Viacom Inc.
11 Stanwix Street
Pittsburgh, PA 15222-1312

Richard K. Smith
Vice President – Environmental Remediation

Telephone: 412-642-3285
Facsimile: 412-642-3957
E-mail: Richard.Smith@viacom.com

VIACOM.

October 29, 2002

U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001
ATTN: Document Control Desk

Subject: Viacom Inc.
Westinghouse Test Reactor TR-2, Docket No. 50-22
Application for NRC Orders (1) Terminating 10 CFR Part 50 Portion of TR-2 License and (2) Declaring that Decommissioning of TR-2 Structures Has Been Satisfactorily Completed

References:

1. "Westinghouse Test Reactor TR-2 Final Decommissioning Plan" (approved by Amendment 8 of utilization facility license No. TR-2), as revised (Revision 1) January 2000.

2. NRC letter (Mr. Patrick M. Madden, Office of Nuclear Reactor Regulation) to Viacom (Mr. Richard K. Smith) dated September 6, 2002, enclosing NRC Inspection Report No. 50-22/1999-202.

3. Westinghouse letter (Marlene W. Jackson, Esq.) to Viacom (William D. Wall, Esq.), dated December 20, 2000.

4. Michael F. McBride, Esq. (on behalf of Viacom) letter to F. Ramsey Coates, Esq. (Westinghouse), dated September 19, 2002.

5. Richard G. Murphy, Jr., Esq. (on behalf of Westinghouse) letter to Michael F. McBride, Esq., dated September 30, 2002.

Dear Sirs:

Viacom Inc. ("Viacom") hereby submits an application for two related NRC Orders: (1) an Order terminating the 10 CFR Part 50 utilization facility license (TR-2) held by Viacom for the Westinghouse Test Reactor facility ("WTR") on the Waltz Mill Site in Pennsylvania; and (2) an Order declaring that Viacom's obligations to decommission the WTR in accordance with the NRC approved final decommissioning plan (Reference 1) ("DP") have been completed satisfactorily except for actions which require the cooperation of Westinghouse Electric Company LLC ("Westinghouse"). The purposes of this application are (1) to recognize that, with the completion the decommissioning work on WTR structures required by the DP and described further below, the WTR no longer constitutes a utilization facility within the meaning of section 11cc. of the Atomic Energy Act of 1954, as amended, and 10 CFR § 50.2 of NRC's

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regulations, and (2) to document the progress that has been made in decommissioning of the WTR in accordance with the DP.

The WTR pressure vessel and pressure vessel internals have been removed in accordance with the DP, as has all of the biological shield that needed to be removed in order to remove the pressure vessel. This is documented in NRC Inspection Report 50-22/1999-202 (Reference 2). The referenced Inspection Report concludes, however, that two provisions of the DP still need to be accomplished--determining the residual radioactivity remaining *in situ* and preparing the necessary documents to effect the transfer of these materials to the SNM-770 license.[1]

At the time the DP was submitted and approved, the same entity held both the TR-2 and SNM-770 licenses, and so the transfer of the residual radioactive material from one materials license (the Part 30 portion of TR-2) to another (SNM-770) held by the same licensee on the same site was straightforward. The two licenses are now held by entirely different entities (Viacom holds TR-2 while Westinghouse holds SNM-770), and Viacom's and Westinghouse's decommissioning responsibilities to each other at the Waltz Mill Site are set forth in an Asset Purchase Agreement, as NRC recognized on March 10, 1999 when it approved of various transfers and amendments and acknowledged various financial assurances provided by Viacom. Westinghouse does not currently agree with Viacom about the scope of Viacom's decommissioning responsibilities at the Waltz Mill Site and so, unfortunately, Westinghouse not only refuses to accept any transfer of the residual materials to its SNM-770 license (Reference 3), but also refuses to supply Viacom with the survey of residual radioactive materials which it prepared as Viacom's Project Manager for the TR-2 decommissioning project (NRC References 4 and 5). These refusals prevent Viacom from complying fully with the DP.

However, several useful steps are possible to document and make progress on decommissioning of TR-2 notwithstanding Westinghouse's refusals, and without unnecessarily embroiling NRC in a commercial dispute between parties. First, the 10 CFR Part 50 portion of the TR-2 license can be terminated by NRC Order, and Viacom so requests. The TR-2 license, like all other NRC facility licenses, is actually a combination of two different types of licenses. TR-2 authorizes Viacom to possess, but not operate, the WTR facility as a utilization facility (nuclear reactor) under 10 CFR Part 50, and TR-2 also authorizes Viacom to possess such byproduct materials "as may be contained in the structural parts of the facility" under 10 CFR Part 30. As stated above, the WTR pressure vessel and pressure vessel internals have been removed in accordance with the DP, as has all of the biological shield that needed to be removed in order to remove the pressure vessel. With the completion of these decommissioning activities (if not before), the WTR no longer constitutes a utilization facility requiring a Part 50 license, that is, it is no longer "an apparatus ... designed or used to sustain nuclear fission in a self-supporting chain reaction." 10 CFR § 50.2. Since a 10 CFR Part 50 utilization facility license is no longer authorized or required for the WTR, the Part 50 portion of TR-2 can and should be terminated.

10 CFR § 50.82(b)(6) is not strictly applicable to this request since the TR-2 license is not requested to be terminated in its entirety, and termination as requested will not result in a

[1] The DP is not clear on the need for a survey of the *in situ* radioactive materials before the materials are transferred. Section 4 of the DP, "Proposed Final Survey," states that "[u]pon completion, no materials covered by the 10 CFR 50 license will exist" and that [t]he method for determining that the WTR facility has met the decommissioning objectives and prerequisites for license termination will be an independent verification that the reactor vessel internal contents, the reactor vessel, and the biological shield have been removed." Nevertheless Viacom understands NRC to be requesting a survey of *in situ* materials.

unrestricted or restricted release of the Waltz Mill Site under 10 CFR Part 20, subpart E. It was never the concept of the DP that completion of decommissioning under it would lead to application of Part 20, subpart E. Instead, as the DP provides, it was envisioned that Part 20, subpart E would be applied at a later date, some 25 years into the future, when active operations at the Waltz Mill Site under SNM-770 ceased.

However, if NRC should decide that 10 CFR § 50.82 (b)(6) applies, it should grant an exemption from the requirements of 10 CFR § 50.82(b)(6) in accordance with 10 CFR § 50.12. Exhibit A sets forth a justification for such an exemption. NRC granted a similar exemption in connection with the termination of the University of Illinois reactor license No. R-117, Docket No. 50-356, by order dated July 16, 1997, and so the requested exemption would be in accord with NRC practice.

Termination of the 10 CFR Part 50 portion of the TR-2 license will not result in a loss of NRC licensing authority over the WTR because the 10 CFR Part 30 portion of the license will remain. After termination of the Part 50 portion of the TR-2 license (paragraph 2.a.), paragraph 2.b. of the license (authorizing possession of residual radioactive materials in structural parts of the facility) will remain. Alternatively, those portions of the TR-2 license applicable to byproduct materials could be restated in the usual format for an NRC materials license and a new materials license number could be assigned. The "Authorized use" in paragraph 9 of NRC Form 374 would be "possession only," as in the TR-2 license, and all of the applicable conditions in TR-2 can be recast as license conditions in NRC Form 374, paragraphs 10 and following.[2]

Since all of the requirements in TR-2 applicable to the Part 30 portion of the license will remain in effect, and will merely be restated in a new license format, no amendments to the Part 30 license are intended or necessary. A formal application for a materials license for the residual materials is also not needed since, as explained above, Viacom already has one as part of TR-2. However, to facilitate NRC's consideration of this request, a description of the remaining TR-2 structures is provided in Exhibit B.

Viacom maintains a level of decommissioning financial assurance for its TR-2 license. Viacom does not propose, as part this request, to change that amount which, upon the grant of our request, would be associated with the Viacom materials license for the residual materials. However, once the Part 50 portion of the TR-2 license is terminated, no indemnity agreement is required by 10 CFR Part 140.

Our second requested Order follows logically from the first. If, as requested, NRC terminates the 10 CFR Part 50 portion of the TR-2 license, it will be because NRC has determined that all of the requirements in the DP has been satisfied, except for the transfer of residual materials to SNM-770 and the filing of the survey of the TR-2 residual materials, neither of which can be accomplished at present because of Westinghouse's position. Our requested Order would simply have NRC declare that this is the case. Specifically, Viacom asks NRC to determine (a) that all

[2] The "Technical Specifications" in the TR-2 license would need to be re-titled as "License Conditions" and made expressly applicable, subject to the licensee's ability to make changes in accordance with limitations which are the functional equivalent of the limitations in 10 CFR 50.59. (Suitable limitation language can be borrowed from condition 25 of SNM-770.) Similarly, the DP should also be included as a license condition in the Part 30 license since enforceability of the DP under 10 CFR 50.59 will no longer be appropriate once the Part 50 portion of TR-2 is terminated and only a Part 30 license remains. Moreover, incorporation of the DP into the license will serve to document all of the steps that are now required in order to terminate the Viacom Part 30 license--the survey and the transfer of the residual materials to SNM-770. Further decommissioning activities for these materials can then be carried out under SNM-770, as the DP contemplated.

of Viacom's obligations under the DP have been satisfied, except for the transfer of residual radioactive materials formerly held under TR-2 and submission of the survey, and (b) that it is prepared to entertain the appropriate submissions from Westinghouse (if necessary through Viacom) that are needed to complete decommissioning under the DP. Orders of this sort are authorized by section 161 of the Atomic Energy Act of 1954, as amended. Moreover, the requested order is consistent with the NRC's policy to issue such orders as are needed to avoid unnecessary delay and uncertainty. *See Public Service Co. of New Hampshire (Seabrook Station, Units 1 and 2)*, CLI-778, 5 NRC 503, 516 (1977).

Since the requested orders will merely document the status quo, and will not change any requirements applicable to the residual radioactive materials now covered by the TR-2 license, there will be no environmental impact resulting from the requested actions.

This application should not embroil NRC in the commercial dispute between Viacom and Westinghouse. The license termination will simply reflect the status quo as NRC sees it, and NRC's decision that it is prepared to receive the survey and to entertain the request for license transfer to SNM-770 will not constitute an NRC opinion that Westinghouse is contractually obligated to do these things under the Asset Purchase Agreement or any other agreement between the parties. However, as reference 2 indicates, while Viacom believes that it has completed all of the decommissioning required under the DP, except for submission of the survey of residual materials and transfer of those residual materials, Westinghouse has a contrary view that the remainder of the biological shield must still be removed. As this dispute involves interpretation of an NRC requirement (the DP), Viacom believes NRC is uniquely capable of resolving it. Whether the DP requires that all of the biological shield be removed or whether, as Viacom believes, only requires removal of that portion of the shield that was necessary for removal of the pressure vessel, can be determined from a review of the DP, especially Revision 1, January 2000, sections 2.1, 2.2.2, 2.2.2.4, and 2.2.3.

On this date Viacom is also filing a petition under 10 CFR § 2.206 which asks NRC to issue an order to Westinghouse that would require Westinghouse to accept the residual radioactive materials from TR-2 under its SNM-770 license (including if necessary submitting any necessary amendments to SNM-770), and to submit the residual radiation survey data to NRC, as NRC requested. The application herein can and should be considered by NRC as a matter separate from the 2.206 petition.

For NRC's convenience, copies of references 3, 4 and 5 are enclosed.

Sincerely,

Viacom Inc.

By Richard K. Smith

Richard K. Smith
Vice President - Environmental Remediation
Viacom Inc.

Enclosures

cc: U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406-1415

Dr. Ronald R. Bellamy
Chief, Decommissioning and Laboratory Branch
Division of Nuclear Materials Safety
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Patrick M. Madden, Section Chief
Mr. Alexander Adams, Jr., Senior Project Manager
Mr. Stephen W. Holmes, Reactor Inspector
Research and Test Reactors Section
Operating Reactor Improvements Program
Office of Nuclear Reactor Regulation
Division of Regulatory Improvement Programs
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Mr. James G. Yusko
Pennsylvania Department of Environmental Protection
400 Waterfront Drive
Pittsburgh, PA 15222

Mr. A. Joseph Nardi, Licensing Administrator
Westinghouse Electric Company, LLC
P.O. Box 355
Pittsburgh, PA 15203-0355

Martin G. Malsch, Esq.
Michael F. McBride, Esq.
John W. Lawrence, Esq.
LeBoeuf, Lamb, Greene & MacRae, L.L.P.
1875 Connecticut Avenue, N.W., Suite 1200
Washington, DC 20009-5728

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF ALLEGHENY)

Before me, the undersigned notary public, this day personally appeared Richard K. Smith, Vice President, Environmental Remediation, Viacom Inc., 11 Stanwix Street, Second Floor, Pittsburgh, PA 15222-1384 to me known, who being duly sworn according to law, deposes and says that the statements sworn to in this letter and enclosures are correct and accurate to the best of his knowledge.

Richard K. Smith

Signature of Affiant

Subscribed and sworn to before me this 29th day of October, 2002

Marie A. Podvorec

Notary Public

Notarial Seal
Marie A. Podvorec, Notary Public
Pittsburgh, Allegheny County
My Commission Expires Dec. 8, 2003
Member, Pennsylvania Association of Notaries

WESTINGHOUSE ELECTRIC COMPANY

WESTINGHOUSE TEST REACTOR

TR-2

FINAL DECOMMISSIONING PLAN

Revision 1
January 2000

WALTZ MILL
REMEDIATION PROJECT
CONTROLLED DOCUMENT
COPY NO 001
ISSUE DATE 01/2000

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LIST OF ACRONYMS

ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
CFR	Code of Federal Regulations
CRD	Control Rod Drive
CRDM	Control Rod Drive Mechanism
DOT	Department of Transportation
EPA	Environmental Protection Agency
GST	General Site Training
HEPA	High Efficiency Particulate Air
HP	Health Physics
LSA	Low Specific Activity
MDA	Minimum Detectable Activity
MWt	Megawatts Thermal
NIOSH	National Institute of Safety & Occupational Health
NIST	National Institute of Standards and Technology
NRC	Nuclear Regulatory Commission
NSD	Nuclear Services Division
NVLAP	National Voluntary Laboratory Accreditation Program
PADEP	Pennsylvania Department of Environmental Protection
PCM	Project Control Manual
PMP	Project Management Plan
PQP	Project Quality Plan
QA	Quality Assurance
RP	Radiation Protection
RSO	Radiation Safety Officer
RWP	Radiation Work Permit
RWT	Radiation Worker Training
SEG	Scientific Ecology Group
SNM	Special Nuclear Material
SRD	Self-Reading Dosimeters
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent Dosimeter
TR	Test Reactor
WTR	Westinghouse Test Reactor

SECTION 1
GENERAL INFORMATION

The Westinghouse Electric Corporation (Westinghouse) Test Reactor (WTR) is located on the Waltz Mill Site near Madison, Pennsylvania. The retired WTR is currently licensed under Nuclear Regulatory Commission (NRC) License TR-2. The balance of the Waltz Mill Site is licensed and operated under NRC License SNM-770.

Westinghouse has developed a detailed Decommissioning Plan (Plan), based on a Conceptual Decommissioning Plan (Ref. 1), to address the activities required to terminate the TR-2 License. It is considered reasonable and prudent that the activities required for license termination are: removal of the remaining reactor vessel internal contents, the reactor vessel, and the biological shield. Following these decommissioning activities, Westinghouse will request transfer of the remaining residual radioactivity and WTR facilities to the SNM-770 License. This Plan describes these decommissioning activities and the required interfaces with the SNM-770 licensed site.

Prior to or during TR-2 license termination, the SNM-770 License will be amended to include the plans and costs for remediation of the structures, materials, and equipment transferred from the TR-2 License. Future use of these structures, materials, and equipment shall be in accordance with the SNM-770 license conditions and site procedures controlling occupational exposure and exposure to the public.

This Plan has been prepared using Draft Regulatory Guide DG-1005, "Standard Format and Content for Decommissioning Plans for Nuclear Reactors" (Ref. 2) and the applicable regulatory requirements associated with 10 CFR 50.82(b). Although DG-1005 is still in draft form, it is considered appropriate for the development and general format of the Plan. The standard format of DG-1005 has been slightly altered for consistency with the Waltz Mill Facility SNM-770 Remediation Plan, previously submitted to the NRC on November 27, 1996 (Ref. 3).

1.1 LICENSE INFORMATION

License: TR-2

Docket Number: 50-22

Location of Use: Westinghouse Electric Corporation
Waltz Mill Site
Interstate 70 - Madison Exit 25A
P.O. Box 158
Madison, PA 15663

Licensee of Use: Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, PA 15230

Licensee Contact: Mr. A. Joseph Nardi
Westinghouse Electric Corporation
P.O. Box 355
Pittsburgh, PA 15230

The initial WTR operating license was issued on June 19, 1959. Amendment Number 1 to the operating license, dated January 8, 1960, authorized maximum thermal power to be raised from 20 MWt to 60 MWt. Westinghouse informed the NRC that the WTR had permanently ceased operations on March 22, 1962. The following license amendments were issued after permanent cessation of operations:

Amendment Number 2 dated March 25, 1963 - The TR-2 License was amended to allow possession, but not use of the reactor (Possession Only License).

Amendment Number 3 dated April 22, 1970 - The TR-2 License was amended to transfer the Truck Lock Building to the SNM-770 License.

Amendment Number 4 dated June 24, 1970 - The TR-2 License was amended to transfer the Facilities Operations Building to the SNM-770 License.

Amendment Number 5 dated April 17, 1974 - The TR-2 License was amended to extend the license termination to November 30, 1993. Since then a timely license renewal letter was sent to the NRC on December 8, 1992 requesting license extension to November 30, 2003. NRC action is pending on this request.

Amendment Number 6 dated June 14, 1993 - The TR-2 License was amended to transfer the three WTR Basins (No. 1, 2, and 3) to the SNM-770 License.

1.2 DECOMMISSIONING OVERVIEW

This Plan describes the objectives, activities, and controls that will apply to the decommissioning of the WTR. The ultimate objective is to terminate the TR-2 License. To accomplish this, the following are the major decommissioning activities:

1. Remove the remaining reactor vessel internal contents, the reactor vessel, and the biological shield.

2. Provide the NRC with sufficient documentation to demonstrate that license termination requirements have been met. This would include all documentation that is required for transfer of the remaining residual radioactivity and WTR facilities to the SNM-770 License.

Additionally, decontamination and dismantlement activities of other structures and equipment associated with TR-2 may be performed in accordance with this plan. Those activities not completed under this plan will be completed after being transferred to the SNM-770 license. The approved acceptance criteria associated with the retired facilities in the SNM-770 Remediation Plan will also be used for these other areas.

1.3 FACILITY AND SITE DESCRIPTION

The Waltz Mill site is located approximately 30 miles southeast of Pittsburgh in Westmoreland County, Pennsylvania (see Figure 1-1). The site is approximately 850 acres and is located about three miles west of the town of New Stanton between the towns of Madison and Yukon (see Figure 1-2). The WTR facility is located in the northwest portion of the Waltz Mill site, north of the G Building (see Figure 1-3).

The Waltz Mill site is operated by the Nuclear Services Division of the Westinghouse Energy Systems Business Unit. The WTR is maintained under NRC License Number TR-2 (Possession Only), encompassing the requirements of 10 CFR 50. The WTR license includes the reactor structure, reactor systems, the reactor containment building, the rabbit pump room, the sub-pile room, the polar crane, and the WTR portion of the transfer canal.

The WTR was a low pressure, low temperature, water cooled 60 MWt reactor housed in a cylindrical vapor containment structure (see Figure 1-4). Since permanent shutdown in 1962, all fuel and some of the reactor internal contents have been removed from the reactor vessel and from the Waltz Mill site. The reactor vessel has been drained of all water and the vessel head is secured on the vessel. The Site, including the WTR facility, has been extensively characterized and is controlled to not pose a threat to the health and safety of the site worker or the general public.

1.4 ADMINISTRATION OF THE DECOMMISSIONING PLAN

This Decommissioning Plan provides sufficient detail of the WTR decommissioning activities to allow NRC review and approval. The provisions of 10 CFR 50.59(e) shall apply to the NRC approved Decommissioning Plan and the criteria to be used in evaluating changes to the Plan will be included in project procedures.

REFERENCES FOR SECTION 1

1. Westinghouse letter, Nardi to NRC, dated April 7, 1997; Subject: "Submittal of Remediation Plan for the Westinghouse Test Reactor, USNRC License Number TR-2, Docket 50-22."

2. Westinghouse letter, Nardi to Bellamy (NRC), dated November 27, 1996; Subject: "Submittal of Remediation Plan for the Westinghouse Waltz Mill Site, USNRC License Number SNM-770, Docket 70-698."

3. NRC Draft Regulatory Guide DG-1005, "Standard Format and Content for Decommissioning Plans for Nuclear Reactors," September 1989.

Figure 1-1
MAP OF AREA SURROUNDING WALTZ MILL SITE

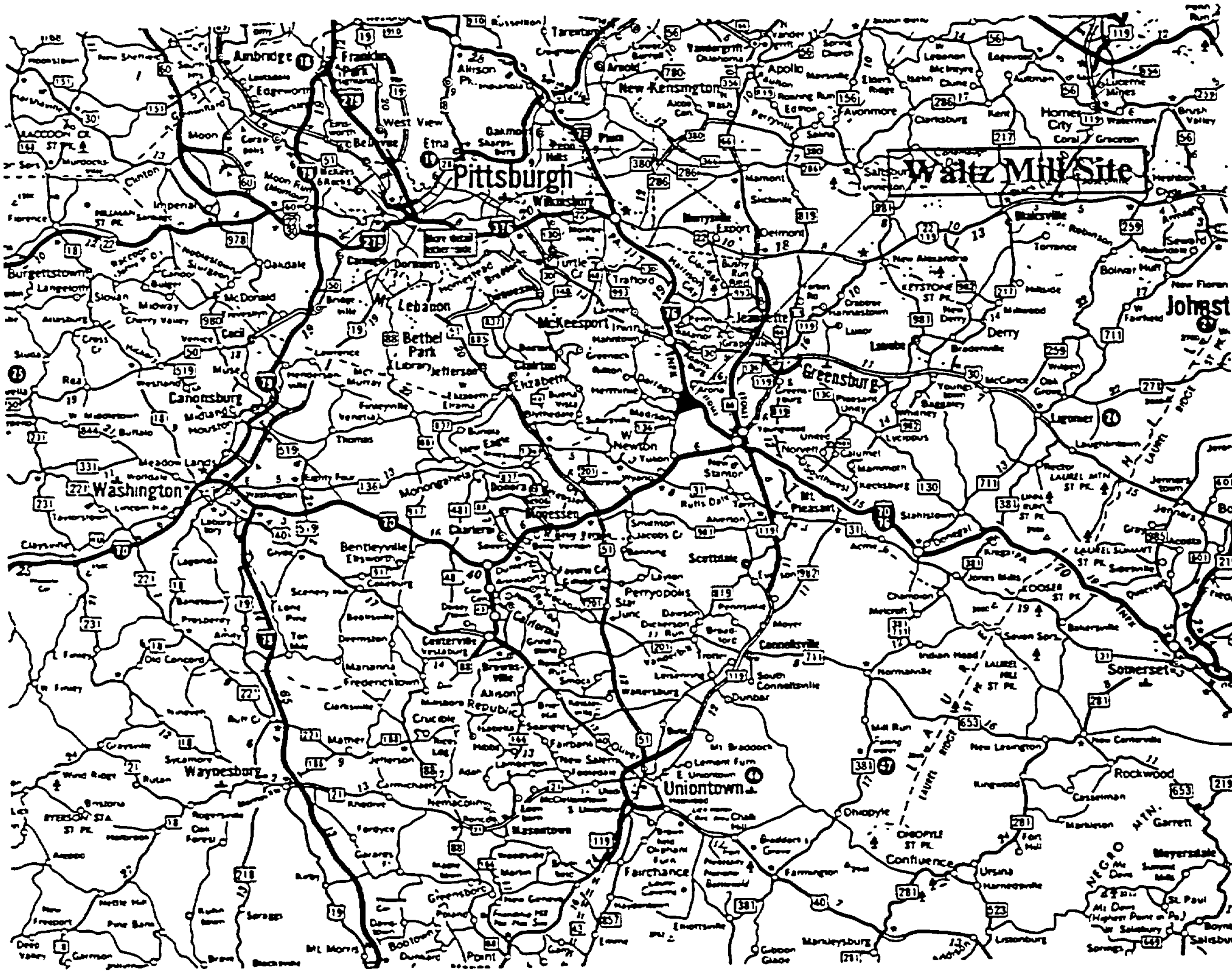


Figure 1-2
OVERALL MAP OF WALTZ MILL SITE

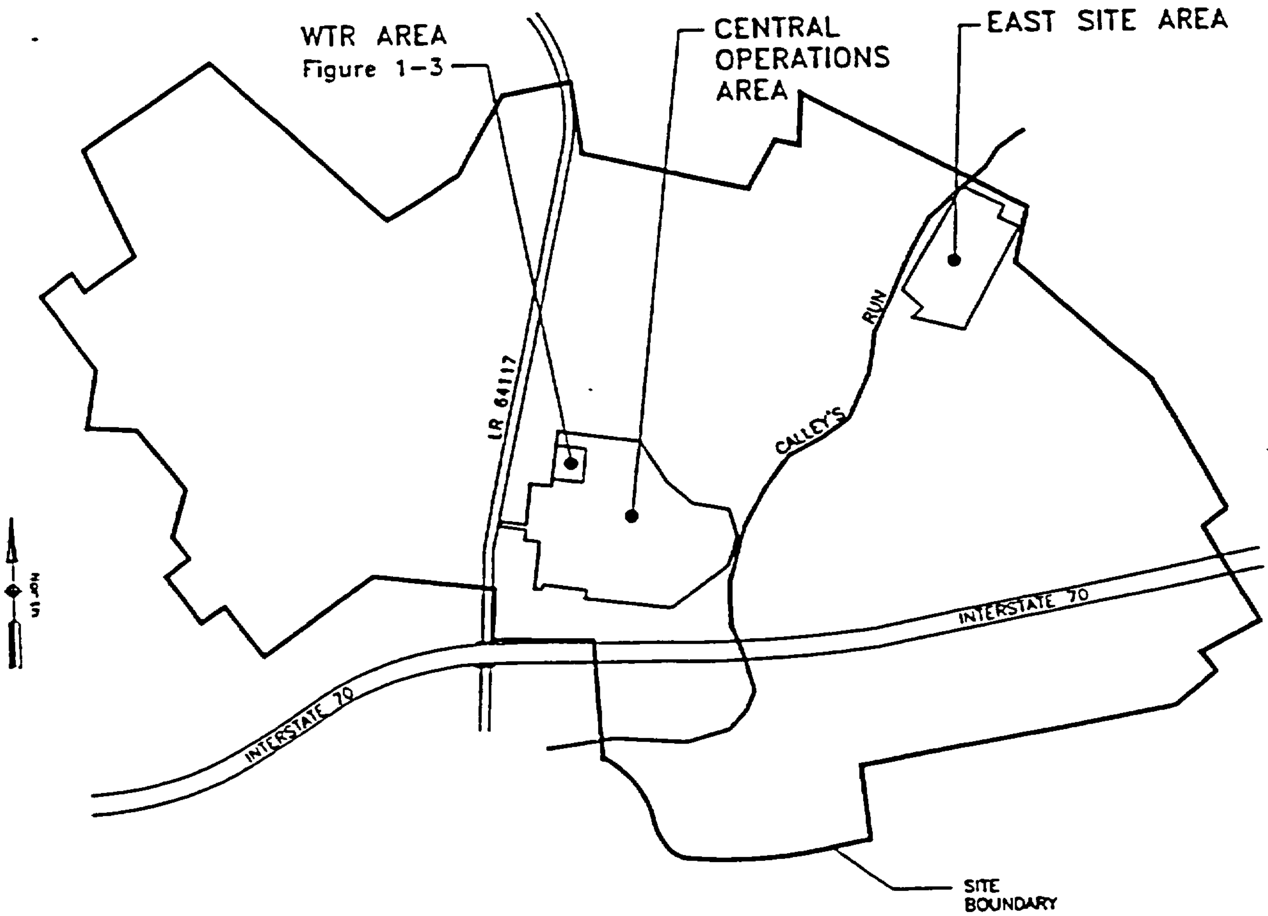


Figure 1-3
WTR AREA

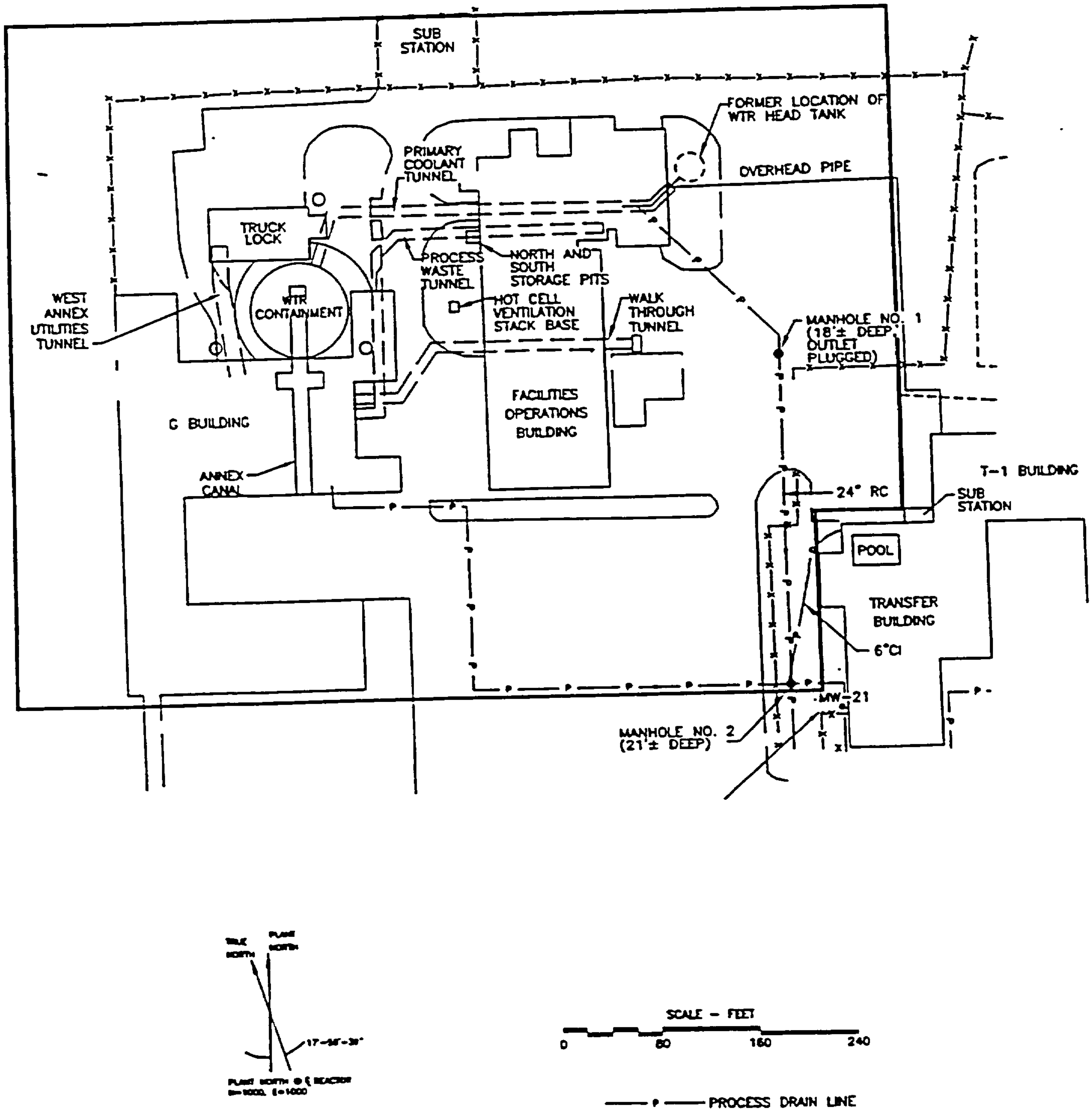
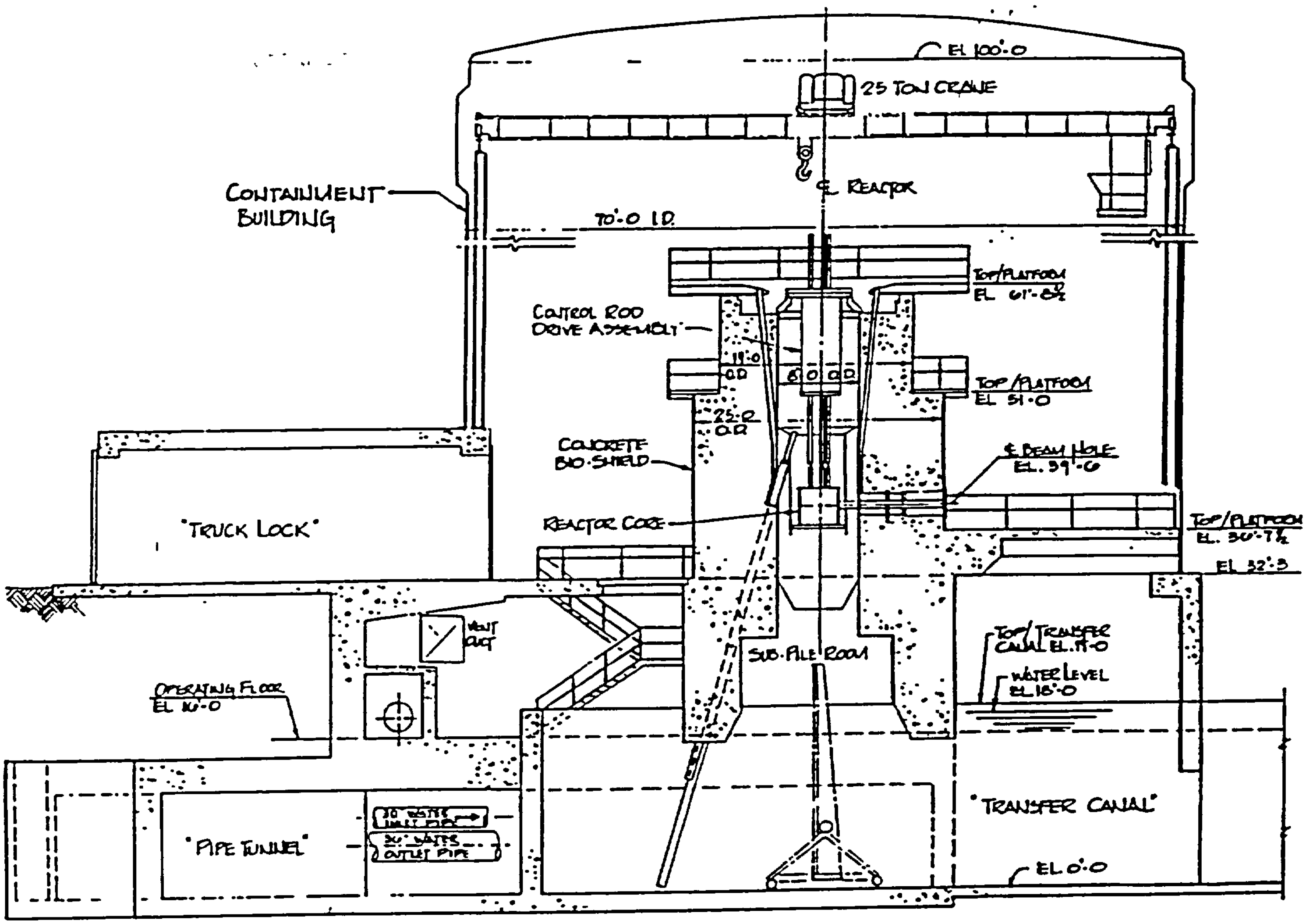


FIGURE 1-4
CROSS-SECTIONAL VIEW OF WTR LOOKING EAST



SECTION 2
CHOICE OF DECOMMISSIONING METHOD AND DESCRIPTION OF ACTIVITIES

2.1 DECOMMISSIONING METHOD

Decommissioning, as described in this Plan, will be accomplished by removal and disposal of portions of the biological shield, the reactor vessel and the reactor vessel internal contents. The balance of the WTR facility components and the remaining residual radioactivity will be transferred to the SNM-770 License. There are no radiological limits applicable to the transfer of structures, materials, and equipment to the SNM-770 License, other than the radioactive materials possession limits specified in the SNM-770 License. Prior to the transfer, the SNM-770 License will be amended as necessary to include the remaining WTR associated radioactive material inventory. Additionally, any other document revisions required as a result of this transfer will be performed. Future use of these structures, materials, and equipment shall be appropriately maintained in accordance with the SNM-770 license conditions and site procedures controlling occupational and public exposure.

In addition to removing portions of the biological shield, the reactor vessel and the reactor vessel internal contents, decontamination and dismantlement activities may be performed on other structures and equipment located within the WTR containment building. These other activities are not required for WTR decommissioning; however, they are addressed herein as optional activities that may be undertaken under the authority of the TR-2 Decommissioning Plan, prior to transfer of remaining residual radioactivity and WTR facilities to the SNM-770 License. The approved acceptance criteria associated with the retired facilities in the SNM-770 Remediation Plan will also be used for these other areas.

Precedent for transferring the residual radioactivity to the SNM-770 License has already been established by Amendment Numbers 3, 4, and 6 to the TR-2 License. These Amendments transferred previous WTR facilities to the SNM-770 License (Truck Lock Building, Facilities Operations Building, and WTR Basins).

2.2 DECOMMISSIONING OBJECTIVE, ACTIVITIES, METHODS AND SCHEDULE

2.2.1 Decommissioning Objectives

The objective of this Decommissioning Plan is to outline the activities for removal of the WTR reactor vessel internal contents, the reactor vessel, and the biological shield, to the point where the TR-2 License can be terminated by transferring the remaining residual radioactivity and WTR facilities to the SNM-770 License. Decommissioning will be by removal, dismantlement, decontamination, release of clean items and disposal of contaminated waste

2.2.2 Decommissioning Activities

The general activities needed to complete the Plan objectives are:

- Remove portions of the biological shield, the reactor vessel and the reactor vessel internal contents.
- Prepare the decommissioning generated material for release or disposal; either decontaminate and release as non-radioactive waste, or package for transport as radioactive waste.
- Ship all radioactive waste off-site to a licensed waste processor or disposal facility. In the event that no acceptable licensed disposal facility is available, waste may be retained onsite or, after processing, returned to the site for interim storage.
- Determine the residual radioactivity remaining and prepare the necessary amendments to the SNM-770 License.
- Request transfer of the remaining residual radioactivity and WTR facilities to the SNM-770 License.
- Request termination of the TR-2 License.

The Plan includes examples of decontamination techniques, equipment and materials which may be used, a schedule, special training requirements for workers, radiation protection and occupational safety and health practices. Selection of decommissioning methods is heavily influenced by worker and public ALARA considerations. A list of WTR facilities, planned decommissioning and decontamination activities and estimated worker exposure (person-rem) is presented in Table 2-1.

Work plans will be prepared to address issues such as asbestos, lead, or other known hazardous materials in the area of work. The final decommissioning methods will utilize the best, most economical means to minimize hazardous, mixed and radioactive waste volume requiring licensed disposal. From the standpoint of cost-effectiveness, contaminated equipment, materials, etc., may be decontaminated, allowing release for unrestricted use, or packaged for transport and disposal. This Plan allows flexibility in the choice of decontamination procedure/technique and sequence.

2.2.2.1 Pre-decommissioning Activities

Three alternative methods for removing the WTR reactor vessel are under consideration that potentially affect pre-decommissioning activities. These three options are 1) one-piece removal through the containment dome , 2) multiple piece removal through the truck lock, and 3) one-piece removal through the truck lock. The first removal option involves cutting an opening in the containment building and lifting the reactor vessel and part of the biological shield out of the containment building with an external crane. The second removal option involves sectioning the reactor vessel and the biological shield concrete into pieces that can be removed through the truck lock with the existing overhead crane. The third removal option involves cutting away a portion of the biological shield and downending the reactor vessel out through the truck lock. As discussed below, these three alternatives have different impacts on activities to upgrade the existing crane, maintain integrity of the containment building, and install a filtered ventilation system.

Access Control

Initial access to the WTR facility will be established through the existing air locks that separate the WTR from the G Building Annex. This access could be used for equipment and material required for the installation of a new HEPA filter system in the existing containment building and for any required repairs to the interior truck lock door. Following HEPA installation and operational verification of the filtration system, the majority of equipment and material access to the WTR will be through the adjacent truck lock on the north side of the reactor, except for any materials removed through a temporary containment building access opening, if the first removal method is used.

The east air lock will continue to be used as the main control point for personnel access to the containment building. A change area will be provided at the entrance to the east air lock in the annex to route personnel upstairs and out through the annex building (see Figure 2-1). Personnel access to the containment building may also be provided through the truck lock.

HEPA Filtration/Ventilation System

A HEPA filtration/ventilation system will be installed. This system will be capable of creating a negative air pressure within the containment building when personnel access airlock doors are open. In addition, this system will be capable of maintaining an inward airflow within the containment building during times when a large component removal hole (if installed) or if the truck lock door is open in the containment building.

Truck Lock Door

Electrical service will be re-established and repairs made to the truck lock door motor and hoist to allow controlled equipment and material access to and from the containment area.

Temporary Utilities

Temporary lighting and power will be installed in accordance with applicable requirements, as well as local safety codes. Some existing electrical systems may be used, after inspection and repairs.

Polar/Mobile Crane

The polar crane and components may be repaired and/or upgraded, as necessary, to allow for safe operation throughout the decommissioning activities. Prior to use, the manufacturer or qualified inspector will certify the crane and components for safe operation, including performance of necessary load tests.

An alternative to using the overhead crane is using a mobile crane operated from outside of the containment building for reactor vessel one-piece removal as described in the first option. This requires that a hole be cut in the containment building roof to allow the crane to access the sectioned components for lifting. If cutting is required after the containment vessel is breached, additional engineering or administrative controls will be used.

Decommissioning Activity and Associated Person-rem

Each decommissioning activity has an estimated worker exposure calculated for that task which is dependent on labor loading, decommissioning method, and known radiological conditions. The decommissioning methods selected strive for ALARA exposures to the workers. These estimated doses are presented in Table 2-1 at the end of this section.

2.2.2.2 Additional Material Handling Capabilities

General

To facilitate safe and efficient material handling capabilities, temporary support structures may be assembled and installed. This may include providing a method for easy transportation of heavy and/or bulky materials and equipment out of the containment building, as well as providing an additional temporary containment (auxiliary area) adjacent to the truck lock.

Temporary Transportation System

Heavy and/or bulky materials which require removal from the containment building, such as sectioned concrete, reactor vessel and components, etc., may require additional transportation capabilities. The addition of a rail cart or similar capacity transportation device in the truck lock area will allow the safe and efficient removal of the material to a staging area in preparation for transport or to other areas for further processing. An example of such a transport system, in this case a rail cart, is shown in Figures 2-2 and 2-3.

Adjacent Auxiliary Area

A temporary auxiliary area adjacent to the truck lock building (shown as Tented Area in Figure 2-2) may be utilized to process material removed from the containment building. This auxiliary area will be covered by a temporary building. The area may be used to decontaminate material, survey and/or sample material, section or segregate clean from contaminated material, and/or package material for transportation to an off site processing location or a licensed disposal facility. This area may be necessary due to space constraints within the containment building and will allow dismantling activities to progress with minimal interruption.

The temporary auxiliary area will be fully contained and provided with a HEPA ventilation system sufficient to maintain a negative pressure within the area while materials are processed. Procedures and/or work plans will describe acceptable methods for movement of materials into and out of the auxiliary area.

2.2.2.3 Removal of Hazardous Materials

Lead

Approximately 266,000 pounds (385 cubic feet) of lead in the form of brick, sheet, shot and other casting remain in the reactor area. Some of the lead material may require decontamination prior to final disposition. Lead will be surveyed and/or sampled for radioactive contamination in order to segregate clean material from contaminated material. The material will be packaged in transport containers, as necessary, and removed from the containment building. Contaminated lead may be decontaminated on site or transported to a licensed facility for treatment. Options for the beneficial re-use of lead will be evaluated and the most cost effective method for final disposition pursued.

Lead-Containing Coatings

Demolition work performed during the TR-2 decommissioning project may require the removal of lead-containing coatings, or remediation in areas where lead dust may have accumulated. Upon identification of these areas, a qualified lead abatement subcontractor, or qualified remediation team workers, will be used to remove the lead containing coatings or dust.

Any work performed that requires a torch to metal that has a lead-containing coating, will have the coating removed by a qualified abatement subcontractor or qualified remediation team worker. The activities will be performed prior to any torch to metal work or grinding of lead containing coatings and will comply with the Waltz Mill Remediation Project Site Specific Safety and Health Manual.

Asbestos Abatement

Asbestos containing materials will be removed and packaged for disposal prior to any decommissioning activities in areas where these materials exist, provided these activities can be conducted safely and radiation exposure can be maintained ALARA. Asbestos has been identified in the floor tiles on the operating floor (elevation 16'0"), on several of the intermediate reactor platforms (elevation 32'3" and 36'7 1/2"), and in the test reactor piping systems insulation. Removal and disposal of asbestos will be accomplished by a licensed asbestos abatement contractor. Additional asbestos materials discovered in the course of decontamination activities will be abated by the asbestos contractor, as needed.

2.2.2.4 Reactor Vessel and Biological Shield

General

The stainless steel reactor vessel is centered and encased within the biological shield. The vessel is approximately 8 feet in diameter and 32 feet in length, extending vertically from approximately elevation 62' down to the top of the sub-pile room at elevation 29'. Access to the reactor vessel is available from the top, at the reactor head, and the bottom at the vessel's bottom flange. The interior of the vessel may be accessed by removing either the reactor head or the bottom flange in the sub-pile room. Radioactive contamination is present as surface contamination and component activation within the reactor vessel. The most feasible means of access would be through the reactor head or the access plugs in the reactor head.

Removal of the WTR reactor vessel and biological shield will proceed following either one of two options:

Option 1- One-Piece Reactor Vessel Removal Through the Containment Dome
Option 2- Multiple Piece Reactor Vessel Removal Through the Truck Lock
Option 3- One-Piece Reactor Vessel Removal Through the Truck Lock

All of the options are presented in this Decommissioning Plan to allow overall project flexibility. The final course of action will be determined based on engineering, licensing, and ALARA considerations.

Option 1- One-Piece Reactor Vessel Removal Through the Containment Dome

Option 1 involves removing a portion of the biological shield, and lifting the entire reactor vessel and internal components intact out of an opening cut into the top of the containment building. Details of the rigging and lifting are provided in Section 2.2.3.1; a conceptual drawing is provided as Figure 2-4 sheets 1 of 6, 2 of 6, and 3 of 6.

Option 1 requires the following actions for one-piece removal of the reactor vessel and internal components:

a) Remove portions of the biological shield;
b) Inject low density grout into the reactor vessel;
c) Fix external contamination and prepare the vessel for rigging;
d) Cut an opening in the dome of the containment building;
e) Lift the reactor vessel and remaining biological shield out of the containment building;
f) Prepare and ship the vessel to a licensed disposal facility.

Remove Excess Biological Shield

The portion of the biological shield beyond approximately one foot from the vessel exterior will be cut, removed, and staged for final disposition in a safe and secure manner. With a large portion of the biological shield removed, the reactor vessel and remaining biological shield attached to the vessel will be approximately 32 feet tall by 10 feet in diameter, and will weight approximately 148 tons.

Inject Low Density Grout into the Vessel

A low density cellular grout (approximately 20-25 pounds per cubic feet wet density) may be used for stabilizing components and fixing contamination inside the reactor vessel. The reactor piping may also be removed and the control rod drive mechanisms will be removed from the reactor vessel. Covers will be positioned and welded to the cut/prepared reactor vessel openings. Once the major openings are sealed, the reactor vessel may be filled with low density grout. Some opening will have to be used to inject the grout.

The grout mix, equipment, materials, personnel and methods to be employed for this operation will be substantially the same as those previously used for other large nuclear steam supply system component removals. The grouting equipment will be kept outside the containment as much as possible to avoid contamination and minimize waste volumes.

Fix External Contamination and Prepare for Rigging

A paint or similar coating will be applied to the outside surface of the remaining biological shield to fix contamination in place. This paint/coating will be a high solids encapsulating paint/coating. (This application has been used in similar processes for steam generator component removal.) The paint/coating may be applied to the surfaces with minimal surface preparation. In addition to or as a replacement for the painting/coating, the remaining biological shield may be placed in a container or sleeve to control the spread of contamination.

Cut an Opening in Dome of the Containment Building

A layout plan will be prepared for accurate alignment of the dome cutting operations to minimize the size of the opening required for the large component removal. After the cut layout is marked on the dome, the cut may be made, using torch or equivalent method. A temporary closure will then be installed over the opening after the cut is complete; the opening will be uncovered only during actual rigging and lifting of the components. This temporary closure will allow a negative pressure to be maintained in the containment building when the enclosure is installed.

Rigging and Lifting the Reactor Vessel and Remaining Biological Shield

After the vessel has been prepared, the outside crane will be positioned and the rigging attached to the reactor vessel and remaining biological shield. The lifting rig will be designed to lift the total calculated loads. Once the lifting arrangement has been attached, the rigging slack will be taken up, and the load transferred to the outside crane. The reactor vessel will then be lifted from the containment building and staged in a safe and secure manner.

Prepare and Ship the Reactor Vessel/Remaining Biological Shield to a Disposal Facility

After the reactor vessel and remaining biological shield have been lifted out of the containment building, it will be prepared for shipping. Either the vessel/biological shield will be modified so that it becomes the waste package, or it will be placed inside a cask/container. Packaging, shipping, and transportation will comply with all applicable licensing and shipping regulations. Safety analyses and radiological surveys will be performed, and special permits will be obtained before shipping the vessel/biological shield to a licensed disposal facility, as required.

Option 2 - Multiple-Piece Reactor Vessel Removal Through the Truck Lock

The multiple-piece option involves cutting the biological shield off of the reactor vessel using a diamond wire saw, removing the upper and lower reactor internals, and cutting the upper, middle, and lower vessel into sections. All of the sections will be within the capacity of the interior polar crane to allow moving the sections from the work area onto a transport system and then out of the containment building. The process for the multiple piece removal is described as follows:

Upper Vessel Internal Components

Prior to removal activities an interim HEPA filtration system, capable of creating a negative pressure within the vessel, will be installed at one or more inspection ports at the reactor head (elevation 61' 8 1/2", see Figure 2-5). From the sub-pile room (elevation 29' 8"), access ports will be removed from the bottom flange of the vessel to allow installation of HEPA filtration ducting at the bottom of the reactor vessel. The interim HEPA system, at the reactor head, can be removed once ventilation through the bottom flange is established. The reactor head can then be removed using a crane.

The head can be placed on the head stand located on the second platform (elevation 51'0"). Depending on radiological conditions, construction of a temporary containment and air lock over the vessel may be required at the reactor head while the internal components are removed.

Removal of upper internal components from the reactor vessel can be done manually using long handled tools, as appropriate, to maintain exposure ALARA. Figure 2-5 depicts the reactor with internal components in place. Control rods, guides, flanges and piping penetrations will be dismantled with hand tools and/or cutting, as appropriate. Once the reactor internal components are removed, reducing the dose rates within the vessel, access can be allowed provided that exposure can be maintained ALARA. Welded components within the reactor vessel will be removed using appropriate cutting equipment (e.g., plasma torch). A lay down area for the internal components will be located on the platform adjacent to the reactor head (elevation 61'8 1/2"). Debris will be handled manually, again using long handled tools, as appropriate, or lifted out using a crane. Waste containers will be positioned on the platform for material packaging and removal. Filled waste containers will be removed from the platform using a crane and positioned on the transport system for transfer to the staging area.

Upper Platforms and Biological Shield

The upper platforms and the upper biological shield will be removed after the upper vessel internal components are removed. Figure 2-6 illustrates how the upper portion of the biological shield will be removed by sectioning. The sectioning plan is based on the results of concrete core samples, which allows for separation of activated from non-activated concrete. Concrete blocks will be sectioned to stay within the load limits of the crane. Blocks of removed concrete will be moved by the crane and placed on the floor at elevation 16'0" in a designated low background area. The blocks will be surveyed and sampled for contamination and prepared for removal from the containment building. Contaminated blocks will be transported to an auxiliary area for decontamination or packaging for disposal. Concrete blocks meeting the unrestricted release criteria may be transported to an appropriately permitted landfill. This procedure will be repeated throughout the removal of the remainder of the biological shield and upper platforms.

Mid Biological Shield Area

The mid biological shield area will be removed from the perimeter of the vessel leaving a center square column of concrete around the core of the reactor. This column of concrete will remain, acting as shielding, until removal of the vessel's internal components is complete. The mid biological shield will be removed from elevation 51'0" down to approximately 34'0" (see Figure 2-7). Some of these sections of concrete blocks will require additional sectioning to remove contamination on the sides nearest the reactor vessel. This activity will take place in the Auxiliary Area after the blocks are removed from the containment building.

The center column containing the vessel will be reduced as shown in Figure 2-8. These blocks of concrete and the portions of the vessel contained within are contaminated, or contain activated materials. These sections are not economical to decontaminate or further volume reduce. Some of these sections will require special containers to shield higher levels of radioactivity. The containerization of this waste will take place inside the containment building. These containers will then be transferred out of the containment building by the transport system and moved to a temporary storage area on site or to a licensed disposal site.

Lower Internal Components

The lower internal components and the remainder of the mid biological shield will be removed as shown in Figures 2-9 and 2-10. Decontamination of a majority of the contaminated/activated internal components using existing technology is not feasible and they will therefore be containerized prior to leaving the containment building. The truck lock platform may need to be removed if the lower biological shield and sub-pile area are removed (see the following section). If it is necessary to remove the lower biological shield and sub-pile sections and, consequently, the truck lock platform, a new structural steel replacement platform may be required. If the biological shield below elevation 32'3" can be decontaminated without disassembly, the truck lock platform will remain.

Lower Biological Shield

Due to the levels of contamination in areas within the lower section of the biological shield, it may be necessary to remove the entire base as opposed to portions, or decontaminate in place. This is a decision that will require further consideration as the area is exposed during the decommissioning effort. Figure 2-11 illustrates the methods of removal of this section. The blocks of concrete will be staged and removed as previously discussed with the exception of the utilization of the newly constructed structural steel platform in the place of the removed truck lock platform. This approach will leave the lower level base elevation at approximately 19'. It will then be determined whether further reduction will be necessary. The remaining contaminated portions could be either cut away or decontaminated in place.

Option 3- One-Piece Reactor Vessel Removal Through the Truck Lock

Option 3 involves removing the majority of the biological shield, and lifting and downending the entire reactor vessel and internal components intact out of the containment building through the truck lock. Details of the rigging and lifting are provided in Section 2.2.3.1; a conceptual drawing is provided as Figure 2-4 sheets 4 of 6, 5 of 6, and 6 of 6.

Option 3 requires the following actions for one-piece removal of the reactor vessel and internal components:

a) Remove portions of the biological shield;
b) Fix internals inside the reactor vessel;
c) Apply shielding and cover penetration openings;
d) Fix external contamination and prepare the vessel for rigging;
e) Lift the reactor vessel and downend and remove from the containment building through the truck lock
f) Prepare and ship the vessel to a licensed disposal facility.

Remove Biological Shield

The biological shield will be cut from the vessel, removed, and staged for final disposition in a safe and secure manner. With the biological shield removed, the reactor vessel will be approximately 32 feet tall by 8 feet in diameter, and will weight approximately 80 tons.

Fix Internals Inside the Reactor Vessel

The reactor piping exterior to the vessel and the control rod drive mechanisms will be cut off and removed from the reactor vessel. Covers will be positioned and welded to the cut/prepared reactor vessel openings prior to removal from the containment building.

The reactor vessel openings may be used for access to the reactor vessel internals. Attachments may be made to the internals through the vessel openings, the reactor head or the access plugs in the reactor head, to prevent movement of the internals during lifting, downending and shipment.

To minimize the movement of articles inserted into the reactor core structure and the radial reflector structures, an encapsulating material may be applied. This encapsulating material would be sprayed on, covering the internal structures in a manner which would preclude their movement during lifting, downending, and shipment. The encapsulating material employed for this operation will be substantially the same as materials used to coat and decontaminate other nuclear components, using a stripping technique, such as Master-Lee's Instacote® or IceSolv's TLC.

Apply Shielding and Cover Penetration Openings

Shielding may be applied, as needed, to reduce exposure rates during the removal process and to meet applicable DOT requirements during transportation. In addition, all open penetrations will be covered with adequate shielding and to secure the contents of the reactor vessel.

Fix External Contamination and Prepare for Rigging

A paint or similar coating will be applied to the outside surface of the reactor vessel to fix contamination in place. This paint/coating may be a high solids encapsulating paint/coating or other suitable fixative. (This application has been used in similar processes for steam generator component removal.) The paint/coating may be applied to the surfaces with minimal surface preparation.

Rigging and Lifting the Reactor Vessel

After the vessel has been prepared, a lifting attachment or lifting lugs will be secured to the reactor vessel. A lifting device, such as a jacking tower, will be positioned inside containment above the reactor vessel and the rigging attached to lifting attachment or lugs on the reactor vessel. The lifting device, the lift rigging and the lifting attachments will be designed to lift the total calculated loads. Once the lifting arrangement has been attached to the reactor vessel, the rigging slack will be taken up, and the load transferred to the lifting device. The reactor vessel will then be detached from it's mounting and lifted in a safe and secure manner. The reactor vessel will then be lowered and downended into the horizontal position and transported out of the containment building through the truck lock.

Prepare and Ship the Reactor Vessel to a Disposal Facility

After the reactor vessel has been removed from the containment building, it will be prepared for shipping. Any additional shielding required to meet applicable DOT requirements for shipment will be attached to the vessel or added to the shipping cradle. The reactor vessel will be enclosed in a protective barrier, such as a High Density PolyEthylene (HDPE) wrap and placed in the shipping cradle. Packaging, shipping, and transportation will comply with all applicable licensing and shipping regulations. Safety analyses and radiological surveys will be performed, and special permits will be obtained before shipping the vessel to a licensed disposal facility, as required.

2.2.3 Decommissioning Methods

WTR Decommissioning involves removal and disposal of portions of the biological shield, the reactor vessel and the reactor vessel internal contents. This includes the following activities:

1. Remove and dispose of material as radioactive waste
2. Remove, decontaminate as necessary, and release material for unrestricted use (this will generally involve disposal at a landfill or processing at a scrap/recycling facility)

Activities that may be undertaken to dismantle and decontaminate other areas within the containment building are described in Section 2.7, and will involve additional decontamination and removal processes. These areas will be left in place and transferred to the SNM-770 License.

Each major equipment item and area will be evaluated to determine the best method(s) for removal, for decontamination, and to determine whether to decontaminate or dispose of as radioactive waste. Criteria to be used in the evaluations include: availability of a burial facility; the cost of decontamination versus the cost of burial; radiological and occupational hazards involved; and site operations in progress or planned.

Removal of structures, equipment and components can be achieved using proven mechanical/thermal cutting and demolition equipment. Mechanical methods such as diamond wire cutting, saw cutting, concrete scabbling, expandable grout, the use of jackhammers, and machining may be utilized. Thermal methods such as metal cutting with an oxy-acetylene torch method may also be used.

2.2.3.1 Demolition and Component Removal

Decommissioning of TR-2 involves removal of the reactor vessel, the biological shield, and the vessel internals.

Methods used for the removal of concrete include jackhammers, expandable grout, concrete saws, and diamond wire saws. These methods are described as follows:

Jackhammer
Equipment can range in size from hand held units to large hoe rams mounted on tracked excavators. The concrete is degraded through constant pneumatic impact of a chisel pointed bit. This method works well but it is noisy and produces large quantities of dust and debris. Typically a containment tent is constructed over the area and supplemental roughing filter and HEPA filter ventilation is used to control airborne dust and radioactivity.

Hoe Ram
Where large areas of concrete require removal, it may be cost effective to decontaminate the concrete to acceptable levels by other means and then use large hoe rams to remove the concrete. Appropriate controls will be used to minimize the spread of airborne dust and radioactivity.

Expandable Grout
Expandable grout may be used for demolition of concrete structures or removal of predetermined layers of concrete from structures. Holes are systematically drilled into the concrete in preparation for the addition of the grout. The grout is then mixed with the appropriate quantity of water and poured into the pre-drilled holes. As the grout hardens, it expands and cracks the concrete apart.

Concrete Saws
Concrete saws may be used for accurate cutting of concrete for general demolition and dismantlement. Also this method may be used to cut large slabs of concrete for waste volume reduction or packaging.

Diamond Wire Cutting
Diamond wire cutting techniques can be used to remove large segments of concrete. A diamond-studded cable is circulated by a hydraulic pulley drive system through the concrete, cutting through concrete, steel rebar and other steel members in the concrete. Hydraulic cylinders control the tension of the cable. Holes are drilled through the concrete to enable stringing the cable into cutting target areas that would otherwise be inaccessible. Water applied to cool and lubricate the cable also aids in control of airborne dust. A slurry collection system is installed to collect contaminated cutting slurry, decant the slurry and recycle the water.

Pipe Removal
Various reactor system pipes and sample loop piping will be removed as part of TR-2 decommissioning. Steel pipes are generally removed using mechanical or thermal cutting methods, such as hand-held band or reciprocating saws and oxy-acetylene cutting torch. Commercially

available oxy-lance and plasma arc cutting methods may also be used. Plasma arc cutting equipment can be track-mounted and operated remotely, minimizing personnel exposure in high radiation areas. It also can cut underwater.

Rigging and Lifting

Plans will be developed for removing equipment and material from inside the containment building to a safe and secured area outside of the containment building. These plans will include the integration of equipment, methodology, and training of personnel to enhance total safety as much as practical. All rigging and lifting will be performed in accordance with industry standard safe practices and lifting equipment will be designed to comply with ANSI/ASME specifications.

The rigging and lifting method selected will depend upon whether one-piece removal through the containment dome, the multiple piece reactor vessel removal, or the one-piece removal through the truck lock method is selected.

The first method, one piece removal through the containment dome, involves a large capacity external crane to lift the reactor vessel and large slabs of the biological shield through an opening cut in the containment building dome. The advantages of the one piece removal over a multiple piece removal, are that less cutting and packaging is required and worker exposures are reduced. However, this method involves greater rigging challenges and a hole has to be cut into the containment building.

The second method, multiple piece removal method through the truck lock, involves cutting the reactor vessel and biological shield into pieces small enough to be handled by the existing interior polar crane. The advantages are lower waste volumes and ease of handling/packaging smaller pieces with the existing polar crane, and then moving them to the truck lock and out for packaging and shipment offsite. The disadvantage is that cutting the reactor vessel may result in increased dose exposures and may not be ALARA.

The third method, one-piece removal through the truck lock, utilizes jacking and downending techniques commonly used to remove components in close quarters of the size and weight as that of the reactor vessel. The advantages of less cutting and packaging and lower exposures of Option 1 are again realized in addition to the advantages of maintaining the boundary of the containment building and avoiding a high heavy-lift scenario.

Radiological Control/Equipment Decontamination

Equipment will be checked for residual contamination before exiting designated restricted areas. Any equipment utilized within a designated restricted area will be decontaminated before removal from the work area. The restricted area will be demarcated by flagging, physical barricades or fencing as deemed appropriate.

Loading/Shipping

Loading of shipping containers and hauling equipment will be controlled to minimize contamination on external surfaces. Containers/loads will be secured/covered. Material designated for off-site disposal will be placed in packagings which meet DOT requirements, and staged in a secured area to prevent inadvertent removal from the site.

2.2.3.2 General Surface Decontamination Methods

The methods described below are typical, other processes and technologies may be used.

Strippable Coatings
Strippable coatings may be used to assist in the removal of loose radionuclides from large surface areas. Strippable coating is a simple, effective means of removing loose radionuclides or protecting areas that may possibly be contaminated during scheduled work activities. Once the surface is dry, the strippable coating serves as a barrier preventing radionuclides from reaching the surface below. If the barrier becomes contaminated, it can be stripped away, or the radionuclides can be sealed in place with a second layer and subsequently stripped away. Any method normally used to apply coatings (airless sprayers, paint rollers or brushes) may be used to apply strippable coatings.

Vacuuming/Scrubbing/Wiping
These techniques are generally used when gross loose radionuclides are visible on the targeted surface(s). Vacuum operations use systems equipped with a HEPA filter. If a wet vacuum is required for liquid retrieval, the vacuum system also includes an automatic water shut-off system to prevent destruction of the HEPA filter when the unit is full. Scrubbing and wiping techniques are used where access is limited or can not be reached with a vacuum unit. It should be noted that vacuuming can be used on any type of surface but scrubbing/wiping are normally used on smooth, non-porous surfaces.

Pressurized Water
Pressurized water spraying may be used for general area decontamination or decontamination of items and components in a confined space. This method will only be utilized in areas where the spent water can be directed into a drain, sump or some other means of collection. The contaminated water will be treated and monitored to ensure compliance with discharge limits before discharge. Descriptions of pressure spray methods follow:

Low Pressure Spray (Power Wash)
Water is sprayed on the surface to be decontaminated with the objective of removing loosely adhered contamination. This technique is effective on coated surfaces that allow the contamination to be removed easily. Water pressure is generally in the 1,500 to 5,000 psi range with water consumption typically 3 to 6 gallons per minute.

High Pressure Spray (Hydrolaser)
A powerful stream of water is applied to the surface in a side to side, top to bottom fashion. This method is used on large surfaces and complex structures or equipment. This technique is effective on coated and uncoated contaminated surfaces. The water can be applied in various temperature ranges and the addition of chemical agents may increase the overall decontamination factor; chemical agents will be carefully selected to

reduce the possibility of creating mixed wastes. High pressure sprays typically operate over a pressure range of 5,000 to 20,000 psi, with a water consumption rate of about 5 gallons per minute.

Ultra High Pressure
Ultra High Pressure water can be utilized in two ways. The first method is to direct a precise stream of water from a multi-jet rotating nozzle at the target surface. This method is capable of removing loose as well as fixed contamination from the surface. The second method utilizes a single nozzle that is capable of cutting material from the surface. Water consumption varies with nozzle selection; rates from 2 to 6 gallons per minute are typical with water pressure approaching 40,000 psi.

Steam Lance
Saturated steam is directed on the surface to be decontaminated. Crystalline materials can be solubilized and particulates removed using this technique. This method may be useful when the surface is sludge or oil coated.

2.2.3.3 Concrete Surface Removal Methods

Scabbling
When coating removal and/or surface removal is required, scabbling may be the preferred decontamination method. Scabblers remove the surface by impacting the area with air driven tungsten carbide tipped bits. Scabbler sizes range from single-piston units suited for small constricted or isolated areas, to multi-piston units designed for operation in large open areas. Surface removal can vary from a light single pass removing 1/16 inch to multiple activities removing 1 inch or more. HEPA filter vacuum units will be attached to shrouds around the scabbler heads to control airborne radioactivity, where necessary.

Scarification
Scarification is the process of removing a surface layer of material from concrete floor slabs or similar surfaces. This equipment is generally utilized for projects where wide open floor areas are contaminated and require surface removal. A scarifier is a mechanically powered (electric, gas or propane) device that removes surface layers of material with a rotating drum equipped with tungsten carbide tipped cutters. When the unit is operated the bits are forced against the surface at a predetermined depth and lateral speed.

A HEPA filtered vacuum system operated in conjunction with a vacuum shroud attached to the scarifier is used in controlling airborne radioactivity during operation. Typical surface removal depths vary between 1/16 to 1/4 of an inch per pass.

Grinding
On a smaller scale, hand held grinders can also be used to remove surface coatings or concrete.

Needle Gun

A needle gun operates by pneumatically driving specially hardened needles into the surface being cleaned. The needle gun is designed to remove surface material from small areas or restricted spaces. The process takes place within a vacuum shroud, preventing the escape of dust, debris and airborne contamination. The vacuum shroud is connected to a HEPA filtered vacuum system that provides the negative pressure required.

Abrasive Blasting

Abrasive blasting is a preferred metal surface removal method, and is described below. However, it can also be effectively used for concrete surface removal. Blastrac (discussed in following section) is commonly used for decontamination of concrete floors.

2.2.3.4 Metal Surface Removal Methods

Abrasive Blast

Coating and/or surface removal can be achieved using an abrasive blast method. This technique is capable of removing loose and fixed contamination with a high production rate.

Abrasive blast techniques use non-hazardous abrasive material suspended in a medium (air or water) that is propelled against the targeted surface. The result is a fairly uniform removal of surface material. High production rates are common. Overhead and vertical surfaces can be decontaminated with relative ease. Depending on the equipment used and radionuclide levels encountered, the blasting medium may be reused.

Blasting media include sand, steel, aluminum oxide, walnut shells and plastic. Supplemental HEPA filter ventilation is used when necessary to control airborne dust and radioactivity.

Recycled Abrasive Blast (Blastrac)

Shot blasting is an airless method that strips, cleans, and prepares the surface for coating application. Surface removal can be achieved by selecting the proper shot size and residence time. The shot is propelled at the surface using a centrifugal blast wheel. As the wheel spins, the abrasive is hurled from the blades, blasting the surface with a barrage of media. The abrasive is continuously recycled using a vacuum system in conjunction with a separation system.

Supplemental HEPA filter ventilation is required to control airborne dust and radioactivity.

2.2.4 Decommissioning Schedule

The WTR Decommissioning Project is currently scheduled from February 1998 to 2003. The decommissioning project schedule assumes NRC approval of the Decommissioning Plan by January 1998. See Figure 2-12, entitled "WTR Decommissioning Schedule."

Changes to the schedule may be made at Westinghouse's discretion as a result of resource allocation, availability of a radioactive waste burial site, interference with ongoing Waltz Mill operations, ALARA considerations, further characterization measurements and/or temporary on-site radioactive waste storage operations.

2.3 DECOMMISSIONING WORK CONTROLS

Work controls will be established to ensure remediation work is safely performed in accordance with the Decommissioning Plan, Waltz Mill license requirements and established procedures.

A Project Management Plan (PMP) will be prepared that describes the approach and methods to be used to ensure the successful decommissioning of Waltz Mill facilities. The PMP will provide descriptions of the management philosophy, approach, and techniques to be used on the project. The system of work controls described above will be proceduralized in a Project Control Manual (PCM), which will include implementing procedures and supporting information for preparation of the Work Breakdown Structure, Work Specifications and Work Packages, in accordance with requirements of the Decommissioning Plan.

A General Work Specification will be developed to establish the basic requirements and provide the planning information for the performance of work activities. In addition to the General Work Specification, other Work Specifications may be prepared for activities that require special controls (e.g., water treatment).

Work Packages will be prepared based upon the Work Specifications and will contain the detailed instructions for accomplishing the defined tasks.

2.4 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

The Westinghouse Test Reactor Decommissioning organization complies with the existing license and applicable regulatory requirements.

The direct responsibility for operational oversight of activities conducted under the TR-2 License and the TR-2 Protection Program rests with the CBS TR-2 Decommissioning Project Director who is located on-site and reports directly to CBS Corporation's Director of Environmental Remediation. The Project Director will have overall responsibility for the TR-2 licensed facility. The Waltz Mill Site will continue to provide support for operations, engineering, industrial hygiene, safety, security, environmental compliance, facilities support, radiation protection and quality assurance through a contractual arrangement with CBS.

Reporting through a contract arrangement to CBS is the Industrial Hygiene, Safety and Environmental Compliance Manager to whom the Radiation Safety Officer (RSO) reports. The RSO is responsible for the establishment and guidance of programs in radiation protection. To maintain a uniform program that addresses public safety for the Waltz Mill Site, the RSO position for the TR-2 License will be filled by the RSO for the SNM-770 License. The RSO also evaluates potential and/or actual radiation exposures, establishes appropriate control measures, approves written procedures, and ensures compliance with pertinent policies and regulations. Under the RSO's direction, health physics personnel administer the established site policy, collect samples, perform analyses, take measurements, maintain records, and generally assist in performing the technical aspects of the radiation protection program. The RSO will be supported by adequate staff, facilities and equipment and will hold a position within the organizational structure providing direct access to senior management.

The Decommissioning Team Program Manager reports to CBS through a contract arrangement. The Decommissioning Team Program Manager will coordinate the elements of the functional groups of the Waltz Mill decommissioning organization, Decommissioning Team, and decommissioning contractors, as it applies to decommissioning activities. The Decommissioning Team reports to the Decommissioning Team Program Manager.

The Radiation Safety Committee has been established to monitor decommissioning operations to ensure they are being performed safely and according to federal, state, and local regulatory requirements (NRC, EPA, PADEP, DOT, etc.). Members of this committee are appointed by the CBS Director of Environmental Remediation and will include members of the Radiation Safety Committee (RSC) for the SNM-770 license. The RSC will review major decommissioning activities dealing with radioactive material and radiological controls. In addition, the Radiation Safety Committee will review and approve changes to the Decommissioning Plan that do not require prior NRC approval.

The number and titles of key functions/positions shown on Figure 2-13 may be modified during the course of the decommissioning project. However, the following key functions/positions will not be eliminated while decommissioning activities are in progress, without prior NRC approval:

- CBS TR-2 Decommissioning Project Director
- Radiation Safety Officer
- Radiation Safety Committee

2.5 CONTRACTOR ASSISTANCE

Westinghouse management has selected a team of qualified contractors to perform the WTR Decommissioning project. The team consists of Westinghouse-Nuclear Services Division (NSD), Morrison-Knudsen, and GTS-Duratek (formerly SEG). Westinghouse-NSD will be in charge of the overall project management and engineering; Morrison-Knudsen will manage the craft laborers who will do the physical work; and GTS-Duratek is responsible for Health Physics support, radiation surveys, and waste packaging, processing, and shipping. Other contractors may be added to the team as-needed throughout the project.

Contractors and subcontractors performing work under this Decommissioning Plan will be required to comply with the applicable Waltz Mill site policies and procedures.

2.6 TRAINING PROGRAM

Individuals (employees, contractors, and visitors) who require access to the work areas or a radiologically restricted area will receive training commensurate with the potential hazards to which they may be exposed.

Radiation protection training will be provided to personnel who will be performing remediation work in radiological areas or handling radioactive materials. The training will ensure that decommissioning project personnel have sufficient knowledge to perform work activities in accordance with the requirements of the radiation protection program and accomplish ALARA goals and objectives. The principle objective of the training program is to ensure that personnel understand the responsibilities and the required techniques for safe handling of radioactive materials and for minimizing exposure to radiation.

Records of training will be maintained which include trainees name, date of training, type of training, test results, authorization for protective equipment use, and instructor's name. Radiation protection training will provide the necessary information for workers to implement sound radiation protection practices. The following are examples of the training programs applicable to remediation activities.

2.6.1 General Site Training

A general training program designed to provide orientation to project personnel and meet the requirements of 10 CFR 19 will be implemented. General Site Training (GST) will be required for all personnel assigned on a regular basis to the decommissioning project. This training will include:

- Project orientation/access control
- Introduction to radiation protection
- Quality assurance
- Industrial safety
- Emergency procedures

2.6.2 Radiation Worker Training

Radiation Worker Training (RWT) will be required for decommissioning project personnel working in restricted areas and will be commensurate with the duties and responsibilities being performed. Personnel completing RWT will be required to pass a written examination on the material presented. Completion of this training will qualify an individual for unescorted access to radiologically controlled areas. RWT will include the following topics:

- Fundamentals of radiation

- Biological effects of radiation
- External radiation exposure limits and controls
- Internal radiation exposure limits and controls
- Contamination limits and controls
- Management and control of radioactive waste, including waste minimization practices
- Response to emergencies
- Worker rights and responsibilities

In addition to a presentation of the topics identified above, participants in RWT will be required to participate in the following demonstrations:

- The proper procedures for donning and removing a complete set of protective clothing (excluding respiratory protection equipment)
- The ability to read and interpret self-reading and/or electronic dosimeters
- The proper procedures for entering and exiting a contaminated area, including use of proper frisking techniques
- An understanding of the use of a Radiation Work Permit (RWP) by working within the requirements of a given RWP

Personnel who have documented equivalent RWT from another site may be waived from taking training except for training on Waltz Mill administrative limits and emergency response, and will be required to pass the written examination and demonstration exercises.

2.6.3 Respiratory Protection Training

Individuals whose work assignments require the use of respiratory protection devices will receive respiratory protection training in the devices and techniques that they will be required to use. The training program will comply with the requirements of 10 CFR 20 Subpart H, Regulatory Guide 8.15 (Ref. 2), NUREG-0041 (Ref. 3) and 29 CFR 1910.134. Training will consist of a lecture session and a simulated work session. Personnel who have documented equivalent respiratory protection training may be waived from this training.

2.7 OPTIONAL DECONTAMINATION AND DISMANTLEMENT ACTIVITIES WITHIN THE WTR CONTAINMENT BUILDING

In addition to removal of the reactor vessel internal contents, the reactor vessel, and the biological shield, decontamination and dismantlement activities may be performed in other areas within the WTR containment building. These activities are not required for TR-2 decommissioning; however, they may be performed prior to transfer of remaining residual radioactivity to the SNM-770 License.

The decontamination techniques and methods described in Sections 2.2.3.2 through 2.2.3.4, and the dismantlement techniques described in Section 2.2.3.1 may be used to decontaminate and dismantle equipment and structures in these areas.

These optional activities are discussed as follows:

2.7.1 Sub-pile Room

General

The sub-pile room is a 15' x 15' room located below the reactor vessel. This room has a ¼-inch steel liner on all four walls covering the concrete biological shield. The floor is uncoated concrete. The WTR canal runs through the sub-pile room (north-south), separating the room into two areas (east and west). The two doors to the sub-pile room consist of a steel liner filled with 12 inches of poured lead. One permits accesses to the east side of the canal and the other to the west side. The WTR fuel chute is accessible in the northeast corner of the room through a shielded opening in the fuel chute pipe chase. The sub-pile room contains primary system piping, rabbit tubes, test loop piping and instrumentation piping.

Internal Equipment

The sub-pile room will be cleaned and all remaining piping will be dismantled and/or cut-out in disposable sized sections and removed.

Floor and Walls

Following removal of remaining loose debris, the area will be re-surveyed for loose and fixed radioactive contamination to determine the appropriate floor, wall and ceiling surface decontamination method. Destructive methods such as scabbling, full or partial demolition may be performed. If scabbling equipment incorporates a self-contained ventilation and filtration system (HEPA), additional containment of the work area may not be required. Demolition of the intact structure may be performed with the demolition of the biological shield.

2.7.2 Rabbit Pump Room

General

The Rabbit Pump Room measures approximately 6'6" by 10'0" by 7'6" high and is located on the operating floor along the north wall of the containment building. The Rabbit Pump Room contains pumps and valves that delivered the rabbits (test material samples) in a container, to the reactor core via the rabbit tubes.

Internal Equipment

Decommissioning activities within the Rabbit Pump Room consist of the dismantling and removal of the pumps, valves, piping and control assemblies. To control airborne radioactive contamination, the Rabbit Pump Room may be contained and fitted with a HEPA filtration system capable of creating a negative pressure within the room during equipment removal operations.

Floor, Walls and Ceiling

Following removal of remaining loose debris, the area will be re-surveyed for loose and fixed radioactive contamination to determine the appropriate floor, wall and ceiling surface decontamination method. Destructive methods such as scabbling, full or partial demolition may be performed. If scabbling equipment incorporates a self contained ventilation and filtration system (HEPA), additional containment of the work area may not be required. Demolition of the intact structure may be performed with the demolition of the biological shield.

2.7.3 Test Loop Cubicles

General

Three test loop cubicles are located along the west side of the reactor vessel adjacent to the reactor biological shield. Each cubicle is constructed of concrete of varying dimensions and all cubicles are currently vacant.

Floors, Walls and Ceilings

Fixed and transferable contamination is found on the cubicle floors and fixed contamination on the walls and ceilings of the cubicle. Following removal of loose debris, destructive methods such as scabbling, full or partial demolition may be performed. Additional containment may not be necessary if equipment utilizes self contained filtration and ventilation.

Demolition of the intact cubicle structures may be performed with the demolition of the biological shield

2.7.4 Test Loop Dump Tank Pits

General
Two 8'0" by 9'0" by 13'0" high Test Loop Dump Tank Pits are located below the operating floor on the east and west side of the transfer canal below the reactor vessel. The west tank pit contains three steel tanks approximately 12' tall and 4' in diameter. The east pit is flooded with water and the pit interior currently inaccessible.

Tank (and Pit) Water Removal
The water in the flooded east pit will be pumped out and routed to either the site radioactive water processing facility or treated with a portable system. Once the water in the pit has been removed, the floors, walls, tanks, and tank internals will be surveyed for fixed and transferable contamination.

Any remaining water in the tanks located in the east and west pits will be pumped out to either the site radioactive water processing facility or treated using a portable system.

All liquids removed and treated from both tanks and the flooded pits will be sampled and analyzed for determination of the proper disposal mechanism.

Tank Demolition
Following removal of the water within the tanks, the tanks will be removed from the pits intact, decontaminated and cut into appropriately sized dimensions for packaging and disposal. The tanks may also be shipped off-site intact following decontamination should a satisfactory salvage opportunity be identified, or shipped off-site intact to a licensed waste processor.

Pit Areas
Fixed and transferable radioactive contamination has been found on the floor and walls (and exposed area of the concrete shield plugs) and is also anticipated to be found on the surfaces of the flooded pit once the water has been removed. Destructive methods such as scabbling, full or partial demolition may be performed. The pits may be left in place.

Duct Decontamination and/or Removal
Supply and exhaust air ducts may be decontaminated in place or removed, decontaminated and released. If warranted by radiological conditions, a temporary HEPA filtration system may be attached to the ventilation ducts to ensure that any loose contamination is drawn away from workers during these operations. Contaminated ducts which can not be decontaminated efficiently and economically may be removed, cut and sized for packaging and disposal. Resulting penetrations through the containment building will be sealed.

2.7.5 Utilities

General

Prior to removing electrical, service water, service air, fire or HVAC systems, each system will be inspected by a qualified individual. All efforts will be made to review the existing status of the respective utilities with Westinghouse service personnel who have a working knowledge of the utilities to prevent service disruption to other site facilities. Emergency utilities, such as fire alarm systems, will be maintained, as required.

Utility Removal

Initially, electrical systems will be disconnected. Piping systems will then be removed in areas where electrical systems have been disconnected/removed. Reactor and containment building utilities can be removed simultaneously. As lines are disconnected, provisions will allow the collection of any remaining fluid.

Characterization results indicate the presence of fixed and transferable contamination on some portions of electrical wiring, conduit, cable trays, electrical boxes and piping. As the utility systems are removed, contaminated piping, conduit, cables, etc., will be separated from non-contaminated systems. Fluids collected from piping systems will be sampled and analyzed for determination of the proper disposal method.

It is expected that dry and/or wet wiping techniques will be sufficient to decontaminate those portions of the materials initially found contaminated. Contaminated materials which can not be successfully decontaminated for unrestricted release or if decontamination is not feasible or cost effective can be volume reduced to the maximum extent practicable and packaged for disposal. Clean material will be disposed of at a local landfill or recycled, if appropriate.

2.7.6 Primary Coolant Pipe Tunnels

General

The primary coolant pipe tunnels surround the north end of the transfer canal along the east and west sides of the reactor vessel below the operating floor. Each tunnel measures approximately 5'0" wide by 10'0" high by 39'0" long and merge into a common tunnel at the north side of the containment building. The tunnel continues below grade to the northeast to the Facilities Operations Building. The pipe tunnels contain the primary coolant circulation supply and return lines, demineralizer, emergency coolant and various other piping systems.

Pipe Removal

Initial work in the pipe tunnels consists of installing temporary lighting and a HEPA filtration system. Identification and tagging of all piping systems will be performed prior to any pipe removal. Any water contained within the tunnels will be pumped out to either the Waltz Mill radioactive water processing facility or treated using a portable system. The piping components identified as contaminated will be dismantled and/or cut into disposal dimensions or separated for decontamination. Pipe ends should be wrapped as they are dismantled/cut. The processed or treated liquids will be sampled and analyzed to determine the proper disposal mechanism.

Tunnel Floors, Walls and Ceilings

Following removal of all contaminated piping systems, the pipe tunnels floor, wall and ceiling concrete surfaces will be surveyed to determine the extent of contamination. Following removal of loose debris, radiological conditions will determine the appropriate floor, wall and ceiling surface decontamination method for the tunnels. Destructive methods such as scabbling, full or partial demolition may be performed. The tunnels may be left in place.

2.7.7 Transfer Canal

General

The transfer canal is approximately 19 feet deep, varies in width from 7 feet to 10 feet and is approximately 160 feet long north-south down the axis of the reactor. The canal begins north of the biological shield and continues beneath the reactor vessel to the south, through the G building Annex, ending beneath the Hot Cell area. The transfer canal was the means of transporting spent fuel rods from the reactor vessel to the Annex Building and irradiated test specimens to the Hot Cell area (see Figure 2-14). The fuel rod conveyor, storage racks, thimble loading machine, transfer chute, rabbit tubes, piping and pipe supports were left in the canal following the 1962 shut down. All irradiated material was removed and properly dispositioned.

Exterior Equipment and Materials

The initial phase of canal remediation will require removal of exterior appurtenances above the canal (between the 15' and 19' elevations). This may include removal and decontamination of the drive mechanisms, platforms and existing wire mesh and foam covers.

Sediment Removal

The transfer canal has sediment attached to the walls, floor and structural debris system, in addition, the concrete sealant is peeling off. This sediment is generally contaminated and in some locations highly contaminated. A filter system will be designed to remove, safely contain the sediment, and shield workers prior to or during lowering the water. Figure 2-15 illustrates one of many ways to accomplish the removal of sediment.

Canal Water Removal and Interior Wall Surfaces

The existing water within the canal will be pumped through a water filtration system to remove fine particles suspended in water. After the water is cleared of solids, the existing or a supplemental liquid radioactive waste treatment system will be utilized to treat the canal water. The water level can be lowered in stages and the walls cleaned, as required, from a platform suspended from the canal walls at elevation 19', Figure 2-16. This method will not allow large portions of contaminated surfaces to be exposed above the water level. As water is removed from the canal, the radiological conditions within the canal will be monitored. Appropriate precautions will be taken to prevent or minimize the potential for airborne radioactive contamination. This may include containment and HEPA filtration, maintaining contaminated surfaces wet, or both. Destructive methods, such as scabbling, may be required to remove the fixed contamination on the walls.

2.7.8 Containment Building

The WTR was a low pressure, low temperature, water cooled 60 MWt reactor housed in a cylindrical vapor containment building. There are two airlocks, and a large overhead door that provides access from the truck lock to the WTR. A schematic of the WTR is shown on Figure 2-1.

The reactor core support structure is 29 feet in diameter and 36 feet tall, which houses the reactor pressure vessel. The biological shield surrounding the reactor vessel is made of magnetite bearing concrete, is a total of 44 feet in height and is up to eight feet thick from the 32 to 51 foot elevations. The operating floor is on the 16 foot elevation and is constructed of concrete. The containment is 90 feet in diameter, with a total floor area of 5000 square feet. There are four support platforms: the truck lock, the reactor head stand, reactor head, and the beam port platforms. As part of the materials testing that was included in the WTR's operational charter, there were several controlled environment test loops installed in concrete cubicles and in an underground test loop vault. Since the shut down most of these loops have been removed.

The containment building also houses the rabbit pump room, polar crane, and other support systems such as: piping, electrical conduit and boxes, plant and instrument air lines, hydraulic lines, steam and condensate lines, and ventilation ductwork.

Decontamination of the interior of the structure will be conducted only after all other major components have been removed or addressed. Decontamination of the structure will use non-destructive methods if it is to be left in place. If the structure will be removed completely, then it will be shipped to a licensed scrap metal processing facility according to their license requirements. All remaining piping, platforms, and ductwork will be dismantled and either cleaned, and free released, or sent off site to a licensed waste processor. The cleanup or dismantling should proceed from the top of the dome downward using erected staging or power manlifts for access. Finally, the surface of the operating floor will be decontaminated.

REFERENCES FOR SECTION 2

1. "Westinghouse Electric Corporation, Waltz Mill Facility, Characterization Report; Nuclear Material License TR-2, Test Reactor License," Scientific Ecology Group, dated February 1994.

2. NRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," October 1976.

3. NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials," October 1976.

Table 2-1

WTR FACILITIES, DECOMMISSIONING ACTIVITIES AND ESTIMATED WORKER EXPOSURE

WTR FACILITY AREA	PROPOSED DECOMMISSIONING ACTIVITIES	ESTIMATED EXPOSURE (Person-rem)
Pre-decommissioning Activities	Establish radiological controls.	0.05
Reactor Vessel, Internal Contents, and Biological Shield [1]	Remove internal contents. Use a diamond wire saw to section the biological shield into slabs and section reactor vessel. (Option 2)	26.14
Sub-pile Room [2]	Components removed, concrete decontamination, and partial or full demolition.	0.85
Rabbit Pump Room [2]	Components removed, concrete decontamination, and partial or full demolition	0.08
Test Loop Cubicles [2]	Components removed, concrete decontamination, and partial or full demolition.	0.13
Test Loop Dump Tank Pits [2]	Components removed, concrete decontamination, and partial or full demolition.	0.28
Primary Coolant Pipe Tunnel [2]	Piping removed, concrete decontamination, and partial or full demolition.	1.88
Transfer Canal [2]	Water drained, sediment removed, concrete decontaminated, and partial or full demolition.	7.93
Vapor Containment Building and Misc. Systems and Components [2][3]	Miscellaneous systems and components decontaminated and/or removed, concrete and structure surfaces decontaminated, and polar crane decontaminated.	0.89
	TOTAL	38.23

[1] The total exposure estimate for the one piece reactor vessel removal, internal component removal, and biological shield sectioning and removal (Option 1) is 18.25 person-rem.

[2] Decommissioning of these and other structures may be undertaken as part of the WTR Decommissioning project, and will be completed in conjunction with remediation of SNM-770 facilities.

[3] See Table 2-1(a) for complete list of miscellaneous systems and components.

TABLE 2-1(A)

LIST OF TR-2 MISCELLANEOUS SYSTEM AND COMPONENTS CONSIDERED

Transfer Building Pool
HVAC Ducts (2)
Experimental Cooling Water
LLRW Liquid Drain
Process Vent
Electrical Cond & Boxes
Plant & Instrument Air
Dionized Water
Steam & Condensate Lines
Polar Crane
Containment Building
Final Surveys
Operating Floor 16' Elev
Truck Lock Platform
Beam Port Platform
WTR Head Stand Platform
WTR Head Platform

Table 2-2

EQUIPMENT SELECTION MATRIX

Decontamination Method	Loose Contamination - Large Surface Area	Loose Contamination - Small Surface Area	Loose Contamination - w/o Water Treatment or Containment	Fixed Contamination - Floors/Walls - Large Areas	Fixed Contamination - Floors/Walls - Isolated Areas	Fixed Contamination - Concrete Removal Required	Compressed Air Required for Operation	Electrical Service Required for Operation	HEPA Vacuum Required and/or Containment for Operation	Water Source Required for Operation
Blastrac Unit				X				X	X	
Abrasive Blaster	X		X	X			X			
Pentek Corner Cutter					X		X		X	
McDonald HF 1-Piston Scabbler					X		X			
McDonald 3WCD 3-Piston Scabbler					X		X		X	
McDonald U-5 5-Piston Scabbler				X			X		X	
Pentek Squirrel III Scabbler					X		X		X	
EDCO Floor Plane				X					X	
Power Washer 1-5K psi	X							X		X
Hydrolaser 5-10K psi	X							X		X
ULTRA High Pressure Unit 35-40K psi	X			X	X			X		X
Jack Hammer						X	X			
Bristar Expandable Grout						X				
Hands on Decon		X	X							
Strippable Coatings	X	X	X				X			

FIGURE 2-1

ACCESS CONTROL POINTS TO WTR

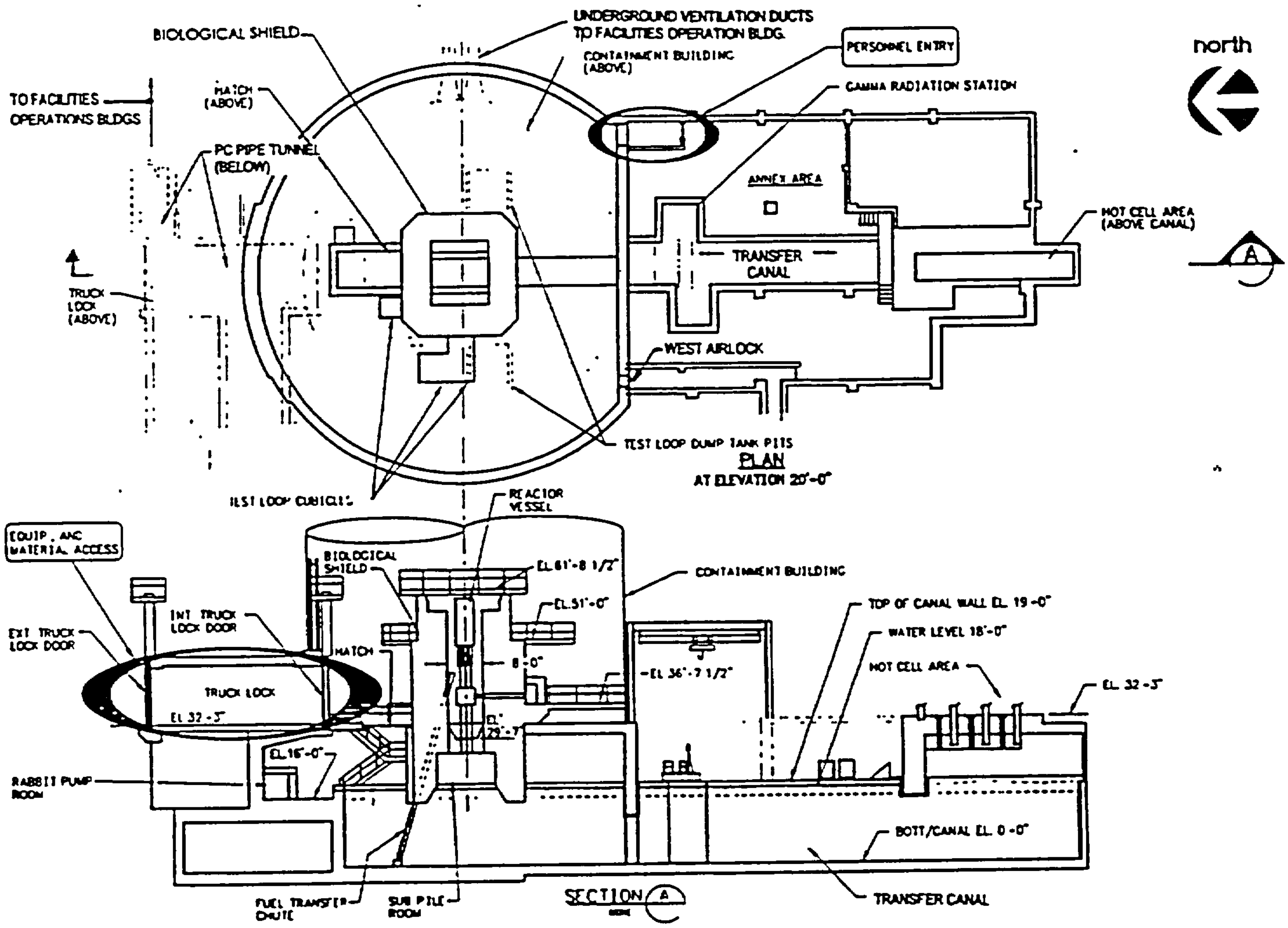


FIGURE 2-2

CONCEPTUAL DRAWING OF RAIL TRANSPORT SYSTEM TO REMOVE MATERIAL FROM WTR

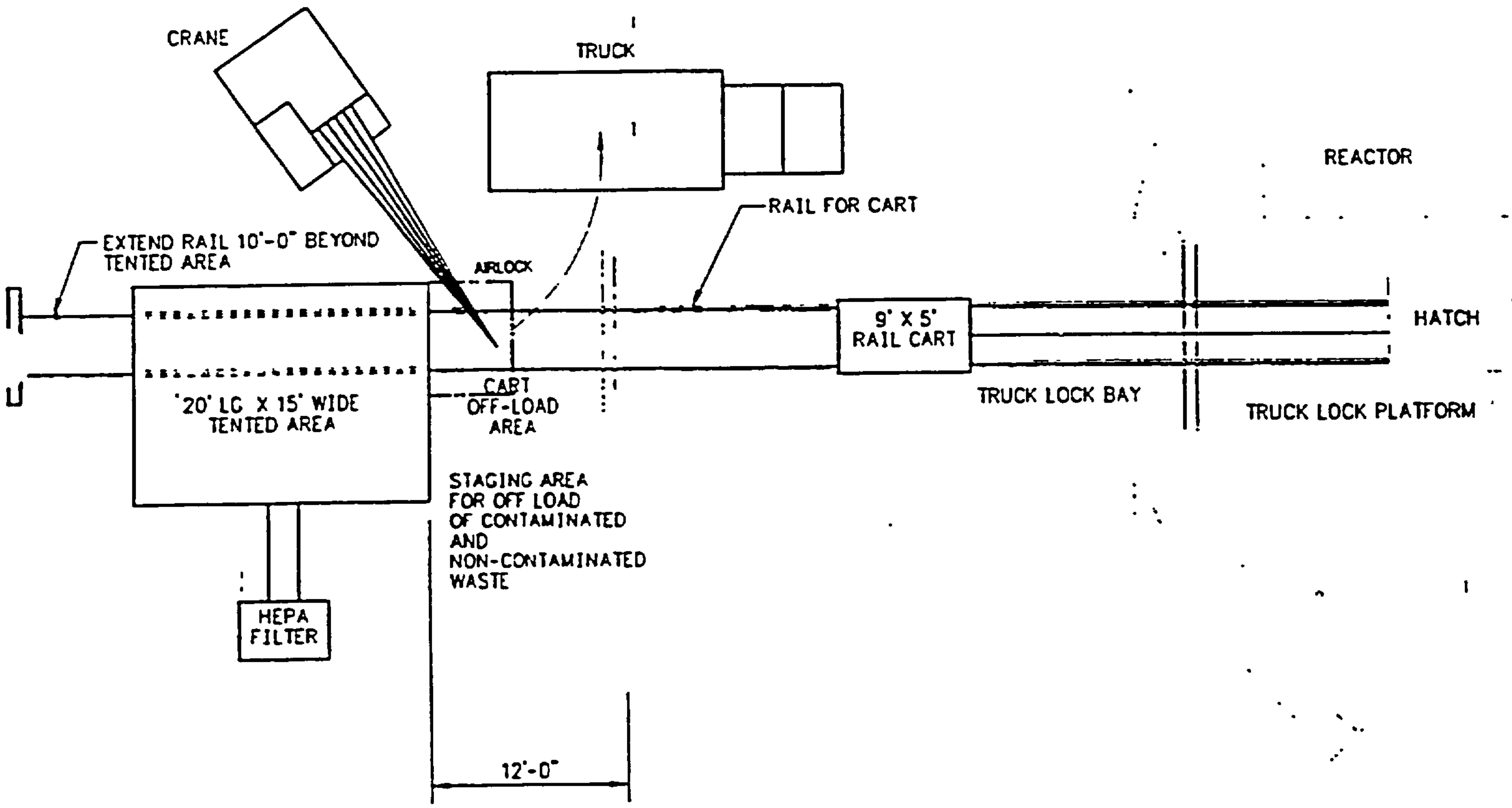


FIGURE 2-3

CONCEPTUAL PLAN HAUL CHART

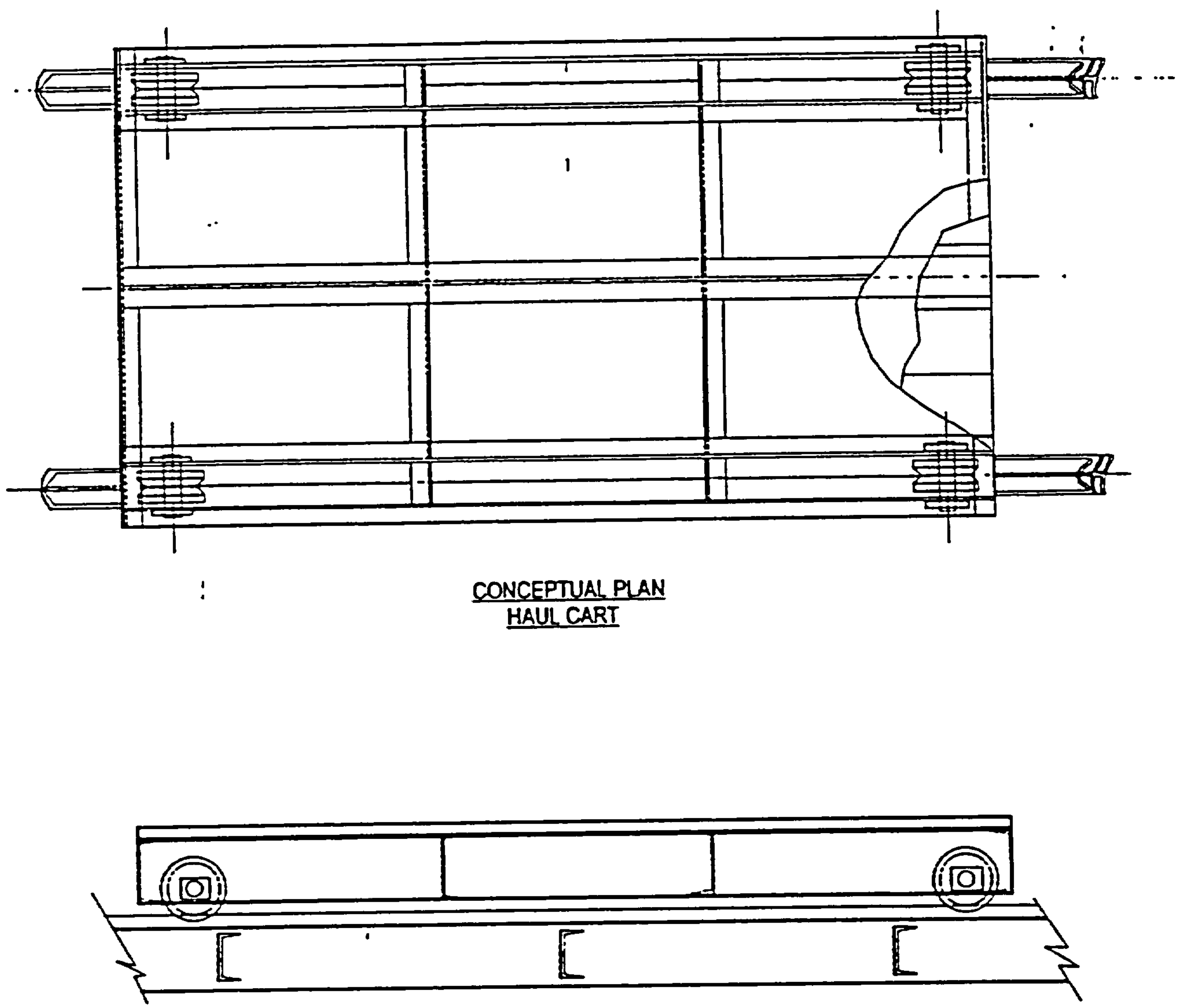


FIGURE 2-4
(Sheet 1 of 6)

CONCEPTUAL DRAWING
ONE PIECE REACTOR VESSEL REMOVAL
THROUGH THE CONTAINMENT DOME

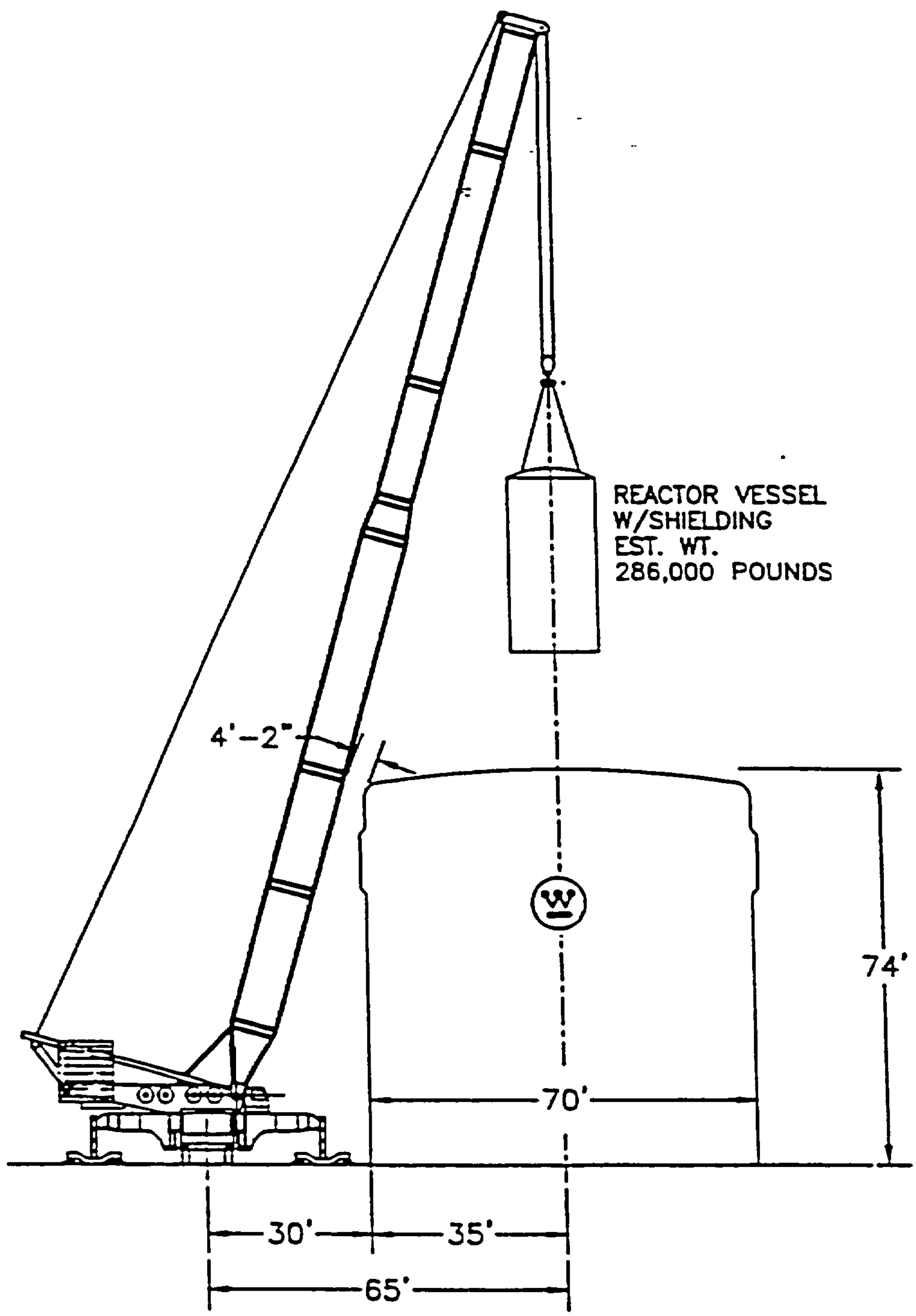


FIGURE 2-4
(Sheet 2 of 6)

AREA PLAN FOR ONE PIECE REMOVAL FROM CONTAINMENT THROUGH THE CONTAINMENT DOME

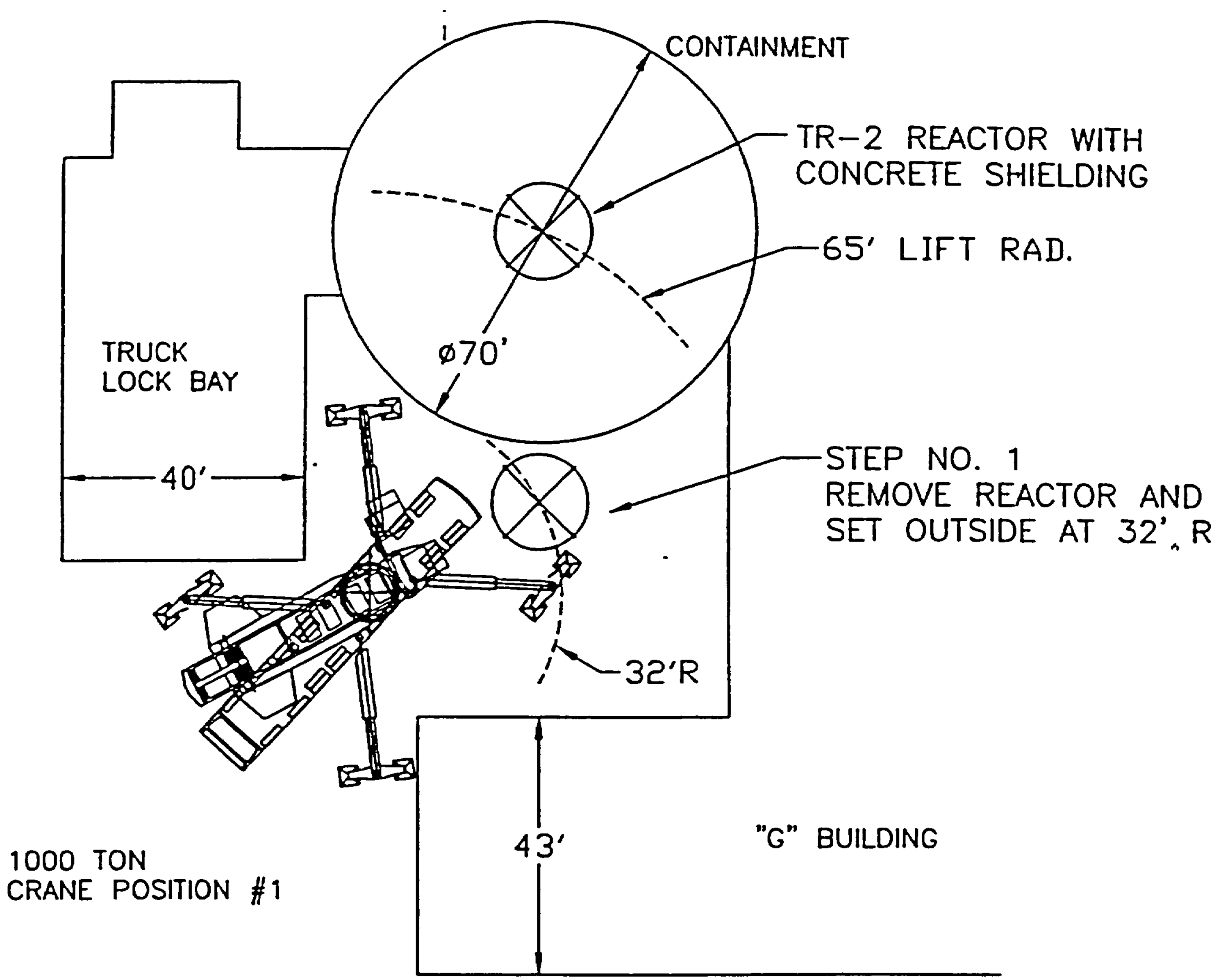


FIGURE 2-4
(Sheet 3 of 6)

AREA PLAN FOR ONE PIECE REMOVAL THROUGH THE CONTAINMENT DOME /LOADING OUT TO TRUCK

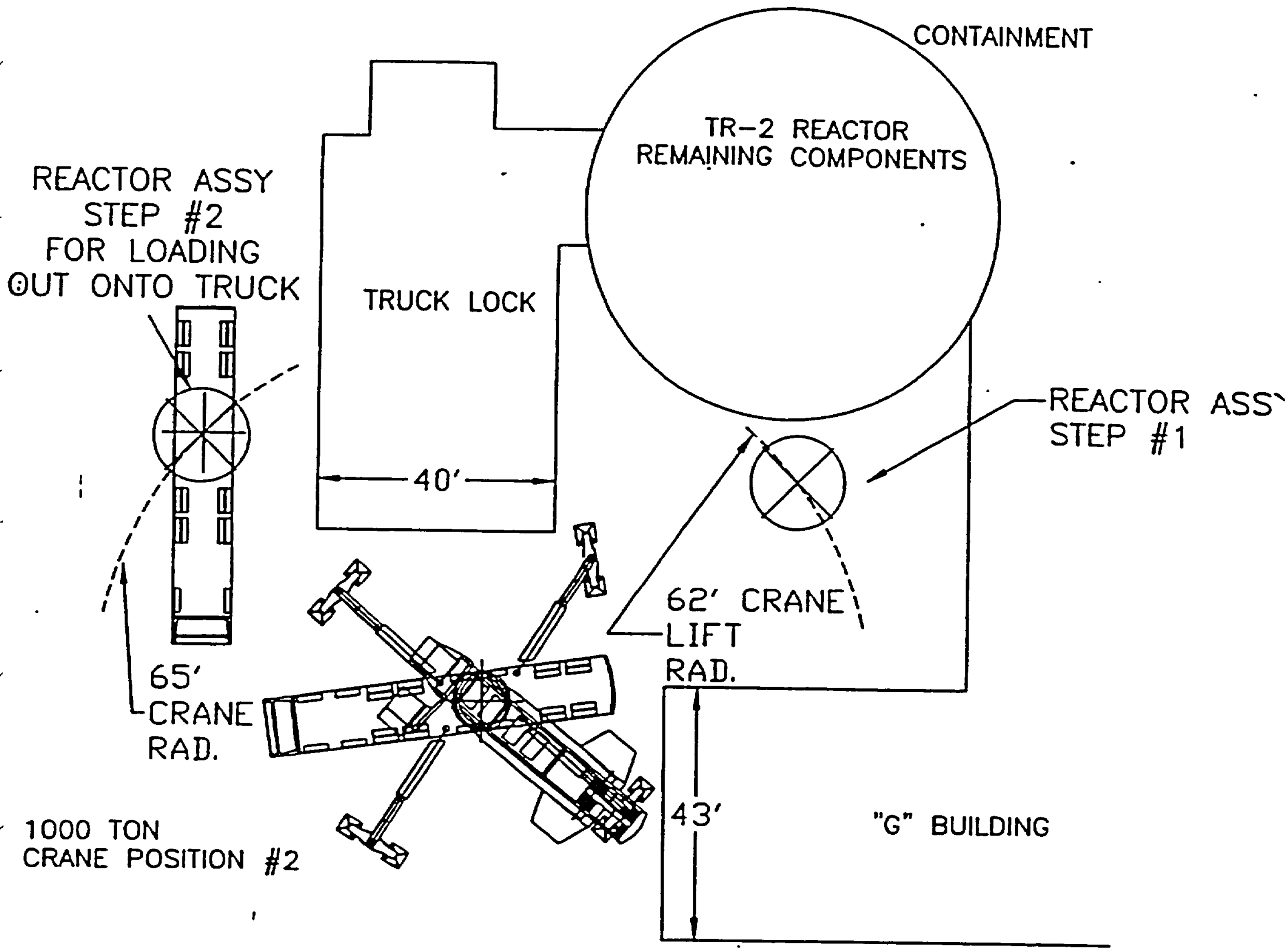


FIGURE 2-4
(Sheet 4 of 6)

ONE PIECE REMOVAL THROUGH THE TRUCK LOCK
LIFTING ASSEMBLY INSIDE CONTAINMENT

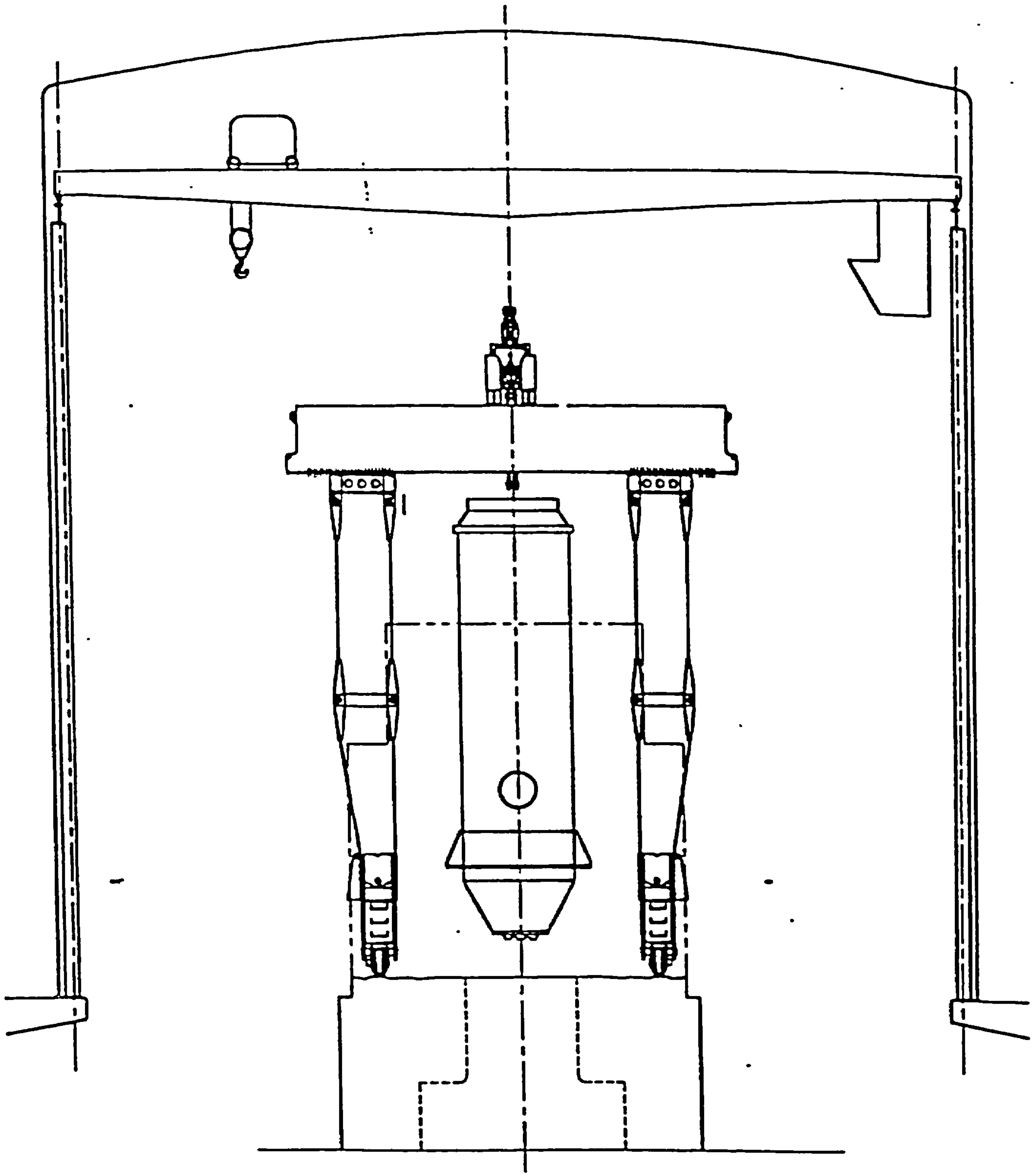


FIGURE 2-4
(Sheet 5 of 6)

ONE PIECE REMOVAL THROUGH THE TRUCK LOCK
DOWNEND REACTOR TANK TO HORIZONTAL POSITION

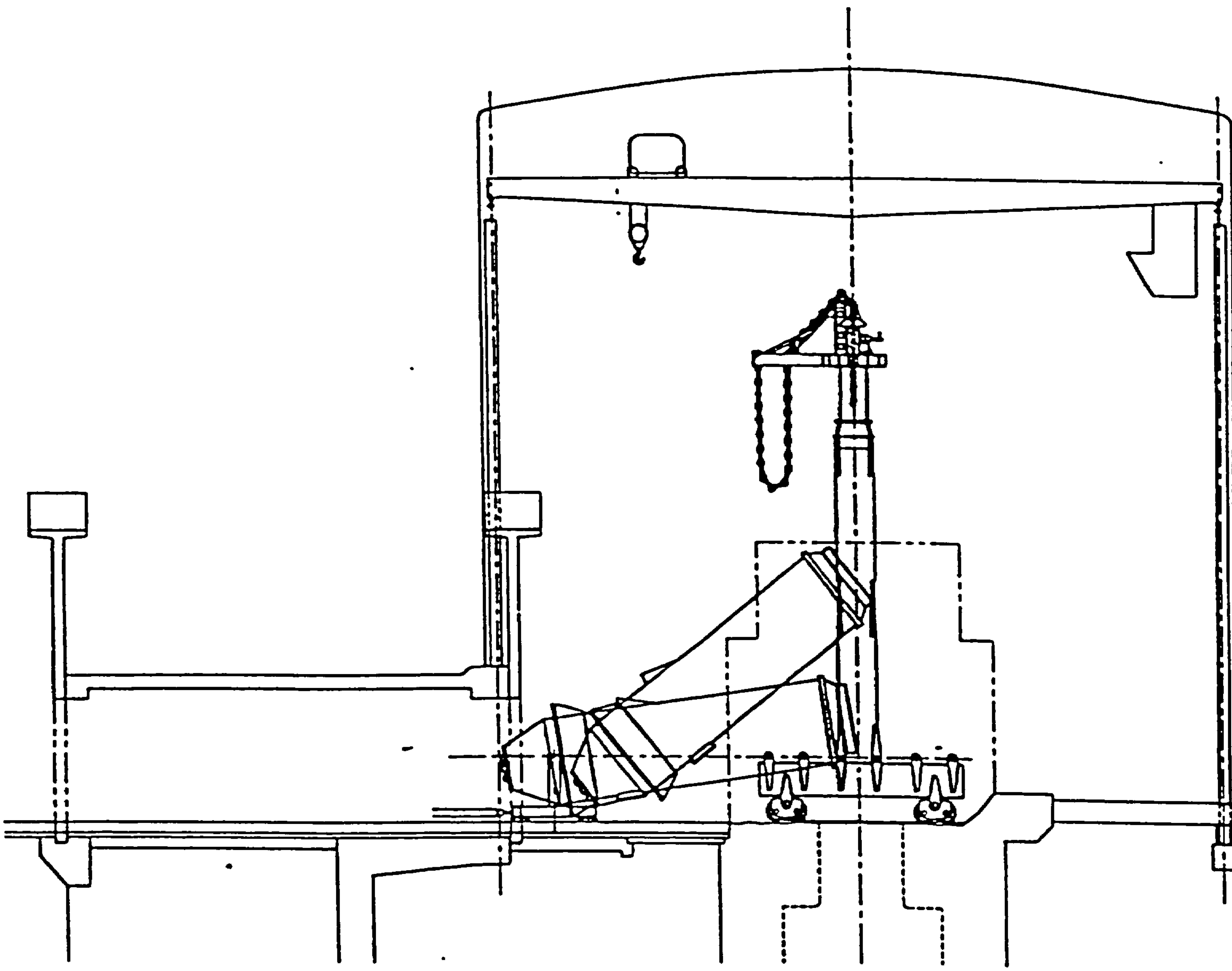


FIGURE 2-4
(Sheet 6 of 6)

ONE PIECE REMOVAL THROUGH THE TRUCK LOCK

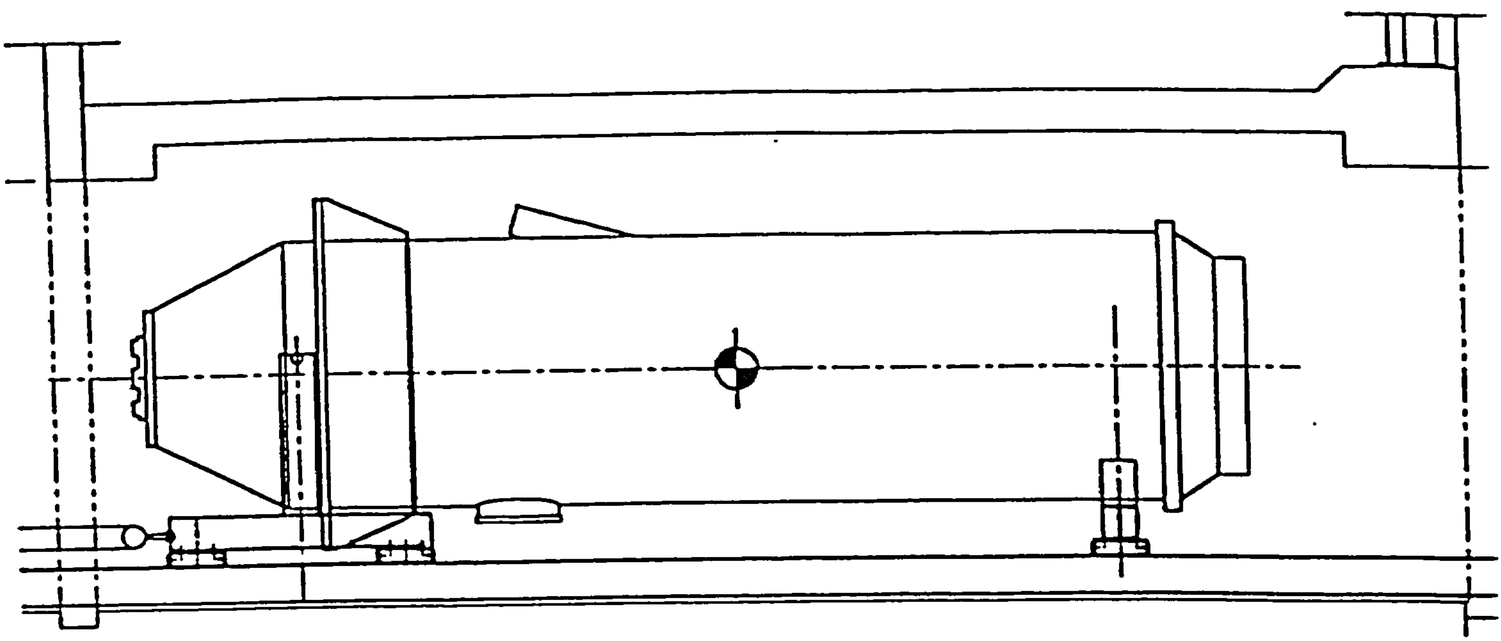
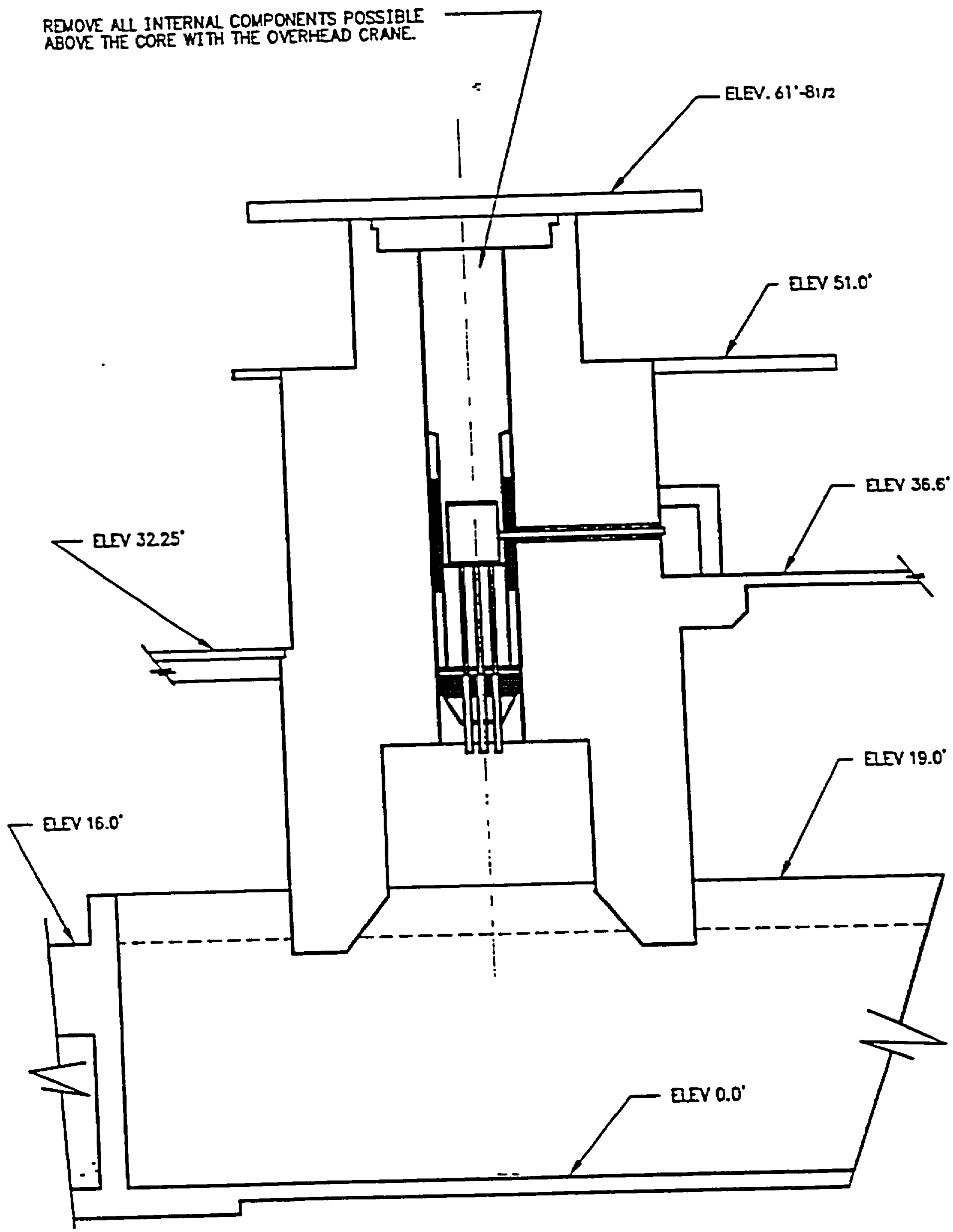


FIGURE 2-5

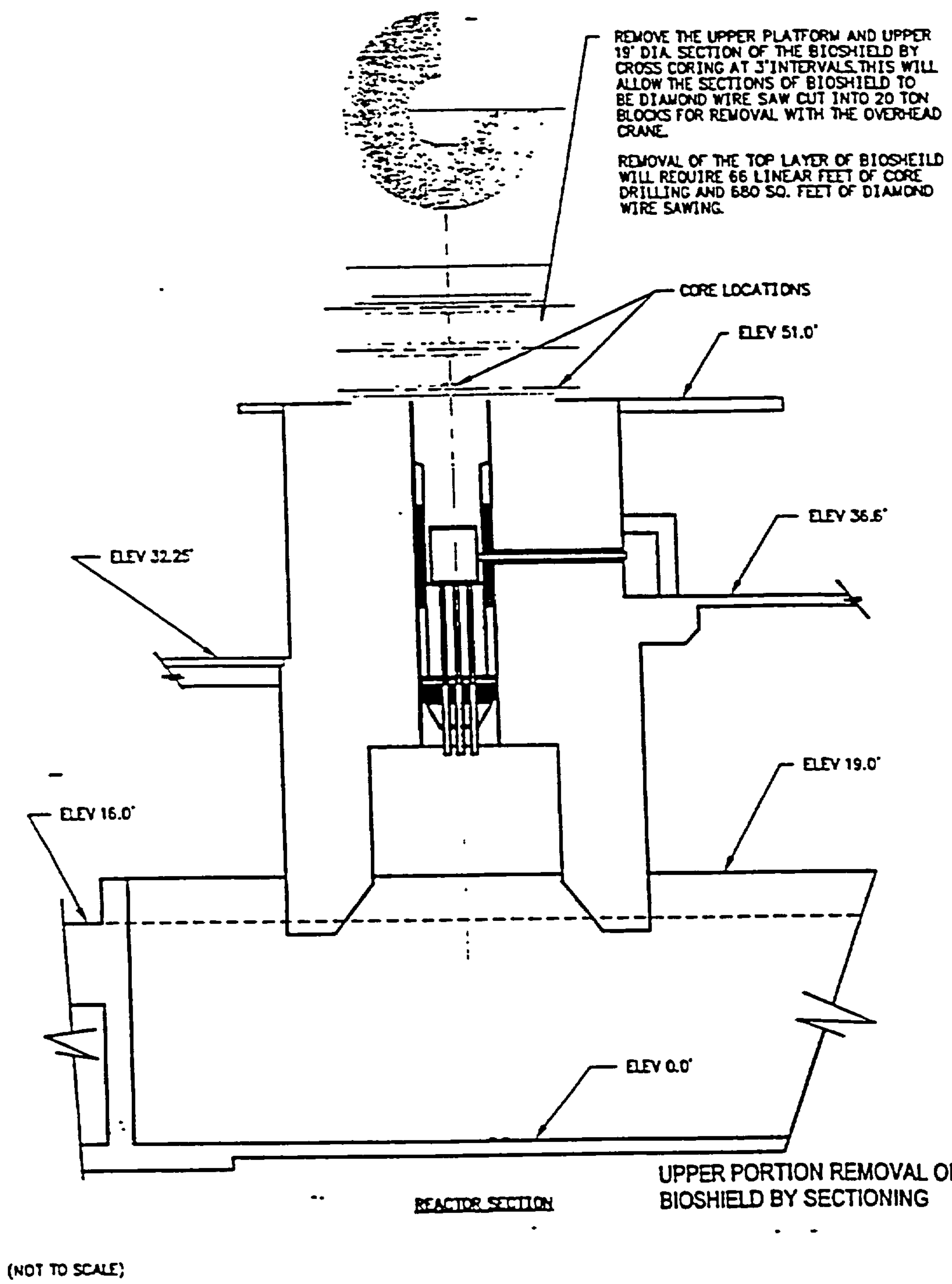
ELEVATIONS



REACTOR SECTION

FIGURE 2-6

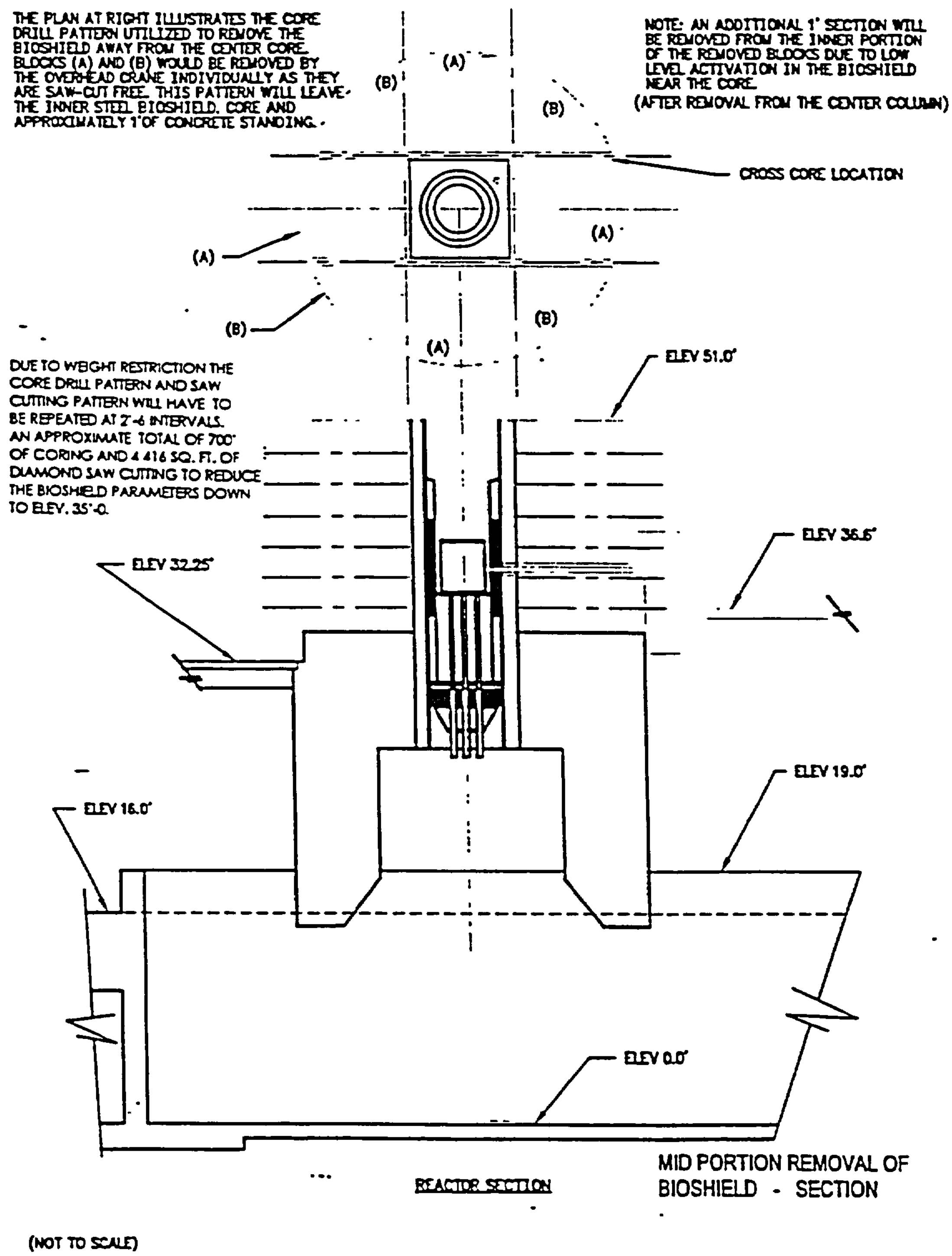
UPPER PORTION REMOVAL OF BIOSHIELD BY SECTIONING



UPPER PORTION REMOVAL OF BIOSHIELD BY SECTIONING

FIGURE 2-7

MID PORTION REMOVAL OF BIOSHIELD - SECTION



MID PORTION REMOVAL OF BIOSHIELD - SECTION

FIGURE 2-8

REACTOR VESSEL REMOVAL SECTIONS

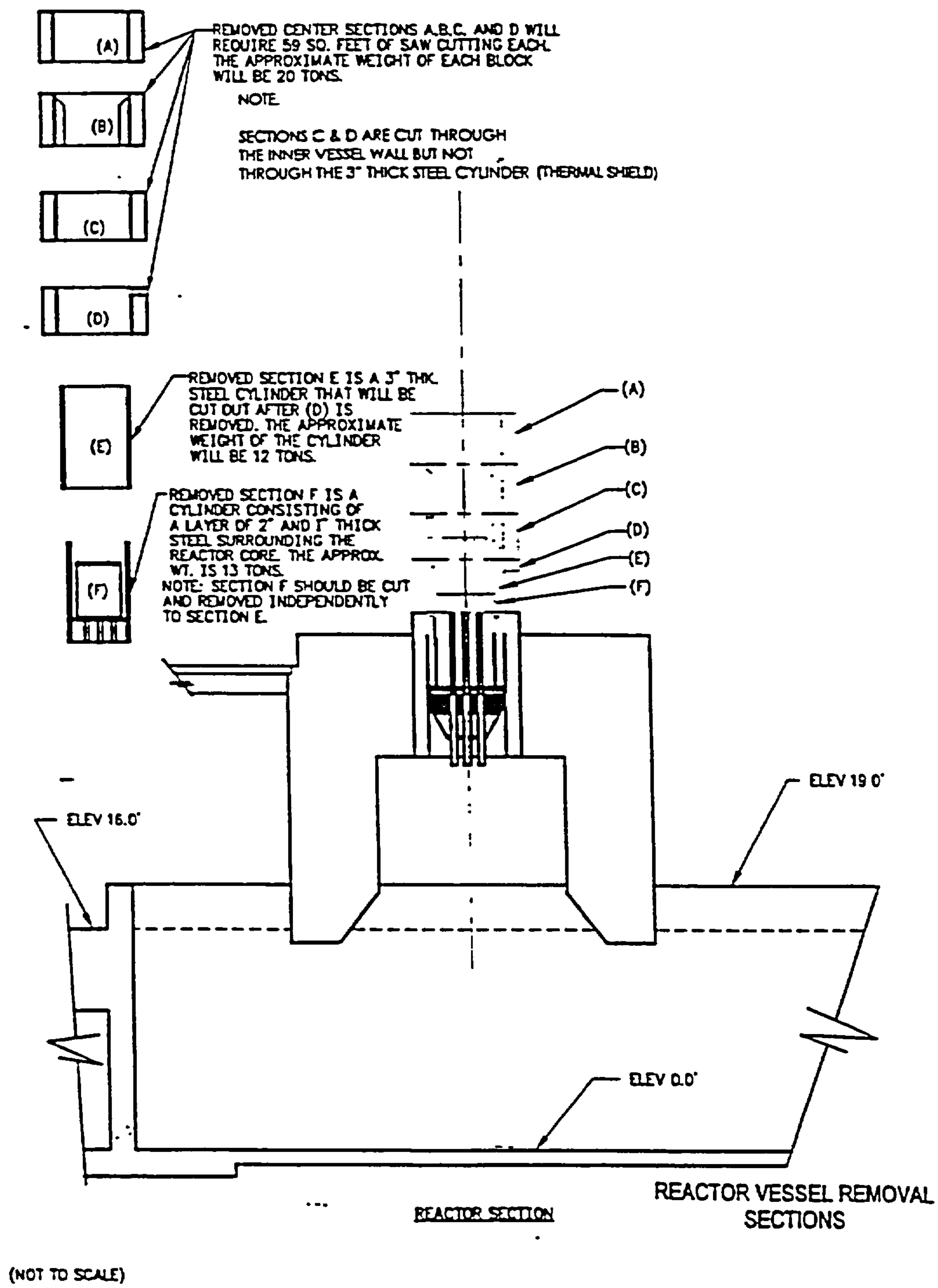
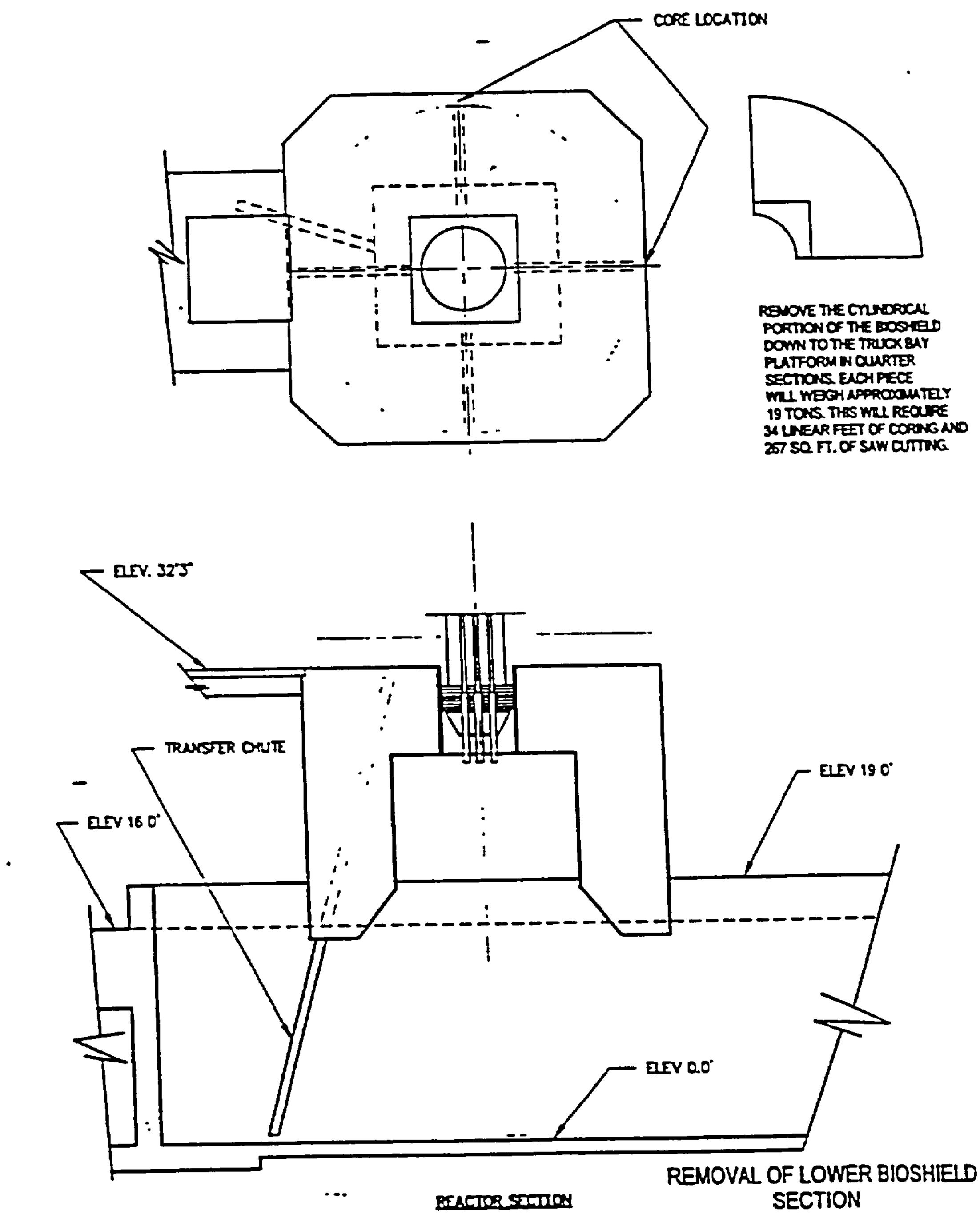


FIGURE 2-9

REMOVAL OF LOWER BIOSHIELD SECTION

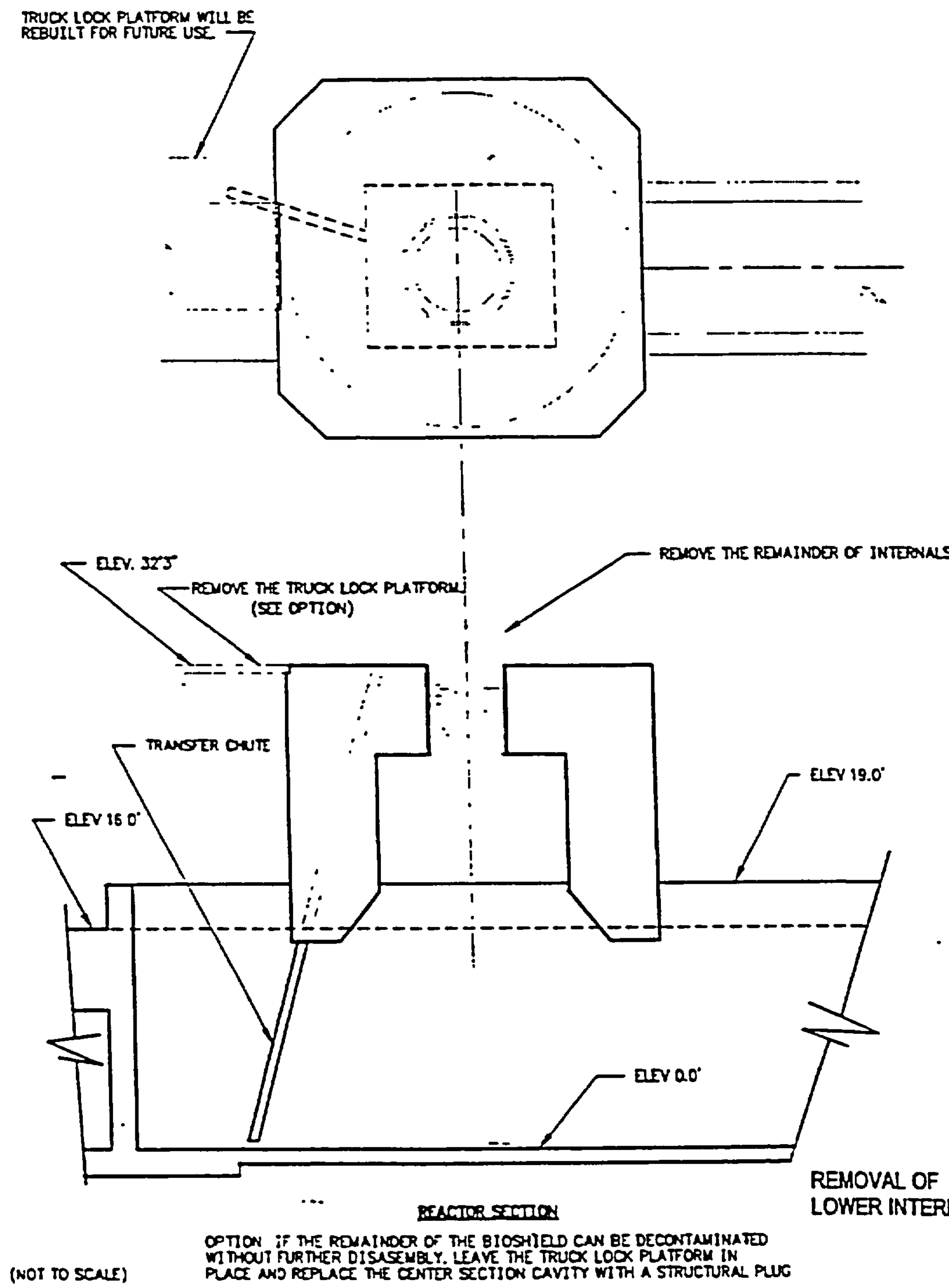


REMOVAL OF LOWER BIOSHIELD SECTION

(NOT TO SCALE)

FIGURE 2-10

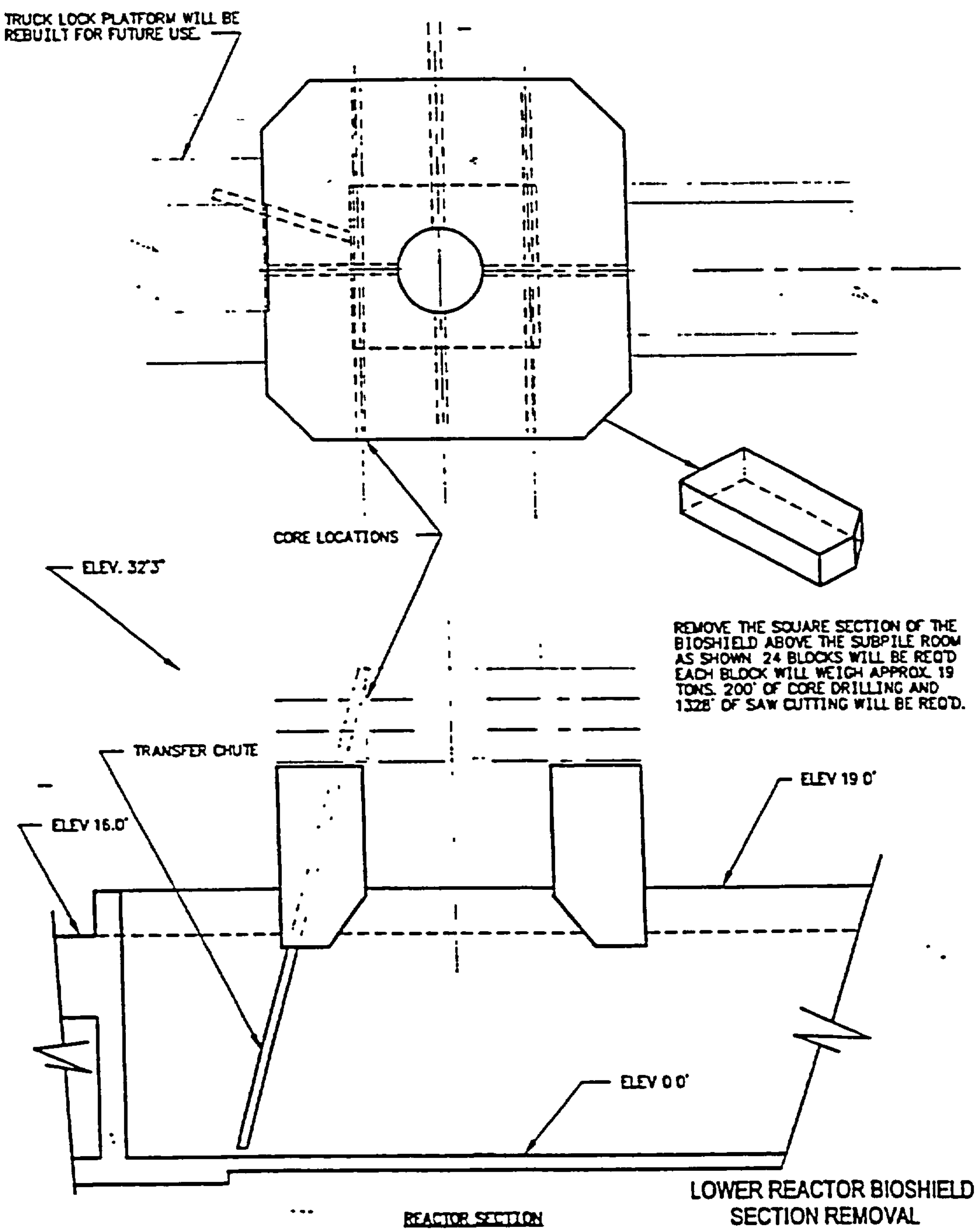
REMOVAL OF LOWER INTERNALS



REMOVAL OF LOWER INTERNALS

FIGURE 2-11

LOWER REACTOR BIOSHIELD SECTION REMOVAL



LOWER REACTOR BIOSHIELD SECTION REMOVAL

NOTE: IF OPTION DESCRIBED IN FIGURE 8 IS NOT UTILIZED CONTINUE AS SHOWN.

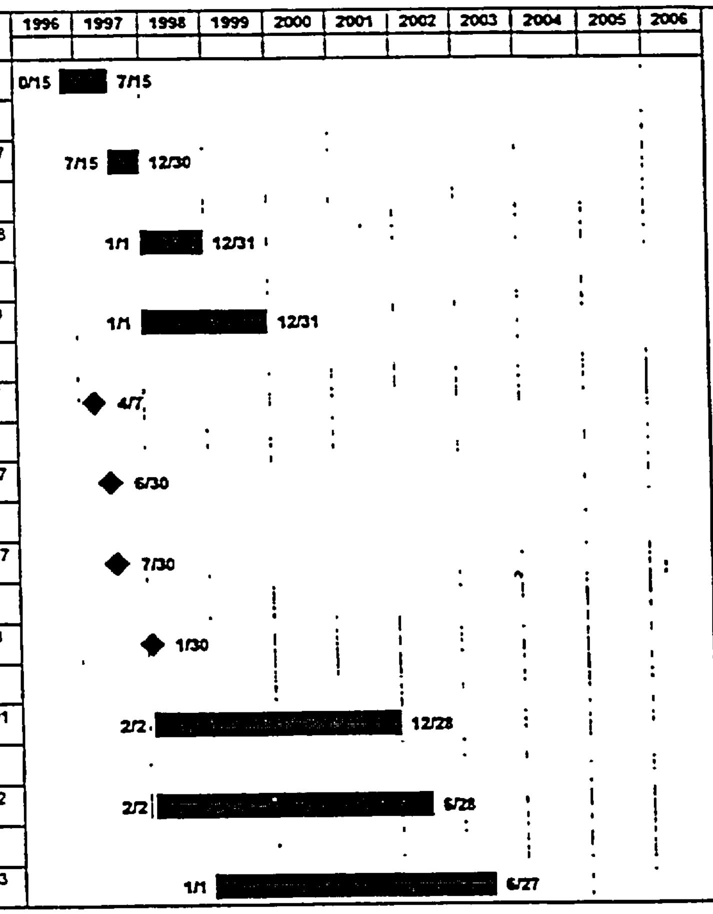
(NOT TO SCALE)

FIGURE 2-12

WTR DECOMMISSIONING SCHEDULE

WTR Decommissioning Schedule

ID	Task Name	Duration	Start	Finish
1	Project Preparation (SNM-770 & TR-2)	196d	Tue 10/15/96	Tue 7/15/97
2				
3	Remove Structures and Interferences (SNM-770)	121d	Tue 7/15/97	Tue 12/30/97
4				
5	Remediate Contaminated Soil	261d	Thu 1/1/98	Thu 12/31/98
6				
7	Remediate Retired Facilities	522d	Thu 1/1/98	Fri 12/31/99
8				
9	Submit Conceptual WTR Decommissioning Plan	1d	Mon 4/7/97	Mon 4/7/97
10				
11	NRC Conceptual Decommissioning Plan Concurrence	1d	Mon 6/30/97	Mon 6/30/97
12				
13	Submit WTR Decommissioning Plan	1d	Wed 7/30/97	Wed 7/30/97
14				
15	NRC Decommissioning Plan Approval	1d	Fri 1/30/98	Fri 1/30/98
16				
17	Remove Reactor Vessel/Biological Shield	1020d	Mon 2/2/98	Fri 12/28/01
18				
19	Waste Packaging/Shipping	1150d	Mon 2/2/98	Fri 6/28/02
20				
21	WTR License Termination/Transfer to SNM-770	1171d	Fri 1/1/99	Fri 6/27/03

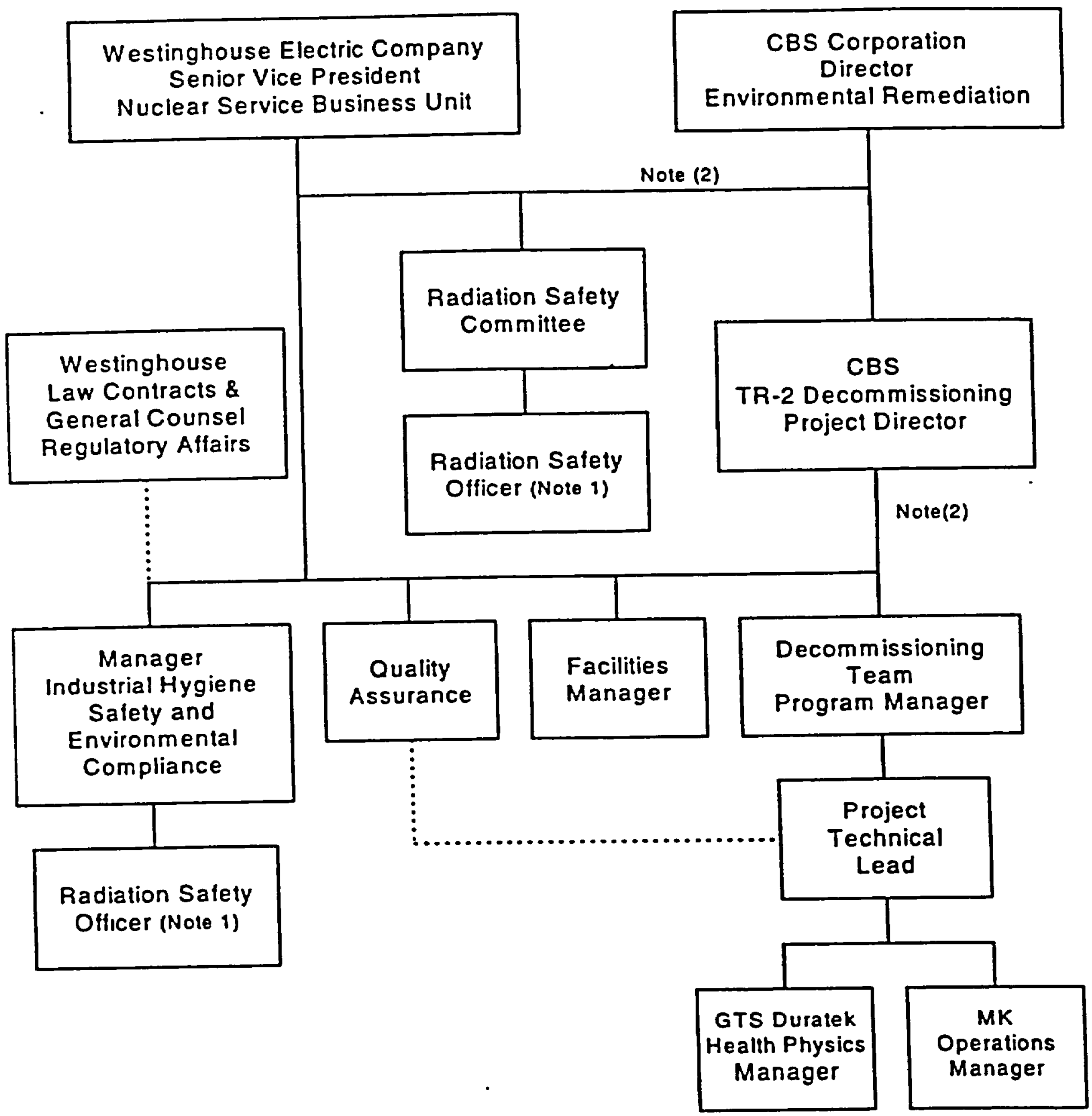


Project: WMRT.MPP
Date: Wed 7/16/97

Task | Progress | Milestone | Summary | Rolled Up Task | Rolled Up Milestone | Rolled Up Progress

FIGURE 2-13 (Rev. 2)

WTR DECOMMISSIONING PROJECT RESPONSIBILITY MATRIX



Notes:
1) The Radiation Safety Officer reports to the Industrial Hygiene, Safety and Environmental Compliance Manager and is also the Secretary of the Radiation Safety Committee.
2) Denotes a contractual relationship.

FIGURE 2-14

LOCATION OF TRANSFER CANAL

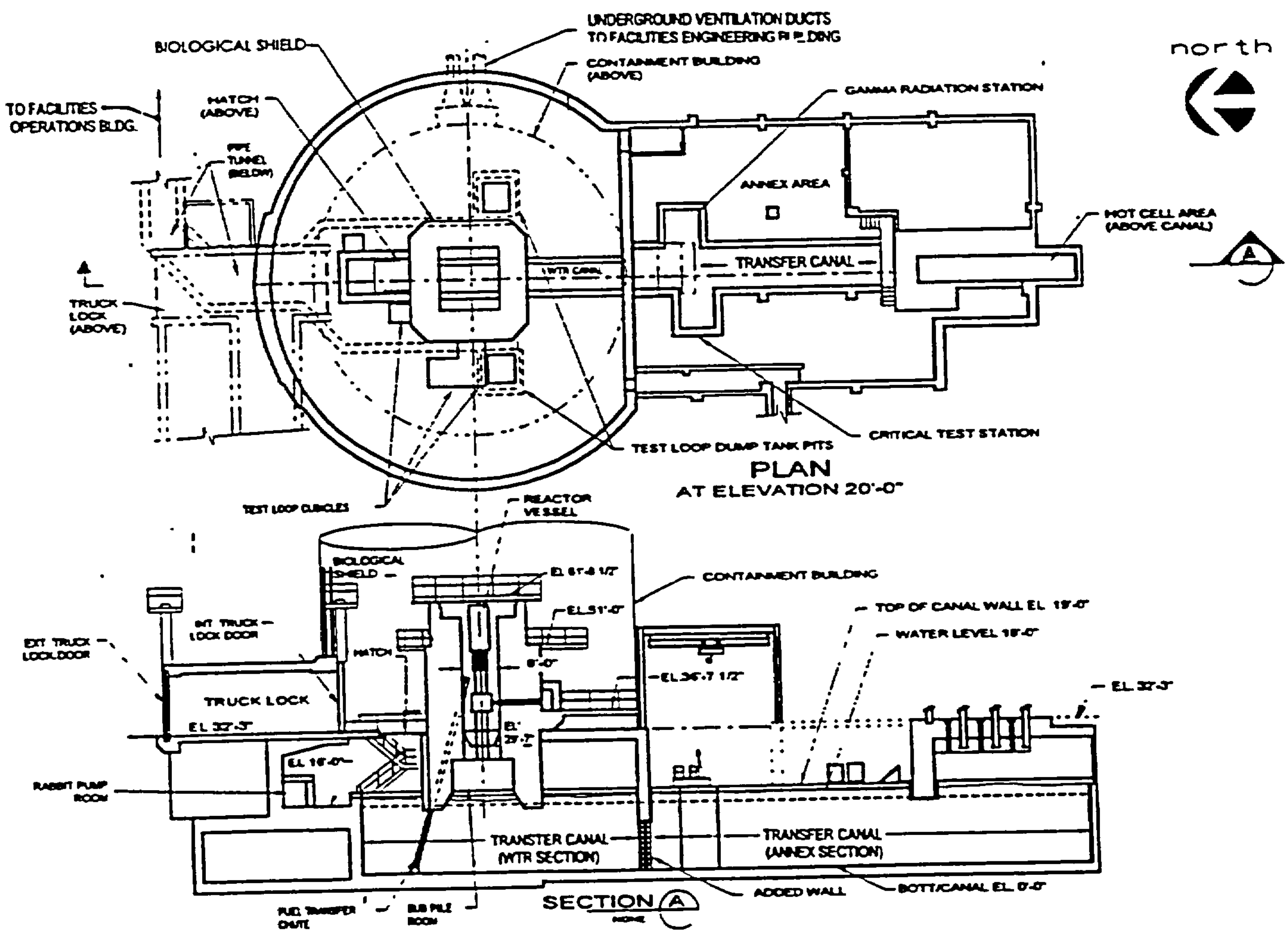


FIGURE 2-15

TRANSFER CANAL DECONTAMINATION SEDIMENT REMOVAL CONCEPT

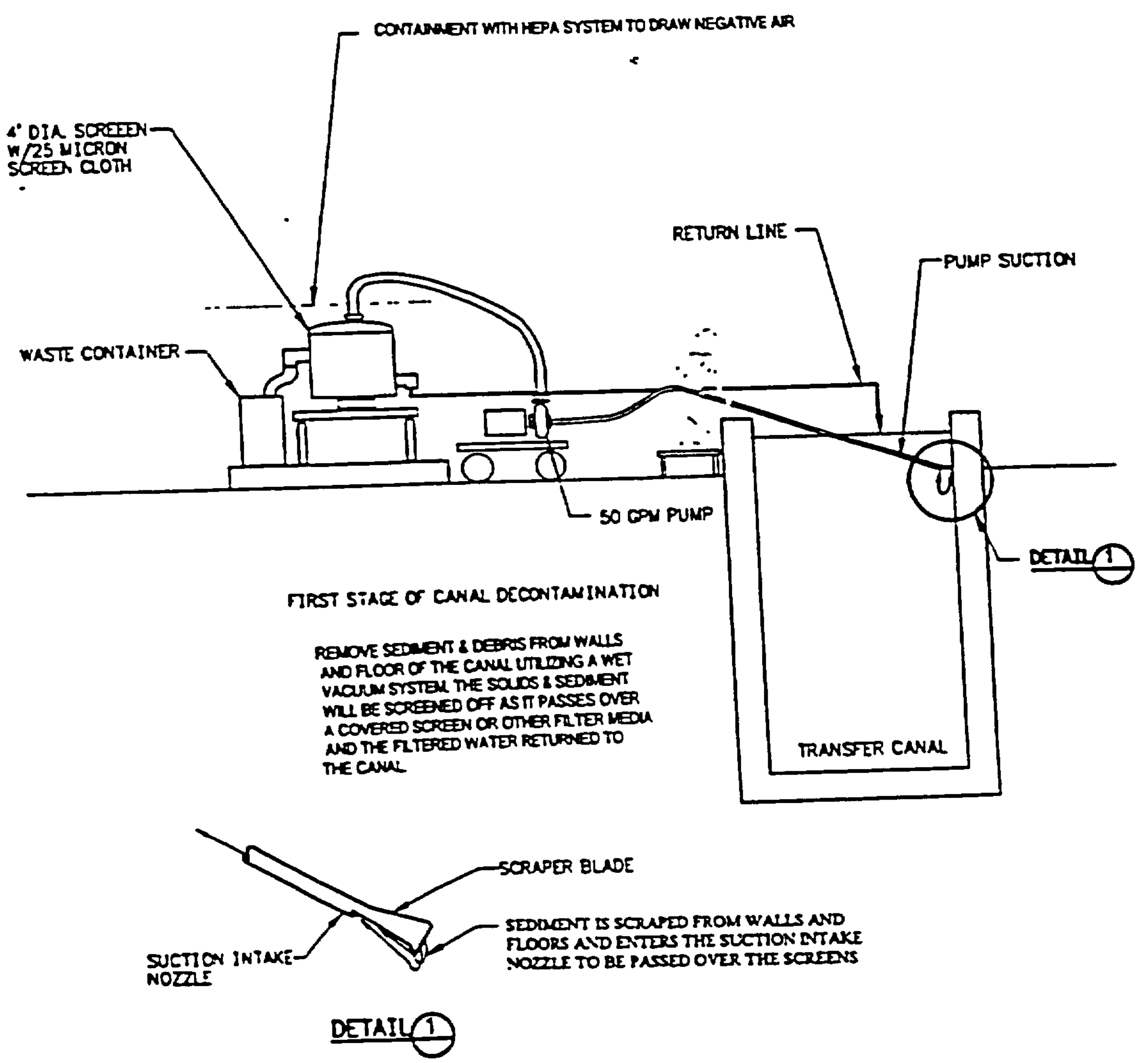
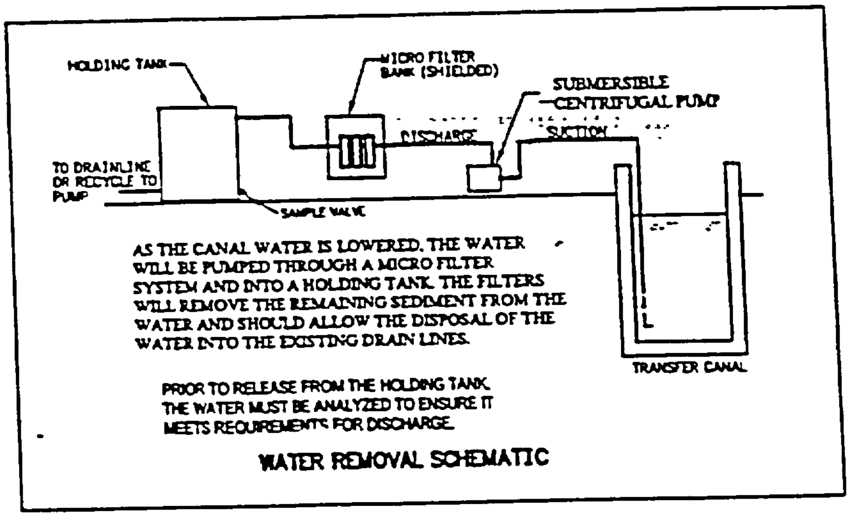
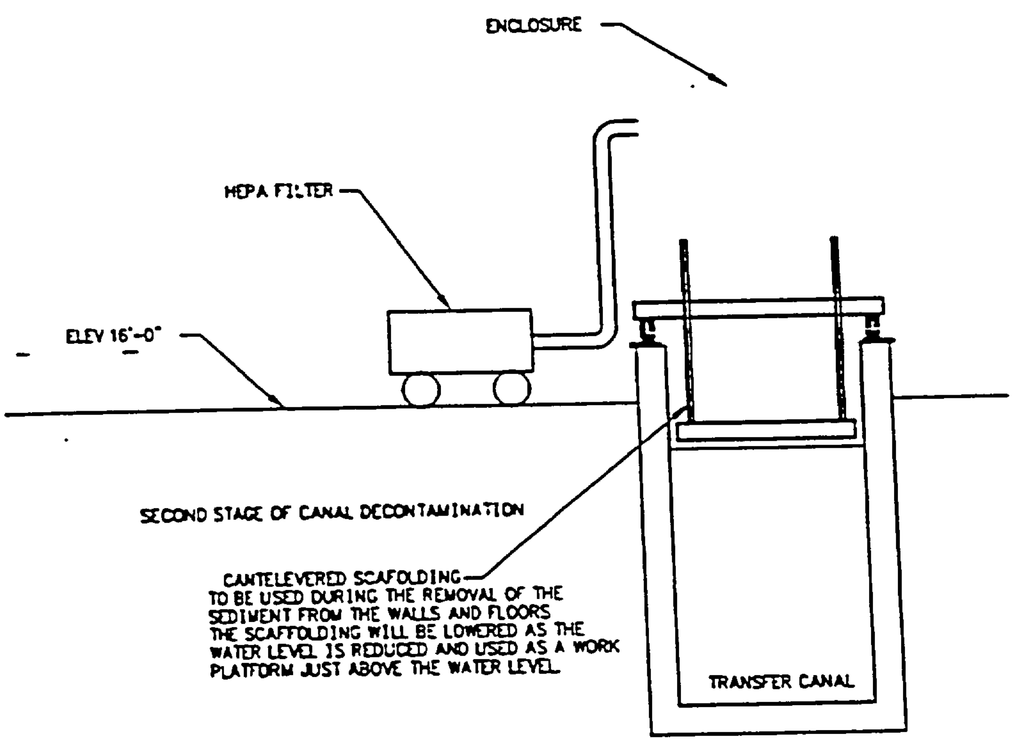


FIGURE 2-16

TRANSFER CANAL DECONTAMINATION





SECTION 3
PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

3.1 FACILITY RADIOLOGICAL STATUS

3.1.1 Facility Operating History

The WTR began operations in 1959, initially operating at a power level of 20 MWt and ultimately operating at 60 MWt. The WTR operated on commercial contracts in which various materials were inserted into the core and removed at the end of a 21-day cycle. Additional capabilities, such as the rabbit facilities, allowed specimens to be inserted into the core and withdrawn during reactor operations, independent of the 21-day cycle.

3.1.2 Current Radiological Status of Facility

Following final shut down of the WTR in March 1962, all spent fuel was removed from the site and shipped to Idaho Falls, Idaho for reprocessing. Unirradiated fuel (new fuel in storage) was returned to vendors for cold reprocessing, and irradiated specimens were returned to experimenters or disposed as radioactive waste. The reactor facility was partially dismantled, but not completely decontaminated. Some of the equipment and tooling was left in the transfer canal and reactor internals remained in the pressure vessel. The reactor head was replaced and secured in accordance with standard procedures. The vessel and primary coolant system were drained and doors were sealed or secured to prevent unauthorized entry.

During 1993, a complete radiological characterization of the remaining WTR structures and components was conducted. The primary objective of this effort was to provide sufficient radiological information to develop the WTR Decommissioning Plan and facilitate realistic cost benefit analysis in support of assessing decontamination and decommissioning options.

Numerous measurements and samples were obtained and analyzed to characterize the extent of neutron activation, the radioactive contamination present on the internal surface of components, piping, etc., and the extent of fixed and transferable contamination present on internal and external surfaces of the WTR structures and systems. This included the radioactive contamination present in the WTR transfer canal water, in the canal sediment, and on the surfaces of the canal walls and components/materials present in the transfer canal.

The following sections summarize the radiological characterization of major WTR structures and systems. This summary identifies the average value for the population of gamma exposure rate, surface contamination (fixed and transferable), material activation, and volume contamination meassurements, where appropriate. A more detailed presentation of the radiological data, as well as the survey and sample collection methodology, characterization, Quality Assurance/Quality

Controls, and the determination and documentation of background radiation measurements are found in Reference 1 of this Section.

In the following summary of radiological data, "N/A" identifies a specific parameter which was not measured; MDA indicates a value which was below the measurement Minimum Detectable Activity (MDA). MDA values for the specific measurements are provided in the WTR Characterization Report. Additional systems, not identified in this section, are also included in the WTR Characterization Report. These consist of Electrical Conduit and Boxes, Plant and Instrument Air Lines, Deionized Water Lines, Hydraulic Lines, CRDM Gland Seal Piping, and Steam and Condensate Lines.

3.1.2.1 WTR Structures

Operating Floor (16' Elevation):

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Floor	MDA	15,000
Walls @ floor level and 6'	N/A	620
Wall Hot Spots	N/A	23,000

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Floor	24	550
Perimeter Wall	MDA	MDA

Gamma Exposure Rates (μR/hr)

Description	Gamma
Floor @ 1 meter	59

Concrete Core Samples
Total Gamma Activity (µCi/g)

Description	0.5" Depth	1.0" Depth
NW Quadrant	1.7E-6	3.6E-7
NE Quadrant	9.03E-5	9.84E-6
SE Quadrant	3.3E-6	7.9E-7
SW Quadrant	3.1E-6	2.7E-7
Personnel Hatch	8.7E-5	N/A

The primary contaminant identified by gamma isotopic analysis in all core samples was Cs-137 (Co-60 was identified in cores from the Northeast Quadrant and Personnel Hatch, and accounted for 4% and 2%, respectively, of the total activity).

Composite of 0.5" Concrete Core Samples
Gamma and Beta Emitting Radionuclides (µCi/gram)

Cs-137	4.7E-5
Ni-63	1.6E-5
Sr-90	7.6E-5
Total Activity	1.39E-4

Composite of Paint Scraping Samples (µCi/gram)

Cs-137	7.6E-6
Co-60	3.4E-6

Sub-Pile Room:

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Floor	MDA	14,000
Floor beneath east valve bank	28	20,000

Gamma Exposure Rates (mrem/hr)

Description	Gamma
Floor @ contact	< 1

Composite of 0.5"and 1.5" Concrete Core Samples
Gamma Activity (μCi/g)

Nuclide	0.5" Depth	1.5" Depth
Cs-137	2.7E-5	1.7E-6
Co-60	1.1E-5	N/A
Sr-90	7.6E-5	N/A
U-234	2.7E-5	N/A
Pu-238	1.9E-5	N/A
Pu-239,240	2.6E-6	N/A
Total Activity	1.6E-4	1.7E-5

Composite of Paint Scraping Samples
Gamma Activity (μCi/g)

Cs-137	8.9E-6
Co-60	2.8E-6

Composite of Lead Scraping Samples
Gamma Activity (μCi/g)

Cs-137	7.3E-6
Co-60	7.9E-6

Rabbit Pump Room:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Floor	N/A	34,000
Walls	N/A	440
Ceiling	N/A	530

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Floor	MDA	570
Walls	MDA	MDA

Gamma Exposure Rates (mrem/hr)

Description	Gamma
Floor @ 1 meter	17

Concrete Core Samples
Gamma Activity (µCi/g)

Nuclide	0.5" Depth	1.0" Depth
Cs-137	3.1E-5	3.5E-6
Co-60	3.5E-6	N/A

Test Loop Shield Cubicles:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Gas Loop Test Cubicle Floor	MDA	65,000
Walls	MDA	227
West Test Cubicle Floor	N/A	240
Walls	MDA	235
North Test Cubicle Floor	N/A	N/A
Walls	230	590

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
North Test Cubicle Walls	MDA	MDA
West Test Cubicle Walls	MDA	MDA
Gas Test Cubicle Walls	MDA	MDA
Gas Test Cubicle Walls	MDA	490

Gamma Exposure Rates (μR/hr)

Description	Gamma
Floor @ contact	86
Floor @ 1 meter	18

Concrete Core Samples
Gamma Activity (μCi/g)

Description	Nuclide	0.5" Depth	1.0" Depth
North Cubicle Walls	Cs-137	2.1E-6	2.0E-6
	Co-60	1.5E-6	1.7E-6
	Eu-154	N/A	1.5E-6
West Cubicle Walls	Cs-137	1.7E-6	1.1E-6
	Co-60	2.0E-6	6.9E-7
Gas Test Cubicle Floor	Cs-137	3.5E-4	1.4E-6

Test Loop Dump Tank Pits:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Walls	MDA	67,000

Transferable Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Pit Floor	MDA	3,700
Pit Walls	24	1,600

Gamma Exposure Rates (μR/hr)

Description	Gamma
Floor @ 1 meter	99

Concrete Core Samples
Gamma Activity (μCi/g)

Description	Nuclide	0.5" Depth	1.0" Depth
Pit Walls	Cs-137	1.6E-5	2.5E-6
	Co-60	1.1E-5	2.6E-6
Pit Floor	Cs-137	1.6E-5	8.0E-7
	Co-60	1.2E-5	5.5E-7

Truck Lock Platform:

Direct Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Floor Tile Surface	94	1,600

Transferable Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Floor	MDA	120

Gamma Exposure Rates (µR/hr)

Description	Gamma
Platform @ 1 meter	13
East of Bioshield @ 1 meter	13
West of Bioshield @ 1 meter	8

Concrete Core Samples
Gamma Activity (µCi/g)

Description	Nuclide	0.5" Depth	1.0" Depth
Platform Floor	Cs-137	3.06E-7	N/A

Beam Port Platform:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Floor Tile Surface	N/A	1,200
Beam Port Shield	N/A	MDA
Shield Blocks	N/A	MDA
Pump Skid	N/A	MDA

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Floor	MDA	110
Beam port shield and shield blocks	MDA	MDA
Pump Skid	MDA	170

Gamma Exposure Rates (µR/hr)

Description	Gamma
Platform @ 1 meter	9
Pump Skid @ contact	8

Concrete Core Samples
Gamma Activity (μCi/g)

Description	Nuclide	0.5" Depth	1.0" Depth
Platform Floor	Cs-137	2.0E-7	1.5E-6
	Co-60	N/A	1.0E-6

Reactor Head Stand Platform:

Direct Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Platform & Catwalk	N/A	740
Reactor Head Stand	63	1,200
Wiring Raceway/Cable Chase	MDA	MDA

Transferable Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Platform, Stand and Cable Chase	MDA	MDA

Gamma Exposure Rates (μR/hr)

Description	Gamma
Platform @ 1 meter	7
Head Stand @ contact	12
Bioshield Catwalk @ contact	10
Cable Chase @ contact	8

Reactor Head Platform:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Deck Plates	96	860
Platform Storage Area	98	710
CRDMs	59	5,300
Reactor Head	84	MDA

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Deck Plates	MDA	MDA
Platform Storage Area	MDA	34
Reactor Head	MDA	300
Top of Bioshield	MDA	120

Composite of Smear Samples
Fraction of Total Gamma Activity (µCi/g)

Cs-137	0.73
Co-60	0.27

3.1.2.2 WTR Systems

Reactor Vessel and Internals:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description		Alpha	Beta
Reactor Head Access Plugs and Flange	Internal surface	220	86,000
	External surface	MDA	420
Lower Reactor Vessel Flange	Sub-pile platform and canal sides	MDA	19,000
	Bottom of vessel	MDA	7,222
	Flanges in sub-pile room	N/A	MDA
Internal Thermal Shield	3" thermal shield	620	N/A

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description		Alpha	Beta
Reactor Head Access Plugs and Flange	Plugs - Internal	38	88,000
	Plugs - External	MDA	250
	Rx head and CRD Externals	MDA	430
	Rx head and CRD Internals	MDA	890
Upper Reactor Vessel Internals	Vessel walls, 54.5' to 58'	MDA	1,700 - 2,100
	Sides and bottom of CRD	46	3,600
	Core plate top	MDA	990
	Fuel chute	MDA	550
Lower Reactor Vessel Flange	Sub-pile platform and canal sides	MDA	920
	Lower vessel internals	19	5,400
Lower Vessel Drain Internals	1" Thermal shield	MDA	240
	2" Thermal shield	MDA	430
	3" Thermal shield	MDA	570

Gamma Exposure Rates

1. Upper Reactor Vessel Internals

A gamma radiation profile was obtained at one foot intervals, starting at the head access ports and terminating approximately 12 feet into the reactor vessel. Attempts were made to maintain the detector vessel geometry constant for all measurements.

Gamma exposure rates, starting at the reactor head, range from 113 mR/hr to 1800 mR/hr.

Gamma exposure rates were also obtained for the CRD and Fuel chutes. In both cases, the gamma exposure rates dropped to below 100 mR/hr approximately 5 to 6 feet into the chute (CRD chute rate dropped to 1 mR/hr).

2. Lower Reactor Vessel

Gamma exposure rates as high as 400 mR/hr were found on some components, with a general area gamma exposure rate of 1-3 mR/hr in the Sub-Pile Room.

3. Reactor Internal Thermal Shields and Core Lattice

Thermal shield contact gamma exposure rates range from <0.2 mr/hr to 8,250 mr/hr. Core lattice gamma exposure rates range from 27 R/hr to 62 R/hr.

Isotopic Analysis of Composite Smear Samples and Material Scrapings

Smear composites, metal scrapings/cores, and samples of miscellaneous materials removed from the reactor vessel were analyzed to identify the radionuclides present and the relative abundance of each. In all cases, the primary gamma emitting radionuclides included Cs-137 and Co-60. As expected, analysis results for samples obtained from areas not directly influenced by the core neutron flux indicated that Cs-137 was the predominant gamma emitting radionuclide and for samples from areas which were in close proximity to the core, Co-60 predominated. A trace amount of Sr-90, generally much less than 10% of the total activity, was also identified in a number of the samples.

Biological Shield:

Direct Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Magnetic Plugs	MDA	380
Instrument Penetration Covers (External)	77	2,300
Pipe Sleeve Plugs	MDA	2,300
Bioshield @ Top of Reactor	MDA	21,000

Transferable Alpha and Beta Radioactivity

All average transferable alpha and beta contamination was at or below the detection MDA with the exception of the exterior surface of the instrument penetration covers which had an average beta activity of 76 dpm/100 cm^2.

Gamma Exposure Rates

Gamma radiation profiles were obtained over the penetration length of several instrument penetration tubes at intervals of one foot from the top of the bioshield down to the sub-pile room. Gamma exposure rates range from <1 mR/hr to 65 mR/hr.

Isotopic Analysis of Concrete Cores, Miscellaneous Samples and Material Scrapings

Metal scrapings/shavings, concrete cores, and samples of miscellaneous materials removed from the bioshield at various locations were analyzed to identify the radionuclides present and the relative abundance of each. In all cases, the primary gamma emitting radionuclides included Cs-137 and Co-60.

Transfer Canal:

The transfer canal remains filled with water, therefore, direct alpha and beta radioactivity measurements were not possible.

Transferable Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Canal Wall, North	5,900	270,000
Canal Wall, Sub-Pile	6,100	330,000
Canal Wall, South	7,000	320,000

Gamma exposure rate profiles were taken along the canal walls at several locations, including the north, south, and sub-pile canal sections. Gamma exposure rates were also taken at specified locations along the canal walls, bottom and on canal components, the rabbit tube and indexing station. The following are typical gamma exposure rates:

Gamma Exposure Rates (mrem/hr)

Description	Gamma
Canal Wall Top	0-2
Canal Wall Bottom	50
Rabbit Tube Indexing System	30
Canal Bottom	50
Canal Components	50

Isotopic Analysis of Composite Smear Samples and Material/Sediment

Smear composites and samples of various materials from within the canal were analyzed to identify the radionuclides present and the relative abundance of each. In all cases, the primary gamma emitting radionuclides were Cs-137 and Co-60, with Co-60 being the predominant contaminant.

Concrete Core Samples
Gamma Activity (µCi/g)

Description	Nuclide	0.5" Depth	1.0" Depth
Canal Floor	Cs-137	1.5E-3	5.0E-3
	Co-60	4.8E-5	1.2E-5
	Am-241	N/A	3.2E-4

Primary Coolant Piping Systems:

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Piping Exterior	MDA	510
Lead Bricks	MDA	1,600

Gamma Exposure Rates

The average gamma exposure rate for the primary coolant piping is 4,300 µR/hr.

Isotopic Analysis of Various Material, Sediment Samples

Samples of various materials from within the primary coolant pipe tunnel were analyzed to identify the radionuclides present and the relative abundance of each. Samples included water, sediment, scale, lead scrapings, and pipe insulation. In all cases, the primary gamma emitting radionuclides were Cs-137 and Co-60, with Cs-137 being the predominant contaminant.

Sub-Pile Room Thimble Valve Banks:

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Exterior East & West Valve Banks	MDA	280
Interior East & West Valve Banks	MDA	14,000

Gamma Exposure Rates

The average gamma exposure rate for the west valve bank is 1-3 mR/hr (indistinguishable from the sub-pile room general area gamma exposure rate).

Isotopic Analysis of Composite Smear Samples and Material/Sediment

Samples of sediment and metal from the test thimble bank were analyzed to identify the radionuclides present and the relative abundance of each. In all cases, the primary gamma emitting radionuclides were Cs-137 and Co-60, with Cs-137 being the predominant contaminant for a majority of samples.

Rabbit Pump and Piping:

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Rabbit Tube Internals	310	210,000
Rabbit Pump Internals	19	670
Rabbit Pump Externals	MDA	MDA

Gamma Exposure Rates

The average gamma exposure rate for the rabbit pump is 24 μR/hr.

Isotopic Analysis of Composite Smear Samples and Material/Sediment

Samples of the rabbit tube and from the rabbit pump drain plug were analyzed to identify the radionuclides present and the relative abundance of each. In both cases, the primary gamma emitting radionuclides were Cs-137 and Co-60.

Gas Test Loop Tanks and Piping:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Test Loop Pipe Exterior	N/A	590

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Test Loop Pipe Exterior	MDA	MDA

Gamma Exposure Rates

Average gamma exposure rates for the test loop tank exterior and interior, piping exterior, and open ended pipe range from 6 μR/hr to 12 μR/hr.

Isotopic Analysis of Smear Samples

Gas test loop smears were analyzed to identify the radionuclides present and the relative abundance of each. No activity was detected.

Chemistry Test Loop Piping:

Direct surveys for alpha and beta radioactivity indicated no elevated levels of contamination. Smear samples analysis results were less than MDA.

Test Loop Dump Tanks and Piping:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Dump Tanks and Piping Exterior	MDA	71,000
Dump Tank Interior	MDA	MDA

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Exterior of each tank	MDA	370

Gamma Exposure Rates

A survey of the tanks and piping was conducted and identified an average gamma exposure rate of 0.5 mR/hr and an average beta exposure rate of 2 mrad/hr.

Isotopic Analysis of Miscellaneous Samples and Material Scrapings

Metal scrapings/shavings and samples of miscellaneous materials removed from the west pit were analyzed to identify the radionuclides present and the relative abundance of each. In all cases, the primary gamma emitting radionuclides included Cs-137 and Co-60.

Test Loop Primary Piping:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Open-Ended Pipe - Return	N/A	4,200
Open-Ended Pipe - Supply	N/A	11,000

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Open-Ended Pipe	MDA	500
Outside of Pipe	MDA	42

Gamma Exposure Rates

Average gamma exposure rates for the open-ended supply and return piping range from 68 μR/hr to 78 μR/hr.

Isotopic Analysis of Smear Samples

Test loop primary piping smears were analyzed to identify the radionuclides present and the relative abundance of each. The primary radionuclide identified is Co-60.

Heating and Ventilation Ductwork:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Exhaust Ducts	86	980
Supply Ducts	MDA	450

Transferable Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Exhaust Ducts	MDA	MDA
Supply Ducts	MDA	MDA

Isotopic Analysis of Smear Samples

Large area wipes were obtained and composited for isotopic analysis. The primary radionuclides identified are Co-60 and Cs-137.

Low-Level Radioactive Liquid Drain:

Direct Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Water Trap Drain	N/A	2,600
External Overhead Cylinder	N/A	290

Transferable Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Water Trap Drain	MDA	MDA
External Overhead Cylinder	MDA	MDA

Isotopic Analysis of Smear Samples

Metal, water and sediment samples were collected for isotopic analysis. The primary radionuclides identified include Co-60, Cs-137, and Ag-108m.

Process Vent:

Direct Alpha and Beta Radioactivity (dpm/100 cm²)

Description	Alpha	Beta
Vent Interior	N/A	380
Vent Exterior (East)	N/A	570
Vent Exterior (West)	N/A	510

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Vent Interior (West)	N/A	320
Vent Exterior	N/A	130

Isotopic Analysis of Smear Samples

Smear composites were obtained for isotopic analysis. The primary radionuclides identified are Co-60 and Cs-137.

Polar Crane:

Direct Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Polar Crane Catwalk	91	N/A
Polar Crane Cab	N/A	N/A
Hot Spots	94	8,000

Transferable Alpha and Beta Radioactivity (dpm/100 cm^2)

Description	Alpha	Beta
Catwalk	MDA	37
Cab	MDA	35
Polar Crane Equipment	MDA	38

Gamma Exposure Rates

Average gamma exposure rates on the polar crane catwalk are 530 μR/hr on contact and 20 μR/hr at 1 meter. The general area inside the polar crane cab is 13 μR/hr.

Isotopic Analysis of Debris

Isotopic analysis of debris from the polar crane identified the primary gamma emitting radionuclides as Co-60 and Cs-137.

3.2 RADIATION PROTECTION PROGRAM

The responsibility for the site radiation program rests with the Radiation Safety Officer and Radiation Safety Committee as established under NRC License No. SNM-770, and as continued under TR-2.

The Waltz Mill radiation protection program will ensure that all radiological activities conducted during the Decommissioning Project comply with regulatory requirements. Radiological hazards will be monitored and evaluated on a routine basis to maintain radiation exposures and the release of radioactive materials to unrestricted areas as far below specified limits as reasonably achievable.

The radiation protection program will be integrated into all remediation project work activities, and each element of the program will be defined and implemented by approved policies, procedures and guidelines.

The Waltz Mill Radiation Protection (RP) Manual describes the essential elements of the program. It provides the responsibilities, authorities and qualifications, administrative policies, program objectives and standards to implement the radiation protection program. Included in the RP manual is the commitment of management to incorporate ALARA principles and philosophy into all radiological work activities. This commitment will ensure that the occupational radiation exposures for individual and collective doses and the releases of radioactive effluents are ALARA.

Established Health Physics (HP) procedures will provide guidance for performing specific tasks and methods used to maintain a radiologically safe working environment. HP procedures specify the types of instrumentation and the methods to be employed when performing surveys and obtaining samples. Examples of typical HP procedures for surveillance include:

- Radiation, surface and airborne radioactive material surveys
- Identification and posting of radiation, high radiation, surface and airborne radioactivity areas
- Access controls for radiation, high radiation, surface and airborne radioactivity areas
- Hot particle area posting and control
- Protective clothing selection, issue, donning and removal
- Protective clothing collection, cleaning, survey and reissue
- Personnel radioactivity monitoring and decontamination
- Radiological protection incidents and reports
- Radiation protection surveillance, evaluation and assessment programs

The remediation project may involve work activities which are not normally performed during site operations at Waltz Mill. To ensure the current Waltz Mill radiation protection program is adequate to protect the health and safety of workers during the remediation project, a review of the current program has been performed. As necessary, program enhancements will be implemented prior to the start of field remediation activities.

The following sections describe major elements of the program.

3.2.1 Radiological Surveillance and Work Area Controls

3.2.1.1 Radiological Evaluations

Radiological area surveys will be performed to monitor and record the radiological conditions of the work area and adjacent unrestricted areas, and to meet the site administrative guidelines and the requirements of 10 CFR 20 (Ref. 2). These surveys will identify and measure direct radiation levels, airborne radioactivity, and surface radioactivity.

Results will be documented on survey forms or recorded in logs. This information will be available to personnel entering the radiological area. The information on the survey form may include a sketch or map of the area, contact and general area dose rates, radioactivity levels, identification of specific hazards such as hot spots, and the location of radiological boundaries.

A supervisory review will be performed on all evaluations to ensure that they are adequate to assess the radiological hazards in the area, and that all information is properly recorded. The supervisor reviewing the survey will ensure that the results are consistent with those anticipated and, if not, will determine the reason for the variance.

Survey frequencies will be established at the direction of the RSO and will be based on the hazard which may be encountered, the potential for changing radiological conditions, and the frequency of occupation. Evaluations will be performed to provide positive verification that radioactive materials are being adequately controlled and are not spreading to unrestricted areas.

3.2.1.2 Radiation Work Permits

A Radiation Work Permit (RWP) system will be used for the administrative control of personnel entering or working in areas that have radiological hazards present. Work techniques will be specified in such a manner that the exposures for all personnel, individually and collectively, are maintained ALARA. RWPs do not replace work procedures, but act as a supplement to procedures. Radiation work practices will be considered when procedures are developed for work which will take place in a radiologically controlled area.

Project RWPs will describe the job to be performed, define protective clothing and equipment to be used, and personnel monitoring requirements. RWPs will also specify any special instructions or precautions pertinent to radiation hazards in the area including listing the radiological hazards present, area dose rates and the presence and intensity of hot spots, loose surface radioactivity, and other hazards as appropriate. The HP organization will ensure that radiation, surface radioactivity, and airborne surveys are performed as required to define and document the radiological conditions for each job.

RWPs for jobs with low dose and minimal hazards will be approved at the HP technician or HP supervisory level while RWPs for jobs with potentially high dose or significant radiological hazards will be approved by the RSO. Also, collective RWPs for the completion of specific work evolutions with high estimated collective or individual exposures or in high hazard work locations may require review and approval by the Waltz Mill Radiation Safety Committee, in addition to the

RSO. Examples of topics covered by implementing procedures for the Radiation Work Permits are:

- Requirements, classifications and scope for RWPs
- Initiating, preparing and using RWPs
- Extending expiration dates of an RWP
- Terminating RWPs

3.2.2 Access Control

Areas at Waltz Mill which present a radiological hazard will be posted in such a manner that personnel are made aware of the presence and extent of the hazards in the area. Areas will be posted and access controlled based on the hazard evaluation and will be in accordance with 10 CFR 20 requirements. Access restrictions and entry requirements for areas will be based on the degree of hazard present.

3.2.3 Facilities and Equipment

Sufficient facilities, equipment, and instrumentation will be available to permit the radiation protection staff to function effectively. The facilities, and types and quantities of instruments provided will be adequate to meet activity needs for the duration of the project. Radiation protection facilities could include the following functions or work areas:

- Sample analysis
- Bioassay
- Dosimetry issue, storage and calibration
- Instrument issue, storage and calibration
- Access and egress control
- Protective equipment cleaning, maintenance, storage and issue
- Personnel change areas
- Personnel, equipment and materials decontamination area(s)

Radiation protection equipment will include sample counting equipment, portable survey instruments, dosimetry and dosimetry processing equipment, protective equipment, and consumables such as smears and decontamination supplies. A nearby whole body counter is available under a continuing service contract.

Areas will be provided for the storage, repair, calibration, and issue of the project instrumentation. The operation, repair and calibration of instruments will be performed according to ANSI standards and manufacturers' recommendations as detailed in procedures or manufacturers' instructions. These procedures describe the proper techniques and the limitations for the specific piece of equipment. Calibrations will be appropriate for the anticipated radiation fields and will be traceable to the National Institute of Standards and Technology (NIST).

3.2.4 Exposure Control

Exposure control includes both the monitoring and regulation of radiation exposure. Personnel monitoring devices (dosimetry) will be required for all personnel meeting the exposure conditions specified in 10 CFR 20 and in the administrative radiation protection procedures.

Administrative exposure limits will be used to ensure that personnel do not exceed the exposure limits specified in 10 CFR 20. The administrative limits will also serve as a management tool to ensure that individual and collective doses are maintained ALARA. Administrative exposure limits have been established in such a manner that increasing exposure levels require increasing levels of management approval.

3.2.4.1 External Whole Body Monitoring

External radiation monitoring will be accomplished through the use of primary and secondary dosimeters, such as thermoluminescent dosimeters (TLDs), self-reading dosimeters (SRDs), and electronic dosimeters. The official record of accumulated external exposure will normally be obtained from the TLD primary dosimeter. Secondary dosimetry will be used as a back-up to the primary and as a means for tracking exposure between processing periods.

Personnel are required to wear external radiation monitoring devices whenever work assignments require access to radiologically controlled areas. Primary and secondary dosimeters will be worn on the trunk of the body between the neck and waist in close proximity to each other unless the RWP specifies otherwise.

Multiple whole body monitoring may be required when work is performed in a non-uniform radiation field and when the portion of the body receiving the highest exposure is not easily determined or is subject to change. RWPs will be used to specify multiple dosimetry requirements (location and position) for personnel working in these areas.

Primary dosimeters will be processed by a facility which has current accreditation from the National Voluntary Laboratory Accreditation Program (NVLAP). Secondary dosimeters will be calibrated on a semi-annual frequency using NIST traceable radiation fields.

Dose information from other sources may replace or supplement primary dosimeter results. Such action may be necessary if the primary dosimeter results are unavailable due to loss or damage or if the results are suspect. In these cases, the actions taken and the justification for such actions will be documented and approved by the RSO according to procedures for evaluating exposure when dosimetry is lost, damaged or contaminated.

3.2.4.2 Special Monitoring

Extremity monitoring devices will be worn when exposure conditions warrant their use. Specific criteria to be used in determining the need for extremity monitoring and for determining the extremity dose will be identified in procedures or provided on the RWP.

3.2.4.3 Skin Monitoring

Monitoring of the skin of the whole body will normally be accomplished with whole body dosimetry. Instructions regarding the proper method for wearing the dosimeter so that the skin dose will be properly measured will be provided in procedures, RWP and/or personnel training. Guidance may also be provided in radiation protection procedures and specified on RWPs for use of protective clothing (or other tools for reducing beta, x-ray, and/or low energy gamma radiation) to reduce skin exposure.

When it is suspected that the whole body dosimeter does not provide proper measurements of the skin dose, calculations will be performed according to applicable radiation protection procedures or using accepted industry computational models.

3.2.4.4 Internal Exposures

Internal radiation exposure will be minimized by establishing exposure limits and administrative exposure controls, identifying and controlling sources or potential sources of airborne radioactivity, and through the use of engineering controls. Respirators may be used when engineering controls may not adequately protect the worker.

Breathing zone air sampling, and/or bioassay measurements will be used to determine intakes of radionuclides. Breathing zone air sampling will be the principle means for determining the amount of intake of radioactive material.

The bioassay program will provide a quality control check of the success of the program in minimizing internal radioactivity of personnel. Bioassay may include whole body counting (in-vivo) and/or analysis of excreta (in-vitro). Bioassay results will be used to evaluate potential intakes, estimate the magnitude of intakes, and provide data necessary to assess the committed effective dose equivalent.

Procedures will include criteria for the performance of bioassay, as well as methods for data analyses, interpretation, and dose assessment. The methods and techniques prescribed by these procedures will follow the applicable guidance documents (Regulatory Guides 8.9 and 8.26) (Refs. 3 and 4).

3.2.5 Respiratory Protection Program

The respiratory protection program will be established in accordance with 10 CFR 20 Subpart H (Ref. 2) and 29 CFR 1910.134 (Ref. 5), and will be based on the guidance provided in ANSI Z88.2 (Ref. 6), NRC Regulatory Guide 8.15 (Ref. 7), and NUREG-0041 (Ref. 8).

Elements of the respiratory protection program include:

- Respirator selection and use
- Training programs
- Medical evaluations
- Fit testing
- Respiratory protection equipment maintenance and issue records

- Air quality standards for supplied breathing air systems
- Bioassay

3.2.5.1 Respirator User Qualification

Respirator users will be screened and/or examined to establish physical and psychological capabilities necessary to perform tasks using a respirator. Personnel will be trained before using respiratory protection devices. Personnel will be fit tested before the first use of respirators requiring a face piece-to-face seal and on an annual basis.

3.2.5.2 Respiratory Protection Equipment Description and Selection

The requirement for and selection of respiratory protection equipment for radiological purposes will normally be determined and approved as part of the RWP process. Project supervisors will be responsible for worker compliance with RWP requirements. Only National Institute of Safety and Occupational Health (NIOSH) certified respiratory protection equipment will be used. Routine and emergency issue of respirators will be performed according to applicable procedures. Breathing air may be supplied to respirators from compressed breathing air cylinders, air compressors, or the plant breathing air system. All sources of supplied air will meet the requirements for Grade D or better breathing air.

3.2.5.3 Equipment Inspection and Maintenance

Requirements and techniques for inspection and maintenance of respiratory protection equipment will be performed according to manufacturers' and regulatory requirements. Respirators will be cleaned, sanitized, inspected and maintained according to approved procedures. Respirator repairs will be performed by qualified personnel with parts designed for the respirator. Respirators ready for use will be stored to protect against dust, sunlight, heat, extreme cold, excessive moisture, and damaging chemicals.

3.2.6 Radioactive Materials Control Program

Radioactive material control will be implemented through procedures. The radioactive materials control program will be effective in preventing the spread of radioactive materials.

3.2.6.1 Radioactive Material Storage

Radioactive material will be stored in specially designated restricted areas. These areas may contain reusable equipment and tools, waste awaiting processing, wastes or other materials prepared for shipment, or equipment and tools awaiting decontamination or reuse. Procedures describe posting requirements, access controls, survey requirements and controls placed on the movement of equipment/materials to and from the storage area.

3.2.6.2 Contamination Control Program

The contamination control program is designed to reduce and minimize contaminated areas, tools, and components, prevent the spread of radioactivity and maintain releases of radioactive materials

ALARA. Major components of this program include the identification, posting and control of contaminated areas, decontamination of tools, equipment and areas, engineering controls, and monitoring. Procedures provide guidance for identifying the extent of contaminated areas, reducing contaminated areas, release criteria, and decontamination of components, tools, equipment and material.

3.2.6.3 Byproduct Material Control

Procedures specify the requirements for the overall control and accountability of byproduct materials including material receipt, leak testing, accountability, safe handling, and disposal.

3.2.7 Ensuring that Occupational Radiation Exposures are As Low As Reasonably Achievable

All remediation activities will be planned and conducted in accordance with the ALARA policy and comprehensive safety programs. Westinghouse places the highest priority on conducting the Waltz Mill remediation project safely and maintaining exposures to ionizing radiation ALARA.

The primary objective of the ALARA program is to minimize exposures of workers, visitors and the general public to ionizing radiation to the maximum extent practicable, taking social, technical and economic factors into consideration. Remediation activities will be conducted in a manner that ensures the health and safety of all employees, contractors, and the general public. Westinghouse shall ensure that radiation exposures to workers and the public, as well as releases of radioactive material to the environment, are maintained below regulatory limits and that deliberate efforts are taken to further reduce exposures and releases in accordance with procedures that seek to make any such exposures or releases as low as reasonably achievable.

The ALARA plan consists of several essential elements which will be incorporated into the remediation project and have the full support of management. These elements include:

3.2.7.1 Management Commitment

Management will provide full support and commitment to reducing individual and collective exposures and ensuring appropriate controls to minimize the potential for release of radioactive material to the environment. All Westinghouse and remediation contractor project management will be held responsible for strictly adhering to the ALARA policy.

All project personnel will be made aware of management's commitment and instructed on their responsibility to execute project activities in accordance with the ALARA policy. This commitment will be regularly affirmed through training programs.

3.2.7.2 Radiation Safety Committee

The Radiation Safety Committee's main responsibility is to review and participate in the implementation of the ALARA program by reviewing ALARA job evaluations when prescribed by procedure and ensuring the program is in compliance with the terms and conditions of NRC License No. SNM-770. The committee will also ensure activities are conducted under the terms & conditions of the TR-2 license.

3.2.7.3 Radiological Performance Goals

Radiological performance goals will be established for the remediation project. Performance goals will be reviewed to ensure that they are challenging, yet achievable and set by the Radiation Safety Committee.

Remediation project performance goals may include:

- Collective exposure for the remediation project
- Collective exposure for remediation project organizations
- Maximum annual individual exposure
- Number and type of project personnel contamination events
- Number of radiological occurrences

3.2.7.4 Plans and Procedures

The organization, responsibilities, and method of operation of the ALARA program will be addressed in supporting procedures. In addition to the ALARA policy statement, procedures and/or guidance documents will include:

- Responsibility for and performance of ALARA job reviews
- Radiation Safety Committee charter
- Radiation Work Permits

3.2.7.5 Radiological Work Planning

Waltz Mill remediation activities inside the restricted area (radiological work) will be conducted according to radiation protection procedures to control workers' exposure and the spread of radioactive material. Work involving significant radiation exposure will be planned as far in advance as practical. During planning, unnecessary work steps will be deleted, radiation and radioactivity levels in the work area will be determined, and collective exposure estimated. The following considerations may be factored into all work plans:

- Determine needed tools, parts, equipment, etc. before the work begins and stage them to minimize delays
- Coordinate efforts of different groups, such as decontamination, construction, radiation protection, so work can proceed in a systematic and efficient manner

- Minimize the number of workers assigned to a particular job
- Coordinate work by area so that work, such as scaffold and shielding installation and removal, is not duplicated for multiple tasks to be performed in the same area
- Perform as much work outside of radiation areas as possible
- Remove or shield sources of radiation to allow work to proceed in lower gamma exposure rate areas
- Identify required procedures and facilitate review and concurrence before work may proceed
- Identify special tools (including robotics, remote handling equipment, video monitors, etc.) and temporary services (including auxiliary lighting, power, communications) and ensure availability and operational status at the work location

All radiological work will be conducted according to the specific requirements of a RWP. RWPs are effective tools for communicating radiological information; providing instruction to workers regarding radiological work requirements; defining radiation protection practices, instruments, and equipment to be employed; and special requirements, such as specific procedure requirements, engineering controls, containments, shielding, etc. For ALARA purposes, a preliminary estimate of time and exposure for the activity and any special ALARA controls may be provided, as appropriate. The RWP will be reviewed by the supervisor of the organization responsible for the work and, if necessary, reviewed by the Radiation Safety Committee or designated HP personnel, as appropriate, before beginning work activities.

A formal ALARA job review will be conducted for work that has the potential to exceed radiological thresholds established by the Radiation Safety Committee and incorporated in implementing procedures.

3.3 RADIOACTIVE WASTE MANAGEMENT

This section addresses the technologies, equipment, and procedures to be implemented for the management of radioactive waste during the project. These technical approaches are based upon experience and address facets of planning, decontamination, packaging, storage, transportation, volume reduction or beneficial reuse, and final disposition of the waste materials, while minimizing secondary wastes and radiation exposure.

In developing the radioactive waste management program, the following elements will be considered:

- Location and availability of disposal facilities
- Potential for off-site release during remediation operations
- Preventing contamination of uncontaminated areas
- Use of existing facilities to support the waste packaging operations
- Methods of approach related to waste type, waste class, and impact on safety
- Cost effectiveness
- Logical approach to remediation operations
- Ensuring that the occupational exposures are maintained ALARA
- Minimizing the impact on the health and safety of the general public
- Maintaining flexibility for waste management to allow for unexpected wastes and changes in available technology
- Minimization of radioactive waste
- Quality Control

On-site packaging or processing of radioactive waste prior to transportation will be performed in areas designated for these activities. Except for lead shielding, no sources of mixed waste have been identified. No chemicals or other substances will be used during remediation operations that may become hazardous wastes or result in mixed waste. To reduce or avoid the generation of mixed wastes, project management will control the use of any chemical or other substance that could become a mixed waste concern.

If hazardous materials containing radioactive material are identified during remediation, they will be classified and stored on-site until declassified or approved for disposition. These materials will be managed according to Federal, state, local and site permitting requirements to the extent it is not inconsistent with NRC handling, storage and transportation regulations.

3.3.1 Radioactive Waste Processing

3.3.1.1 On-Site Radioactive Waste Volume Minimization

Minimization of the quantity of radioactive waste requiring disposal is a high priority during the project. Project management will incorporate the radioactive waste volume minimization practices into work procedures. The following elements will be included as appropriate:

- Radiation worker training will identify policies and practices to prevent the unnecessary generation of mixed or radioactive wastes.
- Unnecessary generation of radioactive and mixed wastes will be minimized by controlling chemicals brought on-site, and preventing unnecessary packaging, tools and equipment from entering radiologically controlled areas.
- Some materials will be reused during the remediation project. This typically includes contaminated tools, equipment, and clothing (after laundering).
- The volume of radioactive waste will be minimized by decontaminating areas and equipment where practical, and by segregating waste as radioactive and non-radioactive, where practical.
- Decontamination activities will be planned to minimize the generation of secondary waste volumes as a result of decontamination processes.
- Bulky material may be dismantled or cut up to reduce volume. Metal materials, incinerables and compactible waste may be sent off-site for processing.
- Waste containers for direct burial will be packaged so that void space is minimized. Space around large bulky objects will be filled with small items and debris, if conducive to ALARA.

3.3.1.2 Off-Site Shipments of Radioactive Materials for Further Processing

The project will result in the accumulation of significant volumes of contaminated and/or activated material and debris. To minimize the volume of material for disposal, licensed waste processors, capable of reducing this volume to the maximum extent practical, will be utilized when cost effective. This will be determined through cost benefit analyses performed for each waste stream and/or waste category.

3.3.1.3 Liquid Waste Processing System

Contaminated water may be generated as a result of draining, decontamination, and cutting processes. The contaminated liquids will be processed either in the existing liquid waste processing system located in the basement of the Facilities Operations Building or in a temporary treatment system (e.g., ion exchange and filtration system), solidified or sent to a licensed waste processor. All liquid radioactive waste will be processed according to approved procedures for waste collection and discharge. Liquids released from site are monitored and controlled to ensure all releases of radioactivity to the environment are as low as is reasonably achievable and that the releases meet established administrative controls, and regulatory criteria.

3.3.1.4 Local Ventilation

Local HEPA filtration systems will be used when activities could result in the generation of significant airborne radioactive particulate activity. If the filtered air is exhausted directly to an unrestricted area, appropriate air sampling will be performed at the point of discharge.

3.3.2 Radioactive Waste Disposal

3.3.2.1 Waste Classification

Proper classification of waste for disposal will be conducted using procedures which implement the requirements of federal regulations and disposal site criteria. Procedures will ensure that a realistic representation of the distribution of radionuclides in waste is known and that waste classification is performed in a consistent manner. Any of the following basic methods, used individually or in combination, will be used to achieve this goal: materials accountability (including process knowledge and activation analysis), classification by source, gross radioactivity measurements, and measurement of specific radionuclides.

Waste characterization will be performed on samples from each area (building), room or process. Individual waste stream designations will be established for areas and processes that have dissimilar radionuclide distributions and physical properties (dry active waste, liquids, sludges, etc.)

Appropriate instrumentation will be used to determine the type and quantity of radioactive material in each waste stream. Samples for radionuclide distribution will be obtained from wastes, whenever practicable. The curie content of each package can be calculated when radionuclide concentrations are determined, or by using a dose rate to curie conversion factor. Characterization will be performed by monitoring and/or sampling before packaging, and the activity of each radionuclide present in the mixture will then be used to estimate the activity in the final package. Radioactive waste will be classified as A, B, C or greater than Class C according to 10 CFR 61.

Table 3-1 contains an estimate of the radioactive waste anticipated to be generated during WTR decommissioning. This Table includes the total volume anticipated for one piece reactor vessel, biological shield, and vessel internal removal (2050 ft^3), the total volume for multiple piece reactor vessel, biological shield, and vessel internal removal (2584 ft^3), and those areas described in Section 2.7. It is anticipated that the majority of this waste will be Class A.

3.3.2.2 Waste Packaging, Transfer and Storage

Waste will be packaged at the point of generation or at a designated location on site. Packaging may include approved disposal containers. Other forms of packaging may include radioactivity control measures, such as bagging and/or wrapping, to facilitate safe transportation to other locations within the site for further processing. Heavily contaminated items may require further radioactivity control measures, such as the application of a fixative, in addition to bagging or wrapping.

After packaging, the waste will be transported to an on-site staging area and prepared for shipment. If necessary, the waste container will be placed in a storage area. Greater than Class C waste, if any, will be stored until it can be transported to a facility licensed to accept greater than Class C waste.

Waste storage facilities planned for use during remediation activities include:

- Trailers and sea/land containers may be stored and used on-site to temporarily house dry and solid low level waste
- Selected yard areas may be used for short term storage of packaged waste staged for transport
- WTR Trucklock, or other designated onsite facility for extended interim storage
- Temporary storage areas for building rubble and soil

Once all waste is removed from the storage locations, the areas will be surveyed and decontaminated, if necessary, and temporary structures removed.

3.3.2.3 Waste Transportation

Before waste is shipped from Waltz Mill, each package will be inspected to ensure it meets all applicable design and/or certification requirements and the container is not damaged or impaired. Most shipments are expected to be low specific activity (LSA) and will be shipped in exclusive-use vehicles. Radioactive material and waste will be transported by truck or rail, depending on the volume of material, vehicle availability and packaging requirements. In some cases, approved shielded casks will be employed due to radiation levels or limits for quantities of radioactivity in a package.

Some of the relevant regulatory requirements are discussed below.

- DOT Regulatory Requirements - All radioactive material/waste shipments shall be performed in accordance with DOT and other applicable federal regulations, as well as burial site requirements.
- Documentation - Radioactive waste shall only be shipped to a licensed waste processing or disposal facility. Where applicable, a state user's permit may be required. All required documents shall be complete and legible and shall meet the requirements in 49 CFR, 10 CFR 20, and the receiving facility license requirements.
- Shipping Routes - The actual routing of shipments may vary with weather and highway conditions. Additionally, local and state restrictions pertaining to radioactive material transport may affect some route selections, particularly in congested metropolitan areas. The carrier is responsible for selecting the appropriate route, which must conform to applicable federal, state, and local shipping regulations and requirements.
- Burial Site or Waste Processor Acceptance Criteria - All packaging, waste form and transportation methods will be in accordance with the criteria for the intended burial or processing facility prior to shipment. The most current, applicable regulations and specific facility terms and license conditions will be used when the shipments are made.
- Quality Control - The quality control program for waste packaging and shipping will provide assurance and verification of compliance with radioactive waste shipping regulations. The remediation contractor will have the appropriate DOT certification for all waste packaging.

3.3.3 Disposal of Non-Radioactive Waste

Non-radioactive wastes will be disposed of by release to appropriate disposal facilities such as landfills, scrap yards and scrap recovery facilities. Materials that are inappropriate for surface surveys will be sampled and appropriately analyzed. Materials found to be non-contaminated will be disposed of as non-radioactive waste.

3.3.4 Release of Material for Unrestricted Use

Surface contamination surveys will be conducted for both removable and fixed contamination before potentially contaminated equipment is released from restricted to unrestricted areas. Release of equipment and packages from the Waltz Mill site to unrestricted areas shall be in accordance with Reference 9. The survey methodology used will be sufficient to detect the levels specified in Table 1 of Reference 9.

3.3.5 Hazardous Waste

All hazardous waste generated as a result of this activity, will be handled, packaged, transported, and disposed of in compliance with Pennsylvania Code, Title 25, Environmental Resources, Chapters 260--270, Hazardous Waste Regulations. Activity will be identified with current EPA ID Number PAD074953241 issued to the Waltz Mill Site.

3.3.6 Mixed Waste

All hazardous waste generated as a result of this activity, will be handled, packaged, transported, and disposed of in compliance with the consent order and agreement between Westinghouse Electric Corporation and the Department of Environmental Protection, dated August 28, 1996. At all times, Westinghouse will manage radioactive mixed waste in a manner that complies with the Solid Waste Management Act, 35 P.D. ss 6018.101 et seq, and the Department's hazardous waste regulations, 25 PA Code ss 260.1 et seq.

3.4 ACCIDENT ANALYSIS

The risk of accidents resulting in a significant radiological release and off-site dose during decommissioning activities is very low. Since decommissioning activities will not involve any irradiated fuel, only non-fuel related accident scenarios are evaluated in this section. The focus of these decommissioning accident analyses is on public health and safety.

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

1. Dropping of contaminated concrete block/rubble
2. Fire/Explosion
3. Canal Sediment Criticality and Handling
4. Rupture of a HEPA vacuum bag

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

The activity concentrations of the various components used in the following accident analyses were derived from the characterization data summarized in Section 3.1 and described in Reference 1.

3.4.1 Assumptions

The following are the major assumptions used in the following accident analyses:

1. All irradiated fuel is removed from the Waltz Mill site.
2. Although local HEPA filtration may be used for decommissioning activities, no filtration of radiological effluent is assumed for the accident analyses.
3. A worst case atmospheric dispersion factor of 3.53 E-02 sec/m;. This atmospheric dispersion factor was calculated using the guidelines presented in Regulatory Guide 1.145 (Ref. 10) and is based on the assumption that the dose receptor is located 100 meters from the radioactivity release point, the wind speed is one mile per hour, and the atmospheric stability condition is extremely stable (Pasquill's turbulence type G). A distance of 100 meters from the release location is within the 850 acre Westinghouse site property and the Waltz Mill average annual wind speed is greater than one mile per hour.
4. All releases to the environment are assumed to be ground level releases.
5. A standard breathing rate of 3.33 E-04 m^3/sec (Ref. 11).
6. The worst case effective committed dose equivalent dose conversion factors for inhalation (Ref. 11).

7. Any radiological contaminants as a result of activation of the metallic reactor vessel would remain intact and are not available for release in any of the postulated accidents, except for a release due to HEPA vacuum filter bag rupture.

3.4.2 Dropping of a Contaminated Concrete Block/Rubble Accident

Decommissioning activities at the WTR will result in the accumulation of contaminated concrete and the handling of contaminated concrete. This contaminated concrete may be in the form of a concrete block or rubblized concrete. This accident analysis used the worst case radiological activity from samples of the biological shield. This radiological activity concentration was assumed to be homogeneous in all of the concrete block, even though the actual activity levels were next to the reactor vessel. It was assumed that a 50 ton block of concrete is handled. The anticipated capacity of the containment polar crane is 25 tons and the postulated scenario of removing the activated biological shield and activated vessel in its entirety results in approximately 44 tons of concrete (out of the total lift of 148 tons). As this concrete block is being handled with a crane, it is postulated to be dropped and one percent of the activity becomes airborne and is released to the atmosphere. The resulting dose to an individual located 100 meters down wind of the release is approximately 22 mrem total effective dose equivalent (TEDE). To show the conservatism in this accident analysis, the postulated drop results in 1000 pounds of concrete dust becoming airborne and released to the atmosphere.

3.4.3 Fire/Explosion Accident

The majority of the WTR containment is comprised of non-combustible material and fire detection and suppression methods used during any thermal cutting activities make the possibility of a fire or explosion during decommissioning activities low. However, it is postulated that combustible waste (rags, wipes, anti-contamination clothing, etc.) have come in contact with contaminated surfaces and hold small quantities of radionuclides. This material used for decommissioning is placed in a sealand container, postulated to ignite and burn. It is also assumed that 0.1 percent of this entire activity is released to the atmosphere. This release percentage is conservative in that Reference 12 states that 0.015 percent of the activity would be released. The resulting dose to an individual located 100 meters down wind of the release is very small (less than 1 mrem).

It was determined that the consequences of the concrete drop scenario bound those for a postulated local explosion which involves the biological shield concrete or an explosion which would release the accumulated contents of any HEPA filters. It is assumed that a cloud of contaminated concrete, based on uniform radionuclides from the same worst case biological shield concrete samples, is created and carried off site. The drop scenario envelopes the consequences of an explosion due to the conservatism in the drop scenario calculations.

3.4.4 Canal Sediment Criticality and Handling

Since the canal sediment and canal water are categorized as optional areas to decommission in accordance with the TR-2 Decommissioning Plan (see Section 2.7), the analyses required to assure subcriticality of the canal sediment will be performed prior to remediating the canal. Additionally, the handling, stabilization, shipment, and disposal of the sediment will be evaluated prior to remediating the canal.

3.4.5 Rupture of a HEPA Vacuum Bag

For this scenario it is assumed that a HEPA vacuum collection bag ruptures at the time it is full, just prior to change-out. The resulting dose to an individual located 100 meters down wind of the release is very small (less than 1 mrem).

REFERENCES FOR SECTION 3

1. Westinghouse Electric Corporation, Waltz Mill Facility, Characterization Report, Nuclear Materials License TR-2, Test Reactor, Volumes 1 and 2, dated February 9, 1994.

2. 10 CFR 20, "Standards for Protection Against Radiation."

3. NRC Regulatory Guide 8.9, "Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program," dated July 1993.

4. NRC Regulatory Guide 8.26, "Application of Bioassay for Fission and Activation Products," dated September 1980.

5. 29 CFR 1910, "Occupational Safety and Health Standards."

6. ANSI Z88.2, "American National Standard for Respiratory Protection," dated August 1992.

7. NRC Regulatory Guide 8.15, "Acceptable Programs for Respiratory Protection," dated October 1976.

8. NRC NUREG-0041, "Manual of Respiratory Protection Against Airborne Radioactive Materials," dated October 1976.

9. NRC document, "Guidelines for Decontamination of Facilities and Equipment Prior to Release for Unrestricted Use or Termination of Licenses for Byproduct, Source, or Special Nuclear Material," dated May 1987.

10. NRC Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," November 1982.

11. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake And Air Concentration and Dose Conversion Factors For Inhalation, Submersion, and Ingestion," September 1988.

12. NRC NUREG/CR-1756, "Technology, Safety and costs of Decommissioning Reference Nuclear Research and Test Reactors," March 1982.

Table 3-1
ESTIMATE OF RADIOACTIVE WASTE VOLUME

COMPONENT	VOLUME (ft^3)
Reactor Vessel Body	210
Access Plug & Flange	97
Upper Vessel Internals	265
Lower Vessel Flange Area	85
Internal Shields & Lattice	742
Biological Shield	1,185
REACTOR VESSEL, BIOLOGICAL SHIELD, and INTERNALS SUBTOTAL (1)	**2,584**
Transfer Canal	110
Primary Coolant Piping	4,337
Thimble Valve Banks	7
Rabbit Pump & Piping	6
Gas Test Loop & Piping	2
Dump Tank & Piping	1,703
Test Loop Primary Pipe	4
Experimental Cooling Water Pipe	6
Low Level Radiological Drain	156
Process Vent	26
Electrical Conduit & Boxes	432
Plant & Instrument Air	9
Steam & Condensate Piping & Valves	15
Polar Crane	155
Sub-Pile Room	246
Beam Port Platform (elevation 37')	119
Head Stand Platform	12
Top Platform (elevation 61')	10
OPTIONAL AREAS SUBTOTAL	**7,355**
TOTAL	**9,939**

(1) The total volume for one piece reactor vessel, biological shield, and vessel internal removal (Option 1) is 2050 ft^3. The total volume for multiple piece reactor vessel, biological shield, and vessel internal removal (Option 2) is 2584 ft^3.

NOTE: It is anticipated that the majority of the waste will be Class A.

SECTION 4
PROPOSED FINAL SURVEY

The WTR decommissioning activities will result in the removal of the reactor biological shield, vessel and internal components. Also, decontamination and dismantlement activities of other structures and equipment associated with TR-2 may be performed under the provisions of the WTR Decommissioning Plan. After removal of the reactor vessel internal contents, the reactor vessel, and the biological shield, all remaining residual radioactivity and WTR facilities will be transferred to the SNM-770 License, where it will be addressed by the SNM-770 Remediation Plan. Upon completion, no materials covered by the 10 CFR 50 license will exist.

The method for determining that the WTR facility has met the decommissioning objectives and prerequisites for license termination will be an independent verification that the reactor vessel internal contents, the reactor vessel, and the biological shield have been removed. This independent verification will be performed and documented in accordance with the Project Quality Plan.

SECTION 5
FUNDING

Westinghouse has established one Financial Assurance Mechanism that encompasses all of the Westinghouse facilities that hold NRC licenses. The Financial Assurance Mechanism established by this approach meets all the requirements of the NRC's decommissioning financial assurance regulations contained in 10 CFR 50. Appropriate updates have been submitted to the NRC to maintain adequate levels of financial assurance.

In March of 1997, Westinghouse submitted to the NRC a revision to the Financial Assurance Mechanism for Decommissioning its NRC licensed facilities (Ref. 1). This submittal was reviewed by the NRC staff and found to be in compliance with the regulations (Ref. 2). Future updates will be made as appropriate.

REFERENCES FOR SECTION 5

1. Westinghouse letter, Nardi to NRC, dated March 6, 1997; Subject: "Revised Financial Assurance Mechanism for Decommissioning."

2. NRC letter, Hickey to Nardi (Westinghouse), dated April 23, 1997; Subject: "Response to Revised Financial Assurance Mechanism for Decommissioning."

SECTION 6
TECHNICAL AND ENVIRONMENTAL SPECIFICATIONS

The proposed WTR Technical and Environmental Specifications are included as Appendix A to the TR-2 Decommissioning Plan.

The WTR technical specifications are applicable to all remaining activities at the WTR, including safe storage and decommissioning activities. This is consistent with the requirements of 10 CFR 50.36.(c)(6), which provides for case-by-case development of technical specifications for non-power reactors that are not authorized to operate.

Limiting conditions for operation and surveillance requirements are identified for confinement, for ventilation systems, and for radiation and effluent monitors. These provisions provide reasonable assurance that radioactive contamination will not be released to the environment in excess of 10 CFR Part 20 limits. Flexibility is provided to permit dismantlement activities and removal of components and material from the reactor building, while maintaining ventilation and administrative controls on activities that could result in airborne contamination.

The accident analyses in Section 3.4 show that the source terms at the WTR are sufficiently small that no equipment features, such as filtered ventilation, are relied upon to maintain public doses well below regulatory guidance limits. Therefore, the requirements of the technical specifications are conservative limitations to protect workers and the public from any radioactive material sources discovered during decommissioning that might substantially exceed expected activity levels.

The environmental specifications included in Appendix A are the same sampling and monitoring requirements currently being performed for the Waltz Mill site under License No. SNM-770, with the exception that a weekly gaseous air monitoring requirement is added during activities that could produce airborne contamination in the WTR reactor building in excess of 10 CFR 20 limits. The current program periodically monitors air, water, soil, sediment, and vegetation representative environmental samples.

Administrative controls include requirements for independent review and examination of WTR programs and activities through a Review Committee. Procedures, reports, and records requirements are also identified, as appropriate, for decommissioning activities.

SECTION 7
QUALITY ASSURANCE PLAN

A Project Quality Plan (PQP) will be developed to incorporate the applicable portions of 10 CFR 50, Appendix B. In addition, the PQP will identify additional procedures and requirements that are applicable based on government and regulatory requirements, contractual commitments and supplemental quality standards.

The following is a list of the sections from the Westinghouse Quality Management System and whether they are applicable to decommissioning activities at the WTR:

1.0 Management Responsibilities - Applicable
2.0 Quality Systems - Applicable
3.0 Contract Review - Applicable
4.0 Design Control - Applicable
5.0 Document and Data Control - Applicable
6.0 Purchasing - Applicable
7.0 Control of Customer-Supplied Product - Not Applicable
8.0 Product Identification and Traceability - Applicable
9.0 Process Control - Applicable
10.0 Inspection and Testing - Applicable
11.0 Control of Inspection, Measuring, and Test Equipment - Applicable
12.0 Inspection and Test Status - Applicable
13.0 Control of Nonconforming Product - Applicable
14.0 Corrective and Preventive Action - Applicable
15.0 Handling, Storage, Packaging, Preservation and Delivery - Applicable
16.0 Control of Quality Records - Applicable
17.0 Internal Quality Assessments - Applicable
18.0 Training - Applicable
19.0 Servicing - Not Applicable
20.0 Statistical Techniques - Applicable

An extensive quality assurance program will be carried on throughout the TR-2 decommissioning effort to assure that work does not endanger public safety, and to assure the safety of the decommissioning staff.

Quality Assurance efforts during the TR-2 decommissioning period will include the following:

- performing QA functions for procurement
- qualifying suppliers
- auditing all project activities
- monitoring worker performance for compliance with work procedures
- verifying compliance of radioactive shipments with appropriate procedures and regulations
- performing dimensional, visual, nondestructive examinations or other required inspection services to assure compliance with work plans
- maintaining auditable files on the QA audits
- preparing a final report on overall performance of the TR-2 Decommissioning Project with regard to the QA function

SECTION 8
ACCESS CONTROL PLAN

8.1 CURRENT PROVISIONS

Access to the Waltz Mill site is currently controlled in accordance with an industrial security program. The entire facility is surrounded by a security chain link fence and is protected by a security guard force.

Entrances into the WTR containment building are locked and access is controlled by the RSO. Entrances to certain areas within the containment building are also locked for radiological control purposes to preclude inadvertent entry.

The Access Control Plan described in this section will address controls related to decommissioning activities in the WTR containment building. Access control requirements into radiologically controlled areas are based on 10 CFR 20 requirements and are described in Section 3.2, Radiation Protection Program.

8.2 ACCESS CONTROL PLAN

8.2.1 WTR Access Control Organization

The Waltz Mill Site Operations Manager is responsible for site access control, including:

- Gatehouse and vehicle access into the decommissioning area, and
- Emergency, medical, and fire reporting.

Access control personnel will be properly trained and will demonstrate understanding of decommissioning area access control requirements and responsibilities. Access control personnel will be unarmed and equipped for continuous on-site and off-site communications. Local law enforcement authorities should be familiarized with procedures and plant layout, and arrangements will be made to obtain their services, in the event they may be required.

8.2.2 Physical Security Measures

8.2.2.1 Physical Barriers

Physical barriers will be used to control access to the decommissioning area as follows:

- The security chain link fence that surrounds the entire Waltz Mill facility will be maintained during decommissioning.
- A personnel access gatehouse is and will be located at the main plant entry and will normally be occupied by access control personnel.
- A vehicle access gate is and will be located in the immediate vicinity of the personnel access gatehouse at the main plant entry. Use of this gate will also be controlled by access control personnel.
- The decommissioning area is and will be surrounded by a continuous permanent fence to prevent unauthorized access to restricted areas.
- Other plant gates associated with the decommissioning area will be kept locked or continuously monitored by access control personnel.

8.2.2.2 Access Authorization

Access to the WTR decommissioning areas will be controlled and permitted only to those individuals authorized by the Waltz Mill Site Radiation Safety Officer, or an authorized representative.

All persons passing through the gatehouses will be required to demonstrate valid access authorization. To ensure that only authorized individuals are granted access to the decommissioning site restricted area, decommissioning workers will be controlled through positive identification (e.g., picture badges, controlled access lists, or other means).

Visitor access to the decommissioning area must also be approved by the Waltz Mill Site Radiation Safety Officer or a designated representative.

Access to the decommissioning site restricted area does not guarantee access to radiologically controlled areas. The radiation protection staff will continue to administer the radiologically controlled area access control program. Specific requirements that must be met prior to accessing radiologically controlled areas are identified in Section 3.

8.2.3 Communications

Telephone service will be available at the main plant entry access gatehouse to allow local law enforcement authorities and other local emergency services to be contacted. Radio communications should be available for access control personnel in the event it becomes necessary to limit access to the decommissioning area or if it becomes necessary to contact local emergency services.

8.2.4 Procedures

Written procedures will be prepared and implemented to provide access control personnel with guidance for routine and abnormal occurrences. These procedures will include:

- Criteria for identifying abnormal conditions within the decommissioning area,
- Access control personnel actions, and
- Required notifications.

The types of routine occurrences to be addressed in procedures include:

- Personnel access control
- Vehicle access control
- Communications equipment and routine testing requirements
- Surveillance/inspection of decommissioning area physical barriers
- Recordkeeping requirements

The types of abnormal occurrences to be addressed in procedures include:

- Fire or explosion
- Site evacuation
- Site radiological emergencies
- Personnel disturbance
- Acts of perceived threat of sabotage
- Civil disturbance
- Suspected or confirmed intrusion sabotage attempt
- Breached security area barrier
- Unidentified person in security area
- Medical emergencies
- Theft of material

8.2.5 Changes to Current Program

The Access Control Program for decommissioning does not involve any changes to the current program that may reduce its effectiveness.

8.2.6 Access Control Transition

Upon completion of decommissioning activities in the WTR reactor building, all access control program requirements will be transferred to the access control program for the remainder of the Waltz Mill site.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 6, 2002

Mr. Richard K. Smith
Director - Environmental Remediation
Viacom Inc.
11 Stanwix Street
Pittsburgh, PA 15222-1312

SUBJECT: NRC INSPECTION REPORT NO. 50-22/1999-202

Dear Mr. Smith:

This letter refers to the inspection which was conducted on October 27-28, 1999, and January 17-20, April 19-21, and May 14-16, 2000, at the Westinghouse Test Reactor at Waltz Mill. The enclosed report presents the results of that inspection.

Various aspects of your decommissioning and safety programs were inspected, including selective examinations of procedures and representative records, interviews with personnel, and observations of the facility.

Based on the results of this inspection, no safety concern or noncompliance with Nuclear Regulatory Commission (NRC) requirements was identified. No response to this letter is required.

Although this inspection documents the removal of the reactor vessel internal contents, the reactor vessel, and the biological shield, you should note that two provisions of the Final Decommission Plan still need to be accomplished prior to termination of the TR-2 license. These are determining the residual radioactivity remaining in-situ and preparing the necessary amendments for and requesting the transfer of the remaining residual radioactivity and WTR facilities to the SNM-770 License.

We thank you for your letter of March 25, 2002 updating us on the current status of decommissioning activities and the discussions between Viacom and Westinghouse Electric Company on the transfer of the remaining residual radioactivity and WTR facilities to the SNM-770 License. We encourage you to continue focusing on completing the TR-2 decommissioning plan, as you have described in your letter. Please inform us if the situation described in you letter of March 25, 2002, changes.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at (the Public Electronic Reading Room) http://www.nrc.gov/reading-rm/adams.html. Should you have any questions concerning this inspection, please contact Mr. Stephen Holmes at 301-415-8583.

Sincerely,

Patrick M. Madden, Section Chief
Research and Test Reactors Section
Operating Reactor Improvements Program
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Docket No. 50-22
Enclosure: NRC Inspection Report No. 50-22/1999-202

cc w/enclosure: Please see next page

Westinghouse/Waltz Mill

Docket No. 50-22

cc:

Mr. James G. Yusko
Pennsylvania Department of
Environmental Protection
400 Waterfront Drive
Pittsburgh, PA 15222

Joseph Nardi, License Administrator
Westinghouse Electric Company
Box 355
Pittsburgh, PA 15230-0355

Test, Research, and Training
Reactor Newsletter
University of Florida
202 Nuclear Sciences Center
Gainesville, FL 32611

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Docket No:	50-22
Report No:	50-22/1999-202
Licensee:	VIACOM /Westinghouse Electric Company Division
Facility:	Westinghouse Test Reactor
Location:	Waltz Mill, Pennsylvania
Dates:	October 27-28, 1999, January 17-20, April 19-21, and May 14-16, 2000
Inspector:	Stephen W. Holmes, Reactor Inspector
Approved by:	Patrick M. Madden, Section Chief Research and Test Reactors Section Operating Reactor Improvements Program Division of Regulatory Improvement Programs Office of Nuclear Reactor Regulation

EXECUTIVE SUMMARY

Viacom
Westinghouse Test Reactor at Waltz Mill Facility
Report No: 50-22/1999-202

The primary focus of this routine, announced inspection conducted on October 27-28, 1999, and January 17-20, April 19-21, and May 14-16, 2000, was the on-site review of selected decommissioning activities at the Viacom, Westinghouse Test Reactor at the Waltz Mill facility. This facility is a Class III test reactor. The activities audited during this inspection included: organization and staffing; review and audit functions; plant decommissioning; procedures; maintenance and surveillance; radiation protection program; effluent and environmental monitoring; the shipment of radioactive material; and training.

Organizational and Staffing

- The decommissioning organizational structure and functions were consistent with Technical Specification Section 6.0-Administrative Controls and the Final Decommissioning Plan.

Review and Audit Functions

- The review and audit program satisfied Technical Specification requirements.

Plant Decommissioning

- Reactor decommissioning, shift turnover, and logs were acceptable.
- The control and performance of decommissioning activities were being performed in accordance with procedural requirements.
- Removal and disposal of the biological shield and reactor tank were performed as required by the Final Decommissioning Plan, licensee procedures and Department of Transportation requirements.

Procedures

- Based on the procedures and records reviewed and observations of staff during the inspection, the procedural control and implementation program satisfied Technical Specification requirements.

Maintenance and Surveillance

- The licensee's program for surveillance and limiting conditions for operation confirmations satisfied Technical Specification requirements.
- The maintenance program was being implemented as required by Westinghouse Test Reactor procedures.
- The licensee's design change procedures were in place and were implemented as required.

Radiation Protection Program

- The radiation protection program satisfied the requirements of 10 CFR 19.12 and 10 CFR 20.1101.
- Radiological postings satisfied regulatory requirements.
- Surveys were performed and documented as required by 10 CFR 20.1501(a), Technical Specifications, and licensee procedures.
- The personnel dosimetry program was acceptably implemented and doses were in conformance with licensee and 10 CFR Part 20 limits.
- Portable survey meters, radiation monitoring, and counting lab instruments were being maintained according to Technical Specifications, industry/equipment manufacturer standards, and licensee procedures.
- The respiratory protection program implemented by the licensee was acceptable and in compliance with 29 CFR 1910.134.

Effluent Monitoring

- The effluent monitoring and release program satisfied NRC requirements.
- The stack continuous air monitor event was handled and documented as required.

Transportation of Radioactive Materials

- Transportation of byproduct material by the licensee satisfied the applicable NRC and Department of Transportation regulations and in accordance with Westinghouse Test Reactor procedures.

Emergency Preparedness

- The emergency preparedness program was conducted and implemented in accordance with Westinghouse Test Reactor procedures.

Security

- The physical protection features, procedures, equipment, and access control of the Waltz Mill site and the Westinghouse Test Reactor, satisfied site procedures and the access control plan.

Training

- The 10 CFR Part 19 and Final Decommissioning Plan required radiation worker training was performed in accordance with established procedures.

REPORT DETAILS

Summary of Plant Status

During this inspection the Westinghouse Test Reactor (WTR) was undergoing active decommissioning in accordance with the TR-2 Final Decommissioning Plan dated July 25, 1997, as supplemented on March 20 and July 10, 1998, and authorized by License Amendment No. 8, dated September 30, 1998. At the time of this inspection, the Licensee's activities were specifically focused on the dismantling and removal of the reactor tank, its internal contents, and the surrounding biological shield. The Licensee's determination of residual radioactive material that will be left in-situ and its transfer to the SNM-770 materials license was ongoing at the time of this inspection.

Introduction

1. Changes, Organization, and Staffing

a. Inspection Scope (Inspection Procedure [IPs] 39745 and 69001)

The inspector reviewed selected aspects of:

- organizational structure
- staffing requirements for safe operation of the research reactor facility
- qualifications
- administrative controls

b. Observations and Findings

During the time covered by this report, the license was transferred from CBS Corporation to VIACOM. However, the Waltz Mill site organizational staff had not functionally changed. The management and decommissioning structure consisted of Level 1- Individual responsible for the license, Level 2-Individual responsible for facility activities, and Level 3-Individual responsible for day-to-day supervision as outlined in Technical Specification (TS) Section 6.1, "Organization", authorized by Amendment No. 10 dated November 23, 1999, and charted in Table 2-13 (revision 2) of the Final Decommissioning Plan (FDP). The inspector verified that these positions were filled, interviewed the individuals, and determined that they were knowledgeable about their duties and responsibilities as required by FDP Section 2.4.

c. Conclusions

The decommissioning organizational structure and functions were consistent with TS Section 6.0, Administrative Controls and the FDP.

2. Review and Audit Functions

a. Inspection Scope (IPs 40745 and 69001)

The inspector reviewed selected aspects of:

- Radiation Safety Committee (RSC) minutes
- safety review records
- audit records
- responses to safety reviews and audits
- review and audit personnel qualifications

b. Observations and Findings

Review of the RSC membership and quarterly meeting schedule confirmed that they met TS Section 6.2, "Radiation Safety Committee", and the Committee's charter requirements. The inspector reviewed the 1999 and 2000 minutes of the RSC and determined that they provided guidance, direction, and decommissioning oversight. The RSC reviewed and approved new procedures and radiological safety significant revisions to existing ones, FDP and facility 10 CFR 50.59 changes, and TS and license change requests, as required by TS Section 6.2.3, "Review Requirements", FDP Section 2.4, "Decommissioning Organization and Responsibilities", and NRC regulations.

Committee minutes and audit records showed that safety reviews and audits were conducted as required by TS Section 6.2.4, "Audit Requirements", and the Committee's charter. The content of the safety reviews were found by the inspector to be consistent with the TS. These reviews provided guidance, direction, and oversight to ensure satisfactory decommissioning of the reactor.

The inspector reviewed the committee's approval of work package 626-08-"New opening and reenforced bumper spacer" (for the reactor vessel transport skid), the 50.59 change to the FDP increasing the size of bioshield blocks allowed to be moved in containment from 6.5 tons to 15 tons, and audits of the decommissioning and surveillance programs. The inspector determined through these examinations that the safety reviews and audits and associated findings were satisfactory and that the licensee took the appropriate corrective actions in response to these findings.

c. Conclusions

Audits, reviews, and approvals being conducted by the RSC were found to be in accordance with the requirements specified in TS and the RSC charter.

3. Plant Decommissioning

a. Inspection Scope (IPs 39745 and 69001)

The inspector reviewed selected aspects of:

- decommissioning logs and records
- staffing for decommissioning
- selected operational activities
- decommissioning procedures
- RSC minutes

- biological shield removal
- reactor tank down-ending and removal
- reactor tank shipment

b. Observations and Findings

(1) General Decommissioning Activities

The inspector reviewed randomly selected entrance, operations, health physics, and daily containment logs since February 1999. The inspector observed drilling, cutting, sawing and other decommissioning activities being performed during sampling or removal of ducting, piping, biological shield, and other reactor and auxiliary components. Decommissioning activities were carried out following written procedures as required by TS Section 6.3, "Procedures", and FDP Section 2.4.1, "Procedures." Information on operational status of the facility was recorded clearly in log books and/or checklists as required by procedures, providing a record of operational activities and events. During shift turnovers, the oncoming staff was briefed on the status of decommissioning and health physics (HP) activities.

(2) Biological Shield Removal

The reactor tank was surrounded by a "biological shield" made of magnetite bearing concrete. Although this high density concrete provided shielding from the radiation produced in the reactor core, it provided no structural support to the reactor tank. Prior to removing the reactor tank from the containment for disposal, this biological shield needed to be removed.

The shield was removed from the tank using primarily diamond wire cutting techniques. The inspector reviewed the cutting procedures and observed numerous vertical and horizontal sawing operations using the diamond wire cutting techniques. Additionally, the inspector observed the lifting and removal of the first and a number of subsequent concrete shield blocks cut away from the reactor tank. As experience was gained, the licensee appropriately modified its cutting, lifting and removal procedures. In addition, the inspector observed that the operations were performed deliberately, in accordance with the FDP, approved procedures, and with safety as a fundamental interest.

(3) Reactor Tank "down-ending" and Removal

After the surrounding concrete biological shield was removed, the penetrations in the reactor tank were sealed and the tank internals were fixed in place, the tank itself was now ready to be removed.

The tank was first separated from the concrete containment floor by severing its hold-down bolts. Next it was lifted off the floor by use of a heavy lifting crane and pulled on its side and onto a transport skid by cables attached to its base. The tank was then packaged for rail shipment as described in the licensee's Department of Transportation (DOT) exemption request dated December 15, 1999, (DOT exemption DOT-E 12404, issued April 28, 2000, expiring April 30, 2001). The package was then transported to the onsite rail siding to await shipment.

The inspector reviewed the lifting, down-ending, and packaging procedures. The inspector observed the lifting and down-ending of the tank onto its transport skid and found them being performed in a controlled manner and in accordance with the FDP and approved procedures.

(4) Rail shipment for Disposal

The shipped reactor tank and its internals was being shipped to Alaron Corporation, a disposal facility in Wampum, Pennsylvania, just north of Pittsburgh, for processing and final disposal. The shipping package was first transferred to a rail car at the onsite rail siding, then assembled into a special train, and finally transported to the Alaron Corporation site, a distance of approximately 100 miles. The trip entailed one change of engines and crews, as the route covered two individual railroad lines.

The Inspector reviewed the DOT exemption and WTR shipping requirements, inspected the shipping package on its rail car, and accompanied the shipment to its destination. During the trip, the inspector observed the Waltz Mill escort personnel performing radiation surveys, inspecting placarding, providing instructions to the railroad crews, and issuing dosimetry. Additionally, the inspector interviewed one railroad crew in regards to their understanding and responsibilities for the shipment.

Based on the observations and interviews, the inspector determined that the radiation surveys, placarding, dosimetry issuance, and training provided to the railroad crews, met the DOT exemption and WTR shipping requirements.

c. Conclusions

Based on the procedures and records reviewed, observations made, and interviews performed during the inspection, the inspector determined that the control and performance of decommissioning activities related to the removal and disposal of the biological shield and reactor tank were acceptable and in accordance with the DOT exemption, FDP, licensee procedures, and TS requirements.

4. **Procedures**

a. <u>Inspection Scope (IP 69001)</u>

The inspector reviewed selected aspects of:

- administrative controls
- records for changes and temporary changes
- decommissioning procedures
- procedural implementation
- logs and records
- Waltz Mill Westinghouse Test Reactor Decommissioning Project Control Manual (WMDT) Rev 11, Issued October 2001.

b. <u>Observations and Findings</u>

Over the course of the inspection, the inspector reviewed the following work packages and procedures regarding the cutting and removal of the biological shield, the down-ending and removal of the reactor tank, the packaging and transport of the reactor tank, and applicable HP activities:

WP-1	Project Management
WP-3	Establish Access Through Truck Lock
WP-615	HEPA Filtration/Ventilation Systems
WP-620	Immobilize Reactor Tank Internals
WP-621	Partial Removal of the Bio-Shield
WP-623	Stabilize Core Region
WP-625	Structural Removal
WP-626	Remove and Package Reactor Tank
WMDT-004	Reactor Tank Shipment Transportation, Health and Safety Plan
WM-HP-ADMIN-105	Analytical Laboratory Administrative Procedure
WM-HP-ADMIN-106	Dosimetry Program Requirements
WM-HP-ADMIN-109	Radiation Protection Instrumentation Program
WM-HP-ADMIN-112	Survey and Surveillance Program Requirements
WM-HP-ADMIN-118	Radioactive Material Management Program Requirements
WM-HP-ALARA-134	Request, Authorization, and Issue of Radiation Work Permits
WM-HP-ENVIR-22	Environmental Radiation Surveillance
WM-HP-ENVIR-47	Stack Monitoring
WM-HP-INST-7	Calibration of Laboratory Counters
WM-HP-INST-11	Calibration of Continuous Air Monitors
WM-HP-INST-12	Calibration of Count Rate Meters
WM-HP-INST-17	Calibration of Portable Radiation Survey Meters
WM-HP-INST-470	Response Checking Instruments.
WM-HP-41	Shipping of Byproduct, Source, and Special Nuclear Materials
WM-HP-REM-168	Visitor Dosimeter Issuance and Usage
WM-HP-REM-324	TLD Issue Procedure
WM-HP-SURV-220	Radiation Surveys
WM-HP-SURV-227	Surveying Material for Unconditional Release

WM-HP-SURV-230 Routine Surveillances
WM-HP-SURV-254 Air Sampling Using Portable Air Samplers

The inspector confirmed that written HP and decommissioning procedures were available for those tasks and items required by TS Section 6.3. The procedures were routinely updated while minor modifications to the procedures were made as temporary changes. Temporary changes to procedures could be made by Level 3 management and higher; however, those affecting radiation safety must also be reviewed by the RSC within 45 days. Substantial changes were effective only after approval by appropriate management or the RSC.

After review of the 1999 and 2000 training records and interviews with staff, the inspector determined that the training of personnel on procedures was adequate. During the inspector's tours of the facility, it was observed that personnel performing radiation surveys, conducting instrument checks, issuing dosimetry, performing cutting, drilling, sawing, lifting, removal, and other decommissioning activities were doing so in accordance with applicable procedures.

c. Conclusions

Based on the procedures and records reviewed and observations of staff during the inspection, the inspector determined that the procedural control and implementation program was being acceptably implemented.

5. Maintenance and Surveillance

a. Inspection Scope (IP 69001)

The inspector reviewed selected aspects of:

- maintenance procedures
- equipment maintenance records
- surveillance and calibration procedures
- surveillance, calibration, and test data sheets and records
- reactor decommissioning, periodic checks, tests, and verifications were observed
- facility design and FDP changes and records
- facility configuration
- WMDT
- Procedure WM-DT-6.2, "Decommissioning Licensing evaluation"

b. Observations and Findings

(1) General Maintenance

During decommissioning, general maintenance was focused on the support services and equipment and not on any reactor systems. Those reactor systems still required were covered under the surveillance program. The inspector interviewed logistics staff and observed minor maintenance performed on lighting, water and air service, pumps, electrical and other support equipment. Burnt out bulbs were replaced with new ones, gaskets and filters were replaced, lubricant was topped off, and standard industrial maintenance was performed on equipment. Based on the inspector's interviews and observations, general maintenance was performed as expected on an industrial site.

(2) Surveillance

The WMDT, procedure WM-DT-4.7, "Surveillance Schedule" was used to track surveillance checks, and required system/component inspections. The procedure was found to provide adequate control over the TS required tests and surveillance checks during decommissioning.

The inspector reviewed records of all TS Section 3 and Section 4 required limiting conditions for operation (LCO) and surveillance verifications performed since February 1999. Additionally, the inspector observed: 1) TS Section 3.1.3.2 required ventilation LCO verifications performed prior to restricted operations, 2) TS Section 3.3.3.2 required monitoring system LCO verifications, and 3) TS Section 4.4.3.2 required air particulate monitor surveillance. The results of the surveillances observed or reviewed were within prescribed TS limits and procedure parameters and in close agreement with the previous surveillance results.

(3) Change Control

Reactor, TS, or FDP related 50.59 changes require review by the RSC in accordance with TS Section 6.2.3, "Review Requirements" and FDP Section 1.4, "Administration of the Decommissioning Plan." The reviews are controlled by procedure WM-DT-6.2, "Decommissioning Licensing Evaluation."

The inspector reviewed the RSC approved change packages for increasing the size of the biological shield blocks which could be moved in containment and for modifying the method used to remove the reactor tank from containment and ship it. From these reviews, the inspector determined that change evaluations were technically complete and adequately documented. Additionally, the inspector concluded that RSC 10 CFR 50.59 reviews and approvals were focused on safety, and met licensee program requirements.

c. <u>Conclusions</u>

The licensee's program for surveillance and LCO verification satisfied TS and FDP requirements. The licensee's maintenance and design change programs were in place and were being implemented as required by licensee procedures.

6. Radiation Protection

a. Inspection Scope (IPs 83743 and 69001)

The inspector reviewed selected aspects of the radiation protection program (RPP):

- The Radiation Protection Manual (RPM)
- As Low As Reasonably Achievable (ALARA) reviews
- Radiation Protection Training
- radiological signs and posting
- facility and equipment during tours
- routine surveys and monitoring
- survey and monitoring procedures
- dosimetry records
- maintenance and calibration of radiation monitoring equipment
- periodic checks, quality control, and test source certification records
- event/incident records

b. Observations and Findings

(1) Radiation Protection Program

Although the RPM and individual procedures had been revised and some added, the RPP had not structurally changed since the last inspection. The licensee reviewed the RPP at least annually in accordance with 10 CFR 20.1101(c). This review and oversight was provided by the RSC as required by TS Section 6.2.4.

The inspector's review of procedure change records, RPM revisions, and radiation work permits (RWP), confirmed that the RSO specifically reviewed and approved RPP changes, experiments, and radiation protection related events/conditions as required by TS 6.1.2, FDP Section 3.2 and the RPP.

(2) Radiation Protection Postings

The inspector observed that caution signs, postings and controls to radiation, high radiation, and contaminated areas at the WTR were acceptable for the hazards involved and were being implemented as required by 10 CFR Part 20, Subpart J. The inspector observed licensee and contractor personnel and verified that they complied with the indicated precautions for access to such areas. The inspector confirmed that current copies of NRC Form-3 and notices to workers were posted in appropriate areas in the facility as required by 10 CFR Part 19.

(3) Radiation Protection Surveys

The inspector audited the daily, weekly, monthly, quarterly, and other periodic contamination and radiation surveys, including airborne activity sampling. They were performed and documented as required by TS Section 6.3, "Radiation Safety", FDP Section 3.2.1 and WTR survey procedures. HP surveys required for special decommissioning activities, such as truck door openings, RWPs, etc. were also performed and documented as required. Results were evaluated and corrective actions taken and documented when readings/results exceeded set action levels.

The inspector's review of the survey records since November 1999 confirmed that contamination, radiation, and airborne surveys were being performed as required by the RPM and individual procedures. Results were reviewed and corrective action taken when results exceeded facility action levels. Resurveys were performed to ensure corrective actions were effective.

(4) Dosimetry

The Inspector confirmed that dosimetry was being issued to staff, contractors, and visitors as outlined in licensee procedures. The licensee's dosimetry issuing criteria specifies that dosimetry be issued to individuals who might receive a dose equivalent exceeding 10 percent of the annual limits specified in 10 CFR 20.1201(a). This criteria meets the requirements of 10 CFR 20.1502 for individual monitoring. During the inspection the inspector observed that visitors, workers, and staff wore their dosimetry, including extremity dosimeters, as required.

The licensee used a National Voluntary Laboratory Accreditation Program-accredited vendor to process personnel thermoluminescent dosimetry. Dosimetry results were reviewed by the RSO and doses above the facility's ALARA limits were investigated or referred to the RSC as required. The inspector's review of 12 individual radiological exposure records, each covering the period from February 1999, to April 2000, verified that occupational doses were within 10 CFR Part 20 limitations.

(5) Radiation Monitoring Equipment

The calibration and periodic checks of the portable survey meters, radiation monitoring, air sampling, and counting lab instruments were performed by facility staff or by certified contractors. The inspector confirmed that the licensee's calibration procedures and annual, quarterly, semiannual and monthly calibration, test, and check frequencies satisfied TS Section 4.3.3, FDP Section 3.2.3, and 10 CFR 20.1501(b) requirements, and the American National Standards Institute (ANSI) N323, "Radiation Protection Instrumentation Test and

Calibration" or the instrument's manufacturers' recommendations. The inspector verified that the calibration and check sources used were traceable to the National Institute of Standards and Technology and that the sources' geometry and energies matched those used in actual detection/analyses.

The inspector reviewed the facility calibration list for 1999 and 2000 and confirmed that the calibrations for the radiation monitoring and counting lab equipment in use had been performed. The inspector verified the calibration of the lab multichannel analyzer, a containment continuous air monitor, a ventilation effluent monitor, two count rate meters, and one portable ion chamber. All instruments checked had current calibrations appropriate for the types and energies of radiation they were used to detect and/or measure.

(6) Respiratory Protection

FDP Section 3.2.5 describes how the WTR respiratory protection will be established. It will meet 10 CFR Part 20, Subpart H and 29 CFR 1910.134 requirements and use ANSI Z88.2, NRC Regulatory Guide 8.15, and NUREG-0041 for guidance.

During the time covered by the inspection, the licensee had not implemented the respiratory protection program under 10 CFR Part 20, Subpart H. The licensee determined that, based on calculations and air sampling results, respiratory protection was not required to limit intake of radioactive material. The inspector reviewed the licensee's sampling and calculation results, interviewed staff, and observed ongoing air sampling during tours. The inspector confirmed the licensee's conclusion that respiratory protection was not required to limit intake of radioactive material and determined that the continuing air sampling program was adequate to evaluate the need for 10 CFR Part 20, Subpart H respiratory protection.

Although respiratory protection was not used to limit intake of radioactive material, it was used during welding, cutting, drilling, and other decommissioning activities. The licensee provided respiratory protection based on the specific work being performed, air sampling results, calculations, or worker request as appropriate. Individual breathing zone and general area air sampling were performed for lead and silica, while airborne dust was monitored using a MIE DataRam dust monitor. All results, during the inspection, were less than 5% of the occupational limits. The inspector reviewed the licensee's and contractors' program, procedures, and training for 29 CFR 1910.134 related respiratory protection, air sampling and calculation results, observed ongoing air sampling during tours, and interviewed staff. Workers using respiratory protection were knowledgeable of its maintenance and use. All respirators checked were National Institute for Occupational Safety and Health approved and within maintenance dates. Based on this review, the inspector determined that the respiratory protection program implemented by the licensee satisfied 29 CFR 1910.134 requirements and that the continuing air sampling program was adequate to evaluate the need for respiratory protection.

c. Conclusions

The inspector determined that, because: 1) surveys were being completed and documented as required by 10 CFR 20.1501(a), Technical Specifications, and licensee procedures, 2) postings met regulatory requirements, 3) the personnel dosimetry program was acceptably implemented and doses were in conformance with licensee and 10 CFR Part 20 limits, 4) portable survey meters, radiation monitoring, and counting lab instruments were being maintained and calibrated as required, and 5) the evaluation and administration of the respiratory programs were adequately being performed, the RPP being implemented by the licensee satisfied regulatory requirements.

7. Effluent Monitoring

a. Inspection Scope (IP 69001)

The inspector reviewed selected aspects of:

- release records
- counting and analysis program
- maintenance and calibration records
- annual and periodic reports
- procedures

b. Observations and Findings

The inspector audited the gaseous releases for 1999 and 2000. The results were calculated using the Environmental Protection Agency COMPLY code. The inspector's review of these releases confirmed that they met the annual dose constraint specified by 10 CFR 20.1101(d), 10 CFR Part 20, Appendix B concentrations, and TS Section 3.7.2 discharge limits.

The inspector verified that radioactive liquid releases were below 10 CFR Part 20, Appendix B limits. Liquid release records since February 1999 were reviewed through May 2000, confirming that these releases met 10 CFR 20.2003 and 10 CFR Part 20, Appendix B limits.

c. Conclusions

Effluent monitoring satisfied license and regulatory requirements and releases were within the specified regulatory and TS limits.

8. Transportation of Radioactive Materials

a. Inspection Scope (IP 86740)

The inspector reviewed selected aspects of:

- radioactive materials shipping procedures
- radioactive materials transportation and transfer records for 1999-2000
- interviewed staff
- observed package preparations and transport of the reactor tank

b. Observations and Findings

The requirements of 10 CFR 30.41, "Transfer of Byproduct Material", obligates the shipper, prior to transferring byproduct material to another entity, to verify that the transferee is authorized under 10 CFR 30.41(b) (1)-(7) to receive byproduct material and that their license authorizes the receipt of the type, form, and quantity of byproduct material being transferred.

The inspector reviewed ten shipments in 1999 and 2000 and confirmed compliance with the requirements of 10 CFR 30.41(d) (1)-(5).

The WTR, in addition to NRC regulations, is required by 10 CFR Part 71, "Packaging and Transportation of Radioactive Material", to comply with the applicable requirements of the DOT regulations in 49 CFR Parts 170 through 189.

Shipping paper documentation required by 49 CFR must include the proper shipping name and hazard class, the words "Radioactive Material," the applicable identification number (UNXXXX), and the name, physical/chemical form/description, and activity in SI units of each nuclide. Additionally, the category of label applied to each package and the TI assigned to each Yellow-II or III package must be included. If tendered to a common carrier an appropriate signed shipper's certificate is required and if by aircraft additional statements as to acceptability are also needed.

The inspector confirmed by review of shipping records since February 1999, that the licensee properly prepared the shipping paper documentation. Emergency response information and monitored telephone contacts were as required.

The 49 CFR Part 173 requires that each shipper of a type 7A package maintain on file, a written document of the test and engineering evaluation or other data showing the package complies with the specification. Additionally, if the shipper makes any changes to the packaging, a supplemental evaluation must be performed and documented. The documentation must address that the change to the packaging is in conformance with the specifications.

Packages used at the WTR are normally purchased from a vendor or provided by the contractor involved with the material being shipped. The inspector confirmed that the manufacturers' testing and evaluation documentation along with their packaging instructions were on file.

c. Conclusions

Based on the records reviewed, the inspector found the transportation of byproduct material by the licensee satisfied the applicable NRC and DOT regulations.

9. Emergency Preparedness

a. Inspection Scope (IP 69001)

The inspector reviewed selected aspects of:

- the emergency plan
- implementing procedures
- emergency response facilities, supplies, equipment and instrumentation training records
- offsite support
- emergency drills and exercises

b. Observations and Findings

Although a NRC approved emergency plan was not required, the Waltz Mill site maintains it own fire brigade and separate medical clinic. Training and tours of the facility were provided yearly to the staff. Interviews with staff and review of training records indicate that they are knowledgeable of the hazards involved in responding to emergencies at the reactor. Emergency call lists at the guard station were available and accurate.

c. Conclusions

Emergency response staffing was sufficient for the radiological hazards involved in a fire or injury involving the reactor facility.

10. Security

a. Inspection Scope (IP 81401)

The inspector reviewed selected aspects of:

- the access control plans
- security systems, equipment and instrumentations
- implementation of the access control plan
- audits
- general site security

b. Observations and Findings

The Waltz Mill site is an industrial complex and as such access to the facility is controlled by use of fences and barriers, gates and secured access points, and identification and badging procedures. The inspector toured the Waltz Mill site and confirmed that the physical protection systems (barriers and alarms), equipment, and instrumentation were as required by procedures.

Access to the reactor was described in FDP Section 2.2.2.1 and performed in accordance with WMDT Section 1.6, "Access Control Plans." During the inspection, the inspector reviewed the access procedures, observed personnel entries into the reactor containment, and interviewed staff. Additionally, the inspector himself made numerous entries into containment through both the east air lock and the north truck lock. Through the reviews, observations, interviews, and entries into the reactor containment, the inspector confirmed that access was controlled as required.

c. Conclusions

Based on the observations, the inspector found the physical protection features of the Waltz Mill site and WTR, the equipment, procedures and access control, satisfied site procedures and the access control plan.

11. Training

a. Inspection Scope (IP 69001)

The inspector reviewed selected aspects of:

- radiation protection training records and rosters
- radiation protection training procedures
- FDP

b. Observations and Findings

FDP Section 2.6 outlines the training requirements for individuals who require access to the work or radiological areas of the site. The training consisted of two parts. The first part being general site training to provide orientation and meet the requirements of 10 CFR Part 19, and the second being radiation worker training (RWT) commensurate with the potential hazards to which the individual could be exposed based on their specific duties and responsibilities. The training records maintained by the licensee include the trainee's name, date of training, type of training, test results, authorizations for protective equipment use, and instructor's name.

The inspector's reviews of the training records since February 1999 confirmed that 10 CFR Part 19 and specific training appropriate to individual status and work requirements had been provided to staff and contractors. The inspector confirmed by interviewing and observing individuals performing maintenance and decommissioning activities, calibrations, and surveys, that the training had been effective. Additionally, the inspector verified the training of two contractors and one Westinghouse employee. All training records reviewed were current.

c. Conclusions

The 10 CFR Part 19 and FDP required RWT training were performed in accordance with established procedures.

12. Exit Meeting Summary

The inspector presented the inspection results to members of licensee management at the conclusion of each inspection period. (i.e., October 28, 1999, and January 20, April 21, and May 16, 2000). The licensee acknowledged the findings presented and did not identify any of the material provided to or reviewed by the inspector during the inspection as proprietary.

PARTIAL LIST OF PERSONS CONTACTED

Licensee

R. Banning	GTS Duratek, HP Manager
*B. Bowman	Viacom (CBS) Project Director
*R. Cline	Manager, Industrial Hygiene, Safety & Environmental Compliance
E. Hume	GTS Duratek, HP Technician
*W. Lavallee	Westinghouse, Project Manager
*A. J. Nardi	Westinghouse License Administrator
*D. Reese	Morrison-Knudsen, Operations Manager
D. Robin	Morrison-Knudsen, General Superintendent
*R. Sisk	Westinghouse, Licensing Engineer
J. Smith	GTS Duratek, HP Technician
S. Thompson	Technical Director, Antech Ltd.
*W. D. Vogel	Waltz Mill Radiation Safety Officer

(*Attended Exit Meeting)

INSPECTION PROCEDURE (IP) USED

IP39745	Class I Organization and Operations and Maintenance Activities
IP 40745	Class I Review -Audit & Design Change
IP 69001	Class II Non-Power Reactors
IP 81401	Plans, Procedures, and Reviews
IP 83743	Class I Radiation Protection
IP 86740	Transportation Activities

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

NONE

Closed

NONE

DISCUSSED

NONE

PARTIAL LIST OF ACRONYMS USED

ALARA	As Low As Reasonably Achievable
ANSI	American National Standards Institute
DOT	Department of Transportation
FDP	Final Decommissioning Plan
HP	Health Physics
LCO	Limiting Conditions for Operation
RSC	Radiation Safety Committee
WTR	Westinghouse Test Reactor
NRC	Nuclear Regulatory Commission
RPM	The Radiation Protection Manual
RPP	Radiation Protection Program
RWP	Radiation Work Permits
RWT	Radiation Worker Training
TS	Technical Specifications
WMDT	Waltz Mill Westinghouse Test Reactor Decommissioning Project Control Manual

Westinghouse Electric Company LLC

Marlene W. Jackson
Assistant General Counsel
Law Department

(412) 374-5570
(412) 374-6122 (Fax)
jacksonmw@westinghouse.com

Box 355
Pittsburgh Pennsylvania 15230-0355

Shipping Address
4350 Northern Pike
Monroeville Pennsylvania 15146-2886

December 20, 2000

William D. Wall, Esq.
Viacom Inc.
11 Stanwix Street
Pittsburgh, PA 15222

Subject: Transfer of Remaining Westinghouse Test Reactor (TR-2) Facilities to the Westinghouse Waltz Mill SNM-770 license

Dear Mr. Wall:

On July 5, 2000, Viacom Inc. wrote to Dr. Charles R. Pryor requesting that Westinghouse Electric Company LLC accept transfer of the remaining contamination associated with the Viacom TR-2 License at the Waltz Mill Facility into the Waltz Mill SNM-770 License. When Ramsey Coates corresponded with Mr. Michael Sweeney on this subject, he expressed several concerns on behalf of Westinghouse. First, Viacom has only removed those portions of the biological shield necessary to take away the reactor vessel and did not attempt to remove any portion of the lower shield. Second, given that the reactor containment building does not lend itself to reuse, we requested that Viacom confirm that it understood that the transferred facilities would be added to the retired facilities list and remediated in accordance with the free release criteria set forth in the SNM-770 Remediation Plan.

As you are aware, in accordance with Section 8.1 of the Asset Purchase Agreement dated June 25, 2000 ("APA"), Westinghouse agreed to accept certain portions of WTR facilities into the SNM-770 license, assuming that Viacom met several preconditions set forth in the TR-2 Decommissioning Plan and continued remediation of the residual contamination associated with the TR-2 Facilities under the SNM-770 Remediation Plan. As we stated previously, Westinghouse will live up to its obligation to accept the transfer of the TR-2 areas. Following our recent meetings, we are prepared to effectuate the retirement of the TR-2 license, based upon your assurance that Viacom will meet all of its obligations with respect to the TR-2 facilities once such facilities come within the purview of the Retired Facilities under the SNM-770 Remediation.

Please have the project personnel prepare the paperwork necessary to effectuate the transfer of the remaining TR-2 facilities into the SNM-770 license, together with the amendment to the SNM-770 Remediation Plan retired facilities list, and we will forward them to the NRC for approval.

I look forward to hearing from you on these issues. Please feel free to call me if you have any additional questions.

Very truly yours,

R. Cline, WM

LD812

Westinghouse Electric Company LLC

Box 355
Pittsburgh Pennsylvania 15230-0355

Marlene W. Jackson
Assistant General Counsel
Law Department

(412) 374-5570
(412) 374-6122 (Fax)
jacksonmw@westinghouse.com

Shipping Address
4350 Northern Pike
Monroeville Pennsylvania 15146-2886

December 20, 2000

William D. Wall, Esq.
Viacom Inc.
11 Stanwix Street
Pittsburgh, PA 15222

Subject: Transfer of Remaining Westinghouse Test Reactor (TR-2) Facilities to the Westinghouse Waltz Mill SNM-770 license

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Please have the project personnel prepare the paperwork necessary to effectuate the transfer of the remaining TR-2 facilities into the SNM-770 license, together with the amendment to the SNM-770 Remediation Plan retired facilities list, and we will forward them to the NRC for approval.

I look forward to hearing from you on these issues. Please feel free to call me if you have any additional questions.

Very truly yours,

R. Cline, WM

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SAN FRANCISCO

1875 CONNECTICUT AVENUE, N.W.
WASHINGTON, DC 20009-5728
(202) 986-8000
TELEX: 440274 FACSIMILE: (202) 986-8102

WRITER'S DIRECT DIAL:
(202)986-8050
E-MAIL ADDRESS: MFMCBRID@LLGM.COM

LONDON
(A LONDON BASED MULTINATIONAL PARTNERSHIP)
PARIS
BRUSSELS
MOSCOW
RIYADH
(AFFILIATED OFFICE)
TASHKENT
BISHKEK
ALMATY
BEIJING

September 19, 2002

Via Facsimile and First-Class Mail

F. Ramsey Coates, Esq.
Vice President and General Counsel
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

Re: Your Letter of September 17, 2002 and NRC Letter of September 6, 2002 Regarding Waltz Mill Site

Dear Mr. Coates:

Thank you for your letter of September 17, 2002. Preliminarily, let me say that you misunderstood a portion of my September 9, 2002 letter to you. I did not say I would not communicate further with you about all matters, but rather that I would not continue to debate the matters set forth in your letter of August 29, 2002 and to which I had responded on September 9. There is a new matter, which arises because of the NRC's September 6, 2002 letter and inspection report addressed to Mr. Richard Smith of Viacom, that was not addressed in my September 9, 2002 letter, and which is the subject of this letter. (Mr. Smith is separately communicating with Mr. Lavallee of Westinghouse regarding the process drain line matter.)

In its September 6, 2002 letter, the NRC stated that it contemplates the submission of data documenting the residual radioactivity remaining in situ, i.e., residual radioactivity in the test reactor structures covered by the TR-2 license. I am advised by Viacom that the data contemplated by NRC exists and is solely in the possession of Westinghouse, but that Westinghouse has not provided that data to Viacom.

In light of the NRC's recent request for that radiation data, I hereby formally request that Westinghouse provide that data to me by September 25, 2002, so that Viacom can provide it to the NRC. If I do not hear from you by September 25, 2002, we will assume that you do not intend to provide that data to Viacom, and we will proceed accordingly.

Very truly yours,

Michael F. McBride

Michael F. McBride

Attorney for Viacom, Inc.

Sutherland
Asbill &
Brennan LLP
ATTORNEYS AT LAW

1275 Pennsylvania Avenue, NW
Washington, DC 20004-2415
202 383 0100
fax 202.637.3593
www sablaw.com

RICHARD G. MURPHY, JR.
DIRECT DIAL: (202) 383-0635
Internet:rgmurphy@sablaw com

September 30, 2002

BY FACSIMILE (202) 986-8102
HARD COPY TO FOLLOW

Michael F. McBride, Esq.
LeBoeuf Lamb Greene & MacRae
1875 Connecticut Avenue, N.W.
Washington, D.C. 20009-5728

Re: Westinghouse Electric Company LLC
- Claims Against Viacom Inc.

Dear Mr. McBride:

Your letter of September 27, 2002 to Ramsey Coates, Vice-President and General Counsel of Westinghouse Electric Company LLC, has been referred to me for reply. In that letter you renew Viacom's request that Westinghouse deliver to Viacom data concerning residual radioactivity remaining *in-situ* in the former WTR facilities at the Waltz Mill site. Westinghouse declines to accede to your request, principally because Westinghouse is concerned that Viacom will use the information in support of a misleading application to the NRC for termination of the TR-2 license held by Viacom, notwithstanding the fact that Viacom has not completed the requirements of the TR-2 Decontamination and Decommissioning Plan (the "TR-2 Plan").

My client's belief that Viacom might misuse the information stems from misstatements and omissions in a letter written by Richard K. Smith of Viacom to the NRC on March 25, 2002. In that letter, Mr. Smith stated that the work required by the TR-2 Plan had been completed without informing the NRC that there was (and is) substantial disagreement with respect to whether the activities called for under the terms of the TR-2 Plan had been completed. Mr. Smith then went on to suggest that documents necessary to transfer the residual radioactivity to the SNM-770 license held by Westinghouse had not been completed because of a dispute concerning "various issues associated with the sale of the business," in an obvious attempt to conceal from the NRC the fact that the dispute is really about whether Viacom has done what is required under the terms of the TR-2 Plan. The dispute between our clients is not about "various issues associated with the sale of the business." The dispute is about whether Viacom has discharged its obligations under the TR-2 Plan and the companion SNM-770 Plan.

The data you request would be a necessary part of an application for transfer of the residual radioactivity and the WTR facility now covered by the TR-2 license to the SNM-770 license. Westinghouse will not agree to the transfer of the WTR facilities until Viacom makes a clear commitment concerning what additional remediation of those facilities would be conducted by Viacom under the SNM-770 Plan after the transfer. Viacom has consistently stated that no further remediation work would be done, and Westinghouse has made it clear that it will not accept transfer of the residual radioactivity until Viacom commits to complete the remediation in a manner consistent with the requirements of the TR-2 and SNM-770 Plans. Given Westinghouse's unwillingness to accept transfer of the residual radioactivity now covered by the TR-2 license, we fail to understand how Viacom intends to use the requested information. The NRC has not requested the information; it has merely noted that the residual radioactivity remaining *in-situ* at the WTR facility will have to be determined prior to termination of the TR-2 license. Westinghouse will provide the information to Viacom once there is a resolution of the dispute concerning Viacom's obligations under the Plan with respect to remediation of the residual radioactivity Viacom desires to transfer to the SNM-770 license.

As you note, there is a secondary reason for Westinghouse's refusal to deliver the information to Viacom. If Westinghouse were to deliver the information to Viacom, it would do so in its capacity as Project Manager pursuant to the "Agreement for Radiological Project Management Engineering and Field Services Provided by Westinghouse Electric Company LLC for Waltz Mill Remediation Project" (the "Agreement"). Westinghouse's obligations under the Agreement are dependant on Viacom's performance of its reciprocal promises under the Agreement. Viacom has breached the Agreement by refusing to pay Westinghouse more than $3,000,000, primarily to reimburse Westinghouse for expenses incurred and paid under the terms of the Agreement. Quite simply, Westinghouse has been relieved of any obligation to provide any information to Viacom under the terms of the Agreement by Viacom's unjustified refusal to pay Westinghouse what Westinghouse is owed.

Please direct any future correspondence with respect to the above matter to the undersigned.

Very truly yours,

Richard G. Murphy, Jr.

Richard G. Murphy, Jr.

RGMjr/epb
cc: F. Ramsey Coates, Esq.
Marlene W. Jackson, Esq.
Kevin Brode, Esq.
Mr. A. Joseph Nardi

LeBoeuf, Lamb, Greene & MacRae
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PITTSBURGH
SALT LAKE CITY
SAN FRANCISCO

1875 CONNECTICUT AVE., N.W.
SUITE 1200
WASHINGTON, D.C. 20009-5728
(202) 986-8000
FACSIMILE (202) 986-8102
E-MAIL ADDRESS MMALSCH@LLGM.COM
WRITER'S DIRECT DIAL: (202) 986-8059
WRITER'S DIRECT FAX: (202) 986-8102

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(A LONDON-BASED MULTINATIONAL PARTNERSHIP)
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BRUSSELS
JOHANNESBURG
(PTY) LTD
MOSCOW
RIYADH
(AFFILIATED OFFICE)
TASHKENT
BISHKEK
ALMATY
BEIJING

October 30, 2002

Dr. William D. Travers
Executive Director for Operations
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Re: <u>Petition Pursuant to 10 CFR 2.206</u>

Dear Dr. Travers:

Enclosed is a "Petition Pursuant to 10 CFR 2.206," on behalf of Viacom Inc., for the consideration and decision of the U.S. Nuclear Regulatory Commission.

Also enclosed is an additional copy of the Petition for date and return to us in the self-addressed, postage-prepaid envelope.

Respectfully submitted,

Martin G. Malsch
Michael F. McBride
John W. Lawrence

Attorneys for Viacom Inc.

Enclosures

cc: U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406-1415

Dr. Ronald R. Bellamy
Chief, Decommissioning and Laboratory Branch
Division of Nuclear Materials Safety
U.S. Nuclear Regulatory Commission
Region I
475 Allendale Road
King of Prussia, PA 19406-1415

Mr. Patrick M. Madden, Section Chief
Mr. Alexander Adams, Jr., Senior Project Manager
Mr. Stephen W. Holmes, Reactor Inspector
Research and Test Reactors Section
Operating Reactor Improvements Program
Office of Nuclear Reactor Regulation
Division of Regulatory Improvement Programs
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Mr. James G.Yusko
Pennsylvania Department of Environmental Protection
400 Waterfront Drive
Pittsburgh, PA 15222

Mr. A. Joseph Nardi, Licensing Administrator
Westinghouse Electric Company, LLC
P.O. Box 355
Pittsburgh, PA 15203-0355

Mr. Richard K. Smith
Vice President - Environmental Remediation
Robert A. Noethiger, Esq.
Vice President, Counsel
Viacom Inc.
11 Stanwix Street
Pittsburgh, PA 15222-1312

Exhibit A

Pursuant to 10 CFR 50.12, the Commission may, upon application by any interested person or upon its own initiative, grant exemptions from the requirement s of 10 CFR Part 50 when (1) the exemptions are authorized by law, will not present an undue risk to public health or safety, and are consistent with the common defense and security and (2) when special circumstances are present. Special circumstances are present, according to 10 CFR 50.12(a)(2)(ii), whenever "application of the regulation in the particular circumstances would not serve the underlying purpose of the rule or is not necessary to achieve the underlying purpose of the rule."

The underlying purpose of 10 CFR 50.82 (b) (6) is to describe the requirements that must usually be met for license termination, one of which is that the results of the terminal radiation survey and other documentation show that the facility and site meet the requirements for release in 10 CFR Part 20, subpart E. In this case, the reactor structures, components and fuel have all been removed, but residual materials will remain under license, and the site will not be released because of continued operations of Westinghouse under license No. SNM-770. Therefore, application of the rule is not necessary in order to terminate the Part 50 portion of the license because termination will still leave other licenses in effect on the site and termination will not result in site release under 10 CFR Part 20, subpart E. Moreover, since the residual materials will remain under NRC license, under the same conditions of use, the grant of an exemption from 10 CFR 50.82 (b)(6) will be in accord with public health and safety and the common defense and security, as well as the public interest, and there will be no environmental impact.

Exhibit B

A Description of the TR-2 Facilities And Residual Radioactivity to be Transferred To the SNM-770 License

Waltz Mill Facility
Westinghouse Test Reactor
License No. TR-2
Docket No. 50-22

Viacom Inc.

October 2002

The TR-2 facilities are described in Section 1.3 and Section 2 of the "Westinghouse Electric Corporation, Westinghouse Test Reactor, TR-2 Final Decommissioning Plan, Revision 1, January 2000" ("Plan"). In accordance with the Plan; the reactor vessel, the reactor internals, and a large portion of the biological shield have been removed from containment, packaged and shipped to a licensed low level waste processor or disposal facility. In conjunction with these activities some "optional decontamination and dismantlement activities" were conducted in the Test Reactor Containment Building in accordance with the Plan. Descriptions of the existing facilities are provided below. Attachment 1 provides a characterization of the facilities to be transferred to the SNM-770 License.

FACILITY AND SITE DESCRIPTION

The Waltz Mill site is located approximately 30 miles southeast of Pittsburgh in Westmoreland County, Pennsylvania (see Figure 1-1 of the Plan). The site is approximately 850 acres and is located about three miles west of the town of New Stanton between the towns of Madison and Yukon (see Figure 1-2 of the Plan). The WTR facility is located in the northwest portion of the Waltz Mill site, north of the G Building (see Figure 1-3 of the Plan).

Westinghouse Electric Company LLC operates the Waltz Mill Facility under the SNM-770 License. The Test Reactor facility is currently maintained under NRC License Number TR-2 (Possession Only). The Test Reactor at Waltz Mill was a low pressure, low temperature, water cooled 60 MWt reactor housed in a cylindrical vapor containment structure (see Figure 1-4 of the Plan). Since permanent shutdown in 1962, all fuel, the reactor internal contents, the reactor vessel and a large portion of the biological shield have been removed from the remaining reactor containment, or vapor shell, and from the Waltz Mill site. The remaining equipment and structures include the reactor containment building, the rabbit pump room, the sub-pile room, the polar crane, and a portion of the transfer canal. These have been extensively characterized (see Attachment 1) and are controlled to not pose a threat to the health and safety of the site worker or the general public.

DESCRIPTION OF OPTIONAL DECONTAMINATION AND DISMANTLEMENT ACTIVITIES CONDUCTED WITHIN THE TR-2 FACILITIES

In addition to removal of the reactor vessel internal contents, the reactor vessel, and portions of the biological shield, decontamination and dismantlement activities have been performed in other areas within the TR-2 facilities. These activities were not required for TR-2 decommissioning; however, they were performed in accordance with the NRC approved Decommissioning Plan prior to transferring the remaining residual radioactivity to the SNM-770 License.

The specific areas where the optional decontamination and dismantling activities were conducted are described below. Any residual radioactivity remaining in equipment and structures under the TR-2 license will, after termination of the Part 50 portion of the TR-2 license, remain under the Part 30 portion of that license.

Sub-pile Room

General

The sub-pile room is a 15' x 15' room located below the reactor vessel. This room has a ¼-inch steel liner on all four walls covering the concrete biological shield. The floor is uncoated concrete. The WTR canal runs through the sub-pile room (north-south), separating the room into two areas (east and west). The two doors to the sub-pile room consist of a steel liner filled with 12 inches of poured lead. One permits accesses to the east side of the canal and the other to the west side. The WTR fuel chute is accessible in the northeast corner of the room through a shielded opening in the fuel chute pipe chase. The sub-pile room contains primary system piping, rabbit tubes, test loop piping and instrumentation piping.

Remedial action taken

All remaining piping and the metal walkway were dismantled and/or cutout in disposable sized sections and removed for disposal. Asbestos floor tiling, lead shot in the Reactor lining and lead brick from around the Fuel Chute and walls were removed and disposed. A section of the Fuel Chute and the Fuel Chute liner were removed and disposed. A general cleanup was performed and, following removal of the Reactor, the keyway area was cleaned.

Rabbit Pump Room

General

The Rabbit Pump Room measures approximately 6'6" by 10'0" by 7'6" high and is located on the operating floor along the north wall of the containment building. The Rabbit Pump Room contains pumps and valves that delivered the rabbits (test material samples) in a container, to the reactor core via the rabbit tubes.

Remedial action taken

The remaining piping, valves, pumps and control assemblies were dismantled and/or cutout in disposable sized sections and removed. Aggressive decontamination was conducted on concrete surfaces of the floor and portions of the walls. All penetrations were surveyed, mechanically plugged and labeled.

Test Loop Cubicles

General

Three test loop cubicles are located along the west side of the reactor vessel adjacent to the reactor biological shield. Each cubicle is constructed of concrete of varying dimensions and all cubicles are currently vacant.

Remedial action taken

The remaining loose debris was removed and a general cleaning conducted. All penetrations were surveyed, mechanically plugged and labeled.

Test Loop Dump Tank Pits

General

Two 8'0" by 9'0" by 13'0" high Test Loop Dump Tank Pits are located below the operating floor on the east and west side of the transfer canal below the reactor vessel. The west tank pit contained three steel tanks approximately 12' tall and 4' in diameter. The east pit was flooded with water and the pit interior was inaccessible.

Remedial action taken

The East Test Loop Cubicle was dewatered and the remaining equipment and loose debris was removed. The remaining equipment, piping, tanks and loose debris in the West Loop Test Cubicle were also removed. Aggressive decontamination of the concrete surfaces was conducted.

Decontamination and/or Removal of Miscellaneous Piping, Ducting and Utilities

General

Prior to removing electrical, service water, service air, fire or HVAC systems, each system was inspected, their existing operational status verified. Existing electrical systems were disconnected and other inactive systems were identified. Piping systems were removed in areas where electrical systems have been disconnected/removed. The original supply and exhaust air ducts consist of 4' by 4' ducting which encircles the in inside of containment and runs to the Facility Operations Building. Emergency utilities, such as fire alarm systems, were either maintained or replaced, as required

Remedial action taken

Inactive ventilation, piping and electrical systems have been removed from the WTR facilities. A new HEPA ventilation system was installed in the containment building along with new electrical and fire water systems. The remaining systems were decontaminated.

Primary Coolant Pipe Tunnels

General

The primary coolant pipe tunnels surround the north end of the transfer canal along the east and west sides of the reactor vessel below the operating floor. Each tunnel measures approximately 5'0" wide by 10'0" high by 39'0" long and merge into a common tunnel at the north side of the containment building. The tunnel continues below grade to the northeast to the Facilities Operations Building. The pipe tunnels contain the primary coolant circulation supply and return lines, demineralizer, emergency coolant and various other piping systems.

Remedial action taken

The piping components were dismantled and/or cut into sections for disposal. A general cleaning was done with the removal of dust and loose debris. Aggressive decontamination was performed on the walls and floors. All penetrations were surveyed, mechanically plugged and labeled.

Transfer Canal

General

The transfer canal is approximately 19 feet deep, varies in width from 7 feet to 10 feet and is approximately 160 feet long north-south down the axis of the reactor. The canal begins north of the biological shield and continues beneath the reactor vessel to the south, through the G building Annex, ending beneath the Hot Cell area. The transfer canal was the means of transporting spent fuel rods from the reactor vessel to the Annex Building and irradiated test specimens to the Hot Cell area (see Figure 2-14 of the Plan). The fuel rod conveyor, storage racks, thimble loading machine, transfer chute, rabbit tubes, piping and pipe supports were left in the canal following the 1962 shut down. All irradiated material was removed and properly dispositioned.

Remedial action taken

The water was drained from the canal, filtered and processed for disposal. Freestanding equipment was removed and the walls were hosed down with clean water during the drain down. Sediment was removed from the canal and processed for disposal. Remaining equipment was removed, such as fuel racks bolted on the walls, one cask stand, metal tracks used for the conveyor, miscellaneous piping and some piping that was embedded on the south end of the canal. The interior concrete surfaces of the canal were scrubbed and pressure washed and then aggressively scabbled. A cinderblock barrier separating the WTR portion from the Annex portion was removed and a new barrier was constructed following the work.

Containment Building

General

The WTR was a low pressure, low temperature, water cooled 60 MWt reactor housed in a cylindrical vapor containment building. There are two airlocks, and a large overhead door that provides access from the truck lock to the WTR. A schematic of the WTR is shown on Figure 2-1 of the Plan.

The operating floor is on the 16-foot elevation and is constructed of concrete. The containment is 90 feet in diameter, with a total floor area of 5000 square feet. There are four support platforms: the truck lock, the reactor headstand, reactor head, and the beam port platforms. As part of the materials testing that was included in the WTR's operational charter, there were several controlled environment test loops installed in concrete cubicles and in an underground test loop vault. Since the shut down these loops have been removed.

The containment building also houses the rabbit pump room, polar crane, and other support systems such as: piping, electrical conduit and boxes, plant and instrument air lines, hydraulic lines, steam and condensate lines, and ventilation ductwork.

Remedial action taken

The reactor core support structure was 29 feet in diameter and 36 feet tall, and housed the reactor pressure vessel. The reactor vessel, the reactor internals and a large portion of the reactor core support structure have been removed. The biological shield surrounding the reactor vessel which was made of magnetite bearing concrete, and stood approximately 44 feet in height and was up to eight feet thick from the 32 to 51 foot elevations has also been removed. The Polar Crane and the interior of the dome were cleaned. The platforms at the 32 foot elevation were decontaminated as was the remaining biological shield. Asbestos containing tiles were removed from the floor and aggressive decontamination was performed on the concrete. All penetrations in the floor and the biological shield were surveyed, mechanically plugged and labeled.

Attachment 1
Current Status of TR-2 Facility
A summary of the decommissioning/remediation work
conducted on the WTR facility since 1994

The following systems were characterized as part of the Westinghouse Waltz Mill Characterization Study completed in 1994 by Scientific Ecology Group, Inc. As a result of the decommissioning work conducted since 1994 the following changes to the Characterization Study should be noted:

Reactor Vessel Body
The reactor vessel body has been removed and no residual radioactive activity remains.

Reactor Head Access Plug and Flange
The reactor head access plug and flange has been removed and no residual radioactive activity remains.

Upper Reactor Vessel Internals
The upper reactor vessel internals have been removed and no residual radioactive activity remains.

Lower Reactor Vessel Flange
The lower reactor vessel flange has been removed and no residual radioactive activity remains.

Reactor Internal Thermal Shields and Core Lattice
The reactor internal thermal shields and core lattice have been removed and no residual radioactive activity remains.

WTR Biological Shield
The portion of the WTR biological shield surrounding the reactor has been removed. The remaining concrete has been decontaminated and all penetrations were surveyed, mechanically plugged and labeled. Residual activity levels have been surveyed and documented.

Transfer Canal (WTR)
The water was drained from the canal, filtered and processed for disposal. Freestanding equipment was removed and the walls were hosed down with clean water during the drain down. Sediment was removed from the canal and processed for disposal. Remaining equipment was removed, such as fuel racks bolted on the walls, one cask stand, metal tracks used for the conveyor, miscellaneous piping and some piping that was embedded on the south end of the canal. The interior concrete surfaces of the canal were scrubbed and pressure washed and then aggressively scabbled. A cinderblock barrier separating the WTR portion from the Annex portion was removed and a new barrier was constructed following the work. Residual activity levels have been surveyed and documented.

Reactor Primary Coolant Piping and Other Snake Pit Systems
The primary coolant piping and other inactive systems in the Primary Coolant Tunnel have been removed and no residual radioactive activity remains.

Sub-Pile Room Thimble Valve Banks
The inactive systems in the Sub-Pile room have been removed and no residual radioactive activity remains.

Rabbit Pump and Piping
The Rabbit Pump and piping have been removed and no residual radioactive activity remains.

Gas Test Loop Tanks and Piping
The Gas Test Loop tanks and piping have been removed and no residual radioactive remains.

Chemistry Test Loop Piping
The Chemistry Test Loop piping have been removed and no residual radioactive activity remains.

Test Loop Dump Tanks and Piping
The Test Loop Dump Tanks and Piping have been removed and no residual radioactive activity remains.

Test Loop Primary Piping (East Side)
The Test Loop Primary Piping has been removed and no residual radioactive activity remains.

Heating and Ventilation Ductwork
The major portion of the original Heating and Ventilation Ductwork has been removed. The remaining ductwork (the portion of the risers that goes around the Polar Crane) and the newly installed HEPA ventilation system were decontaminated and residual activity levels have been surveyed and documented.

Experimental Cooling Water System Piping
The Experimental Cooling Water System Piping has been removed and no residual radioactive activity remains.

Low-Level Radioactive Liquid Drain
The Low-Level Radioactive Liquid Drain has been removed and no residual radioactive activity remains.

Process Vent
The Process Vent piping has been removed and no residual radioactive activity remains.

Electrical Conduit and Boxes
The inactive electrical conduit and boxes have been removed from the containment building. The newly installed electrical systems have been decontaminated and residual activity levels have been surveyed and documented.

Plant and Instrument Air Lines
The Plant and Instrument Air Lines have been removed and no residual radioactive activity remains.

Deionized Water Line
The Deionized Water Lines have been removed and no residual radioactive activity remains.

Hydraulic Lines
The Hydraulic Lines were removed during Transfer Canal remediation and no residual radioactive activity remains.

CRDM Gland Seal Piping
The CRDM Gland Seal Piping has been removed and no residual radioactive activity remains.

Steam and Condensate Lines
The Steam and Condensate Lines have been removed and no residual radioactive activity remains.

Polar Crane
The Polar Crane was decontaminated and residual activity levels have been surveyed and documented.

The following structures were characterized as part of the WEC Waltz Mill Characterization Study completed in 1994 by Scientific Ecology Group, Inc. As a result of the decommissioning work conducted since 1994 the following changes to the Characterization Study should be noted:

Operating Floor
The Operating Floor has had the asbestos containing tile removed and a general cleaning. The floor was decontaminated and residual activity levels have been surveyed and documented.

Sub-Pile Room
All remaining piping and the metal walkway were dismantled and/or cutout in disposable sized sections and removed for disposal. Asbestos floor tiling, lead shot in the Reactor lining and lead brick from around the Fuel Chute and walls were removed and disposed. A section of the Fuel Chute and the Fuel Chute liner were removed and disposed. A general cleanup was performed and, following removal of the Reactor, the keyway area was cleaned. Residual activity levels have been surveyed and documented.

Rabbit Pump Room
The remaining piping, valves, pumps and control assemblies were dismantled and/or cutout in disposable sized sections and removed. Aggressive decontamination was conducted on concrete surfaces of the floor and portions of the walls. All penetrations were surveyed, mechanically plugged and labeled. Residual activity levels have been surveyed and documented.

Test Loop Shield Cubicles
The remaining loose debris was removed and a general cleaning conducted. All penetrations were surveyed, mechanically plugged and labeled. Residual activity levels have been surveyed and documented.

Test Loop Dump Tank Pits
The East Test Loop Cubicle was dewatered and the remaining equipment and loose debris was removed. The remaining equipment, piping, tanks and loose debris in the West Loop Test Cubicle were also removed. Aggressive decontamination of the concrete surfaces was conducted. Residual activity levels have been surveyed and documented.

Primary Coolant Pipe Tunnel (Snake Pit)
The piping components were dismantled and/or cut into sections for disposal. A general cleaning was done with the removal of dust and loose debris. Aggressive decontamination was performed on the walls and floors. All penetrations were surveyed, mechanically plugged and labeled. Residual activity levels have been surveyed and documented.

Truck Lock Platform
The Truck Lock Platform was decontaminated and residual activity levels have been surveyed and documented.

Beam Port Platform
The Beam Port Platform was decontaminated and residual activity levels have been surveyed and documented.

Reactor Head Stand Platform
The Reactor Headstand Platform has been removed and no residual radioactive activity remains.

Reactor Head Platform
The Reactor Head Platform has been removed and no residual radioactive activity remains.