**Portland General Electric Company**
*Trojan Nuclear Plant*
*71760 Columbia River Hwy*
*Rainier, OR 97048*
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March 6, 2001

VPN-004-2001

Trojan Nuclear Plant
Docket 50-344
License NPF-1

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

PGE-1061, "Trojan Nuclear Plant Decommissioning Plan
and License Termination Plan (PGE-1078)," Revision 9

The enclosure to this letter provides Revision 9 to Portland General Electric Company's "Trojan Nuclear Plant Decommissioning Plan and License Termination Plan." The primary purpose of this revision to PGE-1061 is to combine the Decommissioning Plan and the recently approved License Termination Plan into a single document. A summary of the changes included in Revision 9 is provided in the attachment to this letter. Revised portions of the Decommissioning Plan and License Termination Plan are denoted by sidebars.

Any questions concerning this revision can be directed to Mr. L. G. Dusek, of my staff, at (503) 556-7409.

Sincerely,

Stephen M Quennoz

Stephen M. Quennoz
Vice President,
Power Supply/Generation

Attachment

Enclosure

c: Director, NRC, Region IV, DNMS
D. J. Wrona, NRC, NRR
D. Stewart-Smith, OOE
A. Bless, OOE

A001

## SUMMARY OF CHANGES

The following provides a summary of the changes incorporated into the Trojan Nuclear Plant (TNP) Decommissioning Plan and License Termination Plan, Revision 9. These changes were evaluated and determined to not require prior NRC approval pursuant to 10 CFR 50.59. Changes are identified below by Licensing Document Change Request (LDCR) number.

LDCR 2001-002 PGE-1061, formerly the Decommissioning Plan (which serves as the Post-Shutdown Decommissioning Activities Report (PSDAR)) is revised to incorporate the approved License Termination Plan (LTP), PGE-1078, due to a significant amount of overlap in content between the two documents. The LTP received NRC approval on February 12, 2001. PGE-1061, Revision 9, incorporates both the PSDAR and the LTP required by 10 CFR 50.82. NRC regulatory content requirements for the PSDAR and the LTP continue to be satisfied, and NRC regulatory guidance for format and content of the LTP continues to be satisfied.

Administrative changes are made to former Section 2.3.1 and Figure 2-10 to indicate that the General Manager, Engineering/Decommissioning, is responsible for cost control.

Section 1.4 is revised to clarify that additional change criteria, consistent with the revised 10 CFR 50.59 rule, apply to both the Decommissioning Plan and the LTP. Sections 2.2.5.31 and 9.1 are revised to clarify that Fire Protection Program changes are subject to the change control stipulations in the TNP facility operating license.

Sections 1.2.3, 2.2.2, and Figure 2-11 are revised to reflect the impact of delays in Independent Spent Fuel Storage Installation (ISFSI) design and fabrication activities on the decommissioning schedule.

Sections 2.2.5.14, 2.2.5.3, 2.2.5.4, 2.2.5.33, and various sections throughout PGE-1061 are revised to reflect removal of the reactor vessel with internals intact in 1999, Containment Building internal concrete demolition, and fuel transfer tube removal. Table 2.2-4 is deleted. The changes to Section 2.2 are not required since this section is not required to be updated; however, these specific progress items are incorporated due to their magnitude and scope. These changes reflect approved changes that were implemented in the facility as part of decommissioning activities in accordance with approved procedures. These changes to the Decommissioning Plan are administrative in nature.

Figures 3-1 and 3-2 are no longer applicable and are deleted.

Numerous editorial changes are made throughout PGE-1061 to enhance readability and clarity, to correct typographical, grammatical, and other errors, and to eliminate redundancy.

# TROJAN NUCLEAR PLANT

PGE-1061, Trojan Nuclear Plant Decommissioning Plan and License Termination Plan (PGE-1078)

Revision 9

The following information is provided as a guide for revising PGE-1061, "Trojan Nuclear Plant Decommissioning Plan and License Termination Plan (PGE-1078)." With Revision 9, PGE-1061 becomes a 2-volume document, with each volume in its own binder. You received Volume 1 of the document with this letter. Volume 2 will be formed from your existing copy of PGE-1061 by replacing the pages as follows:

**PGE-1061, Volume 2 of 2 (i.e., your existing controlled copy of PGE-1061)**

| Remove Page | Insert Page |
|---|---|
| Binder Cover Insert and Spine | New Binder Cover Insert and Spine |
| Title Page | ----- |
| Table of Contents and Front Matter | ----- |
| Sections 1 through 9 | ----- |
| List of Abbreviations and Acronyms | ----- |
| Index | ----- |
| Title Page for Addendum | New Title Page for Addendum |
| Title Page for Appendix A | New Title Page for Appendix A |
| Title Page for Appendix B | New Title Page for Appendix B |

NOTE: Leave the Addendum documents, Appendix A, and Appendix B in Volume 2 of 2.

**PGE-1078, Trojan Nuclear Plant License Termination Plan**

Your existing controlled copy of PGE-1078 is no longer an active document and should be destroyed.

LGD-009-2001

TO: Distribution

FROM: L. G. Dusek

DATE: March 6, 2001

SUBJECT: Transmittal of Revision 9 to PGE-1061, "Trojan Nuclear Plant Decommissioning Plan and License Termination Plan (PGE-1078)"

Enclosed are replacement pages for your Controlled Copy of PGE-1061, "Trojan Nuclear Plant Decommissioning Plan and License Termination Plan (PGE-1078)." The pages are to be arranged in accordance with the accompanying instruction sheet.

Please acknowledge receipt of your copy by completing the lower portion of this transmittal and returning it to the location given below.

LGD/CKC

Enclosure

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3/6/2001

D. Stewart-Smith

ACKNOWLEDGMENT

PGE-1061, "Trojan Nuclear Plant Decommissioning Plan and License Termination Plan (PGE-1078)"

Revision 9

I hereby acknowledge receipt of Controlled Copy number 20 the subject document. All changes have been made in accordance with the instructions, and superseded pages have been destroyed.

Signature of Copy Holder Date

Return to: Pat Schaffran/TCB-3
Trojan Nuclear Plant
71760 Columbia River Highway
Rainier, Oregon 97048

PGE-1061

# Trojan Nuclear Plant Decommissioning Plan and License Termination Plan (PGE-1078)

VOLUME 1 OF 2

PGE-1061

# Trojan Nuclear Plant

# Decommissioning Plan
# and
# License Termination Plan (PGE-1078)

Portland General Electric

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# 1. SUMMARY OF PLAN

## 1.1 PURPOSE

This Decommissioning Plan incorporates both the Trojan Nuclear Plant (TNP) Post-Shutdown Decommissioning Activities Report (PSDAR) and the TNP License Termination Plan required by 10 CFR 50.82 (Reference 1-1). This plan also represents the TNP Decommissioning Plan required by the Oregon Office of Energy (OOE) in Oregon Administrative Rule (OAR) 345-026-0370 (Reference 1-2). Finally, this plan contains the Spent Fuel Management Plan for the TNP in accordance with 10 CFR 50.54(bb) (Reference 1-3). This Decommissioning Plan is maintained as a supplement to the TNP Defueled Safety Analysis Report (DSAR) (Reference 1-4) in accordance with 10 CFR 50.82(a)(9)(i).

Prepared using the guidance provided in Draft Regulatory Guide DG-1005, "Standard Format and Content for Decommissioning Plans for Nuclear Reactors" (Reference 1-5), the Decommissioning Plan discusses TNP decommissioning methodology and organization, estimated costs and available funds, major tasks and schedules, protection of occupational and public health and safety including site characterization, radiation protection, waste management, and analyses of hypothetical decommissioning events. It also addresses a number of additional areas and programs such as quality assurance, fire protection, and physical security provisions. Reflecting the choice of "DECON" as the decommissioning alternative at TNP, the objective of the Decommissioning Plan is to demonstrate TNP can be decommissioned in a safe manner and to describe plans for demonstrating the facility and site meet the criteria for release for unrestricted use.

The portions of this Decommissioning Plan representing the TNP License Termination Plan are consistent in format and content with the guidance provided in Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors" (Reference 1-6); NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans" (Reference 1-7); and NUREG-1727, "NMSS Decommissioning Standard Review Plan" (Reference 1-8). The TNP License Termination Plan demonstrates that the remainder of the decommissioning activities at the TNP site will be performed in accordance with the regulations in 10 CFR 50.82; will not be inimical to the common defense and security nor to the health and safety of the public; and will not have a significant effect on the quality of the environment.

## 1.2 HISTORICAL BACKGROUND

TNP is located in Columbia County, Oregon, approximately 42 miles north of Portland, Oregon. TNP is jointly owned by Portland General Electric (PGE), 67.5 percent; the City of Eugene, 30 percent through the Eugene Water and Electric Board (EWEB); and Pacific Power and Light/PacifiCorp (PP&L), 2.5 percent. PGE is the majority owner and has responsibility for operating and maintaining TNP. The Bonneville Power Administration (BPA), a power marketing agency under the United States Department of Energy (DOE), is obligated through Net Billing Agreements to pay costs associated with EWEB's share of TNP, including decommissioning and spent fuel management costs.

TNP, Docket 50-344, achieved initial criticality in December 1975 and began commercial operation in May 1976. The reactor output was rated at 3411 MWt with an approximate net electrical output rating of 1130 MWe. The nuclear steam supply system was a four-loop pressurized water reactor designed by Westinghouse Electric Corporation.

TNP was shut down for the last time on November 9, 1992. On January 27, 1993, after approximately 17 years of operation, PGE notified the Nuclear Regulatory Commission (NRC) of its decision to permanently cease power operations. The owners' decision was predicated on both financial and reliability considerations. The NRC amended the TNP Facility Operating License (NPF-1) to a Possession Only License on May 5, 1993. On July 31, 1993, PGE submitted a request to revise the TNP Technical Specifications to reflect the permanently defueled status of the plant (Reference 1-9). That request was supplemented by PGE on March 8, 1994 (Reference 1-10). On March 31, 1995, the NRC issued Amendment #194 to Facility Operating (Possession Only) License NPF-1 (Reference 1-11). This amendment revised the TNP Technical Specifications to reflect the permanently defueled condition of the plant, and regulatory requirements and operating restrictions to ensure the safe storage of spent fuel in the spent fuel pool. On October 7, 1993, PGE transmitted an updated Safety Analysis Report for the Defueled Condition (DSAR).

PGE submitted a proposed TNP Decommissioning Plan and Supplement to the Environmental Report (Reference 1-12) on January 26, 1995, which were approved by the NRC on April 15, 1996 (Reference 1-13). The TNP Decommissioning Plan was submitted and approved in accordance with the NRC's rule governing decommissioning and termination of license, 10 CFR 50.82. This rule was revised effective August 28, 1996. The revised rule resulted in the TNP Decommissioning Plan being considered the (newly required) PSDAR, and also added a new requirement for all power reactor licensees to develop and submit a license termination plan for NRC approval at least two years before termination of the license date. The NRC approved the TNP License Termination Plan, PGE-1078, on February 12, 2001 (Reference 1-14). Because of the significant overlap in content between the two documents, the approved TNP License Termination Plan has been incorporated into this Decommissioning Plan.

## 1.3 SUMMARY OF MAJOR ACTIVITIES AND SCHEDULE

TNP decommissioning is divided into two broad periods: a Transition Period and a Decontamination and Dismantlement Period . Decommissioning will be followed by site restoration. This section provides a brief description of these activities. Details are provided in Sections 2.2 and 2.3.

### 1.3.1 DESCRIPTION OF MAJOR ACTIVITIES

The Transition Period began with permanent plant shutdown in January 1993 and will continue until spent fuel is transferred to an ISFSI. The Decontamination and Dismantlement Period will begin once the spent fuel is transferred to the ISFSI. Site restoration will begin following termination of the 10 CFR 50 license and involves the final disposition of structures, systems, and components.

Storing fuel at TNP during and after plant decommissioning significantly impacts both the process and costs associated with decommissioning. The TNP contract with DOE, "Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste," provides the basis for the schedule forecast in DOE's annual acceptance priority ranking for receipt of spent fuel and/or high level waste. The published schedule specifies the first TNP shipment to be in 2002, and the final shipment is projected for 2018. Recognizing the uncertainty, but with no better formal estimate, the contract dates for fuel shipment are currently being used for planning purposes. The spent fuel management plan is discussed in Section 3.3.

#### 1.3.1.1 Transition Period

The Transition Period of TNP decommissioning is nearing completion. PGE continues to maintain systems and components required to support decommissioning and spent fuel storage in accordance with the Facility Operating (Possession Only) License NPF-1 and administrative procedures. The facility currently is maintaining its spent fuel in the spent fuel pool (SFP) and undergoing active decontamination and dismantlement and activities in accordance with the approved TNP DSAR and this Decommissioning Plan and License Termination Plan. Final survey of areas for which remediation has been completed may also begin towards the end of the Transition Period.

A concurrent effort is underway to complete the construction of an Independent Spent Fuel Storage Installation (ISFSI) at the Trojan site to facilitate decommissioning of the TNP. Fuel transfer to the ISFSI is planned to begin in 2002. The completion of fuel transfer to the ISFSI will allow the removal or decontamination in place of systems and components that support the SFP or wet fuel storage, including the SFP itself.

#### 1.3.1.2 Decontamination and Dismantlement

Once the spent fuel is transferred to the ISFSI, the Transition Period ends and the Decontamination and Dismantlement Period begins. Major activities planned during the Decontamination and Dismantlement Period include removing the remaining contaminated systems and components, and continuing the decontamination of structures and final radiation surveys. The final survey, described in Section 4, is performed to demonstrate that radiological conditions at TNP satisfy the final site release criteria of 10 CFR 20.1402 (Reference 1-15) to support unrestricted release of the TNP site and license termination. Upon completion of the final survey, a final survey report will be submitted to the NRC.

#### 1.3.1.3 Site Restoration

Nonradiological site remediation activities are scheduled to be completed following termination of the Facility Operating (Possession Only) License NPF-1. The primary nonradiological site remediation effort is scheduled to begin around 2018 and conclude in 2019. Some site restoration activities have been completed, and some may continue to be conducted during the Transition and Decontamination and Dismantlement Periods of decommissioning.

A listing and schedule of remaining major license termination activities is provided in Sections 2.2 and 2.3. According to this schedule, PGE anticipates the completion of decommissioning activities, including the final survey and license termination, by early 2005.

### 1.3.2 FINAL RELEASE CRITERIA

TNP decommissioning will safely reduce radioactivity at the site to levels meeting the unrestricted release criteria of 10 CFR 20.1402. The TNP final survey plan provided in Section 4 describes the scope and methodology of the final survey process, quality assurance measures, access control procedures, and how implementation of the plan will demonstrate that the plant and site will meet the 10 CFR 20.1402 criteria for unrestricted release of the site.

### 1.3.3 SCHEDULE FOR DECOMMISSIONING/SITE RESTORATION ACTIVITIES

A detailed schedule for decommissioning/site restoration activities is discussed in Section 2.2. The following is an overview of the current TNP decommissioning/site restoration project schedule.

| | |
|---|---|
| January 1993 - Mid 2003 | Transition Period |
| Late 1994 - Late 1995 | Large Component Removal Project |
| Late 1996 - Mid 2002 | Complete planning/building an ISFSI |
| Early 1997 - Late 1999 | Reactor Vessel and Internals Removal |
| Late 2002 - Mid 2003 | Transfer spent nuclear fuel to the ISFSI |
| Mid 2003 - Early 2005 | Decontamination and Dismantlement Period |
| Late 2004 | Complete final radiation survey |
| Late 2004 | Submit application for license termination |
| Early 2005 | License Termination |
| Early 2005 - Mid 2018 | Caretaking |
| Mid 2018 - Late 2019 | Demolish buildings |

## 1.4 EVALUATION OF CHANGES, TESTS, AND EXPERIMENTS

This Decommissioning Plan and License Termination Plan is an extension of the DSAR. PGE may make changes to this plan without prior NRC approval provided the proposed changes do not:

1. Meet any of the change criteria incorporated into the TNP Facility Operating (Possession Only) License NPF-1, License Condition 2.C (11); or

2. Cause a significant increase in the consequences of a decommissioning event as described in Section 3.4 of this plan, or create the potential for a new or different kind of decommissioning event from those previously analyzed;

In taking actions permitted under 10 CFR 50.59, PGE is required to notify the NRC and the State of Oregon, in writing, before performing any decommissioning activity inconsistent with, or making any significant schedule change from, those actions and schedules described in this Decommissioning Plan, including changes that significantly increase the decommissioning cost. [10 CFR 50.82(a)(7)]

## 1.5 PLAN SUMMARY

This Decommissioning Plan describes the process by which decommissioning will be completed and the TNP site released for unrestricted use. The following subsections provide a brief summary of the sections presented in this Decommissioning Plan.

### 1.5.1 SUMMARY OF SECTION 1 – GENERAL INFORMATION

This section provides the purpose of and regulatory basis for PGE-1061, TNP Decommissioning Plan and License Termination Plan (PGE-1078), as well as a brief overview of each section contained in the plan. A brief historical background and a summary description and schedule of major activities also are provided.

### 1.5.2 SUMMARY OF SECTION 2 – CHOICE OF DECOMMISSIONING ALTERNATIVE AND DECOMMISSIONING ACTIVITIES, TASKS, AND SCHEDULES

This section provides the basis for selecting the DECON alternative for decommissioning the TNP. In accordance with 10 CFR 50.82(a)(9)(ii)(B), this section also identifies the major dismantlement and decontamination activities that remain at TNP as of early-1999. This information details those areas and equipment that need further remediation to allow an estimation of the radiological conditions that may be encountered during remediation. Included herein are schedules for implementation of decommissioning and dismantlement activities, estimates of associated occupational radiation dose, and projected volumes of radioactive waste. The TNP decommissioning organization is also described in this section.

Finally, in accordance with 10 CFR 50.82(a)(9)(ii)(C), this section describes how remediation actions may be applied to various areas on the TNP site, identifies the remediation methodology to be used, and demonstrates that the remediation methodology is adequate to ensure that the site release criteria of 10 CFR 20.1402 are met. Verification of the site release criteria is detailed further in Section 4, Final Survey Plan.

### 1.5.3 SUMMARY OF SECTION 3 – PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

In accordance with 10 CFR 50.82(a)(9)(ii)(A), Section 3.1 provides a description of the radiological conditions at the TNP site. The TNP site characterization incorporates the results of scoping and characterization surveys conducted to quantify the extent and nature of contamination at TNP. The results of the scoping and characterization surveys have been and continue to be used to identify areas of the site that will require remediation, as well as to plan remediation methodologies and costs.

Section 3.2 provides a description of the TNP Radiation Protection Program, and Section 3.3 describes radioactive waste management at TNP, including the TNP Spent Fuel Management Program in accordance with 10 CFR 50.54(bb). Section 3.4 presents the results of evaluations and analyses of postulated decommissioning events, and evaluates the potential for adverse effects on public health and safety. This evaluation includes postulated events that could be significantly different from accidents that have previously been evaluated for plant operations or maintenance. The analyses consider events related to decommissioning activities, loss of support systems, internal events, and external phenomena. The results of the analyses indicate that decommissioning activities can be conducted in a manner that does not significantly affect public health and safety. Finally, Sections 3.5 and 3.6 discuss industrial safety and nonradioactive waste management.

### 1.5.4 SUMMARY OF SECTION 4 – FINAL SURVEY PLAN

In accordance with 10 CFR 50.82(a)(9)(ii)(D), the TNP Final Survey Plan describes the methods and criteria that will be used to demonstrate that the TNP site meets the radiological release criteria for unrestricted use specified in 10 CFR 20.1402. This plan includes a description of control measures implemented in accordance with approved plant procedures to preclude the recontamination of clean areas. The TNP Final Survey Plan also incorporates measures to ensure that final survey activities are planned and discussed with the NRC and the OOE sufficiently in advance to allow the scheduling of inspection activities.

### 1.5.5 SUMMARY OF SECTION 5 – UPDATED SITE-SPECIFIC ESTIMATE OF REMAINING DECOMMISSIONING COSTS

In accordance with 10 CFR 50.82(a)(9)(ii)(F), this section provides an updated site-specific estimate of remaining decommissioning costs, a comparison of these estimated costs with the present funds set aside for decommissioning, and a description of the means for ensuring adequate funds to complete decommissioning.

### 1.5.6 SUMMARY OF SECTION 6 – TECHNICAL SPECIFICATIONS AND ENVIRONMENTAL PROTECTION PLAN

This section references and summarizes the Permanently Defueled Technical Specifications and the Non-Radiological Technical Specifications (Environmental Protection Plan) that are Appendices A and B, respectively, to the TNP Facility Operating (Possession Only) License NPF-1.

### 1.5.7 SUMMARY OF SECTION 7 – QUALITY ASSURANCE, PHYSICAL SECURITY, AND FIRE PROTECTION PROVISIONS

This section references and summarizes the TNP topical reports that contain the Nuclear Quality Assurance Program, the Physical Security Plan, and the Fire Protection Program.

### 1.5.8 SUMMARY OF SECTION 8 – EVALUATION OF ENVIRONMENTAL EFFECTS OF LICENSE TERMINATION

In accordance with 10 CFR 50.82(a)(9)(ii)(G), this section compares the impacts associated with TNP site-specific license termination activities identified during preparation of the TNP License Termination Plan with environmental impacts previously analyzed in the approved TNP Supplement to the Environmental Report that accompanied the TNP Decommissioning Plan.

The evaluation in this section finds that the activities proposed as part of the TNP License Termination Plan result in no significant environmental changes not bounded by the original TNP Decommissioning Plan and the previously approved Supplement to the Environmental Report. This evaluation satisfies the requirement of 10 CFR 51.53(d) (Reference 1-16), to reflect any new information or significant environmental change associated with proposed decommissioning or fuel storage activities.

## 1.6 REFERENCES FOR SECTION 1


1-1 Code of Federal Regulations, Title 10, Part 50.82, "Application for Termination of License," August 28, 1996.

1-2 Oregon Administrative Rule 345-026-0370, "Standards for Council Approval of the Decommissioning Plan."

1-3 Code of Federal Regulations, Title 10, Part 50.54(bb).

1-4 Portland General Electric, "Trojan Nuclear Plant Defueled Safety Analysis Report," Revision 7.

1-5 Draft Regulatory Guide DG-1005, "Standard Format and Content for Decommissioning Plans for Nuclear Reactors," September 1989.

1-6 Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," January 1999.

1-7 NUREG-1700, "Standard Review Plan for Evaluating Nuclear Power Reactor License Termination Plans," April 2000.

1-8 NUREG-1727, "NMSS Decommissioning Standard Review Plan," September 2000.

1-9 Portland General Electric Letter VPN-114-93, "License Change Application 234 – Permanently Defueled Technical Specifications," July 31, 1993.

1-10 Portland General Electric Letter VPN-009-94, "Supplemental Information Regarding the Proposed Permanently Defueled Technical Specifications for the Trojan Nuclear Plant," March 8, 1994.

1-11 NRC Letter, M. T. Masnik to S. M. Quennoz, "Issuance of Amendment for Trojan Nuclear Plant (TAC No. M87167)," March 31, 1995.

1-12 Portland General Electric Topical Report PGE-1063, "Trojan Nuclear Plant Supplement to Applicant's Environmental Report – Post Operating License Stage," Revision 3.

1-13 NRC Letter, M. T. Masnik to S. M. Quennoz, "Order Approving the Decommissioning Plan and Authorizing Decommissioning of the Trojan Nuclear Plant," April 15, 1996.

1-14 NRC Letter, D. J. Wrona to S. M. Quennoz, "Issuance of Amendment (TAC No. MA6216)," dated February 12, 2001.

1-15 Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

1-16 Code of Federal Regulations, Title 10, Part 51.53, "Post-Operating License Stage Environmental Reports."

# 2. CHOICE OF DECOMMISSIONING ALTERNATIVE AND DESCRIPTION OF ACTIVITIES

## 2.1 DECOMMISSIONING ALTERNATIVE

PGE reviewed the decommissioning alternatives described in NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities" (Reference 2-1). Although the NUREG discusses three alternatives[1], DECON, SAFSTOR, and ENTOMB, the NRC concluded the ENTOMB option was less desirable and would result in a decommissioning period greater than 60 years. Consequently, PGE selected several DECON and SAFSTOR implementation methods for detailed review and analysis.

PGE chose the DECON alternative for decommissioning since this alternative minimizes financial uncertainties associated with waste disposal and other decommissioning costs. The DECON alternative will also use experienced plant personnel and allows for prompt site remediation and release for unrestricted use.

The TNP decommissioning schedule is consistent with the 60 year decommissioning time limit specified in 10 CFR 50.82 (a)(3) (Reference 2-2).

[1] NUREG-0586 defines the three alternatives as follows.

> DECON is the alternative in which equipment, structures, and portions of a facility and site containing radioactive contaminants are removed or decontaminated to a level that permits the property to be released for unrestricted use shortly after cessation of facility operations.
>
> SAFSTOR is the alternative in which the nuclear facility is placed and maintained in a condition that allows the nuclear facility to be safely stored and subsequently decontaminated to levels that permit release for unrestricted use.
>
> ENTOMB is the alternative in which radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombed structure is appropriately maintained, and continued surveillance is carried out until the radioactivity decays to a level permitting unrestricted release of the property.

## 2.2 DECOMMISSIONING ACTIVITIES, TASKS, AND SCHEDULES

### 2.2.1 INTRODUCTION

In accordance with 10 CFR 50.82(a)(9)(ii)(B), this section identifies the major dismantlement and decontamination activities that remain at TNP as of early-1999. This information details those areas and equipment that need further remediation to allow an estimation of the radiological conditions that may be encountered during remediation. Included herein are schedules for implementation of decommissioning and dismantlement activities, estimates of associated occupational radiation dose, and projected volumes of radioactive waste.

TNP decommissioning is divided into two broad periods: a Transition Period and a Decontamination and Dismantlement Period. Figure 2-11 illustrates how these periods are incorporated into the overall decommissioning schedule. This schedule was used in the preparation of the decommissioning cost estimate and funding plan discussed in Section 5. More detailed scheduling is prepared as part of pre-job planning.

The Transition Period began with permanent plant closure in January 1993 and will continue until the spent fuel is transferred to the ISFSI. Decontamination and dismantlement of the remaining facility radioactive systems, components, and structures are scheduled to be conducted upon completion of the transfer of spent fuel to the ISFSI, which is currently scheduled for around mid 2003. Some decontamination and dismantlement activities have occurred and will continue to occur during the Transition Period. Major activities completed or planned during the Transition Period are described in Section 2.2.2.

Following the Transition Period, the remaining decontamination and dismantlement activities are expected to last from mid 2003 to early 2005. Major activities planned during the Decontamination and Dismantlement Period are discussed in Section 2.2.3.

Nonradiological site restoration activities involving the final disposition of structures, systems, and components are scheduled to be completed following the termination of Facility Operating (Possession Only) License NPF-1. Some site restoration activities have been completed and others may continue to be conducted during the Transition and Decontamination and Dismantlement Periods.

### 2.2.2 TRANSITION PERIOD

Plant closure activities were initiated following the decision to permanently cease TNP power operations in January 1993. These activities culminated with the plant in a safe transition state awaiting decontamination and dismantlement. Detailed project planning and engineering activities for the Decontamination and Dismantlement Period, as discussed in Section 2.2.3.2, will continue during the Transition Period. Plant activities will continue to be implemented in compliance with the existing possession-only license and other regulatory requirements.

Removal of the four steam generators and pressurizer has been completed. These components were disposed of at the US Ecology low level radioactive waste disposal facility near Richland, Washington. Removal of the steam generators and pressurizer was accomplished through a new opening in the south face of the Containment Building. That opening is covered so that the Containment Building can be maintained in a closed condition except during active component removal. Each component was moved onto a barge at the TNP barge slip and shipped up the Columbia River to the Port of Benton, Washington, where it was off-loaded and transported for disposal at the US Ecology low level radioactive waste disposal facility near Richland, Washington. The Containment Building door will be controlled in accordance with the TNP security plan.

During 1999, the reactor vessel with internals intact (reactor vessel package) was also transported for disposal at the US Ecology near Richland, Washington. Removal of the reactor vessel package from the 10 CFR 50 licensed area of the TNP site eliminated approximately 2 million curies of activity from the TNP. Not including the spent nuclear fuel that will be transferred to the ISFSI, removal of the reactor vessel and internals resulted in elimination of greater than 99 percent of the remaining activity (curies) at the TNP facility.

Additional activities that were completed or are in process during the Transition Period include, but are not limited to the following:

1. Assessment of the functional requirements for plant systems, structures, and components.

   Plant systems, structures, and components needed to support safe storage of the spent fuel, support Spent Fuel Pool (SFP) cooling, and facilitate ongoing plant activities have been identified.

2. Deactivation/removal of plant systems, structures, and components.

   A comprehensive plant lay-up program was developed and is being implemented as described in Section 2.2.3.7. Systems, structures, and components not required to support decommissioning and spent fuel storage continue to be decontaminated and/or removed in accordance with the possession only license and approved plant procedures.

3. Redefinition of regulatory basis for the defueled plant.

   As summarized in Section 1.2, the licensing basis has and continues to be redefined to reflect the current permanently shutdown and decommissioning condition of the TNP.

4. Assessment of the plant's radiological status.

   Section 3.1 presents the results of an initial assessment of the radiological status of TNP. This assessment was used in developing this TNP Decommissioning Plan and License Termination Plan. Additional site characterization activities continue as structures, systems, and components are removed or remediated, as summarized in Section 3.1 and Section 4.2.1.

5. Final Survey Activities

   Final survey activities will begin towards the end of the Transition Period as the progress of remediation allows. Final survey activities are discussed further in Section 4.

6. Licensing and Construction of the TNP ISFSI.

   The NRC issued Materials License No. SNM-2509 for the Trojan ISFSI on March 31, 1999 (Reference 2-3). Spent fuel will be transferred from the SFP to the ISFSI in order to facilitate decontamination and dismantlement. Once the fuel is transferred to the ISFSI, the Transition Period ends and the Decontamination and Dismantlement Period begins.

## 2.2.3 DECONTAMINATION AND DISMANTLEMENT PERIOD

### 2.2.3.1 Overview

This section presents a general description of the Decontamination and Dismantlement Period activities for TNP decommissioning. These activities involve the reduction of radioactivity to acceptable levels, allowing for release of the site for unrestricted use. This information provides the basis for development of programs and procedures for ensuring safe decommissioning and a basis for detailed planning and preparation of decontamination and dismantlement activities.

During this period, the remaining contaminated systems and components will be decontaminated or removed, packaged, and either shipped to an offsite processing facility, shipped directly to a low level radioactive waste disposal facility, or handled by other alternatives in accordance with applicable regulations.

Decontamination of plant structures may be completed concurrently with equipment removal. Decontamination of structures may include a variety of techniques ranging from water washing to surface material removal. Contaminated structural material may be packaged and either shipped to a processing facility, or shipped directly to a low-level radioactive waste disposal facility. Alternative disposal methods, in accordance with applicable regulations, may also be used.

Following the removal or decontamination of contaminated systems, components, and structures, a comprehensive final radiation survey will be completed as described in Section 4. The survey will verify that radioactivity has been reduced to sufficiently low levels, as stipulated in 10 CFR 20.1402 (Reference 2-4), to allow the release of the site for unrestricted use. Upon completion of the final survey, PGE will submit a final survey report to support license termination.

#### 2.2.3.2 Detailed Planning and Engineering Activities

Detailed project plans will continue to be developed in accordance with design control procedures to support the decontamination and dismantlement activities. The plans are used to develop work packages, support ALARA reviews, aid in estimating labor and resource requirements, and track decommissioning costs and schedule.

Work packages are used to implement the detailed plans and provide instructions for actual field implementation. The work packages address discrete units of work and include appropriate hold and inspection points. Administrative procedures control work package format and content, as well as the review and approval process.

#### 2.2.3.3 General Decontamination and Dismantlement Considerations

As has been the practice during the Transition Period and in accordance with this Decommissioning Plan, the following general decontamination and dismantlement considerations, as applicable, will continue to be incorporated into decommissioning work packages during the decontamination and dismantlement period. Specific considerations are presented in Section 2.3.

Dismantlement activities are currently reviewed to ensure that they do not impact the safe storage of fuel in the SFP. During the decontamination and dismantlement period, dismantlement activities will be reviewed to ensure they do not impact the safe storage of fuel in the ISFSI licensed under 10 CFR 72. Design change work packages are implemented in accordance with administrative controls that require evaluations in accordance with the requirements of 10 CFR 50.59.

Temporary shielding is used where practical for ALARA purposes during decommissioning activities. Some dismantlement activities may be performed under water for shielding purposes as well as contamination control.

As currently practiced at TNP in accordance with this approved Decommissioning Plan, the capability to isolate or to mitigate the consequences of a radioactive release will continue to be maintained during decontamination and dismantlement activities. Isolation is the closure of penetrations and openings to restrict transport of radioactivity to the environment. This consideration should not preclude the removal of penetrations and attachments to the Containment Building, provided that openings are closed in a timely manner.

Airborne radioactive particulate emissions will continue to be filtered, and effluent discharges monitored and quantified. Consideration is given to the following items:

1. Operation of the appropriate portions of the containment ventilation and purge system, or an approved alternate system, during decontamination and dismantlement activities in the Containment Building;

2. Operation of the appropriate portions of the Auxiliary Building and Fuel Building ventilation system, or an approved alternate system, during decontamination and dismantlement activities in the Auxiliary and Fuel Buildings;

3. Operation of the Condensate Demineralizer Building ventilation exhaust system during decontamination and dismantlement activities in the Condensate Demineralizer Building; and

4. Use of local high efficiency particulate air (HEPA) filtration systems for activities expected to result in the generation of airborne radioactive particulates (e.g. grinding, chemical decontamination, or thermal cutting of contaminated components).

Work activities are planned to minimize the spread of contamination. Contaminated liquids are contained within existing or supplemental barriers and processed by a liquid waste processing system prior to release. To minimize the potential for spread of contamination, the following considerations will continue to be incorporated into the planning of decommissioning work activities:

1. Covering of openings in internally contaminated components to confine internal contamination;

2. Decontamination and dismantlement of contaminated systems, structures, and components by decontamination in place, removal and decontamination, or removal and disposal;

3. Removal of contaminated supports in conjunction with equipment removal or decontamination of supports in conjunction with building decontamination;

4. Removal of contaminated systems and components from areas and buildings prior to structural decontamination (block shield walls, or portions of other walls, ceilings, or floors may be removed to permit removal of systems and components);

5. Removal or decontamination of embedded contaminated piping, conduit, ducts, plates, channels, anchors, sumps, and sleeves during area and building structural decontamination activities;

6. Use of local or centralized processing and cutting stations to facilitate packaging of components removed in large pieces; and

7. Removal of small or compact plant components and parts intact, where feasible. (This includes most valves, smaller pumps, some small tanks, and heat exchangers. These components could then be decontaminated in whole or part, and reduced to smaller dimensions in preparation for disposal or release.)

### 2.2.3.4 Decontamination (Remediation) Methods

The process of identifying areas that need remediation is ongoing and will continue throughout decommissioning. Remediation methods that may be applied are described below. The extent to which these remediation methods are applied is based on a determination using the methodology described in Section 2.4.

For remediation purposes, areas are categorized as one of three types:

1. Structures, which include building interiors and exteriors, major freestanding exterior structures, exterior surfaces of plant systems, and paved exterior ground surfaces;

2. Land areas, which include unpaved exterior ground surfaces; and

3. Plant systems, which include interior surfaces of process piping and components.

Remediating structures may include the use of a variety of techniques ranging from water washing to surface material removal. Several factors determine the choice of the remediation method for a given application, including the extent of the contaminated area, surface material, depth of contamination, and access considerations. Demolishing certain structures may be necessary based on degraded structural integrity as a result of remediation efforts and/or removal of systems and components, surrounding walls, or other barriers.

Remediation actions performed on exposed surfaces vary with respect to the amount of residual radioactivity previously identified. Remediation activities may include wiping down an area, vacuuming to collect dirt and contamination from recesses and corners, and low or high pressure water washing of an area. Surfaces may also be remediated by scabbling or grinding the surface. Surface removal is performed using methods that control the removal depth to minimize the waste volume produced.

For concrete surfaces, remediation methods may include core drilling, concrete sawing, and scarifying or scabbling walls. This last method removes the surface of walls by bush heads, roto-peen devices, flappers, etc., and is good for removal of material close to the surface. Other methods include abrasive blasting, which is good for contamination removal from surfaces that are not necessarily smooth. Also, chipping guns and jackhammers may be used for removal of concrete surfaces as deep as the first mat of rebar. Certain surfaces with exposed cracks may require more aggressive means of removing concrete. Demolition equipment such as the BROKK and Dynahoe are used to remove concrete walls and to reduce to rubble any concrete blocks which already have been removed.

Strippable coatings can be used to remove contaminants from surfaces on which other methods don't work. Waterjet, $CO_2$ blasting, and other means may be used to clean surfaces which tend to trap contamination using other methods of remediation. Abrasive blasting may also be used to remove contaminants from steel and other surfaces besides concrete.

As discussed in Section 3.1, the remediation of surface and subsurface soils, surface water and groundwater, bottom sediment, and pavement is not expected to be necessary.

Contaminated systems and components typically are removed and sent to an offsite processing facility, sent to a low-level radioactive waste disposal facility, or decontaminated onsite and released. Although large-scale chemical decontamination is not anticipated as part of the TNP decommissioning, limited application has been and may be used to reduce radiation dose rates prior to dismantlement or general area decontamination.

Other decontamination methods typically include wiping, washing, vacuuming, scabbling, spalling, and abrasive blasting. Selection of the preferred method is based on the specific situation. Other decontamination technologies may be considered and used if appropriate.

Application of coatings and hand wiping may be used to stabilize or remove loose surface contamination. Airborne contamination control and waste processing systems are used as necessary to control and monitor releases. If structural surfaces are washed to remove contamination, controls are implemented in accordance with approved plant procedures to ensure that wastewater is collected for processing by liquid waste processing systems.

Tanks and vessels are evaluated and, if required, flushed or cleaned prior to sectioning and/or removal to reduce contamination levels and remove sludge. The following considerations are incorporated into tank and vessel sludge removal activities:

1. Precautions are taken to ensure that liquid inadvertently discharged from the tank is captured for processing by a liquid waste processing system;

2. Sludge removed from the tank is stabilized prior to shipment in conjunction with the TNP plant process control program; and

3. Wastewater will be processed and analyzed before being discharged.

#### 2.2.3.5 Dismantlement Methods

Dismantlement methods can be divided into two basic types: disassembly, and cutting or other destructive methods. Disassembly generally means removing fasteners and components in an orderly non-destructive manner (the reverse of the original assembly). Cutting methods include flame cutting, abrasive cutting, and cold cutting.

Flame cutting includes the use of oxyacetylene and other gas torches, carbon arc torches, air or oxy arc torches, plasma arc torches, cutting electrodes, or combinations of these. Most of the torches can either be handheld or operated remotely with the appropriate devices. Abrasive cutting includes the use of grinders, abrasive saw blades, most wire saws, water lasers, grit blast, and other techniques that wear away metal. Cold cutting includes the use of bandsaws, bladesaws, drilling, machining, shears, and bolt/pipe/tubing cutters.

Selection of the preferred method depends on the specific situation. Other dismantlement technologies may be considered and used if appropriate. Dismantling of systems includes the removal of valves and piping for disposal. Most valves can be removed with the piping. Larger valves and valves with actuators may be removed separately for handling purposes. Valve actuators that can be decontaminated are removed from the valves prior to pipe removal where practical.

2.2.3.6 Removal Sequence and Material Handling

Removal sequence may be dictated by access and material handling restrictions or by personnel exposure considerations. In some cases, a top-down approach is used; materials and structures at the highest elevations are removed first to allow access to components in lower levels. In other cases, different approaches may prove more efficient.

In most cases, the first items removed are those that are not contaminated, or are only slightly contaminated, to preclude contamination by other equipment. However, personnel exposure considerations may not always allow this option. Where non-contaminated equipment cannot be removed first, covers or other protection methods may be used. Similarly, non-contaminated piping should be removed from pipe chases and horizontal pipeways before cutting contaminated pipes. If this is not possible, other precautions, such as covers, are used to minimize the spread of contamination.

Where rapid cutting techniques are available, pipes and equipment can be sectioned into pieces that are manageable using light rigging or by manual lifting. Where slow cutting techniques are used, the largest manageable pieces will typically be freed and further reduced at a more convenient location.

Material removed from the Containment Building will be mainly passed through the equipment hatch into the Fuel Building, through the Containment penetration cut for large component and reactor vessel removal, or through the 45 ft Containment Roll-Up Door (CRUD) to the sorting area outside the Containment Building. The Fuel Building operating floor provides a convenient location for handling and processing of materials, but will later become congested due to the ISFSI work activities. The Fuel Building areas above the holdup tanks are surrounded by shielding walls and can be used for decontamination, sorting, or packaging. The cask wash pit, SFP, and cask load pit may be available for wash down of components.

The plant is equipped with multiple cranes, hoists, and lifting and transport systems. These systems can be used to lift and transport components and equipment to support plant decommissioning activities. Forklifts, mobile cranes, front-end loaders, and other lifting and transport devices also can be used for plant decommissioning activities. The major installed plant cranes, hoists, and lifting and transport devices that are available to support decommissioning include:

1. Containment Building polar crane;
2. Fuel Building overhead crane;
3. Auxiliary Building electric hoist;
4. Auxiliary Building elevator;
5. Equipment room monorails;
6. SFP bridge crane; and
7. Condensate Demineralizer Building bridge crane.

Inspection requirements for the Containment Building Polar Crane, Fuel Building Overhead Crane, Auxiliary Building Electric Hoist, SFP Bridge Crane, and the Condensate Demineralizer Building Bridge Crane are specified in Trojan Plant Maintenance Procedure MP 1-20, "Cranes, Hoists and Winches." Chainfalls and other temporary hoists are inspected and verified to be in good working condition at the time of issue from the Tool Room. The Auxiliary Building Elevator is inspected by a State Inspector in accordance with State of Oregon requirements.

The Containment Building polar crane is capable of reaching most locations inside the Containment Building and can handle large, heavy loads. The Fuel Building overhead crane has access to a hoistway open to plant grade at the 45 ft elevation. The Auxiliary Building elevator has access to upper floors in the building and can carry small loads.

Installed cranes and hoists may be used in conjunction with temporary or mobile lifting and transport devices to support decommissioning. The installed plant cranes, hoists, and other lifting devices can be decontaminated and dismantled when they no longer are required to support decommissioning activities.

#### 2.2.3.7 System Deactivation

Systems or components will continue to be deactivated prior to decontamination and dismantlement. In general, deactivation is implemented by mechanical isolation of interfaces with operating plant systems, draining piping/components, and de-energizing electrical supplies. Combustible material (e.g., charcoal from filters, lube oil) is removed from the deactivated components where possible. Chemicals used in, or resulting from, decommissioning activities are controlled in accordance with the plant chemical safety program. Plant drawings are revised to indicate deactivated portions of systems. Plant procedures are modified to reflect the changes.

Deactivation of plant systems is administratively controlled by approved plant procedures. The design change process is used to remove components, lift electrical leads, install electrical jumpers, cut and cap piping systems, or install blank flanges.

#### 2.2.3.8 Temporary Systems to Support Decommissioning

Decontamination and dismantlement of systems, structures, and components often require the removal of interferences. Removal of some of these interferences may eliminate power, service air, and other services used for decommissioning. It may also become impractical at some point to continue using installed plant systems. Temporary services and systems can be provided to support decommissioning activities. Temporary modifications to plant structures, systems, and components are controlled by plant design control procedures.

Portable electric power packs can be powered from motor control centers, load centers, or the yard loop. These portable load centers can supply cutting, hoisting, temporary lighting, or other power needs. Service air can be provided by portable air compressors using hoses or temporary air manifolds. Demineralized water is available from portable demineralizer skids or portable tankers brought from offsite. Portable hydraulic power centers can be used to power hydraulic equipment.

Temporary liquid and solid waste processing systems may be used during decommissioning for processing plant waste. These systems may include filters and/or demineralizers, and may be used at one or more locations in the waste-processing path.

Portable radiation monitors and air monitoring equipment can provide localized radiation monitoring. Localized temporary ventilation equipment and HEPA filtration may be used to supplement building ventilation and minimize the spread of radioactive particulate contamination.

## 2.3 REMAINING DISMANTLEMENT ACTIVITIES

### 2.3.1 IDENTIFICATION OF REMAINING SYSTEMS, STRUCTURES, AND COMPONENTS

This section summarizes the structures, systems, and components remaining at TNP and the associated schedule and status of dismantlement and decontamination. Radiological implications, such as occupational exposure estimates, radioactive waste characterization, and radioactive material release estimates, associated with the remaining TNP site dismantlement activities are summarized in Section 2.5.

The remaining dismantlement and decontamination activities can be classified into several phases, the implementation of which may overlap. The first phase includes removal or in-place decontamination of contaminated systems and components not required for support of fuel storage or subsequent decontamination activities. The second phase includes the removal or decontamination in place of systems and components that support the SFP or fuel storage. The third phase involves the remediation (removal or decontamination in place) of remaining structures, systems, and components, including the SFP itself. These phases primarily are implemented on an area-by-area basis. Using this approach, often only a part of the system will be removed, with the remaining portion awaiting removal in an adjacent area or being maintained in service to support spent fuel storage and defueled plant operation.

The following Sections 2.3.1.1 through 2.3.1.3 list the major structures, systems, and components remaining at TNP as of January 1999, as categorized into one of the three phases described above. The status of decontamination and dismantlement of these systems, structures, and components, as of January 1999, is summarized in Table 2-1. Table 2-2 contains a list of major components removed during each year since 1996. As indicated by Table 2-1 and Table 2-2, the majority of radiologically contaminated systems and components not required to support the storage of spent fuel have been deactivated, dismantled, and disposed of in accordance with the TNP Decommissioning Plan. Additional detail regarding decontamination, dismantlement, and radiological controls for these structures, systems, and components is provided in Section 2.3.2.

#### 2.3.1.1 Remaining Structures, Systems, And Components Not Required For Spent Fuel Storage (Phase 1)

The remaining systems and components listed below are not necessary to support spent fuel storage. These systems and components will continue to be dismantled and decontaminated in accordance with Sections 2.2 and 2.3.2.

The following systems are classified as Phase 1, with the applicable section of this Decommissioning Plan referenced in parentheses:

- Reactor Vessel and Internals (Section 2.3.2.1)
- Component Cooling Water System (Section 2.3.2.3)
- Portions of the Service Water System (Section 2.3.2.4)
- Original Spent Fuel Cooling and Demineralizer System (Section 2.3.2.6)

- Remaining Portions of Condensate Demineralizer System (Section 2.3.2.8)
- Refueling Water Storage Tank (Section 2.3.2.10)
- Portions of the Plant Effluent System (Section 2.3.2.11)
- Portions of Containment Ventilation Systems (Section 2.3.2.12)
- Portions of the Fuel and Auxiliary Building Ventilation Systems (Section 2.3.2.13)
- Portions of the Compressed Air Systems (Section 2.3.2.15)
- Gaseous Radioactive Waste System (Section 2.3.2.16)
- Solid Radioactive Waste System (Section 2.3.2.17)
- Portions of the Liquid Radioactive Waste System (Sections 2.3.2.18)
- Portions of the Radiation Monitoring System (Section 2.3.2.19)
- Process Sample System (Section 2.3.2.20)
- Portions of the Fire Protection System (Section 2.3.2.21)
- Portions of the Electrical Systems (Section 2.3.2.22)
- Portions of contaminated embedded piping[2]

#### 2.3.1.2 Remaining Systems, Structures, and Components Associated With Spent Fuel Storage (Phase 2)

After the spent fuel and spent fuel debris have been removed from the SFP and transferred to the ISFSI, components and systems listed below will be dismantled and/or decontaminated in accordance with Sections 2.2 and 2.3.2. Some components within the SFP may be removed prior to completion of spent fuel removal activities. Appropriate administrative controls will continue to be used to ensure removal activities are performed in a manner that will not adversely impact the safe storage of spent fuel.

The following list of structures, systems, and components are classified as Phase 2, with the applicable section of this Decommissioning Plan referenced in parentheses:

- Contaminated embedded piping[2]
- Boric Acid Batch Tank (portion of CVCS System) (Section 2.3.2.2)
- Portions of the Service Water System (Section 2.3.2.4)
- Spent Fuel Storage and Handling Equipment (Section 2.3.2.5)
- Modular SFP Cooling and Cleanup System (Section 2.3.2.7)
- Portions of the Fuel and Auxiliary Building Ventilation Systems (Section 2.3.2.13)
- Portions of the Compressed Air Systems (Section 2.3.2.15)
- Portions of the Liquid Radioactive Waste System (Sections 2.3.2.18)

[2] The majority of contaminated embedded piping, which primarily includes piping from various radioactive waste drain systems, embedded ventilation ductwork, and buried process piping, is expected to be decontaminated to acceptable levels as adopted in Section 4, Final Survey Plan, to satisfy site release criteria. Portions of the piping may be removed.

- Portions of the Radiation Monitoring System (Section 2.3.2.19)
- Portions of the Fire Protection System (Section 2.3.2.21)
- Portions of the Electrical Systems (Section 2.3.2.22)

#### 2.3.1.3 Remaining Structures, Systems, And Components (Phase 3)

Contaminated structural concrete, steel, and other building materials and remaining components of systems listed below will be removed or decontaminated in place in a manner consistent with Sections 2.2 and 2.3.2. The SFP itself will be remediated following removal of fuel and draining of the SFP water. Decontamination of the structures will be performed on an area-by-area basis, with the majority of the decontamination activities occurring following completion of equipment removal from an area. Demolition of the building structures is not anticipated prior to termination of the TNP 10 CFR 50 license.

The following list of structures, systems, and components are classified as Phase 3, with the applicable section of this Decommissioning Plan referenced in parentheses.

- Service Water System (Section 2.3.2.4)
- SFP (Section 2.3.2.5)
- Plant Effluent System (Section 2.3.2.11)
- Containment Ventilation System (Section 2.3.2.12)
- Fuel and Auxiliary Building Ventilation Systems (Section 2.3.2.13)
- Condensate Building Ventilation System (Section 2.3.2.14)
- Compressed Air Systems (Section 2.3.2.15)
- Liquid Radioactive Waste System (Sections 2.3.2.18)
- Radiation Monitoring System (Section 2.3.2.19)
- Fire Protection System (Section 2.3.2.21)
- Electrical Systems (Section 2.3.2.22)
- Containment Building (Section 2.3.2.23)
- Auxiliary Building (Section 2.3.2.24)
- Fuel Building (Section 2.3.2.25)
- Other Buildings (Section 2.3.2.26)

### 2.3.2 GENERAL DESCRIPTION OF AND REMEDIATION CONSIDERATIONS FOR REMAINING SYSTEMS, STRUCTURES, AND COMPONENTS

This section presents a summary description of the TNP systems, components, and structures remaining that are known or considered to be internally contaminated or that may be used to support decommissioning activities. This discussion includes general activities and remediation considerations associated with decommissioning these systems, structures, and components. For reference, a site area map current as of January 1999 is provided in Figures 2-1 and 2-1a. Plant layout and general arrangement drawings reflecting the TNP facility at the time of plant shutdown are provided in Figures 2-2 through 2-9.

Because external contamination is generally considered to exist on systems, structures and components located in the RCA's of the plant, it is not specifically noted in the following system discussions. However, systems, components, and structures that are externally contaminated will be decontaminated for release or disposed of as radioactive waste.

This section is intended to provide general information and guidance for work package planning and is not required to be updated to reflect equipment removal. However, Table 2-2 is updated annually to provide a list of major plant systems and system components that were removed the previous year (beginning with 1996).

#### 2.3.2.1 Reactor Vessel and Internals

During 1999, PGE removed the reactor vessel with internals intact (reactor vessel package) from the 10 CFR 50 licensed area of the TNP site. The reactor vessel package was transported for disposal at the US Ecology low level radioactive waste facility near Richland, Washington. Removal of the reactor vessel package from the 10 CFR 50 licensed area of the TNP site eliminated approximately 2 million curies of activity from the TNP. Not including the spent nuclear fuel that will be transferred to the ISFSI, removal of the reactor vessel and internals resulted in elimination of greater than 99 percent of the remaining activity (curies) at the TNP facility.

#### 2.3.2.2 Chemical and Volume Control System

Located in the Fuel Building, the remaining portion of the CVCS primarily supports boron addition to the SFP. The only remaining major component is the boric acid batching tank. The boric acid batching tank is not internally contaminated. The tank can be removed in one piece or can be segmented.

#### 2.3.2.3 Component Cooling Water System

The major components of the Component Cooling Water (CCW) system have been removed. Portions of the system piping remain. The system piping is not internally contaminated.

No specific considerations apply to the remaining system equipment.

#### 2.3.2.4 Service Water System

The service water system supplies raw water from the Columbia River via the intake structure for emergency make-up to the spent fuel pool and for dilution flow for plant liquid discharges. Remaining major system components include three service water pumps and associated valves, piping, fittings, and instrumentation. The active portion of the system piping is located in outside areas. The system is not internally contaminated.

No specific considerations apply to the remaining system equipment.

#### 2.3.2.5 SFP and Fuel Handling Equipment

The SFP and fuel storage structures consist of the SFP, the spent fuel storage racks, the fuel transfer canal, the cask load pit, and the new fuel storage area. The SFP provides for irradiated fuel storage. Additionally, the SFP provides a transparent radiation shield for personnel. Fuel assemblies are stored in stainless steel spent fuel storage racks located at the bottom of the SFP. The fuel transfer canal and cask loading pit facilitated handling of irradiated fuel by providing isolable underwater operating areas for fuel transfer evolutions. The new fuel storage area provided a protected area for dry, subcritical storage of new fuel assemblies, and now houses portions of the modular SFP cooling and cleanup system (Section 2.3.2.7). The fuel transfer canal and cask loading pit are connected to the SFP by transfer slots which can be closed and sealed by leak-tight gates. The SFP, fuel transfer canal, and the cask loading pit are reinforced concrete structures with seam-welded stainless steel linings.

The SFP and remaining fuel handling equipment are located in the Fuel Building. The SFP supports spent fuel storage and will be required until the fuel is moved to the ISFSI. The fuel handling equipment in the SFP may be required to transfer the fuel to the ISFSI. Additionally, the SFP bridge crane may be required to support the decommissioning of the SFP. The system is contaminated. The following specific considerations apply.

The potential for high levels of contamination exists for components removed from the SFP. The spent fuel storage racks are accessible with the Fuel Building crane and could be removed from the SFP intact for sectioning and packaging at another location. The liner of the SFP, fuel transfer canal, and cask load pit may be sectioned for removal. It may be possible to decontaminate and release sections of the liners, and the spent fuel storage racks.

Dismantlement of the majority of the SFP and fuel handling radioactive systems, components, and structures is scheduled to be conducted upon completion of the transfer of spent fuel to the ISFSI. Spent fuel will be in the ISFSI, and other items stored in the SFP will be moved to alternate storage locations or disposed of, prior to the SFP liner being sectioned for removal or decontaminated in place.

#### 2.3.2.6 SFP Cooling and Demineralizer System (Original System)

The SFP cooling and demineralizer system was replaced by the modular SFP cooling and cleanup system in 1998. Remaining major system components include a skimmer pump and associated valves, piping, and instrumentation. Portions of the system piping which formerly entered the SFP have been capped on the exterior of the SFP wall, limiting the SFP piping wetted surface area to a minimum. The modular SFP cooling and cleanup system now provides cooling and water purification functions.

No specific considerations apply to the remaining system equipment. The skimmer filter housing and pump are easily accessible and can be removed intact.

#### 2.3.2.7 Modular SFP Cooling and Cleanup System

The Modular SFP Cooling and Cleanup System removes the decay heat from the spent fuel elements stored in the SFP and purifies the system water inventory. Major equipment associated with the system includes two pumps, two water-to-air coolers, a demineralizer, a filter, and associated valves, piping, fittings, and instrumentation.

No specific considerations apply to removal of this system or its components. The system is leased and the components are skid mounted and will likely be removed intact for return shipment to the supplier when their SFP support function is no longer required.

#### 2.3.2.8 Condensate Demineralizers

The remaining major system components include a backwash system and associated valves, piping, fittings, and instrumentation. The system equipment is located in the Condensate Demineralizer Building. The backwash receiver tank is potentially internally contaminated. Other portions of the system have detectable levels of contamination.

No specific considerations apply to the remaining system components.

#### 2.3.2.9 Steam Generator Blowdown System

The major components of the steam generator blowdown system have been removed. Portions of the system piping remain and are included in the embedded piping scope.

No specific considerations apply to the remaining system equipment.

#### 2.3.2.10 Primary Makeup Water System and Refueling Water Storage Tank

The major system components remaining as of January 1999 include the refueling water storage tank (RWST) and portions of the associated system piping, which is included in the embedded piping scope. The RWST is a vertical tank constructed of austenitic stainless steel with immersion heaters. The RWST is located in the tank farm (south of the Containment Building). The RWST and system piping is internally contaminated.

The following specific considerations apply. Due to its size, the RWST will be sectioned to facilitate packaging and shipping. It may be possible to decontaminate and release sections of the tanks.

#### 2.3.2.11 Plant Effluent System

The Turbine Building sump, oily water separator, solids settling basin, and discharge and dilution structure together comprise the plant effluent system. The plant effluent system provides a means of discharging plant liquid wastes while ensuring compliance with the National Pollutant Discharge Elimination System Waste Discharge Permit.

The plant effluent system components are located throughout the plant site. Portions of the system are required to support collection and disposal of waste generated by decommissioning activities. The Turbine Building sump is contaminated. Other portions of the system, such as the oily water separator, may also be contaminated; they are still being used to support ongoing plant activities. These components can be sampled for contamination when they are no longer in use.

The following specific considerations apply. Turbine Building sump contamination will be removed using concrete decontamination techniques described in Section 2.2.3.4. Sump input and discharge piping will be checked for contamination to determine the proper method of disposal. Buried and embedded piping may be left in place if it meets site release criteria. The effluent diffusion pipe can be removed by divers if determined to be necessary.

#### 2.3.2.12 Containment Ventilation Systems

The remaining Containment ventilation systems consist of the purge supply and purge exhaust systems. Containment purge exhaust is directed to the primary vent stack that is attached to the outside of the Containment Building. Exhaust air is monitored for radiation and is exhausted through a vent at the top of the Containment Building.

Portions of the Containment ventilation systems are located inside the Containment Building, the Auxiliary Building, and the Main Steam Support Structure. The primary vent stack is attached to the outside of the Containment Building. Portions of the system will be used to maintain a habitable environment and control contamination during decommissioning. The systems are internally contaminated. The following specific considerations apply.

The systems will remain in service until:

1. The individual system or component has been evaluated as not required to support further decommissioning activities; or

2. An alternate system has been established; or

3. Contaminated components for all systems in the building have been removed or remediated, and the building has been decontaminated.

#### 2.3.2.13 Fuel Building and Auxiliary Building Ventilation Systems

The Fuel Building and Auxiliary Building ventilation systems provide for the supply, heating, cooling, and exhaust of air for the Fuel and Auxiliary Buildings. The remaining systems include several subsystems: Fuel and Auxiliary Building supply system, Fuel and Auxiliary Building exhaust system, SFP exhaust system, space heating system, radioactive waste annex supply and return, and pump cooling units. Air is exhausted through the primary vent stack that is attached to the outside of the Containment Building. Exhaust air is monitored for radiation. The other systems provide heating and cooling for specific areas in the buildings.

The Fuel and Auxiliary Building ventilation systems are located inside the Fuel and Auxiliary Buildings. Portions of the systems will be used to maintain a habitable environment and control contamination during decommissioning. The systems are internally contaminated. The following specific considerations apply.

The systems will remain in service until:

1. The individual system or component has been evaluated as not required to support further decommissioning activities; or

2. An alternate system has been established; or

3. Contaminated components for all systems in the building have been removed or remediated, and the building has been decontaminated.

2.3.2.14 Condensate Demineralizer Building Ventilation System

The Condensate Demineralizer Building ventilation system provides for supply and exhaust air in the building. Supply air is provided through infiltration and, as appropriate, roof supply fans. Exhaust air is monitored using a sample pump, sample probe, and a radioactive airborne particulate monitoring filter.

The system will remain available for service until the Condensate Demineralizer Building is no longer used to process radioactive waste, and the building is decontaminated.

2.3.2.15 Instrument and Service Air System

The instrument and service air system supplies compressed air required for pneumatic instruments, valves, and service air outlets throughout the plant. The system has four air compressors, aftercoolers, air receivers, filters, dryers, and associated valves, piping, fittings, and instrumentation.

The instrument and service air system is located in buildings throughout the plant. The system may be used for operation of control valves, dampers, tools, and breathing air. As a portion of the instrument and service air system is determined not to be required to support further decommissioning, it may be deactivated and removed. The system is not considered to be internally contaminated.

2.3.2.16 Gaseous Radioactive Waste System

The majority of gaseous radioactive waste system components have been removed and the entire system has been removed from service. Remaining major equipment includes the vent collection header exhaust fan and associated piping, fittings, filters, and instrumentation. The remaining gaseous radioactive waste systems are located in the Auxiliary and Fuel Buildings.

The following specific considerations apply. The gas collection header and vent collection header exhaust filter housings can be removed intact.

2.3.2.17 Solid Radioactive Waste System

Major remaining equipment associated with the solid radioactive waste system includes a filter handling vehicle, and some piping. Solid wastes generated as a result of plant system operation or decommissioning activities are processed in accordance with the site Process Control

Program. The solid radioactive waste system components are located in the Auxiliary and Fuel Buildings. The system is contaminated.

No specific considerations apply for the remaining system equipment.

2.3.2.18 Liquid Radioactive Waste System

The liquid radioactive waste system collects, stores, processes, and disposes of contaminated liquids, including radioactive wastewater generated during decommissioning activities. System components are used to process water and monitor it during discharge. Major remaining system equipment includes two treated waste-monitor tanks and pumps, the dirty waste drain tank and pumps, the Auxiliary Building sumps and pumps, Containment sumps, various filters, and associated valves, piping, fittings, and instrumentation. The liquid waste system components and piping are located primarily in the Auxiliary and Fuel Buildings and the Containment Building, with several floor and equipment drains located within the Main Steam Support Structure and Control Building. The system is internally contaminated.

The following specific considerations apply. System tanks can be sectioned to facilitate removal. System pumps are small and can be removed intact. The filter skids will be dismantled and the bag filter housings removed intact. System sumps are concrete pits that will be decontaminated. Techniques for concrete decontamination and demolition are noted in Section 2.2.3.4.

Temporary water cleanup systems may be used to reduce the amount of installed equipment required to remain operational and still maintain the ability to collect, store, process and discharge radioactive liquid waste. Plumbing modifications may be required to use temporary systems. Temporary systems to support decommissioning are discussed in Section 2.2.3.8. Radioactive liquid effluents will be monitored and released in accordance with the requirements of topical report PGE-1021, "Offsite Dose Calculation Manual" (ODCM) (Reference 2-5).

2.3.2.19 Radiation Monitoring System

The radiation monitoring system consists of the process and effluent radiological monitoring systems (PERMS) and the area radiation monitoring system (ARMS). The PERMS provide monitoring of gaseous and liquid effluent release paths and selected gaseous and liquid plant process systems. The ARMS provide remote monitoring of selected areas at the plant. Portions of the radiation monitoring system are considered to be contaminated.

The following specific considerations apply. The following PERMS channels will be used during decommissioning activities for effluent monitoring. Temporary power or monitoring channels may be used to support decommissioning.

1. PRM-9 (liquid radioactive waste effluent discharge) monitors liquid waste effluent during discharges. The PRM may be moved from its current location due to possible discharge line plumbing modifications.

2. PRM-2A (Auxiliary Building ventilation discharge - particulate channel) is designed to monitor the discharge of radioactive particulates through the Auxiliary Building exhaust ventilation stack.

3. PRM-1A (Containment Building ventilation discharge - particulate channel) is designed to monitor the discharge of radioactive particulates through the Containment Building exhaust ventilation stack.

While spent fuel is in the SFP, PERMS channels PRM-2A and PRM-2C (Auxiliary Building ventilation exhaust low level gas channel) and ARMS channels ARM-12 (Fuel Building, elevation 93 ft, machine shop access) and ARM-13 (Fuel Building, elevation 93 ft, new fuel storage area) will be required. Localized radiation monitoring will be provided in work areas by temporary monitoring instrumentation, when necessary.

Due to the mounting configuration of many of the PERMS detectors, they are not likely to be significantly contaminated. The sample tubing and supply piping for PERM-1 (Containment Ventilation) and PERM-2 (Auxiliary Building Ventilation) are potentially contaminated and will be sectioned as necessary for removal. The skid housings should be dismantled to facilitate removal of various detectors, lead detector housings, and sample piping. Sample pumps should be removed intact for disposal.

#### 2.3.2.20 Process Sampling System

The major components of the process sampling system have been removed and required sampling is performed locally. Portions of the system piping remain. Portions of the system are internally contaminated. No specific considerations apply to the remaining system equipment.

#### 2.3.2.21 Fire Protection System

The fire protection system provides manual and automatic fire suppression and automatic fire detection for plant areas. The fire protection system includes the following: portable fire extinguishers, water supply and distribution systems, fire suppression system, emergency lighting, and the fire detection and alarm system. The main fire pumps are located in the intake structure.

The fire protection system is located in buildings and areas throughout the plant site. The system is not considered to be internally contaminated. The following specific considerations apply.

Sections of the fire protection system may be deactivated and removed from service when no longer required to support further decommissioning activities or after alternate fire detection and suppression capability has been established. Such changes are changes to the TNP Fire Protection Program, and will be made in accordance with the provision of License Condition C.(8) of Facility Operating (Possession Only) License NPF-1 (Reference 2-6).

2.3.2.22 Electrical Systems

The electrical system includes the main generator, the switchyard, main and auxiliary transformers, and the 230 kV ac, 12.47 kV ac, 4160 V ac, 480 V ac, 120 V ac, 250 V dc, 125 V dc, and lighting distribution systems.

The electrical systems are located in buildings and areas throughout the plant site. Portions of the systems (primarily 12.47 kV ac yard loop, 480 V ac, and 120 V ac) may be used to support decommissioning activities. The systems are not considered to be internally contaminated. The following specific considerations apply.

Temporary electrical services may be used as required during decommissioning to facilitate dismantling and removal of plant components. When a system or component is no longer required, the electrical supply to the component may be isolated and removed. Plant lighting may also be required until building demolition. Lighting and electrical power may be provided by temporary services as discussed in Section 2.2.3.8.

2.3.2.23 Containment Building

Most equipment has been removed from the Containment Building. The only major equipment remaining in the Containment Building is the overhead polar crane. The Containment Building is a concrete structure in the shape of a cylinder with a hemispherical roof and flat foundation. The approximate dimensions of the Containment Building are: 124-ft inside diameter, 203-ft inside height, 3½-ft wall thickness and 2½-ft dome thickness.

The following discussion provides general information about the construction details of the Containment Building. The cylindrical section has a post-tensioning system consisting of vertical and hoop tendons. The dome has a two-way post-tensioning system consisting of hoop tendons and continuous vertical tendons. Containment tendons have been detensioned, and some tendons have been removed. The inside of the concrete shell is steel-lined. During plant operation, the liner plate was coated with an epoxy-phenolic finish that is approximately 5 mils thick generally to a height of 6 ft above the floors, and 2 to 3 mils inorganic topcoat above that. Penetrations in the Containment Building include openings for the equipment hatch, two personnel air locks, and numerous smaller electrical and mechanical penetrations. Additionally, since decommissioning activities started, openings with doors have been created in the south wall of containment to support removal of large components and for general material removal considerations.

The following specific considerations apply.

The Containment Building internal concrete has been removed. Plate steel, structural steel, grating, ladders, and platforms may be decontaminated in place or may be removed by unbolting or cutting, and rigged out for decontamination or disposal. The polar crane may be removed or decontaminated in place.

### 2.3.2.24 Auxiliary Building (Including Pipe Facade)

The Auxiliary Building has two floors below grade, one at grade (elevation 45 ft), and three floors above grade. The portion at or above grade is structurally connected to the Fuel Building on the east and to the Control Building on the west. A number of framing members in the Auxiliary Building are supported by the Containment Building wall.

The exterior walls below grade and slabs are constructed of reinforced concrete. Interior framing members below grade and framing members above grade are structural steel. Exterior walls above grade are generally constructed of concrete masonry block with exterior precast concrete panels (elevation 45 ft) or metal siding at the upper floors. Interior walls are constructed of concrete block masonry. Portions of the Auxiliary Building are coated with an epoxy surface. The surface is generally applied to the floors and to a height of 12 inches above the floor, but may extend up to 6 ft in corridors and selected rooms.

Most of the major contaminated equipment has been removed from the Auxiliary Building since the start of decommissioning activities. Major plant equipment remaining in the Auxiliary Building includes the following items:

1. Ventilation equipment supporting the Auxiliary, Fuel, and Containment Buildings;

2. Filters; and

3. Liquid radioactive waste system components.

Portions of exposed surfaces in the Auxiliary Building are contaminated. The following specific considerations apply.

The surfaces of walls and slabs in traffic areas have protective coatings. Concrete can be decontaminated by water or chemical washing. Surfaces that cannot be decontaminated can be scabbled or surface ground down to non-contaminated depths.

### 2.3.2.25 Fuel Building

The Fuel Building contains facilities for storage of spent fuel and systems used for processing liquid wastes generated by plant operation and decommissioning activities. Remaining areas of note in the building include four floors above grade, the SFP, cask loading pit, new fuel storage pit, and cask wash pit. Portions of the Fuel Building are coated with an epoxy surface. The surface is generally applied to the floor and to a height of 12 inches above the floor, but extends up to 6 ft in corridors and selected rooms. The walls and base slab of the SFP are constructed of thick (approximately 5-ft to 6½-ft) reinforced concrete. The SFP and fuel handling equipment are discussed in Section 2.3.2.5.

Most of the major contaminated equipment has been removed from the Fuel Building since the start of decommissioning activities. Major plant equipment remaining in the Fuel Building includes the following items:

1. SFP;

2. Modular SFP cooling and cleanup system; and

3. Liquid radioactive waste system components.

Portions of exposed surfaces inside the Fuel Building are contaminated. The following specific considerations apply. The surfaces of walls and slabs in traffic areas have protective coatings. Concrete can be decontaminated by water or chemical washing. Surfaces that can not be decontaminated can be scabbled or surface ground down to non-contaminated depths. Removal of entire walls or portions of walls may require evaluation of the building's structural integrity.

2.3.2.26 Other Buildings

The Main Steam Support Structure (MSSS) consists of two floors, one at grade (Elevation 45 ft) and one at Elevation 63 ft. It is located between the Containment Building and Turbine Building and provided protection and support for the main steam isolation, power-operated relief and safety valves, as well as main steam and feedwater piping. The structure is constructed of reinforced concrete and structural steel. Portions of the MSSS are potentially contaminated.

The Condensate Demineralizer Building is a three-story, partially below grade structure located west of the Turbine Building. The building is used for temporary storage and processing of low-level radioactive waste prior to disposal. Portions of the building will require decontamination from radwaste processing activities.

The Steam Generator Blowdown (SGBD) Building is located south of the MSSS, between the Containment Building and Turbine Building. The building has a reinforced concrete slab floor, reinforced masonry block walls, and a reinforced concrete roof supported by steel beams and metal decking. The building has a foundation curb designed to contain liquid spills. Decontamination of the SGBD building has been completed.

The Radwaste Annex is a single-story windowless structure adjacent to the north wall of the Fuel Building. It was utilized for laundry sorting, storage, and frisking, as well as solid waste compaction and drum storage.

The Wright-Schuchart-Harbor (WSH) Warehouse is a Quonset type building that has been partially removed and is currently used as a staging area to support ISFSI activities. The WSH Warehouse has undergone final survey as documented in PGE-1074, "Trojan Nuclear Plant Final Survey Report for the ISFSI Site" (Reference 2-7). The WSH Warehouse will eventually be dismantled and portions of its concrete slab demolished or modified as required.

The Turbine Building is located west of the Containment and Control buildings. It houses the plant's secondary side components, including the turbine generator, condenser, and feedwater equipment. Portions of the Turbine Building are potentially contaminated.

## 2.4 SITE REMEDIATION PLANS

### 2.4.1 INTRODUCTION

The purpose of this section is to describe how remediation actions may be applied to various areas on the TNP site, identify the remediation methodology to be used, and demonstrate that the remediation methodology is adequate to ensure that the site release criteria of 10 CFR 20.1402 are met. Verification of the site release criteria is detailed further in Section 4.

### 2.4.2 REMEDIATION LEVELS

The ALARA evaluation uses action levels, referred to as remediation levels, that are established for various types of remediation actions such as chemical decontamination, wiping, washing, vacuuming, scabbling, spalling, abrasive blasting, and high pressure washing. A remediation level is the level of residual radioactivity at which the desired beneficial effects due to the performance of a given remediation action are equal to the undesirable effects or costs of the action. The methodology and equations used here for calculating remediation levels are from draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination" (Reference 2-8).

Remediation levels are developed using an unbiased analysis of remediation actions which can both avert future dose (a benefit to society) and cost money (a potential detriment to society). In order to compare the benefits and costs of a remediation action, the benefits and costs are given a monetary value. The monetary value of the collective averted dose (the benefit) is compared with the monetary value of the undesirable effects (the cost). The remediation level is the point at which the benefits of the remediation action equal the costs.

#### 2.4.2.1 Remediation Level Calculation

Remediation levels are calculated using Equation 2-1:

$$RL = \frac{Cost_T}{(2000)\,(PD)\,(0.025)\,(F)\,(A)} \times \frac{r + \lambda}{1 - e^{-(r+\lambda)Y}} \qquad \text{(Equation 2-1)}$$

where:

| | | |
|---|---|---|
| RL | = | remediation level fraction of DCGL (dimensionless) |
| $Cost_T$ | = | total monetary cost of remediation action ($) |
| 2000 | = | value of a person-rem averted ($/person-rem) |
| PD | = | population density for the critical group scenario (persons/$m^2$) |
| 0.025 | = | annual dose to an average member of the critical group from residual radioactivity at the DCGL concentration (rem/yr) |
| F | = | remediation action effectiveness (dimensionless) |
| A | = | area being evaluated ($m^2$) |
| r | = | monetary discount rate ($yr^{-1}$) |
| $\lambda$ | = | radiological decay constant for the radionuclide ($yr^{-1}$) |
| Y | = | number of years over which collective averted dose is calculated (years) |

Acceptable values for the equation parameters of population density, PD; the monetary discount rate, r; and the number of years, Y, are taken from Table 3.1 of draft Regulatory Guide DG-4006 and are given in the table which follows. The development of the equation parameters of total cost, $Cost_T$, and remediation action effectiveness, F, are described in Sections 2.4.2.2 and 2.4.2.3. A justification is provided where values are used other than those given in the following table or calculated in accordance with Sections 2.4.2.2 or 2.4.2.3.

Remediation Level Equation Values

| **Equation Parameter** | **Acceptable Value** | |
|---|---|---|
| | **Building** | **Land** |
| PD | 0.09 | 0.0004 |
| r | 0.07 | 0.03 |
| Y | 70 | 1000 |

2.4.2.2 Calculation Of Total Cost

The total monetary cost for performing a given remediation action, $Cost_T$, is calculated using Equation 2-2:

$$Cost_T = Cost_R + Cost_{WD} + Cost_{Acc} + Cost_{TF} + Cost_{WDose} \qquad \text{(Equation 2-2)}$$

where:

$Cost_R$ = monetary cost of the remediation action, including mobilization costs
$Cost_{WD}$ = monetary cost for transport and disposal of waste generated by the remediation action
$Cost_{Acc}$ = monetary cost of worker accidents during the remediation action
$Cost_{TF}$ = monetary cost of traffic fatalities during transporting of waste
$Cost_{WDose}$ = monetary cost of dose received by workers performing the remediation action and transporting waste to the disposal facility

Other monetary costs may be included as appropriate for the particular situation.

The monetary cost of the remediation action, $Cost_R$, is the cost of performing the remediation action, including equipment mobilization and demobilization and labor costs.

The cost of waste transport and disposal, $Cost_{WD}$, is calculated using Equation 2-3:

$$Cost_{WD} = (V_A)(Cost_V) \qquad \text{(Equation 2-3)}$$

where:

$V_A$ = volume of remediation waste produced ($m^3$)
$Cost_V$ = cost of waste disposal ($\$/m^3$)

The cost of workplace accidents, $Cost_{Acc}$, is calculated using Equation 2-4:

$$Cost_{Acc} = (3{,}000{,}000)\ (4.2 \times 10^{-8})\ (T_A) \qquad \text{(Equation 2-4)}$$

where:

| | | |
|---|---|---|
| 3,000,000 | = | monetary value of a fatality equivalent to \$2,000/person-rem (\$) |
| $4.2 \times 10^{-8}$ | = | workplace fatality rate ($hrs^{-1}$) |
| $T_A$ | = | worker time required for remediation (person-hrs) |

The cost of traffic fatalities incurred during the shipment of waste, $Cost_{TF}$, is calculated using Equation 2-5:

$$Cost_{TF} = \frac{(3{,}000{,}000)\ (V_A)\ (3.8 \times 10^{-8})\ (890)}{13.6} \qquad \text{(Equation 2-5)}$$

where:

| | | |
|---|---|---|
| 3,000,000 | = | monetary value of a fatality equivalent to \$2,000/person-rem (\$) |
| $V_A$ | = | volume of remediation waste produced ($m^3$) |
| $3.8 \times 10^{-8}$ | = | truck fatality rate per kilometer traveled ($km^{-1}$) |
| 890 | = | distance traveled by truck (km) |
| 13.6 | = | volume of a truck shipment ($m^3$) |

The cost of remediation worker dose, $Cost_{WDose}$, is calculated using Equation 2-6:

$$Cost_{WDose} = (2{,}000)\ (D_R)\ (T_A) \qquad \text{(Equation 2-6)}$$

where:

| | | |
|---|---|---|
| 2,000 | = | monetary cost of dose received (\$/person-rem) |
| $D_R$ | = | TEDE rate to remediation workers (rem/hr) |
| $T_A$ | = | worker time required for remediation (person-hrs) |

#### 2.4.2.3 Determination Of Remediation Action Effectiveness

The remediation action effectiveness, F, is expressed in terms of the fraction of the residual radioactivity removed by the remediation action. It is determined by collecting and analyzing pre-remediation and post-remediation measurements in an area in which the remediation action is performed. A sufficient number of measurements are made to establish a consistent fraction.

### 2.4.3 ALARA EVALUATION

The ALARA evaluation compares residual radioactivity levels to calculated remediation levels for possible remediation actions. Where the level of residual radioactivity exceeds the remediation level, the remediation action is considered to have a net benefit. Therefore, it is considered cost effective and must be taken for the residual radioactivity to be considered ALARA. Conversely, if the concentration is less than the remediation level, the level of residual radioactivity is already considered ALARA and the remediation action is not required to be performed.

The ALARA evaluation is needed only to justify not taking a remediation action. If a decision has already been made to perform a given remediation action, there is no need to evaluate whether the action is necessary to meet the ALARA requirement. For example, if wiping down surfaces with loose radioactive contamination is a good practice that is applied regardless of radioactive contamination levels, then an ALARA evaluation is not required. For those remediation actions considered but not taken, the ALARA evaluation includes the levels of residual radioactivity above which those remediation actions would have been justified.

Remediation levels do not represent concentration limits that cannot be exceeded. Rather, they represent the threshold at which the given remediation action is taken. The ALARA requirement is met by performing the appropriate remediation action and not by being below a specified concentration after the action is taken. The ALARA evaluation ensures that efforts to remove residual radioactivity are commensurate with the level of risk the residual radioactivity poses.

## 2.5 RADIOLOGICAL IMPACTS OF DECOMMISSIONING ACTIVITIES

The decommissioning activities described herein are conducted under the auspices of the approved TNP Radiation Protection Program and Radioactive Waste Management Program. These programs continue to be implemented as described in Sections 3.2 and 3.3, respectively. The TNP Radiation Protection Program implements the regulatory requirements of 10 CFR 20 (Reference 2-9) through approved plant procedures established to maintain radiation exposures ALARA. The Radioactive Waste Management Program controls generation, characterization, processing, handling, shipping, and disposal of radioactive waste per approved TNP Radiation Protection Program, Process Control Program, and plant procedures.

### 2.5.1 OCCUPATIONAL EXPOSURE

Table 2-3 documents personnel exposure projections for various decommissioning and fuel storage activities. The total radiation exposure impact for decommissioning and spent fuel management is estimated in Table 2-3 to total approximately 551 person-rem. Originally projected to be approximately 591 person-rem, this value has been updated to reflect actual exposure savings for specific projects and revised estimates for normal plant operations and fuel transfer to the ISFSI. The estimates contained in Table 2-3 incorporate the following assumptions and bases:

1. Area dose rates are based on radiological surveys that have been adjusted to account for radioactive decay to the estimated start of decommissioning activities;

2. The projected exposure for decommissioning activities is based on TNP site information;

3. The exposure for removal of the steam generators and pressurizer reflects actual values;

4. Personnel radiation exposure during the Transition Period is estimated to be approximately 4 person-rem per year, excluding some dismantlement activities; and

5. Estimated personnel exposure due to the transfer of fuel to the ISFSI is approximately 2.9 person-rem for each of the estimated 34 casks, for a total of approximately 99 person-rem.

As of January 1, 1999, the actual total exposure for decommissioning activities was approximately 224 person-rem. Detailed exposure estimates and exposure controls for specific activities are developed during detailed planning per Radiation Protection Program procedures.

### 2.5.2 DECOMMISSIONING RADIOACTIVE WASTE PROJECTIONS

The radioactive waste management program (Section 3.3) is used to control the generation, processing, handling, shipping, and disposal of radioactive waste during decommissioning. Activated and contaminated systems, structures, and components represent the largest volume of low level radioactive waste expected to be generated during decommissioning. Other forms of waste generated during decommissioning include:

1. Contaminated water;
2. Used disposable protective clothing;
3. Expended abrasive and absorbent materials;
4. Expended resins and filters;
5. Contamination control materials (e.g., strippable coatings, plastic enclosures); and
6. Contaminated equipment used in the decommissioning process.

Table 2-4 provides projections of waste volumes for decommissioning. The waste volume projections are conservative estimates obtained from the decommissioning cost estimate, and include actual waste volume amounts for removal of the steam generators and pressurizer. As reflected in this table, approximately 343,162 $ft^3$ of low-level radioactive waste will be generated as a result of decommissioning and spent fuel management activities. As of January 1, 1999, an approximate volume of 181,076 $ft^3$ of low level radioactive waste had been shipped in 304 shipments.

Decommissioning planning at TNP incorporates the assumption that cost-effective waste volume reduction methods are limited. It also assumes significantly contaminated or activated materials are sent directly to a disposal facility. However, alternative processing methods may be evaluated during decommissioning.

## 2.6 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

### 2.6.1 DECOMMISSIONING ORGANIZATION

The TNP organization (General Manager and above) is shown in Figure 2-10. The Trojan Site Executive and Plant General Manager has corporate responsibility for overall nuclear safety and decommissioning activities at TNP. The General Manager, Trojan Plant is responsible for operations, maintenance, personnel/radiation protection, including the ALARA and onsite safety and hazardous materials programs, and emergency preparedness. Reporting to the General Manager, Trojan Plant are the General Manager, Nuclear Oversight; General Manager, Engineering/Decommissioning; General Manager, Plant Support and Technical Functions; Manager, Operations; Manager, Personnel/Radiation Protection; and Manager, Maintenance. The Independent Review and Audit Committee (IRAC) reports to and advises the General Manager, Trojan Plant.

The General Manager, Nuclear Oversight, is responsible for quality assurance and quality control. The Nuclear Oversight Department is independent of other departments performing quality-related activities. The General Manager, Engineering/Decommissioning, is responsible for decommissioning planning, engineering, and cost control. The General Manager, Plant Support and Technical Functions, is responsible for purchasing, nuclear security, licensing, and training.

Experienced and knowledgeable personnel will be utilized to perform the technical and administrative tasks required during TNP decommissioning. To the extent practicable, the decommissioning organization will include staff previously employed at TNP to capitalize on their knowledge and familiarity with the facility. Contractors may be used to provide specialized services, or to supplement the facility staff, when warranted.

Each member of the facility staff will meet or exceed the minimum qualifications of ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel" (Reference 2-10) for comparable positions, except for the Manager, Radiation Protection, who shall meet or exceed the qualifications of Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants" (Reference 2-11).

### 2.6.2 REVIEWS AND AUDITS

The Independent Review and Audit Committee (IRAC) is responsible for reviews and audits in accordance with the TNP Technical Specifications, Appendix A to Facility Operating (Possession Only) License NPF-1. The IRAC is responsible for advising the General Manager, Trojan Plant on matters relating to safe storage of irradiated fuel. This review and audit function is independent of the line organization responsibilities.

As specified in the TNP Technical Specifications, the Independent Review and Audit Committee will review and/or audit evaluations completed under the provisions of 10 CFR 50.59, special nuclear material control, radiation protection activities, radioactive waste controls, and reportable occurrences.

## 2.7 TRAINING PROGRAM

The TNP Training Program is designed to provide the necessary instruction to ensure that individuals have adequate knowledge and skills to perform their job functions safely. Training programs are conducted in accordance with appropriate plant procedures. Initial training programs prepare entry-level employees to assume their assigned tasks; retraining programs enable employees to maintain their proficiency.

Individuals requiring access to TNP, including radiologically controlled areas, will receive training commensurate with the potential hazards to which they will be exposed. This applies to PGE employees, contractors, and visitors.

Training applicable to specific activities, tasks, and conditions will be developed or discontinued, as appropriate, as decommissioning progresses. Since decommissioning activities will occur while fuel remains stored at TNP, PGE will retain those elements of the TNP Training Program necessary to ensure safe fuel storage and handling, including protection of workers from hazards associated with such activities.

### 2.7.1 PROGRAMS

TNP will maintain training and retraining programs throughout decommissioning as necessary to provide the TNP staff with the specialized training and technical skills necessary to maintain the plant in a safe condition.

#### 2.7.1.1 General Employee Training

Individuals requiring unescorted access to the TNP Industrial Area will receive General Employee Training, which includes the following representative topics:

1. TNP introduction;
2. Radiological protection fundamentals;
3. Emergency response plan;
4. Plant safety;
5. Fire protection;
6. Chemical safety;
7. Security;
8. Quality assurance; and
9. Corporate drug awareness.

Individuals requiring unescorted access to the protected area will receive additional training, as necessary, in fitness for duty and TNP site-specific radiation protection. Individuals requiring unescorted access to RCAs will receive additional training, as necessary, in TNP site-specific radiation protection. Escorted individuals will receive appropriate training for the areas they will be entering.

General Employee Retraining is conducted annually and includes subject material from General Employee Training. General Employee Training and Retraining Programs consist of lectures and demonstrations that may be augmented with selected audiovisual aids. The content of the course may be revised, as needed, during decommissioning.

#### 2.7.1.2 Certified Fuel Handler Training

Training for operators is described in topical report PGE-1057, "Certified Fuel Handler Training Program" (Reference 2-12). The Certified Fuel Handler Training Program is based on a systems approach to training. The program ensures that staff members are adequately trained to perform activities that support the proper handling, storing, and cooling of the fuel.

#### 2.7.1.3 Work-Specific Training

Work specific training for selected activities will include the appropriate level of training in decontamination and other decommissioning activities, health physics, and the use and maintenance of radiation surveillance and monitoring equipment. Cognizant managers will ensure that employees and contractors who perform decommissioning activities are properly trained, qualified, and proficient in the principles and techniques of activities necessary to perform their assigned tasks, in accordance with approved procedures.

### 2.7.2 TRAINING RECORDS

Records of training on quality-related activities will be maintained as quality assurance records.

### 2.7.3 INSTRUCTOR QUALIFICATION

Training may be conducted by PGE employees or contractors. The background, qualifications, and experience of instructors will be appropriate for the subject matter. Instructor qualifications are administratively controlled by approved procedures. Instructors are responsible for ensuring training materials are technically accurate and applicable prior to their use or issuance. Instructors are also responsible for documenting training sessions.

## 2.8 CONTRACTOR ASSISTANCE

### 2.8.1 CONTRACTOR SCOPE OF WORK

During decommissioning PGE may use contractors to provide specialized services or to supplement the facility staff when warranted. Tasks where contractors may be used to provide support during decommissioning include, but are not limited to, the following:

1. Processing, packaging, transportation, and disposal of radioactive material;
2. Decontamination and recycling of radioactively contaminated material;
3. Radiation protection staff augmentation;
4. Design and fabrication of special dismantling equipment;
5. Engineering and design services such as heavy loads management and transportation engineering; and
6. Dismantlement and demolition of components, systems, and structures.

### 2.8.2 CONTRACTOR ADMINISTRATIVE CONTROLS

PGE has the responsibility for contractor control, including the contractor's effectiveness in performing to bid specifications. PGE will provide the necessary management oversight to ensure that tasks performed by the contractors are in full compliance with the TNP Nuclear Quality Assurance Program (Reference 2-13), the purchase agreement, and applicable regulatory requirements.

### 2.8.3 CONTRACTOR QUALIFICATIONS AND EXPERIENCE

#### 2.8.3.1 General

Potential contractors for activities will be required to supply their qualifications as part of bid specifications. These qualifications will be evaluated and reviewed for:

1. Demonstrated experience in providing services on similar projects;
2. Cost and schedule compliance;
3. Technical and operational capability; and
4. Ability to meet regulatory requirements.

#### 2.8.3.2 TLG Services, Inc.

TLG Services, Inc. was contracted to perform the TNP Activation Analysis for use in site characterization and Decommissioning Cost Analysis to support the development of the Decommissioning Plan.

TLG Services, Inc. is an engineering firm with extensive experience in performing decommissioning cost estimates and other aspects of nuclear facility decommissioning. TLG Services Inc.'s experience includes:

1. More than 60 utility and government sponsored decommissioning studies for more than 90 units; providing costs, occupational exposure, and waste generation estimates;

2. Development of NUREG/CR-3587, "Identification and Evaluation of Facilitation Techniques for Decommissioning Light Water Power Reactors," June 1986;

3. Development of industry-accepted reference manual, AIF/NESP-036, "Guidelines to Producing Decommissioning Cost Estimates;"

4. Active participation on four Industry Standards Committees associated with decommissioning; and

5. Performance of activation analyses for five nuclear facilities.

## 2.9 REFERENCES FOR SECTION 2


2-1 NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," August 1988.

2-2 Code of Federal Regulations, Title 10, Part 50.82, "Application for Termination of License," August 28, 1996.

2-3 NRC Letter, E. W. Brach to S. M. Quennoz, "Issuance of Materials License SNM-2509 for the Trojan Independent Spent Fuel Storage Installation (TAC Nos. L22102 and L22834)," March 31, 1999.

2-4 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use."

2-5 Portland General Electric Topical Report PGE-1021, "Offsite Dose Calculation Manual," Revision 17.

2-6 Trojan Nuclear Plant Facility Operating License No. NPF-1, Amendment 192, September 22, 1993.

2-7 Portland General Electric Topical Report PGE-1074, "Trojan Nuclear Plant Final Survey Report for the ISFSI Site," Revision 0.

2-8 Draft Regulatory Guide DG-4006, "Demonstrating Compliance with the Radiological Criteria for License Termination," August 1998.

2-9 Code of Federal Regulations, Title 10, Part 20, "Standards for Protection Against Radiation."

2-10 American National Standards Institute Committee N18 Design Standard ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel."

2-11 Regulatory Guide 1.8, "Qualification and Training of Personnel for Nuclear Power Plants," Revision 2, April 1987.

2-12 Portland General Electric Topical Report PGE-1057, "Certified Fuel Handler Training Program."

2-13 Portland General Electric Topical Report PGE-8010, "PGE Nuclear Quality Assurance Program for Trojan Nuclear Plant."

Table 2-1

**Status of Major TNP Systems, Structures, and Components as of January 1999**

| System, Structure or Component | Required to Support Fuel Storage | Status |
|---|---|---|
| Reactor Coolant System | NO | Removed. |
| Reactor Vessel Internals | NO | Preparations for removal in 1999 are underway. |
| Reactor Vessel | NO | Preparations for removal in 1999 are underway. |
| Steam Generators | NO | Removed. |
| Reactor Coolant Pumps | NO | Removed. |
| Pressurizer and Pressurizer Relief Tank | NO | Removed. |
| Chemical and Volume Control System | YES | Partially removed. Boric Acid Batch Tank remains in service to support boration of the SFP if required. |
| Safety Injection System | NO | Removed. |
| Residual Heat Removal System | NO | Removed. |
| Containment Spray System | NO | Removed. |
| Component Cooling Water System | NO | Partially removed. |
| Service Water System | YES | Partially removed. Portion of system remains in service to supply dilution flow for liquid discharges and alternate SFP make-up. |
| SFP and Fuel Handling Equipment | YES | Partially removed; SFP portion in service. |
| SFP Cooling and Demineralizer System (Original) | NO | Removed. |
| Modular SFP Cooling and Cleanup System | YES | In service. This system replaced the original system. |
| Condensate Demineralizers | NO | Partially removed. |
| Steam Generator Blowdown System | NO | Partially removed. Some piping is included in embedded piping scope. |
| Primary Makeup Water System | NO | Partially removed. Some system piping is included in embedded piping scope. |
| Refueling Water Storage Tank | NO | To be removed in 1999. Some related piping is included in embedded piping scope. |

Table 2-1

**Status of Major TNP Systems, Structures, and Components as of January 1999**

| System, Structure or Component | Required to Support Fuel Storage | Status |
|---|---|---|
| Plant Effluent System | NO | Partially removed. System decontamination activities are in progress in conjunction with Turbine Building activities. System includes a portion of piping in embedded piping scope. |
| Containment Ventilation Systems | NO | Partially removed; in service. |
| Hydrogen Recombiners | NO | Removed. |
| Fuel Building Ventilation System | YES | Partially removed; in service. |
| Auxiliary Building Ventilation System | YES | Partially removed; in service. |
| Condensate Demineralizer Building Ventilation System | NO | In service. Supports use of building as radwaste storage and processing facility. |
| Instrument and Service Air Systems | YES | Partially removed; in service. Air currently supplies pressure to SFP gates for sealing. |
| Gaseous Radioactive Waste System | NO | Partially removed. |
| Solid Radioactive Waste System | NO | Partially removed. |
| Liquid Radioactive Waste System | YES | Partially removed. Portion of remaining system is used for processing liquid radwaste prior to discharge from plant. System drain piping and some process piping is included in embedded piping scope. |
| Radiation Monitoring System | YES | Partially removed. Portions of system remains in service for effluent monitoring of liquid and ventilation exhausts and criticality monitors at SFP area. |
| Process Sampling System | NO | Partially removed. |
| Fire Protection System | YES | Partially removed. Portions of system remain in service for fire detection and suppression and alternate SFP make-up. |
| Electrical Systems | YES | Partially removed. Portions of the systems remain in service to support ongoing plant operation and decommissioning activities. |
| Containment Building | NO | In service. Most equipment has been removed with exception of Reactor Vessel and Internals. |

Table 2-1

**Status of Major TNP Systems, Structures, and Components as of January 1999**

| System, Structure or Component | Required to Support Fuel Storage | Status |
|---|---|---|
| Auxiliary Building (Including the Pipe Facade) | YES | In service. Part of seismically qualified Control-Auxiliary-Fuel Building Structure |
| Fuel Building | YES | In service. Part of seismically qualified Control-Auxiliary-Fuel Building Structure. |
| Main Steam Support Structure | NO | Equipment removed. Decontamination activities are in progress. |
| Condensate Demineralizer Building | NO | In service. Used for temporary radwaste storage and processing of low level radioactive waste. |
| Steam Generator Blowdown Building | NO | Equipment removed. Building decontaminated |
| Radwaste Annex | NO | In service. |
| Wright-Schuchart-Harbor (WSH) Warehouse | NO | NRC approved final survey to remove from 10 CFR 50 licensed area; Currently used for ISFSI preparation activities. |
| Turbine Building | NO | Affected equipment has been removed. Decontamination of affected areas is in progress. Contains a portion of drain and process piping included in embedded piping scope. |

Table 2-2

**Major Components Removed (By Year)**

1996:

Reactor Coolant Pumps and Motors - (4)
RCS Piping (up to bioshield wall)
RCS and Steam Generator support structural steel
Containment Main Steam and Feedwater Piping and Supports (all B/C loops; partial A/D loops)
A Residual Heat Removal Heat Exchanger
B Residual Heat Removal Heat Exchanger
Positive Displacement Charging Pump/Motor
A Centrifugal Charging Pump Motor
B Centrifugal Charging Pump Motor
A Safety Injection Pump/Motor
B Safety Injection Pump/Motor
A Containment Spray Pump/Motor
B Containment Spray Pump/Motor
B Component Cooling Water Pump/Motor (Note: replaced with smaller pump)
Condensate Demineralizer vessels
Decontamination Shop Equipment
Portions of Outside Buildings (WSH warehouse, Maintenance Shop)
Low Level Radwaste Storage Building (Replaced by Condensate Demin Bldg)

Table 2-2

**Major Components Removed (By Year)**

1997:

A/B Residual Heat Removal (RHR) Pumps/Motors
A/C Service Water Booster Pumps/Motors
Letdown Heat Exchanger
A/B Boric Acid Evaporator skids
Clean Radioactive Waste Evaporator skid
Seal Water Heat Exchanger
Dirty Waste Monitor Tank
Auxiliary Building Drain Tank and Pumps
Waste Concentrate Hold Tanks and Pump
CVCS Concentrates Hold Tank and Pumps
Tiger Lock Storage Tank
Steam Generator Blowdown Tank and Pump
Boric Acid Storage Tanks and Pumps
Boric Acid Blender and Chemical Addition Tank
Containment Post-Accident Sampling System (PASS)
Containment Hydrogen Analysis System (CHAS)
Hold-Up Tank Recirculation Pump
Gas Stripper Feed Pumps
Spent Fuel Storage Rack (1)
New Fuel Storage Racks
Decontamination Area Ventilation Supply Cooling System (AB-5)
B/C Safety Injection Accumulators
Control Rod Drive Mechanism Patch Panels
Reactor Vessel Neutron Water Bag Racks
Flux Thimble Drive Units and Detectors
Manipulator Crane
Portions of Containment Ventilation Systems CS-1, CS-2, CS-3, CS-4, CS-5, CS-6
Portions of Activated Concrete around Reactor Vessel
Portions of Containment Wall for new 10'x10' roll-up access door
Bioshield Structural Steel and Equipment Supports
RCP Oil Collection System
Remaining Containment Main Steam and Feedwater Piping and Supports

Table 2-2

**Major Components Removed (By Year)**

1998:

Service Water Booster Pumps/Motors (B/D)
Sodium Hydroxide Tank
Boron Injection Tank
Clean Waste Receiver Tank Pumps / Motors
Clean Waste Receiver Tanks (A/B)
Chemical and Volume Control System (CVCS) Monitor Tanks and Pumps
Primary Makeup Water (PMW) Pumps
Reactor Coolant Drain Tank (RCDT) pumps
Chemical Waste Drain Tank and Pumps
Waste Gas Decay Tanks (A/B/C/D)
Waste Gas Surge Tank
Waste Gas Compressors (A/B)
CVCS Volume Control Tank
Boric Acid Heat Trace Panels
CVCS Evaporator Condensate Demineralizers (T-220 A/B)
CVCS Evaporator Feed Demineralizers (T-219 A/B/C)
CVCS Mixed Bed Demineralizers (T-210 A/B)
CVCS Cation Bed Demineralizer (T-211)
SFP Demineralizer (T-224)
SGBD Demineralizers (T-316 A/B)
Reactor Coolant Filter (F-204)
Seal Injection Filters (F-210A/B)
Seal Water Return Filter (F-209)
Reactor Coolant Drain Filter (F-307)
CVCS Evaporator Concentrates Filter (F-208)
CVCS Evaporator Condensate Filter (F-207)
CVCS Ion Exchange Filter (F-206)
Resin Backflush Filter (F-305)
SFP Demin Pre-Filter (F-201)
SFP Demin After-Filter (F-211)
Clean Waste Filter (F-304)
SGBD Filter (F-306)
Automatic Gas Analyzer
RCS Post Accident Sampling System Panel
S/G Sample Panel
Steam Jet Air Ejector (SJAE) associated vessels and piping

Table 2-2

**Major Components Removed (By Year)**

1998 (continued):

SJAE Effluent Monitor (PERM-6)
CVCS Holdup Tanks (A/B/C)
Primary Makeup Water Tank
Contaminated Piping in Various Pipe Chases
Component Cooling Water (CCW) Heat Exchangers (A/B)
SFP Heat Exchangers (A/B)
SFP Cooling Pumps / Motors (A/B)
SFP Purification Pump / Motor
Reactor Coolant Drain Tank
Containment Air Coolers (8)
Hydrogen Mixing Fans
Reactor Cavity Cooling Fans
Reactor Cavity Exhaust Fans
Containment Sump Pumps (2)
Pressurizer Relief Tank
Containment Motor Control Centers (Containment has temporary power only)
Reactor Coolant Pump fans for "A" and "D" RCP's
Component Cooling Water and Containment Spray Piping/Supports
Equipment Hatch Trolley Rail System
Control Rod Drive Mechanisms
Electrical Penetration Assemblies (partial)
Containment Piping Penetrations (partial)

Table 2-2

**Major Components Removed (By Year)**

1999:

Hot Sample Sink
Liquid Radioactive Waste Ion Exchangers (T-326A/B)
Dirty Waste Monitor Tank Pumps/Motors
A/C/D Component Cooling Water (CCW) Pumps/Motors
A/B/C CCW Pump Area Coolers
A/B CCW Makeup Water Pumps
B/D Service Water Booster Pump Area Cooler
Auxiliary Building Sump Pumps/Motors
Auxiliary Building Passageway Pump/Motor
Dirty Waste Drain Tank Pumps/Motors
Dirty Waste Drain Tank
Spent Fuel Pool Skimmer Pump
Spent Fuel Pool Skimmer Filter
Auxiliary Radioactive Waste Control Panels (Panels C-150 and C-151)
Solid Radioactive Waste System Process Module
Drum Compactor
Spent Resin Storage Tank
Spent Resin Transfer Pump and Resin Transfer Piping
Filter Handling Vehicle
Radioactive Waste Annex HVAC Equipment
Refueling Water Storage Tank
Resin Fill Tank
Portions of Outside Contaminated Embedded Piping (Tank Farm Area)
Portions of Exposed Piping in Contaminated Embedded Piping Scope
Spent Fuel Storage Rack (1)
Turbine Building Sump Pumps and Exposed Discharge Piping
Containment Roll-Up Door (Used for Reactor Vessel and Steam Generator Removal)
Reactor Cavity Sump Pump
Reactor Vessel and Internals
Remaining Portions of RCS Piping
45' Elevation Containment Personnel Airlock and Vestibule
Remaining Electrical Penetration Assemblies
Remaining Containment Piping Penetrations
Containment Recirculation Sump Piping, Screens, Racks, and Supporting Structure
Portions of Containment Building HVAC Systems
Portions of Auxiliary/Fuel Building HVAC Systems
Portions of Auxiliary/Fuel Building Instrument and Service Air Systems
Portions of Vent Collection Header Piping

Table 2-2

**Major Components Removed (By Year)**

1999 (continued):

Portions of Auxiliary/Fuel Building Fire Protection System
Portions of Auxiliary/Fuel Building Electrical Systems
Portions of Containment Internal Structural Materials
Portions of Auxiliary/Fuel Building Structure and Building Surfaces
Portions of Containment Building Structure and Building Surfaces

Table 2-3

**Radiation Exposure Projections**

| Activity | Exposure (person-rem) |
|---|---|
| Steam generators and pressurizer removal | 54[a] |
| Reactor vessel and internals removal | 67 |
| Dismantlement | |
| Nuclear steam supply system | 51 |
| Spent fuel racks | 19 |
| Balance of plant systems | 165 |
| Structures | 46 |
| Miscellaneous | 20 |
| Subtotal | 301 |
| Normal plant operations | 30 |
| Fuel transfer to ISFSI | 99 |
| Total | 551[b] |

[a] This value reflects actual total project exposures.
[b] This value represents a reduction in the original projection by approximately 40 person-rem.

Table 2-4

**Decommissioning Waste Classification and Volume Projections**

| Item | Class A Burial Volume ($ft^3$) | Class B Burial Volume ($ft^3$) | Class C Burial Volume ($ft^3$) |
|---|---|---|---|
| Reactor coolant piping | 5,894 | 0 | 0 |
| Pressurizer relief tank | 625 | 0 | 0 |
| Reactor coolant pumps and motors | 3,044 | 0 | 0 |
| CRDMs/incore instrumentation/service structure removal | 1,726 | 0 | 0 |
| Steam Generators and Pressurizer (Large Component Removal) | 57,800[a] | 0 | 0 |
| Reactor vessel and internals | 0 | 0 | 8,341 |
| Spent fuel racks | 16,551 | 0 | 0 |
| 120 V ac preferred instrument ac | 1,400 | 0 | 0 |
| 125 V dc power | 175 | 0 | 0 |
| 4.16 kV ac power | 726 | 0 | 0 |
| 480 V ac auxiliary load center | 5,080 | 0 | 0 |
| 480 V ac motor control center | 8,426 | 0 | 0 |
| Chemical and volume control | 10,968 | 0 | 0 |
| Clean radwaste | 5,423 | 0 | 0 |
| Containment Building penetrations | 188 | 0 | 0 |
| Control rod drive | 85 | 0 | 0 |
| Dirty radwaste | 1,613 | 0 | 0 |
| Electric heat tracing | 164 | 0 | 0 |
| Electrical (Cables/Tray/Conduit) | 60,139 | 0 | 0 |
| Fuel handling system | 339 | 0 | 0 |
| Fuel pool cooling and demineralizer | 4,632 | 0 | 0 |
| Fuel and Auxiliary Building heating, ventilation, and air conditioning (HVAC) | 3,661 | 0 | 0 |

Table 2-4

**Decommissioning Waste Classification and Volume Projections**

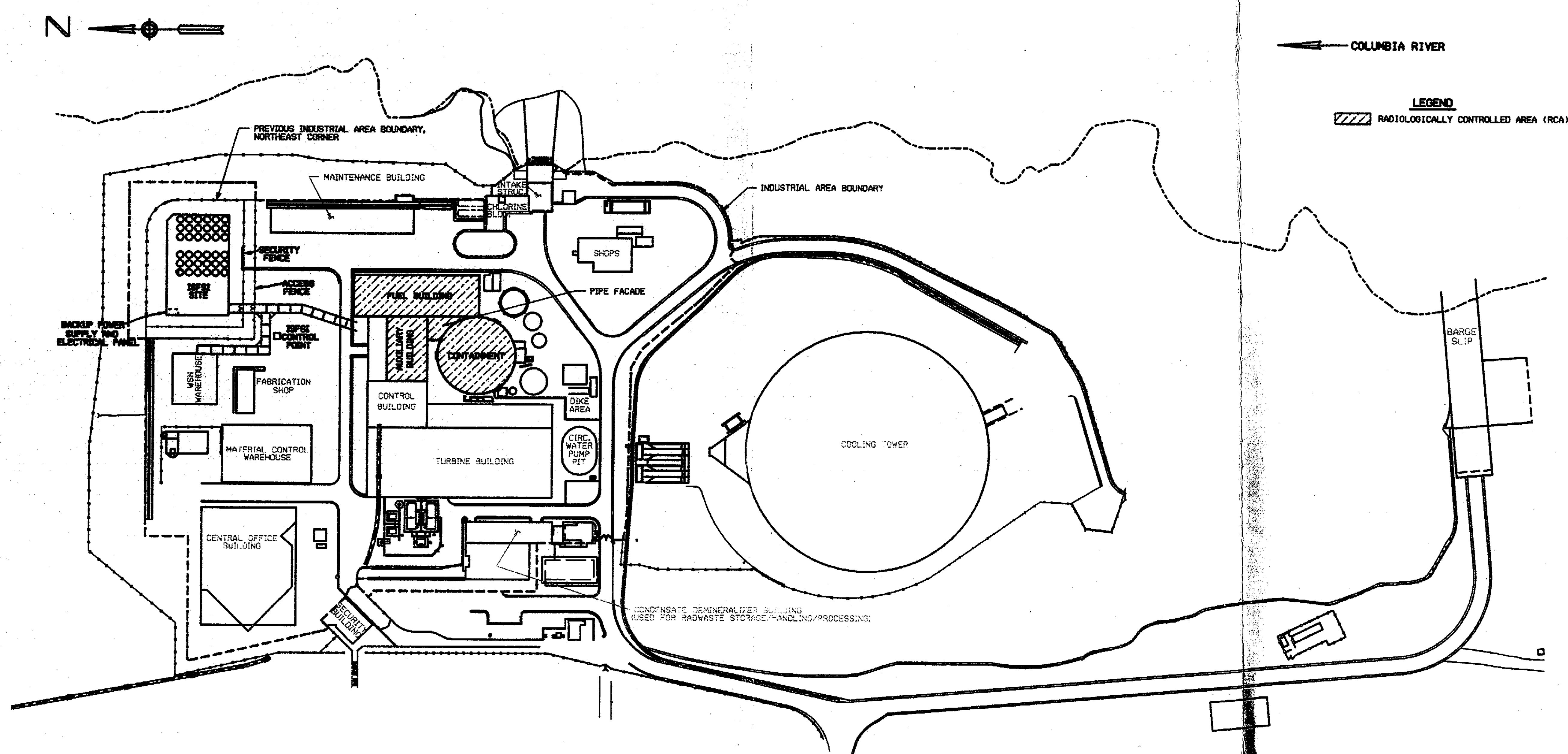
| Item | Class A Burial Volume ($ft^3$) | Class B Burial Volume ($ft^3$) | Class C Burial Volume ($ft^3$) |
|---|---|---|---|
| Gaseous radwaste | 2,529 | 0 | 0 |
| HVAC | 6,635 | 0 | 0 |
| Hydrogen recombiners | 576 | 0 | 0 |
| Integrated leak rate test instrument line | 106 | 0 | 0 |
| Instrument and service air | 1,327 | 0 | 0 |
| Lighting panel supply | 997 | 0 | 0 |
| Miscellaneous components | 1,936 | 0 | 0 |
| Miscellaneous reactor coolant | 3,418 | 0 | 0 |
| Nuclear instrumentation | 193 | 0 | 0 |
| Oily waste and storm drains | 1,882 | 0 | 0 |
| Containment HVAC | 18,869 | 0 | 0 |
| Primary makeup water | 3,615 | 0 | 0 |
| Primary sampling | 114 | 0 | 0 |
| Radiation monitoring | 134 | 0 | 0 |
| Reactor nonnuclear instruments | 245 | 0 | 0 |
| Reactor vessel system | 116 | 0 | 0 |
| Residual heat removal | 7,649 | 0 | 0 |
| Safety injection system | 7,149 | 0 | 0 |
| Solid radwaste | 370 | 0 | 0 |
| Spent fuel pool | 754 | 0 | 0 |
| Steam generator system | 3,562 | 0 | 0 |
| Turbine Building sump pumps and miscellaneous | 639 | 0 | 0 |
| Component cooling water | 6,115 | 0 | 0 |

Table 2-4

**Decommissioning Waste Classification and Volume Projections**

| Item | Class A Burial Volume ($ft^3$) | Class B Burial Volume ($ft^3$) | Class C Burial Volume ($ft^3$) |
|---|---|---|---|
| Condensate demineralizer | 2,262 | 0 | 0 |
| Discharge and dilution | 3,834 | 0 | 0 |
| Containment spray | 1,563 | 0 | 0 |
| Containment Building | 50,901[b] | 0 | 0 |
| Auxiliary Building | 2,650 | 0 | 0 |
| Fuel Building | 4,711 | 0 | 0 |
| Main steam supply system and electrical penetration area | 629 | 0 | 0 |
| Turbine Building | 1,054 | 0 | 0 |
| Process liquid radwaste | 0 | 3,686 | 0 |
| Disposal of dry active waste generated | 5,942 | 0 | 0 |
| Total | 331,135 | 3,686 | 8,341 |

[a] This value reflects actual burial volumes for the large component removal project.
[b] The projected burial volume as Class A material is 50,901 $ft^3$. An additional 187,809 $ft^3$ of licensed material is projected to be sent offsite for processing.




* *Figure is representative of plant areas as of January 1999.*

*Trojan Nuclear Plant*

DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 2-1
Site Plan
Revision 9

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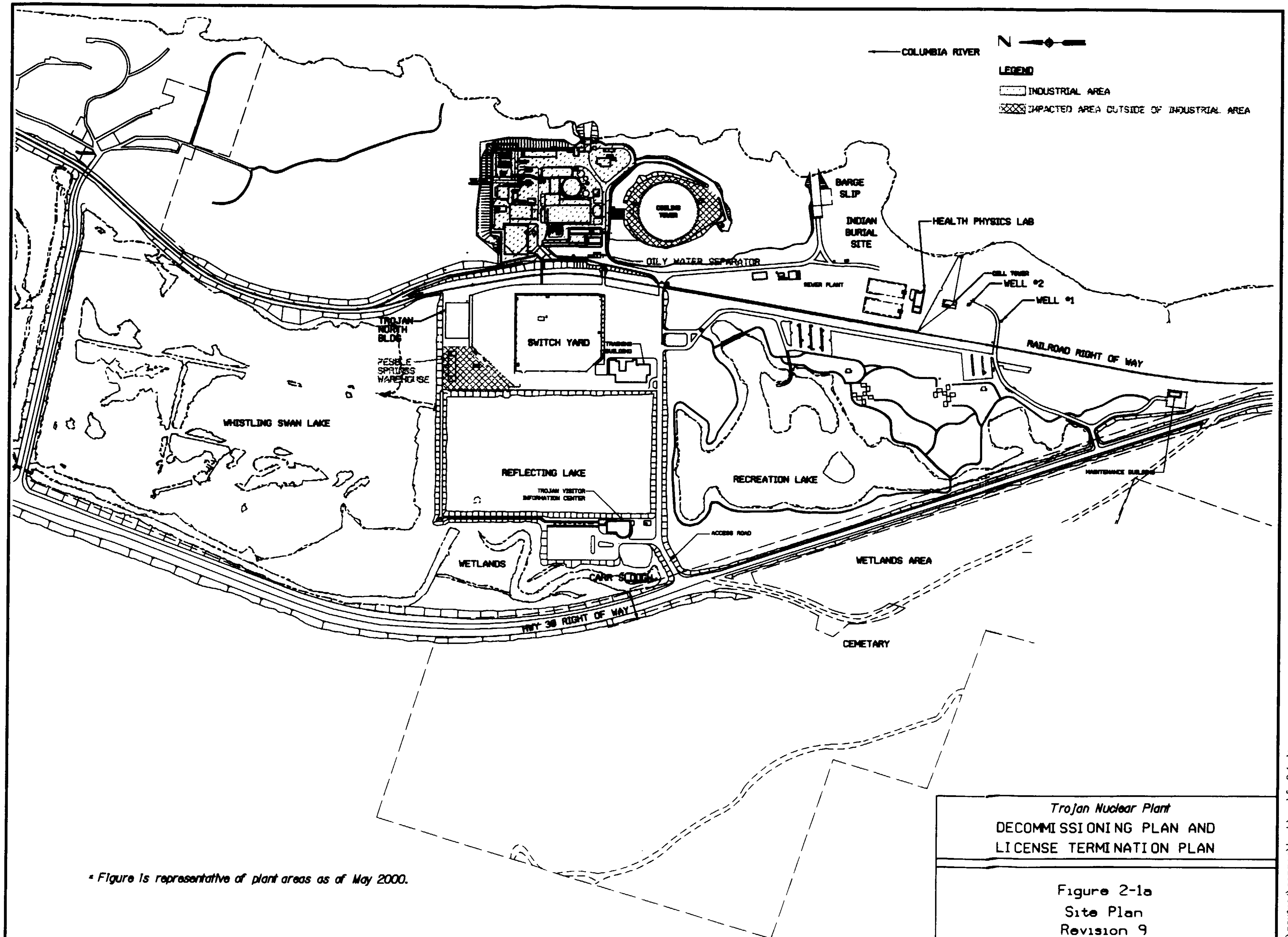
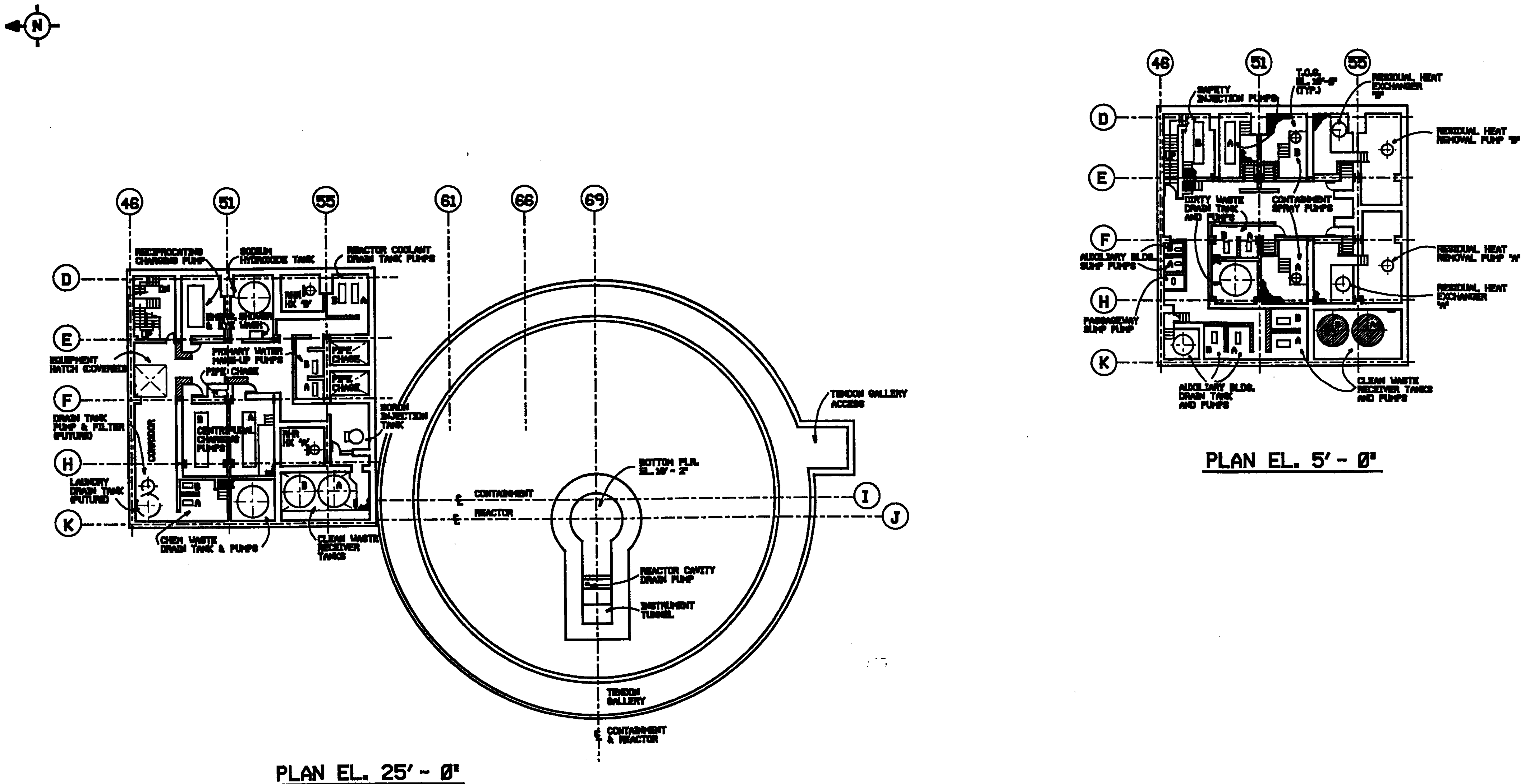



* *Figure is representative of plant areas as of May 2000.*

*Trojan Nuclear Plant*

DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 2-1a
Site Plan
Revision 9




* *Figure is representative of plant areas at the time of shut down. Equipment shown may have been deactivated or removed.*

Trojan Nuclear Plant
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 2-2
Containment And Auxiliary Buildings Plan Below Ground Floor

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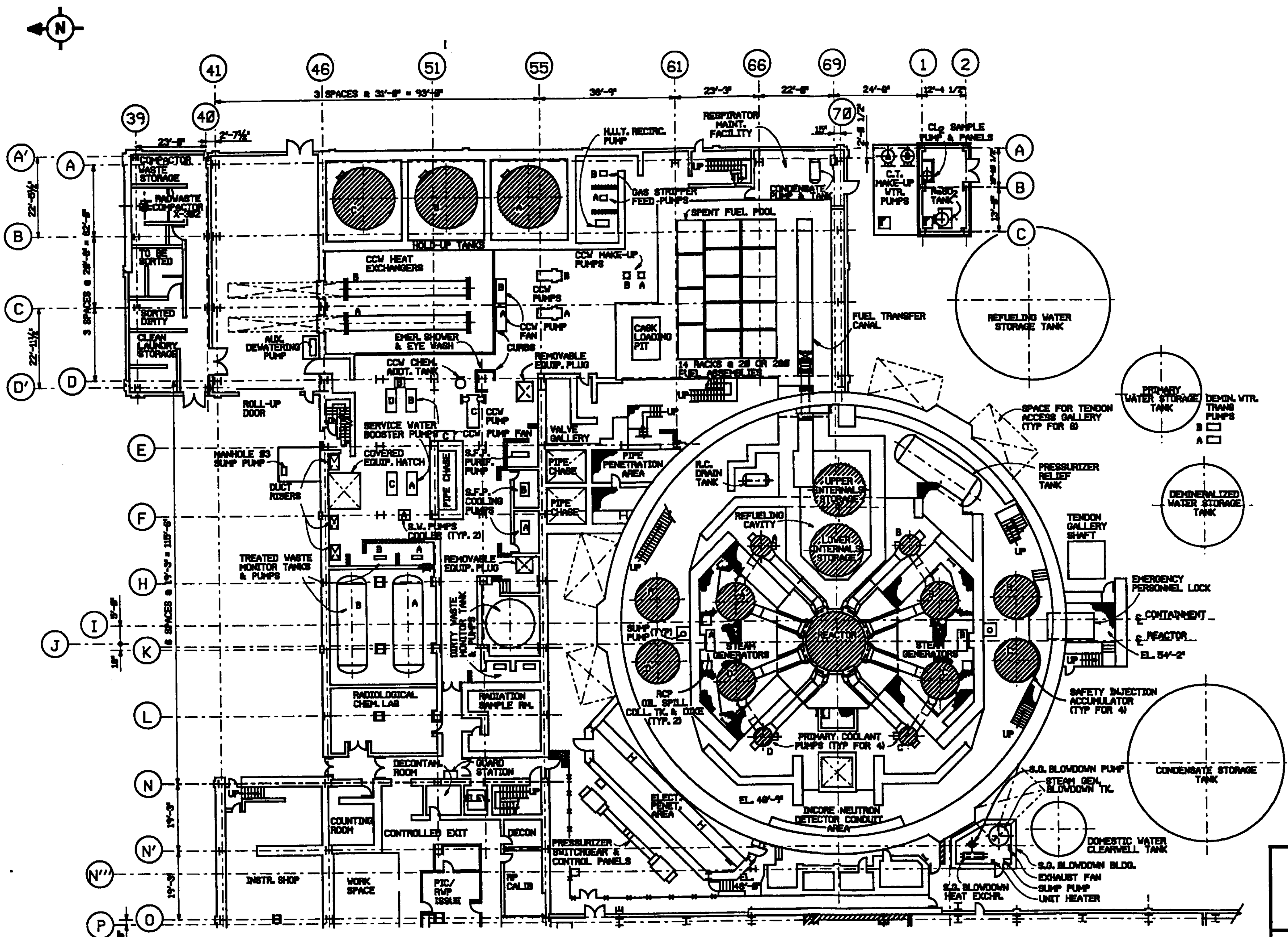


* *Figure is representative of plant areas at the time of shut down. Equipment shown may have been deactivated or removed.*

***Trojan Nuclear Plant***

DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 2-3
Containment And Auxiliary Buildings
Elevation 45 FT

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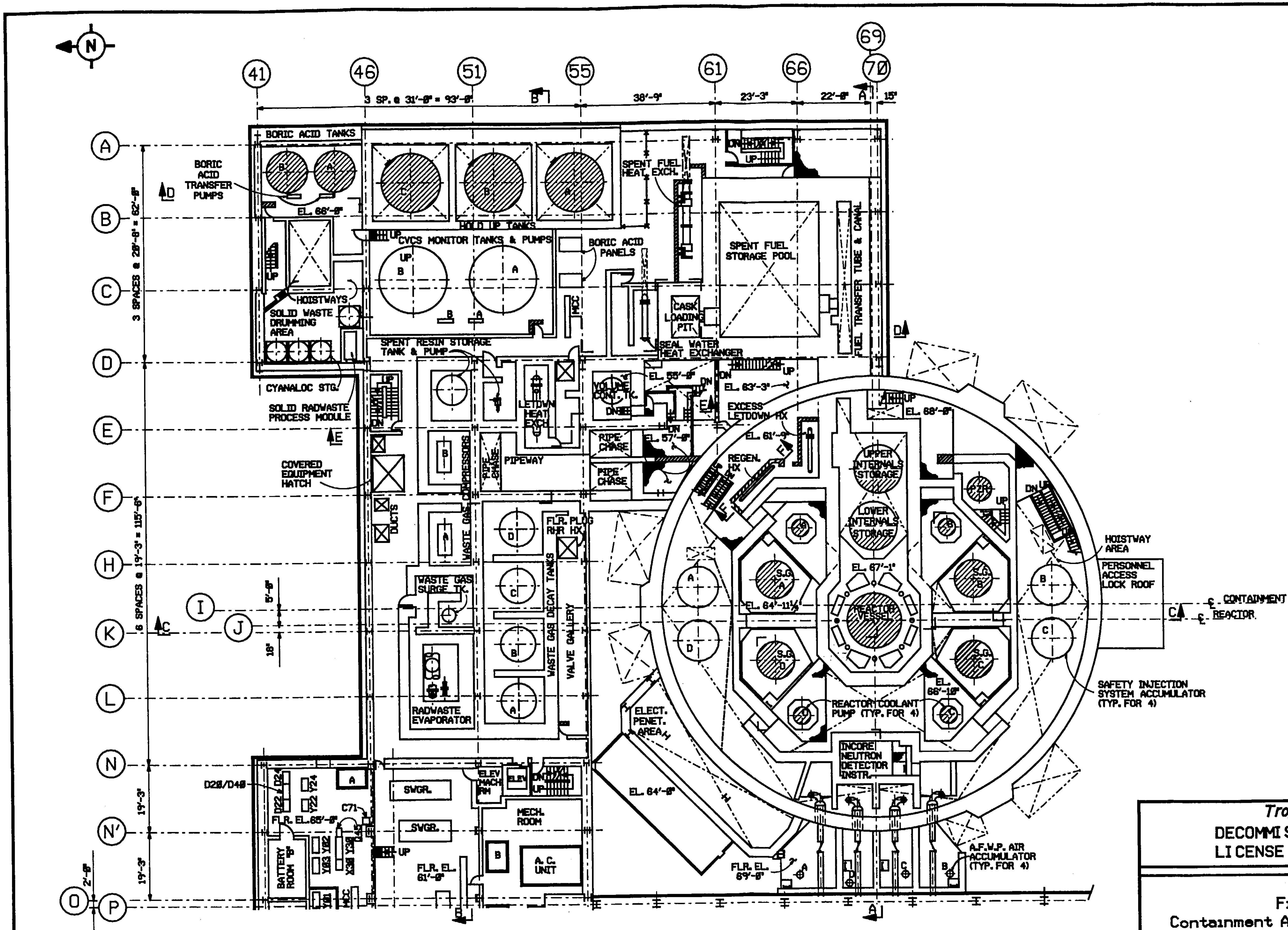


* Figure is representative of plant areas at the time of shut down. Equipment shown may have been deactivated or removed.

Figure 2-4
Containment And Auxiliary Buildings
Elevation 61 FT

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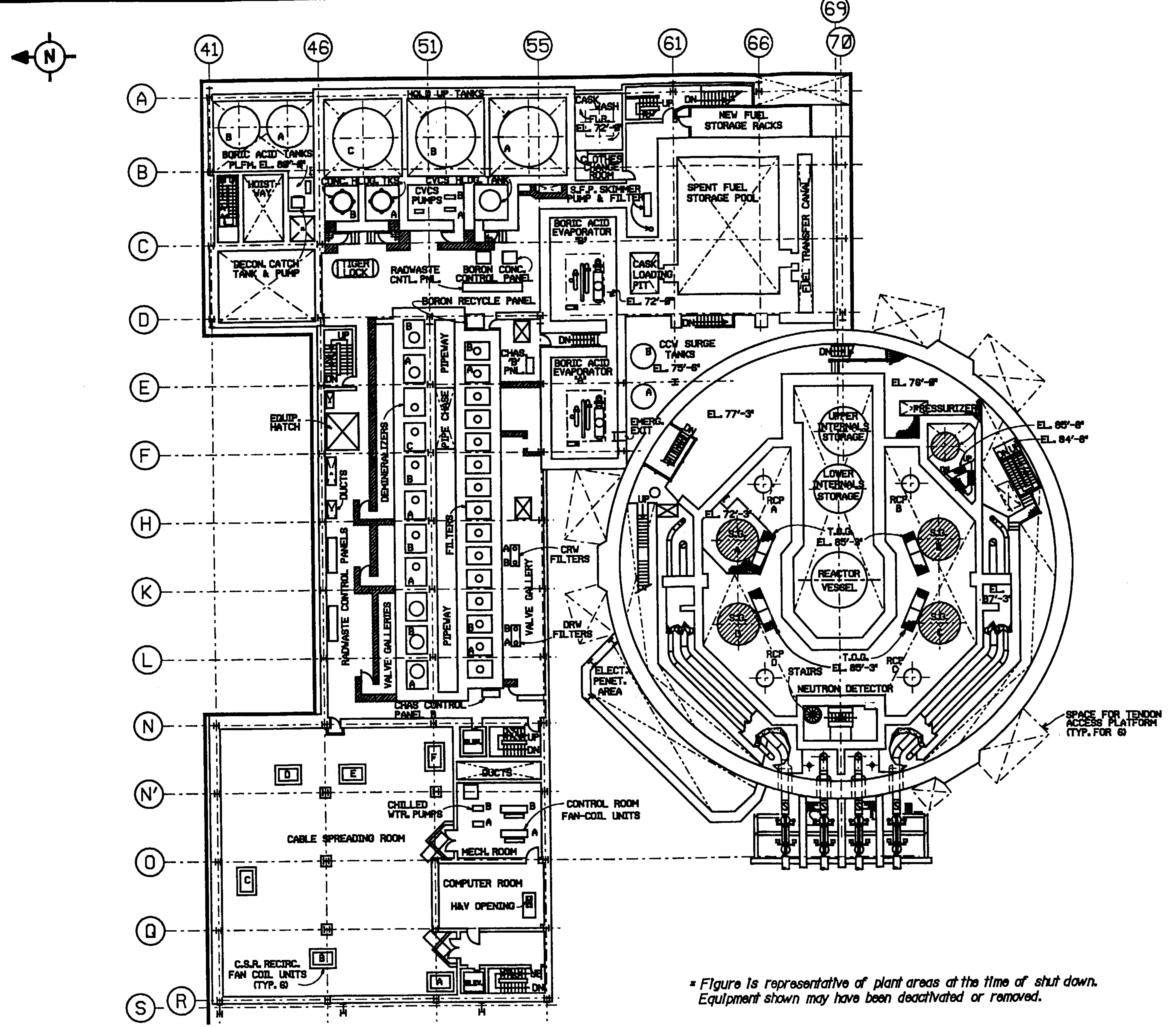


* Figure is representative of plant areas at the time of shut down. Equipment shown may have been deactivated or removed.

Trojan Nuclear Plant
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 2-5
Containment And Auxiliary Buildings
Elevation 77 FT

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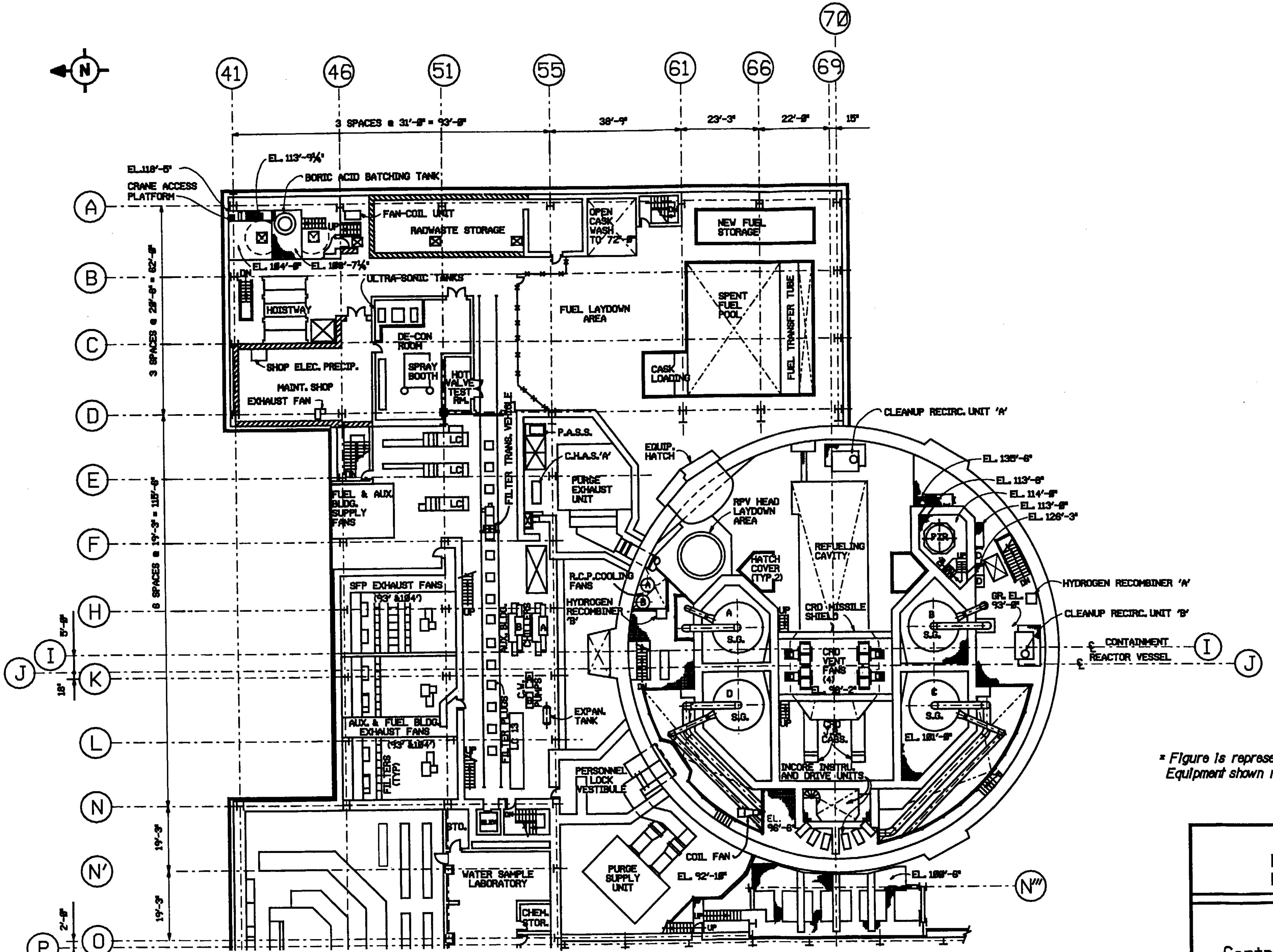


* Figure is representative of plant areas at the time of shut down. Equipment shown may have been deactivated or removed.

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 2-6
Containment And Auxiliary Buildings
Operating Floor And Above

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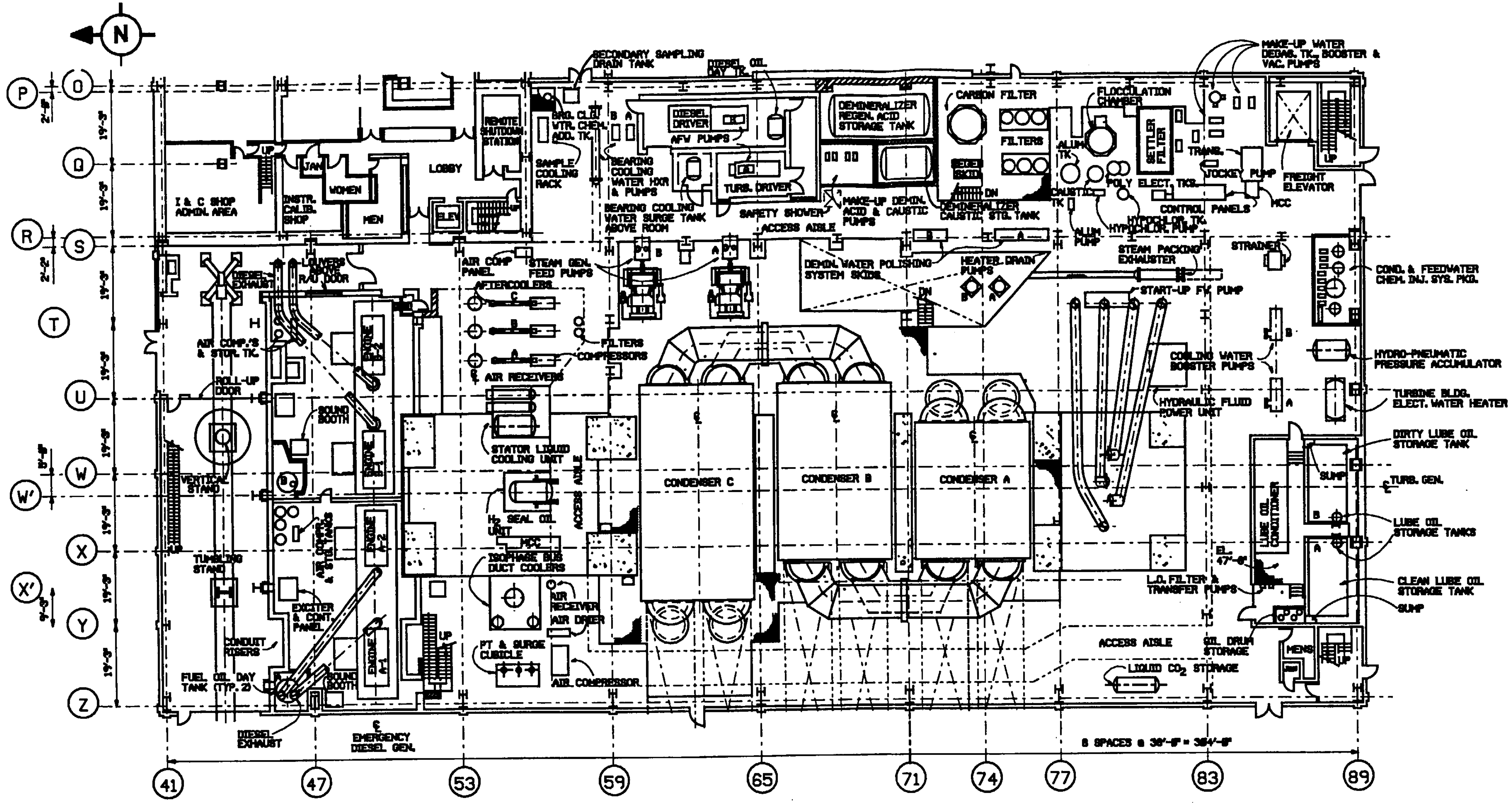


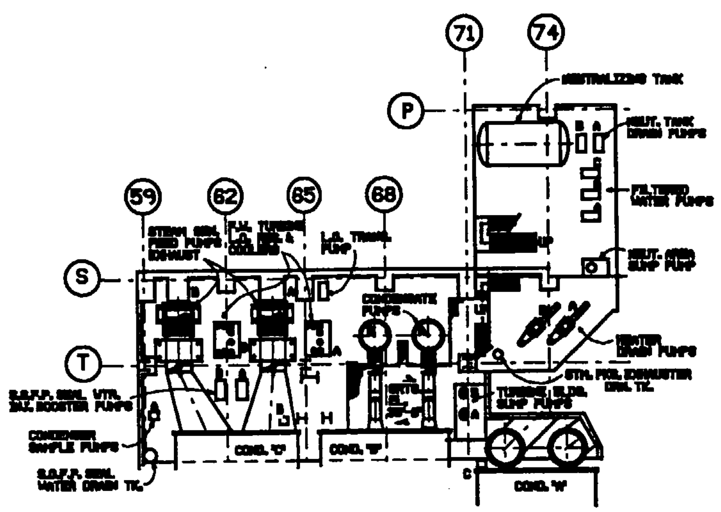


PARTIAL PLAN EL. 27'- 0"

* Figure is representative of plant areas at the time of shut down. Equipment shown may have been deactivated or removed.

*Trojan Nuclear Plant*

DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 2-7
Turbine Building
Elevation 27 FT and 45 FT

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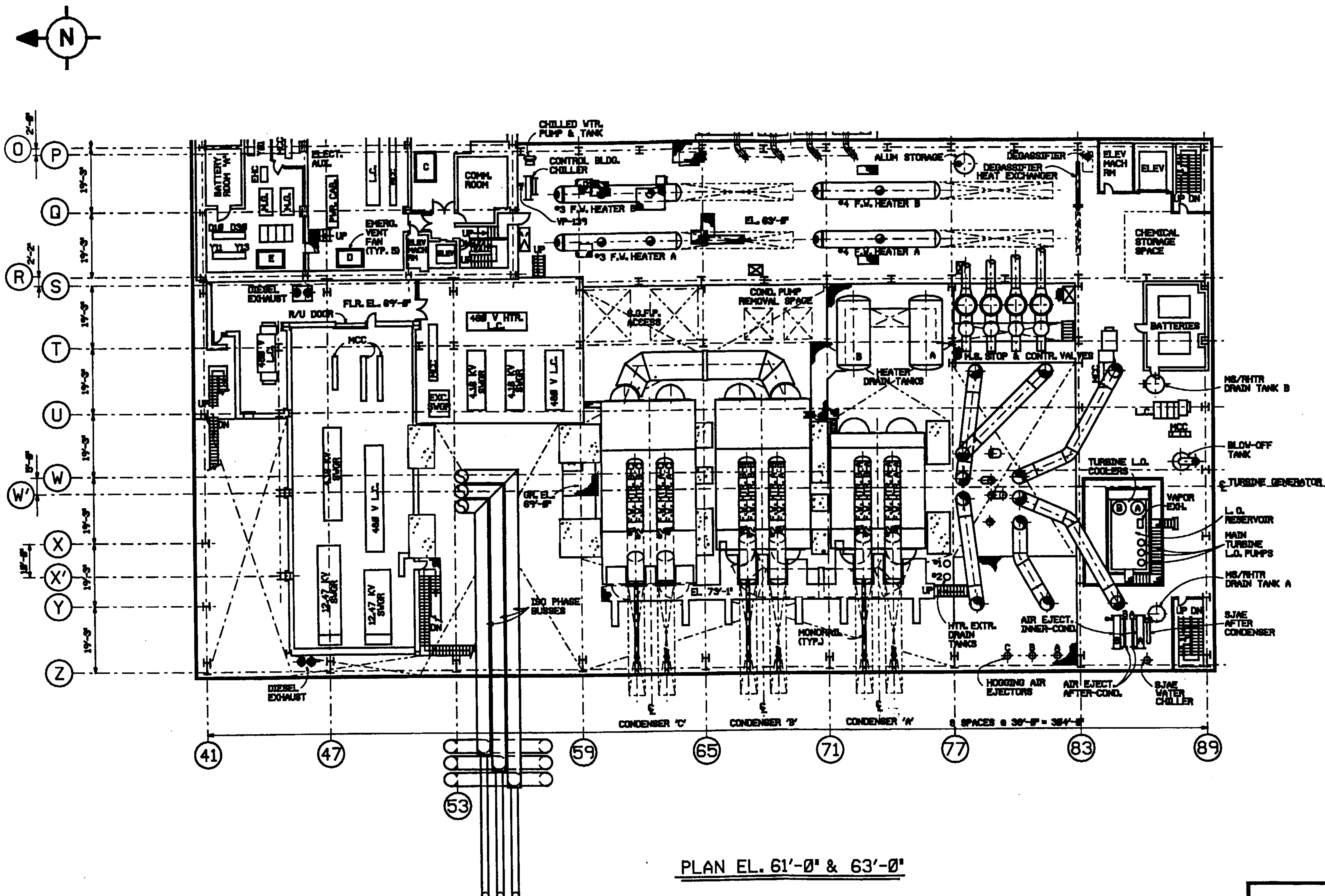


PLAN EL. 61'-0" & 63'-0"

* Figure is representative of plant areas at the time of shut down. Equipment shown may have been deactivated or removed.

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 2-8
Turbine Building
Elevation 61 FT and 63 FT

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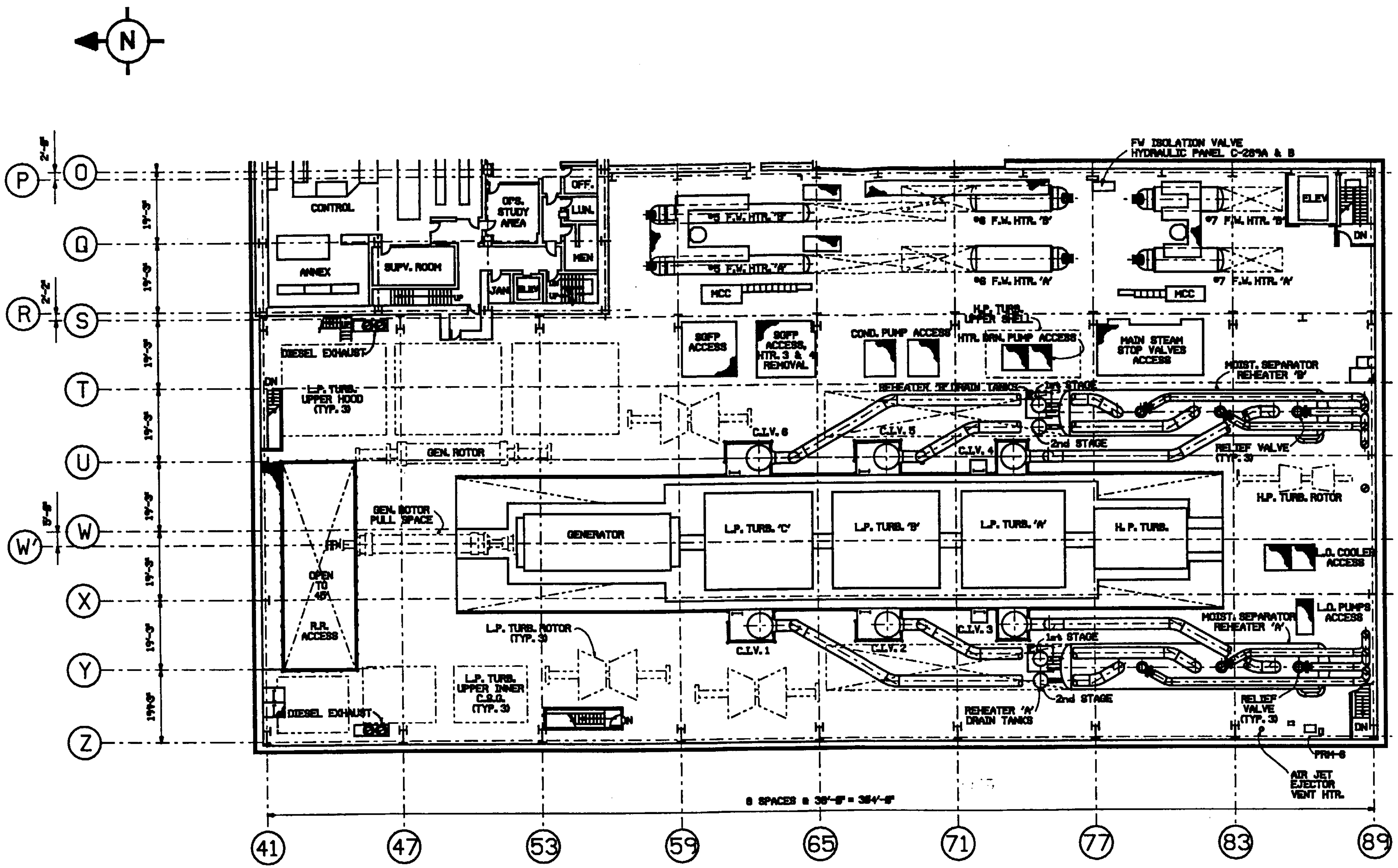


PLAN EL. 93'-0"

* Figure is representative of plant areas at the time of shut down. Equipment shown may have been deactivated or removed.

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 2-9
Turbine Building
Elevation 93 FT

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**Figure 2-10**

**TROJAN ORGANIZATION**

**TROJAN SITE EXECUTIVE AND PLANT GENERAL MANAGER**

**INDEPENDENT REVIEW AND AUDIT COMMITTEE (IRAC)**

**Key Decommissioning Responsibilities Are:**

**Maintenance**

**Operations**

**Personnel/Radiation Protection**

**Emergency Preparedness**

**GENERAL MANAGER NUCLEAR OVERSIGHT**

**Key Decommissioning Responsibilities Are:**

**Quality Assurance**

**Quality Control**

**GENERAL MANAGER ENGINEERING/ DECOMMISSIONING**

**Key Decommissioning Responsibilities Are:**

**Engineering**

**Decommissioning**

**Cost Control**

**GENERAL MANAGER PLANT SUPPORT AND TECHNICAL FUNCTIONS**

**Key Decommissioning Responsibilities Are:**

**Nuclear Security**

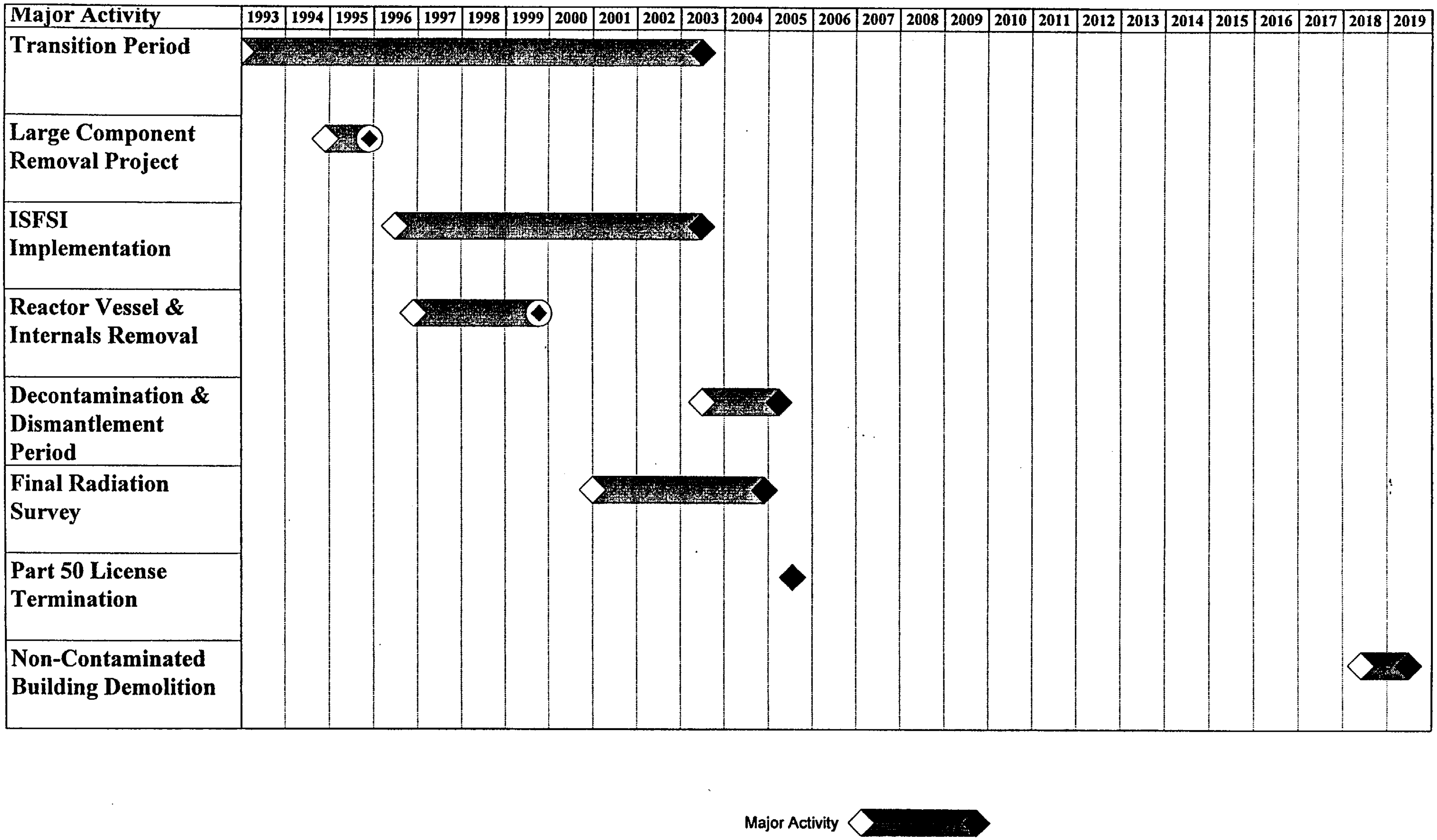
**Licensing**

**Purchasing**

**Training**

**Figure 2-11**

**Decommissioning/Site Restoration Schedule**



# 3. PROTECTION OF OCCUPATIONAL HEALTH AND SAFETY

## 3.1 FACILITY RADIOLOGICAL STATUS

In accordance with the requirements of 10 CFR 50.82(a)(9)(ii)(A) (Reference 3-1) and guidance of Regulatory Guide 1.179 (Reference 3-2), this section provides a description of the radiological conditions at the TNP site. This TNP site characterization incorporates the results of surveys conducted to quantify the extent and nature of contamination at TNP. The results of TNP site characterization surveys and analyses have been and continue to be used to identify areas of the site that will require remediation, as well as to plan remediation methodologies and costs.

### 3.1.1 DEVELOPMENT OF SITE CHARACTERIZATION METHODOLOGY

TNP's site characterization plan was developed and implemented following permanent shutdown of the plant in 1993 using the guidance available at the time the characterization surveys were conducted. This guidance included Regulatory Guide 1.86 (Reference 3-3), NUREG/CR-5849 (Reference 3-4) and NUREG/CR-5512 (Reference 3-5). Following the development and implementation of the site characterization plan, the revised release criteria of 10 CFR 20, Subpart E (Reference 3-6), and guidance of NUREG-1575 (MARSSIM) (Reference 3-7) were issued. Although the TNP site characterization was conducted under the previous guidance, PGE has elected to conduct final surveys using the most recent 10 CFR 20.1402 release criteria and the MARSSIM approach of applying derived concentration guideline levels to verify that allowable release criteria are met.

NUREG-1575 incorporates a data quality objectives (DQO) process to ensure that survey results are of sufficient quality and quantity to support the decision to release the area for unrestricted use. The DQO process is introduced as a systematic planning tool to determine the type, quantity, and quality of data needed to support decision-making. Because the DQO process is intended to be used in the planning and development effort, applying this process to the completed TNP site characterization is not practicable. Moreover, for areas of the TNP site classified as impacted, this action would have no significant benefit since: 1) many of the contaminated systems and components have been removed from the site, making much of the site characterization data historical in nature; and 2) these areas will undergo final survey to verify that the unrestricted release criteria are met.

The portion of the TNP site outside the current industrial area was determined from the results of the site characterization to be non-impacted, thus requiring no additional surveys. This conclusion was reached without application of the DQO process since, as stated above, the guidance in use at the time did not incorporate the DQO concept. PGE has determined that the site characterization results, including classification of the non-impacted areas, remain valid since the TNP site characterization activities were planned and conducted using a rigorous systematic method in accordance with the available regulatory guidance as well as the approved Trojan Nuclear QA Program (Reference 3-8).

In accordance with the available guidance, the TNP Site Characterization Plan incorporated the following objectives:

1. Determine the initial (post operation) radiological status of the facility;

2. Estimate the site source term and isotopic mixture to support decommissioning cost estimation and decision-making; and

3. Determine the location and extent of any contamination outside the radiologically controlled areas.

In the plan, the primary decision-maker and key team members were identified. Available resources were specified and relevant deadlines for the survey established. To assure representative data, the site characterization plan identified the method for selecting the type and number of measurements, locating those measurements, and determining the background contribution.

The site characterization process was divided into four areas: structures, systems, activation, and environment. Quality assurance requirements were imposed on the process, which included training and qualifications, instrumentation, procedures, records, and audits and surveillances. These measures, along with quality control methods for data collection, were implemented to assure data quality.

A comprehensive report of the results of the TNP site characterization was prepared and made available for review by the NRC. A description of the TNP site characterization results is incorporated into this section.

### 3.1.2 FACILITY HISTORY

#### 3.1.2.1 Operating History

TNP achieved initial criticality in December 1975 and began commercial operation in May 1976. The reactor output was licensed at 3411 MWt with an approximate net electrical output of 1130 MWe. TNP shut down for the last time in November 1992, because of a steam generator tube leak precipitated by a failed sleeve. Commercial operation was formally discontinued in January 1993, after approximately 17 years of operation. The plant operated for 14 fuel cycles and approximately 3300 effective full power days.

#### 3.1.2.2 Radiological History

##### 3.1.2.2.1 Effluents

TNP operation resulted in limited release of radioactive material through two main pathways: gaseous and liquid effluents. The plant routinely monitored these releases. Monitored gaseous, or airborne, pathways during power operation included the Auxiliary/Fuel Building exhaust, Containment Building exhaust (purge), and condenser offgas system exhaust. Other potential airborne release pathways, not specifically monitored, included the main steam relief valves, steam packing exhauster blower discharge, and turbine building exhaust. Liquid discharge pathways included liquid radwaste discharge, steam generator blowdown, and Turbine Building sump/oily water separator. Effluents were quantified and reported to the NRC in the Semiannual Radioactive Effluent Release Reports. Tables 3-1 and 3-2 contain information on gaseous releases (noble gases, iodines, and particulates) and Table 3-3 provides information on liquid releases by calendar quarter.

3.1.2.2.2 Operational Events

PGE conducted a review of operational events to determine which could potentially impact decommissioning. Primary sources used to determine TNP radiological history included corrective action system documents (e.g., event reports, radiological event reports, and nonconformance reports) and reports to the Atomic Energy Commission, NRC, and the State of Oregon (e.g., licensee event reports). Interviews were also conducted. A discussion of several notable events follows.

Between 1981 and 1982, fuel assembly damage occurred during plant operation which resulted in loose fuel pellets being released into the RCS. The damage was attributed to mechanical wearing of the fuel cladding (water-jet-induced vibration or baffle jetting) caused by excessive gaps in the lower internal baffle plate joints.

Fuel assembly damage resulted in high levels of transuranic radionuclides being found in plant systems during operation. Although this did not result in radioactivity releases greater than TNP Technical Specifications limits, consideration of the implications was necessary while developing the site characterization survey plan. Baffle jetting was eliminated with modifications to selected fuel assemblies, peening of baffle plate gaps, and an upflow modification to the lower vessel internals. No further baffle jetting damage was observed.

The fuel assembly damage necessitated implementing special measures to prevent spreading discrete radioactive particles created by these fuel fragments. However, approximately five years later, high levels of loose surface and airborne contamination occurred while removing the reactor head (during stud hole cleaning). Subsequent investigation revealed a partial fuel pellet on the reactor vessel flange and pellet fragments in the lower refueling cavity area. During stud hole cleaning, air dispersed the fuel pellet ash, contaminating many surfaces in the Containment, Fuel, and Auxiliary Buildings.

Since this event occurred, much of the contamination has been remediated. Prior to remediation, many areas of TNP required special radiation protection technician coverage and additional protective clothing when accessing locations where discrete radioactive particles could be found. As a result, surfaces in the Containment, Auxiliary, and Fuel Buildings are assumed to be potentially contaminated and will require, as a minimum, a wipe/wash down to remove loose surface contamination. PGE recognizes the potential for discovering additional discrete radioactive particles during decontamination and dismantlement activities, and will implement enhanced monitoring techniques (e.g., increased surface area swipes) to ensure detection.

Releases of radioactive material from the RCS to secondary systems of the plant occurred through steam generator tube leaks during several operating cycles. Leaking tubes were identified in 1978, 1979, 1981, and 1992. Radiological monitoring of the condenser offgas also indicated continued low-level leakage from one or more steam generators from cycle 3 until the final shutdown of TNP. Plant shutdowns to correct the primary-to-secondary leakage were made to ensure the plant complied with TNP Technical Specifications operating and effluent release limits. Primary-to-secondary leakage resulted in some contamination of secondary systems. Areas and systems affected by leakage were included in routine surveys and the site radiological characterization survey. Normal leakage from secondary systems also resulted in the contamination of areas outside the Radiologically Controlled Area (RCA), including the Turbine and Condensate Demineralizer Buildings. Radiological controls were established to prevent spreading contamination.

Although these events were noted, minimal or no offsite radiological consequences resulted from them. In each instance, PGE implemented actions to remove and control contamination and instituted corrective actions to prevent recurrence. These events were used to select additional sampling locations during site characterization, which is discussed later in Section 3.1, with survey results. Appendix 3-1 contains a description of several TNP radiological contamination events.

### 3.1.3 RADIOLOGICAL STATUS OF TNP

TNP site characterization is being completed in two phases: 1) Phase I, scoping survey/site characterization; and 2) Phase II, radiological surveys to support TNP dismantlement and decommissioning.

Phase I, which is complete and compiled in the "Trojan Nuclear Plant Radiological Site Characterization Report," Revision 0.1 dated February 8, 1995 (Reference 3-9), was used to characterize the radiological status of the facility; estimate the site source term and isotopic mixture to support decommissioning cost estimates and decision making; determine the location and extent of contamination outside the RCA; and collect background information to help facilitate release of the site for unrestricted use. Phase II is ongoing and involves routine radiological surveys in support of PGE's current 10 CFR 50 license. Phase II will be used to help support facility decontamination and dismantlement. Phase II will continue using the existing radiation protection program and procedures. Areas that were not, or could not be, surveyed during Phase I have been or will be surveyed during Phase II.

Phase I methodology and survey results are summarized in this section. The discussion is divided into four general areas: structures, systems, activation, and environment. An estimate of the disposal volume is also provided for contaminated/activated systems, structures, and components. Phase II characterization efforts are briefly described in Section 4.2.1, and are addressed further in this section only as additional detail is necessary.

The total waste volume and activity discussed in this section does not include the following materials since these materials are not considered decommissioning waste:

1. Nuclear fuel;

2. Control rod elements;

3. Incore instrumentation hardware installed in fuel elements; and

4. Radioactive fluids, filter media, and resins contained in piping, equipment, sumps, etc.

Nuclear fuel, control rod elements, and incore instrumentation hardware installed in the fuel elements will be stored in the ISFSI.

A calculated radioactivity inventory, excluding the items listed above, is furnished that includes both contamination and activation. Various mixtures of radionuclides were evident in survey samples throughout the affected areas. Predominant nuclides are shown in Table 3-7. These nuclides are typical of those found in pressurized water reactor plants and are similar to those discussed in NUREG/CR-0130, "Technology, Safety and Costs of Decommissioning a Referenced Pressurized Water Reactor" (Reference 3-10). This was expected since TNP was the reference plant used for NUREG/CR-0130.

Site characterization survey maps are included in Figures 3-1 through 3-36. These maps show radiation protection survey data collected during the first quarter of 1994. This data is a "snapshot" of the radiological conditions during the survey period. Normally accessed locations are surveyed periodically for radiation/contamination. Current survey data are maintained as qualify-related documents.

Survey instrumentation included the following:

1. Static and scan alpha surface contamination measurements: Eberline ESP-2 instrument with an Eberline AC-3-8 alpha scintillation probe;

2. Static and scan beta-gamma surface contamination measurements: Eberline ESP-2 instrument with a National Nuclear BP-100 beta-gamma scintillation probe; and

3. Gamma exposure rates: Eberline ESP-2 instrument with a SPA-8 gamma scintillation probe and/or a Reuter-Stokes RSS-112 pressurized ion chamber.

To assist the reader in locating specific components or areas while reviewing survey data, equipment locations for major TNP plant equipment are provided in Section 2, Figures 2-2 through 2-9. Reflecting plant configuration at the time of permanent shutdown, these figures are provided for information only. The equipment shown on these figures may have been deactivated or removed.

The following table summarizes the scoping and characterization survey results for structures and systems as discussed in the following sections. Environmental survey results indicated no activity above release criteria.

| Section | Activity (Ci) |
|---|---|
| Structures | 0.031 |
| Systems | 1070.5[a] |
| Activation | $4.2 \times 10^{6}$ [b] |
| TOTAL | $4.2 \times 10^{6}$ |

[a] Not including steam generators, pressurizer, or activation.

[b] Most activity is contained in the vessel internals. Activation curies for reactor vessel, clad, insulation, and concrete are approximately $3.1 \times 10^{4}$ Ci.

### 3.1.3.1 Structures

Structures were surveyed to determine contamination levels found in TNP buildings. Operational radiation protection survey data was supplemented by additional surveys to determine the presence and/or level of contamination. The survey focused on areas outside the present RCA to determine remediation needed to release the areas for unrestricted use. Structures with known contamination were surveyed to characterize the extent of contamination, including area and depth of contamination penetration. External system surfaces were considered structures for the purpose of this survey.

Using TNP radiological history, biased surveys were conducted to quantify radioactivity based on suspected, or known, contamination at a given location. Contamination in excess of the cleanup criteria was identified in a limited number of biased survey locations, including Turbine Building locations where primary-to-secondary leakage caused fixed contamination to build up in floor concrete.

Unbiased locations of unaffected areas were selected based on random selection of sampling locations within areas of TNP where radioactivity above background was not expected. Unbiased areas included office buildings, Turbine Building, Maintenance Building, etc. Only two of the sample points had detectable removable contamination levels. At these locations, the beta-gamma contamination was below 1 percent of the Regulatory Guide 1.86 contamination limit and the alpha contamination was below 40 percent of the limit.

Contamination includes both removable and fixed radioactivity. Removable contamination will be decontaminated through simple means such as wiping or mopping. Fixed contamination appears to be deposited in the upper 1 cm of the concrete and can be removed using surface destruction techniques (e.g., scabbling). Table 3-4 contains estimates of volume and radioactivity contained in structures requiring remediation. Fixed contamination levels were not measurable in many biased survey room locations (e.g., Fuel and Auxiliary Buildings) because system radioactivity masked the surface contamination. Additional surveys for fixed contamination will be performed as radiation levels are reduced by decay or removal of the radiation source.

Radioactive waste volume estimates for each building elevation were calculated based on the total area of the elevation, the estimated percentage of contaminated area greater than Regulatory Guide 1.86 limits and an assumed penetration depth of 1 cm. The estimated area of contamination was based on historical data for spills and industry experience and knowledge. Additional data will be collected, as necessary, to ensure the assumptions used are accurate and conservative.

An estimate of the Containment Building concrete volume requiring removal was made through analyses of samples and direct surveys. Sample and survey results indicated the Containment Building internal structural concrete would need to be removed to assure compliance with dose objectives. Samples taken at a number of structural joints, cracks in concrete and in-bed interfaces indicated the contamination might exceed release criteria. A cost analysis indicated removal of the concrete would be the lowest cost option. The Containment Building dome, which consists of the Containment Building concrete structure lined on its interior by an approximately ¼-in. steel liner, will remain following demolition activities. The activated concrete wall that provided structural support and shielding for the reactor vessel has been removed for disposal.

As summarized in Table 3-4, the total estimated radioactivity on structural surfaces attributed to contamination was approximately 0.031 Ci. Survey maps showing RCA radiological conditions are also included at the end of this section as Figures 3-13 through 3-36. A summary of radiological conditions for each building, by elevation, is contained in Table 3-5 and a description summarizing the results is contained in Appendix 3-2. Removable contamination, general radiation levels, and maximum contact dose rates are identified.

#### 3.1.3.2 Systems

Systems were surveyed to determine contaminated systems and estimate the quantity of contamination. Each plant system was evaluated for its likelihood to be contaminated and sampled by direct survey, loose surface swipe, or metal scrapings. Detected activity that could not be identified as naturally occurring was attributed to plant operations and the system was classified as contaminated. The approach involved grouping plant systems into four categories: C1 or contaminated; C2 or potentially contaminated due to cross contamination; I or indeterminate (need more data); and N or free of contamination.

Samples of systems from these classifications were collected. The samples were predominantly taken from systems classified as C2, I, or N to determine the extent of plant system contamination. Less attention was directed at contaminated systems since data was collected for these systems during plant operation and was still representative.

Systems with potential internal contamination were also sampled as part of the site characterization scoping survey. Some systems which were classified as N, and were expected to be only externally contaminated, were not sampled (as mentioned previously, the external surfaces of a system were considered structures).

If a system is sufficiently contaminated, its curie content can be estimated by measuring the dose rate from its piping. For systems classified as C1, plant surveys determined the dose rate levels for each system. Conservative values (i.e., maximum average dose rate) were assumed for each system to provide a bounding estimate of curie content. Dose rate and pipe size were used to calculate the activity deposition ($Ci/m^2$). The result was multiplied by the total contaminated surface area of the system to conservatively estimate the system's curie content.

For potentially contaminated systems (i.e., C2 or I), it was not possible to approximate activity deposition by field dose rate measurements. Scrapings from the system were used, or an estimate was assumed, to determine activity deposition and curie content.

Where ambient radiation areas and physical configuration allowed, the surface activity of systems was determined by direct surveys with beta-gamma detection equipment. Direct surveys were not performed on systems in high ambient radiation areas or where physical configuration prevented the survey.

Swipes were taken to measure removable surface contamination and scrapings were collected to determine fixed surface contamination. Swipes and scrapings were analyzed in a low background area. Some swipes were limited to one location because of physical restrictions, while others were composite swipes used to check large areas and/or various individual locations.

Alpha contamination at TNP has been detected only when accompanied by detectable beta-gamma activity. Consequently, only samples that showed detectable beta-gamma activity were typically counted for alpha contamination.

Individual system scrapings were counted in the laboratory to determine a qualitative radionuclide spectrum. Fixed and removable contamination was found in the contaminated systems. The total radioactivity is not expected to be substantially reduced through nonaggressive decontamination methods. Operational experience, during activities such as steam generator primary bowl hydrolasing, indicates contamination is tightly adhered to surfaces and will probably require component disposal.

Potential burial waste volumes and surface activities by system are contained in Table 3-6. The total system burial volume is estimated at 215,789 $ft^3$. The total surface activity is conservatively estimated to be 1070.5 Ci. The volume and activity estimates assume that the four steam generators and pressurizer are disposed of as part of the large component removal project. The volume estimate was based on a model developed by TLG Services, Inc (TLG). The estimate includes the volume of externally contaminated systems (e.g., electrical cable, conduit, piping) not sampled as part of the scoping survey.

### 3.1.3.3 Activation

Plant component activation occurred during normal plant operation due to neutron irradiation. Estimates of plant component activation were made using operational data. TLG performed neutron transport calculations using TNP-specific data. Measurements were made to verify the accuracy of the calculations.

Calculations for components activated by neutron irradiation consist of one-dimensional neutron transport and point neutron activation analyses. Calculations indicate the reactor vessel, vessel internals, and concrete shielding have levels of radioactivity that will require remediation. These calculations were performed using TLG's FISSPEC and O2FLUX computer codes and the ANISN and ORIGEN computer codes obtained through the Oak Ridge National Laboratories Radiation Shielding Information Center. Ancillary calculations were performed using TLG's ANISNOUT and O2READ computer codes and Microsoft's EXCEL computer program.

The one-dimensional neutron transport model was normalized with data from a Westinghouse Electric Corporation report, "Analysis of Capsule V from the Portland General Electric Company Trojan Nuclear Plant Reactor Vessel Radiation Surveillance Program." The radionuclide inventories evaluated in these analyses were for the four major structural material compositions including Type 304 stainless steel, pressure vessel carbon steel, concrete, and plate/rebar carbon steel.

A listing of 10 CFR 61 classification by component, one and five years after shutdown, is included in Tables 3-8 and 3-9. One year following shutdown, the radioactivity content of activated components was estimated at $4.2 \times 10^6$ Ci. Five years following shutdown, the calculated activity of the activated components was approximately $2 \times 10^6$ Ci. Predominant radionuclides include $^{55}Fe$, $^{60}Co$, and $^{63}Ni$.

During 1999, PGE removed the reactor vessel with internals intact (reactor vessel package) from the 10 CFR 50 licensed area of the TNP site (Reference 3-11). The reactor vessel package was transported for disposal at the US Ecology low level radioactive waste facility near Richland, Washington. Removal of the reactor vessel package from the 10 CFR 50 licensed area of the TNP site eliminated approximately 2 million curies of activity from the TNP. Not including the spent nuclear fuel that will be transferred to the ISFSI, removal of the reactor vessel and internals resulted in removal of greater than 99 percent of the remaining activity (curies) at the TNP facility.

### 3.1.3.4 Environment

The environmental survey, which included representative outdoor areas, focused on the impact of TNP operation on the environment due to the release of radioactive material. Operational and pre-operational environmental monitoring data were used to measure and evaluate the impact. Additional sampling was conducted to augment, or better define, areas requiring biased surveys. Survey results were compared to background data to determine the overall consequences of TNP operation.

During Phase I of site characterization, soil, sediment, and surface water were sampled. Exposure rates were measured wherever soil was sampled, except where exposure rates were influenced by onsite structures. Paved areas onsite were scanned for beta contamination or sampled and analyzed for gamma emitters. A general site map, Figure 3-1, shows the site divided into zones. Survey maps depicting sample points by grid location are provided in Figures 3-2 through 3-12. Survey maps were not included for zones where no samples were collected (i.e., Zones 4, 12, 13, 15, and 16).

Biased sample locations were determined from reviewing plant records that documented radiological events at TNP from 1975 to 1993 (see Section 3.1.2.2.2 and Appendix 3-1). Corrective action programs were reviewed and interviews conducted with PGE personnel to help determine potential sample locations.

3.1.3.4.1 Surface Soil Survey

Surface soil samples were obtained at locations on PGE property contiguous with TNP. The locations of the soil samples are outlined in the Site Characterization Report, Section 5, and are shown in Figures 3-2 through 3-12 of this Decommissioning Plan. Each sample contained approximately 1 liter of material collected from a 1 $ft^2$ area. Soil samples were analyzed in-house for gamma emitters by gamma spectrometry and control samples were analyzed by TMA/Eberline in Albuquerque, New Mexico. Selected samples were also analyzed for $^{90}$Sr.

Background soil samples were collected from four locations around TNP. Two samples were used as quality control checks and were not included with the data. The four background locations were:

1. PGE owned property in Prescott, Oregon (approximately 0.75 miles north-northwest of TNP containment);

2. Water treatment facility in Rainier, Oregon, near radiological environmental sample location 2 (approximately 3.8 miles northwest of TNP containment);

3. PGE owned property west of Highway 30 (approximately 1 mile west of TNP containment); and

4. Northwest of Kalama, Washington, near radiological environmental sample location 11B (approximately 1.4 miles east-northeast of TNP containment).

For soil background measurements, the mean background $^{137}$Cs concentration was 0.49 pCi/g with a standard deviation of 0.4 pCi/g and a range of 0.01 to 1.3 pCi/g. Substantial variation in background $^{137}$Cs concentrations was observed between varying soil types. Sandy soils found near the river contained low $^{137}$Cs concentrations, while clay soils contained higher concentrations.

For the survey of unaffected soil areas, the mean $^{137}$Cs concentration was 0.77 pCi/g with a standard deviation of 0.86 pCi/g and a range of 0.01 to 2.94 pCi/g. Primarily, the nonnaturally occurring isotopes found in soil samples were $^{137}Cs$ and $^{90}Sr$. Fallout from atmospheric weapons tests and the Chernobyl accident are the major sources of $^{137}Cs$ and $^{90}Sr$ in the environment. $^{90}Sr$ results averaged 0.2 pCi/g with a standard deviation of 0.16 pCi/g and a range of 0.02 to 0.32 pCi/g. The $^{90}Sr$ levels measured during the preoperational period ranged from 0.01 to 1.28 pCi/g with a mean of 0.30 pCi/g.

Biased survey soil samples were taken onsite where potential soil contamination may have occurred. Subsurface soil samples taken in 1991 from the tank farm area were also reviewed as part of the analysis. Samples were taken at 1, 2, and 3 ft depths at 5 locations.

The predominant nonnaturally occurring isotope detected was $^{137}Cs$. One surface sample, taken from the tank farm area, also contained low levels of $^{134}Cs$ (0.010 pCi/g) and $^{60}Co$ (0.044 pCi/g). The mean value for $^{137}Cs$ in affected soil samples was 0.10 pCi/g with a standard deviation of 0.098 pCi/g. The $^{137}Cs$ content of the 1991 samples was below the cleanup criteria.

3.1.3.4.2 Groundwater Survey

A groundwater monitoring program is being developed based on the specific geological and hydrogeological characteristics of the TNP site. As part of this effort, the geological and hydrogeological studies that were incorporated during plant operation in the TNP Final Safety Analysis Report (Reference 3-12) and the TNP Environmental Report (Reference 3-13) have been updated. The updated evaluation is documented in a Hydrogeology Evaluation Report, dated October 2000, by Cornforth Consultants (Reference 3-14). With the assistance of consultant experts in the field of geology and hydrogeology, the conclusions and recommendations of the updated evaluation will be used to ensure that the groundwater monitoring program implementation is representative and appropriate for measuring potential groundwater contamination on the TNP site.

Site Geology

The TNP was constructed on an elongated rocky ridge or hill referred to as "Trojan Hill." The Trojan Hill is approximately 1,000 feet wide, extends about 5,500 feet along the Columbia River, and has a maximum elevation of approximately 100 feet. Trojan Hill principally consists of bedrock of the Goble Volcanic Series. The TNP was constructed directly on bedrock that is composed of basaltic flow breccia, basalt flows and dikes, vesicular basalt, agglomerate, tuff breccia and tuff. The tuffs and basalt are the predominant rock types. The tuffs are soft to medium hard and porous. The basalt is very hard, except where its surface has been weathered. The agglomerate is moderately hard and well cemented. These rock types occur in discontinuous layers that range from 2 to 25 feet in thickness. The dip of the layering is about 10 to 30 degrees to the southwest. Basalt dikes trend northwest-southeast and dip about 70 to 80 degrees toward the northeast.

Based on drillers logs for rock cores collected as part of the foundation evaluation for the TNP, the basalt and tuff underlying the site are competent with small fracture zones at varying depths below ground surface. Many of the fracture zones have been partially to fully filled through the deposition of secondary minerals. There is little horizontal correlation between fracture zones.

Two areas of the TNP site were filled, with sand and silt derived from dredging the Columbia River, to raise the ground surface to an elevation of 40 to 44 feet. Approximately 30 feet of dredged material was placed 100 feet east/southeast of the Auxiliary/Fuel/Containment Building Complex (i.e., power block), just south of the water intake structure. A similar depth of fill material was placed approximately 200 feet west of the power block, under what is now a portion of the entrance road and an the employee parking lot. West of Trojan Hill is a buried valley that was naturally filled with over 300 feet of alluvial sediments. The sediments consist of silt, sandy silt, and silty fine sand. East of Trojan Hill is the Columbia River whose bottom sediments mainly consist of silts and sand.

Site Hydrogeology

Groundwater elevations measured in 1968 indicate that saturated zones are present in Trojan Hill. These zones are not expected to yield sufficient water supply for long-term human use. The uppermost water-bearing zone was found to be present at depths ranging from 5 to 20 feet below ground surface. This zone likely corresponds to weathered basalt surfaces or unhealed fracture zones capable of transmitting groundwater.

As is stated in the Cornforth Consultants hydrogeology evaluation, the Trojan Hill has been characterized as "essentially impervious." Coefficients of permeability likely range from $1 \times 10^{-7}$ to $1 \times 10^{-8}$ cm/sec, except in localized zones with a high degree of unhealed fractures or weathered surfaces. The conclusion that groundwater movement in Trojan Hill is limited is corroborated by the fact that groundwater did not seep into deep excavations during plant construction and rainwater tends to pond at the land surface and does not seep into the ground.

Based on the 1968 water level measurements, the direction of groundwater flow along the west side of Trojan Hill is to the west toward the buried valley. A comparable eastern component of groundwater flow on the east side of Trojan Hill is expected given the Trojan Plant site topography and location of the Columbia River. Thus, a groundwater divide is expected to be present near the center of Trojan Hill. Construction-related excavations may have caused a shift in the location of the divide to one side or the other of Trojan Hill. Groundwater that discharges to the west enters the buried valley where it tends to flow toward either the north or south end of the valley and then discharge into the Columbia River.

Groundwater Monitoring Program

The groundwater monitoring program will be implemented in a phased manner. The first phase (i.e., Phase I) of the program will involve the installation of monitoring wells in Trojan Hill to determine the following:

- Directions of groundwater movement in the vicinity of the power block
- Rates of groundwater movement in the vicinity of the power block
- Presence above background, if any, of radionuclides associated with plant operations

A second phase (i.e., Phase II) will be implemented if radionuclides are detected above background levels and if it is determined that there is a potential for these radionuclides to migrate into the buried valley at concentrations that would require additional groundwater monitoring. Each phase of the groundwater monitoring program is discussed below.

Phase I

Phase I will consist of the following:

- Groundwater monitoring plan preparation
- Monitoring well installation and development
- Monitoring well sampling and water level measurement
- Aquifer testing
- Monitoring data analysis and reporting

*Groundwater Monitoring Plan*

A groundwater monitoring plan will be prepared that covers field work planning, project management, sampling and analysis, health and safety, and waste management.

*Monitoring Well Installation and Development*

Seven monitoring wells will be installed in the vicinity of the power block. Two background wells also will be installed, one at the far north end of the TNP site near the town of Prescott, and the other near the Cooling Tower south of the TNP. The background wells will be installed at a similar depth to the monitoring wells installed near the power block to allow for the collection of background water quality data for the uppermost water-bearing zone in an area unaffected by plant operations. The two existing wells in the town of Prescott will not be used because of their depth (280 and 560 feet below ground surface). The TNP water supply wells also will not be used because they are screened in valley fill sediments other than the Goble Volcanic Series.

Historical data and soil characterization data indicate that minor releases occurred on the south side of the power block in the area of the plant tank farm. A summary description of the contamination events is contained in Appendix 3-1. This historical information and characterization data is used as the basis for selecting the monitoring well locations. Four of the monitoring wells will be installed on the east side of the power block given the locations of the two known releases in that area of the TNP site. In addition, the principal portions of the power block where other releases could have occurred (e.g., containment building and fuel storage building) are located on the east side of the power block. One of the four wells will be installed just to the east of the containment building to provide an upgradient control point for measuring groundwater elevations. The remaining three wells will be placed approximately 100 to 200 feet east of the power block between the southern end of the dry fuel storage area and the water intake structure.

The remaining three wells will be located to the west of the power block. One well will be located just west of the turbine building to provide an upgradient control point for measuring groundwater elevations. The other two wells will be located 100 to 200 feet west of the turbine building. One of the two wells will be located west of the southwest corner of the turbine building, in the fill material that was placed in this area. The second well will be placed further to the north near the security building.

Each monitoring well will be installed with air rotary, cable tool, or other acceptable drilling methods. In well locations where fill material is present, a temporary steel casing will be advanced through the fill to the top of bedrock. An open borehole will be advanced into the bedrock until the first water-bearing zone is encountered. Borehole cuttings will be logged by a qualified geologist. A 2-inch diameter monitoring well constructed of Schedule 80 PVC casing, screen and end cap will be placed in each borehole. Well construction will be completed by installing a filter pack, seal, annual seal, and flush-mounted traffic-rated surface monument.

Each monitoring well will be developed to improve its hydraulic connection with the water-bearing zone. Consistent with Oregon Water Resources Department rules, the well will not be developed for at least 12 hours after construction if the seal was constructed of dry bentonite or 24 hours after completion if the well seal was constructed of bentonite grout. The well will be developed using surging or bailing methods. Well development will continue until turbidity clears and the pH, temperature, and specific conductance stabilize. All water generated during well development will be managed in accordance with approved TNP waste management procedures.

*Monitoring Well Sampling and Water Level Measurement*

After at least seven days have elapsed since completing well development, the water level in each well will be measured with a decontaminated water-level probe. The well will be purged until at least three casing volumes are removed, until field water quality parameters stabilize, or until the well purges dry. Field parameters will include pH, temperature and specific conductance. Groundwater samples will be collected, and sample analysis will include the following radionuclides: tritium, Fe-55, Co-60, Ni-63, Cs-137, Sr-90, U-234, U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-242, and Cm-243/244. The samples will also be analyzed for nonradioactive boron. All purge water will be managed in accordance with approved TNP waste management procedures.

*Aquifer Testing*

Aquifer tests will be performed in each monitoring well installed near the power block to determine aquifer properties (e.g., hydraulic conductivity). Slug tests will be performed. Water level data will be measured using a pressure transducer with an electronic data recorder.

*Monitoring Data Analysis and Reporting*

Water level measurements made in each monitoring well will be used to determine the direction of groundwater movement on both the east and west sides of the power block. Hydraulic conductivity measurements made in each well will be used along with gradients estimated from the water level measurements to calculate the rate of groundwater movement.

The analytical results for each monitoring well will be compared with background groundwater quality to determine if any radionuclides are present above background levels. If so, available site data and literature information regarding radionuclide retardation and decay will be used to determine if there is a potential for radionuclide migration from Trojan Hill into the buried valley at concentrations that indicate there is a need to conduct a second phase of groundwater monitoring.

Phase II

The scope of the Phase II groundwater monitoring program will be based on the Phase I results. Phase II could include, for instance, the installation of additional monitoring wells in the buried valley (to the west of Trojan Hill), if radionuclides are detected during Phase I at concentrations above background levels and if there is a potential for these radionuclides to migrate to potential exposure points at concentrations that would exceed background levels.

3.1.3.4.3 Surface Water Survey

Surface water was sampled from indicator sites on PGE property surrounding TNP. A 1 gallon sample was obtained from each site for gamma and $^{90}$Sr analysis and a 60 ml sample for tritium analysis. The water samples were analyzed for gamma emitters using a gamma spectroscopy system located onsite. Water samples were analyzed for tritium in the onsite counting laboratory. Selected samples were analyzed for $^{90}$Sr.

To determine background, water samples were collected from four locations around TNP. The locations included:

1. Fishhawk Lake (approximately 18 miles west of TNP containment);
2. Ponds at the intersection of Goble and Bishop Roads (approximately 3 miles southwest of TNP containment);
3. Kress Lake (approximately 1 mile east-northeast of TNP containment); and
4. Deer Island ponds (approximately 7 miles south of TNP containment).

Analyses for gamma emitters and tritium were completed on the samples. No gamma emitters other than naturally occurring radionuclides were identified in the samples. Tritium values were less than detectable. The four samples analyzed for $^{90}$Sr were less than detectable. Minimum detectable activity (MDA) for $^{137}$Cs, tritium, and $^{90}$Sr was approximately 4, 450, and 0.3 pCi/l, respectively.

For the survey of unaffected water areas, samples were collected from random locations in Whistling Swan and Reflection Lakes located on PGE-owned property surrounding the TNP site. No nonnaturally occurring radionuclides were detected in the samples by gamma spectrometry. Neither tritium nor $^{90}$Sr was detected in the samples.

For the biased survey, samples were taken from the potentially affected Recreation Lake, also located on PGE-owned property surrounding the TNP site. No nonnaturally occurring radionuclides were detected in the samples. MDAs for the biased and unbiased survey analyses were the same.

#### 3.1.3.4.4 Bottom Sediment Survey

Bottom sediment samples were taken from PGE property around TNP. Approximately 1 liter of sediment was obtained at each sampling site. The sediment samples were dried and analyzed for gamma emitters using a gamma spectroscopy system located onsite. Selected sediment samples were analyzed for $^{90}$Sr by TMA/Eberline.

Specific isotopic background sediment samples were not collected. Instead, soil background results were used as sediment background. Background soil samples were analyzed as part of the site characterization effort, and the mean $^{137}$Cs concentration was 0.49 pCi/g. A comparison of the $^{137}$Cs concentration in preoperational sediment samples to the background soil samples showed a high correlation with the sediment mean equal to 0.51 pCi/g and the soil mean equal to 0.49 pCi/g.

In conducting the survey of unaffected sediment areas, samples were taken from Whistling Swan and Reflection Lakes. The mean value for $^{137}$Cs was 0.36 pCi/g with a standard deviation of 0.22 pCi/g and a range of 0.02 to 0.86 pCi/g. The unaffected area sediment samples contain $^{137}$Cs at levels below the release value for $^{137}$Cs. $^{90}$Sr content of the two sediment samples sent to TMA/Eberline were 0.05 and 0.03 pCi/g. The lower level of detectability for the $^{90}$Sr analysis was 0.02 pCi/g. These results are within the preoperational range of $^{90}$Sr which was from 0.01 to 0.44 pCi/g with a mean of 0.08 pCi/g. The $^{90}$Sr content of the sediment samples was also below the corresponding screening release level.

For the biased sediment survey sample population, samples were taken from the berm and main areas of Recreation Lake. Results of the analyses indicate a mean of 0.28 pCi/g with a standard deviation of 0.37 pCi/g and a range of 0.04 to 1.12 pCi/g. The affected area samples contain $^{137}$Cs in amounts below the release level. No other gamma emitters were detected.

#### 3.1.3.4.5 Pavement Survey

Pavement scans and sampling were performed. Pavement was scanned for beta contamination. In areas where there was interference from the RWST, a 1 ft$^2$ sample was collected and analyzed using a gamma spectroscopy system located onsite.

No specific background pavement locations were monitored for this survey. Sample locations located in the TNP park and recreational areas were used to estimate background levels. Since these areas were unaffected by TNP operation, the survey data for these locations was determined to be an acceptable estimate of background levels of radioactive material in pavement. The mean gross beta reading was 610 dpm/100 cm$^2$ with a standard deviation of 94 dpm/100 cm$^2$ and a range of 456 to 764 dpm/100 cm$^2$.

For the survey of unaffected pavement areas, randomly selected 100 $ft^2$ sections of pavement in other areas of the TNP site which were unaffected by operations were scanned with an ESP-2 and BP-100 detector. The mean value was 657 dpm/100 $cm^2$ with a standard deviation of 74 dpm/100 $cm^2$. The range of measurements was from 542 to 788 dpm/100 $cm^2$.

For the biased pavement survey, the affected areas consisted of pavement around the tank farm and its drainage to the west, pavement around the oily water separator, and the paved equipment laydown area around the cooling tower. Pavement samples were taken from affected areas with at least two samples from each affected area. The only detectable nonnaturally occurring radionuclide found in the pavement samples was $^{137}Cs$ in low concentrations. The results of the biased samples exhibited a mean of 0.16 pCi/g with a standard deviation of 0.40 pCi/g and a range of 0.019 to 1.5 pCi/g. $^{137}Cs$ content of the biased pavement samples was similar to that found in background and indicator soil samples obtained for site characterization. One sample, taken from the curb at the southeast corner of the circulating water pump pit area, had the highest $^{137}Cs$ concentration of 1.5 pCi/g. For comparison, conservatively assuming the $^{137}Cs$ was from the top 1 cm of the concrete and covered a 100 $cm^2$ area, then the calculated contamination level would be 799 dpm/100 $cm^2$.

3.1.3.4.6 Exposure Rate Survey

Exposure rates were measured at locations where affected and unaffected site characterization indicator soil samples had been collected. The measurements were made with a Reuter-Stokes pressurized ion chamber instrument positioned 1 meter above the sample site.

Data for exposure rate background was collected during preoperational surveys at TNP using a high pressure ion chamber, the same type of instrument used during the site characterization survey. The preoperational mean reading was 7.1 μR/hr with a standard deviation of 1.0 μR/hr and a range of 5.6 to 9.4 μR/hr. The survey locations coincide with the Radiological Environmental Monitoring Program locations.

For the exposure rate survey of unaffected areas, surveys were taken at the unaffected soil sampling locations. Exposure rates ranged from 5.2 to 9.0 μR/hr at the unaffected area locations. The mean exposure rate was 6.4 μR/hr. Data compared favorably with preoperational data, indicating no effect from TNP operation.

For the biased survey, exposure rates were measured at affected area soil sample sites where it was determined that radioactive content of surrounding structures would not influence the measurements. Measurements made at two locations were influenced by the RWST and were not included. Exposure rates at four locations were not measured because of radiation levels from the RWST. Exposure rates at two locations were not measured because of radiation levels from the Low-Level Radioactive Waste Storage and Fuel Buildings. The values at the remaining locations ranged from 6.0 to 8.3 μR/hr with a mean of 6.8 μR/hr. This is consistent with background data.

### 3.1.4 SITE CHARACTERIZATION QUALITY ASSURANCE

TNP site characterization activities are conducted under the auspices of PGE-8010, "Trojan Nuclear Plant Quality Assurance Program," as incorporated into Section 7 of this Decommissioning Plan. TNP's Nuclear Quality Assurance Program ensures that survey activities are performed in a manner that assures the results are accurate and that uncertainties have been adequately considered. Surveys are performed by trained individuals who follow standard written procedures and are using properly calibrated and source-checked instruments. The custody of samples is tracked from collection to analysis, with every step of the process documented in a way that can be audited. In addition, QA practices ensure that offsite laboratory analyses are conducted using approved Radiological Environmental Monitoring Program (Reference 3-12) procedures. Finally, characterization data, as well as calibration and source check documentation, are maintained as quality-related decommissioning records.

### 3.1.5 CONCLUSION

In summary, several general overall conclusions regarding the site characterization survey can be made about the four sections: structures, systems, activation, and environment. First, plant structures contain radioactive material that will require removal prior to license termination. The contamination consists of radioactive material incorporated (fixed) into the upper layer of concrete/block and deposited on the surface (loose). Although the levels of radioactivity are generally low, structures within what was the RCA in 1993, including building surfaces and piping, are considered potentially contaminated and will require, as a minimum, a wipe or wash down.

Second, some plant systems contain deposited radioactive material due to plant operation. The majority of the radioactive material is contained in RCS piping and systems directly connected to the RCS (e.g., CVCS, safety injection system, and residual heat removal [RHR] system). Although some systems contain contamination, the systems are not expected to be greater than Class A waste.

Third, activated components contain the vast majority of the radioactive material not contained in fuel. Most activity is primarily concentrated in the vessel internals and shield wall. The reactor vessel lower internals contain the highest activity. Although radionuclide distributions are provided for the reactor vessel and vessel internals, they will have been removed before final survey data collection begins in the Containment. Neutron activation products have been found in samples of containment concrete in various structures, including the reactor vessel shield wall, steam generator missile shields, and the containment wall itself. Remediation of the activated components will be required to meet the site release criteria and facilitate license termination.

Fourth, and finally, the environmental survey results indicated that no radioactive material requiring remediation is present in the various materials sampled, and that no radioactivity requiring remediation has been spread to the environment outside the TNP industrial area. The final survey may require additional background data for a number of the sample media. Preliminary results indicate no radioactivity at TNP has been spread to the environment inside the industrial area in quantities requiring remediation.

## 3.2 RADIATION PROTECTION PROGRAM

### 3.2.1 INTRODUCTION

This section summarizes the Radiation Protection Program that will be used during decommissioning of TNP. The TNP Radiation Protection Program implements the regulatory requirements of 10 CFR 20, "Standards for protection against radiation," through plant procedures established to maintain radiation exposures ALARA. These ALARA principles are incorporated into the Radiation Protection Program.

TNP's current Radiation Protection Program, which maintains the essential elements of the Radiation Protection Program used when TNP was licensed for power operations, will continue to be implemented during decommissioning through existing TNP procedures. Changes to the Radiation Protection Program and implementing procedures will be made as necessary to support decommissioning activities.

### 3.2.2 RADIATION PROTECTION OBJECTIVE

The primary objective of the Radiation Protection Program is to be prepared to protect workers, visitors, and the general public from radiological hazards that have the potential of developing during plant activities. PGE provides qualified staff, facilities, and equipment to maintain TNP in a radiologically safe manner. PGE is committed to compliance with regulatory requirements, radiation dose limits, and limits regarding release of radioactive materials. Plant and radiation protection implementing procedures reflect PGE's policy of maintaining radiation doses and releases of radioactive materials in effluents to unrestricted areas ALARA and ensuring the Radiation Protection Program objective is met.

### 3.2.3 RADIATION PROTECTION AND ALARA PROGRAM POLICIES

TNP's radiation protection policies are established by management based on radiation protection standards, and are expected to be followed in activities and decisions related to radiation protection and radiological controls. The policies and requirements of the Radiation Protection Program are described in plant and radiation protection implementing procedures.

It is the policy of PGE to maintain total radiation exposures to plant personnel and the general public ALARA. TNP's ALARA policy complies with the regulatory requirements of 10 CFR 20 and 10 CFR 50, "Domestic licensing of production and utilization facilities," as well as the applicable guidance provided in Regulatory Guides 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable," Revision 3, and 8.10, "Operating Philosophy for Maintaining Occupational Exposures As Low As Is Reasonably Achievable," Revision 1.

Radiological hazards are monitored and evaluated on a routine basis to maintain radiation exposures and the release of radioactive materials to unrestricted areas ALARA. Air sampling is required whenever work which could cause the generation of airborne radioactivity is performed.

Local grab sampling equipment or continuous air monitors are available to measure airborne radioactivity. Radiation protection training is provided to occupationally exposed individuals to ensure they understand their responsibility to follow procedures and to maintain their radiation doses ALARA.

### 3.2.4 RADIATION PROTECTION ORGANIZATION

The PGE organizational structure that is used to implement the Radiation Protection Program during decommissioning is shown in Figure 3-37.

### 3.2.5 MANAGEMENT RESPONSIBILITIES

PGE management establishes specific ALARA goals and objectives for the Radiation Protection Program and ensures that work specifications, designs, and work packages involving radiation exposure or handling of radioactive materials incorporate effective radiological controls. Implementation of specific ALARA actions, as incorporated into daily work activities, is the responsibility of each individual manager, supervisor, and worker. Specific management responsibilities for radiation protection and for maintaining exposures ALARA are summarized in this section.

#### 3.2.5.1 General Manager, Trojan Plant

The General Manager, Trojan Plant has the ultimate responsibility for assuring that an effective Radiation Protection Program is implemented and is responsible for ensuring a coordinated and effective approach to the minimization of individual and collective dose and the control of radioactive material. He is also responsible for establishing the goals and objectives for the plant and for ensuring that personnel comply with radiation protection requirements.

#### 3.2.5.2 Manager, Personnel/Radiation Protection

TNP Radiation Protection Program development and implementation is under the direct authority of the Manager, Personnel/Radiation Protection, who reports to the General Manager, Trojan Plant. The Manager, Personnel/Radiation Protection is responsible for ensuring that radiation protection activities of the Radiation Protection Department are effectively implemented and comply with the Facility Operating (Possession Only) License NPF-1, regulatory requirements, and TNP procedures. The Manager, Personnel/Radiation Protection has overall responsibility for ensuring that radiation exposures are ALARA.

#### 3.2.5.3 Engineering Management

Engineering Management is responsible for ensuring that radiation exposures during planned engineering activities are maintained ALARA, and that engineering personnel comply with radiation protection requirements and maintain their radiation exposures ALARA.

#### 3.2.5.4 Managers and Supervisors

Managers and supervisors have responsibilities related to the Radiation Protection Program including the following:

1. Maintaining ALARA awareness and cooperating with Radiation Protection to provide individual personnel with the understanding and the means to minimize their own exposures;

2. Ensuring that personnel assigned to work with radioactive material attend required training; and

3. Ensuring personnel under their direction comply with radiation protection requirements.

### 3.2.6 RADIATION PROTECTION PROGRAM IMPLEMENTATION

The purpose of this section is to summarize how TNP's Radiation Protection Program will be implemented during decommissioning to maintain radiation exposure ALARA. The Radiation Protection Program is implemented and audited in accordance with approved plant procedures. Additional details concerning TNP's Radiation Protection Program are provided in the DSAR and in radiation protection implementing procedures.

#### 3.2.6.1 Radiation Protection Equipment and Instrumentation

The various equipment and instrumentation for conducting radiation surveys and measuring and minimizing personnel exposure are summarized in this section. Additional information on the facilities for radiation protection activities, and the procedures and equipment employed for measuring and minimizing personnel exposure, is provided in the DSAR.

##### 3.2.6.1.1 Laboratory Radiation Protection Instrumentation

The laboratory-type radiation instrumentation includes the following:

1. High-resolution solid-state detector(s) provided with lead shielding and a multichannel analyzer;

2. A beta counting system using Geiger-Mueller detectors;

3. A liquid scintillation counter;

4. A low-background thin-window gas flow proportional counter; and

5. An alpha counting system (solid-state detector).

Counting efficiencies of laboratory radiation detectors have been determined with certified radionuclide standards. A periodic calibration check is performed to check the efficiency of "in use" laboratory radiation detectors. Additional detail regarding calibration, testing, and maintenance of laboratory radiation protection instrumentation is provided in the DSAR and in radiation protection implementing procedures.

3.2.6.1.2 Portable Radiation Detection Instrumentation

The portable radiation detection instrumentation available for use within the plant includes the following:

1. Alpha detectors having count rate output;

2. Ionization chamber instruments equipped with a beta window and correction factor for beta measurement; and

3. Wide-range Geiger-Mueller instruments having dose rate and count rate output.

Additional detail regarding the use, storage, calibration, testing, and maintenance of portable radiation detection instrumentation is provided in the DSAR and in radiation protection implementing procedures.

3.2.6.1.3 Portable Air Sampling Instrumentation

The portable air sampling instrumentation includes the following:

1. Grab air samplers; and

2. Air samplers equipped with a filter and detector for the collection and counting of particulates.

The grab air samplers are used to collect samples of radioactive particulates for subsequent analysis in the laboratory. Periodic sampling of localized areas is conducted prior to and during personnel entry in accordance with radiation protection implementing procedures.

The air monitors collect and measure gross activity concentrations of airborne radioactivity. These monitors are stationed as necessary in airborne radioactivity areas during personnel occupancy, and will warn of increasing radioactivity levels. The air monitors can also be employed for routine surveys of gross airborne radioactivity levels throughout the plant.

Additional detail regarding the use, storage, calibration, testing, and maintenance of portable air sampling instrumentation is provided in the DSAR and in radiation protection implementing procedures.

3.2.6.1.4 Personnel Radiation Monitoring Instrumentation

Worker radiation monitoring instrumentation includes the following:

1. Direct-reading pocket ion chambers;
2. Thermoluminescent dosimeters (TLDs); and
3. Digital alarming dosimeters (DADs).

Additional detail regarding the use, storage, calibration, testing, and maintenance of personnel radiation monitoring instrumentation is provided in the DSAR and in radiation protection implementing procedures.

3.2.6.1.5 Area Radiation Monitoring Instrumentation

The ARMS supplements the personnel and area radiation monitoring of the plant Radiation Protection Program. Radiation detectors provide local and/or remote indication and alarm of direct radiation dose rate. The ARMS measures radiation levels over the range of $1.0 \times 10^{-1}$ to $1.0 \times 10^{7}$ mR/hr. The ARMS, as well as temporary area radiation monitoring instrumentation, are available as necessary to monitor the SFP and support decontamination and dismantlement activities during the decommissioning of TNP.

The radiation monitors have been calibrated by the manufacturer. The manufacturer's calibration is traceable to certified National Bureau of Standards/National Institute of Standards and Technology or commercial radionuclide standards. Following repairs or modifications, the monitors are recalibrated at the plant with the secondary radionuclide standards. Additional details on the use, calibration, testing, and maintenance of area radiation monitoring instrumentation are provided in radiation protection implementing procedures.

3.2.6.2 <u>Control of Radiation Exposure to the Public</u>

3.2.6.2.1 Radiological Effluent Monitoring

The ODCM contains the Radioactive Effluent Controls Program required by TNP Technical Specifications. Implementation of this program ensures compliance with the requirements of 10 CFR 50.36a, "Technical specifications on effluents from nuclear power reactors;" 10 CFR 20; 10 CFR 50 Appendix I, "Numerical guides for design objectives and limiting conditions for operation to meet the criterion "as low as is reasonably achievable" for radioactive material in light-water-cooled nuclear power reactor effluents;" and 40 CFR 190, "Environmental radiation protection standards for nuclear power operations."

Installed and temporary process and effluent monitoring systems necessary to support decommissioning activities are described in Section 2.3.2.19. These systems monitor liquid and airborne effluent discharges from the plant per the requirements of the TNP Radiological Effluent Controls Program. The effluent sampling and analysis schedules comply with the NRC positions described in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents for Light-Water-Cooled Nuclear Power Plants," Revision 1. Additional details of the policy, methods, frequency, and procedures for effluent monitoring are provided in the ODCM and radiation protection implementing procedures.

3.2.6.2.2 Radiological Environmental Monitoring

Monitoring, sampling, analyzing, and reporting radiation and radionuclides in the environment is performed in accordance with the Radiological Environmental Monitoring Program, which is required by TNP Technical Specifications and is incorporated into the ODCM. The Radiological Environmental Monitoring Program provides representative measurements of radioactivity in the highest potential exposure pathways and verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The Radiological Environmental Monitoring Program is periodically reviewed to address changing plant conditions and regulatory requirements in accordance with plant procedures. Additional details of the policy, methods, and procedures associated with the Radiological Environmental Monitoring Program are provided in the ODCM and radiation protection implementing procedures.

3.2.6.3 Control of Personnel Radiation Exposure

Personnel radiation exposure is maintained ALARA by a combination of shielding, access control, contamination control, surveys and monitoring, work planning, training, and sound radiation protection practices implemented by TNP plant procedures. As specified in TNP Technical Specifications, the procedures for personnel radiation protection are prepared consistent with the requirements of 10 CFR 20 and are approved, maintained, and adhered to for activities involving personnel radiation exposure.

3.2.6.3.1 Shielding

The objective of facility radiation shielding is to reduce external doses to plant personnel, in conjunction with a program for controlling personnel access and occupancy in radiation areas, to levels which are both ALARA and within the regulations defined in 10 CFR 20. Radiation protection implementing procedures provide for evaluation of the use of temporary shielding for activities involving high dose rates.

3.2.6.3.2 Access Control and Area Designations

In general, access to plant buildings and the Industrial Area is controlled by locked doors or gates. Radiologically controlled access within the Industrial Area of the plant is determined by the radiation level, the degree of contamination, or the presence of radioactive materials in the various areas.

A RCA is an area where access is controlled for the purpose of protecting individuals from exposure to radiation. Within the RCA, access to areas of higher radiation or contamination levels is further controlled and defined in accordance with 10 CFR 20 and radiation protection implementing procedures. Plant procedures also describe the requirements for radiological postings advising workers of potential radiological hazards at the entrance and boundaries of radiologically controlled areas.

3.2.6.3.3 Facility Contamination Control

Plant and radiation protection implementing procedures direct the use of various practices and equipment to ensure general plant area contamination is controlled at the source to the greatest extent possible. Additional contamination controls are specified for jobs involving high levels of contamination (e.g., a double step-off pad, additional surveys, etc.). Appropriate contamination controls are used when carrying contaminated tools and equipment between areas. Geiger-Mueller count rate meters (friskers) are located within the plant so that personnel can determine if they have been contaminated prior to entering another area of the plant. The final checkpoint for personnel leaving controlled areas of the plant is the access control point. Temporary exit points may be established at remote control areas as needed.

Airborne contamination is minimized by minimizing loose contamination levels and their sources. The use of installed and temporary ventilation systems prevents the build-up of air contamination concentrations. These systems are described further in Sections 2.2.3.3, 2.3.2.12, 2.3.2.13, and 2.3.2.14.

Additional details on the policy and methods for controlling general area and airborne contamination are contained in radiation protection implementing procedures.

3.2.6.3.4 Personnel Contamination Control

Contamination of personnel is controlled by the use of several types of protective clothing when entering contaminated areas. In the event that levels of airborne contamination approach or exceed applicable limits, provision is made for personnel to use respiratory protective equipment. Allowances are made for the use of respiratory protective equipment, as specifically authorized by the NRC, in determining whether individuals in restricted areas are exposed to concentrations in excess of the values specified in 10 CFR 20. The use of respiratory protection equipment is consistent with the goal of maintaining the total effective dose to personnel ALARA.

Additional details on the policy and methods for controlling personnel contamination are contained in radiation protection implementing procedures.

3.2.6.3.5 Area Surveys

Radiation protection personnel perform routine radiation surveys of accessible areas of the plant. These surveys consist of contamination surveys, air samples, and external radiation measurements as appropriate for the specific area. Additionally, specific surveys are performed as needed for operational and maintenance functions involving potential exposure of personnel to radiation or radioactive materials. Specific activities requiring these non-routine surveys include initial system opening, equipment release to uncontrolled areas, and response to radiation alarms. Additional details on the policy, methods, frequencies, and requirements for conducting both routine and non-routine radiation surveys are contained in radiation protection implementing procedures. These procedures specify the types and suitability of instrumentation and methods to be employed when performing surveys and actions required when abnormal radiological conditions are discovered.

3.2.6.3.6 Personnel Monitoring

TLDs are worn by plant personnel within radiologically controlled areas to measure radiation dose. Personnel assigned TLDs are also required to wear a direct-reading pocket ion chamber or a DAD when entering RCAs. The internal deposition of radioactive materials in personnel working in RCAs of the plant is evaluated primarily by a whole body count. Urinalyses are performed on plant personnel involved in plant activities where airborne concentrations of tritium exceed the limits of 10 CFR 20.

Additional details on the policy, methods, and frequency of personnel monitoring are contained in radiation protection implementing procedures.

3.2.6.3.7 Radiation Work Permits

Work in RCAs is performed under the authorization of radiation work permits issued by radiation protection personnel. These permits state protective clothing and dosimetry requirements, monitoring requirements, and special notes or cautions pertinent to the job. These permits also specify the maximum contamination level, radiation level (including hot spot contact radiation level), and airborne radiation level for the worker to be entered under that radiation work permit, or will instruct the worker where to obtain such information. Additional details on the use of radiation work permits are contained in radiation protection implementing procedures.

3.2.6.3.8 Training

Workers requiring unescorted access to the Industrial Area receive General Employee Training which includes radiological protection fundamentals. Personnel who require access to RCAs at TNP receive radiation protection training in accordance with 10 CFR 19, "Notices, instructions and reports to workers: inspection and investigations," and commensurate with the individual's responsibilities. The training process and requirements for general employees and radiation workers are summarized in Section 2.7 and are described in plant and radiation protection implementing procedures.

In addition to radiation worker training, separate and detailed instruction in advanced radiation work practices is provided to those workers performing tasks that involve significant exposure to radiation or quantities of radioactive material. The need for specialized ALARA training is evaluated during ALARA reviews or radiation work permit preparation in accordance with radiation protection implementing procedures. Specialized ALARA training includes such items as mock-ups, dry runs, pre-job briefings, and other special training classes. Specialized respiratory protection training is required for radiation workers who use respiratory protection devices.

3.2.6.3.9 Controls, Practices, and Special Techniques

Radiation protection implementing procedures specify that during the planning phase for activities in high dose rate areas or high airborne areas, various engineering controls to minimize exposures should be evaluated and/or implemented. These engineering controls and practices include, but are not limited to: temporary shielding; remote surveillance equipment; multi-discipline input regarding ALARA goals; pre-job, in-progress, and post-job briefings; and adequate lighting, ventilation, work space, and work area accessibility. Additional details on the implementation of controls, practices, and other techniques that are used to meet the radiation protection standards of 10 CFR 20, including ALARA, are contained in radiation protection implementing procedures.

3.2.6.3.10 Radioactive Materials Safety

Equipment and fluids in certain plant systems became contaminated during plant operation. These contaminated materials, together with radioactive materials contained in spent fuel, sealed sources, and instrument calibration devices, can result in radiation exposure to plant personnel. Procedures, facilities, and equipment for handling, processing, and disposing of radioactive gaseous, liquid, and solid wastes are described in Sections 3.3.2 and 3.3.3. Recognized methods for the safe handling of radioactive materials will be implemented to maintain potential external and internal doses ALARA.

External doses are minimized by a combination of time, distance, and shielding considerations. Internal doses are minimized by the measurement and control of loose contamination. The materials safety program is defined by written radiation protection procedures. Additional details on the materials safety program are contained in radiation protection implementing procedures.

## 3.3 RADIOACTIVE WASTE MANAGEMENT

Radioactive waste management activities during the TNP decommissioning include activities related to spent fuel management; and gaseous, liquid, and solid radioactive waste processing and disposal.

Spent fuel will be stored onsite in accordance with the requirements of the TNP Technical Specifications. The processing and disposal of gaseous, liquid, and solid radioactive waste will be managed in accordance with the Radiation Protection Program, Process Control Program, ODCM, Radioactive Effluent Controls Program, Radiological Environmental Monitoring Program, and Storage Tank Radioactivity Monitoring Program.

TNP policy for control of radioactive wastes is to minimize the amount of waste material generated, and to maintain the discharge of radioactive material below the design objectives provided in the ODCM. To ensure waste minimization goals are achieved during decommissioning, radiation workers will receive training in waste minimization procedures and practices. The TNP radioactive waste control program defines responsibilities and provides guidance for the minimization of radioactive wastes.

### 3.3.1 SPENT FUEL MANAGEMENT PROGRAM

This section describes the program for management of the spent fuel at TNP during and following decommissioning until title and possession of the spent fuel is transferred to the DOE. The estimated costs and associated funding plan for implementation of the TNP fuel management program are described in Section 5.

The descriptions of the spent fuel management program and associated funding plan provided in this section and in Section 5, respectively, fulfill the requirements of 10 CFR 50.54, "Conditions of licenses," Paragraph (bb) (Reference 3-15), which stipulates that "nuclear power reactors licensed by the NRC...shall, within 2 years following permanent cessation of operation of the reactor..., submit written notification to the Commission...of the program by which the licensee intends to manage and provide funding for the management of all irradiated fuel at the reactor until title to the irradiated fuel and possession of the fuel is transferred to the Secretary of Energy for its ultimate disposal in a repository."

#### 3.3.1.1 Spent Fuel Management Program Description

The spent fuel (irradiated fuel) currently stored in the SFP will remain there until licensing and construction of an ISFSI is completed on the TNP site. Use of an ISFSI was determined to be the most economical method for the temporary storage of the TNP spent fuel until a DOE or other offsite facility is available. Relocation of the spent fuel, and other high-level radioactive waste stored in the pool, to the ISFSI would allow decontamination and dismantlement of structures, systems, and components throughout TNP to proceed without impacting the safe storage of the spent fuel. This action will allow earlier termination of TNP's Part 50 license.

The spent fuel, and other high-level radioactive waste stored in the pool, will be relocated to the ISFSI prior to:

1. Beginning decommissioning activities that have the potential to adversely affect the spent fuel pool and its contents; and

2. Beginning decommissioning of systems and components needed for moving and loading spent fuel into casks for storage in the ISFSI.

The ISFSI will include the capability to transfer spent fuel from a storage cask to a shipping cask for shipment directly to an offsite repository. Since the spent fuel pool will be decontaminated and dismantled prior to the shipment of spent fuel to a permanent offsite facility, the capability to transfer the fuel from a storage cask to a shipping cask using the SFP will no longer exist.

#### 3.3.1.2 Effects of Permanent Repository Schedule on Spent Fuel Management Plan

Under the terms of the "Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste" executed between PGE and the DOE, the DOE has the responsibility, following commencement of operation of a repository, to take title to and possession of the TNP spent fuel and high-level radioactive waste as expeditiously as practicable upon the request of PGE. The scope of the contract states, in part, that the services to be provided by DOE shall begin, after commencement of facility operations, not later than January 31, 1998. This contract clause provides the basis for the schedule forecast in the DOE's annual Acceptance Priority Ranking and Annual Capacity Report for receipt of spent fuel and/or high-level waste.

If a DOE facility is in operation as of January 31, 1998, then the first shipment from TNP would occur in 2002. Shipments are forecast to continue through 2018. This projection is based on DOE's 1991 Acceptance Priority Ranking (DOE/RW-0331P, December 1991), 1991 Annual Capacity Report (DOE/RW-0328P, December 1991), and an extrapolation beyond the 10-year DOE outlook. This schedule was used to develop the decommissioning cost estimate described in Section 5.

#### 3.3.1.3 Licensing Activities to Support the Spent Fuel Management Plan

PGE will file an application for a license for an ISFSI in accordance with 10 CFR 72, Subpart B, which specifies the NRC licensing requirements for the independent storage of spent fuel and high-level radioactive waste. PGE will also petition the State of Oregon to adopt rules allowing storage of spent fuel in an ISFSI. Transfer of spent fuel and high-level radioactive waste from the spent fuel pool to the ISFSI will commence after NRC issuance of the 10 CFR 72 license, and issuance of required State of Oregon approvals.

### 3.3.2 RADIOACTIVE WASTE PROCESSING

#### 3.3.2.1 Gaseous Radioactivity

Gaseous radioactivity is expected to be limited primarily to airborne radioactive particulates generated during decontamination and dismantlement activities.

Airborne radioactive particulates will be filtered through HEPA filters in the containment ventilation system, the Auxiliary Building and Fuel Building ventilation systems, and the Condensate Demineralizer Building ventilation system, portions of which will be maintained in operation during decontamination and dismantlement activities in those buildings (see Sections 2.2.3.3, 2.3.2.12, 2.3.2.13, and 2.3.2.14). Local temporary ventilation systems with HEPA filtration, or other approved alternate systems, may be used in lieu of or to supplement building ventilation for activities expected to result in the generation of airborne radioactive particulates.

Radioactive gaseous effluents will be monitored and release limits adhered to in accordance with the methodology and parameters in the ODCM.

#### 3.3.2.2 Liquid Radioactive Waste

Liquid radioactive waste will be generated as a result of draining, decontamination, and cutting processes during plant decommissioning.

Portions of the existing liquid radioactive waste treatment systems (plant effluent system, clean radioactive waste system, and dirty radioactive waste system) will be maintained in operation during decommissioning to process liquid radioactive wastes by filtering, demineralizing, and providing for holdup or decay of the radioactive wastes for the purpose of reducing the total radioactivity prior to release to the environment (see Sections 2.3.2.11, 2.3.2.17, and 2.3.2.18). Temporary liquid waste processing systems may also be used to process liquid radioactive waste.

Radioactive liquid effluents will be processed in accordance with the ODCM.

#### 3.3.2.3 Solid Radioactive Waste

Solid radioactive waste generated during decommissioning will include neutron-activated materials, contaminated materials, and radioactive wastes. Neutron-activated materials include the reactor pressure vessel, reactor vessel internals components, and the concrete biological shield. Contaminated material and radioactive wastes include pipe sections, valves, tanks, other plant equipment, concrete surfaces, contaminated air filters, wet solid wastes from the processing of contaminated water volumes (ion exchange resins, cartridge filters), and dry solid wastes (rags and wipes, plastic sheeting, contaminated tools, disposable protective clothing).

The solid radioactive waste system spent resin transfer system, filter handling vehicle, solid waste compactor, and spent resin compactor will be maintained in operation as necessary during decommissioning to process solid waste (see Section 2.3.2.17). Temporary solid waste processing systems may also be used.

Solid radioactive waste will be processed in accordance with the TNP Radiation Protection Program, Process Control Program, and plant procedures. The Process Control Program provides requirements for processing radioactive wastes requiring solidification, radioactive wastes requiring high-integrity containers, and low activity dewatered resins and other wet wastes to ensure that shipping and burial ground requirements are met with respect to solidification and dewatering. To the maximum extent practicable, solid radioactive waste will be decontaminated and compacted to reduce the volume to be packaged for shipment to an offsite disposal facility.

Waste container selection will be determined by the type, size, weight, classification, and activity level of the material to be packaged. Examples of containers used at TNP include drums, metal boxes, C-vans (container vans), and high-integrity containers. Other special containers may be used as required.

#### 3.3.2.4 Mixed Wastes

Mixed wastes are wastes that contain both a hazardous waste component regulated under Subtitle C of the Resource Conservation and Recovery Act and a radioactive component consisting of source, special nuclear, or byproduct material regulated under the Atomic Energy Act. Plant procedures provide guidance for the minimization, control, and storage of mixed waste in accordance with the Environmental Protection Agency (EPA) and NRC regulations. The use of potentially hazardous materials in radiologically controlled areas will be reviewed to minimize the generation of mixed waste. TNP currently has approximately 60 $ft^3$ of mixed waste stored onsite. Mixed waste will continue to be stored onsite until a permanent storage or disposal facility becomes available.

### 3.3.3 RADIOACTIVE WASTE DISPOSAL

Radioactive waste will be appropriately packaged and will either be shipped to an offsite processing facility, shipped directly to a low-level radioactive waste disposal facility, or otherwise handled in accordance with applicable regulations. Packaging, storage, and shipment of radioactive waste generated during decommissioning will be controlled by the TNP Radiation Protection and Process Control Programs, and plant procedures. Plant procedures include requirements for:

1. Sorting and segregation of radioactive waste, and processing to an acceptable form;

2. Classification of radioactive waste in accordance with Department of Transportation (DOT) and NRC requirements;

3. Packaging, labeling, and marking of radioactive waste in accordance with DOT and disposal site criteria;

4. Storage of radioactive waste;

5. Receipt survey of vehicles used to transport radioactive waste;

6. Contamination surveys to ensure packages shipped meet DOT requirements for smearable contamination levels;

7. Radiation surveys, e.g., package contact, vehicle contact, specified distances from the package and the vehicle, and normally occupied positions in the vehicle cab for the material and package and for the transport vehicle depending on the type of shipment (e.g, low-specific activity, exclusive-use low-specific activity, etc.);

8. Shipment of radioactive waste in accordance with DOT and NRC requirements; and

9. Disposal and offsite volume reduction arrangements.

Radioactive waste storage facilities onsite include the Condensate Demineralizer Building, and, depending on the type and radiation levels, may also include areas of the Auxiliary and Fuel Buildings. Other temporary radioactive waste storage areas may be established as necessary.

A projection of radioactive waste generation (projected waste volumes, radionuclide concentrations, waste forms, and classification) is contained in Sections 2.5.2 and/or 3.1.3.

## 3.4 EVENT ANALYSIS

### 3.4.1 OVERVIEW

This section presents the results of evaluations and analyses of postulated decommissioning events and evaluates the potential for adverse effects on public health and safety. This evaluation includes postulated events that could be significantly different from accidents that have previously been evaluated for plant operations or maintenance. The analyses consider events related to decommissioning activities, loss of support systems, internal events, and external phenomena. The results of the analyses indicate that decommissioning activities can be conducted in a manner that does not significantly affect public health and safety. Section 3.4.5 discusses radiological occupational safety during decommissioning.

### 3.4.2 INTRODUCTION

The evaluation methodology was developed by generally following the methodology presented in NUREG/CR-0130, "Technology, Safety and Costs of Decommissioning a Referenced Pressurized Water Reactor." Decommissioning activities and other occurrences which could cause radiological events with the potential for release of radioactive material beyond the Exclusion Area Boundary were identified.

Potential accidents involving the storage and handling of spent fuel are not within the scope of these analyses. TNP spent fuel is currently stored in the spent fuel pool. Potential accidents involving the storage and handling of spent fuel are addressed in the DSAR. Spent fuel will later be removed from the spent fuel pool and transferred to an ISFSI. Postulated accidents involving the transfer of spent fuel to the ISFSI and potential interactions with decommissioning activities will be addressed as part of the license submittal for construction and operation of an ISFSI in accordance with 10 CFR 72.

The decommissioning activities evaluated included events with the potential for liquid and/or airborne radioactive releases. During decommissioning activities, contaminated liquids will primarily be generated inside buildings. If there is leakage of these contaminated liquids, the liquid would flow to the floor and equipment drain system, and would be disposed of through the normal plant discharge system, which is a monitored release pathway. Engineering and/or administrative controls will be established to ensure the quantity of radioactive liquids stored within plant buildings does not exceed the capacity of the available liquid waste processing equipment. Temporary tanks will be used when installed equipment is removed or unavailable.

In general, the volume of water generated by decontamination efforts is expected to be relatively small. A disposable liner could be used in an existing sump cavity to collect any water. A portable pump could be used to transfer the water to a temporary holding tank for sampling and processing prior to final discharge. As is the case with the installed system, a tank level alarm and indication system could be used to notify Operations personnel when the tank is filled to some pre-determined level.

Section 2.2.3.8 describes that temporary systems for decommissioning support may be utilized, including possible use of temporary liquid processing systems. As described in this section, plant design change procedures would control temporary modifications to plant structures, systems, and components.

Administrative controls contain restrictions on the amounts of radioactive material that may be stored in temporary tanks (tanks used for temporary storage of contaminated liquids located exterior to buildings). These restrictions ensure that potential liquid releases from such temporary tanks are within the limits of 10 CFR 20, Appendix B, at the nearest potable water and surface water supply in unrestricted areas. Storage of liquid wastes during decommissioning activities will be subject to these same restrictions, or alternatively, the wastes will be stored such that releases would be contained by appropriate engineered features such as dikes, dams, and overflows routed to plant drains. Design of the engineered features as described will generally follow the guidance of Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," and Generic Letter 81-38, "Storage of Low-Level Radioactive Wastes at Power Reactor Sites" where applicable or appropriate. In all cases, good engineering practices will be followed.

It was determined that potential consequences of such events are less than the calculated doses at the Exclusion Area Boundary from an airborne release of radioactive material. Consequently, only events with the potential for airborne releases are discussed in detail in this section.

The events that could involve airborne releases were then grouped into one of four categories:

1. Decommissioning activity events, including decontamination, dismantlement and materials handling events (Sections 3.4.4.1 through 3.4.4.3);

2. Loss of support system events, including loss of offsite power, cooling water, and compressed air (Section 3.4.4.4);

3. Fires and explosions (Sections 3.4.4.5 and 3.4.4.6); and,

4. External events (Section 3.4.4.7).

Each of these events was evaluated for its potential offsite radiological consequences. Each was evaluated to ensure that the resulting doses at the Exclusion Area Boundary would be less than 0.5 rem TEDE. Previous calculations have indicated that a 2.07 Curie release outside a building has the potential to result in a 0.5 rem dose at the Exclusion Area Boundary. The analyses in this section use this value as the limiting release quantity. The major assumptions that were used in the analyses are described in Section 3.4.3.2.

The sections describing each type of event contain a brief explanation of the reason a particular scenario was determined to be the bounding case. This discussion is followed by a summary of the analysis of the particular scenario.

### 3.4.3 LIMITS AND ASSUMPTIONS

#### 3.4.3.1 Radionuclide Release Limits

The EPA has established protective action guidelines, EPA 400-R-92-001, "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," October 1991, that specify the potential offsite dose levels at which actions should be taken to protect the health and safety of the public. The EPA protective action guidelines (PAGs) are limiting values based on the sum of the effective dose equivalent resulting from exposure to external sources and from the committed effective dose equivalent incurred from the significant inhalation pathways during the early phase of an event. The EPA PAG limits are:

| | EPA PAGs, rem |
|---|---|
| TEDE | 1 |
| Thyroid Committed Dose Equivalent (CDE) | 5 |
| Skin CDE | 50 |

Following permanent shutdown of Trojan, PGE analyzed the potential accidents that could occur in a permanently defueled state. PGE concluded there were no potential accident scenarios that could lead to Exclusion Area Boundary doses in excess of EPA PAGs. Based on the results of these analyses, PGE requested exemption from the offsite emergency preparedness requirements of 10 CFR 50.54(q). The NRC subsequently granted this exemption.

The Food and Drug Administration has established preventive PAGs for low impact protective actions at projected radiation doses of 0.5 rem TEDE, bone marrow CDE, or other organ CDE and 1.5 rem thyroid CDE.

PGE has conducted additional evaluations to ensure that decommissioning activities described in this plan will not create the potential for new or different events that could cause doses at the Exclusion Area Boundary to exceed preventive PAG levels.

To ensure that the maximum dose at the Exclusion Area Boundary would be maintained less than or equal to these limits, calculations were performed to determine the amount of radioactive material that would have to be released to result in a limiting dose of 0.5 rem at the Exclusion Area Boundary. This calculation was performed for three possible release locations.

| Release Location | Airborne Activity Limit |
|---|---|
| Inside a building with filtered ventilation | 2,840 Ci |
| Inside a building without filtered ventilation | 34.5 Ci |
| Outside a building | 2.07 Ci |

Airborne releases of the magnitude shown above potentially result in radiation exposures of 0.5 rem at the Exclusion Area Boundary using the conservative assumptions discussed in Section 3.4.3.2.

3.4.3.2 Assumptions

The following assumptions have been incorporated into the TNP decommissioning event analysis:

1. Dose calculations for releases of airborne activity from an event involving contaminated components are based on dislodging ten percent of the contamination, of which one percent becomes airborne (except $^{129}I$ and tritium, of which 100 percent of that dislodged becomes airborne.);

2. Airborne releases of radioactive materials are assumed to pass through a HEPA filtration system when the release originates inside a building with filtered ventilation. HEPA filter efficiency is assumed to be 99.97%;

3. For releases that originate inside a building without filtered ventilation, the guidance of Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," Revision 0, was used to obtain an activity decay rate of $6.00 \times 10^{-2}$/hr;

4. Conservative meteorological conditions (Pasquall class F with wind speed 1.13 m/sec) and release elevations are assumed;

5. Dose conversion factors were taken from EPA-520, Table 2.1, Revision 2.

6. Doses are early phase projections during the first two hours or less.

7. The north sector site boundary at 662 meters is the location for which doses are calculated.

8. Breathing rate is 3.33E-4 $m^3$/sec.

9. A plume depletion factor of 0.92 is used.

10. Credit was taken for diffusion due to building wake effect.

11. Isotopic concentrations found in the primary side of the Trojan steam generators projected to June 1994 were used for dose predictions. This isotopic mixture is conservative for contaminated components and activation of concrete components. The amount of activity for each isotope was determined as a percentage of the total activity present in the isotopic mixture. This percentage was then normalized to 1 curie, and the projected dose from an airborne release was calculated. The isotopic mixture of activated metallic components was not utilized since credible events did not result in airborne release of these materials.

### 3.4.4 RADIOLOGICAL EVENT IDENTIFICATION

Decommissioning activities and other occurrences which could cause radiological events that were considered include the following:

1. Decommissioning activity events, including decontamination, dismantlement, and materials handling events;

2. Loss of support system events, including loss of offsite power, cooling water, and compressed air;

3. Fires and explosions; and

4. External events (e.g., earthquakes, tornadoes, etc.).

Each of these types of events was evaluated for its potential offsite radiological consequences. The results of those evaluations are presented below.

Potential accidents involving the storage and handling of spent fuel are not within the scope of these analyses. Spent fuel from Trojan is currently stored in the onsite spent fuel pool. Potential accidents involving the storage and handling of spent fuel are addressed in the DSAR. Spent fuel will later be removed from the spent fuel pool and transferred to an ISFSI. Postulated accidents involving the transfer of spent fuel to the ISFSI and potential interactions with decommissioning activities will be addressed as part of the license submittal for construction and operation of an ISFSI in accordance with 10 CFR 72.

To ensure dismantlement activities will not impact the safe storage of spent fuel, dismantlement will be implemented in accordance with administrative controls that require, in part, an evaluation of activities in accordance with the requirements of 10 CFR 50.59. Work packages will include specific steps to physically protect the systems, structures, and components supporting spent fuel storage, or establish safe load paths and protective zones around these systems and structures.

Sources of radioactive material associated with decommissioning activities can be separated into two categories; contamination and activation products. Activation products are contained within structures and components and are therefore not available for airborne release during most credible events. Contamination is considered to be more readily available for airborne release and constitutes the major concern for offsite doses. Section 3.1 provides a more detailed discussion of the radiological characterization of the site.

3.4.4.1 Decontamination Events

Plant systems, structures, and components may be decontaminated in order to reduce worker radiation exposure rates or to allow release of materials from the plant. Decommissioning decontamination events relate to activities associated with hand-held water lance surface cleaning operations and chemical decontamination practices. The events could involve items such as:

1. Gross leakage of in situ decontamination equipment; and

2. Accidental spraying of liquids containing concentrated contamination.

Decontamination methods typically use liquids to remove radioactivity from the surface (e.g., chemical decontamination, high-pressure water washing). Contaminated liquid wastes generated during decommissioning operations will be sent to the plant liquid waste storage system or to other tanks that are designated for temporary storage. Administrative controls contain restrictions on the amounts of radioactive liquids that may be stored in temporary tanks. These restrictions ensure that potential liquid releases from such temporary tanks are within the limits of 10 CFR 20, Appendix B at the nearest potable water and surface water supply in unrestricted areas. Storage of liquid radioactive waste generated during decommissioning activities will be subject to these same restrictions, or alternatively, the wastes will be stored such that releases would be contained by appropriate engineered features such as dikes, dams, and overflows routed to plant drains.

Liquid radioactive wastes generated during decommissioning will be filtered and/or demineralized in a liquid radioactive waste system so that liquid releases remain within the limits established by the Facility Operating (Possession Only) License NPF-1, the ODCM, and 10 CFR 20.

3.4.4.1.1 In Situ Decontamination of Systems

Although large scale chemical decontamination of systems is not anticipated as part of TNP decommissioning, limited application may be used on systems or tanks to reduce radiation dose rates prior to dismantlement or general area decontamination.

Chemical decontamination is typically performed by recirculation of a decontamination solution throughout a system or tank until analysis of samples indicates that the desired decontamination level has been achieved. One system volume is used to minimize the quantity of contaminated liquids, thus an extended recirculation time of several days may be required. The system is typically heated to 80°C and may be pressurized.

This in situ decontamination method is performed on piping systems which form a closed loop. There is normally no interface between the circulating solution and the air outside the system. This provides a low potential for airborne release outside the system. However, any fluid system has the potential for leaks.

To provide a bounding analysis for chemical decontamination the methodology and assumptions of NUREG/CR-0130 are applied. The assumptions used in the calculation are:

1. Chemical decontamination of any system or component will not exceed one week of continuous operation;

2. A leak rate of 1 gpm (3.8 l/min) may go undetected during decontamination operation;

3. The decontamination solution density is approximately that of water (62.4 $lb/ft^3$ or 1 $g/cm^3$);

4. Airborne droplet concentration generated due to stream or drop type leaks is assumed to be 10 $mg/m^3$ (this results in an airborne release fraction of $10^{-5}$); and,

5. For spray type leaks, a maximum of 0.3% of the solution will be in the size range that could be transported as airborne activity.

A leak during chemical decontamination would be in the form of a stream, drops, or spray. At 80°C, it is anticipated that a solution is in the vapor phase. A stream or drops of liquid do not present a significant airborne release potential. Using the above assumptions, a maximum concentration (Ci/l) for decontamination solutions can be determined to ensure the releases do not result in conditions which would exceed applicable limits. No credit is taken for building ventilation or filtration.

For conservatism, it was assumed that 2.07 Ci (the limit for an outside release location) is the maximum airborne activity limit. The maximum solution activity associated with a decontamination leak is calculated by the equation:

$$\text{Maximum Concentration} = \frac{\text{Airborne Activity Limit}}{\text{(Leak Rate)(Airborne Release Fraction)(Leak Duration)}}$$

Stream or Drops Type Leak

For leaks in the form of streams or drops, using a release fraction of $10^{-5}$ and a leak duration of one week, the maximum concentration of the decontamination solution is:

$$\text{Maximum Concentration} = \frac{2.07 \text{ Ci}}{(3.8 \text{ l/min})(10^{-5})(10{,}080 \text{ min})}$$

Maximum Concentration = 5.4 Ci/l (20.5 Ci/gal)

Assuming the entire system contamination inventory was contained within the volume of fluid released (10,080 gal), systems containing $2.1 \times 10^5$ Ci would not result in airborne activities in excess of applicable limits. Excluding the reactor vessel internals, there are no single systems or structures that contain activity of this magnitude.

Spray Type Leak

A spray type leak would provide the maximum airborne release potential since the contamination would already be airborne and can be immediately entrained in local airflow. Assuming an airborne release fraction of 0.3% (0.003) and a leak duration of one week, the maximum concentration of the decontamination solution is:

$$\text{Maximum Concentration} = \frac{2.07 \text{ Ci}}{(3.8 \text{ l/min})(0.003)(10{,}080 \text{ min})}$$

Maximum Concentration = $1.8 \times 10^4$ µCi/l ($6.8 \times 10^4$ µCi/gal).

Assuming the entire contamination was contained within the volume of fluid released (10,080 gal), system activities 690 Ci would not result in airborne activities in excess of applicable limits. Chemical decontamination on systems which contain 690 Ci will not result in airborne release activities which exceed applicable limits. There are no single systems at TNP with contamination levels as great as 690 Ci.

The above calculations are conservative. They assume that the entire decontamination solution, up to a maximum of 10,080 gal (1 gpm for 1 week) are released to the environment, as spray. No credit is taken for operator action. Spray type leaks could be terminated by securing recirculation and depressurizing the system. For stream or drop type leaks, system depressurization should result in decreased leak rate. Additional measures such as leak isolation or collection could be used to further minimize consequences.

3.4.4.1.2 Surface Cleaning Techniques

Several different techniques can be employed in decontamination of surfaces. These typically include wiping, washing, vacuuming, and water jets (e.g., the use of hand-held high-pressure water jets). The high-pressure water cleaning is considered to provide the greatest potential for airborne activity generation and is discussed below.

The principle mechanism for airborne activity entrainment is the suspension of liquid droplets containing contamination. When the lance is sprayed initially, it produces droplets up to 300 μm in size, the size of fogs or mists. The spray particles then break into smaller particles when they impact on a surface. Thus there are significant amounts of small airborne droplets with considerable variation in the amount airborne.

Direct data is not available to define the quantity of droplets formed. A conservative estimate is made by assuming that a sufficient quantity of droplets are generated to maintain an airborne liquid concentration of 10 mg/m$^3$ with vigorous mixing in air. This is the maximum mass concentration that is found in air velocities less than 0.046 m/s. There is fairly constant weight in a spray of 10 to 20 μm particles of about 10 mg/m$^3$ even with gross entrainment of larger particles. Since 10 μm particles and smaller are in the respirable size range, they are potentially hazardous. The quantity of radioactivity in these airborne droplets is influenced by many factors such as the quantity of radioactivity on the surface, the ability of the liquid to entrain radioactivity, and the contact between the liquid and the surface.

The operating parameters for the high-pressure spray vary with the requirements of the situation. For the surfaces considered in this analysis, 23 to 30 l/min and 46.5 to 65 m$^2$/hr are reasonable estimates of the range of solution flow rate and surface cleaning rate respectively. To determine the maximum release potential, the largest cleaning rate (65 m$^2$/hr) with the lowest solution flow rate (23 l/min) is used in the calculation.

Tables 3-4, 3-5, and 3-6 provide a summary of contamination levels of various systems, buildings, and components and estimated contaminated surface areas. The SFP contains the highest activity level of systems, components, or structures which could be expected to be decontaminated by high-pressure water lance. Using the methodology provided in NUREG/CR-0130 and the TNP specific values, the following values associated with water lance cleaning of the SFP can be calculated:

| | |
|---|---|
| Total Surface Area (m$^2$) | 24,071 |
| μCi/m$^2$ | 4150 |
| Contamination Removed (μCi/min) | 4.5 x 10$^3$ |
| Solution Concentration (μCi/l) | 195 |
| Solution Concentration (μCi/cm$^3$) | 0.195 |
| Airborne Concentration (μCi/m$^3$) | 1.95 x 10$^{-3}$ |
| Aerosol Generation Rate (μCi/min) | 52.8 |
| Total Operation Time (min) | 22,220 |
| Total Generated Airborne Activity | 1.2 Ci |

The above values are greater than could be expected based on several conservative assumptions. NUREG/CR-0130 assumes a decontamination efficiency of 90%; for conservatism 100% of the activity was assumed to be removed. The total airborne activity assumes that airborne activity generated remains suspended and is not removed by ventilation filtration, nor does it precipitate out for removal by water collection system. The above analysis also assumes water lance operations are continuous for approximately 370 hours and are not terminated upon detection of increased airborne activity. Even under these conservative assumptions, the 0.5 rem TEDE limits would not be exceeded.

3.4.4.2 Dismantlement Events

Dismantlement events relate to activities associated with segmenting of components or structures and the removal of concrete. The potential for airborne release for each of these activities must be reviewed separately.

3.4.4.2.1 Segmentation of Components or Structures

Segmentation of components or structures can be accomplished by disassembly, or cutting or other destructive methods.

Disassembly of components or structures does not result in destruction of material. The potential for radioactive material release is limited to dislodging contamination. Disassembly events are therefore considered bounded by the material handling event discussed in Section 3.4.4.3.

Segmentation of components or structures by cutting or other destructive methods (e.g., sawing, grinding, or plasma cutting) can result in releasing airborne activation products in addition to dislodging contamination. Detailed planning of dismantlement activities will ensure that systems, structures, and components that are contaminated or activated will be dismantled using methods that minimize the release and spread of contamination.

Although activated metallic components contain the greatest activity levels, the potential for airborne release is limited to small fractions of material directly affected by cutting operations. The reactor vessel internal components, containing the largest quantity of activated materials, were removed intact with the reactor vessel for off-site disposal in 1999.

The dismantlement of the RCS piping is considered to provide the bounding analysis for generation of airborne activity. Dismantlement activities cannot be performed underwater, as may be done with the reactor vessel internals, and the RCS piping contains the greatest contamination of the systems to be dismantled, 221 Ci. The guidance provided by NUREG/CR-0130 was used to determine the amount of activity that could be generated during segmentation. To determine the total activity generated from segmentation the following equation was used:

Total Activity Generated = (Surface Contamination Level)(Kerf width) x
$\pi$ (Length of ID per cut)(Number of Cuts)

To determine the maximum generated activity the following values were used:

Surface contamination activity for the RCS was 55 $\mu Ci/cm^2$. This is the highest activity level calculated for the RCS and was used for determining contamination levels of CRDMs and reactor coolant pumps.

Kerf width was 0.95 cm. This is conservative since it is the largest kerf of possible cutting methods that may be employed.

The length of ID per cut 78.7 cm (31.7 in). This is conservative since it is the largest sized section of piping.

The number of cuts is assumed to be 55. This is an estimated number of cuts needed to segment the RCS piping and is considered conservative for this analysis since the airborne release during a segmentation event would only involve the cuts being performed just prior to the release.

Using the above equation a maximum release to the Containment Building atmosphere is calculated to be 0.71 Ci. This activity is conservatively assumed to become airborne. This limit is significantly below the 2,840 Ci limit for a release from inside a building with filtered ventilation such as the containment building and is well below the 2.07 Ci assumed for releases outside a building. Based on this analysis it is concluded that dismantlement activities cannot result in events which would exceed the 0.5 rem TEDE limits.

3.4.4.2.2 Removal of Concrete

Three techniques are available for the removal of concrete. These are rock splitting, explosives, and electric or pneumatic hammers. Each of these techniques has a different potential for the release of airborne radioactivity and may be used within different areas of the plant.

The bounding analysis for concrete removal is considered to be the primary shield wall. The primary shield wall, located inside the Containment Building, contained an estimated 351 Ci of total activity one year after plant shutdown. For the purposes of this bounding analysis explosives are assumed to be used in dismantlement of the primary shield wall.

Airborne activity would be generated during the various activities of primary shield wall dismantlement. The airborne release potential for these activities is discussed below.

Long, deep holes would be drilled for the insertion of explosives. They would be drilled from the top of the primary shield wall, down, parallel to the inside of the concrete surface.

The primary shield wall contains an estimated 351 Ci within the 885,000 kg (1,950,000 lb) of concrete, resulting in an average activity of 0.397 $\mu$Ci/g. If the airborne concentration of dry material is assumed to be 10 mg/m$^3$, and it is conservatively assumed this can be dispersed throughout the $5.66 \times 10^4$ m$^3$ containment volume, then a maximum of 566 g of dust could be suspended. The maximum activity that could be airborne would be 225 $\mu$Ci. This maximum suspension of activity is well below the 2.07 Ci limit discussed in Section 3.4.3.1 for maintaining Exclusion Area Boundary exposures below 0.5 rem.

The above calculations are conservative since they do not account for local contamination control methods. These methods could include providing local tenting and HEPA filtration of the concrete drilling area, as well as dust minimization by spraying water on the area. Based on the above, it is not credible that drilling of concrete could result in a release which would exceed the applicable limits.

For the blasting of the primary shield wall, a postulated aerosol concentration of 100 $mg/m^3$ is used. This concentration is ten times that assumed in the discussion of drilling above. The entire volume of the Containment Building ($5.66 \times 10^4$ $m^3$) is considered available for suspension of material. The concrete is assumed to have 0.397 μCi/g of activity. Based on the above assumptions the total Containment Building atmosphere could contain 0.0023 Ci. This activity is significantly less than 2.07 Ci limit discussed in Section 3.4.3.1 for maintaining Exclusion Area Boundary exposures below 0.5 rem.

#### 3.4.4.3 Material Handling Events

Material handling events involve activities associated with lifting and transporting parts of systems, structures and components once removed from the facility. Material handling events encompass those events that could potentially occur during movement of radioactive materials from their installed location to a location outside of the structure. The events could involve such items as:

1. Dropping of contaminated components;
2. Dropping of concrete rubble; and
3. Dropping of filters or packages of particulate material.

##### 3.4.4.3.1 Dropping of Contaminated Components

Table 3-6 contains a listing of contaminated systems and provides activity levels of contamination. Radioactive material contained as activation products within the metal lattice matrix of plant components or structures is not considered to be releasable as airborne particles via a materials handling event. Therefore, no further consideration of materials handling events involving activated metal components is necessary. The highest contamination activity within a system (not including activation) is contained in the RCS piping, approximately 221 Ci.

The assumptions contained in Section 3.4.3.2 conservatively assume that 10% of the contained activity of a contaminated component becomes dislodged, of which one percent becomes airborne. To provide an extremely conservative analysis, it is assumed that the entire activity of the RCS piping is contained within a single component being handled. Using this 221 Ci as the bounding activity, 0.221 Ci could become airborne. This amount of airborne activity is below the limit of 2.07 Ci which was determined to result in offsite radiation exposures of 0.5 rem for a release location outside a building. Therefore, dropping a contaminated component can not result in Exclusion Area Boundary doses exceeding applicable limits.

3.4.4.3.2 Dropping of Concrete Rubble

The primary shield wall contains an estimated 351 Ci of activity. Due to physical and process constraints the primary shield wall will be dismantled and packaged in portions. It is not credible that the entire volume of concrete rubble associated with the primary shield wall would be involved in a drop event, however, it does provide a bounding case for the analysis. The assumptions contained in Section 3.4.3.2 conservatively assume that 10% of the contamination of a contaminated component becomes dislodged, of which one percent becomes airborne. The majority of activity in the primary shield wall consists of activation products which are not easily dislodged or as likely to become airborne as contamination on a surface. Therefore, it is conservative to assume 10% of the contained activity of the primary shield wall becomes dislodged, of which 1% becomes airborne. These assumptions would result in 0.351 Ci becoming airborne. This amount of airborne activity is below the limit of 2.07 Ci which was determined to result in Exclusion Area Boundary radiation exposures of 0.5 rem for a release location outside a building. Therefore, dropping of concrete rubble cannot result in Exclusion Area Boundary doses exceeding applicable limits.

3.4.4.3.3 Dropping of Filters or Packages of Particulate Material

Dropping of filters or packages containing particulate material has the potential to create an airborne release by dislodging material. To minimize this potential the Radiation Protection Program provides administrative controls on the packaging and movement of radioactive material which include:

1. Radioactive materials removed from contaminated areas will be contained, surveyed and labeled to allow appropriate control of the material;

2. Radioactive liquid samples or sources will be properly contained and should be transported by, or under the cognizance of, radiation protection personnel; and

3. Movement of radioactive material should be made by the most practical direct route.

The worst case scenario would involve dropping material in a radioactive waste storage area. This location is considered bounding since the potential release could include additional containers affected by the impact of the drop.

The consequences associated with the dropping of filters or packages of particulate material is considered bounded by the worst case fire scenario discussed in Section 3.4.4.5 for the following reasons:

1. The fire scenario is assumed to affect all containers within the storage area, whereas a drop event would be limited to the impact area; and

2. A fire results in greater release of airborne activity than drop events.

Therefore, dropping of filters or packages of particulate material will not result in a release of radioactive material that would exceed 0.5 rem limit as discussed in Section 3.4.3.1.

#### 3.4.4.4 Loss of Support System Events

Electric power, cooling water, and compressed air systems provide support to decommissioning activities. Loss of these systems could potentially affect many other systems and plant areas simultaneously. Each of these events is evaluated below.

##### 3.4.4.4.1 Loss of Offsite Power

Offsite power is used to energize tools, cranes, lighting, and air filtering equipment used during decommissioning activities. The following events can result from a loss of offsite power:

1. Tools, lighting, and air filtering equipment are de-energized; and

2. Cranes are de-energized.

A loss of power to tools and lighting being used for decommissioning will result in the interruption of work activities, but does not result in the release of radioactivity.

A loss of power to plant ventilation and filtering systems could result in the disruption of air flow paths and effective utilization of HEPA filters. In the event of loss of offsite power, work activities with the potential for airborne contamination will be suspended.

A loss of offsite power could result in loss of power to material handling equipment. Occupational Safety and Health Administration (OSHA) regulations require that crane hoisting units be equipped with a holding brake. A holding brake is a brake that automatically prevents motion when power is off. The Containment Building Polar Crane, Fuel Building Overhead Crane, Auxiliary Building Electric Hoist, Spent Fuel Pool Bridge Crane, and the Condensate Demineralizer Building Bridge Crane are equipped with holding brakes. Although loss of power is not expected to result in crane or hoists failure, this event would be bounded by the material handling events analyses provided in Section 3.4.4.3.

##### 3.4.4.4.2 Loss of Cooling Water

Cooling water may be supplied to air compressors and the decommissioning cutting equipment and tools. The following events result from a loss of cooling water:

1. Compressed air is lost if an alternate cooling water supply is not established to the station air compressors within a short time period. The consequences of a loss of compressed air are presented in Section 3.4.4.4.3; and

2. Cutting operations that use cooling water will stop. This does not adversely affect contamination control.

A loss of cooling water does not result in events leading to a significant release of radioactive material to the environment during decommissioning activities. Therefore, public health and safety are not adversely affected by a loss of cooling water event.

3.4.4.4.3 Loss of Compressed Air

Compressed air is supplied by the station air compressors to operate pneumatic valves and dampers and to power pneumatic tools. The following events occur upon a loss of compressed air:

1. The liquid discharge control valve for plant effluents fails in a closed position terminating liquid releases;

2. Decommissioning pneumatic tools shutdown. This terminates potential releases from activities using these tools; and

3. Pneumatic ventilation exhaust fan dampers fail in a closed position terminating airborne and gaseous release via those paths. Since this event is not postulated to occur coincident with an event involving abnormal releases of radioactive material, there would be no significant impact on offsite releases.

A loss of compressed air does not result in events leading to significant releases of radioactive material to the environment during decommissioning activities. Therefore, public health and safety are not adversely affected by a loss of compressed air event.

3.4.4.5 Fire Events

A fire event could affect several plant systems, structures, and components simultaneously. Combustible materials can be ignited by either external ignition sources (e.g., oxyacetylene torches) or internal ignition sources (e.g., spontaneous combustion). Adequate fire protection features will be maintained through implementation of the fire protection program discussed in Section 9, thereby minimizing the potential of occurrence of a fire. These features include:

1. Fire detection and suppression systems and equipment;

2. Fire barrier maintenance and control;

3. Personnel training and qualification;

4. Fire Protection Program procedures; and

5. Control of transient combustible materials and ignition sources.

Following permanent shutdown of TNP, a deactivation program was undertaken which included, in part, the removal of carbon from ventilation filters, the removal of oil from non-essential pumps and motors, and the deenergization of electrical power to non-essential equipment. This has reduced the amount of combustibles in the facility and the potential for fires.

A calculation was performed to determine the maximum expected release that would occur in the event of a fire. The worst case fire is assumed to occur in the Condensate Demineralizer Building, which stores low-level radioactive waste. A source term based on historical waste generation was assumed. This was considered conservative as discussed below.

Waste generated during decommissioning includes contaminated clothing, decontamination materials resulting from physical or chemical removal of radioactive contamination, work containment materials (plastic tents, etc.), contaminated residue from cutting and grinding, and contaminated components. These activities are basically similar to the types of work and materials generated during refueling and maintenance outages. The size of the work force that will be generating radioactive waste during decommissioning activities is expected to be less than the number typically employed during a refueling and maintenance outage. It is unlikely that a greater volume of combustible radioactive material will be generated during decommissioning than was present during plant operations. Wastes of this type will be produced at a slower rate than during normal outages and will be shipped offsite on a regular basis rather than accumulated over long periods of time, further reducing the probability of a large source term being available for conflagration.

The results of this analysis determined that the offsite dose from a worst case fire involving radioactive material was approximately 5 mrem whole body. This is well below the 0.5 rem limit as discussed in Section 3.4.3.1.

#### 3.4.4.6 Explosion Events

During decontamination and dismantlement activities portable gas bottles may be used in support of welding or cutting activities or as fuel for vehicles such as fork lifts. The use, movement, and storage of portable bottles of combustible gases (e. g., acetylene, hydrogen, and propane) in the plant will be controlled by administrative procedures. These procedures will establish requirements ensuring that the planned use, movement, or storage meets appropriate fire protection codes and safety standards. In addition, administrative procedures will control the use of ignition sources within the plant. These measures will serve to limit both the residence time of portable bottles within the plant and the potential for ignition sources in close proximity to the bottles.

NUREG/CR-5759, "Risk Analysis of Highly Combustible Gas Storage, Supply, and Distribution in PWR Plants" provides a discussion of the risks associated with the use of bottled gases in plants. The report states that a review of historical information for safety-related plant areas did not identify any incidents of explosions of bottles. The report further explains that based on discussions with explosion experts, the explosion of an individual bottle of hydrogen containing 200-250 standard cubic feet of gas in a confined space, could result in the breach of fire doors and concrete block walls. However, this would serve to dissipate the energy and no widespread damage would result. NUREG/CR-5759 concludes the risk to plant safety from the explosion of portable gas bottles is not significant.

The aforementioned administrative controls, in conjunction with the information provided in NUREG/CR-5759, provide the basis for a determination that appropriate measures are/will be provided to minimize the potential for explosion events. Given the limited potential for widespread damage as described in NUREG/CR-5759, it can be concluded that the consequences of the explosion event are bounded by the postulated fire event discussed in Section 3.4.4.5.

3.4.4.7 External Events

A review of external events was done to evaluate the effects of natural and manmade events on the radiological consequences of decommissioning activities. The hazards associated with these events are assumed to be consistent with those that could have occurred during TNP operation. Several external events were identified as having potential applicability to TNP decommissioning:

1. Earthquake;
2. External flooding;
3. Tornadoes and extreme winds;
4. Volcanic activity;
5. Lightning; and
6. Toxic chemical event.

Such events are of extremely low probability. A discussion for each of the above listed events follows.

3.4.4.7.1 Earthquake

A seismic event during decommissioning could initiate a materials handling event similar to those described in Section 3.4.4.3. The analysis in Section 3.4.4.3 concludes that the bounding material handling event results in an Exclusion Area Boundary dose that is significantly less than the 0.5 rem limit.

Structures whose failure during a seismic event could significantly affect the spent fuel pool or spent fuel integrity are seismically qualified. Decommissioning activities which could impact the seismic qualification of these structures/components will be evaluated. One of the purposes of these evaluations is to ensure that the dismantling activities do not result in a configuration that could fail during a seismic event collapsing onto or into the spent fuel pool or affect the seismic qualification of the spent fuel pool or spent fuel integrity. The consequences of a seismic event on the safe storage of spent fuel has already been analyzed with the results provided in Section 6.3 of the DSAR. This analysis concluded that a seismic event would not result in Exclusion Area Boundary doses which would exceed the applicable limits.

Following transfer of the spent nuclear fuel and high level radioactive waste to the ISFSI, seismic qualification of the spent fuel pool will no longer be required.

During the Large Component Removal Project, the Containment Building was detensioned and the opening in the south face was covered by a roll-up door. The door has since been removed and replaced with a sheet metal cover. An evaluation was performed on the Containment Building structural integrity in this configuration. It was concluded that under bounding environmental loading conditions, the Containment Building will remain stable and the reinforced concrete and liner plate integrity will be maintained.

3.4.4.7.2 Flooding

As discussed in the DSAR, the water surface level of the maximum flood level is calculated to be 41 ft mean sea level (MSL). The TNP yard elevation is 45 ft MSL. This level is sufficient to be considered safe from projected floods. Access to the Auxiliary Building, Fuel Building, Containment Building, and Condensate Demineralizer Building are above the maximum expected flood level. If storage of radioactive material outside of the structures becomes necessary, it will be limited to areas with an elevation or protection equivalent to an elevation of 45 ft MSL. Alternatively, the dedicated capability to relocate stored radioactive material to a protected elevation within 60 hours (minimum warning time for flood peak following failure of Grand Coulee dam) will be maintained. In the unlikely event that a maximum flood level was experienced during decommissioning, loss of offsite power is considered to be the only potentially significant resultant decommissioning event. The consequences associated with loss of offsite power were discussed in Section 3.4.4.4.1.

3.4.4.7.3 Tornadoes and Extreme Winds

Dismantling activities take place within structures that were designed to withstand credible meteorological conditions for the area. The Containment Building was modified by the Large Component Removal Project as described below.

During the Large Component Removal Project, the Containment Building was detensioned and the opening in the south face was covered by a roll-up door. The door has since been removed and replaced with a sheet metal cover. An evaluation was performed on the Containment Building structural integrity in this configuration. It was concluded that this configuration does not adversely affect the ability of the structure to withstand tornado forces. Postulated radiological releases as a result of breaching the opening door during a tornado or extremely high winds are bounded by other analyzed events.

Storage of radioactive material is normally limited to locations within the Fuel Building, Radwaste Annex, and Condensate Demineralizer Building. The storage of radioactive material will be administratively controlled to ensure adequate protection to prevent airborne releases. These administrative controls are contained in the Radiation Protection Program.

Based on the administrative controls, tornadoes and extreme winds are not expected to initiate conditions that would result in releases that would exceed 0.5 rem TEDE limit.

Tornadoes or extreme winds could initiate a loss of offsite power event. The analysis in Section 3.4.4.4.1 concludes that the 0.5 rem TEDE limit is not exceeded in the event of a loss of offsite power.

3.4.4.7.4 Volcanic Activity

Section 2.5.6 of the DSAR provides a discussion of the probability and credible effects of volcanic activity. The credible effects associated with volcanic activity are identified as:

1. Ash fall;

2. Air blast, debris avalanche and pyroclastic flows; and

3. Mud flow - flooding.

An impending ash fall at TNP (irrespective of volume) would activate preparations for cleanup and maintenance of necessary systems. If necessary, dismantling activities could be suspended to minimize activities that could result in airborne contamination. Ash fall is not considered to be an initiating event for any event resulting in offsite radiological release. The distance of TNP from areas of known volcanic activity makes damage from air blast, debris avalanche, and pyroclastic flows remote and is not considered a credible initiating event for decommissioning events.

Mud flow and flooding resulting from volcanic activity have been analyzed and peak flood levels in the TNP area are not expected to exceed 30 ft MSL. This level is less than that expected for the Probable Maximum Flood level discussed in Section 3.4.4.7.2.

3.4.4.7.5 Lightning

A lightning strike could result in the loss of offsite power or fire event. The loss of offsite power event is discussed in Section 3.4.4.4.1 and the effect of an onsite fire is provided in Section 3.4.4.5.

3.4.4.7.6 Toxic Chemical Event

Section 2.2.3.2 of the DSAR provides a description of toxic chemical hazards. Toxic chemicals are a personnel safety concern. In the event of a toxic chemical event affecting plant personnel, decommissioning activities would be suspended and personnel evacuated as necessary. A toxic chemical event has the potential to initiate a radiological event. The most severe radiological event that could be initiated would be if a personnel injury resulted in an event involving a loaded crane or hoist. A toxic chemical event is therefore considered an initiating event for a material handling event which is discussed in Section 3.4.4.3.

### 3.4.5 RADIOLOGICAL OCCUPATIONAL SAFETY

Radiological events could occur which result in increased exposure of decommissioning workers to radiation. However, the occurrences of these events are minimized or the consequences are mitigated through the implementation of the Radiation Protection Program and the Permanently Defueled Emergency Plan.

The Radiation Protection Program is applied to activities performed onsite involving radioactive materials. A primary objective of the Radiation Protection Program is to protect workers and visitors to the site from radiological hazards during decommissioning. The program requires PGE and its contractors to provide sufficient qualified staff, facilities, and equipment to perform decommissioning activities in a radiologically safe manner.

Activities conducted during decommissioning that have the potential for exposure of personnel to either radiation or radioactive materials will be managed by qualified individuals who will implement program requirements in accordance with established procedures. Radiological hazards will be monitored and evaluated to maintain radiation exposures ALARA.
The Radiation Protection Program at TNP implements administrative dose guidelines for TEDE to ensure personnel do not exceed federal 10 CFR 20 dose limits for occupational exposure to ionizing radiation. Radiation protection training will be provided to occupationally exposed individuals to ensure that they understand their responsibility to follow procedures and to maintain their individual radiation dose ALARA.

TNP work control procedures will ensure that work specifications, designs, work packages, and radiation work permits involving potential radiation exposure or handling of radioactive materials incorporate effective radiological controls.

The Permanently Defueled Emergency Plan retains onsite emergency response capability. This capability includes relocation of personnel from a radiologically affected area, if necessary.

Implementation of the Radiation Protection Program and the Permanently Defueled Emergency Plan ensures that potential radiological events affecting occupational health and safety will be sufficiently minimized and mitigated.

### 3.4.6 OFFSITE RADIOLOGICAL EVENTS

Offsite radiological events related to decommissioning activities are limited to those associated with the shipment of radioactive materials. Radioactive shipments will be made in accordance with applicable regulatory requirements. The radioactive waste management program and the Nuclear Quality Assurance Program assure compliance with these requirements. Compliance with these requirements ensures that both the probability of occurrence and the consequences of an offsite event do not significantly affect the public health and safety.

### 3.4.7 NONRADIOLOGICAL EVENTS

Decommissioning TNP may require different work activities than were typically conducted during normal plant operations. However, effective application of the TNP safety program to decommissioning activities will ensure worker safety. No decommissioning events were identified that would be initiated from nonradiological sources that could significantly impact public health and safety.

Hazardous materials handling will be controlled by the Hazardous Material Control Program using approved plant procedures. There are no chemicals stored onsite in quantities which, if released, could significantly threaten public health and safety.

Flammable gases stored onsite include combustible gases used for cutting and welding. Safe storage and use of these gases and other flammable materials is controlled through the Fire Protection Program and plant safety procedures.

Plant safety procedures and off-normal instructions have been established which would be implemented if a nonradiological event occurred at TNP. Implementation of these programs and procedures ensures that the probability of occurrence and consequence of onsite nonradiological events do not significantly affect occupational or public health and safety. Plant safety procedures provide personnel safety rules and responsibilities and control both chemical and hazardous waste identification, inventory, handling, storage, use, and disposal.

## 3.5 OCCUPATIONAL SAFETY

TNP has, and will continue to maintain, an industrial occupational safety program. The intent of the safety program is to ensure a safe and healthy working environment, maintain employee safety awareness, promote safety as an integral part of facility activities, and comply with occupational safety and health regulatory standards. Safety at TNP is viewed as part of the overall plant performance and is an integral part of the activities at the plant. Individuals requiring unescorted access to the Industrial Area receive training in plant safety and chemical safety as part of General Employee Training.

The TNP safety program is addressed in plant safety procedures, which reflect local, state, and federal safety codes and standards. The safety procedures outline acceptable safe work and housekeeping practices, ensure that employees are adequately trained and prepared to perform their jobs safely, and ensure that employees are adequately informed of chemical hazards they may encounter on the job. Personnel are responsible for maintaining a safe work environment.

Individuals assigned the responsibility for the industrial safety programs at the plant will meet with the plant staff on a periodic basis to discuss safety program activities, procedure changes, accident causes, safe procedures, and other subjects pertinent to promoting safety.

## 3.6 NONRADIOACTIVE WASTE MANAGEMENT

Nonradioactive regulated waste materials expected to be handled during decommissioning include asbestos, polychlorinated biphenyls (PCBs), mercury, and lead. Other nonradioactive waste materials include steel components (e.g., piping, valves), electrical components (e.g., wiring, motors), and structural materials (e.g., concrete, beams). Handling and disposal of nonradioactive regulated waste materials will be controlled by the TNP chemical safety program. This program provides for evaluation of regulated substances and approval of methods for their handling and disposal. Work will be done in accordance with the TNP work control process. This process ensures that decommissioning activities receive appropriate safety and technical reviews. The proposed work will be reviewed for ALARA concerns if the systems, structures, or components are in the RCA, and for coordination with other projects.

Nonradioactive waste materials will be transported by approved or licensed transporters as required, and shipped to permitted solid waste landfills or licensed hazardous waste facilities.

### 3.6.1 ASBESTOS

Asbestos containing materials include Marinite board, used in the plant as a fire barrier; electrical cable with a wrap containing asbestos; piping systems with a wrap containing asbestos; the cooling tower mist eliminators and distribution piping fabricated from an asbestos cement material; and roof flashing sealant containing asbestos fibers. Other materials that are suspected of containing asbestos will be sampled and analyzed before work is done on the material.

Asbestos material will be removed and disposed of in accordance with plant safety procedures, federal and state OSHA regulations, and federal and state hazardous air pollutant and solid waste regulations.

### 3.6.2 POLYCHLORINATED BIPHENYLS (PCB)

The rod control cabinet capacitors in the Control Building may contain a small amount (approximately two liters) of PCBs.

PCBs and PCB items will be handled and disposed of in accordance with federal and state PCB regulations.

### 3.6.3 MERCURY

Mercury is contained in some plant components. Vendor technical manuals and plant walkdowns will be used to identify components that contain mercury metal.

Mercury metal will be collected and sent for recycling as plant equipment is removed. If not sent for recycling or reclamation as scrap metal, the mercury will be disposed of as a hazardous waste in accordance with federal and state hazardous waste regulations.

### 3.6.4 LEAD

Lead is contained in lead based paints, which may have been used as a primer for some steel surfaces at TNP, and lead sheets used as a radiological shield material.

Lead containing materials will be removed and disposed of in accordance with plant safety procedures, OSHA regulations, and federal and state hazardous waste regulations. Nonradioactive metals with lead based paints or coatings will be sent as scrap metal to a dealer who will accept lead painted metal; otherwise they will be disposed of as hazardous waste. Sandblast materials used to remove lead based paints will be handled and disposed of as hazardous waste unless they pass a toxic characteristic leaching procedure test.

### 3.6.5 OTHER PLANT WASTE MATERIALS

Other plant waste materials, including batteries (e.g., lead-acid, nicad) and refrigerants from chillers and air conditioners, will be sent to a recycling facility, or disposed of in accordance with normal waste disposal practices for nonradioactive nonregulated solid waste. Some industrial solid wastes (e.g., treated wood poles) may need special permits before disposal in solid waste landfills.

## 3.7 REFERENCES FOR SECTION 3


3-1 Code of Federal Regulations, Title 10, Part 50.82, "Application for Termination of License," August 28, 1996.

3-2 Regulatory Guide 1.179, "Standard Format and Content of License Termination Plans for Nuclear Power Reactors," January 1999.

3-3 Regulatory Guide 1.86, "Termination of Operating Licenses for Nuclear Reactors," June 1974.

3-4 NUREG/CR-5849, "Manual for Conducting Radiological Surveys in Support of License Termination," June 1992.

3-5 Draft NUREG/CR-5512, "Residual Radioactive Contamination from Decommissioning," January 1990.

3-6 Code of Federal Regulations, Title 10, Part 20.1402, "Radiological Criteria for Unrestricted Use."

3-7 NUREG-1575, "Multi-Agency Radiation Survey and Site Investigation Manual (MARSSIM)," December 1997.

3-8 Portland General Electric Topical Report PGE-8010, "Trojan Nuclear Plant Quality Assurance Program."

3-9 Portland General Electric "Trojan Nuclear Plant Radiological Site Characterization Report," Revision 0.1, February 8, 1995

3-10 NUREG/CR-0130, "Technology, Safety and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station," June 1978.

3-11 NRC Letter, W. F. Kane to S. M. Quennoz, "Authorization of the Trojan Reactor Vessel Package for Transport," October 29, 1998.

3-12 Portland General Electric, "Trojan Nuclear Plant Updated Final Safety Analysis Report," Amendment 14, Volume 2, October 1990.

3-13 Portland General Electric, "Trojan Nuclear Plant Environmental Report," Amendment 3, July 24, 1972.

3-14 Cornforth Consultants Report to Portland General Electric, "Hydrogeology Evaluation – Trojan Nuclear Plant," October 2000.

3-15 Code of Federal Regulations, Title 10, Part 50.54, "Conditions of Licenses," Paragraph (bb).

# Appendix 3-1
# Summary of Notable Radiological Contamination Events

This appendix contains a description of the notable radiological contamination events that occurred during the operation of the Trojan Nuclear Plant. This information is not intended to be an all-inclusive listing, but rather provides summaries of the more significant operational event data that is used, as appropriate, to select additional sampling locations during site characterization.

**Date**: Various

**Description**:

Spills of radioactive liquids on the 45 ft elevation of the Containment Building occurred on several occasions during plant operation. Events include draining reactor coolant through a steam generator manway during maintenance. Other spills occurred during leaks from miscellaneous valves.

**Radiological Consequences**:

The radioactive water may have caused contamination of the concrete surfaces in the area. Fixed contamination requiring remediation may be present in the area. Additional samples/surveys, including concrete samples, are planned for this area.

**Date**: Various

**Description**:

The storm drain systems which discharged to Recreation Lake and the Columbia River were contaminated from a number of sources. For example, the oily water separator and start-up boiler, which are addressed later in this appendix, both contributed to past storm drain contamination. Other contamination sources included leaks in the electrical facade, contamination of building roof surfaces from main steam relief and air ejector discharges, and rain "rinse out" of activity releases from the PWST and RWST vents, which are also discussed later in this appendix.

**Radiological Consequences**:

Elevated levels of tritium on the river side drains, and $^{137}Cs$, $^{58}Co$, and $^{60}Co$ in sludge from the drains, were identified several years ago. Storm drains are still considered potentially contaminated from past discharges of slightly radioactive liquids. The drains are too small to allow access for surveys; however, they may be addressed under Phase II of site characterization or as part of the embedded pipe survey program. Samples of the water and sediment from the drain discharge at Recreation Lake were collected as part of the scoping survey and the results indicated no activity above background.

**Date**: 1975 to early 1980s

**Description**:

Steam generator primary-to-secondary leakage resulted in contamination of the secondary system, which was sampled in the secondary chemical laboratory. Secondary chemical laboratory drains were routed to the wastewater treatment plant.

**Radiological Consequences**:

Contaminated water was discharged to the waste treatment plant. The water from the treatment plant was sampled and then discharged to the Columbia River. Discharges were included in the Semiannual Effluent Release Reports. The system was modified to direct the drains to the oily water separator, which was a monitored release pathway.

**Date**: 1975 - 1979

**Description**:

Based on oral interviews, it appears that on several occasions, a radwaste system tank overflowed to the Auxiliary Building floors/sumps prior to 1980. The floors and walls were decontaminated (removable activity) and returned to access without protective clothing.

**Radiological Consequences**:

These events may have resulted in fixed contamination that will require remediation to meet the final survey release criteria. Operational survey data will be collected to determine the extent of contamination in this area. The radioactive material was confined to the building.

**Date**: 1978 - 1980

**Description**:

The plant start-up boiler was contaminated by primary-to-secondary leaks. The start-up boiler used makeup water from the condensate storage tank. The condensate storage tank was contaminated due to primary-to-secondary leakage. Start-up boiler blowdown was discharged to the pavement around the start-up boiler. The blowdown flowed to the storm drain and Recreation Lake. Contamination was found in the boiler and on the pavement near the boiler blowdown line. A plant modification changed the makeup source to demineralized water and rerouted blowdown to the discharge and dilution structure.

**Radiological Consequences**:

Blowdown from the start-up boiler caused contamination of the curb going to the storm drain and Recreation Lake. A concrete slab was removed and replaced with new concrete. The releases did not exceed effluent release limits. Scoping surveys indicated that these areas (i.e., pavement, start-up boiler, and Recreation Lake) do not contain radioactivity levels greater than background.

**Date**: Prior to 1980

**Description**:

The CVCS was over-pressurized releasing contaminated water to the 77 ft elevation of the Auxiliary Building.

**Radiological Consequences**:

The contaminated water sprayed onto untreated concrete and concrete block surfaces. The removable contamination was removed. Levels of fixed contamination remained in the concrete. The area was repainted and marked to indicate that fixed contamination was present under the paint. The levels of fixed contamination are not measurable due to the presence of radioactive material that masks the contamination. Operational samples/surveys, including concrete samples, are planned for this area.

**Date**: October 1980

**Description**:

Water samples from the Recreation Lake berm contained radioactive material. This was caused by water from the oily water separator, which collected potentially contaminated oily wastewater from the Turbine and Condensate Demineralizer Buildings, being discharged to Recreation Lake. The design of the system directed the water to the lake. The system was modified to direct the oily water separator discharge to the discharge and dilution structure which was the normal release path for liquid wastes. The oily water separator overflowed on at least one occasion following the modification. The overflow spilled over ground to the site storm drains and consequently, to Recreation Lake.

**Radiological Consequences**:

At the time of the event, radioactive iodine ($^{131}I$), cesium ($^{137}Cs$), cobalt ($^{58}Co$ and $^{60}Co$), and tritium were found in the water samples in the berm area of Recreation Lake. The releases did not exceed plant effluent limits. More recent site characterization scoping survey results indicate that these areas no longer contain activity levels greater than background.

**Date**: 1981

**Description**:

The Main Steam Support Structure and the gravel area surrounding the plant tank farm (south of containment) were contaminated when a main steam relief valve opened during a steam generator hydrostatic test. This occurrence was noted due to steam generator tube leakage.

**Radiological Consequences**:

The release resulted in the contamination of a large area (>1000 $ft^2$) of the Main Steam Support Structure. The contamination was fixed in concrete. The release did not exceed effluent release limits. Operational surveys indicate the need for remediation.

**Date**: 1986, 1987, and 1989

**Description**:

Tritium contamination was identified in the sewer treatment plant, storm drains, and Recreation Lake. The source of tritium was attributed to minor flange leakage from the RWST and seal leakoff from condensate transfer pumps, which was allowed to spill on the ground prior to rerouting the leakoff. Another possible source was from the oily water separator which is located above a sewer treatment system manway. The 1989 tritium releases appear to have been caused by condensation from the PWST and RWST atmospheric vents which allow tritium in the air space above the tank to be released to atmosphere. Actions taken to stop the releases included sealing the RWST flanges and rerouting the condensate transfer pump seal leakoff.

**Radiological Consequences**:

Tritium levels in the waters released from TNP in the storm and sanitary sewer system were less than the permissible concentration specified in 10 CFR 20, "Standards for Protection Against Radiation."

**Date**: April 1987

**Description**:

As previously discussed in Section 3.1.2.2.2, during refueling activities in April 1987 an airborne contamination event occurred that resulted in the dispersal of fuel/fission product activity in the Containment Building. High levels of removable contamination were found from the 93 ft to the 205 ft elevations. An investigation revealed several partial fuel pellets and pellet fragments on the reactor vessel flange and in the lower refueling cavity area.

**Radiological Consequences**:

The Containment Building inner surfaces were decontaminated to remove the material. Surveys indicated contamination levels following the incident were consistent with pre-event levels. Discrete radioactive particles continued to be found in the plant following the incident. The presence of these particles will require detailed surveys prior to free release of material from the RCA during decommissioning activities. The contamination is expected to increase the difficulty in decontaminating the materials in the Containment Building for release.

# Appendix 3-2
# Summary of Structural Survey Results

A summary of radiological conditions and components in plant areas is presented below. Descriptions of plant areas and component locations are indicative of conditions during early 1994. In some cases, the description below may differ from the survey maps at the end of Section 3. In part, this occurs because the survey maps are intended to be a "snapshot" of the radiological conditions during the first quarter of 1994, whereas, the individual summaries below may contain more current data, professional insight, estimates, and assumptions. Recognizing that radiological conditions change, as may building and structures configuration, PGE has attempted to provide additional insight via the summary below. If current radiological conditions are needed, the most recent survey maps should be used.

## Biased Survey Results

## Structures Within the RCA

Buildings in the RCA include the Containment, Auxiliary, and Fuel Buildings, Radwaste Annex, Main Steam Support Structure, electrical facade, and the Low-Level Radioactive Waste Storage Building. Because of past problems with discrete radioactive particles, surfaces in the Containment, Auxiliary, and Fuel Buildings are assumed to be affected by plant operations and, therefore, potentially contaminated. It is anticipated that the surfaces will require a minimal wipe/wash down to remove loose surface contamination.

Containment Building

Samples of containment concrete were collected by core boring at four locations. The bores were collected from the reactor shield wall, reactor vessel missile shield, secondary shield wall, and containment dome. The samples consisted of 3 inch diameter bores. The bores were segmented and counted for radioactivity using gamma isotopic analysis. The cores were used to validate the neutron activation analysis, to determine penetrating depth of activation products in concrete structures (other than the primary reactor vessel shield), and to estimate the area extent and levels of fixed surface contamination.

Containment Building - 205 ft Elevation

The major components on this elevation include the containment air coolers and the polar crane. The highest general area radiation reading is 1.8 mrem/hr. Removable contamination levels vary from less than 1k to 150k dpm/100 $cm^2$. Fixed contamination levels are expected to be 1-5k dpm/100 $cm^2$. Material on this level consists of structural steel and grating.

Containment Building - 136/105 ft Elevation

This elevation has ventilation units and is primarily a storage area for the missile shields and head ventilation duct work during refueling operations. With no installed equipment, dose rates range from 0.3 to 0.5 mrem/hr general area. Highest removable contamination level is 2k dpm/100 $cm^2$. Fixed contamination levels are expected to be 1-5 k dpm/100 $cm^2$. Untreated concrete and structural steel are located on this level.

Containment Building - 93 ft Elevation

This elevation is the main refueling floor and the normal access path for containment. Access is provided to the refueling cavities, top levels of the steam generators and pressurizer shed, and to the seal table for the incore detectors. Storage area for the reactor vessel head is also provided. Installed equipment includes the incore detector drive boxes, ventilation units, and electrical panels. General area dose rates range from less than 0.2 to 100 mrem/hr. Highest removable contamination levels are on the refueling upender which has up to 200k dpm/100 $cm^2$. Fixed contamination levels of 5-25k dpm/100 $cm^2$ are estimated. Surfaces on this level consist of treated and untreated concrete, structural steel, and grating. Neutron activation of the concrete in the containment wall, secondary shield wall, and reactor vessel missile shield were determined by core bore analysis. The primary radionuclides identified were $^{60}Co$ and $^{152}Eu$.

Containment Building - 77 ft Elevation

This elevation provides access to feedwater piping. General area dose rates range from less than 0.2 to 1.5 mrem/hr. Removable contamination levels range up to 2k dpm/100 $cm^2$. Fixed contamination levels of 5-25k dpm/100 $cm^2$ are estimated. Surfaces on this level consist of treated and untreated concrete, structural steel, and grating.

Containment Building - 61 ft Elevation

This elevation contains the regenerative heat exchanger, excess letdown heat exchanger, letdown piping, and access to the bottom of the pressurizer shed. The regenerative heat exchanger room is not normally accessed. However, based on past surveys, the room will need remediation. The letdown piping west of the regenerative heat exchanger room has an 800 mrem/hr hot spot. The pressurizer shed has contact dose rates up to 280 mrem/hr and general area dose rates up to 120 mrem/hr. Elsewhere, general area dose rates range from less than 0.2 to 8 mrem/hr. Highest removable contamination levels measured on this level are 60k dpm/100 $cm^2$ in the pressurizer shed. Fixed contamination levels of 5-50k dpm/100 $cm^2$ are estimated for this level. Surfaces on this level consist of treated and untreated concrete, structural steel, and grating.

Containment Building - 45 ft Elevation

This elevation contains the emergency airlock for containment, safety injection accumulators, pressurizer relief tank, reactor coolant drain tank, both recirculation sumps, and primary access to the bioshield area. General area dose rates range from 0.2 to 170 mrem/hr outside of the bioshield. Removable contamination levels range up to 30k dpm/100 $cm^2$ at the safety injection line area near accumulators B and C. Fixed contamination levels of 5-100k dpm/100 $cm^2$ are estimated for this level. Surfaces on this level consist of treated and untreated concrete, structural steel, and grating.

Containment Building - Bioshield Area

This area contains four steam generators, four reactor coolant pumps, and access to the under vessel area. Contact dose rates are up to 1000 mrem/hr on the RTD bypass manifolds. General area dose rates range from 10 to 250 mrem/hr. Removable contamination levels up to 110k dpm/100 $cm^2$ are found in the bioshield. The under vessel area has not been surveyed since shutdown. The incore detectors are partially withdrawn making this area a high radiation exclusion area. Fixed contamination levels of 5-500k dpm/100 $cm^2$ are estimated for this level. Treated concrete and structural steel make up the surfaces on this level. A 3 inch diameter concrete core bore was taken from the reactor vessel shield wall at approximately the 50 ft elevation (corresponding to near centerline of the reactor vessel). The core was segmented and analyzed for gamma ray emitters. Predominant radionuclides identified were $^{60}Co$, $^{152}Eu$, $^{154}Eu$, $^{155}Eu$, and $^{134}Cs$. Radioactivity was detected to a depth of approximately 50 inches (total shield thickness is 102 inches). Close agreement was noted between the calculated neutron activation results and the measured activation values. The comparison was good for $^{152}Eu$ in the first 3 inch segment. The calculated value was 0.29 μCi/g while the measured value was 0.25 μCi/g. The agreement was not as good for segments farther from the inner wall, although the calculated values were conservative.

Auxiliary Building - 104 ft Elevation

This elevation contains the supply and exhaust ventilation filters for the Auxiliary and Fuel Buildings. Dose rates are less than 0.2 mrem/hr. Removable contamination levels are below 1k dpm/100 $cm^2$ (see 93 ft elevation below).

Auxiliary Building - 93 ft Elevation

This elevation contains supply and exhaust fans for the Auxiliary and Fuel Buildings, containment purge exhaust unit, containment access point, and access to the filter pits. Dose rates are less than or equal to 0.2 mrem/hr. Dose rates in the filter pits vary by system and age of filter. These will be surveyed at filter changes or when other access is required. Removable contamination levels outside of the filter pits are less than 1k dpm/100 $cm^2$. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated at 19% of the surface area. An average fixed contamination level of 10k dpm/100 $cm^2$ is estimated.

Auxiliary Building - 77 ft Elevation

Major equipment on this elevation includes the demineralizer valve galleries and access to the demineralizer vaults, radioactive waste filter glove boxes, and two boric acid evaporators. The highest dose rates are in the demineralizer valve galleries and are up to 350 mrem/hr contact and up to 50 mrem/hr general area. Removable contamination levels are up to 9k dpm/100 $cm^2$ in the filter valve gallery. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to exist over 19% of the surface area. An average fixed contamination level of 10k dpm/100 $cm^2$ is estimated.

Auxiliary Building - 61 ft Elevation

This elevation contains the radioactive waste evaporator, waste gas surge tank, waste compressors, waste gas decay tanks, spent resin storage tank and pump, and letdown heat exchanger. Dose rates in the letdown heat exchanger valve gallery range from 3 to 40 mrem/hr. Dose rates at the spent resin storage tank pump room range from 1.3 to 18 mrem/hr. Contact dose rates up to 400 mrem/hr are found on this elevation. Removable contamination levels are up to 7k dpm/100 $cm^2$ in the spent resin storage tank pump room. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to exist over 11% of the surface area. An average fixed contamination level of 10k dpm/100 $cm^2$ is estimated.

Auxiliary Building - 45 ft Elevation

Major components on this elevation include treated waste monitor tanks and pumps, dirty waste monitoring tank and pumps, spent fuel pool cooling pumps, spent fuel pool purification pump, chemistry hot lab, and hot sample room. Dose rates range from less than 0.2 to 5 mrem/hr. Removable contamination levels are less than 1k dpm/100 $cm^2$ with the exception of the hot sample sinks. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to cover 15% of the surface area. An average fixed contamination level of 10k dpm/100 $cm^2$ is estimated.

Auxiliary Building - 25 ft Elevation

This elevation is below grade and contained the positive displacement pump, centrifugal charging pumps, sodium hydroxide tank, primary water makeup pumps, boron injection tank, reactor coolant drain tank pumps, chemical waste tank and pumps, and access to the clean waste receiver tanks. Accessible dose rates are up to 60 mrem/hr contact at the boron injection tank and range from less than 0.2 to 12 mrem/hr at the clean waste receiver tanks. Removable contamination levels are up to 3k dpm/100 $cm^2$ in the boron injection tank area. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to exist over 15% of the surface area. An average fixed contamination level of 25k dpm/100 $cm^2$ is estimated.

Auxiliary Building - 5 ft Elevation

This elevation contains residual heat removal pumps and heat exchangers, clean waste receiver tanks and pumps, dirty waste drain tank and pumps, Auxiliary Building drain tank and pumps, safety injection pumps, containment spray pumps, and the Auxiliary Building and passageway sumps. The bottom level of the clean waste receiver tanks was not routinely accessed, however, the area was posted as a high radiation area. Contact dose rates range up to 100 mrem/hr, with general area dose rates ranging from less than 0.2 to 45 mrem/hr. Removable contamination levels are up to 35k dpm/100 $cm^2$ at the residual heat removal pumps. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to exist over 28% of the surface area. An average fixed contamination level of 25k dpm/100 $cm^2$ is estimated.

Pipe Facade - 77 ft Elevation

This elevation contains the component cooling water (CCW) surge tanks, an emergency escape hatch between the pipe facade and the Auxiliary Building and the CCW penetrations into the containment. Dose rates are less than 0.2 to 1 mrem/hr general area. Dose rates have ranged up to 200 mrem/hr general area at the resin header. Removable contamination levels are less than 1k dpm/100 $cm^2$. Fixed contamination estimate for this elevation are included in the Auxiliary Building 77 ft elevation.

Pipe Facade - 61 ft Elevation

Major components include the volume control tank, letdown system piping and penetrations, access to the vertical residual heat removal pipe chase, and the fuel transfer tube. Dose rates are 1 to 40 mrem/hr general area, with some hot spots on the letdown lines and volume control tank. Contamination levels are up to 20k dpm/100 $cm^2$ at the letdown line area. A fixed contamination estimate is included in the Auxiliary Building 61 ft elevation.

Pipe Facade - 45 ft Elevation

This level is the main access to the pipe facade. Components include the boric acid blender, residual heat removal piping, and containment penetrations. General area dose rates range from less than 0.2 to 12 mrem/hr. Contamination levels of up to 2k dpm/100 $cm^2$ are found in this area. Fixed contamination estimate for this elevation are included in the Auxiliary Building 45 ft elevation.

Fuel Building - 118 ft Elevation

This is the Fuel Building crane elevation. Dose rates depend highly on location of the crane. General area dose rates are normally less than 0.2 mrem/hr. Contamination levels are less than 1k dpm/100 $cm^2$. This elevation consists of structural steel. No fixed contamination is estimated.

Fuel Building - 104 ft Elevation

This elevation is used primarily for equipment (both contaminated and clean) storage. Dose rates depend on material stored in the area. With the absence of material, dose rates are less than 0.2 mrem/hr. Contamination levels are less than 1k dpm/100 $cm^2$. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to exist over 5% of the surface area. An average contamination level of 10k dpm/100 $cm^2$ is estimated.

Fuel Building - 93 ft Elevation

This is the main operating floor for the Fuel Building and includes the spent fuel pool, decon shop, hot machine shop, radioactive material storage areas, the top of the boric acid storage tanks, and access to the containment equipment hatch. General area dose rates range from less than 0.2 to 0.8 mrem/hr. Contact dose rates above this depend on storage of radioactive material in the area. Removable contamination levels range to 1k dpm/100 $cm^2$. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to exist over 19% of the surface area. An average fixed contamination level of 10k dpm/100 $cm^2$ is estimated.

Fuel Building - 77 ft Elevation

Major components on this level include radioactive waste control panels, evaporator concentrates holding tanks, CVCS surge tanks, CVCS concentrates pump, spent fuel pool skimmer pump, new fuel storage, and access to the cask washing pit. General area dose rates on the level range from less than 0.2 to 40 mrem/hr. Items stored in the cask wash pit could be high radiation sources. The evaporator concentrates holding tanks are not routinely accessed. Removable contamination levels are up to 3k dpm/100 $cm^2$ in the cask wash pit. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to exist over 15% of the surface area. Average fixed contamination is estimated at 10k dpm/100 $cm^2$.

Fuel Building - 61 ft Elevation

Major components on this elevation include the CVCS monitor tanks, boric acid storage tanks and pumps, seal water heat exchanger, spent fuel pool heat exchanger, I&C hot shop, a respirator wash facility, and the radioactive waste solidification room. The solidification room is designed for use with a urea formaldehyde solidification system. This system was abandoned in place and the room was used for storing and decontaminating equipment. Dose rates on this elevation depend highly on material stored in the area. General areas range from less than 0.2 to 18 mrem/hr and contact dose rates on installed equipment are up to 35 mrem/hr. Removable contamination levels are less than 1k dpm/100 $cm^2$. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to exist over 15% of the surface area. An average fixed contamination level of 10k dpm/100 $cm^2$ is estimated.

Fuel Building - 45 ft Elevation

This elevation contains the CVCS hold up tanks, hold up tank recirculation pump, gas stripper pumps, CCW pumps and heat exchangers, a respirator maintenance facility, a crane bay, and the Radwaste Annex, which is used for radioactive material storage and radioactive waste compacting. General area dose rates on this elevation are primarily less than 0.2 mrem/hr. The highest contact dose rate is 180 mrem/hr. Removable contamination levels are less than 1k dpm/100 $cm^2$. Fixed contamination at levels greater than 5k dpm/100 $cm^2$ is estimated to exist over 8% of the surface area. An average fixed contamination level of 10k dpm/100 $cm^2$ is estimated.

Electrical Penetration Area

Containment electrical penetrations are located in this area. The area is posted as a fixed contamination area. General area dose rates range from less than 0.2 to 5 mrem/hr. Removable contamination levels are less than 1k dpm/100 $cm^2$. Fixed contamination is known to exist under the RCS sample line isolation valves. Fixed contamination levels cannot be measured at this time due to background radiation levels.

Main Steam Support Structure

This area contains main steam and steam generator blowdown penetrations and is a transition area between the Containment and Turbine Buildings. The Main Steam Support Structure and the gravel area surrounding the plant tank farm were contaminated when a main steam relief valve opened during a steam generator hydrostatic test. The steam generator contained a mixture of primary and secondary liquid following a steam generator tube leak. This event caused fixed contamination throughout this structure. General area dose rates are less than 0.2 mrem/hr. Although removable contamination levels are less than 1k dpm/100 $cm^2$, there is fixed contamination and the area below the floor grating on the 45 ft elevation is posted as contaminated. Fixed contamination surveys indicate a 1300 $ft^2$ area is contaminated to levels greater than 5k dpm/100 $cm^2$. The average fixed contamination level is approximately 50k dpm/100 $cm^2$.

Steam Generator Blowdown Building

Due to primary-to-secondary leakage, this building has some fixed contamination. General area dose rates are less than 0.2 to 0.4 mrem/hr and removable contamination levels are less than 1k dpm/100 $cm^2$.

Low Level Radioactive Waste Storage Building

This building no longer exists. The building was constructed in 1991 as a long-term storage area for waste ready for shipment. General area dose rates were less than 0.2 to 20 mrem/hr and depended highly on building contents. Removable contamination levels were less than 1k dpm/100 $cm^2$.

Wright-Schuchart-Harbor (WSH) Radioactive Material Storage Area

The WSH Radioactive Material Storage Area has been free-released in accordance with plant procedures and regulatory requirements.

**Unbiased Survey Results**

Structures selected for characterization were divided into manageable survey areas such as a building elevation or group of buildings depending on size. For each survey area, a minimum of 30 randomly selected sampling locations were chosen or an average of 1 measurement location per 500 $ft^2$, whichever produced the greater number of locations. These locations were restricted to the floor and lower 6 ft of wall surface unless surveys indicated the potential for additional contaminated surfaces. A total of 840 sample points were established in the area surveyed for Phase I of the site characterization.

Only two of the sample points had detectable removable contamination levels. One point is on the roof of the Maintenance Building and had detectable alpha and beta activity. The other point is on the roof of the Turbine Building and had detectable alpha activity. Neither location is expected to require remediation.

**Structures Within the Industrial Area**

The following buildings are outside of the RCA, but inside the Industrial Area, and were surveyed specifically for the site characterization scoping survey.

Turbine Building - Roof

Exposure rates at the southeast corner are higher than 5 net μR/hr due to "shine" from the RWST. Contamination survey results were less than 1k dpm/100 $cm^2$.

Turbine Building - 93 ft Elevation

This elevation is the main operating floor of the plant. The turbine, generator, moisture separator/reheaters, feedwater heaters, and process radiation monitor for the condenser off gas system are on this level. General area exposure rates are below 5 net μR/hr, except at the south end of the building where exposure rates are slightly higher due to the RWST. Biased contamination survey results identified fixed contamination under the condenser off-gas grab sample rack measured at 15k dpm/100 $cm^2$ over an area of 25 $ft^2$. All other survey location measurements were below 1k dpm/100 $cm^2$.

Turbine Building - 63 ft Elevation

This elevation contains the steam jet air ejector, lube oil reservoir, main steam stop and throttle valves, feed water heaters, and the switchgear room. Three areas were identified with exposure rates above 5 net μR/hr. The first area is the alum tank. Exposure rates up to 12 μR/hr were measured at the tank due to the uranium content of the alum. The second area is a location on the floor about 30 ft west and 15 ft south of the alum tank near the grating over the main steam stop valves. The exposure rate at this floor location is 28 μR/hr, probably due to fixed contamination under the paint. The third location is on the floor near the floor drain east of the steam jet air ejector. This area was remediated, but exposure rates can still be measured to 22 μR/hr. The drain is marked by a fixed contamination label. Biased contamination survey results identified fixed contamination in two areas on this level: 5k dpm/100 $cm^2$ over approximately 10 $ft^2$ near the drain in the southeast corner and 50k dpm/100 $cm^2$ over approximately 32 $ft^2$ near the drain on the south end. All other survey location measurements were below 5k dpm/100 $cm^2$.

Turbine Building - 45 ft Elevation

This is the grade-level floor of the Turbine Building. The primary components on this floor include the main condensers, make up water treatment system and the air compressors. One area is identified with exposure rates above 5 net μR/hr. This area is on floor near the flow transmitter stand by the southwest corner of condenser A. Exposure rates at this floor location are up to 18 net μR/hr. Biased contamination survey results identified fixed contamination measured in the pipe trough between the electric AFW pump and the condensate pump pit. Contamination levels of 50k dpm/100 $cm^2$ over approximately 40 $ft^2$ were found. All other survey location measurements were below 1k dpm/100 $cm^2$.

Turbine Building - 35 and 27 ft Elevation

These elevations contain the condensate pumps, neutralizing tank, and the Turbine Building sump and pump. The Turbine Building sump is a contaminated area. Site characterization loose contamination surveys were taken from areas other than the sump and results were below 1k dpm/100 $cm^2$. General area exposure rates were below 5 net μR/hr. However, exposure rates at the floor on the south end of the 35 ft elevation ranged up to 6 net μR/hr. Fixed contamination levels of 5-50k dpm/100 $cm^2$ were identified over an area of 2181 $ft^2$. The contamination is a result of sump overflows and condensate pump and heater drain pump leaks during periods of primary-to-secondary leakage.

Control Building Roof

General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$.

Control Building - 105 ft Elevation

This elevation is for control room ventilation systems and contains the control room viewing gallery. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$.

Control Building - 93 ft Elevation

The main areas include the main control room and the chemistry cold lab. The lab had numerous systems which were either contaminated or potentially contaminated. Equipment in the chemistry cold lab contained low levels of radioactivity due to primary-to-secondary leakage. However, the survey results of the room (structure) indicated levels less than 1k dpm/100 $cm^2$. Since the site characterization survey, most of the equipment from the lab has been removed.

Control Building - 77 ft Elevation

The cable spreading room, computer room, and mechanical room occupy this entire elevation. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$.

Control Building - 61 ft Elevation

Primary equipment on this level includes the electrical auxiliaries, emergency batteries, mechanical room, and the telephone equipment. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$.

Control Building - 54 ft Elevation

This level contains some ventilation equipment and office areas. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$.

Control Building - 45 ft Elevation

This elevation is the primary access area for the RCA. Also on this level are the Radiation Protection Department offices, counting rooms, and calibration facility. Site characterization surveys were taken outside of the RCA. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$.

Security Building

This is the current access and egress point for the Industrial Area. Previous surveys have shown no radiological impact on this building. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$.

Administration Building

This building is used only for office space. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$. The roof was not surveyed due to access difficulty.

Central Building

This building is used for office space. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$.

Chlorine Building

This building was used to treat the service water system prior to use in the plant. The contamination survey results were below 1k dpm/100 $cm^2$. Exposure rates over most of the building were influenced by the RWST.

Condensate Demineralizer Building

This building was used to treat condensate prior to return to the feedwater system. Due to primary-to-secondary leakage during plant operations, many of the systems in this building are potentially contaminated. The structure survey results were below 1k dpm/100 $cm^2$. Recent survey results of the hopper room on the 19 ft elevation indicate contamination levels below 1k dpm/100 $cm^2$ and dose rates below 0.2 mrem/hr.

Discharge and Dilution Structure

This is the main liquid effluent release point for the plant. Exposure rate measurements in this structure were above 5 net μR/hr due to proximity of the RWST. Results of contamination surveys taken on the 45 level are below 1k dpm/100 cm$^2$.

Guard House

This was the primary access and egress point for the plant until replaced by a new access control facility in 1992. Survey results for this building have shown no historical impact from plant operations. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 cm$^2$.

Intake Structure

Exposure rate measurements in this structure were above 5 net μR/hr due to proximity of the RWST and will be resurveyed following RWST remediation. The contamination survey results were below 1k dpm/100 cm$^2$.

Maintenance Building

This is the primary maintenance support area for the plant. Exposure rates at the north end of the building were above 5 net μR/hr due to proximity to the Low Level Radioactive Waste Storage Building. Exposure rates at the south end of the building were above 5 net μR/hr due to proximity of the RWST. Exposure rates by a granite slab in the tool room were up to 7 μR/hr. Results of contamination surveys were below 1k dpm/100 cm$^2$.

Materials Building

Contamination survey results were below 1k dpm/100 cm$^2$. Exposure rates at the radioactive material storage area in the southwest corner of the building were close to 5 net μR/hr. Exposure rates were above 5 net μR/hr in the weld rod and radioactive material storage area at the north end of the building because of the radioactive material stored in the building.

Plant Modification Shop

This building was used for craft support during refueling outages. The contamination survey results were below 1k dpm/100 cm$^2$. Exposure rates were above 5 net μR/hr at the northwest part of the building due to proximity of the RWST.

Startup Boiler

Exposure rates at this structure were higher than 5 net μR/hr due to proximity of the RWST. Contamination survey results were below 1k dpm/100 cm$^2$.

Technical Support Center

This building is adjacent to the Condensate Demineralizer Building, but does not share a wall or have through-wall penetrations. The building has been used as office space and record storage, as well as for emergency response. The site contamination survey results were below 1k dpm/100 $cm^2$. Exposure rates were below 5 net μR/hr except near the check source in radiation monitor PRM-25 in the basement.

Wright-Schuchart-Harbor (WSH) Warehouse

This was one of the original structures built on the plant site. One small corner of this building was used for storage of radioactive tools and equipment. The contamination survey results from outside of the radioactive material storage area were below 1k dpm/100 $cm^2$. Inside the storage area, dose rates were from 2.5 to 5 mrem/hr and removable contamination levels were less than 1k dpm/100 $cm^2$. Exposure rates were above 5 net μR/hr in the entire east half of the building due to radiation from the radioactive material storage area. This building has since undergone final survey and has been found to satisfy the site release criteria as documented in the approved PGE-1074, "Trojan Nuclear Plant Final Survey Report for the ISFSI Site."

**Structures Outside of the Industrial Area**

Dosimetry Lab

This building contains the dosimetry and environmental monitoring groups. Sealed sources are stored in this building which caused exposure rates to be measured above 5 net μR/hr in the surrounding area during gamma surveys. Once the sealed sources were removed, nothing else was identified that would impact radiological conditions in the building.

Park Structures

These buildings were not used for plant activities and should be below contamination levels of 1k dpm/100 $cm^2$. A gamma scan performed on the park office did not identify exposure rates above 5 net μR/hr.

Pebble Springs Warehouse

A gamma scan of the building found no exposure rates above 5 net μR/hr.

Sewer Treatment Plant

A gamma scan of the building did not identify exposure rates above 5 net μR/hr.

South Maintenance Building

This building was surveyed at selected sites to provide background data. General area exposure rates were below 5 net μR/hr and contamination levels were less than 1k dpm/100 $cm^2$.

Training Building

This building was used only as a training facility. A gamma scan of the building did not identify exposure rates above 5 net μR/hr.

Trojan North Building

This building was used for engineering and administrative office space. A gamma scan of the building found no exposure rates above 5 net μR/hr.

Trojan Visitors Information Center

This building was surveyed at selected sites to provide background data. General area exposure rates were below 5 net μR/hr except near the check source for the radiation monitor. Contamination levels were below 1k dpm/100 $cm^2$.

Table 3-1

**Radioactive Effluent Summary, Noble Gases**

| | Release (Ci) | | | |
|---|---|---|---|---|
| Year | First Quarter | Second Quarter | Third Quarter | Fourth Quarter |
| 1993 | 26.5 | 13.1 | 7.3 | 6.5 |
| 1992 | 24.7 | 33.1 | 46.8 | 102.0 |
| 1991 | 75.7 | 23.9 | 40.3 | 26.6 |
| 1990 | 86.4 | 32.9 | 43.2 | 43.9 |
| 1989 | 14.3 | 228.0 | 179.0 | 42.4 |
| 1988 | 61.3 | 145.0 | 67.2 | 126.0 |
| 1987 | 85.4 | 61.3 | 9.6 | 93.0 |
| 1986 | 475.0 | 246.0 | 117.0 | 75.0 |
| 1985 | 340.0 | 277.0 | 166.0 | 278.0 |
| 1984 | 368.0 | 331.0 | 0.183 | 138.0 |
| 1983 | 65.7 | 0.0 | 31.9 | 131.0 |
| 1982 | 347.0 | 162.0 | 80.4 | 275.0 |
| 1981 | 316.0 | 399.0 | 153.0 | 293.0 |
| 1980 | 161.0 | 90.1 | 53.2 | 86.4 |
| 1979 | 33.9 | 252.0 | 131.0 | 510.0 |
| 1978 | 290.0 | 129.0 | 1.1 | 1.4 |
| 1977 | 650.0 | 1960.0 | 300.0 | 160.0 |
| 1976 | 170.0 | 360.0 | 26.1 | 111.0 |
| 1975 | [a] | [a] | [a] | 30.0 |

[a] Operating license granted November 1975.

Table 3-2

**Radioactive Effluent Summary,**
**Iodine and Particulates (excluding tritium)**

| Year | Release (Ci) First Quarter | Second Quarter | Third Quarter | Fourth Quarter |
|---|---|---|---|---|
| 1993 | [a] | [a] | [a] | [a] |
| 1992 | 0.0000003 | 0.00008 | 0.000002 | 0.00019 |
| 1991 | 0.00009 | 0.00038 | 0.00004 | 0.00008 |
| 1990 | 0.00092 | 0.00067 | 0.00005 | 0.000008 |
| 1989 | 0.00009 | 0.00389 | 0.00031 | 0.00003 |
| 1988 | 0.00008 | 0.00093 | 0.00018 | 0.00194 |
| 1987 | 0.00050 | 0.00150 | 0.00004 | 0.00016 |
| 1986 | 0.00022 | 0.00333 | 0.00216 | 0.00042 |
| 1985 | 0.00264 | 0.00184 | 0.00062 | 0.00039 |
| 1984 | 0.00387 | 0.00396 | 0.00054 | 0.00094 |
| 1983 | 0.00365 | 0.00095 | 0.00070 | 0.00082 |
| 1982 | 0.00318 | 0.00721 | 0.00132 | 0.00278 |
| 1981 | 0.0294 | 0.0419 | 0.00167 | 0.00369 |
| 1980 | 0.00265 | 0.02038 | 0.00089 | 0.00120 |
| 1979 | 0.00763 | 0.00789 | 0.00131 | 0.01707 |
| 1978 | 0.00600 | 0.00289 | 0.00064 | 0.00044 |
| 1977 | 0.0231 | 0.0244 | 0.0023 | 0.00065 |
| 1976 | 0.0097 | 0.00032 | 0.00066 | 0.00037 |
| 1975 | [b] | [b] | [b] | 0.00016 |

[a] Minimum Detectable Activity.
[b] Operating license granted November 1975.

Table 3-3

**Radioactive Effluent Summary, Liquids**

| Year | Release (Ci) First Quarter | Second Quarter | Third Quarter | Fourth Quarter |
|---|---|---|---|---|
| 1993 | 0.0453 | 0.0240 | 0.0490 | 0.0087 |
| 1992 | 0.0241 | 0.0305 | 0.0150 | 0.0214 |
| 1991 | 0.0118 | 0.0261 | 0.0114 | 0.0087 |
| 1990 | 0.0084 | 0.0560 | 0.0675 | 0.0123 |
| 1989 | 0.0308 | 0.0468 | 0.0684 | 0.0156 |
| 1988 | 0.0543 | 0.0511 | 0.0570 | 0.0382 |
| 1987 | 0.0334 | 0.0657 | 0.0539 | 0.0564 |
| 1986 | 0.0460 | 0.0847 | 0.0877 | 0.0459 |
| 1985 | 0.0757 | 0.154 | 0.0870 | 0.148 |
| 1984 | 0.0618 | 0.107 | 0.0696 | 0.111 |
| 1983 | 0.0463 | 0.0981 | 0.107 | 0.0589 |
| 1982 | 0.298 | 0.215 | 0.264 | 0.0789 |
| 1981 | 0.265 | 0.318 | 0.218 | 0.193 |
| 1980 | 0.127 | 0.381 | 0.101 | 0.178 |
| 1979 | 0.141 | 0.0456 | 0.0410 | 0.327 |
| 1978 | 0.279 | 0.165 | 0.0782 | 0.185 |
| 1977 | 0.450 | 2.65 | 1.00 | 0.0920 |
| 1976 | 0.26 | 0.94 | 0.89 | 0.63 |
| 1975 | [a] | [a] | [a] | 0.02 |

[a] Operating license granted November 1975.

Table 3-4

**Structures Burial Volume and Contamination[a] Activity Projections**

| Building | Volume ($ft^3$) | Activity (mCi)[b] |
|---|---|---|
| Containment Building | 50,901 | 24 |
| Auxiliary | 2,650 | 2 |
| Fuel | 4,711 | 1 |
| MSSS/EP | 629 | 1 |
| Turbine | 1054 | 2 |
| Total | 59,945 | 31 |

[a] Includes removable and fixed contamination, but not activation.
[b] Column may not total exactly due to rounding activity estimates.

Table 3-5

**Status of Buildings in the Radiologically Controlled Area as of 1994**

| Building | Elevation (ft) | Room or Component | β-γ Removable Contamination Level (dpm/100 cm$^2$) | General Area Dose Rate (mrem/hr) | Maximum Contact Dose Rate (mrem/hr) |
|---|---|---|---|---|---|
| Containment | 205 | Containment air coolers | <2k to 150K | <0.2 to 1.8 | NA |
| | | Grating | <1k to 3k | 0.4 to 0.6 | NA |
| | 105 | General area | <1k to 2k | 0.3 to 0.5 | NA |
| | 93 | General area | <1k to 3k | <0.2 to 8 | NA |
| | | Refueling bridge | 4k | 3 to 4 | NA |
| | | Upper refueling cavity | 5k to 50k | 40 to 70 | NA |
| | | Lower refueling cavity | 1k to 20k | 1.5 to 5 | NA |
| | | Refueling upender | 6k to 200k | 20 to 100 | [a] |
| | 77 | General area | <1 to 2k | <0.2 to 1.5 | NA |
| | 63 | Seal table | 4k to 30k | 2 to 6 | NA |
| | 61 | General area | <1k to 4k | <0.2 to 8 | 800 |
| | | Pressurizer shed | 1k to 60 k | 5 to 120 | 280 |
| | | Regenerative heat exchanger | NA | 1000 to 6000 | NA |
| | | Excess letdown heat exchanger | NA | 600 | NA |

Table 3-5

**Status of Buildings in the Radiologically Controlled Area as of 1994**

| Building | Elevation (ft) | Room or Component | β-γ Removable Contamination Level (dpm/100 $cm^2$) | General Area Dose Rate (mrem/hr) | Maximum Contact Dose Rate (mrem/hr) |
|---|---|---|---|---|---|
| | 45 Outside bioshield | General area | <1k to 3k | 0.2 to 10 | NA |
| | | Safety injection line area | <1k to 30k | 20 to 170 | 600 |
| Containment | 45 Outside bioshield | Reactor coolant drain tank and recirculation sump | 2k to 5k | 1.2 to 20 | NA |
| | | Pressurizer relief tank | 8k to 70k<br>220 dpm alpha | 5 to 30 | NA |
| | | Tendon gallery | <1k | <0.2 | NA |
| | 45 Inside bioshield | General area | 2k to 110k | 10 to 250 | 1000 |
| | 58 | A/D steam generator platform | <1k | 18 to 250 | 700 |
| | | B/C steam generator platform | <1k | 18 to 100 | 1000 |
| | | A/D resistance temperature detector (RTD) platform | 10k to 30k | 12 to 80 | NA |
| | | B/C RTD platform | 3k to 5k | 10 to 80 | NA |

Table 3-5

**Status of Buildings in the Radiologically Controlled Area as of 1994**

| Building | Elevation (ft) | Room or Component | β-γ Removable Contamination Level (dpm/100 $cm^2$) | General Area Dose Rate (mrem/hr) | Maximum Contact Dose Rate (mrem/hr) |
|---|---|---|---|---|---|
| | 67 | A/D reactor coolant pump (RCP) platform | 2k | 6 to 30 | 34 |
| | | B/C RCP platform | 4k | 15 to 20 | NA |
| Auxiliary | 104 | General area | <1k | <0.2 | NA |
| | 93 | General area | <1k | <0.2 | NA |
| | 77 | General area | <1k | <0.2 | NA |
| | | Demineralizer valve galleries | <1k to 3k | <0.2 to 50 | NA |
| | | Filter valve gallery | <1k to 9k | <0.2 to 30 | 350 |
| | 77 | Boric acid evaporators | <1k to 1k | 0.5 to 8 | NA |
| | 61 | General area | <1k | <0.2 | NA |
| | | Waste gas decay tanks | <1k | <0.2 to 0.4 | 20 |
| | | Waste gas compressors | <1k | <0.2 to 0.3 | NA |
| | | Spent resin storage tank pump room | <1k to 7k | 1.3 to 18 | NA |
| | | Letdown heat exchanger valve gallery | <1k to 6k | 3 to 40 | 400 |
| | 45 | General area | <1k, 2k at hot sample sinks | <0.2 to 1.0 | NA |

Table 3-5

**Status of Buildings in the Radiologically Controlled Area as of 1994**

| Building | Elevation (ft) | Room or Component | β-γ Removable Contamination Level (dpm/100 $cm^2$) | General Area Dose Rate (mrem/hr) | Maximum Contact Dose Rate (mrem/hr) |
|---|---|---|---|---|---|
| | | Dirty waste monitor tank | <1k | 1 to 5 | NA |
| | | Treated waste monitor tanks | <1k | 1 to 1.8 | NA |
| | | Spent fuel pool cooling pumps | <1k | <0.2 to 1.0 | NA |
| | 25 | General area | <1k | <0.2 to 0.4 | NA |
| | | Centrifugal charging pumps | <1k | 0.5 to 0.7 | 10 |
| | | Boron injection tank | <1k to 3k | 0.2 to 8 | 60 |
| | | Clean waste receiver tanks | <1k | 4 to 12 | 35 |
| | 5 | General area | <1k | <0.2 to 5 | NA |
| Auxiliary | 5 | Residual heat removal (RHR) pumps | <1k to 35k | 3 to 70 | NA |
| | | RHR heat exchanger | <1k to 30k | 1 to 60 | NA |
| | | Clean waste pumps | <1k | <0.2 to 25 | 100 |
| | | Auxiliary building drain tank and pumps | <1k | 0.5 to 45 | 50 |
| | | Dirty waste drain tank and pumps | <1k | 0.2 to 30 | NA |
| Pipe facade | 77 | General area | <1k | <0.2 to 1 | NA |
| | | Pipe chase, resin header | NA | 1 to 200 | 800 |

Table 3-5

**Status of Buildings in the Radiologically Controlled Area as of 1994**

| Building | Elevation (ft) | Room or Component | β-γ Removable Contamination Level (dpm/100 $cm^2$) | General Area Dose Rate (mrem/hr) | Maximum Contact Dose Rate (mrem/hr) |
|---|---|---|---|---|---|
| | 61 | General area | <1k to 4k | 1 to 12 | NA |
| | | Letdown line | 1k to 20k | 2 to 12 | 8 |
| | | Pipe chase | 1k | 2 to 4 | 80 |
| | | Volume control tank | <1k | 2 to 40 | 200 |
| | 45 | General area | <1k | <0.2 to 12 | NA |
| | 40 | Pipe chase | 2k | 0.6 to 3 | NA |
| | 25 | Pipe chase | <1k | 1 | NA |
| Fuel | 118 | Fuel building crane | <1k | <0.2 | NA |
| | 104 | General area | <1k | <0.2 to 2.5 | NA |
| | 93 | General area | <1k to 1k | <0.2 to 0.8 | 7 |
| | | Spent fuel pool | <1k | <0.2 to 0.5 | 200 |
| | 77 | General area | <1k | <0.2 to 1 | NA |
| | | Chemical volume control system surge tank | <1k | 0.5 | NA |
| | | CVCS pump room | <1k to 2k | 1 to 6 | NA |

Table 3-5

**Status of Buildings in the Radiologically Controlled Area as of 1994**

| Building | Elevation (ft) | Room or Component | $\beta$-$\gamma$ Removable Contamination Level (dpm/100 $cm^2$) | General Area Dose Rate (mrem/hr) | Maximum Contact Dose Rate (mrem/hr) |
|---|---|---|---|---|---|
| | | Cask wash pit | 3k | 10 to 40 | NA |
| | 61 | General area | <1k | <0.2 to 1.4 | 8 |
| | | Spent fuel pool heat exchanger | <1k | 0.2 to 1 | NA |
| | | Seal water heat exchanger | <1k | 4 to 18 | 35 |
| | | CVCS monitor tanks | <1k | <0.2 | NA |
| | | Boric acid storage tanks | <1k | 0.5 to 10 | NA |
| | 45 | General area | <1k to 1k at CCW heat exchanger A | <0.2 to 1 | NA |
| | | Radwaste annex | <1k | <0.2 | NA |
| | | CVCS holdup tank pumps | <1k | 0.2 to 1 | 180 |
| Electrical penetration | 70 and 77 | General area | <1k | <0.2 | NA |
| | 61 | General area | <1k | <0.2 to 1.0 | NA |
| | 51 | General area | <1k | <0.2 | NA |
| | 45 | General area | <1k outside controlled area | <0.2 to 5 | 30 |

Table 3-5

**Status of Buildings in the Radiologically Controlled Area as of 1994**

| Building | Elevation (ft) | Room or Component | β-γ Removable Contamination Level (dpm/100 cm²) | General Area Dose Rate (mrem/hr) | Maximum Contact Dose Rate (mrem/hr) |
|---|---|---|---|---|---|
| Main steam support structure | 77 | General area | <1k | <0.2 | NA |
| | 69 | General area | <1k | <0.2 | NA |
| | 59 | General area | <1k | <0.2 | NA |
| | 45 | General area | <1k | <0.2 | NA |
| Steam generator blowdown building | 55 | | <1k | <0.2 | NA |
| | 45 | | <1k | <0.2 to 0.4 | NA |
| Wright-Schuchart-Harbor RMSA | 45 | General area | <1k | <0.2 to 5 | NA |
| Refueling water storage tank | 45 | Exterior inside fence | <1k | 2.5 to 5 | NA |
| Primary water storage tank | 45 | Exterior | <1k | <0.2 | NA |

Table 3-5

**Status of Buildings in the Radiologically Controlled Area as of 1994**

| Building | Elevation (ft) | Room or Component | β-γ Removable Contamination Level (dpm/100 $cm^2$) | General Area Dose Rate (mrem/hr) | Maximum Contact Dose Rate (mrem/hr) |
|---|---|---|---|---|---|
| Condensate demineralizer building | 33 | Hopper room, sump not included | <1k | NA | NA |
| Radwaste storage building | 45 | General area | <1k | <0.2 to 20 | NA |

NA - Not applicable

[a] Recent survey data is not available.

Table 3-6

**System Burial Volume and Surface Activity Projections**

| System | Volume ($ft^3$) | Activity[a] (Ci) |
|---|---|---|
| Reactor coolant piping | 5,894 | 221 |
| Pressurizer relief tank | 625 | <1 |
| Reactor coolant pumps and motors | 3,044 | 134 |
| Control rod drive mechanisms/incore instrumentation/service structure | 1,726 | 83 |
| Reactor vessel and internals | 8,341 | 357.9 |
| Spent fuel pool and racks | 17,305 | 150+ |
| 120-V ac preferred instrument ac | 1,400 | <1 |
| 125-V dc power | 175 | <1 |
| 4.16-kV ac power | 726 | <1 |
| 480-V ac auxiliary load center | 5,080 | <1 |
| 480-V ac motor control center | 8,426 | <1 |
| Chemical and volume control | 10,968 | 25 |
| Clean radwaste | 5,423 | 14 |
| Containment building penetrations | 188 | <1 |
| Control rod drive | 85 | <1 |
| Dirty radwaste | 1,613 | <1 |
| Electric heat tracing | 164 | <1 |
| Electrical (Cable/Tray/Conduit) | 60,139 | <1 |
| Fuel handling system | 339 | b |
| Fuel pool cooling and demineralizer | 4,632 | 5.6 |
| Fuel and auxiliary building heating, ventilation, and air conditioning (HVAC) | 3,661 | <1 |
| Gaseous radwaste | 2,529 | <1 |
| HVAC | 6,635 | <1 |
| Hydrogen recombiners | 576 | <1 |
| Integrated leak rate test instrument line | 106 | <1 |
| Instrument and service air | 1,327 | <1 |

Table 3-6

**System Burial Volume and Surface Activity Projections**

| System | Volume ($ft^3$) | Activity[a] (Ci) |
|---|---|---|
| Lighting panel supply | 997 | <1 |
| Miscellaneous components | 1,936 | <1 |
| Miscellaneous reactor coolant | 3,418 | <1 |
| Nuclear instrumentation | 193 | <1 |
| Oily waste and storm drains | 1,882 | <1 |
| Containment HVAC | 18,869 | <1 |
| Primary makeup water | 3,615 | <1 |
| Process sampling | 114 | 4 |
| Radiation monitoring | 134 | <1 |
| Reactor nonnuclear instruments | 245 | <1 |
| Reactor vessel system | 116 | c |
| Residual heat removal | 7,649 | 36 |
| Safety injection system | 7,149 | 7 |
| Solid radwaste | 370 | <1 |
| Steam generator system | 3,562 | <1 |
| Turbine building sump pumps and miscellaneous | 639 | <1 |
| Component cooling water[d] | 6,115 | <1 |
| Condensate demineralizers[d] | 2,262 | <1 |
| Discharge and dilution[d] | 3,834 | <1 |
| Containment spray[d] | 1,563 | <1 |
| Total | 215,789 | 1070.5 |

a Does not include activation.
b To be determined.
c Activity included with reactor coolant piping.
d Site characterization survey results identified these systems as contaminated.

Table 3-7

**Isotopic Distribution**
(Decay Corrected to 1994 and 1998)

| Radionuclide | % Activity[a] Auxiliary Systems 1994 / 1998 | % Activity[b] Primary Systems 1994 / 1998 |
|---|---|---|
| Magnesium-54 | 1.0 / <1 | <1 / <1 |
| Iron-55 | 5.9 / 2.6 | 61 / 43 |
| Cobalt-57 | <1 / <1 | c |
| Cobalt-58 | <1 / <1 | <1 / <1 |
| Cobalt-60 | 23 / 18 | 15 / 18 |
| Nickel-63 | 52 / 66 | 19 / 38 |
| Strontium-89 | <1 / <1 | <1 / <1 |
| Strontium-90 | <1 / <1 | <1 / <1 |
| Zirconium-95 | <1 / <1 | <1 / <1 |
| Ruthenium-106 | 3.7 / <1 | <1 / <1 |
| Silver-108m | <1 / <1 | c |
| Silver-110m | <1 / <1 | c |
| Cadmium-109 | 1.3 / <1 | c |
| Tin-113 | <1 / <1 | c |
| Antimony-125 | 1.2 / <1 | <1 / <1 |
| Iodine-129 | <1 / <1 | <1 / <1 |
| Cesium-134 | <1 / <1 | c |
| Cesium-137 | <1 / <1 | <1 / <1 |
| Cerium-144 | <1 / <1 | <1 / <1 |
| Plutonium-238 | <1 / <1 | <1 / <1 |
| Plutonium-239/240 | <1 / <1 | <1 / <1 |
| Plutonium-241 | 11 / 11 | <1 / <1 |
| Americium-241 | <1 / <1 | c |
| Curium-242 | <1 / <1 | <1 / <1 |
| Curium-243/244 | <1 / <1 | c |

a Based on clean waste filter analysis (10 CFR 61 Waste Stream, 1992).
b Based on 1992 steam generator tube analysis.
c Not identified.

## Table 3-8

## 10 CFR Part 61 Classification by Component One Year After Shutdown

| Component | 10 CFR 61 Classification | H-3 (Ci/m) | C-14 | Ca-45 | Mn-54 | Fe-55 | Co-60 | Ni-59 | Ni-63 | Nb-94 | Tc-99 | Sn-119m | Sb-125 | Te-125m | Sum of Fractions Tab. 1 | Tab. 2 Col. 1 | Tab. 2 Col. 2 | Tab. 2 Col. 3 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Core Baffle | >C | 2.494E+02 | 1.012E+02 | 1.650E+00 | 3.556E+04 | 9.244E+05 | 1.046E+06 | 4.544E+02 | 7.635E+04 | 1.897E+00 | 4.341E-01 | 1.631E-10 | 7.702E-01 | 5.999E-03 | 12.818 | 5053.209 | 109.073 | 10.907 |
| Core Formers | >C | 1.517E+02 | 9.709E+01 | 1.626E+00 | 7.463E+03 | 9.219E+05 | 5.652E+05 | 3.619E+02 | 6.821E+04 | 7.868E-01 | 1.059E-01 | 1.317E-11 | 1.924E-01 | 1.941E-03 | 6.793 | 4087.766 | 97.441 | 9.744 |
| Lower Core Barrel | C | 1.951E+01 | 3.376E+00 | 5.473E-02 | 1.190E+03 | 3.092E+04 | 4.129E+04 | 1.859E+01 | 2.691E+03 | 6.870E-02 | 1.740E-02 | 6.478E-17 | 1.225E-03 | 8.980E-06 | 0.470 | 182.215 | 3.844 | 0.384 |
| Upper Core Barrel | A | 7.963E-04 | 1.378E-04 | 2.234E-06 | 4.858E-02 | 1.262E+00 | 1.685E+00 | 7.589E-04 | 1.098E-01 | 2.804E-06 | 7.104E-07 | 2.644E-21 | 5.001E-08 | 3.666E-10 | 0.000 | 0.007 | 0.000 | 0.000 |
| Thermal Shield Pads | C | 1.373E+01 | 2.385E+00 | 3.865E-02 | 8.428E+02 | 2.185E+04 | 2.932E+04 | 1.319E+01 | 1.903E+03 | 4.872E-02 | 1.237E-02 | 3.449E-16 | 4.588E-04 | 3.359E-06 | 0.333 | 129.011 | 2.719 | 0.272 |
| Vessel Clad | C | 1.504E+01 | 2.628E+00 | 4.356E-02 | 3.603E+02 | 2.493E+04 | 2.167E+04 | 1.573E+01 | 2.186E+03 | 2.936E-02 | 5.366E-03 | 1.474E-16 | 3.752E-04 | 3.469E-06 | 0.251 | 129.929 | 3.123 | 0.312 |
| Vessel Wall | A | 2.293E-01 | 2.716E-04 | 2.038E-04 | 8.962E+00 | 2.298E+02 | 2.900E+01 | 5.836E-03 | 8.439E-01 | 7.638E-05 | 1.939E-04 | 1.142E-03 | 1.581E-03 | 5.869E-06 | 0.000 | 0.412 | 0.001 | 0.000 |
| Vessel Insulation | A | 8.655E-02 | 1.432E-02 | 2.360E-04 | 2.753E+00 | 1.347E+02 | 1.340E+02 | 8.627E-02 | 1.188E+01 | 1.943E-04 | 4.270E-05 | 7.968E-21 | 1.513E-08 | 1.307E-10 | 0.002 | 0.729 | 0.017 | 0.002 |
| 1st 3" Bioshield | A | 6.825E+00 | 2.217E-03 | 2.702E-01 | 1.098E-02 | 1.386E+00 | 3.508E-01 | 1.700E-05 | 2.286E-03 | 2.915E-06 | 3.358E-08 | 1.017E-04 | 1.515E-04 | 5.620E-07 | 0.000 | 0.174 | 0.000 | 0.000 |
| 2nd 3" Bioshield | A | 2.597E+01 | 8.305E-03 | 1.019E+00 | 1.397E-02 | 5.232E+00 | 1.143E+00 | 6.500E-05 | 8.621E-03 | 8.182E-06 | 4.623E-08 | 1.400E-04 | 2.405E-04 | 8.922E-07 | 0.000 | 0.660 | 0.000 | 0.000 |
| 1st Bioshield Rebar/Liner | A | 3.821E-01 | 4.795E-03 | 3.181E-04 | 9.640E-01 | 3.609E+02 | 1.397E+01 | 1.108E-02 | 1.468E+00 | 3.509E-05 | 2.465E-09 | 9.324E-22 | 9.968E-09 | 1.084E-10 | 0.000 | 0.588 | 0.002 | 0.000 |
| 2nd 6" Bioshield | A | 1.243E+01 | 3.959E-03 | 4.865E-01 | 2.759E-03 | 2.499E+00 | 5.204E-01 | 3.117E-05 | 4.118E-03 | 3.506E-06 | 1.061E-08 | 3.210E-05 | 6.772E-05 | 2.509E-07 | 0.000 | 0.316 | 0.000 | 0.000 |
| 3rd 6" Bioshield | A | 2.544E+00 | 8.086E-04 | 9.952E-02 | 2.387E-04 | 5.111E-01 | 1.043E-01 | 6.387E-06 | 8.419E-04 | 6.840E-07 | 1.215E-09 | 3.679E-06 | 9.929E-06 | 3.677E-08 | 0.000 | 0.065 | 0.000 | 0.000 |
| 4th 6" Bioshield | A | 3.961E-01 | 1.259E-04 | 1.550E-02 | 3.156E-05 | 7.960E-02 | 1.621E-02 | 9.949E-07 | 1.311E-04 | 1.060E-07 | 1.728E-10 | 5.230E-07 | 1.479E-06 | 5.476E-09 | 0.000 | 0.010 | 0.000 | 0.000 |
| 2nd Bioshield Rebar | A | 6.169E-03 | 7.268E-05 | 2.285E-05 | 4.894E-03 | 7.169E+00 | 2.258E-01 | 1.696E-04 | 2.252E-02 | 4.544E-07 | 9.221E-12 | 6.141E-26 | 1.675E-12 | 1.425E-12 | 0.000 | 0.011 | 0.000 | 0.000 |
| 5th 6" Bioshield | A | 5.954E-02 | 1.893E-05 | 2.328E-03 | 6.215E-06 | 1.196E-02 | 2.445E-03 | 1.495E-07 | 1.971E-05 | 1.607E-08 | 3.029E-11 | 9.164E-08 | 2.400E-07 | 8.888E-10 | 0.000 | 0.002 | 0.000 | 0.000 |
| 6th 6" Bioshield | A | 9.559E-03 | 3.042E-06 | 3.741E-04 | 1.406E-06 | 1.922E-03 | 3.955E-04 | 2.400E-08 | 3.166E-06 | 2.624E-09 | 6.060E-12 | 1.834E-08 | 4.346E-08 | 1.610E-10 | 0.000 | 0.000 | 0.000 | 0.000 |

## Table 3-8

## 10 CFR Part 61 Classification by Component One Year After Shutdown

| | 10 CFR 61 | H-3 | C-14 | Ca-45 | Mn-54 | Fe-55 | Co-60 | Ni-59 | Ni-63 | Nb-94 | Tc-99 | Sn-119m | Sb-125 | Te-125m | Sum of Fractions | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Component | Classification | (Ci/m) | | | | | | | | | | | | | Tab. 1 | Tab. 2 Col. 1 | Tab. 2 Col. 2 | Tab. 2 Col. 3 |
| 4th 12" Bioshield | A | 1.008E-03 | 3.212E-07 | 3.948E-05 | 2.027E-07 | 2.028E-04 | 4.210E-05 | 2.530E-09 | 3.342E-07 | 2.824E-10 | 7.986E-13 | 2.418E-09 | 5.239E-09 | 1.943E-11 | 0.000 | 0.000 | 0.000 | 0.000 |
| 5th 12" Bioshield | A | 4.210E-05 | 1.342E-08 | 1.650E-06 | 1.128E-08 | 8.471E-06 | 1.777E-06 | 1.056E-10 | 1.396E-08 | 1.209E-11 | 4.161E-14 | 1.260E-10 | 2.528E-10 | 9.346E-13 | 0.000 | 0.000 | 0.000 | 0.000 |
| 6th 12" Bioshield | A | 2.067E-06 | 6.591E-10 | 8.102E-08 | 6.166E-10 | 4.161E-07 | 8.767E-08 | 5.184E-12 | 6.854E-10 | 6.000E-13 | 2.227E-15 | 6.785E-12 | 1.329E-11 | 4.644E-14 | 0.000 | 0.000 | 0.000 | 0.000 |
| Lower Core Plate | >C | 9.614E+01 | 2.474E+01 | 4.073E-01 | 5.392E+03 | 2.311E+05 | 2.291E+05 | 1.252E+02 | 1.934E+04 | 3.557E-01 | 7.724E-02 | 4.980E-12 | 3.097E-02 | 2.605E-04 | 2.657 | 1220.184 | 27.632 | 2.763 |
| Lower Core Support Columns | C | 3.785E+01 | 6.318E+00 | 1.056E-01 | 3.896E+02 | 6.069E+04 | 4.335E+04 | 3.921E+01 | 5.349E+03 | 5.026E-02 | 5.982E-03 | 1.221E-16 | 6.073E-04 | 6.239E-06 | 0.508 | 302.959 | 7.642 | 0.764 |
| Lower Core Support | A | 7.425E-03 | 1.281E-03 | 2.087E-05 | 3.832E-01 | 1.183E+01 | 1.452E+01 | 7.321E-03 | 1.037E+00 | 2.325E-05 | 5.908E-06 | 9.228E-23 | 1.304E-10 | 1.011E-12 | 0.000 | 0.068 | 0.001 | 0.000 |
| Below Lower Core Support | A | 4.101E-05 | 6.465E-06 | 1.080E-07 | 3.751E-04 | 6.215E-02 | 4.443E-02 | 4.148E-05 | 5.532E-03 | 5.058E-08 | 6.047E-09 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 0.000 | 0.000 | 0.000 | 0.000 |
| Upper Core Plate | C | 2.078E+01 | 3.699E+00 | 6.067E-02 | 8.779E+02 | 3.452E+04 | 3.736E+04 | 2.125E+01 | 3.014E+03 | 5.715E-02 | 1.293E-02 | 4.549E-16 | 7.761E-04 | 6.373E-06 | 0.429 | 190.561 | 4.305 | 0.431 |
| Upper Core Support Columns | B | 2.670E+00 | 4.186E-01 | 6.989E-03 | 2.643E+01 | 4.020E+03 | 2.913E+03 | 2.673E+00 | 3.575E+02 | 3.366E-03 | 4.116E-04 | 5.553E-19 | 2.707E-06 | 2.776E-08 | 0.034 | 20.224 | 0.511 | 0.051 |

Table 3-9

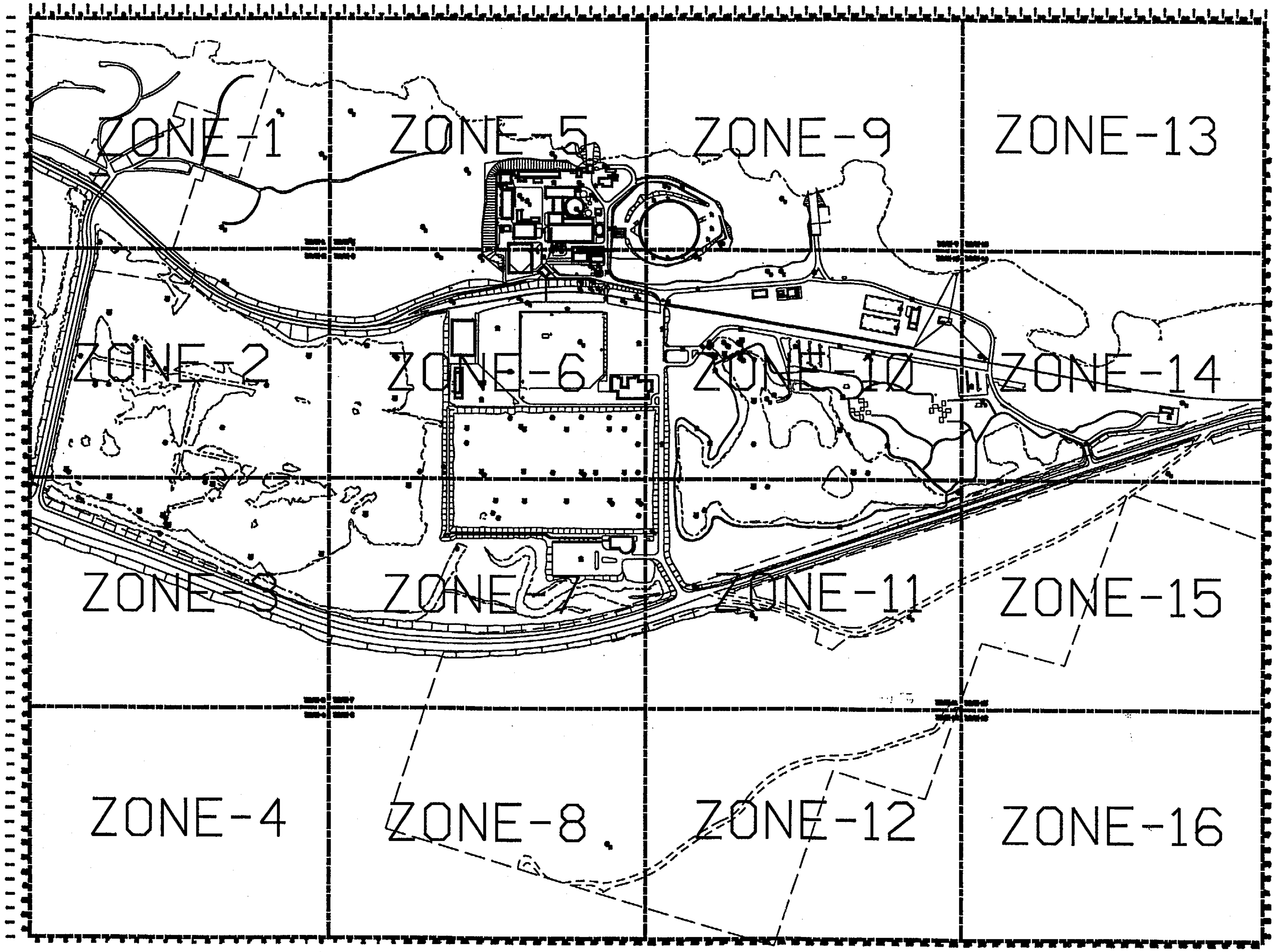
**10 CFR Part 61 Classification by Component Five Years After Shutdown**

| Component | 10 CFR 61 Classification | H-3 (Ci/m) | C-14 | Ca-45 | Mn-54 | Fe-55 | Co-60 | Ni-59 | Ni-63 | Nb-94 | Tc-99 | Sn-119m | Sb-125 | Te-125m | SUM OF FRACTIONS Tab. 1 | Tab. 2 Col. 1 | Tab. 2 Col. 2 | Tab. 2 Col. 3 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Core Shroud | >C | 1.992E+02 | 1.011E+02 | 3.301E-03 | 1.392E+03 | 3.182E+05 | 6.180E+05 | 4.544E+02 | 7.408E+04 | 1.897E+00 | 4.341E-01 | 2.615E-12 | 2.831E-01 | 1.566E-10 | 12.959 | 3461.137 | 105.834 | 10.583 |
| Core Formers | >C | 1.212E+02 | 9.705E+01 | 3.253E-03 | 2.921E+02 | 3.174E+05 | 3.340E+05 | 3.618E+02 | 6.618E+04 | 7.867E-01 | 1.059E-01 | 2.111E-13 | 7.071E-02 | 5.069E-11 | 6.826 | 2824.912 | 94.548 | 9.455 |
| Lower Core Barrel | C | 1.558E+01 | 3.374E+00 | 1.095E-04 | 4.658E+01 | 1.064E+04 | 2.439E+04 | 1.859E+01 | 2.611E+03 | 6.869E-02 | 1.740E-02 | 1.039E-18 | 4.503E-04 | 2.345E-13 | 0.476 | 125.107 | 3.730 | 0.373 |
| Upper Core Barrel | A | 6.362E-04 | 1.378E-04 | 4.470E-09 | 1.901E-03 | 4.345E-01 | 9.959E-01 | 7.589E-04 | 1.066E-01 | 2.804E-06 | 7.104E-07 | 4.240E-23 | 1.838E-08 | 9.573E-18 | 0.000 | 0.005 | 0.000 | 0.000 |
| Thermal Shield Pads | C | 1.097E+01 | 2.384E+00 | 7.734E-05 | 3.299E+01 | 7.520E+03 | 1.732E+04 | 1.319E+01 | 1.846E+03 | 4.871E-02 | 1.237E-02 | 5.530E-18 | 1.686E-04 | 8.771E-14 | 0.337 | 88.570 | 2.638 | 0.264 |
| Vessel Clad | C | 1.202E+01 | 2.626E+00 | 8.715E-05 | 1.410E+01 | 8.583E+03 | 1.280E+04 | 1.573E+01 | 2.121E+03 | 2.936E-02 | 5.366E-03 | 2.364E-18 | 1.379E-04 | 9.058E-14 | 0.253 | 91.483 | 3.030 | 0.303 |
| Vessel Wall | A | 1.832E-01 | 2.715E-04 | 4.078E-07 | 3.508E-01 | 7.911E+01 | 1.714E+01 | 5.836E-03 | 8.189E-01 | 7.637E-05 | 1.939E-04 | 1.830E-05 | 5.811E-04 | 1.533E-13 | 0.000 | 0.166 | 0.001 | 0.000 |
| Vessel Insulation | A | 6.914E-02 | 1.431E-02 | 4.721E-07 | 1.077E-01 | 4.637E+01 | 7.918E+01 | 8.627E-02 | 1.153E+01 | 1.943E-04 | 4.270E-05 | 1.278E-22 | 5.560E-09 | 3.413E-18 | 0.002 | 0.511 | 0.016 | 0.002 |
| 1st 3" Bioshield | A | 5.453E+00 | 2.216E-03 | 5.405E-04 | 4.298E-04 | 4.773E-01 | 2.073E-01 | 1.700E-05 | 2.218E-03 | 2.914E-06 | 3.358E-08 | 1.631E-06 | 5.567E-05 | 1.468E-14 | 0.000 | 0.137 | 0.000 | 0.000 |
| 2nd 3" Bioshield | A | 2.074E+01 | 8.301E-03 | 2.038E-03 | 5.468E-04 | 1.801E+00 | 6.756E-01 | 6.500E-05 | 8.365E-03 | 8.181E-06 | 4.623E-08 | 2.245E-06 | 8.839E-05 | 2.330E-14 | 0.000 | 0.522 | 0.000 | 0.000 |
| 1st Bioshield Rebar/Liner | A | 3.053E-01 | 4.793E-03 | 6.365E-07 | 3.773E-02 | 1.242E+02 | 8.254E+00 | 1.108E-02 | 1.425E+00 | 3.509E-05 | 2.465E-09 | 1.495E-23 | 3.663E-09 | 2.831E-18 | 0.000 | 0.238 | 0.002 | 0.000 |
| 2nd 6" Bioshield | A | 9.928E+00 | 3.957E-03 | 9.733E-04 | 1.080E-04 | 8.604E-01 | 3.075E-01 | 3.117E-05 | 3.996E-03 | 3.505E-06 | 1.061E-08 | 5.148E-07 | 2.489E-05 | 6.553E-15 | 0.000 | 0.250 | 0.000 | 0.000 |
| 3rd 6" Bioshield | A | 2.032E+00 | 8.082E-04 | 1.991E-04 | 9.343E-06 | 1.760E-01 | 6.163E-02 | 6.387E-06 | 8.169E-04 | 6.840E-07 | 1.215E-09 | 5.898E-08 | 3.649E-06 | 9.602E-16 | 0.000 | 0.051 | 0.000 | 0.000 |
| 4th 6" Bioshield | A | 3.164E-01 | 1.259E-04 | 3.100E-05 | 1.235E-06 | 2.740E-02 | 9.578E-03 | 9.948E-07 | 1.272E-04 | 1.059E-07 | 1.728E-10 | 8.387E-09 | 5.434E-07 | 1.430E-16 | 0.000 | 0.008 | 0.000 | 0.000 |
| 2nd Bioshield Rebar | A | 6.169E-03 | 7.268E-05 | 2.285E-05 | 4.894E-03 | 7.169E+00 | 2.258E-01 | 1.696E-04 | 2.252E-02 | 4.544E-07 | 9.221E-12 | 6.141E-26 | 1.675E-12 | 1.425E-12 | 0.000 | 0.011 | 0.000 | 0.000 |
| 5th 6" Bioshield | A | 4.756E-02 | 1.892E-05 | 4.659E-06 | 2.432E-07 | 4.118E-03 | 1.445E-03 | 1.494E-07 | 1.912E-05 | 1.607E-08 | 3.029E-11 | 1.469E-09 | 8.819E-08 | 2.321E-17 | 0.000 | 0.001 | 0.000 | 0.000 |

Table 3-9

**10 CFR Part 61 Classification by Component Five Years After Shutdown**

| Component | 10 CFR 61 Classification | H-3 (Ci/m) | C-14 | Ca-45 | Mn-54 | Fe-55 | Co-60 | Ni-59 | Ni-63 | Nb-94 | Tc-99 | Sn-119m | Sb-125 | Te-125m | SUM OF FRACTIONS Tab. 1 | Tab. 2 Col. 1 | Tab. 2 Col. 2 | Tab. 2 Col. 3 |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 6th 6" Bioshield | A | 7.636E-03 | 3.041E-06 | 7.485E-07 | 5.504E-08 | 6.615E-04 | 2.337E-04 | 2.400E-08 | 3.072E-06 | 2.623E-09 | 6.060E-12 | 2.941E-10 | 1.597E-08 | 4.205E-18 | 0.000 | 0.000 | 0.000 | 0.000 |
| 4th 12" Bioshield | A | 8.056E-04 | 3.210E-07 | 7.899E-08 | 7.934E-09 | 6.980E-05 | 2.487E-05 | 2.530E-09 | 3.243E-07 | 2.824E-10 | 7.986E-13 | 3.877E-11 | 1.926E-09 | 5.073E-19 | 0.000 | 0.000 | 0.000 | 0.000 |
| 5th 12" Bioshield | A | 3.363E-05 | 1.341E-08 | 3.301E-09 | 4.415E-10 | 2.916E-06 | 1.050E-06 | 1.056E-10 | 1.354E-08 | 1.208E-11 | 4.161E-14 | 2.020E-12 | 9.289E-11 | 2.440E-20 | 0.000 | 0.000 | 0.000 | 0.000 |
| 6th 12" Bioshield | A | 1.651E-06 | 6.588E-10 | 1.621E-10 | 2.414E-11 | 1.432E-07 | 5.181E-08 | 5.184E-12 | 6.651E-10 | 5.999E-13 | 2.227E-15 | 1.088E-13 | 4.883E-12 | 1.213E-21 | 0.000 | 0.000 | 0.000 | 0.000 |
| Lower Core Plate | >C | 7.681E+01 | 2.472E+01 | 8.149E-04 | 2.110E+02 | 7.955E+04 | 1.354E+05 | 1.252E+02 | 1.877E+04 | 3.557E-01 | 7.724E-02 | 7.986E-14 | 1.138E-02 | 6.801E-12 | 2.682 | 845.506 | 26.812 | 2.681 |
| Lower Core Support Columns | C | 3.023E+01 | 6.315E+00 | 2.113E-04 | 1.525E+01 | 2.089E+04 | 2.561E+04 | 3.921E+01 | 5.190E+03 | 5.025E-02 | 5.982E-03 | 1.957E-18 | 2.232E-04 | 1.629E-13 | 0.510 | 215.509 | 7.415 | 0.741 |
| Lower Core Support | A | 5.932E-03 | 1.280E-03 | 4.176E-08 | 1.500E-02 | 4.074E+00 | 8.582E+00 | 7.321E-03 | 1.006E+00 | 2.324E-05 | 5.908E-06 | 1.480E-24 | 4.791E-11 | 2.641E-20 | 0.000 | 0.047 | 0.001 | 0.000 |
| Below Lower Core Support | A | 3.276E-05 | 6.462E-06 | 2.161E-10 | 1.468E-05 | 2.139E-02 | 2.626E-02 | 4.148E-05 | 5.367E-03 | 5.058E-08 | 6.047E-09 | 0.000E+00 | 0.000E+00 | 0.000E+00 | 0.000 | 0.000 | 0.000 | 0.000 |
| Upper Core Plate | C | 1.660E+01 | 3.697E+00 | 1.214E-04 | 3.436E+01 | 1.188E+04 | 2.208E+04 | 2.124E+01 | 2.924E+03 | 5.715E-02 | 1.293E-02 | 7.295E-18 | 2.852E-04 | 1.664E-13 | 0.433 | 132.522 | 4.177 | 0.418 |
| Upper Core Support Columns | B | 2.133E+00 | 4.184E-01 | 1.398E-05 | 1.035E+00 | 1.384E+03 | 1.721E+03 | 2.673E+00 | 3.469E+02 | 3.365E-03 | 4.116E-04 | 8.903E-21 | 9.950E-07 | 7.248E-16 | 0.034 | 14.402 | 0.496 | 0.050 |

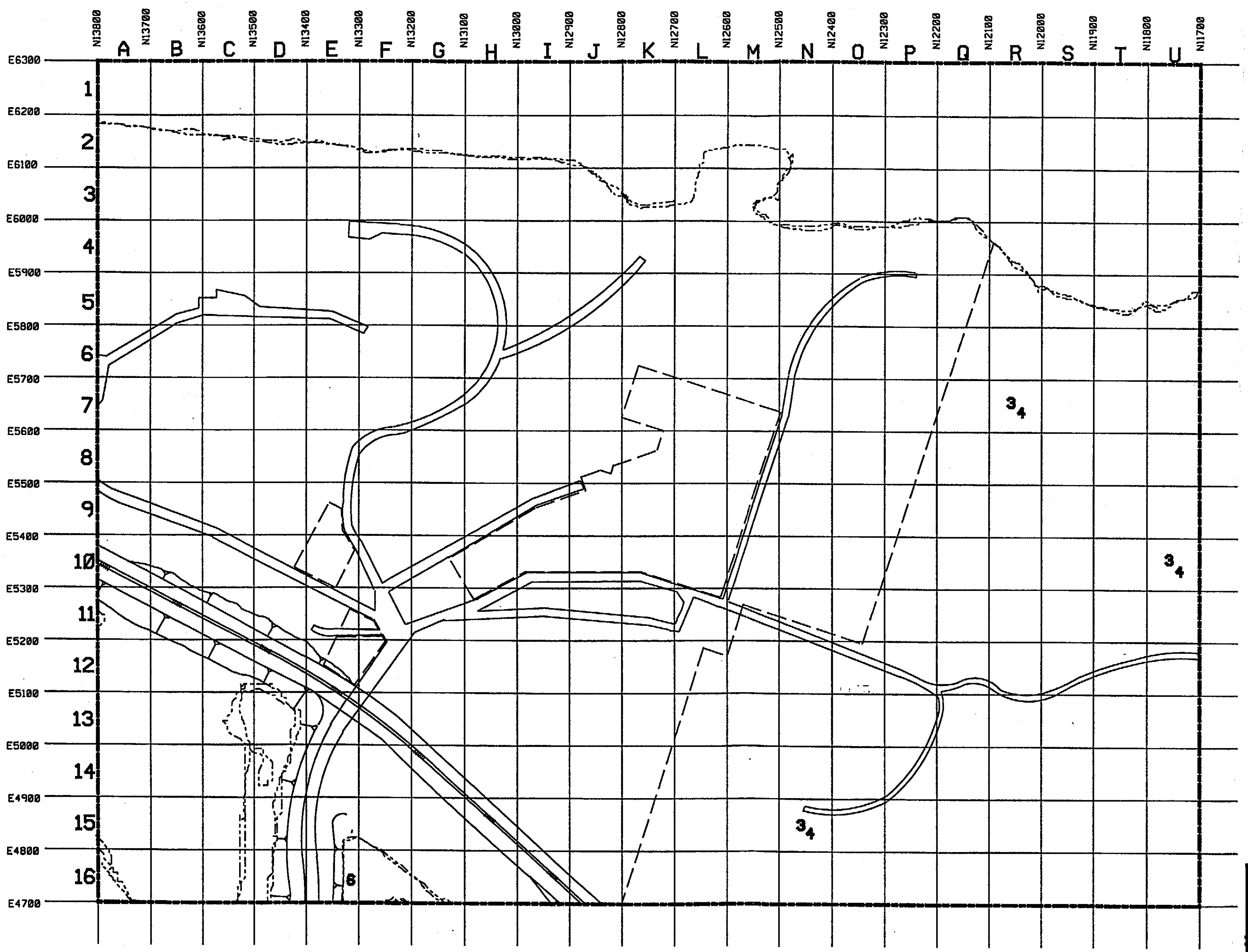



LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-1
Radiological Analysis Samples
Trojan Site, Zones 1 Through 16

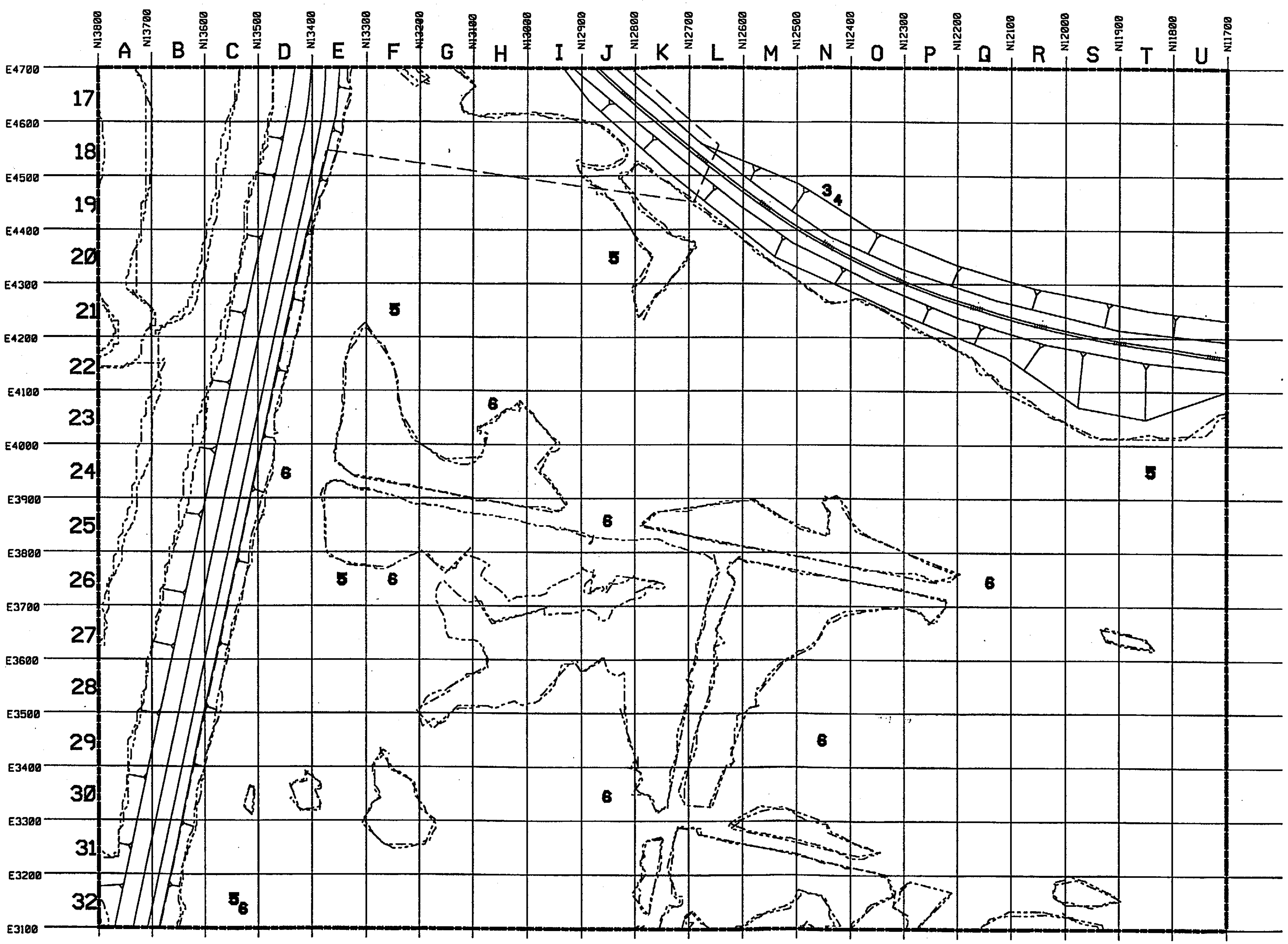



## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-2
Radiological Analysis Samples
Zone 1




## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-3
Radiological Analysis Samples
Zone 2

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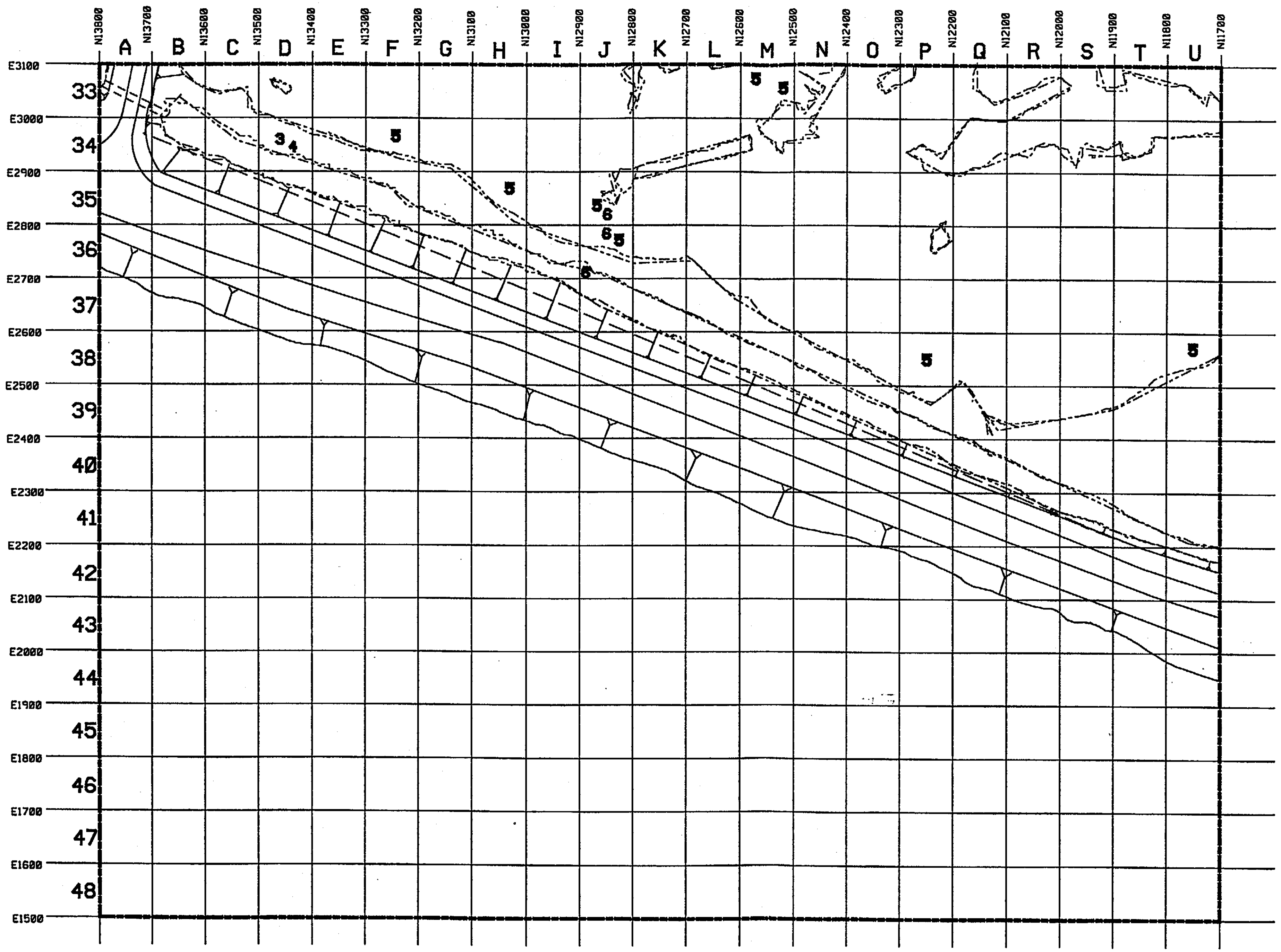


## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-4
Radiological Analysis Samples
Zone 3

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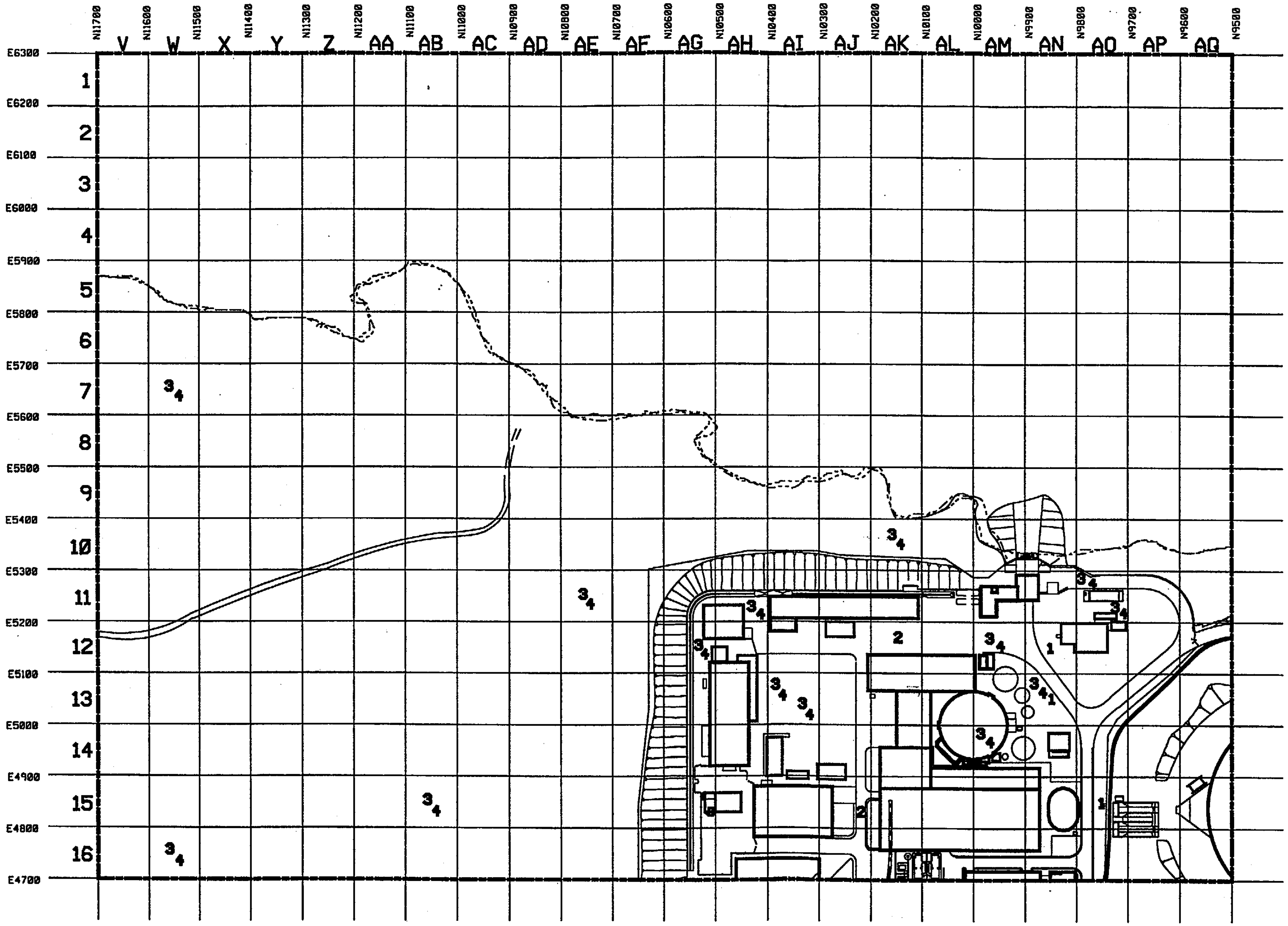


## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-5
Radiological Analysis Samples
Zone 5

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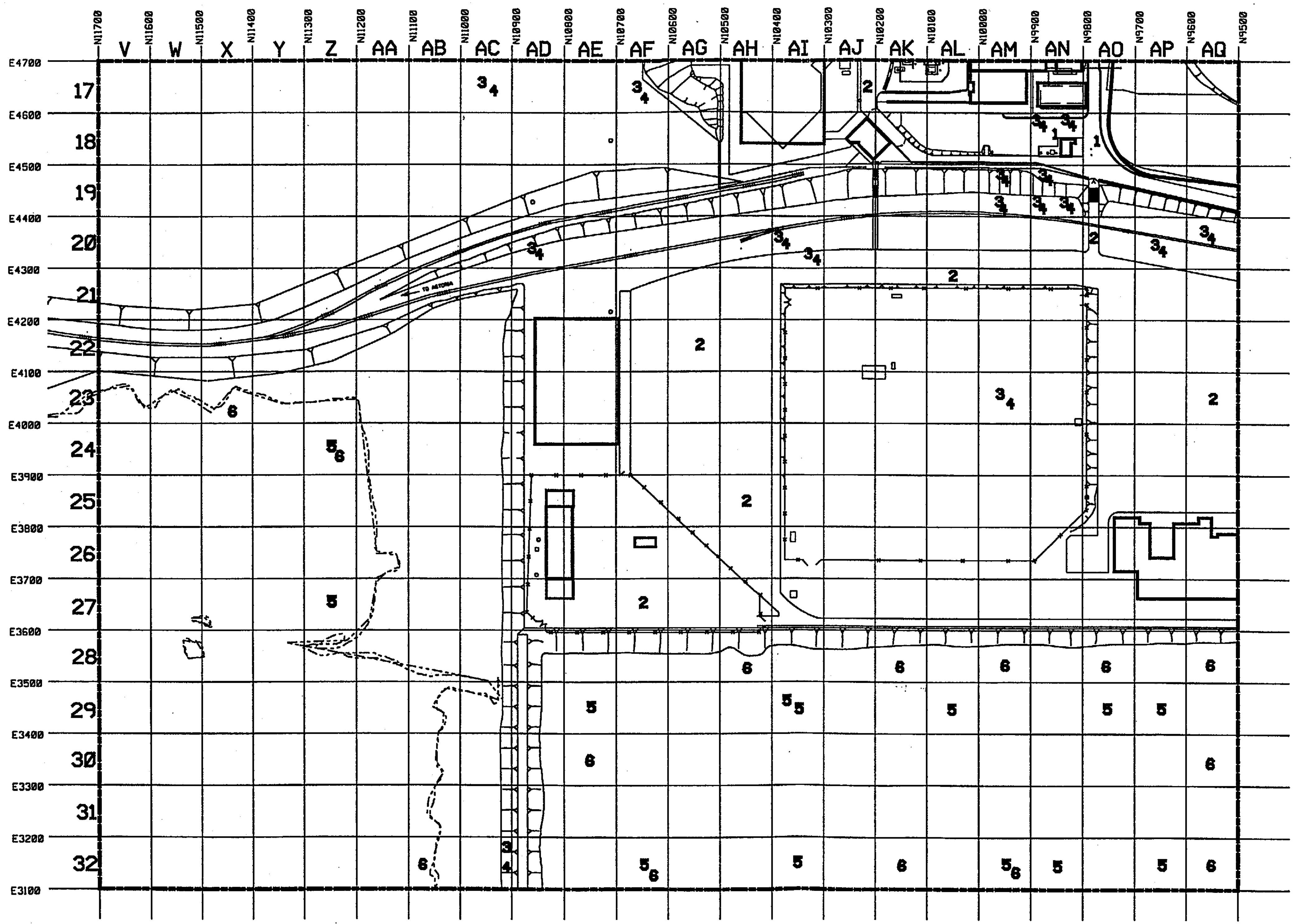


## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-6
Radiological Analysis Samples
Zone 6

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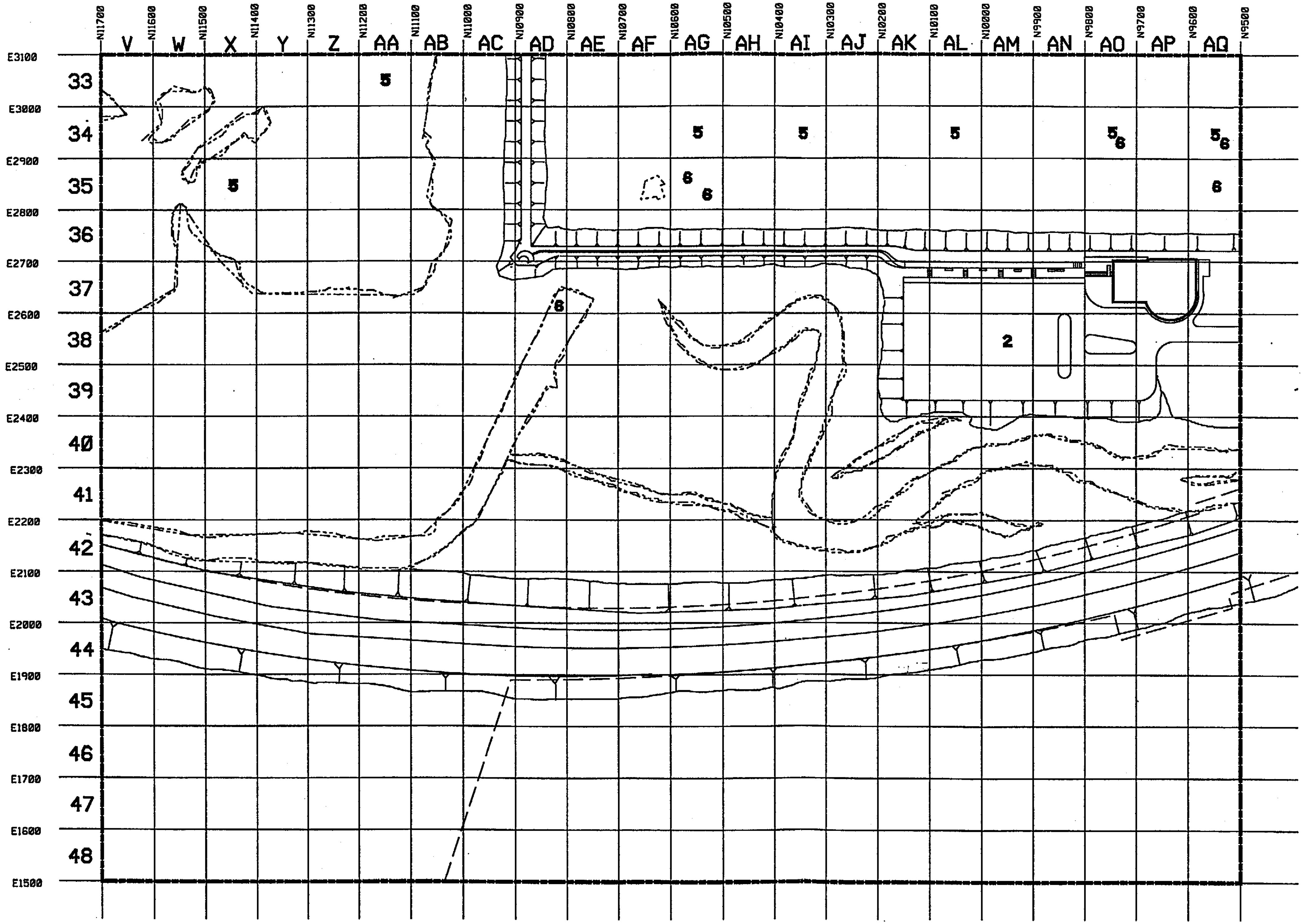


## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-7
Radiological Analysis Samples
Zone 7

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LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-8
Radiological Analysis Samples
Zone 8

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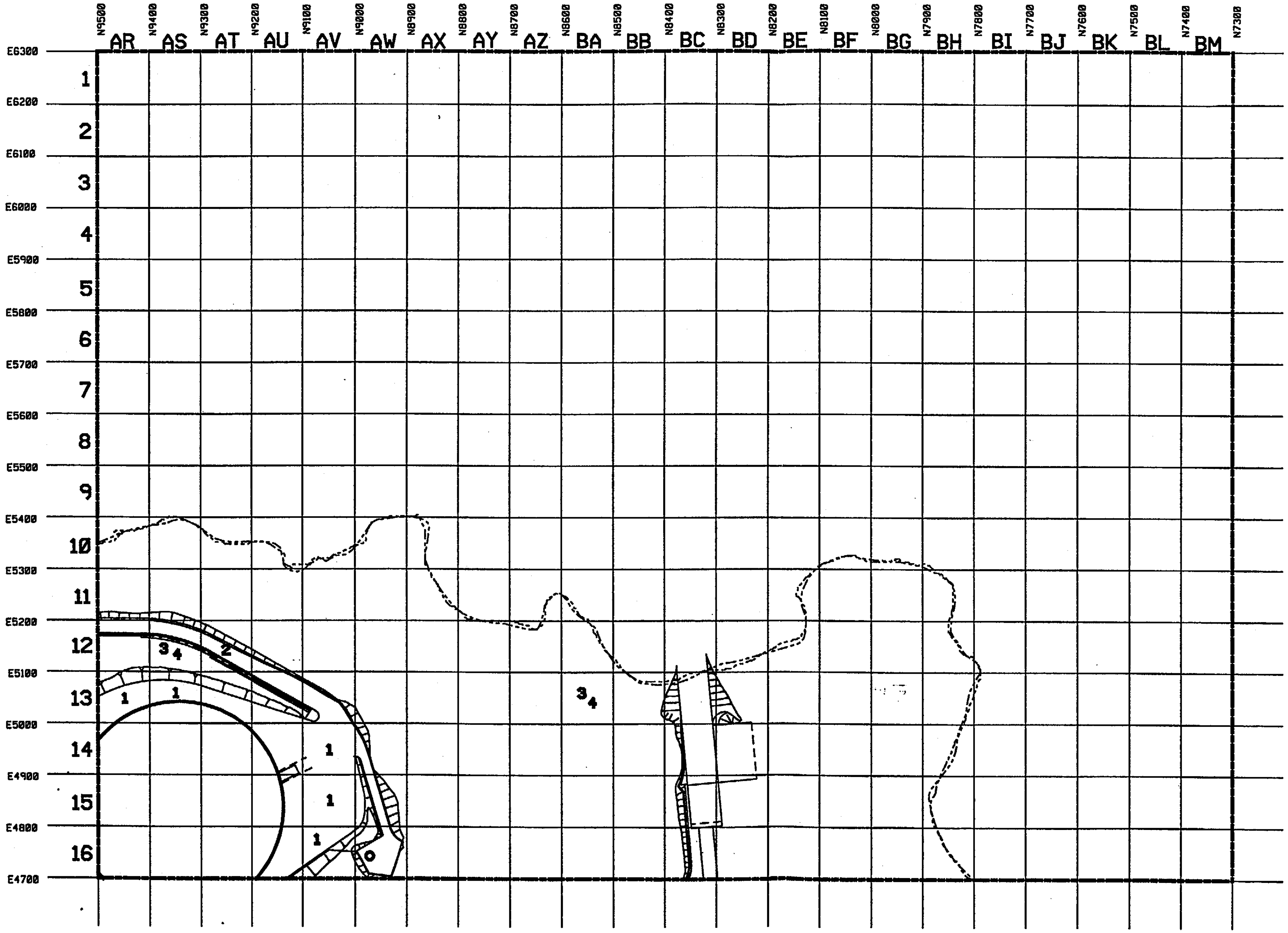


## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-9
Radiological Analysis Samples
Zone 9

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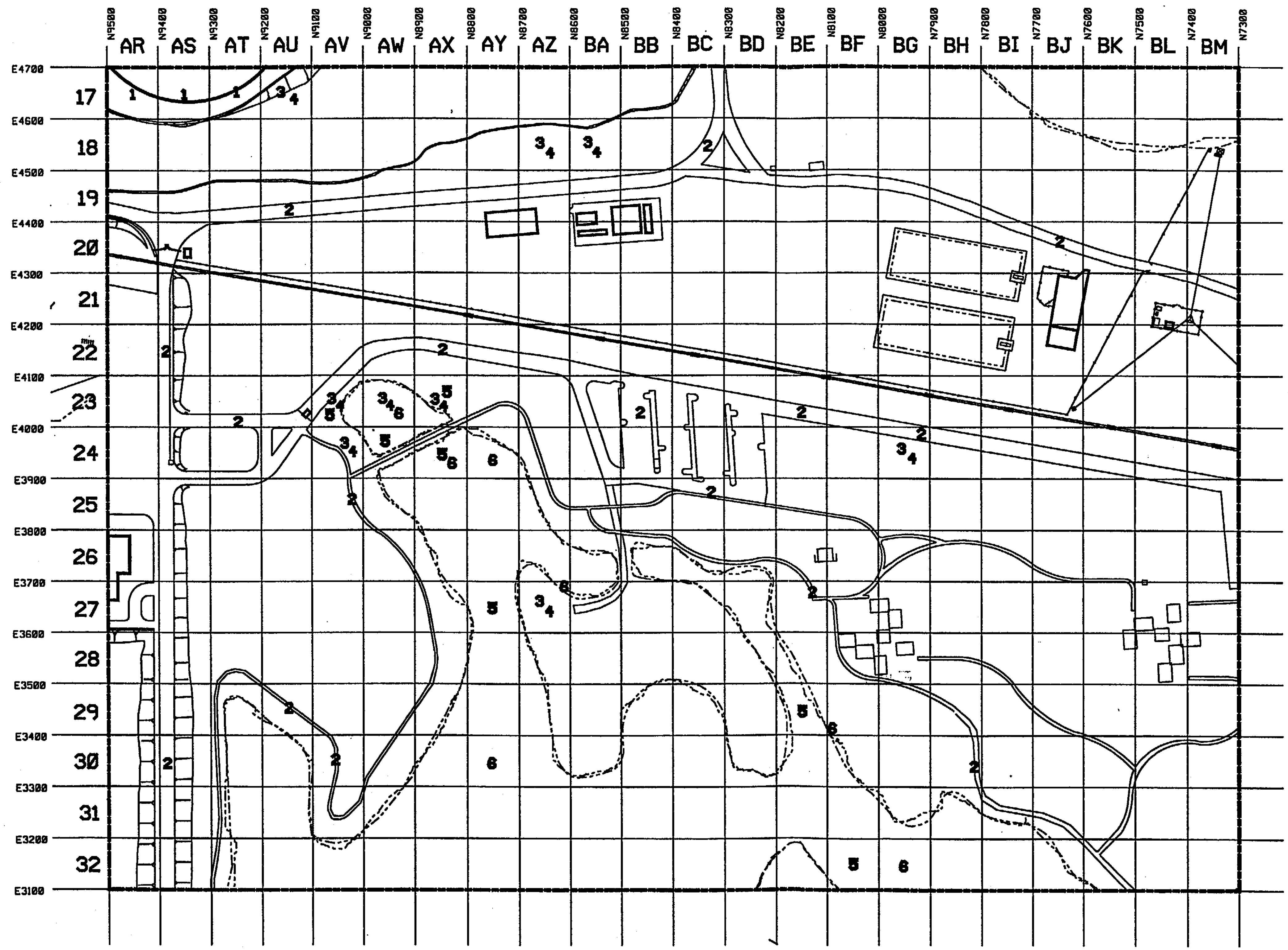


## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-10
Radiological Analysis Samples
Zone 10

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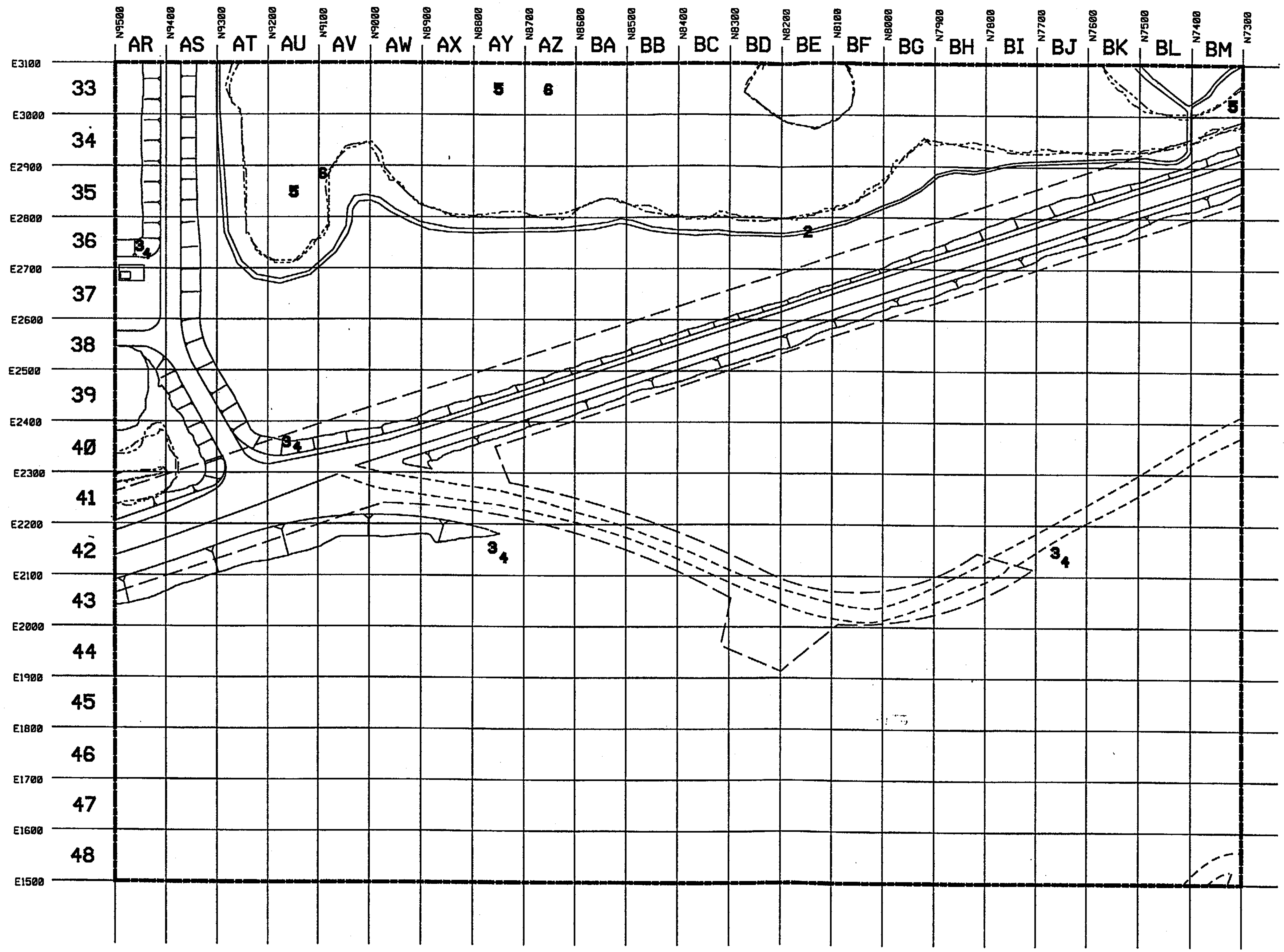


## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-11
Radiological Analysis Samples
Zone 11

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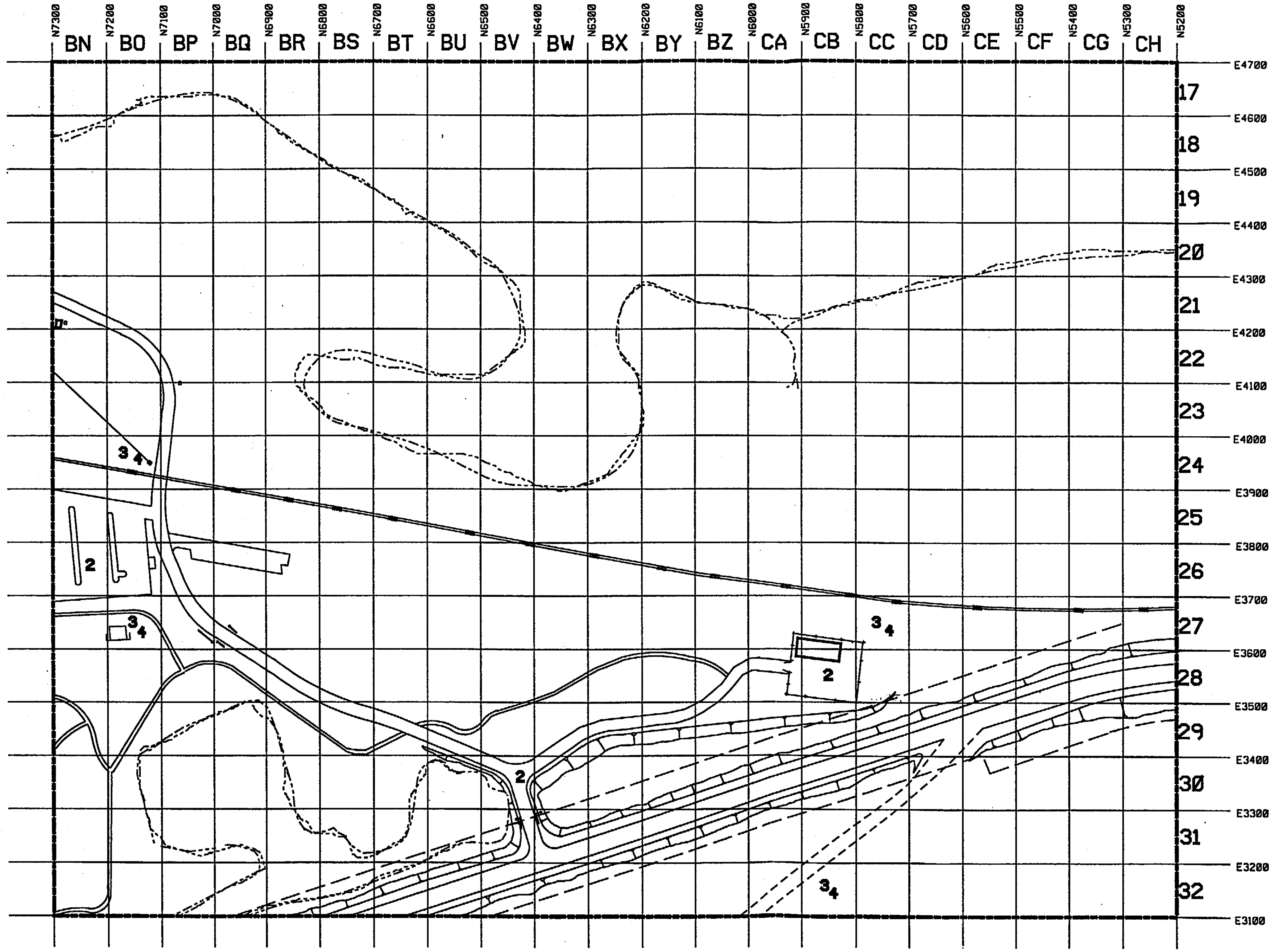


## LEGEND

1 - PAVEMENT SAMPLE
2 - PAVEMENT/DIRECT SURFACE MEASUREMENT
3 - SOIL SAMPLE
4 - DIRECT RADIATION MEASUREMENT
5 - BOTTOM SEDIMENT SAMPLE
6 - SURFACE WATER SAMPLE

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-12
Radiological Analysis Samples
Zone 14

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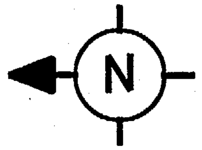


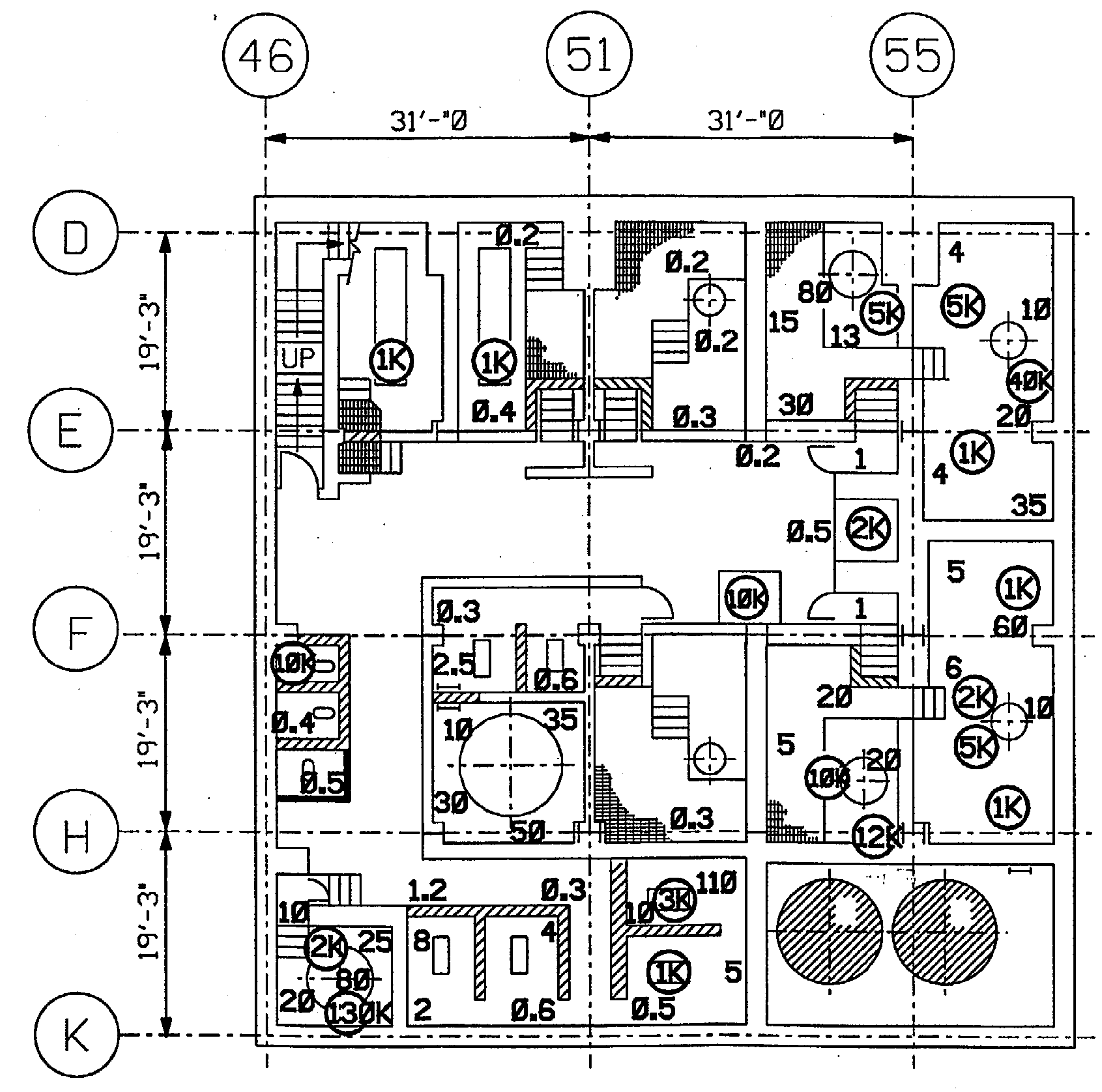


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-13
Radiological Survey Data
Auxiliary Building Elevation 5 FT

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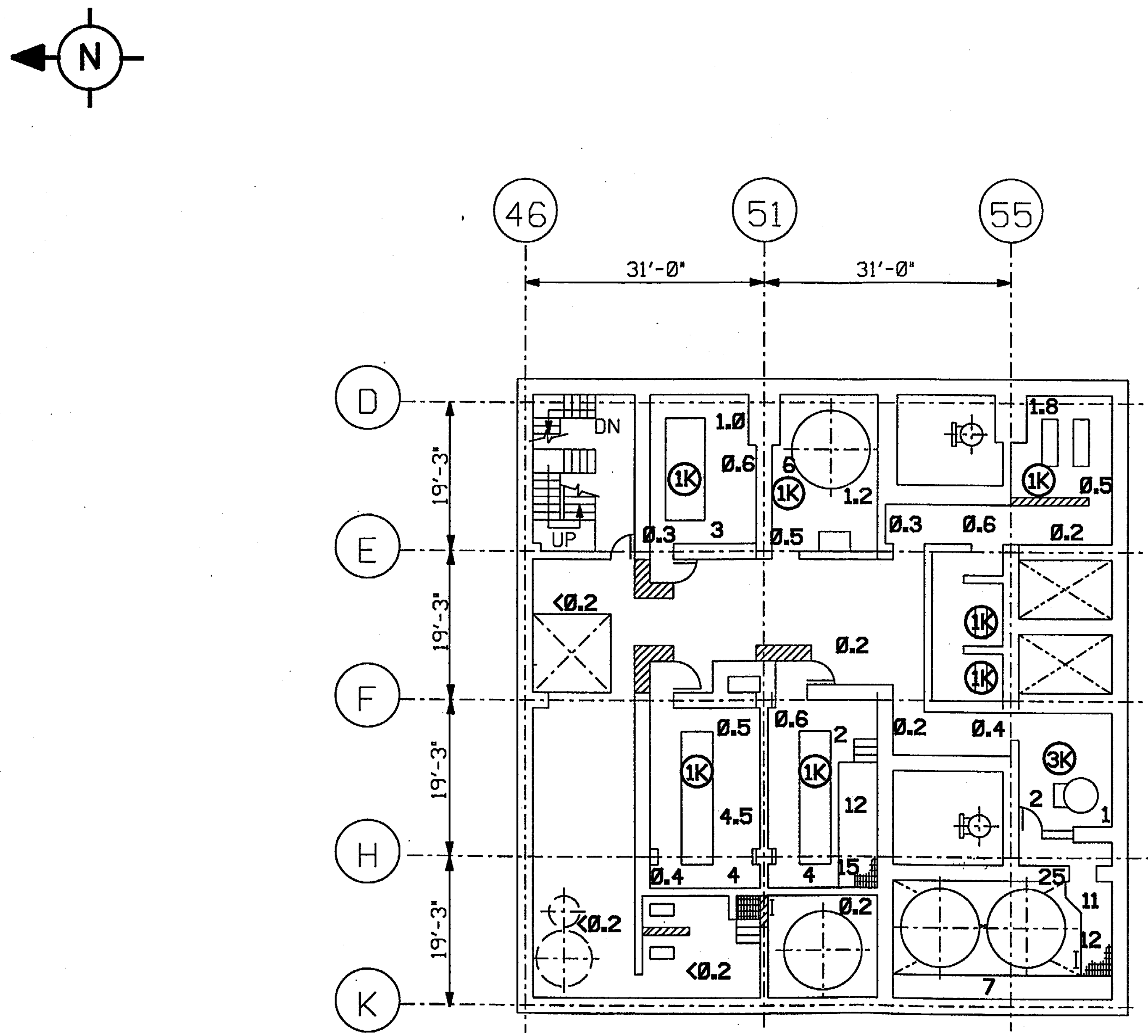


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND LICENSING TERMINATION PLAN

Figure3-14
Radiological Survey Data
Auxiliary Building Elevation 25 FT

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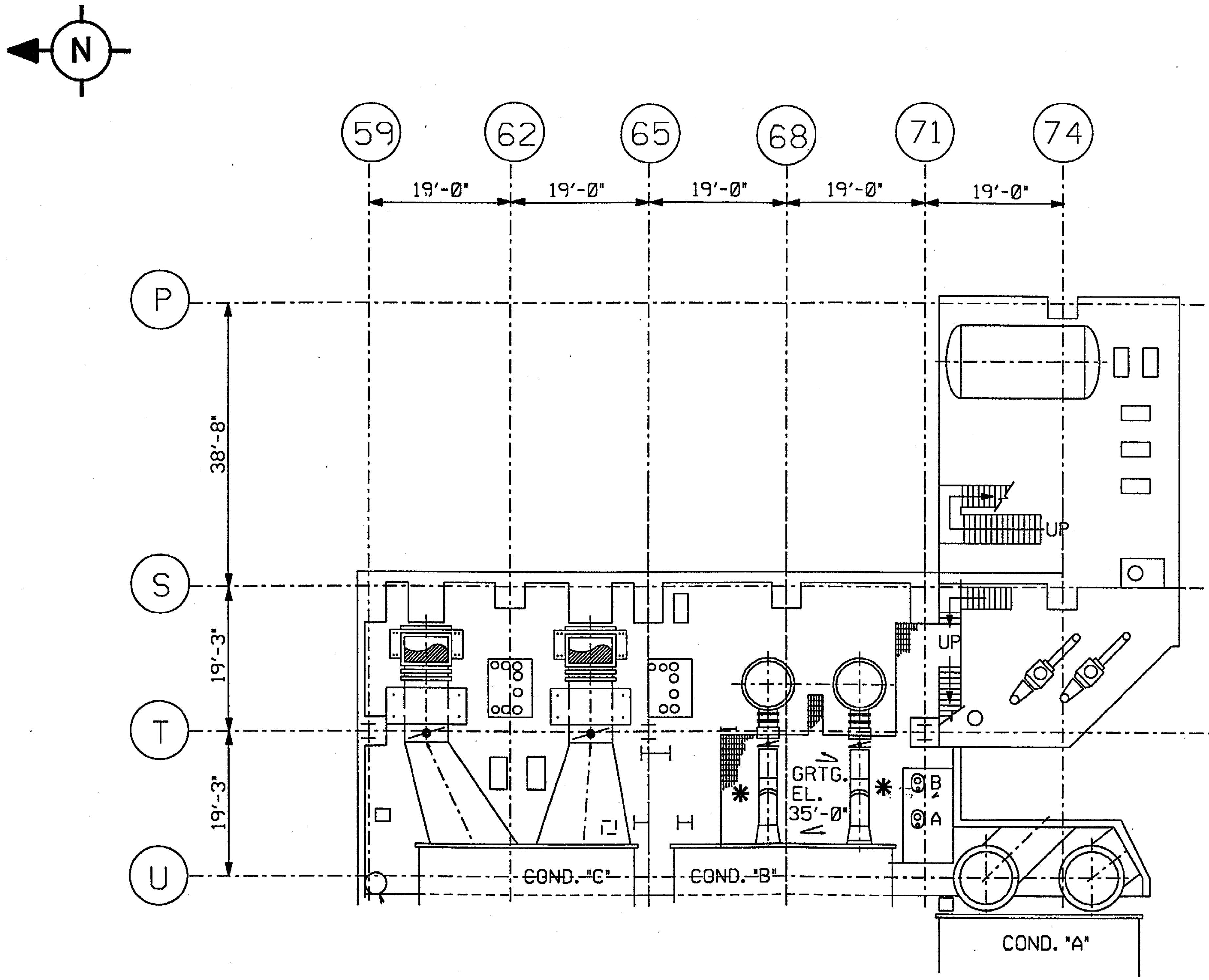


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

* TURBINE BUILDING SUMP SURFACES ARE POSSIBLY CONTAMINATED BUT NOT SURVEYED DUE TO WATER IN THE SUMP.

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-15
Radiological Survey Data
Turbine Building Elevation 27 FT

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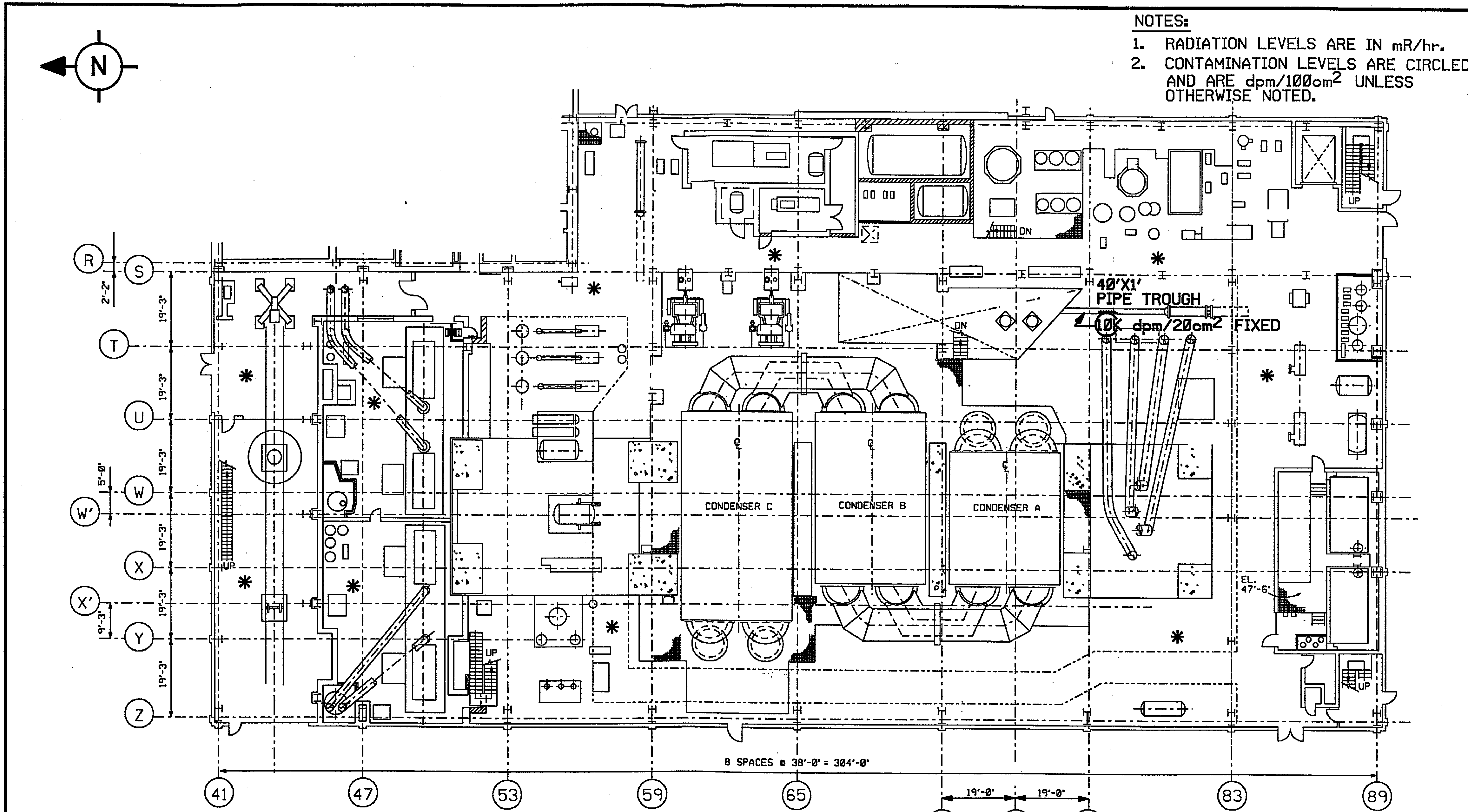


✱ SMEAR SURVEYS OF ALL 45FT TURBINE BUILDING SURFACES ≤ 10.1 dpm/100cm$^2$ ; ALL RADIATION LEVELS ≤ 0.2 mR/hr.

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DECOMMISSIONING PLAN AND
LICENSE TERMINATION PLAN

Figure 3-16
Radiological Survey Data
Turbine Building Elevation 45 FT

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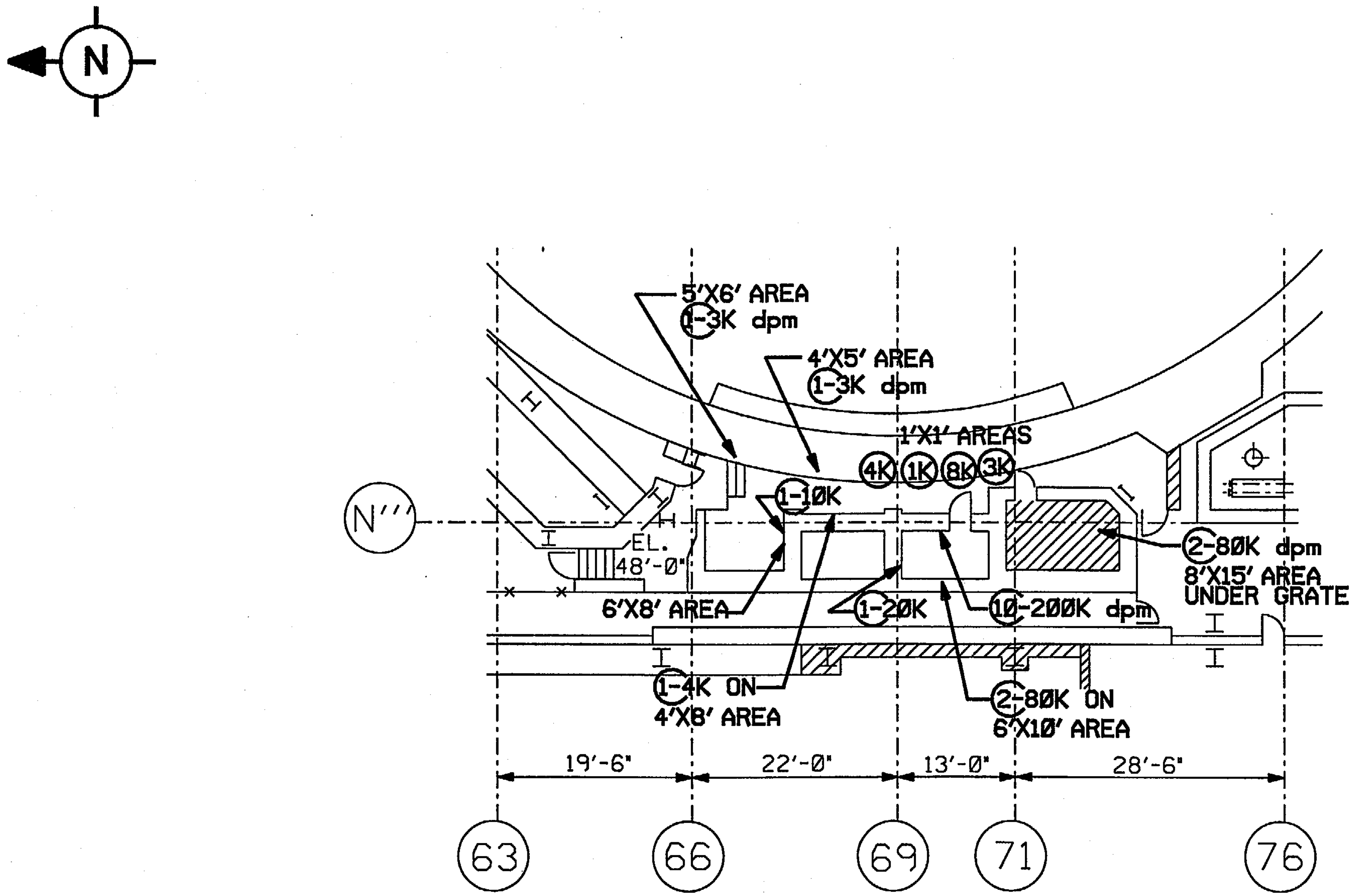


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE dpm/20cm$^2$ UNLESS OTHERWISE NOTED.

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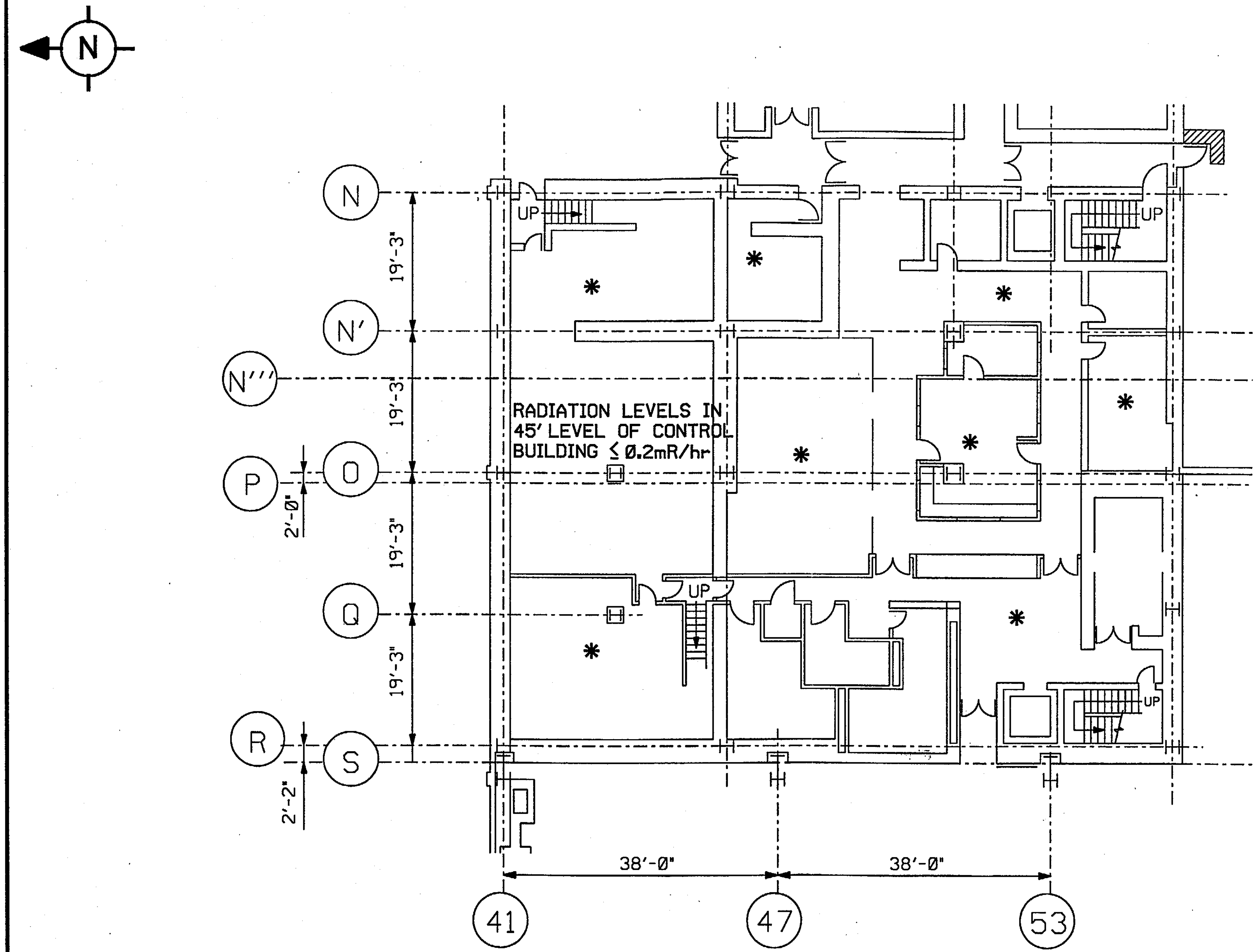
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-17
Radiological Survey Data
Main Steam Support Structure
Elevation 45 FT




* ALL CONTROL BUILDING SURFACES SMEAR SURVEY RESULTS ≤ 12.6 dpm/100cm$^2$

NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

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DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-18
Radiological Survey Data
Control Building Elevation 45 FT

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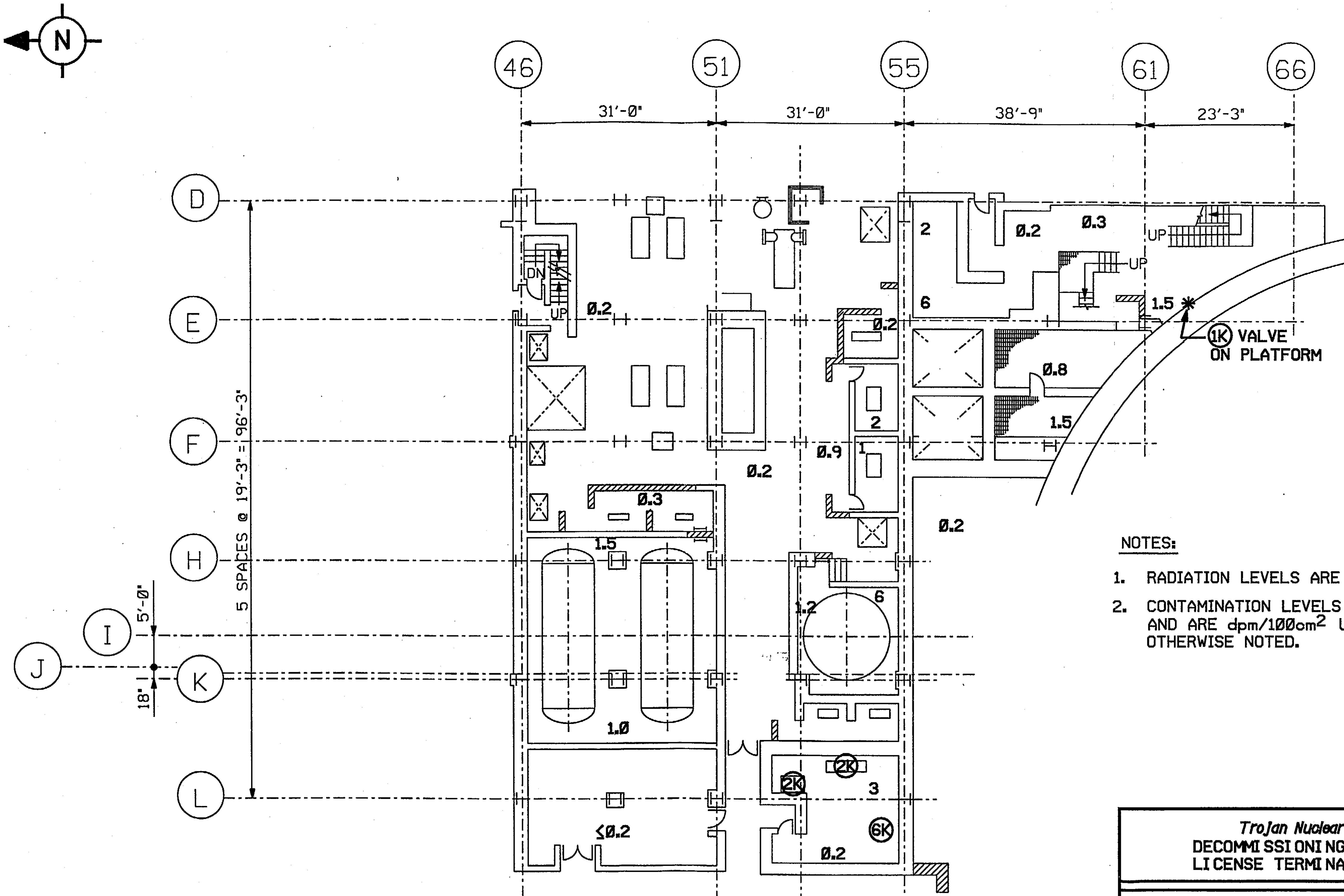


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

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Figure 3-19
Radiological Survey Data
Auxiliary Building Elevation 45 FT

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NOTES:
1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

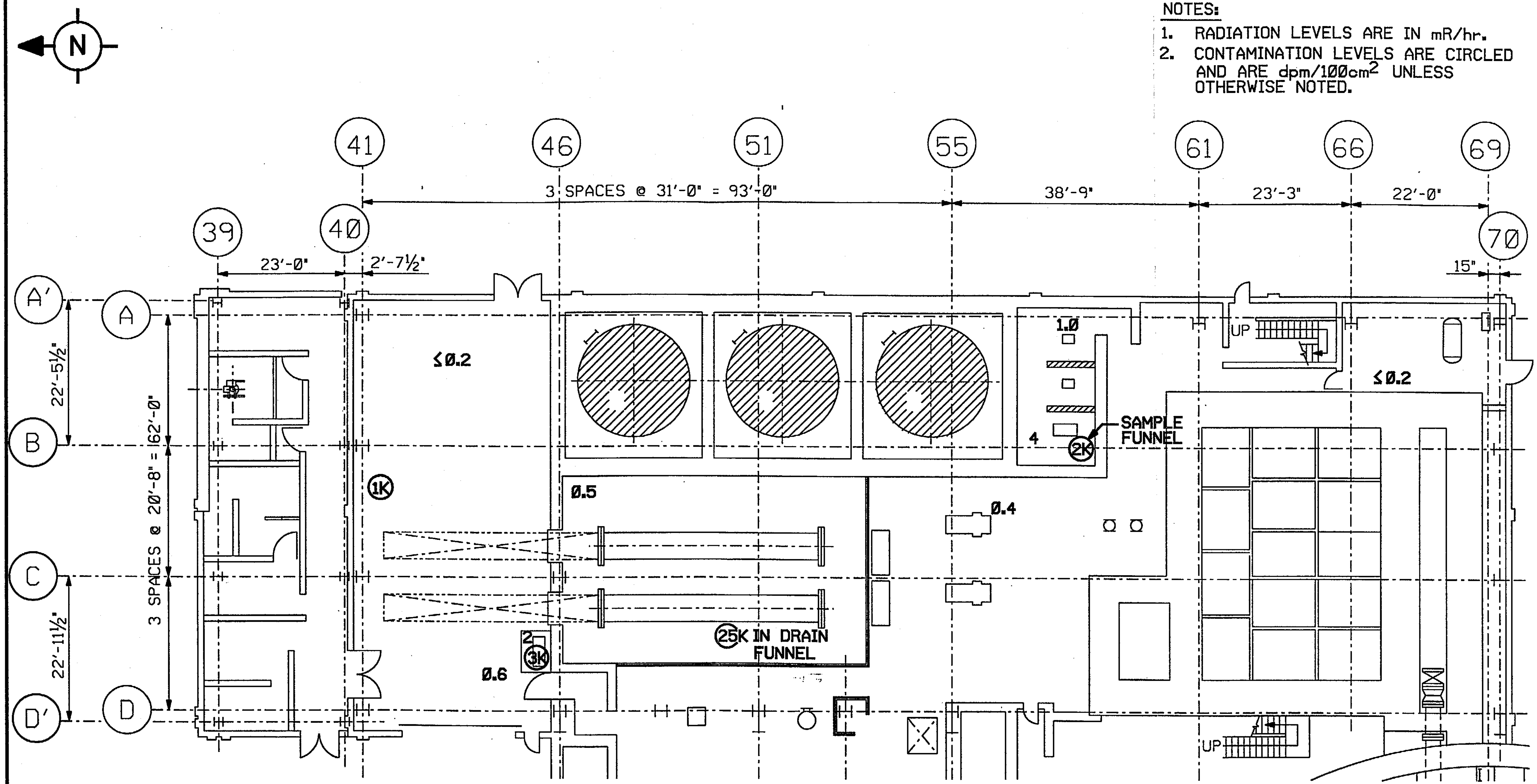


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Figure 3-20
Radiological Survey Data
Fuel Building Elevation 45 FT

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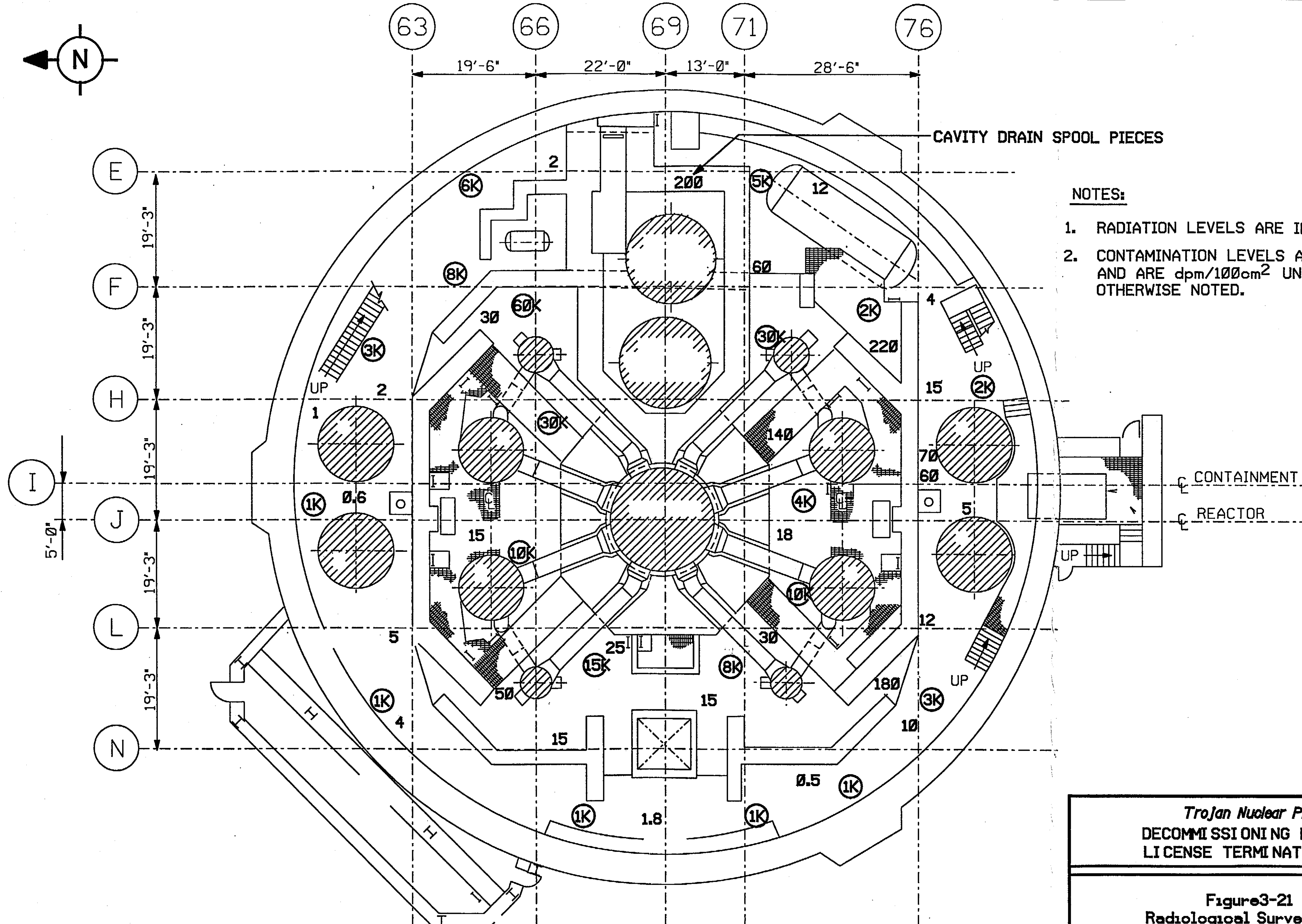


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

*Trojan Nuclear Plant*
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Figure3-21
Radiological Survey Data
Containment Elevation 45 FT

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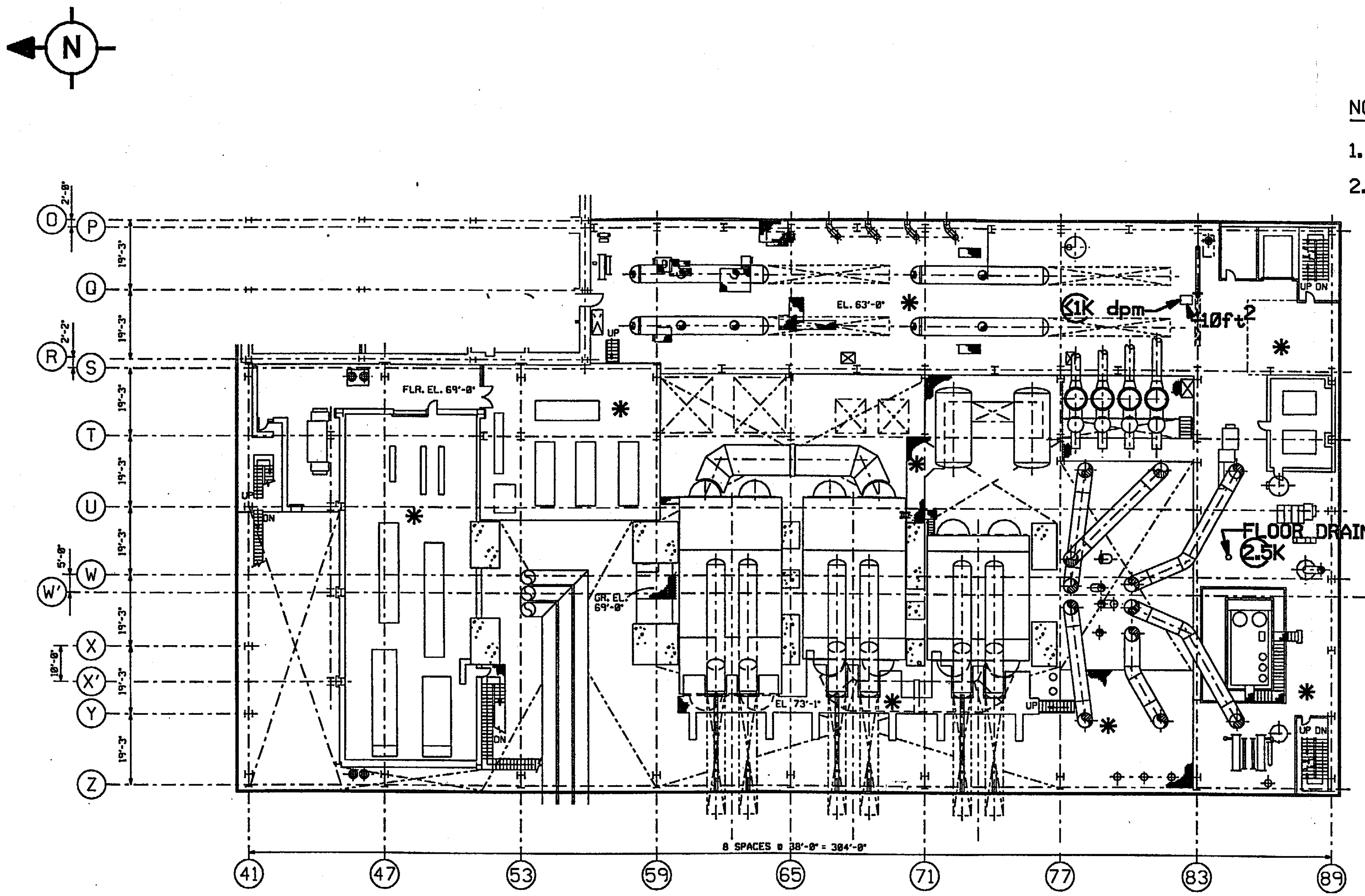


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/20cm$^2$ UNLESS OTHERWISE NOTED.

* SMEAR SURVEYS OF ALL 63' AND 69' TURBINE BUILDING SURFACES ≤13.7 dpm/100cm$^2$.

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Figure 3-22
Radiological Survey Data
Turbine Building Elevation 63 FT

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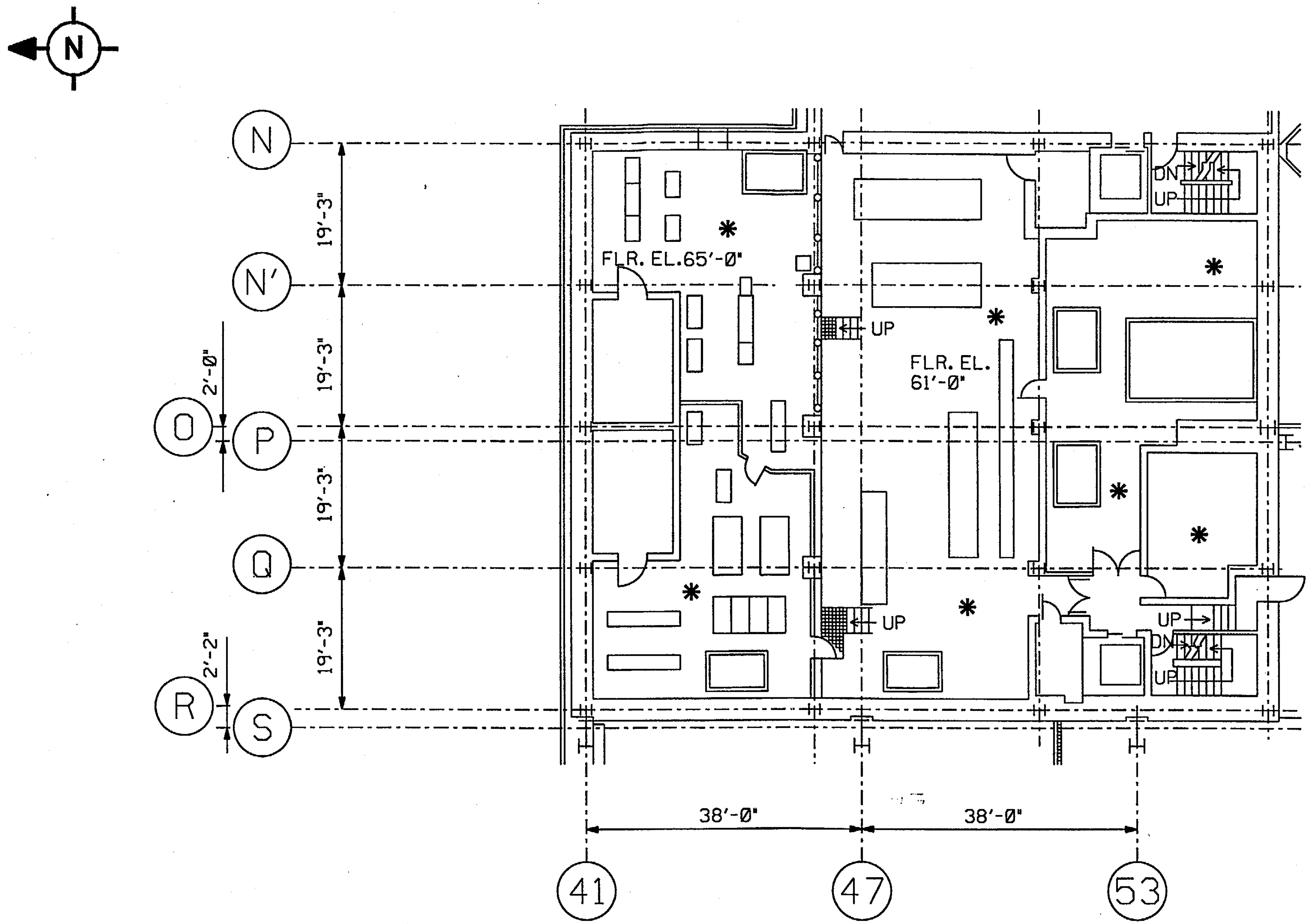


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

* SMEAR SURVEYS OF ALL 61' AND 65' CONTROL BUILDING SURFACES $\leq$12.3 dpm/100cm$^2$

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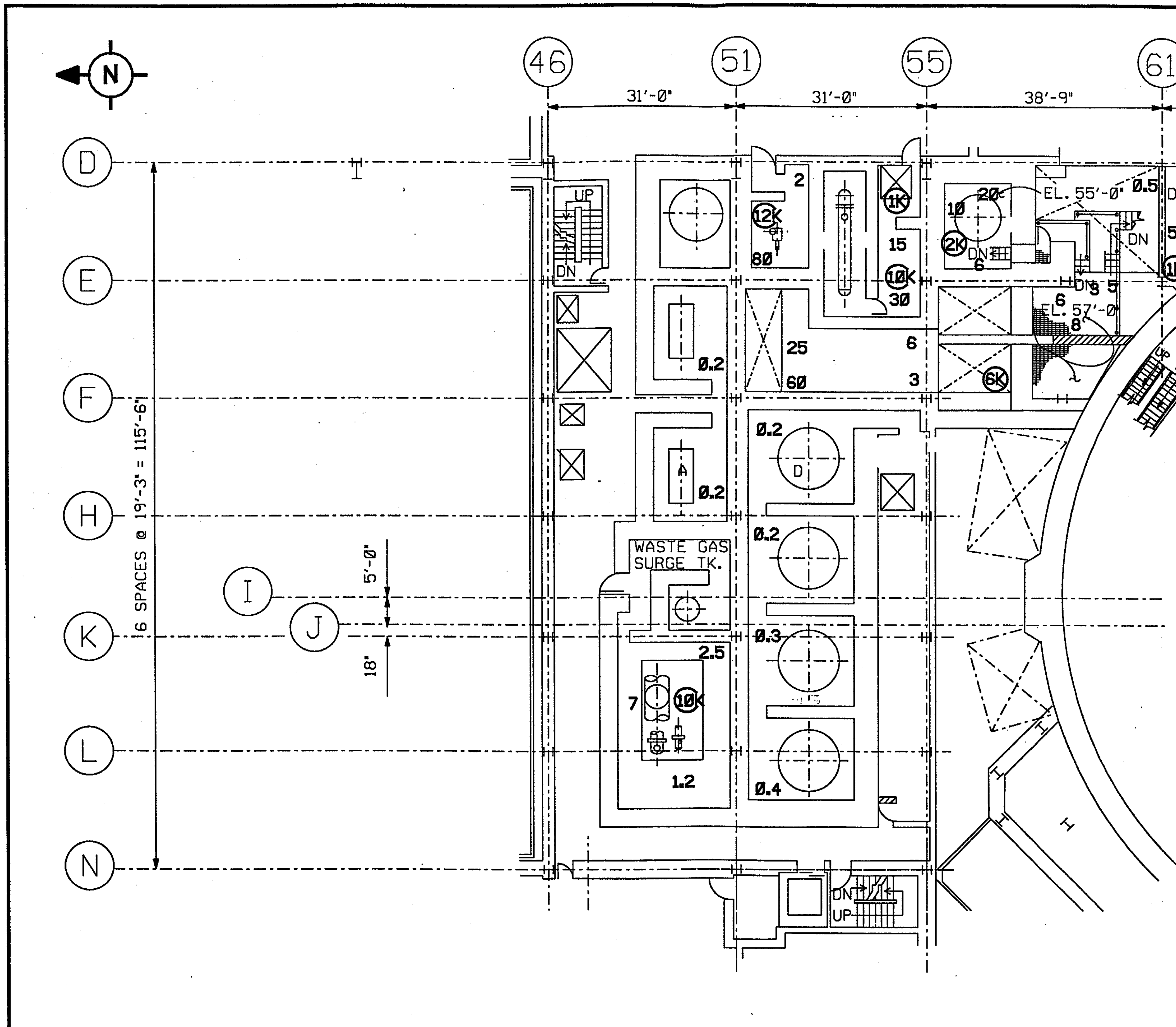
Figure 3-23
Radiological Survey Data
Control Building
Elevation 61 FT And 65 FT




NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

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DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-24
Radiological Survey Data
Auxiliary Building Elevation 61 FT

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NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

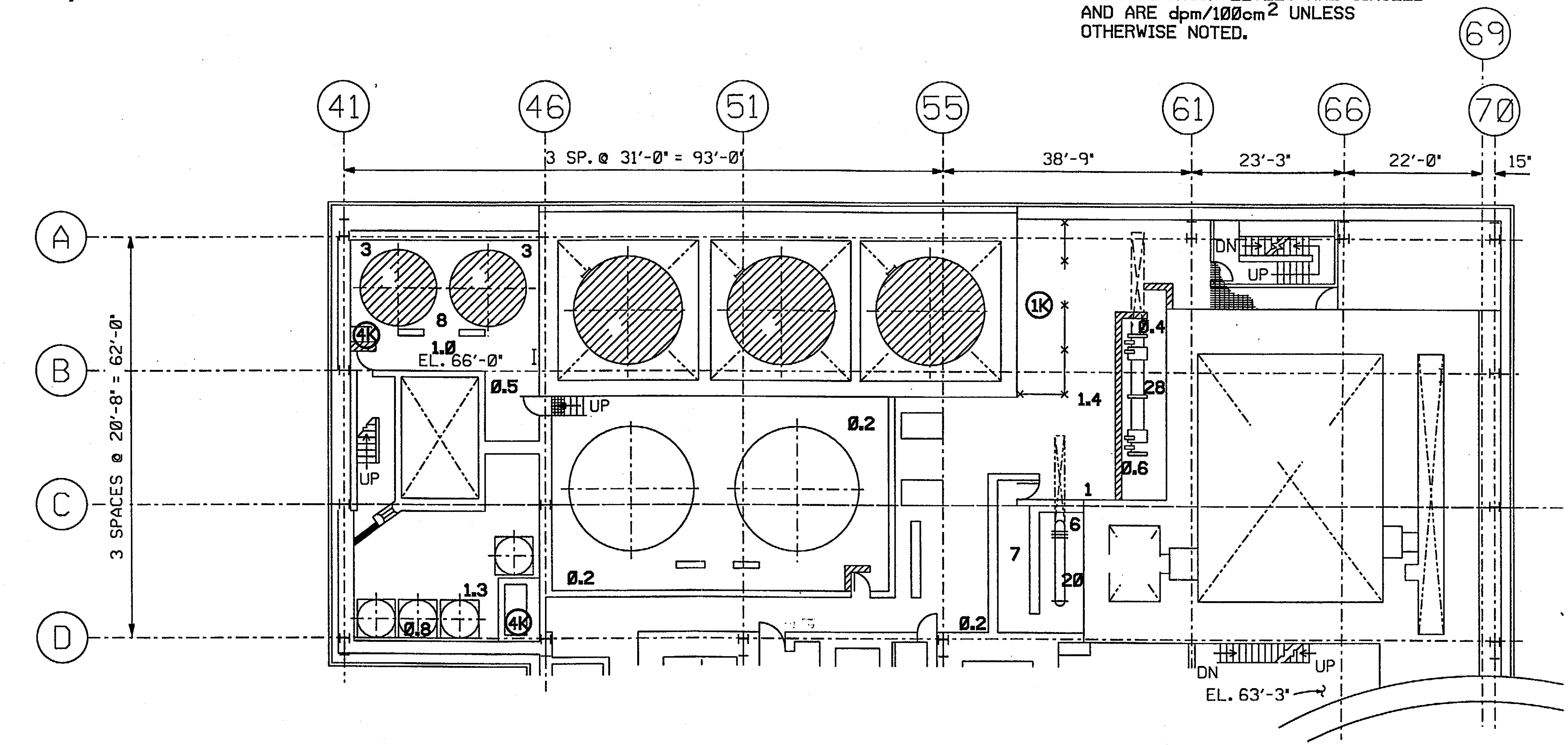


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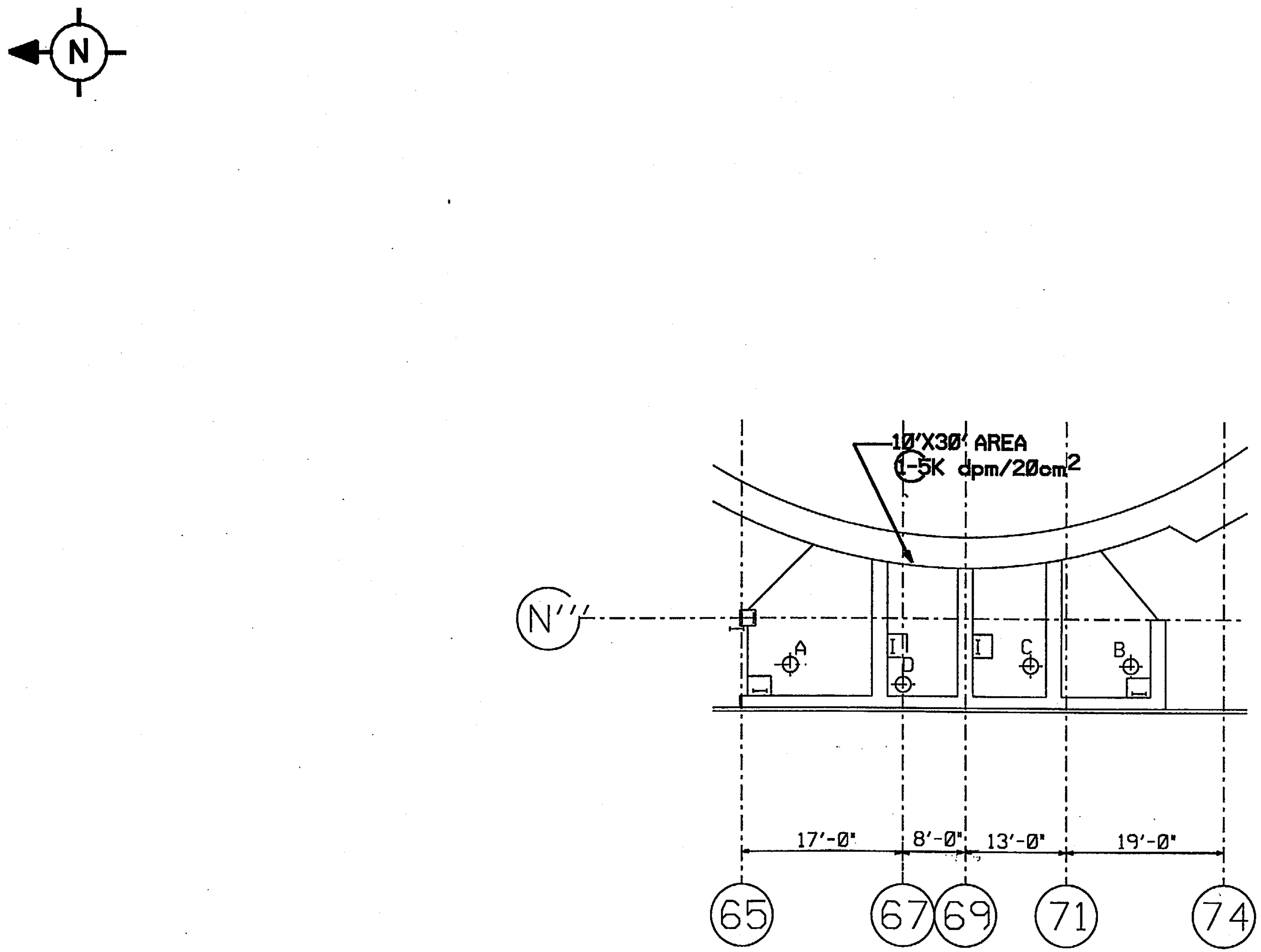
Figure 3-25
Radiological Survey Data
Fuel Building Elevation 61 FT




NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/20cm² UNLESS OTHERWISE NOTED.

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DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-26
Radiological Survey Data
Main Steam Support Structure
Elevation 69 FT

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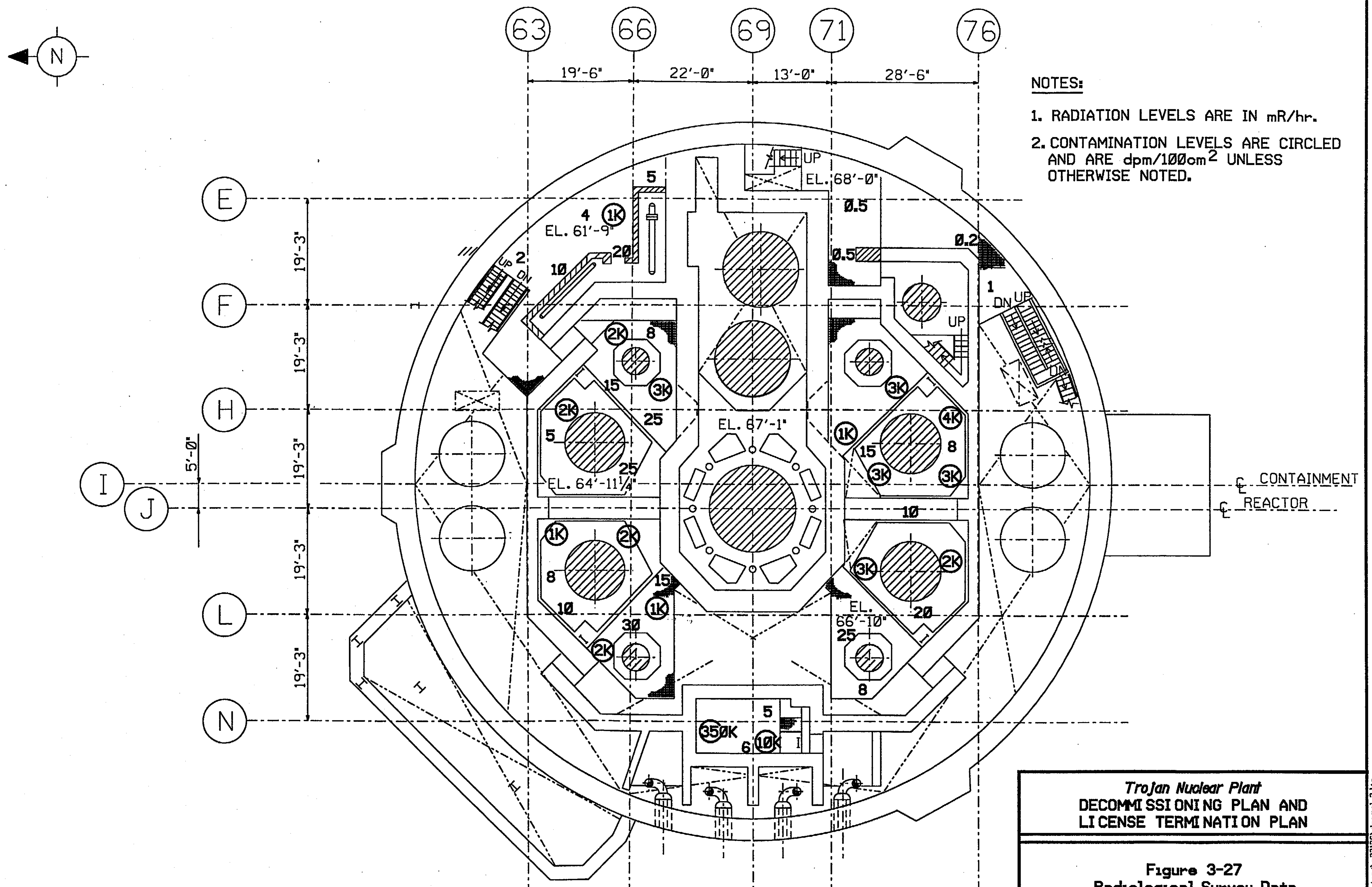


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

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Figure 3-27
Radiological Survey Data
Containment Elevation 61 FT

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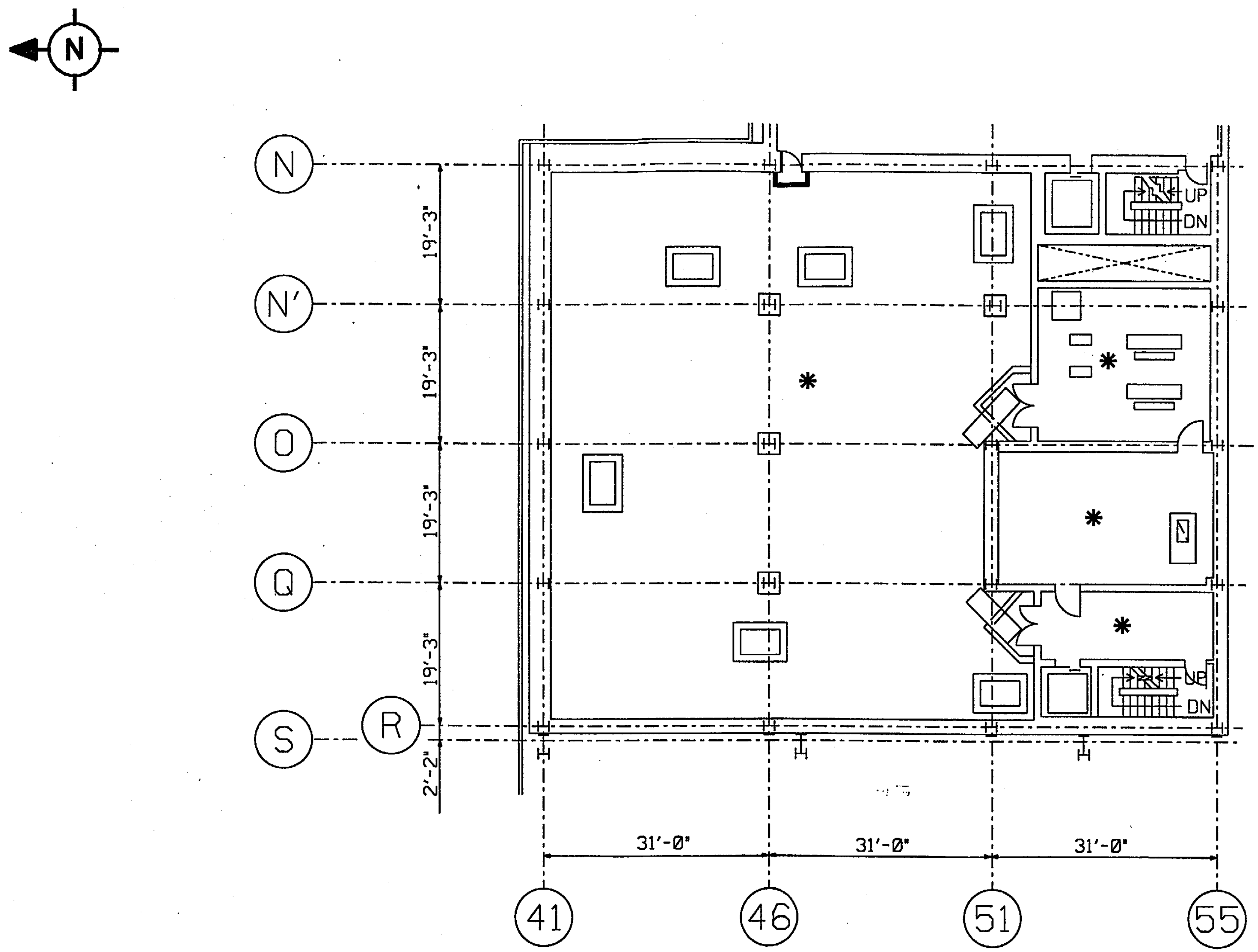


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

* SMEAR SURVEYS OF ALL 77' CONTROL BUILDING SURFACES $\leq$12.3 dpm/100cm$^2$.

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Figure 3-28
Radiological Survey Data
Control Building Elevation 77 FT

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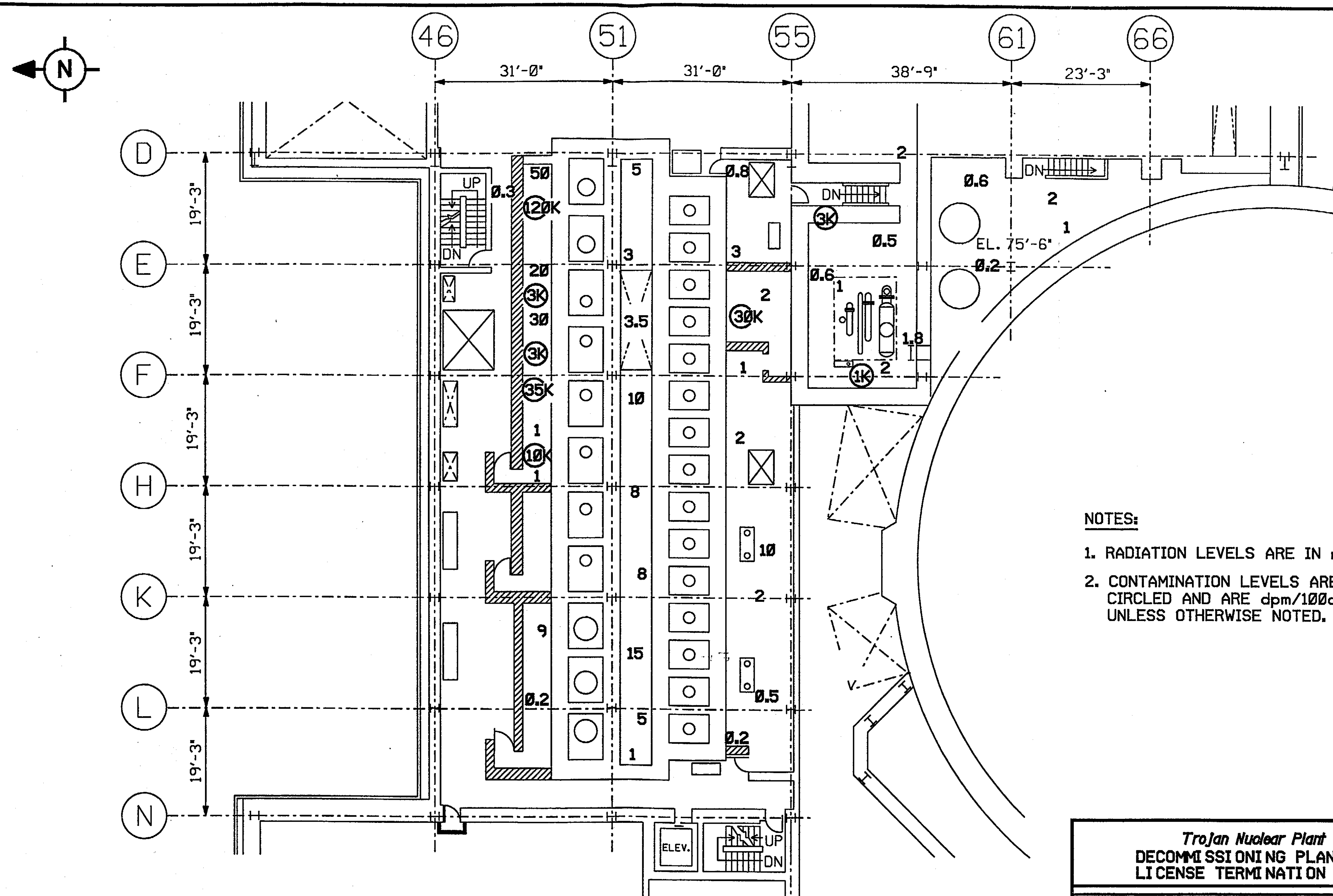


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-29
Radiological Survey Data
Auxiliary Building Elevation 77 FT

NOTES:

1. RADIATION LEVELS ARE IN mR/hr.

2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

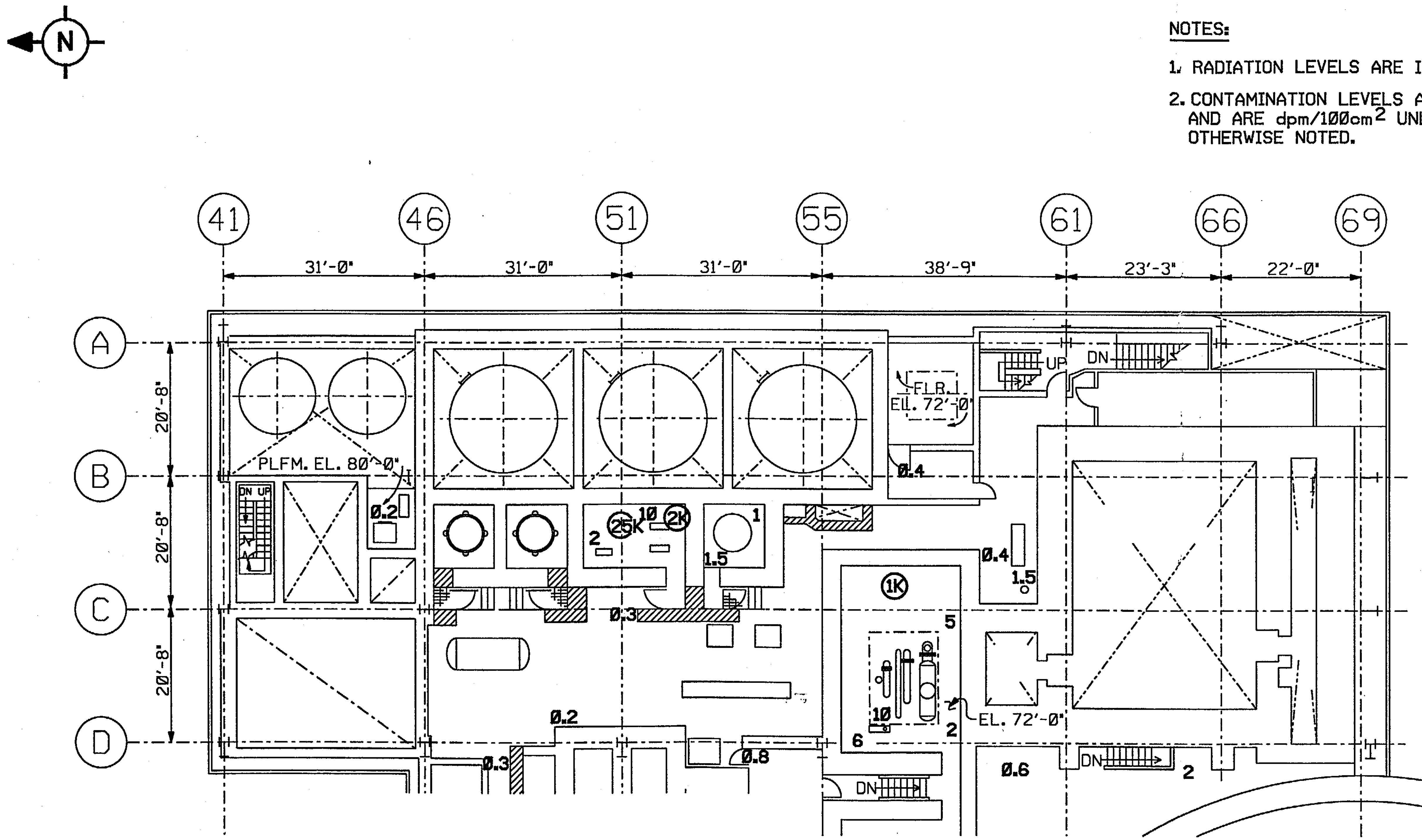


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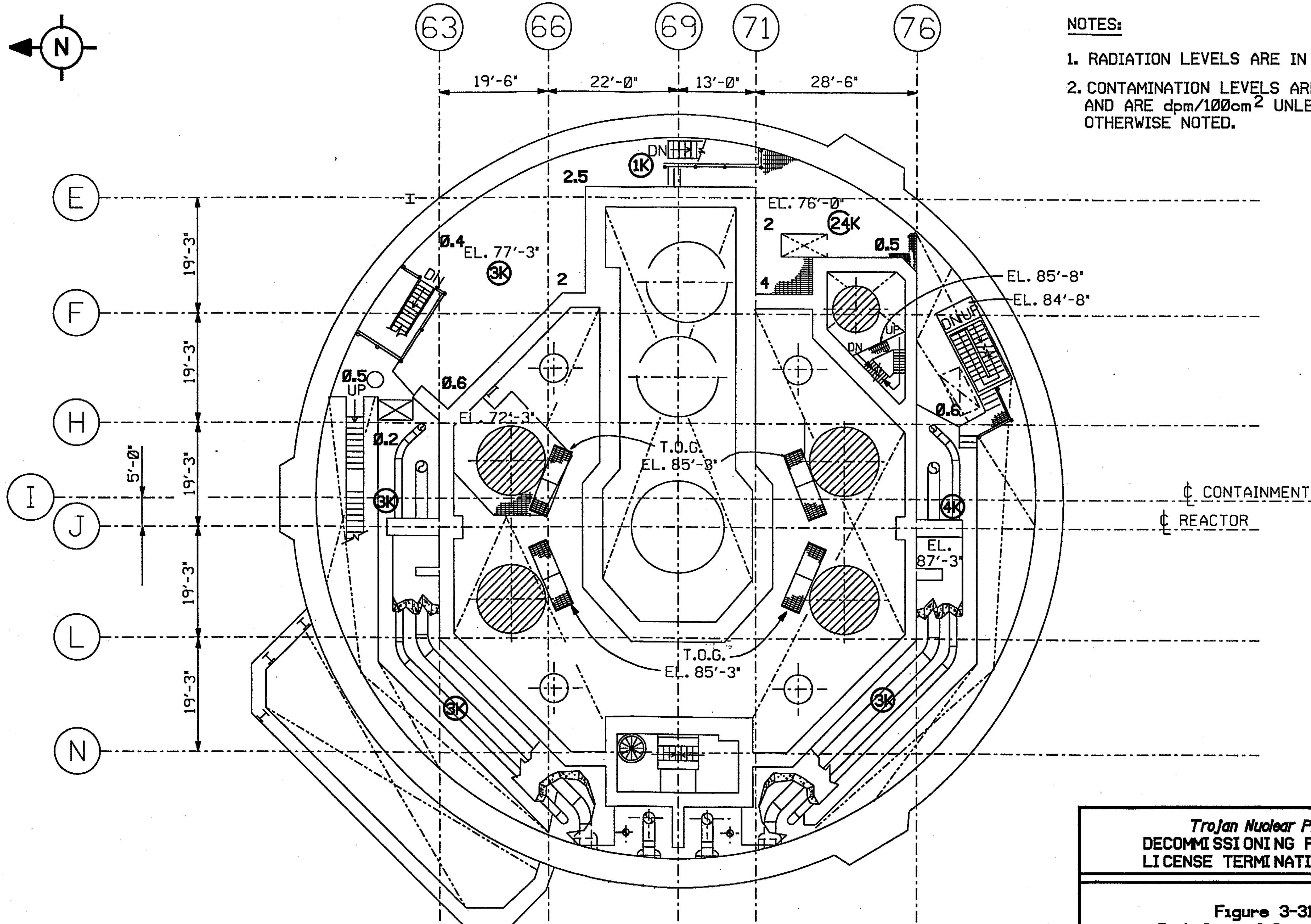
Figure 3-30
Radiological Survey Data
Fuel Building Elevation 77 FT




NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-31
Radiological Survey Data
Containment Elevation 77 FT

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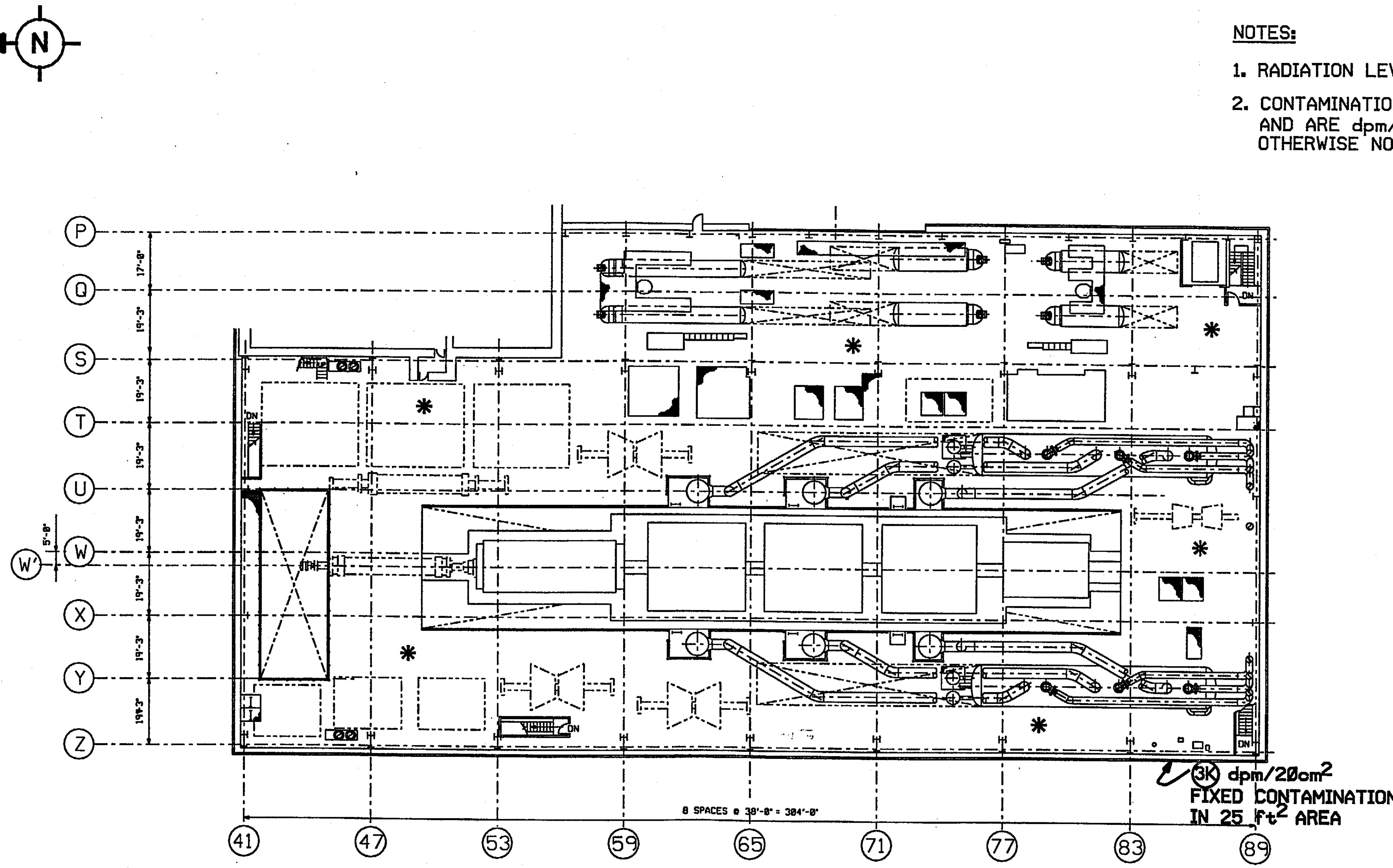
NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.




✱ SMEARS SURVEYS OF ALL 93' TURBINE BUILDING SURFACES ≤12.4 dpm/100cm$^2$.

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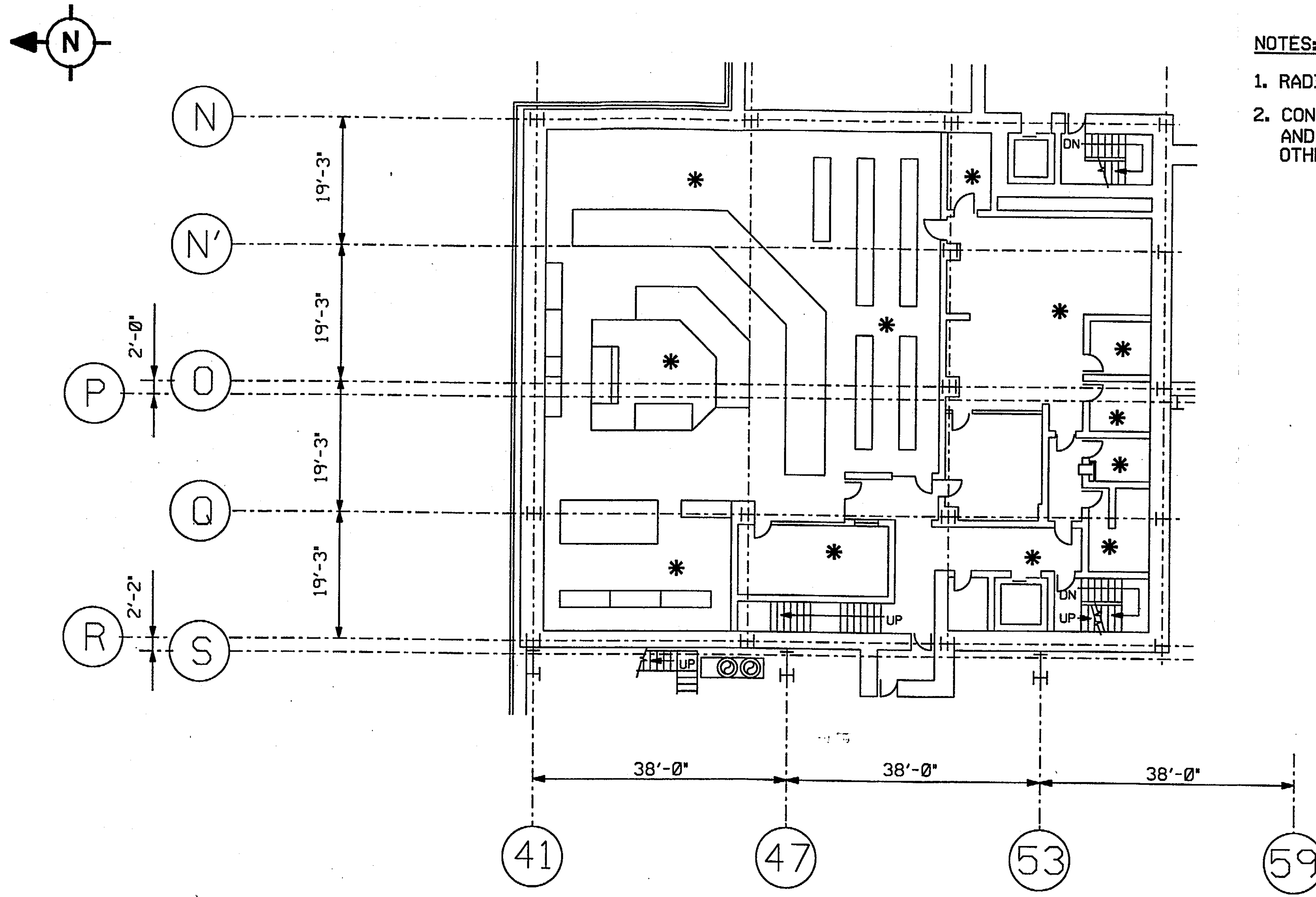
Figure 3-32
Radiological Survey Data
Turbine Building Elevation 93 FT




NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

* SMEAR SURVEYS OF ALL 93' CONTROL BUILDING SURFACES $\leq$10.5dpm/100cm$^2$

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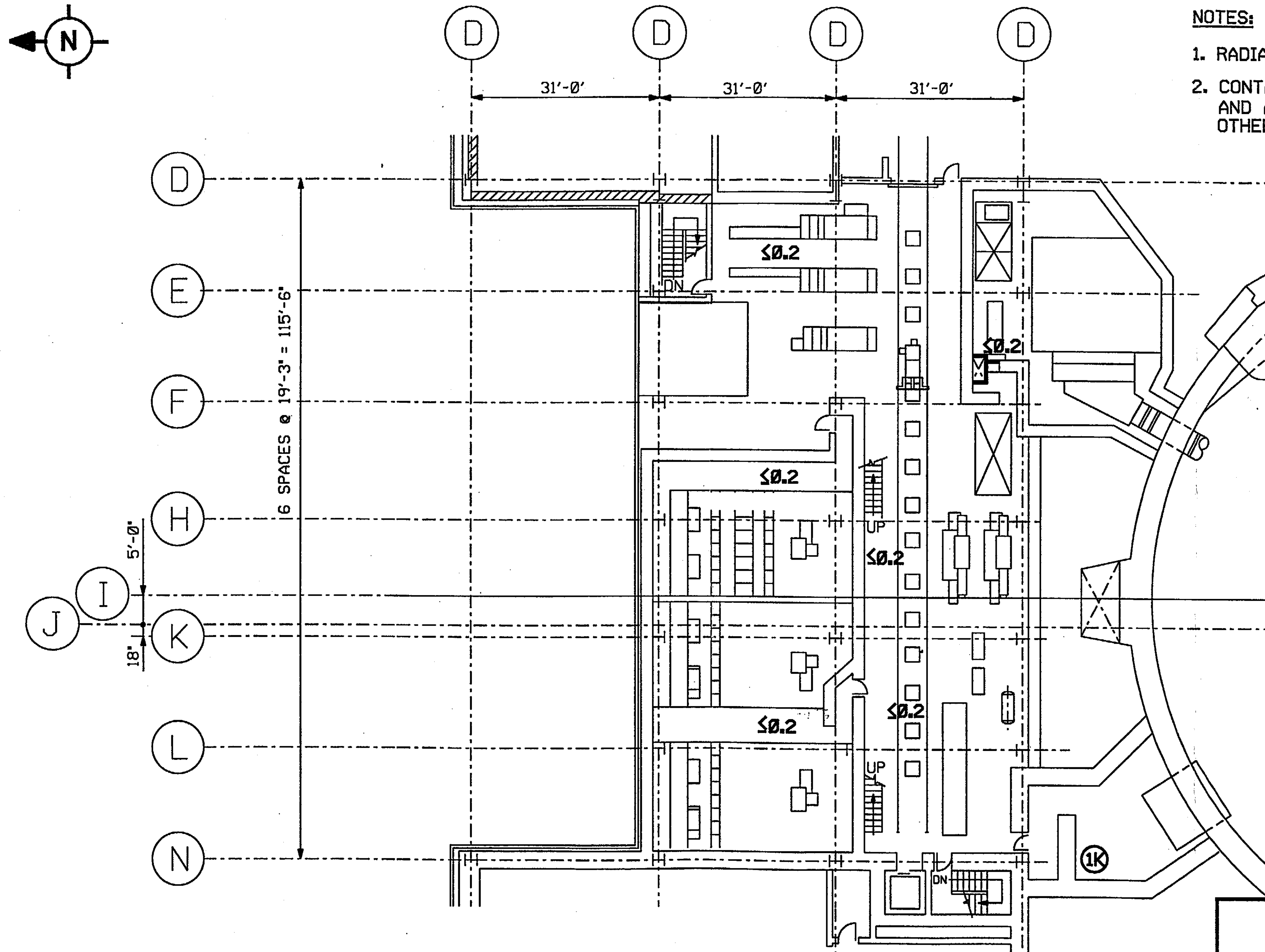
Figure 3-33
Radiological Survey Data
Control Building Elevation 93 FT




NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

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Figure 3-34
Radiological Survey Data
Auxiliary Building Elevation 93 FT

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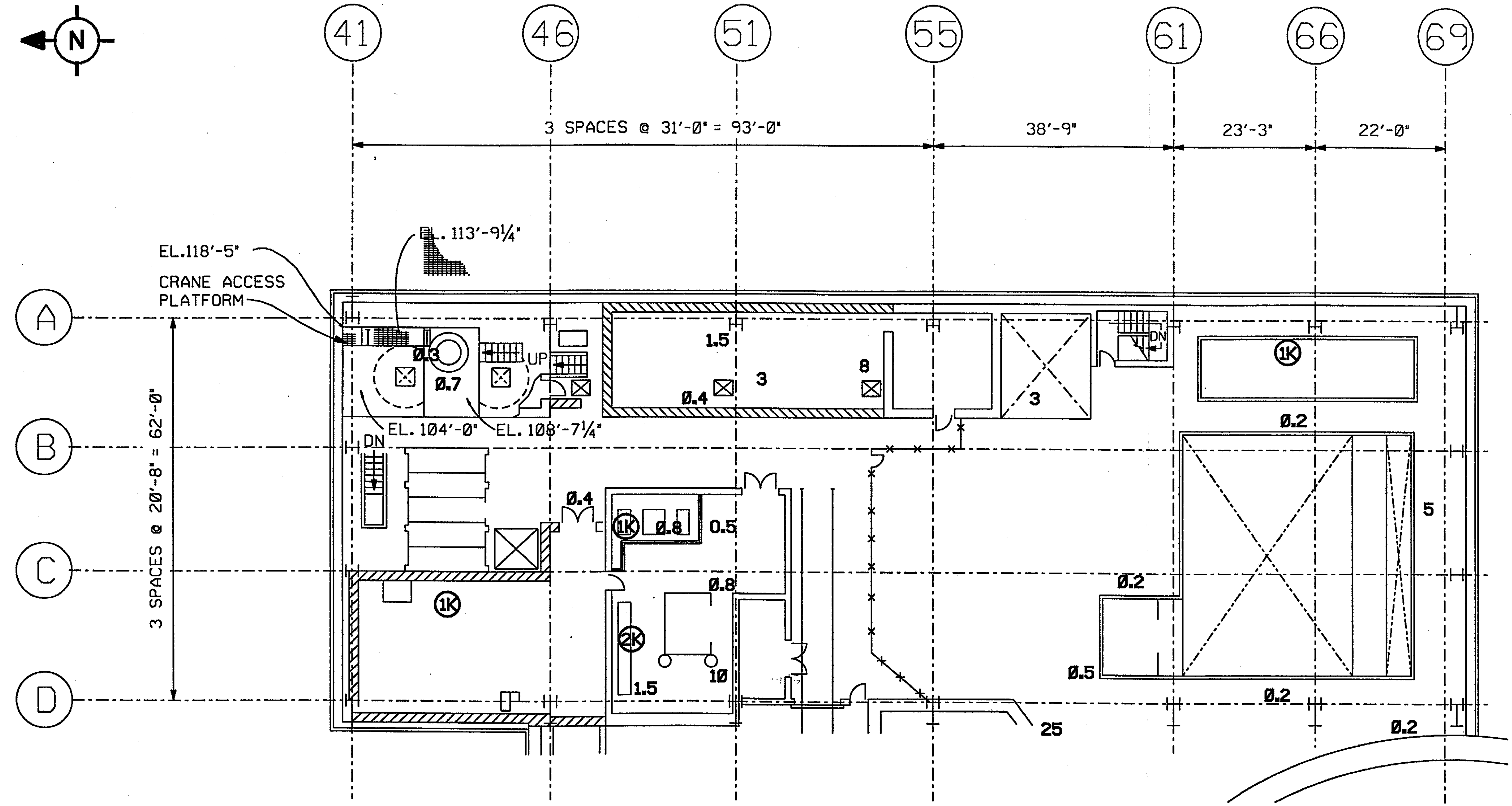


NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm$^2$ UNLESS OTHERWISE NOTED.

*Trojan Nuclear Plant*
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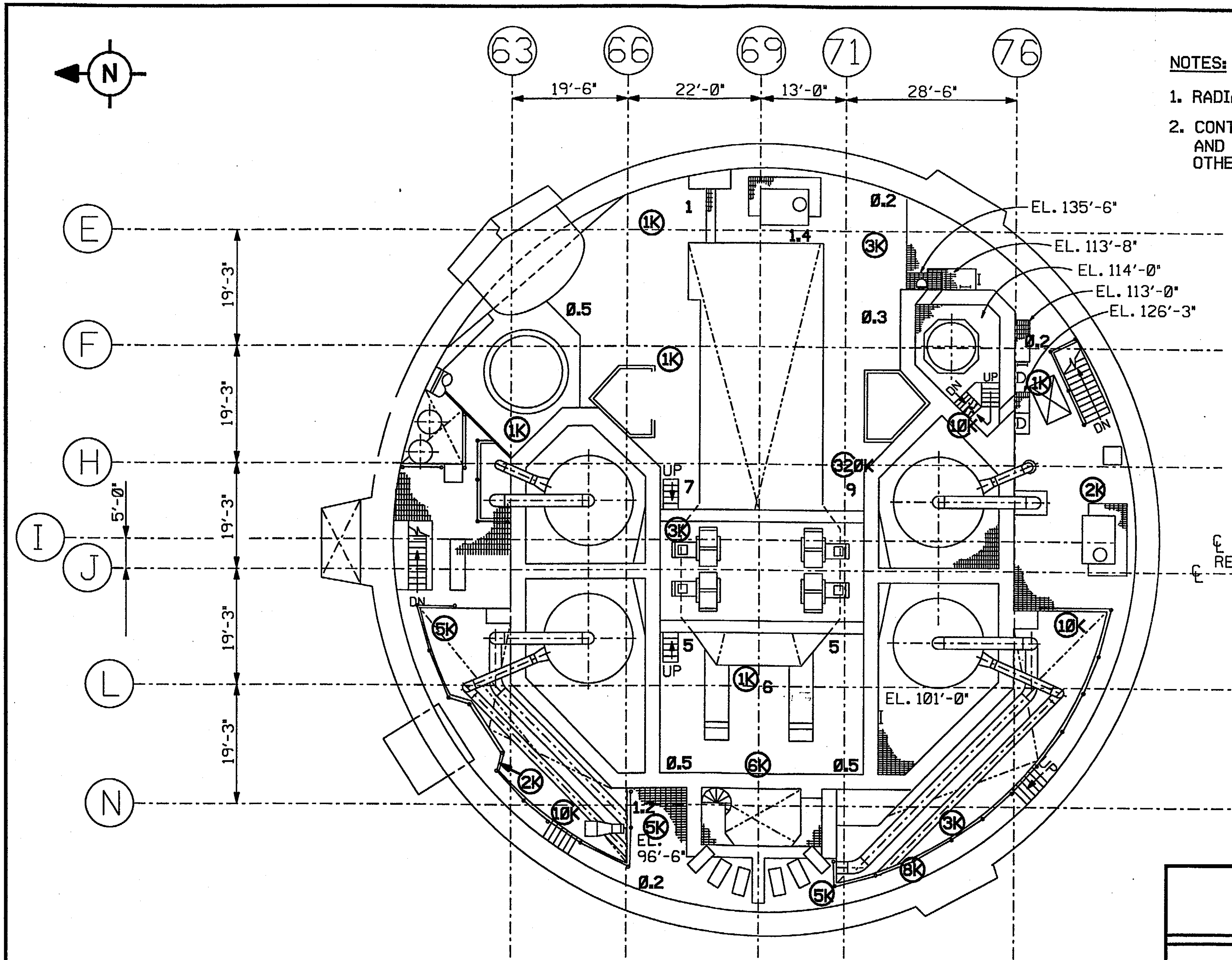
Figure 3-35
Radiological Survey Data
Fuel Building Elevation 93 FT




NOTES:

1. RADIATION LEVELS ARE IN mR/hr.
2. CONTAMINATION LEVELS ARE CIRCLED AND ARE dpm/100cm² UNLESS OTHERWISE NOTED.

*Trojan Nuclear Plant*
DECOMMISSIONING PLAN AND LICENSE TERMINATION PLAN

Figure 3-36
Radiological Survey Data
Containment Elevation 93 FT

Figure 3-37

**RADIATION PROTECTION ORGANIZATION**