Public Service

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February 18, 1992
Fort St. Vrain
Unit No. 1
P-92064

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 - QUESTIONS NO. 9, 12, & 14**

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 the 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 submittal provides specific PSC responses to NRC Questions No. 9 (PCRV Dismantlement Activities), No. 12 (PCRV Top Head Concrete and Liner Removal), and No. 14 (Core Barrel Removal).

If you have any questions related to the contents of this letter, please contact Mr. M. H. Holmes at (303) 620-1701.

Sincerely yours,

D. W. Warembourg

D. W. Warembourg
Manager, Nuclear Operations

A001 1/1

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-92064

PSC RESPONSE
TO NRC QUESTIONS NO. 9, 12, & 14

FROM THE NRC RAI
DATED AUGUST 30, 1991

NRC Question No. 9 (Section 2.3.3.1; PCRV Dismantlement Activities)

While PSC has addressed some of our concerns, PSC must s' 'mit the final dismantling methods and a supporting safety analysis for NRC review. Altern ely, provide description and safety analysis for potential options that may be used. Include valuations of and methods to minimize personnel exposure in your safety analysis.

NRC Question No. 12 (Section 2.3.3.7; PCRV Top Head Concrete and Liner Removal)

"Provide a safety analysis of procedures being developed to minimize personnel exposure. What maximum radiation levels are expected at worker locations during removal of radioactive components? Neither the April 26, 1991 response nor the July 1 revision to the Decommissioning Plan provide this information."

NRC Question No. 14 (Section 2.3.3.9; Core Barrel Removal)

"The July 1, 1991 revision selects a thermal cutting method for core barrel removal. Provide procedures and related safety analysis for minimization of occupational exposure to personnel."

PSC Response:

In PSC's response to NRC RAI Question No. 9 in the PSC letter dated December 6, 1991 [1], PSC and the Westinghouse team identified nine specific dismantlement activities to be performed inside the PCRV for which detailed descriptions and safety analyses would be prepared. PSC committed to provide detailed evaluations for the following activities:

- Water Cleanup and Clarification System (NRC RAI Questions 11 and 38)
- Removal of Top Head Concrete and Liner (Question No. 12)
- Removal of Hex Blocks with Hastelloy Cans (Question No. 13)
- Removal of Core Barrel and Keys (Question No. 14)
- Removal of Hex Blocks without Hastelloy Cans
- Removal of Side Spacer Blocks with Boron Pins
- Removal of Large Permanent Side Reflectors
- Removal of Core Support Floor
- Removal of Steam Generator Primary Assemblies

These activities were selected on the basis of criteria discussed with the NRC: (1) potential for high radiation levels or high occupational exposures, or (2) unique evolutions with a high degree of difficulty. Per conversations with the NRC staff, both the criteria and the list of proposed evolutions for detailed evaluation appear to be reasonable.

A detailed evaluation was provided for the PCRV Water Cleanup and Clarification System (Shield Water System) in PSC's response dated January 9, 1991 [2]. A detailed evaluation of the Removal of Hexagonal Blocks with Hastelloy Cans was provided in PSC's response dated December 6, 1991 [1]. This submittal provides PSC's remaining detailed evaluations of the following dismantlement activities to be performed inside the PCRV in response to NRC RAI Question No. 9, as well as responses to the specific concerns identified above in NRC RAI Questions No. 12 and 14:

RAI Response Section	Description
9.1	Removal of the PCRV Top Head Concrete and Liner
9.2	Removal of Core Graphite Blocks
	- Hex Blocks without Hastelloy Cans
	- Large Permanent Side Reflectors
	- Side Spacer Blocks with Boron Pins
9.3	Removal of Core Barrel and Keys
9.4	Removal of the Core Support Floor
9.5	Removal of the Steam Generator Primary Assemblies

Attachment to P-92064
February 18, 1992

9.1 PCRV TOP HEAD CONCRETE REMOVAL

I. GENERAL DESCRIPTION - TOP HEAD CONCRETE AND LINER

The dismantling approach described in Section 2.3 of the Proposed Decommissioning Plan [3] requires the removal of the PCRV top head to gain access to the reactor core components. This activity is unique due to its safety concerns related to handling of heavy components, potential for the spread of contamination and occupational radiation exposure. Therefore, the removal of the hexagonal concrete prism (37½ feet across the flats, 15½ feet thick, and approximately 1500 tons) is discussed in this evaluation. A discussion of the removal of the 31-foot diameter section of PCRV liner plate, insulation and concrete that remains after the hexagonal prism is removed and that forms the top of the reactor cavity is also provided.

As described in Section 2.2.2 of the PDP, the 15½ foot-thick PCRV top head is constructed around a 3/4-inch low carbon steel liner which forms the internal PCRV cavity. This liner is anchored to the concrete at frequent intervals. The prestressing tendons are located in conduits made of carbon steel tubing and embedded in the PCRV concrete. The cutting of the top head concrete will be performed by a diamond wire cutting process. The proposed dismantling process is described in Section 2.3.3.7 of the PDP (excerpts of Section 2.3.3.7 of the PDP are attached for information and as a reference). This process has been developed to limit the weight of the top head segments to be less than the rated capacity (170 tons) of the existing Reactor Building crane. Specially engineered lifting attachments will be used to safely handle the heavy components, consistent with the requirements of 29 CFR 1926 and referenced ANSI standards.

II. DESCRIPTION OF TOP HEAD AND LINER REMOVAL PROCEDURE

A. Prerequisites For Concrete and Liner Removal

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

1. Detensioning and removal of selected tendons as discussed in the April 26, 1991 response to NRC RAI Question No. 7 [4] and in

Section 2.3.3.5.1 of the PDP [3].

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

It should be noted that removal of the highly radioactive components with the FHM is the preferred method to maintain personnel exposure ALARA. However, use of the FHM for this purpose is dependent on its operability, and its availability has not been relied upon as the basis for removal of these components.

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

B. Removal Activities

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

- Removal of 10 large sections of PCRV top head concrete.
- Removal of a final 2-3 inch thick layer of activated concrete and the top of the carbon steel PCRV liner.

The first phase, removal of the ten large sections of top head concrete, is described in Section 2.3.3.7 of the PDP and the cutting sequence is shown in PDP Figures 2.3-6 through 2.3-9. The concrete sections will be cut using the diamond wire cutting process. The diamond wire cutting process will utilize appropriate radiological engineering controls to contain the cutting slurry and control airborne radioactivity. The diamond wire saw uses water as a coolant and lubricant for the cutting process. However, the coolant water is independent of the PCRV Shield Water System. A water collection system will collect the cutting slurry, decant the slurry and recycle the water. In addition, airborne and loose surface contamination control will be achieved by containing the diamond wire path and drive units(s) in a containment tent(s) served by HEPA ventilation.

The first phase activity will account for the majority of the effort that will be spent to remove the top head concrete and liner. Occupational exposure is expected to be negligible during the core boring and concrete cutting operations on the top head due to the relatively low radiation fields external to the PCRV.

The second phase, removal of the final thin (2-3 inch thick) concrete layer and the top of the PCRV carbon steel liner, is also discussed in Section 2.3.3.7 of the PDP and is depicted in PDP Figure 2.3-9. In this activity, the PCRV liner plate and adjoining concrete will be sectioned using a combination of circular water-cooled concrete saws, a hydraulic concrete breaker and a thermal torch (oxy-acetylene or oxy-lance). The hydraulic breaker will be remotely operated and therefore is not expected to involve significant personnel exposure. The concrete saw is expected to be a hydraulic-driven, track-mounted, manually-operated saw. Thermal cutting will be performed using a long-handled torch (3 to 4 feet).

The concrete saws will be used to score the cut lines around the circumference and for segmenting cuts. The hydraulic breaker will then chip away the concrete to expose the carbon steel liner. The thermal torch (oxy-acetylene or oxy-lance) will then be used to cut through the liner, insulation and cover plate to free sections for removal. The layout and sequencing of cuts will take into consideration the structural stability during the disassembly process. The concrete/liner/insulation disk, after possible further segmentation, will be removed to a waste processing area for further segmenting (if necessary for disposal), segregation and preparation for disposal. Radiological engineering controls will be utilized to control the dust, smoke and potential airborne contamination related to this process. A containment will be constructed across the top of the PCRV, and HEPA ventilation will be provided during these operations. Personnel required to work within the containments will be required to wear the appropriate protective clothing and respiratory protection per ALARA review and RWP requirements.

Of the two phases, the second phase represents the greatest potential for personnel exposure. The PCRV liner plate and the few remaining inches of activated concrete will be uncovered as the final segments of the top head are removed. The PCRV liner plate is estimated to have radiation levels of up to 600 mRem/hr on contact. This estimated exposure rate is a conservative interpretation of information provided in the activation analysis for the bottom side of the cover plate, insulation, liner plate and activated concrete. As indicated in Table 9.1-1, workers on the top side of this composite disk are expected to experience a lower exposure rate since the activated concrete will provide shielding from the more highly activated ferrous materials of the liner plate. Shielding for the workers will be utilized as appropriate for the close operations such as installing the saw tracks, operation of the saw and thermal cutting.

The procedures and controls to be used to maintain a safe operation and minimize occupational exposure have been factored into the removal of the liner plate. These include the following:

1. Access to the exposed PCRV liner plate will be controlled by Radiation Protection personnel to ensure positive control and maintain personnel exposure ALARA.
2. The dose rates on the PCRV liner plate will be determined by Radiation Protection personnel as it is uncovered, so that appropriate controls can be instituted.
3. Temporary shielding will be used as necessary to reduce work area dose rates and maintain personnel exposures ALARA.

4. Remotely operated equipment will be used, when necessary, to minimize the time workers spend directly over the liner.
5. HEPA ventilation and containments will be utilized to control dust, smoke, and potential airborne contaminants during concrete and thermal cutting operations.
6. Protective clothing, respiratory protection and dosimetry required for the various phases of the liner removal will be evaluated and prescribed, per approved Radiation Protection procedures, during the pre-job ALARA review and Radiation Work Permit (RWP) generation.

III. RADIOLOGICAL CONSIDERATIONS

A. Occupational Radiation Exposures

The person-hour estimate for top head cutting and liner removal, WBS Element 2.3.2.7, includes 21,660 man hours spent on a multi-shift operation over a period of approximately 5 months. It is estimated that 10,830 of those hours will involve personnel exposure to radiation. The total occupational exposure for the top head cutting and liner removal is estimated at 11.91 person-rem (See Table 9.1-1). As indicated in this table, over half of that total will be expended on the liner removal operation (7.5 person-rem).

B. Offsite Exposures From Accidents

PDP Section 3.4.3, "Dropping of Contaminated Concrete Rubble Accident", postulates the dropping of the concrete rubble from the concrete disc of the top head. This accident scenario assumes that radioactivity is released from the drop of a transport container that contains the rubble removed from the six inches of concrete just above the PCRV top head liner. For this accident scenario it is assumed that the 6-inch concrete wafer adjacent to the PCRV top head liner contains 5.9 Curies of Eu-154, 3.93 Curies of tritium, 32.8 Curies of Fe-55 and 1.43 Curies of Co-60. As analyzed, 10 percent (or approximately 7,500 lbs) of the concrete in the 6-inch concrete wafer has been rubblized and is involved in the accident. No credit was taken for particulate filtration by the Reactor Building ventilation system.

The whole body and bone doses to an individual standing at a point on the Emergency Planning Zone (EPZ) 100 meters from the Reactor Building were calculated to be 4.92 mRem and 54.7 mRem, respectively. 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" [6]. Dose conversion factors were taken from NUREG-0172 [7]. This analysis determined that the radiation exposure to the general public as a result of the dropping of contaminated concrete rubble is very low. 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. The radiological consequences are also 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 [8].

As presented in Table 3.4-3 of the PDP, the activation analysis indicates that the highest concentration of radioactivity in the PCRV concrete is in the 6-inch increment of the PCRV top head immediately above the top head liner. Therefore, it can be concluded that the worst accident involving the dropping of contaminated concrete rubble has been postulated and analyzed in Section 3.4.3 of the PDP.

C. Radioactive Waste Generated

Figure 3.2-1 of the Decommissioning Cost Estimate provides a tabular listing of the estimated radioactive waste disposal volume for each WBS element. The estimated disposal volume for the contaminated top head concrete is 3744 cubic feet and the burial classification is "Class A".

IV. SAFETY ANALYSIS CONCLUSIONS

During the PCRV top head removal activities, the radiological hazards will be monitored and evaluated on a routine basis. All work activities associated with the removal of the top head concrete and liner will incorporate effective radiological controls to maintain occupational radiation exposures within regulatory limits and as low as reasonably achievable (ALARA), consistent with dose-limiting provisions of 10 CFR 20, as well as both Regulatory Guides 8.8 [9] and 8.10 [10]. All workers will be provided instructions in radiation protection concepts commensurate with the radiological hazards that they will encounter during the removal of the top head concrete and liner, including instructions concerning actions required during unusual conditions.

It is estimated that the removal of the top head concrete and liner will result in a total occupational radiological exposure of 11.91 person-Rem. Personnel occupational exposures will not be in excess of allowable 10 CFR 20 limits for occupational radiation exposure. Furthermore, postulated accidents involving the top head concrete and liner will result in offsite radiological consequences that are a small fraction of the guidelines established in both 10 CFR 100 and EPA Protective Action Guidelines [8]. It is therefore concluded that the activities associated with the removal and disposal of the top head concrete and liner will not pose an undue risk to the health and safety of the general public nor to the occupationally exposed decommissioning workers.

TABLE 9.1-1
OCCUPATIONAL RADIATION EXPOSURE ESTIMATE
TOP HEAD CONCRETE AND LINER REMOVAL

OPERATION	NO. OF WORKERS	EFFECTIVE EXPOSURE RATE (mR/hr)	TASK DURATION (hrs)	WORKER EXPOSURE TIME (hrs)	ESTIMATED EXPOSURE (person-Rem)
Core bore, install seals and plugs, and setup diamond wire equipment	5	0.25	265	1325	0.33
Concrete cutting and rigging	8	0.44	1160	9280	4.08
Liner removal					
o Shielding, installation and equipment setup	2	50.0	12.5	25	1.25
o Concrete saw operation	2	25.0	75.0	150	3.75
o Thermal torch Operations	2	50.0	25.0	50	2.50
TOTALS				10,830	11.91

9.2 GRAPHITE BLOCK REMOVAL

I. GENERAL DESCRIPTION - GRAPHITE BLOCKS

The following types of graphite blocks will be removed from the PCRV during decommissioning activities:

GRAPHITE BLOCK TYPE	RADIATION LEVEL (on contact without shielding)
1. Hex Reflector Blocks with Hastelloy Cans	300 R/hr
2. Hex Reflector Blocks without Hastelloy Cans	500 mR/hr
3. Large Permanent Side Reflectors	<30 R/hr
4. Side Spacer Blocks	
(a) with Boronated Pins	30 R/hr
(b) without Boronated Pins	<3 R/hr
(c) Boronated Pins (removed from block)	60 R/hr

The PSC response to NRC RAI Question No. 13 previously submitted to the NRC [1] addressed removal of the hex reflector blocks with hastelloy cans. Therefore, this discussion addresses the removal of hex reflector blocks without hastelloy cans, removal of side spacers with boronated pins, and removal of large permanent side reflectors. A general discussion is provided for the activities that are common to the removal of all types of graphite blocks. The activities that are specific for one type of block are discussed at the end of the general discussion.

II. DESCRIPTION OF THE GRAPHITE BLOCK REMOVAL PROCEDURE

A. Prerequisites for Graphite Block Removal

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

1. Flooding the PCRV with shield water as described in Section 2.3.3.6 of the PDP [3].
2. Removal of the top head concrete and liner as discussed in Section 2.3.3.7 of the PDP and in Section 9.1 of this response.
3. Installing the PCRV work platform.
4. Radiological survey of the work area and installation of temporary shielding if necessary for ALARA purposes.

B. General Arrangement of Work Area for Graphite Block Removal

The arrangement of the work area that will be typically used for removal of all types of graphite blocks was provided in response to NRC RAI Question No. 13 (Figure 13-4) and is provided again in Figure 9.2-1 for reference. The PCRV will have been filled with shield water to a level approximately 4 feet above the graphite blocks, but below the top of the PCRV liner. The Work Platform will have been installed on the ledge at the bottom of the hex opening in the PCRV. The Work Platform will be designed with the capability of rotating to provide access to all areas of the core. It will have three access openings to allow insertion and removal of tools and components. This will permit up to three operations to proceed in parallel. A floor will be installed between the platform and the walls of the PCRV at the level of the Work Platform. There will be three jib cranes installed on the refueling floor level to service the access openings in the platform. The Reactor Building crane will also be available to service the platform and the remainder of the refueling floor area. A ventilation system will be installed to provide control of airborne contamination, including tritium. Air will be drawn from the refueling floor to the Work Platform, down through the access openings in the platform, and then exhausted to the Reactor Building Ventilation System (System 73). The discharges from the Reactor Building Ventilation System will be monitored in accordance with the FSV Offsite Dose Calculation Manual (ODCM) [11]. The airflow from uncontaminated areas to contaminated areas through the Work Platform will minimize personnel exposure to airborne contamination.

The area on the Work Platform will be quite large, approximately 43 feet across the corners of the hexagonal opening. This will provide the capability to move personnel on the Work Platform to a considerable distance away from an operation if a significant radiation field is encountered.

B. General Graphite Block Removal Sequence

The sequence of operations for removal of all graphite blocks are similar to those described in Reference 1 for the removal of the hastelloy can hex reflector blocks. The following is the general sequence of operations for removal of the graphite blocks:

1.	Removing	5.	Unloading
2.	Staging	6.	Dewatering
3.	Loading	7.	Drying
4.	Transferring	8.	Packaging

Since this general sequence of operations will be used for the removal of all types of graphite blocks, the discussion of the eight steps provided below are applicable to the removal of the hex reflectors without hastelloy cans, the large permanent reflector blocks, and side spacer blocks with boronated pins.

(1) Removing:

The blocks will be lifted from their position in the PCRV core area, and placed in an intermediate staging area that is below the surface of the water (see Figures 9.2-2 and 9.2-3). This will be accomplished using remotely engaged long handled tools (LHT's) attached to an overhead crane that is operated by personnel on the Work Platform. The workers will be working from the Work Platform that will be installed over the flooded PCRV. The tool for handling the graphite blocks (except the side spacers) will be an expanding collet type similar to that used in the Fuel Handling Machine (FHM). The end of the tool will be inserted into the reverse counterbored hole in the top of the block with the end of the tool retracted. The end of the tool will then be expanded in the larger diameter in the lower portion of the hole and the block will be lifted utilizing an overhead crane. The side spacers will be handled by attaching a lifting bail to the top of the block using the existing threaded holes in the graphite block.

(2) Staging:

After removal from the PCRV core area and while still submerged, the blocks will be lifted and placed on an intermediate stand attached to the work platform (see Figures 9.2-2 and 9.2-3). During this operation, the block will remain submerged underwater. The LHT will be disengaged and removed, leaving the block temporarily stored on the stand.

(3) Loading:

A short handling tool with integral shielding bell will then be lowered into position on guide pins as shown in Figure 9.2-3. A grappling tool will be lowered from the inside of the shielding bell, engage the block, and lift the block into the shielding bell. The shielding bell guide pins and the storage stand will provide the necessary alignment for engagement of the tool. The actual raising of the block will be accomplished in a few minutes.

After the block has been loaded into the bell and the shielding bell has been lifted to just above the floor of the Work Platform, a catch pan with absorbent material (see Figure 9.2-4) will be installed under the shielding bell to contain possible drippings of contaminated water during transport to the dryer/shipping liner. The catch pan will be strong enough to retain the block in the shielding bell in the unlikely event that the grappling mechanism should fail. The catch pan will also provide limited shielding at the bottom of the shield bell. However, this shielding will not be sufficient to fully shield and protect the workers on the platform from the indirect scattering that will occur out of the bottom of the shielding bell. Therefore, during loading operations, radiation levels in the immediate vicinity of the shielding bell will be closely monitored and personnel access to the affected area will be limited by administrative procedural controls.

The expected dose rates on the Work Platform, both with and without the shielding bell, are shown in Figure 9.2-5 for the large permanent side reflector block, in Figure 9.2-6 for the hex reflector blocks without hastelloy cans, and in Figure 9.2-7 for the side spacers without boronated pins.

(4) Transfer:

As the shielding bell is moved from the work platform (using a jib crane or the Reactor Building crane) to the dewatering device and to the dryer, nonessential personnel will be required to stay clear of the area to create a clear path for movement of the load.

(5) Unloading:

Unloading of the shielding bell into the dewatering device will be accomplished by removing the catch pan and lowering the block from the shield bell into an opening in the dewatering device. An alignment fixture will be used as necessary to lower the block into the dewatering device.

(6) Dewatering:

Since some of the graphite blocks contain blind holes that will have collected a small amount of water while submerged in the PCRV, it will be necessary to place the blocks in a dewatering device to drain the water from these blind holes. A schematic of the dewatering device is shown in Figure 9.2-8. After the blocks are unloaded from the shielding bell into the dewatering device, the blocks will be rotated (tipped) approximately 90 degrees. The water that is drained from the graphite blocks will drain from the dewatering device back into the PCRV. The routing of the drain line will be such to avoid crud traps or stagnant water. The blocks will then be moved from the dewatering device into the shielding bell, the catch pan installed, and then moved to a block dryer.

(7) Drying:

Dryer units will be set up on the refueling floor. A schematic of the block dryer arrangement, and loading and unloading positions is shown in Figure 9.2-9. The shielding bell will be positioned on alignment pins over the dryer, the catch pan removed, and the block lowered into the dryer. Pneumatic push rods will progressively push the blocks through the dryer. Air will be drawn into the dryer through spring loaded louvers near the block exit end and exhausted near the block entry end to the existing Reactor Building Exhaust System. It is not planned to heat the air for drying.

(8) Packaging:

The graphite blocks will be re-loaded into the shielding bell from the dryer and transferred to the packaging area. The graphite blocks will be discharged into a shielded shipping container (see Figure 9.2-10), using an alignment fixture as necessary to assure placement for efficient use of available space. After the shipping container is filled, the top will be installed.

III. DESCRIPTION OF ACTIVITIES SPECIFIC TO BLOCK TYPE

A. Large Side Reflector Blocks

A typical large side reflector block is shown in Figure 9.2-11. There are 312 large side reflectors, ranging from approximately 522 to 2030 lbs. each, located in layers 1 through 12 around the circumference of the core as shown in Figure 9.2-12. The large side reflector block keys will be removed by unbolting the restraining bolts (or by underwater thermal cutting if required) to allow removal of the large side reflectors and the boronated side spacer blocks. The large side reflector blocks will then be removed and processed using the general steps described above. The large side reflector blocks will be handled using a dual collet tool inserted into the reverse counterbored holes. Sectioning of the large side reflector blocks may be required because of packaging requirements. If sectioning is required, it will be accomplished in the Hot Service Facility (HSF), as shown in Figure 9.2-13, after the blocks have been dried. The blocks will be sectioned as necessary for packaging. The blocks will be transferred into and out of the HSF in a shielding bell with catch pan as appropriate.

B. Hex Reflector Blocks Without Hastelloy Cans

Removal of the bottom, side and top hex reflector blocks without hastelloy cans will be handled in much the same way as the hastelloy can hex reflector blocks. The eight steps in the general sequence remain the same. However, a different shielding bell will be used for removal of the hex reflector blocks without Hastelloy cans. This shielding bell will also have a catch pan with absorbent material that will be installed under the shielding bell to contain possible drippings of contaminated water during transport to the dryer/shipping liner. The hex reflector blocks without hastelloy cans are shown in Figure 9.2-14 and their position in the core is shown in Figure 9.2-12.

C. Side Spacer Blocks With Boronated Pins

The side spacer blocks with boronated pins are shown in Figure 9.2-15. There are 1152 boronated side spacer blocks weighing approximately 100 - 150 lbs each. Their location in the core is shown in Figure 9.2-12 and dimensions of the pins are shown in Figure 9.2-16. The following changes to the eight steps in the general sequence of operations will be used for the removal of the side spacer blocks with boronated pins, due to the dumping of the boronated pins (Figure 9.2-17 is provided to illustrate the steps involved in removing the pins from the side spacer blocks):

(1) Removing

In order to remove the side spacer blocks and dump the boronated pins, a lifting bail will be installed on the top of the side spacer block. These lifting bails will be designed to fit all 19 variations of side spacer blocks and will also provide the means for upending the block to dump the boronated pins and water from the holes (as described in the following paragraphs).

The bail will be positioned on top of the block while it is underwater and will be attached by captured bolts engaging the existing threaded holes in the block. The bolts will be engaged using a long handled torque tool. The bail will be attached to a lifting tool and the block will be moved to a pin dumping station (intermediate stand) that is suspended underwater, under the Work Platform (see Sketch 1 of Figure 9.2-17).

(2) Staging/Dumping

The block will be staged underwater in the pin dumper and the lifting tool will be disengaged. A second tool with integral shielding bell will be reattached to the bail for upending (see Figure 9.2-17, Sketch 2). As the block is upended (Sketches 3 through 5), the pins will fall out onto the intermediate stand. The pins will then be pushed into a hole in the bottom of the intermediate stand and will travel through a chute to a cask liner located beneath the intermediate stand (see Figure 9.2-18). There is sufficient clearance for the pins in the spacer blocks to ensure that they are loose and will drop out of the spacer blocks easily.

The block will then be raised out of the water and into the integral shielding bell and a catch pan attached under the shield bell. The block will be surveyed as it is lifted out of the water and into the bell to confirm that no pins remain in the block. Expected contact radiation levels of the blocks with the pins removed is approximately 3 R/hr, as compared to the radiation levels of blocks containing the pins which are much greater, up to 30 R/hr. Since the block will be removed from the water in the inverted position, it will not be necessary to take it to the dewatering station. The block will be transferred directly to the dryer in the shielding bell.

(3) Drying

The shield bell containing the block will be moved to the dryer and the catch pan will be removed. The shield bell will be aligned on top of the dryer and the block will be lowered to remove the bolts attaching the block to the lifting bail. As noted above, the contact radiation level is expected to be 3 R/hr. The remainder of the operations will be performed the same as outlined in the general sequence of operations.

(4) Boronated Pin Handling

When the pins (60 R/hr on contact) are removed from the block, they will be pushed into a chute and will slide into a shipping cask liner. After a specified number of blocks have had the pins dumped into the liner, the liner will be removed from the PCRV into a shielding bell. The cask liner will have holes in the bottom to allow water to drain out when the liner is removed from the PCRV. A catch pan will be installed, and the shielding bell will be moved and the cask liner transferred to a shipping cask.

IV. RADIOLOGICAL CONSIDERATIONS

A. Occupational Radiation Exposures

The occupational radiation exposures that will result from the removal of the graphite blocks have been estimated and are presented in Figure 3.5-2 of the Cost Estimate (WBS Nos. 2.3.3.4, 2.3.3.5 and 2.3.3.6). Detailed breakdowns of the occupational radiation exposure estimate for the removal of all of the side spacer blocks with boronated pins, large side reflector blocks and hex reflector blocks without hastelloy cans are provided in Tables 9.2-1, 9.2-2 and 9.2-3.

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

- Use of the Work Platform will improve worker efficiency and safety.
- Installation of a HEPA ventilation system to remove evaporated tritium and other airborne contaminants away from the work area under and around the platform.
- Use of long handled tools and submerged staging areas to perform high exposure activities underwater.
- Use of automated drying equipment to minimize the need for operation by personnel.
- Use of temporary shielding as appropriate to maintain exposures ALARA.
- Installation of additional area radiation monitors (ARMS) with local alarm features to detect unexpected dose rates around the platform work areas.
- Audio-visual communication equipment to support remote surveillance of activities and equipment operations.
- Use of shielding bells.

B. Offsite Exposures From Accidents

Section 3.4.5 of the PDP analyzed a postulated heavy load drop accident that assumed that a container with a single unsectioned large side reflector block, with a total releasable activity of 1477 Curies, falls 100 feet to the level of the truck loading bay, and spills its entire contents on the truck bay floor. The whole body and lung doses to an adult standing at a point on the EPZ 100 meters from the Reactor Building were calculated to be 4.66 mRem and 133 mRem, respectively. 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" [6]. NUREG-0172 [7] dose conversion factors were also used in the calculations. This analysis determined that the radiation exposure to the general public as a result of a heavy load drop are very low. 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. The radiological consequences are also 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 [8].

As shown in Table 3.4-5 of the PDP, the graphite blocks contain the following activity that is available for release during postulated accidents:

	GRAPHITE BLOCK TYPE	Curies/Item
1.	Hex Reflector Blocks without Hastelloy Cans	
	- Bottom reflector blocks	4.43
	- Top reflector blocks	4.58
	- Side reflector blocks	8.30
2.	Large Permanent Side Reflectors	1477
3.	Side Spacer Blocks without Boronated Pins	16.1
4.	Boronated Pins	0.12

Note: As shown in WBS No. 2.3.3.6 of the Cost Estimate, the estimated Curie content of the side spacers without boronated pins range from 9.5 to 28.8 Curies. However, this is still significantly lower than the 1477 Curies estimated to be contained in the large permanent side reflector blocks.

As identified above, the large permanent side reflector blocks have the greatest amount of activity available for release during postulated accidents. Therefore, the dropping of an unsectioned large permanent side reflector block would be the maximum credible accident that could be postulated during the handling of any single graphite block. As such, the activity in any of the other graphite blocks is bounded by the 1477 Curies that was assumed in the accident analysis presented in Section 3.4.5 of the PDP. Moreover, any hypothetical dropping of a hex reflector without hastelloy cans or side spacer blocks with boronated pins would be bounded by the consequences predicted for the Heavy Load Drop accident as presented in Section 3.4.5 of the PDP. Although accident scenarios involving the dropping of a radwaste container with several graphite blocks can be postulated, the releasable Curie content of the radwaste containers will be limited to less than that analyzed in Section 3.4.5 of the PDP.

In addition, the Decommissioning Technical Specifications [12] specify requirements on the integrity of the Reactor Building and operation of the Reactor Building ventilation exhaust system to ensure that the offsite doses under abnormal conditions during decommissioning activities are well below 10 CFR 100 guidelines. Therefore, it can be concluded that the radiological consequences from any postulated handling accident involving the large permanent side reflector blocks, the hex reflector blocks without hastelloy cans, or the side spacers blocks with boronated pins removed would also be a small fraction of the limits established in 10 CFR 100.

C. Radioactive Waste Generated

A tabular listing of the estimated radioactive waste disposal volume for each WBS element was provided in Figure 3.2-1 of the Cost Estimate. As presented in Figure 3.2-1, the estimated disposal volume and burial class for the graphite blocks are:

GRAPHITE TYPE	DISPOSAL VOLUME (Cubic Feet)	BURIAL CLASS
Large Side Reflectors	12,600	B
Spacer Elements and boronated pins	2,393	B
Bottom hex reflector blocks w/o cans	1,414	A
Top hex reflector blocks w/o cans	1,515	A
Radial hex reflector blocks w/o cans	1.903	A

V. SAFETY ANALYSIS CONCLUSIONS

Since the block removal tasks are repetitive in nature, it is expected that the continued incorporation of "lessons learned" will reduce the cumulative exposure for this task. During the removal activities, the radiological hazards will be monitored and evaluated on a routine basis. All work activities associated with the removal of the graphite blocks will incorporate effective radiological controls to maintain occupational radiation exposures within regulatory limits and as iow as reasonably achievable (ALARA), consistent with dose-limiting provisions of 10 CFR 20, as well as both Regulatory Guides 8.8 [9] and 8.10 [10]. All workers will be provided instructions in radiation protection concepts commensurate with the radiological hazards that they will encounter during the removal of the grapnite blocks, including instructions concerning actions required during unusual conditions.

It is estimated that the removal of the large side reflector blocks, hex reflector blocks without hastelloy cans and side spacer blocks with boronated pins will result in total occupational radiological exposure of 50.0 person-Rem, 30.86 person-Rem and 15.85 person-Rem, respectively. Personnel occupational exposures will not be in excess of allowable 10 CFR 20 limits for occupational radiation exposure. Furthermore, postulated accidents involving the graphite blocks will result in offsite radiological consequences that are a small fraction of the guidelines established in both 10 CFR 100 and the EPA Protective Action Guidelines [8]. It is therefore concluded that the activities associated with the removal and disposal of the graphite blocks will not pose an undue risk to the health and safety of the general public nor to occupationally exposed decommissioning workers.

TABLE 9.2-1
OCCUPATIONAL RADIATION EXPOSURE ESTIMATE
LARGE SIDE REFLECTOR BLOCK REMOVAL

OPERATION	NO. OF WORKERS	EFFECTIVE EXPOSURE RATE (mR/hr)	TASK DURATION (Min.)	WORKER EXPOSURE TIME (hrs)	ESTIMATED EXPOSURE (person-mRem)
1. Move Block From Core to Intermediate Position	4	2	220	14.7	29.33
2. Move Shield Bell into Position	3	2	15	0.8	1.5
3. Load Block into Shield Bell	3	7	20	1.0	7.0
4. Raise Bell, Install Catch Pan	3	7	15	0.8	5.25
5. Move Loaded Bell to Refueling Floor	3	7	15	0.8	5.25
6. Move Shield Bell to Dewatering Device and Dewater	3	7	45	2.3	15.75
7. Load Block Back to Shield Bell, Move to HSF, Segment	3	4	400	20.0	80.0
8. Load Block Back to Shield Bell, Move to Dryer	3	7	15	0.8	5.25
9. Discharge Block to Dryer	3	7	20	1.0	7.0
10. Remove from Dryer, Move to Cask	3	2	15	0.8	1.5
11. Load into Cask	3	1.8	27	1.4	2.4
TOTALS				44.0	160.26

Note: Multiplying the 160.26 mRem by 312 to account for all of the large side reflector blocks yields the 50.0 person-Rem shown in Table 3.5-2 of the Decommissioning Cost Estimate.

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TABLE 9.2-2
OCCUPATIONAL RADIATION EXPOSURE ESTIMATE
HEX REFLECTOR BLOCKS W/O HASTELLOY CANS REMOVAL

OPERATION	NO. OF WORKERS	EFFECTIVE EXPOSURE RATE (mR/hr)	TASK DURATION (Min.)	WORKER EXPOSURE TIME (hrs)	ESTIMATED EXPOSURE (person-mRem)
1. Move Block From Core to Intermediate Position	3	2	20	1.0	2.00
2. Move Shield Bell into Position	3	2	10	0.5	1.00
3. Load Block into Shield Bell	3	3.5	12	0.6	2.10
4. Raise Bell, Install Catch Pan	3	3.5	10	0.5	1.75
5. Move Loaded Bell to Refueling Floor	3	3.5	10	0.5	1.75
6. Move Shield Bell to Dewatering Device and Dewater	3	3.5	20	1.0	3.50
7. Load Block Back to Shield Bell, Move to Dryer	3	3.5	10	0.5	1.75
8. Discharge Block to Dryer	3	3.5	15	0.8	2.625
9. Remove from Dryer, Move to Cask	3	2	10	0.5	1.00
10. Load into Cask	3	1.25	12	0.6	0.75
TOTALS				6.5	18.23

Note: Multiplying the 18.23 mRem by 1693 to account for all of the hex reflector blocks without hastelloy cans yields the 30.86 person-Rem shown in Table 3.5-2 of the Decommissioning Cost Estimate.

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TABLE 9.2-3
OCCUPATIONAL RADIATION EXPOSURE ESTIMATE
SIDE SPACER BLOCKS WITH BORONATED PINS REMOVAL

	OPERATION	NO. OF WORKERS	EFFECTIVE EXPOSURE RATE (mR/hr)	TASK DURATION (Min.)	WORKER EXPOSURE TIME (hrs)	ESTIMATED EXPOSURE (person-mRem)
1.	Move Block From Core to Intermediate Position	2	2	20	0.7	1.33
2.	Install Dumping Tool with Integral Bell	2	2	15	0.5	1.00
3.	Dump Boronated Pins into Cask Liner	2	2	3	0.1	0.20
4.	Stage Bell	2	2	10	0.3	0.67
5.	Load Block into Shield Bell	2	3	10	0.3	1.00
6.	Raise Bell, Install Catch Pan	2	3	10	0.3	1.00
7.	Move Loaded Bell to Refueling Floor	2	3	10	0.3	1.00
8.	Move Shield Bell to Dryer	2	3	10	0.3	1.00
9.	Discharge Block to Dryer	2	3	20	0.7	2.00
10.	Remove from Dryer, Move to Cask	2	2	10	0.3	0.67
11.	Load Block into Cask	3	1.5	14	0.7	1.05
12.	Remove Bail	2	3	4	0.1	0.40
13.	Remove Pin Cask Liner	2	65	12	0.4	26.00
14.	Dewater Pin Cask Liner	2	65	20	0.7	43.33
15.	Move Pin Cask Liner to Cask	2	65	12	0.4	26.00
16.	Load Liner into Cask	3	22	20	1.0	22.00
	TOTALS				7.2	128.65

Note: It is assumed that there will be a total of 24 pin cask liners. Therefore, multiplying the block removal activities by 1152 to account for all of the boronated side spacer blocks and multiplying the packaging of the pin cask liner by 24 yields the 15.85 person-Rem shown in Table 3.5-2 of the Decommissioning Cost Estimate.

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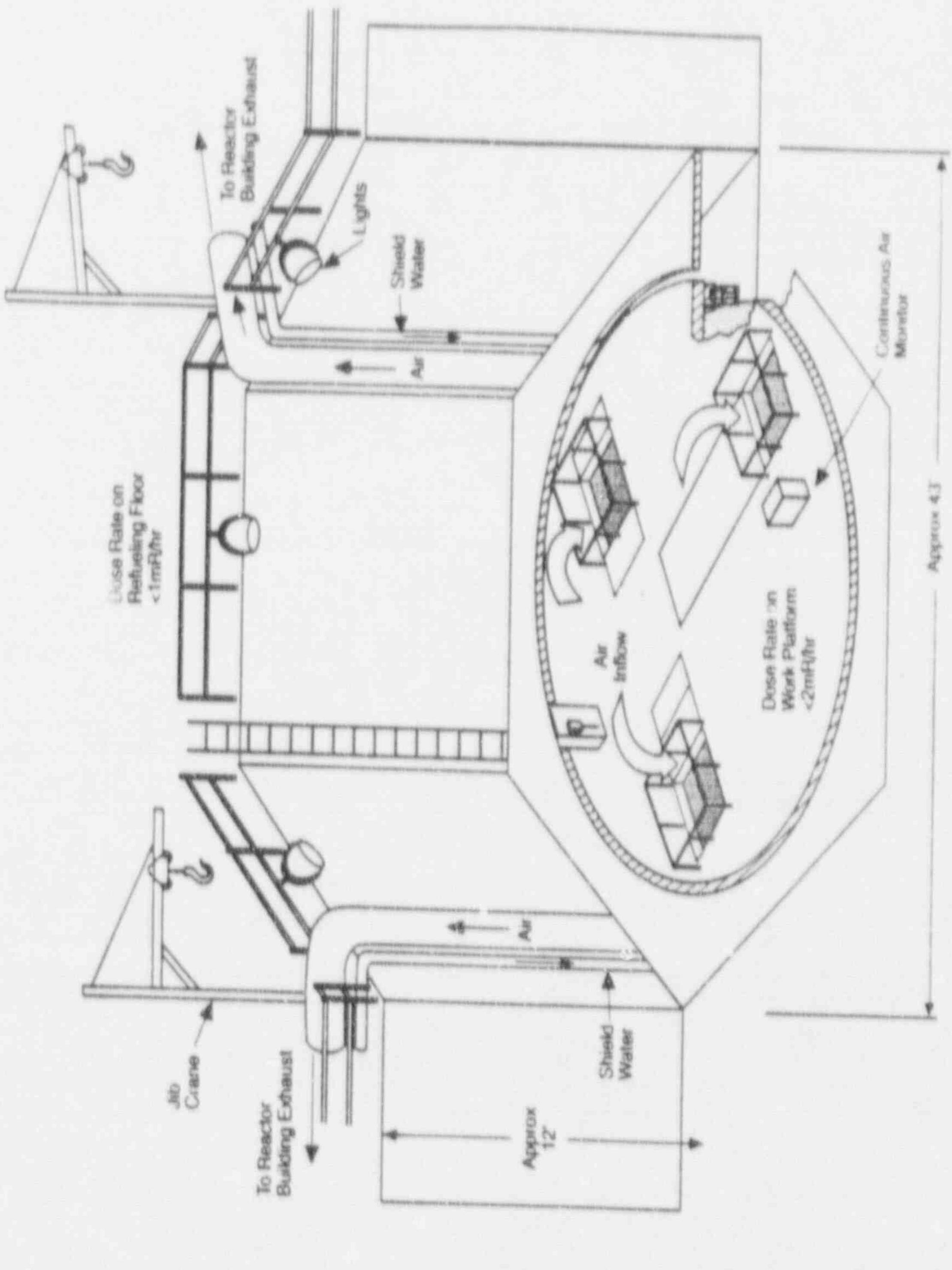

Figure 9.2-1 Rotary Work Platform

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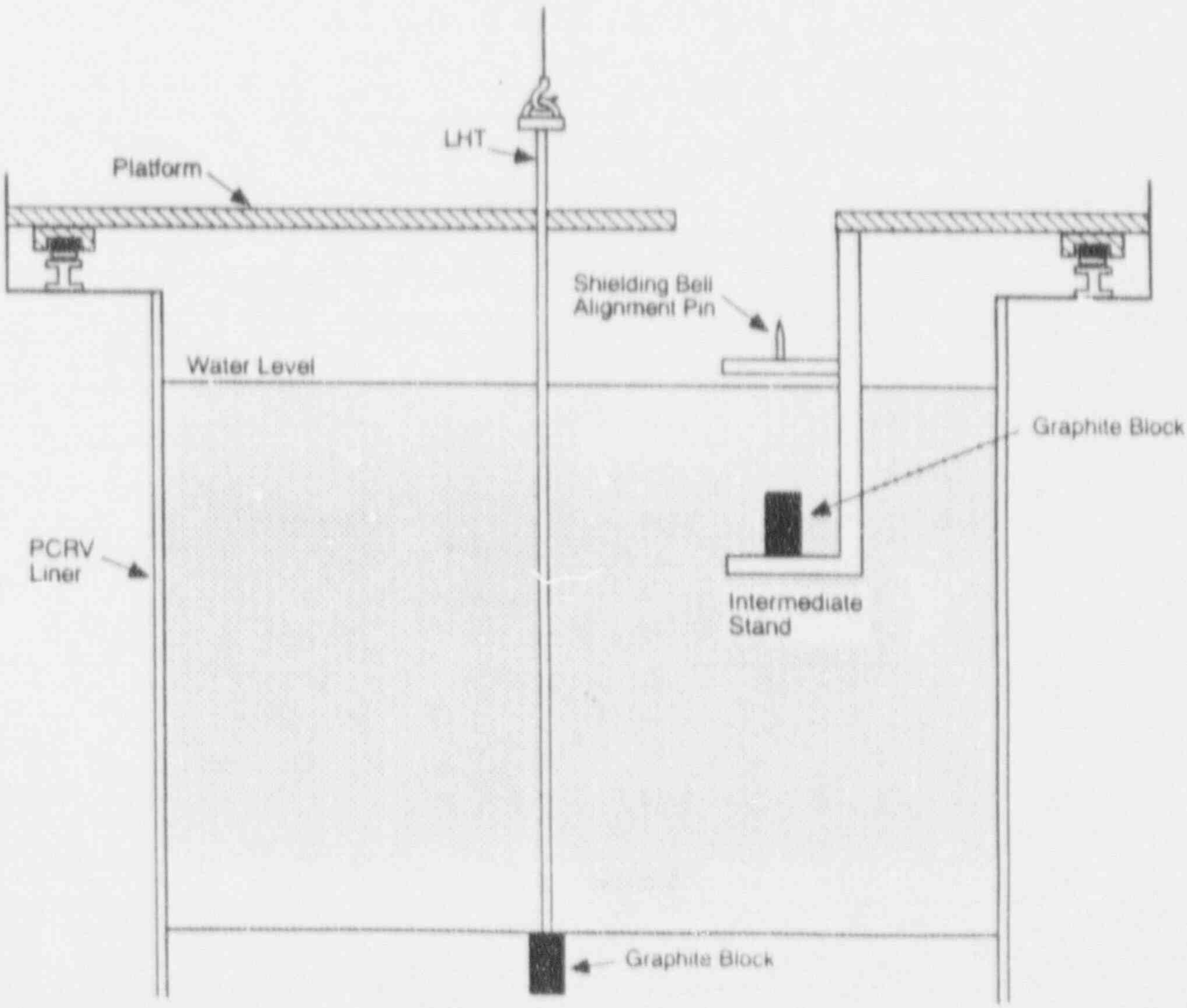

Figure 9.2-2 Graphite Block Intermediate Stand

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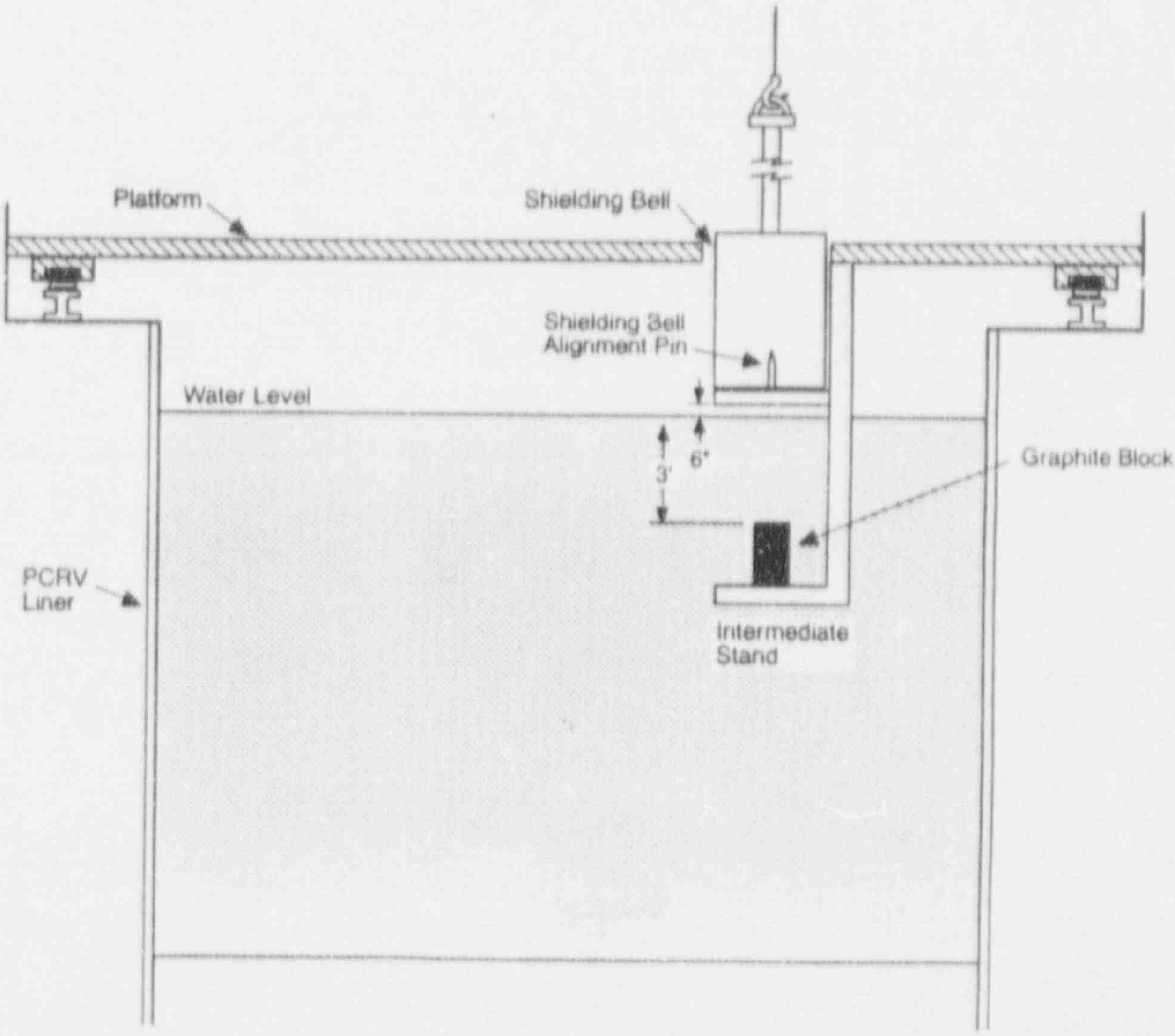

Figure 9.2-3 Loading Shielding Bell

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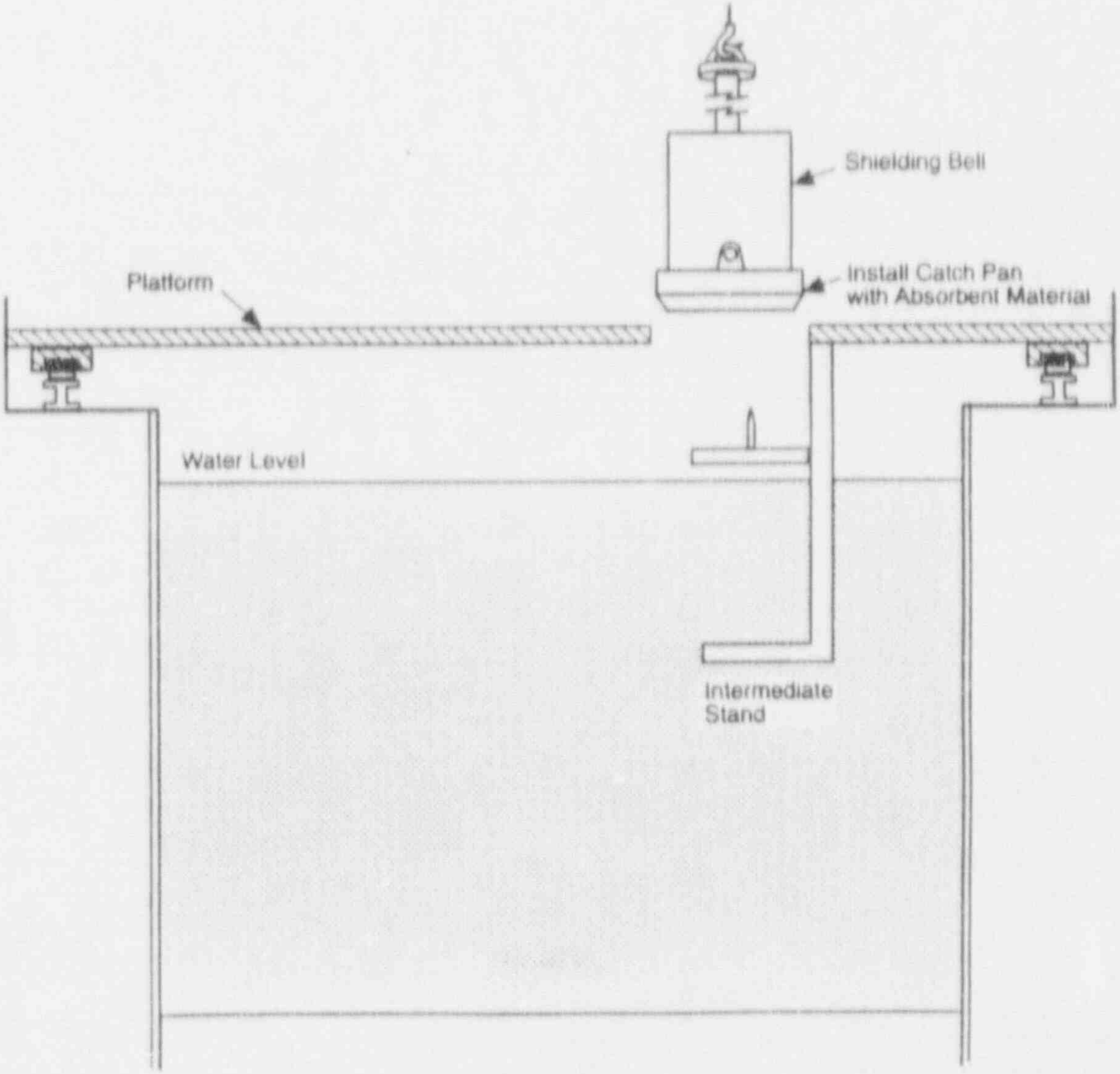

Figure 9.2-4 Shielding Bell With Catch Pan

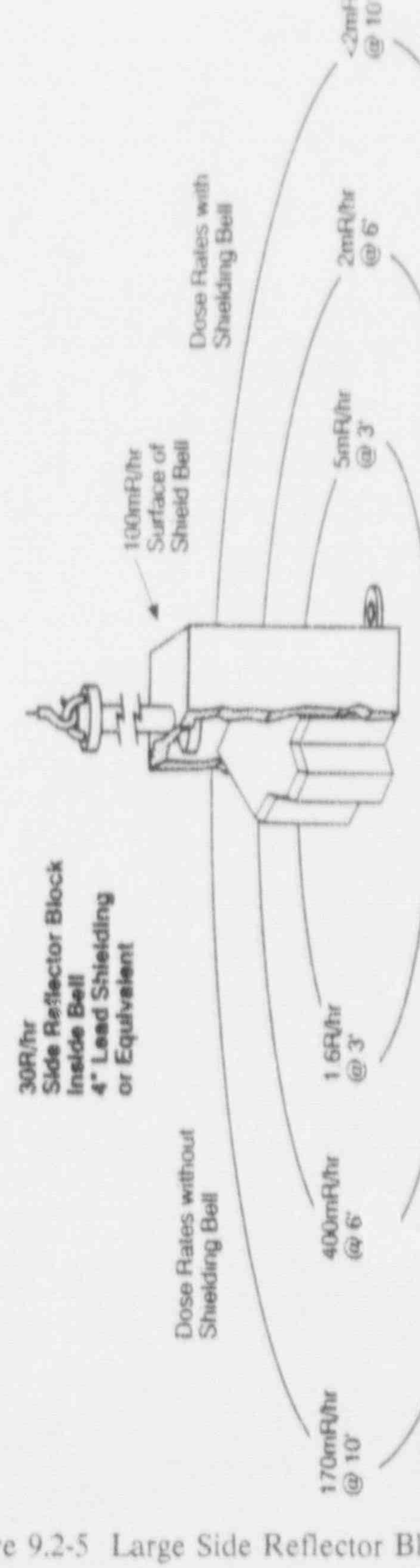

Figure 9.2-5 Large Side Reflector Block Dose Rates

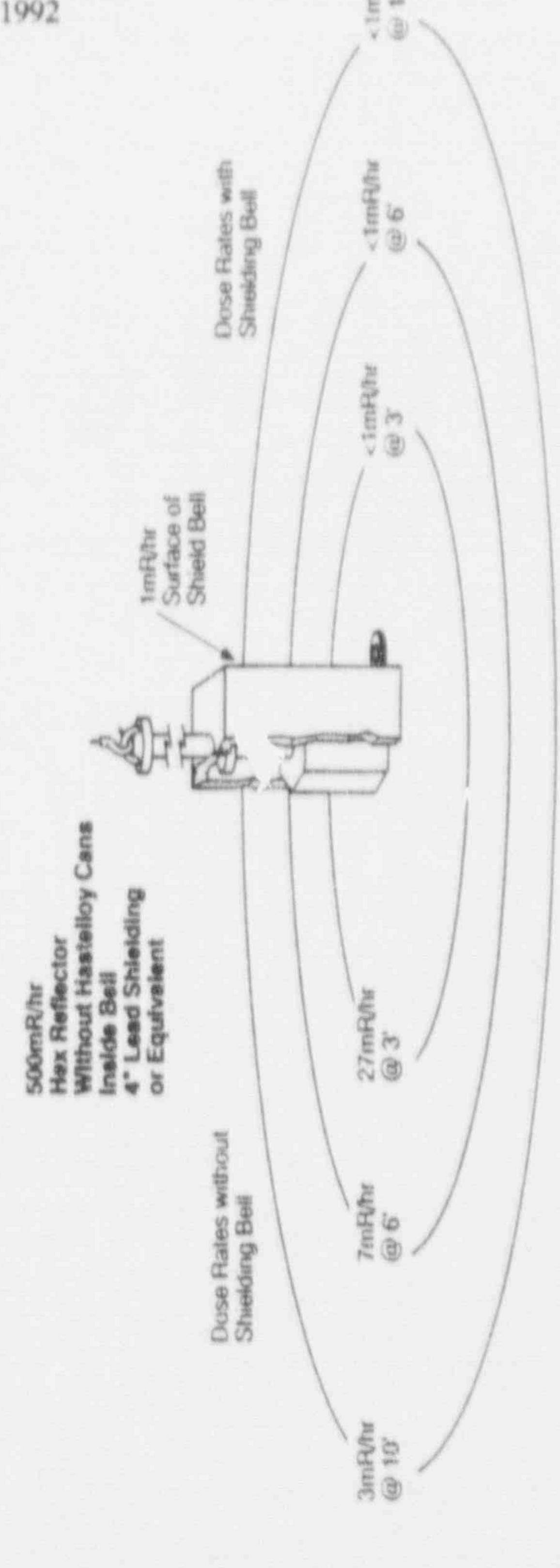

Figure 9.2-6 Hex Reflector Block (Without Hastelloy Cans) Dose Rates

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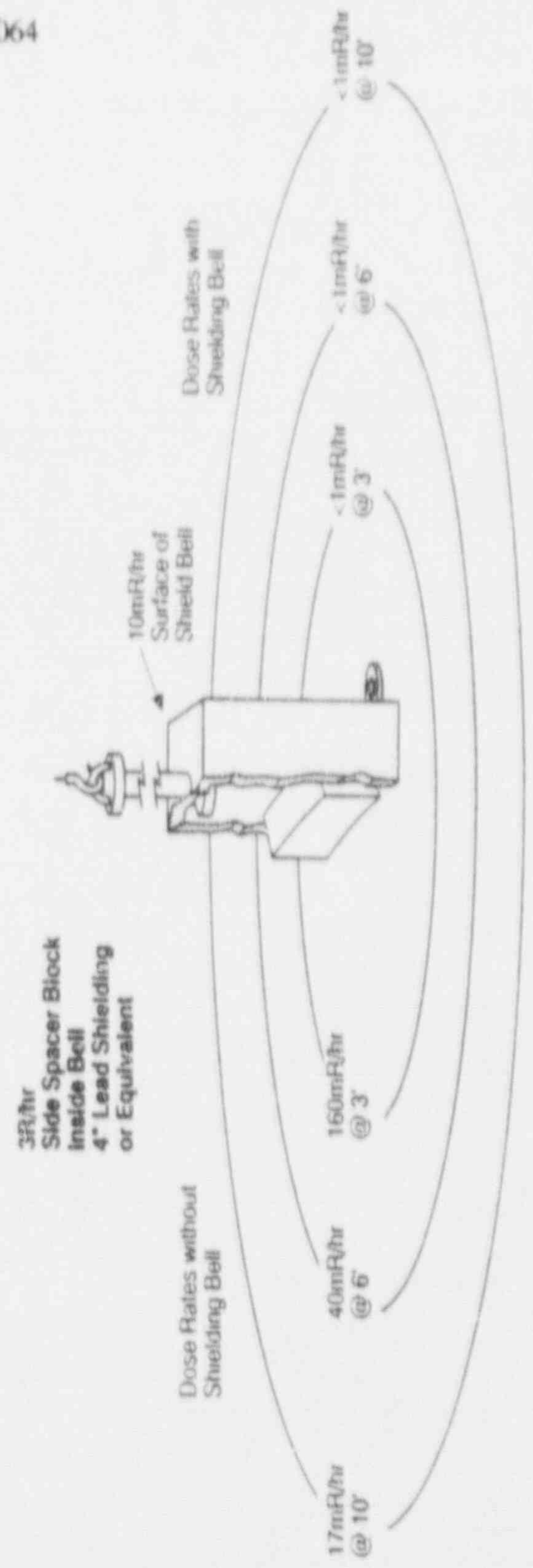

Figure 9.2-7 Side Spacer Blocks (Without
Boronated Pins) Dose Rates

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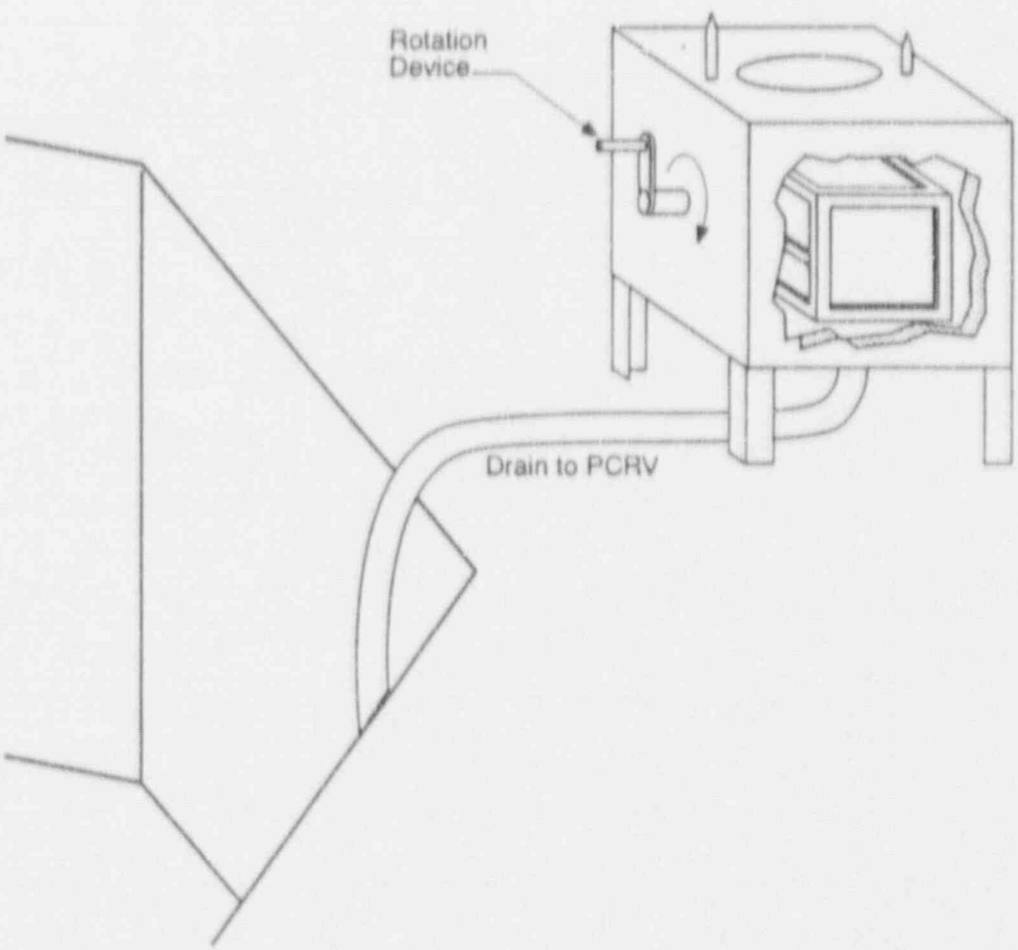

Figure 9.2-8 Dewatering Device

Attachment to P-92064
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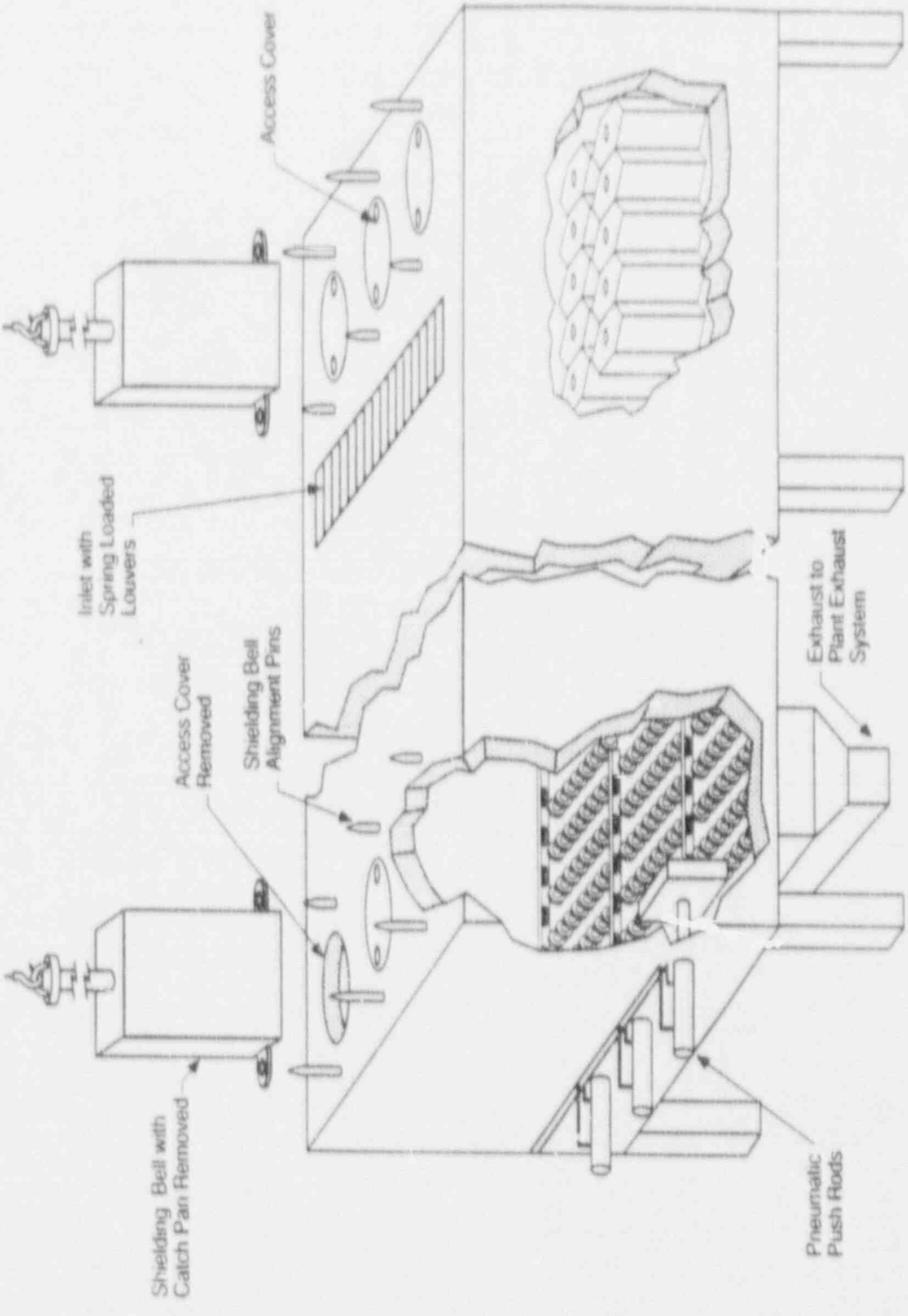

Figure 9.2-9 Block Dryer Arrangement
(Loading/Unloading)

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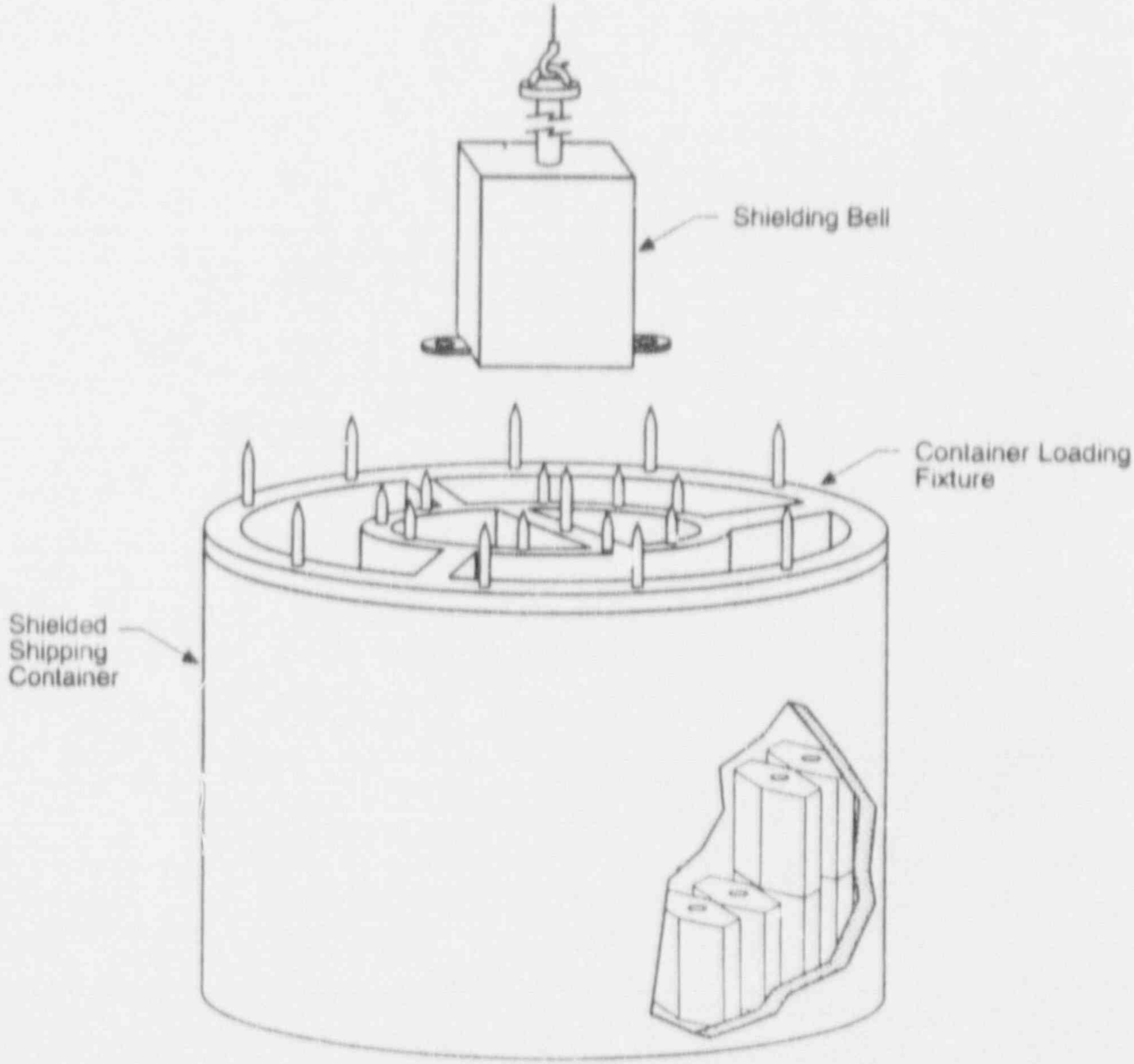

Figure 9.2-10 Loading Shipping Container

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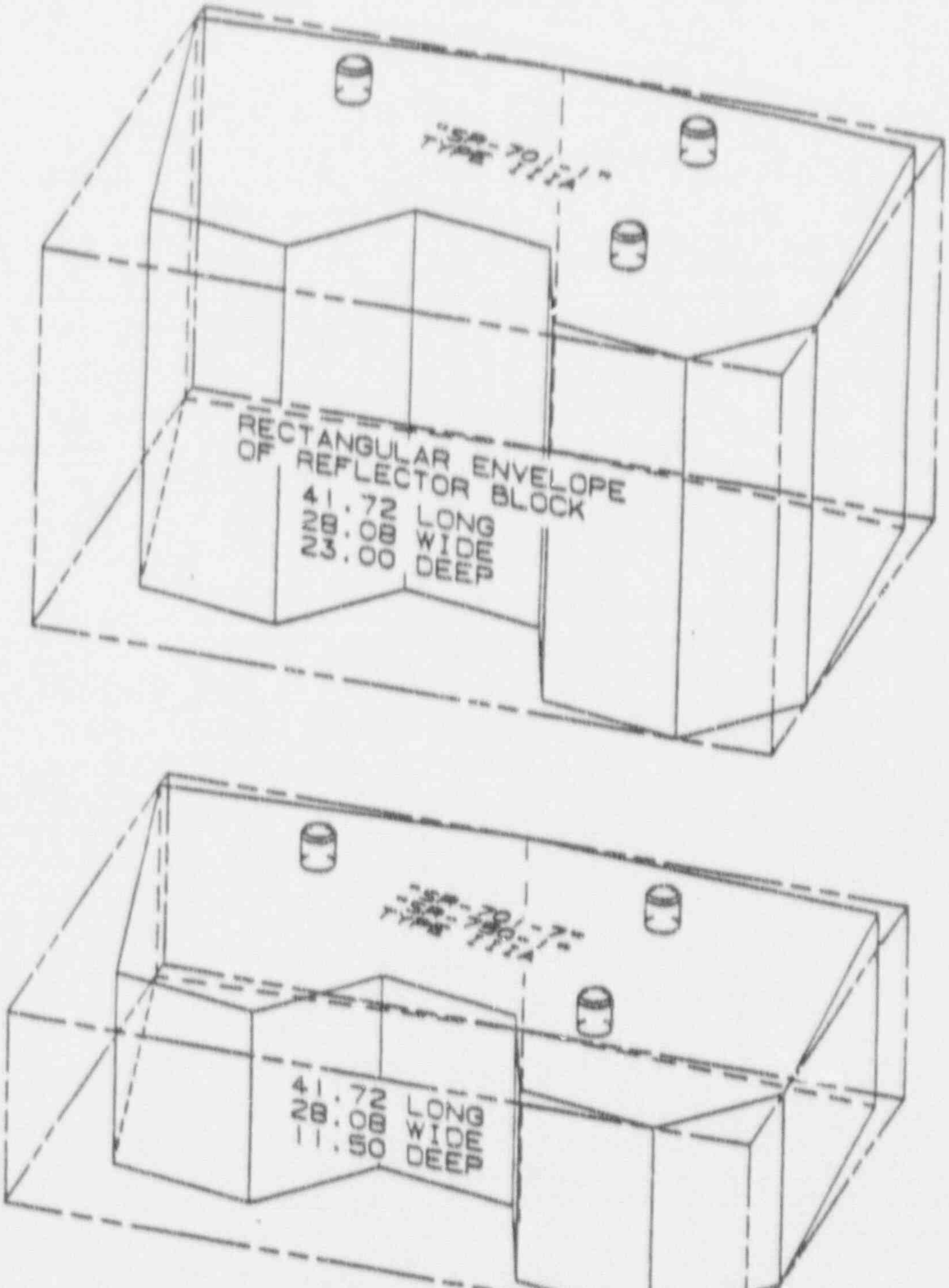

Figure 9.2-11 Large Side Reflector Block
(Typical)

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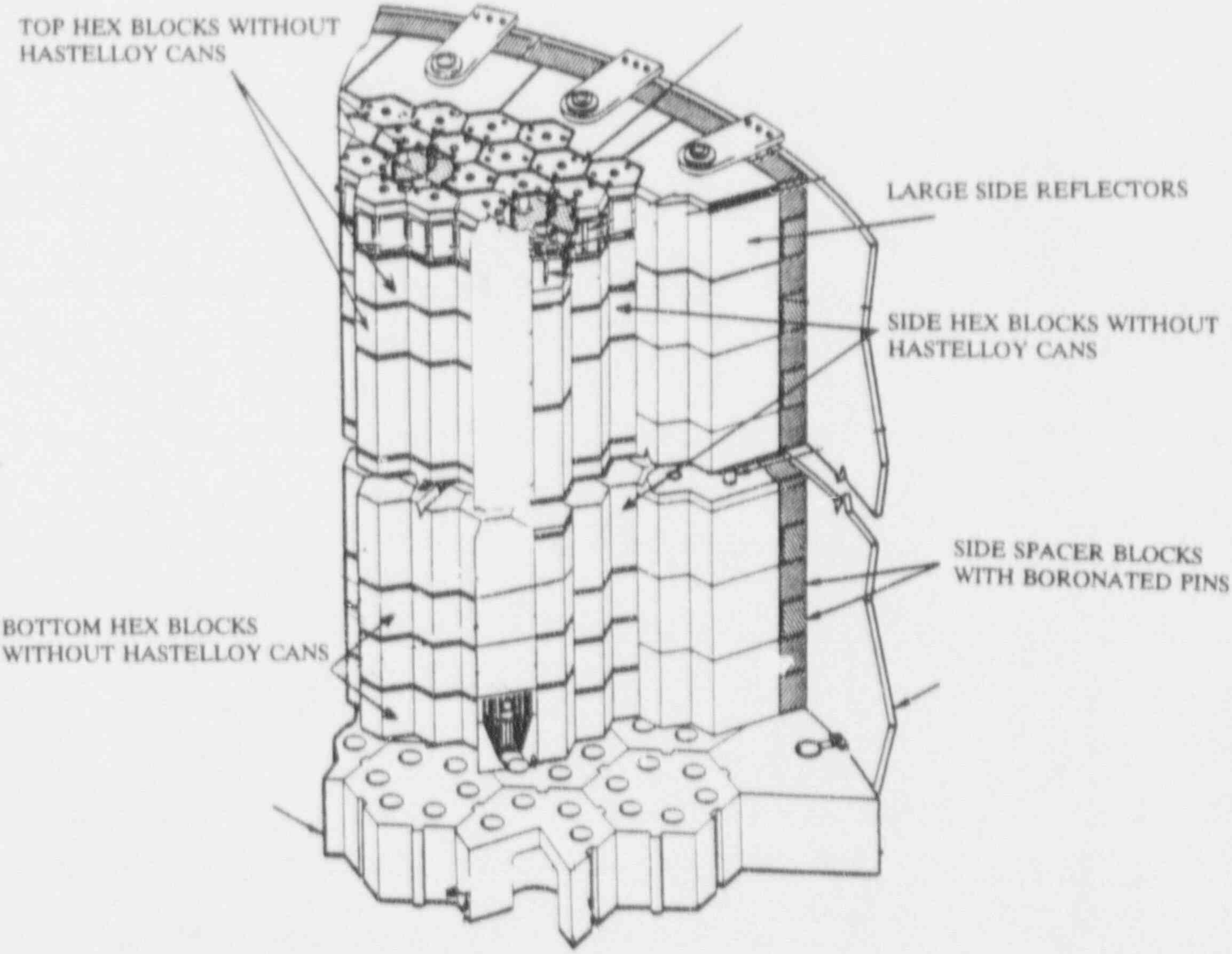

Figure 9.2-12 Location In Core of Graphite Blocks

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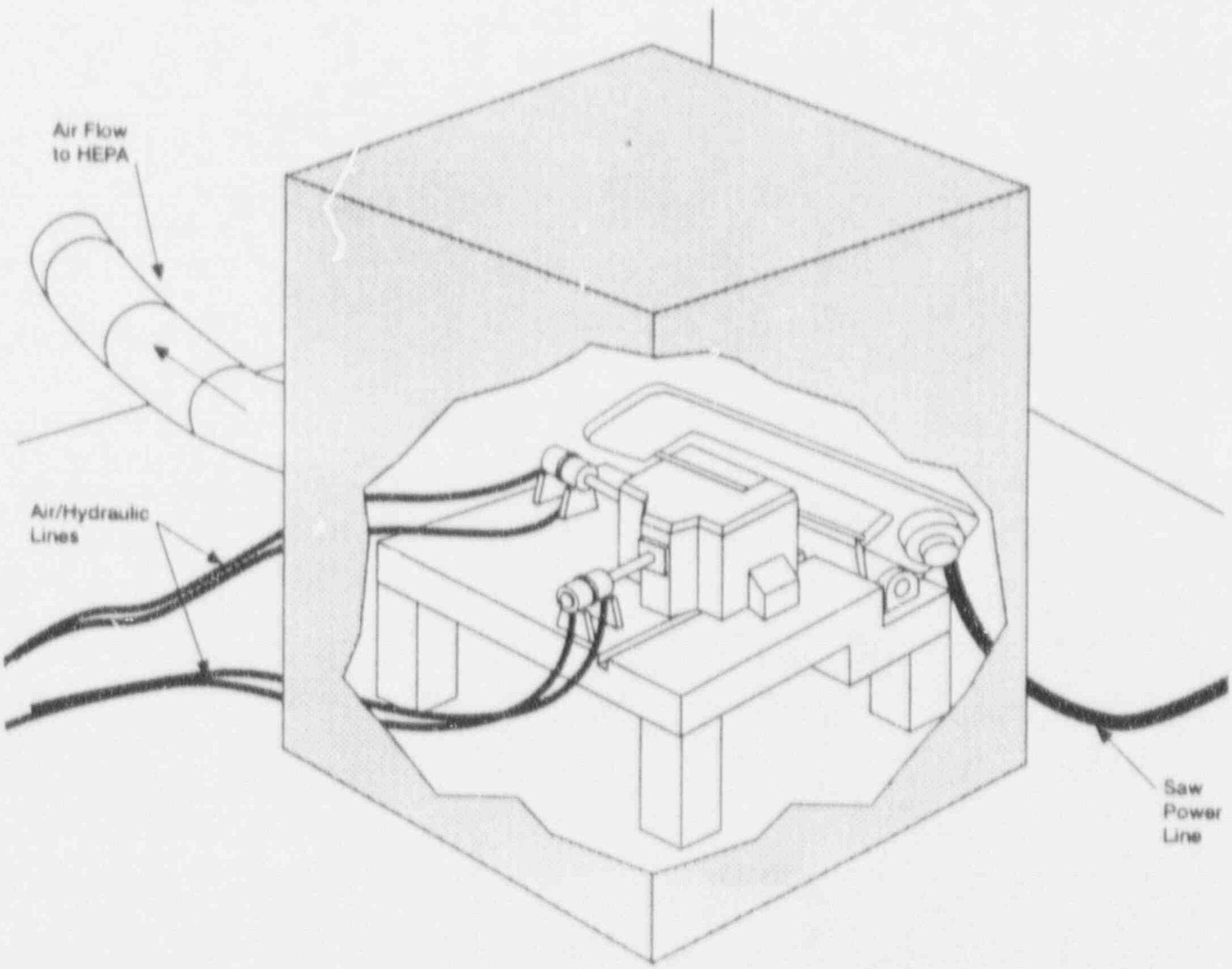

Figure 9.2-13 Large Side Reflector Block
Sectioning Station

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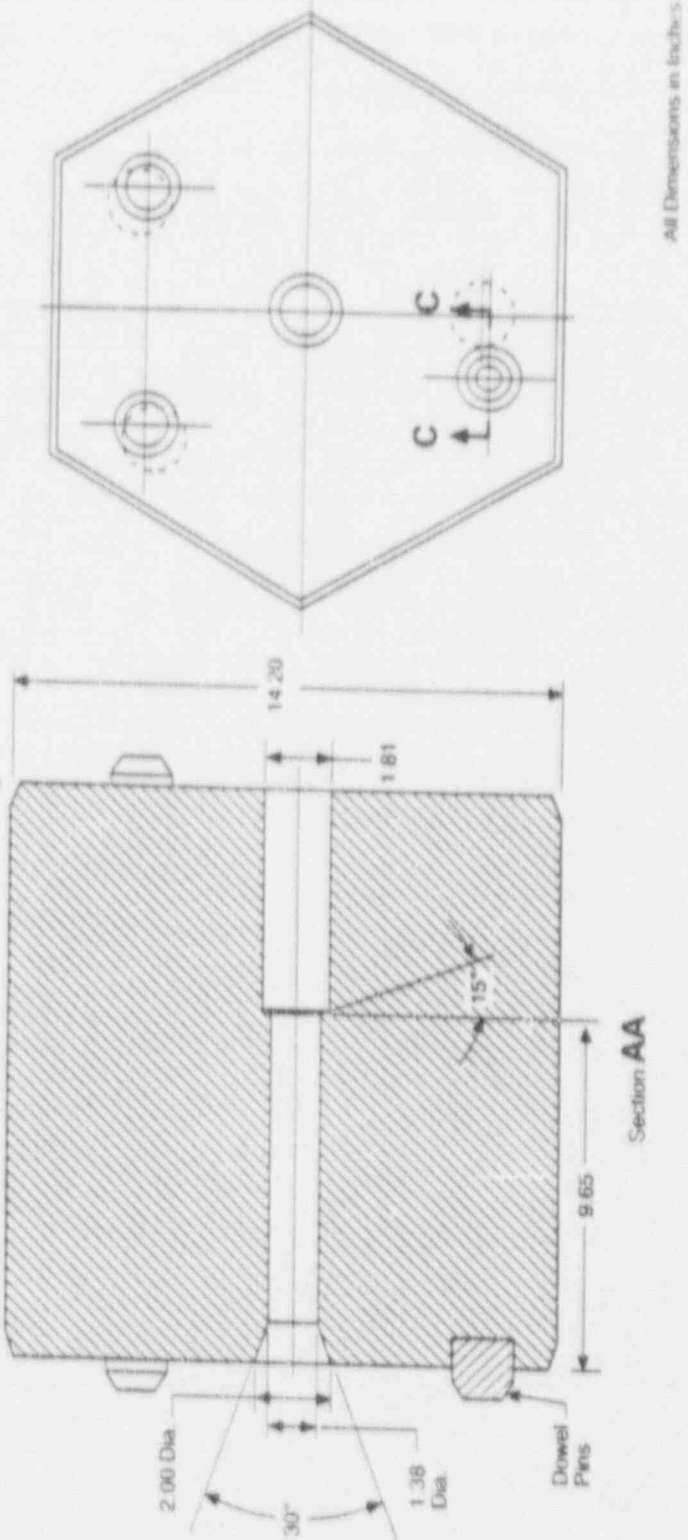

Figure 9.2-14 Hex Reflector Block without Hastelloy Cans

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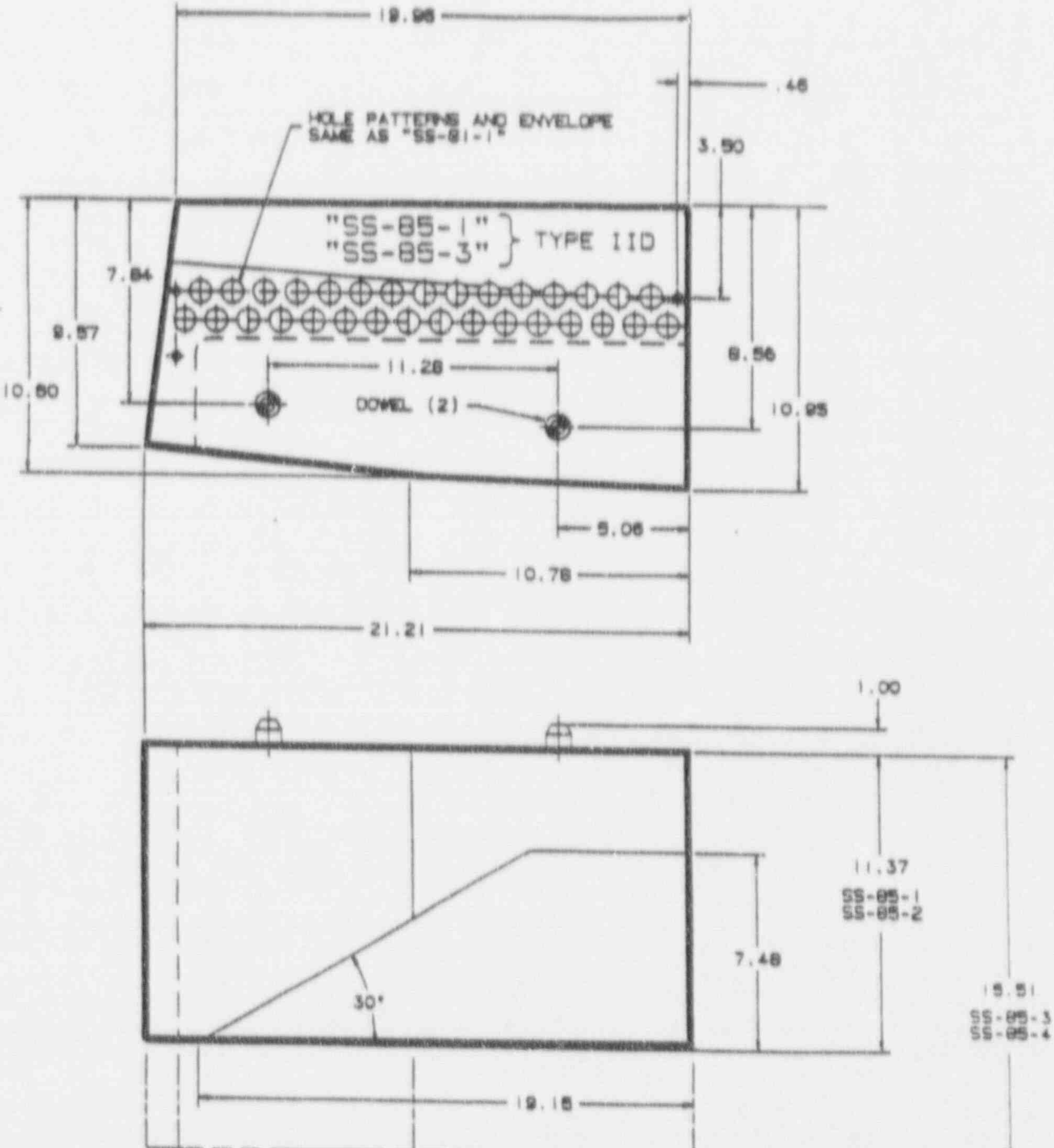

Figure 9.2-15 Side Spacer Block
(Typical)

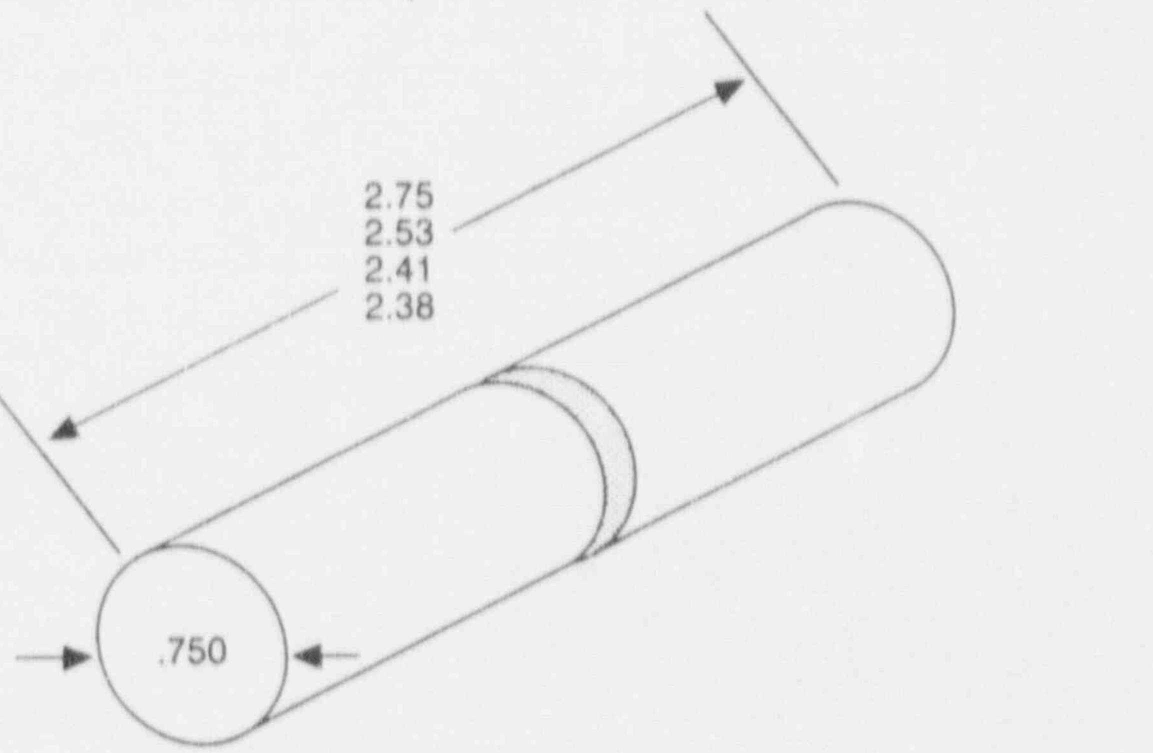

Figure 9.2-16 Boronated Rod

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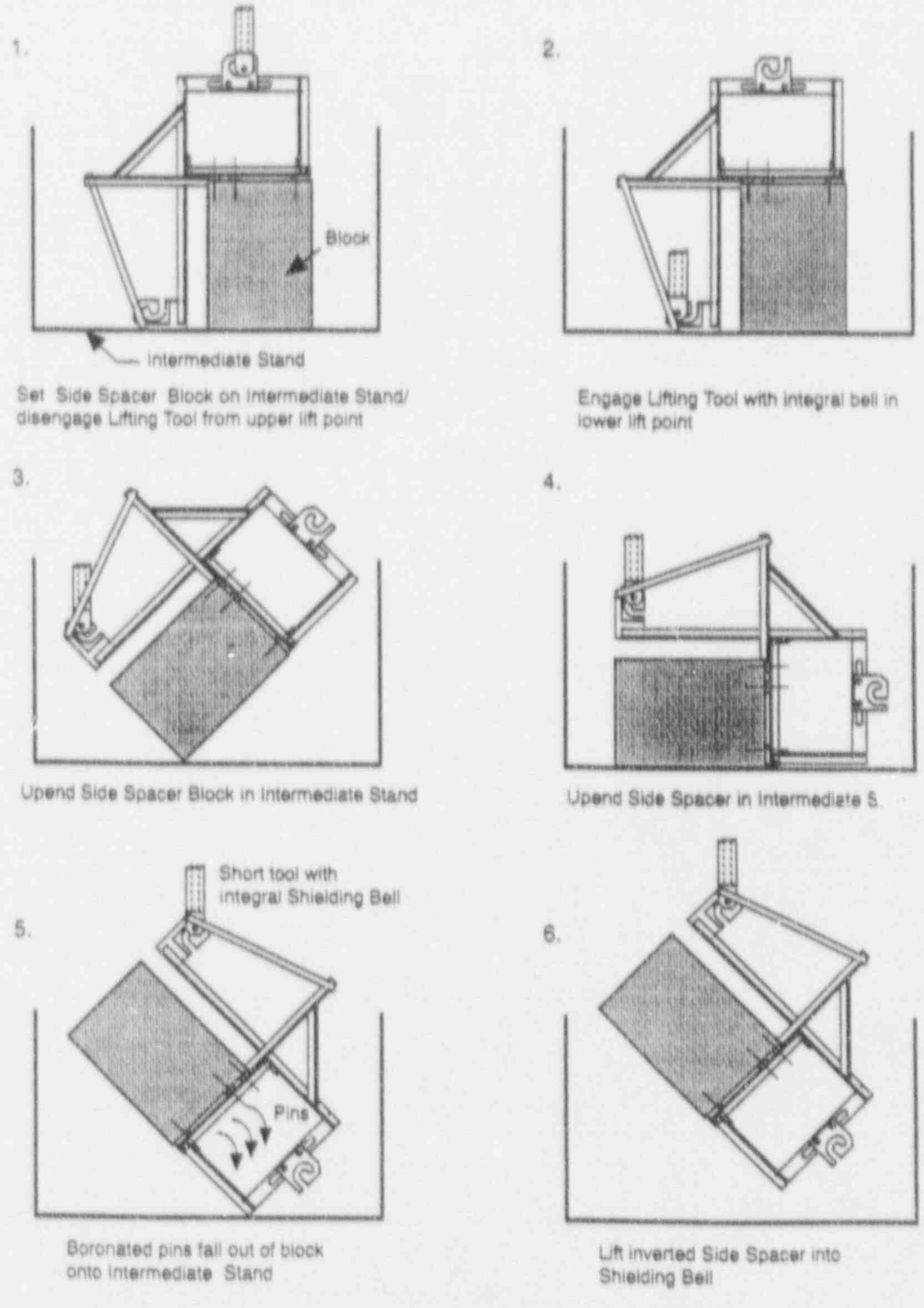

Figure 9.2-17 Side Spacer Block Pin Dumping

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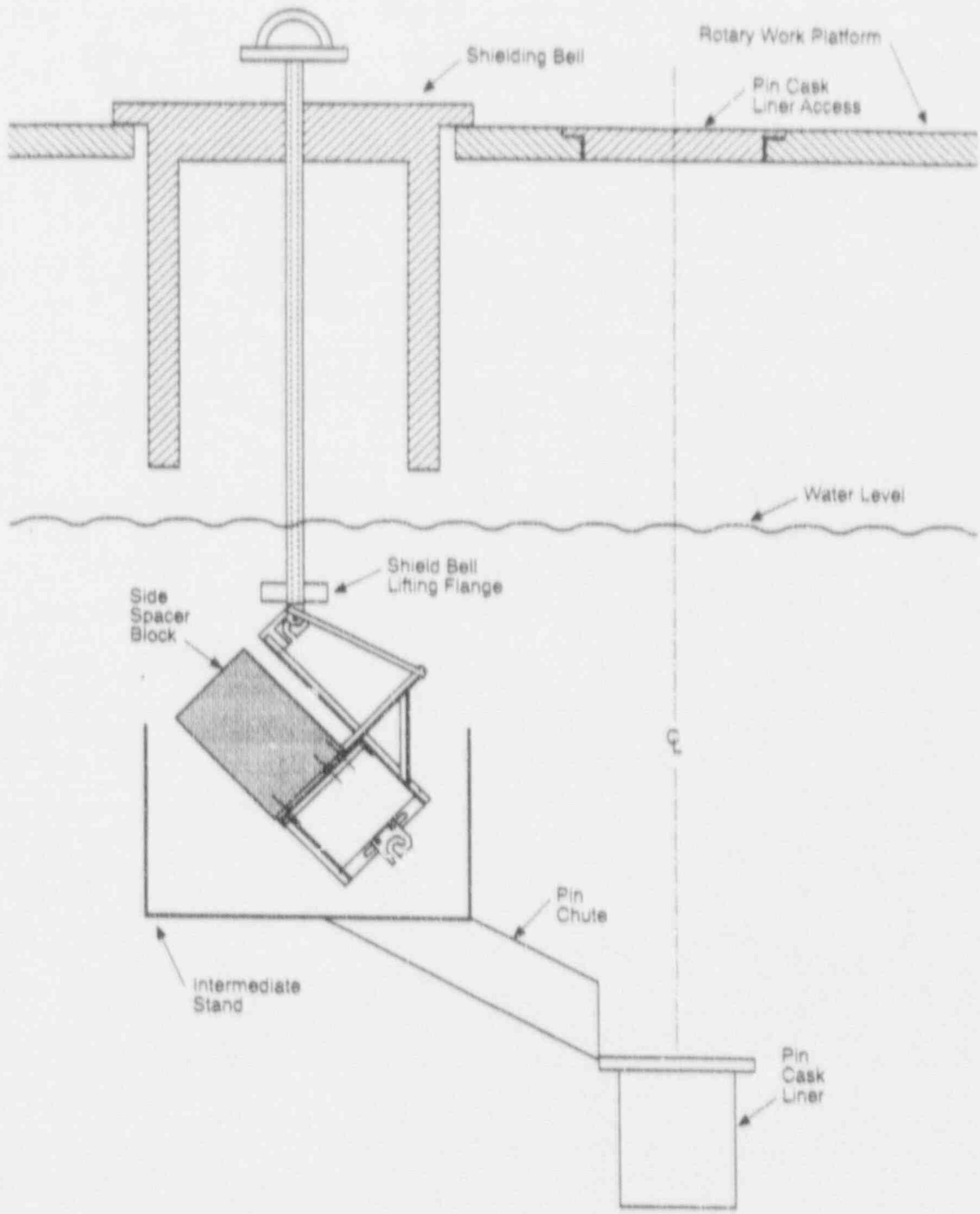

Figure 9.2-18 Side Spacer Pin Collection

9.3 CORE BARREL REMOVAL

I. GENERAL DESCRIPTION - CORE BARREL AND OUTER KEYS

The core barrel, Figure 9.3-1, is a steel cylinder approximately 27 feet 4 inches inside diameter and 29 feet high. Its thickness steps from 2.25 inches thick at the top to 2.75 inches at the bottom. The core barrel has 12 upper outer keys and 12 lower outer keys which center the core barrel to the PCRV liner. The lower three feet of both the inside and outside surface of the core barrel is insulated. In addition, there are seven thermocouple penetrations located about four feet above the bottom of the core barrel that are between the PCRV and the core barrel.

II. DESCRIPTION OF THE CORE BARREL AND OUTER KEYS REMOVAL PROCEDURE

A. Prerequisites For Core Barrel and Outer Keys Removal

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

1. The PCRV top head concrete has been removed as discussed in Section 9.1 of this response.
2. The PCRV has been flooded above the core barrel with shield water and water clarity has been established as discussed in Section 2.3.3.6 of the PDP [3].
3. Reactor core graphite blocks have been removed from the PCRV to a level low enough to permit the cutting of a core barrel section as discussed in Section 9.2 of this response.

B. Removal Activities

The methods for removal of the core barrel and outer keys, as described in Section 2.3.3.9 of the PDP, were selected with consideration of worker safety and minimizing occupational exposure. WBS Element No. 2.3.4.3.1 of the Decommissioning Cost Estimate provided further details of the dismantlement operations.

The core barrel and core barrel keys will be segmented underwater using remotely operated cutting equipment after the graphite core components are removed. However, as stated in Section 2.3.3.9 of the PDP, if radiological surveys in the core barrel indicate that actual radiation and contamination levels are low, the PCRV water level will be progressively lowered and the core barrel and outer keys will be thermally cut above the water line. While cutting of the core barrel above the water line appears to have a schedule advantage over the underwater cutting, it will only be considered if it can be justified by an ALARA review.

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

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

1. Rigging the core barrel sections for removal.
2. Making horizontal and vertical cuts in the core barrel to segment it into sections suitable for handling.
3. Removing the core barrel segments out of the PCRV.
4. Progressively removing the outer keys and thermocouple expansion joint assembly that is between the PCRV liner and the core barrel. (See the core barrel key detail on Figure 9.3-1.)

cutting of the core barrel will be performed with the Work Platform in
For underwater cutting, a mast or a remotely positioned track-mounted
ool will be operated from the Work Platform to make the vertical cuts
e core barrel. When the vertical cuts are complete, rigging will be
the core barrel segments prior to making the horizontal cuts. The
will then be made and the core barrel segment removed. The jib
used to lift the segments to awaiting LSA boxes positioned
opening on the work platform. The cut pattern will be
d upon the size of LSA containers selected and the features
g system. The assumptions used in the PDP and cost

The core barrel and core barrel keys will be segmented underwater using remotely operated cutting equipment after the graphite core components are removed. However, as stated in Section 2.3.3.9 of the PDP, if radiological surveys in the core barrel indicate that actual radiation and contamination levels are low, the PCRV water level will be progressively lowered and the core barrel and outer keys will be thermally cut above the water line. While cutting of the core barrel above the water line appears to have a schedule advantage over the underwater cutting, it will only be considered if it can be justified by an ALARA review.

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1. Rigging the core barrel sections for removal.
2. Making horizontal and vertical cuts in the core barrel to segment it into sections suitable for handling.
3. Removing the core barrel segments out of the PCRV.
4. Progressively removing the outer keys and thermocouple expansion joint assembly that is between the PCRV liner and the core barrel. (See the core barrel key detail on Figure 9.3-1.)

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

III. RADIOLOGICAL CONSIDERATIONS

A. Occupational Radiation Exposure (ORE)

The procedures and controls to be used to maintain a safe operation and minimize occupational exposure have been factored into the removal of the core barrel and keys. They include the following:

1. The Work Platform ventilation system will draw air from the refueling floor to the Work Platform, down through the access openings in the platform, and then exhaust it to the Reactor Building Ventilation (exhaust) System where it will be discharged in accordance with the Fort St. Vrain ODCM [11]. With the platform in place and the ventilation system operating, the cutting fumes and any potential airborne contamination will be contained below the platform.

2. Exhaust hoods, powered by HEPA-filtered air handlers, will be positioned at the water surface or, if the cut is performed dry, in close proximity to the cut. These exhaust hoods will capture the majority of the fumes at their source.

3. The tooling will allow the workers to perform their activities on the platform in a relatively low radiation field, estimated to be 2 mR/hr.

4. Protective clothing, respiratory protection, HEPA ventilation and temporary shielding required for the various phases of this removal

operation will be evaluated and prescribed during the pre-job ALARA review and RWP preparation process.

5. The core barrel sections will be surveyed as they break the water to determine exposure rates before being handled. The segments are expected to have a contact dose rate of 40 mR/hr. Loose contamination is expected to be moderate (100,000 - 300,000 dpm/100 cm).

6. Loose surface contamination from pieces removed from the water will be controlled by a combination of pressure washing, rinsing with clean water, wet vacuuming and swabbing. These measures will control the spread of contamination and minimize potential for airborne contamination.

7. Access to the work platform will be controlled by Radiation Protection personnel during cutting operations and when core barrel segments are removed from the pool. This will ensure positive control and maintain personnel exposure ALARA.

The person-hour estimate for the core barrel removal, described in WBS Element No. 2.3.4.3.1, includes 4913 person-hours. It is estimated that 2456 of these hours will involve occupational radiation exposure. Workers are expected to be exposed to the highest radiation fields during the handling operation as segments of the core barrel are removed from the water (40 mR/hr on contact). This operation will be carefully monitored by radiation protection personnel. The total occupational radiation exposure (ORE) for the removal of the core barrel and the 24 outer keys is estimated to be 22.59 person-Rem. The ORE estimates for individual core barrel removal tasks are provided in Table 9.3-1.

B. Offsite Exposures From Accidents

As presented in Table 3.4-5 of the PDP, the core barrel is expected to have 8.4 Curies of activity. However, this activity is tightly bound in the steel as activation products and is not readily dispersible. Furthermore, since the core barrel will be cut and removed in segments, the total activity that could be involved in an accident during the lifting and handling of the core barrel segments will be a very small percentage of the total 8.4 Curies. As such, it is concluded that an accident involving the removing and handling of the core barrel segments would be bounded by the 9.83 (98.3 x 10%) Curies that was assumed to be involved in the Contaminated Concrete Rubble Drop Accident scenario described in Section 3.4.3 of the PDP. The radiological consequences from a postulated accident involving the handling of the core barrel segments would be well within the 25 Rem whole body dose and 300 Rem to any specific organ guidelines established in 10 CFR 100.

C. <u>Radioactive Waste Generated</u>

As tabulated in Figure 3.2-1 of the Decommissioning Cost Estimate, the estimated radioactive waste disposal volume is 1400 cubic feet for the core barrel and outer keys. The burial classification for both the outer keys and the core barrel is expected to be "Class A".

IV. SAFETY ANALYSIS CONCLUSIONS

During the core barrel removal activities, the radiological hazards will be monitored and evaluated on a routine basis. All work activities associated with the removal of the core barrel and keys will incorporate effective radiological controls to maintain occupational radiation exposures within regulatory limits and as low as reasonably achievable (ALARA), consistent with dose-limiting provisions of 10 CFR 20, as well as both Regulatory Guides 8.8 [9] and 8.10 [10]. All workers will be provided instructions in radiation protection concepts commensurate with the radiological hazards that they will encounter during the removal of the core barrel and keys, including instructions concerning actions required during unusual conditions.

It is estimated that the removal of the core barrel and keys will result in a total occupational radiological exposure of 22.59 person-Rem. Personnel occupational exposures will not be in excess of allowable 10 CFR 20 limits for occupational radiation exposure. Furthermore, postulated accidents involving the core barrel and keys will result in offsite radiological consequences that are a small fraction of the guidelines established in both 10 CFR 100 and the EPA Protective Action Guidelines [8]. It is therefore concluded that the activities associated with the removal and disposal of the core barrel and keys will not pose an undue risk to the health and safety of the general public nor to occupationally exposed decommissioning workers.

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TABLE 9.3-1
OCCUPATIONAL RADIATION EXPOSURE ESTIMATES
CORE BARREL AND OUTER KEY REMOVAL

OPERATION	NO. OF WORKERS	EFFECTIVE EXPOSURE RATE (mR/hr)	TASK DURATION (hrs)	WORKER EXPOSURE TIME (hrs)	ESTIMATED EXPOSURE (person-Rem)
Setup and Prep area to remove core barrel	2	2	55.0	110	0.22
Cut core barrel	4	6	477.25	1909	11.45
Lift sections out of water and move sections to [illegible]A boxes and package	4	25	109.25	437	10.92
TOTALS				2456	22.59

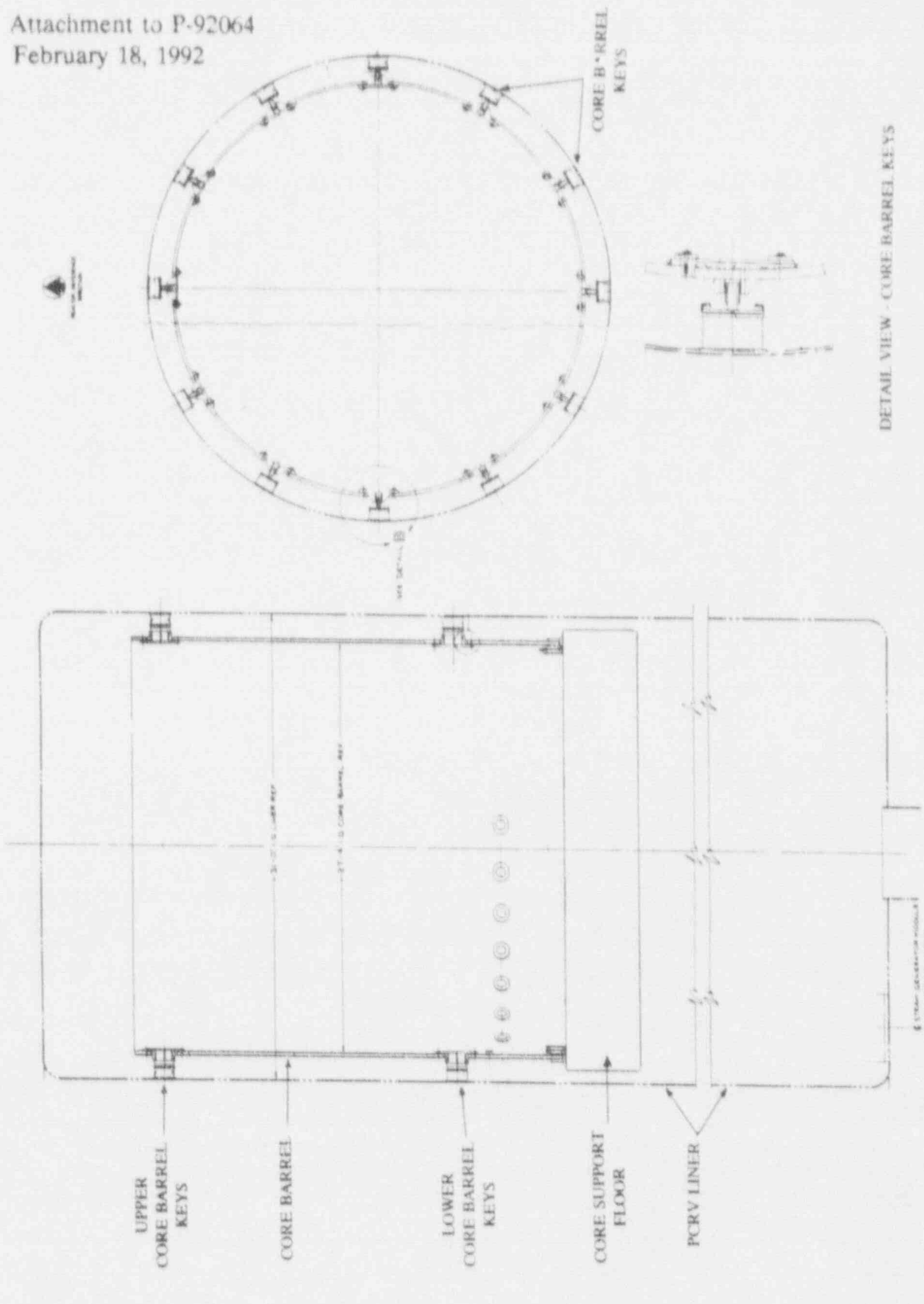

Figure 9.3-1 Core Barrel

9.4 CORE SUPPORT FLOOR

I. GENERAL DESCRIPTION - CORE SUPPORT FLOOR (CSF)

The removal of the Core Support Floor (CSF) is necessary during decommissioning because it is both contaminated and activated, and to gain access to the lower chamber of the PCRV. The method of removing the CSF is discussed in Section 2.3.3.10.2 of the PDP. As stated in PSC's response to NRC RAI Question No. 15, dated April 26, 1991 [4], the removal of the CSF is considered a difficult operation because of its size and weight. Figure 9.4-1 provides a simplified cut-away view of the CSF, showing the steam generator ducts, CSF support columns and thermal insulation.

Although Section 2.2.2 of the PDP simply describes the CSF as the component that bears the weight of the reactor core, the CSF is a complex component that includes the following features:

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

II. DESCRIPTION OF THE CSF REMOVAL PROCEDURE

A. Prerequisites For CSF Removal

The prerequisites necessary to begin work on the CSF are as follows:

1. All core components have been removed from the PCRV.
2. The core barrel has been removed to within a few feet of the top of the CSF.
3. The shield water has been drained down to within a few feet of the top of the CSF.
4. Loose contamination has been removed from or stabilized on the interior walls (insulation cover plate) of the PCRV.
5. The Work Platform is removed prior to lifting the CSF to the upper PCRV area.

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

B. Removal Activities

The removal of the CSF top insulation using a remotely-operated electro-hydraulic ram hoe from a work platform above the CSF is discussed in Section 2.3.3.10.1 of the PDP. Once the insulation on top of the CSF has been removed, the remaining two feet of the core barrel will be removed in a manner similar to that described in Section 9.3 of this response. Cutting of the steam generator ducts and CSF columns were discussed in the response to NRC RAI Question No. 15 [4] and Section 2.3.3.10.2 of the PDP. However, both of these documents indicate that in-air thermal cutting would be used to detach the CSF columns, which is unlikely due to the expected CSF dose rates discussed above. Unless dose rates are determined to be significantly below those estimated, it will be necessary to perform these cuts underwater to obtain adequate shielding from the PCRV shield water. The use of underwater cutting, in combination with exhaust hoods and respiratory protection, will provide the worker a safe environment. A localized containment may be used to prevent the spread of airborne contamination to other areas or workers.

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

1. Raising the CSF

Prior to lifting the CSF, workers will require access to the area immediately above the CSF inside the PCRV to perform the cutting of the steam generator ducts and CSF columns, and to attach the lifting cables to the CSF. These activities will be accomplished using a man-basket suspended from the Reactor Building crane. This method will minimize the time that will be spent in the radiation field and minimize the resultant exposure. The use of the man-basket will comply with the requirements of 29 CFR 1926.550(g) and will also be coordinated with the containments that will be in place during the various phases of the work.

After the CSF has been cut free from the steel support columns, the CSF will be lifted and supported inside the PCRV. The CSF will be raised to the PCRV top head region using a strand jacking system, which uses multiple cables attached both to the CSF and to the jacking stations that have been established on top of the PCRV (see Figure 9.4-2). After raising the CSF, supports will be installed on the PCRV ledge where the PCRV top head was previously cut and removed, and the CSF will then be lowered onto these supports.

2. Segmenting the CSF

With the CSF supported in the top head area, segmenting the CSF into sections will begin. Prior to initiation of segmenting activities, radiological surveys will be performed to determine the extent of the activation in the

CSF. Based on the results of the radiological surveys, shielding may be placed over the top of the CSF to reduce radiation levels to acceptable levels. Radiological containments may also be constructed if determined necessary.

Segmenting the CSF will be performed using the diamond wire cutting operation. The primary work area for the segmenting activity will be around the perimeter of the CSF. This will keep the workers away from the top of the CSF which is the significant source of radiation exposure. The diamond wire cutting process is adequate to segment the CSF and the monorail spider located under the CSF, eliminating the need to remove this monorail separately. Individual segments of the CSF will be removed by the Reactor Building crane to the fuel deck staging area, where the segments will be prepared for disposal.

III. RADIOLOGICAL CONSIDERATIONS

A. Occupational Radiation Exposures

The estimate includes 7,762 person-hours for the Removal of the Core Support Floor (WBS No. 2.3.4.4.), of which 3881 person-hours involve radiation exposure. The exposure rate from the top activated surfaces of the CSF is expected to be approximately 360 mR/hr after removing the insulation from the top surface of the CSF. The occupational radiation exposure that will result from the cutting and removal of the CSF has been estimated to be 48.51 person-Rem. A breakdown of the exposure for each of the major tasks that will be performed to remove the CSF is shown in Table 9.4-1.

B. Offsite Exposures From Postulated Accidents

As previously stated, the graphite core components and the core barrel will have been removed from the PCRV prior to removing the CSF. As such, accidents involving activated graphite blocks will no longer be a concern. Since the CSF is predicted to have very low levels of radioactivity, the risk of accidents resulting in offsite radiological releases during the removal of the CSF is considerably less than the postulated decommissioning accident scenarios presented in Section 3.4 of the PDP.

Although the exposure rate from the top surface of the CSF is expected to be approximately 400 mRem/hr, the CSF concrete is predicted to contain only 6 Curies of activity as shown in Table 3.4-3 of the PDP. Furthermore, as shown in Table 3.4-5 of the PDP, the total Curie content expected to be found in the various components that will be severed or removed as part of the CSF activities are identified as follows:

Component	Total Curies
Core support floor columns	1
Misc. steel from beneath CSF	2
Misc. inconel parts on CSF	15
CSF Liner	142

Based on these limited Curie contents and the fact that activity is embedded in the steel liner and is not readily dispersible, any postulated accidents created during CSF removal activities would involve the release of smaller amounts of activity than the concrete rubble that was postulated in the accident scenario analyzed in Section 3.4.3 of the Decommissioning Plan. Therefore, in the unlikely event that an accident occurred during removal of the CSF, the offsite consequences would be bounded by those predicted for the postulated concrete rubble drop accident analyzed in Section 3.4.3 of the PDP.

C. Radioactive Waste Generated

A tabular listing of the estimated radioactive waste disposal volume for each WBS was provided in Figure 3.2-1 of the Decommissioning Cost Estimate. As presented in Figure 3.2-1, the estimated disposal volume of the activated concrete in the CSF is 6,240 cubic feet and 200 cubic feet of concrete cutting debris. The activated concrete from the CSF is expected to be classified as Class "A" waste.

IV. SAFETY ANALYSIS CONCLUSIONS

During the CSF removal activities, the radiological hazards will be monitored and evaluated on a routine basis. All work activities associated with the removal of the core support floor will incorporate effective radiological controls to maintain occupational radiation exposures within regulatory limits and as low as reasonably achievable (ALARA), consistent with dose-limiting provisions of 10 CFR 20, as well as both Regulatory Guides 8.8 [9] and 8.10 [10]. All workers will be provided instructions in radiation protection concepts commensurate with the radiological hazards that they will encounter during the removal of the core support floor, including instructions concerning actions required during unusual conditions.

It is estimated that the removal of the core support floor will result in a total occupational radiological exposure of 48.51 person-Rem. Personnel occupational exposures will not be in excess of allowable 10 CFR 20 limits for occupational radiation exposure. Furthermore, postulated accidents involving the core support floor will be bounded by the postulated concrete rubble drop accident (PDP Section 3.4.3) and will result in offsite radiological consequences that are a small fraction of the guidelines established in both 10 CFR 100 and EPA Protective Action Guidelines [8]. It is therefore concluded that the activities associated with the removal and disposal of the core support floor will not pose an undue risk to the health and safety of the general public nor to occupationally exposed decommissioning workers.

TABLE 9.4-1
OCCUPATIONAL RADIATION EXPOSURE ESTIMATE
CORE SUPPORT FLOOR REMOVAL

OPERATION	NO. OF WORKERS	EFFECTIVE EXPOSURE RATE (mR/hr)	TASK DURATION (hrs)	WORKER EXPOSURE TIME (hrs)	ESTIMATED EXPOSURE (person-Rem)
Install CSF Jacking System	4	4	78.75	315	1.26
Cut ducts and columns	5	15	212.0	1060	15.90
Jack CSF up and support on ledge	4	35	35.0	140	4.90
Remove insulation from CSF	4	30	35.0	140	4.20
Segment concrete	5	10	282.0	1410	14.10
Rig and Remove concrete	4	10	81.25	325	3.25
Package concrete	4	10	122.5	490	4.90
TOTALS				3880	48.51

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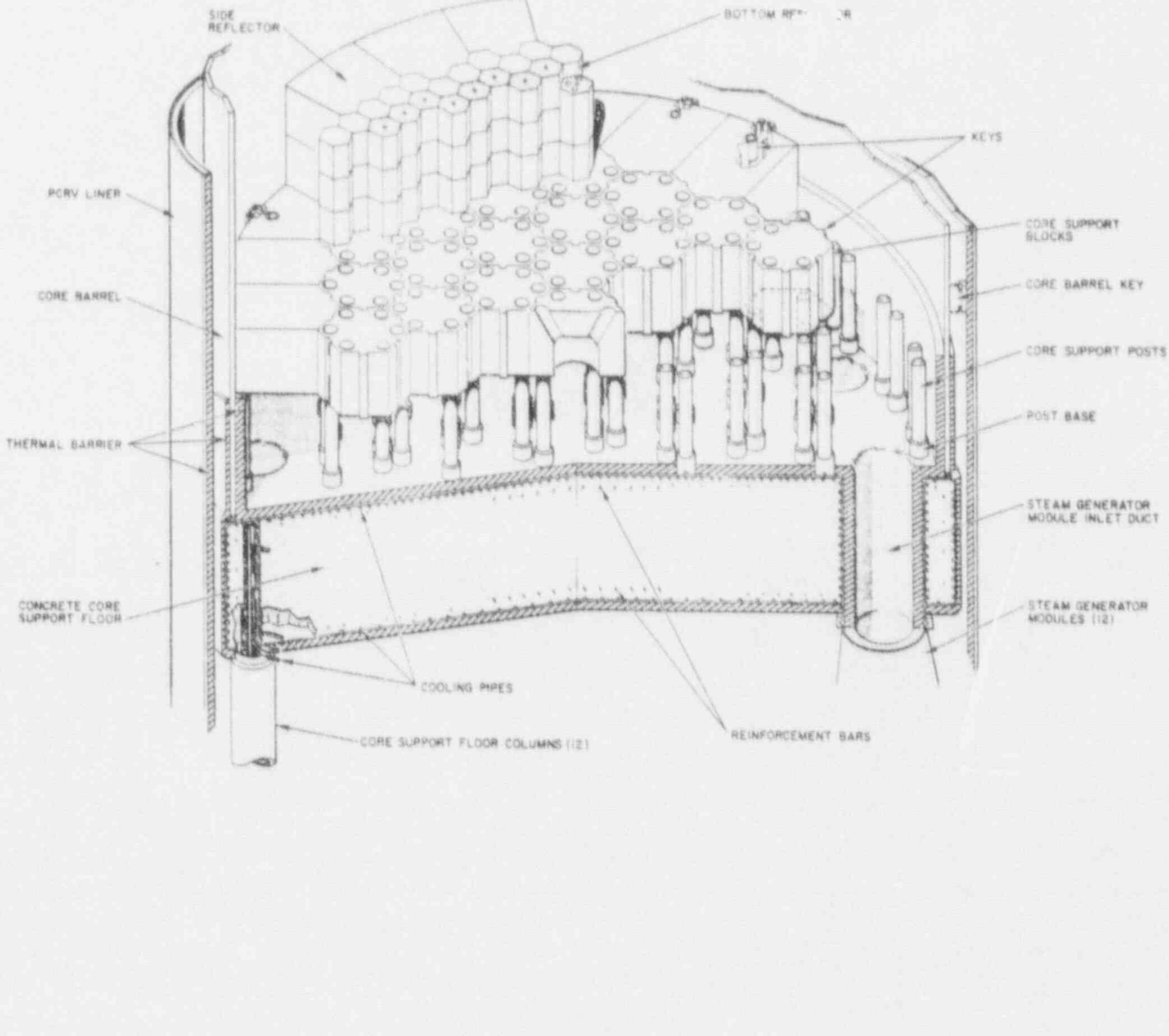
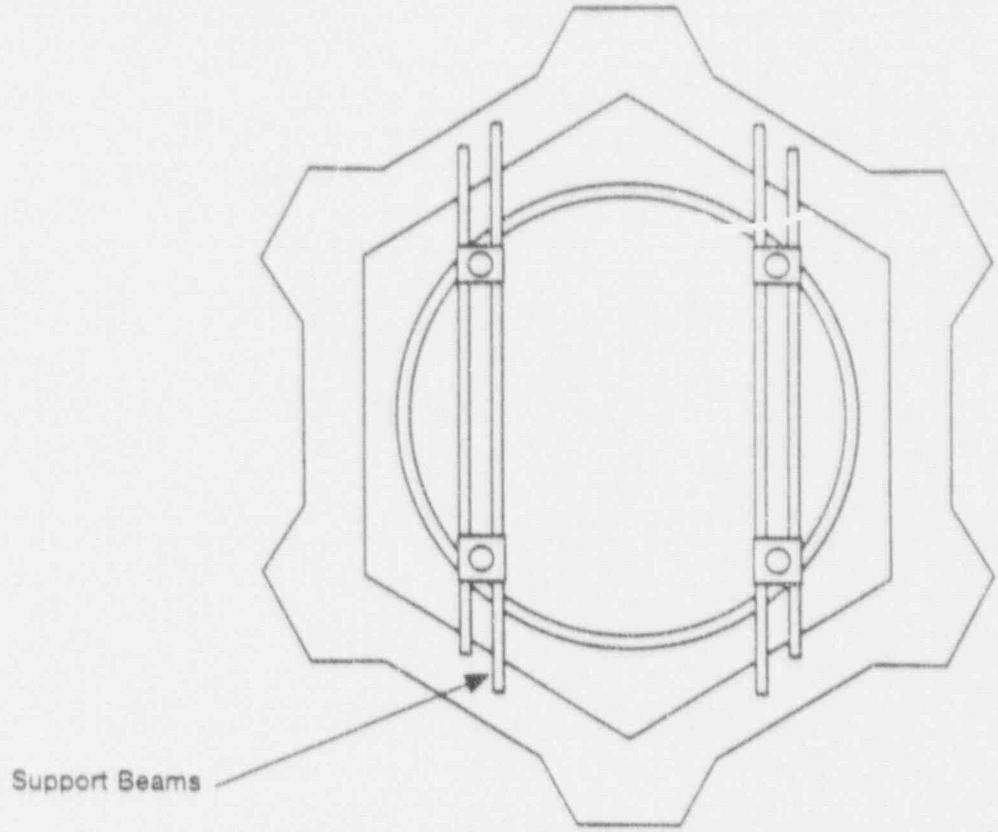


Figure 9.4-1 Core Support Floor - Cut Away View



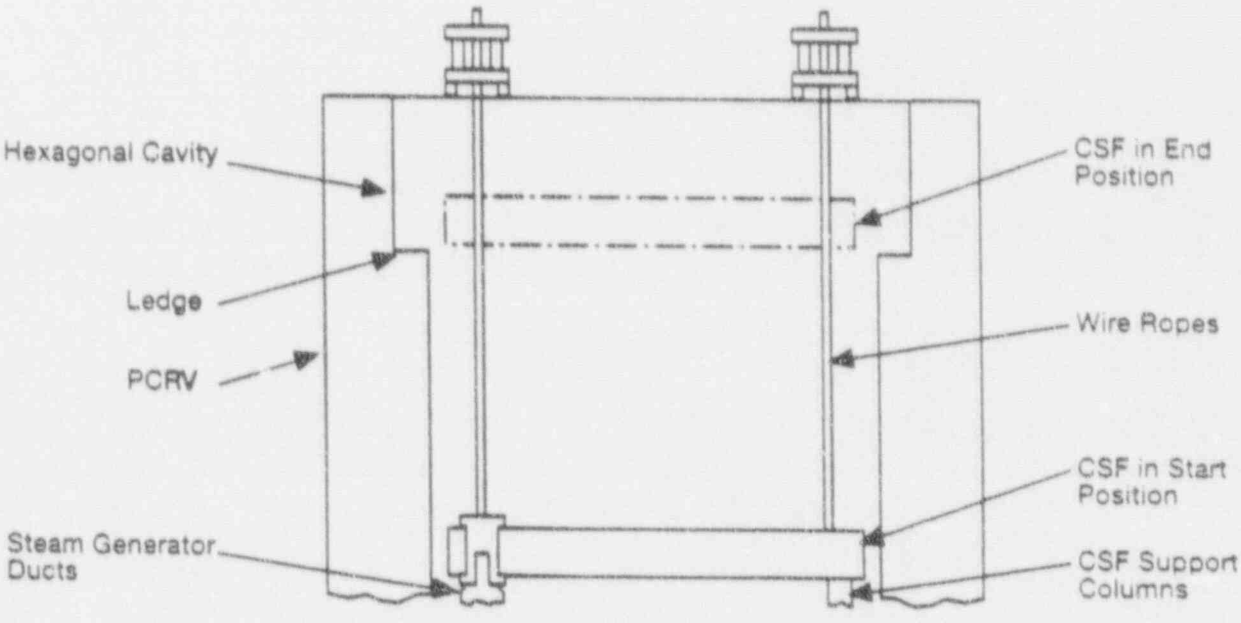


Figure 9.4-2 CSF Four Point Jacking System

9.5 REMOVAL OF STEAM GENERATOR PRIMARY ASSEMBLIES

I. GENERAL DESCRIPTION - STEAM GENERATOR PRIMARY ASSEMBLIES

The twelve (12) steam generator primary modules are contaminated components that will require removal and disposal as part of the decommissioning of the Fort St. Vrain plant. The steam generators are mentioned, without description other than location, in Section 2.2.2 of the PDP. Section 2.3.3.11 and Section 2.3.3.3 of the PDP discusses the removal of the steam generator and Figure 2.3-2 is provided for descriptive purposes. Each primary module is approximately 6 feet in diameter, 26 feet in height and weighs 65,000 pounds. The radiation source is primarily attributable to plate out contamination with a minor contribution from activation. The uppermost portion of the primary steam generator is estimated to have a contact dose rate of 700 mR/hr. The lower portion of the primary steam generator is estimated to have a contact dose rate of 50 mR/hr. Localized hot spots on the generators are estimated to be 2 R/hr on contact.

II. DESCRIPTION OF THE STEAM GENERATOR PRIMARY MODULES REMOVAL PROCEDURE

A. Prerequisites For Primary Modules Removal

The prerequisites necessary to begin work on the steam generator primary modules are as follows:

1. The steam generators have been disconnected from the PCRV penetration flanges.
2. The core support floor has been removed from the PCRV.
3. The steam generator secondary assemblies outside the PCRV have been removed.

B. Removal Activities

The primary modules will have been structurally disconnected from the PCRV penetration flanges during the activity which removed the uncontaminated steam generator secondary assemblies. That task (see PDP Section 2.3.3.3.2) will have been accomplished in an uncontaminated environment with a low radiation field. The primary modules remain connected to the PCRV internals by the connection of the steam generator shrouds to the plenum floor in the lower portion of the PCRV. This connection needs to be detached to allow the lifting of the primary modules from the PCRV. The separation of the steam generator primary modules from the plenum floor is the most complex task associated with the primary steam generator removal tasks.

The PCRV water level will be maintained above the top of the primary module to reduce radiation levels while it is being separated from the plenum floor using remotely operated cutting equipment. However, if radiological surveys of the primary modules indicate that actual radiation and contamination levels are low, the PCRV water level may be lowered to obtain more direct control during the separation of the primary module from the lower plenum. Lowering the PCRV water level will only be considered if it can be justified by an ALARA review.

Due to the expected radiation levels associated with the steam generator primary modules and the limited access in the area of the joint between the primary modules and the lower plenum floor, there is no simple means of making the separation. The method indicated in the PDP is the cutting of the clamp remotely. An alternative method for separating the steam generators from the plenum floor which requires similar precautions and effort is to remotely cut the plenum floor around the attachment location. The cutting would be performed by oxy-lance or plasma torch. The method that is chosen would be based on an evaluation of the performance characteristics of both methods in the limited access in which it will be used. However, either method would be utilized underwater to derive the benefit of the water for shielding the workers from radiation. Fumes from cutting and any potential airborne contamination will be collected by an exhaust hood at the surface of the water.

Prior to separating a steam generator module from the plenum floor, it will be rigged to the Reactor Building crane for lifting. Upon separation, the lifting of the steam generator primary module will begin. As the module is lifted, it will be allowed to drain and will be enveloped with poly film or Herculite to prevent the spread of loose contamination. As discussed in Section 2.3.3.11 of the PDP, the steam generators will be moved to pre-staged containers for packaging for shipment. During the movements of the modules, radiation protection personnel will ensure that distance is maintained between the workers and the source to keep exposures ALARA. The container design will take into account stay times such that radiation exposure will be minimized during packaging of the modules. Local shielding will be utilized as appropriate.

III. RADIOLOGICAL CONSIDERATIONS

A. Occupational Radiation Exposures

The occupational radiation exposures that will result from the cutting and removing of the steam generator primary modules have been estimated to be 17.85 person rems. A breakdown of the total 17.85 person rems into the major tasks that will be performed to remove the CSF is provided in Table 9.5-1.

B. Offsite Exposures From Postulated Accidents

Decommissioning accidents that could result in radiation exposure at the site boundary were postulated and analyzed in Section 3.4 of the PDP. The accident scenarios that were postulated include the following:

- o Dropping of contaminated concrete rubble
- o Heavy load drop
- o Fire
- o Loss of PCRV Shielding Water
- o Loss of power
- o Natural disasters

Section 3.4.5 of the PDP analyzed the most severe heavy load drop accident by postulating the dropping of the component with the largest inventory of dispersible radioactive material, the large side reflector blocks. Credit is taken for decontamination of the particulate afforded by the Reactor Building

Ventilation exhaust system. However, Decommissioning Technical Specification 3.2 [12], "Reactor Building Ventilation Exhaust System", is applicable only when activated graphite blocks have been removed from the PCRV shield water and remain inside the Reactor Building. Consequently, credit cannot be taken for the Reactor Building Ventilation exhaust system in a postulated primary steam generator drop accident. Therefore, an evaluation has been performed to assess the offsite radiological consequences that could result from a postulated accident involving a drop of one primary steam generator module. To analyze this postulated accident, the following major assumptions were used:

1. A worst case atmospheric dispersion factor of 3.53 E-2 sec/m^3 was used.
2. All releases to the environment are ground level releases.
3. The major pathway is air inhalation.
4. An individual is located at the decommissioning Emergency Planning Zone (a minimum of 100 meters from the Reactor Building, the Fuel Storage Building, and the Radioactive Waste Compactor Building) for the duration of the release.

The atmospheric dispersion factor was calculated using the guidelines provided in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants" [6]. NUREG-0172 [7] dose conversion factors for an adult were also used in the calculations. These assumptions are consistent with the assumptions that were used in the analysis of the decommissioning accident scenarios described in Section 3.4 of the PDP.

In order to conservatively estimate the activity levels of radionuclides that are present on the surfaces of the steam generators, the plateout levels predicted by the revised FSV Plateout Analysis for Decommissioning Study [13] were used. This revised plateout analysis fulfills the commitment made in Section 3.1.4.2 of the PDP that, "Additional plateout analyses will be performed to predict primary system inventories based on the actual end of life burnup. The accuracy of the predicted fuel performance and gas release will be assessed by comparison to measured R/B (Release to Birth Rate) data. The accuracy of the predicted fission metal release data will be assessed by comparison to measured plateout probe data."

PDP Table 3.1-5 identifies projected plateout levels of six key radionuclides that are either fission products or come from fission products (Sr-90, Te-127m, I-129, I-131, Cs-134 and Cs-137). The values presented in this table are based on the original FSV Plateout Analysis for Decommissioning Study that is described in Section 3.1.4.2 of the PDP and include projected plateout levels on the reheater, superheater, economizer and evaporator tube bundles of all 12 steam generator modules. PDP Section 3.1.4.2 and Table 3.1-5 will be updated in the next PDP revision to account for the revised FSV Plateout Analysis for Decommissioning Study.

In addition to the six key radionuclides associated with fission products discussed above, it is also necessary to consider activation products which may have accumulated on the steam generator surfaces, such as Mn-54, Fe-55, Co-60, Ni-63, etc. While it is not presently feasible to obtain samples from the surfaces of the steam generator tube bundles, surface concentrations of activation products can be estimated based on samples taken from the surfaces of helium circulators, which are in the primary coolant flow stream.

Although measured surface concentrations of Co-60 were a factor of 15 lower than those measured for Cs-137 on a helium circulator, it was conservatively assumed that the Co-60 inventory on a steam generator module was equal to the Cs-137 inventory that is predicted by the revised FSV Plateout Analysis for Decommissioning Study to be on a steam generator module. The ratio of the surface concentrations of remaining activation products to Co-60 was assumed to be identical to those measured on the surfaces of a helium circulator [14]. Based on these assumptions, the inventory of various radionuclides plated out on the surfaces of a steam generator module were derived and are shown in Table 9.5-2.

It was then conservatively assumed that none of the activity on the surfaces of the steam generator modules is removed while the steam generators are underwater. It was also assumed that 1% of the total surface activity inventory of one steam generator module becomes airborne as a result of the postulated drop accident. Furthermore, a three year decay period was assumed, although the steam generators are not scheduled for removal until about two years into decommissioning, or about 5 years after reactor shutdown. The radionuclides assumed to be released as the result of a postulated drop accident are thus 1% of the values presented in the 3 year decay column of Table 9.5-2.

The whole body and lung doses to an individual standing at a point 100 meters from the Reactor Building were calculated to be 8.3 mRem whole body and 90.7 mRem to the maximum organ (lung). A breakdown of the doses produced by individual radionuclides is presented in Table 9.5-3.

Based on the above, it is concluded that the radiation exposure to the general public as a result of a postulated steam generator module drop accident would be very low. The radiological consequences from the postulated accident are well within the 25 Rem whole body dose and 300 Rem to any specific organ guidelines established in 10 CFR 100. The radiological consequences are also 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 [8].

Section 3.4 of the PDP will be revised in the next PDP update to include a discussion of the postulated steam generator module drop accident and its consequences.

C. <u>Radioactive Waste Generated</u>

A tabular listing of the estimated radioactive waste disposal volume for each WBS was provided in Figure 3.2-1 of the Cost Estimate. As presented in Figure 3.2-1 the estimated disposal volume of radioactive waste from the steam generators (including both the primary and secondary assemblies) is 20,736 cubic feet. The radioactive waste is expected to have a burial class of "A".

IV. SAFETY ANALYSIS CONCLUSIONS

During the removal activities, the radiological hazards will be monitored and evaluated on a routine basis. All work activities associated with the removal of the steam generators will incorporate effective radiological controls to maintain occupational radiation exposures within regulatory limits and as low as reasonably achievable (ALARA), consistent with dose-limiting provisions of 10 CFR 20, as well as both Regulatory Guides 8.8 [9] and 8.10 [10]. All workers will be provided instructions in radiation protection concepts commensurate with the radiological hazards that they will encounter during the removal of the steam generators, including instructions concerning actions required during unusual conditions.

It is estimated that the removal of the steam generator primary modules will result in a total occupational radiological exposure of 17.85 person-Rem. Occupational exposures will not be in excess of allowable 10 CFR 20 limits for occupational radiation exposure. Furthermore, postulated accidents involving the steam generator primary assemblies will result in offsite radiological consequences that are a small fraction of the guidelines established in both 10 CFR 100 and EPA Protective Action Guidelines [8]. It is therefore concluded that the activities associated with the removal and disposal of the steam generator primary modules will not pose an undue risk to the health and safety of the general public nor to the decommissioning workers.

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TABLE 9.5-1
OCCUPATIONAL RADIATION EXPOSURE ESTIMATE
STEAM GENERATOR PRIMARY MODULES REMOVAL

OPERATION	NO. OF WORKERS	EFFECTIVE EXPOSURE RATE (mR/hr)	TASK DURATION (hrs)	WORKER EXPOSURE TIME (hrs)	ESTIMATED EXPOSURE (person-mrem)
Install rigging System	3	2	14 [illegible] 7	44	88.0
Cut modules free	3	2	30.00	90	180.0
Remove segments	4	10	14.67	58.7	587.0
Package	4	5	31.67	126.7	633.0
TOTALS				319.4	1488

Note: Multiplying the 1488 mrem by 12 to account for all of the steam generator primary modules yields the 17.85 person-Rem shown in Table 3.5-2 of the Decommissioning Cost Estimate.

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TABLE 9.5-2
INVENTORY OF RADIONUCLIDES
ON ONE STEAM GENERATOR MODULE[(1)]

ISOTOPE	INITIAL CURIES	HALF-LIFE (years)	FINAL CURIES (3 year decay)
Mn-54	1.74 E+05	8.55 E-01	1.53 E+04
Fe-55	5.20 E+06	2.70 E+00	2.41 E+06
Co-60	5.71 E+05	5.26 E+00	8.48 E+05
Ni-63	9.31 E+04	9.20 E+01	9.10 E+04
Sr-90	3.94 E+04	2.81 E+01	3.66 E+04
Y-90*	1.68 E+05	64 Hours	1.41 E+05
Ru-106	1.47 E+04	1.01 E+00	1.88 E+03
Ag-110m	1.83 E+05	6.90 E-01	9.00 E+03
Cs-134	2.34 E+05	2.06 E+00	8.53 E+04
Cs-137	5.71 E+05	3.01 E+01	5.33 E+05
Ce-144	2.18 E+05	7.78 E-01	1.50 E+04
Pm-147	1.83 E+05	2.62 E+00	8.42 E+04

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

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

TABLE 9.5-3
CONTRIBUTION TO OFFSITE RADIOLOGICAL CONSEQUENCES

SIGNIFICANT ISOTOPE	WHOLE BODY DOSE (mRem)	MAXIMUM ORGAN DOSE - LUNG (mRem)
Mn-54	1.47 E-3	3.28 E-1
Fe-55	1.46 E-1	2.66 E+0
Co-60	1.92 E-1	7.75 E+1
Ni-63	2.02 E-2	2.49 E-1
Sr-90	3.42 E+0	5.38 E+0
Y-90	1.21 E-4	3.66 E-1
Ru-106	2.50 E-4	2.68 E-1
Ag-110m	8.20 E-4	6.38 E-1
Cs-134	9.51 E-1	1.27 E-1
Cs-137	3.49 E+0	6.14 E-1
Ce-144	4.22 E-2	1.79 E+0
Pm-147	3.30 E-2	6.80 E-1
Other	≤0.01	0.10
TOTALS	8.30	90.7

9.6 REFERENCES

1. PSC letter, Crawford to Weiss, dated December 6, 1991, "PSC Response to NRC RAI on the Fort St. Vrain Decommissioning", (P-91423).

2. PSC letter, Crawford to Weiss, dated January 9, 1992, "PSC Response to NRC R on the Fort St. Vrain Decommissioning", (P-92014).

3. 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).

4. PSC letter, Crawford to Weiss, dated April 26, 1991, "PSC Response to NRC RAI on the Fort St. Vrain Decommissioning", (P-91118).

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

6. USNRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants", Rev. 1, November 1982.

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

8. 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.

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

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