YANKEE NUCLEAR POWER STATION

DECOMMISSIONING PLAN

REVISION: 0.0

Prepared by:

Yankee Atomic Electric Company

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ABREVIATIONS, TERMS AND UNITS

AC	Alternating Current
ALARA	As Low As is Reasonably Achievable
ANSI	American National Standards Institute
AOR	Abnormal Occurance Report
ASME	American Society of Mechanical Engineers
BOP	Balance of Plant
CFH	Certified Fuel Handler
CFR	Code of Federal Regulations
Ci	Curie; unit of radioactivity = 3.7E10 disintegrations per second
CRP	Component Removal Project
DC	Direct Current
DECON	Immediate Decontamination and Dismantlement Option
DOC	Decommissioning Operations Contractor
DOE	Department of Energy
DPM	Disintegrations per Minute
EDM	Electrical Discharge Machining
ENTOMB	Encasement in Concrete with Future Dismantlement Option
EPA	Environmental Protection Agency
FERC	Federal Energy Regulatory Commission
FSAR	Final Safety Analysis Report
GET	General Employee Training
GM	Geiger-Mueller
HEPA	High Efficiency Particulate Air (filter)
ICRP	International Council on Radiation Protection and Measurement
INPO	Institute of Nuclear Power Operators
kV	Kilovolt
LLD	Lower Limit of Detection
LLW	Low Level Waste
LPG	Liquid Propane Gas
M&TE	Measuring and Test Equipment
μCi	0.000001 Ci
MCS	Main Coolant System
MDA	Minimum Detectable Activity
MDC	Minimum Detectable Concentration
MDM	Metal Disintigration Machining
MPC	Maximum Permissible Concentration
mR	0.001 R
μR	0.000001 R

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ABREVIATIONS, TERMS AND UNITS

MWe	Megawatts Electric
MWt	Megawatts Thermal
NCRP	National Council on Radiation Protection and Measurement
NIST	National Institute of Standards and Technology
NPDES	National Pollutant Discharge Elimination System
NRC	Nuclear Regulatory Commission
NSARC	Nuclear Safety Audit and Review Committee
NSSS	Nuclear Steam Supply System
NST	Neutron Shield Tank
ODCM	Off-Site Dose Calculation Manual
OSHA	Occupational Safety and Health Administration
PAG	Protective Action Guides
PAB	Primary Auxiliary Building
PASS	Post Accident Sample System
PCA	Potentially Contaminated Area
PCB	Polychlorinated Biphenyls
pCi	0.000000000001 Ci
Person-rem	Collective Radiation Dose to a Population
PIC	Pressurized Ion Chamber
PIR	Plant Information Report
POL	Possession Only License
PORC	Plant Operation Review Committee
R	Roentgen; Unit of radiation exposure
RCA	Radiation Control Area
RCRA	Resource Conservation and Recovery Act
Rem	Unit of dose equivalent
REMP	Radiological Environmental Monitoring Program
RETS	Radiological Effluent Technical Specifications
RWP	Radiation Work Permit
SAFSTOR	Delayed Decontamination and Dismantlement Option
SFP	Spent Fuel Pit
TEDE	Total Effective Dose Equivalent
TLD	Thermoluminescent Dosimeter
TLG	TLG Engineering, Inc.
TS	Technical Specifications
VC	Vapor Container
WD	Waste Disposal
YAEC	Yankee Atomic Electric Company
YAEL	Yankee Atomic Environmental Lab

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ABREVIATIONS, TERMS AND UNITS

YNPS	Yankee Nuclear Power Station
YNSD	Yankee Nuclear Services Division

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SECTION 1
SUMMARY OF PLAN

1.1 DESCRIPTION OF DECOMMISSIONING PLAN AND DECOMMISSIONING ALTERNATIVE

1.1.1 Introduction

By letter dated February 27, 1992 (Reference 1.1-1), Yankee Atomic Electric Company (Yankee, YAEC) notified the Nuclear Regulatory Commission (NRC) of the company's decision to permanently cease power operations at the Yankee Nuclear Power Station (YNPS). Following notification of the NRC, Yankee initiated decommissioning planning and other plant closure activities. The objective of the YNPS decommissioning is to safely reduce radioactivity at the site to residual levels, allowing release of the site for unrestricted access.

On August 5, 1992, the NRC amended the YNPS Facility Operating License (DPR-3) to a possession only status. This, combined with other amendments and program changes, described in Section 1.2, forms the basis of this decommissioning plan. The plan is submitted by YAEC in accordance with the requirements of 10 CFR 50.82(a), which requires submittal of a proposed decommissioning plan within two years of the permanent cessation of operations.

Yankee proposes to defer dismantlement of YNPS until a low level radioactive waste disposal facility is available. The plant will be maintained in a safe storage condition following approval of the decommissioning plan. The safe storage period will extend until such time that a firm contract for disposal of low level radioactive waste generated during decommissioning activities can be secured. At that time, final dismantlement of the facility will commence. For the purposes of this decommissioning plan, it is assumed that the safe storage period will extend until the year 2000. Limited access to low level radioactive waste disposal facilities may also occur during the safe storage period. Yankee intends to use these opportunities to remove components and structures consistent with the guidance presented in Section 1.4.

Plant closure activities, which were commenced following the decision to cease power operations, will continue in accordance with applicable regulatory requirements and Yankee's commitment to maintain the facility in a safe and economical manner. These activities have included decontamination, disposal of radioactive components, lay-up of plant equipment, and facility modifications to improve plant operations.

Yankee, with the concurrence of the NRC, has initiated a project to remove and dispose of the steam generators, pressurizer, and reactor vessel internals before July 1994. These components will be removed from the site with the exception of certain high level radioactive material from the reactor vessel internals which will be stored with the fuel. Appendix A presents a description of the Component Removal Project. The decommissioning plan incorporates the plant configuration changes resulting from the implementation of the Component Removal Project.

A total of 533 fuel assemblies are stored in the YNPS Spent Fuel Pit. Yankee is currently seeking accelerated acceptance of the YNPS spent fuel by the Department of Energy in accordance with the current fuel disposal contract. However, it is unlikely that the Department of Energy will accept all YNPS spent fuel before the beginning of the dismantlement period in 2000. Although the Department of Energy may start taking fuel as early as 1998, for planning purposes, fuel shipments are assumed to be completed in 2018.

In recognition of the ultimate goal of decommissioning, this plan presents information on plant dismantlement activities in addition to safe storage period implementation activities. Where appropriate the plan presents radiological information corresponding to the effect of completing all activities in early 1994. Although this is not realistic based on schedule and low level radioactive waste facility constraints, it provides a bounding estimate of the radiological impact of earlier execution of decommissioning activities.

1.1.2 Background

YNPS achieved initial criticality in 1960 and began commercial operations in 1961. The Nuclear Steam Supply System is a four loop pressurized water reactor designed by Westinghouse Electric Corporation. The original thermal power design limit of 485 MWt was upgraded to 600 MWt in 1963. The Turbine Generator, also designed by Westinghouse, is rated to produce 185 MWe. Commercial operation ceased in 1992 after about 31 years of operation. During its operation, YNPS achieved an average capacity factor of about 74%.

YNPS shut down on October 1, 1991, in response to regulatory uncertainties associated with the integrity of the Reactor Vessel. During the outage and before February 26, 1992, all fuel assemblies, control rods, and neutron sources were removed from the Reactor Vessel and stored in the Spent Fuel Pit. A total of 533 fuel assemblies are stored in the Spent Fuel Pit. Plant systems required to support spent fuel storage and to support permanently defueled operations are in service.

On February 26, 1992, the YAEC Board of Directors decided to cease power operations permanently at YNPS. This decision was based on the following factors:

- Economic analyses indicated that shutdown of the plant before expiration of its NRC operating license in July, 2000 could produce a substantial savings to the electricity purchasers.

- Significant regulatory uncertainty associated with the timing of the completion of the NRC's review of the integrity of the YNPS Reactor Vessel.

In addition to decommissioning planning, Yankee has completed several plant closure activities, including site security modifications, control rod disposal, decontamination, and miscellaneous facility improvement projects. In 1993, YNPS initiated a project to remove the steam generators, the pressurizer, and the Reactor Vessel internal components. These components will be removed from the site with the exception of certain high level radioactive material from the Reactor Vessel internals which will be stored with the fuel. The project will be completed before July 1994.

1.1.3 Contents of the Decommissioning Plan

The decommissioning plan has been prepared in accordance with the requirements of 10 CFR 50.82(b) using guidance presented in Draft Regulatory Guide DG-1005 (Reference 1.1-2). The following is a brief description of the sections presented in the decommissioning plan:

- Section 1: Summary of Plan - This section presents a summary of the information presented in the decommissioning plan. The section also presents a description of the process used to administer and implement the decommissioning plan.

- Section 2: Choice of Decommissioning Alternatives and Description of Activities - This section presents detailed information regarding the safe storage and dismantlement phases of decommissioning. This section includes a description of the facility, the decommissioning process, and the decommissioning organization.

- Section 3: Protection of Occupational and Public Health and Safety - This section presents a description of the radiological status of YNPS, the Radiation Protection Program, the Radioactive Waste Management Program, and the Occupational Safety Program. This section also presents an accident analysis that bounds events that could occur during decommissioning.

- Section 4: Final Radiation Survey Plan - This section presents the site release criteria and a description of the objectives, criteria, and methodologies that will be used to develop a final radiation survey plan.

- Section 5: Decommissioning Cost Estimate And Funding Plan - This section presents the decommissioning cost estimate and funding plan.

- Section 6: Decommissioning Technical Specifications - This section presents a description of the technical specifications that will be in effect during decommissioning.

- Section 7: Decommissioning Quality Assurance Plan - This section presents a description of the quality assurance plan that will be in effect during decommissioning.

- Section 8: Decommissioning Security Plan - This section describes access control and other security plan requirements that will be in effect during decommissioning.

- Section 9: Fire Protection - This section presents the fire protection program that will be in effect during decommissioning.

REFERENCES

1.1-1 BYR 92-024, Permanent Cessation of Power Operations at the Yankee Nuclear Power Station, A.C. Kadak to T.E. Murley (USNRC), February 27, 1992.

1.1-2 Draft Regulatory Guide DG-1005, "Standard Format and Content For Decommissioning Plans for Nuclear Reactors."

1.2 MAJOR TASKS, SCHEDULES AND ACTIVITIES

1.2.1 Description of Major Activities

YNPS decommissioning is comprised of four periods: activities prior to decommissioning plan approval, safe storage, dismantlement, and site restoration. This section presents a description of each phase.

1.2.1.1 Activities Prior to Decommissioning Plan Approval

Following the decision to permanently cease power operations at YNPS, Yankee began activities associated with plant closure.

In March 1992, YAEC proposed to the NRC a change to modify the plant full power operating license to a possession only license status. The change removed YNPS' authority to load fuel into the reactor and to operate the reactor. A possession only license amendment was issued by the NRC in August 1992 (Reference 1.2-1).

YAEC reviewed the plant licensing basis to evaluate the applicability of existing technical specifications and NRC regulations to a permanently defueled condition. The technical specifications which were applicable only to an operating mode that corresponded to fuel in the reactor were determined to be not applicable to the permanently defueled condition. Specific technical specification changes were proposed based on the evaluation. Table 1.2-1 summarizes the technical specification amendments proposed by YAEC, including their approval status. Section 6 summarizes the technical specifications that will be in effect during decommissioning.

Other NRC regulations and programs were reviewed to assess their applicability to the permanently defueled condition. This process included detailed engineering and licensing studies unique to the permanently defueled condition. Where appropriate, relief was requested from regulations determined to be no longer appropriate or applicable. Table 1.2-1 summarizes the regulatory relief requests, including their approval status.

Plant systems, structures, and components were evaluated to identify those needed to support plant operations (e.g., Spent Fuel Pit cooling). The systems, structures, and components not required to support plant operation were further evaluated for inclusion in the plant lay-up program based on their potential future use during plant decommissioning. A comprehensive plant lay-up program was developed and implemented to support decommissioning. Plant procedures were modified as appropriate to reflect the new operating requirements.

An engineering task force was created to develop a plan for the disposition of spent fuel stored on-site. The likelihood of storing fuel on-site during and after plant decommissioning significantly impacts both the decommissioning process and the decommissioning cost. Detailed engineering evaluations were prepared to investigate spent fuel disposition alternatives (Reference 1.2-2).

Based on the task force review, the following spent fuel management strategy was implemented:

- Continue operation of the Spent Fuel Pit and implement any economically attractive improvements
- Urge the Department of Energy to accelerate acceptance of spent fuel or to accept financial responsibility for on-site spent fuel storage
- Continue evaluations of wet and dry storage options to reflect Yankee and industry developments
- Initiate preliminary design of a dry storage facility

Yankee is currently seeking accelerated acceptance of YNPS' spent fuel by the Department of Energy in accordance with the current fuel disposal contract. The Department of Energy's current position is that they have not yet determined whether priority will be accorded shutdown reactors, or if priority is granted, under what specific circumstances it might be granted. A rulemaking is scheduled that will include the issue of acceptance priority for shutdown plants. YAEC will participate in the rulemaking process. However, it appears that priority for shutdown plants may not be supported by the majority of participants in this process.

It is unlikely that the Department of Energy will accept all YNPS spent fuel before the beginning of the dismantlement period in 2000. Although the Department of Energy may start taking fuel as early as 1998, for planning purposes, fuel shipments are assumed to be completed in 2018. This projection is based on the Department of Energy's Acceptance Priority Ranking, Annual Capacity Report, and an extrapolation beyond the 10-year Department of Energy outlook. For planning purposes, YAEC's current decommissioning cost estimate assumes storage of fuel in the Spent Fuel Pit until 1996, at which time it will be transferred to an on-site dry storage facility. Spent fuel is projected to remain in the dry storage facility until 2018.

YAEC evaluated the option to store fuel in the Spent Fuel Pit during a portion of the dismantlement phase of decommissioning. The purpose of the evaluation was to identify safety considerations and limitations on decommissioning activities associated with operating the Spent Fuel Pit concurrent with dismantlement activities. Operation of the Spent Fuel Pit during the dismantlement phase would allow YAEC additional time to pursue early transfer of spent fuel to the Department of Energy without incurring a significant investment associated with a dry cask facility. This option also allows additional time for the development of a multi-purpose canister system that is compatible with YNPS limitations. Fuel storage flexibility has been incorporated in the dismantlement plan and accident analyses.

Yankee has also initiated a project to remove and dispose of the steam generators, pressurizer, and Reactor Vessel internals before June 1994. These components will be removed from the site with the exception of certain high level radioactive material from the Reactor Vessel internals which will be stored with the fuel. Appendix A presents a description of the Component Removal Project.

1.2.1.2 Safe Storage Period

The safe storage period includes activities associated with establishing and maintaining the facility in a safe condition following approval of the decommissioning plan. The goal of the safe storage period is to prevent inadvertent exposure to radiation and to prevent spread of contamination. Most of the activities required to establish the safe storage condition will be completed before decommissioning plan approval.

During the safe storage period, plant operations will include those needed to monitor the radiological status of the facility, to maintain systems in a dormant condition, and to support operation of the spent fuel storage facility. The Radiation Protection Program implemented during the safe storage period is intended to support routine clean-up and decontamination operations. The radiation protection staff will be augmented with operating and security personnel who will perform full-time on-site monitoring. The objective of the monitoring is to control access to the facility, preventing inadvertent exposure to radiation or the spread of contamination.

Yankee will maintain the systems and components required to support decommissioning and spent fuel storage in accordance with the possession only license and other administrative and implementing procedures. The maintenance program in effect during the safe storage period consists of corrective maintenance, preventive maintenance, and surveillances. Yankee will continue preventative maintenance practices for systems and components required by technical specifications. Preventative maintenance will continue for other systems at a level commensurate with the functional requirements described in Section 2.2.

Yankee will continue to seek potential low level radioactive waste disposal sites for YNPS decommissioning waste during the safe storage period. If a site is identified that can support the decommissioning waste, Yankee will proceed with earlier dismantlement. Limited access to low level radioactive waste disposal facilities may also occur. Yankee intends to use these opportunities to remove components and structures consistent with the guidance presented in Section 1.4.

1.2.1.3 Dismantlement Period

About 12 to 18 months before the start of decontamination and dismantlement activities, the decommissioning administrative and engineering organization will be mobilized. During the first several months the following activities will occur:

- Initiation of detailed project planning
- Preparation of engineering specifications and procedures
- Procurement of special equipment needed to support decommissioning
- Negotiation of service contracts required for decommissioning activities
- Reactivation and return to service of systems required for decommissioning

The engineering and preparation phase is followed by plant dismantlement activities. The contaminated systems will be removed, packaged, and either shipped to an off-site processing facility or shipped directly to a low level radioactive waste disposal facility. Decontamination of plant structures will be completed concurrently with the equipment and system removal process. Structure decontamination will include a variety of techniques ranging from high pressure water washing to selective removal of concrete to allow release of the structures. Contaminated structural material will be packaged and either shipped to a processing facility for decontamination or shipped directly to a low level radioactive waste disposal facility.

Following the removal of contaminated systems, structures, and components, a comprehensive final radiation survey will be conducted. The survey will verify that radioactivity has been reduced to sufficiently low levels allowing unrestricted release of the site. Successful completion of the final survey will be demonstrated through a verification survey completed by an independent contractor selected by the NRC.

1.2.1.4 Site Restoration

Site restoration activities will be completed following termination of the YNPS possession only license by the NRC. Some of these activities may be completed during the safe storage and dismantlement periods. Activities associated with the Vapor

Container will include removal of internal structures, disassembly of the Vapor Container shell, and demolition of the Reactor Support Structure and other concrete structures. Instrumentation will be removed from the Control Room and other remote control areas after the instrumentation is not needed to support plant activities. The remaining plant structures will be demolished. All building foundations will be back filled with concrete rubble and structural fill. Site areas will be graded and landscaped as necessary.

1.2.2 Final Release Criteria

The ultimate goal of decommissioning is to release the site for unrestricted use. This requires assurance that future uses of the facility will not result in exposing individuals to unacceptable levels of radiation. The release criteria are presented in Section 4.1.

1.2.3 Schedule for Decommissioning Activities

A detailed schedule for decommissioning activities is presented in Section 2.3.6. The decommissioning schedule assumes for planning purposes that a low level radioactive waste facility will be available for YNPS decommissioning wastes in 2000 and that fuel will be transferred to a dry cask storage facility in 1996. Based on these assumptions, the following is a summary of the YNPS decommissioning project schedule:

- Detailed site radiological characterization of the plant systems, structures, components, soil, and groundwater will be completed to support dismantlement activities.

- NRC approval of the decommissioning plan is expected before January 1, 1995. Following approval, the facility will be maintained in a safe storage condition until 2000.

- A dry cask spent fuel storage facility will be constructed in 1995 and loaded with fuel in 1996. Spent fuel will be transferred to the Department of Energy beginning in 1998. The final spent fuel assembly will be removed from the site in 2018.

- Detailed engineering and planning for plant decontamination and dismantlement activities will begin in 1999. Actual decontamination and dismantlement activities are scheduled to begin in July 2000 and continue through January 2002. Site restoration will be completed by December 2002.

REFERENCES

1.2-1 NYR 92-148, Issuance of Amendment No. 142 to Facility License DPR-3 - Yankee Nuclear Power Station (TAC No. M83024), M.B. Fairtile (USNRC) to J.M. Grant, August 8, 1992.

1.2-2 YRP 303/93, Impact of Wet Spent Fuel Storage on Decommissioning, P.A. Rainey/K.J. Heider to R.A. Mellor, September 10, 1993.

TABLE 1.2-1

REGULATORY RELIEF ACTIVITIES
AS OF DECEMBER 1, 1993

Submittal	Approval Date	Reference
Possession Only License	8/8/92	NYR 92-148, Issuance of Amendment No. 142 to Facility License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M83024).
Containment Testing Exemption	6/15/92	NYR 92-116, Exemption from 10 CFR Part 50 - Appendix J Leak Rate Testing at the Yankee Nuclear Power Station (TAC No. M83341).
Minimum Shift Staffing	7/22/92	NYR 92-143, Issuance of Amendment No. 141 to Facility Operating License No. DRP-3 - Yankee Nuclear Power Station (TAC No. M83383).
Certified Fuel Handler Certification Program	6/16/92	NYR 92-122, NRC Approval of Certified Fuel Handler Program and Termination of Operator Licensed (TAC No. M83384).
Emergency Planning Exercise Exemption	7/24/92	NYR 92-144, Exemption from 10 CFR 50 Appendix E Emergency Preparedness Training Exercise at Yankee Nuclear Power Station (TAC No. M83415).
Fire Protection TS Transfer	8/20/92	NYR 92-156, Issuance of Amendment 144 to Possession Only License for Yankee Nuclear Power Station (TAC No. M83746).
FSAR Annual Update Exemption	7/23/92	NYR 92-142, Exemption from 10 CFR 50.71(e) Periodic Submittal of an Updated FSAR (TAC No. M83789).
Defueled Emergency Plan (Exemption Included)	10/30/92	NYR 92-178, Exemption from 10 CFR 50.54(q) and Approval of the Defueled Emergency Plan for Yankee Nuclear Power Station (TAC No. M83941).
PORC Activities TS Change	9/4/92	NYR 92-182, Issuance of Amendment No. 145 to Possession Only License for Yankee Nuclear Power Station (TAC No. M84005).
Defueled Security Plan (Exemption Included)	11/24/92	NYR 92-194, Exemption from Certain Requirements of 10 CFR 73.55 for the Yankee Nuclear Power Station (TAC No. M84267).

TABLE 1.2-1 (CONTINUED)

Submittal	Approval Date	Reference
Chemistry TS Change	2/19/93	NYR 93-095, Issuance of Amendment No. 147 to Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. 84519).
RETS/REMP TS Transfer	11/5/92	NYR 92-183, Issuance of Amendment No. 146 to Possession Only License for Yankee Nuclear Power Station (TAC No. M84372).
Permanently Defueled TS	6/11/93	NYR 93-062, Issuance of Amendment No. 148 to Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M85244).
Property Insurance Exemption	4/8/93	NYR 93-037, Exemption from the Requirements of 10 CFR 50.54(w) for the Yankee Nuclear Power Station (TAC No. M84579).
Liability Insurance Exclusion Letter (Part 140)	Pending	
Annual Fees Exclusion Letter (171.15)	8/14/92	BYR 92-083, Annual Fee Requirements of 10 CFR Part 171.

1.3 DECOMMISSIONING COST ESTIMATE AND AVAILABILITY OF FUNDS

1.3.1 Decommissioning Cost

In May 1992, YAEC and TLG Engineering, Inc. prepared an updated cost estimate for the YNPS decommissioning (Reference 1.3-1). Unlike previous YNPS decommissioning cost studies, the updated study was site specific, based on an analysis of all plant systems, structures, and components. In addition, specific consideration was given to unique YNPS features (e.g., plant size) in developing the decommissioning organization.

The study evaluated the cost associated with both immediate dismantlement (DECON case) and deferred dismantlement (SAFSTOR case) alternatives. As discussed in Section 1.1, YAEC proposes to defer dismantlement until after a contract for the disposal of low level radioactive waste is secured. This plan assumes that a low level radioactive waste disposal site will not be available until 2000.

The following is a summary of the costs, presented on a January 1992 cost basis, associated with the various phases of decommissioning, including the effect of the Component Removal Project:

Component Removal Project (CRP)	$ 26.66 million
Safe Storage Period	$ 32.66 million
Low Level Radioactive Waste Disposal (non-CRP)	$ 43.78 million
Spent Fuel Storage	$ 56.49 million
Dismantlement and Site Restoration	$ 83.66 million
Total Cost	$ 243.25 million

The revised cost estimate reflects a significant credit for removing the steam generators, pressurizer, and reactor vessel internals in 1993 and 1994. The total cost is about $4 million less than the cost estimate for the deferred dismantlement alternative, $247 million.

1.3.2 Funding Plan

In May 1981, FERC authorized a decommissioning fund accumulation from customers over a ten year period ending in 1991. Since that time, YAEC has filed periodic collection updates based on revised cost estimates and collection periods. Through June 30, 1993, YAEC accumulated $99.6 million in cash and anticipated tax credits for decommissioning. The decommissioning collections are made through YAEC's Power Contracts and are deposited in an independent and irrevocable trust at a commercial bank, with the principle and interest to be used to discharge future decommissioning obligations as incurred. YAEC has provided a copy of the trust document to the NRC (Reference 1.3-2). This trust is executed in compliance with 10 CFR 50.75(e)(1)(ii).

On June 1, 1992, YAEC submitted a rate filing with FERC, in part, to increase decommissioning collections to fund the balance of the revised cost estimate. The increased collections would be recovered through June 2000 utilizing YAEC's existing Power Contracts. In March 1993, FERC approved a settlement of $235 million with respect to all parties, with the exception of the town of Norwood, Massachusetts, a customer of one of YAEC's sponsors which is obligated for 0.413% of YAEC's charges. A separate proceeding regarding Norwood is now pending.

The settlement of $235 million is lower than the deferred dismantlement cost estimate (SAFSTOR case) of $247 million submitted to FERC. The lower value was chosen for the purposes of the settlement and does not establish any principle or precedent for future proceedings regarding the cost or timing of a collection for YNPS decommissioning. The settlement also requires YAEC to file a revised decommissioning funding schedule, supported by a revised cost study, within 90 days of NRC approval of the decommissioning plan. The revised cost estimate will include an updated cost and collection schedule consistent with the NRC approved decommissioning plan.

REFERENCES

1.3-1 Decommissioning Cost Estimate for the Yankee Nuclear Power Station, TLG Engineering Inc., May 1992.

1.3-2 BYR 90-102, Decommissioning Funding Assurance Report and Certification, H.T. Tracy to Document Control Desk (NRC), July 25, 1990.

1.4 REGULATORY BASIS FOR ADMINISTRATION OF THE DECOMMISSIONING PLAN

The decommissioning plan has been prepared and submitted in response to the requirements of 10 CFR 50.82, "Application for Termination of License," using guidance from Draft Regulatory Guide DG-1005 (Reference 1.4-1). The decommissioning plan is intended to provide the framework for the decommissioning process by which the YNPS site will be returned to an unrestricted use condition. The plan will be maintained current through controlled revision, as described below.

The following documents will constitute the decommissioning licensing basis of YNPS following NRC approval of this plan:

- NRC-approved Decommissioning Plan
- NRC-approved Possession Only License for YNPS
- Technical Specifications (Section 6).

Following NRC approval, the decommissioning plan will supersede and replace the YNPS Final Safety Analysis Report (FSAR). The FSAR will be retained as a historical document. All operational descriptions and commitments described in the FSAR will be superseded by the approved decommissioning plan. Essential safety features and functions which will be relied upon during decommissioning are described in this plan.

For the purposes of YNPS decommissioning, the following provisions, similar to those in 10 CFR 50.59, will be used to control changes, tests, and experiments:

- YNPS may (i) make changes in the facility or procedures as described in the decommissioning plan and (ii) conduct tests or experiments not described in the decommissioning plan, without prior NRC approval, if the proposed changes, tests, or experiments do not involve a change to the technical specifications or do not result in an unreviewed safety question.
- A proposed change, test, or experiment shall be deemed to involve an unreviewed safety question (i) if the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the decommissioning plan may be increased, or (ii) if the possibility for an accident or malfunction of a different type than evaluated previously in the decommissioning plan may be created, or (iii) if the margin of safety as defined in the basis for a technical specification is reduced.

- Records will be maintained of changes to the facility and of changes to procedures to the extent that the changes constitute changes to the facility or to the procedures as described in the decommissioning plan. Records will also be maintained of tests and experiments as described above. The records will include a written safety evaluation which provides the basis for the determination that the changes, tests, or experiments do not involve an unreviewed safety question.

- An annual report will be submitted, as specified in 10 CFR 50.4, containing a brief description of any changes, tests, or experiments. The report will include a summary of the safety evaluation of each change.

- Records of changes to the facility will be maintained until the termination of the license. Records of changes to procedures, tests, and experiments will be maintained for five years.

Descriptions of the following plans are included in the decommissioning plan:

• Final Radiation Survey Plan (Section 4)	10 CFR 50.82(b)(3)
• Decommissioning Funding Plan (Section 5)	10 CFR 50.82(b)(4) 10 CFR 50.82(b)(5)
• Quality Assurance Plan (Section 7)	10 CFR 50.54(a) 10 CFR 50 Appendix A

REFERENCES

1.4-1 Draft Regulatory Guide DG-1005, "Standard Format and Content For Decommissioning Plans for Nuclear Reactors".

1.5 ADMINISTRATIVE CONTROLS DURING THE TRANSITION PERIOD PRIOR TO APPROVAL OF THE DECOMMISSIONING PLAN

Administrative controls have been implemented to control activities prior to approval of the decommissioning plan. The purpose of the controls is to ensure that YNPS is operated within the existing licensee authority. The requirements and controls are implemented by plant procedures.

The letter transmitting the possession only license (Reference 1.5-1) provided guidance to YNPS regarding those activities which could proceed prior to NRC approval of a decommissioning plan. This guidance was clarified in a subsequent letter to YNPS (Reference 1.5-2). The guidance specifically states that YNPS may undertake activities before approval of the decommissioning plan that do not:

- Foreclose the release of the site for possible unrestricted use
- Significantly increase decommissioning costs
- Cause any significant environmental impact not previously evaluated
- Violate the terms of the existing license or 10 CFR 50.59

Activities performed prior to approval of the decommissioning plan will be performed within existing licensee authority, subject to the criteria and clarification presented above. Any proposed activities that are not within existing licensee authority, subject to the criteria and clarifications above, will be submitted for NRC approval prior to their implementation.

REFERENCES

1.5-1 NYR 92-148, Issuance of Amendment No. 142 to Facility License DPR-3 - Yankee Nuclear Power Station (TAC No. M83024), M.B. Fairtile (USNRC) to J.M. Grant, August 8, 1992.

1.5-2 NYR 93-028, YNPS Activities Prior to Decommissioning Plan Approval, T.E. Murley (USNRC) to A.C. Kadak, March 29, 1993.

SECTION 2
CHOICE OF DECOMMISSIONING ALTERNATIVE AND DESCRIPTION OF ACTIVITIES

2.1 ALTERNATIVE SELECTION AND JUSTIFICATION

2.1.1 Decommissioning Alternative

Yankee has evaluated the three decommissioning alternatives described in the generic environmental impact statement on decommissioning of nuclear facilities, NUREG-0586 (Reference 2.1-1): DECON, SAFSTOR, and ENTOMB. The most appropriate alternative for decommissioning YNPS is to defer dismantlement until a low level radioactive waste disposal facility is available to receive low level radioactive waste from decommissioning. A disposal facility is not expected to be available to YNPS until 2000. This results in a decommissioning duration in excess of the approximately six year period associated with the DECON alternative in the generic environmental impact statement. Therefore, the YNPS decommissioning alternative is most similar to the SAFSTOR alternative with a relatively short period before final dismantlement and license termination.

2.1.2 Decommissioning Alternative Justification

Access to a low level radioactive waste disposal facility is unlikely for the period immediately after NRC approval of the decommissioning plan. Significant dismantlement activities cannot proceed without sustained access to a low level radioactive waste disposal facility. Therefore, the plant must be placed in a safe storage condition until a facility becomes available. Limited dismantlement may be possible during the safe storage period using a combination of off-site reclamation facilities, on-site storage, and limited access to waste disposal facilities.

Under the provisions of the Low Level Radioactive Waste Policy Amendments Act of 1985, existing disposal sites are permitted to deny access to waste generated in states that have neither joined a compact with an established disposal site nor established their own sites by January 1, 1993. Existing sites have begun to exercise their right to exclude low level radioactive waste disposal sites from non-compliant states. The Commonwealth of Massachusetts has neither joined a regional compact nor sited its own low level radioactive waste disposal facility. Although Massachusetts continues to develop plans, progress to date indicates that a facility in Massachusetts will not be available to accept YNPS decommissioning waste in June 1994 when the Barnwell, South Carolina Waste Management Facility closes.

YNPS was excluded from all disposal sites after December 1992 with the exception of the Barnwell, South Carolina Waste Management Facility. Barnwell is available to YNPS through June 1994. After that date, the disposal site at Barnwell is not available to YNPS. Although new low level radioactive waste disposal facilities are being planned by several regional compacts, each facility has encountered numerous delays and significant progress appears doubtful. None of the sites appear to be capable of operation by the mid 1990's. In addition, there is a high probability that regional compacts may specifically exclude out-of-compact waste from their facilities.

Massachusetts has not entered into a compact or developed an in-state low level radiological waste disposal facility. In 1993, the Governor filed a bond authorization bill that would allow the Commonwealth to expend funds to site a facility in-state or to use the monies to enter into a compact arrangement with other states. The Commonwealth's Low Level Radioactive Waste Management Board will decide in early 1994 whether to site a facility for its own use or to enter into a compact agreement where a facility would be shared with other states. Compact regulations and legislation have expressly or by implication rejected accepting decommissioning waste, particularly from out-of-compact generators. However, if Massachusetts starts the process in 1994 to either join a compact, form a new compact, or build a disposal facility for its own use, there is a likelihood that a disposal facility under these arrangements will be available in 2000 to accept the YNPS decommissioning waste.

Therefore, Yankee has concluded, for planning purposes, that the option of complete dismantlement beginning in 1995 is not viable. YNPS will be placed in a safe storage condition until the year 2000. Detailed planning and engineering for dismantlement will begin in 1999, with decontamination and dismantlement activities beginning in 2000 and completed by 2002. This proposed schedule is within the 60 year limit (after cessation of operation) in 10 CFR 50.82(b)(1).

Yankee will continue to seek potential low level radioactive waste disposal sites for YNPS decommissioning waste. If a site is identified that can support the decommissioning waste, Yankee will proceed with earlier dismantlement. Limited access to low level radioactive waste facilities may also become available during the safe storage period. Yankee intends to use these opportunities to remove components and structures consistent with the guidance presented in Section 1.4.

Yankee's choice of decommissioning alternative is consistent with NUREG-0586. In that document, the NRC concluded that DECON and SAFSTOR alternatives are reasonable for decommissioning a pressurized water reactor. Although there are advantages and disadvantages to either alternative, there is nothing that causes one alternative to be preferred from an environmental impact perspective. ENTOMB was determined to be less desirable than either DECON or SAFSTOR. The presence of long lived radioisotopes in nuclear power plants would require implementation of the ENTOMB alternative for a period significantly longer than the 60 year limit.

Earlier YNPS decommissioning studies assumed that DECON was the preferred decommissioning alternative. If a low level radioactive waste disposal site were available, DECON would be the preferred alternative for several reasons:

- The site is remediated as soon as possible after cessation of power operations, allowing unrestricted use of the site.
- Experienced plant personnel are more likely to be available, facilitating incorporation of detailed operating experience into the decommissioning staff.
- Decommissioning cost is lower, minimizing the cost to the consumer.
- Financial exposure from escalation of low level radioactive waste disposal cost and other decommissioning costs is minimized.

The incentives above form the basis for Yankee pursuing removal of components at the earliest possible time, consistent with our goal of decommissioning YNPS as safely and economically as possible. Yankee intends to take advantage of low level radioactive waste reclamation facilities, on-site storage, and limited access to disposal facilities to pursue timely removal of components, structures, and systems at YNPS. All activities will be reviewed within established guidelines presented in Section 1.4 to ensure that they meet all regulatory requirements.

REFERENCES

2.1-1 NUREG-0586, Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities, August 1988.

2.2 FACILITY DESCRIPTION

2.2.1 General Description

The Yankee Nuclear Power Station (YNPS) is located on the east bank of the Deerfield River in Rowe, Massachusetts. YNPS achieved initial criticality in 1960 and began commercial operations in 1961. The nuclear steam supply system is a four loop pressurized water reactor designed by Westinghouse Electric Corporation. The original thermal power design limit of 485 MWt was upgraded to 600 MWt in 1963. The turbine generator, also designed by Westinghouse, is rated to produce 185 MWe. Commercial operation ceased in 1992 after greater than 31 years of operation.

YNPS is located on a 2000 acre site which straddles the Deerfield River in the towns of Rowe and Monroe Bridge, Massachusetts. About 10 acres of the site have been developed for plant use. Drawing 9699-FY-6A presents a plot plan of the developed area.

This section presents descriptions of systems, components, and structures that are contaminated, potentially contaminated, or are needed to support decommissioning activities. A list of plant systems not included in this section is presented in Table 2.2-1. These systems have no interaction with the decommissioning of YNPS. All drawings referenced in this section are presented in Appendix C.

2.2.2 Reactor Vessel

The Reactor Vessel contained and supported the uranium-fueled core, directed pressurized water through the core, and facilitated the operation, control, and handling of the core components. The Reactor Vessel consists of an integral lower head, a removable upper head, and various internal plates and barrels (Figure 2.2-1). All of the fuel assemblies, control rods, shim rods, and neutron source vanes were removed from the Reactor Vessel and stored in the Spent Fuel Pit prior to plant closure. The control rods and shim rods were disposed of after plant closure.

During reactor operations, main coolant entered the Reactor Vessel through four inlet nozzles and flowed down through the annuli between the vessel wall, the thermal shield, and the core barrel to the lower core plenum. From the lower core plenum, the main coolant flowed up through the core and exited the Reactor Vessel through four outlet nozzles.

All Reactor Vessel internals, with the exception of the thermal shield, are hung from a support ledge near the top of the Reactor Vessel. The thermal shield was supported from lugs mounted on the Reactor Vessel inside wall. The internals were secured in place by the Reactor Vessel head which compressed the core hold down ring. The core and the guide tubes were positioned between two core support plates and two guide tube plates. Control rod drive mechanisms, mounted on the Reactor Vessel head, positioned control rods in the core.

2.2.2.1 Reactor Vessel Design

The Reactor Vessel is fabricated from carbon steel with internal stainless steel cladding. Table 2.2-2 presents material properties of the major sections of the Reactor Vessel. The outside of the Reactor Vessel is insulated with 3 to 4 inch thick calcium silicate blocks covered with either stainless steel cladding or stainless steel cloth. The calcium silicate contains asbestos.

The Reactor Vessel is supported by 28 support lugs. The lugs, with a welded, grooved pad, are seated on a continuous steel ring, which is also grooved. Support pins are located in the grooves to provide vertical support while allowing for radial movement from thermal effects. The continuous steel ring is an integral part of the reactor support ring, which is supported on a ledge formed on the inner wall of the primary shield. The inner wall of the primary shield supports both the Neutron Shield Tank and the Reactor Vessel. The primary shield wall transfers its load to the main support structure.

2.2.2.2 Reactor Vessel Internal Components On-site Storage

All Reactor Vessel internal components, with the exception of the core baffle and the lower core support plate, were disposed of as a part of the Component Removal Project (Appendix A). The core baffle and the lower core support plate were packaged in containers and stored in the Spent Fuel Pit. These components contained radionuclide concentrations in excess of 10 CFR Part 61 Class C limits and will be shipped to a high level radioactive waste disposal facility. The storage container geometry and packaged weight were designed to be similar to those of a YNPS fuel assembly (Reference 2.2-1). This maximized compatibility with the storage, shipment, and disposal systems used for YNPS spent nuclear fuel.

2.2.2.3 Decommissioning Configuration

The Reactor Vessel is not required to support plant operations during the safe storage period and will be maintained in lay-up until it is dismantled. During the safe storage period, the Reactor Vessel may either be completely drained or filled with water to below the nozzles. The Reactor Vessel closure head will be installed on the Reactor Vessel flange. Access to the Vapor Container will be controlled during the safe storage period to preclude inadvertent exposure to radiation from the Reactor Vessel.

2.2.3 Steam Generators

The Steam Generators used heat generated in the reactor core to produce dry, saturated steam which flowed through the turbine generator to produce electricity. The four Steam Generators were vertical shell, U-tube generators with an integral steam drum containing moisture separation and drying components. The Steam Generators were removed as a part of the Component Removal Project (Appendix A).

2.2.4 Main Coolant System

The Main Coolant System supported plant operations by providing the following major functions:

- Transferred heat generated in the reactor core to the Steam Generators to produce steam for the turbine generator.

- Circulated cooling water to the reactor to meet heat transfer and moderating requirements

- Provided a barrier between the fuel cladding and the Vapor Container

The system consists of four parallel piping loops connected to the Reactor Vessel. Each loop consists of a Steam Generator, two motor operated isolation valves, a canned-motor circulating pump, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Numbers 9699-FM-6A, 6B, 25A, 25B).

The Main Coolant Pumps are canned-motor, hermetically sealed, single stage, centrifugal pumps. Each pump unit consists of a hermetically sealed motor and a centrifugal pump impeller mounted on a single shaft as an integral unit complete with heat exchanger, volute, and motor terminals. The unit is mounted with the motor shaft vertical and above the pump. Both component cooling water and main coolant were used to remove heat from internal pump components.

Two motor operated isolation valves are installed to isolate each coolant loop from the Reactor Vessel. The valves are designed to open against a differential pressure of 200 psi and close against a differential pressure of 500 psi and are capable of withstanding about 3450 psi differential pressure in either direction when closed. The valves are not capable of providing a leak-tight seal under low differential pressure conditions. Leakage through the valve seat into the main coolant loop piping is removed by the Primary Plant Vent and Drain System.

The Steam Generators were removed during the Component Removal Project (Appendix A). Piping connections that were cut to support project implementation were sealed with welded caps.

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until it is dismantled. Although not anticipated at this time, sections of the Main Coolant System may be chemically decontaminated if lower activity levels are needed to reduce personnel exposure during dismantlement activities.

2.2.5 Pressure Control and Relief System

The Pressure Control and Relief System supported plant operations by providing the following major functions:

- Maintained the Main Coolant System pressure during steady-state operations
- Controlled Main Coolant System pressure changes during plant heat up, cool down, and load transients
- Prevented Main Coolant System pressure from exceeding the design pressure

The system consisted of a pressurizer vessel, immersion heaters, safety and relief valves, a spray system, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Numbers 9699-FM-6A, 6B). The Pressurizer was removed and the connecting piping was capped as a part of the Component Removal Project (Appendix A). Piping connections that were cut to support project implementation were sealed with welded caps.

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.6 Charging and Volume Control System

The Charging and Volume Control System supported Main Coolant System operations by providing the following major functions: inventory control (make-up and letdown), boric acid concentration control, chemical injection, and flow path between the Main Coolant System and the Purification System.

The system consists of three charging pumps, a low pressure surge tank with dedicated cooling system (heat exchanger and pump), two low pressure surge tank make-up pumps, a feed and bleed heat exchanger (four shells), and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-8A).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled. Temporary facilities will be used to fill the Reactor Vessel with water and to purify water in the Reactor Vessel, as a part of the reactor removal activity.

2.2.7 Chemical Shutdown System

The Chemical Shutdown System provided strong boric acid solution to the Main Coolant System for negative reactivity. Boron was required in the Main Coolant System during most operating conditions for reactivity control. The system consists of a boric acid mix tank, transfer pump, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-8A).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.8 Purification System

The Purification System removed impurities from the Main Coolant System. Lower impurity concentration decreases main coolant radioactivity levels and reduces fouling of heat exchange surfaces. The system consists of two purification pumps, ion exchange resin beds, filters, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-8B).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled. Temporary facilities will be used to fill the Reactor Vessel with water and to purify water in the Reactor Vessel as required during the reactor removal activity.

2.2.9 Component Cooling Water System

The Component Cooling Water System provides an intermediate cooling system between contaminated systems and the Service Water System. As an intermediate system it contains leakage of radioactive water from cooled components, preventing contamination from entering the environment. Chromates were used to inhibit corrosion in the Component Cooling Water System. The chromates were removed from the system after plant closure. The system consists of two circulating pumps, two heat exchangers, a surge tank, a chemical addition tank, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-9A, 9B, 9C).

Portions of the system support plant operations during the safe storage period. The portions of the system that are not required will be isolated until they are dismantled. The system will be maintained to provide the following functions during the safe storage period:

- Circulate component cooling water through the Spent Fuel Pit heat exchanger until either all fuel is removed from the Spent Fuel Pit or an alternative Spent Fuel Pit cooling scheme is implemented.

- Circulate component cooling water through the Liquid Waste Evaporator overhead condenser as required, until an alternative cooling water scheme is implemented.

Portions of the system that are not required may be isolated, drained, and sealed.

The Component Cooling Water System may be used to support dismantlement activities by providing cooling water to the Liquid Waste Evaporator Overhead Condenser. This service may be replaced with an alternate cooling system. Temporary liquid waste processing equipment may be installed, eliminating the need to cool the Liquid Waste Evaporator Overhead Condenser.

2.2.10 Primary Plant Corrosion Control System

The Primary Plant Corrosion Control System provided facilities to inject chemicals into the main coolant to minimize the corrosion rate. Corrosion control chemicals were injected into the charging pump suction and hydrogen was injected into the Low Pressure Surge Tank. The system consists of injection pumps, chemical feed tanks, gas bottles, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-92A).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.11 Primary Plant Sample System

The Primary Plant Sample System provided the capability to sample liquids and gases from various plant systems such as the Main Coolant, secondary steam generator blowdown, and the Low Pressure Surge Tank. Samples collected were evaluated for both chemical and radiochemical properties. The system also included the Post Accident Sample System (PASS) which was designed to sample highly radioactive streams from the Main Coolant System following postulated events causing significant fuel failure. The system consists of sample lines, a contaminated sample sink, a sample hood which vents to the Primary Vent Stack, a PASS shielded control station, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number YM-H-8).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled. Manual samples can be obtained from plant systems to support decommissioning activities, if necessary.

2.2.12 Waste Disposal System

The Waste Disposal System receives, contains, treats, and safely disposes all radioactive wastes. The system is divided into liquid, gaseous, and solid waste processing subsystems. The system uses the following processes:

- Natural decay of radioactive isotopes
- Filtration of particulate matter from liquids and gases and of iodine from gases
- Evaporation of liquid wastes and solidification in cement
- Dilution of liquids and gases
- Compaction of solids

The Liquid Waste Processing Subsystem was designed to separately process liquids that contain hydrogen and fission product gases and liquids that contain dissolved air. All systems in the plant that contained hydrogen and fission product gases have been purged and vented to atmosphere. There are no liquids that contain hydrogen and fission product gases remaining at YNPS. All liquid wastes are now treated as wastes containing dissolved air, which allows storage of liquids containing air in tanks previously restricted to liquids containing hydrogen and fission product gases. The subsystem consists of three feed tanks, a Liquid Waste Evaporator, three heat exchangers, two product tanks, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Numbers 9699-FM-41A, 41B, 41C, 41D).

The Liquid Waste Processing Subsystem uses three feed tanks: Activity Dilution Decay Tank, Waste Hold-Up Tank, and Gravity Drain Tank. Liquid from one of these tanks is pumped to the Liquid Waste Evaporator through the Feed-distillate Heat Exchanger. The bottoms stream from the evaporator reservoir is circulated through a steam heated reboiler which discharges to a steam separator. Liquid from the steam separator returns to the evaporator reservoir. Circulation through the reboiler continues until the evaporator bottoms are concentrated such that, when solidified in steel drums, the drum and contents meet all applicable transportation and disposal regulations in accordance with the Process Control Program and plant procedures.

Steam from the steam separator and the evaporator reservoir passes through the evaporator tower to the Evaporator Overhead Condenser. The steam is condensed using component cooling water. The distillate is pumped through the Feed-distillate Heat Exchanger to two product tanks (Test Tanks). Sufficient tank capacity is provided to permit analysis of processed liquid wastes before releasing these wastes to the environment. The contents of the Test Tanks are released to the Sherman Reservoir using established procedures (Section 3.3). All releases are monitored with a radiation detector that is interfaced with an isolation valve. The valve will close on high radiation level or on a loss of instrument air pressure.

The Liquid Waste Processing Subsystem is monitored to assure safe operation, storage, drumming, and controlled release of all radioactivity either to the environment or to approved waste disposal sites. Setpoints for liquid effluent radiation monitoring instrumentation are based on the methodology in the Off-Site Dose Calculation Manual (ODCM). The methodology in the ODCM ensures that members of the public at and beyond the site boundary are not exposed to annual average radionuclide concentrations exceeding maximum permissible concentrations.

The Liquid Waste Processing Subsystem will be used during both the safe storage and dismantlement periods to process liquids drained from radioactive systems and collected from contaminated drains and sumps. This capability could be supplemented or replaced by temporary liquid waste processing facilities. Any changes to the Liquid Waste Processing Subsystem will be reflected in the Process Control Program (Section 3.3.3).

The Gaseous Waste Processing Subsystem was designed to process hydrogen and radioactive fission product gases that were either dissolved in the liquid discharged to the Waste Disposal System or vented from plant systems. There are no plant systems that contain hydrogen and fission product gases. The Gaseous Waste Processing Subsystem was purged and vented during the outage prior to the plant closure in preparation for maintenance activities. The subsystem consists of a surge drum, waste gas compressors, and the necessary associated valves, piping, fittings, and instrumentation.

The Gaseous Waste Processing Subsystem is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

The Solid Waste Processing Subsystem provides the facilities to pack, compact, and store solid waste. The subsystem consists of a solid waste compactor and a storage warehouse. The solid waste processing subsystem will be used during both the safe storage and the dismantlement activities. This capability could be supplemented by temporary solid waste processing facilities or off-site waste processing systems.

2.2.13 Shutdown Cooling System

The Shutdown Cooling System provided a cooling loop to remove heat generated by the radioactive decay of fission products in the reactor core during extended shutdown periods. The decay heat was transferred from the system to the Component Cooling Water System through the Shutdown Cooling Heat Exchanger. The Shutdown Cooling Heat Exchanger also provides redundant cooling capability for the SFP Cooling System. The system consists of a heat exchanger, a circulation pump, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-9C).

The Shutdown Cooling Heat Exchanger and associated valves and lines that provide back-up cooling to the Spent Fuel Pit will be maintained until fuel is permanently removed from the Spent Fuel Pit or an alternate cooling capability is implemented. The portions of the systems that are not required to provide back-up cooling to the Spent Fuel Pit will be drained and sealed until they are dismantled.

2.2.14 Primary Plant Vent and Drain System

The Primary Plant Vent and Drain System supported Main Coolant System and primary auxiliary systems filling, draining, boration, dilution, and flushing operations. Both liquid and gaseous effluent streams from the system discharge to the Waste Disposal System. The system consists of vent and drain headers, the Primary Drain Collection Tank, the Vapor Container Drain Tank, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Numbers 9699-FM-6A, 6B).

The Vapor Container Drain Tank receives liquids that drain from several areas in the Vapor Container. Significant drainage is not expected during the safe storage period. The remainder of the system is not required to support plant operations during the safe storage period and will be maintained in lay-up until it is required to support decommissioning activities.

The Primary Plant Vent and Drain System may be used to support Reactor Vessel dismantlement activities. Water from the Reactor Vessel may be pumped into the system using a temporary connection between the Reactor Vessel and the drain header.

2.2.15 Emergency Core Cooling System

The Emergency Core Cooling System supported plant normal and emergency operations by providing the following major functions:

- Flow path for filling and draining the Shield Tank Cavity

- Flow path for filling Main Coolant System loop piping

- Injection of borated water to the Reactor Vessel to prevent core damage in the event of a loss of coolant accident or inadvertent cooldown

- Long term recirculation of safety injection water to remove decay heat following a loss of coolant accident

The system consists of three trains of high and low pressure pumps, an accumulator, eighteen high pressure nitrogen storage containers, a borated water storage tank, a tank steam heating system (exchanger and circulating pump), an abandoned borated water storage tank, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Numbers 9699-FM-83A, 83B).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.16 Radiation Monitoring System

The Radiation Monitoring System monitors plant radiological conditions through two subsystems:

- Process Radiation Monitoring Subsystem - measures radiation levels of selected systems or processes, providing indication, alarms, and limited control functions

- Area Radiation Monitoring Subsystem - measures radiation levels in selected areas throughout the plant, providing local and remote indication and alarms

The system consists of detectors, and associated instrumentation and controls.

The following components of the Process Radiation Monitoring Subsystem are required to support plant operations during the safe storage and dismantlement periods:

- Liquid waste effluent channel - This channel monitors liquid waste before it is discharged to Sherman Reservoir. If any of the following conditions occur, a valve on the liquid waste discharge line will close terminating the release:

  - A high or failure alarm from the liquid waste discharge rate meter
  - Loss of power to the liquid waste discharge tank flow rate meter or control circuit

Liquid waste releases will continue throughout decommissioning to support dismantlement and decontamination activities.

- Primary Vent Stack channels - These channels monitor airborne releases from ventilated areas of the plant before release to the environment. Airborne release monitoring will continue throughout decommissioning until all ventilated areas are decontaminated (Drawing Number 9699-FK-13A).

The following process radiation monitors are not required to support plant operations during the safe storage and dismantlement periods:

- Steam Generator blowdown tank effluent channel - This channel monitored water drained from the Steam Generator secondary side before it was discharged to Sherman Reservoir. An isolation valve on the Blowdown Tank discharge line was automatically closed whenever the channel alarm setpoint was exceeded. The Steam Generators were removed as part of the Component Removal Project (Appendix A).

- Steam Generator blowdown (four channels) - These channels monitor the Steam Generator blowdown flow paths to detect primary to secondary system leakage. Individual monitoring is not required. The Steam Generators were removed as part of the Component Removal Project (Appendix A).

- Loop seal - This channel monitors releases from the waste gas surge drum to the Primary Vent Stack. The Gaseous Waste Processing Subsystem was purged and vented during the outage prior to the plant closure in preparation for maintenance activities and will remain in that condition until dismantled.

- Post-accident hydrogen vent - This channel monitors controlled gaseous releases from the Vapor Container following an accident that generates significant quantities of hydrogen. The reactor is permanently defueled, therefore, an accident requiring the use of the hydrogen vent is not possible.

- Component cooling water - This channel monitors the Component Cooling Water System to detect leakage of main coolant into the system. There are no sources of highly contaminated water at sufficient pressure to leak into the Component Cooling Water System.

- Main coolant bleed line - This channel monitors the bleed line to detect significant increases in the radiation level of the main coolant. The reactor is permanently defueled, therefore, monitoring of bleed line radiation levels is not needed.

- Vapor Container air particulate - This channel monitors leakage of main coolant into the Vapor Container atmosphere. The reactor is permanently defueled; air particulate monitoring to indicate main coolant leakage is not needed. Ad hoc air monitoring will be employed to support decommissioning activities.

- Air ejector - This channel monitors leakage from the main coolant into the steam system through the Steam Generators. The Steam Generators were removed as a part of the Component Removal Project (Appendix A).

- Main steam - This channel monitors leakage from the Main Coolant System into the secondary side of the Steam Generators. The Steam Generators were removed as a part of the Component Removal Project (Appendix A).

The following area radiation monitoring channels, located in potentially contaminated areas, will be used to monitor conditions during both the safe storage period and system and component dismantling activities:

- Vapor Container and Spent Fuel Pit manipulator cranes during component movement activities

- Waste Disposal Building

- Radiation Control Area Control Point

- Primary Auxiliary Building (Charging Pump cubicles, cubicle corridor, valve room, chemical sample room, and fan room)

- Safety Injection Building

- New Fuel Vault

These channels provide local and remote indication as well as alarm capability.

The following area radiation monitoring channels, located outside of the potentially contaminated areas, will not be used during either the safe storage period or dismantlement activities:

- Accident emergency gamma guard (Turbine Hall)

- Post-accident area monitor (Turbine Hall)

- Auxiliary Boiler Feed Pump room

- Boiler Feed Pump area
- Turbine Hall
- Control Room
- Switchgear Room
- Gatehouse
- Vapor Container High Range

Portions of the system are required to support plant operations during the safe storage and dismantlement periods. The portions of the system that are not required will be dismantled.

2.2.17 VC Ventilation and Purge System

The VC Ventilation and Purge System maintained the Vapor Container atmosphere radionuclide concentration at levels as low as was reasonably achievable. The air removed from the Vapor Container is filtered and released to the Primary Vent Stack. Radiation detection channels from the Radiation Monitoring System are located in the Primary Vent Stack to monitor airborne releases. Filtered and heated air replaces air that is removed by the system. The system consists of inlet air filters and heating coils, an inlet fan, and the necessary associated valves, piping, ducting, fittings, and instrumentation (Drawing Number 9699-FB-3H).

The system is required to support plant operations during the safe storage period. The VC Ventilation and Purge System will be used to support dismantlement activities in the Vapor Container. Airborne particulate radionuclide activity will be controlled and monitored as systems and structures are decontaminated and dismantled.

2.2.18 VC Heating and Cooling System

The VC Heating and Cooling System maintained Vapor Container air temperature within acceptable limits during plant operations. The cooling subsystem used Service Water to cool the air that was heated from hot process surfaces and solar radiation. The heating subsystem uses steam to heat the Vapor Container air during cold season operation. The system consists of heaters, coolers, fans, and the necessary associated ducting, fittings, and instrumentation (Drawing Number 9699-FB-3H).

The system is not required to support plant operations during the safe storage period and will be drained and sealed until it is required to support decommissioning activities. The system will not be returned to service unless it is required to support work activities in the Vapor Container.

The VC Heating and Cooling System may be used to support dismantlement activities inside the Vapor Container. Vapor Container air temperature control is needed to ensure an acceptable working environment. This capability would not be required if a temporary air heating and cooling system is installed.

2.2.19 Post-Accident Hydrogen Control System

The Post-Accident Hydrogen Control System prevented the formation of explosive mixtures of hydrogen and oxygen gases in the Vapor Container following a loss of coolant accident. The system prevented explosive mixtures by circulating air in the Vapor Container and venting hydrogen through a recombiner to the Primary Vent Stack. The system consists of three recirculation fans, two hydrogen analyzers, and the necessary associated valves, piping, ducting, fittings, and instrumentation (Drawing Number 9699-FB-3H).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.20 Containment Isolation System

The Containment Isolation System isolated piping entering and exiting the Vapor Container following an event that significantly increased Vapor Container pressure. The system actuates solenoid valves that vent air from the isolation valve actuators, closing the Vapor Container isolation valves. The system consists of pressure sensors and the necessary associated valves, tubing, fittings, and instrumentation. Vapor Container isolation valves are considered part of their respective systems.

The system is not required to support plant operations during the safe storage and dismantlement periods and will be purged and sealed until dismantled. The capability to isolate the Vapor Container will be maintained to preclude exposure to radiation and inadvertent spread of contamination (Section 2.3.4.3).

2.2.21 Fuel Handling Equipment System

The Fuel Handling Equipment System supports the handling of fuel and irradiated components by providing the following major functions:

- Underwater handling of irradiated fuel in the reactor and Spent Fuel Pit
- Containment of water to support handling and storage of irradiated components
- Transferring fuel between the new fuel vault, Spent Fuel Pit, Shield Tank Cavity, and Reactor Vessel
- Storing irradiated fuel in the Spent Fuel Pit and new fuel in the new fuel vault
- Movement of casks to and from the Spent Fuel Pit
- Handling of Reactor Vessel components

The system consists of several cranes (Spent Fuel Pit and Vapor Container manipulator cranes, polar crane, yard area crane, new fuel hoists), fuel inspection equipment, fuel upenders and transfer carriage, grappling fixtures, reactor internal component lifting devices, fuel storage racks, and the necessary associated controls and instrumentation.

The Spent Fuel Pit Manipulator Crane is a trolley mounted on a bridge that moves along rails set in concrete at the top of the Spent Fuel Pit. A rotating turret is mounted on the trolley that supports a rigid tool boom. The tool boom is fitted with a grappling tool which can engage the top end of a fuel assembly, control rod, or shim rod. The grappling tool is latched by spring action and unlatched pneumatically. The tool is designed to not disengage when loaded. The crane may also be fitted with a manual fuel assembly lifting fixture.

The Vapor Container Manipulator Crane was removed during the Component Removal Project (Appendix A).

The Spent Fuel Pit fuel handling equipment will be used intermittently until fuel is removed permanently from the Spent Fuel Pit. If fuel is stored in the Spent Fuel Pit during dismantlement activities, special precautions will be instituted to protect the Spent Fuel Pit and associated support systems from physical damage and other adverse conditions that could occur (Section 3.3.1).

2.2.22 SFP Cooling and Purification System

The SFP Cooling and Purification System cools and purifies Spent Fuel Pit water. The system transfers decay heat from the fuel stored in the Spent Fuel Pit to the Component Cooling Water System and controls the concentration of radioactive corrosion products in the Spent Fuel Pit water. The system consists of two circulating pumps, one heat exchanger, one ion exchange bed or filter, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-9C).

The system uses one of two circulating pumps to move water between the Spent Fuel Pit and a heat exchanger, maintaining water temperature below the design limit of 150°F. Water can also be circulated through an ion exchanger or filter located in the Ion Exchange Pit. Either cooling pump can take suction from either end of the Spent Fuel Pit and discharge into the south end of the Spent Fuel Pit. The Shutdown Cooling Heat Exchanger provides redundant cooling capability, if the Spent Fuel Pit Cooling Heat Exchanger is unavailable. The system is required to support plant operations until fuel is removed permanently from the Spent Fuel Pit.

The SFP Cooling and Purification System is not required to support dismantlement activities. If fuel is stored in the Spent Fuel Pit during dismantlement activities, special precautions will be instituted to protect the system from physical damage and other adverse conditions that could occur (Section 3.3.1). After the system is not required, it will be drained and sealed until dismantled.

2.2.23 Main Steam System

The Main Steam System supported normal and emergency operations by providing the following major functions:

- Flow path from the Steam Generators in the Vapor Container to the steam turbine in the turbine building

- Safety valves for over-pressure protection

- Non-return valves for steam line break protection

The system consists of the main steam lines, non-return valves, safety valves, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Numbers 9699-FM-3A, 3B).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.24 Feedwater System

The Feedwater System pumped condensate from the Condensate System to the Steam Generators where it was evaporated and returned as steam to the turbine generator. The system consists of three pumps and the necessary associated valves, piping, fittings, and instrumentation (Drawing Numbers 9699-FM-4B, 4C).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.25 Steam Generator Blowdown System

The Steam Generator Blowdown System controlled Steam Generator secondary side solids concentration by providing a monitored flow path from the Steam Generator secondary side water phase to the plant discharge structure. The flow path was also used to drain the Steam Generator secondary side. The system consists of a blowdown tank, blowdown control valves, combined blowdown radiation detector, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-3B).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.26 Emergency Feedwater System

The Emergency Feedwater System provided feedwater to the Steam Generators in the event of a failure of the Feedwater System. The system is capable of injecting feedwater through two flow paths: normal feedwater flow path, and Steam Generator blowdown flow path. The system consists of two motor driven pumps, one steam turbine driven pump, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-3B).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.27 Service Water System

The Service Water System supports plant operations by supplying water from Sherman Reservoir for the following major uses:

- Cooling water to plant components and systems
- Feed water to the Demineralized Water System
- Dilution water for waste water releases
- Fire water to the Vapor Container hose reels

The system consists of three service water pumps, two booster pumps, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Numbers 9699-FM-27B, 27C). One of the booster pumps, which supplied the Vapor Container Service Water, was removed from service and installed to operate in parallel with the three Service Water pumps in the Screenwell House.

Service Water will be provided for the following uses during the safe storage and dismantlement periods:

- Component Cooling Water System cooling to remove heat from the Spent Fuel Pit and Liquid Waste Evaporator
- Control Air Compressor cooling water
- Waste water effluent dilution
- Vapor Container fire water hose reels
- Miscellaneous cooling loads (e.g. air conditioning units)

The portions of the system that are not required during safe storage and decommissioning periods will be drained and sealed until they are dismantled.

2.2.28 Demineralized Water System

The Demineralized Water System supports plant operations by providing demineralized water for the following major uses:

- Make-up water to Spent Fuel Pit
- Water for decontamination activities
- Feed water make-up to the Heating System

The system consists of a water treatment facility, two water storage tanks, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-4D).

The system is required to support plant operations during the safe storage period. Demineralized water will be used for flushing and decontamination to support plant system lay-up activities. Demineralized water is also needed to meet make-up requirements for the Spent Fuel Pit and the Heating System.

The Demineralized Water System will be used to support dismantlement activities. Demineralized Water System make-up requirements may increase during dismantlement activities to support filling of the Reactor Vessel and other systems for shielding as well as increases in decontamination activities. The on-site water treatment system may be supplemented or replaced by a temporary facility.

2.2.29 Compressed Air System

The Compressed Air System provides air for plant use through three subsystems:

- Control Air Subsystem - provides compressed air to instrumentation and valves
- Service Air Subsystem - provides compressed air for pneumatic tools and the damper controls of the Ventilation System
- Breathing Air Subsystem - provides compressed air for the breathing air stations in the Vapor Container and the air bottle filling station

The system consists of station air compressors, breathing air compressors, receiver tanks, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-26A, 26B, 26D).

The Compressed Air System is required to support plant operations during the safe storage period and will be used to support dismantlement activities. The system also is needed to provide compressed air to the fuel handling equipment until fuel is removed permanently from the Spent Fuel Pit or a temporary compressed air system is available. The Breathing Air Subsystem will be maintained to support supplied air requirements for dismantlement activities. The Breathing Air Subsystem is not required if a temporary breathing air system is installed.

Portions of the Compressed Air System will be isolated and dismantled as the systems that they support are removed permanently from service.

2.2.30 Electrical System

The Electrical System provided equipment to generate power and to deliver it to a transmission system (115 kV). The system also distributes AC power throughout the plant. Back-up capability is supplied through a diesel generator and a DC station battery. The system consists of transformers, motor control stations, distribution boxes, and the necessary associated instrumentation and controls (Drawing Numbers 9699-RE-1F, 9699-FE-1B, 1C, 1D, 1G, 1H, 1J, 1K, 1L, 1M).

The Electrical System is required to support plant operations during the safe storage period and will be used to support decommissioning activities. Portions of the system are also needed to provide power and back-up power to support spent fuel storage until fuel is removed permanently from the Spent Fuel Pit. Portions of the system will be isolated and dismantled as the systems that they support are removed permanently from service.

A diesel generator is available to supply power in the unlikely event that off site power is unavailable. Automatic starting capability is not needed to support plant operation in a permanently defueled condition (Reference 2.2-2). The requirements will be reduced further as the decay heat of the spent fuel stored in the Spent Fuel Pit continues to decrease or the spent fuel is removed permanently from the Spent Fuel Pit.

Prior to final dismantlement of the Electrical System components, a temporary supply will be installed to support dismantlement activities and fuel storage power requirements. The permanent system will be fully de-energized during Electrical System removal activities to preclude access to energized components.

2.2.31 Heating System

The Heating System supports plant operations by supplying steam from the Auxiliary Boilers for the following major uses:

- Domestic hot water, including decontamination showers
- Tank and building heating
- Liquid Waste Evaporator reboiler heating

The system consists of two auxiliary boilers, condensate tanks, heat exchangers, circulating pumps, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FB-7B).

The Heating System is required to support plant operations during the safe storage and dismantlement periods. The system also is needed to heat areas of the plant with systems and structures that support fuel storage until fuel is removed permanently from the Spent Fuel Pit. Portions of the system will be isolated and dismantled as the systems that they support are removed permanently from service.

2.2.32 Ventilation System

The Ventilation System supports plant operations by providing the following functions:

- Filtration of air ventilated from enclosed areas
- Heating and cooling of building spaces

Exhaust air from the Primary Auxiliary Building cubicle area, Waste Disposal Building, and Spent Fuel Pit Building is filtered through a high efficiency filter assembly before monitoring and discharging to the Primary Vent Stack. The system consists of ventilators, fans, charcoal and HEPA filters, heaters, and the necessary associated ducting, fittings, and instrumentation (Drawing Number 9699-FB-7K).

The Ventilation System with the exception of the charcoal filters, is required to support plant operations during the safe storage period. The system also is needed to ventilate areas of the plant with systems and structures that support fuel storage until fuel is removed permanently from the Spent Fuel Pit. The system will be used to support dismantlement and decontamination activities. Airborne releases in the areas that are ventilated by this system will be filtered and monitored before they are discharged to the environment.

2.2.33 Fire Protection and Detection System

The Fire Protection and Detection System provides the equipment needed to detect and respond to fires that could occur in the plant. The system has the following subsystems:

- Sprinkler subsystem (wet and dry)
- Foam and deluge subsystem
- Carbon dioxide flooding subsystem
- HALON flooding subsystem

The system consists of electric and diesel driven fire pumps, pressure maintenance tank, hydrants, hoses, detectors, and necessary associated valves, piping, fittings, and instrumentation (Drawing Numbers 9699-FM-90A, 90C).

Portions of the Fire Protection and Detection System are required to support plant operations during the safe storage and dismantlement periods. The system also is needed to protect areas of the plant with systems and structures that support fuel storage until fuel is removed permanently from the Spent Fuel Pit. The Fire Protection Technical Requirements Manual presents system availability requirements (Reference 2.2-3). Portions of the system will be isolated and dismantled as the systems that they support are removed permanently from service.

2.2.34 Primary Pump Seal Water System

The Primary Pump Seal Water System supplied demineralized water to the following pump seals: Shutdown Cooling Pump, Low Pressure Surge Tank Cooling Pump, and Purification Pumps. The seal water system operates at a higher pressure than the maximum working pressure of the pumps that are supplied, preventing leakage from the pump seals to the environment. The system consists of a supply tank, pump, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-60A).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.35 Safe Shutdown System

The Safe Shutdown System provided emergency make-up water to the Main Coolant System and the Feedwater System in the event that all normal and emergency systems became unavailable. The system was designed to allow transition from hot shutdown conditions to cold shutdown conditions following a seismic event, flood, fire, or a tornado. The system consists of a dedicated motor control center, two pumps, instrumentation panel, and the necessary associated valves, piping, fittings, and instrumentation (Drawing Number 9699-FM-100B).

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.36 Water Cleanup System

The Water Cleanup System was designed to remove radionuclides from water recirculated to the Reactor Vessel from the Vapor Container sump during a loss of coolant accident. The system consists of ion exchange capsules, filters, heat exchanger, and the necessary associated valves, piping, fittings, and instrumentation.

The system is not required to support plant operations during the safe storage and dismantlement periods and will be drained and sealed until dismantled.

2.2.37 Vapor Container

The Vapor Container is a spherical steel structure that surrounds the Reactor Support Structure, Reactor Vessel, Main Coolant System, and miscellaneous Vapor Container support systems (Drawing Numbers 9699-FV-1A, 9699-FM-1A, 1B, 1C, 46A). The Vapor Container provided one of the barriers credited in the accident analyses for the prevention of a fission product release to the environment.

The Vapor Container is an elevated steel sphere located about 23 feet above grade and is supported by 16 steel columns. Each column is braced by steel rods that provide cross bracing for lateral loads. The steel columns are supported by reinforced concrete pedestals. The Vapor Container is an unfired pressure vessel that was designed, built, and tested in accordance with applicable ASME Boiler and Pressure Vessel Codes.

There are no significant structural attachments to the Vapor Container. Attachments are limited to minor platform framing, exterior stairs, and lightly loaded supports for pipes and cable trays.

The following comprise major Vapor Container penetrations:

- Reactor Support Structure support columns - The Reactor Support Structure support columns carry the Reactor Support Structure load. Each column is isolated from the Vapor Container by bellows located at the Vapor Container shell.

- Fuel Transfer Chute - The Fuel Transfer Chute is a concrete shielded pipe connecting the Shield Tank Cavity and the Spent Fuel Pit. The chute was used to transfer fuel assemblies, control rods, and miscellaneous irradiated components.

- Equipment Hatch - The Equipment Hatch provides an opening, about 14 feet in diameter, in the Vapor Container lower hemisphere. The hatch allows transport of equipment into and out of the Vapor Container. A temporary cover may be used to isolate the Vapor Container.

- Personnel Hatch - The Personnel Hatch provided a dual airlock for personnel access into and out of the Vapor Container when containment integrity was required. A temporary door may be used in place of the Personnel Hatch to isolate the Vapor Container.

- Piping penetrations - The Vapor Container piping penetrations are reinforced enclosures for process piping entering and exiting the Vapor Container. High temperature piping is isolated from the Vapor Container shell by steel thermal sleeves which are attached to the Vapor Container shell.

- Electrical penetrations - Electrical penetrations provide gas-tight fittings for electrical power and instrumentation control leads entering and exiting the Vapor Container. Electrical penetrations are made from steel pipe penetrations welded into the Vapor Container with bolted and gasketed flanges. An electrical penetration cartridge with matching flange is drilled and tapped to receive sealed fittings for one or several conductors.

The Vapor Container shell prevents inadvertent exposure of plant personnel to radioactivity and the spread of contamination. Access to the Vapor Container will be controlled to limit access to significantly irradiated systems and components. The Vapor Container shell will be used as a barrier to control the release of radioactive materials that may become airborne during dismantlement activities.

2.2.38 Reactor Support Structure

The Reactor Support Structure is a reinforced concrete structure which supports the reactor pressure vessel, the Pressurizer, the Main Coolant System, and the polar crane (Drawing Number 9699-FM-1A, 1B, 1C). The structure also provides shielding to attenuate radiation from systems located inside the Vapor Container.

The Reactor Support Structure consists of two concentric reinforced concrete cylinders. The cylinders are connected together with reinforced concrete radial walls which form compartments for the main coolant loops, the Pressurizer, and the Equipment Hatch. The compartment radial concrete walls have several ports to limit differential pressure across the walls in the event of a major loss of coolant accident.

The inner concrete wall supports the Reactor Vessel and the Neutron Shield Tank. The inner wall also forms the Shield Tank Cavity above the vessel. The Shield Tank Cavity is clad with a stainless steel liner to limit leakage of cavity water.

The operating compartments are covered by a reinforced concrete charging floor. The charging floor is composed of removable concrete slabs which allow crane access to equipment in the main coolant loop compartments. Spiral stairs provide personnel access between the charging floor and the main coolant loop compartments.

The Reactor Support Structure is supported on eight reinforced concrete steel encased columns which penetrate the Vapor Container shell. The Vapor Container penetrations are sealed by stainless steel expansion joints. An annular space is provided to permit the Vapor Container and the internal concrete structure to move independently.

2.2.39 Vapor Container Polar Crane

The Vapor Container polar crane was used to support refueling and maintenance related activities inside the Vapor Container. The crane was originally designed for the installation of the Reactor Vessel and the Steam Generators. However, crane capacity was reduced during plant operations by converting one hook to a smaller capacity to increase hook travel speed. The smaller hook was replaced with a larger hook as a part of the Component Removal Project (Appendix A), returning the polar crane to its original capacity.

The crane consists of a bridge which rides on a 75 foot diameter crane rail with a common trolley rigged with two hooks. The rated capacity of the bridge and common trolley is 150 ton. The installed hooks have rated capacities of 75 ton each.

The Vapor Container polar crane will be to support dismantlement activities in the Vapor Container.

2.2.40 Radiation Shielding

Radiation shielding is installed for both personnel protection and equipment protection. The radiation shielding is comprised of several categories according to function:

- Neutron Shield Tank - The Neutron Shield Tank is described in Section 2.2.41.

- Primary Shield - The Primary Shield is a cylindrical, reinforced concrete structure immediately adjacent to and surrounding the exterior of the Neutron Shield Tank. Combined with the shield tank, the Primary Shield attenuated neutron and gamma radiation from the reactor core. The shield also transfers the weight of the Reactor Vessel and the Neutron Shield Tank to the main support structure.

  The lower portion of the primary shield is 4.5 to 5 feet thick and extends from the Reactor Support Structure to the Shield Tank Cavity floor. The upper portion of the shield forms the walls of the Shield Tank Cavity. The Primary Shield is 6 feet thick in the area adjacent to the Equipment Hatch to provide additional shielding during refueling activities.

- Secondary Shield - The Secondary Shield surrounds the entire primary plant inside the Vapor Container. The shield reduces radiation dose outside of the Vapor Container to below acceptable levels. The bottom portion of the shield forms a section of the main structural support for the primary plant. The portion of the shield below the charging floor is 5 feet thick. The structure also shielded portions of the Steam Generator above the charging floor. The upper portion of the secondary shield is 2 feet thick and extends above the charging floor to support the polar crane.

Additional shield walls located between the primary and secondary shields segregate the main coolant loop, the Steam Generators, the Pressurizer, and the Equipment Hatch compartments. These partitions are designed to reduce radiation from adjacent compartments during plant shutdown and equipment maintenance periods. The partitions separating the main coolant loops and Steam Generators are 2 feet thick. The Pressurizer partitions are 1.5 feet thick. The walls surrounding the Equipment Hatch taper from 5 feet thick at the base to 2 feet thick at the top.

- Fuel Handling Shield - The fuel handling shield attenuates the radiation from the transfer of fuel assemblies, control rods, and other irradiated components through the Fuel Transfer Chute. Both concrete and lead are used to shield the spent fuel chute. Where space permits, the Fuel Transfer Chute is shielded with 4 feet of concrete; where space is limited, 14 inches of lead is used. Transfer of irradiated fuel and control rods from the Reactor Vessel to the Spent Fuel Pit was completed during the outage prior to plant closure. Sections of the core baffle and lower core support plate were transferred to the Spent Fuel Pit during the Component Removal Project (Appendix A). The possession only license precludes any future transfers of fuel to the Vapor Container.

- Auxiliary Shielding - Auxiliary shielding protects personnel during operating and maintenance activities in several plant areas where radiation exposure could be excessive. Shielding is provided in the following areas: the Control Room, sampling room, in the vicinity of the Waste Disposal System, the Purification System, and the Chemical Shutdown System. This shielding will be maintained as necessary during the safe storage and dismantlement periods to prevent inadvertent exposure to radiation.

- Supplemental Shielding - Supplemental shielding is designed to reduce the dose rates from neutron streaming from the Reactor Vessel. The shielding was comprised of blocks surrounding the Reactor Vessel flange area. The reactor is permanently defueled. This shielding is not needed during the safe storage and dismantlement periods.

2.2.41 Neutron Shield Tank

The Neutron Shield Tank is an annular, carbon steel tank that surrounds the Reactor Vessel. The tank was filled with component cooling water during plant operations.

The tank is constructed in two sections. The lower section surrounds the Reactor Vessel from the vessel bottom to the vessel support ring. The top section surrounds the Reactor Vessel from the vessel support ring to just above the main coolant inlet and outlet nozzles. The upper portion is connected to the lower portion with eight sets of circulation pipes.

The Neutron Shield Tank supported plant operations by providing the following major functions:

- Shielded plant components within the Vapor Container from neutron irradiation
- Prevented overheating and dehydration of the primary concrete shield
- Moderated neutrons for the excore nuclear detectors

The Neutron Shield Tank contains eight fin-tube coolers which provided cooling to the tank. The tank also contains eight instrumentation wells. Seven of these are neutron detector wells and one is a Neutron Shield Tank temperature well.

A surge tank located at an elevation higher than the top of the Neutron Shield Tank maintained water level in the Neutron Shield Tank. The top of the Neutron Shield Tank is vented to the surge tank which is vented to the Vapor Container atmosphere.
Two annular dry wells are located above the tank. The dry well directly above the tank provides an area for the neutron detector cabling, the shield tank piping, and the shield tank heat exchanger piping and connections. This dry well is accessed through eight manhole covers. The manhole covers are accessed from the upper dry well. The upper dry well is the area immediately below the walkway that surrounds the Reactor Vessel. The dry wells are sealed to prevent water entry from the Shield Tank Cavity.

The tank has been drained and vented and will be maintained in that condition until dismantled.

2.2.42 Pipe Chases

Two pipe chases are used as piping corridors between the Primary Auxiliary Building and the Vapor Container:

- Lower Pipe Chase - The Lower Pipe Chase is a corridor that runs between the second story of the Primary Auxiliary Building and the Vapor Container lower hemisphere. The chase is constructed of reinforced concrete.

- Upper Pipe Chase - The Upper Pipe Chase is a corridor that runs from the Primary Auxiliary Building roof to the Vapor Container lower hemisphere. The chase is constructed of concrete masonry units and is supported by the lower pipe chase.

The pipe chases are required to prevent inadvertent exposure to radiation and spread of contamination until piping inside the chases is removed.

2.2.43 Fuel Transfer Chute

The Fuel Transfer Chute was used to transfer new and spent fuel as well as irradiated components between the Spent Fuel Pit and the Vapor Container. The chute is a series of stainless steel pipe sections connected by bolted flanges enclosed in a reinforced concrete tunnel. The chute is structurally isolated from the Vapor Container by a metal bellows expansion joint. The Fuel Transfer Chute is accessed through a below grade manhole tank.

All transfers of irradiated fuel and control rods between the Reactor Vessel and Spent Fuel Pit were completed during the outage prior to plant closure. The possession only license precludes any future transfers of fuel.

The Fuel Transfer Chute will be isolated from the Spent Fuel Pit if fuel is stored in the Spent Fuel Pit coincident with dismantlement activities that could interact with the Fuel Transfer Chute (e.g., heavy lifts near Fuel Transfer Chute). The Fuel Transfer Chute will be isolated by capping the chute at the inside wall of the Spent Fuel Pit and filling the Fuel Transfer Chute between the lower lock valve and the cover with a structurally stable material (e.g., grout, hydraulic cement).

2.2.44 Yard Area Crane And Support Structure

The yard area crane support structure is a braced steel frame structure that supports a crane that services the Ion Exchange Pit, the Spent Fuel Pit, and the Decontamination Room. The crane support structure also provides lateral support to the Vapor Container service elevator tower. The crane support structure is about 34 feet by 151 feet by 73 feet high with a design capacity of 60 tons.

The Yard Area Crane and support structure will be used to support fuel management activities until fuel is permanently removed from the Spent Fuel Pit. The crane will also be used to support activities associated with the Spent Fuel Pit, Ion Exchange Pit, and other heavy lifts.

2.2.45 Ion Exchange Pit

The Ion Exchange Pit is a reinforced concrete structure that contains the ion exchange vessels which are used to purify the Spent Fuel Pit and Main Coolant System (Drawing Number 9699-FM-35B). The pit is divided into two compartments by an interior wall. The south compartment is a nominal 6 feet by 30 feet by 5 feet deep and contains the valves used to isolate and control ion exchanger flow. The north compartment is a nominal 20 feet by 28 feet by 18 feet deep and contains five ion exchanger storage bays. Each bay may hold one operating ion exchanger and three stored ion exchangers. The north compartment is filled with water to provide additional radiation shielding. The water contains a residual amount of chromates, which were inadvertently added during plant operations. The top of the Ion Exchange Pit is covered by removable concrete slabs on steel plates.

The Ion Exchange Pit supports Spent Fuel Pit operation and will remain in operation until fuel is removed from the Spent Fuel Pit. This capability would not be required if alternate filtering and ion exchange equipment are installed. The common wall between the Ion Exchange Pit and the Spent Fuel Pit will be maintained until fuel is permanently removed from the Spent Fuel Pit. Chromates will be removed from the water in the Ion Exchange Pit before water is drained from the pit.

2.2.46 Primary Vent Stack

The Primary Vent Stack is a steel stack that vents monitored airborne releases from the Ventilation System and the VC Ventilation and Purge System. The stack is a nominal 5 feet outside diameter and 130 feet high. The bottom of the stack is supported by a steel frame that is supported by the Primary Auxiliary Building.

The Primary Vent Stack is required during the safe storage and dismantlement periods to vent air processed by both the Ventilation System and the VC Ventilation and Purge System.

2.2.47 Spent Fuel Pit And Spent Fuel Pit Building

The Spent Fuel Pit is a reinforced concrete structure that provides underwater storage of irradiated fuel and control rods and associated fuel transfer equipment (Drawing Number 9699-FM-21A). The Spent Fuel Pit inside dimensions are about 16 feet by 34 feet by 33 feet deep with a wall thickness that varies between 5 and 6 feet. The Spent Fuel Pit floor is constructed from a 3 foot mat located about 17 feet below grade. The Spent Fuel Pit walls and floor are lined with stainless steel to prevent leakage.

Fuel assemblies, neutron sources, canisters containing portions of the Reactor Vessel internals (core baffle and lower core support plate), and other irradiated components are stored in a two tier rack system. Currently, there are 533 fuel assemblies stored in the Spent Fuel Pit. The lower racks are an anodized aluminum support structure, to which are attached welded Boral sheets. The upper racks are modules supported on intermediate columns attached to the Spent Fuel Pit floor. The modules are comprised of boral contained between an inner and outer canister wall. Grating is installed between upper and lower racks. The racks are designed to maintain proper spacing and structural integrity after being impacted by a fuel assembly dropped onto any location from a height of six inches above the top of the racks.

The Spent Fuel Pit Building is a steel-braced frame, metal sided structure that supports the superstructure to both the new fuel vault and the Spent Fuel Pit. The building provides an enclosed work area and contains the Spent Fuel Manipulator Crane, the New Fuel Hoist, and the SFP Cooling System pumps. Roof hatches are provided for equipment and cask access using the Yard Area Crane which is located directly above the building.

The Spent Fuel Pit will be maintained until fuel is permanently removed from the pit. If fuel is stored in the Spent Fuel Pit during dismantlement activities, special precautions will be instituted to protect the Spent Fuel Pit and associated support systems from physical damage and other adverse conditions that could occur (Section 3.3.1).

2.2.48 New Fuel Vault

The New Fuel Vault is a reinforced concrete and concrete masonry structure that provides secure storage for new fuel (Drawing Number 9699-FM-21B). The vault is contained within a lower section of the Spent Fuel Pit Building. The vault has a bridge crane used for new fuel assembly handling activities. The west and south walls of the New Fuel Vault are common to the Spent Fuel Pit and the Ion Exchange Pit respectively.

The New Fuel Vault is required to support plant activities until fuel is permanently removed from the Spent Fuel Pit. The common walls between the vault and the Spent Fuel Pit and Ion Exchange Pit will be maintained.

2.2.49 Primary Auxiliary Building

The Primary Auxiliary Building (Drawing Numbers 9699-FM-57A, 57B) contains many of the plant systems required to support primary plant operations (e.g. Charging and Volume Control System, Shutdown Cooling System, Purification System, Component Cooling System, Ventilation System). The Primary Auxiliary Building is a concrete masonry building with two stories and a partial basement at the southeast corner.

The Primary Auxiliary Building will be required during the safe storage and dismantlement periods to prevent inadvertent exposure to radiation and spread of contamination until the contaminated systems and components within the building are dismantled and the building is decontaminated.

2.2.50 Diesel Generator Building

The Diesel Generator Building (Drawing Numbers 9699-FM-19A, 81A) contains the plant systems that support plant response to postulated accidents and transients: Emergency Core Cooling System, emergency diesel generators, station service battery number 3, and safety injection accumulator. The structure is a braced steel framed building with concrete masonry unit walls. A nitrogen storage tank support frame is connected to the building roof and extends above the roof. This support frame holds the nitrogen supply bottles for the safety injection accumulator. Nitrogen was vented from the bottles as a part of plant closure activities.

The Diesel Generator Building will be required during the safe storage and dismantlement periods to prevent inadvertent exposure to radiation and spread of contamination until the contaminated systems and components within the building are dismantled and the building is decontaminated.

2.2.51 Waste Disposal Building

The Waste Disposal Building contains systems and structures for processing, packaging, and temporarily storing prior to shipment low level radioactive waste (9699-FA-17A). The structure is a steel framed building with concrete masonry unit walls. The walls in the southeast corner of the building are constructed from filled masonry block to provide shielding from the Waste Holdup and Activity Dilution Decay Tanks.

The Waste Disposal Building will be required during the safe storage and dismantlement periods to prevent inadvertent exposure to radiation and spread of contamination until the contaminated systems and components in the building are removed and the building is decontaminated.

2.2.52 Safe Shutdown System Building

The Safe Shutdown System Building contains the Safe Shutdown System (Drawing Number 9699-FM-100A). The structure is constructed of reinforced concrete walls.

The Safe Shutdown System Building will be required during the safe storage and dismantlement periods to prevent inadvertent exposure to radiation and spread of contamination until the contaminated components within the building are dismantled and the building is decontaminated.

2.2.53 Potentially Contaminated Area Storage Buildings 1 & 2 and Warehouse

There are three major areas located on the plant site for the storage of radioactive materials and waste awaiting shipment:

- Potentially Contaminated Area (PCA) Storage Building 1 - PCA Storage Building 1 is used primarily for the storage of low level radioactive material prior to shipping. The structure's walls are constructed from concrete masonry units.

- PCA Storage Building 2 - PCA Storage Building 2 is used for the storage of contaminated tools and equipment. The structure's walls are constructed from uninsulated corrugated metal panels.

- PCA Warehouse - The PCA Warehouse is used for storage of low level radioactive waste, waste containers, and contaminated equipment prior to shipment. The structure is a steel framed building with reinforced concrete masonry unit walls.

The storage areas will be used during the safe storage and dismantlement periods to prevent inadvertent exposure to radiation and spread of contamination. The structures will be decontaminated after the contaminated materials stored within the areas are removed permanently.

2.2.54 Compactor Building

The Compactor Building contains two solid waste compactors and provides a packaging area for radioactive waste shipping containers. The structure's walls are constructed from reinforced concrete masonry units.

The building will be required during the safe storage and dismantlement periods to prevent inadvertent exposure to radiation and spread of contamination. The structure will be decontaminated after contaminated material processing is not required.

2.2.55 Service Building

The Service Building is divided into two sections. One of those sections is located in the Radiation Control Area of the plant. The section in the Radiation Control Area contains the primary side machine shops, decontamination rooms, control point, primary side chemistry laboratory, counting room, and decontamination showers. The structure's walls are constructed from reinforced concrete masonry units.

The building will be required during the safe storage and dismantlement periods to support dismantlement and decontamination activities. The structure will be decontaminated after most of the site decommissioning activities have been completed.

2.2.56 Miscellaneous Tanks

This section presents descriptions of plant tanks. These tanks are contaminated, potentially contaminated, or are needed to support decommissioning activities:

- Safety Injection Tank - The Safety Injection Tank provided a source of borated water to the Safety Injection Pumps and Low Pressure Surge Tank during plant operations and provided a source of borated water to the Shield Tank Cavity during refueling. The Safety Injection Tank is a component of the Emergency Core Cooling System (Section 2.2.15). The tank is an atmospheric, cylindrical stainless steel shell vessel with a fixed dome roof and flat bottom. An overflow line is installed to route excess inventory to a floor drain in the Waste Disposal Building. This tank may be used for temporary storage of demineralized water or mildly contaminated water to support decommissioning activities.

- Abandoned Safety Injection Tank - The abandoned Safety Injection Tank was removed from service in 1990 due to minor leakage. The tank is constructed from aluminum. This tank is empty and will not be used to support decommissioning.

- Primary Water Storage Tank - The Primary Water Storage Tank stored demineralized water to support plant operations. The tank is a component of the Demineralized Water System (Section 2.2.28). The tank is constructed of aluminum and has an inner floating roof which restricts the absorption of air into the water. The tank may be used to store demineralized water to support Spent Fuel Pit operations and the safe storage and dismantlement period activities.

- Waste Processing System Tanks - The Waste Processing System (Section 2.2.12) uses several tanks for feed and product storage:

  - Monitored Waste Tanks - The two Monitored Waste Tanks were designed to store waste liquid that contained dissolved air. Each tank is constructed of steel and has an approximate diameter of 5 feet and a nominal height of 8 feet. These tanks are not used and will not be used to support the safe storage and dismantlement period activities.

- Waste Holdup and Activity Dilution Decay Tanks - The Waste Holdup and Activity Dilution Decay Tanks are large capacity tanks that store contaminated liquids prior to processing. Each tank is constructed of steel and is surrounded by a moat. The moat is designed to contain the tank inventory in the event of tank failure. These tanks will be used to support waste processing during the safe storage and dismantlement periods.

- Test Tanks - The two Test Tanks store processed liquids prior to testing and discharge from the plant. Each tank is constructed of steel. These tanks will be used to support waste processing during the safe storage and dismantlement periods.

- Waste Gas Surge Drum - The Waste Gas Surge Drum stored hydrogen and fission gases removed from contaminated liquids and plant systems. The drum is constructed of steel and is surrounded by a reinforced concrete shield wall. All plant systems, including the Waste Gas Surge Drum, were purged and vented of hydrogen and fission gasses during the outage prior to the plant closure in preparation for maintenance activities. The Waste Gas Surge Drum is not required to support the safe storage and dismantlement period activities.

- Waste Gas Decay Drums - The three Waste Gas Decay Drums were designed to store excess hydrogen and fission gases recovered by the Gaseous Waste Processing Subsystem. The drums are constructed of steel and are surrounded by a reinforced concrete shield wall. Use of these drums was discontinued after a few years operation due to operational and maintenance considerations. The drums will not be used to support the safe storage and dismantlement period activities.

- Service Building Radioactive Sump Tank - The Service Building Radioactive Sump Tank temporarily stores water from the Radiation Control Area, chemistry laboratory, and machine shop drains and sinks. Water collected in the tank is pumped to the gravity drain tank as necessary. This tank will be maintained until Service Building decontamination activities are completed.

• Demineralized Water Storage Tank - The Demineralized Water Storage Tank provides a source of demineralized water to support secondary plant operations. The tank is a component of the Demineralized Water System (Section 2.2.28). The tank is constructed of aluminum and is connected by a gravity feed line with the primary water storage tank which is at a higher elevation. The tank may be used to provide demineralized water to support Spent Fuel Pit operations, auxiliary boiler operation, and the safe storage and dismantlement period activities.

- Fire Water Storage Tank - The Fire Water Storage Tank provides a back-up source of water to the Fire Protection and Detection System (Section 2.2.33). The tank is constructed of steel. The tank will be used to support operation of the Fire Protection and Detection System during the safe storage and dismantlement period activities.

- Fuel Oil Storage Tank - The Fuel Oil Storage Tank stores fuel oil used by the auxiliary boilers and the emergency diesel generators. The tank is constructed of steel. The tank will be used to support the safe storage and dismantlement period activities until after the Auxiliary Boilers and the Emergency Diesel Generators are not required.

2.2.57 Meteorological Tower

The Meteorological Tower is used to support routine effluent controls identified in the Off-site Dose Calculation Manual and determination of wind speed and direction as required by the Defueled Emergency Plan. The tower is a 200 ft structures with the capability to measure wind speed and direction at about 33 ft and 195 ft above ground level, and vertical temperature difference between about 33 ft and 195 ft.

The tower will be required during the safe storage and dismantlement periods to provide data for both the Off-site Dose Calculation Manual and the Defueled Emergency Plan.

2.2.58 Site Characteristics

2.2.58.1 Demography

The population density in the rural area surrounding the site is low. Within one mile of the site, for example, the population is 48 (based on 1989 Massachusetts municipal census counts). The populations of the two closest towns, Rowe and Monroe, are 354 and 141, respectively. The nearest population center of 25,000 or more is Pittsfield, Massachusetts, located about 21 miles southwest of the site. The regional population is expected to remain virtually unchanged over the next decade.

The nearest resident is located in Monroe, approximately 1500 feet northwest of the site along the western shore of the Sherman Reservoir. This is well beyond the plant's protected area.

2.2.58.2 Geography and Land Use

YNPS is located in the Berkshire Hills of Franklin County in Rowe, Massachusetts. The site is at the bottom of a deep valley along the Deerfield River on the southeast bank of

Sherman Reservoir. The area surrounding the site is mostly wooded with very steep gradients on both sides of the Deerfield River.

Land use near the site is made up of a few farms and some commercial businesses. The centers of Rowe and Monroe have small, mixed clusters of local businesses, municipal buildings, and residences, with homes scattered throughout the area. There is no large industrial activity within five miles of the site; the only industry in the area is the YNPS and several hydroelectric facilities along the Deerfield River.

The nearest highway and railroad are each about five miles south of the site. The closest airport is in North Adams, Massachusetts, about ten miles west of the site.

2.2.58.3 Geography and Seismology

The regional bedrock geology is complex; bedrock age ranges from 100 million to over one billion years old and is comprised of mostly a mosaic of metamorphic and igneous rocks. The youngest deposits are glacial soils, 10 to 12 thousand years old. Most volcanic and sedimentary rock in the area are now metamorphosed.

The site is situated on dense Wisconsinian-aged glacial till. YNPS structures are founded on this till, which ranges from 0 to 140 feet thick across the site. Bedrock under the till is part of the lower Cambrian Hoosac formation and consists of quartz-albite-biotite gneiss and a rusty gneiss in adjacent areas. Underlying these are garnet schist and a layered gneiss with some dolomitic marble, with the latter units belonging to the lower Cambrian or older Cavendish formation. A south-plunging anticline, whose axis is just east of the site, defines local bedrock structure.

Site bedrock is hard, internally welded metamorphic rock, not subject to significant deterioration. Bedrock fracturing is not a prominent structural feature of this bedrock; outcrops exhibit either no joints or minor, discontinuous joint surfaces. Fracture pattern analysis of site vicinity joints, joint sets, and faults show no anomalous trends for fractures. This suggests the absence of any through-going zones of post-metamorphic faulting or shear.

The site is in the Western New England Fold Belt province. It borders the Adirondack Uplift province to the west, the Valley and Ridge province to the southwest, and the New York Recess to the south. The Southeast New England Platform and Merrimack Synclinorium occur to the east and northeast, respectively.

Regional seismic events are very infrequent and do not cause surface faulting. Only two events in the province were greater than Intensity V (MM). These events were 125 miles and 210 miles from the site. The site seismic design level for new seismically qualified installations at the plant is a peak ground acceleration of 0.19 g. The return period is between 10,000 and 100,000 years.

2.2.58.4 Hydrology

The site is located on the east bank of the Deerfield River adjacent to Sherman Reservoir, which serves as the source of the cooling water for the plant. The drainage area upstream of the plant is characterized by a dendritic pattern, is 236 square miles, and has an average annual rainfall of between 40 and 50 inches.

The Deerfield River flow is highly regulated by two large, upstream hydroelectric reservoirs. The average long-term flow near the plant is 738 ft^3/sec.

Bedrock in the region is not a significant source of groundwater nor are there major bedrock aquifers within the site area. The direction of groundwater flow under the site is from the recharge areas on the slopes surrounding the plant toward the Deerfield River.

2.2.58.5 Plant Water Supply

Cooling water supply for the plant comes from the Sherman Reservoir, adjacent to the site. Potable water is supplied by an on site bedrock well and stored in a 10,000 gallon, above ground tank. The well is 437 feet deep and has a yield of 13 gpm.

2.2.58.6 Plant Effluent

The same liquid effluent release path will be used during decommissioning as was used during plant operation. All liquid discharges will be controlled in accordance with the National Pollutant Discharge Elimination System (NPDES) permit. Dilution water will be taken from the Sherman Reservoir via the Service Water System.

Surface water use downstream from the site is mostly for recreation and sport fishing, with limited irrigation. Water supply for the municipalities within five miles downstream of the plant is from private wells. The closest public water supplies are well fields 20 to 25 miles downstream of the plant.

2.2.58.7 Meteorology (General Climate, Severe Weather)

The site lies in the prevailing westerlies, the belt of eastward moving air that is found in middle latitudes. Many storms pass over Massachusetts compared to other parts of the United States. This is a result of extensive air masses originating at higher and lower latitudes. The three major air mass types that affect the site are cold, dry, subartic air from Canada; warm, moist air from the Gulf of Mexico; and cool, damp air from the North Atlantic Ocean.

The hills on either side of the site rise about 1000 feet above ground level within one mile and extend from 12 miles north to 8 miles south southeast of the site. This feature affects the winds. There is, for instance, a high frequency of occurrence of "channel flow" up and down the Deerfield River valley. Also, night time drainage flow down the east side of the river valley occurs frequently.

Normal daily temperatures range from 10°F in January to 80°F in July. Recorded extreme temperatures for the site region are -25°F and 98°F. Thunderstorms occur about 28 days per year; the annual flash density of ground lightning strikes is four flashes per square kilometer.

The site design wind speed (defined as the "fastest-mile" wind speed at 30 feet above the ground with a 100 year return period) is 80 mph. Hail storms occur about two days annually and freezing rain about 12 days per year. The maximum radial ice thickness expected for the region is 1.25 inches. Mean annual snowfall at the site totals 100 inches, with maximum snow depth on the ground totaling about 40 inches.

REFERENCES

2.2-1 EDCR 93-304, Core Baffle Storage Containers.

2.2-2 Safety Classification of Systems Manual, Appendix A, Component, Equipment, and System Classification at YNPS for the Permanently Defueled Condition, Revision 70, June 1, 1991.

2.2-3 Fire Protection Technical Requirements Manual, September 11, 1992.

TABLE 2.2-1

PLANT SYSTEMS NOT INCLUDED IN DECOMMISSIONING PLAN

The following systems and structures are not included in Section 2.2 of the Decommissioning Plan:

Auxiliary Steam System (excluding the Auxiliary Boilers)
Condensate System
Circulating Water System
Turbine Oil System
Generator Hydrogen and Seal Oil System

These systems have no significant interaction with the decommissioning of YNPS. However, the systems will be dismantled during either the safe storage or dismantlement periods.

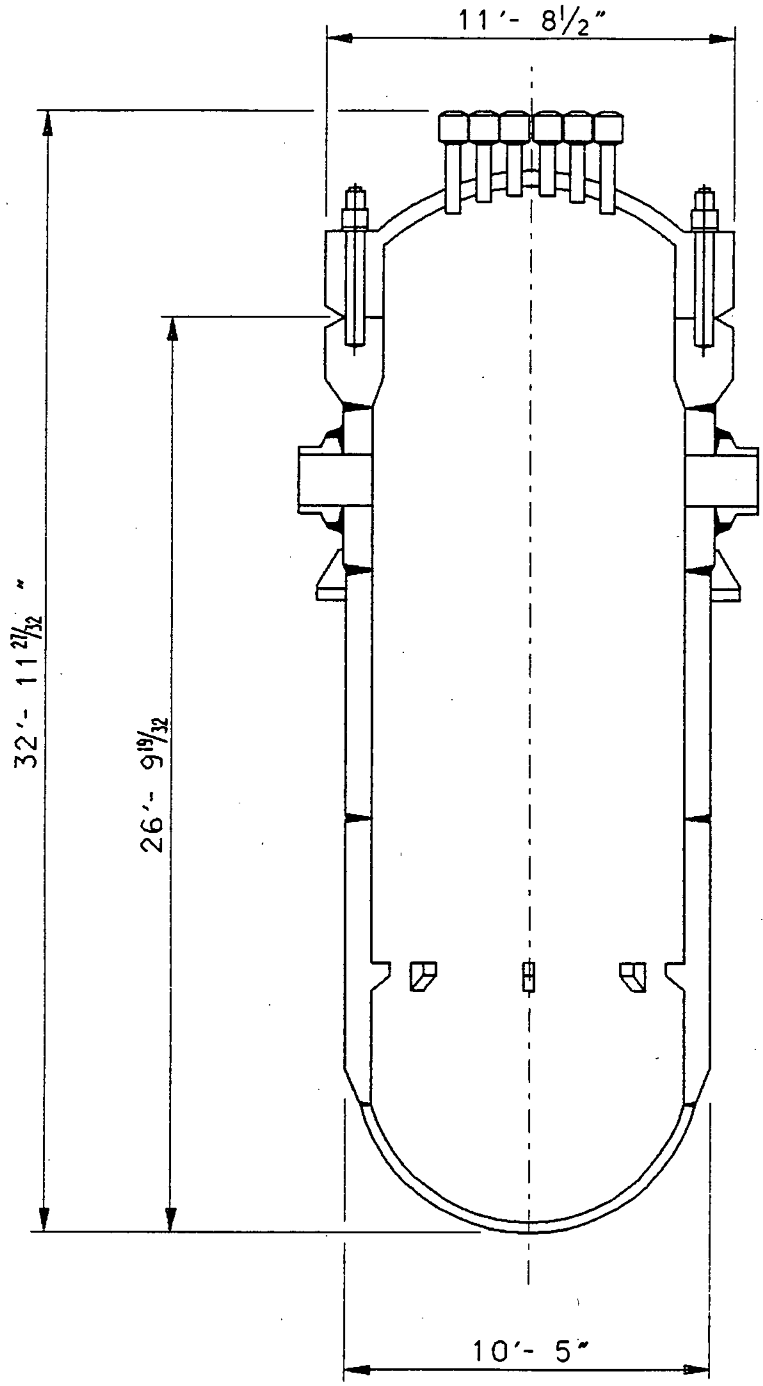
TABLE 2.2-2

REACTOR VESSEL MATERIAL COMPOSITION

Section	Base Material	Clad
Hemispherical Bottom Head	SA-302 Grade B carbon steel plate: 3.875 inch minimum thickness	SA-240 Grade S Type 304 stainless steel: 0.109 inch thickness
Lower Cylindrical Shell	SA-302 Grade B carbon steel plate: 7.875 inch minimum thickness (2 rolled courses)	SA-240 Grade S Type 304 stainless steel: 0.109 inch thickness
Upper Cylindrical Shell	SA-302 Grade B carbon steel plate: 9.625 inch minimum thickness (3 formed segments)	SA-240 Grade S Type 304 stainless steel: 0.250 inch thickness
Inlet / Outlet Nozzles	SA-336 carbon steel modified to SA-302 Grade B carbon steel requirements	Type 308L stainless steel: 0.250 inch thickness, weld deposited
Bolting Flanges	SA-105 Grade 2 carbon steel forging	Type 308L stainless steel: 0.250 inch thickness, weld deposited
Reactor Vessel Head Center Disc	SA-302 Grade B carbon steel plate: 7 inch minimum thickness	SA-240 Grade S Type 304 stainless steel: 0.109 inch thickness
Reactor Vessel Head Adapters (32 penetrations)	Type 304 stainless steel top w/ co-extruded tube of SA-106 Grade C carbon steel tube	Type 304 stainless steel

FIGURE 2.2-1

REACTOR VESSEL



E:/DGN/YRDECOM/FIG221.DGN

2.3 DECOMMISSIONING ACTIVITIES AND PLANNING

2.3.1 Introduction

This section presents a description of decommissioning activities and tasks as well as a schedule for the implementation of the safe storage and dismantlement period activities at YNPS. The information presented in this section reflects initial planning of decommissioning activities. Yankee intends to complete more detailed planning prior to initiating each decommissioning activity. Detailed planning includes engineering design; ALARA planning; and cost, schedule, and resource refinement.

YNPS decommissioning is comprised of four periods: activities prior to decommissioning plan approval, safe storage, dismantlement, and site restoration. The safe storage period will begin following approval of the decommissioning plan and will extend until such time that a firm contract for disposal of low level radioactive waste generated during dismantlement activities can be secured. For planning purposes, safe storage is assumed to extend until 2000. The dismantlement period will extend about two years. After successful completion of a final radiation survey, site restoration activities will begin. Site restoration will be completed in about one year.

2.3.2 Activities Prior to Decommissioning Plan Approval

Plant closure activities, described in Section 1.2.1, were initiated following the decision to permanently cease YNPS power operations in February 1992. These activities will culminate with YNPS being placed in a safe storage condition until final decontamination and dismantlement activities begin. Plant activities have, and will continue to be, implemented through compliance with the existing possession only license and other NRC requirements. It is expected that most preparations for the safe storage period will be completed before formal NRC approval of the decommissioning plan.

Limited component removal activities are also being conducted at YNPS, consistent with NRC requirements and the availability of radioactive materials processing or disposal options. A project (Component Removal Project) was initiated in 1993 to remove and dispose of the steam generators, pressurizer, and reactor vessel internals before June 1994. These components were all removed from the site with the exception of certain high level radioactive material from the reactor vessel internals which were stored with the fuel. Appendix A presents a description of the Component Removal Project.

Preparation for safe storage includes the following major activities:

- Assess the functional requirements for all plant systems, structures, and components for all phases of decommissioning. This activity has been completed and is documented in Section 2.2. Plant procedures were modified as appropriate to reflect the new operating requirements.

- Deactivate, drain, and either lay-up or isolate plant systems, structures, and components. The functional evaluations presented in Section 2.2 were used to assess the future operating requirements for plant systems, structures, and components. A lay-up program was developed and implemented to preserve, as appropriate, plant systems, structures, and components for future use.

  Plant Procedure AP-1902, "Plant Lay-up Program," presents the lay-up program. The procedure guides the development and implementation of lay-up packages and establishes a surveillance program to ensure the long term integrity of the lay-up packages. The lay-up program also establishes an equipment tagging program that identifies all plant equipment that is required to support plant operations. All equipment that will remain in service has been tagged by the Operations Department. Equipment that has not been tagged will either be incorporated in the lay-up program or be drained and isolated until they are dismantled.

- Define and modify as necessary technical specifications and other regulatory requirements which are applicable to the safe storage period. Section 1.2.1.1 presents changes made to the YNPS licensing basis during the plant closure period. These changes include a technical specification amendment creating permanently defueled technical specifications. No additional changes are envisioned to support decommissioning activities. Section 6 presents the decommissioning technical specifications.

- Perform an assessment of the current radiological condition of YNPS. Section 3.1 presents an assessment of the radiological condition of YNPS. This assessment was used to develop the decommissioning plan.

- Design and implement a site radiological characterization program. Section 3.1 presents a description of the continuing site radiological characterization program that will be implemented.

- Develop and implement a plan for long term storage of spent fuel. An engineering task force developed a plan for the disposition of spent nuclear fuel stored on site (Reference 2.3-1). The following spent fuel storage strategy resulted from the study:

  - Continue operation of the Spent Fuel Pit and implement any safe, but economically attractive improvements

- Urge the Department of Energy to accelerate acceptance of spent fuel or to accept financial responsibility for on-site fuel storage

- Continue evaluations of wet versus dry storage options to reflect Yankee and industry progress

- Initiate preliminary design activities for a dry storage facility

For planning purposes, YAEC's current decommissioning cost estimate assumes storage of fuel in the Spent Fuel Pit until 1996, at which time it is transferred to an on-site dry storage facility. The dry storage facility remains until 2018 when the last fuel assembly is transferred to the Department of Energy. The dry storage facility will be decommissioned after the fuel is permanently removed.

Yankee re-evaluated the spent fuel storage options to determine the impact of storing fuel in the Spent Fuel Pit during the dismantlement period (Reference 2.3-2). The evaluation indicated that a significant amount of decommissioning activities could be completed without interfering with fuel stored in the Spent Fuel Pit. The decommissioning plan allows for these activities.

Section 3.3.1 presents a more detailed description of YAEC's plans for spent fuel disposition.

- Design and implement a security plan for all phases of decommissioning. Yankee completed a comprehensive review of the security requirements for a permanently defueled condition with spent fuel stored in the Spent Fuel Pit. This evaluation provided the basis for the YNPS Defueled Security Plan (Section 8).

  As a result of the security analysis, the protected area boundary was reduced to the Spent Fuel Pit Building outside wall. The area within the original protected area boundary was reclassified as an industrial security area. No changes to the Defueled Security Plan are envisioned to meet plant needs during decommissioning. The plan will remain in effect until all spent fuel is removed permanently from the Spent Fuel Pit. At that time the security plan will be modified as necessary.

- Develop and implement a Decommissioning Quality Assurance Plan. Section 7 presents a description of the Decommissioning Quality Assurance Plan. The plan will remain in effect during all phases of decommissioning.

- Process, package, and ship liquid and solid radiological waste generated during plant closure activities. This activity was completed to the maximum extent possible as systems were deactivated and drained. The Waste Disposal System will remain operational during the safe storage period. Since access to a low level radioactive waste disposal facility may be limited, processed waste may be stored temporarily in an on-site storage facility. Yankee has completed a safety evaluation to justify limited on-site storage of low level radioactive waste (Reference 2.3-3).

- Develop and implement a decommissioning organizational structure. Section 2.4 presents a description of the organizational structure that will be implemented during decommissioning. The organizational structure for the period before decommissioning plan approval and the safe storage period has been implemented.

- Develop and implement decommissioning radiation protection and ALARA programs. Section 3.2 presents a description of the Radiation Protection Program and ALARA Program that will be implemented during decommissioning. These programs are based on the programs successfully used during YNPS operations.

2.3.3 Safe Storage Period

The safe storage period begins after the tasks and activities identified in Section 2.3.2 have been implemented and the decommissioning plan has been approved by the NRC. During the safe storage period, plant operations will include those needed to monitor the radiological status of the facility, to maintain systems in a dormant condition, and to support operation of the spent fuel storage facility. Limited component removal activities may also occur during the safe storage period consistent with regulatory requirements and the availability of radioactive materials processing or disposal options.

The Radiation Protection Program implemented during the safe storage period is intended to support clean-up and decontamination operations. The radiation protection staff will be augmented with operating and security personnel who will perform full-time on-site monitoring. The objective of the monitoring is to control access to the facility, preventing inadvertent exposure to radiation or the spread of contamination. Yankee will continue both on-site and off-site radiation monitoring programs during the safe storage period as described in Section 3.2.

Yankee will maintain the systems and components required to support decommissioning and spent fuel storage in accordance with the possession only license and other administrative and implementing procedures. The maintenance program in effect during the safe storage period consists of corrective maintenance, preventive maintenance, and surveillances. Yankee will continue preventative maintenance practices for systems and components required by technical specifications. Preventative maintenance will continue for other systems at a level commensurate with the functional requirements described in Section 2.2. The maintenance program will be changed as necessary to reflect implementation of the Maintenance Rule, per 10 CFR 50.65. If changes to decommissioning and spent fuel storage maintenance activities are necessary, the changes will be implemented prior to the rule becoming effective. The maintenance program is controlled by administrative procedures and maintenance activities are accomplished by procedures and task specific work plans. These procedures are similar to procedures used during plant operation.

Yankee will continue to seek potential low level radioactive waste disposal sites for YNPS decommissioning waste. If a site is identified that can support the decommissioning waste, Yankee will proceed with earlier dismantlement. Limited access to low level radioactive waste disposal facilities may also occur during the safe storage period. Yankee intends to use these opportunities to remove components and structures consistent with the guidance presented in Section 1.4.

The plant and corporate organizational structure initiated before decommissioning plan approval will remain in effect during the safe storage period as described in Section 2.4.

During the safe storage period, Yankee will decide whether or not long-term on-site fuel storage is necessary, based on off-site alternatives. If availability of an off-site alternative appears favorable, Yankee will store fuel in the Spent Fuel Pit until the fuel is transferred off of the site. If long-term on-site fuel storage appears necessary, Yankee will initiate a project to transfer fuel to a dry storage facility. Yankee will incorporate necessary restrictions and modifications to decommissioning activities to ensure safe fuel storage at YNPS. For planning purposes, Yankee has incorporated the cost of a dry storage facility that would be loaded in 1996 into the decommissioning cost estimate. This reflects the current uncertainty regarding spent fuel disposal site availability.

2.3.4 Decontamination and Dismantlement Plan: General Information

2.3.4.1 Plan Overview

This section presents a general description of the decontamination and dismantlement activities that are necessary to decommission YNPS. The information presented in this section provides sufficient detail to address the adequacy of the programs and procedures, ensuring safe and economic decommissioning of YNPS. The information in this section will be incorporated into the more detailed planning that will be completed prior to initiating each decommissioning activity. Detailed planning includes engineering design, ALARA planning, and cost, schedule, and resource refinement. Section 2.3.5 presents specific information for YNPS systems, structures, and components.

The description of decontamination and dismantlement options includes the words "should" and "must" to describe alternatives. The word "should" implies that the alternative is preferred, however, alternatives may be available that are equally acceptable. The word "must" implies that the alternative is based on a programmatic, regulatory, or safety analysis requirement. If another alternative is chosen, the corresponding requirements must be re-evaluated to ensure that the original intent is not adversely affected.

Before the start of the dismantlement period, the decommissioning administrative and engineering organization will be mobilized. During the first several months the following activities will occur:

- Initiation of detailed project planning
- Preparation of engineering specifications and procedures
- Procurement of special equipment needed to support decommissioning
- Negotiation of service contracts required for decommissioning activities
- Reactivation and return to service of systems required for decommissioning

The engineering and preparation phase is followed by the initiation of plant dismantlement activities. The contaminated systems will be removed, packaged, and either shipped to an off-site processing facility or shipped directly to a low level radioactive waste disposal facility. Decontamination of plant structures will be completed concurrently with the equipment and system removal process. Structure decontamination will include a variety of techniques ranging from high pressure water washing to removal of concrete to allow release of the structures. Contaminated structural material will be packaged and either shipped to a processing facility for decontamination or shipped directly to a low level radioactive waste disposal facility.

Following the removal of contaminated systems, structures, and components, a comprehensive final radiation survey will be conducted. The survey will verify that radioactivity has been reduced to sufficiently low levels allowing unrestricted release of the site. Successful completion of the final survey will be demonstrated through a verification survey completed by an independent contractor selected by the NRC.

2.3.4.2 Detailed Planning and Engineering Activities

YAEC intends to be the prime contractor (Decommissioning Operations Contractor) responsible for YNPS decommissioning. In this position Yankee will have direct control and oversight over all decommissioning activities. This role is similar to that taken by YAEC during the 31 year operation of YNPS. In that role YAEC provided operational, technical, licensing, and project management for all plant systems, structures, and components for all phases of decommissioning.

Detailed project implementation plans will be developed to support decontamination and dismantlement activities before these activities are initiated. The plans will be used as a project management tool to support detailed engineering activities and ALARA Program implementation, to estimate decommissioning labor requirements, and to manage decommissioning cost and schedule.

Decommissioning work packages will be completed for all decommissioning activities. The work packages will be developed using similar controls to those used to support YNPS operations:

- Engineering Specifications - Engineering specifications will be developed for all system, structure, and component removal activities. A procedure will be developed to provide guidance for the preparation of engineering specifications for decommissioning activities. The process used to develop the packages will be similar to the YAEC Engineering Design Change Request process (Reference 2.3-4). Use of this process ensures that all significant decommissioning activities receive appropriate safety and technical reviews. The engineering specification will be reviewed by the Plant Operation Review Committee prior to implementation.

- Decommissioning Procedures - All decommissioning activities will be completed using Plant Operations Review Committee reviewed procedures. The procedures will be developed and controlled in accordance with the Decommissioning Quality Assurance Plan (Section 7) and Plant Procedure AP-0001, "Plant Procedures." This procedure presents administrative controls for the format, content, review, and approval of all procedures used at the plant. Applicable elements of the Radiation Protection Program and Occupational Safety Program will be integrated into the procedures (e.g., ALARA Program, Radioactive Waste Minimization Program).

2.3.4.3 General Decontamination and Dismantlement Considerations

The following are general decontamination and dismantlement considerations that will be incorporated into the decommissioning work packages for the systems, components, and structures at YNPS. Specific and unique considerations are presented in Section 2.3.5.

- Caution must be used when working in areas which contain systems or structures that support spent fuel cooling and storage (e.g., Spent Fuel Pit Building, Primary Auxiliary Building). The systems and structures cannot be affected by removal activities. Work packages must include specific steps either to physically protect the systems and structures or to establish safe load paths and protective zones around the systems and structures (Section 3.3.1.3).

- Dismantlement activity planning must consider the impact of seismic events on components that are affected by removal activities. These components should be evaluated and physically supported, as appropriate, to limit the off-site dose resulting from a release of radioactivity to within the accident analysis limits (Section 3.4). The impact of seismic events must also be evaluated during planning of dismantlement activities in proximity of the Spent Fuel Pit. The purpose of this evaluation is to ensure that partial dismantling of equipment and structures does not result in a configuration that could fail during a seismic event, subsequently collapsing onto or into the Spent Fuel Pit or Fuel Transfer Chute.

- Hazardous materials and wastes must be processed in accordance with the YNPS Hazardous Waste Management Program (Section 3.6):

  - Asbestos containing material (e.g. insulation) must be removed and processed in accordance with the YNPS Asbestos Control Program (Section 3.6.4.6). This activity should be scheduled prior to initiating equipment removal in an area.

  - Structures and components containing lead based coatings must be processed in accordance with the guidance presented in Section 3.6.4.7.

  - Systems and components which contained or were immersed in chromated solutions must be free of chromate residue before shipment off site. The systems and components must be rinsed, if necessary, to ensure that loose surface chromate residue is removed.

- The capability to isolate the Vapor Container from the environment, to mitigate the consequences of a significant radioactive release, must be maintained during decontamination and dismantlement activities in the Vapor Container. Vapor Container isolation is the closure of all penetrations and openings to restrict transport of airborne radioactivity from the Vapor Container atmosphere to the environment. Pressure retention capability is not necessary.

  If this capability cannot be established, activities involving significantly contaminated or activated components must be suspended. This consideration should not preclude the removal of penetrations and attachments to the Vapor Container shell providing that the opening is closed in a timely manner.

- Decommissioning activities that use liquids must ensure that contaminated liquids will be processed by the liquid waste processing system. Additionally, existing or supplemental barriers must be used to ensure that inadvertent spills from these activities are contained within the liquid waste processing system.

- The following considerations must be incorporated into tank and vessel sludge removal activities:

    - The method used must ensure that any liquid inadvertently discharged from the system is contained in the plant liquid waste processing system.

    - Sludge removed from the system must be stabilized prior to shipment.

    - Waste water must be processed and analyzed before discharging from the facility.

    - The use of a high pressure water rinse should be used before dismantlement if necessary to reduce internal contamination levels.

- Radioactive particulate emissions must be filtered and monitored to the maximum extent practicable. The following must be implemented:

    - The VC Ventilation and Purge System must be maintained in operation during decontamination and dismantlement activities in the Vapor Container.

    - The Ventilation System must be maintained in operation during decontamination and dismantlement activities in the Primary Auxiliary Building cubicle area, Waste Disposal Building, and Spent Fuel Pit Building.

    - The PAB and Diesel Generator Building Roof Fans must be secured and not operated during decontamination and dismantlement activities in the PAB and Diesel Generator Building.

    - Local HEPA filtration systems must be used when activities could result in the release of significant radioactive particulates. The local HEPA filtration systems should exhaust to areas served by the Ventilation System when used outside of the Vapor Container to monitor particulate releases.

- Electrical and pneumatic services must be isolated from the systems, components, and structures prior to dismantlement.

The following should be implemented:

- Pumps, fans, heaters, motor operated valves, motor operated dampers and instrumentation power sources should be isolated and disconnected from station electrical and control systems at the motor control centers, supply breakers, fuse blocks, and at the equipment.

- Pneumatically operated components and instrumentation should be isolated from the Compressed Air System at the root and equipment isolation valves.

- Openings in components must be enclosed with a protective cover to confine internal contamination.

- Explosive methods must not be used during the YNPS decommissioning to remove contamination.

- Before removing contaminated systems, structures, and components with significant external contamination, they should be wiped down to remove external contamination or painted with a coating to stabilize external contamination.

- Contaminated piping and tubing should be dismantled as follows:

    - Large bore piping (greater than 2 1/2 inch diameter) should be cut into manageable lengths. Significantly contaminated systems (i.e., Main Coolant System, Bleed Line in Vapor Containment, Feed and Bleed Heat Exchanger connections) must be cut using mechanical methods to minimize the generation of airborne contamination.

    - Small bore piping (less than 2 1/2 inch diameter) and tubing should be cut using an appropriate method based on the radiological conditions.

    - Remote cutting systems should be used as appropriate to maintain worker exposure as low as is reasonably achievable.

    - Open pipe ends must be enclosed with a protective cover to confine contamination to the inside of the pipe.

    - Piping penetrations should be cut as close as practicable to the Vapor Container shell. The opening in the Vapor Container shell should be covered once the piping is removed.

  - Underground piping must be visually examined after it is excavated to ensure that it is physically sound prior to cutting and removal.

- Contaminated supports should be removed in conjunction with equipment removal activities.

- Systems and components should be removed from areas and buildings prior to the commencement of area structural decontamination activities. The block shield walls in the Primary Auxiliary and Waste Disposal Buildings should be removed as necessary to permit removal of systems and components.

- Embedded contaminated piping, conduit, ducts, plates, channels, anchors, sumps and sleeves should be removed or decontaminated during area and building structural decontamination activities.

- Centralized processing and cutting stations should be considered to facilitate packaging of components for shipment to an off-site processing facility or a low level radioactive waste disposal facility.

2.3.4.4 <u>Decontamination and Dismantlement Process</u>

The decontamination and dismantlement of contaminated systems, structures and components may be accomplished by either decontamination in place, removal and decontamination, or removal and disposal. A combination of these methods may be utilized to reduce contamination levels and reduce worker radiation exposures.

In general, contaminated and potentially contaminated systems, structures, and components will be removed as follows:

- Radiological characterization survey data will be used to identify the systems, structures, and components to be decontaminated and dismantled. The type of contamination associated with the systems and components is presented in Table 2.3-1.

- Detailed decommissioning work packages will be developed, reviewed and approved in accordance with project and plant programs and procedures (Section 2.3.4.2)

- Plant tag-out procedures will be used to de-energize electrical and control equipment, isolate and drain fluid systems, and isolate and depressure pneumatic systems. Radiation Protection procedures will be used to ensure radiological requirements for control of contamination, worker protection, and ALARA program are satisfied. Occupational Safety standards will be observed.

- All components will be identified prior to removal. The components will then be removed using the techniques and methods as specified in the decommissioning work packages. All openings in components will be covered and sealed to minimize the spread of contamination. The components may be moved to a processing area for volume reduction and packaging into containers for shipment to a processing facility for decontamination or low level radioactive waste disposal facility.

- Contaminated concrete and structural steel components will be decontaminated or removed when all contaminated and uncontaminated systems and equipment have been removed from the area or building. The contaminated concrete will then be removed and packaged into containers for shipment to a low level radioactive waste disposal facility. Contaminated structural steel components may be moved to a processing area for volume reduction and packaging into containers for shipment to a processing facility for decontamination or low level radioactive waste disposal facility.

- Buried contaminated components (e.g., piping, drains, conduit) will be excavated. After excavation, the components will be examined to ensure that they are physically sound prior to cutting and removal. Most buried contaminated piping is located in steel conduits (i.e., pipes enclosed in pipes). Contamination controls will be modified as necessary if the components are significantly degraded.

- A final termination survey will be performed to verify removal of contamination to below release levels.

2.3.4.5 Decontamination Methods

Contaminated systems and components will be removed and sent to an off-site processing facility or to a low level radioactive waste disposal facility. On-site decontamination of systems and components will be limited to activities needed to maintain personnel exposure as low as is reasonably achievable, to expedite equipment removal, and to control the spread of contamination.

Large scale chemical decontamination of internally contaminated systems will not be used during YNPS decommissioning, since major dismantlement may be delayed until 2000. However, selective chemical decontamination may be used in localized high contamination areas to reduce radiation dose rates. Radioactive decay of cobalt-60 diminishes the benefit of chemical decontamination when dismantlement activities are delayed from the initial shutdown date (cobalt-60 is the primary contributor to radiation dose rate at YNPS). Delaying YNPS dismantlement activities until 1996 reduces post-shutdown radiation dose rates by about 50% (70% reduction by 2000). For comparison, large scale chemical decontamination soon after shutdown could reduce radiation dose rates by about 90%. The principles of the YNPS ALARA Program (Section 3.2.5) will be implemented during detailed planning of decommissioning activities to determine whether chemical decontamination is justified.

Application of coatings and hand wiping will be the preferred methods for stabilizing or removing loose surface contamination. If other methods are utilized, airborne contamination control and waste processing systems will be used as necessary to control and monitor any releases of contamination. If structural surfaces are washed to remove contamination, barriers will be established to ensure that wash water is collected and processed in the plant liquid waste processing system.

Contaminated and activated concrete as well as other contaminated materials will be removed and sent to a low level radioactive waste disposal facility. Removal of concrete should be performed using a method which controls the removal depth to minimize the waste volume produced (e.g., scabbling, scarifying). Vacuum removal of the dust and debris with HEPA filtration of the effluent should be used to minimize the need for additional respiratory protection control measures. Explosive methods for the removal of contaminated concrete or other structural materials will not be used.

A summary of currently available methods for decontamination of plant equipment and structures is presented in Table 2.3-2. The methods presented in this section are the most practicable and widely utilized at the time that this plan was generated. However, new decontamination technologies developed prior to the commencement of actual decommissioning activities will be considered and used if appropriate.

2.3.4.6 Dismantlement Methods

Dismantlement methods can be divided into two basic types:

- Mechanical Methods - Mechanical methods machine the surfaces of the material that is being cut. The methods typically are capable of cutting remotely without generating significant amounts of airborne contamination. This attribute makes this method attractive for most of the contaminated piping, equipment, and components that will be removed at YNPS. The outside diameter machining method is best suited for cutting large bore contaminated piping. Smaller bore contaminated piping, tubing, and supports can be cut using any of the mechanical methods (e.g., band saws, reciprocating saws).

- Thermal Methods - Thermal methods melt or vaporize the surfaces of material that is being cut. The cutting debris is transported from the cut region with a gas jet or water spray. Although thermal methods are significantly quicker than mechanical methods, they have high power requirements and generate airborne contamination when used on contaminated systems in air. Generation of airborne contamination can be easily controlled when the method is used underwater. Thermal methods are suitable for segmenting large vessels in areas that can easily be sealed, filtered, or maintained underwater. The method is also suitable for use at a cutting station with air filtration. Thermal methods are appropriate for removing structural steel if it has been decontaminated or if a local contamination envelope with HEPA filtration is established. Adequate lead paint removal controls must also be implemented.

A summary of currently available methods for cutting plant equipment and structures is presented in Tables 2.3-3 and 2.3-4. The methods presented in this section are the most practicable and widely utilized at the time that this plan was generated. However, new dismantlement technologies developed prior to the commencement of actual decommissioning activities will be considered and used if appropriate.

2.3.4.7 Materials Cutting Station

A centralized processing and cutting station may be implemented to facilitate packaging of components for shipment to an off-site processing facility or a low level radioactive waste disposal facility. One possible location for the station is the east side of the PCA Warehouse, which could be sealed to prevent the spread of contamination and to provide airborne containment. The processing and cutting station atmospheric control must use a HEPA filtration system which exhausts to the Ventilation System to ensure that gaseous releases are monitored.

2.3.5 Decontamination and Dismantlement Plan: Systems, Structures, and Components

2.3.5.1 Overview

This section presents a specific description of the decontamination and dismantlement activities that are necessary to decommission each potentially contaminated system, structure, and component at YNPS. The information presented in this section reflects initial planning of decommissioning activities and will be incorporated into the more detailed planning that will be completed prior to initiating any decommissioning activity. Detailed planning includes engineering design, ALARA planning, and cost, schedule, and resource refinement.

2.3.5.2 Reactor Vessel (Section 2.2.2)

The Reactor Vessel is located inside the Vapor Container. The Reactor Vessel is not required in a permanently defueled condition or to support decommissioning activities. The Reactor Vessel internal components were removed during the Component Removal Project.

Neutron irradiation from the reactor core generated activation products in both the stainless steel vessel liner and the carbon steel vessel shell (Section 3.1.3.4). Table 3.1-9 presents an estimate of contact and stand-off dose rates resulting from the neutron activation. Based on these estimates, the radionuclide concentrations do not exceed 10 CFR Part 61 Class A limits. Although the radionuclide content exceeds the 10 CFR Part 71 Type A package limits, the specific activity is less than the 10 CFR Part 71 low specific activity material limits. This allows the Reactor Vessel to be qualified for normal conditions of transport and to be exempted from other accident design and qualification requirements defined in 10 CFR Part 71. The radionuclide content estimates will be verified with a radiation survey of the Reactor Vessel after completion of the Component Removal Project. Detailed classification evaluations will be completed as a part of detailed planning of the Reactor Vessel removal activity.

An engineering evaluation was performed to investigate potential Reactor Vessel removal alternatives (Reference 2.3-5). The evaluation identified two technically feasible alternatives: intact vessel removal and segmented vessel removal. The intact vessel removal alternative proposes shipment of the vessel to a low level radioactive waste disposal facility as one piece inside an exclusive use cask. The segmented vessel removal option proposes shipment of vessel sections to a low level radioactive waste disposal facility inside standard shipping casks.

The Reactor Vessel upper head will be decontaminated, packaged, and shipped to a low level radioactive waste disposal facility. The packaging activities include sealing of all openings with closure plates.

The engineering evaluation concluded that neither option was significantly preferred, compared to the alternative option. Final alternative selection will be based on an evaluation of parameters associated with project planning and execution: ease of execution, personnel exposure, schedule impact, disposal facility availability, and cost. Final alternative selection will be made as a part of detailed engineering and planning.

This section summarizes both alternatives including the following information: description, issues, implementation assumptions, and sequence of events. For planning purposes, the decommissioning cost estimate assumes that the Reactor Vessel is removed using the segmented vessel removal alternative.

2.3.5.2.1 Intact Reactor Vessel Removal Alternative

The intact vessel removal alternative proposes shipment of the vessel to a low level radioactive waste disposal facility as one piece inside an exclusive use cask.

Intact removal requires the construction and certification of an exclusive use shipping cask for transporting the Reactor Vessel and associated insulation. The cask will be designed to meet the requirements of a 10 CFR Part 71 Type A shipping cask based on a low specific activity exemption from Type B requirements. The activation analysis of the Reactor Vessel indicates that the specific activity is less than the low specific activity material limits.

Intact removal also requires movement of the Reactor Vessel inside the Vapor Container with the Polar Crane. This will require a one-time upgrade of the Polar Crane to allow a lift that exceeds the name plate capacity by about 13%. The Reactor Vessel removal engineering evaluation presents a preliminary assessment indicating that an upgrade is feasible if several administrative and inspection prerequisites are met.

High dose rates from the activated surfaces of the Reactor Vessel require the use of special shielding and handling methods to ensure that personnel exposure is maintained as low as is reasonably achievable. A temporary shield will be placed on the Reactor Vessel flange when the piping is cut and the vessel water level is lowered. The flange will have an air filtration port which will be connected to a HEPA filter to control the release of airborne radioactivity. The Reactor Vessel will be moved inside the Vapor Container without additional shielding, therefore, Charging Floor access will be restricted to limit personnel exposure. The Polar Crane controls will be modified to allow remote crane operation and cameras will be installed to allow remote spotting of the load as it is moved into the shipping cask.

The Reactor Vessel is insulated with calcium silicate that contains asbestos. The calcium silicate is covered by a thin stainless steel sheet. The only insulation that may be removed from the vessel is located between the vessel flange and the nozzles. Asbestos insulation may also be disturbed during Main Coolant System pipe cutting operations. Any activities associated with asbestos containing materials will be completed in accordance with the Asbestos Control Program (Section 3.6.4.6).

Implementation of this alternative is based on the following assumptions:

- A low level waste disposal site is available with the capability to handle large components.

- The reactor vessel can be classified as low specific activity material and packaged in a shipping cask that is exempted from 10 CFR Part 71 Type B packaging requirements (10 CFR 71.52).

- An acceptable transportation route is available. This includes upgrading the Sherman Dam bridge and accessing the railroad at the Hoosac Tunnel.

- The Fuel Transfer Chute is isolated from the Spent Fuel Pit and either cut above the lower lock valve or structurally reinforced before loading the Reactor Vessel into the shipping cask.

The following is a summary of the sequence of events for removal of the Reactor Vessel using the intact vessel removal alternative:

- Prepare the Vapor Container for Reactor Vessel Removal - Several Vapor Container modifications are needed to support removal of the Reactor Vessel as a single component:

  - Enlarge the equipment hatch opening to allow movement of the shipping cask under the hatch

  - Remove a section of the wall between the Shield Tank Cavity and the equipment hatch to allow transit of the Reactor Vessel

  - Remove the Vapor Container struts and bracing that interfere with shipping cask movement

  - Modify the Vapor Container Polar Crane for remote operation

- Prepare Reactor Vessel For Removal - Several activities must be completed in preparation for the Reactor Vessel removal:

  - Install a shield plate on the Reactor Vessel flange to reduce dose rate and to control airborne radioactivity

  - Cut the Main Coolant System piping from the Reactor Vessel at the nozzles using an internally mounted pipe cutter

  - Remove water from within the Reactor Vessel and wash the internal surfaces to remove loose activity

- Apply a fixative within the Reactor Vessel to stabilize the remaining internal surface contamination

- Lift Reactor Vessel in Vapor Container - The Reactor Vessel will be lifted using the Polar Crane. The Polar Crane will be modified for remote operation by installing pendant controls and spotting cameras. The Reactor Vessel nozzles will be covered after the vessel has been lifted several feet to allow access. The higher radioactivity regions will be below the Shield Tank Cavity floor level during nozzle covering activities. The external surface of the Reactor Vessel will be coated with a fixative, using a remotely controlled applicator, as it is lifted from the Reactor Vessel cavity to preclude spread of contamination from the vessel surface.

- Load Reactor Vessel in Cask - The Reactor Vessel will be moved over the equipment hatch and lowered into the shipping cask. The cask cover will be installed after the Reactor Vessel is seated in the cask. The cask will then be moved from under the Vapor Container, lowered to a horizontal position, and prepared for transportation. The transporter limits presented in Section 3.4.9.7 must be implemented to limit operation near the Spent Fuel Pit.

- Transport Cask to Disposal Facility - The cask will be transported to the railway line at the Hoosac Tunnel and placed on a rail car. The cask will be transported to the disposal facility using the railway.

2.3.5.2.2 Segmented Reactor Vessel Removal Alternative

The segmented vessel removal option proposes shipment of vessel sections to a low level radioactive waste disposal facility inside standard shipping casks.

The Reactor Vessel will be segmented using a combination of cutting methods. A track mounted milling machine, attached to the inner wall of the vessel, will be used to make circumferential cuts in the vessel shell. These cuts will be about 90% through wall cuts. Metal Disintegration Machining (MDM) will be used to complete the circumferential cuts and to make axial cuts in the vessel shell. These methods were chosen based on their ability to cut relatively thick components without generating significant amounts of airborne radioactivity. HEPA filtration will be used to remove any particulates that are generated during cutting operations.

High dose rates from the activated surfaces of the Reactor Vessel require the use of special shielding and handling methods to ensure that personnel exposure is maintained as low as is reasonably achievable. A temporary shield will be placed on the Reactor Vessel flange when the piping is cut and the vessel water level is lowered. The flange will be have an air filtration port which will be connected to a HEPA filter to control the release of airborne radioactivity.

Water in the Reactor Vessel will be purified during cutting operations to minimize activity levels in the water and to maintain water clarity. A two train water clean-up system will be installed in opposite Main Coolant System loop areas. The two train system ensures continuous water purification if one train is out of service or moved to facilitate cutting operations. The water clean-up system location provides shielding for the ion exchange capsules.

The Main Coolant System piping penetrations through the bioshield wall will be sealed to prevent the transport of airborne contamination or contaminated water into the Main Coolant System loop areas. In addition, the water level in the Reactor Vessel and the Neutron Shield Tank will be monitored closely to ensure that it does not overflow into the Neutron Shield Tank Cavity.

The Neutron Shield Tank inner wall will be used to contain water during underwater cutting operations. The drains on the bottom of the tank must be plugged to ensure that tank integrity is maintained. Grout or concrete may be poured into the bottom portion of the Neutron Shield Tank to prevent deformation of the inner wall and to transfer loads from the water and the vessel segments to the Neutron Shield Tank outer wall.

The Reactor Vessel is insulated with calcium silicate that contains asbestos. The calcium silicate is covered by a thin stainless steel sheet. Most of the Reactor Vessel insulation will be removed underwater. Asbestos insulation may also be disturbed during Main Coolant System pipe cutting operations. Any activities associated with asbestos containing materials will be completed in accordance with the Asbestos Control Program (Section 3.6.4.6).

Implementation of this alternative is based on the following assumptions:

- Existing field machining technology can be modified and demonstrated to operate efficiently and reliably in the YNPS Reactor Vessel.

- Asbestos contained in the Reactor Vessel insulation can be processed efficiently without imposition of requirements incompatible with segmenting operations.

The following is a summary of the sequence of events for removal of the Reactor Vessel using the segmented vessel removal alternative:

- <u>Prepare the Reactor Vessel for Segmentation</u> - Several activities must be completed in preparation for Reactor Vessel segmentation:

- Install a shield plate on the Reactor Vessel flange to reduce dose rate and to control airborne radioactivity

- Cut the Main Coolant System piping from the Reactor Vessel at the nozzles using an internal pipe lathe

- Install a two train water clean-up system in opposite Main Coolant System loop areas to purify water during vessel segmentation

- Install a decontamination system on the Shield Tank Cavity floor to clean cutting equipment and underwater tools

- Seal Main Coolant System piping penetrations in the bioshield wall

- Implement appropriate asbestos controls for insulation removal

- Plug the Neutron Shield Tank drain lines to allow filling of the annulus between the Reactor Vessel and the Neutron Shield Tank with water

- Segment the Reactor Vessel - The Reactor Vessel will be segmented into seven rings (i.e., flange ring, nozzle ring, four core area rings, bottom head) with six 90% through wall cuts using a track mounted milling machine. The bottom head and the flange ring will not be segmented axially. The nozzle ring will be segmented into eight axial regions and the four core area rings will be segmented into seven axial regions. MDM will be used to complete the circumferential cuts and to make the axial cuts. The first segment that will be separated from the vessel is the lower head, which will be placed on the Neutron Shield Tank inner wall. Subsequent segments will be removed as they are separated from the vessel. Asbestos insulation will be cut with a rotating disk and removed as it becomes exposed.

- Package Segmented Components - Segments will be removed from the Reactor Vessel in a shielded enclosure and placed in a liner on the Shield Tank Cavity floor. The liner will be shielded to reduce the Shield Tank Cavity area dose rate. After the liner is filled, it will be lifted into a shipping cask and prepared for shipment to a disposal facility. The reactor flange ring will be removed as one piece, decontaminated to remove loose surface contamination, and coated with a fixative to stabilize any remaining activity. The reactor flange ring will be packaged for disposal as a single piece. The bottom head will be lifted and the asbestos will be stripped before lifting it out of the water. The bottom head will be processed using the same methods as the reactor flange ring. Reactor Vessel casks must be transported separately on-site until they are prepared for final shipment.

- Cleanup Neutron Shield Tank - After the Reactor Vessel has been removed, the Neutron Shield Tank inner wall will be cleaned to remove loose asbestos and cutting debris. The water will be retained in the Neutron Shield Tank in preparation for removal of the Neutron Shield Tank inner wall.

2.3.5.3 Main Coolant System (Section 2.2.4)

The Main Coolant System and components are located in the Vapor Container Loop areas. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact operations of the Spent Fuel Pit.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Main Coolant System should be isolated at the connections to the Charging and Volume Control, Shutdown Cooling, Component Cooling, Emergency Core Cooling, Primary Plant Sample, and Primary Plant Vent and Drain systems.

- Main Coolant System piping should be cut using a mechanical method. The piping should be cut and capped as close to the Biological Shield Wall as practicable. The Main Coolant System piping between the Reactor Vessel and the Biological Shield Wall will be cut and removed after the nozzle cuts have been made as part of the Reactor Vessel removal activity.

- The motor operators should be removed from the isolation valves prior to final packaging. When the operators have been removed, the isolation valves should be packaged intact without further disassembly.

- Each pump motor and casing and container of Main Coolant System piping and valves should be transported on-site individually due to their weight and to minimize the release of airborne radioactivity in the unlikely event that a handling accident occurs.

2.3.5.4 Pressure Control and Relief System (Section 2.2.5)

The Pressure Control and Relief System and components are located in the Vapor Container, Lower Pipe Chase and the PAB Valve Room. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific system consideration:

- The Pressure Control and Relief System should be isolated at the connections to the Charging and Volume Control, Waste Disposal, and Primary Plant Vent and Drain systems.

2.3.5.5 Charging and Volume Control System (Section 2.2.6)

The Charging and Volume Control System and components are located in the Vapor Container, Lower Pipe Chase, and Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not directly impact Spent Fuel Pit operations. However, sections of this system are interconnected with and are in close proximity to the SFP Cooling and Purification System. Safe load paths and protective zones must be established to prevent interactions with spent fuel cooling.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Charging and Volume Control System should be isolated at the connections to the Shutdown Cooling, Component Cooling, Purification, Chemical Shutdown, Emergency Core Cooling, Primary Plant Corrosion Control, Service Water, Demineralized Water, Main Coolant, Pressure Control and Relief, Primary Plant Sample, Safe Shutdown, Emergency Feedwater, and Primary Plant Vent and Drain systems.

- Sludge should be removed from the Low Pressure Surge Tank prior to dismantlement of the system. The tank should be dismantled into manageable sections in the tank cubicle.

- The Feed and Bleed Heat Exchangers and Vapor Container Bleed Line piping contain a significant amount of contamination and should be cut using a mechanical method. Each of the four shells and containers of Vapor Container Bleed Line piping must be handled individually to minimize the release of airborne radioactivity in the unlikely event that a handling accident occurs.

- The charging pumps, motors, and drive units should be separated from the baseplate prior to removal from the pump cubicles.

- The Low Pressure Surge Tank Cooling pump and Low Pressure Surge Tank Cooler should be removed intact through the Primary Auxiliary Building corridor and double doors.

2.3.5.6 Chemical Shutdown System (Section 2.2.7)

The Chemical Shutdown System and components are located in the Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Chemical Shutdown System should be isolated at the connections to the Charging and Volume Control, Emergency Core Cooling, Demineralized Water, and Heating systems.

- The Boric Acid Mix Tank should be removed from the Primary Auxiliary Building intact and moved to a cutting station to be dismantled into manageable sections.

2.3.5.7 Purification System (Section 2.2.8)

The Purification System and components are located in the Ion Exchange Pit and Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not directly impact Spent Fuel Pit operations. However, sections of this system are interconnected with and are in close proximity to the SFP Cooling and Purification System. Safe load paths and protective zones must be established to prevent interactions with spent fuel cooling.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Purification System should be isolated at the connections to the Charging and Volume Control, Primary Plant Sample, Shutdown Cooling, Emergency Core Cooling, Service Water, Waste Disposal, SFP Cooling and Purification, and Primary Plant Vent and Drain systems.

- All resin should be sluiced from the ion exchange capsules and dewatered. The capsules should be removed from the Ion Exchange Pit and rinsed to remove chromate residue from the external surfaces.

2.3.5.8 Component Cooling Water System (Section 2.2.9)

The Component Cooling Water System and components are located in the Vapor Container, Upper Pipe Chase, Waste Disposal Building and Primary Auxiliary Building. The system is required to support decommissioning activities. Portions of the Component Cooling Water System are required to support Spent Fuel Pit operations and liquid waste processing.

The Component Cooling Water System may be dismantled in stages by isolating sections of the system that are not required to support decommissioning activities or Spent Fuel Pit cooling. Controls must be implemented to clearly identify the section of the system which are required to support decommissioning. Piping at the interfaces of systems and sections being removed must be sealed with welded caps to ensure system integrity.

The supply and return lines to the Vapor Container may be isolated in the upper level of the Primary Auxiliary Building permitting removal of the Component Cooling Water System lines in the Upper Pipe Chase and the Vapor Container. The supply and return lines to the Waste Disposal Building can be isolated in the lower level of the Primary Auxiliary Building permitting the removal of the Component Cooling Water System lines in the Waste Disposal Building. The lines serving the Waste Disposal Building will only be removed when the liquid waste evaporator is not required to support decommissioning or a temporary liquid waste processing system has been installed or an alternate cooling system for the evaporator is available.

The balance of the Component Cooling Water System will remain in service until either the Spent Fuel Pit Cooling System is not required or an alternate source of cooling water to the Spent Fuel Pit cooler is installed.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Component Cooling Water System should be isolated at the connections to the Charging and Volume Control, Primary Plant Sample, Shutdown Cooling (only when Spent Fuel Pit Cooling is no longer required), Main Coolant, Spent Fuel Pit Cooling and Purification (only when Spent Fuel Pit Cooling is no longer required), Neutron Shield Tank, Waste Disposal (only when the evaporator is no longer required), Service Water, Water Cleanup, and Primary Plant Vent and Drain systems.

- The non-contaminated portions of the Component Cooling Water System will be removed prior to the commencement of area and building dismantlement activities.

- Dead legs and low flow sections of piping should be checked for chromate residue prior to disposal.

2.3.5.9 Primary Plant Corrosion Control System (Section 2.2.10)

The Primary Plant Corrosion Control System and components are located in the Water Treatment Room and Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

The contaminated portions of the Primary Plant Corrosion Control System are located in the charging pump cubicles, under the floor blocks in the Primary Auxiliary Building cubicle corridor, and in the Low Pressure Surge Tank Room.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific system consideration:

- The Primary Plant Corrosion Control System should be isolated at the connections to the Charging and Volume Control, Water Treatment, and Condensate Systems.

2.3.5.10 Primary Plant Sample System (Section 2.2.11)

The Primary Plant Sample System and components are located in the Vapor Container, Lower Pipe Chase, and Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Primary Plant Sample System should be isolated at connections to the Main Coolant, Charging and Volume Control, Service Water, Purification, Shutdown Cooling, Component Cooling, Waste Disposal, Neutron Shield Tank, Steam Generator Blowdown, and Primary Plant Vent and Drain systems.

- The seven Primary Plant Sample System coolers should be removed intact.

- The Post Accident Sample System panel may be cut into sections to permit removal from the building.

- The contaminated sample sink and hood should be cut into sections for removal. The duct should be left in place until the Ventilation System is not required.

2.3.5.11 Waste Disposal System (Section 2.2.12)

The Waste Disposal System and components are located in the Tank Farm Area, Waste Disposal Building, and Primary Auxiliary Building. Portions of the system will be used to support both decommissioning and Spent Fuel Pit operations. The gaseous waste subsystem is not required. The liquid water subsystem is required to process water from Spent Fuel Pit operations and decommissioning activities. The liquid waste evaporator and drumming equipment may be dismantled if a temporary liquid waste processing system is installed. The solid waste processing subsystem is required until area and building decontamination is completed or alternate processing capability is established.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- Waste Disposal Liquid Waste Processing Subsystem:
    - The liquid waste subsystem should be isolated at the connections to the Component Cooling, Heating, Service Water, and Primary Plant Vent and Drain systems.
    - Sludge should be removed from the Activity Decay, Waste Hold-up, Primary Building Sump, and Gravity Drain Tanks prior to dismantlement of the subsystem.
    - The Gravity Drain and Primary Building Sump Tank should be dismantled into manageable sections in the tank cubicle.
    - The evaporator should be dismantled into the largest sections practicable which can be removed and processed for shipment. Segmentation of the vessel, tower, reboiler and heat exchangers should not be required.
    - The drumming equipment should be dismantled into the largest sections practicable which can be removed and processed for shipment.
    - The dismantlement of the Activity Decay, Waste Hold-up, Monitored Waste, and Test Tanks are presented in Section 2.3.5.55.
- Waste Disposal Gaseous Waste Processing Subsystem:
    - The gaseous waste subsystem should be isolated at the connections to the Component Cooling, Service Water, Ventilation, and Primary Plant Vent and Drain systems.

- The 6 inch line from the loop seal to the Primary Vent Stack must be capped and sealed after the piping has been cut as close as practicable to the Primary Vent Stack. The vent line attached to the Primary Vent Stack should be removed when the Ventilation System is dismantled.

- The dismantlement of the Waste Gas Surge Drum and Gas Decay Drums is presented in Section 2.3.5.55.

- Waste Disposal Solid Waste Processing Subsystem:

    - No specific considerations.

2.3.5.12 Shutdown Cooling System (Section 2.2.13)

The Shutdown Cooling System and components are located in the Vapor Container, Lower Pipe Chase, and the Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. The cooler and associated piping are required to support Spent Fuel Pit operations.

The Shutdown Cooling System can be dismantled in stages depending on the need to maintain the system for redundant Spent Fuel Pit cooling capabilities. The supply and return lines to the Vapor Container can be isolated in the Primary Auxiliary Building Valve Room by cutting and capping the lines which will permit the removal of the contaminated lines in the Lower Pipe Chase and the Vapor Container. The balance of the system must remain in service until the fuel is permanently removed from the Spent Fuel Pit or an alternate to the Shutdown Cooling Heat Exchanger is installed.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Shutdown Cooling System should be isolated at the connections to the Charging and Volume Control, Primary Plant Sample, Main Coolant, Spent Fuel Pit Cooling and Purification (only after the Shutdown Cooling System Heat Exchanger is not required), Service Water, and Primary Plant Vent and Drain systems.

- The Shutdown Cooling Pump and Shutdown Cooling Cooler should be removed intact through the Primary Auxiliary Building corridor and double doors (only after the Shutdown Cooling System Heat Exchanger is not required).

2.3.5.13 Primary Plant Vent and Drain System (Section 2.2.14)

The Primary Plant Vent and Drain System and components are located in the Vapor Container, Lower Pipe Chase and Primary Auxiliary Building. The system will be used during both safe storage period and Reactor Vessel dismantlement activities. Upon completion of the Reactor Vessel dismantlement the system can be removed. This system has no impact on Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3.
The following are specific system considerations:

- The Primary Plant Vent and Drain System should be isolated at the connections to the Main Coolant, Charging and Volume Control, Emergency Core Cooling, Shutdown Cooling, Pressure Control and Relief, Purification, Component Cooling, Primary Plant Sample, and Waste Disposal.

- Sludge should be removed from the Primary Drain Collecting and VC Drain Tanks prior to dismantlement of the system. The Primary Drain Tank should be dismantled into manageable sections in the tank cubicle. The VC Drain Tank should be removed intact.

2.3.5.14 Emergency Core Cooling System (Section 2.2.15)

The Emergency Core Cooling System and components are located in the Vapor Container, Lower Pipe Chase, Diesel Generator Building, Primary Auxiliary Building, and Yard Area. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

The dismantlement of the Safety Injection and abandoned Safety Injection Storage Tanks is presented in Section 2.3.5.55.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Emergency Core Cooling System should be isolated at the connections to the Shutdown Cooling, Chemical Shutdown, Purification, Chemical Shutdown, Main Coolant, Demineralized Water, Primary Plant Vent and Drain, Water Cleanup, and Emergency Feedwater systems.

- The accumulator should be dismantled into manageable sections in the accumulator cubicle.

- The block walls and roof of the accumulator cubicle should be removed as required to permit the removal of the sections.

- The pumps and motors should be separated from the baseplates prior to removal.

- The uncontaminated nitrogen supply portions of the Emergency Core Cooling System should be removed prior to the commencement of area and building dismantlement activities.

2.3.5.15 Radiation Monitoring System (Section 2.2.16)

The Radiation Monitoring System and components are located in the Vapor Container, Lower Pipe Chase, Diesel Generator Building, Primary Auxiliary Building, Turbine Building, Control Room, and Yard Area.

The process radiation monitoring equipment will be removed as the systems that they monitored are dismantled. The area radiation monitoring equipment will remain in operation until all contaminated process systems have been removed from the area and will then be removed prior to the commencement of area and building decontamination activities.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific system consideration:

- The Radiation Monitoring System in uncontaminated areas of the plant should be removed as part of the site dismantlement and restoration process.

2.3.5.16 VC Ventilation and Purge System (Section 2.2.17)

The VC Ventilation and Purge System and components are located in the Vapor Container, Primary Auxiliary Building, and Yard Area. The system is required to support decommissioning activities and is interconnected with the Ventilation System which supports Spent Fuel Pit operations.

The VC Ventilation and Purge System should remain in service until all contaminated systems in the Vapor Container have been dismantled and decontamination activities in the Vapor Container have been completed.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The fan and motor should be separated from the baseplate prior to removal.

- The filter unit should be dismantled into manageable sections.

2.3.5.17 VC Heating and Cooling System (Section 2.2.18)

The contaminated portions of the VC Heating and Cooling System are located in the Vapor Container. The system is required to support decommissioning activities in the Vapor Container. This system is not required to support Spent Fuel Pit operations.

The VC Heating and Cooling System should remain operational to support environmental heating and cooling requirements during contaminated system removal activities. Temporary heating and cooling may be required during Vapor Container structural dismantlement activities.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The fans and motors should be separated from the baseplates prior to removal.

- The coolers should be disassembled into manageable sections prior to removal from the Vapor Container.

2.3.5.18 Post-Accident Hydrogen Control System (Section 2.2.19)

The contaminated portions Post-Accident Hydrogen Control System and components are located in the Vapor Container and Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- No specific considerations.

2.3.5.19 Containment Isolation System (Section 2.2.20)

The Containment Isolation System and components are located in the Vapor Container and Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- No specific considerations.

2.3.5.20 Fuel Handling Equipment System (Section 2.2.21)

The Fuel Handling Equipment System and components are located in the Vapor Container and Spent Fuel Pit Building. The system is not required in a permanently defueled condition or to support decommissioning activities. The portions of the Vapor Container fuel handling components will be dismantled as part of Component Removal Project. The portions of this system in the Spent Fuel Pit are required for the removal of fuel and irradiated materials from the Spent Fuel Pit.

Dismantlement of the Spent Fuel Pit fuel handling components, Vapor Container Polar Crane, Yard Area Crane are presented in Sections 2.3.5.46, 2.3.5.38, and 2.3.5.43.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The remaining Vapor Container fuel handling components (manipulator, upender, Reactor Vessel internal lifting devices, transfer carriage) should be dismantled into manageable sections.

2.3.5.21 SFP Cooling and Purification System (Section 2.2.22)

The SFP Cooling and Purification System and components are located in the Spent Fuel Pit Building, Primary Auxiliary Building, and Ion Exchange Pit. The SFP Cooling and Purification System will remain in service until the fuel is permanently removed from the Spent Fuel Pit.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The SFP Cooling and Purification System should be isolated at the connections to the Shutdown Cooling, Component Cooling, and Demineralized Water systems.

- All resin should be sluiced from ion exchange capsules and dewatered. The capsules should be removed from the Ion Exchange Pit and rinsed to remove chromate residue from the external surfaces.

2.3.5.22 Main Steam System (Section 2.2.23)

The Main Steam System and components are located in the Vapor Container, Turbine Building, and Yard Area. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- No specific considerations.

2.3.5.23 Feedwater System (Section 2.2.24)

The Feedwater System and components are located in the Vapor Container, Turbine Building, and Yard Area. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- No specific considerations.

2.3.5.24 Steam Generator Blowdown System (Section 2.2.25)

The Steam Generator Blowdown System and components are located in the Vapor Container, Upper Pipe Chase, and Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- No specific considerations.

2.3.5.25 Emergency Feedwater System (Section 2.2.26)

The Emergency Feedwater System and components are located in the Turbine Building, Primary Auxiliary Building, and Yard Area. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific system consideration:

- The Emergency Feedwater System should be isolated at the connections to the Charging and Volume Control, Emergency Core Cooling, and Safe Shutdown systems.

2.3.5.26 Service Water System (Section 2.2.27)

The Service Water System and components are located in the Vapor Container, Upper Pipe Chase, Turbine Building, Primary Auxiliary Building, Screenwell House, Waste Disposal Building, and Yard Area. The system will be used to support decontamination and dismantlement activities. Portions of the Service Water System are required to support Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific system consideration:

- The Service Water System should be isolated at the connections to the Shutdown Cooling, Waste Disposal (when the evaporator is not required), Component Cooling (when component cooling is not required), Charging and Volume Control, Purification, Primary Plant Sample, and Demineralized Water systems.

2.3.5.27 Demineralized Water System (Section 2.2.28)

The Demineralized Water System and components are located in the Vapor Container, Upper Pipe Chase, Turbine Building, Primary Auxiliary Building, Service Building, Waste Disposal Building, and Yard Area. The system will be used to support decontamination and dismantlement activities. Portions of the Demineralized Water System are required to support Spent Fuel Pit operations.

The dismantlement of the Primary Water Storage and Demineralized Water Storage Tanks is presented in Section 2.3.5.55.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3.

The following is a specific system consideration:

- The Demineralized Water System should be isolated at the connections to the plant systems as they are isolated and dismantled.

2.3.5.28 Compressed Air System (Section 2.2.29)

The Compressed Air System and components are located in the Vapor Container, Turbine Building, Primary Auxiliary Building, Service Building, Waste Disposal Building, and Yard Area. The system should remain in service to support decommissioning activities. Portions of the system will be isolated and removed as the systems and areas that they support are dismantled. The breathing air subsystem will be maintained to support area and building decontamination and dismantlement activities.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- No specific considerations.

2.3.5.29 Electrical System (Section 2.2.30)

The Electrical System and components are located in all plant buildings and areas.

The Electrical System should remain in service until all contaminated process systems have been dismantled. Portions of the Electrical System will be disconnected and isolated as the systems that they support are dismantled. Prior to area and building decontamination and dismantlement activities, the Electrical System will be physically isolated from the power supply grid and a temporary electrical supply system will be installed to provide electrical power. An alternate power supply will be installed to support the systems required for Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific system consideration:

- Electrical penetrations in the Vapor Container should be disconnected and left for removal with the shell plating.

2.3.5.30 Heating System (Section 2.2.31)

The Heating System and components are located in all plant buildings. The system should remain operational to support environmental heating requirements during contaminated system removal activities. Temporary heating may be required during area and building dismantlement activities.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- No specific considerations.

2.3.5.31 Ventilation System (Section 2.2.32)

The Ventilation System and components are located in the Primary Auxiliary Building, Waste Disposal Building, and Yard Area. The Ventilation System is required to support safe storage and decommissioning activities. Portions of the Ventilation System are required for Spent Fuel Pit operations.

The Ventilation System should remain in service until all contaminated process systems have been dismantled and a majority of the area and building decontamination and dismantlement is complete. Portions of the Ventilation System will remain in service until the fuel is permanently removed from the Spent Fuel Pit.

The dismantlement of the Primary Vent Stack is presented in Section 2.3.5.45.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The fans and motors should be separated from the baseplates prior to removal.

- The filter units should be dismantled into manageable sections.

2.3.5.32 Fire Protection and Detection System (Section 2.2.33)

The Fire Protection and Detection System and components are located in plant buildings and structures as described in the Fire Protection Technical Requirements Manual.

The system should remain in service until all contaminated process systems have been dismantled and fuel is permanently removed from the Spent Fuel Pit. However, portions of the Fire Protection and Detection System may be disconnected, isolated and removed when they are no longer required to support fire protection requirements.

The dismantlement of the Fire Water Storage Tank is presented in Section 2.3.5.55.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3.

The following is a specific system consideration:

- Modifications to the Fire Protection and Detection System require review and modifications, as necessary, to the YNPS Fire Protection Plan (Section 9).

2.3.5.33 Primary Pump Seal Water System (Section 2.2.34)

The Primary Pump Seal Water System and components are located in the Primary Auxiliary Building. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Primary Pump Seal Water System should be isolated at the connections to the Demineralized Water, Charging and Volume Control, and Shutdown Cooling systems.

- The Seal Water Tank should be removed in one piece.

2.3.5.34 Safe Shutdown System (Section 2.2.35)

The Safe Shutdown System and components are located in the Safe Shutdown Building, Primary Auxiliary Building and Yard area. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific system consideration:

- The Safe Shutdown System should be isolated at the connections to the Fire Protection and Detection, Charging and Volume Control, and Emergency Feedwater systems.

2.3.5.35 Water Cleanup System (Section 2.2.36)

The Water Cleanup System and components are located in the Ion Exchange Pit, Primary Auxiliary Building, and Yard area. The system is not required in a permanently defueled condition or to support decommissioning activities. This system does not impact Spent Fuel Pit operations. The heat exchanger may be modified for use in the SFP Cooling and Purification System.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Water Cleanup System should be isolated at the connections to the Component Cooling, and Emergency Core Cooling systems.
- The sand should be sluiced out of the sand filters prior to removal from the Ion Exchange Pit.
- Components removed from the Ion Exchange Pit should be rinsed to remove chromates from external surfaces.

2.3.5.36 Vapor Container (Section 2.2.37)

The Vapor Container is required to prevent exposure to radiation and the inadvertent spread of contamination. The capability to establish Vapor Container isolation will be maintained during decommissioning activities inside the containment (Section 2.3.4.3). The reactor support structure support column and Fuel Transfer Chute decontamination activities are presented in Sections 2.2.4.37 and 2.2.4.42.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- The internal surface of the Vapor Container shell should be decontaminated after all contaminated process systems have been removed from the Vapor Container, but before commencement of the reactor support structure decontamination. A strippable coating may be applied to the skin to prevent recontamination.
- The internal surfaces of the Equipment Hatch should be decontaminated at the same time as the Vapor Container shell.
- The Personnel Hatch should be decontaminated at the same time as the Vapor Container shell.
- Piping penetrations should be cut off as close as practicable to the Vapor Container shell when the process system which passes through it is dismantled. The opening in the Vapor Container shell should be closed once the piping is removed.
- Electrical penetrations should be cut off as close as practicable to the Vapor Container shell after all cables in the penetration have been disconnected and

removed. The opening in the Vapor Container shell should be closed once the penetration is removed.

- Platforms, ladders and stairs along with the supporting steel members should be removed in conjunction with area decontamination and dismantlement activities.

- Dismantlement of the Vapor Container shell should not commence until the fuel chute has been physically disconnected from the Spent Fuel Pit.

2.3.5.37 Reactor Support Structure (Section 2.2.38)

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- The shield tank cavity liner should be removed to permit decontamination of the underlying concrete surfaces.

- The steel casings of the support columns from the shell to the expansion joint should be removed to permit access to the concrete column (Figure 2.3-1).

- The concrete columns should be decontaminated by removing the contaminated concrete.

- All contaminated equipment should be removed prior to decontamination or removal of concrete on the walls, floors and ceilings.

- Concrete and reinforcing bar on the inner section of the inner support wall, which was behind the Neutron Shield Tank, is slightly activated and will require removal to a depth of about 4 inches. The removal zone extends from the top of the Neutron Shield Tank support ledge down to the floor of the cavity, a total height of approximately 17 feet (Figure 2.3-2).

2.3.5.38 Vapor Container Polar Crane (Section 2.2.39)

The Vapor Container Polar Crane is required to support Vapor Container decontamination and dismantlement activities.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- The Vapor Container Polar Crane should be decontaminated at the same time as the Vapor Container shell.

- The hoist trolley, motors, and control cab should be removed from the girders.
- The girders, drive trollies, and crane rails should be removed when the Vapor Container shell is being dismantled and a temporary crane is available.

2.3.5.39 Radiation Shielding (Section 2.2.40)

The Neutron Shield Tank decontamination and dismantlement activities are presented in Section 2.3.5.39. The Primary Shield decontamination and dismantlement activities are presented in Section 2.3.5.36. The Secondary Shield decontamination and dismantlement activities are presented in Section 2.3.5.36. The Fuel Handling Shield decontamination and dismantlement activities are presented in Section 2.3.5.41.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- Auxiliary shielding will be decontaminated and dismantled as part of the area and building decontamination and dismantlement activity
- Supplemental shielding may be decontaminated and dismantled at any time.

2.3.5.40 Neutron Shield Tank (Section 2.2.41)

Neutron irradiation from the reactor core generated activation products in the Neutron Shield Tank inner wall (Section 3.1.3.4). The radionuclide concentrations do not exceed 10 CFR Part 61 Class A limits. Although the radionuclide content exceeds the 10 CFR Part 71 Type A package limits, the specific activity is less than the 10 CFR Part 71 low specific activity limits. The outer wall has significantly lower radioactivity.

The Neutron Shield Tank contained chromated water during plant operations.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific system considerations:

- The Neutron Shield Tank and Coolers should be isolated by closing the valves at the connections to the Component Cooling system.
- The Reactor Vessel must be removed prior to the commencement of Neutron Shield Tank dismantlement activities.
- The Neutron Shield Tank should be filled with demineralized water, connected to a de-chromating ion exchange unit, and recirculated until all residual chromates have

been removed from the tank. This activity must be completed prior to the start of Reactor Vessel removal activities if concrete is to be placed in the bottom of the tank for stabilization during vessel segmentation.

- The Neutron Shield Tank Surge Tank and associated piping can be removed after the flushing activity has been completed.

- Any debris from the Reactor Vessel removal activity should be removed from the inner Neutron Shield Tank annular space.

- The inner drain line should be plugged prior to filling the annular space with water.

- Low activity sections of the Neutron Shield Tank internals, lower head, and outer shell should be wrapped in protective coverings prior to removal from the Shield Tank Cavity to reduce the spread of contamination.

- The Neutron Shield Tank should be dismantled as follows:

  - Fill the inner annular space with demineralized water. The water level should be maintained below the bottom of the nozzle openings to prevent spillage into the loop areas

  - Install a water filtration system to maintain water clarity during cutting activities.

  - Establish airborne containment measures over the Neutron Shield Tank as required.

  - Using underwater thermal techniques, segment the Neutron Shield Tank inner shell.

  - The inner shell sections, excluding the lower head, should be placed in a liner located in the Shield Tank Cavity.

  - When the inner shell has been removed, drain the Neutron Shield Tank and annular space.

  - Segment and remove the internals and outer shell of the Neutron Shield Tank using thermal techniques.

  - The lead shielding should be removed prior to segmenting the upper section of the tank and the detector sleeves.

- Any debris from the Neutron Shield Tank removal activity should be removed from the reactor cavity.

2.3.5.41 Pipe Chases (Section 2.2.42)

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- No specific considerations.

2.3.5.42 Fuel Transfer Chute (Section 2.2.43)

The Fuel Transfer Chute contains two valves adjacent to the Spent Fuel Pit wall that are significantly contaminated. In addition, integrity of the Fuel Transfer Chute is necessary to maintain Spent Fuel Pit integrity. This integrity can be permanently established by capping the end of the chute in the Spent Fuel Pit and filling the Fuel Transfer Chute between the cap and the lower lock valve with structurally stable material. Additionally, the Fuel Transfer Chute should either be cut above the lower lock valve (after isolating the chute from the Spent Fuel Pit) or structurally reinforced before commencement of activities that could affect chute integrity (e.g., heavy lifts over or in close proximity to the chute). Removal of the Fuel Transfer Chute and lower lock valve from the Spent Fuel Pit wall can only be performed after the fuel has been removed from the pit.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- The Fuel Transfer Chute must be capped and filled with a low density grout from the Shield Tank Cavity to the lower lock valve and from the Spent Fuel Pit to the lower lock valve from the SFP side.

- The Fuel Transfer Chute must be cut as close as practicable to the Spent Fuel Pit prior to commencement of dismantlement activities which could affect chute integrity.

- The Fuel Transfer Chute should be dismantled as follows:

  - Coat the internal surfaces of the Fuel Transfer Chute Structure and entrance manhole with a fixative to bind the loose contamination.

  - Remove the removable shield blocks from the access openings on top of the Fuel Transfer Chute Structure. Decontaminate the blocks as required.

- The lead shielding should be removed from the Fuel Transfer Chute Structure at the Vapor Container shell penetration.

- Encase the lower lock valve in a high density concrete enclosure to provide shielding and protection for the valve and Fuel Transfer Chute to the Spent Fuel Pit. (Figure 2.3-3)

- Segment the Fuel Transfer Chute and Structure into manageable sections using a diamond wire saw or a similar method. Start segmenting at the Vapor Container shell and stop as close as practicable to the lower lock valve concrete enclosure.

- Remove the fuel chute, lower lock valve and concrete enclosure intact with a section of the Spent Fuel Pit wall and place it into a shipping cask for disposal.

- Excavate the entrance manhole and move it to a cutting station for segmentation.

- Close the opening in the Vapor Container shell at the location of the Fuel Transfer Chute penetration.

2.3.5.43 Yard Area Crane And Support Structure (Section 2.2.44)

The yard area crane and support structure will be used to support fuel management activities until fuel is permanently removed from the Spent Fuel Pit. The crane will also be used to support activities associated with the Spent Fuel Pit, ion exchange pit, and other heavy lifts.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- No specific considerations.

2.3.5.44 Ion Exchange Pit (Section 2.2.45)

The Ion Exchange Pit supports Spent Fuel Pit operation and will remain in operation until fuel is removed from the Spent Fuel Pit or the capsules are relocated. No dismantlement activities should be performed on the Ion Exchange Pit walls or floor until fuel is permanently removed from the Spent Fuel Pit.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- Chromates should be removed from the water in the Ion Exchange Pit before the water is drained from the pit.
- The sludge at the bottom of the pit may contain chromates and should be evaluated for mixed waste potential.
- All resin should be sluiced from the ion exchange capsules and dewatered. The capsules and Water Cleanup System components should be removed from the Ion Exchange Pit prior to draining the pit. All components should be rinsed when moved from the pit to remove chromate residue from the external surfaces.
- Spent Fuel Pit security requirements must be re-evaluated if the Ion Exchange Pit is emptied while fuel is in the Spent Fuel Pit.
- Concrete and soil removal may be required due to known leakage from the Ion Exchange Pit.

2.3.5.45 Primary Vent Stack (Section 2.2.46)

The Primary Vent Stack is required to support decommissioning activities and to vent air processed by both the Ventilation System and the VC Ventilation and Purge System. The Primary Vent Stack must not be dismantled until the fuel has been permanently removed from the Spent Fuel Pit and decontamination activities in the Vapor Container, Primary Auxiliary Building, Waste Disposal Building, and Spent Fuel Pit Building have been completed.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific area consideration:

- The Primary Vent Stack should be removed in one piece, capped, and lowered to the ground for dismantlement.

2.3.5.46 Spent Fuel Pit And Spent Fuel Pit Building (Section 2.2.47)

Fuel, secondary neutron sources, Reactor Vessel internals (high level radioactive segments), and other irradiated components are stored in a two tier rack system. The upper racks are modules supported on intermediate columns attached to the Spent Fuel Pit floor. Grating is installed between upper and lower racks. The Spent Fuel Pit

Building is a steel-braced frame, metal sided structure that supports the superstructure to both the New Fuel Vault and the Spent Fuel Pit.

The Spent Fuel Pit and Spent Fuel Pit Building must be maintained until fuel is permanently removed from the Spent Fuel Pit.

The Fuel Transfer Chute decontamination and dismantlement activities are presented in Section 2.3.5.42.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- If fuel is stored in the Spent Fuel Pit during the dismantlement of systems and structure in proximity of the Spent Fuel Pit Building, the special precautions presented in Section 3.3.1.3 will be instituted to restrict activities that could cause physical damage or other adverse consequences to the Spent Fuel Pit and associated support systems.

- The fuel racks and Spent Fuel Pit liner should be decontaminated prior to dismantlement.

- The Spent Fuel Pit fuel handling components (manipulator, manipulator rails, upender, new fuel elevator, and fuel inspection elevator) should be dismantled into manageable sections.

2.3.5.47 New Fuel Vault (Section 2.2.48)

The New Fuel Vault is required to support plant activities until fuel is permanently removed from the Spent Fuel Pit.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific area consideration:

- The walls of the New Fuel Vault must remain intact until fuel is permanently removed from the Spent Fuel Pit.

2.3.5.48 Primary Auxiliary Building (Section 2.2.49)

The Primary Auxiliary Building will be required during decommissioning to prevent inadvertent exposure to radiation and spread of contamination until the contaminated systems and components within the building are dismantled.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- The removable shield blocks in the Primary Auxiliary Building corridor floor should be removed to permit access to the trench below.
- The block shield wall between the corridor and the Vertical Pipe Chase should be removed to permit greater access to the pipe chase.

2.3.5.49 Diesel Generator Building (Section 2.2.50)

The Diesel Generator Building will be required during decommissioning to prevent inadvertent spread of contamination until the contaminated systems and components within the building are dismantled.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following is a specific area consideration:

- The carbon dioxide fire protection system should be removed prior to commencement of decontamination activities in Manhole No. 3.

2.3.5.50 Waste Disposal Building (Section 2.2.51)

The Waste Disposal Building will be required during decommissioning to prevent inadvertent exposure to radiation and spread of contamination until the contaminated systems and components within the building are dismantled.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- No specific considerations.

2.3.5.51 Safe Shutdown System Building (Section 2.2.52)

The Safe Shutdown System Building will be required during decommissioning to prevent inadvertent spread of contamination until the contaminated components within the building are dismantled.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- No specific considerations.

2.3.5.52 Potentially Contaminated Area Storage Buildings 1 & 2 and Warehouse (Section 2.2.53)

These storage areas will be required during decommissioning to prevent inadvertent exposure to radiation and spread of contamination. All radioactive materials stored in these buildings will be removed before the buildings are decontaminated.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- No specific considerations.

2.3.5.53 Compactor Building (Section 2.2.54)

The building will be required during decommissioning to prevent inadvertent exposure to radiation and spread of contamination. The structure will be decontaminated after contaminated material processing is no longer required. The following are specific area considerations:

- No specific considerations.

2.3.5.54 Service Building (Section 2.2.55)

The building will be required during decommissioning to support dismantlement and decontamination activities. The structure will be decontaminated after most of the site decommissioning activities have been completed.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- No specific considerations.

2.3.5.55 Miscellaneous Tanks (Section 2.2.56)

The following tanks will not be used to support decommissioning: Abandoned Safety Injection Tank, two Monitored Waste Tanks, Waste Gas Surge Drum, and three Waste Gas Decay Drums. The following tanks will be used to support decommissioning activities:

- The Safety Injection Tank may be used to temporarily store mildly contaminated liquids or demineralized water during decommissioning activities.

- The Primary Water Storage Tank is used to store demineralized water to support Spent Fuel Pit operations and decommissioning activities.

- The Waste Holdup and Activity Dilution Decay Tanks will be used to support waste processing during plant decommissioning.

- The two Test Tanks will be used to support waste processing during plant decommissioning.

- The Demineralized Water Storage Tank may be used to provide demineralized water to support Spent Fuel Pit operations, auxiliary boiler operation, and decommissioning activities.

- The Fire Water Storage Tank will be used to support operation of the Fire Protection and Detection System during decommissioning activities.

- The Fuel Oil Storage Tank will be used to support decommissioning activities until after the auxiliary boilers and the emergency diesel generators are dismantled.

Section 2.2.56 presents more detailed descriptions of the miscellaneous tanks at YNPS.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- Temporary holding tank(s) may be used to permit the removal of the Test Tanks.

- The tanks and drums should be desludged and dismantled into large manageable sections.

2.3.5.56 Meteorological Tower (Section 2.2.57)

The meteorological tower will be used to support safe storage and dismantlement period activities. The tower will be removed after dismantlement period activities are completed and the Off-site Dose Calculation Manual and the Defueled Emergency Plan are modified.

General decontamination and dismantlement considerations are presented in Section 2.3.4.3. The following are specific area considerations:

- No specific considerations.

2.3.6 Schedule of Decommissioning Activities

The decommissioning schedule presented in Table 2.3-5 is based on durations derived from the Decommissioning Cost Estimate (Section 5). It is anticipated that the Decommissioning Plan will be approved within one year of submittal to the NRC. Following approval, the safe storage period will begin. The Dismantlement Period is expected to begin in early 1999 with detailed engineering activities and continue through site restoration in 2003. For planning purposes, the last spent fuel shipment to a disposal facility is scheduled for 2018.

Yankee will perform detailed, task specific scheduling as part of the detailed planning.

2.3.7 Decommissioning Exposure Projections

The total decommissioning exposure from all sources is estimated to be about 744 person-rem. This estimate is based on completing all activities in January 1994 and therefore represents a bounding estimate of exposure for the YNPS decommissioning. A summary of the estimated exposures for the decommissioning activities is presented in Table 2.3-6. This estimate is for planning purposes only. Detailed exposure estimates and exposure controls will be developed in accordance with the requirements of the ALARA Program (Section 3.2.5) during detailed planning of decommissioning activities.

The estimate incorporates the following assumptions:

- Area dose rates are based on radiological scoping survey results measured during the first half of 1993 (Section 3.1). No credit has been taken for radioactive decay.

- The exposure hours for decommissioning activities are based on the labor estimates from the decommissioning cost estimate (Section 5). The exposure hours reflect time spent in the radiation field with an adjustment for work difficulty.

- The Component Removal Project personnel exposure is based on current project estimates accounting for project implementation experience (Reference 3.4-6).

- Safe storage period personnel exposure is not included in the limiting estimate. Personnel exposure would be about 48 person-rem for a safe storage period between 1994 and 2000. This value is based on actual YNPS exposure data from the first quarter of 1993 with an annual adjustment for the decay of cobalt-60.

- Fuel transfer personnel exposure is based on an exposure of 1 person-rem per cask loaded. This rate is based on recent industry experience in transferring fuel from a wet to dry storage facility. The exposure also includes decontamination of the Spent Fuel Pit after removal of the spent fuel is completed.

Yankee is committed to implement the principles of the ALARA Program to reduce personnel exposures to as low as is reasonably achievable.

2.3.8 Decommissioning Radioactive Waste Projections

The radioactive waste management program (Section 3.3) will be used to control radioactive waste handling during decommissioning. The largest volume of low level radioactive waste will be generated during the dismantlement of activated and contaminated systems and components. Another significant contributor is waste produced from the removal of contaminated concrete and structural components. Additional waste generated during the support of decontamination activities will include the following:

- Contaminated water
- Used disposable protective clothing
- Expended abrasive and absorbent materials
- Contamination control materials (e.g., strippable coatings, plastic enclosures, expended filters)

The waste volume projection is based on data obtained during the radiological scoping survey and detailed plant system and commodity reviews. Packaging, shipping, and volume reduction factors were derived from the cost estimate. Tables 2.3-7, 2.3-8, and 2.3-9 present an estimate of waste volumes and the number of containers and shipments needed to transport the waste to a low level radioactive disposal facility. No credit is taken for the decay of system radioactivity contamination levels or material activation. The waste volume projection is similar to the independent estimate incorporated into the decommissioning cost estimate.

A significant waste volume reduction can be achieved by using the several off-site radioactive materials processing options that are currently available to YNPS. Decontamination and release rates of up to 85% for selected commodities could reduce YNPS burial volumes by about 55% (Table 2.3-10). However, more stringent radioactive material release criteria could limit the availability of processing alternatives when YNPS decommissioning actually occurs. The current decommissioning cost estimate (Section 5) assumes that the availability of processing alternatives is limited and that all significantly contaminated and activated materials are sent to a disposal facility. However, all processing alternatives will be evaluated during decommissioning to determine the most effective processing options for radioactive materials.

REFERENCES

2.3-1 YRP 435/92, Spent Nuclear Fuel Storage Study Report and Recommendations, B. W. Holmgren, J. M. Buchheit, R. A. Mellor to J. K. Thayer, October 9, 1992.

2.3-2 YRP 303/93, Impact of Wet Spent Fuel Storage On Decommissioning, P. A. Rainey to R. A. Mellor, July 9, 1993.

2.3-3 REG 227/93, Low Level Waste Storage Safety Evaluation, J. Bisson to R. A. Mellor, September 1, 1993.

2.3-4 WE-101, Engineering Design Change Request, YAEC Engineering Manual.

2.3-5 YRP 436/93, Reactor Vessel Removal Option Study for YNPS Decommissioning Plan, K.J. Heider, W.J. Pananos to R.A. Mellor, September 24, 1993.

2.3-6 CRP ALARA Committee Meeting 93-3 Agenda Memo, B Cox to G. Babineau, August 28, 1993.

PROCEDURE REFERENCES

AP-1902 "Plant Lay-up Program"

TABLE 2.3-1

SYSTEM CONTAMINATION

SECTION	SYSTEM	INTERNAL	EXTERNAL	EXTENT OF CONTAMINATION
2.3.5.3	MAIN COOLANT SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.4	PRESSURE CONTROL AND RELIEF SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.5	CHARGING AND VOLUME CONTROL SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.6	CHEMICAL SHUTDOWN SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.7	PURIFICATION SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.8	COMPONENT COOLING WATER SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.9	PRIMARY PLANT CORROSION CONTROL SYSTEM		X	ENTIRE SYSTEM
2.3.5.10	PRIMARY PLANT SAMPLE SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.11	WASTE DISPOSAL SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.12	SHUTDOWN COOLING SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.13	PRIMARY PLANT VENT AND DRAIN SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.14	EMERGENCY CORE COOLING SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.15	RADIATION MONITORING SYSTEM		X	ENTIRE SYSTEM
2.3.5.16	VC VENTILATION AND PURGE SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.17	VC HEATING AND COOLING SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.18	POST-ACCIDENT HYDROGEN CONTROL SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.19	CONTAINMENT ISOLATION SYSTEM		X	ENTIRE SYSTEM
2.3.5.20	FUEL HANDLING EQUIPMENT SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.21	SFP COOLING AND PURIFICATION SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.22	MAIN STEAM SYSTEM		X	PARTIAL SYSTEM
2.3.5.23	FEEDWATER SYSTEM		X	PARTIAL SYSTEM
2.3.5.24	STEAM GENERATOR BLOWDOWN SYSTEM	X	X	PARTIAL SYSTEM
2.3.5.25	EMERGENCY FEEDWATER SYSTEM		X	PARTIAL SYSTEM
2.3.5.26	SERVICE WATER SYSTEM		X	PARTIAL SYSTEM
2.3.5.27	DEMINERALIZED WATER SYSTEM		X	PARTIAL SYSTEM

TABLE 2.3-1 (Continued)

SYSTEM CONTAMINATION

SECTION	SYSTEM	INTERNAL	EXTERNAL	EXTENT OF CONTAMINATION
2.3.5.28	COMPRESSED AIR SYSTEM		X	PARTIAL SYSTEM
2.3.5.29	ELECTRICAL SYSTEM		X	PARTIAL SYSTEM
2.3.5.30	HEATING SYSTEM	X	X	PARTIAL SYSTEM
2.3.5.31	VENTILATION SYSTEM	X	X	ENTIRE SYSTEM
2.3.5.32	FIRE PROTECTION AND DETECTION SYSTEM		X	PARTIAL SYSTEM
2.3.5.33	PRIMARY PUMP SEAL WATER SYSTEM		X	ENTIRE SYSTEM
2.3.5.34	SAFE SHUTDOWN SYSTEM	X	X	PARTIAL SYSTEM
2.3.5.35	WATER CLEANUP SYSTEM		X	PARTIAL SYSTEM

TABLE 2.3-2

DECONTAMINATION METHODS

METHOD	ADVANTAGES	DISADVANTAGES
Carbon Dioxide Blasting	Low waste volume	High operating costs HEPA filtration required to control airborne contamination High setup costs
Abrasive Blasting	Very effective for surface contamination	Large waste volumes HEPA filtration required to control airborne contamination
Hydro Blasting	Remote operation possible Very effective for surface contamination Easy to use	Large waste volumes HEPA filtration required to control airborne contamination
Strippable coatings	Easy to use Good for fixing loose surface contamination	Only effective on loose contamination
Scarifying	Effective on coated and uncoated surfaces Removes concrete to 1/4" deep	Cannot cut rebar Vacuum with HEPA must be used to control waste and dust
Scabbling	Removes concrete to 1/4 inch deep per pass Easy to control removal depth per pass	Cannot cut rebar Vacuum with HEPA must be used to control waste and dust Not as effective on coated surfaces
Spalling	Low airborne contamination Good for limited access and small areas	Slow process Cannot cut rebar
Vacuum Cleaning	Easy to use Fast removal times	Only effective on loose contamination HEPA filtration required to control airborne contamination.

TABLE 2.3-3

MECHANICAL CUTTING/REMOVAL METHODS

METHOD	APPLICABILITY	ADVANTAGES	DISADVANTAGES
Machining	Contaminated pipe cutting Vessel segmentation	Minimum airborne contamination Easy setup Remote operation Quick cutting times	Unit is heavy in larger sizes
Abrasive Water-Jet	Tank segmentation Concrete cutting	Remote operation possible	Large amounts of liquid waste HEPA filtration required to control airborne contamination Slow cutting times
Abrasive Wheel	Steel cutting	Inexpensive Easy to setup and operate Can cut rebar and imbedded steel	HEPA filtration required to control airborne contamination Slow Cutting times
Diamond Wire	Concrete cutting	Capable of cutting thick concrete Can cut rebar and imbedded steel Remote operation possible	Slow cutting times Large amounts of liquid waste
Metal Disintegration Machining (MDM)	Vessel segmentation	Capable of cutting thick steel High precision obtainable Remote operation possible	Slow cutting times Complex control system required
Mechanical Shears	Tubing, cable, and pipe cutting	Quick cutting times Best for tubing and sheet metal	Cannot be used on large items
Band and Reciprocating Saws	Contaminated pipe and steel cutting	Easy to use	Slow cutting times Cannot be used on large items
Impact Hammer	Concrete removal	Inexpensive Fast removal rates	HEPA filtration required to control airborne contamination Creates rubble

TABLE 2.3-4

THERMAL CUTTING/REMOVAL METHODS

METHOD	APPLICABILITY	ADVANTAGES	DISADVANTAGES
Plasma Arc	Tank segmentation Uncontaminated pipe cutting Structural steel cutting Contaminated piping (with appropraite radiological controls)	Fast cutting times Remote operation possible	HEPA filtration required to control airborne contamination from contaminated component removal Large power supply required
Oxy-Fuel	Tank segmentation Uncontaminated pipe cutting Structural steel cutting Contaminated piping (with appropraite radiological controls)	Easy to use Fast cutting times Remote operation possible	HEPA filtration required to control airborne contamination from contaminated component removal Ineffective on stainless steel Uses inflammable gases
Electric Discharge Machining (EDM)	Vessel segmentation	Capable of cutting thick steel High precision obtainable Remote operation possible	Slow cutting times Complex control system required
Flame Cutting	Concrete cutting	Capable of cutting thick concrete structures Low waste volumes	HEPA filtration required to control airborne contamination Difficult to use
Arc Saw	None at this time	Capable of cutting thick steel Remote operation possible	HEPA filtration required to control airborne contamination Large power supply required Heavy weight of unit Limited experience with unit

TABLE 2.3-5
DECOMMISSIONING SCHEDULE

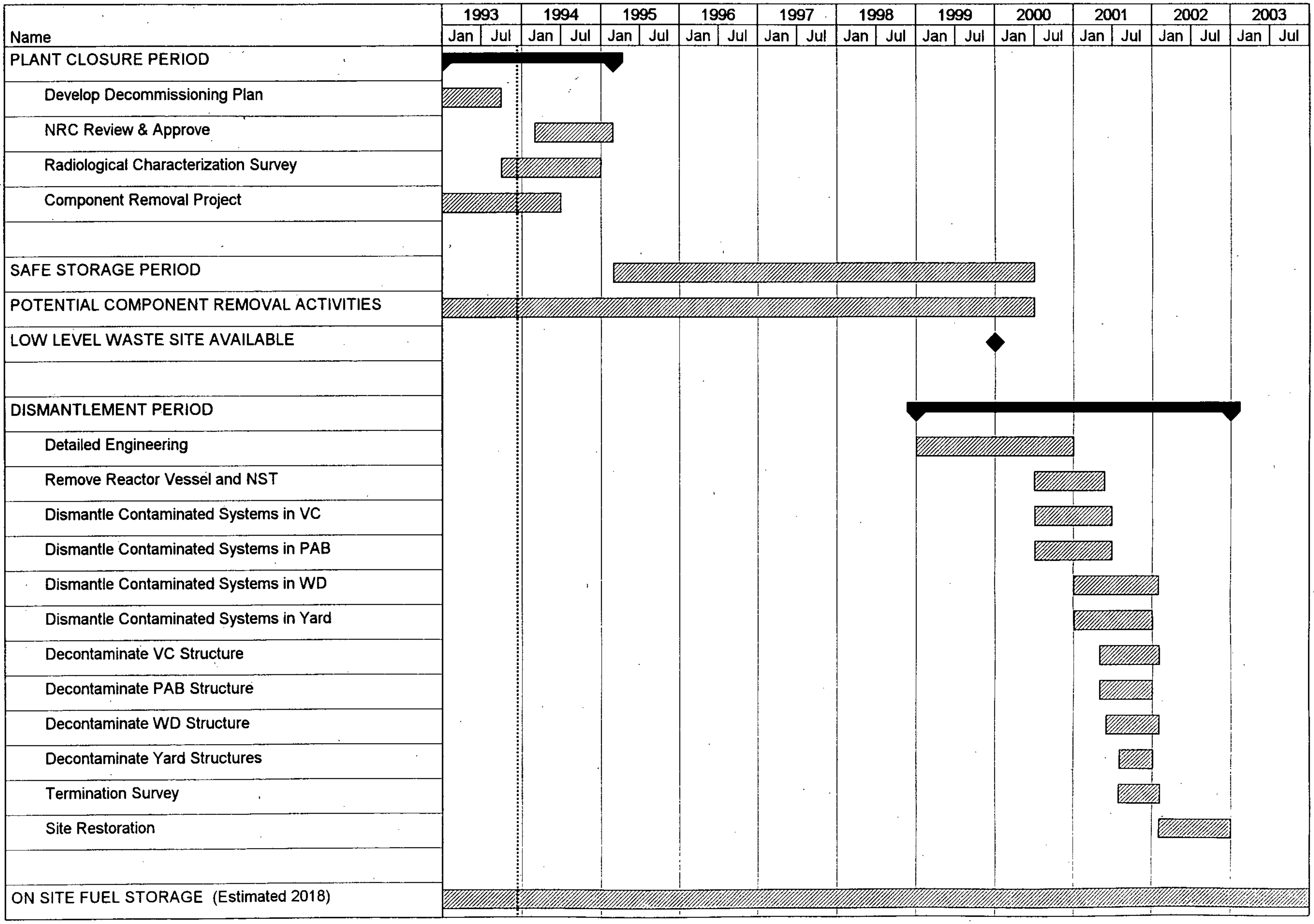


TABLE 2.3-6

RADIATION EXPOSURE PROJECTIONS

ACTIVITY	EXPOSURE (Person-rem)
COMPONENT REMOVAL PROJECT	
Asbestos Abatement	73
Steam Generators & Pressurizer	62
Reactor Vessel Internals	25
Subtotal:	**160**
FUEL TRANSFER	41
DISMANTLEMENT (1)	
Reactor Vessel	48
Main Coolant System	50
Other Systems in the Vapor Container	84
Balance of Plant Systems	98
Asbestos Abatement	90
Structures	50
Miscellaneous (e.g., packaging, inspections)	82
Subtotal:	**502**
TRANSPORTATION (2)	41
PLANT EFFLUENT	<1
TOTAL:	**744**

NOTES
1: Conservatively assumes all activities occur in 1994, maximizing exposure projections
2: Based on maximum allowable dose rates on packages

TABLE 2.3-7

DECOMMISSIONING WASTE CLASSIFICATION AND VOLUME PROJECTIONS

COMMODITY	10CFR61 CLASSIFICATION	BURIAL VOLUME (ft^3)	NUMBER OF CONTAINERS	NUMBER OF DRUMS	NUMBER OF CASKS
REACTOR VESSEL	A	4162	3		23
REACTOR VESSEL INTERNALS	>C	50	14		
FUEL RACKS	A	8478	24		
NEUTRON SHIELD TANK	A	1515	25		2
MAIN COOLANT PIPING AND SUPPORTS	A	2468	23		
MAIN COOLANT PUMPS	A	1040	8		
MAIN COOLANT VALVES	A	135	6.5		
PIPE, TUBING, AND SUPPORTS	A	13743	134.5		
VALVES	A	4036	41.5		
MECHANICAL EQUIPMENT	A	7512	66.5		
TANKS	A	4527	44		
DUCT AND SUPPORTS	A	2112	21.5		
HVAC EQUIPMENT	A	9189	85		
CABLE, CONDUIT, CABLE TRAY, AND SUPPORTS	A	6646	59		
ELECTRICAL EQUIPMENT	A	51	1		
CONCRETE	A	8502	170		
STRUCTURAL STEEL (Only 15% of total volume to be buried)	A	1324	66		
MISCELLANEOUS MATERIALS (E.G., TOOLS, EQUIPMENT)	A	7150	70		
DRUMS (SOLIDIFIED WASTE)	A	5730		764	
PCA WAREHOUSE No. 1 material	A	666	2	64	
TOTALS		**89,036**	**865**	**828**	**25**

TABLE 2.3-8

COMPONENT REMOVAL PROJECT WASTE CLASSIFICATION AND VOLUME PROJECTIONS

COMMODITY	10CFR61 CLASSIFICATION	BURIAL VOLUME (ft^3)	NUMBER OF CONTAINERS	NUMBER OF DRUMS	NUMBER OF CASKS
STEAM GENERATORS	A	7068			
PRESSURIZER	A	682			
REACTOR VESSEL INTERNALS	B,C	2110			24
ASBESTOS	A	600	6		
DRUMS (SOLIDIFIED WASTE)	A	555		75	
MISCELLANEOUS WASTES (e.g., containment materials, disposable protective clothing)	A	4720	47		
TOTALS		15,735	53	75	24

TABLE 2.3-9

WASTE SHIPMENTS

	Quantity	Shipments
Component Removal Project:		
Casks	24	24
Containers	53	11
Drums	75	3
Vessels	5	3
Subtotal:		41
Decommissioning:		
Casks	23	25
Containers	865	174
Drums	828	28
Vessels	0	0
Subtotal:		227
TOTAL:		268

Notes:

1. Shipment numbers were estimated using the following values from the cost estimate:

- 5 containers per shipment
- 30 drums per shipment
- 1 cask per shipment
- 2 Steam Generators per shipment

TABLE 2.3-10

DECOMMISSIONING WASTE BURIAL AND RECLAMATION VOLUME PROJECTIONS

COMMODITY	DECONTAMINATION AND RECLAIM Y/N	RECLAMATION VOLUME (ft^3)	FINAL BURIAL VOLUME (ft^3)	VOLUME REDUCTION (%)
REACTOR VESSEL	N	0	4162	0
REACTOR VESSEL INTERNALS	N	0	50	0
FUEL RACKS	Y	7206	1272	85
NEUTRON SHIELD TANK	N	0	1515	0
MAIN COOLANT PIPING AND SUPPORTS	Y	1728	740	70
MAIN COOLANT PUMPS	N	0	1040	0
MAIN COOLANT VALVES	N	0	135	0
PIPE, TUBING, AND SUPPORTS	Y	11682	2061	85
VALVES	Y	2018	2018	50
MECHANICAL EQUIPMENT	Y	3765	3765	50
TANKS	Y	3848	679	85
DUCT AND SUPPORTS	Y	1795	317	85
HVAC EQUIPMENT	Y	4595	4595	50
CABLE, CONDUIT, CABLE TRAY, AND SUPPORTS	Y	3323	3323	50
ELECTRICAL EQUIPMENT	Y	25	25	50
CONCRETE	N	0	8502	0
STRUCTURAL STEEL	Y	7500	1324	85
MISCELLANEOUS MATERIALS (E.G., TOOLS, EQUIPMENT)	Y	3575	3575	50
DRUMS (SOLIDIFIED WASTE)	N	0	5730	0
PCA WAREHOUSE No.1 material	N	66	600	0
TOTALS		**51,126**	**45,428**	**58**

FIGURE 2.3.1

TYPICAL V.C. SUPPORT COLUMN DETAIL

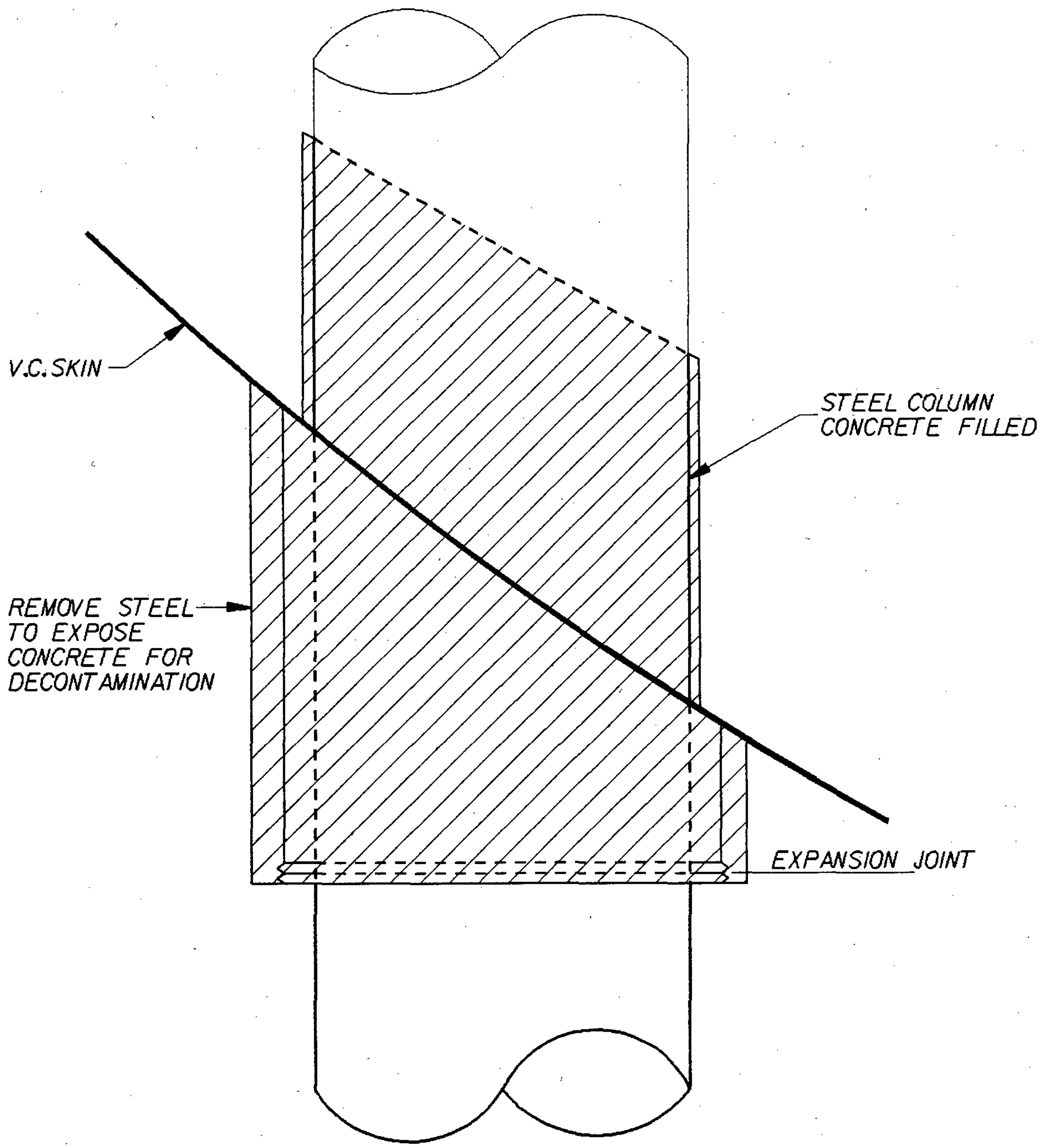


FIGURE 2.3.2

ACTIVATED CONCRETE REMOVAL ZONE

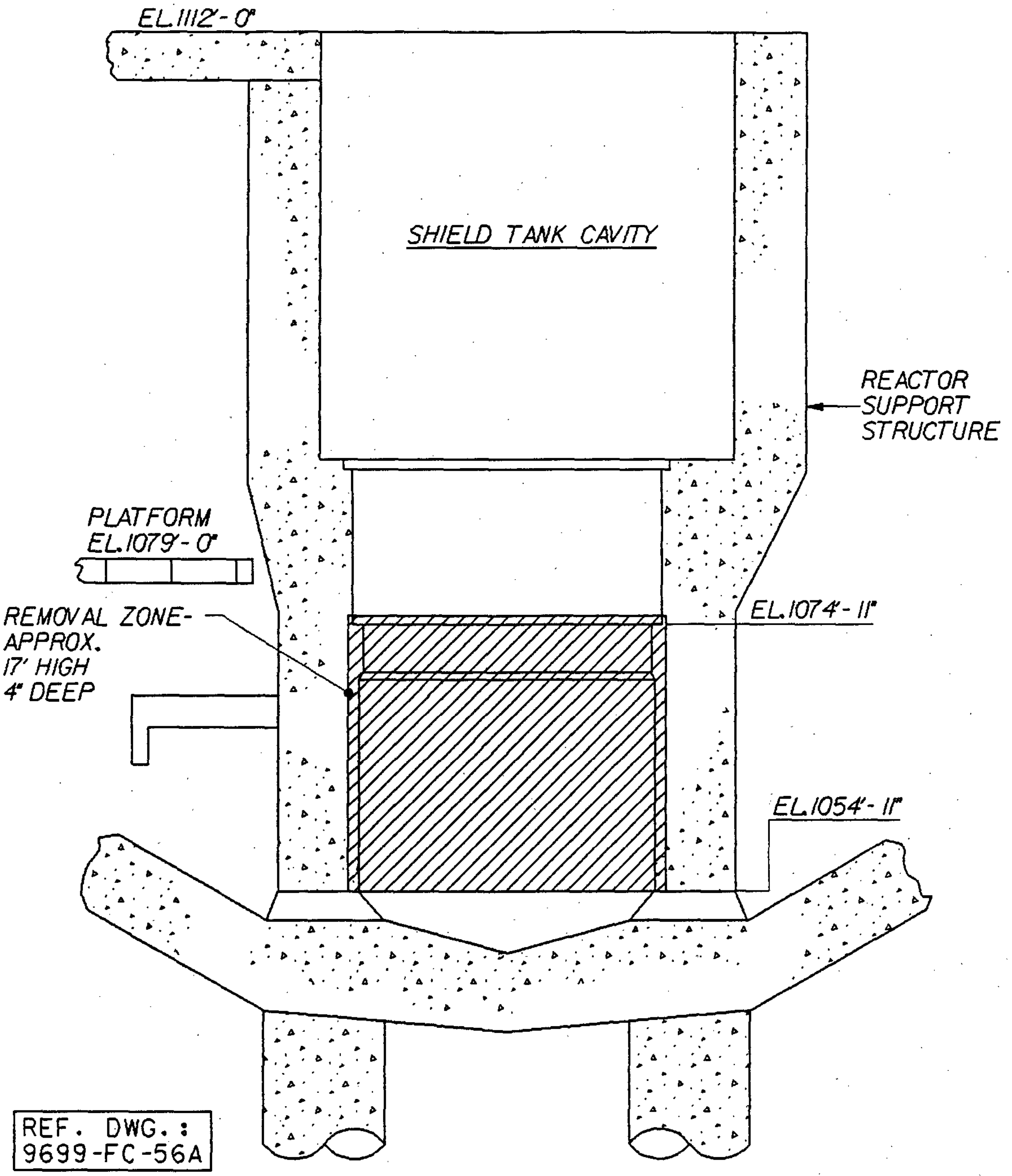
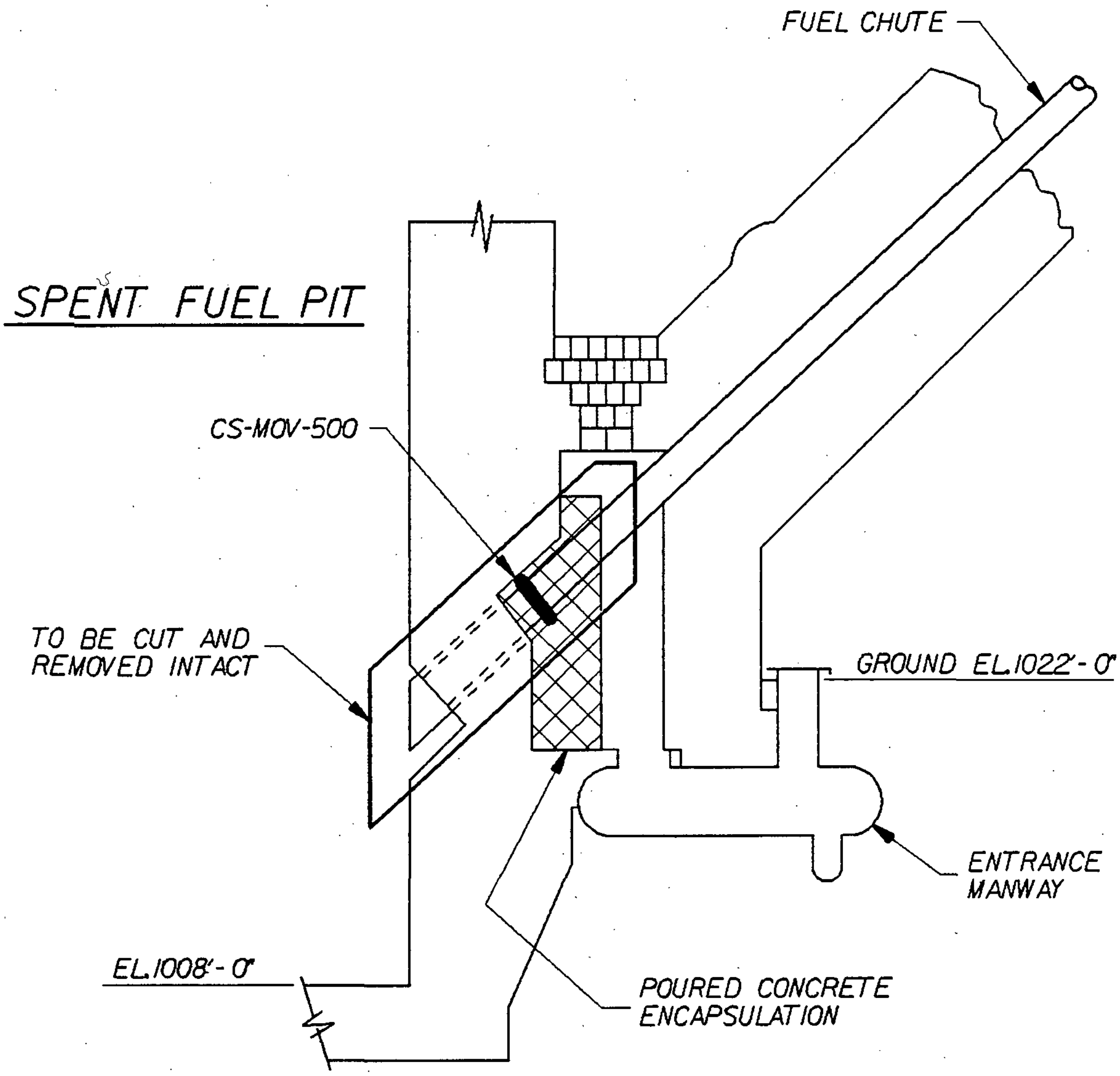


FIGURE 2.3.3

SPENT FUEL PIT AND FUEL CHUTE REMOVAL DETAIL



2.4 DECOMMISSIONING ORGANIZATION AND RESPONSIBILITIES

2.4.1 Yankee Atomic Commitment

Yankee Atomic Electric Company is committed fully to compliance with the existing license and applicable regulatory requirements during all phases of YNPS decommissioning. Yankee's commitment to the safe decommissioning of the facility will be accomplished with diligence and quality. Corporate principles, policies, and goals will be followed to ensure performance excellence, management competence, and high standards in every facet of the decommissioning.

2.4.1.1 Goals

The primary goals of the YNPS decommissioning are 1) to safely remove the nuclear facilities from service and to reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of the license and 2) to safely store spent nuclear fuel until it can be removed from the site. While achieving these primary goals, Yankee will conduct all decommissioning operations consistent with applicable regulations and a focus on the following considerations:

- Maintain radiation exposure to the public and on-site personnel as low as is reasonably achievable
- Ensure occupational safety for all on-site personnel
- Minimize environmental impact
- Minimize radioactive waste generation
- Ensure prudent expenditure of decommissioning funds

2.4.1.2 Organizational Strategy

The combination of Yankee's projected five year safe storage period and Yankee's desire to proceed with plant dismantlement as soon as a low level radioactive waste disposal facility becomes available requires establishment of a flexible organizational staffing strategy. The strategy must ensure that adequate numbers of experienced and knowledgeable personnel are available to perform the technical and administrative tasks required to decommission YNPS.

The YNPS decommissioning organization is currently staffed with experienced Yankee employees who have worked at and are very knowledgeable of the plant. The organization, initiated during the plant closure period, will remain in effect through the decommissioning. Yankee believes that retaining personnel with intimate knowledge of YNPS is important to the success of decommissioning.

The YNPS organization is supported technically and administratively by the Yankee Nuclear Services Division (YNSD) located in Bolton, Massachusetts. YNSD is a full service engineering organization which has supported YNPS operations since its start-up. YNSD also provides support services to several other nuclear power stations. YNPS, through YNSD, has access to significant in-house resources capable of responding to changing project conditions and schedules.

YAEC intends to be the prime contractor (Decommissioning Operations Contractor) responsible for YNPS decommissioning. In this position Yankee will have direct control and oversight over all decommissioning activities. This role is similar to that taken by YAEC during the 31 year operation of YNPS. In that role YAEC provided operational, technical, licensing, and project management support of YNPS. Yankee will contract services to supplement its capabilities as necessary.

Organizational impacts associated with the transition from plant closure to safe storage to dismantlement periods will be evaluated periodically to assure that adequate and appropriate staffing levels and capabilities are preserved. Yankee will maintain level staffing, to the extent possible, to optimize resource utilization and to capitalize on positive project momentum. Both of these are characteristics of successful project teams.

Technical Specification 6.3.1 requires that YNPS management personnel meet or exceed the minimum qualifications for education, training, and experience outlined in ANSI N18.1-1971, for comparable positions.

2.4.2 Yankee Organization and Functions During Safe Storage

The YAEC organizational structure that will be implemented during safe storage is presented in Figure 2.4-1. To the extent practicable, the initial safe storage organization will consist of staff, previously employed at YNPS, to capitalize on their detailed knowledge of and familiarity with YNPS. The staff may change in response to the activities being performed. For example, staff augmentation may be necessary to support component removal activities. Contractors may be utilized to provide specialized services or to supplement the decommissioning organization when warranted.

The YAEC Vice President and Manager of Operations is the corporate officer responsible for YNPS nuclear safety. This person may take any measures necessary to assure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant so that continued nuclear safety is assured (TS 6.2.1.b).

The on-site decommissioning organization is directed by the YNPS Superintendent who reports directly to the Vice President and Manager of Operations. The off-site decommissioning organization is directed by the YNSD Decommissioning Project Manager who reports directly to the Vice President and Manager of Operations. Quality assurance oversight of the decommissioning organization is directed by the Quality Assurance Department which reports directly to the YAEC President.

The Plant Superintendent has the overall responsibility for safe operation of the plant and has control over those on-site resources necessary to meet this objective (TS 6.2.1.c). Included in this responsibility is 1) assurance of safe, orderly, and efficient operation, and 2) compliance of operations with the requirements of the license, and applicable federal, state, and local laws and regulations. The YNPS Superintendent is the chairperson of the Plant Operation Review Committee and directs the on-site implementation of the Quality Assurance Program.

Three departments report to the YNPS Superintendent:

- <u>Administration Department</u> - The Administration Department provides overall management and planning of all plant administrative functions. In addition, the department also coordinates site security, health and safety, and stores functions.

- <u>Operations and Technical Department</u> - The Operations and Technical Department ensures safe operation of YNPS and effective implementation of the Radiation Protection Program (Section 3.2). In addition, the department also coordinates on-site training, special nuclear materials, licensing, hazardous waste, fire protection, and emergency planning programs.

- <u>Maintenance Department</u> - The Maintenance Department maintains plant equipment, implements the plant lay-up program, and implements the preventive maintenance program. The department is comprised of maintenance engineering and mechanical, electrical, instrumentation, and control maintenance functions.

Daily YNPS operations are directed by the Shift Supervisor, who is responsible for safety, security, and proper operation. The Shift Supervisor is the direct representative of the YNPS Superintendent for all matters related to operation of the plant. The Shift Supervisor is required to possess and maintain a Certified Fuel Handler qualification.

The YNSD Decommissioning Project Manager directs the engineering, licensing, and planning activities associated with the YNPS decommissioning. The project manager position is closely integrated with the YNPS Superintendent position to assure effective development and implementation of engineering and licensing activities. The project manager is responsible for coordinating YAEC resources to meet YNPS needs.

2.4.3 Organization and Functions During Dismantlement

Near the end of the safe storage period, the organization in place during that period will be augmented with additional management and staff personnel. Figure 2.4-2 presents the organizational structure for the dismantlement phase of YNPS decommissioning. The structure reflects addition of a YNPS Decommissioning Department. This organizational structure is similar to that used for the Component Removal Project.

The YNPS Decommissioning Department is responsible for ensuring the safe and effective decommissioning of YNPS. The interface between the department and the balance of the YNPS organization is a matrix relationship. Therefore, the decommissioning organization will have direct responsibility for dismantling the facility, while utilizing the existing plant organization for those capabilities and support services already established and functioning during the safe storage phase. This approach is intended to enhance overall project and organizational effectiveness following transition from the safe storage phase.

The YNPS Decommissioning Department will incorporate the following functions:

- Construction Services - The Construction Services function will ensure effective management of the daily activities of the on-site project team, craft personnel, and subcontractors.

- Project Controls - The Project Controls function will control the decommissioning cost and schedule control. This will be accomplished through establishment of a budget and schedule control system. The function also assures effective implementation of contracts executed to support decommissioning activities.

- Engineering Services - The Engineering Services function will provide on-site engineering support of decommissioning activities. This includes interaction with specialty contractors and development of procedures to support decommissioning activities.

The YNPS Decommissioning Department staffing levels will be reviewed periodically to ensure that adequate staffing levels are maintained consistent with current and planned decommissioning activities. Staffing levels will be adjusted as necessary to ensure that the activities are implemented effectively.

2.4.4 Safety Review Committees

Technical Specification 6.5.1 establishes a Plant Operation Review Committee (PORC) to advise the YNPS Superintendent on all matters related to nuclear and decommissioning safety. The PORC is composed of a chairperson and a minimum of four management or supervisory staff members appointed by the YNPS Superintendent. The PORC is responsible for the review of proposed changes to the technical specifications or license; all reportable events; proposed changes or modifications to plant systems, equipment, and procedures that could affect nuclear safety.

Technical Specification 6.5.2 establishes a Nuclear Safety Audit and Review Committee (NSARC) to provide an independent review and audit of all aspects of plant safety. The NSARC is composed of at least four persons with the committee membership and its chairperson appointed by the Manager of Operations. The NSARC is responsible for reviewing those items referenced by Technical Specification 6.5.2.8.

Figure 2.4-1
ORGANIZATIONAL STRUCTURE: SAFE STORAGE

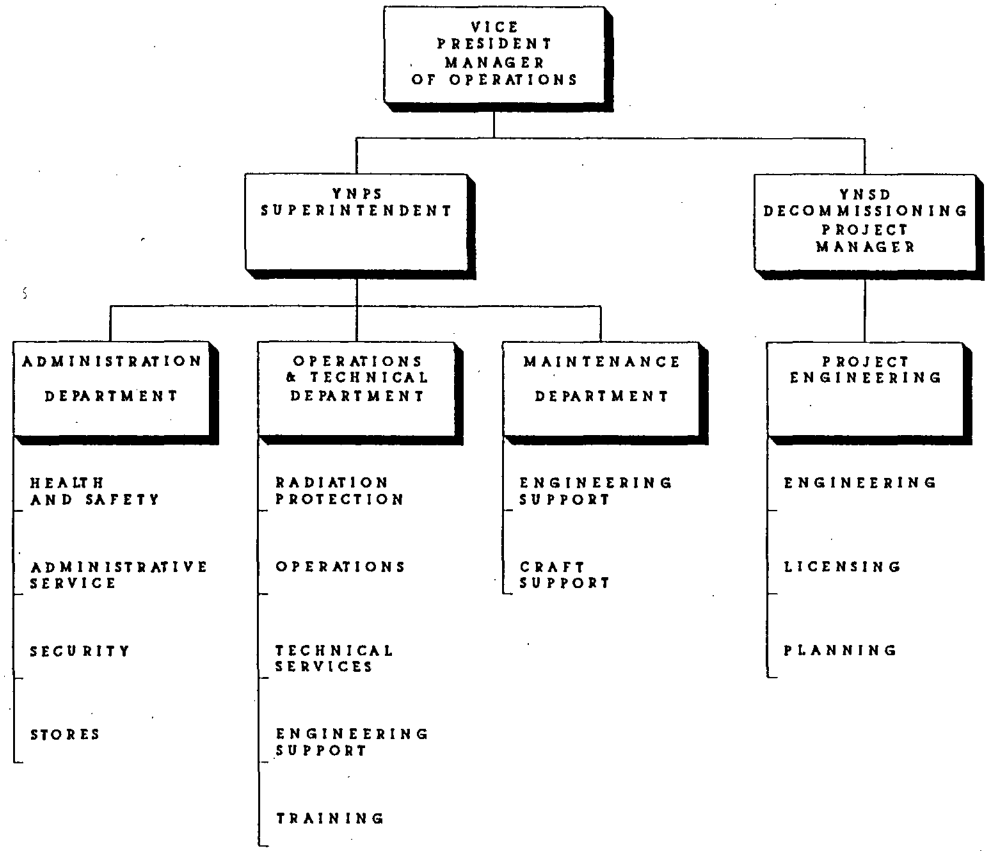
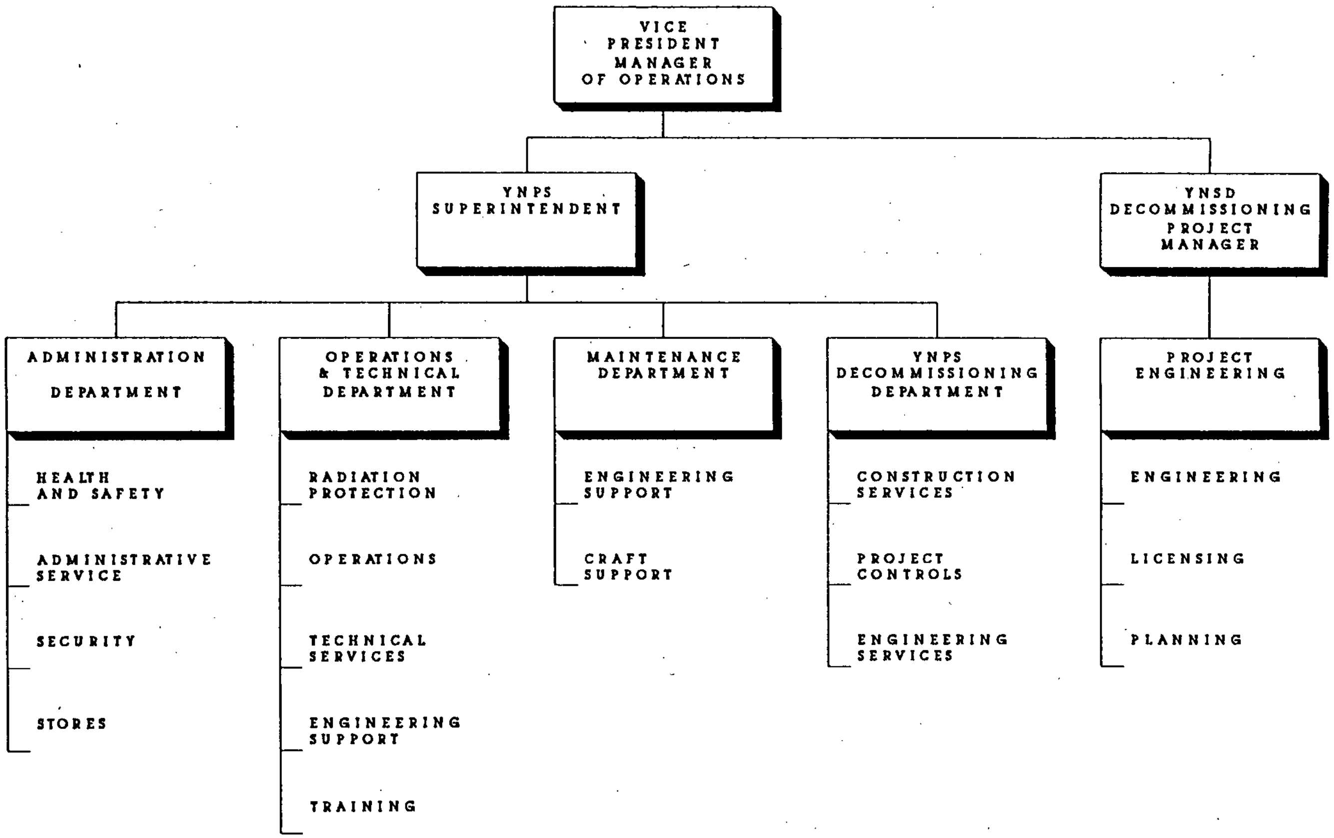


Figure 2.4-2
ORGANIZATIONAL STRUCTURE: DISMANTLEMENT



2.5 CONTRACTOR RESPONSIBILITIES

2.5.1 Use of Contractors

As the Decommissioning Operations Contractor, Yankee intends to manage safe storage and dismantlement activities. Consistent with past practices at YNPS, contractors will be used on an as-needed basis during the decommissioning. The use of contractors will be complementary in nature, including specialty support and staff augmentation.

2.5.2 Contractor Scope of Work

Tasks where specialty contractors may be utilized in support of decommissioning include but are not limited to the following:

- Packaging, transportation and disposal of radioactive materials
- Decontamination and recycling of radioactively contaminated materials
- Design and fabrication of specialty dismantling tooling and equipment
- Asbestos removal and disposal
- Health physics and radiological staff augmentation
- Specialty engineering and design services such as heavy loads management and transportation engineering

2.5.3 Contractor Qualifications and Experience

Each contractor used at YNPS will be evaluated to ensure that they have the appropriate qualifications for the tasks that they will perform. Contractors will be evaluated based on the following criteria: technical and operational capability, cost and schedule compliance, demonstrated experience in providing quality services on similar projects, and ability to meet regulatory requirements.

2.5.4 Contractor Administrative Controls

Yankee will retain responsibility for the performance of all contractors during decommissioning. Yankee will also provide the necessary management oversight to assure that tasks performed by the contractors are in full compliance with the Quality Assurance Program, the purchase agreement, and applicable regulatory requirements.

2.6 TRAINING PROGRAM

Yankee will maintain a training program commensurate with the needs of the various phases of decommissioning and the Training Rule 10 CFR 50.120. The training program, in conjunction with other administrative programs and controls, will ensure that qualified individuals are available to operate and maintain the facility in a safe manner. The training programs will be based on a systematic analysis of job performance requirements. The analysis will ensure that personnel will have qualifications commensurate with the performance requirements of their jobs.

All Yankee and contractor personnel and visitors, who require access to work areas in the Radiation Control Area will receive appropriate training commensurate with the potential hazards to which they may be exposed.

2.6.1 General Employee Training

General Employee Training will be provided to all personnel who have unescorted access to the YNPS industrial area. Training will be completed in accordance with Plant Procedure AP-0501, "General Employee Training Program."

Initial training will include the following topics:

- Plant organization and administration
- Plant description
- Occupational safety
- Quality assurance
- Fire Protection
- Emergency response
- Radiation protection
- Security

Periodic requalification training will be administered to plant personnel and contractors assigned to the plant on a long-term basis. The requalification training will include the following:

- Changes to the plant policies and procedures
- Relevant plant and industry operating experiences
- Annual training to fulfill regulatory requirements

2.6.2 Radiation Worker Training

Initial radiation worker training will be provided to personnel who enter the Radiation Control Area. Initial radiation worker training includes topics such as the following:

- Fundamentals of radiation
- Radiation and contamination measurement and control
- Maintaining radiation dose as low as is reasonably achievable
- Radioactive waste minimization
- Radiation work permits
- Radiation protection issues associated with decommissioning

In addition to classroom training, participants in the initial radiation worker training will receive practical abilities training including the following:

- Donning and removing protective clothing
- Reading and interpreting direct reading and electronic dosimetry
- Entering and exiting contaminated areas, including frisking techniques
- Preparation and implementation of a Radiation Work Permit

Periodic requalification training will be administered in conjunction with the General Employee Training requalification training.

2.6.3 Certified Fuel Handler

YNPS has implemented, with NRC review and approval, a Certified Fuel Handler training program which replaced the 10 CFR Part 55 NRC licensed operator training program (Reference 2.6-2). The program includes provisions for training, proficiency testing, certification and recertification of the Certified Fuel Handler position. Certified Fuel Handlers are required by Technical Specifications to be on site at all times as part of the minimum shift crew composition, with additional staffing required for supervising fuel handling operations.

2.6.4 Specific Job Training

YAEC has evaluated the training requirements of 10 CFR 50.120 relative to the YNPS decommissioning. Based on the evaluation, YAEC determined that the Shift Technical Advisor training program was not applicable to YNPS after fuel permanently removed from the Reactor Vessel. Yankee has requested an exemption from the 10 CFR 50.120 Shift Technical Advisor training requirements (Reference 2.6-2). The balance of the YNPS training programs will be based on a systematic analysis of job performance requirements using Plant Procedure AP-0550, "Performance Based Training Program."

YNPS training programs will be developed to assure the following:

- Personnel responsible for performing activities are instructed as to the purpose, scope, and implementation of applicable controlling procedures

- Personnel performing activities are trained as appropriate, in the principles and techniques of the activity being performed

- The scope, objectives and methods of implementing the training programs are documented

2.6.5 Non-Radiation Worker Indoctrination

Personnel who are not qualified radiation workers may be granted limited access to the Radiation Control Area through an exemption process in Plant Procedure AP-0809, "Requirements for Radiation Control Area Access and Egress." All personnel without radiation worker training require escorts that are qualified radiation workers. The escort is responsible for assuring adherence to plant policies and radiation protection practices.

2.6.6 Training Staff Qualifications

Plant Procedure AP-0512, "Instructor Training and Qualification," presents qualification requirements for the YNPS training staff. This procedure presents both initial and continuing training requirements for instructors.

2.6.7 <u>Training Records</u>

Training records will be maintained in accordance with Plant Procedure AP-0508, "Maintenance of Training Records." These records will be retained in accordance with the requirements in Plant Procedure AP-0221, "Plant Records Management."

REFERENCES

2.6-1 BYR 93-055, Implementation of the Training Rule, 10 CFR 50.120, J. K. Thayer to M. B. Fairtile (USNRC), July 28, 1993.

2.6-2 NYR 92-122, NRC Approval of Certified Fuel Handler Program and Termination of Operator Licenses (TAC No. M83384), M. B. Fairtile (USNRC) to J. M. Grant, June 26, 1992.

PROCEDURE REFERENCES

AP-0221 "Plant Records Management"

AP-0501 "General Employee Training Program"

AP-0512 "Instructor Training and Qualification"

AP-0550 "Performance Based Training Program"

AP-0809 "Requirements for Radiation Control Area Access and Egress"

SECTION 3
PROTECTION OF OCCUPATIONAL AND PUBLIC HEALTH AND SAFETY

3.1 FACILITY RADIOLOGICAL STATUS

3.1.1 Facility Operating History

Yankee Nuclear Power Station achieved initial criticality in 1960 and began commercial operations in 1961. On October 1, 1991, YNPS shut down after operating 31 years. On February 26, 1992, the YAEC Board of Directors decided to cease power operations permanently. YNPS operated with an average lifetime capacity factor of about 74%. Table 3.1-1 presents a summary of the YNPS operating history.

There were occurrences, during the operation of YNPS, that resulted in contamination of structures and components inside buildings located in the Radiation Control Area. In addition, several occurrences resulted in the contamination of the grounds outside of the buildings but inside the Radiation Control Area. Most of these events were minor, resulting in minimal contamination. Following detection of an event, actions were taken to remove and control contamination and to institute corrective actions to preclude future occurrences.

Appendix B presents a summary of significant radiological contamination occurrences at YNPS. The primary sources used to compile and to review events were: Control Room Logs; Abnormal Occurrence Reports; Licensee Event Reports; Plant Information Reports; and reports to the Atomic Energy Commission, Nuclear Regulatory Commission, and the Commonwealth of Massachusetts. Interviews were also conducted with present employees, recently discharged employees, and retired long-term employees. The purpose of the interviews was to gain additional information concerning plant occurrences and operations that may have resulted in residual contamination.

The most significant contamination event at YNPS was leakage from the Ion Exchange Pit. Significant leakage was first identified in May 1964. An unsuccessful attempt to isolate the leak was made in July 1964 by installing and sealing a concrete plug in the pit sump. Following that attempt, a decision was made to empty the resin capsules and to drain the pit. This activity was completed in April 1965. In May 1965, a crack in a vertical joint at the northwest corner of the Ion Exchange Pit was found after draining the pit. The crack was repaired and the Ion Exchange Pit floor and walls were sealed to prevent further leakage.

In 1965, tritium was detected in Sherman Spring. The presence of tritium was attributed to migration of tritium from the Ion Exchange Pit into the groundwater. At the present time, the tritium concentration is significantly below the Environmental Protection Agency community water system limit (Section 3.1.5). In addition, the water from Sherman Spring, as well as the Deerfield River, into which it flows, is not used for human consumption.

Systems in contact with the main coolant have been contaminated with activated corrosion products and fuel residue. In 1977, YNPS began converting from stainless steel clad fuel to zircaloy clad fuel. Following conversion, fuel clad failures began occurring. Most of the failures were minor failures of clad integrity, releasing iodines and other fission gasses into the main coolant. However, during Cycles 14, 16, and 18 several peripheral fuel rods failed as a result of damage caused by water jetting from the core baffle spacer plates. These fuel failures were sufficient to cause contamination of the Reactor Vessel and Main Coolant System with fuel residue. Baffle jetting damage was eliminated by the addition of spacer plugs at the bottom of the core baffle spacer plates and modifying fuel assembly design. No significant fuel failures occurred after the modifications were implemented.

Primary system integrity was very good during plant operations. Minor leakage from Main Coolant System flanges and valve stems to the Vapor Container occurred during plant operations. However, leakage rates were maintained below technical specification limits. The Vapor Container was decontaminated during refueling outages to remove contamination that may have accumulated during previous operating cycles. The YNPS steam generators performed well during the 31 year operating life. There was no significant leakage between the primary and secondary sides of the generators.

The results of the historical review were incorporated into the design and implementation of the radiological scoping survey and sampling program.

3.1.2 Radiological Scoping Survey

3.1.2.1 Scoping Survey Basis

NUREG/CR-5849 (Reference 3.1-1) presents guidance for implementing radiological surveys during the decommissioning of nuclear facilities. One of the surveys identified in this document is a scoping survey, which provides "the basis for initial estimates of the level of effort required for decommissioning and for planning the characterization survey." Yankee completed a scoping survey to provide the basis for development of this decommissioning plan.

Yankee's scoping survey consisted of measurements and samples of the plant systems, structures, and components and of the soil and groundwater beneath the plant site and surrounding areas. The objective of the scoping survey was to provide a preliminary assessment of site conditions and to classify the site into radiologically affected and unaffected areas. The objective was achieved by developing a survey method that, when combined with analytical methods, achieved the following goals:

- Identification of plant areas affected and not affected by radiological contamination and activation

- Identification of radionuclide contaminants located on site

- Determination of the distribution of radionuclides in contaminated and activated materials

- Determination of the general extent of radiological contamination and activation - both in terms of activity and volume

The scoping survey sample location selection process identified both biased and unbiased locations. Biased sampling locations were areas of suspected contamination that were identified using information obtained from plant records and employee interviews. A grid system was established for systematically identifying unbiased sampling locations in areas known to be contaminated and in areas of potential contamination.

3.1.2.2 Scoping Survey Procedures

Several plant procedures were developed to implement, to document, and to ensure the completion of an accurate and reliable scoping survey:

- AP-0830: "Administrative Program For The Radiological Scoping Survey To Support The YNPS Decommissioning Plan" - This procedure is the implementing procedure for the YNPS radiological scoping survey. It presents a general overview of the scoping survey and administrative requirements for implementation of the survey.

- DP-8120: "Collection of Surface Soil and Asphalt Samples" - This procedure presents implementing instructions for the collection of near-surface and asphalt grab samples.

- DP-8121: "In-Plant Radiological Surveys To Support The Scoping Program" - This procedure presents implementing instructions for radiological surveys and sampling of YNPS systems, structures, and components.

- DP-8122: "Subsurface Soil Sampling and Monitoring Well Installation" - This procedure presents implementing instructions for the installation of monitoring wells and the collection of subsurface soil samples during the installation of the monitoring wells.

- DP-8123: "Sample Chain of Custody" - This procedure presents implementing instructions to ensure sample integrity throughout all phases of sample collection, analysis, and retention.

- DP-9745: "Ground Water Level Measurement and Sample Collection in Observation Wells" - This procedure presents implementing instructions to obtain groundwater characterization data.

In addition to these specific procedures, many other plant radiation protection procedures and laboratory analytical procedures were implemented to complete scoping survey data gathering and analysis activities.

The scoping survey procedures incorporate the following administrative requirements to assure the quality of the scoping survey:

- Equipment and techniques for measuring, sampling, and analyzing data

- Types, numbers, and locations of measurements and samples

- Quality control and audit requirements

- Personnel qualification

- Sample chain of custody requirements

3.1.2.3 Summary of Scoping Survey

Surveys completed inside the facility consisted of the following: 1) area and contact dose measurements of contaminated and uncontaminated systems, structures, and components; 2) samples of plant piping; and 3) samples of paint and concrete. A list of potentially contaminated systems was developed and a survey package was developed for each system evaluated. The survey packages included drawings, system descriptions, and lists of piping and components. Potentially uncontaminated systems were also surveyed to verify their radiological status.

Surveys completed outside of the facility buildings consisted of the following: 1) samples of soil and groundwater from test wells, 2) samples of asphalt and near-surface soil, and 3) surface gamma spectroscopy. Soil and asphalt covered areas were analyzed to estimate the extent and distribution of contamination that occurred from plant operation.

Groundwater samples were analyzed to determine if contamination was transported into the groundwater.

Personnel performing the surveys received training to qualify them in the procedures being performed. The extent of the training was commensurate with the education and experience of the individual and the scope and complexity of the task. A significant portion of the surveys were performed by trained and qualified YNPS Radiation Protection Technicians under the direction of the Site Characterization Coordinator.

Samples collected during the scoping survey were evaluated by the Yankee Atomic Environmental Laboratory and the Yankee Plant Chemistry and Radiation Protection Laboratories. Instrumentation used in the field and laboratory were calibrated against sources and standards that were traceable to the National Institute of Standards and Technology (NIST). The calibration sources and standards that were used contained the nuclides or representative nuclide mixes encountered at the site. Calibrations were conducted using industry recognized standards (e.g. ANSI, NCRP, and INPO) using approved procedures.

A chain of custody process was established to ensure sample process integrity. The chain of custody provided an accurate record of the sample collection, transport, analysis and disposal. This ensured that the sample analyzed in the laboratory was actually the sample taken from a specific location in the field. Sample custody was assigned to one individual at a time, preventing confusion of responsibility.

A computerized database was developed to process the results from the scoping survey. Use of the database facilitated data storage and retrieval. The radiological survey data was integrated with physical data for on-site systems, structures, and components to calculate the volume and activity of radiological materials present on site. The database will continue to be extended to support site characterization and final survey activities.

The Quality Assurance Department independently reviewed selected portions of the survey results. The review included evaluation of field and analytical data and a review of the data analysis methods.

3.1.3 Scoping Survey Results: Systems, Structures, and Components

This section presents the results of radiological scoping survey of the YNPS systems, structures, and components. The survey was conducted inside and outside the Radiation Control Area. The data collection consisted of samples of components and structures, radiation readings, and contamination measurements. Additionally, an activation analysis of neutron irradiated components was completed.

3.1.3.1 Area Survey Maps

Figures 3.1-1 through 3.1-25 summarize the current (early 1993) radiological status of YNPS. Survey maps are presented for both contaminated and uncontaminated areas of the plant. Each map presents general area dose rates and average removable surface contamination levels. Contact radiation levels have been included if they affect the local general area readings.

Routine radiological surveillances will continue to be conducted during decommissioning to monitor radiation sources, to determine radiological conditions, and to comply with the requirements of 10 CFR Part 20. Surveys will be conducted in the Vapor Container following completion of the Component Removal Project to determine the radiological condition after project completion. Surveys also will be performed to evaluate radiological conditions in support of decommissioning work activities.

3.1.3.2 Radionuclide Distribution

Sections of pipe were removed from three plant systems as a part of the radiological scoping survey: the Safety Injection Pump Discharge Header, the Number 2 Charging Pump Discharge Line, and the Bleed Line Header in the Valve Room. These samples are representative of a wide range of process conditions and contamination levels at YNPS. Direct internal and external radiation measurements were made on each pipe sample. A section of the internal surface of each pipe sample was wiped to collect the removable activity and acid-etched to collect the fixed activity. The acid-etching process removed essentially all activity from the pipe surface. The smear and acid-etched samples were analyzed to determine the radionuclide distributions. A complete analysis of alpha, beta, and gamma emitting nuclides was performed on several of these samples.

Evaluation of the pipe sample data indicated that between 20% to 50% of the total activity removed from the internal surface of the pipe samples was removed by wiping or smearing the surface (Reference 3.1-2). This indicates that between 50% to 80% of the contamination on the internal surfaces of YNPS piping is tightly bound and cannot be removed by wiping the surface.

The radionuclide distribution data were evaluated to determine the variability of the radionuclide distribution between plant systems and to compare the results with previous plant data. The radionuclide distribution data from all samples and previous data were similar, indicating that the results were representative of a single nuclide distribution. This conclusion enabled the scoping results to be combined with other plant radionuclide distribution data into a single radionuclide distribution.

The combined radionuclide distribution was used for evaluations of all contaminated

YNPS systems, structures, and components with the exception of the Main Coolant System and adjoining systems. A specific Main Coolant System distribution was used for these systems. The Main Coolant System distribution is consistent with the distribution used to characterize the steam generators and the pressurizer for the Component Removal Project. Use of a separate Main Coolant System distribution results in a more conservative accident analysis. A release of airborne radioactivity with the Main Coolant System distribution results in a higher off-site dose compared to an equivalent release of material with the combined radionuclide distribution.

Two radionuclide distributions are presented in Table 3.1-2. One distribution is applicable for the Main Coolant System and adjoining systems. The other distribution is applicable for the balance of plant systems, structures, and components. Table 3.1-3 presents the projected radionuclide distribution for the balance of plant contamination from 1994 to 2000.

3.1.3.3 Contamination Results

Contamination levels for plant systems and components were estimated using several methods: direct measurement, comparison with adjacent or similar systems, and calculation from external dose rate. Table 3.1-4 presents the system average contamination levels. External contamination levels are presented for those systems without significant internal contamination.

Internal contamination levels were calculated for several systems and components based on external dose rate data. The calculation used a matrix of dose rate conversion factors based on radiation shielding calculations for selected geometries (Reference 3.1-3). Use of the dose rate conversion factor is based on the assumption that external dose rates are due primarily to cobalt-60 deposited in the contamination layer on the inside of the component. The radionuclide distribution data indicate that cobalt-60 is the most significant contributor to the gamma activity (77% of total gamma activity). The total internal contamination level was determined by scaling individual radionuclides to cobalt-60 using the radionuclide distributions described in Section 3.1.3.2.

System activity inventories were calculated by combining the average predicted contamination level with surface area data from a plant physical inventory database. Components with activity contents that were not representative of the average system contamination level (e.g., Low Pressure Surge Tank, Activity Decay and Dilution Tank) were estimated using survey data and individual shielding calculations (Reference 3.1-2). These values were input separately into the database for each affected component.

Contamination levels for plant structures were estimated from contamination data and samples of contaminated paint and concrete. Paint and paint/concrete samples were obtained from various surfaces by using a small chiseling tool, which removed a thin layer of material. Typically, a 100 cm^2 area was sampled. These samples were weighed and then analyzed with a Geiger-Mueller (GM) detector.

Three inch long core bore samples were obtained from various concrete floors and walls primarily within the Radiation Control Area. Several samples were also taken outside of the Radiation Control Area. The core bore samples were sliced into thin wafers about 5 mm thick. Adjacent wafers were analyzed for gamma emitting radionuclides to estimate the depth of the contamination in each core bore.

Table 3.1-5 summarizes results of the concrete core bore analyses. The Vapor Container sample locations indicated that contamination was limited to the first slice (<5 mm). The Brass Drain Box floor and the Waste Disposal Building floor had contamination down to the second slice (<15 mm). The Evaporator Cubicle floor had contamination down to the third slice (<32 mm).

Table 3.1-6 presents concrete contamination data for plant structures based on radiological scoping surveys and concrete core bore data. The depth of contamination estimates were based on samples from nearby or similar areas of the plant. The depth of contamination in the concrete walls and floors of the following inaccessible areas were estimated: Fuel Transfer Chute (4 inches), Spent Fuel Pit (4 inches), Ion Exchange Pit (6 inches). The contamination level data were combined with the structural inventory database to estimate the activity inventory.

Structural steel contamination levels were estimated based on radiological scoping survey results. The contamination level data were combined with the structural inventory database to estimate the activity inventory.

3.1.3.4 Activation Analysis

Neutron activation of the Reactor Vessel, the Neutron Shield Tank, and the bioshield wall were estimated analytically. Three-dimensional neutron activation estimates were derived based on synthesis of two-dimensional neutron transport calculations using fine detailed R-θ and R-Z geometries (Reference 3.1-4). The DORT discrete ordinates neutron transport computer program in conjunction with the SAILOR cross section library were used to determine estimates of neutron flux and 47-group spectra for each structural region. The ORIGEN-II computer program and associated activation libraries were used to calculate neutron activation by isotope from the neutron flux data, reactor operating history, and material properties of the individual components. A simplified weighing of one group cross sections based on a pressurized water reactor fuel spectrum and a thermal spectrum was used to provide region average activation estimates for each component region. Material impurities were taken from NUREG/CR-3474 (Reference

3.1-5) which provided average values of component material compositions based on measured samples from other nuclear plants.

DORT calculations were also performed in R-θ geometry using a 27-group library collapsed from the 218-group SCALE library so that spectrum weighted one group cross sections for key activation reactions (e.g., cobalt-59 (n,γ)) could be determined for all component structures. This calculation was performed to determine an approximate error band associated with using a simplified spectral weighing scheme to determine the activities for each of the structural regions of interest. The simplified spectral weighted values for cobalt activation exceeded the more discrete estimate by up to a factor of two in the Reactor Vessel wall and bioshield wall. In these regions the neutron spectrum was shifted significantly to higher energies (i.e., harder spectrum). However, the two methods compared within about 25% for cobalt activation in the balance of the regions.

Table 3.1-2 presents radionuclide distribution estimates for the Reactor Vessel, Neutron Shield Tank, and bioshield wall in the regions adjacent to the reactor core. Table 3.1-7 and 3.1-8 present the radial and axial radionuclide concentration estimates for these components. The Reactor Vessel wall (stainless steel cladding and carbon steel base metal) and the Neutron Shield Tank inner wall (carbon steel) are estimated to be significantly activated. However, the water in the Neutron Shield Tank provided adequate shielding to reduce the activation of the outer Neutron Shield Tank wall by about five to six orders of magnitude. The Neutron Shield Tank also shielded the bioshield wall, which is only mildly activated. Concrete removal estimates are based on removing about four inches of activated concrete from the bioshield wall. Final removal depth will be determined based on core samples taken after Reactor Vessel removal.

Table 3.1-9 presents an estimate of Reactor Vessel contact and stand-off dose rates resulting from neutron activation of the vessel. Reactor Vessel activation results in a dose rate in 2000 of about 1.4 R/hr at a distance 2 meters from the external surface (Reference 3.1-6). Preliminary verification of this value has been made through limited external dose rate measurements through the Neutron Shield Tank neutron detector wells. These will be verified through more extensive measurements taken inside the Reactor Vessel after the Component Removal Project is completed.

Two additional samples were obtained and analyzed for neutron activation: a diaphragm from the manway hatch at the top of the Vapor Container and a section from the missile shield above the Reactor Vessel. These samples were decontaminated extensively before analysis. A direct measurement of the samples with a GM detector and smears counted on a proportional counter indicated no positive activity. Subsequent gamma spectroscopy analysis indicated the presence of cobalt-60. Further analyses and samples will be obtained during decommissioning to determine the extent of activation of components external to the Reactor Vessel.

3.1.3.5 Facility/Component Radionuclide Activity Inventory

An estimate of the YNPS systems, structures, and components activity inventory is presented in Table 3.1-10. The inventory includes all materials except radioactive materials (boxes and drums) awaiting shipment in the warehouse, 533 spent fuel assemblies, and the greater than 10 CFR Part 61 Class C Reactor Vessel internal components stored in the Spent Fuel Pit.

3.1.4 Scoping Survey Results: Soil and Groundwater

This section presents the results of radiological scoping surveys of the soil, asphalt, and groundwater (Reference 3.1-7). The surveys were conducted inside and outside the Radiation Control Area. Soil samples were collected in both surface and subsurface locations. Groundwater samples were collected from on-site observation wells. All of the samples were evaluated for gamma emitting radionuclides using HPGe or GeLi detectors. The groundwater samples were also analyzed for beta activity and tritium concentrations. The detection capabilities used for the analyses were based on the environmental lower limits of detection presented in the YNPS Off-Site Dose Calculation Manual (Reference 3.1-8). Soil was also evaluated using in situ gamma spectroscopy and pressurized ion chamber measurements.

A grid system was established for systematically identifying unbiased sampling locations. This system used a 10 and 20 meter grid spacing for areas inside and outside of the Radiation Control Area, respectively. Biased sampling locations were added in areas of suspected contamination.

3.1.4.1 Surface Soil and Asphalt Sampling

Near-surface samples of soil and asphalt were collected from 79 locations on and around the YNPS site as a part of the radiological scoping survey (Figures 3.1-26 and 3.1-27). Most locations were based on a sample grid, although several were biased (i.e. collected in areas of suspected contamination). Hand tools were used to sample soil to a depth of about six inches. Any layer covering the soil also was collected as a separate sample (e.g., asphalt, sod including matted root and dirt, humus). Table 3.1-11 summarizes the distribution of the 133 samples that were collected. A description was recorded of the conditions found at each sample location.

The results of the radiological analyses of the soil and asphalt samples are presented in Tables 3.1-12 through 3.1-14. All samples were analyzed without drying. Wet-to-dry ratios were also determined for most of the soil samples to allow conversion of data from wet to dry concentrations.

Cesium-137 was detected in many soil and asphalt samples. Cesium-137 is a fission product produced from atmospheric testing of nuclear weapons and the operation of a reactor. Cesium-137 was distributed in soils throughout the region as a result of atmospheric testing of nuclear weapons. Environmental concentrations differ widely due to the varying affinity of different soil types for cesium. Table 3.1-14 presents off-site control samples collected up to 22 km from the site. These samples were collected as part of this scoping survey and the Radiological Environmental Monitoring Program. Cesium-137 was detected in all but one of the control samples. With the exception of the one location described below, the cesium-137 concentrations from the scoping survey results are consistent with those detected in the control samples.

Cobalt-60 was detected in several soil and asphalt samples. Cobalt-60 is an activation product produced from neutron irradiation of corrosion products in the Reactor Vessel during operation and is contained in the contamination layers on YNPS systems, structures, and components. Analyses of the soil and asphalt samples indicated the following number of locations with cobalt-60 detected:

Inside Radiation Control Area .... 5 of 20 (25%)
Outside Radiation Control Area .... 5 of 45 (11%)

All of the locations outside the Radiation Control Area with detectable cobalt-60 were inside the Owner Controlled Area fence.

Analysis of samples from a roped off storage area behind the Potentially Contaminated Area (PCA) Storage Building Number 1 indicated low levels of cobalt-58, silver-108m, and cesium-137 in addition to cobalt-60. This is the only location in which cobalt-58 and silver-108m were detected. The non-occupational exposure limit of 100 mrem/yr (10 CFR Part 20) is not exceeded in this area, based on direct ground plane exposure from the measured radioactivity. Additional sampling will be completed in this area to determine the extent of the contamination as a part of the characterization program.

3.1.4.2 Soil Boring and Test Pits

Subsurface samples were collected from four soil borings and four test pits. Test pits were dug in the location where subsurface conditions made soil boring impractical. The core bores and test pits are located in the following areas (Figure 3.1-28):

- CB-1: Northeast of Spent Fuel Pit Building
- CB-2: North of Site Administration Building
- CB-3: East side of Fire Tank

- CB-4: Abandoned septic leach field (northwest, beyond Industrial Area Fence)
- Test Pits (4): Miscellaneous Fill Area (southeast, beyond Industrial Area Fence)

Table 3.1-15 summarizes the location and depth profile for each of these locations.

The soil boring samples were collected using a continuous large diameter split spoon, sampling from ground level to about 10 feet below the groundwater table. The test pit samples were collected using a backhoe.

Tables 3.1-16 and 3.1-17 present an analysis of the gamma emitting radionuclides in the subsurface samples. The 0.3 ft core bore sample from the north side of the Spent Fuel Pit Building indicated the only detectable cobalt-60 result. Five samples had detectable levels of cesium-137, all within a range expected from nuclear weapons test fallout. The cobalt-60 and cesium-137 results were all near the lower limit of detection of the measurement system.

3.1.4.3 Soil from Construction Excavations

Contaminated soil from two recent construction excavations is currently stored on site. These projects were the installation of the new Safety Injection Tank during May and June 1990 and the installation of the new Spent Fuel Building security wall during September and October 1992.

The construction for the new Safety Injection Tank resulted in approximately 200 cubic yards of contaminated dirt being excavated and removed. Forty-seven samples were obtained and analyzed for gamma emitting radionuclides. Nine of the samples indicated low levels of cobalt-60. Using conservative assumptions, the total quantity of cobalt-60 in the 200 cubic yards of soil is approximately 44 μCi.

The construction for the Spent Fuel Building security wall resulted in approximately 67 cubic yards of contaminated dirt being excavated and removed. Twenty four samples were obtained and analyzed for gamma emitting radionuclides. Using conservative assumptions, the total quantity of manganese-54, cobalt-60, cesium-134, and cesium-137 in the 67 cubic yards is approximately 228 μCi.

3.1.4.4 Groundwater Samples from Observation Wells

Groundwater samples were collected from 12 observation wells. Ten of the observation wells were installed as a part of the radiological scoping program. Two existing observation wells were reactivated. Figure 3.1-28 presents the location of the observation wells, and Table 3.1-18 presents characteristic data for each observation well.

The ten new observation wells were installed using 2.5 inch PVC screens and piping in previously drilled 5-inch diameter holes. Each well extended at least 10 feet below the groundwater table as measured at the time of installation. Filter sand was installed around the PVC screened portion of the wells and a clay seal was installed above each filter to enhance well operation and prevent leakage from the surface. All installation work was subject to continuous engineering inspection. Each observation well was developed by pumping water from the well before sampling.

Groundwater samples were analyzed for beta and gamma emitting radionuclides. Table 3.1-19 presents the results of these analyses. With the exception of naturally-occurring K-40 and radon daughters, no gamma emitting radionuclides were detected in any of the samples.

Beta measurements were higher than typically found in routine REMP samples collected from established potable water wells, Sherman Spring, or surface water. It was noted, however, that most of the samples had visible solids suspended in the water due to the nature of the site soils. Following filtration of the original 12 samples, beta concentrations in the water were reduced in some samples by a factor of up to 14, and in some not at all. Gamma spectroscopy analyses done on both the original water samples and the filtered solids for these samples showed no detectable plant-related radioactivity. Based on these results and visual inspection of the water samples, the beta radioactivity is from naturally-occurring radionuclides in the water and in the suspended solids.

Tritium was detected in groundwater samples from three related wells. Two of the wells are near the Spent Fuel Pit Building and one down-gradient well is near the Administration Building. The tritium concentrations were 3,000 - 8,000 pCi/kg, which is well within the Environmental Protection Agency limit for man-made radionuclides in community water systems of 20,000 pCi/kg (40 CFR 141.16). The tritium is most likely leaching from the soil under the Ion Exchange Pit as a result of the leakage described in Section 3.1.1.

3.1.4.5 In Situ Gamma Spectroscopy and Pressurized Ion Chamber Measurements

In situ gamma spectroscopy analysis was performed to evaluate large soil surface areas with a single measurement (or in high background areas, with two measurements). This method identified gamma emitting radionuclides at the ground plane or near the surface and estimated the terrestrial exposure rate from gamma emitting radionuclides. Quantitative determination of the average radioactivity concentration at each location was made by integrating soil radioactivity depth profile data with the in situ gamma spectroscopy results. Pressurized ion chamber measurements were used to verify the sum of the calculated dose rates from the detected radionuclides.

The in situ gamma spectroscopy system utilized a down-looking HPGe detector system mounted on a tripod and positioned one meter above the surface. The detector was operated in one of three modes: 1) with no shield, 2) with a two inch thick collimated lead shield focusing the viewing angle of the detector downward, or 3) with a two inch thick lead shield plug in the end of the collimator. In the unshielded configuration, the effective field of view was a radius about 10 meters. With the collimated shield in place, the effective field of view was a radius about 5 meters.

Photons penetrating the collimator from sources outside the region of interest were compensated for by subtracting the fully-shielded spectrum from the partially-shielded spectrum. This method was used whenever the initial exposure rate at the measurement location was greater than 15 μR/hr (0.015 mR/hr), indicating that the gamma flux may be from sources other than the ground below the detector. The method was also used when the unshielded or partially-shielded gamma spectrum indicated the presence of fission or activation products.

Depth profiles of significant gamma emitting radionuclides were measured to determine the areal and volume radioactivity concentrations. Soil core samples (10 to 12 inches deep) were taken at several in situ gamma spectroscopy measurement locations. The cores were segmented into layers several inches thick for gamma spectroscopy analysis. Naturally- occurring radionuclides were found to be uniformly distributed in the surface layers of soil. Fission and activation products had a profile of decreasing activity with depth. For the purposes of the scoping survey, all radionuclides in asphalt were assumed to be uniform with depth.

In situ gamma spectroscopy and pressurized ion chamber exposure rate measurements were performed at 110 on-site locations (Figures 3.1-29 and 3.1-30). The location selection was based on the grid system described in Section 3.1.4. The middle of each grid square was selected as the survey point unless a physical structure precluded access.

Tables 3.1-20 and 3.1-21 present in situ gamma spectroscopy results, in terms of radionuclide concentrations, for locations outside and inside of the Radiation Control Area, respectively. The following summarizes the results for cobalt-60, the most prevelant man-made radionuclide detected:

Outside Radiation Control Area

Number of locations Co-60 detected 7/40 (18%)

Range of Results .... 35 - 90 pCi/kg

Average Minimum Detectable Concentration (MDC) .... 51 pCi/kg

Inside Radiation Control Area

Number of locations Co-60 detected	26/70 (37%)
Range of Results	99 - 3,900 pCi/kg
Average Minimum Detectable Concentration (MDC)	154 pCi/kg

The cobalt-60 concentrations for the areas outside the Radiation Control Area are near the minimum detectable concentration. All locations with detectable cobalt-60 were within the Owner Controlled Area fence. Cesium-137 was also detected inside and outside of the Radiation Control Area. With the exception of three locations inside the Radiation Control Area, the Cs-137 results are similar to those measured in the off-site control samples presented in Section 3.1.4.1. Trace concentrations of several other gamma emitting radionuclides were also identified inside the Radiation Control Area: Fe-59 (1 location), Mn-54 (1 location), and Ag-110m (1 location).

Tables 3.1-22 and 3.1-23 summarize the terrestrial exposure rate contribution of radionuclides for areas outside and inside of the Radiation Control Area, respectively. These exposure rates were calculated by summing the individual calculated dose rates from each radionuclide, based on the in situ measurements. With the exception of several areas inside the Radiation Control Area, the tables indicate that the contribution from man-made radionuclides (which include plant-related radionuclides and those from atmospheric nuclear weapons testing fallout) is a very small fraction of the total calculated terrestrial exposure rate. In the excepted areas, the man-made component of the exposure rate is also less than the natural terrestrial component.

Tables 3.1-24 and 3.1-25 present the pressurized ion chamber measurements. The measurements made inside the Radiation Control Area indicate a direct exposure component from plant structures. The higher measurements from outside of the Radiation Control Area (53.6 μR/hr, or 0.0536 mR/hr, at locations very close to the Radiation Control Area) indicates that direct radiation from plant structures was being detected.

3.1.5 Radiological Environmental Monitoring Program

Technical Specification 6.7.5.b establishes the Radiological Environmental Monitoring Program. The purpose of the program is to monitor the radiation and radionuclides in the environs of the plant. The program includes the following elements:

- Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters presented in the YNPS Off-Site Dose Calculation Manual.

- Performance of a land use census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required.

- Participation in an inter-laboratory comparison program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the Quality Assurance Program for environmental monitoring.

Technical Specification 6.8.2 requires an annual report to the NRC of the results, interpretation, and analysis of trends from the Radiological Environmental Monitoring Program.

The Radiological Environmental Monitoring Program was initiated in 1958, about two years before YNPS began commercial operation. The program has been executed continuously since its inception. In recent years, the program has incorporated continuous monitoring of air and automatic composite sampling of river water, as well as sampling of groundwater, river sediment, fish, cow milk, goat milk, vegetables, and maple syrup. Direct radiation exposure measurements, both in the immediate plant vicinity and at more distant locations, have been made continuously through the use of a network of thermoluminescent dosimeters (TLDs). Greater than 1,000 samples are collected annually, with greater than 23,000 samples collected since 1958.

The Radiological Environmental Monitoring Program will continue to be updated and modified to reflect changes to the potential source term, the plant environs, and the surrounding land use during YNPS decommissioning. All changes made to the program will include a safety evaluation in accordance with 10 CFR 50.59, if appropriate.

Naturally-occurring radionuclides and fallout from atmospheric nuclear weapons testing have been the predominant radionuclides detected in the environs surrounding YNPS. The predominant radionuclides in this category are:

- <u>Potassium-40</u> - This naturally-occurring radionuclide is found in all biological media; including milk, fish and vegetation.

- <u>Thorium-232</u> - This naturally-occurring radionuclide is found in soil and sediment.

- <u>Cesium-137</u> - This radionuclide resulted from nuclear weapons testing fallout and is detected in milk, vegetation, soil and sediment.

- <u>Strontium-90</u> - This radionuclide resulted from nuclear weapons testing fallout and is detected in milk.

Two media, in addition to those presented in the preceding sections, have been monitored closely through the Radiological Environmental Monitoring Program:

- Detection of cobalt-60 in Sherman Pond bottom sediment near the plant discharge In addition to cesium-137, most of which can be attributed to nuclear weapons testing fallout, cobalt-60 has been detected in the sediment near the plant discharge structure in Sherman Pond (Figures 3.1-31 and 3.1-32). The cobalt-60 was deposited during controlled plant discharges through a licensed and monitored discharge point.

  The cobalt-60 results were very low with the exception of one set of samples. The cobalt-60 concentration in the 0-5 cm layer of sediment collected in May 1993 was greater than the other values presented in Figure 3.1-31. Subsequent samples taken in October 1993 indicated the typical concentrations of cobalt-60.

  Variability in sediment sample results at low concentrations is not unexpected. In addition, there is no significant pathway of exposure from cobalt-60 in the sediment. The dose would be less than 2 mrem/yr if the sediment was brought above water and a person was exposed to a uniform 15 cm thick ground plane concentration of cobalt-60 at the May 1993 sample value for the time spent outdoors in the residential scenario of Reference 3.1-9 (about 1700 hours per year). This is significantly less than the 25 mrem per year dose limit in 40 CFR Part 190 or the 100 mrem per year total effective dose equivalent limit in 10 CFR Part 20.

  Cobalt-60 is an insoluble radionuclide and typically is distributed non-homogeneously in the sediment. This was verified by splitting the May 1993 sample into two portions. The results indicated that one sample showed no cobalt-60. The sample with the cobalt-60 was split again with the results indicating a non-uniform distribution.

- Detection of tritium in Sherman Spring. Tritium was detected initially in Sherman Spring in 1965 (Figure 3.1-33). Sherman Spring is located on site property below the Sherman Dam. The concentration has decreased exponentially since its discovery. The source of tritium is believed to be the Ion Exchange Pit leak described in Section 3.1.1.

TLDs have been used to continuously measure direct radiation from plant sources and natural background. In recent years, five TLDs have been located at each of forty locations near YNPS. These have been collected and processed on a quarterly basis. These monitors demonstrate a large variation in background exposure rate as a function of time and location. For example, in 1992, quarterly background radiation levels varied from 4.5 to 8.8 μR/hr (0.0045 to 0.0088 mR/hr) at off-site locations (Reference 3.1-10).

The data also indicated no significant difference between near-site and far-site locations.

In situ gamma spectroscopy measurements were taken in the vicinity of YNPS in the years 1981, 1984 and 1987. Soil core samples were also taken at many locations along with the in situ measurements. The core samples were segmented into 5 cm increments and analyzed. The cesium-137 concentration in the soil core segments from seven locations surveyed in 1987 varied from 20 to 1150 pCi/kg (wet). Several naturally-occurring radionuclides also were detected in these samples: thorium-232, uranium-238, and potassium-40. Direct radiation measurements were also taken with a pressurized ion chamber coincident with the in situ gamma spectroscopy measurements. The exposure rates varied from 8.9 to 11.0 μR/hr (0.0089 to 0.011 mR/hr) at the seven locations surveyed in 1987.

3.1.6 Radiologically Affected Area Identification

NUREG/CR-5849 presents classifications for characterization of plant areas for radiological significance:

- Affected Area - This designation includes areas that have either potential or known radioactive contamination. Areas immediately surrounding or adjacent to locations where radioactive materials were used or stored, spilled or buried are included in this classification because of the potential for inadvertent spread of contamination.

- Unaffected Area - This designation includes areas that are not expected to contain residual radioactivity.

Based on a review of the operating history and scoping survey results, the following YNPS areas are considered Affected Areas:

- Radiation Control Area - All structures and soil in the Radiation Control Area. Radioactive materials were used and stored in the Radiation Control Area.

- Outside the Radiation Control Area - Structures and soil located in areas outside the Radiation Control Area but within the Industrial Area fence. Several plant occurrances resulted in low level contamination of some secondary side systems. The scoping survey results indicated several locations with plant related radioactivity in the soil.

- Soil between the RCA and Owner Controlled Fence - Soil located on the south side of the site between the Radiation Control Area fence and the Owner Controlled fence indicated plant related radioactivity.

These three areas collectively make up all areas within the Owner Controlled Area fence as shown on Figure 3.1-34.

3.1.7 Site Characterization Surveys

Site characterization is the next phase in the radiological survey process. The purpose of the site characterization surveys is to define more precisely the extent and magnitude of the contamination on site. Site characterization surveys will be used to supplement the scoping survey data in areas where data are missing or where the data indicate contamination levels are at or near the release criteria. The level of effort is related to the data needs for the item or area being surveyed. Items or areas that are highly contaminated require less data than items or areas that are near the release criteria to make decisions regarding their radiological status. Scoping and characterization data will be used to plan and complete decommissioning activities.

3.1.7.1 Program Description

The site characterization survey will collect additional radiological data and samples in areas to facilitate decommissioning planning. The following items will be considered in the site characterization survey for systems, structures, and components:

- Measure the activity and contamination levels associated with inaccessible plant areas before removal activities.
- Perform additional testing on contaminated concrete to improve estimates of contamination levels and depth of contamination.
- Perform a detailed radiological survey of the Reactor Vessel after completion of the Component Removal Project.
- Determine the extent of contamination of non-radiological plant systems, structures, and components.
- Determine the extent of activated structures and components external to the Reactor Vessel.

The following items will be considered in the site characterization survey for soil and groundwater:

- Perform additional surface and subsurface soil sampling as well as in situ measurements to determine the lateral and vertical extent of contamination identified during the radiological scoping surveys. The sampling should include areas identified with significant soil contamination as well as soil under buildings to the extent practicable.

- Establish the expected cesium-137 background concentration for the plant site. This determination should include analysis of additional off-site samples of varying soil types.

- Develop a computer model to define the groundwater regime for the site. Additional wells may be installed to provide the database needed to perform the modelling calculations.

3.1.7.2 Implementation Schedule

Site characterization surveys of YNPS systems, structures, and components will be completed to support detailed planning activities associated with their decontamination and dismantlement. The surveys for soil and groundwater will be completed to support detailed planning activities associated with preparation of the site for the final radiation survey.

REFERENCES

3.1-1 NUREG/CR-5849 (Draft), "Manual For Conducting Radiological Surveys in Support of License Termination", June 1992.

3.1-2 YRC-1024, "Basis for the Radiological Status of Plant Systems and Structures", P. Hollenbeck, September 1993.

3.1-3 REG 90/93, "Dose Conversion Factors For Yankee Rowe Piping", Y. J. Yu to R. A. Mellor, April 2, 1993.

3.1-4 YRC-1031, "Yankee Rowe Component Activation Analysis", K.J. Morrissey, September 1993.

3.1-5 NUREG/CR-3474, "Long-Lived Activation Products in Reactor Materials", August 1984.

3.1-6 REG 147/93, "Dose Rate Estimate of the YNPS Reactor Vessel", Y. J. Yu to R. A. Mellor, June 4, 1993.

3.1-7 "YNPS Outdoor Site Scoping Data", F.X. Bellini, E.R. Cumming to P. Hollenbeck, September 24, 1993

3.1-8 YNPS Off-Site Dose Calculation Manual, Revision 10, June 1993.

3.1-9 NUREG/CR-5512, Vol. 1, "Residual Radioactive Contamination from Decommissioning - Technical Basis for Translating Contamination Levels to Annual Total Effective Dose Equivalent," Final Report, October 1992.

3.1-10 1992 Annual Radiological Environmental Operating Report.

PROCEDURE REFERENCES

AP-0830, "Administrative Program For The Radiological Scoping Survey To Support The YNPS Decommissioning Plan"

DP-8120, "Collection of Surface Soil and Asphalt Samples"

DP-8121, "In-Plant Radiological Surveys To Support The Scoping Program"

DP-8122, "Subsurface Soil Sampling and Monitoring Well Installation"

DP-8123, "Sample Chain of Custody"

DP-9745, "Ground Water Level Measurement and Sample Collection in Observation Wells"

YAEL-510, "Identification and Quantitative Determination of Radionuclides in Soil by Gamma-Ray In-Situ Spectrometry"

YAEL-512, " Operation and Calibration of the Reuter-Stokes Pressurized Ion Chamber (PIC)"

TABLE 3.1-1

YANKEE NUCLEAR POWER STATION FACILITY OPERATING HISTORY

Cycle Number	Start-up Date	Shutdown Date	Cycle Burn-up (MWD/MTU)	Effective Full Power Days	Average Power Level (%)
1	8/19/60	5/18/62	8,470	234.9	36.9
2	9/20/62	9/3/63	7,866	273.2	78.5
3	11/12/63	8/2/64	6,329	220.5	83.5
4	9/5/64	8/9/65	8,734	302.0	89.4
5	11/9/65	10/4/66	8,893	307.3	93.4
6	11/7/66	3/23/68	12,419	430.0	85.7
7	5/1/68	8/2/69	11,963	412.0	90.0
8	9/24/69	10/24/70	10,142	346.1	87.6
9	11/20/70	2/12/72	11,946	413.6	92.1
10	5/17/72	5/11/74	15,148	461.3	63.7
11	8/25/74	10/18/75	12,869	391.9	93.5
12	12/18/75	6/4/77	14,879	446.9	83.7
13	9/1/77	10/21/78	12,890	382.3	92.1
14	12/21/78	5/2/81	16,114	478.8	55.5
15	7/30/81	9/11/82	13,000	392.6	96.2
16	12/16/82	3/31/84	14,168	418.4	88.8
17	6/8/84	10/19/85	15,116	444.9	89.3
18	12/10/85	5/2/87	16,020	471.5	92.8
19	7/3/87	11/12/88	15,518	449.4	90.2
20	1/5/89	6/23/90	16,830	486.1	92.8
21	11/11/90	10/1/91	9,936	288.6	89.2

TABLE 3.1-2

NUCLIDE DISTRIBUTIONS FOR SYSTEMS AND STRUCTURES

Fractional Nuclide Activity Distributions as of 1/94						
Nuclide	Contamination		Activation			
	MCS and Bleed Line in VC	Remaining Systems & Structures	Vessel Cladding	Vessel Wall	Neutron Shield Tank	Bio Shield Concrete
H-3			5.09e-04	1.42e-03	1.49e-03	6.48e-01
C-14			2.42e-04	2.87e-05	2.87e-05	
Ar-39						7.49e-03
Ca-41						2.05e-03
Ca-45						1.43e-02
Fe-55	5.00e-01	7.72e-01	4.67e-01	8.88e-01	9.09e-01	2.39e-01
Ni-59	2.18e-03	6.16e-04	6.29e-04	1.71e-05	1.81e-05	
Ni-63	2.52e-01	7.63e-02	8.03e-02	2.39e-03	2.43e-03	
Sr-89		2.87e-07				
Sr-90		1.26e-03				
Pu-238	6.20e-05	5.47e-05				
Pu-239/240	1.87e-04	1.20e-04				
Pu-241	8.57e-03	5.27e-03				
Am-241	1.04e-04	6.84e-05				
Cm-242	7.08e-06	9.88e-06				
Cm-243/244	1.26e-04	1.14e-04				
Mn-54	1.21e-02	1.92e-02	2.62e-03	1.90e-02	1.30e-02	3.36e-03
Fe-59	1.05e-07	1.74e-07				
Co-57	6.68e-05	3.36e-05				
Co-58	7.02e-05	2.67e-05				
Co-60	2.17e-01	1.13e-01	4.49e-01	8.88e-02	7.38e-02	3.87e-02
Nb-94			1.04e-06	6.73e-07	5.14e-07	
Zr-95	4.42e-06	2.51e-06				
Nb-95	1.18e-05	5.97e-06				
Tc-99			1.62e-07	1.58e-06	1.08e-06	
Ru-103	2.04e-08	4.86e-09				
Ru-106	3.19e-03	3.20e-03				
Ag-108m		1.62e-05				
Ag-110m		9.10e-04				
Sb-124	5.59e-07	1.92e-06				
Sb-125	4.24e-04	3.98e-03				
Cs-134		2.26e-04				2.14e-03
Cs-137		1.79e-03				
Ce-144	4.66e-03	2.57e-03				
Eu-152			8.92e-05	3.78e-04	3.50e-04	4.19e-02
Eu-154			1.22e-05	3.55e-05	3.15e-05	3.68e-03
Eu-155			9.01e-07	2.89e-07	2.38e-07	
TOTAL	1.000	1.000	1.000	1.000	1.000	1.000

NOTE: Activation values are based upon an average flux at the core centerline

TABLE 3.1-3

DECAYED FRACTIONAL NUCLIDE ACTIVITY DISTRIBUTIONS

Nuclide	Year			
	1/1/94	1/1/96	1/1/98	1/1/2000
Fe-55	7.72e-01	7.22e-01	6.46e-01	5.54e-01
Ni-59	6.16e-04	9.63e-04	1.44e-03	2.06e-03
Ni-63	7.63e-02	1.18e-01	1.74e-01	2.45e-01
Sr-89	2.87e-07	2.01e-11	1.34e-15	8.62e-20
Sr-90	1.26e-03	1.87e-03	2.67e-03	3.65e-03
Pu-238	5.47e-05	8.42e-05	1.24e-04	1.75e-04
Pu-239/240	1.20e-04	1.88e-04	2.81e-04	4.02e-04
Pu-241	5.27e-03	7.47e-03	1.02e-02	1.32e-02
Am-241	6.84e-05	1.07e-04	1.59e-04	2.27e-04
Cm-242	9.88e-06	6.94e-07	4.67e-08	3.01e-09
Cm-243/244	1.14e-04	1.70e-04	2.42e-04	3.30e-04
Mn-54	1.92e-02	5.95e-03	1.76e-03	5.01e-04
Fe-59	1.74e-07	3.23e-12	5.72e-17	9.72e-22
Co-57	3.36e-05	8.09e-06	1.87e-06	4.13e-07
Co-58	2.67e-05	3.27e-08	3.84e-11	4.32e-14
Co-60	1.13e-01	1.35e-01	1.55e-01	1.71e-01
Zr-95	2.51e-06	1.44e-09	7.94e-13	4.19e-16
Nb-95	5.97e-06	3.43e-09	1.89e-12	9.97e-16
Ru-103	4.86e-09	1.96e-14	7.60e-20	2.82e-25
Ru-106	3.20e-03	1.27e-03	4.79e-04	1.74e-04
Ag-108m	1.62e-05	2.51e-05	3.71e-05	5.26e-05
Ag-110m	9.10e-04	1.87e-04	3.70e-05	6.99e-06
Sb-124	1.92e-06	6.70e-10	2.23e-13	7.13e-17
Sb-125	3.98e-03	3.77e-03	3.42e-03	2.97e-03
Cs-134	2.26e-04	1.80e-04	1.37e-04	1.01e-04
Cs-137	1.79e-03	2.66e-03	3.81e-03	5.21e-03
Ce-144	2.57e-03	6.78e-04	1.71e-04	4.13e-05
SUM OF FRAC	1.000	1.000	1.000	1.000
RELATIVE ACT	1.000	0.640	0.428	0.298

Based on system and structure contamination distribution

TABLE 3.1-4

SYSTEM AVERAGE CONTAMINATION LEVELS

SYSTEM	dpm/100 cm^2	$\mu Ci/cm^2$	Internal Contamination	External Contamination
Steam Generator Blowdown	1.0e+3	4.5e-6	X	X
Compressed Air	1.0e+4	4.5e-5		X
Component Cooling	1.0e+3	4.5e-6	X	X
Charging and Volume Control	1.2e+7	5.3e-2	X	X
Containment Isolation	1.0e+3	4.5e-6		X
Primary Plant Corrosion Control	1.2e+4	5.4e-5		X
Chemical Shutdown	1.1e+4	5.0e-5	X	X
Demineralized Water	5.0e+3	2.3e-5		X
Emergency Feedwater	1.0e+3	4.5e-6		X
Fuel Handling	1.7e+6	7.8e-3	X	X
Spent Fuel Cooling	3.3e+8	1.5e+0	X	X
Fire Protection	1.0e+3	4.5e-6		X
Feedwater	1.0e+3	4.5e-6		X
Heating	1.0e+3	4.5e-6	X	X
Ventilation	5.0e+3	2.3e-5	X	X
Main Coolant	7.1e+9	3.2e+1	X	X
Main Steam	1.0e+3	4.5e-6		X
Miscellaneous Vent and Drain	5.0e+3	2.3e-5	X	X
Primary Pump Seal Water	5.0e+3	2.3e-5		X
Pressure Control and Relief	1.0e+4	4.5e-5	X	X
Purification	1.4e+6	6.1e-3	X	X
Primary Plant Sampling	1.4e+6	6.1e-3	X	X
Shutdown Cooling	1.2e+7	5.3e-2	X	X
Safety Injection	1.4e+5	6.5e-4	X	X
Safe Shutdown	1.4e+5	6.5e-4	X	X
Service Water	5.0e+3	2.3e-5		X
Primary Plant Vent and Drain	1.2e+7	5.3e-2	X	X
Post Accident Hydrogen Control	1.2e+4	5.4e-5	X	X
VC Heating and Cooling	1.2e+4	5.4e-5	X	X
VC Ventilation and Purge	1.2e+4	5.4e-5	X	X
Water Cleanup	5.3e+5	2.4e-3		X
Waste Disposal	1.2e+7	5.3e-2	X	X

TABLE 3.1-5

CONCRETE CORE BORE RESULTS

Sample ID	Location Description	Contamination Depth (mm)
CB-1	Service Bldg, Clean Side Machine Shop, floor near lathe	0
CB-2	Turbine Bldg, Lower Level, floor near #1 condensate pump	0
CB-3	Service Bldg, RCA Machine Shop, floor near lathe	0
CB-4	PAB, Lower Level, floor near component cooling pumps	0
CB-5	PAB, Upper Level, floor near Boric Acid Mix Tank	0
CB-6	PAB, Valve Room, floor near verticle pipe chase	6
CB-7	Waste Disposal Bldg, Evap Cub, floor near access ladder	32
CB-8	Waste Disposal Bldg, Evap Cub, east wall	6
CB-9	PAB, Shutdown Cooling HX Cub, floor under HX	5
CB-10	PAB, Shutdown Cooling HX Cub, east wall	0
CB-11	PAB, PDCT, floor near #2 pump	5
CB-12	PAB, #1 Chrg Pump Cub, floor near drain	13
CB-13	VC, Charging Floor, near control point	5
CB-14	VC, Loop 4, inner wall	5
CB-15	VC, Loop 1, inner wall	5
CB-16	VC, Brass Drain Box, floor under Loop 1 and PZR	15
CB-17	VC, Loop 2, inner wall	3
CB-18	VC, Charging Floor, floor west side	5
CB-19	VC, Loop 3, inner wall	6
CB-20	Old PCA, floor near bath tub	12
CB-21	Waste Disposal Bldg, Drumming Area, floor near rolling machine	14
CB-22	Service Bldg, South Decon Room, floor near moat	6
CB-23	VC, Loop 2 Dog House, 1st block closest to cavity	7
CB-24	VC, Loop 2 Dog House, 2nd block	5
CB-25	VC, Loop 2 Dog House, 3rd block	6
CB-26	VC, Loop 2 Dog House, back (5th) block	7

TABLE 3.1-6
CONCRETE AVERAGE CONTAMINATION LEVELS

Building	Description	dpm/100 cm^2	Depth(mm)
Diesel Generator Building	SI Pump Room	200	5
Diesel Generator Building	SI Accumulator Room	200	5
Primary Auxilary Building (PAB)	Gravity Drain Tank Cubicle	2500	5
PAB	Primary Building Sump Cubicle	2000	5
PAB	Primary Drain Collecting Tank Cubicle	50000	5
PAB	Lower Level	0	0
PAB	Cubicle Corridor	500	5
PAB	Low Pressure Surge Tank Pump Cubicle	2000	5
PAB	Shut Down Cooling Heat Exchanger Cubicle	2000	5
PAB	Shut Down Cooling Heat Exchanger Pump Cubicle	2000	5
PAB	Low Pressure Surge Tank Heat Exchanger Cubicle	2000	5
PAB	No.1 Charging Pump Cubicle	2000	15
PAB	No.2 Charging Pump Cubicle	2000	15
PAB	No.3 Charging Pump Cubicle	2000	15
PAB	No.1 Purification Pump Cubicle	2000	15
PAB	No.2 Purification Pump Cubicle	2000	15
PAB	Pipe Trench	50000	15
PAB	Upper Level	0	0
PAB	Valve Room	500	5
PAB	Vertical Pipe Chase	5000	5
PAB	Lower Pipe Chase	500	5
PAB	Low Pressure Surge Tank Cubicle	2000	5
PAB	Chemistry Sample Room	1000	5
Potentially Contaminated Area Storage Building	Potentially Contaminated Area Storage Building 1	5000	15
Potentially Contaminated Area Storage Building	Potentially Contaminated Area Storage Building 2	500	5
Service Building (SB)	North Decon Room	1000	5
SB	South Decon Room	5000	5
SB	Primary Side Machine Shop	0	0
SB	Welding Booth	0	0

TABLE 3.1-6 (Continued)
CONCRETE AVERAGE CONTAMINATION LEVELS

Building	Description	dpm/100 cm^2	Depth(mm)
SB	Tool Decontamination Room	0	0
SB	Primary Side Chemistry Lab	0	0
SB	Rad Protection Calibration Lab	0	0
Spent Fuel Building	Spent Fuel Pit	10000	102
Spent Fuel Building	New Fuel Vault	0	0
Vapor Container (VC)	Shield Tank Cavity	10000	5
VC	Charging Floor	1000	5
VC	Loop 1	10000	5
VC	Loop 2	10000	5
VC	Loop 3	10000	5
VC	Loop 4	10000	5
VC	Pressurizer Cubicle	10000	5
VC	Brass Drain Box	10000	15
VC	Feed and Bleed Heat Exchanger Cubicle	10000	5
VC	Equipment Hatch Area	10000	5
VC	Broadway	1000	5
VC	Shell Area	10000	5
Waste Disposal Building (WD)	Gas Compressor Room	0	0
WD	Drumming Pit	1000	15
WD	Corridor Area	200	5
WD	Liquid Water Transfer Pump Cubicle	200	5
WD	Evaporator Cubicle	4000	32
WD	Stripper Cubicle	9000	32
WD	Sump Room	500	15
Compactor Building	Compactor Building	500	5
Yard Area in RCA	ION EXCHANGE PIT	5.3E+5	152
Yard Area in RCA	IEP PIPE TUNNEL	20000	5
Yard Area in RCA	FUEL CHUTE	20000	102
Yard Area in RCA	TANK FARM AREA	5000	5

TABLE 3.1-7

RADIAL ACTIVATION ANALYSES RESULTS AS OF 1/94 (Ci/m^3)

Nuclide	Reactor Vessel Cladding	Reactor Vessel Wall	Inner Wall NST	Outer Wall NST	Bioshield Concrete (Inner 2")
H-3	4.63e+00	1.16e+00	5.09e-01	4.72e-07	3.94e-06
C-14	2.20e+00	2.34e-02	9.81e-03	8.75e-09	
Ar-39					4.56e-08
Ca-41					1.25e-08
Ca-45					8.68e-08
Mn-54	2.38e+01	1.55e+01	4.44e+00	2.03e-06	2.05e-08
Fe-55	4.25e+03	7.26e+02	3.11e+02	2.84e-04	1.45e-06
Ni-59	5.73e+00	1.40e-02	6.21e-03	5.87e-09	
Co-60	4.08e+03	7.25e+01	2.53e+01	1.78e-05	2.36e-07
Ni-63	7.30e+02	1.95e+00	8.32e-01	7.57e-07	
Nb-94	9.46e-03	5.49e-04	1.76e-04	1.07e-10	
Tc-99	1.47e-03	1.29e-03	3.70e-04	1.71e-10	
Cs-134					1.30e-08
Eu-152	8.12e-01	3.09e-01	1.20e-01	1.01e-07	2.55e-07
Eu-154	1.10e-01	2.90e-02	1.08e-02	8.44e-09	2.24e-08
Eu-155	8.23e-03	2.36e-04	8.18e-05		
TOTAL	9.09e+03	8.17e+02	3.42e+02	3.05e-04	6.09e-06

Note: Data are the average values over the active fuel length

TABLE 3.1-8

AXIAL ACTIVATION ANALYSES RESULTS AS OF 1/94

Axial Position Above Bottom Tangent (cm)	Relative Nuclide Distribution		
	Reactor Vessel Wall	Inner NST Wall	Bioshield Concrete
150	2.8e-05	4.6e-04	7.3e-03
175	2.7e-04	2.8e-03	2.4e-02
200	8.2e-04	8.0e-03	5.0e-02
225	2.9e-03	2.3e-02	9.7e-02
250	1.2e-02	6.9e-02	1.8e-01
275	7.8e-02	1.8e-01	3.2e-01
300	3.1e-01	4.0e-01	5.1e-01
325	7.2e-01	7.2e-01	7.3e-01
350	1.0e+00	9.9e-01	9.4e-01
375	1.2e+00	1.1e+00	1.1e+00
400	1.2e+00	1.2e+00	1.2e+00
425	1.2e+00	1.2e+00	1.2e+00
450	1.2e+00	1.2e+00	1.2e+00
475	1.1e+00	1.1e+00	1.1e+00
500	9.5e-01	9.0e-01	9.2e-01
525	6.1e-01	6.2e-01	7.6e-01
550	3.7e-01	3.5e-01	6.4e-01
575	1.9e-01	1.9e-01	5.6e-01
600	6.6e-02	1.3e-01	4.3e-01
625	1.4e-02	7.0e-02	2.8e-01
650	2.8e-03	4.2e-02	1.5e-01
675	9.8e-04	2.8e-02	8.0e-02
700	4.6e-04	1.5e-02	3.6e-02
725	4.4e-05	4.4e-03	8.9e-03

Note: Data are relative to values presented in Table 3.1-7

TABLE 3.1-9

CALCULATED DOSE RATES FROM THE YNPS REACTOR VESSEL (R/hr)

DOSE POINT	DOSE RATE	
	1/1/94	1/1/2000
Contact with outer surface	1.1e+01	5.1e+00
2 meters from outer surface	3.1e+00	1.4e+00
5 meters from outer surface	1.2e+00	5.3e-01
Contact with inner surface	3.8e+02	1.7e+02
Center of vessel	1.9e+02	8.4e+01

NOTE:

1. All dose points are located at the mid-height of the vessel
2. The activation levels were assumed to be in the entire 26.8 foot vessel height

TABLE 3.1-10

FACILITY/COMPONENT RADIONUCLIDE INVENTORY (1/94)

Material / Source	Activity (Ci)
Contaminated System Components	175
Contaminated Concrete	0.43
Contaminated Steel	0.03
Miscellaneous Tools, Equipment, Duct, Conduit	39
Miscellaneous Drums and Solidified Waste	41
Reactor Vessel	4700
Neutron Shield Tank	180
Instrument Calibration Sources	32
Water in the Spent Fuel Pit and Ion Exchange Pit	4.8
ROUNDED TOTAL	5172

NOTE:

1. Does not include activity in 533 fuel assemblies stored on site
2. Does not include Greater Than Class C reactor internal material stored on site

TABLE 3.1-11

TOTAL NUMBER OF SURFACE SOIL AND ASPHALT SAMPLES COLLECTED

Area	Asphalt Samples	Sod or Humus Samples	Soil Samples	Total Samples
Inside RCA	21	0	22	43
Outside RCA	5	23	54	82
Control	0	4	4	8
Total	26	27	80	133

TABLE 3.1-12

SURFACE SOIL AND ASPHALT RESULTS - INSIDE RCA
(pCi/kg-wet)

Location on Fig. 3.1-27	Sample Type	Man-made				Naturally-occurring	
		Co-60	Cs-137	Co-58	Ag-108m	K-40	Th-232
33	Asphalt	-	-	-	-	6960	396
	Soil	-	-	-	-	14100	578
34	Asphalt	-	209	-	-	5780	362
	Soil	-	-	-	-	13700	493
35	Asphalt	400	178	-	-	5800	447
	Soil	-	-	-	-	14400	884
36	Asphalt	-	-	-	-	8480	335
	Soil	-	-	-	-	12300	318
37	Asphalt	-	-	-	-	6910	316
	Soil	-	-	-	-	12300	645
38	Asphalt	-	-	-	-	7980	438
	Soil	-	-	-	-	11200	484
39	Asphalt	-	-	-	-	6100	247
	Soil	-	133	-	-	12200	648
40	Asphalt	-	-	-	-	3540	-
	Soil	-	-	-	-	13700	542
41	Asphalt	-	-	-	-	4310	-
	Soil	-	-	-	-	11200	491
42	Asphalt	566	534	-	-	7690	-
	Soil	-	105	-	-	16200	897
43	Asphalt	-	-	-	-	5280	-
	Soil	-	-	-	-	11200	468

TABLE 3.1-12 (Continued)

SURFACE SOIL AND ASPHALT RESULTS - INSIDE RCA
(pCi/kg-wet)

Location on Fig. 3.1-27	Sample Type	Man-made				Naturally-occurring	
		Co-60	Cs-137	Co-58	Ag-108m	K-40	Th-232
44	Asphalt	-	-	-	-	9960	649
	Soil	-	324	-	-	18100	876
45	Asphalt	-	-	-	-	5460	358
	Soil	-	-	-	-	13700	347
46	Asphalt	356	499	-	-	4880	-
	Soil	19300	8450	212	503	13700	868
47	Asphalt	-	-	-	-	9170	615
	Soil	692	431	-	-	10900	490
48	Asphalt	-	-	-	-	8180	310
	Soil	-	288	-	-	15200	329
49	Asphalt	-	107	-	-	8200	-
	Soil	-	-	-	-	12500	579
50	Asphalt	-	-	-	-	7650	-
	Soil	-	-	-	-	16800	521
51	Soil	-	-	-	-	15900	552
59	Asphalt	1650	1840	-	2500	13000	805
	Soil	1500	1680	-	2280	11900	733
60	Asphalt	3090	923	-	3920	4090	342
	Soil	13500	11300	-	7590	14200	587
61	Asphalt	296	234	-	-	4660	-
	Soil	3630	916	-	234	13800	294

NOTE: A hyphen in the table indicates that radioactivity was not detected.

TABLE 3.1-13

SURFACE SOIL AND ASPHALT RESULTS - OUTSIDE RCA
(pCi/kg-wet)

Location on Fig.3.1-26	Sample Type	Man-made				Naturally-occurring	
		Co-60	Cs-137	Co-58	Ag-108m	K-40	Th-232
1	Soil	-	-	-	-	15300	582
2	Soil	-	-	-	-	10400	434
3	Asphalt	-	-	-	-	7780	597
	Soil	-	-	-	-	13800	587
4	Asphalt	-	-	-	-	6420	-
	Soil	-	-	-	-	13500	692
5	Sod	-	-	-	-	17300	672
	Soil	-	-	-	-	11100	308
6	Asphalt	-	-	-	-	5490	407
	Soil	-	-	-	-	11500	603
7	Sod	-	745	-	-	14000	951
	Soil	-	263	-	-	9210	603
8	Sod	-	356	-	-	12200	789
	Soil	-	141	-	-	12200	602
9	Sod	-	94.8	-	-	8410	409
	Soil	-	-	-	-	6540	308
10	Sod	707	351	-	-	13600	983
	Soil	-	-	-	-	13300	729
11	Soil	-	-	-	-	13900	613
12	Sod	637	274	-	-	17800	736
	Soil	-	-	-	-	16700	795
13	Sod	-	226	-	-	20900	713
	Soil	-	-	-	-	17400	835

TABLE 3.1-13 (Continued)

SURFACE SOIL AND ASPHALT RESULTS - OUTSIDE RCA
(pCi/kg-wet)

Location on Fig. 3.1-26	Sample Type	Man-made				Naturally-occurring	
		Co-60	Cs-137	Co-58	Ag-108m	K-40	Th-232
14	Humus	-	518	-	-	-	-
	Soil	-	302	-	-	2030	-
	Soil	-	259	-	-	7930	-
15	Humus	-	1480	-	-	3920	-
	Soil	-	66.9	-	-	6530	474
16	Soil	-	662	-	-	-	-
17	Soil	-	639	-	-	-	-
18	Soil	-	-	-	-	13500	619
19	Soil	-	169	-	-	13800	440
20	Soil	-	629	-	-	7540	366
21	Soil	-	412	-	-	9240	430
22	Soil	-	337	-	-	9080	162
23	Soil	-	467	-	-	10400	328
24	Soil	-	208	-	-	9510	465
25	Soil	-	-	-	-	12000	578
26	Sod	-	1170	-	-	15200	430
	Soil	-	-	-	-	16400	688
27	Soil	-	136	-	-	18900	754
28	Soil	-	-	-	-	17800	918
29	Soil	-	-	-	-	16400	622
30	Sod	-	-	-	-	18400	932
	Soil	-	-	-	-	19600	670
31	Soil	-	-	-	-	18100	605
32	Soil	-	-	-	-	16500	535

TABLE 3.1-13 (Continued)

SURFACE SOIL AND ASPHALT RESULTS - OUTSIDE RCA
(pCi/kg-wet)

Location on Fig. 3.1-26	Sample Type	Man-made				Naturally-occurring	
		Co-60	Cs-137	Co-58	Ag-108m	K-40	Th-232
52	Asphalt	-	-	-	-	11400	477
	Soil	-	-	-	-	16000	798
53	Asphalt	-	-	-	-	6990	-
	Soil	-	157	-	-	14900	458
54	Sod	-	-	-	-	16000	960
	Soil	-	-	-	-	12900	467
55	Soil	-	-	-	-	13800	335
56	Soil	-	411	-	-	5020	514
57	Sediment	-	105	-	-	9670	352
58	Soil	-	372	-	-	9680	614
62	Sod	-	-	-	-	11000	774
	Soil	-	-	-	-	14300	734
63	Sod	-	-	-	-	15100	489
	Soil	-	-	-	-	15100	500
64	Sod	-	195	-	-	8870	650
	Soil	-	-	-	-	14300	589
65	Sod	256	-	-	-	10300	350
	Soil	-	-	-	-	12600	497
66	Sod	-	457	-	-	8530	552
	Soil	-	282	-	-	9920	871
67	Sod	-	220	-	-	8270	510
	Soil	-	114	-	-	10400	510
68	Sod	-	692	-	-	15000	676
	Soil	-	-	-	-	16600	700

TABLE 3.1-13 (Continued)

SURFACE SOIL AND ASPHALT RESULTS - OUTSIDE RCA
(pCi/kg-wet)

Location on Fig. 3.1-26	Sample Type	Man-made				Naturally-occurring	
		Co-60	Cs-137	Co-58	Ag-108m	K-40	Th-232
69	Sod	133	2950	-	-	10500	250
	Soil	-	188	-	-	16900	374
70	Sod	-	605	-	-	-	-
	Soil	-	332	-	-	6230	385
71	Sod	134	1460	-	-	2710	-
	Soil	-	-	-	-	8100	446
72	Soil	-	-	-	-	15200	734
77	Humus	290	1060	-	-	-	-
	Soil	-	302	-	-	802	-
78	Sediment	-	-	-	-	11400	608
79	Sediment	-	-	-	-	-	371

NOTE: A hyphen in the table indicates that radioactivity was not detected.

TABLE 3.1-14

SURFACE SOIL RESULTS - OFFSITE CONTROLS
(pCi/kg-wet)

Location Sector, Dist.	Date	Description/ Core Depth	Cs-137	K-40	Th-232
SW, 6.6 km	6/10/93	Humus	2790	-	-
		Soil	553	16600	537
SSW, 11.9 km	6/10/93	Humus	945	1870	-
		Soil	367	10800	460
SSW, 4.9 km	6/10/93	Sod	422	7560	477
		Soil	-	8650	259
SSW, 6.4 km	6/10/93	Sod	455	10100	700
		Soil	259	10500	192
N, 6.1 km	9/5/91	Tilled Soil	268	14300	522
S, 8.5 km	9/6/91	Tilled Soil	339	8890	563
SE, 4.2 km	9/22/87	0-5 cm	318	9730	497
		5-10 cm	273	12900	632
		10-15 cm	163	14900	747
N, 3.2 km	9/22/87	0-5 cm	461	14100	643
		5-10 cm	604	11900	648
		10-15 cm	328	12400	643
NNW, 2.0 km	9/22/87	0-5 cm	220	10500	410
		5-10 cm	169	12800	587
		10-15 cm	110	14400	977
SW, 1.1 km	9/23/87	0-5 cm	811	12300	764
		5-10 cm	259	13800	609
W, 22.2 km	9/23/87	0-5 cm	372	14800	740
		5-10 cm	385	17500	928
		10-15 cm	203	19600	923

NOTE: A hyphen in the table indicates that radioactivity was not detected.

TABLE 3.1-15

SUBSURFACE SOIL SAMPLING LOCATIONS

Boring / Test Pit Number	Depth to Groundwater (ft)	Total Depth (ft)	No. of Samples Analyzed	Location on Site
CB-1	9.4	25.5	19	NE of Spent Fuel Building
CB-2	11.0	25.0	19	N of Admin Building
CB-3	3.9	15.0	14	E of Fire Water Tank
CB-4	11.3	20.0	17	Old Septic Leaching Field
TP-1	-	6.0	3	E Misc. Fill Area
TP-2	-	6.0	4	Center Misc. Fill Area
TP-3	-	4.0	3	N Misc. Fill Area
TP-4	-	7.0	4	W Misc. Fill Area

NOTE: Groundwater depth shown was determined at the time of drilling.

TABLE 3.1-16

SUBSURFACE SOIL SAMPLE RADIOLOGICAL RESULTS
BORING CB-1

Sample Number	Depths Represented (ft)	Average Depth (ft)	Co-60 (pCi/kg)	Cs-137 (pCi/kg)
S-1	0-0.5	0.3	90	79
S-2A	0.5-0.9	0.7	-	-
S-2B	0.9-1.4	1.2	-	-
S-2C	1.4-1.9	1.7	-	-
S-3A	2.0-2.6	2.3	-	-
S-3B	2.6-3.1	2.9	-	-
S-4	4.0-4.5	4.3	-	-
S-5A	6.5-7.5	7.0	-	-
S-5B	7.5-8.3	7.9	-	-
S-6	8.5	8.5	NR	NR
S-7A	13.0-13.8	13.4	-	-
S-7B	13.8-14.2	14.0	-	-
S-8A	16.0-16.7	16.4	-	-
S-8B	16.7-17.1	16.9	-	-
S-9A	18.0-18.5	18.3	-	-
S-9B	18.5-18.9	18.7	-	-
S-10A	21.0-21.7	21.4	-	-
S-10B	21.7-22.1	21.9	-	-
S-11	23.0-23.6	23.6	-	-
S-12	25.0-25.5	25.3	-	-

TABLE 3.1-16 (Continued)

SUBSURFACE SOIL SAMPLE RADIOLOGICAL RESULTS
BORING CB-2

Sample Number	Depths Represented (ft)	Average Depth (ft)	Co-60 (pCi/kg)	Cs-137 (pCi/kg)
S-1	0-0.5	0.3	-	-
S-2A	0.5-1.0	0.8	-	-
S-2B	1.0-1.5	1.3	-	-
S-2C	1.5-2.0	1.8	-	-
S-3A	2.0-2.7	2.4	-	-
S-3B	2.7-3.5	3.1	-	-
S-4A	4.0-4.6	4.3	-	-
S-4B	4.6-5.2	4.9	-	-
S-5	6.0-6.5	6.3	-	-
S-6A	6.5-7.3	6.9	-	-
S-6B	7.3-8.0	7.7	-	-
S-7A	9.0-9.4	9.2	-	-
S-7B	9.4-10.2	9.8	-	-
S-8A	11.0-11.6	11.3	-	-
S-8B	11.6-12.2	11.9	-	-
S-9A	13.0-13.7	13.4	-	-
S-9B	13.7-14.4	14.1	-	-
S-10	15.0	15.0	NR	NR
S-11A	20.5-21.2	20.9	-	-
S-11B	21.2-21.8	21.5	-	-

TABLE 3.1-16 (Continued)

SUBSURFACE SOIL SAMPLE RADIOLOGICAL RESULTS BORING CB-3

Sample Number	Depths Represented (ft)	Average Depth (ft)	Co-60 (pCi/kg)	Cs-137 (pCi/kg)
S-1	0-0.5	0.3	-	-
S-2A	1.0-1.5	1.3	-	76
S-2B	1.5-1.9	1.7	-	216
S-2C	1.9-2.3	2.1	-	-
S-3	2.5-3.5	3.0	-	-
S-4	5.0	5.0	NR	NR
S-5A	6.0-6.4	6.2	-	-
S-5B	6.4-6.9	6.6	-	-
S-5C	6.9-7.4	7.2	-	-
S-6A	8.0-8.5	8.3	-	-
S-6B	8.5-8.9	8.7	-	-
S-6C	8.9-9.3	9.1	-	-
S-7	10.0-10.4	10.2	-	-
S-8	10.8-11.5	11.2	-	-
S-9	14.0-14.8	14.4	-	-

TABLE 3.1-16 (Continued)

SUBSURFACE SOIL SAMPLE RADIOLOGICAL RESULTS
BORING CB-4

Sample Number	Depths Represented (ft)	Average Depth (ft)	Co-60 (pCi/kg)	Cs-137 (pCi/kg)
S-1	0.0-0.5	0.3	-	-
S-2A	2.0-2.8	2.4	-	-
S-2B	2.8-3.6	2.1	-	-
S-3A	4.0-4.4	4.2	-	-
S-3B	4.4-4.9	4.7	-	-
S-4A	5.0-5.6	5.3	-	-
S-4B	5.6-6.0	5.8	-	-
S-5	7.0	7.0	NR	NR
S-6A	9.0-10.0	9.5	-	-
S-6B	10.0-11.0	10.5	-	-
S-7A	11.0-12.0	11.5	-	-
S-7B	12.0-13.0	12.5	-	-
S-8	13.0	13.0	NR	NR
S-9A	14.0-14.7	14.4	-	-
S-9B	14.7-15.4	15.1	-	-
S-10A	16.0-16.6	16.3	-	-
S-10B	16.6-17.1	16.9	-	-
S-11A	18-18.5	18.3	-	-
S-11B	18.5-19.0	18.8	-	-

NOTES:
1. NR indicates no recovery of sample during boring.
2. A hyphen indicates sample tested but radionuclide not detected. A posteriori MDCs for Co-60 ranged from 34 to 52 pCi/kg and for Cs-137 from 35 to 57 pCi/kg.
3. Radionuclides other than Co-60, Cs-137 and naturally-occurring radionuclides not detected in samples.

TABLE 3.1-17

SUMMARY OF TEST PIT SOIL SAMPLE ANALYSES

Test Pit Number	Sample Number	Depths Represented (ft)	Cs-137 (pCi/kg)
TP-1	S-1	0-1	-
TP-1	S-2	2-4	119
TP-1	S-3	5-6	-
TP-2	S-1	0-1	-
TP-2	S-2	1-2	-
TP-2	S-3	3-4	-
TP-2	S-4	5-6	-
TP-3	S-1	0-1	-
TP-3	S-2	2-3	-
TP-3	S-3	3-4	-
TP-4	S-1	0-1	95
TP-4	S-2	2-3	64
TP-4	S-3	4-5	-
TP-4	S-4	6-7	-

NOTES:

1. A hyphen indicates radionuclide not detected. A posteriori MDCs for Cs-137 ranges from 27 to 72 pCi/kg.
2. Radionuclides other than Cs-137 and naturally-occurring radionuclides not detected in samples.

TABLE 3.1-18

GROUNDWATER OBSERVATION WELLS AT THE SITE

Observation Well Number	Depth to Groundwater (ft)	Total Depth (ft)	Location on Site
CB-1	9.4	25.5	NE of Fuel Building
CB-2	11.0	25.0	NE of Office Building
CB-3	3.9	15.0	E of Fire Water Tank
CB-4	11.3	20.0	Old Septic Leaching Field
CW-1	7.2	21.0	NW of Diesel Building
CW-2	11.8	21.0	SE of Ion Exchange Pit
CW-3	8.3	23.0	NE of Plant Warehouse
CW-4	7.7	17.0	NW of Plant Warehouse
CW-5	6.3	21.5	SE of Service Building
CW-6	11.3	24.0	SW of Turbine Building
B-1	17.2	78.5	NE of Fuel Building
B-3	3.8	46.2	NE of Fire Water Tank

NOTES:

1. Wells B-1 and B-3 installed for prior studies 12/14/77 and 10/4/79, respectively.
2. Groundwater depth as determined at the time of drilling.

TABLE 3.1-19

SUMMARY OF GROUNDWATER ANALYSES

Well Number	Groundwater Depth (ft)	Sample Date	Unfiltered Gross Beta (pCi/kg)	Filtered Gross Beta (pCi/kg)	Gamma (pCi/kg)	Tritium (pCi/kg)
CB-1	11.1	6/18/93	17.2	17.6	-	8045
CB-2	12.8	6/18/93	16.9	13.9	-	3038
CB-3	4.1	6/28/93	74.0	5.2	K-40 83	-
CB-4	11.3	6/30/93	12.1	4.9	-	-
CW-1	7.5	6/28/93	11.4	12.3	-	-
CW-2	8.0	6/28/93	43.6	6.2	-	-
CW-3	8.2	6/18/93	25.4	26.8	-	-
CW-4	7.9	6/28/93	79.0	1.7	K-40 56	-
CW-5	8.5	6/18/93	31.5	25.3	-	-
CW-6	11.8	6/18/93	15.4	13.3	-	-
B-1	19.6	6/30/93	16.0	4.7	-	5435
B-3	5.4	6/30/93	11.9	6.8	-	-

NOTES:

1. A hyphen indicates that plant-related radioactivity was not detected.
2. Groundwater depth measured at the time of well installation.

TABLE 3.1-20

IN SITU GAMMA-RAY SPECTROMETRY MEASURMENTS SITE AREAS OUTSIDE THE RCA (pCi/kg)

Location on Figure 3.1-30	Ground Type	Naturally-occurring			Man-made				
		K-40	Th-232	U-238	Co-60	Cs-137	Ag-110m	Mn-54	Fe-59
IS-1	Soil	15,990	795	469		112			
IS-2	Soil	17,030	773	417		259			
IS-3	Soil	15,410	773	506		96			
IS-4	Soil	16,070	866	535					
IS-5	Soil	16,740	800	449	43	213			
IS-6	Soil	16,400	796	550					
IS-7	Gravel	10,900	531	336	80				
IS-8	Asphalt	10,950	584	520					
IS-9	Asphalt	11,570	737	531	45				
IS-10	Asphalt	11,650	715	461					
IS-11	Asphalt	9,940	587	413					
IS-12	Asphalt	9,520	548	450					
IS-13	Asphalt	10,520	582	517					
IS-14	Asphalt	11,060	568	489					
IS-15	Asphalt	7,120	344	374					
IS-16	Asphalt	7,460	516	388					
IS-17	Soil	10,530	581	351	57	140			
IS-18	Asphalt	11,570	688	548					
IS-19	Asphalt	11,730	667	562					
IS-20	Asphalt	11,840	589	482					
IS-21	Asphalt	9,720	462	456					
IS-22	Asphalt	9,640	455	431					
IS-23	Asphalt	10,530	546	478					
IS-24	Asphalt	10,630	478	504		82			

TABLE 3.1-20 (Continued)

IN SITU GAMMA-RAY SPECTROMETRY MEASUREMENTS SITE AREAS OUTSIDE THE RCA (pCi/kg)

Location on Figure 3.1-30	Ground Type	Naturally-occurring			Man-made				
		K-40	Th-232	U-238	Co-60	Cs-137	Ag-110m	Mn-54	Fe-59
IS-25	Asphalt	9,480	541	542		61			
IS-26	Soil	11,030	652	585		239			
IS-27	Asphalt	9,840	520	488	35	106			
IS-28	Asphalt	9,610	498	389					
IS-29	Soil	11,720	700	504	90	276			
IS-30	Asphalt	9,670	500	458					
IS-31	Soil	6,130	423			289			
IS-32	Soil	5,110	215	267		332			
IS-33	Soil	14,490	591	409		394			
IS-34	Soil	15,790	636	429		161			
IS-35	Soil	14,970	717	357	80				
IS-36	Soil	19,570	1,081	663					
IS-37	Soil	10,430	592	403		842			
IS-38	Soil	13,270	535	480		406			
IS-117	Soil	13,570	622	389					
IS-118	Soil	13,850	747	476		73			
	AVG	11,826	614	463	61	240			
	MAX	19,570	1,081	663	90	842			
	MIN	5,110	215	267	35	61			

TABLE 3.1-21

IN SITU GAMMA-RAY SPECTROMETRY MEASUREMENTS SITE AREAS INSIDE THE RCA (pCi/kg)

Location on Figure 3.1-29	Ground Type	Naturally-occurring			Man-made				
		K-40	Th-232	U-238	Co-60	Cs-137	Ag-110m	Mn-54	Fe-59
IS-39	Asphalt	8,260	514	437					
IS-40	Asphalt	9,270	519	471					
IS-41	Asphalt	7,990	388						
IS-42	Asphalt	8,260	598	319		170			
IS-44	Asphalt	9,270	730	650	212				
IS-45	Asphalt	9,160	513	337	198				
IS-47	Asphalt	9,750	606	510	339				
IS-48	Asphalt	9,750	700		1,515	10,030			
IS-49	Asphalt	10,280	514	379					
IS-50	Asphalt	10,300	584	490	304				
IS-51	Asphalt	11,130	649	551					
IS-52	Asphalt	9,590	610	490	251				
IS-53	Asphalt	9,590	600	380					
IS-54	Asphalt	10,120	319		3,900	1,640		360	
IS-55	Asphalt	9,370	480	450	352	385			
IS-56	Gravel	7,880	490						
IS-57	Gravel	9,850	830		860				
IS-58	Asphalt	10,120		279	204				
IS-59	Asphalt	11,400	542	277	142				
IS-60	Asphalt	9,640	434	315					
IS-61	Asphalt	9,690	621	530					
IS-62	Asphalt	8,730	341	600	226				
IS-63	Asphalt	8,680	364	509					
IS-64	Asphalt	9,530	583	463					
IS-65	Asphalt	6,280	600	442					
IS-66	Asphalt	7,240	421	296					
IS-67	Asphalt	9,690	527	357					
IS-68	Asphalt	7,940	359	472	201				

TABLE 3.1-21 (Continued)

IN SITU GAMMA-RAY SPECTROMETRY MEASUREMENTS
SITE AREAS INSIDE THE RCA
(pCi/kg)

Location on Figure 3.1-29	Ground Type	Naturally-occurring			Man-made				
		K-40	Th-232	U-238	Co-60	Cs-137	Ag-110m	Mn-54	Fe-59
IS-69	Asphalt	6,440	429	324					
IS-70	Asphalt	8,730	518	338					
IS-71	Asphalt	9,050	486	585	99	197			
IS-72	Asphalt	8,260	414	392					
IS-73	Asphalt	9,480	465	471					198
IS-74	Asphalt	6,820	348	341					
IS-75	Asphalt	6,340	335	317					
IS-76	Asphalt	6,280	356	390					
IS-77	Asphalt	5,910	315	324					
IS-78	Asphalt	5,060	457	230					
IS-79	Asphalt	5,110	283	370					
IS-80	Asphalt	6,440	389	421					
IS-81	Asphalt	6,390	424	381					
IS-82	Asphalt	5,490	331	346					
IS-83	Asphalt	7,140		410					
IS-85	Asphalt	6,820	405	340					
IS-86	Asphalt	7,720	518	286					
IS-87	Asphalt	7,940	440	301					
IS-88	Asphalt	5,590	319	251					
IS-89	Asphalt	10,970	432	490	291				
IS-91	Asphalt	11,340	690	465	486				
IS-92	Asphalt	9,050	620				570		
IS-93	Asphalt	10,120	630		1,375	2,370			
IS-94	Asphalt	8,200		580	890	820			
IS-95	Asphalt	8,630	850	550	284				
IS-96	Asphalt	11,770	570		960				
IS-97	Asphalt	6,760	418	296					
IS-98	Asphalt	6,920	404	360					

TABLE 3.1-21 (Continued)

IN SITU GAMMA-RAY SPECTROMETRY MEASUREMENTS SITE AREAS INSIDE THE RCA (pCi/kg)

Location on Figure 3.1-29	Ground Type	Naturally-occurring			Man-made				
		K-40	Th-232	U-238	Co-60	Cs-137	Ag-110m	Mn-54	Fe-59
IS-99	Asphalt	12,620	650	440	258				
IS-101	Asphalt	10,330	636	532	421	273			
IS-102	Asphalt	10,390	531	473					
IS-103	Asphalt	9,530	514	359					
IS-106	Asphalt	9,910	560	510					
IS-108	Asphalt	10,700	535	410		229			
IS-109	Asphalt	7,240	421	556		211			
IS-110	Asphalt	7,400	710	392					
IS-111	Asphalt	9,800	509	452	177	332			
IS-112	Asphalt	8,470	390						
IS-113	Asphalt	9,750	572	490					
IS-114	Gravel	11,980	543	349	364	206			
IS-115	Asphalt	9,480	550	450	423				
IS-116	Asphalt	9,850	458	390	228	542			
	AVG	8,728	505	416	575	1,339	570	360	198
	MAX	12,620	850	650	3,900	10,030	570	360	198
	MIN	5,060	283	230	99	170	570	360	198

TABLE 3.1-22

TERRESTRIAL EXPOSURE RATE ESTIMATES
SITE AREAS OUTSIDE THE RCA (μR/hr)

Location on Figure 3.1-30	Ground Type	Naturally-occurring			Total Natural Exposure Rate	Man-made					Total Man-made Exposure Rate	Total Combined Exposure Rate	% Man-made to Total Combined
		K-40	Th-232	U-238		Co-60	Cs-137	Ag-110m	Mn-54	Fe-59			
IS-1	SOIL	2.85	2.11	0.93	5.89		0.007				0.01	5.90	0.12%
IS-2	SOIL	3.03	2.06	0.83	5.92		0.016				0.02	5.93	0.27%
IS-3	SOIL	2.75	2.06	1.00	5.81		0.006				0.01	5.81	0.10%
IS-4	SOIL	2.86	2.30	1.06	6.23						0.00	6.23	0.00%
IS-5	SOIL	2.98	2.13	0.89	6.00	0.012	0.013				0.03	6.03	0.42%
IS-6	SOIL	2.92	2.12	1.09	6.13						0.00	6.13	0.00%
IS-7	GRAVEL	1.94	1.41	0.67	4.02	0.023					0.02	4.04	0.57%
IS-8	ASPHALT	1.95	1.55	1.03	4.53						0.00	4.53	0.00%
IS-9	ASPHALT	2.06	1.96	1.05	5.08	0.013					0.01	5.09	0.26%
IS-10	ASPHALT	2.08	1.90	0.91	4.89						0.00	4.89	0.00%
IS-11	ASPHALT	1.77	1.56	0.82	4.15						0.00	4.15	0.00%
IS-12	ASPHALT	1.70	1.46	0.89	4.04						0.00	4.04	0.00%
IS-13	ASPHALT	1.87	1.55	1.02	4.45						0.00	4.45	0.00%
IS-14	ASPHALT	1.97	1.51	0.97	4.45						0.00	4.45	0.00%
IS-15	ASPHALT	1.27	0.92	0.74	2.93						0.00	2.93	0.00%
IS-16	ASPHALT	1.33	1.37	0.77	3.47						0.00	3.47	0.00%
IS-17	SOIL	1.88	1.55	0.70	4.12	0.016	0.009				0.03	4.14	0.60%
IS-18	ASPHALT	2.06	1.83	1.09	4.98						0.00	4.98	0.00%
IS-19	ASPHALT	2.09	1.77	1.11	4.98						0.00	4.98	0.00%

TABLE 3.1-22 (Continued)

TERRESTRIAL EXPOSURE RATE ESTIMATES
SITE AREAS OUTSIDE THE RCA (μR/hr)

Location on Figure 3.1-30	Ground Type	Naturally-occurring			Total Natural Exposure Rate	Man-made					Total Man-made Exposure Rate	Total Combined Exposure Rate	% Man-made to Total Combined
		K-40	Th-232	U-238		Co-60	Cs-137	Ag-110m	Mn-54	Fe-59			
IS-20	ASPHALT	2.11	1.57	0.96	4.63						0.00	4.63	0.00%
IS-21	ASPHALT	1.73	1.23	0.90	3.86						0.00	3.86	0.00%
IS-22	ASPHALT	1.72	1.21	0.86	3.78						0.00	3.78	0.00%
IS-23	ASPHALT	1.88	1.45	0.95	4.28						0.00	4.28	0.00%
IS-24	ASPHALT	1.89	1.27	1.00	4.16		0.005				0.01	4.17	0.12%
IS-25	ASPHALT	1.69	1.44	1.08	4.20		0.004				0.00	4.21	0.09%
IS-26	SOIL	1.96	1.74	1.16	4.86		0.015				0.01	4.87	0.30%
IS-27	ASPHALT	1.75	1.39	0.97	4.11	0.010	0.007				0.02	4.12	0.40%
IS-28	ASPHALT	1.71	1.33	0.77	3.81						0.00	3.81	0.00%
IS-29	SOIL	2.09	1.86	1.00	4.95	0.026	0.017				0.04	4.99	0.86%
IS-30	ASPHALT	1.72	1.33	0.91	3.96						0.00	3.96	0.00%
IS-31	SOIL	1.09	1.12		2.21		0.018				0.02	2.23	0.80%
IS-32	SOIL	0.91	0.57	0.53	2.01		0.021				0.02	2.03	1.01%
IS-33	SOIL	2.58	1.57	0.81	4.96		0.024				0.02	4.98	0.49%
IS-34	SOIL	2.81	1.69	0.85	5.35		0.010				0.01	5.36	0.19%
IS-35	SOIL	2.67	1.91	0.71	5.29	0.023					0.02	5.31	0.43%
IS-36	SOIL	3.49	2.88	1.31	7.68						0.00	7.68	0.00%
IS-37	SOIL	1.86	1.58	0.80	4.23		0.052				0.05	4.28	1.21%
IS-38	SOIL	2.36	1.42	0.95	4.74		0.025				0.03	4.76	0.52%

TABLE 3.1-22 (Continued)

TERRESTRIAL EXPOSURE RATE ESTIMATES SITE AREAS OUTSIDE THE RCA (μR/hr)

Location on Figure 3.1-30	Ground Type	Naturally-occurring			Total Natural Exposure Rate	Man-made					Total Man-made Exposure Rate	Total Combined Exposure Rate	% Man-made to Total Combined
		K-40	Th-232	U-238		Co-60	Cs-137	Ag-110m	Mn-54	Fe-59			
IS-117	SOIL	2.42	1.65	0.77	4.84						0.00	4.84	0.00%
IS-118	SOIL	2.47	1.99	0.94	5.40		0.005				0.00	5.40	0.08%
	AVG	2.11	1.63	0.92	4.63	0.018	0.015				0.01	4.64	0.22%
	MAX	3.49	2.88	1.31	7.68	0.026	0.052				0.05	7.68	1.21%
	MIN	0.91	0.57	0.53	2.01	0.010	0.004				0.00	2.03	0.00%

Note: Terrestrial exposure rates derived from in situ gamma-ray spectra obtained with a HPGe detector (see Table 3.1-20)

TABLE 3.1-23

TERRESTRIAL EXPOSURE RATE ESTIMATES SITE AREAS INSIDE THE RCA (μR/hr)

Location on Figure 3.1-29	Ground Type	Naturally-occurring			Total Natural Exposure Rate	Man-made					Total Man-made Exposure Rate	Total Combined Exposure Rate	% Man-made to Total Combined
		K-40	Th-232	U-238		Co-60	Cs-137	Ag-110m	Mn-54	Fe-59			
IS-39	ASPHALT	1.47	1.37	0.87	3.71						0.00	3.71	0.00%
IS-40	ASPHALT	1.65	1.38	0.93	3.96						0.00	3.96	0.00%
IS-41	ASPHALT	1.42	1.03		2.45						0.00	2.45	0.00%
IS-42	ASPHALT	1.47	1.59	0.63	3.69		0.011				0.01	3.70	0.28%
IS-44	ASPHALT	1.65	1.95	1.28	4.88	0.061					0.06	4.94	1.23%
IS-45	ASPHALT	1.63	1.37	0.67	3.67	0.057					0.06	3.73	1.53%
IS-47	ASPHALT	1.74	1.61	1.01	4.36	0.098					0.10	4.46	2.20%
IS-48	ASPHALT	1.74	1.87		3.61	0.437	0.619				1.06	4.67	22.63%
IS-49	ASPHALT	1.83	1.37	0.75	3.95						0.00	3.95	0.00%
IS-50	ASPHALT	1.84	1.56	0.97	4.37	0.088					0.09	4.46	1.97%
IS-51	ASPHALT	1.98	1.73	1.09	4.80						0.00	4.80	0.00%
IS-52	ASPHALT	1.71	1.62	0.97	4.30	0.072					0.07	4.37	1.65%
IS-53	ASPHALT	1.71	1.60		3.31						0.00	3.31	0.00%
IS-54	ASPHALT	1.80	0.85		2.65	1.126	0.101		0.034		1.26	3.91	32.24%
IS-55	ASPHALT	1.67	1.27	0.90	3.84	0.101	0.024				0.12	3.96	3.15%
IS-56	GRAVEL	1.40	1.30		2.70						0.00	2.70	0.00%
IS-57	GRAVEL	1.75	2.21		3.96	0.248					0.25	4.21	5.89%
IS-58	ASPHALT	1.80		0.55	2.35	0.059					0.06	2.41	2.45%
IS-59	ASPHALT	2.03	1.44	0.55	4.02	0.041					0.04	4.06	1.01%

TABLE 3.1-23 (Continued)

TERRESTRIAL EXPOSURE RATE ESTIMATES
SITE AREAS INSIDE THE RCA (μR/hr)

Location on Figure 3.1-29	Ground Type	Naturally-occurring			Total Natural Exposure Rate	Man-made					Total Man-made Exposure Rate	Total Combined Exposure Rate	% Man-made to Total Combined
		K-40	Th-232	U-238		Co-60	Cs-137	Ag-110m	Mn-54	Fe-59			
IS-60	ASPHALT	1.72	1.15	0.62	3.49						0.00	3.49	0.00%
IS-61	ASPHALT	1.73	1.65	1.06	4.44						0.00	4.44	0.00%
IS-62	ASPHALT	1.56	0.91	1.18	3.65	0.065					0.07	3.72	1.75%
IS-63	ASPHALT	1.55	0.97	1.01	3.53						0.00	3.53	0.00%
IS-64	ASPHALT	1.70	1.55	0.92	4.17						0.00	4.17	0.00%
IS-65	ASPHALT	1.12	1.59	0.88	3.59						0.00	3.59	0.00%
IS-66	ASPHALT	1.29	1.12	0.59	3.00						0.00	3.00	0.00%
IS-67	ASPHALT	1.73	1.40	0.71	3.84						0.00	3.84	0.00%
IS-68	ASPHALT	1.41	0.95	0.94	3.30	0.058					0.06	3.36	1.73%
IS-69	ASPHALT	1.15	1.14	0.64	2.93						0.00	2.93	0.00%
IS-70	ASPHALT	1.56	1.38	0.67	3.61						0.00	3.61	0.00%
IS-71	ASPHALT	1.61	1.29	1.16	4.06	0.029	0.012				0.04	4.10	0.99%
IS-72	ASPHALT	1.47	1.10	0.78	3.35						0.00	3.35	0.00%
IS-73	ASPHALT	1.69	1.24	0.93	3.86					0.027	0.03	3.89	0.69%
IS-74	ASPHALT	1.21	0.92	0.68	2.81						0.00	2.81	0.00%
IS-75	ASPHALT	1.13	0.89	0.63	2.65						0.00	2.65	0.00%
IS-76	ASPHALT	1.12	0.95	0.77	2.84						0.00	2.84	0.00%
IS-77	ASPHALT	1.05	0.84	0.64	2.53						0.00	2.53	0.00%
IS-78	ASPHALT	0.90	1.22	0.46	2.58						0.00	2.58	0.00%

TABLE 3.1-23 (Continued)

TERRESTRIAL EXPOSURE RATE ESTIMATES
SITE AREAS INSIDE THE RCA (μR/hr)

Location on Figure 3.1-29	Ground Type	Naturally-occurring			Total Natural Exposure Rate	Man-made					Total Man-made Exposure Rate	Total Combined Exposure Rate	% Man-made to Total Combined
		K-40	Th-232	U-238		Co-60	Cs-137	Ag-110m	Mn-54	Fe-59			
IS-79	ASPHALT	0.91	0.75	0.73	2.39						0.00	2.39	0.00%
IS-80	ASPHALT	1.15	1.04	0.83	3.02						0.00	3.02	0.00%
IS-81	ASPHALT	1.14	1.13	0.76	3.03						0.00	3.03	0.00%
IS-82	ASPHALT	0.98	0.88	0.69	2.55						0.00	2.55	0.00%
IS-83	ASPHALT	1.27		0.81	2.08						0.00	2.08	0.00%
IS-85	ASPHALT	1.21	1.08	0.67	2.96						0.00	2.96	0.00%
IS-86	ASPHALT	1.38	1.38	0.57	3.33						0.00	3.33	0.00%
IS-87	ASPHALT	1.41	1.17	0.60	3.18						0.00	3.18	0.00%
IS-88	ASPHALT	1.00	0.85	0.50	2.35						0.00	2.35	0.00%
IS-89	ASPHALT	1.95	1.15	0.97	4.07	0.084					0.08	4.15	2.02%
IS-91	ASPHALT	2.02	1.84	0.92	4.78	0.140					0.14	4.92	2.85%
IS-92	ASPHALT	1.61	1.64		3.25			0.172			0.17	3.42	5.03%
IS-93	ASPHALT	1.80	1.69		3.49	0.397	0.146				0.54	4.03	13.46%
IS-94	ASPHALT	1.46		1.14	2.60	0.257	0.051				0.31	2.91	10.58%
IS-95	ASPHALT	1.54	2.26	1.08	4.88	0.082					0.08	4.96	1.65%
IS-96	ASPHALT	2.10	1.52		3.62	0.277					0.28	3.90	7.11%
IS-97	ASPHALT	1.20	1.11	0.59	2.90						0.00	2.90	0.00%
IS-98	ASPHALT	1.23	1.08	0.72	3.03						0.00	3.03	0.00%
IS-99	ASPHALT	2.25	1.74	0.87	4.86	0.075					0.08	4.94	1.52%

TABLE 3.1-23 (Continued)

TERRESTRIAL EXPOSURE RATE ESTIMATES
SITE AREAS INSIDE THE RCA (μR/hr)

Location on Figure 3.1-29	Ground Type	Naturally-occurring			Total Natural Exposure Rate	Man-made					Total Man-made Exposure Rate	Total Combined Exposure Rate	% Man-made to Total Combined
		K-40	Th-232	U-238		Co-60	Cs-137	Ag-110m	Mn-54	Fe-59			
IS-101	ASPHALT	1.84	1.69	1.05	4.58	0.122	0.017				0.14	4.72	2.94%
IS-102	ASPHALT	1.85	1.41	0.94	4.20						0.00	4.20	0.00%
IS-103	ASPHALT	1.70	1.37	0.71	3.78						0.00	3.78	0.00%
IS-106	ASPHALT	1.76	1.49	1.01	4.26						0.00	4.26	0.00%
IS-108	ASPHALT	1.91	1.42	0.81	4.14		0.014				0.01	4.15	0.34%
IS-109	ASPHALT	1.29	1.12	1.10	3.51		0.013				0.01	3.52	0.37%
IS-110	ASPHALT	1.32	1.89	0.78	3.99						0.00	3.99	0.00%
IS-111	ASPHALT	1.75	1.35	0.90	4.00	0.051	0.021				0.07	4.07	1.76%
IS-112	ASPHALT	1.51	1.03		2.54						0.00	2.54	0.00%
IS-113	ASPHALT	1.74	1.52	0.97	4.23						0.00	4.23	0.00%
IS-114	GRAVEL	2.13	1.44	0.69	4.26	0.105	0.013				0.12	4.38	2.69%
IS-115	ASPHALT	1.69	1.47	0.90	4.06	0.122					0.12	4.18	2.92%
IS-116	ASPHALT	1.75	1.22	0.77	3.74	0.066	0.033				0.10	3.84	2.59%
	AVG	1.55	1.34	0.83	3.55	0.166	0.083	0.172	0.034	0.027	0.08	3.63	1.99%
	MAX	2.25	2.26	1.28	4.88	1.126	0.619	0.172	0.034	0.027	1.26	4.96	32.24%
	MIN	0.90	0.75	0.46	2.08	0.029	0.011	0.172	0.034	0.027	0.00	2.08	0.00%

Note: Terrestrial exposure rates derived from in situ gamma-ray spectra obtained with a HPGe detector (see Table 3.1-21)

TABLE 3.1-24

ENVIRONMENTAL EXPOSURE RATE ESTIMATES
SITE AREAS OUTSIDE THE RCA
(μR/hr)

Location on Figure 3.1-30	Ground Type	Natural Terrestrial (a)	Cosmic (b)	Total Natural Bkgrd (a)+(b)= (c)	Man-made Terrestrial (d)	Environmental Exposure Rate (c)+(d)= (e)	PIC Results (f)	PIC vs Environmental (f)-(e)= (g)
IS-1	Soil	5.89	4.26	10.15	0.01	10.16	12.0	1.84
IS-2	Soil	5.92	4.26	10.18	0.02	10.20	11.6	1.40
IS-3	Soil	5.81	4.26	10.07	0.01	10.08	11.2	1.12
IS-4	Soil	6.23	4.26	10.49	0.00	10.49	11.9	1.41
IS-5	Soil	6.00	4.26	10.26	0.03	10.29	12.1	1.81
IS-6	Soil	6.13	4.26	10.39	0.00	10.39	12.6	2.21
IS-7	Gravel	4.02	4.26	8.28	0.02	8.30	12.2	3.90
IS-8	Asphalt	4.53	4.26	8.79	0.00	8.79	13.0	4.21
IS-9	Asphalt	5.08	4.26	9.34	0.01	9.35	12.2	2.85
IS-10	Asphalt	4.89	4.26	9.15	0.00	9.15	13.2	4.05
IS-11	Asphalt	4.15	4.26	8.41	0.00	8.41	11.5	3.09
IS-12	Asphalt	4.04	4.26	8.30	0.00	8.30	12.3	4.00
IS-13	Asphalt	4.45	4.26	8.71	0.00	8.71	10.9	2.19
IS-14	Asphalt	4.45	4.26	8.71	0.00	8.71	11.5	2.79
IS-15	Asphalt	2.93	4.26	7.19	0.00	7.19	10.4	3.21
IS-16	Asphalt	3.47	4.26	7.73	0.00	7.73	11.5	3.77
IS-17	Soil	4.12	4.26	8.38	0.03	8.41	11.3	2.89
IS-18	Asphalt	4.98	4.26	9.24	0.00	9.24	12.3	3.06
IS-19	Asphalt	4.98	4.26	9.24	0.00	9.24	12.0	2.76
IS-20	Asphalt	4.63	4.26	8.89	0.00	8.89	11.4	2.51
IS-21	Asphalt	3.86	4.26	8.12	0.00	8.12	10.0	1.88
IS-22	Asphalt	3.78	4.26	8.04	0.00	8.04	10.3	2.26
IS-23	Asphalt	4.28	4.26	8.54	0.00	8.54	NM	
IS-24	Asphalt	4.16	4.30	8.46	0.01	8.47	10.1	1.63
IS-25	Asphalt	4.20	4.42	8.62	0.00	8.62	10.3	1.68
IS-26	Soil	4.86	4.42	9.28	0.01	9.29	11.0	1.71
IS-27	Asphalt	4.11	4.42	8.53	0.02	8.55	11.0	2.45

TABLE 3.1-24 (Continued)

ENVIRONMENTAL EXPOSURE RATE ESTIMATES SITE AREAS OUTSIDE THE RCA (μR/hr)

Location on Figure 3.1-30	Ground Type	Natural Terrestrial (a)	Cosmic (b)	Total Natural Bkgrd (a)+(b)= (c)	Man-made Terrestrial (d)	Environmental Exposure Rate (c)+(d)= (e)	PIC Results (f)	PIC vs Environmental (f)-(e)= (g)
IS-28	Asphalt	3.81	4.42	8.23	0.00	8.23	12.1	3.87
IS-29	Soil	4.95	4.42	9.37	0.04	9.41	14.0	4.59
IS-30	Asphalt	3.96	4.42	8.38	0.00	8.38	15.7	7.32
IS-31	Soil	2.21	4.28	6.49	0.02	6.51	32.8	26.29
IS-32	Soil	2.01	4.28	6.29	0.02	6.31	24.5	18.19
IS-33	Soil	4.96	4.32	9.28	0.02	9.30	28.3	19.00
IS-34	Soil	5.35	4.32	9.67	0.01	9.68	20.6	10.92
IS-35	Soil	5.29	4.32	9.61	0.02	9.63	53.6	43.97
IS-36	Soil	7.68	3.98	11.66	0.00	11.66	11.3	-0.36
IS-37	Soil	4.23	4.10	8.33	0.05	8.38	8.9	0.52
IS-38	Soil	4.74	4.08	8.82	0.03	8.85	9.1	0.25
IS-117	Soil	4.84	4.08	8.92	0.00	8.92	9.4	0.48
IS-118	Soil	5.40	4.08	9.48	0.00	9.48	9.9	0.42
	AVG	4.63	4.27	8.90	0.01	8.91	14.10	5.18
	MAX	7.68	4.42	11.66	0.05	11.66	53.60	43.97
	MIN	2.01	3.98	6.29	0.00	6.31	8.90	-0.36

NM: No measurement made at this location.
(a): Calculated from HPGe measurements for naturally-occuring nuclides (Table 3.1-22).
(b): Calculated value based on site barometric pressure and temperature data taken on day of measurement.
(d): Calculated from HPGe measurements for fission and activation products (Table 3.1-22).
(f): Direct measurement of exposure rate using a Pressurized Ion Chamber (PIC).
(g): Difference between measured exposure rate (PIC) and calculated environmental exposure rate from HPGe measurements. Includes any instrument bias between PIC and HPGe (typically +/- 15%) and any direct radiation from plant structures and components.

TABLE 3.1-25

ENVIRONMENTAL EXPOSURE RATE ESTIMATES SITE AREAS INSIDE THE RCA (μR/hr)

Location on Figure 3.1-29	Ground Type	Natural Terrestrial (a)	Cosmic (b)	Total Natural Bkgrnd (a)+(b)= (c)	Man-made Terrestrial (d)	Environmental Exposure Rate (c)+(d)= (e)	PIC Results (f)	PIC vs Environmental (f)-(e)= (g)
IS-39	Asphalt	3.71	4.09	7.80	0.00	7.80	113.6	105.80
IS-40	Asphalt	3.96	4.09	8.05	0.00	8.05	68.5	60.45
IS-41	Asphalt	2.45	4.09	6.54	0.00	6.54	60.8	54.26
IS-42	Asphalt	3.69	4.09	7.78	0.01	7.79	90.2	82.41
IS-44	Asphalt	4.88	4.42	9.30	0.06	9.36	148.2	138.84
IS-45	Asphalt	3.67	4.09	7.76	0.06	7.82	141.4	133.58
IS-47	Asphalt	4.36	4.09	8.45	0.10	8.55	189	180.45
IS-48	Asphalt	3.61	4.42	8.03	1.06	9.09	129.9	120.81
IS-49	Asphalt	3.95	4.42	8.37	0.00	8.37	74.8	66.43
IS-50	Asphalt	4.37	4.42	8.79	0.09	8.88	60.2	51.32
IS-51	Asphalt	4.80	4.42	9.22	0.00	9.22	69.6	60.38
IS-52	Asphalt	4.30	4.42	8.72	0.07	8.79	101.6	92.81
IS-53	Asphalt	3.31	4.42	7.73	0.00	7.73	131.9	124.17
IS-54	Asphalt	2.65	4.42	7.07	1.26	8.33	248.4	240.07
IS-55	Asphalt	3.84	4.42	8.26	0.12	8.38	164.7	156.32
IS-56	Gravel	2.70	4.25	6.95	0.00	6.95	235	228.05
IS-57	Gravel	3.96	4.25	8.21	0.25	8.46	274.1	265.64
IS-58	Asphalt	2.35	4.25	6.60	0.06	6.66	90.5	83.84
IS-59	Asphalt	4.02	4.25	8.27	0.04	8.31	42.8	34.49
IS-60	Asphalt	3.49	4.25	7.74	0.00	7.74	52.8	45.06
IS-61	Asphalt	4.44	4.25	8.69	0.00	8.69	64.9	56.21
IS-62	Asphalt	3.65	4.35	8.00	0.07	8.07	116.2	108.14
IS-63	Asphalt	3.53	4.35	7.88	0.00	7.88	47.6	39.72
IS-64	Asphalt	4.17	4.35	8.52	0.00	8.52	62.5	53.98
IS-65	Asphalt	3.59	4.35	7.94	0.00	7.94	85.3	77.36
IS-66	Asphalt	3.00	4.35	7.35	0.00	7.35	58.3	50.95
IS-67	Asphalt	3.84	4.35	8.19	0.00	8.19	33.2	25.01

TABLE 3.1-25 (Continued)

ENVIRONMENTAL EXPOSURE RATE ESTIMATES
SITE AREAS INSIDE THE RCA
(μR/hr)

Location on Figure 3.1-29	Ground Type	Natural Terrestrial (a)	Cosmic (b)	Total Natural Bkgrnd (a)+(b)= (c)	Man-made Terrestrial (d)	Environmental Exposure Rate (c)+(d)= (e)	PIC Results (f)	PIC vs Environmental (f)-(e)= (g)
IS-68	Asphalt	3.30	4.35	7.65	0.06	7.71	37.4	29.69
IS-69	Asphalt	2.93	4.35	7.28	0.00	7.28	30.8	23.52
IS-70	Asphalt	3.61	4.35	7.96	0.00	7.96	28.9	20.94
IS-71	Asphalt	4.06	4.36	8.42	0.04	8.46	33.7	25.24
IS-72	Asphalt	3.35	4.36	7.71	0.00	7.71	30.3	22.59
IS-73	Asphalt	3.86	4.36	8.22	0.03	8.25	27.5	19.25
IS-74	Asphalt	2.81	4.01	6.82	0.00	6.82	22.8	15.98
IS-75	Asphalt	2.65	4.01	6.66	0.00	6.66	54.4	47.74
IS-76	Asphalt	2.84	4.01	6.85	0.00	6.85	38.4	31.55
IS-77	Asphalt	2.53	4.14	6.67	0.00	6.67	21.6	14.93
IS-78	Asphalt	2.58	4.14	6.72	0.00	6.72	20.5	13.78
IS-79	Asphalt	2.39	4.14	6.53	0.00	6.53	17.3	10.77
IS-80	Asphalt	3.02	4.14	7.16	0.00	7.16	35.7	28.54
IS-81	Asphalt	3.03	4.14	7.17	0.00	7.17	43.1	35.93
IS-82	Asphalt	2.55	4.25	6.80	0.00	6.80	47.5	40.70
IS-83	Asphalt	2.08	3.88	5.96	0.00	5.96	142.3	136.34
IS-85	Asphalt	2.96	3.88	6.84	0.00	6.84	120.3	113.46
IS-86	Asphalt	3.33	3.88	7.21	0.00	7.21	100.8	93.59
IS-87	Asphalt	3.18	4.19	7.37	0.00	7.37	97.2	89.83
IS-88	Asphalt	2.35	4.01	6.36	0.00	6.36	22.5	16.14
IS-89	Asphalt	4.07	4.19	8.26	0.08	8.34	257.7	249.36
IS-91	Asphalt	4.78	4.19	8.97	0.14	9.11	162.6	153.49
IS-92	Asphalt	3.25	4.26	7.51	0.17	7.68	259	251.32
IS-93	Asphalt	3.49	4.19	7.68	0.54	8.22	135	126.78
IS-94	Asphalt	2.60	4.19	6.79	0.31	7.10	440	432.90
IS-95	Asphalt	4.88	4.26	9.14	0.08	9.22	394.3	385.08
IS-96	Asphalt	3.62	4.26	7.88	0.28	8.16	>500	491.84

TABLE 3.1-25 (Continued)

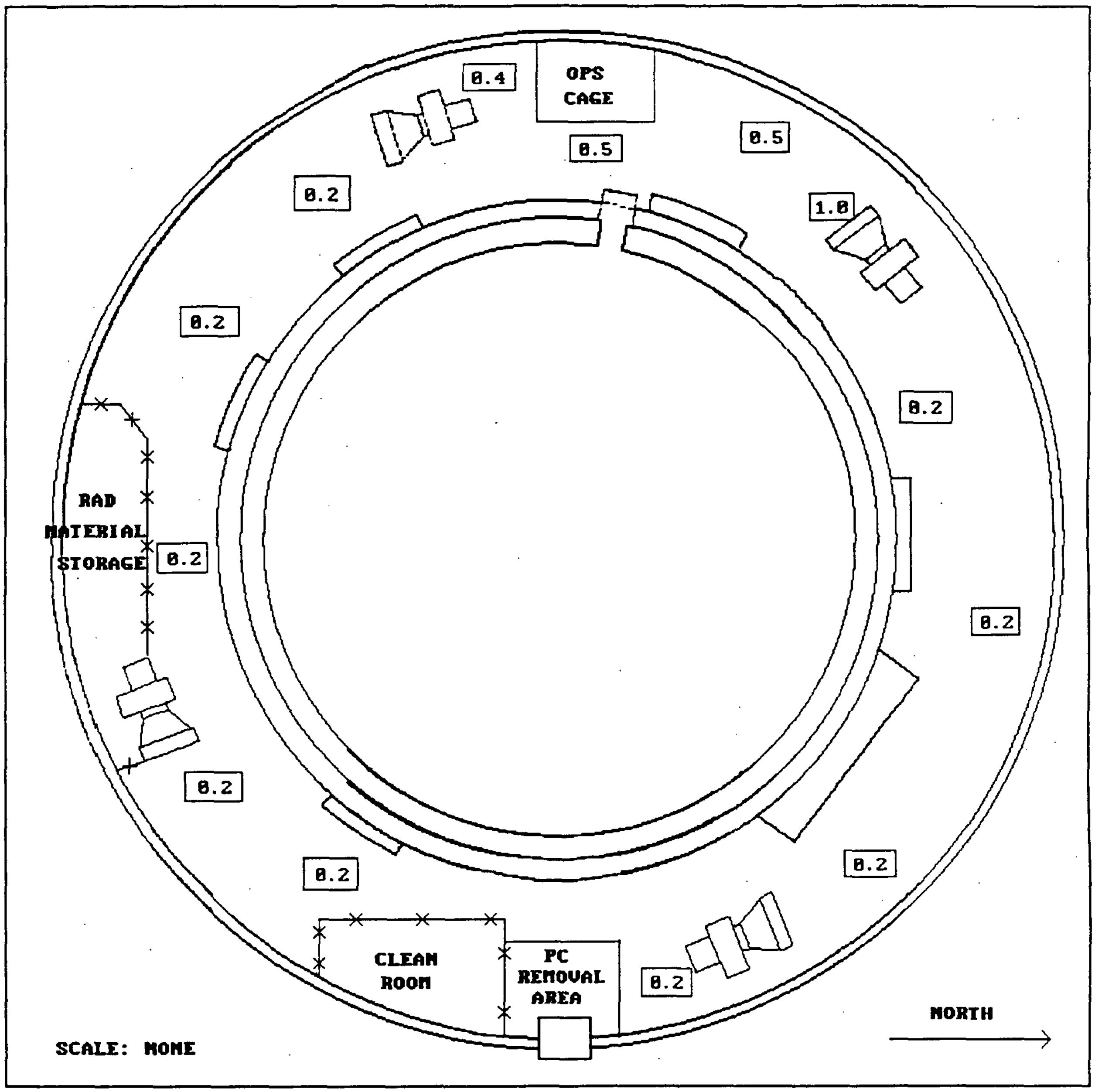
ENVIRONMENTAL EXPOSURE RATE ESTIMATES
SITE AREAS INSIDE THE RCA
(μR/hr)

Location on Figure 3.1-29	Ground Type	Natural Terrestrial (a)	Cosmic (b)	Total Natural Bkgrnd (a)+(b)= (c)	Man-made Terrestrial (d)	Environmental Exposure Rate (c)+(d)= (e)	PIC Results (f)	PIC vs Environmental (f)-(e)= (g)
IS-97	Asphalt	2.90	4.19	7.09	0.00	7.09	69.6	62.51
IS-98	Asphalt	3.03	4.26	7.29	0.00	7.29	224.2	216.91
IS-99	Asphalt	4.86	4.26	9.12	0.08	9.20	220.3	211.11
IS-101	Asphalt	4.58	4.45	9.03	0.14	9.17	21.5	12.33
IS-102	Asphalt	4.20	4.33	8.53	0.00	8.53	20.5	11.97
IS-103	Asphalt	3.78	4.45	8.23	0.00	8.23	24	15.77
IS-106	Asphalt	4.26	4.45	8.71	0.00	8.71	54.6	45.89
IS-108	Asphalt	4.14	4.26	8.40	0.01	8.41	124.1	115.69
IS-109	Asphalt	3.51	4.33	7.84	0.01	7.85	80.4	72.55
IS-110	Asphalt	3.99	4.33	8.32	0.00	8.32	155.3	146.98
IS-111	Asphalt	4.00	4.26	8.26	0.07	8.33	93.5	85.17
IS-112	Asphalt	2.54	4.19	6.73	0.00	6.73	>500	493.27
IS-113	Asphalt	4.23	4.45	8.68	0.00	8.68	20.5	11.82
IS-114	Gravel	4.26	4.45	8.71	0.12	8.83	25.7	16.87
IS-115	Asphalt	4.06	4.45	8.51	0.12	8.63	70.2	61.57
IS-116	Asphalt	3.74	4.45	8.19	0.10	8.29	41.3	33.01
	AVG	3.55	4.26	7.81	0.08	7.89	111.76	103.87
	MAX	4.88	4.45	9.30	1.26	9.36	>500.00	493.27
	MIN	2.08	3.88	5.96	0.00	5.96	17.30	10.77

(a): Calculated from HPGe measurements for naturally-occuring nuclides (Table 3.1-23).
(b): Calculated value based on site barometric pressure and temperature data taken on day of measurement.
(d): Calculated from HPGe measurements for fission and activation products (Table 3.1-23).
(f): Direct measurement of exposure rate using a Pressurized Ion Chamber (PIC).
(g): Difference between measured exposure rate (PIC) and calculated environmental exposure rate from HPGe measurements. Includes any instrument bias between PIC and HPGe (typically +/- 15%) and any direct radiation from plant structures and components.

FIGURE 3.1-1

VC BROADWAY

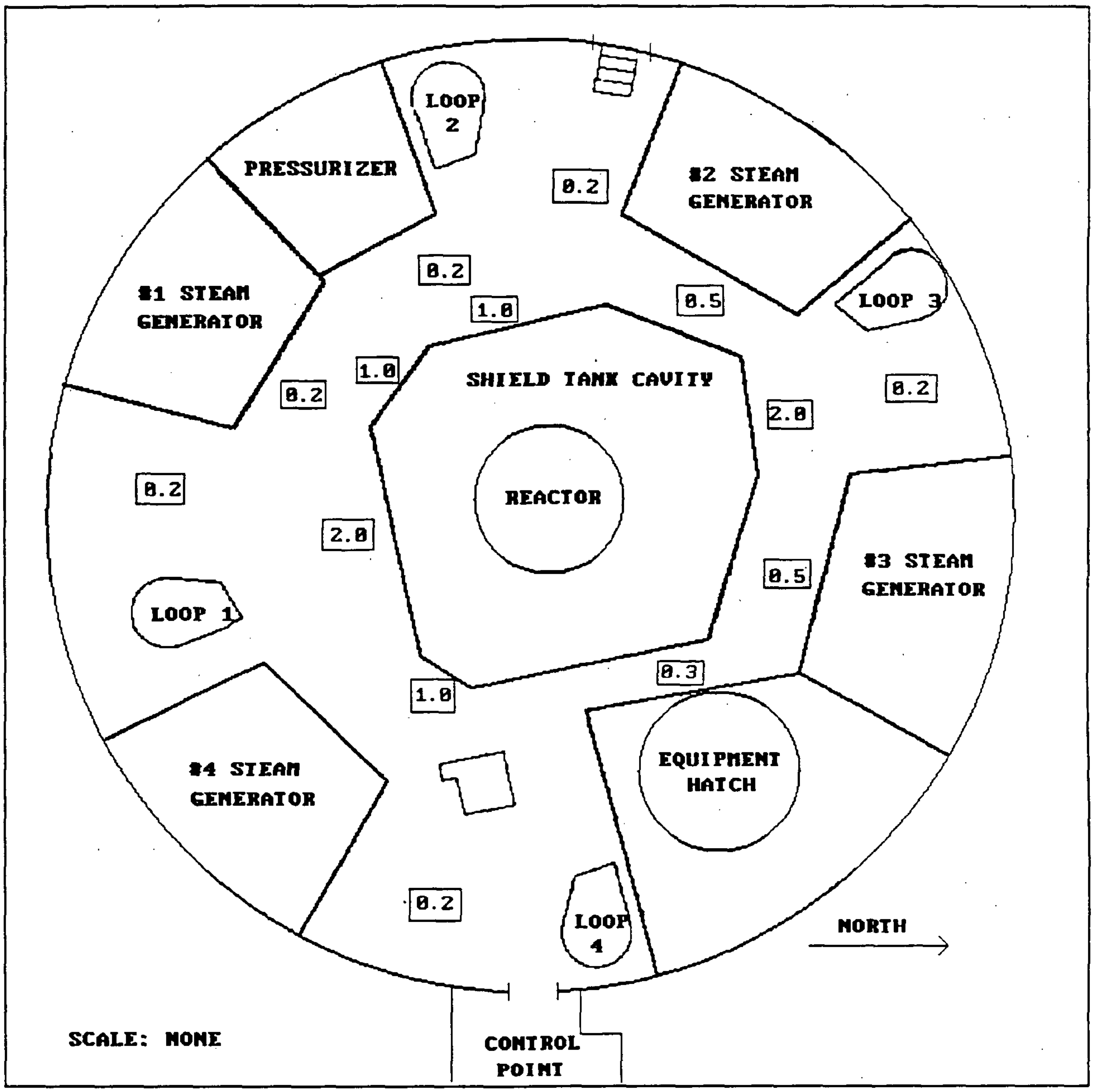


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 1K - 2K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-2

VC CHARGING FLOOR

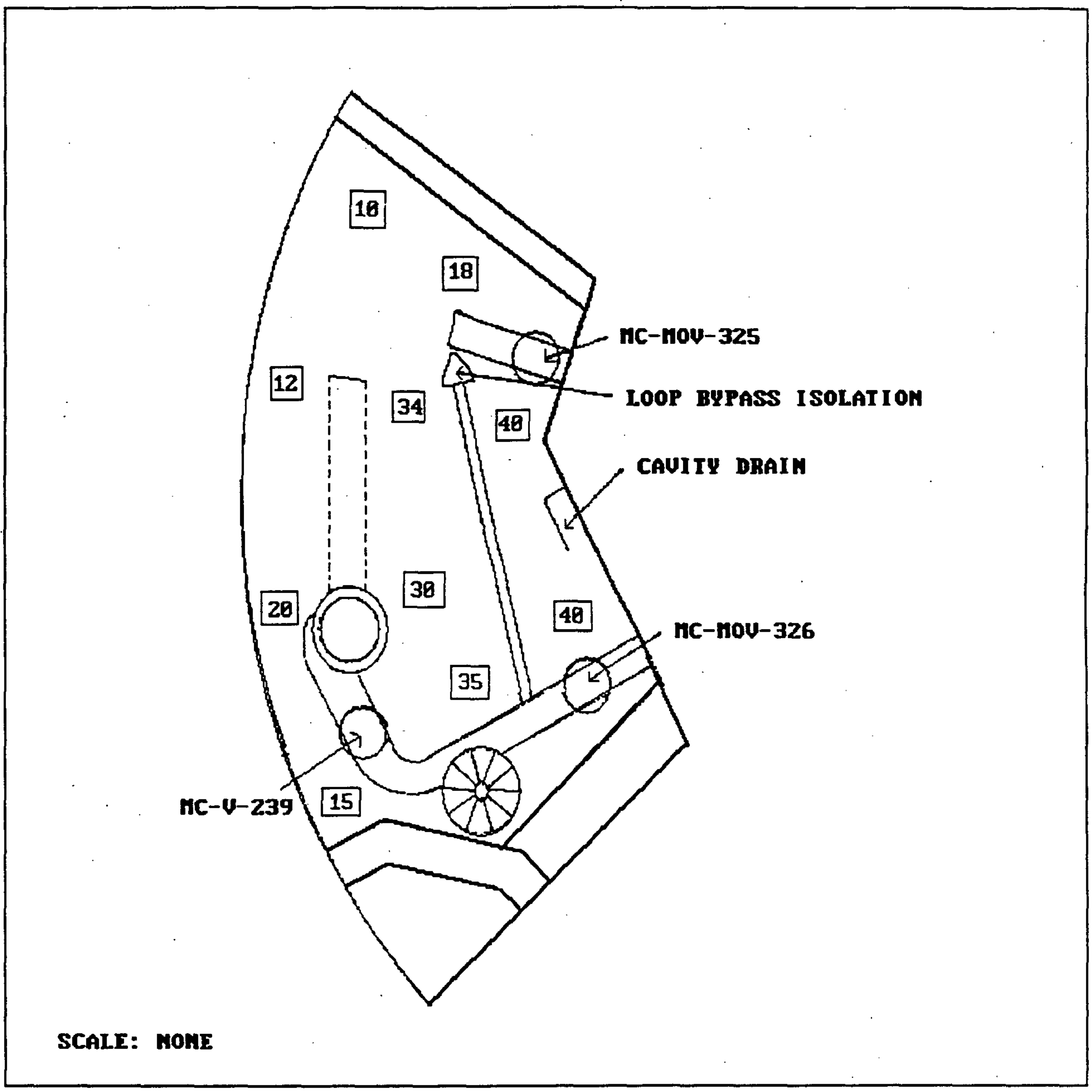


☐ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 1K - 2K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-3

VC LOOP 1

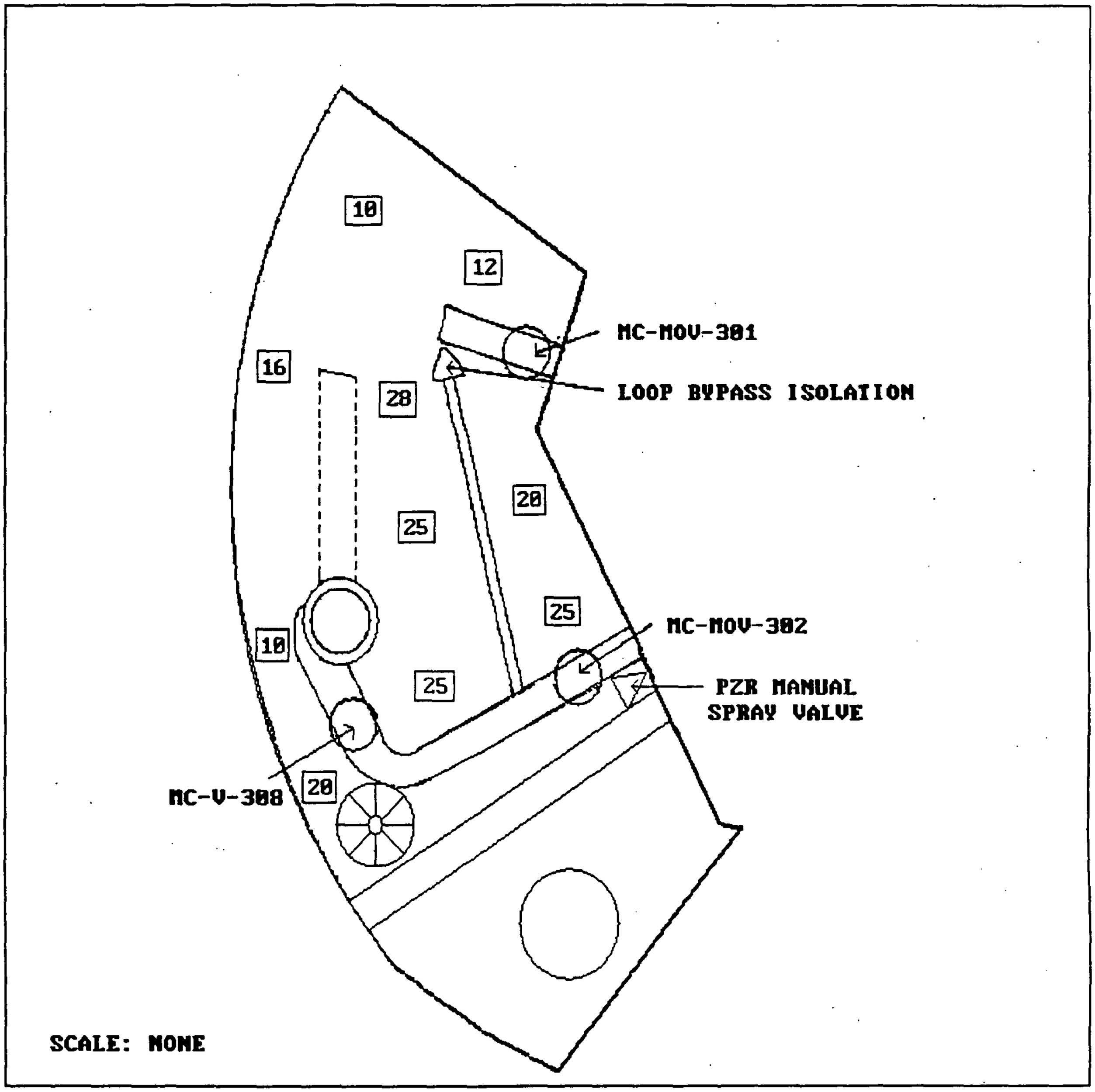


☐ = General Area Radiation Level (mR/h)
○ = Contact Radiation Level (mR/h)
Average Contamination Level = 2K - 10K dpm/100 cm^2

SURVEY DATA AS OF 11/93

FIGURE 3.1-4

VC LOOP 2

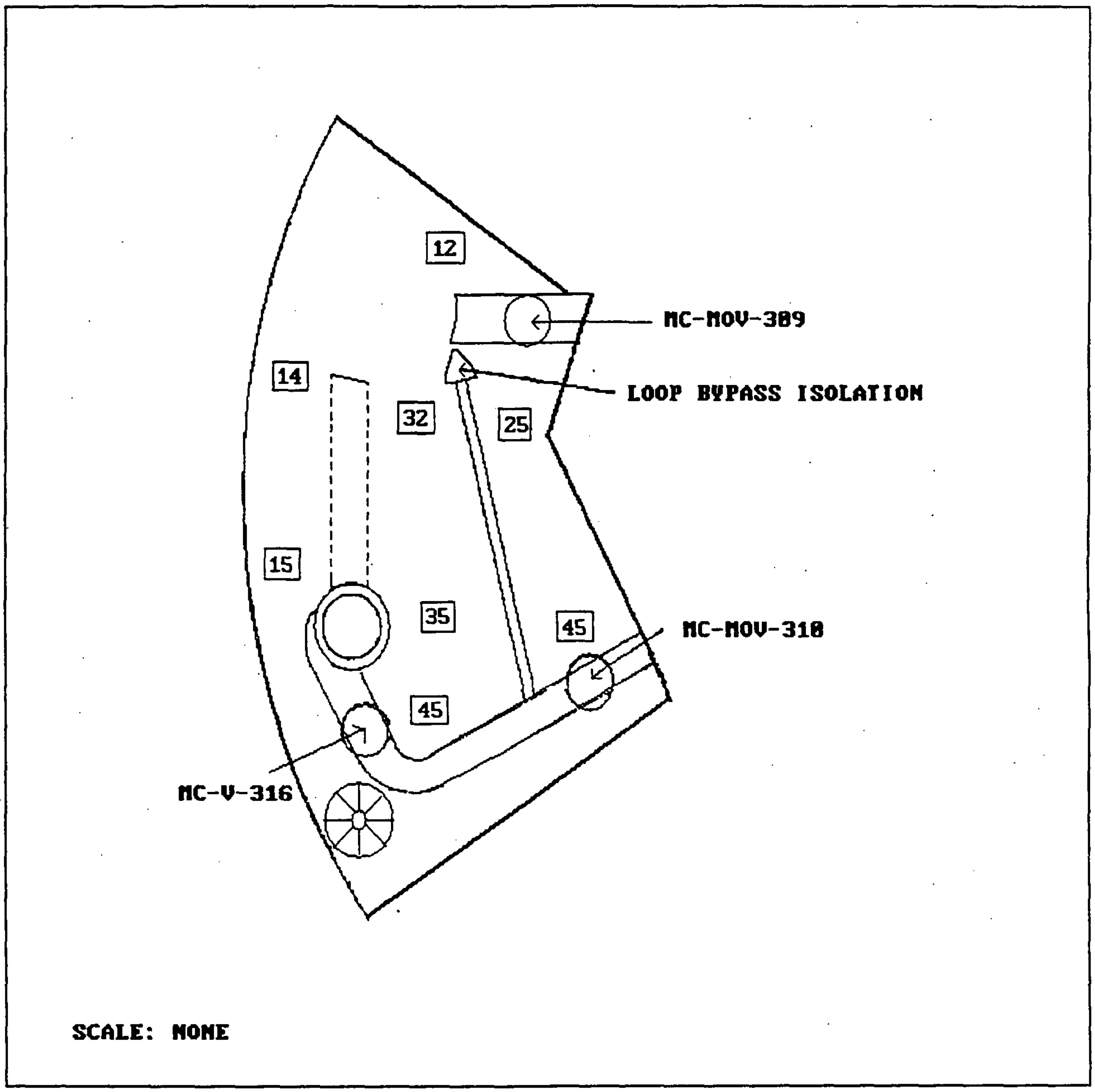


□ = General Area Radiation Level (mR/h)
○ = Contact Radiation Level (mR/h)
Average Contamination Level = 2K - 10K dpm/100 cm^2

SURVEY DATA AS OF 11/93

FIGURE 3.1-5

VC LOOP 3

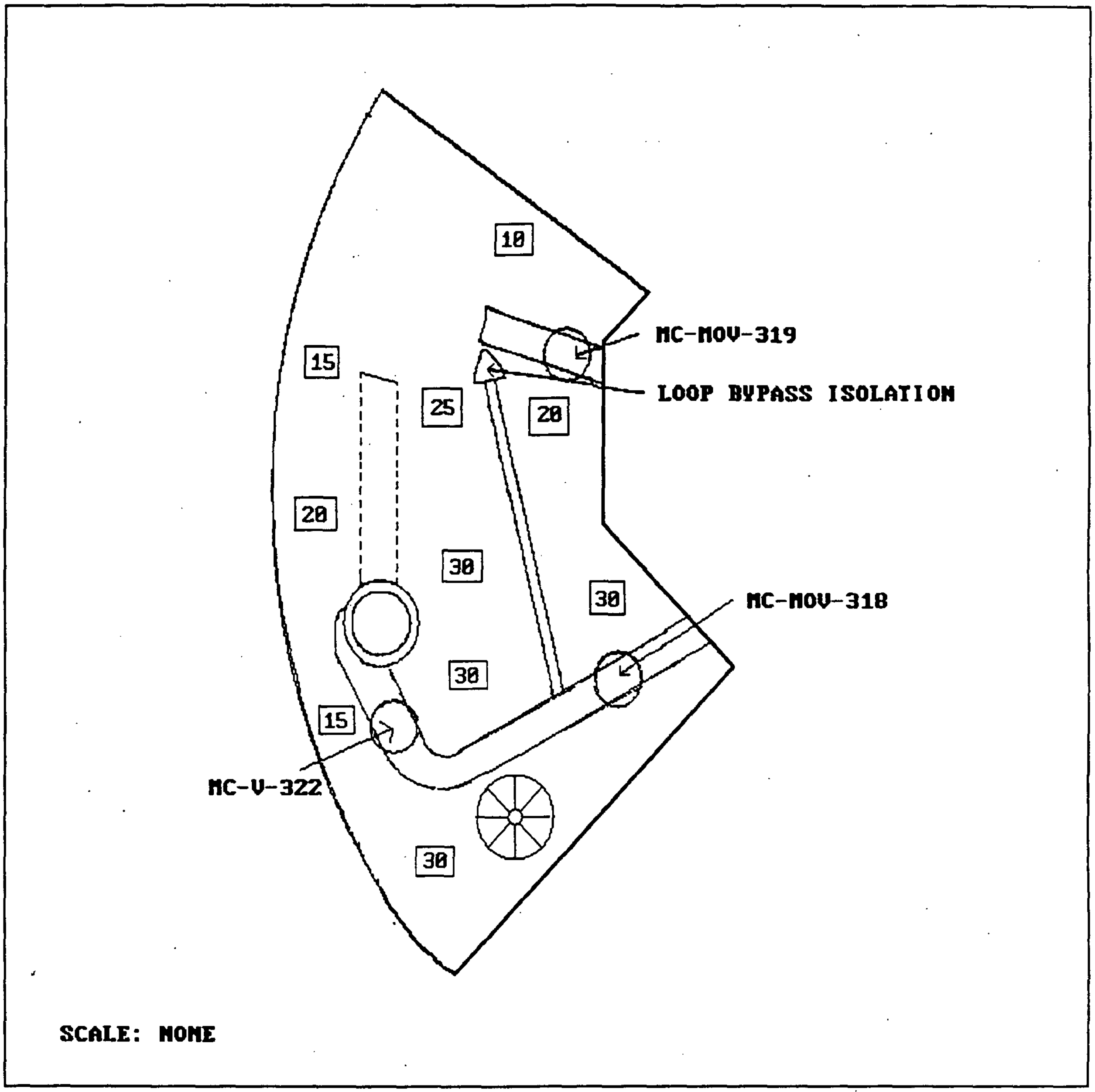


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 2K - 10K dpm/100 cm^2

SURVEY DATA AS OF 11/93

FIGURE 3.1-6

VC LOOP 4

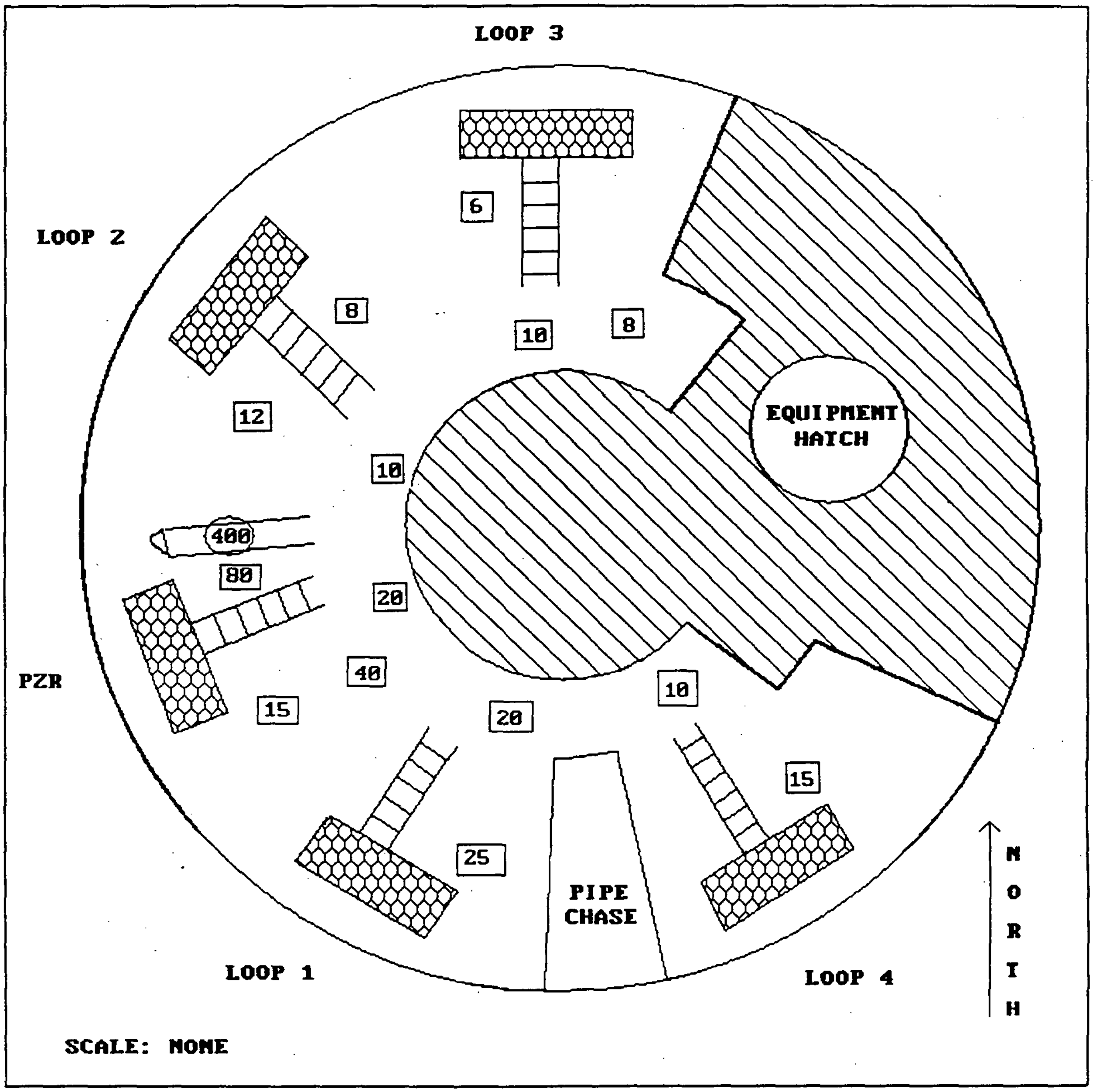


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 2K - 10K dpm/100 cm^2

SURVEY DATA AS OF 11/93

FIGURE 3.1-7

VC BRASS DRAIN BOX

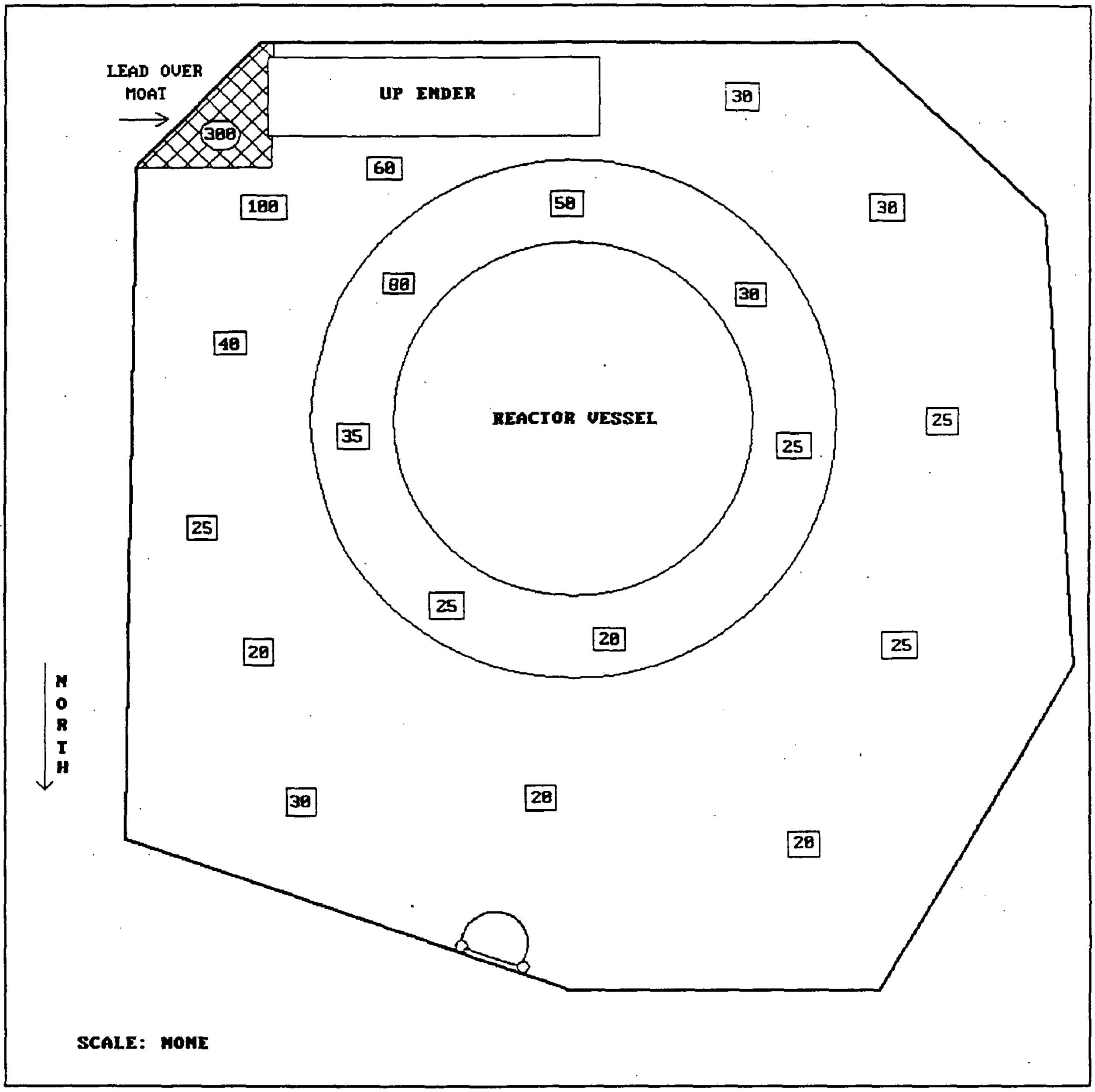


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 5K - 15K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-8

VC SHIELD TANK CAVITY

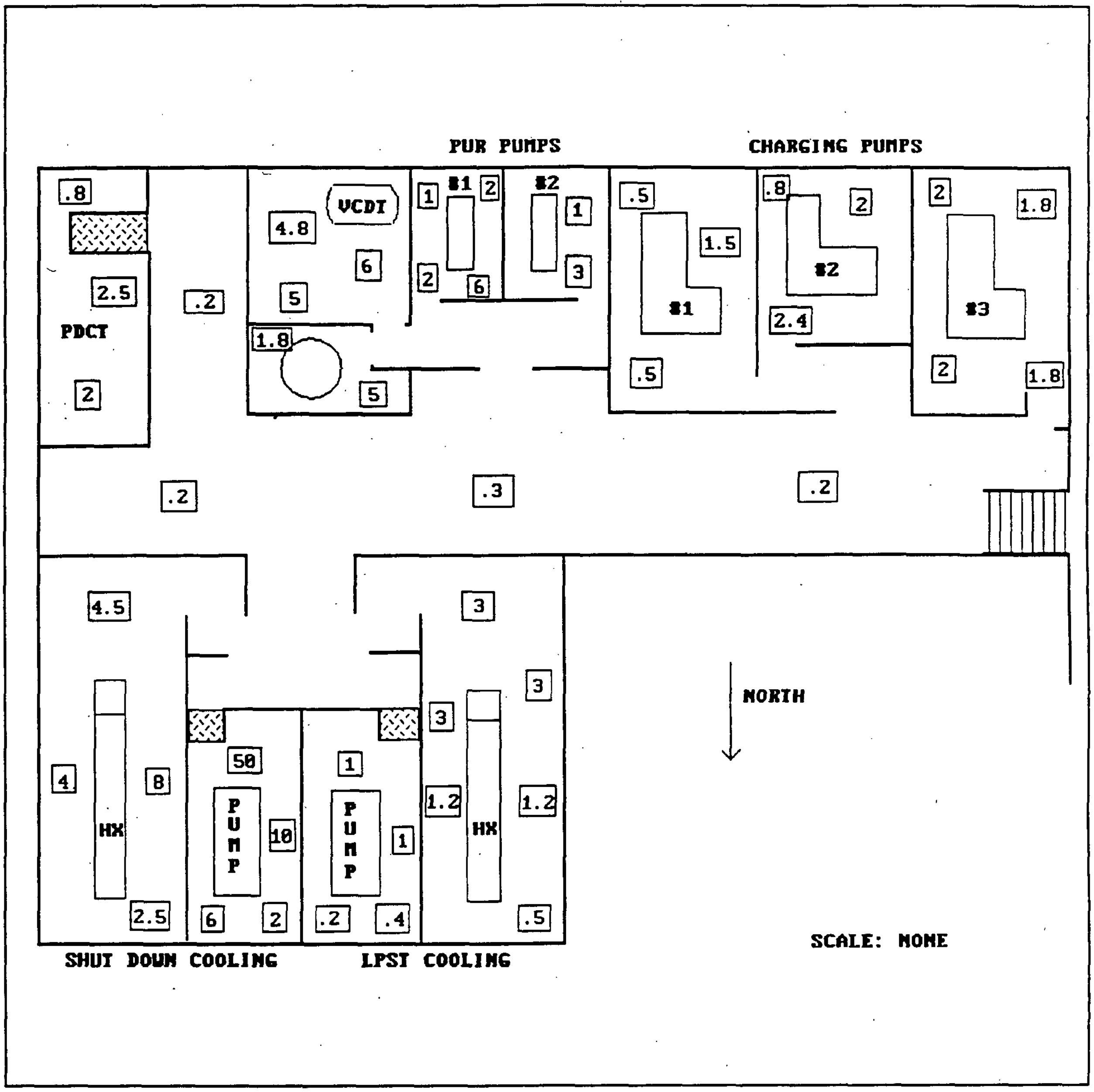


☐ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 5K - 15K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-9

PAB CUBICLE CORRIDOR

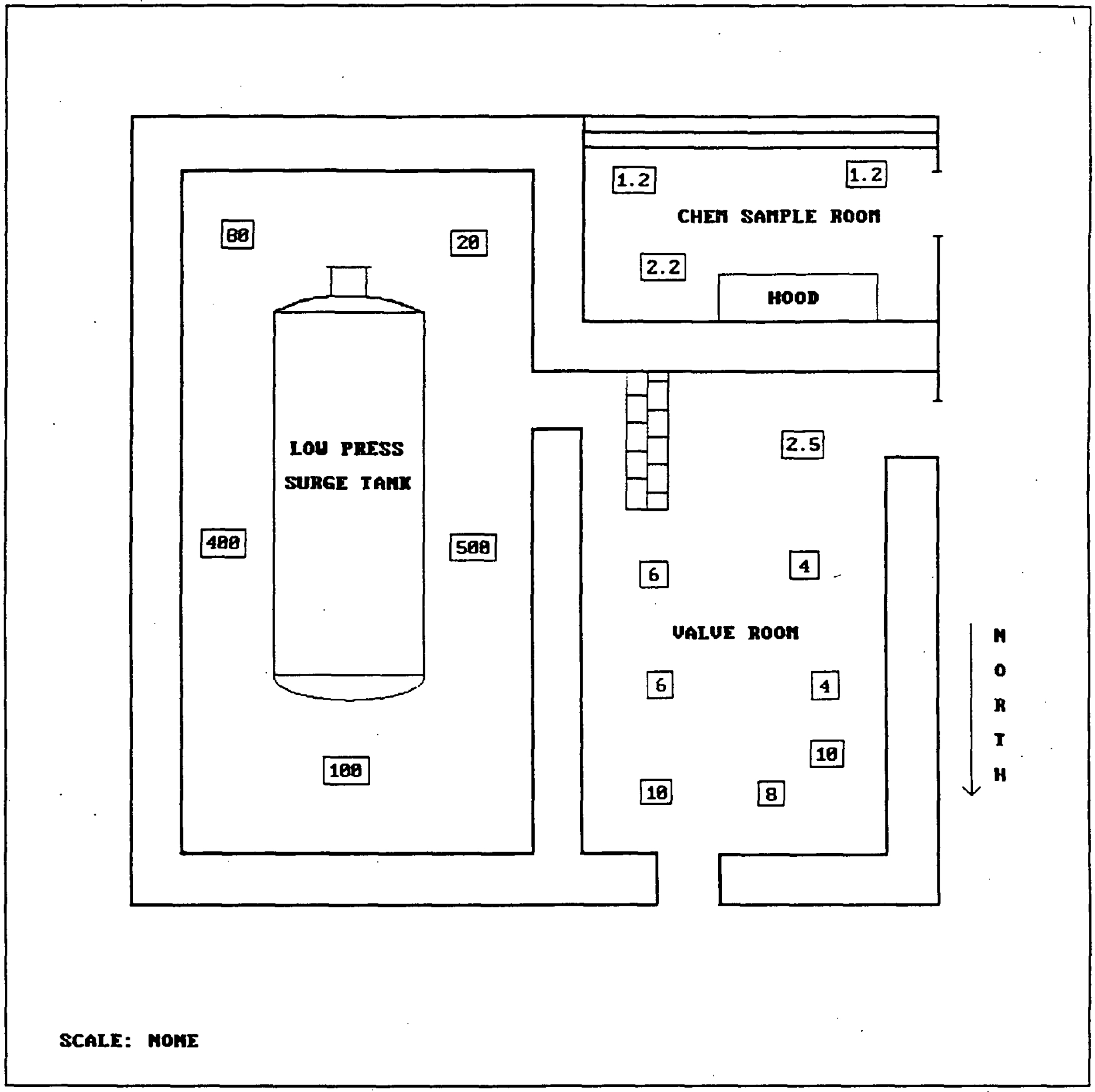


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 2K - 10K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-10

PAB VALVE ROOM, LPST, & SAMPLE ROOMS

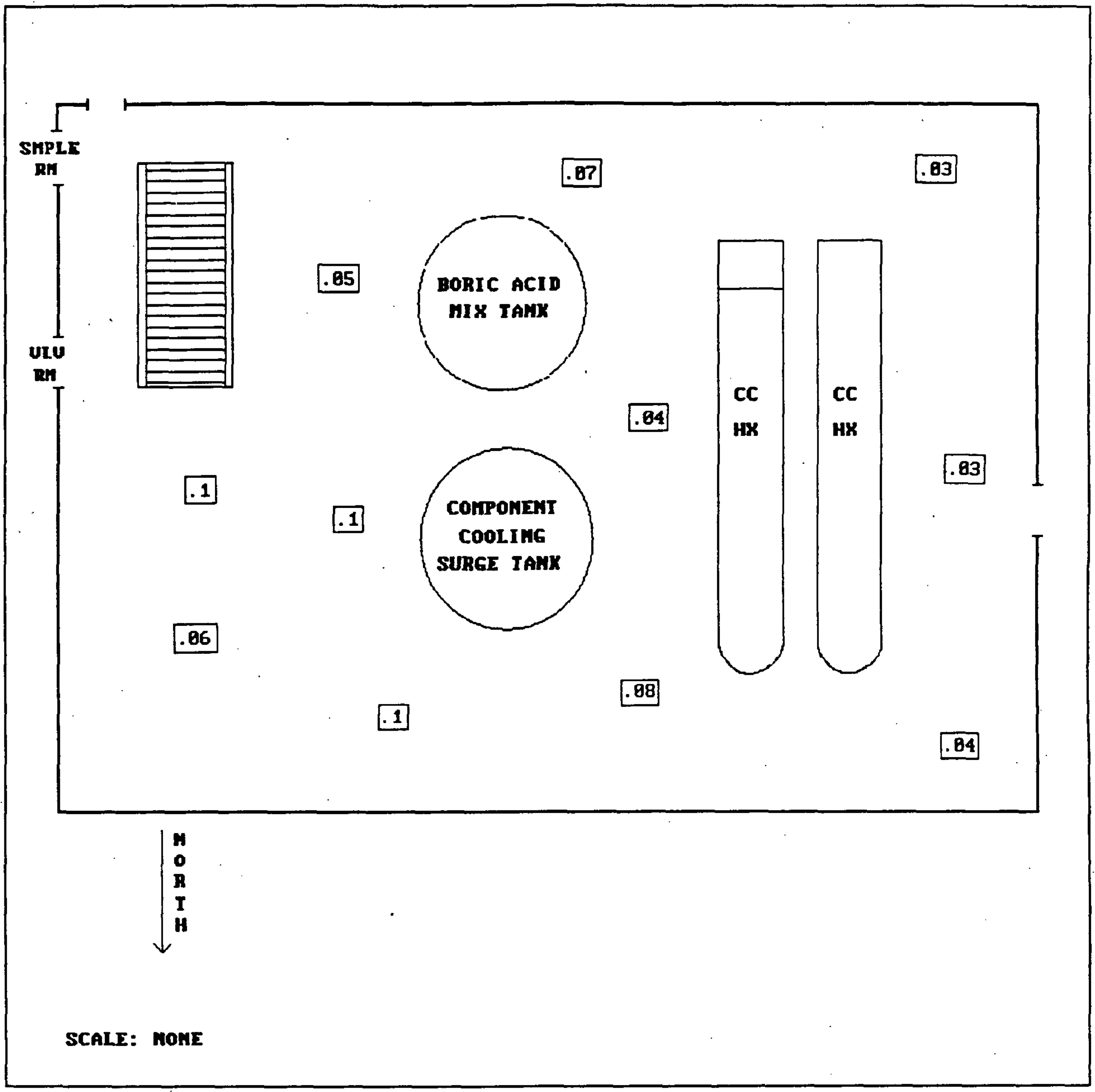


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 1K - 5K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-11

UPPER LEVEL PAB

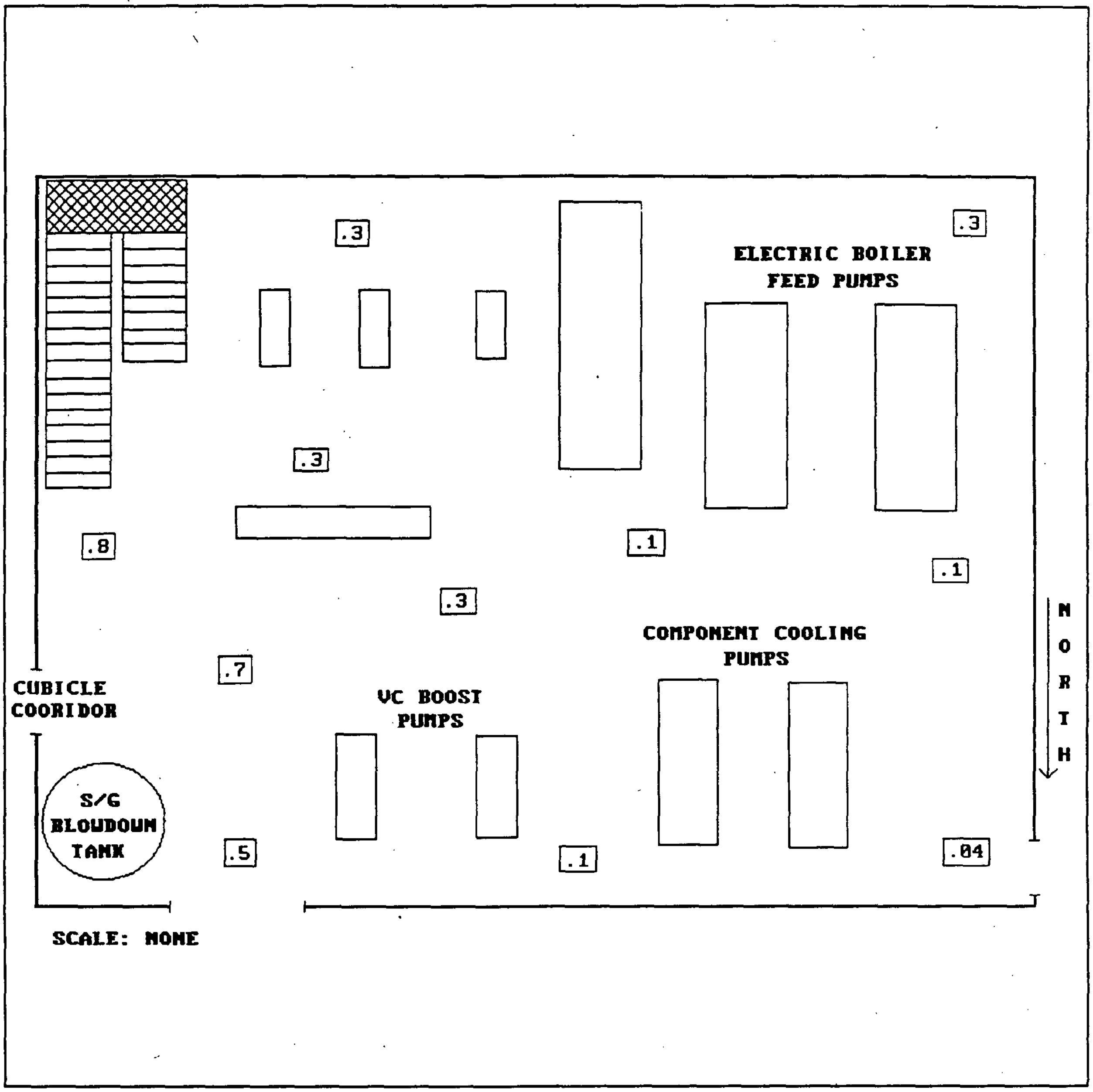


□ = General Area Radiation Level (mR/h)
○ = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-12

LOWER LEVEL PAB

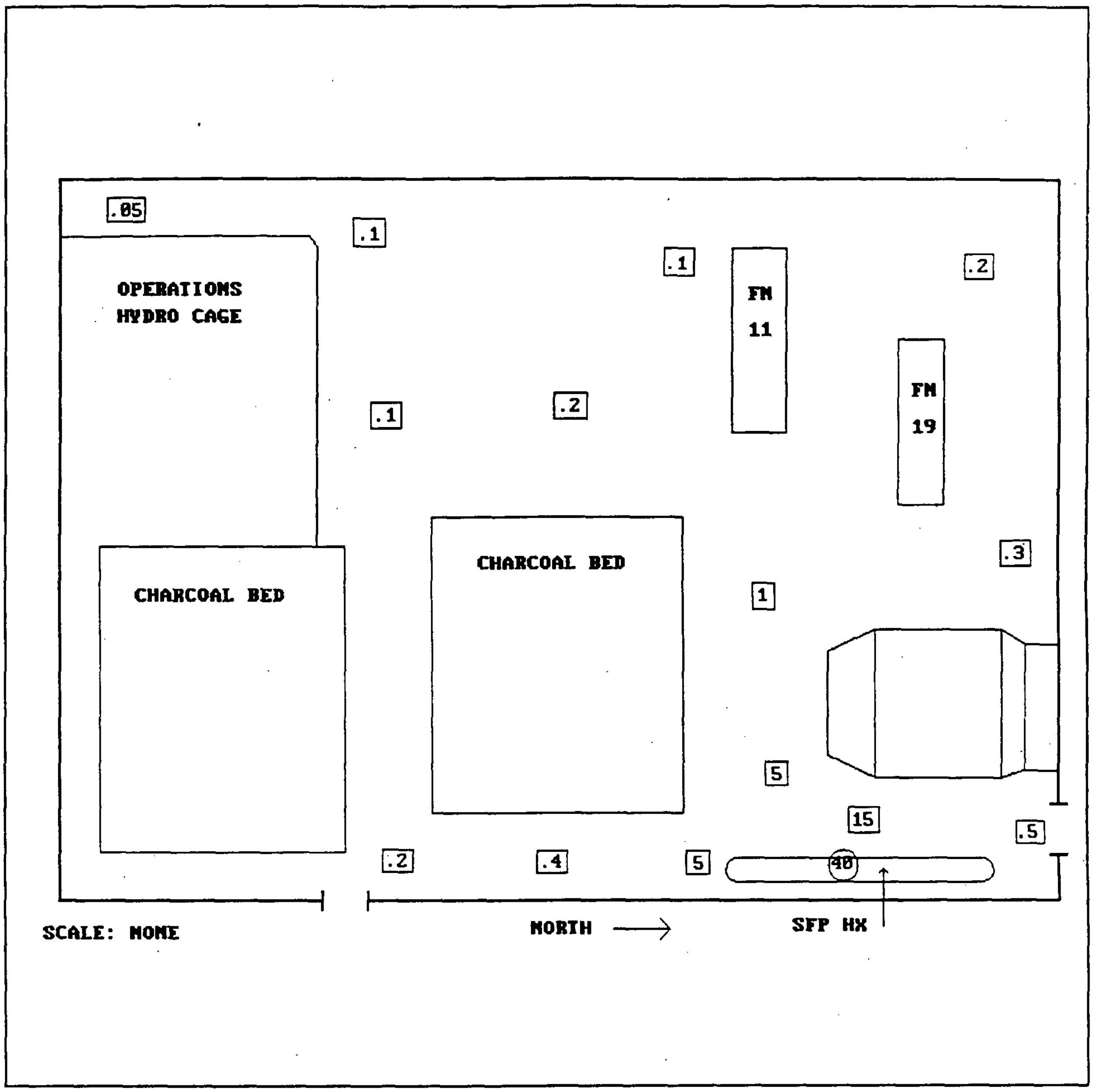


☐ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-13

MECHANICAL EQUIPMENT ROOM

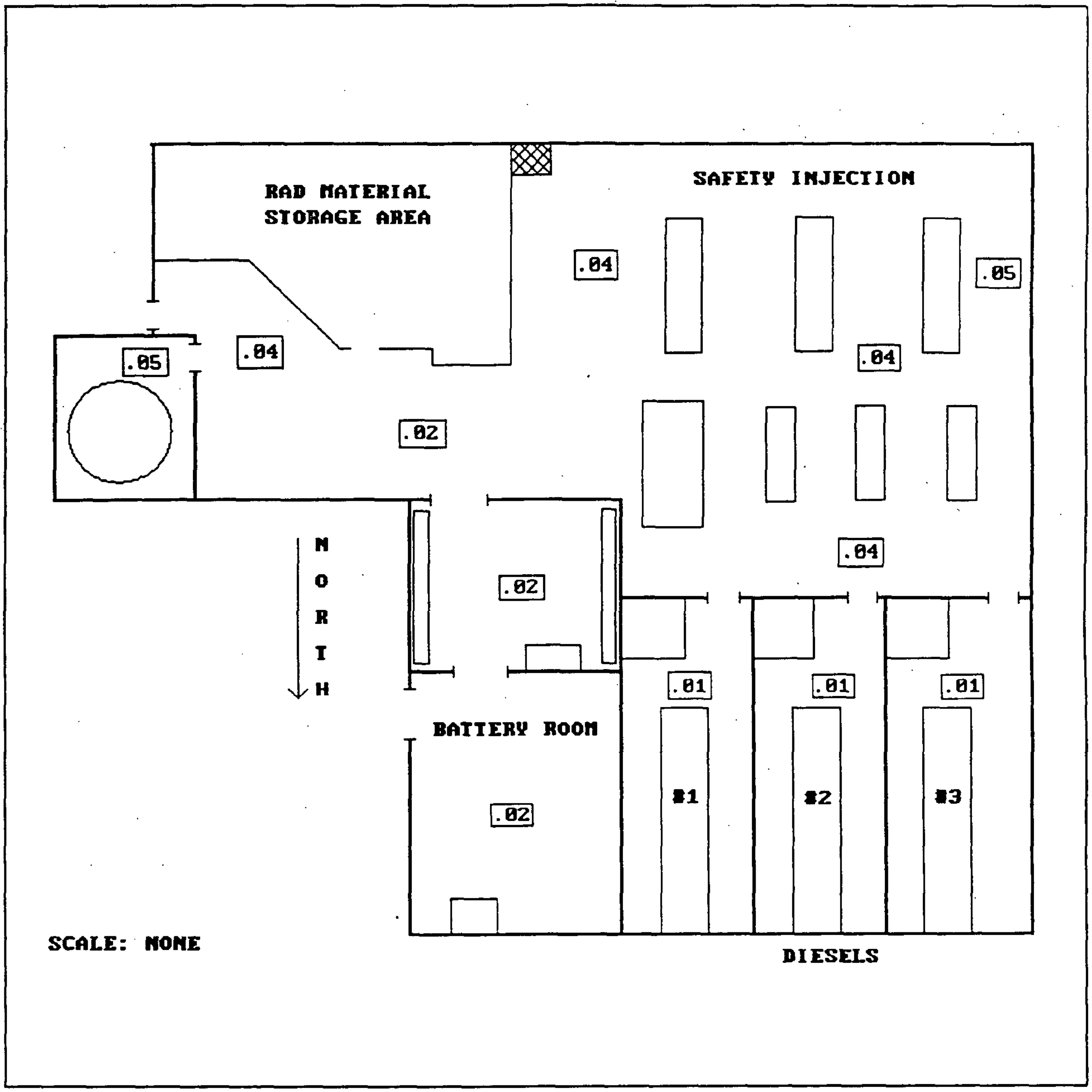


☐ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-14

DIESEL GENERATOR BUILDING

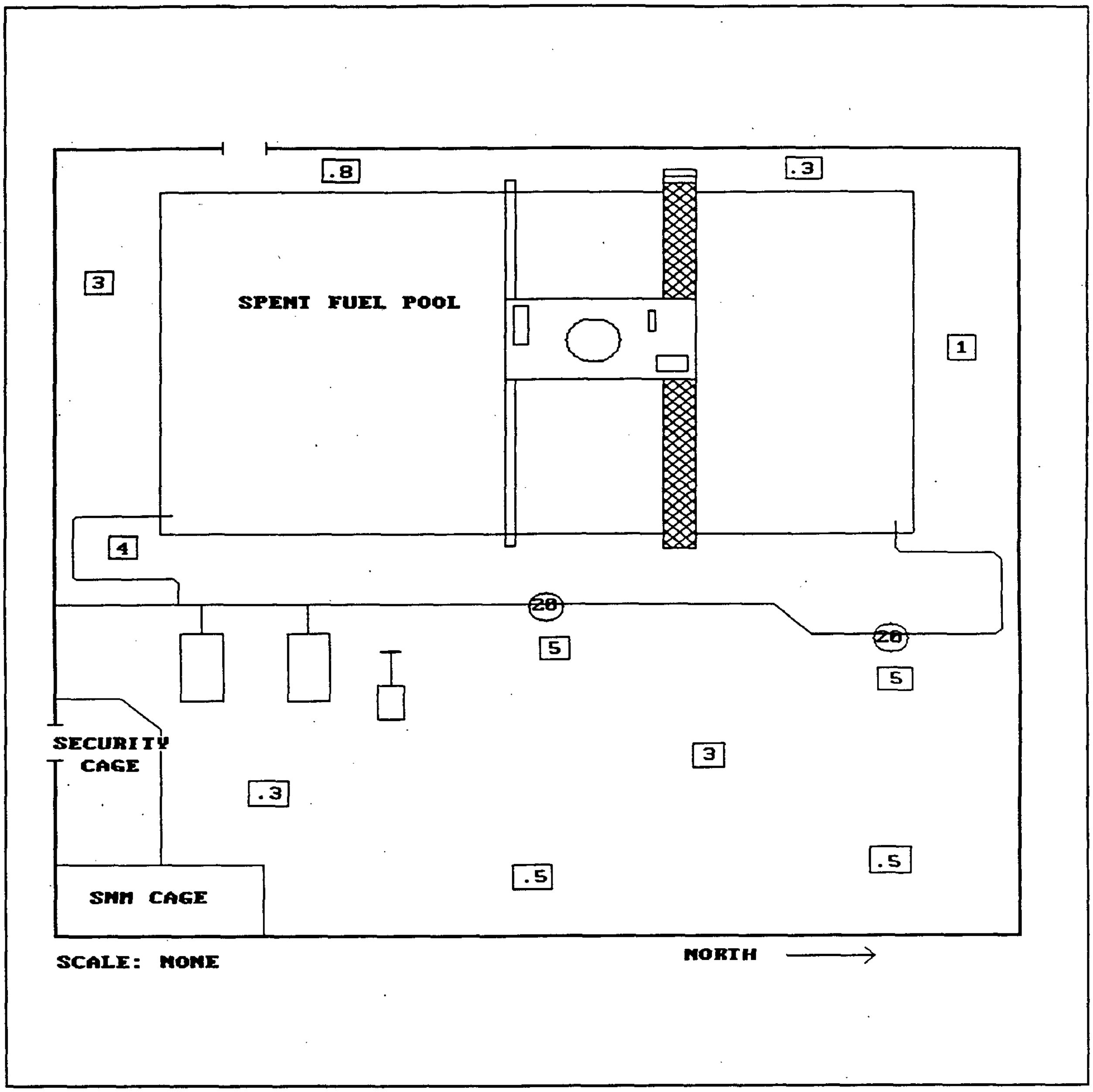


☐ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-15

SPENT FUEL PIT

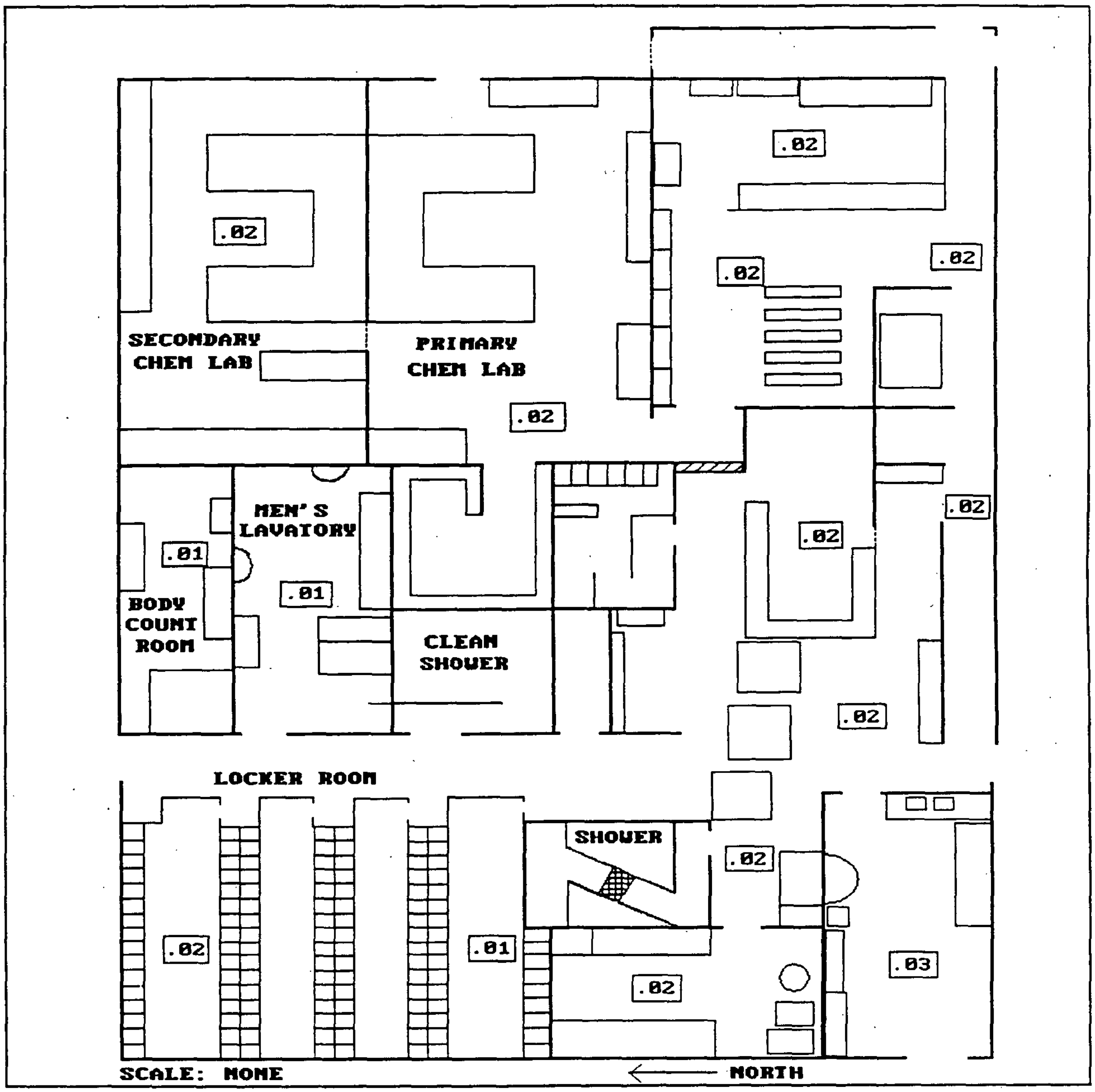


☐ = General Area Radiation Level (mR/h)
○ = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-16

CONTROL POINT AND PRIMARY CHEMISTRY

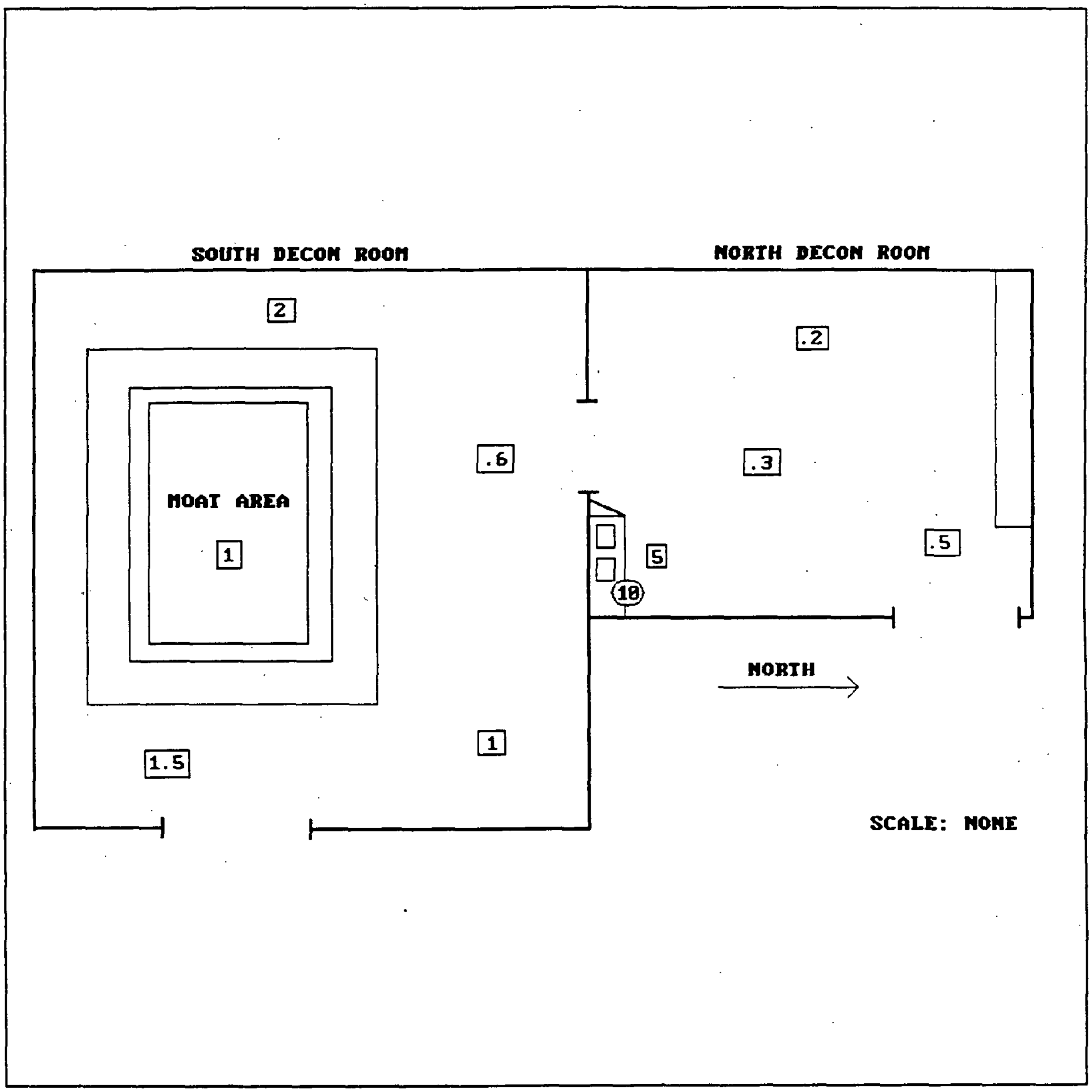


☐ = General Area Radiation Level (mR/h)
○ = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-17

NORTH AND SOUTH DECON ROOMS

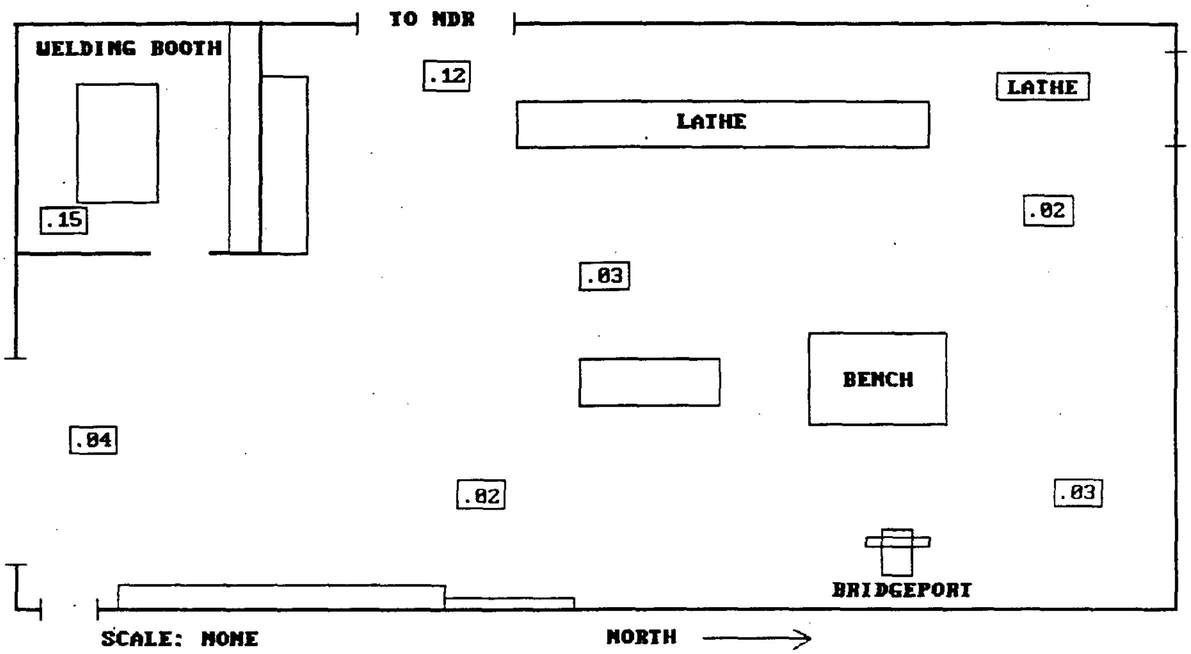


☐ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 2K - 5K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-18

RCA MACHINE SHOP

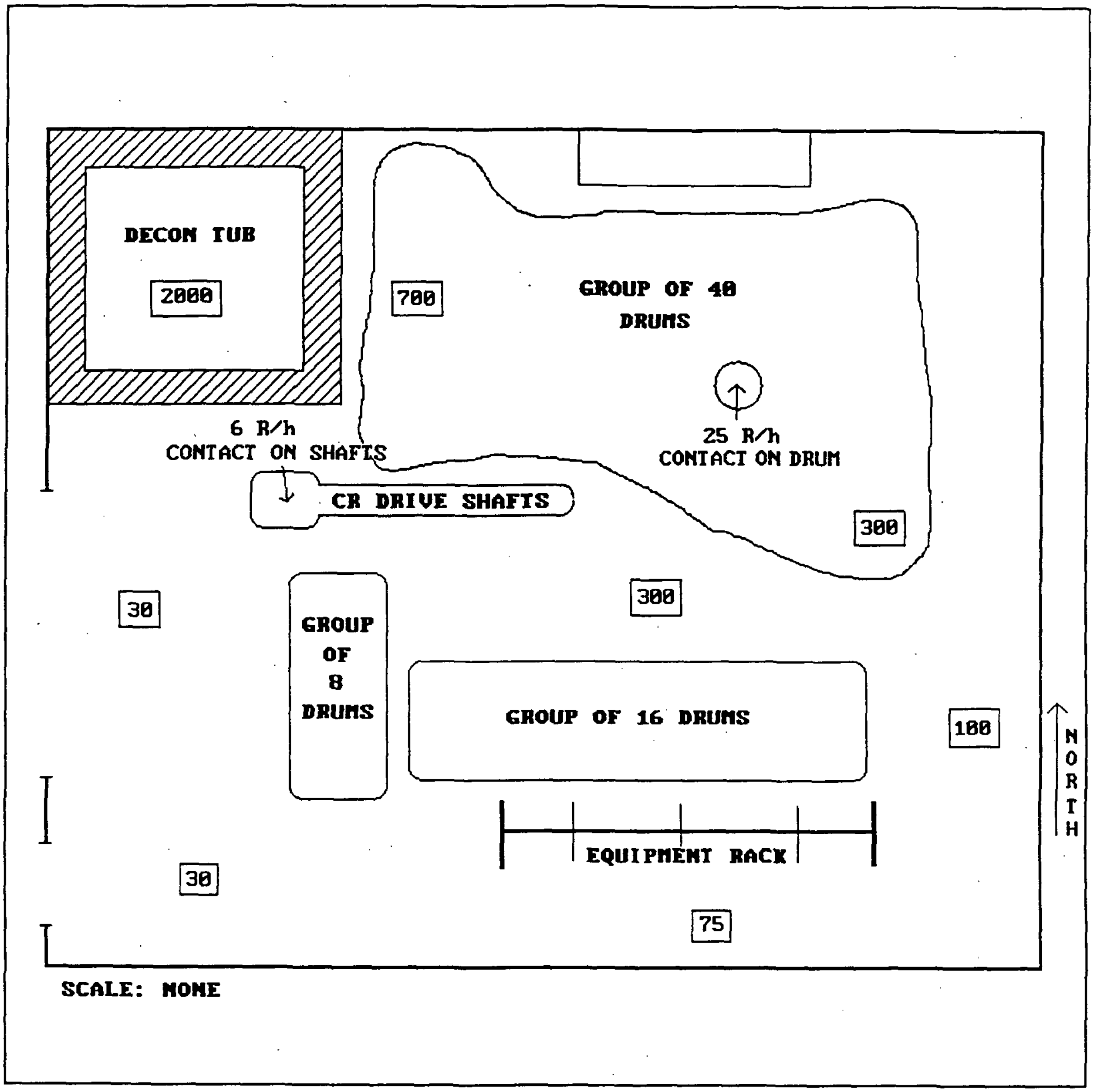


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = <1K $dpm/100\ cm^2$

SURVEY DATA AS OF 6/93

FIGURE 3.1-19

PCA STORAGE BUILDING 1

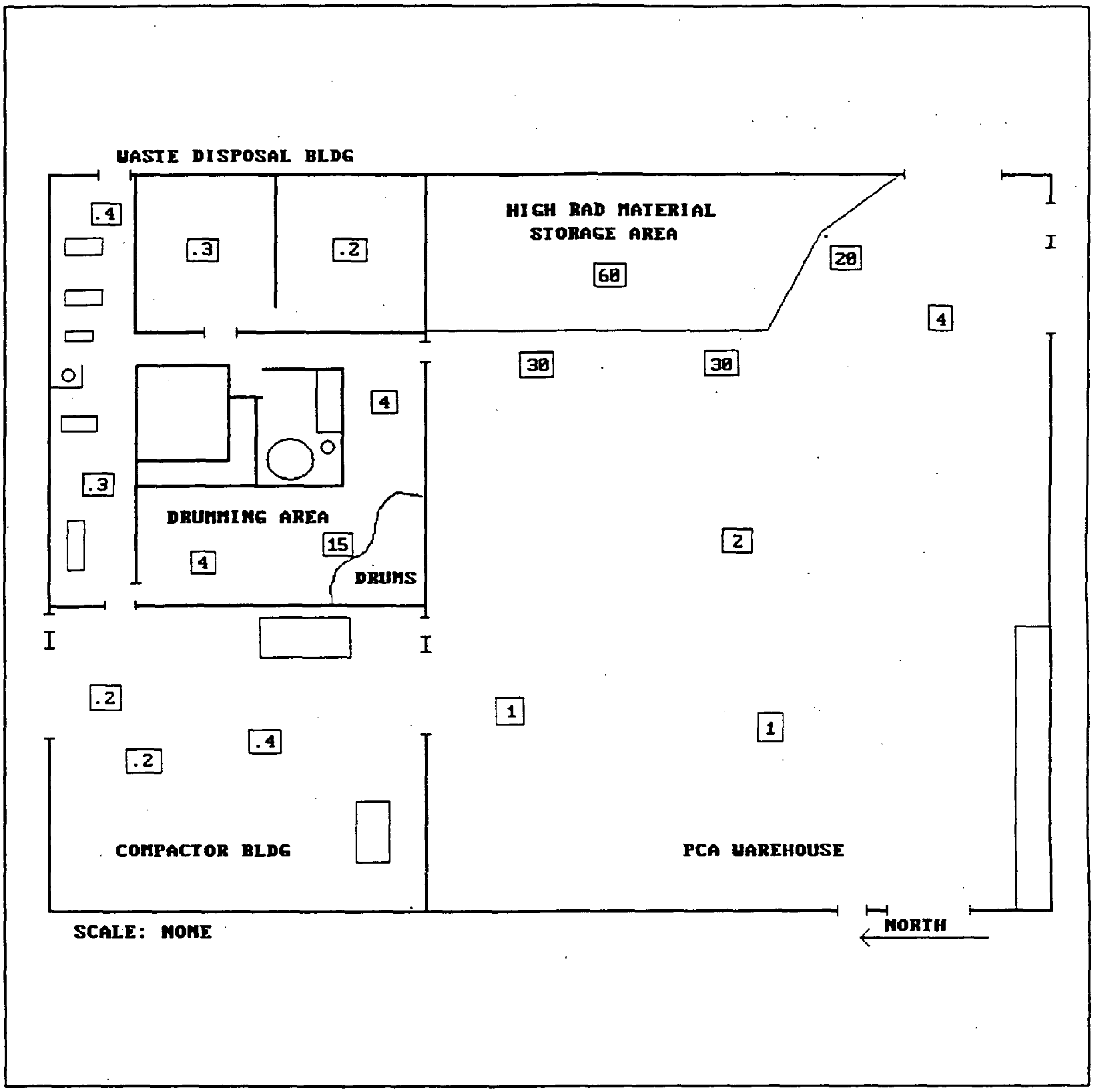


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = 2K - 5K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-20

WASTE DISPOSAL, COMPACTOR & PCA WAREHOUSE BUILDINGS

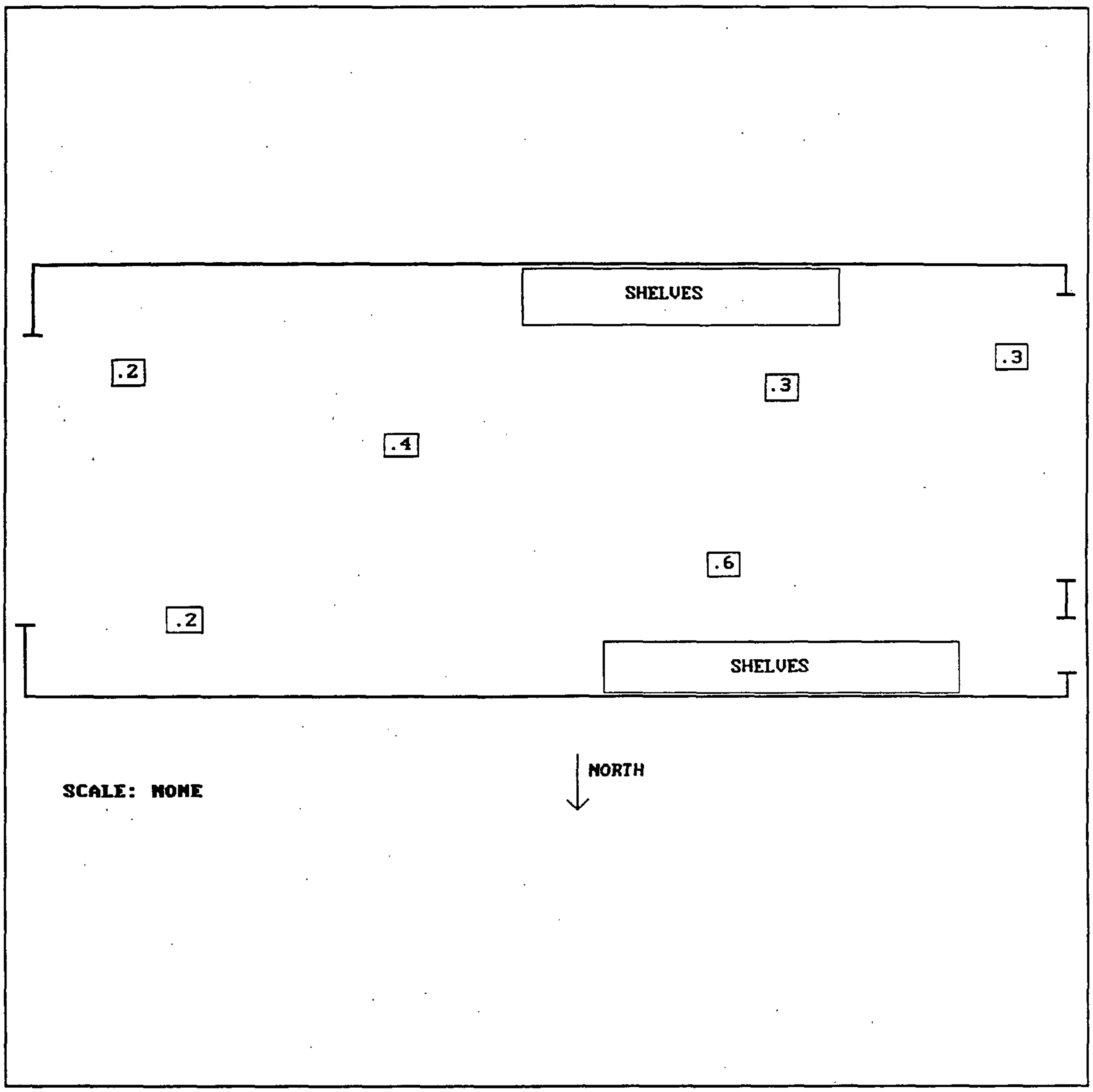


☐ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-21

PCA STORAGE BUILDING 2

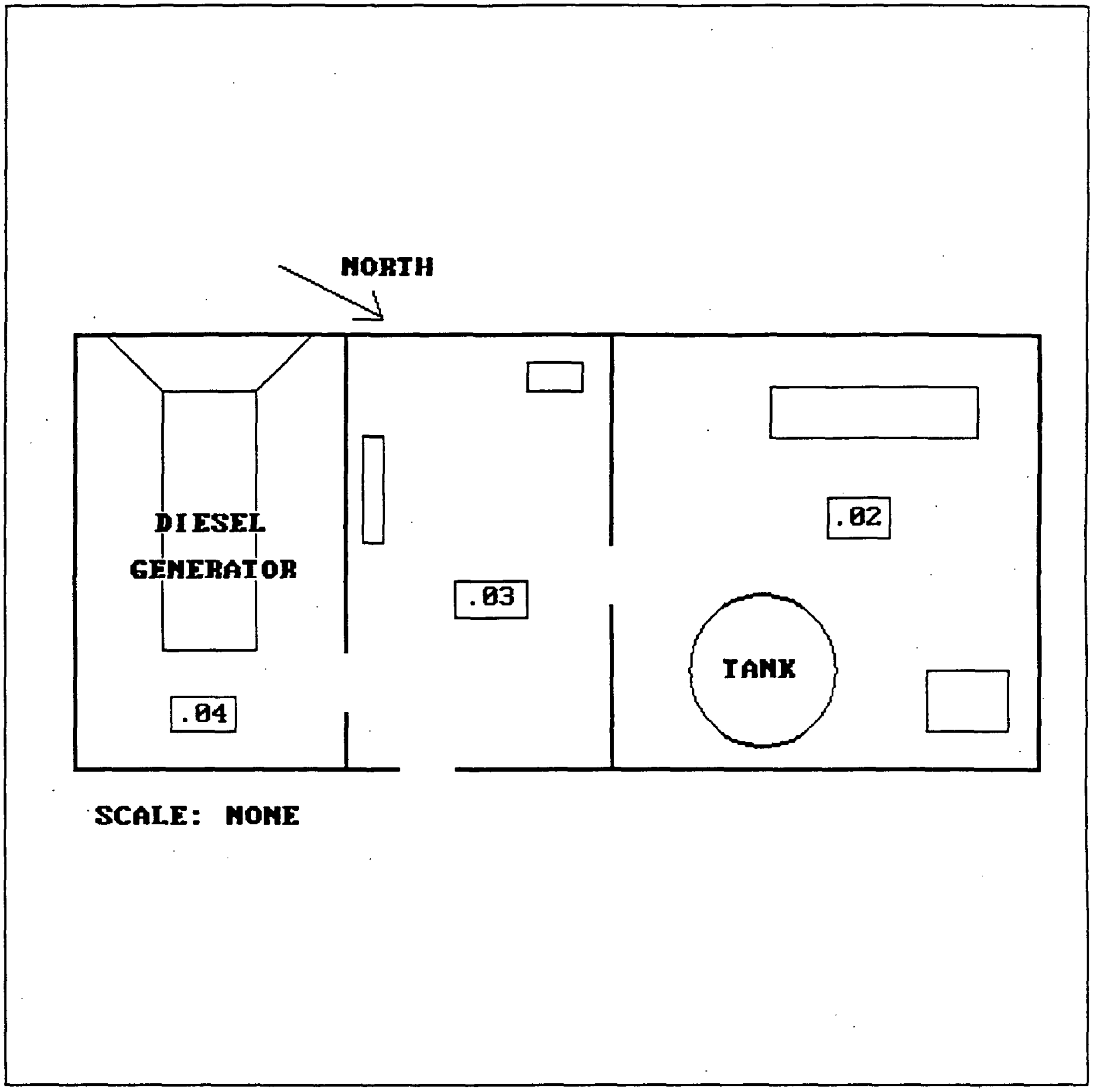


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-22

SAFE SHUTDOWN BUILDING

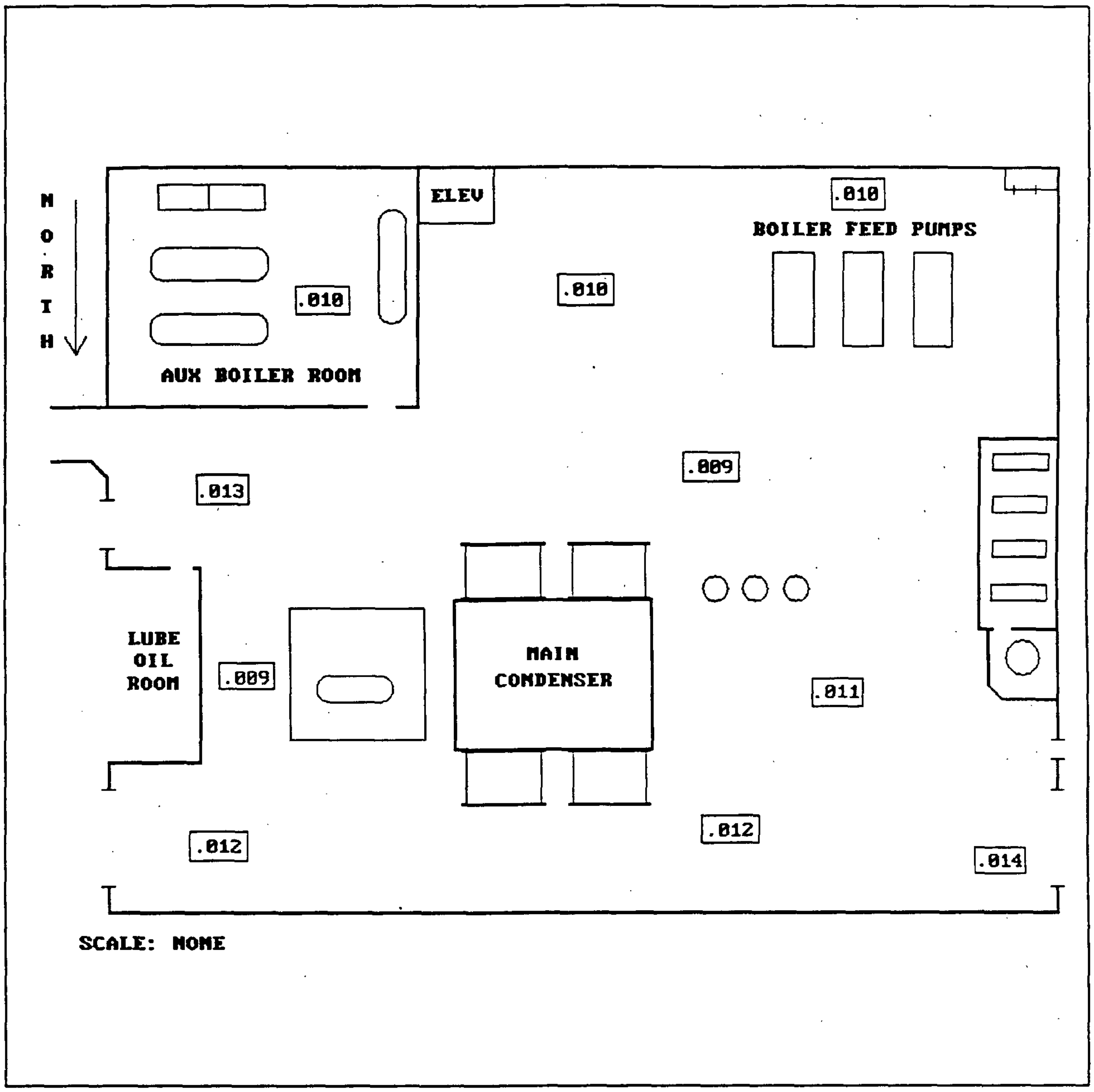


□ = General Area Radiation Level (mR/h)
O = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-23

TURBINE BUILDING - GROUND FLOOR

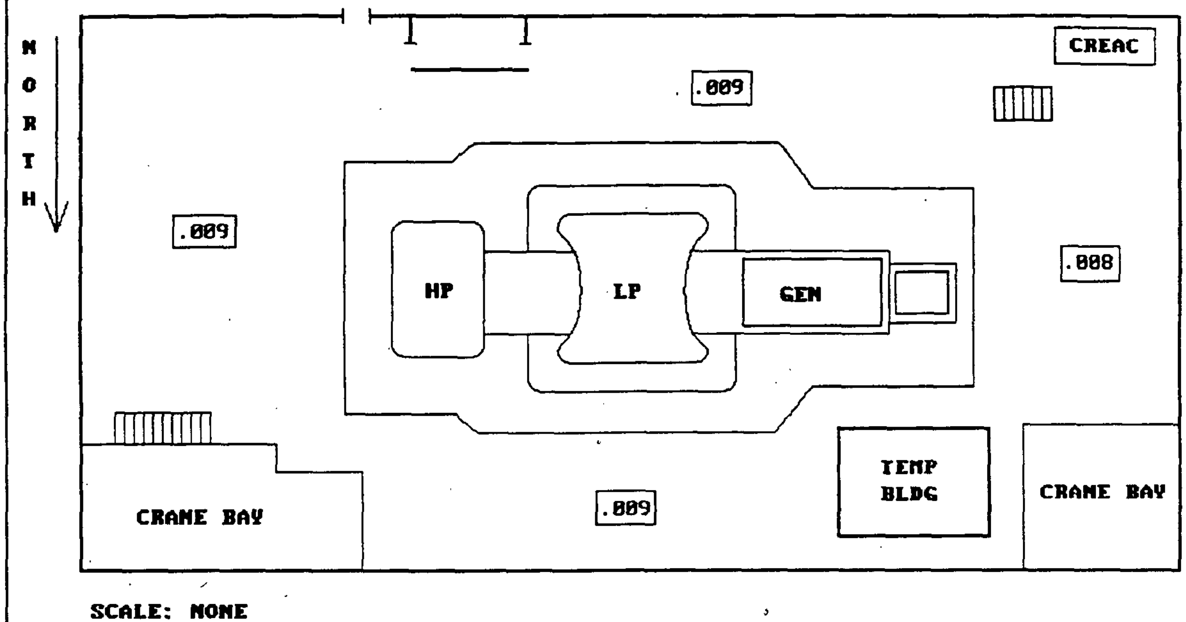


□ = General Area Radiation Level (mR/h)
○ = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-24

TURBINE BUILDING - TURBINE FLOOR

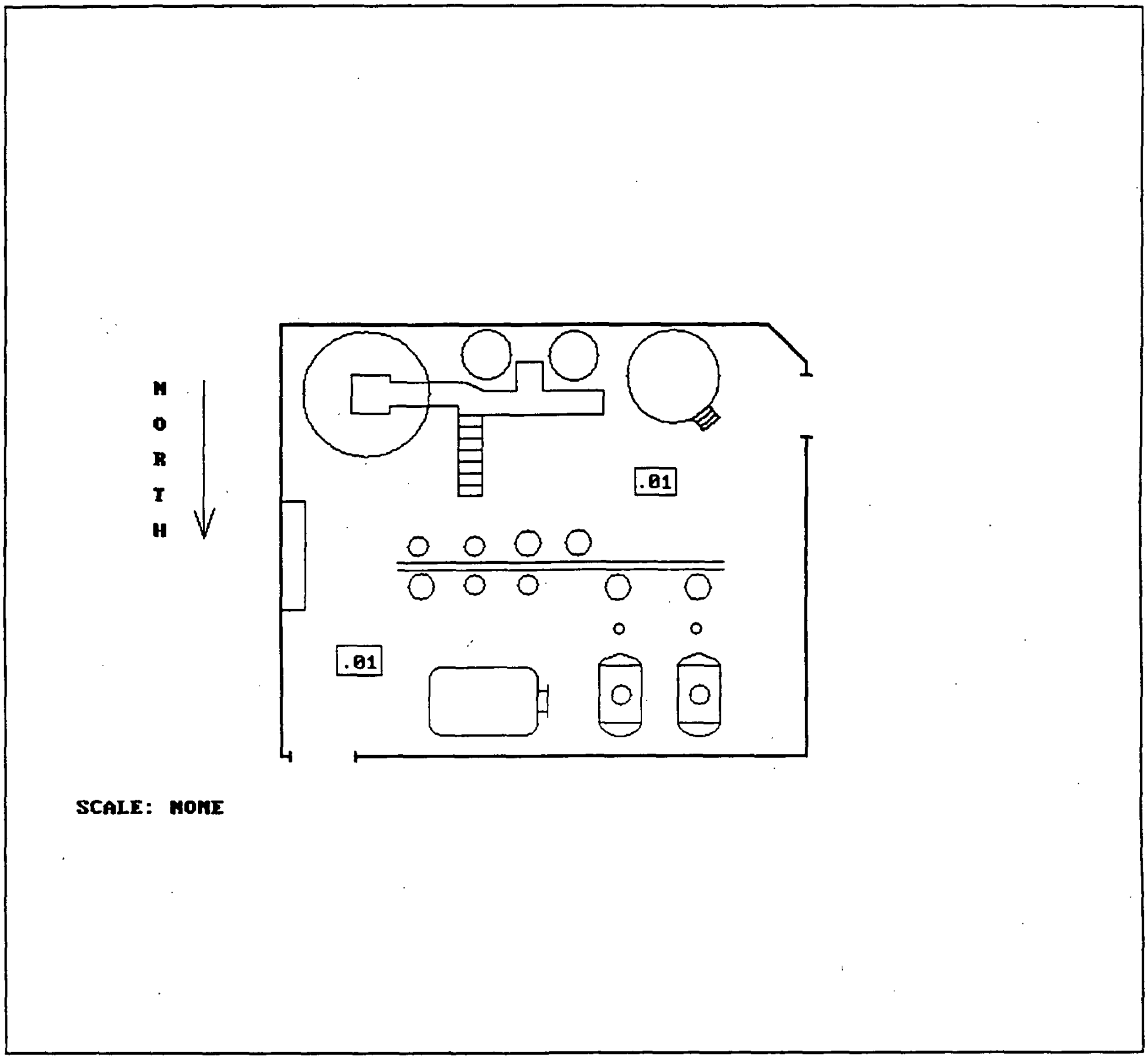


□ = General Area Radiation Level (mR/h)
○ = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

FIGURE 3.1-25

WATER TREATMENT



☐ = General Area Radiation Level (mR/h)
○ = Contact Radiation Level (mR/h)
Average Contamination Level = <1K dpm/100 cm^2

SURVEY DATA AS OF 6/93

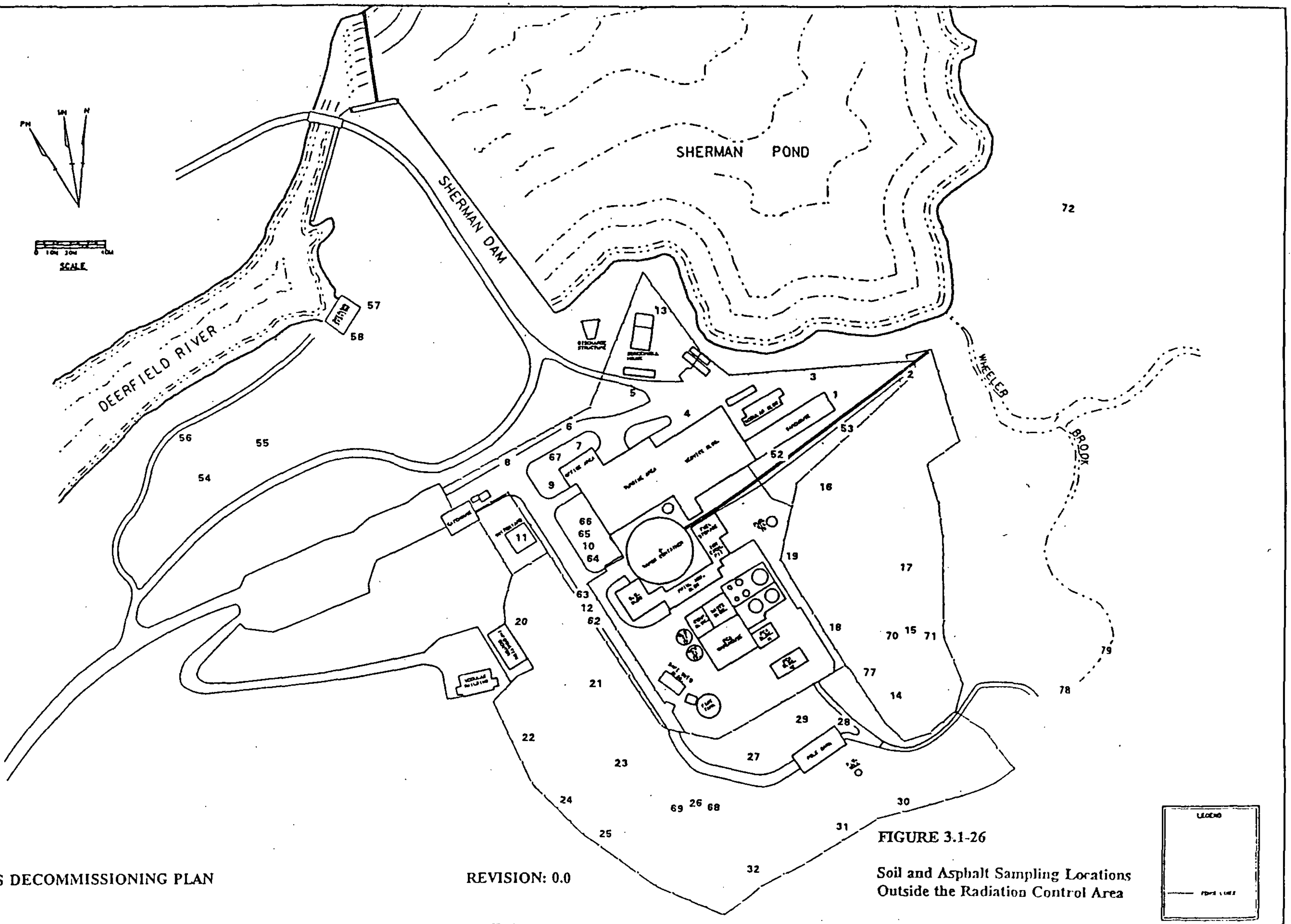


FIGURE 3.1-26

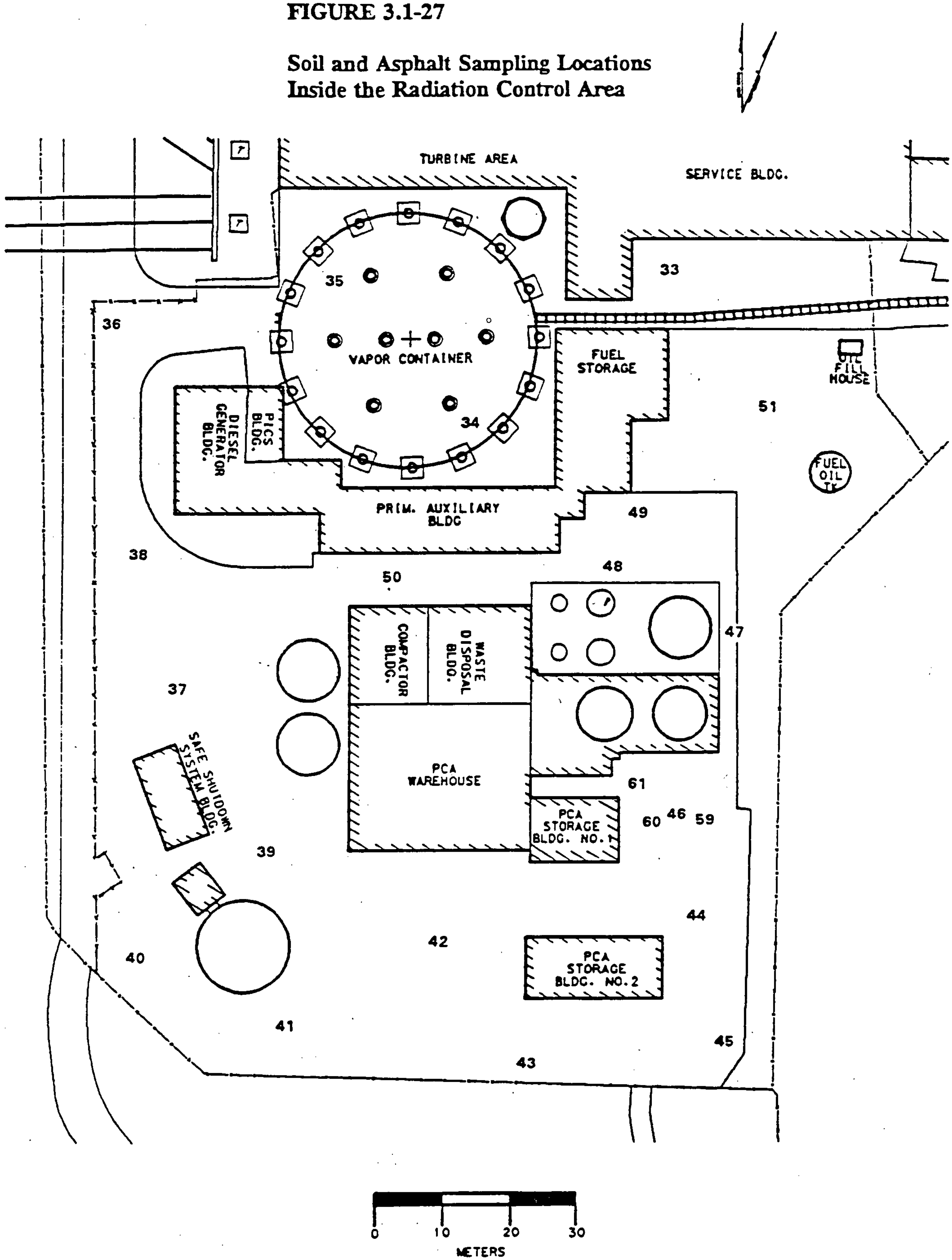
Soil and Asphalt Sampling Locations Outside the Radiation Control Area

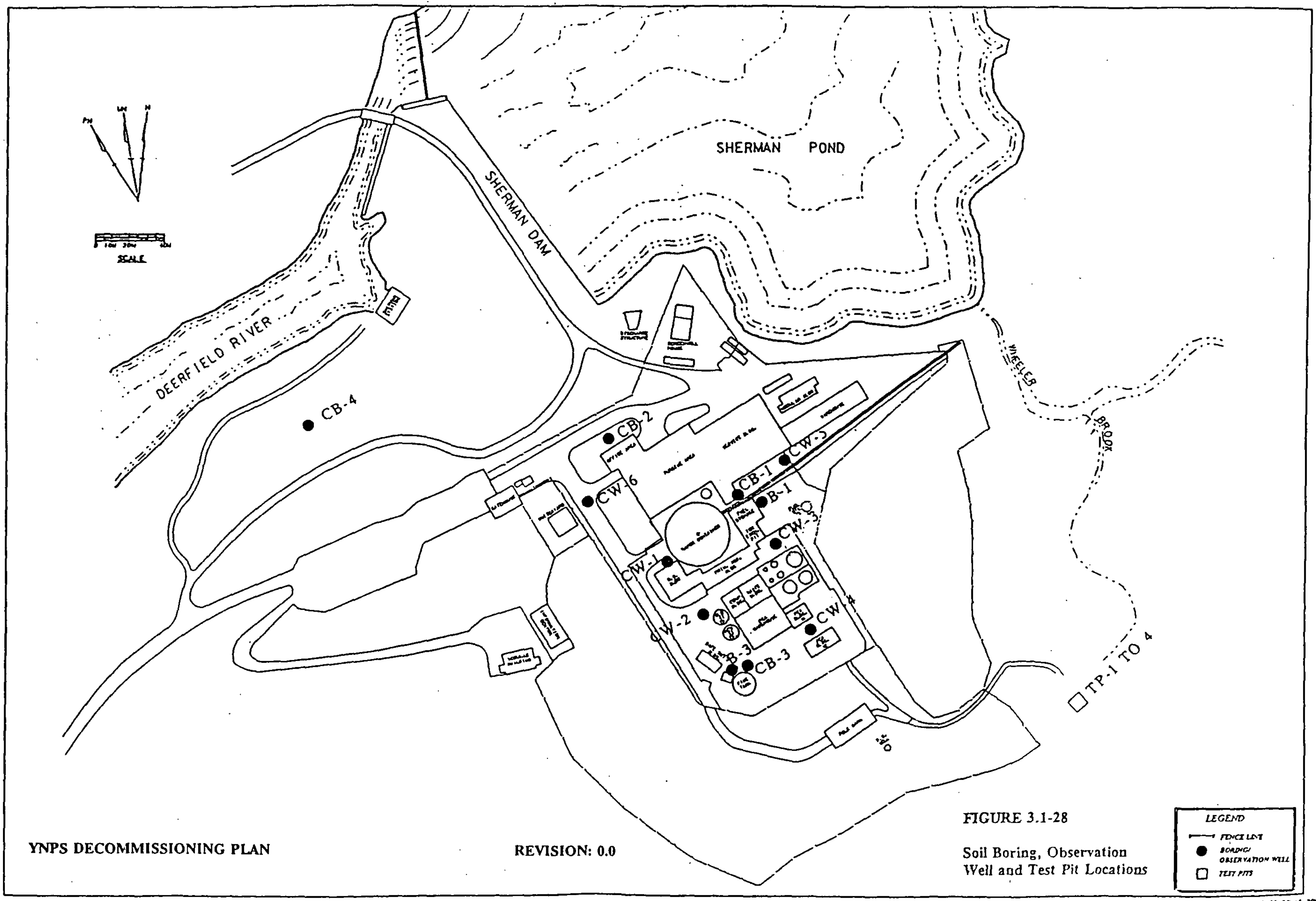
YNPS DECOMMISSIONING PLAN

REVISION: 0.0

FIGURE 3.1-27

Soil and Asphalt Sampling Locations Inside the Radiation Control Area





YNPS DECOMMISSIONING PLAN

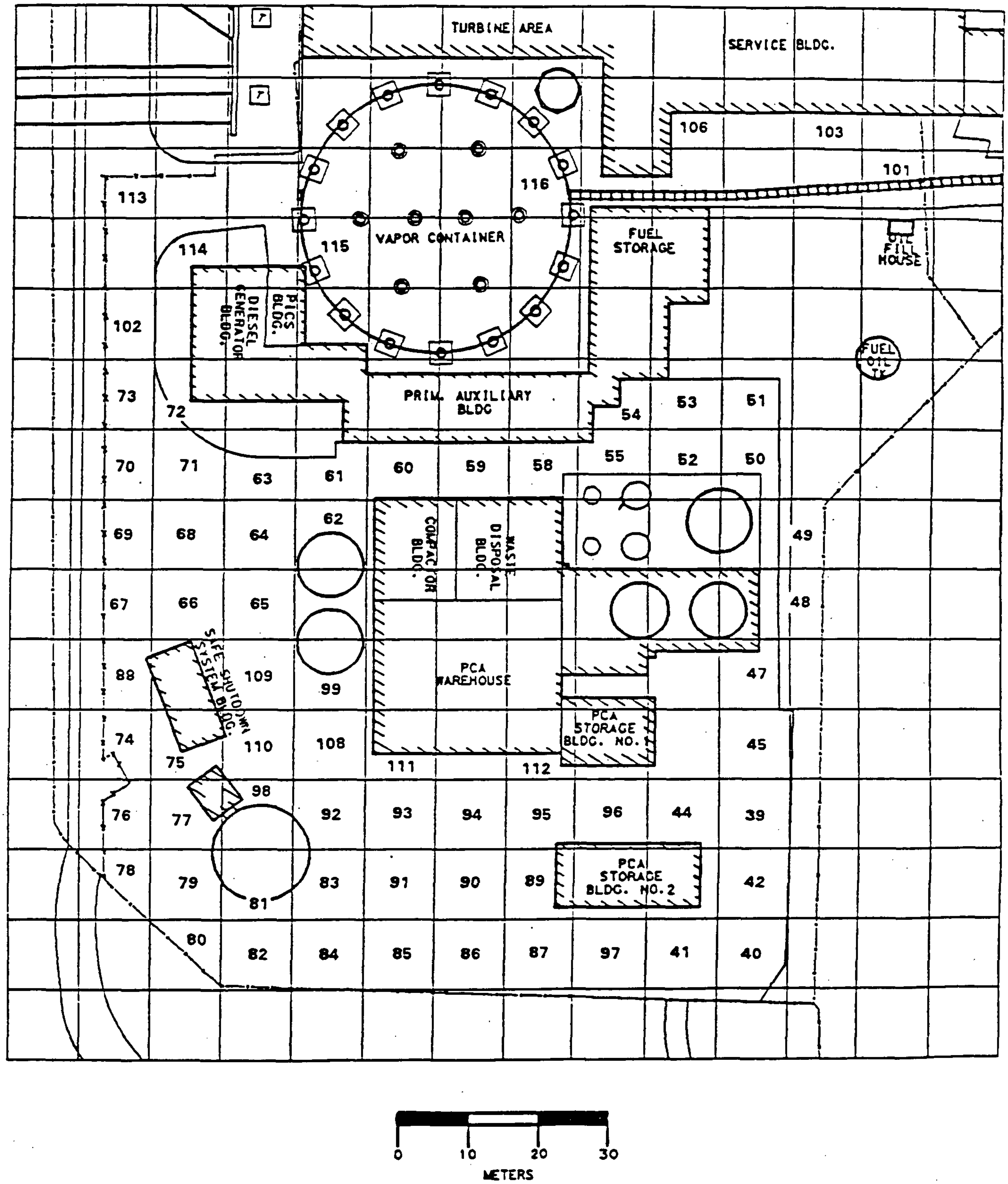
REVISION: 0.0

FIGURE 3.1-28

Soil Boring, Observation Well and Test Pit Locations

FIGURE 3.1-29

In Situ Gamma-Ray Spectroscopy Locations
Inside the Radiation Control Area



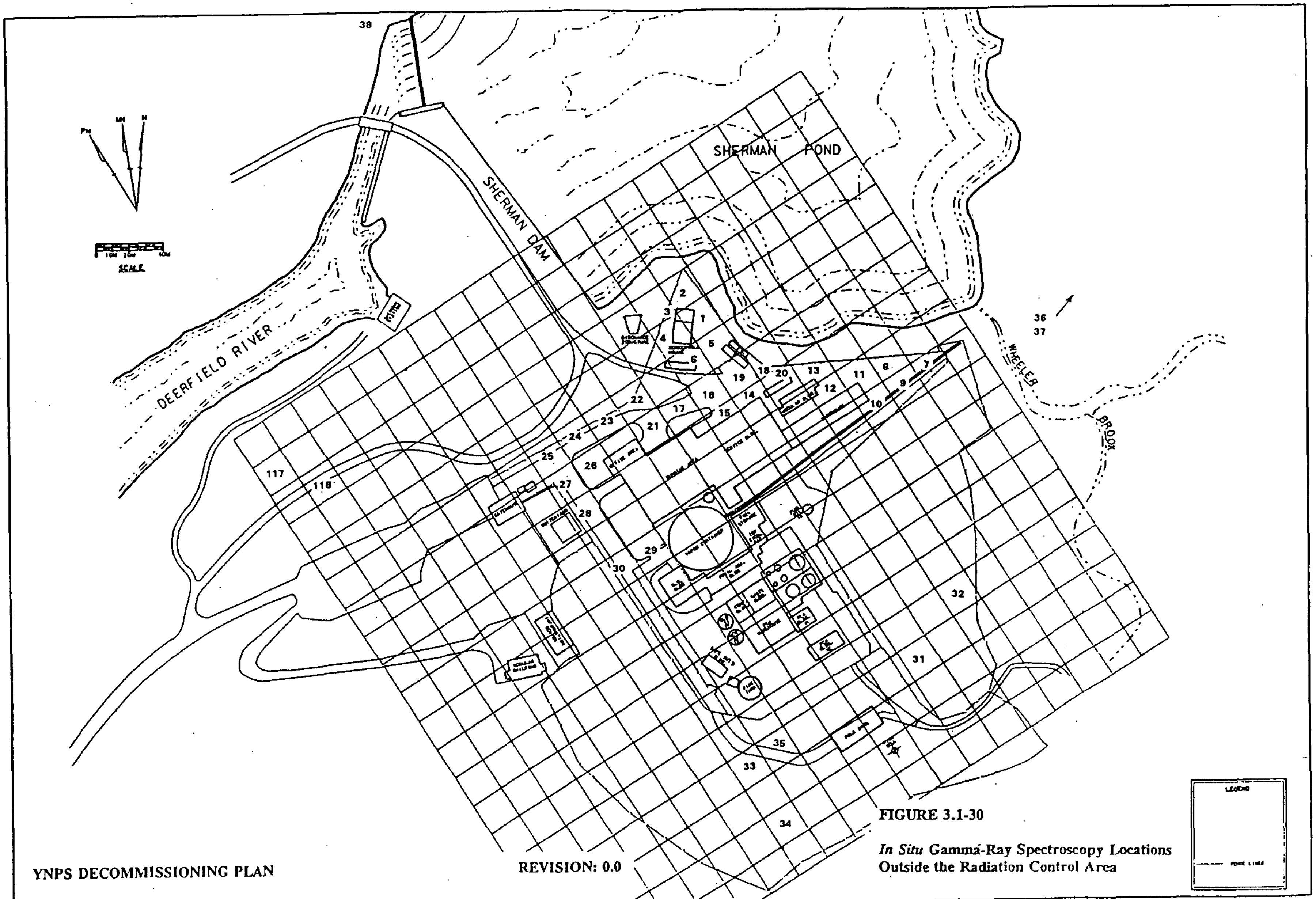


FIGURE 3.1-30

In Situ Gamma-Ray Spectroscopy Locations Outside the Radiation Control Area

FIGURE 3.1 - 31
CO-60 IN BOTTOM SEDIMENT
STATION SE-91, SHERMAN POND

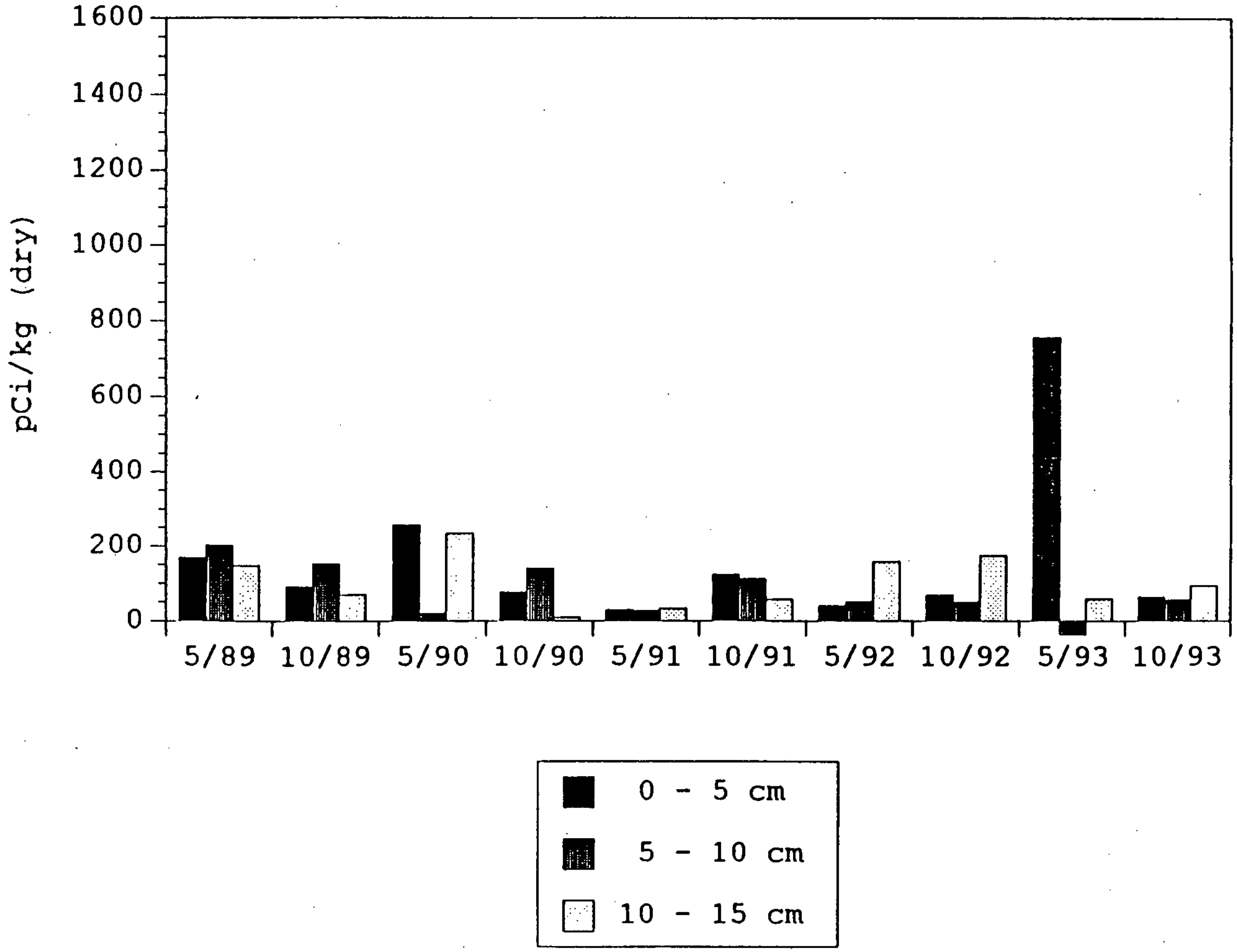


FIGURE 3.1 - 32

CESIUM-137 IN BOTTOM SEDIMENT

STATION SE-91, SHERMAN POND

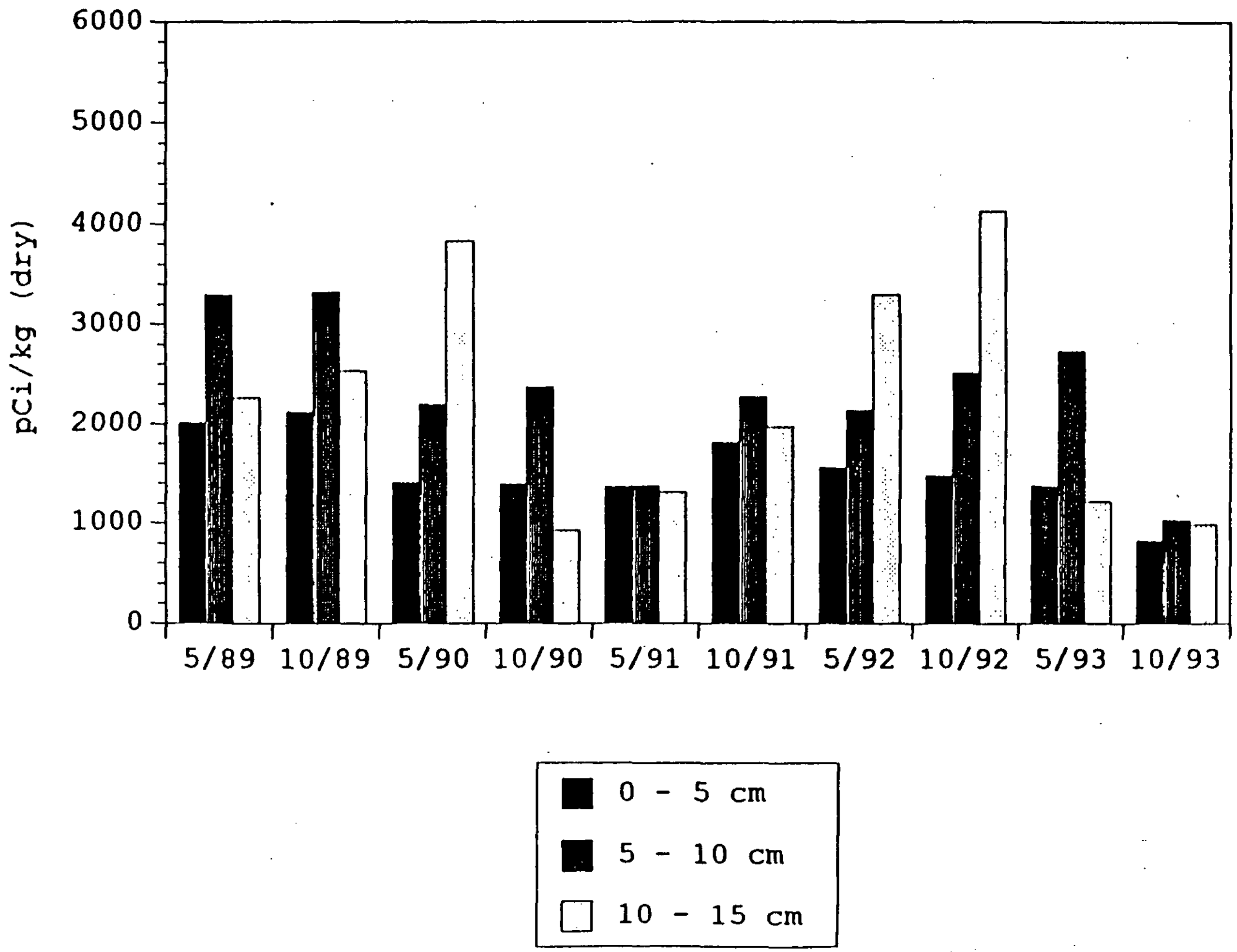


FIGURE 3.1-33
YANKEE NUCLEAR POWER STATION
H-3 IN SHERMAN SPRING WATER

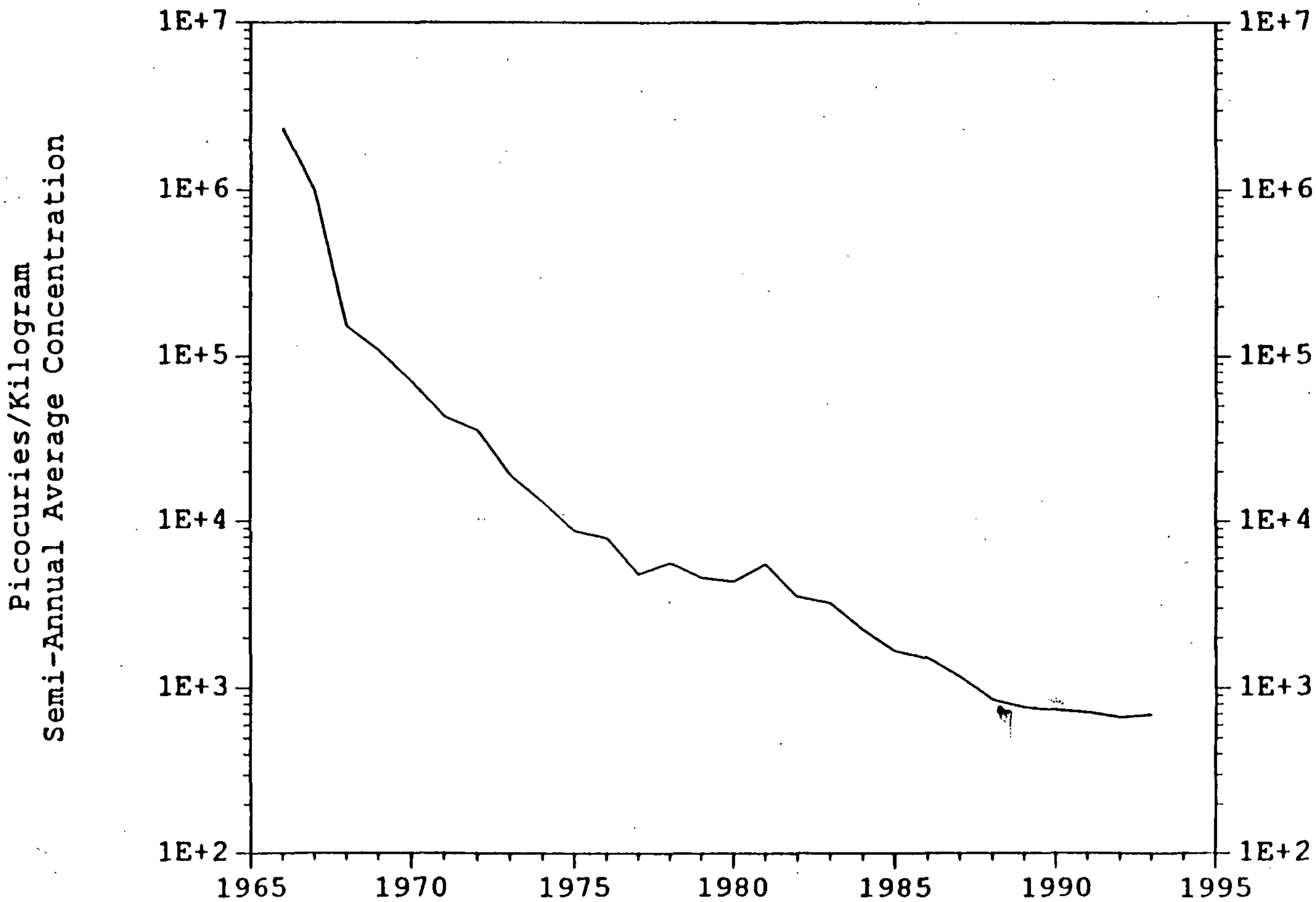
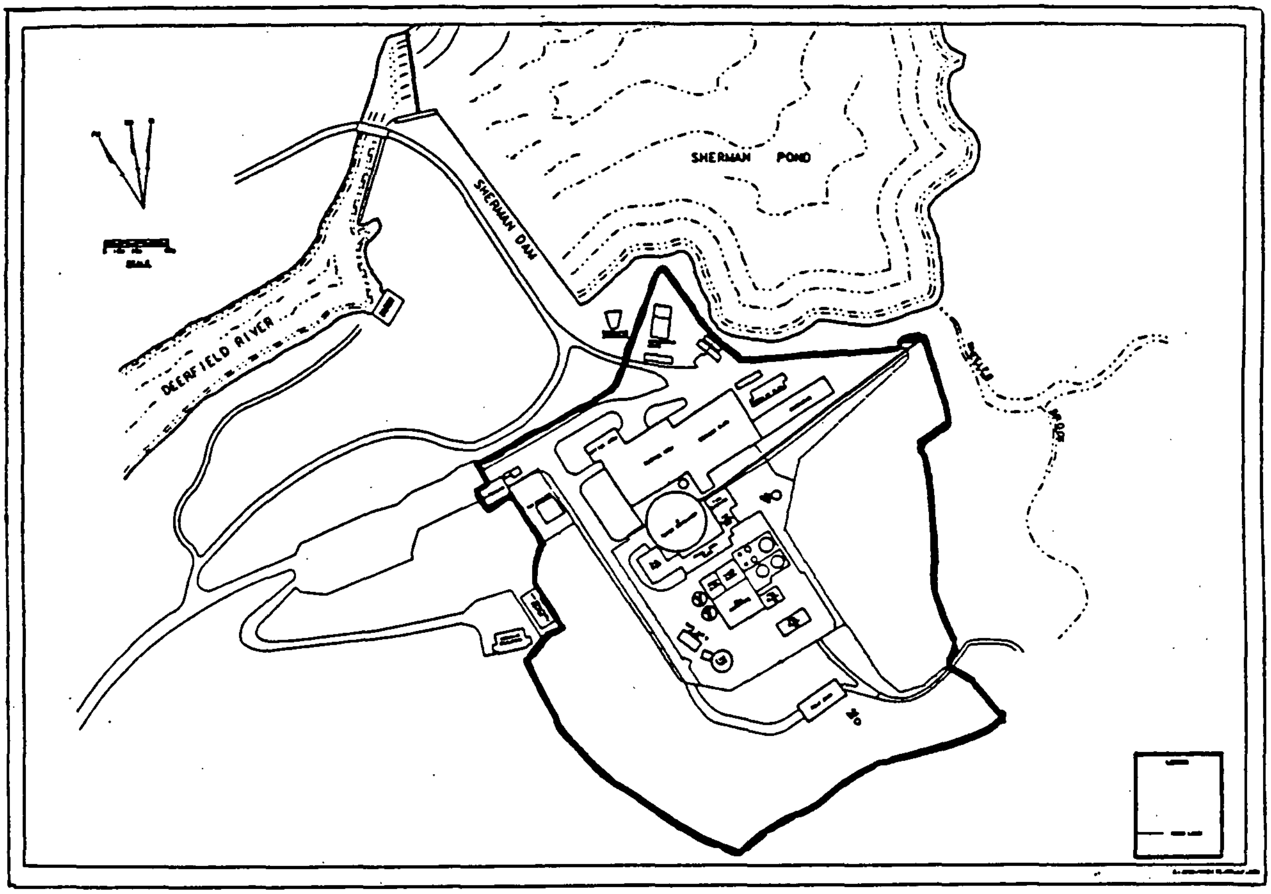


FIGURE 3.1-34

RADIOLOGICALLY AFFECTED AREA



▬▬▬ = AFFECTED AREA BOUNDARY

3.2 RADIATION PROTECTION PROGRAM

3.2.1 Introduction

The Code of Federal Regulations defines decommissioning as the activities necessary "to remove (as a facility) safely from service and reduce residual radioactivity to a level that permits release of the property for unrestricted use and termination of license" (10 CFR 50.2). A comprehensive Radiation Protection Program is needed to meet the objectives presented in the definition. YNPS intends to maintain essential elements of the Radiation Protection Program that were implemented successfully during its 31 year operating life to meet this goal. Changes to the program will be made as necessary to meet the needs of decommissioning.

The Radiation Protection Program has undergone and continues to undergo inspections and audits from both the NRC and the YAEC Quality Assurance Department. The purpose of these reviews is to ensure that the program complies with the Code of Federal Regulations, applicable regulatory guidance documents, and industry standards. The Radiation Protection Program consistently has been rated well by the NRC in the Systematic Assessment of Licensee Performance. In the most recent assessment (Reference 3.2-1), the NRC rated the YNPS Radiological Controls Program as a Category 1 program, indicating a superior level of performance. YAEC is committed to maintaining a high level of performance and to enhancing the quality of the Radiation Protection Program throughout the YNPS decommissioning.

The YNPS Radiation Protection Program for decommissioning will continue to be implemented through existing YNPS administrative procedures. These procedures constitute the highest tier documentation of the Radiation Protection Program and define the radiation protection organization, responsibilities, authorities, administrative policies, program objectives, and standards to implement the Radiation Protection Program.

This section of the plan presents an overview of the Radiation Protection Program administrative and implementing procedures that will be used during decommissioning.

3.2.2 Management Policies

3.2.2.1 Management Policy Statement

YAEC is committed to the safe decommissioning of YNPS. The primary objective of the Radiation Protection Program is to minimize the actual and potential exposure of workers, visitors, and general public to radiation. YAEC and its contractors will provide sufficient qualified staff, facilities, and equipment to conduct a radiologically safe decommissioning. YAEC will continue to comply with regulatory requirements, radiation exposure limits, and radioactive material release limits. In addition, YAEC will make every effort to maintain radiation exposures and releases of radioactive materials in effluents to unrestricted areas as low as is reasonably achievable (ALARA). The ALARA philosophy will be incorporated into all decommissioning activities and will have full management support.

YAEC requires good radiation work practices as a condition of employment. Each radiation worker is responsible for performing work in a radiologically safe manner, consistent with the standards of conduct described in the Radiation Protection Program procedures.

This management policy will continue to be communicated to all radiation workers through General Employee Training and will continue to be incorporated into all applicable procedures.

3.2.2.2 Administration Policy

YAEC will ensure that activities conducted during decommissioning will be managed by qualified individuals who will perform program operations in accordance with established procedures. Radiological hazards will be monitored and evaluated on a routine basis to maintain radiation exposures and the release of radioactive materials to unrestricted areas as far below specified limits as is reasonably achievable. Each element of the Radiation Protection Program will be defined and implemented using written procedures. Radiation protection training will be provided to all occupationally exposed individuals to ensure that they understand and accept their responsibility to follow procedures and to maintain their individual radiation dose as low as is reasonably achievable.

YAEC project management will ensure that work specifications, designs, and work packages involving potential radiation exposure or handling of radioactive materials incorporate effective radiological controls. Project supervisors will incorporate radiation protection considerations in the work activities under their control.

Radiation protection records will be prepared and maintained using high standards of accuracy, traceability, and legibility to meet the requirements of regulatory agencies and company procedures.

3.2.2.3 ALARA Policy

YAEC is committed to maintain an ALARA Program that is implemented based on guidance provided in Regulatory Guides 8.8 and 8.10 (References 3.2-2 and 3.2-3). All activities at YNPS involving radiation and radioactive materials will be conducted such that exposure of employees, contractors, and the general public to radiation is maintained as low as is reasonably achievable. This determination will consider the current state of technology and the economics of improvements in relation to their benefit (i.e., reduction of dose).

Appropriate ALARA considerations will be incorporated into decommissioning activity planning and design activities at an early stage to allow full consideration of reasonable alternatives. Final plant modifications also will be reviewed to ensure that ALARA was incorporated into the activities.

YNPS management will establish and monitor progress towards specific goals and objectives for the YNPS decommissioning ALARA Program.

3.2.2.4 Regulatory Compliance Policy

YAEC is committed to maintain the Radiation Protection Program in compliance with the requirements of the Code of Federal Regulations and, to the extent practical, information contained in industry standards, regulatory guides, and other guidance documents. The Radiation Protection Program will be assessed against all new regulatory guidance and the program will be modified as necessary. YAEC intends to implement the revised 10 CFR Part 20 effective as of January 1, 1994.

3.2.2.5 Waste Minimization and Disposal Policy

YAEC will ensure appropriate processing, packaging, and monitoring of solid, liquid, and gaseous wastes during decommissioning by continuing to implement the Process Control Program, the Radiological Effluent Control Program, and the Radiological Environmental Monitoring Program. These programs will be maintained in strict compliance with Technical Specification and Off-Site Dose Calculation Manual requirements to meet the requirements of 10 CFR Parts 20, 50, 61, and 71; 49 CFR; state regulations; disposal site requirements; and any other applicable requirements. Implementing procedures will be maintained for the classification, treatment, packaging, and shipment of radioactive material.

YNPS will continue to implement and enforce a Waste Minimization Program to minimize the generation of radioactive wastes. YNPS management will establish and monitor waste minimization goals for decommissioning. All decommissioning personnel will receive training in the applicable procedures and practices to minimize the generation of radioactive waste.

3.2.2.6 Respiratory Protection Policy

YAEC is committed to minimizing the inhalation of air contaminated with dusts, mists, fumes, gases, vapors, and radionuclides at YNPS. The primary means of achieving this goal is to prevent or mitigate the hazardous condition at the source. Every reasonable effort will be made to achieve this objective by using engineering controls, including process modification, containment, and ventilation techniques. Respiratory protection equipment usage will be considered after engineering controls have been evaluated. Use of respiratory protection equipment will be consistent with the goal of maintaining the total effective dose to personnel as low as is reasonably achievable.

The existing respiratory protection program will continue to be implemented and maintained in accordance with 10 CFR Part 20 and other applicable regulatory guidance.

3.2.3 Radiation Protection Organization and Functions

3.2.3.1 Radiation Protection Organization

The purpose of the radiation protection organization is to ensure a high level of performance in radiation protection practices through effective Radiation Protection Program implementation and control of all work activities involving radioactivity or radioactive materials. Figure 3.2-1 presents the YAEC management structure that will oversee and control the Radiation Protection Program during decommissioning. The positions shown on Figure 3.2-1 may be modified during the course of decommissioning to support changing requirements.

3.2.3.2 Functional Descriptions

The effective implementation of the Radiation Protection Program is the responsibility of all decommissioning personnel. Table 3.2-1 summarizes the Radiation Protection Organization functional descriptions. The following are the responsibilities of YAEC personnel under the Radiation Protection Organization:

- YAEC Vice President and Manager of Operations - The YAEC Vice President and Manager of Operations is the corporate officer responsible for YNPS nuclear safety. This person may take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant so that continued nuclear safety is assured (TS 6.2.1.b).

- Plant Superintendent - The Plant Superintendent has the overall responsibility for safe operation of the plant and has control over those on-site resources necessary to meet this objective (TS 6.2.1.c). Included in this is the responsibility for assuring effective implementation of the Radiation Protection Program and assuring that all organizations involved with decommissioning are coordinated to achieve the goals of minimizing individual and collective dose and controlling radioactive materials. The Plant Superintendent is the chairperson of the Plant Operation Review Committee and the ALARA Committee.

- Radiation Protection Manager - The Radiation Protection Manager has responsibility for the development and implementation of the Radiation Protection Program policies and standards. The Radiation Protection Manager has the authority to cease any work activity when worker radiological safety is jeopardized or when unnecessary personnel exposure is occurring (TS 6.2.1.e). The Radiation Protection Manager serves as a member of the Plant Operation Review Committee and the ALARA Committee. The Radiation Protection Manager must meet the minimum qualifications of Regulatory Guide 1.8, Revision 1 (TS 6.3.1, Reference 3.2-4).

- Radiation Protection Personnel - Radiation Protection Personnel report to the Radiation Protection Manager and are comprised of the following positions: Radiation Protection Engineer, Radwaste Supervisor, ALARA Supervisor, Radiation Protection Supervisor, Chemist and Environmental Engineers, Radiation Protection Technicians. These positions may be combined dependent on scheduled workload. Radiation Protection Personnel have the authority to cease any work activity when worker radiological safety is jeopardized or when unnecessary personnel exposure is occurring (Reference 3.2-4).

- Shift Supervisor - The Shift Supervisor is responsible for safe plant operations. The Shift Supervisor ensures that planned radiological effluent releases are performed in accordance with plant procedures. The Shift Supervisor is also required to possess and maintain a Certified Fuel Handler qualification.

- Operating Crew Personnel - The Operating Crew Personnel positions fulfill the Technical Specification 6.2.2.d requirement to maintain an individual qualified in radiation protection procedures on site when fuel is stored in the Spent Fuel Pit. The Operating Crew Personnel are responsible for reporting to the Shift Supervisor any contamination or radiation levels which may be detrimental to plant personnel or plant operations.

- Decommissioning Supervisors and Workers - Decommissioning supervisors and workers are responsible for strictly complying with all aspects of the Radiation Protection Program.

- Quality Assurance Personnel - Quality Assurance personnel support the Radiation Protection Manager by providing independent oversight of the implementation of the Radiation Protection Program, including the performance of audits, surveillances, and routine inspections of work areas.

3.2.3.3 Radiation Protection Organization Staffing

The radiation protection organization will continue to provide appropriate personnel and resources to ensure compliance with the Radiation Protection Program and its implementing procedures during decommissioning. Increased staffing will be needed during the dismantlement phase of decommissioning. Specific staffing levels for this phase of decommissioning will be developed by decommissioning project planning personnel and the Radiation Protection Manager during the detailed planning phase of decommissioning. The Radiation Protection Manager will establish guidelines for adequate radiation protection organization staffing based on the site radiological condition and work scope.

Radiation protection organization staffing levels will be reviewed periodically by the Radiation Protection Manager to ensure that adequate staffing levels are maintained consistent with current and planned decommissioning activities. Staffing levels will be adjusted as necessary to ensure that the Radiation Protection Program is implemented effectively.

3.2.3.4 Radiation Protection Program Procedures

The Radiation Protection Program will be integrated into all applicable decommissioning activities. Each element of the Radiation Protection Program, as well as all decommissioning activities, will be defined and implemented using written procedures and instructions.

The radiation protection procedures will be developed and controlled in accordance with the Decommissioning Quality Assurance Plan (Section 7) and Plant Procedure AP-0001, "Plant Procedures." This procedure presents administrative controls for the format, content, review, and approval of all procedures used at the plant.

3.2.3.5 Contracted Radiation Protection Services

Staff augmentation of the radiation protection organization will be necessary to support the dismantlement phase of decommissioning. Contracted services will be used to provide the additional staff. Procurement of contracted radiation protection services will be provided in accordance with the Decommissioning Quality Assurance Plan (Section 7), the Radiation Protection Program, and bid specifications developed for decommissioning. All radiation protection contractors will receive training commensurate with their duties and responsibilities.

3.2.4 Radiation Protection Training and Qualification

All decommissioning workers will be provided radiation protection training commensurate with the radiological hazards that they may encounter during decommissioning. Training is essential to maintaining Yankee's high level of performance in radiation protection activities.

Radiation protection training will be provided to three basic work groups: Non-Radiation Workers, Radiation Workers, and Radiation Protection Personnel. Training for each work group will including the following:

- Non-radiation workers
  - Introduction to Radiation Protection
  - Non-Radiation Worker Indoctrination

- Radiation Workers
  - Radiation Worker Training
  - ALARA Program
  - Respiratory Protection Program
  - Radioactive Waste Reduction Program

- Radiation Protection Personnel
  - Radiation Protection Technician Training
  - Radiation Protection Support Staff Training
  - Operating Crew Personnel Radiation Protection Training

Radiation protection training programs will be developed based upon a systematic approach to training. Radiation protection training for both radiation workers and radiation protection personnel will include the following information regarding decommissioning at YNPS: typical decommissioning activities, radiological concerns during decommissioning, radioactive waste minimization, and decommissioning procedural responsibilities.

All classroom training will be conducted using lesson plans approved by the Radiation Protection Manager. On-the-job training will be administered through procedure AP-0520, "On-The-Job Training Program" or AP-0524, "Radiation Protection Technician Training Program." Personnel assigned to perform training will be qualified as instructors and/or evaluators in accordance with the Radiation Protection Training

Program. Plant Procedure AP-8001, "Radiation Protection Department Organization," presents qualification and training requirements for radiation protection personnel.

The radiation protection training procedures specify the types of training records to be maintained. Training records will be maintained in accordance with Plant Procedures AP-0221, "Plant Record Management," and AP-0508, "Maintenance of Training Records."

3.2.5 ALARA Program

3.2.5.1 General Program Description

All activities at YNPS involving radiation and radioactive material will be conducted such that the radiation doses received by employees, contractors, and the general public are maintained as low as is reasonably achievable. This determination will consider the current state of technology and the economics of improvements in relation to the benefit (i.e., reduction of dose). The YNPS ALARA Program is implemented in Plant Procedure OP-8020, "Implementation and Documentation of ALARA Reviews," and is based on guidance provided in Regulatory Guides 8.8 and 8.10.

The following criteria will continue to be used to determine when an activity requires a specific ALARA review:

- The dose for the total completion of the activity exceeds 1 person-rem.
- The whole body dose rate field exceeds 5 R/hr for an activity other than a surveillance or inspection.
- The loose surface contamination exceeds 500,000 dpm/100cm^2 (beta/gamma) for an activity other than surveillance or inspection.
- The airborne radioactivity concentration exceeds 40 MPC as a result of an activity or in the area of an activity.
- The Radiation protection staff or ALARA Coordinator request an ALARA review.

In addition to these criteria, ALARA considerations will be incorporated into decommissioning activity planning and design activities at an early stage to allow full consideration of reasonable alternatives. Plant modifications also will be reviewed to ensure that ALARA was incorporated into the activities. YNPS management will establish and monitor progress towards specific goals and objectives for the YNPS decommissioning ALARA Program.

3.2.5.2 ALARA Program Organization and Responsibilities

The Radiation Protection Manager coordinates the ALARA Program scope and implementation. The ALARA Coordinator is responsible for completing the ALARA reviews. However, the actual implementation of specific ALARA actions, as incorporated into daily work activities, is the responsibility of each individual manager, supervisor, and worker.

Plant Procedure AP-0820, "Plant ALARA Committee," defines the responsibilities and authorities of the ALARA Committee. The primary responsibility of the ALARA Committee is to advise the Plant Superintendent on matters related to exposure and contamination reduction. The committee will review the following:

- All plant decommissioning activities, maintenance activities, and modifications that have an estimated dose expenditure in excess of 10 person-rem.

- All ALARA post-job reviews for activities exceeding 10 person-rem.

- Any individual's dose which exceeds the quarterly or annual plant administrative limits.

- Any exposure or contamination issue of concern requested by committee members.

The ALARA Committee is chaired by the Plant Superintendent. Other members of the committee include the Assistant Plant Superintendent, Radiation Protection Manager, ALARA Coordinator, and other designated managers and supervisors involved in decommissioning activities. Members are given the appropriate authority and responsibility necessary to implement an effective ALARA Program.

3.2.5.3 ALARA Training and Instruction

Commitment to the principles of the ALARA Program will be reflected in all radiation protection training. Training courses will be evaluated by the Radiation Protection Manager to ensure that ALARA principles are incorporated into lesson plans.

3.2.6 Administrative Dose Control

Administrative radiation dose controls will continue to be implemented during decommissioning. Dose controls ensure the following:

- Personnel do not exceed regulatory dose limits

- Equitable distribution of dose among available qualified workers

- Collective dose to workers is as low as is reasonably achievable

The following procedures implement the program to control and limit external and internal radiation exposure: Plant Procedures AP-0801, "Radiological Work Practices Program," AP-0803, "External Radiation Exposure Control," AP-0804, "Internal Radiation Exposure Control," and AP-0809, "Requirements for Radiation Control Area Access and Egress." These procedures include the following elements:

- A summary of administrative and regulatory dose limits

- A description of Radiation Control Area Postings and Controls

- A description of radiological survey data available on site

- Instructions on the use and care of dosimetry

- Instructions on the conduct of work in the Radiation Control Area

- Instructions on personal monitoring for contamination

Personnel dose reports are prepared weekly with more frequent reports during periods of high work activity. The reports are distributed to each plant department and are posted in the Radiation Control Area Control Point. Decommissioning supervisors are responsible for reviewing the dose reports and planning high dose activities such that the dose is distributed as evenly as possible among available qualified personnel.

3.2.7 Radiation Work Permits

Radiation Work Permits will continue to be used to administratively control personnel entering or working in areas that have, or potentially have, radiological hazards present. The primary function of the Radiation Work Permit is to allow authorized activities to be conducted in radiologically controlled areas using safe and radiologically sound practices. The permit documents the work description, the worker names, the radiological conditions, and the radiological precautions and requirements. The permit also is an element of the ALARA program where it is used to screen activities to determine if a specific ALARA review is necessary and to track personnel and job exposure data.

Plant Procedure AP-0806, "Radiation Work Permits, Request and Use," presents the requirements for requesting, using, and terminating a Radiation Work Permit. Plant

Procedure OP-8415, "Radiation Work Permit Issue, Update, and Closeout," presents the process used by the radiation protection staff to prepare, issue, and monitor a Radiation Work Permit.

Radiation Work Permits are required for the following activities:

- Entry into a high radiation area, the vapor container, an airborne radioactivity area, or any area posted with a sign stating that a Radiation Work Permit is required
- All fuel handling operations
- Maintenance on or inspections of equipment with loose surface contamination levels in excess of 10,000 dpm/100 cm^2 (beta-gamma)
- When prudent radiation protection practices warrant the use of a Radiation Work Permit, as determined by the Radiation Protection Manager

3.2.8 Area Definitions and Postings

Plant Procedure OP-8100, "Establishing and Posting Controlled Areas," describes the requirements for radiological postings at the entrance and boundaries of radiologically controlled areas. The purpose of the postings is to advise workers of radiological hazards that may be encountered in the areas. Informational postings may also be used to provide additional radiological instructions to workers. Each worker is responsible for the observance of the area postings and compliance with the indicated requirements.

3.2.9 External Dosimetry

3.2.9.1 General Considerations

External radiation dose will be monitored through the use of thermoluminescent dosimeters (TLD), direct reading dosimeters, and digital alarming dosimeters. The official record of external dose from beta and gamma radiation normally will be obtained from the TLD readings. Direct reading or digital alarming dosimeters will be used as a means for tracking dose between TLD processing and may also be used as a back-up to the TLDs. TLDs will be processed at a frequency that ensures personnel dose limits are not exceeded.

3.2.9.2 Monitoring Whole Body Dose

All decommissioning workers are required to wear external radiation monitoring devices whenever they enter the Radiation Control Area. Radiation workers are instructed to read the direct reading and digital alarming dosimeters prior to use and periodically during the work activity. The TLD and the direct reading or digital alarming dosimeters are worn typically in close proximity to each other on the trunk of the body between the neck and waist. Under certain conditions, where the chest or trunk may not be the location of highest whole body dose, dosimetry devices may be relocated.

Multiple whole body dosimetry may be used if work is to be performed in a nonuniform radiation field in which the dose to a portion of the body that is exposed to the highest dose source cannot easily be determined. In these cases, multiple sets of dosimeters will be worn on those portions of the body expected to receive the highest dose. Guidance for conducting the evaluation and criteria for determining when multiple dosimetry is required is provided in Plant Procedures OP-8415, "Radiation Work Permit Issue, Update, and Closeout," and DP-8433, "Extremity and Multiple Whole Body Dosimetry Dose Tracking."

Dosimetry requirements are specified on the Radiation Work Permit.

3.2.9.3 Dosimetry Quality Control

Periodic quality assurance checks of dosimetry will be conducted by exposing whole body, extremity, and environmental dosimetry to known radiation doses and then sent to the Yankee Atomic Environmental Laboratory (YAEL) for processing. Plant Procedure DP-8404, "Dosimetry Service Quality Control Program," implements the quality control program for plant dosimetry. Discrepancies between the expected exposure and the laboratory results will be reconciled and documented. The YAEL is accredited by the National Institute of Standards and Technology (NIST) under the National Voluntary Laboratory Accreditation Program (NVLAP) for dosimetry.

3.2.10 Internal Dosimetry Control and Monitoring

3.2.10.1 General Considerations

Internal radiation dose inherently is more difficult to measure than external radiation dose, but it is generally much easier to prevent. Therefore, major emphasis is placed on preventing internal radiation dose, as long as it is consistent with the goal of keeping total effective dose as low as is reasonably achievable.

The primary methods for controlling the intake of radioactive material into the body is identifying and minimizing the sources of airborne radioactivity and applying engineering controls to reduce airborne radioactivity concentrations. The use of respiratory protection will be used after the primary methods have been implemented to the extent practicable.

Plant Procedure AP-0804, "Internal Radiation Exposure Control," describes the program that is implemented to control and limit the potential for internal radiation exposure.

3.2.10.2 Bioassay Program

Whole body counting (in vivo) is the primary method that is used to determine the identity and quantity of gamma emitting radionuclides present in the body. Radiation workers will receive, as a minimum, an initial, a semi-annual, and a termination whole body count. In addition, personnel will receive a whole body count after a suspected intake of radioactive materials. Radiation protection implementing procedures provide guidance on whole body counter operation, calibration, and quality control.

Indirect bioassay (in vitro) measurements will be made, as necessary, to monitor for alpha and beta emitting radionuclides and to provide data for calculation/determination of internal dose. This method of bioassay will typically be used only for radionuclides which cannot be determined by whole body counting or when additional information on an intake is required. Radiation protection implementing procedures include criteria for indirect bioassay and methods for data analysis and interpretation.

Plant Procedure OP-8405, "Bioassay Program," will be used to implement the bioassay program at YNPS.

3.2.11 Respiratory Protection Program

A Respiratory Protection Program will continue to be maintained in accordance with 10 CFR Part 20 and other applicable regulatory guidance. The primary means of providing respiratory protection is to prevent or mitigate the hazardous condition at the source. Every reasonable effort will be made to achieve this objective by using engineering controls, including process modification, containment, and ventilation techniques. Respiratory protection equipment usage will be considered after engineering controls have been evaluated. Use of respiratory protection equipment will be consistent with the goal of maintaining the total effective dose to personnel as low as is reasonably achievable.

Plant Procedures AP-0804, "Internal Radiation Exposure Control," and AP-0810, "Respirator Selection and Use," are used to implement the Respiratory Protection Program at YNPS.

3.2.12 Radioactive Material Controls

The Radiation Protection Program establishes radioactive material controls that ensure the following:

- Prevention of inadvertent radioactive material release to uncontrolled areas
- Assurance that personnel are not exposed inadvertently to radiation from radioactive materials
- Minimization of the amount of radioactive waste material generated during decommissioning

Radioactive material is defined as any of the following: 1) material activated by YNPS reactor operation, 2) material contaminated from the operation or decommissioning of YNPS, or 3) licensed material procured and used to support the operation and decommissioning of YNPS.

All materials leaving the Radiation Control Area and the YNPS site will be surveyed to ensure that radioactive materials are not inadvertently discharged from the facility. Plant Procedure AP-0052, "Radiation Protection Release of Equipment, Materials, and Vehicles," will be used to ensure that all potentially radioactive or contaminated items removed from the Radiation Control Area or the YNPS site are surveyed. This procedure was written to incorporate the guidance presented in NRC Circular No. 81-07 and NRC Information Notice No. 85-92 (References 3.2-5 and 3.2-6). The following survey methods will be used:

- Materials and Equipment - Direct frisking with a portable Geiger-Mueller or a gas flow proportional detector.
- Smear Samples - Analysis with a Geiger-Mueller or a gas flow proportional detector.
- Bulk Liquids or Soil - Analysis with high resolution gamma spectrometry system to the environmental lower limit of detection.

Materials will be released if no discernable plant-related activity is detected within the capability of the survey methods presented above. Any radioactive material that is shipped from the site is handled in accordance with Plant Procedure OP-8301, "Radioactive Material Shipment," which ensures compliance with NRC and Department of Transportation requirements.

Plant Procedure AP-0812, "Control of Radioactive Contaminated Tools and Equipment," provides instructions regarding the proper handling and storage of contaminated tools and equipment. This procedure ensures that tools and equipment are decontaminated promptly. Tools and equipment that are not fully decontaminated are stored in designated radioactive material storage areas. Plant Procedure OP-8100, "Establishing and Posting Controlled Areas," ensures that the areas where radioactive materials are stored are posted clearly.

3.2.13 Surveillance

Routine radiological surveillances will continue to be conducted during decommissioning to monitor radiation sources, to determine radiological conditions, and to comply with the requirements of 10 CFR Part 20. Surveys also will be performed to evaluate radiological conditions in support of decommissioning work activities. Plant Procedures OP-8101, "Plant Radiological Surveys," and OP-8102, "Plant Airborne Radioactivity Surveys," will be used to implement the surveys. These procedures specify the types of instrumentation, survey methods, and review requirements for each survey performed.

The final radiation survey that will be completed following decontamination and dismantlement activities is described in Section 4 of this plan.

3.2.14 Instrumentation

A sufficient inventory and variety of operable and calibrated portable, semi-portable, and fixed radiological instrumentation will be maintained on site to allow for effective measurement and control of radiation exposure and radioactive material and to provide back-up capability for inoperable equipment. Equipment will be capable of measuring the range of gamma, beta, and alpha dose rates and radioactivity concentrations expected. Instrumentation will be calibrated at prescribed intervals or prior to use against certified equipment having known valid relationships to nationally recognized standards. Plant Procedure AP-8006, "Control of Radiation Protection Instrumentation," will be used to control the use of radiation protection instrumentation.

Installed process and effluent monitors are used in accordance with the Off-Site Dose Calculation Manual (Reference 3.2-7).

3.2.15 Review and Audit

To ensure the Radiation Protection Program is effectively implemented and maintained, an organized system of reviews and audits will continue to be implemented during decommissioning in accordance with the Quality Assurance Plan presented in Section 7 of this plan.

3.2.16 Radiation Protection Program Performance Analysis

Plant Procedure AP-0802, "Radiological Occurrence Reports," will be used during decommissioning to evaluate the causes of unacceptable radiation protection performance, to initiate corrective actions, and to trend overall performance. This process will be used to address the following types of deficiencies:

- Work activities generating unnecessary radiation exposure or contamination
- Procedural actions resulting in unacceptable radiological performance
- Unacceptable radiological work practices resulting in personnel contamination, spread of contamination, or unnecessary radiation exposure
- Activities resulting in unnecessary generation of liquid or solid radioactive waste
- Activities violating Radiation Work Permit instructions, postings, signs, and radiation protection implementing procedures

Incidents that result in more serious radiological events will be reported using Plant Procedure AP-0008, "Event Reportability Evaluation Process." These events include overexposures, large intakes of radioactive material, unplanned radioactive releases, and significant radioactive spills. AP-0008 ensures that immediate and written notifications are made in accordance with regulatory requirements. The procedure also provides for initiation of a Plant Information Report in accordance with Plant Procedure AP-0004, "Plant Information Reports," if appropriate.

REFERENCES

3.2-1 NYR 91-097, Systematic Assessment of Licensee Performance (SALP) Final Report for Yankee Nuclear Power Station for the Period August 1, 1989 to January 15, 1991 (50-29/89-99), T. T. Martin (USNRC) to A. C. Kadak, May 20, 1991.

3.2-2 Regulatory Guide 8.8, Information Relevant to Ensuring That Occupational Radiation Exposure at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable.

3.2-3 Regulatory Guide 8.10, Operating Philosophy For Maintaining Occupational Radiation Exposure As Low As Is Reasonably Achievable.

3.2-4 Regulatory Guide 1.8, Personnel Selection and Training.

3.2-5 NRC IE Circular No. 81-07: "Control of Radioactively Contaminated Material", May 14, 1981.

3.2-6 NRC IE Information Notice No. 85-92: "Surveys of Wastes Before Disposal From Nuclear Reactor Facilities", December 2, 1985.

3.2-7 Off-Site Dose Calculation Manual, Revision 10, June 1993.

PROCEDURE REFERENCES

AP-0001, "Plant Procedures"

AP-0004, "Plant Information Reports

AP-0008, "Event Reportability Evaluation Process"

AP-0052, "Radiation Protection Release of Equipment, Materials, and Vehicles"

AP-0221, "Plant Record Management"

DP-0508, "Maintenance of Training Records"

AP-0520, "On-The-Job Training Program"

AP-0524, "Radiation Protection Technician Training Program"

AP-0801, "Radiological Work Practices Program"

AP-0802, "Radiological Occurrence Reports,"

AP-0803, "External Radiation Exposure Control"

AP-0804, "Internal Radiation Exposure Control"

AP-0806, "Radiation Work Permits, Request and Use"

AP-0809, "Requirements for Radiation Control Area Access and Egress"

AP-0810, "Respirator Selection and Use"

AP-0812, "Control of Radioactive Contaminated Tools and Equipment,"

AP-0820, "Plant ALARA Committee"

AP-8001, "Radiation Protection Department Organization"

AP-8006, "Control of Radiation Protection Instrumentation"

OP-8020, "Implementation and Documentation of ALARA Reviews"

OP-8100, "Establishing and Posting Controlled Areas"

OP-8101, "Plant Radiological Surveys"

OP-8102, "Plant Airborne Radioactivity Surveys"

OP-8301, "Radioactive Material Shipment"

DP-8404, "Dosimetry Service Quality Control Program"

OP-8405, "Bioassay Program"

OP-8415, "Radiation Work Permit Issue, Update, and Closeout"

DP-8433, "Extremity and Multiple Whole Body Dosimetry Dose Tracking"

TABLE 3.2-1

RADIATION PROTECTION PROGRAM RESPONSIBILITIES MATRIX

Position	Responsibility Number							
	1	2	3	4	5	6	7	8
Plant Superintendent	X	X	X	X				
Radiation Protection Manager	X	X	X	X	X			
Radiation Protection Engineer	X	X	X		X			
Radwaste Supervisor	X	X	X			X		
ALARA Supervisor	X	X	X	X			X	
Radiation Protection Supervisor	X	X	X		X			
Chemistry & Environmental Eng.	X	X			X			
Radiation Protection Technicians	X	X	X		X			X
Shift Supervisors	X	X	X					
Operating Crew Personnel (1)	X	X	X					
Decommissioning Supervisors	X							
Decommissioning Personnel	X							
Quality Assurance Personnel	X							

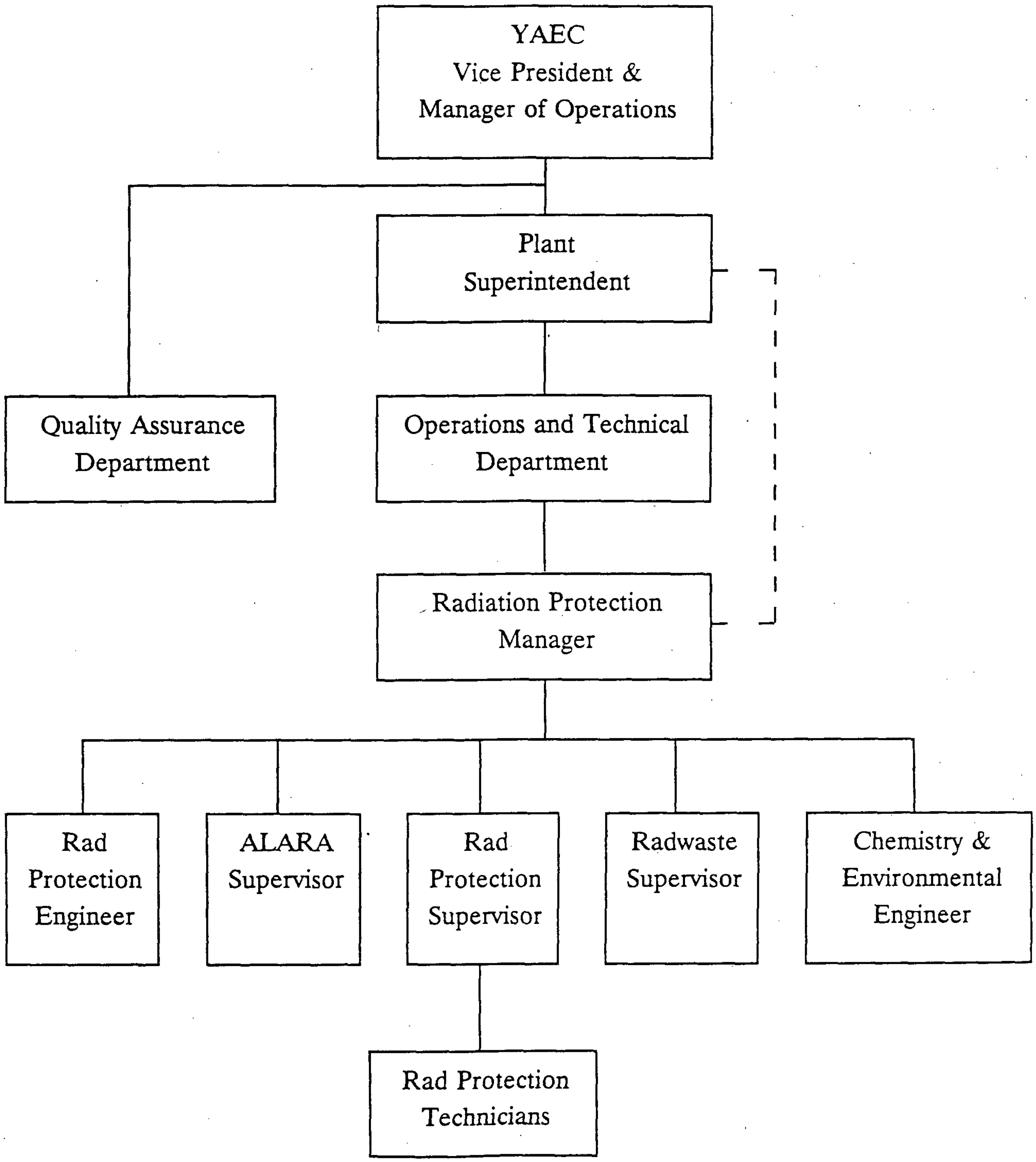
Responsibility Key:

1 = Comply with all aspects of the Radiation Protection Program
2 = Ensure effective implementation of the Radiation Protection Program
3 = Have "stop work" authority
4 = Participate on the ALARA Committee
5 = Ensure proper calibration and use of radiation monitoring equipment
6 = Coordinate waste minimization, classification, packaging, and shipping
7 = Coordinate the ALARA program, including ALARA reviews, exposure trending and tracking
8 = Perform radiation surveys, provide job coverage, prepare Radiation Work Permits, monitor work practices

Note: 1. This position meets the requirements of Technical Specification 6.2.2.d during back shifts.

FIGURE 3.2-1

RADIATION PROTECTION ORGANIZATIONAL STRUCTURE



3.3 RADIOACTIVE WASTE MANAGEMENT

YNPS decommissioning requires handling of a large volume of radioactive materials to reduce residual radioactivity to a level permitting release of the site for unrestricted use and termination of license. Materials that are not decontaminated and released will be processed as radioactive waste. This section of the decommissioning plan presents the programs used to manage spent fuel and to control the processing of solid, liquid, and gaseous radioactive waste.

YAEC will continue to ensure appropriate processing, packaging, and monitoring of solid, liquid, and gaseous wastes during decommissioning by implementing the Radiation Protection procedures, the Process Control Program, the Radioactive Effluent Controls Program, and the Radiological Environmental Monitoring Program. These programs will be maintained in compliance with technical specification requirements to meet federal and state regulations, disposal site requirements, and any other applicable requirements. The YNPS radioactive waste management program is implemented through the Radiation Protection Program (Section 3.2). Implementing procedures will be used to control the classification, treatment, packaging, and shipment of radioactive material.

3.3.1 Spent Fuel Management Plan

3.3.1.1 Background

The NRC has proposed a rule to ensure that licensees of prematurely shutdown facilities notify the NRC of their spent fuel management and funding plans in a timely manner (Reference 3.3-1). The proposed rule recognizes the significant impact of spent fuel storage on decommissioning safety and cost. This section of the plan documents the YNPS spent fuel management plan. The spent fuel storage cost estimate is presented in Section 5.

Currently, there are 533 fuel assemblies stored in double tier racks in the Spent Fuel Pit. These fuel assemblies were discharged from the reactor between 1972 and 1992. There also are several failed fuel pins that must be consolidated before they are moved from the Spent Fuel Pit. The Spent Fuel Pit also contains high level waste in the form of canisters containing reactor vessel internals (core baffle and lower core support plate). The canisters have the same external dimensions as fuel assemblies. Several miscellaneous low level radioactive items also are stored in the Spent Fuel Pit (e.g. neutron sources, filter cartridges, material from reconstitution activities).

Yankee is currently seeking accelerated acceptance of YNPS's spent fuel by the Department of Energy in accordance with the current fuel disposal contract. The Department of Energy's current position is that they have not yet determined whether priority will be accorded shutdown reactors, or if priority is granted, under what specific circumstances it might be granted. A rulemaking is scheduled that will include the issue of acceptance priority for shutdown plants. YAEC will participate in the rulemaking process, however, it appears that priority for shutdown plants may not be supported by the majority of participants in this process.

It is unlikely that the Department of Energy will accept all YNPS spent fuel before the beginning of the dismantlement period in 2000. Although the Department of Energy may start taking fuel as early as 1998, for planning purposes fuel shipments are assumed to be completed in 2018. This projection is based on the Department of Energy's Acceptance Priority Ranking, Annual Capacity Report, and an extrapolation beyond the 10 year Department of Energy outlook. For planning purposes, YAEC's current decommissioning cost estimate assumes storage of fuel in the Spent Fuel Pit until 1996, at which time it will be transferred to an on-site dry storage facility. Spent fuel is assumed to remain in the dry storage facility until 2018.

3.3.1.2 Spent Fuel Management Strategy

A task force was formed, following the decision to permanently cease power operations, to develop a spent fuel management strategy. The task force completed a comprehensive evaluation of the following spent fuel storage and disposal alternatives (Reference 3.3-2):

- Continued operation of the Spent Fuel Pit and required auxiliary systems
- Construction of an independent wet spent fuel storage facility
- Construction of a dry cask spent fuel storage facility
- Shipment of the spent fuel off site

Based on the task force review, the following spent fuel management strategy was implemented:

- Continue operation of the Spent Fuel Pit and implement any safe, but economically attractive enhancements
- Urge the Department of Energy to accelerate acceptance of spent fuel or to accept financial responsibility for on-site spent fuel storage
- Continue evaluations of wet and dry storage options to reflect Yankee and industry developments

- Initiate preliminary design activities for a dry storage facility

The decision to initiate preliminary design activities for a dry storage facility is based on the reduced operating costs of the dry storage facility compared to those of the Spent Fuel Pit. The wet storage option also requires operation of support systems, restricting dismantlement activities. Spent Fuel Pit operations will continue until fuel is removed permanently to either an on-site or off-site facility. Section 3.3.1.3 presents limitations on Spent Fuel Pit operations during decommissioning activities.

Compatibility of the dry cask primary containment canister with shipping containers and disposal site requirements are significant issues. Currently there is no primary containment canister that is approved for either shipment to or disposal in a disposal facility. This issue is significant for YNPS because the capability to reconfigure the fuel elements within the primary containment canister will be eliminated (or severely reduced) after the Spent Fuel Pit is decontaminated and dismantled. The Department of Energy is currently in the process of developing a multi-purpose canister system that addresses the compatibility issue, however, progress is slow. In addition, the YNPS Spent Fuel Pit and Yard Area Crane designs limit the applicability of the transfer systems currently proposed by the Department of Energy.

YAEC has concluded that, if spent fuel is not transferred to the Department of Energy in the short term, a down-sized canister system, compatible with the Department of Energy transport cask, is appropriate for YNPS. A custom-designed transport cask is also considered a viable option. The down-sized system would be compatible with the on-site handling systems eliminating the need for extensive structural modifications or increased crane capacity. A proposed location for a new storage location is at the south end of the YNPS site. An additional benefit to construction and operation of the dry cask facility is that it will permit decommissioning of the Spent Fuel Pit structure coincident with other dismantlement activities, leaving removal of the relatively simple and essentially uncontaminated dry cask facility when fuel is removed by the Department of Energy.

3.3.1.3 Spent Fuel Pit Operation Limitations

YAEC also evaluated the option to store fuel in the Spent Fuel Pit during a portion of the dismantlement phase of decommissioning (Reference 3.3-3). The purpose of the evaluation was to identify safety considerations and limitations on decommissioning activities associated with operating the Spent Fuel Pit concurrent with dismantlement activities. Operation of the Spent Fuel Pit during the dismantlement phase would allow YAEC additional time to pursue early transfer of spent fuel to the Department of Energy without incurring a significant investment associated with the dry cask facility.

This option also allows additional time for the development of a multi-purpose canister system that is compatible with YNPS limitations.

Dismantlement activities are limited when spent fuel is stored in the Spent Fuel Pit. Activities cannot be pursued that could result in the loss of Spent Fuel Pit integrity or in physical damage to the fuel that would reduce subcriticality margin or cause a loss of a coolable geometry. These events can be precluded by incorporating the following provisions in the plan:

- Delay dismantlement of the Ion Exchange Pit until after the fuel is removed permanently from the Spent Fuel Pit

- Limit cask usage in and over the Spent Fuel Pit to those activities that have acceptable cask drop consequences. Technical Specification 3.2 currently limits cask usage over the Spent Fuel Pit to a shipping cask weighing less than 35 tons.

- Isolate the Fuel Transfer Chute from the Spent Fuel Pit by capping the chute inside the Spent Fuel Pit and filling the chute with structurally stable material between the cap and the lower lock valve. This must be completed before initiating activities that could adversely interact with the Fuel Transfer Chute.

- Ensure that detailed work planning excludes activities that could result in a drop of a heavy load onto or into the Spent Fuel Pit or Fuel Transfer Chute. Ensure that partial dismantling of equipment and structures does not result in a configuration that could result in failure during an external event (e.g., seismic event) and subsequent collapse onto or into the Spent Fuel Pit or Fuel Transfer Chute. Alternatively, consider modifications to protect the Spent Fuel Pit and the Fuel Transfer Chute from heavy load drops. Technical Specification 3.2 limits movement of loads over the Spent Fuel Pit to those less than 900 pounds.

- Ensure that demolition explosives that could affect the Spent Fuel Pit structure are not permitted for use until either fuel is removed permanently from the Spent Fuel Pit or an analysis of the impact of explosives on the Spent Fuel Pit structure is completed.

Movement of fuel from the Spent Fuel Pit to either an on-site or off-site fuel storage facility will require a separate safety analysis to ensure that there are no unacceptable interactions between fuel movement and decommissioning activities.

Although it is important to take actions to prevent a loss of spent fuel cooling capability, the consequences of such an event are not severe. As of January 1994, greater than four weeks must elapse without re-establishing cooling or adding make-up water before the water remaining in the Spent Fuel Pit is insufficient to provide shielding adequate for operator response in the Spent Fuel Pit Building. At that time a make-up water flow rate of less than 1 gpm is needed to replace water lost through evaporation. Several diverse sources of make-up are available including demineralized water, fire water, service water, or Sherman Reservoir water. Water may be injected to the Spent Fuel Pit through installed or portable pumps as well as gravity feed. Adequate time is available either to re-establish Spent Fuel Pit cooling or to provide make-up water to maintain Spent Fuel Pit inventory.

Although scenarios that result in a loss of spent fuel cooling capability can be mitigated without any significant consequence, the capability should be protected during decommissioning. Dismantlement activities near and around the Spent Fuel Pit Cooling System and other support systems should be controlled to prevent damage to these systems. This may be accomplished by physically protecting the systems or by establishing safe load paths and protective zones around the systems.

3.3.2 Solid Radioactive Waste Processing

Solid radioactive waste handling at YNPS is divided into three phases: packaging, on-site storage awaiting shipment, and shipment. Each of these phases will be implemented in strict compliance with technical specifications and applicable federal, state, and disposal site requirements. In addition, all waste processing activities will be completed in accordance with the requirements of the Decommissioning Quality Assurance Plan (Section 7).

YNPS decommissioning solid waste is comprised of both high and low level radioactive waste. Several of the reactor vessel internal components (e.g., core baffle and lower core support plate) have radionuclide concentrations in excess of the 10 CFR Part 61 Class C limits. These materials are not generally acceptable for near-surface disposal and have been classified as high level radioactive waste. High level radioactive waste will be stored with the fuel and will be shipped to an appropriate disposal facility after a location becomes available.

3.3.2.1 Solid Radioactive Waste Packaging

Radioactive waste packaging at YNPS will be performed in areas that minimize radiation exposure to personnel, control the spread of contamination, and are adequate for packaging activities. Examples of potential on-site waste packaging areas are: Compactor Building and Potentially Contaminated Area Warehouse, Vapor Container, Primary Auxiliary Building, and Spent Fuel Pit Building. Temporary facilities, that meet the requirements above, also may be constructed for waste packaging.

Radioactive waste packaging operations will be implemented through plant procedures that ensure the following:

- A Radiation Work Permit has been issued for handling radioactive materials
- Specific packaging requirements are identified
- Quality assurance personnel have been notified of packaging operations
- Containers are surveyed for external contamination

Quality assurance personnel will observe final loading and container preparation activities in accordance with the requirements of the Decommissioning Quality Assurance Plan (Section 7).

Waste packages will meet the requirements for transportation and disposal for each decommissioning waste stream. Examples of the waste containers that may be used are drums, boxes, liners, high integrity containers, sea-land containers, shielded casks, and other specialty containers. Waste container selection will be determined by the size, weight, classification, and activity level of the material to be packaged. Selection of the appropriate packaging is the responsibility of the Radwaste Supervisor. In all cases, packaging will comply with requirements specified by 49 CFR, 10 CFR Part 71, and the disposal facility site criteria, as applicable.

Plant Procedure OP-8301, "Radioactive Material Shipment," provides instructions for determining the 10 CFR Part 61 classification of low level radioactive waste. The procedure is used to determine the radionuclide content of a container through a combination of direct measurements and radiation shielding calculations.

Spent resins, filter media, and other wetted wastes requiring stabilization will be processed in accordance with the Process Control Program (Reference 3.3-4), which is implemented through plant operating procedures. Whenever possible, stabilization will be completed inside the disposal package or liner to minimize additional waste handling prior to disposal.

3.3.2.2 Solid Radioactive Waste Storage Awaiting Shipment

Solid radioactive waste awaiting shipment to a disposal facility normally will be stored in the following locations:

- Potentially Contaminated Area Buildings and Warehouse

- Compactor Building
- Sea-Land Containers for short-term storage

Radioactive materials will be stored such that in the event of a fire or an explosion, sufficient fire detection, response, and suppression capability in conjunction with spatial separation will ensure that any radiological release would be bounded by the accident analyses presented in Section 3.4.

Large components awaiting shipment may be stored in the Yard Area prior to shipment. Precautions will be taken to ensure that the components are within barriers, as necessary, and adequately protected from on-site hazards (e.g., heavy load movement).

3.3.2.3 Solid Radioactive Waste Shipment

Solid radioactive wastes will be shipped in compliance with applicable federal and state regulations. Plant Procedures OP-8301, "Radioactive Material Shipment," and OP-8302, "Final Preparation, Inspection, and Loading of Radwaste Containers for Transport and Disposal," present the requirements for radioactive materials shipment. These procedures ensure the following:

- Appropriate labeling and documentation of shipping containers
- Quality assurance oversight of shipment preparation
- Verification of acceptable package physical condition and contamination levels
- Appropriate permits and licenses for waste shipment
- Notification of appropriate governmental agencies prior to shipment

Most of the radioactive material and waste shipments will be completed over roads. Rail transportation may be used for heavy shipments (e.g., Reactor Vessel, if intact removal is chosen). The routing of shipments may vary with weather and highway conditions. Additionally, local and state restrictions pertaining to radioactive material transport may affect some route selections. The carrier is responsible for selecting the appropriate route, which must conform to applicable federal, state, and local shipping requirements and be in accordance with Department of Transportation and NRC regulations.

3.3.3 Liquid Radioactive Waste Processing

Contaminated water will be generated during YNPS decommissioning as a result of draining, decontamination, and cutting processes. The contaminated liquids will be processed either in the liquid waste evaporator or in a temporary facility (e.g., ion exchange and filtration system, solidification system). All liquid radioactive waste will be processed in accordance with the Process Control Program, the Off-Site Dose Calculation Manual (ODCM), applicable technical specifications, and plant procedures.

The Process Control Program presents the administrative and technical controls for the liquid radioactive waste solidification system to assure that solidified waste meets shipment and disposal facility requirements. Liquid waste processing is monitored to assure safe operation, storage, drumming, and disposal of waste to approved waste disposal sites. Liquids released from the site are monitored and controlled to ensure all releases of radioactivity to the environment are as low as is reasonably achievable. The Process Control Program is maintained in accordance with Technical Specification 6.12.

Technical Specification 6.7.5 establishes two programs affecting radioactive liquids processing: Radioactive Effluent Controls Program, Radiological Environmental Monitoring Program. The Radioactive Effluent Controls Program conforms with 10 CFR 50.36a requirements to control radioactive effluents and to maintain dose to members of the public from radioactive effluents as low as is reasonably achievable. This program is presented in the ODCM (Reference 3.3-5) and implemented through several plant procedures. This program complies with the requirements of Technical Specifications 6.7.5 and 6.13.1.

The ODCM contains methodologies and parameters used in the following:

- Calculation of off-site doses resulting from radioactive gaseous and liquid effluents
- Calculation of gaseous and liquid effluent monitoring alarm and trip setpoints
- Conduct of the Radiological Environmental Monitoring Program

The ODCM forms the basis of plant procedures which document the off-site doses due to plant operation. The off-site dose calculations demonstrate compliance with the numerical guides for design controls of 10 CFR Part 50, Appendix I. Several plant procedures implement the ODCM requirements:

- AP-4900: "Chemistry Surveillance Tests" - Administrative procedure presenting required frequency of chemistry analyses to support ODCM required analyses. Each chemistry analysis is implemented through a plant procedure.

- OP-4952: "Radiological Environmental Sampling" - Implementing procedure for the Radiological Environmental Monitoring Program that determines the effectiveness of the Radioactive Effluent Controls Program.

- OP-8040: "Implementation of Radioactive Discharge Permits" - Implementing procedure to establish release limits that assure that the dose to the public is maintained as low as is reasonably achievable.

- OP-9246: "Radioactive Liquid Releases" - Implementing procedure to ensure that all liquid releases are monitored and maintained within the limits of the ODCM requirements.

3.3.4 Airborne Radioactive Waste Processing

Airborne radioactive waste processing is limited to radioactive particulate emissions during decontamination and dismantlement activities. Exhaust air from the Vapor Container, Primary Auxiliary Building cubicle area, Waste Disposal Building, and Spent Fuel Pit Building is filtered through a high efficiency filter assembly before discharging to the Primary Vent Stack. Instrumentation channels monitor gas released through the Primary Vent Stack. Plant Procedure OP-9247, "Sampling, Measuring, and Reporting Radioactive Airborne Releases," ensures that airborne releases are monitored and maintained within the limits of the ODCM.

Dismantlement activities will be designed to ensure that airborne releases are monitored to the maximum extent practicable by implementing the following considerations during detailed planning of decommissioning activities:

- The VC Ventilation and Purge System will be maintained in operation during decontamination and dismantlement activities in the Vapor Container.

- The Ventilation System will be maintained in operation during decontamination and dismantlement activities in the Primary Auxiliary Building (PAB) cubicle area, Spent Fuel Pit Building, and Waste Disposal Building.

- The PAB and Diesel Generator Building Roof Fans will be secured and not operated during decontamination and dismantlement activities in the PAB and Diesel Generator Building.

- Local HEPA filtration systems will be used when activities could result in the release of significant radioactive particulates. The local HEPA filtration systems should exhaust to areas served by the Ventilation System when used outside of the Vapor Container to monitor airborne releases.

Airborne effluents from the Primary Vent Stack will be monitored and be reported using installed plant equipment and established procedures in accordance with Off-Site Dose Calculation Manual requirements. Local supplemental air monitoring will be performed to support decommissioning activities.

3.3.5 Mixed Waste

Yankee has a non-radioactive waste management program to ensure compliance with all the federal and state hazardous waste regulatory requirements. The use of hazardous materials and the generation of hazardous wastes are controlled through the non-radioactive hazardous waste management program. This program is presented in Section 3.6.

No chemicals or other substances are anticipated to be used during decommissioning operations that could become mixed waste. If mixed wastes are generated, they will be managed according to Subtitle C of the Resource Conservation and Recovery Act (RCRA) to the extent it is not inconsistent with NRC handling, storage, and transportation regulations.

Mixed wastes from the YNPS will be transported only by authorized and licensed transporters and shipped only to authorized and licensed facilities. If technology, resources, and approved processes are available, processes will be evaluated to render the mixed waste nonhazardous.

3.3.6 Radioactive Waste Minimization

3.3.6.1 Radioactive Waste Reduction Program

YNPS will continue to implement and enforce a Radioactive Waste Reduction Program through implementation of Plant Procedure AP-0807, "Radioactive Waste Reduction Program." YNPS management will establish and monitor waste minimization goals for decommissioning. All decommissioning personnel will receive training in the applicable procedures and practices to minimize the generation of radioactive waste.

All workers entering the Radiation Control Area will receive radiation worker training. This training will include a review of work techniques that prevent unnecessary contamination of areas and equipment, practices for reuse of materials, and policies to prevent the unnecessary generation of mixed or radioactive wastes.

Performance indicators will be developed to track total radioactive waste generated during decommissioning. The actual waste generation rates will be compared to the estimates made during detailed work planning.

The following radioactive waste volume reduction methods will be incorporated into YNPS decommissioning activities:

- Prevention of Waste - Unnecessary generation of radioactive wastes will be controlled by procedures established to prevent unnecessary packaging, tools, and equipment from entering Radiation Control Area.

- Decontamination and Re-Use of Materials - Materials will be re-used to the maximum extent practicable. Typical materials reused during the decommissioning include contaminated tools, equipment, and clothing. Contaminated tools and equipment storage areas will be maintained. Protective clothing and collection bags will be laundered, repaired, and made available for re-use.

Voids in disposal containers will be filled with other contaminated material to reduce the total volume of waste for disposal to the maximum extent practicable. This produces a better waste form, maximizing burial efficiency, and minimizing project cost, disposal site usage, and transportation risk.

3.3.6.2 On-Site Decontamination Methods

On-site decontamination techniques will be used for processing and volume reduction of radioactive materials. The following are currently available decontamination methods that may be used during decommissioning:

- Strippable Coatings - Strippable coatings may be used to lift radionuclides from contaminated surfaces. A strippable coating is typically applied in a manner similar to spray painting a surface. Additives in the coating are designed to attract and to combine chemically with radioactive contaminants. Once the coating is dry, the contaminant is locked in the coating. The dried coating is easily removed from the surface, stripping the film containing the contamination. The stripped film is packaged and buried as a solid waste. Strippable coating may also be used to protect surfaces from becoming contaminated.

- Chemical or Solvent Decontamination - Chemical and solvent decontamination methods remove contamination by creating a solution of the radionuclides and the solvent used. This type of decontamination may be difficult to control because of the aggressiveness of the chemicals and solvents. However, the systems decontaminated at YNPS will not be returned to service after decontamination, therefore, excessive metal wastage is not significant. Chemicals used for decontamination will be evaluated for hazardous constituents. If the chemical could become a listed or characteristic hazardous mixed waste it will not be used.

- Dry Abrasive Impingement - Dry abrasive impingement is effective for removing heavy or tightly adhering oxide films. Examples of this technology are sandblasting and dry ice blasting.

- Water Washing - High pressure water washing is effective for removing surface contamination and for sluicing sludge from tanks. Barriers must be established to ensure that wash water is collected and processed in the plant liquid waste processing system.

- Vacuum Cleaning: HEPA filtered vacuum cleaners may be used in areas of high loose surface contamination.

3.3.6.3 Off-Site Radioactive Materials Processing

Several off-site radioactive materials processing options are currently available. However, more stringent radioactive material release criteria could limit the availability of processing alternatives. The current decommissioning cost estimate (Section 5) assumes that the availability of processing alternatives is limited and that all significantly contaminated and activated materials are sent to a low level radioactive waste disposal facility. However, all processing alternatives will be evaluated during decommissioning to determine the most effective processing of radioactive materials.

The following are currently available off-site processing alternatives for radioactive materials removed during YNPS decommissioning:

- Decontamination - Decontamination facilities provide a wide range of decontamination technologies at centralized locations. The variety allows selection of appropriate technologies for each component of the decommissioning waste stream.

- Volume Reduction - Volume reduction facilities provide various processes (e.g., sorting, super-compaction) to reduce the volume of material that is sent to the disposal facility. This processing alternative is attractive for asbestos compaction which requires specialized containment during processing.

- Incineration - Incineration facilities safely incinerate materials resulting in very high volume reduction rates. Appropriate materials may include paper, certain plastics, lubricating oils, and solvents.

- Metal Melting - Metal melting materials process low specific activity metals. The processed metal is recycled to the nuclear industry as shielding and potentially in the future as cask liners and fuel canisters.

Waste packages will be transported to off-site facilities primarily in sea-land containers selected to meet transportation and receipt requirements of the off-site processing facility. Voids in transport containers are not a significant concern. However, efficient management of transportation resources is an important consideration to minimize the total number of shipments and decommissioning costs.

Radioactive material control and accountability procedures will be implemented to track material originating from YNPS during receipt, sorting, processing, and packaging for disposal. Off-site processing facilities will be selected that provide adequate radioactive material control and accountability procedures.

REFERENCES

3.3-1 58-FR-34947, Notification of Spent Fuel Management and Funding Plans By Licensees of Prematurely Shut Down Power Reactors, June 30, 1993.

3.3-2 YRP 435/92, Spent Nuclear Fuel Storage Study Report and Recommendations, B. W. Holmgren, J. M. Buchheit, R. A. Mellor to J. K. Thayer, October 9, 1992.

3.3-3 YRP 303/93, Impact of Wet Spent Fuel Storage on Decommissioning, P. A. Rainey to R. A. Mellor, July 15, 1993.

3.3-4 Yankee Nuclear Power Station Process Control Program.

3.3-5 Yankee Nuclear Power Station Off-site Dose Calculation Manual.

PROCEDURE REFERENCES

AP-0807, "Radioactive Waste Reduction Program"

AP-4900, "Chemistry Surveillance Tests"

OP-4952, "Radiological Environmental Sampling"

OP-8040, "Implementation of Radioactive Discharge Permits"

OP-8301, "Radioactive Material Shipment"

OP-8302, "Final Preparation, Inspection, and Loading of Radwaste Containers for Transport and Disposal"

OP-9246, "Radioactive Liquid Releases"

OP-9247, "Sampling, Measuring, and Reporting Radioactive Airborne Releases"

3.4 ACCIDENT ANALYSIS

3.4.1 Overview

This section of the decommissioning plan presents an accident analysis that assesses the impact of decommissioning on both occupational and public health and safety. A structured, comprehensive process was used to identify and to evaluate events that could occur during the period from approval of the decommissioning plan through completion of the final radiation surveys (Reference 3.4-1). The accident analysis considered decommissioning events, fuel storage events, and external events. Analysis of decommissioning events included all phases of decommissioning activities: decontamination, dismantlement, packaging, storage, and radioactive material handling.

The risk of accidents resulting in a significant radiological release during decommissioning activities is considerably less than that during plant operations. YAEC evaluated all of the Final Safety Analysis Report Section 400 safety analyses for applicability to a permanently defueled condition (Reference 3.4-2). The only design basis event remaining applicable was the Spent Fuel Pit fuel handling accident. The remaining events which could occur in a permanently defueled condition and impact the health and safety of the public are related to the release of airborne radioactive materials during decommissioning activities.

3.4.1.1 Radionuclide Release Limits

Prior to the decision to permanently cease power operations, radiological releases resulting from design basis accidents postulated in safety evaluations and the Final Safety Analysis Report were evaluated using dose reference values from 10 CFR Part 100. The 10 CFR Part 100 reference values limit dose to an individual at the Exclusion Area Boundary during the first two hours following the onset of a postulated fission product release. The dose limits are less than 25 rem total dose to the whole body and less than 300 rem thyroid dose from radioactive iodine.

Since the decision to permanently cease power operations, YAEC requested and received an exemption from the emergency preparedness requirements of 10 CFR 50.54(q) (Reference 3.4-3). This exemption allowed YAEC to discontinue off-site emergency response activities and to refocus the scope of on-site response capability. NRC approval of the exemption was predicated on the absence of accidents at a level of severity where the off-site dose could exceed the Environmental Protection Agency (EPA) Protective Action Guides (PAGs). Off-site protective actions are not warranted if the off-site dose following a postulated accident is less than the EPA PAGs.

The EPA 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 all significant inhalation pathways during the early phase of an event (Reference 3.4-4):

	EPA PAGs, rem
Whole Body	1
Thyroid	5
Skin	50

Releases resulting from accidents postulated in the decommissioning accident analysis were evaluated using the EPA PAGs as an upper limit. This ensures that the current Defueled Emergency Plan remains adequate for decommissioning and eliminates the need to re-institute an off-site emergency response capability. Use of the EPA PAGs as an administrative limit also ensures that postulated accident off-site doses are significantly less than the 10 CFR Part 100 reference values.

3.4.1.2 Assumptions

The following assumptions have been incorporated into the accident analysis:

- Special Nuclear Material used as reactor fuel will not be moved into the Reactor Vessel. This a condition of the YNPS possession only license.

- Removal of the steam generators, pressurizer, and Reactor Vessel internal components through the Component Removal Project has been completed prior to commencing decommissioning activities evaluated in this safety analysis. Any activities initiated before approval of the decommissioning plan will be evaluated in accordance with the requirements presented in Section 1.5.

- The airborne pathway is the dominant radioactivity release pathway. Activities that could result in release of radioactive liquids will be designed to contain the releases within the liquid waste processing system using existing or supplemental barriers.

- Airborne releases are assumed to occur at ground level with a conservative dispersion factor of 2.84E-04 sec/m^3 (Reference 3.4-5).

- Direct failures and consequences of initiating events were considered in the consequence analyses. Separate, coincident, random failures were not considered.

- Decommissioning activities are independent from each other. There are no credible common cause mechanisms that could result in the simultaneous release of radioactivity from multiple activities that would exceed the equivalent release of the radioactive contents of the single, bounding container or component that results in the highest off-site dose. Interactions between systems during radioactive material handling activities will be precluded by the maintenance of safe load paths; protective zones around limiting systems, structures, and components; and single handling criteria for higher contamination items. The consequences of fire and explosions may impact several activities simultaneously. These events are considered separately in Sections 3.4.7 and 3.4.8.

- Decommissioning activities may be performed in the Vapor Container without isolating the Vapor Container from the environment. However, the capability to isolate the Vapor Container will be retained to mitigate the consequences of a significant radioactive release. If an accident occurs, the Vapor Container will be isolated expeditiously. Vapor Container isolation is the closure of all penetrations and openings to restrict transport of airborne radioactivity from the Vapor Container atmosphere to the environment. Pressure retention capability is not necessary.

3.4.2 Event Identification Process

A structured, comprehensive process was used to identify accident initiating events which could lead to radionuclide releases during the YNPS decommissioning (Reference 3.4-1). The process included development of a logic diagram to evaluate all phases of decontamination, dismantlement, and fuel management activities as well as to identify non-radiological events.

Accident initiating events were grouped by structures, systems, and components within a plant area. The accident initiating events were compared using previous YNPS accident analyses, as well as current evaluations and calculations to identify the dominant accident initiating events within each plant area and then among plant areas. Accident scenarios were developed for these dominant accident initiating events. The scenarios formed the bases and inputs for radiological dose calculations to determine the impact on the health and safety of the public.

The following events were considered in the accident analysis and are presented below:

- Events affecting occupational health and safety, including radiological and non-radiological events

- Off-site events affecting public health and safety

- Non-radiological events affecting public health and safety

- Radiological events affecting public health and safety, including the following:
  - Decommissioning activity events, including decontamination, dismantlement, packaging, storage, and materials handling
  - Loss of support system events, including loss of off-site power, cooling water, and compressed air
  - Fire and explosion events
  - External events
  - Spent fuel storage events, including fuel handling event, loss of spent fuel cooling capability, and interactions between spent fuel and decommissioning activities

3.4.3 Events Affecting Occupational Health and Safety

3.4.3.1 Radiological Events

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 (Section 3.2) and the Defueled Emergency Plan (Reference 3.4-16).

The Radiation Protection Program is applied to all activities performed on site involving radioactive materials. A primary objective of the Radiation Protection Program is to protect workers and visitors to the site from radiological hazards that have the potential to develop during decommissioning. The program requires YNPS and its contractors to provide sufficient qualified staff, facilities, and equipment to perform decommissioning 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 on a routine basis to maintain radiation exposures and the release of radioactive materials to unrestricted areas as far below specified limits as is reasonably achievable. Radiation protection training will be provided to all occupationally exposed individuals to ensure that they understand and accept the responsibility to follow procedures and to maintain their individual radiation dose as low as is reasonably achievable.

Project management will ensure that work specifications, designs, and work packages involving potential radiation exposure or handling of radioactive materials incorporate effective radiological controls. Task planning will include consideration of the potential adverse events. The objective of this planning is to ensure that protective measures and contingency plans are developed to address the potential occurrence of these events and to minimize their impact on the workers as well as the public health and safety.

The Defueled Emergency Plan retains an on-site emergency response capability. This capability includes removal of personnel from an affected area, including site evacuation, if necessary. The plan is implemented by the control room personnel.

Implementation of these programs ensures that potential radiological events affecting occupational health and safety will be sufficiently minimized and mitigated to not warrant further consideration in this analysis.

3.4.3.2 Non-Radiological Events

Decommissioning YNPS may require different work activities than were typically conducted during normal plant operations. Effective implementation of the Occupational Safety Program (Section 3.5) to decommissioning activities will ensure worker safety. The goal of the Occupational Safety Program is to provide a hazard-free environment for employees. The program incorporates safety into every phase of decommissioning from early design through implementation.

Implementation of the Occupational Safety Program will ensure that industrial safety events are eliminated to the maximum extent possible.

3.4.4 Events Affecting Public Health and Safety

3.4.4.1 Offsite Radiological Events

The Environmental Supplement for the YNPS decommissioning presents the impact of decommissioning on the public and the environs. Off-site events related to decommissioning activities are limited to those associated with the shipment of radioactive materials. Radioactive shipments will be made in accordance with all applicable requirements (e.g., NRC, Department of Transportation). The radioactive waste management program (Section 3.3) and the Decommissioning Quality Assurance Plan (Section 7) assure compliance with these requirements.

Compliance with these requirements ensures that both the probability of occurrence and the consequences of an off-site event do not significantly affect the public health and safety.

3.4.4.2 Non-Radiological Events

There are no decommissioning events that can be initiated from non-radiological sources that could significantly impact public health and safety.

Hazardous materials handling will be controlled through the Non-Radioactive Hazardous Materials Program and the Chemical Control Program (Section 3.6). There are no chemicals stored on-site which, after release, could significantly threaten public health and safety. Inflammable gases stored on-site include combustible gases used for cutting and welding and liquid propane gas (LPG) used for operation of forklift trucks. Safe storage and use of these gases and any other inflammable materials is controlled through the Occupational Safety Program and Fire Protection Program (Section 3.5 and 9).

The programs described above are implemented through procedures that control material identification, inventory, handling, storage, use, and disposal, minimizing the probability of on-site non-radiological events. In addition, procedures present mitigative measures that would be implemented if an event occurred.

Implementation of these programs ensures that the probability of occurrence and consequence of on-site non-radiological events do not significantly affect public health and safety.

3.4.4.3 Radiological Events

Radiological events that affect the health and safety of the public are considered to be those that could result in a release of radioactive materials exceeding the EPA PAGs. These events are divided into several categories: decommissioning activity events, external events, and fuel storage events. Sections 3.4.5 through 3.4.10 present the dominant radiological events.

3.4.4.4 Radiological Analysis Basis

The consequences of postulated decommissioning accident scenarios on the health and safety of the public were determined by calculating the potential dose at the Exclusion Area Boundary. The location of the Exclusion Area Boundary is presented in Technical Specification 5.1.1 as a point 3100 ft from the center of the Vapor Container. The airborne release is assumed to result from release of the entire potential airborne radioactivity in a container or component. Releases to the environment are assumed to be at ground level. Activities that could result in release of radioactive liquids will be designed to contain the releases within the liquid waste processing system using existing or supplemental barriers.

An atmospheric dispersion factor of 2.84E-04 sec/m^3 (Reference 3.4-5) was used to estimate the two hour dose at the Exclusion Area Boundary resulting from a ground level release of radioactivity. The ground level release assumption results in selection of a conservative atmospheric dispersion factor, increasing the calculated dose. The radionuclide distributions used to evaluate postulated releases are estimated from the radiological scoping survey data (Table 3.1-2). Radionuclide distributions, contamination levels, and radioactivity contents are calculated as of January 1, 1994, with the exception of the fuel handling accident which is based on May 31, 1992 (Reference 3.4-6).

The ELISA computer program (Reference 3.4-7) was used to calculate the Exclusion Area Boundary doses relative to the radioactivity released (Reference 3.4-8). The following doses were calculated: total effective dose equivalent, thyroid dose, and skin dose. In each case, the total effective dose equivalent was the limiting value. Table 3.4-1 presents the dose to radioactivity conversion factors used in the analysis.

The off-site dose was determined based on the airborne radioactivity released in each accident scenario. Most of the calculations for the dominant scenarios used the highest dose to radioactivity conversion factors, which were based on the Main Coolant System and Bleed Line radionuclide distribution. Doses estimated for activated components were based on a combined release of loose activated base material (e.g., fine cutting debris, concrete dust) and surface contamination. Table 3.4-2 presents a summary of the materials that were assumed to be released.

The radiological analysis results in conservative, bounding estimates of the radiological consequences of the events considered in this analysis based on the following:

- The atmospheric dispersion factor was based on conservative meteorology. Realistic meteorology would increase dispersion, decreasing the dose by about a factor of 3 (Reference 3.4-7).

- The radioactivity release estimates assume that all of the radioactivity released to the environment is incorporated into a plume and is transported to the Exclusion Area Boundary. Only a fraction of the radioactivity released will form a plume and a portion of the plume will drop out prior to reaching the Exclusion Area Boundary, decreasing the dose.

- The release fractions from the systems, structures, and components were assumed to be 100% of the surface contamination and loose activated base metal. This could only result as a non-mechanistic release of radioactivity following a significant force being applied to the container or component. A significant fraction of the contamination that could be released is tightly bound to the surface. Radiological scoping analysis indicated that this fraction was between 50% to 80% (Reference

3.4-6). Only a fraction of the total radioactivity would be released as a result of the energy imparted during an impact on the component or container, if the component or container is breached. More realistic release fractions of 1% to 10% are most likely justifiable. These fractions are consistent with drop, fire, and explosion scenarios in comparable evaluations (Reference 3.4-14).

Releases resulting from accidents postulated in the decommissioning accident analysis were evaluated against the EPA PAGs. Events less than the guides were classified as not having a significant effect on the public health and safety.

3.4.5 Decommissioning Activity Events

Decommissioning activities were identified on a location-by-location basis. Nine plant areas containing radiologically contaminated and activated systems, structures, and components were identified. Dominant systems, structures, and components were identified based on the amount of potential airborne radioactivity that could be released during decommissioning activities.

For each of the dominant systems, structures, and components in each plant area, five decommissioning process steps were considered: decontamination, dismantlement, packaging, storage, and material handling.

3.4.5.1 Decontamination Events

Selected systems, structures, and components will be decontaminated during decommissioning to remove radioactivity from or stabilize radioactivity on external and internal surfaces. Decontamination methods that may be applied during YNPS decommissioning are presented in Section 2.3.4. External contamination levels are significantly lower than the contamination levels on internal surfaces of systems and components. Therefore, the bounding decontamination event is based on decontamination of internal surfaces.

Internal decontamination methods typically use liquids to remove radioactivity from the surface (e.g., chemical decontamination, high pressure water washing). Detailed planning of decommissioning activities that use liquids will ensure that contaminated liquids will be processed by the liquid waste processing system. Additionally, existing or supplemental barriers will be used to ensure that inadvertent spills from these activities will be contained within the liquid waste processing system. These precautions prevent an unmonitored release of radioactive liquids to the environment.

If the Reactor Vessel is removed as a single component (Section 2.3.5.2), a fixative will be applied to the internal surface to stabilize the contamination after the vessel is drained. Surface contamination could be disturbed during application of the coating, potentially creating airborne radioactivity. A local HEPA filtration unit will be used to remove airborne radioactivity generated by this process. Additionally, routine air sampling performed as part of the Radiation Protection Program would detect increased radioactivity in the Vapor Container atmosphere and the inadvertent release of radioactivity would be identified and stopped.

A bounding analysis was completed for the purposes of this accident analysis to estimate the consequences of generating airborne radioactivity during decontamination activities. The dominant system, structure, or component that could cause the highest off-site dose as a result of a release of airborne radioactivity during decontamination is the Reactor Vessel. The bounding analysis conservatively assumes that all of the radioactivity in the contamination layer on the internal surface of the Reactor Vessel is non-mechanistically released to the Vapor Container atmosphere.

Based on the evaluation of a gas bottle explosion event in the Vapor Container presented in Section 3.4.8, the amount of radioactivity that would be transported from the Vapor Container to the environment following the event was about 10%. The motive forces associated with the release of contamination during a decontamination event are significantly less than those during an explosion. However, an instantaneous release of 10% of the potential airborne radioactivity from the component to the environment was used as a bounding value for the purposes of the calculation.

The estimated radioactivity content of the Reactor Vessel internal surface contamination is about 23 Ci. A release of 10% of this material to the environment would result in an off-site dose at the Exclusion Area Boundary of about 0.078 rem. This is significantly less than the EPA PAGs. In addition, radiological scoping analyses indicated that between 50% and 80% of the internal surface contamination is tightly bound. Incorporation of this effect would reduce the dose to less than 0.039 rem.

The radiological consequences of the bounding decontamination event result in a dose at the Exclusion Area Boundary significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by the potential decontamination events that could occur during decommissioning.

3.4.5.2 <u>Dismantlement Events</u>

Systems, structures, and components will be dismantled during decommissioning to remove radioactive materials from the site. Dismantlement methods that may be applied during YNPS decommissioning are presented in Section 2.3.4. Detailed planning will ensure that the following systems with high internal contamination will be dismantled using mechanical methods (e.g., split frame machining): Main Coolant System, Feed and

Bleed Heat Exchanger, Vapor Container Bleed Line Piping. This restriction limits the amount of airborne contamination generated during dismantlement activities. Thermal dismantlement methods (e.g., plasma arc, oxy-fuel) may be used on systems with lower radioactivity levels.

If the Reactor Vessel is segmented, a combination of cutting methods will be implemented: metal disintegration machining and milling (Section 2.3.5.2).. Most of the Reactor Vessel cutting will be conducted under water, which minimizes the generation of airborne radioactivity. In addition, local HEPA filtration will be used to remove any gases and airborne radioactivity generated during cutting operations.

The portion of the Reactor Vessel above the vessel support lugs will be segmented above water. Prior to cutting, the internal surface will be decontaminated to remove loose contamination. The radioactivity from neutron activation of the vessel wall in this region is significantly lower compared to the core region. A contamination envelope with HEPA filtration will be used to preclude release of airborne radioactivity to the Vapor Container atmosphere.

A bounding analysis was completed for the purposes of this accident analysis to estimate the consequences of generating airborne radioactivity during dismantlement activities. The dominant system, structure, or component that could cause the highest off-site dose as a result of a release of airborne radioactivity during dismantlement is one Main Coolant System loop. The bounding analysis conservatively assumes that all of the radioactivity on the internal surface of one Main Coolant System Loop is non-mechanistically released to the Vapor Container atmosphere. This event is bounding because underwater cutting of the highest radioactivity regions of the Reactor Vessel reduces the potential airborne contamination generation from that source.

Based on the evaluation of a gas bottle explosion event in the Vapor Container presented in Section 3.4.8, the amount of radioactivity that would be transported from the Vapor Container to the environment following the event was about 10%. The motive forces associated with the release of contamination during a dismantlement event are significantly less than those during an explosion. However, an instantaneous release of 10% of the potential airborne radioactivity from the Vapor Container to the environment was used as a bounding value for the purposes of the calculation.

The estimated radioactivity content of a Main Coolant System loop internal surface contamination is about 15 Ci. A release of 10% of this material to the environment would result in an off-site dose at the Exclusion Area Boundary of about 0.051 rem. This is significantly less than the EPA PAGs. In addition, radiological scoping analyses indicated that between 50% and 80% of the internal surface contamination is tightly bound. Incorporation of this effect would reduce the dose to less than 0.026 rem. A non-mechanistic release of the cuttings from segmentation of the upper portion of the Reactor Vessel segmentation would result in an off-site dose of less than 0.013 rem.

The radiological consequences of the bounding dismantlement event result in a dose at the Exclusion Area Boundary significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by the potential dismantlement events that could occur during decommissioning.

3.4.5.3 Packaging Events

Radioactive materials are packaged prior to shipment from YNPS to either a disposal facility or an off-site processing facility. Intermediate packaging may also be used prior to transporting radioactive materials from their removal area to a final packaging area. The materials handling event presented in Section 3.4.5.5 non-mechanistically assumes that the potential airborne radioactivity in a bounding container or component is released directly to the environment. This event also bounds any packaging event.

If the Reactor Vessel is removed as a single component, the vessel will be packaged in a shipping cask located directly under the Vapor Container Equipment Hatch. The internal and external surfaces of the Reactor Vessel will be coated with a fixative to ensure that contamination on the surfaces of the vessel are stabilized and will not become airborne during placement in the shipping cask. A lifting fixture/cover and nozzle covers will be installed to provide additional barriers to the release of radioactivity. These considerations preclude any significant release of radioactivity during packaging of the Reactor Vessel.

The radiological consequences of a packaging event are bounded by the consequences of a materials handling event. Therefore, the public health and safety are not significantly affected by the potential packaging events that could occur during decommissioning.

3.4.5.4 Storage Events

Containers and components will be stored on-site prior to shipment to either a disposal facility or an off-site processing facility. Intermediate storage locations may also be used before the radioactive materials are moved to a packaging area. Several evaluations are presented in the accident analysis regarding the storage of radioactive materials:

- Fire Events (Section 3.4.7)
- Explosion Events (Section 3.4.8)
- External Events (Section 3.4.9)

Each of these sections presents restrictions to ensure that adequate separation by barrier, distance, or radioactivity content is employed to preclude an event causing a release that exceeds the bounding materials handling event presented in Section 3.4.5.5. The materials handling event presented in Section 3.4.5.5 non-mechanistically assumes the breach of a single container or component, which releases the total potential airborne radioactivity directly to the environment. Storage areas will be located and arranged such that multiple containers or components could not be affected by a single event causing a release of airborne radioactivity that exceeds the bounding materials handling event.

The radiological consequences of a storage event are bounded by the consequences of a material handling event. Therefore, the public health and safety are not significantly affected by the potential storage events that could occur during decommissioning.

3.4.5.5 Materials Handling Events

Materials handling events encompass those events that could potentially occur during movement of radioactive materials from their removal location to a staging location outside of the structure containing the materials. Subsequent handling of these materials is considered by the on-site transportation external event presented in Section 3.4.9.7.

After removal, all openings in components will be covered to minimize the spread of contamination. Components will then either be placed in containers for on-site transportation or be transported individually. The following components containing high potential airborne radioactivity will be handled as single containers or components to reduce the consequences of a materials handling event: Reactor Vessel Casks; Main Coolant System Piping Containers, Valve Containers, and Pumps; Feed and Bleed Heat Exchanger Shells; Vapor Container Bleed Line Piping Containers.

A bounding analysis was completed for the purposes of this accident analysis to estimate the consequences of generating airborne radioactivity resulting from a materials handling event. The dominant system, structure, or component that could cause the highest off-site dose as a result of a release of airborne radioactivity during handling is one of the four Feed and Bleed Heat Exchanger shells. The bounding analysis conservatively assumes that all of the radioactivity on the internal surfaces of a Feed and Bleed Heat Exchanger shell is non-mechanistically released to the environment. If the Reactor Vessel is removed as a single component, a fixative will be applied to the internal surface to stabilize the contamination. This would significantly reduce the amount of radioactivity that could be released during handling.

The estimated radioactivity content of one Feed and Bleed Heat Exchanger shell internal surface contamination is about 9.5 Ci. A release of this material to the environment would result in an off-site dose at the Exclusion Area Boundary of about 0.320 rem. This is significantly less than the EPA PAGs. In addition, radiological scoping analyses indicated that between 50% and 80% of the internal surface contamination is tightly bound. Incorporation of this effect would reduce the dose to less than 0.160 rem. The non-mechanistic release assumes that the heat exchanger shell fails catastrophically, releasing radioactivity from all surfaces to the environment. Realistically, the shell is structurally stable and total failure is highly unlikely, further reducing the release of radioactivity.

The following are the potential off-site doses at the Exclusion Area Boundary for the non-mechanistic release of the radioactivity from high radioactivity containers presented above:

Vapor Container Bleed Line Piping	0.160 rem
Main Coolant Pump (1)	0.100 rem
Main Coolant Pipe Container	0.100 rem
Reactor Vessel Segment Cask	0.078 rem
Main Coolant Valve Container	0.041 rem

The radiological consequences of the bounding materials handling event result in a dose at the Exclusion Area Boundary significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by the potential materials handling events that could occur during decommissioning.

3.4.6 Loss of Support System Events

The electric power, cooling water, and compressed air systems provide support to both spent fuel storage and decommissioning activities. Loss of these systems could potentially affect many systems and plant areas simultaneously. However, none of these support systems are safety related, with the exception of the cooling water supply for spent fuel storage (Reference 3.4-9).

3.4.6.1 Loss of Off-Site Power

Off-site power is used to energize components of the Spent Fuel Pit Cooling System and to energize tools, cranes, lighting, and air filtering equipment used during decommissioning activities. The following results from a loss of off-site power:

- Spent fuel cooling capability is lost. The consequences of a loss of spent fuel cooling capability are presented in Section 3.4.10.

- Decommissioning tools, lighting, and air filtering equipment are de-energized. All decommissioning activities will be terminated.

- Cranes are de-energized and lifting operations are terminated. Cranes fail in a safe condition when they are de-energized.

Back-up power sources will be maintained to support spent fuel cooling requirements, however, back-up power is not needed to support decommissioning activities.

Loss of off-site power will not result in the failure of containment systems designed to preclude the spread of contamination (e.g., local contamination control envelopes, HEPA filters). Although the HEPA filter fans will stop, the filter will remain intact and the contamination control envelope will not be breached, preventing unfiltered releases. A breach of the contamination envelope is an independent event and is not considered in this evaluation. Any significant breach of the contamination envelope would be detected and closed independent of a loss of off-site power.

A loss of off-site power does not result directly in a release of radioactive material to the environment during decommissioning activities. Section 3.4.10 demonstrates that the public health and safety are not affected adversely from a loss of spent fuel cooling capability. Therefore, public health and safety are not adversely affected by a loss of off-site power event.

3.4.6.2 Loss of Cooling Water

Cooling water is supplied by the Service Water System to the Component Cooling Water System (which cools the Spent Fuel Pit Cooling System), the station air compressors, and the decommissioning cutting equipment. The following results from a loss of cooling water:

- Spent fuel cooling capability is lost. The consequences of a loss of spent fuel cooling capability are presented in Section 3.4.10.

- Compressed air is lost if an alternate cooling water supply is not established to the station air compressors within a short period. The consequences of a loss of compressed air are presented in Section 3.4.6.3.

- Cutting operations that use service water will stop. This does not adversely affect contamination control.

- Service water supply to the Vapor Container fire hose reels is lost. If service water is not available, an alternate source of water must be established within one hour in accordance with the Fire Protection Plan (Section 9).

A loss of cooling water does not result directly in a release of radioactive material to the environment during decommissioning activities. Section 3.4.10 and 3.4.6.3 demonstrate that the public health and safety would not be affected adversely from a loss of spent fuel cooling capability and compressed air, respectively. Therefore, public health and safety are not adversely affected by a loss of cooling water event.

3.4.6.3 Loss of Compressed Air

Compressed air is supplied by the station air compressors to operate fuel handling equipment, to operate pneumatic valves and dampers, and to power pneumatic tools. The following occurs following a loss of compressed air:

- Fuel handling equipment fails in a safe condition. Compressed air is required to unlatch a load from the fuel handling tool.

- Pneumatic valves and dampers fail in safe positions. The liquid discharge control valve fails in a closed position following a loss of air, terminating any liquid releases. The ventilation fan dampers fail in a closed position terminating any gaseous releases. Abnormal Operations Procedure OP-3002, "Loss of Station Air Supply," directs an operator to re-establish ventilation by blocking the dampers in an open position if compressed air cannot be re-established to the dampers and the VC Ventilation and Purge System purge fans are available.

- Decommissioning pneumatic tools shut down. This stops any potential releases from these activities.

Decommissioning activities in areas ventilated by the Ventilation System or the VC Ventilation and Purge System will be suspended if ventilation capability is lost (e.g., loss of compressed air to dampers). Any generation of airborne radioactivity that may occur in the interim period before the decommissioning activities are suspended will be small. Decommissioning activities that result in significant airborne releases will be contained within contamination envelopes with HEPA filtration. These systems are not affected by a loss of compressed air.

The radiological consequences of a loss of compressed air event are small and are bounded by the materials handling event presented in Section 3.4.5.5. Therefore, public health and safety are not adversely affected by a loss of compressed air event.

3.4.7 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 levels of the following fire protection features will be maintained through implementation of the Fire Protection Program (Section 9) minimizing the potential of occurrence of a fire:

- Fire detection equipment and systems
- Fire barrier maintenance and control
- Personnel training and qualification programs
- Fire Protection Program procedures
- Control of transient combustible materials and ignition sources

In addition, if a fire occurs, the following fire protection features will be employed to limit the consequences to those of the decommissioning materials handling event presented in Section 3.4.5.5:

- Maintain sufficient fire detection, response, and suppression capability.
- Separate containers, as necessary, by distance, barrier, or radioactivity content.

Higher radioactivity containers are unlikely to fail and cause a significant release due to a fire as these containers are designed for greater levels of integrity. Spontaneous combustion inside a container is highly unlikely as most containers are filled with noncombustible materials. Sea-land containers may be used to ship combustible radioactive materials. The estimated radioactivity level of a sea-land container filled with combustible radioactive material is about 2.9 Ci. Release of all of the radioactivity as a result of a fire would result in an off-site dose of 0.100 rem at the Exclusion Area Boundary. The assumption of a total release of radioactivity is very conservative, Reference 3.4-14 presents a release fraction of 0.00015 for a similar event. Incorporation of this assumption would significantly reduce the release.

Implementation of the Fire Protection Program minimizes the probability of occurrence of a fire. Implementation of the restrictions presented above limits radiological consequences at the Exclusion Area Boundary to a value significantly less than the EPA PAGs. Therefore, public health and safety are not adversely affected by a fire event.

3.4.8 Explosion Events

An explosion event could affect several plant systems, structures, and components simultaneously. Explosions are possible from the following sources:

- Ion exchange resin off-gases
- Explosives
- Inflammable gas storage bottles

Processing and limitation of ion exchange resin off-gases will continue to be controlled by procedures that have successfully precluded this type of accident throughout the greater than 30 year operation of YNPS. Explosives will not be used at YNPS without completion of a separate safety analysis. The analysis must include the effects of the use of explosives both on the Spent Fuel Pit Building and Fuel Transfer Chute structural integrity and on the potential release of airborne radioactivity.

Inflammable gases (e.g., acetylene, LPG) may be used during decommissioning for thermal cutting or to power material handling equipment. Inflammable gas cylinders will be used, located, and stored in quantities such that the possibility of explosion is minimized. Additionally, the quantities of the inflammable gas cylinders will be such that the radiological consequences of an explosion are bounded by the events described below.

3.4.8.1 Explosion Events: Vapor Container

Inflammable gases may be used in the Vapor Container during thermal cutting activities. The decommissioning plan recommends mechanical cutting of significantly contaminated systems, structures, and components to minimize the generation of airborne radioactivity. However, thermal cutting methods utilizing inflammable gases could be used for dismantlement activities on components with lower contamination levels.

An engineering evaluation was completed to determine the physical consequences of an explosion in the Vapor Container (Reference 3.4-10). If the Vapor Container is closed, the peak pressure resulting from an explosion of a standard cylinder of acetylene is less than 5% of the Vapor Container design pressure. If the Vapor Container is not isolated (e.g., Equipment Hatch open), about 10% of the Vapor Container air mass could be released through existing openings before the pressure inside the Vapor Container equalizes with the outside atmospheric pressure. It is assumed that radiological contamination controls are re-established after the explosion by closing Vapor Container openings to terminate the release to the environment.

The explosion evaluation assumes that all of the energy from the gas explosion is converted into a pressure increase. This is a highly conservative assumption, most of the energy would be lost to other effects (e.g., heat generation, mechanical deformation). Less than 5% of the energy most likely would be converted into the pressure increase.

If an explosion non-mechanistically releases all of the internal surface contamination contained in systems inside the Vapor Container to the Vapor Container atmosphere, the release to the environment would not exceed the EPA PAGs at the Exclusion Area Boundary. The radioactivity content of the internal surface contamination in systems inside the Vapor Container is less than 130 Ci. Release of 10% of this material to the environment would result in a dose at the Exclusion Area Boundary of about 0.440 rem.

It is highly unlikely that all systems either will be opened or will be in the Vapor Container such that an explosion could remove the internal radioactivity. In addition, a significant fraction of the internal surface contamination is tightly bound. Radiological scoping analyses indicated that this fraction varied between 50% and 80%. The combination of these effects would reduce the release by 70% to 90%, resulting in an off-site dose of less than 0.100 rem.

In order to minimize the overall risk of an inflammable gas-air explosion in the Vapor Container, the Vapor Container should not be used as a general storage location for inflammable gas cylinders. Only cylinders in use or required in the near term should be located inside the structure at any given time. Additionally thermal cutting methods using inflammable gases will not be used to dismantle the following components with high contamination levels: Reactor Vessel, Main Coolant System, Feed and Bleed Heat Exchanger, Vapor Container Bleed Line Piping.

The consequences of an explosion in the Vapor Container are significantly less than the EPA PAGs. Therefore, there is no significant impact on pubic health and safety.

3.4.8.2 Explosion Events: Potentially Contaminated Area Warehouse

Inflammable gases may be used in the Potentially Contaminated Area Warehouse to power forklift trucks and to support thermal cutting activities. In the unlikely event that a single inflammable gas cylinder explodes in the warehouse, the pressure would be in excess of the capacity of the warehouse if the structure is sealed. Doors, portions of the roof, or portions of the walls would likely fail in order to relieve excess pressure. Explosion released overpressure would not cause the containers to explode since the explosion is external to the containers. Some damage of the containers would be likely, but a release of significant portions of the contents of the containers as airborne radioactivity is not likely. In addition, the segregation and separation practices implemented for fire protection purposes would also limit the effects on storage containers.

In order to minimize the overall risk of an inflammable gas-air explosion in the Potentially Contaminated Area Warehouse, the warehouse should not be used as a general storage location for inflammable gas cylinders. Only cylinders in use or required in the near term should be located inside the structure at any given time.

The consequences of an explosion in the Potentially Contaminated Area Warehouse are bounded by those of the decommissioning materials handling events presented in Section 3.4.5.5. Therefore, there is no significant impact on pubic health and safety.

3.4.9 External Events

A systematic assessment of external events was made to evaluate the effects of natural and manmade events on decommissioning activities. The hazards associated with these events are assumed to be consistent with those that could have occurred while YNPS was in operation. Seven external events were identified as having potential applicability to the YNPS decommissioning based on a review of the natural and manmade external events presented in NUREG/CR-2300 (Reference 3.4-11).

3.4.9.1 Aircraft Impact

YAEC evaluated the significance of a potential aircraft hazard on YNPS in response to Systematic Evaluation Program Topic III-4 (Reference 3.4-17). The analysis concluded that the annual probability of an aircraft impact was very low. Further consideration of the interaction between an aircraft impact and decommissioning is not warranted.

3.4.9.2 Earthquake

A seismic event during decommissioning could initiate a materials handling event similar to those described in Section 3.4.5.5. The analysis in Section 3.4.5.5 concludes that the bounding material handling event results in an off-site dose that is significantly less than the EPA PAGs. In addition, detailed planning of dismantlement activities will consider the impact of seismic events on components that are affected by removal activities. These components will be evaluated and physically supported, as appropriate, to limit the off-site dose resulting from a release of radioactivity to less than the EPA PAGs.

The Spent Fuel Pit, Fuel Transfer Chute, and the overhead crane structure were analyzed for seismic considerations and found acceptable as part of the Systematic Evaluation Program. All structures whose failure during a seismic event that could significantly affect either the Spent Fuel Pit structural integrity or the spent fuel integrity are seismically qualified. The impact of seismic events will be evaluated during planning of dismantlement activities in proximity of the Spent Fuel Pit. The purpose of the evaluation is to ensure that partial dismantling of equipment and structures does not result in a configuration that could fail during a seismic event, subsequently collapsing onto or into the Spent Fuel Pit or Fuel Transfer Chute.

The potential for radiological consequences from a seismic event resulting in a dose at the Exclusion Area Boundary greater than the EPA PAGs is extremely low. In the unlikely event that a materials handling event is initiated, the consequences would be significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by a potential seismic event during decommissioning.

3.4.9.3 External Flooding

A flooding event at YNPS typically would be preceded by a sufficient warning period to prepare the site for the event by securing decommissioning activities. Most of the potentially removable radioactivity at YNPS is located in the Vapor Container, well above the potential flood height. Most of the balance of contaminated systems, structures, and components would either be packaged for shipment or secured inside buildings. Containers that hold high radioactivity materials are designed for greater levels of structural integrity, providing additional protection. In the unlikely event that a lower radioactivity container or component is exposed to flood waters and radioactive material is dispersed, the flooding dilution effect results in a radiological consequence significantly less than an airborne release of a similar amount of radioactivity.

Flooding could initiate a loss of off-site power event. The analysis in Section 3.4.6.1 concludes that public health and safety are not adversely affected from a loss of off-site power event.

Flooding events at YNPS do not result in a significant radiological release, therefore, public health and safety are not adversely affected.

3.4.9.4 Tornadoes and Extreme Winds

The annual strike probability of a tornado that could cause a significant release of radioactivity from a container or component is very low. In addition, most components and containers that would be vulnerable to a tornado will be packaged awaiting shipment. The integrity of these containers would limit the probability and consequences of a significant release of radioactivity. Further consideration of the interaction between a tornado and decommissioning is not warranted.

An extreme winds event (i.e., hurricanes above 74 mph, thunderstorms above 100 mph) at YNPS typically would be preceded by a sufficient warning period to prepare the site for the event by securing decommissioning activities. Most of the potentially airborne radioactivity at YNPS is located in the Vapor Container, which protects the components from the effects of extreme winds and associated missiles. Most of the balance of contaminated systems, structures, and components would either be packaged for shipment or secured inside buildings with system integrity, providing protection from the extreme winds and associated missiles.

Containers that hold higher radioactivity materials are designed for greater levels of structural integrity, providing additional protection. In the unlikely event that a lower radioactivity container or component is unprotected and is exposed to extreme winds and radioactive material is dispersed, the combination of low radioactivity content and significant dispersion by the wind would result in an off-site dose that is bounded by the limiting release of the material handling event presented in Section 3.4.5.5.

Tornadoes or extreme winds could initiate a loss of off-site power event. The analysis in Section 3.4.6.1 concludes that public health and safety are not adversely affected from a loss of off-site power event.

Extreme tornadoes and winds events at YNPS will not result in a significant radiological release, therefore, public health and safety are not adversely affected.

3.4.9.5 Forest Fire

A forest fire event at YNPS typically would be preceded by a sufficient warning period to prepare the site for the event by securing decommissioning activities and closing Vapor Container openings, if necessary. In addition, YNPS is protected from the effects of a forest fire by a buffer zone surrounding the facility. The probability of a forest fire of sufficient intensity to bridge the zone is very low (Reference 3.4-15).

In the unlikely event that a forest fire causes an on-site fire, the fire protection features described in Section 3.4.7 would be sufficient to mitigate the consequences of the fire. A forest fire could also initiate a loss of off-site power event. The analysis in Section 3.4.6.1 concludes that public health and safety are not adversely affected from a loss of off-site power event.

Forest fire events at YNPS will not result in a significant radiological release, therefore, public health and safety are not adversely affected.

3.4.9.6 Lightning Event

The lightning strike annual probability for a decommissioning activity or single exposed container or component is very low. In addition, YNPS structures provide protection against lightning. Although the effects of lightning generally are localized, a lightning strike could initiate a loss of off-site power event or a fire. The analyses in Sections

3.4.6.1 and 3.4.7 conclude that public health and safety are not adversely affected from a loss of off-site power event or fire event. Further consideration of the interaction between decommissioning and a lightning event are not warranted.

3.4.9.7 On-site Transportation Accidents

On-site transportation accident events consist of those events occurring after removal of a container or components from a structure, but, before final packaging for shipment. These events could occur during transportation of a container or component to a packaging area.

Detailed planning will ensure that only one container or component from the following components with high potential airborne radioactivity is transported simultaneously by the same vehicle until they are prepared for final shipment: Reactor Vessel Casks; Main Coolant System Piping Containers, Valve Containers, and Pumps; Feed and Bleed Heat Exchanger Shells; Vapor Container Bleed Line Piping Containers. This constraint ensures that the off-site dose resulting from an on-site transportation event is bounded by the materials handling event presented in Section 3.4.5.5. Additionally, detailed planning will limit the maximum number of containers transported on-site to ensure that they are also bounded by this analysis.

Containers and components stored in materials storage areas awaiting final packaging will be located to preclude impact from a runaway vehicle. The location selection will consider separation by distance, barrier, or radioactivity to ensure that the off-site dose resulting from an impact is bounded by the dose resulting from the materials handling event presented in Section 3.4.5.5.

If the Reactor Vessel is removed as a single component, it will be transported in a cask in close proximity of the Spent Fuel Pit. Impact of the Reactor Vessel cask or the transporter with the Spent Fuel Pit could challenge the structural integrity of the Spent Fuel Pit. The following events associated with movement of the Reactor Vessel Cask were considered:

- Transporter Failure From Flat Tires - Failure of the tires on one side of the transporter could affect load stability. However, a flat tire would be isolated to one tire on an axle that has several other tires. The loss of one tire would not cause the Reactor Vessel cask to tilt such that the load becomes unstable. Common mode failure of all tires on one side of the transporter is not credible.

- Transporter Failure From a Loss of Hydraulic Compensation - Two parallel hydraulic compensation trains extend the full length of the transporter. A failure of one train would cause the transporter flatbed to tilt. Limits will be placed on the maximum elevation of the flatbed above its minimum elevation (zero pressure configuration) during travel near the Spent Fuel Pit to ensure that the load remains stable.

- Transporter Collision With Spent Fuel Pit Wall - A collision of the transporter into the Spent Fuel Pit wall could adversely affect the structural integrity of the wall. Limits will be placed on the speed of the transporter to limit the kinetic energy of impact to a level ensuring that the wall is not damaged structurally.

The radiological consequences of an on-site transportation event are bounded by a decommissioning materials handling event. The physical consequences of an on-site transportation event does not adversely affect the Spent Fuel Pit structural integrity. Therefore, public health and safety are not adversely affected by an on-site transportation event.

3.4.10 Spent Fuel Storage Events

This section evaluates potential accident scenarios that could impact spent fuel storage during decommissioning. The events below correspond to storage of spent fuel in the Spent Fuel Pit. If fuel is stored in an on-site dry storage facility, the facility will be located such that it is a safe distance from decommissioning activities, precluding any significant interactions.

3.4.10.1 Fuel Handling Event

YAEC re-evaluated the Spent Fuel Pit fuel handling event during preparation for the transition to a possession only license (Reference 3.4-2). The analysis demonstrated that the release resulting from a fuel handling event would be significantly less than the EPA PAGs. This analysis bounds any fuel handling event that could occur during decommissioning. The NRC performed an independent review of the analysis, confirming the results (Reference 3.4-12). The results of the analysis are summarized below.

The fuel handling accident in the Spent Fuel Pit assumes that a fuel assembly is dropped, strikes the floor, rotates towards the horizontal, and strikes a sharp object in the most vulnerable area. The analysis applied the available kinetic energy as a linear point force to the fuel pins and assumed that the pins would fail when 1% of the yield stress is exceeded. No credit was taken for the fuel pellets within the fuel pins (i.e., the pins are assumed to be hollow). For this case, approximately 6 rows (100 pins) were estimated to fail. However, for the radiological analysis, it was conservatively assumed that all of the fuel pins in one assembly failed.

The fuel handling accident off-site dose at the Exclusion Area Boundary is less than 0.001 rem. The radionuclide inventory used in the estimate was based on the worst case inventory on May 31, 1992. The worst case radionuclide inventory will continue to decrease due to radioactive decay.

Movement of fuel from the Spent Fuel Pit to either an on-site or off-site fuel storage facility is not bounded by this analysis. A separate safety analysis is needed to ensure that there are no unacceptable consequences during movement from the Spent Fuel Pit. This analysis must also include any interactions between fuel movement and decommissioning activities. The analysis will be completed before movement of fuel out of the Spent Fuel Pit.

The radiological consequences of the fuel handling accident result in a dose at the Exclusion Area Boundary significantly less than the EPA PAGs. Therefore, the public health and safety are not significantly affected by a fuel handling accident.

3.4.10.2 Loss of Spent Fuel Cooling Capability

Although spent fuel cooling capability must be maintained, the consequences of a loss of that capability are not severe. As of January 1994, greater than four weeks must elapse without re-establishing cooling or adding make-up water before the water remaining in the Spent Fuel Pit is insufficient to provide adequate shielding. A make-up water flow rate of about 1 gpm is needed to replace water lost through evaporation.

Several diverse sources of make-up are available including demineralized water, fire water, service water, or Sherman Reservoir water. Water may be injected to the Spent Fuel Pit through installed or portable pumps as well as gravity feed. Adequate time is available either to re-establish Spent Fuel Pit cooling or to provide make-up water to maintain Spent Fuel Pit inventory.

Although scenarios that result in a loss of spent fuel cooling capability can be mitigated without any significant consequence, cooling capability will be protected during decommissioning. Decommissioning activities near and around the Spent Fuel Pit Cooling System and other support systems will be controlled to prevent damage to these systems. This may be accomplished by physically protecting the systems or by establishing safe load paths and protective zones around the systems.

The time available to respond to a loss of spent fuel cooling capability is sufficient to ensure that the event is terminated precluding any impact on the public health and safety. Implementation of the restrictions presented above will reduce the probability of occurrence of this event.

3.4.10.3 Interactions Between Spent Fuel and Decommissioning

YAEC has evaluated the option to store fuel in the Spent Fuel Pit during the decommissioning dismantlement phase (Reference 3.4-13). The evaluation identified safety considerations and limitations on decommissioning activities associated with operating the Spent Fuel Pit concurrent with dismantlement activities.

Potential dismantlement activities are limited when spent fuel is stored in the Spent Fuel Pit. Activities cannot be implemented that could result in the loss of Spent Fuel Pit integrity or could cause physical damage to the fuel that would reduce subcriticality margin or adversely affect the ability to cool the fuel.

Adverse spent fuel and decommissioning interactions can be precluded by incorporating the following restrictions:

- Delay dismantlement of the Ion Exchange Pit until after the fuel is removed permanently from the Spent Fuel Pit

- Limit cask usage over the Spent Fuel Pit to those activities that have acceptable cask drop consequences. Technical Specification 3.2 currently limits cask usage in and over the Spent Fuel Pit to a shipping cask weighing less than 35 tons.

- Isolate the Fuel Transfer Chute from the Spent Fuel Pit by capping the chute inside the Spent Fuel Pit and filling the chute with structurally stable material between the cap and the lower lock valve. This must be completed before initiating activities that could adversely interact with the Fuel Transfer Chute.

- Ensure that detailed work planning excludes activities that could result in a drop of a heavy load onto or into the Spent Fuel Pit or Fuel Transfer Chute. Ensure that partial dismantling of equipment and structures does not result in a configuration that could result in failure during an external event (e.g., seismic event) and subsequent collapse onto or into the Spent Fuel Pit or Fuel Transfer Chute. Alternatively, consider modifications to protect the Spent Fuel Pit and the Fuel Transfer Chute from heavy load drops. Technical Specification 3.2 limits movement of loads over the Spent Fuel Pit to those less than 900 lb.

- Ensure that demolition explosives that could affect the Spent Fuel Pit structural integrity are not permitted for use until either fuel is removed permanently from the Spent Fuel Pit or an analysis of the impact of explosives on the Spent Fuel Pit structure is completed.

Incorporation of the restrictions presented above is sufficient to preclude significant events resulting from interactions between fuel storage and decommissioning activities. There is no significant affect on public health and safety.

3.4.11 Summary of Results

The accident analysis assessed the impact of decommissioning on both occupational and public health and safety by considering decommissioning and fuel storage events. The evaluation of events that could affect occupational health and safety indicated that implementation of the Radiation Protection and Occupational Safety Programs and Defueled Emergency Plan ensures that these events are sufficiently minimized. Analysis of events that could affect public health and safety indicated that there were no events that could significantly affect public health and safety.

The analysis presented above identified requirements that must be implemented during decommissioning to ensure that the accident analysis basis is maintained. Table 3.4-3 summarizes these items and provides references for the sections of the decommissioning plan. The analysis also presents planning considerations that reduce the probability of occurrence or consequences of the events that were evaluated. Table 3.4-4 summarizes these items and provides references for the sections of the decommissioning plan.

REFERENCES

3.4-1 YRP 437/93, Decommissioning Safety Analysis, September 28, 1993.

3.4-2 BYR 92-057, Request for Exemption From Annual and Biennial Emergency Preparedness Exercise in 1992, S. P. Schultz to M. B. Fairtile (USNRC), May 22, 1992.

3.4-3 NYR 92-178, Exemption From The Emergency Preparedness Rule 10 CFR 50.54(q) and Approval of The Defueled Emergency Plan At The Yankee Nuclear Power Station (TAC No. M83991), M. B. Fairtile (USNRC) to J. M. Grant, October 30, 1992.

3.4-4 EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents, US Environmental Protection Agency, October 1991.

3.4-5 YRC-182, Yankee Rowe Accident Atmospheric Diffusion Factors, January 5, 1982.

3.4-6 YRC-1024, Basis for the Radiological Status of Plant Systems and Structures, September 1993.

3.4-7 ELISA - Technical Description, A Computer Code for the Radiological Evaluation of Licensing and Severe Accidents at Light-Water Nuclear Power Plants, J. N. Hamawi, April 1991.

3.4-8 YRC-1014 Revision 1, Decommissioning Accident Source Terms.

3.4-9 Safety Classification of Systems Manual, Revision 2.

3.4-10 ESG 90/93, Flammable Gas Explosion Evaluation For Decommissioning Safety Analysis, G. A. Harper to R. A. Mellor, August 20, 1993.

3.4-11 NUREG/CR-2300, PRA Procedures Guide, January 1983.

3.4-12 NYR 92-144, Exemption From 10 CFR Part 50 - Appendix E - Emergency Preparedness Training Exercises at the Yankee Nuclear Power Station (TAC No. M83415), M. B. Fairtile (USNRC) to J. M. Grant, July 24, 1992.

3.4-13 YRP 303/93, Impact of Wet Spent Fuel Storage on Decommissioning, P. A. Rainey to R. A. Mellor, September 27, 1993.

3.4-14 NUREG/CR-130, Technology, Safety, and Costs of Decommissioning a Reference Pressurized Water Reactor Power Station, Battelle Pacific Northwest Laboratory, June 1978.

3.4-15 YAEC-1711, Yankee Nuclear Power Station Severe Accident Closure Submittal, December 21, 1989.

3.4-16 NYR 92-178, Exemption From The Emergency Preparedness Rule 10 CFR 50.54(q) and Approval of the Defueled Emergency Plan at the Yankee Nuclear Power Station (TAC No. M83991), M. B. Fairtile (USNRC) to J. M. Grant, October 30, 1992.

3.4-17 NUREG-0825, Integrated Plant Safety Assessment Systematic Evaluation Program, Yankee Nuclear Power Station, June 1983.

PROCEDURE REFERENCES

OP-3002 Loss of Station Air Supply

TABLE 3.4-1

EXCLUSION AREA BOUNDARY DOSE PER CURIE CONVERSION FACTORS
(BASIS: 1/94)

	Dose Conversion Factors, rem/Ci		
	TEDE	Thyroid	Skin
Contamination:			
MCS and Bleed Line in VC	3.4E-02	1.4E-03	8.6E-05
Remaining Systems and Structures	2.4E-02	7.8E-04	4.9E-05
Activation:			
Bioshield Concrete	1.9E-03	4.1E-04	2.5E-05
Vessel Cladding	9.9E-03	2.8E-03	1.6E-04
Vessel Wall	2.1E-03	6.2E-04	3.4E-05

Notes:

1. TEDE is total effective dose equivalent.

2. Activation results are based on the average value in the core region.

TABLE 3.4-2

SUMMARY OF MATERIALS RELEASED IN POSTULATED EVENTS

Item	Radioactivity Content (1)	Source of Radioactivity
Main Coolant System:		
Piping (1 Loop)	11 Ci	Internal Contamination
Valves (1 Loop)	1.2 Ci	Internal Contamination
Pumps (1 Pump)	3.0 Ci	Internal Contamination
Charging & Volume Control System:		
Bleed Line Piping (Total in VC)	4.6 Ci	Internal Contamination
Feed & Bleed Heat Exchanger (1 Shell)	9.5 Ci	Internal Contamination
Reactor Vessel:		
Intact Removal (Internal Contamination)	23 Ci	Internal Contamination
Segmented Removal (Cutting Debris)	120 Ci	Activated Metal
Radioactive Materials Container:		
Main Coolant System Piping Container	2.9 Ci	Internal Contamination
Reactor Vessel Segment Cask	2.3 Ci	Internal Contamination
Sea-Land Container With Combustible Material	2.9 Ci	Internal Contamination

Note:

1. Radioactivity content is based on January 1994.

TABLE 3.4-3

SUMMARY OF SAFETY ANALYSIS REQUIREMENTS

Accident Analysis Section	Analysis Requirement	Plan Section Reference
3.4.1.2	Special Nuclear Material used as reactor fuel will not be moved into the Reactor Vessel.	1.2.1.1
3.4.1.2	Activities initiated before approval of the decommissioning plan will be evaluated prior to implementation.	1.5
3.4.1.2	The capability to isolate the Vapor Container will be retained to mitigate the consequences of a significant radioactive release. If an accident occurs, the Vapor Container will be isolated expeditiously.	2.3.4.3
3.4.5.1	Detailed planning of activities that use liquids will ensure that contaminated liquids will be processed by the liquid waste processing system. Existing or supplemental barriers will be used to ensure that inadvertent spills are contained within the liquid waste processing system.	2.3.4.3
3.4.5.1 3.4.5.3 3.4.9.7	The following will be performed if the Reactor Vessel is removed as a single component: • A fixative will be applied to the internal surface to stabilize the contamination. • Limitations will be placed on the Reactor Vessel cask transporter operation to ensure that Spent Fuel Pit structural integrity is not adversely affected.	2.3.5.2
3.4.5.4 3.4.8.2 3.4.9.7	Radioactive materials storage areas will be located and arranged such that multiple containers or components will not be significantly affected by a fire event causing a release of airborne radioactivity exceeding the bounding materials handling event.	3.3.2.2
3.4.5.5 3.4.9.7	The following components will be handled and transported on-site as single containers or components to reduce the consequences of a materials handling event: Reactor Vessel Casks; Main Coolant System Piping Containers, Valve Containers, and Pumps; Feed and Bleed Heat Exchanger Shells; Bleed Line Piping Containers.	2.3.5.2 2.3.5.3 2.3.5.5

TABLE 3.4-3 (CONTINUED)

Accident Analysis Section	Analysis Requirement	Plan Section Reference
3.4.8	Explosives will not be used at YNPS without completion of a separate safety analysis. The analysis must include the effects of the use of explosives both on the Spent Fuel Pit Building and Fuel Transfer Chute structural integrity and on the potential release of airborne radioactivity.	2.3.4.3
3.4.8.1	The quantity of inflammable gases stored on-site will be limited as follows: • The Vapor Container and Potentially Contaminated Area Warehouse will not be used as a general storage location for inflammable gas cylinders. Only cylinders in use or required in the near term should be located inside the structures at any given time. • Acetylene cylinders used in the Vapor Container will be limited to standard size cylinders.	2.3.4.3
3.4.10	If fuel is stored in an on-site dry storage facility, the facility will be located such that it is a safe distance from decommissioning activities, precluding any significant interactions.	3.3.1.3
3.4.10.1	A separate safety analysis will be completed before movement of fuel from the Spent Fuel Pit to ensure that there are no unacceptable interactions between fuel movement and decommissioning activities.	3.3.1.3
3.4.10.2	Decommissioning activities near and around the Spent Fuel Pit Cooling System and other support systems will be controlled to prevent damage to these systems.	2.3.4.3
3.4.10.3	Activities will not be implemented that could result in the loss of Spent Fuel Pit integrity or could cause physical damage to the fuel that would reduce subcriticality margin or adversely affect the ability to cool the fuel.	3.3.1.3

TABLE 3.4-4

SUMMARY OF DECOMMISSIONING PLANNING CONSIDERATIONS FROM THE SAFETY ANALYSIS

Accident Analysis Section	Decommissioning Planning Considerations	Plan Section Reference
3.4.3.1	The Radiation Protection Program will be applied to all activities performed on site involving radioactive materials: • Activities will be managed by qualified individuals who will implement program requirements in accordance with established procedures. • Radiation exposures and the release of radioactive materials to unrestricted areas will be maintained as far below specified limits as is reasonably achievable. • Radiation protection training will be provided to all occupationally exposed individuals.	3.2.2
3.4.3.1	Project management will ensure that work specifications, designs, and work packages involving radiation exposure or radioactive materials incorporate effective radiological controls.	2.3.4.2
3.4.3.2	The Occupational Safety Program will be implemented during YNPS decommissioning.	3.5
3.4.4.2	Hazardous materials handling will be controlled through the Non-Radioactive Hazardous Materials Program and the Chemical Control Program.	3.6
3.4.4.2	Safe storage and use of inflammable gases and any other inflammable materials will be controlled through the Occupational Safety Program and the Fire Protection Program.	3.5 9.1
3.4.5.1 3.4.5.3 3.4.9.7	The following will be performed if the Reactor Vessel is removed as a single component: • A local HEPA filtration unit will be used as necessary to remove airborne radioactivity. • A lifting fixture/cover and nozzle covers will be installed to provide additional barriers to the release of radioactivity.	2.3.5.2

TABLE 3.4-4 (CONTINUED)

Accident Analysis Section	Decommissioning Planning Considerations	Plan Section Reference
3.4.5.2	Detailed planning will ensure that the following systems with high internal contamination will be dismantled using mechanical methods: Main Coolant System, Feed and Bleed Heat Exchanger, Bleed Line Piping.	2.3.5.3 2.3.5.5
3.4.5.2	The following will be performed if the Reactor Vessel is segmented: • Mechanical cutting methods will be implemented to cut the Reactor Vessel. • Cutting of the Reactor Vessel beltline area will be conducted under water. • A local HEPA filtration unit will be used to remove airborne radioactivity.	2.3.5.2
3.4.5.5	Openings in components will be covered and sealed to minimize the spread of contamination after dismantlement and before on-site transportation.	2.3.4.3
3.4.6.1	Back-up power sources will be maintained to support spent fuel cooling requirements.	2.2.30
3.4.6.2	If service water is not available to the Vapor Container fire hose reels, an alternate source of water will be re-established in accordance with the Fire Protection Plan.	9.1
3.4.6.3	Decommissioning activities in areas ventilated by the VC Ventilation and Purge System or the Ventilation System will be suspended if ventilation capability is lost.	2.3.4.3
3.4.7	The following fire protection features will be maintained through implementation of the Fire Protection Program: • Fire detection equipment and systems • Fire barrier maintenance and control • Personnel training and qualification programs • Fire Protection Program procedures • Control of transient combustible materials and ignition sources	9.1

TABLE 3.4-4 (CONTINUED)

Accident Analysis Section	Decommissioning Planning Considerations	Plan Section Reference
3.4.9.2	The impact of seismic events will be evaluated during planning of dismantlement activities. The evaluation will consider the possibility of the spread of contamination and of the impact on Spent Fuel Pit structural integrity.	2.3.4.3

3.5 OCCUPATIONAL SAFETY PROGRAM

3.5.1 Introduction

This section provides an overview of the Yankee Occupational Safety Program as provided in the Yankee Safety Manual and applicable plant procedures.

3.5.2 Management Policy Statement

YAEC and its management are committed to the safe decommissioning of YNPS. The primary objective of the Occupational Safety Program is to protect workers and visitors from industrial hazards that have the potential of developing during decommissioning activities. YAEC and its contractors will provide sufficient qualified staff, facilities, and equipment to perform decommissioning in a safe and effective manner. YAEC is committed to compliance with federal and state requirements and to the guidance provided through industry standards and good work practices.

3.5.3 Health and Safety Organization and Functions

The existing Occupational Safety Program provides the basis for controlling safety during decommissioning activities. The purpose of the health and safety organization is to ensure that the standards of safety are maintained through effective implementation of the Occupational Safety Program. The effective implementation of the Occupational Safety Program is the responsibility of all decommissioning personnel:

- Plant Superintendent - The Plant Superintendent has the overall responsibility for safe operation of the plant and has control over those on-site resources necessary to meet this objective (TS 6.2.1.c). Included in this is the responsibility for assuring effective implementation of the Occupational Safety Program and assuring that all organizations involved with decommissioning are coordinated to achieve the goals of providing a safe work place and the reduction of industrial hazards.

- Health and Safety Supervisor - The Health and Safety Supervisor is responsible for the development and implementation of the Occupational Safety Program policies and standards. The Health and Safety Supervisor has the authority to cease any work activity when worker safety is jeopardized or an unsafe condition occurs.

- Health and Safety Personnel - Health and Safety personnel report to the Health and Safety Supervisor and are responsible for the day to day function of the organization. Health and Safety Personnel have the authority to cease any work activity when worker safety is jeopardized or an unsafe condition occurs.

- Decommissioning Supervisors - All supervisory personnel are responsible for the supervision and direction of safety practices during decommissioning activities.

- Decommissioning Workers - All plant and decommissioning workers are responsible for their own safe work practices as presented in the Safety Manual and OSHA Standards.

3.5.4 Yankee Occupational Safety Program

The Yankee Occupational Safety Program was developed to establish and maintain a safe work place for YAEC workers, contractors, and visitors. The program provides guidelines and procedures to be used to reduce industrial hazards and risks.

The Yankee Safety Manual provides guidelines and requirements which will be incorporated into the detailed decommissioning planning process.

The following areas are discussed in the manual:

- Personnel protection and safety equipment
- Prevention of falls
- Safe use of ladders and scaffolding
- Safe handling of hazardous substances and materials
- Safe use of hand and portable powered tools and equipment
- Welding, cutting and brazing safety
- Electrical safety
- Confined space safety requirements
- Heat stress prevention

Additional safety guidelines and instructions are included in plant procedures which receive a Health and Safety Organization review during the approval process. Health and Safety will review Decommissioning Work Packages prior to commencement of work activities (Section 2.3.4.2).

3.5.5 Safety Training and Meetings

Safety training is conducted as part of the General Employee Training process and during routine safety meetings. The safety meetings focus on current safety issues and events as well as providing a forum for workers to ask questions and provide feedback.

3.6 NON-RADIOACTIVE WASTE MANAGEMENT

3.6.1 Introduction

This section provides an overview of the Yankee Non-Radioactive Waste Management Program and applicable plant procedures.

Site materials that are routinely handled as hazardous waste during disposal are fuel oil, lubricating oil, 1,1,1-trichloroethane, and any absorbent materials used with these items. Non-routine wastes such as oil containing polychlorinated biphenyls (PCBs), laboratory chemicals, mercury, paint, and battery acid are also handled and disposed of through this program.

3.6.2 Management Policy Statement

YAEC and its management are committed to the safe decommissioning of YNPS. The primary objective of the Yankee Non-Radioactive Waste Management Program is to protect workers, visitors, and the environment from the potential effects of hazardous materials. YAEC is committed to strict compliance with all federal and state hazardous waste handling and disposal requirements.

3.6.3 Hazardous Material Management

YAEC is required by the OSHA Hazard Communication Standard (29 CFR 1910.1200) to provide information to its employees and contractors concerning the hazardous substances to which they may be exposed. Plant Procedures AP-0043, "Hazard Communication Program" and AP-0031, "Control of Chemicals" were implemented to meet these requirements. General Employee Training was revised to apprise employees and contractors of hazardous materials used at YNPS.

3.6.4 Hazardous Waste Management

The Yankee Non-Radioactive Waste Management Program was established to assure compliance with all the federal and state hazardous waste regulatory requirements. Non-radioactive hazardous wastes from YNPS are transported only by authorized and licensed transporters and shipped only to authorized and licensed facilities.

The program is implemented through Plant Procedure AP-0030, "Non-Radioactive Hazardous Waste Management." This procedure provides direction for the handling, temporary storage, and preparation for shipment of non-radioactive hazardous waste. Routine preventive and emergency response procedures have been developed for precluding and containing hazardous material incidents.

The following are the hazardous materials procedures:

- OP-3011, "Oil Spill Incident"
- OP-3012, "Hazardous Waste, Asbestos, or Chemical Spill"
- OP-4236, "Weekly Oil Spill Prevention/Hazardous Waste Storage Inspection"
- OP-MEMO 2DD-2, "YAEC Spill Prevention Control and Countermeasure Plan"

3.6.4.1 Above Ground Fuel Oil Storage Tanks

There are six above-ground oil storage tanks at YNPS. The Fuel Oil Storage Tank has a capacity of 30,450 gallons and supplies oil to the auxiliary boilers and the Emergency Diesel Generator Day Tanks. The three Emergency Diesel Generator Day Tanks have a capacity of 275 gallons each. The Diesel Fuel Oil Overflow Tank, which collects any overflow from the Emergency Diesel Generator Day Tanks, has a capacity of 250 gallons. The Diesel Powered Fire Pump tank has a capacity of 275 gallons. These tanks will remain in service throughout the safe storage period and into the dismantlement phase. When no longer required, the tanks will be emptied, cleaned, and disposed of by an authorized and licensed contractor

3.6.4.2 Underground Fuel Oil Storage Tanks

Two underground fuel oil storage tanks are located at the YNPS. One supplied fuel oil to the Safe Shutdown System Emergency Diesel Generator and the other supplied the Security Diesel Generator. Both tanks are no longer in service and have been emptied. The tanks will be excavated, removed and disposed of by an authorized and licensed contractor.

3.6.4.3 Underground Waste Oil Tank

The Underground Waste Oil Tank was used to store uncontaminated waste oil prior to shipment to a licensed disposal facility. The tank has been removed from service, emptied and cleaned. Uncontaminated waste oil is and will continue to be stored in drums in the Lube Oil Storage Room until disposal. The tank will be excavated, removed and disposed of by a licensed contractor.

3.6.4.4 PCB Contaminated Transformer Oil

PCBs are present in the transformer oil in the No. 4, 5, and 6 Station Service Transformers. Each transformer contains approximately 370 gallons of oil. This oil will be removed from the transformers as they are removed from service. The oil will be processed and disposed of by an authorized and licensed contractor.

3.6.4.5 Mercury

Mercury contained in instruments and switches will be removed and collected prior to final disposal of the equipment. The mercury will be reclaimed or processed by an authorized and licensed contractor.

3.6.4.6 Asbestos Containing Materials

The original plant insulation, contains 6-8% asbestos with a calcium silicate binder. Asbestos insulating material has been identified on many plant systems and in most areas and buildings. Asbestos insulating material was replaced with non-asbestos material as a part of maintenance activities that required insulation removal and labeled accordingly. Most of the systems originally covered with asbestos insulating material now have portions which are covered with non-asbestos insulating material. Minor quantities of non-insulation asbestos containing materials, in the form of gaskets and packing, remain present in numerous systems at the plant.

Asbestos material will be removed by a licensed contractor prior to the start of dismantlement activities. Insulating material will be considered to be asbestos material unless marked "NON-ASBESTOS". All asbestos containing materials will be removed and processed in accordance with plant procedure AP-0709, "Asbestos Control Program," which assures compliance with federal and state regulations.

Radiologically contaminated asbestos containing material and non-asbestos material will be disposed of in accordance with the requirements Section 3.3 of this plan.

3.6.4.7 Lead Based Paints

Lead based paints were used at YNPS to coat many steel components, some concrete structures, and underground carbon steel piping. During the operating life of the plant, some of the lead based paints have been covered with several coats of non-lead based paint. In other cases, non-lead based painted surfaces have been coated or touched up with a lead based paint. Controls of the lead based paint identification and removal process will be implemented to ensure proper handling of lead materials. Steel components coated with lead based paint will be sent to a reprocessing facility. The lead based paints on non-recyclable components (e.g., concrete) will be removed, processed, and disposed of by an authorized and licensed contractor.

3.6.5 Sampling and Remedial Actions

The areas which contained non-radioactive hazardous materials will be evaluated by appropriate sampling and analytical protocols. Remedial actions will be implemented as necessary to meet all federal, state, and local environmental quality requirements.

3.6.6 Industrial Waste Management

The final dismantlement of YNPS will require the handling and disposal of system and building wastes. These wastes will include materials that were never radiologically contaminated, have been decontaminated to meet release criteria, or do not contain asbestos or other hazardous materials. Non-radioactive non-hazardous wastes are expected to include the following:

- System piping and components (e.g., pumps, valves, tanks, non-asbestos insulation, heat exchanges and supports)
- Duct-work and associated equipment (e.g., duct, fans, filters and supports)
- Electrical systems and equipment (e.g., cables and trays, conduit, motor control centers, generators, motors, and panels)
- Buildings and structures (e.g., concrete, structural steel, roofing materials, siding, doors, and windows)

The materials presented above will be processed in accordance with the rules and regulations governing the disposal of non-radioactive, non-hazardous wastes.

3.6.7 Training

All personnel involved in the handling of hazardous materials and wastes will receive initial and annual training. This training will include information on types of hazardous materials and wastes, handling precautions, temporary storage locations, and emergency response procedures.

PROCEDURE REFERENCES

AP-0030, "Non-Radioactive Hazardous Waste Management"

AP-0031, "Control of Chemicals"

AP-0043, "Hazard Communication Program"

AP-0709, "Asbestos Control Program"

OP-3011, "Oil Spill Incident"

OP-3012, "Hazardous Waste, Asbestos, or Chemical Spill"

OP-4236, "Weekly Oil Spill Prevention/Hazardous Waste Storage Inspection"

OP-MEMO 2DD-2, "YAEC Spill Prevention Control and Countermeasure Plan"

SECTION 4
FINAL RADIATION SURVEY PLAN

4.1 FINAL RELEASE CRITERIA

4.1.1 Site Release Criteria

The release of the site, facilities, and materials remaining on site will be based on application of criteria for surface contamination, soil/water concentration, and exposure rate. The NRC is in the process of developing a rule for decommissioned site release criteria. A final rule is not expected before the end of 1994.

The NRC provided YAEC with release criteria that will be applied to sites encompassed by the Site Decommissioning Management Plan (Reference 4.1-1). Yankee has reviewed these criteria and compared them with YAEC comments presented as a participant in the rulemaking on radiological criteria for decommissioning NRC-licensed facilities. Yankee participated as a member of the NUMARC Ad Hoc Advisory Committee on Residual Radioactivity and as a panelist at the NRC Workshop held in Boston on March 12, 1993.

Based on the review of the Site Decommissioning Management Plan release criteria and the development of a position for the participatory rulemaking (Reference 4.1-2), YAEC will use the following release criteria for the decommissioning of YNPS:

- Surface contamination must not exceed the values presented in Table 1 of Regulatory Guide 1.86 (Reference 4.1-3) for average, maximum, and removable contamination. Table 4.1-1 presents the limits from the regulatory guide.

- Direct exposure from gamma emitting radionuclides (e.g. cobalt-60, cesium-137, europium-152), created as a result of reactor operation and located in concrete, components, and structures, must not exceed 5 μR/hr above natural background measured 1 meter from the surface.

- Total effective dose equivalent to the critical population group from residual contamination in the components, structures, soil, and water must be maintained below 30 mrem/yr. The dose determination is based on the methods described in NUREG/CR-5512 (Reference 4.1-4).

- Migration of contamination into the groundwater and potential airborne contamination of streams must not exceed the Environmental Protection Agency regulation 40 CFR Part 141, National Primary Drinking Water Standards.

The selection of the surface contamination, direct exposure, and water quality limits is consistent with those presented by the NRC in Reference 4.1-1. The total effective dose equivalent is based on the ICRP and NCRP recommendation of 100 mrem/yr dose to members of the public from all sources of man made radiation with the exception of radon and medical exposure. A conservative value of 30 mrem/yr was chosen to ensure that the total dose from residual site radioactivity and other man made sources would not exceed the ICRP and NCRP value.

A total effective dose equivalent of less than 30 mrem/yr is consistent with the variability of background gamma radiation levels in New England (30 - 40 mrem/yr from site to site). No health effects have ever been observed at this level of background variability.

An optimization process, based on ALARA, will be used to reduce the levels of radioactivity on the site commensurate with the minimization of total risk. Based on the NRC Issues Paper supporting the rulemaking, the ALARA analysis will take into account the following:

- Radiation doses and environmental impacts from the decommissioning process and from the residual radiation remaining on the site after completion of decommissioning

- All of the costs and other risks associated with decontamination and decommissioning the site

An additional screening level of 10 mrem/yr will be established below which further optimization analyses are not necessary. This level is well below background variability, and below the level where expenditure of additional efforts will likely result in a net reduction in risk.

It is unlikely that YNPS decommissioning and final survey activities will be completed prior to completion of the scheduled rulemaking. Yankee will continue to monitor and participate in the rulemaking activities. Changes will be made to the final release criteria, if appropriate, after the final rule is issued. However, until such time, the final release criteria above will be used as the basis for detailed decommissioning planning.

4.1.2 Material Release Criteria

All materials leaving the Radiation Control Area and the YNPS site will be surveyed to ensure that radioactive materials are not inadvertently discharged from the facility. Plant Procedure AP-0052, "Radiation Protection Release of Equipment, Materials, and Vehicles," will be used to ensure that all potentially radioactive or contaminated items removed from the Radiation Control Area or the YNPS site are surveyed. This procedure was written to incorporate the guidance presented in NRC Circular No. 81-07 and NRC Information Notice No. 85-92 (References 4.1-5 and 4.1-6). The following survey methods will be used:

- Materials and Equipment - Direct frisking with a portable Geiger-Mueller or a gas flow proportional detector.
- Smear Samples - Analysis with a Geiger-Mueller or a gas flow proportional detector.
- Bulk Liquids or Soil - Analysis with high resolution gamma spectrometry system to the environmental lower limit of detection.

Materials will be released if no discernable plant-related activity is detected within the capability of the survey methods presented above.

4.2 FINAL SURVEY METHODOLOGY

4.2.1 Overview

The purpose of the final radiation survey is to demonstrate that the radiological condition of the facility is within the final site release criteria. Demonstrating that this has been achieved requires collection of radiological data to determine the following:

- Surface activity levels (total and removable)
- Direct exposure rates
- Radionuclide concentrations in soil and groundwater

In addition, final site release criteria verification requires determination of total site inventory of residual radioactive material.

The data must be accurate and representative of site conditions. These objectives will be ensured through the execution of a well-documented, statistically based survey plan. This section describes the methodology and criteria that will be used in developing and executing a final radiation survey plan.

The NRC has issued draft guidelines in NUREG/CR-5849 for conducting radiological surveys in support of license termination (Reference 4.2-1). Yankee will use these guidelines to develop a final survey plan for YNPS that incorporates the following items:

- A detailed description of the types, extent, and locations of measurements and samples that will be obtained.
- A description of the equipment and techniques that will be used for measuring, sampling, and analyzing data.
- A description of the methods for interpreting and evaluating the data.
- A list of quality control requirements for ensuring data quality.

The final survey plan will be implemented through plant procedures, which will be developed prior to the start of survey activities.

4.2.2 Instrumentation

Selection and use of radiation detection instrumentation during the final survey are critical elements of determining the site radiological condition. Final survey instrumentation will be selected based upon the need to ensure that the residual radiation remaining on-site will not exceed the site release criteria. This requires the use of diverse instrumentation types with the capability of measuring levels below 75% (preferably below 10%) of the guideline values. Instrumentation selection will be based on both general guidance presented in Section 5.0 of NUREG/CR-5849 and industry experience.

Instrumentation used in the surveys will be calibrated against sources and standards that are traceable to the National Institute of Standards and Technology (NIST). The calibration sources and standards will contain nuclides or representative nuclide mixes encountered at the site. Calibrations will be conducted using industry recognized standards (e.g., ANSI, NCRP, INPO) and approved procedures.

Personnel performing the surveys will receive training to qualify them in the tasks being performed. The extent of the training will be commensurate with the education and experience of the technician and the scope and complexity of the task.

4.2.3 Documentation

All aspects of the survey will be documented in accordance with plant approved procedures. These procedures will become part of the administrative record of the survey.

A final survey report will be prepared using the guidance presented in NUREG/CR-5849. The final survey report will provide a complete record of the facility radiological condition and comparison of the condition to the site release criteria. Additionally, sufficient information and data will be provided to enable an independent re-creation and evaluation of both the survey activities and the derived results.

4.2.4 Quality Assurance

Applicable portions of the quality assurance program (Section 7) will be implemented during all phases of the final radiation survey to ensure the validity of the data and analysis results. Periodic audits will be performed to verify that survey activities comply with established procedures and applicable aspects of the quality assurance plan.

4.2.5 Independent Verification

After acceptance of the YNPS termination survey report, the NRC will select an independent contractor to perform a confirmatory survey to verify the adequacy of the final radiation survey. The confirmatory survey results will be used in conjunction with decommissioning documentation to support a decision to terminate the license and release the site for unrestricted use. The radiation survey programs and procedures in place during decommissioning will be developed and implemented to ensure that they are adequate to support the independent verification process.

REFERENCES

4.1-1 NYR 93-028, Site Release Criteria For Decommissioned Nuclear Plants, S. H. Weiss (USNRC) to A. C. Kadak, February 25, 1993.

4.1-2 FYC 93-017, Yankee Atomic Electric Company's Comments on "Rulemaking Issues Paper on Radiological Criteria for Decommissioning of NRC-licensed Facilities" (57 FR 58727, comment period extended by 58 FR 29998), D. W. Edwards to S. Chilk (USNRC), June 28, 1993.

4.1-3 Regulatory Guide 1.86, Termination of Operating Licenses For Nuclear Reactors, June 1974.

4.1-4 NUREG/CR-5512, Residual Radioactive Contamination From Decommissioning, September 1992.

4.1-5 NRC IE Circular No. 81-07, Control of Radioactively Contaminated Material, May 14, 1981.

4.1-6 NRC IE Information Notice No. 85-92, Surveys of Wastes Before Disposal From Nuclear Reactor Facilities, December 2, 1985.

4.2-1 NUREG/CR-5849 (Draft), Manual For Conducting Radiological Surveys In Support of License Termination, May 1992.

PROCEDURE REFERENCES

AP-0052, "Radiation Protection Release of Equipment, Materials and Vehicles"

TABLE 4.1-1

REGULATORY GUIDE 1.86 ACCEPTABLE SURFACE CONTAMINATION LEVELS

Nuclides (1)	Average (2,3)	Maximum (2,4)	Removable (2,5)
U-nat, U-235, U-238, and associated decay products	5,000 dpm α / 100 cm^2	15,000 dpm α / 100 cm^2	1,000 dpm α / 100 cm^2
Transuranics, Ra-226, Ra-228, Th-230, Th-228, Pa-231, Ac-227, I-125, I-129	100 dpm / 100 cm^2	300 dpm / 100 cm^2	20 dpm / 100 cm^2
Th-nat, Th-232, Sr-90, Ra-223, Ra-224, U-232, I-126, I-131, I-133	1,000 dpm / 100 cm^2	3,000 dpm / 100 cm^2	200 dpm / 100 cm^2
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except Sr-90 and others noted above.	5,000 dpm β-γ / 100 cm^2	15,000 dpm β-γ / 100 cm^2	1,000 dpm β-γ / 100 cm^2

Notes:

1. Where surface contamination by both alpha- and beta-gamma-emitting nuclides exists, the limits established for alpha- and beta-gamma-emitting nuclides should apply independently.

2. As used in this table, dpm (disintegrations per minute) means the rate of emission by radioactive material as determined by correcting the counts per minute observed by an appropriate detector for background, efficiency, and geometric factors associated with the instrumentation.

3. Measurements of average contaminant should not be averaged over more than 1 square meter. For objects of less surface area, the average should be derived for each such object.

4. The maximum contamination level applies to area of not more than 100 cm^2.

5. The amount of removable material per 100 cm^2 of surface area should be determined by wiping that area with dry filter or soft absorbent paper, applying moderate pressure, ans assessing the amount of radioactive material on the wipe with an appropriate instrument of known efficiency. When removable contamination on objects of less surface area is determined, the pertinent levels should be reduced proportionally and the entire surface should be wiped.

6. Source: Regulatory Guide 1.86, Table 1.

SECTION 5
DECOMMISSIONING COST ESTIMATE AND FUNDING PLAN

5.1 DECOMMISSIONING COST ESTIMATE

In May 1992, YAEC and TLG Engineering, Inc. prepared an updated cost estimate for the YNPS decommissioning (Reference 5.1-1). Unlike previous YNPS decommissioning cost studies, the updated study was site specific, based on an analysis of all plant systems, structures, and components. In addition, specific consideration was given to unique YNPS features (e.g., plant size) in developing the decommissioning organization.

The structure of the cost estimate was based on the guidelines presented in the AIF/NESP-36 "Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates" (Reference 5.1-2). TLG supplemented the estimate with specific decommissioning and decommissioning cost estimating experience gained subsequent to the publication. The estimate included the following information:

- Cost estimates for the immediate dismantlement (DECON case) and deferred dismantlement (SAFSTOR case) alternatives
- Detailed schedules and staff requirements
- Costs for remediation, labor, equipment, and plant operation

As discussed in Section 2.1, YAEC proposes to defer dismantlement until after a contract for the disposal of low level radioactive waste is secured. The deferred dismantlement cost estimate assumed that a low level radioactive waste disposal site would be available in 2000. The cost estimate also assumed, for planning purposes, that spent fuel would be transferred to a dry storage facility in 1996 and that the final spent fuel shipment to the Department of Energy would occur in 2018.

YAEC has revised the deferred dismantlement cost estimate to reflect execution of the Component Removal Project. The revised cost estimate indicates a significant credit for removing the steam generators, pressurizer, and reactor vessel internals in 1993 and 1994. A summary of the cost estimates is presented in Table 5.1-1. The cost estimate, with the incorporation of the Component Removal Project, remains representative of the cost of decommissioning YNPS. The cost estimate will be revised during preparation of the FERC submittal required by the settlement described in Section 5.2.

5.2 DECOMMISSIONING FUNDING

The following decommissioning funding requirements from 10 CFR 50.82 must be incorporated into the decommissioning plan:

- 10 CFR 50.82 (b) - A decommissioning plan must include an updated cost estimate for the decommissioning alternative, a comparison of the estimate with present funds set aside for decommissioning, and a plan for assuring the availability of adequate funds for completion of decommissioning.

- 10 CFR 50.82 (c) - A decommissioning plan that proposes the SAFSTOR alternative must provide the following:

  (1) Assurance either that funds needed to complete decommissioning are placed into an account segregated from licensee assets and outside of the licensee's administrative control during the SAFSTOR period, or that a surety method of fund statement of intent must be maintained in accordance with the criteria of 10 CFR 50.75 (e).

  (2) Assurance that a method is established for adjusting cost estimates and associated funding levels over the SAFSTOR period.

In 1980, YAEC initiated a proceeding before the Federal Energy Regulatory Commission (FERC) for approval to establish a decommissioning fund. In May 1981, FERC authorized a decommissioning fund accumulation from customers over a ten year period ending in 1991. Since that time, YAEC has filed periodic collection updates based on revised cost estimates and collection periods.

From March 1981 through June 30, 1993, YAEC accumulated $99.6 million in cash and anticipated tax credits for decommissioning. The decommissioning collections are made through YAEC's Power Contracts and are deposited in an independent and irrevocable trust at a commercial bank, with the principle and interest to be used to discharge future decommissioning obligations as incurred. YAEC has provided a copy of the trust document to the NRC (Reference 5.2-1). This trust is executed in compliance with 10 CFR 50.75(e)(ii).

On June 1, 1992, YAEC submitted a rate filing with FERC, in part to increase decommissioning collections to fund the balance of the revised cost estimate. The increased collections would be recovered through June 2000 utilizing YAEC's existing Power Contracts. In March 1993, FERC approved a settlement of $235 million with respect to all parties, with the exception of the town of Norwood, Massachusetts, a customer of one of YAEC's sponsors which is obligated for 0.413% of YAEC's charges.

A separate proceeding regarding Norwood is now pending.

The settlement of $235 million is lower than the deferred dismantlement cost estimate (SAFSTOR case) of $247 million submitted to FERC. The lower value was chosen for the purposes of the settlement and does not establish any principle or precedent for future proceedings regarding the cost or timing of a collection for YNPS decommissioning. The settlement also requires YAEC to file a revised decommissioning funding schedule, supported by a revised cost study, within 90 days of NRC approval of the decommissioning plan. The revised cost estimate will include an updated cost and collection schedule consistent with the NRC approved plan.

Decommissioning cash flows were determined by escalating the January 1992 dollar estimate ($235 million) to the period when decommissioning activities are expected to take place. Safe storage period activities begin in 1995 and site restoration ends in 2002, with on-site spent fuel storage extending to 2018. The weighted escalation factor used to determine the current dollar decommissioning cost was 5.65% per year.

The annual funding requirement, which is necessary to accumulate the amount needed for decommissioning, accounts for several funding sources:

- Payments to the fund from customers pursuant to the Power Contracts
- Earnings realized from the investment of those contributions
- Tax loss carrybacks

The combination of funds from these three sources (net of taxes and fees) must equal the cost of decommissioning activities expected to take place during the period from 1995 to 2018. The cash flow estimate contains assumptions of the timing of decommissioning, investment rate of return, federal and state tax rates, and escalation.

The FERC rate filing also included an estimate of facility operating and maintenance costs for the period from January 1993 until NRC approval of the Decommissioning Plan. The estimate is based on Decommissioning Plan approval by January 1995. These costs are additional to those that have specifically been categorized as decommissioning costs. The pre-decommissioning costs do not include Component Removal Project costs, which are an early expenditure of decommissioning costs.

As indicated by the foregoing summary, YAEC's Power Contracts with its sponsors constitute FERC approved tariffs which are binding upon the sponsors. The Power Contracts obligate the sponsors to continue payments into the decommissioning fund until decommissioning is completed. The FERC orders referred to above have acknowledged that continuing obligation of the sponsors as a cost of the power generated by YAEC and the periodic reviews of revised decommissioning cost estimates mandated by FERC provide the mechanism for updating the required payments under the Power Contracts to assure adequate funds for that purpose.

In 1993, YAEC requested that the NRC approve use of decommissioning funds for the Component Removal Project (Reference 5.2-2). This request was in response to NRC guidance which authorized the staff to approve withdrawals from decommissioning funds for legitimate decommissioning activities implemented before NRC approval of the decommissioning plan (Reference 5.2-3). The Component Removal Project activities are legitimate decommissioning activities and were included in the original cost estimate (Table 5.1-1). The NRC concurred with YAEC concluding that the proposed activities were legitimate decommissioning activities and that YAEC would maintain the decommissioning funds at an adequate level throughout the Component Removal Project. Based on their review, the NRC did not object to the use of the decommissioning funds for the project (Reference 5.2-4).

In accordance with 10 CFR 50.82(c), YAEC will review the cost of decommissioning on an annual basis during the safe storage period. The purpose of this review is to identify any significant change in the cost of decommissioning that would require modification of the decommissioning trust fund collections.

REFERENCES

5.1-1 Decommissioning Cost Estimate for the Yankee Nuclear Power Station, TLG Engineering, Inc., May 1992.

5.1-2 AIF/NESP-036 (Volumes 1 and 2), Guidelines for Producing Commercial Nuclear Power Plant Decommissioning Cost Estimates, May 1986.

5.2-1 BYR 90-102, Decommissioning Funding-Assurance Report and Certification, H. T. Tracy to Document Control Desk (NRC), July 25, 1990.

5.2-2 BYR 93-014, Decommissioning Funds, H. T. Tracy to M. B. Fairtile (USNRC), March 5, 1993.

5.2-3 SECY-93-137, Use of Decommissioning Trust Funds Before Decommissioning Plan Approval, May 18, 1993.

5.2-4 NYR 93-045, Decommissioning Funds (TAC No. M85897), M. B. Fairtile (USNRC) to H. T. Tracy, April 16, 1993.

TABLE 5.1-1

DECOMMISSIONING COST ESTIMATE SUMMARY

	SAFSTOR (1)	SAFSTOR With Component Removal Project (2)
Component Removal Project Low Level Radioactive Waste Balance of Project Total	 0.00	 11.23 15.43 26.66
Safe Storage Period	32.66	32.66
Low Level Radioactive Waste	53.18	43.78
Spent Fuel Storage (3)	56.49	56.49
Basic Dismantlement	104.79	83.66
Total (4)	247.12	243.25

Note:

1. The SAFSTOR (deferred dismantlement) case assumes that the plant is placed in a storage mode until 2000 when dismantlement occurs. This case was submitted to FERC in 1992.

2. The SAFSTOR/CRP case assumes execution of the Component Removal Project (CRP) followed by a storage mode until 2000 when dismantlement is resumed.

3. The cost estimates assume that the last fuel assembly is removed from the site in 2018.

4. All costs are based on January 1992.

SECTION 6
DECOMMISSIONING TECHNICAL SPECIFICATIONS

6.1 DESCRIPTION

After the decision to permanently cease YNPS reactor operations, YAEC submitted a proposed change in March 1992 to the NRC modifying the plant full power operating license to a possession only license status (Reference 6.1-1). The proposed change removed authority of YNPS to operate the reactor and to move fuel back into either the Reactor Vessel or the Vapor Container. The facility license was amended in August 1992 to a possession only status (Reference 6.1-2).

Coincident with the possession only license submittal, Yankee reviewed the plant licensing basis to determine the applicability of existing technical specifications to a permanently defueled condition. The technical specifications which were applicable to a defined mode associated with fuel in the reactor were determined to be not applicable to the permanently defueled condition. Technical specifications that were determined to be applicable at all times were reviewed to assess their relevance to the permanently defueled condition. Individual technical specification changes were proposed for the higher priority items identified by the review. Table 6.1-1 summarizes the technical specification changes proposed by YAEC.

On December 23, 1992, a technical specification change was proposed to complete the transition of the technical specifications to be applicable to the permanently defueled condition (Reference 6.1-3). The proposed change eliminated specifications that were not applicable and reformatted the specifications that remained applicable. The NRC approved the license amendment on June 11, 1993 (Reference 6.1-4).

Limiting conditions for operation and surveillance requirements were reduced to the following items:

- Applicability - This specification describes the implementation of the limiting conditions for operation and surveillance requirements.

- Spent Fuel Pit Water Level - This specification presents a minimum spent fuel pit water level requirement to ensure the basis of the fuel handling accident and to minimize exposure to personnel during fuel handling.

- Crane Travel Over Spent Fuel Pit - This specification presents maximum load and load path restrictions for the spent fuel pit crane to protect fuel stored in the spent fuel pit.

- Spent Fuel Storage Area Radiation Monitor - This specification presents radiation monitoring operability requirements to ensure detection of inadvertent criticality during fuel handling.

- Liquid Hold-Up Tanks - This specification presents radioactive material curie limits for tanks that are not protected by engineered features that would contain inadvertent release of their contents.

- Sealed Source Contamination - This specification presents maximum contamination limits for sealed sources to ensure that allowable intake limits are not exceeded.

The design features and administrative controls were modified to delete portions that were not applicable to a permanently defueled condition.

No additional technical specification changes are needed to control decommissioning operations.

REFERENCES

6.1-1 BYR 92-037, Request For Modification of Yankee Nuclear Power Station's Operating License To Remove Authorization For Reactor Power Operation - Possession Only License, J. K. Thayer to T. Murley (USNRC), March 27, 1992.

6.1-2 NYR 92-148, Issuance of Amendment No. 142 To Facility License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M83024), M. B. Fairtile (USNRC) to J. M. Grant, August 8, 1992.

6.1-3 BYR 92-112, Permanently Defueled Technical Specifications, J. K. Thayer to M. Fairtile (USNRC), December 21, 1992.

6.1-4 NYR 93-062, Issuance of Amendment No. 148 To Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M85244), M. B. Fairtile (USNRC) to J. M. Grant, June 11, 1993.

TABLE 6.1-1

TECHNICAL SPECIFICATION CHANGES FOR

TRANSITION TO PERMANENTLY DEFUELED CONDITION

Submittal	Approval Date	Amendment Number	Reference
Minimum Shift Staffing	July 22, 1992	141	NYR 92-143, Issuance of Amendment No. 141 To Facility Operating License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M83383).
Possession Only License	August 8, 1992	142	NYR 92-148, Issuance of Amendment No. 142 To Facility License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M83024).
Transfer of Fire Protection Specifications To Manual	August 20, 1992	144	NYR 92-156, Issuance of Amendment No. 144 To Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M83746).
PORC Administration Change	September 4, 1992	145	NYR 92-182, Issuance of Amendment No. 145 To Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M84005).
RETS/REMP Transfer	November 5, 1992	146	NYR 92-183, Issuance of Amendment No. 146 To Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M84372).
Permanently Defueled Technical Specifications	June 11, 1993	148	NYR 93-062, Issuance of Amendment No. 148 To Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M85244).

SECTION 7
DECOMMISSIONING QUALITY ASSURANCE PLAN

7.1 POLICY STATEMENT

The YNPS Decommissioning Quality Assurance Plan is based on the requirements of 10 CFR Part 50 Appendix B and will be applied to selected activities during the decommissioning of YNPS. The plan is intended to assure both occupational and public health and safety.

Procedures shall ensure compliance with appropriate regulatory, statutory, and license requirements. Specific quality assurance requirements and organizational responsibilities for implementation of these requirements shall be specified in the appropriate procedures.

Compliance with this plan and procedures is mandatory for personnel involved with decommissioning activities which may affect quality and the health and safety of project personnel and the general public. Personnel shall, therefore, be familiar with plan requirements and responsibilities that are applicable to their individual activities and interfaces.

7.2 INTRODUCTION

The Decommissioning Quality Assurance Plan is applicable to and is structured to assure the following:

- Plant configuration is maintained in accordance with the YNPS design requirements for those structures, systems, and components that prevent or mitigate the consequences of a fuel handling accident.

- Compliance with applicable license conditions and technical specifications.

- Compliance with regulatory requirements and design bases for the following programs:

  Radioactive Effluent Control
  Radiological Environmental Monitoring
  Radiation Protection
  Radioactive Waste Handling and Shipping (including Process Control)
  Security
  Fire Protection
  Emergency Preparedness
  Training

7.3 ORGANIZATION

YAEC is responsible for the Quality Assurance Plan implementation. Verification of effective plan implementation is the responsibility of the Yankee Nuclear Services Division (YNSD) Quality Assurance organization. Quality Assurance personnel shall have the authority and organizational freedom to identify quality problems; to take action to stop unsatisfactory work and control further processing, delivery, installation or use of nonconforming items; to initiate, to recommend, or to provide solutions; and to verify implementation of solutions. The persons and organizations performing quality assurance functions report to a management level that assures the required authority and organizational freedom are provided, including sufficient independence from cost and schedule. The individuals assigned the responsibility for assuring effective execution of any portion of the Quality Assurance Plan have direct access to the levels of management necessary to perform quality assurance functions.

The YAEC decommissioning organization is presented in Section 2.4. An organizational chart is included along with a description of responsibilities of significant positions.

7.4 DECOMMISSIONING QUALITY ASSURANCE PLAN

The Decommissioning Quality Assurance Plan shall apply to systems, structures, and components necessary to prevent or mitigate the consequences of a fuel handling accident. Applicable portions shall also apply to programs for Radioactive Effluent Control, Radiological Environmental Monitoring, Radiation Protection, Radioactive Waste Handling and Shipping (including Process Control), Security, Fire Protection, Emergency Preparedness, and Training. Specific details are presented in Table 7.4-1.

Implementation of the plan shall be controlled by written procedures. The adequacy and status of the plan shall be regularly reviewed.

Training programs shall be established for those personnel performing activities affecting quality such that they are knowledgeable in the quality assurance documents and their requirements, and proficient in implementing these requirements. These training programs are described in Section 2.6.

7.4.1 Design Control

Design controls shall be established commensurate with regulatory requirements, the potential impact on quality, and the health and safety of project personnel and the general public. These controls shall apply to those activities presented in Table 7.4-1.

Appropriate provisions of design control shall include specification of design input, correct translation of input in design documents, verification of design by persons other than the originator, and assurance that changes to design are properly reviewed, controlled, and documented.

7.4.2 Procurement Document Control

Measures shall be established to assure that applicable regulatory requirements, design bases, and other requirements necessary to assure adequate quality are included or referenced in the documents (including changes to the documents) for procurement of material, equipment, and services. The program applies to those activities presented in Table 7.4-1 shall have provisions for the following:

- Evaluating prospective suppliers.
- Accepting purchased items and services.
- Ensuring that procurement, inspection, and test requirements have been satisfied before an item is placed in service or used.

- Ensuring that appropriate controls for the selection, determination of suitability for intended use (critical characteristics), evaluation, receipt, and quality evaluation of commercial-grade items are to be imposed to ensure that they will perform satisfactorily in service.

7.4.3 Instructions, Procedures and Drawings

Instructions, procedures, and drawings of a type appropriate to the circumstances shall be provided for the control and performance of activities important to quality, health, and safety. These shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Provisions shall be made to establish procedures which define responsibilities and delineate the sequence of actions to be accomplished in the preparation, review, approval, and control of instructions, procedures, or drawings, including associated changes.

The following typical procedures shall be provided as appropriate for those activities presented in Table 7.4-1. This list includes procedures for executing quality assurance functions:

- Calibration procedures
- Radiation protection procedures
- Special process procedures
- Work control documents that provide work instructions accompanied with pertinent technical data to perform a specific task
- Radioactive material processing, packaging and transporting procedures
- Audit and QA surveillance procedures
- Administrative control procedures
- Emergency response procedures
- Inspection procedures
- Training/qualification/certification procedures

- Procurement procedures
- Design and design document control procedures
- Nonconformance/corrective action procedures
- Quality records procedures
- Access control procedures
- Material/equipment control procedures
- Site characterization procedures including final site survey procedures
- Fire prevention/protection procedures

7.4.4 Document Control

Measures shall be established to control the issuance of and changes to documents such as procedures, drawings and specifications that prescribe those activities affecting quality as presented in Table 7.4-1. Measures shall be established to assure that these documents are reviewed for adequacy and approved for release by authorized personnel. Completed documents will be distributed to and used at the location where the prescribed activity is performed. Provisions shall be established such that changes to these documents are reviewed and approved in the same manner as the original.

7.4.5 Control of Purchased Material, Equipment and Services

Measures shall be established to assure that purchased material, equipment and services conform to the procurement documents for those activities presented in Table 7.4-1. These measures shall include provisions, as appropriate, for vendor evaluation and selection, objective evidence of quality materials furnished by the vendor, surveillance at the vendor source, and inspection of products upon delivery.

The effectiveness of the control of contractor services shall be assessed at intervals consistent with the importance of the service.

The adequacy of the vendor's Quality Assurance Plan specified in procurement documentation shall be verified prior to use when appropriate. Vendor adherence to their Quality Assurance Plan shall also be verified as appropriate.

Appropriate material and equipment shall be inspected upon receipt at the site prior to use or storage to determine that procurement requirements are satisfied.

Material, parts, and components that are to be utilized for packaging and transporting of radioactive materials shall be inspected upon receipt to assure that associated procurement document provisions have been satisfied.

Measures shall be established for identifying nonconforming material, parts and components.

7.4.6 Identification and Control of Materials, Parts, and Components

Measures shall be established for the identification and control of critical materials, parts, components, and equipment to prevent the use of any incorrect or defective item.

7.4.7 Control of Special Processes

Measures shall be established so that special processes such as welding and nondestructive examination are controlled and accomplished by qualified personnel for those activities presented in Table 7.4-1.

Welding activities shall be performed in accordance with qualified procedures that are qualified in accordance with applicable codes and standards and reviewed to assure their technical adequacy.

Nondestructive examinations shall be performed in accordance with procedures formulated in accordance with applicable codes and standards and shall be reviewed to assure their technical adequacy.

7.4.8 Inspection

Measures shall be established for inspection of those activities presented in Table 7.4-1 to verify conformance with specified requirements.

Inspection planning activities are to identify the characteristics and activities to be inspected and the organization responsible for performing the inspection. Planning shall include measures for the establishment of inspection hold points in appropriate work documents. Associated work shall not proceed beyond a hold point without prior consent.

Required inspections shall be performed in accordance with written instructions or checklists that identify the acceptance criteria and any special equipment or calibrations required to conduct the inspection.

Personnel performing inspections shall be qualified based upon experience and training in inspection methods. Required inspections shall not be performed by individuals who performed the activity.

7.4.9 Test Control

A test control program is to be established to demonstrate that items associated with the activities presented in Table 7.4-1 will perform satisfactorily in service. This includes proof tests before installation, pre-operational tests, post-modification tests, post-maintenance tests, and operational tests.

The program is to be controlled by procedures that include instructions and prerequisites for performing the test, the use of proper test equipment, acceptance criteria, and mandatory hold points as required.

Test results shall be documented and evaluated by responsible authorities assure that test requirements have been satisfied.

7.4.10 Control of Measuring and Test Equipment (M&TE)

Measures shall be established to assure that appropriate tools, gauges, instruments, and other measuring and testing devices used in activities important to quality, health and safety are properly controlled, calibrated and adjusted at specified periods to maintain accuracy within necessary limits, and maintain traceability to National Institute of Standards and Technology (NIST), or other known standard.

7.4.11 Handling, Storage, and Shipping

Measures shall be established to control the handling, storage, and shipping of items associated with the activities presented in Table 7.4-1 to prevent their damage, deterioration, or loss. Special protective measures are to be specified. Items are to be marked and labeled to identify, maintain, and preserve the items' integrity and indicate the need for special control.

Areas shall be provided for storage of radioactive material that assure physical protection, radiation exposure to personnel as low as reasonable achievable, and control of the stored material.

Handling, storage, and shipment of radioactive material shall be controlled based upon the following criteria:

- Established safety requirements concerning the handling, storage, and shipping of packages for radioactive material shall be followed.

- Shipments shall not be made unless all tests, certifications, acceptances, and final inspections have been completed.

- Procedures shall be provided for handling, storage, and shipping operations.

Shipping and packaging documents for radioactive material shall be consistent with pertinent regulatory requirements.

7.4.12 Inspection, Test, and Operating Status

Appropriate controls shall be established for the control of radioactive material, systems configuration, as well as personnel exposure.

Inspection, test, and operating status of equipment and components associated with radioactive material, systems configuration, and personnel exposure shall be established based upon the following criteria:

- Inspection, test, and operating status for radioactive material, system configuration, and personnel exposure shall be indicated and controlled by established procedures.

- Status shall be indicated by tag, label marking or log entry.

- Status of nonconforming items or packages shall be positively maintained by established procedures.

7.4.13 Nonconforming Materials, Parts or Components

Measures shall be established to control materials, parts, or components that do not conform to requirements in order to prevent their inadvertent use or release for shipment. These measures shall include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations.

Nonconformance items shall be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures.

7.4.14 Corrective Action

Measures shall be established to assure that conditioned adverse to quality, health and safety are promptly identified and corrected.

In the event of significant conditions adverse to quality, the measures shall assure that the cause of the conditions is determined, and corrective action taken to preclude repetitions These conditions shall be documented and reported to appropriate levels of management.

7.4.15 Quality Assurance Records

Sufficient records shall be maintained to furnish evidence of activities important to safe decommissioning as required by code, standard, specification or project procedures. Typical records would include:

- Decommissioning Plan
- Procedures
- Reports
- Personnel qualification records
- Radiological and environmental site characterization records, including final site release records
- Dismantlement records
- Inspection, surveillance, audit and assessment records

Records shall be identifiable, available, and retrievable.

Requirements shall be established concerning record collection, safekeeping, retention, maintenance, updating, location, storage, preservation, administration, and assigned responsibility. Such requirements shall be consistent with the potential impact on quality, radiation exposure to the workers and the public, and applicable regulations.

Records shall be reviewed to ensure their completeness and ability to serve their intended function.

7.4.16 Audits

A system of planned audits shall be carried out to verify compliance with appropriate requirements of the Quality Assurance Plan and to determine the effectiveness of the plan. The audits shall be performed in accordance with written procedures or checklists by trained and qualified personnel not having direct responsibility in the areas being audited.

Audit reports shall be prepared that include a description of the area audited, identification of individuals responsible for implementation of the audited provisions, identification of individuals responsible for performance of the audit, and identification of discrepant areas. Audit reports shall be distributed to the appropriate level of management and to those individuals responsible for implementation of audited provisions. Measures shall be established to assure that discrepancies identified by audits are resolved. These measures shall include notification of the manager responsible for the discrepancy, and verification of satisfactory resolution. Discrepancies shall be resolved by the manager responsible for the discrepancy. Higher levels of management shall resolve disputed discrepancies. Follow-up action, including re-audit of deficient areas, shall be taken where indicated.

REFERENCES

7.1-1 Facility Operating License No. DPR-3 - Yankee Nuclear Power Station

7.1-2 YOQAP-1-A; Yankee Operational Quality Assurance Manual

7.1-3 10 CFR Part 50 Appendix B; Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants

7.1-4 NRC Inspection & Enforcement Manual, Chapter 2561; Reactor Inspection Program -Post Operational Phase

TABLE 7.4-1

APPLICATION OF QUALITY ASSURANCE PROGRAM

Program/Activities	Appendix B Criteria																	
	I	II	III	IV	V	VI	VII	VIII	IX	X	XI	XII	XIII	XIV	XV	XVI	XVII	XVIII
Structures, Systems, Components presented in Section 7.2	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Plant Operations/Technical Specifications	X	X			X	X					X	X	X	X	X	X	X	X
Radiation Protection	X	X			X	X									X	X	X	X
Radioactive Effluent Control	X	X	X	X	X	X	X								X	X	X	X
Radioactive Environmental Monitoring	X	X	X	X	X	X	X								X	X	X	X
Radwaste Handling & Shipment (including Process Control)	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
Fire Protection	X	X	X	X	X	X	X			X	X			X	X	X	X	X
Security	X	X	X		X	X									X	X	X	X
Emergency Preparedness	X	X			X	X									X	X	X	X
Training	X	X			X	X									X	X	X	X

Appendix B Criteria

I Organization
II Quality Assurance Program
III Design Control
IV Procurement Document Control
V Instructions, Procedures, & Drawings
VI Document Control
VII Control of Purchased Material, Equipment, & Services
VIII Identification & Control of Materials, Parts, & Components
IX Control of Special Processes
X Inspection
XI Test Control
XII Control of Measuring & Test Equipment
XIII Handling, Storage, & Shipping
XIV Inspection, Test, & Operating Status
XV Nonconforming Materials, Parts, or Components
XVI Corrective Actions
XVII Quality Assurance Records
XVIII Audits

SECTION 8
DECOMMISSIONING ACCESS CONTROL PLAN

8.1 PERMANENTLY DEFUELED SECURITY PLAN

After the decision to permanently cease YNPS reactor operations, Yankee completed a comprehensive review of the security requirements for YNPS. The review incorporated the following:

- Re-evaluation of the design basis threat matrix to determine security needs for a permanently defueled condition.

- Evaluation of the Vital and Access Control Areas to determine the appropriate level of protection for the permanently defueled condition. This analysis reviewed existing safety analyses, the potential consequences of radiological sabotage events, and the potential consequences of fuel storage events. The evaluation indicated that nuclear security could be focussed on spent fuel storage.

- Physical survey of the site to determine physical modifications needed to implement a reduced protected zone. Implementation of the recommendations of this survey allowed reduction of the protected area and the implementation of an industrial security zone for most of the plant site.

Based on this evaluation, the security plan was modified creating a YNPS Defueled Security Plan (Reference 8.1-1 and 8.1-2). The NRC approved the plan and, on December 18, 1992, the plan was implemented by YNPS (Reference 8.1-3). The Defueled Security Plan reduces the protected area boundary to the Spent Fuel Pit Building outside wall. The area that was within the original protected area boundary (with the exception of the spent fuel complex) was reclassified as an industrial security area. Plant Procedure AP-0400, "Defueled Physical Security Organization," identifies individuals responsible for the management of the security program and for the supervision of the security force.

The NRC has inspected the implementation of the Defueled Security Plan to ensure that the changes had been implemented satisfactorily (Reference 8.1-4). The inspection concluded that the plan, "as implemented, was directed toward the protection of public health and safety."

YNPS will maintain the Defueled Security Plan throughout decommissioning. However, changes to the plan will be necessary if fuel is transferred to an alternate facility on-site.

8.2 SITE ACCESS CONTROL

YNPS access control requirements are presented in plant procedure AP-0482, "Unescorted Access Authorization." This procedure presents requirements for access to both the industrial security area and the protected area.

YNPS will maintain a Fitness For Duty program for personnel granted unescorted access to either the industrial security area or the protected area. The objective of the program is to provide a drug and alcohol free work environment. The program is essential to maintaining the health and safety of those working on site. The Fitness For Duty program is implemented through the YAEC Personnel Administrative Guideline Sections 12.1 - 12.9 (Reference 8.2-1). The guideline implements the policies presented in 10 CFR Part 26, Fitness For Duty Programs.

REFERENCES

8.1-1 BYR 92-077, Defueled Security and Training and Qualification Plans, J. K. Thayer to M. B. Fairtile (USNRC), August 11, 1992.

8.1-2 BYR 92-102, Defueled Security and Training and Qualification Plans, J. K. Thayer to M. B. Fairtile (USNRC), October 22, 1992.

8.1-3 NYR 92-194, Exemptions From Certain Requirements of 10 CFR 73.55 For The Yankee Nuclear Power Station (YNPS) (TAC No. M84267), M. B. Fairtile to J. M. Grant, November 24, 1992.

8.1-4 NYR 93-027, NRC Inspection No. 50-29/93-03, J. H. Joyner (USNRC) to J. K. Thayer, March 26, 1993.

8.2-1 YAEC Personnel Administrative Guidelines.

PROCEDURES REFERENCES

AP-0400 "Defueled Physical Security Organization"

AP-0482 "Unescorted Access Authorization"

SECTION 9
FIRE PROTECTION

9.1 PROGRAM DESCRIPTION

On August 20, 1992, the NRC approved a Technical Specification change to remove the fire protection technical specifications (Reference 9.1-1). The fire protection technical specifications were replaced with a set of administrative controls. The Fire Protection Technical Requirements Manual (Reference 9.1-2) was developed to set forth the operational and surveillance requirements of the Fire Protection Plan as approved by NRC Safety Evaluation Reports dated March 15, 1979, and as supplemented October 1, 1980 and August 27, 1986.

The Fire Protection Plan is based on defense in depth. The plan will maintain the following features, as appropriate, during decommissioning:

- Fire detection equipment and systems
- Fire barrier maintenance and control
- Personnel training and qualification program
- Fire Protection Program procedures
- Control of transient combustible materials and ignition sources, including limitations on inflammable gases as described in Section 3.4.8

Decommissioning activities will be reviewed during the safe storage and dismantlement periods to ensure that the appropriate level of fire protection is being implemented. In addition, fire protection requirements will be evaluated during detailed planning of decommissioning activities.

The following plant procedures implement the Fire Protection Plan:

- <u>AP-0050: Fire Protection Plan</u> - This procedure defines the Fire Protection Plan for the YNPS, describes the organization and structure of the fire protection program, and defines the administrative and functional responsibilities of assigned personnel.
- <u>AP-0051: Plant Fire Protection</u> - This procedure establishes the administrative controls for materials and events that could create potential fire hazards. Materials and processes, which represent a potential fire hazard, are controlled to minimize the possibility of a fire and the effect of a fire on the following:
    - Operation of the SFP cooling and support systems

- The release of radioactive material to the environment
- Plant personnel safety
- Balance of the facility

This procedure establishes housekeeping requirements, provides guidelines for hot work, designates fire areas and defines the process for control of combustibles.

Additional fire protection related programs and procedures are as follows:

- AP-0053, "Plant Fire Protection Inspection Program"
- AP-0503, "Fire Protection Training"
- OP-3017, "Fire Emergency"

9.2 FIRE PROTECTION PROGRAM IMPLEMENTATION

9.2.1 Safe Storage Period

Continued operability of required fire detection systems ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. The operability of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in those protected facilities. Off-site assistance is provided by the Town of Rowe. The Rowe Fire Department is available within a 30 minute response time. These systems and procedures will remain in effect during the safe storage period.

9.2.2 Dismantlement Period

Fire detection and suppression capabilities, as presented in Section 9.2.1, will remain in place during the decontamination and dismantlement of contaminated systems and equipment and until fire loading has been substantially reduced. In addition, spatial separation, fire barriers, and the grouping by activity level will be used to minimize the radiological consequences of a fire in radioactive materials awaiting shipment from the site. Prior to final building and area decontamination and dismantlement activities, fire detection and suppression systems will be isolated and removed. Supplemental fire detection and suppression measures may be employed as required.

REFERENCES

9.1-1 NYR 92-156, Issuance of Amendment No. 142 To Facility Possession Only License No. DPR-3 - Yankee Nuclear Power Station (TAC No. M83746), M. B. Fairtile (USNRC) to J. M. Grant, August 20, 1992.

9.1-2 Fire Protection Technical Requirements Manual

PROCEDURE REFERENCES

AP-0050 "Fire Protection Plan"

AP-0051 "Plant Fire Protection"

AP-0053 "Plant Fire Protection Inspection Program"

AP-0503 "Fire Protection Training"

OP-3017 "Fire Emergency"

APPENDIX A
COMPONENT REMOVAL PROJECT SUMMARY

The YNPS Decommissioning Plan does not include the dismantlement of the four steam generators, the pressurizer, and the Reactor Vessel internal components. These components will be removed in a project that will be completed before June 30, 1994. The plan is written from the perspective that these activities have been completed. Any changes resulting from the execution of the project will be incorporated in future revisions of this plan.

A.1 PROJECT BACKGROUND

In late September 1992, YAEC initiated an evaluation of the feasibility of removing and disposing of the four steam generators, the pressurizer and the Reactor Vessel internal components at YNPS. This evaluation was prompted, in part, by an extension of access to the Barnwell low level radioactive waste disposal facility for the period between January 1, 1993 and June 30, 1994. The scope of the project was based on a careful evaluation of several factors including the impact on future decommissioning activities, duration of the disposal facility access, and the total project costs. In March 1993, a decision was made to proceed with the Component Removal Project.

The project feasibility evaluation included significant interaction with the NRC to establish a basis for completing decommissioning activities prior to NRC approval of the decommissioning plan. The possession only license amendment (Reference A.1-1) allows certain decommissioning activities to proceed. However, its application to the proposed Component Removal Project required NRC clarification regarding the type of activities that could proceed without an approved decommissioning plan.

The NRC Commissioners issued guidance to the NRC Staff on decommissioning activities in the form of a Staff Requirements Memorandum in January 1993 (Reference A.1-2). The Staff Requirements Memorandum included clarification of the nature and extent of activities that can be accomplished prior to NRC approval. Activities may be conducted if they:

- Do not violate the terms of the existing license with consideration of 10 CFR 50.59
- Do not foreclose release of the site for possible unrestricted use
- Do not significantly increase the cost of decommissioning
- Do not cause any significant environmental impacts not previously evaluated

Yankee performed a comprehensive review of the project scope to verify that all aspects were in compliance with the guidance above as well as all other applicable regulatory and plant licensing requirements. Yankee provided to the NRC the methodology that will be used to implement the Component Removal Project (Reference A.1-3). The NRC concluded that YAEC had properly addressed all of the criteria and guidance necessary to proceed with the project (Reference A.1-4).

A.2 COMPONENT DESCRIPTIONS

This section presents a brief description of the components removed during the Component Removal Project.

A.2.1 Steam Generator

The steam generators used heat generated in the reactor core to produce dry saturated steam which flowed through the turbine generator to produce electricity. The four steam generators were vertical shell, U-tube generators with an integral steam drum containing moisture separation components.

The steam generators were carbon steel vessels. Generator surfaces that contact main coolant were clad with stainless steel. Each generator tube bundle contained 1620 U-tubes constructed from 0.75 inch Type 304 stainless steel. The steam generators were constructed in accordance with the requirements of the ASME Code, Section VIII, under which they were classified as unfired pressure vessels.

A.2.2 Pressurizer

The pressurizer vessel was constructed of carbon steel with stainless steel cladding on the inside surfaces. The vessel contained 4 heater assemblies each with 12 bundles which resulted in classification as an electrically heated steam generator. Penetrations were provided at the top and the bottom of the vessel to accommodate the spray and surge lines respectively. The vessel had several penetrations in the upper head to accommodate two safety valves and one power operated relief valve.

A.2.3 Reactor Vessel Internal Components

The Reactor Vessel internal components were constructed from Type 304 stainless steel. The following were the major reactor vessel internal components:

- Thermal shield - The thermal shield was a 3 inch thick cylindrical sheet constructed from four 90° bolted segments. Each bolted joint was reinforced with a seam clamp. The thermal shield was supported by eight lugs welded to the inside wall of the reactor vessel. The thermal shield was designed to limit thermal stresses on the reactor vessel wall from gamma heating and to attenuate fast neutron flux that escaped the core.

- Lower core support barrel - The lower core support barrel was composed of two cylinders bolted together, the lower core support plate, and the control rod shroud tubes. The upper cylinder had a machined lip that fit into a groove machined in the reactor vessel flange. The upper cylinder wall contained four penetrations corresponding to the reactor vessel outlet nozzles. The wall also formed the inner wall of the cold leg downcomer. The lower cylinder was bolted to the upper cylinder and contained the core baffle assembly.

- Lower core support plate - The lower core support plate was constructed of two perforated plates joined together by sleeves and a spacer ring to form an 8 inch high support structure. The plate supported the fuel assemblies. Openings in the plate allowed main coolant flow into the fuel assembly inlet nozzles and into the interior of the plate to minimize thermal stresses. Additional openings in the plate guided the movement of control rod followers into and out of the core.

- Control rod shroud tubes - The control rod shroud tubes consisted of 24 cylinders bolted to the bottom of the shroud tube support plate. A lower tie plate secured the bottom of the shrouds preventing excess movement. The entire assembly was bolted to the bottom of the lower core support plate. The shroud tubes guided the movement of control rod followers when they were not inserted in the core.

- Core baffle - The core baffle was a form fit structure that transitioned from the round core barrel to the squared-off periphery of the core. Horizontal baffle plates were bolted into circular grooves located in the core barrel wall. Vertical baffle plates were bolted to the horizontal plates. Twenty spacers were attached to the vertical plates to prevent coolant bypass through channels formed by cut-outs on the outside of periphery fuel assemblies. A flow restricting plug was located between the lower core support plate and the spacer to preclude impingement of fluid from behind the baffle on fuel assemblies located adjacent to the baffle.

- Core barrel - The core barrel was a cylindrical shell that hung from a ledge in the reactor vessel between the core baffle structure and the thermal shield. The upper rim of the core barrel and the upper flange of the baffle were doweled and bolted to the lower rim of the lower core support barrel. The core barrel supported and contained the fuel assemblies and directed coolant flow through the core.

- Upper core support plate - The upper core support plate was constructed of two perforated plates joined together by sleeves and a spacer ring to form an 8 inch high support structure. Openings in the plate allowed main coolant flow from the fuel assembly outlet nozzles and into the interior of the plate to minimize thermal stresses. Additional openings in the plate guided the movement of control rods into and out of the core. The upper core support plate was doweled and bolted to the lower rim of the upper core support barrel. The upper core support plate transmitted uplift forces from the core flow and spring forces from the fuel assemblies to the lower core support barrel.

- Upper core support barrel - The upper core support barrel consisted of a cylinder and the upper core support plate. The top flange of the upper core support barrel rested on the upper flange of the lower core support barrel. The upper core support barrel contained the incore instrumentation package and supported the guide tube support plate.

- Incore instrumentation support structure - The incore instrumentation support structure consisted of three frames, a support plate, four guide columns, two thermocouple columns, two flux detector columns, 22 flux detector thimbles, and 26 thermocouple thimbles. The three frames, which were constructed to form a lattice, combined with the support plate to provide lateral support for the structure. The guide columns provided vertical support.

  The incore instrumentation support structure telescoped the flux detector thimbles into and out of the core during structure installation and removal. For removal, the movable frames and support plate were pulled upwards, removing the thimbles from the core. The structure was then locked in position. The incore instrumentation support structure was removed by lifting the upper core support barrel which contained the support structure.

- Guide tubes - Twenty four control rod guide tubes were positioned between the upper core support plate and the guide tube support and hold down plate. The guide tubes enclosed and guided movement of the control rods and the control rod drive shafts.

- Drive shafts - Twenty four drive shafts were connected to the top of the control rod absorber section. The drive shafts interfaced with the control rod drive mechanisms located in the reactor head to position the control rods in the reactor core.

- Guide Tube Support Plate and Hold Down Ring - The guide tube support plate and hold down ring was located on the top of the upper core support barrel. The plate and hold down ring maintained the guide tubes in position above the reactor core.

A.3 PROJECT OVERVIEW

The Component Removal Project was implemented to remove the four steam generators, the pressurizer, and the Reactor Vessel internal components. Each of these components with the exception of certain Reactor Vessel internal components were removed from the site. The Reactor Vessel internal components (i.e., core baffle, lower core support plates) that have radionuclide concentrations in excess of the 10 CFR Part 61 Class C limits were loaded into containers and stored in the Spent Fuel Pit. These components will be shipped to a high level waste disposal site.

All major piping connections to the steam generators and the pressurizer were cut and capped with welded plates before removal of the components. The main steam and feedwater lines to the steam generators were cut and capped with a welded carbon steel plate at the bioshield wall. The main coolant lines to the steam generators were cut and capped with a welded stainless steel plate near the steam generator inlet and outlet nozzles. The pressurizer surge and drain lines were cut and capped with welded stainless steel plates. The pressurizer spray line was cut and covered with a temporary cover.

Several other activities were also completed during the Component Removal Project. The Polar Crane was returned to its original design lifting capacity of 150 tons. A new crane was installed in the Vapor Container to facilitate movement of materials through the equipment hatch. Asbestos was removed from the components removed in the project as well as the main steam and feed lines in the Vapor Container.

All removal activities were controlled through the Engineering Design Change Request (EDCR) process. The following EDCRs were initiated to complete the Component Removal Project:

- <u>EDCR 93-301: VC Jib Crane</u> - This EDCR controlled the installation of an auxiliary crane in the Vapor Container. The crane facilitates transfer of materials through the equipment hatch.

- <u>EDCR 93-302: Steam Generators and Pressurizer Removal</u> - This EDCR controlled the removal and the on-site preparation for shipment of the steam generators and pressurizer.

- <u>EDCR 93-303: Reactor Vessel Internals Segmentation</u> - This EDCR controlled the removal, packaging, and transportation off-site of the Reactor Vessel internal components including the thermal shield. The EDCR also controlled site installations necessary for internals segmentation (e.g., cavity filtration system, materials cutting bridge, local ventilation system).

- EDCR 93-304: Core Baffle Containers - This EDCR controlled the design and fabrication of containers for the Reactor Vessel internal components that exceeded 10 CFR Part 61 Class C limits. The container external geometry is similar to that of a fuel assembly.

- EDCR 93-305: Shield Tank Cavity Modifications - This EDCR controlled modifications to the Shield Tank Cavity to reduce leakage when the cavity was filled with water to support Reactor Vessel internal components removal activities. The modification installed steel plates over the gap between the Reactor Vessel and the Neutron Shield Tank.

After completion of the Component Removal Project, the Reactor Vessel head was reinstalled. The Main Coolant System was drained and vented. The Reactor Vessel was drained and vented.

REFERENCES

A.1-1 Issuance of Amendment No. 142 to Facility Operating License No, DPR-3 - Yankee Nuclear Power Station (Rowe) (TAC No. M83024), M. B. Fairtile (USNRC) to J. M. Grant, August 5, 1992.

A.1-2 M921124, Staff Requirements - Briefing By OGC on Regulatory Issues and Options for Decommissioning Proceedings (SECY-92-382), S. J. Chilk (USNRC) to W. C. Parler (USNRC) / J. M. Taylor (USNRC), January 14, 1993.

A.1-3 BYR 93-031, Activities Prior to Decommissioning Plan Approval - NRC Request for Additional Information, J. K. Thayer to M. B. Fairtile (USNRC), April 23, 1993.

A.1-4 Activities Prior to Decommissioning Plan Approval (TAC No. M86283), S. H. Weiss (USNRC) to J. K. Thayer, July 15, 1993.

APPENDIX B

SUMMARY OF SIGNIFICANT RADIOLOGICAL CONTAMINATION EVENTS

SUMMARY OF SIGNIFICANT RADIOLOGICAL CONTAMINATION EVENTS

Date	Description of Occurrence	Radiological Consequences
9/20/61 Incident Report No. 61-15	A main coolant sample container was dropped between the Primary Auxiliary Building and the Waste Disposal Building. The contents of the container spilled on to the asphalt surface.	About 35 µCi of radioactivity was released as a result of the spill on the asphalt surface. Fixed contamination remaining following decontamination resulted in a contact dose of about 0.05 mR/hr.
9/18/63 Incident Report No. 63-12	A sampling valve located over the Ion Exchange Pit was inadvertently left open when filling the Shield Tank Cavity from the Safety Injection Tank. Water from the tank spilled onto the cover of the Ion Exchange Pit. A portion of the water drained from the cover onto the asphalt on the west side of the Ion Exchange Pit.	Radiation levels of the spill area were 70-100 mR/hr measured 1 inch from the surface. Ion Exchange Pit cover contamination levels were 1,000,000-10,000,000 dpm measured over several square inches. Asphalt contamination levels were 20,000-60,000 dpm/ft^2. Both the Ion Exchange Pit cover and the asphalt were decontaminated.
10/8/63 Incident Report No. 63-17	Several small holes were detected on the bottom of a leakage collection drum located in a storm drain catch basin outside of the Spent Fuel Building. The drum was used to collect leakage from the Fuel Transfer Chute Dewatering Pump.	Water with a radioactivity level of about 6E-5 µCi/ml leaked from the drum into the catch basin. The basin was pumped to the Waste Disposal System and flushed with Service Water.
9/3/64 Incident Report No. 64-8	A spill of main coolant occurred while filling and pressurizing the shutdown cooling pump gland seal water tank. Main coolant flowed out the vent connection into a relief valve discharge header and on to the Primary Auxiliary Building roof.	About 270 µCi of radioactivity was released as a result of the spill. Several samples were taken: Seal tank drain 2E-3 µCi/ml PAB roof 1E-3 µCi/ml Storm drain 1E-6 µCi/ml Predominant isotopes were Co-58, Co-60 and Mn-54. The area was decontaminated. Several years after the event, a new roof was installed and the old roofing material was disposed of as radioactive material.

SUMMARY OF SIGNIFICANT RADIOLOGICAL CONTAMINATION EVENTS

Date	Description of Occurrence	Radiological Consequences
10/3/64 AOR No. 64-13	A valve was inadvertently left open after adding water to the Ion Exchange Pit. Water continued to flow into the Ion Exchange Pit from the Primary Water Storage Tank. Water leaked from the Ion Exchange Pit into the ground resulting in water coming up through the asphalt on the west side of the pit.	Leakage from the Ion Exchange Pit contaminated the ground and asphalt on the west side of the Ion Exchange Pit. The water had a specific activity of 8E-8 μCi/ml and was below the maximum permissible concentration. The area was flushed with service water to dilute the contamination as it entered the storm sewer.
2/17/65 AOR No. 65-6	A routine chemistry sample of Ion Exchange Pit water indicated an activity level of 1.2E-5 μCi/ml. This was about 100 times above the normal level. The cause was identified as a leaking outlet connection on a cation bed that was in main coolant service. The activity decreased by about 50% within a day after isolating the leak. The connection was installed 19 days before the leak was identified.	Analysis indicated that the activity was comprised primarily of F-18 (half life = 1.8 hr) and an unknown isotope with a gamma energy of 0.14 MeV (half life = 40 hr). Based on an existing leak from the Ion Exchange Pit to the soil, about 3,400 μCi in 74,000 gallons of water was released to the area beneath the Ion Exchange Pit over the 19 day period.
3/25/65 AOR No. 65-8	A sample of Ion Exchange Pit water indicated a high activity level, 2E-4 μCi/ml. The cause was identified as a leak from an anion exchange capsule and filter that were installed the previous day. Main Coolant System purification flow was stopped and all capsules were isolated.	Main coolant leaked into the Ion Exchange Pit, increasing the radioactivity level. After the leak was isolated, a feed and bleed dilution of the Ion Exchange Pit was established. About 88,000 gallons of demineralized water was added to the Ion Exchange Pit.

SUMMARY OF SIGNIFICANT RADIOLOGICAL CONTAMINATION EVENTS

Date	Description of Occurrence	Radiological Consequences
5/64 - 5/65 Operation Reports Nos. 44-53	Significant Ion Exchange Pit leakage was identified in May 1964. An unsuccessful attempt to isolate the leak was made in July 1964 by installing and sealing a concrete plug in the pit sump. Following that attempt, a decision was made to empty the resin capsules and to drain the pit. This activity was completed in April 1965. In May 1965, a crack in a vertical joint at the northwest corner of the Ion Exchange Pit was found after draining the pit. The crack was repaired and the Ion Exchange Pit floor and walls were sealed to prevent further leakage. In 1965, tritium was detected in Sherman Spring. The presence of tritium was attributed to migration of tritium from the Ion Exchange Pit into the groundwater. Section 3.1.5 presents additional information regarding the activity in Sherman Spring.	Water from the Ion Exchange Pit leaked into the soil below the pit during the period between May 1964 and April 1965. The average activity during this period was 7E-6 µCi/ml resulting in a total release of about 36,000 µCi of identifiable activity. Greater than 95% of this activity was due to Cr-51, Mn-54, Fe-59, Co-58, Co-60, Ag-110m, and Hf-181. Based on the average activity and the lowest maximum permissible concentration, the release from the Ion Exchange Pit was less than 12% of the maximum permissible concentration for unrestricted areas. Based on soil characteristics it is unlikely that this activity migrated to an unrestricted area. The amount of tritium attributed to this leakage has been estimated to be less than 60 Ci.
4/8/66 AOR No. 66-3	Main coolant was inadvertently aspirated into the Primary Vent Stack fan from the Low Pressure Vent Header during outage related Reactor Vessel venting operations. Main coolant leaked from the riveted joints in the fan duct work onto the Primary Auxiliary Building Mechanical Equipment Room floor.	Main coolant was spilled with a radioactivity level of 2.26 µCi/ml tritium and 3.4E-2 µCi/ml gross beta-gamma. Smears of the affected area indicated 100,000-1,000,000 dpm/ft^2. A survey of the vent line and the duct work indicated contact radiation levels of 10-70 mR/hr on horizontal runs and 10-25 mR/hr on vertical runs. The area was decontaminated.

SUMMARY OF SIGNIFICANT RADIOLOGICAL CONTAMINATION EVENTS

Date	Description of Occurrence	Radiological Consequences
9/27/66 AOR No. 66-7	The Spent Fuel Pit overflowed when a Primary Water Storage Tank valve was inadvertently left open. Most of the water flowed from the building, down the north exterior wall, over a small section of asphalt paving and into a storm drain. A few gallons leaked into the New Fuel Vault.	A sample of the water that overflowed from the Spent Fuel Pit indicated 3.2E-5 μCi/ml gross activity and 5.4E-3 μCi/ml tritium. This occurrence resulted in a total release of 4 μCi gross beta-gamma activity and 670 μCi of tritium. Affected areas were decontaminated, the storm drain was flushed with about 75,000 gallons of service water.
9/27/66 AOR No. 66-8	A sample of the west side storm drain culverts indicated elevated activity levels. This culvert was not affected by the Spent Fuel Pit spill. The source of activity was traced to a leak from the Safety Injection Tank heating system safety valve located in the Primary Auxiliary Building. Floor drains in this section of the building discharge to a storm drain on the south side of the building which discharge into the west culvert.	Samples collected from the west side culvert indicated 6.7 E-7 μCi/ml average gross beta-gamma activity. The Safety Injection Tank water analysis indicated 3E-5 μCi/ml gross beta-gamma activity and 1.1E-1 μCi/ml tritium. About 0.8 μCi gross beta-gamma activity and 3.32 μCi of tritium were released into the west culvert. Affected surface areas were decontaminated.
11/1/66 AOR No. 66-9	A temporary hose failed during a routine drainage operation on the Fuel Transfer Chute pump discharge line. The spilled water drained into a storm drain served by the east culvert.	The spilled liquid had an activity of 3E-3 μCi/ml, resulting in a release of about 113 μCi into the storm drain. The storm drain was flushed with about 250 gallons of water.
2/18/72 AOR No. 72-3	Moisture was detected under insulation on the Test Tank level indicator. A leak developed from the indicator to the ground at the base of the tank coincident with the discovery. The leak was contained within 15 minutes of discovery.	The leakage from the tank had a radioactivity content of 0.0004 μCi beta-gamma and 2,017 μCi tritium. About 4 ft^3 of crushed stone and soil at the base of the tank was removed for disposal. Analysis of the soil following removal indicated no radioactivity above background levels.

SUMMARY OF SIGNIFICANT RADIOLOGICAL CONTAMINATION EVENTS

Date	Description of Occurrence	Radiological Consequences
7/16/75 PIR No. 75-7	While walking near the Ion Exchange Pit a technician received an alarm on a portable radiation detection monitor. Upon investigation, an area of bare soil with radioactive contamination of about 500,000 dpm was detected. Contamination appeared to be limited to several square feet of soil with most of the activity limited to an area of about 1 ft^2. A complete surface survey within the plant protected area fence was initiated using RM-14 radiation detectors with HP-210 probes held approximately 1 inch off the ground. Fourteen additional areas of contamination were identified. Ten of the fourteen areas were located on the non-radiological side of the plant.	Several sources of contamination were identified: 1) single specks of radioactive material, 2) soil contaminated with relatively low activity to a depth of several inches, and 3) pieces of contaminated materials (e.g. wood, concrete, polyethylene) on top of the ground. Low levels of radioactivity were also detected in the normal rain water runoff paths. Analyses of several soil samples indicated that the radionuclides consisted of nuclides with half-lives greater than one year (predominately Co-60). The contamination most likely was deposited several years prior to detection. Contaminated soil and debris were removed, including soil at the bottom of the storm drain catch basins. Any remaining soil contaminated with residual, low level radioactivity was sprayed with an asphalt sealer and covered with a thin layer of soil. Paved areas were swept and sealed with an asphalt sealer.
8/10/77 PIR No. 77-10	Routine analysis of the laboratory demineralized water supply indicated the presence of tritium. The cause was inferred to be a valve line-up error that allowed main coolant to flow into the demineralized water header.	Contaminated water was detected in the Demineralized Water Storage Tank and the Auxiliary Boiler Condensate Tank. The estimated activity released to the secondary plant was 4,000 μCi tritium and 75 μCi non-volatile activity. The Demineralized Water Storage Tank was discharged as a permitted release and refilled with clean water. Normal boiler drum blowdown reduced the radioactivity to undetectable levels.

SUMMARY OF SIGNIFICANT RADIOLOGICAL CONTAMINATION EVENTS

Date	Description of Occurrence	Radiological Consequences
12/21/77 PIR No. 77-16	A pipe from the Service Building sump to the Gravity Drain Tank was severed during core boring activities. The sump tanks receive water from the Radiation Control Area sinks, chemistry laboratory, and machine shop drains. A mechanical float-switch assembly starts and stops the pump which transfers liquid from the sump tank to the gravity drain tank. Water was released from the system before the pump could be secured.	The contaminated water released the following radioactivity into the soil: I-131 16.50 μCi I-133 2.76 μCi Cs-134 0.34 μCi Cs-137 0.50 μCi Co-60 0.58 μCi The soil at the location of the rupture was removed and disposed of.
10/24/79 PIR No. 79-4	Routine analysis of the auxiliary plant systems indicated the presence of tritium. The cause was inferred to be valve leakage between the Test Tank transfer pumps and the Primary Water Storage Tank. Water was transferred to the Auxiliary Boiler Condensate Tank, and the Demineralized Water Storage Tank during normal make-up operations.	The radioactivity level of water in the demineralized water header was very low: 5E-5 - 3E-4 μCi/ml tritium and 1E-8 - 1E-7 μCi/ml Cs-134, Cs-137, Co-60, Mn-54 activity.
8/6/80 PIR No. 80-9	Several gallons of contaminated water and about one quart of resin were released through a pinhole leak in a hose while transferring spent resin from the Ion Exchange Pit into a shipping cask.	Resin contact radiation levels were less than 1 mR/hr and spilled liquid readings were about 300,000 dpm/100 cm^2. The spill area was decontaminated to remove contamination, including excavation of some asphalt.

SUMMARY OF SIGNIFICANT RADIOLOGICAL CONTAMINATION EVENTS

Date	Description of Occurrence	Radiological Consequences
5/15/81 PIR No. 81-9	The reactor vessel head was bumped against the side of the equipment hatch during removal from the Vapor Container. Contaminated material from the underside of the head was released and fell to the asphalt below the equipment hatch.	General contamination levels on the asphalt below the equipment hatch were 1,000-500,000 dpm/100 cm^2. An area about 30 ft by 50 ft was contaminated with a total activity of about 250 μCi. About 10 μCi was discharged to Sherman Pond when rain washed the radioactive material into the east storm drain before the area could be decontaminated.
9/10/84 PIR No. 84-16	A failed PVC drain line from the Waste Disposal Building was discovered during soil excavation activities. The drain line had six pipe joints, each of which leaked apparently due to failure of the solvent welds.	Soil below one of the pipe joints was significantly contaminated. Analysis indicated 50,000 dpm over the area of maximum contamination. One "hot spot" contained 29,300 pCi/g Co-60. Average Co-60 activity, measured 2 feet below the pipe joint was about 2,100 pCi/g. Average Cs-137 activity at this location was about 17 times less than the average Co-60 activity. Soil under all pipe joints was removed to a depth of 5-7 feet below plant grade. About 420 cubic feet of soil and rock were removed and disposed of as radioactive waste. All areas above the excavation were sealed under a concrete cap.
12/14/91 PIR No. 91-7	Analysis of the laboratory demineralized water supply, the Demineralized Water Storage Tank, and the Auxiliary Boiler Condensate Tank indicated the presence of tritium and boron. No activity was found in the Primary Water Storage Tank. The root cause of the event could not be determined.	About 840 μCi of tritium was leaked into the demineralized water supply. Water was discharged from the Demineralized Water Storage Tank as a permitted release and the demineralized water header was flushed.

APPENDIX C

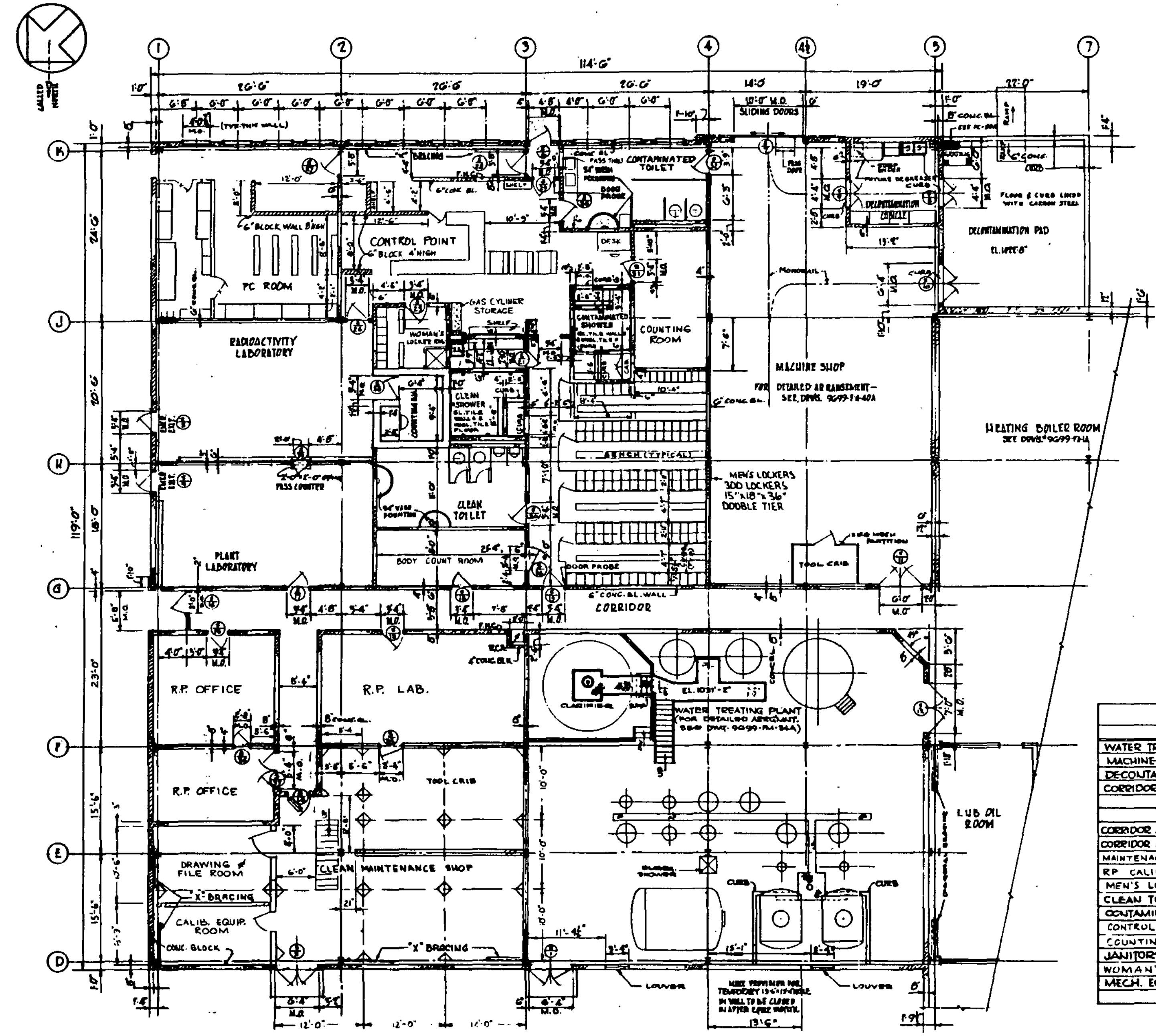
PLANT DRAWINGS REFERENCED IN THE DECOMMISSIONING PLAN

APPENDIX C

APPENDIX C

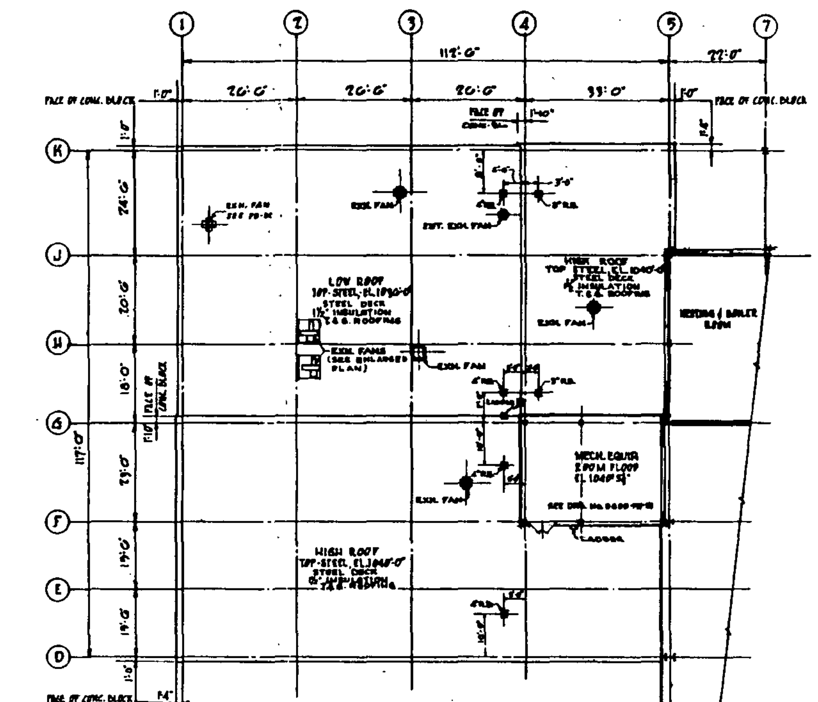
APPENDIX C

9699-FA-1E



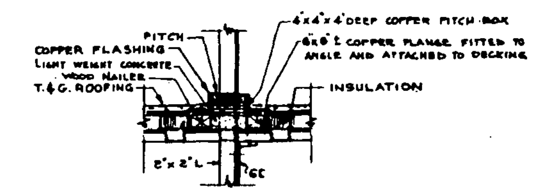
FLOOR PLAN-EL.1022'-8"

NOTES:
1. NO COPPER SCREEN SHIELDING TO BE PROVIDED FOR COUNTING ROOM.
2. FOR LABORATORY FURNITURE LAYOUTS SEE DRWG. NO 9699-FA-12A

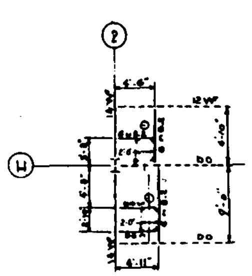


ROOF PLAN

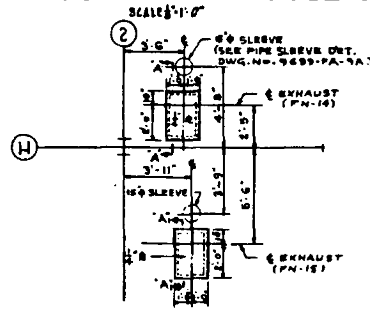
SCHEDULE OF FLOOR FINISHES				CEILING SCHEDULE	
ROOM	FLOOR FINISH	ROOM	FLOOR FINISH	TYPE OF CEILING	CLEAR HGT.
WATER TREATING PLANT	1" GRANOLITHIC	CORRIDOR ALONG 2 LINE BETW. K & J	EPOXY		
MACHINE SHOP	" "	CORRIDOR ALONG K LINE BET 2 & 3 LINE	EPOXY		
DECONTAMINATION CUBICLE	" "	RADIOACTIVITY LABORATORY	EPOXY		
CORRIDOR ALONG G LINE BET. 4 & 5 LINES	" "	PLANT LABORATORY	EPOXY		
	" "	PC ROOM & RESPIRATOR STORAGE	EPOXY		
		COUNTING ROOM	EPOXY		
CORRIDOR ALONG G LINE BETW 1 & 4 LINES	MONOLITHIC GRANOLITHIC				
CORRIDOR ALONG 3 LINE	EPOXY				
MAINTENANCE SHOP & OFFICE	[illegible]	RP OFFICES	ASPHALT TILE	ACOUSTIC TILE (SUSPENDED)	9'-0"
RP CALIBRATION	SHEET VINYL				
MEN'S LOCKER ROOM	[illegible]	BODY COUNT ROOM	GREASE PROOF ASP TILE	[illegible]	9'-0"
CLEAN TOILET					
CONTAMINATED TOILET	EPOXY				
CONTROL POINT	" "	CONTAMINATED SHOWER	[illegible] CERAMIC TILE	CEMENT PLASTER (SUSPENDED)	8'-0"
COUNTING ROOM	" "	CLEAN SHOWER	" " " "	"	8'-0"
JANITORS CLOSETS	[illegible]				
WOMAN'S LOCKER ROOM	EPOXY				
MECH. EQUIP. ROOM	MONOLITHIC GRANOLITHIC				



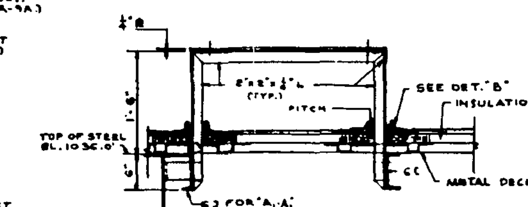
DETAIL "B"



FRAMING PLAN



ENLARGED PLAN (ROOF FANS)



SECTION "A-A"
SECT. A1-A1 SIMILAR EXCEPT AS NOTED

ISSUE	DESCRIPTION	BY	CHKD	APPR
14	REVISED PER PDCR 87-04		GC	
13	REVISED PER ME 89-2054	MM		
12	REVISED PER P.A. 8C-027 AND GENERAL UPDATE		GC	
11	REDRAWN FOR CLARITY			

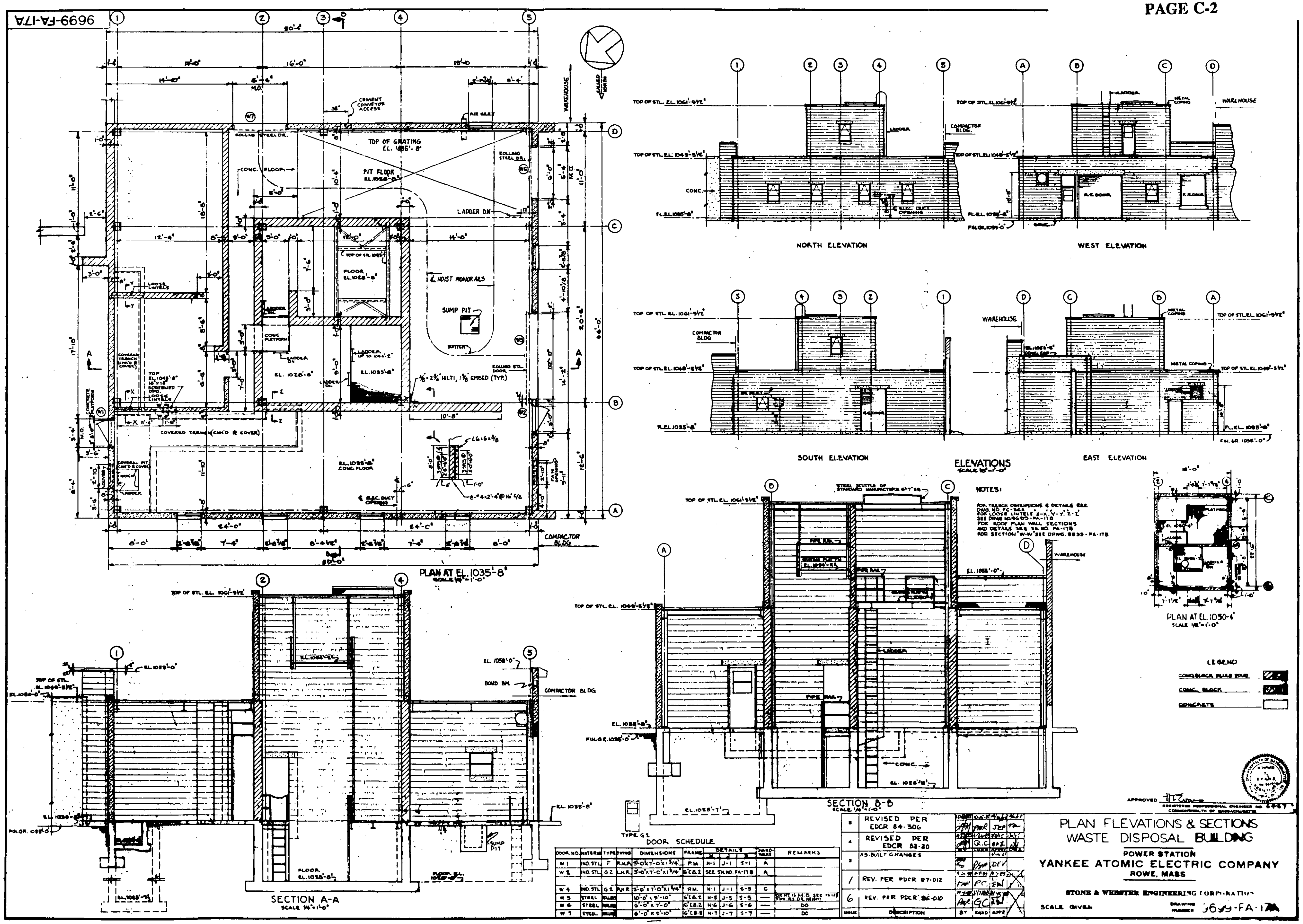
ORIGINAL DRAWING SEALED AND SIGNED BY Howard T. Evans P.E. NO. 6467

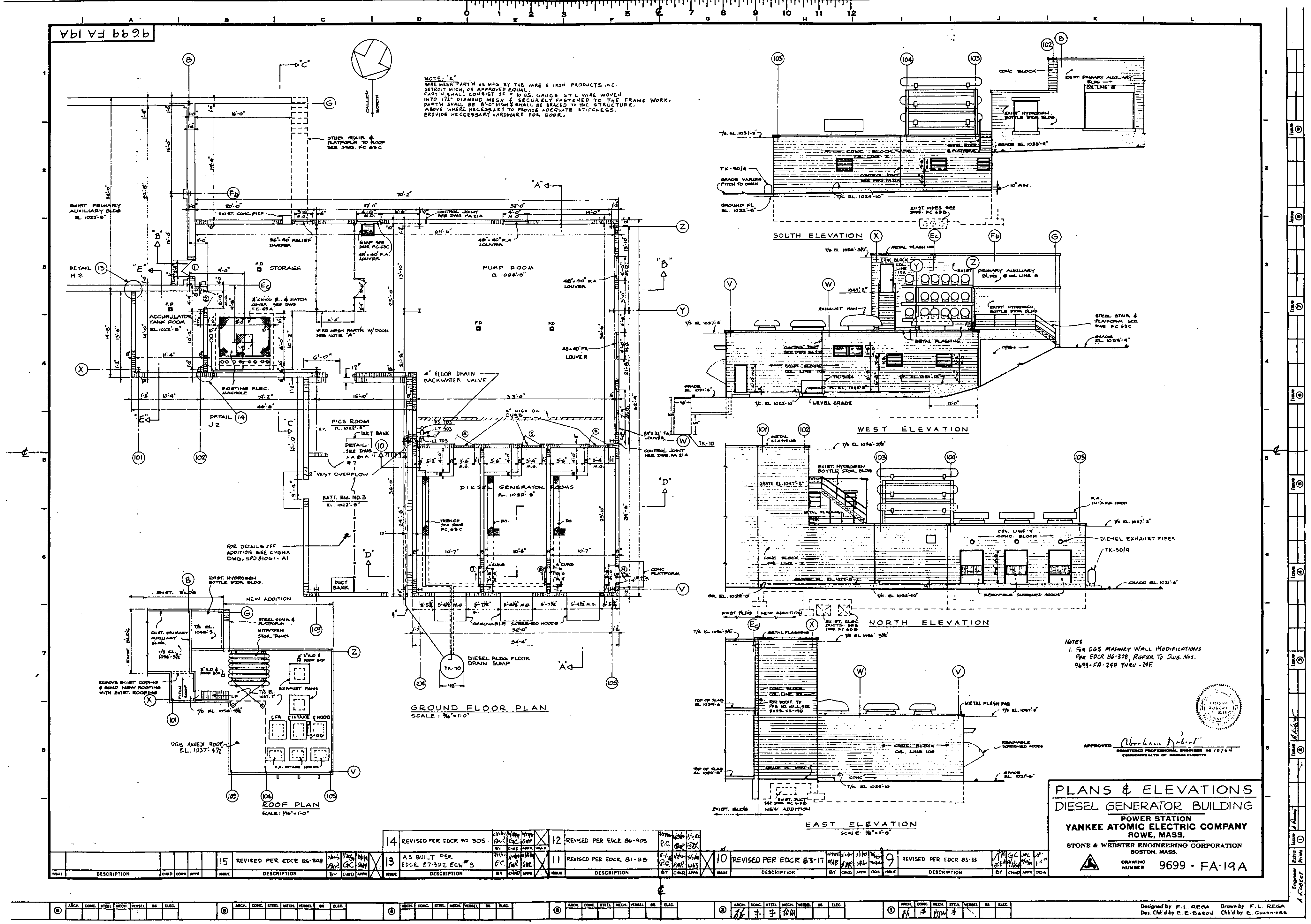
FLOOR AND ROOF PLAN
SERVICE BUILDING
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.

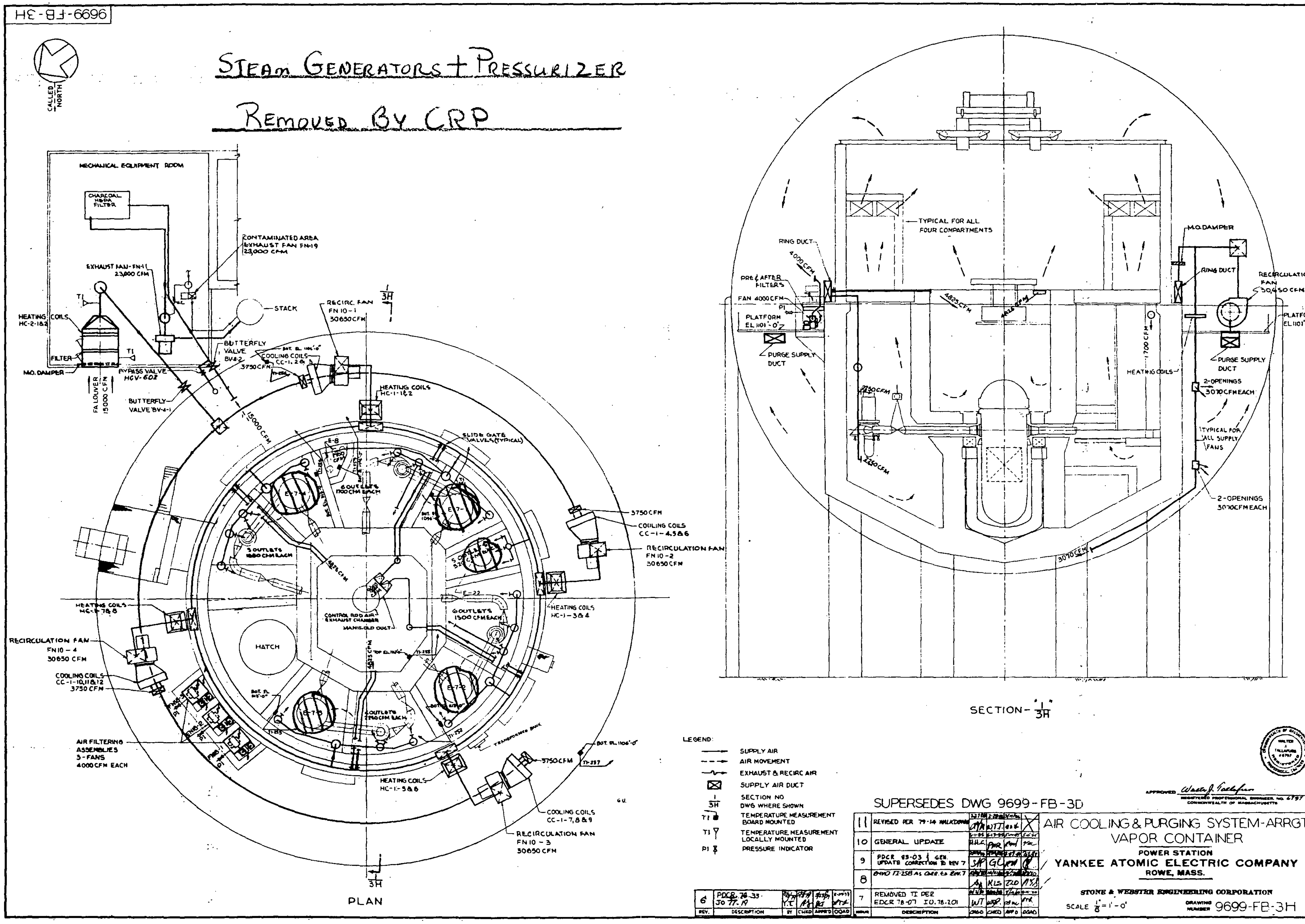
STONE & WEBSTER ENGINEERING CORPORATION

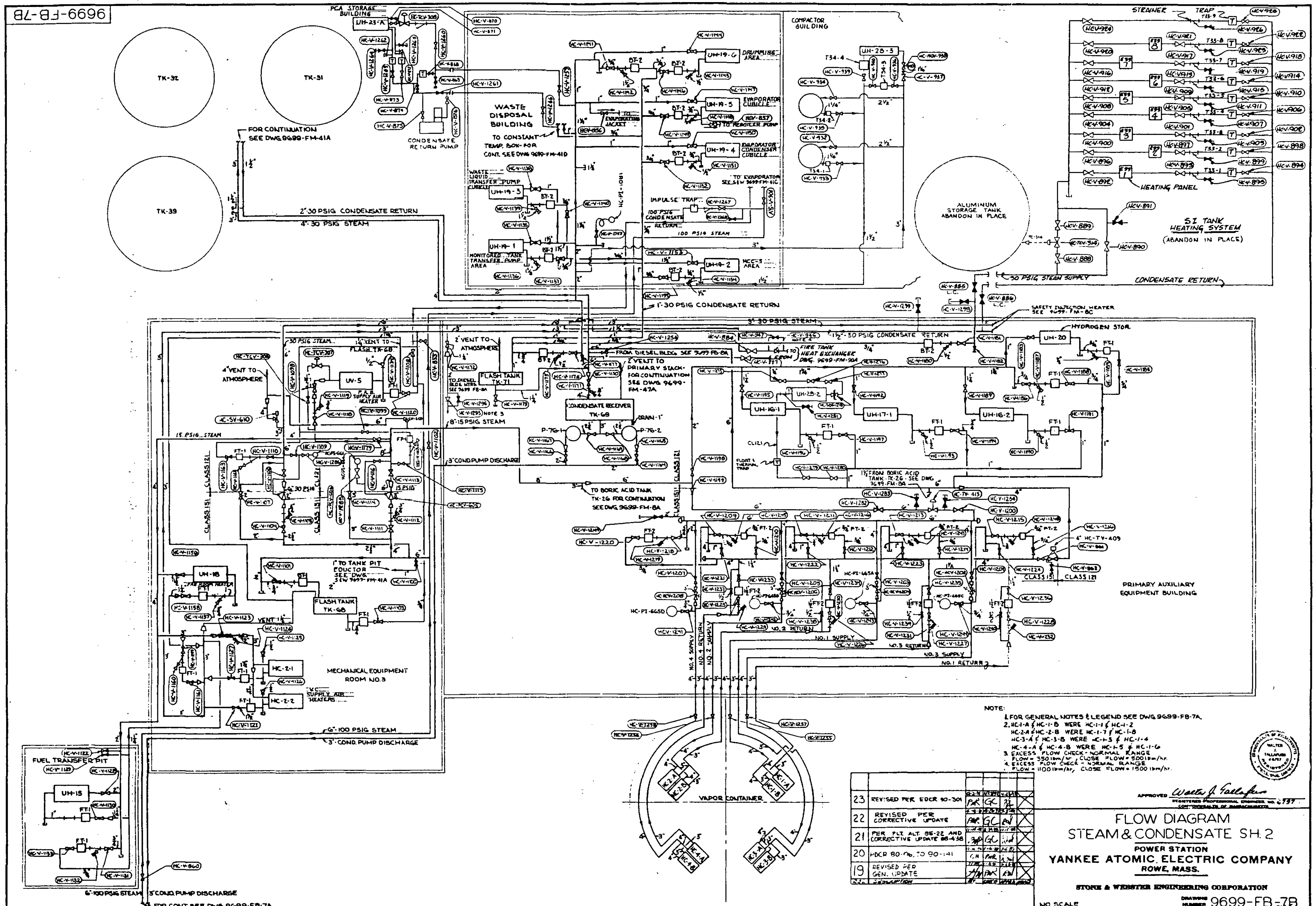
DRAWING NUMBER 9699-FA-1E

FA-17A


9699-FA-17A
PLAN AT EL. 1035'-8"
NORTH ELEVATION
WEST ELEVATION
SOUTH ELEVATION
EAST ELEVATION
ELEVATIONS
PLAN AT EL. 1050'-6"
SECTION A-A
SECTION B-B
NOTES:
LEGEND
DOOR SCHEDULE
TYPE G2
PLAN FLEVATIONS & SECTIONS
WASTE DISPOSAL BUILDING
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS
STONE & WEBSTER ENGINEERING CORPORATION
SCALE GIVEN
REVISED PER EDCR 84-306
REVISED PER EDCR 83-30
AS BUILT CHANGES
REV. PER PDCR 87-012
REV. PER PDCR 86-010
APPROVED


9699 FA 19A
NOTE: "A"
WIRE MESH PART'N AS MFG BY THE WIRE & IRON PRODUCTS INC. DETROIT MICH, OR APPROVED EQUAL.
PART'N SHALL CONSIST OF #10 U.S. GAUGE STL WIRE WOVEN INTO 1½" DIAMOND MESH & SECURELY FASTENED TO THE FRAME WORK.
PART'N SHALL BE 8'-0" HIGH & SHALL BE BRACED TO THE STRUCTURE ABOVE WHERE NECESSARY TO PROVIDE ADEQUATE STIFFNESS.
PROVIDE NECESSARY HARDWARE FOR DOOR.
PUMP ROOM
EL. 1022'-8"
ACCUMULATOR TANK ROOM
STORAGE
DIESEL GENERATOR ROOMS
PICS ROOM
BATT. RM. NO. 3
DIESEL BLDG FLOOR DRAIN SUMP
GROUND FLOOR PLAN
SCALE: 3/16"=1'-0"
ROOF PLAN
SCALE: 1/8"=1'-0"
SOUTH ELEVATION
WEST ELEVATION
NORTH ELEVATION
EAST ELEVATION
SCALE: 1/8"=1'-0"
NOTES
1. FOR DGB MASONRY WALL MODIFICATIONS PER EDCR 86-308, REFER TO DWG. NOS. 9699-FA-24A THRU -24F.
14 REVISED PER EDCR 90-305
13 AS BUILT PER EDCR 87-302 ECN#3
12 REVISED PER EDCR 86-305
11 REVISED PER EDCR 81-58
15 REVISED PER EDCR 86-308
10 REVISED PER EDCR 83-17
9 REVISED PER EDCR 83-13
ISSUE
DESCRIPTION
BY
CHKD
APPR
APPROVED
PLANS & ELEVATIONS
DIESEL GENERATOR BUILDING
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER
9699 - FA-19A
Designed by F. L. REGA
Drawn by F. L. REGA
Des. Chk'd by E. E. BARON
Chk'd by E. GUARNIERE


9699-FB-3H
CALLED NORTH
STEAM GENERATORS + PRESSURIZER
REMOVED BY CRP
MECHANICAL EQUIPMENT ROOM
CHARCOAL HEPA FILTER
CONTAMINATED AREA EXHAUST FAN FN-19 23,000 CFM
EXHAUST FAN-FN-11 23,000 CFM
STACK
HEATING COILS HC-2-1&2
FILTER
M.O. DAMPER
BYPASS VALVE HCV-602
BUTTERFLY VALVE BV-4-1
BUTTERFLY VALVE BV-4-2
FA LOUVER 15000 CFM
15000 CFM
RECIRC. FAN FN 10-1 30650 CFM
COOLING COILS CC-1-1,2 & 3
3750 CFM
HEATING COILS HC-1-1&2
SLIDE GATE VALVES (TYPICAL)
6 OUTLETS 1100 CFM EACH
3750 CFM
COOLING COILS CC-1-4,5 & 6
RECIRCULATION FAN FN 10-2 30650 CFM
5 OUTLETS 1880 CFM EACH
CONTROL ROD AIR EXHAUST CHAMBER
MANIFOLD DUCT
HEATING COILS HC-1-3&4
6 OUTLETS 1500 CFM EACH
HEATING COILS HC-1-7&8
RECIRCULATION FAN FN 10-4 30650 CFM
COOLING COILS CC-1-10,11 & 12 3750 CFM
HATCH
AIR FILTERING ASSEMBLIES 3-FANS 4000 CFM EACH
6 OUTLETS 2750 CFM EACH
HEATING COILS HC-1-5&6
3750 CFM
COOLING COILS CC-1-7,8 & 9
RECIRCULATION FAN FN 10-3 30650 CFM
PLAN
TYPICAL FOR ALL FOUR COMPARTMENTS
RING DUCT
PRE & AFTER FILTERS
FAN 4000 CFM
PLATFORM EL 1101'-0"
PURGE SUPPLY DUCT
M.O. DAMPER
RECIRCULATION FAN 30,650 CFM
HEATING COILS
2-OPENINGS 3070 CFM EACH
TYPICAL FOR ALL SUPPLY FANS
SECTION - 1/3H
LEGEND:
SUPPLY AIR
AIR MOVEMENT
EXHAUST & RECIRC AIR
SUPPLY AIR DUCT
SECTION NO. DWG WHERE SHOWN
TEMPERATURE MEASUREMENT BOARD MOUNTED
TEMPERATURE MEASUREMENT LOCALLY MOUNTED
PRESSURE INDICATOR
SUPERSEDES DWG 9699-FB-3D
11 REVISED PER 79-14 WALKDOWN
10 GENERAL UPDATE
9 PDCR 83-03 & GEN UPDATE CORRECTION TO REV 7
7 REMOVED TI PER EDCR 78-07 IO.78-201
6 PDCR 74-33 JO 77.19
AIR COOLING & PURGING SYSTEM-ARRGT.
VAPOR CONTAINER
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
SCALE 1/8" = 1'-0"
DRAWING NUMBER 9699-FB-3H



NOTE:
1. FOR GENERAL NOTES & LEGEND SEE DWG 9699-FB-7A.
2. HC-1-A & HC-1-B WERE HC-1-1 & HC-1-2
HC-2-A & HC-2-B WERE HC-1-7 & HC-1-8
HC-3-A & HC-3-B WERE HC-1-3 & HC-1-4
HC-4-A & HC-4-B WERE HC-1-5 & HC-1-6
3. EXCESS FLOW CHECK - NORMAL RANGE FLOW = 350 lbm/hr, CLOSE FLOW = 500 lbm/hr.
4. EXCESS FLOW CHECK - NORMAL RANGE FLOW = 1100 lbm/hr, CLOSE FLOW = 1500 lbm/hr.

23	REVISED PER EDCR 90-301	PAR	GC		
22	REVISED PER CORRECTIVE UPDATE	PAR	GC		
21	PER PLT. ALT. 85-22 AND CORRECTIVE UPDATE 86-438		GC		
20	PDCR 80-06, TO 90-141	CN	PAR		
19	REVISED PER GEN. UPDATE		PAR		
	DESCRIPTION	BY	CHK'D	APP'D	

APPROVED Walter J. Tallafus
REGISTERED PROFESSIONAL ENGINEER NO. 6737 COMMONWEALTH OF MASSACHUSETTS

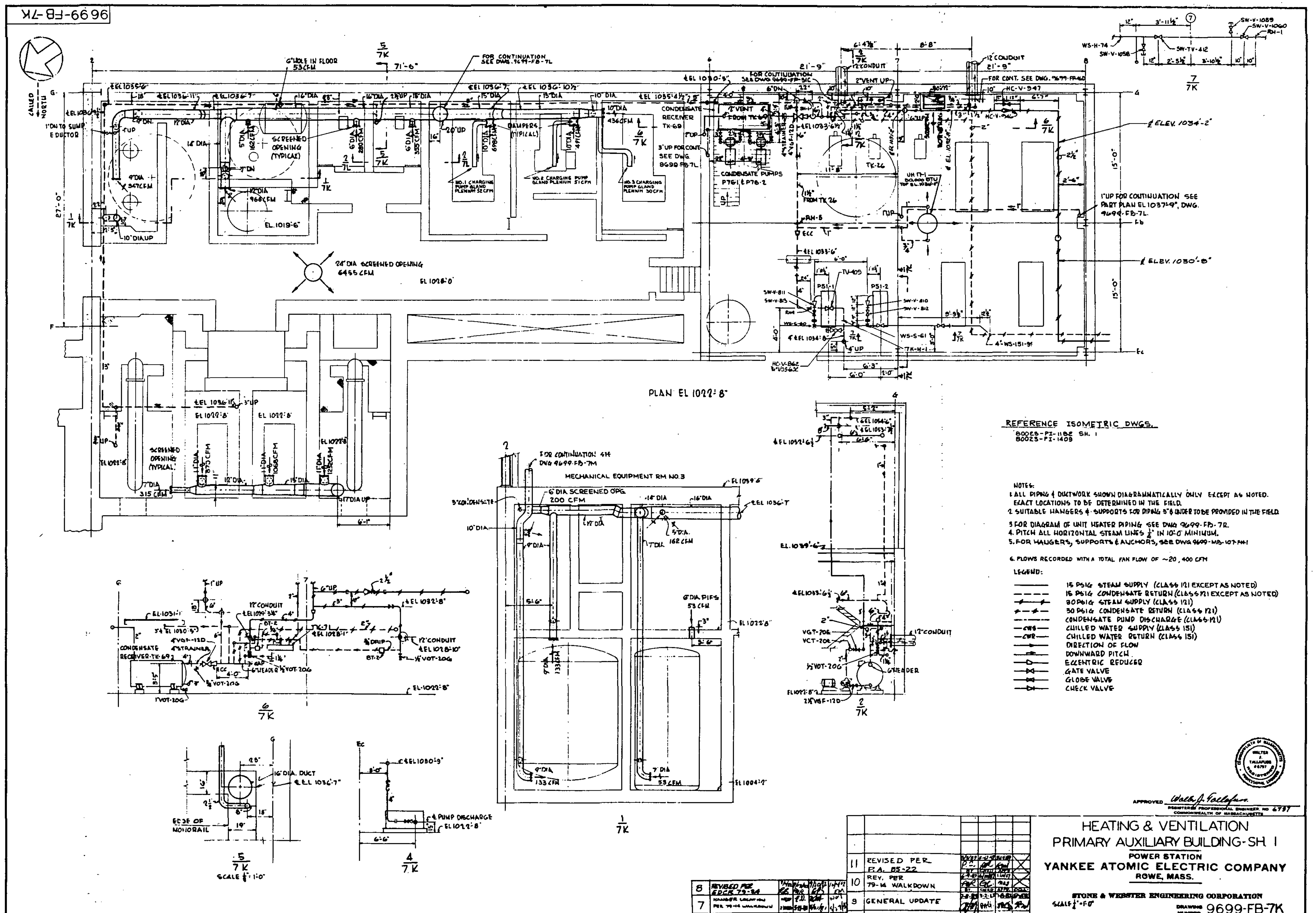
FLOW DIAGRAM
STEAM & CONDENSATE SH. 2

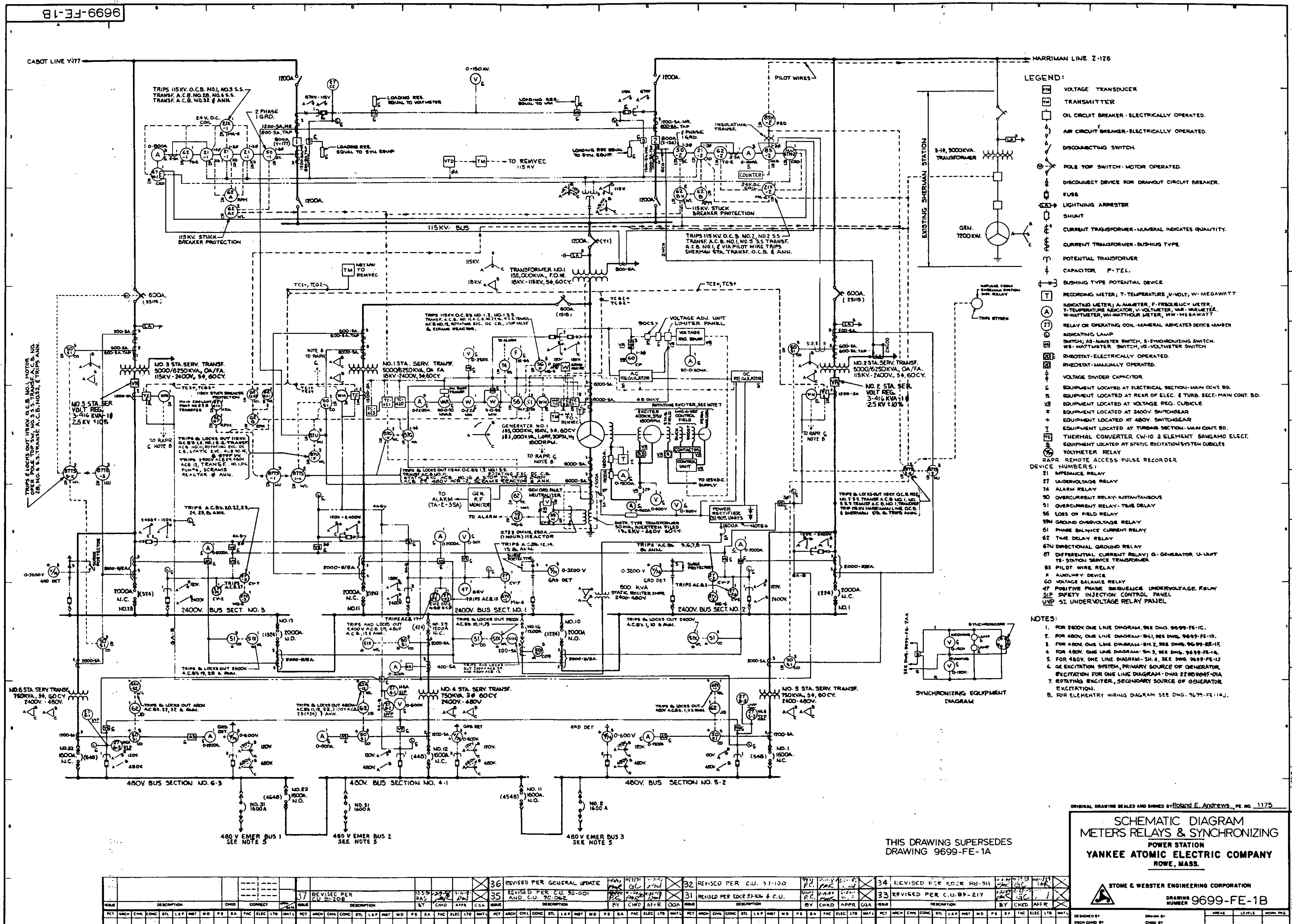
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.

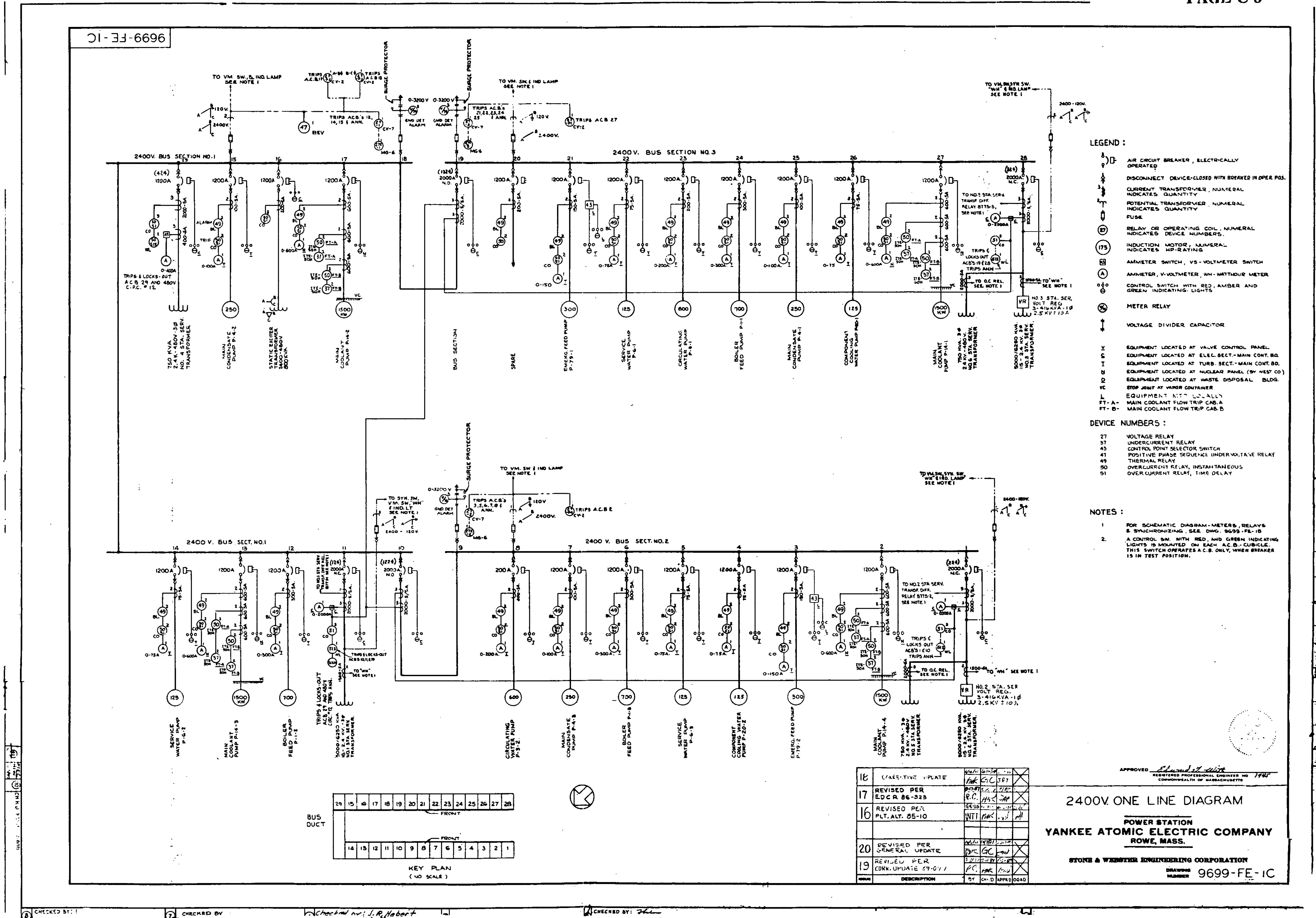
STONE & WEBSTER ENGINEERING CORPORATION

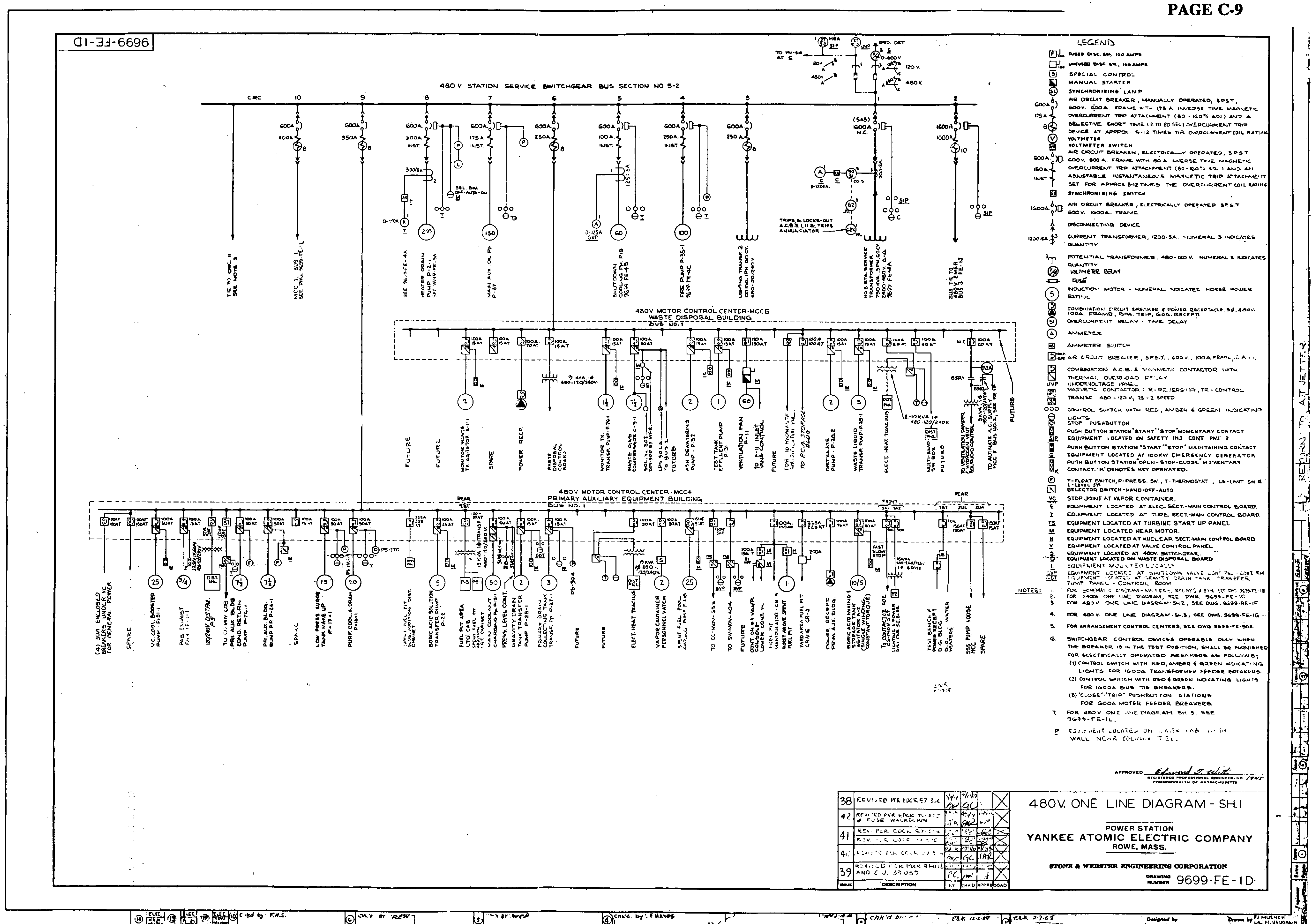
NO SCALE

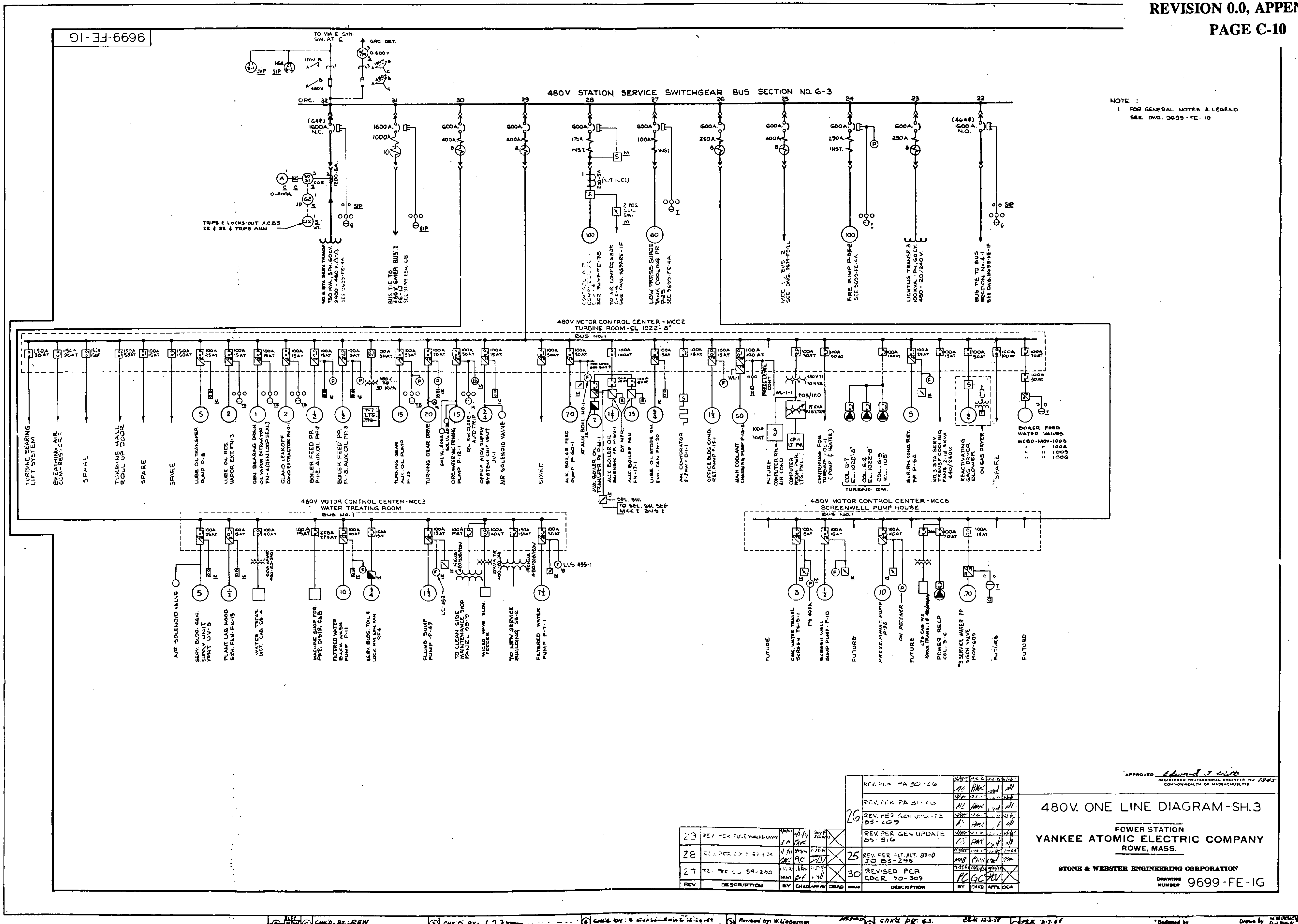
DRAWING NUMBER 9699-FB-7B

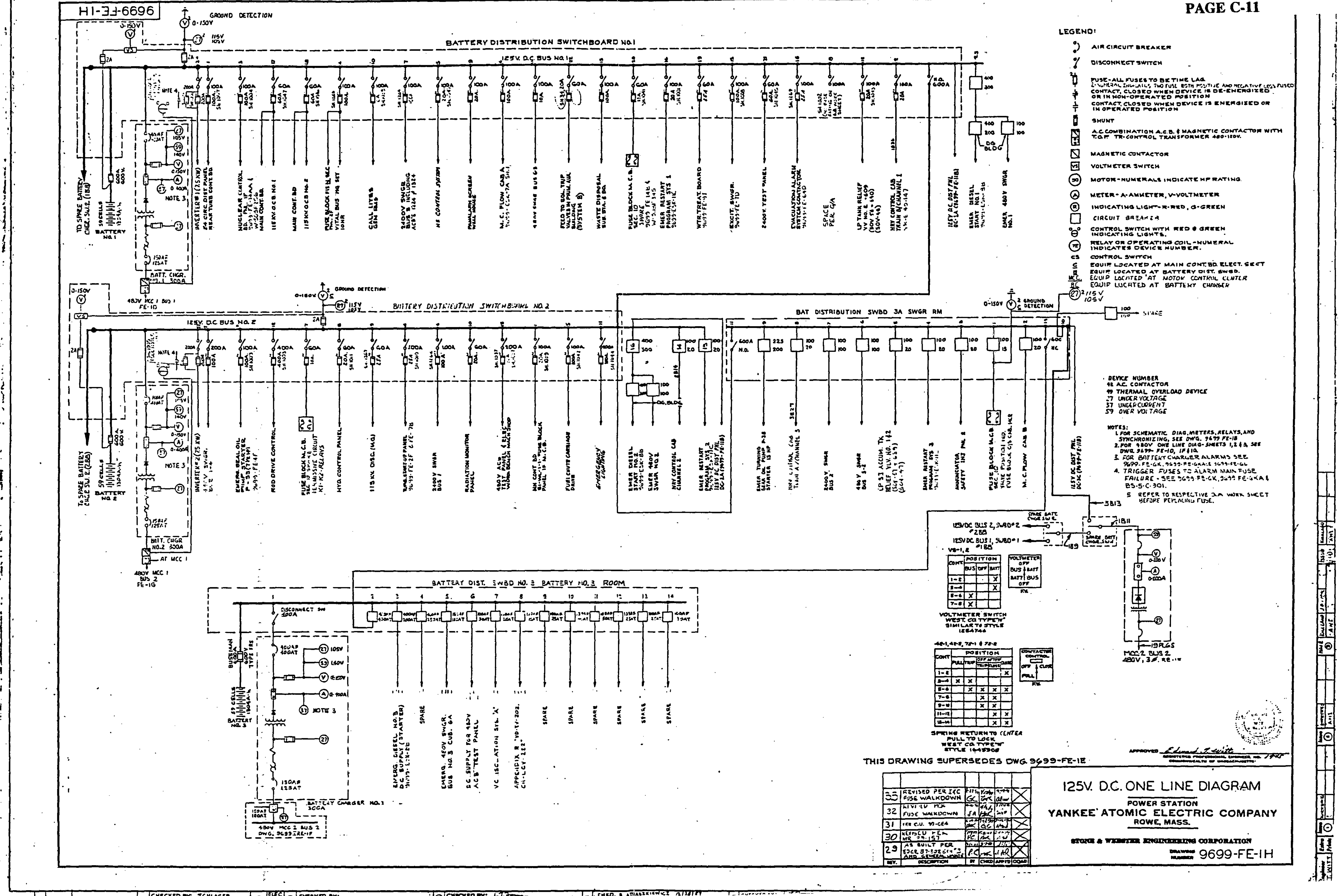

9699-FB-7K
PLAN EL 1022'-8"
REFERENCE ISOMETRIC DWGS.
HEATING & VENTILATION
PRIMARY AUXILIARY BUILDING-SH. 1
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
9699-FB-7K

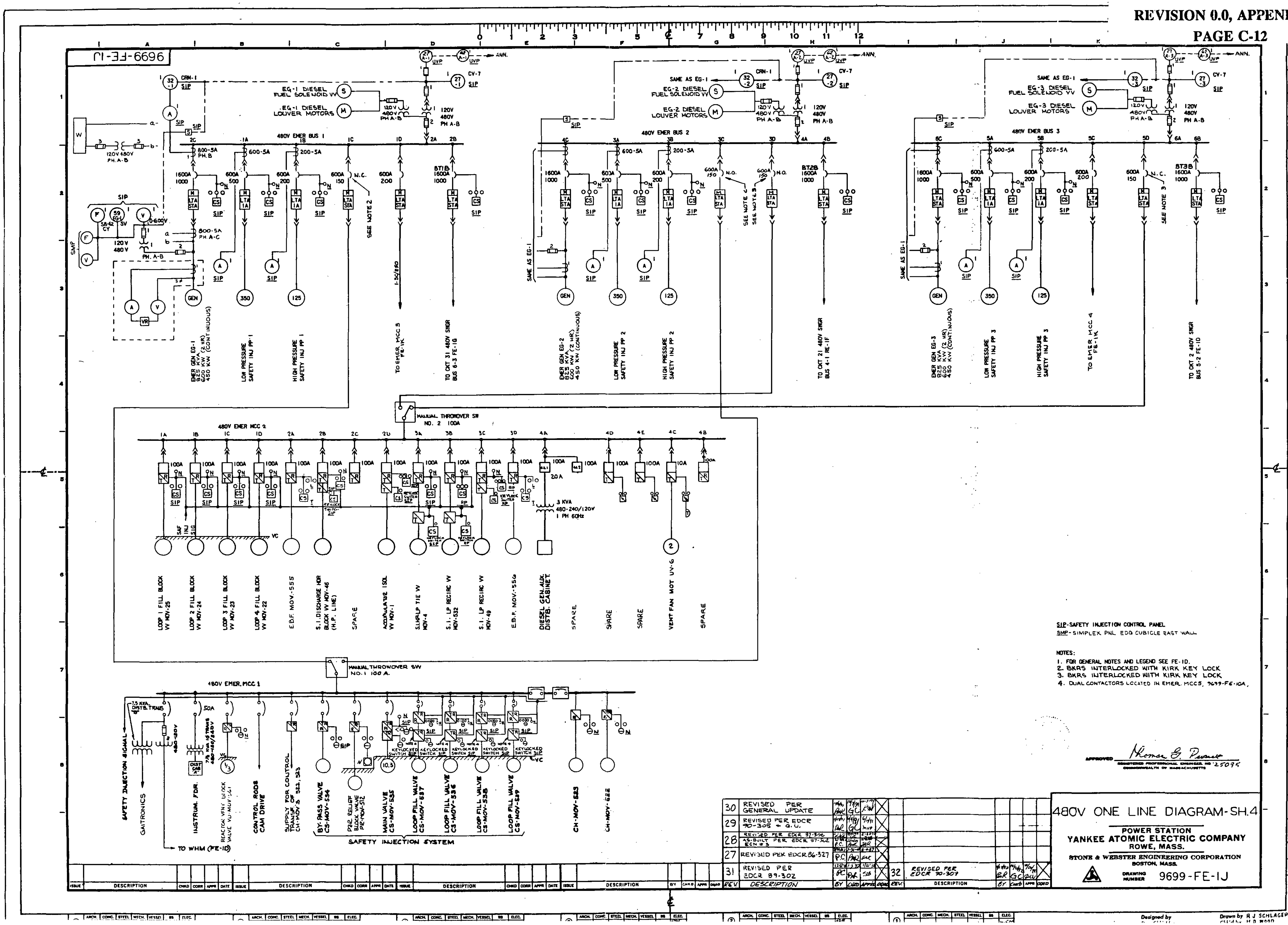

9699-FE-1B
CABOT LINE Y177
HARRIMAN LINE Z-126
115KV. BUS
TRANSFORMER NO.1
2400V. BUS SECT. NO. 3
2400V. BUS SECT. NO. 1
2400V. BUS SECT. NO. 2
480V. BUS SECTION NO. 6-3
480V. BUS SECTION NO. 4-1
480V. BUS SECTION NO. 5-2
480 V EMER BUS 1
SEE NOTE 5
480 V EMER BUS 2
SEE NOTE 5
480 V EMER BUS 3
SEE NOTE 5
EXISTING SHERMAN STATION
LEGEND:
DEVICE NUMBERS:
NOTES:
SYNCHRONIZING EQUIPMENT DIAGRAM
THIS DRAWING SUPERSEDES
DRAWING 9699-FE-1A
SCHEMATIC DIAGRAM
METERS RELAYS & SYNCHRONIZING
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
9699-FE-1B
CHARGE J.O. NO. 15386

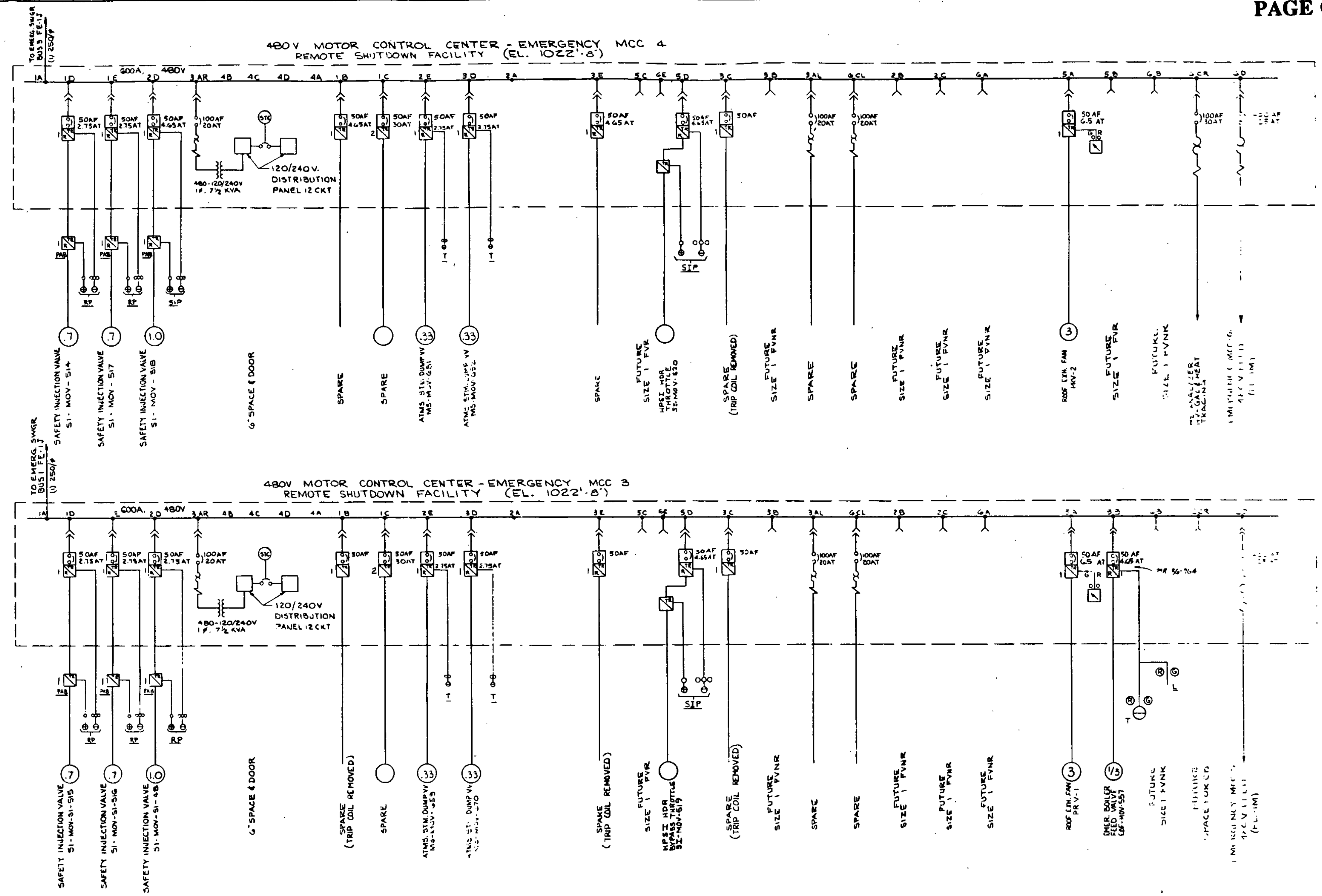

9699-FE-1C
2400V. BUS SECTION NO. 1
2400V. BUS SECTION NO. 3
2400 V. BUS SECT. NO. 1
2400 V. BUS SECT. NO. 2
LEGEND:
AIR CIRCUIT BREAKER, ELECTRICALLY OPERATED
DISCONNECT DEVICE-CLOSED WITH BREAKER IN OPER POS.
CURRENT TRANSFORMER, NUMERAL INDICATES QUANTITY
POTENTIAL TRANSFORMER, NUMERAL INDICATES QUANTITY
FUSE
RELAY OR OPERATING COIL, NUMERAL INDICATES DEVICE NUMBERS.
INDUCTION MOTOR, NUMERAL INDICATES HP-RATING
AMMETER SWITCH, VS - VOLTMETER SWITCH
AMMETER, V-VOLTMETER, WH-WATTHOUR METER
CONTROL SWITCH WITH RED, AMBER AND GREEN INDICATING LIGHTS
METER RELAY
VOLTAGE DIVIDER CAPACITOR
DEVICE NUMBERS:
27 VOLTAGE RELAY
37 UNDERCURRENT RELAY
43 CONTROL POINT SELECTOR SWITCH
47 POSITIVE PHASE SEQUENCE UNDERVOLTAGE RELAY
49 THERMAL RELAY
50 OVERCURRENT RELAY, INSTANTANEOUS
51 OVERCURRENT RELAY, TIME DELAY
NOTES:
1. FOR SCHEMATIC DIAGRAM-METERS, RELAYS & SYNCHRONIZING, SEE DWG. 9699-FE-1B
2. A CONTROL SW. WITH RED, AND GREEN INDICATING LIGHTS IS MOUNTED ON EACH A.C.B.-CUBICLE. THIS SWITCH OPERATES A.C.B. ONLY, WHEN BREAKER IS IN TEST POSITION.
RETURN TO ATLETTER
KEY PLAN (NO SCALE)
BUS DUCT
FRONT
18 CORRECTIVE UPDATE
17 REVISED PER EDCR 86-323
16 REVISED PER PLT. ALT. 85-10
20 REVISED PER GENERAL UPDATE
19 REVISED PER CONN. UPDATE
DESCRIPTION
2400V. ONE LINE DIAGRAM
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
9699-FE-1C
CHECKED BY


9699-FE-1D
480V STATION SERVICE SWITCHGEAR BUS SECTION NO. 5-2
480V MOTOR CONTROL CENTER-MCC5
WASTE DISPOSAL BUILDING
BUS NO. 1
480V MOTOR CONTROL CENTER-MCC4
PRIMARY AUXILIARY EQUIPMENT BUILDING
BUS NO. 1
LEGEND
NOTES:
480V. ONE LINE DIAGRAM - SH.1
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-FE-1D


9699-FE-1G
480V STATION SERVICE SWITCHGEAR BUS SECTION NO. G-3
NOTE: 1. FOR GENERAL NOTES & LEGEND SEE DWG. 9699-FE-1D
480V MOTOR CONTROL CENTER - MCC2
TURBINE ROOM - EL. 1022'-8"
BUS NO. 1
480V MOTOR CONTROL CENTER - MCC3
WATER TREATING ROOM
BUS NO. 1
480V MOTOR CONTROL CENTER - MCC6
SCREENWELL PUMP HOUSE
BUS NO. 1
TRIPS & LOCKS-OUT ACB'S 22 & 32 & TRIPS ANN
TURBINE BEARING LIFT SYSTEM
BREATHING AIR COMP. RES. C.C.
SPARE
TURBINE HALL ROLL UP DOOR
SPARE
SPARE
LUBE OIL TRANSFER PUMP P-8
LUBE OIL RES. VAPOR EXT. FN-3
BOILER FEED PP. P-1-2, AUX. OIL PP.2
BOILER FEED PP. P-1-3, AUX. OIL PP.3
TURNING GEAR AUX. OIL PUMP P-33
TURNING GEAR DRIVE
AIR SOLENOID VALVE
SPARE
AUX. BOILER FEED PUMP P-60-1
LUBE OIL STORE RM. EXH. FAN FN-20
OFFICE BLDG COND. RET. PUMP P-151-1
MAIN COOLANT CHARGING PUMP P-15-2
TURBINE RM.
SPARE
BOILER FEED WATER VALVES WCBO-MOV-1003, 1004, 1005, 1006
AIR SOLENOID VALVE
SERV. BLDG. GEN. SUPPLY UNIT VENT UV-5
PLANT LAB HOOD EXH. FAN FN-15
WATER TREAT. DIST. CAB. 6B-4
MACHINE SHOP FOR PWR. DISTR. CAB
FILTERED WATER BACKWASH PUMP P-11
SUMP PUMP P-47
TO CLEAN SIDE MAINTENANCE SHOP PANEL PD-9
MICRO WAVE BLDG FEEDER
TO NEW SERVICE BUILDING SB-2
FILTERED WATER PUMP P-7-1
FUTURE
CIRC. WATER TRAVEL. SCREEN TS-3-1
SCREEN WELL SUMP PUMP P-10
FUTURE
PRESS. MAINT. PUMP P-76
FUTURE
POWER RECP. COL. 5-C
SERVICE WATER PP. DISCH. VALVE MOV-609
FUTURE
FUTURE
APPROVED Edward J. Witt
REGISTERED PROFESSIONAL ENGINEER NO. 1945
COMMONWEALTH OF MASSACHUSETTS
480V. ONE LINE DIAGRAM - SH.3
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-FE-1G
REV. PER PA 50-66
REV. PER PA 51-66
REV. PER GEN. UPDATE 85-609
REV. PER GEN. UPDATE 85-516
REV. PER FLT. ALT. 83-10 JO 83-295
REVISED PER EDCR 90-309
REV DESCRIPTION BY CHKD APPR OQA


9699-FE-1H
GROUND DETECTION
BATTERY DISTRIBUTION SWITCHBOARD NO.1
125V. D.C. BUS NO.1
BATTERY DISTRIBUTION SWITCHBOARD NO. 2
125V. D.C. BUS NO. 2
BAT DISTRIBUTION SWBD 3A SWGR RM
BATTERY DIST. SWBD NO. 3 BATTERY NO. 3 ROOM
BATTERY NO. 1
BATTERY NO. 2
BATTERY NO. 3
BATT. CHGR. NO. 2 300A
BATTERY CHARGER NO. 3
LEGEND:
AIR CIRCUIT BREAKER
DISCONNECT SWITCH
SHUNT
MAGNETIC CONTACTOR
VOLTMETER SWITCH
MOTOR-NUMERALS INDICATE HP RATING
METER-A-AMMETER, V-VOLTMETER
INDICATING LIGHT-R-RED, G-GREEN
CIRCUIT BREAKER
CONTROL SWITCH
DEVICE NUMBER
48 A.C. CONTACTOR
49 THERMAL OVERLOAD DEVICE
27 UNDER VOLTAGE
37 UNDER CURRENT
59 OVER VOLTAGE
NOTES:
VOLTMETER SWITCH WEST. CO. TYPE "W" SIMILAR TO STYLE
SPRING RETURN TO CENTER PULL TO LOCK
SPARE
THIS DRAWING SUPERSEDES DWG. 9699-FE-1E
125V. D.C. ONE LINE DIAGRAM
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
9699-FE-1H
REVISED PER IEC FUSE WALKDOWN
FUSE WALKDOWN
AS BUILT PER
CHECKED BY: SCHLAGER


9699-FE-1J
480V EMER BUS 1
480V EMER BUS 2
480V EMER BUS 3
480V EMER MCC 2
480V EMER. MCC 1
MANUAL THROWOVER SW NO. 2 100A
MANUAL THROWOVER SW NO. 1 100A
EG-1 DIESEL FUEL SOLENOID VV
EG-1 DIESEL LOUVER MOTORS
EG-2 DIESEL FUEL SOLENOID VV
EG-2 DIESEL LOUVER MOTORS
EG-3 DIESEL FUEL SOLENOID VV
EG-3 DIESEL LOUVER MOTORS
LOW PRESSURE SAFETY INJ PP 1
HIGH PRESSURE SAFETY INJ PP 1
LOW PRESSURE SAFETY INJ PP 2
HIGH PRESSURE SAFETY INJ PP 2
LOW PRESSURE SAFETY INJ PP 3
HIGH PRESSURE SAFETY INJ PP 3
SAFETY INJECTION SYSTEM
SIP-SAFETY INJECTION CONTROL PANEL
SMP-SIMPLEX PNL EDG CUBICLE EAST WALL
NOTES:
1. FOR GENERAL NOTES AND LEGEND SEE FE-1D.
2. BKRS INTERLOCKED WITH KIRK KEY LOCK
3. BKRS INTERLOCKED WITH KIRK KEY LOCK
4. DUAL CONTACTORS LOCATED IN EMER. MCC5, 9699-FE-10A,
APPROVED
REGISTERED PROFESSIONAL ENGINEER NO 25095
COMMONWEALTH OF MASSACHUSETTS
30 REVISED PER GENERAL UPDATE
29 REVISED PER EDCR 90-305 + Q.U.
28 REVISED PER EDCR 87-306 AS-BUILT PER EDCR 87-302 ECN #3
27 REVISED PER EDCR 86-327
31 REVISED PER EDCR 89-302
32 REVISED PER EDCR 90-309
480V ONE LINE DIAGRAM-SH.4
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER 9699-FE-1J



LEGEND

100AF 20AT MOLDED-CASE CIRCUIT BREAKER, 3 P.S.T. 600V, 100A FRAME WITH THERMAL MAGNETIC OVERCURRENT TRIP DEVICE - 20A RATING

COMBINATION CIRCUIT BREAKER AND MAGNETIC CONTACTOR WITH THERMAL OVERLOAD RELAY AND CONTROL TRANSFORMER.

MAGNETIC CONTACTOR : TR - CONTROL TRANSFORMER 480 - 120V; R - REVERSING; 1 - SIZE 1

CONTROL SWITCH WITH RED, WHITE AND GREEN INDICATING LIGHTS.

KEY-LOCKED CONTROL SWITCH WITH WHITE INDICATING LIGHT

PAB EQUIPMENT LOCATED IN PRIMARY AUX. BLDG.

RP EQUIPMENT LOCATED AT RECIRCULATION PANEL.

SIP EQUIPMENT LOCATED AT SAFETY INJECTION PANEL.

CONTROL SWITCH WITH RED AND GREEN INDICATING LIGHTS.

SELECTOR SWITCH AND RED AND GREEN PILOT LIGHTS.

NOTES:

FOR GENERAL NOTES & LEGEND ~ SEE DWG. 9699-FE-1D

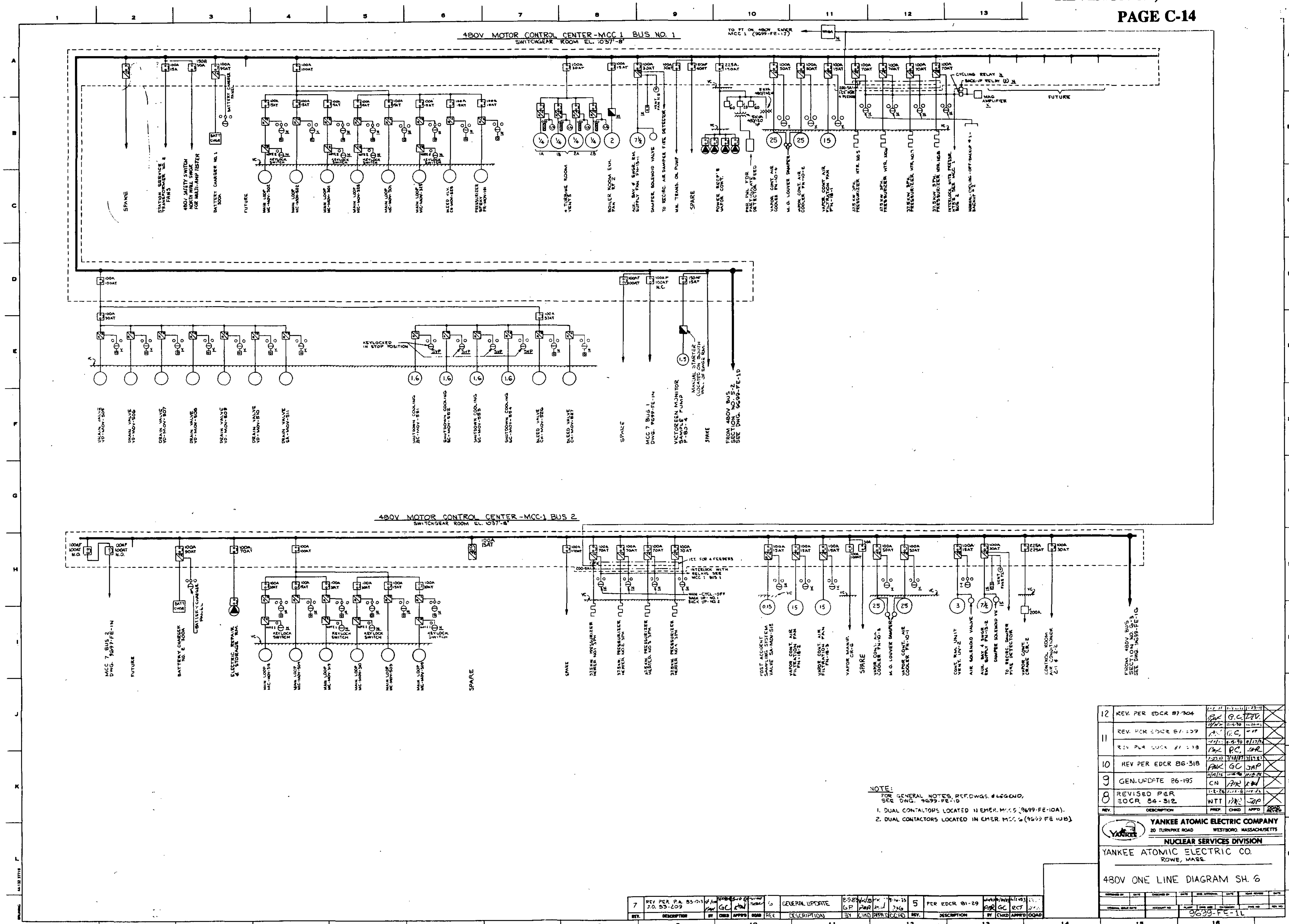
REV	DESCRIPTION	PREP	CHKD	APPD	QOAD REVIEW
5	REVISED PER MR 86-964 & G.U.				
4	REVISED PER EDCR 84-312				
3	REVISED PER EDCR 84-[illegible]		FMR		
2	REVISED PER EDCR 83-33		GC		
1	REV. PER EDCR 81-35	RC	GC		
6	REVISED PER EDCR 87-302	PC			

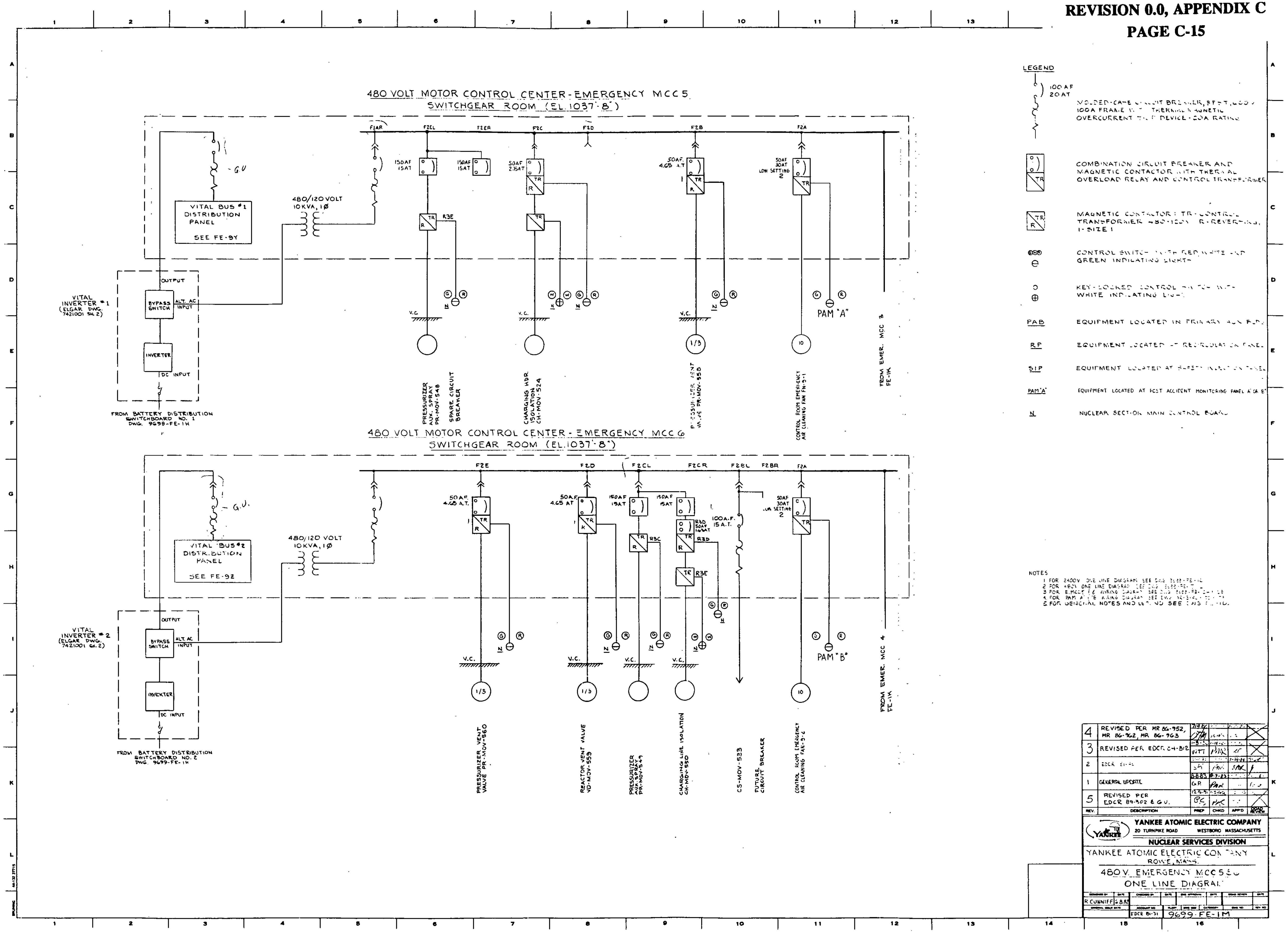
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD WESTBORO MASSACHUSETTS
NUCLEAR SERVICES DIVISION

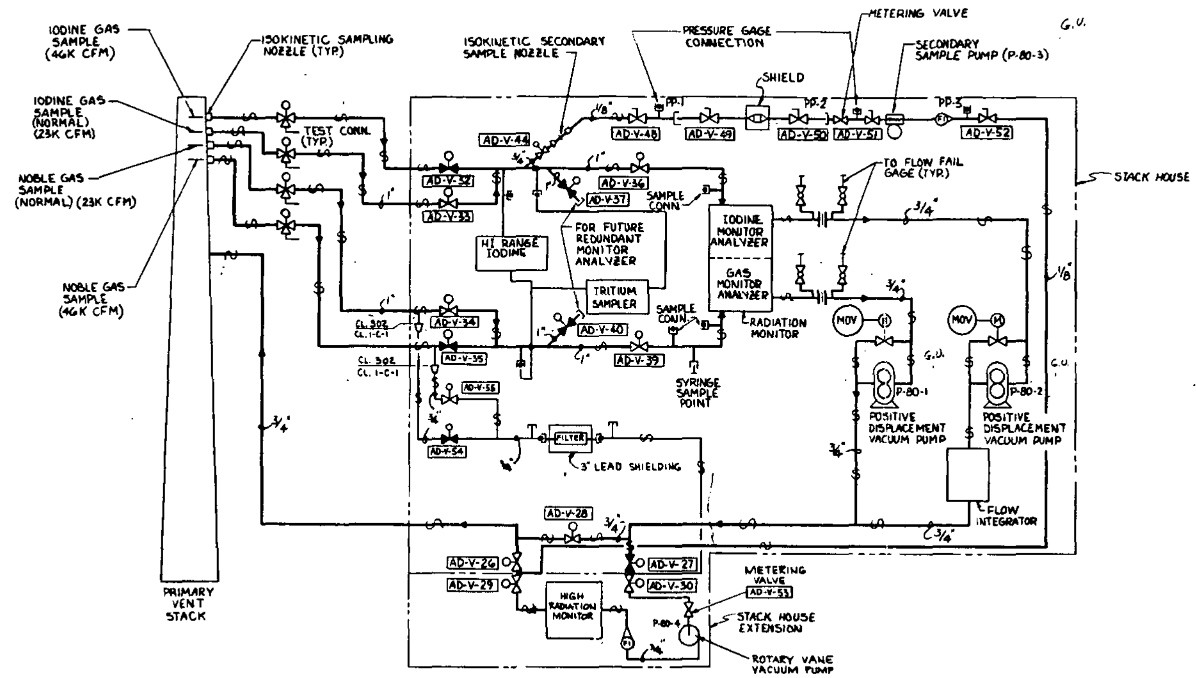
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.

480V EMERGENCY MCC 3 - 4
ONE LINE DIAGRAM

EDCR 81-37 9699-FE-[illegible]


480V MOTOR CONTROL CENTER-MCC 1 BUS NO. 1
SWITCHGEAR ROOM EL. 1037'-8"
480V MOTOR CONTROL CENTER-MCC-1 BUS 2
SWITCHGEAR ROOM EL. 1037'-8"
NOTE:
FOR GENERAL NOTES, REF. DWGS. & LEGEND, SEE DWG. 9699-FE-1D
1. DUAL CONTACTORS LOCATED IN EMER. MCC 5 (9699-FE-10A).
2. DUAL CONTACTORS LOCATED IN EMER. MCC 6 (9699-FE-10B).
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC CO.
ROWE, MASS.
480V ONE LINE DIAGRAM SH. 6
9699-FE-1L


480 VOLT MOTOR CONTROL CENTER - EMERGENCY MCC 5
SWITCHGEAR ROOM (EL. 1037'-8")
480 VOLT MOTOR CONTROL CENTER - EMERGENCY MCC 6
SWITCHGEAR ROOM (EL. 1037'-8")
VITAL BUS #1 DISTRIBUTION PANEL SEE FE-9Y
VITAL BUS #2 DISTRIBUTION PANEL SEE FE-92
480/120 VOLT 10KVA, 1Ø
VITAL INVERTER #1 (ELGAR DWG. 7421001 SH. 2)
VITAL INVERTER #2 (ELGAR DWG. 7421001 SH. 2)
FROM BATTERY DISTRIBUTION SWITCHBOARD NO. 1 DWG. 9699-FE-1H
FROM BATTERY DISTRIBUTION SWITCHBOARD NO. 2 DWG. 9699-FE-1H
PRESSURIZER AUX. SPRAY PR-MOV-548
SPARE CIRCUIT BREAKER
CHARGING HDR ISOLATION CH-MOV-524
PRESSURIZER VENT VALVE PR-MOV-558
CONTROL ROOM EMERGENCY AIR CLEANING FAN FN-5-1
FROM EMER. MCC 3 FE-1K
PAM "A"
PRESSURIZER VENT VALVE PR-MOV-560
REACTOR VENT VALVE VD-MOV-559
PRESSURIZER AUX. SPRAY PR-MOV-549
CHARGING LINE ISOLATION CH-MOV-550
CS-MOV-523
FUTURE CIRCUIT BREAKER
CONTROL ROOM EMERGENCY AIR CLEANING FAN-5-2
FROM EMER. MCC 4 FE-1K
PAM "B"
LEGEND
100 AF 20 AT
MOLDED-CASE CIRCUIT BREAKER, 3 POLE, 600V, 100A FRAME WITH THERMAL MAGNETIC OVERCURRENT TRIP DEVICE - 20A RATING
COMBINATION CIRCUIT BREAKER AND MAGNETIC CONTACTOR WITH THERMAL OVERLOAD RELAY AND CONTROL TRANSFORMER
MAGNETIC CONTACTOR: TR - CONTROL TRANSFORMER 480-120V, R - REVERSING, 1 - SIZE 1
CONTROL SWITCH WITH RED, WHITE AND GREEN INDICATING LIGHTS
KEY-LOCKED CONTROL SWITCH WITH WHITE INDICATING LIGHT
PAB EQUIPMENT LOCATED IN PRIMARY AUX. BLDG.
RP EQUIPMENT LOCATED AT RECIRCULATION PANEL
SIP EQUIPMENT LOCATED AT SAFETY INJECTION PANEL
PAM'A' EQUIPMENT LOCATED AT POST ACCIDENT MONITORING PANEL A OR B
N NUCLEAR SECTION MAIN CONTROL BOARD
NOTES
5 FOR GENERAL NOTES AND LEGEND SEE THIS DWG.
4 REVISED PER MR 86-952, MR 86-962, MR 86-963
3 REVISED PER EDCR 84-812
1 GENERAL UPDATE
5 REVISED PER EDCR 89-302 & G.U.
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD WESTBORO MASSACHUSETTS
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY ROWE, MASS.
480 V. EMERGENCY MCC 5 & 6 ONE LINE DIAGRAM
9699-FE-1M



NOTES:

1. ALL LINES SHOWN AS 1" ARE CLASS 302 PIPING AND FITTINGS AS SPECIFIED PER YANKEE PIPE SPEC YS-497
2. ALL LINES SHOWN AS 3/4" OR SMALLER ARE CLASS 1-C-1 STAINLESS STEEL TUBING AND FITTINGS AS SPECIFIED PER YANKEE PIPE SPEC YS-497

REFERENCE DWGS.

STACK MOD. SYS. PIPING MODS. ~ 9699-FK-13B & C
ELEC HT. TRACING STACK MON. ~ 9699-FK-13D

LEGEND:

- T.C. 3 WAY BALL VALVE W/TEST CONN.
- BALL VALVE
- QUICK CONNECT
- MOV MOTOR OPERATED VALVE
- HEAT TRACING
- SAMPLE CARTRIDGE HOLDER
- PLUG VALVE
- ROTOMETER
- PP PRESSURE POINT

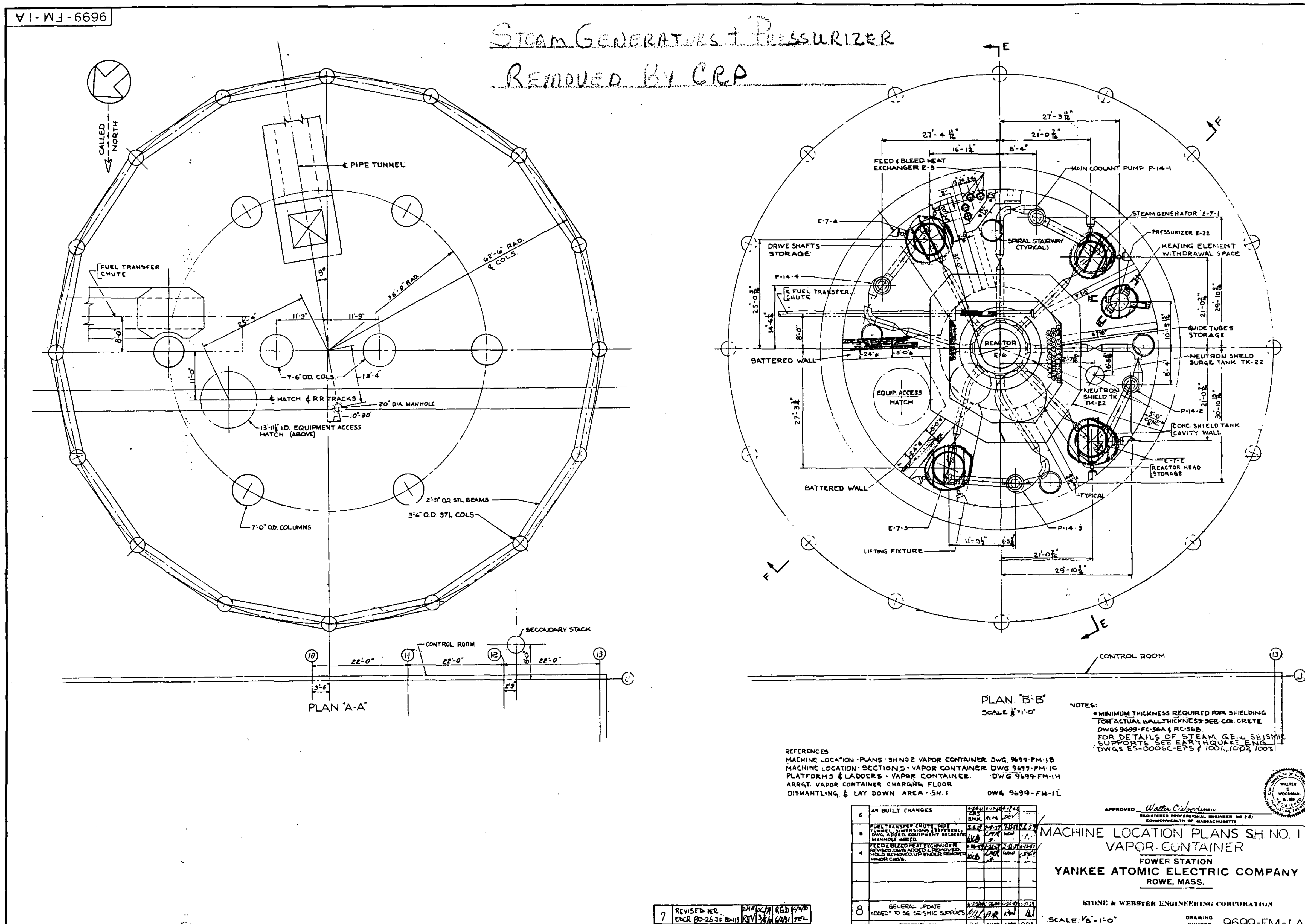
REV.	DESCRIPTION	PREP.	CHKD.	APP'D	
5	REVISED PER EDCR 87-309 AND G.U.	11-15-90	11-15-90	11-20-90	
4	REVISED PER EDCR 84-326	3-25-86	3-25-86	3-26-86	
3	THIS DWG SUPERSEDES 9699-FK-13A REV. 2 PER EDCR 84-314 & EDCR 84-326				

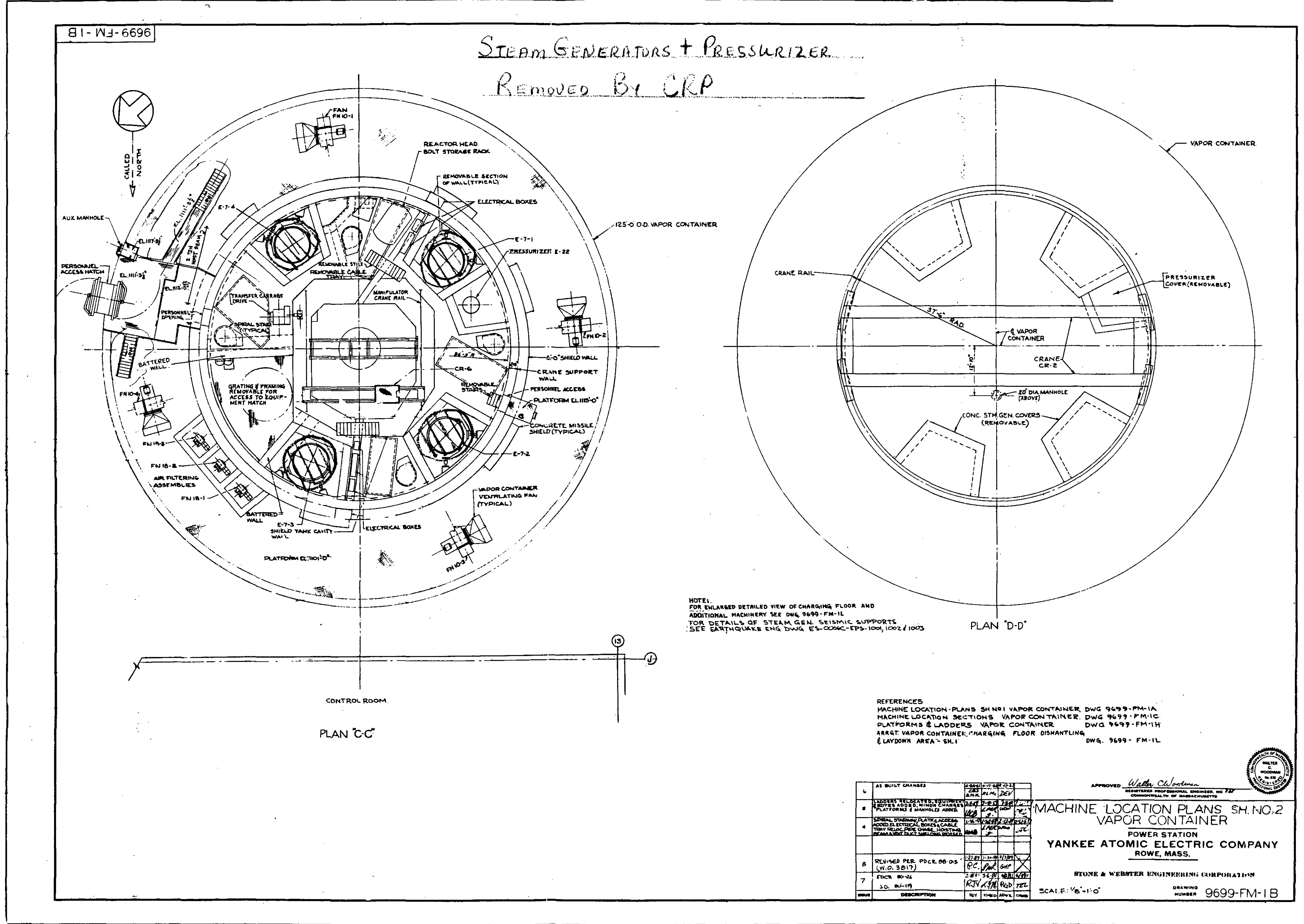
YANKEE ATOMIC ELECTRIC COMPANY
NUCLEAR SERVICES DIVISION

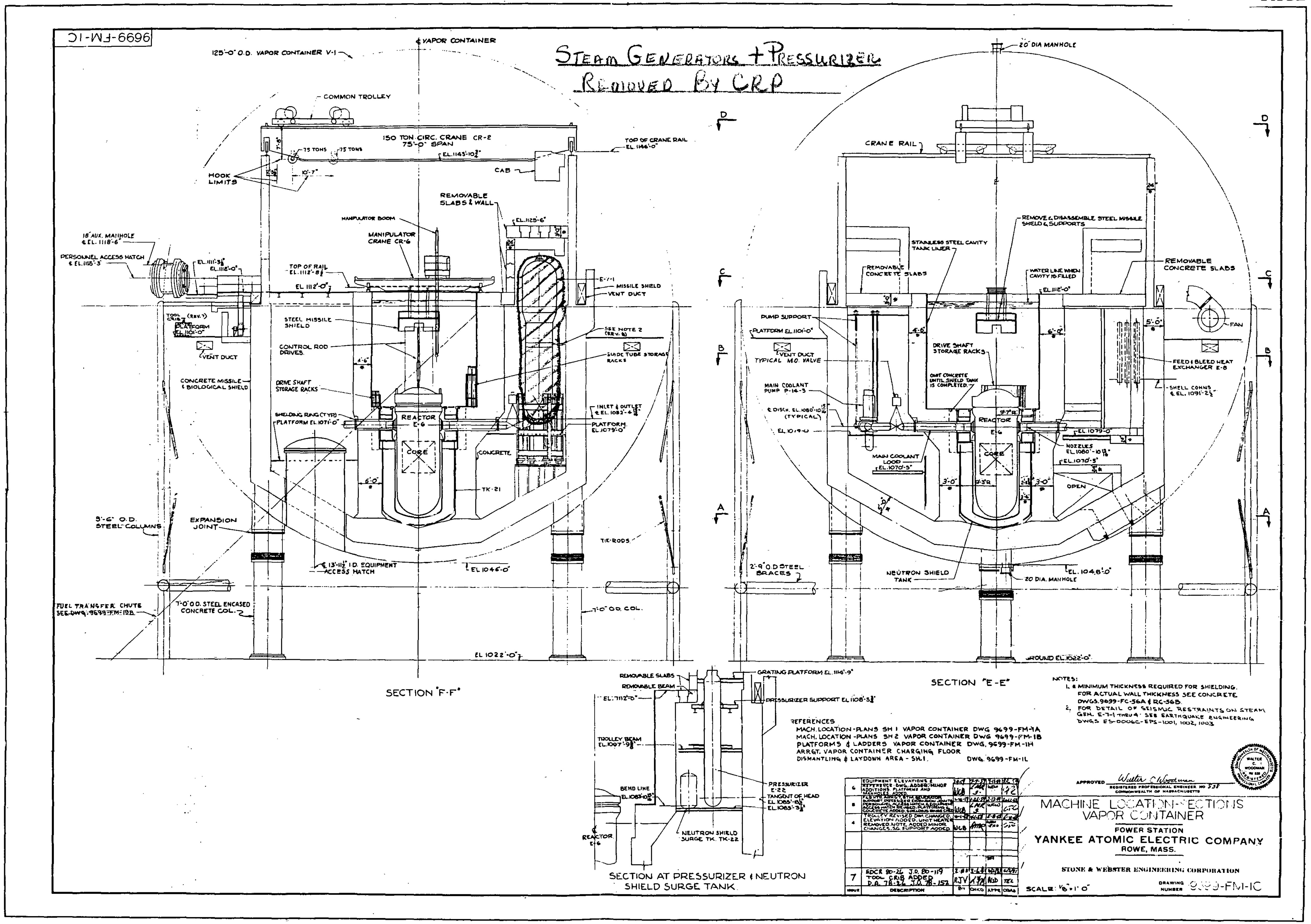
YANKEE ATOMIC ELECTRIC CO.
ROWE, MASS.

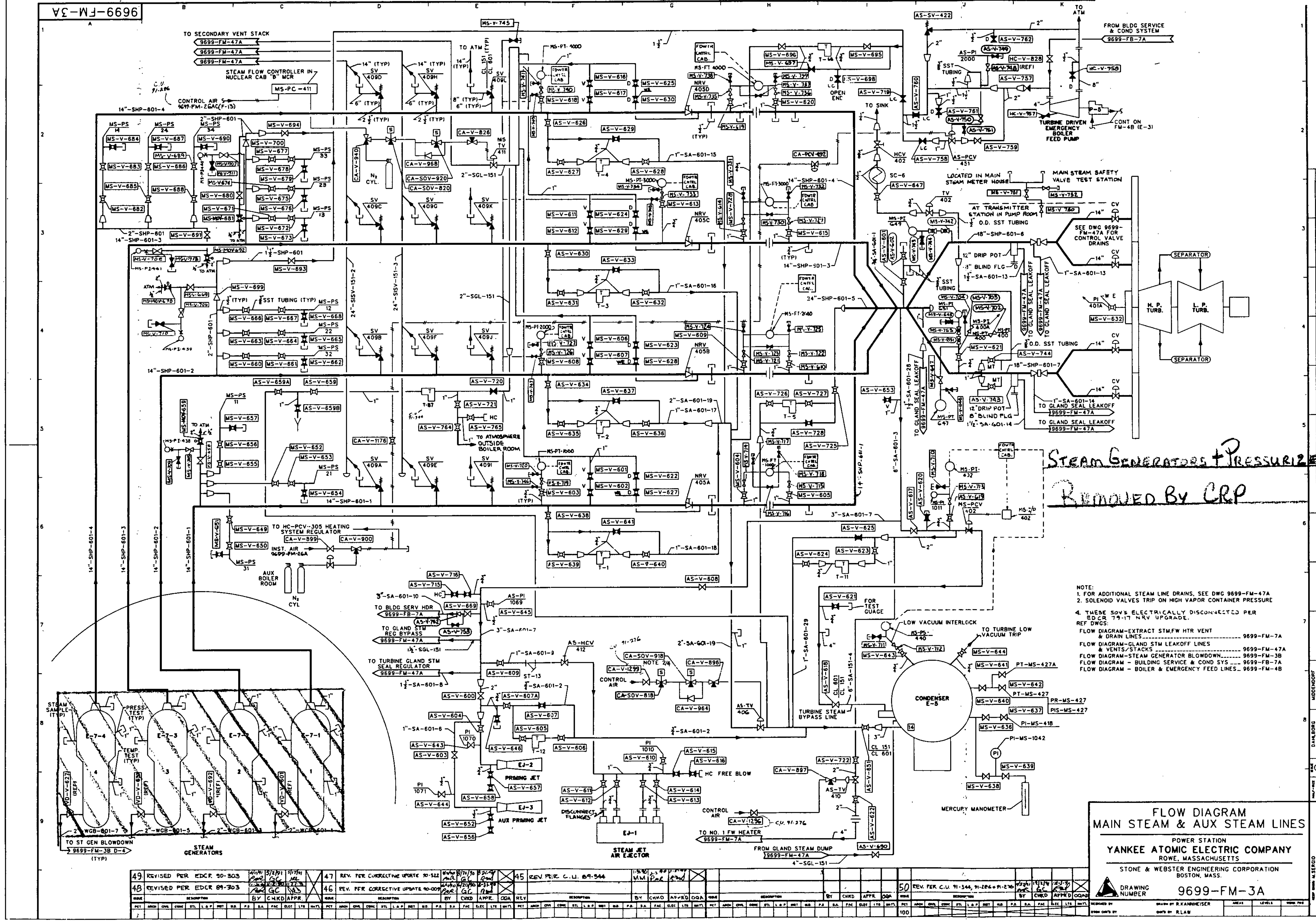
P & I DIAGRAM
STACK MONITORING SYSTEM

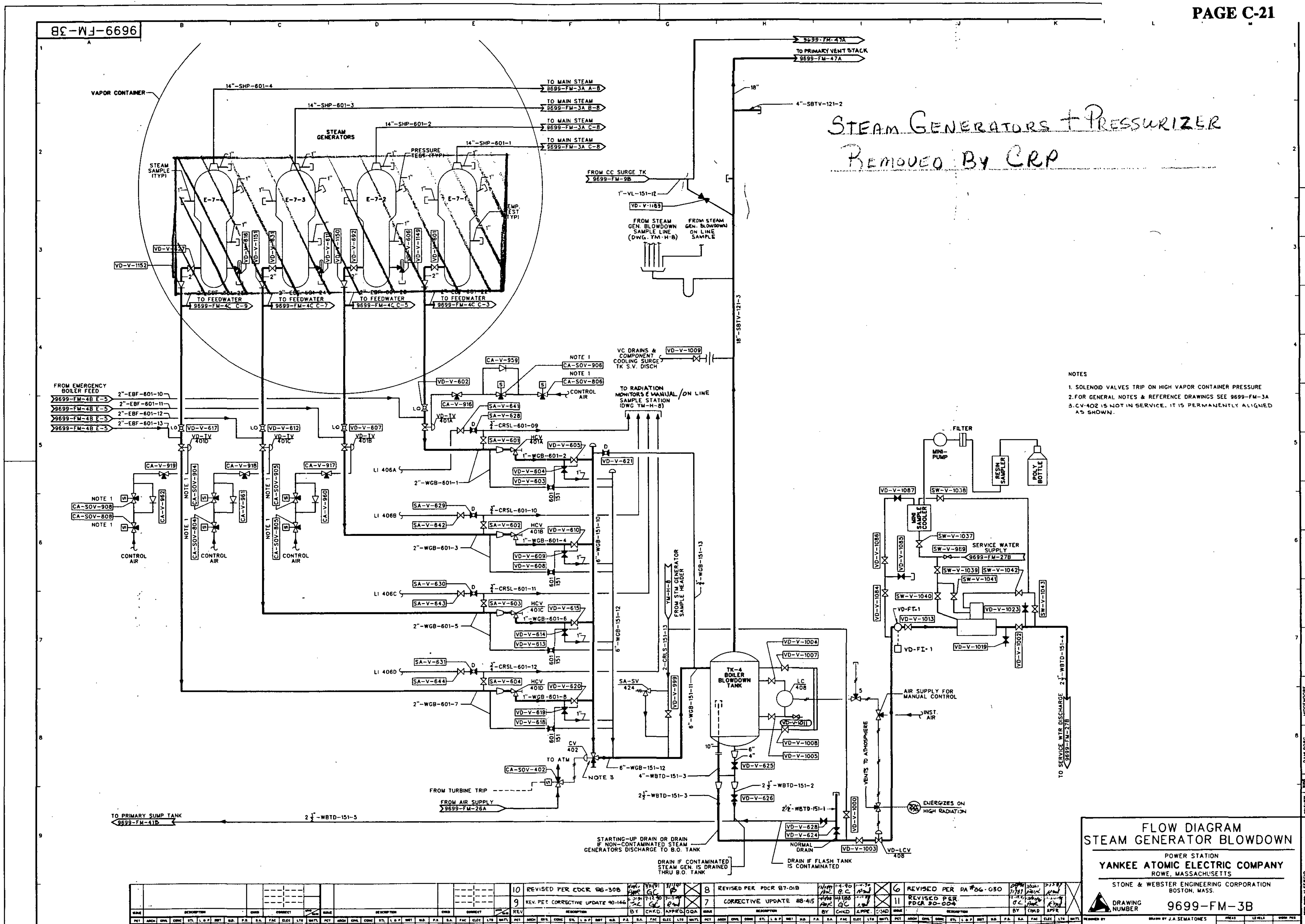
9699-FK-13A


9699-FM-1A
Steam Generators + Pressurizer
Removed By CRP
CALLED NORTH
PIPE TUNNEL
FUEL TRANSFER CHUTE
62'-6" RAD. ℄ COLS.
36'-9" RAD.
7'-6" O.D. COLS.
℄ HATCH & R.R. TRACKS
20" DIA. MANHOLE
13'-11⅝" I.D. EQUIPMENT ACCESS HATCH (ABOVE)
2'-9" O.D. STL BEAMS
3'-6" O.D. STL COLS
7'-0" O.D. COLUMNS
SECONDARY STACK
CONTROL ROOM
PLAN "A-A"
FEED & BLEED HEAT EXCHANGER E-5
MAIN COOLANT PUMP P-14-1
STEAM GENERATOR E-7-1
PRESSURIZER E-22
HEATING ELEMENT WITHDRAWAL SPACE
SPIRAL STAIRWAY (TYPICAL)
DRIVE SHAFTS STORAGE
GUIDE TUBES STORAGE
NEUTRON SHIELD SURGE TANK TK-22
REACTOR
BATTERED WALL
EQUIP. ACCESS HATCH
NEUTRON SHIELD TK TK-22
CONC. SHIELD TANK CAVITY WALL
REACTOR HEAD STORAGE
LIFTING FIXTURE
CONTROL ROOM
PLAN "B-B"
SCALE ⅛"=1'-0"
NOTES:
* MINIMUM THICKNESS REQUIRED FOR SHIELDING
FOR ACTUAL WALL THICKNESS SEE CONCRETE
DWGS 9699-FC-56A & RC-56B.
FOR DETAILS OF STEAM GE. & SEISMIC
SUPPORTS SEE EARTHQUAKE ENG.
DWGS ES-0006C-EPS & 1001, 1002, 1003
REFERENCES
MACHINE LOCATION - PLANS - SH NO 2 VAPOR CONTAINER DWG. 9699-FM-1B
MACHINE LOCATION - SECTIONS - VAPOR CONTAINER DWG 9699-FM-1C
PLATFORMS & LADDERS - VAPOR CONTAINER DWG 9699-FM-1H
ARRGT. VAPOR CONTAINER CHARGING FLOOR
DISMANTLING & LAY DOWN AREA - SH. 1 DWG 9699-FM-1L
6 AS BUILT CHANGES
8 FUEL TRANSFER CHUTE, PIPE TUNNEL, DIMENSIONS & REFERENCE DWG ADDED. EQUIPMENT RELOCATED. MANHOLE ADDED.
4 FEED & BLEED HEAT EXCHANGER REVISED. DWGS ADDED & REMOVED. HOLD REMOVED. UP ENDER REMOVED. MINOR CHG'S.
8 GENERAL UPDATE ADDED * TO SG SEISMIC SUPPORTS
7 REVISED PER EDCR 80-26 & 80-113
ISSUE DESCRIPTION BY CHKD APPR OQA
APPROVED Walter C. Woodman
REGISTERED PROFESSIONAL ENGINEER
COMMONWEALTH OF MASSACHUSETTS
MACHINE LOCATION PLANS SH. NO. 1
VAPOR CONTAINER
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
SCALE: ⅛"=1'-0"
DRAWING NUMBER 9699-FM-1A


9699-FM-1B
STEAM GENERATORS + PRESSURIZER
REMOVED BY CRP
CALLED NORTH
FAN FN 10-1
REACTOR HEAD BOLT STORAGE RACK
REMOVABLE SECTION OF WALL (TYPICAL)
ELECTRICAL BOXES
125'-0 O.D. VAPOR CONTAINER
E-7-4
E-7-1
PRESSURIZER E-22
AUX MANHOLE
PERSONNEL ACCESS HATCH
REMOVABLE STILE
REMOVABLE CABLE TRAY
TRANSFER CARRIAGE DRIVE
MANIPULATOR CRANE RAIL
PERSONNEL OPENING
SPIRAL STAIR (TYPICAL)
FN 10-2
BATTERED WALL
SHIELD WALL
CR-6
CRANE SUPPORT WALL
REMOVABLE STAIRS
PERSONNEL ACCESS
PLATFORM EL.1115'-0"
GRATING & FRAMING REMOVABLE FOR ACCESS TO EQUIPMENT HATCH
CONCRETE MISSILE SHIELD (TYPICAL)
E-7-2
FN 18-3
FN 18-2
AIR FILTERING ASSEMBLIES
FN 18-1
VAPOR CONTAINER VENTILATING FAN (TYPICAL)
BATTERED WALL
E-7-3
SHIELD TANK CAVITY WALL
ELECTRICAL BOXES
PLATFORM EL.1101'-0"
FN 10-3
CONTROL ROOM
PLAN "C-C"
VAPOR CONTAINER
CRANE RAIL
PRESSURIZER COVER (REMOVABLE)
37'-6" RAD
℄ VAPOR CONTAINER
CRANE CR-2
20' DIA MANHOLE (ABOVE)
CONC. STM GEN. COVERS (REMOVABLE)
PLAN "D-D"
NOTE:
FOR ENLARGED DETAILED VIEW OF CHARGING FLOOR AND ADDITIONAL MACHINERY SEE DWG 9699-FM-1L
FOR DETAILS OF STEAM GEN. SEISMIC SUPPORTS SEE EARTHQUAKE ENG DWG ES-0006C-EPS-1001, 1002 & 1003
REFERENCES
MACHINE LOCATION PLANS SH NO 1 VAPOR CONTAINER DWG 9699-FM-1A
MACHINE LOCATION SECTIONS VAPOR CONTAINER DWG 9699-FM-1C
PLATFORMS & LADDERS VAPOR CONTAINER DWG 9699-FM-1H
ARRGT. VAPOR CONTAINER CHARGING FLOOR DISMANTLING & LAYDOWN AREA - SH. 1 DWG. 9699-FM-1L
APPROVED Walter C. Woodman
REGISTERED PROFESSIONAL ENGINEER
COMMONWEALTH OF MASSACHUSETTS
AS BUILT CHANGES
REVISED PER PDCR 88-05 (W.O. 3817)
DESCRIPTION
MACHINE LOCATION PLANS SH. NO. 2
VAPOR CONTAINER
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
SCALE: 1/8"=1'-0"
DRAWING NUMBER 9699-FM-1B


9699-FM-1C
Steam Generators + Pressurizer
Removed By CRP
℄ VAPOR CONTAINER
125'-0" O.D. VAPOR CONTAINER V-1
COMMON TROLLEY
150 TON CIRC. CRANE CR-2
75'-0" SPAN
TOP OF CRANE RAIL EL. 1146'-0"
CAB
HOOK LIMITS
REMOVABLE SLABS & WALL
MANIPULATOR CRANE CR-6
PERSONNEL ACCESS HATCH
STEEL MISSILE SHIELD
CONTROL ROD DRIVES
DRIVE SHAFT STORAGE RACKS
CONCRETE MISSILE & BIOLOGICAL SHIELD
VENT DUCT
REACTOR E-6
CORE
TK-21
CONCRETE
5'-6" O.D. STEEL COLUMNS
EXPANSION JOINT
TIE-RODS
FUEL TRANSFER CHUTE
7'-0" O.D. STEEL ENCASED CONCRETE COL.
7'-0" O.D. COL.
EL 1022'-0"
SECTION "F-F"
20" DIA MANHOLE
CRANE RAIL
REMOVABLE CONCRETE SLABS
PUMP SUPPORT
MAIN COOLANT PUMP P-14-3
MAIN COOLANT LOOP
FEED & BLEED HEAT EXCHANGER E-8
FAN
NEUTRON SHIELD TANK
2'-9" O.D. STEEL BRACES
20 DIA. MANHOLE
GROUND EL. 1022'-0"
SECTION "E-E"
SECTION AT PRESSURIZER & NEUTRON SHIELD SURGE TANK
PRESSURIZER E-22
NEUTRON SHIELD SURGE TK. TK-22
NOTES:
1. ✱ MINIMUM THICKNESS REQUIRED FOR SHIELDING. FOR ACTUAL WALL THICKNESS SEE CONCRETE DWGS. 9699-FC-56A & RC-56B.
2. FOR DETAIL OF SEISMIC RESTRAINTS ON STEAM GEN. E-7-1 THRU 4 SEE EARTHQUAKE ENGINEERING DWGS ES-0006C-EPS-1001, 1002, 1003
REFERENCES
MACH. LOCATION-PLANS SH 1 VAPOR CONTAINER DWG 9699-FM-1A
MACH. LOCATION-PLANS SH 2 VAPOR CONTAINER DWG 9699-FM-1B
PLATFORMS & LADDERS VAPOR CONTAINER DWG. 9699-FM-1H
ARRGT. VAPOR CONTAINER CHARGING FLOOR
DISMANTLING & LAYDOWN AREA - SH.1. DWG. 9699-FM-1L
MACHINE LOCATION-SECTIONS
VAPOR CONTAINER
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
SCALE: 1/8"=1'-0"
DRAWING NUMBER 9699-FM-1C


9699-FM-3A
Steam Generators + Pressurizer
Removed By CRP
NOTE:
1. FOR ADDITIONAL STEAM LINE DRAINS, SEE DWG 9699-FM-47A
2. SOLENOID VALVES TRIP ON HIGH VAPOR CONTAINER PRESSURE
4. THESE SOVS ELECTRICALLY DISCONNECTED PER EDCR 79-17 NRV UPGRADE.
REF DWGS:
FLOW DIAGRAM-EXTRACT STM,FW HTR VENT & DRAIN LINES 9699-FM-7A
FLOW DIAGRAM-GLAND STM LEAKOFF LINES & VENTS/STACKS 9699-FM-47A
FLOW DIAGRAM-STEAM GENERATOR BLOWDOWN 9699-FM-3B
FLOW DIAGRAM - BUILDING SERVICE & COND SYS 9699-FB-7A
FLOW DIAGRAM - BOILER & EMERGENCY FEED LINES 9699-FM-4B
CONDENSER E-5
STEAM GENERATORS
H.P. TURB.
L.P. TURB.
FLOW DIAGRAM
MAIN STEAM & AUX STEAM LINES
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASSACHUSETTS
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER 9699-FM-3A
49 REVISED PER EDCR 90-303
48 REVISED PER EDCR 89-303
47 REV. PER CORRECTIVE UPDATE 90-322
46 REV. PER CORRECTIVE UPDATE 90-009
45 REV PER C.U. 89-344
50 REV. PER C.U. 91-344, 91-286 + 91-276



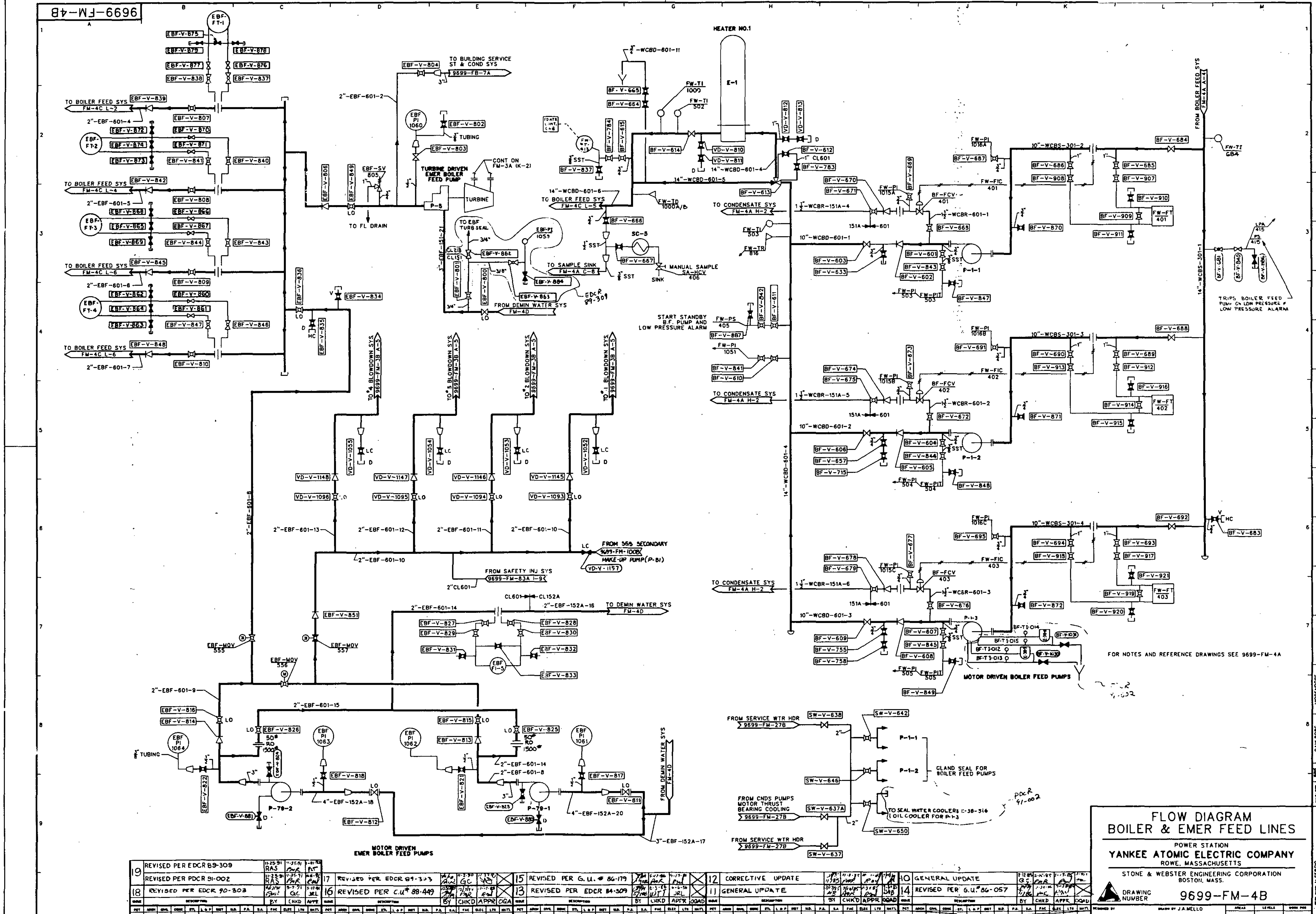
STEAM GENERATORS + PRESSURIZER REMOVED BY CRP

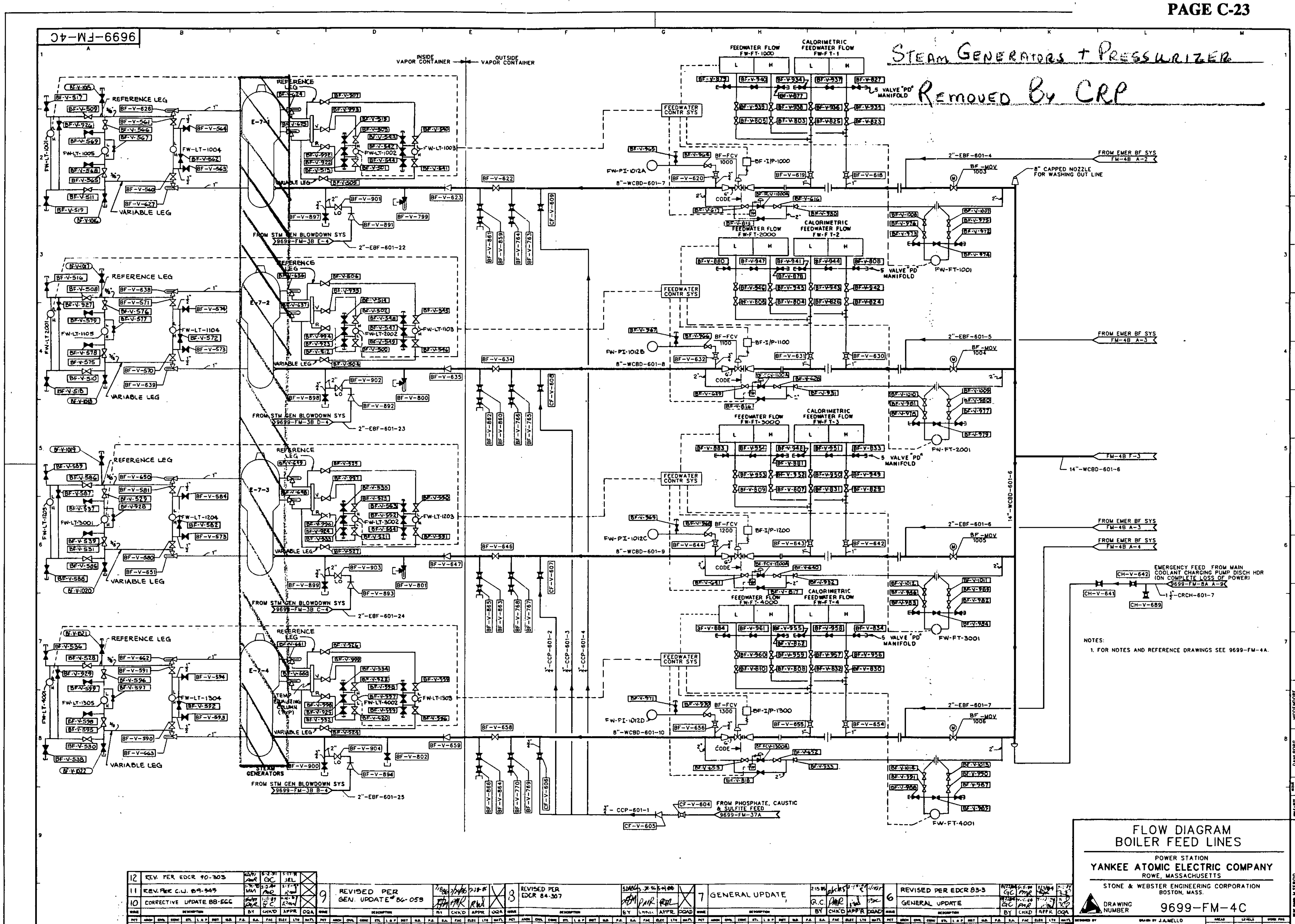
NOTES

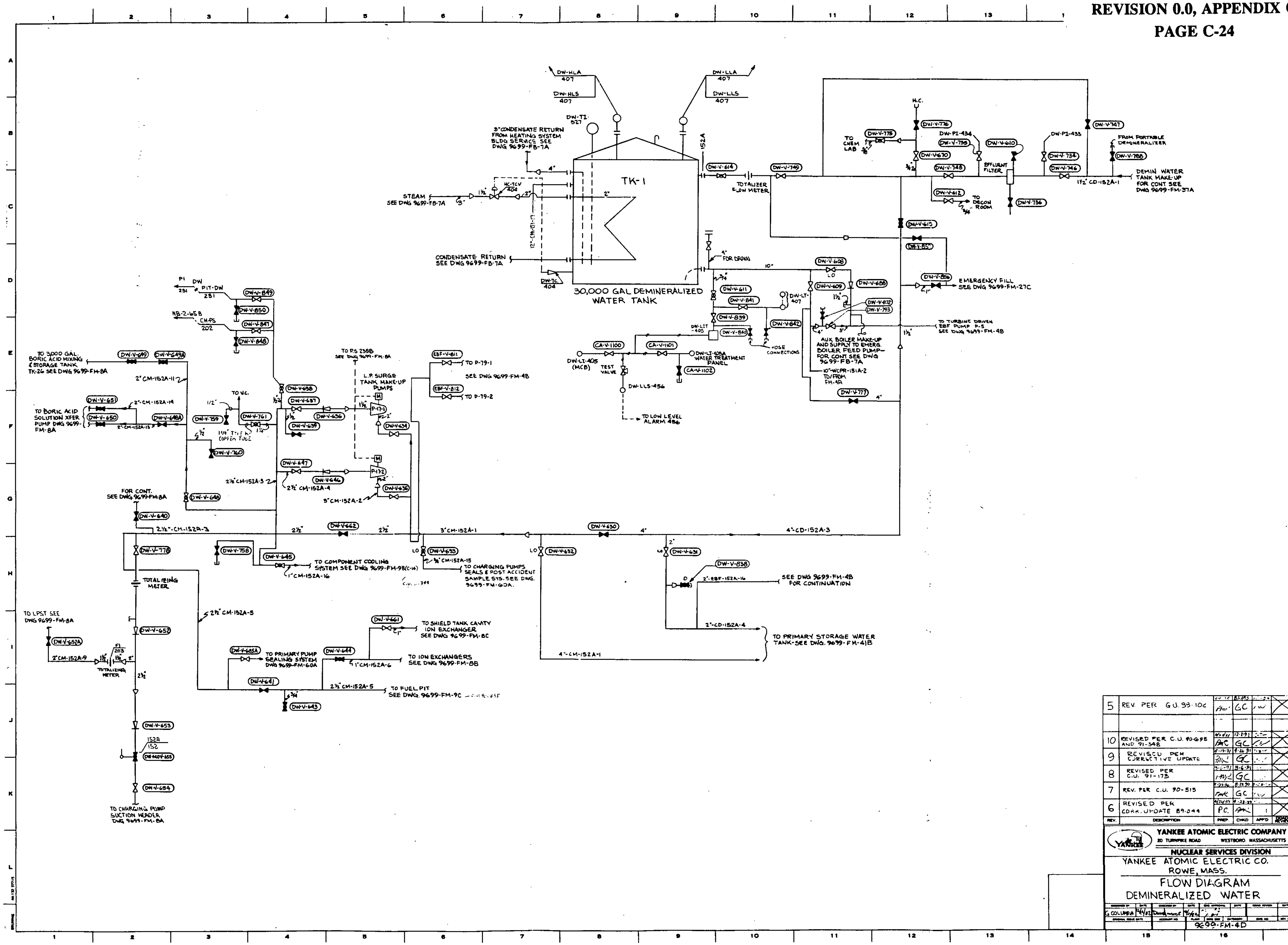
1. SOLENOID VALVES TRIP ON HIGH VAPOR CONTAINER PRESSURE
2. FOR GENERAL NOTES & REFERENCE DRAWINGS SEE 9699-FM-3A
3. CV-402 IS NOT IN SERVICE. IT IS PERMANENTLY ALIGNED AS SHOWN.

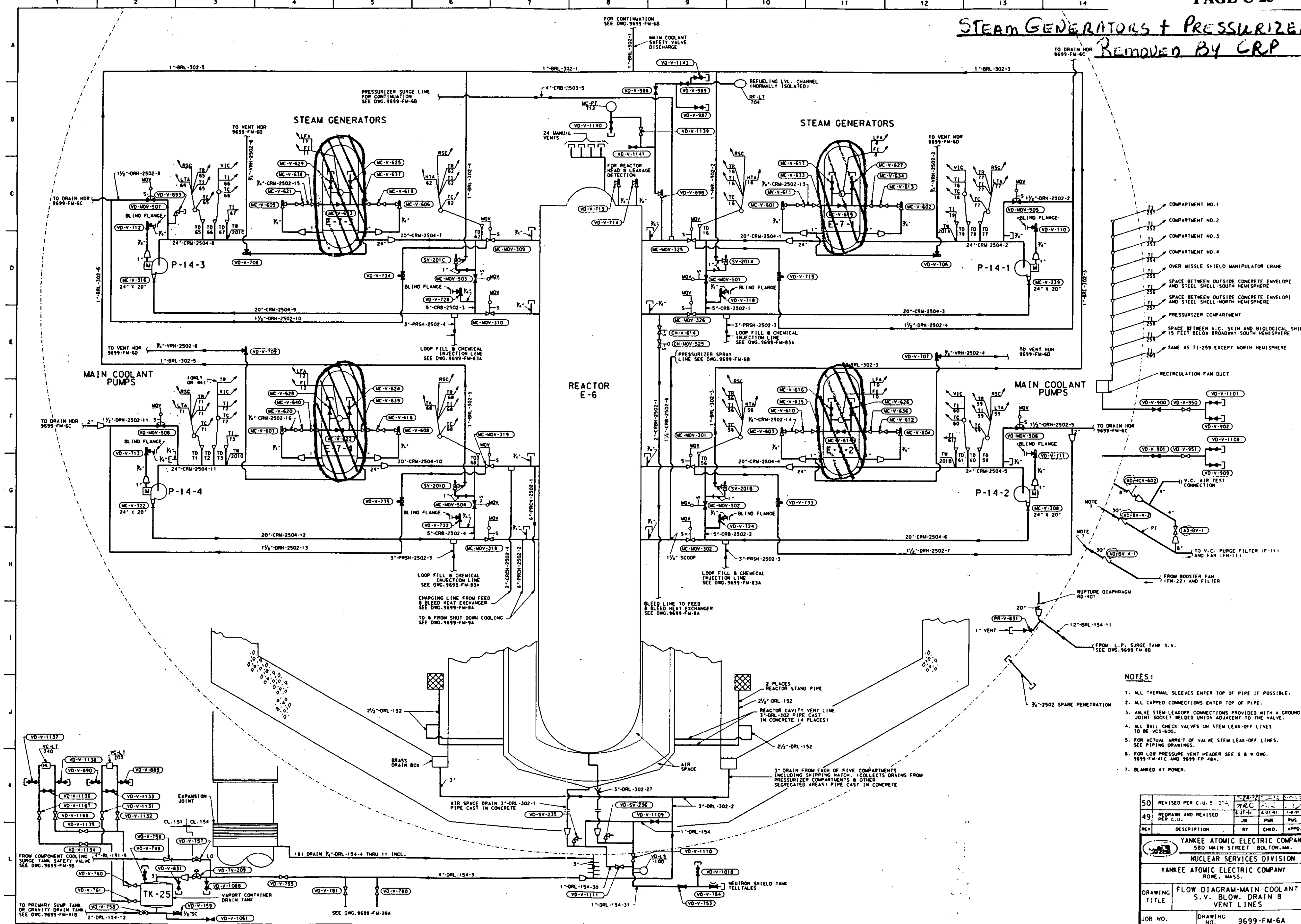
10	REVISED PER EDCR 96-308	8	REVISED PER PDCR 87-018	6	REVISED PER PA #86-030
9	REV. PER CORRECTIVE UPDATE 90-146	7	CORRECTIVE UPDATE #8-415	11	REVISED PER PDCR 90-004

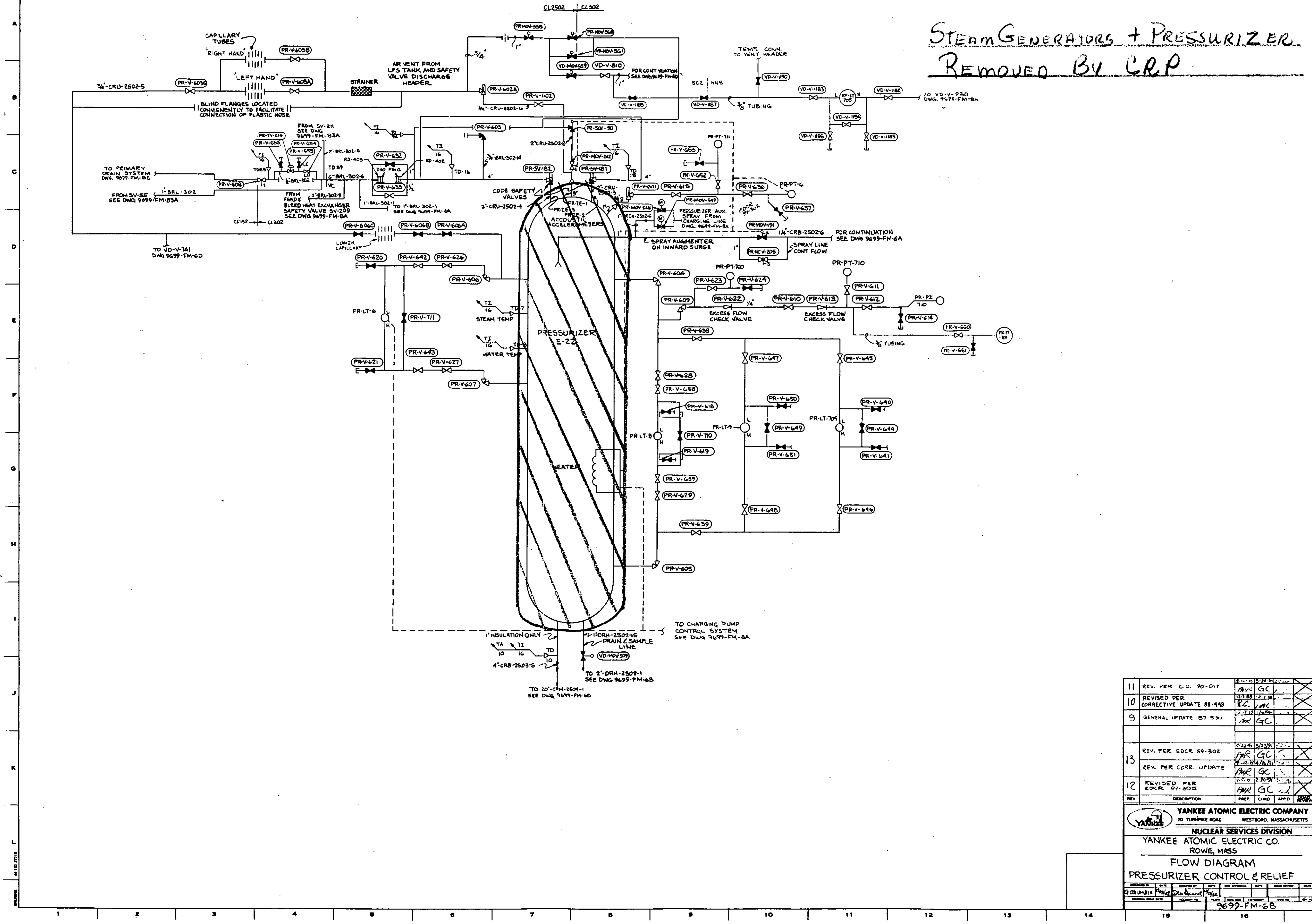
FLOW DIAGRAM
STEAM GENERATOR BLOWDOWN
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASSACHUSETTS
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER 9699-FM-3B
DRAWN BY J.A. SEMATONES
CHECKED BY R.T. LAW

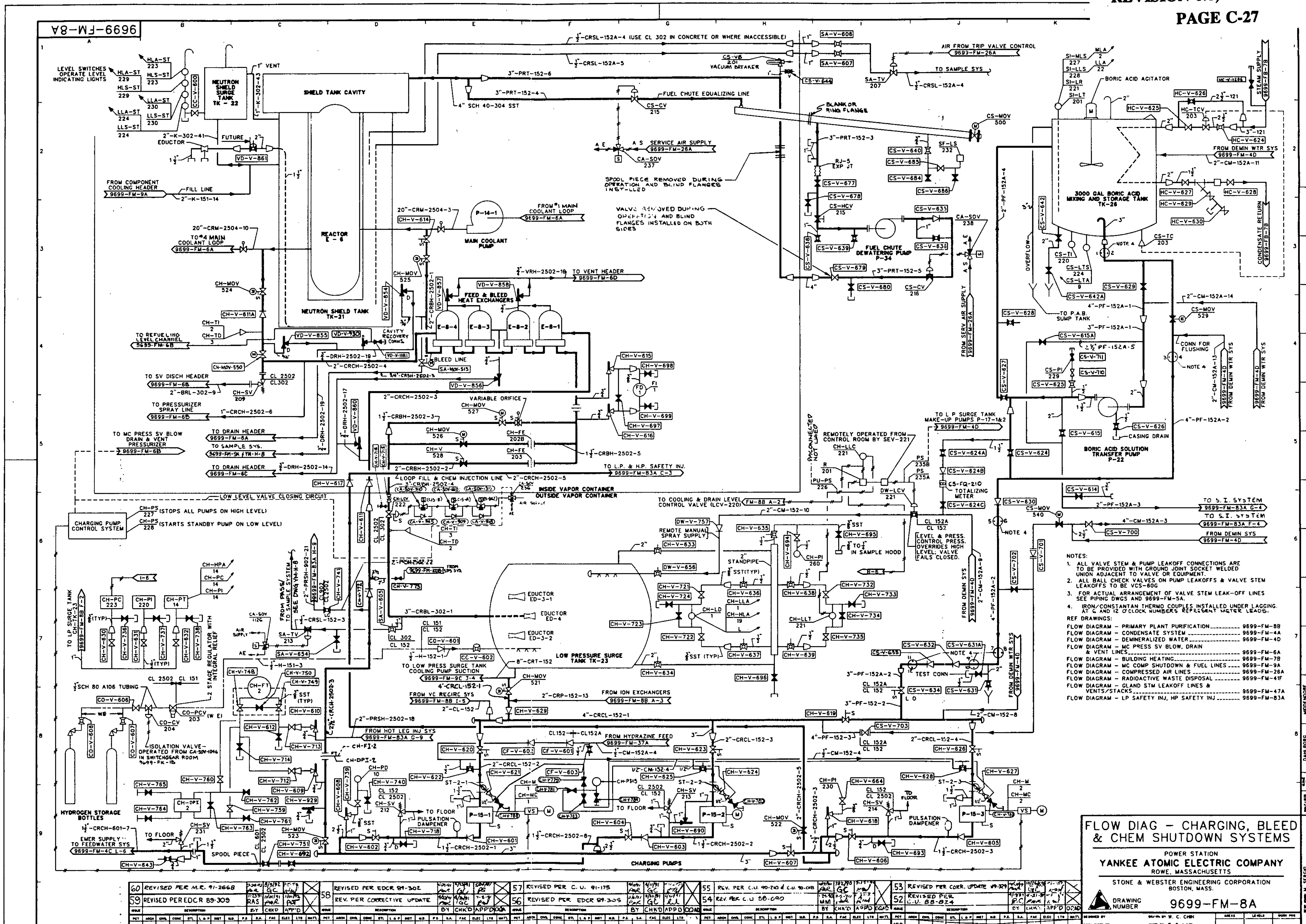

9699-FM-4B
HEATER NO.1
TURBINE DRIVEN EMER BOILER FEED PUMP
MOTOR DRIVEN BOILER FEED PUMPS
MOTOR DRIVEN EMER BOILER FEED PUMPS
GLAND SEAL FOR BOILER FEED PUMPS
START STANDBY B.F. PUMP AND LOW PRESSURE ALARM
TRIPS BOILER FEED PUMP ON LOW PRESSURE & LOW PRESSURE ALARM
FOR NOTES AND REFERENCE DRAWINGS SEE 9699-FM-4A
19 REVISED PER EDCR 89-309
REVISED PER PDCR 91-002
18 REVISED PER EDCR 90-303
17 REVISED PER EDCR 89-303
16 REVISED PER C.U.# 88-449
15 REVISED PER G.U.# 86-179
13 REVISED PER EDCR 84-309
12 CORRECTIVE UPDATE
11 GENERAL UPDATE
10 GENERAL UPDATE
14 REVISED PER G.U.# 86-057
FLOW DIAGRAM
BOILER & EMER FEED LINES
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASSACHUSETTS
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER 9699-FM-4B


9699-FM-4C
Steam Generators + Pressurizer
Removed By CRP
INSIDE VAPOR CONTAINER
OUTSIDE VAPOR CONTAINER
STEAM GENERATORS
NOTES:
1. FOR NOTES AND REFERENCE DRAWINGS SEE 9699-FM-4A.
EMERGENCY FEED FROM MAIN COOLANT CHARGING PUMP DISCH HDR (ON COMPLETE LOSS OF POWER)
FROM PHOSPHATE, CAUSTIC & SULFITE FEED 9699-FM-37A
FLOW DIAGRAM
BOILER FEED LINES
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASSACHUSETTS
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER
9699-FM-4C
12 REV. PER EDCR 90-303
11 REV. PER C.J. 89-945
10 CORRECTIVE UPDATE 88-666
9 REVISED PER GEN. UPDATE 86-059
8 REVISED PER EDCR 84-307
7 GENERAL UPDATE
6 REVISED PER EDCR 83-3
GENERAL UPDATE

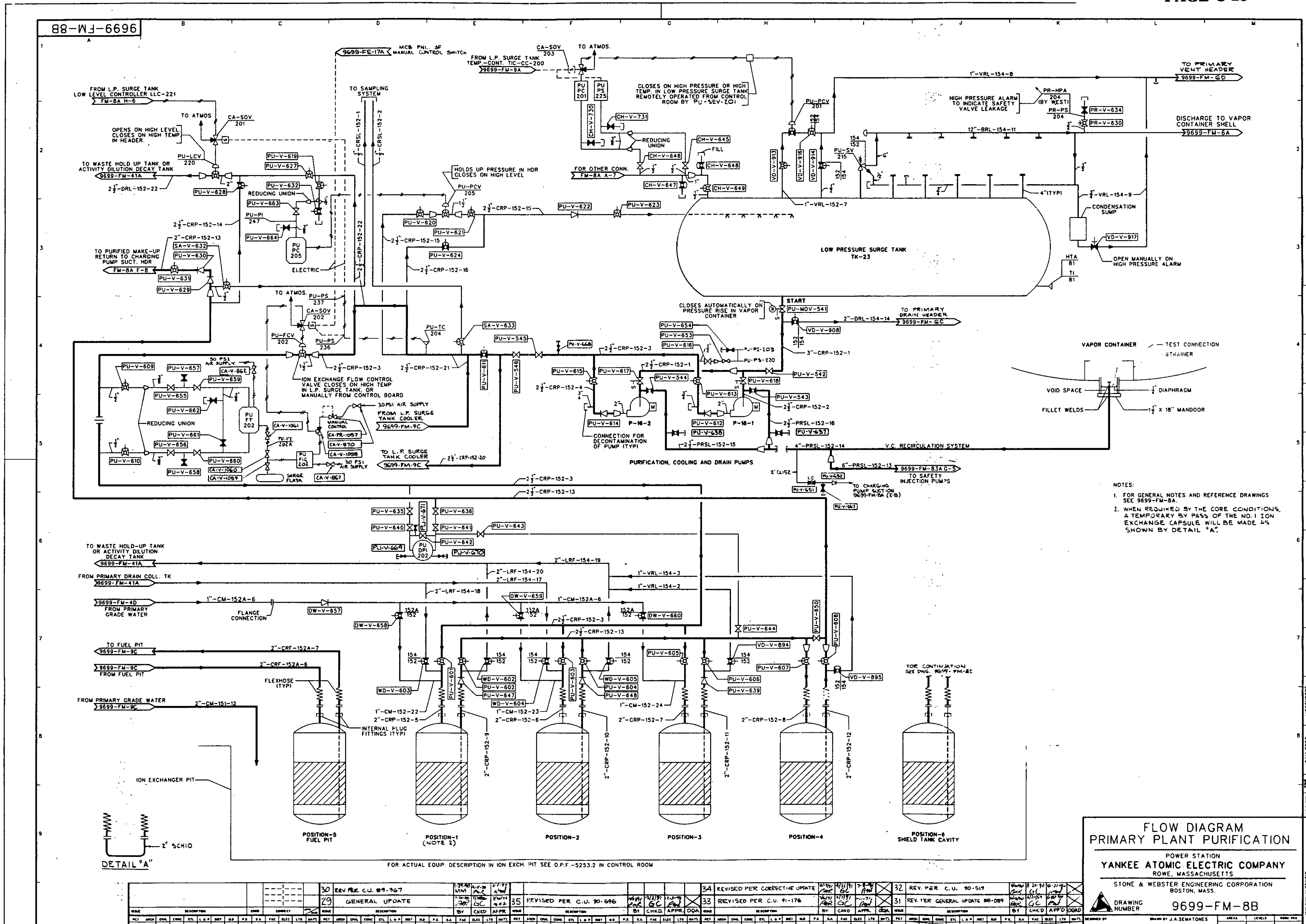

TK-1
30,000 GAL DEMINERALIZED WATER TANK
STEAM SEE DWG 9699-FB-7A
CONDENSATE RETURN SEE DWG 9699-FB-7A
3" CONDENSATE RETURN FROM HEATING SYSTEM BLDG SERVICE SEE DWG 9699-FB-7A
TOTALIZER FLOW METER
TO CHEM LAB
EFFLUENT FILTER
TO DECON ROOM
FROM PORTABLE DEMINERALIZER
DEMIN WATER TANK MAKE-UP FOR CONT SEE DWG 9699-FM-37A
EMERGENCY FILL SEE DWG 9699-FM-27C
TO TURBINE DRIVEN EBF PUMP P-5 SEE DWG 9699-FM-4B
AUX BOILER MAKE-UP AND SUPPLY TO EMERG BOILER FEED PUMP-FOR CONT SEE DWG 9699-FB-7A
WATER TREATMENT PANEL
HOSE CONNECTIONS
TO LOW LEVEL ALARM
TO 3,000 GAL. BORIC ACID MIXING & STORAGE TANK TK-26 SEE DWG 9699-FM-8A
TO BORIC ACID SOLUTION XFER PUMP DWG 9699-FM-8A
L.P. SURGE TANK MAKE-UP PUMPS
TO P-19-1
TO P-19-2
SEE DWG 9699-FM-4B
FOR CONT. SEE DWG 9699-FM-8A
TO COMPONENT COOLING SYSTEM SEE DWG 9699-FM-9B
TO CHARGING PUMPS SEALS & POST ACCIDENT SAMPLE SYS. SEE DWG. 9699-FM-60A
SEE DWG 9699-FM-4B FOR CONTINUATION
TO PRIMARY STORAGE WATER TANK-SEE DWG. 9699-FM-41B
TO LPST SEE DWG 9699-FM-8A
TOTALIZING METER
TO SHIELD TANK CAVITY ION EXCHANGER SEE DWG 9699-FM-8C
TO PRIMARY PUMP SEALING SYSTEM DWG 9699-FM-60A
TO ION EXCHANGERS SEE DWG 9699-FM-8B
TO FUEL PIT SEE DWG. 9699-FM-9C
TO CHARGING PUMP SUCTION HEADER DWG 9699-FM-8A
YANKEE ATOMIC ELECTRIC COMPANY
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC CO. ROWE, MASS.
FLOW DIAGRAM DEMINERALIZED WATER
9699-FM-4D

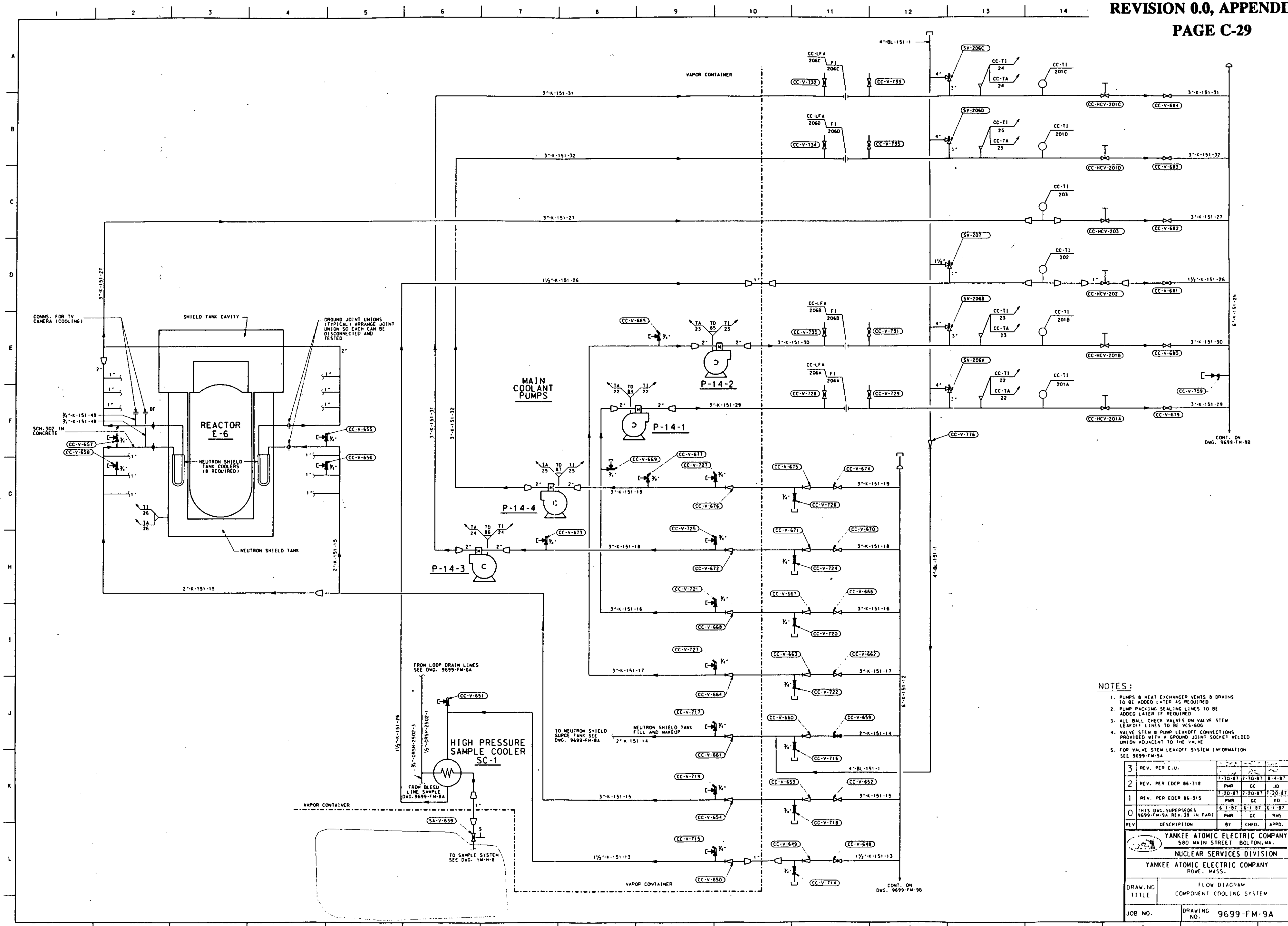

STEAM GENERATORS + PRESSURIZER
REMOVED BY CRP
STEAM GENERATORS
STEAM GENERATORS
MAIN COOLANT PUMPS
MAIN COOLANT PUMPS
REACTOR E-6
P-14-1
P-14-2
P-14-3
P-14-4
TK-25
NOTES:
1. ALL THERMAL SLEEVES ENTER TOP OF PIPE IF POSSIBLE.
2. ALL CAPPED CONNECTIONS ENTER TOP OF PIPE.
3. VALVE STEM LEAKOFF CONNECTIONS PROVIDED WITH A GROUND JOINT SOCKET WELDED UNION ADJACENT TO THE VALVE.
4. ALL BALL CHECK VALVES ON STEM LEAK-OFF LINES TO BE VCS-60G.
5. FOR ACTUAL ARRG'T OF VALVE STEM LEAK-OFF LINES. SEE PIPING DRAWINGS.
6. FOR LOW PRESSURE VENT HEADER SEE S & W DWG. 9699-FM-41C AND 9699-FP-48A.
7. BLANKED AT POWER.
50 REVISED PER C.U.
49 REDRAWN AND REVISED PER C.U. 6-27-91 JW 6-27-91 PMR 7-8-91 RMS
REV DESCRIPTION BY CHKD. APPD.
YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
DRAWING TITLE
FLOW DIAGRAM-MAIN COOLANT S.V. BLOW, DRAIN & VENT LINES
JOB NO.
DRAWING NO. 9699-FM-6A

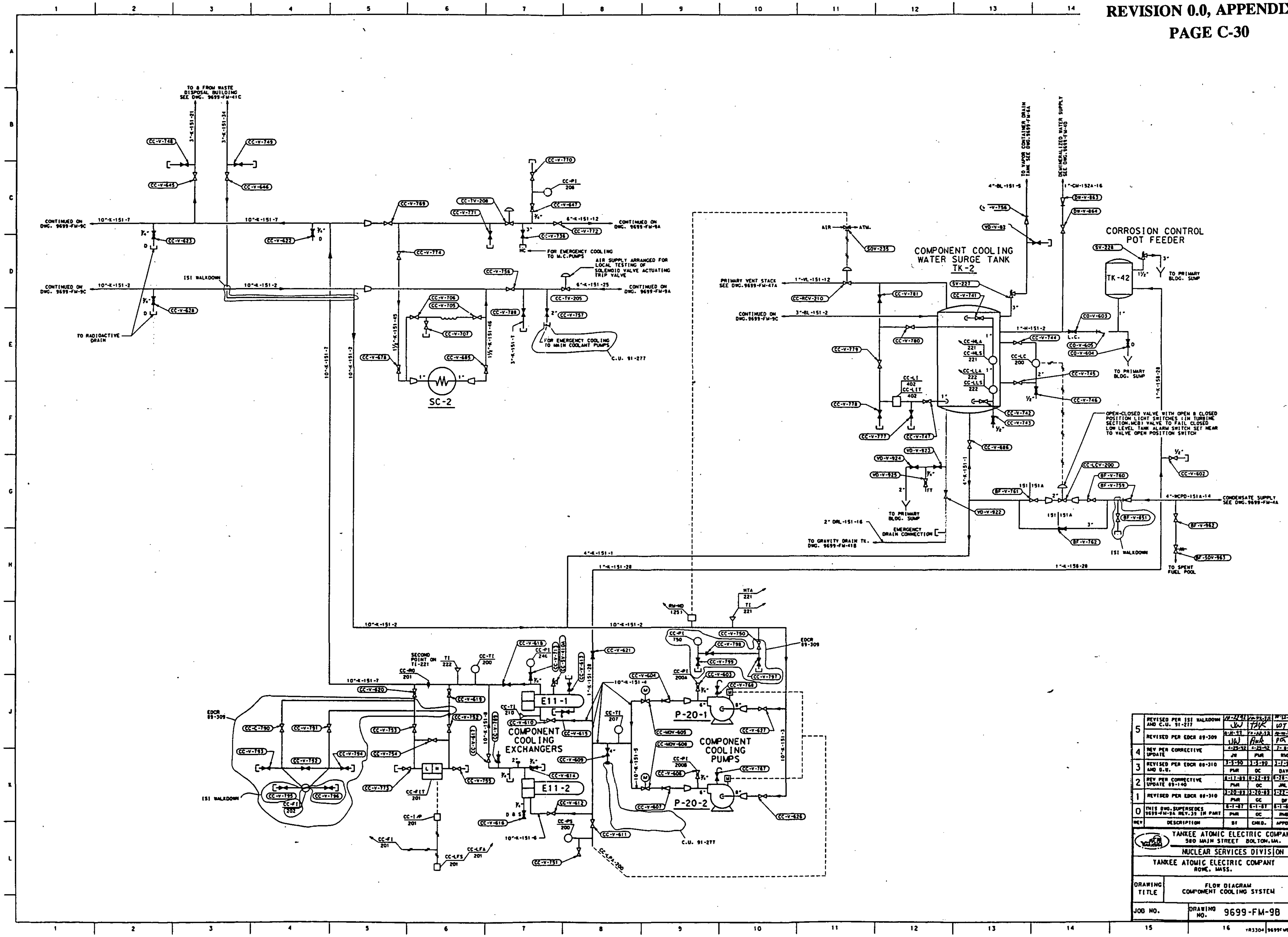

STEAM GENERATORS + PRESSURIZER
REMOVED BY CRP
PRESSURIZER E-22
HEATER
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC CO.
ROWE, MASS
FLOW DIAGRAM
PRESSURIZER CONTROL & RELIEF
9699-FM-6B



(CONTROL ROOM DRAWING)


9699-FM-8B
LOW PRESSURE SURGE TANK
TK-23
PURIFICATION, COOLING AND DRAIN PUMPS
V.C. RECIRCULATION SYSTEM
TO PRIMARY VENT HEADER
DISCHARGE TO VAPOR CONTAINER SHELL
TO PRIMARY DRAIN HEADER
TO SAFETY INJECTION PUMPS
TO CHARGING PUMP SUCTION
VAPOR CONTAINER
TEST CONNECTION
STRAINER
VOID SPACE
FILLET WELDS
DIAPHRAGM
MANDOOR
ION EXCHANGER PIT
ION EXCHANGE FLOW CONTROL VALVE CLOSES ON HIGH TEMP IN L.P. SURGE TANK, OR MANUALLY FROM CONTROL BOARD
CLOSES ON HIGH PRESSURE OR HIGH TEMP. IN LOW PRESSURE SURGE TANK REMOTELY OPERATED FROM CONTROL ROOM BY PU-SEV-201
HIGH PRESSURE ALARM TO INDICATE SAFETY VALVE LEAKAGE
CLOSES AUTOMATICALLY ON PRESSURE RISE IN VAPOR CONTAINER
OPEN MANUALLY ON HIGH PRESSURE ALARM
CONDENSATION SUMP
HOLDS UP PRESSURE IN HDR CLOSES ON HIGH LEVEL
OPENS ON HIGH LEVEL CLOSES ON HIGH TEMP IN HEADER
TO WASTE HOLD UP TANK OR ACTIVITY DILUTION DECAY TANK
TO PURIFIED MAKE-UP RETURN TO CHARGING PUMP SUCT. HDR
FROM L.P. SURGE TANK LOW LEVEL CONTROLLER LLC-221
TO SAMPLING SYSTEM
CONNECTION FOR DECONTAMINATION OF PUMP (TYP)
FROM PRIMARY DRAIN COLL. TK
FROM PRIMARY GRADE WATER
TO FUEL PIT
FROM FUEL PIT
FLEXHOSE (TYP)
INTERNAL PLUG FITTINGS (TYP)
FLANGE CONNECTION
SURGE FLASK
POSITION-5 FUEL PIT
POSITION-1 (NOTE 2)
POSITION-2
POSITION-3
POSITION-4
POSITION-6 SHIELD TANK CAVITY
FOR CONTINUATION SEE DWG. 9699-FM-8C
DETAIL "A"
2" SCH10
FOR ACTUAL EQUIP. DESCRIPTION IN ION EXCH. PIT SEE O.P.F.-5253.2 IN CONTROL ROOM
NOTES:
1. FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE 9699-FM-8A.
2. WHEN REQUIRED BY THE CORE CONDITIONS, A TEMPORARY BY PASS OF THE NO. 1 ION EXCHANGE CAPSULE WILL BE MADE AS SHOWN BY DETAIL "A".
FLOW DIAGRAM
PRIMARY PLANT PURIFICATION
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASSACHUSETTS
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER
9699-FM-8B
30 REV. PER C.U. 89-967
29 GENERAL UPDATE
35 REVISED PER C.U. 90-696
34 REVISED PER CORRECTIVE UPDATE
33 REVISED PER C.U. 91-176
32 REV. PER C.U. 90-519
31 REV. PER GENERAL UPDATE 88-089


VAPOR CONTAINER
MAIN COOLANT PUMPS
P-14-1
P-14-2
P-14-3
P-14-4
REACTOR E-6
SHIELD TANK CAVITY
NEUTRON SHIELD TANK
NEUTRON SHIELD TANK COOLERS (8 REQUIRED)
CONNS. FOR TV CAMERA (COOLING)
GROUND JOINT UNIONS (TYPICAL) ARRANGE JOINT UNION SO EACH CAN BE DISCONNECTED AND TESTED
SCH. 302 IN CONCRETE
HIGH PRESSURE SAMPLE COOLER SC-1
FROM LOOP DRAIN LINES SEE DWG. 9699-FM-6A
FROM BLEED LINE SAMPLE DWG. 9699-FM-8A
TO SAMPLE SYSTEM SEE DWG. 1M-H-8
TO NEUTRON SHIELD SURGE TANK SEE DWG. 9699-FM-8A
NEUTRON SHIELD TANK FILL AND MAKEUP
CONT. ON DWG. 9699-FM-9B
NOTES:
1. PUMPS & HEAT EXCHANGER VENTS & DRAINS TO BE ADDED LATER AS REQUIRED
2. PUMP PACKING SEALING LINES TO BE ADDED LATER IF REQUIRED
3. ALL BALL CHECK VALVES ON VALVE STEM LEAKOFF LINES TO BE VCS-60G
4. VALVE STEM & PUMP LEAKOFF CONNECTIONS PROVIDED WITH A GROUND JOINT SOCKET WELDED UNION ADJACENT TO THE VALVE
5. FOR VALVE STEM LEAKOFF SYSTEM INFORMATION SEE 9699-FM-5A
3 REV. PER C.U.
2 REV. PER EDCP 86-318 7-30-87 PMR 7-30-87 GC 8-4-87 JD
1 REV. PER EDCR 86-315 7-20-87 PMR 7-20-87 GC 7-20-87 KD
0 THIS DWG. SUPERSEDES 9699-FM-9A REV.39 IN PART 6-1-87 PMR 6-1-87 GC 6-1-87 RMS
REV DESCRIPTION BY CHKD. APPD.
YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
DRAWING TITLE
FLOW DIAGRAM
COMPONENT COOLING SYSTEM
JOB NO.
DRAWING NO. 9699-FM-9A



REV	DESCRIPTION	BY	CHKD.	APPD.
5	REVISED PER ISI WALKDOWN AND C.U. 91-277	10-22-92 JW	10-22-92 TJK	10-23-92 WTT
	REVISED PER EDCR 89-309	9-21-92 JW	10-22-92 RHR	10-22-92 PG
4	REV PER CORRECTIVE UPDATE	4-23-92 JR	4-23-92 PWR	7-8-92 RMS
3	REVISED PER EDCR 88-310 AND O.U.	3-5-90 PWR	3-5-90 GC	3-7-90 DAV.
2	REV PER CORRECTIVE UPDATE 89-140	8-17-89 PWR	8-22-89 GC	8-28-89 JRL
1	REVISED PER EDCR 88-310	3-20-89 PWR	3-20-89 GC	3-22-89 DF
0	THIS DWG. SUPERSEDES 9699-FM-9A REV.39 IN PART	6-1-87 PWR	6-1-87 GC	6-1-87 RMS

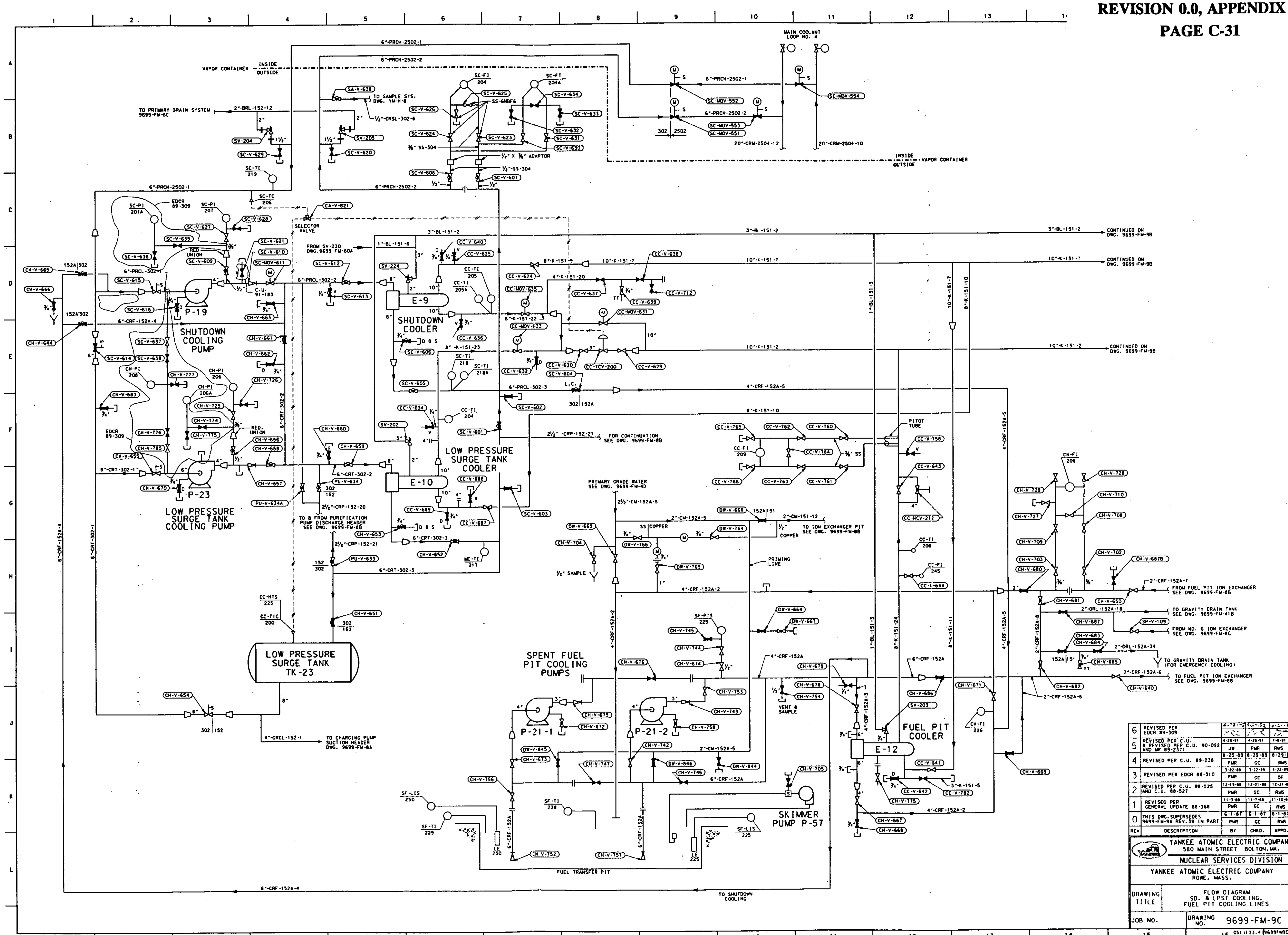
YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION

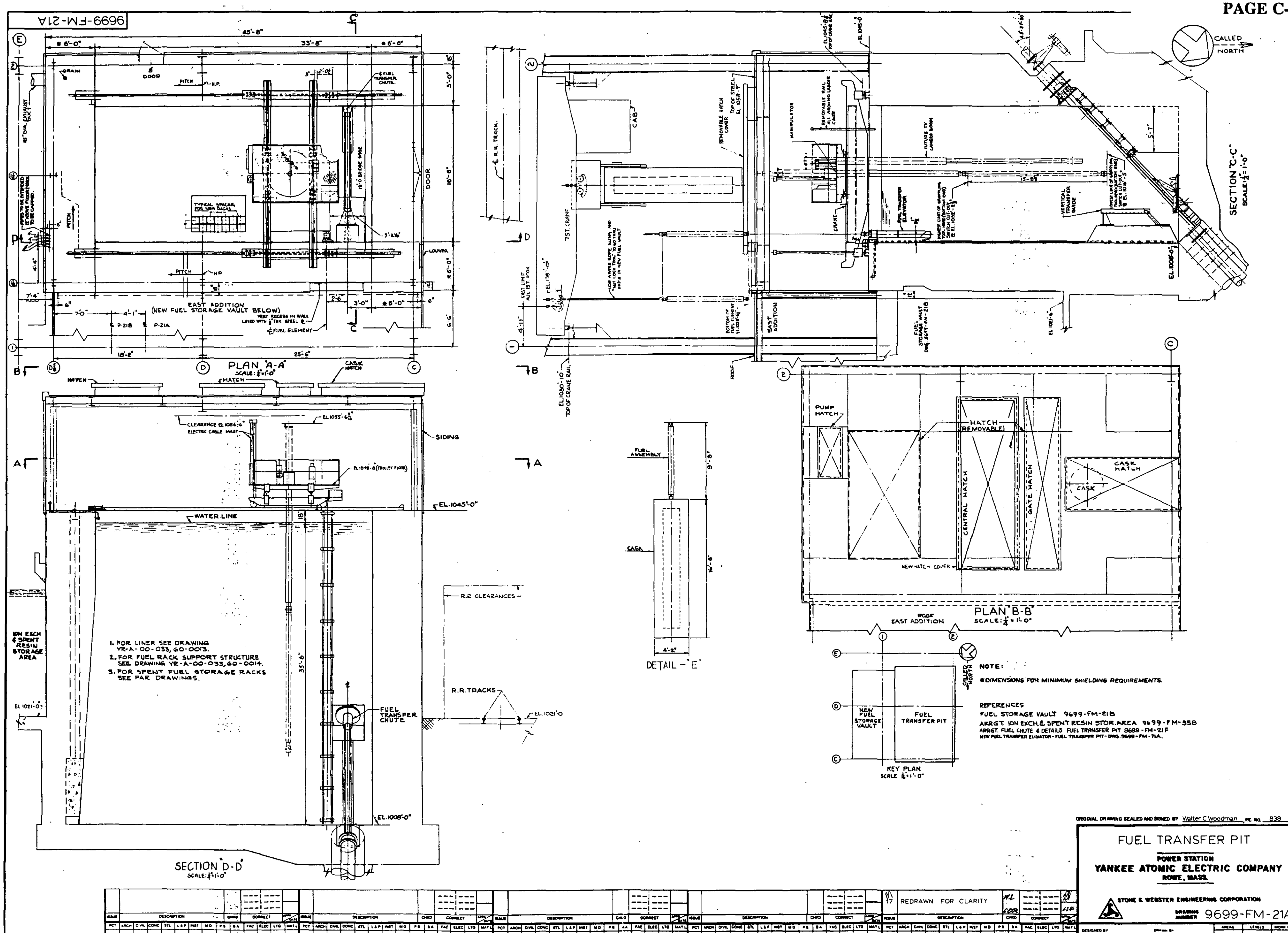
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.

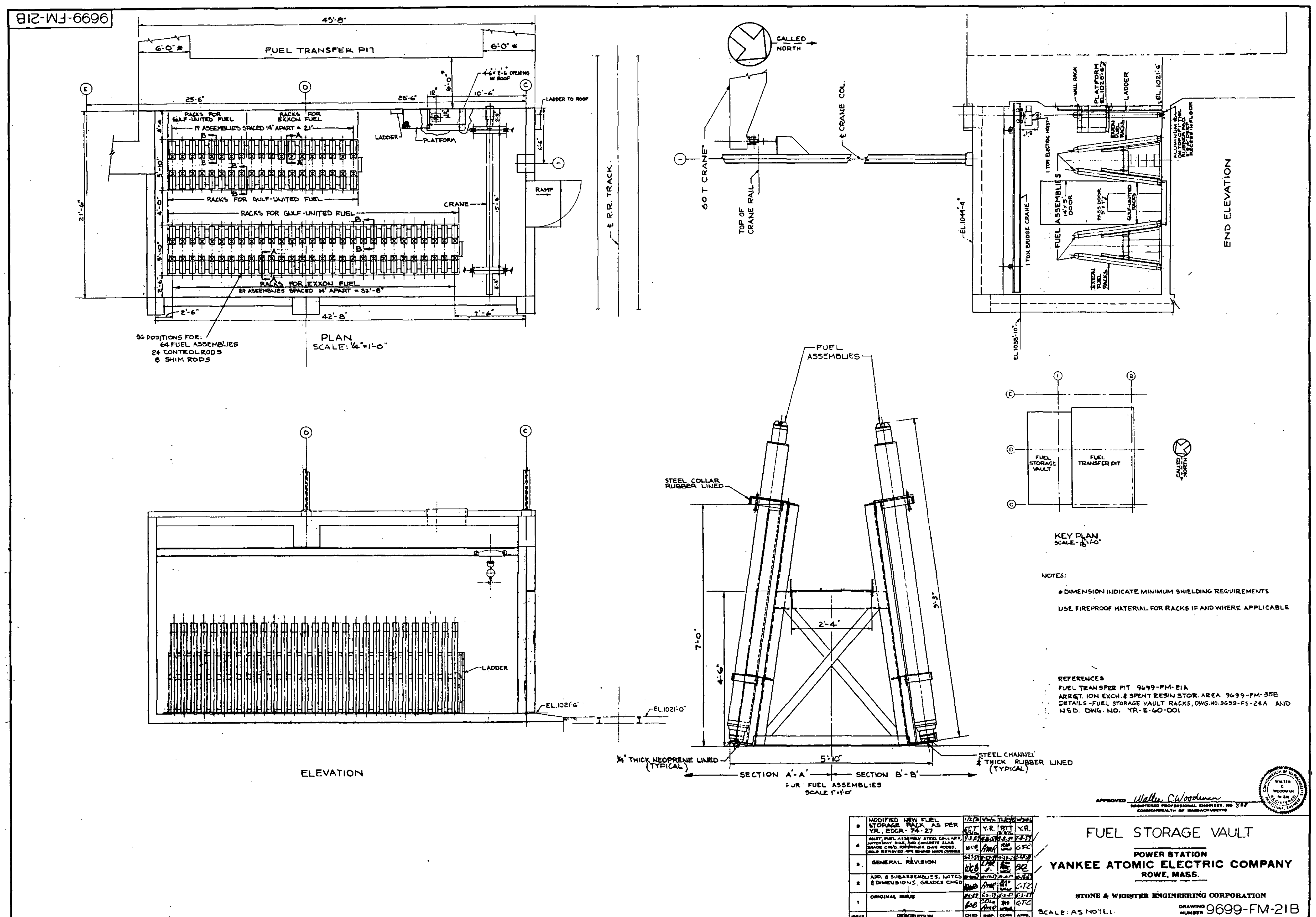
DRAWING TITLE: FLOW DIAGRAM COMPONENT COOLING SYSTEM

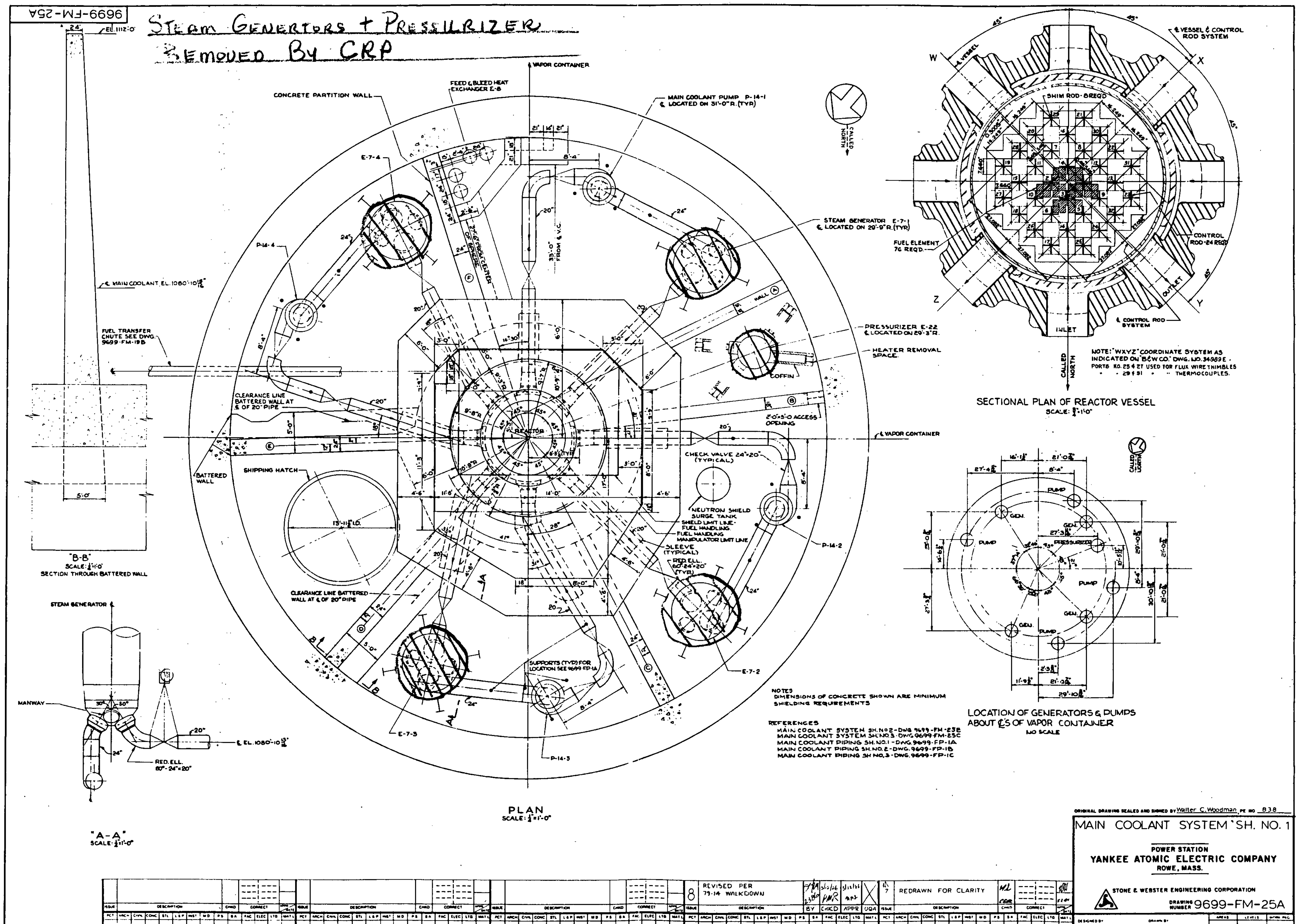
JOB NO. | DRAWING NO. 9699-FM-98

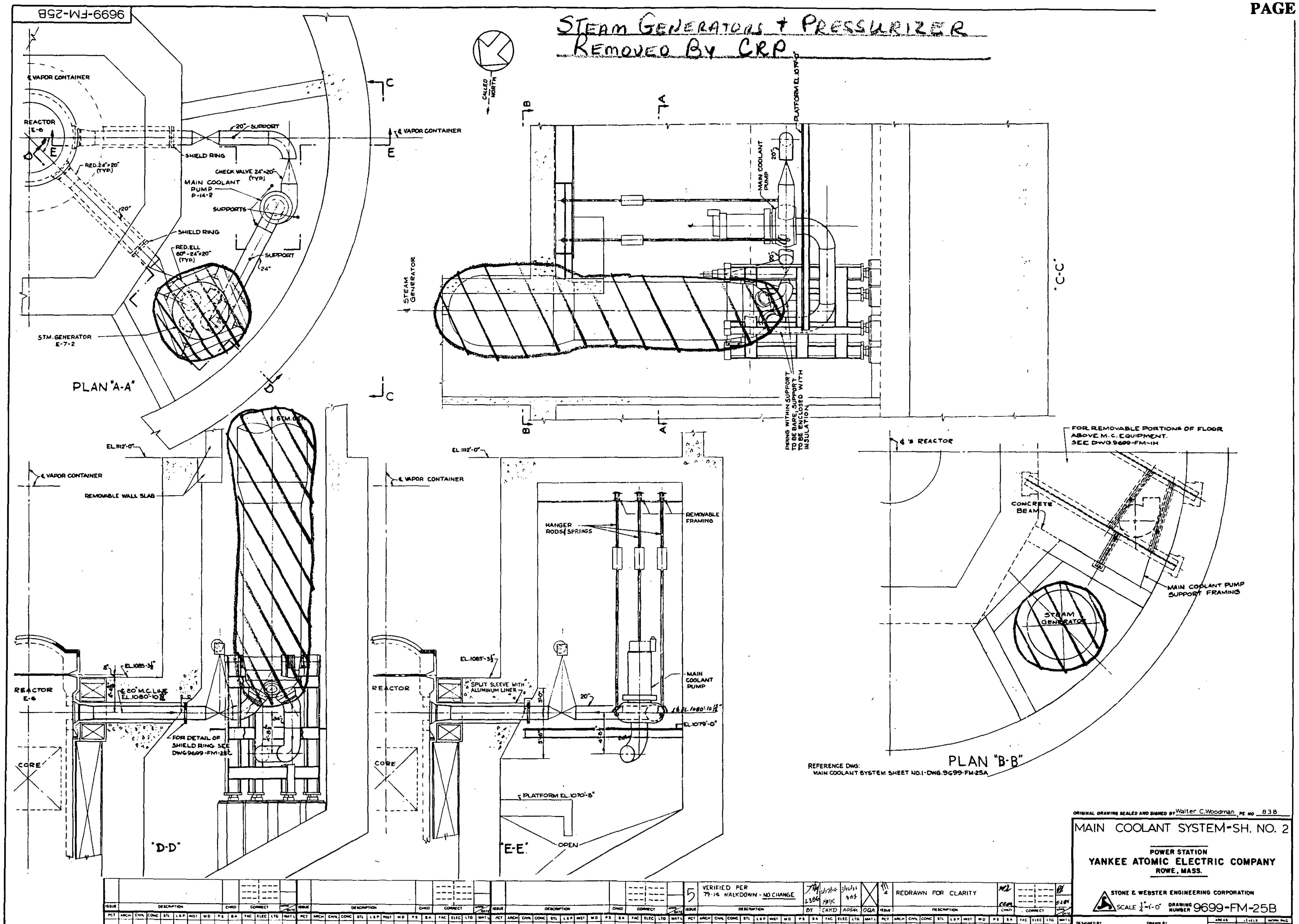
YR3304 | 9699FM98.DGN

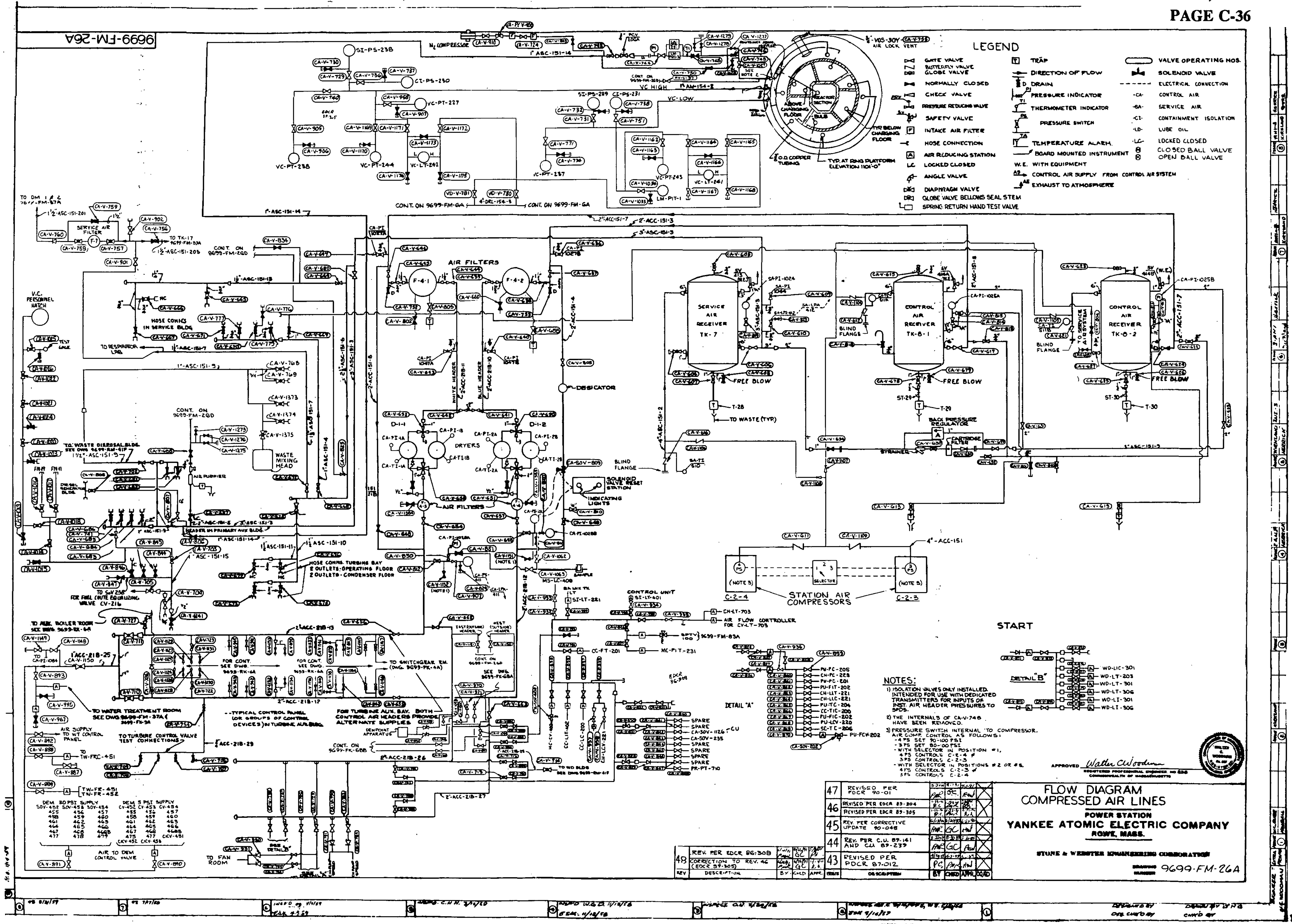

SHUTDOWN COOLING PUMP
P-19
SHUTDOWN COOLER
E-9
LOW PRESSURE SURGE TANK COOLER
E-10
LOW PRESSURE SURGE TANK COOLING PUMP
P-23
LOW PRESSURE SURGE TANK TK-23
SPENT FUEL PIT COOLING PUMPS
P-21-1
P-21-2
SKIMMER PUMP P-57
FUEL PIT COOLER
E-12
FUEL TRANSFER PIT
TO SHUTDOWN COOLING
TO CHARGING PUMP SUCTION HEADER DWG. 9699-FM-8A
TO PRIMARY DRAIN SYSTEM 9699-FM-6C
VAPOR CONTAINER
INSIDE
OUTSIDE
MAIN COOLANT LOOP NO. 4
CONTINUED ON DWG. 9699-FM-9B
YANKEE ATOMIC ELECTRIC COMPANY
580 MAIN STREET BOLTON, MA.
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
DRAWING TITLE
FLOW DIAGRAM
SD. & LPST COOLING,
FUEL PIT COOLING LINES
JOB NO.
DRAWING NO.
9699-FM-9C

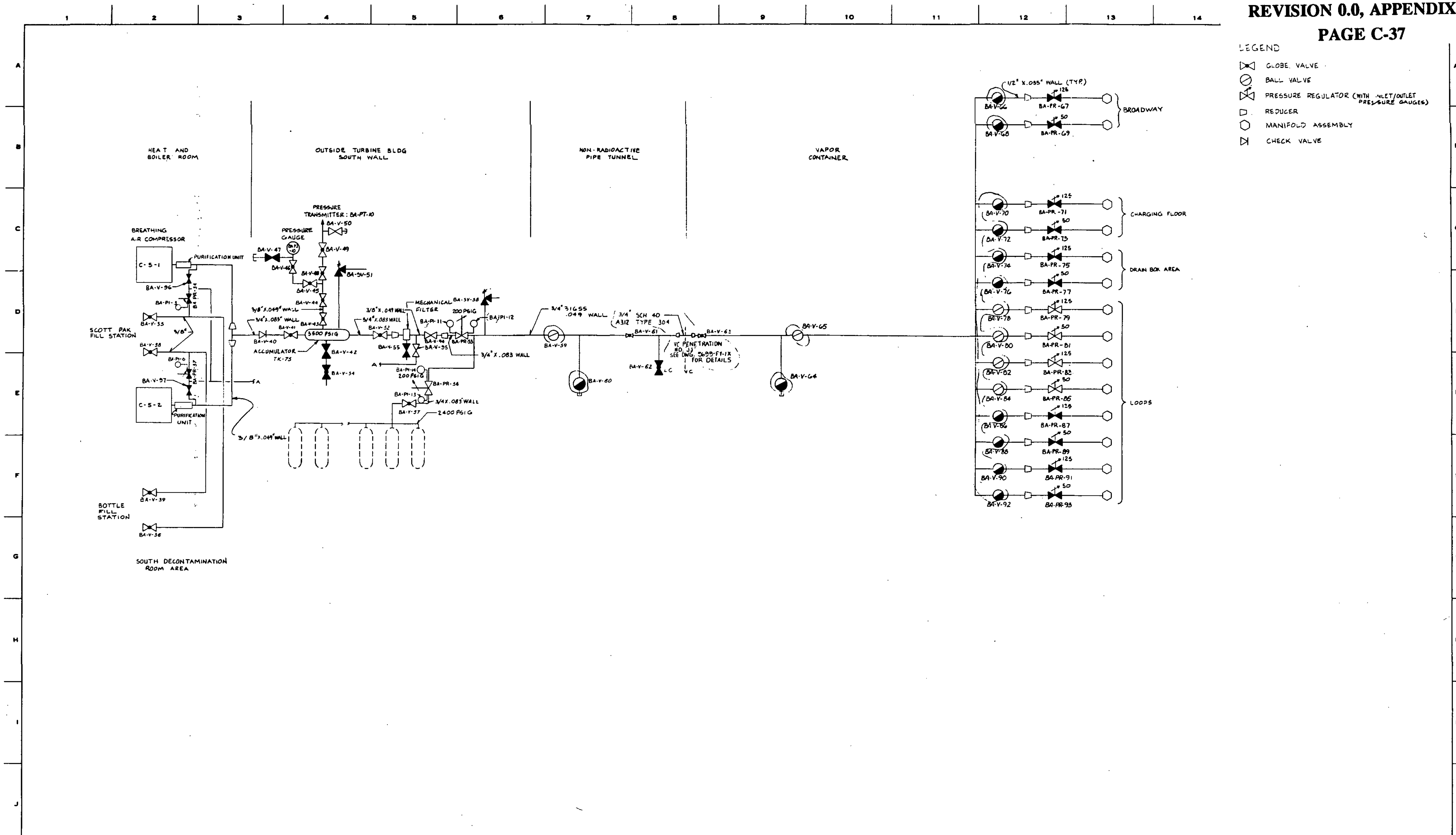

9699-FM-21A
PLAN "A-A"
SCALE: 1/8"=1'-0"
EAST ADDITION
(NEW FUEL STORAGE VAULT BELOW)
SECTION "C-C"
SCALE: 1/8"=1'-0"
CALLED NORTH
PLAN "B-B"
SCALE: 1/4"=1'-0"
PUMP HATCH
HATCH (REMOVABLE)
CENTRAL HATCH
GATE HATCH
CASK HATCH
CASK
NEW HATCH COVER
ROOF
EAST ADDITION
SECTION "D-D"
SCALE: 1/8"=1'-0"
DETAIL - "E"
FUEL ASSEMBLY
CASK
SIDING
WATER LINE
EL. 1045'-0"
EL. 1008'-0"
EL. 1021'-0"
R.R. CLEARANCES
R.R. TRACKS
FUEL TRANSFER CHUTE
ION EXCH & SPENT RESIN STORAGE AREA
1. FOR LINER SEE DRAWING YR-A-00-033, 60-0013.
2. FOR FUEL RACK SUPPORT STRUCTURE SEE DRAWING YR-A-00-033, 60-0014.
3. FOR SPENT FUEL STORAGE RACKS SEE PAR DRAWINGS.
NOTE:
* DIMENSIONS FOR MINIMUM SHIELDING REQUIREMENTS.
REFERENCES
FUEL STORAGE VAULT 9699-FM-21B
ARRGT. ION EXCH. & SPENT RESIN STOR. AREA 9699-FM-35B
ARRGT. FUEL CHUTE & DETAILS FUEL TRANSFER PIT 9699-FM-21F
NEW FUEL TRANSFER ELEVATOR - FUEL TRANSFER PIT - DWG. 9699-FM-71A
NEW FUEL STORAGE VAULT
FUEL TRANSFER PIT
KEY PLAN
SCALE 1/8"=1'-0"
REDRAWN FOR CLARITY
ORIGINAL DRAWING SEALED AND SIGNED BY Walter C Woodman PE NO. 838
FUEL TRANSFER PIT
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-FM-21A


9699-FM-21B
FUEL TRANSFER PIT
RACKS FOR GULF-UNITED FUEL
RACKS FOR EXXON FUEL
RACKS FOR EXXON FUEL
LADDER
PLATFORM
CRANE
RAMP
LADDER TO ROOF
℄ R.R. TRACK
96 POSITIONS FOR:
64 FUEL ASSEMBLIES
24 CONTROL RODS
8 SHIM RODS
PLAN
SCALE: 1/4"=1'-0"
CALLED NORTH
60 T CRANE
TOP OF CRANE RAIL
℄ CRANE COL.
END ELEVATION
FUEL ASSEMBLIES
1 TON BRIDGE CRANE
STEEL COLLAR RUBBER LINED
1/4" THICK NEOPRENE LINED (TYPICAL)
STEEL CHANNEL 1/8" THICK RUBBER LINED (TYPICAL)
SECTION A'-A'
SECTION B'-B'
FOR FUEL ASSEMBLIES
SCALE 1"=1'-0"
ELEVATION
EL. 1021'-6"
EL. 1021'-0"
FUEL STORAGE VAULT
FUEL TRANSFER PIT
KEY PLAN
NOTES:
* DIMENSION INDICATE MINIMUM SHIELDING REQUIREMENTS
USE FIREPROOF MATERIAL FOR RACKS IF AND WHERE APPLICABLE
REFERENCES
FUEL TRANSFER PIT 9699-FM-21A
ARRGT. ION EXCH. & SPENT RESIN STOR. AREA 9699-FM-35B
DETAILS - FUEL STORAGE VAULT RACKS, DWG. NO. 9699-FS-24A AND
N.S.D. DWG. NO. YR-E-60-001
GENERAL REVISION
ORIGINAL ISSUE
DESCRIPTION
FUEL STORAGE VAULT
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
SCALE: AS NOTED
DRAWING NUMBER 9699-FM-21B


9699-FM-25A
Steam Generators + Pressurizer
Removed By CRP
VAPOR CONTAINER
FEED & BLEED HEAT EXCHANGER E-8
CONCRETE PARTITION WALL
MAIN COOLANT PUMP P-14-1 ℄ LOCATED ON 31'-0" R. (TYP)
CALLED NORTH
STEAM GENERATOR E-7-1 ℄ LOCATED ON 29'-9" R. (TYP)
PRESSURIZER E-22 ℄ LOCATED ON 29'-3" R.
HEATER REMOVAL SPACE
COFFIN
2'-0"×5'-0" ACCESS OPENING
℄ VAPOR CONTAINER
CHECK VALVE 24"×20" (TYPICAL)
NEUTRON SHIELD SURGE TANK
SHIELD LIMIT LINE-FUEL HANDLING
FUEL HANDLING MANIPULATOR LIMIT LINE
SLEEVE (TYPICAL)
RED. ELL. 60°-24"×20" (TYP.)
P-14-2
E-7-2
E-7-3
P-14-3
P-14-4
E-7-4
REACTOR
SHIPPING HATCH
13'-11½" I.D.
BATTERED WALL
CLEARANCE LINE BATTERED WALL AT ℄ OF 20" PIPE
CLEARANCE LINE BATTERED WALL AT ℄ OF 20" PIPE
SUPPORTS (TYP) FOR LOCATION SEE 9699-FP-1A
℄ MAIN COOLANT EL. 1080'-10⅝"
FUEL TRANSFER CHUTE SEE DWG. 9699-FM-19B
EL. 1112'-0"
"B-B"
SCALE: ¼"=1'-0"
SECTION THROUGH BATTERED WALL
STEAM GENERATOR ℄
MANWAY
RED. ELL. 60°-24"×20"
℄ EL. 1080'-10⅝"
"A-A"
SCALE: ¼"=1'-0"
PLAN
SCALE: ¼"=1'-0"
℄ VESSEL & CONTROL ROD SYSTEM
SHIM ROD-8 REQD
FUEL ELEMENT 76 REQD.
CONTROL ROD-24 REQD
OUTLET
INLET
℄ CONTROL ROD SYSTEM
CALLED NORTH
NOTE: "WXYZ" COORDINATE SYSTEM AS INDICATED ON "B&W CO." DWG. NO. 34989E. PORTS NO. 25 & 27 USED FOR FLUX WIRE THIMBLES • 29 & 31 • THERMOCOUPLES.
SECTIONAL PLAN OF REACTOR VESSEL
SCALE: ⅜"=1'-0"
PUMP
GEN.
PRESSURIZER
LOCATION OF GENERATORS & PUMPS ABOUT ℄'S OF VAPOR CONTAINER
NO SCALE
NOTES
DIMENSIONS OF CONCRETE SHOWN ARE MINIMUM SHIELDING REQUIREMENTS
REFERENCES
MAIN COOLANT SYSTEM SH. NO. 2 - DWG. 9699-FM-25B
MAIN COOLANT SYSTEM SH. NO. 3 - DWG. 9699-FM-25C
MAIN COOLANT PIPING SH. NO. 1 - DWG. 9699-FP-1A
MAIN COOLANT PIPING SH. NO. 2 - DWG. 9699-FP-1B
MAIN COOLANT PIPING SH. NO. 3 - DWG. 9699-FP-1C
ORIGINAL DRAWING SEALED AND SIGNED BY Walter C. Woodman PE NO 838
MAIN COOLANT SYSTEM SH. NO. 1
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-FM-25A
8 REVISED PER 79-14 WALKDOWN
7 REDRAWN FOR CLARITY


9699-FM-25B
STEAM GENERATORS + PRESSURIZER
REMOVED BY CRP
CALLED NORTH
℄ VAPOR CONTAINER
REACTOR E-6
20" SUPPORT
SHIELD RING
RED. 24"×20" (TYP.)
CHECK VALVE 24"×20" (TYP.)
MAIN COOLANT PUMP P-14-2
SUPPORTS
RED. ELL 60°-24"×20" (TYP.)
SUPPORT
24"
STM. GENERATOR E-7-2
PLAN "A-A"
STEAM GENERATOR
MAIN COOLANT PUMP
PLATFORM EL. 1070'-8"
PIPING WITHIN SUPPORT TO BE BARE, SUPPORT TO BE ENCLOSED WITH INSULATION
"C-C"
EL. 1112'-0"
REMOVABLE WALL SLAB
EL. 1085'-3½"
℄ 20" M.C. LINE EL. 1080'-10½"
FOR DETAIL OF SHIELD RING SEE DWG 9699-FM-25C
CORE
"D-D"
SPLIT SLEEVE WITH ALUMINUM LINER
HANGER RODS & SPRINGS
REMOVABLE FRAMING
MAIN COOLANT PUMP
EL. 1079'-0"
PLATFORM EL. 1070'-8"
OPEN
"E-E"
℄ REACTOR
FOR REMOVABLE PORTIONS OF FLOOR ABOVE M.C. EQUIPMENT. SEE DWG. 9699-FM-1H
CONCRETE BEAM
MAIN COOLANT PUMP SUPPORT FRAMING
STEAM GENERATOR
PLAN "B-B"
REFERENCE DWG: MAIN COOLANT SYSTEM SHEET NO. 1-DWG. 9699-FM-25A
ORIGINAL DRAWING SEALED AND SIGNED BY Walter C. Woodman PE NO. 838
MAIN COOLANT SYSTEM-SH. NO. 2
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
SCALE ¼"=1'-0"
DRAWING NUMBER 9699-FM-25B
5 VERIFIED PER 79-14 WALKDOWN - NO CHANGE
REDRAWN FOR CLARITY


LEGEND
GATE VALVE
BUTTERFLY VALVE
GLOBE VALVE
NORMALLY CLOSED
CHECK VALVE
PRESSURE REDUCING VALVE
SAFETY VALVE
INTAKE AIR FILTER
HOSE CONNECTION
AIR REDUCING STATION
LOCKED CLOSED
ANGLE VALVE
DIAPHRAGM VALVE
GLOBE VALVE BELLOWS SEAL STEM
SPRING RETURN HAND TEST VALVE
TRAP
DIRECTION OF FLOW
DRAIN
PRESSURE INDICATOR
THERMOMETER INDICATOR
PRESSURE SWITCH
TEMPERATURE ALARM
BOARD MOUNTED INSTRUMENT
W.E. WITH EQUIPMENT
CONTROL AIR SUPPLY FROM CONTROL AIR SYSTEM
EXHAUST TO ATMOSPHERE
VALVE OPERATING NOS.
SOLENOID VALVE
ELECTRICAL CONNECTION
-CA- CONTROL AIR
-SA- SERVICE AIR
-CI- CONTAINMENT ISOLATION
-LO- LUBE OIL
-LC- LOCKED CLOSED
CLOSED BALL VALVE
OPEN BALL VALVE
AIR FILTERS
DRYERS
SERVICE AIR RECEIVER TK-7
CONTROL AIR RECEIVER TK-8-1
CONTROL AIR RECEIVER TK-8-2
FREE BLOW
STATION AIR COMPRESSORS
C-2-4
C-2-3
START
DETAIL "A"
DETAIL "B"
NOTES:
1) ISOLATION VALVES ONLY INSTALLED. INTENDED FOR USE WITH DEDICATED TRANSMITTERS FOR INPUTS OF INST. AIR HEADER PRESSURES TO SPDS.
2) THE INTERNALS OF CA-V-748 HAVE BEEN REMOVED.
3) PRESSURE SWITCH INTERNAL TO COMPRESSOR. AIR COMP. CONTROL AS FOLLOWS:
-4 PS SET 90-100 PSI
-3 PS SET 80-90 PSI
-WITH SELECTOR IN POSITION #1, 4 PS CONTROLS C-2-4, 3 PS CONTROLS C-2-3
-WITH SELECTOR IN POSITIONS #2 OR #3, 4 PS CONTROLS C-2-3, 3 PS CONTROLS C-2-4
APPROVED Walter C. Woodman
REGISTERED PROFESSIONAL ENGINEER NO 636 COMMONWEALTH OF MASSACHUSETTS
47 REVISED PER PDCR 90-01
46 REVISED PER EDCR 89-304
REVISED PER EDCR 89-305
45 REV. PER CORRECTIVE UPDATE 90-048
44 REV. PER C.U. 89-141 AND C.U. 89-239
43 REVISED PER PDCR 87-012
48 REV. PER EDCR 86-308 CORRECTION TO REV. 46 (EDCR 89-305)
REV DESCRIPTION BY CHKD APPR.
DESCRIPTION BY CHKD APPR. COORD
FLOW DIAGRAM
COMPRESSED AIR LINES
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-FM-26A



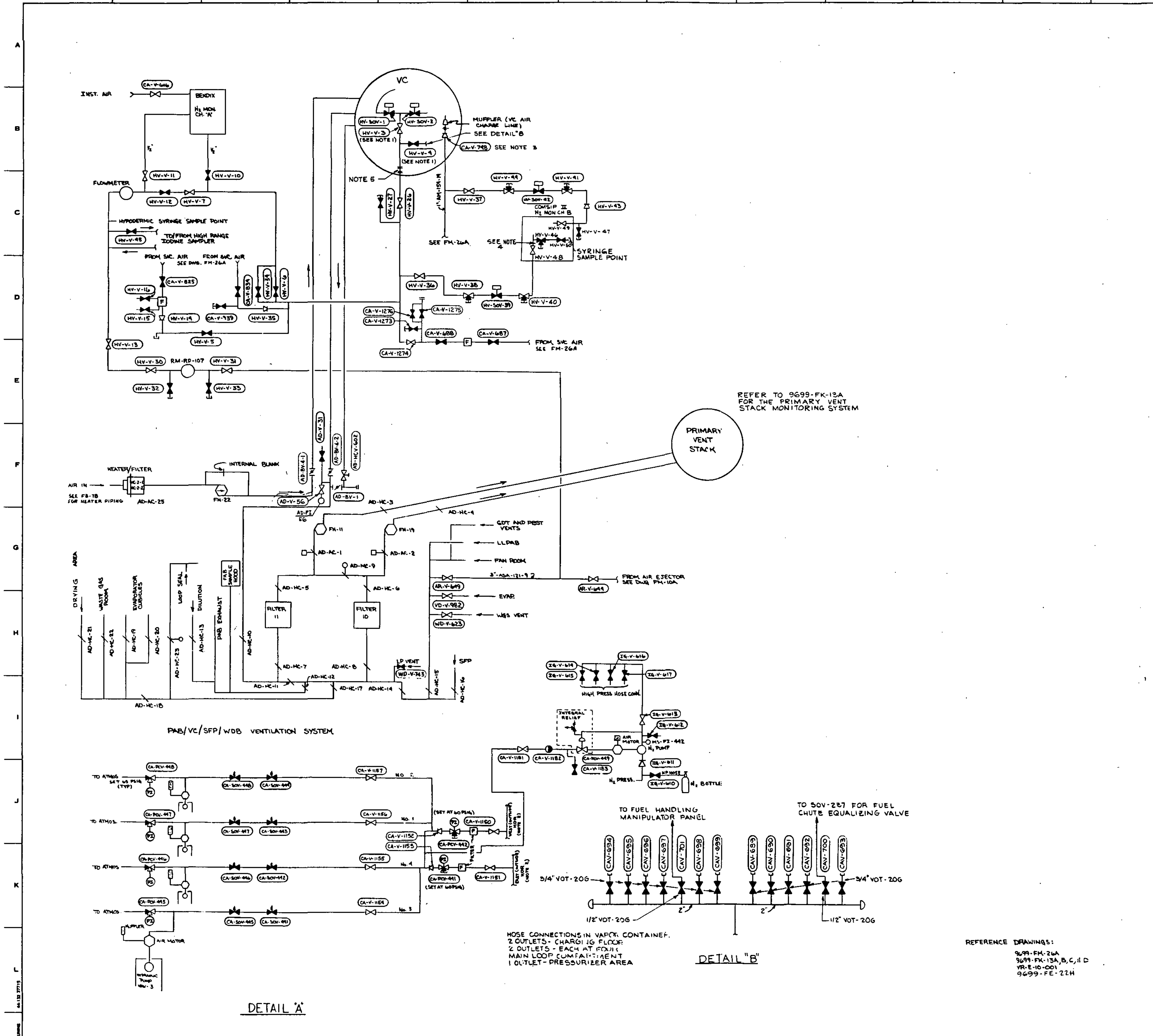
REV.	DESCRIPTION	PREP.	CHKD	APP'D	QDAD REVIEW
3	REVISED PER CORR. UPDATE 89-046	4/24/89 P.C.	PMK		
2	REVISED PER PLT. ALT. 86-026	6/22/87 P.C.	PMK		
1	REV. PER GEN. UPDATE 87-005	4-1-87 PMK	GC	4-3-87	
0	ORIGINAL ISSUE PER EDCR 84-322	6-11-86 RLP	6-11-86 PMK	7-3-86 JRL	

YANKEE ATOMIC ELECTRIC COMPANY
NUCLEAR SERVICES DIVISION

YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.

FLOW DIAGRAM/BREATHING AIR SYSTEM

DESIGNED BY: P. CARTER 1-7-85
ACCOUNT NO.: EXR 84-322
DWG NO.: 9699-FM-26B



NOTES:

1. WHEN INTEGRITY SET, HV-V-3 OPEN AND HV-V-4 CLOSED
   WHEN INTEGRITY IS NOT SET HV-V-3 CLOSED AND HV-V-4 OPEN
2. SEE DWG 9699-FM-26A FOR CONTINUATION.
3. THE INTERNALS OF CA-V-748 HAVE BEEN REMOVED.
4. HV-V-46 IS A SPECIAL PORTED WHITEY VALVE MODEL SS-43XHF4. ALL PORTS ARE EITHER OPEN OR CLOSED.
5. FLANGED SPOOL PIECE INSTALLED TO FACILITATE TYPE C TESTING OF THE LINE.

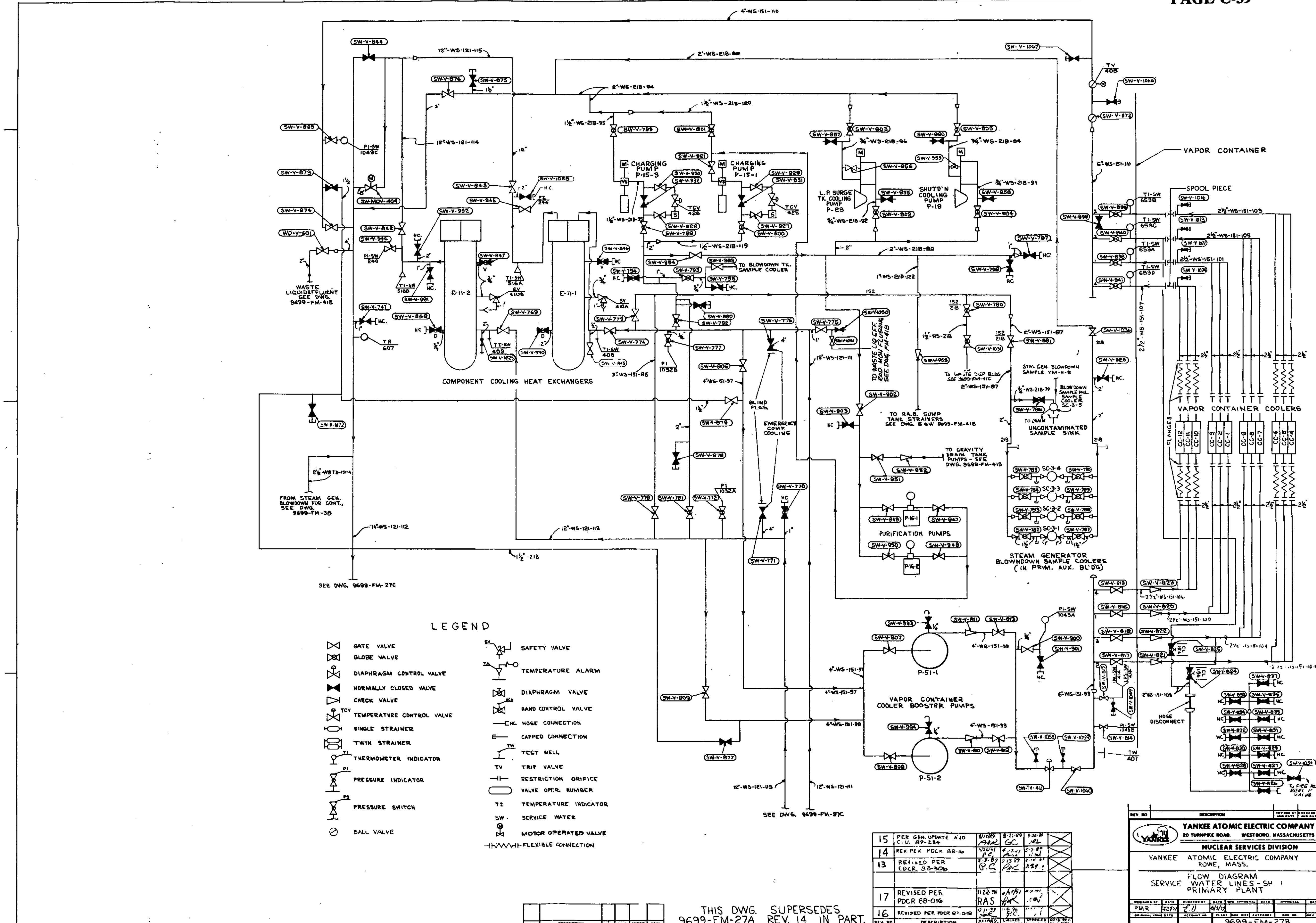
REV.	DESCRIPTION	PREP.	CHKD	APP'D	QUAD REVIEW
15	REV. PER C.U. 91-177	3-27-91 PMR	4-1-91 BF	4-1-91	
14	REV. PER EDCR 87-309	11-15-90 PMR	11-15-90 BC	11-20-90	
13	REVISED PER C.U. 90-535 AND C.U. 90-524	PMR	8-10-90 GC		
12	REVISED PER CORRECTIVE UPDATE 90-053	5-27-90 PMR	5-30-90 GC		
11	REVISED PER C.U. 84-251	BG	PMR		

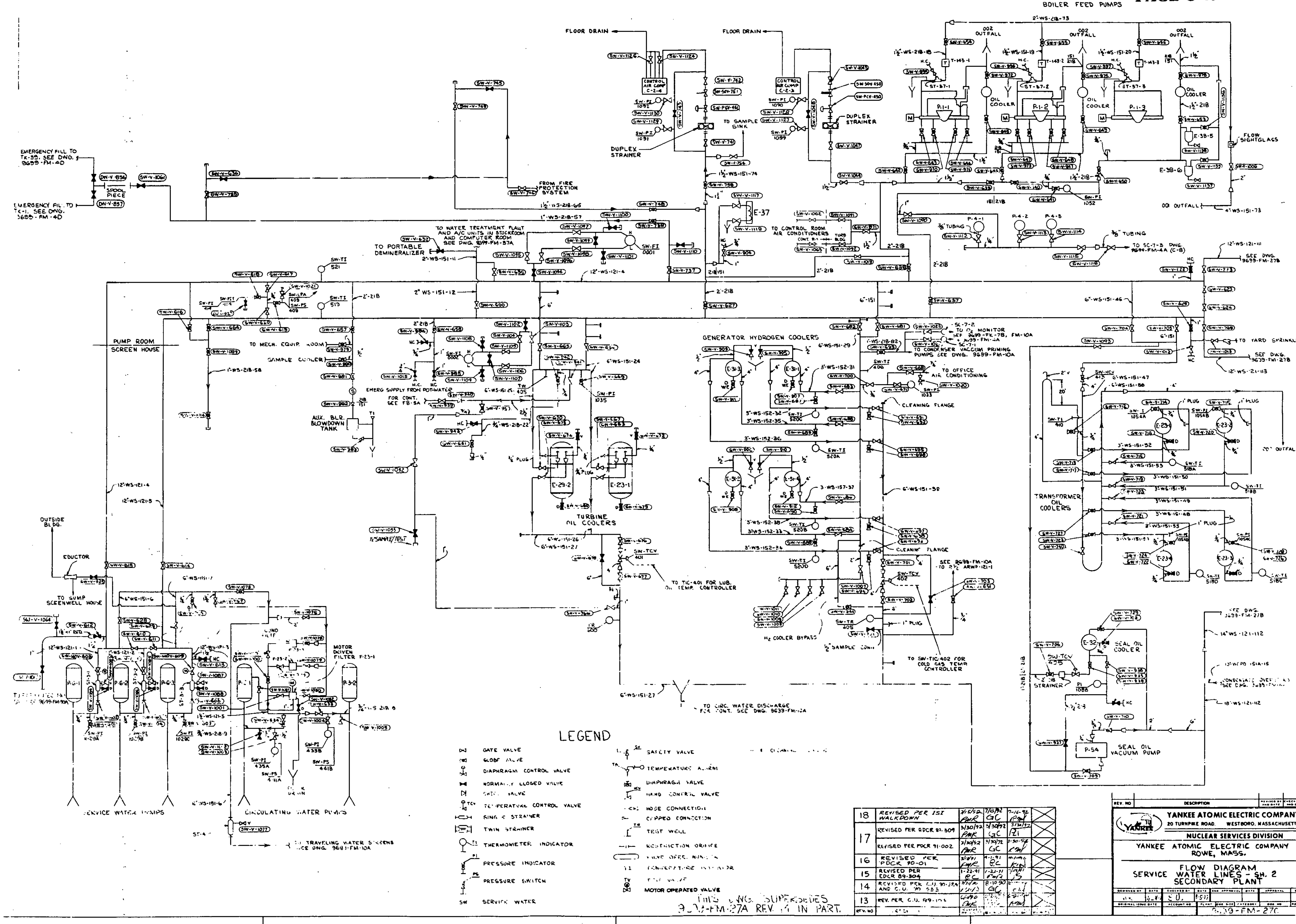
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION

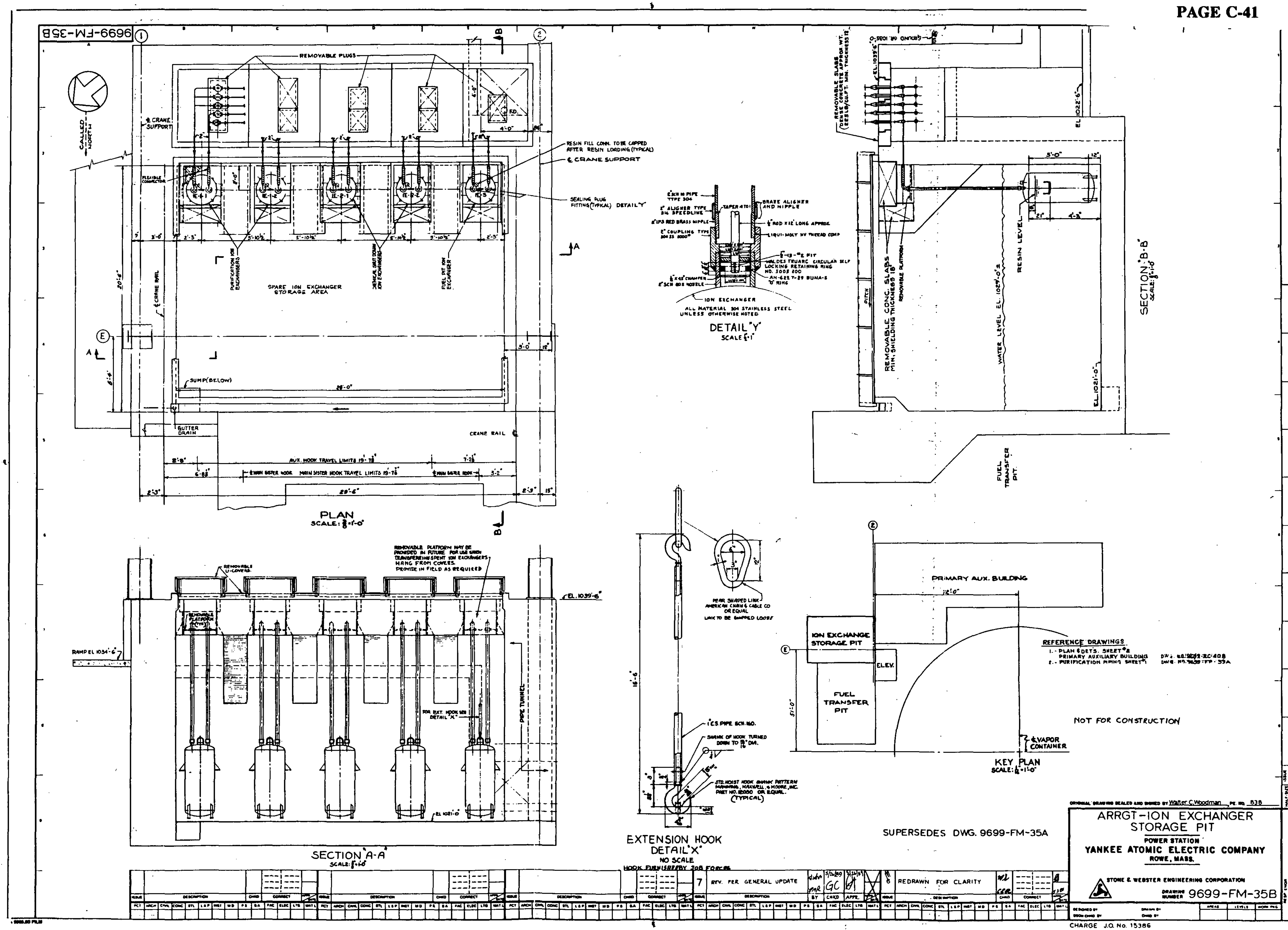
YANKEE ATOMIC ELECTRIC COMPANY
ROWE MA

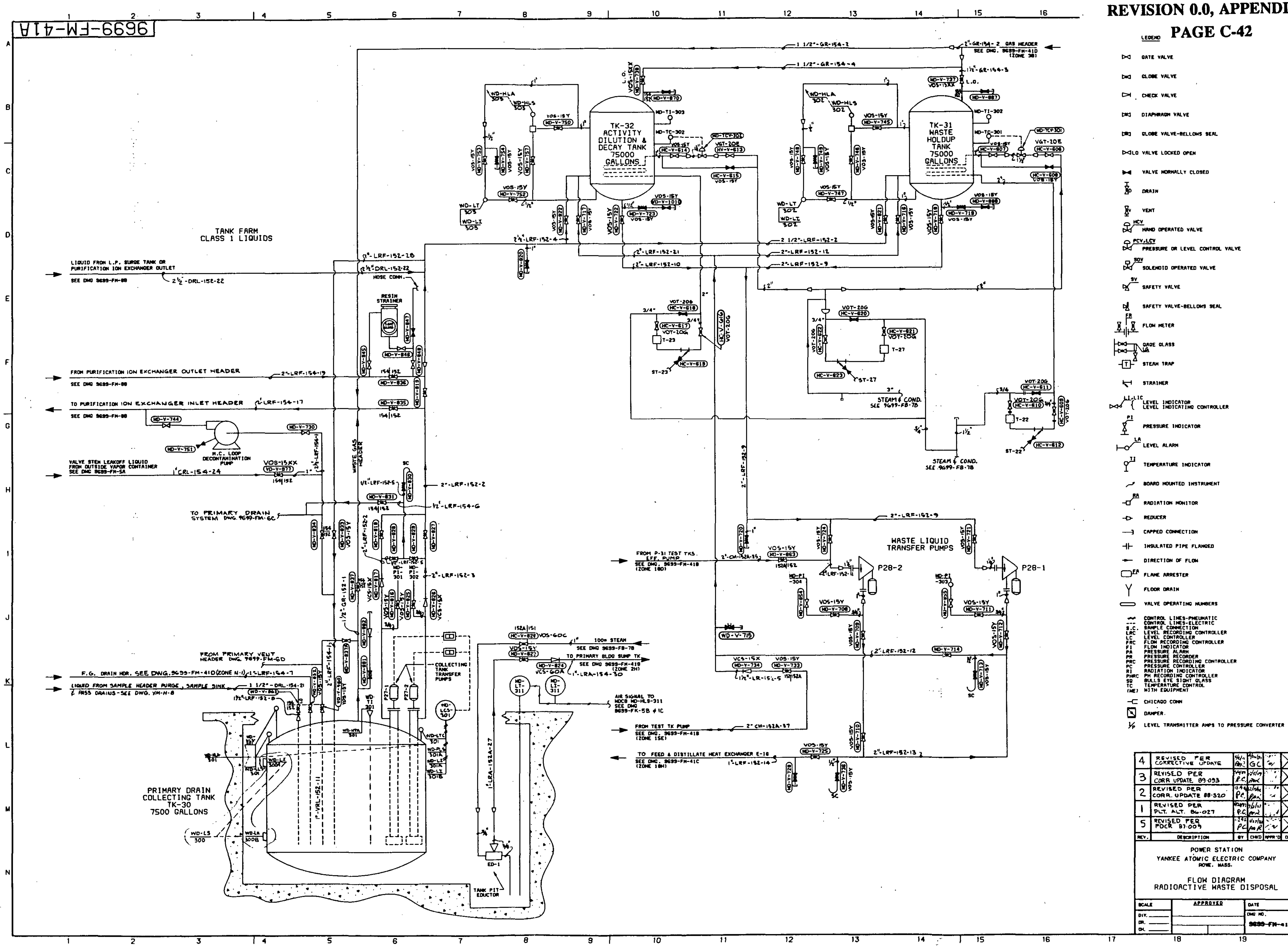
PRIMARY AIR DISPOSAL & HYDROGEN MONITORING WITH MISCELLANEOUS SECTIONS

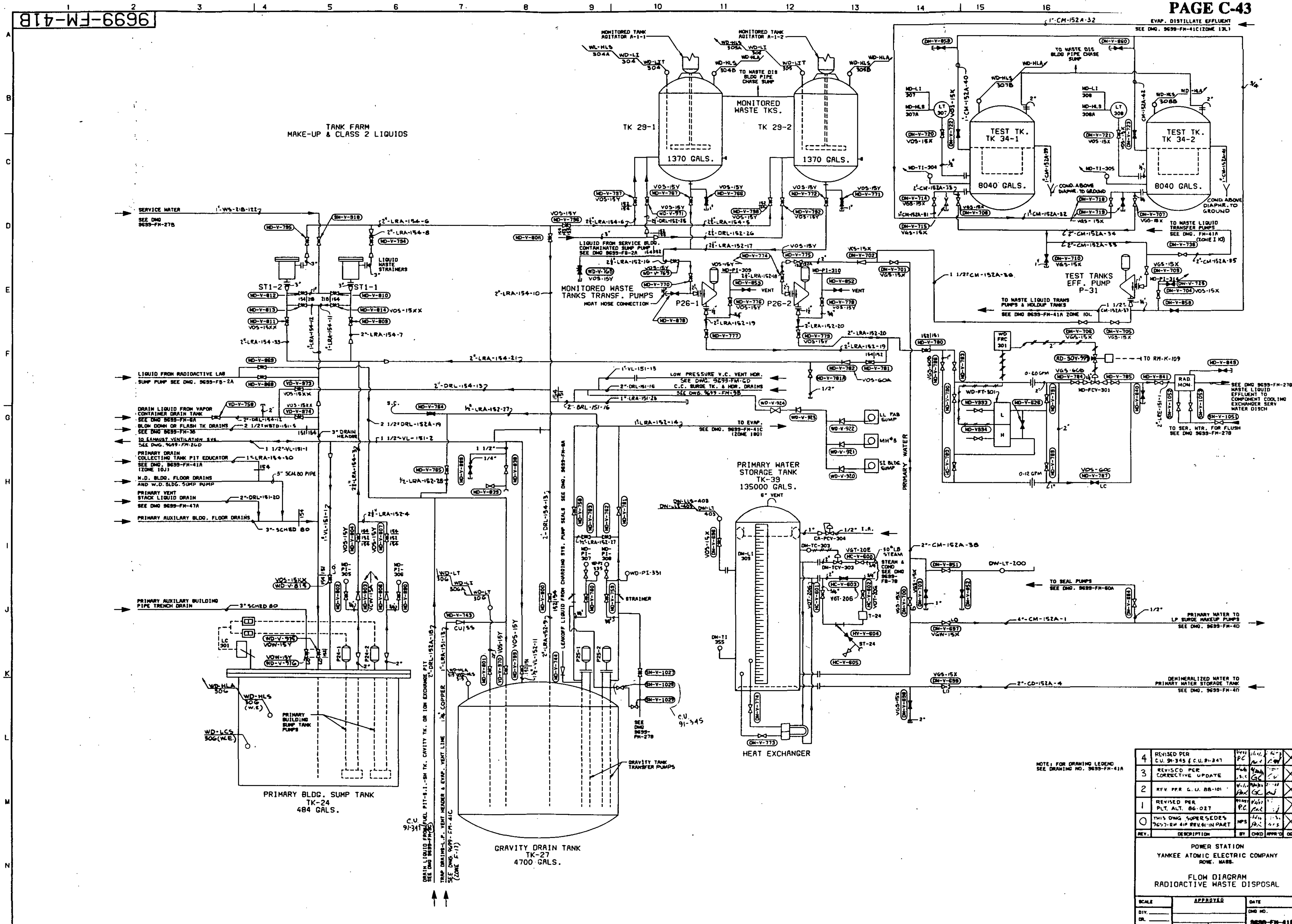
9699 — FM — 26D

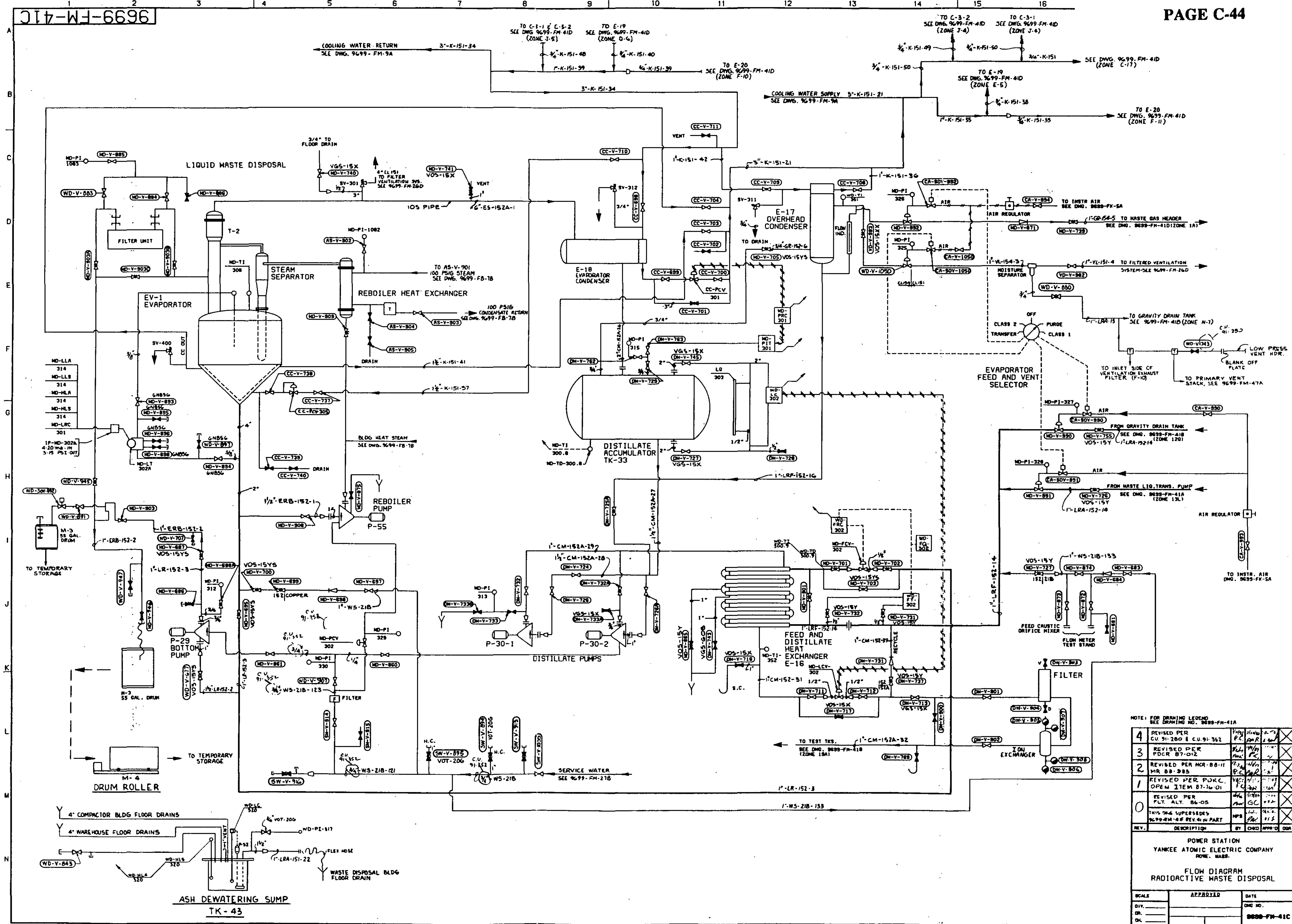

VAPOR CONTAINER
SPOOL PIECE
CHARGING PUMP P-15-3
CHARGING PUMP P-15-1
L.P. SURGE TK. COOLING PUMP P-23
SHUTD'N COOLING PUMP P-19
COMPONENT COOLING HEAT EXCHANGERS
E-11-2
E-11-1
WASTE LIQUID EFFLUENT SEE DWG. 9699-FM-41B
FROM STEAM GEN. BLOWDOWN FOR CONT., SEE DWG. 9699-FM-38
SEE DWG. 9699-FM-27C
TO BLOWDOWN TK. SAMPLE COOLER
BLIND FLGS.
EMERGENCY COMP. COOLING
TO R.A.B. SUMP TANK STRAINERS SEE DWG. 9699-FM-41B
TO GRAVITY DRAIN TANK PUMPS - SEE DWG. 9699-FM-41B
PURIFICATION PUMPS
STM. GEN. BLOWDOWN SAMPLE YM-H-3
BLOWDOWN SAMPLE PNL. SAMPLE COOLER SC-3-5
UNCONTAMINATED SAMPLE SINK
STEAM GENERATOR BLOWDOWN SAMPLE COOLERS (IN PRIM. AUX. BL'DG)
VAPOR CONTAINER COOLERS
FLANGES
VAPOR CONTAINER COOLER BOOSTER PUMPS
P-51-1
P-51-2
HOSE DISCONNECT
LEGEND
GATE VALVE
GLOBE VALVE
DIAPHRAGM CONTROL VALVE
NORMALLY CLOSED VALVE
CHECK VALVE
TEMPERATURE CONTROL VALVE
SINGLE STRAINER
TWIN STRAINER
THERMOMETER INDICATOR
PRESSURE INDICATOR
PRESSURE SWITCH
BALL VALVE
SAFETY VALVE
TEMPERATURE ALARM
DIAPHRAGM VALVE
HAND CONTROL VALVE
HOSE CONNECTION
CAPPED CONNECTION
TEST WELL
TV TRIP VALVE
RESTRICTION ORIFICE
VALVE OPT'R. NUMBER
TI TEMPERATURE INDICATOR
SW SERVICE WATER
MOTOR OPERATED VALVE
FLEXIBLE CONNECTION
THIS DWG. SUPERSEDES 9699-FM-27A REV. 14 IN PART.
15 PER GEN. UPDATE AND C.U. 89-234
14 REV. PER PDCR 88-16
13 REVISED PER EDCR 88-306
17 REVISED PER PDCR 88-016
16 REVISED PER PDCR 87-018
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD, WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY ROWE, MASS.
FLOW DIAGRAM SERVICE WATER LINES - SH. 1 PRIMARY PLANT
9699-FM-27B

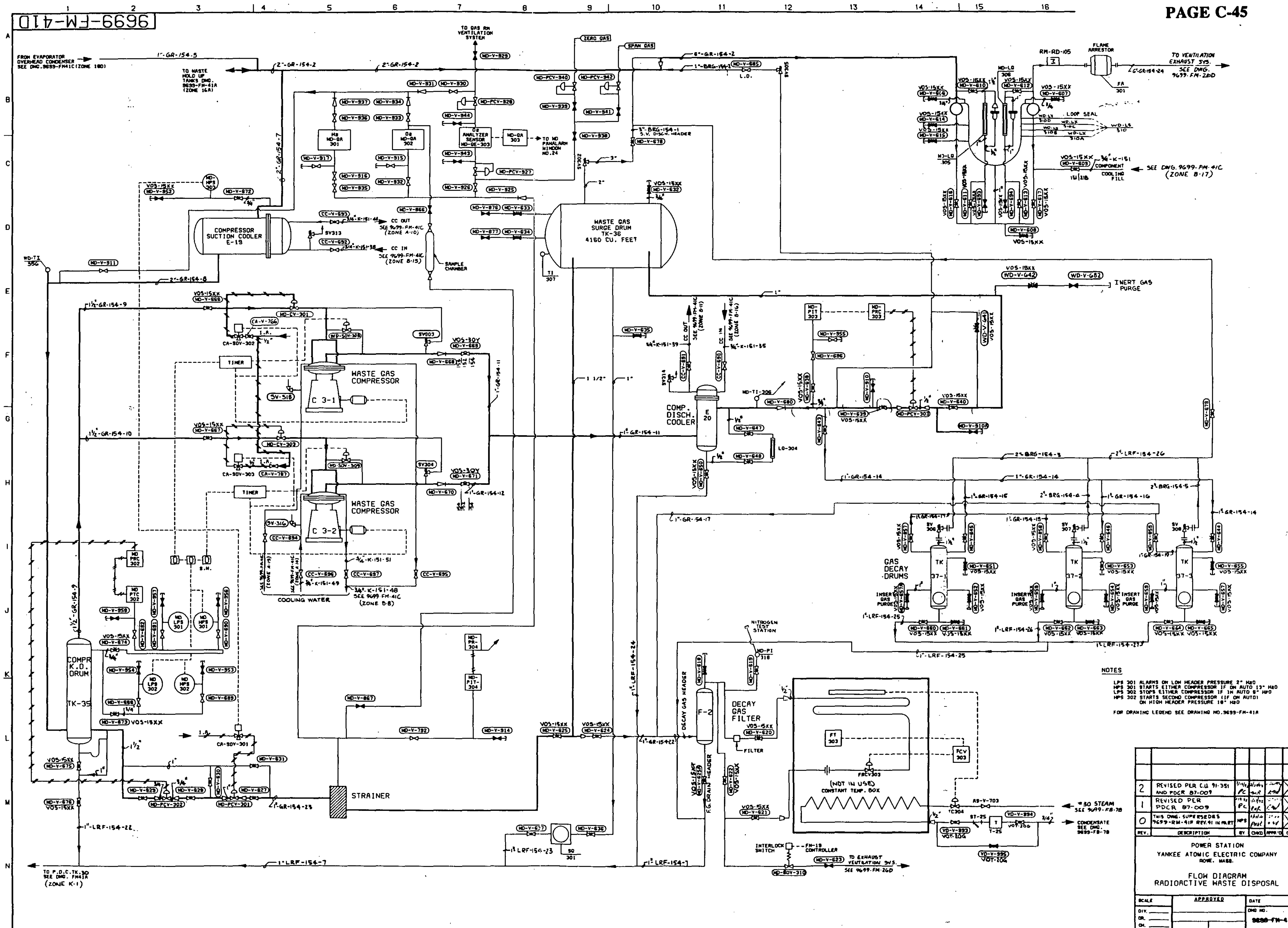

BOILER FEED PUMPS
FLOOR DRAIN
DUPLEX STRAINER
PUMP ROOM
SCREEN HOUSE
AUX. BLR. BLOWDOWN TANK
GENERATOR HYDROGEN COOLERS
TURBINE OIL COOLERS
TRANSFORMER OIL COOLERS
SEAL OIL COOLER
SEAL OIL VACUUM PUMP
SERVICE WATER PUMPS
CIRCULATING WATER PUMPS
LEGEND
GATE VALVE
GLOBE VALVE
DIAPHRAGM CONTROL VALVE
NORMALLY CLOSED VALVE
CHECK VALVE
TEMPERATURE CONTROL VALVE
RING & STRAINER
TWIN STRAINER
THERMOMETER INDICATOR
PRESSURE INDICATOR
PRESSURE SWITCH
SERVICE WATER
SAFETY VALVE
TEMPERATURE ALARM
DIAPHRAGM VALVE
HAND CONTROL VALVE
HOSE CONNECTION
CAPPED CONNECTION
TEST WELL
RESTRICTION ORIFICE
MOTOR OPERATED VALVE
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD, WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY ROWE, MASS.
FLOW DIAGRAM
SERVICE WATER LINES - SH. 2
SECONDARY PLANT

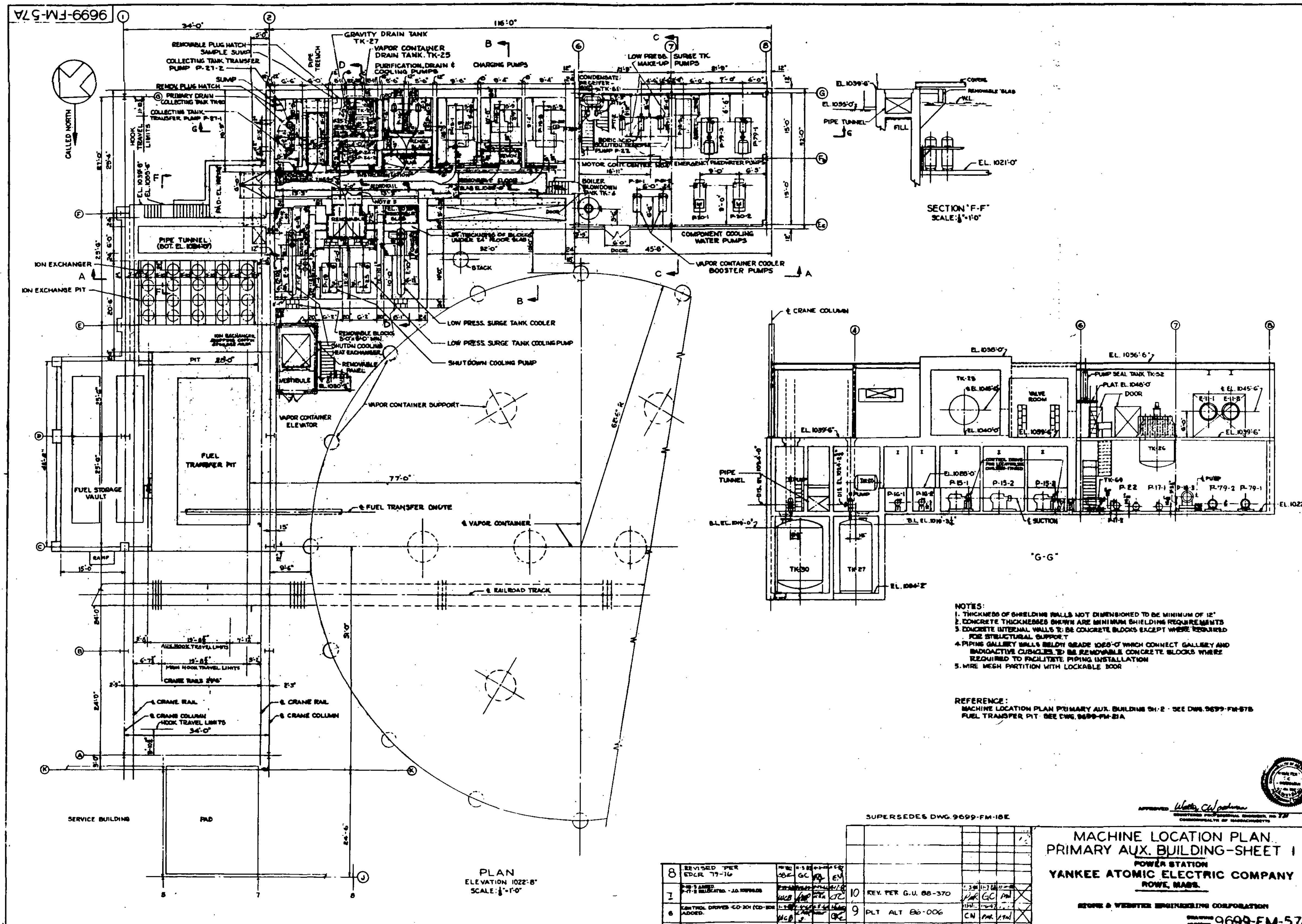

9699-FM-35B
REMOVABLE PLUGS
CALLED NORTH
℄ CRANE SUPPORT
RESIN FILL CONN. TO BE CAPPED AFTER RESIN LOADING (TYPICAL)
℄ CRANE SUPPORT
SEALING PLUG FITTING (TYPICAL) DETAIL "Y"
SPARE ION EXCHANGER STORAGE AREA
PURIFICATION ION EXCHANGERS
FUEL PIT ION EXCHANGER
SUMP (BELOW)
GUTTER DRAIN
CRANE RAIL ℄
AUX. HOOK TRAVEL LIMITS
MAIN SISTER HOOK TRAVEL LIMITS
PLAN
SCALE: 1/8"=1'-0"
DETAIL "Y"
SCALE 1"=1'
ION EXCHANGER
ALL MATERIAL 304 STAINLESS STEEL UNLESS OTHERWISE NOTED
REMOVABLE SLABS
REMOVABLE CONC. SLABS MIN. SHIELDING THICKNESS 18"
REMOVABLE PLATFORM
WATER LEVEL EL. 1029'-0"
RESIN LEVEL
EL. 1022'-6"
EL. 1021'-0"
SECTION "B-B"
SCALE: 1/8"=1'-0"
FUEL TRANSFER PIT
REMOVABLE PLATFORM MAY BE PROVIDED IN FUTURE FOR USE WHEN TRANSFERRING SPENT ION EXCHANGERS HANG FROM COVERS. PROVIDE IN FIELD AS REQUIRED
REMOVABLE U-COVERS
EL. 1039'-6"
RAMP EL. 1034'-6"
PIPE TUNNEL
FOR EXT. HOOK SEE DETAIL "X"
EL. 1021'-0"
SECTION "A-A"
SCALE: 1/8"=1'-0"
PEAR SHAPED LINK AMERICAN CHAIN & CABLE CO OR EQUAL
1" C.S. PIPE SCH. 160.
EXTENSION HOOK DETAIL "X"
NO SCALE
HOOK FURNISHED BY JOB FORCES
PRIMARY AUX. BUILDING
ION EXCHANGE STORAGE PIT
ELEV.
FUEL TRANSFER PIT
℄ VAPOR CONTAINER
KEY PLAN
SCALE: 1/64"=1'-0"
REFERENCE DRAWINGS
1. - PLAN & DETS. SHEET #2 PRIMARY AUXILIARY BUILDING
2. - PURIFICATION PIPING SHEET #1
NOT FOR CONSTRUCTION
SUPERSEDES DWG. 9699-FM-35A
7 REV. PER GENERAL UPDATE
6 REDRAWN FOR CLARITY
ORIGINAL DRAWING SEALED AND SIGNED BY Walter C. Woodman
ARRGT-ION EXCHANGER STORAGE PIT
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-FM-35B
CHARGE J.O. No. 15386

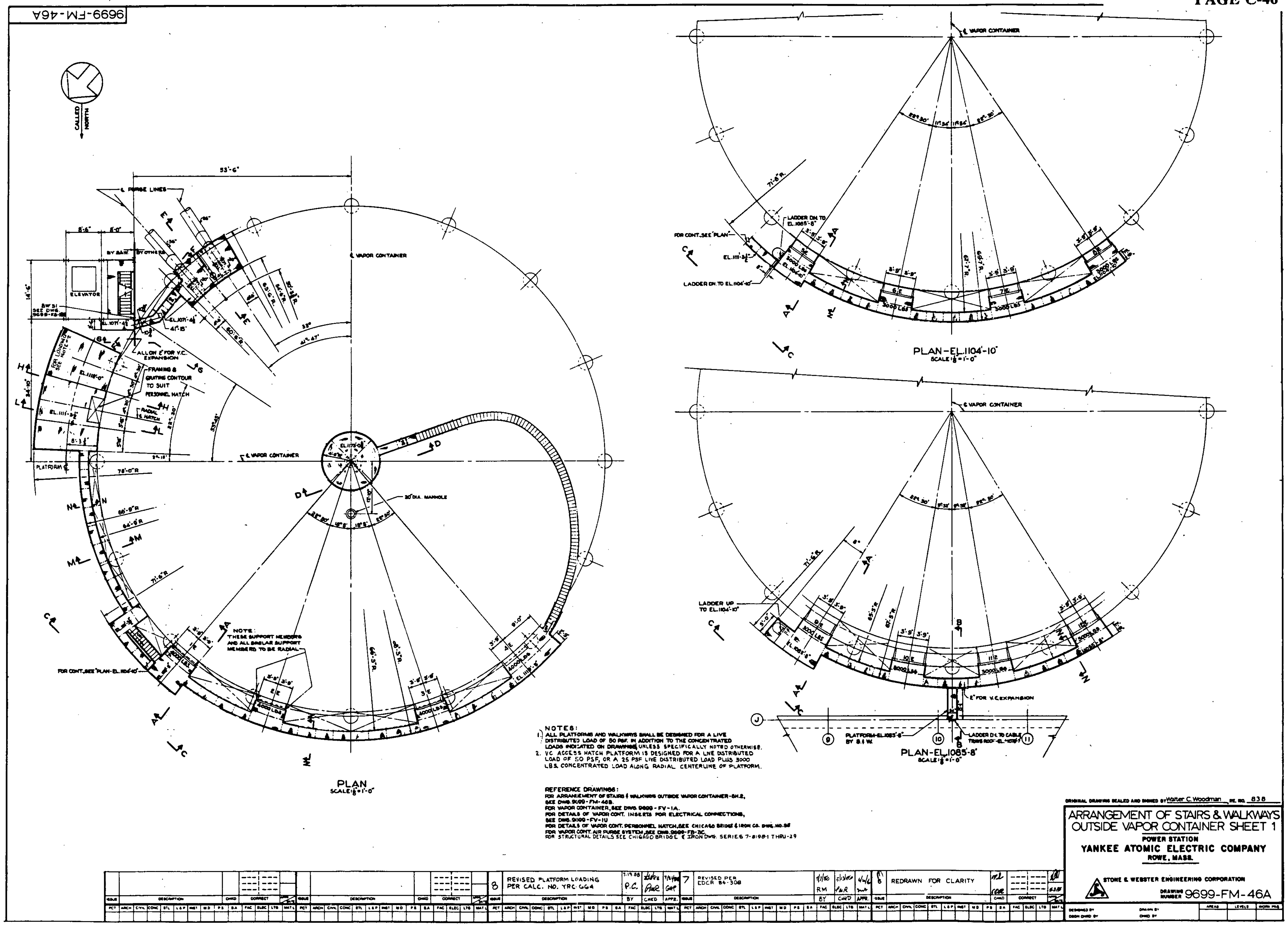

9699-FM-41A
TANK FARM
CLASS 1 LIQUIDS
TK-32
ACTIVITY
DILUTION &
DECAY TANK
75000
GALLONS
TK-31
WASTE
HOLDUP
TANK
75000
GALLONS
WASTE LIQUID
TRANSFER PUMPS
P28-2
P28-1
PRIMARY DRAIN
COLLECTING TANK
TK-30
7500 GALLONS
COLLECTING
TANK
TRANSFER
PUMPS
TANK PIT
EDUCTOR
RESIN
STRAINER
WASTE GAS
HEADER
H.C. LOOP
DECONTAMINATION
PUMP
FROM PURIFICATION ION EXCHANGER OUTLET HEADER
TO PURIFICATION ION EXCHANGER INLET HEADER
TO PRIMARY DRAIN
SYSTEM
FROM PRIMARY VENT
HEADER
STEAM & COND.
LEGEND
GATE VALVE
GLOBE VALVE
CHECK VALVE
DIAPHRAGM VALVE
GLOBE VALVE-BELLOWS SEAL
VALVE LOCKED OPEN
VALVE NORMALLY CLOSED
DRAIN
VENT
HAND OPERATED VALVE
PRESSURE OR LEVEL CONTROL VALVE
SOLENOID OPERATED VALVE
SAFETY VALVE
SAFETY VALVE-BELLOWS SEAL
FLOW METER
GAGE GLASS
STEAM TRAP
STRAINER
LEVEL INDICATOR
LEVEL INDICATING CONTROLLER
PRESSURE INDICATOR
LEVEL ALARM
TEMPERATURE INDICATOR
BOARD MOUNTED INSTRUMENT
RADIATION MONITOR
REDUCER
CAPPED CONNECTION
INSULATED PIPE FLANGED
DIRECTION OF FLOW
FLAME ARRESTER
FLOOR DRAIN
VALVE OPERATING NUMBERS
CHICAGO CONN
DAMPER
LEVEL TRANSMITTER AMPS TO PRESSURE CONVERTER
REVISED PER CORRECTIVE UPDATE
REVISED PER CORR. UPDATE 89-093
REVISED PER CORR. UPDATE 88-320
REVISED PER PLT. ALT. 86-027
REVISED PER PDCR 87-009
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
FLOW DIAGRAM
RADIOACTIVE WASTE DISPOSAL

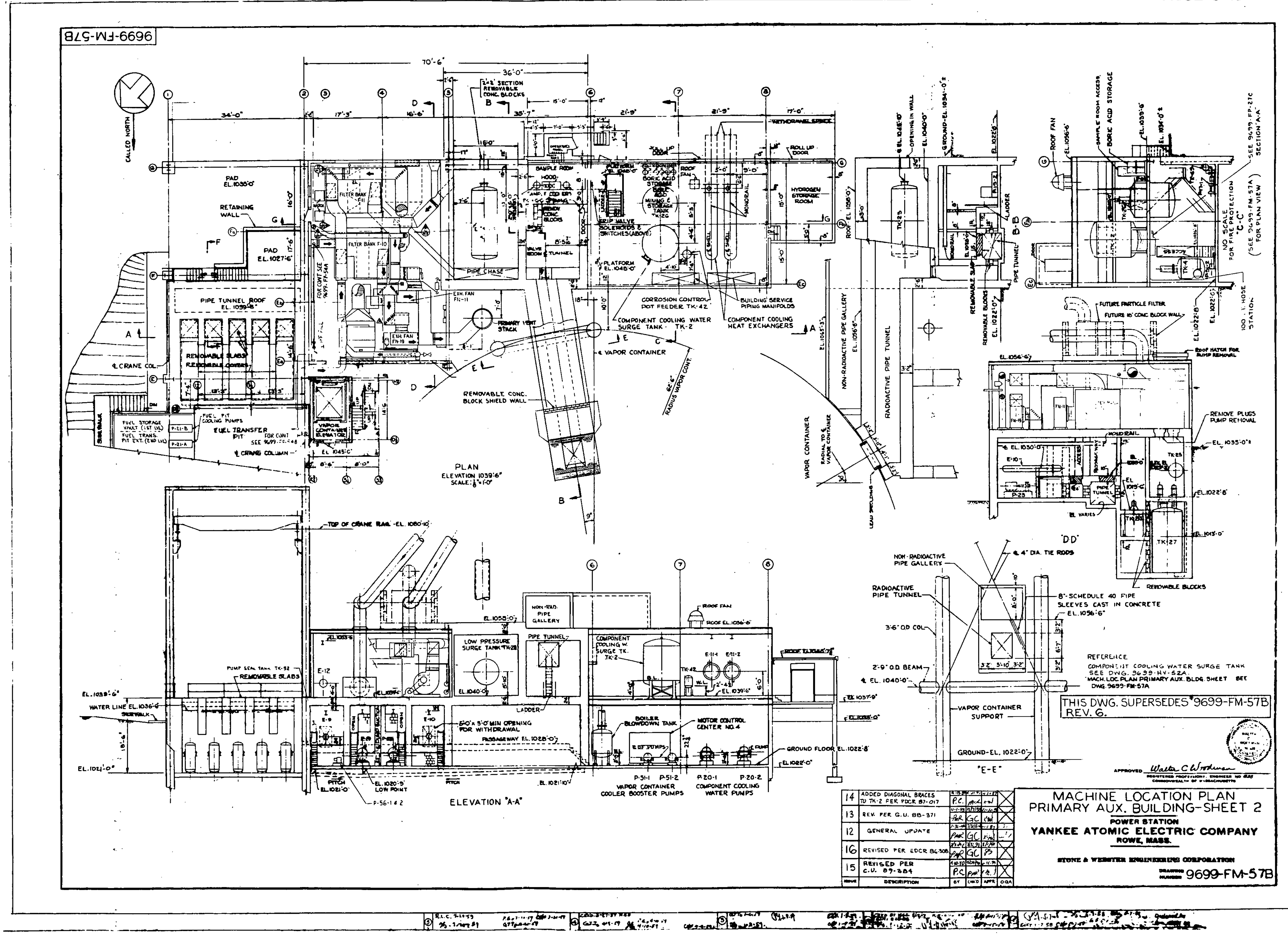

9699-FM-41B
TANK FARM
MAKE-UP & CLASS 2 LIQUIDS
MONITORED
WASTE TKS.
TK 29-1
TK 29-2
1370 GALS.
1370 GALS.
TEST TK.
TK 34-1
TEST TK.
TK 34-2
8040 GALS.
8040 GALS.
MONITORED WASTE
TANKS TRANSF. PUMPS
P26-1
P26-2
TEST TANKS
EFF. PUMP
P-31
ST1-2
ST1-1
PRIMARY WATER
STORAGE TANK
TK-39
135000 GALS.
HEAT EXCHANGER
PRIMARY BLDG. SUMP TANK
TK-24
484 GALS.
GRAVITY DRAIN TANK
TK-27
4700 GALS.
SERVICE WATER
LIQUID FROM RADIOACTIVE LAB
SUMP PUMP
PRIMARY AUXILIARY BLDG. FLOOR DRAINS
PRIMARY AUXILIARY BUILDING
PIPE TRENCH DRAIN
TO SEAL PUMPS
PRIMARY WATER TO
LP SURGE MAKEUP PUMPS
DEMINERALIZED WATER TO
PRIMARY WATER STORAGE TANK
NOTE: FOR DRAWING LEGEND
SEE DRAWING NO. 9699-FM-41A
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
FLOW DIAGRAM
RADIOACTIVE WASTE DISPOSAL


9699-FM-41C
LIQUID WASTE DISPOSAL
EV-1 EVAPORATOR
STEAM SEPARATOR
REBOILER HEAT EXCHANGER
E-17 OVERHEAD CONDENSER
E-18 EVAPORATOR CONDENSER
DISTILLATE ACCUMULATOR TK-33
EVAPORATOR FEED AND VENT SELECTOR
REBOILER PUMP P-55
P-29 BOTTOM PUMP
P-30-1
P-30-2
DISTILLATE PUMPS
FEED AND DISTILLATE HEAT EXCHANGER E-16
FILTER
ION EXCHANGER
M-4 DRUM ROLLER
ASH DEWATERING SUMP TK-43
COOLING WATER RETURN SEE DWG. 9699-FM-9A
COOLING WATER SUPPLY SEE DWG. 9699-FM-9A
SERVICE WATER SEE 9699-FM-21B
4" COMPACTOR BLDG FLOOR DRAINS
4" WAREHOUSE FLOOR DRAINS
WASTE DISPOSAL BLDG FLOOR DRAIN
TO TEMPORARY STORAGE
NOTE: FOR DRAWING LEGEND SEE DRAWING NO. 9699-FM-41A
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
FLOW DIAGRAM
RADIOACTIVE WASTE DISPOSAL
9699-FM-41C

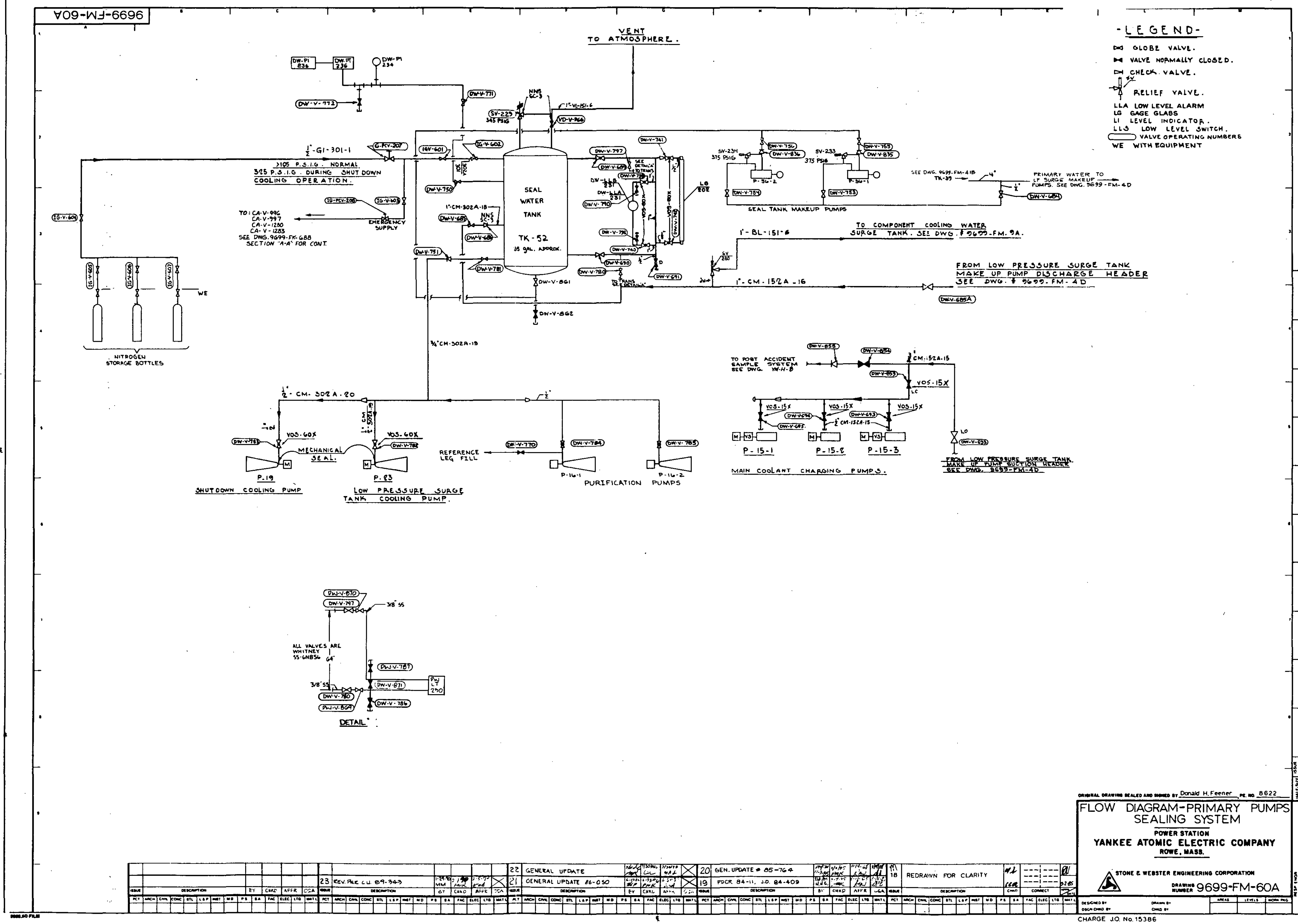

9699-FM-41D
FROM EVAPORATOR OVERHEAD CONDENSER SEE DWG. 9699-FM-41C (ZONE 18D)
TO WASTE HOLD UP TANKS DWG. 9699-FM-41A (ZONE 16A)
TO GAS RM VENTILATION SYSTEM
ZERO GAS
SPAN GAS
TO VENTILATION EXHAUST SYS. SEE DWG. 9699-FM-26D
FLAME ARRESTOR
LOOP SEAL
COMPONENT COOLING FILL
SEE DWG. 9699-FM-41C (ZONE B-17)
COMPRESSOR SUCTION COOLER E-19
ANALYZER SENSOR
TO MD PANALARM WINDOW NO. 24
SAMPLE CHAMBER
WASTE GAS SURGE DRUM TK-36 4160 CU. FEET
INERT GAS PURGE
WASTE GAS COMPRESSOR
C 3-1
WASTE GAS COMPRESSOR
C 3-2
TIMER
COOLING WATER
COMP. DISCH. COOLER E 20
GAS DECAY DRUMS
INSERT GAS PURGE
COMPR K.O. DRUM TK-35
STRAINER
NITROGEN TEST STATION
DECAY GAS FILTER
F-2
FILTER
(NOT IN USE) CONSTANT TEMP. BOX
#30 STEAM SEE 9699-FB-7B
CONDENSATE SEE DWG. 9699-FB-7B
INTERLOCK SWITCH
FM-19 CONTROLLER
TO EXHAUST VENTILATION SYS. SEE 9699-FM-26D
TO P.D.C. TK-30 SEE DWG. FM41A (ZONE K-1)
NOTES
LPS 301 ALARMS ON LOW HEADER PRESSURE 2" H2O
HPS 301 STARTS EITHER COMPRESSOR IF ON AUTO 13" H2O
LPS 302 STOPS EITHER COMPRESSOR IF IN AUTO 8" H2O
HPS 302 STARTS SECOND COMPRESSOR (IF ON AUTO) ON HIGH HEADER PRESSURE 18" H2O
FOR DRAWING LEGEND SEE DRAWING NO. 9699-FM-41A
2 REVISED PER C.U. 91-351 AND PDCR 87-009
1 REVISED PER PDCR 87-009
0 THIS DWG. SUPERSEDES 9699-RM-41F REV. 41 IN PART
REV. DESCRIPTION BY CHKD APPR'D
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
FLOW DIAGRAM
RADIOACTIVE WASTE DISPOSAL
SCALE
APPROVED
DATE
DIV.
DR.
CH.
DWG NO.
9699-FM-41D

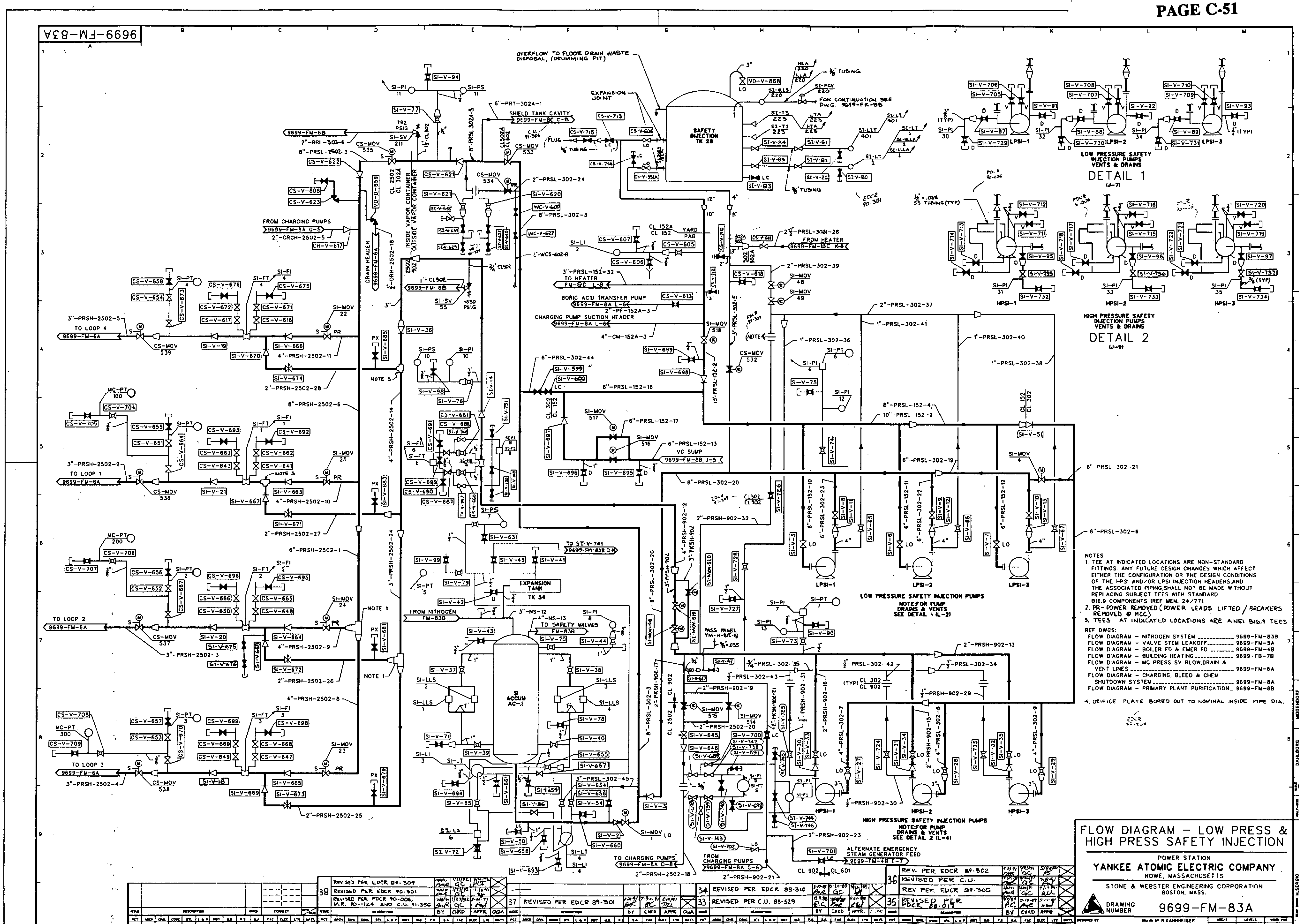

9699-FM-57A
CALLED NORTH
GRAVITY DRAIN TANK TK-27
VAPOR CONTAINER DRAIN TANK TK-25
PURIFICATION, DRAIN & COOLING PUMPS
CHARGING PUMPS
LOW PRESS. SURGE TK. MAKE-UP PUMPS
REMOVABLE PLUG HATCH
SAMPLE SUMP
COLLECTING TANK TRANSFER PUMP P-27-2
SUMP
PRIMARY DRAIN COLLECTING TANK TK-30
MOTOR CONT. CENTER
EMERGENCY FEEDWATER PUMPS
BOILER BLOWDOWN TANK
COMPONENT COOLING WATER PUMPS
VAPOR CONTAINER COOLER BOOSTER PUMPS
PIPE TUNNEL
ION EXCHANGER
ION EXCHANGE PIT
STACK
LOW PRESS. SURGE TANK COOLER
LOW PRESS. SURGE TANK COOLING PUMP
SHUTDOWN COOLING PUMP
VAPOR CONTAINER SUPPORT
VAPOR CONTAINER ELEVATOR
FUEL STORAGE VAULT
FUEL TRANSFER PIT
FUEL TRANSFER CHUTE
VAPOR CONTAINER
RAILROAD TRACK
CRANE RAIL
CRANE COLUMN
SERVICE BUILDING
PAD
SECTION "F-F"
SCALE: 1/4" = 1'-0"
"G-G"
PLAN
ELEVATION 1022'-8"
SCALE: 1/8" = 1'-0"
NOTES:
1. THICKNESS OF SHIELDING WALLS NOT DIMENSIONED TO BE MINIMUM OF 12"
2. CONCRETE THICKNESSES SHOWN ARE MINIMUM SHIELDING REQUIREMENTS
3. CONCRETE INTERNAL WALLS TO BE CONCRETE BLOCKS EXCEPT WHERE REQUIRED FOR STRUCTURAL SUPPORT
4. PIPING GALLERY WALLS BELOW GRADE 1026'-0" WHICH CONNECT GALLERY AND RADIOACTIVE CUBICLES TO BE REMOVABLE CONCRETE BLOCKS WHERE REQUIRED TO FACILITATE PIPING INSTALLATION
5. WIRE MESH PARTITION WITH LOCKABLE DOOR
REFERENCE:
MACHINE LOCATION PLAN PRIMARY AUX. BUILDING SH. 2 - SEE DWG. 9699-FM-57B
FUEL TRANSFER PIT - SEE DWG. 9699-FM-21A
SUPERSEDES DWG. 9699-FM-18E
8 REVISED PER EDCR 79-16
10 REV. PER G.U. 88-370
9 PLT ALT 86-006
DESCRIPTION
MACHINE LOCATION PLAN
PRIMARY AUX. BUILDING-SHEET 1
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
9699-FM-57A

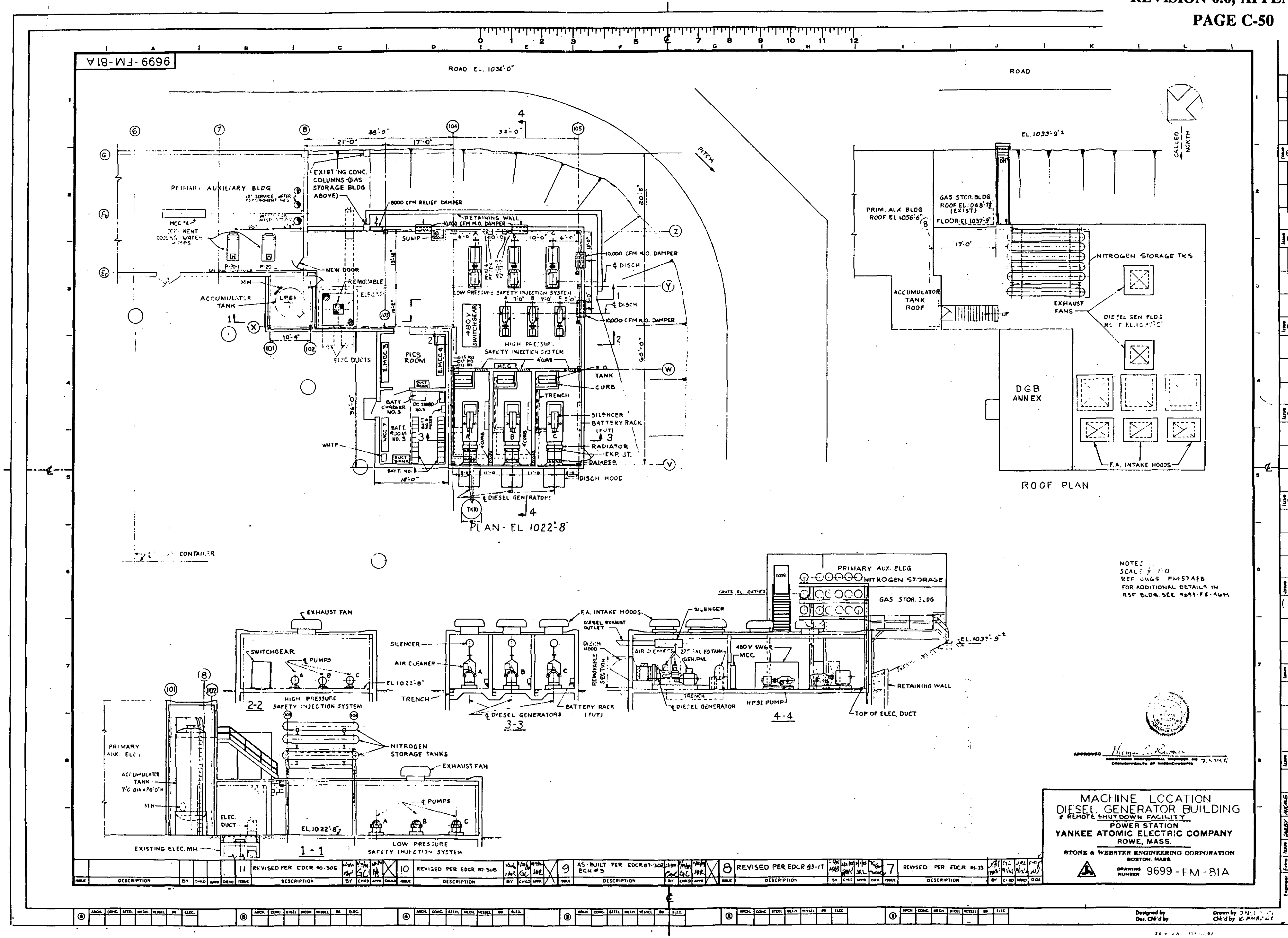

9699-FM-46A
CALLED NORTH
PLAN
SCALE: 1/8"=1'-0"
PLAN-EL.1104'-10"
SCALE: 1/8"=1'-0"
PLAN-EL.1085'-8"
SCALE: 1/8"=1'-0"
NOTES:
1. ALL PLATFORMS AND WALKWAYS SHALL BE DESIGNED FOR A LIVE DISTRIBUTED LOAD OF 50 PSF. IN ADDITION TO THE CONCENTRATED LOADS INDICATED ON DRAWINGS UNLESS SPECIFICALLY NOTED OTHERWISE.
2. VC ACCESS HATCH PLATFORM IS DESIGNED FOR A LIVE DISTRIBUTED LOAD OF 50 PSF, OR A 25 PSF LIVE DISTRIBUTED LOAD PLUS 3000 LBS. CONCENTRATED LOAD ALONG RADIAL CENTERLINE OF PLATFORM.
REFERENCE DRAWINGS:
FOR ARRANGEMENT OF STAIRS & WALKWAYS OUTSIDE VAPOR CONTAINER-SH.2, SEE DWG.9699-FM-46B.
FOR VAPOR CONTAINER, SEE DWG. 9699-FV-1A.
FOR DETAILS OF VAPOR CONT. INSERTS FOR ELECTRICAL CONNECTIONS, SEE DWG.9699-FV-1U
FOR DETAILS OF VAPOR CONT. PERSONNEL HATCH, SEE CHICAGO BRIDGE & IRON CO. DWG.NO.35
FOR VAPOR CONT. AIR PURGE SYSTEM, SEE DWG.9699-FB-3C.
FOR STRUCTURAL DETAILS SEE CHICAGO BRIDGE & IRON DWG. SERIES 7-8198-1 THRU-29
8 REVISED PLATFORM LOADING PER CALC. NO. YRC-664 7-19-88 P.C.
7 REVISED PER EDCR 84-308 RM
6 REDRAWN FOR CLARITY
ORIGINAL DRAWING SEALED AND SIGNED BY Walter C. Woodman PE. NO. 838
ARRANGEMENT OF STAIRS & WALKWAYS OUTSIDE VAPOR CONTAINER SHEET 1
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-FM-46A

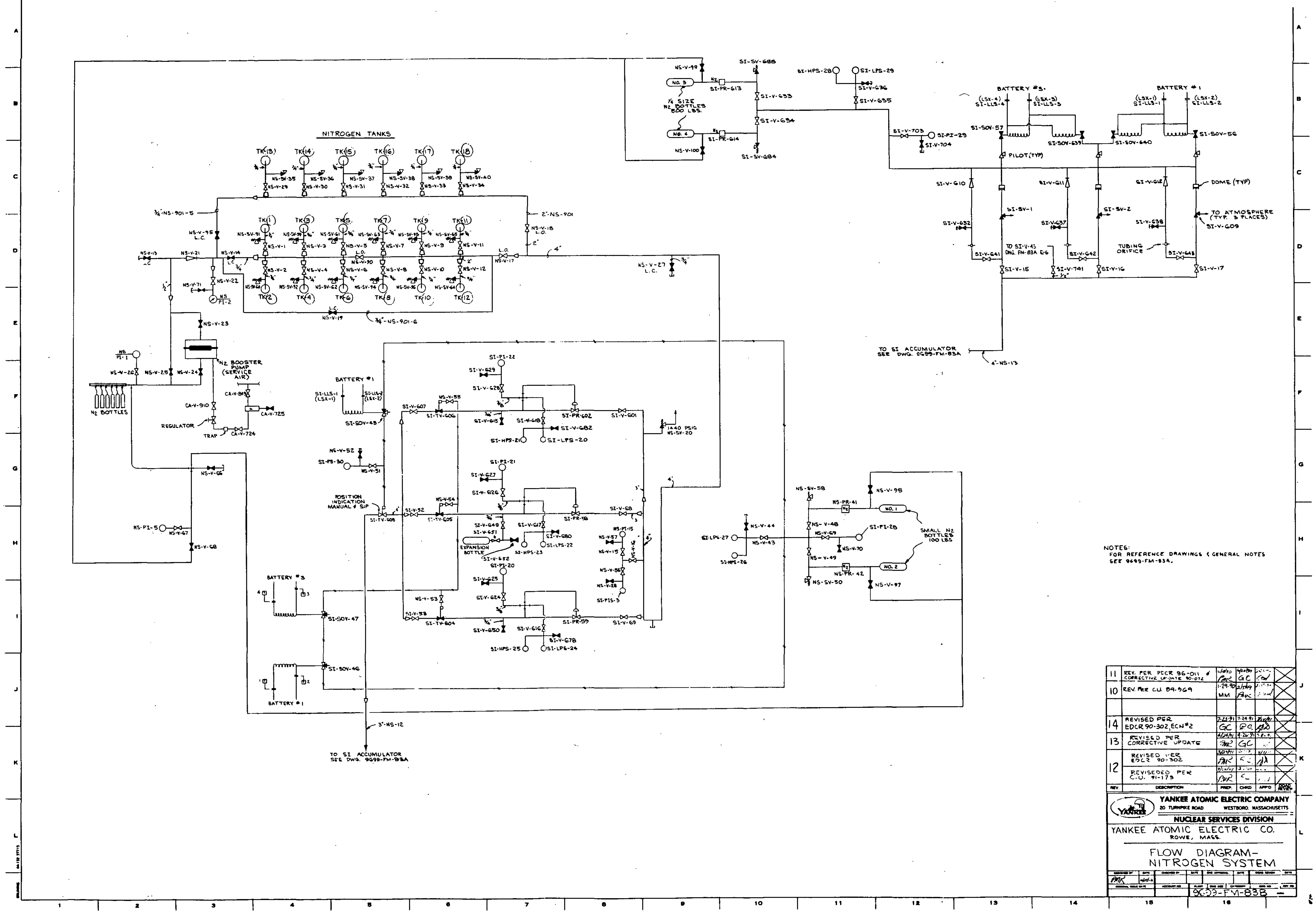

9699-FM-57B
PLAN
ELEVATION 1039'-6"
ELEVATION "A-A"
"DD"
"E-E"
"C-C"
THIS DWG. SUPERSEDES 9699-FM-57B REV. 6.
MACHINE LOCATION PLAN
PRIMARY AUX. BUILDING-SHEET 2
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
9699-FM-57B

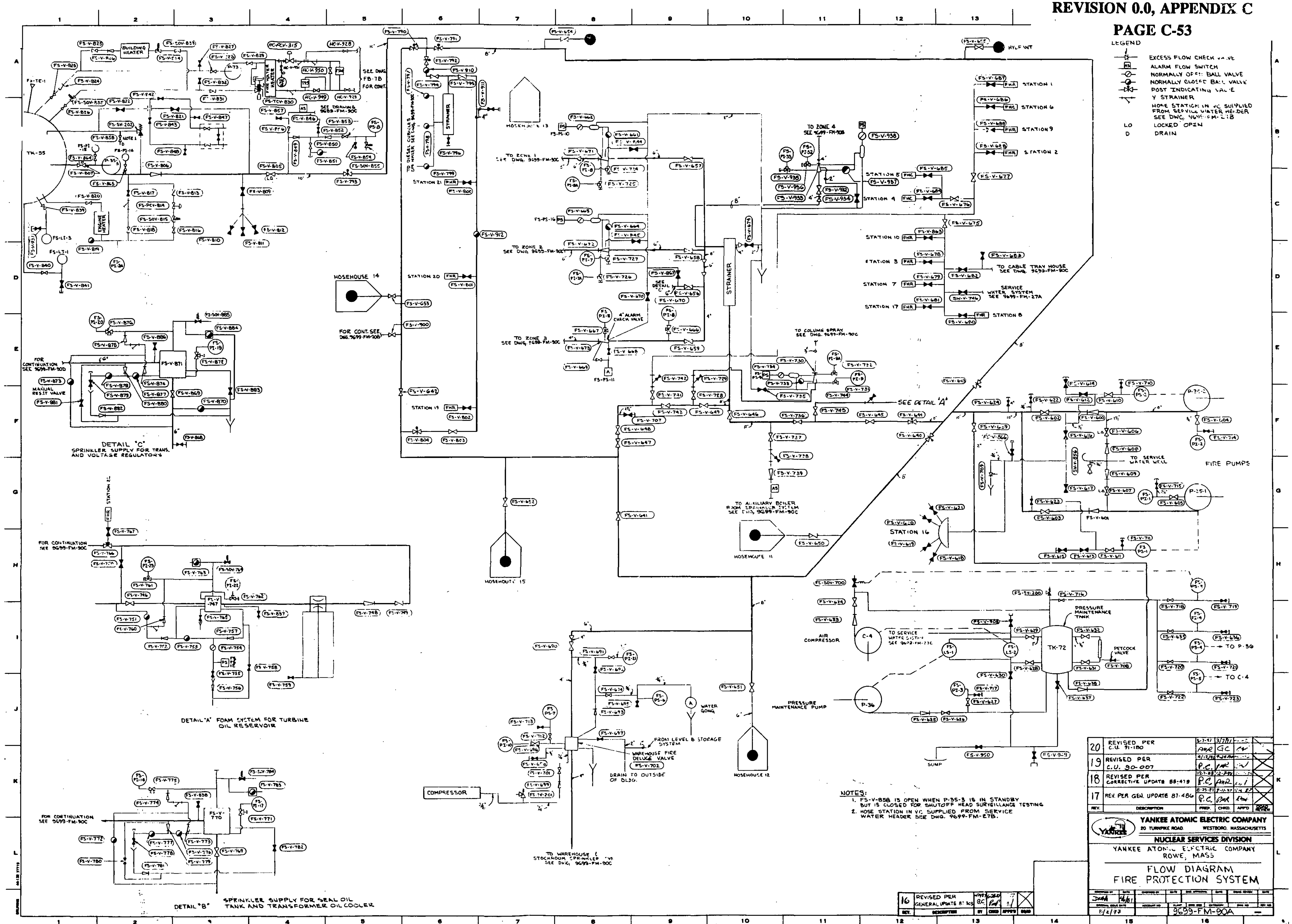
9699-FM-60A

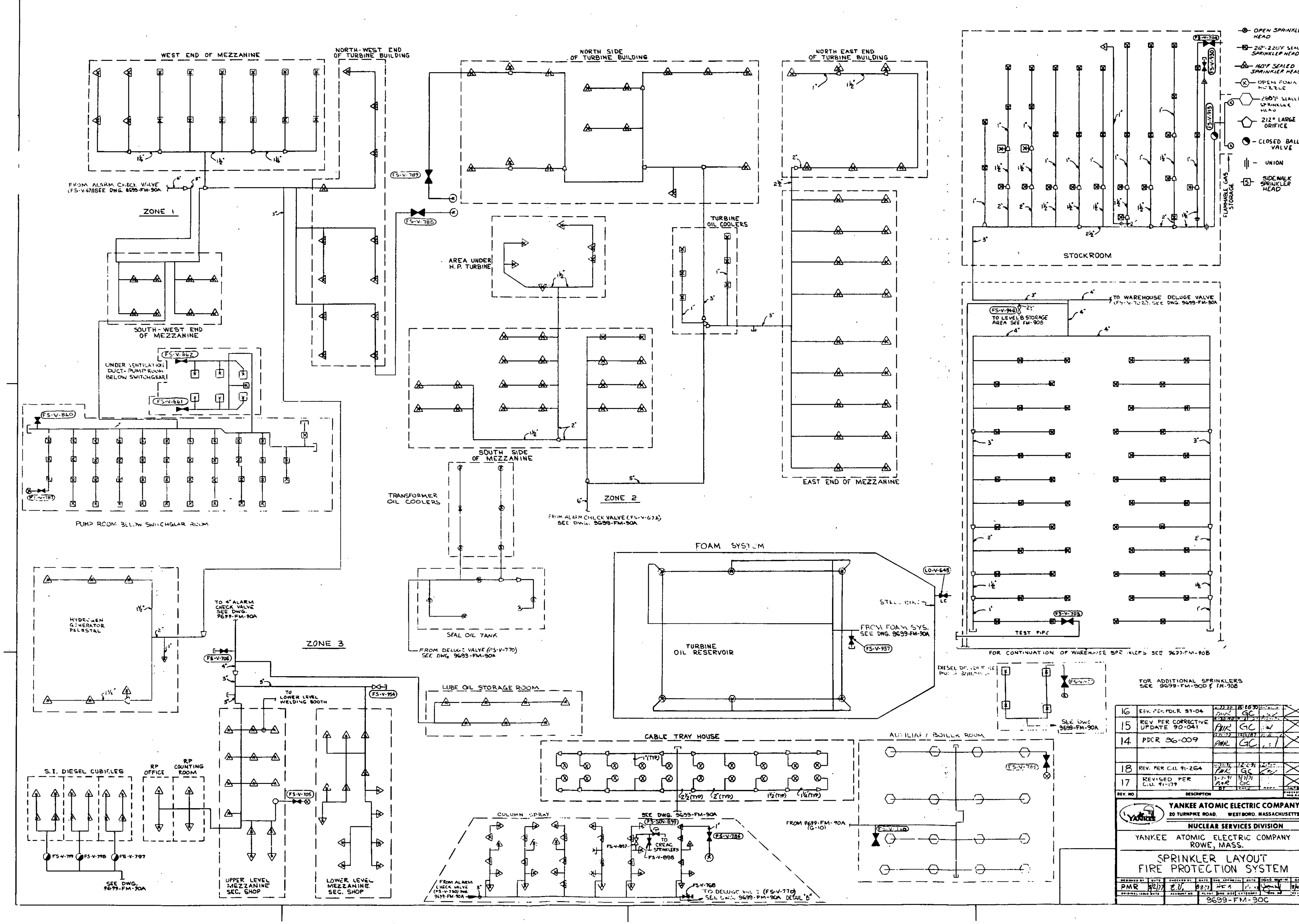

9699-FM-60A
VENT TO ATMOSPHERE.
-LEGEND-
GLOBE VALVE.
VALVE NORMALLY CLOSED.
CHECK VALVE.
RELIEF VALVE.
LLA LOW LEVEL ALARM
LG GAGE GLASS
LI LEVEL INDICATOR.
LLS LOW LEVEL SWITCH.
VALVE OPERATING NUMBERS
WE WITH EQUIPMENT
SEAL WATER TANK
TK-52
>105 P.S.I.G. NORMAL
325 P.S.I.G. DURING SHUT DOWN COOLING OPERATION.
EMERGENCY SUPPLY
NITROGEN STORAGE BOTTLES
SEAL TANK MAKEUP PUMPS
PRIMARY WATER TO LP SURGE MAKEUP PUMPS. SEE DWG. 9699-FM-4D
TO COMPONENT COOLING WATER SURGE TANK. SEE DWG. # 9699-FM-9A.
FROM LOW PRESSURE SURGE TANK MAKE UP PUMP DISCHARGE HEADER SEE DWG. # 9699-FM-4D
TO POST ACCIDENT SAMPLE SYSTEM SEE DWG.
MECHANICAL SEAL.
P-19
SHUTDOWN COOLING PUMP
P-23
LOW PRESSURE SURGE TANK COOLING PUMP.
REFERENCE LEG FILL
P-16-1
P-16-2
PURIFICATION PUMPS
P-15-1
P-15-2
P-15-3
MAIN COOLANT CHARGING PUMPS.
FROM LOW PRESSURE SURGE TANK MAKE UP PUMP SUCTION HEADER SEE DWG. 9699-FM-4D
ALL VALVES ARE WHITNEY SS-6NBS4
DETAIL
23 REV. PAE CU 89-343
22 GENERAL UPDATE
21 GENERAL UPDATE 86-050
20 GEN. UPDATE # 85-764
19 PDCR 84-11, J.O. 84-409
18 REDRAWN FOR CLARITY
ORIGINAL DRAWING SEALED AND SIGNED BY Donald H. Feener PE NO. 8622
FLOW DIAGRAM-PRIMARY PUMPS SEALING SYSTEM
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-FM-60A
CHARGE J.O. No. 15386

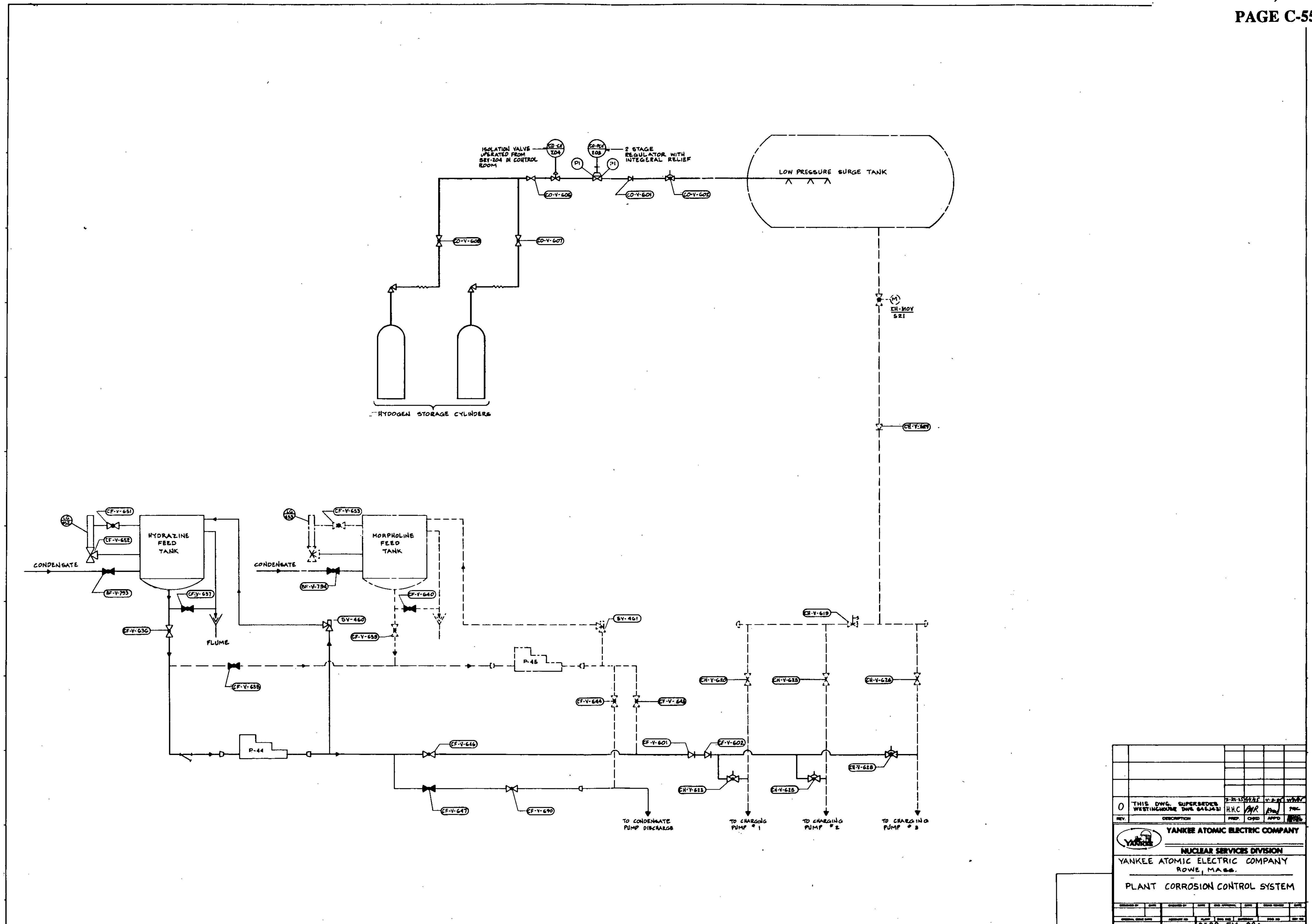

9699-FM-83A
SAFETY INJECTION TK 28
SI ACCUM AC-1
EXPANSION TANK TK 54
LOW PRESSURE SAFETY INJECTION PUMPS VENTS & DRAINS
DETAIL 1
HIGH PRESSURE SAFETY INJECTION PUMPS VENTS & DRAINS
DETAIL 2
LOW PRESSURE SAFETY INJECTION PUMPS
NOTE: FOR PUMP DRAINS & VENTS SEE DETAIL 1 (L-2)
HIGH PRESSURE SAFETY INJECTION PUMPS
NOTE: FOR PUMP DRAINS & VENTS SEE DETAIL 2 (L-4)
NOTES
1. TEE AT INDICATED LOCATIONS ARE NON-STANDARD FITTINGS. ANY FUTURE DESIGN CHANGES WHICH AFFECT EITHER THE CONFIGURATION OR THE DESIGN CONDITIONS OF THE HPSI AND/OR LPSI INJECTION HEADERS, AND THE ASSOCIATED PIPING, SHALL NOT BE MADE WITHOUT REPLACING SUBJECT TEES WITH STANDARD B16.9 COMPONENTS (REF MEM. 24/77).
2. PR - POWER REMOVED (POWER LEADS LIFTED / BREAKERS REMOVED @ MCC)
3. TEES AT INDICATED LOCATIONS ARE ANSI B16.9 TEES
REF DWGS:
FLOW DIAGRAM - NITROGEN SYSTEM 9699-FM-83B
FLOW DIAGRAM - VALVE STEM LEAKOFF 9699-FM-5A
FLOW DIAGRAM - BOILER FD & EMER FD 9699-FM-4B
FLOW DIAGRAM - BUILDING HEATING 9699-FB-7B
FLOW DIAGRAM - MC PRESS SV BLOW, DRAIN & VENT LINES 9699-FM-6A
FLOW DIAGRAM - CHARGING, BLEED & CHEM SHUTDOWN SYSTEM 9699-FM-8A
FLOW DIAGRAM - PRIMARY PLANT PURIFICATION ... 9699-FM-8B
4. ORIFICE PLATE BORED OUT TO NOMINAL INSIDE PIPE DIA.
FLOW DIAGRAM - LOW PRESS & HIGH PRESS SAFETY INJECTION
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASSACHUSETTS
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
DRAWING NUMBER 9699-FM-83A

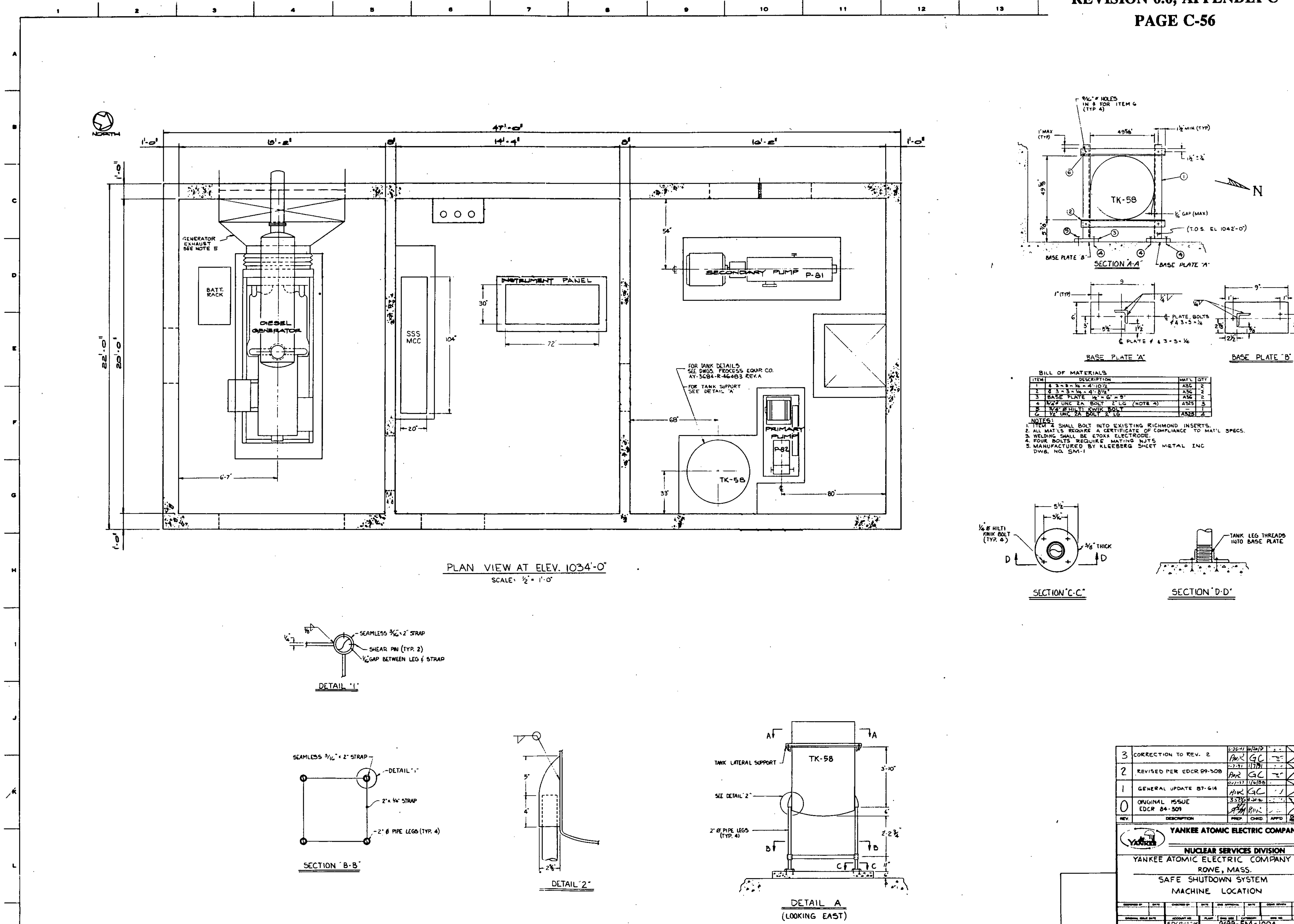

PLAN - EL 1022'-8"
ROOF PLAN
2-2
3-3
4-4
1-1
MACHINE LOCATION
DIESEL GENERATOR BUILDING
& REMOTE SHUTDOWN FACILITY
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
BOSTON, MASS.
9699-FM-81A


NITROGEN TANKS
N_2 BOTTLES
N_2 BOOSTER PUMP (SERVICE AIR)
REGULATOR
TRAP
BATTERY #1
BATTERY #3
POSITION INDICATION MANUAL & SIP
EXPANSION BOTTLE
3"-NS-12
TO SI ACCUMULATOR SEE DWG. 9699-FM-83A
TO SI ACCUMULATOR SEE DWG. 9699-FM-83A
4"-NS-13
SMALL N_2 BOTTLES 100 LBS
BATTERY #1
BATTERY #3
PILOT (TYP)
DOME (TYP)
TO ATMOSPHERE (TYP. 3 PLACES)
TUBING ORIFICE
1440 PSIG NS-SV-20
NOTES:
FOR REFERENCE DRAWINGS & GENERAL NOTES SEE 9699-FM-83A.
11 REV. PER PDCR 96-011 & CORRECTIVE UPDATE 90-072
10 REV. PER C.U. 89-369
14 REVISED PER EDCR 90-302, ECN #2
13 REVISED PER CORRECTIVE UPDATE
12 REVISED PER EDCR 90-302
REVISED PER C.U. 91-173
REV DESCRIPTION
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC CO.
ROWE, MASS.
FLOW DIAGRAM-
NITROGEN SYSTEM
9699-FM-83B


LEGEND
EXCESS FLOW CHECK VALVE
ALARM FLOW SWITCH
NORMALLY OPEN BALL VALVE
NORMALLY CLOSED BALL VALVE
POST INDICATING VALVE
Y STRAINER
LO LOCKED OPEN
D DRAIN
DETAIL "C" SPRINKLER SUPPLY FOR TRANS. AND VOLTAGE REGULATORS
DETAIL "A" FOAM SYSTEM FOR TURBINE OIL RESERVOIR
DETAIL "B" SPRINKLER SUPPLY FOR SEAL OIL TANK AND TRANSFORMER OIL COOLER
FIRE PUMPS
PRESSURE MAINTENANCE TANK
TK-72
AIR COMPRESSOR
PRESSURE MAINTENANCE PUMP
STRAINER
COMPRESSOR
NOTES:
1. FS-V-858 IS OPEN WHEN P-35-3 IS IN STANDBY BUT IS CLOSED FOR SHUTOFF HEAD SURVEILLANCE TESTING
2. HOSE STATION IN VC SUPPLIED FROM SERVICE WATER HEADER SEE DWG. 9699-FM-27B.
20 REVISED PER C.U. 91-180
19 REVISED PER C.U. 90-007
18 REVISED PER CORRECTIVE UPDATE 88-479
17 REV. PER GEN. UPDATE 87-486
16 REVISED PER GENERAL UPDATE
YANKEE ATOMIC ELECTRIC COMPANY
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS
FLOW DIAGRAM
FIRE PROTECTION SYSTEM
9699-FM-90A


WEST END OF MEZZANINE
NORTH-WEST END OF TURBINE BUILDING
NORTH SIDE OF TURBINE BUILDING
NORTH EAST END OF TURBINE BUILDING
STOCKROOM
ZONE 1
ZONE 2
ZONE 3
SOUTH-WEST END OF MEZZANINE
AREA UNDER H.P. TURBINE
TURBINE OIL COOLERS
EAST END OF MEZZANINE
SOUTH SIDE OF MEZZANINE
TRANSFORMER OIL COOLERS
SEAL OIL TANK
PUMP ROOM BELOW SWITCHGEAR ROOM
HYDROGEN GENERATOR PEDESTAL
FOAM SYSTEM
TURBINE OIL RESERVOIR
LUBE OIL STORAGE ROOM
CABLE TRAY HOUSE
AUXILIARY BOILER ROOM
COLUMN SPRAY
S.I. DIESEL CUBICLES
RP OFFICE
RP COUNTING ROOM
UPPER LEVEL MEZZANINE SEC. SHOP
LOWER LEVEL MEZZANINE SEC. SHOP
TEST PIPE
FOR CONTINUATION OF WAREHOUSE SPRINKLERS SEE 9699-FM-90B
FOR ADDITIONAL SPRINKLERS SEE 9699-FM-90D & FM-90B
OPEN SPRINKLER HEAD
212°-220°F SEALED SPRINKLER HEAD
160°F SEALED SPRINKLER HEAD
OPEN FOAM NOZZLE
212° LARGE ORIFICE
CLOSED BALL VALVE
UNION
SIDEWALK SPRINKLER HEAD
16 REV. PER PDCR 93-04
15 REV PER CORRECTIVE UPDATE 90-041
14 PDCR 86-009
18 REV. PER C.U. 91-264
17 REVISED PER C.U. 91-179
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD, WESTBORO, MASSACHUSETTS
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY ROWE, MASS.
SPRINKLER LAYOUT FIRE PROTECTION SYSTEM
9699-FM-90C


ISOLATION VALVE OPERATED FROM SBY-204 IN CONTROL ROOM
2 STAGE REGULATOR WITH INTEGERAL RELIEF
LOW PRESSURE SURGE TANK
HYDROGEN STORAGE CYLINDERS
HYDRAZINE FEED TANK
MORPHOLINE FEED TANK
CONDENSATE
CONDENSATE
FLUME
P-44
P-45
TO CONDENSATE PUMP DISCHARGE
TO CHARGING PUMP #1
TO CHARGING PUMP #2
TO CHARGING PUMP #3
THIS DWG. SUPERSEDES WESTINGHOUSE DWG 542J431
YANKEE ATOMIC ELECTRIC COMPANY
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
PLANT CORROSION CONTROL SYSTEM
9699-FM-92A

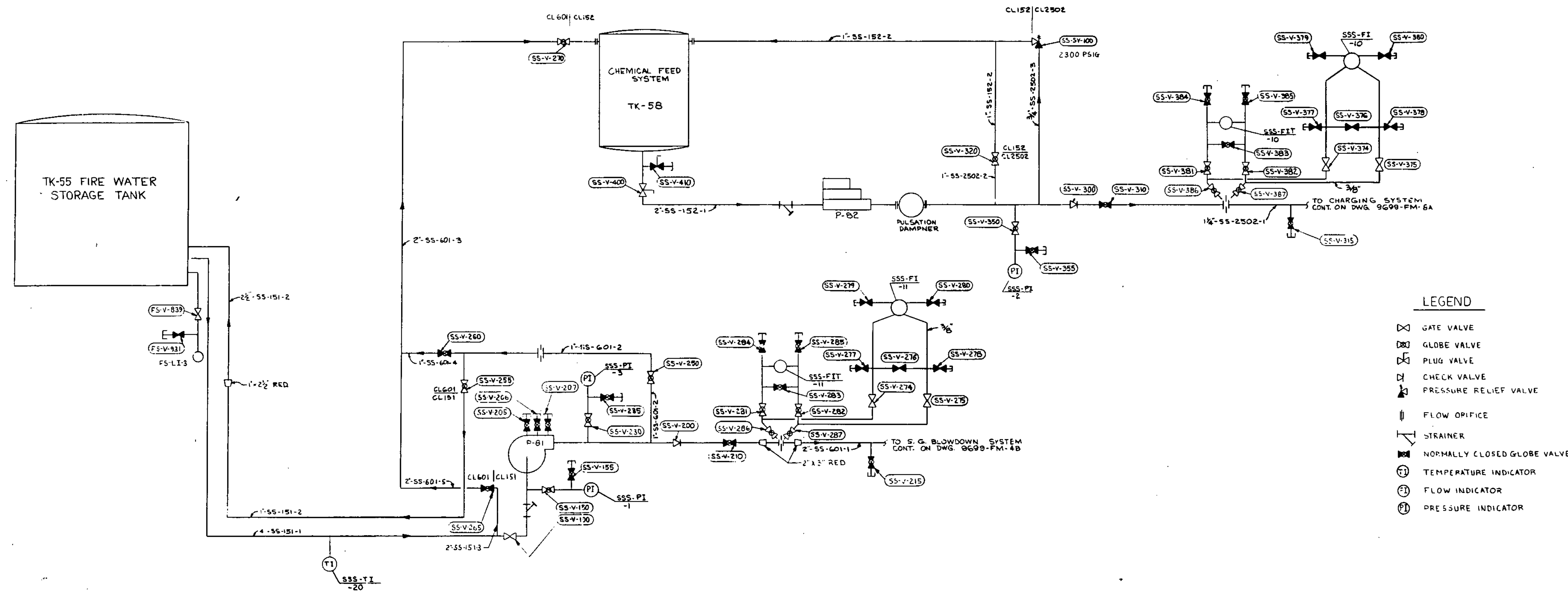


ITEM	DESCRIPTION	MAT'L	QTY
1	∠ 3 x 3 x 1/4 x 4'-10 1/2"	A36	2
2	∠ 3 x 3 x 1/4 x 4'-8 1/2"	A36	2
3	BASE PLATE 1/2" x 6" x 9"	A36	2
4	3/4" Ø UNC 2A BOLT 2" LG (NOTE 4)	A325	3
5	3/4" Ø HILTI KWIK BOLT	—	1
6	1/2" UNC 2A BOLT 2" LG	A325	4

NOTES:
1. ITEM 4 SHALL BOLT INTO EXISTING RICHMOND INSERTS.
2. ALL MAT'LS REQUIRE A CERTIFICATE OF COMPLIANCE TO MAT'L SPECS.
3. WELDING SHALL BE E70XX ELECTRODE.
4. FOUR BOLTS REQUIRE MATING NUTS
5. MANUFACTURED BY KLEEBERG SHEET METAL INC DWG. NO. SM-1

REV.	DESCRIPTION	PREP.	CHKD.	APP'D	QUAD REVIEW
3	CORRECTION TO REV. 2	6-26-91 PMR	6/26/91 GC		
2	REVISED PER EDCR 89-308	1-7-91 PMR	1/7/91 GC		
1	GENERAL UPDATE 87-614	12-17-87 PMR	1/6/88 GC		
0	ORIGINAL ISSUE EDCR 84-309				

YANKEE ATOMIC ELECTRIC COMPANY
NUCLEAR SERVICES DIVISION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
SAFE SHUTDOWN SYSTEM
MACHINE LOCATION
EDCR 84-309
9699-FM-100A



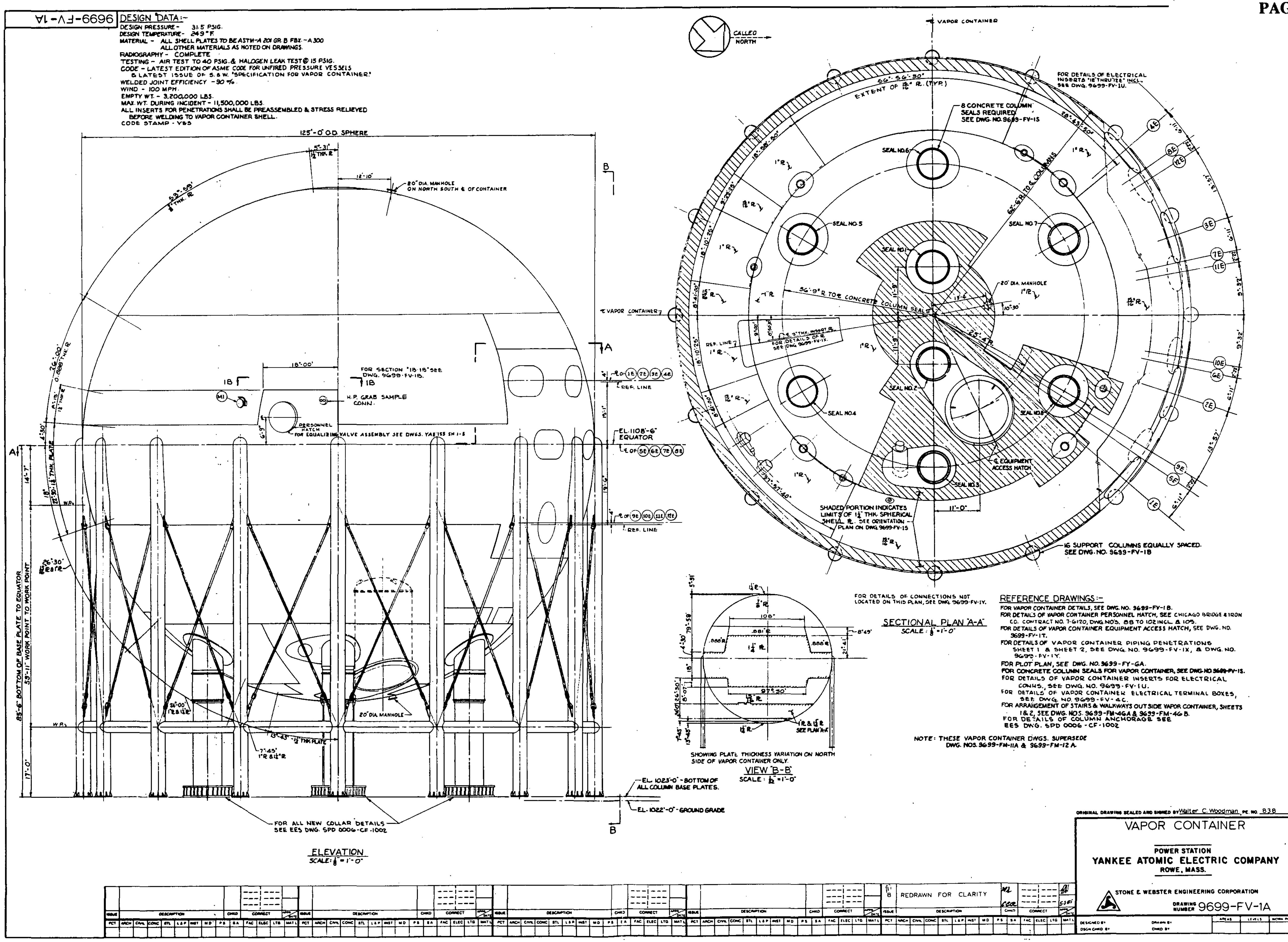
REV.	DESCRIPTION	PREP.	CHKD	APP'D	QQAD REVIEW
1	REVISED PER EDCR 89-309	5-9-92	7-16-92		
0	ORIGINAL ISSUE PER EDCR 84-309	RM	WTT		

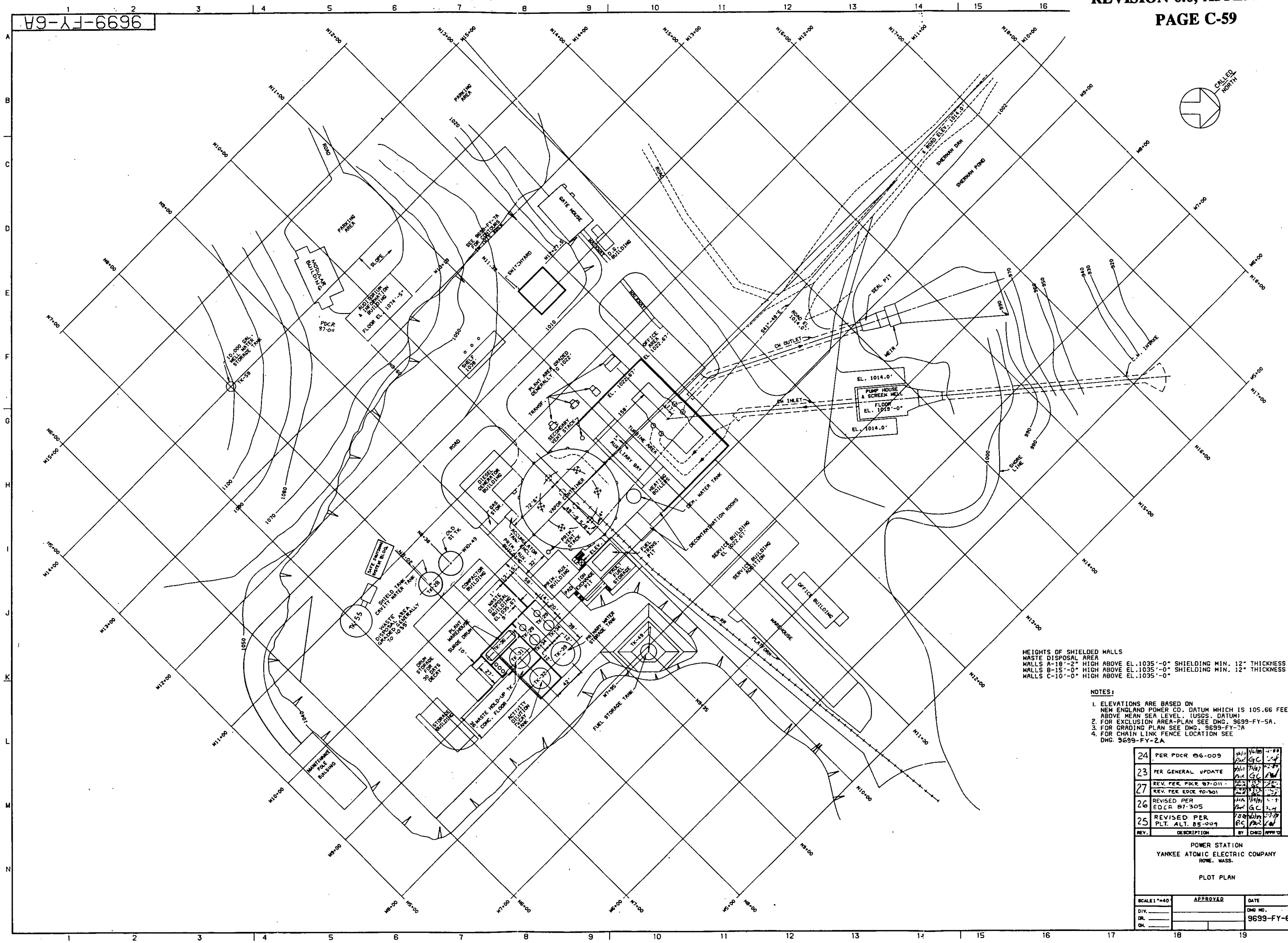
YANKEE ATOMIC ELECTRIC COMPANY
20 TURNPIKE ROAD WESTBORO MASSACHUSETTS
NUCLEAR SERVICES DIVISION

YANKEE ATOMIC ELECTRIC CO.
ROWE, MA

FLOW DIAGRAM - SAFE SHUTDOWN SYSTEM

EDCR 84-309 9699-FM-100B


9699-FV-1A
DESIGN DATA:-
DESIGN PRESSURE - 31.5 PSIG.
DESIGN TEMPERATURE - 249°F.
MATERIAL - ALL SHELL PLATES TO BE ASTM-A 201 GR. B FBX-A300
ALL OTHER MATERIALS AS NOTED ON DRAWINGS.
RADIOGRAPHY - COMPLETE
TESTING - AIR TEST TO 40 PSIG. & HALOGEN LEAK TEST @ 15 PSIG.
CODE - LATEST EDITION OF ASME CODE FOR UNFIRED PRESSURE VESSELS
& LATEST ISSUE OF S.&W. "SPECIFICATION FOR VAPOR CONTAINER."
WELDED JOINT EFFICIENCY - 90%
WIND - 100 MPH.
EMPTY WT. - 3,200,000 LBS.
MAX. WT. DURING INCIDENT - 11,500,000 LBS.
ALL INSERTS FOR PENETRATIONS SHALL BE PREASSEMBLED & STRESS RELIEVED BEFORE WELDING TO VAPOR CONTAINER SHELL.
CODE STAMP - YES
125'-0" O.D. SPHERE
20" DIA. MANHOLE ON NORTH SOUTH ℄ OF CONTAINER
FOR SECTION "1B-1B" SEE DWG. 9699-FV-1B.
H.P. GRAB SAMPLE CONN.
PERSONNEL HATCH
FOR EQUALIZING VALVE ASSEMBLY SEE DWGS. YAE 153 SH 1-5
EL. 1108'-6" EQUATOR
REF. LINE
20" DIA. MANHOLE
EL. 1023'-0" - BOTTOM OF ALL COLUMN BASE PLATES.
EL. 1022'-0" - GROUND GRADE
FOR ALL NEW COLLAR DETAILS SEE EES DWG. SPD 0006-CF-1002
85'-6" BOTTOM OF BASE PLATE TO EQUATOR
59'-11" WORK POINT TO WORK POINT
ELEVATION
SCALE: 1/8" = 1'-0"
CALLED NORTH
℄ VAPOR CONTAINER
EXTENT OF 1 1/2" R. (TYP.)
8 CONCRETE COLUMN SEALS REQUIRED SEE DWG. NO. 9699-FV-15
FOR DETAILS OF ELECTRICAL INSERTS "ETHRUTEE" INCL. SEE DWG. 9699-FV-1U.
SEAL NO. 1
SEAL NO. 2
SEAL NO. 3
SEAL NO. 4
SEAL NO. 5
SEAL NO. 6
SEAL NO. 7
SEAL NO. 8
56'-9" R TOE CONCRETE COLUMN SEALS
20" DIA. MANHOLE
℄ EQUIPMENT ACCESS HATCH
SHADED PORTION INDICATES LIMITS OF 1 1/2" THK. SPHERICAL SHELL R. SEE ORIENTATION PLAN ON DWG. 9699-FV-15
16 SUPPORT COLUMNS EQUALLY SPACED. SEE DWG. NO. 9699-FV-1B
FOR DETAILS OF CONNECTIONS NOT LOCATED ON THIS PLAN, SEE DWG. 9699-FV-1Y.
SECTIONAL PLAN "A-A"
SCALE: 1/8" = 1'-0"
SHOWING PLATE THICKNESS VARIATION ON NORTH SIDE OF VAPOR CONTAINER ONLY.
VIEW "B-B"
SCALE: 1/32" = 1'-0"
REFERENCE DRAWINGS:-
FOR VAPOR CONTAINER DETAILS, SEE DWG. NO. 9699-FV-1B.
FOR DETAILS OF VAPOR CONTAINER PERSONNEL HATCH, SEE CHICAGO BRIDGE & IRON CO. CONTRACT NO. 7-6120, DWG. NOS. 88 TO 102 INCL. & 105.
FOR DETAILS OF VAPOR CONTAINER EQUIPMENT ACCESS HATCH, SEE DWG. NO. 9699-FV-1T.
FOR DETAILS OF VAPOR CONTAINER PIPING PENETRATIONS SHEET 1 & SHEET 2, SEE DWG. NO. 9699-FV-1X, & DWG. NO. 9699-FV-1Y.
FOR PLOT PLAN, SEE DWG. NO. 9699-FY-GA.
FOR CONCRETE COLUMN SEALS FOR VAPOR CONTAINER, SEE DWG. NO. 9699-FV-1S.
FOR DETAILS OF VAPOR CONTAINER INSERTS FOR ELECTRICAL CONNS., SEE DWG. NO. 9699-FV-1U.
FOR DETAILS OF VAPOR CONTAINER ELECTRICAL TERMINAL BOXES, SEE DWG. NO. 9699-FV-4C.
FOR ARRANGEMENT OF STAIRS & WALKWAYS OUTSIDE VAPOR CONTAINER, SHEETS 1 & 2, SEE DWG. NOS. 9699-FM-4GA & 9699-FM-4GB.
FOR DETAILS OF COLUMN ANCHORAGE SEE EES DWG. SPD 0006-CF-1002
NOTE: THESE VAPOR CONTAINER DWGS. SUPERSEDE DWG. NOS. 9699-FM-11A & 9699-FM-12A.
ORIGINAL DRAWING SEALED AND SIGNED BY Walter C. Woodman PE NO. 838
VAPOR CONTAINER
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-FV-1A
REDRAWN FOR CLARITY



HEIGHTS OF SHIELDED WALLS
WASTE DISPOSAL AREA
WALLS A-18'-2" HIGH ABOVE EL.1035'-0" SHIELDING MIN. 12" THICKNESS
WALLS B-15'-0" HIGH ABOVE EL.1035'-0" SHIELDING MIN. 12" THICKNESS
WALLS C-10'-0" HIGH ABOVE EL.1035'-0"

NOTES:

1. ELEVATIONS ARE BASED ON NEW ENGLAND POWER CO. DATUM WHICH IS 105.66 FEET ABOVE MEAN SEA LEVEL. (USGS. DATUM)
2. FOR EXCLUSION AREA-PLAN SEE DWG. 9699-FY-5A.
3. FOR GRADING PLAN SEE DWG. 9699-FY-7A
4. FOR CHAIN LINK FENCE LOCATION SEE DWG. 9699-FY-2A

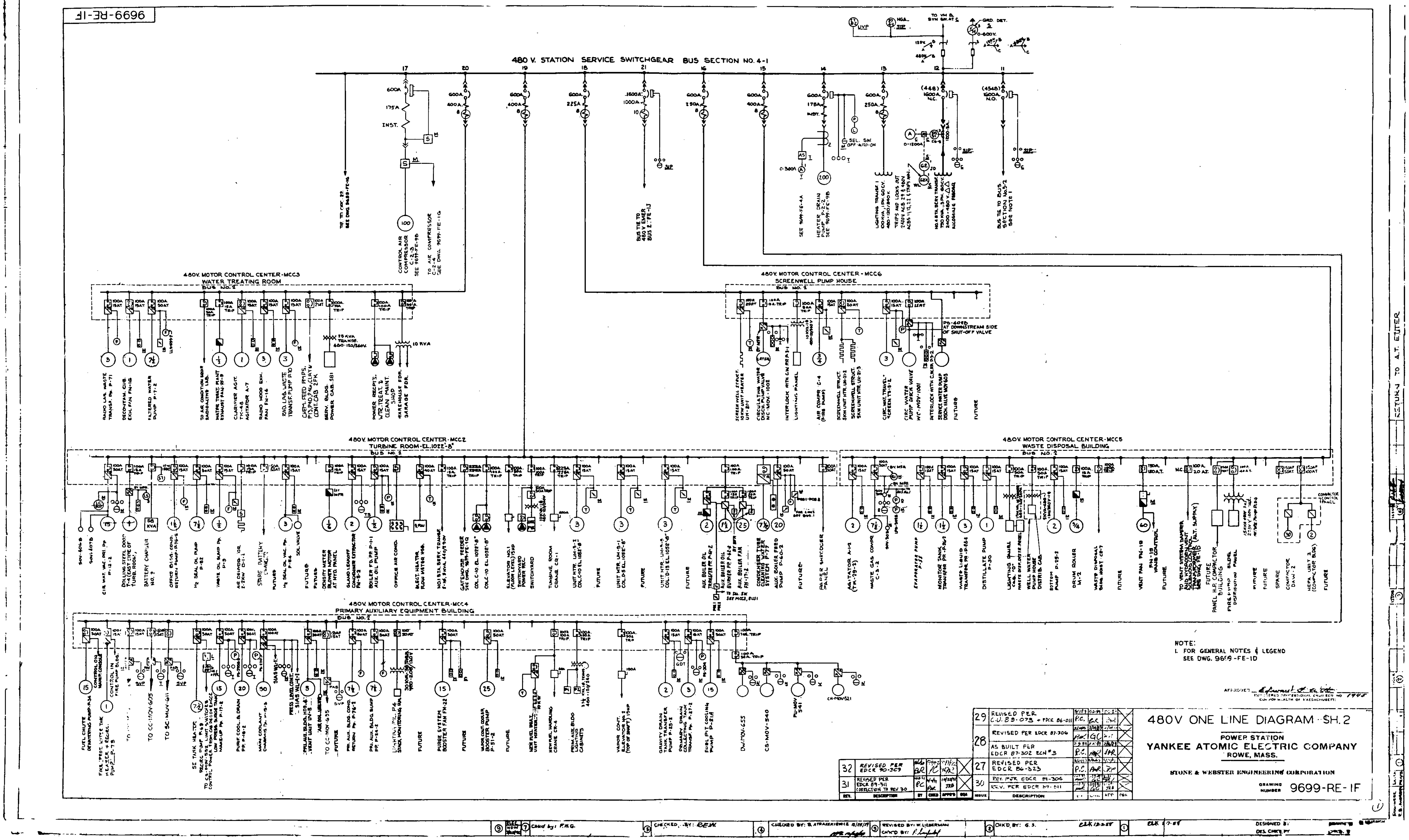
REV.	DESCRIPTION	BY	CHKD	APPR'D
24	PER PDCR 86-009	PAR	GC	[illegible]
23	PER GENERAL UPDATE	PAR	GC	[illegible]
27	REV. PER PDCR 87-011	[illegible]	GC	[illegible]
	REV. PER EDCR 90-301	[illegible]	[illegible]	[illegible]
26	REVISED PER EDCR 87-305	[illegible]	GC	[illegible]
25	REVISED PER PLT. ALT. 85-009	P.C	PAR	[illegible]

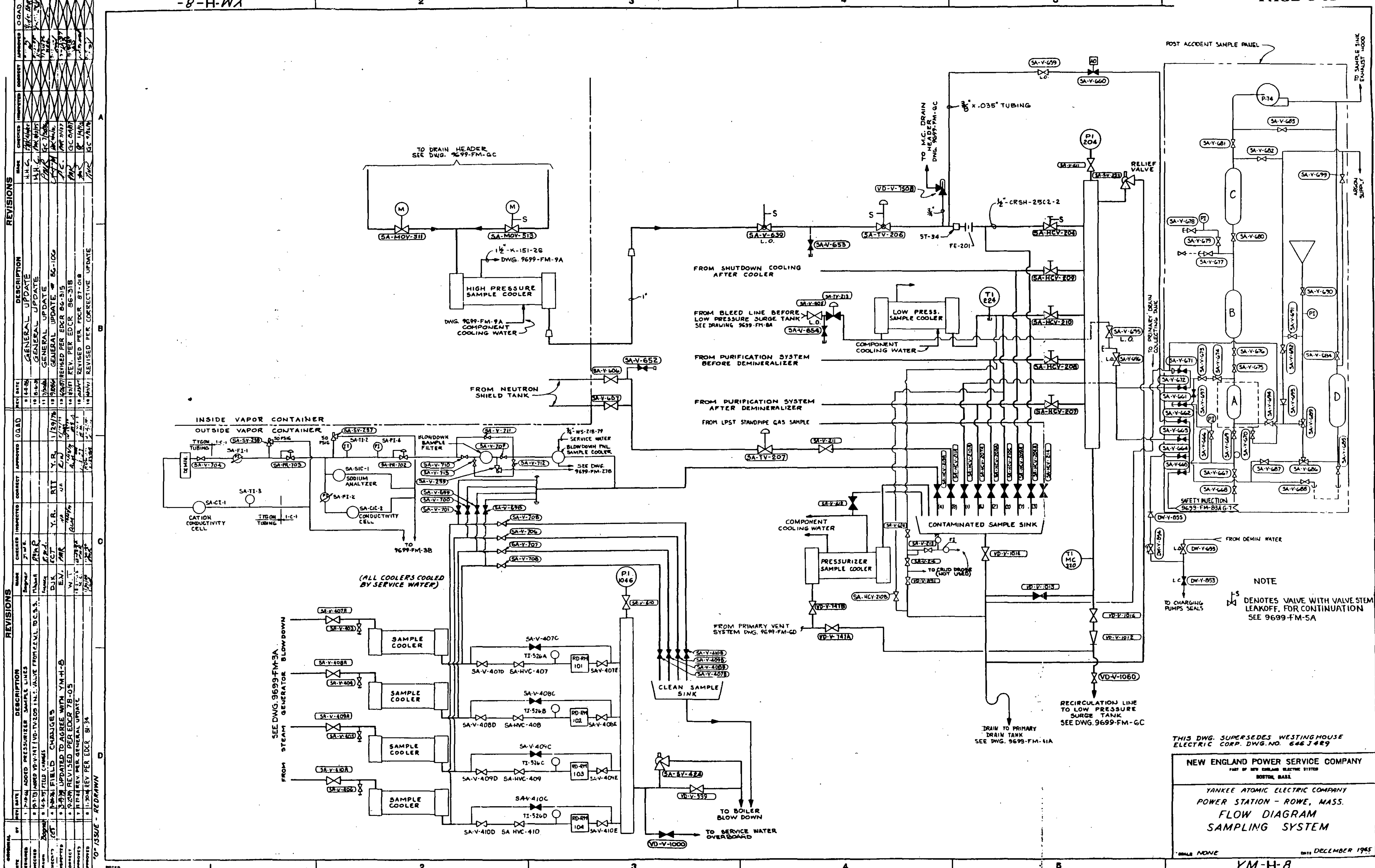
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.

PLOT PLAN

SCALE 1"=40' | APPROVED | DATE
DIV. ______
DR. ______
CH. ______
DWG NO. 9699-FY-6A

9L909 FCD 05/03/95 MPS AUTOMATION SERVICES


9699-RE-1F
480 V. STATION SERVICE SWITCHGEAR BUS SECTION NO. 4-1
480V. MOTOR CONTROL CENTER-MCC3
WATER TREATING ROOM
480V. MOTOR CONTROL CENTER-MCC6
SCREENWELL PUMP HOUSE
480V. MOTOR CONTROL CENTER-MCC2
TURBINE ROOM-EL.1022'-8"
480V. MOTOR CONTROL CENTER-MCC5
WASTE DISPOSAL BUILDING
480V. MOTOR CONTROL CENTER-MCC4
PRIMARY AUXILIARY EQUIPMENT BUILDING
NOTE:
1. FOR GENERAL NOTES & LEGEND SEE DWG. 9699-FE-1D
480V ONE LINE DIAGRAM SH. 2
POWER STATION
YANKEE ATOMIC ELECTRIC COMPANY
ROWE, MASS.
STONE & WEBSTER ENGINEERING CORPORATION
DRAWING NUMBER 9699-RE-1F
RETURN TO A.T. EUTER


REVISIONS
TO DRAIN HEADER SEE DWG. 9699-FM-GC
HIGH PRESSURE SAMPLE COOLER
FROM NEUTRON SHIELD TANK
INSIDE VAPOR CONTAINER
OUTSIDE VAPOR CONTAINER
SODIUM ANALYZER
CATION CONDUCTIVITY CELL
(ALL COOLERS COOLED BY SERVICE WATER)
SAMPLE COOLER
SEE DWG. 9699-FM-3A STEAM GENERATOR BLOWDOWN
FROM
CLEAN SAMPLE SINK
TO BOILER BLOW DOWN
TO SERVICE WATER OVERBOARD
FROM SHUTDOWN COOLING AFTER COOLER
FROM BLEED LINE BEFORE LOW PRESSURE SURGE TANK
FROM PURIFICATION SYSTEM BEFORE DEMINERALIZER
FROM PURIFICATION SYSTEM AFTER DEMINERALIZER
FROM LPST STANDPIPE GAS SAMPLE
LOW PRESS. SAMPLE COOLER
COMPONENT COOLING WATER
CONTAMINATED SAMPLE SINK
PRESSURIZER SAMPLE COOLER
FROM PRIMARY VENT SYSTEM DWG. 9699-FM-GD
DRAIN TO PRIMARY DRAIN TANK SEE DWG. 9699-FM-4A
RECIRCULATION LINE TO LOW PRESSURE SURGE TANK SEE DWG. 9699-FM-GC
RELIEF VALVE
POST ACCIDENT SAMPLE PANEL
SAFETY INJECTION 9699-FM-83A G-7
FROM DEMIN WATER
TO CHARGING PUMPS SEALS
NOTE
DENOTES VALVE WITH VALVE STEM LEAKOFF. FOR CONTINUATION SEE 9699-FM-5A
THIS DWG. SUPERSEDES WESTINGHOUSE ELECTRIC CORP. DWG. NO. 646J429
NEW ENGLAND POWER SERVICE COMPANY
YANKEE ATOMIC ELECTRIC COMPANY
POWER STATION - ROWE, MASS.
FLOW DIAGRAM
SAMPLING SYSTEM
SCALE NONE
DATE DECEMBER 1965
YM-H-8