Public Service

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January 9, 1992
Fort St. Vrain
Unit No. 1
P-92014

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

ATTN: Dr. Seymour H. Weiss, Director
Non-Power Reactor, Decommissioning and
Environmental Project Directorate

Docket No. 50-267

SUBJECT: **PSC RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION ON THE FORT ST. VRAIN PROPOSED DECOMMISSIONING PLAN**

REFERENCES: (See Attached)

Dear Dr. Weiss:

The purpose of this letter is to respond to the NRC's Request for Additional Information (RAI), forwarded to Public Service Company of Colorado (PSC) in Reference 1. The NRC RAI was developed based on the NRC review of a revision to the Proposed Decommissioning Plan for Fort St. Vrain Nuclear Generating Station and a PSC response to the previous NRC RAI (dated February 8, 1991), that were submitted to the NRC in References 2 and 3. As committed in Reference 4, this response provides specific PSC responses to NRC Questions No. 11 (Water Cleanup and Clarification System) and No. 38 (Liquid Wastes).

If you have any questions related to the contents of this letter, please contact Mr. M. H. Holmes at (303) 430-6960.

Very truly yours,

A. Clegg Crawford

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Vice President
Nuclear Operations

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ACC:CRB/cb

Attachment

cc: Regional Administrator, Region IV

Mr. J.B. Baird
Senior Resident Inspector
Fort St. Vrain

Mr. Robert M. Quillin, Director
Radiation Control Division
Colorado Department of Health
4210 East 11th Avenue
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REFERENCES

(1) NRC letter, Erickson to Crawford, dated August 30, 1991 (G-91178)

(2) PSC letter, Crawford to Weiss, dated July 1, 1991 (P-91217)

(3) PSC letter, Crawford to Weiss, dated April 26, 1991 (P-91118)

(4) PSC letter, Crawford to Weiss, dated December 6, 1991 (P-91423)

ATTACHMENT TO P-92014

PSC RESPONSE
TO NRC QUESTIONS NO. 11 & 38

FROM THE NRC RAI
DATED AUGUST 30, 1991

Attachment to P-92014
January 9, 1992

NRC Question 11 (Section 2.3.3.6.2; Water Cleanup and Clarification System)

"Provide analysis of radiological consequences of system operation. Further, provide a safety analysis for potential accident scenarios with regard to occupational and public exposure. Include analysis of radioactive material (including tritium) that is released to this system."

PSC Response:

I. SYSTEM DESIGN CONSIDERATIONS:

A. Background Information:

Decommissioning of Fort St. Vrain requires the dismantlement of the PCRV and PCRV internal components. The sequence of PCRV dismantlement evolutions are described in Sections 2.3.3.7 through 2.3.3.12 of the Fort St. Vrain Proposed Decommissioning Plan [1]. PCRV dismantlement operations will be performed underwater in order to take advantage of the water shielding to maintain radiation exposures ALARA. After flooding the PCRV, dismantlement of the radioactive portions of the PCRV and removal of the PCRV internal components will commence with the PCRV top head removal and then progress downward. This writeup provides a discussion of the expected conditions within the flooded PCRV, design considerations, and description of components and operations of the PCRV Shield Water System (PCRV Water Cleanup and Clarification System), as well as PSC's responses to the specific NRC concerns identified above.

B. Expected Conditions Within the Flooded PCRV:

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

The principal radionuclides of concern are Co-60, Fe-55 and Cs-137. Of these, Co-60 will provide the majority of the whole body exposure to occupational workers as

a result of dismantlement operations. These radionuclides will appear in particulate and ionic form, and the PCRV Shield Water System will be designed to remove the principal radionuclides.

Although not a major contributor to whole body exposures, the other major radionuclide of concern is tritium. Since the tritium cannot be removed by processing through filters or demineralizers, it will be processed and released using liquid effluent discharge operations in accordance with the revised 10 CFR 20 limits. The maximum tritium concentration shall not exceed the limit specified in the Decommissioning Technical Specifications. [2]

2. Particulates: During PCRV dismantlement evolutions, debris will be generated from handling graphite blocks, concrete cutting operations, insulation, and dross from underwater cutting operations. Relatively large particles of debris (½ to 1-inch in diameter) are expected to be generated from the various cutting methods to be employed during PCRV dismantlement operations, including diamond wire cutting (PCRV top head), oxy-acetylene cutting, thermitic rod cutting, and underwater plasma arc cutting. This debris will settle downward in the PCRV water and will be removed by the PCRV Shield Water System. Suitable provisions will be included in the system design to collect this debris and prevent it from damaging system components.

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

C. Design Considerations

The primary function of the system is to provide water shielding to minimize personnel exposure during dismantlement operations internal to the PCRV. The system will also be designed to provide a means to meet 10 CFR 20 discharge limits for the radionuclides identified above and ensure compliance with 10 CFR 50

Appendix I guidance for radioactive liquid waste discharges to unrestricted areas. Specifically, the system design will provide:

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

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

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

(1) maintain acceptable water clarity to conduct underwater dismantlement operations.

(2) minimize corrosion and biological fouling by suitable chemistry control.

(3) provide a means of initial fill of the PCRV, as well as the ability for makeup with demineralized water to compensate for system losses due to effluent discharges.

The recommendations of Regulatory Guide 1.143 "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants" [3] will be used in system design.

This s[illegible]m will be designed to maintain occupational radiation exposures within 10 CF[illegible] regulatory limits and as low as is reasonably achievable (ALARA). The r[illegible]endations of Regulatory Guide 8.8 "Information Relevant to Ensuring That [illegible]ational Radiation Exposures at Nuclear Power Stations Will Be ALARA" [4] [illegible] also be incorporated (to the degree applicable) into system design.

D. <u>Design Experience from Similar Operations</u>

The design of the PCRV Shield Water System incorporates experience gained by the Westinghouse team from the design and maintenance of PWR nuclear power plants, and from the design of similar systems used during the post-accident cleanup efforts at Three Mile Island. The system design has been reviewed with engineers and

chemists who were involved with the design and operation of the Three Mile Island cleanup system and other similar water systems to insure incorporation of "lessons learned" into the Fort St. Vrain design. Experienced gained in the following areas has been incorporated into the design of the PCRV Shield Water System:

- Syste n turnover rate
- Chemistry and biofouling controls
- Maintenance requirements
- Water clarity to support underwater operations

TMI experience was reviewed to determine a suitable system turnover to remove particulate debris from underwater dismantlement activities and to reduce radionuclide concentrations. A system turnover rate of at least 2.5 times per day was determined to be acceptable at TMI and this experience will be factored in the PCRV Shield Water System design. Similarly, consistent with TMI experience and in consideration of the PCRV carbon steel liner, chemistry control will maintain the pH of the shield wate between 9.0 and 10.0. A hydrogen peroxide system was used at TMI to minimize biofouling. Results of this experience indicated that maintaining a concentration of hydrogen peroxide would adequately minimize biofouling in the open-air, atmospheric shield water system.

An evaluation was performed to determine the system filtration and water clarity requirements necessary to conduct underwater operations. Experience demonstrated the usefulness of a surface skimmer system with a capability to perform underwater vacuuming which will be incorporated into system design.

Maintenance requirements, including approximate numbers of filter and demineralizer replacements and associated occupational radiation exposure, were also evaluated in comparison with relevant experience.

II. GENERAL SYSTEM DESIGN INFORMATION:

The PCRV Shield Water System is shown in Figure 11-1 (Westinghouse Flow Diagram 2011E26). The system will consist of two parallel trains of equipment, each sized for 500 gpm (or 50%) of the total flow. This total design flow rate (1000 gpm) will provide a turnover rate of approximately five PCRV volumes per day. Based on discussions with personnel involved in the TMI cleanup, this rate is considered

adequate. The system will be designed to allow a complete train or individual components to be removed from service for preventive or corrective maintenance. Provisions will be included for the addition of another complete train if required. The trains are cross-connected to permit the pumps and filters to be used interchangeably between the two trains.

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

A. Filtration Trains:

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

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

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

3. Filter Trains: Each filter train will consist of two filters, with the filters mounted in a series arrangement. Bypasses will be provided to allow each filter to be operated individually or in series with other filters. Five micron filters will be used to remove the graphite particulate expected while

the PCRV is flooded. The series arrangement for filters will be specified to remove 98% of all particles larger than 5-micron at a maximum differential pressure of 15 psid. Provisions will be included for shielding and radiation monitoring.

B. Demineralizer Train:

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

1. Demineralizers: Two demineralizers capable of removing the principal radionuclides of concern (Co^{++}, Fe^{++}, Cs^{+}) will be provided in parallel. The demineralizer train will be sized for a minimum of 10% of the total process flow (100 gpm) and will remove dissolved radionuclides from the PCRV water inventory. The demineralizer will be enclosed in a shielded housing and will include provisions for remote radiation monitoring. One demineralizer is planned to be in service at all times.

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

3. Demineralizer Pumps: A demineralizer pump will be provided to take suction fro the demineralizer surge tank and provide the necessary flow through the demineralizers. After exiting the demineralizers and the resin fines filters, the demineralizer booster pump will return the processed water to the top of the PCRV or to the radioactive liquid waste effluent release path when effluent discharge operations are in progress.

4. Demineralizer Surge Tank: A demineralizer surge tank will be provided to isolate the low pressure demineralizer sidestream from the higher pressure clarification filtration trains.

C. Chemical Addition Train:

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

1. Chemical Addition Tanks: Two 100-gallon chemical addition tanks will be included, one for sodium hydroxide and the other for hydrogen peroxide. The tanks will be used to add chemicals to the system for the maintenance of proper chemistry and to control biological fouling.

2. Chemical Addition Pumps: Two chemical addition pumps will be included. One pump will be provided for injection of sodium hydroxide into the system to maintain chemistry control and pH balance, and the other pump will be provided for injection of hydrogen peroxide to control biological fouling.

D. Skimmer System:

Consistent with experience gained conducting underwater operations at TMI, a skimmer subsystem has been included in the system design to maintain adequate surface visibility and to reduce surface contamination. The skimmer subsystem will include a duplex strainer, a skimmer pump, a filter and a floating strainer. By changing the valve positions, this system can also be aligned to provide underwater vacuuming capability.

E. System Controls:

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

F. General Design Information:

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

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

III. SYSTEM OPERATION

A. Initial Fill of the PCRV

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

As the PCRV is being filled, the displaced air and gas will be passed through a portable HEPA filter system attached to a refueling penetration in the PCRV head. Using temporary ventilation ducting, the displaced air from the HEPA filter will then

be routed to the installed Radioactive Gas Waste System (System 63) for sampling, and then to the existing Reactor Building Ventilation Exhaust System (System 73) if concentrations permit direct discharge. All gaseous discharges will be in compliance with the Fort St. Vrain Offsite Dose Calculation Manual (ODCM) [5]. The PCRV will be inspected for leaks after the initial fill process is begun. Filling of the PCRV will be stopped at predetermined levels (1/4 core submergence increments) to allow tritium sampling and analysis. The fill operation is expected to take several days.

The Decommissioning Technical Specifications [2] require that the PCRV water be sampled and analyzed daily for tritium concentration during the initial fill of the PCRV. Sample frequency may be reduced to weekly after the tritium concentration has decreased to less than 0.1 μCi/cc. Limits have been established in the Decommissioning Technical Specifications [2] to assure that tritium activity concentrations in the PCRV Shield Water System will not exceed those postulated in the decommissioning accident analyses.

B. Normal System Operation

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

1. System Recirculation to the PCRV:

The normal operational mode of the system will be with both trains processing PCRV water at a total flow rate of approximately 1000 gpm. Both clarifying pumps will take suction from the bottom of the PCRV (total flow rate - 1000 gpm) and will process the water through two parallel trains of filters. During system operation, water flows from the flooded PCRV through duplex strainers to the suction of the clarifying pumps. From the pumps, the water is processed through the filters before returning to the PCRV. A minimum side stream flow of 10% of the water flow rate will be processed through the

demineralizers for removal of radionuclides, namely Co-60, Fe-55 and Cs-137. The filter trains will be set up in a series arrangement depending on the turbidity conditions in the PCRV. During initial operation, 50-micron elements will be loaded into both filter housings in each train. As water clarity improves, filter elements and filter alignment will be changed as needed to support ongoing operating conditions. The clarified water will be returned to the top of the PCRV through a sparger arrangement which will minimize surface disturbance.

2. System Processing Via The Demineralizer:
A minimum sidestream flow of approximately 10% of the total flow (100 gpm) will be taken from the return line downstream of the filtration units and processed through demineralizers. The flow will be adjusted as required to maintain acceptable radiation levels to minimize personnel exposure. Effluent from the demineralizer train can also be routed back to the system return lines for recirculation to the PCRV or, after sampling, routed to the effluent discharge connections as described in the following paragraph. Suitable provisions will also be provided for additional demineralizer capacity as required.

3. Discharge Via Radioactive Liquid Waste System (System 62):
All liquid waste from the Fort St. Vrain decommissioning will be routed through the existing radioactive liquid waste system (System 62) discharge line that was utilized during normal plant operations. Further details of this system are provided in Figure 2.2-23 of the Proposed Decommissioning Plan [1] and Section 11.1 of the Fort St. Vrain FSAR [6]. Discharges will also be performed in compliance with the NPDES permit in effect at that time.

4. Sampling Operations:
Initially, releases will be batch mode releases. Prior to liquid effluent discharge operations, representative samples obtained from the PCRV Shield Water System will be analyzed for principal radionuclides to ensure that the concentrations of radionuclides discharged to the environment do not exceed the values specified in 10 CFR 20. Samples will also be taken to verify maintenance of suitable water chemistry. Sample locations will be included (see Figure 11-1) at the suction line from the PCRV, at the outlet of the filter trains, and at the outlet of the demineralizers.

5. Chemistry Control:

In order to minimize corrosion from the carbon steel components within the PCRV, it will be necessary to maintain the pH balance of the PCRV water to a control band of between 9.0 and 10.0 by the addition of suitable caustic. Additionally, to control biological fouling, it will be necessary to maintain a hydrogen peroxide concentration. A system will be provided for this purpose consisting of two 100-gallon tanks and two chemical addition pumps. The system will initially be used to add chemical during the initial fill of the PCRV. The system will then be used to batch feed chemicals as required until tritium levels are reduced to a level where continuous effluent discharge operations are acceptable. To support a continuous discharge operation, it will be necessary to continuously feed chemicals.

6. System Interfaces:

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

(1) Demineralized Water System (System 31)

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

(2) Radioactive Liquid Waste System (System 62)

Tritium inventories will be initially controlled and subsequently reduced using effluent discharge operations. The discharge from the PCRV Shield Water System will initially be to the existing plant liquid waste holdup and monitoring tanks for processing and subsequent discharge. As tritium levels are reduced, the discharge will be directed to the Reactor Building sump discharge line.

(3) Electrical Power

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

(4) Compressed Air System

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

(5) Heating, Ventilation and Air Conditioning

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

C. <u>System Maintenance</u>

Methods for handling the replacement of radioactive strainers, filter elements and the change out of demineralizer resins will be designed for ease of replacement and will incorporate ALARA concepts, consistent with the recommendations of Regulatory Guide 8.8 [4]. These components will be shielded as necessary to minimize occupational radiation exposures.

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

D. System Removal

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

IV. RADIOLOGICAL CONSIDERATIONS

A. Releases from the System

Tritium concentrations in the system will be controlled by liquid effluent discharge operations. (See PSC response to NRC RAI Question No. 38 attached.) Concentrations of other radionuclides, primarily Co-60, Fe-55 and Cs-137, will be controlled and reduced by reprocessing the PCRV water through the PCRV Shield Water System or the existing Radioactive Liquid Waste System demineralizers. There are three methods for releasing water from the system:

(1) From the discharge of the PCRV Shield Water System demineralizers (primary release path);
(2) From the PCRV Shield Water System return line downstream of the filter trains; and
(3) Directly from the PCRV to the Radioactive Liquid Waste System (with no processing) via a cross-connection from the PCRV suction line;

All three paths will be capable of discharge to: (1) the existing radioactive liquid waste tanks; (2) the Reactor Building sump discharge line; or (3) a PCRV Shield Water System waste holdup tank (see sheet 2 of Figure 11-1).

Initially, the estimated tritium concentration in the PCRV will limit allowable discharges to less than 10 gpm. During this period, system output will be discharged to the existing radioactive liquid waste tanks. Samples will be taken and analyzed for tritium as well as other principal radionuclides. Based on sample results and the limits established in the Fort St. Vrain ODCM [5], an allowable release rate will be determined using the cooling tower blowdown flow to dilute the discharge. Once concentrations of tritium and other radionuclides have stabilized and are reduced to acceptable levels to allow direct discharge from the PCRV, the second discharge flow

path (to the Reactor Building sump discharge line) will be used. The third discharge path provides a path for releasing collected drains from the PCRV Shield Water System after sampling and verification of the allowable discharge rate.

B. Control of Tritium

Tritium levels in the PCRV will be controlled by effluent discharge operations. It is estimated that 500 Curies of tritium will be released from the graphite blocks to the shield water system in the first month of operation. (See PSC's response to NRC RAI Question No. 38 for additional analysis of the amount of tritium projected to be released to the PCRV Shield Water System.) As conditions permit, the PCRV Shield Water System will discharge to the outlet piping of the Reactor Building sump pumps during the discharge operations. The controls and administrative procedures that governed liquid effluent releases from the radioactive liquid waste system during plant operation will also govern the releases from the PCRV during decommissioning. The revised 10 CFR 20 MPC limit for tritium discharge to the environment is 1 E(-3) μCi/ml. Sampling and analysis prior to discharge or during releases will ensure that this limit is met. The PCRV water will be mixed with cooling tower blowdown water for discharge. The discharge flow rate will be established in accordance with the Fort St. Vrain ODCM [5] prior to radioactive liquid release to assure the limits of 10 CFR 20 and 10 CFR 50 Appendix I are not exceeded. At a dilution of 2000 gpm, up to 10.9 Curies can be discharged per day. As the total curie content in the PCRV decreases, the allowable bleed rate is expected to increase.

C. Non-Occupational Radiation Exposure

The doses from the projected liquid tritium discharge have been calculated based on the guidelines established in Regulatory Guide 1.109 [7], and are presented in Section 4.2.4.1 and 4.2.4.2 of the "Supplement to Applicant's Environmental Report - Post Operating License Stage" [8].

D. Occupational Exposure

The occupational radiation exposure that will result from the normal operation and maintenance of the PCRV Shield Water System has been calculated to be a total of 2.5 Person-Rem. Table 11-1 (attached) identifies assumptions regarding maintenance requirements for system components during the dismantlement period. (See also

Section 3.2 of the PDP Detailed Decommissioning Cost Estimate [9].) The whole body exposure is primarily due to Co-60 exposure, since the exposure due to tritium will be a small fraction of the total external exposure. This total represents the radiation exposure associated with system maintenance, filter replacements, demineralizer resin change out, and final removal of the system. The filters and demineralizers will be shielded during operation and will be designed for ease of opening, removal and replacement to minimize occupational exposure. (The projected occupational exposure to personnel from accident scenarios due to tritium and Co-60 are provided in PSC's response to NRC RAI Question No. 38.) The details of the occupational radiation exposure estimate for filter replacement are shown in Table 11-1.

V. SYSTEM SAFETY ANALYSIS

The maximum credible accident involving the PCRV Shield Water System would be the rupture of the system, resulting in the release of the entire liquid contents of the flooded PCRV. The accident scenario has been postulated and analyzed in Section 3.4.7 of the PDP [1]. This accident scenario conservatively assumed that the theoretical maximum amount of tritium (1 E5 Curies) is transferred to the PCRV shield water from the graphite blocks. Furthermore, it is assumed that the entire PCRV inventory of tritiated water spills into the Reactor Building sump/keyway and floods the basement floor to a height of two feet. In analyzing this accident, atmospheric dispersion factors were calculated using the guidelines provided in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequences Assessments at Nuclear Power Plants" [10]. Furthermore, it was conservatively assumed that all releases to the environment were ground level releases. Dose conversion factors were taken from NUREG-0172 [11]. The dose to an individual standing at a point on the EPZ 100 meters from the Reactor Building was calculated to be 34.8 mrem whole body and lung dose for a two hour period. This accident scenario was also analyzed in Section 5.0 of the Supplement to the Applicant's Environmental Report Post Operating License Stage [8], using the AIRDOS-EPA computer model to assess the radiological impact on the general public within 30 miles of Fort St. Vrain. For this analysis, the terrestrial food chain model presented in Regulatory Guide 1.109 [7] was used. The maximum individual dose was calculated to be 4.6 E(-6) mrem red bone marrow dose and 4.6 E(-6) mrem to the lung. These analyses determined that the radiation exposure to the general public as a result of an accident involving the PCRV Shield Water System is very low. In

all cases, the radiological consequences from the postulated accident scenario are well within the 25 Rem whole body dose and 300 Rem to any specific organ guidelines established in 10 CFR 100. Moreover, the radiological consequences are a small fraction of the one Rem whole body dose and five Rem to any specific organ guidelines cited in the EPA Protective Action Guidelines [12].

Limits have been established in the Decommissioning Technical Specifications [2] to assure that tritium activity concentrations in the PCRV Shield Water System will not exceed those postulated in the decommissioning accident analyses.

VI. SAFETY ANALYSIS CONCLUSIONS

The proposed PCRV Shield Water System will be designed for operation in a safe, analyzed condition and the system will not pose an undue risk to the health and safety of the general public and occupationally exposed personnel. Based on the design, operation, radiological considerations, and bounding accident conditions for the system, the following conclusions are determined to be applicable:

A. The design of the PCRV Shield Water System will meet the anticipated processing requirements for the decommissioning of Fort St. Vrain, including periods when major processing equipment may be down for maintenance. Adequate processing, recirculation, and temporary holdup capabilities will also be available during periods of liquid waste generation. In accordance with the guidelines of Regulatory Guide 1.143, system design will also provide suitable provisions to (1) control leakage and facilitate system operation and maintenance, and (2) prevent and collect spills from storage tanks.

B. Administrative controls will be incorporated into the system operation so that the total quantity of all radioactive material released during decommissioning activities to unrestricted areas will not exceed limits established in the revised 10 CFR 20, and therefore will not exceed limits for non-occupationally exposed personnel.

C. Administrative controls will be incorporated into the system operation so that the total radiation exposure will not result in estimated annual dose or dose commitment to occupationally exposed workers associated with

normal system operation and maintenance in excess of allowable 10 CFR 20 occupational exposure limits.

D. Administrative controls will be incorporated into the system operation so that the total concentration of radioactive materials in liquid effluents released to unrestricted areas during liquid discharge operations will not exceed 10 CFR 20 limits.

Attachment to P-92014
January 9, 1992

REFERENCES

1. PSC letter, Crawford to Weiss, dated November 5, 1990, "Proposed Decommissioning Plan for the Fort St. Vrain Nuclear Generating Station", (P-90318) (Revised July 1991, P-91217).

2. PSC letter, Crawford to Weiss, dated August 30, 1991, "Decommissioning Technical Specifications" (P-91278).

3. USNRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants", Rev. 1, October 1979.

4. USNRC Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be ALARA", Rev. 3, June 1978.

5. Fort St. Vrain Offsite Dose Calculation Manual, dated August 29, 1990.

6. Fort St. Vrain Final Safety Analysis Report, Rev. 9, July 1991.

7. USNRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50 Appendix I", Rev. 1, October 1977.

8. PSC letter, Crawford to Weiss, dated July 10, 1991; "Supplement to Applicant's Environmental Report - Post Operating License Stage, for Proposed Decommissioning of the Fort St. Vrain Nuclear Generating Station" (P-91219).

9. PSC letter, Crawford to Weiss, dated June 6, 1991; "Fort St. Vrain Decommissioning Cost Estimate - PROPRIETARY" (P-91198).

10. USNRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", Rev. 1, February 1983.

11. NUREG-0172, "Age Specific Radiation Dose Commitment Factors for a One Year Chronic Intake", October 1977.

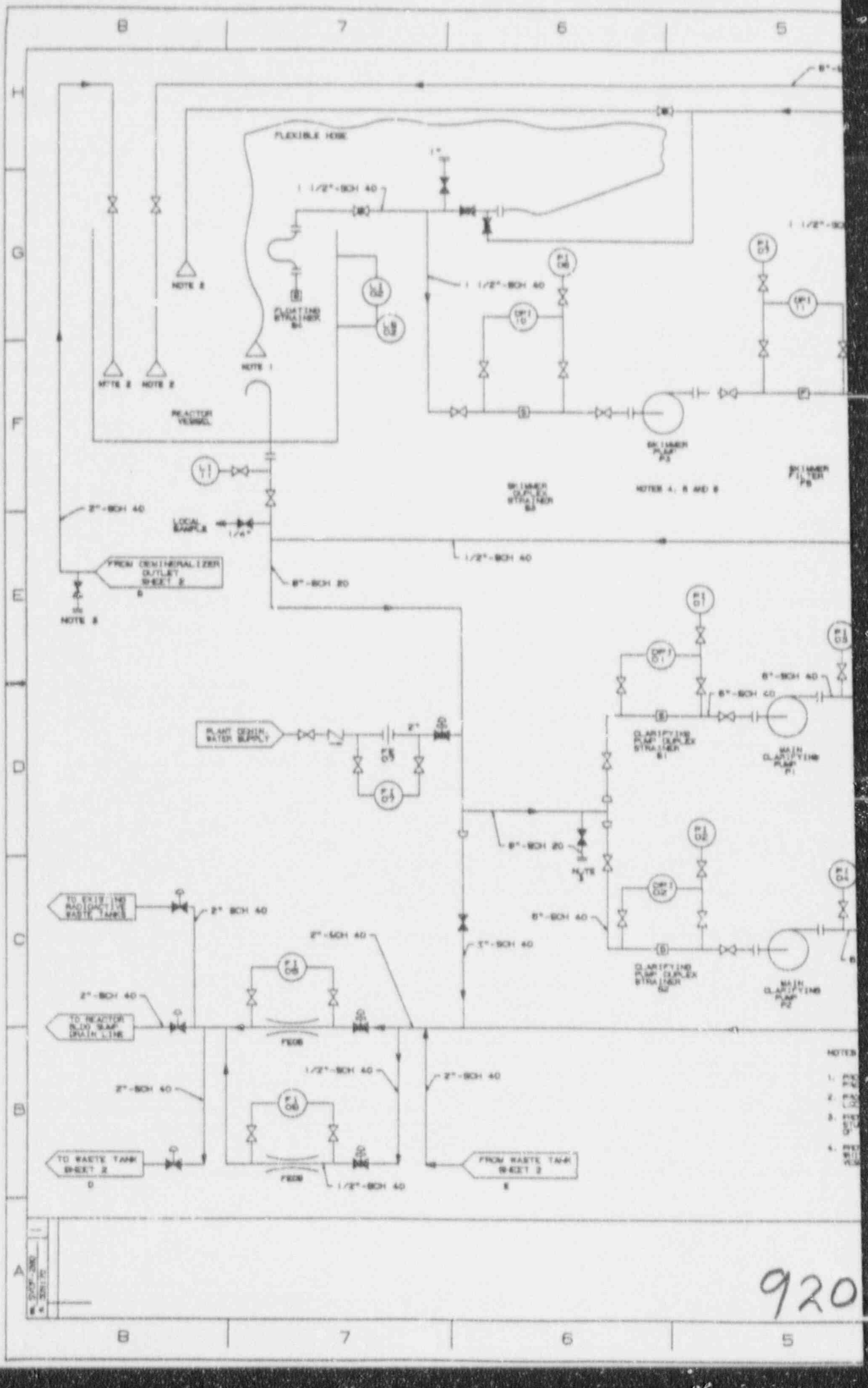
12. EPA "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents", EPA-520/1-75-001-A, U.S. Environmental Protection Agency, January 1990.

TAB...
FILTER REPLACEMENT OCCUPATIONAL RADIATION EXPOSURE ESTIMATE

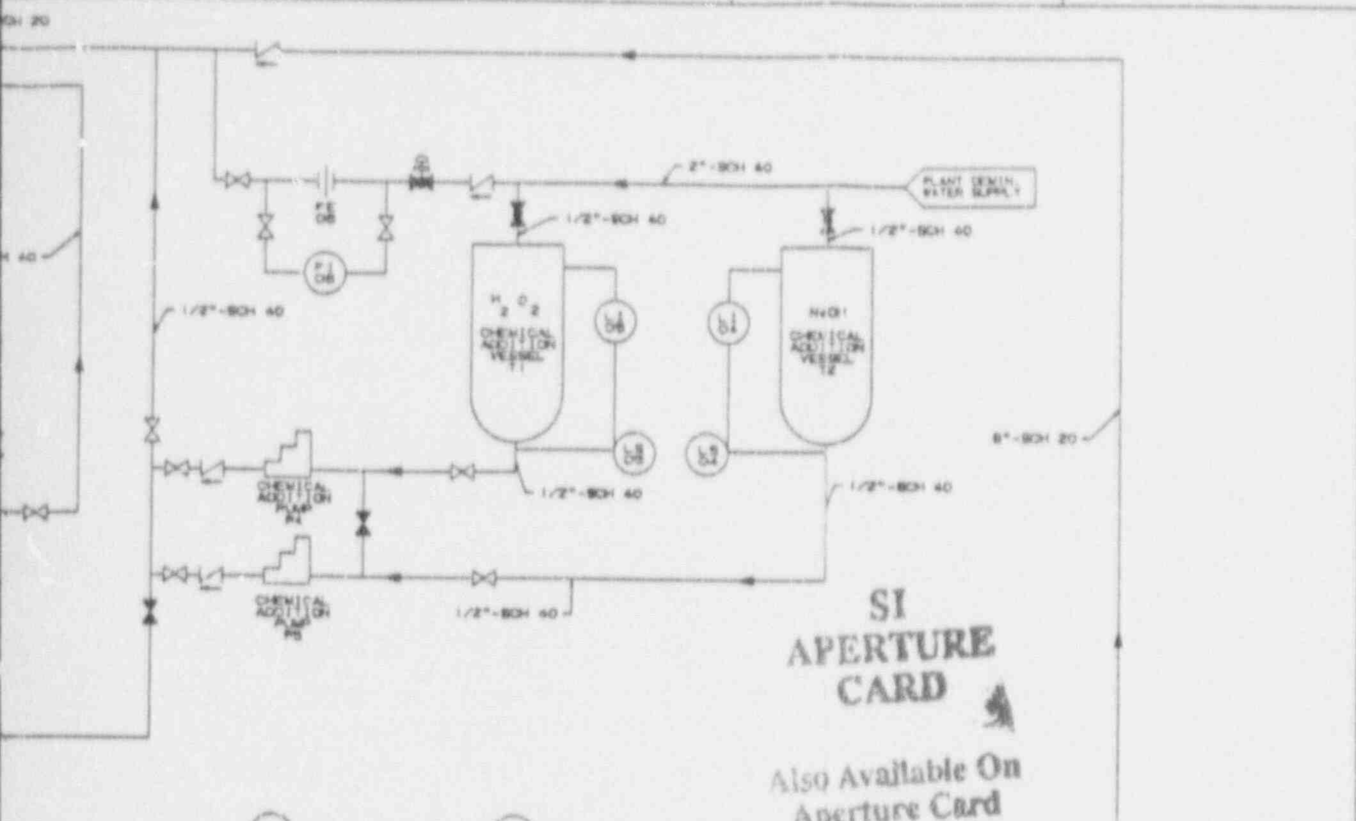
	Operation	No. of Workers	Field (mr/hr)	Time	Person Rem
1.	Open Shield	2	36	2 min	0.0012
2.	Open Filter Housing and Attach Crane	1 1	150 36	2 min 2 min	0.006
3.	Lift Filter to Shielded Cask	2	36	5 min	0.006
4.	Close Shielded Cask	2	36	1 min	0.0012
	TOTAL				0.0144

TOTAL OCCUPATIONAL RADIATION EXPOSURE FOR THE PCRV SHIELD WATER SYSTEM (MAINTENANCE ITEMS)

	Maintenance Operation	Person-Rem
1.	Filter Replacements: • Approx. 0.015 Person Rem per replacement • 100 replacements	1.5 Person-Rem
2.	Demineralizer Resin Change-out: • 0.050 Person Rem per change out • 12 change outs	0.6 Person-Rem
3.	Other Maintenance and Removal:	0.4 Person-Rem
	TOTAL MAINTENANCE ORE BUDGET	2.5 Person-Rem



FLEXIBLE HOSE
1 1/2"-SCH 40
FLOATING STRAINER
NOTE 1
NOTE 2
REACTOR VESSEL
SKIMMER PUMP P3
SKIMMER DUPLEX STRAINER
NOTES 4, 5 AND 6
SKIMMER FILTER
LOCAL SAMPLE
1/4"
2"-SCH 40
1/2"-SCH 40
FROM DEMINERALIZER OUTLET SHEET 2
NOTE 3
8"-SCH 20
PLANT DEMIN. WATER SUPPLY
CLARIFYING PUMP DUPLEX STRAINER S1
MAIN CLARIFYING PUMP P1
CLARIFYING PUMP DUPLEX STRAINER S2
MAIN CLARIFYING PUMP P2
8"-SCH 40
3"-SCH 40
TO EXISTING RADIOACTIVE WASTE TANKS
2" SCH 40
TO REACTOR BLDG SUMP DRAIN LINE
TO WASTE TANK SHEET 2
FROM WASTE TANK SHEET 2
NOTES
920



SI APERTURE CARD

Also Available On Aperture Card

...VIDE DOWNWARD FACING SCOOP ON INLET PIPE TO ...CLUDE INTAKE OF LARGE CUTTING DEBRIS FRAGMENTS.

...VIDE SPARGER PIPES ON RETURN LINES TO MINIMIZE ...AL WATER VELOCITIES AND TURBULENCE.

...VIDE SMALL DRAIN LINES AND DRAIN VALVES ON PIPE ...D FITTED WITH BLANK FLANGES TO ALLOW FLUSHING ...PARTICULATES WHICH MAY COLLECT IN THESE AREAS.

...VIDE DRIP PANS AND LIQUID HOLDUP DAMS ON THE SKIDS TO ...CH THE PUMPS, STRAINERS, FILTERS AND DEMINERALIZER ...SEL ARE MOUNTED.

5. PROVIDE PORTABLE RADIATION SHIELDING FOR STRAINERS, FILTERS AND DEMINERALIZER VESSELS AS REQUIRED.

6. PROVIDE AREA RADIATION MONITORS IN THE VICINITY OF STRAINERS, FILTERS AND DEMINERALIZER VESSELS FOR PERSONNEL PROTECTION.

7. PROVIDE ADDITIONAL VENT AND DRAIN VALVES TO FACILITATE DRAINING AND REFILL OF VARIOUS ISOLABLE PIPING SEGMENTS.

BY	DATE	BY	DATE	BY	DATE
PRELIMINARY INFO ONLY		APPROVED FOR LAYOUT		APPROVED FOR CONSTRUCTION	

DRAWING STATUS

INTERACTIVE GRAPHICS

Westinghouse Electric Corporation

FT. ST. VRAIN DECOMMISSIONING PCRV SHIELDING WATER SYSTEM FLOW DIAGRAM

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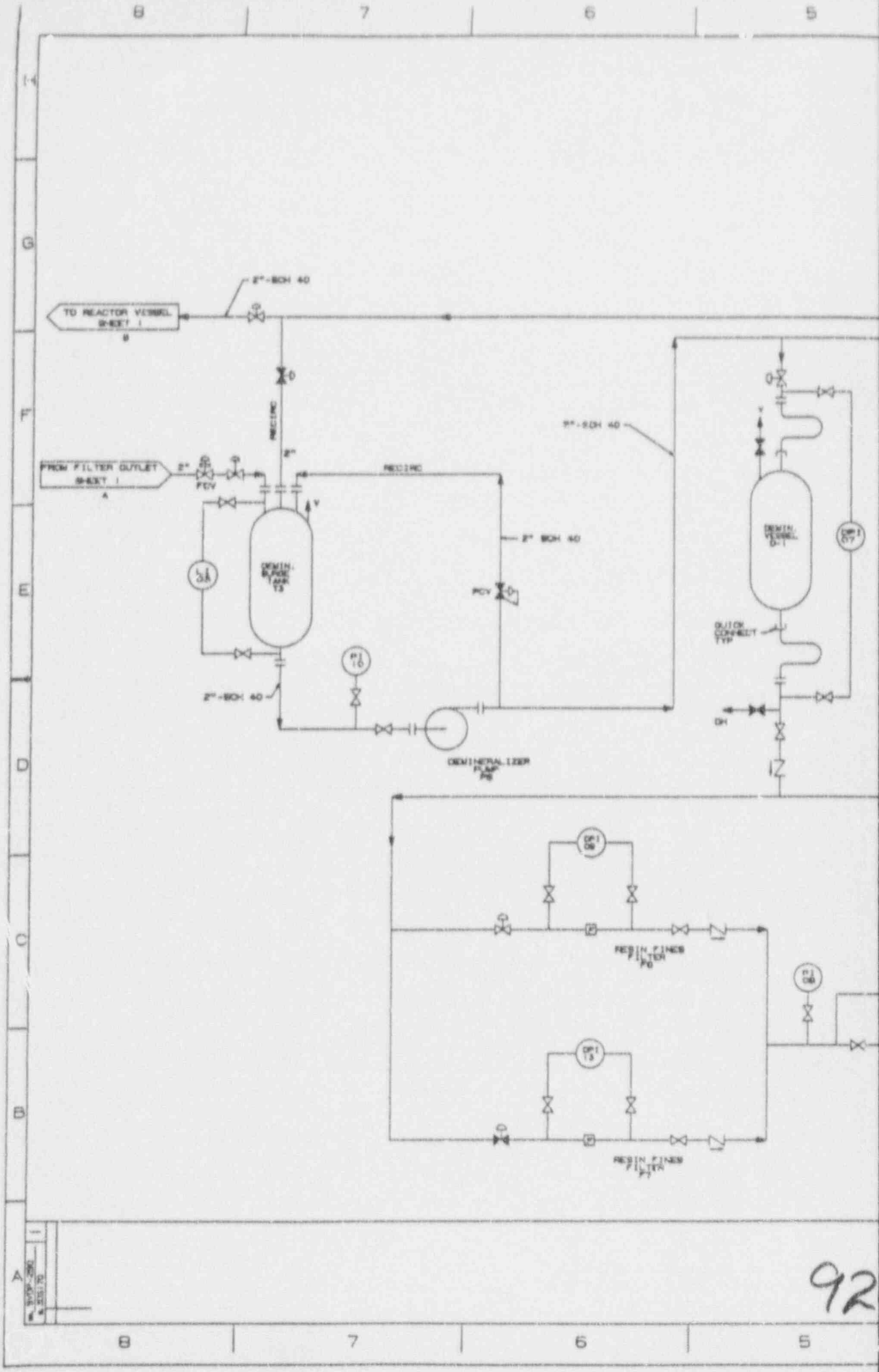
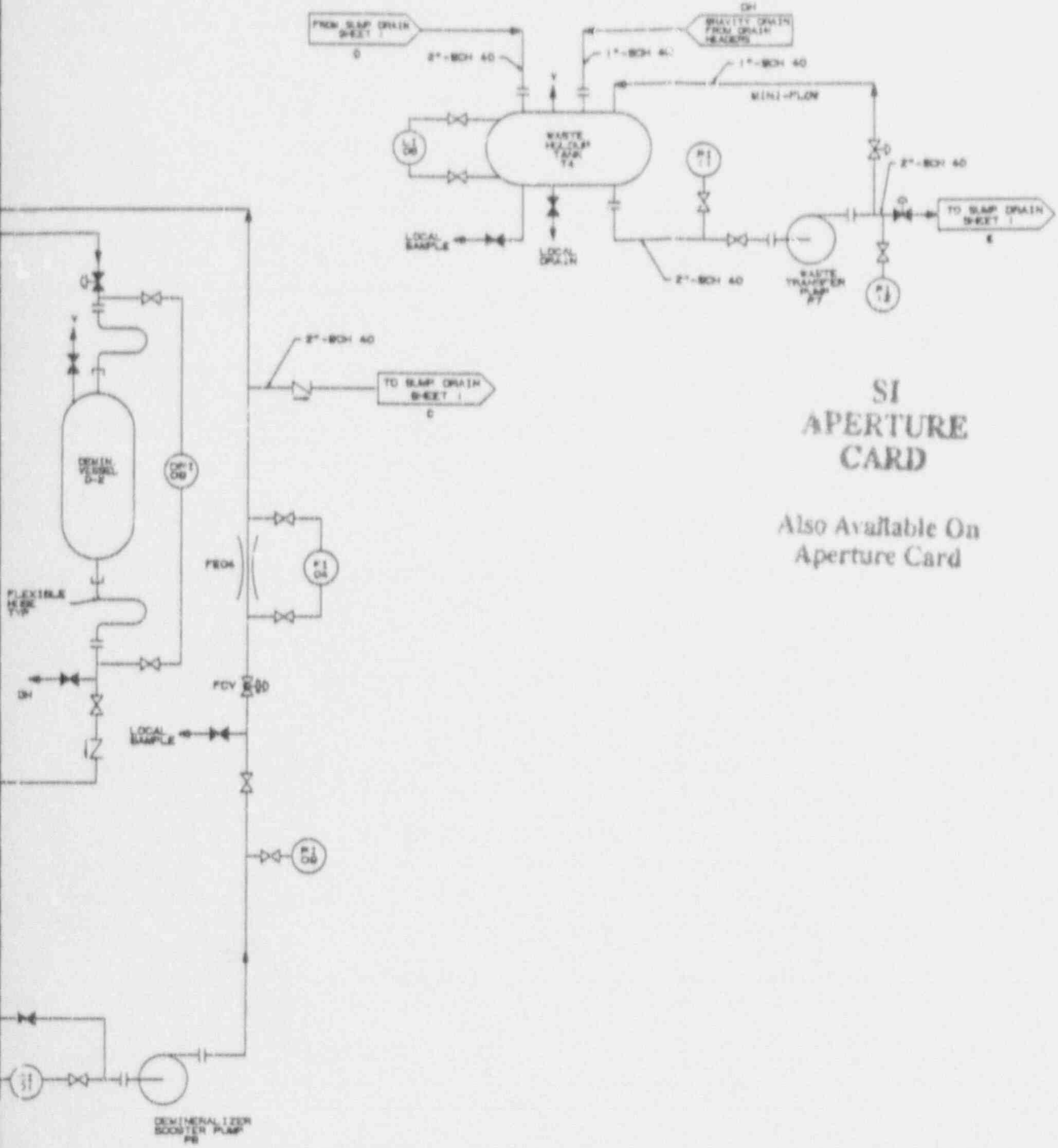


TO REACTOR VESSEL
SHEET 1
2"-SCH 40
FROM FILTER OUTLET
SHEET 1
A
FCV
RECIRC
2"
DEMIN.
SURGE
TANK
T3
2" SCH 40
2"-SCH 40
PI
10
DEMINERALIZER
PUMP
DEMIN.
VESSEL
D-1
QUICK
CONNECT
TYP
DH
RESIN FINES
FILTER
RESIN FINES
FILTER
92



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BY	DATE	BY	DATE	BY	DATE
PRELIMINARY INFO ONLY		APPROVED FOR LAYOUT		APPROVED FOR CONSTRUCTION	

DRAWING STATUS

INTERACTIVE GRAPHICS

Westinghouse Electric Corporation

FT. ST. VRAIN DECOMMISSIONING
PCRV SHIELDING WATER SYSTEM
FLOW DIAGRAM

2011E26

SHT. 2 OF 2

9011150117-02

Attachment to P-92014
January 9, 1992

NRC Question No. 38 (Section 3.3.2.2; Liquid Wastes)

A. *PSC estimates that about 500 curies of tritium would be released to the PCRV shield water from the 100,000 curies of tritium in the graphite (one-half of one percent). PSC's April 26th response and July 1st revision to the Decommissioning Plan states that the estimated tritium release from graphite blocks to PCRV shield water is "based on data." The source of this data must be provided and its accuracy and applicability justified. Include in these evaluations the structure of graphite blocks, e.g. unclad and material composition.*

B. *PSC's July 10, 1991 Supplement to Environmental Report, page 4-12 states that feed and bleed operations would be used to dilute 535 curies of tritium to one-half of 10 CFR Part 20 limits using 2000 gallons of water per minute for about one month. This is based on the data evaluation above. Any change in this evaluation must be reflected in this release plan. Provide verification that the planned releases are consistent with ALARA principles and Environmental Protection Regulations related to 10 CFR Part 51. Discuss other potential release options considered.*

C. *Also evaluate the potential contamination of large volumes of concrete with tritium from water leaks in PCRV penetrations or liner, from drying of wet graphite blocks, from water spills during cask and radioactive meterial handling and from evaporation of water from open PCRV pool surface. As discussed with your staff, tritiated water of hydration in the concrete of a reactor room at a 5 MW, heavy water, research reactor prevented its release for unrestricted use after extensive decontamination efforts (NUREG/CR-3336 "Summary Report Ames Laboratory Research Reactor").*

PSC Response:

I. SUMMARY OF 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. However, since 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, 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 process this tritium inventory for discharge 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 proceed and will be accomplished within the decommissioning cost estimate previously submitted to the NRC, as will be discussed in this response. In addition, with considerations for the worst credible accident and this extreme case, decommissioning will also be accomplished without undue risk to the safety of the public.

II. ANALYSIS OF TRITIUM GENERATION IN FORT ST. VRAIN

A. Sources of Tritium From Reactor Operation

Chapter 11 of the Fort St. Vrain Final Safety Analysis Report (FSAR) [1] provides a detailed discussion of tritium generation in the Fort St. Vrain reactor. As stated in the FSAR, the main sources of tritium are:

(1) ternary fission in the fuel particles
(2) n,p reaction with He-3 in the coolant
(3) n,α reaction by Li-6 impurities in the graphite and other components

The tritium produced in the fuel by ternary fission totals about 10,000 Curies and is expected to be contained by the fuel particle coatings. This tritium will be removed when the fuel is removed prior to decommissioning. The tritium produced from He-3 reactions was minimized by using helium low in He-3 and totaled about 3000 Curies. This tritium was either removed by the gas purification system or was absorbed in the core. Most of this absorption would have occurred in fuel graphite blocks near the coolant channels, and the majority of the tritium generated by this reaction will also be removed when the reactor is defueled prior to decommissioning.

Table 3.1.2 of the Fort St. Vrain Proposed Decommissioning Plan (PDP) [2] identifies the amounts of tritium determined to be present in the PCRV as a result of the activation analysis. However, the removable graphite reflector blocks located at the top, bottom and sides of the core were not included in the results of the activation analysis since it was assumed at that time that these removable reflectors would be removed. However, current plans are for these removable reflector blocks to remain in the PCRV after the fuel elements are removed and replaced with defueling elements. Therefore, it is possible that these removable reflector blocks could be in the PCRV when it is flooded, and the total tritium inventory should be revised to account for the contents of these blocks.

Table 38-1 provides a summary of the key parameters of these hexagonal removable reflector blocks. These blocks are comprised of H-327 and H-451 graphites. Although more than one-third of these removable reflector blocks were removed from the PCRV during the three core refuelings and replaced with unirradiated blocks, it is conservatively assumed that all blocks were irradiated over the full 890 EFPD reactor operating lifetime and were exposed to the same relatively high integrated thermal neutron flux as the large side reflectors. Based on these assumptions (and a Lithium impurity of 0.1 ppm), a tritium inventory of 3,500 Curies is estimated to be contained in the removable hexagonal reflector block graphite.

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

Large permanent side reflectors	82,588 Ci	(84.5%)
Boronated side spacer blocks	11,532	(11.8%)
Removable hexagonal reflector blocks	3,500	(3.6%)
Core support blocks and bottom reflectors with hastelloy cans	48	(<0.1%)
TOTAL	97,638 Ci	

For the purposes of estimating the amount of tritium in the graphite, a tritium inventory of 100,000 Curies is assumed. Per the terms of the fixed price contract, Westinghouse is obligated to remove and dispose of the entire tritium inventory of up to 100,000 Curies.

B. <u>Description of Fort St. Vrain Graphites Retaining Tritium</u>

Figures 38-1 and 38-2 provide plan and elevation views of the core area that identify the key components and type of graphite used for each. From the standpoint of tritium generation, the graphite in Fort St. Vrain is primarily of two types. The primary graphite of interest is the HLM graphite, which was used for the large side reflector blocks and the side spacer blocks. HLM graphite is less pure than the graphite used in fuel elements, with a maximum specification for lithium of 2 ppm.

The second graphite of interest is the relatively high purity graphite (either H-327 or H-451) that was used for the fuel elements and removable hexagonal reflector blocks. This high purity graphite is specified to have less than 0.1 ppm of lithium. The amount of tritium contained in these blocks is expected to be relatively low in comparison to the tritium activity in the HLM graphite blocks.

Table 38-1 provides a summary comparison of the major properties of the graphites found in the core area. As seen from the table and the above description of sources of tritium, over 96.4% of the tritium produced from the Li-6 reaction is produced in the HLM graphite. In particular, due to the neutron flux depression caused by the boron pins in the side spacer blocks and lithium's thermal neutron capture cross

section, most of this tritium will have been generated in the large permanent side reflector blocks. These blocks are solid (with no coolant holes) and have a relatively low surface-to-volume ratio, which is expected to result in lower release rates, as discussed later. Figure 38-2 also identifies the outer extreme limits of hexagonal reflector blocks that can be removed by the Fuel Handling Machine (FHM) prior to removal of the PCRV top head. This figure clearly shows that removal of any of the HLM graphite blocks to obtain tritium data is not possible with the FHM due to the size (weighing approximately 1500 lbs each) and location of these blocks. Additional information can be obtained by reviewing Figures 2.2-5 and 2.2-10 of the PDP. [2]

Conservative calculations of the total amount of tritium that could potentially be in the graphite blocks have been made. Based on the maximum specification value for lithium impurity in the HLM graphite, the theoretical maximum amount of tritium that can be in the total system is approximately 100,000 Curies. This estimate was derived in the activation analysis [2] using the calculated neutron flux in the graphite and the actual reactor power history. The calculated flux was derived from one-dimensional neutron transport calculations in the axial up, axial down, and radial directions. These calculations produce the maximum flux in each direction and therefore are expected to overestimate the activation, especially in corner regions. An additional conservative assumption is that all of the tritium produced in the graphite has been retained, i.e., none has diffused out during reactor operation or subsequent shutdown.

The activation analysis also estimated the curie content of other radionuclides produced in the graphite. Of these, Co-60 will be the major contributor to worker dose (a total of 3,000 to 10,000 Curies of Co-60 may also be retained in the graphite). The Co-60, like the tritium, is also expected to slowly enter the water system, but the cobalt is capable of being removed by the PCRV Shield Water System demineralizers. Another radionuclide of concern is Europium (Eu); however, previous measurements of Fort St. Vrain graphite [2] indicate that Europium will be only a small fraction of the Co-60 activity and therefore is not a major concern during an early dismantlement process.

III. ANALYSIS OF TRITIUM RELEASE INTO THE PCRV SHIELD WATER SYSTEM

A. Description of British Graphite Testing

Data on tritium leaching from graphite obtained by the British [3] 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 also included in Table 38-1 for comparison with data for Fort St. Vrain graphites.

The British leach rate measurements were carried out following IAEA guidelines [4] with a slight modification to expose a larger graphite surface area to the leachant. The British data were taken for several cases; those most appropriate for the Fort St. Vrain case are measurements of leaching in simulated ground water and in demineralized water. Two graphite samples were tested in each test. The measurements were made by placing a graphite sample into a relatively small amount of water in order to produce tritium concentrations that could be easily and accurately measured. However, since the number of hydrogen atoms in the water dwarfs the number of tritium atoms in the graphite samples, any effect of the amount of water or its circulation should be negligible.

For the two samples of British graphite that were tested in demineralized water, the leach rate of the tritium was measured to decrease with time starting at about 0.1% per day and declining to below 0.0001% per day after several months, as shown in Figure 38-4 (Figure 6.6 from [3]). Table 6.6b [3] identifies the cumulative fraction of tritium leached after 100 days in demineralized water to be 0.52%. The fraction leached versus time in demineralized water is shown in Figure 38-3 (Figure 6.2 taken from [3]).

B. Results of Graphite Testing Performed by the French

Measurements of tritium leaching from irradiated graphite in distilled water were performed by the French [5]. These measurements were carried out on two types of

graphite with different porosities obtained from gas cooled reactors. Leaching of the unimpregnated graphite for greater than 90 days resulted in fractional tritium amounts in the water of 0.004% to 0.3%, which are less than the results observed in the British testing. In addition, the French results support an interpretation that reducing graphite porosity (i.e., higher density) will retard the tritium leach rate.

C. Determination of the Fort St. Vrain Leach Rate

As noted previously, 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, as shown in Figure 38-4 (Figure 6.6 from [3]). 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 and absorbed by 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.

The tritium in the PCRV Shield Water System will be gradually removed by effluent discharge operations. Based on allowable discharge limits and the tritium release curve constructed from the British data, an estimate was made of the amount of tritium in the PCRV water system. This estimate is shown in Figure 38-5. This curve is revised from that previously provided in Figure 3.3-1 of the PDP [2] and Figure 4.2-1 of the Supplement to the Environmental Report - Post Operating Stage [6], and assumes that discharges will be in accordance with the revised 10 CFR 20 limits as discussed below. This curve is based on the tritium leaching and discharge rates discussed in the following section on tritium release and monitoring.

D. Comparison of British Test Results vs. Fort St. Vrain Assumptions

The impurities in the British graphite are lower than those in the Fort St. Vrain large side reflector graphite. Therefore, typical tritium levels in the British graphite are expected to be more than 10 times less than expected in the large side reflector blocks, even though the irradiation exposure of the British test samples was higher. The tritium may, however, be assumed to be distributed in a similar fashion in the

two cases since its formation method (i.e., n,α reaction with Li-6) is the same.

Several other factors are expected to affect the leach rates as determined by the British testing. These factors have been evaluated in comparing the British test samples and the Fort St. Vrain HLM graphite, which is the principal graphite of concern since it contains over 96.4% of the tritium retained in the core. These factors include: (1) surface-to-volume ratio; (2) graphite density; (3) reactor power history; and (4) type of primary coolant.

(1) Surface-to-Volume Ratio: One of the most important differences between the graphite in the British tests and the Fort St. Vrain HLM graphite in the PCRV is the significantly smaller surface-to-volume ratio of the Fort St. Vrain HLM graphite. This ratio is significant since the actual diffusion of tritium from the graphite depends on the amount of surface area exposed to the water. Greater surface area will allow greater water ingress into the graphite matrix and reduce the tritium migration distance before it will dissociate into the water. A higher surface area will therefore result in a higher leach rate and greater fractional release. However, the relationship between surface-to-volume ratio and tritium leach rate is not linear since the water ingress into the graphite is a complicated function.

The surface-to-volume ratios for the large side reflectors and the boronated side spacer blocks are approximately 1/20th and one-half, respectively, of the surface-to-volume ratio of the British test samples. Therefore, tritium leach rates for these HLM graphites should be less than those determined in the British testing. Consequently, use of the British leach rate would be expected to bound the tritium release rate from the larger Fort St. Vrain graphite blocks, both in terms of rate of release and total release fraction.

(2) Graphite Density

Appreciably lower density implies that the graphite would have a higher porosity within its graphite matrix. Higher porosity, as indicated in French data, allows more sites for the tritium to collect in interstitial holes with the graphite matrix. This, in turn, implies weaker bonding for this tritium, and more rapid release. Higher porosity will also allow water to more easily penetrate the graphite matrix, which will also allow more rapid release of the tritium (i.e., higher leach rate and greater fractional release).

Table 38-1 compares the Fort St. Vrain HLM graphite with the British Magnox test samples, and unirradiated densities of both are nearly equal, while the irradiated density of the British graphite is less than the HLM graphite. Therefore, based on density properties of irradiated graphite, the leach rate for the HLM graphite should be less than that determined by the British testing.

(3) Reactor Power History

Reactor power operation will subject the graphite matrix to damaging irradiation effects. One possible result is increased radiation damage to the interstitial graphite matrix, which can increase the porosity of this matrix.

Comparing the HLM graphite with the British test samples, both were exposed to a thermal neutron flux of approximately 3 E13 neutrons/(cm^2*sec). However, due to the low power history of Fort St. Vrain, the HLM graphite was subjected to only 890 EFPD, whereas the British test samples were exposed to over 3500 EFPD. Therefore, the potential damage to the British test samples would be expected to be greater than that experienced by the HLM graphite, resulting in more irradiation damage to the British graphite, increased porosity, and a higher leach rate than the HLM graphite under these circumstances.

(4) Type of Primary Coolant

The effects of primary coolant were also evaluated. British Magnox reactors use carbon dioxide as the primary coolant, whereas helium was used as the primary coolant in the Fort St. Vrain reactor. Carbon dioxide, when used as a coolant at elevated temperatures within a graphite-moderated reactor, will react and remove carbon from the graphite moderator and produce carbon monoxide. Over time, this process will decrease the graphite's density and increase its porosity. As noted above, this increased porosity will increase both the tritium leach rate and the fractional release of the total tritium inventory.

Helium, when used as the primary coolant, will not react with the graphite and therefore will not affect the density or porosity of the graphite. (As noted previously, He-3 will react to create tritium, some of which will be retained within the core graphites; however, as discussed, it is expected that nearly all of this tritium is retained in the fuel element graphite that will be removed prior to beginning decommissioning.) As noted with reactor power history, the potential damage to the British test samples would be expected to be greater than that

experienced by the HLM graphite, resulting in increased porosity and a higher leach rate than the HLM graphite under these circumstances.

(5) Comparison of Key Parameters

For each of the above parameters, the properties of the Fort St. Vrain HLM graphite are expected to result in a more conservative behavior than the British Magnox test samples 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 is 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.

IV. TRITIUM PROCESSING, RELEASE AND DISPOSAL OPTIONS

A. Tritium Release Alternatives

Since tritium cannot be removed from the water by processing, it must either be diluted to releasable levels or disposed of as radioactive waste. The release option was chosen for the low tritium concentrations expected in the PCRV Shield Water System. The following tritium release options were evaluated for the Fort St. Vrain decommissioning program.

(1) Construction of a large shallow water impoundment for solar evaporation of tritium.
(2) Installation of a series of mechanical evaporators for forced evaporation of tritium.
(3) Use of a discharge system (PCRV Shield Water System) to discharge tritium through the existing Fort St. Vrain liquid effluent release pathway.

Solar evaporation was evaluated and deemed inappropriate at this time for the following reasons:

1. The evaporation rate can not be controlled.
2. Rainfall or snow could add to the amount of tritiated liquid to be evaporated.
3. Required security and access fencing.
4. Difficulty in preventing migrating birds and small animals from entering the area.
5. Potential occupational radiation exposure to decommissioning personnel.
6. The impoundment presents an additional accident source term, specifically in the event of rupture of the impoundment.
7. Costs associated with lining the impoundment.
8. Disposal of sludge and ultimate decontamination of the impoundment area.

Mechanical evaporators were also evaluated and deemed inappropriate at this time for the following reasons:

1. Large throughput mechanical evaporators are expensive and require constant attention.
2. Large number of evaporators are required for the necessary throughput and operational flexibility in the event of evaporator breakdown.
3. Concern over input stream to evaporators (i.e., oils and detergents).
4. Potential occupational radiation exposure to decommissioning personnel.
5. Eventual decontamination and disposal of evaporators is required.

The PCRV Shield Water System, discharging to the existing Fort St. Vrain plant liquid effluent stream, was selected as the best possible release path for tritium in the PCRV shield water. In addition, occupational and public doses from effluent discharge operations are the lowest of all the alternatives evaluated. This effluent discharge path was fully analyzed in the Fort St. Vrain Safety Analysis Report [1] and the original Supplement to the Environmental Report [7], and the impacts during normal plant operations were determined to be acceptable to the NRC. Moreover, the PSC environmental monitoring program, which complies with the Regulatory Guide 1.21 [8], has confirmed no significant impacts to the environment due to discharges of tritiated water over the last fifteen years of operation.

The current tritium discharge pathway is modeled by the Fort St. Vrain Offsite Dose Calculation Manual (ODCM) from this release pathway. This pathway currently has adequate measuring and monitoring capabilities for the anticipated discharge level (both water quantity and curie content). This pathway provides adequate water to dilute the anticipated quantity of tritium to below the revised 10 CFR 20 MPC limits.

In summary, the discharge of tritium to the existing Fort St. Vrain liquid effluent stream provides the most advantageous method for tritium release during decommissioning. In addition, it is an accepted and demonstrated safe method that will minimize both occupational and public doses.

B. Solidification of Highly Tritiated Water

In the unlikely event that the amount of tritium entering the water greatly exceeds the expected levels, and the effluent discharge release method cannot be used, alternate disposal methods are available. In this case, after tritium pickup by the water is complete and suitable containers are in place, a feasible contingency plan is to remove the water from the PCRV in its entirety and solidify it for disposal. For cost estimating purposes, an acceptable solidification process would be to use Aquaset, which has a solidification efficiency of 45 gallons in a 55 gallon drum. The disposal effort requires about 1 hour per drum.

Appropriate radiological controls would be implemented during the solidification of the tritiated water to maintain external and internal radiation exposures ALARA. The occupational dose resulting from this operation would be negligible and any incident involving a small spill of tritiated water during solidification operations would be bounded by the scenario described in Accident Scenario 3 "Small Spill" (described in paragraph V.D. of this response.)

Major costs associated with this method include the Aquaset ($250 per drum) and additional waste disposal costs of $300 - $1100 per drum, based on projected disposal costs between $40/ft^3 - $140/ft^3. The total disposal cost for the water in the PCRV could range from $5 million - $10.0 million (including labor, materials, transportation, and burial costs). Because of the increased costs and inability to continuously improve water quality, the solidification method would not be used unless the tritium level greatly exceeds expected levels. Solidification of highly tritiated water is discussed here to demonstrate that suitable technology is currently

available, and to establish a bounding cost for disposal should the preferred method not be suitable.

C. Tritium Release and Monitoring

The PCRV will be filled with approximately 325,000 gallons of water. Filling of the PCRV will be stopped at predetermined levels (1/4 core submergence increments) to allow tritium sampling and analysis. No discharge will be made until the trend of tritium concentration is determined. The initial concentration of tritium in the PCRV (approximately 5 days after fill) is estimated to be less than 0.40 μCi/ml, based on 500 Ci of tritium diluted in 325,000 gallons of water.

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

Once the trend of tritium concentration in the PCRV is established, discharge will begin. Water from the PCRV will be processed through the PCRV Shield Water System (See RAI #11 for details) and a side stream will be transferred to a liquid waste holdup tank in the existing plant Radioactive Liquid Waste System (System 62). The tank will be sampled for tritium and other principal radionuclides. Based on sample results and the limits prescribed in the Fort St. Vrain Offsite Dose Calculation Manual (ODCM) [9], an allowable release rate will be determined. This method of redundant monitoring will ensure that the desired discharge concentration (less than MPC) is not exceeded.

After sampling, the liquid in the liquid waste holdup tank will be initially discharged at a rate from 1.4 - 10 gpm and diluted by the cooling tower blowdown flow prior to release to the surrounding surface water. The minimum cooling tower blowdown flow of 1100 gpm defined in the ODCM will ensure a dilution factor of more than 100. Figure 38-5 shows the projected decrease in PCRV tritium concentration assuming a discharge of up to 10.9 Curies per day (2000 gpm cooling tower blowdown discharge) until tritium concentrations drop below levels where this rate

can be maintained without requiring dilution to meet the revised 10 CFR 20 MPC limits. Tritium concentration will continue to be reduced, and after approximately 3 months of effluent discharge operations, the PCRV water tritium concentrations will be low enough to allow direct discharge to the environment (i.e., less than 10 CFR 20 MPC limits). Discharge at a slower rate (1100 gpm, resulting in a release rate of 6 Curies/day or less) would extend the time to reach the 10 CFR 20 concentration limits.

Water consumption requirements to support the dilution of the vessel water may range from the minimum flow rate of 1100 gallons per minute to a flow rate of more than 2000 gallons per minute. Existing site capacity can accommodate these requirements and no additional water sources are required. FSAR Section 2.5.1 details the make up water supply which includes water diverted from the South Platte River and St. Vrain Creek.

In summary, the estimated 500 Curies of tritium released from the graphite blocks into the PCRV shield water will be sampled, analyzed and discharged through the existing Fort St. Vrain liquid effluent release pathway. The established liquid effluent pathway will be used to dilute the effluent to less than the revised 10 CFR 20 limits. The liquid effluent pathway uses cooling tower blowdown (nominally 1100 to 2000 gpm) to dilute releases to ensure that the releases are consistent with 10 CFR 50 Appendix I and 10 CFR 51. The controls and administrative procedures that governed releases of radioactive liquid wastes during plant operation will also govern the releases from the PCRV during decommissioning. Sampling and analysis prior to discharge or during releases will ensure that the limits of 10 CFR 20 and 10 CFR 50 Appendix I are not exceeded.

V. RADIOLOGICAL CONSIDERATIONS

A. Non Occupational Radiation Exposure

The effluent discharge operation will be capable of discharging up to 6 Curies per day using an 1100 gpm blowdown rate, or up to 10.9 Curies per day using a 2000 gpm blowdown rate. Using the existing Fort St. Vrain ODCM [9], the projected annual dose to a member of the public is estimated to be less than 1.0 mRem. This assumes that all 500 Curies are discharged in a manner consistent with the revised 10 CFR 20 limit in approximately 3 months, and a dose rate conversion factor of

0.226 mRem/hr/μCi/ml is used. The resulting dose to any member of the public due to the proposed discharge is a fraction of the guideline provided in 10 CFR 50 Appendix I (i.e., less than 3 mRem total body or 10 mRem to any organ).

B. Occupational Radiation Exposure From Normal Operations

Some limited occupational exposure to tritium is expected to occur while performing decommissioning work activities located over the flooded PCRV. The resulting personnel dose from tritium is expected to be a small fraction of any doses resulting from external radiation.

The work activities most likely to involve exposure to tritium will be those performed on or around the rotary work platform over the PCRV. (See the previous PSC response to NRC RAI Question No. 13 [10] for conceptual drawing.) The work platform will be equipped with an air handling system to exhaust the air from the space between the work platform and the open PCRV pool surface, and discharge it to the Radioactive Gas Waste System (System 63) for sampling and to the Reactor Building Ventilation System (System 73) for exhaust. This air flow will virtually eliminate any exposure to workers on the work platform due to tritium evaporation from the open PCRV pool surface. In addition, workers handling wet items will be provided protective clothing to minimize direct contact with tritiated water.

A tritium bioassay program will be implemented as part of the Decommissioning Radiation Protection Program and tritium air sampling equipment will be available to ensure the timely detection and assessment of individuals likely to be exposed to tritium. Personnel exposure to tritium during decommissioning is expected to steadily decline as the PCRV water tritium concentration decreases due to the effluent discharge operation. In fact, the tritium concentration in the PCRV water is projected to be reduced to 0.01 μCi/cc after about 2 months of discharge assuming a 2000 gpm blowdown rate (See Figure 38-5). According to Regulatory Guide 8.32, "Criteria for Establishing a Tritium Bioassay Program" [11], operations involving tritium concentrations below 0.01 μCi/cc are sufficiently low that a bioassay program is not specifically required. Additionally, the bulk of the activities involving workers handling wet, contaminated items will occur after this 2-month period.

C. Evaluation of Worst Case Accident Conditions

The maximum credible accident involving the PCRV Shield Water System would be the rupture of the system, resulting in the liquid release of the entire content of the flooded PCRV. Although such a release is considered to be improbable, this accident scenario has been postulated and was analyzed in Section 3.4.7 of the PDP [2]. This accident scenario was analyzed to bound the offsite radiological consequences that could result from the installation and operation of the PCRV Shield Water System during normal operation and from potential accidents.

This accident scenario conservatively assumed that the theoretical maximum amount of tritium (100,000 Curies) is transferred to the PCRV shielding water from the graphite blocks, resulting in a tritium concentration in water of 62.4 μCi/cc. Furthermore, it is also assumed that the entire inventory of the PCRV water spills into the Reactor Building sump/keyway and floods the basement floor to a height of two feet.

The dose to an individual standing at a point on the Emergency Planning Zone (EPZ) boundary 100 meters from the Reactor Building as a result of this accident was calculated to be 34.8 mRem whole body and lung dose for a two hour period. The radiological consequences are well within the 25 Rem whole body and 300 Rem to any specific organ guidelines established in 10 CFR 100.

D. Occupational Radiation Exposure from Accidents

Additional concerns also exist relating to personnel exposure as a result of accidents and routine operations, contamination of concrete with tritiated water, and for disposal of tritiated water. An evaluation of occupational radiation exposures for three accident scenarios involving the PCRV Shield Water System were made. These accident scenarios were:

Scenario 1: A worker falls from the work platform into the flooded PCRV.

Scenario 2: A worker is exposed to the tritium-contaminated water in the Reactor Building sump following a catastrophic rupture of the PCRV Shield Water System.

Scenario 3: A worker is exposed to a small spill of tritiated water as a result of a routine maintenance mishap.

Each scenario was evaluated for two tritium inventories in the PCRV. These are the maximum theoretical value (100,000 Curies) and the expected value (500 Curies). The assumed Co-60 water inventory of 1000 Curies in both cases is twice the amount of Co-60 expected to enter the water. No credit is taken for removal by the demineralizer systems to maintain the Co-60 level at a small fraction of this value.

Accident Scenario 1: Worker Falls into Flooded PCRV

In this scenario, a worker falls through the work platform access openings into the flooded PCRV. Personnel exposures were evaluated based on the assumption that the worker is rescued and dried within: (1) 10 or (2) 60 minutes of the fall. The Reactor Building and work platform ventilation systems are assumed inoperable at the time of the accident.

For this accident scenario, individual exposures due to tritium are assumed to occur as a result of (1) inhalation of air (86°F at 70% relative humidity); (2) absorption through the skin as a result of 100% wetting; and (3) ingestion of 10 ml of water. Exposure due to Co-60 is assumed to occur as a result of (1) external irradiation from the water itself, and (2) ingestion of 10 ml of water.

For the Scenario 1 accident involving the maximum theoretical value (100,000 Curie case), the following exposures were determined for the 10 minute and 60 minute immersion cases, respectively:

	10 minute exposure (Rem)		60 minute exposure (Rem)	
Route	tritium	Co-60	tritium	Co-60
Inhalation	0.01563	---	0.0937	---
Wetted Skin	0.02253	---	0.0981	---
Swallowing	0.03900	0.198	0.0390	0.198
External	---	0.734	---	4.402
TOTAL (Whole Body)	1.009 Rem		4.832 Rem	

For the Scenario 1 accident involving the expected value (500 Curie), the corresponding totals are 0.932 Rem (WBE) for the 10 minute exposure and 4.602 Rem (WBE) for the 60 minute exposure.

Accident Scenario 2: Worker exposed at the Reactor Building Sump after a postulated Loss of PCRV Shield Water.

This exposure is postulated to occur following complete loss of all water in the PCRV water to the Reactor Building sump. An emergency worker is sent to the area of the Reactor Building sump containing the entire contents of the PCRV and remains in the vicinity of the flooded sump area. The exposure is due both to direct shine (beta and gamma from the water), as well as inhalation of the tritiated water vapor. The dominant contributor to exposure is Co-60.

For the Scenario 2 accident involving maximum theoretical values (100,000 Curies), the worker would be exposed to an external dose rate of 2.2 Rem/hour and the worker would receive an internal exposure of 0.097 Rem due to inhalation of tritiated water vapor. The total 1-hour exposure would be 2.3 Rem (WBE) for the maximum (100,000 Curies) tritium inventory scenario.

For the Scenario 2 accident involving expected values (500 Curies), the total exposure would be approximately 2.2 Rem external and 4.85 E(-4) Rem internal exposure due to tritium inhalation, for a total of 2.2 Rem (WBE).

Accident Scenario 3: Small Spill of Tritiated Water

In this accident scenario, it is postulated that a worker stands 1 foot from a two gallon spill of PCRV shield water (from a clarifying pump). It is assumed that the worker becomes contaminated with 1 E(-5) (0.001%) of the spill [12], resulting in exposure from (1) direct (external) exposure from the spill; (2) internal Committed Effective Dose Equivalent (50 yr CEDE) due to Co-60; and (3) internal Committed Effective Dose Equivalent (50 yr CEDE) from tritium. As noted for Accident Scenario 2, the dominant contributor to exposure is Co-60.

For the Scenario 3 accident involving maximum theoretical values (100,000 Curies), the worker would receive an external exposure of 87.4 mRem and an internal exposure of 10.6 mRem, primarily due to the Co-60. The total exposure would be

98 mRem (WBE) for the maximum (100,000 Curies) tritium inventory scenario.

For the Scenario 3 accident involving expected values (500 Curies), the total exposure is approximately 97.7 mRem (WBE).

E. Accident Scenario Conclusions:

Since these exposures are postulated to occur as a result of accidents, the occupational exposure limits of 10 CFR 20 are not applicable. However, none of the accident scenarios identified above will result in life-threatening exposures to workers, even with the most conservative assumptions as to radionuclide concentrations in the water. Fort St. Vrain decommissioning operating procedures will minimize the likelihood of these accidents.

VI. EVALUATION OF TRITIATED CONCRETE AT THE AMES LABORATORY RESEARCH REACTOR

Based on the decommissioning experience gained from the Ames Laboratory Research Reactor (ALRR), the NRC identified a concern for potential contamination of large volumes of concrete with tritiated water. The ALRR operated for about 12 years and had a tritium concentration in the water of 1.8 Ci/liter (1800 μCi/cc) at shutdown [13,14]. This represents a much longer potential exposure time and a much higher tritium concentration than the maximum that could exist at Fort St. Vrain. Thus the analysis described below predicts an impact of worst case concrete contamination of only a small fraction of the Ames case.

In order to assess the effect on Fort St. Vrain decommissioning of spills of tritiated water on concrete, the worst case of a total release of the water in the PCRV shield water system was considered. This release was postulated to occur with the water containing the maximum possible concentration of tritium (62.4 μCi/cc) permitted by the Decommissioning Technical Specifications [15]. Such a release is very improbable, but could be postulated to occur as a result of a major break in the water system. It was further assumed that the water would enter the sump and remain there for one month until repairs could be effected and the tritiated water pumped back into the PCRV.

The concrete in the lower portion of the Fort St. Vrain Reactor Building sump has been exposed to water during operations and will continue to be exposed during decommissioning. The sump can be periodically wetted down during decommissioning as necessary to prevent dryout and inhibit possible diffusion into dry concrete. Therefore, the concrete can be considered to be saturated with water. The primary mechanism for tritium to enter the concrete as a result of exposure to tritiated water will be by diffusion in accordance with Fick's law of diffusion. Diffusion coefficients for water into concrete vary with parameters such as aggregate size, porosity, and hydrostatic pressure, and therefore only an approximate analysis is possible. The values of the diffusion coefficient for unpainted concrete were measured at about 1.0 E(-5) cm^2/s [16] for small samples. Another set of measurements was made on an actual concrete wall and floor of a heavy water research reactor [17]. These measurements gave a much lower value of 3 E(-7) cm^2/s. Using the conservative higher value, calculations were performed to estimate the tritium distribution in the concrete as a function of time. It was assumed that the surface of the concrete was washed with clean water after removal of the tritiated water, and that the surface was kept wet to allow migration of the tritium back out of the concrete by exchange with the surface moisture. Results of the diffusion calculations are shown in Figures 38-6 and 38-7. These calculations indicate that one year after the exposure, the tritium concentration in the water of hydration throughout the concrete will be less than 2% of the 62.4 μCi/cc maximum. The surface concentration will be even lower and the tritium in the top centimeter of concrete will average less than 0.015 μCi/cc.

Based on the diffusion analysis a maximum concentration of 0.02 times 62.4 μCi/cc or 1.25 μCi/cc of tritium in the water in the concrete will remain. For typical concrete density and water fraction, this results in a maximum of 0.11 μCi/gm. Evaluation of the impact of exposures to such residual radioactive contamination is contained in the report "Residual Radioactive Contamination from Decommissioning" [18]. This report contains the results of analyses to support the technical basis for translating contamination levels to annual dose. For tritium in concrete, the worst case exposures are concluded to occur as a result of the "building renovation scenario". The value derived for tritium is 2.9 E(-7) total effective dose equivalent (TEDE) in mRem per pCi/gm for an exposure of 500 hours (Table 3.1 of [18]). Therefore, based on 500 hours exposure, the Fort St. Vrain exposure for this activity would be 0.032 mRem, which is well below the allowable maximum of 10 mRem/yr.

It is therefore concluded that exposure of the concrete to tritiated water at or below the limit of 62.4 μCi/cc will not result in the necessity to remove any contaminated concrete, nor will it create any problems in the decommissioning schedule. The case considered above will bound cases of smaller spills that can be cleaned up in shorter times and that will expose more limited volumes of concrete.

VII. CONCLUSIONS

The dominant source of tritium that will remain in the graphite after defueling was generated during plant operations by neutron capture by Li-6 impurities in the HLM graphite. Based on the maximum specification for lithium impurity in the HLM graphite and the actual power history, the activation analysis predicts that the theoretical maximum amount of tritium that can be in the PCRV Shield Water System is approximately 100,000 Curies.

Testing performed by the British has shown that a total of 0.5% of the contained tritium leaches into the water after 100 days. In addition, tritium leach testing performed by the French on unimpregnated graphite in distilled water has also shown that the maximum amount of tritium that leaches into the water after 90 days is 0.3%. The 500 curies that are estimated to be leached into the PCRV water is based on British test data that has since been substantiated by independent testing performed by the French.

The preceding evaluations have also shown that even if the entire inventory of tritium (100,000 curies) is released into the PCRV water, the radiological consequences to the public and decommissioning workers from normal and postulated accidents scenarios would not exceed the guidelines established in 10 CFR 100. Finally, this tritiated water could be disposed of by solidification if the preferred release method of effluent discharge release is determined to be not viable. Allowing for this extreme case, decommissioning can proceed and will be accomplished within the decommissioning cost estimate previously submitted to the NRC. In addition, with considerations for the worst credible accident and this extreme case, decommissioning will also be accomplished without undue risk to the safety of the public.

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TABLE 38-1
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 (g/cc)						
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	3.395 E7	8 E7	4.5 E7	376*	* - Actual sample size; 2 samples tested
Total Tritium Content (Ci)	82,558	11,532	3500	17		
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 ($n/cm^2/sec$)	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	He	He	He	He	CO_2	

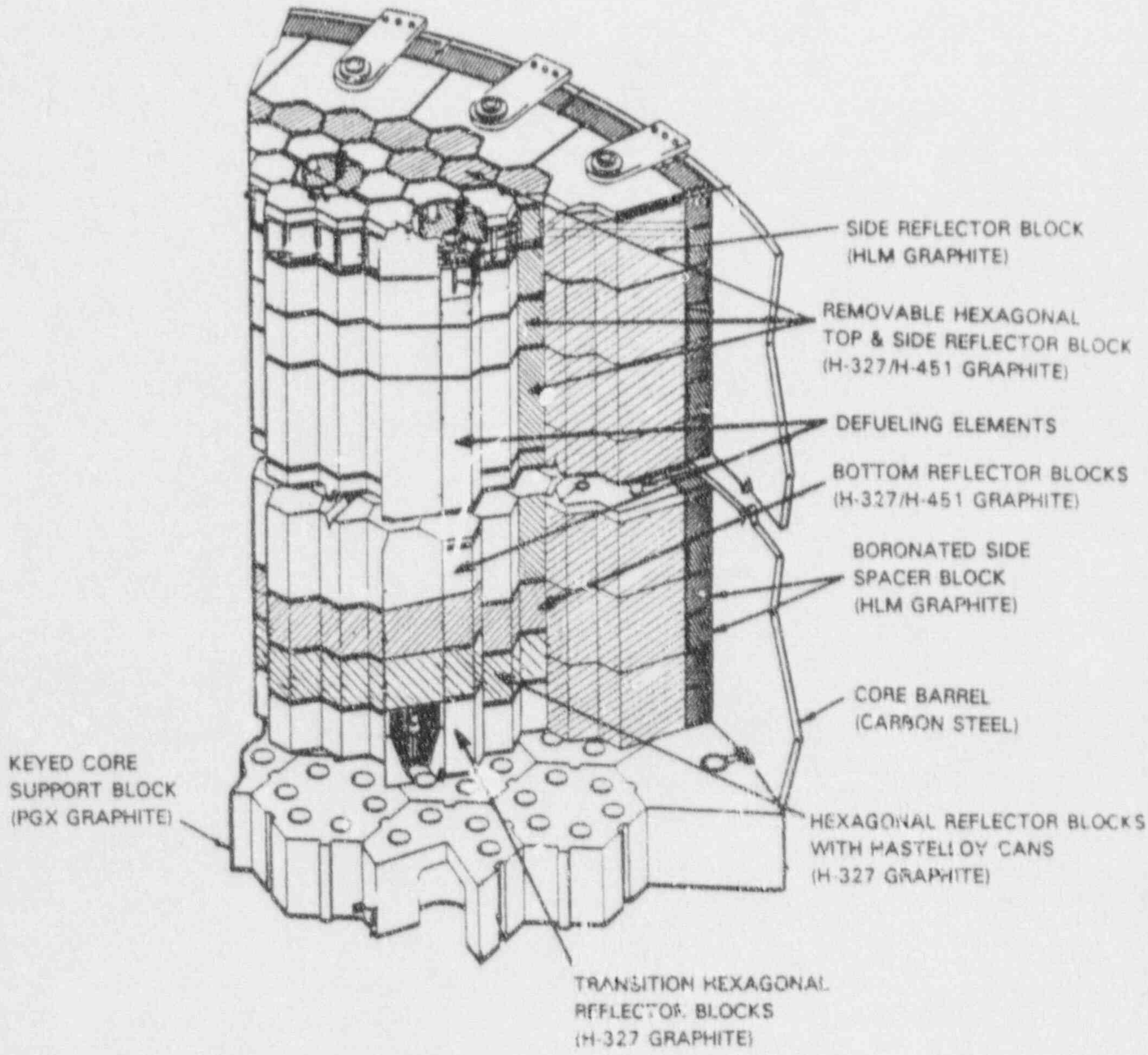

Figure 38-1 Core Graphite Components - Elevation View

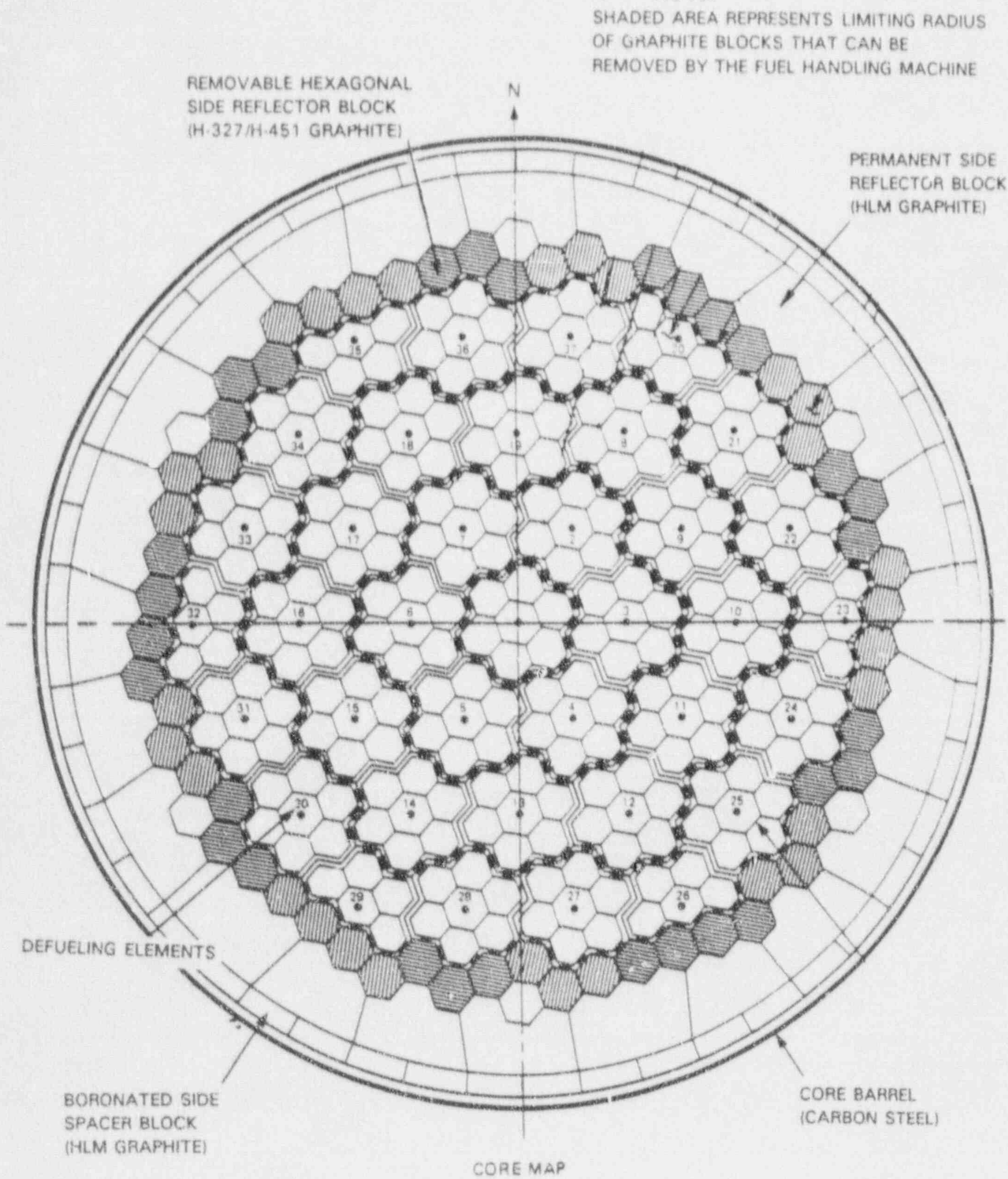

Figure 38-2 Core Graphite Components - Plan View

[Figure 6.2 of Reference 3]

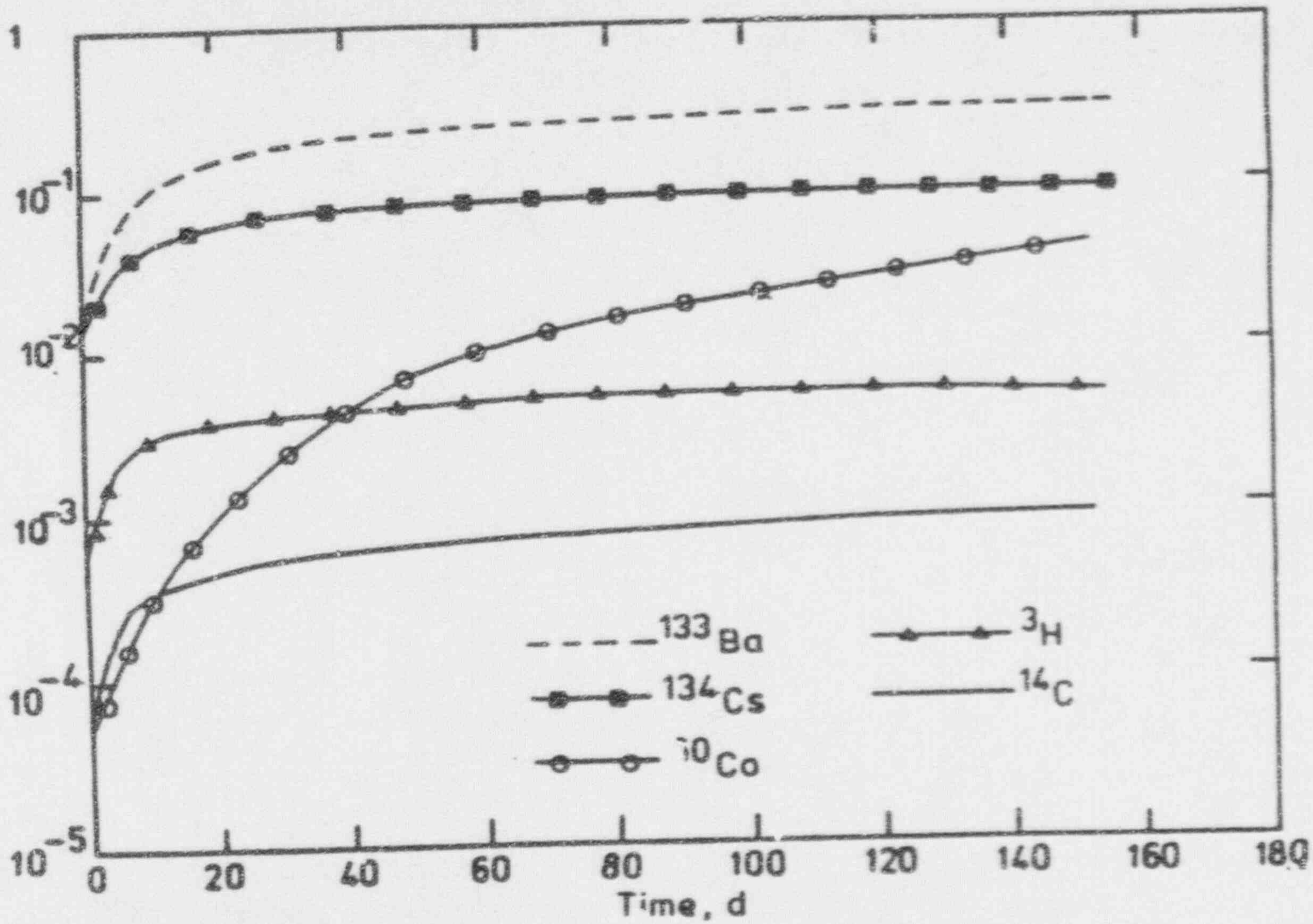

Figure 38-3 Cumulative Fraction of Activity Leached in Demineralized Water (1 bar, 25°C)

[Figure 6.6 of Reference 3]

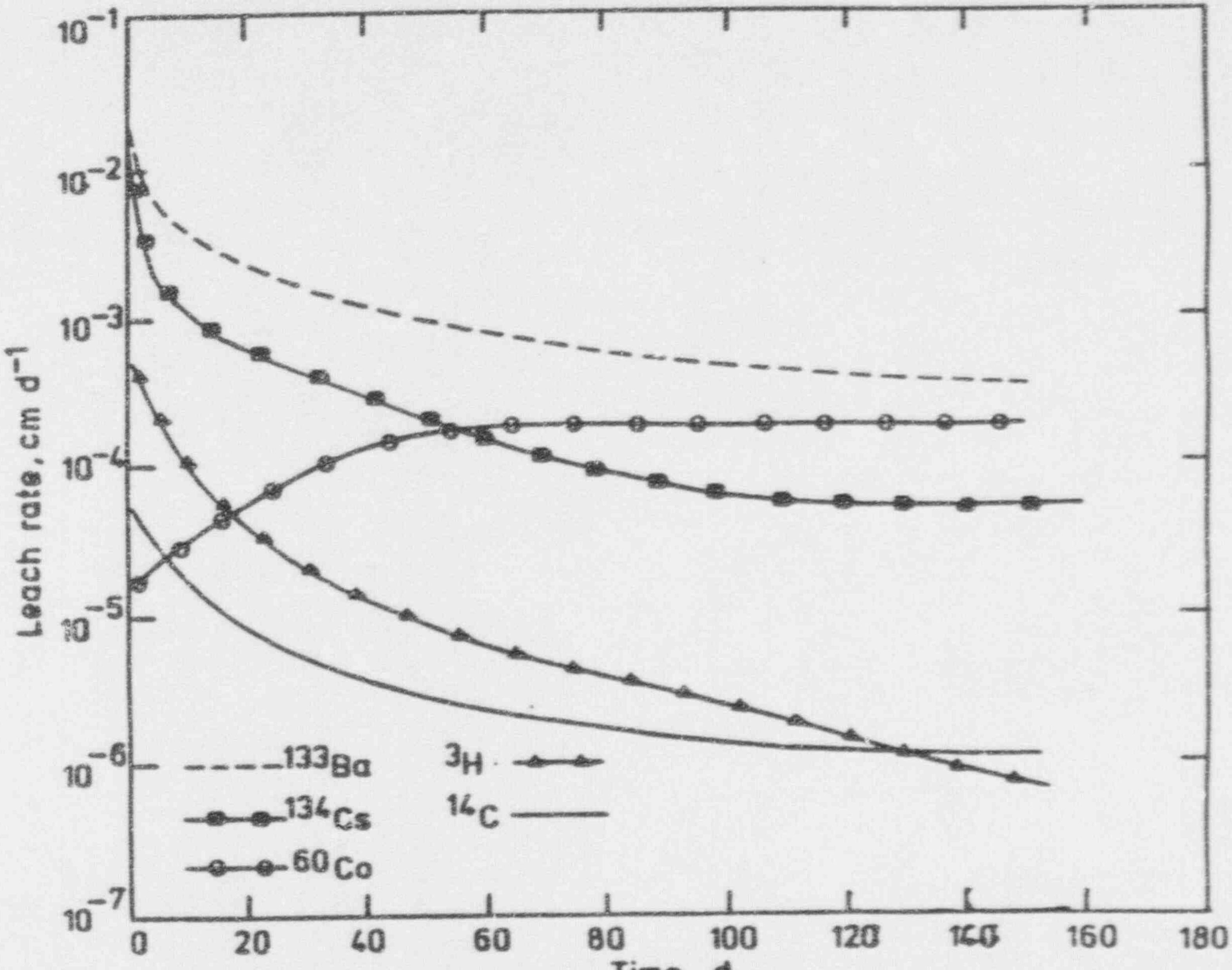

Figure 38-4 Leach Rate Curves in Demineralized Water (1 bar, 25°C)

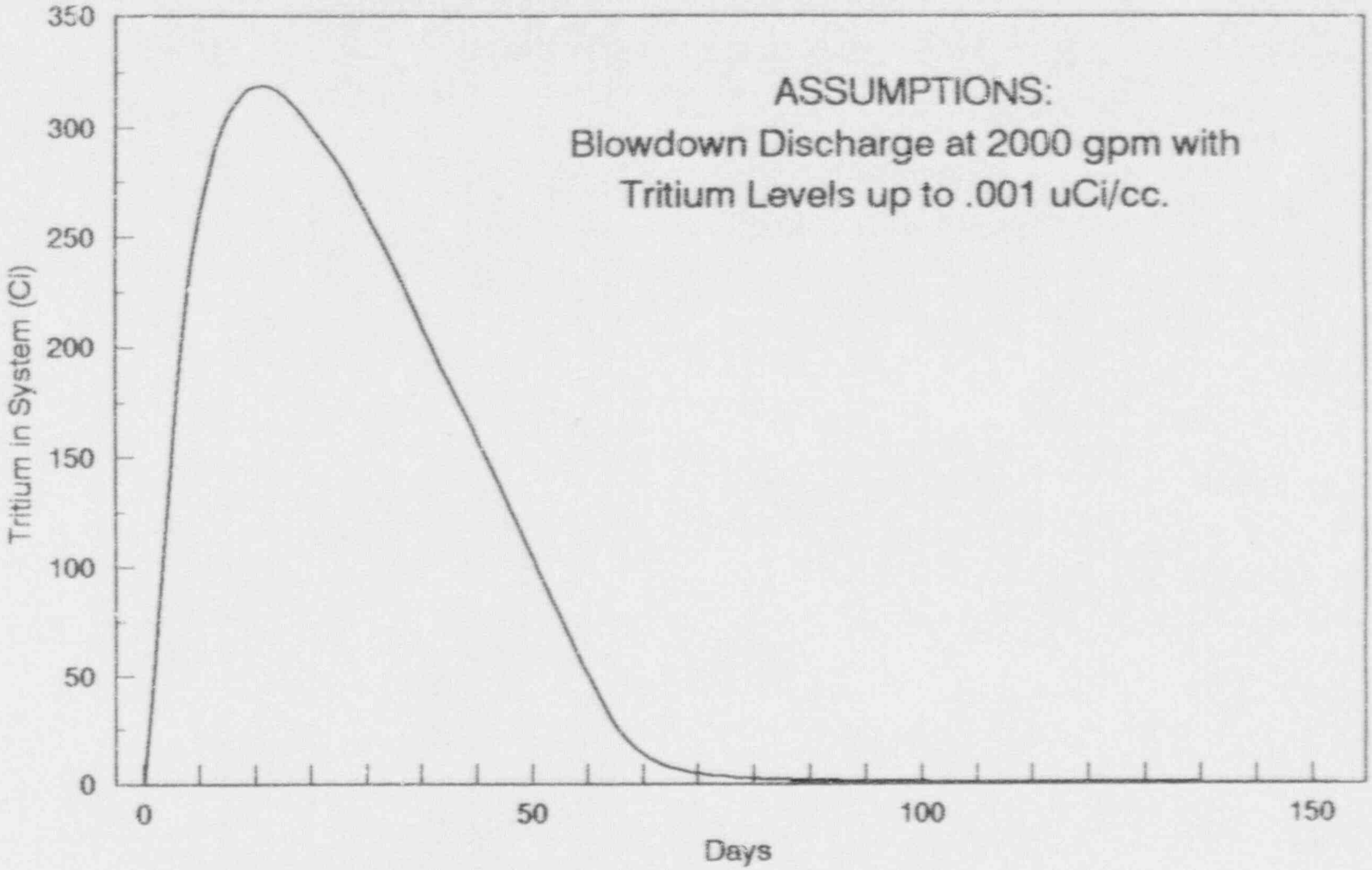

Figure 38-5 Tritium in the PCRV System

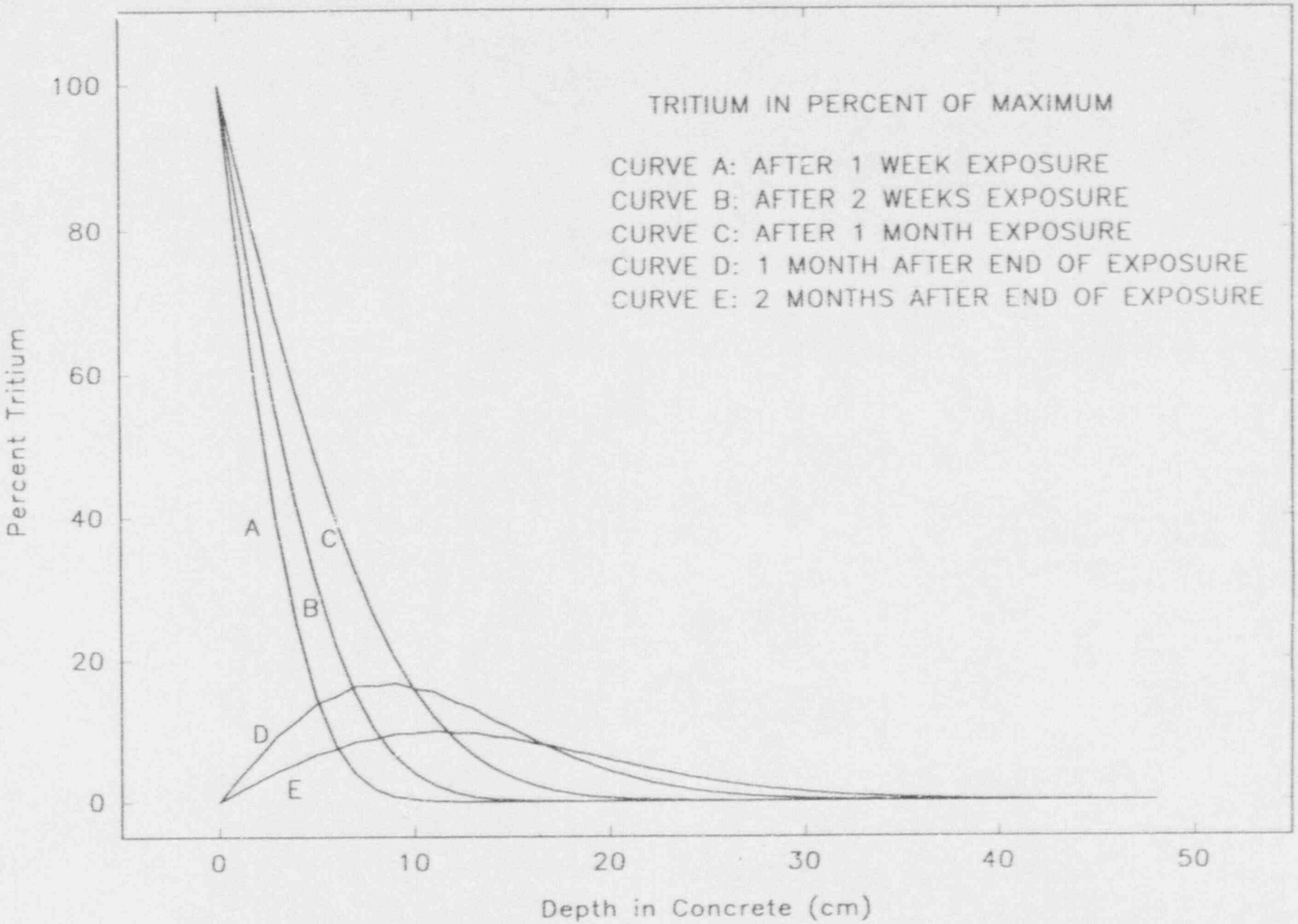

Figure 38-6 Tritium Distribution

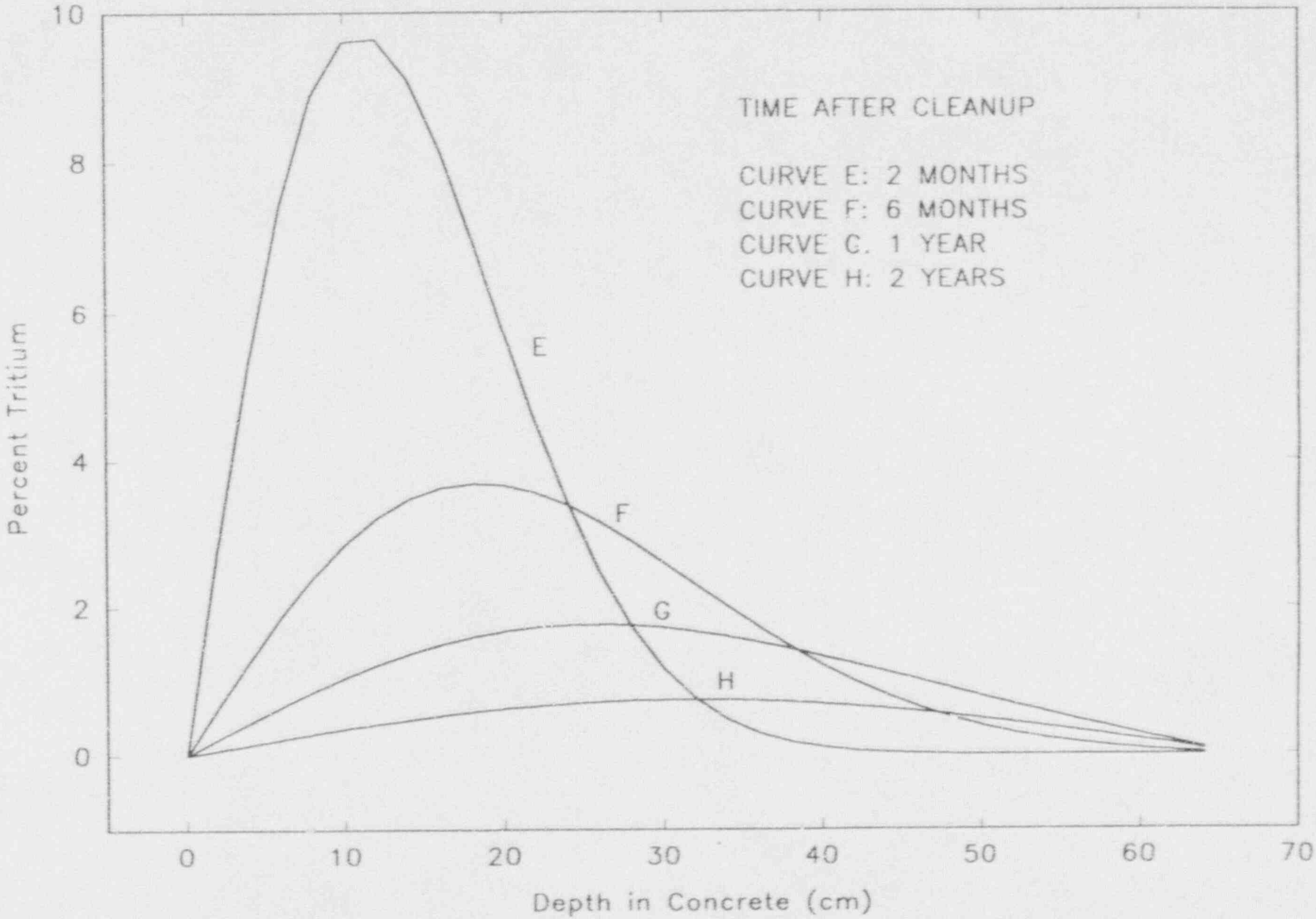

Figure 38-7 Tritium Distribution