial, they are low and very unlikely. The number of fire accidents involving transportation of the most highly radioactive material (graphite side spacer blocks) from Fort St. Vrain is estimated to be 4.2 E(-5). This value is based on accident probabilities (Reference 22) for 5 shipments with a one-way distance of 913 miles.

It is estimated that to dispose of <u>all</u> radioactive waste from Fort St. Vrain decommissioning will require 160 one way and 225 round trip truck trips. The one way distance to the disposal site in Beatty, Nevada, is 913 miles, although other disposal sites may be used. The number of traffic accidents was estimated to be seven with two nonradiological injuries and no fatalities. These values were also based on Reference 22.

It is currently planned to ship the upper primary (contaminated) portions of the twelve steam generators by rail to Richland, Washington, for burial. A shipping container will consist of a metal culvert section seven feet in diameter by 27 feet long. The culvert section will be cut in half lengthwise to provide a hollow half-cylinder. Structural supports will be welded to the half section of culvert.

The region between the outside of the steam generator shroud and the inside of the culvert can be filled with grout for shielding. The combined weight of the shipping container, steam generator assembly and grout could be as much as 195,000 pounds.

It is possible that a railroad accident could result in the steam generator shipping container falling into a river enroute to Richland, Washington. However, this accident has not been analyzed because the risk from this accident scenario is very low. First, the probability of a railroad accident occurring near a river or large stream is very low. The estimated probability of immersion due to accidents occurring on railroad bridges over deep rivers is 2 E(-11) per car mile (Reference 23). Additionally, the consequences from this accident would be minuscule, i.e., less than 0.001 mrem whole body. The low consequence is due to the fact that the steam generators are shipped in strong containers and loose crud and surface activation associated with the steam generators will have already been removed during the period that the steam generators are purposefully submerged in the PCRV water.

Results of transportation accident scenarios postulated for Fort St. Vrain decommissioning activities demonstrate that radiation exposure to the general public will be very low. Thus, activities related to the Fort St. Vrain decommissioning project will not pose any undue risk to the health and safety of the public.

5.3 Other Impacts

We have evaluated the impacts of radiological accidents on the health and safety of the public in the two previous sections. There are no large supplies of hazardous chemicals for decom[illegible]ng that could result in a catastrophic release. Approximately 20,00[illegible] cubic yards of non-radioactive concrete and metallic scrap will be disposed of in local land fills. We conclude that there are no other significant activities resulting from decommissioning operations that could result in a substantial impact on the environment.

5.4 References

1. Proposed Decommissioning Plan for the Fort St. Vrain Nuclear Generating Station, Public Service Company of Colorado.

2. AIRDOSE-EPA: A Computerized Methodology for Estimating Environmental Concentrations and Dose to Man from Airborne Releases of Radionuclides, U.S. Environmental Protection Agency, December, 1979.

3. Regulatory Guide 1.109 "Calculation of Doses to Man From Routine Releases of Reactor Effluents", Revision 1, October 1977.

4. "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA-520/1-75-001-A, U.S. Environmental Protection Agency, January 1990.

5. Engineering Evaluation, "Fort St. Vrain Activation Analysis," EE-DEC 0010, Rev. D, Public Service of Colorado, March 1992.

6. General Atomics Internal Correspondence, M.B. Richards to F.C. Dahms, "Tritium Source Terms in Fort St. Vrain Permanent Side Reflector and Spacer Block Graphite," dated October 10, 1990; attachment to General Atomics Letter GP-3487, dated October 11, 1990.

7. NRC Letter, Erickson to Crawford, dated May 21, 1991 (G-91106); Subject: "Fort St. Vrain Nuclear Generating Station Amendment No. 82 to Facility Operating License No. DPR-34."

8. "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612, USNRC, July 1980.

9. "Single-Failure-Proof Cranes for Nuclear Power Plants," NUREG-0554, USNRC, May 1979.

10. "Waste Isolation Pilot Plant Integrated Risk Assessment," DOE 89-10, June 1990.

11. "Special Analysis: Heavy Truck Fires, 1982-1986," National Fire Protection Association Fire Analysis and Research Division.

12. "Standard Test Methods for Fire Tests of Building Construction and Materials," ASTM E119-88.

13. Mishma, J. and Schwendiman; "Fractional Airborne Release of Uranium

During the Burning of Contaminated Waste," BNWL-1730, April 1973.

14. Westinghouse WCAP-11002, "Evaluation of Steam Generator Overfill Due to a Steam Generator Tube Rupture Accident," February 1986 (proprietary).

15. Fort St. Vrain Updated Final Safety Analysis Report, Revision 9, Public Service Company of Colorado.

16. Draft Individual Plant Examination of External Events (IPEEE), NUREG-1407, USNRC, 1990.

17. Abbey, R.F., and Fujita, T.T.; "Regionalization of the Tornado Hazard, Tenth Conference on Severe Local Storms," American Meteorological Society, October 1977.

18. "Tornado Missile Simulation and Design Methodology, Volume 2: Model Verification and Data Base Updates," EPRI NP-2005, Volume 2, Project 616-2, Final Report, August 1981.

19. GA International Services Corporation, Report 909658, "Fort St. Vrain Plateout Analysis for Decommissioning Study," Issue B, February 28, 1992 (GP-3538).

20. FSV 10 CFR 61 Compliance Program Update, dated April 1991.

21. NUREG/CR-0672, Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, Pacific Northwest Laboratory, 1984.

22. WASH 1238, Environmental Safety of Transportation of Radioactive Materials to and from Nuclear Power Plants, U.S. AEC, Directorate of Regulatory Standards, 1972.

23. Dennis A W. at al. "Severities of Transportation Accidents Involving Large Packages", Sandia Laboratories SAND77-000, May 1978.

TABLE 5.1-1
SUMMARY OF POSTULATED ACCIDENT SCENARIOS

Postulated Accident	Description
Dropping of Contaminated Concrete Rubble	Rubble from PCRV top head concrete is dropped during processing.
Conversion Construction Near PCRV Dismantlement	Natural gas explosion/ crane falling.
Heavy Load Drop	Container drop to loading bay.
Fire	Truck diesel fuel pool fire.
Loss of PCRV Shielding Water	Pipe rupture in the PCRV Shield Water System.
Loss of Power	Release of graphite cutting debris from refueling floor work station.
Natural Disasters	Tornado-generated missile striking LSA waste container.
Steam Generator Module Drop	Primary steam generator module drop to loading bay.

TABLE 5.1-2

DOSES DUE TO POSTULATED DECOMMISSIONING ACCIDENTS
1 Year Committed Effective Dose Equivalent (CEDE)

Maximum Postulated Accident	Maximum Individual Red Bone Marrow Dose (mrem)	Maximum Individual Organ Dose (mrem)
Dropping of Concrete Rubble	7.1 E(-7)	1.8 E(-6) (endosteal)
Heavy Load Drop	1.1E(-6)	2.7 E(-6) (lung)
Fire	2.1 E(-5)	3.1 E(-5) (lung)
Loss of PCRV Shielding Water	4.6 E(-6)	4.6 E(-6) (lung)
Loss of Power	2.4 E(-7)	5.8 F(-7) (lung)
Natural Disaster (Tornado)	7.2 E(-5)	2.0 E(-4) (lung)
Steam Generator Module Drop	1.1 E(-6)	2.3 E(-6) (lung)

1. All maxumum doses occur in the first sector 4828 m WSW of the plant. Refer to Figure 5.1-1 for location.

2. The most recent release of AIRDOSE-EPA does not calculate a whole body dose. Red bone marrow doses are presented in its place.

TABLE 5.1-3
CURIE TOTALS IN ACTIVATED PCRV CONCRETE
(3 YEARS DECAY)

Location	Depth		Curies
Top Head Axial Up	1st	6 inches	9.83 E(1)
	2nd	6 inches	2.56 E(1)
	3rd	6 inches	2.52 E(0)
	4th	6 inches	2.70 E(-1)
	5th	6 inches	3.68 E(-2)
	6th	6 inches	6.35 E(-3)
	7th	6 inches	1.31 E(-3)
	8th	6 inches	2.85 E(-4)
Radial	1st	6 inches	8.89 E(0)
	2nd	6 inches	3.13 E(0)
	3rd	6 inches	3.66 E(-1)
	4th	6 inches	4.10 E(-2)
	5th	6 inches	5.94 E(-3)
	6th	6 inches	1.08 E(-3)
	7th	6 inches	2.31 E(-4)
	8th	6 inches	5.22 E(-5)
Core Support Floor Axial Down	1st	6 inches	5.69 E(0)
	2nd	6 inches	3.80 E(-1)
	3rd	6 inches	3.33 E(-2)
	4th	6 inches	3.60 E(-3)
	5th	6 inches	4.66 E(-4)
	6th	6 inches	7.67 E(-5)
	7th	6 inches	1.42 E(-5)
	8th	6 inches	3.08 E(-6)
	9th	6 inches	6.69 E(-7)
	10th	6 inches	1.25 E(-7)

TABLE 5.1-4
PERCENTAGE CONTRIBUTION OF ACTIVATION PRODUCTS IN FIRST 6 INCHES OF TOP HEAD CONCRETE (3 YEARS DECAY)

Significant Nuclides	Percent of Total (1)
H-3	2.89
Ca-41	0.05
Ca-45	0.18
Fe-55	89.29
Co-60	3.43
Cs-134	0.24
Eu-152	3.51
Eu-154	0.36
	99.95

Note 1: 98.3 curies total in 1.44 E7 cc of top head concrete.

TABLE 5.1-5
WASTE VOLUME/ACTIVITIES
ESTIMATES FOR THE PCRV
(3-YEARS DECAY)

Item/System	Number	Total Curies	Curies/ Item
Region constraint device & pins	84	122	1.4
Metal clad reflector block - CR	37	23100	624
Metal clad reflector block - NCR	270	173000	640
Defueling blocks	1482	<0.01	
Top reflector graphite blocks	589	2700	4.58
Bottom reflector graphite blocks	902	4000	4.43
Radial reflector hex graphite blocks - removable & permanent	396	3300	8.3
Large reflector blocks	312	354500*	1477*
Half-size reflector blocks	96	83700	872
Upper reflector keys (carbon steel)	24	0.02	<0.01
Side spacer blocks (no rods)	1152	18550*	16.1
Boron rods	309792	36800	0.12
Lower reflector keys (Hastelloy)	24	470	19.6
Core support blocks	61	120	2.0
Core support posts	183	36.5	0.2
Core support floor columns	12	1	0.08
Misc. steel from beneath CSF		2	
Metal on large side reflectors	24	0.02	<0.01
Core barrel	1	8.4	8.4
Lower plenum insulation		<0.01	
Silica blocks (25,000 lbs.)		250	
Concrete - top		130	
Concrete - CSF		6	

* These values are different than the values computed in the Activation Analysis, as explained in Section 5.1.1.

TABLE 5.1-5 (Continued)

Item/System	Number	Total Curies	Curies/ Item
Concrete - side		12	
Misc. Inconel parts on CSF		15	
Hastelloy Cans	20061	3800	0.19
Concrete cutting debris - top		15	
Concrete cutting debris - CSF		0.45	
Concrete cutting debris - side		0.44	
Helium purifiers in PCRV head	10	0.9	0.09
Helium diffusers	4	20	5
Helium circ. shutoff valve assy	4	2	0.5
Helium bellows	12	20	1.66
Thermocouples & guide tubes		0.8	
Steam generators	12	52.1	4.34
Lower floor appurtenances		2	
Platform/handling tools/jib cranes		<0.01	
Crane cable/drum/3 buck[illegible] inverters		<0.01	
Helium circulators	5	0.13	0.03
Orifice valves	37	415	11.2
Control rod drive assembly	44	233	5.3
Control rod absorber assembly	88	2.8	0.03
CSF Kaowool & Cover Plates		90	90
CSF Liner		142	142
Radial PCRV Liner		10	10
Top Cover Plates		5.7	5.7
Top Kaowool		<0.01	<0.01
Top Head Liner		105	105

TABLE 5.1-6
INVENTORY OF RADIONUCLIDES ON ONE STEAM GENERATOR MODULE (1)

ISOTOPE	INITIAL ACTIVITY (μCi)	HALF-LIFE (years)	3 YEAR DECAY ACTIVITY (μCi)
Mn-54	1.74 E+05	8.55 E-01	1.53 E+04
Fe-55	5.20 E+06	2.70 E+00	2.41 E+06
Co-60	5.71 E+05	5.26 E+00	8.48 E+05
Ni-63	9.71 E+04	9.20 E+01	9.10 E+04
Sr-90	3.94 E+04	2.81 E+01	3.66 E+04
Y-90*	1.68 E+05	64 Hours	1.41 E+05
Ru-106	1.47 E+04	1.01 E+00	1.88 E+03
Ag-110m	1.83 E+05	6.90 E-01	9.00 E+03
Cs-134	2.34 E+05	2.06 E+00	8.53 E+04
Cs-137	5.71 E+05	3.01 E+01	5.33 E+05
Ce-144	2.18 E+05	7.78 E-01	1.50 E+04
Pm-147	1.83 E+05	2.62 E+00	8.42 E+04

* - Y-90 continues to be produced from the radioactive decay of Sr-90

(1) - Nuclides that contribute at least 0.1% to the offsite radiological consequences (Whole body or maximum organ)

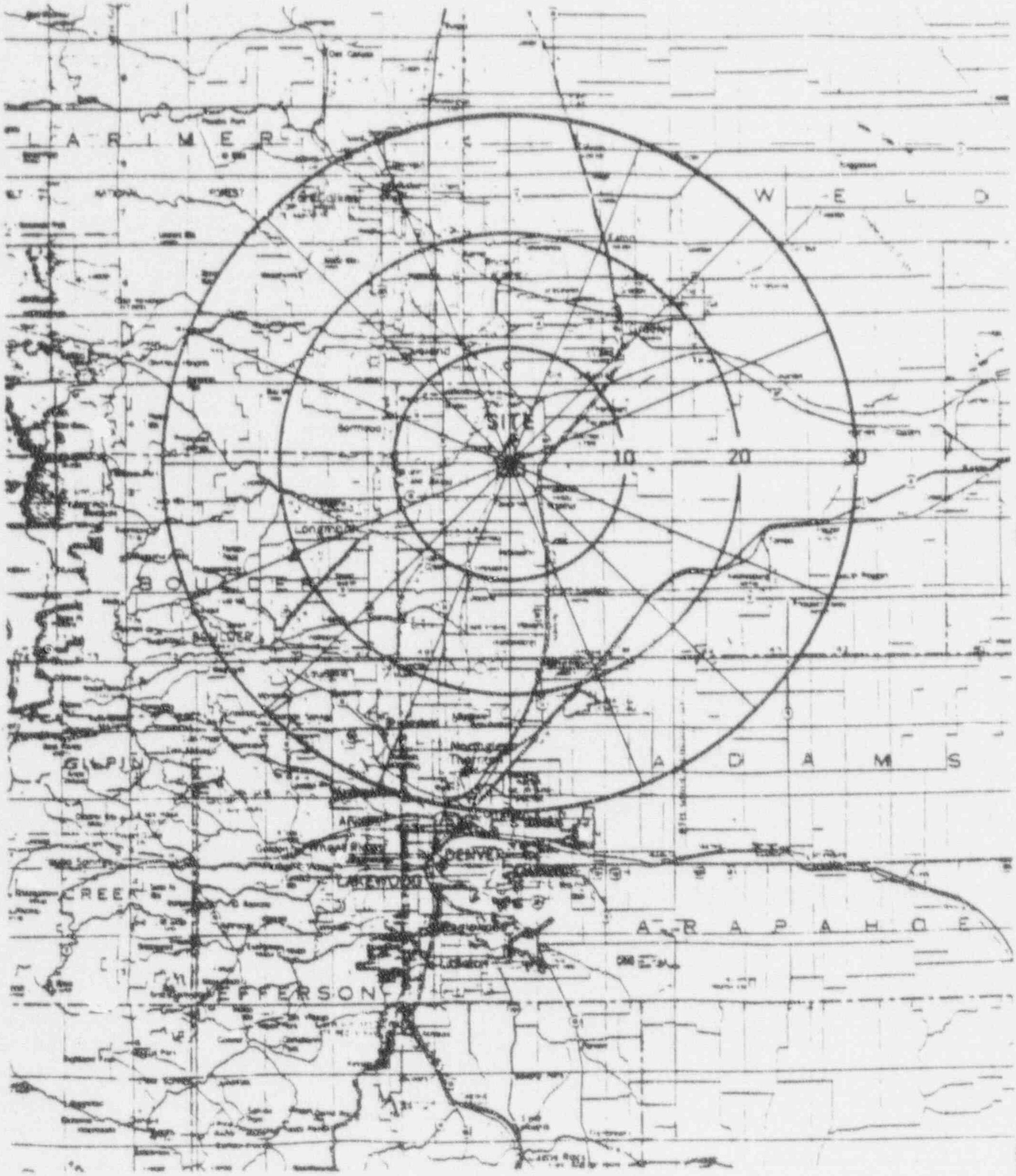


Figure 5.1-1
Map Centered on Fort St. Vrain Site

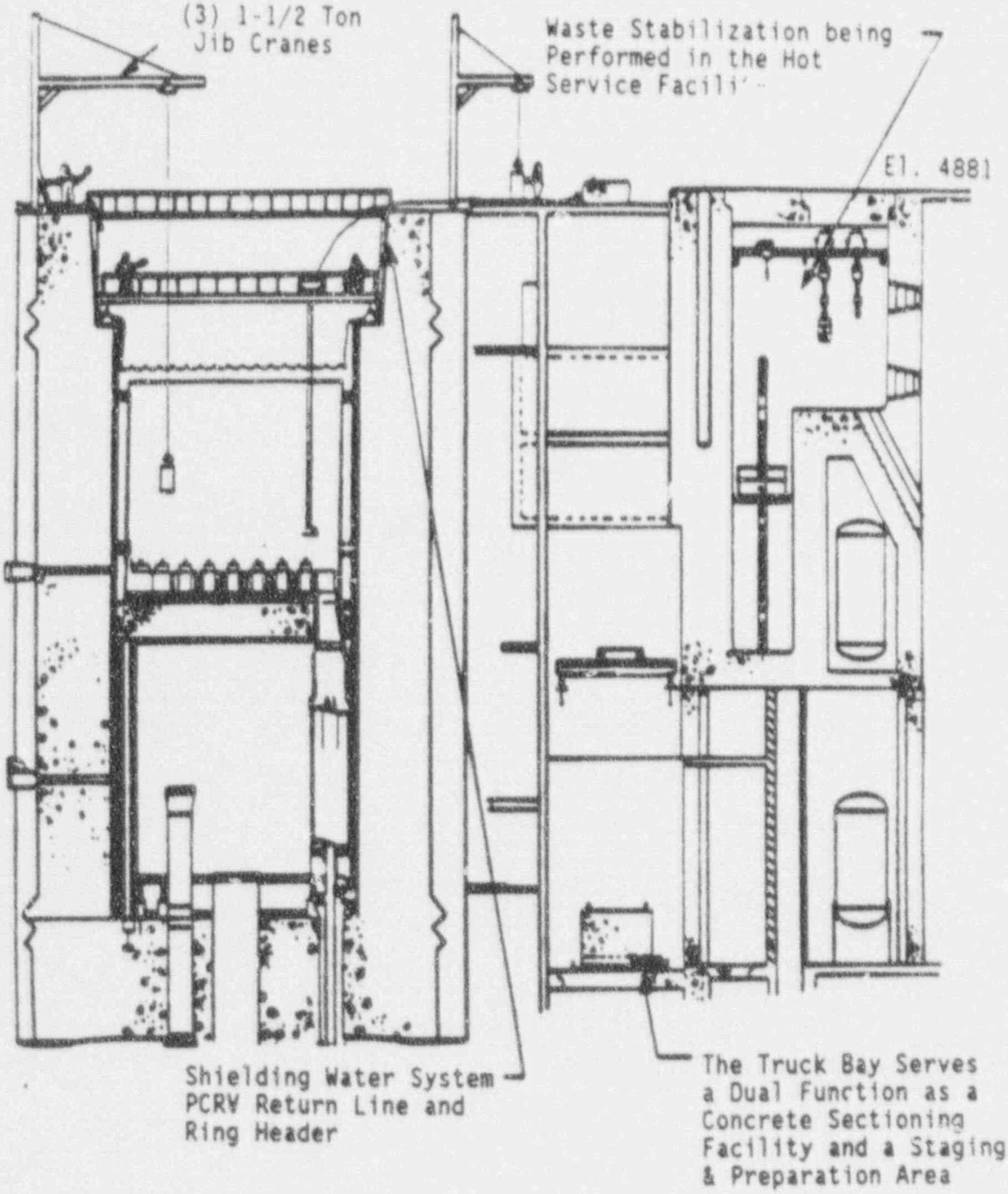
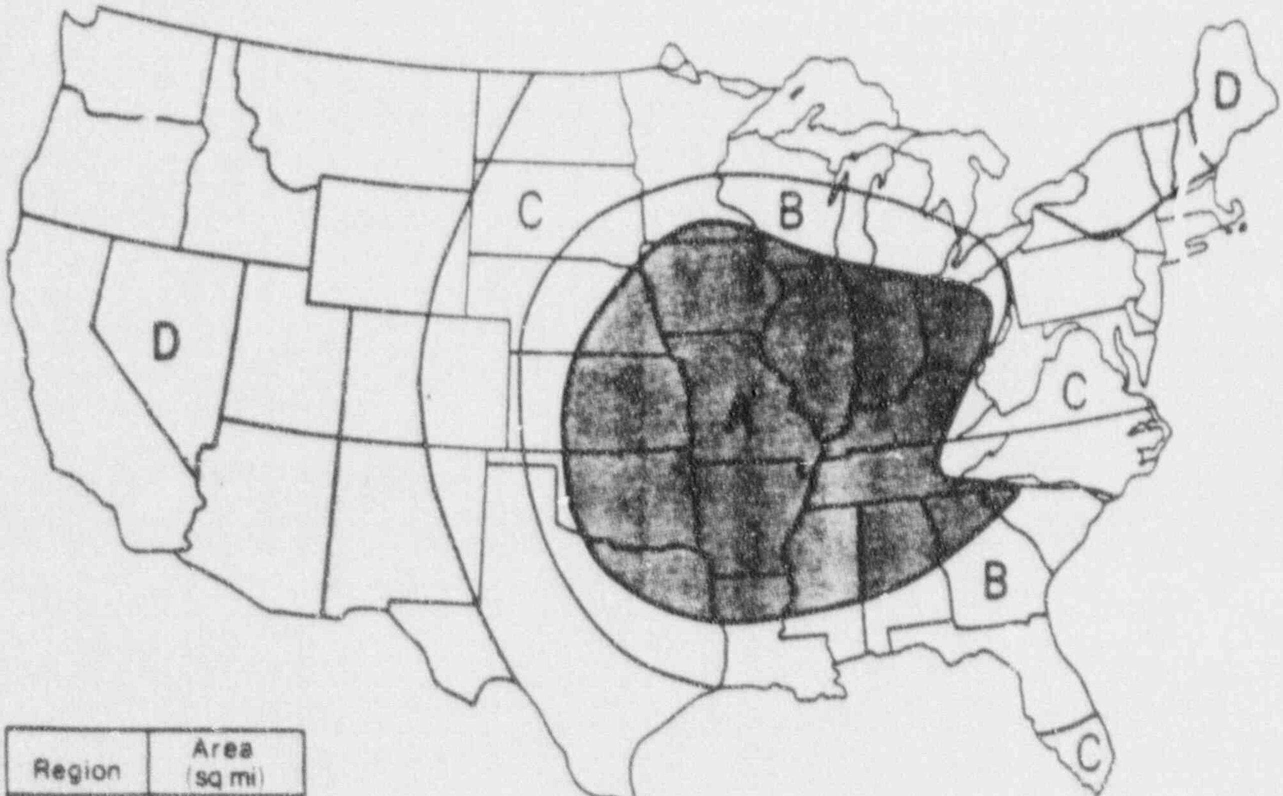


Figure 5.1-2
PCRV Work Area - Elevation View



Region	Area (sq mi)
A	625,628
B	365,180
C	775,569
D	1,205,117

Figure 5.1.3
Large Scale Regionalization for Tornado Risk Analysis

6.0 ALTERNATIVES TO PROPOSED ACTION

6.1 Available Alternatives

The proposed decommissioning alternative for Fort St. Vrain is the DECON, or immediate dismantlement, alternative. Upon approval of the Proposed Decommissioning Plan and removal of all irradiated fuel from the Reactor Building, this alternative will be implemented to immediately decontaminate and dismantle as necessary all plant systems and areas to allow release of the facility for unrestricted use.

The following were also considered as possible alternatives to the DECON decommissioning alternative, and will be evaluated in the following paragraphs:

- No action
- The SAFSTOR Decommissioning Alternative
- The ENTOMB Decommissioning Alternative

The Decommissioning final rule allows the licensee to select any of the approved decommissioning alternatives. As each of these alternatives have been evaluated and determined to be acceptable (with limited exceptions; see ENTOMB below), licensees may choose any approved decommissioning alternative. Current regulations in 10 CFR 50.82 do not require that the licensee provide justification for selection of one approved decommissioning alternative over another.

6.2 No Action

Decommissioning of the Fort St. Vrain Nuclear Generating Station is a regulatory requirement under 10 CFR 50.82, which requires that a licensee must apply for authority to surrender its license and to decommission the facility within two years following permanent cessation of operations, and in no case later than one year prior to expiration of the operating license.

Therefore, taking no action is an unacceptable alternative.

6.3 Alternative Decontamination and Decommissioning Plans

6.3.1 Delayed Dismantlement (SAFSTOR Option)

An alternative to the DECON (immediate dismantlement) decommissioning alternative is SAFSTOR, in which the plant is placed and maintained in a condition that allows the nuclear facility to be safely stored and subsequently decontaminated to levels that permit release for unrestricted use. SAFSTOR can be utilized for a period of up 60 years, based on the expected amount of radioactive decay during an approximate 50-year storage period. In the Generic Environmental Impact Statement (GEIS) on the decommissioning of nuclear facilities (Reference 1), the NRC noted that most of the reduction in occupational radiation exposures occurs in the first 30 years after shutdown. This 30-year reduction is due to the decay of Co-60, which is a large contributor to worker exposure. SAFSTOR beyond 60 years will only be allowed when necessary to protect public health and safety.

PSC has considered SAFSTOR, which was the decommissioning alternative originally selected and submitted to the NRC in the Preliminary Decommissioning Plan (Reference 2). PSC evaluated a SAFSTOR period until 2043, after which the plant would be dismantled, decontaminated, and released for unrestricted use by 2046. In the 54 years after shutdown, Co-60 (with a half-life of 5.26 years) would largely decay away, but europium (Eu-154) would still be present in the PCRV concrete to the degree that concrete removal would be required. Also, remote handling equipment would be required for some of the internal core components. Tritium, with a half-life of 12.3 years will be reduced over four half-lives, but would still be a consideration.

SAFSTOR Preparation Activities

The unique nature of the HTGR design, with all primary coolant system equipment located inside the Prestressed Concrete Reactor Vessel (PCRV), allows the plant to be prepared for an extended decay period with relatively little effort. Unlike LWR facilities, the FSV reactor building was accessible during normal operations, so extensive chemical decontamination of plant systems is not required to allow access during the decay period.

The initial FSV SAFSTOR preparation activities would involve removal of certain accessible components (e.g., control rod drive assemblies and helium circulators) as soon as practicable after shutdown. This would reduce burial costs which are expected to increase over time. For components with special handling equipment like the helium circulators, early removal would also allow use of knowledgeable operating personnel. All PCRV penetrations will be capped and seal welded shut unless the penetrating component remains in place (e.g., steam generators).

SAFSTOR Decay Period Activities

Activities during the decay period include routine industrial surveillance, building maintenance, and security. A minimal level of dedicated staff, familiar with nuclear requirements, will be maintained. An environmental monitoring program will be maintained, periodic NRC inspections are expected, and an appropriate level of ANI insurance will be maintained.

SAFSTOR Delayed Dismantlement Activities

Since remote handling equipment would still be required at the end of the decay period, the SAFSTOR dismantlement methodologies would be the same as are planned for DECON. The PCRV sectioning operation would still involve cutting concrete through the inner tendon tubes, so concrete waste volumes (which are a major part of the decommissioning waste volumes) will be largely the same as for DECON. It is possible that further concrete cutting and volume reduction techniques will be attractive in 2043, so that only a portion of the PCRV concrete inside the inner tendon tubes would be shipped for disposal; however, the feasibility of these techniques is speculative at this time and no credit was taken for a significant reduction in waste volumes after the decay period. The assumed SAFSTOR waste volumes at FSV represent a substantial difference from the reference PWR and BWR studies, which assumed a major reduction (to 10%) in waste volume for SAFSTOR, due to the decay of many contamination and activation products to background levels. This also results in higher estimated costs for FSV SAFSTOR than are estimated for the LWRs.

Engineering evaluations performed by PSC and independent contractors, confirmed during the competitive bid process for selecting the decommissioning contractor, have clearly demonstrated that the technology is available today to dismantle, decontaminate, and decommission Fort St. Vrain in a cost-effective manner. PSC does not need to rely on any major technological breakthrough or advances to perform the complete decommissioning and site release of Fort St. Vrain for unrestricted use.

6.3.2 Encasement for Radioactive Decay (ENTOMB Option)

A second decommissioning alternative to DECON is the ENTOMB decommissioning alternative. ENTOMB is defined as the alternative in which radioactive contaminants are encased in a structurally long-lived material, such as concrete. The barriers must prevent the escape of radioactivity and prevent deliberate or inadvertent intrusion. The entombed structure is appropriately maintained and continued surveillance is carried out until the radioactivity decays to a level permitting unrestricted release of the property. The length of the entombment period depends on the inventory of radioactive nuclides present.

Although the Decommissioning Rule does not specifically preclude selection of the ENTOMB alternative for power reactors, ENTOMB was provided to allow the NRC flexibility when dealing with smaller reactor facilities, reactors which do not run to the end of their lifetimes, or other situations where long-lived isotopes do not build up to significant levels or where there are other site specific factors affecting the safe decommissioning of the facility (e.g., presence of other nuclear facilities at the site for extended periods). Additionally, the NRC has indicated that the ENTOMB alternative is not a viable choice for power reactors.

PSC has completed preliminary activation analyses (Reference 3) of the PCRV and reactor core region, and determined that even after 60-100 years of decay, sufficient radioactive contaminants will be present to preclude site release for unrestricted use. Therefore, ENTOMB is not considered to be a viable decommissioning alternative due to the presence of long-lived nuclides on the Fort St. Vrain site, and the reluctance of the regulatory agency to allow encasement of the large radioactive quantities on-site for an extended period beyond 100 years.

Therefore, the ENTOMB decommissioning alternative is not considered to be an acceptable alternative for decommissioning Fort St. Vrain, and was not considered.

6.4 Radioactive Waste Transportation Alternatives

The primary method of transportation of the radioactive waste from the decontamination and decommissioning of Fort St. Vrain will be truck.

Because of size and weight constraints, the special shielded shipping containers with the steam generator modules will be transported by rail for disposal.

The location of Fort St. Vrain precludes the consideration of conveyance by water as an alternative mode of transportation.

6.5 References

1. Final Generic Environmental Impact Statement on Decommissioning Nuclear Facilities, USNRC, NUREG-0586, August 1988.

2. Preliminary Decommissioning Plan for Fort St. Vrain Nuclear Generating Station, submitted via PSC Letter, Crawford to NRC, dated June 30, 1989, P-89228.

3. Engineering Evaluation - Fort St. Vrain Activation Analysis, EE-DEC-0010 Rev. D, Public Service of Colorado, March 1992.

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7.0 ANALYSIS

7.1 Proposed Action

The proposed decommissioning plan to decontaminate and dismantle activated and/or contaminated portions of the facility will result in several positive impacts. All radioactive contamination as well as asbestos in the areas to be decommissioned will be removed from the site and transferred to an approved disposal facility, thereby allowing the FSV site to be released for unrestricted use. Because of the existing low radiation levels at the plant and considering the escalating costs and uncertainties associated with disposal, there is no advantage in delaying decommissioning activities.

Approved disposal sites and space are currently available. Regulatory requirements are constantly changing, and the availability of a disposal site in the future is uncertain. Over the past several years, the costs of disposal have been increasing about 11.9% per year, and at this rate, they will almost double every six years. Disposal costs are a significant portion of decommissioning costs, so delays in decommissioning will most certainly result in higher costs for completing this project.

No radiological impacts are expected to the public in the vicinity of the FSV site as the result of decommissioning activities.

There will be some worker exposure to radiation during various decommissioning activities. It is estimated that plant workers will be exposed to an estimated total dose of approximately 433 person-rem from various dismantling and cleanup activities. Procedures will be implemented to ensure that worker exposure will be maintained as low as reasonably achievable (ALARA).

There will be a very low level radiation exposure to transportation personnel and the general public from the transportation of radioactive wastes to the disposal site. An estimated 385 truck shipments (160 one way and 225 round trip) will be made over a one way distance of 913 miles. These shipments will be packaged in accordance with Department of Transportation requirements for allowable radiation levels, and the radiation dose impact to workers and the public will be minimal.

Asbestos removal will be carried out in a safe manner according to the requirements of 29 CFR 1910.1010, and no exposure to the general public is expected. Work will be carried out in ventilated areas, and the exhaust will be filtered to remove asbestos particles.

The exhaust from vehicles and equipment will have minor impact on air quality during dismantling operations. These impacts will be minor and limited to the duration of the dismantling activities.

Some fugitive dust will be generated by the dismantling activities. Impacts on air quality will be minimal and temporary in duration.

No known archaeological or historic resources will be impacted by the project.

There are no known threatened or endangered plant or animal species on or near the FSV site that will be impacted.

The expenditure of greater than \$137 million will contribute to the economy of the project. Assuming that 20 percent of the project cost will be spent locally, and using a total impact multiplier of 1.3, the project's total impact on the local economy will be in excess of \$35 million.

7.2 Alternative Decommissioning Plans

Alternative decommissioning plans evaluated were found not to be as acceptable technically, economically, or environmentally as the proposed plan. Each of the three decommissioning alternatives, DECON, SAFSTOR, and ENTOMB, are described in Section 6. The major advantages and disadvantages of each alternative are discussed below.

7.2.1 <u>DECON</u>

As discussed in Section 3.1, this alternative has been chosen for the proposed decommissioning alternative. It involves the decontamination and dismantlement as necessary of plant systems and areas to allow release of the facility for unrestricted use.

The DECON alternative is advantageous because it allows termination of the NRC license shortly after cessation of defueling operations and eliminates a radioactive site. In the Generic Environmental Impact Statement on Decommissioning Nuclear Facilities (GEIS, Reference 1), the NRC concludes that DECON is advantageous if the site is required for other purposes or if the site is extremely valuable; both of these reasons apply to the FSV site because of its potential for conversion to a gas-fired generating station and because the FSV switchyard is valuable for power distribution within the PSC system.

DECON is advantageous in that the reactor operating staff is available to assist with decommissioning and that continued surveillance and maintenance is not required. The disadvantage of DECON, as stated in the GEIS, is the higher occupational radiation dose compared to the other alternatives.

An intangible benefit of DECON, which was a consideration in the decision to undertake this alternative at Fort St. Vrain, is that DECON is more equitable to future generations. The current generation decided to build Fort St. Vrain, and the current generation received the power generated by the plant. DECON does not encumber future generations with the risks, costs, and responsibility for dismantling Fort St. Vrain.

This alternative results in worker radiation exposures that are greater than the SAFSTOR alternative, but the exposures are still substantially less than the exposure estimates for light water reactor dismantlement projects.

7.2.2 SAFSTOR

As discussed in Section 6.3.1, this alternative involves maintaining the plant in a condition that allows the nuclear facility to be safely stored and subsequently decontaminated to levels that permit release for unrestricted use. Based on the 60 year allowance, PSC evaluated placing FSV into a SAFSTOR condition until 2043, and then decommissioning the facility from 2043 until 2046.

SAFSTOR is advantageous in that it results in reduced occupational radiation exposure. Although radiation levels would decrease by 2043, they would still require remote handling techniques.

There are numerous disadvantages that were evaluated in determining the proposed FSV decommissioning alternative:

1. A Possession Only License must be maintained and met under 10 CFR Part 50 at all times. Also, continuing inspections are required and the experienced staff will not be available at the end of the safe storage period to assist in decontamination.

2. Selection of DECON reduces the risk associated with the uncertainties in the regulatory process, a process still very much in a state of development. Regulatory changes that could affect release criteria or disposal requirements

are difficult to predict and may change due to public perceptions and economic realities. Historically, the impact of regulatory changes has been to increase costs to licensees and this potential must be acknowledged for the future. This process could have a significant impact on future cost of decommissioning or duration of the approval process.

3. Financial Risks:

   a. Low Level Radioactive Waste (LLRW): much uncertainty exists and is associated with future costs of low level radioactive waste (LLRW) disposal. Waste disposal costs will likely increase faster than the rate of inflation. Historically these costs have been increasing at about 11.9% per year, which may cause additional financial uncertainties in the final cost of decommissioning the longer the plant is placed in SAFSTOR.

   b. Decommissioning Cost Estimates: uncertainties and contingencies can be much more accurately predicted in the immediate future, than for 50 - 60 years into the future.

   c. Nuclear Insurance Costs. Under the Price-Anderson Act, PSC remains liable for any accident occurring at the site or any other power reactor site (up to $63 million per occurrence) for as long as PSC possesses a Part 50 operating reactor license.

7.2.3 ENTOMB

As discussed in Section 6.3.2, this alternative involves encasing the radioactive contaminants in a structurally long-lived material, such as the concrete PCRV, until the radioactivity decays to a level permitting unrestricted release of the property.

The ENTOMB option is advantageous because of reduced occupational and public exposure to radiation compared to DECON, because little surveillance is required, and because little land is required. It is disadvantageous because the integrity of the entombing structure must be assured in some cases for hundreds or thousands of years, because a possession-only license under 10 CFR Part 50 would be required, and because entombing contributes to the number of sites permanently dedicated to radioactive materials containment.

As indicated in Section 6.3.2, the ENTOMB alternative may be useful when dealing with smaller reactor facilities, but the NRC has indicated that this alternative is not a viable choice for power reactors such as Fort St. Vrain. ENTOMB is not a realistic decommissioning alternative and it is not further evaluated.

7.3 Analysis of Alternatives

This Section evaluates the two viable decommissioning alternatives desc ibed above, DECON and SAFSTOR. The relative costs, radiation exposures, and other environmental impacts are identified in a manner similar to that used in the NRC's GEIS on Decommissioning Nuclear Facilities (Reference 1).

This analysis demonstrates that the environmental impacts of decommissioning the Fort St. Vrain facility are bounded by the previously accepted impacts of either (1) operating Fort St. Vrain, (2) decommissi , Light Water Reactor (LWR), or (3) operating an LWR. This is the same approach taken in the GEIS to show the acceptability of the environmental impacts of decommissioning nuclear facilities.

7.3.1 Cost Comparison

The costs of DECON and SAFSTOR at Fort St. Vrain are identified in Table 7.3-1. For comparison, costs for comparable decommissioning activities at PWR and BWR facilities are provided, based on information provided in the GEIS (Reference 1), NUREG/CR-0130, Addendum 4 (Reference 2) for a reference PWR facility, and in NUREG/CR-0672, Addendum 3 (Reference 3) for a reference BWR facility.

The commitment of financial resources is not the most important environmental impact of decommissioning, although it is an important consideration. The costs identified in Table 7.3-1 illustrate that DECON is a reasonable alternative for Fort St. Vrain. For ease of comparison, the costs in Table 7.3-1 are listed in two ways: (1) as they were published in reference documents and (2) converted to constant 1991 dollars. In order to provide a common basis for comparison, PSC adjusted certain costs as follows:

a. Table 7.3-1 identifies costs for future activities in terms of constant dollars, without taking into account any discounting factors. This affects SAFSTOR costs and is the methodology used in the NRC's decommissioning studies for LWRs (References 1, 2 and 3). However, it is a common practice to reduce future costs by accounting for the cost of capital, so that the present value of work in the future is less than the cost of work performed in the present.

   Using cost of capital methods, the costs of SAFSTOR dismantlement activities performed in 50 years would escalate at the rate of inflation (say for example, 5 percent). However, the SAFSTOR funds set aside to cover these activities are invested at a normally higher rate (for example, 9 percent), so that the funds to be set aside to cover dismantlement activities in 50 years are less than the funds that would be needed to pay for the work today. Using this method of discounting future costs, the $172.7 million FSV SAFSTOR cost identified in Table 7.3-1 would be only about $58.7 million in discounted dollars; this is comparable to the $80.9 million estimated SAFSTOR cost previously identified in the FSV Preliminary Decommissioning Plan (which was a discounted figure).

b. Table 7.3-1 identifies PWR and BWR decommissioning costs that have been escalated from the 1986 costs identified in References 1, 2, and 3. PSC escalated the 1986 costs using actual inflation rates of 1.90% (1986), 3.66% (1987), 4.08% (1988), 4.83% (1989), 5.40% (1990), and 4.23% (1991). This method escalates all cost components (e.g., labor, energy, and waste disposal costs) at the same rate. Although this method is not strictly accurate, it is supportive of a general cost comparison.

c. Table 7.3-1 identifies cost differentials that would result from use of an outside contractor, since that is the method chosen by PSC for the FSV dismantlement. The decommissioning studies in References 2 and 3 estimate that the cost differential for use of an outside contractor for DECON is $14.8 million for a reference PWR (an additional 17%) and $22.9 million for a reference BWR (an additional 21%). To estimate the costs for SAFSTOR dismantlement with an outside contractor, PSC increased the PWR and BWR SAFSTOR dismantlement costs identified in References 2 and 3 by the same percentages.

d. Table 7.3-1 identifies SAFSTOR costs for LWR facilities for a 50 year decay period. This decay period is basically the same as the 52 year decay period planned for Fort St. Vrain. The 50 year decay period costs were obtained from the PWR and BWR studies in References 2 and 3, since the GEIS (Reference 1) only identifies costs for 10, 30, and 100 year decay periods.

e. Table 7.3-1 identifies an FSV DECON cost of $138.3 million. This cost was based on the $157.5 million cost estimate provided in the PDP, converted from future value dollars to 1991 dollars for ease of comparison. This cost includes 18 months for planning and engineering and 39 months to complete dismantlement and decontamination/ disposal activities. As described in the PDP, this cost is based on the fixed price contract between PSC and the Westinghouse project team.

The cost information discussed above and summarized in Table 7.3-1 illustrates that DECON is a reasonable decommissioning alternative for Fort St. Vrain. The estimated cost for DECON activities at Fort St. Vrain is comparable to similar activities at a reference PWR or BWR facility. As identified in the GEIS, these costs are substantial but, in comparison to the costs of building and operating a nuclear power plant, the decommissioning costs are substantially less than the $2-plus billion and 12 years it takes to license and construct a new power plant.

The FSV SAFSTOR cost estimate differs from the LWR SAFSTOR estimates in several areas. The FSV SAFSTOR Preparation costs are substantially lower than the comparable LWR costs because, unlike the LWRs, substantial system decontamination efforts are not required to prepare the FSV reactor building for access. FSV Decay Period costs are greater than those estimated for the LWRs, as the FSV estimate includes costs to retain a minimal level of qualified nuclear personnel and a minimum dedicated industrial security force. Also, FSV's annual ANI insurance premium estimate alone is greater than the annual costs estimated for the LWRs. FSV dismantlement costs are greater than the LWR costs largely because FSV waste disposal volumes will not be reduced as substantially as LWR wastes, as discussed in Section 7.2.2.

It is noted here that the costs of waste disposal in the future could greatly impact the costs of the SAFSTOR option at all plants, although actual future disposal costs are difficult to predict. Waste disposal costs have historically been increasing at about 11.9% per year. Because of various uncertainties in future regulations, volume reduction techniques, and disposal site availability, this escalation rate was not taken into consideration in the Table 7.3-1 costs. However, an 11.9% escalation rate has a cost impact that is considerably greater than the normal rate of inflation, and would substantially impact the FSV waste disposal costs that were based on $140 per cubic foot.

7.3.2 Personnel Exposure Comparison

This discussion addresses occupational exposures, exposures due to waste transportation (worker and public), and public exposures due to accidents.

7.3.2.1 Occupational Exposure

The collective worker doses due to DECON and SAFSTOR activities at FSV are identified in Table 7.3-3, along with doses for comparable activities at the reference PWR and BWR provided in the GEIS.

The primary disadvantage of the DECON alternative is the higher occupational doses when compared to other alternatives. This disadvantage at FSV is perhaps more noticeable than at LWRs because of the exceptionally low occupational doses experienced during FSV reactor operations. Due to the unique HTGR design, annual occupational doses were normally less than 3 person-rem, with the maximum annual dose being 35 person-rem during an extended outage when all 37 control rod drive mechanisms were rebuilt. Thus, the 499 person-rem that PSC estimates as a collective dose for the entire 39 month dismantlement period (433 person-rem for dismantlement plus 33 person-rem for transportation) is greater than previous or projected operational doses. Although the FSV DECON occupational dose is not bounded by normal HTGR plant operations, it is bounded by normal LWR plant operations and also by estimated LWR decommissioning doses.

The dose of 499 person-rem over 39 months for this one-time project is bounded by normal operating doses at LWRs. According to the NRC's report entitled "LWR Occupational Dose Data for 1990," the average collective annual dose at a BWR in 1990 was 433 person rem and the average collective annual dose at a PWR in 1990 was 291 person-rems. Therefore, the 499 person-rem collective dose for the 39 months of FSV DECON is less than that experienced during normal operations at an LWR.

The 499 person-rem occupational dose for DECON is also substantially less than the estimated doses for decommissioning an LWR, as provided in the GEIS. As shown in Table 7.3-3, the estimated occupational doses for the reference PWR are 1215 person-rem for DECON and 333 person-rem for SAFSTOR. For the reference BWR, the estimated occupational doses are 1874 person-rem for DECON and 361 person-rem for SAFSTOR.

The transportation dose estimates due to the shipment of FSV waste will be described in the following section, and are based on methodology used in the GEIS.

The SAFSTOR occupational dose estimate of 22 person-rem is based on the assumption that the major remaining dose contributors will be tritium remaining in the core graphite and europium remaining in the PCRV concrete. The Co-60 dose contribution will essentially disappear. With a 8.56 year half-life for Eu-154 and a 12.3 year half-life for tritium, in 54 years the doses can be expected to decrease to less than 5% of the doses determined for DECON. The 433 person-rem DECON dose is used as a base since it is assumed that the same methodology will be used for SAFSTOR disassembly.

For decommissioning workers, external exposure to radioactive materials is the dominant exposure pathway, since inhalation and ingestion are minimized or eliminated as pathways by protective techniques, clothing and breathing apparatus.

7.3.2.2 Transportation Doses

The FSV occupational exposure data described above includes doses from the transportation of radioactive wastes. These are estimated to be 66 person-rem to truck and train transportation workers from DECON waste shipments and 59 person-rem for SAFSTOR shipments. These estimates for radiation doses from truck and rail transport of radioactive material are based on the method given in Reference 4, which was used in the review of PWR decommissioning costs and technology (Reference 2). Consistent with Reference 2, the following assumptions are made:

1) Shipments will be made in accordance with Department of Transportation regulations (49 CFR 173.393) that set the following limits:

   - 1000 Mr/hr at 3 ft (1 m) from the external surface of the package (provided the package is transported in a closed vehicle)

   - 200 mR/hr at the external surface of the vehicle

   - 10 mR/hr at any point 6 ft (2 m) from the vehicle

   - 2 mR/hr at any normally occupied position in the vehicle.

2) Two truck drivers during a 1200 mile trip from FSV to Richland, Washington would probably spend no more than 29 hours inside the cab (12 hours assumed in Reference 2 for a 500 mile trip ratioed to a 1200 mile trip) and 2 hours outside the cab (twice the time assumed in Reference 2 for a 500 mile trip) at an average distance of about 6 feet (2 m) from the truck.

3) Normal truck servicing enroute would require that two garage men spend no more than 10 minutes about 6 ft (2 m) from a shipment.

4) Onlookers from the general public might be exposed to radiation when a truck stops for fuel or for the drivers to eat. The onlooker dose is calculated on the basis than 10 people spend an average of 3 minutes each at a distance of about 6 feet (2 m) from a shipment (0.00025 person-rem per shipment).

5) The cumulative dose to the general public from truck shipments is based on a population dose of 1.2 E-5 person-rem per km (Reference 4).

6) Train brakemen during the 1200 mile trip would probably spend 10 minutes in the vicinity of a shipping cask car during each stop. Assume an exposure rate of 25 mrem/hr at an average distance of about 3 feet (1 m) from the cask

car (Reference 2). Assume two brakemen at each stop and a stop every 100 miles.

7) The onlooker dose for train shipments is based on an onlooker population of 10 people who each spend 3 minutes at an average distance of about 6 feet (2 m) from a shipment (0.00025 person-rem per shipment).

8) The cumulative dose to the general public from rail shipments is based on a population dose of 1.2 E-5 person-rem/km (Reference 4). This is the same population dose assumed for truck shipments.

The occupational and public transportation doses are presented in Table 7.3-4. These doses are based on 385 truck shipments and 12 rail shipments (one for each steam generator) made to the radioactive waste disposal facility in Richland, Washington. LWR decommissioning estimates in the GEIS include at least five times the waste projected for FSV DECON. This results in greater doses to transportation workers and the general public. It is noted that the transportation route from FSV passes through rural areas (the only urban area along the route is Ogden, Utah), so the estimated public doses are considered very conservative.

The FSV DECON occupational doses for transportation are less than those for LWR DECON (66 person-rem for FSV versus 102 person-rem for a PWR and 110 person-rem for a BWR. However, the FSV SAFSTOR transportation doses are greater (59 person-rem for FSV versus 12 person-rem for a PWR and 24 person-rem for a BWR). This is because the FSV waste volume assumptions for SAFSTOR are based on the concrete cutting methodology and not on the remaining activity level.

PSC considers that all other risks associated with transportation of FSV wastes, whether from DECON or SAFSTOR, are bounded by the analyses in the GEIS. The number of shipments is less than that considered for the reference PWR and BWR facilities. Also, the types of shipments and packaging, and the limits on shipment inventories are both bounded by the same DOT regulations. PSC plans to make about 385 truck shipments over a distance of 1200 miles. This distance is greater than the 500 miles evaluated in the GEIS, but the number of shipments is much less than the 1363 shipments evaluated for the reference PWR and 1495 for the reference BWR. Therefore the accident risks to workers and the public for FSV waste shipments will be no greater than that evaluated in the GEIS.

7.3.2.3 Public Doses - Normal

Three important radiation exposure pathways need to be considered in the evaluation of the radiation safety of normal reactor decommissioning activities: inhalation, ingestion, and external exposure to radioactive materials. The GEIS concludes that inhalation would be the dominant pathway for public radiation exposure at an LWR. As described in Section 4.2, at FSV the dominant pathway for public exposure is ingestion of foods containing tritium, transported from tritiated water released into the environment. Although this is the dominant pathway, it is not significant since releases will be controlled to maintain river concentrations within EPA safe drinking water limits and to ensure that public doses do not exceed the limits of 40 CFR 190.

The amount of tritium estimated to be released as liquid effluent is 500 Curies for DECON. This is greater than the liquid effluent tritium releases made during plant operations. FSV typically released around 200 Curies of tritium in liquid effluent during plant operation, with a peak annual release of 370 Curies in 1983. This release is bounded, however, by annual releases typically made at many PWRs. As identified in Reference 5, at least a dozen PWRs in the United States typically release over 500 Curies per year and two facilities typically release several thousand Curies of tritium in liquid effluent each year.

For SAFSTOR, much of the radioactivity would have decayed and shielding water may not be needed. If PCRV shielding water were used, the amount of tritium released in liquid effluent would be substantially reduced. Based on a 54 year decay of 100,000 Curies of tritium in the core graphite, and assuming a 0.5% release rate, 28 Curies of tritium would be released as liquid effluent during SAFSTOR.

The Committed Effective Dose Equivalent (CEDE) to the maximally exposed member of the public has been calculated to be 0.27 mrem from all reasonable pathways during DECON, due to a release of 500 Curies of tritium in liquid effluent. An additional calculation was performed for the hypothetical case where one of the beef cattle that graze near the plant and could drink liquid effluent is consumed; the CEDE from consumption of this beef is calculated to be 2.3 mrem. These doses are described in the discussion on tritium releases provided elsewhere in this submittal.

The inhalation radiation dose to the public from airborne radionuclide releases during DECON is estimated to be negligible, as it was in the GEIS. As described in the PDP and in the Decommissioning Technical Specifications, PSC will maintain the reactor building in a subatmospheric condition with filtered ventilation during activities that could hypothetically create airborne releases. This will minimize the release of airborne contaminants to the public.

7.3.2.4 Public Doses - Accidents

Potential doses to the maximally exposed member of the public from postulated accidents are identified in PDP Section 3.4. For DECON, the first year doses are a maximum of:

121 mrem whole body and
215 mrem lung,

due to a fire of packaged waste outside the reactor building and within the Emergency Planning Zone where the airborne releases cannot be filtered.

For SAFSTOR at FSV, the maximum public doses have been determined to be:

4.75 mrem whole body and
4.77 mrem lung,

also due to a fire outside the reactor building.

These doses are greater than the LWR decommissioning accident doses presented in the GEIS, where the highest accident doses were:

3.6 E-2 mrem for the reference PWR and
9 E-2 mrem for the reference BWR.

The FSV postulated accident doses are considerably greater than the LWR accident doses, largely due to conservatisms included in the FSV calculations. For example, PSC assumed an atmospheric dispersion factor of 3.5 E-2, where the LWR calculations used a factor of 6.5 E-4; use of the LWR dispersion factor would reduce the FSV calculated doses by a factor of 50.

The postulated FSV DECON accident doses are substantially less than the potential doses to the public from postulated operational accidents. The highest bounding dose identified in the FSV FSAR is a 2.5 rem whole body dose. This dose is identified in FSAR Section 14.11 for the maximum hypothetical accident, a rapid depressurization of the PCRV.

7.3.3 Waste Volume Comparison

Waste volumes for decommissioning are estimated as follows (in cubic feet):

FSV DECON	128,000
FSV SAFSTOR ([illegible] yrs)	115,000
PWR DECON	648,000
PWR SAFSTOR (50 yrs)	65,000
BWR DECON	670,000
BWR SAFSTOR (50 yrs)	63,000

LWR waste values were obtained from the GEIS, converted from cubic meters.

As discussed in Section 7.2.2, the FSV SAFSTOR waste volume estimate is not substantially reduced from the DECON waste volume, which is significantly different from the LWR assumptions. This is largely due to the fact that the continued presence of Eu-154 in the PCRV concrete will require its removal during SAFSTOR dismantlement activities. PSC would plan to use the same concrete removal methodology for SAFSTOR as will be used for DECON. This involves removing the PCRV concrete inside the inner row of tendon tubes and will not significantly reduce the required disposal volumes.

The FSV DECON waste estimate is greater than the waste generated during plant operation. The Final Environmental Statement estimated that approximately 400 cubic feet of solid waste per year would be released. However, the FSV DECON waste volume is considerably less than the waste volumes projected for LWR DECON activities.

7.3.4 Comparison of Environmental Consequences

Radiation doses and costs associated with possible FSV decommissioning alternatives are discussed above. In the cases of both DECON and SAFSTOR, the FSV environmental effects of greatest concern (i.e., radiation dose and radioactivity released to the environment) are substantially less than the same effects resulting from normal LWR operation and maintenance. They are also substantially less than the effects of LWR decommissioning.

A major environmental consequence of decommissioning, other than radiation dose and dollar cost, is the commitment of land area to the disposal of radioactive waste. The waste volumes identified for FSV DECON or SAFSTOR can be accommodated in less than 2 acres of shallow-land burial. Two acres is a small area in comparison with the 2798 acres at the FSV site, and also in comparison with the approximately 50 acres occupied by the fenced area of the plant and other plant buildings, storage ponds, parking lots, and switchyard equipment.

Other environmental consequences of decommissioning are minor compared to the environmental consequences of building and operating a nuclear facility, particularly an LWR. Water use and evaporation are reduced. The Final Environmental Statement discusses an operating cooling tower evaporation rate of as much as 2300 gpm, and this is the largest flow rate that is contemplated for dilution of tritiated water released during decommissioning. This water will not be lost since it will be returned to the downstream surface waters.

The number of workers on site at any time will be no greater than when FSV was in operation. The transportation network is generally in place, but would likely require some future maintenance if SAFSTOR were chosen. Transportation of steam generators by rail will require upgrading the rail spur into the site for any dismantlement option, but this is considered a minor activity.

Disturbance of the ground cover will not take place to any appreciable extent, except for some small degree of soil remediation that may be required to remove any contamination that might be found. The Initial Radiological Site Characterization Program results are still inconclusive about whether any soil remediation will be required. This activity would likely be required for both DECON and SAFSTOR.

The socioeconomic impacts on the surrounding communities are not significantly different for either DECON or SAFSTOR. In either case, PSC plans to convert the facility to a natural gas-fired generating station, so the local tax base will not be greatly affected. There will be significant activity shortly after the completion of reactor defueling for either alternative. For DECON, radioactive components and systems will be dismantled and the plant will be converted for generation. For SAFSTOR, the PCRV and certain radioactive components will be prepared for long term safe storage and the plant will be converted for generation. This would then be followed by a dismantlement period from 2043 until 2046.

7.4 Summary Comparison of Decommissioning Alternatives

The decommissioning alternative proposed for Fort St. Vrain is DECON. The environmental consequences of both DECON and SAFSTOR are acceptable and, with the exception of radiation exposures, are basically the same, largely due to the similarity in dismantlement methodologies for the two alternatives.

Table 7.3-5 provides a summary comparison of the DECON and SAFSTOR alternatives for Fort St. Vrain and for reference PWR and BWR facilities. The following conclusions can be drawn from this table:

- The cost of FSV DECON is less than the cost of FSV SAFSTOR on a constant dollar basis. However, when the cost of capital is taken into consideration, the present value FSV SAFSTOR cost is less than the cost of FSV DECON. The FSV DECON cost is comparable to the DECON costs identified for LWRs.

- Although FSV DECON results in greater worker and public exposures than FSV SAFSTOR, the worker doses are less than those for DECON activities or even for normal operations at LWRs.

- The potential public doses resulting from postulated DECON accidents at Fort St. Vrain are greater than those postulated for LWRs, largely due to conservatisms included in the FSV dose calculations. However, the postulated FSV DECON accident doses are less than those identified for accidents while Fort St. Vrain was operating.

- The amount of tritium released in liquid effluent is considerably greater for FSV DECON than it is for FSV SAFSTOR, but is still within regulatory limits.

The chief advantage of DECON over SAFSTOR is that it removes the uncertainties associated with future disposal costs and regulatory requirements. Also, DECON does not encumber future generations with the responsibility for this action.

In summary, the DECON alternative is the most acceptable of the various feasible alternatives evaluated. The proposed plan will result in the decontamination or removal of contaminated equipment and materials from the FSV site, to an approved waste disposal facility with minimal adverse environmental impacts. The overall positive environmental impacts resulting from the proposed plan will far outweigh adverse impacts. The FSV facility will undergo a thorough cleanup, and unrestricted access will be permitted. The expenditure of greater than $137 million will also have a positive environmental impact.

The environmental impacts of decommissioning Fort St. Vrain are minimal in comparison with accepted impacts of operating and decommissioning LWR power plants. No site specific characteristics were identified that would result in environmental impacts that would be significantly different from those studied generically. The impacts of DECON at Fort St. Vrain are not significantly different from the impacts of DECON at reference LWR facilities (as identified in the GEIS) and these impacts do not significantly affect the human environment.

7.5 References

1. Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, NUREG-0586, August 1988

2. Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, NUREG/CR-0130, 1978, through Addendum 4, 1988

3. Technology, Safety and Costs of Decommissioning a Reference Boiling Water Reactor Power Station, NUREG/CR-0672, 1980, through Addendum 3, 1988

4. Environmental Safety of Transportation of Radioactive Materials to and From Nuclear Power Plants, Directorate of Regulatory Standards, WASH-1238, 1972

5. Radioactive Materials Released from Nuclear Power Plants, Brookhaven National Laboratory, NUREG/CR-2907, BNL-NUREG-51481, 1988

TABLE 7.3-1
DECOMMISSIONING COSTS

(Costs are in millions)

Activity	Previously Published Cost: Utility Staffing	Previously Published Cost: Use of External Contractor	PSC Determined Levelized Cost(1): - 1991 Dollars - External Contractor
FSV			
DECON	--	157.5 (2)	138.3
SAFSTOR			
- Preparation	--	--	2.6 (3)
- Decay Period	--	--	37.6 (4)
- Dismantlement	--	--	132.5
SAFSTOR TOTAL	--	--	172.7 (3)
SAFSTOR Present Value			58.7 (5)
PWR (6)			
DECON	88.7	103.5	131.0
SAFSTOR			
- Preparation	21.8	27.5	34.8
- Decay Period (7)	6.4	6.4	8.1
- Dismantlement	40.5	47.4 (8)	60.0
- Other Costs (9)	4.8	4.8	6.1
SAFSTOR TOTAL	73.5	86.1 (8)	109.0
BWR (10)			
DECON	108.9	131.8	166.9
SAFSTOR			
- Preparation	41.0	50.9	64.4
- Decay Period (11)	6.0	6.0	7.6
- Dismantlement	48.3	58.4 (12)	73.9
- Other Costs (13)	4.6	4.6	5.8
SAFSTOR TOTAL	99.9	119.9 (12)	151.8

Explanatory Notes are on the following page.

Notes for Table 7.3-1:

1. The Levelized Costs have been adjusted from published data for ease of comparison. All levelized costs are provided in 1991 dollars and reflect use of an external contractor.
2. The $157.5 million cost for FSV DECON is in future dollars, based on when they are spent. Reference PDP Section 5.2.3.
3. The FSV SAFSTOR costs do not include the sunk costs and costs to terminate the Westinghouse contract that would be actual PSC SAFSTOR costs if the SAFSTOR alternative were selected today. These additional costs would bring the total FSV SAFSTOR cost to $196.6 million, in 1991 dollars.
4. FSV Decay Period costs based on $723,000 per year for 52 years, as illustrated on the following Table 7.3-2. Includes NRC Licensing charges.
5. FSV SAFSTOR Present Value determined by escalating 1991 cost at projected inflation rates of about 5%, and discounting that figure back at 9%.
6. Published PWR costs obtained from NUREG/CR-0130, Addendum 4 and NUREG-0586, except for contractor costs for dismantlement. SAFSTOR costs are for a 50 year SAFSTOR period. Published Costs are in 1986 dollars.
7. Decay Period costs based on $128,000 per year identified in NUREG/CR-0130, Addendum 4, for 50 years.
8. Dismantlement contractor cost determined by increasing the 40.5 Million cost for utility staffing by 17%, which is the cost differential between utility staff and contractor costs for PWR DECON dismantlement activities.
9. Other Costs include pre-decommissioning engineering costs and NRC Licensing Activity charges (approximately $0.01 million per year, per NUREG-0586, Table 4.3-1).
10. Published BWR costs obtained from NUREG/CR-0672, Addendum 3 and NUREG-0586, except for contractor costs for dismantlement. SAFSTOR costs are for a 50 year SAFSTOR period. Published costs are in 1986 dollars.
11. Decay Period costs based on $120,000 per year identified in NUREG/CR-0672, Addendum 3, for 50 years.
12. Dismantlement contractor cost determined by increasing the 48.3 Million cost for utility staffing by 21%, which is the cost differential between utility staff and contractor costs for BWR DECON dismantlement activities.
13. Other Costs include pre-decommissioning engineering costs and NRC Licensing Activity charges (approximately $0.01 million per year, per NUREG-0586, Table 5.3-1).

TABLE 7.3-2
ANNUAL FSV SAFSTOR DECAY PERIOD COSTS

Staffing w/overheads (1 professional, 1 craft, minor management involvement)	130,000
Security (2 guards/shift, 4 shifts)	240,000
Insurance (Based on ANI conversations)	200,000
Environmental Monitoring (50% of current program)	68,000
Utilities (domestic water, housepower)	25,000
Maintenance (Building maintenance)	40,000
NRC Charges (Annual inspections)	20,000
TOTAL Annual Costs	723,000

TABLE 7.3-3
OCCUPATIONAL DOSE COMPARISON

Activity	Dose (person-rem) DECON	SAFSTOR
FSV		
Dismantlement/Decon	433	22
Transportation	66	59
Total	499	81
PWR		
Dismantlement/Decon	1115	321
Transportation	100	12
Total	1215	333
BWR		
Dismantlement/Decon	1764	337
Transportation	110	24
Total	1874	361

Notes:
1. LWR data from NUREG-0586, Final Generic Environmental Impact Statement.
2. LWR SAFSTOR data is for 30 year storage period, the closest data given in the GEIS to the planned 54 year period at FSV.

TABLE 7.3-4
TRANSPORTATION DOSES

	FSV		PWR		BWR	
	DECON	SAFSTOR	DECON	SAFSTOR	DECON	SAFSTOR
Occupational (person-rem)						
Truck Drivers	62	55	95	10	100	21
Garage men	3	3	4	0.4	5	1
Trainmen	1	1	3	2	5	2
Total	66	59	102	12	110	24
Public (person-rem)						
Onlookers	0.1	0.1	7	1	7	2
General Public	9	8	15	2	3	1
Total	9	8	22	3	10	3
Number of Shipments						
Truck	385	358	1363	139	1495	318
Train	12	12	28	-	43	-
Distance (miles)						
Truck	1200	1200	500	500	500	500
Train	1200	1200	1500	1500	1500	1500

Notes:

1. PWR data from NUREG/CR-0130, Decommissioning the Reference PWR, Section 11
2. BWR data from NUREG/CR-0672, Decommissioning the Reference BWR, Appendix N
3. "-" means information is not identified in the references

TABLE 7.3-5
DECOMMISSIONING SUMMARY TABLE

	FSV		PWR		BWR	
	DECON	SAFSTOR	DECON	SAFSTOR(5)	DECON	SAFSTOR(5)
Cost						
(in $ millions) (1,2)	138.3	172.7(3)	131.0	109.0	166.9	151.8
Worker Exposure						
(collective, person-rem)						
Dismantlement/Decon	433	22	1115	321	1764	337
Transportation	66	59	100	12	110	24
Total 499	81	1215		333	1864	361
Public Exposure						
Transportation (collective, person-rem)	9	8	22	3	10	3
Accident(4) (individual, mrem)	121	5	3.6E-2	2.1E-4	9.0E-2	9.0E-2
Waste Volume						
(cubic feet)	128,000	115,000	648,000	65,000	670,000	63,000

(1) Constant 1991 Dollars

(2) Includes use of an outside contractor

(3) Does not include $23.9 million in sunk costs and costs required to terminate current DECON contracts; present value is $58.7 million in discounted dollars.

(4) Worst case possible accident exposures to maximally exposed individual. Doses are first year whole body exposure data, except BWR doses are lung doses.

(5) 50 year SAFSTOR data for costs and waste volumes, 30 year SAFSTOR exposure data.

8.0 ENVIRONMENTAL APPROVALS

Decommissioning of Fort St. Vrain will require the authorization of several Federal, State and local agencies. Some activities, including the decommissioning itself, will require specific authorization. Others may involve permits and approvals already in effect for operation of the facility. Federal, State and local requirements are identified, and the status for each is reviewed below.

8.1 Federal Requirements

Decommissioning activities that are subject to Federal regulations, permits, licenses, notification or approvals include:

- Initiation of decommissioning
- Handling, packaging and shipment of radioactive waste
- Radio communications
- Worker health and safety
- Worker radiation protection
- Handling and removal of asbestos
- Hazardous waste generation

The majority of these activities fall under the purview of Nuclear Regulatory Commission (NRC) regulations: Title 10 of the Code of Federal Regulations (CFR). Applicable Title 10 regulations are:

- Part 50 - for decommissioning
- Part 20 - for protection against radiation
- Part 51 - for environmental protection
- Part 61 - for disposal of radioactive waste
- Part 71 - (and 49 CFR Parts 171 through 174) for packaging and transportation of radioactive waste.

The EPA Safe Drinking Water Standards in 40 CFR 141 will also be met for surface waters located downstream of the liquid effluent release location.

The Decommissioning Plan requires review and approval by the NRC. Once the Decommissioning Plan is approved and all irradiated fuel has been removed from the Reactor Building, decommissioning will proceed under the conditions established by the Plan. The proposed Decommissioning Plan for Fort St. Vrain was submitted to the NRC in November 1990. Decommissioning will be performed under the existing 10 CFR 50 regulations.

Worker health and safety protection during decommissioning falls under Occupational Safety and Health Administration (OSHA) regulations. These are 29 CFR Parts 1910 and 1926 regulations applicable to construction activities. These regulations include requirements for respiratory protection (non-radiological), hearing protection, illumination, scaffold safety, crane and rigging safety.

Asbestos handling and removal falls under OSHA regulations 29 CFR 1910 and 1926, and Environmental Protection Agency (EPA) regulations 40 CFR Parts 61, Subpart M. In the State of Colorado, the State Health Department administers the EPA regulations dealing with asbestos handling and removal.

Federal Communications Commission (FCC) licenses are required for radio communications equipment used at the Fort St. Vrain site. This would include any radio communications equipment used in the reactor dismantlement and radwaste processing areas.

8.2 State and Local Requirements

Permits and approvals from or notifications to several State and local agencies are required for safety and environmental protection purposes. Some of these are for specific decommissioning activities, and others are for existing FSV site facilities and ongoing activities that are necessary to support decommissioning. Decommissioning activities and related site operations that fall under State and local jurisdiction include:

- Asbestos removal
- Asbestos disposal
- Fuel oil storage
- Air emissions
- Plant service water wells
- Site liquid effluents (non-radiological)
- Building permits
- Hazardous waste generation

At the State level, Colorado Department of Health's (CDOH) Air Quality Control Division regulates the installation, removal and encapsulation of friable asbestos containing materials.

Diesel fuel used during decommissioning is expected to be drawn from existing underground on-site storage tanks. These are regulated by the CDOH's Hazardous Material and Waste Management Division.

Air emissions from the burning of diesel fuel is regulated by the CDOH's Air Quality Control Division.

The site make-up water wells are operated under permits from the Colorado Department of Natural Resources.

The site sewage and non-radioactive liquid effluents are regulated by the CDOH's Water Quality Control Division.

At the local level, building permits will be required from Weld County for any temporary field office facilities constructed on the plant site to support decommissioning activities. Weld County uses the Uniform Building Code for evaluating permit applications.

Hazardous waste generation is regulated by the Colorado Department of Health's Hazardous Materials and Waste Management Division. Notification of the generator status and annual reporting are conducted in accordance with Colorado state regulations.

8.3 References

1. 10 CFR Part 20, Standards for Protection Against Radiation

2. 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities.

3. 10 CFR Part 51, Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.

4. 10 CFR Part 61, Licensing Requirements for Land Disposal of Radioactive Waste.

5. 10 CFR Part 71, Packaging and Transportation of Radioactive Material.

6. 49 CFR parts 171 through 174, Department of Transportation, Hazardous Materials Regulations.

7. 29 CFR Part 1910, Occupational Safety and Health Standards.

8. 29 CFR Part 1926, Safety and Health Regulations for Construction.

9. 40 CFR Part 61, National Emission Standards for Hazardous Air Pollutants.

10. 40 CFR Part 141, National Primary Drinking Water Regulations.

9.0 SUMMARY AND CONCLUSIONS

Public Service Company of Colorado proposes to decommission (DECON) nuclear facilities at the Fort St. Vrain Plant. The Fort St. Vrain High Temperature Gas Cooled Reactor operated from 1977 through 1989. Following this decommissioning the 10 CFR 50 operating license will be terminated and the reactor site restored to unrestricted use.

The major dismantlement and decontamination activities to be performed during decommissioning are described in detail in Chapter 3. The decommissioning project is divided into the following major work areas:

1. Decontamination and dismantlement of the PCRV.
2. Decontamination and dismantlement of the contaminated balance of plant (BOP) systems.
3. Site cleanup and final site radiation survey.
4. Terminate the 10 CFR 50 operating license.

Site cleanup is described in Section 3.15 and the final site radiation survey is described in Section 2.3.

The twelve upper, primary portions of the steam generators will be shipped to a disposal site by rail. Other waste will be transported by truck to waste disposal sites.

Baseline radiation and contamination surveys were performed on the Fort St. Vrain Reactor and Turbine buildings in August 1990. The results showed that fixed contamination levels are generally less than 1000 dpm/15 cm^2 and loose surface contamination levels are generally less than 1000 dpm/100 cm^2 in the Reactor Building. In the Turbine Building, contamination levels (both fixed and loose) are less than 1000 dpm/100 cm^2 in all locations and are generally less than 100 dpm/100 cm^2.

No future nuclear power operations at Fort St. Vrain are planned.

Radiological impacts to the public are expected to be insignificant. Routine dismantling and packaging operations at the Fort St. Vrain site will not result in releases above 10 CFR 20 MPC and 10 CFR 50 Appendix I dose limits.

The planned decommissioning operations that potentially could produce radioactive releases will be conducted only when plant and local ventilation systems are in service.

Workers carrying out various dismantling and demolition activities will be exposed to some radiation. However, radiation exposures of occupationally exposed personnel will be maintained as low as reasonably achievable (ALARA) and in compliance with

10 CFR 20. The total estimated dose for all decommissioning activities at the site has been estimated at 433 person rem. Decommissioning personnel will be protected against airborne radioactivity by the health physics controls and local environmental controls such as portable ventilation exhaust systems with HEPA filters. An NRC-approved Respiratory Protection Program will be used during the decommissioning project.

Accident scenarios were developed for onsite decommissioning accidents. These included: dropping of contaminated concrete rubble, conversion construction near PCRV dismantlement, heavy load drop, fire, loss of PCRV shielding water, loss of power, natural disasters such as tornadoes, and dropping a steam generator primary module. The components with the highest radioactive inventories were used in the accident analyses.

No postulated accident has potential onsite or offsite radiological consequences in excess of a small fraction of the Environmental Protection Agency's Protective Action Guide or above 10 CFR Part 20 limits for routine occupational exposure.

The maximum potential onsite accident dose is due to a truck transportation accident and ensuing fire involving graphite side spacer blocks. Based on worst case site meteorology the postulated dose is 121 mrem whole body and 215 mrem lung dose at the Emergency Planning Zone boundary. Based on generic worst case meteorology, such a transport accident occurring offsite could lead to exposure of an onlooker situated 100 meters from the accident location of 103 mrem whole body and 183 mrem lung dose.

It is not expected that decommissioning activities will have an adverse impact on air quality. The reduced staff and derating of the auxiliary boiler will actually improve air quality relative to past power operation. The impact of the planned natural gas-fired power plant on air quality will be the subject of a separate environmental evaluation.

Asbestos removal operations will only involve that asbestos which must be disturbed to decommission the plant, and are expected to result in no public exposure to asbestos. Work areas will be ventilated and the exhaust filtered. All requirements of 29 CFR 1910.1010 concerning asbestos removal will be met.

Noise impacts from decommissioning will be minimal. The bulk of PCRV concrete will be removed by diamond wire cutting rather than by jackhammer. Demolition will be conducted during the day and the nearest population center, Platteville, is located 3 1/2 miles southeast of the site.

There should be no impacts on wildlife or plants. No threatened or endangered species will be impacted.

Radioactive discharge of tritium is planned. Approximately 500 Curies of tritium are expected to be released during processing of the PCRV cavity water over a one year period. This process will utilize the existing liquid waste discharge system. Water from the plant blowdown will be used to sufficiently dilute the tritiated water to ensure compliance with the maximum concentration limits in 10 CFR 20, the public dose limits in 10 CFR 50 Appendix I, and the Safe Drinking Water Standards in 40 CFR 141 for downstream surface waters.

The proposed decommissioning plan will have a positive socioeconomic impact. A force of approximately 300 PSC and contract workers will be required for decommissioning activities. Over $137 million will be spent on the decommissioning project and as much as 20 percent of that amount is expected to accrue to the local economy.

The proposed decommissioning plan for Fort St. Vrain is environmentally sound and is expected to have no significant impact on the environment. The decommissioning plan will result in the removal of radioactively contaminated equipment, materials, and waste from the site and permit unrestricted use of the decommissioned facilities.

The environmental consequences of decommissioning Fort St. Vrain, by use of early dismantlement methods and techniques described in this Supplement to the Environmental Report, are not significant when compared with accepted consequences of operating and decommissioning Light Water Reactor (LWR) plants. No site specific characteristics were identified that would result in environmental impacts that would be significantly different from those studied generically by the NRC. The impacts of decommissioning the Fort St. Vrain facility are not significantly different from the impacts of decommissioning LWR facilities, and they do not significantly affect the human environment.

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APPENDIX 1

10 CFR 51.45(b) CONSIDERATIONS

This Appendix discusses each of the considerations identified in 10 CFR 51.45(b) for environmental reports. 10 CFR 51.45(b) states that "The environmental report shall contain a description of the proposed action, a statement of its purposes, a description of the environment affected, and discuss the following considerations:

(1) The impact of the proposed action on the environment. Impacts shall be discussed in proportion to their significance;

(2) Any adverse environmental affects which cannot be avoided should the proposal be implemented;

(3) Alternatives to the proposed action. The discussion of alternatives shall be sufficiently complete to aid the Commission in developing and exploring, pursuant to section 102(2)(E) of NEPA, "appropriate alternatives to recommended courses of action in any proposal which involves unresolved conflicts concerning alternative uses of available resources." To the extent practicable, the environmental impacts of the proposal and the alternatives should be presented in comparative form;

(4) The relationship between local short-term uses of the human environment and the maintenance and enhancement of long-term productivity; and

(5) Any irreversible and irretrievable commitments of resources which would be involved in the proposed action should it be implemented."

The discussion of many of these considerations has been provided previously in this ER Supplement, but is repeated or referenced here to provide a concise documentation of compliance with the provisions of 10 CFR 51.45(b). Due to the often lengthy nature of these earlier discussions, information provided previously will generally be incorporated by reference and will not be repeated in its entirety.

1. DESCRIPTION OF PROPOSED ACTION

The proposed action is to decommission the Fort St. Vrain Nuclear Generating Station by use of the early dismantlement and decontamination alternative (DECON). The description of PSC's planned DECON activities is provided in Section 3 of the ER Supplement, and is further described in the Proposed Decommissioning Plan and in PSC's written responses to NRC questions.

DECON is the immediate removal and disposal of all radioactivity in excess of levels which would permit release of the facility for unrestricted use. As described in the PDP, the FSV site will not be completely torn down and restored to a corn field. The non-radioactive equipment and structures will generally be retained and converted, as necessary, for potential future use as a natural gas-fired generating station. Much of the piping and equipment in the reactor building will not be removed even though it is not needed for the converted plant, because it is expected to be readily decontaminated and released for unrestricted use. The end result is the release of the site and any remaining structures for unrestricted use as early as 39 months after the end of reactor defueling.

2. PURPOSE OF PROPOSED ACTION

The purpose of DECON, as identified in Section 1.1 of the ER Supplement, is to reduce residual radioactivity to a level that permits release of the Fort St. Vrain facility for unrestricted use, and that also permits termination of the license.

3. DESCRIPTION OF ENVIRONMENT AFFECTED

A description of the affected environment is provided in Section 4.1 of the ER Supplement. Overall, the areas directly affected by decommissioning Fort St. Vrain include the following:

1) The same areas on the Fort St. Vrain site that were disturbed during plant construction and operation. This site is located in Weld County, Colorado, between the South Platte River and the St.Vrain Creek, approximately 3-1/2 miles northwest of Platteville and 35 miles north of Denver.

2) Radioactive low level waste disposal site(s) that will likely be the same sites used during plant operation, and in any case will be licensed for this purpose.

3) The same transportation routes that were used during plant construction or during plant operation.

4) The same replacement power sources that were utilized during plant operation.

4. DISCUSSION OF ENVIRONMENTAL CONSIDERATIONS

4.1. IMPACTS OF PROPOSED ACTION

The impacts of DECON activities at Fort St. Vrain are addressed in Sections 4 and 5 of the ER Supplement and in the information provided in previous sections of this submittal. In general, the impacts of decommissioning the facility will be positive. The facility will be released for unrestricted use and relatively small amounts of burial space (approximately 2 acres) will be committed for waste disposal. This space is small compared to the approximately 50 acres occupied by plant buildings, ponds, and other facilities.

The costs in dollars and occupational radiation exposure are generally small in comparison to decommissioning costs and operational exposures at light water reactor facilities. Also, the impacts on the environment due to planned releases of tritium are minimized by ensuring that downstream surface water tritium concentrations do not exceed the EPA Safe Drinking Water Standard of 20,000 pCi/l (40 CFR 141).

4.2. ADVERSE ENVIRONMENTAL EFFECTS

Detailed descriptions of the impact of the Fort St. Vrain decommissioning are found in the following sections of the ER Supplement:

- 4.4 Ambient Air Quality
- 4.5 Effects of Chemical and Biocide Discharges
- 4.6 Effects of Sanitary Waste Discharges
- 4.7 Endangered Species
- 4.8 Other Effects (noise, runoff, socioeconomic)
- 5.0 Environmental Impacts of Accidents

Additionally there are a number of decommissioning related environmental impact drivers. The term "impact drivers" as used here refers to the precursors to possible environmental impacts. For example, the incremental work force needed to accomplish decommissioning activities is not an environmental impact, per se, but the resultant effects on housing, transportation, schools, etc., are environmental impacts (Reference 1). The environmental impact drivers for decommissioning are:

- Labor hours and work force size

- Labor costs
- Occupational radiation exposure
- Capital costs (hardware, materials and equipment used during decommissioning)
- Radioactive waste volumes

Labor Hours and Work Force Size

The staffing plan for professionals and crafts labor includes a total of 487 man-years of support by professionals and up to 120 craftsmen per quarter involved in the decommissioning operations. These figures are a net decrease from operations during nuclear power generation.

Labor Costs

It is estimated that over $138 million will be spent on the decommissioning project and as much as 20 percent of that amount is expected to accrue to the local economy.

Occupational Radiation Exposure

A summary of occupational radiation exposure (ORE) estimates for Fort St. Vrain decommissioning is provided in Table 4.3-1 of the ER Supplement. The total exposure is 433 person-rem for dismantlement, as detailed in ER Supplement Table 4.3-2. An additional 66 person-rem is estimated for transportation, for a total of 499 person-rem. These numbers, while not large, are greater than the actual ORE history for a similar period of time during FSV power operation. However, these ORE rates are comparable to ORE rates occurring during normal operations and outage activities at Light Water Reactor power plants.

Capital Costs

Most of the hardware, materials, and equipment used during decommissioning will be purchased out of state by the project team. Thus, no positive impact is expected on the local tax base from purchase of major system components. There will, however, be a positive impact on the local tax base from the mobilization of professionals and craft labor to the FSV decommissioning site.

Radioactive Waste Volumes

The expected volume of solid radioactive waste produced is nearly 128,000 cubic feet. This consists of 113,972 cubic feet from the PCRV and 13,991 cubic feet from the Balance of Plant (BOP). These values are detailed in ER Supplement Tables 3.4-3 and 3.4-4.

The radioactive waste volumes associated with a reference BWR and PWR are 670,000 cubic feet and 648,000 cubic feet respectively (Reference 2). This represents a burial ground area of less than two acres for a reference LWR.

The expected burial ground area for the radioactive wastes from Fort St. Vrain decommissioning is considerably less than two acres. This is a small area when compared to the release for unrestricted use of the approximately 50 fenced acres occupied by the plant buildings and structures.

4.3. ENVIRONMENTAL IMPACTS FROM ALTERNATIVES TO PROPOSED ACTION

The environmental impacts from both the DECON and SAFSTOR alternatives are discussed in Sections 6 and 7 of the ER Supplement.

The SAFSTOR alternative considered for Fort St. Vrain involves a brief SAFSTOR preparation period, a decay period until 2043, and a dismantlement period from 2043 to 2046. The FSV plant can be prepared for the 52 year decay period by removing a few components from the prestressed concrete reactor vessel (PCRV) such as the helium circulators and the control rod drives, and then seal welding shut all of the penetrations to the PCRV. The decay period involves general industrial security provisions, typical industrial facility inspections, and general building maintenance. The final dismantlement methodology will be the same for SAFSTOR as is planned for DECON.

The decay period allows most of the radioactive materials to decay away, but remote handling techniques will still be required. Because of the reduced levels of radiation, SAFSTOR results in lower doses to workers (22 person-rem versus 433 person-rem for dismantlement workers), and lower levels of tritium released in liquid effluent (28 Curies versus 500 Curies). The waste volume is not significantly reduced, however, because the dismantlement methodology will remove the same amount of PCRV concrete as for DECON. The cost of SAFSTOR in constant 1991 dollars is $172.7 million, which is greater than the $138.3 million cost of DECON in constant 1991 dollars. However, by accounting for the cost of capital, the present value cost of SAFSTOR in discounted 1991 dollars is $58.7 million.

4. LOCAL SHORT-TERM USES VERSUS LONG-TERM PRODUCTIVITY

Short-Term Uses

Section VIII of the Final Environmental Statement (FES) Related to the Operation of the Fort St. Vrain Nuclear Generating Station concluded that the short term effects from construction included erosion from freshly graded ground, unavoidable heavy traffic, and noise. While it is expected that decommissioning will also result in some of the same short term effects, these effects, as well as additional ones, were evaluated in Section 4 of the ER Supplement. Specifically, the following effects were evaluated:

1. Ambient air quality
2. Noise
3. Effects of chemical and biocide discharge
4. Effects of sanitary waste discharge
5. Endangered species
6. Runoff
7. Socioeconomic

As discussed in Section 4.0 of the ER Supplement, these short term effects are expected to be less pronounced than the effects from construction and possible continued plant operation. Therefore, it is concluded that the short term use of the decommissioning area by decommissioning workers will not adversely affect the productivity of the local area.

Long-Term Productivity

The Fort St. Vrain site consists of 2798 acres owned by PSC. Approximately 600 acres within the site area is designated as the industrial complex, containing the reactor and power producing facility, of which only about 50 acres are fenced. The reactor building occupies only a few acres within the industrial complex. Farming has been continued on the remaining site areas, including a portion of the industrial area.

PSC does not intend to make the reactor area available for any use other than power production. The facility includes a switchyard and transmission lines that are an integral part of PSC's electric power distribution system, and will not change with

FSV decommissioning. Also, the FSV site includes an Independent Spent Fuel Storage Facility (ISFSI) which is currently in use and is licensed through November 30, 2011. The decommissioned site will be retained indefinitely as an industrial complex. Therefore, there is no conflict between short term uses and the long term productivity of the FSV site.

4.5. IRREVERSIBLE AND IRRETRIEVABLE COMMITMENTS OF RESOURCES

The generic implications of decommissioning power reactors were evaluated by the NRC in NUREG-0586 (Reference 2). The NRC concluded that:

"Decommissioning of nuclear facilities is not an imminent health and safety problem. Decommissioning of a nuclear facility generally has a positive environmental impact. At the end of facility life, termination of a nuclear license is the goal. Termination requires decontamination of the facility so that the level of any residual radioactivity remaining in the facility or the site is low enough to allow unrestricted use of the facility and site. Commitment of resources, compared to operational aspects, is generally small. The major environmental impact of decommissioning is the commitment of small amounts of land for waste burial in exchange for reuse of the facility and site for other purposes."

This is also true for Fort St. Vrain since decommissioning of the facility does not involve the commitment of any significant amount of resources. The facility, after decommissioning, will continue to occupy the same space that was previously evaluated in the NRC's Final Environmental Statement (FES) for operation of the plant, with the addition of approximately two acres for waste disposal at a licensed waste burial site. No resources not previously considered in the FES are required for decommissioning.

4.6. PROPOSED ACTION AND ALTERNATIVES

The alternatives considered for FSV decommissioning are DECON and SAFSTOR. These alternatives and the associated environmental impacts are discussed in Sections 6 and 7 of the ER Supplement. The conclusion of the discussion is that DECON is a reasonable decommissioning alternative for Fort St. Vrain, and the associated environmental impacts are less than those evaluated for decommissioning and operation of typical Light Water Reactor facilities.

The cost of DECON at FSV is less than the cost of SAFSTOR on a constant dollar basis, but when the cost of capital is taken into consideration, the present value SAFSTOR cost is less expensive than DECON. The FSV DECON cost is comparable to the DECON costs identified for LWRs.

Although DECON results in greater worker and public exposures than SAFSTOR, the worker doses are less than those for DECON or even for normal operations at LWRs. The potential public doses resulting from postulated DECON accidents at FSV are greater than those postulated for LWRs, but they are less than those identified for hypothetical accidents while FSV was operating. DECON results in greater amounts of tritium released in liquid effluent than SAFSTOR, but the planned release of 500 Curies of tritium during DECON is comparable to typical releases at numerous operating PWR facilities.

The chief advantage of DECON over SAFSTOR is that it removes the uncertainties associated with future disposal costs and regulatory requirements. Also, DECON does not encumber future generations with the responsibility for this action.

In general, the environmental consequences of DECON at Fort St. Vrain are not significant when compared with accepted consequences of operating and decommissioning LWR power plants. No site specific characteristics were identified that would result in environmental impacts that would be significantly different from those studied generically. The impacts of DECON at Fort St. Vrain are not significantly different from the impacts of DECON at LWRs, and they do not significantly affect the human environment.

5. REFERENCES

1. Generic Environmental Impact Statement for License Renewal of Nuclear Plants, NUREG-1437, NRC, August 1991

2. Final Generic Environmental Impact Statement on Decommissioning Nuclear Facilities, NUREG-0586, NRC, 1988