# CLINCH RIVER

# BREEDER REACTOR PROJECT

# PRELIMINARY SAFETY ANALYSIS REPORT

## VOLUME 1

PROJECT MANAGEMENT CORPORATION

TABLE OF CONTENTS

## TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)





Amend. 68
May 1982

TABLE OF CONTENTS (Cont'd.)

## TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

## TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)



Amend. 68
May 1982

## TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

## TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

TABLE OF CONTENTS (Cont'd.)

# CLINCH RIVER

# BREEDER REACTOR PROJECT

# PRELIMINARY SAFETY ANALYSIS REPORT

## CHAPTER 1

## INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

PROJECT MANAGEMENT CORPORATION

## TABLE OF CONTENTS

TABLE OF CONTENTS

Page No,

Amend. 49
Apr. 1979

# LIST OF TABLES

## LIST OF REFERENCES

## LIST OF FIGURES

LIST OF FIGURES (cont'd)

LIST OF FIGURES (cont'd)



1 1

46 39

## LIST OF FIGURES (cont'd)



39

## LIST OF FIGURES (cont'd)



1 1 44 46 39

Amend. 46
August 1978

LIST OF FIGURES (cont'd)





Amend. 46
August 1978

LIST OF FIGURES (cont'd)



46 24 39

Amend. 46
August 1978

## LIST OF FIGURES



44 35

# NOTICE

The Energy Reorganization Act of 1974 (Public Law 93-438) establishing the Energy Research and Development Administration (E.R.D.A.) and the Nuclear Regulatory Commission (N.R.C.) became effective on January 19, 1975.

Throughout this Preliminary Safety Analysis Report, appearance of or reference to the Atomic Energy Commission (A.E.C.) (with the exception of the Directorate of Regulation) will now mean the Energy Research and Development Administration.

Appearance of or reference to the Atomic Energy Commission (Directorate of Regulation) will now mean the Nuclear Regulatory Commission.

## Chapter 1.0 - INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

### 1.1 Introduction

The Clinch River Breeder Reactor Plant (CRBRP) will provide a vital step in the United States' reactor development program. The objective of the U.S. Energy Research and Development Administration (ERDA) Liquid Metal Fast Breeder Reactor (LMFBR) program is to develop, on a broad, proven technological and engineering base, with joint utility and industry participation, a commercial breeder reactor industry.

In keeping with the objective of the LMFBR program, the objectives of the CRBRP are as follows:

1. To confirm and demonstrate the potential value and environmental desirability of the LMFBR concept as a practical and economic future option for generating electical power;

2. Confirm the value of this concept for conserving important nonrenewable national resources;

3. To develop, for the benefit of government, industry and the public, important technological and economic data;

4. To provide a broad base of experience and information important for commercial and industrial application of the LMFBR concept; and

5. To verify certain key characteristics and capabilities of LMFBR plants for operation on utility systems such as licensability and safety, operability, reliability, availability, maintainability, flexibility and prospect for economy.

Since there is limited experience within the present-day licensing framework which is directly relatable to a first-of-a-kind demonstration plant such as the CRBRP, the information presented in this introductory section is more extensive than normally found in light water reactor PSARs.

41

### 1.1.1 General Information

This PSAR is submitted in support of a joint application by the
23 United States Energy Research and Development Administration (ERDA),
Project Management Corporation (PMC) and the Tennessee Valley Authority
(TVA) for a CP and Class 104 (b) Operating License to construct and operate
the Nation's first large-scale LMFBR Demonstration Plant.

The plant will consist of a single generating unit, employing a
liquid metal cooled fast breeder reactor Nuclear Steam Supply System
(NSSS). Westinghouse Electric Corporation (Advanced Reactors Division)
is responsible for the design of the NSSS and of the steel containment,
under the technical direction of the United States Energy Research and
Development Administration's (ERDA) Division of Reactor Development and
23 Demonstration. The General Electric Company and the Atomics International
Division of Rockwell International Corporation have major subcontracts,
related to the NSSS, from Westinghouse. Burns & Roe is responsible for the
design of the balance-of-blant (BOP) and other functions normally associated
with the architect-engineer (e.g., characterization of the site seismology,
etc.), and Stone and Webster Engineering Corporation is responsible for
41 construction of the plant and site facilities, both under the direction
of ERDA. The plant will be operated by TVA through contractual arrange-
23 ments with ERDA. Further amplification of the relationship between these
participants in the Project is found in Section 1.4 of this PSAR.

The Clinch River site is in east central Tennessee in the eastern part of Roane County and within the town limits of Oak Ridge, approximately 25 miles west of Knoxville. The site is on a peninsula bounded on the north by ERDA's Oak Ridge Reservation and on the remaining sides by the Clinch River. Complete details of the site location, layout and characteristics are given in Chapter 2 of this PSAR.

The design power level for the plant is 975 MW(th), corresponding to a gross generation level of 380 MW(e). This power level is discussed under the terms "thermal/hydraulic" (T/H) conditions in various sections of the PSAR. It is this power level which forms the basis for the present application, and for the safety analyses presented in Chapter 15. However, the permanent components of the plant (heat transport system, core support structure, BOP, etc.) have been designed for additional capability, namely for a power level of 1121 MW(th) corresponding to a gross generation level of 439 MW(e). These latter conditions are referred to as "stretch" conditions in this PSAR. In various sections, components are shown to be capable of accommodating "stretch" conditions. Although "stretch" conditions do not form the basis for the present application, subsequent to issuance of the Construction Permit, a supplementary application may be made to increase the power level to thse "stretch" conditions. However, for purposes of the CP review, the additional capability of permanent plant components should be treated as an inherent margin in the plant design.

The plant is designed with three main coolant loops and the intended mode of operation is that all three loops should be continuously in service.

While the system design is intended to be capable of allowing for operation at power on two loops, the applicant is not requesting NRC review of this operational mode. If, at some time in the future, the applicant considers that all safety requirements can be met under two-loop operatioin without significant additional design features, the applicant may elect to apply to NRC for a two-loop operation capability. This should not constrain the NRC review of the construction permit application.

The construction completion data for the plant was originally scheduled for September, 1981. Until current Congressional and Administration actions are completed, the reactor criticality data cannot be re-scheduled.

### 1.1.2 Overview of Safety Design Approach

The design of the CRBRP is based on the defense-in-depth safety philosophy, commonly known as the Three Levels of Safety design approach. A summary of the design safety approach for the CRBRP is provided in Tables 1.1-1 and -2.

#### Level 1 Design

The first level of safety provides reliable plant operation and prevention of accidents during normal operating conditions through the intrinsic features of the design, such as quality assurance, redundancy, maintainability, testability, inspectibility, and fail-safe characteristics. The plant is designed not only to accommodate steady-state power conditions, but also to have adequate tolerance for normal operating transients, such as start-up, shutdown, and load-following. As a basic part of the CRBRP development program, a number of large-scale engineering proof tests are being performed to verify the design concepts. This testing process provides predictability of performance and, hence, safety through assurance of the use of proven methods, materials, and technology.

Extensive pre-operational test programs will be conducted in the plant to assure conformance of components and systems to the established performance requirements. Key parameters will be monitored continuously or routinely and well-define surveillance, in-service inspection, and preventive maintenance programs will be carried out by a trained operating and maintenance staff to provide assurance that as-built quality is maintained through the life of the plant.

#### Level 2 Design

The second level of safety provides protection against Anticipated and Unlikely Faults (such as partial loss of flow, reactivity insertions, failure of parts of the control system, or fuel handling errors - Faults are defined in Table 1.1-1A) which might occur in spite of the care taken in design, construction, and operation of the plant. This level of safety for the public is provided by redundancy of critical components as well as by protection devices and systems designed to assure that such events will be arrested. The requirements for these protection systems are based on a spectrum of occurrences which the plant design must safely accommodate. Conservative design practices, including providing redundant detecting and actuating equipment, are incorporated in the protection systems to assure both the effectiveness and reliability of this second level of design. These systems are designed to be routinely monitored and tested to provide full assurance that if they are required to operate, they will do so reliably.

Level 3 Design

The third level of safety supplements the first two levels by providing acceptable plant response to extremely unlikely faults such as large pipe leaks, large sodium fires, or large sodium-water reactions. A list of extremely unlikely faults used as design cases is provided in Table 1.1-3.
41 Although occurrence of these faults is of low probability, appropriate engineered safety features are incorporated into the CRBRP design to safely accommodate such events. Conservative assumptions and evaluation methods, such as assumed failure of any single active component, are used to develop adequate designs. In addition, conditions associated with extremely unlikely natural phenomena, which bound the most severe that have been historically reported for the site and the surroundings, are used as design bases for the plant. These include such low probability events as severe earthquakes, tornadoes, and floods. These faults and natural phenomena combine to define the design basis envelope.

Reliability Program

Primary emphasis in the CRBRP design is placed on adherence to the first two levels of safety; part of this emphasis is carried by the Reliability Program. The basic objective of the Reliability Program is to provide assurance beyond the normal design process that the probability of exceeding the guidelines for radiological release is acceptably low.

The details of the Reliability Program are discussed in PSAR Appendix C.

Design Margins Beyond the Design Base

As discussed above, the safety design philosophy of the plant provides for mitigation of the full range of events from relatively trivial events to postulated design basis accidents.

Because of the extremely conservative approach to safety of the plant and the extensive safety features provided, it has been concluded by the Project and concurred in by the NRC staff that core disruptive accidents can and must be excluded from the design basis accidents for the plant. This is documented in References 1 and 2.

41

However, to provide extra margin, because of the first-of-a-kind nature of the plant, certain hypothetical core disruptive accidents have been analyzed. As a unique feature of the design of this plant, additional Design Margins are provided such that even the consequences of postulated HCDAs will also be adequately accommodated. These include both structural and thermal margins.

Structural Margin Beyond the Design Basis (SMBDB) assure that extra margins exist to accommodate structural loadings on the reactor vessel system and the PHTS components from postulated HCDAs. Thermal Margins Beyond the Design Basis (TMBDB) assure that radiological consequences of HCDAs will be mitigated to acceptable levels. Preliminary details are provided in Reference 10 of PSAR Section 1.6.

Piping Integrity Considerations

The integrity of the sodium piping in containment is of special concern because of the function of the sodium heat transport systems in maintaining the reactor core in a safe condition. A multifacetted program has been undertaken by the Project to assure and document the high integrity of the in-containment sodium piping. The details of the piping integrity program are discussed in Reference 2 of PSAR Section 1.6. The program will show that with the appropriate materials surveillance, in-service inspection and leak detection provisions, double-ended pipe ruptures of the sodium piping in containment can be excluded from the CRBRP design basis.

### 1.1.3 Applicability of Regulatory Guides

The NRC Regulatory Guides provide Regulatory guidance as to how the requirements of NRC regulations may be satisfied. These requirements are set forth in Appendix A to 10CFR Part 50 for design of nuclear power plants and in various parts of 10 CFR for design, construction, operation, and quality assurance. Many of the detailed requirements address directly the light-water-cooled nuclear power plants. Consequently, a number of the requirements of the existing Regulatory Guides may or may not apply to the CRBRP, due to the differences in designs between the LMFBR plants and the LWR plants.

The Regulatory Guides have been reviewed for applicability to CRBRP and the NRC guidance is followed as appropriate to an LMFBR. A summary of this applicability to CRBRP is provided in Appendix I. The Regulatory Guides have been designated: applicable in total, applicable in principle and intent, or application to CRBRP is inappropriate. Table 1 will be revised for the FSAR to add references to Sections where specific applicability of the Regulatory Guide is discussed.

References to Section 1.1

1. Letter from R. P. Denise (NRC) to L. W. Caffey (Director of CRBRP Project), May 6, 1976.

2. Final Environmental Statement for the Clinch River Breeder Reactor Plant (Docket No. 50-537), NUREG-0139, February 1977.

TABLE 1.1-1

SUMMARY OF DESIGN SAFETY APPROACH FOR THE CRBRP

This table represents the CRBRP Project Design Safety Approach.

41 1. The following three levels of safety approach used in CRBRP design is generally consistent with the three levels of safety concept used by Regulatory to evaluate the adequacy, for licensing purposes, of nuclear power reactors. However, beyond these three levels, the CRBRP approach includes structural and thermal margins beyond the design base as indicated in reference 10 of Section 1.6. 41

a. The first level focuses on the reliability of operation and prevention of accidents through the intrinsic features of the design, construction, and operation of the plant, including quality assurance, redundancy, testability, inspectability, maintainability, and failsafe features of the components and systems of the entire plant.

b. The second level focuses on the protection against Anticipated Faults and Unlikely Faults (as defined in Table 1.1-1A) which might occur despite the care taken in design, construction, and operation of the plant set forth in Level One above. This protection will ensure that the plant is placed in a safe condition following one of these faults.

c. The third level focuses on the determination of events to be classified as Extremely Unlikely Faults (as defined in Table 1.1-1A) and their inclusion in the design basis. Table 1.1-3 contains a list of such "Extremely Unlikely Faults". These faults are of low probability and no such events are expected to occur during the plant lifetime. Even though they represent extreme and unlikely cases of failures, they have been analyzed using the same conservative assumptions as those employed in consideration of second level events.

41

TABLE 1.1-1A

DEFINITION OF TRANSIENTS

1. Anticipated Fault

   An off-normal condition which individually may be expected to occur once or more during the plant lifetime.

2. Unlikely Fault

   An off-normal condition which individually is not expected to occur during the plant lifetime; however, when integrated over all components and systems, events in this category may be expected to occur once or more during the life of the plant.

3. Extremely Unlikely Fault

   An off-normal condition of such low probability that no events in this category are expected to occur during the plant lifetime, but which nevertheless represents extreme or limiting cases of failures.

TABLE 1.1-2

MULTI-LEVEL DESIGN OF CRBRP

| 'EL | OBJECTIVE | OPERATING CATEGORY | TYPICAL EVENTS WITHIN OPERATING CATEGORY | OCCURRENCE PROBABILITY | TYPICAL DESIGN FEATURES |
|---|---|---|---|---|---|
| | Provision of simple, reliable and functional design free of defects, with inherent safe performance, fabricated and operated to the highest proven standards. | Normal operation | • Full power operation<br>• Startup & Shutdown<br>• Refueling | Normally occurring | • Fuel assembly designed to prevent flow blockage<br>• Fuel of proven performance through life<br>• Core restraint to provide negative power coefficient<br>• Adequate Doppler coefficient<br>• Low pressure coolant systems with wide margin to boiling<br>• Maximum use of proven technology and hands-on maintenance<br>• Radioactive waste treatment system<br>• Multiple reactor coolant loops<br>• Decay heat removal redundancy<br>• Battery power supplies for vital services<br>• Guard vessels for leak protection<br>• Inert atmosphere in Sodium Cells |
| 2 | Provision of protection systems for adequate response of system in the event of all identified transients. | Anticipated Faults* | • Loss of off-site power<br>• Loss of power to one pump<br>• Spurious scrams | Each event is expected to occur at least once in the life of the plant | • Two independent shutdown systems<br>• Sodium water reaction protection system |
| | | Unlikely Faults* | • Pump seizure<br>• Liquid Radwaste system failure<br>• Small Steam Generator leak | These events are individually not expected to occur, but based on the total list of such events, one might occur once during the life of the plant. | |

15

## TABLE 1.1-2

## MULTI-LEVEL DESIGN OF CRBRP

Page 2

| LEVEL | OBJECTIVE | OPERATING CATEGORY | TYPICAL EVENTS WITHIN OPERATING CATEGORY | OCCURRENCE PROBABILITY | TYPICAL DESIGN FEATURES |
|---|---|---|---|---|---|
| 3 | Provision of extra capability to cope with extremely unlikely events which are never expected to occur, and additional design fequirements to provide prudent margin for unforeseen events. | Extremely Unlikely Faults* | • Large sodium-fire<br>• Large sodium water reaction<br>• Large steam system pipe rupture | Never expected to occur | • Containment isolation<br>• Low leakage containment<br>• SWR pressure relief system<br>• Pipe rupture criteria |
| | | Hypothetical Events | Unforeseen events, hypothesized to involve a radiological release | The reliability program shall confirm that the probability of exceeding 10CFR100 site dose criteria is acceptab,y low. | • Capability to accept extra thermal loads in the core support structure<br>• Capability to accept dynamic loads in vessel and primary system components<br>• Geometric requirements in and around the vessel<br>• Control room radiological protection |
| | Provision of capability to withstand natural phenomena to a degree consistent with established NRC guidelines. | Design Basis Environmental Events | • Flood<br>• Earthquake<br>• Tornado | These events at design bases magnitudes are never expected to occur. | • Site selection, Flood barriers<br>• Seismic design criteria<br>• Tornado design criteria |

*Defined in Table 1.1-A

TABLE 1.1-3

LIST OF EXTREMELY UNLIKELY FAULTS USED AS DESIGN BASES

Design Basis Earthquake, Flood or Tornado

Large Steam System Pipe Rupture

41

Large Sodium Spills Inside and Outside Containment

Large $Na-H_2O$ Reactions in the Steam Generator

TABLE I

EVALUATION OF APPLICABILITIES OF EXISTING NRC REGULATORY GUIDES TO THE CLINCH RIVER BREEDER REACTOR PLANT

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| - | 1.1 | Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | CRBRP design precludes the need for an ECCS or other safety related systems with pumps whose performance depends on containment pressure. |
| - | 1.2 | Thermal Shock to Reactor Pressure Vessels | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | The CRBRP Reactor Vessel is not subjected to thermal shocks with significant potential to cause brittle fracture even after a 30 year accumulation of irradiation damage. |
| 2 | 1.3 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | Refer to Regulatory Guide 1.4. The guide is applicable to BWRs. |
| 2 | 1.4 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident to Pressurized Water Reactors. | Applicable in principle and intent. | Accident conditions and source terms delineated in Regulatory Position C.1 are specific to PWR technology. Positions C.2a through C.2f are applicable to CRBRP. |
| - | 1.5 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | These regulatory requirements are for BWR plants. Reg. Guides 1.109 and 1.145 identify the assumptions to be used for radiological analysis of accidents. |
| - | 1.6 | Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems. | Applicable in Total. | |
| - | 1.7 | Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | CRBRP is not subject to a design basis accidents with potential to produce significant amounts of hydrogen in the containment. |
| - | 1.8 | Personnel Selection and Training | Applicable in Total. | |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| 2 | 1.9 | Selection, Design and Qualification of Diesel-Generator Units Used as Standby (Onsite) Electric Power Systems at Nuclear Power Plants. | Applicable in Total. | |
| 1 | 1.10 | Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures | Applicable in Principle and Intent | Superceded by R.G. 1.136, 1.142, 1.94 |
| - | 1.11 | Instrument Lines Penetrating Primary Reactor Containment | Applicable in total. | |
| | | Supplement to Safety Guide 11, Back fitting Considerations | Applicable in total. | |
| 1 | 1.12 | Instrumentation for Earthquakes | Applicable in principle and intent. | The location restrictions specified for LWRs are not responsive to the structural and equipment differences in an LMFBR. Appropriate locations for CRBRP are identified. |
| 1 | 1.13 | Spent Fuel Storage Facility Design Basis | Applicable in principle and intent. | Specific elements of the method of implementation as described in this Reg. Guide are not compatible with a liquid metal coolant design. |
| 1 | 1.14 | Reactor Coolant Pump Flywheel Integrity | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | Coolant pumps in CRBRP are not provided with flywheels. |
| 1 | 1.15 | Testing of Reinforcing Bars for Category I Concrete Structures | Applicable in principle and intent | Superceded by R.G. 1.136, 1.142, 1.94 |
| 4 | 1.16 | Reporting of Operating Information - Appendix A Technical Specification | Applicable in Total. | |
| 1 | 1.17 | Protection of Nuclear Plant Against Industrial Sabotage | Applicable in Total. | |
| 1 | 1.18 | Structural Acceptance Test for Concrete Primary Reactor Containments. | Applicable in Principle and Intent. | Superceded by R. G. 1.136 |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| 1 | 1.19 | Nondestructive Examination of Primary Containment Liner Welds | Applicable in Principle and intent | Superceded by R.G. 1.136 |
| 2 | 1.20 | Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing. | Applicable in principle and intent. | The high temperature sodium environment in CRBRP precludes in place inspection and the use of instruments to measure UIS vibration. Extra emphasis is being placed on the vibration analysis program and the testing of scale models to compensate for this limitation. |
| 1 | 1.21 | Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive materials in Liquid and Gaseous Effluents from Light Water-Cooled Nuclear Power Plants | Applicable in principle and intent | Evaluation and reporting of the quantity of radioactive effluents pertaining to LMFBR technology are applied. Operating information is covered under R.G. 1.16. |
| - | 1.22 | Periodic Testing of Protection System Actuation Functions. | Applicable in total. | |
| - | 1.23 | Onsite Meteorological Programs. | Applicable in total. | |
| - | 1.24 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure. | Applicable in principle and intent. | Gas storage tanks for noble gas decay are located inside controlled ventilation protective structures. Offsite radiological consequences account for these protective structures. |
| - | 1.25 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | This R.G. discusses pool water absorption parameters and is primarily concerned with iodine-based thyroid doses to offsite individuals. CRBRP does not include a fuel storage (water pool) nor any other significant iodine source. |
| 3 | 1.26 | Quality Group Classifications and Standards for Water, Steam-and Radioactive-Waste-Containing Components of Nuclear Power Plants. | Applicable in principle and intent. | CRBRP meets the intent of this Guide as detailed in the response to NRC Question 001.274. |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| 2 | 1.27 | Ultimate Heat Sink for Nuclear Power Plants | Applicable in principle and intent. | CRBRP utilizes both water and air as ultimate heat sinks. |
| 2 | 1.28 | Quality Assurance Program Requirements (Design and Construction) | Applicable in principle and intent. | See note 1. |
| 3 | 1.29 | Seismic Design Classification | Applicable in principle and intent. | The Reg. Guide refers to LWRs and and their technology. CRBRP utilizes LMFBR equivalents. |
| - | 1.30 | Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment. | Applicable in principle and intent. | See note 1. |
| 3 | 1.31 | Control of Ferrite Content in Stainless Steel Weld Metal. | Applicable in principle and intent | The intent is met by assuring a range of 5 to 9% delta ferrite. |
| 2 | 1.32 | Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants. | Applicable in Total. | |
| 2 | 1.33 | Quality Assurance Program Requirements (Operation). | Applicable in principle and intent. | See note 1. |
| - | 1.34 | Control of Electroslag Weld Properties | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | This weld technique is not used on safety related equipment, however if electroslag welding is used on safety related equipment the requirements of this R.G. will be met. |
| 2 | 1.35 | Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | CRBRP containment is not a prestressed concrete containment. |
| - | 1.36 | Nonmetallic Thermal Insulation for Austenitic Stainless Steel. | Applicable in Total. | |
| - | 1.37 | Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants. | Applicable in principle and intent. | See note 1. |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| 2 | 1.38 | Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants. | Applicable in principle and intent. | See note 1 |
| 2 | 1.39 | Housekeeping Requirements for Water-Cooled Nuclear Power Plants. | Applicable in principle and intent. | See note 1 |
| - | 1.40 | Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants. | Applicable in principle and intent. | Applicable except accident qualification environments will be based on CRBRP design basis events. |
| - | 1.41 | Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments. | Applicable in Total. | |
| - | 1.42 | Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors. | Withdrawn by NRC | |
| - | 1.43 | Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | The Regulatory Guide is concerned with SS Welds of low-alloy steel components under water at high pressures. The situations of concern as described in the Regulatory Guide do not exist in CRBRP |
| - | 1.44 | Control of the Use of Sensitized Stainless Steel. | Applicable in principle and intent. | Due to the high temperature of operation, sensitization cannot be avoided. Low internal system pressures and selection of operating environments, including insulation, will minimize the potential for stress corrosion cracking. |
| - | 1.45 | Reactor Coolant Pressure Boundary Leakage Detection Systems. | Applicable in principle and intent. | Leakage characteristics for CRBRP primary sodium coolant are not the same as for LWRs. Detectors and detection methods are different. |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| - | 1.46 | Protection Against Pipe Whip Inside Containment | Applicable in principle and intent. | CRBRP meets the intent of this Regulatory Guide, however, the primary coolant system is a low pressure system and not subject to pipe whip. |
| - | 1.47 | Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems. | Applicable in Total. | |
| - | 1.48 | Design Limits and Loading Combinations for Seismic Category I Fluid Systems Components. | Applicable in principle and intent. | The loadings associated with design basis events in CRBRP are different from those associated with design basis events in LWRs. Leakage from IHTS and SGS is the only leakage with significant potential to produce dynamic system loadings. |
| 1 | 1.49 | Power Levels of Nuclear Power Plants. | Applicable in Total. | |
| - | 1.50 | Control of Preheat Temperature for Welding of Low-Alloy Steel. | Applicable in Total. | |
| - | 1.51 | Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components. | Withdrawn by NRC | |
| 2 | 1.52 | Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants. | Applicable in Total. | |
| - | 1.53 | Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems. | Applicable in Total. | |
| - | 1.54 | Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants. | Applicable in principle and intent. | See note 1. |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| - | 1.55 | Concrete Placement in Category I Structures | Applicable in principle and intent. | Superceded by R.G. 1.136, 1.142, 1.94 |
| 0 | 1.56 | Maintenance of Water Purity in Boiling Water Reactors | Applicable in principle and intent. | CRBRP uses Sodium Technology to establish purity limits for the reactor coolant. |
| - | 1.57 | Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components. | Applicable in principle and intent. | Design limits and load combinations have been applied for CRBRP design basis events. |
| 1 | 1.58 | Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel. | Applicable in principle and intent. | See note 1. |
| 2 | 1.59 | Design Basis Floods for Nuclear Power Plants. | Applicable in Total. | |
| | | Errata to Regulatory Guide 1.59. | Applicable in Total. | |
| 1 | 1.60 | Design Response Spectra for Seismic Design of Nuclear Power Plants. | Applicable in Total. | |
| - | 1.61 | Damping Values of Seismic Design of Nuclear Power Plants. | Applicable in Total. | |
| - | 1.62 | Manual Initiation of Protective Actions. | Applicable in Total. | |
| 2 | 1.63 | Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants. | Applicable in Total. | |
| 2 | 1.64 | Quality Assurance Requirements for the Design of Nuclear Power Plants. | Applicable in principle and intent. | See note 1. |
| - | 1.65 | Materials and Inspection for Reactor Vessel Closure Studs | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | The CRBRP reactor vessel closure head is not subjected to high internal pressures for normal operation or design basis events. |
| - | 1.66 | Nondestructive Examination of Tubular Products. | Withdrawn by NRC | |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| - | 1.67 | Installation of Overpressure Protective Devices. | Applicable in Total. | |
| 2 | 1.68 | Initial Test Programs for Water-Cooled Nuclear Power Plants. | Applicable in principle and intent. | CRBRP will use applicable portions of the Reg. Guides in developing an initial startup test program tailored to an LMFBR type plant. |
| 1 | 1.68.1 | Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Plants. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | This guide is pertinent to BWR's only. |
| 1 | 1.68.2 | Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants. | Applicable in principle and intent. | CRBRP will use applicable portions of the Reg. Guide in developing an initial startup test program tailored to an LMFBR type plant. |
| - | 1.69 | Concrete Radiation Shields for Nuclear Power Plants. | Applicable in Total. | |
| 3 | 1.70 | Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants. | Applicable in Principle and Intent. | The CRBR PSAR used "Standard Format and Content of Safety Analysis Report for Nuclear Power Plants: "LMFBR Edition" (Issued by NRC 1974). |
| - | 1.71 | Welder Qualification for Areas of Limited Accessibility. | Applicable in principle and intent. | See note 1. |
| 2 | 1.72 | Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | There is no spray pond in CRBRP. |
| - | 1.73 | Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants. | Applicable in principle and intent. | Due to the low pressure sodium coolant systems located in inerted cells, CRBRP is not subject to a LOCA involving high containment pressure, temperature, humidity and radiation. Other appropriate design basis events have been identified for CRBRP equipment qualification. |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| - | 1.74 | Quality Assurance Terms and Definitions. | Applicable in principle and intent. | See note 1. |
| 2 | 1.75 | Physical Independence of Electric Systems. | Applicable in Total. | |
| - | 1.76 | Design Basis Tornado for Nuclear Power Plants. | Applicable in Total. | |
| - | 1.77 | Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors. | Applicable in principle and intent. | Appropriate reactivity transients and analysis assumptions have been specified for CRBRP. |
| - | 1.78 | Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release. | Applicable in Total. | |
| 1 | 1.79 | Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | The CRBRP design precludes the need for an ECCS. |
| - | 1.80 | Preoperational Testing of Instrument Air Systems. | Applicable in Total. | |
| 1 | 1.81 | Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | CRBRP is a single unit plant. |
| - | 1.82 | Sumps for Emergency Core Cooling and Containment Spray Systems | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | CRBRP design precludes the need for either ECCS or a containment spray system or other safety related systems taking suction on the containment sump(s). |
| 1 | 1.83 | Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes. | Applicable in principle and intent. | The Steam Generators operate at high temperatures and in liquid metal environments. They have an inservice inspection program designed for an LMFBR. |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| 18 | 1.84 | Design and Fabrication Code Case Acceptability ASME Section III Division 1. | Applicable in principle and intent. | Some components are built to a previous Regulatory Guide revision. |
| 18 | 1.85 | Materials Code Case Acceptability - ASME Section III Division 1. | Applicable in principle and intent. | Some components are built to a previous Regulatory Guide revision. |
| - | 1.86 | Termination of Operating Licenses for Nuclear Reactors. | Applicable in Total. | |
| 1 | 1.87 | Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Classes 1592, 1593, 1594, 1595, and 1596). | Applicable in Total. | |
| 2 | 1.88 | Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records. | Applicable in principle and intent. | See note 1. |
| - | 1.89 | Qualification of Class IE Equipment for Nuclear Power Plants. | Applicable in principle and intent. | Due to the low pressure primary coolant system located in inerted cells, CRBRP is not subject to a LOCA involving high containment pressures, temperatures, humidity and radiation levels. Other appropriate design basis events have been identified for CRBRP equipment qualification. |
| 1 | 1.90 | Inservice Inspection of Prestressed Containment Structures with Grouted Tendons. | The Regulatory guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | CRBRP containment is not a prestressed containment. |
| 0 | 1.91 | Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants. | Applicable in Total. | |
| 1 | 1.92 | Combining Modal Responses and Spatial Components in Seismic Response Analysis. | Applicable in Total. | |
| - | 1.93 | Availability of Electric Power Sources. | Applicable in Total. | |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| 1 | 1.94 | Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants. | Applicable in principle and intent. | See note 1. |
| 1 | 1.95 | Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release | Applicable in principle and intent. | Liquid or gaseous chlorine will not be used for treatment of circulating water and will not be stored on site. Possible off-site release control requirements will be as per Reg. Guide 1.78. |
| 1 | 1.96 | Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | CRBRP design precludes the need for a main steam isolation valve leakage control system or a similar system as part of the containment system. |
| 2 | 1.97 | Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident. | Applicable in principle and intent. | The parameters to be monitored for accidents in CRBRP are different than those for LWRs due to the difference in technology. Parameters appropriate for an LMFBR will be monitored. |
| - | 1.98 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor. | Issued for comment. | |
| 1 | 1.99 | Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | The CRBRP reactor vessel does not contain ferritic materials subject to significant irradiation damage. Reactor vessel material in the belt-line region is austinitic stainless steel not subject to brittle fracture under any normal or accident conditions. |
| 1 | 1.100 | Seismic Qualification of Electric Equipment for Nuclear Power Plants. | Applicable in Total. | |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| 2 | 1.101 | Emergency Planning for Nuclear Power Plants. | Applicable in Total. | |
| 1 | 1.102 | Flood Protection for Nuclear Power Plants. | Applicable in Total. | |
| 1 | 1.103 | Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments. | The Regulatory Guide has no commonality with CRBRP design, therefore, application to CRBRP is inappropriate. | CRBRP does not utilize a post-tensioned prestressing system. |
| - | 1.104 | Overhead Crane Handling Systems for Nuclear Power Plants. | Withdrawn by NRC | |
| 1 | 1.105 | Instrument Setpoints | Applicable in Total. | |
| 1 | 1.106 | Thermal Overload Protection for Electric Motors on Motor-Operated Valves. | Applicable in Total. | |
| 1 | 1.107 | Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | Cement grouting for pre-stressing tendons in containment structures is not used in CRBRP. |
| 1 | 1.108 | Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants. | Applicable in Total. | |
| | | Errata to Regulatory Guide 1.108. | | |
| 1 | 1.109 | Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I. | Applicable in principle and intent. | This Guide is met in practice and intent. More advanced assessment techniques are used as appropriate. |
| - | 1.110 | Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors. | Issued for comment | |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| 1 | 1.111 | Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors. | Applicable in Total. | |
| - | 1.112 | Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors. | Applicable in Total. | |
| 1 | 1.113 | Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I. | Applicable in Total. | |
| 1 | 1.114 | Guidance on Being Operator at the Controls of a Nuclear Power Plant. | Applicable in Total. | |
| 1 | 1.115 | Protection Against Low-Trajectory Turbine Missiles. | Applicable in Total. | |
| - | 1.116 | Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems. | Applicable in principle and intent. | See note 1. |
| 1 | 1.117 | Tornado Design Classification. | Applicable in Total. | |
| 2 | 1.118 | Periodic Testing of Electric Power and Protection Systems. | Applicable in Total. | |
| - | 1.119 | Surveillance Program for New Fuel Assembly Designs. | Withdrawn by NRC | |
| 1 | 1.120 | Fire Protection Guidelines for Nuclear Power Plants. | Issued for Comment. | |
| - | 1.121 | Bases for Plugging Degraded PWR Steam Generator Tubes. | Issued for Comment. | |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| 1 | 1.122 | Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components. | Applicable in Total. | |
| 1 | 1.123 | Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants. | Applicable in principle and intent. | See note 1. |
| 1 | 1.124 | Service Limits and Loading Combinations for Class 1 Linear Type Component Supports. | Applicable in principle and intent. | Stress limitations specified in the R.G. do not apply to high temperature operation and are appropriately adjusted. |
| 1 | 1.125 | Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants. | Applicable in Total. | |
| 1 | 1.126 | An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification. | The Regulatory Guide has no commonality with CRBRP design; therefore, application to CRBRP is inappropriate. | For LMFBR technology, experimental results show negative densification. |
| 1 | 1.127 | Inspection of Water-Control Structures Associated with Nuclear Power Plants. | Applicable in Total. | |
| 1 | 1.128 | Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants. | Applicable in Total. | |
| 1 | 1.129 | Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants. | Applicable in Total. | |
| 1 | 1.130 | Service Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports. | Applicable in Total. | |

TABLE 1 (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| - | 1.131 | Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants. | Issued for Comment | |
| 1 | 1.132 | Site Investigations for Foundations of Nuclear Power Plants. | Applicable in Total. | |
| 1 | 1.133 | Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors. | Under Project Review | |
| 1 | 1.134 | Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator License. | Applicable in Total. | |
| 1 | 1.135 | Normal Water Level and Discharge of Nuclear Power Plants. | Issued for Comment | |
| 1 | 1.136 | Material for Concrete Containments (Article CC-2000 of the "Code for Concrete Reactor Vessels and Containments. | Applicable in principle and intent with exceptions as noted. | See PSAR Section 3.8 |
| 1 | 1.137 | Fuel-Oil Systems for Standby Diesel Generators. | Applicable in Total. | |
| - | 1.138 | Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants. | Issued for Comment. | |
| 0 | 1.139 | Guidance for Residual Heat Removal. | Issued for Comment. | |
| 1 | 1.140 | Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants. | Applicable in principle aned intent. | CRBRP has incorporated requirements of Reg. Guide 1.52 |

TABLE I (Continued)

| REV. | NUMBER | TITLE | APPLICABILITY | JUSTIFICATION/EXPLANATION |
|---|---|---|---|---|
| - | 1.141 | Containment Isolation Provisions for Fluid Systems. | Issued for Comment. | |
| 1 | 1.142 | Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments). | Under Project Review. | |
| 1 | 1.143 | Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed In Light-Water-Cooled Nuclear Power Plants. | Applicable in Principle and Intent. | Calculated off-site releases of CRBRP, during any accident conditions, are well below 10 CFR 20 limits. |
| 1 | 1.144 | Auditing of Quality Assurance Programs for Nuclear Power Plants. | Applicable in principle and intent. | See note 1. |
| - | 1.145 | Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants. | Issued for Comment. | |
| - | 1.146 | Qualifications of Quality Assurance Program Audit Personnel for Nuclear Power Plants. | Applicable in principle and intent. | See note 1. |
| - | 1.147 | Inservice Inspection Code Case Acceptability-ASME Section XI Division 1 | Applicable in principle and intent. | Those portions of affected systems that function at high temperatures and in liquid metal environments have the inservice inspection programs designed for an LMFBR. |
| - | 1.148 | Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants. | Under Project Review | |
| - | 1.149 | Nuclear Power Plant Simulators for Use In Operating Training. | Under Project Review | |
| - | 1.150 | Ultrasonic Testing of Reactor Vessel Welding During Pre-Service and In-Service Examinations. | Under Project Review | |

Note 1:

The CRBRP Quality Assurance Program complies with the Quality Assurance criteria of the Federal Regulations as defined in 10CFR50, Appendix B, "Qualitly Assurance Requirements for Nuclear Power Plants and Fuel Reprocessing Plants." The Project's method of complying with 10CFR50, Appendix B is through the implementation of its overall Quality Assurance program developed in accordance with RDT F2-2, "Quality Assurance Program Requirements." The Program also complies with supporting standards RDT F1-2, "Preparation of System Design Descriptions," RDT F1-3, "Preparation of Unusual Occurrence Reports," and RDT F3-2, "Calibration Program Requirements."

The various NRC Regulatory Guides are not established as requirements for the CRBRP Quality Assurance Program. However, the program will accept practices that comply with these guides as fulfilling the like requirements of RDT F2-2 insofar as the requirements of RDT F2-2 are met. Maximum recognition will be made of quality assurance practices described in Regulatory Guides and other nationally recognized codes and standards as part of the implementation of the CRBRP Quality Assurance Program.

TABLE I

EVALUATION OF APPLICABILITIES OF EXISTING AEC
REGULATORY GUIDES TO THE CLINCH RIVER BREEDER REACTOR PLANT

| No. | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR BEING NOT APPLICABLE) |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.1 | Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (formerly Safety Guide 1) | 0.0 | 0.0 | (No equivalent system pumps in the CRBRP) |
| 1.2 | Thermal Shock to Reactor Pressure Vessels (formerly Safety Guide 2) | 0.0 | 0.0 | (No comparable emergency core cooling system, nor any large quantities of cold coolant injection involved on the CRBR) |
| 1.3 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors (Revision 1, 6/73, of Safety Guide 3) | 0.0 | 0.0 | A separate new guide for LMFBRs needs to be developed. Major changes required include:<br>1. Emphasis on loss of coolant accident is not applicable to the CRBRP.<br>2. Acceptable assumptions related to the accident release, taking into consideration the LMFBR characteristics as appropriate, and<br>3. Addition of provisions to allow credit for reduction in the amount of release available for leakage(s) due to plate-out and settling. |
| 1.4 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors (Revision 1, 6/73, of former Safety Guide 4) | 0.0 | 0.0 | Same as 1.3 above |
| 1.5 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (formerly Safety Guide 5) | 0.0 | 0.0 | (No comparable radiological consequences involved for a steam line break in the CRBRP) |
| 1.6 | Independence Between Redundant Standby (Onsite) Power Sources & Between Their Distribution Systems (formerly Safety Guide 6) | 100% | 100% | Consistent with the (Proposed) CRBRP, GDC 17. |
| 1.7 | Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident (formerly Safety Guide 7) | 0.0 | 0.0 | There is no zirconium-water reaction, nor containment spray reaction with metals in the CRBRP. Also, emphasis on loss of coolant accident is not applicable to the CRBRP.<br>However, need for monitoring of combustible gases is to be assessed. |

TABLE I (Cont'd)

| No. | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE) |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.8 | Personnel Selection and Training (formerly Safety Guide 8) | 100% | 100% | ANSI N18.1 equally applies to the CRBRP |
| 1.9 | Selection of Diesel Generator Set Capacity for Standby Power Supplies (formerly Safety Guide 9) | 100% | 100% | Intent consistent with the Proposed GDC 17 The detailed provisions are equally applicable to the CRBRP |
| 1.10 | Mechanical (Caldweld) Splices in Reinforcing Bars of Category I Concrete Structures (Revision 1, 1/2/73, of former Safety Guide 10) | 100% | 100% | This Guide is directly applicable.<br>The procedures set forth in this Guide for testing & sampling of mechanical splices in reinforcing bars are considered equally applicable to the Category I concrete structures of any nuclear power plant. |
| 1.11 | Instrument Lines Penetrating Primary Reactor Containment (formerly Safety Guide 11) | 100% | | The intent of this Guide is consistent with GDC 55 and GDC 56 of the Proposed CRBRP GDC.<br>However, for the current design selections of the CRBRP, there are no instrument lines penetrating the containment. |
| 1.12 | Instrumentation for Earthquakes (formerly Safety Guide 12) | 100% | 100% | The intent of this Guide is consistent with 10 CFR 50.36(c), which applies equally to any nuclear power plant.<br>The provisions set forth in this Guide relating to a suitable program for the seismic instrumentation required are considered equally applicable to the CRBRP as appropriate. |
| 1.13 | Fuel Storage Facility Design Basis (formerly Safety Guide 13) | 100% | 50% | The intent of this Guide is consistent with GDC 61 of the Proposed CRBRP GDC.<br>The detailed provisions of this Guide would be 90% applicable to an LMFBR plant using ex-containment water pool spent fuel storage. The only modification required would be related to Provision C.4 in that the inventory of radioactive materials available from leakage should be based on assumptions consistent with the characteristics of an LMFBR, rather than Regulatory Guide 1.25 (also see evaluation of Regulatory Guide 1.25 below).<br>The CRBRP is presently using an ex-containment sodium-cooled EVST design. Consequently the detailed provisions of this Guide is estimated to be about 50% applicable.<br>To make the Guide fully applicable to the CRBRP, appropriate changes are required to supplement and/or modify Provisions C.3, C.4 and C.8. |
| 1.14 | Reactor Coolant Pump Flywheel Integrity (formerly Safety Guide 14) | 0.0 | 0.0 | (This Guide is related to flywheels of reactor coolant pump motors in LWRs and is not applicable to the CRBRP.) |
| 1.15 | Testing of Reinforcing Bars for Category I Concrete Structures (Revision 1) 12/28/72, of former Safety Guide 15) | 100% | 100% | This Guide is wholly applicable to the CRBRP. |

TABLE I (Cont'd)

| No. | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE) |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.16 | Reporting of Operating Information (Revision 1, 10/73, of former Safety Guide 16) | 100% | 50% | This Guide is partially applicable to the CRBRP.<br>The changes required include the following:<br>1. The parameter list in Provision C.1.a.(3).(f) needs minor modification.<br>2. In Table 1, the report items related to "Fracture Toughness" and "Reactor Vessel Material Surveillance" need modification for full applicability to the CRBRP. This is due to the reason that both Appendices G and H to 10 CFR 50 may be not applicable or only partially applicable. This in turn depends on the materials selection for the vessel system which is not yet firm in certain areas. |
| 1.17 | Protection of Nuclear Plants Against Industrial Sabotage, (Revision 1, 6/73, of former Safety Guide 17) | 100% | 100% | This Guide is considered fully applicable to the CRBRP. |
| 1.18 | Structural Acceptance Test for Concrete Primary Reactor Containments (Revision 1, 12/28/72 of former Safety Guide 18) | 0.0 | 0.0 | The containment design selection is steel so that this is not applicable to the CRBRP. |
| 1.19 | Nondestructive Examination of Primary Containment Liner Welds (Revision 1, 8/11/72, of former Safety Guide 19) | 100% | 0.0 | For the bottom liner in the concrete base, ~~ASME-III~~, Division 2 provisions will be followed. (Note: This is so in order to be consistent with E-Spec) |
| 1.20 | Vibration Measurements on Reactor Internals (formerly Safety Guide 20) | 100% | 50% | The intent of this Guide is applicable, however the testing details given are not appropriate to LMFBR's. |
| 1.21 | Measuring & Reporting of Effluents from Nuclear Power Plants (formerly Safety Guide 21) | 100% | 75% | The intent of this Guide is equally applicable to the CRBRP.<br>The provisions in this Guide are only applicable to the CRBRP, where appropriate. |
| 1.22 | Periodic Testing of Protection System Actuation Function (formerly Safety Guide 22) | 100% | 100% | The intent of this Guide is consistent with the Proposed CRBRP GDC. |
| 1.23 | Onsite Meteorological Programs (formerly Safety Guide 23) | 100% | 95% | The intent and provisions of this Guide are considered generally applicable.<br>Although in the "Discussion" section of this Guide references are made to Safety Guides 3 and 4 which were prepared for LWRs, the detailed provisions as set forth in the "Regulatory Position" section of the Guide have no requirements strictly and exclusively based upon these two LWR guides. (Also see Regulatory Position C.6.d of this Guide.) |
| 1.24 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Gas Storage Tank Failure (formerly Safety Guide 24) | 100% | 0.0 | This Guide was specifically prepared for PWR plants, although the basic intent is considered generally applicable.<br>The detailed provisions are considered not applicable to the CRBRP. |
| 1.25 | Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling & Storage Facility for Boiling & Pressurized Water Reactors (formerly Safety Guide 25) | 50% | 0.0 | For applicability to LMFBRs, major changes in Provisions C.1 and C.3 of this Guide are needed.<br>Due to basic differences in fuel handling and storage designs between the CRBRP and the LWRs, the detailed provisions of the Guide are largely not applicable. |

TABLE I (Cont'd)

| No. | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE) |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.26 | Quality Control Classifications & Standards (formerly Safety Guide 26) | 100% | 25% | The intent of this Guide is equally applicable to the LMFBR plants. The detailed provisions of this Guide are basically not applicable to the CRBRP. This will be addressed in the PSAR per Section 3.2.2 of the SFAC. |
| 1.27 | Ultimate Heat Sink (formerly Safety Guide 27) | 100% | 100% | The intent of this Guide is considered generally applicable. Due to design differences, however, the detailed provisions of this Guide are applicable only where appropriate. |
| 1.28 | Quality Assurance Program Requirements (Design & Construction) (formerly Safety Guide 28) | 100% | 0.0 | This Guide is mainly to concur on the requirements as set forth in ANSI N45.2.11 (Draft No. 3, Rev. 1, July 1973). The intent is applicable. For the detailed provisions the CRBRP QA program will be followed. |
| 1.29 | Seismic Design Classification (Revision 1, 8/73, of former Safety Guide 29) | 100% | 50% | The basic intent of this Guide is equally applicable to the CRBRP. In their present version, the detailed provisions described in this Guide are not directly applicable to the CRBRP. This will be addressed in the PSAR per Section 3.2.1 of SFAC. |
| 1.30 | Quality Assurance Requirements for the Installation, Inspection, & Testing of Instrumentation & Electric Equipment (formerly Safety Guide 30) | 100% | 0.0 | The intent is applicable. For the detailed provisions, the CRBRP QA Program will be followed. |
| 1.31 | Control of Stainless Steel Welding (Revision 1, 6/73, of former Safety Guide 31) | 100% | 100% | Although this Guide was prepared for application to LWRs, it is equally applicable to the CRBRP. |
| 1.32 | Use of IEEE Std 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generation Stations" (formerly Safety Guide 32) | 100% | 100% | The intent and provisions of this Guide are equally applicable to the CRBRP, as appropriate. |
| 1.33 | Quality Assurance Program Requirements (Operation) (formerly Safety Guide 33) | 100% | 0.0 | The intent of this Guide is applicable. For the detailed provisions, the CRBRP QA Program will be followed. |

TABLE I (Cont'd)

| No. | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE) |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.34 | Control of Electroslag Weld Properties (12/28/72) | 100% | 100% | This Guide, describing an acceptable method for assuring materials control & control of special process related to fabricating electroslag welds for nuclear components, is equally applicable to the CRBRP.<br>Actual use of this Guide, however, is expected to be very limited, if any. One possible use is for the core support. It is anticipated that "Up-John" or "Subvert" will be the special process to be used on the CRBRP. |
| 1.35 | Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures (2/5/73) | 0.0 | 0.0 | (This Guide, relating to Prestressed Concrete Containment, is not applicable to the CRBRP.) |
| 1.36 | Nonmetallic Thermal Insulation for Austenitic Stainless Steel (2/23/73) | 100% | 50% | This Guide addresses the selection and use of nonmetallic Thermal insulation to minimize promotion of stress-corrosion cracking in the stainless steel portions of the reactor coolant boundary and other systems important to safety. Parts of the detailed provisions of the Guide are applicable where appropriate to the CRBRP. |
| 1.37 | Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (3/16/73) | 0.0 | 0.0 | In the context of "on-site cleaning" as intended by this Guide, the provisions set forth in ANSI N45.2.1-1973 which forms the basis of this Guide are not expected to be applicable to most of the liquid-metal systems of this plant.<br>At this point in time, it is anticipated that these fluid systems components will be cleaned, prior to installation, in the fabricator's shop. This shop cleaning may be water cleaning, and the requirements and control will be comparable to ANSI N45.2.1-1973. On site pre-operation cleaning, to which this Guide refers, if any, will be minimal and will be done by hand.<br>Because of the above reasons, this Guide is not rated. |
| 1.38 | Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, & Handling of Items for Water-Cooled Nuclear Power Plants (3/16/73) | 100% | 0.0 | The intent of this Guide is consistent with Appendix B to 10CFR50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants".<br>For the detailed provisions, the CRBRP QA Program will be followed. |

TABLE I (Cont'd)

| No. | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE) |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.39 | Housekeeping Requirements for Water-Cooled Nuclear Power Plants (3/16/73) | 100% | 0.0 | The intent of this Guide is consistent with Appendix B to 10 CFR 50.<br>For the detailed provisions, the CRBRP QA Program will be followed. |
| 1.40 | Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (3/16/73) | 100% | 25% | This Guide is intended mainly to concur on the requirements set forth in IEEE Std-334-1971, subject to additional provisions.<br>The basic intent of the guide is generally applicable. However, changes and supplements to IEEE Std-334-1971 appropriate to LMFBRs are needed in order to be applicable to the CRBRP. |
| 1.41 | Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments (3/16/73) | 100% | 100% | This Guide describes an acceptable method of verifying the proper assignments of redundant load groups to the related on-site power sources.<br>It is considered equally applicable to the CRBRP. |
| 1.42 | Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors (6/73) | 50% | 0.0 | The detailed provisions, developed primarily for LWR plants, do not apply to the CRBRP. |
| 1.43 | Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (5/73) | 100% | 100% | This Guide is related to selection and control of welding processes used for cladding ferritic steel components with austenitic stainless steel.<br>It is equally applicable to the CRBRP, as appropriate. |
| 1.44 | Control of the Use of Sensitized Stainless Steel (5/73) | 0.0 | 0.0 | The intent of this Guide relates to control of the application and processing of stainless steel to avoid severe sensitization that could lead to stress corrosion. It was developed primarily for LWRs.<br>For the S.S. materials to be used for the primary system components in the CRBRP, sensitization will occur. On the other hand, the high operating temperatures limit the use of materials of low carbon content.<br>The solution is therefore mainly to rely upon control for cleanliness and protection against contaminants. |

TABLE I (Cont'd)

| No. | TITLE | % RATING OF APPLICABILITY: INTENT | DETAILED PROVISIONS | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE) |
|---|---|---|---|---|
| 1.45 | Reactor Coolant Pressure Boundary Leakage Detection System (5/73) | 50% | 0.0 | The basic intent of this Guide is considered generally applicable, but the Guide was prepared to address the LWR coolant systems.<br><br>The detailed provisions of this Guide are largely not applicable to an LMFBR plant. |
| 1.46 | **Protection Against Pipe Whip Inside Containment (5/73)** | **100%** | 0.0 | **The basic intent of this Guide is considered generally applicable.**<br><br>The detailed provisions of this Guide, however, was developed primarily for LWR plants. |
| 1.47 | Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (5/73) | 100% | 100% | This Guide is considered equally applicable to CRBRP. |
| 1.48 | Design Limits and Loading Combinations for Seismic Category I Fluid System Components (5/73) | 100% | 50% | The basic intent of delineating acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents and specified seismic events for the design of Seismic Category I fluid system components is considered generally applicable to all nuclear power plants.<br><br>The detailed provisions of this Guide were developed primarily for LWR plants. They need to be supplemented and/or modified for direct application to the CRBRP. |
| 1.49 | Power Levels of Nuclear Power Plants (Revision 1, 12/73) | 100% | 100% | This Guide is generally applicable. (It should be noted that, due to the projected power levels of this plant, this Guide has no impact on the CRBRP.) |
| 1.50 | Control of Preheat Temperature for Welding of Low-Alloy Steel (5/73) | 100% | **95%** | **This Guide is considered applicable to ASME Section III, Class 1 components. The provisions of this Guide are considered applicable to the CRBRP with the following exception:**<br>**Regulatory Position C.2 requires that the preheat temperature for production welds be maintained until a postweld heat treatment has been performed. This will be complied with whenever practicable or required by RDT E15-2NB-T, unless the need and acceptability of an alternate procedure has been demonstrated.** |
| 1.51 | Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components (5/73) | -- | -- | **(This Guide has been withdrawn by the NRC).** |

25

TABLE I (Cont'd)

| No. | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE) |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.52 | Design, Testing, & Maintenance Criteria for Atmosphere Clean-up System Air Filtration and absorption Units of Light-Water-Cooled Nuclear Power Plants (6/73) | 100% | 100% | |
| 1.53 | Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (6/73) | 100% | 100% | This Guide is considered applicable to CRBRP. |
| 1.54 | Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants (6/73) | 100% | 0.0 | The intent of this Guide is considered applicable. For the detailed provisions, the CRBRP QA Program will be followed. |
| 1.55 | Concrete Placement in Category I Structures (6/73) | 100% | 100% | This Guide is considered equally applicable to any nuclear power plant. |
| 1.56 | Maintenance of Water Purity in Boiling Water Reactors (6/73) | 0.0 | 0.0 | (This Guide was developed for BWRs and is not applicable to the CRBRP.) |
| 1.57 | Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (6/73) | 0.0 | 0.0 | This Guide was specifically prepared for and limited to those LWR plants of which the containment system comprises a metal containment that is completely enclosed within a Seismic Category I structure (e.g.) a concrete shield building). It is, therefore, generally applicable to those plants which use this particular type of containment system.<br>Due to containment selection, this Guide is not rated as it is not applicable. |
| 1.58 | Qualification of Nuclear Power Plant Inspection, Examination, & Testing Personnel (8/73) | 100% | 0.0 | The intent of this Guide is considered applicable. For the detailed provisions, the CRBRP QA Program will be followed. |
| 1.59 | Design Basis Floods for Nuclear Power Plants (8/73) | 100% | 100% | This Guide is equally applicable to CRBRP, as appropriate. |
| 1.60 | Design Response Spectra for Seismic Design of Nuclear Power Plants (Revision 1, 12/73) | 100% | 100% | This Guide is considered equally applicable to CRBRP, as appropriate. |
| 1.61 | Damping Values for Seismic Design of Nuclear Power Plants (10/73) | 100% | 100% | This Guide is equally applicable to CRBRP, as appropriate. |
| 1.52 | Manual Initiation of Protective Actions (10/73) | 100% | 100% | This Guide describes an acceptable method for complying with the requirements of IEEE Std 279-1971 (Section 4.17). It is considered equally applicable to the CRBRP. |
| 1.63 | Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants (10/73) | 100% | 100% | This Guide concurs with IEEE Std 317-1972 and supplements it with four additional provisions.<br>It is considered equally applicable to CRBRP, as appropriate. |
| 1.64 | Quality Assurance Program Requirements for the Design of Nuclear Power Plants (10/73) | 100% | 0.0 | The intent of this Guide is considered applicable. For the detailed provisions, the CRBRP QA Program will be followed. |

TABLE I (Cont'd)

| No. | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.65 | Materials & Inspection for Reactor Vessel Closure Studs (10/73) | 0.0 | 0.0 | This Guide was prepared primarily for LWRs.<br>Due to differences in loading characteristics, it is considered essentially not directly applicable to the CRBRP. |
| 1.66 | Nondestructive Examination of Tubular Products (10/73) | 50% | 50% | This Guide was developed and intended primarily for application to tubular products used for ASME-III Code Class 1 components on LWRs.<br>The corresponding CRBRP components will be of austenitic steel. The state-of-the-art of the UT examination, as specified by the Guide, has not been capable of producing meaningful results. The CRBRP, however, is anticipated to meet the requirements as set forth in NB-2550 of ASME-III for the examination addressed by the Guide. |
| 1.67 | Installation of Over-Pressure Protection Devices (10/73) | 100% | 50% | Code Case 1569, which forms the basis of this Guide, has covered four categories. Only the open systems, however, are treated in detail. Closed discharge systems are essentially left undefined.<br>According to the selected design of the CRBRP at this time, the Guide is expected to be applicable only in the design of steam line safety valves. The Guide is therefore considered as partially applicable to the CRBRP in terms of the detailed provisions. |
| 1.68 | Preoperational & Initial Start-up Test Programs for Water-Cooled Power Reactors (11/73) | 50% | 25% | This Guide was developed primarily for LWR plants.<br>In order to properly cover the LMFBR plants, the detailed provisions of this Guide need to be supplemented and modified by taking into consideration characteristics of LMFBR plants.<br>Specifically, this includes modifications of and supplements to appropriate items included in Appendices A and C to this Guide. |
| 1.69 | Concrete Radiation Shields for Nuclear Power Plants (1/74) | 100% | 100% | This Guide is considered applicable to CRBRP. |
| 1.70 | Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LMFBR Edition | --- | --- | (See Note 1.) |
| 1.71 | Welder Qualification for Limited Accessibility Areas (1/74) | 100% | 100% | This Guide relates to control of welding for nuclear components and is considered generally applicable. |
| 1.72 | Spray Pond Plastic Piping (1/74) | 0.0 | 0.0 | There will be no spray pond in the CRBRP. |

TABLE I (Cont'd)

| No. | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE) |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.73 | Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (1/74) | 100% | 75% | This Guide is mainly based upon IEEE Std. 382-1972 and is considered equally applicable to any nuclear power plant, where appropriate.<br>In order to be properly applicable to LMFBRs. modifications and supplements to IEEE Std. 382-1972 appropriate to LMFBRs are required. |
| 1.74 | Quality Assurance Terms and Definitions | 100% | 0.0 | The intent of this Guide is applicable. |
| 1.75 | Physical Independence of Electric Systems (1/75) | 100% | 75% | This Guide is mainly intended to concur with the requirements set forth in IEEE-384 (1974), subject to 16 exceptions.<br>With regard to IEEE-384 Sections 5.1.1.1, 5.1.3, and 5.7 requirements, the CRBRP will conform with IEEE - 384.<br>Regulatory Position C.1 (on IEEE-384, Section 3) is still being assessed. |
| 1.76 | Design Basis Tornado for Nuclear Power Plants | 100% | 100% | This Guide describes design basis tornadoes for nuclear power plants, acceptable to the Regulatory for three regions within the contiguous United States.<br>It is generally applicable and is applicable to the CRBRP as appropriate. |
| 1.77 | Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors | 0.0 | 0.0 | This Guide was specifically prepared for PWR plants in regard to acceptable analytical methods and assumptions that may be used in evaluating the consequences of a rod ejection accident in uranium oxide fueled cores.<br>It is not applicable to the CRBRP. |
| 1.78 | Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release | 100% | 50% | This Guide describes acceptable assumptions and criteria to be used in the evaluation of control room habitability during and after a postulated hazardous chemical release. Requirements of the Guide are dependent upon actual or projected presence of certain specified chemicals within five miles of the plant or in frequent transit within the same distance.<br>Preliminary design of the CRBRP control room habitability system has been assessed for a hypothetical and most limiting radiological consequence. Chemical toxicity will be assessed. |
| 1.79 | Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors | 0.0 | 0.0 | This Guide was specifically prepared for PWR plants in regard to acceptable preoperational testing programs for ECCs.<br>It is not applicable to the CRBRP. |
| 1.80 | Preoperational Testing of Instrument Air Systems | --- | --- | This Guide describes an acceptable preoperational testing program for verifying the operability of safety-related instrument air system.<br>On the CRBRP, except those portions penetrating the containment and being considered as parts and appurtenance thereof, safety-related instrument air system parts are yet to be identified |

TABLE I - (Cont'd)

| No. | TITLE | %RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDNEITFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 1.81 | Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants | 0.0 | 0.0 | This Guide addresses the USAEC's requirements with regard to the sharing of onsite emergency and shutdown electric systems for multi-unit nuclear power plants.<br>It is not applicable to the CRBRP. |
| 1.82 | Sumps for Emergency Core Cooling and Containment Spray Systems | 0.0 | 0.0 | This Guide applies to PWRs only.<br>It is not applicable to the CRBRP. |
| 1.83 | Inservice Inspection of Presurized Water Reactor Steam Generator Tubes | 0.0 | 0.0 | This Guide applies only to PWRs.<br>It is not applicable to the CRBRP. |
| 1.84 | Code Case Acceptability ASME Section III Design and Fabrication (4/75) | 100% | 0.0 | This Guide was prepared specifically for LWR plants. A separate Guide addressing LMFBR plants appears to be desirable. |
| 1.85 | Code Case Acceptability ASME Section III Materials (4/75) | 100% | 0.0 | Same as for Guide 1.84. |
| 1.86 | Termination of Operating Licenses for Nuclear Reactors (6/74) | 100% | 95% | The intent of this Guide is equally applicable to LMFBR plants.<br>The detailed provisions of this Guide are essentially fully applicable to LMFBRs. |
| 1.87 | Construction Criteria for Class 1 Components in Elevated Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595 and 1596) (6/74) | 100% | 100% | This Guide, 1.87 (6/74,Rev.0) is considered applicable to the CRBRP. Detailed applications of this Guide are discussed in Chapter 5.0 of this PSAR. |
| 1.88 | Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (8/74) | 100% | 0.0 | The intent of this Guide is applicable.<br>For the detailed provisions, the CRBRP QA program will be followed. |
| 1.89 | Qualification of Class IE Equipment for Nuclear Power Plants (11/74) | 100% | 50% | The intent of this Guide is equally applicable to LMFBR plants.<br>The detailed provisions of this Guide will have to be expanded to include LMFBRs considerations in the radiological source terms. |
| 1.90 | Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons (11/74) | 100% | 0.0 | The intent is applicable. The detailed provisions may also apply to LMFBRs; but it is not applicable to the CRBRP, due to design selection. |
| 1.91 | Evaluation of Explosions Postulated to Occur on Transportation Routes Near the Nuclear Power Plant Sites (1/75) | 100% | 100% | This Guide is considered generally applicable to all types of nuclear power plants, as appropriate. |
| 1.92 | Combination of Modes and Spatial Components in Seismic Response Analysis (12/74) | 100% | 100% | The intent and detailed provisions of this Guide are considered equally applicable to all types of nuclear power plants.<br>It should be noted that detailed seismic analysis procedures are provided in PSAR Section 3.7 for the CRBRP. |

41

1

TABLE 1 - (Cont'd)

| No | TITLE | % RATING OF APPLICABILITY | | REASONS FOR APPLICABILITY AND/OR IDENTIFICATIONS OF CHANGES REQUIRED (OR REASONS FOR NOT BEING APPLICABLE |
|---|---|---|---|---|
| | | INTENT | DETAILED PROVISIONS | |
| 93 | Availability of Electric Power Sources (12/74) | 95% | --- | The intent of this Guide is considered generally applicable except for the reference to LOCA, which is not applicable to CRBRP.<br><br>The System proposed for the CRBRP has a higher redundancy than the one assumed and used as the basis in the Guide. In addition, the consequences of electric power failures are different from those of LWRs. The detailed provisions of this Guide, and its applicability to CRBRP, are currently being evaluated. |
| 94 | Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (4/75) | 100% | 0.0 | The intent of this Guide is generally applicable<br><br>For the detailed provisions, the CRBRP QA Program will be followed. |
| 95 | Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release (2/75) | 100% | 100% | The Guide is considered generally applicable, as appropriate. |
| 96 | Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants (5/75) | 0.0 | 0.0 | (This Guide, intended for BWRs only, is not applicable to the CRBRP.) |

1

Note (1):

At this review, the Commission has issued 34 Guides in the 1.70.XX series. Additional guides in this series are still being issued. These Guides, so far issued, have been prepared primarily for the LWR Edition of the SFAC. Applicabilities of these Guides to the LMFBR Edition of the SFAC are being assessed. The applicant will report on the evaluation at a proper stage in a later amendment.

## 1.2 GENERAL PLANT DESCRIPTION

The Clinch River Breeder Reactor Plant is a liquid sodium cooled fast breeder reactor nuclear power plant. The major systems of the plant are the Reactor, Heat Transport and related systems, Steam Generator and related systems, Turbine/Generator and related systems, Fuel Handling System, Power Transmission and Plant Electrical System, Auxiliary Systems, and Instrumentation and Control System. The major plant operational parameters are given below.

| | |
|---|---|
| Thermal Output | 975 MW (rated) |
| Electrical Output | 380 MW Gross (rated) |
| | 350 MW Net |
| Steam Production | $3.34 \times 10^6 \frac{lbs.}{hr.}$ |
| Steam Temperature at Turbine | 900°F |
| Steam Pressure at Turbine | 1450 psi |
| Total Primary Coolant Flow Rate | $41.5 \times 10^6 \frac{lbs.}{hr.}$ |
| Breeding Ratio (initial cycle) | 1.29 |
| (equilibrium core) | 1.24 |
| Average Burnup (initial core) | 50,000 MWD/T |
| (future core) | 80,000 MWD/T |

51 The thermal hydraulic design parameters for the plant are based on a thermal power rating of 975 MW. The recommended design margins create a possibility that the plant may attain a rated thermal power of 1121 MWt without exceeding the system design basis values. Accordingly, structural analysis for all non-replaceable systems is based on the 1121 MWt rating. The plant will initially be licensed for 975 MWt.

### 1.2.1 Site

The Clinch River site is on a peninsula bounded on the south by the Clinch River and on the north by the AEC Oak Ridge Reservation and within the city limits of Oak Ridge, Tennessee. The 1364 acre site is owned by the United States Government and is in the custody of the Tennessee Valley Authority. The site is 12 miles southwest of downtown Oak Ridge and 25 miles west of Knoxville, Tennessee. The ground-level on which the CRBRP will be situated is 815 feet above sea level, 74 feet above the mean Clinch River water level of 741 feet. Figure 1.2-1 is an artist conception of the plant location.

The Reactor Confinement/Containment Buildings are centrally located in a complex of major plant buildings. The complex is enclosed by a paved road. The cooling tower and circulating water pumphouse are located approximately 700 feet east of the reactor building. The power transmission facilities are located approximately 900 feet northeast of the reactor building. The riverwater pumphouse is on the Clinch River northeast of the plant complex. Highway and railroad access will be from the North.

### 1.2.2 Engineered Safety Features

The CRBRP design includes engineered safety features that are provided to mitigate the consequences of postulated accidents. These features are discussed in Chapter 6.0. Examples of such features are: Containment/Confinement Systems, Reactor Guard Vessel, Guard Vessels for Primary Heat Transport System major components, Steam Generator Auxiliary Heat Removal System (SGAHRS) and Habitability Systems.

### 1.2.3 Reactor, Heat Transport and Related Systems

A system of three identically configured piped circuits transport heat from the reactor, through primary and intermediate sodium loops, to steam generator modules which produce steam for the turbine. The three loops are independent with only the Primary Heat Transport System (PHTS) having common flow paths through the reactor (See Figure 1.2-2). The PHTS removes the heat generated in the fuel assemblies, blanket assemblies, control rod assemblies, and structural elements. Each of the three independent Intermediate Heat Transport Systems (IHTS) receive heat from the PHTS through an Intermediate Heat Exchanger. The IHTS transfers heat outside containment with non-radioactive sodium. The Intermediate Heat Exchanger acts as a barrier for the transfer of radioactive materials between the PHTS and IHTS. The IHTS is maintained at higher pressure than the PHTS to inhibit leakage from the radioactive PHTS into the non-radioactive IHTS. Each primary loop contains a hot leg pump, an intermediate heat exchanger, a cold leg check valve and interconnecting piping between the above-mentioned components and the reactor vessel inlet and outlet nozzles. Each intermediate loop has a cold leg pump, intermediate sodium expansion tank and interconnecting piping to transport the sodium from the tube side of the IHX to the superheater inlet and from the evaporator outlets back to the IHX tube side inlet (Reference Chapter 5 for details of the Heat Transport System).

The Primary Heat Transport System (PHTS) piping and components are located in cells within the containment building. The components and piping for each loop are located within three vaults (cells) in the containment building: (1) an HTS cell which contains all of the major loop components; (2) an IHTS pipeway; and (3) the reactor cavity which houses the reactor vessel and the associated primary loop piping. The cells are separated from each other by concrete shielding walls and are inerted with nitrogen which is circulated for cooling. Those parts of the PHTS equipment which come in contact with sodium are located in a nitrogen atmosphere below the level of the containment building operating floor. Each HTS cell has a separate atmosphere and the reactor cavity and the HTS pipeways have a common atmosphere. The pump drive systems (motors, speed controllers, and heating and seal assemblies) are located in an air environment above the operating floor. Separation of the equipment cells provides the capability of deinerting individual vaults for independent access for maintenance or inspections.

The reactor has 156 fuel assemblies, 82 inner blanket assemblies, 126 radial blanket assemblies and 15 control assemblies (9 primary assemblies and 6 secondary assemblies). The reactor fuel assemblies are about 14 feet long, with an active core height of 36 inches, upper and lower axial blankets of 14 inches each, and a fission gas plenum of 48 inches. Each fuel assembly contains 217 stainless steel clad fuel pins. Each blanket assembly contains 61 stainless steel clad fuel pins. The fuel in the active core is mixed oxides of plutonium and uranium ($PuO_2/UO_2$). The blanket rods are 116.5 inches long with 64 inches of depleted uranium oxide pellets and a 48 inch long plenum. The control rod absorber material is enriched boron carbide ($B_4C$). Each primary control assembly contains 37 absorber pins and each secondary assembly contains 31 pins. The core is designed for annual refueling. The coolant flow is upwards through the core. The free sodium surface in the upper plenum is covered by argon.

The reactor is located in a stainless steel reactor vessel of nominal inside diameter 20'3", and 58'8" high from the bottom of the vessel to the top of the support ring. The vessel is provided with a closure head designed to accommodate through-the-head refueling (See Figure 1.2-3). The reactor vessel, IHX, and primary sodium pumps are enclosed by free-standing, structurally independent guard vessels. Chapter 4 of this PSAR provides a thorough description of the reactor.

### 1.2.4 Steam Generator - Turbine and Related Systems

The Steam Generator System provides independent steam generation capability for each of the three reactor heat transport loops. Steam, combined from all three loops, supplies the single turbine-generator. The Steam Generator System is of a modular forced recirculation configuration. The recirculation ratio is 2 to 1. Each of the three independent loops consists of the following:

Steam Generator Evaporator/Superheater Modules
Feedwater System
Sodium-Water Reaction Pressure Relief System

Water Isolation and Dump System
Leak Detection System

The steam generator evaporator/superheater is of the shell and tube type. Sodium flow is on the shell side, counter-flow to the water/steam flow on the tube side.

The Feedwater System supplies feedwater to the steam drum where it mixes with and subcools the saturated water from the evaporators. The subcooled water flows through the tube side of the two evaporators where it is partially vaporized by the higher temperature sodium flowing on the shell side. The steam/water mixture then flows to the steam drum where the steam is separated and the water continues to recirculate. Entrained moisture in the steam is removed by dryers and separators, internal to the drum. The dry saturated steam is superheated, to the desired temperature, in the tube side of the superheaters. The superheated steam flows to main steam headers and then to the turbine.

The Sodium-Water Reactor Pressure Relief System is a passive system which only becomes operational in the event of a steam tube leak within an evaporator or superheater module large enough to cause a rapid pressure rise due to the sodium/water reaction. In the event of a large sodium-water reaction, the system protects the sodium side of the evaporator and superheater modules, the IHTS and the IHX from over-pressure by the use of rupture discs on the piping adjacent to the modules. Sodium and/or sodium/water reaction products expelled through the rupture discs are directed by the Sodium-Water Reaction Relief System piping to a separation tank where gross separation of liquid, solid and gaseous products takes place. Gaseous reaction products (primarily hydrogen) then flow through a centrifugal separator where additional separation takes place. The gaseous reaction products are then vented via a flare stack to the atmosphere. The flare stack burns any hydrogen that may be present in the gas.

In order to reduce the amount of water which may be admitted to the IHTS in the event of a large sodium/water reaction in an evaporator module, blowdown of the evaporator modules through power relief/safety valves is accelerated by the Water Dump System. Quick opening water dump valves are located at the inlet to each evaporator module. Water dump piping directs the water/steam to a water dump tank.

In addition to the above, in the event of a large sodium/water reaction, sodium dump capability is provided for the Intermediate Heat Transport System (IHTS) and the sodium side of the Steam Generator System. The sodium can be drained rapidly to a sodium dump tank.

The Steam Generator Leak Detection System monitors: (1) sodium exiting from each evaporator and superheater.

Monitoring is by hydrogen and oxygen meters which provide a measure of the hydrogen and oxygen concentration levels in the sodium. In the event of a water-to-sodium leak, changes in the hydrogen and oxygen concentration levels are detected and off-normal conditions are annunciated.

The Steam Generator Auxiliary Heat Removal System (SGAHRS) removes reactor-generated heat following reactor shutdown or trip when the main NSSS heat sink or normal feedwater supply is unavailable. The heat is initially rejected to the environment through a direct steam dump and by condenser tubes heating atmospheric air for long term heat removal. The SGAHRS can also function to remove shutdown heat loads for refueling and other long term outages.

A Direct Heat Removal Service (DHRS) is also provided to remove decay heat. In the remote event that the steam generator decay heat removal paths are not available, the reactor decay heat will be dumped through the reactor overflow system to an overflow heat exchanger. From the heat exchanger the heat flows through the ex-vessel storage tank (EVST) NaK system to air blast heat exchangers where the heat is transferred to the atmosphere (See Section 5.6.2 for details of the DHRS).

### 1.2.5 Offsite and Onsite Power

The Offsite Power Systems deliver the power to and from the site and include transformers, switchgear, structures, overhead, and underground conduit. Included are devices by which the main generator is connected or isolated from the TVA distribution grid. In addition, this system provides the unit station service transformers, reserve transformers, and related primary side switchgear through which the station auxiliary loads are supplied power from the power grid.

The Systems distribute and control the electrical energy for the site. The systems interface with the Offsite Power System at the secondary terminals of the unit station service and reserve transformers. The Onsite Power Systems provide the following functions:

a. Receive power from the offsite power supplies and transforms the voltage to the utilization levels of the Nuclear Island, building, lighting and site service loads;

b. Provide diesel generators for standby power; and batteries and invertors for vital AC and DC power;

c. Provide cable, conduit, raceway, and shielded penetration systems for interconnecting wiring for electrical power and control, instrumentation, lighting and communication;

d. Provide control and interlocking operations when these functions are not provided by other systems.

The Offsite and Onsite Power Systems are discussed in detail in Chapter 8.

### 1.2.6 Instrumentation, Control and Protection

The plant control system integrates the various plant instrumentation and control systems into a coordinated, centralized control room and local control stations (as applicable), for the purpose of providing safe, effective and economic plant operation. The system also provides the control and instrumentation circuits and devices required to provide overall plant control which automatically maintains essentially constant steam temperature and pressure at the turbine inlet over the 40 to 100 percent load range of operation. The plant control system is designed to provide load follow capability as required in practical utility operation; that is, maintaining plant parameters within scheduled variations such that near constant temperature and pressure are maintained at the turbine inlet and rates-of-temperature change are minimized in system components during both plant steady-state and load change periods. Refer to Chapter 7 for complete description of these systems.

This system interfaces with the reactor and heat transport system through their instrumentation systems. Thus the plant control system utilizes signal inputs and control equipment provided by the reactor system, the heat transport system, the reactor and vessel instrumentation systems, fuel failure monitoring system, flux monitoring system, radiation monitoring system as well as the building electrical power system, power transmission system, auxiliary power system, steam turbine system, and data handling and display system.

The Plant Protection System (PPS) assures that the results of postulated plant fault conditions do not exceed the specified limits for the fuel or release of radioactivity by initiating reactor trip, sodium pump trip, turbine generator trip, containment isolation, or decay heat removal. The PPS includes the shutdown system, containment isolation system and interfaces with the shutdown heat removal system. The PPS does
41 not require the reactor operator or control system to implement a protective action.

The shutdown system consists of two independent diverse systems, either of which is capable of reactor shutdown. The primary shutdown system is configured using a local coincidence logic arrangement while the secondary shutdown system is arranged using a general coincidence logic. These logics are described in Section 7.2.1. Primary and secondary systems are electrically and mechanically isolated. Sufficient redundancy is included within each system to assure that single random failures will not degrade protection by either system.

### 1.2.7 Auxiliary Systems

The Auxiliary Liquid Metal System provides the facilities for receipt, storage and purification of all liquid metal used in the CRBRP. It also provides the capability for controlling reactor sodium level variations, accommodates primary sodium volumetric changes, provides cooling for the core components stored in the Ex-Vessel Storage Tank (EVST), and by means of the Direct Heat Removal Service (DHRS) gives a means of long term reactor decay heat removal that is independent of the
26 intermediate heat transport system and steam generator system loops.

The Compressed Gas System processes ambient air to provide compressed dry air for pneumatic instruments, maintenance systems, unloading devices, tooling, and miscellaneous cleaning and inspection services. This system provides for sodium removal systems and as required for plant usage.

The Recirculating Gas Cooling System provides cooling service to cells and equipment located in the Reactor Containment Building and the Reactor Service Building.

15 The Chilled Water Systems provide heat removal capability from certain equipment and areas in the Reactor Containment Building and the Reactor Service Building.

The Inert Gas Receiving and Processing System (IGRPS) provides inert gases as required by other systems of the CRBRP, including cover gas, cell inerting atmosphere, valve actuation gas in inerted cells, cooling
59 gas, gas for certain seals, for component cleaning and other services, and vacuum for out-gassing and gas-collection purposes. In addition, the IGRP System provides for the control of reactor cover gas radioactivity and for the processing of gases to be released from the system to remove their contained radioactivity.

The Impurity Monitoring and Analysis System provides for the sampling, monitoring, and analysis of the sodium, NaK, and argon cover gas systems in the plant, and acceptance sampling and analysis of incoming
56 sodium, NaK, argon, and nitrogen.

The Treated Water System includes the domestic (potable) water system, the closed cooling water system, water (makeup) treatment system and the cooling water makeup system.

The River Water Service System handles and treats river water for the plant. The system includes the river water pumps and piping, intake filtration equipment and the plant service water system.

The Heat Rejection System provides the heat sink using the main cooling tower
for waste heat loads from the turbine condensers, and from the various plant
1 auxiliary and service systems such as sodium pump oil coolers, air conditions,
44 air compressors, pump coolers and the turbine oil coolers. The Emergency Plant
Service Water System emergency cooling tower structure provides the heat sink
for the safety related components listed in Table 9.9-3. Details of the
33 auxiliary system are given in Chapter 9.

1.2.8 <u>Refueling System</u>

The Reactor Core is designed to be refueled annually. Under equilibrium
conditions, all fuel and inner blanket assemblies are replaced as a batch every
two years, with a planned mid-term interchange of 6 inner blanket assemblies
41 20 for 6 fresh fuel assemblies designed to add sufficient excess reactivity to the
system to complete the (550 fpd) burnup. The radial blanket assemblies in the
59 first and second rows are replaced as a batch at 4 and 5 year intervals,
51 respectively.

The In-Vessel Handling Subsystem (IVHS) provides for the transfer of core
44 assemblies in the reactor vessel, between their normal positions in the reactor
41 core and the storage positions outside the core accessible by the Ex-Vessel
Transfer Machine. The major equipment comprising the IVHS are the In-Vessel
Transfer Machine (IVTM), Auxiliary Handling Machine (AHM), AHM Floor Valves
(FV), IVTM Port Adaptors, and associated maintenance and storage facilities and
equipment. The IVTM is installed in the small rotating plug in the reactor
44 head after reactor shutdown. The machine raises or lowers core assemblies by
means of a grapple. Translation to a new position is by rotation of the
41 44 reactor head rotatable plugs. The AHM is used to install and remove the
control rod drivelines, port plugs, and in-vessel section of the IVTM in the
reactor. The port adaptors and floor valves provide a means for closure of the
61 reactor and storage ports during the transfer of refueling equipment in
59 preparation for refueling operation.

The Ex-Vessel Handling Subsystem (EVHS) provides for the transfer of core
assemblies between the reactor, the Ex-Vessel Storage Tank (EVST), and the Fuel
41 44 Handling Cell (FHC) located in the Reactor Service Building (RSB). The system
consists of the Ex-Vessel Transfer Machine (EVTM) mounted on a Gantry-Trolley
44 (G-T), EVTM Floor Valves (FV), Core Component Pots (CCP), port plugs and
adaptors, and associated maintenance and storage equipment and facilities.

59 The Ex-Vessel Storage Subsystem (EVSS) consists of the Ex-Vessel Storage Tank
(EVST). The EVST is a sodium-filled tank used to store and cool spent fuel
44 prior to shipment offsite, and preheat new core assemblies. The capacity of
41 the EVST is about 650 assemblies.

41 44

20

The Conditioning and Service Subsystem (CSS) and Receiving and
Shipping Subsystem (RSS) consist primarily of the facilities necessary to un-
load, inspect and prepare the new core assemblies prior to loading in the
44 EVST; and to handle, inspect and load the spent core assemblies in shipping
41 casks for shipment off-site. The facilities include a Fuel Handling Cell
44 (FHC) which is a shielded inerted hot cell. The major equipment consists of
20 cask handling and transporting machinery.
59

Amend. 59
Dec. 1980

20

The Refueling System is discussed in detail in Section 9.1.

### 1.2.9 Radwaste Disposal System

The Radwaste Disposal System (RWDS) provides all equipment necessary to collect, process, store, monitor and dispose of liquid and solid radioactive waste. The RWDS consists of two subsystems, the Liquid Radwaste System (LRS) and the Solid Radwaste System (SRS). (Radioactive gases are processed by the Inert Gas Receiving and Processing System as discussed in Section 1.2.7).

The LRS consists of two liquid waste concentration and distillation flow processes. One system is designed for processing intermediate activity level liquid containing tritium; the other is designed for processing low activity level liquid with little or no tritium. The design is such that evaporators are used to decontaminate the input streams for either level liquid waste system. The intermediate activity level liquids may consist of water or neutralized acids from the Large Component Cleaning Cell. The liquids are processed in separate batches following the infrequent maintenance operations.

The SRS collects dry compactible waste, disposable solid waste
such as filters, scrap metal and support tools, and solid waste containing
induced radioactivity. In addition, solid materials contaminated with
fission and corrosion products and radioactive sodium are collected.
The solid waste process system provides a solidification station for
liquid waste, a compacting station for dry compactible waste, a storage
area, and loadout station. Unusually high activity liquids or solid
components will be transferred to DOT licensed shipping containers
available from commercial firms licensed by the NRC for transport and for
41 transfer to NRC licensed processing or burial sites. Solid components
contaminated with radioactive sodium and surface deposited activity will
be treated and shipped to special processing centers for disposal. This
is discussed in detail in Chapter 11.

### 1.2.10 Reactor Confinement/Containment System

The Reactor Confinement/Containment System provides a protective boun-
18 dary between the plant and the surrounding environment in the event of a
serious radioactive release. The containment consists of a steel pressure
vessel with inner and outer concrete shields below the operating floor.
The free standing steel vessel is cylindrical in shape with an ellipsoidal
spherical top and flat bottom. The cylindrical steel vessel is approximately
1.5 inches thick, 186 feet inner diameter with the ellipsoidal spherical top 158.3
feet above the operating floor. The containment houses the reactor vessel
32 and the primary heat transport system components, including the

intermediate heat exchanger, primary pumps, primary piping, sodium overflow tank, portions of the compartment inerting and primary cover gas systems, one primary sodium storage tank, and some of the fuel handling
39 equipment. (Reference Figures 1.2-4 through 1.2-20).
The building provides space and facilities necessary for the cleaning of sodium wetted equipment housed within the containment structure. The cell structure arrangement provides shielding for protection of equipment and personnel, prevents damage to adjacent equipment, and facilitates maintenance activities, fire-fighting and cleanup. The containment system provides protection from natural phenomena, including seismic events, for systems required for safe operation and shutdown of the reactor, and provides isolation of redundant systems required for safe operation and shutdown of the reactor.

The Reactor Containment Building will have a maximum leakage rate of 0.1% volume per day at an internal pressure of 10 psig, whereas the maximum calculated internal pressure under any condition is approximately 2 psig.

The Reactor Containment Building is surrounded by a low leakage concrete confinement structure, with an annulus space separating the two structures. The annular space between the containment and confinement is maintained at a negative pressure relative to atmospheric pressure during normal operation and exhausted through high efficiency filters should an accident occur. The Concrete confinement will be designed to meet tornado missile and Seismic Category 1 Criteria. A discussion of the Reactor Confinement/Containment is provided in Section 6.2.

18

### 1.2.11 Major Structures

The major buildings of the CRBRP are the Reactor Confinement/Containment
18 Building, Reactor Service Building, Steam Generator Building, Control Building and Diesel Generator Building. These buildings are discussed in detail in Section 6.1 and 3.A. The general layout of these buildings
39 is shown in Figure 1.2- 4.

18 a. Reactor Confinement/Containment Building-
This building houses the reactor and primary heat transport system and is discussed in the previous section.

b. Reactor Service Building (RSB) -
This building is designed to house portions of several auxiliary systems,
39 reactor refueling system, and maintenance systems associated with operation of the Nuclear Island(Reference
39 Figures 1.2-21 through 1.2-38).

c. Steam Generator Building (SGB) -
This building houses the major components for the Steam Generator System, the Intermediate Heat Transport System, the Auxiliary Heat Removal System, the Sodium-Water Reaction Pressure Relief System, and primary sodium storage. (Reference Figures 1.2-39 through 1.2-59).

39 d. Control Building -
This building houses the main control room which contains both Nuclear Steam Supply System and the BOP controls, the life supporting heating and ventilating systems for the main control room, the cable spreading room, AC/DC bus room, control rod drive and the motor generator sets for the mechanism switchgear primary and intermediate pumps
33 speed control and the NSS switchgear. (Reference Figures 1.2-60 through 1.2-65)

39 e. Diesel Generator Building -
This building houses safety related emergency electrical supply equipment to assure safe shutdown of the plant in the event of the loss of external power, and the Emergency Plant Service Water System equipment to assure safe shutdown and the maintenance of the safe shutdown condition in the event of loss of the Normal Plant Service Water
33 39 System. (Reference Figures 1.2-66 through 1.2-72)

In addition to the above major buildings, the Radwaste Building (Reference Figures 1.2-73 through 1.2-81) and other balance of plant buildings are supplied. These include the Turbine Generator Building
15 (Reference Figures 1.2-82 through 1.2-89), the Circulation Water Pumphouse, the River Water Pumphouse, the Plant Service Building (Reference Figure 1.2-90), the Gatehouse, and the Maintenance Shop and Warehouse Building (Reference
41 5 Figures 1.2-91 and 92).

c. Steam Generator Building (SGB)-
This building houses the major components for the Steam Generator System, the Intermediate Heat Transport System, the Auxiliary Heat Removal System, the Sodium-Water Reaction Pressure Relief System, and primary sodium storage (reference Figures
47 45 39 1.2-48 through 1.2-69).

d. Control Building -
This building houses the main control room which contains both Nuclear Steam Supply System and the BOP controls, the life supporting heating and ventilating systems for the main control room, the cable spreading room, AC/DC bus room, control rod drive and the motor generator sets for the mechanism switchgear, primary and intermediate pumps speed control and the NSS switchgear
47 33 39 (reference Figures 1.2-70 through 1.2-75).

e. Diesel Generator Building -
This building houses safety related emergency electrical supply equipment to assure safe shutdown of the plant in the event of the loss of external power. The building also houses breakers for the PHTS and IHTS sodium pumps, the motor generators for loop #3 PHTS and IHTS sodium pumps, and the 13.8KV and 4.16KV switchgear (reference Figures 1.2-76
47 45 33 39 through 1.2-80).

In addition to the above major buildings, the Radwaste
47 Area (reference Figures 1.2-39 through 1.2-47) and other Balance of Plant buildings are supplied. These include the Turbine
47 15 Generator Building (reference Figures 1.2-81 through 1.2-88), the
42 Circulation Water Pumphouse, the Switchyard Relay House, the River Water Pumphouse, the Plant Service Building (reference Figure
5 1.2-90), the Gatehouse, and the Maintenance Shop and Warehouse
41 33 Building (reference Figures 1.2-91 and 1.2-92).

6642-1

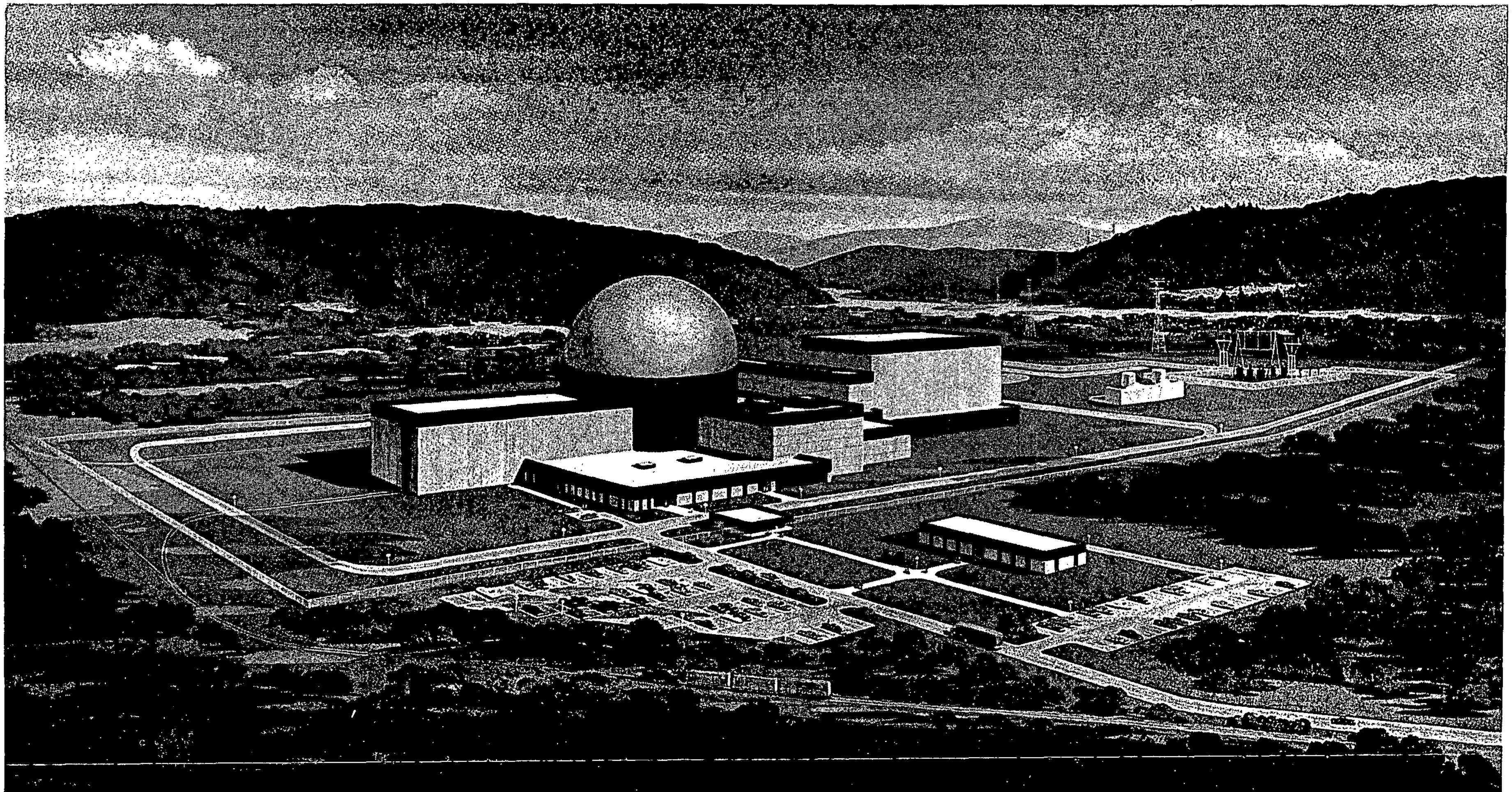


Figure 1.2-1. Artist's Conception of The Clinch River Breeder Reactor Plant

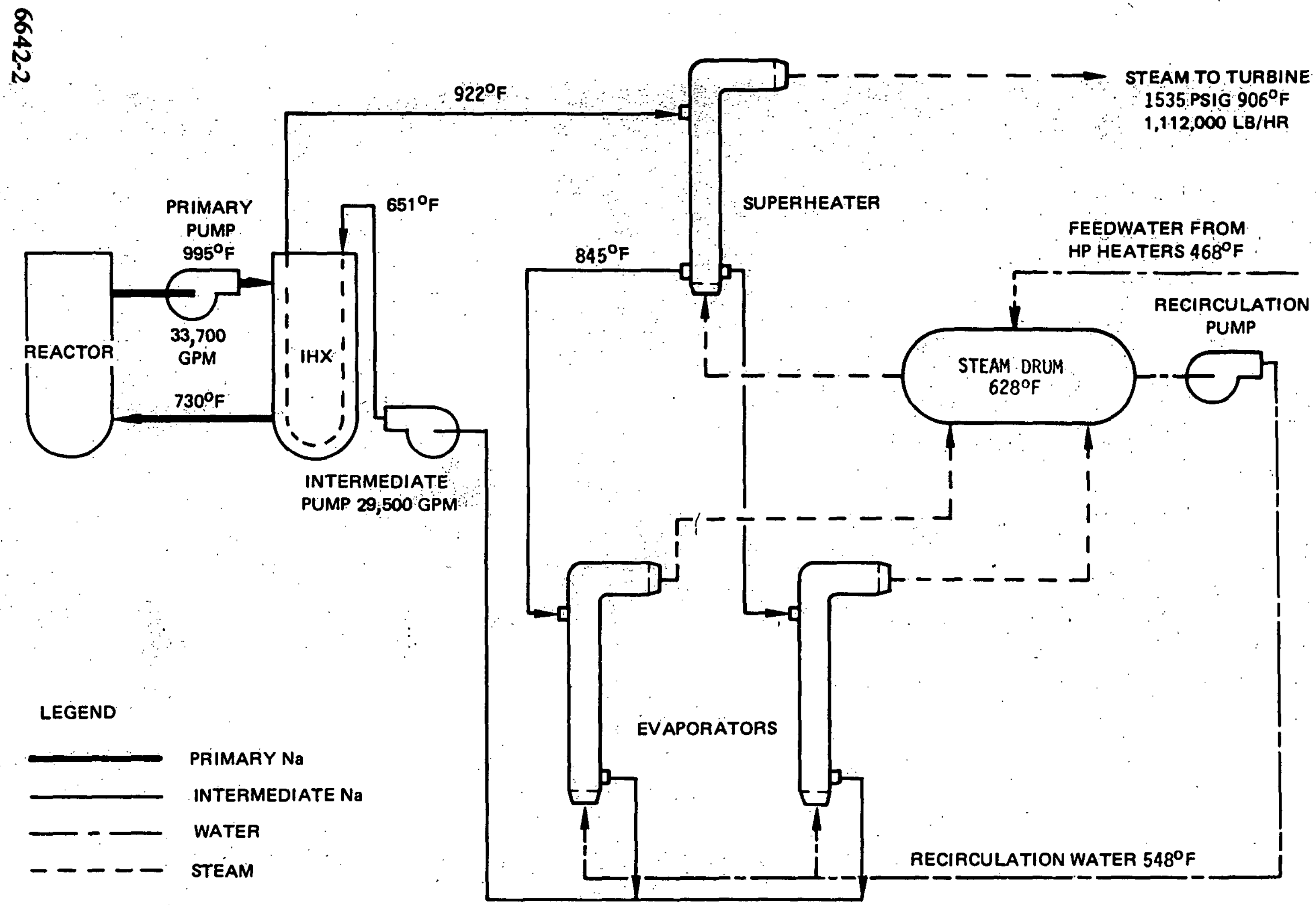



Figure 1.2-2. General Configuration of the Heat Transport System, One of Three Loops

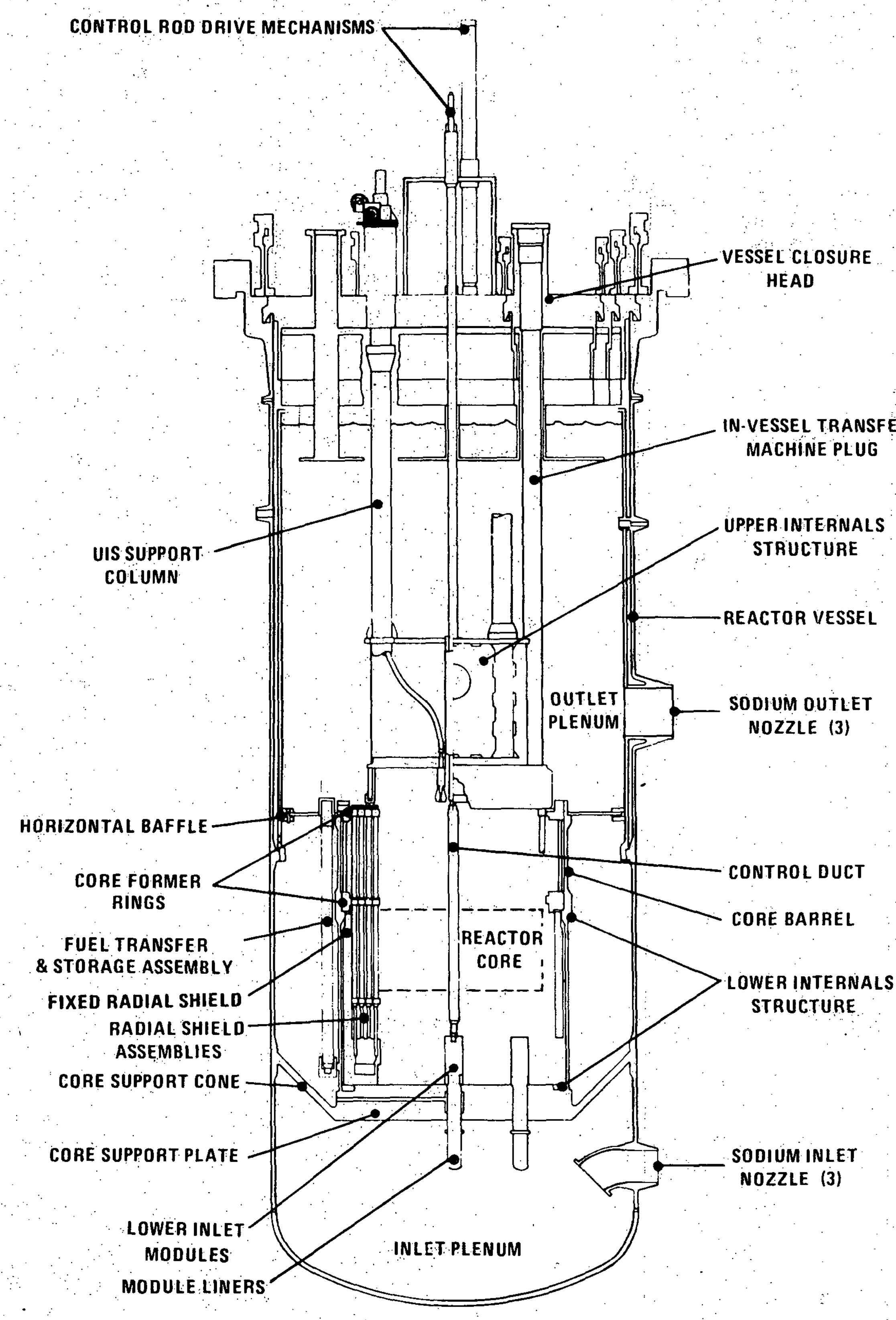
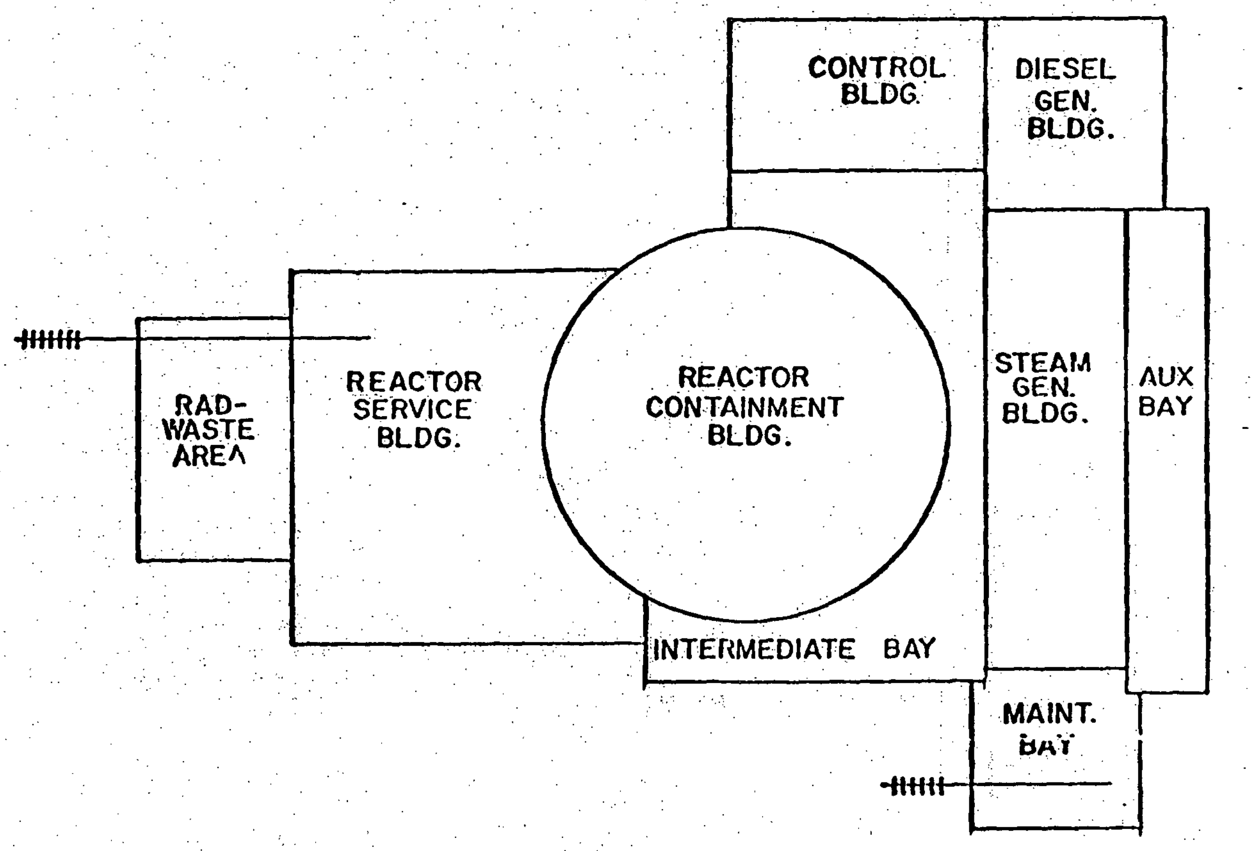



Figure 1.2-3. Reactor and Closure Head Schematic

4419-5



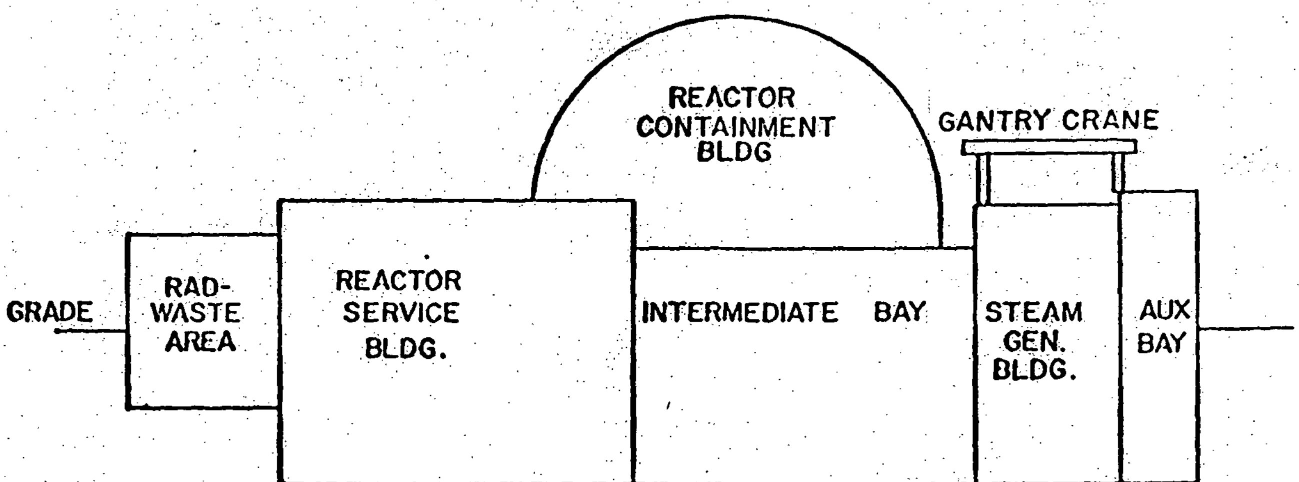
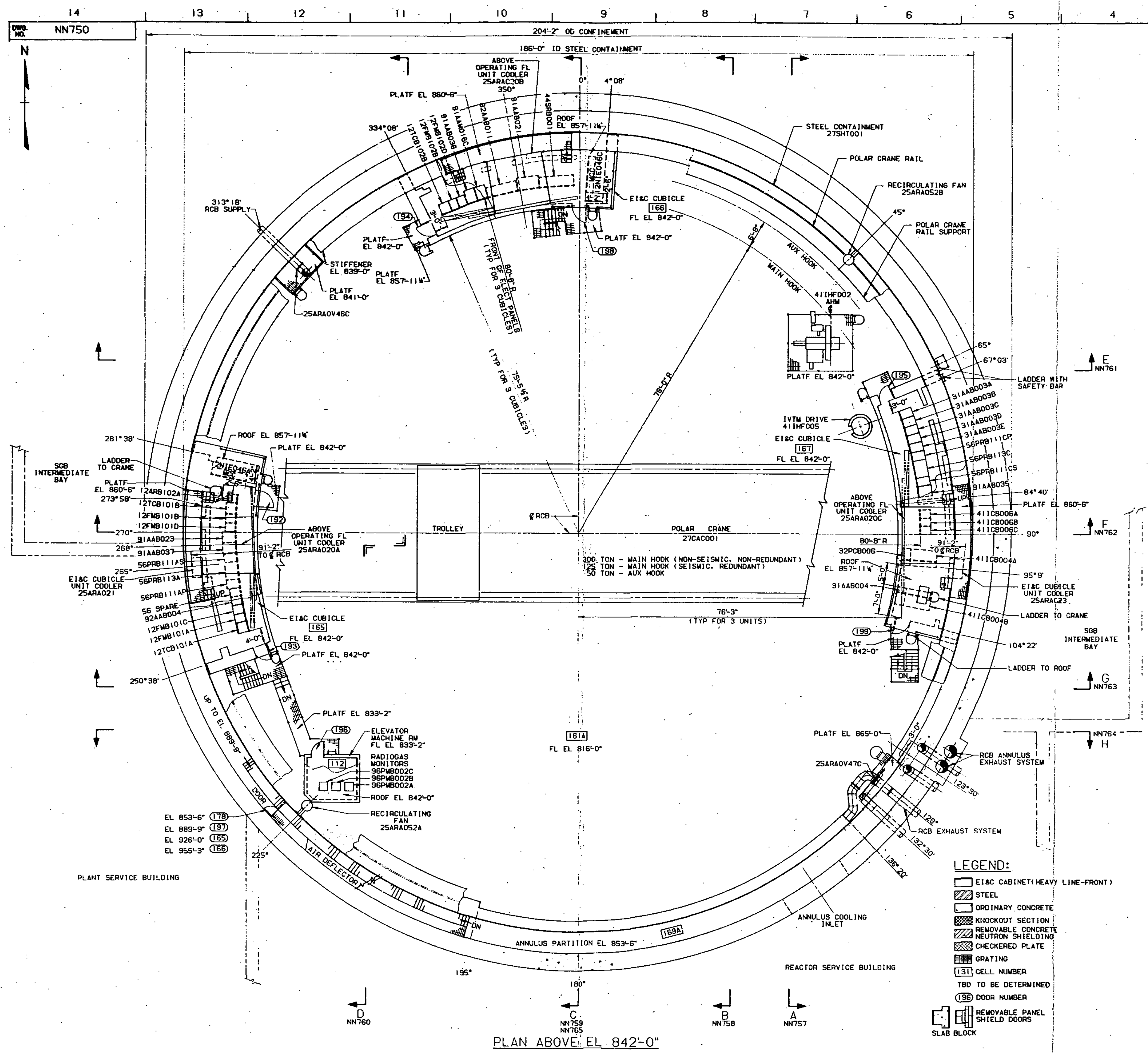




Figure 1.2- 4 Basic Layout of the Nuclear Island Building




## GENERAL NOTES

1. FOR RADIATION ZONING, SEE "DESIGN RADIATION DOSE RATES" DWGS NN7399-NN7405

## REFERENCE DRAWINGS

| | | |
|---|---|---|
| NN751 | RCB-GEN ARRGT-PLAN ABV | EL 816'-0" |
| NN752 | | EL 794'-0" |
| NN753 | | EL 780'-0" |
| NN754 | | EL 766'-0" |
| NN755 | | EL 752'-8" |
| NN756 | | EL 733'-0" |
| NN757 | SECTION | A-A&A1-A1 |
| NN758 | | B-B&B1-B1 |
| NN759 | | C-C |
| NN760 | | D-D&D1-D1 |
| NN761 | | E-E |
| NN762 | | F-F |
| NN763 | | G-G |
| NN764 | | H-H |
| NN765 | | OF DOME ABOVE EL 899'-0" |

## LEGEND:

- EI&C CABINET (HEAVY LINE-FRONT)
- STEEL
- ORDINARY CONCRETE
- KNOCKOUT SECTION
- REMOVABLE CONCRETE NEUTRON SHIELDING
- CHECKERED PLATE
- GRATING
- 131 CELL NUMBER
- TBD TO BE DETERMINED
- 196 DOOR NUMBER
- SLAB BLOCK
- REMOVABLE PANEL SHIELD DOORS

KEY PLAN

GRAPHIC SCALE 0 4' 8' 12' 16' 20' 24' 1/8" = 1'-0"

Figure 1.2-5
General Arrangement
Reactor Containment Building
Plan El. 842'-0"

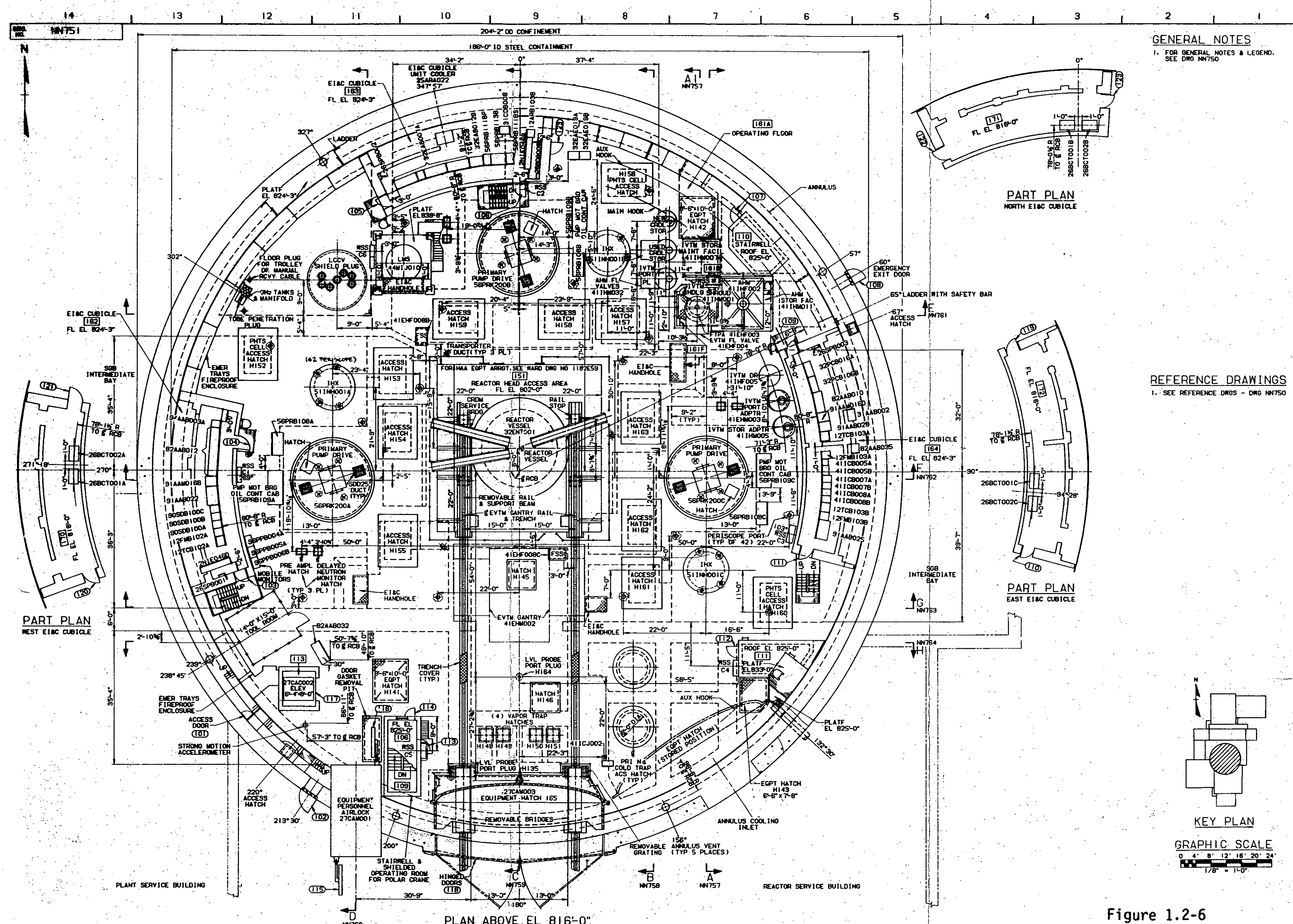



Figure 1.2-6
General Arrangement
Reactor Containment Building
Plan El. 816'-0"

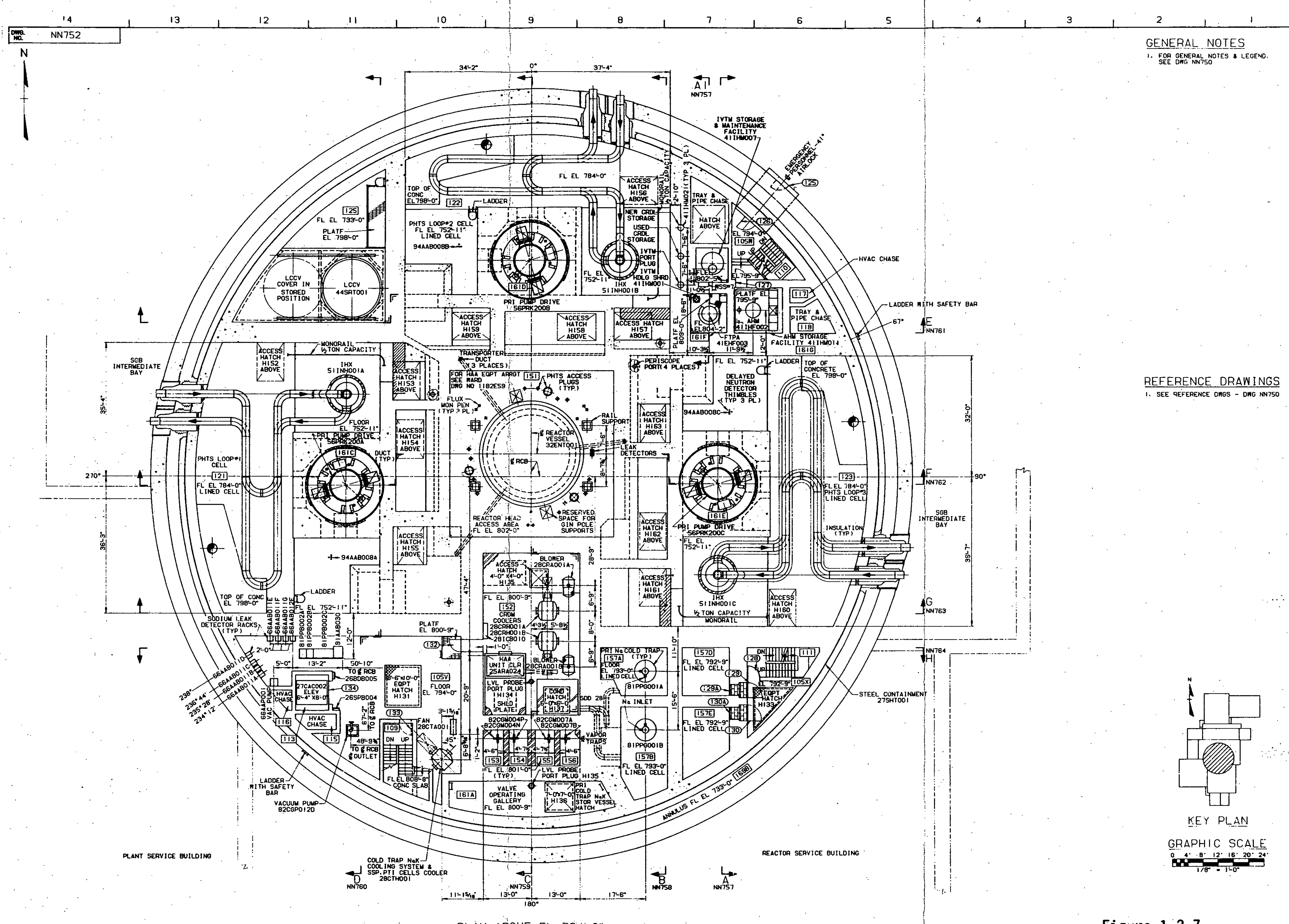



Figure 1.2-7
General Arrangement
Reactor Containment Building
Plan El. 794'-0"

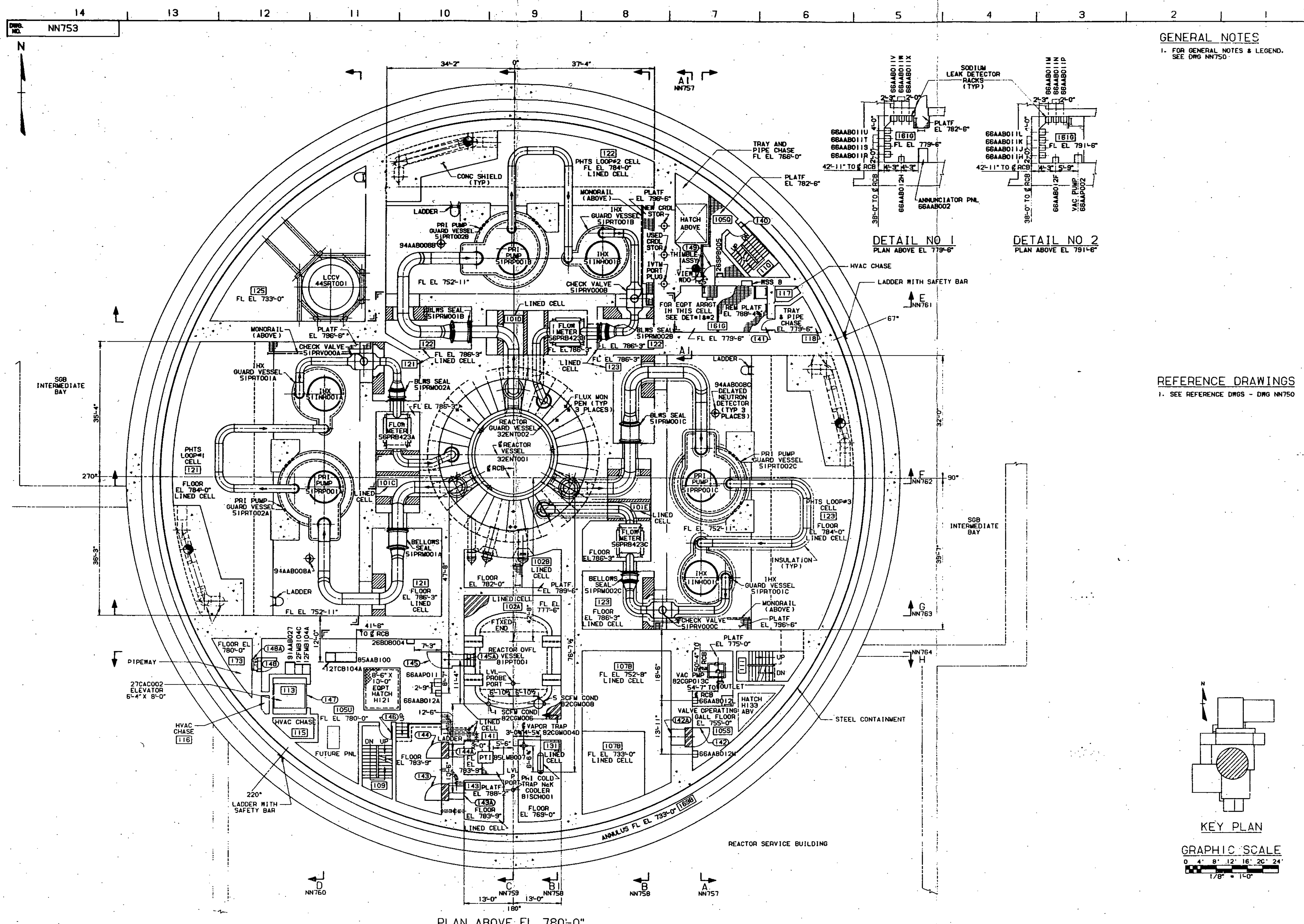
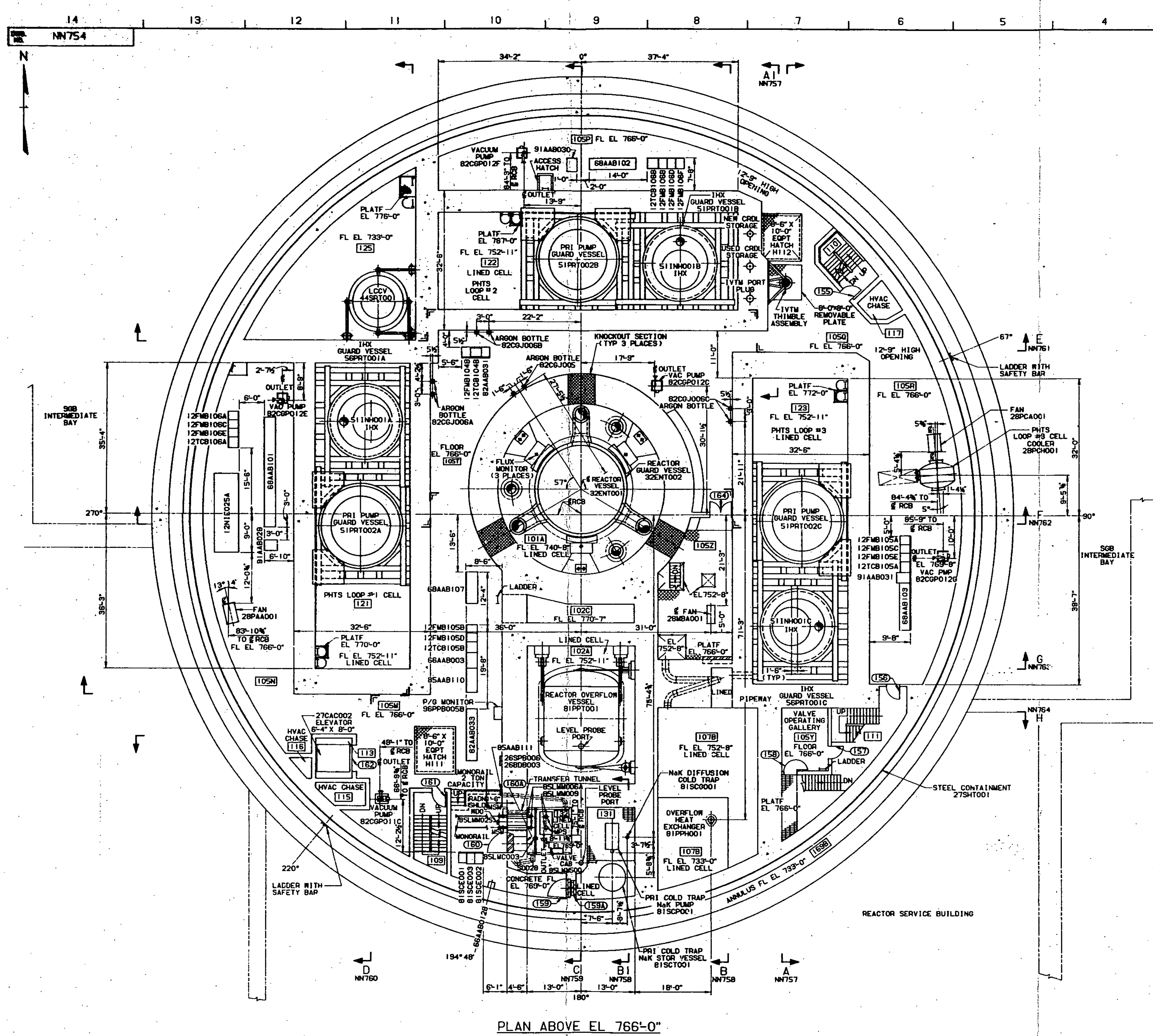



Figure 1.2-8
General Arrangement
Reactor Containment Building
Plan El. 780'-0"



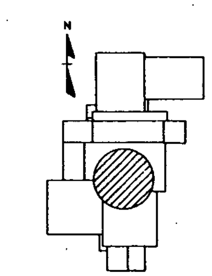

PLAN ABOVE EL 766'-0"

GENERAL NOTES

1. FOR GENERAL NOTES & LEGEND, SEE DWG NN750

REFERENCE DRAWINGS

1. SEE REFERENCE DWGS - DWG NN750



KEY PLAN

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

Figure 1.2-9
General Arrangement
Reactor Containment Building
Plan El. 766'-0"

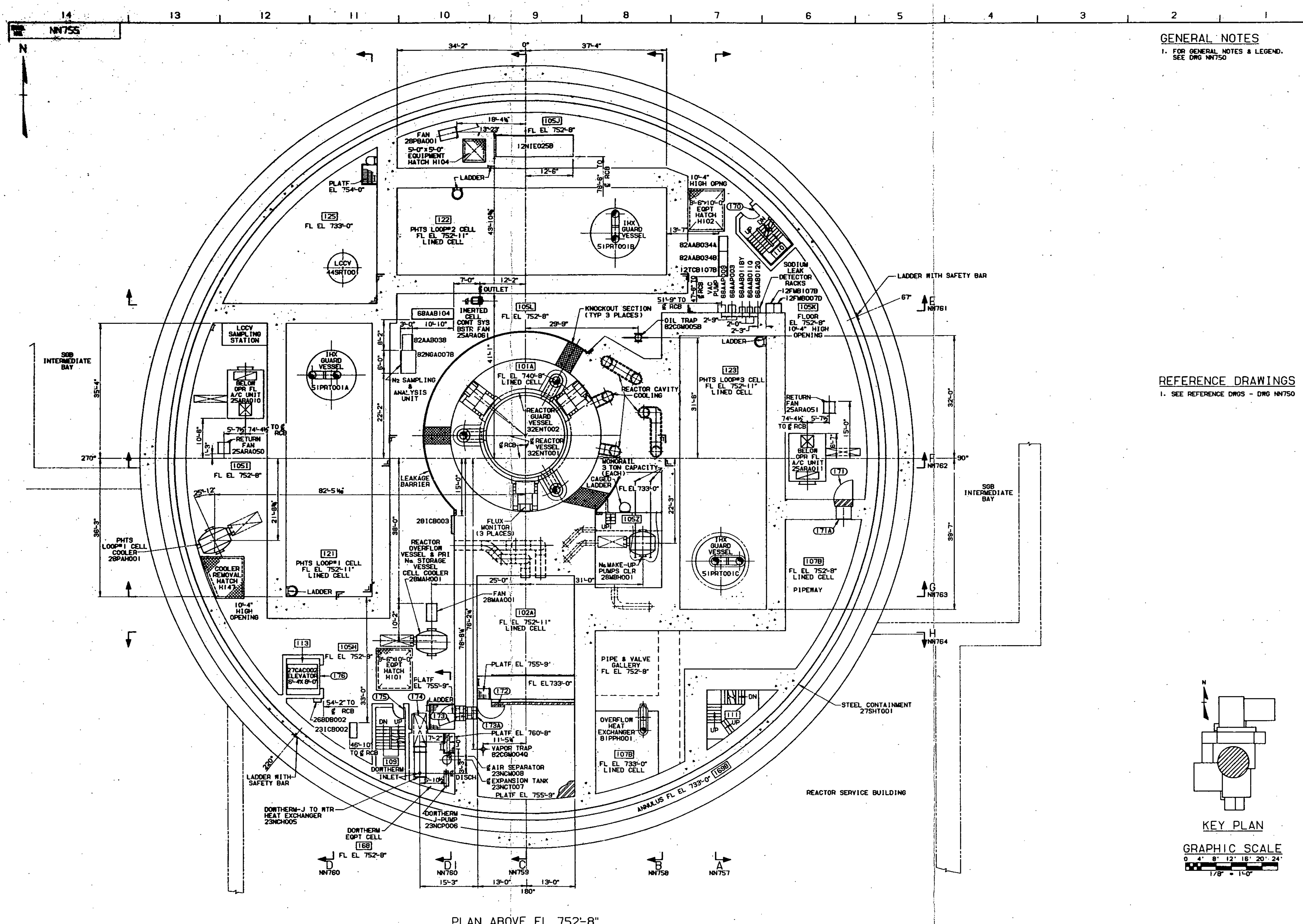



Figure 1.2-10
General Arrangement
Reactor Containment Building
Plan El. 752'-8"

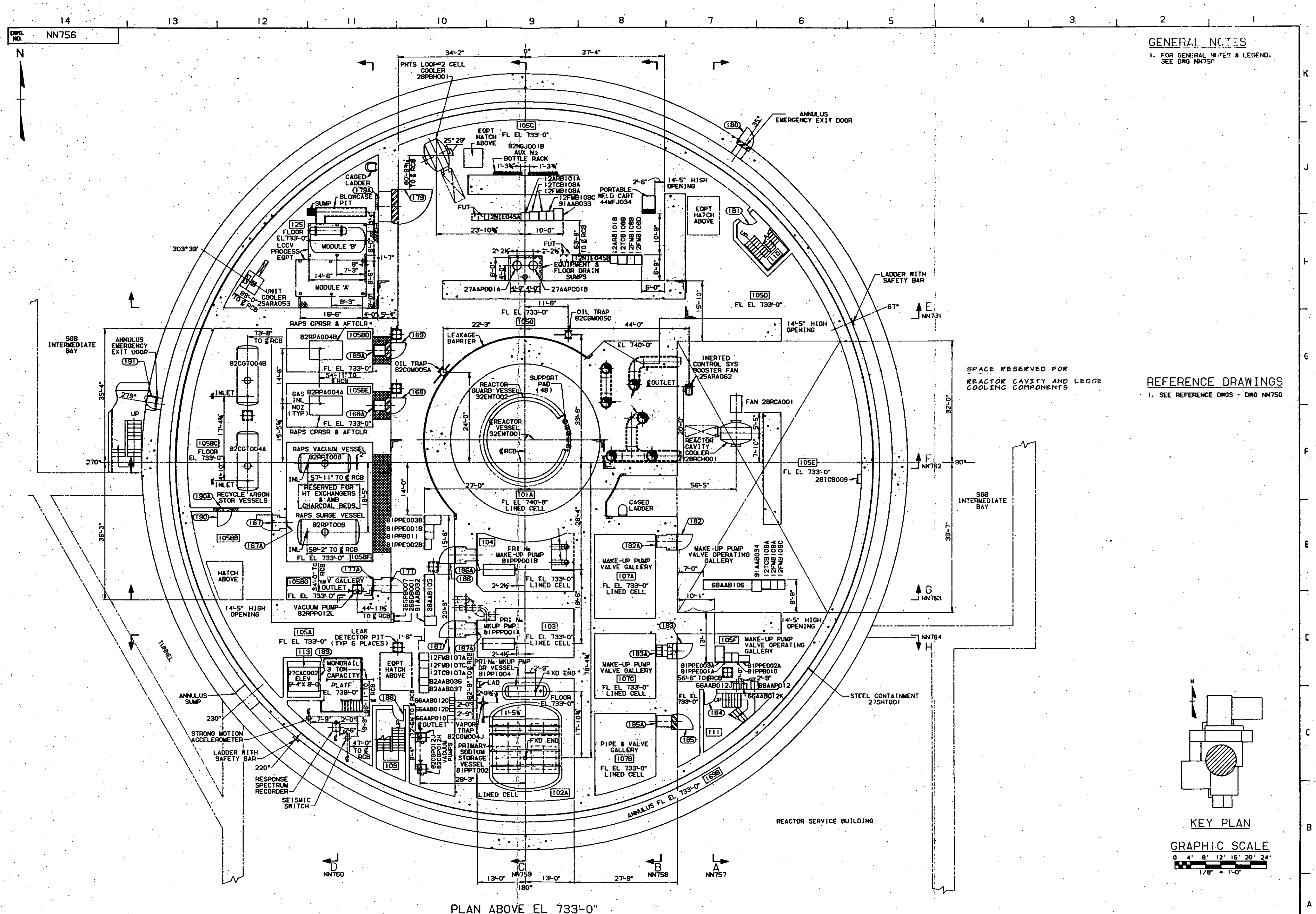



**Figure 1.2-11**
**General Arrangement**
**Reactor Containment Building**
**Plan El. 733'-0"**

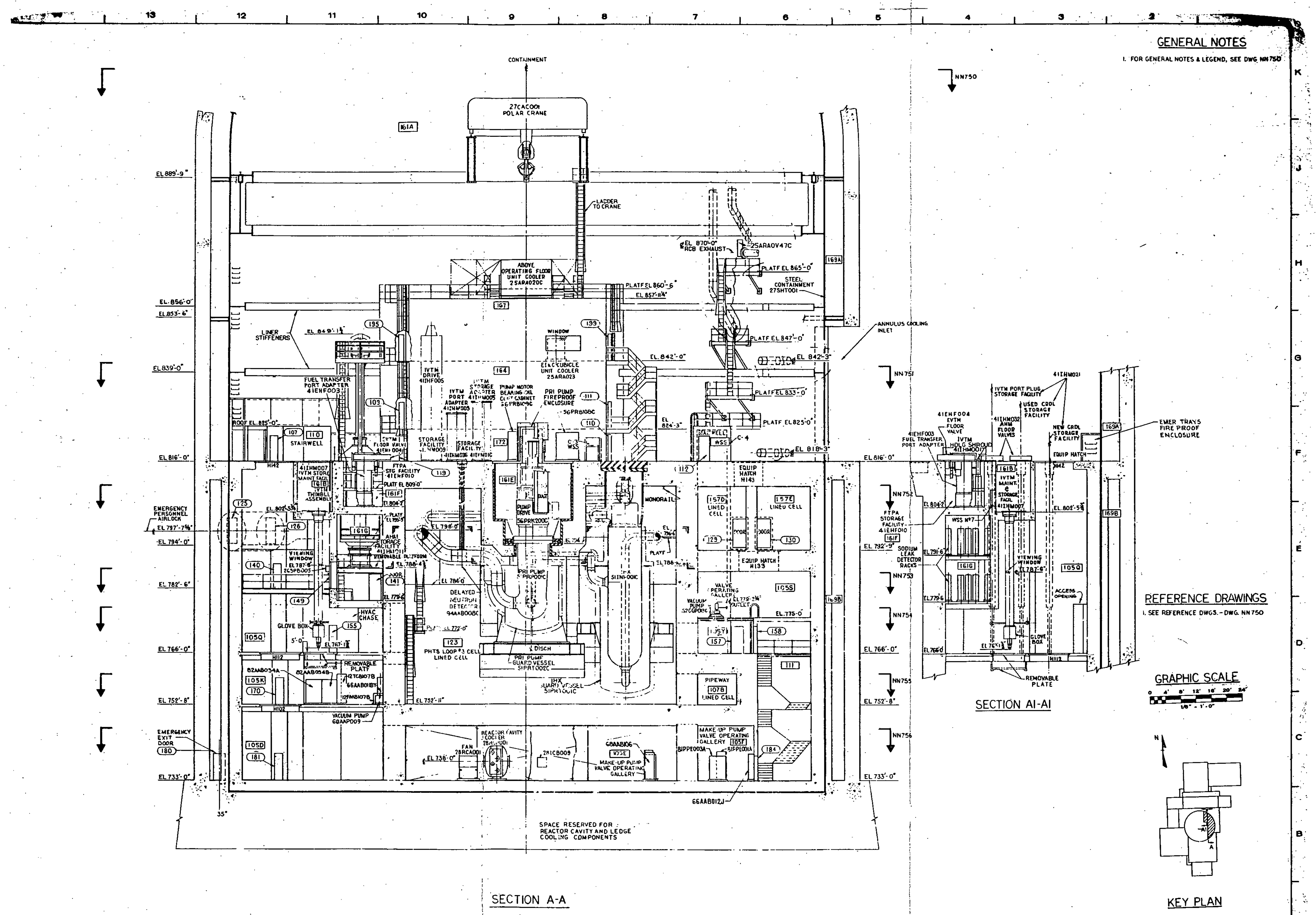



Figure 1.2-12
General Arrangement
Reactor Containment Building
Section A-A and A1-A1

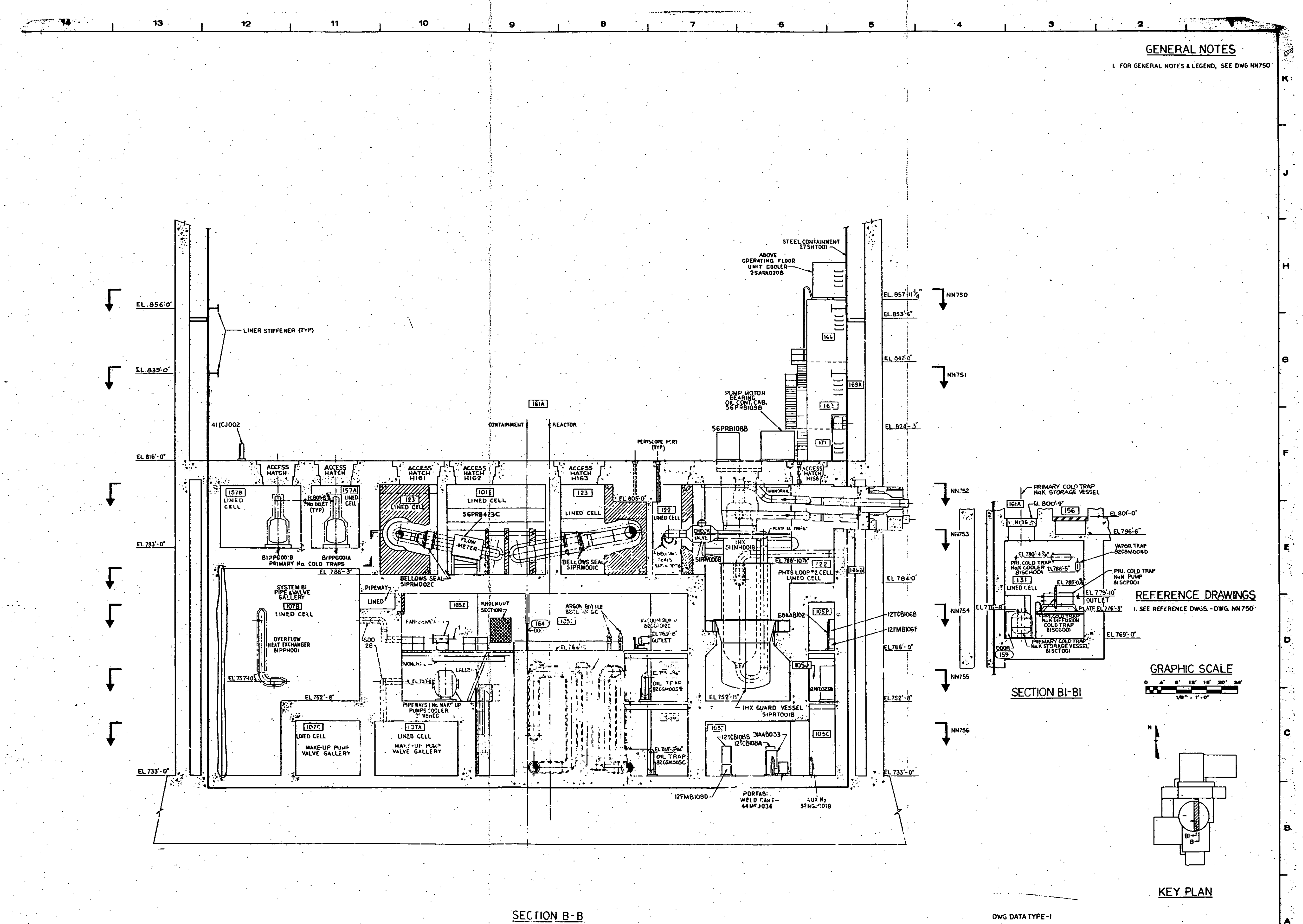
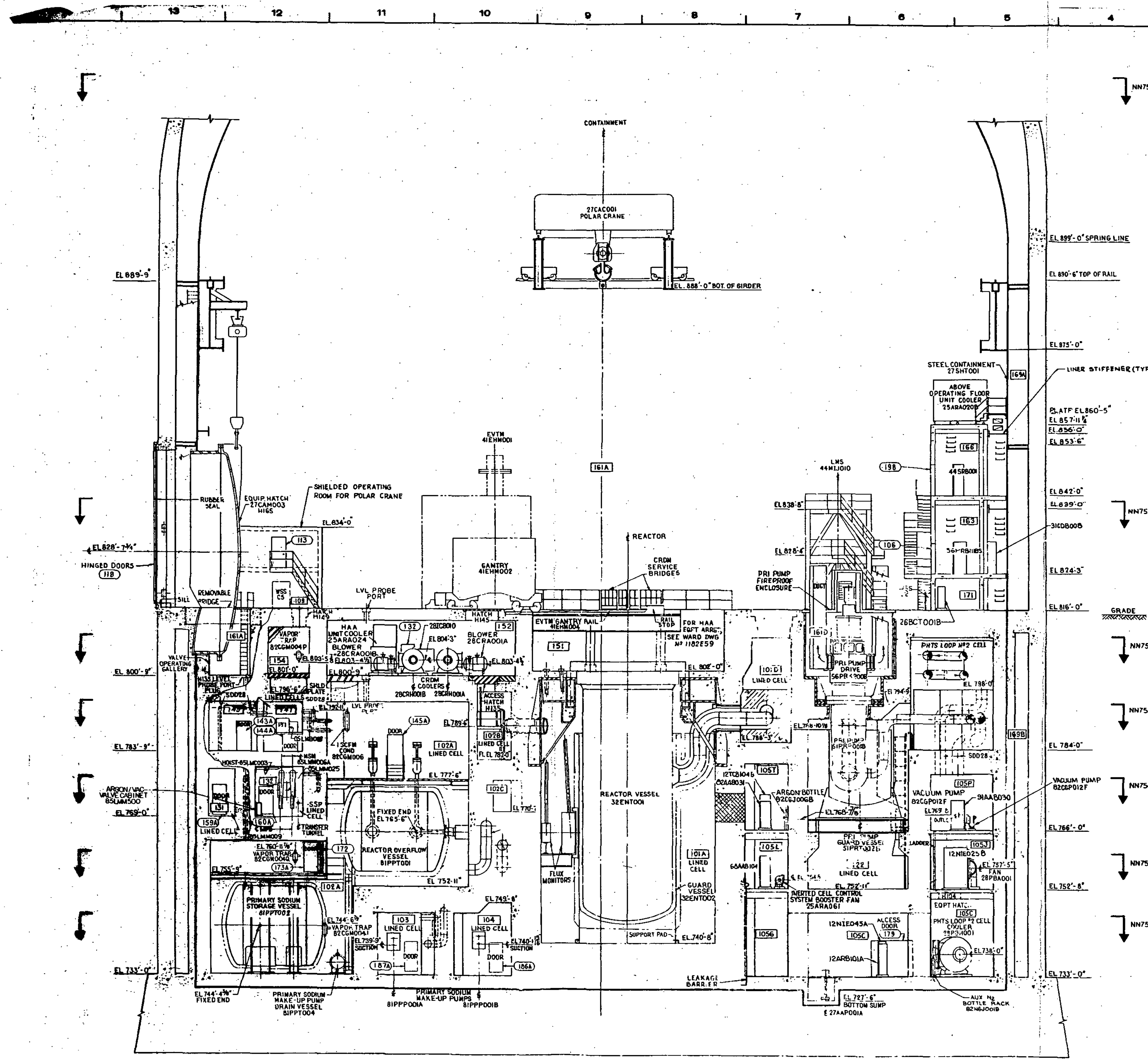



**Figure 1.2-13**
**General Arrangement**
**Reactor Containment Building**
**Section B-B and B1-B1**

## GENERAL NOTES

1. FOR GENERAL NOTES & LEGEND, SEE DWG NN750



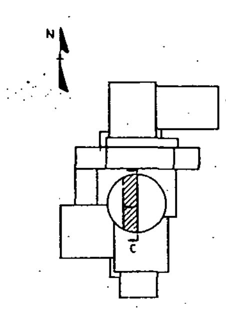

SECTION C-C

## REFERENCE DRAWINGS

1. SEE REFERENCE DWGS. - DWG. NN750

GRAPHIC SCALE

1/8" = 1'-0"



KEY PLAN

Figure 1.2-14
General Arrangement
Reactor Containment Building
Section C-C

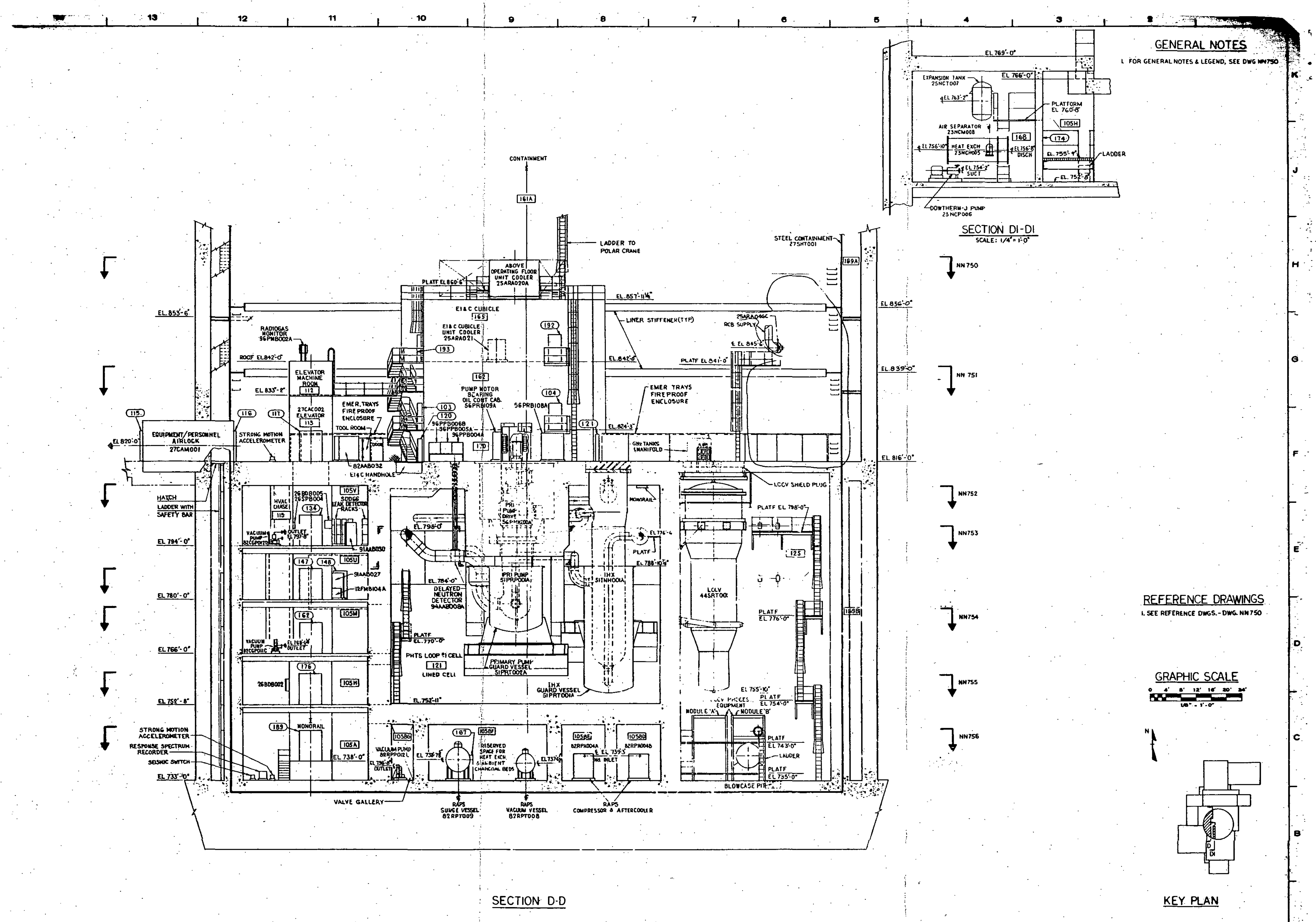



Figure 1.2-15
General Arrangement
Reactor Containment Building
Section D-D and D1-D1

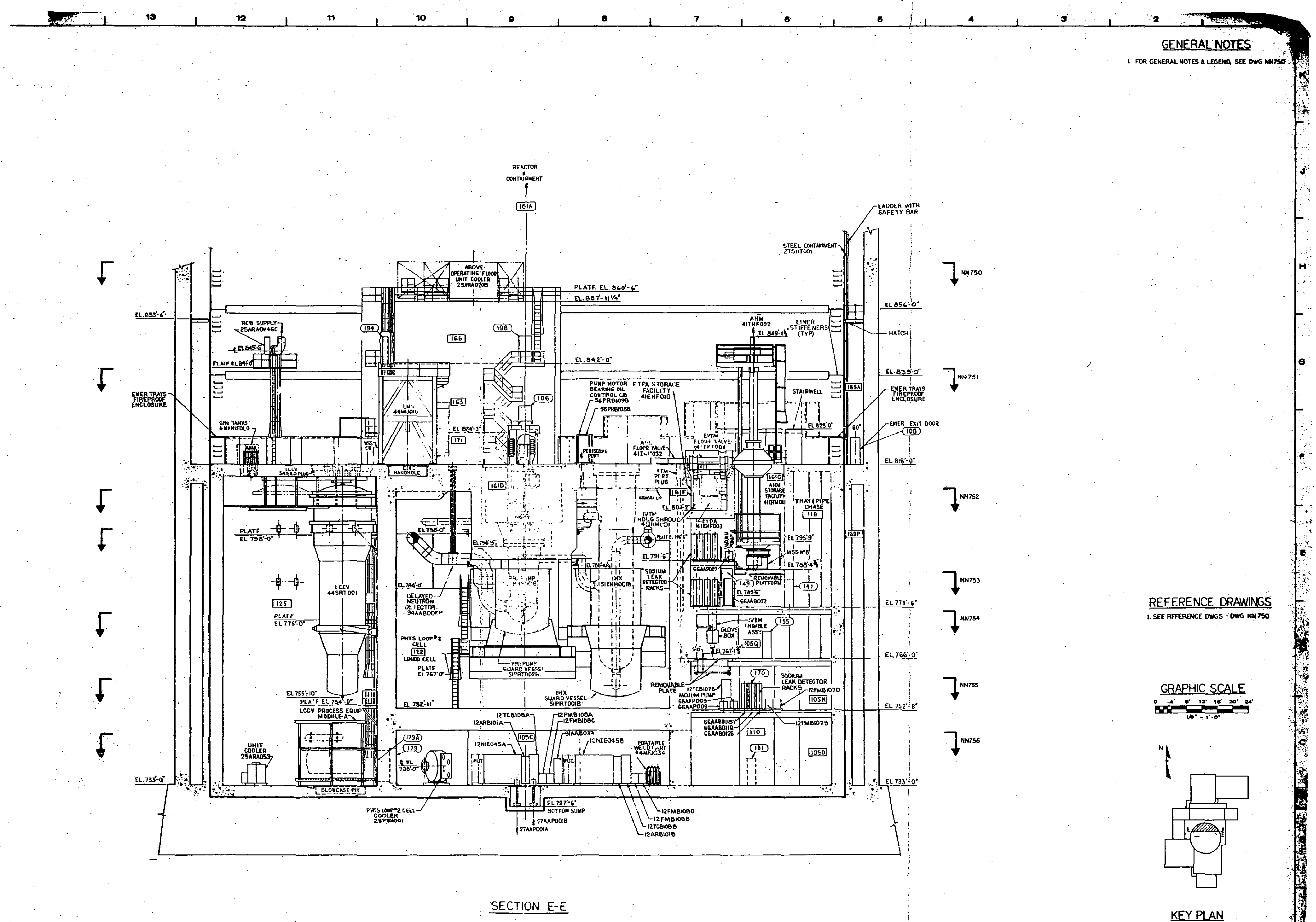



Figure 1.2-16
General Arrangement
Reactor Containment Building
Section E-E

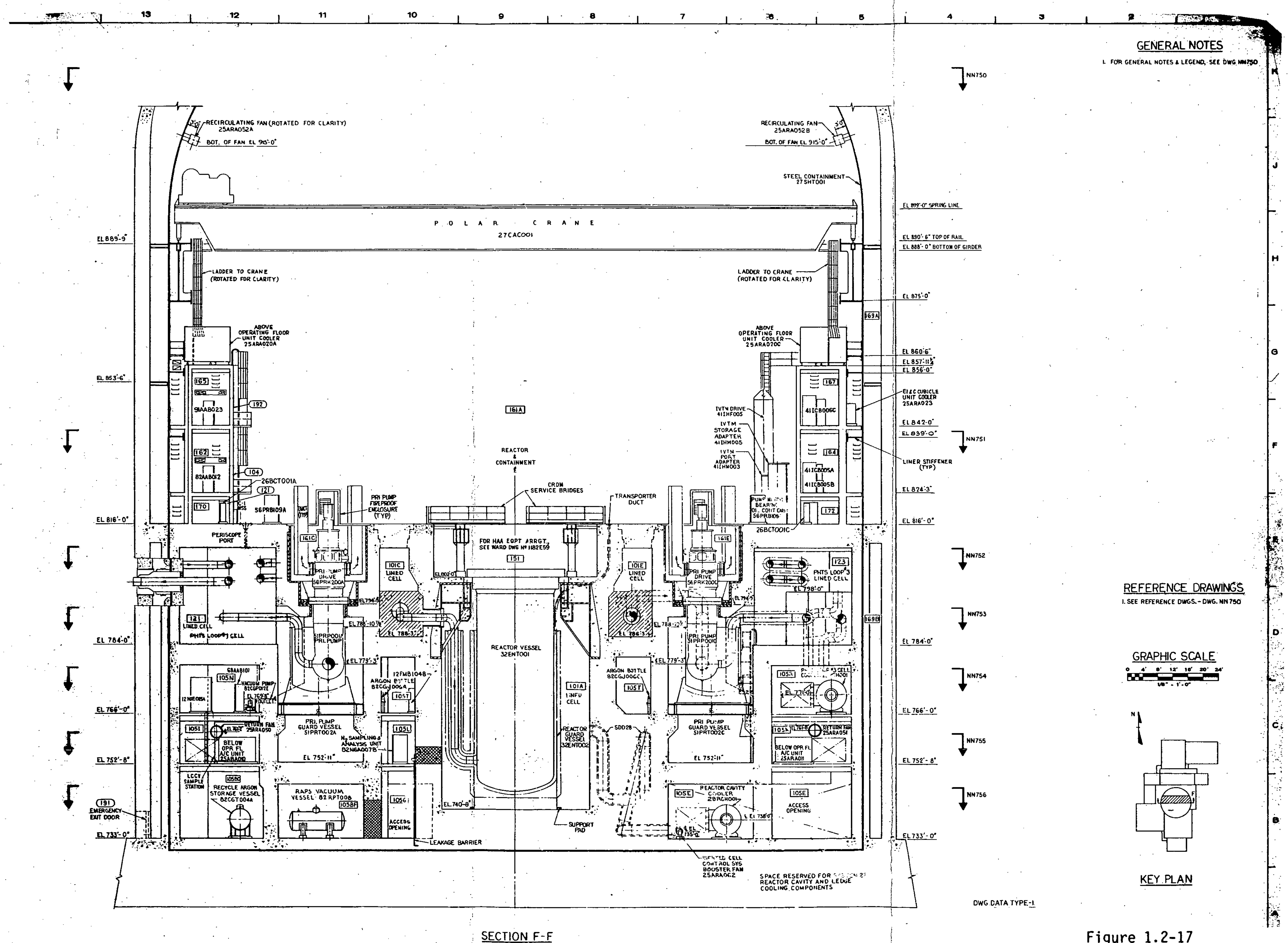
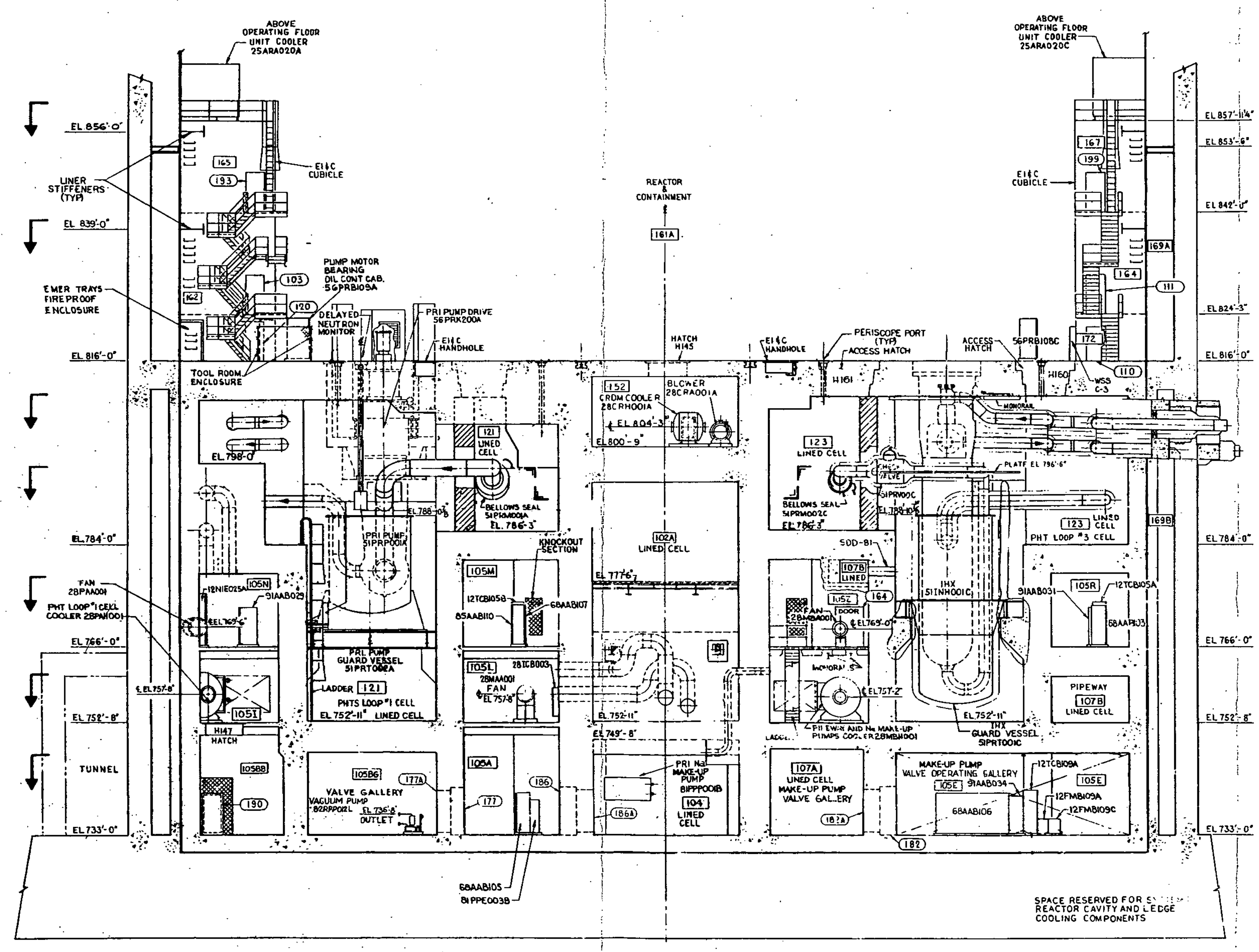



Figure 1.2-17
General Arrangement
Reactor Containment Building
Section F-F

GENERAL NOTES

1. FOR GENERAL NOTES & LEGEND, SEE DWG NN750

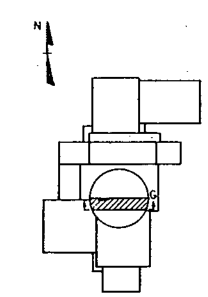



SECTION G-G

REFERENCE DRAWINGS

1. SEE REFERENCE DWGS.—DWG. NN 750

GRAPHIC SCALE



KEY PLAN

DWG DATA TYPE-I

Figure 1.2-18
General Arrangement
Reactor Containment Building
Section G-G

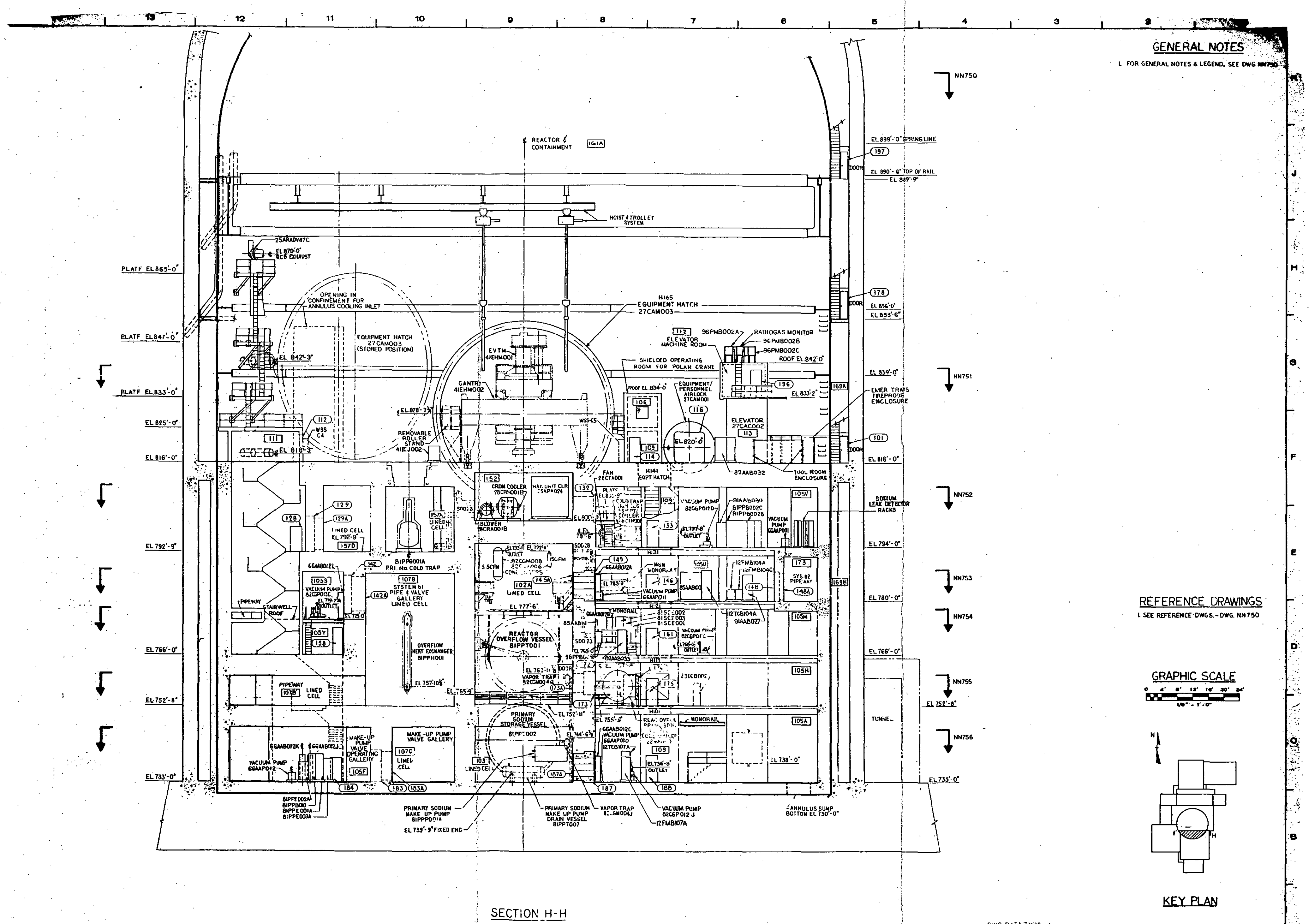



Figure 1.2-19
General Arrangement
Reactor Containment Building
Section H-H

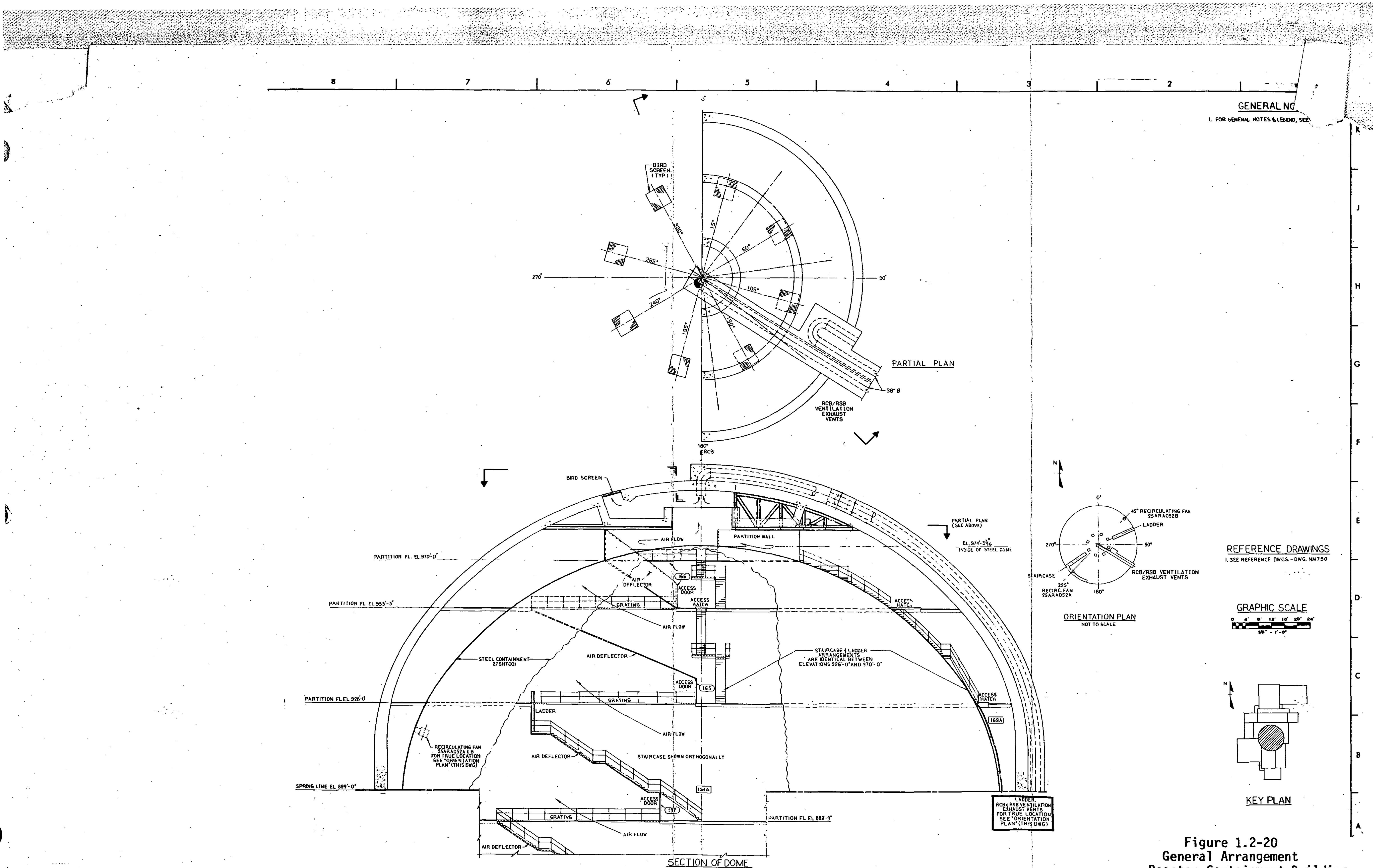
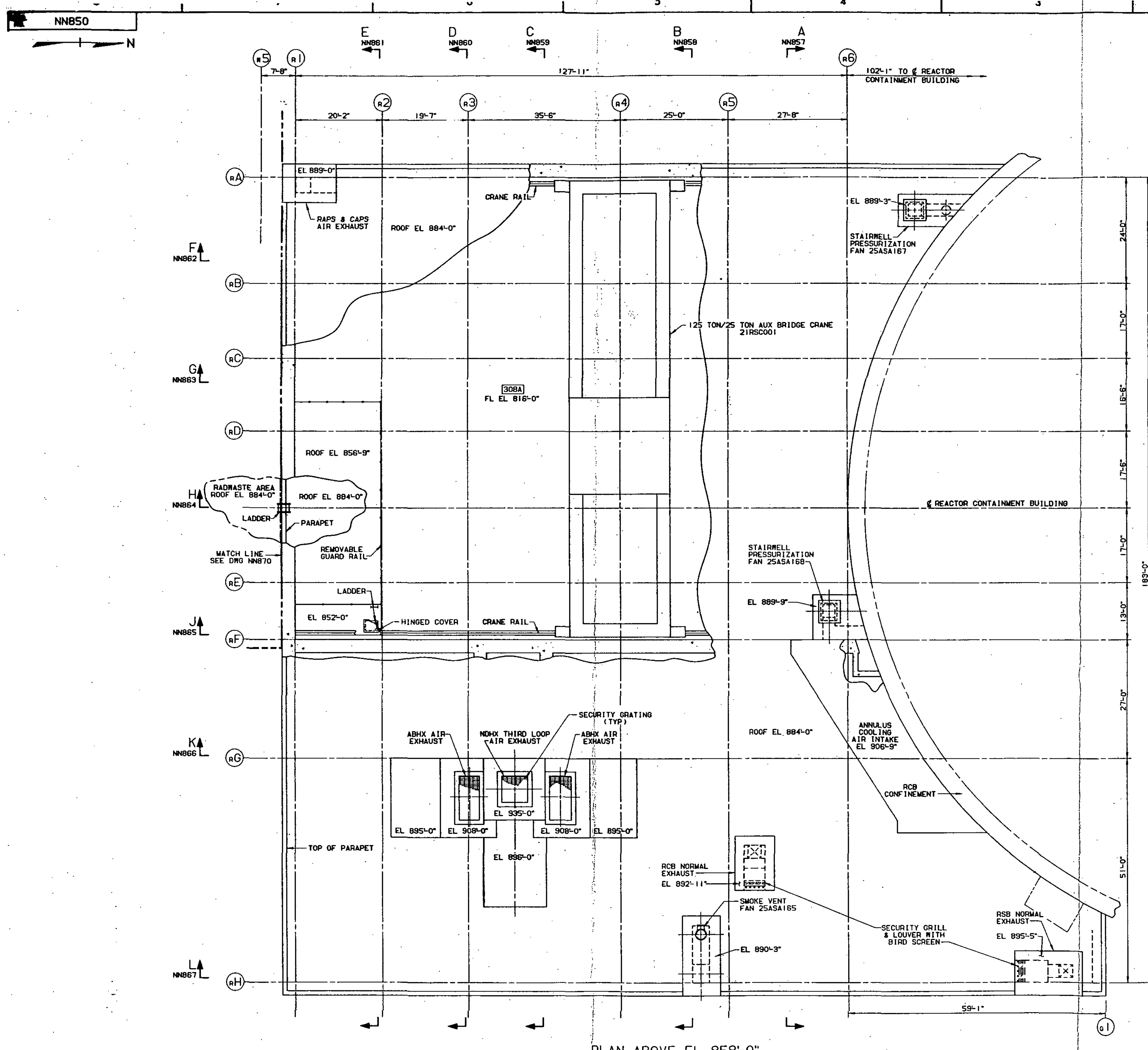



Figure 1.2-20
General Arrangement
Reactor Containment Building
Section C-C
above plan, El. 899-0"




PLAN ABOVE EL 858'-0"

## GENERAL NOTES

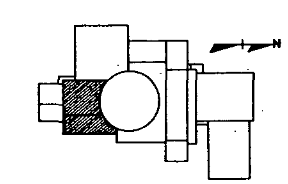
1. ALL HATCH DIMENSIONS ARE CLEAR OPENING SIZE
2. ALL EQUIPMENT/ACCESS HATCHES WILL BE PROVIDED WITH REMOVABLE GUARD RAILS FOR PERSONNEL SAFETY

## REFERENCE DRAWINGS

NN851 RSB-GEN ARRGT-PLAN ABV EL 816'-0"
NN852 EL 792'-0"
NN853 EL 779'-0"
NN854 EL 755'-0"
NN855 PARTIAL EL 749'-0"
EL 840'-0"
EL 864'-6"
NN856 -PLAN ABV EL 733'-0"
NN857 SECTION A-A
NN858 B-B
NN859 C-C
NN860 D-D
NN861 E-E
NN862 F-F
NN863 G-G
NN864 H-H
NN865 J-J
NN866 K-K
NN867 L-L
NN870 RSB-GEN ARRGT-PLAN ABV EL 867'-0" RADWASTE AREA & EL 838'-0"
NN871 EL 816'-0" & EL 795'-0"
NN872 EL 775'-0"
NN873 SECTION A-A
NN874 B-B
NN875 C-C
NN876 D-D
NN877 E-E & F-F
NN878 G-G & H-H

## LEGEND:

- E&C CABINET(HEAVY LINE FRONT)
- HIGH DENSITY CONCRETE
- KNOCKOUT SECTION
- STEEL
- ORDINARY CONCRETE
- CHECKERED PLATE
- GRATING
- 301 CELL NUMBER
- CONCRETE BLOCK
- 200 DOOR NUMBER
- CATCH PAN GRATING
- REMOVABLE PANEL SHIELD DOORS



KEY PLAN

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

Figure 1.2-21
General Arrangement
Reactor Service Building
El. 858'-0"

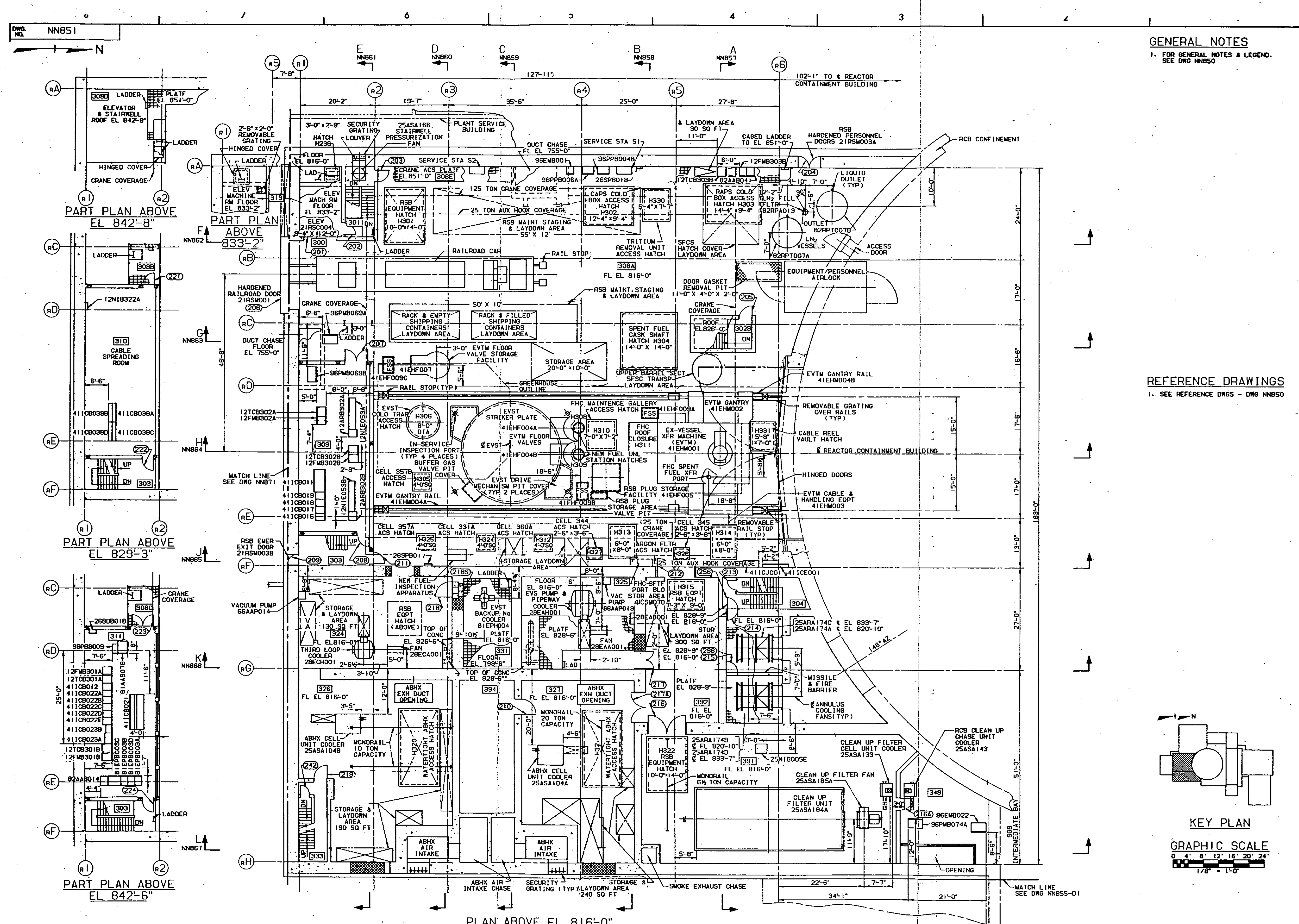



Figure 1.2-22
General Arrangement
Reactor Service Building
El. 816'-0"

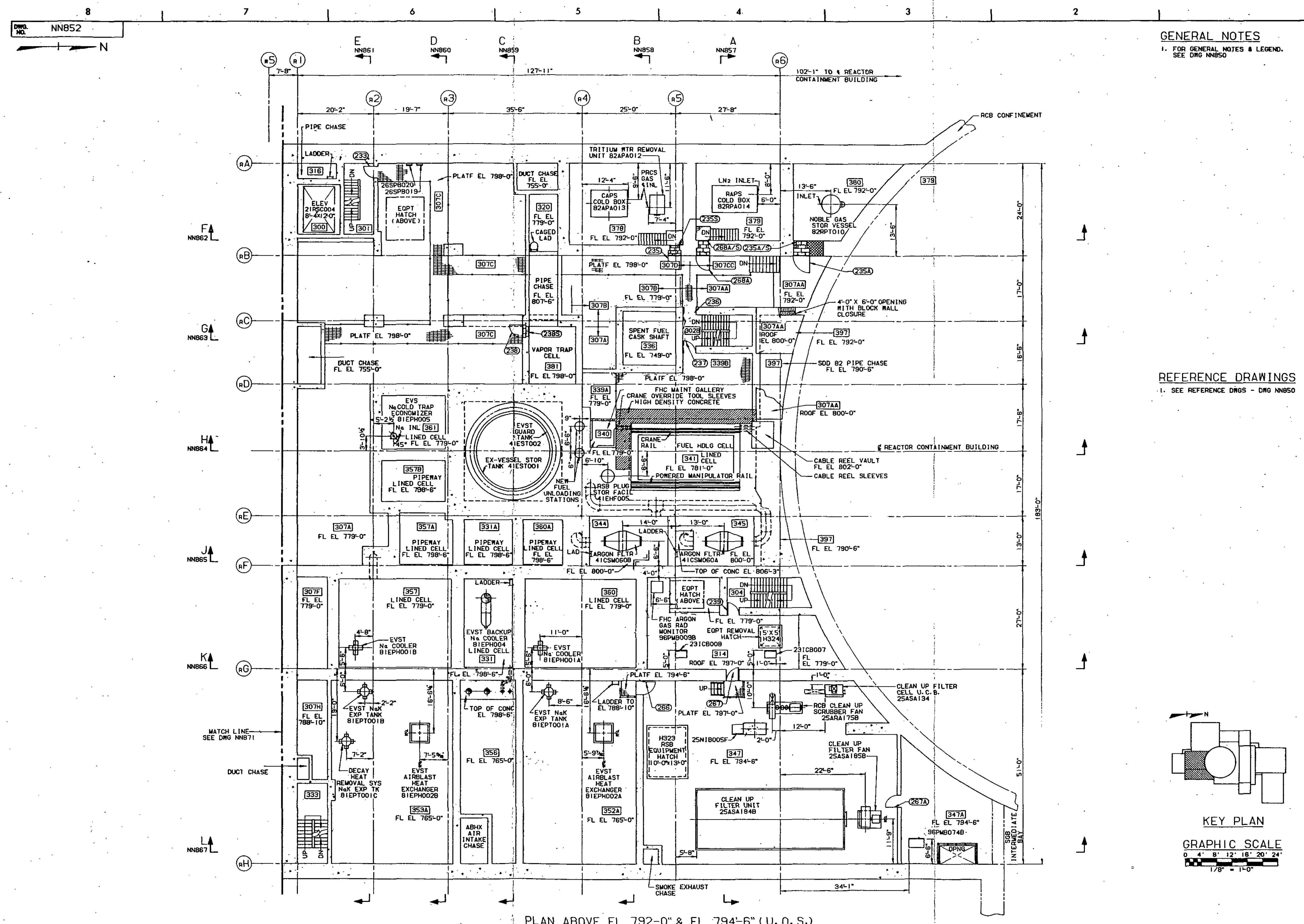



Figure 1.2-23
General Arrangement
Reactor Service Building
El. 792'-0"

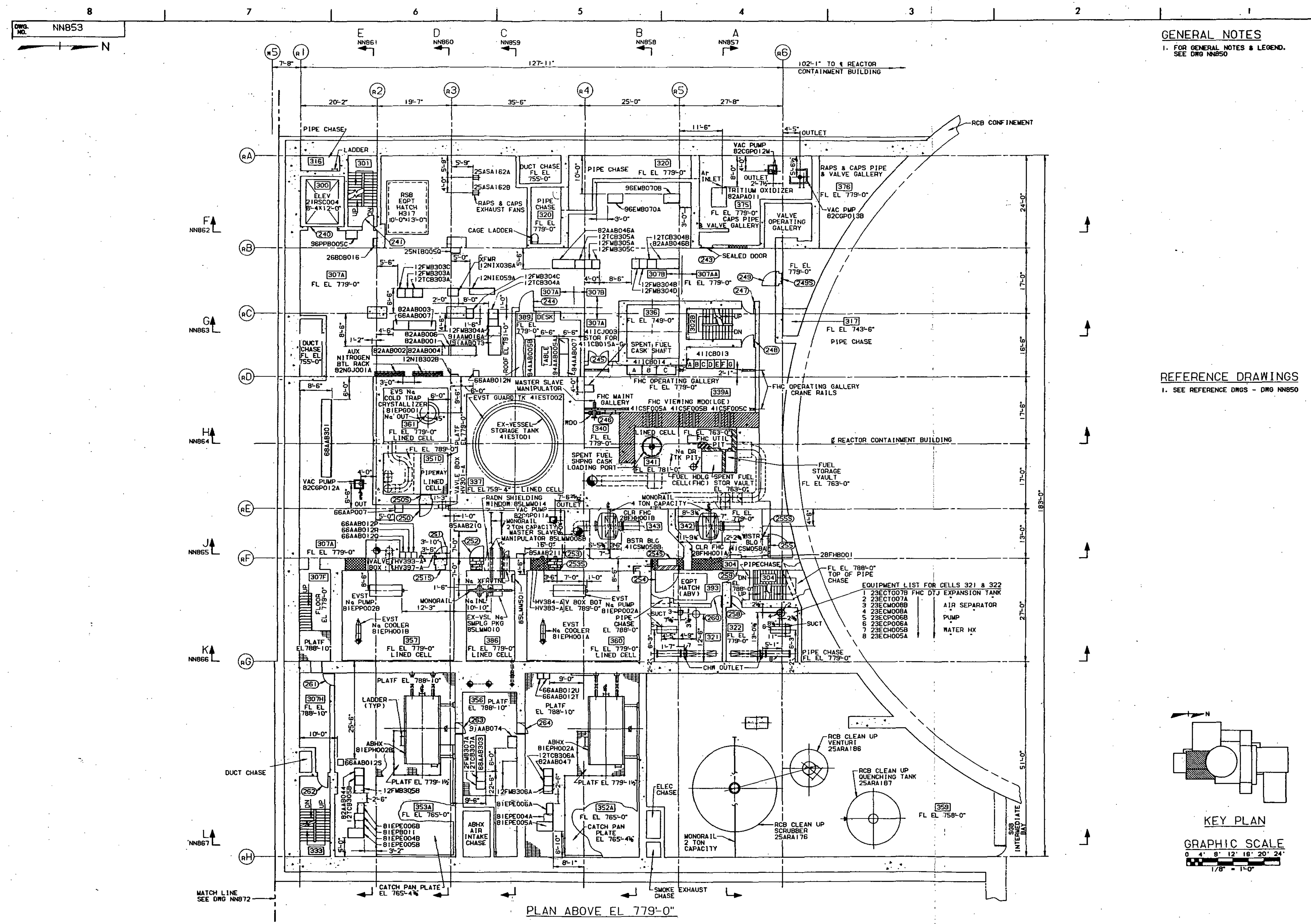



Figure 1.2-24
General Arrangement
Reactor Service Building
El. 779'-0"

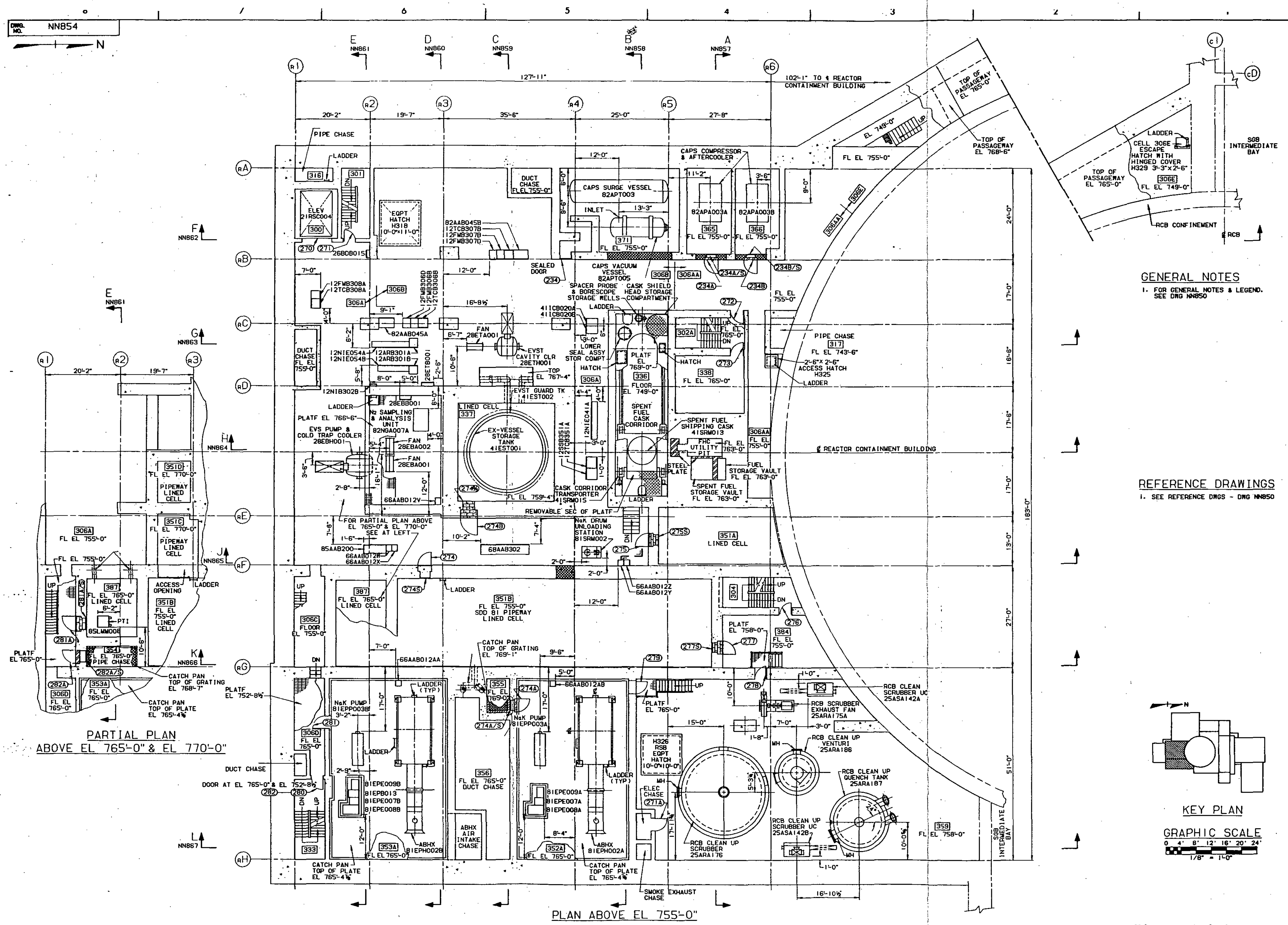



Figure 1.2-25
General Arrangement
Reactor Service Building
El. 755'-0"

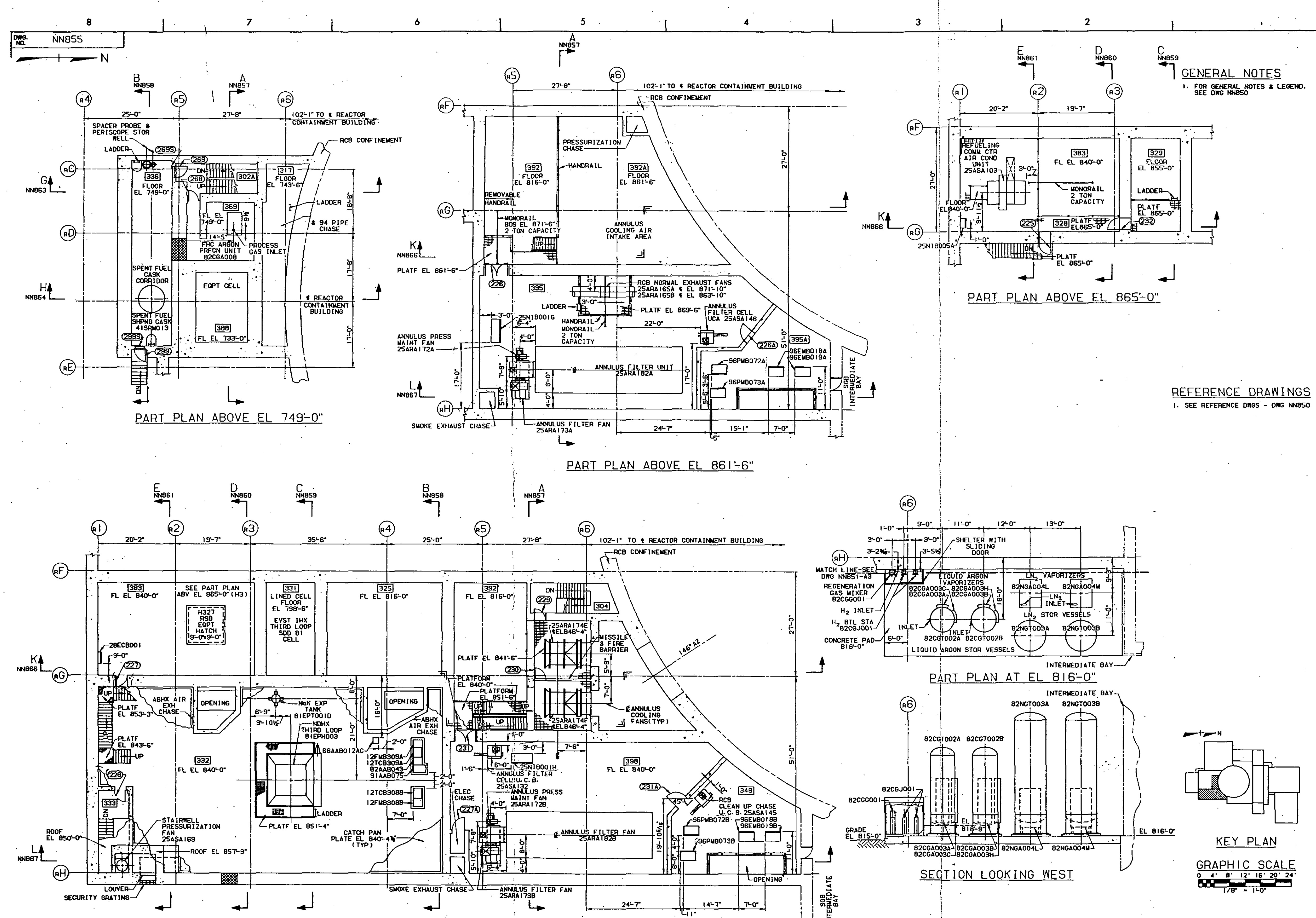



Figure 1.2-26
General Arrangement
Reactor Service Building
Above El. 749'-0", El. 840'-0"
and 864'-6"

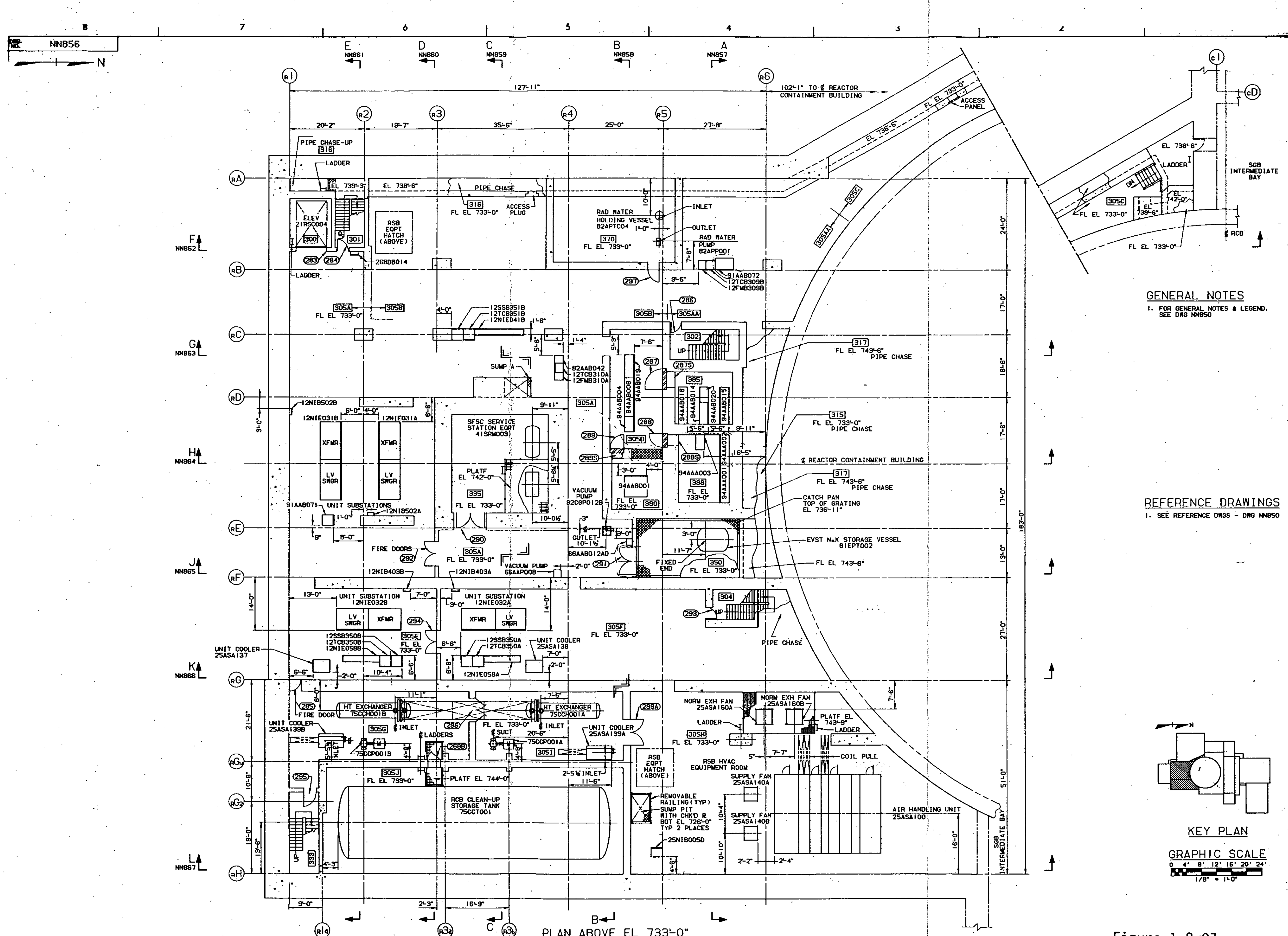



Figure 1.2-27
General Arrangement
Reactor Service Building
El. 733'-0"

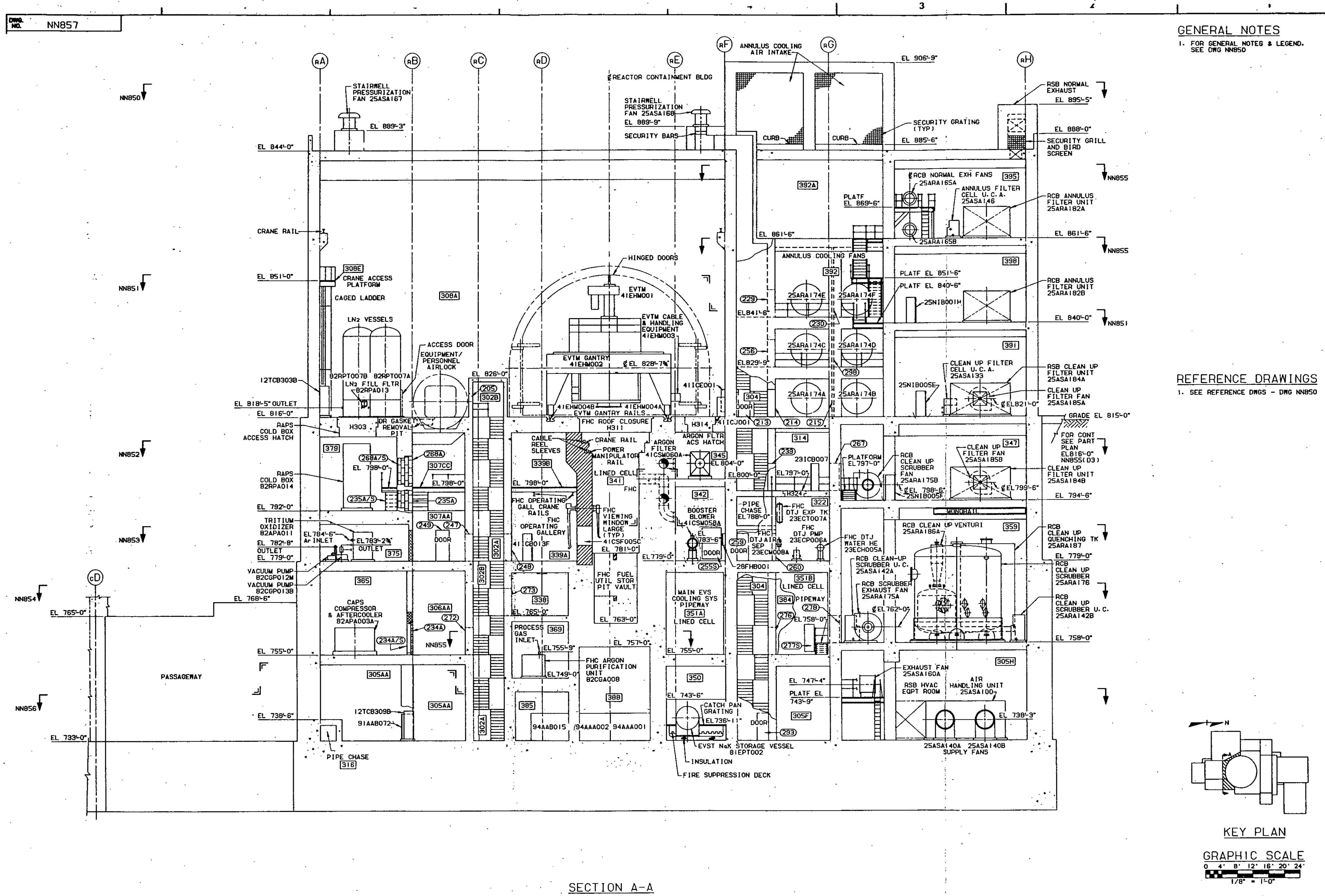



Figure 1.2-28
General Arrangement
Reactor Service Building
Section A-A

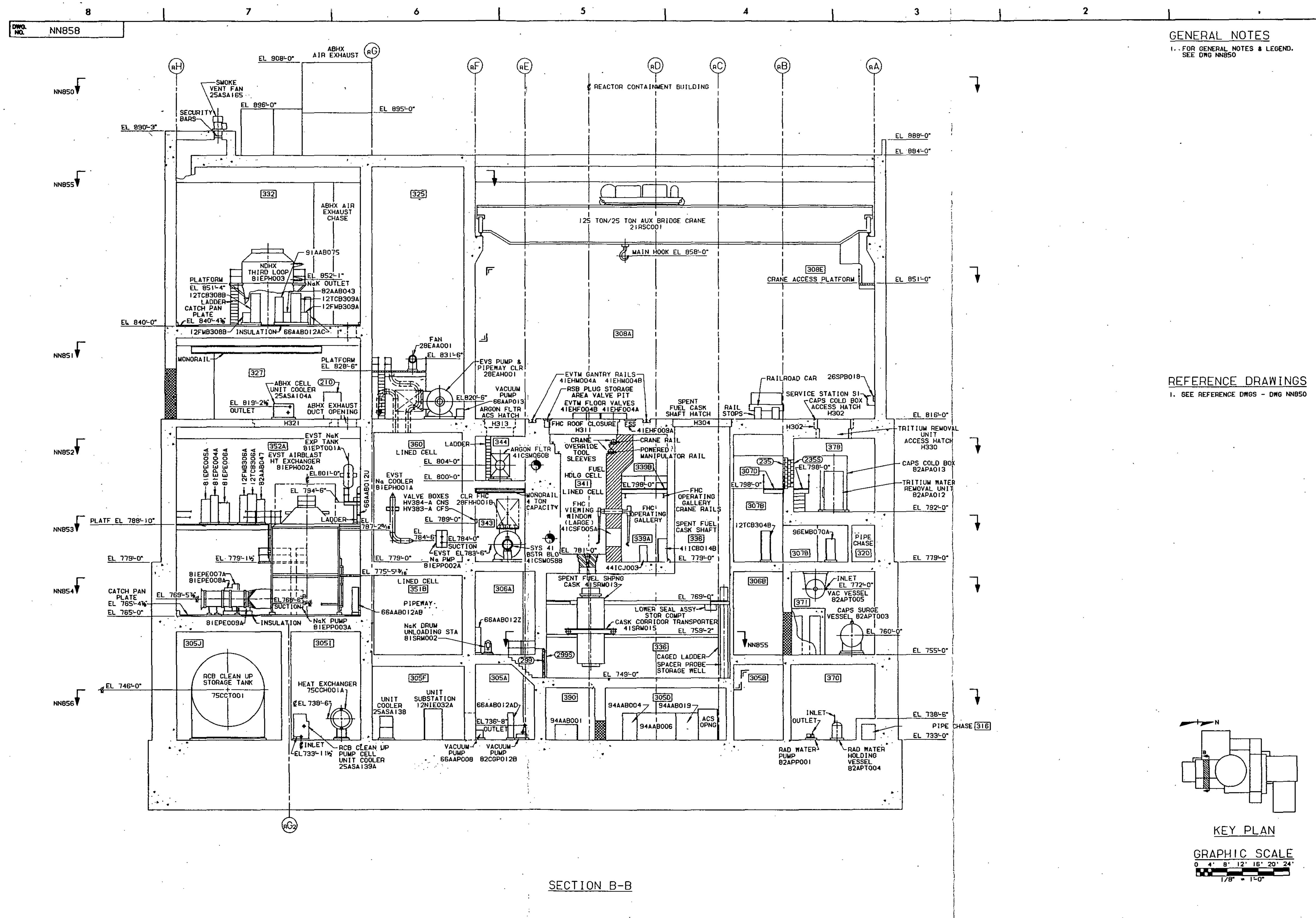



Figure 1.2-29
General Arrangement
Reactor Service Building
Section B-B

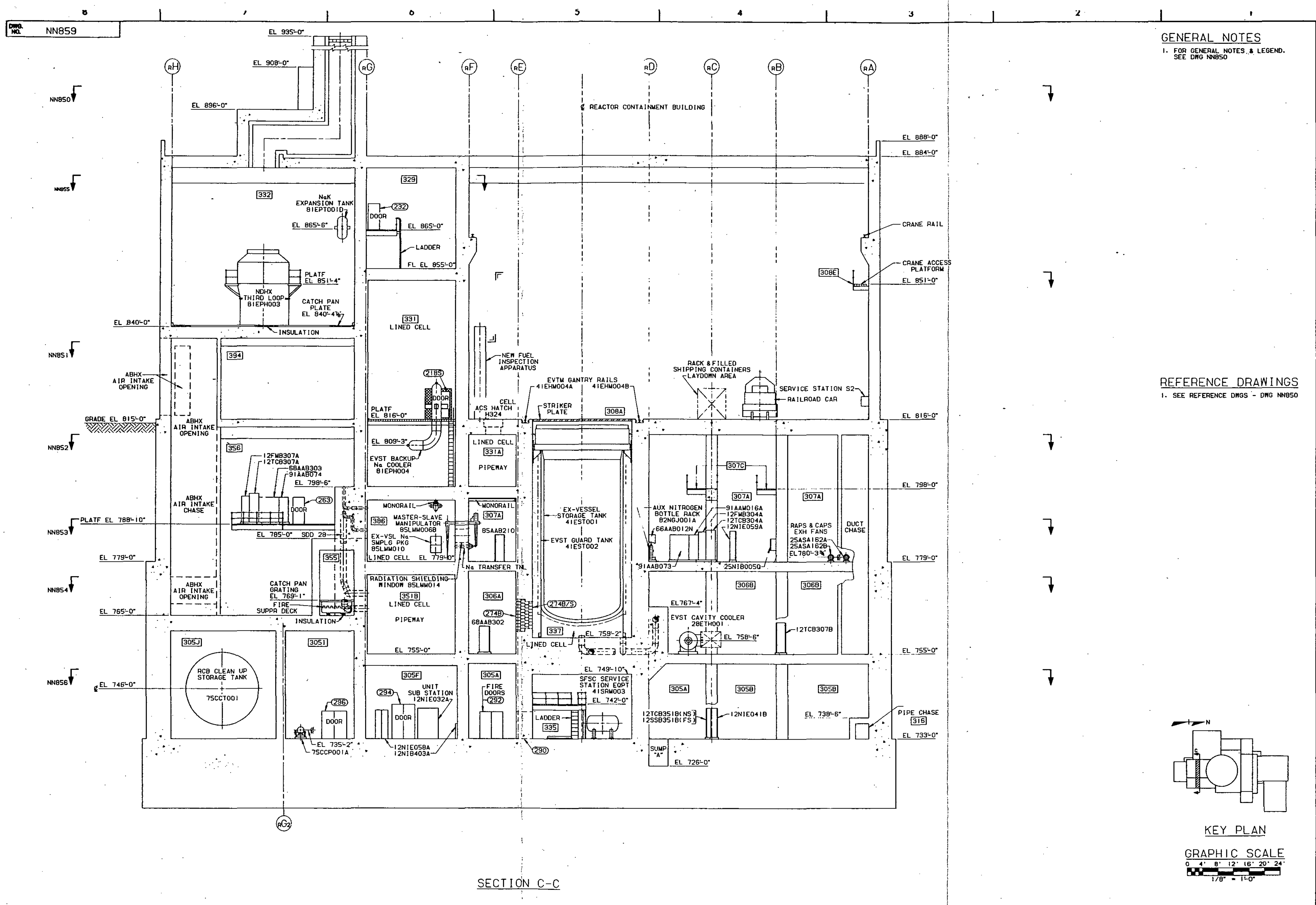
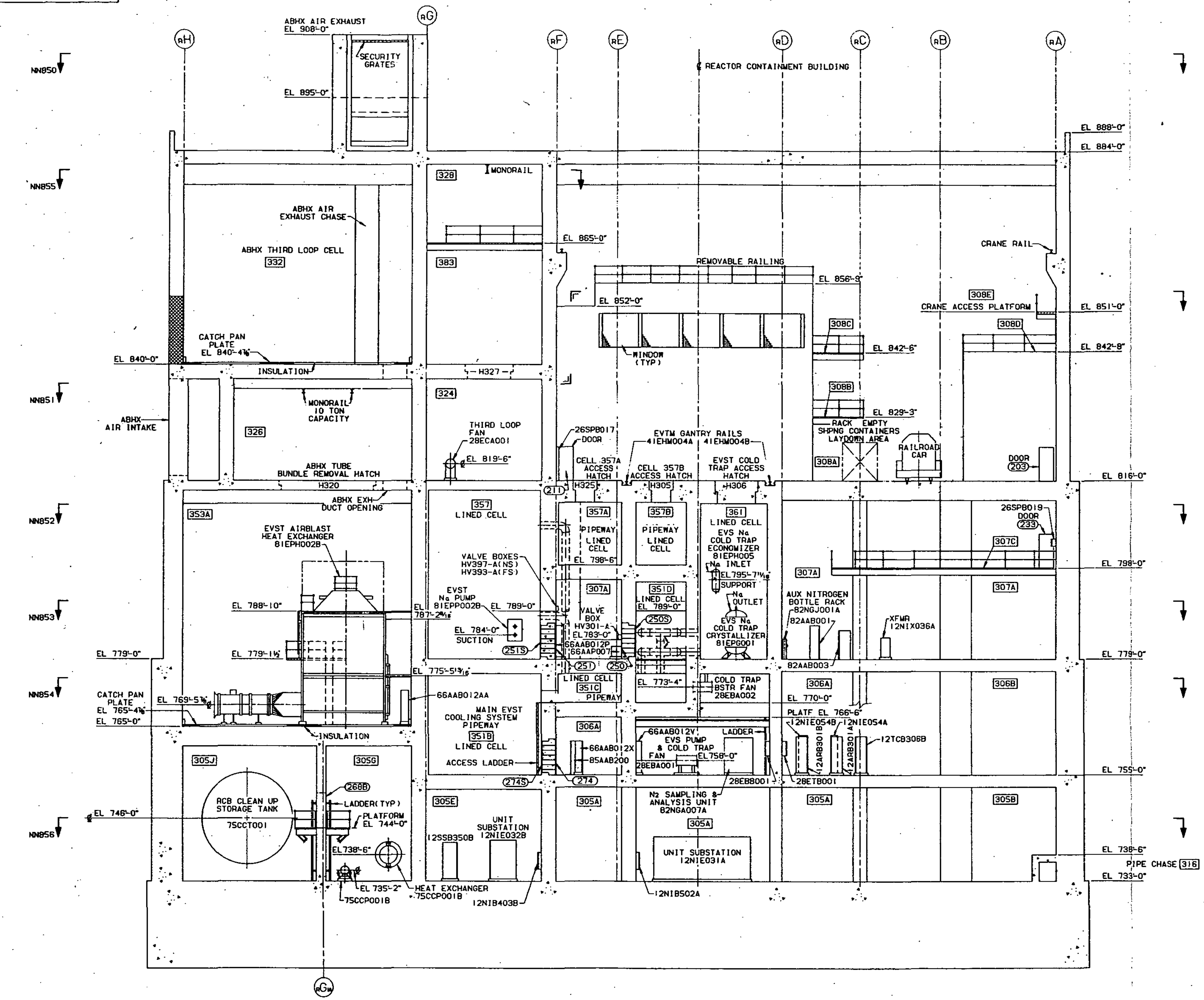



Figure 1.2-30
General Arrangement
Reactor Service Building
Section C-C

DWG. NO. NN860

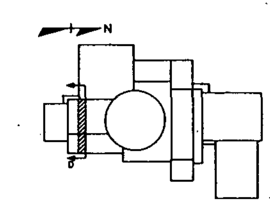



GENERAL NOTES

1. FOR GENERAL NOTES & LEGEND, SEE DWG NN850

REFERENCE DRAWINGS

1. SEE REFERENCE DWGS - DWG NN850



KEY PLAN

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

Figure 1.2-31
General Arrangement
Reactor Service Building
Section D-D

Amend. 64

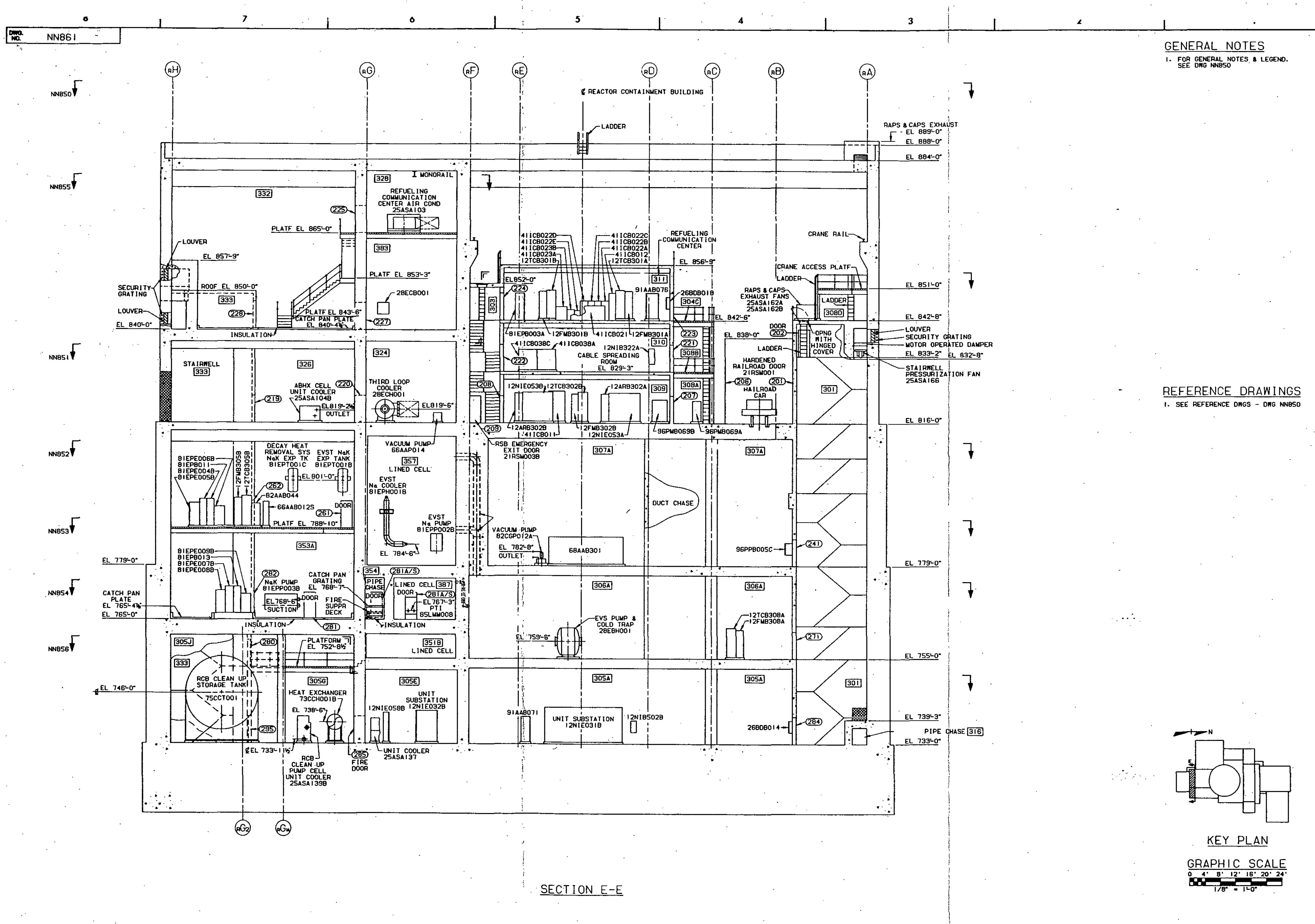



Figure 1.2-32
General Arrangement
Reactor Service Building
Section E-E

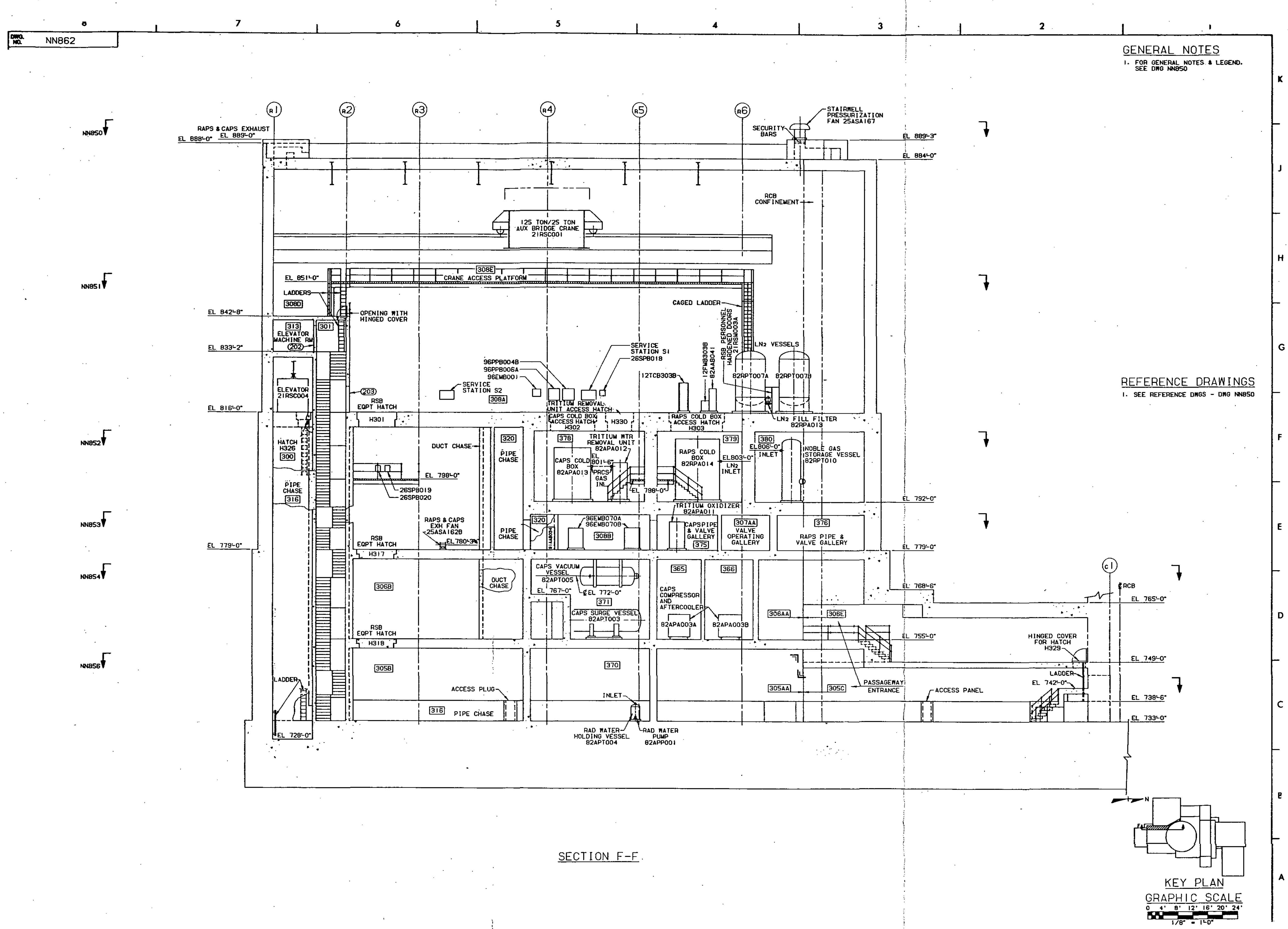
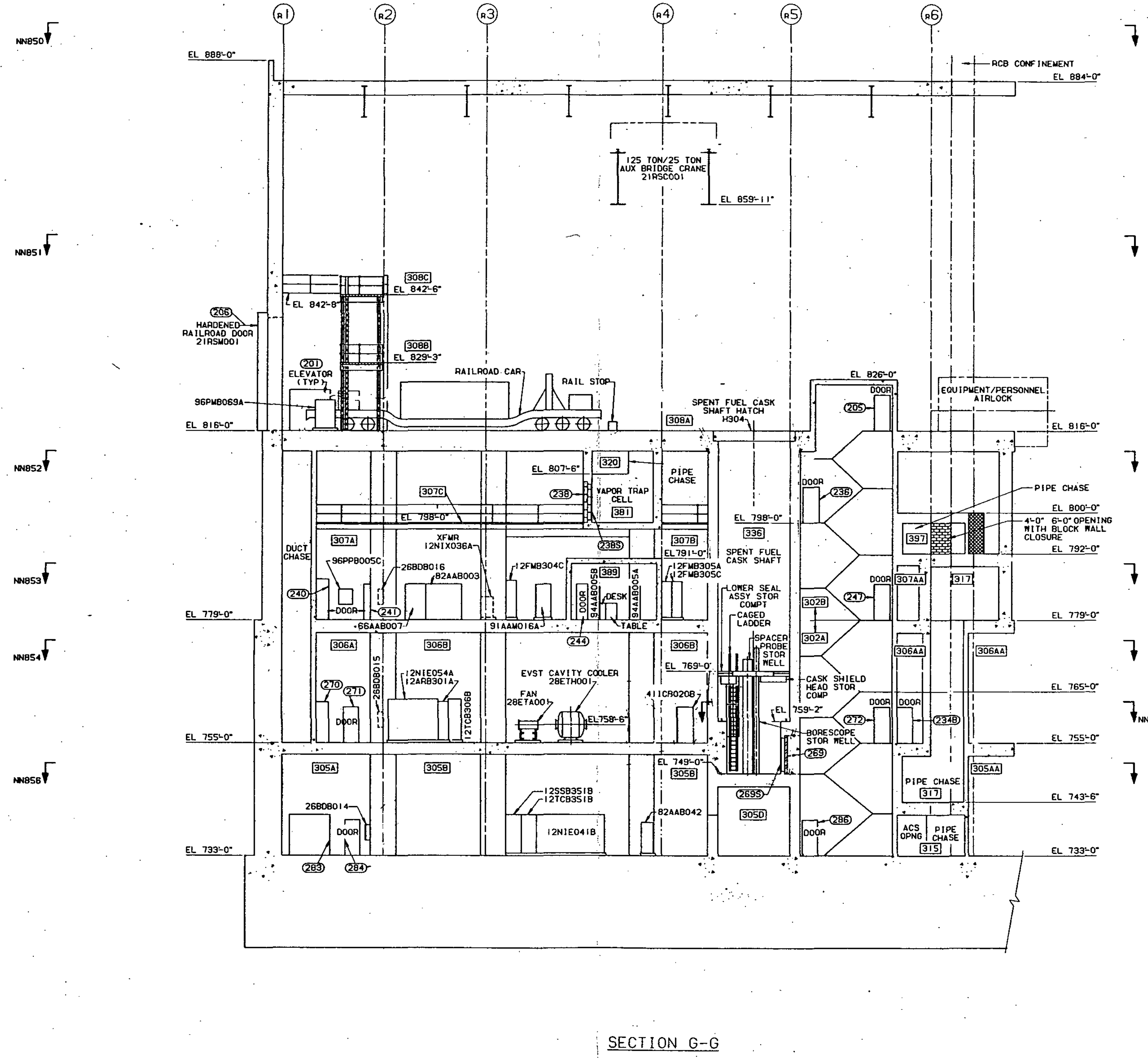



Figure 1.2-33
General Arrangement
Reactor Service Building
Section F-F

DWG. NO. NN863

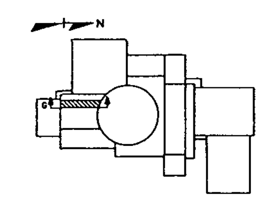



GENERAL NOTES

1. FOR GENERAL NOTES & LEGEND, SEE DWG NN850

REFERENCE DRAWINGS

1. SEE REFERENCE DWGS - DWG NN850



KEY PLAN

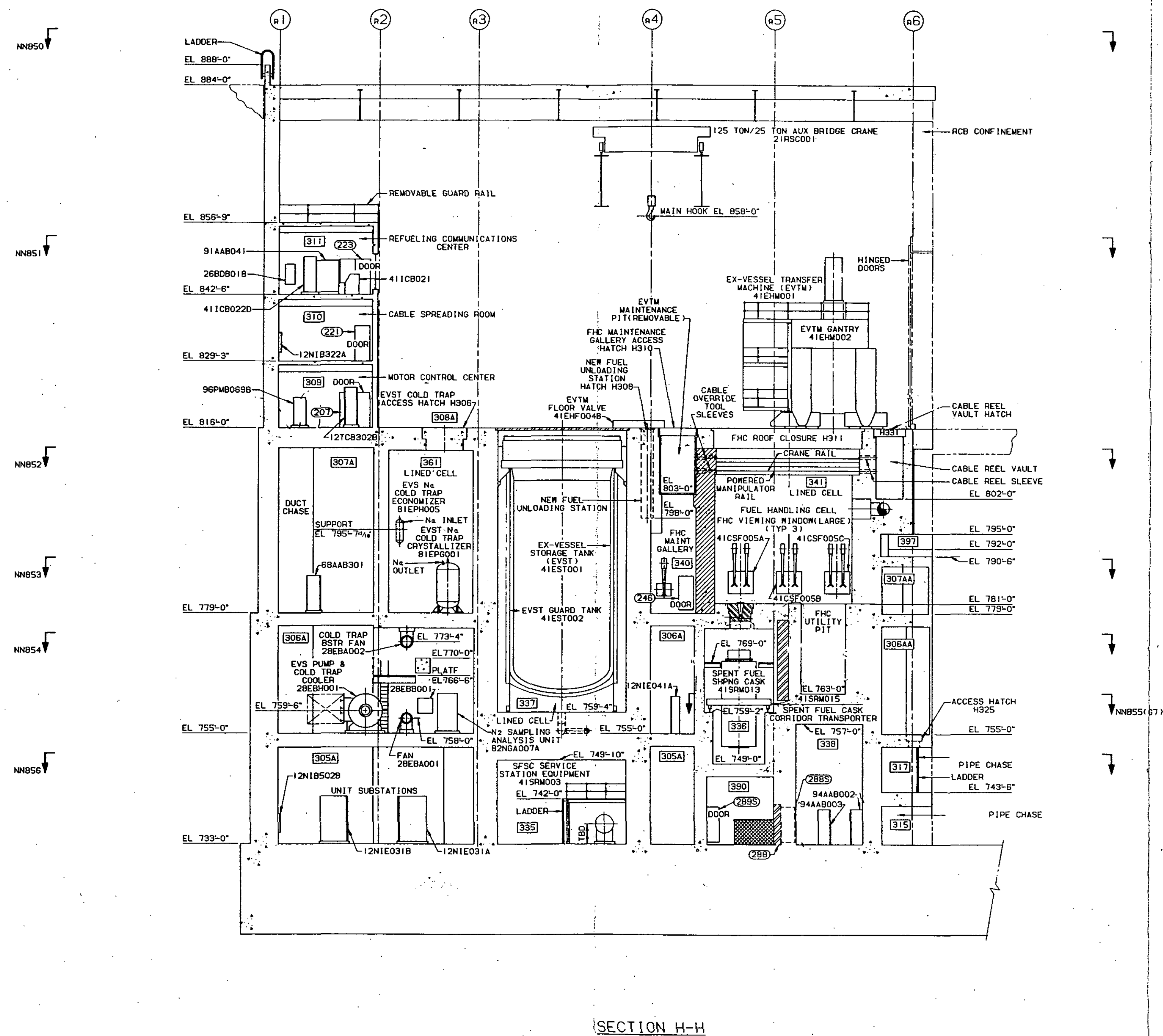
GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

Figure 1.2-34
General Arrangement
Reactor Service Building
Section G-G

DWG. NO. NN864

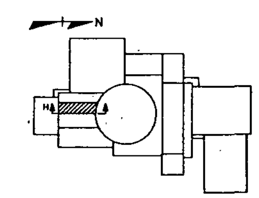



## GENERAL NOTES

1. FOR GENERAL NOTES & LEGEND. SEE DWG NN850

## REFERENCE DRAWINGS

1. SEE REFERENCE DWGS - DWG NN850




KEY PLAN

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

Figure 1.2-35
General Arrangement
Reactor Service Building
Section H-H

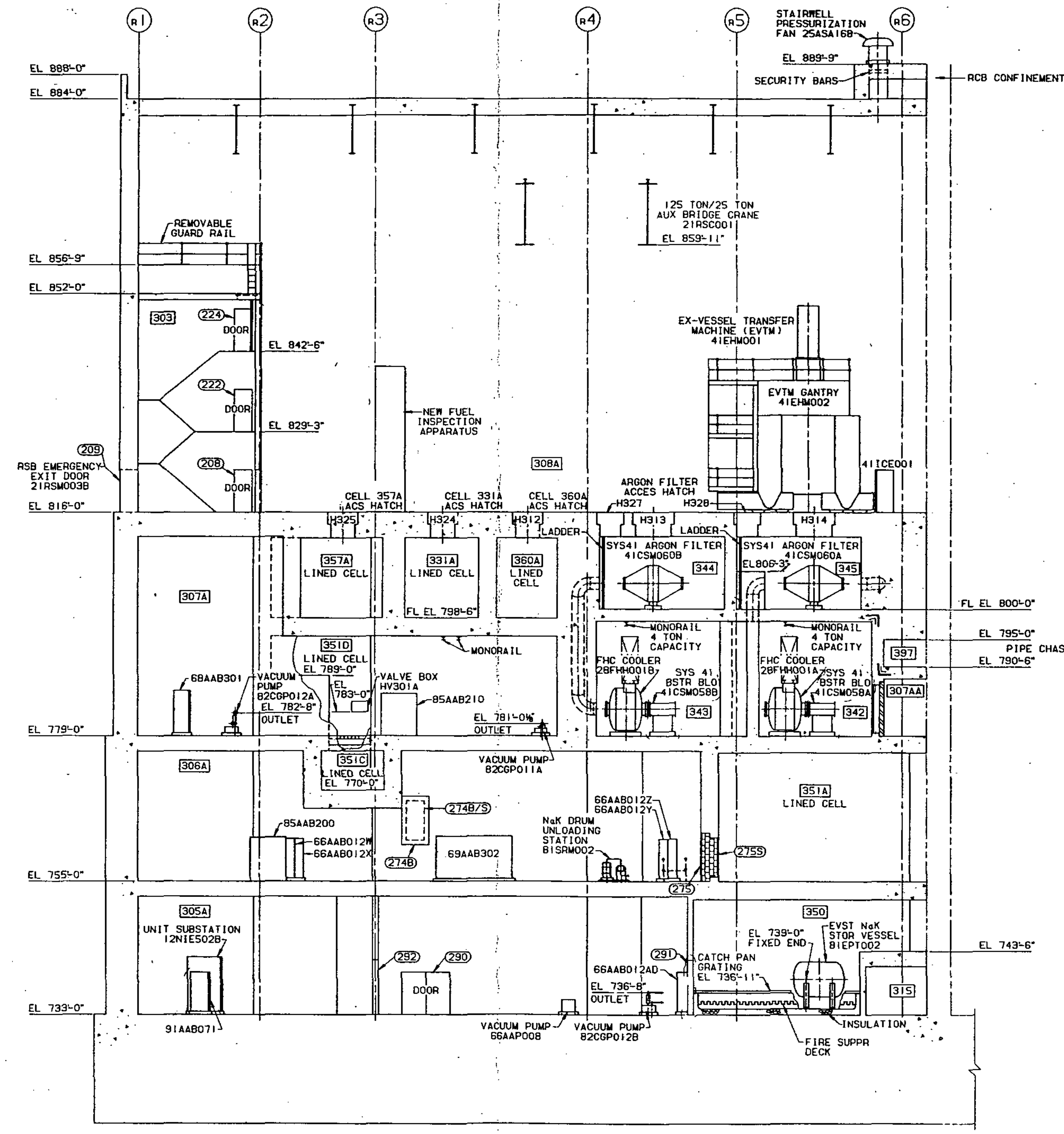
DWG. NO. NN865

## GENERAL NOTES

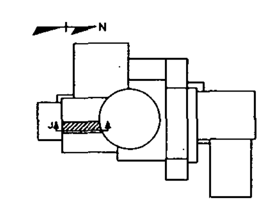
1. FOR GENERAL NOTES & LEGEND, SEE DWG NN850

## REFERENCE DRAWINGS

1. SEE REFERENCE DWGS - DWG NN850




SECTION J-J



KEY PLAN

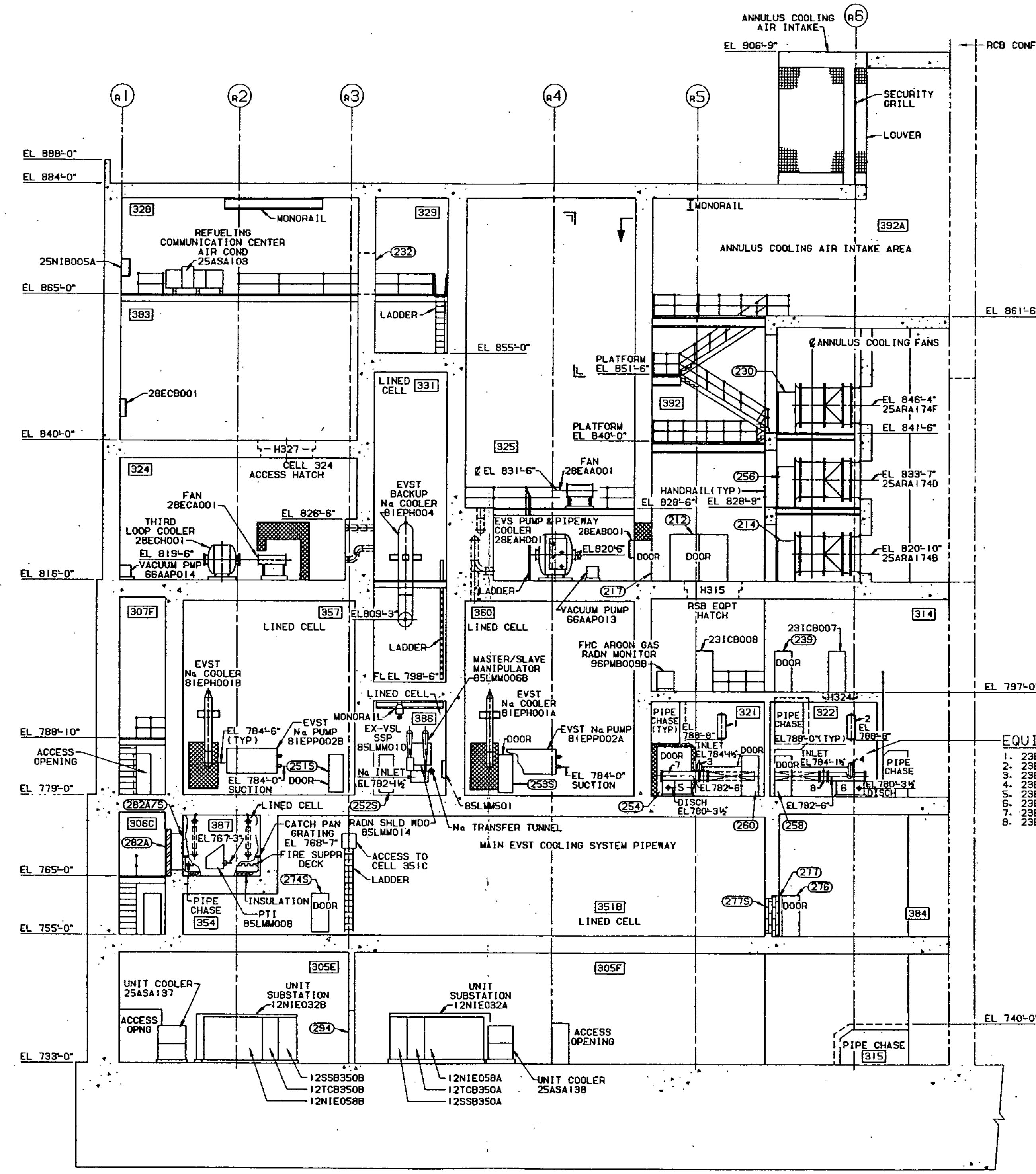
GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

Figure 1.2-36
General Arrangement
Reactor Service Building
Section J-J

DWG. NO. NN866




SECTION K-K

EQUIPMENT LIST: CELL 321 & 322

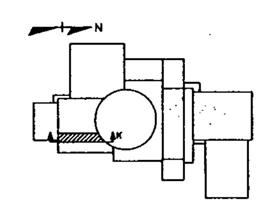
1. 23ECT007B FHC DTJ EXPANSION TANK
2. 23ECT007A
3. 23ECM008B FHC DTJ AIR SEPARATOR
4. 23ECM008A
5. 23ECP006B FHC DTJ PUMP
6. 23ECP006A
7. 23ECH005B FHC DTJ WATER HX
8. 23ECH005A

GENERAL NOTES

1. FOR GENERAL NOTES & LEGEND. SEE DWG NN850

REFERENCE DRAWINGS

1. SEE REFERENCE DWGS - DWG NN850



KEY PLAN

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

Figure 1.2-37
General Arrangement
Reactor Service Building
Section K-K

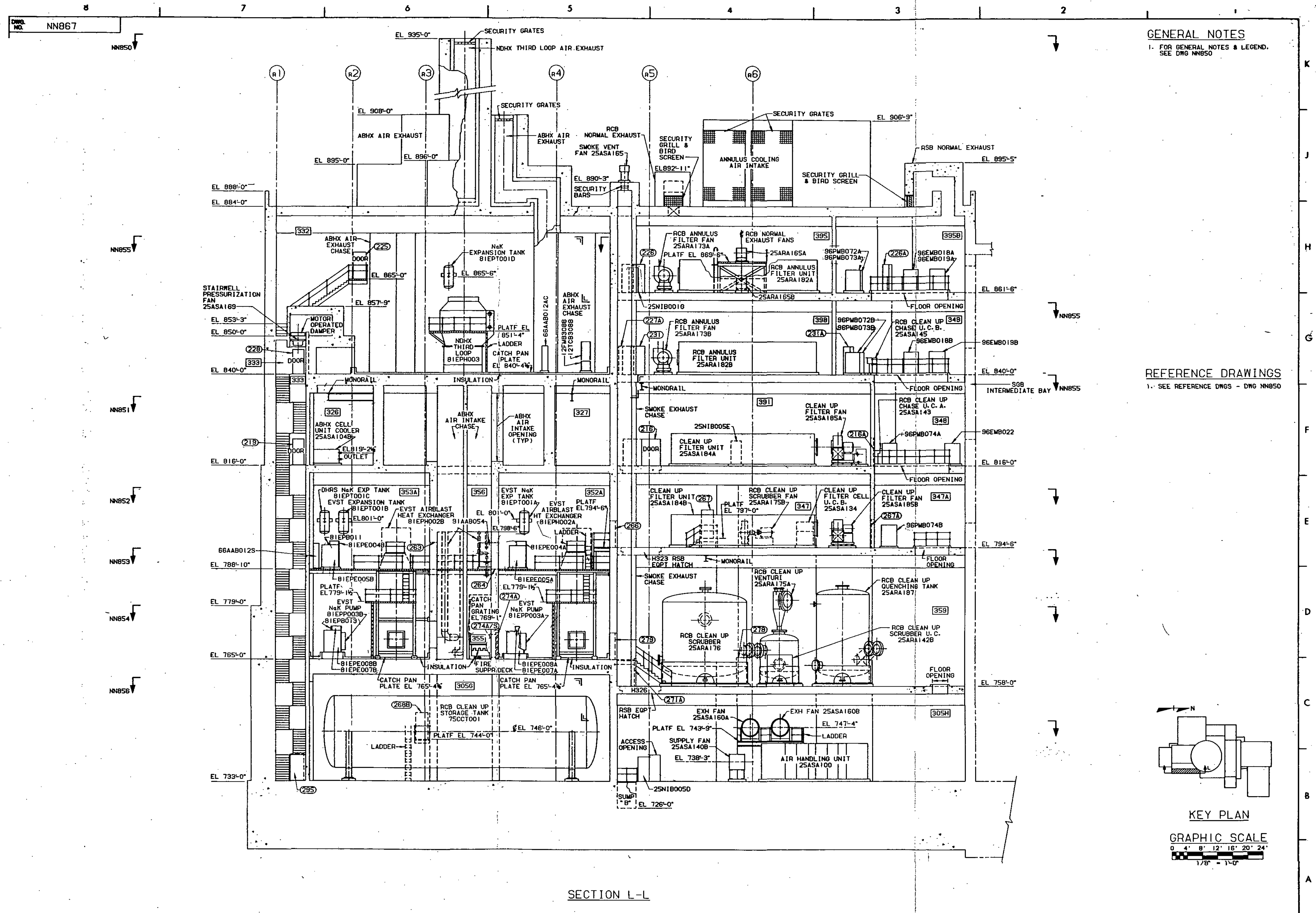



Figure 1.2-38
General Arrangement
Reactor Service Building
Section L-L


Amend. 64
Jan. 1982

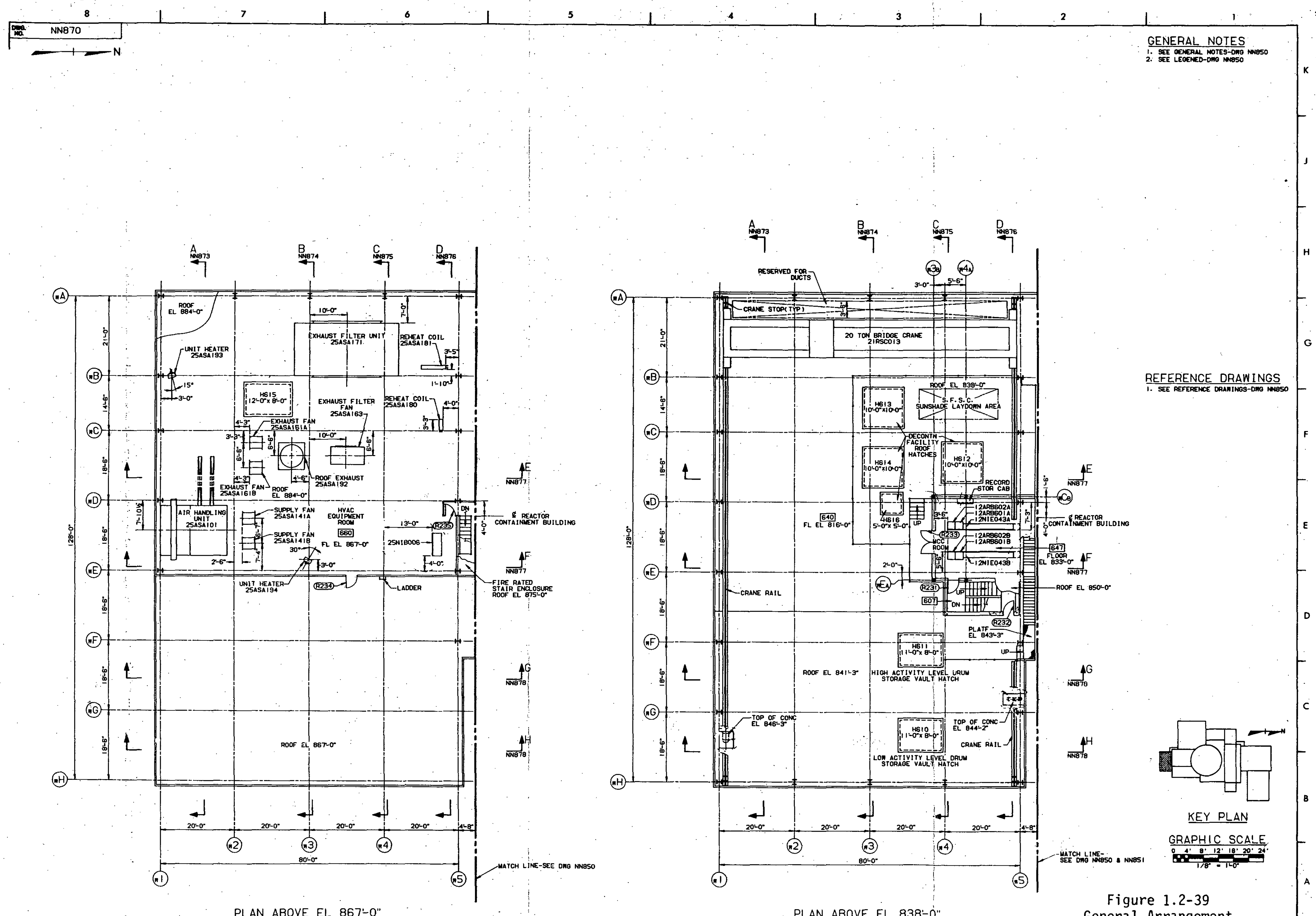



Figure 1.2-39
General Arrangement
Reactor Service Building
Radwaste Area
El. 867'-0" and 838'-0"

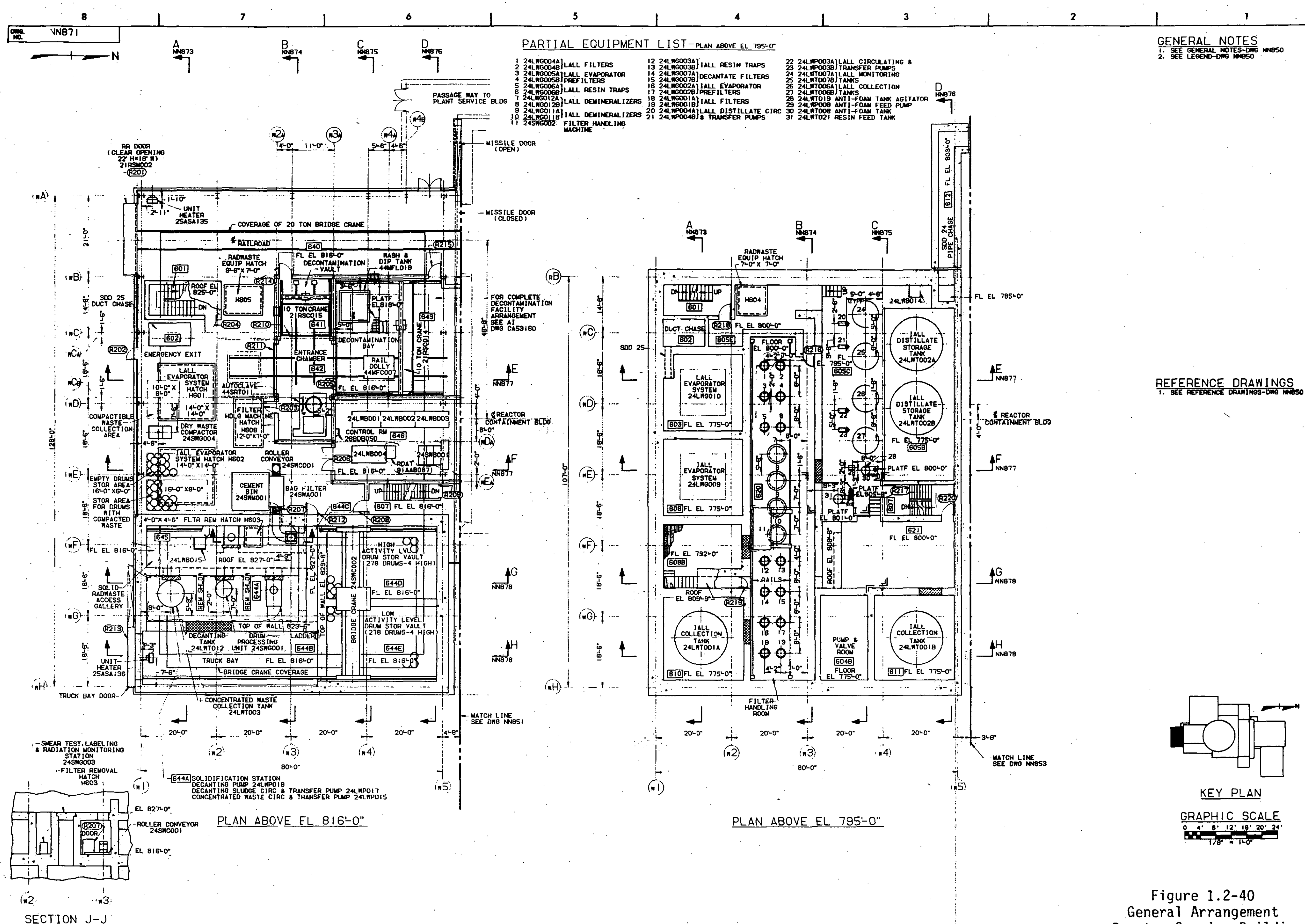



Figure 1.2-40
General Arrangement
Reactor Service Building
Radwaste Area
El. 816'-0" and 795'-0"

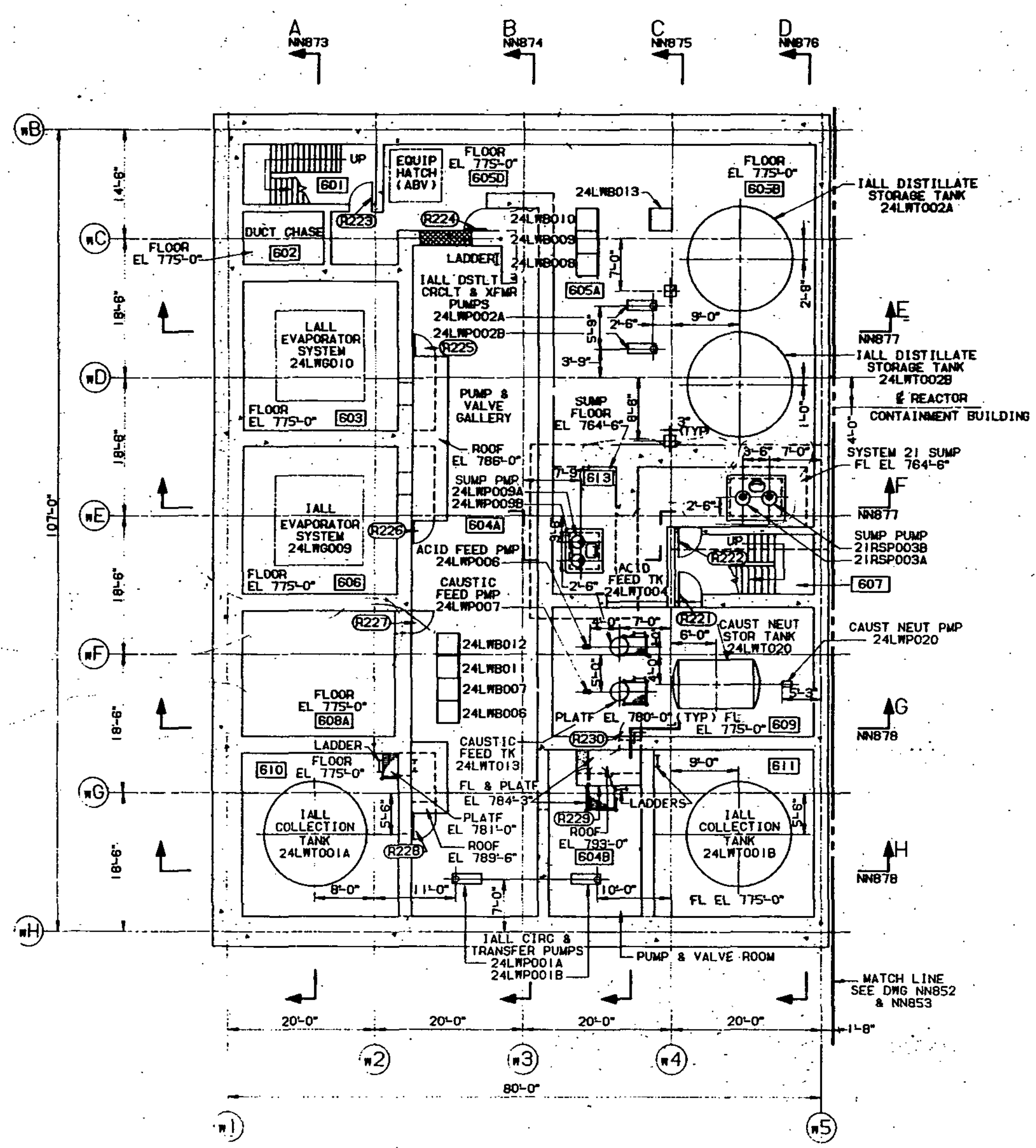
DWG. NO. NN872

## GENERAL NOTES

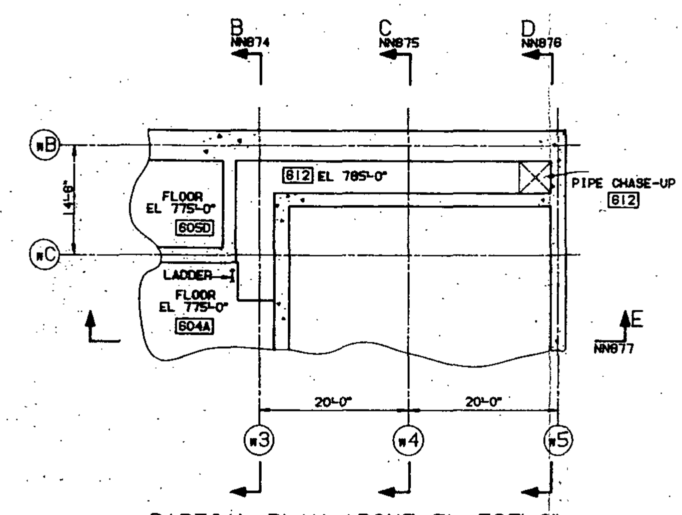
1. SEE GENERAL NOTES-DWG NN850
2. SEE LEGEND-DWG NN850

## REFERENCE DRAWINGS

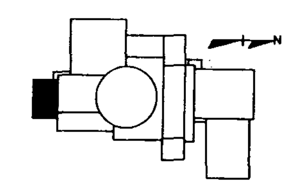
1. SEE REFERENCE DRAWINGS-DWG NN850




PLAN ABOVE EL 775'-0"




PARTIAL PLAN ABOVE EL 785'-0"



KEY PLAN

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

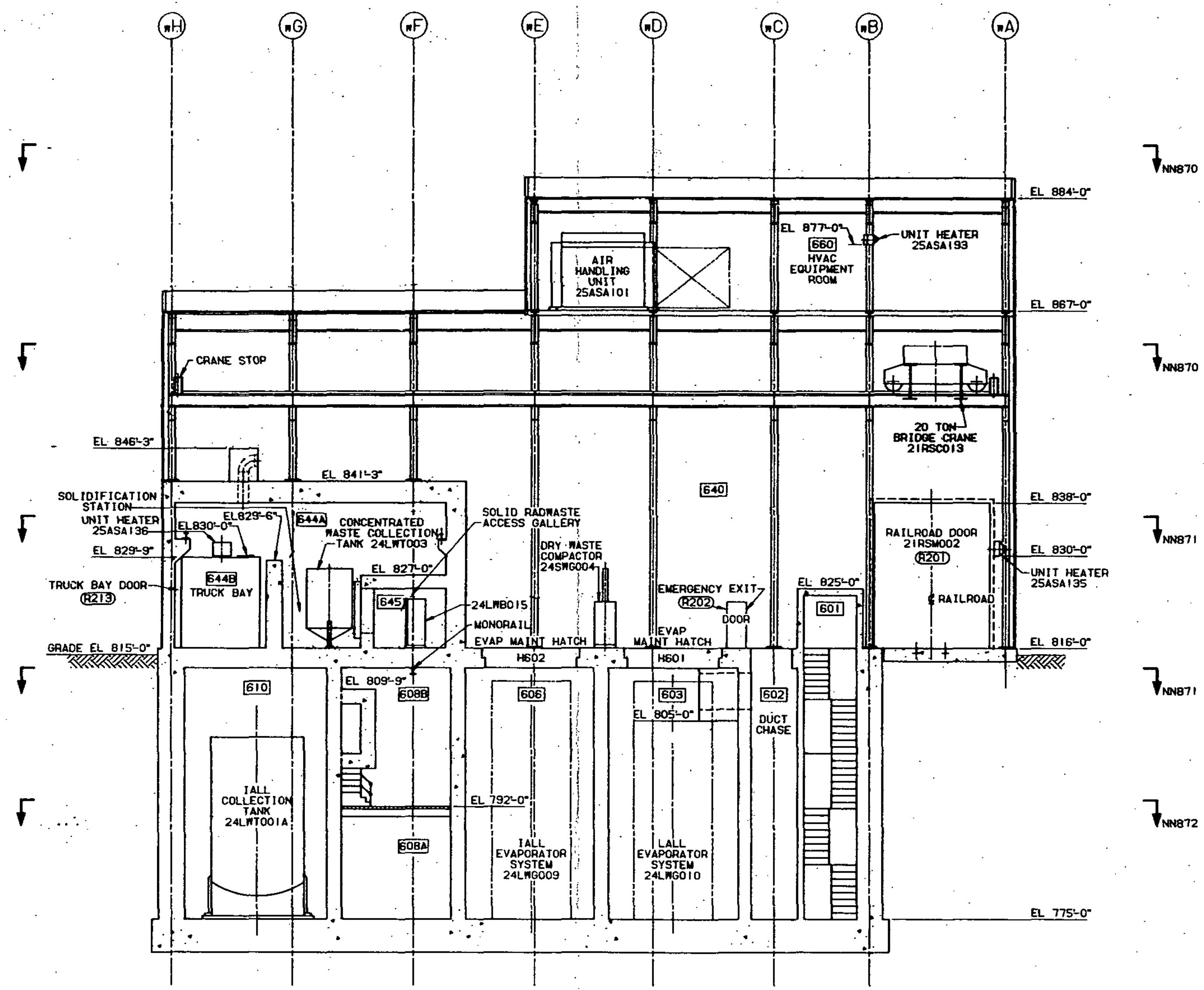
1/8" = 1'-0"

Figure 1.2-41
General Arrangement
Reactor Service Building
Radwaste Area
Plan El. 775'-0"

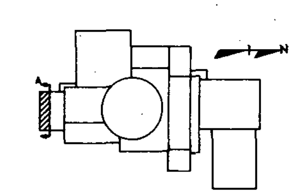
DWG. NO. NN873

GENERAL NOTES
1. SEE GENERAL NOTES-DWG NN850
2. SEE LEGEND-DWG NN850

REFERENCE DRAWINGS
1. SEE REFERENCE DRAWINGS-DWG NN850




SECTION A-A



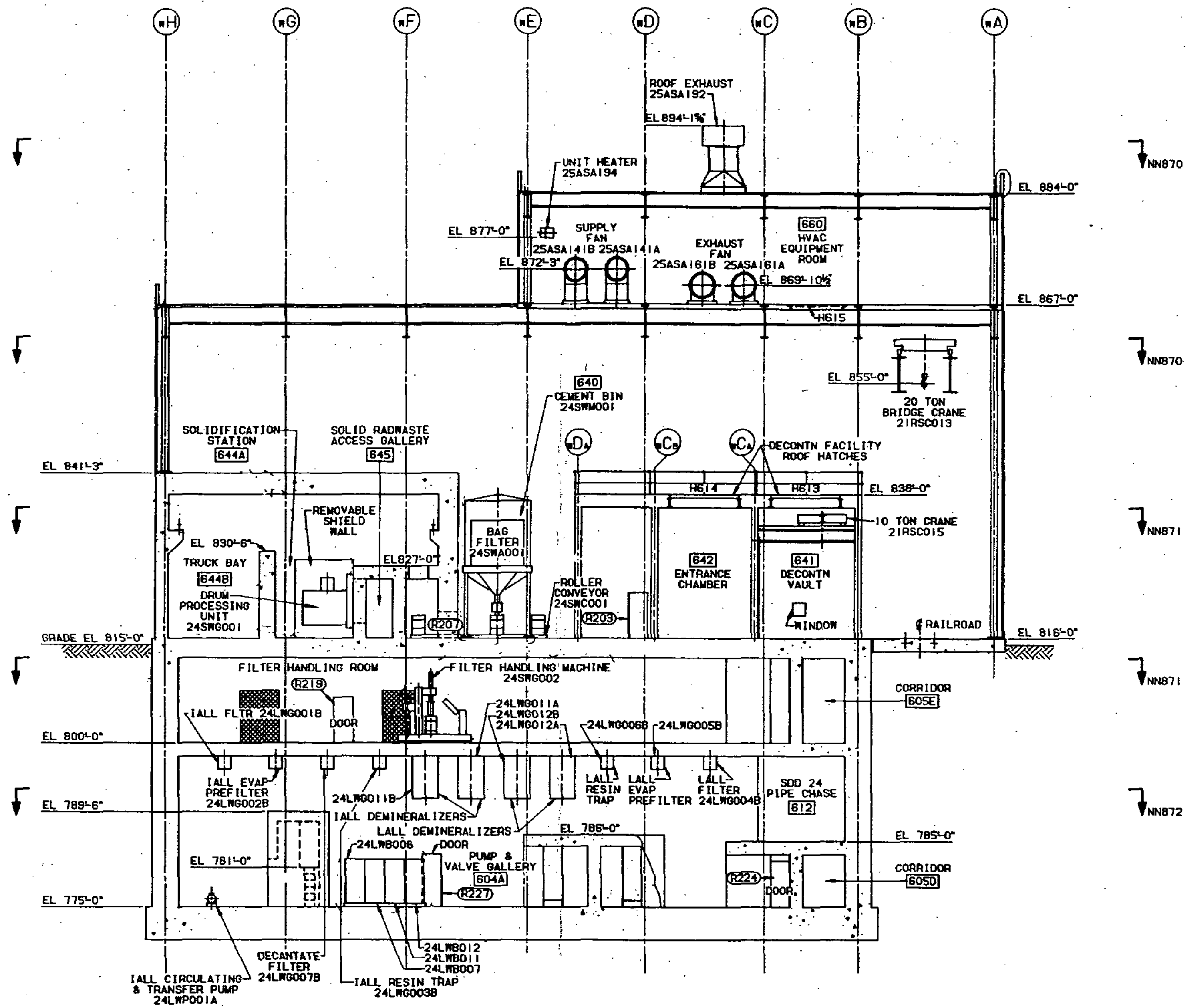

GRAPHIC SCALE
0 4' 8' 12' 16' 20' 24'
1/8" = 1'-0"

Figure 1.2-42
General Arrangement
Reactor Service Building
Radwaste Area
Section A-A

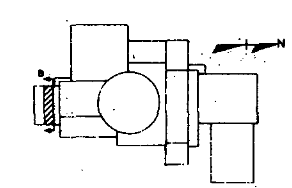
DWG. NO. NN874

GENERAL NOTES
1. SEE GENERAL NOTES-DWG NN850
2. SEE LEGEND-DWG NN850

REFERENCE DRAWINGS
1. SEE REFERENCE DRAWINGS-DWG NN850




SECTION B-B



KEY PLAN

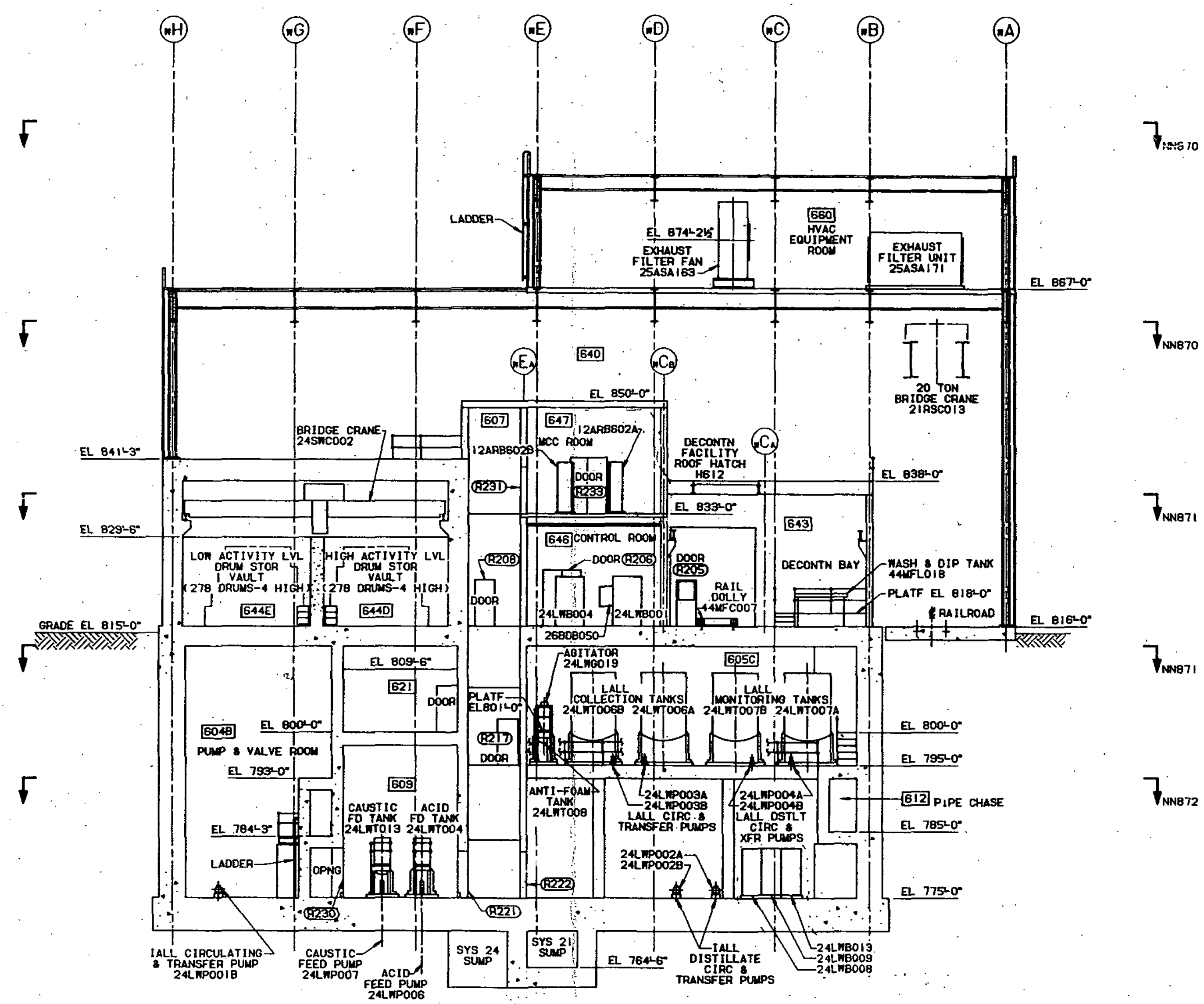
GRAPHIC SCALE
0 4' 8' 12' 16' 20' 24'
1/8" = 1'-0"

Figure 1.2-43
General Arrangement
Reactor Service Building
Radwaste Area
Section B-B

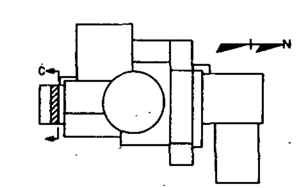
NN875

GENERAL NOTES
1. SEE GENERAL NOTES-DWG NN850
2. SEE LEGEND-DWG NN850

REFERENCE DRAWINGS
1. SEE REFERENCE DRAWINGS-DWG NN850




SECTION C-C



KEY PLAN

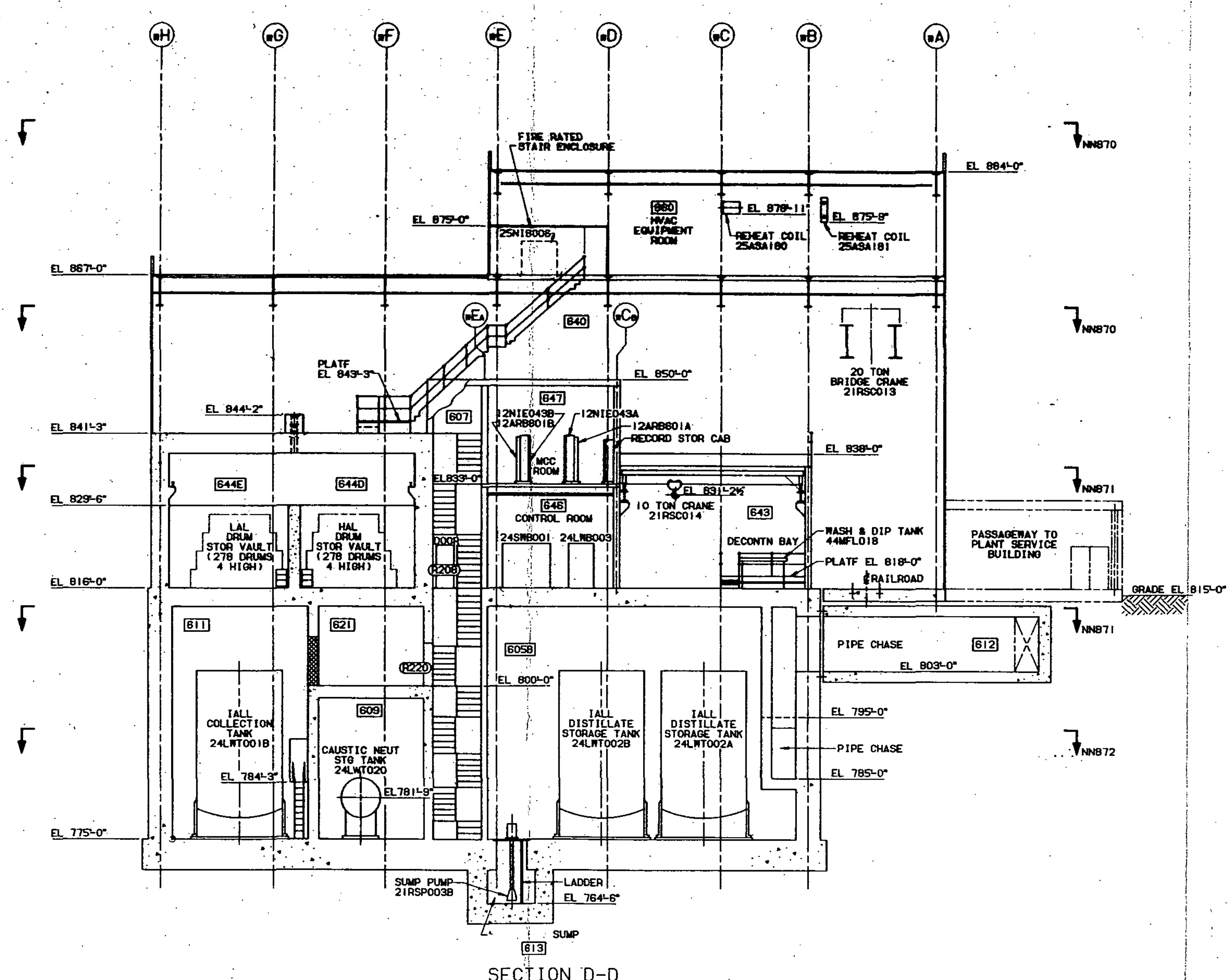
GRAPHIC SCALE
0 4' 8' 12' 16' 20' 24'
1/8" = 1'-0"

Figure 1.2-44
General Arrangement
Reactor Service Building
Radwaste Area
Section C-C

NN876

GENERAL NOTES
1. SEE GENERAL NOTES-DWG NN850
2. SEE LEGEND-DWG NN850

REFERENCE DRAWINGS
1. SEE REFERENCE DRAWINGS-DWG NN850




KEY PLAN

GRAPHIC SCALE
0 4' 8' 12' 16' 20' 24'
1/8" = 1'-0"

Figure 1.2-45
General Arrangement
Reactor Service Building
Radwaste Area
Section D-D

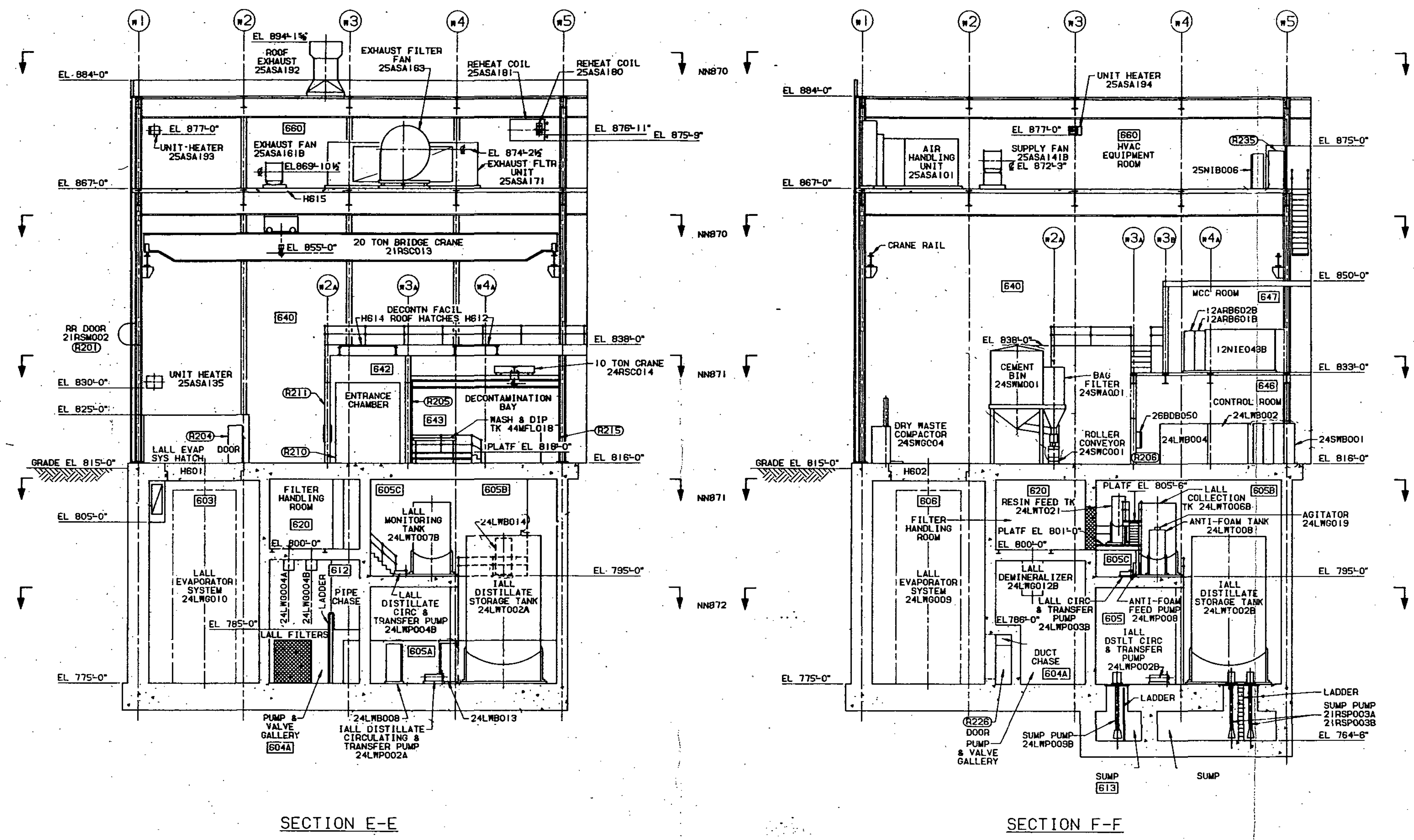
DWG. NO. NN877

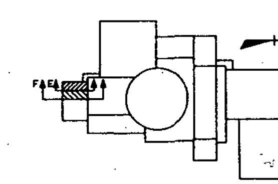
## GENERAL NOTES

1. SEE GENERAL NOTES-DWG NN850
2. SEE LEGEND-DWG NN850

## REFERENCE DRAWINGS

1. SEE REFERENCE DRAWINGS-DWG NN850



KEY PLAN

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

Figure 1.2-46
General Arrangement
Reactor Service Building
Radwaste Area
Section E-E and F-F

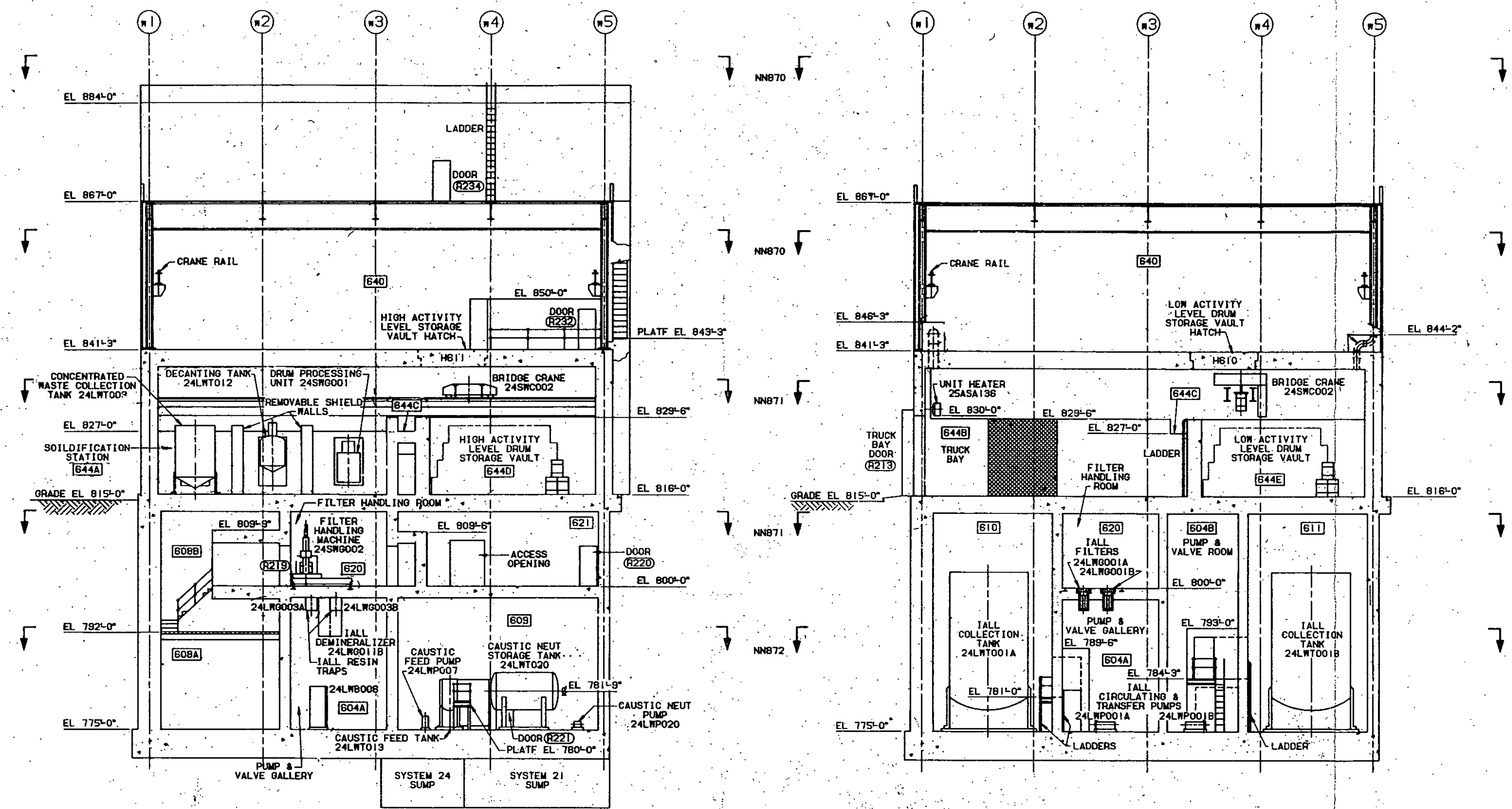
DWG. NO. NN878

GENERAL NOTES

1. SEE GENERAL NOTES-DWG NN850
2. SEE LEGEND-DWG NN850

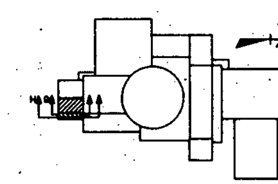
REFERENCE DRAWINGS

1. SEE REFERENCE DRAWINGS-DWG NN850




SECTION G-G

SECTION H-H



KEY PLAN

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

Figure 1.2-47
General Arrangement
Reactor Service Building
Radwaste Area
Section G-G and H-H

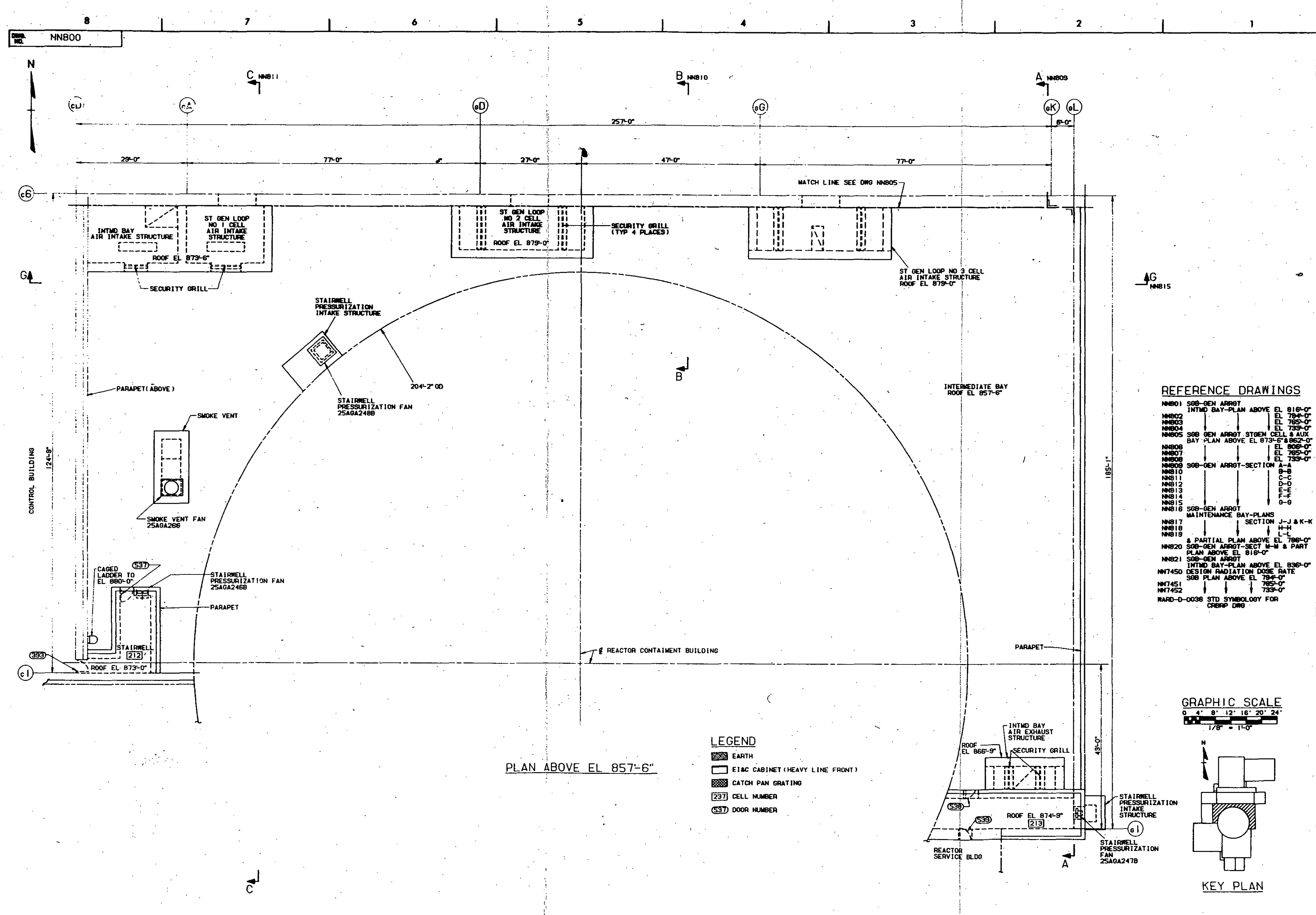



Figure 1.2-48
General Arrangement
Steam Generator Building
Plan El. 857'-6"

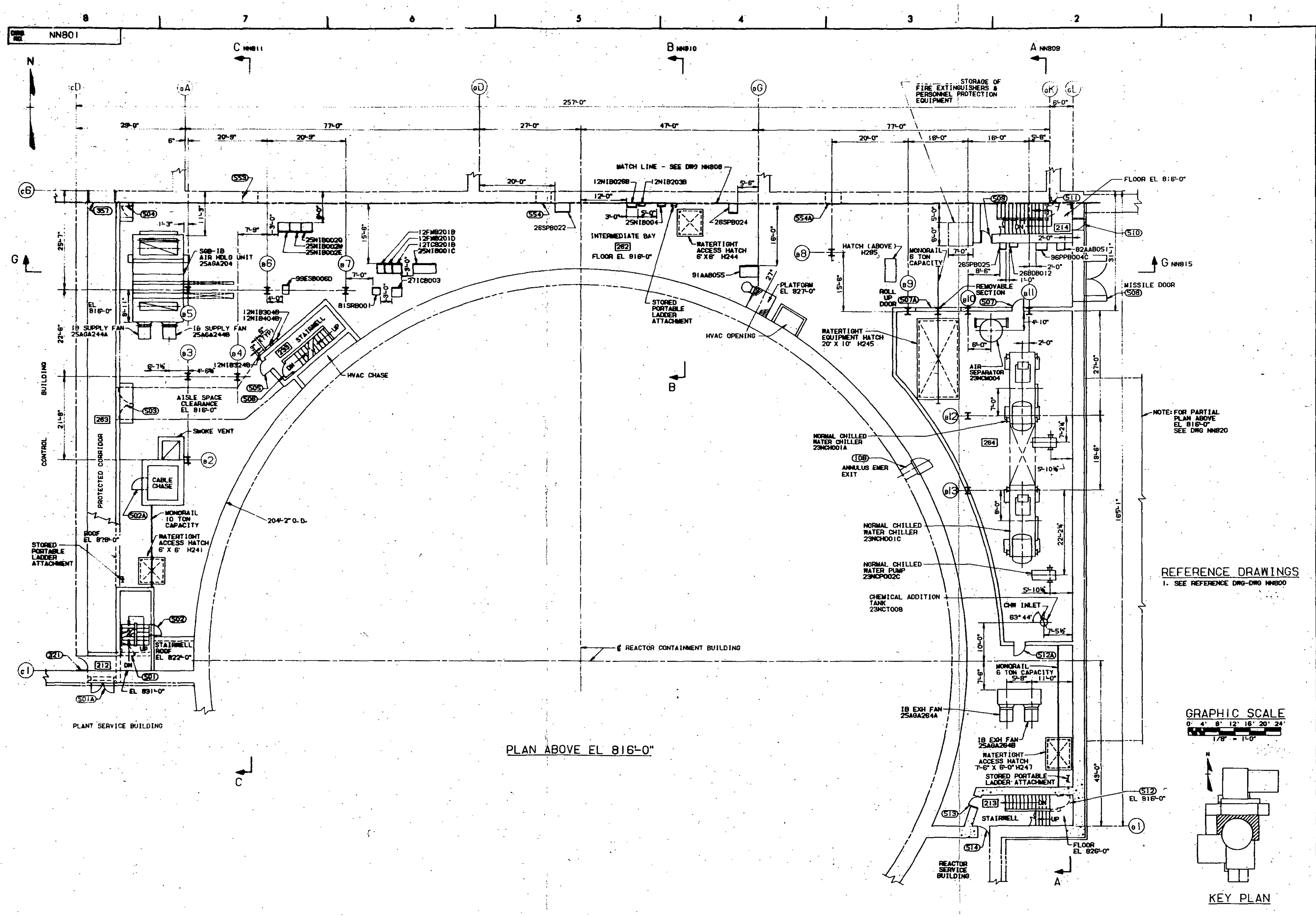



Figure 1.2-49
General Arrangement
Steam Generator Building
Plan El. 816'-0"

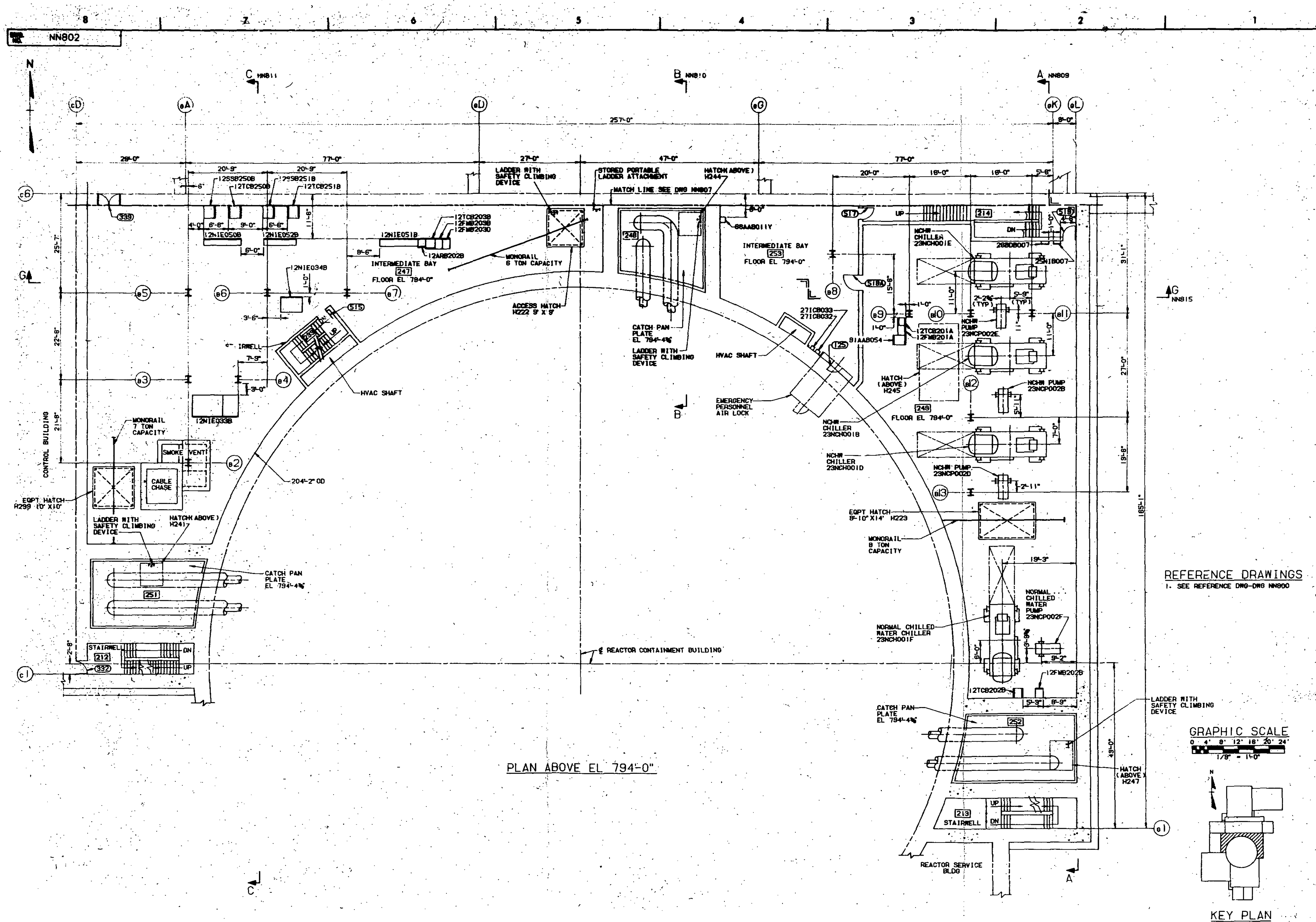



Figure 1.2-50
General Arrangement
Steam Generator Building
Plan El. 794'-0"

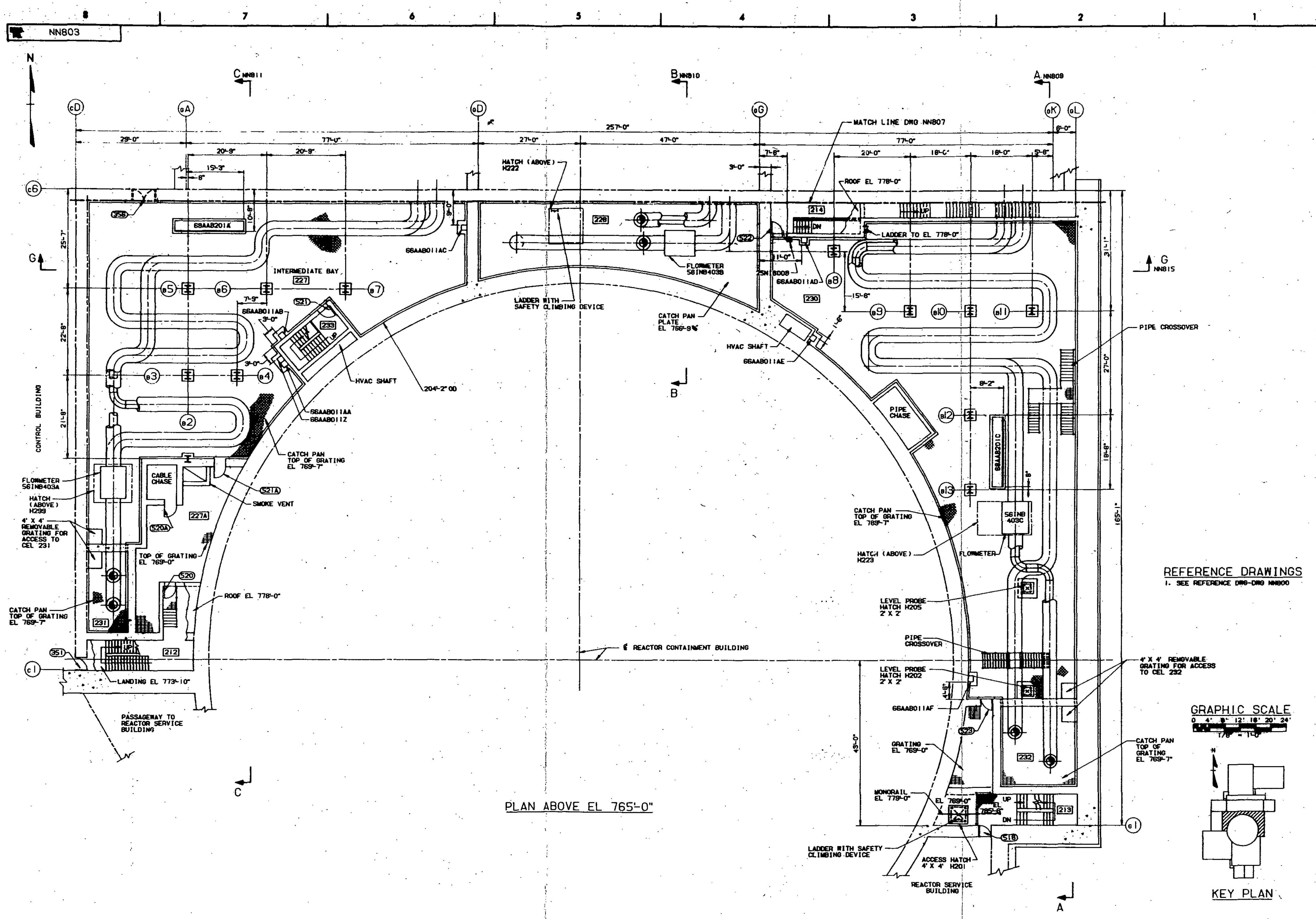



Figure 1.2-51
General Arrangement
Steam Generator Building
Plan El. 765'-0"

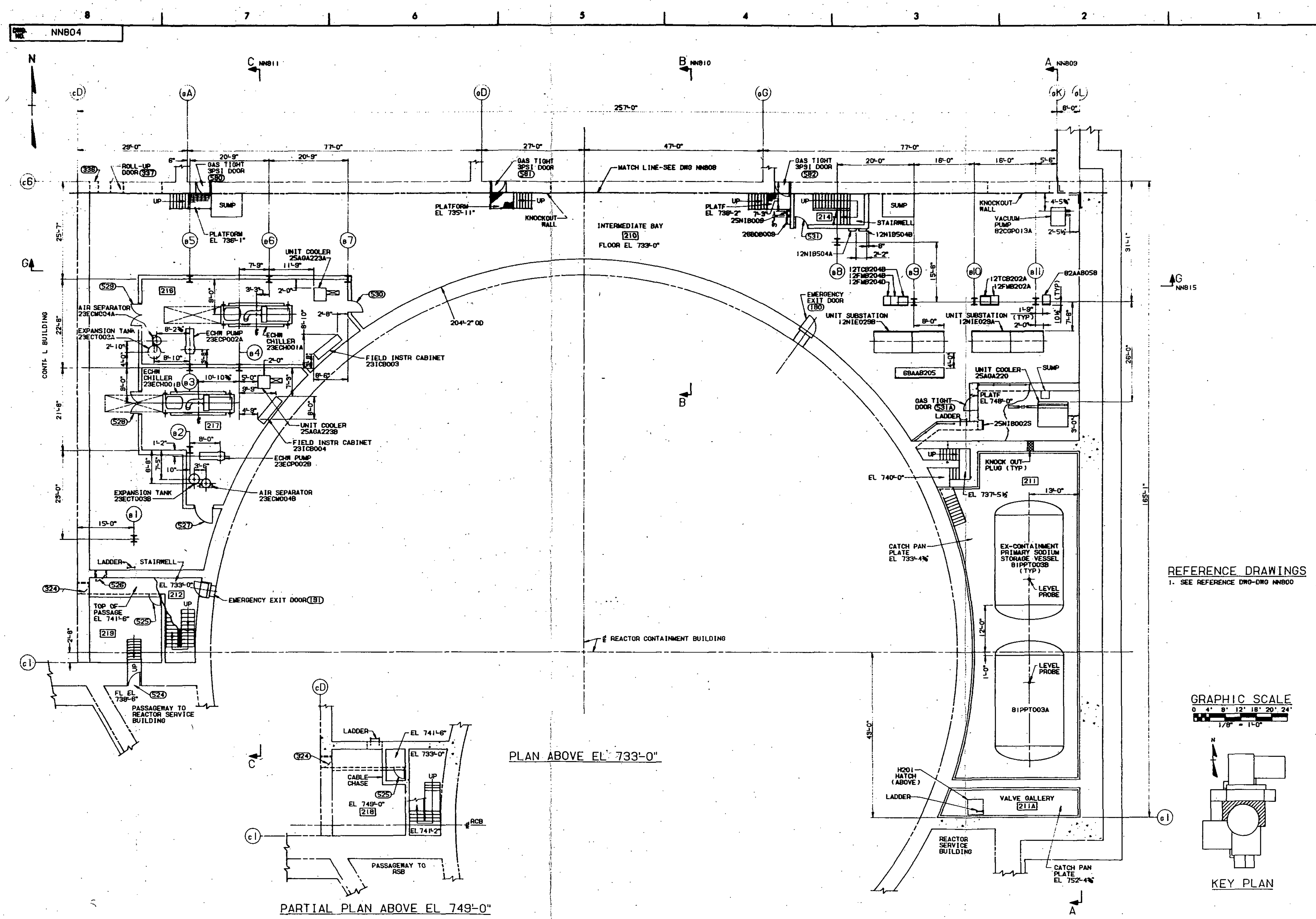



Figure 1.2-52
General Arrangement
Steam Generator Building
Plan El. 733'-0"

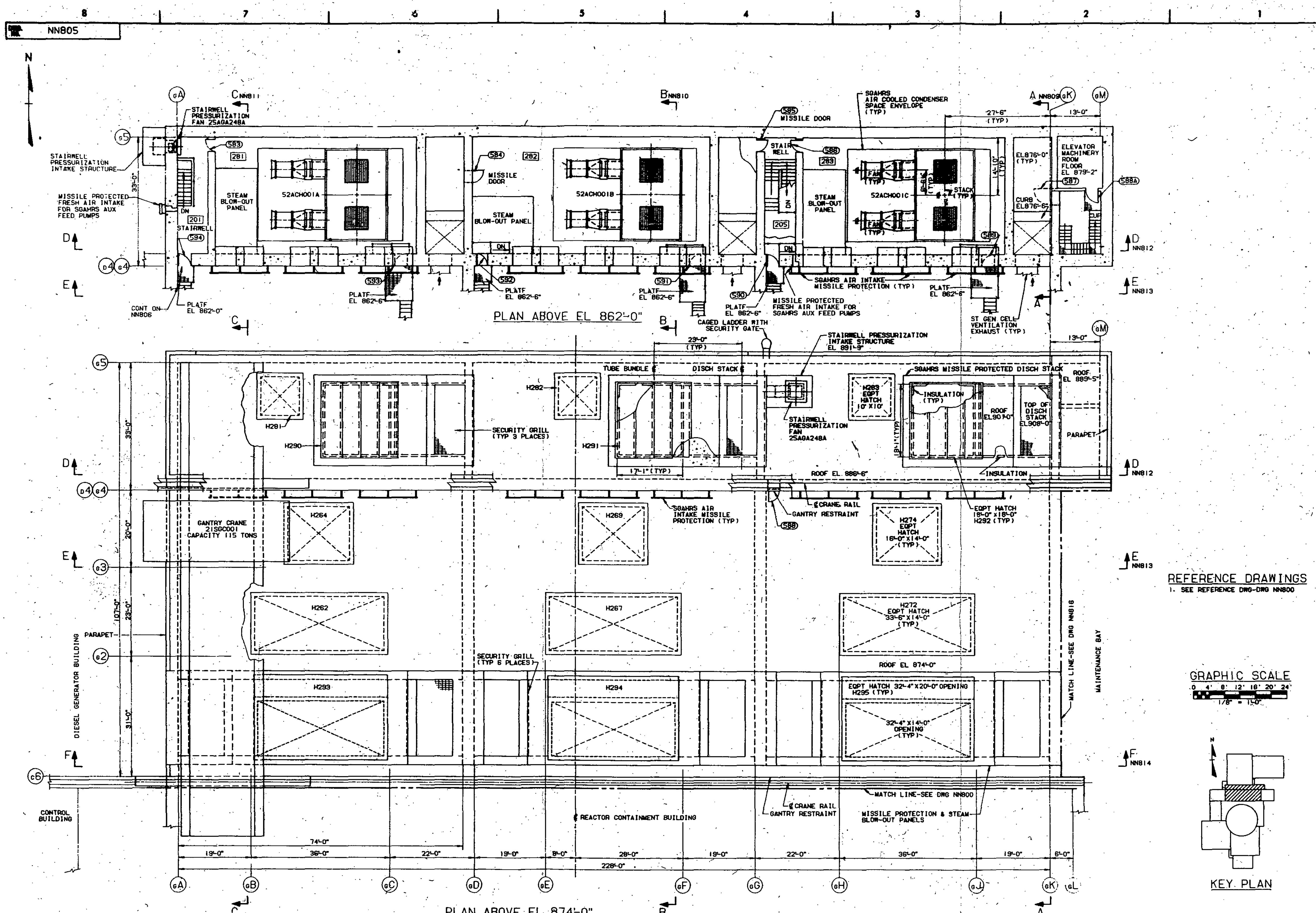



Figure 1.2-53
General Arrangement
Steam Generator Building
Plan El. 874'-0" and
El. 862'-0"

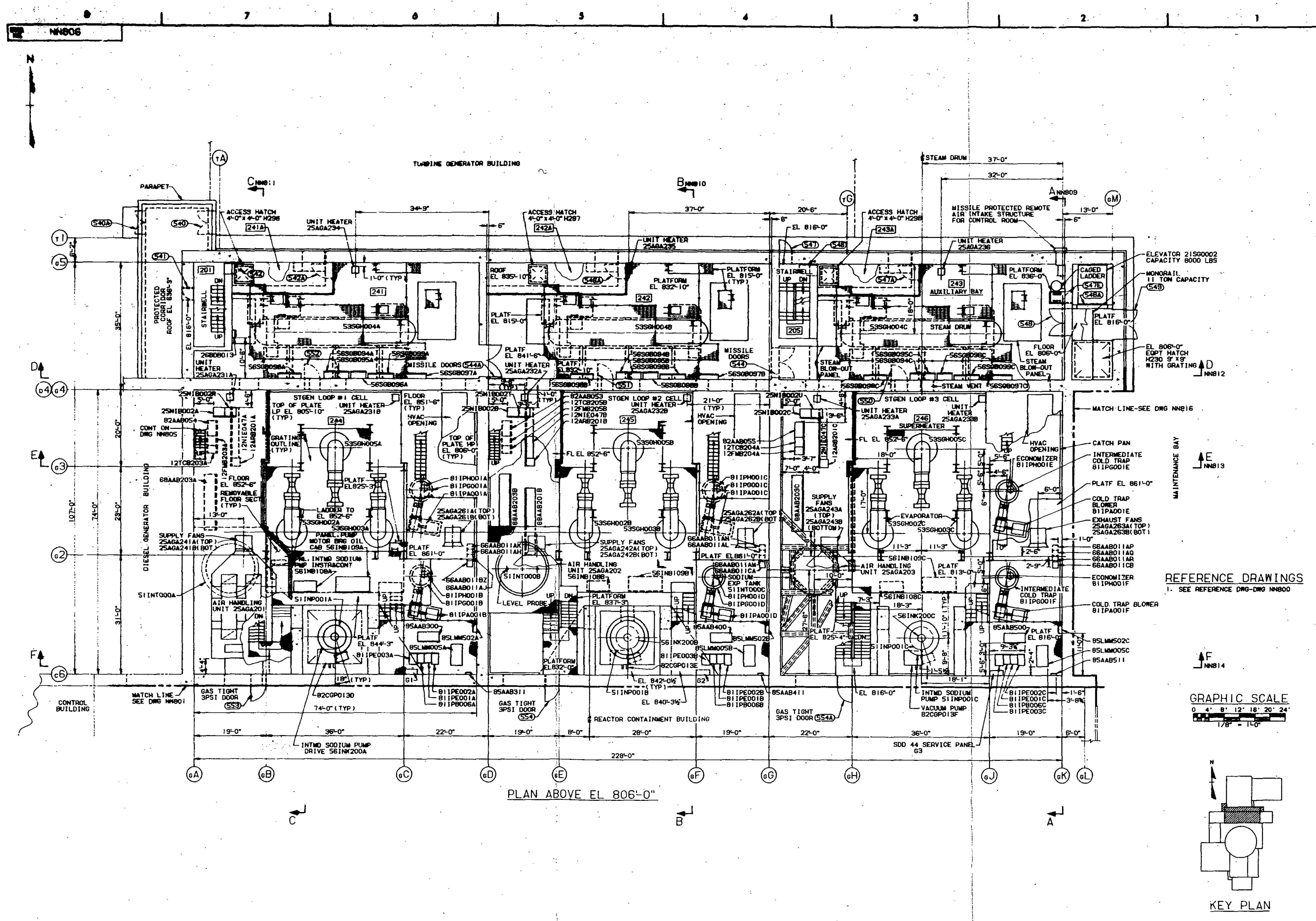



Figure 1.2-54
General Arrangement
Steam Generator Building
Plan El. 806'-0"

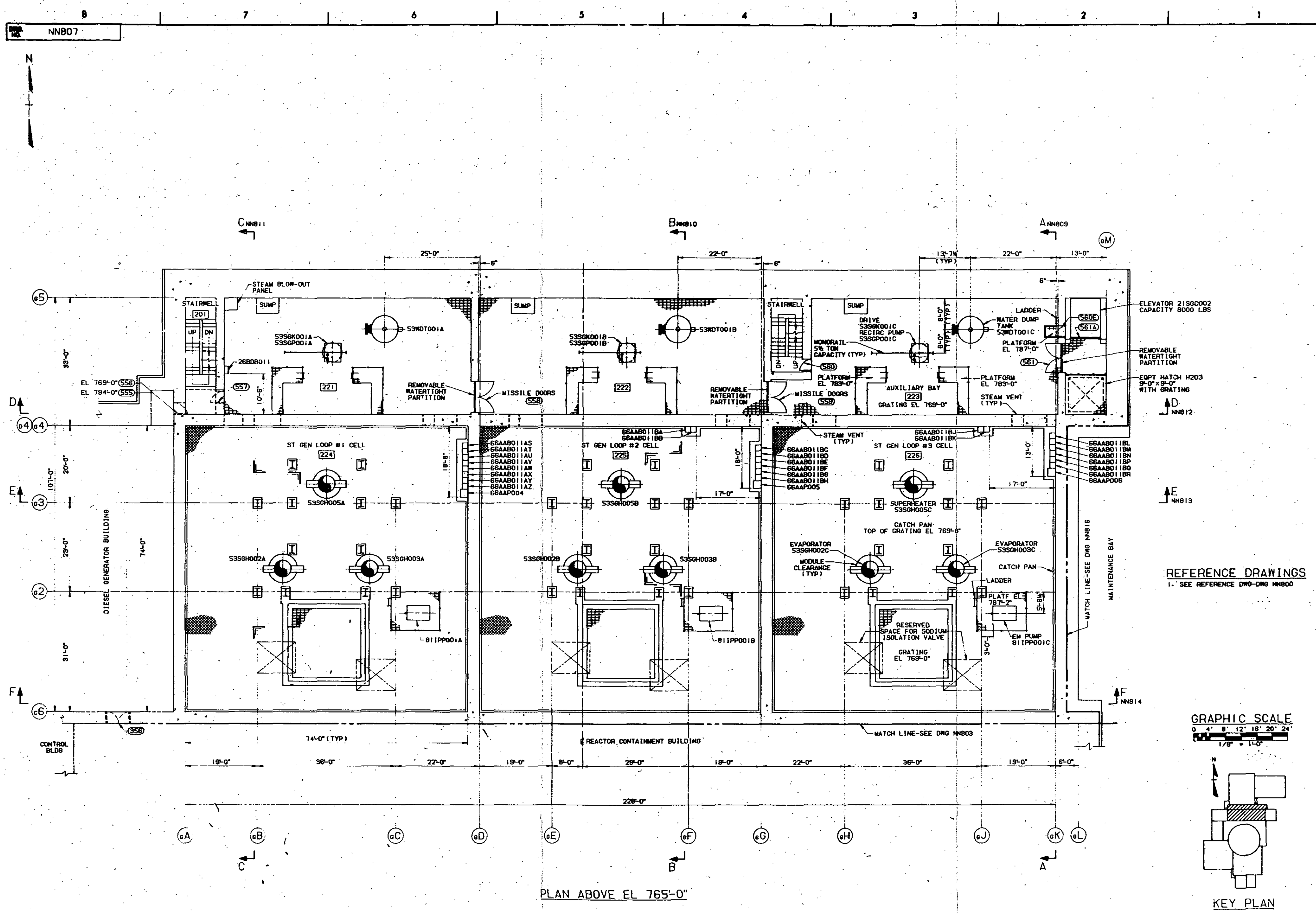



Figure 1.2-55
General Arrangement
Steam Generator Building
Plan 765'-0"

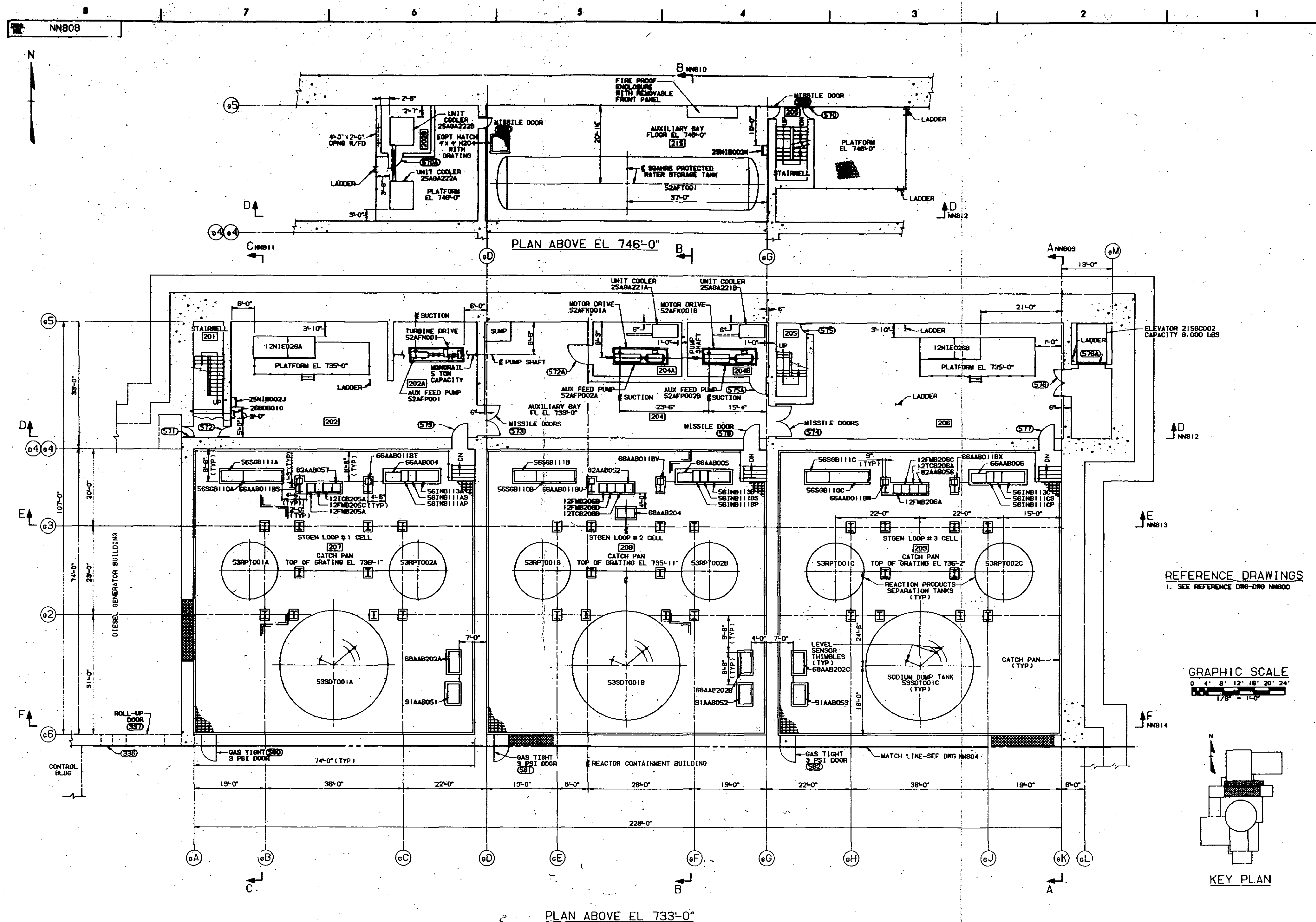



Figure 1.2-56
General Arrangement
Steam Generator Building
Plan 733'-0" and
El. 746'-0"

Amend. 64

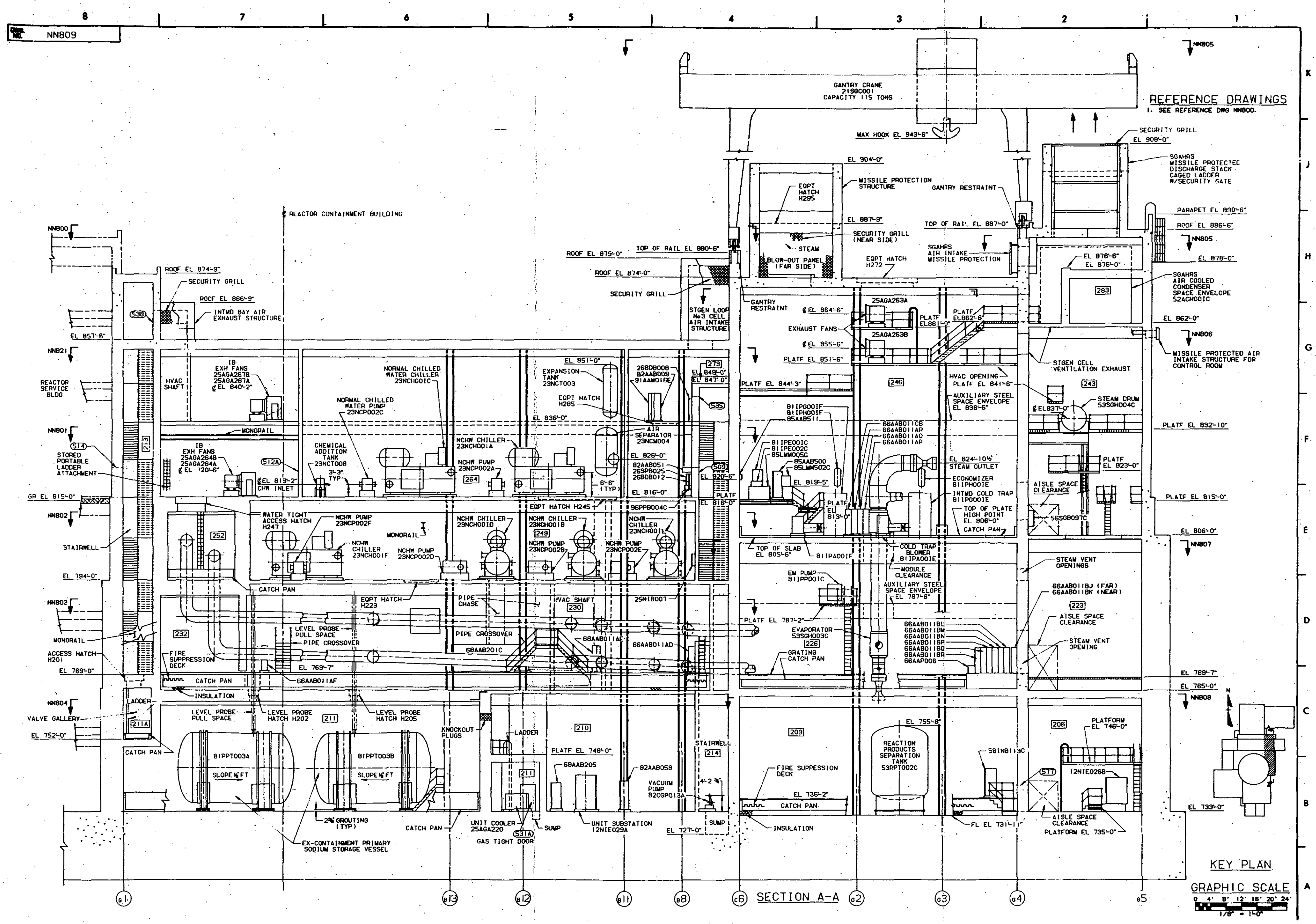



Figure 1.2-57
General Arrangement
Steam Generator Building
Section A-A

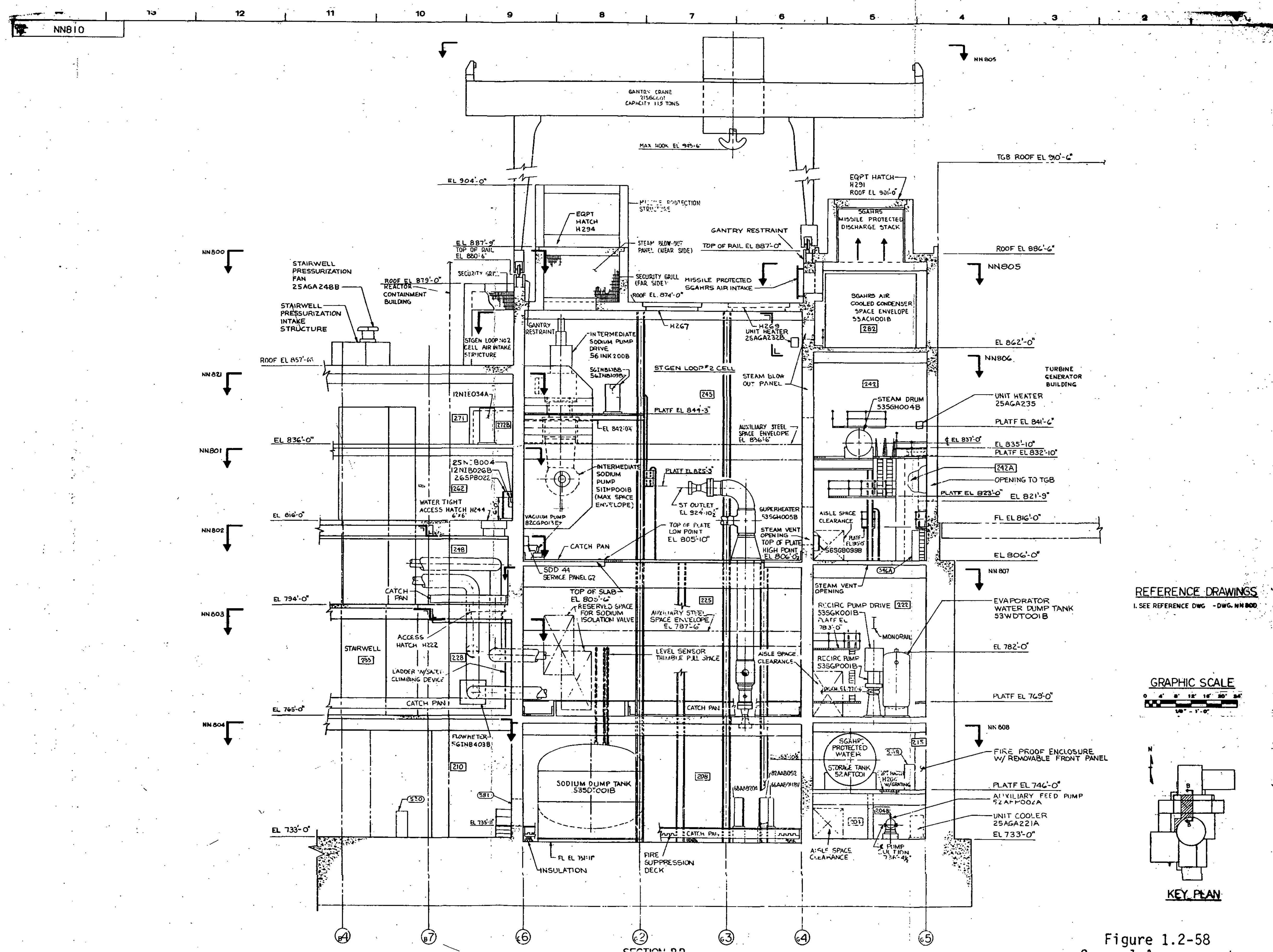


Figure 1.2-58
General Arrangement
Steam Generator Building
Section B-B

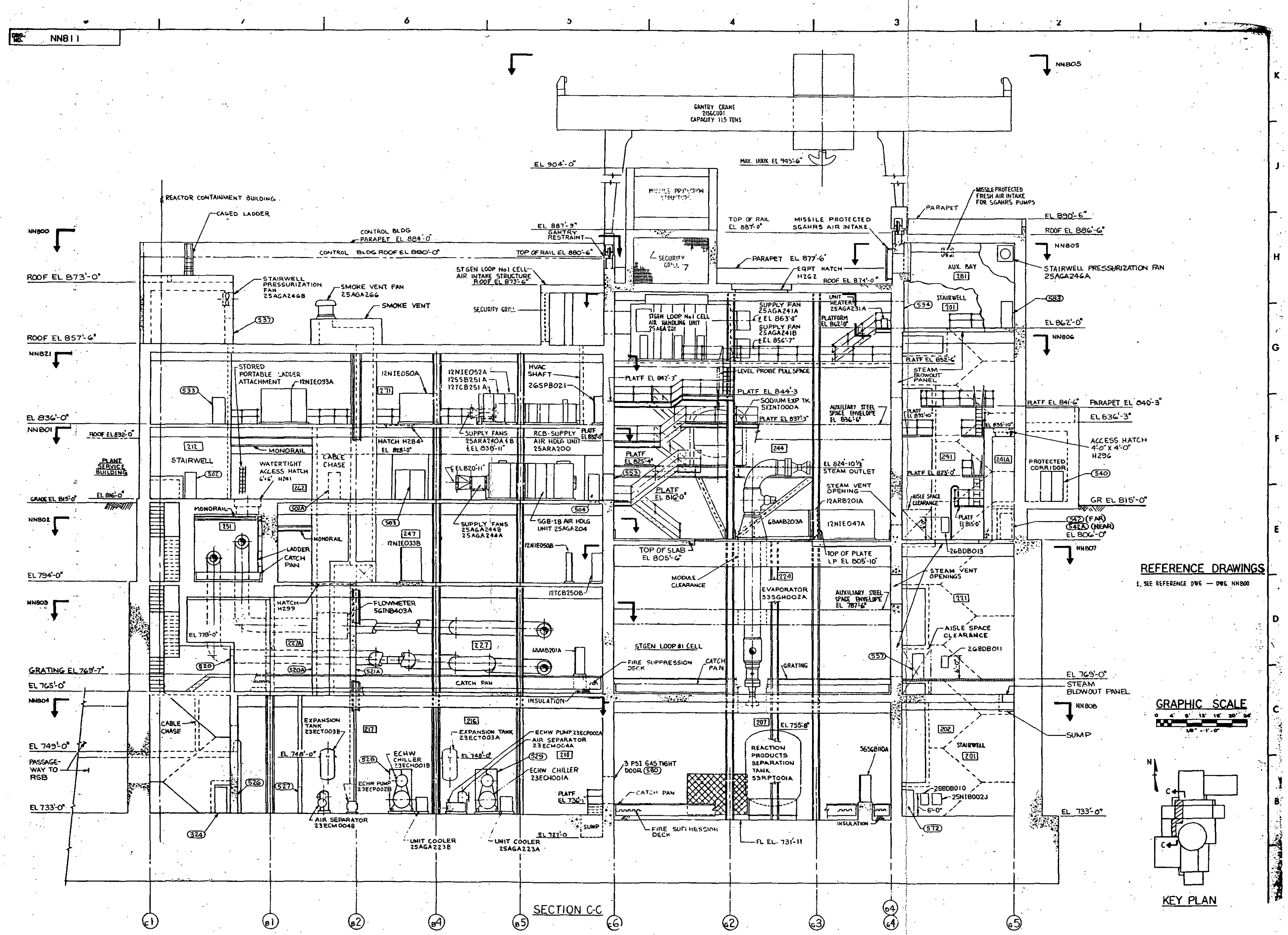



Figure 1.2-59
General Arrangement
Steam Generator Building
Section C-C

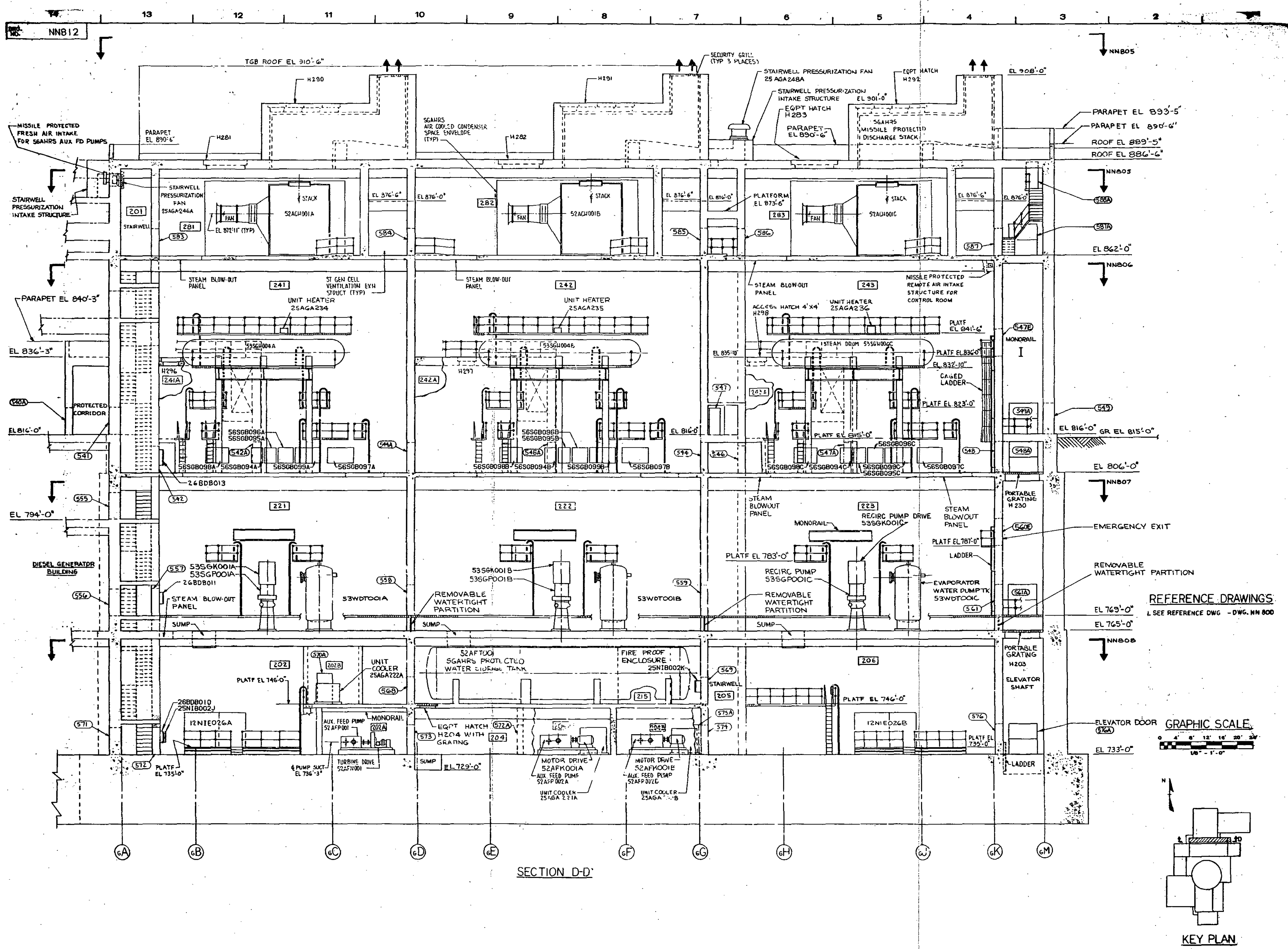



Figure 1.2-60
General Arrangement
Steam Generator Building
Section D-D

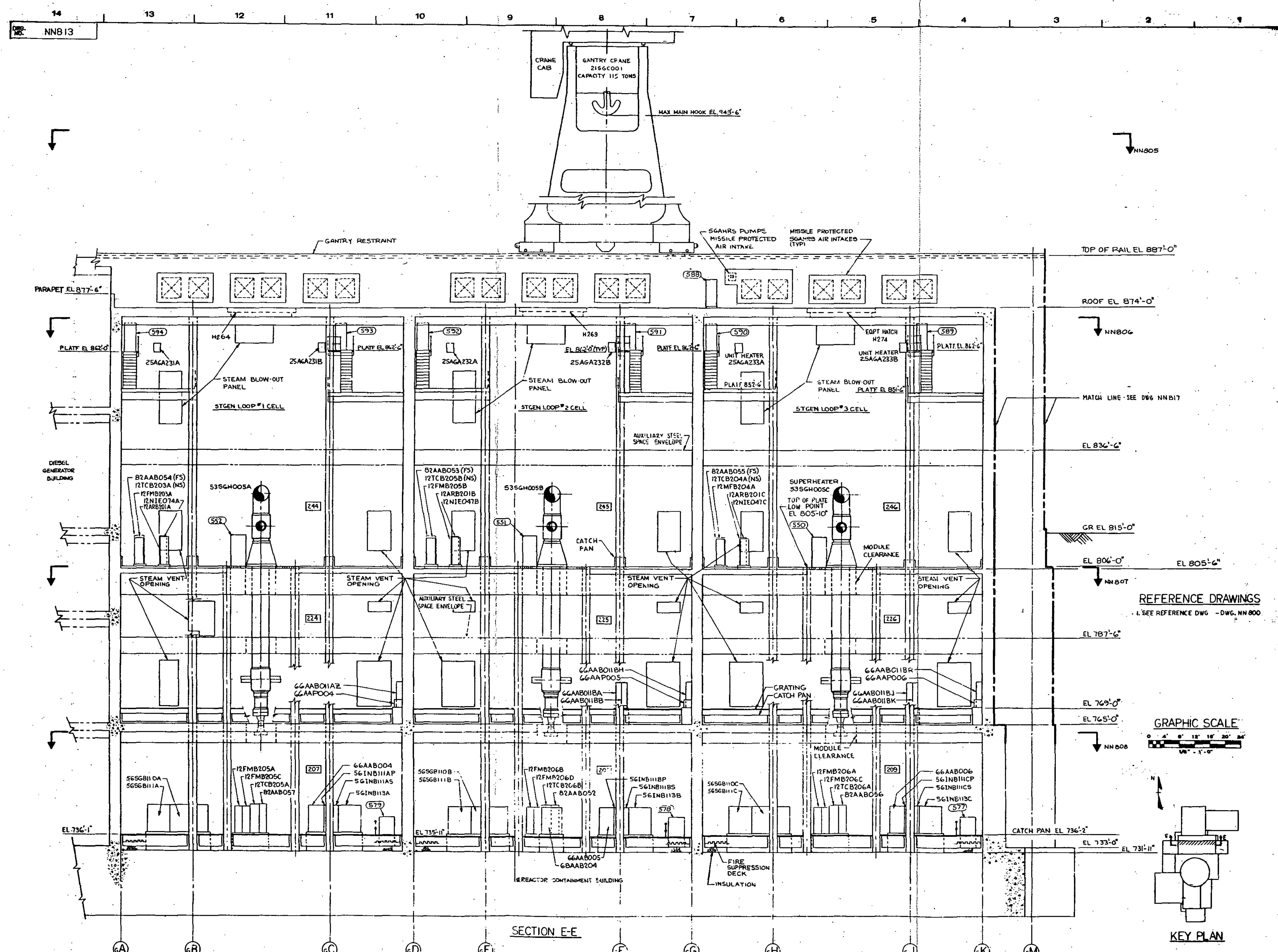



Figure 1.2-61
General Arrangement
Steam Generator Building
Section E-E

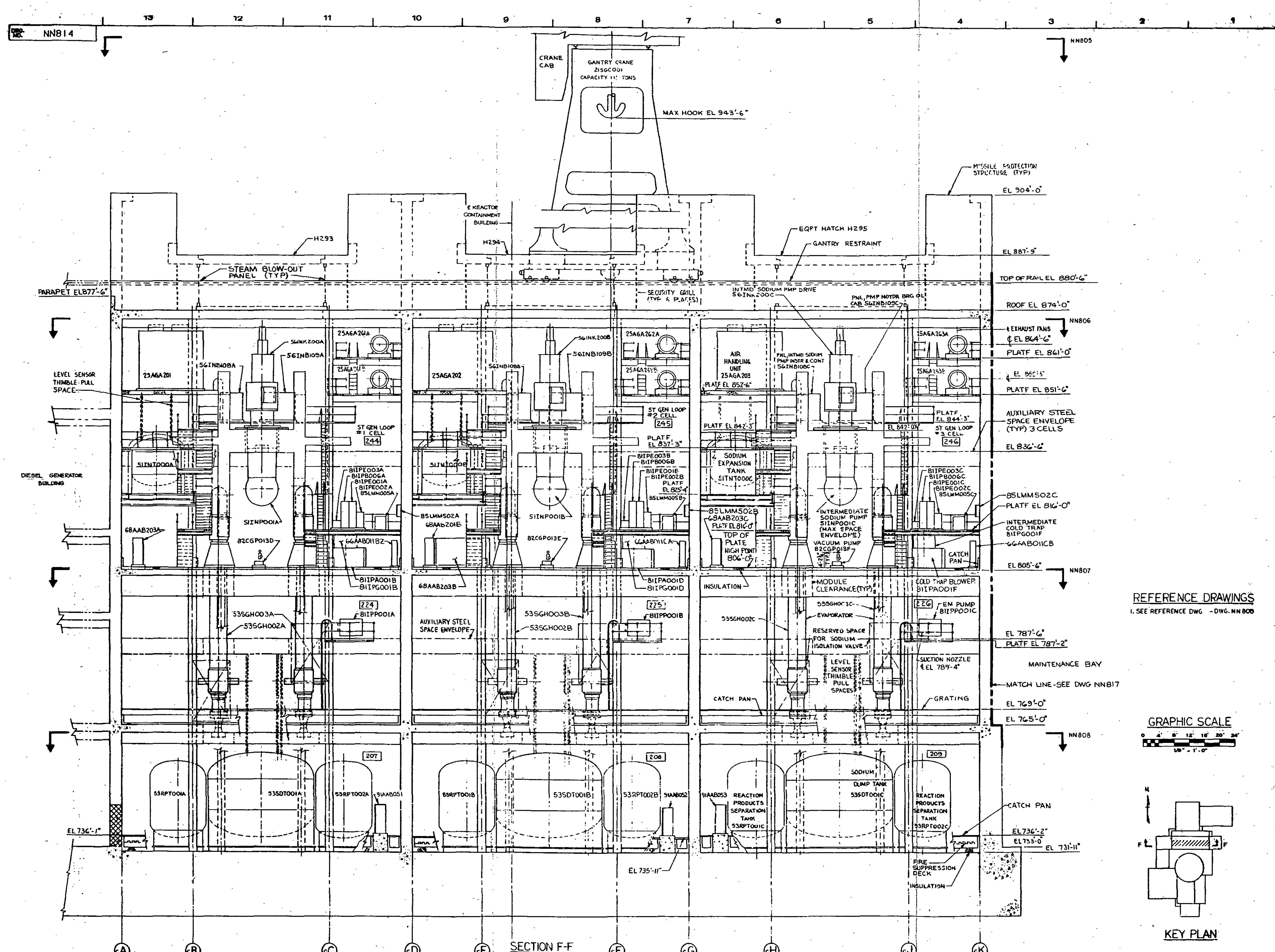



Figure 1.2-62
General Arrangement
Steam Generator Building
Section F-F

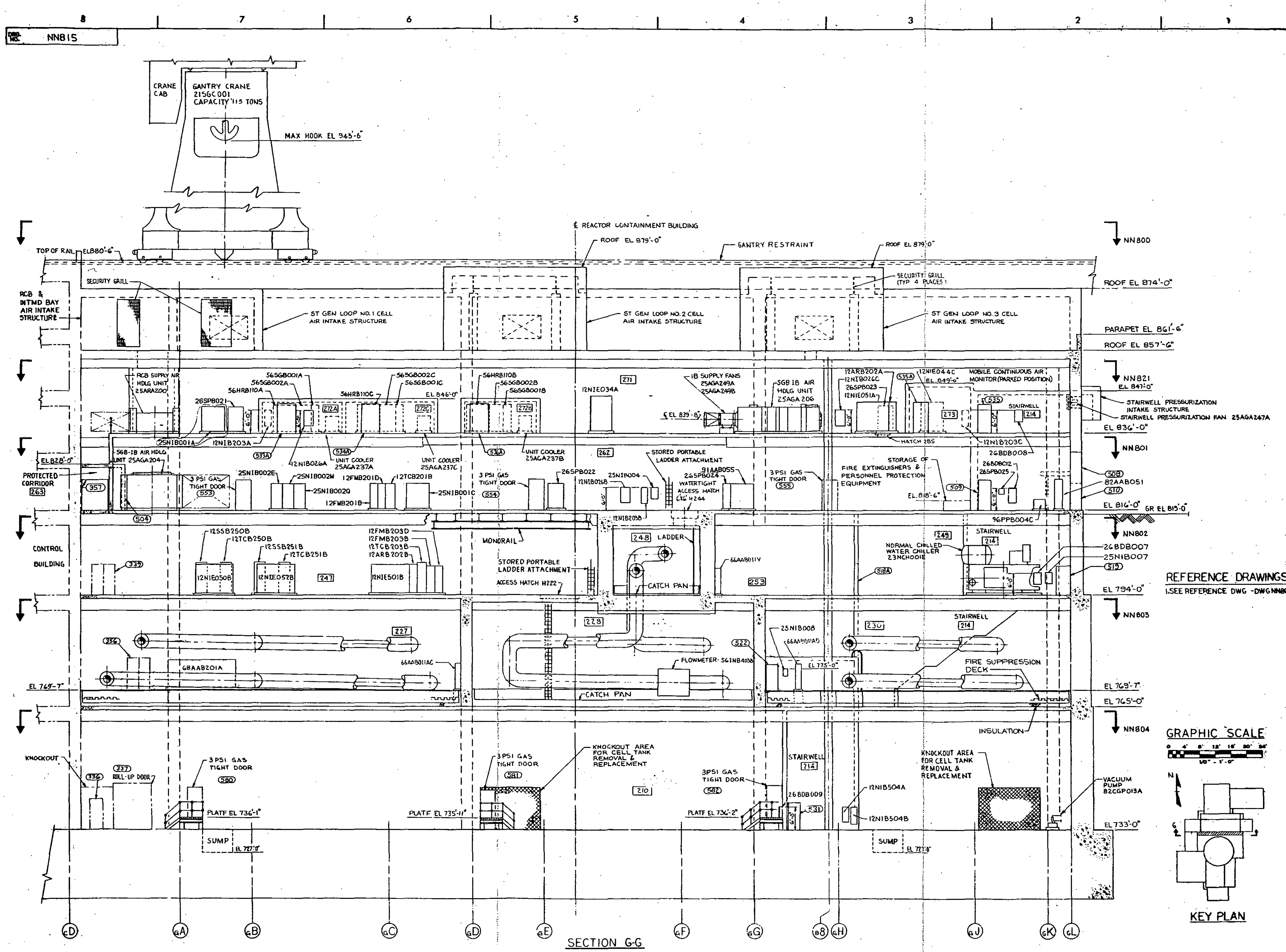



Figure 1.2-63
General Arrangement
Steam Generator Building
Section G-G

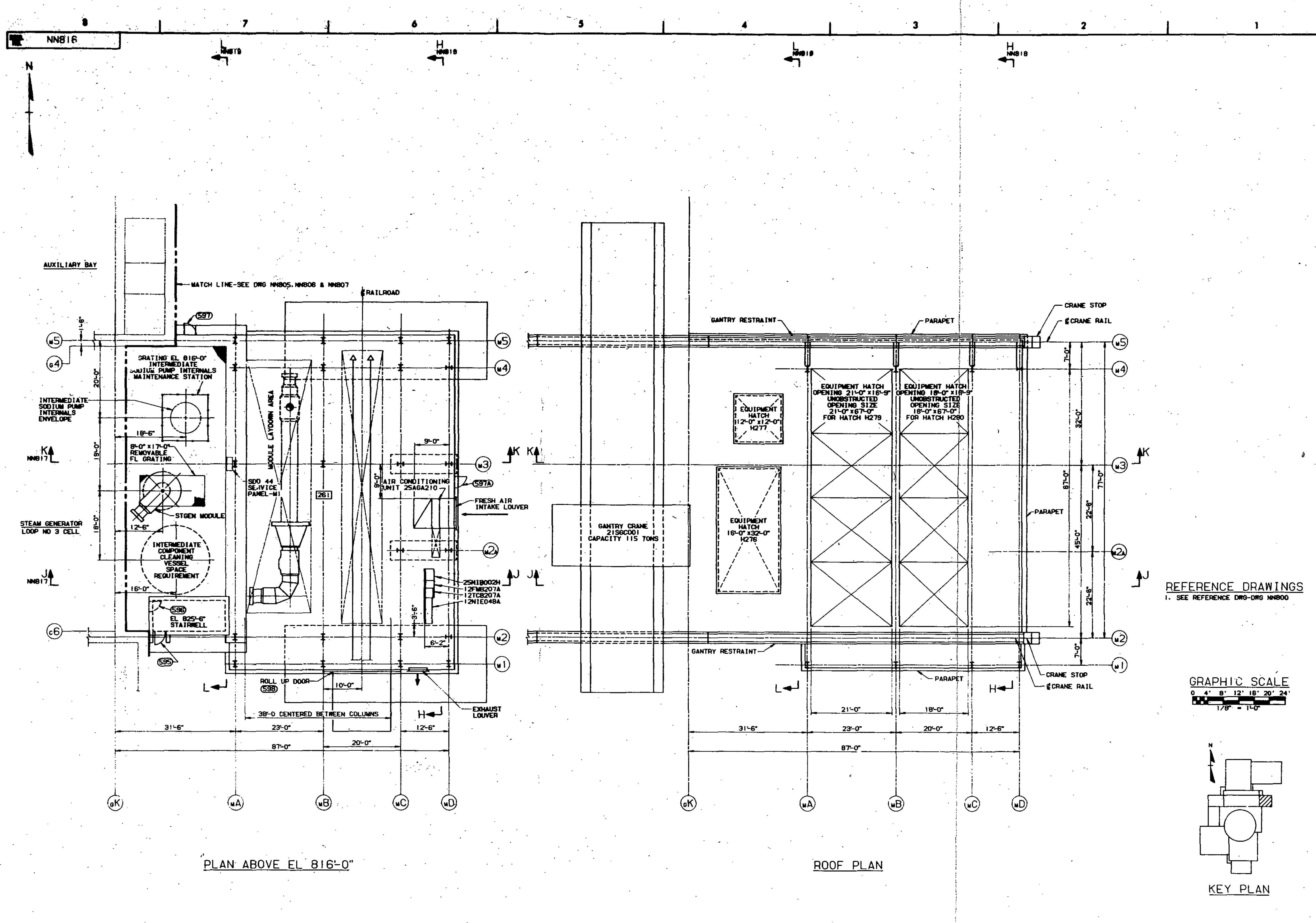



Figure 1.2-64
General Arrangement
Steam Generator Building
Plan El. 816'-0" and Roof Plan

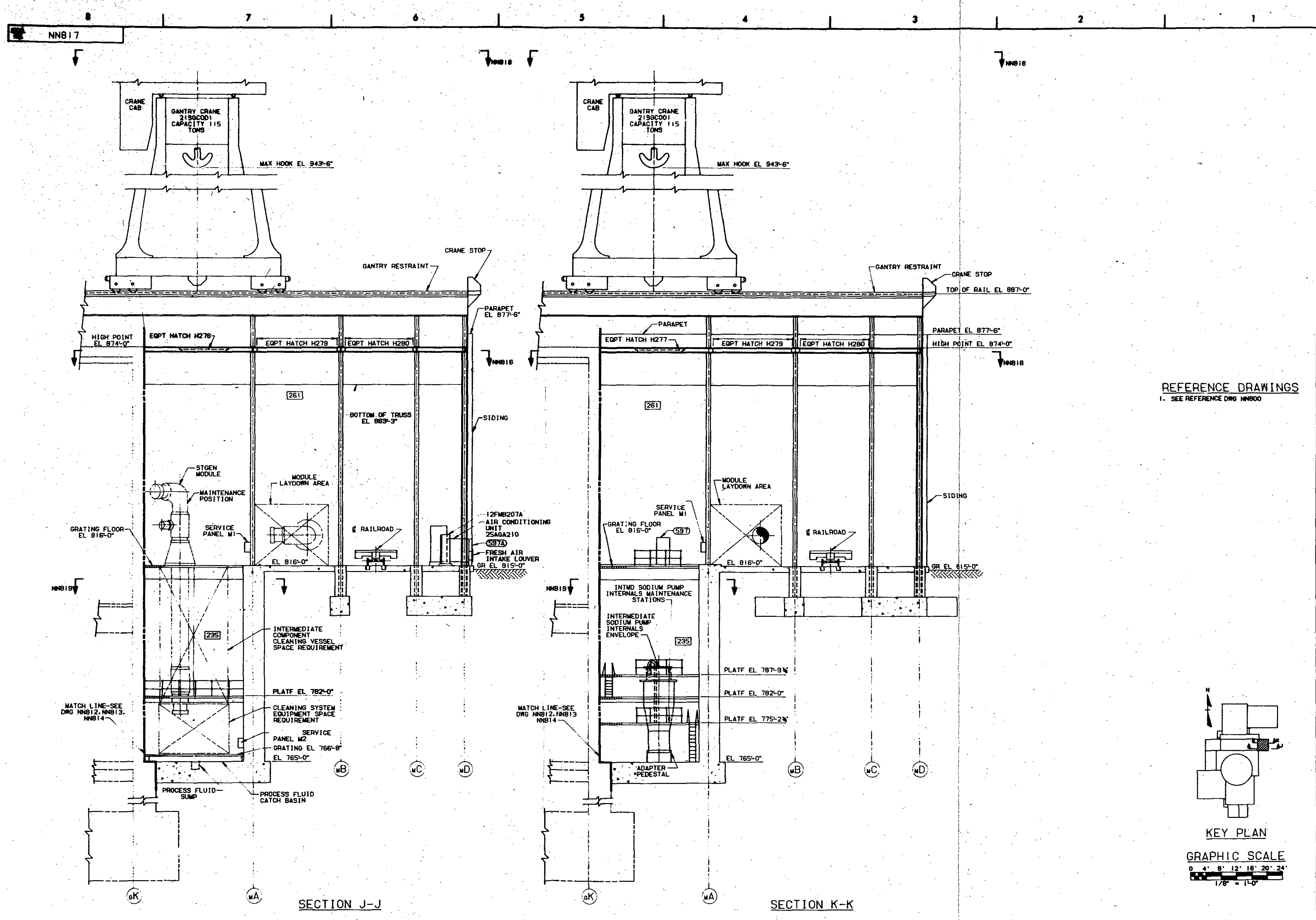



Figure 1.2-65
General Arrangement
Steam Generator Building
Section J-J and K-K

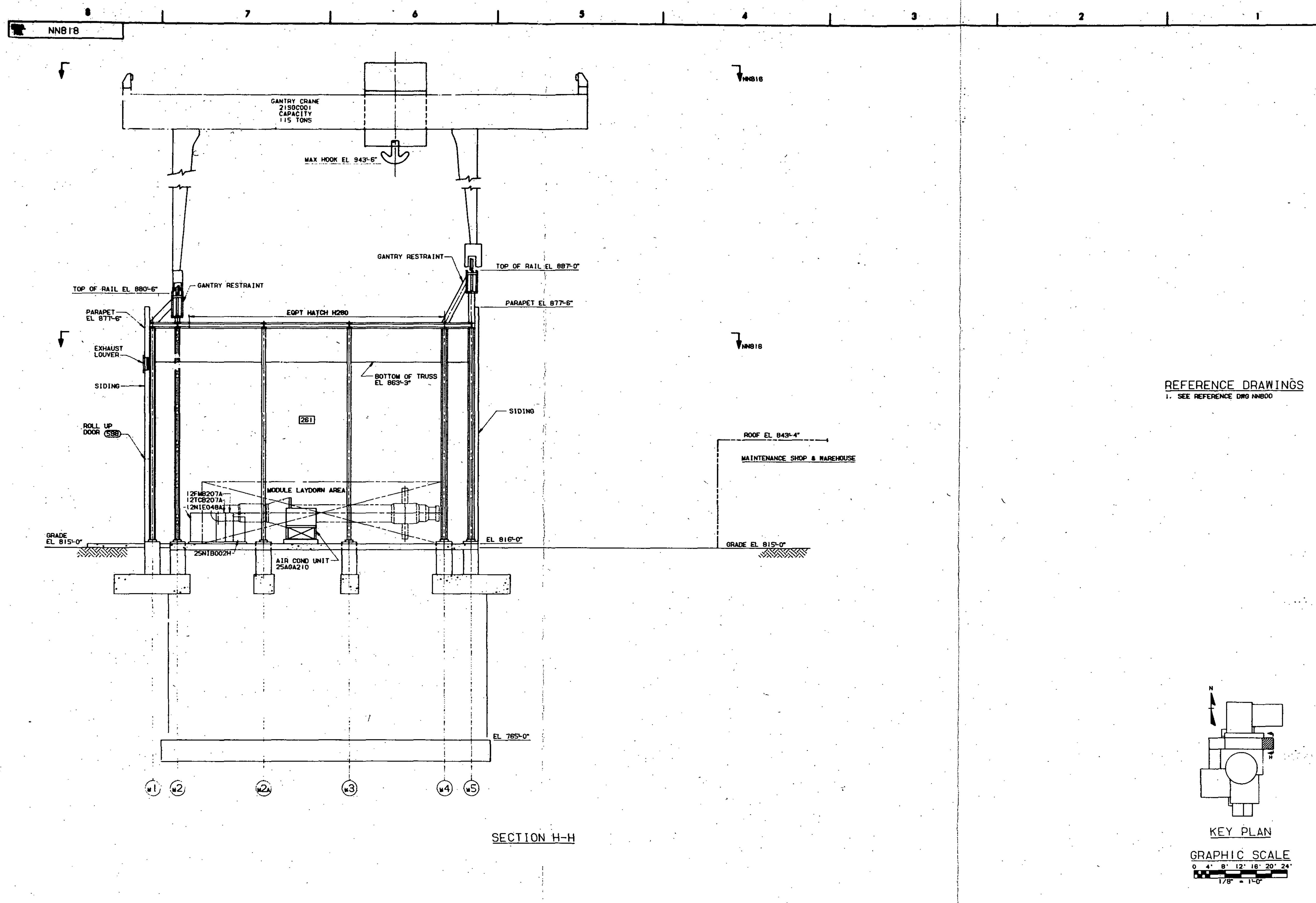



Figure 1.2-66
General Arrangement
Steam Generator Building
Section H-H

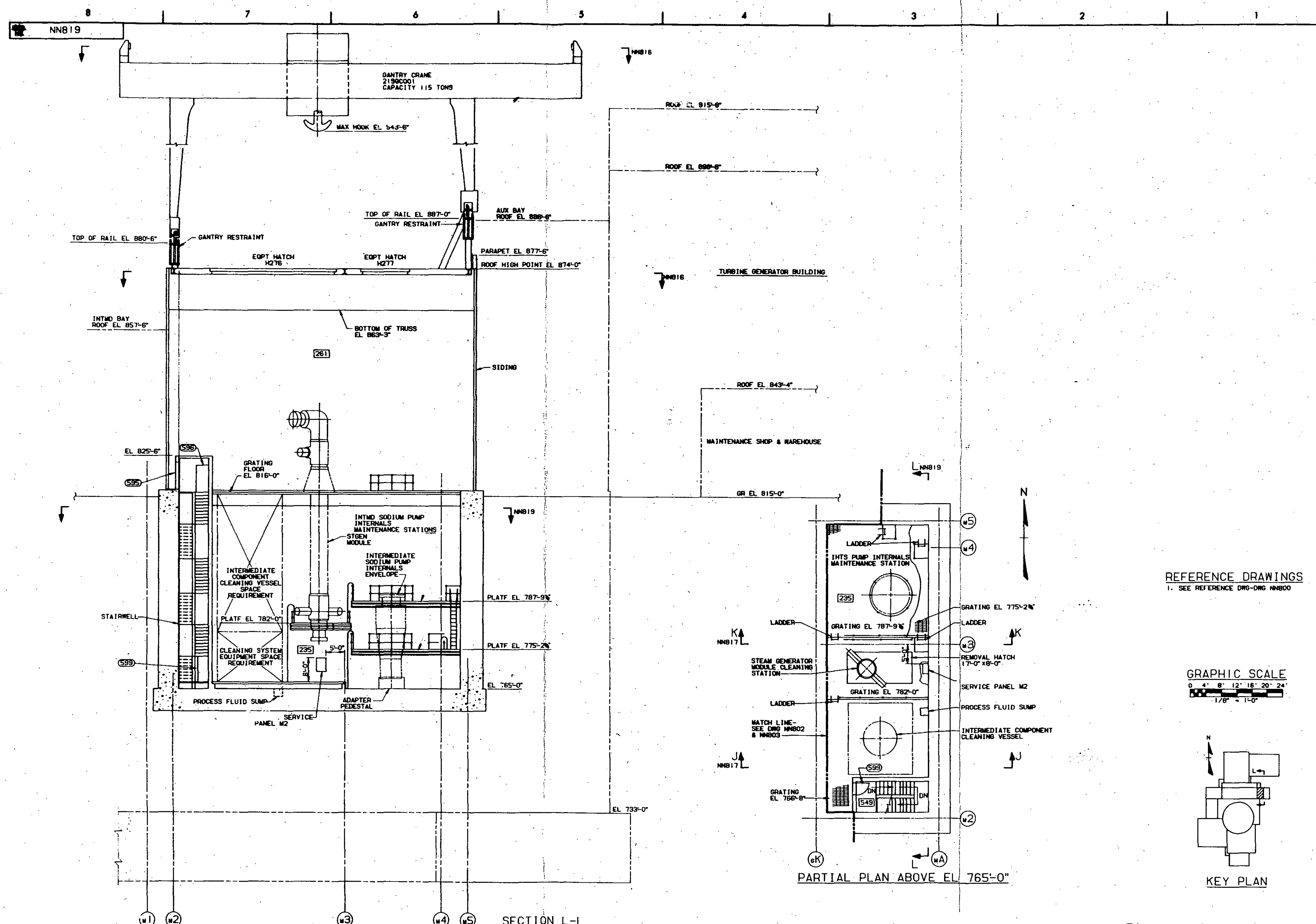



Figure 1.2-67
General Arrangement
Steam Generator Building
Section L-L Part Plan
El. 765'-0"

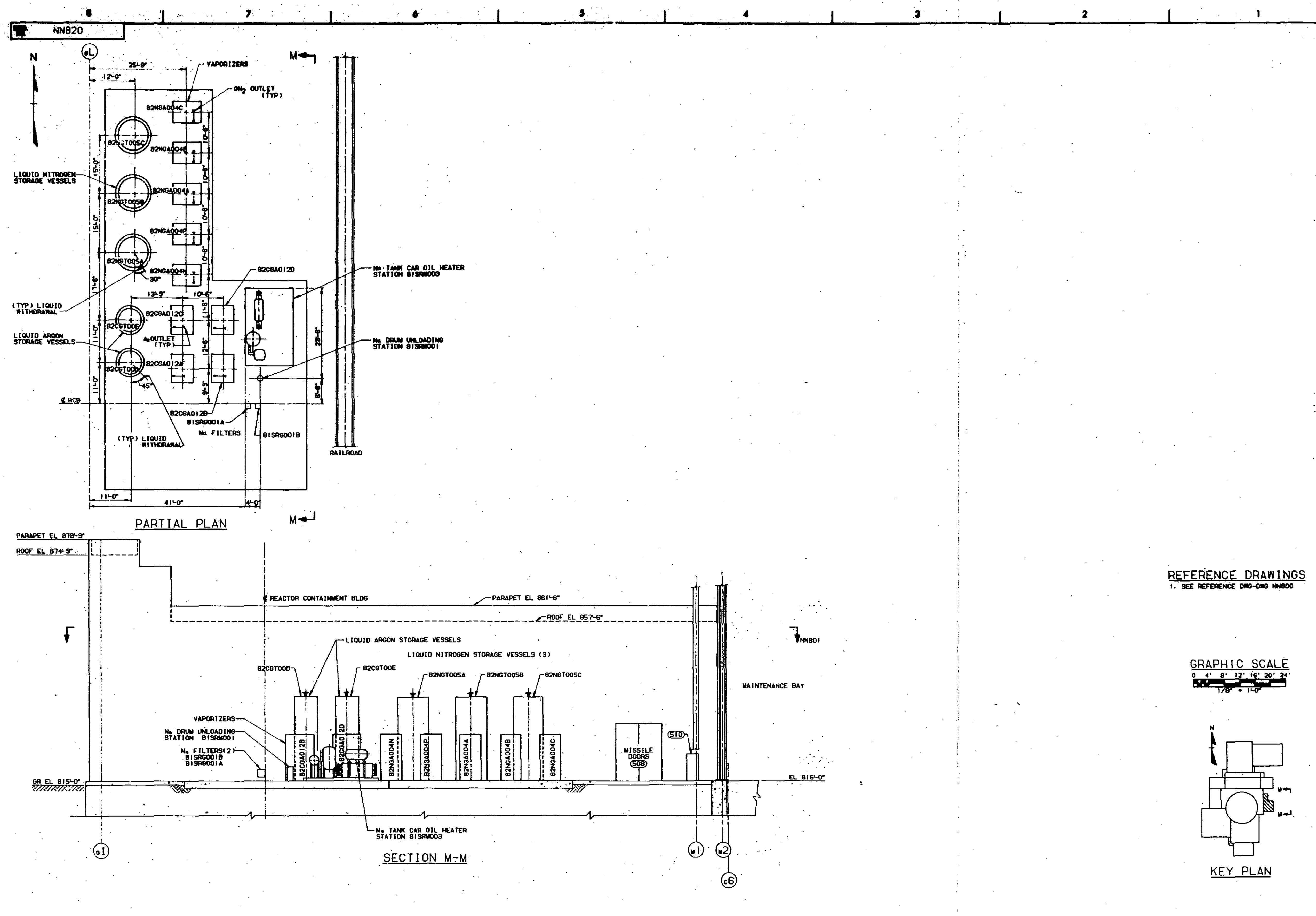



Figure 1.2-68
General Arrangement
Steam Generator Building
Plan El. 816'-0" and 840'-0"



NN820.10

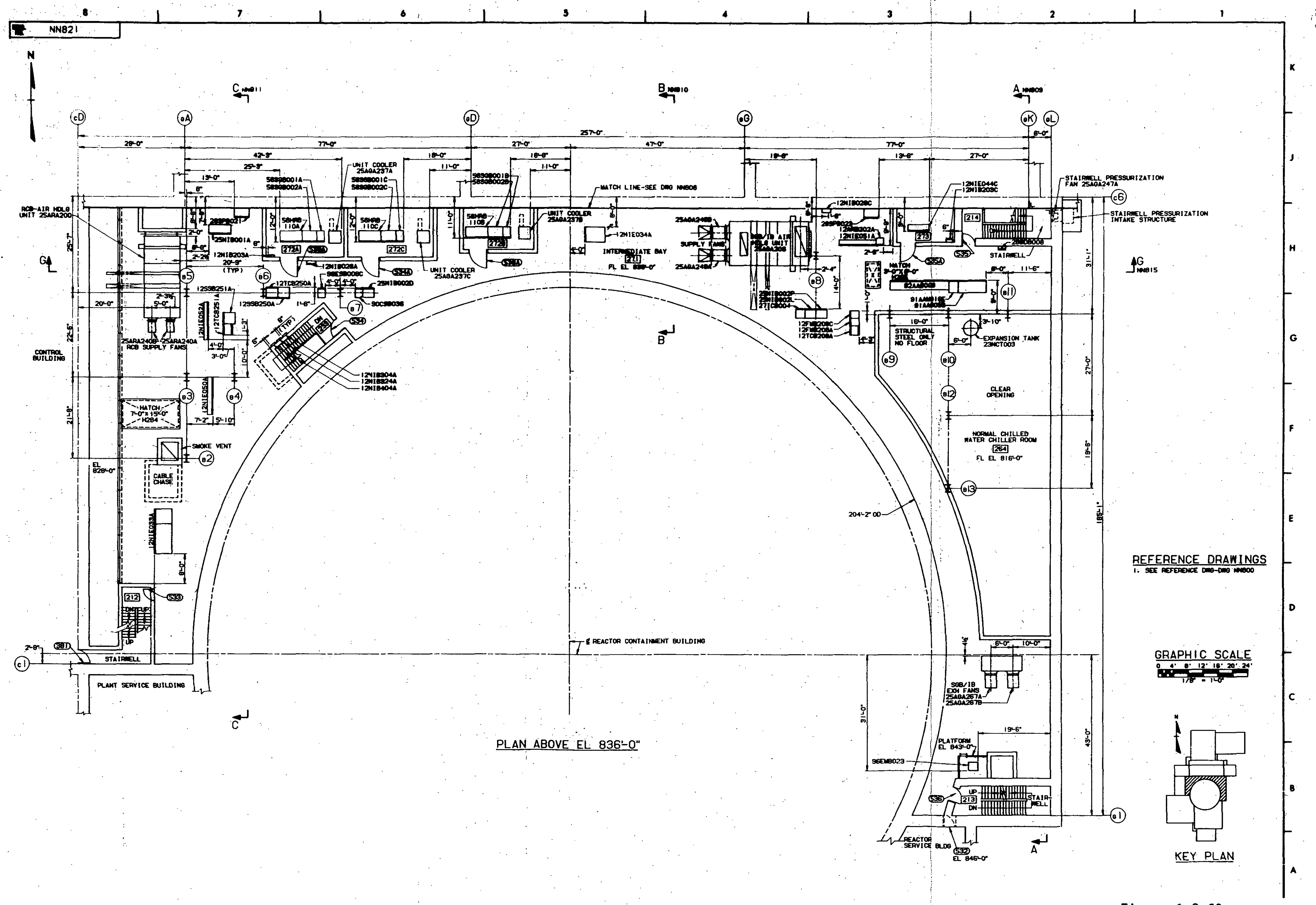
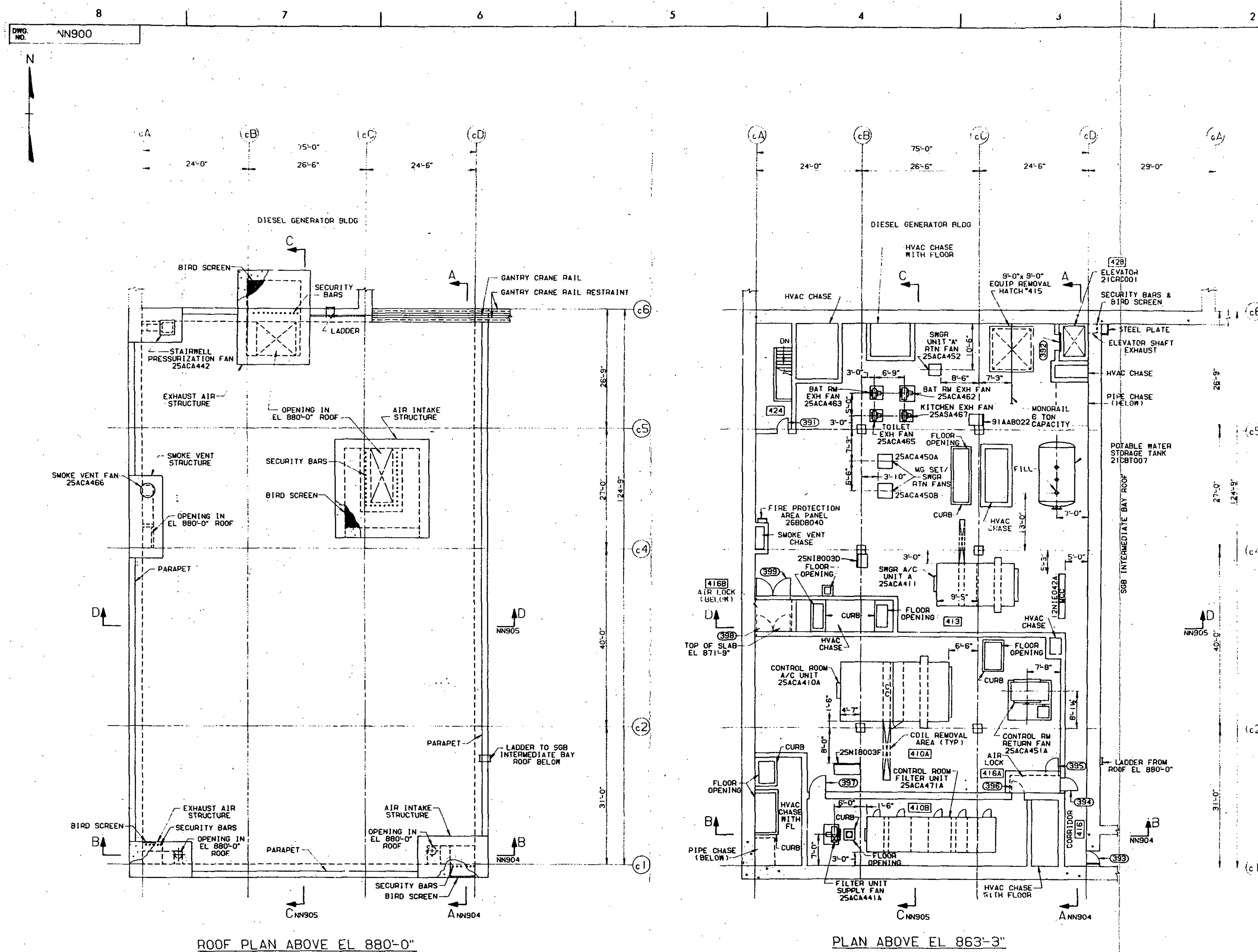



Figure 1.2-69
General Arrangement
Steam Generator Building
Intermediate Bay
El. 836'-0"




## GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD DOC-D-0036
2. ALL PARTITION WALLS ON EL 765'-0" & EL 733'-0" ARE FIREWALLS

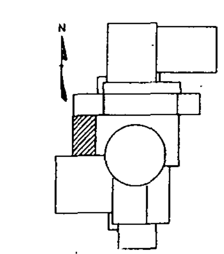
## REFERENCE DRAWINGS

NN901-CB GENERAL ARRANGEMENT PLAN ABOVE EL 847'-3" & EL 831'-0" SDD210
NN902-CB GENERAL ARRANGEMENT PLAN ABOVE EL 816'-0" & EL 794'-0" SDD210
NN903-CB GENERAL ARRANGEMENT PLAN ABOVE EL 765'-0" & EL 733'-0" SDD210
NN904-CB GENERAL ARRANGEMENT SECTION A-A & B-B SDD210
NN905-CB GENERAL ARRANGEMENT SECTION C-C & D-D SDD210

## LEGEND

- EI&C CABINET SWGR (HEAVY LINE FRONT)
- GRATING
- 410 CELL NUMBER
- KNOCKOUT SECTION
- CONCRETE
- 394 DOOR NUMBER

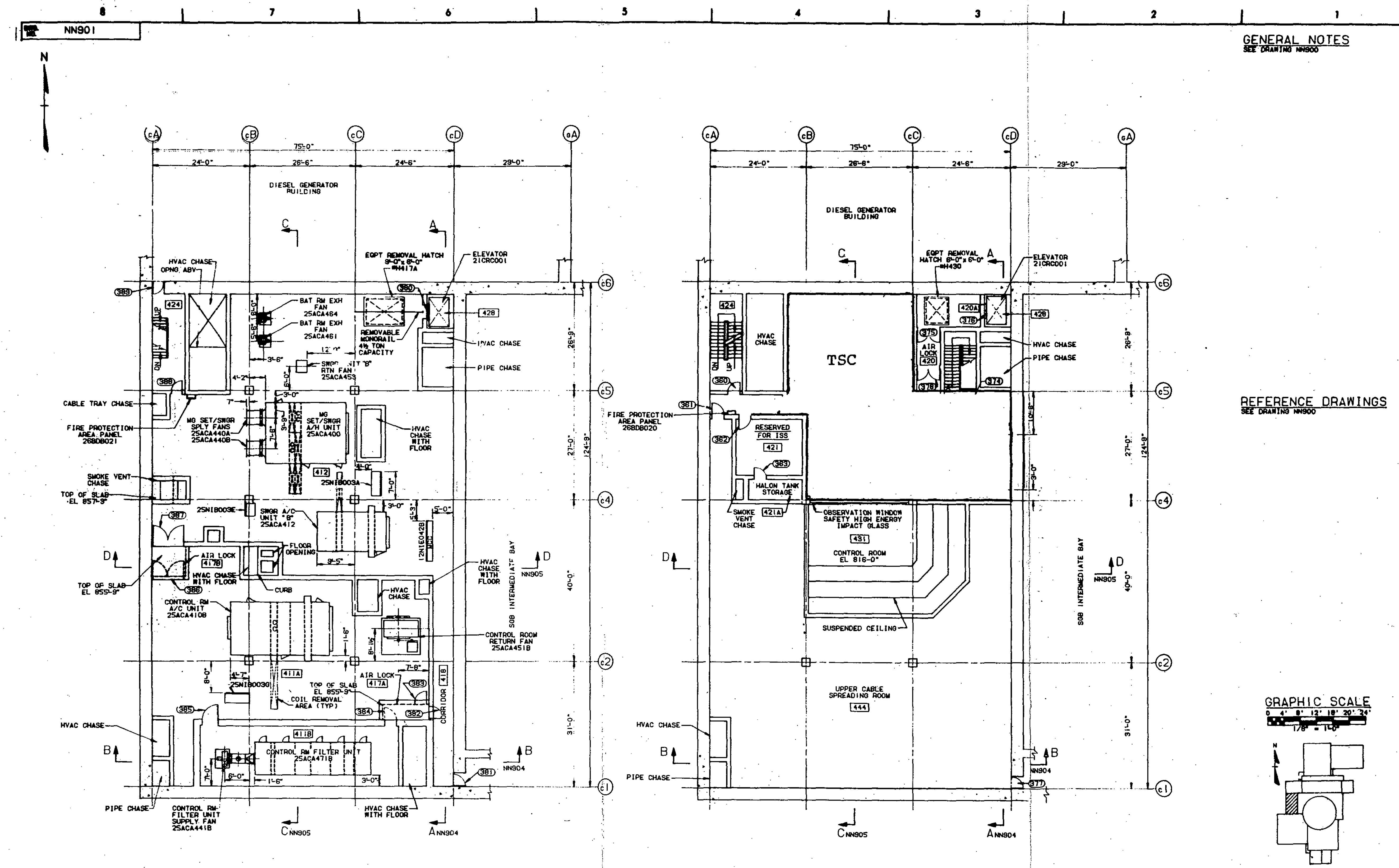
GRAPHIC SCALE
0 4' 8' 12' 16' 20' 24'
1/8" = 1'-0"



KEY PLAN

Figure 1.2-70
General Arrangement
Control Building
El. 880'-0" and 863'-3"

GENERAL NOTES
SEE DRAWING NN900




PLAN ABOVE EL 847'-3"

PLAN ABOVE EL 831'-0"

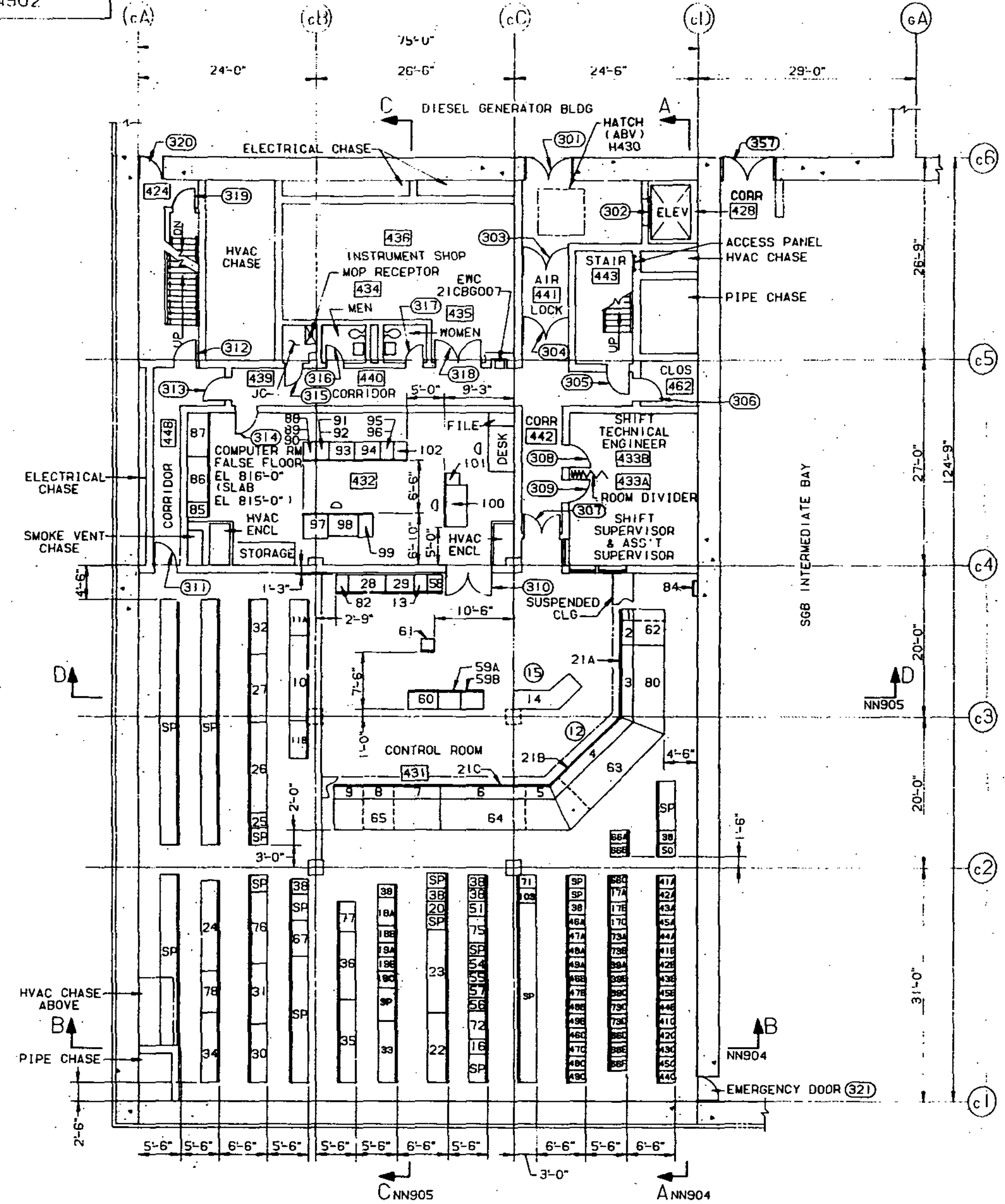
REFERENCE DRAWINGS
SEE DRAWING NN900

GRAPHIC SCALE

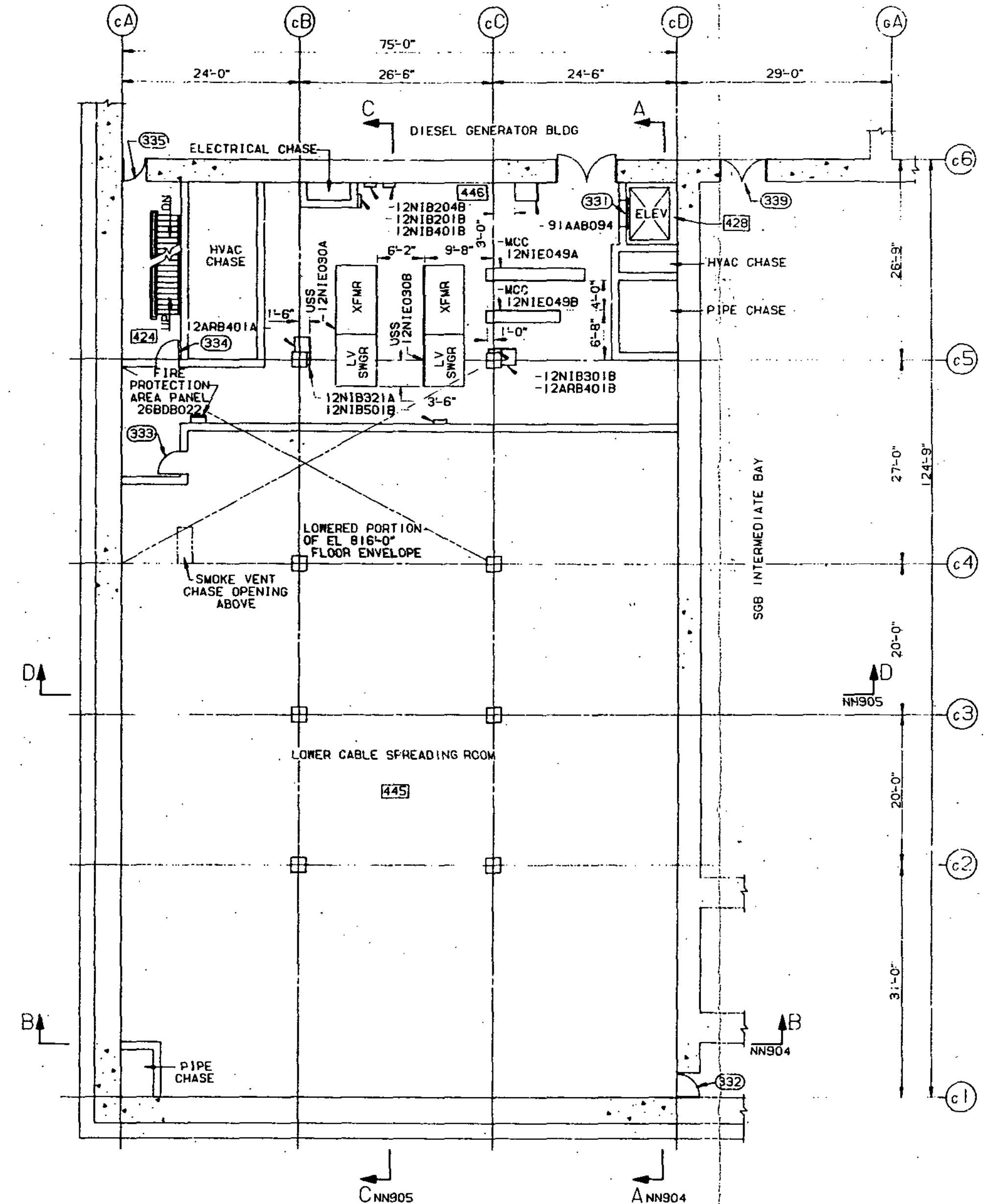
KEY PLAN

Figure 1.2-71
General Arrangement
Control Building
El. 847'-3" and 831'-0"

DWG. NO. NN902




PLAN ABOVE EL 816'-0"

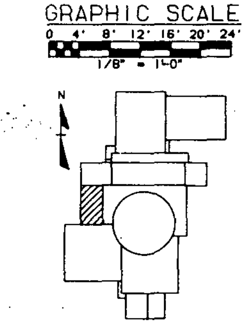



PLAN ABOVE EL 794'-0"

GENERAL NOTES
SEE DRAWING NN900

REFERENCE DRAWINGS
SEE DRAWING NN900

GRAPHIC SCALE



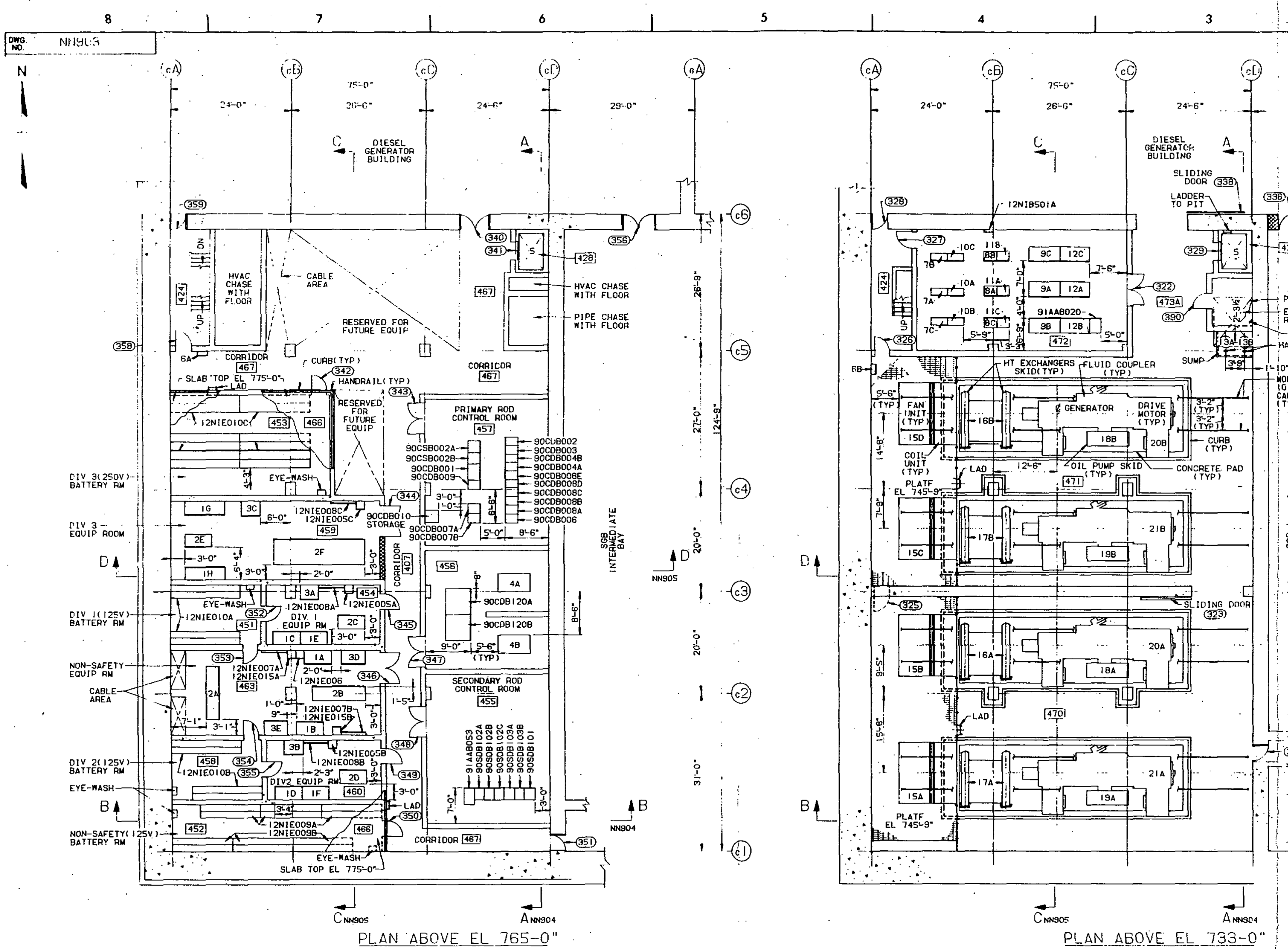

KEY PLAN

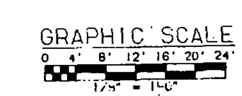
| ITEM | DESCRIPTION | EQUIPMENT NO | ITEM | DESCRIPTION | EQUIPMENT NO | ITEM | DESCRIPTION | EQUIPMENT NO | ITEM | DESCRIPTION | EQUIPMENT NO | ITEM | DESCRIPTION | EQUIPMENT NO |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| 1 | REACTOR SUPPORT SYSTEMS | 90CSB016 | 21C | COMMUNICATIONS JACK & SWITCH | | 43A | PRIMARY PPS COMPARATOR PNL | 99PSB001C | 57 | LOAD DISPATCH PANEL | 90CSB005B | 76 | HEAT REMOVAL & CONDTN LGC RACK | 56HRB020 |
| 2 | REACTOR SUPPORT SYSTEMS | 90CSB016 | 22 | HVAC SYS RECIRC AIR STATUS CAB | 59AAB001 | 43B | PRIMARY PPS COMPARATOR PNL | 99PSB001F | 58 | FAILED FUEL READOUT PANEL | 94AAB012 | 77 | PDH & DS REM DATA ACQ TERM | 91AAB056 |
| 3 | ENGINEERED SAFETY SYSTEMS | 90CSB016 | 23 | HVAC CONTROL RACK | 59AAB002 | 43C | PRIMARY PPS COMPARATOR PNL | 99PSB001J | 59A | COMPUTER TYPEWRITER | 91AAM015A | 78 | REMOTE ANNUNCIATOR CABINETS | 67NIB003A |
| 4 | REACTOR PRI HEAT TRANSPORT SYSTEMS | 90CSB016 | 24 | GEN PROTECTION | 12NIB002 | 44A | PRIMARY PPS LOGIC RACK | 99PSB003B | 59B | COMPUTER TYPEWRITER | 91AAM015B | 79 | MANUAL TEL SWBD & PAIC HAND SET | ——— |
| 5 | INTMD Na HEAT TRANSPORT SYSTEMS | 90CSB016 | 25 | TURB GEN SUPERVISORY PANEL | 69AAB025 | 44B | PRIMARY PPS LOGIC RACK | 99PSB003D | 60 | COMPUTER LINE PRINTER/PLOTTER | 91AAM013 | 80 | CRT DISPLAY & KEYBOARD | 91AAM001F |
| 6 | STEAM GEN & ASSOCIATED SYSTEMS | 90CSB016 | 26 | CABINET EHC EQUIPMENT | 69AAB028 | 44C | PRIMARY PPS LOGIC RACK | 99PSB003F | 61 | CRT DISPLAY & KEYBOARD | 91AAM001A | 81 | | |
| 7 | TURBINE SYSTEMS | 90CSB016 | 27 | BOP BACK PANEL | 69AAB024 | 45A | PRIMARY PPS ISOLATION RACK | 99PSB003A | 62 | CRT DISPLAY & KEYBOARD | 91AAM001B | 82 | BLDG FIRE PROTECTION PANEL | 26BDB023 |
| 8 | GENERATOR SYSTEMS | 90CSB016 | 28 | NON-SODIUM FIRE PROTECTION PANEL | 26BDB042 | 45B | PRIMARY PPS ISOLATION RACK | 99PSB003C | 63 | CRT DISPLAY & KEYBOARD | 91AAM001C | | | |
| 9 | SYNC & MAIN UNIT CONTROL | 90CSB016 | 29 | SODIUM FIRE PROT Z-IND PANEL | 26SPB015 | 45C | PRIMARY PPS ISOLATION RACK | 99PSB003E | 64 | CRT DISPLAY & KEYBOARD | 91AAM001D | 84 | PATCH PANEL | |
| 10 | SWYD & STA ELEC DISTR | 12NIB020 | 30 | HEAT REMOVAL & CONDTN LGC RACK | 56PRB020 | 46A | SECONDARY PPS BUFFER | 99PSB002A | 65 | CRT DISPLAY & KEYBOARD | 91AAM001E | 85 | CABINET CARD STORAGE | 91AAJ003 |
| 11A | EMER DIESEL GEN PANEL | 12NIB019 | 31 | STEAM GEN LGC RACK | 56SGB100 | 46B | SECONDARY PPS BUFFER | 99PSB002D | 66A | PDH & DS REM DATA ACQ TERM | 91AAB052A | 86 | CABINET PAPER STORGE | 91AAJ002 |
| 11B | EMER DIESEL GEN PANEL | 12NIB019 | 32 | CHILLED WATER CONTROL CABINET | 23ICB001 | 46C | SECONDARY PPS BUFFER | 99PSB0026 | 66B | PDH & DS REM DATA ACQ TERM | 91AAB052B | 87 | CABINET MAGNETIC TAPE STORAGE | 91AAJ001 |
| 12 | REACTOR OPERATOR | ——— | 33 | AUX LIQ METAL CONTROL CABINET | 81AAB018 | 47A | SECONDARY PPS TERMINATION CAB | 99PSB002B | 66C | PDH & DS REM DATA ACQ TERM | 91AAB052C | 88 | MAGNETIC TAPE UNIT NO 3 | 91AAM010C |
| 13 | SODIUM LEAK DETECTOR PANEL | 66AAB008 | 34 | REMOTE ANNUNCIATOR CABINETS | 90CSB019 | 47B | SECONDARY PPS TERMINATION CAB | 99PSB002E | 66D | PDH & DS REM DATA ACQ TERM | 91AAB055A | 89 | SERIAL LINE INTERFACE | 91AAB005 |
| 14 | DESK | ——— | 35 | BOP STEAM PLANT | 69AAB026 | 47C | SECONDARY PPS TERMINATION CAB | 99PSB002H | 66E | PDH & DS REM DATA ACQ TERM | 91AAB055B | 90 | FLEXIBLE DISC | 91AAM014 |
| 15 | CHAIR | ——— | 36 | LOGIC CABINET STEAM PLANT | 69AAB027 | 48A | SECONDARY PPS COMPARATOR CAB | 99PSB002C | 66F | PDH & DS REM DATA ACQ TERM | 91AAB055C | 91 | FIXED HEAD DISC | 91AAM008B |
| 16 | SEISMIC INSTRUMENT PANEL | 27ICB001 | 37 | | | 48B | SECONDARY PPS COMPARATOR CAB | 99PSB002F | 67 | RECIRCULATING GAS CONTROL CAB | 28ICB001 | 92 | MAGNETIC TAPE NO 2 | 91AAM010B |
| 17A | FLUX MONITORING | 95AAB001A | 38 | TERMINATION RACKS | 90CSB015 | 48C | SECONDARY PPS CONPARATOR CAB | 99PSB002J | 68 | | | 93 | CENTRAL PROCESSOR UNIT NO 1 | 91AAB002A |
| 17B | FLUX MONITORING | 95AAB001B | 39A | PPS CONTAINMENT ISOL INSTR RACK | 99ESB005A | 49A | SECONDARY PPS SOLENOID DRIVE | 99PSB004A | 69 | | | 94 | CENTRAL PROCESSOR UNIT NO 2 | 91AAB002B |
| 17C | FLUX MONITORING | 95AAB001C | 39B | PPS CONTAINMENT ISOL INSTR RACK | 99ESB005B | 49B | SECONDARY PPS SOLENOID DRIVE | 99PSB004B | | | | 95 | FIXED HEAD DISC | 91AAM008A |
| 18A | CR RAD MON PRINTER | 96AAB002A | 39C | PPS CONTAINMENT ISOL INSTR RACK | 99ESB005C | 49C | SECONDARY PPS SOLENOID DRIVE | 99PSB004C | 71 | CONTAINMENT INSTRUMENTATION | 27ICB002 | 96 | MAGNETIC TAPE UNIT NO 1 | 91AAM010A |
| 18B | CR RAD MON CONSOLE | 96AAB001A | | | | 50 | PPS MONITOR RACK | 99AAB007 | 72 | PCS SWITCHING LOGIC | 90CSB014 | 97 | CARD PUNCH WITH KEY BOARD | 91AAM012 |
| 19A | CLASS IE PANEL A | 96AAB003A | 41A | PRIMARY PPS BUFFER | 99PSB001A | 51 | PRIMARY ROD CONTROL & INSTR | 90CSB010 | 73A | PPS AUX EQUIPMENT ISOLATION & LOGIC | 99PSB005A | 98 | LINE PRINTER | 91AAM009 |
| 19B | CLASS IE PANEL B | 96AAB003B | 41B | PRIMARY PPS BUFFER | 99PSB001D | | | | 73B | PPS AUX EQUIPMENT ISOLATION & LOGIC | 99PSB005B | 99 | CARD READER | 91AAM005 |
| 19C | CLASS IE PANEL C | 96AAB003C | 41C | PRIMARY PPS BUFFER | 99PSB001G | 53 | DESK TOP RADIO | ——— | 73C | PPS AUX EQUIPMENT ISOLATION & LOGIC | 99PSB005C | 100 | PROGRAMER CONSOLE | 91AAB001 |
| 20 | WEATHER STATION SPACE | | 42A | PRIMARY PPS TERMINATION CAB | 99PSB001B | 54 | REACTOR CONTROL PANEL | 90CSB013 | 73D | PPS AUX EQUIPMENT ISOLATION & LOGIC | 99PSB005D | 101 | COLOR CRT DISPLAY UNIT | 91AAM001F |
| 21A | PAX TELEPHONE | | 42B | PRIMARY PPS TERMINATION CAB | 99PSB001E | 55 | SUPERVISORY CONTROL PANEL | 90AAB005A | 74 | PORTABLE RADIO COMMUNICATION | TBD | 102 | MAGNETIC TAPE UNIT NO 4 | 91AAM010D |
| 21B | PAIC | | 42C | PRIMARY PPS TERMINATION CAB | 99PSB001H | 56 | FLOW CONTROL PANEL | 90CSB011 | 75 | SEC ROD CONTROL CABINET | TBD | 103 | CONTAINMENT INSTM DIVISION II | 27ICB005 |

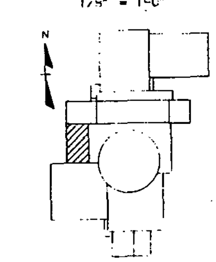
Figure 1.2-72
General Arrangement
Control Building
El. 816'-0" and 794'-0"

GENERAL NOTES
SEE DRAWING NN900

REFERENCE DRAWINGS
SEE DRAWING NN900




| ITEM | DESCRIPTION | EQUIPMENT NO | ITEM | DESCRIPTION | EQUIPMENT NO | ITEM | DESCRIPTION | EQUIPMENT NO |
|---|---|---|---|---|---|---|---|---|
| 1A | CHARGER | 12NIE011A | 8B | INTERMEDIATE Na PUMP #2 | 56INB207C | 19B | OIL PUMP SKID | 56PRP206B |
| 1B | CHARGER | 12NIE011B | 8C | INTERMEDIATE Na PUMP #3 | 56INB207B | 20A | INTERMEDIATE Na PUMP #1 MG SET | 56INK202A |
| 1C | CHARGER | 12NIE012A | 9A | INTERMEDIATE Na PUMP #1 | 56INB209A | 20B | INTERMEDIATE Na PUMP #2 MG SET | 56INK202B |
| 1D | CHARGER | 12NIE012B | 9B | INTERMEDIATE Na PUMP #2 | 56INB209B | 21A | PRIMARY Na PUMP #1 MG SET | 56PRK202A |
| 1E | CHARGER | 12NIE012C | 9C | INTERMEDIATE Na PUMP #3 | 56INB209C | 21B | PRIMARY Na PUMP #2 MG SET | 56PRK202B |
| 1F | CHARGER | 12NIE012D | 10A | PRIMARY Na PUMP #1 | 56PRB203A | | | |
| 1G | CHARGER | 12NIE012E | 10B | PRIMARY Na PUMP #2 | 56PRB203B | | | |
| 1H | CHARGER | 12NIE012F | 10C | PRIMARY Na PUMP #3 | 56PRB203C | | | |
| 2A | INVERTER | 12NIE013A | 11A | PRIMARY Na PUMP #1 | 56PRB207A | | | |
| 2B | INVERTER | 12NIE013B | 11B | PRIMARY Na PUMP #2 | 56PRB207C | | | |
| 2C | INVERTER | 12NIE014A | 11C | PRIMARY Na PUMP #3 | 56PRB207B | | | |
| 2D | INVERTER | 12NIE014B | 12A | PRIMARY Na PUMP #1 | 56PRB209A | | | |
| 2E | INVERTER | 12NIE014C | 12B | PRIMARY Na PUMP #2 | 56PRB209B | | | |
| 2F | INVERTER | 12NIE014D | 12C | PRIMARY Na PUMP #3 | 56PRB209C | | | |
| 3A | TRANSFORMER | 12NIX040A | 13A | SUMP PUMP | 21CBP002A | | | |
| 3B | TRANSFORMER | 12NIX040B | 13B | SUMP PUMP | 21CBP002B | | | |
| 3C | TRANSFORMER | 12NIX040C | | | | | | |
| 3D | TRANSFORMER | 12NIX050A | 15A | MG SET UNIT COOLER | 25ACA421 | | | |
| 3E | TRANSFORMER | 12NIX050B | 15B | MG SET UNIT COOLER | 25ACA422 | | | |
| 4A | PRIMARY CONTROL ROD MG SET | 90CDK001A | 15C | MG SET UNIT COOLER | 25ACA423 | | | |
| 4B | PRIMARY CONTROL ROD MG SET | 90CDK001B | 15D | MG SET UNIT COOLER | 25ACA424 | | | |
| 5 | ELEVATOR | 21CRC001 | 16A | HEAT EXCHANGERS SKID (TYP) | 56INH205A | | | |
| 6A | FIRE PROTECTION AREA PANEL | 26BDB041 | 16B | HEAT EXCHANGERS SKID (TYP) | 56INH205B | | | |
| 6B | FIRE PROTECTION AREA PANEL | 26BDB019 | 17A | HEAT EXCHANGERS SKID (TYP) | 56PRB205A | | | |
| 7A | INTERMEDIATE Na PUMP #1 | 56INB203A | 17B | HEAT EXCHANGERS SKID (TYP) | 56PRB205B | | | |
| 7B | INTERMEDIATE Na PUMP #2 | 56INB203C | 18A | OIL PUMP SKID | 56INP206A | | | |
| 7C | INTERMEDIATE Na PUMP #3 | 56INB203B | 18B | OIL PUMP SKID | 56INP206B | | | |
| 8A | INTERMEDIATE Na PUMP #1 | 56INB207A | 19A | OIL PUMP SKID | 56PRP206A | | | |

Figure 1.2-73
General Arrangement
Control Building
El. 765'-0" and 733'-0"

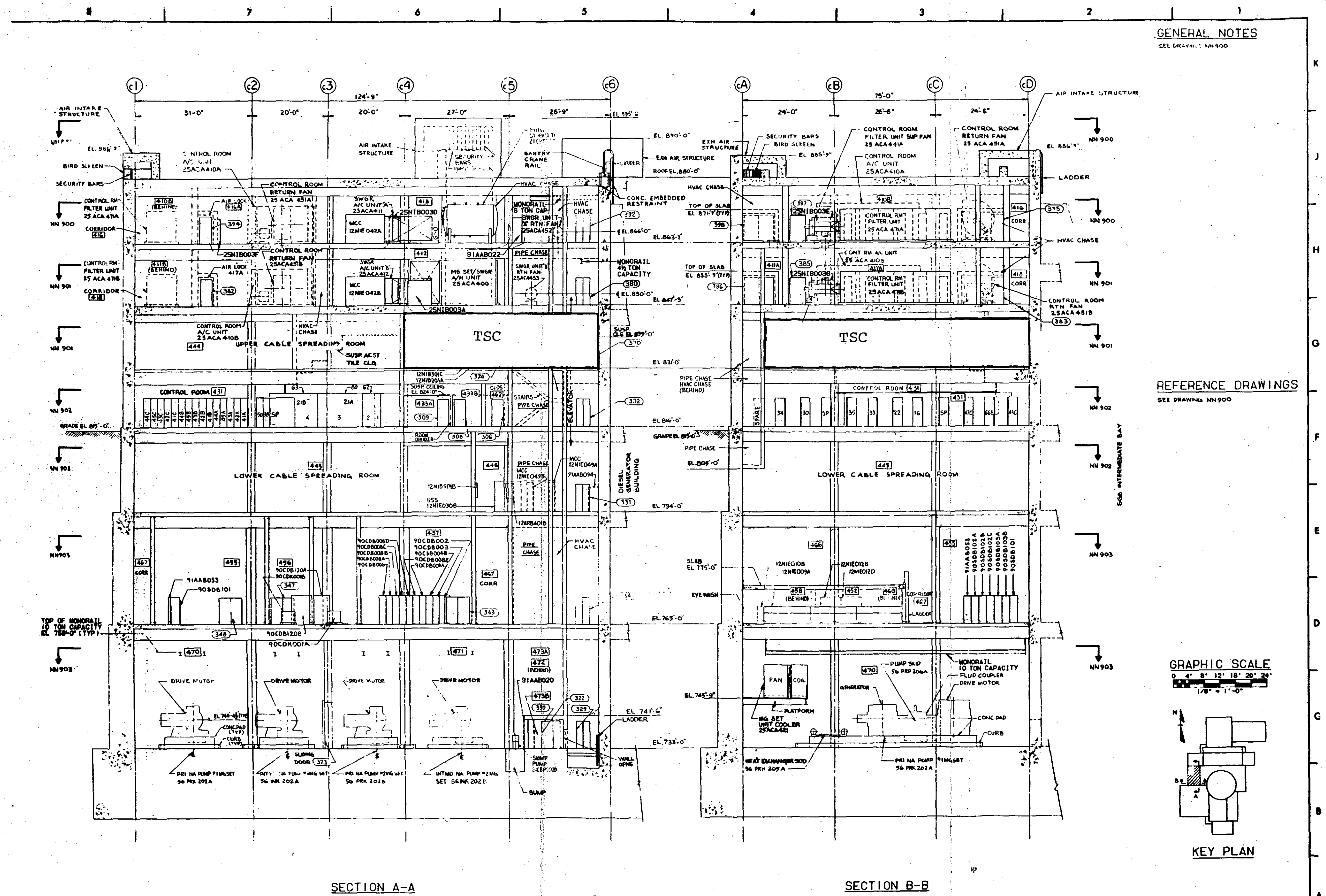



Figure 1.2-74
General Arrangement
Control Building
Section A-A and B-B

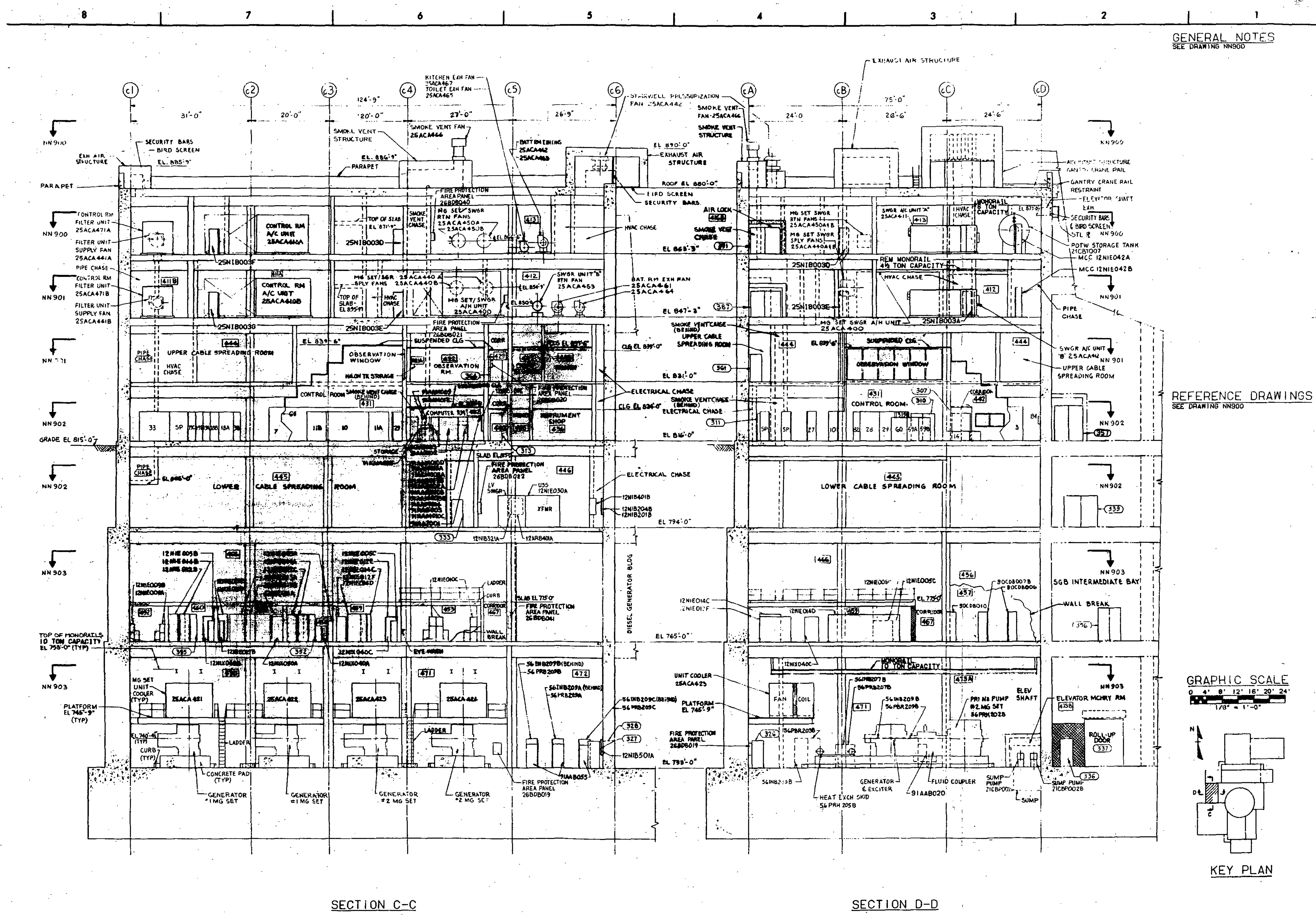
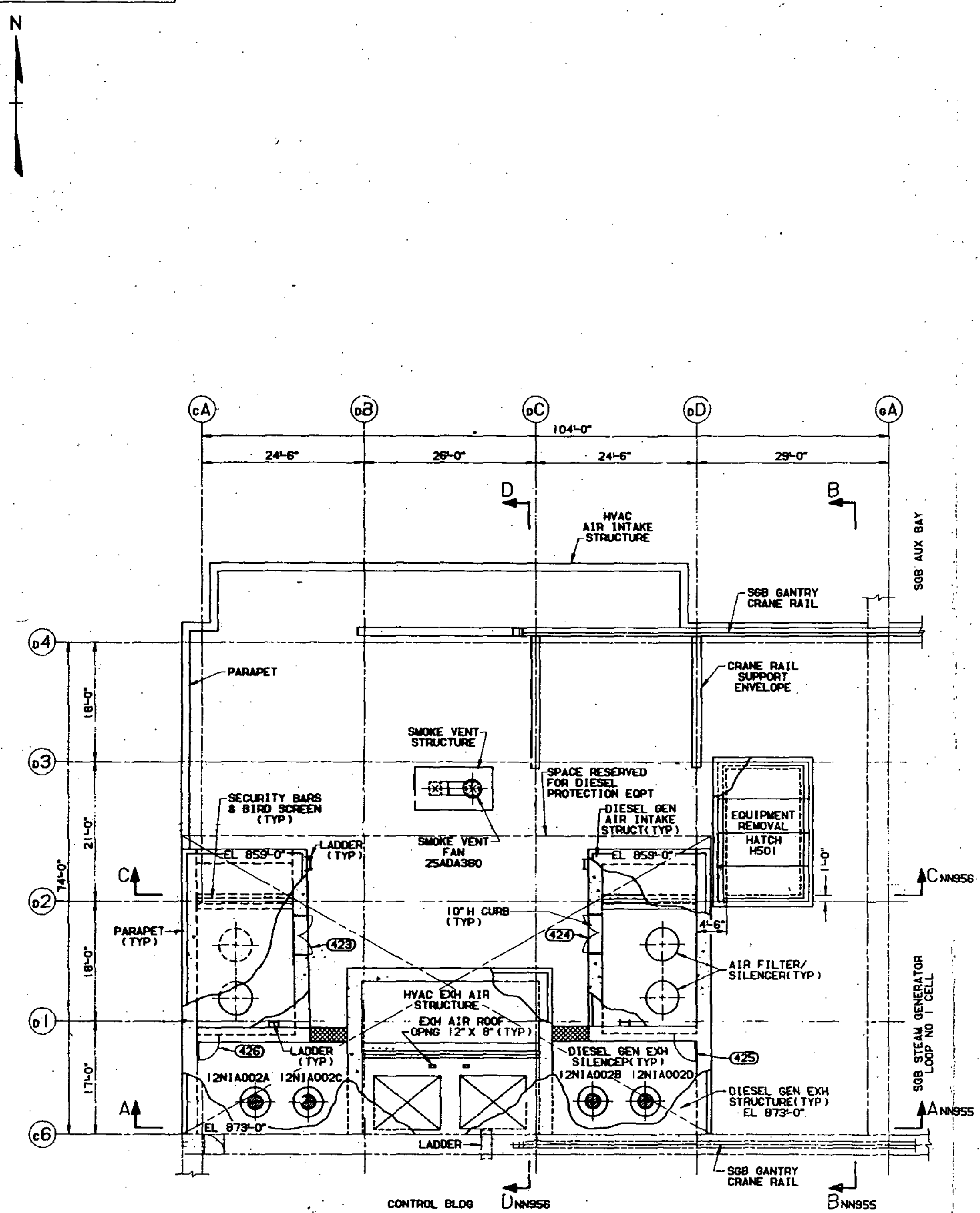



Figure 1.2-75
General Arrangement
Control Building
Section C-C and D-D

DWG NO NN952




ROOF PLAN ABOVE EL 847'-3"

## GENERAL NOTES

1. SYMBOLS & ABBREVIATIONS PER WARD DOC-D-0036
2. ALL LEVELS IN THIS BUILDING ARE CLASSIFIED AS "UNRESTRICTED RADIATION ZONE"
3. EQUIPMENT REMOVAL HATCH WILL BE SERVICED BY THE EXTENSION OF THE SGB GANTRY CRANE

## REFERENCE DRAWINGS

NN953 DGB GENERAL ARRANGEMENT-PLAN ABOVE EL 816'-0" & EL 794'-0"

NN954 DGB GENERAL ARRANGEMENT-PLAN ABOVE EL 765'-0" & EL 733'-0"

NN955 DGB GENERAL ARRANGEMENT SECTION A-A & B-B

NN956 DGB GENERAL ARRANGMENT SECTION C-C & D-D

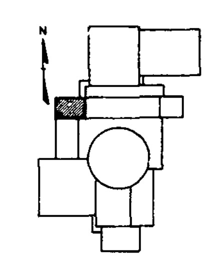
## LEGEND

- EI&C CABINET.SWGR (HEAVY LINE FRONT)
- GRATING
- 410 CELL NUMBER
- KNOCKOUT SECTION
- CONCRETE
- CONCRETE BLOCK
- CHECKERED PLATE
- 400 DOOR NUMBER

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"



KEY PLAN

Figure 1.2-76
General Arrangement
Diesel Generator Building
El. 847'-3"

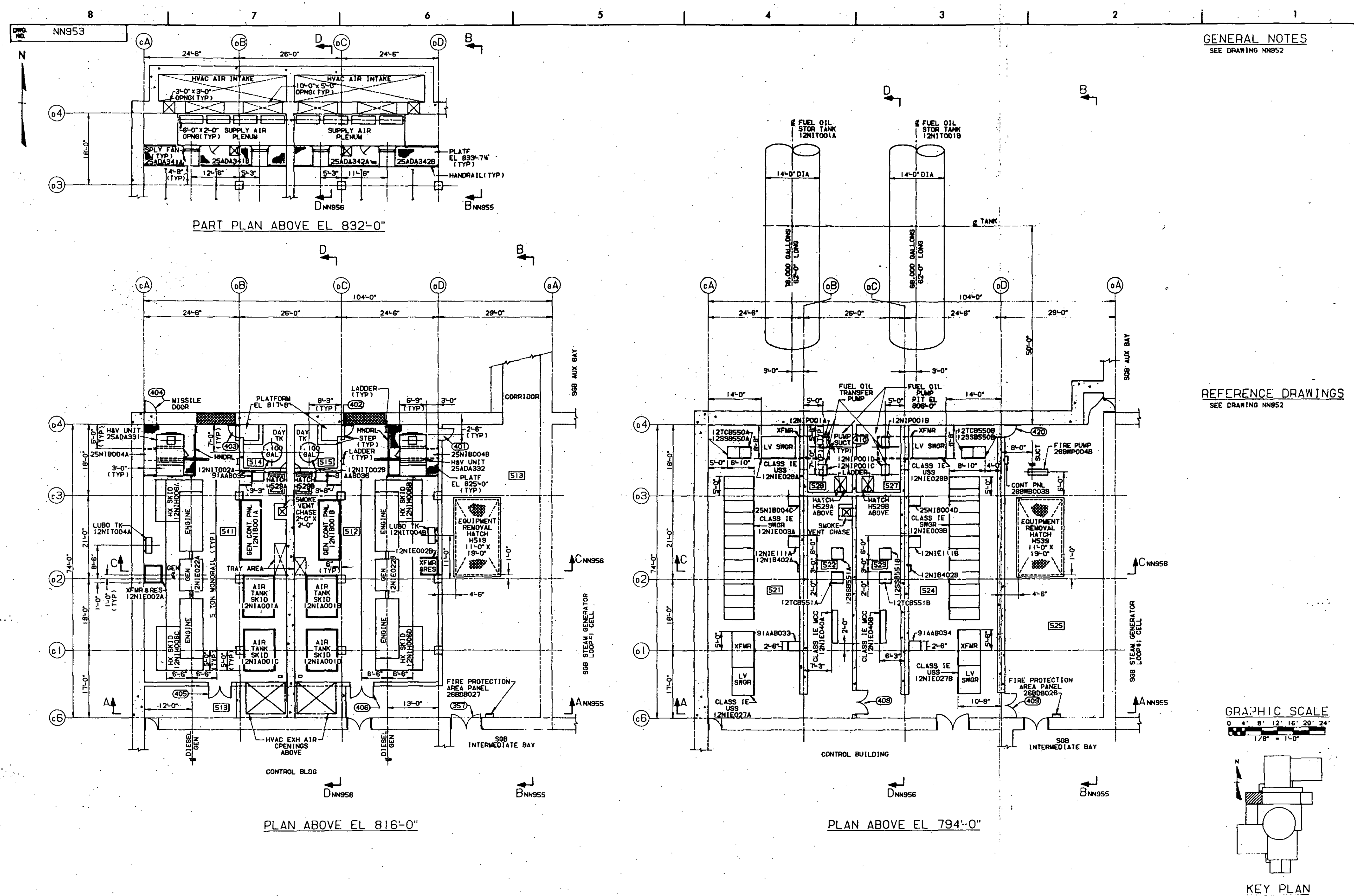
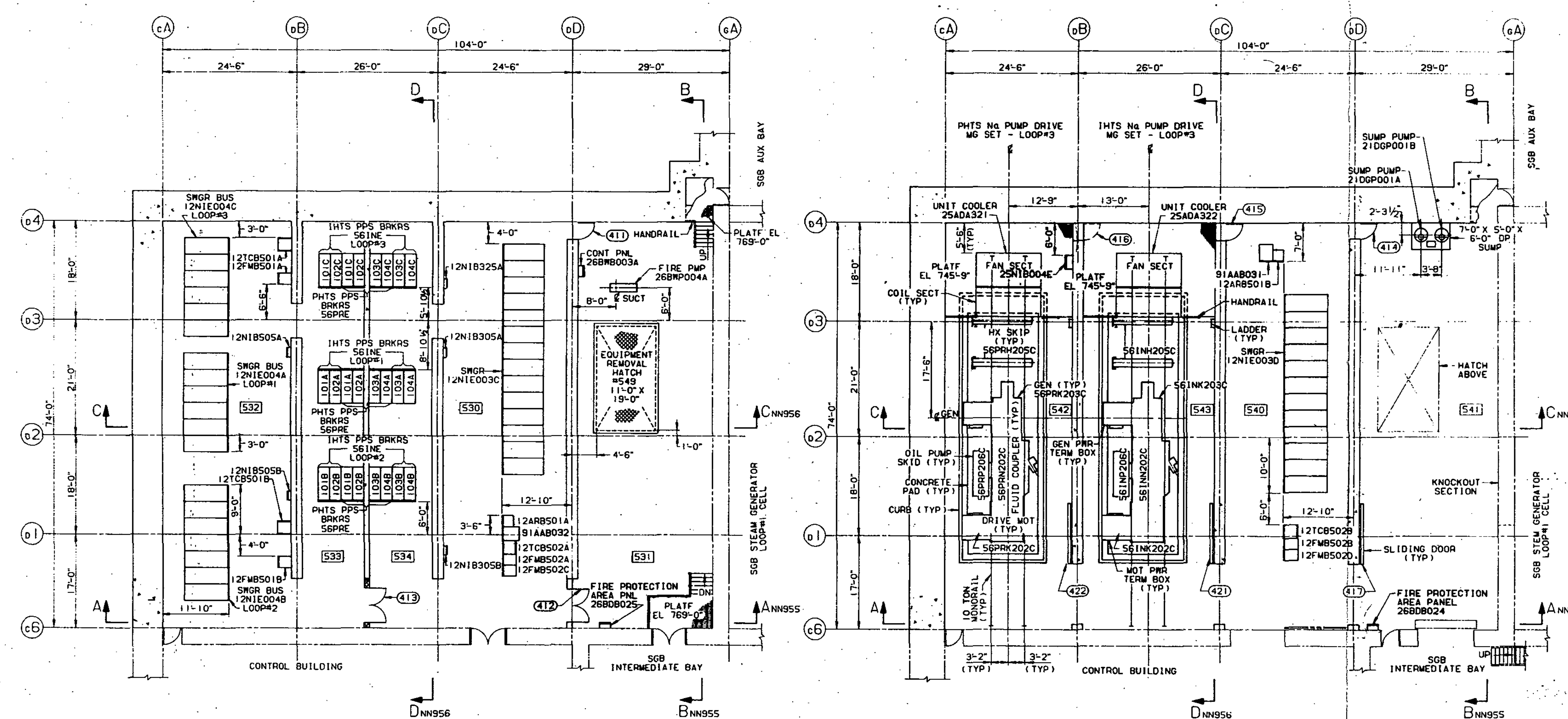



Figure 1.2-77
General Arrangement
Diesel Generator Building
El. 816'-0" and 794'-0"

DWG. NO. NN954

GENERAL NOTES
SEE DRAWING NN952

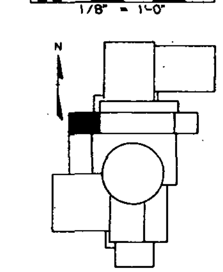
REFERENCE DRAWINGS
SEE DRAWING NN952




PLAN ABOVE EL 765'-0"

PLAN ABOVE EL 733'-0"

GRAPHIC SCALE
1/8" = 1'-0"



KEY PLAN

Figure 1.2-78
General Arrangement
Diesel Generator Building
El. 765'-0" and 733'-0"

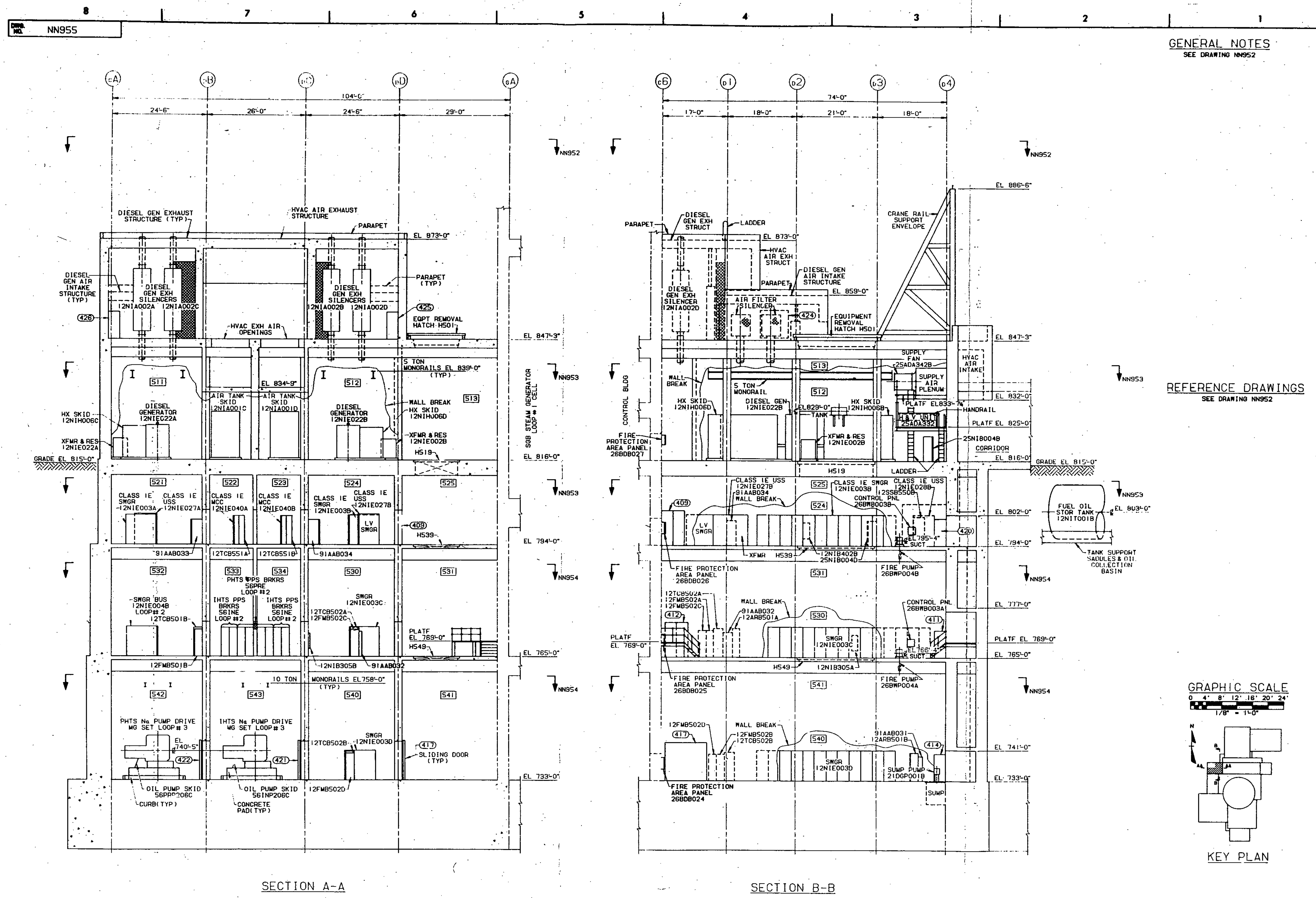



Figure 1.2-79
General Arrangement
Diesel Generator Building
Section A-A and B-B

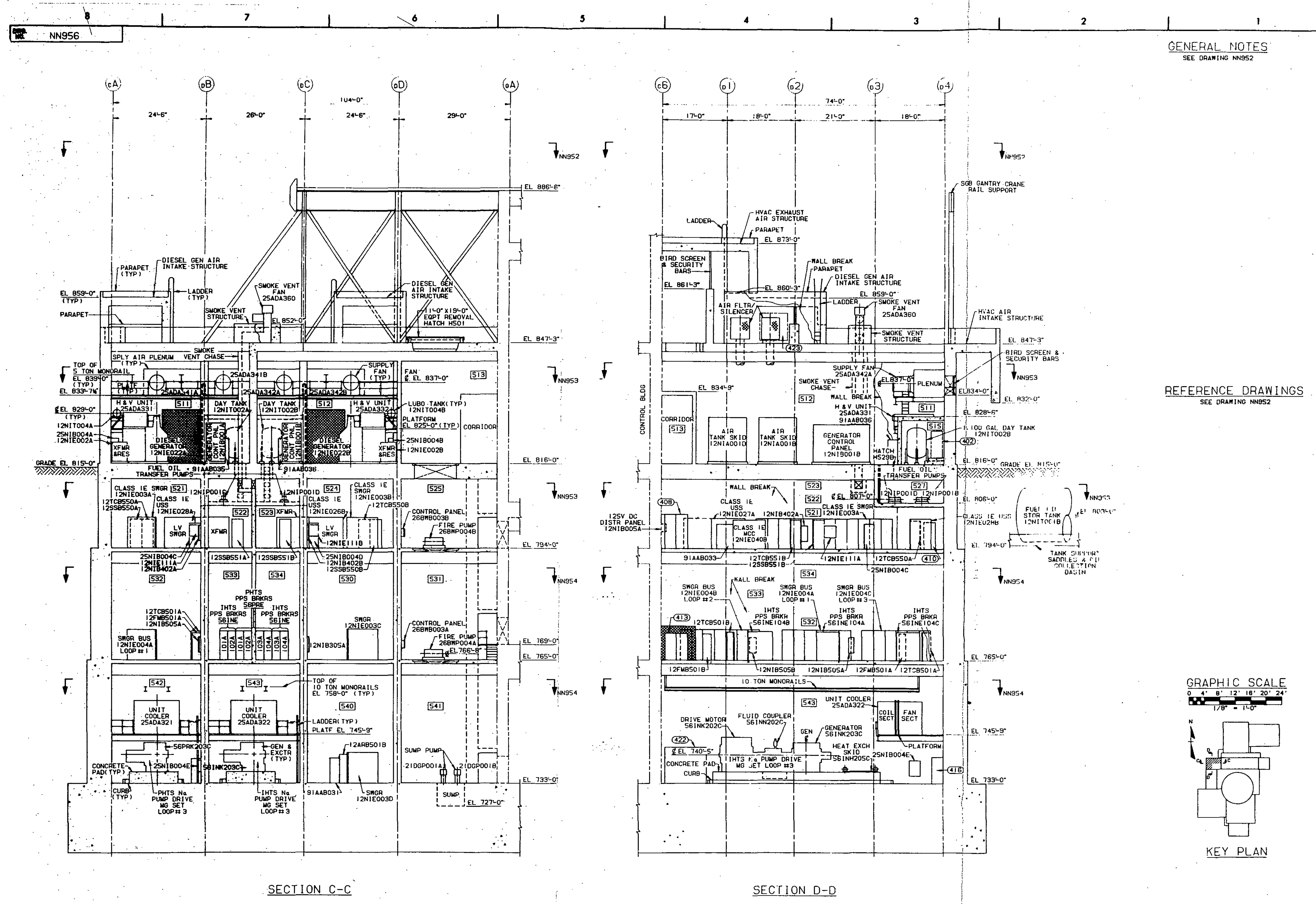
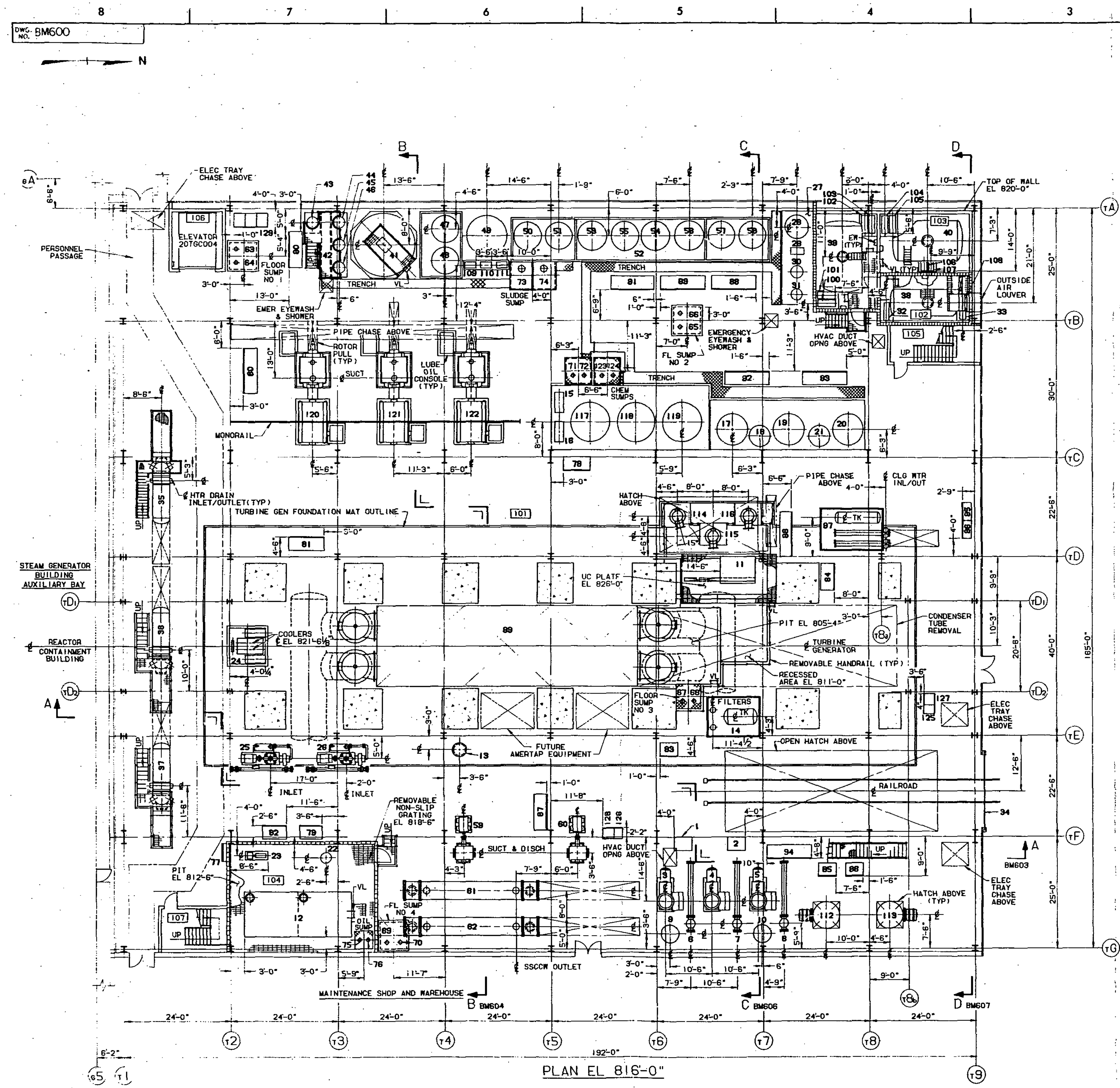



Figure 1.2-80
General Arrangement
Diesel Generator Building
Section C-C and D-D




| ITEM NO | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
|---|---|---|---|
| 1 | C5 | 221AA001A | SERVICE & INSTRUMENT AIR DRYERS |
| 2 | C5 | 221AA001B | |
| 3 | B5 | 22SAA001A | SERVICE & INSTRUMENT AIR COMPRESSORS |
| 4 | B5 | 22SAA001B | |
| 5 | B5 | 22SAA001C | |
| 6 | B5 | 22SAA002A | SERVICE & INSTRUMENT AIR AFTERCOOLER & SEPARATORS |
| 7 | B5 | 22SAA002B | |
| 8 | B4 | 22SAA002C | |
| 9 | B5 | 22SAT001A | SERVICE & INSTRUMENT AIR RECEIVERS |
| 10 | B5 | 22SAT001B | |
| 11 | E5 | 25BGA321 | UNIT COOLER |
| 12 | B7 | 72TST001 | LUBE OIL STORAGE TANK |
| 13 | D6 | 71CDH008 | DRAIN COOLER |
| 14 | D5 | 72TAM003 | HYDROGEN SEAL OIL SUPPLY UNIT |
| 15 | G6 | 71CPP001A | CONDENSATE POLISHERS RECYCLE PUMPS |
| 16 | G6 | 71CPP001B | |
| 17 | G5 | 71CPT002 | CATION TANK |
| 18 | G5 | 71CPT003 | ANION TANK |
| 19 | G4 | 71CPT004 | MIXED RESIN STORAGE TANK |
| 20 | G4 | 71CPT005 | AMMONIA RECLAIM TANK |
| 21 | G4 | 71CPT006 | HOT WATER TANK |
| 22 | C7 | 72TSM001 | LUBE OIL PURIFIER |
| 23 | C7 | 72TSP001 | LUBE OIL TRANSFER PUMP |
| 24 | E7 | 72TAM004 | EHC HYD FLUID POWER UNIT |
| 25 | D7 | 73CEM100A | VACUUM PUMP UNITS |
| 26 | D7 | 73CEM100B | |
| 27 | H4 | 75CFG001 | CONDENSATE CHEMICAL FEED |
| 28 | J4 | 75MUG009 | HOT WATER TANK |
| 29 | H4 | | BLOWER |
| 30 | H4 | | AMMONIA TANK |
| 31 | H4 | | HYDRAZINE TANK |
| 32 | H4 | 75CFP001A | AMMONIA STORAGE TANK METERING PUMPS |
| 33 | H4 | 75CFP001B | |
| 34 | C3 | 20TGC001 | ROLL-UP DOOR |
| 35 | F8 | 71FWH007A | TOPPING HEATER |
| 36 | E8 | 71FWH007B | |
| 37 | C8 | 71FWH007C | |
| 38 | H4 | 75CFT001 | AMMONIA STORAGE TANK |
| 39 | H4 | 75CFT002 | ACID STORAGE TANK |
| 40 | H4 | 75CFT003 | CAUSTIC STORAGE TANK |
| 41 | H6 | 75MUG001 | CLARIFIER |
| 42 | H7 | 75MUG002 | CLARIFIER CHEMICAL FEED |
| 43 | J7 | | CAUSTIC TANK |
| 44 | J7 | | HYPOCHLORITE TANK |
| 45 | H7 | | COAGULANT TANK |
| 46 | H7 | | ALUM TANK |
| 47 | J6 | 75MUG003A | CLARIFIER FILTERS |
| 48 | H6 | 75MUG003B | |
| 49 | J6 | 75MUG004 | CLEARWELL |
| 50 | J6 | 75MUG005A | ACTIVATED CARBON FILTERS |
| 51 | J6 | 75MUG005B | |
| 52 | H5 | 75MUG006 | MAKE UP DEMINERALIZERS |
| 53 | J5 | | A CATION TANKS |
| 54 | J5 | | B |
| 55 | J5 | | A ANION TANKS |
| 56 | J5 | | B |
| 57 | J5 | | A MIXED BED TANKS |
| 58 | J5 | | B |
| 59 | C6 | 75SWP001A | SEC SVCE CL CLG WTR PUMP |
| 60 | C6 | 75SWP001B | |
| 61 | B6 | 75SWH001A | SEC SVCE CL CLG WTR HT EXCH |
| 62 | B6 | 75SWH001B | |
| 63 | H7 | 76WDP001A | FLOOR DRAIN SUMP PUMPS |
| 64 | H7 | 76WDP001B | |
| 65 | G4 | 76WDP002A | |
| 66 | G4 | 76WDP002B | |
| 67 | D5 | 76WDP003A | |
| 68 | D5 | 76WDP003B | |
| 69 | B6 | 76WDP004A | |
| 70 | B6 | 76WDP004B | |
| 71 | G6 | 76WDP005A | CHEMICAL SUMP PUMPS |
| 72 | G6 | 76WDP005B | |
| 73 | H6 | 76WDP008A | SLUDGE SUMP PUMPS |
| 74 | H6 | 76WDP008B | |
| 75 | B7 | 76WDP009A | LUBE OIL SUMP PUMPS |
| 76 | B7 | 76WDP009B | |
| 77 | C7 | 26BDB032 | LOCAL FIRE PROTECTION PANEL |
| 78 | F5 | 69AAB003 | INSTRUMENT RACKS |
| 79 | C7 | 69AAB004C | |
| 80 | G7 | 69AAB006 | |
| 81 | F7 | 69AAB008A | |
| 82 | C7 | 69AAB008B | |
| 83 | D5 | 69AAB008D | |
| 84 | E4 | 69AAB008E | |
| 85 | B4 | 69AAB010A | |
| 86 | B4 | 69AAB010B | |
| 87 | C6 | 69AAB011 | |
| 88 | H5 | 69AAB013A | |
| 89 | H5 | 69AAB013B | |
| 90 | H7 | 69AAB013C | |
| 91 | H5 | 69AAB013D | |
| 92 | G5 | 69AAB014A | |
| 93 | G4 | 69AAB014B | |
| 94 | C4 | 69AAB016 | |

## EQUIPMENT LEGEND

| ITEM NO | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
|---|---|---|---|
| 95 | F3 | 69AAB018 | INSTRUMENT RACKS |
| 96 | F3 | 69AAB019 | |
| 97 | F4 | 72TAM001 | STATOR WINDING CLG UNIT |
| 98 | F4 | 69AAB008F | HYDROGEN & STATOR COOLING WATER CABINET |
| 99 | E6 | 73CWH001 | CONDENSER |
| 100 | H4 | 71CPP002A | ACID STORAGE TANK METERING PUMPS |
| 101 | H4 | 71CPP002B | |
| 102 | H4 | 75MUP007A | |
| 103 | H4 | 75MUP007B | |
| 104 | H4 | 71CPP003A | CAUSTIC STORAGE TANK METERING PUMPS |
| 105 | H4 | 71CPP003B | |
| 106 | H4 | 75MUP008A | |
| 107 | H4 | 75MUP008B | |
| 108 | H4 | 25BGA003 | HEAT & VENTILATING UNIT |
| 109 | H6 | 75MUP001A | CLEARWELL PUMPS |
| 110 | H6 | 75MUP001B | |
| 111 | H6 | 75MUP001C | |
| 112 | B4 | 72ASH001A | AUXILIARY BOILERS |
| 113 | B4 | 72ASH001B | |
| 114 | F5 | 71CDP001A | CONDENSATE PUMPS |
| 115 | F5 | 71CDP001B | |
| 116 | F5 | 71CDP001C | |
| 117 | G5 | 71CPT001A | CONDENSATE POLISHERS |
| 118 | G5 | 71CPT001B | |
| 119 | G5 | 71CPT001C | |
| 120 | G7 | 71FWP001A | STEAM GENERATOR FEED PUMPS |
| 121 | G6 | 71FWP001B | |
| 122 | G6 | 71FWP001C | |
| 123 | G5 | 76WDP005C | CHEMICAL SUMP PUMPS |
| 124 | G5 | 76WDP005D | |
| 125 | C4 | 12FMB001A | FIELD MULTIPLEXING CABINETS |
| 126 | C5 | 12FMB001B | |
| 127 | C4 | 12TCB001A | FIELD MULTIPLEXING TERMINAL CABINETS |
| 128 | C5 | 12TCB001B | |
| 129 | H7 | 91AAB101 | REMOTE DATA ACQUISITION TERMINAL |

## GENERAL NOTES

SEE DRAWING BM601

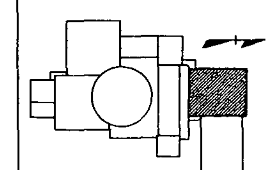
## REFERENCE DRAWINGS

SEE DRAWING BM601

## GRAPHIC SCALE

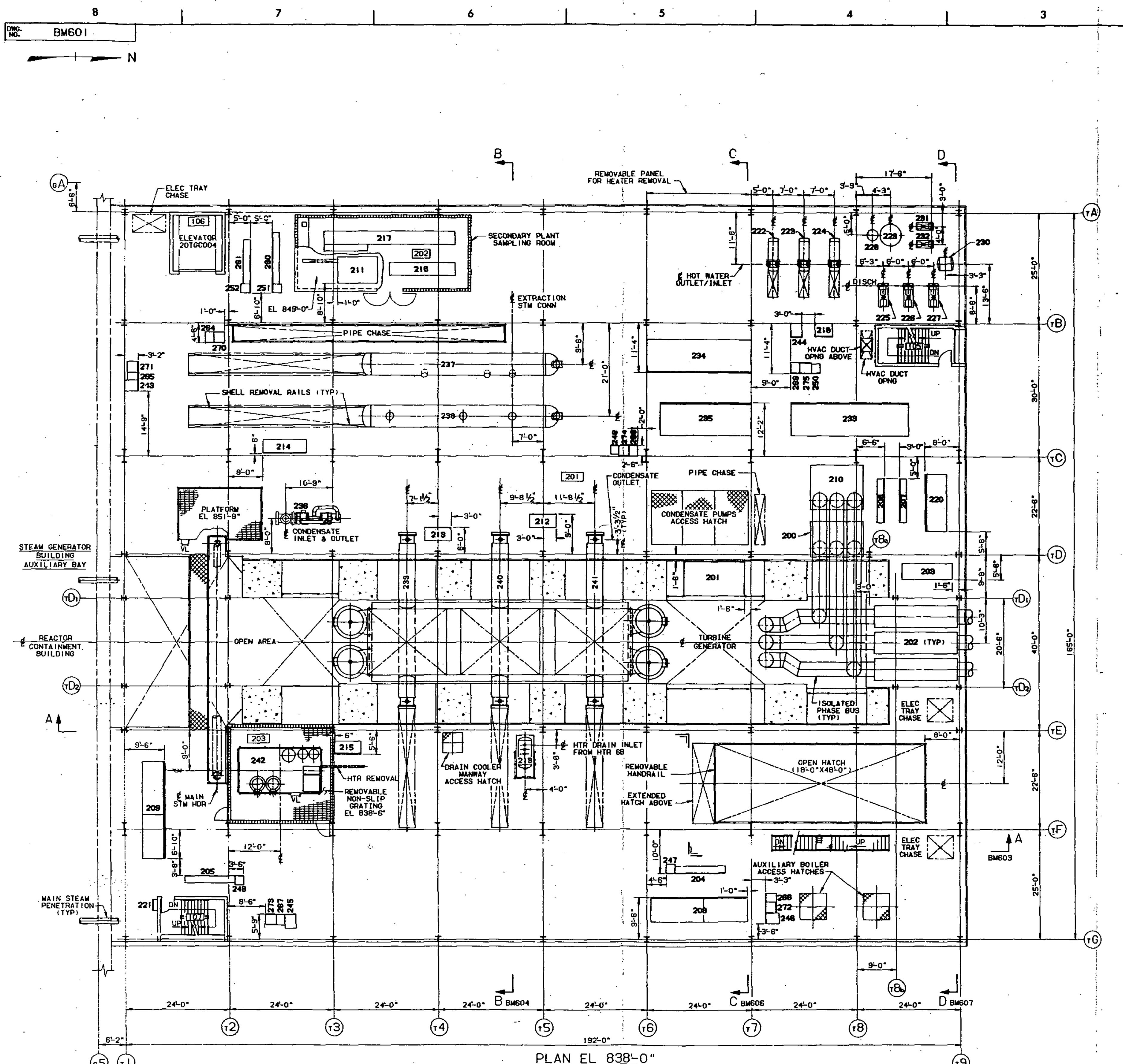
0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"




KEY PLAN

Figure 1.2-81
General Arrangement
Turbine Generator Building
Plan El. 816'-0"




## EQUIPMENT LEGEND

| ITEM NO | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
|---|---|---|---|
| 200 | F4 | 11AAE008 | PT & SURGE PROTECTION CUBICLE |
| 201 | E5 | 11AAE009 | TURBINE GENERATOR GND XFMR & RES |
| 202 | E4 | 11AAE012 | GENERATOR LOAD BREAK SWITCH |
| 203 | E4 | 11AAE013 | COMPRESSOR LOAD BREAK SWITCH |
| 204 | B5 | 12BPE006A | MOTOR CONTROL CENTERS |
| 205 | B8 | 12BPE006B | |
| 206 | F4 | 12BPE007A | |
| 207 | F4 | 12BPE007B | |
| 208 | B5 | 12BPE022A | UNIT SUBSTATION |
| 209 | C8 | 12BPE022B | |
| 210 | F4 | 11AAH001 | ISOLATED PHASE BUS HEAT EXCHANGER |
| 211 | H7 | 25BGA322 | UNIT COOLER |
| 212 | F6 | 69AAB004A | INSTRUMENT RACKS |
| 213 | F6 | 69AAB004B | |
| 214 | G7 | 69AAB007A | |
| 215 | D7 | 69AAB008C | |
| 216 | H6 | 69AAB022 | MONITORING PANEL |
| 217 | H7 | 69AAB023 | SAMPLE PANEL |
| 218 | H4 | 69AAB030 | INSTRUMENT RACK |
| 219 | D6 | 71DVT001 | DRAIN TANK |
| 220 | F4 | 72TAM006 | EXCITATION CUBICLE |
| 221 | B8 | 26BD8031 | LOCAL FIRE PROTECTION PNL |
| 222 | H4 | 72ASH002A | HOT WATER HEAT EXCHANGERS |
| 223 | H4 | 72ASH002B | |
| 224 | H4 | 72ASH002C | |
| 225 | H4 | 75HWP001A | CIRCULATING WATER PUMPS |
| 226 | H4 | 75HWP001B | |
| 227 | H4 | 75HWP001C | |
| 228 | H4 | 75HWM002 | AIR SEPARATOR |
| 229 | H4 | 75HWT001 | EXPANSION TANK |
| 230 | H4 | 75HWH001 | MAKEUP WATER HEATER |
| 231 | H4 | 75HWP002A | MAKEUP WATER PUMPS |
| 232 | H4 | 75HWP002B | |
| 233 | G4 | 12BPE001A | SWITCHGEAR |
| 234 | G5 | 12BPE001B | |
| 235 | G5 | 12BPE001D | |
| 236 | F7 | 72TAH001 | STEAM PACKING EXHAUSTER |
| 237 | G6 | 71CDH004 | LP FEEDWATER HEATERS |
| 238 | G6 | 71CDH005 | |
| 239 | E6 | 71CDH006A | |
| 240 | E6 | 71CDH006B | |
| 241 | E5 | 71CDH006C | |
| 242 | C7 | 72TAM002 | LUBE OIL RESERVOIR |
| 243 | G8 | 91AAB102 | REMOTE DATA ACQUISITION TERMINALS |
| 244 | H4 | 91AAB103 | |
| 245 | B7 | 91AAB104 | |
| 246 | B4 | 91AAB105 | |
| 247 | C5 | 12ARB001A | AUXILIARY RELAY CABINETS |
| 248 | B7 | 12ARB001B | |
| 249 | G5 | 12ARB002A | |
| 250 | G4 | 12ARB002B | |
| 251 | H7 | 12ARB003A | |
| 252 | H7 | 12ARB003B | |
| 260 | H7 | 12BPE015A | MOTOR CONTROL CENTERS |
| 261 | H7 | 12BPE015B | |
| 264 | H7 | 12FMB002A | FIELD MULTIPLEXING CABINETS |
| 265 | G8 | 12FMB002B | |
| 266 | B4 | 12FMB003A | |
| 267 | B7 | 12FMB003B | |
| 268 | G5 | 12FMB004A | |
| 269 | G4 | 12FMB004B | |
| 270 | H7 | 12TCB002A | FIELD MULTIPLEXING TERMINAL CABINETS |
| 271 | G8 | 12TCB002B | |
| 272 | B4 | 12TCB003A | |
| 273 | B7 | 12TCB003B | |
| 274 | G5 | 12TCB004A | |
| 275 | G4 | 12TCB004B | |

## GENERAL NOTES

1. FOR LEGEND AND SYMBOLS, UNLESS OTHERWISE NOTED, REFER TO WARD DOCUMENT DOC-0036-STANDARD SYMBOLOGY FOR CRBRP DRAWINGS
2. [box] HEAVY LINE DENOTES FRONT OF UNIT
3. REMOVABLE HANDRAILS SHALL BE PROVIDED FOR ALL COVERED HATCHES
4. ACCESS AND MAINTENANCE PLATFORMS TO BE PROVIDED WHERE REQUIRED PENDING DEVELOPMENT OF PIPING AND EQUIPMENT DETAILS
5. [symbol] EW-EYE WASH
6. [104] ROOM NUMBER

## REFERENCE DRAWINGS

BM600 GENERAL ARRANGEMENT TURBINE GENERATOR BUILDING FIRST FLOOR PLAN EL 816'-0"

BM602 GENERAL ARRANGEMENT TURBINE GENERATOR BUILDING OPERATING FL PLAN EL 862'-0"

BM603 GENERAL ARRANGEMENT TURBINE GENERATOR BUILDING SECTION A-A

BM604 GENERAL ARRANGEMENT TURBINE GENERATOR BUILDING SECTION B-B

BM605 GENERAL ARRANGEMENT TURBINE GENERATOR BUILDING PARTIAL PLAN EL 892-0" & SECTION E-E

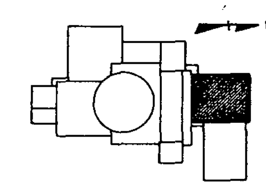
BM606 GENERAL ARRANGEMENT TURBINE GENERATOR BUILDING SECTION C-C

BM607 GENERAL ARRANGEMENT TURBINE GENERATOR BUILDING SECTION D-D

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"

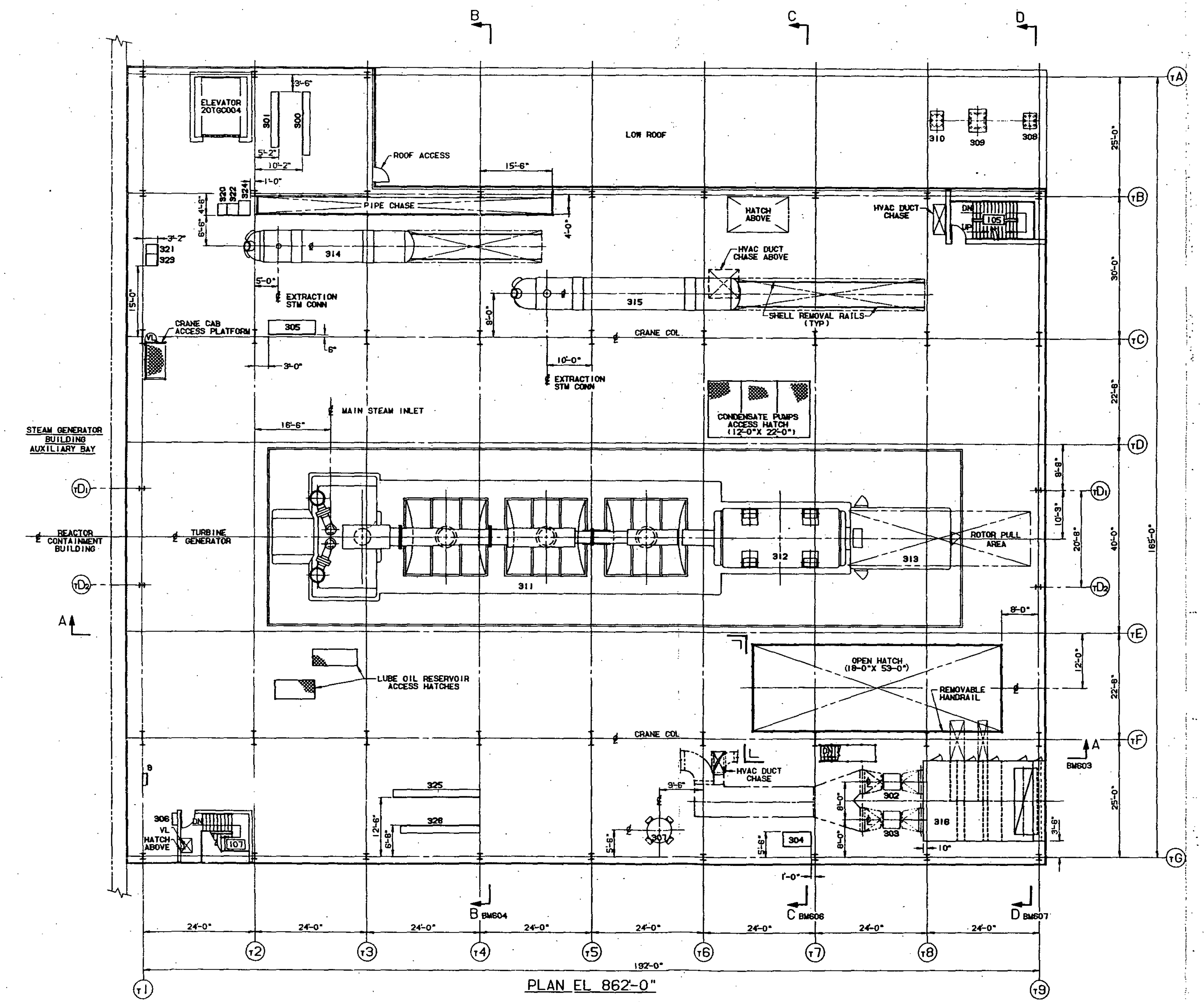


KEY PLAN

Figure 1.2-82
General Arrangement
Turbine Generator Building
Plan El. 838'-0"

Amend. 64

DWG. NO. BM602

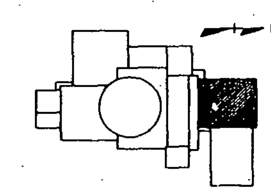



| ITEM NO | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
|---|---|---|---|
| 300 | H7 | 12BPE005A | MOTOR CONTROL CENTERS |
| 301 | H7 | 12BPE005B | |
| 302 | B4 | 25BGA241A | AIR HANDLING UNITS |
| 303 | B4 | 25BGA241B | SUPPLY FANS |
| 304 | B5 | 25BPB001B | INSTRUMENT RACKS |
| 305 | F7 | 69AAB007B | |
| 306 | B8 | 26BDB030 | LOCAL FIRE PROTECTION PNL |
| 307 | B5 | 75SWT001 | SSCCW SURGE TANK |
| 308 | H4 | 25BGA162 | EXHAUST FANS |
| 309 | H4 | 25BGA163 | |
| 310 | H4 | 25BGA361 | |
| 311 | E6 | 72TGM001 | TURBINE |
| 312 | E5 | 72TGM002 | GENERATOR |
| 313 | E4 | 72TAM005 | ALTERNATOR |
| 314 | G7 | 71FWH001 | H.P. FEEDWATER HEATERS |
| 315 | G5 | 71FWH002 | |
| 316 | B4 | 25BGA002 | AIR HANDLING UNIT |
| 320 | H7 | 12FMB005A | FIELD MULTIPLEXING |
| 321 | H8 | 12FMB005B | CABINETS |
| 322 | H7 | 12TCB005A | FIELD MULTIPLEXING |
| 323 | G8 | 12TCB005B | TERMINAL CABINETS |
| 324 | H7 | 91AAB106 | REMOTE DATA ACQUISITION TERMINAL |
| 325 | B6 | 12BPE016A | MOTOR CONTROL CENTERS |
| 326 | B6 | 12BPE016B | |

GENERAL NOTES
SEE DRAWING BM601

REFERENCE DRAWINGS
SEE DRAWING BM601

GRAPHIC SCALE
0 4' 8' 12' 16' 20' 24'
1/8" = 1'-0"



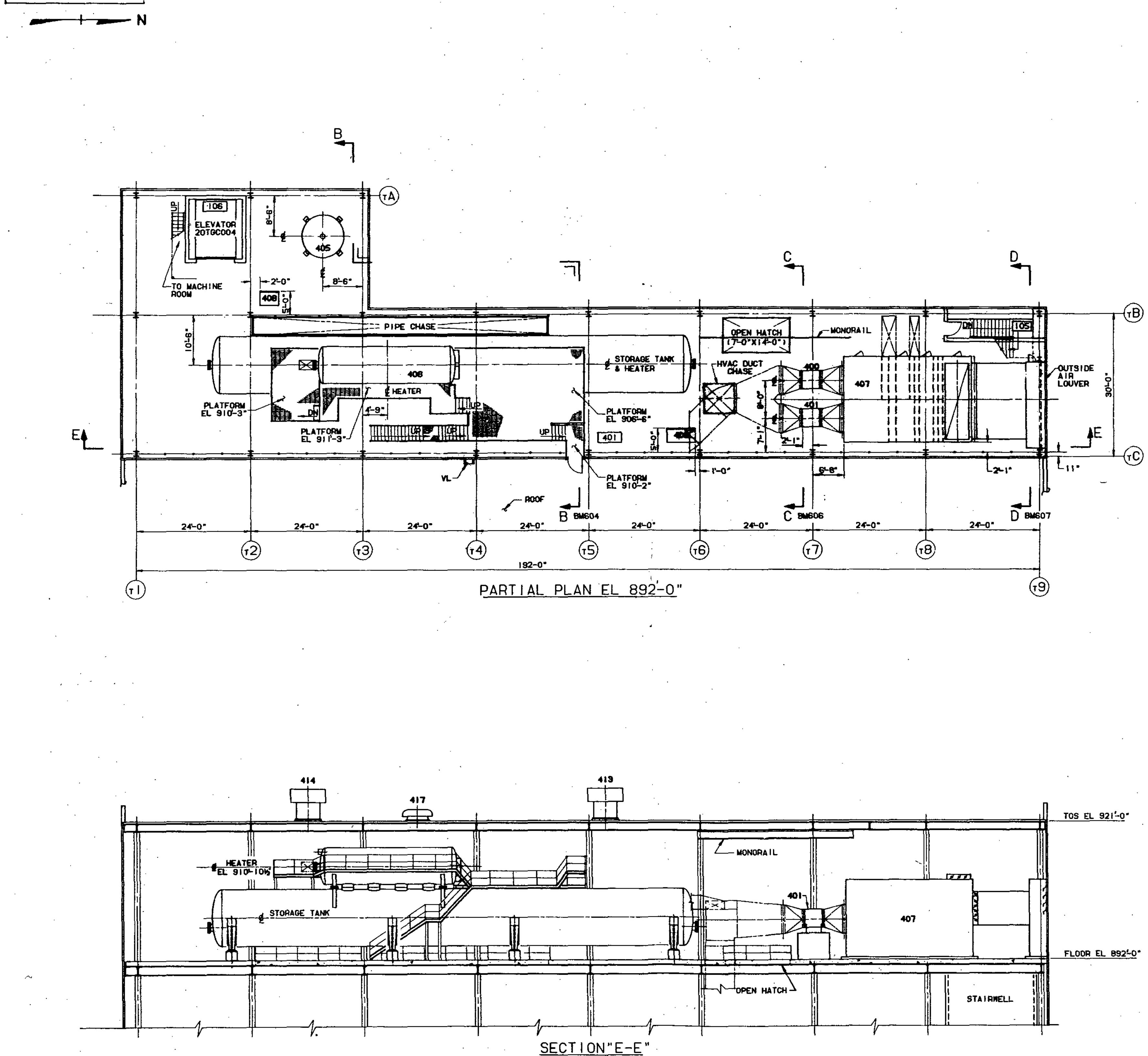
KEY PLAN

Figure 1.2-83
General Arrangement
Turbine Generator Building
Plan El. 862'-0"



BM602-4

DWG. NO. BM605



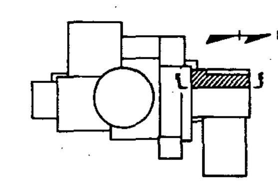

## EQUIPMENT LEGEND

| ITEM NO | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
|---|---|---|---|
| 400 | C4 | 25BGA141A | AIR HANDLING UNIT |
| 401 | G4 | 25BGA141B | SUPPLY FANS |
| 413 | C6 | 25BGA161A | EXHAUST FANS |
| 414 | C7 | 25BGA1619 | |
| 417 | C6 | 25BGA111 | RELIEF HOOD |
| 405 | H7 | 71DVT002 | FLASH TANK |
| 406 | G6 | 71FWH003 | DEAERATOR HEATER |
| 407 | G4 | 25BGA001 | AIR HANDLING UNIT |
| 408 | H7 | 69AAB029 | INSTRUMENT RACKS |
| 409 | G5 | 25BPB001A | |

GENERAL NOTES
SEE DRAWING BM601

REFERENCE DRAWINGS
SEE DRAWING BM601

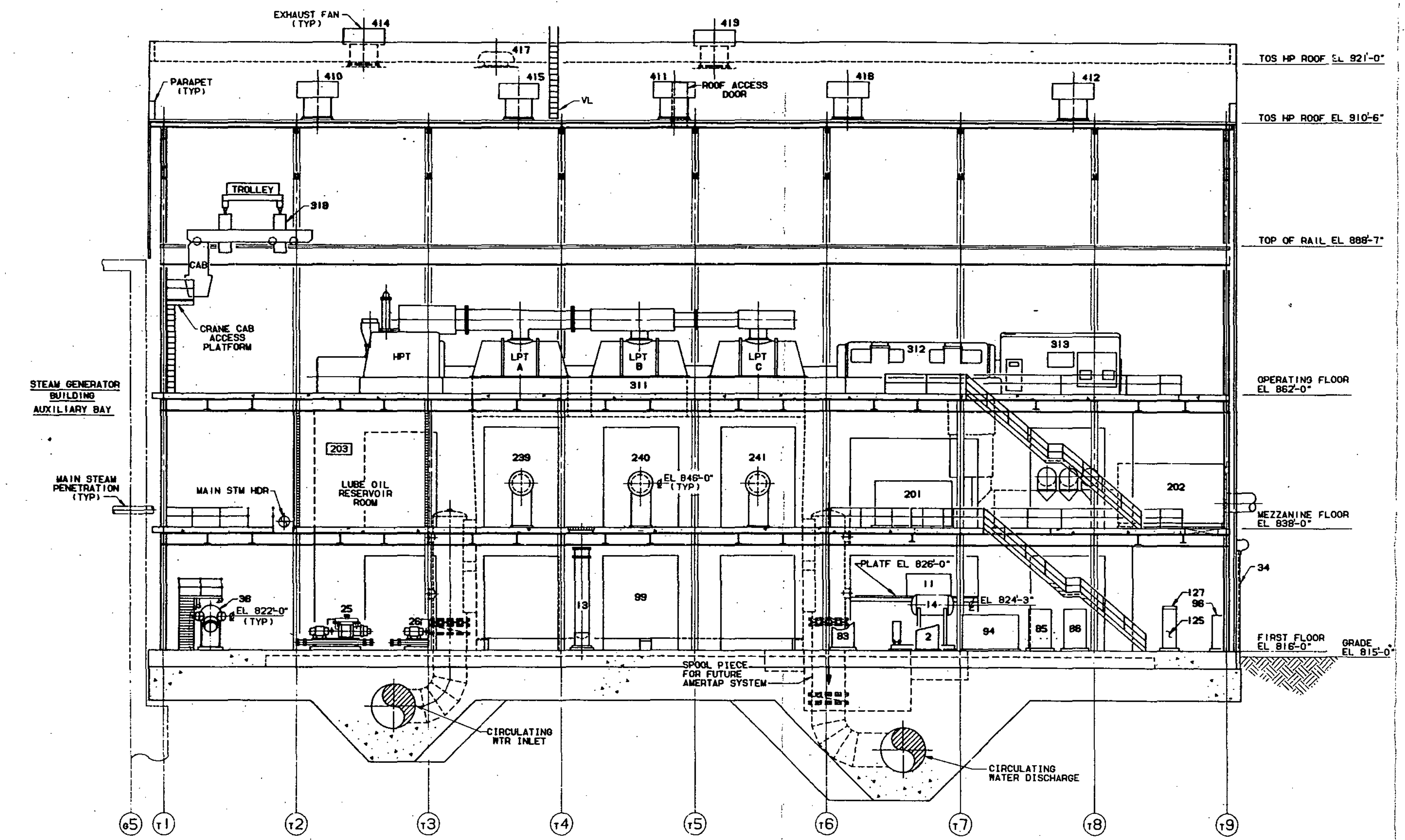
GRAPHIC SCALE
0 4' 8' 12' 16' 20' 24'
1/8" = 1'-0"



KEY PLAN

Figure 1.2-84
General Arrangement
Turbine Generator Building
Partial Plan and Section E-E

DWG. NO. BM603




SECTION "A-A"

## EQUIPMENT LEGEND

| ITEM NO | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
|---|---|---|---|
| 2 | D5 | 221AA001B | AIR DRYER |
| 11 | D5 | 258GA321 | UNIT COOLER |
| 13 | D6 | 71CDH008 | DRAIN COOLER |
| 14 | D5 | 72TAM003 | HYDROGEN SEAL OIL SUPPLY UNIT |
| 25 | D7 | 73CEM100A | VACUUM PUMP UNITS |
| 26 | D7 | 73CEM100B | |
| 34 | D3 | 20TGC001 | ROLL-UP DOOR |
| 36 | D8 | 71FWH007B | TOPPING HEATER |
| 83 | D5 | 69AAB008D | INSTRUMENT RACKS |
| 85 | D4 | 69AAB001A | |
| 86 | D4 | 69AAB001B | |
| 94 | D4 | 69AAB016 | |
| 96 | D4 | 69AAB017 | |
| 99 | D6 | 73CWH001 | CONDENSER |
| 125 | D4 | 12FMB001A | FIELD MULTIPLEXING CABINET |
| 127 | D4 | 12TCB001A | FIELD MULTIPLEXING TERMINAL CABINET |
| 201 | E5 | 11AAE009 | TURBINE GENERATOR GND XFMR & RESISTOR |
| 202 | E4 | 11AAE012 | GENERATOR LOAD BREAK SWITCH |
| 239 | E6 | 71CDH006A | LP FEEDWATER HEATERS |
| 240 | E6 | 71CDH006B | |
| 241 | E5 | 71CDH006C | |
| 311 | F6 | 72TGM001 | TURBINE |
| 312 | F5 | 72TGM002 | GENERATOR |
| 313 | F4 | 72TAM005 | ALTERNATOR |
| 318 | G7 | 20TGC003 | CRANE BRIDGE |
| 410 | H7 | 25BGA261A | EXHAUST FANS |
| 411 | H6 | 25BGA261B | |
| 412 | H4 | 25BGA261C | |
| 413 | H6 | 25BGA161A | |
| 414 | H7 | 25BGA161B | |
| 415 | H7 | 25BGA161C | |
| 416 | H5 | 25BGA161D | |
| 417 | H6 | 25BGA111 | RELIEF HOOD |

## GENERAL NOTES

SEE DRAWING BM601

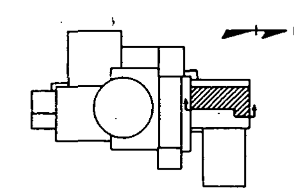
## REFERENCE DRAWINGS

SEE DRAWING BM601

## GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

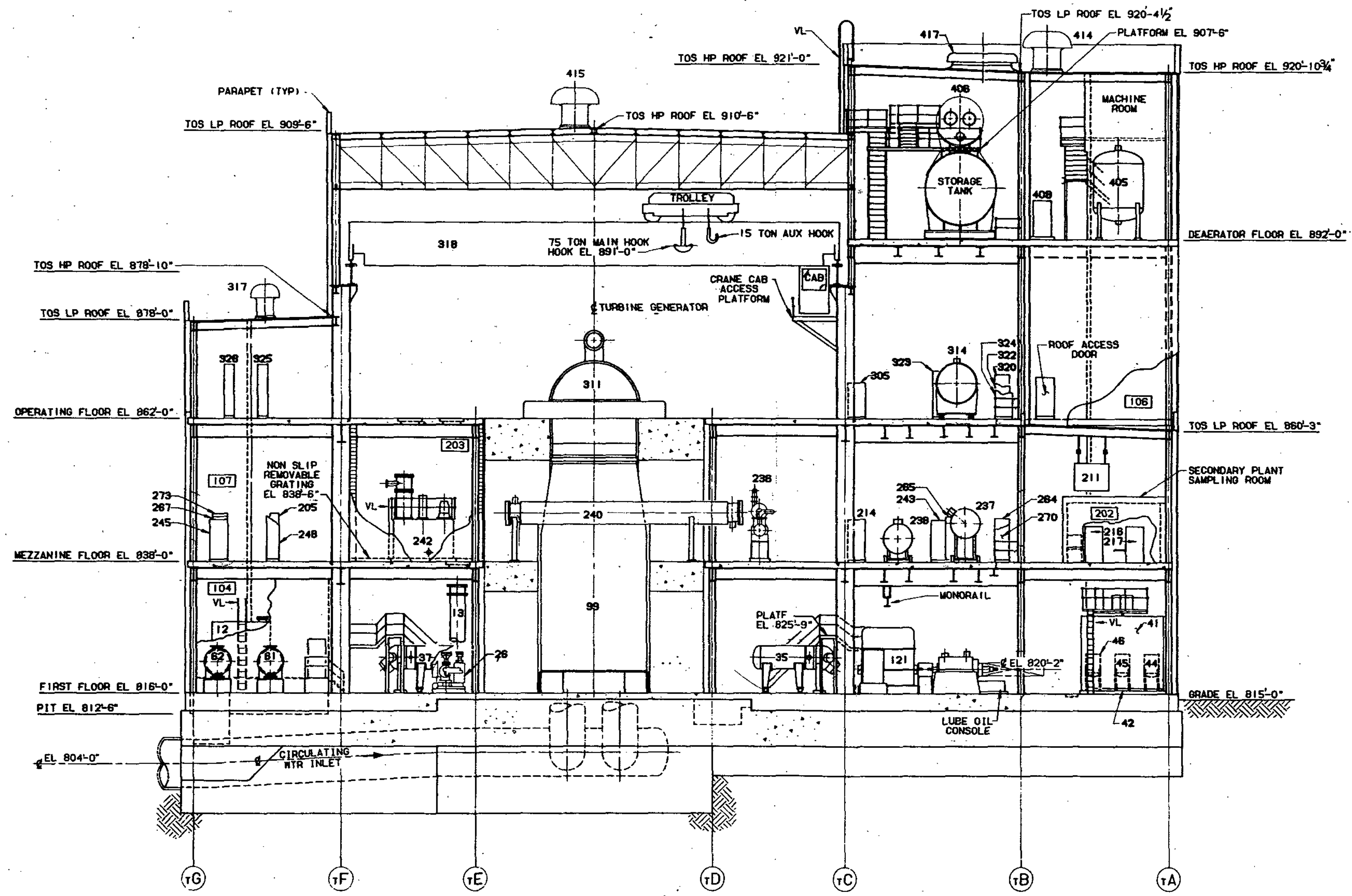
1/8" = 1'-0"




KEY PLAN

Figure 1.2-85
General Arrangement
Turbine Generator Building
Section A-A

| DWG. NO. | BM604 |
|---|---|




SECTION "B-B"

## EQUIPMENT LEGEND

| ITEM NO | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
|---|---|---|---|
| 12 | D7 | 72TST001 | LUBE OIL STORAGE TANK |
| 13 | D6 | 71CDH008 | DRAIN COOLER |
| 26 | D6 | 73CEM100B | VACUUM PUMP UNIT |
| 35 | D5 | 71FWH007A | TOPPING HEATERS |
| 37 | D6 | 71FWH007C | |
| 41 | D4 | 75MUG001 | CLARIFIER |
| 42 | D4 | 75MUG002 | CLARIFIER CHEMICAL FEED |
| 44 | D4 | | HYPOCHLORITE TANK |
| 45 | D4 | | COAGULANT TANK |
| 46 | D4 | | ALUM TANK |
| 61 | D7 | 75SCH001A | SEC SVCE CLOSED CLG WTR HEAT EXCHANGERS |
| 62 | D7 | 75SCH001B | |
| 99 | D6 | 73CWH001 | CONDENSER |
| 121 | D5 | 71FWP001B | STEAM GENERATOR FEED PUMP |
| 205 | E7 | 12BPE006B | MOTOR CONTROL CENTER |
| 211 | E4 | 25BGA322 | UNIT COOLER |
| 214 | E5 | 69AAB007A | INSTRUMENT RACK |
| 216 | E4 | 69AAB022 | MONITORING PANEL |
| 217 | E4 | 69AAB023 | SAMPLE PANEL |
| 236 | E5 | 72TAH001 | STEAM PACKING EXHAUSTER |
| 237 | E5 | 71CDH004 | LP FEEDWATER HEATERS |
| 238 | E5 | 71CDH005 | |
| 240 | E6 | 71CDH006B | |
| 242 | E7 | 72TAM002 | LUBE OIL RESERVOIR |
| 243 | E5 | 91AAB102 | REMOTE DATA ACQUISITION TERMINALS |
| 245 | E7 | 91AAB104 | |
| 248 | E7 | 12ARB001B | AUXILIARY RELAY CABINETS |
| 264 | E4 | 12FMB002A | FIELD MULTIPLEXING CABINETS |
| 265 | E5 | 12FMB002B | |
| 267 | E7 | 12FMB003B | |
| 270 | E4 | 12TCB002A | FIELD MULTIPLEXING TERMINAL CABINETS |
| 273 | E7 | 12TCB003B | |
| 301 | F7 | 12BPE005B | MOTOR CONTROL CENTER |
| 305 | F5 | 69AAB007B | INSTRUMENT RACK |
| 311 | F6 | 72TGM001 | TURBINE |
| 314 | F5 | 71FWH001 | HP FEEDWATER HEATER |
| 317 | F7 | 25BGA262 | EXHAUST FAN |
| 318 | G6 | 20TGC003 | CRANE BRIDGE |
| 320 | F4 | 12FMB005A | FIELD MULTIPLEXING CABINET |
| 322 | F4 | 12TCB005A | FIELD MULTIPLEXING TERMINAL CABINETS |
| 323 | F5 | 12TCB005B | |
| 324 | F4 | 91AAB106 | REMOTE DATA ACQUISITION TERMINAL CABINET |
| 325 | F7 | 12BPE016A | MOTOR CONTROL CENTERS |
| 326 | F7 | 12BPE016B | |
| 405 | G4 | 71DVT002 | FLASH TANK |
| 406 | H5 | 71FWH003 | DEAERATOR HEATER |
| 408 | G4 | 69AAB029 | INSTRUMENT RACK |
| 414 | H4 | 25BGA161B | EXHAUST FANS |
| 415 | H6 | 25BGA161C | |
| 417 | H5 | 25BGA111 | RELIEF HOOD |

## GENERAL NOTES

SEE DRAWING BM801

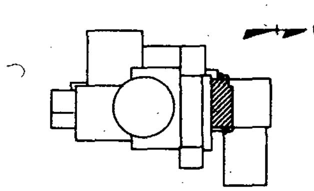
## REFERENCE DRAWINGS

SEE DRAWING BM801

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

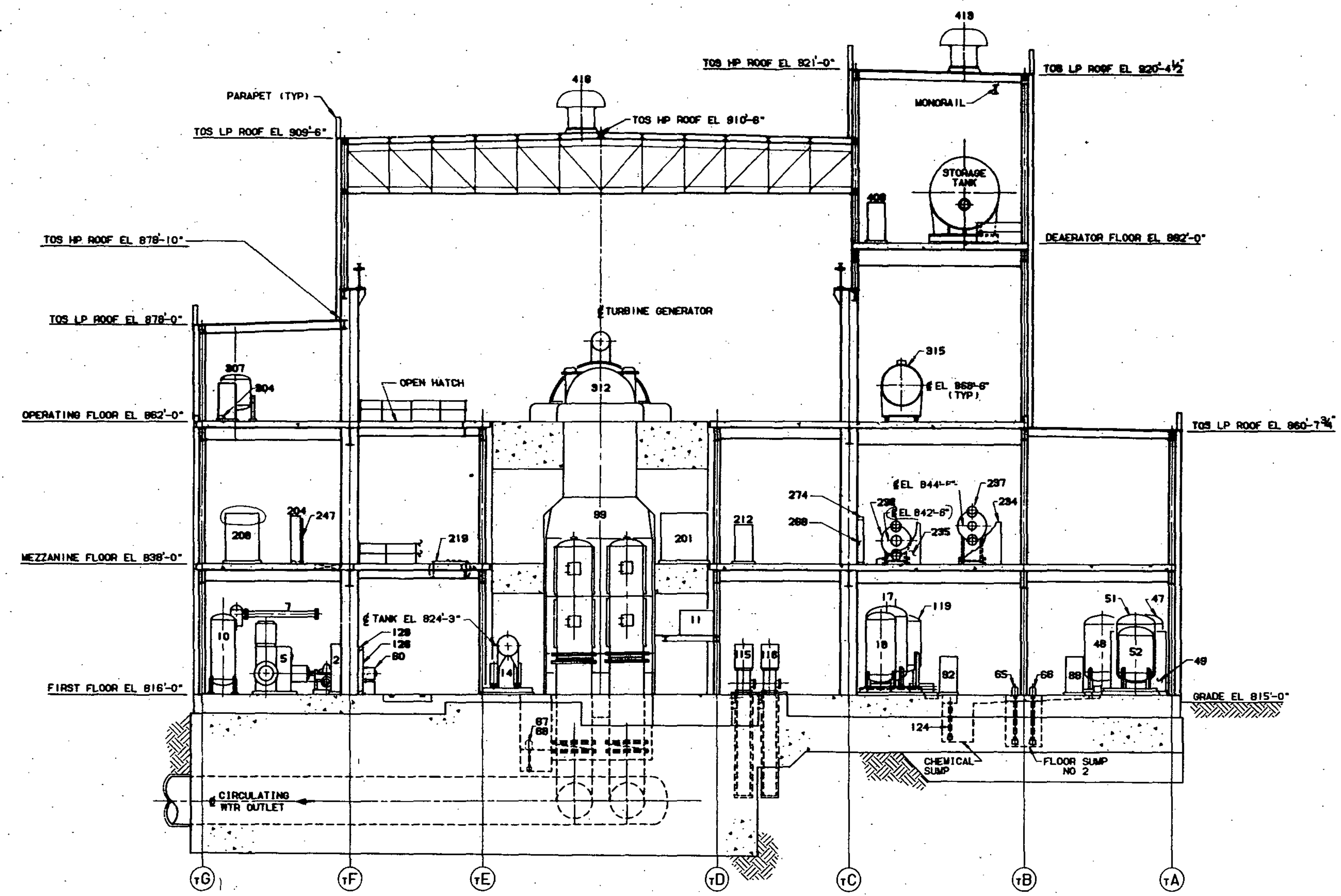
1/8" = 1'-0"




KEY PLAN

Figure 1.2-86
General Arrangement
Turbine Generator Building
Section B-B

DWG. NO. BM606




SECTION "C-C"

## EQUIPMENT LEGEND

| ITEM NO | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
|---|---|---|---|
| 2 | D7 | 221AA001B | SERVICE & INSTRUMENT AIR DRYER |
| 5 | D7 | 22SAA001C | SERVICE & INSTRUMENT AIR COMPRESSOR |
| 7 | D7 | 22SAA002B | SERVICE & INSTRUMENT AIR AFTERCOOLER & SEPARATOR |
| 10 | D7 | 22SAT001B | SERVICE & INSTRUMENT AIR RECEIVER |
| 11 | E6 | 25BGA321 | UNIT COOLER |
| 14 | E6 | 72TAM003 | HYDROGEN SEAL OIL SUPPLY UNIT |
| 17 | E5 | 71CPT002 | CATION TANK |
| 18 | D5 | 71CPT003 | ANION TANK |
| 47 | D4 | 75MUG003A | CLARIFIER FILTERS |
| 48 | D4 | 75MUG003B | |
| 49 | D4 | 75MUG004 | CLEARWELL |
| 51 | D4 | 75MUG005B | ACTIVATED CARBON FILTER |
| 52 | D4 | 75MUG008 | MAKEUP DEMINERALIZER |
| 60 | D7 | 75SWP001B | SEC SVCE CL CLG WTR PUMP |
| 65 | D4 | 76WDP002A | FLOOR DRAIN SUMP PUMPS |
| 66 | D4 | 76WDP002B | |
| 71 | D5 | 76WDP005A | CHEMICAL SUMP PUMP |
| 88 | D4 | 69AAB013A | INSTRUMENT RACKS |
| 92 | D5 | 69AAB014A | |
| 99 | F6 | 73CWH001 | CONDENSER |
| 115 | D5 | 71CDP001B | CONDENSATE PUMPS |
| 116 | D5 | 71CDP001C | |
| 119 | D5 | 71CPT001C | CONDENSATE POLISHER |
| 124 | D4 | 76WDP005D | CHEMICAL SUMP PUMP |
| 134 | D7 | 12FMB001B | FIELD MULTIPLEXING CABINET |
| 138 | D7 | 12TCB001B | FIELD MULTIPLEXING TERMINAL CABINET |
| 201 | E6 | 11AAE009 | TURBINE GENERATOR GND XFMR & RESISTOR |
| 204 | E7 | 12BPE006A | MOTOR CONTROL CENTER |
| 208 | E7 | 12BPE022A | UNIT SUBSTATION |
| 212 | E5 | 69AAB004A | INSTRUMENT RACK |
| 219 | E7 | 71DVT001 | DRAIN TANK |
| 234 | E5 | 12BPE001B | SWITCHGEARS |
| 235 | E5 | 12BPE001D | |
| 237 | E5 | 71CDH004 | LP FEEDWATER HEATER |
| 238 | E5 | 71CDH005 | |
| 247 | E7 | 12ARB001A | AUXILIARY RELAY CABINET |
| 268 | E5 | 12FMB004A | FIELD MULTIPLEXING CABINET |
| 274 | E5 | 12TCB004A | FIELD MULTIPLEXING TERMINAL CABINET |
| 300 | F4 | 12BPE005A | MOTOR CONTROL CENTER |
| 304 | F7 | 25BPB001B | INSTRUMENT RACK |
| 307 | F7 | 75SWT001 | SEC SVCE CL CLG WATER SURGE TANK |
| 312 | F6 | 72TGM002 | GENERATOR |
| 315 | F5 | 71FWH002 | HP FEEDWATER HEATER |
| 413 | H5 | 25BGA161A | EXHAUST FANS |
| 416 | H6 | 25BGA161D | |
| 409 | G5 | 25BPB001A | INSTRUMENT RACK |

## GENERAL NOTES

SEE BM801

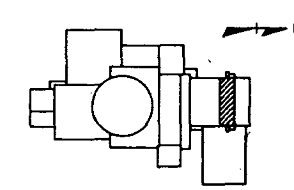
## REFERENCE DRAWINGS

SEE BM801

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

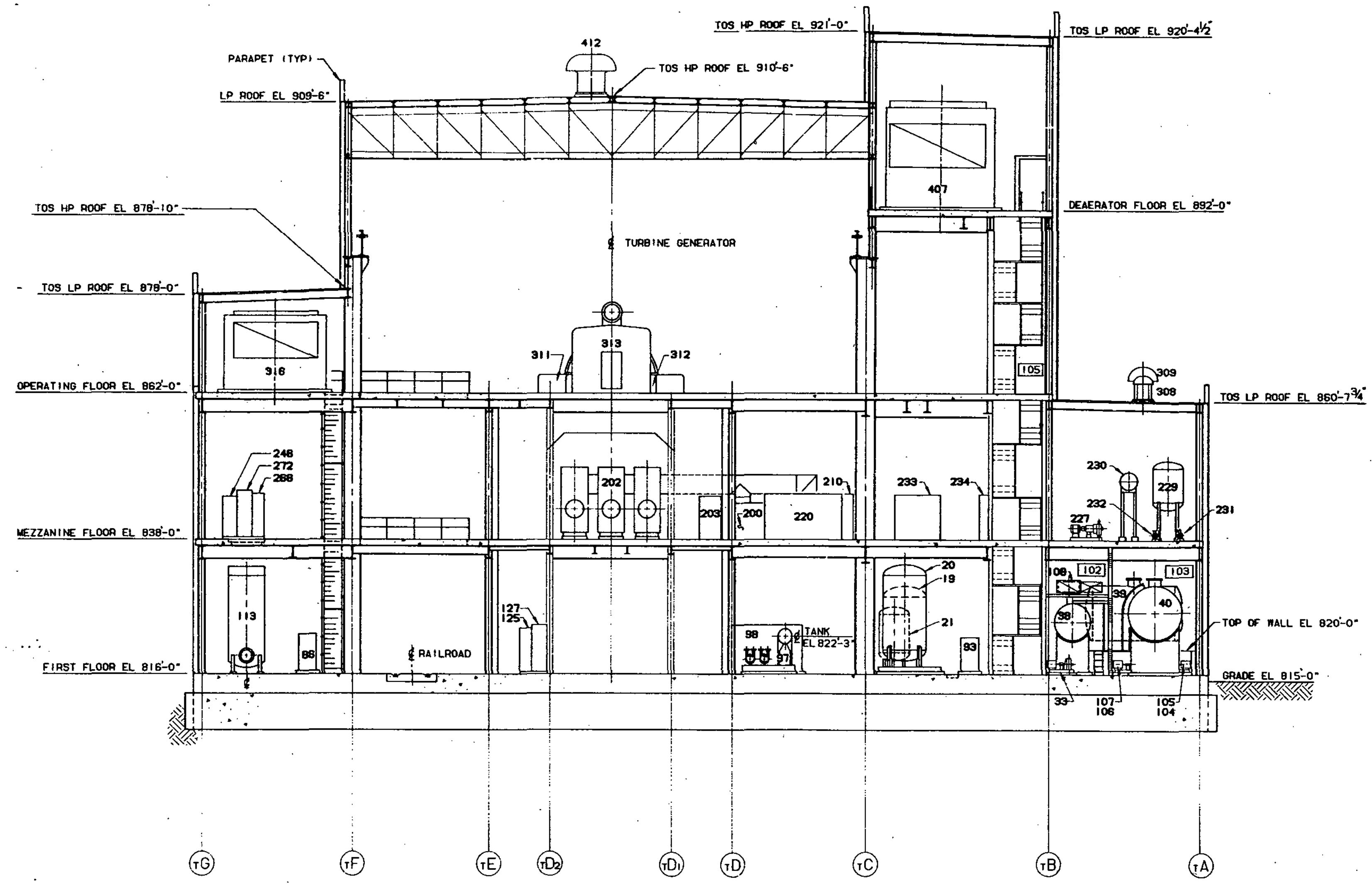
1/8" = 1'-0"




KEY PLAN

Figure 1.2-87
General Arrangement
Turbine Generator Building
Section C-C

DWG. NO. BM607




SECTION "D-D"

## EQUIPMENT LEGEND

| ITEM NO | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
|---|---|---|---|
| 19 | D5 | 71CPT004 | MIXED RESIN STORAGE TANK |
| 20 | D5 | 71CPT005 | AMMONIA RECLAIM TANK |
| 21 | D5 | 71CPT006 | HOT WATER TANK |
| 33 | D4 | 75CFP001B | AMMONIA STORAGE TANK METERING PUMP |
| 38 | D4 | 75CFT001 | AMMONIA STORAGE TANK |
| 39 | D4 | 75CFT002 | ACID STORAGE TANK |
| 40 | D4 | 75CFT003 | CAUSTIC STORAGE TA. |
| 86 | D7 | 69AAB010B | INSTRUMENT RACKS |
| 93 | D5 | 69AAB014B | |
| 97 | D5 | 72TAM001 | STATOR WINDING CLG UNIT |
| 98 | D5 | 69AAB008F | HYDROGEN & STATOR COOLING WATER CABINET |
| 104 | D4 | 71CCP003A | CAUSTIC STORAGE TANK METERING PUMPS |
| 105 | D4 | 71CPP003B | |
| 106 | D4 | 75MUP008A | |
| 107 | D4 | 75MUP008B | |
| 108 | D4 | 25BGA003 | HEAT & VENTILATING UNIT |
| 113 | D7 | 72ASH001B | AUXILIARY BOILER |
| 125 | D6 | 12FMB001B | FIELD MULTIPLEXING CAB |
| 127 | D6 | 12TCB001D | FIELD MULTIPLEXING TERMINAL CAB |
| 200 | E5 | 11AAE008 | PT & SURGE PROTECTION CUBICLE |
| 202 | E6 | 11AAE012 | GENERATOR LOAD BREAK SWITCH |
| 203 | E6 | 11AAE013 | COMPRESSOR LOAD BREAK SWITCH |
| 210 | E5 | 11AAH001 | ISOLATED PHASE BUS HEAT EXCHANGER |
| 220 | E5 | 72TAM006 | EXCITATION CUBICLE |
| 227 | E4 | 75HWP001C | HOT WTR CIRCULATING PUMP |
| 229 | E4 | 75HWT001 | EXPANSION TANK |
| 230 | E4 | 75HWH001 | MAKEUP WATER HEATER |
| 231 | E4 | 75HWP002A | MAKEUP WATER PUMPS |
| 232 | E4 | 75HWP002B | |
| 233 | E5 | 12BPE001A | SWITCHGEAR |
| 234 | E5 | 12BPE001B | |
| 246 | E7 | 91AAB105 | REMOTE DATA ACQUISITION TERMINAL |
| 266 | E7 | 12FMB003A | FIELD MULTIPLEXING CAB |
| 272 | E7 | 12TCB003A | FIELD MULTIPLEXING TERMINAL CAB |
| 308 | F4 | 25BGA162 | EXHAUST FANS |
| 309 | F4 | 25BGA163 | |
| 311 | F6 | 72TGM001 | TURBINE |
| 312 | F6 | 72TGM002 | GENERATOR |
| 313 | F6 | 72TAM005 | ALTERNATOR |
| 316 | F7 | 25BGA002 | AIR HANDLING UNITS |
| 407 | G5 | 25BGA001 | |
| 412 | H6 | 25BGA261C | EXHAUST FAN |

## GENERAL NOTES

SEE BM601

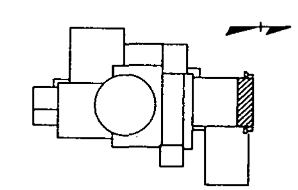
## REFERENCE DRAWINGS

SEE BM601

GRAPHIC SCALE

0 4' 8' 12' 16' 20' 24'

1/8" = 1'-0"




KEY PLAN

Figure 1.2-88
General Arrangement
Turbine Generator Building
Section D-D

Amend. 64
Jan. 1982

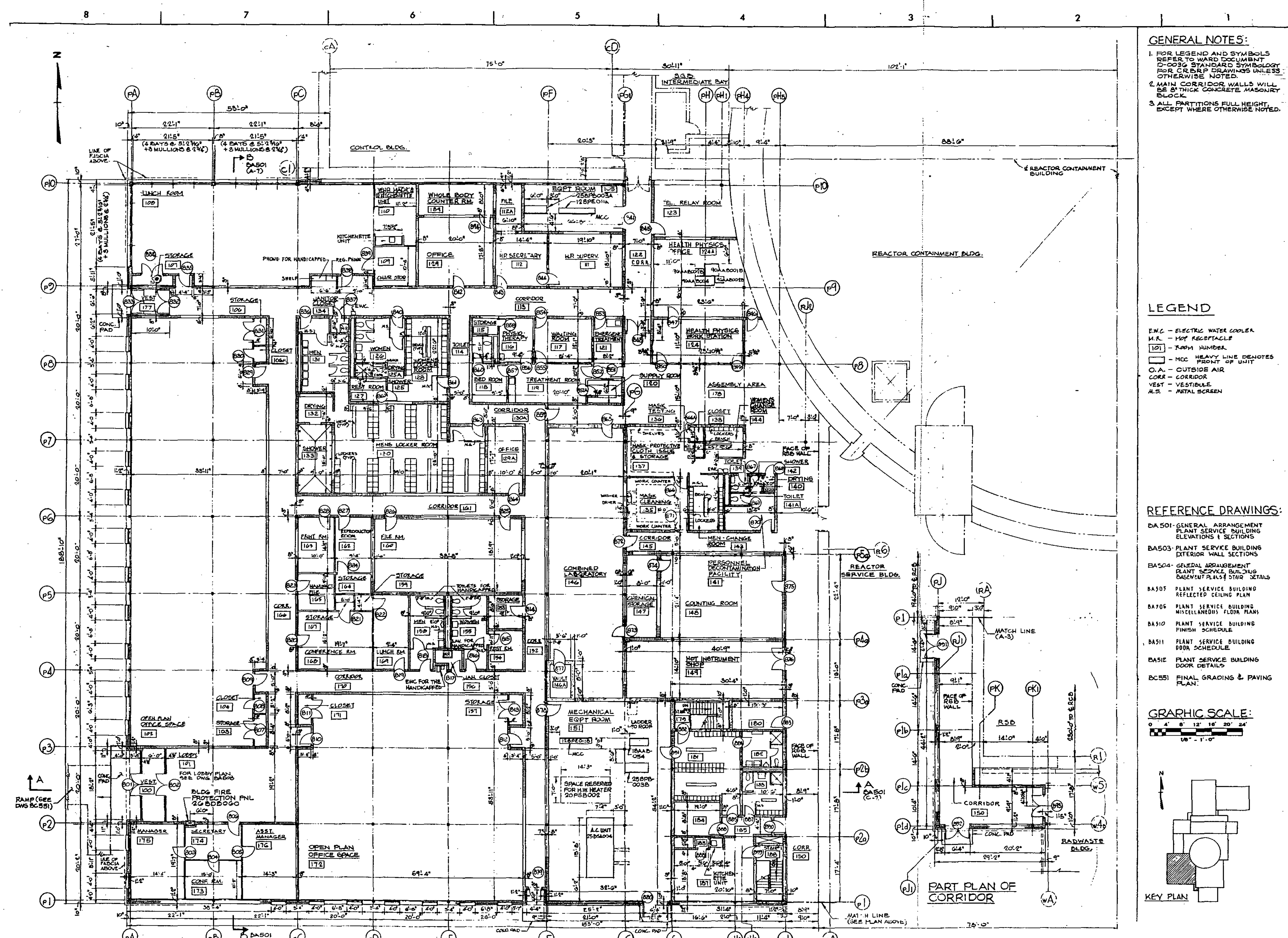



Figure 1.2-89
General Arrangement
Plant Service Building
Floor Plan El. 816'-0"

BA500-7

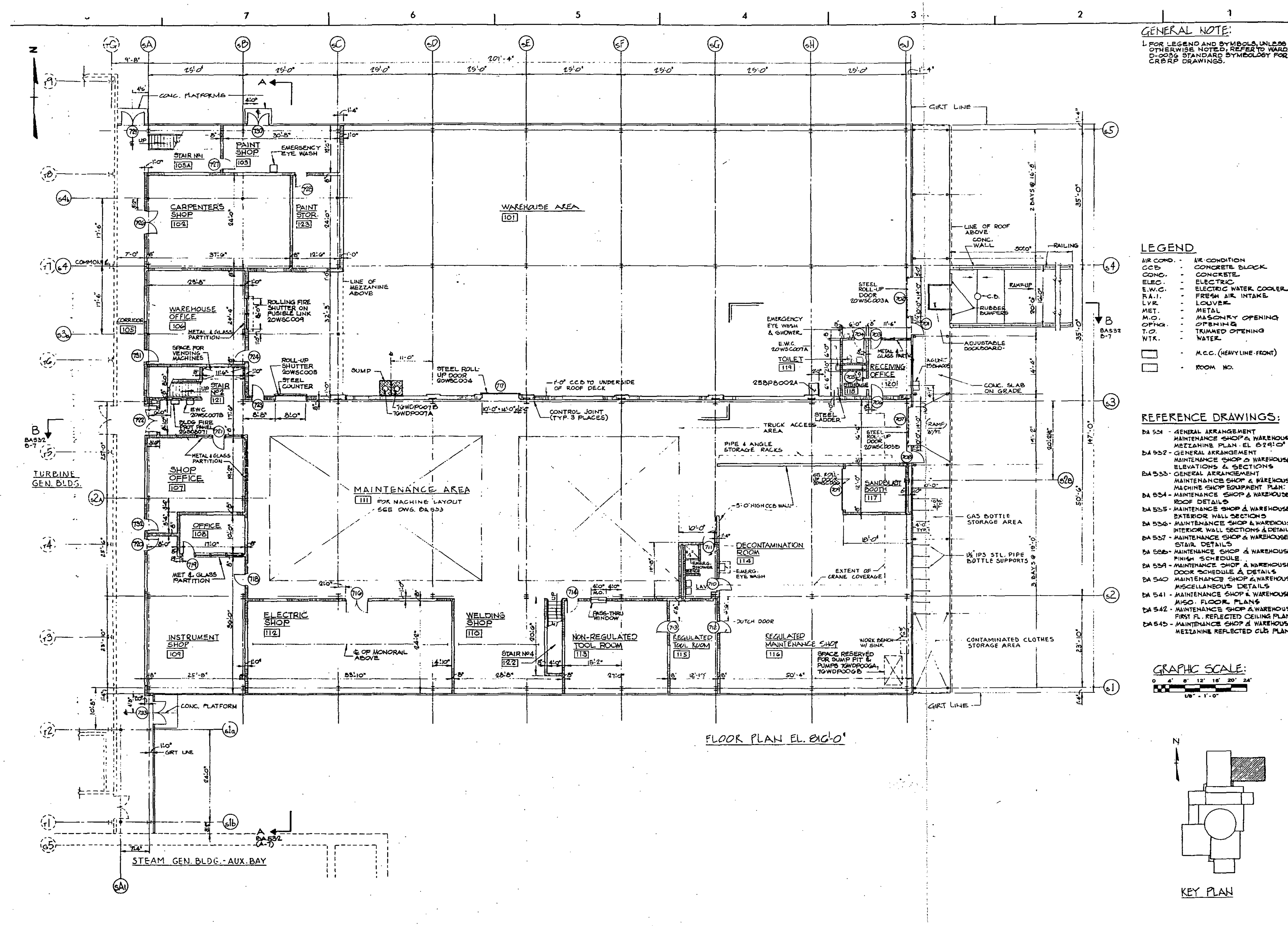



Figure 1.2-90
General Arrangement
Maintenance Shop and Warehouse
First Floor Plan El. 816'-0"





BA530-6

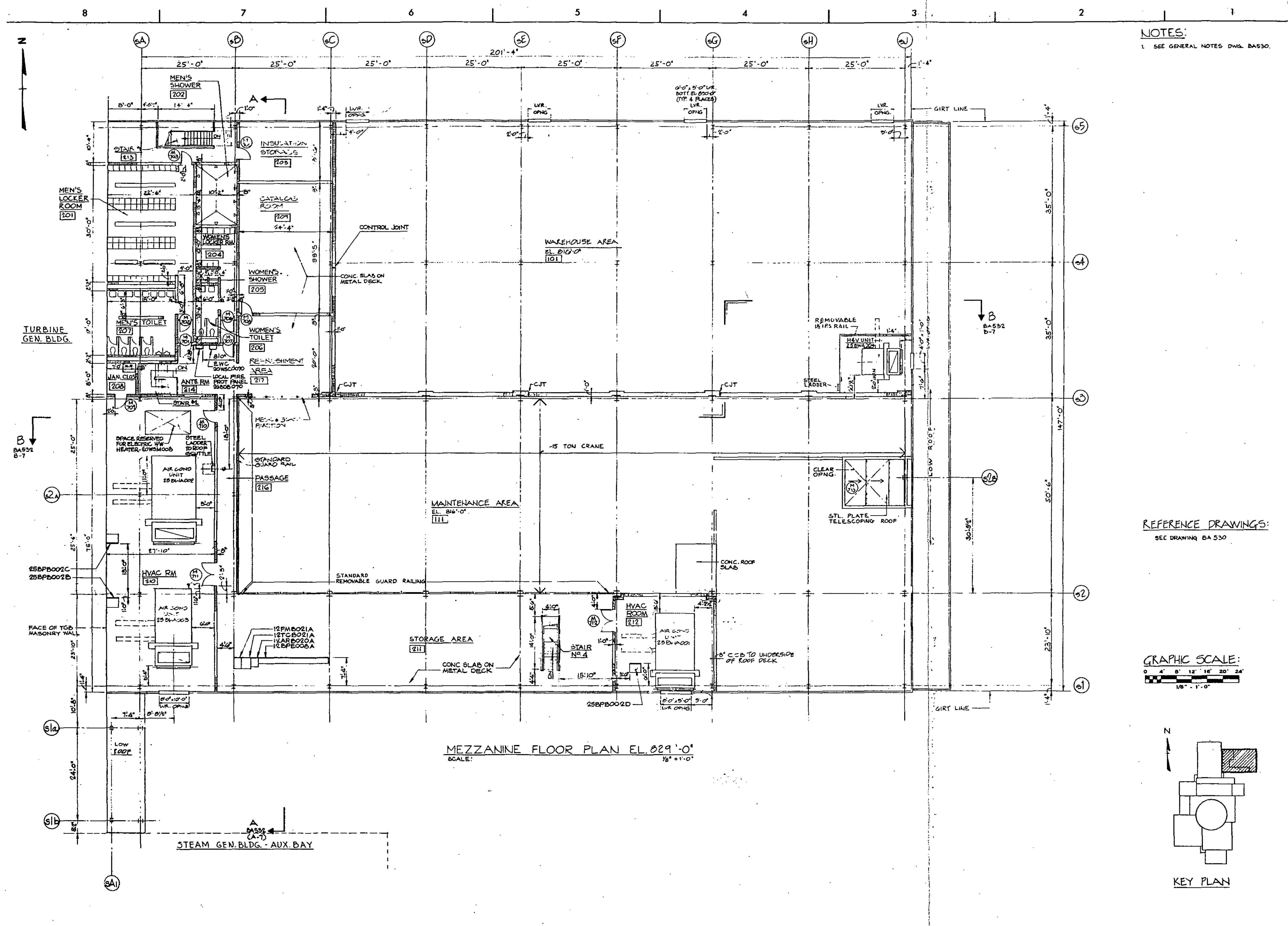
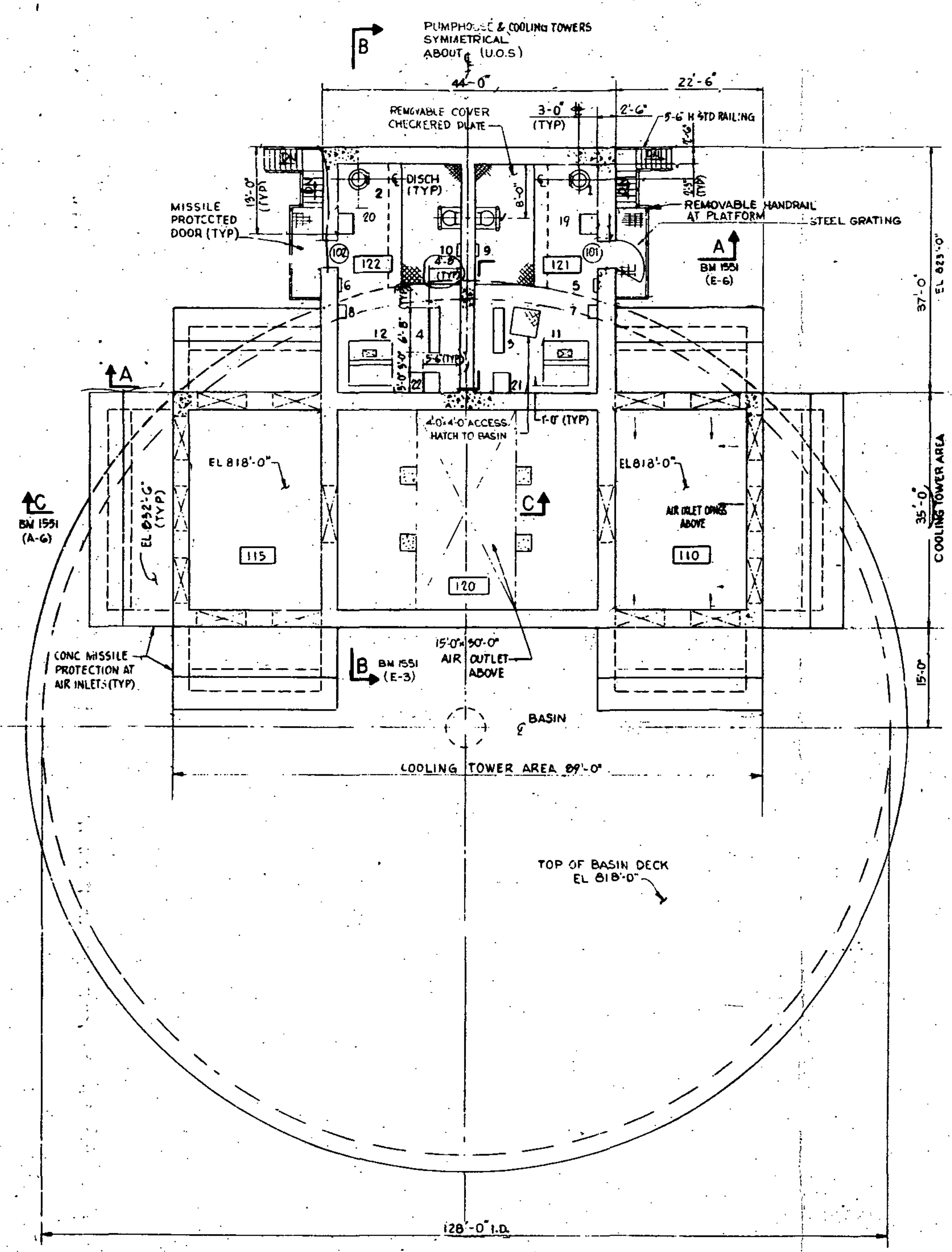
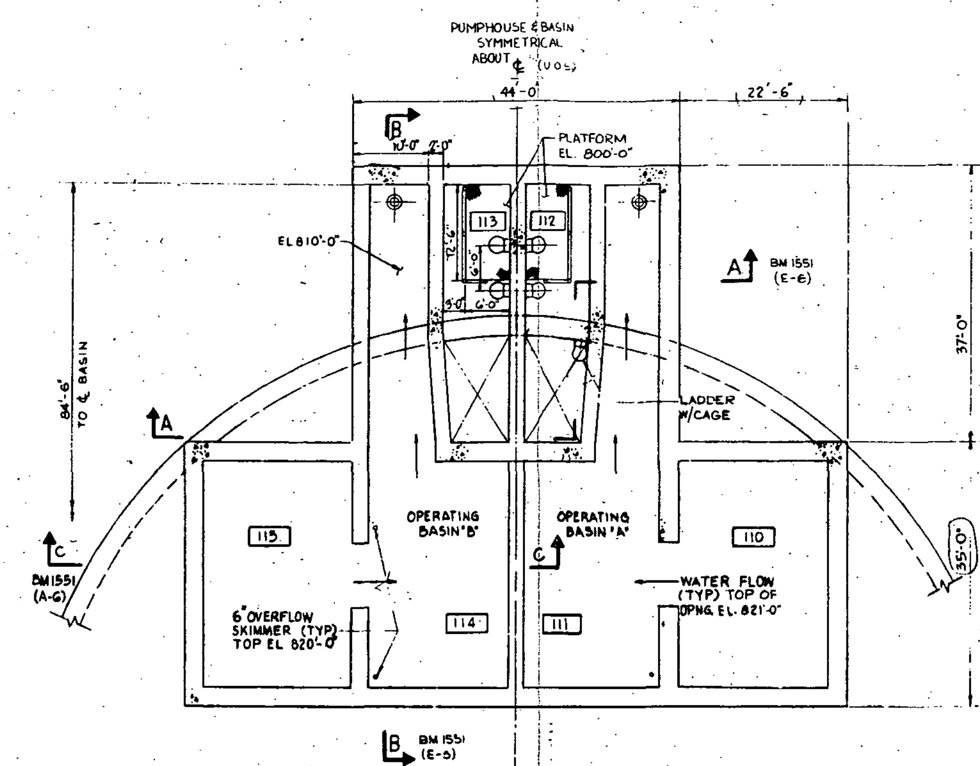



Figure 1.2-91
General Arrangement
Maintenance Shop and Warehouse
Mezzanine Plan El. 829'-0"

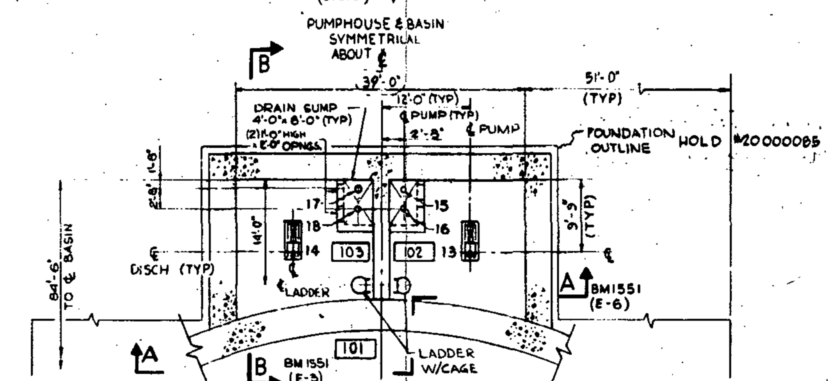
DWG. NO. BM 1550




PLAN EL 823'-0"
U.O.S.




PARTIAL PLAN EL 818'-0"
(U.O.S.)




PARTIAL PLAN EL 771'-0"

| EQUIPMENT LEGEND | | | |
|---|---|---|---|
| ITEM NO. | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
| 1<br>2 | J6<br>J7 | 75EPP001A<br>75EPP001B | EMER PLANT SVCE WTR PUMPS |
| 3<br>4 | H6<br>H7 | 12NIE060A<br>12NIE060B | MOTOR CONTROL CENTERS |
| 5<br>6 | H6<br>H7 | 12NIB039A<br>12NIB039B | LIGHTING PANELS |
| 7<br>8 | H6<br>H7 | 12NIX039A<br>12NIX039B | TRANSFORMERS |
| 9<br>10 | H6<br>H6 | 12NIB040A<br>12NIB040B | DISTRIBUTION PANELS |
| 11<br>12 | H6<br>H7 | 25BEA100A<br>25BEA100B | AIR HANDLING UNITS |
| 13<br>14 | D3<br>D3 | 75EPP003A<br>75EPP003B | EMER PLANT SVCE WTR MKUP PMPS |
| 15<br>16<br>17<br>18 | D3<br>D3<br>D3<br>D3 | 20ECP001A<br>20ECP001B<br>20ECP002A<br>20ECP002B | DRAIN SUMP PUMPS |
| 19<br>20 | J6<br>J7 | 69AAB054<br>69AAB055 | INSTRUMENT RACKS |
| 21<br>22 | H6<br>H7 | 69AAB091A<br>69AAB091B | INSTRUMENT PANELS |

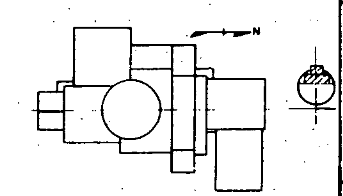
GENERAL NOTES

1. FOR LEGEND, SYMBOLS, & ABBREVIATIONS REFER TO WARD 5-0035
2. HEAVY LINE DENOTES FRONT OF UNIT

REFERENCE DRAWINGS

BM1551 GENERAL ARRANGEMENT EMER. COOLING TOWER PUMPHOUSE AND BASIN SECTIONS

GRAPHIC SCALE



KEY PLAN

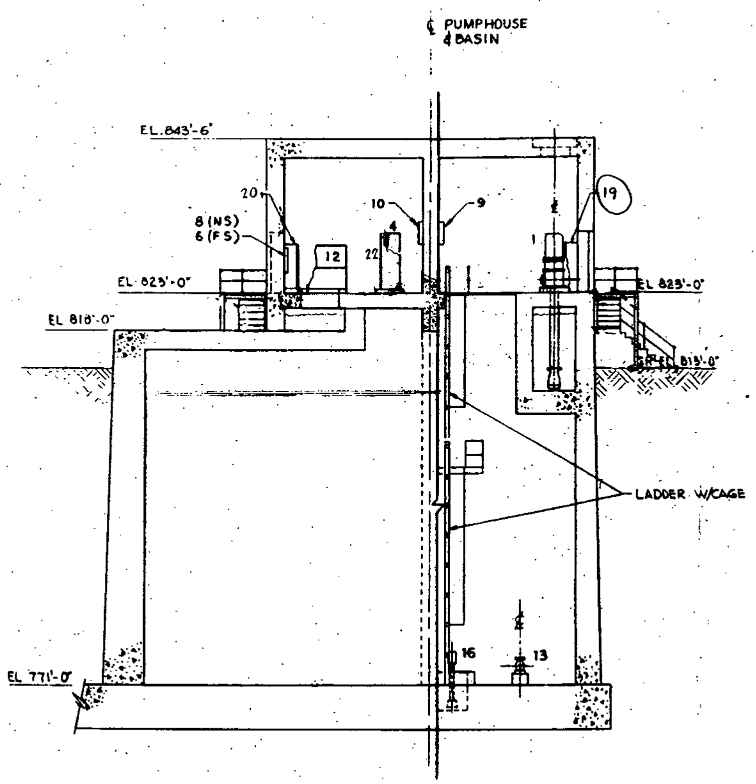
Figure 1.2-92
General Arrangement
Emergency Cooling Tower
Pumphouse and Basin Plan



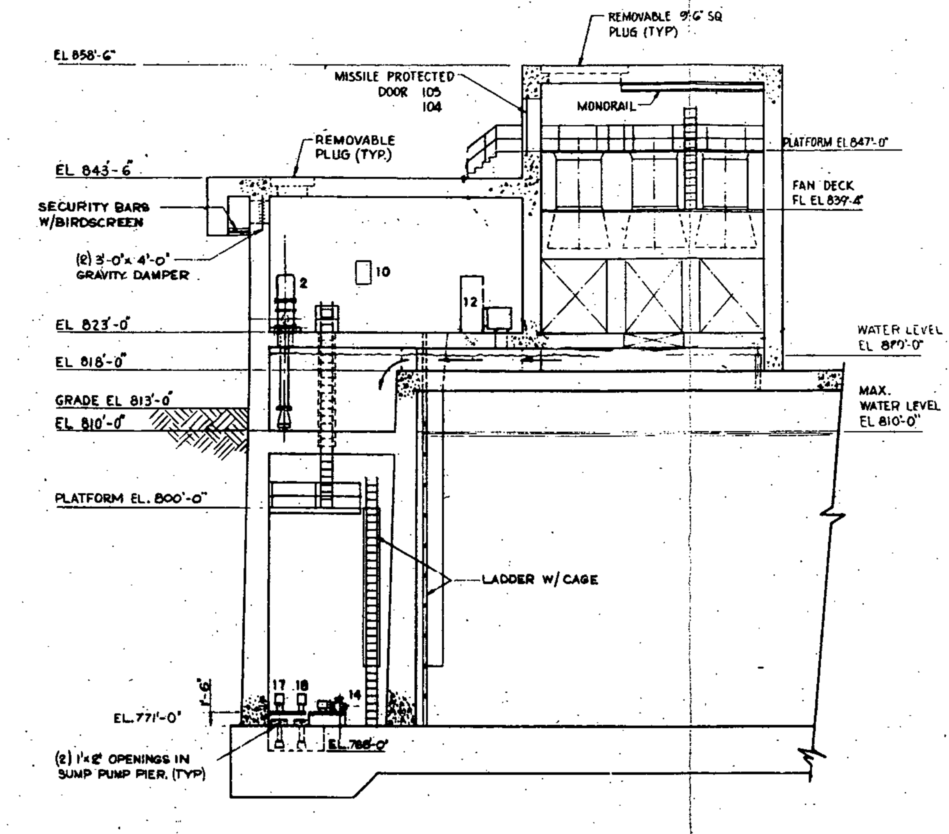
BM1550-3

DWG NO. BM 1551

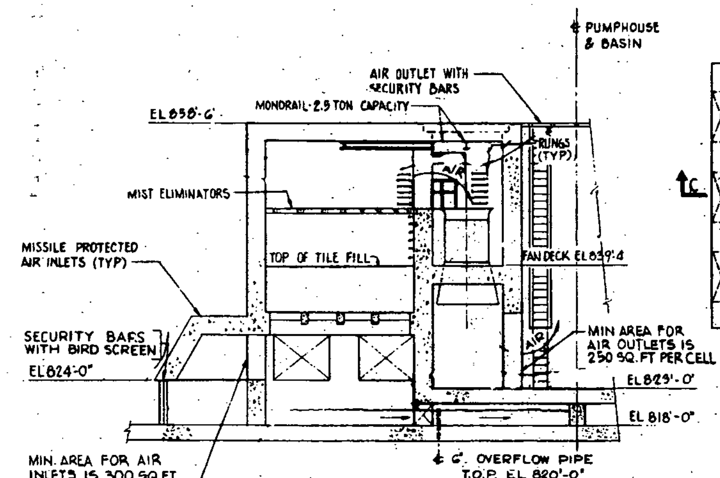
NOTES
FOR GENERAL NOTES AND REFERENCE DRAWINGS SEE DWG BM1550



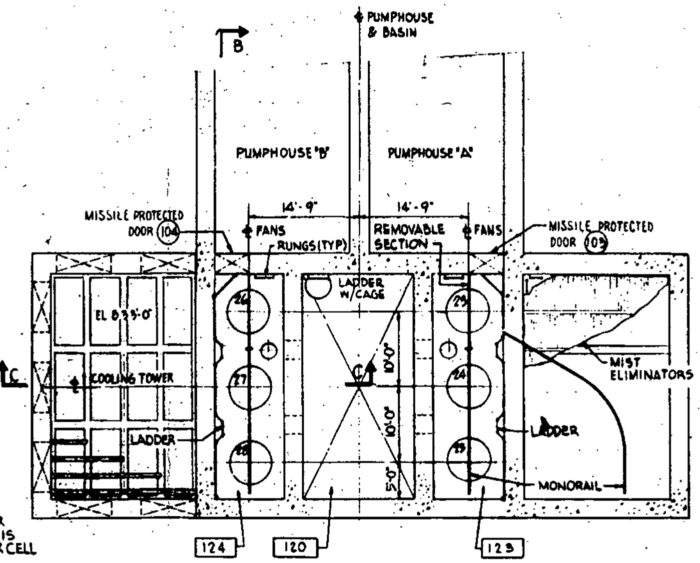

SECTION "A-A"




SECTION "B-B"




SECTION "C-C"




PARTIAL PLAN EL 839'-4"
U.O.S.

| EQUIPMENT LEGEND | | | |
|---|---|---|---|
| ITEM NO. | ZONE | EQUIPMENT NUMBER | DESCRIPTION |
| 1<br>2 | H6<br>H4 | 75EPP001A<br>75EPP001B | EMER PLANT SVCE WTR PUMPS |
| 4 | H6 | 12NIE060B | MOTOR CONTROL CENTER |
| 6 | H6 | 12NIB039B | LIGHTING PANEL |
| 8 | H6 | 12NIX039B | TRANSFORMER |
| 9<br>10 | H6<br>H4 | 12NIB040A<br>12NIB040B | DISTRIBUTION PANELS |
| 12 | H3<br>H6 | 25BEA100B | AIR HANDLING UNIT |
| 13<br>14 | F6<br>F4 | 75EPP003A<br>75EPP003B | EMER PLANT SVCE WTR MKUP PUMP |
| 16<br>17<br>18 | F6<br>F4<br>F4 | 20ECP001B<br>20ECP002A<br>20ECP002B | DRAIN SUMP PUMPS |
| 19<br>20 | J6<br>J7 | 69AAB034<br>69AAB035 | INSTRUMENT RACKS |
| 22 | H6 | 69AAB051B | INSTRUMENT PANEL |
| 23 | D4 | 75EPA002A | FANS |
| 24 | C4 | 75EPA002C | |
| 25 | C4 | 75EPA002E | |
| 26 | D5 | 75EPA002B | |
| 27 | C5 | 75EPA002D | |
| 28 | C5 | 75EPA002F | |

GRAPHIC SCALE

Figure 1.2-93
General Arrangement
Emergency Cooling Tower
Pumphouse and Basin Plan Sections



BM1551-3

## 1.3 COMPARISON TABLES

### 1.3.1 Comparisons with Similar Designs

Table 1.3-1 serves to provide a basis for comparison of selected safety features among large fast reactors of the world.

The following fast reactors have been included because they are comparable in size to the CRBRP.

| | Reactor | Country | Date Critical |
|---|---|---|---|
| 36 | FFTF | U. S. A. | 1979 |
| | SNR 300 | Germany | 1979 |
| | PHENIX | France | August 1973 |
| | PFR | United Kingdom | March 1974 |
| | MONJU | Japan | 1979 |
| | BN 350 | U. S. S. R. | 1972 |

## 1.3.2 Detailed Comparison with Fast Flux Test Facility

This subsection tabulates the principal similarities and differences between the 975 Mw thermal Clinch River Breeder Reactor Plant and the 400 Mw thermal Fast Flux Test Facility.

Table 1.3 -2 compares the principal features of the reactor and associated systems, engineered safety features, and auxiliary systems of each plant.

Table 1.3 -3 compares design data for: (1) reactor core, (2) reactor engineering parameters, (3) reactor physics - initial core, (4) heat transport system, (5) fuel, (6) reactor vessel and enclosure, (7) control rod systems, (8) residual heat removal systems, (9) plant protection system, (10) containment, (11) auxiliary electric power supply, (12) radwaste systems.

Because FFTF is a test reactor and does not generate electricity, there are many fundamental differences between the CRBRP and FFTF which do not lend themselves to comparison. These differences will not be covered by these tables.

## TABLE 1.3-1

## COMPARISON OF THE CRBRP WITH SIMILAR FAST REACTORS

| | CRBRP | FFTF | SNR 300 | PHENIX | PFR | MONJU | BN 350 |
|---|---|---|---|---|---|---|---|
| Thermal Output | 975 MW | 400 Mw | 736 Mw | 563 Mw | 559 Mw | 714 Mw | 1000 Mw |
| Net Electrical Output | 350 Mw | 0 | 282 Mw | 233 Mw | 248 Mw | 248 Mw | 150 Mw + 120,000 tons fresh water per day |
| Pot/Loop | Loop | Loop | Loop | Pot | Pot | Loop | Loop |
| Number Primary Loops | 3 | 3 | 3 | 3 | 3 | 3 | 5 Operational<br>1 Standby |
| Doppler Coefficient | Equilibrium Core - EOL<br>**-0.0087 Tdk/dT, fuel and blankets**<br>**-0.0024 Tdk/dT, Fuel only** | Equilibrium Core - EOL<br>-0.0055 Tdk/dT core | -0.0038 Tdk/dT | -0.0032 Tdk/dT | | -0.0061 Tdk/dT | -.0035 Tdk/dT |
| Containment Concept | Confinement/Containment. A Concrete confinement with an annulus air space surrounding a Steel cylindrical pressure vessel with hemispherical dome and flat bottom. 186 ft. dia. 160 ft. bottom to springline. Steel shell is approx. 1-1/2 in. thick. Concrete shielding below grade. Leak rate of steel containment at design pressure of 10 psig is 0.1% vol. per day. The annulus is maintained at a negative pressure during normal operation and exhausted through filters in the event of an accident. | Single containment. Steel cylindrical pressure vessel with hemi-ellipsoidal top and bottom heads. 135 ft. dia. 187 ft. top to bottom. Steel shell is 1-3/8 in. thick. Concrete shielding below grade. Leak rate at design pressure of 10 psig is 0.1% vol. per day. | Double containment. Inner containment - each radioactive system is contained in a concrete vault with a steel liner on the inside. Vaults are nitrogen inerted. Outer containment - all primary vaults are contained in a rectangular concrete building with steel liner on outside. A closed loop ventilated gap between the two is maintained at -0.15 psig. Allows 0 radioactive release in immediate post-accident time period. Design pressure is 4 psig. | Single containment. Concrete, rectangular, controlled leakage building. Building is 85 ft. by 138 ft. x 115 ft. high. Concrete walls 10 in. thick. Steel reinforced concrete roof. Operated at a pressure of -0.0725 psig. Can withstand pressure of +0.57 psig with a negligible leak rate. | Single containment. Rectangular steel frame building with concrete panel siding, sealed with poly-sulphide. Design leak rate of 50% per day at +0.145 psig. | Double containment configuration. Inner containment All radioactive elements except spent fuel are contained in steel lined nitrogen filled concrete vaults. Outer containment is a steel cylinder with a hemispherical dome on top, 155 ft. dia. Concrete cylinder surrounds entire containment. Design pressure is 4 psig. | No containment building. Reliance on fuel element, reactor vessel and plug. Secondary containment around reactor vessel and piping. Reactor and plug, IHX and pumps are in sealed steel lined, inert gas filled cells. |

18

TABLE 1.3-1 (Cont.)

COMPARISON OF THE CRBRP WITH SIMILAR FAST REACTOR

| | CRBRP | FFTF | SNR 300 | PHENIX | PFR | MONJU | BN 350 |
|---|---|---|---|---|---|---|---|
| Shutdown Heat Removal Scheme | Three systems. Short term, long term and back-up system. Short term rejects heat to atmosphere via direct steam dump from steam drum. Heat rejecting capability approx. 180 Mwt. Long term system condenses steam from steam drum in an air cooled condenser. Long term capacity is about 4.5% of rated power. Back-up system cools sodium from reactor overflow vessel via a Na/NaK heat exchanger. NaK rejects heat to atmosphere via a NaK/Air heat exchanger. | Pony motor provides flow in event of primary/intermediate pump outage. Heat transport system designed to naturally circulate in event of total pump loss. Heat rejected to atmosphere through the Dump Heat Exchanger. | Independent emergency cooling system operates on natural circulation. 6 coolers are submerged between shield tank and reactor vessel. Each cooler has a separate secondary cooling circuit which have EM pumps. Total capacity is 6 Mw. | 3 modes of emergency cooling. Primary pumps can operate on batteries for 5 hours. Natural circulation is the second mode. Third mode is when sodium reaches 700°C cooling circuit outside reactor tank removes residual heat by radiation from the tank. Capacity of third mode is 2 Mw. | 3 independent systems supply 3 NaK/air coolers which extract heat from the IHX. Each circuit is capable of removing 5 Mw. Only 5 Mw must be removed to sufficiently cool reactor. | 3 independent auxiliary systems extract heat from the IHX's. Secondary loops of auxiliary coolant systems have EM Pumps to maintain circulation. | Total decay heat removal capability relies on natural circulation in reactor and steam generator. Design philosophy promotes natural circulation IHX is horizontal and above reactor core. Normal water levels in steam generator constitute one hour of evaporative water. Additional water added later. Calculated heat removal capability is 4-5% of heat in primary circuit. |
| Shutdown and Control Systems | Two independent and diverse systems. Primary system has 9 rods, collapsible roller nut drives, spring assisted gravity insertion. Secondary shutdown system has 6 rods. They have separate sensors, transmitters comparators and logic, ball nut screw drive, hydraulic assisted insertion, below the head release. Primary rods release at roller nut. Either system capable of reactor shutdown with most reactive rod stuck. | Two independent systems. 3 safety rods in inner region. 6 control rods outside of safety rods. All have roller nut drive mechanisms. Either system capable of reactor scram. Rod disconnect at the roller nut. | Two completely independent and diverse systems including sensors and trip points. Primary rods lowered from top by a roller nut mechanism. 3 secondary rods are inserted from bottom by release of magnet which holds down a spring. Absorbers are flexible and can be inserted into slightly deformed channel. | 6 identical control rods are used indiscriminately for operational control, scram capability, shim control, etc. 9 diluents. | 5 Ta control rods in periphery of Zone I. 5 $B_4C$ shutdown rods on periphery of outer core. 1 safety rod in center of core. | 3 regulatory rods (natural $B_4C$) 9 shim rods (slightly enriched $B_4C$) 4 Safety rods (highly enriched $B_4C$ 3 backup shutdown rods (highly enriched $B_4C$). | Normal operational control is accomplished by 6 burn-up rods, 2 regulator rods, and 1 temperature compensator rod. Scram capability is provided by three safety rods. Each of the 3 rods has independent and separate electronic circuits and sensors. Any two rods are capable of reactor shutdown. |

TABLE 1.3-2

COMPARISON OF FFTF AND CRBRP

| 1. Nuclear Steam Supply Systems | CRBRP - 975 Mwt | FFTF - 400 Mwt | CRBRP PSAR Section |
|---|---|---|---|
| Reactor Type | Liquid Metal Fast Breeder Reactor | Liquid Metal Fast Test Reactor | 1.2 |
| Primary Coolant | Sodium | Sodium | 1.2 |
| Intermediate Coolant | Sodium | Sodium | 1.2 |
| Fuel Cycle | U-235/U-238/Pu-239 | U-235/U-238/Pu-239 | 4.2 |
| Reactor Vessel and Head | Stainless Steel Pressure Vessel with Rotating Plug Closure Head | Stainless steel Pressure Vessel | 5.4 |
| Control Systems | Clad Boron Carbide Rods | Clad Boron Carbide Rods | 4.2 |
| Trip/Scram Mechanism | Primary System Gravity drop, spring assist first 14 inches, magnetic release Secondary System gravity drop, hydraulic assist over entire travel, pneumatic release, control rod released at top end of the control assembly. | Both Systems similar gravity drop, spring assist over first 14 inches, magnetic release | 4.2 |

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | CRBRP PSAR Section |
|---|---|---|---|
| Residual Heat Removal Capability | Heat transport is provided by pump pony motor flow in primary and intermediate loops. Natural circulation provides back-up mode of heat transport. Principal mode of heat rejection is a turbine bypass. Alternate short term heat rejection is by direct steam dump from steam drum. Alternate long term heat rejection is by condensing steam in an air cooled condenser. Entirely diverse and independent alternative heat removal capability is provided via cooling of reactor overflow sodium by a Na/NaK heat exchanger. NaK cooled by an NaK/air heat exchanger. | Heat transport is provided by pump pony motor flow in primary and intermediate loops. Natural circulation provides back-up mode of heat transport. Heat rejection to atmosphere is via dump heat exchangers. | 5.6 |
| Sodium Leak Detection | Capability to detect and locate to within a cell or section of piping, small, medium or large liquid metal to gas leaks from any equipment containing liquid metal. | Capability to detect small and large sodium leaks from reactor area and selected piping. | 7.5 |

41

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | CRBRP PSAR Section |
|---|---|---|---|
| Systems Configuration | Primary & Intermediate Loops. IHX, Reactor Vessel, Evap/SPHTR configured so as to promote natural circulation. Elevated loop design prevents sodium from falling below safe levels in reactor due to boundary failure in one loop. Structurally independent free standing guard vessels surround IHX, primary pump and Reactor Vessel to contain sodium in event of large leak. Core remains submerged during all credible accidents. All primary sodium piping is below operating floor. All primary piping contained in nitrogen-inerted steel lined, concrete vaults. | Primary & Intermediate Loops. IHX, Reactor Vessel, DHX, configured so as to promote natural circulation. Elevated loop design prevents sodium from falling below safe levels in reactor due to to boundary failures in one loop. Structurally independent free standing guard vessels surround IHX, Primary Pump, and Reactor Vessel to contain sodium in event of large leak. Core remains submerged during all credible a accidents. All primary sodium piping is below operating floor. All piping contained in nitrogen-inerted steel lined, concrete vaults. | 5.0 |
| Core Instrumentation | Measure temperature at core outlet, temperature in upper plenum, sodium level. | Measure flow temperature at core outlet, temperature upper plenum, sodium level. | 4.4 |

| | | CRBRP - 975 Mwt | FFTF - 400 Mwt | CRBRP PSAR Section |
|---|---|---|---|---|
| 41 | Failed Fuel Detection Capability | Capability to detect and locate failed fuel in the presence of many failures. Each fuel assembly is individually filled with unique isotopic compositions of Xenon and Krypton gas. Mass spectroscopy of cover gas reveals tag gas, indicating leaking fuel assembly. Delayed neutron detectors at reactor vessel outlet dectect fission products in coolant, indicating failed fuel. | Capacity to detect and locate failed fuel in the presence of no other failures. Each fuel assembly is individually filled with unique isotopic compositions of Xenon and Krypton gas. Mass spectroscopy of cover gas reveals tag gas, indicating leaking fuel assembly. Delayed neutron detectors at reactor vessel outlet detect fission products in coolant, indicating failed fuel. | 7.5 |
| 41 41 42 18 | 2. Engineered Safety Features<br>Confinement/Containment | Steel Cylindrically shaped containment with hemispherical or hemiellipsodial top and flat bottom. 186'ID. 160' bottom to springline surrounded by concrete confinement. An annulus space between the containment and confinement maintained at negative pressure during normal operation and exhausted through filters in the event of an accident. | Free standing, structurally independent steel pressure vessel. Cylindrical shape with hemiellipsoidal, top and bottom heads. 135' ID. 186'8" high. Concrete shielding below operating floor. | 3.8 |

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | CRBRP PSAR Section |
|---|---|---|---|
| Containment Isolation System | System provides for complete isolation of all gas and fluid streams into and out of containment in event of leakage of radioactivity into containment atmosphere. System activated by radiation monitors in exhaust stream. (Note - intermediate loops not isolable by CIS.) | System provides for complete isolation of all gas and fluid streams into and out of containments in the event of leakage of radioactivity into containment atmosphere. System activated by radiation monitors in exhaust stream. (Note - Intermediate loops not isolable by CIS.) | 6.2 |
| Radiation Monitoring | Provides sensors on effluent streams for containment isolation system. Support sodium leak detection system. Continuously monitors air for plutonium. | Provides radiation sensors on effluent streams. High trip point trips reactor. Low trip point closes H&V exhaust. | 11.4 |
| 3. Auxiliary Systems | | | |
| Fuel Handling System | Under-the-head refueling by use of triple rotating plug. One in-vessel transfer machine. Access to entire core by movement or triple rotating plug. | Under-the-head refueling 3 in-vessel fuel handling machines access through 3 plugs in head. Each machine services 120° sector. | 9.1 |

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | CRBRP PSAR Section |
|---|---|---|---|
| Irradiated Fuel Storage | Ex-vessel storage tank capacity is 650 assemblies. Storage capacity equal to one full core plus two refuelings (shield assemblies excluded). | In-vessel storage for 57 assemblies. Ex-vessel, Interim Decay Storage capacity is 112 assemblies. (Note that core comprises 74 fuel assemblies and 9 control assemblies.) | 9.1 |
| Auxiliary Power | 3 batteries<br>2 diesel generators (Class 1E) | 2 batteries<br>2 diesel generators | 8.3 |
| Condenser Cooling | Mechanical draft, wet cooling tower. Seismic Category I cooling tower for emergency cooling | Sodium/air dump heat exchanger. | 10.4 |
| Sodium Treatment | Primary and intermediate sodium constantly purified by cold traps. | Primary and intermediate sodium constantly purified by cold traps. | 9.3 |
| Data Handling | Process and co-ordinate data from over 5000 sensor points. Necessary for sodium fill, and refueling operations. No active role in normal power operation. | Process and co-ordinate date from over 1200 sensor points. No active role in plant operation. | 7.8 |

41

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | CRBRP PSAR Section |
|---|---|---|---|
| Fire Protection | Catch pans, oxygen suppression equipment, isolation devices. | Catch pans, oxygen suppression equipment, nitrogen flooding equipment, isolation devices, water prohibited from containment. | 9.13 |
| Emergency Water Service | Provides water to systems essential for plant in safe shut-down condition, in event normal water distribution system is unavailable. Seismic design Category I cooling tower has 30-day supply of evaporative water. | No similar system. | 9.9 |
| Impurity Monitoring and Analysis | Provides for the sampling monitoring and analysis of sodium, NaK and argon cover gas systems in the plant and acceptance sampling and analysis of incoming sodium, NaK, argon, and nitrogen. | Same as CRBRP except that the argon cover gas sampling is an on-line subsystem. | 9.8 |

## TABLE 1.3-3

## DETAILED COMPARISON BETWEEN CRBRP, FFTF, AND MONJU

| | CRBRP 975 Mwt | FFTF 400 Mwt | MONJU* 714 Mwt | PSAR Section |
|---|---|---|---|---|
| 1. Reactor Core | | | | |
| Number Assemblies | | | | |
| Core Zone 1 | 156 | 28 | 108 | 4.3 |
| Core Zone 2 | NA | 45 | 88 | |
| Inner Blanket | 82 | NA | NA | |
| Radial Blanket | 126 | NA | 174 | |
| Removable Radial Shield | 312 | NA | 316 | |
| Primary Control | 9 | 3 | 12 | |
| Secondary Control | 6 | 6 | 7 | |
| Radial Reflector | NA | 108 | - | |
| Core Barrel Inner Diameter (in.) | 150 | 140 | - | |
| Active Core Ht. (in.) | 36 | 36 | 35.4 | |
| Active Core Equiv. Dia. (in.) | 79.50 | 47.23 | 70.08 | |
| 2. Reactor Engineering Parameters | | | | |
| Thermal Power Rating (mw) | 975 | 400 | 714 | 4.3 |
| Gross Electrical Rating (mw) | 380 | NA | 300 | |
| Gross Plant Efficiency (%) | 39 | NA | 42 | |
| Maximum Power (% of Rated Power) | 115 | 115 | 116 | 4.4 |
| Maximum Clad Temp.; Hot Channel, 100% Rated Power, T & H Design Condition, Beginning of Assembly Life, (2 , except where noted) | 1350 | 1180 (600°F Reactor Inlet Temp.)<br>1380 (800°F Reactor Inlet Temp.) | 1209 (nominal) | |

*Notation "-" reflects data not available.
"NA" = not applicable

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*- 714 Mwt | PSAR Section | |
|---|---|---|---|---|---|
| 3. Reactor Physics, Initial Core BOL; except where noted | | | | | |
| Max. Fuel Linear Power at 115% Overpower, (Kw/ft.) | 15.9 | 16.3 | 13.0 | 4.3<br>4.4 | |
| Peak Fuel Linear Power (Kw/ft.) | 12.4 | 12.7 | 11.2 | | |
| Avg. Fuel Linear Power (Kw/ft.) | 8.2 | 7.3<br>(After Cycle 2) | 8.66 | | |
| Peak Neutron Flux - Fuel Zone ($n/cm^2$ sec) | | | | | |
| Total | $5.5 \times 10^{15}$ | $7 \times 10^{15}$ | $4 \times 10^{15}$<br>(average neutron flux for entire reactor) | | |
| Fast (E>.1 Mev) | $3.4 \times 10^{15}$ | | | | 36 |
| Peak Neutron Flux - Radial Blanket ($n/cm^2$ sec) | | | | | |
| Total | $3.9 \times 10^{15}$ | NA | | | |
| Fast (E>.1 Mev) | $2.4 \times 10^{15}$ | NA | | | |
| Pu Inventory (kg) | | | | | |
| Pu-238 | 1.0 | - | - | | 30 |
| Pu-239 | 1468.0 | 552.6 | 1143 | | |
| Pu-240 | 199.7 | 74.7 | 408 | | |
| Pu-241 | 34.0 | 10.7 | 82 | | |
| Pu-242 | 3.4 | 1.12 | - | | 8 |

30

| | CRBRP 975 Mwt | FFTF 400 Mwt | MONJU* 714 Mwt | PSAR Section |
|---|---|---|---|---|
| U Inventory - BOL (kg) | | | | |
| Core U235 | 7.6 | 13.1 | 4160 | |
| U238 | 3476.0 | 1862 | | |
| Blanket U235 | 51.0 | NA | 35 | |
| U238 | 25,150.0 | NA | 17,665 | |
| Peak Discharged Fuel Burn-up (Mwd/Tonne) (First Core) | 74,200 | 80,000 | 80,000 | |
| Control Rod System | | BOEC, Secondary Rods Out of Core: | Regulating Rods | |
| Primary System - BOL Total Available Worth ($) | 23.24 | 18.7 | 13.99 (course rods) 3.60 (fine rods) | |
| Minimum Available Worth with One Rod Stuck ($) | 16.29 | 11.8 | - | |
| Maximum Worth Required for Shutdown ($) | 15.1 | 10.5 | - | |
| Secondary System - BOL | | BOEC, Primary Rods Out of Core: | Safety Rods | |
| Total Available Worth ($) | 13.2 | 21.05 | 8.62 (Safety rods & backup Safety rods) | |
| Minimum Available Worth With One Rod Stuck ($) | 6.75 | Not Available | - | |
| Maximum Worth Required for Shutdown ($) | 6.20 | 20.7 | - | |
| Prompt Neutron Lifetime (Seconds) | $0.4 \times 10^{-6}$ | | $.447 \times 10^{-6}$ | |

| | CRBRP - 975 Mwt | FFTF 400 Mwt | MONJU* 714 Mwt | PSAR Section |
|---|---|---|---|---|
| Doppler Constant - BOL $(-T\frac{dk}{dt})$ (nominal) | | | | |
| Fuel | 0.0026 | 0.0050 | .0061 | |
| Blankets | 0.0059 | NA | (core and blankets combined) | |
| Average Sodium Density Coeff. (¢/°F) | -0.006 | -0.049 | - .0154 | |
| Uniform Axial Expansion Coeff. Coeff. (¢/°F) | -0.038 | -0.038 | -- | |
| Uniform Radial Expansion Coeff. Coeff. (¢/°F) | -0.177 | -0.21 | -- | |
| Core Peaking Factors | | | | |
| Radial - BOL | 1.18 | 1.36 | 1.46 (equil. core) | |
| 51 Axial - BOL | 1.28 | 1.21 | 1.19 (equil. core) | |
| 4. Heat Transport System | | | | 5.0 |
| Primary System Number Loops | 3 | 3 | 3 | |
| Pump Location | Hot Leg | Hot Leg | Cold Leg, | |
| Pump Type | Vertical, single stage, free surface, centri-fugal. | Vertical, single stage, free surface, centri-fugal. | free surface centrifugal | |

8



Amend. 51
Sept. 1979

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*-714 Mwt | PSAR Section |
|---|---|---|---|---|
| Design Head (ft. sodium) | 450 | 500 | 255 | |
| Coolant Flow Direction Through Core | Upward | Upward | Upward | |
| Total FLow Through Reactor (lb./hr.) | $41.45 \times 10^6$ | $17.28 \times 10^6$ | $29.83 \times 10^6$ | |
| Flow Per Loop (gpm) | 33,700 | 14,500 | 23,200 | |
| Reactor Outlet Temp. (°F) | 995 | 858 | 1,005 | |
| Reactor Inlet Temp. (°F) | 730 | 600 | 735 | |
| Nominal Reactor Pressure Drop (Inlet Nozzle to Outlet Nozzle) (psi) | 104.4 | 110 | 65.4 | |
| Intermediate System | | | | |
| Number Loops | 3 | 3 | 3 | |
| Pump Location<br>Pump Type | Cold Leg<br>Vertical, single stage, free surface centri-fugal. | Cold Leg<br>Vertical, single stage, free surface centri-fugal. | Cold Leg, free surface, centrifugal | |
| Design Head (ft. Na.) | 330 | 400 | -- | |
| Coolant Flow Per Loop (gpm) | 29,500 | 14,500 | 17,350 | |
| Hot Leg Temp. (°F) | 922 | 773 | 960 | 8 |

Amend. 41

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*-714 Mwt | PSAR Section |
|---|---|---|---|---|
| Cross Over Leg Temp (°F) | 845 | NA | -- | |
| Cold Leg Temp (°F) | 651 | 515 | 608 | |
| Maximum Cover Gas Pressure in Sodium Expansion Tank (psig) | 115 | 175 | -- | |
| 5. Fuel | | | | 4.2 |
| Composition | $PuO_2/UO_2$ | $PuO_2/UO_2$ | $PuO_2/UO_2$ | |
| Number Fuel Assemblies | 156 | 73 | 196 | |
| Fuel Assembly Ht. (in) | 168 | 144 | 165 | |
| Fuel Assembly Configuration (in) | | | | |
| Active Fuel Height | 36 | 36 | 35.4 | |
| Upper/Lower Axial Blanket | 14/14 | NA | 13.8/13.8 | |
| Fission Gas Plenum | 48 | 42 | 44.5 | |
| Number Fuel Rods per Fuel Assembly | 217 | 217 | 169 | |
| Number Fuel Rods per Blanket Assembly | 61 | NA | 61 | |
| Diameter Fuel Rods in Fuel Assembly | 0.23 | 0.23 | .256 in. | |
| Diameter Fuel Rods in Blanket Assembly | 0.506 | NA | .46 in. | |
| Breeding Ratio (Initial Cycle) | 1.29 | NA | 1.2 | |
| Residence Time of Fuel Assembly Equil. Core (Calendar Years) | 2 | 3 and 4 | 2.5 | |
| Equil. Fuel Residence Time (full power days) | 548 | | | |

8

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*-714 Mwt | PSAR Section |
|---|---|---|---|---|
| Residence Time of Blanket Assembly in Equil. Core (yr.) | 4 or 5 | NA | 2.5 for inner row<br>5.0 for 2 outer rows | 5.1 |
| 6. Reactor Vessel | | | | |
| Type | Stainless Steel Pressure Vessel | Stainless Steel Pressure Vessel | Stainless Steel Pressure Vessel | |
| Dimensions | | | | |
| Height (ft.) | 59 | 43 | 56 | |
| Inner Diameter (ft.) | 20 | 20 | 23 | |
| Material | SS 304 | SS 304 | SS 304 | |
| Thickness (in.) | 2-3/8 | 2-3/8 Vessel Barrel | 1-3/8 | |
| 7. Control Rod Systems | | | | 4.3 |
| Primary System | | | Reactor control is accomplished by 9 coarse regulatory rods and 3 fine regulatory rods | 4.4 |
| No. Assemblies | 9 | 3 | | |
| No. Pins per Assembly | 37 | 61 | | |
| Pin O.D. (in.) | 0.602 | 0.474 | -- | |
| Absorber Ht. (in.) | 36 | 36 | -- | |
| Gas Plenum Ht. (in.) | 28 | 22.36 | -- | |
| Absorber Material (first core) | 92% Enriched $B_4C$ | Natural $B_4C$ | 3 fine rods-natural $B_4C$<br>9 course rods-45% enriched $B_4C$ | |
| Max. Withdrawal Rate (Inches/Min) | 9.0 | 9.0 | -- | |

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*-714 Mwt | PSAR Section |
|---|---|---|---|---|
| Trip Insertion Time (sec.) | <1 | <1 | -- | |
| Drive Mechanism | Collapsible Roller Nut; Disconnect at Top of Drive-line. | Collapsible Roller Nut; Disconnect at Top of Drive-line. | | |
| Operating Mode | Used as Primary Operating Rods. All Rods Drop with Reactor Scram. | All Rods Either Full in Core or Completely Withdrawn. All Rods Drop with Reactor Scram. | Fine rods cover reactivity change from hot zero to rated power; also load follow reactivity changes<br><br>Coarse rods - cover subcriticality and fuel burnup effect. Both course and fine rods are operated as independent banks. | |
| econdary Control Rod System | | | | |
| No. Assemblies | 6 | 6 | There are 4 safety rods and 3 backup safety rods | |
| No. Pins per Assembly | 31 | 61 | | |
| Pin O.D. (in.) | 0.5526 | 0.474 | | |
| Absorber Ht. (in.) | 36 | 36 | | |
| Absorber Material - First Core | 92% Enriched $B_4C$ | Natural $B_4C$ | 90% enriched $B_4C$ | |
| Insertion Mechanism | Gravity, Hydraulic Assist Over Entire Travel. | Gravity, Spring Assist Over First 14 inches for Fully Withdrawn Rods. | | 8 |

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*-714 Mwt | PSAR Section |
|---|---|---|---|---|
| Drive Mechanism | Twin Ball Screw With Translating Carriage. Disconnect at Control Rod Latch Within Control Rod Assembly. | Collapsible Roller Nut; Disconnect at Top of Driveline. | -- | |
| Operating Mode | All Rods Either Full in Core or Completely Withdrawn. All Rods Drop with Reactor Scram. | Used as Primary Operating Rods. All Rods Drop with Reactor Scram. | Used as secondary shutdown system | |
| 8. Shutdown Heat Removal System | Pony motor flow and/or natural circulation in primary and intermediate loops transports heat to steam generator system. Normal method of heat rejection is turbine bypass to the condenser. When this path is unavailable:<br><br>Short term heat rejection accomplished by direct steam dump from | Pony motor flow and/or natural circulation in primary and intermediate loops transports heat to Dump Heat Exchanger (DHX). Heat rejected to atmosphere through the DHX.<br><br>Heat rejection capability depends on temperature differential across primary and intermediate systems. Using reasonable operational temperature differentials. | Primary pump and/or natural circulation transport heat from core to IHX. A separate auxiliary cooling loop removes heat from each IHX. Flow in auxiliary loops is by EM pumps with natural circulation as backup. Final heat rejection is by air coolers. | 5.6 |

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*-714 Mwt | PSAR Section |
|---|---|---|---|---|
| 41 | steam drum to atmosphere. Removes up to ~180 Mwt (18% rated power)<br>• Long term rejection of decay heat accomplished by condensing of steam in an air cooled condenser. Removes up to 4.5% rated power (45 Mwt). When these systems are unavailable, decay/residual heat is removed by cooling of the reactor overflow sodium by a Na/NaK heat exchanger. NaK heat load rejected to atmosphere by a NaK/air heat exchanger. Removes between 10-11 Mwt. | heat rejection capability is ~12% of rated power. (50 Mwt). | | |
| No. Loops | 3 | 3 | 3 | |
| 9. Plant Protection System | | | | 7.2 8 |
| Reactor Trip Action | (1) Release Rods<br>(2) Trip Primary and Intermediate Pumps<br>(3) Provides Turbine Trip Signal | (1) Release Rods<br>(2) Trip Primary and Intermediate Pumps<br>(3) Programs Dump Heat Exchanger Guide Vanes | | |

| | CRBRP 975 Mwt | FFTF 400 Mwt | MONJU* 714 Mwt | PSAR Section |
|---|---|---|---|---|
| Reactor Trip Circuits | | | | |
| No. Circuits Monitored For Trip Actuation | 24-Pri. System<br>16-Sec. System | 23-Pri. System<br>19-Sec. System | -- | |
| Basic Signal and Trip Output Signal Logic | Pri.-2/3 Local Coincidence Logic | Pri.-2/3 Local Coincidence Logic | -- | |
| | Sec.-2/3 General Coincidence Logic | Sec.-1/4 2/3 Hybrid General Local Coincidence Logic | -- | |
| No. External Flux Monitors | 3 | 3 | -- | |
| Max. RSS Logic Response Time (From time RSS senses condition requiring trip to time when rods are released.) (Sec.) | 0.200 | 0.200 | -- | |
| 10. Containment | | | | 3.8 |
| Type/Shape | Single steel vessel, cylindrical shell with flat bottom and hemi-ellipsoidal top. Concrete shielding inside, below operating floor. Steel containment surrounded by concrete confinement building. An annulus space between containment and confinement maintained at negative pressure with respect to outside atmosphere. | Single steel vessel, cylindrical shell with hemi-ellipsoidal top and bottom heads. Concrete shielding below operating floor. | Single steel vessel, cylindrical shell with hemi-spherical top and hemi-ellipsoidal bottom. Concrete cylinder surrounds entire containment. | |

| | | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*-714 Mwt | PSAR Section |
|---|---|---|---|---|---|
| Ht. (ft.) | | 160 from bottom liner to spring line | 186.67 | 262 | |
| Inside Dia. (ft.) | | 186 | 135 | 154 | |
| Internal Design Pressure (PSIG) | | 10 | 10 | 4 | 19 |
| Containment Design Leak Rate @ Design Pressure | | 0.1% Vol. per 24 hours. | 0.1% Vol. per 24 hours. | -- | |
| 18 Confinement/Containment Annulus Design Bypass Leakage | | .001% of containment volume per 24 hours | -- | -- | |
| Design Temp. (°F) | | 250° | 250° | -- | |
| Total Volume (Cu. Ft.) | | $6. \times 10^6$ | $2.265 \times 10^6$ | $4 \times 10^6$ | |
| Tornadic Winds (mph) | | 360 | 175 | -- | |
| Pressure Drop psi | | 3 | 0.75 | -- | |
| psi/sec | | 2 | 0.25 | -- | |
| Missile Protection | Wood | Plank, 4" x 12" x 12' wt.-200 lb., 420 fps | Plank, 2"x12"x12', Wt. 54 lbs. traveling End-on at 100 mph (147 fps) | -- | |
| | Steel | Pipe, Sch40 3" Ø x10' wt.- 78 lb., 210 fps | Plywood sheet, 4'x8'x 3/4", Wt. - 56 lbs., traveling End-on at 150 mph (220 fps) | | 8 |
| | Steel | Pipe, Sch40, 6"Ø x15' wt.-285 lb., 210 fps | | | |
| | Steel | Pipe, Sch40, 12"Ø x15' wt.- 743 lb., 210 fps | | | |
| | Steel | Rod, 1"Ø x 3' wt.- 8 lb., 310 fps | Sheet of corrugated steel siding, 26" x 20'. Wt. 100 lb. traveling End-on at 150 mph (220 fps) | | |
| 36 | | Utility Pole 13.5"Øx35' wt.- 1490 lb., 210 fps | | | 19 |

| | | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*-714 Mwt | PSAR Section |
|---|---|---|---|---|---|
| | | Automobile, 20ft$^2$ (front) wt.- 4000 lb., 100fps | | | 19 |
| | Safe Shutdown Earthquake | 0.25g maximum ground acceleration. | 0.25g maximum ground acceleration. | -- | |
| 36 | Operating Basis Earthquake | 0.125g maximum ground acceleration. | 0.125g maximum ground acceleration. | -- | |
| | 11. Auxiliary Electric Power | | | | |
| | DC Power | | | | |
| | No. Batteries | 2 | 2 | -- | |
| | Voltage (Volts) | 125 | 125 | | |
| 41 | Rating (Amp-Hrs.) | 1350 each | 830 each (8 hr. rating) | | |
| | No. Batteries | 1 | NA | -- | |
| | Voltage (Volts) | 250 | NA | | |
| 42 41 | Rating (Amp. - Hrs.) | 2300 | NA | | |
| | AC Power | | | | |
| | Diesel Generators | | | | |
| | No. | 2 | 2 | -- | |
| | Total Power (kw) | 7600 | 2200 | | |
| 41 | Voltage (Kv) | 4.16 | 0.480 | | 8 |

Amend. 42
Nov. 1977

| | CRBRP - 975 Mwt | FFTF - 400 Mwt | MONJU*-714 Mwt | PSAR Section |
|---|---|---|---|---|
| Time to come on-line (sec.) | 10 | 10 | -- | |
| Class 1-E Standard | Yes | No | -- | |
| 12. Radwaste Systems | | | | |
| Type | Liquid Radwaste System processes low and intermediate level liquid radwaste by evaporation and demineralization, producing purified or decontaminated water for reuse or discharge. Resultant concentrated liquid radwaste is sent to Solid Radwaste system for solidification. SRS processes and packages solid radwaste for shipment to AEC burial sites. The Radioactive Argon Processing System and the Cell Atmospheric Processing System purify contaminated gases and ensure that all gaseous releases are as low as practicable. | All solid and liquid radioactive waste is collected and stored on site until such time as it is shipped to the 200 Area of the Hanford Reservation for permanent storage. There is little processing of liquid waste. | -- | 11.2, 11.3, and 11.5 |

8

TABLE 1.3-2 (Continued)

| | CRBRP - 975 Mwt | FFTF -400 Mwt | MONJU* -714 Mwt | PSAR Section |
|---|---|---|---|---|
| Minimum Exclusion Radius | 2200 ft. | ~4.5 miles | -- | |
| Average Tritium Production | 89 | 40 | -- | |
| Design Basis Quantity of Low and Intermediate Level Waste (gal/yr) | 360,000 | 43,900 | -- | |
| Estimated Design Dose from Liquid Effleunts Discharged to River (mrem/yr) | 0.2 mrem/yr @ 1% Failed Fuel. Calculated for People Living in Close Proximity to Site Boundary | | | |
| Solid Radwaste System Design Quantities (lbs/yr) | | | | |
| Non-Sodium Waste | 242,000 | 14,700 | | |
| Sodium & Sodium Containing Waste | 16,500 | ~50,000 | -- | |
| Radioactive Gaseous Releases at Expected Service Condition of 0.1% Failed Fuel (ci/yr) | | | | |
| Noble gases | 6.4 | 1.2 | -- | |
| Tritium | 3.2 | 16 | -- | |
| Estimated Maximum Total Annual Dose from Gaseous | 0.9 mrem.yr @ 1% Failed Fuel. Calculated for People Living in Close Proximity to Site Boundary | $10^{-3}$ mrem/yr @ 0.1% Failed Fuel. Calculated for Closet Occupied Area - 2 Miles from Reactor. | | |

8

## 1.4 IDENTIFICATION OF PROJECT PARTICIPANTS

BACKGROUND

In July, 1969, Statutory authority was provided by the Congress for the Atomic Energy Commission (AEC) to embark on a two-phase approach for the first LMFBR Demonstration Plant. The first phase, the Project Definition Phase (PDP) work, permitted all participants to better understand and define the technical and economic characteristics of the proposed undertakings. Atomics International, General Electric and Westinghouse and associated utilities participated in this phase of the program. The second phase, the Definitive Cooperative Arrangement phase (DCA), provided for the design, construction and operation of the first LMFBR Demonstration Plant.

Early in 1971, it was determined that, due to the magnitude of the undertaking, the LMFBR Demonstration Plant must have the full support and backing of essentially the entire electric utility industry including the investor-owned and publicly-owned sectors of this very large industry. To seek advice and assistance in obtaining such general support from the utility industry, in April 1971 the AEC established two committees, the Senior Utility Steering Committee and the Senior Utility Technical Advisory Panel. These committees consisted of top senior management and engineering executives from the utility industry.

This coordinated AEC-Utility effort resulted in proposals being made by a group of major utilities from New England, the Pennsylvania-New Jersey- 25
Maryland Interconnection, Empire State Atomic Development Associates, the Southern Company-Middle South Utilities, and Commonwealth Edison-Tennessee Valley Authority for the design, construction, and operation of an LMFBR Demonstration Plant.

The two senior utility advisory committees recommended and the AEC selected the joint submission from Commonwealth Edison (CE) and the Tennessee Valley Authority (TVA) as the basis for negotiations leading to a contract for the design, construction and operation of the first demonstration plant. This action resulted in a shift of program emphasis from the reactor manufacturers 25
to the utility industry.

In March, 1972, based on the CE-TVA proposal, two new corporations - Project Management Corporation (PMC) and the Breeder Reactor Corporation (BRC) - were created. PMC was organized to manage the design, construction, and operation of the demonstration plant and BRC was created to serve as interface between the electric utility industry and the LMFBR Demonstration Plant Project, to provide senior counsel, and to coordinate the electric utility industry's assignment of people and financial contribution to the Project.

The CE-TVA proposal provided for a seven-man Board of Directors comprised of two members each from CE, TVA, and AEC and one member from BRC. However, the Atomic Energy Act of 1954 did not permit AEC employees to serve on the Board of a private corporation. As a result, an interim arrangement with a five-member board was established until legislation could be introduced and passed by Congress to permit AEC participation on the Board. In order to assure the AEC's full participation in the Project, a three-man Project Steering 25

Committee (PSC) with one representative each from TVA, CE and AEC was
established. The Project Steering Committee's function is to implement the 25
general policies established by the PMC Board of Directors.

The arrangements for carrying out the LMFBR Demonstration Plant Project were
formalized in Contract AT(49-18)-12. The parties to the contract were the
Atomic Energy Commission (AEC), the Tennessee Valley Authority (TVA), the
Commonwealth Edison Company (CE), and the Project Management Corporation
45 (PMC). The United States Department of Energy (DOE) has since succeeded to the
role of the AEC in carrying out the Federal Government's responsibilities in
connection with the Project. Contract AT(49-18)-12 identifies the roles of
various participants in the design, construction and operation of the liquid
metal fast breeder reactor demonstration plant.

In June, 1974, the Reference Design for the Clinch River Breeder Reactor Plant
was established. Based on this design, a detailed cost projection for the
Project was prepared. This cost projection was significantly greater than the
estimated cost of the Project that was made in 1972 before the design had been
definitized. 25

Contract AT(49-18)-12 provides for the Federal Government to fund any Project
cost in excess of contributions of the utility industry. As a result of the
significant increase in the cost estimate for the Project, it was agreed that
the Federal Government's authority in the Project was not in proper proportion
to its financial participation. Modifications to the Project arrangements to
increase such authority were agreed to in principle in March, 1975. Changes
in the management structure were planned. A single integrated CRBRP Project
Office, comprised of both government and utility industry personnel has since
been established to manage the Project. 25

### 1.4.1 FUNCTIONS, RESPONSIBILITIES, AND AUTHORITIES OF PROJECT PARTICIPANTS

The general Project management authority and responsibility is now vested in
the Department of Energy (DOE). This authority and responsibility is carried
out on a day-to-day basis by the Clinch River Breeder Reactor Plant (CRBRP)
Project Office. Thus, DOE is responsible for all activities of the Project in
the accomplishment of the design, licensing, construction, testing and
operation of the CRBRP. DOE will provide financial support for the CRBRP
Project as well as support from its LMFBR Base Technology programs. With
respect to the supporting R & D work, DOE will provide information to PMC, TVA
45 and CE and notice of events having a significant potential impact on Project
cost and schedule. DOE will also provide all source and special nuclear
material required for the CRBRP during the term of the Project in the form of
completed fuel assemblies.

DOE, PMC and TVA are co-applicants for the CRBRP Construction Permit and will
receive such support as they may require from the Project contractors in 25
meeting such responsibilities.

PMC is responsible for administering the utilities' interests in the Project and providing utility personnel and financial support for the Project. The PSC will function as a tripartite review mechanism for the overall Project and is responsible for keeping the PMC Board fully and currently informed about Project activities.

TVA and CE will supply key personnel and staff to PMC to meet its obligations for staffing the integrated CRBRP Project Office. In addition, TVA will provide the site and certain transmission facilities and purchase the electrical energy produced.

45 Through separate contractual arrangements between TVA and DOE, TVA will operate and maintain the plant and provide supervision of plant safety, operation and maintenance.

### 1.4.2 DESCRIPTION OF ORGANIZATIONS

45 DOE is a Cabinet level department of the Federal Government with responsibility for policy planning, coordination, support and management of Government research and development programs for all energy sources. The Department is headed by a Secretary who is appointed by the President and
45 confirmed by the Senate.

Project Management Corporation is a non-profit corporation organized to participate in the development, testing and demonstration of generating electric power using the LMFBR concept. The general policies of the Corporation are established by its five-member Board of Directors.

TVA is a corporate agency of the Federal Government with responsibility for planning for the proper use, conservation and development of the natural resources of the Tennessee River drainage basin and its adjoining territory. TVA's major policies, programs and organizations are determined by a full-time, three-member Board of Directors who are appointed by the President and confirmed by the Senate for nine-year terms. TVA operates with a reasonable degree of the autonomy and flexibility which characterizes a private corporation. It is an independent agency, not part of any Federal cabinet department.

Commonwealth Edison is a private corporation primarily concerned with the production of electric power and is especially interested in advancements in the field of nuclear power production. CE's major policies and programs are established by a Board of Directors who are elected by the stockholders of the Corporation.

#### 1.4.2.1 INTEGRATED PROJECT ORGANIZATION

The four parties to the Project Contract AT(49-18)-12 have agreed (Modification No. 1) to the formation of a single, integrated project management organization, staffed by both government and industry personnel.

This management structure has been designated the Clinch River Breeder Reactor Plant (CRBRP) Project Office. The CRBRP Project Director, a DOE official serving under the supervision of the Manager, Oak Ridge Operations Office of DOE will direct the entire CRBRP Project Office staff and manage the Project.

The CRBRP Project Office is expected to consist of about 34 DOE and 120 PMC professional and managerial employees by October, 1982. Most of the PMC professional and managerial staff will be personnel assigned by CE and TVA. In addition, BRC member utilities and other companies may assign personnel to the CRBRP Project Office. Some professional employees will be hired directly by PMC.

The Integrated Project Office includes the Project Director and his staff, fourteen division chiefs and their staffs and a site representative who reports to the Director. The Director's staff consists of a Deputy Director, an Assistant Director and an Executive Assistant. The division chiefs reporting to the Director include Administrative Services, Audit, Automatic Data Processing, Construction, Counsel, Engineering, Financial Management, Information, Operations, Project Control, Public Safety and Quality Assurance. The CRBRP Project Office organization chart is included as Figure 1.4-1.

The Project Office functions and responsibilities are to plan and conduct programs and activities for the design, development, manufacture, licensing, construction and operation of the CRBRP through the demonstration period and to identify and arrange for services for engineering, research, development and testing of systems and components to support successful project completion. The specific functions within the Project Office are:

Project Director's Office

The Project Director directs all activities of the CRBRP Project to accomplish the design, manufacturing, licensing, construction, testing and operation on a utility network of a liquid metal cooled fast breeder reactor demonstration plant. He performs delegated contracting office functions.

The Deputy Director assists the Director in directing, supervising and managing the Project. He performs delegated contracting officer functions. In the absence of the Director, he acts in his stead.

The Assistant Director, as the General Manager of PMC, represents the interest of the utility industry in the Project. He participates actively and closely in reviews of engineering, cost and schedule, planning and execution so as to provide the Director the benefit of the utilities' interest in Project activities. He leads the public information program. He represents the utilities' interest in the formulation of a licensable design that will be commercially viable. He serves as leader of task teams for problem resolution as assigned.

The Executive Assistant acts in a staff capacity in the formulation and executive direction of Project activities, with primary concern for the administrative and managerial aspects of the organization.

## Construction

The Construction Division is responsible for management of the construction of the Project, including the quality, timeliness and cost of the constructed work.

## Engineering

The Engineering Division is responsible for management of the design, engineering and fabrication of systems, processes, equipment and facilities, including quality, cost estimates, schedule and research and development activities.

## Procurement

The Procurement Division is responsible for planning, developing, coordinating and executing policies for contractor selection, contract negotiations, administration of contracts, review and approval of subcontracts and procurement management appraisal. It may also directly procure goods and services.

## Public Safety

The Public Safety Division is responsible for planning, developing, coordinating and executing policies and plans in the areas of public safety, environmental affairs, nuclear safeguards, licensing and reliability. The division is also responsible for management of environmental monitoring activities, including quality, cost, and schedule.

## Operations

The Operations Division is responsible for overseeing TVA's operation of the plant. Before the operating phase, it is responsible for development and implementation of an integrated plant operating program and the orderly transition from the design and construction phases to the operating phase.

## Quality Assurance

The Quality Assurance Division is responsible for planning, developing and assuring effective execution of the integrated quality assurance program including the conduct of the owner program and the integration and coordination of all the quality assurance programs of the Project participants.

## Information

The Information Division is responsible for planning, developing and administering the Project activities in community relations and public education. It coordinates the dissemination of technical information to the utility industry and the general public, and coordinates information activities by all Project participants and major industry organizations.

Project Control

The Project Control Division is responsible for designing, developing and implementing the management control systems for the Project. It monitors the integrated costs, schedule and technical performance of the contractors.

Financial Management

The Financial Management Division is responsible for developing and coordinating policies, programs, and procedures for budgeting and accounting, to ensure financial control.

Audit

The Audit Division is responsible for developing and coordinating policies, programs, and procedures to ensure the conduct of professional audits.

Counsel

The division of Counsel is responsible for providing legal advice and assistance on all matters of law and legal policy for the Project.

Administrative Services

The Administrative Services Division is responsible for planning, developing and implementing support services for the Project Office and support services rendered commonly to other Project participants at Oak Ridge.

Automatic Data Processing

The Automatic Data Processing (ADP) Division is responsible for providing guidance, advice and assistance to the Project Office in technical and business management applications of ADP.

Labor Relations

The Labor Relations Division is responsible for providing guidance, advice and assistance to the Project Office on industrial relations with contractors.

1.4.2.2 PMC ORGANIZATION

The PMC organization is headed by the General Manager and consists of professional employees who perform PMC's contract obligations.

The PMC staff is responsible for administering the utilities' interests in the Project including continuous monitoring of the Project, preparation and dissemination of Project information, arranging for the participation of utility personnel in the Project, investment and disbursement of utility funds and exercising the various contractual rights designed to protect the utilities' interests, including approving any proposed changes in Project scope or deviation from the approved Reference Design or specifications, maintaining access to information and data, either in the possession of the Government or any of the Project contractors, seeing that the conditions for the disbursement of utility funds are met, and exercising the rights of termination of the Project in the event a contractually based termination occasion arises.

### 1.4.2.3 DOE ORGANIZATION

The overall DOE organization is shown in Figure 1.4-2. Prime responsibility for the CRBRP Project is assigned to the Director, CRBRP Project. The line of authority is from the Secretary of Energy to the Under Secretary of Energy to the Manager, Oak Ridge Operations and then to the Project Director as shown on Figure 1.4-2A.

The Assistant Secretary for Nuclear Energy provides program management direction to the Project through the Manager, Oak Ridge Operations. Licensing is the responsibility of the Assistant Secretary for Nuclear Energy under the office of Nuclear Reactor Programs, Plant Development Division. This latter office also provides licensing direction to the CRBRP Project.

The Deputy Assistant for Nuclear Reactor Programs, in consultation with the Project Director, manages the Base Technology program which contributes support to the CRBRP.

### 1.4.2.4 TVA ORGANIZATION

The organization of TVA is shown in Figure 1.4-4. The responsibility for TVA's activities will be met by or through the Office of Power, shown in Figure 1.4-3. The staff and divisions that will carry out, support, or have the potential to support TVA's role as operator are discussed in the following paragraphs:

1.4.2.4.1 OFFICE OF POWER (Figure 1.4-3)

Nuclear Licensing Staff

The Nuclear Licensing Staff (NLS) is responsible for coordinating the licensing activities of TVA nuclear power plants with the Nuclear Regulatory Commission. The NLS is responsible for providing the interface between TVA and all the Project participants for the CRBRP regarding licensing.

Division of Nuclear Power

The Division of Nuclear Power has the responsibility for the operation and maintenance of all TVA nuclear electric generating plants and will have this responsibility for the CRBRP. Additional information about the responsibilities of the Division of Nuclear Power for the CRBRP is included in Section 13.1.

Division of Power System Operations

This division provides the services of its central electrical, instrumentation, and chemical laboratory and technical staff. In addition, field test engineers are provided for chemical and laboratory tests and for solution of special electrical engineering and chemical problems. Engineers and technicians from this division are responsible for the maintenance and testing of the relaying associated with the transmission system and the inter-TVA communications system.

Office of Quality Assurance

The Office Assurance will be responsible for assuring the implementation and maintenance of an effective euality assurance program, including the auditing of all safety-related activities of the CRBRP. Through the audit program, existing and potential deficiencies are identified and appropriate corrective actions are assigned. Through formal audit reports, the Nuclear Safety Review Board and Manager of Power are advised of any identified deviations from procedural requirements and licensing commitments. Its relationship within the TVA corporate structure is indicated in Figure 1.4-4.

1.4.2.4.2 <u>BALANCE OF TVA ORGANIZATION (Figure 1.4-4)</u>

1.4.2.4.2.1 <u>DIVISION/OFFICE/STAFF OUTSIDE THE OFFICE OF POWER THAT PROVIDE A DIRECT SERVICE</u>:

<u>Division of Occupational Health & Safety</u>

The Division of Occupational Health & Safety has corporate responsibility for formulating TVA occupational health and safety plans and policies. It develops and issues criteria and standards for control of hazards in the work-place. It audits and appraises the effectiveness of occupational health and safety programs througout TVA, supports the investigation of serious accidents, investigates employee complaints of unsafe or unhealthful working conditions referred to the "designated Agency safety and health official," and ensures appropriate follow through. It ensures through program evaluation that the occupational health and safety program is adequately implemented in TVA organizations consistent with corporate policies and plans and in compliance with applicable standards and regulations.

It coordinates TVA review of regulatory requirements and industry trends relating to occupational health and safety practices and coordinates the development of Agency comments on proposed regulations.

It provides industrial hygiene services for the Agency, including surveys to measure employee exposure to noise, toxic chemicals, and physical agents, and recommends appropriate administrative and engineering control methods. It plans and coordinates emergency repsonse capability for dealing with major spills or releases of hazardous and toxic materials on TVA property. It is responsible for handling workplace and community noise prevention programs.

It provides accident prevention and safety consulting services, as requested, including fire protection, handling of explosives, and management of hazardous and toxic materials.

<u>Radiological Health Staff</u>

The Radiological Health Staff provides program definition, oversight and performance evaluation for TVA radiological control and protection.

<u>Division of Medical Services</u>

The Division of Medical Services is responsible for TVA's overall health program. This will include employee health services for the CRBRP.

<u>Public Safety Services Staff</u>

The Public Safety Staff will share industrial-radiological security responsibilities for the CRBRP with the Division of Nuclear Power in the Office of Power. The functional relations between these groups and how they share industrial-radiological security responsibilities are discussed in Section 13.7 under Radiological Security Program.

Office of Natural Resources

This office through its Environmental Quality Staff, Air Resoures Program, and Water Resources Program provides environmental technical guidance, assistance, and services as needed to assure activities are in compliance with Federal environmental regulations and legislation.

Nuclear Safety Review Staff

The Nuclear Safety Review Staff is a top-management level group which acts independently of TVA organizations concerned with the design, construction, operation, and support of nuclear plants to monitor, review, and audit TVA's nuclear activities and advise the Board on nuclear safety policy.

1.4.2.4.2.2 Other Organizations

In addition to the organizations listed in Section 1.4.2.4.2.1, any other TVA Organization is available to provide service for the CRBRP.

1.4.2.5 CONTRACTOR ORGANIZATIONS

1.4.2.5.1 WESTINGHOUSE ELECTRIC CORPORATION (Figures 1.4-5,6)

The Advanced Reactors Division (ARD) is part of the Advanced Power Systems Divisions which are responsible for all of the fast reactor programs within the corporation. In the Clinch River Breeder Reactor Plant Project, ARD has the overall responsibility for designing and supplying the entire Nuclear Island (NI) of the plant and for conducting the overall demonstration plant project. This responsibility includes management of Reactor Manufacturer (RM) functions at Atomics International (AI) and General Electric (GE) and interfacing with Burns & Roe regarding the Nuclear Island which is discussed in more detail in Section 1.4.2.5.1.1. The personnel involved with ARD's activities related to CRBRP are located at two sites. The W-OR activities take place in Oak Ridge, Tennessee and the W-WM activities take place at the Waltz Mill site near Madison, Pennsylvania. Division management is located at the Waltz Mill site. The following paragraphs describe functions of senior managers directly concerned with the Clinch River Project. In addition, Westinghouse is able to draw on the expertise of engineering and management personnel associated with the FFTF Project and R & D programs, both at the Advanced Reactors Division and at the Hanford Engineering Development Laboratories, managed by the Westinghouse Hanford Company.

General Manager, Advanced Power Systems Divisions

The General Manager of the Advanced Power Systems Divisions is the senior corporate official responsible for all Liquid Metal Fast Breeder Reactor (LMFBR) activities in Westinghouse. This includes direction of both the Advanced Reactors Division and the Westinghouse Hanford Company. He reports to the Executive Vice President, Nuclear Energy Systems, and is thus able to draw upon the required corporate resources to assure the necessary support of the Project.

General Manager, Advanced Reactors Division

The General Manager of the Advanced Reactors Division reports to the General Manager of the Advanced Power Systems Divisions and is responsible for all the design, development and other activities of the Division. He provides direction and guidance to the CRBRP Project Manager, the Project Management Services Manager, other Project Managers, the Technology Manager, the Product Assurance Manager, the Controller and the Nuclear Safety and Reliability Manager. He conducts reviews of progress being made on the Clinch River Project, and assures that any problems requiring special attention by senior corporate management are immediately made visible.

Product Assurance Manager

The Product Assurance Manager is responsible to the General Manager, Advanced Reactors Division for providing overall ARD Quality Assurance functions. Since the Product Assurance Manager reports directly to the Division General Manager, he has the organizational freedom to initiate and evaluate solutions to product problems and avoid any compromise in product quality resulting from other requirements such as cost, scheduling, production and manufacture. He directs matters of Corporate and Divisional Quality Assurance Policy throughout the Division.

1.4.2.5.1.1 ARD W-OR Organization (Figure 1.4-5)

CRBRP Project Manager

The CRBRP Project Manager reports to the General Manager, Advanced Reactors Division and is responsible for discharging the tasks associated with the Westinghouse role for the Nuclear Steam Supply System (NSSS). The Project Manager is responsible for the NSSS technical integration and program management, for all technical and program planning, contract and project administration, customer liaison, and direction of all NSSS development, design, procurement, component fabrication, testing efforts and the W-OR Quality Assurance Program. In addition, he is responsible for providing the necessary technical requirements to the Architect-Engineer (A-E) regarding NSSS facilities requirements and support, and for providing the necessary construction liaison for the NSSS. He is responsible for the identification and timely resolution of project problems in the above areas.

CRBRP Technical Director

The CRBRP Technical Director reports to the CRBRP Project Manager and is responsible for technical decisions in the discharge of the W-OR NSSS tasks. He is also responsible for NSSS Licensing and Reliability.

W CRBRP Quality Assurance Manager

W CRBRP Quality Assurance Manager reports to the CRBRP Project Manager and has been delegated the authority and execution responsibility by the CRBRP Project Manager for establishing, maintaining, directing and managing the W CRBRP NSSS quality assurance program as described in Chapter 17, Appendix 17D.

CRBRP Program Control Manager

The CRBRP Program Control Manager is responsible to the CRBRP Project Manager for NSSS plans and schedules, estimates, budgets, cost control, development of project policies and requirements, cost reduction efforts, project administration, and data and systems management.

W-OR Procurement Manager

The W-OR Procurement Manager is responsible to the CRBRP Project Manager for W-OR Procurement activities. He administers and controls the W-OR contracts with the RMs and W-OR suppliers to assure that the required systems, structures and components are procured consistent with contract requirements.

GE Program Manager

The GE Program Manager reports to the CRBRP Project Manager and is responsible for the coordination of GE RM activities.

AI Program Manager

The AI Program Manager reports to the CRBRP Project Manager and is responsible for the coordination of AI RM activities.

CRBRP Systems Integration Manager

The CRBRP Systems Integration Manager reports to the CRBRP Project Manager and is responsible for control and integration of the NSSS design and system interface including the W-OR/AE-Constructor interface and development activities. In addition, he is responsible for the plant systems and safety-related design activities.

1.4.2.5.1.2 ARD W-WM Organization (Figure 1.4-6)

Project Management Service Manager

The Project Management Services Manager reports to the General Manager, Advanced Reactors Division and is responsible for divisional program control, engineering services functions, administration of staff functions, preparation and maintenance of divisional policies/procedures, coordination of standards and patent activities, procurement and the functions of the CRBRP Project Deputy Manager.

Program Control Manager

The Program Control Manager is responsible to the Project Management Services Manager for division planning, scheduling, budget and cost control, data management, procedure development and maintenance, cost/schedule analysis and reporting and for standards and patent coordiation.

Procurement Manager

The Procurement Manager is responsible to the Project Management Services manager for all W-WM procurement related to CRBRP.

CRBRP Project Deputy Manager

The CRBRP Project Deputy Manager reports to the Project Management Services Manager, Advanced Reactors Division and is responsible for discharging the tasks assigned to W-WM. In this capacity the Project Deputy Manager has the responsibility for all the technical and financial planning associated with the Westinghouse W-WM activities. The CRBRP Project Deputy Manager takes direction from the CRBRP Project Manger.

CRBRP Program Management Manager

The CRBRP Program Management Manager is responsible to the CRBRP Project Deputy Manager for directing, integrating, coordinating and monitoring the engineering, safety and licensing, program control and external interfacing efforts required at W-WM.

Steam Generator Program Manager

The Steam Generator Program manager is responsible to the CRBRP Project Deputy Manager for all activities related to the steam generator components design and fabrication program.

Engineering Manager

The Engineering Manager reports to the General Manager, Advanced Reactors Division and is responsible for engineering and design of the division-wide technical projects. In this capacity the Engineering Manager takes direction from the CRBRP Project Deputy Manager for the engineering and design of CRBRP NSSS systems and components.

Reactor Engineering Manager

The Reactor Engineering Manager takes direction, through the Engineering Manager, from the CRBRP Project Deputy Manager for establishing system requirements for the reactor enclosure, internals, and control rod systems; and the design, documentation, shipment, and installation support of the reactor vessel, reactor internals, reactor primary control rod system, reactor guard vessel, reactor closure head, and the components for the head access area and the reactor cavity and for the stress and thermal/hydraulic analysis of the permanent reactor components.

Nuclear Systems Engineering Manager

The Nuclear Systems Engineering Manager takes direction, through the Engineering Manager, from the CRBRP Project Deputy Manager for structures analyses, nuclear design, core thermal and hydraulic analyses, shielding analyses, and the design, documentation, and installation support of the fuel and removable assemblies.

Plant Engineering Manager

The Plant Engineering Manager takes direction, through the Engineering Manager, from the CRBRP Project Deputy Manager for establishing system requirements for the reactor heat transport system, plant control, data handling, reactor and vessel instrumentation systems, plant protection systems, as well as the design, fabrication documentation, shipment, and installation support of the components in those systems. In addition, he is responsible for providing overall plant performance analyses, and the manufacturing engineering support for all W-WM NSSS components.

Structural Mechanics and Computer Systems Manager

The Structural Mechanics and Computer Systems Manager takes direction, through the Engineering Manager, from the CRBRP Project Deputy Manager for the performance of structural and stress analysis in support of CRBRP activities at W-WM.

W-WM Safety and Licensing Activities

All W-WM safety and licensing activities are performed by the Division's Nuclear Safety and Reliability Department which is independent from the W-WM Engineering and Procurement Organizations. This department is responsible to the General manager for assuring that all W-WM nuclear safety and licensing requirements have been satisfed. To accomplish this the Nuclear Safety and Reliability Department reviews all W-WM design documentation, prepares and coordinates licensing documentation within W-WM, performs reliability analyses and performs safety analyses.

W-WM Quality Assurance Activities

All W-WM Quality Assurance activities are performed by the Division's Product Assurance Department which is totally independent from the W-WM Engineering and Procurement Organizations. For the description of the Divisional Quality Assurance Organization and its duties and responsibilities with respect to W-WM activities see Chapter 17 Appendix 17D.

1.4.2.5.2 Burns and Roe, Inc. - Breeder Reactor Division (Figure 1.4-7)

Breeder Reactor Division Senior Corporate Vice President

The Senior Corporate Vice President and Director of the Breeder Reactor Division is the senior corporate officer assigned to the project and reports to the President. He draws upon the total resources of the corporation to assure that all necessary actions and support are forthcoming. He provides senior technical guidance as necessary. He assures that any problems requiring attention and resolution are being acted on in a timely manner.

## Breeder Reactor Division Vice President

The Vice President and Deputy Director of the Breeder Reactor Division reports to the senior corporate officer assigned to the project. He provides guidance and direction to the Project Manager and the Project Quality Assurance Manager in the conduct of the project. He performs special reviews of the engineering and design work being conducted on the project and of progress being made. He assures that any problems requiring attention and resolution are being acted on in a timely manner. He contacts senior representatives of DOE and the LRM as necessary to assure satisfactory completion of overall project efforts.

## CRBRP Project - Project Manager

The Project Manager reports to the division Vice President and is assigned overall responsibility and authority for carrying out Burns and Roe's contractual commitments to DOE. He directs and coordinates all project activities in a manner to assure that all Burns and Roe efforts are proceeding in an integrated fashion which will support procurement and construction efforts and will produce a satisfactory technical product, on time, and at minimum cost to DOE. He assures that the engineering and design work by Burns and Roe provides a safe and reliable plant with minimum environmental impact, and a plant which has good operability, availability, maintainability, flexibility, inspectability, and prospect for future economy. He is the official point of contact for the project within Burns and Roe and assures that Burns and Roe's efforts are carried out in a satisfactory manner. He issues management reports and information concerning the project.

## Assistant Project Manager Engineering and Design Services

The Assistant Project Manager, Engineering and Design Services, reports to the Project Manager and is assigned responsibility and authority for the overall direction and coordination of the engineering and design effort including those performed under subcontracts by Holmes and Narver, Inc. He integrates the engineering and design effort in the various areas to assure a sound and technically satisfactory and licensable design which is completed on schedule and within budget. He approves initial issues and revisions, as required, of system design descriptions, drawings, specifications and all technical work on the Project and is assisted in these activities by Project Engineers. He assures that engineering and design efforts are properly interfaced, as to both scope and schedule, with the engineering and design work of the Reactor Manufacturers as applicable.

## Assistant Project Manager, Licensing and Procurement Services

The Assistant Project Manager, Licensing and Procurement Services, reports to the Project Manager and is assigned responsibility and authority for the overall direction of licensing and environmental activities for the SAR and ER. His responsibilities also include the overall direction of procurement management, procurement coordination, vendor documents, and coordinating with engineering and quality assurance in support of such responsibilities.

Project Operations Manager

The Project Operations Manager reports to the Project Manager and is responsible for the administrative, business, planning, scheduling, cost engineering of the Project. For the administrative and business systems, he is responsible for Project cost control and reporting, manpower control, commitment control, and the formulation and monitoring of the Project data bank. He is also responsible for the Management Information Center and for development, custody and control of Project procedures together with the required indoctrination of Project personnel. For the planning, scheduling and cost engineering systems, he provides the necessary controls and monitors overall Project progress and plant capital costs. Under these systems, he is also responsible for the preparation and maintenance of all Project schedules. He is responsible for all project personnel training related to the above systems as required.

Project Office - Resident Manager

The Project Office Resident Manager reports to the Project Manager and coordinates all Burns and Roe operations in Tennessee. He interfaces as necessary and as directed with DOE, PMC, and LRM and the General Constructor. He is supported by a Systems Integration Manager, Planning and Construction Liaison Manger (future Site Manager), Program Manager, and a Licensing and Environmental Representative. He is responsible for the activities conducted at the Project Office and at the construction site, other than Quality Assurance. He shall keep the DOE Project Director advised on as frequent a basis as necessary of status and problems. He is empowered to speak and act for the Burns and Roe Project Manager where necessary.

Contract Supervisor

The Contract Supervisor directs the contract administration functions for the project. He reports to the CRBRP Manager as the central point of contact for the project on contract administration matters. Included in contract administraton matters are preparation of documentation, compliance with notification provision, cost segregation and negotiation.

Quality Assurance Manager

The responsibility and authority of the Quality Assurance Manager is discussed in Section 17E-1.3.

Procurement Manager

The responsibilities of the Procurement Manager, who reports to the Assistant Project Manager, Licensing and Procurement Services, are governed by the scope of work included in Burns and Roe's contract with the CRBRP Project Office. Where Burns and Roe has procurement support responsibility, the Procurement Manager is responsible for the preparation of the potential offeror's lists; review of technical specifications for procurement suitability; administration of Burns and Roe support responsiblities for each subcontract and provides Burns and Roe contact with vendor subcontract administration personnel. Where Burns and Roe has complete procurement responsibility, the Procurement Manager is also responsibile for the conduct of the contracting process including negotiations and award of subcontracts and administration of subcontracts.

Licensing and Environmental Manager

The Licensing and Environmental Manager reports to the Assistant Project Manager, Licensing and Procurement Services and coordinates all Burns and Roe licensing activities for the SAR and the ER. He assures that the requirements of all cognizant regulatory bodies - federal, state and local - are recognized and included in the design. He is responsible for insuring that all revisions to regulations during the course of the work are properly evaluated and included as may be required.

1.4.2.5.3 General Electric Company (Figure 1.4-8)

The Advanced Reactor Systems Department (ARSD) is a part of the Energy Systems and Technology Division (ES&TD) of General Electric Co. (GE). The General Manager of the GE-ARSD reports to the Vice President and General Manager of the ES&TD and is responsible for organizing the resources to carry out such programs and for developing corporate programs that will lead to the eventual commercialization of Advanced Nuclear Power Programs, including LMFBR technology.

The GE-ARSD General Manager conducts review of progress being made on projects within the department and provides direction and guidance to the Section Managers reporting to him. He has the responsibility and authority to issue Department policy and to establish quality goals and objectives. (See Chapter 17, Appendix I for details of the General Managers' QA responsibility).

The GE-ARSD consists of seven sections and the Legal Operation. Each section is headed by a Section Manager who reports to the General Manager and is responsible for an assigned area of responsibility as defined in the following paragraphs.

Clinch River Project Section

The Manager of the Clinch River Project Section is responsible to the Department General Manager, GE-ARSD, for performance of work on the Clinch River Breeder Reactor Plant. Major functional responsibilities related to the plant include engineering, design and supply of hardware for the Intermediate Heat Transport System, Steam Generator System, Auxiliary Heat Removal System, Reactor Heat Transport Instrumentation System, Piping and Equipment Electrical Heating and Control System; and similar responsibilities related to furnishing equipment which is part of systems by others, specifically - the Secondary Control Rod System, Secondary Control System Controls, and the Primary Sodium Pump. Also, Licensing support and Procurement is provided for all Plant related activities. In addition, functional responsibilities include development and test of prototypes for Secondary Control Rod System, Sodium Pump, and Sodium Pump Drive System. The Clinch River Project Section is comprised of six sub-sections with major responsiblities as identified on Figure 1.4-8. Support is provided to the Clinch River Project Section by the other ARSD sections as required.

Engineering Section

The Manager of the Engineering Section is responsible to the Department General Manager. The responsibilities of the Engineering Section in support of the Clinch River Breeder Reactor Plant Project include providing analytical and design engineering services in the areas of structural and thermal hydraulic analyses, safety analyses, reliability engineering and SCRS Design. The Engineering Section provides nuclear engineering support primarily related to the evaluation of critical experiments for the Clinch River Core, and systems Engineering Support.

Product Assurance and Services Section

The Manager of the Product Assurance and Services section is responsible for ensuring an acceptable level of quality in all GE-ARSD products and services. It is the responsibility of Quality Assurance to assure that all technical activities of the Clinch River Project, including those performed by subcontractors, are consistent with the customer quality requirements and company quality policy (see Chapter 17, Appendix I for further detail). He also provides leadership and coordinates development of management systems and procedures to guide and control all Department activities; and provides centralized engineering, technical and administrative support services for the Department. Services include test operations in support of engineering development, plant materials, laboratory activities, and experimental facility design and construction for LMFBR programs. Such activities include work in support of assigned projects as well as the development of new systems and components for future LMFBR product lines.

Applications Engineering and Planning Section

The Manager of Applications Engineering and Planning Section is responsible to the Department General Manager for recommending goals and objectives and formulating and implementing strategies and action plans relating to the marketing of current Department services and products and related contract negotiation and administration and the market development for the Department's new products and services. Applications Engineering and Planning is also responsible for the negotiation and administration of all contractual matters related to the Clinch River Project.

Technology and Special Project Section

The Manager of the Technology and Special Projects Section is responsible for coordinating and directing the overall management and execution of all projects undertaken by the GE-ARSD with the exception of those specifically assigned to other sections in the department by the Department General Manager. Similarly, he is responsible for coordinating the funding, reporting and measurement or progress of the department Development Authorizations (DA's). He provides the primary technical and programmatic interface between the Department and the Department of Energy (DOE) and other customer organizations on projects and related matters. He also provides technical and programmatic leadership and assistance to the Applications Engineering and Planning Section on project proposal and contract activities, product planning, product applications, and market development.

GE-ARSD Legal Operation

The GE-ARSD Legal Operation is staffed by the Department Counsel who is responsible to the Department General Manager for advice and counseling of department management regarding legal implications of contracts and other arrangements which legally bind the Company. In addition, Counsel participates with other members of the staff in the general operation of the business, advises on antitrust, labor, government regulatory, equal employment and other matters of legal significance. Counsel is assisted by patent counsel on matters involving patents and data.

GE-ARSD Financial Section

The Manager of the Financial Section is responsible to the Department General Manager for reporting financial results of the Department, establishing the financial policies of the Department and providing financial service and counsel to the other GE-ARSD sections. In addition, the Financial Section is responsible for interpretation of financial contract language, establishment and negotiation of overhead rates, and development of operating budgets and long range forecasts of GE-ARSD.

GE-ARSD Employee Relations Section

The Manager of the GE-ARSD Employee Relations Section is responsible to the Department General Manager for identifying, developing and implementing relations programs responsive to the Department needs; for establishing goals, objectives and assuring timely employment of qualified personnel. He also provides coordination, counseling and direction for all Department components in relations areas including Manpower Development and Equal Employment Opportunity and Minority Relations and maintains procedures and records and to assure compliance with federal and state laws in the areas of fair employment practices.

1.4.2.5.4 Rockwell International Corporation (Figure 1.4-9)

The LMFBR Program is being undertaken at Atomics International (AI), a division of the Energy Systems Group of Rockwell International Corporation. The principal organizational entities directly involved in this program at ESG are described below.

Atomics International Division Vice President and General Manager

The Atomics International Division Vice President and General Manager is responsible for the management of the CRBRP Program and related LMFBR Programs. Related LMFBR Programs include LMFBR Base Technology Program activities and a steam generator development and manufacturing program. Therefore, the responsibility for ESG's overall performance on the CRBRP is vested in the General Manager.

LMFBR Programs Director

The LMFBR Program Director has overall responsibility for the LMFBR business segment, including CRBRP Program Activities, Large Plant Design projects, and LMFBR Base Technology.

CRBRP Program Director

The CRBRP Program Director is responsible for the management of the CRBRP Program at ESG. In this capacity, he is responsible for managing the CRBRP Program work in accordance with the contract requirements and providing direction to the functional organizations within ESG for CRBRP development, design and procurement with exception of the Steam Generator Program.

CRBRP Steam Generator Program Director

The Steam Generator Program Director is responsible for the management of the Steam Generator Program at ESG. In this capacity, he is responsible for managing the program work in accordance with the contract requirements and providing direction to the functional organizations within ESG for steam
52 generator development, design, procurement and fabrication.

## Quality Assurance Director

The responsibilities of the Quality Assurance Director are discussed in Chapter 17, Appendix J, Section 2.0.

## Research and Engineering Vice President

The Research and Engineering Vice President is responsible for the management of ESG's centralized engineering activities. On the CRBRP Program, engineering work in support of conceptual design, preliminary design, and final design is assigned to the Engineering Department. Engineering and design work conducted by the Engineering Department includes: Mechanical Design, Drafting and Checking, Electrical and Control Engineering, Materials and Process, Piping and Structural Design, Thermal and Process systems Pressure Components Stress Analysis, Structural Systems Stress Analysis, Specifications and Manuals, Engineering Assurance and Data Management and the verification of design through developmental and acceptance testing.

## Operations Director

The Director of Operations is responsible for the product manufacturing, material purchasing and warehousing in support of the CRBRP in accordance with the controlling programmatic documents. The material purchasing function is responsible for selecting sources, procurement, subcontract administration, assuring adherence to work statements, prices and delivery schedules, receiving, inspection, storage, issuance, payment of invoices, and observing the performance quality of the articles purchased. The manufacturing manager is responsible for reviewing engineering and design work performed by ESG to assure manufacturability. On the CRBRP Program, as with other programs, the Manager of Manufacturing Engineering is responsible for conducting on-the-board reviews, participating in design reviews, and reviewing supplier design information to assure component designs can be fabricated and assembled expeditiously and at minimum cost.

## Finance and Administration Vice President and Controller

The CRBRP administration is under the cognizance of the Finance and Administration Vice President. The Finance Controller reports administratively to the Finance and Administration Vice President and organizationally to the AI Division Vice President and General Manager. Within the Finance and Administration Organization, the Program Business Management function is responsible to the individual projects for assistance in the budgeting and planning of manpower and dollar expenditure rate; for maintaining and reporting project costs and remaining balances; for monitoring and satisfying contractural requirements; for maintaining contract data control systems; and for providing assistance in preparation of project schedules. On the CRBRP Program, Program Administration provides the CRBRP Project management with detailed weekly summaries of manpower expenditures, monthly cost information, projection of figure costs at various subaccount levels, commitment control system reports, and various other reports required by the Project and the customer.


(Next page is 1.4-20)

1.4.2.5.6 Stone and Webster Engineering Corporation (Figure 1.4-11)

The construction of the CRBRP is being undertaken by Stone and Webster Engineering Corporation (SWEC) a wholly owned subsidiary of Stone & Webster, Inc. As general contractor, SWEC will prepare the site, construct permanent plant structures and install both NSSS and B.O.P. components, systems and equipment.

CRBRP Senior Project Manager

The Senior Project Manager for the CRBRP construction effort is a SWEC Vice President and is the senior corporate official responsible for SWEC activities on the CRBRP Project. As Senior Project Manager, he will be responsible for coordinating all SWEC headquarters and field operations required to perform the construction of the Project in accordance with contract requirements. He reports to the President of SWEC and is thus able to draw upon the required corporate resources to assure the necessary support for the Project.

CRBRP Deputy Director of Construction

The Deputy Director of Construction is a SWEC Vice President and the Construction Manager of the CRBRP Project. As Construction Manager, he is responsible for the construction organization and assignment of construction personnel. He participates in establishing company-wide SWEC construction policies and procedures.

CRBRP Project Managers

Management of the SWEC CRBRP construction activities is divided into two areas; control and production. Managers of these areas are accountable to the Senior Project Manager and work directly with the Project participants to support the Project schedule and budget. The Project Manager - Control is responsible for establishing Project construction criteria and determining the timing of and directing of all Project construction criteria and determining schedules, estimates and expenditure forecasts. The Project Manager - Production is responsible for providing the necessary manpower and resources to meet the construction goals, coordinating with other groups and for the quality of the work.

CRBRP Project Quality Assurance Manager

The Project Quality Assurance Manager is responsible for assuring that an adeauate quality assuance program is established, implemented, and documented to meet the requirements of Appendix B, 10CFR50, and RDT F2-2, August 1973, with Addenda I dated 12/73, Addenda II, dated 3/74, and Addenda III, dated 7/11/75, within the scope of the SWEC construction effort. He receives quality assurance guidance from the SWEC Manager of Quality Assurance in SWEC Headquarters.

Senior Site Construction Representative

The Senior Site Construction Representative is in charge of the construction organization at the site and directs the day-to-day activities. He responds to the goals set by the SWEC Project Managers and acts under the guidance of the Deputy Director of Construction.

Superintendent of Field Quality Control

The Superintendent of Field Quality Control is in charge of the quality control organization at the site and directs the day-to-day activities. He is responsible for the implementation of the quality assurance program at the construction site and acts under the direction of the SWEC Project QA Manager. Corporate administration, corporate policy, and corporate resource support are received from the Manager, Field Quality Control Division in SWEC Headquarters.

Contract Administrator

The Contract Administrator provides liaison activities related to the SWEC contract with DOE reviews contract related material, monitors performance and provides the interface with DOE on contractual matters related to construction site activity. The Contract Administrator acts under the direction of the Senior Construction Site Representative.

Engineering Liaison

Engineering Liaison is responsible for providing the SWEC interface in the offices of the Architect-Engineer. Acting under the guidance of the Senior Site Construction Representative, he is responsible for providing SWEC input to the design and engineering process and for providing SWEC with timely information on engineering and design matters which impact construction.

Superintendent of Cost and Scheduling

The Superintendent of Cost and Scheduling acts under the direction of the Senior Construction Site Representative and supervises the project site cost and scheduling program to provide coordinated and integrated cost and planning control necessary for the completion of the construction effort in accordance with master schedules and projected costs.

Construction Administrator

The Construction Administrator reviews design for constructibility and furnishes technical assistance to the Senior Site Construction Representative in planning and execution of the construction program with special attention to areas unique to sodium systems. He is responsible for the daily contact with the Reactor Manufacturers.

Superintendent of Construction

The Superintendent of Construction acts under the direction of the Senior Construction Site Representative and is responsible for the construction of a complete and operating plant in accordance with engineering plans and specifications and planned schedules for the least cost consistent with good quality.

Assistant Superintendent of Construction Engineering

The Assistant Superintendent of Construction Engineering acts under the direction of the Senior Construction Site Representative and directs all SWEC Construction engineering activities for the Project. He directs and controls the distribution of engineering documentation, requisitions permanent plan materials and coordinates with Field Quality Control and the Architect-Engineering in the resolution of problems encountered during the construction phase.

Assistant Superintendent of Construction Services

The Assistant Superintendent of Construction acts under the direction of the Senior Construction Site Representative and is responsible for providing the personnel, purchasing, accounting and office service functions necessary to support the construction effort so that it may proceed in accordance with plans and specifications and according to schedules and budgets.

Safety Supervisor

The Safety Supervisor acts under the direction of the Senior Construction Site Representative and is responsible for the administration of the construction site safety, accident, and fire prevention programs, ensuring adherence to Federal, State, and Local safety regulations and fire ordinances and the SWEC safety program.

1.4.3 INTERRELATIONSHIPS WITH CONTRACTORS AND SUPPLIERS

PMC has contracted with Westinghouse Electric Corporation, acting through its Advanced Reactors Division (ARD), to perform the function of Lead Reactor Manufacturer (LRM) for design, manufacture, and provision of test support for the Nuclear Steam Supply System (NSSS) for the Clinch River Breeder Reactor Plant. Westinghouse also has RM responsibilities and has subcontracted with General Electric Company Energy Systems and Technology Division and Rockwell International (Atomics International Division, AI) to provide the design and manufacture of certain systems for the NSSS. PMC has assigned the administration of its contract with ARD to DOE.

PMC has contracted with Burns and Roe, Inc., to provide the architect-engineer services required for the Project. Burns and Roe has subcontracted with Law Engineering and Testing Company to carry out investigations to determine the suitability of the site geology in support of foundation designs for permanent structures. Burns and Roe also has a subcontract with Holmes and Narver, Inc. to provide services in liquid metal engineering technology. PMC has assigned
45 the administration of its contract with Burns and Roe, Inc. to DOE.

PMC has contracted with Westinghouse Electric Corporation to provide services needed in the preparation of the Environmental Report for the Project and to perform certain other associated tasks. PMC has assigned the administration
45 of its contract with Westinghouse to DOE.

PMC has contracted with Stone & Webster for the construction of the plant. Stone and Webster may subcontract portions of the work to others. PMC has assigned the administration of its contract with Stone and Webster Engineering Corporation to DOE.

45 The DOE provides R&D information in support of the CRBRP Project through its LMFBR base technology programs being carried out by its national laboratories and contractors. A description of related base technology programs is provided in Section 1.5.

## 1.4.4 GENERAL QUALIFICATION REQUIREMENTS OF CRBRP PROJECT PARTICIPANTS

The general qualification requirements for key positions in the management organizations of the CRBRP Project Office and its chief contractors are described in the following sections:

### 1.4.4.1 CRBRP Project Office Organization

The general qualification requirements for key positions in the CRBRP Project Office are identified by Project Office Divisions.

#### Project Director's Office

Qualification requirements for the Project Director include a broad knowledge of engineering and construction theory, technology, systems, components, and applications with particular emphasis on the design, development construction, and operation of large advanced, complex, first-of-a-kind facilities. Such knowledge must be sufficient to permit the formulation of broad programs encompassing unique studies and projects that will substantially advance the application of technology to competitive industrial operations.

This broad knowledge of engineering technology shall be sufficient to permit authoritative technical judgements on concepts, proposals and experiments that will have the effect of determining major direction in program activities. This knowledge would be evidenced by: (1) degree and advanced study in metallurgy, mechanical engineering, chemical engineering, thermodynamics; (2) by many years of progressively responsible experience in design, development, operation and testing programs of large, first-of-a-kind facilities; or by (3) a combination of items one and two.

#### Deputy Director

Qualification requirements include a broad knowledge of engineering, instruction, office contract administration which is evidenced by at least a B.S. degree in a scientific field and many years of responsible experience in project and contract administration.

#### Assistant Director

Qualification requirements include ten years of professional experience including five years in a technically-oriented field, two years of supervisory experience of groups of more than ten people, and one year of experience with total responsibility for the conduct of a specific operation or program. A minimum of a B.S. degree or equivalent in a scientific field is required.

#### Executive Assistant

The minimum qualification requirements are eight years experience in the supervision of technical activities, including three years experience in
personnel acquisition and evaluation functions and office administration 25
functions. He must be capable of working effectively and cooperatively

with department heads within the Project, with other Project participants and with the public. The minimum educational requirement is a B.S. degree in a scientific field with experience in the design, maintenance or operation of electric generating stations.

Construction Division

Assistant Director for Construction

A minimum of fifteen (15) years of progressive experience and responsibility in the planning, management, and supervision of all phases of contract construction efforts is required with major emphasis on heavy construction and complex mechanical systems. He must have a thorough knowledge of contract administration particularly with respect to cost reimbursable contracts and must have a Bachelor's degree in engineering.

Engineering Division

Assistant Director for Engineering

The qualification requirements include a broad knowledge of engineering and construction, both nuclear and conventional, with particular emphasis on the various phases of design, procurement, fabrication, construction, testing, and operation. He must have 10 years experience demonstrating progressively more responsible assignments in reactor design, research, development, tests and evaluations and in power plant design, construction, maintenance and operation. He must also have knowledge of the entire reactor development program, utility systems, and major problems besetting the adoption of reactor systems for the economic production of power. He must have the capability to effectively organize the efforts of several technical organizations to perform timely and responsible reviews, evaluations, and work, and maintain close liaison and communication with all participants. A minimum of a Bachelor of Science degree in a scientific field is required.

45 | Public Safety Division

Assistant Director for Public Safety

The qualification requirements include a broad knowledge of the LMFBR technology as it applies to safety, reliability, nuclear safeguards, and environmental concerns. A broad knowledge of licensing procedures for nuclear facilities is required with five years experience in supervision of a technical staff. A Bachelor of Science degree in an engineering or scientific field is required.

## Operations Division

### Assistant Director for Operations

Minimum qualification requirements include eight years of practical training and experience in the operation and maintenance of a Steam Electrical Generation Plant, including a minimum of three years experience in a nuclear power plant. Familiarity with the CRBRP design and the theory upon which its operation is based is required. A Bachelor of Science degree in electrical, mechanical, or chemical engineering is desirable. 25

Quality Assurance Division

Assistant Director for Quality Assurance

Qualification requirements for the Assistant Director for Quality Assurance is contained in Appendix A, Section 1.4.1 of Chapter 17.

Procurement Division

Assistant Director for Procurement

The qualification requirements include a broad knowledge of laws and regulations applicable to Government contracting, procurement, property management, and traffic management functions. He must have a minimum of five years experience in contract administration and negotiations involving supply, construction, engineering, and R&D contracts and a Bachelor's degree with emphasis in business related subjects such as economics, business administration, accounting, law, and public administration.

Project Control Division

Chief of Project Control

Minimum qualification requirements include ten (10) years experience in the installation, operation, and maintenance of management control systems for research and development projects or programs. This experience should cover cost and schedule controls of contractors; financial controls; contracts; analysis of reports; and interfaces with ERDA Headquarters. A Bachelor's degree in Business Administration or Engineering is required.

Financial Management Division

Chief, Financial Management

The qualification requirements include a knowledge of the theories underlying general accounting, industrial cost accounting, construction accounting, and government fiscal accounting sufficient to advise and assist contractors in the establishment and maintenance of accounting systems.

Knowledge of auditing principles and practices adequate to plan and direct a program of examinations of the financial transactions and business practices of contractors.

Knowledge of the principles, theories and techniques of budget administration and analysis required in budget preparation and review of actions proposed or taken in the day-to-day execution of the budget.

To meet these requirements, an individual would normally have a university degree (accounting major) and 15 years experience in government, industry and public accounting.

1.4.4.2 Westinghouse - ARD - LRM Organization

There are no specific qualification requirements identified for the management positions in the Westinghouse - ARD Organization except as defined in Appendices D and H of Chapter 17. However, the capability of their personnel is demonstrated by the experience and qualifications summarized in the following paragraphs.

Over 400 ARD professionals are working directly on CRBRP. Approximately 100 of these are in management positions.

Essentially 100% of all professionals involved in the Project have Bachelor's degrees and approximately 40% have advanced degrees. The average professional has over seven years experience in LMFBR related work. Approximately 50% of the managers have advanced degrees and the average manager has approximately 12 years of experience in LMFBR related work.

The Bachelors and advanced degrees held by the professionals blanket the following fields:

- Chemistry
- Chemical Engineering
- Civil Engineering
- Electrical/Electronics Engineering
- Industrial/Manufacturing Engineering
- Mechanical Engineering
- Materials Engineering
- Nuclear Engineering
- Physics

ARD utilizes consultants and specialists from other Westinghouse divisions whose background and experience are required for independent design reviews, ASME code expertise, manufacturing engineering, metallurgical problems, stress/thermal/inelastic/structural analysis, and safety related activities.

1.4.4.3 Rockwell International Corporation

There are no specific qualification requirements identified for specific management positions at Atomics International (AI), a division of the Energy
52 Systems Group of Rockwell International Corporation except as defined in Appendix J of Chapter 17. However, the capability of their personnel is demonstrated by the experience and qualifications summarized in the following paragraphs.

There are nearly 30 managers and cognizant engineers assigned to the CRBRP program at AI. These people are supported in their activities by many disciplines and skills in other departments at ESG. However, as an indication, the following statistics are provided for these lead individuals.

The individuals have an average of nearly 17 years in the nuclear field; they have an average of over 13 years experience in liquid metal technology. All of the individuals have Bachelor's degrees, and over 50% have advanced degrees. The disciplines of degrees held include the following:

- Mechanical Engineering
- Electrical Engineering
- Civil Engineering
- Chemical Engineering
- Chemistry
- Physics
- Radiological Physics
- Control Systems Engineering

#### 1.4.4.4 General Electric Company (GE)

GE has no pre-determined sets of qualification requirements for management positions at its Advanced Reactor Systems Department (ARSD), with the exception of the managers within the Quality Assurance Section (ref. Ch. 17, App. I). Rather, an evaluation of the requirements is made when a position becomes open and the best qualified candidate is sought to fill it.

Nearly all of the management level personnel have at least Bachelor's degrees in disciplines appropriate to their areas of responsibility, and about half have advanced degrees.

For those situations where the technical expertise does not exist within the department or where an independent assessment may be desired, specialists and consultants from other divisions within GE are available to provide such assistance as may be required. Outside specialists and consultants are utilized under some circumstances, including personnel from other participants on CRP when appropriate.

#### 1.4.4.5 BURNS AND ROE, INC. - BREEDER REACTOR DIVISION

Specific qualification requirements at the Breeder Reactor Division of Burns and Roe, Inc. are described for key positions identified and described in Section 1.4.2.5.2.

Breeder Reactor Division Vice President

A minimum of 15 years of progressive responsibilities in the management and supervision of all phases of engineering efforts is required with primary emphasis in the nuclear field. He must have a working knowledge of the corporation's resources and also have a Bachelor's degree in Business and/or Science with additional education and/or training in nuclear technology.

CRBRP Project Manager

A minimum of 12 years of progressive responsibility for the management and supervision of technical efforts is essential, with primary emphasis on the development of nuclear power plants. He must have at least a Bachelor of Science degree and education and training in nuclear reactor technology with some training in Business Administration or Management preferred.

Assistant Project Manager

A minimum of 8 years experience in progressively responsible positions for the management and/or supervision of technical efforts primarily in nuclear power plant technology. He must have at least a Bachelor's degree in Science or Engineering with some training in business administration or management.

Contract Supervisor

A minimum of 5 years of practical contract administration experience in the administration and negotiation of government and/or commercial contracts and possess a knowledge of federal procurement regulations and policies. He must have a minimum of a Bachelor's degree in Business Administration or Engineering.

Licensing and Environmental Manager

A minimum of 5 years experience in supervision of nuclear power plant licensing and/or engineering is required. At least a Bachelor of Science degree with education and training in nuclear reactor technology is required.

Project Operations Manager

A minimum of 8 years of experience is required in the management of technical efforts with a detailed knowledge of project management techniques. He must have at least a Bachelor of Science degree with education in management principles.

Procurement Manager

A minimum of 8 years of practical procurement experience in the negotiations and administrations of government and/or commerical contracts and possesses a knowledge of federal procurement regulations and policies. He must have a minimum of a Bachelor degree in Business Administration.

Quality Assurance Manager

The minimum requirements for the Quality Assurance Manager are shown in Section 17E 1.4.1.

Project Office Resident Manager

A minimum of 5 years of progressive responsibility for the management and supervision of technical efforts with primary emphasis in nuclear technology. He must have at least a Bachelor of Science degree or equivalent experience and education and training in nuclear reactor technology.

1.4.4.6 Stone and Webster Engineering Corporation

Specific qualification requirements at Stone and Webster Engineering Corporation are identified for key positions identified and described in Section 1.4.2.5.6. For all the qualification requirements, in lieu of a degree, equivalent qualifications may be substituted based on other educational accomplishments, experience in related fields and technical achievements, such as holding a license as a Professional Engineer or Certification as a Quality or Reliability Engineer by the American Society for Quality Control.

CRBRP Senior Project Manager

A minimum of ten years of progressive responsibilities in the supervision and management of various phases of engineering, construction, and/or quality assurance efforts is required, with primary emphasis in the nuclear power plant field. He must have a working knowledge of the Corporation's resources and also have a Bachelor of Science or Arts degree with additional and/or training in power plant technology.

CRBRP Deputy Director of Construction

A minimum of ten years of progressive responsibilities in the supervision and management of heavy construction projects, with emphasis on the construction of power and/or process facilities. He must have a working knowledge of the Corporation's resources and also have a Bachelor of Science or Arts degree.

CRBRP Project Manager

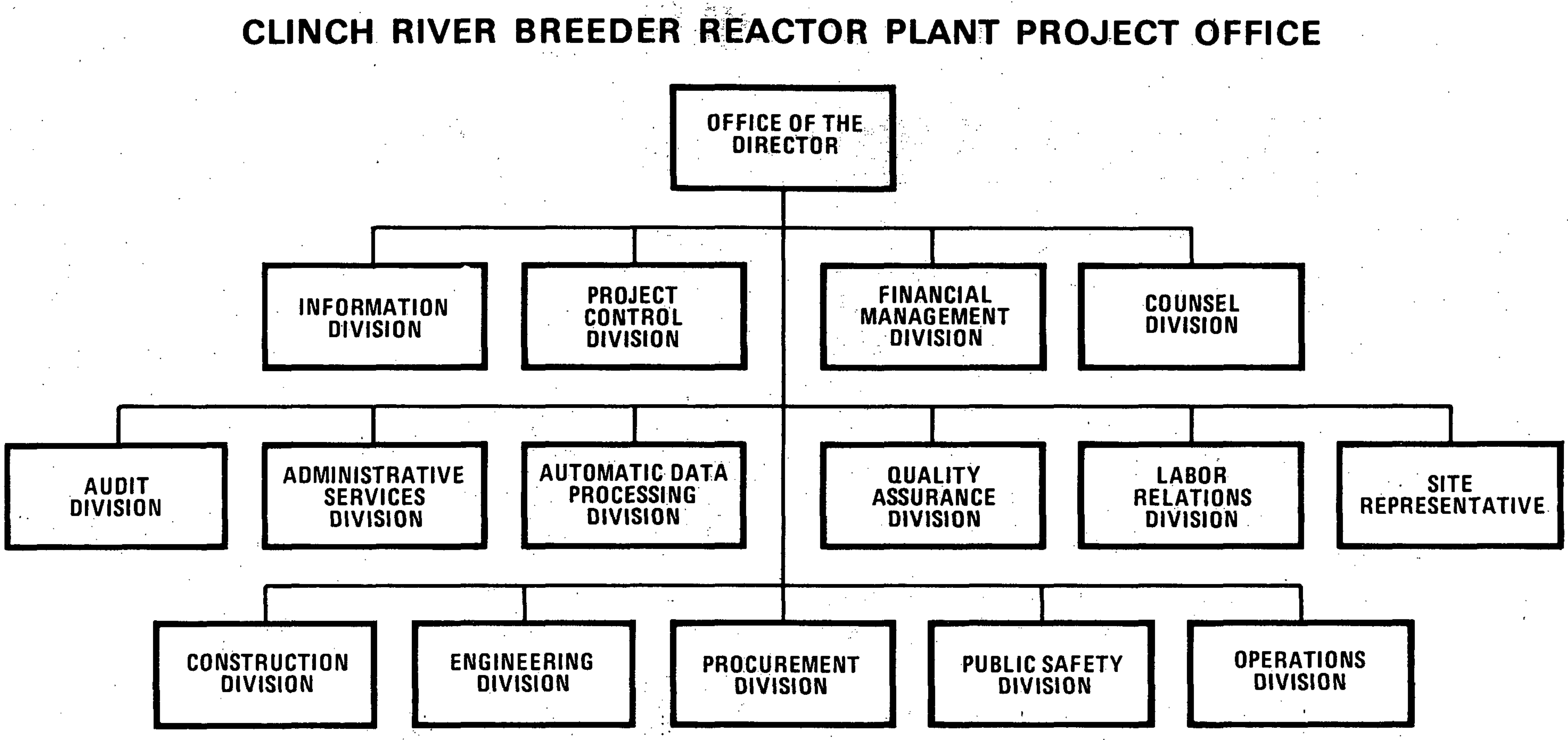
A minimum of ten years of progressive responsibility in the management and supervision of technical efforts is essential, with emphasis in the nuclear power plant field. He must have a Bachelor of Science or Arts degree, with additional education and training in management and power plant technology.

CRBRP Project Quality Assurance Manager

A minimum of ten years in quality assurance and related fields including manufacturing, construction, and/or installation activities. At least two years of this experience shall be associated with the nuclear field in either field or headquarters project quality assurance assignments. He must have a Bachelor of Science or Arts degree.

Senior Site Construction Representative

A minimum of five years in responsible assignments in field engineering and construction activities, with emphasis in the construction of power and/or process facilities. He must have a Bachelor of Science or Arts degree.




Amend. 45

3-77-P0921-7-2

Figure 1.4-1 CRBRP Project Office Organization

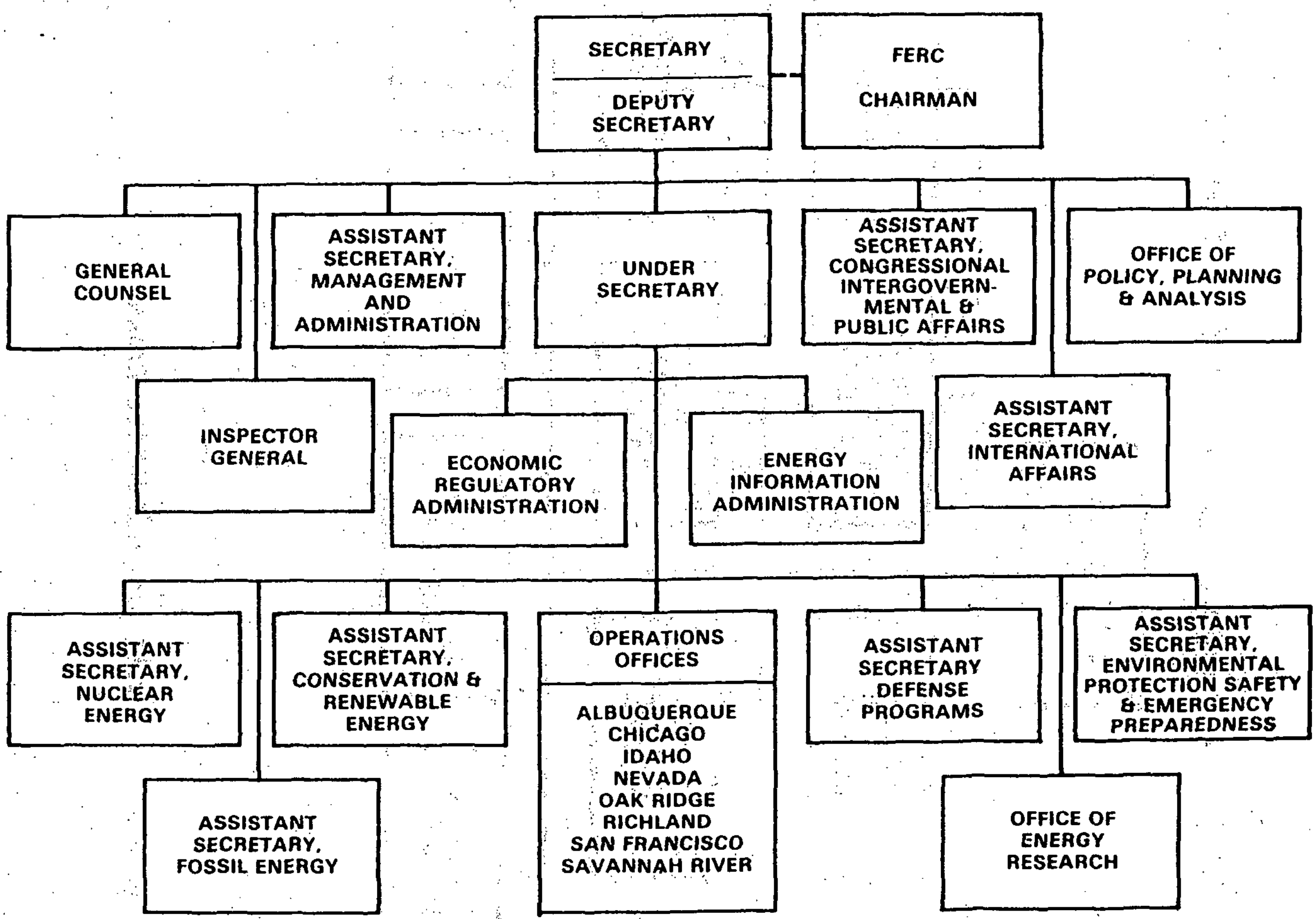



FIGURE 1.4-2 Organization of Department of Energy (DOE)

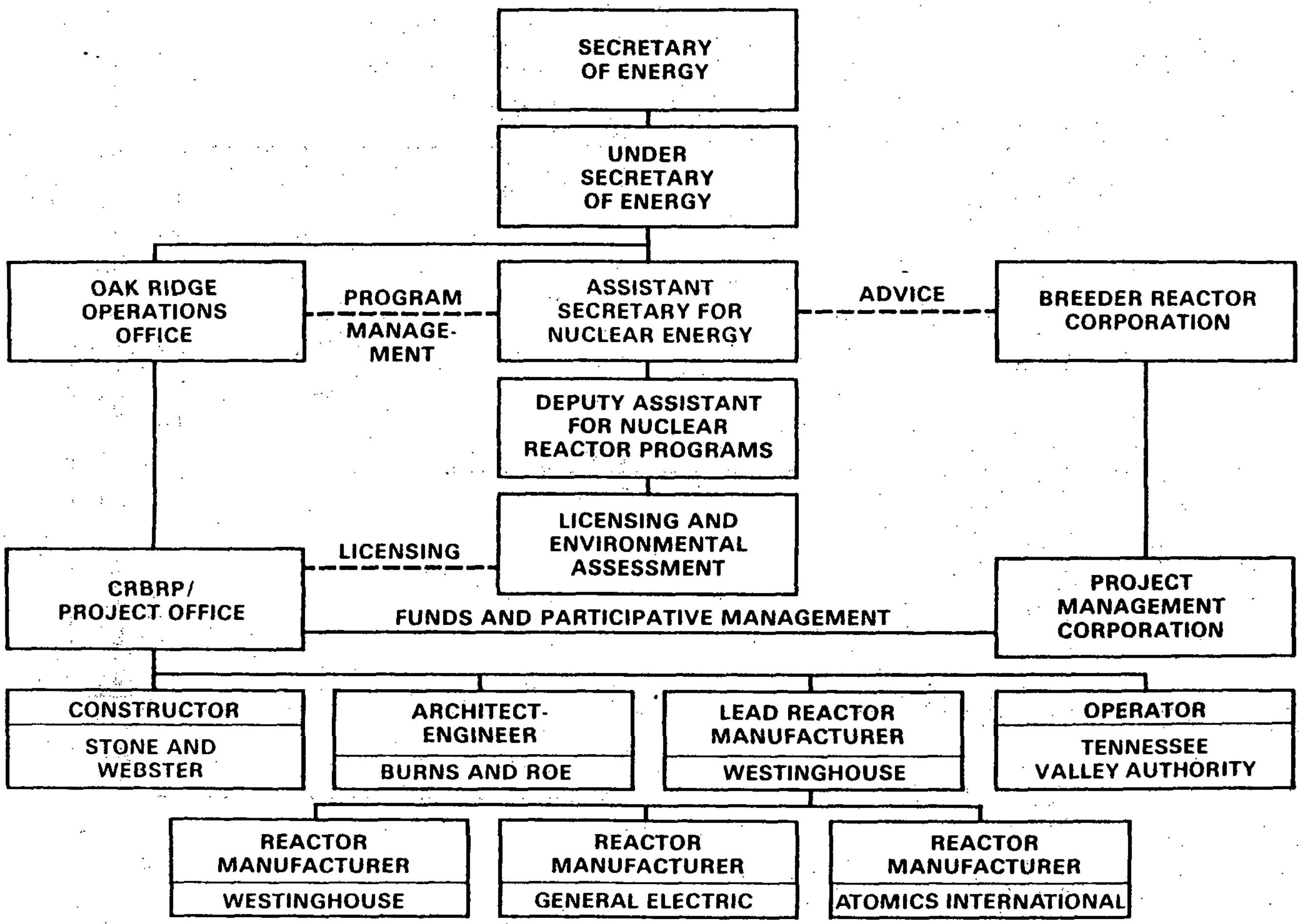
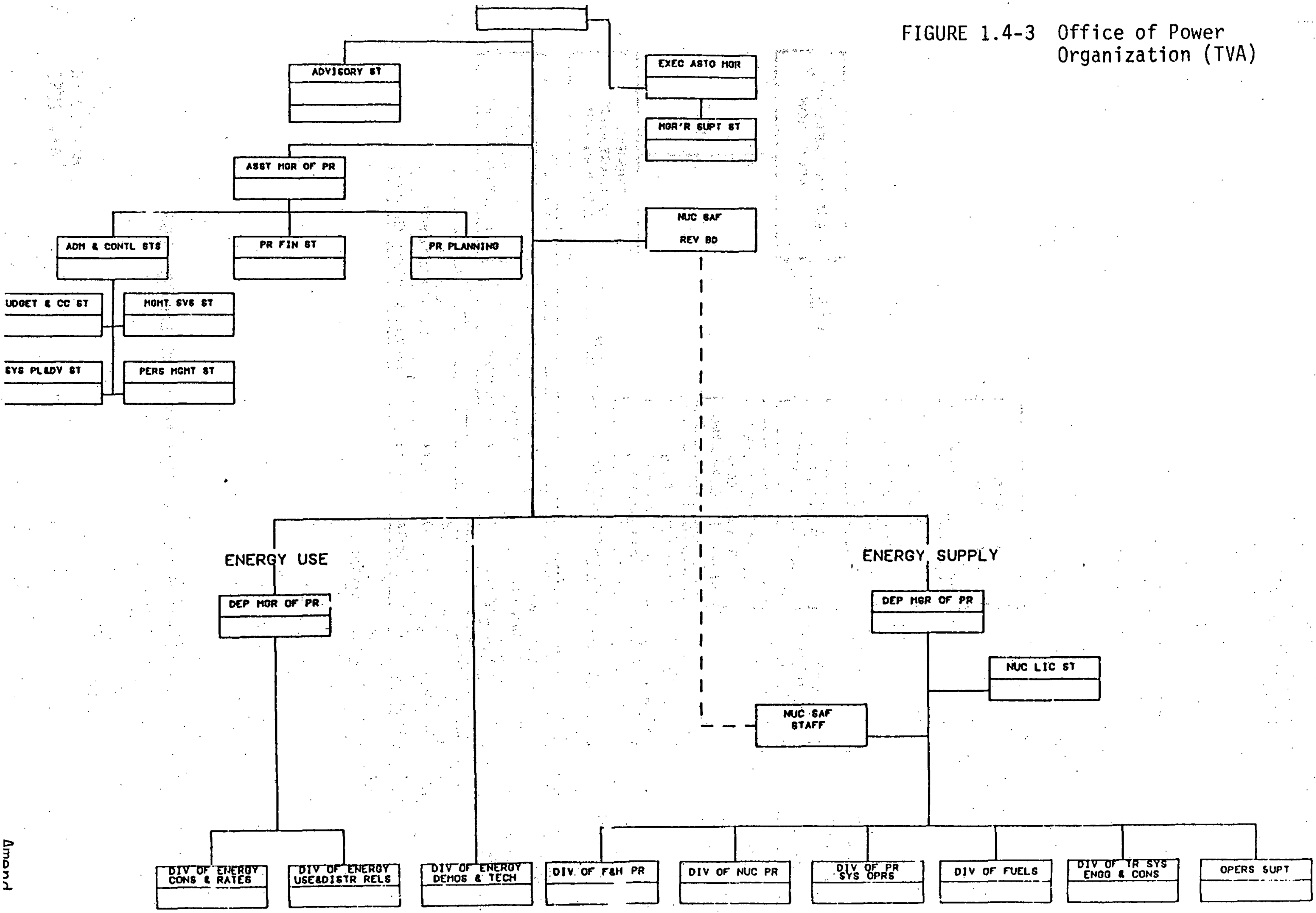



FIGURE 1.4-2a Organization of the Department of Energy (DOE)

FIGURE 1.4-3 Office of Power Organization (TVA)




Amend 75

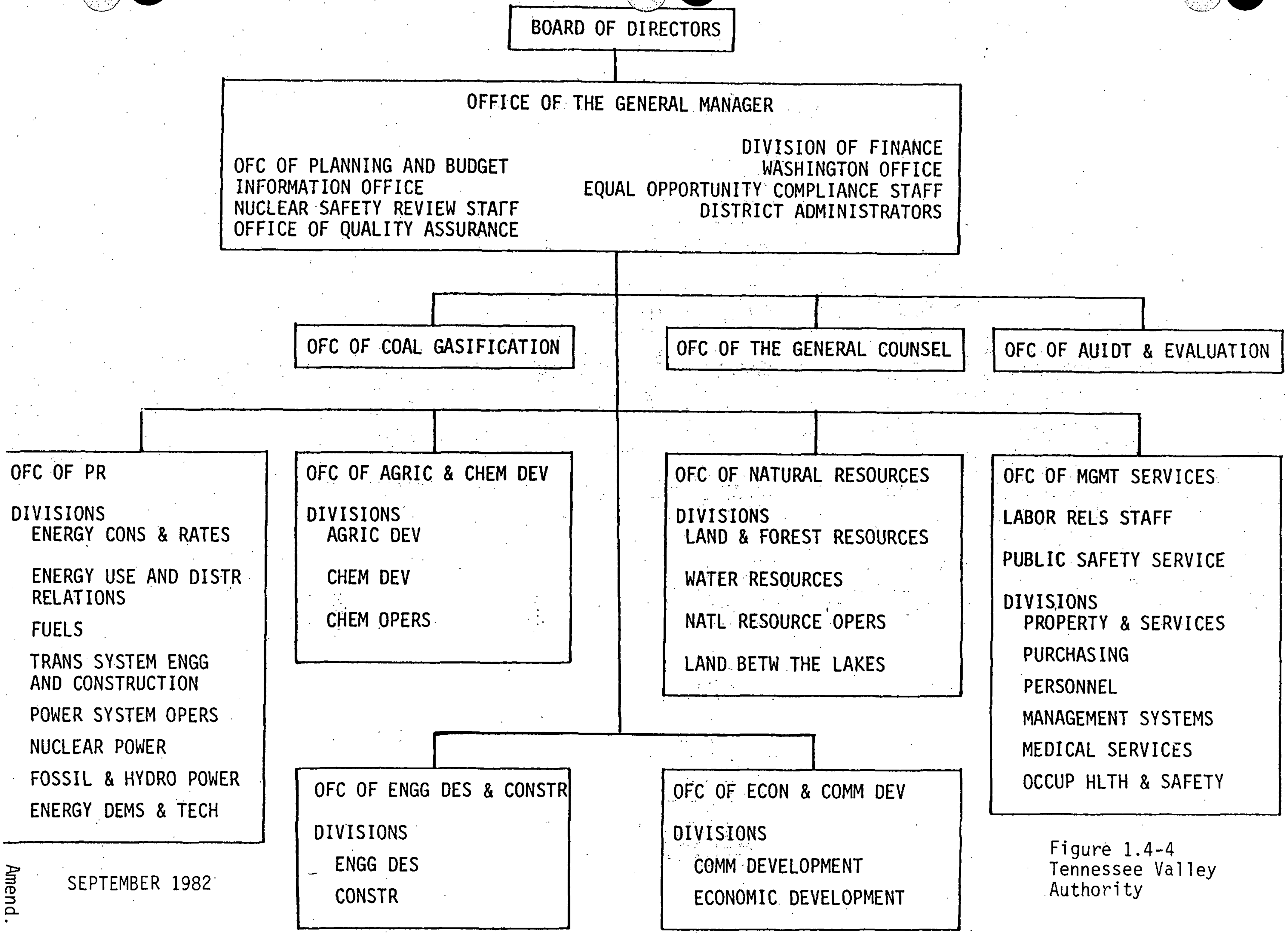



Figure 1.4-4
Tennessee Valley
Authority

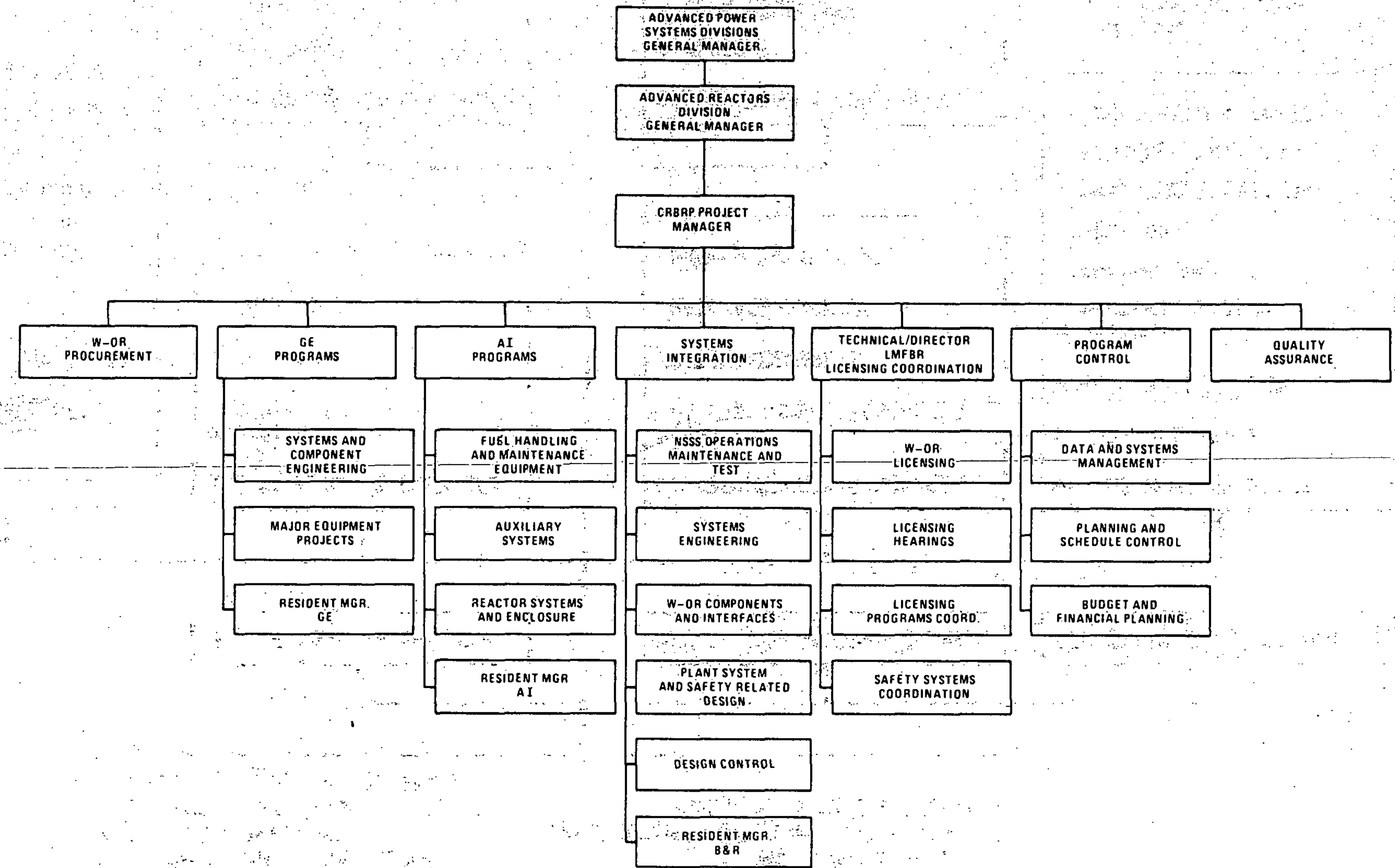



Figure 1.4-5. Westinghouse Electric Corporation – W–OR Organization

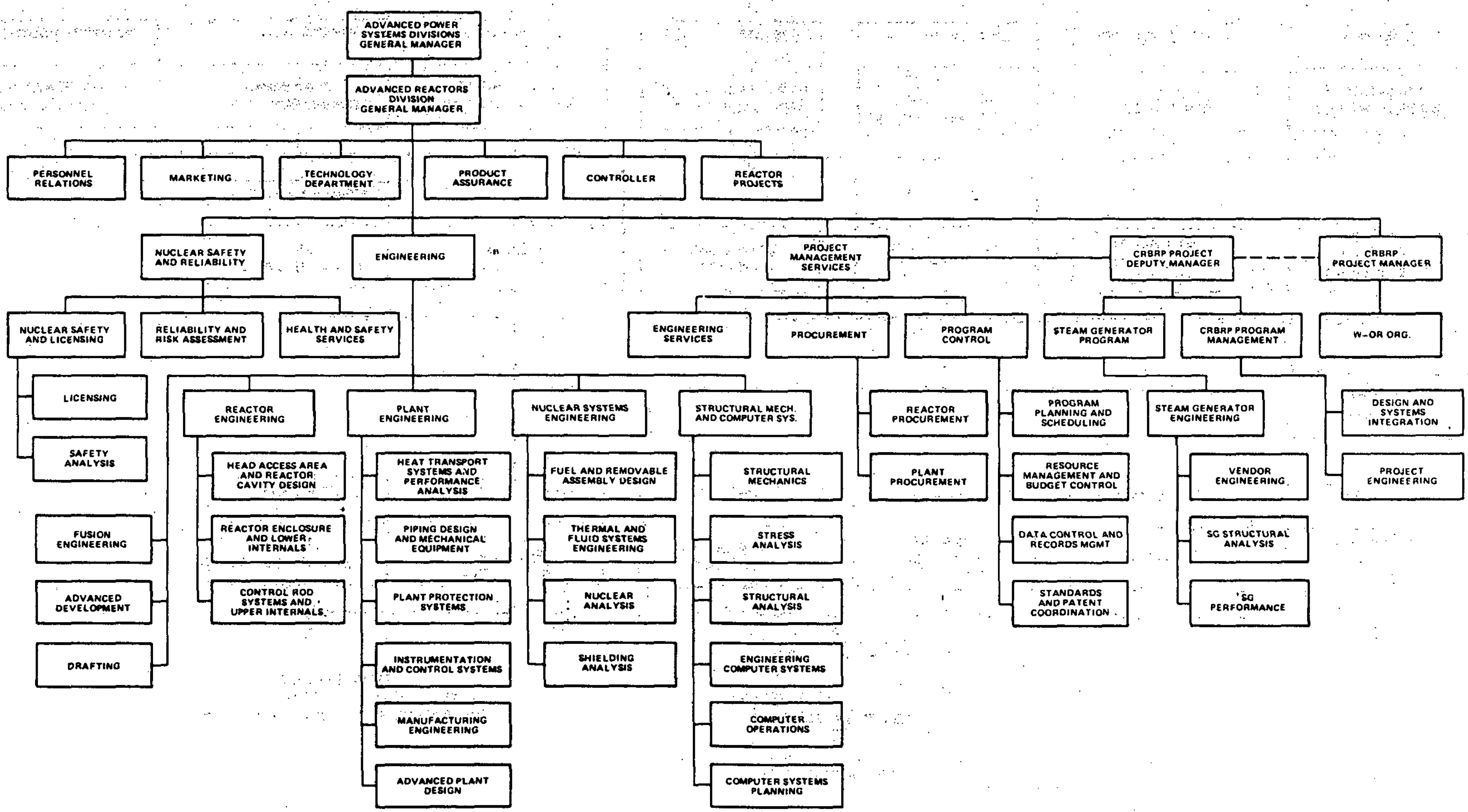



Figure 1.4-6. Westinghouse Electric Corporation – W – WM Organization

7636-3



Amend. 75

## ARCHITECT - ENGINEER ORGANIZATION CHART

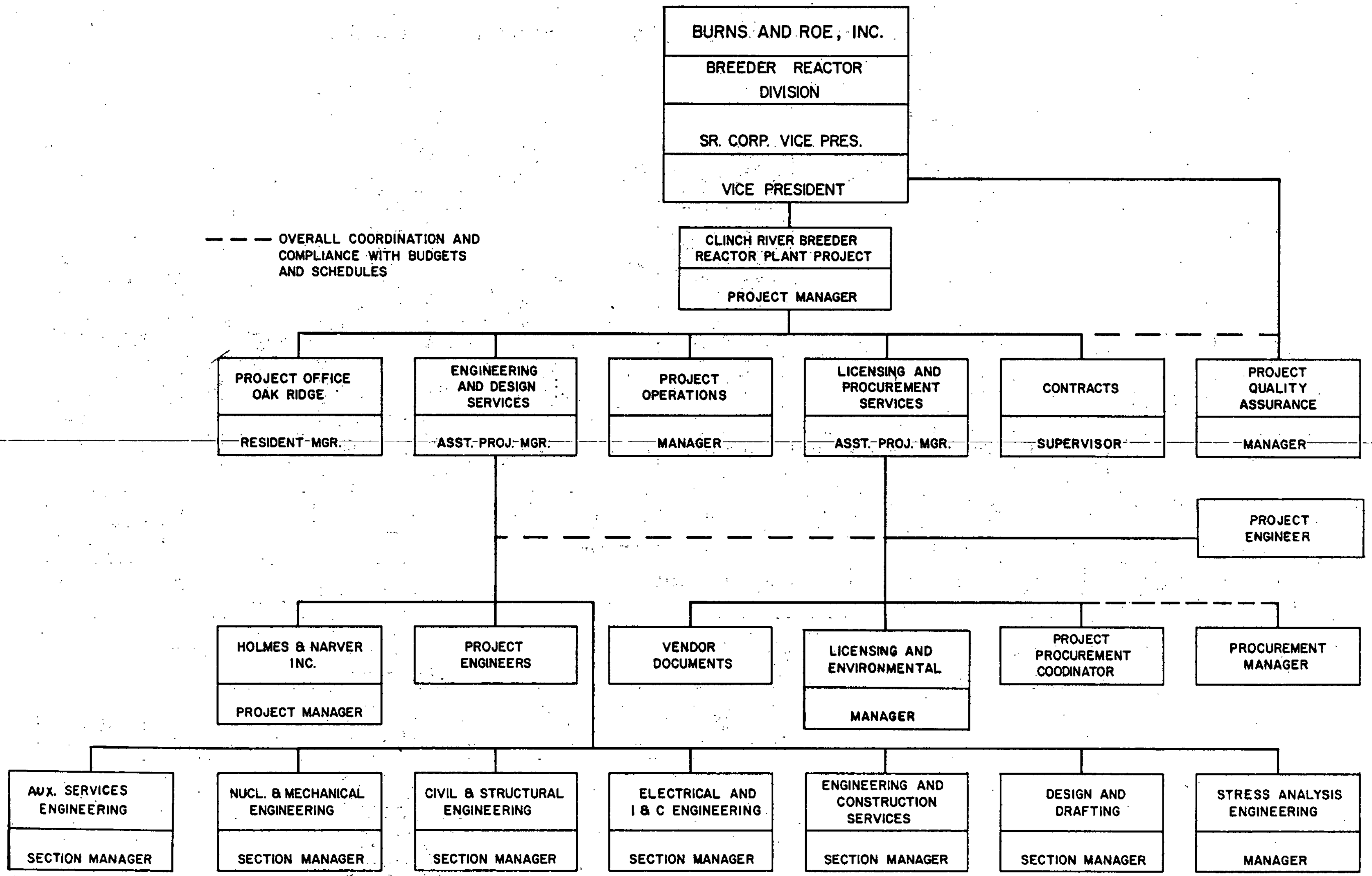



FIGURE 1.4-7 BURNS AND ROE, [illegible]. C[illegible]P PROJECT ORGANIZATION CHART

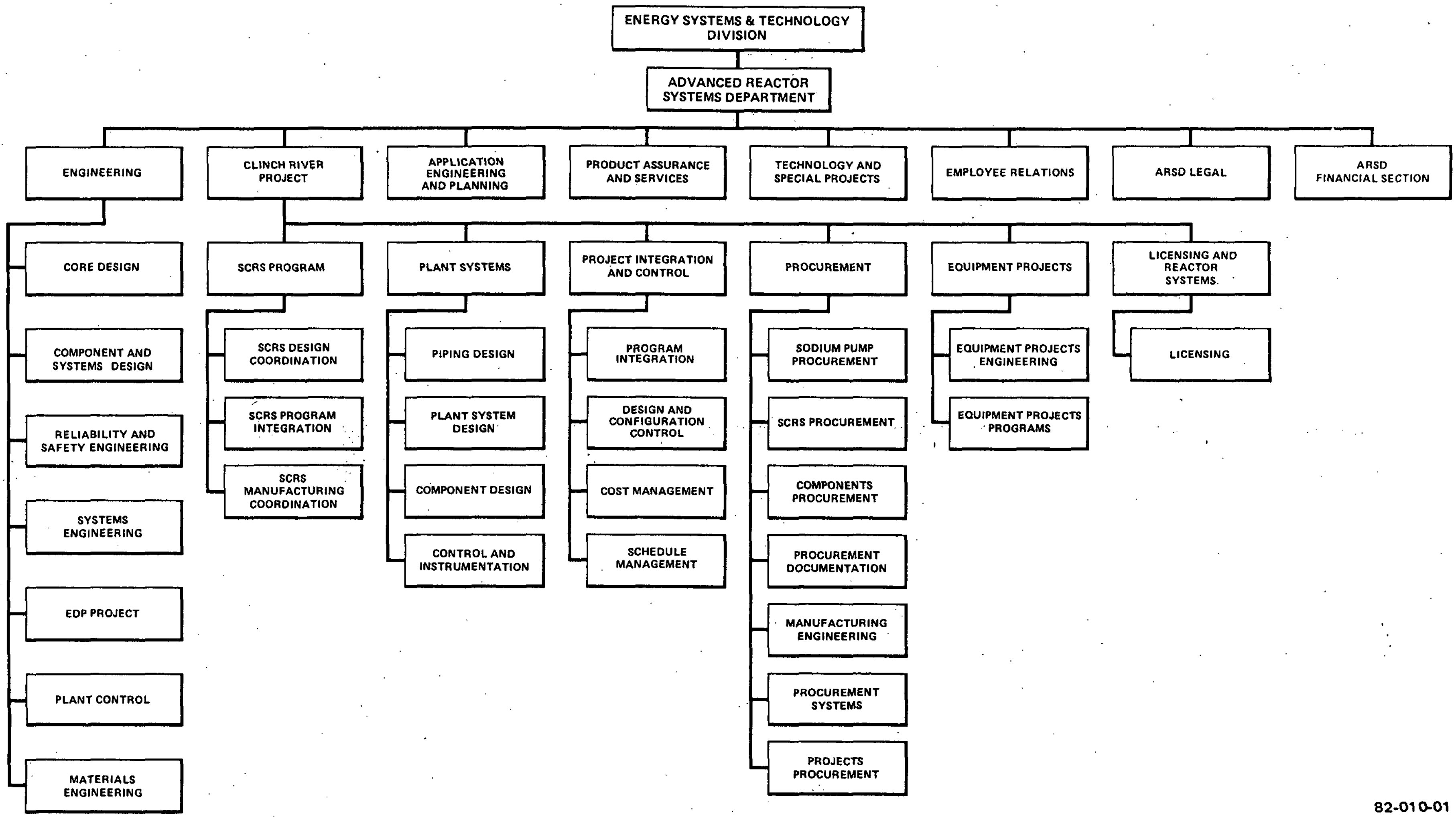



Figure 1.4-8 General Electric Organization



Amend. 75
Feb. 1983

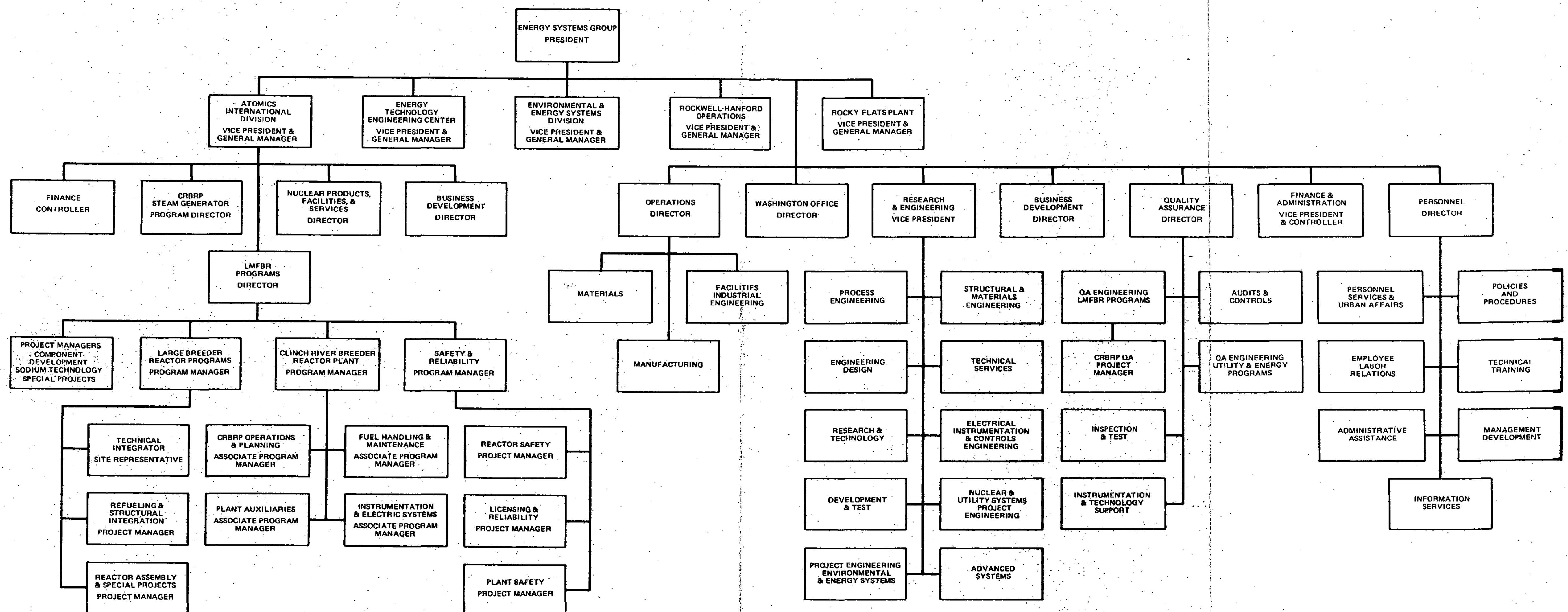
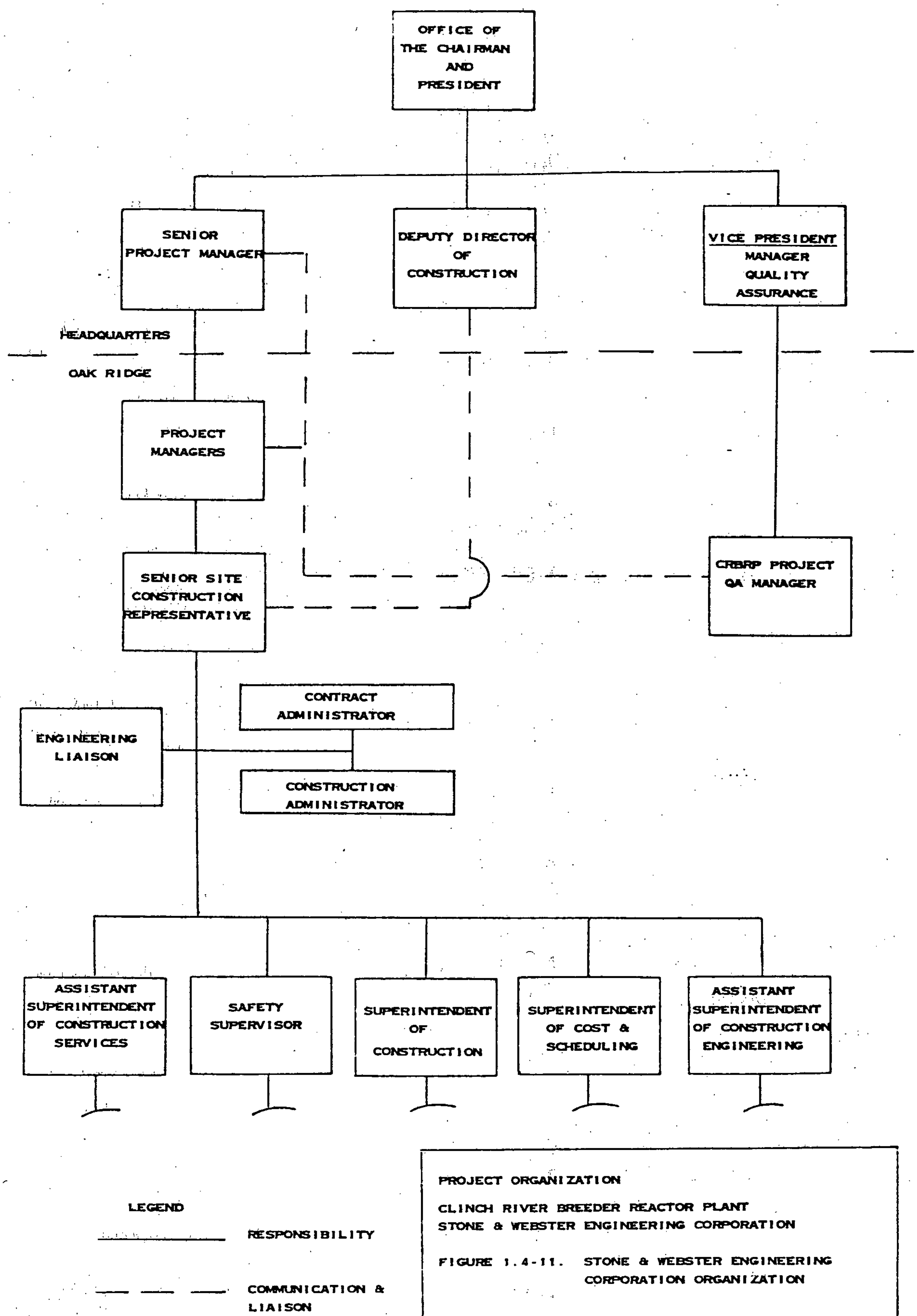



FIGURE 1.4-9 ROCKWELL INTERNATIONAL - ENERGY SYSTEMS GROUP

FIGURE 1.4-10

HAS BEEN DELETED




PROJECT ORGANIZATION

CLINCH RIVER BREEDER REACTOR PLANT
STONE & WEBSTER ENGINEERING CORPORATION

FIGURE 1.4-11. STONE & WEBSTER ENGINEERING CORPORATION ORGANIZATION



Amend. 75
Feb. 1983

## 1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

As a first-of-a-kind plant, it is to be expected that there is a significant quantity of technical information which has yet to be established. Much of this concerns the proving of the design by component life tests, or tests to establish the data necessary to verify the performance of various design features. This section discusses the programs for obtaining necessary technical information to give assurance of the capability of safety features or com-
41 ponents to perform as intended. These programs represent all the technical information development activities necessary to support licensing of the CRBRP. To arrive at the programs described in this section, all safety related systems and components were systemically examined to determine if further development activities were required to support their design. The following criteria were used to determine those safety-related systems and components which need not be subject to Project initiated development
41 programs.

- Directly applicable operating experience or experience which is judged extrapolable to CRBRP conditions.
- Existence of accepted and applicable codes and standards
- Necessary information will be provided by on-going programs without modifications on a time scale compatible with CRBRP schedule.*

41 The remaining programs are presented in the following two groups:

Section 1.5.1 covers the information required to demonstrate the adequacy of a new design. Included in this section is information concerning
26 the Secondary Shutdown System, the Direct Heat Removal Service, radial blanket failure threshold and Sodium-Water Reaction Pressure Relief System.

41 *For example, FFTF fuel technology (Ref. 1) has been extensively utilized in the development of the CRBRP fuel assembly and fuel rod designs. However, CRBRP design and operational variations have required additional design verification tests as described in Subsection 4.2.1.4.1. In the same manner, the LMFBR fuel development program summarized in Subsection 4.2.1.3.1.3 (Irradiation Experience) will be reviewed as the CRBRP steady state and transient operating conditions evolve. When necessary, additional testing has been planned to support CRBRP cladding integrity limits, fuel assembly design bases and performance predictions.
41

Section 1.5.2 treats the information relating to the margin of conservatism of proven designs. Presented in this section are programs directed at confirmation of the extremely low probability of a massive failure
41 of the in-containment sodium boundary. Also included in this section are programs verifying the conservatism provided by the inlet plenum, the core restraint system, the reactivity control, the source range flux monitoring and the ex-vessel transfer machine design.

41

A listing of the areas requiring further technical information to be discussed in the following sections is contained in Table 1.5-1.

## 1.5.1 Information Concerning the Adequacy of a New Design

41

41 Pages 1.5-3 through 1.5-19

Have Been Deleted

Amend. 41
Oct. 1977

- Specification of numerical availability goals and supporting rationale for the two shutdown systems.

- An initial assessment which shows that the reference design will meet the specified goals.

- Issuance of an initial draft of a Reliability Manual describing the methodology to be used in the program.

- A Failure Modes and Effects Analysis of the shutdown systems and pertinent interfaces.

- Preliminary test planning to provide the basis for confirming system availability.

Additional details of work completed or planned is provided in Chapter 7 (Section 7.2.2) and Appendix C.

This confirmation of the reliability is to be provided in the program, by both planning for, and ensuring that reliability requirements are met in the design of the shutdown systems and all interfacing components, and also by confirming the reliability of the final design by test and analysis. All of these functions are to be performed concurrently and iteratively.

The program is a balanced effort of analytical assessment, proof testing, and component and system testing to provide adequate data for system reliability evaluation. Common-mode failures will be identified by a Failure Modes and Effects Analysis (FMEA) supplemented by a Fault Tree Analysis (FTA). Understanding of common-mode failures will be sought by considering functional dependency, parts of similar manufacture, environmental causes, operating and maintenance errors, input and interface parameters, and failures induced by a preceding failure. The analytical program is also designed to suggest remedies for identifiable common-mode failures. These include design diversity, diversity in component fabrication-procurement sources, enhanced testability, reduction in the conditional probability of common mode failures after a casual event has occurred, and stringent procedures designed to eliminate human error in the design, analysis, operation, and maintenance of the CRBRP shutdown system.

The program will extend over 8 to 10 years with most activity in years 2 to 6, as shown in the schedule of Section 1.5.1.1.3. The later work is devoted to long term testing and updating of reliability estimates from data as it is produced. The program utilizes project design experience, disciplined engineering design practices, and reliability techniques, to demonstrate that shutdown system reliability goals are met. Reliability methods used in the program will be published as a manual, complete with examples and guidelines for their use. These include: procedures for model and success-failure criteria development; methods for reliability analysis and code development under the appropriate duty cycle conditions; and procedures for the collection and utilization of data from both the CRBRP and other relevant programs. (See Appendix C) A data bank for general use in LMFBR reliability studies will be developed during the program and include:

a) Shutdown systems operating and development experience

b) Generic failure rate data

c) Vendor experience

d) Fast Test Reactor (FTR) experience

e) National Aeronautics and Space Administration (NASA) and military reliability program experience.

A central Reliability Group monitors and evaluates the various tasks and provides overall guidance for the program to ensure balance and consistency of approach. This provides the close integration with groups such as Design and Test Operations that is necessary to carry out the program effectively. Specifically, feedback from analysis to design is a prime objective of the program.

A more complete and detailed presentation of this program and its rationale is given in Appendix C (Section C3.1.2).

The program for confirmation, verification and assurance of reliability includes the following closely integrated approaches:

I. Reliability Analysis

Two approaches to reliability analysis are being utilized:

A. Qualitative analysis-to establish the fault lines leading to potential failure; to identify the potential for common mode failures; and to establish system failure mode analyses to identify single failure points within each system.

B. Quantitative analysis and procedures-to perform sensitivity analyses; to define reliability goals for subsystems and components; to iteratively perform updated reliability evaluations of components, subsystems and systems; to provide bases for test programs and interpretation of test results; to define a priority listing of component subsystem and system improvement areas.

The following elements are included in the program:

Analysis

Initially, sensitivity studies using the preliminary shutdown reliability model have been used to allocate an initial set of subsystem and component reliability goals for use in preliminary design work. Failure Mode and Effects Analyses (FMEA), qualitative Fault Tree Analyses (FTA) and Common Mode Failure Analyses (CMFA) will be used to identify the failure modes which result in failure to shutdown. Common-modes of failure and a single failure point listing will be developed.

On the basis of the above qualitative analysis, and data available from the data bank including CRBRP test programs to be described below, quantitative shutdown reliability predictions at system, subsystem, and component levels will be provided using the models and codes developed in the task below:

Analytical Models

Reliability block diagrams of the subsystems and components that affect shutdown reliability have been evolved. Preliminary mathematical models depicting the reliability of the shutdown system have been developed which incorporate parameters representing the initial reliability duty cycle and reference maintenance and repair intervals. A manual has been prepared to provide a comprehensive set of reliability methods, complete with examples and appropriate training programs, for project-wide use. Initial activity has concentrated on developing analytical and test planning methods for preliminiary scoping design activities. These include: procedures for management of the program and guidelines for model and success-failure criteria development; methods for qualitative and quantitative reliability analysis and code development under the appropriate duty cycle conditions; and procedures for the collection and utilization of data from both the CRBRP and other relevant programs.

As the CRBRP design matures, the preliminary shutdown reliability model will be updated to incorporate: current Plant Protection System test interval specifications; plant component maintenance procedures; and the results of sensitivity studies. Updated reliability model codes, validated against test data, will be available for project-wide use.

Reliability Envelope

Reliability goals for each shutdown system have been defined by the unavailability (average probability over the test interval that the protection system is in a failed condition at the moment of challenge) of the protection system to respond to plant transient which, without scram, could result in a loss of coolable geometry. The challenges against which the protection system must operate are identified in Chapters 7 and 15, and the plant duty cycle is covered in Appendix B. The transient events and the envelope of conditions (pressure, temperature, loads associated with these events) that can directly affect the reliability of the shutdown system will be defined. Using probabilistic methods, thermal, hydraulic, nuclear effects and environmental data will be calculated for those items which are critical to safe shutdown of the reactor. From these probabilistic loads and environmental data the duty cycles will be developed for use in the reliability analysis of components and subsystems.

II. Test Program

The test program will provide the relevant data necessary to confirm the overall CRBRP shutdown systems reliability when this data is integrated by the reliability analysis with other sources such as existing component and part data, and FFTF and CRBRP design verification testing. The test plan will provide a well-balanced combination of tests involving varying levels of component, subsystem and systems tests of primary and secondary shutdown systems electrical and mechanical hardware. The components and subsystems which will have been tested in the early phase of the program, will be integrated to form a system. The Systems Test program will permit simulation of primary sensor environments to demonstrate the capability of all of the various components to satisfy the shutdown system performance requirements. Abnormal conditions will be simulated in various portions of the subsystems to further demonstrate acceptable systems performance.

The subsystem and systems tests permit an empirical search for common-mode and single random failures to complement qualitative failure mode and effects and fault tree analyses to be performed as part of the design and its review process.

Special tests will be performed under abnormal conditions to verify the absence of common-mode failure mechanisms. These tests will be designed to induce potential common-mode failures related to,

a) internal and external environments

b) design deficiencies

c) functional deficiencies

d) operating and maintenance deficiencies

These special tests will not be statistically oriented but the results may be combined with subsequent analysis to numerically assess the potential for postulated common-mode failures.

The results from all phases of the test program will be transmitted to the shutdown systems design and reliability groups of all organizations involved. The data will be used continuously to update the individual component and overall system numerical reliability assessment. The planned reliability assessments are coordinated with component and plant schedules. Where significant and necessary reliability can result from design changes that become apparent from conduct of tests, these design changes will be made and testing performed, as required, to substantiate reliable performance of the changed design. In summary, those tests described in the following sections will be planned to derive the maximum amount of data related to component, subsystem and systems reliability in a time span compatible with CRBRP design, fabrication, construction, and licensing activities.

Details of the Components and Subsystem Tests which are currently being planned and evaluated are given in Tables 1.5-2 and 1.5-3. The tests in Table 1.5-2 are a mixture of design verification and the reliability testing, both of which will contribute to a reliability assessment.

Component Testing

The purpose of the components test program will be to verify the qualitative reliability of critical primary and secondary system components, and to provide early detection of potential design and manufacturing deficiencies. The data will primarily provide verification that common-mode failures will not be introduced via design deficiencies at the component level. The components test program will also provide early feedback to the design process and will influence the specific direction of the subsystems testing. The selection of hardware to be tested and the test duration will be based on importance to successful scram operation. The required test hardware will be procured and testing will be initiated early in the program. Examples of components planned for extensive tests include: the control rod drive mechanisms, driveline and control assemblies, latches, and solenoid valves.

Refurbished parts will be used as required to extend the test life of a single test component and to ensure relevant reliability data for the reactor system where feasible. The same philosophy will be used in refurbishing individual subsystems, at each level of testing.

The component testing program will include the following:

1. Primary Shutdown System

   a. Control Rod Drive Mechanism testing in argon or air with typical temperature to determine probability of failure of rotor to release the lead screw.

b. Driveline/Control Assembly testing in sodium at prototypic temperatures with various misalignments and bowed ducts to determine drag forces and ability of driveline to meet scram performance requirements.

c. Impact testing of Control Assembly ducts after exposure to radiation and sodium environments in EBR-II.

d. Control Assembly testing in water to determine drag forces, vibration effects, and flow splits as a function of assembly bow with corresponding effects of bow in scram performance.

e. Accelerated life tests and cyclic testing of control rod system scram breakers to determine reliability problem areas under prototypic operating conditions.

2. Secondary Shutdown System

a. Testing of scram valves, cylinders, coil cords and latches to determine the reliability problem areas under prototypic operating conditions.

b. Testing of reference design dampers in sodium and water to verify satisfactory performance of dampers.

9

Additional items will be included in the above if necessary as the testing and analysis programs progress.

Subsystem Testing

Sybsystem testing will be directed toward obtaining significant amounts of pertinent reliability data related to the operation of the various components or subassemblies combined to form a subsystem. The need to reproduce interaction effects such as misalignment, bowing, fully prototypic sodium flow environments, and mechanical and electrical system interactions will be evaluated so that common-mode failure information can be acquired during this phase. The results of the initial reliability assessment and component test program will be utilized in subsystem test planning phases.

The subsystem testing program will include the following:

1. Primary System

   a. Testing of control rod drive mechanisms, drivelines, control assemblies, controllers, electrical interfaces and the primary rod drive power system at prototypic sodium flow, pressure and temperature conditions. Testing includes determination of effects of misalignment, number of scrams, dwell time, travel and start-stop cycles on scram performance.

   b. Testing of the sensors, signal conditioners, comparators, logic train, couplers and scram breakers under thermal cycling and long hold conditions on the performance of the equipment.

2. Secondary System

   a. Testing of integrated mechanical subsystems including drive, latch actuator, driveline latch, and control assemblies under prototypic sodium flow, pressure and temperature conditions utilizing straight and bowed ducts. Testing includes thermal cycling, long holds and life cycling testing.

Additional items will be included in the above if necessary as the testing and analysis programs progress.

Systems Test

The primary purpose of the full systems test phase will be to implement a disciplined search for common mode failure potential in all phases of operation of the Primary and Secondary Shutdown Systems that can be simulated without an actual in-reactor radiation environment. Another objective is to provide a continuation of data accumulation already initiated in the subsystem tests. Output from the system tests will also include trend data which can be compared with similar data obtained from CRBR operation, and can be used to provide early warning of deviations which could lead to system malfunction since the systems test is designed to operate in advance of CRBRP with a 2-3 year lead time. In addition, the testing and improvement of installation, checkout, maintenance, surveillance and operating procedures will be possible. The number and types of control rod components, interface requirements, degree of simulation and duty cycles to be imposed will be factored into the facility design. (See Appendix C)

CRBR Plant Testing

Pre-operational shutdown reliability testing will be performed on all installed components and subsystems after initial installation but before fuel loading. After fuel loading additional testing will be performed to verify component operation with regard to shutdown capability. A program will be developed for testing following initial startup including shutdown, refueling and startup operations with consideration to items such as component test interval, to assure maximum shutdown system reliability. A procedure will be established for collection and rapid disemination of data regarding incidents or problems effecting shutdown reliability encountered during plant operation or testing.

III. Data Collection

Data base development is essential as a source of dependable input for reliability assessment. Existing data forms the basis upon which design, development and testing decisions are made.

A comprehensive program has been initiated for the collection of reliability data from programs other than CRBRP; for the establishment of a CRBRP data bank; for the collection of abnormal operating experience and maintenance problems from all types of reactors.

Computer codes will be adapted or developed for the storage and selective retrieval data from both the CRBRP and other applicable programs such as:

a. Shutdown system operating and development/reliability experience directly applicable to CRBRP.

b. Generic failure rate data

c. Vendor experience

d. FTR experience

e. NASA and military reliability program experience

IV. Design

This program is intended to ensure that reliability requirements are incorporated in the design process, which include the results of qualitative and quantitative analyses and of test programs into the design. The elements of this part of the program are:

## Component Design

Preliminary component design is based on state of the art, FFTF experience and output from preliminary reliability analyses such as FMEA and FTA. Follow-up design revisions will be made to incorporate the results of detailed reliability analyses and associated testing efforts as required.

## Impact Upon System Design Descriptions (SDD) and Engineering Specifications (E-Specs)

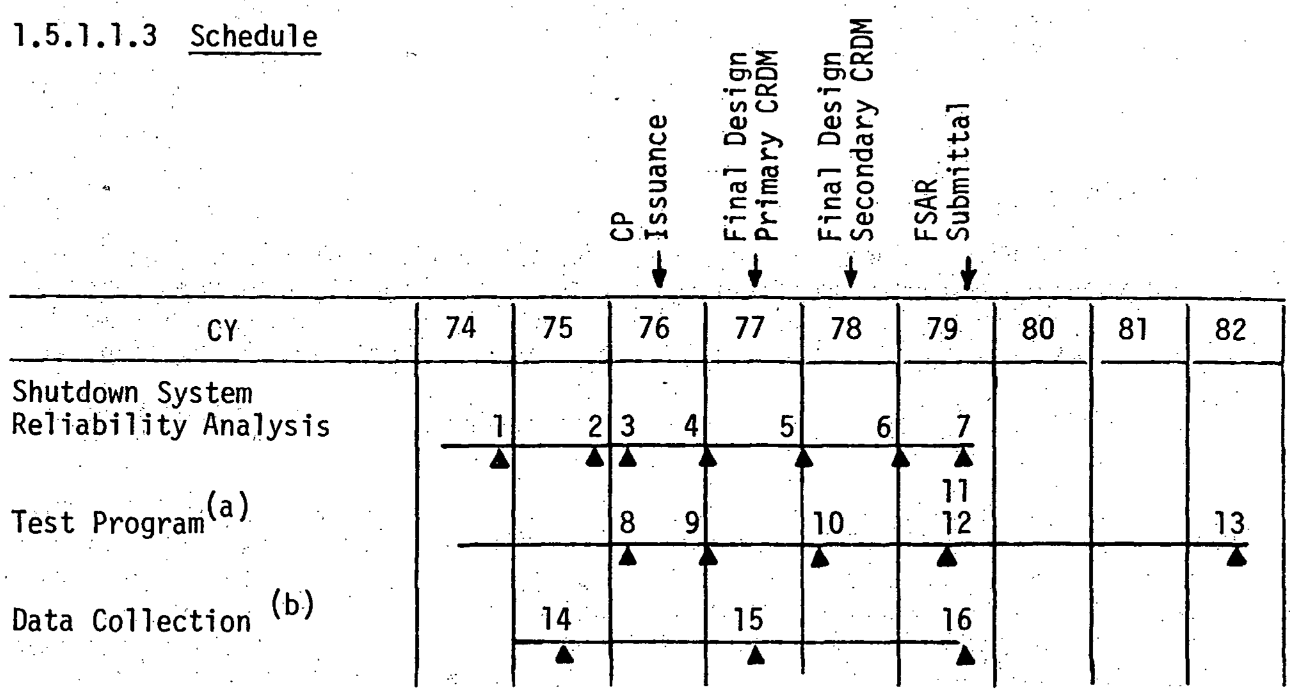
A significant program activity is the review of all CRBRP SDD's for the purpose of identifying those subsystems and components which will affect shutdown reliability. Initial reliability goals have been established for each shutdown system and the initial allocation of these goals to the electrical and mechanical portions of each shutdown system have been defined. On the basis of these goals reliability requirements for the systems are established. Reliability allocations within the various subsystems will be performed by the responsible design groups. Where appropriate, these reliability requirements are being included in system SDD's and in component equipment specifications by procedures established within the CRBR Project.

## Monitoring, Operating and Maintenance Procedures

A review of the shutdown systems and plant maintenance procedures will be performed to identify those areas that could have an effect on shutdown reliability. Procedures will be modified to minimize the potential for maintenance errors and preliminary preventive maintenance tasks will be identified. This information will then be utilized to prepare procedures for the system tests facility as part of the overall system verification plan. The system test facility will in turn provide feedback concerning the adequacy of maintenance procedures which will result in an update of the CRBRP procedures.

The second portion of this task will be the identification of potential trend data that can be used to initiate preventive maintenance for improvement of shutdown reliability. Surveillance procedures will be updated utilizing test data from subsystem and system tests.

#### 1.5.1.1.3 Schedule




1. Initial reliability assessment

2,3,4,5,6. Updated reliability assessments

7. Reliability analysis for FSAR

8. Initiate component tests

9. Initiate subsystem tests

10. Component tests complete

11. Subsystem tests complete

12. Initiate system tests

13. System tests complete

14. Initiate data system operation

15. Update data system operation

16. Data collection for FSAR. Activity continues.

   a) FFTF and CRBR Test Programs continue after CRBR startup

   b) Irradiation tests in FFTF will continue beyond CY-79.

Note: Detailed Schedule given in Appendix C.

#### 1.5.1.1.4 Criteria of Success

The reliability program will assure through testing and analysis that the probability of loss of coolable geometry is $10^{-6}$/reactor year or less.

#### 1.5.1.1.5 Fallback Positions

In the unlikely event that the shutdown reliability program fails to confirm that the overall reliability of the complete CRBRP shutdown system is sufficient to preclude failure to scram concurrently with a plant transient as a design basis accident, components of the shutdown system will be redesigned as necessary to improve reliability. Should redesign be ineffective or inappropriate, the fallback position is the adoption of core disruptive accidents into the design of the plant.

### 1.5.1.2 Shutdown Heat Removal Systems and Structural Reliability Program

#### 1.5.1.2.1 Purpose

The purpose of the Shutdown Heat Removal Systems and Structural Reliability Program is to confirm, in combination with the Shutdown Systems Reliability Program, that loss of in-place core coolable geometry will be of sufficiently low probability under plant transients that a core disruptive accident is not an appropriate design basis event. The combined programs will include all plant hardware and events the failures of which are potential initiators of a core disruptive accident. The Shutdown Heat Removal Systems and Structural Reliability Program will specifically include those components whose failure would lead to lack of adequate core cooling following shutdown.

#### 1.5.1.2.2 Program

This program is in place and is undertaken jointly by Westinghouse and General Electric Company.

The hardware which is covered in the program will include primarily components directly involved in post-shutdown energy removal, both reactor decay heat and stored sensible heat. The equipment covered will be the primary and intermediate sodium loop components, the overflow heat removal system, the steam/water system components (including the steam generator auxiliary heat removal system), and the reactor vessel with its support structure.

The program includes development of reliability methods specifically needed for this program, qualitative and quantitative analysis of the components and systems of concern, failure related data collection, and testing to establish an acceptably low probability of failure which could lead to a core disruptive accident. A more complete and detailed presentation of this program is given in Appendix C (Section C3.2).

Methods Development

Although the lead effort in producing a reliability methods and procedures manual is being undertaken under the Shutdown Systems Reliability Program (Section 1.5.1.1), methods and procedures uniquely required for the reliability analysis of heat removal systems components will be developed under this program. One major procedure will be one to guide reliability analysis of components designed and fabricated to ASME Boiler and Pressure Vessel Code Section III and related high temperature Code Cases. The intent is to make maximum utilization of the existing Code related analytic results for reliability assessment and to improve the failure assessment capability of high temperature structural analysis. The extensive Code-related structural analysis, which will necessarily be accomplished for all Coded components, be utilized to the maximum extent possible in the reliability assessment. A second procedure will be development of a plan of analytic attack in assessing heat exchanger reliability, particularly as regards leak causes and effects. Structural, thermal, and chemical effects will be considered. Further specific methodology may be required for pumps and valves, which play key roles in the dependable operation of the heat removal systems. The results of these method development activities will be incorporated into the overall Reliability Manual being developed under the Shutdown Systems Reliability Program.

Reliability Analyses

An early analysis task is a combination of preliminary component and system reliability estimates and appropriate apportionment of the overall reliability goal among the components and systems covered in the program. The reliability assessment will be upgraded as analytic and test results improve the quality of the assessment and provide feedback to the design. The apportionments likewise will be periodically modified in a dynamic, iterative process. The reliability assessments will be based on:

1. Plant duty cycle description

2. Estimated lifetime of each component of interest

3. Frequencies of external events affecting requirements and performance of heat removal

4. Interface with other plant systems which could affect requirements and performance of heat removal systems components.

5. Man-machine interfaces (operation and maintenance personnel and their relationship to plant equipment).

Other appropriate bases will be included as they are identified in the continuing program.

Detailed qualitative analyses will consist of Failure Mode and Effects Analysis (FMEA), Fault Tree Analysis (FTA), and Common Mode Failure Analysis (CMFA). The FMEA will be completed by component design engineers, monitored and assisted by the reliability specialists. Reliability engineers will participate to a greater extent in the FTA and CMFA, although design engineers will continue direct involvement because of their unique understanding of the component and system designs.

Detailed reliability assessments at several levels of refinement will be performed with emphasis on those items which tend to dictate overall system reliability. The selection of items for detailed study will be decided jointly between design and reliability personnel based on the qualitative analyses just discussed, the conventional thermal and structural analyses which have been performed on the component or system, and available failure data generated within the program or available from other sources. Detailed reliability analyses will be performed using existing methods and those developed within this program and the Shutdown Systems Reliability Program. Example methods for component reliability evaluation are the "stress-strength overlap method" (Ref. 2) and "functional structural reliability analysis," (Ref. 3).

The FMEA, CMFA, and FTA will be applied at the system level to a system model to provide assessments of overall probability of heat removal success, assessments which will be refined as improved input data and the results of more refined analyses become available.

Reliability Data Base Development

Failure data on components of special interest in this program will be collected to complement the main data bank effort (1.5.1.1.2). Example components of special interest are pressure vessels, heat exchangers, pumps, valves, and piping. Data sources to be reviewed include the Liquid Metal Engineering Center (LMEC), Southwest Research Institute (SRI), the Fast Flux Test Facility (FFTF) records to date, LMFBR base development programs, equipment vendor failure data, and the foreign experience. The data bank entries will be prepared according to a format being prepared within the CRBR shutdown reliability programs so that a consistent contribution can be made to the main LMFBR failure data bank.

In addition to component failure data, basic data required for reliability analysis will be collected. This includes such things as statistical descriptions of material properties and test data which assists in the selection of a failure criterion for a given analysis. Two sources of such data are statistical properties collected under the auspices of the Rome Air Development Center (RADC) and the significant mechanical properties data being produced under the program for LMFBR component structural design methods which is under Oak Ridge National Laboratory (ORNL) leadership.

An important additional data base development task is to identify the data most needed for the reliability analysis but which is not available. Recommendations for the most critically needed data will be provided as a source of guidance both to test programs being accomplished within the reliability programs and to other LMFBR programs like the ORNL structural program referred to earlier.

## Test Program

The test program is planned according to two criteria, namely, the test articles should represent components whose proper performance is critical to post-shutdown heat removal reliability and the selected test items should be those for which an established failure data base does not exist. Much data which is of direct interest in the reliability assessment is collected in development testing performed outside the reliability program. Some important examples are steam generator failure data as acquired within the steam generator development program, the large bellows performance as measured in the IHX development test program, and the reactor vessel outlet plenum mixing test which will assure adequate cooling capability of the overflow heat removal system. It has been determined during the analysis to date that data on steam generator tube leakage and on leak detection equipment reliability are of so much importance in the heat removal reliability assessment that testing planned for both these areas in the existing steam generator development program should be supplemented. Consequently, testing is planned within the shutdown heat removal reliability program itself to provide additional data on steam generator leakage and leak detection dependability.

The test program as currently planned is described below. The list of tests may change as a result of more thorough analysis and identification of more critical testing needs.

1. Steam Generator Tube Leakage - In recognition of the importance of steam generator failure rates in the shutdown heat removal reliability analysis, testing will be performed to supplement the tube leakage data already planned in the Steam Generator Testing portion, Task C, of the Steam Generator Development Program.

2. Leak Detection Equipment - Because of the importance of dependable steam generator leak detection indication, this testing will supplement the Sodium-Water Reaction Detection portion, Task B, of the Steam Generator Development Program.

3. Pressure Relief Rupture Discs - The disc assemblies will be tested through cyclic loading/unloading and transient conditions. Also, earthquake loading response will be determined. Particular emphasis during testing will be to identify common mode failures, since this is the most severe situation and would result in draining of the IHTS loops and termination of SGAHRS capability. Finally, the variance about the nominal burst pressure will be measured.

4. Pressure Relief Valves - These valves are for use in relieving superheater and steam drum pressure buildup. The prime interests are variance of actuation pressure with respect to set point pressure, variance of closing pressure with respect to the design value, and reseating behavior of the closed valve. If these valves fail to reseat, the SGAHRS protected water storage will be drained, resulting in a steam generator dryout.

5. SGAHRS Instrumentation and Control - This program would subject the SGAHRS I&C to cyclic conditions representative of the actual application and determine the reliability of performance with particular attention paid to detecting potential common mode failures and variances from nominal performance.

6. SGAHRS Protected Air Cooled Condenser Louver Actuation - The louver actuation mechanism must perform reliably to initiate heat rejection. This testing is to determine performance after many cycles under realistic environmental conditions. Of particular importance is identification of common mode failures. Since the louvers are also adjustable for part heat load rejection, the accuracy of the setting control will be measured.

7. Turbine By-Pass Valve - This series of valves must by-pass the steam flow around the turbine to the main condenser. These valves must be cyclic tested at appropriate environmental conditions to determine the reliability of actuation. Also of interest is the controlling accuracy of these valves for partial steam flows.

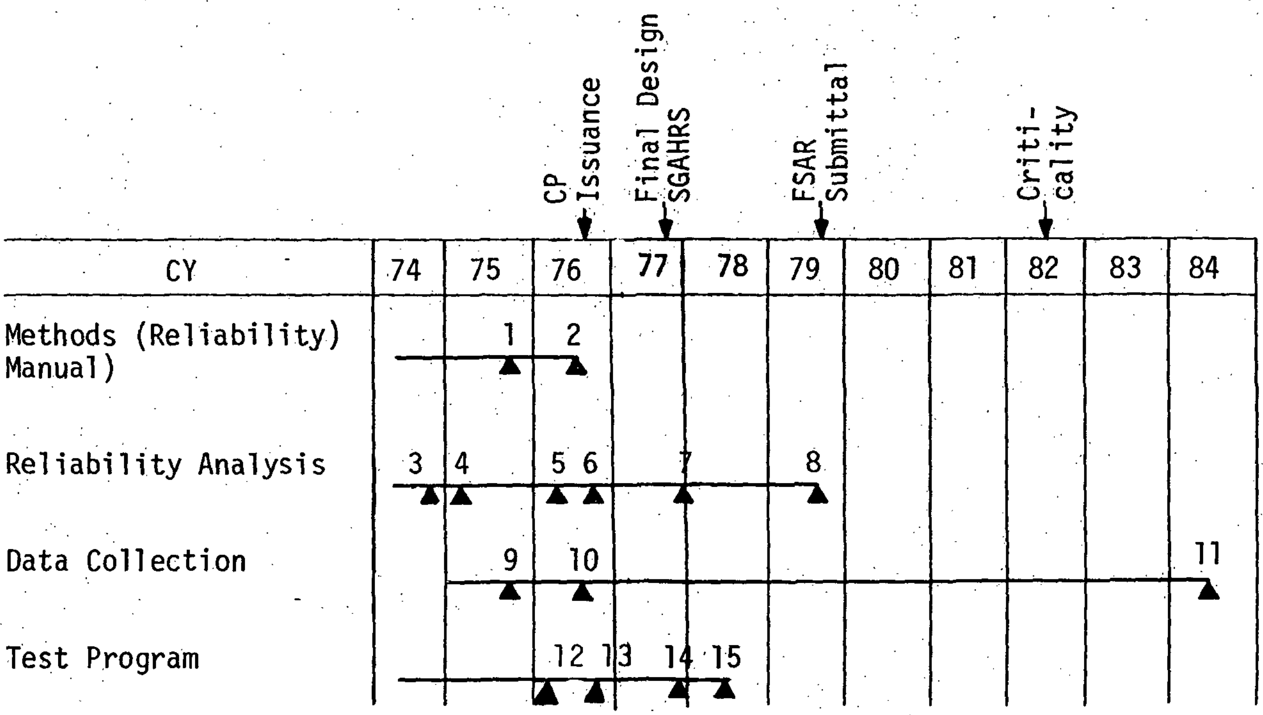
8. SGAHRS Isolation and Control Valves - These valves will be cyclic tested under normal and off design conditions to evaluate potential common mode failures, actuator behavior, and response to abnormal loadings.

9. Pump Bearing/Pony Motor - These components will be subjected to repeated stop/startup, simulating behavior of the pump after scram. The test is necessary to determine the reliability of establishing pony motor flow in the primary and intermediate heat transport loops.

10. Welded HTS Piping - Testing to failure is planned to support prediction of the failure of seam-welded HTS piping.

11. Critical Component Nozzle - The test article will be a nozzle which duplicates the nozzle in any heat removal system component which is judged the most critical in the plant (based on a combination of importance and structural margins). Testing to failure is planned.

1.5.1.2.3 Schedule




1. Draft of reliability procedures for heat transport components including preliminary assessment of reliability implications of ASME Section III

2. Final document on procedures and Section III reliability implications.

3. Preliminary heat removal systems reliability estimate

4,5. Preliminary and final definitions of reliability goal and apportionment.

6,7,8. Two updates and final system reliability/availability assessments

9,10. Preliminary and final recommendations for critical required reliability data.

11. Report on CRBRP operational data

12. Completion of test planning

13,14,15. Completion of initial, critical and final tests

41

### 1.5.1.1 Secondary Control Rod System Test

#### 41 1.5.1.1.1 Purpose

The purpose of the Secondary Control Rod System (SCRS) test program
is to demonstrate that the new design will meet plant performance requirements.
41 A prototype system test program will be performed to verify that the system
meets such requirements. The data obtained from the tests will confirm the
operational characteristics of the selected hardware and the design.

One component that is particularly important to the safety function of
39 the SCRS is the scram release latch mechanism. The latch is an active mechanism
which is required to function reliably throughout its design life. Because of
the safety implications associated with the latch, it will be extensively tested
both as a separate subsystem and as part of the complete shutdown system. Other
component tests of selected items which affect the operation of the SCRS and
contribute to the final mechanical design of the system are described in Section
41 4.2.3.4.1.2 and Appendix C, Section C.5.2.

#### 41 1.5.1.1.2 Program & Schedule

This program is conducted by General Electric Company at the San Jose
facility. A description of the testing program and schedules is provided in
41 Section C.5.2 of Appendix C.



Amend. 41
Oct. 1977

41 THIS PAGE INTENTIONALLY BLANK

### 1.5.1.1.4 Criteria of Success

The latch Component Test in Sodium has been completed. The Inconel gripper/Inconel coupling head performed in accord with specifications for 4 times the required number of cycles. The components are considered acceptable.

The prototype system test will confirm the capability to function reliably throughout its design life.

### 1.5.1.2 Direct Heat Removal

### 1.5.1.2.1 Purpose

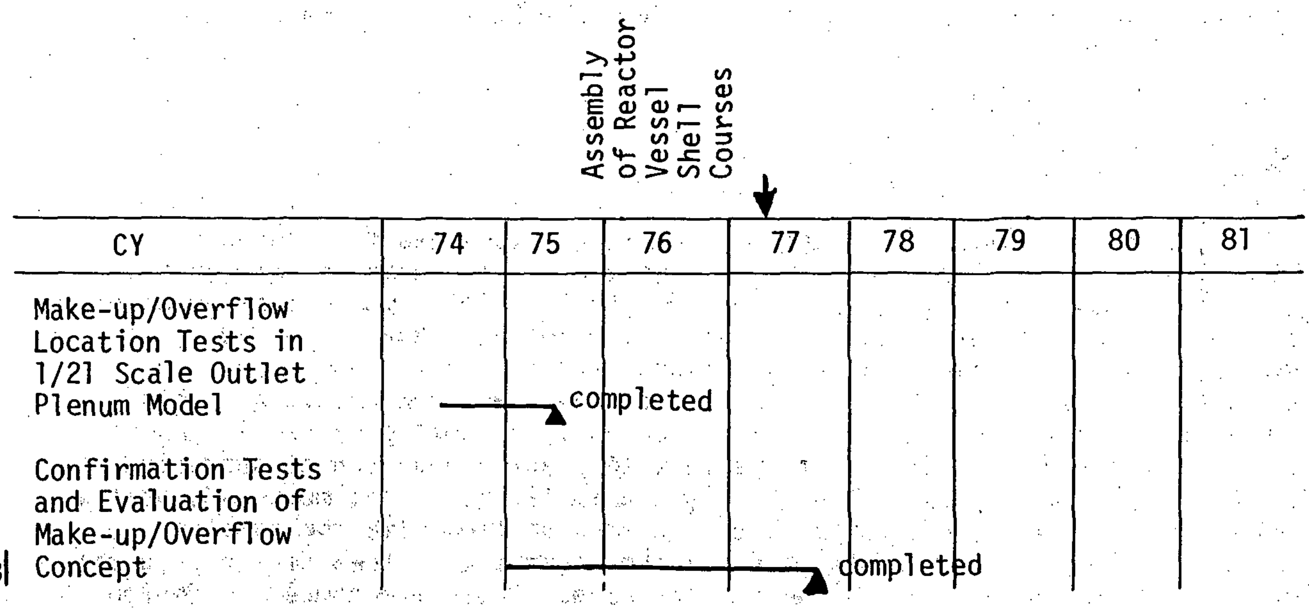
The Direct Heat Removal (DHR) service provides a supplementary means for removing long term decay heat for the remote case when none of the steam generator decay heat removal paths are available. This system must be able to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant boundary are not exceeded. This supplementary decay heat removal is performed by a cooling system incorporated into the sodium make-up/overflow system with plant conditions as specified in Chapter 5 (Section 5.6.2). The principal uncertainty of the make-up/overflow cooling system is short circuiting of the make-up flow with the reactor vessel. Short circuiting would occur if the inlet fluid flows directly to the overflow line without cooling the reactor core. Tests are needed to design the system to ensure short circuiting does not compromise core cooling.

### 1.5.1.2.2 Program

This program is conducted by Hanford Engineering Development Laboratory at the Integral Reactor Flow Model Test Facility. A 1/21 scale outlet plenum model test was used initially to test promising OHR candidate designs for the outlet plenum. Of concern is the location of the make-up and overflow nozzles to reduce short circuiting of make-up flow. This test will conceptually determine overflow nozzle locations.

Confirmation testing of the selected make-up and overflow concept was
53 successfully completed in the Phase I testing of the Integral Reactor Flow Model.

1.5.1.2.3 Schedule




1.5.1.2.4 Success Criteria

The tests demonstrated that the distribution of the make-up flow
53 in the outlet plenum was adequate to assure that the DHR service will function to remove decay heat following a reactor shutdown. This system must be capable of removing heat at a rate such that specified acceptable fuel design limits and design conditions of the reactor coolant boundary are not exceeded.

53 1.5.1.2.5 Results of Tests

53 The 1/21 scale outlet plenum and the HEDL IRFM model tests have been completed. The results show that short circuiting of make-up flow to the over-
53 flow nozzle is limited to approximately 5%. The test and results are discussed in more detail in Response 001.580.

57 1.5.1.3 Blanket Failure Threshold

1.5.1.3.1 Purpose

The CRBRP is being designed to operate with a limited number of failed fuel and blanket rods. This requires demonstration that operation with failed
57 blanket rods exposed to sodium does not result in rod-to-rod failure propagation. This program investigates the potential of blanket material/ sodium reaction to cause swelling, flow blockages, and rod-to-rod failure
57 propagation in blanket assemblies.



Amend. 57
Nov. 1980

57

It needs to be further demonstrated that a relatively small local flow
57 blockage will not lead to failure in a substantial number of blanket
rods. Tests performed for core fuel rod bundle geometries (Ref. 4), indicated
that such propagation is highly unlikely. However, the geometry, thermal
57 conditions and flow conditions in the blanket assemblies are sufficiently
different from that in core fuel assemblies to warrant an independent evalua-
57 tion of flow blockage effects. The variation in coolant flow rates to
blanket assemblies cover a wide flow velocity range from laminar through
transition to turbulent modes of flow. At low flow rates and with steep
temperature gradients across assemblies, buoyancy effects could become a
significant contributor to the temperature and flow distribution within the
blanket assembly.

Efforts are therefore planned to: (1) evaluate the failed
57 blanket rod performance; specifically to verify the performance of
blanket rods with failed cladding and blanket material exposed to sodium,
57 (2) to verify the effect of the high oxygen-to-metal ratio and density on the
probabilities of blanket material sodium reaction, swelling, and flow
57 blockages, (3) to evaluate the cooling rate behind a solid or porous local
flow blockage with tightly arranged rod bundles with pitch to diameter
ratios <1.1.

41 1.5.1.3.2 Program

This program will be conducted by Westinghouse at its ARD facility.

1) Failed Blanket Rod Tests

53

The scope of the blanket rod RBCB (Run Beyond Cladding Breach) portion of the LMFBR Reference Fuel Irradiation Program Includes the design, irradiation, and examination of EBR-II tests and/or TREAT experiments. The scope includes the acquisition, evaluation, analysis and reporting of results to:

- demonstrate performance capability of breached blanket rods at beginning, middle, and end-of-life,
- test the capability to accommodate design transients at end-of-life,
- provide insight into the effect of reactor shutdown on breached blanket rods, and
- establish a theoretical understanding of post-breach bahavior through predictive iterations based on and supported by experiments.

Information developed from the RBCB task will support the following specific areas:

1. Plutonium contamination of sodium.
2. Allowable operating time after breach.
3. Operating procedures for reactor shutdown and restart. 57

Amend. 57
Nov. 1980

4. Delayed neutron detector values for removal of breached rods.

5. Operational transient margins of breached rods.

2) O/M Ratio and Density Effects on Blanket Fuel-Na Reaction

These effects were evaluated as part of the test program on fuel-sodium reaction phenomena conducted by General Electric Company. The results of this program are given in Reference 12.

3) Blanket Assembly Local Flow Blockage Evaluation

The effect of a local flow blockage in a blanket rod bundle have been evaluated with a water flow test and will be documented in a future
57 report.

41 1.5.1.3.3 Schedule

Complete Blanket Final Design Evaluation (milestone at 83)

| 53 41 | CY 79 | 80 | 81 | 82 | 83 | 84 | 85 | 86 |
|---|---|---|---|---|---|---|---|---|
| 53 1. Failed Blanket Rod Tests | | ● — | — | — | — ▲ | | | |
| 2. O/M Ratio and Fuel Density Effects on Blanket Fuel-Na Reaction | | (Complete) | | | | | | |
| 53 41 57 3. Blanket Local Flow Blockage Evaluation | | (Complete) | | | | | | |

41 1.5.1.3.4 Criteria of Success

57 The program will demonstrate that operation with failed blanket rods exposed to sodium does not result in rod-to-rod propagation and that a relatively small local flow blockage will not lead to failure
57 in a substantial number of blanket rods.



Amend. 57
Nov. 1980

### 41 1.5.1.3.5 Fallback Position

57 In the event that operating with failed radial blanket cannot be shown to be satisfactory from a public safety viewpoint, the reactor may be required to shutdown when the blanket material is exposed to the sodium.

### 1.5.1.4 Sodium-Water Reaction Pressure Relief Test

#### 41 1.5.1.4.1 Purpose

The principal concern associated with the large water to sodium leak in steam generators is potential system damage, principally to the IHX by propagation of transient pressure waves through the Intermediate Heat Transport System (IHTS). The objective of the Sodium Water Reaction Pressure Relief Subsystem (SWRPRS) is to relieve pressures from the IHTS and thereby protect the primary coolant boundary from damage in the region of the primary sodium to intermediate sodium heat transfer interface.

61 The approach to design of CRBRP SWRPRS is to assume a conservative design basis
59 water to sodium leak and to use a validated calculational method to predict
pressure loads on the IHX. It is a design requirement that the IHX be able to
withstand the sodium-water reaction pressures without compromising the primary
coolant boundary.

A survey of available existing analytical methods was completed to select the
best method for improvements consistent with CRBRP requirements. The TRANSWRAP
computer program (Ref. 5) was selected for use in the CRBRP analysis. An
improved version of this code was used to establish loads on the IHX for the
reference design IHTS piping and component arrangement and the reference design
SWRPRS. A design basis leak was assumed to consist of an Equivalent Double-
61 Ended Guillotine (EDEG) failure of one steam generator tube followed by the
equivalent of two additional EDEG tube failures. The two additional failures
occur as follows:

- one EDEG failure occurs one second after the initial EDEG failure.

- one additional EDEG failure occurs two seconds after initial EDEG failure

This sequence is superimposed on a system which has been pressurized by an
undetected moderate-sized leak to just below the rupture disk burst pressure.
59 The three tube DEG failure is not intended to represent a realistic event, but
rather it provides a basis for calculating conservatively large pressure loads
for the design of IHX and the pressure relief system. Results of analyses
using this basis are reported in Section 5.5.3.6.

To increase confidence in assuring integrity of the primary coolant boundary even during a large sodium-water reaction, the development program will provide technical information which is not available for inclusion in the PSAR. The safety related objectives of the development program are:

a) to validate the computer program used to predict pressures in the IHX during a postulated sodium water reaction, and

b) to confirm that effects of the design basis leak assumed for determining pressures in the IHX are conservative.

41 ### 1.5.1.4.2 Program

As part of the Steam Generator Development Program, AI has constructed the Large Leak Test Rig (LLTR). The test programs
41 included pulling apart a notched tubular specimen in the sodium filled test article to simulate a DEG failure. A steam/water mixture was forced through
44 41 the burst tube into the sodium. For most tests, surrounding tubes contained stagnant, pressurized steam/water mixtures. In general, the development
44 effort provided technical information regarding the design of pressure relief systems to handle unexpected water-to-sodium leaks.

Measured values of pressure at various locations in the test rig are being compared with calculated pressures obtained using the modified TRANSWRAP computer program to analyze the test rig and test article. It now appears that the computer code predicts values of pressure that are either in agreement with measurements or are conservatively large relative to measured pressures for the test rig and test article. Thus, it appears that the analysis of CRBRP for
41 sodium-water reaction pressures using this code are being conservatively
44 accomplished. This conclusion is still under review and evaluation and therefore subject to adjustment as the remaining test data are examined.

Examination of the test article following intentional bursting of
44 a single tube gives some indication of the nature and extent of damage propagation to other tubes. It is expected that the tests will demonstrate
47 that the calculated loadings from sodium-water reactions are conservative.

41 ### 1.5.1.4.3 Schedule

44 41

| | CY 73 | 74 | 75 | 76 | 77 | 78 | 79 | 80 | 81 |
|---|---|---|---|---|---|---|---|---|---|
| 47 41 LLTR-Module Steam Generator (MSG) test data available | | | | | | ▲ | | | |
| 41 Modified TRANSWRAP validated by test results | | | | | | | ▲ | | |
| 44 41 Extent of damage in MSG evaluated | | | | | | ▲ | | | |

Amend. 47
Nov. 1978

41 1.5.1.4.4 Criteria of Success

The computer program, used to predict pressures in the IHX during a postulated sodium-water reaction, will be validated by comparing the predicted pressures for the test with the measured values.

The conservativeness of loadings associated with the design basis leak will be confirmed by the determination that pressures produced by a sodium-water reaction associated with observed failures are less severe than that predicted by the analysis for the design basis leak.

41 1.5.1.4.5 Fallback Positions

If the results of the computer program are not verified by experimental results, or if there is an indication that calculated results are not conservative for the CRBRP application, then the computer program will be
41 corrected.

If observed damage propagation indicates that actual leakage could produce pressures larger than pressures associated with the design basis leak (equivalent of seven DEG tube failures), then analyses will be done assuming a design basis leak which is larger than presently assumed. If IHX pressures produce loads which have unacceptable margins relative to failure loads, the sodium-water reaction pressures can be reduced by redesign of SWRPRS or by modifications (pipe routing, attenuation baffles, surge tanks, etc.) to IHTS which would protect the primary coolant boundary.

1.5.2 Information Concerning Margin of Conservatism of Proven Design

1.5.2.1 Pipe Integrity Assessment

1.5.2.1.1 Purpose

The purpose of this program is to confirm that the likelihood of a pipe rupture in CRBRP is extremely low, consistent with the project posi-
41 tion that massive failures of the major incontainment sodium piping can be excluded from the plant design basis.

41 The following tasks were defined to demonstrate the low probability.

(a) Develop a model of fatigue crack growth and evaluate crack growth rates. Determine the critical crack size (the size at which the crack will bulge open under operating stresses).

41

41 (b) Determine by experiment the characteristics and detectability of sodium leakage through small cracks in piping. Determine the rate and characteristics of corrosion of piping due to sodium leakage.

41 (c) Develop a reliable leak detection system which will be capable of detecting small leaks in critical areas of the primary
53 and intermediate HTS.

41

53

1.5.2.1.2 Program, Results and Utilization

The details of the program, results and conclusions drawn from the results are discussed in Reference 2 to Section 1.6.

41

Amend. 53
Jan. 1980

41 Pages 1.5-30 and 1.5-31

Have Been Deleted

41 22

### 1.5.2.2 Failed Fuel Assembly Tests for Accident Conditions

#### 1.5.2.2.1 Purpose

The tests are intended to confirm that a loss of boundary integrity in assemblies adjacent to an accident assembly (an assembly affected by accident) is not expected to occur by thermal loads resulting from a partial meltdown in the accident assembly. Further, even in the unlikely event of a loss of boundary integrity, additional failure propagation in these adjacent assemblies would not necessarily occur. Operating experience of fast reactors and mechanistic analyses of failure propagation phenomena have indicated that rapid failure propagation among a large number of the assembly fuel pins is highly unlikely (Refs. 6, 7). At present, more definitive data and analyses are available to evaluate pin-to-pin failure propagation. This experimental program will determine the margins available in the design to accommodate molten fuel within an assembly without leading to widespread fuel assembly failure propagation.

#### 1.5.2.2.2 Program

This program is in-place at Argonne National Laboratory which is applicable to both CRBRP and FFTF.

In order for assembly-to-assembly propagation to occur, the existence of either one of two conditions must be postulated: (a) pressure pulses generated by energetic molten fuel-cladding-coolant interactions, or (b) thermal loads accompanying gross assembly voiding and subsequent massive cladding and fuel melting. Based on recent progress in the understanding of fuel coolant interactions, an energetic interaction is unlikely in the accident assembly (Ref. 8 & 9). Therefore, a gross assembly meltdown that imposes large thermal loads is a necessary condition for assembly-to-assembly propagation.

The deposition of molten fuel in an accident assembly is postulated in order to predict the response of an adjacent assembly duct wall under severe thermal loading conditions. Heat fluxes which could be expected from the deposited molten fuel are frequently based on the "maximum nonboiling fuel thickness," that is, the radial heat flux transferred to the duct wall is generated in that thickness of fuel which is below the fuel boiling point. By using this boundary condition, out-of-pile experiments will be carried out in two phases:

1. Duct wall behavior test utilizing sodium-cooled hexagonal duct with an electrical heater.

41 2. A 217-rod assembly 70 inches long will be tested with the heat flux imposed directly on the adjacent assembly duct. The elements and wire wraps in the test bundle incorporate an array of thermocouples for data acquisition. Deflections caused by thermal stress will be measured by probes or radiography.

The expected information from this development program will consist of the following:

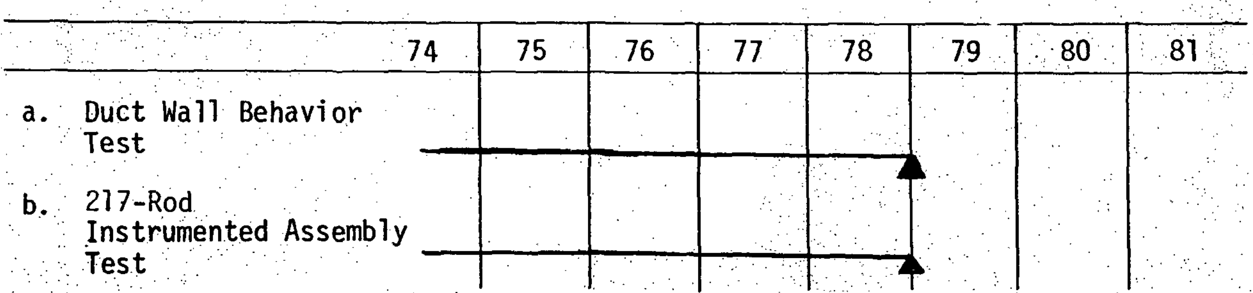
a. Experimental determination of the capability to maintain the normal geometry and flow in the assembly adjacent to the partial meltdown accident assembly.

b. Measurements of the maximum sodium temperature in wall channels having appreciable temperature gradients under the assumed wall heat flux conditions to verify the computer code predictions using lumped nodal representations.

c. Measurements of effects of thermally loading a surface of the duct with and without the heated elements next to the heated surface.

1.5.2.2.3 Schedule

41




41
41

Amend. 41
Oct. 1977

1.5.2.2.4 Criteria of Success

The results should provide experimental verification of the capability to maintain the integrity of the boundary in assemblies adjacent to an accident assembly and/or that determination of the maximum heat flux limit that will not initiate failure propagation in the adjacent assemblies.

1.5.2.2.5 Fallback Position

Even if the results from the experiments and the further analyses conclude that the partial meltdown could propagate to other fuel assemblies, it should be noted that the design has incorporated features to prevent initiation of meltdown. These include a highly reliable shutdown system to terminate all transients without fuel failure, fuel assembly inlet features to preclude blockage and fuel assembly mechanical interlocks to preclude placement of assemblies in positions that could result in undercooling. Therefore, analyses will confirm that the initiating events leading to fuel assembly melting are of such low probability that they need not be the basis for design. If this cannot be demonstrated with the existing design, then those aspects of the design which resulted in the probability being unacceptably high will be reviewed and modified so as to eliminate the initiator.

41

1.5.2.3 Reactor Thermal and Hydraulic Tests

1.5.2.3.1 Purpose

The purpose of the specified thermal and hydraulic tests is to verify the margin of safety in the areas of the behavior of bubbles and the mobility of particles introduced in the inlet plenum. These evaluations have direct impact on the safety feature of inlet plenum design. Although the design of the CRBR fuel assembly in these areas is similar to that of the FFTF driver fuel assembly, differences in the configuration and the arrangement of the lower internals require additional tests with the inlet plenum.

53

53

The core support structure incorporates a feature for preventing possible
41 damaging flow blockage at the entrance to the inlet module flow ports.
This feature is provided by a debris barrier which forms a secondary
inlet plenum with separate, additional flow ports to those providing the
41 normal inlet flow path. A test program was conducted which demonstrated
the capability of this concept to retain structural adequacy and satisfactory flow distribution.

1.5.2.3.2 Program

53 This program consists of three subtasks. Tasks A, B, and C will be conducted by HEDL at its Inlet Plenum Feature Model Test Facility.

53 A. Inlet Plenum Bubble Dispersion Test

A full 360$^{o}$ model of 0.248 scale which contained all hydrodynamically important wetted surfaces was used in this test. This model of CRBRP inlet plenum was tested with water at prototypic velocities.

Test results showed that a gas bubble entering the inlet plenum
from one pipe breaks up before entering the inlet modules, and the bubbles
41 entering the modules are fairly evenly distributed.

53 B. Inlet Module Blockage Prevention Test

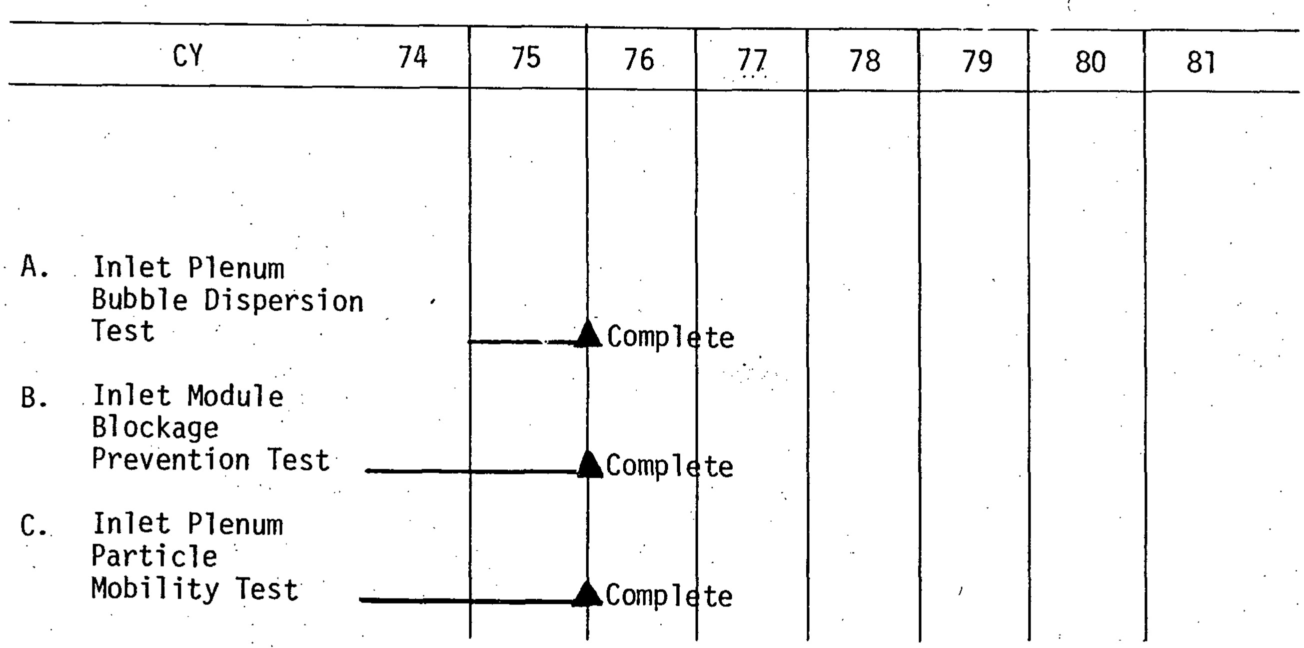
41 A test program was conducted in the CRBRP inlet plenum feature model (IPFM) in (B) to demonstrate the capability to mitigate the consequence of possible flow blockage at the inlet module flow ports. The verification of structural adequacy and a satisfactory flow distribution was demonstrated. The secondary inlet plenum was checked for flow distribution with seven central liners having all three of the primary inlet flow ports fully blocked off to the inlet flow plenum. The results of the test show that the auxiliary ports effectively mitigate the effects of blockage: when seven lines were blocked
41 below the debris barrier, flow was reduced by less than 5%.

53 C. Inlet Plenum Particle Mobility Test

41 The purpose of this test was to provide hydraulic data necessary to determine if debris that may be inadvertently introduced into the reactor vessel inlet plenum area, but not removed during the initial filtration, will remain on the bottom of the reactor vessel or will be lifted into the core
41 53 inlet region.

1.5.2.3.3 Schedule

41




53
53
53
53
41

1.5.2.3.4 Criteria of Success

53

Inlet Plenum Bubble Dispersion Test will confirm the capability of uniform flow distribution and breaking up of the bubbles to preclude a reactivity effect.

Inlet Module Blockage Test will indicate: (1) no damaging vibration of the anti-blockage module liners and (2) the flow distribution is satisfactory in the inlet module.

The data obtained from the Inlet Plenum Particle Mobility Test on the particle transport into the core inlet region and through the core were factored into determining the requirements for in-core filtering of the primary system sodium so that the particles that could be transported into the
41 core are removed from the sodium prior to initial fuel loading.

1.5.2.3.5 Test Results and Fallback Positions

A. The test confirmed bubble dispersal.

B. The Module Blockage Test indicated satisfactory results.

C. The Inlet Plenum Particle Mobility Test indicated that debris not removed during the initial filtration will
41 remain in the bottom of the reactor vessel.

53

### 1.5.2.4 Core Restraint System Tests

#### 1.5.2.4.1 Purpose

In order to provide both axial and radial support for the reactor assemblies, the core restraint system will: (1) maintain fuel, control, blanket and removable shield assemblies in predictable and reproducible positions; (2) assure no damage to the assemblies during refueling, and (3) prevent excessive operating loads from occurring. The key safety related feature is the ability to limit reactivity insertions arising from lateral
53 assembly motion to an acceptable level through control of residual load plane gaps throughout the core.

Variables affecting the core restraint system are the radiation induced
59 53 materials effects of swelling and creep, load pad dimensional and friction characteristics. Irradiation swelling and creep in conjunction with transverse temperature gradients in the core assembly ducts cause bowing deformations of the core assemblies during life. The restraint designs of fast reactors now in operation are among those concepts developed before the recognition of irradiation induced swelling and creep effects in core components. Prototype Fast Reactor and FFTF are among the transitional designs based on recognition of the problem and these systems accommodate design uncertainties due to limited characterization of materials effects. The CRBRP design, effectively the third generation, draws on the design and development experience of the previous two generations. To reduce sensitivity to mechanical
53 effects uncertainties, the following development program is in place.

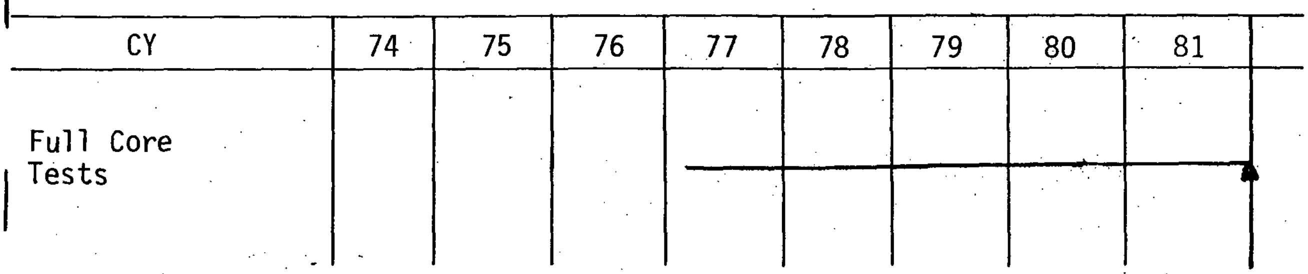
#### 1.5.2.4.2 Program

41 This program has been undertaken by Westinghouse at its ARD facility.
59 The performance of the CRBRP core restraint systems is being simulated in a full scale mechanical test facility. The facility provides the capability of
53 testing a full core array of simulated reactor assemblies. This three dimensional test effort will provide qualitative output for analytical studies, as well as quantitative core restraint system test data.

To provide data to the CRBRP in a timely manner tests consisting of a full core mock-up including fuel, blanket and two rows of removable shielding will be carried out. This will be accomplished through the use of simulated fuel assemblies designed to provide nominal duct bending stiffness,
53 load pad compliance, inlet nozzle clearance and contact surface frictional characteristics. This effort will consist of core compaction tests with bowed core assemblies and prototypical load pad friction during which duct bowing patterns based on analytical predictions for chosen material conditions will be simulated. Inter-assembly loading patterns and geometries will be determined by using multiple "instrumented" load measurement ducts. The compaction characteristics of the core will be determined for core restraint
53 system evaluations.

Amend. 59
Dec. 1980

1.5.2.4.3 Schedule




1.5.2.4.4 Criteria of Success

The test effort is to provide core compaction characterization data and provide a basis for determination of residual load plane gaps within the core under prototypic loading conditions. This will be used to evaluate load
42 41 plane gap related reactivity insertion in the core. The criterion of success
53 is the limitation of reactivity insertions arising from assembly lateral motions to an acceptable level.

1.5.2.4.5 Fallback Position

If the predicted reactivity insertion based on these experimental data is not acceptable, an adjustment of gaps in the load planes will be
53 made.

1.5.2.5 Critical Experiments for Reactivity Coefficients, Control Rod Worth and Fuel Assembly Movement

1.5.2.5.1 Purpose

Critical experiments are required to provide information pertaining to the following parameters and components for the safety analysis of the CRBR: the primary and secondary control systems, the reactivity feedback coefficients and the mechanical motion of core and blanket assemblies.

The following information is required for the determination of the adequacy of the two control systems to meet the safety requirements:

a. The bias factors and associated uncertainties for the control rod worth predictions,

b. The influence of control rod movement on fuel assembly power generation,

c. The rate of reactivity insertion associated with control rod withdrawal,

d. The reactivity effect of control rod materials washout in the sodium coolant.

The following reactivity coefficient information is required to
41 support safety analyses of the CRBRP:

a. The Doppler and sodium voiding reactivity coefficients and associated calculational uncertainties,

b. The detailed spatial distribution of the sodium voiding coefficient.

The following information is needed to support the safety analysis related to mechanical motion effects:

a. Reactivity effects of radial compaction of the core,

b. Reactivity effects of relative axial displacement of the core and control support mechanisms.

1.5.2.5.2 Program

The initial phases of the CRBRP critical mockup experiments program (ZPPR-2, 3, 4, 5, and 6) simulated the homogeneous core configuration. The ZPPR-7 and 8F criticals were pre-enginnering mockups of the heterogeneous CRBRP core configuration. These preliminary series of experiments have been completed and the results have been used to develop design bias factors and uncertainties and to provide validation for the nuclear design methods used in the derivation of the CRBRP PSAR physics characteristics. The Engineering Mockup Critical (EMC), which will closely simulate the final CRBRP core and blanket configuration, will provide final nuclear
41 design bias factors and uncertainties for the FSAR. The experiments are performed by Argonne National Laboratory at Idaho Falls, Idaho. The CRBRP Criticals program is coordinated by General Electric Company in cooperation
51 with Westinghouse Electric Corporation.

The following specific experiments will be performed in support of the safety features and components in CRBRP:

a. The reactivity worth of individual control assemblies,

b. The reactivity worth of control rod banks,

c. The worth vs. insertion profiles for selected control rods,

d. The magnitude of the Doppler reactivity coefficient,



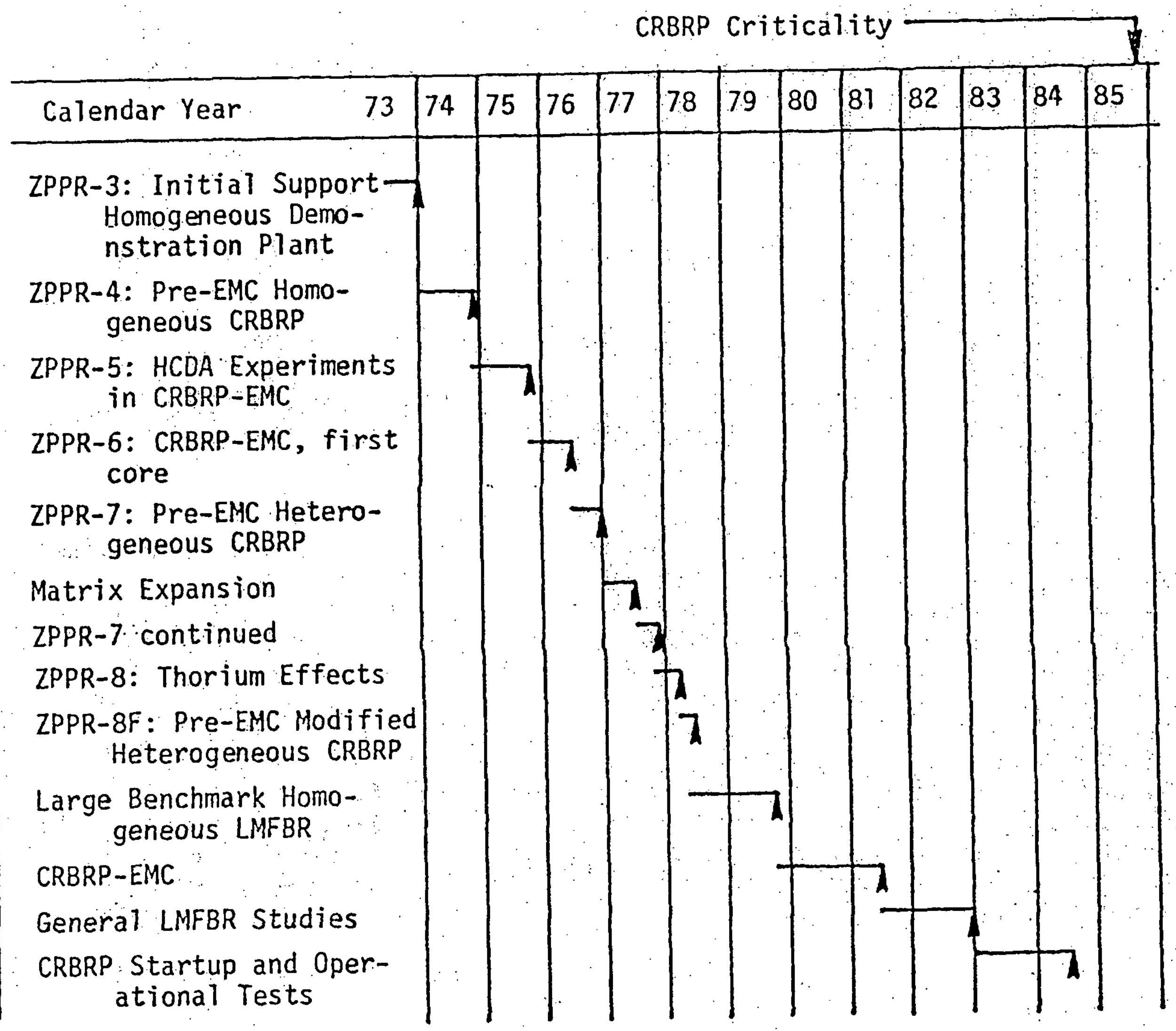
Amend. 51
Sept. 1979

e. The magnitude and spatial distribution of the sodium voiding reactivity effects,

f. Reactivity effects of axial and radial motion of the fuel, blanket and control assemblies.

### 1.5.2.5.3 Schedule

Critical Experiments Schedule




51

### 1.5.2.5.4 Criteria of Success

The planned critical experiments will provide the required information for the adequacy of the control systems, the reactivity coefficients and the mechanical motion effects to support the safety analysis.

Amend. 51
Sept. 1979

#### 1.5.2.5.5 Fallback Position

Flexibility is provided in the on-going critical experiment program described above to provide the best available mockup of the safety-related features and components during the final design phase of the CRBRP. In this way uncertainties in these parameters should be reduced and the confidence level in the final safety analysis increased accordingly. If any of these experiments are not successfully completed, the information used in the safety analysis would have to be based on previous critical mockups that may not be consistent with the final design parameters. If this is the case, larger design margins to fully accommodate these uncertainties would have to be applied to the safety analysis and to the design of the control systems.

### 1.5.2.6 Source Range Flux Monitoring (SRFM) System Experiments

#### 1.5.2.6.1 Purpose

The SRFM System is the instrumentation that will monitor subcritical reactivity during reactor shutdown and refueling operation. Experiments at the Zero Power Plutonium Reactor (ZPPR) facilities will be used to verify the
41 analysis of the ex-vessel low level flux monitoring system and to calibrate the instrumentation prior to use in the CRBRP. The following information is required to meet these objectives:

- The SRFM neutron count rate as a function of subcritical reactivity,
- The SRFM neutron count rate as a function of both reactor geometry and detector environment,
- Performance test of detectors and instrumentation.

#### 1.5.2.6.2 Programs

The SRFM experiments will be performed at ZPPR operated by the Argonne National Laboratory at the National Reactor Test Facility near Idaho Falls, Idaho. The following specific experiments will be performed in support of the analysis of the ex-vessel SRFM.

41

a. Count rate as a function of subcriticality,

b. Count rate as a function of the number of fuel assemblies stored in-vessel,

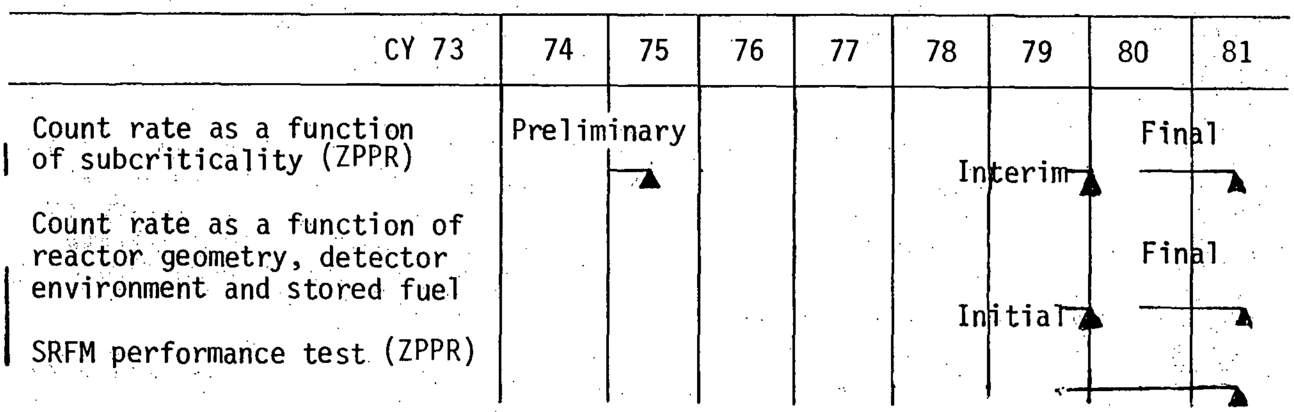
c. Count rate as a function of core and blanket geometry and refueling patterns,

d. Count rate as a function of the detector environment.

Experiments, using detectors and instrumentation which duplicate the actually installed Source Range Flux Monitoring System (SRFMS) equipment, will be performed to verify equipment performance in the expected gamma and neutron fields corresponding to the selected detector locations in CRBRP. These tests may be performed by the FMS equipment supplier at reactor test facilities where suitable gamma environments for the detector can be produced. Information from the foregoing critical experiments will be applied to plan these tests.

1.5.2.6.3 Schedule

41




41

1.5.2.6.4 Success Criteria

The test will demonstrate that the monitoring of subcritical reactivity can be obtained in accordance with the prescribed relationship to guard against inadvertent approach to criticality. The monitoring system should detect gross refueling errors without being overly sensitive to changes in core geometry and refueling patterns. The tests should confirm that measurements at two of the three detector locations should not be affected by fuel assemblies stored in the in-vessel storage.

Tests of the installed FMS equipment will verify its suitability for use in monitoring shutdown, refueling and startup reactor operations.

1.5.2.6.5 Fallback Position

52 SRFM is the essential instrumentation to monitor the subcritical reactivity. In the unlikely event that the in-vessel stored fuel assemblies affect the measurements of the monitors the locations of these monitors and/or shielding arrangement will be modifed. Additionally, if the expected gamma dose rate at the detectors adversely affects the sensitivity of the instrumentation, shielding will be modified to reduce the effect of gamma dose.

### 1.5.2.7 Ex-Vessel Transfer Machine Heat Removal Tests

#### 1.5.2.7.1 Purpose

The CRBRP Ex-Vessel Transfer Machine (EVTM) cooling system must be capable of removing the decay heat generated by the spent fuel assembly during the entire range of fuel handling operations which includes unusual and accident situations. This capability assures the prevention of possible additional fuel pin rupture and release of radioactive fission products. Although adequate heat removal capability is predicted by analysis (Section 9.1.4.3), the multiplicity of possible free convection flow paths within the fuel assembly and Core Component Pot (CCP) introduce considerable uncertainty. This condition is further complicated by the effects of sodium vapor and sodium frost on the emissivities of the CCP and EVTM cold wall. The required technical information will cover the following conditions:

a. Normal operation steady-state heat removal

b. Loss of cooling air

c. Loss of sodium

d. Operational and emergency transient evaluations

e. Reduction in emissivity of CCP and cold wall caused by sodium wetting and frost

These tests will provide data to evaluate the cold wall heat transfer performance when sodium or sodium frost has covered radiant surfaces and thereby establish the EVTM decay heat removal capability under both normal and off-normal conditions.

#### 1.5.2.7.2 Program

This program is in place and is undertaken by HEDL using the Transient Test Loop Facility. The basic concept of this program is to evaluate the cooling of an instrumented and electrically heated 217-rod fuel bundle assembly model contained in a sodium-filled EVTM CCP. Cold wall heat removal capabilities will be determined for normal and off-normal operation by measuring the power input to the fuel bundle assembly at which the maximum fuel cladding temperature reaches 1250°F for normal, forced air convective cooling (1500°F for unlikely and extremely unlikely fault conditions).

The following tests will be performed:

a. Steady-state heat transfer at variable decay power levels under normal design conditions,

b. Steady-state heat transfer with sodium convection being blocked,

c. Steady-state heat transfer under emergency cooling conditions,

d. Steady-state heat transfer with no sodium in the pot,

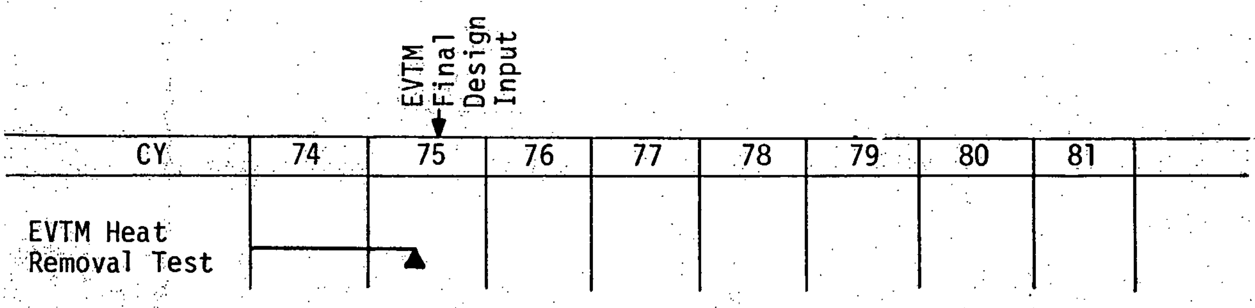
e. Transient thermal performance tests

- Loss of forced cooling air flow
- Normal operation transient
- Simulated CCP failure
- Effect of sodium coating on CCP emissivity

Data to be obtained consist of radial and axial temperature distributions in the fuel bundle, CCP and cold wall, as a function of power input. These data will provide necessary information as stated in Section 1.5.2.7.1.

1.5.2.7.3 <u>Schedule</u>

41




1.5.2.7.4 <u>Criteria of Success</u>

The test results will confirm the EVTM cooling system to be capable of removing the decay heat generated by the spent fuel for the range of fuel handling operations as tested.

1.5.2.7.5 <u>Results of Test</u>

The EVTM Heat Removal test has been completed. As discussed in Section 9.1.4.3, the EVTM cooling system was shown to meet criteria of success.

15

#### 1.5.2.8 Sodium Fires Test Program

##### 1.5.2.8.1 Purpose

The purpose of the sodium fires test program is to verify that plant design features for accommodation of sodium/NaK spills in air-filled cells will result in acceptable cell pressures and structural concrete temperatures. In addition, this test program will be used to demonstrate that the codes used in sodium fire analyses conservatively predict cell accident conditions.

##### 1.5.2.8.2 Programs

The sodium fire experiments have been or will be performed at the Atomics International test facilities in Santa Susana, California. The following small scale tests have been completed:

1) A fast spill (approximately 15 gal/min) of 1000°F sodium onto the fire suppression deck surface

2) A slow spill (approximately 1.5 gal/min) of 1000°F sodium onto the fire suppression deck surface

3) A spray (approximately 15 gal/min) of 1000°F sodium onto the surface of the fire suppression deck

4) A fast spill (approximately 15 gal/min) of 1000°F sodium directly into the catch pan beneath the fire suppression deck

5) A spray (approximately 15 gal/min) of 1000°F sodium, onto the surface of the fire suppression deck, through a walk grating above the deck

6) A spray (approximately 15 gal/min) of 600°F sodium onto the surface of the fire suppression deck, through a walk grating above the deck

The results of the above small scale tests will be documented as the test reports become available. In addition to small tests, a large scale test will be performed using a large-scale model of the CRBRP catch-pan fire suppression deck system to collect spilled sodium under simulated spill conditions. The test facility is designed to accommodate a volume gas as large as 6600 gallons of 1000°F sodium with a sodium discharge flowrate of approximately 70 GPM.

This test will verify the operability of SGB aerosol mitigating dampers by testing under prototypic aerosol conditions.

##### 1.5.2.8.3 Schedule

The small scale tests have been completed. The large scale test is planned to be performed in the last quarter of 1982.

#### 1.5.2.8.4 Success Criteria

The small scale tests successfully demonstrated fire suppression deck design features to ensure drainage capability and fire-suppression effectiveness:

- o No blockage of drain pipes during spill.
- o Post-spill suppression of sodium burning by control of oxygen ingress to sodium pool via oxide plugging of drain pipes and closure of vent lids on vent pipes.
- o No leakage of sodium from catch pan.

The success criteria for the large scale test are that the catch pan shall contain the spilled sodium precluding sodium concrete interactions and that resulting test consequences are enveloped by those calculated with the Project's methodology, and that the aerosol mitigation dampers can reclose as required during the aerosol release.

#### 1.5.2.8.5 Fallback Position

If the effectiveness of the Fire-Suppression Deck/Catch Pan System and damper are not demonstrated, alternative techniques to accommodate design basis liquid metal spill events will be considered and/or prediction of plant design basis accident consequences will be made with alternative methods.

### 1.5.3 References


1. HEDL-SA-771, "Fuel Pin Transient Behavior Technology Applied to Safety Analysis," Presentation to AEC Regulatory Staff, 4th Regulatory Briefing on Safety Technology, November 19-20, 1974.

2. Haugen, E. G., "Probabilistic Approaches to Design," Wiley Book Co., New York (1968).

3. Chang, E. C. C., and C. B. Brown, "Functional Reliability of Structures," Journal of the Franklin Institute, September, 1973.

4. Fontana, M. H., et al, "Effect of Partial Blockages in Simulated LMFBR Fuel Assemblies," Proc. Fast Reactor Safety Meeting CONF-740401-P3, 1139 (1974).

5. Bell, C. R. "TRANSWRAP - A Code for Analyzing the System Effects of Large-Leak Sodium-Water Reactions in LMFBR Steam Generators," Proc. Fast Reactor Safety Meeting, CONF-740401-P1, 124 (1974).

6. Division of Reactor Research and Development, USAEC, RDT Standard M3-7T, "Austenitic Stainless Steel Welded Pipe (ASME SA-358 with Additional Requirements)" November, 1974.

7. Marr, W. W., et al, "Subassembly-to-Subassembly Failure Propagation: Thermal Loading of Adjacent Subassembly," Proc. Fast Reactor Safety Meeting, CONF-740401-P2, 598 (1974).

8. Van Erp, J. B., et al, "An Evaluation of Pin-to-Pin Failure Propagation in LMFBR Fuel Subassemblies," ibid, 615 (1974).

9. Erdman, C. A., et al., "Improvements in Modeling Fuel-Coolant Interactions and Interpretation of the S-11 TREAT Test," ibid, 955 (1974).

60 34 10. Deleted.

41 11. Letter from R. P. Denise (NRC) to L. W. Caffey (ERDA) May 6, 1976.

12. GEFR-00424, UC-79B, "A Physicochemical Model for Predicting Sodium Reaction Swelling in Breached LMFBR Fuel and Breeder elements", R. W. Caputi, M. G.
57 Adamson, and S. K. Evans, March, 1979.

TABLE 1.5-1

FURTHER TECHNICAL INFORMATION REQUIRED

| | PSAR Section | Section Heading and Tasks |
|---|---|---|
| | 1.5 | Introduction |
| | 1.5.1 | Information Concerning the Adequacy of a New Design |
| 41 | 1.5.1.1 | Secondary Control Rod System Test |
| 39 | | |
| | | Latch System Tests |
| | 1.5.1.2 | Direct Heat Removal |
| 41 26 | 1.5.1.3 | Radial Blanket Failure Threshold |
| | | Failed Radial Blanket Rod Evaluations |
| | | Radial Blanket Assembly Local Flow Blockage Evaluation |
| 41 | 1.5.1.4 | Sodium-Water Reaction Pressure Relief Test |
| | 1.5.2 | Information Concerning Margin of Conservatism of Proven Design |
| | 1.5.2.1 | Pipe Integrity Assessment |
| | | Fracture Mechanics Study |
| | | Characteristics of Sodium-Induced Corrosion |
| 41 | | |
| | | Sodium Leak Detection Feature Test |
| | 1.5.2.2 | Failed Fuel Assembly Tests for Accident Conditions |
| | | Duct Wall Behavior Test |
| | | 217- Pin Test |
| 41 | 1.5.2.3 | Reactor Thermal and Hydraulic Tests |
| | | Large Bundle Partial Blockages Evaluations |
| | | Inlet Plenum Bubble Dispersion Test |
| | | Inlet Module Blockage Prevention Test |
| | | Inlet Plenum Particle Mobility Test |
| | 1.5.2.4 | Core Restraint System Tests |
| | | Full-Core Restraint System Test |
| | 1.5.2.5 | Critical Experiments for Reactivity Coefficients, Control Rod Worth and Fuel Assembly Movement |
| | 1.5.2.6 | Source Range Flux Monitoring System Tests |
| | 1.5.2.7 | Ex-Vessel Transfer Machine Heat Removal Tests |

Tables 1.5-2 and 1.5-3 and 1.5-4

Have Been Deleted

41 Tables 1.5-2 and 1.5-3

Have Been Deleted

Table 1.5-4

Phase II Simple Model Experimental Tests (1/20th Scale)

| Experiment Number | Head | Vessel | Core Barrel | Upper Internals | Energetics |
|---|---|---|---|---|---|
| SM-1 | S1 | R1 | None | None | Static Pressure |
| SM-2 | F1 | Flexible | Flexible | None | TLSM |
| SM-3 | F1 | " | " | S3 | TLSM |
| SM-4 | S2 | " | " | S3 | TLSM |
| SM-5 | S2 | " | " | S3 | TLSM |
| SM-6 | S2 | " | " | S3 | Higher than TLSM* |

S1 = Simplified heat including plugs and shear rings

S2 = Simplified head including plugs, shear rings, under-head shielding and risers

S3 = Simplified upper internal structures including lower plate and columns

R1 = Rigid vessel (no plastic deformation)

F1 = Flexible head (plus idealized representation of plug to plug gaps)

TLSM = Third Level Structural Margin conditions considered for CRBR(661 MJ work energy on expansion to one atmosphere)

---

* For energetics higher than TLSM see text for discussion. 34

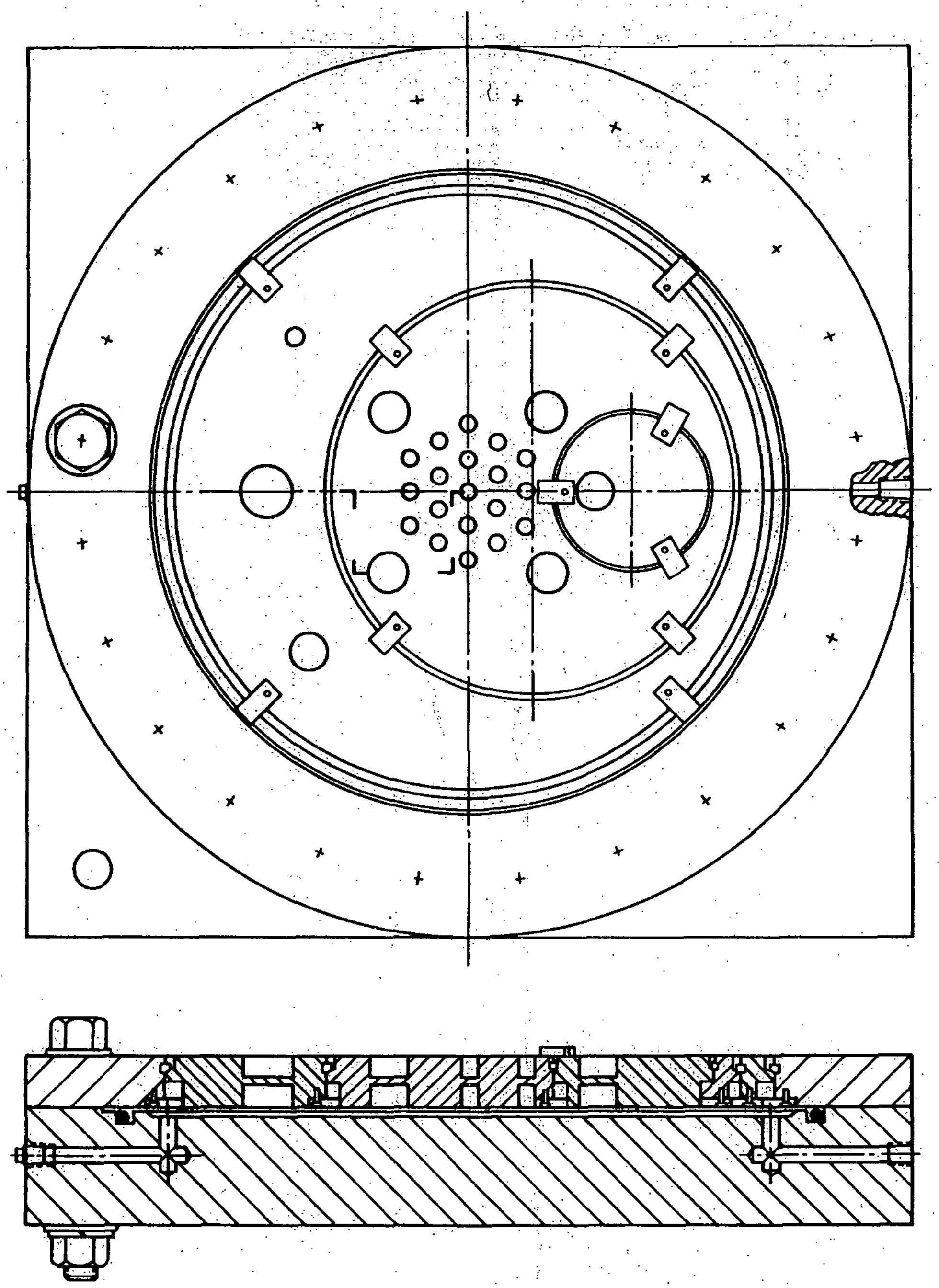


Figure 1.5-1 The SM-1 Model

7683-242

Amend. 34
Feb. 1977

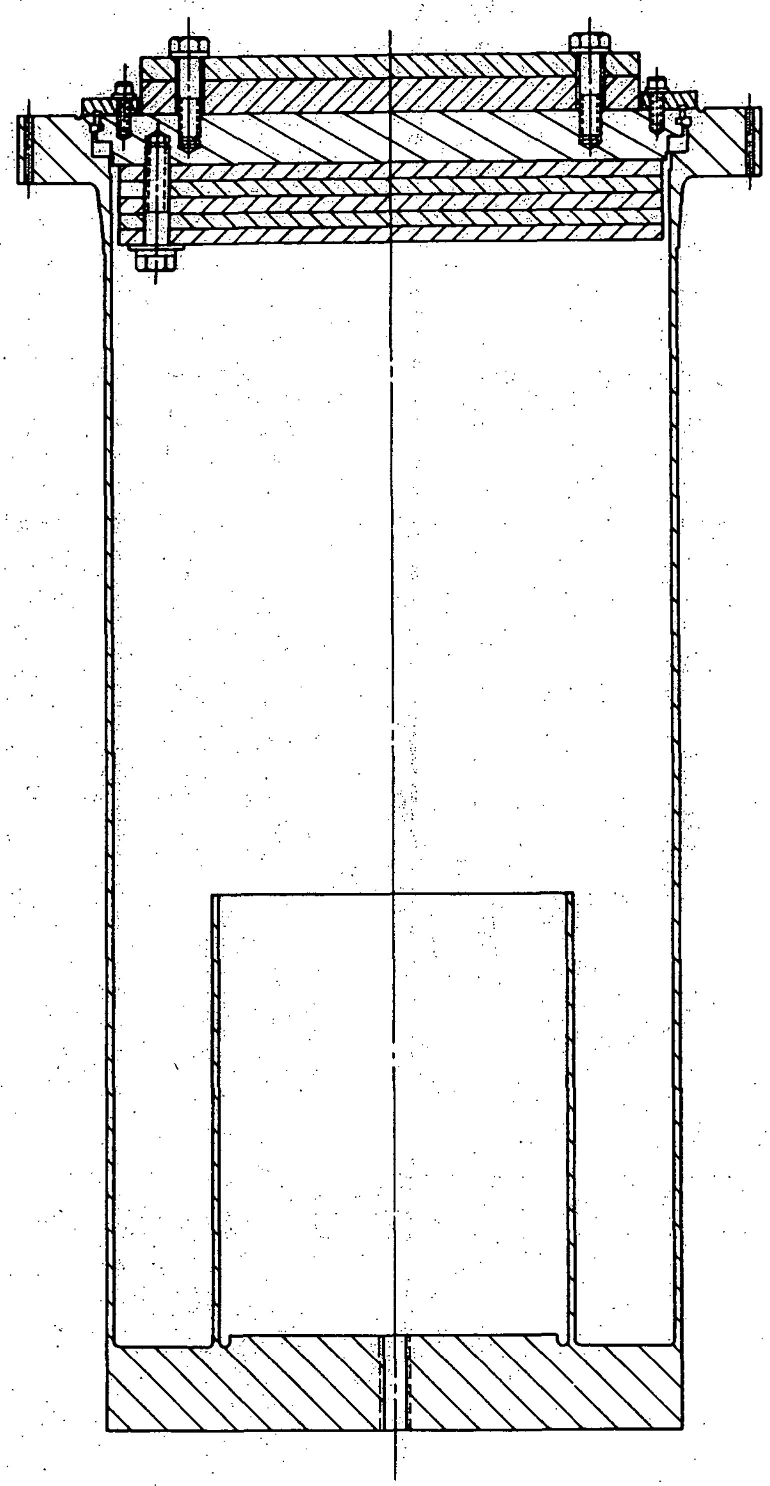


Figure 1.5-2 The SM-2 Model

7683-243

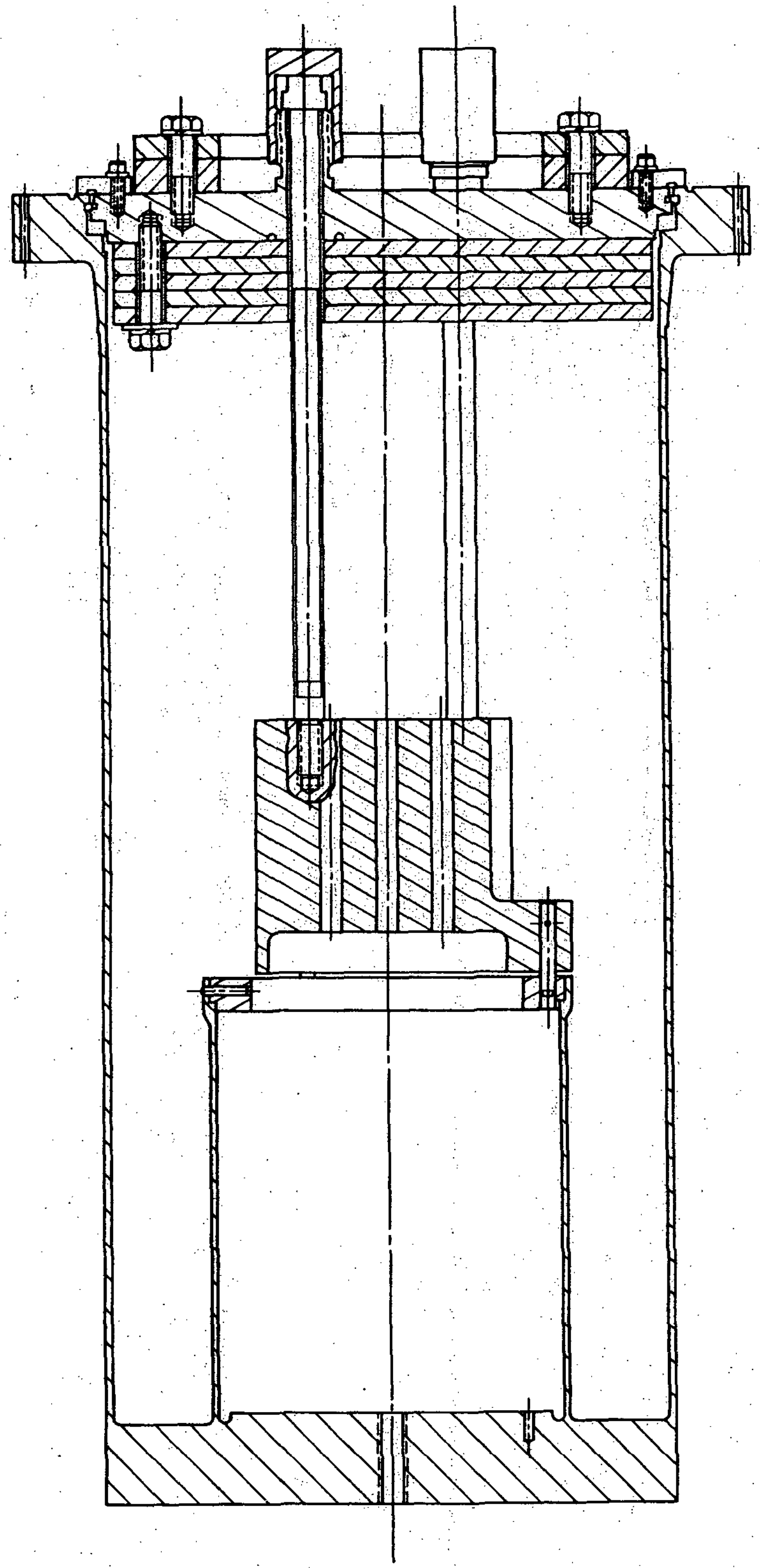


Figure 1.5-3 The SM-3 Model

7683-244

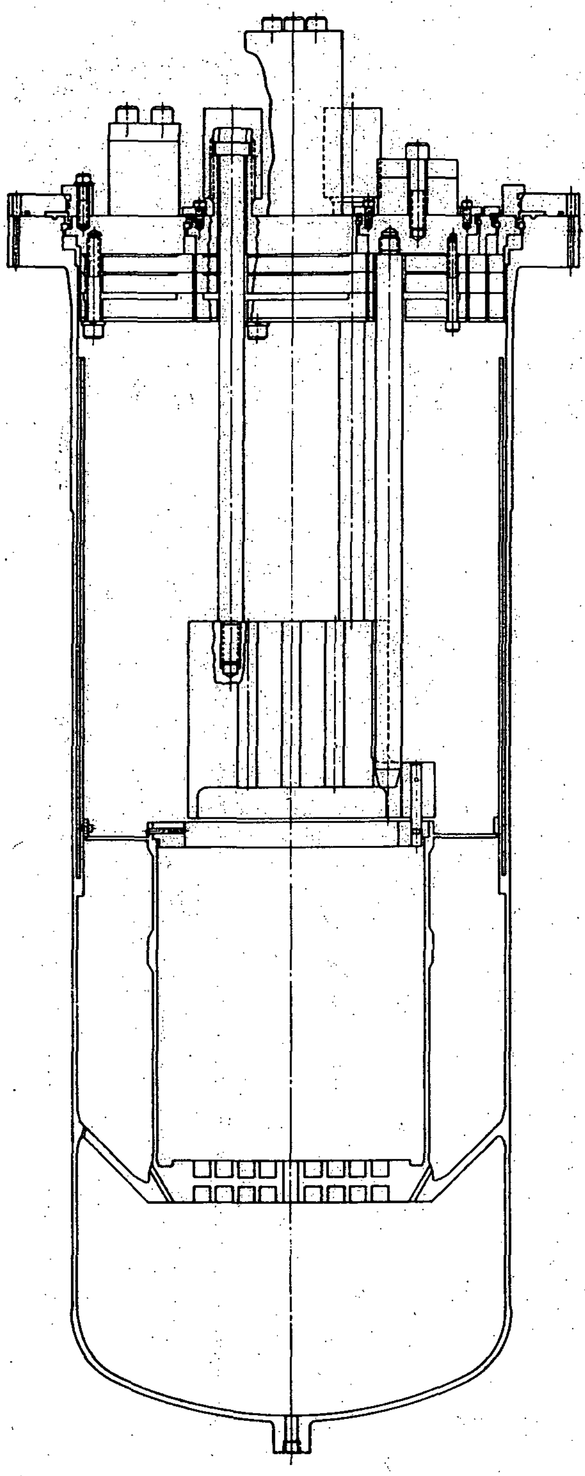


Figure 1.5-4 The SM-4/SM-5/SM-6 Models

7683-245

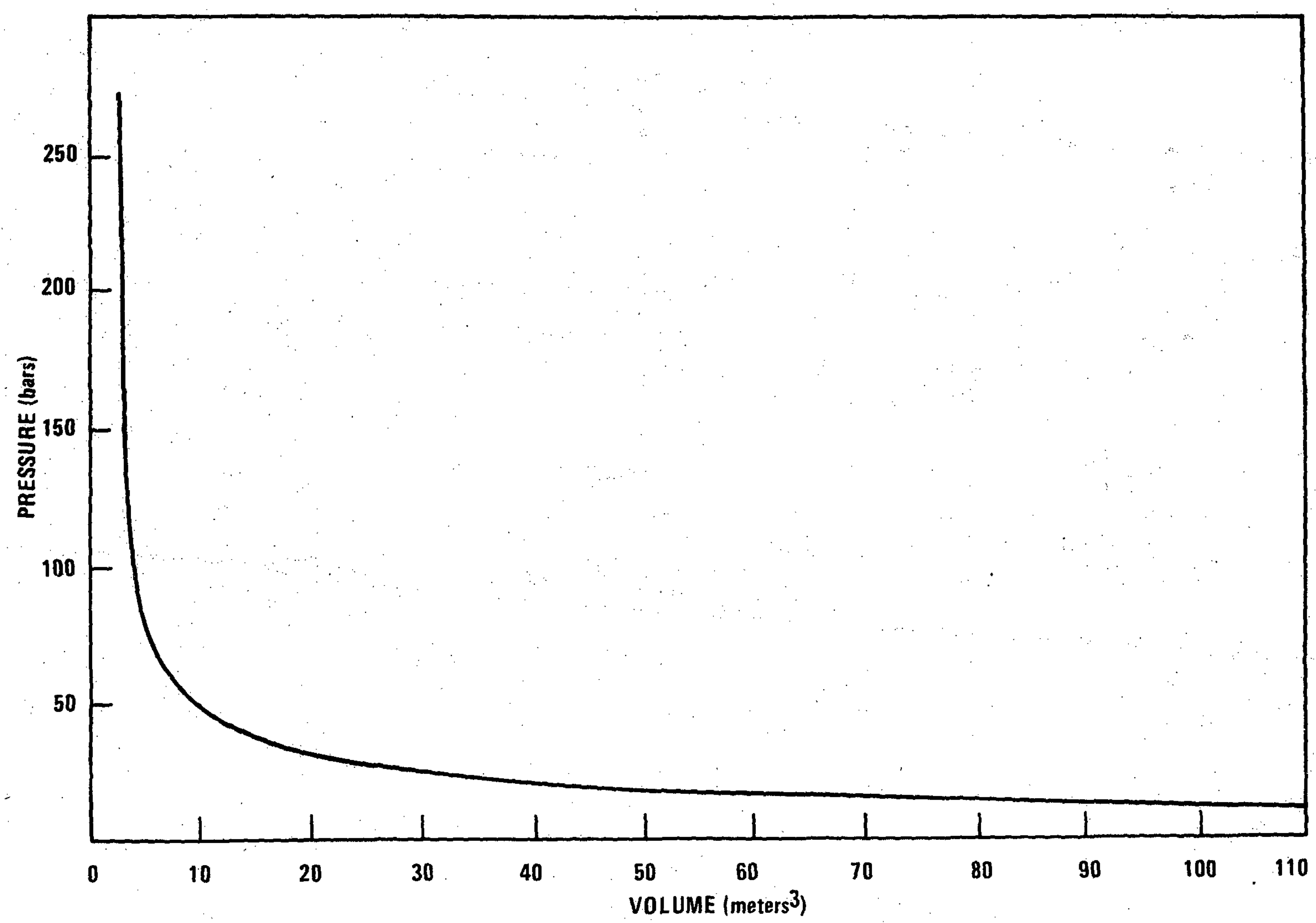



Figure 1.5-5 Pressure Volume Relationship For The 661 MJ HCDA

## 1.6 MATERIAL INCORPORATED BY REFERENCE

### 1.6.1 Introduction

This section identifies technical reports incorporated by reference into the PSAR. Some of the technical reports cited were produced for the LMFBR program under the direction of the Energy Research and Development Administration (ERDA) and, therefore, contain the disclaimer notice as required by ERDA manual Appendix 3201, Part II-D. In support of the construction permit application for the Clinch River Breeder Reactor Plant, however, any such disclaimer notice should be considered to be deleted and therefore of no effect.

### 1.6.2 References


24 36 1. Deleted.

42 2. WARD-D-0185, "Clinch River Breeder Reactor Plant Integrity of Primary and Intermediate Heat Transport System Piping in Containment", September 1977.

3. WARD-D-0115, "Development and Application of a Cumulative Mechanical Damage Function for Fuel Pin Failure Analysis in LMFBR Systems", April 1976.

4. WARD-D-0005, "Demo Code" LMFBR Demonstration Plant Simulation Model, Rev. 4.

5. WARD-D-0090, "CRBRP Decay Power Analysis", January 1976.

36 6. Deleted.

7. AI Report No. 99-TI-413-039, "EVTM/CLEM Full Scale Test Analysis" R.G. Hanson, issued August 15, 1975.

8. AI Report No. 99-TI-413-042, "Subscale Emissivity Test Analysis (EVTM)", D. Vanevenhoven, issued October 17, 1975.

9. "Hypothetical Turbine Missile Data and Probability of Occurrence for 3600-RPM-23-Inch LSB Unit for Use with Liquid Metal Cooled Fast
42 16 Breeder Reactor", General Electric Co., August 4, 1977.

10a. CRBRP-3, Volume 1, "Hypothetical Core Disruptive Accident Considerations in CRBRP; Energetics and Structural Margin Beyond the Design Basis", Rev. 4.

10b. CRBRP-3, Volume 2, "Hypothetical Core Disruptive Accident Considerations in CRBRP, Assessment of Thermal Margin Beyond the Design Base", Rev. 4.

11. WARD-D-0178, "CRBRP; Closure Head Capability for Structural Margin Beyond Design Basis Loading", Revision 3, November 1978.

12. WARD-D-0174, "CRBRP; Active Pump and Valve Operability Verification Plan", April 1977.

13. WARD-D-0165, "Requirements for Environmental Qualification of CRBRP Class 1E Equipment", Rev. 6.

14. WARD-D-0218, "Structural Response of CRBRP Scale Models to a Simulated Hypothetical Core Disruptive Accident", October 1978.

15. CRBRP-GEFR-00103, "CRBRP; An Analysis of Hypothetical Core Disruptive Events in the Clinch River Breeder Reactor Plant", April 1978.

16. HQ:S:82:173, Letter: J. R. Longenecker to P. S. Check, Transmittal of "DEMO Pipe Break Analysis", dated January 1983.

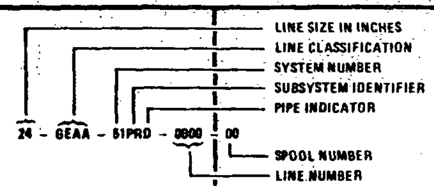
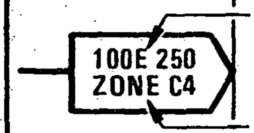
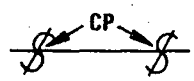
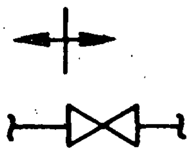
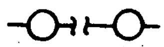
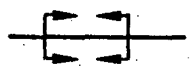
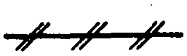
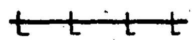
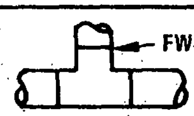
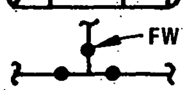
17. HQ:S:82:176, Letter: J. R. Longenecker to P. S. Check, Transmittal of "Methodology for CRBRP's Application of Radiological Source Terms In-Containment", dated January 1983.

18. HQ:S:82:184, Letter: J. R. Longenecker to P. S. Check, Transmittal of "CRBRP Reliability Assurance Activities", dated January 1983.

## Flow Diagram Symbols

### TABLE OF CONTENTS

## LINE DESIGNATIONS

| | | | |
|---|---|---|---|
|  | |  | LINE CONTINUES ON DWG INDICATED<br><br>IN ZONE INDICATED |
| <br><br> | PRINCIPAL FLOW ROUTE<br><br>ALTERNATE FLOW ROUTE |  | CUT PIPE HERE<br>TO REMOVE |
| 57  | PIPELINE<br>SPECIFICATION<br>CHANGE |  | ELECTRICALLY HEATED PIPE<br><br>(FLOW SHEET ONLY) |
|  | ELECTRICAL SIGNAL |  | BURIED PIPE |
|  | PNEUMATIC SIGNAL |  | PNEUMATIC SUPPLY |
|  | CAPILLARY TUBING<br>FILLED SYSTEM |  | HYDRAULIC SIGNAL |
|  | RADIATION OR SONIC<br>SIGNAL WITHOUT WIRING<br>OR TUBING | <br> | FIELD WELD<br><br>FIELD WELD IN SINGLE<br>LINE DRAWING |



Amend. 57
Nov. 1980

# LINE CLASSIFICATIONS

1. PIPING CLASSES ARE DESIGNATED BY A FOUR LETTER CODE. THE FIRST LETTER INDICATES THE PRIMARY VALVE AND FLANGE RATING; THE SECOND LETTER THE TYPE OF MATERIAL; THE THIRD LETTER THE CODE TO WHICH THE PIPING IS DESIGNED AND THE FOURTH LETTER INDICATES THE SYSTEM FLUID

   THE DESIGNATIONS ARE AS FOLLOWS:

   FIRST LETTER: PRESSURE RATING

57 A – 4500 ANSI
B – 2500 ANSI
C – 1500 ANSI
D – 900 ANSI
E – 600 ANSI
F – 400 ANSI
G – 300 ANSI
H – 150 ANSI
J – 125 ANSI B16.1
K – 175 WOG UNDERWRITERS LABORATORIES, INC.
L – 250 ANSI B16.1
X – GRAVITY RATING

   SECOND LETTER: MATERIAL

A – ALLOY
B – CARBON STEEL
C – STAINLESS STEEL (TP 304)
D – COPPER
E – STAINLESS STEEL (TP 316H)
F – CARBON STEEL – COPPER BEARING
G – CARBON STEEL – LINED
H – CAST IRON
J – CONCRETE PIPE
K – VITRIFIED CLAY PIPE
L – CARBON STEEL – IMPACT TESTED
M – DURIRON
N – CARBON STEEL – GALVANIZED
P – CAST IRON – CEMENT LINED
Q – ASBESTOS – CEMENT
57 U – PCV CHROME

   THIRD LETTER: DESIGN CODE

A – NUCLEAR POWER PLANT COMPONENTS, ASME B&PV CODE SEC. III CLASS I
B – NUCLEAR POWER PLANT COMPONENTS, ASME B&PV CODE SEC. III CLASS II
C – NUCLEAR POWER PLANT COMPONENTS, ASME B&PV CODE SEC. III CLASS III
D – POWER PIPING CODE ANSI B31.1.0-1967
F – NATIONAL FIRE PROTECTION ASSOCIATION CODE
G – NATIONAL PLUMBING CODE
H – POWER BOILERS, ASME B&PV CODE SEC. I
J – AMERICAN WATER WORKS STANDARDS

57

## LINE CLASSIFICATIONS (Continued)

FOURTH LETTER: SYSTEM FLUID

A – SODIUM
B – STEAM
C – FEEDWATER
D – NAK
E – DOWTHERM
F – ARGON
G – NITROGEN (GAS)
H – CHILLED WATER
J – WATER
K – AIR
L – LIUQID NITROGEN
M – MIXED CAPS GASES
N – LIQUID ARGON
P – ACID
R – CAUSTIC
S – ALCOHOL
T – HYDROGEN
V – CARBON DIOXIDE
W – OIL

SYSTEM/SUBSYSTEM/PIPE INDICATOR

1. FIRST TWO DIGITS IDENTIFY THE SYSTEM IN WHICH THE PIPE IS LOCATED
2. THE NEXT TWO ALPHAS IDENTIFY THE SUBSYSTEM
3. THE LETTER "D" INDICATES THAT THE ITEM IDENTIFIED IS PIPING.

LINE NUMBER

1. THE LINE NUMBER IS DESIGNATED BY SEQUENTIALLY ASSIGNED SERIAL NUMBERS FOR RUNS OF PIPE WITHIN THAT SYSTEM (THE MAXIMUM OF FOUR DIGITS)

SPOOL NUMBER

1. A SEQUENTIALLY ASSIGNED SERIAL NUMBER DENOTING PIPE SPOOLS WITHIN THE PIPE LINE. THE ASCENDING ORDER OF THE SEQUENCE IS IN THE DIRECTION OF THE FLUID FLOW.

### SPECIAL CASES

1. IN THE PIPELINE IDENTIFICATION SYSTEM THE TWO-DIGIT FIELD INDICATING NOMINAL PIPE DIAMETER MAY BE EXPANDED TO EXPRESS FRACTIONS, e.g. 2½ INCH PIPE IS POSTED AS 2.5.

2. THE PIPELINE IDENTIFICATION MAY BE ABBREVIATED ON DRAWINGS BY REDUCING THE THREE-DIGIT SEQUENTIAL SERIAL NUMBERS FROM 001, 002, 003, ETC. TO 1, 2, 3, ETC.

3. PIPELINE NUMBERS ON PIPING DRAWINGS MAY BE FURTHER ABBREVIATED BY ADDING A GENERAL NOTE STATING THAT "ALL PIPELINE NUMBERS APPEARING ON THIS DRAWING INCORPORATE THE SYSTEM/SUBSYSTEM DESIGNATION XXXX UNLESS OTHERWISE NOTED". THE SYSTEM/SUB-SYSTEM CHARACTERS MAY THEN BE OMITTED FROM PIPELINE IDENTIFICATION NUMBERS POSTED ON THE DRAWING.

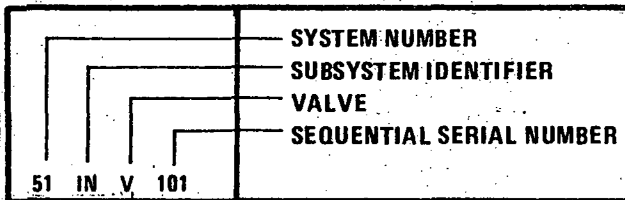
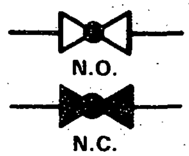
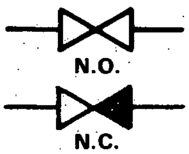
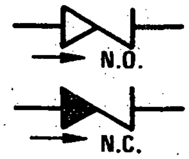
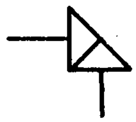
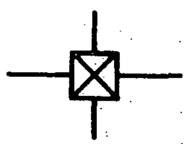
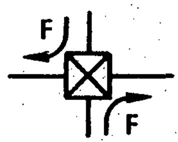
4. THE PIPELINE IDENTIFICATION FOR PIPING DRAWINGS WILL BE ABBREVIATED FOR PIPELINE LISTS BY ELIMINATING THE 2-DIGIT FIELD INDICATING NOMINAL PIPE SIZE AND THE 4-DIGIT FIELD INDICATING PIPELINE CLASSIFICATION.

   THE NOMINAL PIPE SIZE AND THE PIPELINE CLASSIFICATION WILL BE IDENTIFIED AS DATA ELEMENTS IN THE PIPELINE LIST, BUT NOT AS PART OF THE PIPELINE NUMBER.

5. IDENTICAL PIPE LINES IN DIFFERENT LOOPS WITHIN A SYSTEM ARE IDENTIFIED BY LETTER SUFFIX (A, B, C, etc).

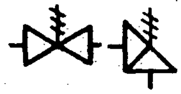
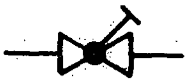
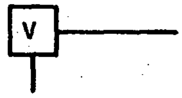
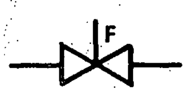
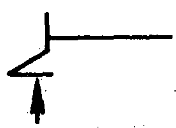
57

## VALVE BODY SYMBOLS

| | | | |
|---|---|---|---|
|  | |  | GLOBE |
|  | GATE |  | STOP CHECK |
|  | CHECK |  | ANGLE |
|  | AUTOMATIC CHECK POSITIVE CLOSING |  | BUTTERFLY |
|  | NEEDLE |  | BALL |
|  | PLUG |  | FOUR-WAY |
|  | THREE-WAY |  | FOUR-WAY INDICATING FAILURE PATH |

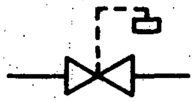
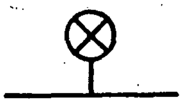
57

## VALVE BODY SYMBOLS (Continued)

| | | | |
|---|---|---|---|
|  | THREE WAY - INDICATING FAILURE PATH |  | SAFETY OR RELIEF |
|  (SAUNDERS TYPE) | DIAPHRAGM |  | AUTOMATIC BALL DRIP CHECK |
|  | "Y" PATTERN GLOBE (BLOWDOWN) |  | AIR VENT |
|  | MANUALLY OPERATED TEST |  | HOSE GLOBE |
|  | HOSE GATE |  | BELLOWS SEAL |
|  | HOSE ANGLE |  | PINCH |
|  | FREEZE SEAL |  | ANGLE CHECK (STOP CHECK) |

57

## VALVE BODY SYMBOLS (Continued)

| | | | |
|---|---|---|---|
|  | SLIDE OR BLAST GATE |  | FIRE HOSE, EXPOSED |
|  | FLOAT OPERATED |  | POST INDICATOR |
|  | HYDRANT |  | PREACTION VALVE |
|  | DRAG DISC VALVE |  | DELUGE VALVE |
|  | ALARM CHECK VALVE | | |
| | | | |
| | | | |

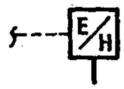
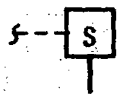
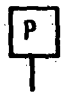
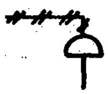
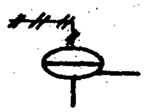
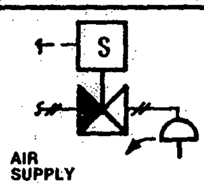
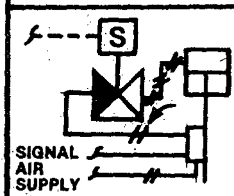
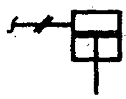
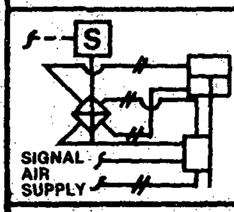
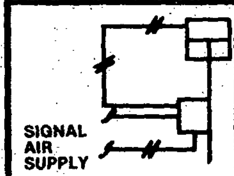
57

## ABBREVIATIONS ASSOCIATED WITH VALVES

| | |
|---|---|
| A.C. | Air Closes |
| A.O. | Air Opens |
| AV | Analytical Element Controlled Valve |
| F.A.I. | Fail As Is |
| F.C. | Fail Closed |
| FCV | Flow Control Valve |
| F.I. | Fail Intermediate |
| F.L. | Fail Locked |
| F.O. | Fail Open |
| FV | Flow Controlled Valve |
| HV | Hand Switch Controlled Valve |
| KV | Time Controlled Valve |
| L.C. | Locked Closed (Sodium Service) |
| L.O. | Locked Open |
| LV | Liquid Controlled Valve |
| N.C. | Normally Closed |
| N.O. | Normally Open |
| PCV | Pressure Controlled Valve |
| PDV | Pressure Differential Controlled Valve |
| PSV | Pressure Safety Valve |
| PV | Pressure Controlled Valve |
| RV | Radiation Controlled Valve |
| TV | Temperature Controlled Valve |
| UV | Multiple Signal Control |
| YV | Shut-off Valve (Sodium Service) |

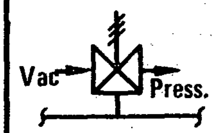
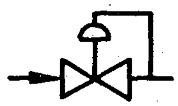
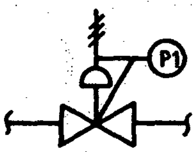
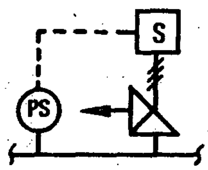
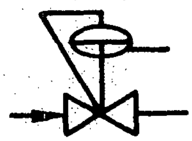
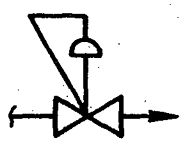
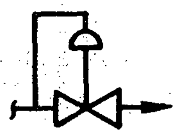
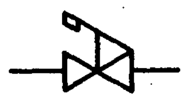
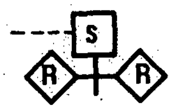
12

## VALVE ACTUATORS

| | | | |
|---|---|---|---|
|  | MOTOR OPERATED |  | HAND OPERATED (Mounted at top side or bottom of valve assembly) |
|  | HYDRAULIC |  | ELECTRO-HYDRAULIC |
|  | SOLENOID OPERATED |  | PNEUMATIC |
|  | DIAPHRAGM: SPRING OPPOSED, WITHOUT POSITIONER OR OTHER PILOT |  | DIAPHRAGM; PRESSURE-BALANCED |
|   | 1,2,3 DIAPHRAGM: SPRING OPPOSED, OVERRIDING PILOT VALVE THAT PRESSURIZES DIAPHRAGM WHEN ACTUATED |  | DIAPHRAGM: SPRING OPPOSED, ASSEMBLED WITH PILOT, ONE CONTROLLED INPUT |
|   | 1,2,3 SINGLE ACTING CYLINDER: CONVERTER, OVERRIDING PILOT VALVE THAT PRESSURIZES DIAPHRAGM WHEN ACTUATED |  | SINGLE ACTING CYLINDER WITHOUT POSITIONER OR OTHER PILOT |
|   | 2,3 DOUBLE ACTING CYLINDER WITH POSITIONER, CONVERTER, OVER-RIDING PILOT VALVE |  | DOUBLE ACTING CYLINDER WITHOUT POSITIONER OR OTHER PILOT |
|   | 2,3 SINGLE ACTING CYLINDER WITH POSITIONER | 1. NORMALLY SHUT PORT IS "FILLED IN." 2. OTHER COMBINATIONS ARE POSSIBLE AND WHEN USED SHALL FOLLOW THE FORMAT ESTABLISHED BY THESE EXAMPLES. 3. ITEMS NOT SHOWN ON P&ID | |

57

## VALVE ACTUATORS (Continued)

| | | | |
|---|---|---|---|
| 57 | |   | PRESSURE AND VACUUM RELIEF VALVE WITH INTEGRAL PILOT. NOTE: ACTUATOR SYMBOL: FOR PRESSURE RELIEF OR SAFETY VALVES ONLY: DENOTES A SPRING WEIGHT OR INTEGRAL PILOT |
|  | PRESSURE REDUCING REGULATOR SELF-CONTAINED |  | PRESSURE REDUCING REGULATOR WITH EXTERNAL PRESSURE TAP |
|   | PRESSURE REDUCING REGULATOR WITH INTEGRAL OUTLET PRESSURE RELIEF VALVE AND OPTIONAL PRESSURE INDICATOR (TYPICAL AIR SET) |   | PRESSURE RELIEF OR SAFETY VALVE, ANGLE PATTERN, TRIPPED BY INTEGRAL SOLENOID |
|  | DIFFERENTIAL PRESSURE REDUCING REGULATOR WITH INTERNAL AND EXTERNAL PRESSURE TAPS |  | BACK PRESSURE REGULATOR SELF-CONTAINED |
|  | BACK PRESSURE REGULATOR WITH EXTERNAL PRESSURE TAPS |  | QUICK OPENER |
|   | SOLENOID RESET (OPTIONAL) | | |
| | | | |

Amend. 57
Nov. 1980

## ABBREVIATIONS ASSOCIATED WITH VALVES

| | | | |
|---|---|---|---|
| N O. | NORMALLY OPEN | N C. | NORMALLY CLOSED |
| L.O. | LOCKED OPEN | L C. | LOCKED CLOSED |
| F O. | FAIL OPEN | F.C | FAIL CLOSED |
| F L | FAIL LOCKED | F.I | FAIL INTERMEDIATE |
| F A I | FAIL AS IS | A O. | AIR OPENS |
| A C | AIR CLOSES | SOV | SOLENOID OPERATED VALVE |
| AOV | AIR OPERATED VALVE | MOV | MOTOR OPERATED VALVE |
| | | NRV | NON RETURN VALVE |

## SPECIAL CASES

1. THE VALVE IDENTIFICATION SYSTEM MAY BE EXPANDED TO DESIGNATE VARIOUS TYPES OF CONTROL VALVES WITH A 4–DIGIT CODE DEFINING THE VALVE FUNCTION. THE VALVE FUNCTION CODE APPEARS IN THE "INSTRUMENTATION IDENTIFICATION TABLE" IN COLUMNS HEADED "CONTROL VALVE" AND SELF "ACTUATED VALVE", e.g.

   FCV SELF ACTUATED FLOW CONTROL VALVE
   FFV FLOW RATIO CONTROL VALVE
   LCV SELF ACTUATED LEVEL CONTROL VALVE
   PDV PRESSURE DIFFERENTIAL CONTROL VALVE
   TDCV SELF ACTUATED TEMPERATURE DIFFERENTIAL CONTROL VALVE

2. THE THREE–DIGIT SEQUENTIAL SERIAL NUMBER IN THE VALVE IDENTIFICATION SYSTEM POSTED AS 001, 002, 003, ETC. ON VALVE LISTS MAY BE SIMPLIFIED TO READ 1, 2, 3, ETC.

3. BY ADDING A GENERAL NOTE TO DRAWINGS STATING THAT ALL VALVE IDENTIFICATION NUMBERS APPEARING ON THE DRAWING "ARE PREFIXED BY SYSTEM/SUB–SYSTEM XXXX UNLESS OTHERWISE NOTED", THE VALVE IDENTIFICATION SYSTEM MAY BE FURTHER SIMPLIFIED ON PIPING DRAWINGS BY ELIMINATION OF THE FOUR–DIGIT SYSTEM/SUB–SYSTEM CHARACTERS.

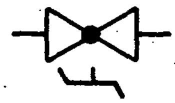
4. ON DRAWINGS INTENDED FOR FABRICATION/ERECTION REVERSE VALVE INSTALLATIONS SHALL BE CONSPICUOUSLY IDENTIFIED ON THE DRAWING BY A SPECIAL NOTE, e. g. "INSTALL BACKWARDS", INDICATING THAT THE VALVE SHALL BE POSITIONED WITH THE FLOW ARROW ON THE VALVE BODY DIRECTED AGAINST NORMAL SYSTEM FLOW.

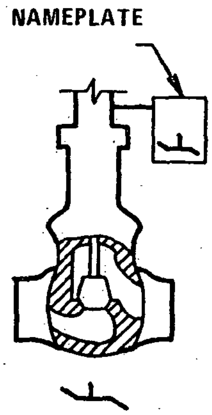
57

## VALVE ORIENTATION SYMBOLOGY

**SEAT – BRIDGE MARK**

THE VALVE ORIENTATION SYMBOL FOR CONVENTIONAL "Y"-PATTERN GLOBE VALVES (⌐┴⌐) DENOTES FLOW UNDER THE PLUG WHEN THE PROCESS FLUID IS MOVING FROM LEFT TO RIGHT AND OVER THE PLUG WHEN FLOW IS IN THE REVERSE DIRECTION. THE SEAT-BRIDGE MARK CLEARLY DENOTES VALVE STEM-SEAT ORIENTATION. RELATIVE TO SYSTEM FLOW FOR ALL GLOBE VALVE.



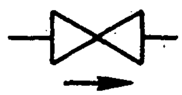


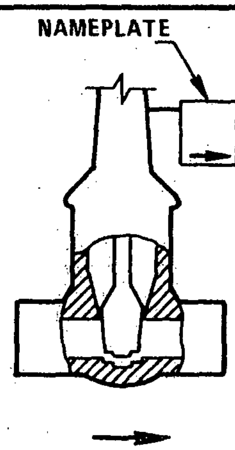

CONVENTIONAL GLOBE VALVE

**FLOW DIRECTION ARROW**

ALL OTHER VALVES EXCEPT THE VALCOR LIQUID METAL VALVES SHALL UTILIZE A FLOW DIRECTION ARROW ( ⟶ ) TO DENOTE THE ORIENTATION OF THE VALVE IN THE LINE.

THE ORIENTATION SYMBOL WILL BE LOCATED BELOW THE VALVE SYMBOL ON THE PIPING & INSTRUMENT DIAGRAMS AND INCLUDED ON THE VALVE NAMEPLATES AT THE TIME OF MANUFACTURE.

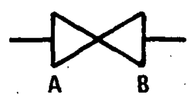


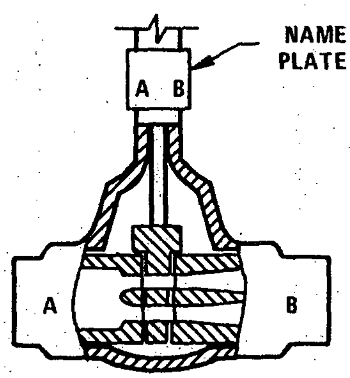



CONVENTIONAL GATE VALVE

**STRAIGHT PATTERN LIQUID METAL VALVE (LMS)**

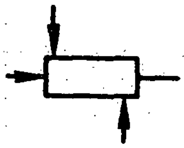
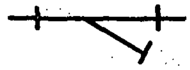
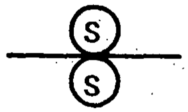
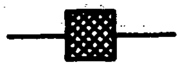
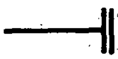
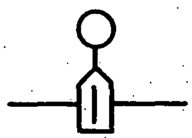
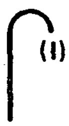
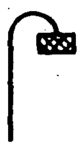
THE VALCOR LIQUID METAL VALVE SHALL IDENTIFY THE INLET PORT AS PORT "A" AND THE OUTLET PORT AS PORT "B". THE DESIGNATION SHALL BE ON THE VALVE BODY & VALVE NAMEPLATE.



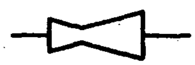




STRAIGHT PATTERN LIQUID METAL VALVE (LMS)

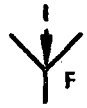
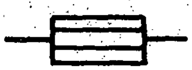
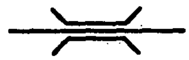
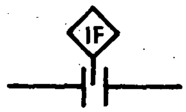
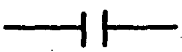
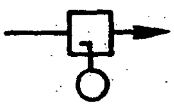
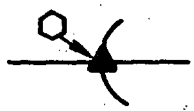
## SPECIALTY SYMBOLS

| | | | |
|---|---|---|---|
|  | IN LINE SELFCLEANING STRAINER |  | Y TYPE STRAINER |
|  | SIMPLEX STRAINER |  | DUPLEX STRAINER |
|  | FILTER |  | TEMPORARY STRAINER |
|  | BLIND FLANGE |  | PLUG OR CAP |
|  | EXPANSION JOINT |  | SPECTACLE FLANGE |
|  | RESTRICTING ORIFICE |  | ORIFICE PLATE IN QUICK CHANGE FITTING |
|  | VENT (I) = INDOORS<br>(O) = OUTDOORS |  | VENT WITH FLAME ARRESTER |
| 57  | TAPERED EXPANSION JOINT | | |

## SPECIALTY SYMBOLS (Continued)

| | | | |
|---|---|---|---|
|   | STEAM TRAP |  | DRIP PAN ELBOW (FOR STEAM RELIEF VALVE) |
|  | LOOP SEAL |  | SPRAY NOZZLES |
|  | IN-LINE SIGHT FLOW GLASS |  | FLEXIBLE CONNECTION |
|  | QUICK DISCONNECT |  | FLOW NOZZLE (TUBE OR VENTURI) |
|  | BEARING |  | REDUCER OR INCREASER |
|  | FLOOR DRAIN, HUB OR TRENCH |  | CHEMICAL SEAL |
|  | CIRCULAR OR HAMMER BLIND |  | VAPOR TRAP, DRY |

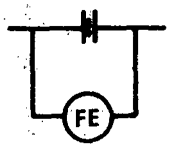
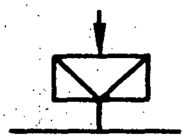
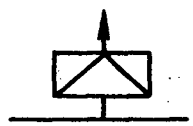
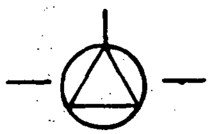
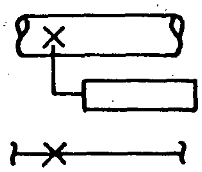
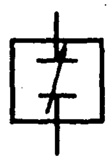
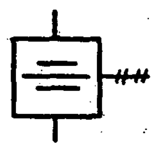
## SPECIALTY SYMBOLS (Continued)

| | | | |
|---|---|---|---|
|  | VAPOR TRAP, WET |  | FUNNEL DRAIN |
|  | EXHAUST HEAD |  | MIXING TEE |
|  | FLOW STRAIGHTENING VANES |  | MAGNETIC FLOWMETER<br>E.M. = ELECTROMAGNETIC<br>P.M. = PERMANENT MAGNET |
|  | SERVICE CONNECTION |  | INSULATING FLANGE |
|  | FLANGE CONNECTION |  | SPOOL PIECE |
|  | PITOT OR PITOT VENTURI TUBE |  | ANCHORED CONTAINMENT PENETRATION |
|  | BELLOWS SEAL CONTAINMENT PENETRATION |  | LINED CELL PENETRATION |



Amend. 57
Nov. 1980

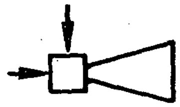
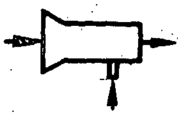
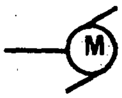
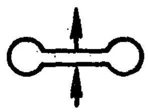
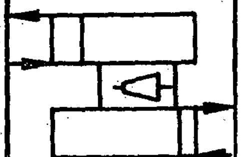
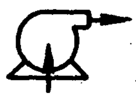
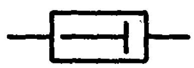
## SPECIALTY SYMBOLS (Continued)

| | | | |
|---|---|---|---|
|  | ORIFICE PLATE WITH VENA CONTRACTA RADIUS OR PIPE TAPS |  | ATTEMPERATOR |
|  | SAFETY HEAD |  | SPRINKLER NOZZLE |
|  | RUPTURE DISK VACUUM RELIEF |  | RUPTURE DISC PRESS RELIEF |
|  | GAS SEPARATOR |  | PIPE SUPPORT<br>PIPE SUPPORT IDENTIFICATION NO.<br>PIPE SUPPORT FOR SINGLE LINE DRAWING |
|  | ACOUSTIC (NOISE) DETECTOR |  | VARIABLE SPEED COUPLING |
| | |  | CLUTCH & BRAKE |
| | | | |

57

Amend. 57
Nov. 1980

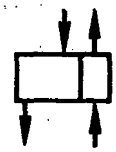
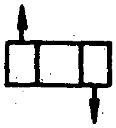
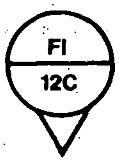
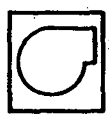
## EQUIPMENT SYMBOLS

| | | | |
|---|---|---|---|
|  | DIESEL DRIVE |  | EDUCTOR OR EJECTOR |
|  | DESUPERHEATER |  | TURBINE DRIVE |
|  | ELECTRIC MOTOR DRIVE |  | CENTRIFUGAL PUMP |
| | |  | POSITIVE DISPLACEMENT PUMP |
|  | CENTRIFUGAL LIQUID CHILLER |  | PLUGGING METER |
|  | GENERAL PURPOSE VERTICAL PUMP |  | GENERAL PURPOSE HORIZONTAL PUMP |
|  | ROTARY PUMP |  | PISTON PUMP |

57

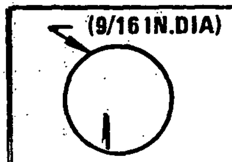
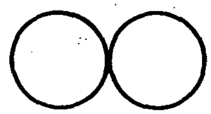
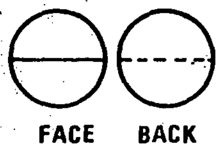
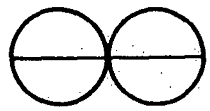
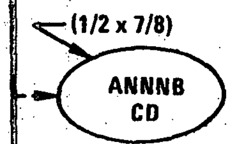
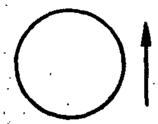
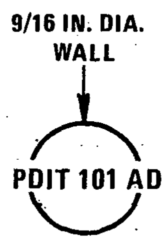
Amend. 57
Nov. 1980

## EQUIPMENT SYMBOLS (Continued)

| | | | |
|---|---|---|---|
|  | ELECTROMAGNETIC PUMP |  | HEAT EXCHANGER (LIQUID - LIQUID) |
|  | HEAT EXCHANGER (LIQUID - GAS) |  | WATER TRAP |
|  | FLOW SENSOR |  | SHOWER HEAD |
|  | EYE WASH |  | FLOW BALANCING VALVE |
|  | VERTICAL PUMP |  | MIXER |
|  | WATER HAMMER ARRESTER |  | BACKFLOW PREVENTER |
|  | ROTOMETER |  | —INLINE PUMP |

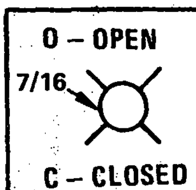
57

## INSTRUMENT DESIGNATIONS

| | | | |
|---|---|---|---|
|  | RELAY OR LOCAL INSTRUMENT INCLUDING TRANSMITTER FOR SINGLE MEASURED VARIABLE |  | LOCAL INSTRUMENT FOR TWO MEASURED VARIABLES OR MORE THAN ONE FUNCTION |
|  | PANEL MOUNTED INSTRUMENT FOR SINGLE MEASURED VARIABLE |  | PANEL MOUNTED INSTRUMENT FOR TWO MEASURED VARIABLES OR MORE THAN ONE FUNCTION |
|  | VARIABLE INTO DATA SYSTEM WHERE:<br>A = TYPE OF VARIABLE OR MEASUREMENT<br>NNN = SERIAL NUMBER (LOOP OR CHANNEL NUMBER WHENEVER POSSIBLE)<br>B = TYPE OF INPUT SIGNAL<br>CD = OPTIONAL PARALLEL OR REDUNDANT MEASUREMENTS AND PLANT PROTECTION SIGNALS<br>WHERE TYPES OF SIGNAL "B" IS DEFINED AS:<br>A – ANALOG<br>D – DIGITAL<br>E – EVENT (CONTACT SENSE)<br>P – PULSE (CONTACT INTERRUPT) |  | CONTROL SIGNAL FROM DATA SYSTEM |
|  | ARROW INDICATES DIRECTION IN WHICH RELAY RESPONDS TO A FAULT.<br>ARROW UP = FORWARD LOOKING<br>ARROW DOWN = REVERSE LOOKING |  | INSTRUMENT BALLOON WITH INSTRUMENT NUMBER (WALL OF BALLOON MAY BE RUPTURED TO ACCOMMODATE INSTRUMENT NUMBER). |

57

## INDICATING LIGHTS

| | | | |
|---|---|---|---|
|  | O - OPEN<br>C - CLOSED<br>H - HIGH<br>L - LOW | COLORS<br>W - WHITE<br>G - GREEN<br>R - RED<br>A - AMBER | |

Amend. 57
Nov. 1980

INSTRUMENT DESIGNATIONS (CONTINUED)

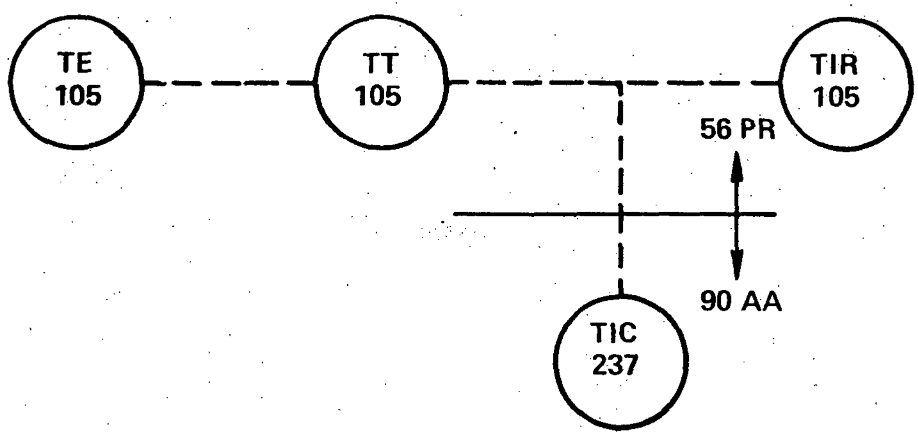
INSTRUMENT IDENTIFIER IS AS FOLLOWS:

NMABCDEFGXYZHJ

WHERE:

| | |
|---|---|
| NM | ARE TWO NUMBERS IDENTIFYING THE SYSTEM DESIGN DESCRIPTION (SDD) |
| AB | ARE TWO LETTERS REPRESENTING THE SDD SUBSYSTEM |
| CDEFG | ARE FIVE LETTERS REPRESENTING THE FUNCTION OF THE INSTRUMENTATION CONSISTENT WITH ISA-S5.1 |
| XYZ | ARE THREE NUMBERS TO REPRESENT LOOP OR CHANNEL NUMBER (ASSIGNED BY THE COGNIZANT ENGINEER |
| HJ | ARE TWO OPTIONAL LETTER(S) TO INDICATE REDUNDANT OR PARALLEL MEASUREMENTS WITHIN A LOOP OR CHANNEL. ALL LETTERS A THROUGH Z CAN BE USED WITH "P" AND "S" RESERVED FOR PLANT PROTECTION SYSTEM. |
| NOTE: | THE SDD AND THE SUBSYSTEM IDENTIFYER ARE NOT PLACED INSIDE THE INSTRUMENT BALLOON ON A DRAWING BUT ARE IDENTIFIED BY A NOTE ON THE DRAWING. IF MORE THAN ONE SYSTEM IS REPRESENTED ON A DRAWING, THE CONVENTION OF ISA-S5.1 SHALL APPLY. (INSTRUMENT BALLOONS IN EXAMPLE ARE ENLARGED RATHER THAN RUPTURED TO ACCOMMODATE INSTRUMENT NUMBER) |

FOR EXAMPLE:




57

Amend. 57
Nov. 1980

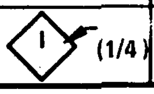
## RELAY AND CONVERSION DEVICE

| | | | |
|---|---|---|---|
| I/P | CURRENT TO PNEUMATIC | E/P | VOLTAGE TO PNEUMATIC |
| E/I | VOLTAGE TO CURRENT | P/I | PNEUMATIC TO CURRENT |
| R/I | RESISTANCE TO CURRENT | E/H | VOLTAGE TO HYDRAULIC |
| P/E | PNEUMATIC TO VOLTAGE | △ | DIFFERENCE |
| Ⓕ | FREQUENCY METER | GE | GENERATOR EXCITER |

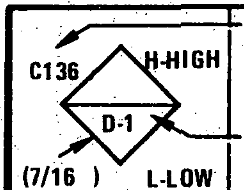
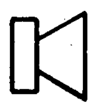
RELAY DESIGNATIONS ← → (SEE ANSI C37.2)

21x TIMER (TO BE USED WITH 21Z2)
21Z1 PHASE DISTANCE RELAY, ZONE 1
- 21Z2 FOR ZONE 2
  21Z3 FOR ZONE 3
  ETC.
27 UNDER VOLTAGE RELAY
40 FIELD CURRENT RELAY
42 CONTRACTOR
46 PHASE BALANCE CURRENT RELAY
49 THERMAL RELAY
50 INSTANTANEOUS OVERCURRENT RELAY
50BF BREAKER FAILURE CURRENT DETECTOR RELAY
50CM CURRENT MONITORING RELAY
50FD PHASE FAULT DETECTOR
50G INSTANTANEOUS GROUND OVERCURRENT RELAY
51 TIME OVERCURRENT RELAY
51G TIME GROUND OVERCURRENT RELAY
51N NEUTRAL INDUCTION TIME OVERCURRENT
51V GENERATOR INVERSE TIME OVERCURRENT WITH VOLTAGE RESTRAIN RELAY
59 OVERVOLTAGE RELAY
60 FUSE FAILURE RELAY
62 TIME DELAY RELAY
63 FAULT PRESSURE RELAY
67 DUAL POLARIZED DIRECTIONAL GROUND RELAY
74 ALARM RELAY
79 RECLOSER
81 FREQUENCY RELAY
83 DROPOUT RELAY
85 CARRIER AUXILIARY RELAYS
85R CARRIER RECEIVER RELAY FOR LINE RELAY CHANNEL
85T CARRIER TRANSMITTER RELAY FOR LINE RELAY CHANNEL
85TTS CARRIER TRANSFER TRIP RELAY SEND
86 HAND RESET LOCKOUT RELAY
87 DIFFERENTIAL RELAY
94 HIGH SPEED TRIPPING RELAY
OSC OSCILLOGRAPH ELEMENT
TT TRANSFER TRIP
57 BFBU BREAKER FAILURE BACKUP

## RELAY AND CONVERSION DEVICE (Continued)

| | | | |
|---|---|---|---|
| ± | BIAS | AVG | AVERAGE |
| AVG-R | AVERAGE - REJECT | 1:1 | BOOST |
| % OR 1:3 | GAIN OR ATTENUATE | X | MULTIPLY |
| ÷ | DIVIDE | f(x) | FUNCTION GENERATOR |
| REV | REVERSING | A/D OR D/A | ANALOG TO DIGITAL OR DIGITAL TO ANALOG |
| ∫ | INTEGRATE | Σ | SUMMER |
| D OR d/dT | DERIVATIVE OR RATE | < | SELECT LOWER |
| > | SELECT HIGHER | CP | COMPUTER |
| LIM | LIMITER | VOT | 2 OUT OF 3 VOTER |
| 1 – 0 | AUTOMATICALLY CONNECT, DIS-CONNECT OR TRAMSFER ONE OR MORE CIRCUITS | n/m | SELECTOR eg 1/4-1 OUT OF 4 |
|  | INTERLOCK |  | PANEL MOUNTED PATCHBOARD OR MATRIX CONNECTION |

## ANNUNCIATOR

| | | | |
|---|---|---|---|
|  | PANEL NO.<br><br>ANNUNCIATOR POINT WINDOW NO. |  | AUDIBLE ALARM |

| FIRST LETTER | | SECOND & SUCCEEDING LETTERS | | | | | | | | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | DISPLAY DEVICES | | | | | | | CONTROLLING DEVICES | | | | | | | SENSING DEVICES | | | | | | | | |
| | | | | | | LOCAL ALARM (See Note 9) | | | | | | | | | | | | | | | | | | |
| MEASURED VARIABLE | SYMBOL FOR MEASURED VARIABLES | INDICATING | RECORDING | INTEGRATING INDICATOR (See Note 4) | SCAN (See Note 6) | LOW | HIGH | LOW HIGH | INDICATING | RECORDING | BLIND | CONTROL VALVE | SELF ACTUATED VALVE | FINAL CONTROL ELEMENT (See Note 8) | SWITCH (See Note 5) | PRIMARY ELEMENT | BLIND TRANSMITTER (See Note 13) | INDICATING TRANSMITTER | LOCAL OBSERVATION GLASS | TEST CONNECTION | RELAY OR CONVERTER (BLIND) (See Note 2) | UNDESIGNATED FUNCTION | BUFFER | CONTROL STATION NOTE 10 |
| TYPICAL SYMBOL | ( ) | ( )I | ( )R | ( )QI | ( )J( ) | ( )AL | ( )AH | ( )AHL | ( )IC | ( )RC | ( )C | ( )V | ( )CV | ( )Z | ( )S( ) | ( )E | ( )T | ( )IT | ( )G | ( )P | ( )Y | ( )X | ( ) B | ( )K |
| ANALYSIS (See Note 1) | * A | AI | AR | | AJ ( ) | AAL | AAH | AAHL | AIC | ARC | AC | AV | | AZ | AS( ) | AE | AT | AIT | | AP | AY | AX | | AK |
| BURNER FLAME | B | BI | BR | | BJ ( ) | BAL | | | | | BC | BV | | | BS( ) | BE | BT | | BG | BP | BY | BX | | |
| CONDUCTIVITY | C | CI | CR | | CJ ( ) | CAL | CAH | CAHL | CIC | CRC | | CV | | CZ | CS( ) | CE | CT | CIT | | CP | CY | CX | | |
| DENSITY | * D | DI | DR | | DJ ( ) | DAL | DAH | DAHL | DIC | DRC | | DV | | DZ | DS( ) | DE | DT | DIT | | DP | DY | DX | | |
| VOLTAGE (EMF) | * E | EI | ER | | EJ ( ) | EAL | EAH | EAHL | EIC | ERC | EC | | | EZ | ES( ) | EE | ET | EIT | | | EY | EX | | |
| FLOW | F | FI | FR | FQI | FJ ( ) | FAL | FAH | FAHL | FIC | FRC | FC | FV | FCV | FZ | FS( ) | FE | FT | FIT | FG | FP | FY | FX | FB | |
| FLOW RATIO | FF | FFI | FFR | | FFJ( ) | | | | FFIC | FFRC | FFC | FFV | | FFZ | | | | | | | | | | |
| GAGING (DIMENSIONAL) | G | GI | GR | | GJ ( ) | GAL | GAH | GAHL | GIC | GRC | GC | GV | | GZ | GS( ) | GE | GT | GIT | | | | | | |
| HAND | H | | | | | | | | HIC | | HC | HV | HCV | HZ | HS( ) | | | | | | | | | |
| CURRENT | I | II | IR | IQI | IJ ( ) | IAL | IAH | IAHL | IIC | IRC | IC | | | IZ | IS( ) | IE | IT | IIT | | | IY | IX | | |
| POWER | J | JI | JR | JQI | JJ ( ) | JAL | JAH | JAHL | JIC | JRC | JC | | | JZ | JS( ) | JE | JT | JIT | | | JY | JX | JB | |
| TIME | K | KI | KR | KQI | KJ ( ) | KAL | KAH | KAHL | KIC | KRC | KC | | | KZ | KS( ) | | KT | KIT | | | KY | KX | | |
| LEVEL | L | LI | LR | | LJ ( ) | LAL | LAH | LAHL | LIC | LRC | LC | LV | LCV | LZ | LS( ) | LE | LT | LIT | LG | LP | LY | LX | LB | |
| MOISTURE | M | MI | MR | | MJ ( ) | MAL | MAH | MAHL | MIC | MRC | MC | MV | | MZ | MS( ) | ME | MT | MIT | | MP | MY | MX | | |
| LEAK DETECTOR | N | NI | NR | | | | | | | | | NV | | | | NE | | | | | | | | |
| TORQUE | O | OI | OR | | OJ ( ) | OAL | OAH | OAHL | OIC | ORC | OC | OV | | OZ | OS( ) | OE | OT | | | | OY | OX | | |
| PRESSURE | P | PI | PR | | PJ ( ) | PAL | PAH | PAHL | PIC | PRC | PC | PV | PCV | PZ | PS( ) | PE | PT | PIT | | PP | PY | PX | PB | |
| PRESSURE DIFFERENTIAL | PD | PDI | PDR | | | PDAL | PDAH | PDAHL | PDIC | PDRC | PDC | PDV | PDCV | PDZ | PDS( ) | | PDT | PDIT | | | | | PDB | |
| QUANTITY OR EVENT | Q | QI | QR | QQI | QJ ( ) | QAL | QAH | QAHL | QIC | QRC | QC | QV | | QZ | QS( ) | | QT | QIT | | | QY | QX | | |
| RADIATION | R | RI | RR | RQI | RJ ( ) | RAL | RAH | RAHL | RIC | RRC | RC | RV | | RZ | RS( ) | RE | RT | RIT | | RP | RY | RX | RB | |
| SPEED OR FREQUENCY | S | SI | SR | SQI | SJ ( ) | SAL | SAH | SAHL | SIC | SRC | SC | | | SZ | SS( ) | | ST | SIT | | | SY | SX | | |
| TEMPERATURE | T | TI | TR | | TJ ( ) | TAL | TAH | TAHL | TIC | TRC | TC | TV | TCV | TZ | TS( ) | TE SEE NOTE 7 | TT | TIT | | TP SEE NOTE 7 | TY | TX | TB | |
| TEMPERATURE DIFFERENTIAL | TD | TDI | TDR | | | TDAL | TDAH | TDAHL | TDIC | TDRC | TDC | TDV | TDCV | TDZ | TDS( ) | | | | | | | | TDB | |
| MULTI-VARIABLE | U | UI | UR | | UJ ( ) | UAL | UAH | UAHL | UIC | URC | UC | UV | | UZ | US( ) | | | | | | UY | UX | UB | |
| VISCOSITY | V | VI | VR | | VJ ( ) | VAL | VAH | VAHL | VIC | VRC | VC | VV | | VZ | VS( ) | VE | VT | VIT | | | VY | VX | | |
| WEIGHT | W | WI | WR | WQI | WJ ( ) | WAL | WAH | WAHL | WIC | WRC | WC | WV | | WZ | WS( ) | WE | WT | WIT | | | WY | WX | | |
| UNCLASSIFIED (See Note 3) | X | XI | XR | | XJ ( ) | XAL | XAH | XAHL | XIC | XRC | XC | XV | | XZ | XS( ) | XE | XT | XIT | | | XY | XX | | |
| VIBRATION | Y | YI | YR | | | | | | | | | | | | YS( ) | YE | | | | | YY | YX | | |
| POSITION | Z | ZI | ZR | | ZJ ( ) | ZAL | ZAH | ZAHL | ZIC | ZRC | ZC | | | ZZ | ZS( ) | ZE | ZT | ZIT | | | ZY | ZX | | |

NOTES

1. 'A' IS USED FOR ALL ANALYTICAL VARIABLES, FOR EXAMPLE: $O_2$, $H_2O$, $CO_2$ pH, OCTAINE IMPROVEMENT, CHROMATOGRAPH ANALYZING ONE OR MORE STREAMS FOR ONE OR MORE COMPOUNDS, BOILING POINT, FREEZING POINT, COMBUSTIBLES ETC. THE CHEMICAL FORMULA RECOGNIZED SYMBOL (SUCH AS pH) OR A DESCRIPTION DENOTING THE FUNCTION OF THE ANALYZER SHOULD BE NOTED ON THE P & ID OUTSIDE THE INSTRUMENT SYMBOL.

2. THE DESCRIPTION OR SYMBOL DENOTING THE FUNCTION OF THE RELAY 'Y' SHOULD BE SHOWN ON THE P & ID.

3. 'X' IS USED TO REPRESENT ANY 'SPECIAL' VARIABLES AND MAY BE DEFINED AS REQUIRED. FOR EXAMPLE. MASS FLOW RECORDERS WHICH RECEIVE A SIGNAL FROM A MULTIPLYING RELAY WHICH COMBINES THE PRODUCT OF DENSITY AND FLOW. THIS ITEM IS NOT TO BE CONFUSED WITH 'U' MULTI-VARIABLE SYMBOL.

4. WHEN 'Q' IS USED AS A SECOND OR SUCCEEDING LETTER IT DENOTES AN INTEGRATING MODIFIER FOR EXAMPLE 'FQI' IS AN INDICATING FLOW INTEGRATOR (OR TOTALIZER) NOTE THAT THE INTEGRATING FUNCTION SHALL BE SHOWN WITH SEPARATE IDENTIFICATION. FOR EXAMPLE: FQI/FRS OR FR/FQIS

5. STARTUP AND SHUTDOWN DEVICES ARE USUALLY BLIND, BUT MAY BE INDICATING OR RECORDING. IF SO, ADD 'I' OR 'R' AFTER MEASURED VARIABLE. FOR EXAMPLE; FIS, TRS. SWITCH FUNCTIONS SHALL BE FURTHER MODIFIED BY 'L' FOR LOW AND 'H' FOR HIGH.

6. THE DESIGNATION 'AJ( )' MAY DENOTE A SCANNING ANALYZER INDICATOR, RECORDER, TRANSMITTER, ETC. BY USING THE DESIGNATION AJI, AJR, AJT, ETC. RESPECTIVELY.

7. 'TW' DENOTES AN EMPTY THERMOWELL. 'TE' DENOTED A THERMOWELL WITH THERMOCOUPLE OR RTD IN A THERMOWELL OR A SURFACE MOUNTED SHEATHED THERMOCOUPLE.

8. FOR DEVICES OTHER THAN CONTROL VALVES, SUCH AS HYDRAULIC COUPLING, VARIABLE SPEED DRIVES, ETC.

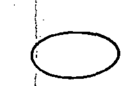
9. HIGH-HIGH ALARMS WILL BE DESIGNATED '( ) AHH' AND LOW-LOW ALARMS '( ) ALL'. FOR EXAMPLE: LAHH DENOTES 'HIGH-HIGH LEVEL ALARM'.

10. ( ) K DENOTES A CONTROL STATION FOR AN ANALYZER OR OTHER INSTRUMENT SUCH AS AK FOR A CHROMATOGRAPHY PROGRAMMER OR INFRARED CONTROL STATION WHICH IS SEPARATE FROM THE ANALYZER ITSELF.

11. WHERE SPECIAL DESIGNATION IS REQUIRED, PILOT LIGHTS SHALL BE IDENTIFIED WITH THE PARTICULAR VARIABLE LETTER, FOLLOWED BY SECOND LETTER 'L'.

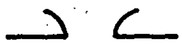
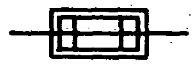
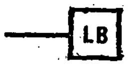
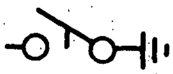
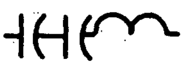
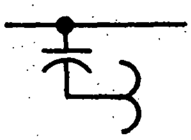
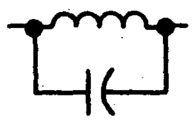
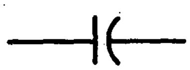
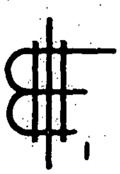
12. PRESSURE RELIEF VALVES AND RUPTURE DISKS SHALL BE IDENTIFIED AS 'PSV' AND 'PSE' RESPECTIVELY.

13. ET REPRESENTS A POTENTIAL TRANSFORMER 57

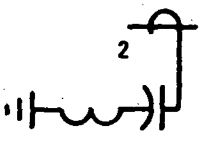
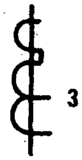
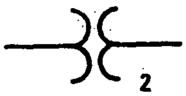
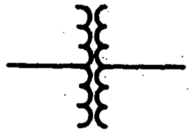
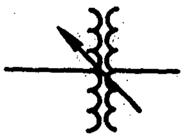
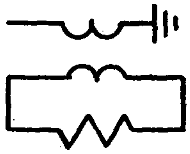
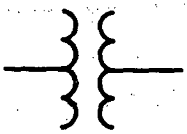
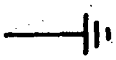
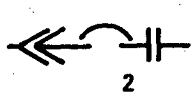
*  WHEN A, D OR E APPEAR IN THIS SYMBOL A = ANALOG; D = DIGITAL; E = EVENT; INPUT TO COMPUTER

INSTRUMENTATION IDENTIFICATION TABLE

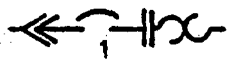
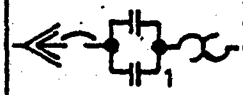
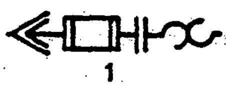
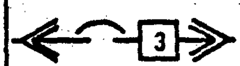
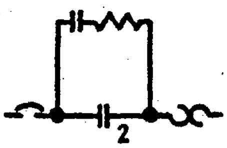
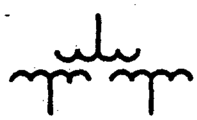
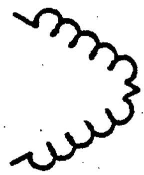
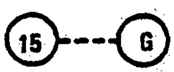
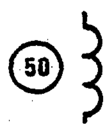
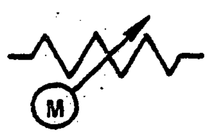
## ONE LINE & ELEMENTARY DIAGRAMS

| | | | | |
|---|---|---|---|---|
| |  | GANG OPERATED HORN GAP SWITCH |  | HIGH VOLTAGE PRIMARY FUSE |
| |  | GANG OPERATED DISC SWITCH |  | MOTOR OPERATOR |
| 57 |  | LOAD BREAK |  | SINGLE POLE |
| |  | GROUNDING SWITCH |  | LIGHTNING ARRESTER |
| |  | COUPLING CAPACITOR FOR CARRIER CURRENT |  | MEDIUM VOLTAGE COMBINATION FUSED DISCONNECT SWITCH & MOTOR CONTROLLER. FULL VOLTAGE. NON-REVERSING |
| |  | COUPLING CAPACITOR WITH POTENTIAL DEVICE |  | CARRIER CURRENT WAVE TRAP |
| |  | GROUND DETECTION CURRENT TRANS. NUMBER INDICATES QUANT. RATIO AS NOTED |  | CAPACITOR |
| 57 |  | GROUND DETECTION CURRENT TRANS. ZERO SEQUENCE TYPE. NUMBER INDICATES QUANTITY RATIO AS NOTED. | | |

## ONE LINE & ELEMENTARY DIAGRAMS (Continued)

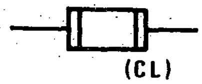
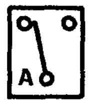
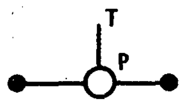
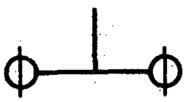
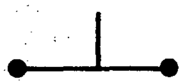
| | | | |
|---|---|---|---|
|  | BUSHING TYPE POTENTIAL DEVICE NUMBER INDICATES QUANT. RATING AS NOTED |  | BUSHING TYPE CURRENT TRANSF. NUMBER INDICATES QUANT. RATIO AS NOTED |
|  | POTENTIAL TRANSF. NUMBER INDICATES QUANT. RATING AS NOTED |  | PWR. TRANSF. SIZE & RATING AS NOTED. AA-OPEN DRY TYPE, GA-SEALED DRY TYPE, DA-NATURAL COOLING, FA-FORCED AIR (FAN) COOLING, FOA-FORCED OIL-AIR (PUMP & AIR) COOLING |
|  | REGULATING PWR. TRANSF. SIZE RATING AS NOTED |  | GROUND TRANSF. & RESISTOR, SIZE & RATING AS NOTED |
|  | GENERAL TRANSFORMER |  | REACTOR, SIZE & RATING AS NOTED |
|  | DISCONNECT LINKS |  | HIGH VOLTAGE CIRCUIT BREAKER, INTERRUPTING CAP., SIZE & RATING AS NOTED |
|  | STRESS CONE |  | CABLE TERMINATOR POTHEAD |
|  | GROUND CONNECTION |  | COMB.3POLE AIR CIRCUIT BREAKER & MAGNETIC CONTACTOR NUMBER INDICATES SIZE |

57

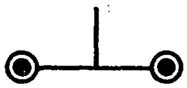
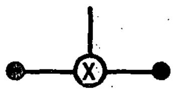
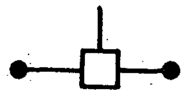
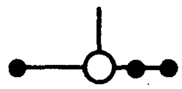
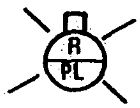
| | | | |
|---|---|---|---|
|  | DRAWOUT DISCONNECT DEVICE |  | 3-POLE CIRCUIT BREAKER DRAWOUT TYPE |
|  | PWR. OPER. AIR CIRCUIT BREAKER DRAWOUT TYPE, MAGNETIC OVER-CURRENT TRIP. AF-AMP FRAME, AT-AMP TRIP, M-MANNUALLY OPER. BREAKER, E-ELECTRICALLY OPER. BREAKER, NA-NON-AUTOMATIC, U-UNDERVOLTAGE ATTACHMENT |  | MOLDED CASE AIR CIRCUIT BREAKER 3 POLE W/THERMAL & MAGNETIC TRIP |
|  | COMB. 3 POLE AIR CKT. BKR., FULL VOLT., NON-REVERSING SINGLE SPEED STARTER W/THERMAL OVERLOAD ELEMENTS. DRAWOUT TYPE NO. INDICATES NEMA SIZE. |  | COMB. 3 POLE AIR CKT. BKR. & FULL VOLT. REVERSING SINGLE SPEED MAG. STARTER WITH THERMAL OVERLOAD ELEMENTS DRAWOUT TYPE, NO. INDICATES NEMA SIZE. |
|  | COMB. 3 POLE FUSED DISCONNECT SW. & FULL VOLTAGE NON-REVERSING SINGLE SPEED STARTER, W/THERMAL OVER-LOAD ELEMENTS. NO. INDICATES SIZE. |  | AIR CIRCUIT BKR. DRAWOUT TYPE SOLID STATE TRIPPING DEVICE NO. INDICATES QUANTITY OF POLES |
|  | PRIMARY RESISTOR REDUCED VOLTAGE STARTER. NO. INDICATES SIZE |  | THREE WINDING, $3\phi$ PWR. TRANSFORMER, SIZE & RATING AS NOTED |
|  | PHASE SHIFTING TRANSFORMER |  | MOTOR – NO. INDICATES HORSEPOWER |
|  | MOTOR GENERATOR SET SIZE & RATING AS SHOWN |  | SYNCHRONOUS MOTOR NO. INDICATES HORSEPOWER |
|  | RHEOSTAT - MANUALLY OPERATED |  | RHEOSTAT-MOTOR OPERATED |

Amend. 57
Nov. 1980

## ONE LINE & ELEMENTARY DIAGRAMS (Continued)

| | | | |
|---|---|---|---|
|  | BATTERY (MULTICELL) |  | FUSE - CL INDICATED CURRENT LIMITING |
|  | SHUNT |  | RESISTOR |
|  | RESISTANCE TEMP. DETECTOR |  | TRANSFER SWITCH M-MANUAL OPERATED, A-AUTOMATICALLY OPERATED |
|  | ELECTRICAL EQUIP. SPACE HEATER |  | POWER RECEPTACLE |
| 57  | GENERATOR – SIZE & RATING AS NOTED |  | CONTROL SW. W/RED & GREEN IND. LIGHTS. P INDICATES PERMISSIVE. T INDICATES COMBINATION CONTROL/TEST SW. |
|  | START – STOP MAINTAINED CONTACT PUSHBUTTON |  | START – STOP MAINTAINED CONTACT PUSHBUTTON W/RED & GREEN IND. LIGHTS |
|  | START – STOP MOMENTARY CONTACT PUSHBUTTON |  | RED & GREEN INDICATING LIGHTS |

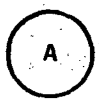
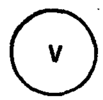
## ONE LINE & ELEMENTARY DIAGRAMS (Continued)

| | | | |
|---|---|---|---|
|  | START – STOP MOMENTARY CONTACT PUSHBUTTON W/RED & GREEN INDICATING LIGHTS |  | HAND–OFF AUTOMATIC SELECTOR SWITCH |
|  | SELECTOR SWITCH W/RED & GREEN INDICATING LIGHTS |  | TEST SWITCH W/RED & GREEN INDICATING LIGHTS |
|  | LOCKOUT PUSHBUTTON |  | CONTROL SWITCH W/RED GREEN & AMBER INDICATING LIGHTS |
| 57  | INDICATING LIGHTS WITH DROPPING RESISTOR. LETTER INDICATES COLOR |  | BELL ALARM CONTACT |
|  | ANNUNCIATOR POINT |  | COMPUTER INPUT OR OUTPUT SIGNAL |
|  | VARIABLE RESISTOR | N O | NORMALLY OPEN |
| N C | NORMALLY CLOSED | ---- E ---- | ELECTRICAL INTERLOCK |



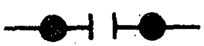
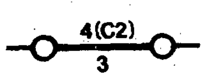
Amend. 57
Nov. 1980

## ONE LINE & ELEMENTARY DIAGRAMS (Continued)

| | | | |
|---|---|---|---|
|  | MECHANICAL INTERLOCK |  | KEY INTERLOCK |
|  | MANUAL MOTOR STARTER, I POLE W/THERMAL OVERLOAD |  | CLEAR INDICATING LIGHT |
|  | AMMETER |  | VOLT METER |

57

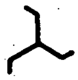
## ADDITIONALLY FOR ELEMENTARY BLOCK DIAGRAMS

| | | | |
|---|---|---|---|
|  | SOLID CIRCLE DENOTES TERMINATION FOR INTERNAL WIRING |  | OPEN CIRCLE DENOTES TERMINAL POINT FOR EXTERNAL WIRING |
|  | NUMBER ON TOP INDICATES CABLE NUMBER. NUMBER BELOW INDICATES CONDUCTOR NUMBER AND COLOR CODE OF THE CABLE ABOVE. EXAMPLE-NUMBER 3 EQUALS "RED". | | |
| | | | |

57

Amend. 57
Nov. 1980

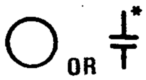
## ONE LINE & ELEMENTARY DIAGRAMS (TRANSFORMER CONNECTIONS)

| | | | |
|---|---|---|---|
|  | 3 $\phi$ ZIG-ZAG UNGROUNDED |  | 3 $\phi$ ZIG-ZAG GROUNDED |
|  | 3 $\phi$, 3W DELTA UNGROUNDED |  | 3 $\phi$, 3W DELTA GROUNDED |
|  | 3 $\phi$, 4W DELTA UNGROUNDED |  | 3$\phi$, 4W DELTA GROUNDED |
|  | 3 $\phi$ OPEN DELTA |  | 3$\phi$ OPEN DELTA, GND. AT COMMON PT. |
|  | 3 $\phi$ OPEN DELTA, GND. AT MID POINT |  | 3$\phi$ BROKEN DELTA |
|  | 3 $\phi$ WYE OR STAR, UNGROUNDED |  | 3$\phi$ WYE OR STAR, GROUNDED NEUTRAL |
|  | 3 $\phi$, 4W WYE OR STAR UNGROUNDED |  | 3$\phi$, 4W WYE OR STAR, RESISTANCE GND. NEUTRAL |

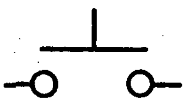
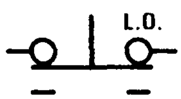
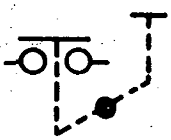
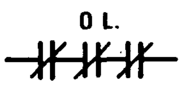
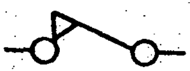
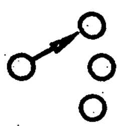
57

Amend. 57
Nov. 1980

## ELEMENTARY & 1 LINE DIAGRAMS (DEVICES)

| | | | |
|---|---|---|---|
|  | ASTERISK INDICATES PLACEMENT OF TYP. ABBREVIATION OF CONTROL DEVICE |   | AS-AMMETER SW.<br>VS-VOLTAGE SW.<br>SS-SYNCHRONIZING SW.<br>MSS-METERING SEL. SW. |
| (3)  | ASTERISK INDICATES PLACEMENT OF TYP. ABBREVIATION OF METER OR INSTRUMENT PREFIX R - RECORDING No. DENOTES QUANT. |  | ASTERISK INDICATES PLACEMENT OF TYP. ABBREVIATION OF RELAY OR DEVICE |

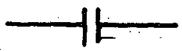
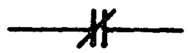
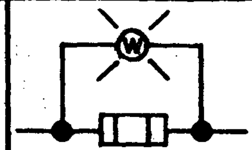
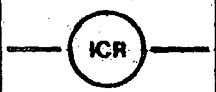
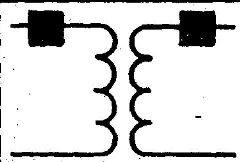
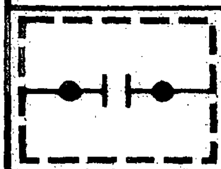
## ELEMENTARY & 1 LINE DIAGRAMS (SWITCH CONTACT & MISCELLANEOUS SYMBOLS)

| | | | |
|---|---|---|---|
|  | DIODE |  | PUSHBUTTON-MOMENTARY CONTACT, NORMALLY CLOSED |
|  | PUSHBUTTON-MOMENTARY CONTACT, NORMALLY OPEN |   | PUSHBUTTON LOCKOUT |
|  | PUSHBUTTON MAINTAINED CONTACT |   | CONTACTS OF OVERLOAD DEVICES |
|  | TORQUE LIMIT SWITCH |  | SELECTOR SWITCH (2 OR 3 POSITION) |
|  | THERMAL ELEMENT |  | PROTECTIVE RELAY OR SOLENOID |

57

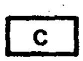
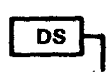
7366-28

## ELEMENTARY & 1 LINE DIAGRAMS (SWITCH CONTACT) (Continued)

| | | | |
|---|---|---|---|
|  | NORMALLY OPEN CONTACT (N.O.) |  | NORMALLY CLOSED CONTACT (N.C.) |
|  | MOTOR OPER. VALVE POS. LIMIT SWITCH |  | INDICATING TYPE FUSE |
|  | CONTACTOR OR AUXILIARY RELAY OPERATING COIL |  | TRANSFORMER WITH POLARITY SIGN |
|  | A DEVICE LOCATED IN A DIFFERENT COMPARTMENT WITHIN THE SWITCH GEAR OR MOTOR CONTROL CENTER | | |

57

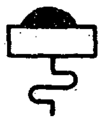
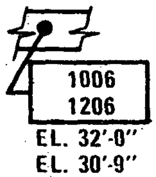
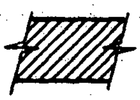
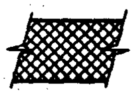
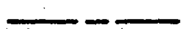
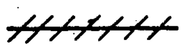
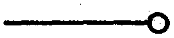
## POWER, GROUNDING & LIGHTING PLANS

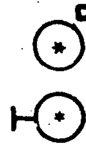
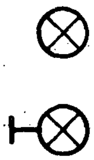
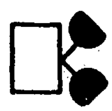
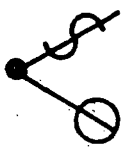
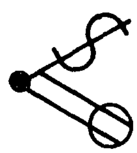
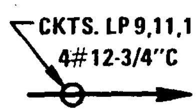
| | | | |
|---|---|---|---|
|  | LIGHTING PANEL |  | PANEL– MISCELLANEOUS |
|  | POWER DISTRIBUTION PANEL |  | MOTOR HORIZONTALLY MTD. |
|  | MOTOR VERTICALLY MTD. |  | TRANSFORMER– SIZE & RATING AS NOTED |
|  | CONTACTOR |  | MOTOR STARTER OR CONTROLLER |
|  | TRANSFER SWITCH SIZE & TYPE AS NOTED |  | DISCONNECT OR SAFETY SWITCH– SIZE & TYPE AS NOTED |

57

57

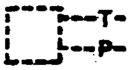
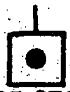
## POWER, GROUNDING & LIGHTING PLANS (Continued)

| | | | |
|---|---|---|---|
|  | POWER RECEPTACLE 100A, 480V, 3Φ, 4W |  | MOTOR OPERATED VALVE |
|  | SOLENOID OPERATED VALVE |  | UNIT HEATER |
|  | RADIANT HEATER |   | CABLE TRAY OR LADDER SYSTEM W/NUMBERS FOR COMPUTER CABLE LOADING. ELEV. ARE TO BOTTOM OF TRAY. |
|  | 5 KV BUS DUCT |  | BUS DUCT OVER 5 KV |
|   | DIRECT BURIAL CABLE |  | RIGID CONDUIT RUN EXPOSED |
|  | RIGID CONDUIT EMBEDDED IN CONCRETE |  | RIGID CONDUIT RUN CONCEALED |
|  | RIGID CONDUIT RUN BELOW EL. SHOWN |  | FLEXIBLE CONDUIT |
|  | CONDUIT OR CABLE TURNING UP OR TOWARDS OBSERVER |  | CONDUIT OR CABLE TURNING DOWN OR AWAY FROM OBSERVER |

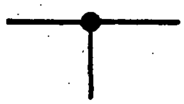
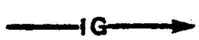
| | | | |
|---|---|---|---|
|  MTG. HGT. ABOVE FINISHED FLOOR | * INDICATES A LETTER WHICH IDENTIFIES FIXTURE TYPE AS SPECIFIED ON LIGHTING FIXTURE SCHEDULE |  CEILING C-2 WALL MTD. | LIGHTING FIXTURE WITH INCANDESCENT OR MERCURY LAMPS. *INDICATES A LETTER WHICH IDENTIFIES FIXTURE TYPE AS SPECIFIED ON FIXTURE SCHEDULE. "C-2" INDICATES POWER SUPPLIED FROM LIGHTING PANEL "C", CIRCUIT NO. "2". |
|  CEILING WALL | EXIT LIGHTING FIXTURE |  CEILING WALL | FLUORESCENT LIGHTING FIXTURE |
|  | BARE LAMP FLUORESCENT STRIP |  | AC/DC EMERGENCY LIGHTING UNIT |
|  | GOOSENECK LIGHTING STANCHION & FIXTURE |  | STREET LIGHTING FIXTURE |
|  | FLOODLIGHT FIXTURE |  a 3 | SINGLE POLE TOGGLE SWITCH a - IND. ASSOCIATED CONTROLLED FIXTURES 3 - IND. 3-WAY SWITCH |
|  | SWITCH & SINGLE CONVENIENCE RECEPTACLE COMBINATION |  | SWITCH & DUPLEX CONVENIENCE RECEPTACLE COMBINATION |
|  | RECEPTACLE - SINGLE CONVENIENCE, VERTICAL SLOTS, 120V, 20A, 3W, GNDED |  | RECEPTACLE - DUPLEX CONVENIENCE, VERTICAL SLOTS, 120V, 20A, 3W, GNDED |
|  | RECEPTACLE - SINGLE PHASE, HORIZONTAL SLOTS, 208V, 20A, 3W, GNDED |  CKTS. LP 9,11,15 4#12-3/4"C | HOMERUN TO PANELBOARD - ALL UNMARKED CONDUITS ARE 3/4" & CONTAIN 2 #12 UNLESS OTHERWISE NOTED. |

Amend. 57
Nov. 1980

## POWER, GROUNDING & LIGHTING PLANS (Continued)

| | | | |
|---|---|---|---|
|   | UNDERFLOOR DUCT<br>W/JUNCTION BOX<br>T-TELEPHONE DUCT<br>P-POWER DUCT |   | J-IND. JUNCTION BOX<br>TB-IND. TERMINAL BOX<br>PB-IND. PULL BOX<br>ADD BOX NUMBER IF<br>REQUIRED |
| <br>PB STA. | PUSHBUTTON STATION | | |

## POWER, GROUNDING & LIGHTING PLANS (GROUNDING)

| | | | |
|---|---|---|---|
|  | GROUND ROD |  | GROUND CONNECTION<br>THERMIT WELD PROCESS |
|   | ANNEALED, BARE<br>STRANDED COPPER GND.<br>CABLE RUN EXPOSED,<br>SIZE AS INDICATED |   | GROUND CABLE<br>RISER UP |
|   | GROUND CABLE<br>RISER DOWN |   | ANNEALED, BARE<br>STRANDED COPPER GND.<br>CABLE RUN CONCEALED,<br>SIZE AS INDICATED |
|  | GROUND CABLE RISER<br>FROM UNDERMAT GND.<br>GRID PER |   | GROUND CABLE RISER,<br>____FT. LONG, TERMINATED<br>AT GRADE FOR FUTURE<br>CONNECTION |
|  | PILE WITH GROUND WIRE |  | GROUND TEST BOX |
| 7   | CONNECTION TO TRAY<br>GROUND |   | CONNECTION TO<br>INSTRUMENT GROUND |

## CONTROL DEVICE CONTACTS (ELEMENTARY)

| —\|\|—* INDICATES PLACEMENT OF CONTROL DEVICE ABBREVIATION (SAME AS BELOW) | CONTROL DEVICE CONTACTS (ELEMENTARY) | | |
|---|---|---|---|
| FLS *COIF FLS *OOIF | FLOW SWITCH | TS *CORT TS *OORT | TEMPERATURE SWITCH |
| COS | CUT-OUT SWITCH | LS *CORL LS *OORL | LEVEL SWITCH |
| 57 PS *CORP PS *OORP | PRESSURE SWITCH | CS | CONTROL SWITCH |
| T | THERMOSTAT | LMS | LIMIT SWITCH |
| TDC<br>TDO | TIME DELAY CLOSE<br>TIME DELAY OPEN | *C01F | INDICATES CLOSES ON INCREASE OF FLOW |
| DPS | DIFFERENTIAL PRESSURE SWITCH | *001F | INDICATES OPENS ON INCREASE OF FLOW |
| EPS | ELECTRO-PNEUMATIC SWITCH | *CORP | INDICATES CLOSES ON RISING PRESSURE |
| H | HUMIDISTAT | *OORP | INDICATES OPENS ON RISING PRESSURE |
| PMS | PERMISSIVE SWITCH | *CORT | INDICATES CLOSES ON RISING TEMPERATURE |
| INST | RELAY INSTANTANEOUS CONTACT | *OORT | INDICATES OPENS ON RISING TEMPERATURE |
| 57 *OORL | INDICATES OPENS ON RISING LEVEL AS TABULATION OR TO APPLICABLE DESCRIPTION COLUMNS. | *CORT | INDICATES CLOSES ON RISING LEVEL |

Amend. 57
Nov. 1980

## DEVICE ABBREVIATIONS (ELEMENTARY)

| | | | |
|---|---|---|---|
| SE | SPECIAL EQUIPMENT FURN. BY MFGR. | PE | PNEUMATIC-ELECTRIC RELAY |
| PT | PRESSURE TRANSMITTER | TE | TEMPERATURE ELEMENT |
| FI | FLOW INDICATOR | FT | FLOW TRANSMITTER |
| TDDO | RELAY TIME DELAY DROPOUT | LT | LEVEL TRANSMITTER |
| T/C | THERMOCOUPLE | TDPU | RELAY TIME DELAY PICKUP |
| TZ | TRANSDUCER | | |
| | | | |
| | | | |
| | | | |
| | | | |
| | | | |
| | | | |
| | | | |
| | | | |
| | | | |
| | | | |
| | | | |
| | | | |
| | | | |

57

Amend. 57
Nov. 1980

## POWER, GROUNDING & LIGHTING PLANS (DEVICE ABBREVIATIONS)

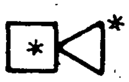
57

| | | | |
|---|---|---|---|
| XFMR | TRANSFORMER | I | INTERLOCK |
| SWGR | SWITCHGEAR | STR | STARTER |
| MCC | MOTOR CONTROL CENTER | HPO | HEALTH PHYSICS OFFICE |
| PC | POWER PANEL (AC) | RHCP | REHEATER CONTACTOR PANEL |
| PD | POWER PANEL (DC) | ATP | AUTOMATIC TEMPERATURE CONTROL PANEL |
| LC | LIGHTING PANEL (AC) | PT | POTENTIAL TRANSFORMER |
| LD | LIGHTING PANEL (DC) | MTS | MANUAL TRANSFER SWITCH |
| GND | GROUND | DB | DIRECT BURIAL CABLE |
| C | CONDUIT | FPC | LOCAL FIRE PROTECTION PUMP CONTROLLER |
| EP | EXPLOSION PROOF | CT | CURRENT TRANSFORMER |
| WP | WEATHER PROOF | POS | POSITIVE |
| VT | VAPOR TIGHT | NEG | NEGATIVE |
| EC | EMPTY CONDUIT | RL | REMOTE LOCATION |
| DT | DUST TIGHT | SPT | SEQUENTIAL PROGRAM TIMER |
| WT | WATERTIGHT (SUBMERSIBLE) | MR | MULTIPLE RATIO |
| RP | RELAY PANEL | IP | ISOLATED PHASE BUS DUCT |
| EBB | ELECTRICAL BENCH BOARD | BD | BUS DUCT |
| MBB | MECHANICAL BENCH BOARD | CP | CONTROL PANEL |
| LTU | LINE TUNING UNIT | PE | PHOTOELECTRIC |

57

Amend. 57
Nov. 1980

## COMMUNICATIONS

| | | | |
|---|---|---|---|
|  | SPEAKER-*REPRESENTS LETTER CORRESPONDING TO SPEAKER AND/OR AMPLIFIER TYPE |  | HANDSET-*REPRESENTS LETTER(S) CORRESPONDING TO HANDSET TYPE |
|  | SPEAKER AMPLIFIER, *REPRESENTS LETTER CORRESPONDING TO SPEAKER AMPLIFIER TYPE |  | SPEAKER, *REPRESENTS LETTER CORRESPONDING TO SPEAKER TYPE |

**SPEAKER TYPE** (USED WITH FOLLOWING DESIGNATIONS)

- **A** DIRECTIONAL TRUMPET, 85° SOUND DISPERSION WITH 15″ HORN (APPROX.); 20″ BELL DIAMETER (APPROX.) & 30W. DRIVER.
- **B** PAGING/TALK BACK SPEAKER, 105° SOUND DISPERSION, 9″ HORN (APPROX.) 10″ BELL DIAMETER (APPROX.)
- **C** WALL MOUNTED CONE SPEAKER ASSEMBLY, WALNUT FINISHED SPEAKER BAFFLE WITH 8 Ohm. 8″ DIAMETER (APPROX.) CONE SPEAKER AND VOLUME CONTROL.
- **D** CORRIDOR TYPE, BI-DIRECTIONAL BAFFLE WITH 8″ DIAMETER SPEAKER AND VOLUME CONTROL.
- **E** FLUSH, WALL OR PANEL MOUNTED CONE SPEAKER ASSEMBLY WITH PROJECTING BAFFLE, 8″ DIAMETER SPEAKER WITH VOLUME CONTROL.
- **F** FLUSH, WALL OR PANEL MOUNTED CONE SPEAKER ASSEMBLY WITH FLUSH BAFFLE, 8″ DIAMETER SPEAKER
- **G** MULTI-DUTY WEATHERPROOF HIGH-FREQUENCY SPEAKER, 120° SOUND DISPERSION.
- **H** DUAL, WIDE ANGLE HORN SPEAKER, 120° × 60° SOUND DISPERSION, 10″ HORN (APPROX.) 18″ × 9″ BELL DIAMETER (APPROX.) WITH 30W. DRIVER.

**SPEAKER AMPLIFIER** (USED WITH FOLLOWING DESIGNATIONS)

- **M** SPEAKER AMPLIFIER ASSEMBLY PUSH-PULL, CLASS B, 12WATT AMPLIFIER WITH INDIVIDUAL VOLUME CONTROL; BAKED ENAMEL ON ZINC CHROMATE PHOSPHATE BONDED ENCLOSURE.
- **N** SPEAKER AMPLIFIER ASSEMBLY (AMPLIFIER TYPE SAME AS M) WEATHERPROOF ENCLOSURE.

**HANDSET TYPE** (USED WITH FOLLOWING DESIGNATIONS).

- **Q** DESK-TOP STATION WITH REMOTE HANDSET SPEAKER, SPEAKER AMPLIFIER, VOLUME CONTROL & ENCLOSURE.
- **K** WALL STATION WITH HANDSET, SPEAKER AMPLIFIER & ENCLOSURE.
- **R** DESK EDGE STATION WITH SUBSET, REMOTE HANDSET, SPEAKER AMPLIFIER & ENCLOSURE.
- **S** FLUSH PANEL STATION WITH SUBSET, REMOTE HANDSET, SPEAKER AMPLIFIER & ENCLOSURE.
- **T** WEATHERPROOF WALL STATION: WITH HANDSET, SPEAKER AMPLIFIER & WEATHERPROOF ENCLOSURE.

57

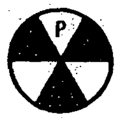
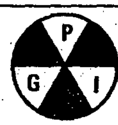
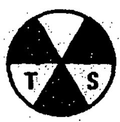
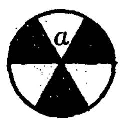
## TELEPHONE

| | | | |
|---|---|---|---|
|  | DIRECT DISPATCH TELEPHONE |  | COMMERCIAL TELEPHONE |
|  | LOAD DISPATCH INTERCOM SYSTEM |  | TELEPHONE SWITCHBOARD |
|  SP | SOUND POWERED TELEPHONE JACK |  | PRIVATE AUTOMATIC EXCHANGE (PAX) TELEPHONE<br>D – INDICATES DESK MTD<br>W – INDICATES WALL MTD |
| | |  | SOUND PROOF ENCLOSED HANDSET |

57

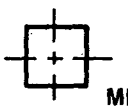
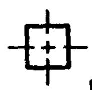
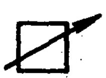
## COMMUNICATIONS (CONT.)

### RADIATION MONITORS

| | | | |
|---|---|---|---|
|   | AREA |   | PARTICULATE |
|   | HAND & FOOT |   | LIQUID MONITOR |
|   | GAS MONITOR |   | FRISKER MONITOR |
|   | LAUNDRY INSPECTION MONITOR |   | PARTICULATE/GASEOUS MONITOR |
|   | PARTICULATE SAMPLER |   | PARTICULATE/GASEOUS/IODINE MONITOR |
|   | TRITIUM SAMPLER |   | ALPHA MONITOR |

57

### UNDERGROUND DISTRIBUTION PLANS

| | | | |
|---|---|---|---|
|   | MANHOLE |   | HANDHOLE |
|  | DUCTBANK |  | SEISMIC JOINT |
|   | SINGLE PHASE TRANSFORMER SIZE & RATING AS NOTED |   | THREE PHASE TRANSFORMER SIZE & RATING AS NOTED |
|  | STREET LIGHTING REGULATOR RATING AS NOTED | | |



Amend. 57
Nov. 1980

## HAZARD MONITORS—LOOP/LOGIC USE

| | | | |
|---|---|---|---|
| HF VAPOR AE | HF ACID VAPOR MONITOR | FIRE—IR AE | INFRARED FLAME DETECTOR |
| FIRE—I AE | IONIZATION SMOKE DETECTOR | FIRE—T AE | THERMAL HEAT DETECTOR— FIXED TYPE |
| FIRE—P AE | PHOTOELECTRIC SMOKE DETECTOR | FIRE—TL AE | THERMAL HEAT DETECTOR— LINE TYPE |
| FIRE—UV AE | ULTRAVIOLET FLAME DETECTOR | FIRE—TR AE | THERMAL HEAT DETECTOR— RATE OF RISE TYPE |

## HAZARD MONITORS—SYSTEM ARRANGEMENT USE

| | | | |
|---|---|---|---|
| AREA I | IONIZATION SMOKE DETECTOR | IR | INFRARED FLAME DETECTOR |
| DUCT I | IONIZATION SMOKE DETECTOR (ON HVAC DUCTING) | FIXED T | THERMAL HEAT DETECTOR— FIXED TYPE |
| P | PHOTOELECTRIC SMOKE DETECTOR | RATE T | THERMAL HEAT DETECTOR— RATE OF RISE TYPE |
| UV | ULTRAVIOLET FLAME DETECTOR | LINE T | THERMAL DETECTOR— LINE TYPE |

57

CHAPTER 2.0 - SITE CHARACTERISTICS

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)





Amend. 71
Sept. 1982

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

## LIST OF TABLES

## LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

## LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)





Amend. 71
Sept. 1982

## CHAPTER 2.0 - SITE CHARACTERISTICS
## LIST OF FIGURES

LIST OF FIGURES (Continued)

## LIST OF FIGURES (Continued)





Amend. 71
Sept. 1982

LIST OF FIGURES (Continued)

## LIST OF FIGURES (Continued)

## LIST OF FIGURES (Continued)

## LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

LIST OF FIGURES (Continued)

## LIST OF FIGURES (Continued)

## LIST OF FIGURES (Continued)

## CHAPTER 2.0 - SITE CHARACTERISTICS
## LIST OF REFERENCES

## CHAPTER 2.0 SITE CHARACTERISTICS

This Chapter of the SAR provides information on the geological seismological, hydrological, and meteorological characteristics of the site and vicinity in conjunction with population distribution, land use, and site activities and controls. The purpose is to indicate how these site characteristics have influenced plant design and operating criteria and to show the adequacy of the site characteristics from a safety viewpoint.

### 2.1 GEOGRAPHY AND DEMOGRAPHY

Section 2.1 provides information concerning a description of the Site location and layout. The geography of the Site is described. Exclusion area control and Site boundaries are discussed. Population distribution, permanent and transient, based on 1970 Census figures and projections thru the year 2010 are provided. Recreation activities within the adjacent lands are described as well as public facilities and institutions. The agriculture of the adjacent lands is also discussed. Uses of nearby bodies of water are also described.

#### 2.1.1 Site Location and Layout

The Clinch River Site is in east central Tennessee in the eastern
15| part of Roane County and within the city limits of Oak Ridge approximately
25 miles west of Knoxville as shown in Figure 2.1-1. The Site is on a
peninsula bounded on the south by the Clinch River between Clinch River
| Mile (CRM) 14.6 and CRM 18.6 and on the north by ERDA's Oak Ridge Reservation. Location of the Site with respect to proximity to populated areas,
railroads, highways and other features is shown in Figure 2.1-2.
| Figure 2.1-2a shows the residential housing zone within the city of Oak Ridge |
15| closest to the CRBRP site. An aerial photograph of the site is shown in |7
Figure 2.1-3.

The coordinates of the center of the containment location for the CRBRP are given below in both latitude and longitude and Universal Transverse Mercator (UTM) coordinates. Latitude and longitude are given to the nearest second and UTM coordinates are given to the nearest 100 meters:

| Latitude and Longitude | UTM Coordinates |
|---|---|
| $35^{o}$ 53' 24" N<br>x$84^{o}$ 22' 57" W | $39_{74}$ 709 N x $7_{36}262_{E}$ |

Plant location is illustrated on figure 2.1-5. It will be situated at 815 feet above mean sea level, placing it 74 feet above the mean Clinch River water level of 741 feet. Chestnut Ridge extends across the north edge of the Site at an elevation of 1100 feet, effectively screening the Site from the industrial park and the Gaseous Diffusion Plant. Pine Ridge, which starts about 2.5 miles north of the Site and runs in a northeasterly direction at an average elevation of 1,100 feet, screens the Site from the city of Oak ridge. A portion of the dome of the Reactor Containment Building may be visible to traffic crossing the Gallagher Bridge on the Oak Ridge Turnpike (Figure 2.1-5) and approximately ten homes on the southern side of the Clinch River will have a limited view of the plant. Figure 2.1-2 shows the relative location of major highways while Figure 2.2-2 contains a topological description of the Site.

### 2.1.2 Site Description

The Clinch River Site is made up of 1364 land acres on a peninsula on the north side of the Clinch River (see Figure 2.1-4). The Site is bounded on the north by ERDA's Oak Ridge Reservation and on the south by the Clinch River. The property is owned by the United States of America and is presently in the custody of the Tennessee Valley Authority (TVA). TVA will transfer to ERDA the custody of those portions of the Site which are reasonably required for the purpose of designing, constructing and operating the CRBRP.

Of the 1364 acres within the Site boundaries, approximately 100 acres will be required for the Clinch River Breeder Reactor Plant (CRBRP) and related facilities such as roads, railroads, and transmission corridors. The Reactor Containment Building and its auxiliary buildings will occupy about four acres. The plant and all associated systems have been located as shown in Figures 2.1-5 and 2.1-5a.

A portion of the Site to the north of the plant, between Bear Creek Road and Grassy Creek, has been set aside for industrial development and is called the Clinch River Consolidated Industrial Park (CRCIP). Approximately 112 acres will be occupied by the CRCIP.

All activities within the Site boundaries (except the CRCIP) will be under the jurisdiction of the applicant.

28

#### 2.1.2.1 Exclusion Area Control

The area in which exclusion control can be exercised in the unlikely event that the need for such control arises is shown in Figure 2.1-5. This area includes the Site (except the CRCIP) and the river adjacent to the Site; the minimum exclusion distance will be 2200 feet from the Containment Building. Control within the Site boundaries (except the CRCIP) will 28
be by the applicant; control of the waterway will be coordinated with the appropriate agency or agencies necessary to guarantee such control as described in the radiological emergency plan presented in Section 13.3 of the PSAR

All activities within the Site (except the CRCIP) are under the jurisdiction of the applicant. The Hensley Cemetery is located on the southern end of the

Site. Hensley heirs will be allowed access for the purpose of visiting and maintaining the cemetery; this access will be controlled by the applicant. There is also an Indian Mound nearby that is of archeological interest. Detailed investigation of the mound is scheduled to be completed prior to start of construction. If further visitation to or study of this mound is necessary, it will be controlled by the applicant.

Road access to the Site will be via River Road from the northwest. At the east boundary of the Site, River Road will be barricaded and locked, to be opened only in case of emergency. The river bank on the plant side will be appropriately marked and posted to prevent any private or commercial use thereof. Activities carried out in the Clinch River may include private or commercial river traffic such as fishing, boating and barge transportation. Signs will be posted to inform river users of the nearby nuclear plant.
15 Railroad access to the Site will be provided from the ERDA's existing facilities at the Oak Ridge Gaseous Diffusion Plant to the north of the Site and will run parallel to River Road and the Site access road, as shown in Figure 2.1-5.

2.1.2.2 <u>Boundaries for Establishing Effluent Release Limits</u>

A protected area will be located within the exclusion area; at no point will the boundary lines of the two areas be coincident. The protected area will enclose all systems and auxiliaries that are essential to safe operation and shutdown of the plant and is shown in Figure 2.1-5 indicated by a security fence that will enclose the protected area.

A patrol road will be provided inside the fence with a clear, unobstructed view inside and outside of the fence. Cleared areas will be maintained on both sides of the fence for a distance of at least 20 feet outside and 50 feet inside the security barrier. The fence and the areas on both sides will be lighted and monitored. Access into the plant will normally be through one guarded gateway, for personnel and vehicular
15 traffic. Railroad access into the Site will be provided from ERDA's existing facilities at the Oak Ridge Gaseous Diffusion Plant to the north of the site and will run parallel to the Site access road. Rail access will be unlocked only when admitting rail cars; the entrance gate will be attended by a guard when unlocked.

The boundary for which routine gaseous effluent dose calculations are performed is shown in Figure 2.1-4. The restricted area boundary (as defined in 10 CFR 20) is defined to be the same as the Site boundary (except the CRCIP) and will be appropriately posted. Doses corresponding to routine 28
releases are computed at a downwind distance of 1800 feet, the shortest distance from the center of containment to the near bank of the Clinch River.

As discussed in Section 15.5, CRBRP PSAR accident analyses will include dose calculations for distances associated with the minimum
32 exclusion distance (2200') and the Low Population Zone (2.5 miles). For comparison purposes, safety calculations for the FFTF project include analyses of doses for 1-1/2 miles and 4-1/2 miles from its reactor containment. The 1-1/2 mile calculation corresponds to the minimun distance from the reactor 8

containment to the site exclusion boundary allowing for future development of the Hanford Reservation as a nuclear center. The 4-1/2 mile calculations correspond to the closest distance of approach by the public
from containment to the Hanford Reservation Boundary. 8

### 2.1.3 Population and Population Distribution

Table 2.1-1 shows the location and size of urban centers (population above 2,500) within a 50-mile radius of the Site. It should be noted that there are only 21 such urban centers and of these, two, Knoxville (population 174,587) and Oak Ridge (population 28,319) have populations exceeding 25,000. The 27 urban centers with populations less than 2,500 within the 50-mile radius are listed in Table 2.1-2.

Further discussion of the 1970 population distribution (Ref. 1) and detailed breakdown of the population into radial-azimuthal sectors within the 50-mile radius follows. Also a projection of this population distribution for the year 2010 is provided and an assessment of the magnitude of the transient population is presented. The projected population figures were baseline values published by the Tennessee Valley Authority, Division of Navigation Development and Regional Studies, Economic Research Staff in October, 1972 (Ref. 2). In providing these values consistent with national, regional, and OBE Economic Area totals, sets of "first approximations" were developed by a Federal team comprising economists from the Bureau of Economic Analysis; the Corps of Engineers; the Tennessee Valley Authority; and the Environmental Protection Agency. These first approximations were then correlated with independently developed county baseline value to be used for each county. Distribution into the sections required for this report was accomplished using local urban vs. rural growth patterns.

#### 2.1.3.1 Resident Population Within Ten Miles

The urban centers of Lenoir City, Kingston, Harriman and Oak Ridge
15 are located within 10 miles of the plant as shown in Figure 2.1-2. Figure
2.1-6 shows the comparative population distribution for the CRBRP area
compared to three other nuclear generating plants.

A detailed analysis of the population distribution within the 10-mile
radius was performed. For this purpose, the region surrounding the plant was
divided into sixteen 22-1/2° azimuthal sectors with radial increments of 1,
2, 3, 4, 5 and 10 miles as illustrated in Figure 2.1-7. The results of this
analysis, as presented in Table 2.1-3, show that the 1970 population out to
10 miles is 41,895, corresponding to an average population density in this
area of 133 persons/square mile. The 1970 population distribution in
15 each azimthal sector for the city of Oak Ridge is shown on Figure 2.1-2a.

Within 5 miles of the Site, there are no significant concentrations
of population. Approximately one-third of this area comprises land owned
by the U.S. government and in custody of the ERDA or TVA (including the
15 Clinch River Site)and is within the city limits of Oak Ridge. Nevertheless,
at least two-thirds of the resident population of Oak Ridge is located

the city of Oak Ridge became self-supporting and self-governing. At this time, the entire "Oak Ridge Reservation" (the entire 80,000 acres) was designated as the city of Oak Ridge though the major portion of the "city" available for the residential development is limited because much of the land is reserved for Government use. The total population of the city was 28,319 in 1970. Harriman is cut by the 10-mile radius to the west-northwest and contains 8,734 people. Two smaller towns are located slightly closer to the Site. Lenoir City is about 9 miles south-east of the Site with 5,324 people and Kingston is about 7 miles to the west with a population of 4,142 in 1970.

Development trends and potential indicate very little change within a 5 miles radius during the lifetime of the Clinch River Plant. The development patterns for the communities in the 5- to 10-mile range indicate that only Oak Ridge has the potential for growth in the direction of the site. The long-range pattern for Oak Ridge (probably beyond 1990) could result in further concentrated development but no closer than 5 miles to the Site due
15 to present zoning, as shown in Figure 2.1-2a.

The 1960 and 1970 census data show that the rural population within 10 miles of the Site has remained essentially constant. The population growth has taken place near present urban centers of Oak Ridge and Kingston. Projections for future growth include Oak Ridge, Harriman, Lenoir City and Kingston.

Population distribution was also projected to the year 2010 (Ref. 2).

The results of this analysis of projected population distribution within 10 miles of the plant are shown in Tables 2.1-4 through 2.1-7 from 1980 thru 2010. From these projections, it is seen that the population within 10 miles of the Site is expected to grow from its present level of 41,895 in 1970 to 65,089 in 2010.

### 2.1.3.2 Resident Population Between 5 and 50 miles

Figure 2.1-9 shows the urban centers of population in relation to the Site. Virtually all of the area within 50 miles of the Site is within Tennessee. Only a small portion of North Carolina and Kentucky are included. The total 1970 population of the 23 counties having 5 percent or more of their population within 50 miles of the Site was 768,955. This is an increase of only 5.9% over 1960 as compared with a population increase of 10% for Tennessee and 13% for the nation. Population distribution for 1970 is shown in Table 2.1-8. Projected population distribution from 1980 thru 2010 is shown in Tables 2.1-9 thru 2.1-12. The Site is located within the city limits of Oak Ridge, as described earlier, with a 1970 population of 28,319. At least two-thirds of the resident population of Oak Ridge is located beyond the 10-mile radius.

One major urban concentration (a population of 50,000 or more) is located within 50 miles of the Site. It is the Knoxville area in the 20- to 30-mile range to the east-northeast with a 1970 population of 174,587.

In addition to Oak Ridge, two smaller population centers (population of 10,000 to 50,000) are within 50 miles. They are the Maryville-Alcoa-Eagleton Village area and Athens. The Maryville-Alcoa-Eagleton Village area is in the 20- to 30-mile range to the east-southeast and contains 26,892 people. Athens is in the 30- to 40-mile range to the south-southwest with a 1970 population of 11,790.

TVA is studying the possibility of development of Timberlake, a new town on the shores of the Tellico Reservoir which will be formed when construction of the dam is completed. Population projections for Timberlake are 3,000 to 5,000 in 1980 and 12,000 to 18,000 in 1990. The range is dependent primarily on general economic conditions. This population should be allocated to the 10- 20-mile SSE sector but it is not included in the data in Tables 2.1-4 through 2.1-7 because of the tentative nature of the plans.

### 2.1.3.3 Low Population Zone

The low population zone associated with the CRBRP extends radially out 2.5 miles from the site. Several smaller communities and crossroads settlements are scattered throughout this region and are surrounded by low density rural development. As shown in Tables 2.1-4 through 2.1-7, zero population growth is expected within 2.5 miles of the site during the life of the plant.

The 2.5 mile designation as the low population zone is based on, and consistent with, the definition offered in 10CFR100. 32

### 2.1.3.4 Transient Population

An investigation was made of the various activities conducted within 10 miles of the Site to determine the magnitude of the transient population in the area. It was established that no major sport facilities or prisons exist within this region and that the only components of the transient population that are significant involve school and industrial activities during week days, and recreational activities primarily over weekends and holidays (Refs. 3,4,5).

Most of the transient population near the Site is due to industrial activities in the area which is discussed in Section 2.2. In addition, some recreation activities occur which add somewhat to the population. Within one mile there is only one sparsely used informal access and bank fishing area at the end of a dirt road. There are two similar informal use areas in the 1- to 3-mile range. There is a 30-unit commercial camping and day use area located about 2-3/4 miles southeast of the Site. The maximum number of people (at any one time) at this camp site

is estimated to be 80 in 1980 and 100 in 1990. A 100-unit commercial
camping site has been developed on the Caney Creek embayment near CRM
15 17. The camp is located slightly over one mile from the southeast boundary
of the Site and estimates of the maximum number of people at this camp site
(at any one time) for 1980 and 1990 are 270 and 340 people, respectively.
Activities at the camp site will include fishing, boating and swimming.
There is a small track where stock cars are raced located southeast of the
site within three miles which may attract 5,500 to 6,000 fans at present
and an estimated 6,500 persons by 1990. The transient population within
5 miles of the Site is shown in Table 2.1-13.

Recreational areas within a 10-mile radius of the Site are shown in Figure 2.1-10. Table 2.1-14 shows the approximate mileage from the Site to each recreational area along with the estimated number of persons that were on each site during peak hour use and the type of site activity. Peak hour use was considered to be July 4 for each year. Projections for peak hours for the years 1980,1990,2000 and 2010 are included. Based on 1970 information, the peak hour recreational use of these facilities could result in 3,565 persons being present, within 10 miles of the Site. This number is increased to 12,885 for the year 2010 projections. Assuming that these visitors resided outside the 10-mile radius from the Site, this would only represent an increase of about 8 percent over the permanent population in the area.

The number of recreational craft locked through the Melton Hill
Dam located about 4.5 miles east of the Site or 6 miles up the Clinch
River for the years 1966-1975 are shown in Table 2.1-15. The total number
15 of visitors to the Melton Hill Dam in 1971 was 225,000 while the total
number of visitors to the Dam since the project opened in 1963 was 2,596,000.
15 The commercial freight traffic on the Clinch River is very sparse and has not
exceeded 10,100 tons in any single year over the past ten years.

15

### 2.1.3.5 Population Center

The nearest population center to the CRBRP is the City of Oak
Ridge with a 1970 population of 28,319. (Projected 2010 population of 54,500).
The population center distance as defined in 10CFR100 is 7.0 miles in the
32 NNE direction (Oak Ridge Country Club).

### 2.1.3.6 Public Facilities and Institutions

Twenty- wo schools located within a 10 mile radius of the Site
15 (Figure2.1-11,) had a 1971 total enrollment of 7901 students (Table 2.1-16).
Oak Ridge anticipates building a new elementary school for their system by
1990. However, this school and other new schools in the foreseeable future
will be built to replace plants presently in use as they become obsolete.

The other school systems included within the 10-mile radius do not forecast any significant expansion beyond that necessary to meet future educational requirements as obsolete plants and facilities are retired or renovated.

According to the Journal of the American Hospital Association,
(Ref. 6) the nearest hospital to the Site is the Harriman City Hospital with
109 beds, located about 10 miles to the west-northwest. The Loudon County
Memorial Hospital, with 50 beds, is located about 10 miles to the south-southeast
and Oak Ridge Hospital of the United Methodist Church, with 220 beds, is
located about 15 miles to the northeast. A tabulation of additional hospital
facilities and their respective capacities within 50 miles of the Site is
shown in Table 2.1-17. No new hospitals are planned within the 10-mile
radius in the foreseeable future. Two primary reasons for this are the scarcity
of medical doctors at the local level and the proximity to the well equipped and
15 staffed hospitals in nearby Knoxville.

Forecasts for public and private recreational area are given in Section 2.1.3.4 and Table 2.1-14.

## 2.1.4 Uses of Adjacent Lands and Waters

The region within a 10-mile radius of the Site encompasses residential, farm, recreation and industrial areas. Schools and hospitals are the only public facilities located within the 10-mile radius. They are shown in Tables 2.1-16 and 2.1-17. There are no airports within the 10-mile radius of the site; it is served primarily by a highway system. The industrial and recreation areas are listed in Section 2.2.1.1 and Table 2.1-14. The eastern Tennessee area within the 10-mile radius has only five commercial dairy farms. There is no mineral production within the 10-mile radius; however, mineral production, primarily in the form of strip mining, does play an important role in the area, particularly in Morgan County. Transportation use of adjacent lands and waters is discussed in Section 2.2.1.3.

### 2.1.4.1 Agriculture

The majority of the region within the 10-mile radius lies within
Roane county, touching only slightly in Morgan, Anderson, Knox and Loudon
15 Counties. Checks with county agents have revealed that there are no farms
located within the 10-mile radius in Morgan, Anderson or Knox Counties (Ref. 7).
There are four dairy farms in Roane County and one dairy farm in Loudon County within 10 miles of the plant; these are shown in Figure 2.1-12 and tabulated in Table 2.1-18. Additionally, the Agricultural Research Laboratory operated by the University of Tennessee is located at the intersection of Bethel Valley and Scarboro Roads in the northeast sector. Since this falls outside the 10-mile radius, the animal population has not been surveyed. During a survey conducted in the spring of 1974, approximately 475 head of beef were counted within 5 miles of the Site. Scattered herds, ranging in size from 20 to 30 head were located in the southeast, southwest, and northwest quadrants. Interspersed with the beef cattle were 61 milk cows. Agricultural crops within the 10 mile radius were reported



Amend. 15
Apr. 1976

15 as grown on scattered small plots for single family use.

In general, farming in eastern Tennessee has followed the national trend of a steadily decreasing number of farms with the remaining farms increasing in average size (Ref. 8). Figure 2.1-13 shows this trend for the Emory River Valley and the county agents agree that this trend is valid in the local area. Because more off-farm employment opportunities exist now than in the past, the trend has been to shift from dairy cows and other forms of farming to raising beef cattle which requires much less labor. The trend mentioned for the Emory River Valley, as shown in Figure 2.1-14, shows that beef cattle production has doubled from 1939 to 1964. The latest agricultural census, 1969, shows that the trend has accelerated and that beef cattle production in that area has nearly doubled in the 5-year period between 1964 and 1969. Also, according to the 1959 census, the number of farms in the general area around the plant has decreased by 12%, but there has been a 10% increase in the size of farms from 1964 to 1969 (Ref. 9). While this trend is believed to apply to the general area around the plant, as indicated above, the number of beef cattle in the immediate area of the Site (within 10 miles) is insignificant.

Although some grains produced outside of the immediate area are
15 used to supplement their diet, dairy cows in the farm nearest the CRBRP
Site are allowed to graze outside all year (Ref. 10). Therefore, dose calculations reported in Section 11 assume that 100% of the annual diet of both cows and cattle comes from the fields. Variables used in the calculation of dose to man from ingestion of leafy vegetables, such as total daily in-
15 take and yield per unit area of cultivated land are provided in the Appendix
to Section 11. All other variables such as total intake of beef and milk and elapsed time between butchering of beef and ingestion are also provided in that appendix.

### 2.1.4.2 Surface Water Use

15 There are 11 public water supplies withdrawing water from surface
sources within a 20-mile radius of the Site. Three of these supplies are located where they could be influenced by the plant's water discharges. The city of Rockwood, Tennessee, has an auxiliary public water intake located on the King Creek embayment of Watts Bar Reservoir where the potential for reverse flow exists. Under certain conditions Clinch River
15 waters could flow upstream in the Emory River. Such flow could possibly
affect the Cumberland Utilities District surface water intake on the Little
15 Emory River and the Harriman water supply intake in the Emory River. Of
the 16 industrial water supplies presently within a 20-mile radius of the Site, only five are located where they could be influenced by waterborne discharges from the Site. The closest of these in located 1.6 miles downstream from the Site at CRM-14.4. This supply is used to provide potable water at the Oak Ridge Gaseous Diffusion Plant and the small industrial park at the north end of the Site property. The second is an ERDA supply at CRM-11.5. This supply is utilized for industrial, non-
15 consumptive purposes. The other supply which could be influenced by discharges
from the Site is the

TVA Kingston Steam Plant. This supply is withdrawn from the Emory River, which is influenced by flow coming down the Clinch River during certain periods of the summer,and it is used for in-plant purposes, including potable uses, as well as cooling. A.B. Long Quarries, Incorporated, and Mead Corporation are both located on the Emory River arm of Watts Bar Reservoir, which could receive upstream flow from the Clinch River, but neither of these supplies is used for potable or sanitary purposes. The location of industrial water supplies is shown in Figure 2.1-15 and additional information is given in table 2.1-19. Water usage by the CRBRP will be such that water quality and use will not be affected.

Downstream from the plant, two other public water supplies can be influenced by water flowing past the Clinch River Site. The nearest, 30-miles from the Site, is Spring City with a population of 1,756 which withdraws 120,000 gallons per day from the Piney River. Piney River is influenced by the backwater from Watts Bar Dam. The second is the city of Dayton, 44 miles from the Site, which withdraws 1,400,000 gallons per day from the Tennessee River. Dayton has a population of 4,361.

Because of the number and proximity of both private and public recreation areas, there is a fairly high degree of recreational usage of the Clinch River during the summer months. As Table 2.1-14 shows, most of the recreational areas are of the day-camp type. Recreational use of the Melton Hill Lock is given in Table 2.1-15.

No quantitative data is currently available on the amount of fish caught in this region by sport fisherman for human consumption (Ref. 11). Several general remarks concerning the fish resources at the Site can be made. Only two game fish are well represented, the Sauger and the White Bass (Ref. 12). Forage and rough fish dominate in both numbers and biomass.
15 Although Watts Bar Reservoir produced a commercial fish harvest of nearly 95,000 pounds in 1973, catches within a 10 mile radius of the site amounted to only 1% of this total (Ref. 13).

15

Dose calculations presented in section 11 from ingestion of aquatic foods assume average consumption of 5 g/day and utilize bioaccumulation factors listed in the Appendix to Section 11.

2.1.4.3 Groundwater Use

Because much of the development has been rural residential, it has not been economically feasible to use public water supplies for every residence. Thus, many individual wells are found in the area near the Site. Additional information is provided in section 2.4. Within a 20-mile radius of the Site, there are 17 public water supplies withdrawing water from wells and springs. These are listed in Table 2.1-20. The locations of public water supplies within 10 miles of the Site are shown in Figure 2.1-16.

## REFERENCES - SECTION 2.1


1. Tenessee General Social and Economic Characteristics - 1970 Census of Population, U.S. Department of Commerce.

2. Population Projections for the Tennessee Valley Region, by County by Decades to 2020. Tennessee Valley Authority, Division of Navigation Development and Regional Studies, Economic Research Staff, October 1972

3. Preliminary Information on Clinch River Site for LMFBR Demonstration Plant.

4. Maps - Recreational Areas on Watts Bar, Melton Hill and Fort Laudon Lakes.

5. Visits to TVA Dams and Steam Plant Project - 1971. Tennessee Valley Authority.

6. Journal of American Hospital Organization.

7. Telecons to County Agents, Emerson Ivers (Anderson Co.), Bob Gilley (Knox Co.), Charles McCall(Laudon Co.), Mr. Jensch (Morgan Co.) and K. Sutton (Roane Co.) from Paul Mattoon, WESD, June 1973.

8. Emory River Valley Summary of Resources, Emory River Watershed Development Associations.

9. Summary Data - 1969 Census of Agriculture, Tennessee. U.S. Department of Commerce.

10. Telecon, Colick J, WESD To Robinette, F.R., Kingston, Tennessee, July 8, 1974

11. Letter, Seawell, William, Tennessee Game and Fish Commission, Knoxville, Tennessee to Colick J., WESD, March 13, 1974

12. TVA Fish Population Monitoring Program - LMFBR Demonstation Project January 11, 1974

13. Letter, Allen, Elmer C., Fishery Reporting Specialist, National Fisheries Service,to Colick, J., WESD, March 1, 1974

TABLE 2.1-1

URBAN CENTERS WITH POPULATION >2500 WITHIN A
50-MILE RADIUS OF THE DEMONSTRATION
PLANT FOR CENSUS YEAR 1970

| Urban Center | County | Distance From Plant (miles) | Approx. Direction | Population |
|---|---|---|---|---|
| Knoxville | Knox | 21.50 | ENE | 174,587 |
| Oak Ridge | Anderson-Roane | 9.00 | NE | 28,319 |
| Maryville | Blount | 25.00 | ESE | 13,808 |
| Athens | McMinn | 32.75 | SSW | 11,790 |
| Harriman | Roane | 9.50 | WNW | 8,734 |
| Alcoa | Blount | 23.50 | ESE | 7,739 |
| La Follette | Campbell | 36.50 | NNE | 6,902 |
| Crossville | Cumberland | 36.50 | W | 5,381 |
| Lenior City | Loudon | 8.75 | SE | 5,324 |
| Rockwood | Roane | 16.00 | W | 5,259 |
| Eagleton Village | Blount | 25.25 | ESE | 5,345 |
| Clinton | Anderson | 20.50 | NE | 4,794 |
| Dayton | Rhea | 44.50 | SW | 4,361 |
| Sweetwater | Monroe | 20.00 | SSW | 4,340 |
| Kingston | Roane | 7.00 | W | 4,142 |
| Etowah | McMinn | 39.25 | SSW | 3,736 |
| Loudon | Loudon | 10.50 | SSE | 3,728 |
| Oliver Springs | Anderson, Morgan, Roane | 10.74 | NNE | 3,405 |
| Sevierville | Sevier | 45.50 | E | 2,661 |
| Madisonville | Monroe | 25.25 | S | 2,614 |
| Oneida | Scott | 42.50 | N | 2,602 |

Distances measured from the edge of the residential areas nearest the site.

TABLE 2.1-2

URBAN AREAS WITH POPULATION <2500 WITHIN A 50-MILE RADIUS OF THE DEMONSTRATION PLANT FOR CENSUS YEAR 1970

| Urban Center | County | Distance from Plant (miles) | Approx. Direction | Population |
|---|---|---|---|---|
| Lake City | Anderson | 25.75 | NNE | 1,923 |
| Norris | Anderson | 27.25 | NE | 1,359 |
| Pikeville | Bledsoe | 49.50 | WSW | 1,454 |
| Friendsville | Blount | 16.75 | ESE | 575 |
| Townsend | Blount | 38.25 | ESE | 267 |
| Charleston | Bradley | 46.75 | SSW | 792 |
| Caryville | Campbell | 29.75 | NNE | 648 |
| Jacksboro | Campbell | 32.25 | NNE | 689 |
| Pleasant Hill | Cumberland | 46.00 | W | 293 |
| Allardt | Fentress | 43.75 | NW | 610 |
| Jamestown | Fentress | 48.00 | NW | 1,899 |
| Greenback | Loudon | 19.75 | SE | 318 |
| Philadelphia | Loudon | 14.75 | S | 554 |
| Calhoun | McMinn | 46.50 | SSW | 624 |
| Englewood | McMinn | 32.75 | S | 1,878 |
| Niota | McMinn | 27.75 | SSW | 629 |
| Decatur | Meigs | 34.75 | SW | 698 |
| Tellico Plains | Monroe | 36.75 | S | 773 |
| Vonore | Monroe | 22.25 | SSE | 524 |
| Oakdale | Morgan | 11.75 | WNW | 376 |
| Wartburg | Morgan | 19.00 | NW | 541 |
| Spring City | Rhea | 29.75 | WSW | 1,756 |
| Huntsville | Scott | 36.25 | N | 337 |
| Gatlinburg | Sevier | 49.50 | ESE | 2,329 |
| Pigeon Forge | Sevier | 45.50 | E | 1,361 |
| Luttrell | Union | 41.75 | ENE | 819 |
| Maynardville | Union | 40.75 | NE | 702 |

Distances measured to edge of residential area nearest the site.

TABLE 2.1-3

POPULATION DISTRIBUTION WITHIN 10 MILES
OF THE DEMONSTRATION PLANT
15 FOR CENSUS YEAR 1970*

| Sector Designation | Radial Interval (miles) | | | | | |
|---|---|---|---|---|---|---|
| | 0-1 | 1-2 | 2-3 | 3-4 | 4-5 | 5-10 |
| N | 0 | 0 | 0 | 0 | 0 | 1,375 |
| NNE | 0 | 0 | 0 | 0 | 0 | 7,850 |
| 33 NE | 0 | 0 | 0 | 0 | 0 | 1,680 |
| ENE | 5 | 5 | 0 | 0 | 0 | 185 |
| E | 10 | 5 | 30 | 30 | 10 | 940 |
| ESE | 5 | 5 | 15 | 65 | 115 | 1,895 |
| SE | 0 | 15 | 45 | 95 | 115 | 8,700 |
| SSE | 0 | 15 | 20 | 120 | 125 | 955 |
| S | 5 | 35 | 20 | 75 | 95 | 335 |
| SSW | 0 | 35 | 5 | 70 | 65 | 175 |
| SW | 5 | 25 | 30 | 100 | 75 | 305 |
| WSW | 0 | 30 | 70 | 115 | 250 | 2,950 |
| W | 0 | 55 | 165 | 105 | 75 | 3,760 |
| WNW | 0 | 60 | 100 | 30 | 55 | 5,545 |
| NW | 0 | 20 | 0 | 0 | 45 | 1,270 |
| NNW | 0 | 0 | 0 | 0 | 75 | 1,235 |
| Sum for Radial Interval | 30 | 305 | 500 | 805 | 1,100 | 39,155 |
| Accumulative Total up to Radius Indicated | 30 | 335 | 835 | 1,640 | 2,740 | 41,895 |
| 35 33 Average Density (people/mi$^2$) in Radial Region | 10 | 32 | 32 | 37 | 39 | 166 |

15 * Based on information prepared by TVA Division of Navigational Development and Regional Studies, Economic Research.



Amend. 35
Feb. 1977

TABLE 2.1-4

PROJECTED POPULATION DISTRIBUTION WITHIN
10 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 1980

| | Sector Designation | Radial Interval (miles) 0-1 | 1-2 | 2-3 | 3-4 | 4-5 | 5-10 |
|---|---|---|---|---|---|---|---|
| | N | 0 | 0 | 0 | 0 | 0 | 1,375 |
| | NNE | 0 | 0 | 0 | 0 | 0 | 7,850 |
| | NE | 0 | 0 | 0 | 0 | 0 | 6,261 |
| 15 | ENE | 5 | 5 | 0 | 0 | 0 | 185 |
| | E | 10 | 5 | 30 | 30 | 10 | 940 |
| | ESE | 5 | 5 | 15 | 65 | 115 | 1,895 |
| | SE | 0 | 15 | 45 | 95 | 115 | 9,605 |
| | SSE | 0 | 15 | 20 | 120 | 125 | 955 |
| | S | 5 | 35 | 20 | 75 | 95 | 335 |
| | SSW | 0 | 35 | 5 | 70 | 65 | 175 |
| | SW | 5 | 25 | 30 | 100 | 75 | 305 |
| | WSW | 0 | 30 | 70 | 115 | 250 | 2,950 |
| | W | 0 | 55 | 165 | 105 | 75 | 4,360 |
| | WNW | 0 | 60 | 100 | 30 | 55 | 7,111 |
| | NW | 0 | 20 | 0 | 0 | 45 | 1,270 |
| | NNW | 0 | 0 | 0 | 0 | 75 | 1,235 |
| | Sum for Radial Interval | 30 | 305 | 500 | 805 | 1,100 | 46,807 |
| | Accumulative Total up to Radius Indicated | 30 | 335 | 835 | 1,640 | 2,740 | 49,547 |
| | Average Density (people/mi$^2$) in Radial Region | 10 | 32 | 32 | 37 | 39 | 199 |

TABLE 2.1-5

PROJECTED POPULATION DISTRIBUTION WITHIN
10 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 1990

| Sector Designation | Radial Interval (miles) 0-1 | 1-2 | 2-3 | 3-4 | 4-5 | 5-10 |
|---|---|---|---|---|---|---|
| N | 0 | 0 | 0 | 0 | 0 | 1,513 |
| NNE | 0 | 0 | 0 | 0 | 0 | 9,027 |
| 15 NE | 0 | 0 | 0 | 0 | 0 | 7,200 |
| ENE | 5 | 5 | 0 | 0 | 0 | 203 |
| E | 10 | 5 | 30 | 30 | 10 | 1,034 |
| ESE | 5 | 5 | 15 | 65 | 115 | 2,122 |
| SE | 0 | 15 | 45 | 95 | 115 | 10,950 |
| SSE | 0 | 15 | 20 | 120 | 125 | 1,003 |
| S | 5 | 35 | 20 | 75 | 95 | 342 |
| SSW | 0 | 35 | 5 | 70 | 65 | 184 |
| SW | 5 | 25 | 30 | 100 | 75 | 342 |
| WSW | 0 | 30 | 70 | 115 | 250 | 2,950 |
| W | 0 | 55 | 165 | 105 | 75 | 4,796 |
| WNW | 0 | 60 | 100 | 30 | 55 | 7,822 |
| NW | 0 | 20 | 0 | 0 | 45 | 1,397 |
| NNW | 0 | 0 | 0 | 0 | 75 | 1,358 |
| Sum for Radial Interval | 30 | 305 | 500 | 805 | 1,100 | 52,243 |
| Accumulative Total up to Radius Indicated | 30 | 335 | 835 | 1,640 | 2,740 | 54,983 |
| Average Density (people/mi$^2$) in Radial Region | 10 | 32 | 32 | 37 | 39 | 222 |

Amend. 15
Apr. 1976

TABLE 2.1-6

PROJECTED POPULATION DISTRIBUTION WITHIN
10 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 2000

| Sector Designation | Radial Interval (miles) | | | | | |
|---|---|---|---|---|---|---|
| | 0-1 | 1-2 | 2-3 | 3-4 | 4-5 | 5-10 |
| N | 0 | 0 | 0 | 0 | 0 | 1,664 |
| NNE | 0 | 0 | 0 | 0 | 0 | 9,929 |
| 15 NE | 0 | 0 | 0 | 0 | 0 | 8,064 |
| ENE | 5 | 5 | 0 | 0 | 0 | 213 |
| E | 10 | 5 | 30 | 30 | 10 | 1,054 |
| ESE | 5 | 5 | 15 | 75 | 115 | 2,334 |
| SE | 0 | 15 | 45 | 95 | 115 | 12,483 |
| SSE | 0 | 15 | 20 | 120 | 125 | 1,053 |
| S | 5 | 35 | 20 | 75 | 95 | 345 |
| SSW | 0 | 35 | 5 | 70 | 65 | 185 |
| SW | 5 | 25 | 30 | 100 | 75 | 345 |
| WSW | 0 | 30 | 70 | 115 | 250 | 3,216 |
| W | 0 | 55 | 165 | 105 | 75 | 5,276 |
| WNW | 0 | 60 | 100 | 30 | 55 | 8,604 |
| NW | 0 | 20 | 0 | 0 | 45 | 1,537 |
| NNW | 0 | 0 | 0 | 0 | 75 | 1,494 |
| Sum for Radial Interval | 30 | 305 | 500 | 805 | 1,100 | 57,796 |
| Accumulative Total up to Radius Indicated | 30 | 335 | 835 | 1,640 | 2,740 | 60,536 |
| Average Density (people/mi$^2$) in Radial Region | 10 | 32 | 32 | 37 | 39 | 245 |

TABLE 2.1-7

PROJECTED POPULATION DISTRIBUTION WITHIN
10 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 2010

| Sector Designation | Radial Interval (miles) 0-1 | 1-2 | 2-3 | 3-4 | 4-5 | 5-10 |
|---|---|---|---|---|---|---|
| N | 0 | 0 | 0 | 0 | 0 | 1,830 |
| NNE | 0 | 0 | 0 | 0 | 0 | 10,921 |
| 15 NE | 0 | 0 | 0 | 0 | 0 | 8,870 |
| ENE | 5 | 5 | 0 | 0 | 0 | 224 |
| E | 10 | 5 | 30 | 30 | 10 | 1,107 |
| ESE | 5 | 5 | 15 | 65 | 115 | 2,521 |
| SE | 0 | 15 | 45 | 95 | 115 | 14,231 |
| SSE | 0 | 15 | 20 | 120 | 125 | 1,085 |
| S | 5 | 35 | 20 | 75 | 95 | 352 |
| SSW | 0 | 35 | 5 | 70 | 65 | 187 |
| SW | 5 | 25 | 30 | 100 | 75 | 376 |
| WSW | 0 | 30 | 70 | 115 | 250 | 3,377 |
| W | 0 | 55 | 165 | 105 | 75 | 5,487 |
| WNW | 0 | 60 | 100 | 30 | 55 | 8,690 |
| NW | 0 | 20 | 0 | 0 | 45 | 1,552 |
| NNW | 0 | 0 | 0 | 0 | 75 | 1,539 |
| Sum for Radial Interval | 30 | 305 | 500 | 805 | 1,100 | 62,349 |
| Accumulative Total up to Radius Indicated | 30 | 335 | 835 | 1,640 | 2,740 | 65,089 |
| Average Density (people/mi$^2$) in Radial Region | 10 | 32 | 32 | 37 | 39 | 265 |

Amend. 15
Apr. 1976

TABLE 2.1-8

POPULATION DISTRIBUTION WITHIN 50 MILES OF THE DEMONSTRATION PLANT FOR CENSUS YEAR 1970

| Sector Designation | Radial Interval (miles) 0-5 | 5-10 | 10-20 | 20-30 | 30-40 | 40-50 |
|---|---|---|---|---|---|---|
| N | 0 | 1,375 | 2,000 | 705 | 3,085 | 6,480 |
| NNE | 0 | 7,850 | 5,845 | 8,515 | 13,575 | 8,160 |
| NE | 0 | 1,680 | 26,955 | 13,110 | 4,675 | 5,665 |
| ENE | 10 | 185 | 13,450 | 129,165 | 31,395 | 9,875 |
| E | 85 | 940 | 21,520 | 74,020 | 15,025 | 15,345 |
| ESE | 205 | 1,895 | 3,890 | 42,620 | 4,325 | 1,700 |
| SE | 270 | 8,700 | 2,220 | 6,280 | 270 | 1,315 |
| SSE | 280 | 955 | 4,385 | 3,045 | 995 | 1,450 |
| S | 230 | 335 | 4,590 | 8,475 | 7,355 | 2,580 |
| SSW | 175 | 175 | 1,725 | 8,255 | 20,045 | 10,480 |
| SW | 235 | 305 | 1,285 | 1,980 | 5,260 | 9,590 |
| WSW | 465 | 2,950 | 1,890 | 2,670 | 3,375 | 3,995 |
| W | 400 | 3,760 | 11,135 | 2,365 | 9,290 | 3,910 |
| WNW | 245 | 5,545 | 3,965 | 230 | 3,290 | 3,915 |
| NW | 65 | 1,270 | 2,730 | 2,490 | 2,205 | 6,365 |
| NNW | 75 | 1,235 | 3,035 | 905 | 4,235 | 2,230 |
| Sum for Radial Interval | 2,740 | 39,155 | 110,620 | 304,830 | 128,400 | 93,055 |
| Accumulative Total up to Radius Indicated | 2,740 | 41,895 | 152,515 | 457,345 | 585,745 | 678,800 |
| Average Density (people/mi$^2$) in Radial Region | 35 | 167 | 117 | 194 | 58 | 33 |

TABLE 2.1-9

PROJECTED POPULATION DISTRIBUTION WITHIN 50 MILES OF THE DEMONSTRATION PLANT FOR CENSUS YEAR 1980

| Sector Designation | Radial Interval (miles) 0-5 | 5-10 | 10-20 | 20-30 | 30-40 | 40-50 |
|---|---|---|---|---|---|---|
| N | 0 | 1,375 | 2,325 | 685 | 3,580 | 7,060 |
| NNE | 0 | 7,850 | 6,430 | 8,260 | 16,190 | 8,405 |
| NE | 0 | 6,261 | 31,045 | 13,240 | 4,815 | 6,685 |
| ENE | 10 | 185 | 16,140 | 142,085 | 34,220 | 10,860 |
| E | 85 | 940 | 23,670 | 77,720 | 16,660 | 17,215 |
| ESE | 205 | 1,895 | 4,160 | 48,585 | 5,840 | 2,040 |
| SE | 270 | 9,605 | 2,310 | 6,405 | 275 | 1,320 |
| SSE | 280 | 955 | 4,520 | 3,555 | 995 | 1,450 |
| S | 230 | 335 | 4,680 | 10,170 | 7,500 | 2,570 |
| SSW | 175 | 175 | 1,760 | 8,585 | 24,055 | 12,365 |
| SW | 235 | 305 | 1,415 | 1,980 | 5,680 | 10,455 |
| WSW | 465 | 2,950 | 2,080 | 2,670 | 3,645 | 4,195 |
| W | 400 | 4,360 | 12,360 | 2,365 | 10,310 | 4,300 |
| WNW | 245 | 7,111 | 4,480 | 230 | 3,290 | 4,070 |
| NW | 65 | 1,270 | 3,000 | 2,490 | 2,205 | 6,430 |
| NNW | 75 | 1,235 | 3,035 | 905 | 4,235 | 2,230 |
| Sum for Radial Interval | 2,740 | 46,807 | 123,410 | 329,930 | 143,495 | 101,650 |
| Accumulative Total up to Radius Indicated | 2,740 | 49,547 | 172,957 | 502,887 | 646,382 | 748,032 |
| Average Density (people/$mi^2$) in Radial Region | 35 | 199 | 130 | 210 | 65 | 36 |

TABLE 2.1-10

PROJECTED POPULATION DISTRIBUTION WITHIN
50 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 1990

| Sector Designation | Radial Interval (miles) 0-5 | 5-10 | 10-20 | 20-30 | 30-40 | 40-50 |
|---|---|---|---|---|---|---|
| N | 0 | 1,513 | 2,560 | 685 | 4,010 | 7,835 |
| NNE | 0 | 9,027 | 7,075 | 8,010 | 17,650 | 8,660 |
| NE | 0 | 7,200 | 36,635 | 13,900 | 5,250 | 8,890 |
| ENE | 10 | 203 | 19,365 | 156,295 | 37,640 | 12,270 |
| E | 85 | 1,034 | 26,040 | 81,605 | 18,325 | 19,280 |
| ESE | 205 | 2,122 | 4,410 | 56,360 | 7,185 | 2,365 |
| SE | 270 | 10,950 | 2,400 | 6,660 | 280 | 1,325 |
| SSE | 280 | 1,003 | 4,655 | 3,730 | 1,000 | 1,450 |
| S | 230 | 342 | 4,775 | 11,190 | 7,650 | 2,560 |
| SSW | 175 | 184 | 1,795 | 9,015 | 26,220 | 14,345 |
| SW | 235 | 342 | 1,585 | 1,985 | 6,360 | 12,130 |
| WSW | 465 | 2,950 | 2,290 | 2,750 | 3,940 | 4,320 |
| W | 400 | 4,796 | 13,595 | 2,365 | 11,240 | 4,685 |
| WNW | 245 | 7,822 | 5,065 | 230 | 3,290 | 4,190 |
| NW | 65 | 1,397 | 3,300 | 2,490 | 2,205 | 6,625 |
| NNW | 75 | 1,358 | 3,035 | 900 | 4,235 | 2,230 |
| Sum for Radial Interval | 2,740 | 52,243 | 138,580 | 358,170 | 156,480 | 113,160 |
| Accumulative Total up to Radius Indicated | 2,740 | 54,983 | 193,563 | 551,733 | 708,213 | 821,373 |
| Average Density (people/mi$^2$) in Radial Region | 35 | 222 | 147 | 227 | 71 | 40 |

TABLE 2.1-11

PROJECTED POPULATION DISTRIBUTION WITHIN 50 MILES OF THE DEMONSTRATION PLANT FOR CENSUS YEAR 2000

| Sector Designation | Radial Interval (miles) 0-5 | 5-10 | 10-20 | 20-30 | 30-40 | 40-50 |
|---|---|---|---|---|---|---|
| N | 0 | 1,664 | 2,815 | 685 | 4,450 | 8,620 |
| NNE | 0 | 9,929 | 7,780 | 7,770 | 18,355 | 8,915 |
| NE | 0 | 8,064 | 42,860 | 14,595 | 5,615 | 11,650 |
| ENE | 10 | 213 | 23,240 | 171,920 | 41,030 | 13,255 |
| E | 85 | 1,054 | 28,640 | 85,685 | 20,160 | 21,600 |
| ESE | 205 | 2,334 | 4,630 | 64,810 | 8,190 | 2,815 |
| SE | 270 | 12,483 | 2,500 | 6,930 | 280 | 1,330 |
| SSE | 280 | 1,053 | 4,795 | 3,955 | 1,000 | 1,450 |
| S | 230 | 345 | 4,870 | 12,530 | 7,650 | 2,550 |
| SSW | 175 | 185 | 1,830 | 9,085 | 30,940 | 16,350 |
| SW | 235 | 345 | 1,710 | 1,985 | 6,360 | 13,220 |
| WSW | 465 | 3,216 | 2,515 | 2,805 | 4,290 | 4,535 |
| W | 400 | 5,276 | 14,820 | 2,365 | 12,135 | 5,205 |
| WNW | 245 | 8,604 | 5,465 | 225 | 3,290 | 4,275 |
| NW | 65 | 1,537 | 3,630 | 2,490 | 2,205 | 6,690 |
| NNW | 75 | 1,494 | 3,035 | 900 | 4,235 | 2,230 |
| Sum for Radial Interval | 2,740 | 57,796 | 155,135 | 388,735 | 170,185 | 124,690 |
| Accumulative Total up to Radius Indicated | 2,740 | 60,536 | 215,671 | 604,406 | 774,591 | 899,281 |
| Average Density (people/mi$^2$) in Radial Region | 35 | 245 | 161 | 247 | 77 | 44 |

TABLE 2.1-12

PROJECTED POPULATION DISTRIBUTION WITHIN
50 MILES OF THE DEMONSTRATION PLANT
FOR CENSUS YEAR 2010

| Sector Designation | Radial Interval (miles) 0-5 | 5-10 | 10-20 | 20-30 | 30-40 | 40-50 |
|---|---|---|---|---|---|---|
| N | 0 | 1,830 | 3,095 | 685 | 4,980 | 9,570 |
| NNE | 0 | 10,921 | 8,560 | 7,690 | 19,275 | 9,185 |
| NE | 0 | 8,870 | 50,575 | 15,180 | 5,955 | 14,560 |
| ENE | 10 | 224 | 27,890 | 189,115 | 45,135 | 14,180 |
| E | 85 | 1,107 | 31,505 | 89,970 | 22,175 | 24,190 |
| ESE | 205 | 2,521 | 4,860 | 73,885 | 9,335 | 3,295 |
| SE | 270 | 14,231 | 2,600 | 7,205 | 290 | 1,330 |
| SSE | 280 | 1,085 | 4,940 | 4,235 | 1,005 | 1,450 |
| S | 230 | 352 | 4,970 | 14,160 | 7,805 | 2,540 |
| SSW | 175 | 187 | 1,870 | 9,720 | 37,130 | 18,315 |
| SW | 235 | 376 | 1,915 | 1,990 | 6,490 | 14,410 |
| WSW | 465 | 3,377 | 2,770 | 2,890 | 4,465 | 4,675 |
| W | 400 | 5,487 | 16,595 | 2,460 | 13,470 | 5,720 |
| WNW | 245 | 8,690 | 6,015 | 225 | 3,290 | 4,360 |
| NW | 65 | 1,552 | 4,065 | 2,490 | 2,205 | 6,825 |
| NNW | 75 | 1,539 | 3,035 | 990 | 4,235 | 2,230 |
| Sum for Radial Interval | 2,740 | 62,349 | 175,260 | 422,890 | 187,240 | 136,835 |
| Accumulative Total up to Radius Indicated | 2,740 | 65,089 | 240,349 | 663,239 | 850,479 | 987,314 |
| Average Density (people/mi$^2$) in Radial Region | 35 | 265 | 186 | 269 | 85 | 48 |

TABLE 2.1-13

TRANSIENT POPULATION WITHIN 5-MILE RADIUS OF SITE

| | Distance (miles) | Activity | Daily Transients |
|---|---|---|---|
| | 1.0 | Recreation Area | 48 |
| 15 | 1.5 | U. S. Nuclear, Inc. | 95 |
| | 1.5 | Nuclear Environmental Engineering, Inc. | 20 |
| | 1.5 | Nuclear Assurance Company | 6 |
| | 2.0 | Recreation Area | 83 |
| | 3.0 | Recreation Areas (6) | 6,423 |
| | 3.0 | Oak Ridge Gaseous Diffusion Plant | 4,607 |
| | 3.5 | Edgewood Elementary School | 150 |
| | 4.0 | Oak Ridge National Laboratory | 4,029 |
| | 4.5 | Melton Hill Dam (TVA) | 512* |
| 15 | 5.0 | Recreation Areas (5) | 248 |

15 *This number includes a daily average of ~500 visitors plus a staff of 12 workers.

Amend. 15
Apr. 1976

TABLE 2.1-14

ESTIMATED AVERAGE PEAK HOUR USE AT RECREATION AREAS WITHIN

10 MILES OF THE CRBRP* 23

| Mileage Zone | Site No.** | Est. No. Persons Present During Peak Hour 1970 | 1980 | 1990 | 2000 | 2010 |
|---|---|---|---|---|---|---|
| 0 to 1 | 1 | 40 | 55 | 70 | 80 | 90 |
| 1 to 2 | 2 | 70 | 95 | 115 | 130 | 145 |
| | 3+ | 15 | 20 | 25 | 30 | 30 |
| | 4+ | 200 | 270 | 340 | 360 | 420 |
| 2 to 3 | 5+ | 5,000 | 6,000 | 6,500 | 7,000 | 7,300 |
| | 6+ | 60 | 80 | 100 | 110 | 120 |
| | 7 | 15 | 20 | 25 | 30 | 30 |
| 3 to 4 | 8 | 70 | 95 | 115 | 130 | 145 |
| 4 to 5 | 9+ | 15 | 20 | 25 | 30 | 30 |
| | 10 | 70 | 95 | 115 | 130 | 145 |
| | 11 | 15 | 20 | 25 | 30 | 30 |
| | 12 | 70 | 1,100 | 1,200 | 1,300 | 1,500 |
| | 13 | 40 | 55 | 70 | 80 | 90 |
| 5 to 6 | 14 | 25 | 35 | 45 | 50 | 55 |
| | 15 | 40 | 55 | 70 | 80 | 90 |
| | 16+ | 15 | 20 | 25 | 30 | 30 |
| | 17 | 40 | 55 | 70 | 80 | 90 |
| 6 to 7 | 18 | 40 | 55 | 70 | 80 | 90 |
| | 19+ | 100 | 135 | 170 | 190 | 210 |
| | 20 | 15 | 20 | 25 | 30 | 30 |
| | 21 | 40 | 55 | 70 | 80 | 90 |
| | 22 | 40 | 55 | 70 | 80 | 90 |
| | 23 | 40 | 55 | 70 | 80 | 90 |
| | 24 | 40 | 55 | 70 | 80 | 90 |
| | 25 | 70 | 95 | 115 | 130 | 145 |
| 15 7 to 8 | 26+ | 20 | 25 | 30 | 35 | 40 |
| | 27 | 40 | 55 | 70 | 80 | 90 |
| | 28 | 40 | 55 | 70 | 80 | 90 |
| | 29 | 70 | 95 | 115 | 130 | 145 |
| | 30 | 40 | 55 | 70 | 80 | 90 |
| | 31 | 70 | 95 | 115 | 130 | 160 |
| | 32+ | 15 | 15 | 20 | 25 | 30 |
| | 33 | 15 | 20 | 25 | 30 | 35 |
| | 34 | 40 | 55 | 70 | 80 | 95 |
| | 35 | 40 | 55 | 70 | 80 | 95 |
| | 36+ | 30 | 45 | 55 | 60 | 75 |
| | 37 | 20 | 25 | 35 | 40 | 45 |
| | 38+ | 70 | 95 | 115 | 130 | 160 |
| | 39+ | 55 | 75 | 90 | 105 | 125 |

Amend. 23
June 1976

TABLE 2.2-14 (Continued)

| Mileage Zone | Site No.** | Est. No. Persons Present During Peak Hour 1970 | 1980 | 1990 | 2000 | 2010 |
|---|---|---|---|---|---|---|
| 7 to 8 | 40 | 15 | 20 | 25 | 30 | 35 |
| | 41 | 70 | 95 | 115 | 130 | 160 |
| | 42 | 15 | 20 | 25 | 30 | 35 |
| | 43 | 5 | 8 | 9 | 10 | 11 |
| | 44+ | 70 | 115 | 130 | 145 | 155 |
| 8 to 9 | 45 | 25 | 35 | 45 | 50 | 60 |
| | 46 | 40 | 55 | 70 | 80 | 95 |
| | 47 | 15 | 20 | 25 | 30 | 35 |
| | 48 | 70 | 95 | 120 | 150 | 165 |
| | 49+ | 70 | 95 | 115 | 145 | 160 |
| | 50+ | 70 | 95 | 115 | 145 | 160 |
| | 51+ | 2 | 3 | 4 | 4 | 4 |
| | 52+ | 35 | 55 | 60 | 65 | 70 |
| 9 to 10 | 53 | 25 | 35 | 45 | 50 | 55 |
| | 54 | 15 | 20 | 25 | 30 | 30 |
| | 55+ | 55 | 75 | 90 | 105 | 115 |
| | 56+ | 70 | 95 | 115 | 130 | 145 |
| | 57 | 40 | 55 | 70 | 80 | 90 |
| | 58+ | 70 | 95 | 115 | 130 | 145 |
| | 59 | 20 | 25 | 30 | 35 | 40 |
| | 60 | 15 | 20 | 25 | 30 | 30 |
| | 61 | 30 | 40 | 50 | 55 | 60 |
| | 62 | 20 | 25 | 35 | 40 | 40 |
| | 63 | 15 | 20 | 25 | 25 | 30 |
| | 64 | 1 | 2 | 2 | 2 | 2 |
| | 65+ | 30 | 45 | 55 | 60 | 65 |
| 15 Total | 65 | 7,658 | 10,643 | 12,085 | 13,321 | 14,442 |

*Information supplied by TVA, Special Studies Section, Recreation Resources Branch

15 **Keyed to Figure 2.2-7

+Activities at the sites are: 3, ORGOP visitors' overlook; 4 and 6, commercial campgrounds; 5, auto raceway; 9, Graphite Reactor; 16, 50 and 56, private club and parks; 19, 51 and 52, wildlife management areas; 26, 36 and 65, boat docks; 32, driving range; 39 and 56, golf courses; and 38, 44, 49, and 58, public parks. All other sites
15 are: public access or incidental use areas.
**Keyed to Figure 2.1-10

Amend. 15
Apr. 1976

TABLE 2.1-15

TRAFFIC LOCKED THROUGH MELTON HILL DAM

| Year | Recreational Craft (number)* | Commercial Traffic Total Tonnage** | Commercial Traffic Average Tons per Barge |
|---|---|---|---|
| 1966 | 1,198 | 1,000 | 500 |
| 1967 | 1,014 | 1,000 | 500 |
| 1968 | 1,256 | 2,000 | 500 |
| 1969 | 1,301 | 1,000 | 500 |
| 1970 | 929 | 4,000 | 800 |
| 1971 | 718 | 10,000 | 715 |
| 1972 | 761 | 3,600 | 720 |
| 1973 | 815 | 10,100 | 720 |
| 1974 | 631 | 4,800 | 800 |
| 1975 | NA+ | 2,956++ | 739 |

28

* Information supplied by TVA Division of Navigational Development and Regional Studies, Navigation Economics Branch

** Information supplied by the U.S. Corps of Engineers

\+ Not available as of January 13, 1976

++ Four (4) barges 28

Amend. 28
Oct. 1976

## TABLE 2.1-16

## INFORMATION ON SCHOOLS WITHIN 10-MILE RADIUS OF CRBRP

| School System | Forecast |
|---|---|
| Anderson County | No schools are within 10 miles of Site and none forecast for 1980 or 1990. |
| Oak Ridge | No schools are within 10 miles of Site. A new elementary school (K-6) is likely by 1990 in western Oak Ridge to accommodate 725 students. |
| Knox County | No schools are within 10 miles of Site and none forecast for 1980 or 1990. |

| School System | No. | School | Grades | 1973 | 1980 | 1990 | Distance in Miles and Direction From Site |
|---|---|---|---|---|---|---|---|
| Loudon County | 1 | Browder | 1-8 | 111 | 200 | 250 | 9.0 SSE |
| | 2 | Eatons | K-8 | 638 | 800 | 850 | 5.5 SE |
| | 3 | Highland Park | K-8 | 380 | 600 | 700 | 9.5 SE |
| Lenoir City | 4 | Lenoir City High School | 9-12 | 910 | 950 | 1,000 | 8.0 SE |
| | 5 | Lenoir City Middle School | 5-8 | 472 | 700 | 800 | 9.5 SE |
| | 6 | Nichols School | K-4 | 401 | 750 | 800 | 9.0 SE |
| | 7 | West Hill | 1-6 | 113 | 250 | 300 | 8.5 SE |
| Morgan County | 8 | Coalfield Elementary | 1-8 | 375 | 375 | 375 | 10.0 NNW |
| | 9 | Coalfield High School | 9-12 | 183 | 200 | 200 | 10.0 NNW |
| Roane County | 10 | Edgewood | 1-6 | 110 | 200 | 200 | 3.5 WSW |
| | 11 | Cherokee | 1-6 | 294 | 500 | 600 | 7.0 NNW |
| | 12 | Dyllis | 1-8 | 211 | 300 | 300 | 5.5 NNW |
| | 13 | Emory | 1-8 | 118 | 200 | 300 | 7.5 NW |
| | 14 | Fairview | K-6 | 200 | 200 | 250 | 9.0 W |
| | 15 | Kingston Elementary | K-6 | 675 | 750 | 900 | 7.5 WSW |
| | 16 | Kingston Junior High School | 7-8 | 351 | 500 | 600 | 8.0 W |
| | 17 | Roane County High School | 9-12 | 814 | 1,000 | 1,200 | 7.5 W |
| Harriman | 18 | Cumberland Junior High School | 7-9 | 345 | 600 | 650 | 9.5 WNW |
| | 19 | Harriman Central Elementary | 1-6 | 362 | 600 | 700 | 9.5 WNW |
| | 20 | Harriman High School | 10-12 | 504 | 850 | 900 | 9.5 WNW |
| | 21 | Margrave | 5-6 | 109 | 125 | 125 | 10.0 WNW |
| | 22 | Walnut Hill | 1-4 | 225 | 500 | 500 | 10.0 WNW |
| Oak Ridge | 23 | New Elementary | K-6 | - | - | 725 | 9.0 NNE |

15

TABLE 2.1-17

HOSPITALS WITHIN 50 MILES OF SITE

| Hospital | City | County | No. of Beds | No. of Bassinets | Distance (miles) and Direction from Plant | |
|---|---|---|---|---|---|---|
| Little Creek Sanitarium and Hospital | Concord | Knox | 25 | - | 21.5 | ENE |
| Cumberland Medical Center | Crossville | Cumberland | 82 | 11 | 36.5 | W |
| Rhea County Hospital | Dayton | Rhea | 45 | 8 | 44.5 | SW |
| Woods Memorial Hospital | Etowah | McMinn | 34 | 10 | 39.25 | SSW |
| Harriman City Hospital | Harriman | Roane | 94 | 12 | 9.5 | WNW |
| Fentress County Hospital | Jamestown | Fentress | 70 | 8 | 48.0 | NW |
| Christenberry Infirmary | Knoxville | Knox | 12 | - | 21.5 | ENE |
| Eastern State Psychiatric | Knoxville | Knox | 2761 | - | 21.5 | ENE |
| East Tennessee Baptist | Knoxville | Knox | 349 | 38 | 21.5 | ENE |
| East Tennessee Chest Disease | Knoxville | Knox | 180 | 4 | 21.5 | ENE |
| East Tennessee Children's | Knoxville | Knox | 52 | - | 21.5 | ENE |
| Fort Sanders Presbyterian | Knoxville | Knox | 374 | 40 | 21.5 | ENE |
| Knoxville Osteopathic | Knoxville | Knox | 25 | 6 | 21.5 | ENE |
| Parkwest | Knoxville | Knox | 200 | 25 | 21.0 | ENE |
| St. Mary's Memorial | Knoxville | Knox | 425 | 30 | 21.5 | ENE |
| Serene Manor Hospital | Knoxville | Knox | 68 | - | 21.5 | ENE |

(Continued)

TABLE 2.1-17 (Continued)

| Hospital | City | County | No. of Beds | No. of Bassinets | Distance (miles) and Direction from Plant | |
|---|---|---|---|---|---|---|
| University of Tennessee Memorial Research Center and Hospital | Knoxville | Knox | 336 | 34 | 21.5 | ENE |
| LaFollette Community | LaFollette | Campbell | 76 | 11 | 36.5 | NNE |
| Charles H. Bacon Hospital | Loudon | Loudon | 47 | 18 | 10.5 | SSE |
| Blount Memorial Hospital | Maryville | Blount | 230 | 30 | 25.0 | ESE |
| Oak Ridge Associated Universities Medical Div. | Oak Ridge | Anderson | 30 | - | 15.0 | NE |
| Oak Ridge Hospital of Methodist Church | Oak Ridge | Anderson | 287 | 20 | 15.0 | NE |
| Chamberlain Memorial | Rockwood | Roane | 55 | 10 | 16.0 | W |
| Sevier County Hospital | Sevierville | Sevier | 48 | 12 | 45.5 | E |
| 15 Lake City | Lake City | Anderson | 20 | -- | 26.0 | NNE |

TABLE 2.1-18

DAIRY HERDS WITHIN 10-MILES OF SITE

| Type | Livestock Population | Distance from Site (miles) | Direction |
|---|---|---|---|
| 1. *Dairy | 45 | 4 | W |
| 2. Dairy | 50 | 4.5 | W |
| 3. Dairy | 50 | 5.5 | W |
| 4. Dairy | 65 | 9.5 | ESE |
| 5. Dairy | 90 | 10 | WNW |

***Table keyed to figure 2.1-12**

TABLE 2.1-19

INDUSTRIAL WATER SUPPLIES WITHIN A 20-MILE RADIUS OF DEMONSTRATION PLANT

| | Supply * | Approx. Radial Distance from Site (miles) | Average Daily Use (gallons) | Source |
|---|---|---|---|---|
| 1. | Atomic Energy Commission | 1.6 | 2,500,000 | Surface (Clinch River Mile 14.4)[1] |
| 2. | Atomic Energy Commission | 3.5 | 5,500,000 | Surface (Clinch River Mile 11.5) |
| 3. | TVA Kingston Steam Plant | 7.8 | 1,400,000,000 | Surface (Emory River Mile 1.9)[3] |
| 4. | Lenoir City Car Works | 9.3 | 30,000 | Ground, Well[3] |
| 5. | The Mead Corp. | 9.7 | 2,900,000 | Surface (Emory River Mile 11.4) |
| 6. | Charles H. Bacon Co. | 9.8 | 255,000 | Ground, Well[3] |
| 7. | Union Carbide | 10.2 | 2,000,000 | Surface (Tennessee River) and Ground, Spring[3] |
| 8. | Charles H. Bacon Co. | 10.2 | 300,000 | Surface (Tennessee River) and Ground, Spring[3] |
| 9. | Atomic Energy Commission | 11.0 | 22,000,000 | Surface (Clinch River Mile 41.5)[2] |
| 10. | Ralph Rogers Co., Inc. | 11.4 | 24,000 | Ground, Well |
| 11. | C. N. O. & T Railway | 12.6 | 27,000 | Surface (Emory River Mile 18.8)[3] |
| 12. | A. B. Long Quarries | 13.4 | 1,500,000 | Surface (Emory River) |
| 13. | John J. Craig Co. | 13.8 | 34,600 | Surface (Small Stream) and Ground, Well |
| 14. | Philadelphia Hosiery Mills | 14.5 | 20,000 | Ground, Well |
| 15. | TVA Bull Run Steam Plant | 15.3 | 572,000,000 | Surface (Clinch River Mile 47.6)[3] |
| 16. | Morgan Apparel Co. | 17.8 | 3,000 | Ground, Well |

[1] Potable water only.
[2] Supplies 3,500,000 gallons per day to City of Oak Ridge (pop. 28,319).
[3] Water supply is also used for potable water within the plant.

*** Numbering keyed to Figure 2.1-15**

TABLE 2.1-20

PUBLIC WATER SUPPLIES WITHIN 20-MILE RADIUS OF DEMONSTRATION PLANT

| | | Supply * | Approx. Radial Distance from Site (miles) | Population Served | Average Daily Use (gallons) | Source |
|---|---|---|---|---|---|---|
| | 1. | Edgewood Elementary School | 3.5 | 196 | 4,900 | Ground, Well |
| | 2. | Cumberland Utility District of Roane and Morgan Co. | 6.7 | 5,000 | 212,000 | Ground, Spring[1] |
| | 3. | Dixie Lee Utility District | 8.6 | 4,500 | 395,000 | Ground, Spring[2] |
| 15 | 4. | Kingston | 8.8 | 5,000 | 350,000 | Ground, Spring[3] |
| 15 | 5. | Lenoir City | 9.7 | 6,500 | 1,000,000 | Surface (Tennessee River Mile 601.3) |
| | 6. | Midtown | 10.8 | 2,090 | 130,000 | Ground, Well |
| 15 | 7. | Harriman | 10.8 | 10,000 | 1,500,000 | Surface (Emory River Mile 12.9)[4] |
| 15 | 8. | Loudon | 11.4 | 5,000 | 650,000 | Ground, Spring |
| | 9. | Piney Utility District | 12.2 | 2,000 | 75,000 | Ground, Spring |
| | 10. | Paint Rock Elementary School | 12.4 | 250 | 6,200 | Ground, Well |
| | 11. | Midway High School | 12.6 | 515 | 12,900 | Ground, Spring |
| 15 | 12. | Oliver Springs | 14.5 | 3,570 | 325,000 | Ground, Spring |
| | 13. | First Utility District of Knox County | 14.5 | 10,500 | 1,051,000 | Surface (Sinking Creek Embayment) |

(Continued)

[1] Also has auxiliary water intake at Little Emory River Mile 3.9.
[2] Includes Martel Utility District.
[3] Also has auxiliary water intake at Tennessee River Mile 568.2.
[4] Includes Swan Pond Utility District.

*** Numbering keyed to figure 2.1-16.**

Amend. 15
Apr. 1976

TABLE 2.1-20 (Continued)

| | | Supply | Approx. Radial Distance from Site (miles) | Population Served | Average Daily Use (gallons) | Source |
|---|---|---|---|---|---|---|
| | 14. | West Knox Utility District | 15.3 and 17.8 | 18,000 | 1,000,000 | Surface (Clinch River Mile 46.9) and Ground, Spring |
| | 15. | Dutch Valley Elementary | 16.8 | 140 | 3,500 | Ground, Well |
| | 16. | First Utility District of Anderson Co. | 17.1 | 3,600 | 270,000 | Ground, Spring |
| | 17. | Hallsdale-Powell Utility District | 17.5 | 22,000 | 1,500,000 | Surface (Bull Run Creek embayment)[5] |
| | 18. | Plateau Utility District | 17.7 | 1,900 | 100,000 | Ground, Well |
| | 19. | Brushy Mountain State Honor Farm | 17.9 | 195 | 60,000 | Ground, Well |
| 15 | 20. | Rockwood | 18.0 | 5,500 | 1,700,000 | Ground, Spring and Surface (King Creek embayment) |
| 15 | 21. | Clinton Utility Board | 20.0 | 17,000 | 870,000 | Surface (Clinch River Mile 59) |
| | 22. | Sweetwater | 20.0 | 5,100 | 700,000 | Ground, Spring and Surface (Sweetwater Creek) |

[5] This figure includes water withdrawn from sources outside of the 20-mile radius.

Amend. 15
Apr. 1976

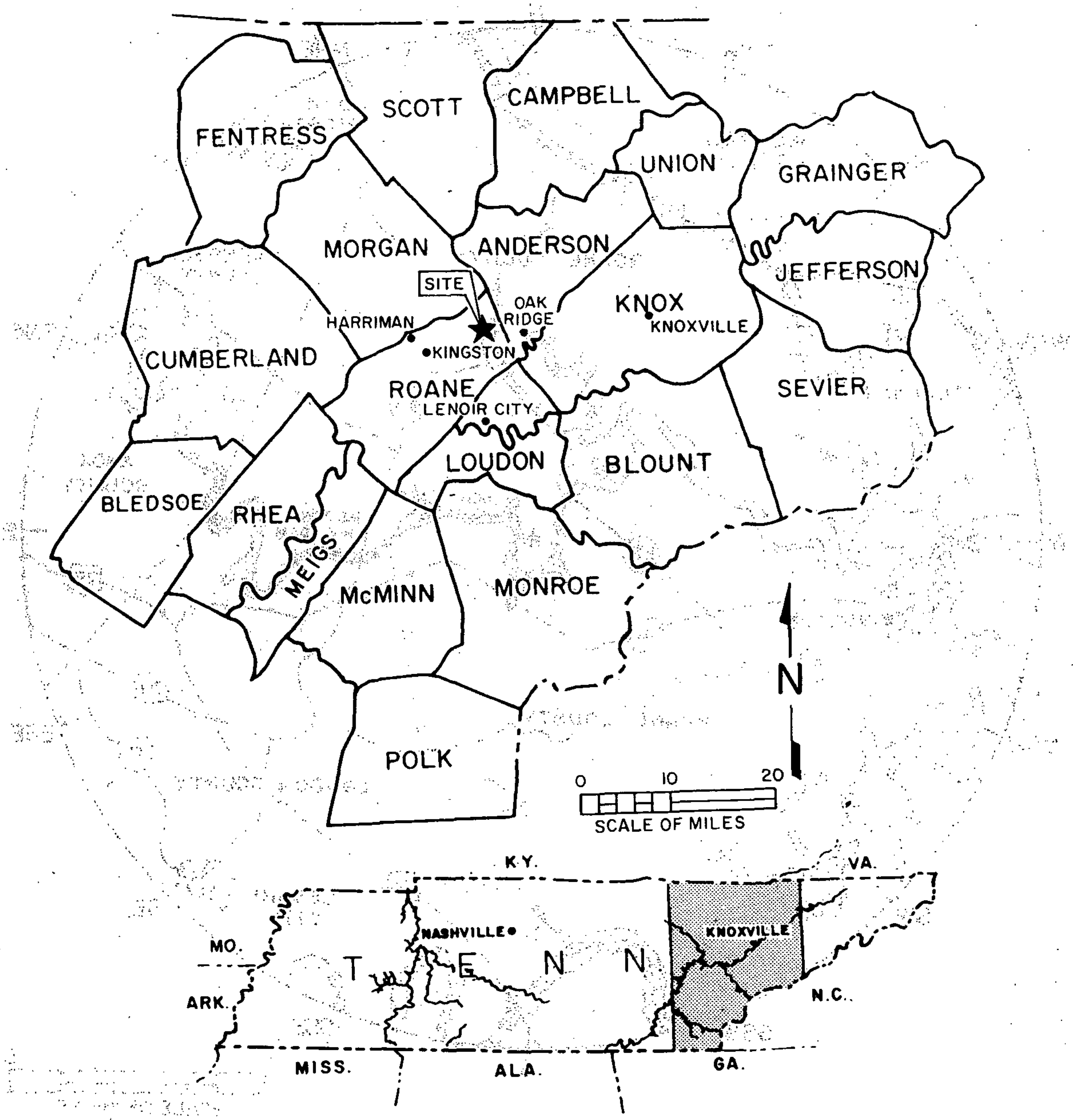



Figure 2.1-1. Location of Clinch River Site in Relation to Counties and State

6647-17

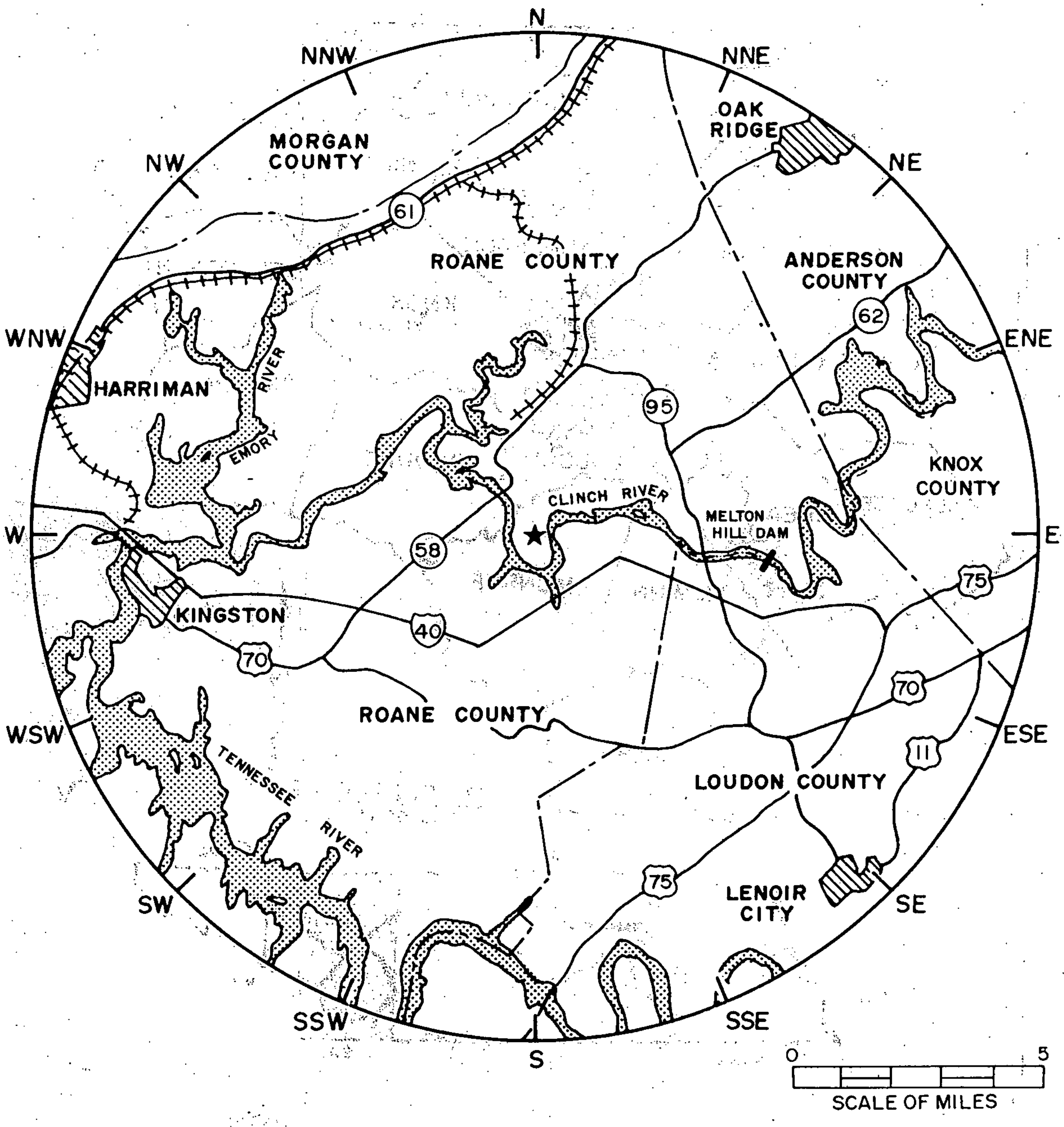



Figure 2.1-2. Urban Areas Within 10 Miles of the Clinch River Site

6647-2

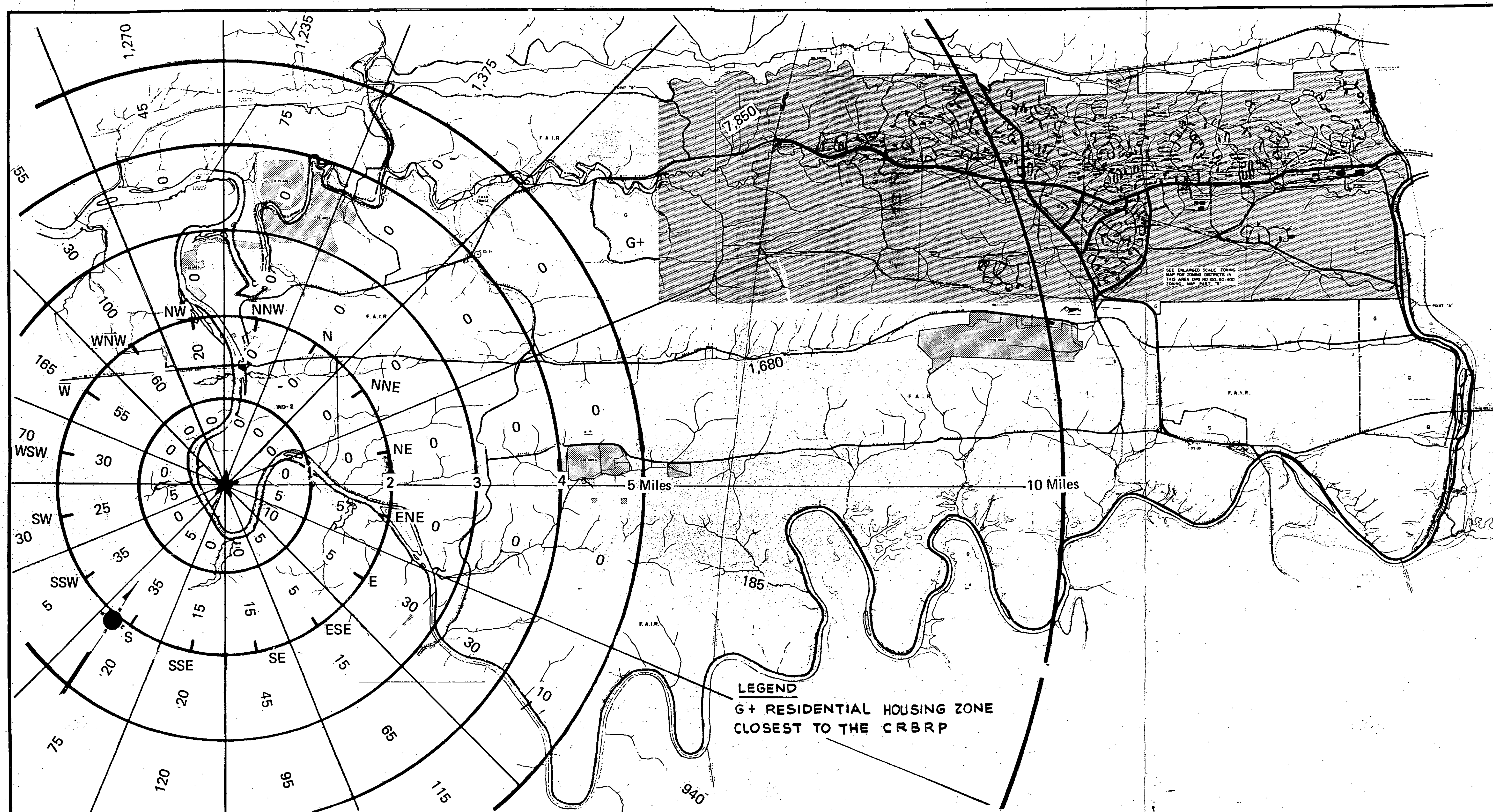



**Figure 2.1-2a. City of Oak Ridge, Tennessee (Reduced Scale Zoning Map Showing Population Distribution Within 10 Miles of the CRBRP)**

Figure 2.1-3. Aerial View of Clinch River Site

6647-3

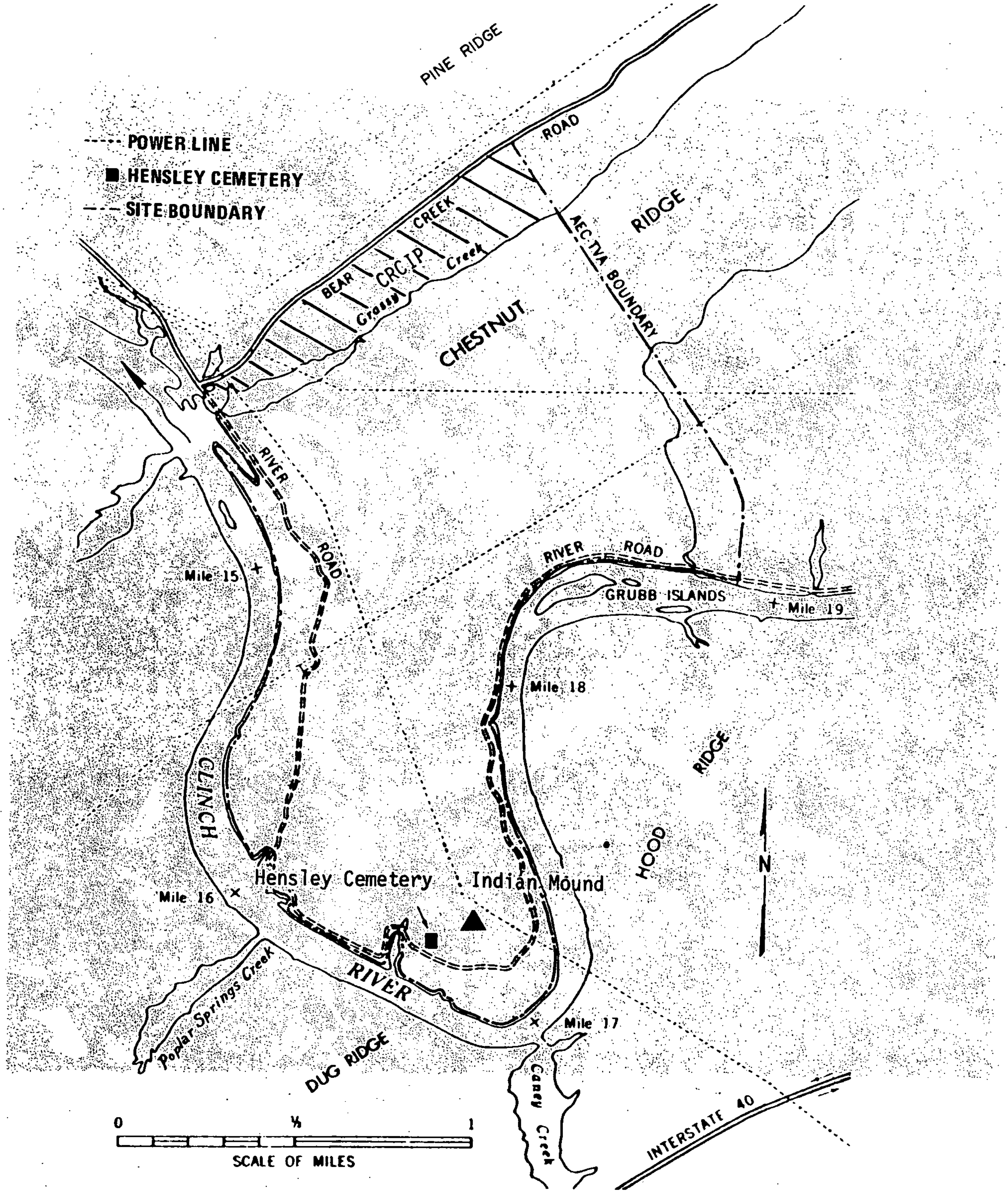
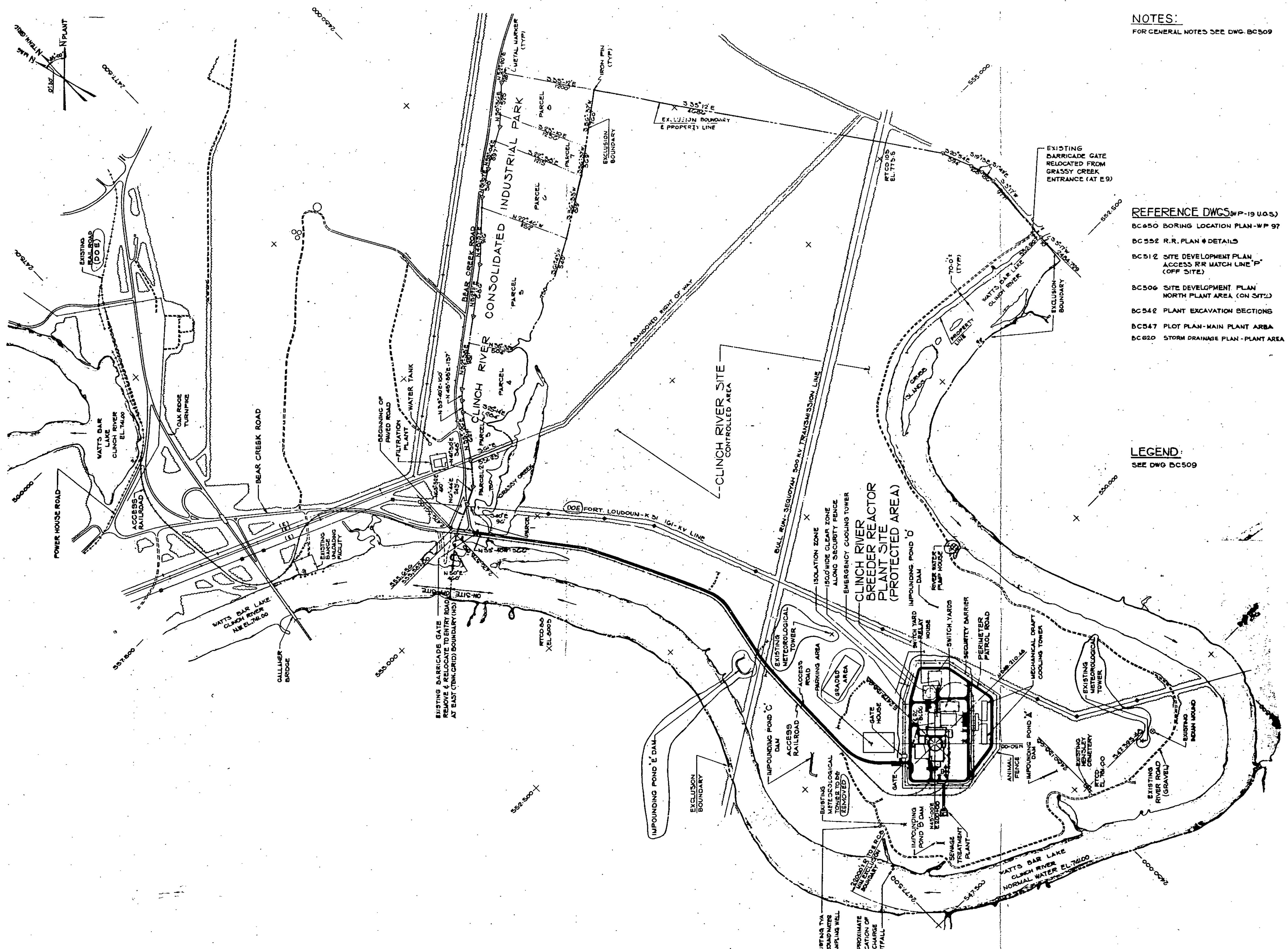



Figure 2.1-4. Clinch River Site

6647-4

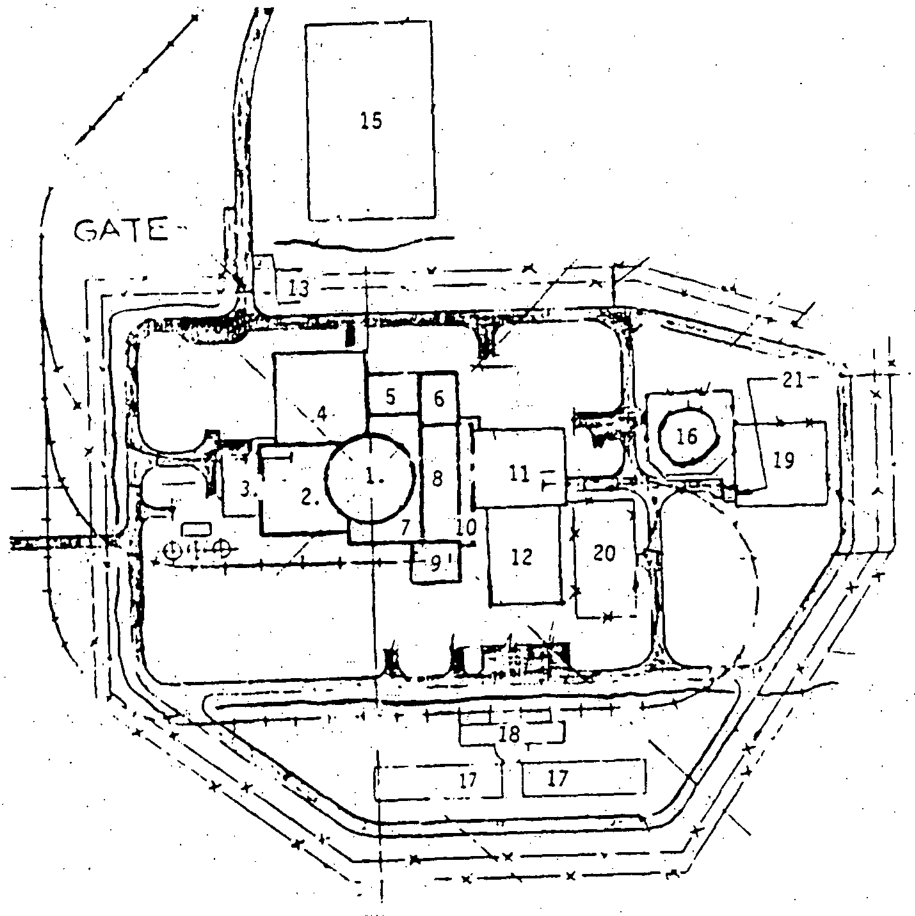



BC-501-4

Figure 2.1-5 - Site Location and CRBRP Layout



Amend. 53
Jan. 1980




1. REACTOR CONTAINMENT BUILDING
2. REACTOR SERVICE BUILDING
3. RADWASTE AREA
4. PLANT SERVICE BUILDING
5. CONTROL BUILDING
6. DIESEL GENERATOR BUILDING
7. INTERMEDIATE BAY
8. STEAM GENERATOR BUILDING
9. MAINTENANCE BAY
10. AUXILIARY BAY
11. TURBINE GENERATOR BUILDING
12. MAINTENANCE SHOP & WAREHOUSE
13. GATE HOUSE
14. 
15. PARKING LOT
16. EMERGENCY COOLING TOWERS
17. COOLING TOWER
18. C. W. PUMP HOUSE
19. GENERATING YARD
20. STARTUP RESERVE YARD
21. SWITCHYARD RELAY HOUSE

NOTE: HEAVY LINES INDICATE SAFETY-RELATED (CATEGORY I) STRUCTURES

Figure 2.1-5A. Arrangement of Plant Structure

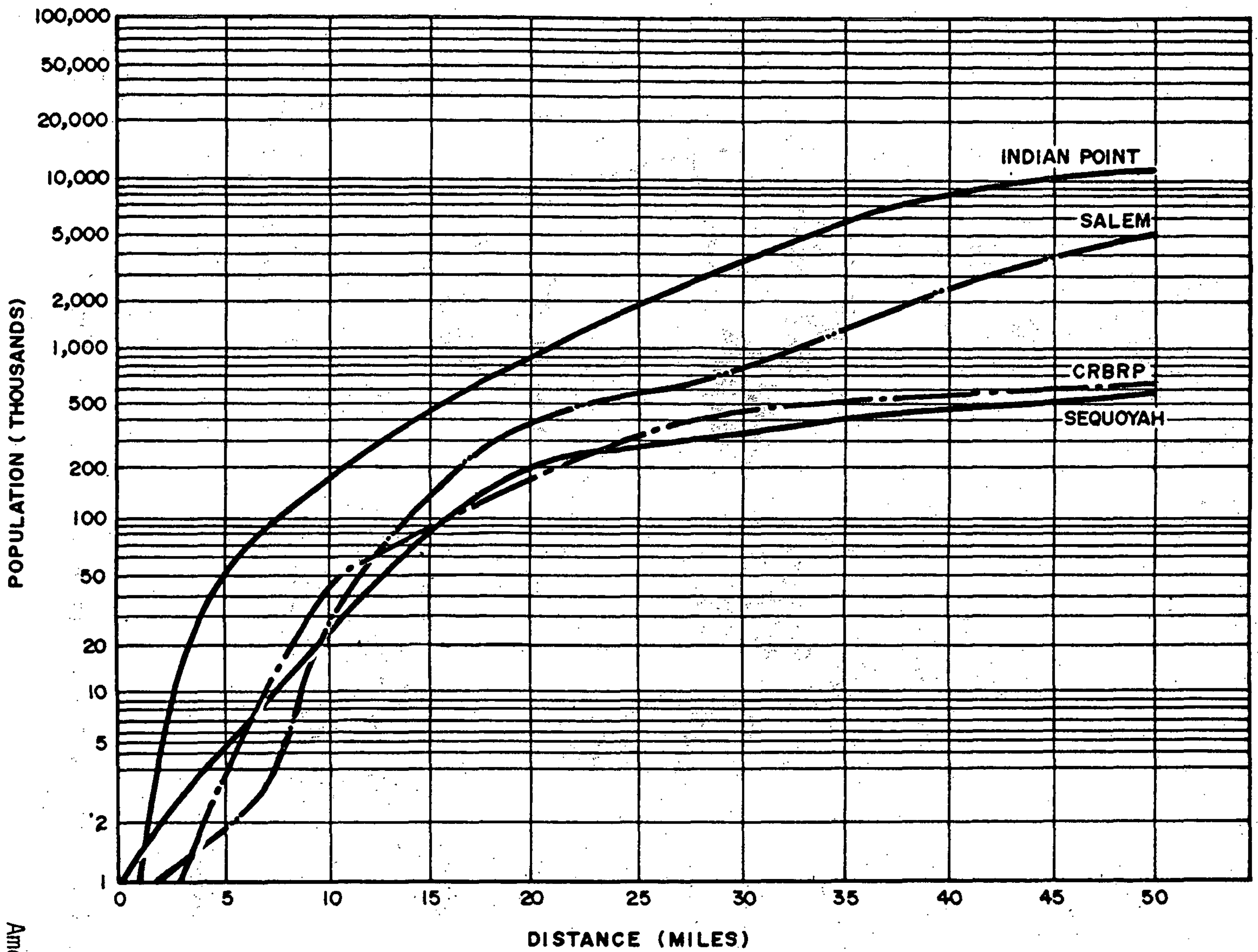



Figure 2.1-6

COMPARATIVE POPULATION DISTRIBUTION SURROUNDING NUCLEAR PLANT SITES



Amend. 15
Apr. 1976

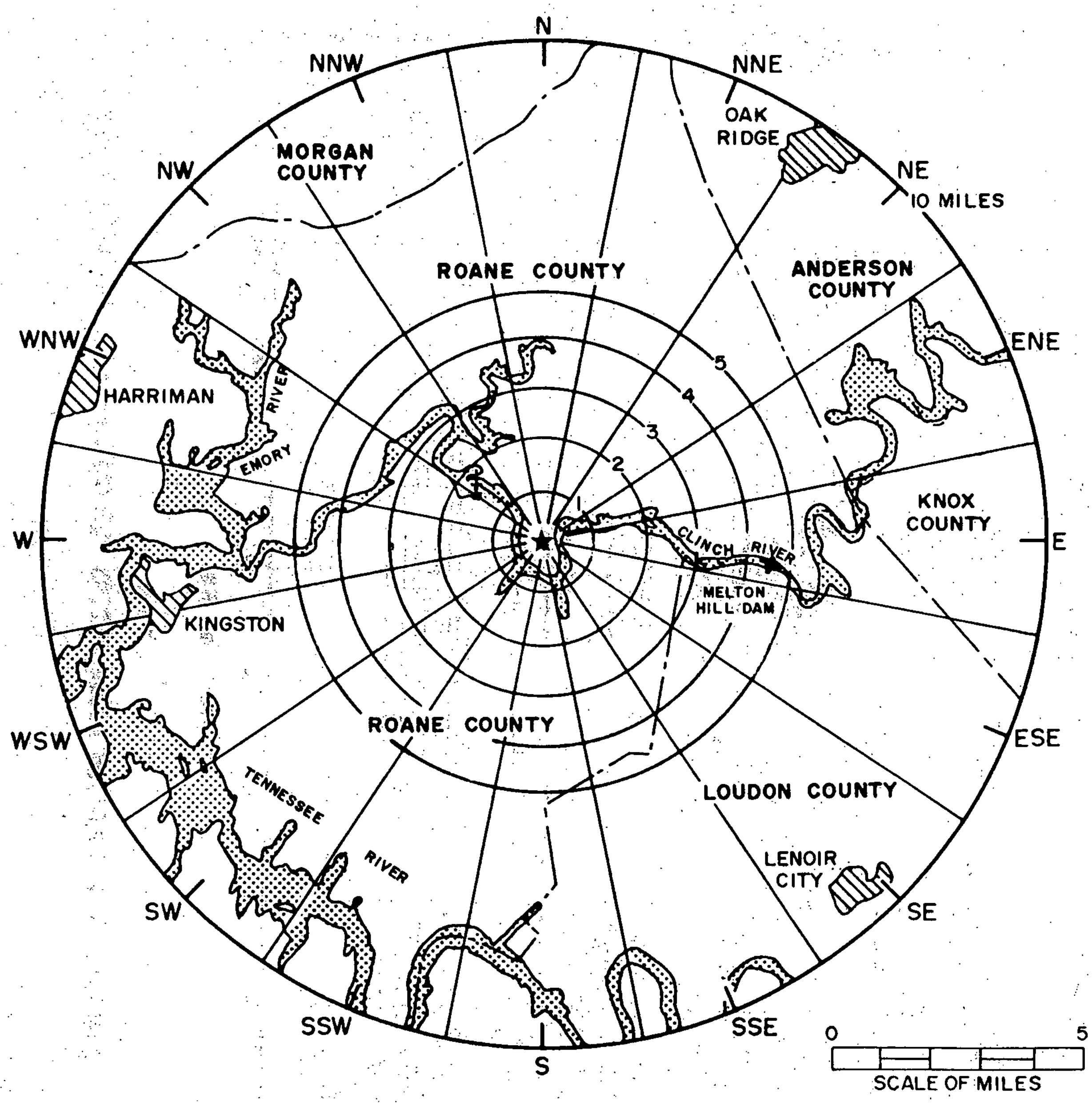



Figure 2.1-7. Sector Designations for 1, 2, 3, 4, 5 and 10 Miles

6647-7

DELETED

Amend. 15
Apr. 1976

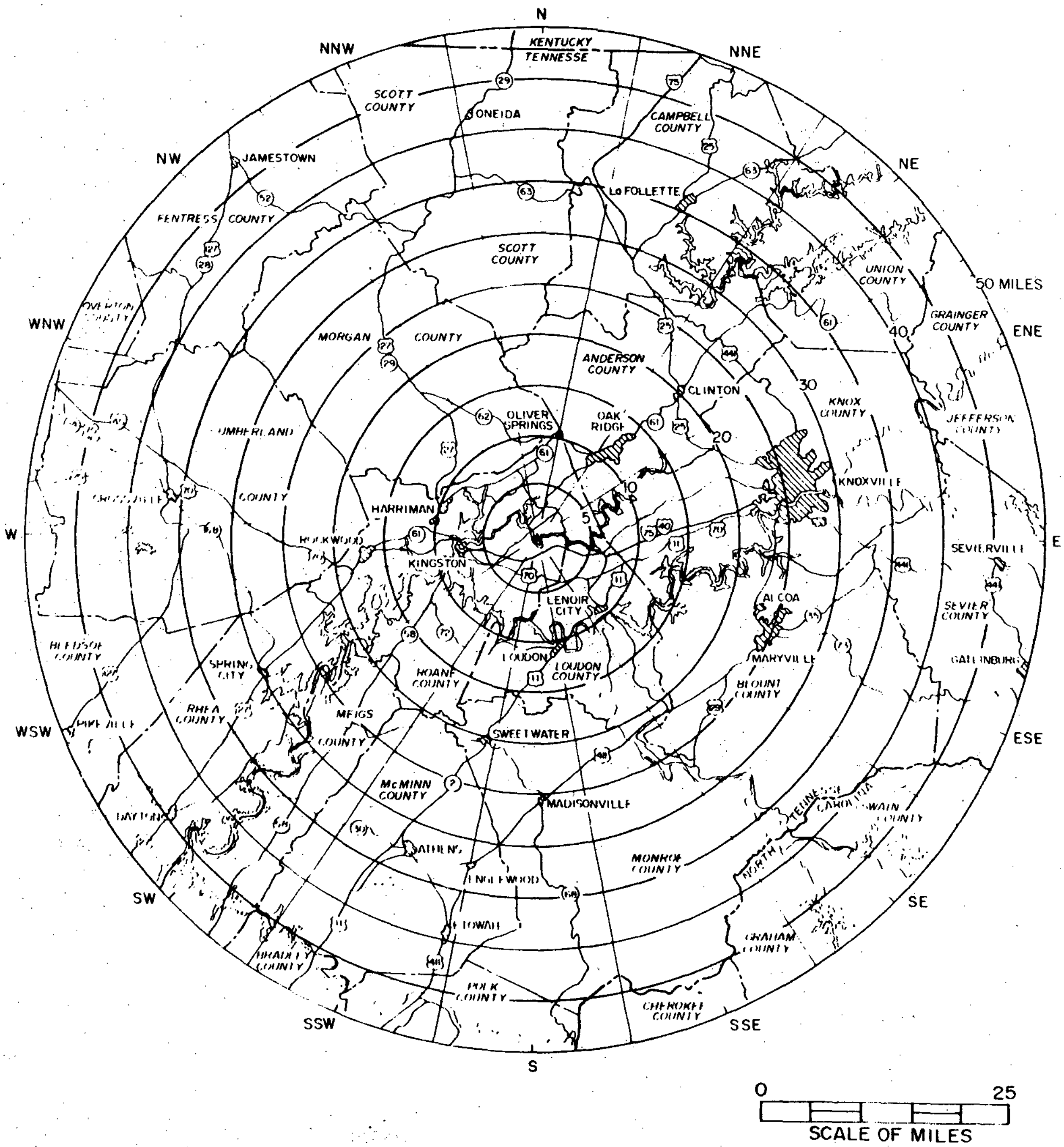



Figure 2.1-9. Urban Centers Within 50 Miles of Clinch River Site

6647-9

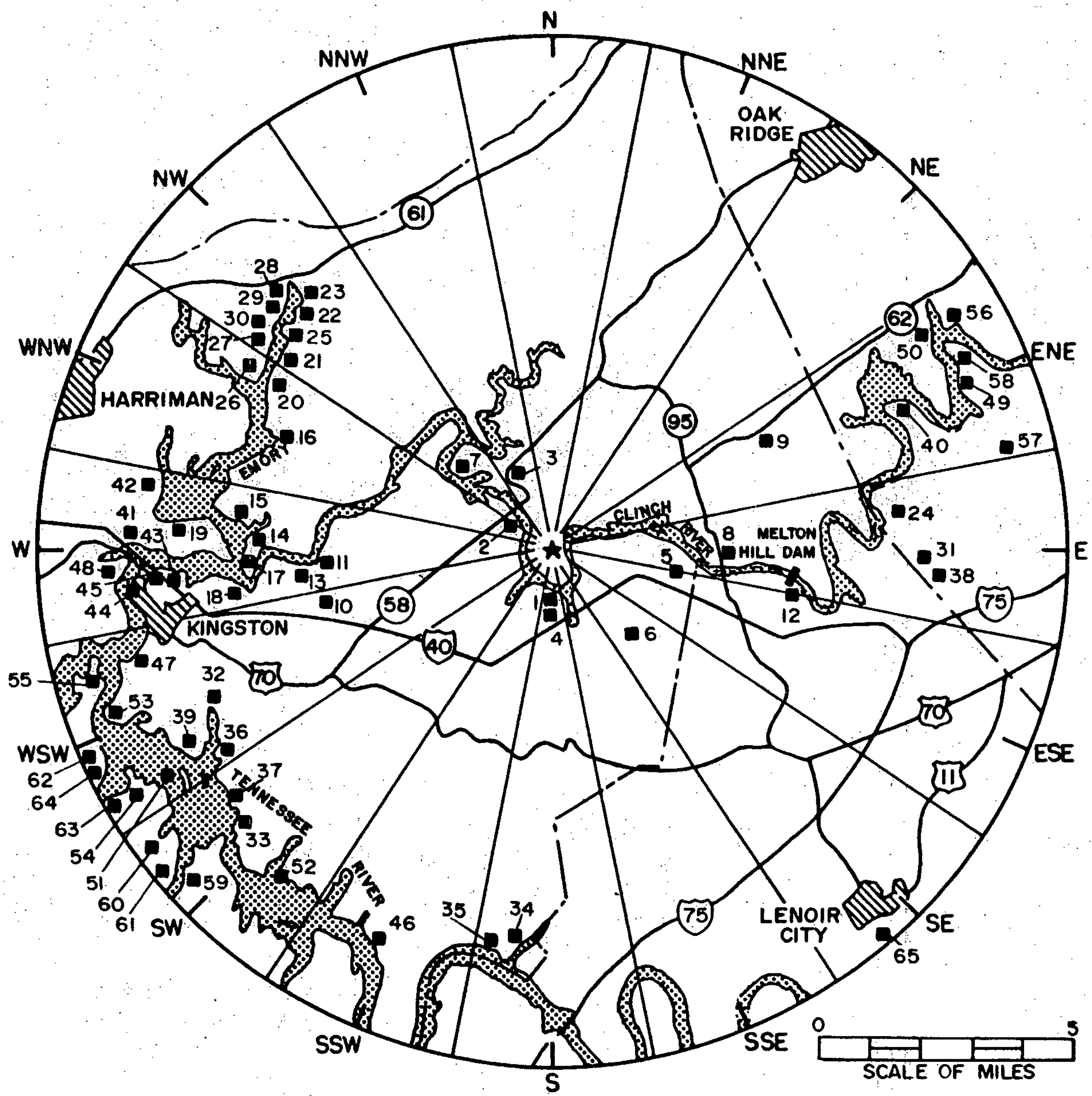



Figure 2.1-10 Recreational Areas Within 10-Mile Radius of CRBRP

Amend. 15
Apr. 1976

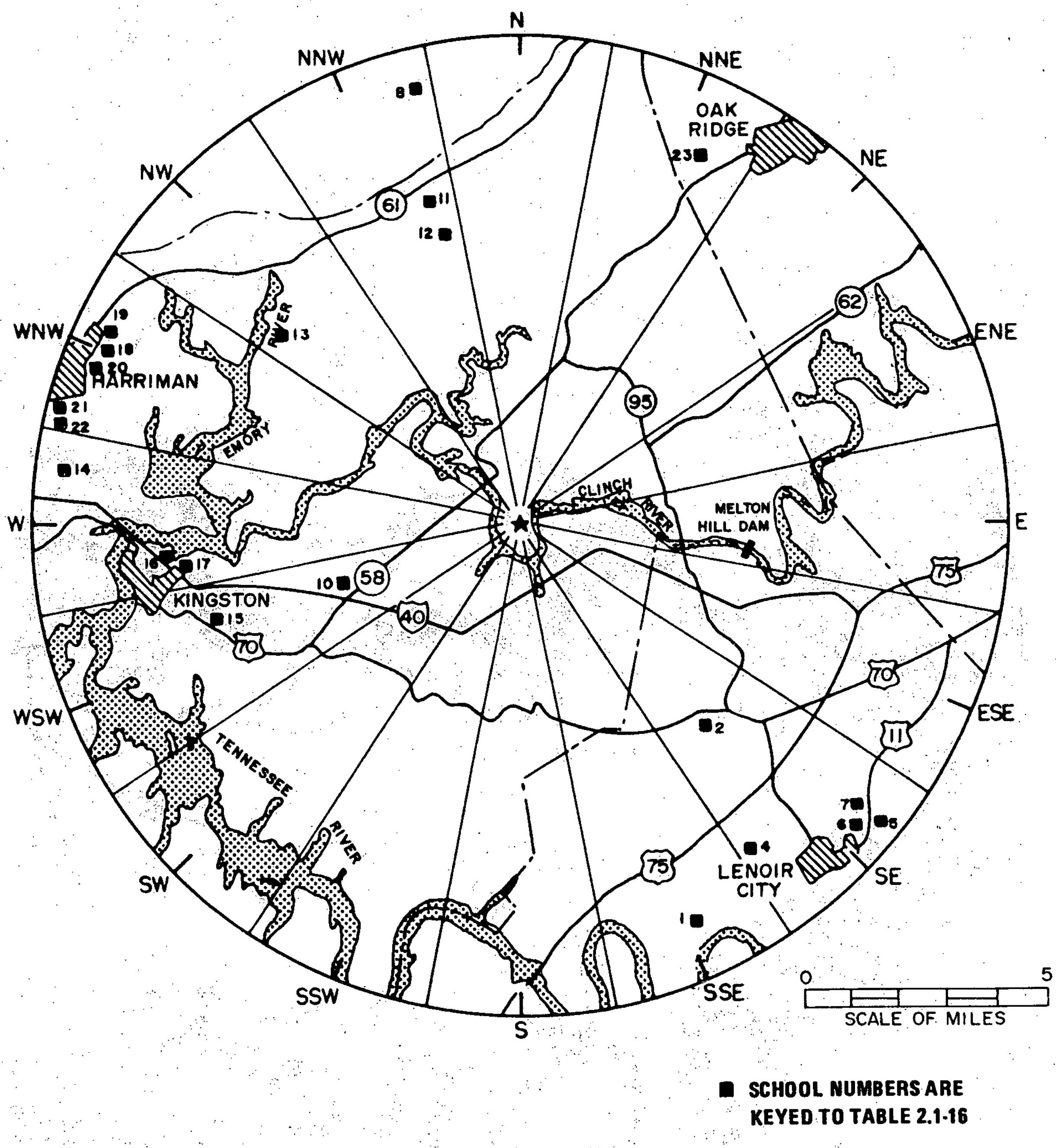



Figure 2.1-11. Schools Within 10-Mile Radius of Clinch River Site

6647-11

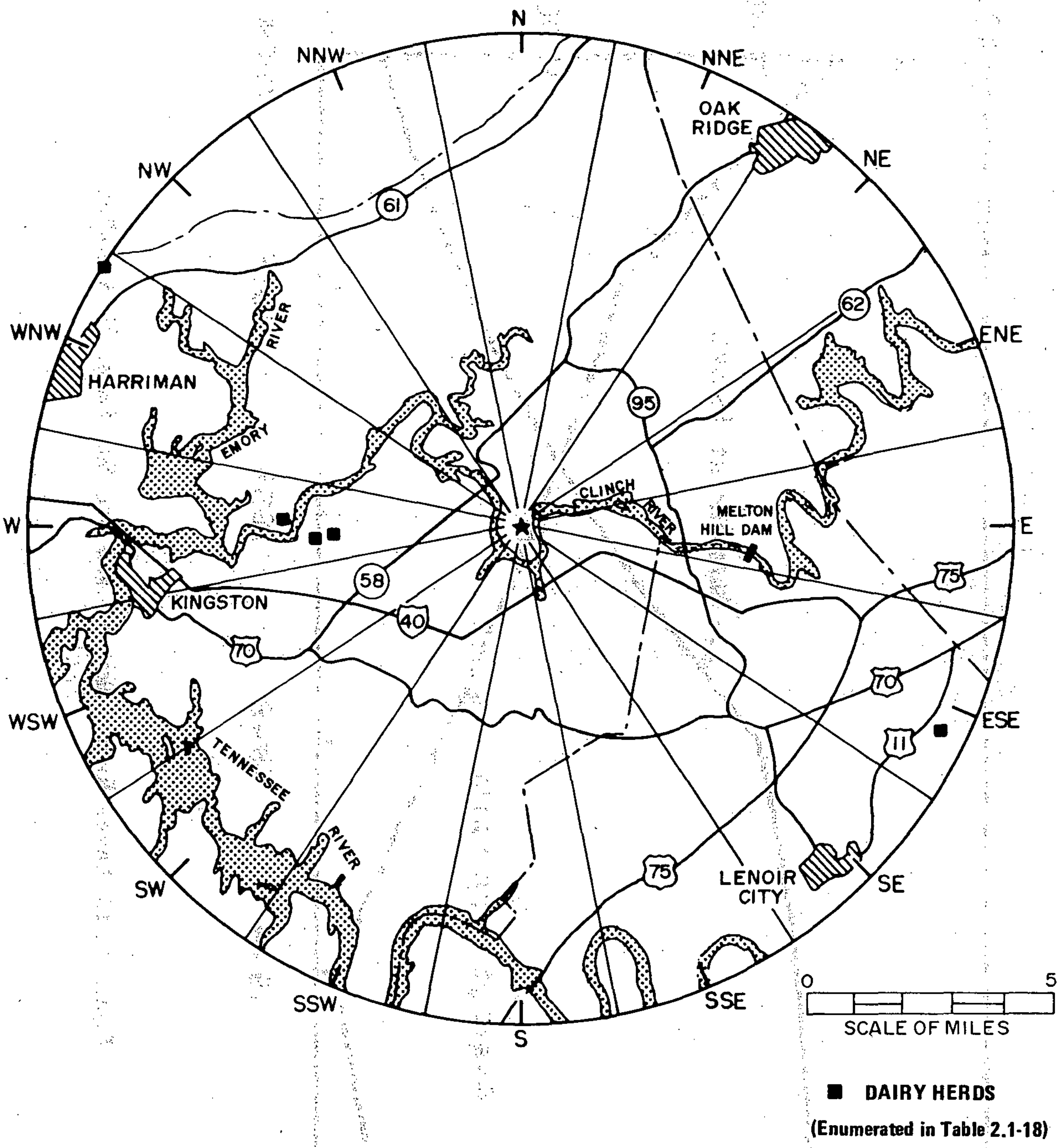



Figure 2.1-12. Dairy Herds Within 10-Mile Radius of Clinch River Site

6647-12

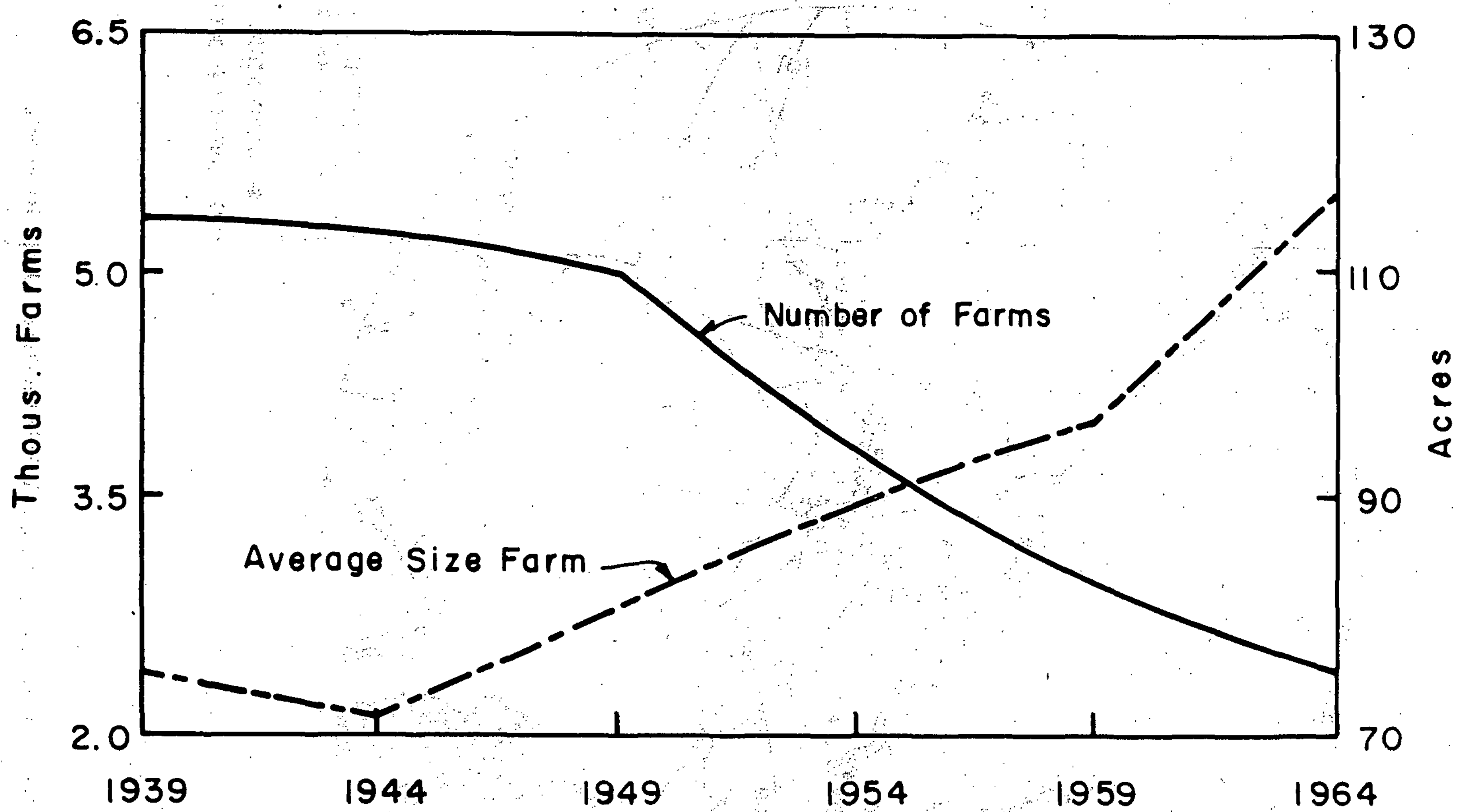



Figure 2.1-13. Emory River Valley, No. and Avg. Size Farms [8] 1939-1964

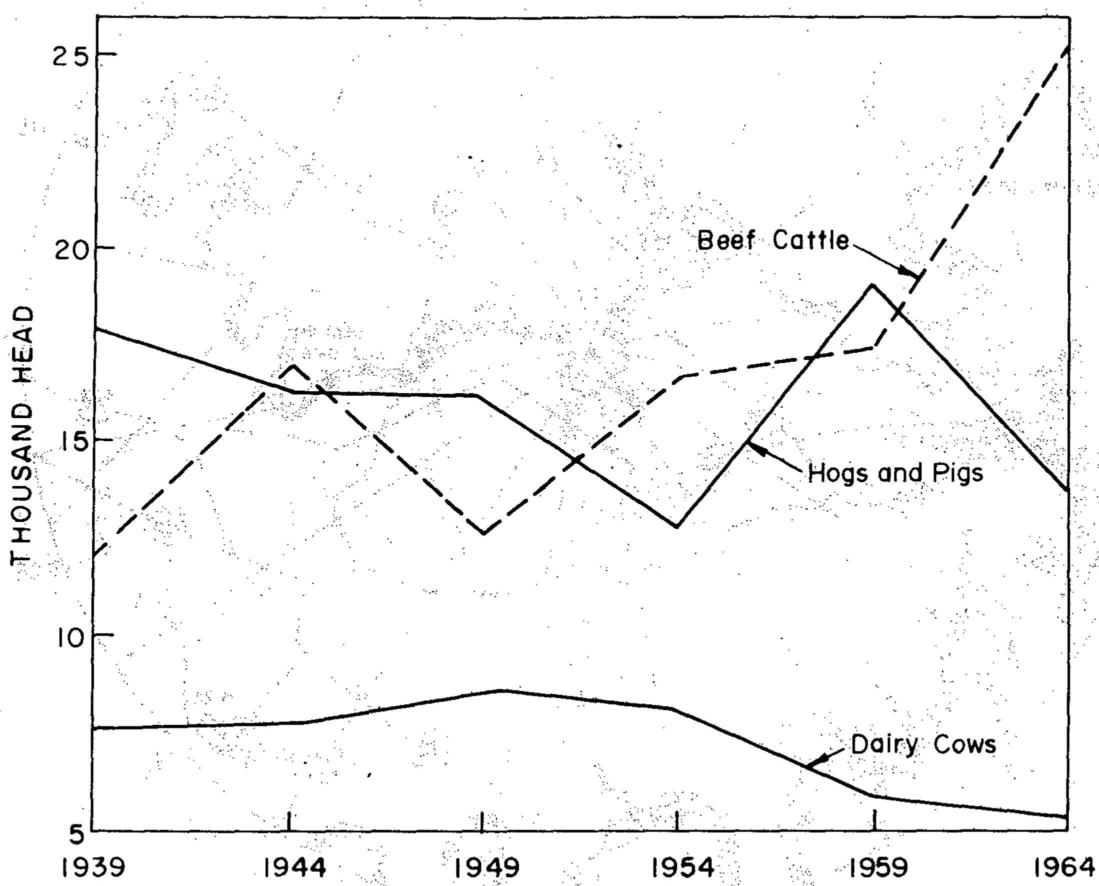
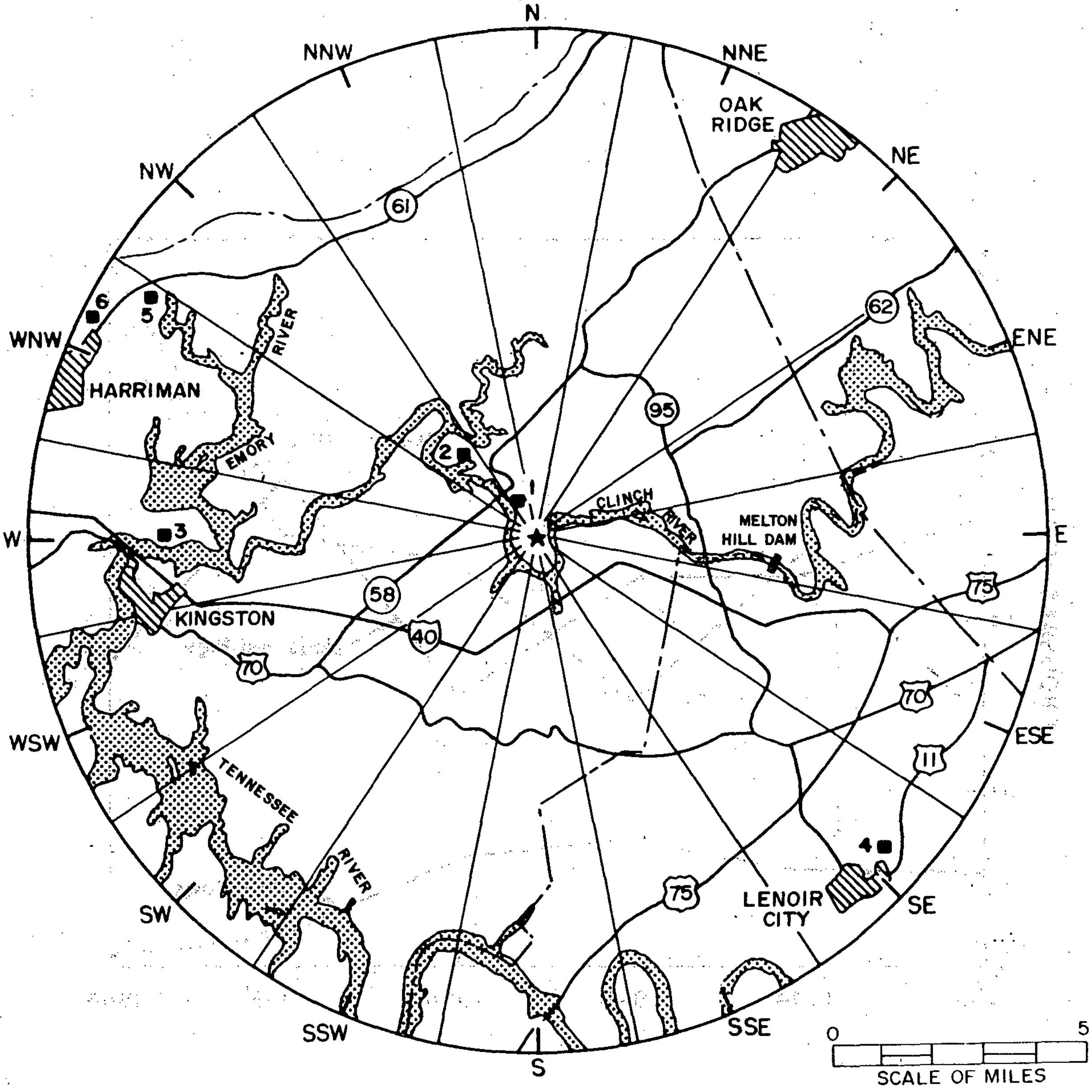



Figure 2.1-14. Emory River Valley, Livestock on Farms (Ref. 8) 1939-1964

6647-14




■ INDUSTRIAL WATER SUPPLY
NUMBERS ARE KEYED TO
TABLE 2.1-19

Figure 2.1-15. Industrial Water Supplies Within 10-Mile Radius of Clinch River Site

6647-15

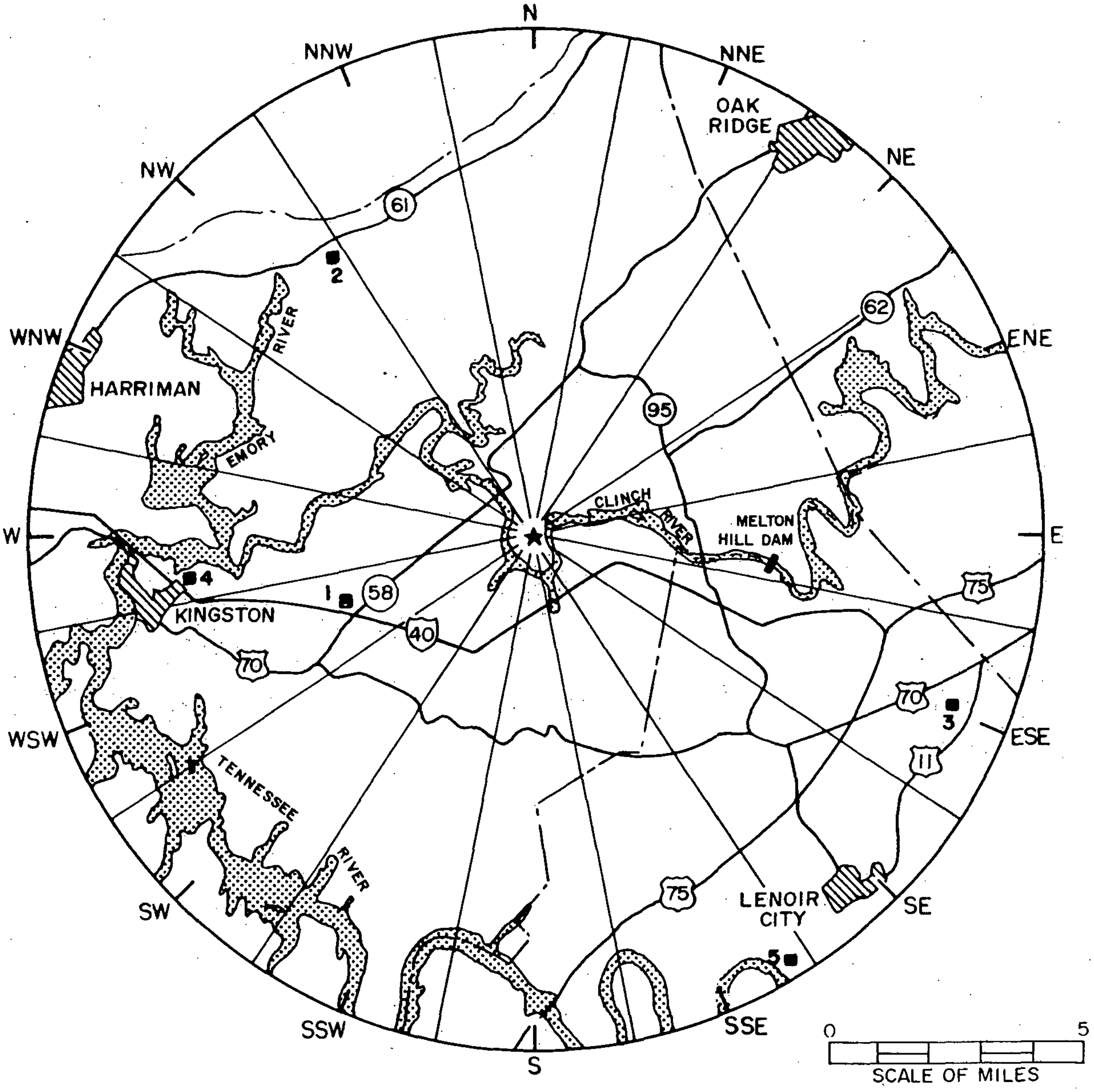



Figure 2.1-16. **Public Water Supplies Within 10-Mile Radius of Clinch River Site**

6647-16

## 2.2 NEARBY INDUSTRIAL, TRANSPORTATION AND MILITARY FACILITIES

### 2.2.1 Locations, Routes and Descriptions

#### 2.2.1.1 Industry

Land adjoining the Site is zoned F.A.I.R., suitable for
6 forestry, agriculture, industry, or research use; the Site 28
is zoned Industrial 2. Three large industrial activities
are located within 5 miles of the plant site, as shown in Figure 2.2-1.
15 These are the Oak Ridge Gaseous Diffusion Plant about 3 miles
north-northwest, the Oak Ridge National Laboratory about
4 miles east-northeast and the TVA Melton Hill Dam about 4.5 miles east.
15

Enriched uranium is produced at the Oak Ridge Gaseous
Diffusion Plant (ORGDP). There are about 4,600 employees
15 at ORGDP. Oak Ridge National Laboratory (ORNL) is a
research and development facility employing approximately
4,000 people. ORNL's work covers reactor and chemical technology,
15 radiation effects, controlled fusion and many other basic and
applied research activities. The Melton Hill Dam provides
hydroelectric power to the TVA system and extends navigation
up the Clinch River. A small crew attends the locks but the
15 power unit operates unattended.

In addition, one small industrial activity is located on a 33-
acre parcel of land in the Clinch River Consolidated Industrial Park (CRCIP)
about one and one-half miles north of the center of the plant site. The
industry, U.S. Nuclear Inc., fabricates neutron absorbers for power reactors
and fuel elements for test reactors and employs 95 people. All nuclear
material handling by this industry is done under controlled conditions in
accordance with governing safety and health regulations. Another firm, Nuc-
lear Environmental Engineering, Inc., has purchased a five-acre parcel. To date,
31 a building site has been prepared on the property but construction has not been
15 initiated. Nuclear Environmental Engineering, Inc., plans to use the
site for the calibration and resale of radioisotopes for use in education,
research, and industry. They also plan to manufacture radioisotopes
generators and radioactive tracers for oil fields and other uses. The
relative location of these two industries within the CRCIP is shown on
Figure 2.2-2. 28

Between 5 and 10 miles there are two additional large
15 industrial activities. They are ERDA's Y-12 facility, 9 miles
northeast, and TVA's Kingston Steam Plant, 7-1/2 miles west. The
15 Kingston Steam Plant is a fossil-fired electrical generating plant
and has about 500 employees. Installed capacity is 1,700,00
kilowatts. It is a major supplier of power to the Oak Ridge
Gaseous Diffusion Plant. The Y-12 facility provides production
15 and research and development facilities for ERDA. This plant
employs about 6,500 people.

2.2.1.2 Minerals and Mining

There is no mineral production within the 10-mile radius; however, mineral production does play an important role in the area, (Ref. 1) particularly in Morgan County where strip mining for coal has been established for many years.

2.2.1.3 Transportation

2.2.1.3.1 Highways

One major highway, Interstate 40, passes approximately 1.25 miles
south of the plant site as shown in Figure 2.2-3 (Ref.2). The closest
interchanges on I-40 are State Routes 58 and 95, which are about four miles
and three miles, respectively, from the plant site location. Existing average
daily traffic near the Site is highest for Interstate Route I-40 and equals
17,530 vehicles per day at the interchange of I-40 and State Route 58.(Ref. 3).
Between this interchange and Oak Ridge, along State Routes 58-95 and US 70
(Kingston Pike) to Kingston, the average daily count equals only 1,970.
Along route 95, between I-40 and the junction of Route 58, the average
daily count equals 3,330. At the interchange of I-40 and Route 95, the
15 average daily traffic count is 14,680.

2.2.1.3.2 Rail

The closest major main rail line to the Site is Harriman Junction, approximately 10 miles northwest of the Site. It is served by both the Cincinnati, New Orleans and Texas Pacific (CNO & TP) and the Southern Railway.

2.2.1.3.3 Water

The U.S. Army Corps of Engineers operates the locks at Meltom Hill Dam
and keeps logs of all barge traffic. Total tonnage and average weight per
barge of commercial traffic through Melton Hill Dam for the period 1966-1975
is given in Table 2.1-15. Barge traffic passing the CRBRP site at the present
time is primarily steel products. None of this traffic contains explosive,
toxic, or hazardous materials. There have been no accidents involving barges
reported near the CRBRP site. 28



Amend. 28
Oct. 1976

There is a potential for an increase in barge traffic due to two proposed coal barge loading facilities and a public use terminal planned for Melton Hill Lake. This will have no impact on the CRBRP since none of this potential increase traffic is expected to contain explosive, toxic, or hazardous materials. 28

2.2.1.3.4 Air

Airports located near the site are as follows:

| | Name | Type | Distance and Direction (miles) |
|---|---|---|---|
| 15 | Meadowlake Air Park | Sport | 10 SW |
| | Oak Ridge Air Park | Sport | 11 NNE |
| | Rockwood Municipal | Business/Sport | 18 W |
| | McGhee-Tyson | Commercial | 28 ESE |

Of the four, only MCGhee-Tyson (Knoxville) has scheduled commercial flights. The nearest flight path (v16) is about 10 miles south of the Site. Checking U.S. Flight information, Figure 2.2-4 indicates that aircraft approaching McGhee-Tyson would be at a minimum altitude of 5,000 feet as they pass 10 miles south of the Site. The nearest holding pattern for McGhee-Tyson is approximately 30 miles northeast of the Site.

#### 2.2.1.4 Military Facilities

There are no military facilities or bases within the 10-mile radius.

### 2.2.2 Evaluations

The CRBRP cooling water intake structure will be located so as to be protected from Clinch River barge traffic. As the site is located in a mild temperature zone, the intake structure should not be subject to ice blockage or damage. The closest source of accidental upstream release of corrosive liquids or oil is Oak Ridge National Laboratory (approximately seven miles distant) via White Oak Creek and the Clinch River. This facility uses nominal quantities of chemicals. The effects of any accidental release or corrosive chemicals on the CRBRP intake structure from ORNL would be inconsequential. As other potential sources of accidental
27 liquid chemical or oil releases are remote in the upper watershed region of the Clinch River system, the effects are not a realistic consideration for the CRBRP.

6 As discussed in Section 2.2.1 there are three nuclear-related
15 facilities located within a 5 mile radius of the CRBRP Site. These 3
27 facilities are designed for, and comply with all appropriate Federal and State regulations.

Existing monitoring programs in the CRBRP area provide characterization of background radiation. Pre-Operational monitoring programs
15 described in Section 6.0 of the CRBRP Environmental Report will provide an assessment of the background in the Site area and will be supplemented by those existing programs of other facilities adjacent to the CRBRP. Such a program will result in an assessment of any significant radiological impact these other nuclear facilities may have on the environment around the CRBR.

The most recent available ORNL Environment Monitoring Program measurements are discussed in Section 2.8 of the CRBR Environmental Report. Results show that air and aquatic activity levels in the site area are well below maximum permissible concentrations as outlined in 10CFR20 Appendix B for unrestricted areas.

Oak Ridge Operations (Ref. 4) has identified a potential accidental release of anhydrous hydrofluoric acid (AHF) as a maximum postulated airborne release from the ORGDP. Such a release is considered to be extremely unlikely, requiring the simultaneous failure of a number of administrative and engineering controls.

Containment for AHF and operations with AHF at ORGDP are designed such that large releases are not considered credible. The CRBRP site is about three air miles south of the ORGDP while the predominant wind current is from the southwest to the northeast parallel to the ridge and valley orientation. This results in the CRBRP site being essentially cross wind to the ORGDP. In addition, there are two ridges (Pine and Chestnut) between the two sites. This topography further reduces the probability of an airborne release from the ORGDP having a significant effect on the CRBRP site.

While it is considered highly unlikely, it is possible that 15,000 pounds of AHF could be lost from containment as a result of an accident due to rupture of a storage tank (40 ton capacity). A release has been postulated assuming the following:

(a) A quantity of 2,000 pounds of the AHF released is evolved as gas to the atmosphere.

(b) The release occurs over a period of 15 minutes before being detected and controlled.

(c) Meteorological conditions are identical with the 0-8 hr. conditions discussed in PSAR Section 2.3 utilized in Chapter 15 for accident analyses.

6

The resulting concentrations of HF downwind from the release for the postulated conditions are indicated in Table 1:

Table 1

| Downwind Distance (Miles) | Peak HF Concentration ($mg/m^3$) (15 minute duration) |
|---|---|
| 1 | 800 |
| 3 (CRBRP) | 240 |
| 10 | 65 |

The National Academy of Sciences - National Research Committee on Toxicology (Reference 5) has recommended emergency exposure limits for HF of 16 $mg/m^3$ for ten minutes, 8 $mg/m^3$ for 60 minutes for military and space operations. Concentrations of HF in the range of 25 $mg/m^3$ for several minutes will cause respiratory discomfort in humans (Ref. 6).

Implications of the postulated release upon the safe operation
15 of the CRBRP involve the effect of such a release upon habitability of the control room. Per Section 2.3 (Table 2.3-24), the 0-8 hr. median wind speed assumed for meteorological conditions is 0.42 m/sec. The 3 miles distance from ORGDP to CRBRP site implies about 3 hours would elapse between time of release and arrival of the AHF cloud at the site. Use of higher mean velocities would reduce the communication time but would increase the dispersion of the AHF and reduce the concentration. Using average annual weather would still provide 42 min. for isolation of the control room and would reduce the peak concentration by more than an order of magnitude. It is evident that there would be ample time for communication between the ORGDP and CRBRP in the interim between release and arrival of the AHF cloud. This communication would result in procedures to isolate the control room and its inhabitants thus resulting in no safety impact on the CRBRP.

Details of the communications between ORGDP and CRBRP assuring immediate contact in the event of such releases will be available as
6 part of the Site Emergency Plan.

Regarding the potential for impact on the CRBRP from the U.S. Nuclear, Inc. facility located in the CRCIP, the Project has examined the facility's special nuclear material license (Ref. 9) and Safety Evaluation Report (Ref. 10) (SER) and had discussions with the U.S. Nuclear, Inc. facility nuclear safety consultants. The major accidents that might expel airborne radioactive material from U.S. Nuclear have been analyzed in the facility's SER and are the potential for a fire and a criticality incident. Neither of these incidents would result in significant doses at the CRBRP, and therefore U.S. Nuclear, Inc. will have no impact on the
safe operation of the CRBRP. 28

Principal highway routes serving the CRBRP Site area are shown in Figure 2.2-3. One major highway, Interstate 40, passes approximately 1.25 miles south of the CRBRP Site, with interchanges at Tennessee Routes 58 and 95 at approximately four and three miles, respectively, from the plant site. Shipments to and from the industrial facilities described in Section 2.2.1.1 would be conducted principally over the following routes: to and from the ORGDP, State Route 58 (Oak Ridge Parkway) and Blair Road; to and from the ORNL, State Route 95 and Bethel Valley Road; to and from Y-12, Bear Creek Road off either Route 95 or Scarbo Road; Melton Hill Dam, Route 95 and Buttermilk Road and to/from the CRCIP, Route 58 and Bear Creek Road. The closest points at which shipments would pass the CRBRP occur at the CRCIP, 1.5 miles and I-40, 1.25 miles. Due to the distances involved, shipments to and from the industrial facilities of the Oak Ridge area will not impose potential adverse impact upon the CRBRP.

Estimates of frequent shipments of toxic materials on I-40 were obtained from the Tennessee Public Service Commission (see Table 2.2-1). Calculations of concentrations of these materials in the CRBRP control room from an accident on I-40 show that hydrogen bromide and hydrogen fluoride could significantly exceed Threshold Limit Values (TLV) assuming major releases and adverse meteorology. A hydrogen fluoride detector is presently included in the control room air intake system design. Further evaluation will determine the need for addition of a hydrogen bromide detector versus the acceptability of reliance on direct operator detection due to its low odor threshold.

Other toxic materials identified by the State which are frequently shipped on I-40 are chloropicrin and acetic anhydride. Highly conservative calculations of control room concentrations from accidents involving these materials resulted in exceeding TLVs by only 15% and 40% respectively. (Conservatisms included complete instaneous release of container contents, G stability, centerline concentration, 0.2 m/sec wind speed, the accident occurring at the closet proximity to the site, and a constant wind direction with no meandering for about three hours.) Detectors for these materials are thus not considered necessary for the control room air intake.

The nearest natural gas pipeline to the CRBRP is a six-inch line which borders the east boundary about one and a third miles away from the facility. This line runs north to south and supplies gas to Lenoir City, Tennessee. Due to the remoteness of the pipe line, no credible explosion of this pipeline could adversely affect the CRBRP.

There are no stone quarries, oil, gasoline plants, or storage facilities near the CRBRP. Consequently, there are no potential effects of explosion or fires from these facilities. The plant building complex is located a minimum of 300 ft. from the nearest tree line in any direction. Due to the separation between plant buildings and the forest and the extensive use of fire retardant construction materials, a local forest fire poses no threat to the integrity of the plant.

There will be no effect from chlorine gas leakage as none is stored on site. Non-hazardous sodium hypochlorite is utilized for plant service instead of chlorine.

There are no on or off site airborne pollutants that may affect plant components.

The closest commercial airport is the McGhee-Tyson (Knoxville) terminal 28 miles east southeast. Checking U.S. flight information indicates that aircraft approaching McGhee-Tyson would be at a minimum altitude of 5000 feet as they pass 10 miles south of the site.

The closest airport is Meadowlake, 10 miles southwest, which handles sport-type aircraft. Therefore, as the CRBRP is not in the vicinity of airport flight holding patterns or flight paths, the impact is considered to be minimal.

There are no tall structures such as natural-draft cooling towers or tall discharge stacks on the facility which may damage critical equipment due to collapse. Discharge vents are located approximately 10 feet above the roofgrade. Cooling structures utilized are wet mechanical draft cooling towers. The dominant feature of the CRBRP is the Reactor Containment Building which rises to an elevation of 984 feet MSL, 169 feet above grade level.

31

### 2.2.3 New Facility/Land Use Requirements

The expanded role of ERDA in developing the country's energy resources will inevitably require expansion of many existing research and development facilities as well as the establishment of new ones.

To aid in identifying long range planning requirements for its Oak Ridge reservation, ERDA requested its Oak Ridge Operations Office to conduct a survey of Federal real property holdings at Oak Ridge, the intent of this study was to "establish a basis for a long-range land use plan to accommodate both present and projected ERDA program requirements." The results of this study are reported in Reference (7) and a brief summary of potential new facilities and land use based on this report are summarized below.

Primary Research Facilities

- Future additional office space and general support facilities for ORNL could require as much as 100 acres. The majority of land required for office space will probably be adjacent to existing ORNL facilities.
- A MSR (Molten Salt Breeder Reactor) test facility proposed for Melton Valley would require approximately 10 acres with no anticipated change in the security/safety buffer zone.
- A superconducting magnet fabrication and test facility, as part of the fusion-research program located at the Y-12 plant, would consist of two structures occupying 40,000 square feet on a two acre site.
- An Experimental Power Reactor (EPR) could be constructed, based on the success of the fusion research program. This reactor would require a minimum of 30 acres for the plant and its exclusion radius. Two potential sites are under consideration. One is the west end of the Y-12 plant, west of the present settling basins. The other is an area of approximately 30 acres adjacent to the former ORGDP powerhouse.
- A demonstration house utilizing solar energy or ice-pond storage for heating and cooling would be located on a yet to be defined 2 to 5 acre site. This facility could be built in conjunction with the University of Tennessee on its land in Knoxville.
- A demonstration plant for the bioconversion of wastes to fuel gas. This would require about 5 acres and could be located either at the sanitary facilities at Oak Ridge or at the sewage treatment facilities at Y-12, ORGDP or CARL (Comparative Animal Research Laboratory).
- A low-temperature heat energy recovery facility. Application to a fossil fuel power plant would require the facility to be located at a TVA plant or at the Y-12 or ORGDP power plants. Additional space required at an existing ERDA facility would be 2 to 3 acres.
- A coal-liquefaction process development facility and a coal equipment test facility would require a fenced area of about 10 acres located close to ORNL to permit use of ORNL support facilities. Preliminary planning has identified a site west of the EGCR (Experimental Gas-Cooled Reactor).
- A hydrogen-production process development plant is tentatively planned at the Y-12 plant adjacent to the existing steam plant. An alternative location would be adjacent to the coal liquefaction facility described above. 13

- This physical sciences program at ORNL is expected to grow to support new ERDA programs. The growth of a complex of facilities to provide space for experimental studies and offices is anticipated in the area adjacent to the present ORIC-ORELA.

- The biological research program at ORNL will expand and approximately 15 acres extending eastward to the east portal boundary and lying north of first street represents the anticipated area for growth.

Uranium Enrichment Activities

- While only in the conceptual design phase, and yet to be approved by Congress, a new gas-centrifuge plant is being considered as part of ERDA contingency planning in the event private industry fails to provide additional separative capacity on a timely basis. Preliminary layouts indicate that the new plant would add to the present ORGDP industrial plant complex approximately 320 acres and approximately 200 acres to the present buffer zone.

The sites indicated for locating the proposed facilities described above are at present being considered in the context of long range land-use planning. The final selection of sites for future facilities on ERDA controlled land will be in accordance with the requirements set forth in ERDA Manual Chapter 6202, "Site Selection", 6203, "Site Development Planning" and other chapter requirements as applicable. Manual Chapters 6202 and 6203 are provided as Appendix 2-E to this chapter. Manual Chapter 6202 and its appendix provide the guidance and criteria to be followed for the selection of new sites, and Manual Chapter 6203 and its appendix provide guidance, planning factors, and criteria for site development planning. With regard to potential impacts on CRBRP operation as well as on the environment in general, the criteria provided in the ERDA manual as well as the applicable requirements of 10CFR11 provide for full consideration to be given to impacts on existing facilities during site selection and site development planning for new or existing sites. These requirements provide the assurance that potential new activities on ERDA controlled Land will not pose an undue risk to the continued safe operation of the CRBRP.

In addition to the projected new facility/land use requirements identified for ERDA-controlled property, Exxon Nuclear Company is presently requesting a 2500 acre site on the Oak Ridge Reservation for "the purpose of spent-fuel reprocessing, spent fuel storage, interim waste storage and possible use for other sectors of the nuclear fuel cycle." Of the 2500 acres, the plant site would require approximately 160 acres, with the remainder required as a buffer area. A water pipeline corridor running west from the plant to the Clinch River with a water pumping station at the Clinch River would also be required, as well as a railroad spur.

While the arrangements regarding acquisition of the site by Exxon are not finalized, the requirement for demonstrating the environmental acceptability of the site and the plant safety will be the responsibility of Exxon (Ref. 8). A Preliminary Safety Analysis Report (PSAR) was submitted to
17 NRC on January 28, 1976 in support of a construction permit for this plant. 13

Other potential future activities involving ERDA controlled land could involve the Oak Ridge community. Examples cited in Reference (7) include such items as highway development, an airstrip, and landfill sites.

If the planned or known use for the land is identified prior to it being excessed, appropriate consideration would be given to potential impacts on both CRBRP operation as well as on operation of other ERDA facilities. Excessed land turned over to the community would then be subject to federal, state, and local regulations regarding its intended use. 13

References - Section 2.2

1. Overall Economic Development Program for Morgan County, Tennessee. Morgan County Overall Economic Development Committee.

2. Comprehensive Plan, Roane County, Tennessee, December 1971.

3. Telecon, October 29, 1973, McClain, B., Tennessee Urban Transportation Department, Traffic Statistics to Matoon, P., ESD.

4. "Request for Information on Accidents that Could Occur at the Gaseous Diffusion Plant for the CRBRP PSAR", R. J. Hart (Oak Ridge Operations) to D. R. Riley (RRD) dated December 24, 1974.

5. National Academy of Sciences-National Research Council, Committee on Toxicology, "Guides for Short-Term Exposures of the Public to Air Pollutants: III, Guide for Gaseous Hydrogen Fluoride," PB-203-465, August 1971, pp. 121-134.

6 6. National Academy of Sciences, Biological Effects of Atmospheric Pollutants: Fluorides, U.S. Government Printing Office, Washington D.C., 1971, pp. 238-239.

7. Oak Ridge Reservation Land-Use Plan, ORO-748 August, 1975. 13

17 8. Federal Register, Vol. 40 No. 233, pages 56477-8.

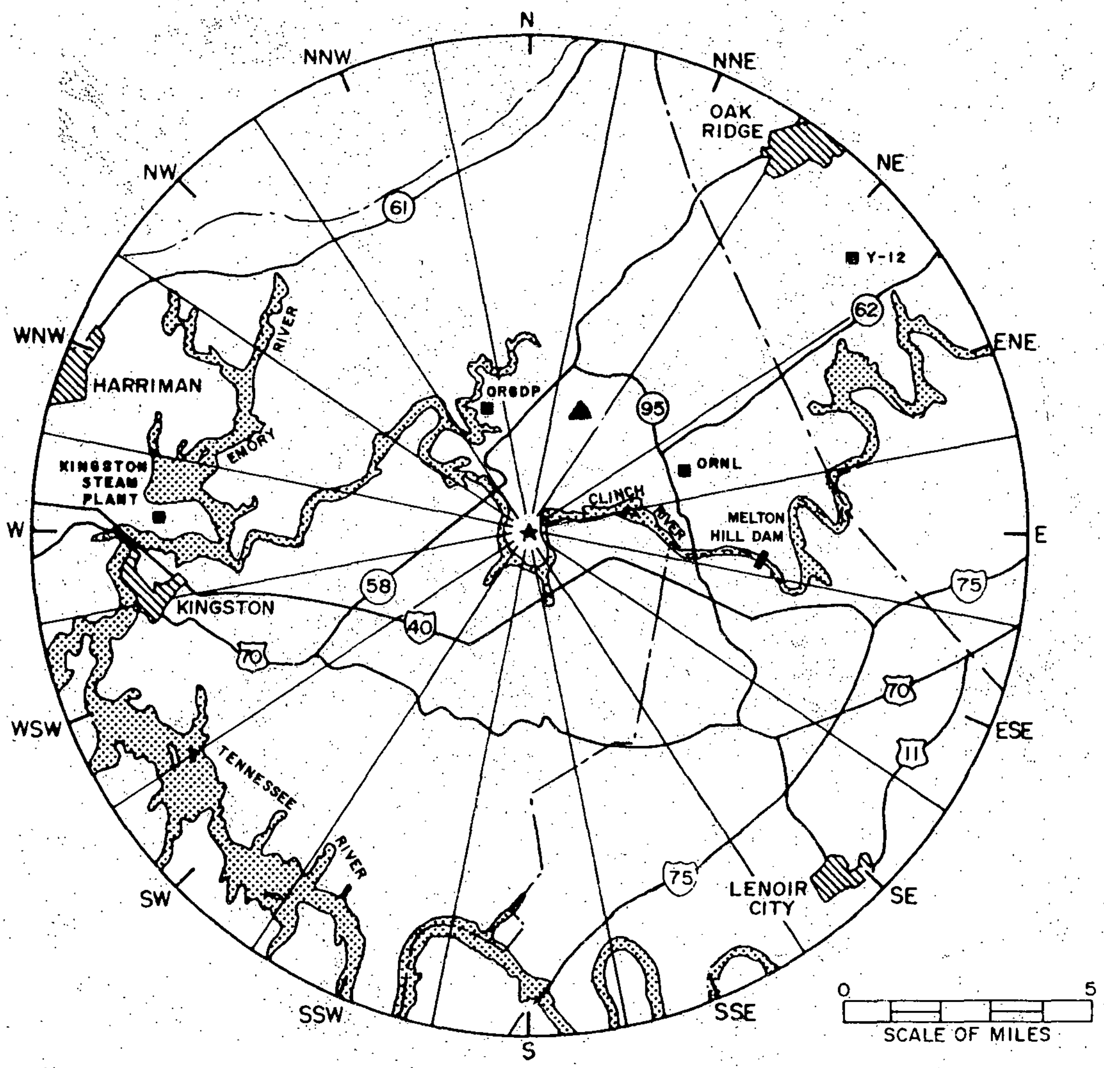
9. U.S. Atomic Energy Commission, Special Nuclear Material License for U.S. Nuclear, Inc., dated November 6, 1973.

10. U.S. Nuclear, Inc., Safety Evaluation Report, Docket No. 70-1319. 28

Table 2.2-1

| Toxic Chemical | Maximum Shipment Weight (lb) | Frequency of Shipment (/wk)* | Concentration in Control Room (mg/m3) | Threshold Limit Value (mg/m3) |
|---|---|---|---|---|
| Bromine | 41,625 | 8 | 245 | 2 |
| Hydrofluoric Acid | 50,000 | 1 | 192 | 2 |
| Chloropicrin | 1,000 | 1 | 2.3 | 2 |
| Acetic Anhydride | 50,000 | 1 | 28 | 20 |

* Estimate obtained from the Tennesee Public Service Commission (1982).




■ **INDUSTRIAL PLANTS**

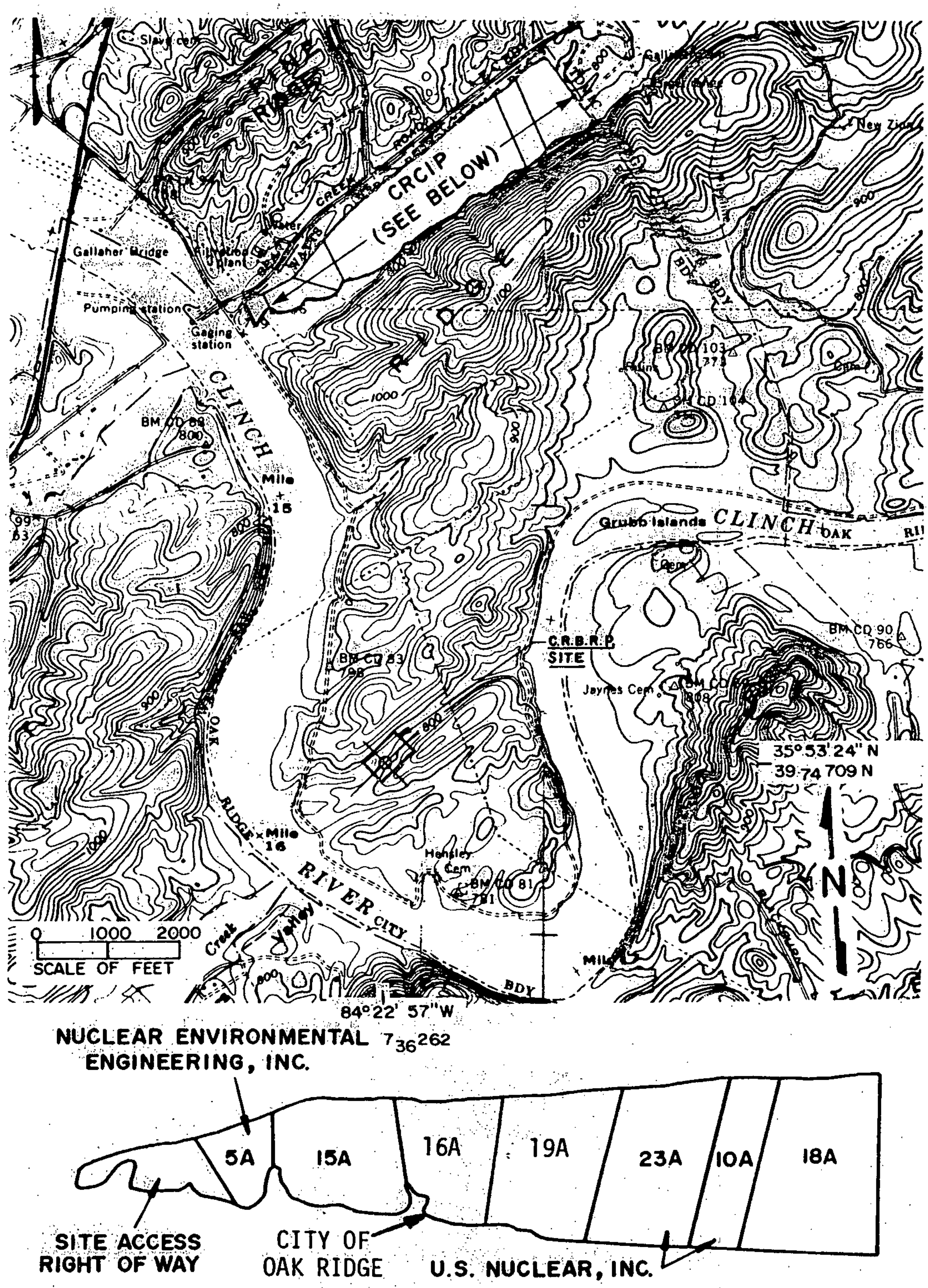
**See Figure 2.2-2 for the Clinch River Consolidated Industrial Park**

▲ Proposed Exxon Nuclear Facility

28

Figure 2.2-1. Industrial Plants within 10-Mile Radius of Clinch River Site

6648-1




28

Amend. 28
Oct. 1976

Figure 2.2-2 Clinch River Consolidated Park

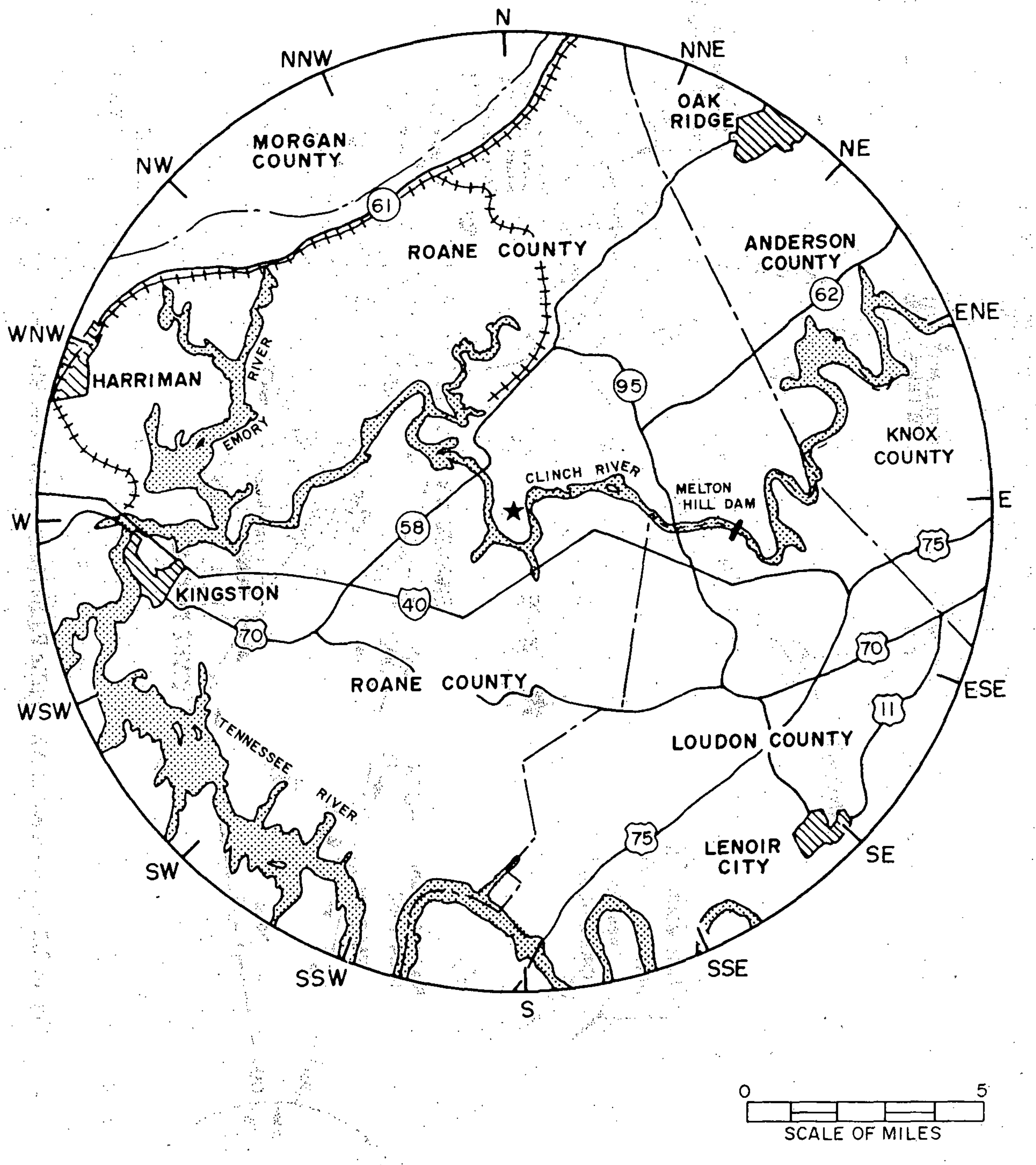



Figure 2.2-3. Clinch River Site Major Highways

6648-3

6648-4



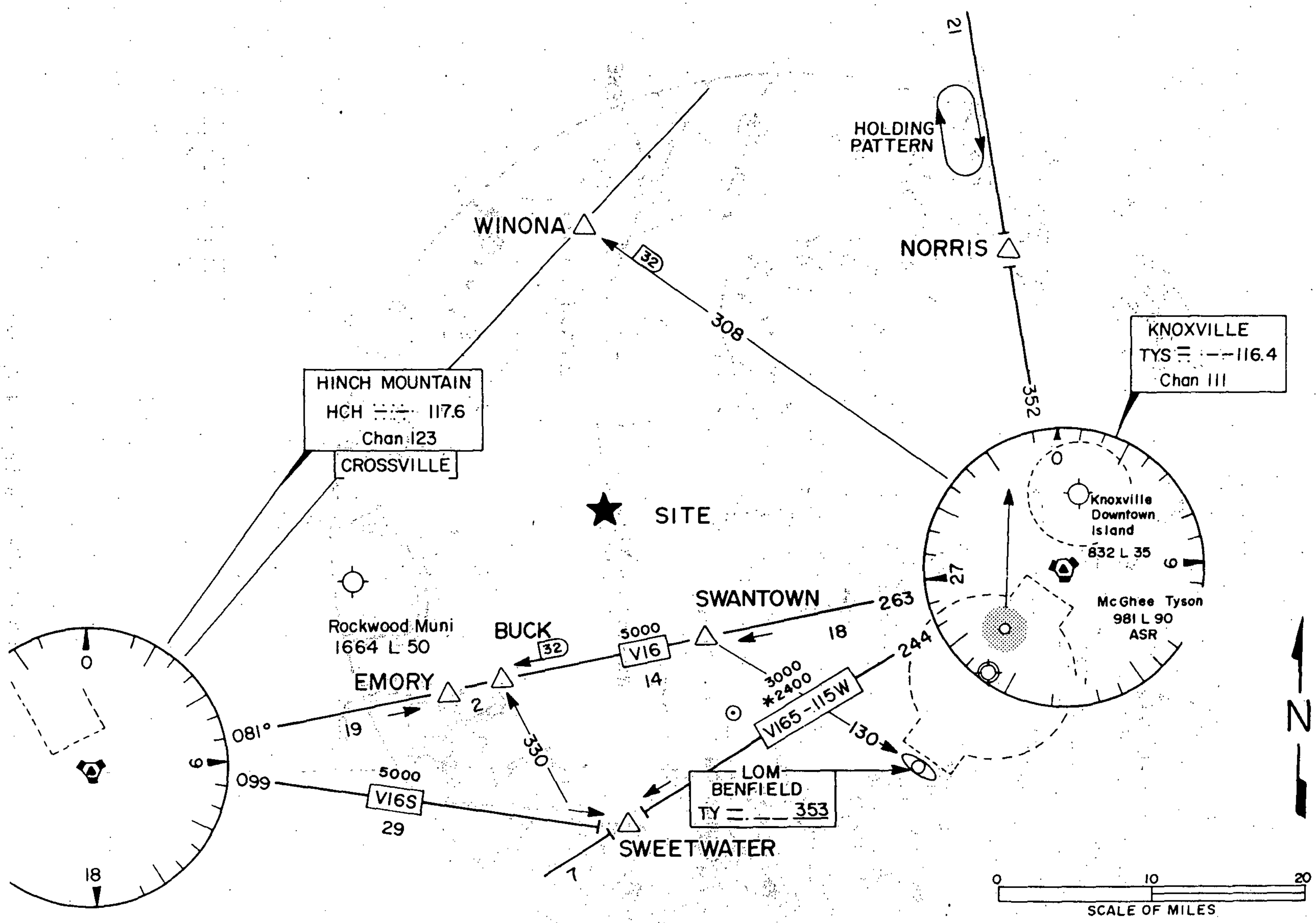



Figure 2.2-4. Commercial Air Traffic Paths Near Clinch River Site

## 2.3 METEOROLOGY

### 2.3.1 Regional Climatology

Meterological data from the Oak Ridge Area Station X-10, (Reference 1 and 2), located 4.5 miles northeast of the Clinch River Breeder Reactor Plant (CRBRP) Site, were used to characterize the Meteorology/Climatology of the region including the Site. Oak Ridge Area Station X-10 was a first order Weather Bureau Station from 1944 - 1964. From 1964 to 1972, only wind, temperature, dewpoint and differential temperature were recorded. The station was discontinued in December 1972. Other climatological data sources used in characterizing the regional climatology were the Knoxville Airport Weather Station (Reference 3), located about 20 miles east of the Site, and the Weather Bureau's Oak Ridge City Office (Reference 4), located 10 miles northeast of the Site. Locations of these weather stations are shown in Figure 2.3-1. General information on the climate of the State is available from the U. S. Weather Bureau (Reference 5). Other sources of specialized data are referenced as they appear in this section.

#### 2.3.1.1 General Climate

The Site is located in Roane County, Tennessee in a broken valley between the Cumberland Mountains to the northwest and the Great Smoky Mountains to the southeast. The topography of the Site is characterized by nearly parallel, northeast-southwest oriented ridges as discussed in Section 2.5. Elevations of the ridge crests range between 900 and 1,200 feet. The Site elevation is approximately 860 feet.

Wind directions in the region often reflect the orientation of valleys and ridges of the southern Appalachians. Mean annual wind speeds are low compared to other areas of Tennessee and the United States (Reference 6). The mean wind speed at the Oak Ridge City Office during the 16-year period of record, was 4.4 miles per hour (Reference 4).

The region has a mild, humid climate, with no distinct dry season. March is normally the wettest month and October the driest. Precipitation is heaviest from December through March when cyclonic activity is high and in July and August when convective showers occur. Maximum recorded rainfall in a 24-hour period was 7.75 inches; this occurred at Oak Ridge Station X-10 in September 1944 (Reference 1). Temperatures above 90 degrees F occurred about 30 days (Reference 4) per year. Zero and sub-zero temperatures (degree F) at the X-10 Station were observed during the months of December, January or February in fewer than half the years from 1945 through 1964.

Synoptic (regional) scale weather systems move through eastern Tennessee with irregularity. These storm systems are most frequent during December and January and cause a maximum monthly number of cloudy days and extensive precipitation. Summer season storm systems are usually weaker and tend to pass to the north, leaving eastern Tennessee with sunshine interspersed with thunderstorm activity.

Usually, 50 to 60 thunderstorm days occur per year, with a peak number of storms occurring in July (Reference 5). About nine thunderstorms per month occur during the period of May through August. The region, including the Site, is subject to only a very small probability of tornado occurrence.

Relative humidity averages lowest in the afternoons and highest at night. Average annual relative humidity in Tennessee is near 70 percent (Reference 6). This is about average for most of the United States east of the 95th meridian.

#### 2.3.1.2 Regional Meteorological Conditions for Design and Operating Bases

##### 2.3.1.2.1 Maximum Rainfall

Maximum recorded point rainfall for the Knoxville Airport, for intervals of 5 minutes to 24 hours, is listed in Table 2.3-1 (Reference 8). Maximum monthly and annual precipitation is also given. Maximum monthly and annual precipitation recorded at the Oak Ridge City Office were 19.27 inches in July 1967 and 76.33 inches in 1973, respectively (Reference 4). Monthly and annual extremes of 14.11 inches in July 1967 and 66.20 inches in 1950, respectively, were recorded at Oak Ridge Area Station X-10 (Reference 2). Maximum measured annual rainfall at Knoxville was 61.49 inches in 1957 (Reference 3). Calculated rainfall for the Site area for time periods of 0.5, 1, 2, 3, 6, 12 and 24 hours, for a recurrence interval of 100 years, is given in Table 2.3-2 (Reference 9).

##### 2.3.1.2.2 Severe Snow and Glaze Storms

Winter storms which produce a snowfall in excess of one inch are uncommon in eastern Tennessee. The area can expect about three significant snowfalls per year (one or more inches) (Reference 5). It is unusual to have snow cover for more than one week at a time (Reference 3). Records over a period of 26 years show that, in March 1960, a maximum of 21 inches accumulated, with 12 inches in a single day. Normal snowfall for March is 1.4 inches (Reference 4). The highest average monthly total was 3.1 inches, occurring in January.

Glaze occurred from three to six times per year during a 28-year survey period ending in 1953 (Reference 10). December through early March is the period with the highest frequency of glaze storms. Occurrences of glaze storms, applicable to the area including the Site, are as follows (Reference 10).

| | |
|---|---|
| Thickness of 0.25 inch or greater | Once every two years |
| Thickness of 0.50 inch or greater | Once every five years |
| Thickness of 0.75 inch or greater | Once every ten years |

##### 2.3.1.2.3 Thunderstorms and Hail

Thunderstorms occured on an average of 53 days per year (Reference 4). The month of July usually had the most. An average of about nine thunderstorms days per month occurred throughout the season from May through August. As can be seen in Table 2.3-3, the months of October through January had the fewest thunderstorms.

Hail is not frequent but it does occur. On an index of potential hail damage to residential property, (calculated for each area formed by one degree of latitude and one degree of longtitude,) the Site is in a region of low potential loss due to hail (Reference 11). Maximum values of the index occur in northwest Kansas where the index is 50. The index in eastern Tennessee is about 5. Therefore, on a geographical basis, the Site is situated in a region where hail is not a significant factor.

2.3.1.2.4 Tornados

The Site is located in an area infrequently affected by tornadoes (Reference 12 and 13). For the purpose of comparison, Tennessee ranked 25th among all states in the number of tornadoes from 1955 to 1967 (Reference 14). Dividing along the 86th Meridian, the western half of the state has reported observing three times as many tornadoes as were observed in the eastern half, which includes the Site (Reference 13). The Oak Ridge-Clinch River area has one of the lowest probabilities of tornado occurrence in the entire State (References 14 and 15).

Tornado frequencies calculated by Thom (Reference 16) for each one-degree square of latitude and longitude, for the period 1953 to 1962, show the Site to be situated in a one-degree square with an annual frequency of 0.5. According to Thom's methodology, the probability that a tornado will strike any point in a particular one-degree square, such as the Site, is calculated to be $3.63 \times 10^{-4}$ per year. The recurrence interval is one divided by the probability, which is once in 2760 years. Raw count data on tornado occurrences for those counties near the Site, for the 57-year period from 1916 to 1972, are presented in Figure 2.3-2 (References 12 and 17). Roane County is only one of several counties which are represented by the one-degree square used for the calculation of the tornado probability. Roane County itself has not recorded a tornado in the 57-year period from 1916 to 1972.

2.3.1.2.5 Strong Winds and Hurricanes

The following table is based on Thom's report (Reference 18) on his analysis of data on fastest-mile wind speeds at 30 feet above ground level, for indicated recurrence intervals, for eastern Tennessee.

CALCULATED FASTEST MILE VS. RECURRENCE INTERVAL (Reference 18)

EASTERN TENNESSEE

| Recurrence Interval (years) | Fastest Mile (mph) |
|---|---|
| 10 | 64 |
| 25 | 73 |
| 50 | 76 |
| 100 | 89 |

The peak gust recorded at the Oak Ridge City Office during a 17-year period was about 59 miles per hour (Reference 4). The fastest mile reported for the Knoxville Airport for a 31-year period was 73 miles per hour (Reference 3). A 33-year record at Chattanooga, Tennessee, shows a fastest-mile of 82 miles per hour (Reference 19).

Hurricanes are in the post hurricane stage with diminished winds by the time they reach the Site area. In the past 70 years, the remnants of nine hurricanes, classified as devastating when crossing the coastline of the U.S., have crossed Tennessee (Reference 20). Flooding in association with the remnants of a hurricane has occurred. (Reference 20). Visible damage associated with tropical storms has been reported about once in 25 years in eastern Tennessee (Reference 5).

2.3.1.2.6 High Air Pollution Potential

According to a study by Holzworth (Reference 21), high air pollution potential can be expected to occur about 5 to 10 days annually. Holzworth's results are based upon the frequency of occurrence of calculated mixing heights combined with concurrent calculated wind speeds. However, since CRBRP releases can be expected to be ground level, the frequency of stable atmospheric conditions in combination with low wind speeds should be more reflective of dispersion conditions at or near the Site. For the year of record from the onsite CRBRP permanent tower, stable conditions (Pasquill stability classes E, F, and G) with wind speeds of about 5 miles per hour or less were reported for about half of the hours.

2.3.2 Local Meteorology

The CRBRP Site meteorological facilities, the Oak Ridge Area Station X-10, the Oak Ridge City Office and the Knoxville Airport (the latter three being the closest NOAA weather stations to the Site) have been used as the primary sources of local meteorological data (References 1, 2, 3, and 4), with a few exceptions noted in the following discussion. Climatological statistics for these stations are believed to be representative of the Site area. Supplementary climatological data were obtained from TVA on relative humidities and fog frequencies (Reference 22).

2.3.2.1 Normal and Extreme Values of Meteorological Parameters

2.3.2.1.1 Temperature

Monthly and annual climatological temperature data for Area Station X-10 and the annual mean temperature data and extremes of temperature for the Oak Ridge City Office and Knoxville vicinity, for comparison purposes, are presented in Table 2.3-4. It is apparent by inspection of these data that the three sites are quite similar with respect to temperature except for the extreme low of -16 degrees F recorded in the Knoxville vicinity. This record low is a part of a much longer observation period, spanning 100 years, in which to observe extremes.

Based on these data one would expect local temperatures to range between about -15 degrees Fahrenheit and 105 degrees Fahrenheit. Temperatures above 90 degrees Fahrenheit should be moderately common from June to early September. Freezing temperatures should be common from December to February, and have occurred in all months from October through May.

2.3.2.1.2 Winds

Data from the permanent meteorological tower have been used to characterize wind conditions at the Site. The period of record is February 17, 1977 through February 16, 1978. These data have been used to construct the joint frequency distributions of wind speed and wind direction by stability classes, presented in Tables 2.3-5 through 2.3-20.

From an examination of available data collected at or near the Site, this one-year summary of on-site wind data appears reasonably representative of average conditions in an average year. The CRBRP meteorological data were used for characterizing dispersion conditions because they are site specific; the measuring heights conform to NRC Regulatory Guide 1.23; TVA has first-hand knowledge of the instrumentation and the quality control and quality assurance procedures applied toward meeting regulatory guide specifications; and the CRBRP wind instruments have lower threshold wind speeds.

An analysis of the one-year summary of on-site wind data shows an average annual wind speed of 3.5 mph at the 33-foot level and 5.6 mph at the 200-foot level. The wind was most frequent from the west-northwest at 33 feet and from the west-southwest at 200 feet. An analysis of the Oak Ridge Area Station X-10 data, where the wind sensor is mounted at a height of 102 feet, shows an average annual wind speed of 4.9 mph and a revailing wind direction of south to southwest (Reference 1). The Oak Ridge City Office shows a prevailing wind from the southwest with a mean speed of 4.4 mph (Reference 4), which is consistent with the other wind data discussed above. Knoxville Airport data show that the prevailing wind is from the northeast, and the mean hourly speed is 7.3 mph. (Reference 3). A summary of these data are provided in Table 2.3-21.

2.3.2.1.3 Humidity

Table 2.3-22 provides monthly and annual average relative humidities from measurements at the 10-meter level at CRBRP. (February 17, 1977 - February 16, 1978). Direct comparisons with respective averages for a 13-year period (1961-1973) at the Knoxville Airport (Table 2.3-23), suggests that the influence of the local CRBRP environment has resulted in higher relative humidity values. During four years (1970-1973) of hourly measurements at the Bull Run Steam plant, (at the one-meter level), relative humidities reached 100 percent about 2 percent of the time and were at least 95 percent about 7 percent of the time; the temperature was less than 50$^{o}$F concurrently with a relative humidity of at least 90 percent, about 4 percent of the time. (Saturation specific humidity at 50$^{o}$F and 1,000 millibars is about 8 grams per kilogram). The Bull Run Steam plant, similar to the CRBRP site, is located along the Clinch River, about 15 direct miles away.

2.3.2.1.4 Precipitation

Average annual precipitation was 51.52 inches at the Oak Ridge Area Station X-10 based upon 21 years of record (Reference 1). As indicated by table 2.3-24, on the average, winter was the wettest season with about 30 percent of the annual precipitation. February and March had the highest monthly average of about 5.4 inches. October averaged the driest (2.82 inches). Maximum observed monthly rainfall and 24-hour precipitation (12.84 and 7.75 inches, respectively), both occurred in September. Monthly onsite precipitation of the period February 17, 1977 through February 16, 1978 is presented in Table 2.3-25.

Snow and ice pellet data for the Oak Ridge City Office are summarized in Table 2.3-26 (Reference 4). Data listed in the table show that the annual snowfall averaged about 10 inches. Maximum annual snowfall in the 26 year period was 41.4 inches, more than four times the annual mean. Snowfalls of more than six inches in 24 hours, were reported at least one for each month from November through March (Reference 4).

2.3.2.1.5 Fog

The incidence of heavy fog (1/4 mile or less visibility) varies greatly around Tennessee (Reference 5). Typical annual values include 31 days at Knoxville (Reference 3), 34 days at Oak Ridge City Office (Reference 4), and 36 days at Chattanooga (Reference 19). Five months of the year had an average fog frequency of three days or more at each of the three stations. At both the Oak Ridge City Office and Knoxville, October had the highest fog incidence, with an average of 8 and 5 occurrences, respectively (Table 2.3-27).

Supplementary fog data (Table 2.3-28) were recorded at two sites along Melton Hill Lake, upstream from the CRBRP site, for the period January 1964 to October 1970. The sites are at Bull Run Creek (about 15 miles northeast of the CRBRP Site) and Melton Hill Dam (about 4.5 miles east of the CRBRP site).

These data show that fogs which restrict visibility to 1,100 yards or less were observed, on the average, 91 days per year at the Bull Run Creek site and 119 days per year at the Melton Hill Dam site. Fog which restricted visibility to less than 550 yards was recorded at the Melton Hill Dam site on an average of 106 days per year (Reference 22). This value is about three times the average for the Oak Ridge City office and Knoxville.

Although these frequencies are not completely comparable, because of differences in visibility limits, it seems probable that a significant portion of the greater reporting frequency at the Melton Hill Sites is a reflection of the proximity of the river.

2.3.2.1.6 Wind and Stability Data

The source of the information for developing a diffusion climatology to represent the Site is a one-year record of wind and temperature difference measurements made on the 370-foot permanent tower at the CRBRP Site. The year of record was from February 17, 1977 through February 16, 1978. The joint recovery rate for wind and stability class was 97 percent (33-foot to 200-foot temperature difference and the winds at both the 33-foot and 200-foot levels).

Pasquill Stability Classes were assigned in accordance with the temperature gradient scheme of NRC Regulatory Guide 1.23. Monthly and annual frequencies of stability classes A-G are shown in Table 2.3-29.

On an annual basis, D was the most frequent single class (36 percent), but the stable classes (E, F, and G) dominated (56 percent). This left only about 8 percent in the unstable categories (A, B, and C). The frequencies of unstable classes for both June and July were greater than 14 percent, but during the minimum month (September) unstable classes were reported for less than 3 percent of the hours. Stable classes were most frequent in September and October (about 64 percent for each month), and the least frequent in January and February (43 and 48 percent, respectively). Almost half of January and February stabilities were neutral D. There was a notable change in the frequencies of neutral and stable classes between December and January and between February and March.

Annual wind records are summarized in Tables 2.3-5 through 2.3-12 for the 33-foot level above ground and in Tables 2.3-13 through 2.3-20 for the 200-foot level above ground. These tables present the joint percentage frequency distribution of wind speed and direction for the seven Pasquill Stability Classes, A through G, and for all stability classes combined.

Annual and seasonal wind rises are shown for the 33-foot level in Figures 2.3-3 through 2.3-7 and for the 200-foot level in Figures 2.3-8 through 2.3-12.

The 33-foot winds for the annual period were within one $22.5^{o}$-sector of the west sector about 29 percent of the time; one sector of the west-southwest sector about 26 percent of the time; and one sector of the west-northwest sector about 35 percent of the time. There was considerable seasonal variation in wind direction frequencies.

For the 33-foot level, on an annual basis, nearly 60 percent of the wind speeds were in the 0.8-3.0 knots (1.5 m/sec) category. More than 80 percent of the wind speeds were less than 5 knots (2-1/2 m/sec). Wind speeds were less than 0.8 knot (0.4 m/sec) (calm) about 3 percent of the hours.

For the 200-foot level, about 80 percent of the wind speeds were less than 7 knots (3-1/2 m/s) and about 90 percent were less than 10 knots (5 m/sec). Less than one percent were less than 0.8 knot (0.4 m/sec).

#### 2.3.2.2 Potential Influence of the Plant and Its Facilities on Local Meteorology

From the safety viewpoint, the impacts of the plant complex (including the cooling towers) and its operation are expected to be insignificant (See CRBRP Final Environmental Statement, Section 5.3.3). It is expected that there will not be measurable deviations in either the extremes or means of relevant meteorological conditions.

Figure 2.3-13 and 2.3-14 are topographic maps showing the area surrounding the Site. The Site is on a peninsula-like body of land, which is bordered on three sites by the Clinch River. The four bends in the river which define this land shape extend approximately between river miles 15 and 18. This region is characterized by a series of nearly parallel ridges oriented approximately along a northeast-southwest axis. The terrain is further complicated by the generally east-southeast to west-northwest orientation of the river valley as it cuts through the ridges for about 8 direct miles. The Site is located approximately midway along this stretch of the river. Normal reservoir pool elevation is about 740 feet MSL. Mean elevation of the Site is 862 feet MSL.

Topographic profile cross sections in eight compass directions radiating from the Site are shown in Figure 2.3-15. A topographical profile cross section indicating the meteorological tower location, sensor heights, and the center of the containment building, with respect to the current topography is given in Figure 2.3-16. The terrain about 3700 feet south of the Site rises abruptly to a height of about 240 feet above the plant grade elevation of 815 feet. Hills or ridges of similar heights are found within two miles of the Site in each of these eight directions except northeast, southwest, and northwest.

The highest point within a radium of five miles of the Site is Melton Hill, elevation 1,356 feet MSL, about 4.75 miles east-northeast of the plant. Lowest points within a radium of five miles of the Site are along the margins of Watts Bar Lake, the surface of which averages about 740 feet MSL.

It is anticipated that the irregular terrain will have an effect on dispersion rates. In stable air, with very light wind pockets of temporary stagnation may develop, which could cause short-term increases in pollutant concentration levels. However, it has been shown that the wind meander, which occurs with light winds, causes ambient pollution concentrations to be much less than the calculated values (Reference 23). As wind speeds increase, turbulence should increase accordingly, with accompanying diminution of ambient pollution levels.

Modifications of the air mass due to travel over water is not considered to be significant because of the limited over-water fetch.

Although it is felt that the mechanical turbulence caused by the terrain irregularities will tend to reduce ambient pollution levels, a more detailed examination of terrain effects is planned.

2.3.2.3 Local Meteorological Conditions For Design and Operating Bases

All plant structures are designed according to the Standard Building Code and ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures." The application of the standards to the CRBRP leads to a (fastest mile) wind speed of 90 miles per hour. However, appropriate structures have been designed to meet the requirements of the design basis tornado (Section 3.3). Roofs are designed to withstand the load caused by one in 100 year rain storm (3.5 inches per hour) (Reference 9). In addition, the roofs of safety-related structures are designed for the Probable Maximum Precipitation (PMP), as described in Section 2.4.2.3.

The temperature extremes for the HVAC system's design vary with their safety class. Safety-related HVAC systems are designed to maintain the building atmosphere temperature within the limits identified in Section 9.6 up to a maximum outside temperature of 105$^{\circ}$F and a minimum outside temperature of -16$^{\circ}$F (see Section 2.3.2.1.1). Non-safety-related systems are designed to maintain building atmosphere temperatures based on outside temperature extremes of 95$^{\circ}$F maximum and 9$^{\circ}$F minimum, which were obtained from Oak Ridge X-10 data from 1966 to 1972.

Design basis meteorological conditions (wind and ice loadings) for the offsite transmission lines are discussed in Section 8.2.1.3).

The main cooling tower has a design dry bulb temperature of 91$^{\circ}$F and wet bulb temperature of 76$^{\circ}$F. The emergency cooling tower design is based on the worst 1 day and worst 30 day periods of record in accordance with Regulatory Guide 1.27 as discussed in response to Question 020.19.

2.3.3 On-Site Meteorological Monitoring Program

See Section 6.1.3.1 of the Environmental Report

2.3.4 Short-Term (Accident) Diffusion Estimates

2.3.4.1 Objectives

The objectives was to provide estimates of $\chi/Q$ values for use in potential accident consequence assessment. Atmospheric dilution factors ($\chi/Q$'s) for accident releases were estimated, using hourly data from the CRBRP 370-foot permanent meteorological tower for the period of February 17, 1977 through February 16, 1978. Releases were assumed to be ground level. Consequently, the Pasquill stability classes were determined by temperature differences between 33 and 200 feet. Wind speed and direction data were taken from the 33-foot level. Joint data recover was 97 percent.

Two different calculational methodologies were used. One followed the guidance given in R. G. 1.145 to determine design basis accident $\chi$/Q's (Reference 24). These $\chi$/Q values are expressed in two ways: (1) the probability of being exceeded 0.5 percent of the time for each of the customary sixteen, 22.5 degree, wind direction sectors, and (2) the probability of being exceeded 5 percent of the time on an overall site basis.

The other methodology followed the guidance given in R. G. 1.70 (which refers to R. G. 1.4) to compute $\chi$/Q values which have the probability of being exceeded 50 percent of the time (References 25 and 26). The $\chi$/Q values reported are for the averaging times and distances specified in P.G. 1.145 and 1.70, as appropriate.

2.3.4.2 Calculations

2.3.4.2.1 Design Basis Accident $\chi$/Q's

Two-hourly $\chi$/Q values were computed for the exlusion area boundary (EAB) and the outer boundary of the low population zone (LPZ). (Hourly average meteorological data were used, and were assumed to apply for the two hour period). A mean wind speed of 0.74 mph was used for all hours for which less than 0.74 mph were recorded. These values were then divided among the wind direction sectors in accordance with the relative frequencies in the next lowest wind speed category.

For Pasquill stability classes D, E, F, and G, and a wind speed less than 13 miles per hour, $\chi$/Q values were computed for each of the following equations (R.G. 1.145):

$$\chi/Q = \frac{1}{\bar{u}\,(\pi\,\sigma_y\sigma_z + A/2)} \quad (1)$$

$$\chi/Q = \frac{1}{\bar{u}\,(3\pi\sigma_y\sigma_z)} \quad (2)$$

$$\chi/Q = \frac{1}{\bar{u}\,\pi\Sigma_y\sigma_z} \quad (3)$$

Where

$\chi$/Q is the relative centerline concentration (sec/$m^3$)

$\chi$ is the centerline ground level concentration (Ci/$m^3$)

Q is the source strength release rate (Ci/sec)

$\bar{u}$ is the hourly average 33 foot wind speed (m/sec)

$\sigma_y$ is the standard deviation of the lateral plume spread (m).

$\sigma_z$ is the standard deviation of the vertical plume spread (m).

$\Sigma_y$ is the standard deviation of the lateral plume spread, including plume meander and building wake effects (m).

A is the minimum vertical-plane cross-sectional area of the reactor building ($m^2$).

The dispersion parameters $\sigma_y$ $\sigma_z$ and $\Sigma_y$ were evaluated in accordance with R. G. 1.145. The minimum vertical plane cross-sectional area (A) of the reactor containment was assumed to be 2415 square meters. The actual minimum cross-section area is somewhat larger in a more recent design of the plant.

For each hour of data, the higher of the respective values from equation (1) and (2) was selected and compared to the result of equation (3). The lower of those latter two values was then selected and used to form the distribution of $\chi/Q$ values for the appropriate 22.5 degree sector and on an overall basis (all sectors combined).

For stabilities A, B, and C, and for all hours with wind speeds greater than 13 miles per hour, meander effects are not considered. Consequently, the larger of equation (1) and (2) was assigned to the appropriate $\chi/Q$ distribution. The $\chi/Q$ value which was exceeded 0.5 percent of the time was selected from each sector distribution. The $\chi/Q$ value which was exceeded 5 percent of the time was selected from the distribution of all $\chi/Q$ values.

The above procedure was applied at the exclusion area boundary (EAB) distance for each of the 16 sectors. The EAB distance and the 0.5 percent $\chi/Q$ value for each sector are given in Table 2.3-30. The 5 percent $\chi/Q$ value and the maximum of the sector values are also given. These two values were compared, and the higher is listed in Table 2.3-31 as the design accident $\chi/Q$ value. The exact same calculational procedure was applied for the Low Population Zone (LPZ) distance (approximately 2.5 miles from the release point) for the averaging period of 0-2 hours. $\chi/Q$ values for the intermediate averaging times, 0-8 hours, 8-24 hours, 1-4 days and 4-30 days, were determined by a logarithmic interpolation (log) between 0-2 hour values and the appropriate annual average values. The sector dependent intermedite averaging time 0.5 percent values were determined by interpolating between the 0-2 hour and the annual average values for each of the sixteen sectors. The overall site intermediate averaging time 5 percent values were determined by interpolating between the overall 5 percent 0-2 hour X/Q and the maximum sector annual average X/Q (see section 2.3.5 for the annual average calculation procedure). The 0.5 percent values for each sector, the 5 percent overall site values and the design accident X/Q value for each time period are given in Table 2.3-30.

2.3.4.2.2 <u>Fifty Percent Calculation</u>

For each wind direction sector, two-hourly X/Q values were calculating using the permanent tower hourly meteorological data and equations derived from R.G. 1.4. The dispersion parameters $\sigma_y$ and $\sigma_z$ were evaluated in accordance with the Pasquill-Gifford curves (Reference 26) except that for stability class G, $\sigma_y$ and $\sigma_z$ were obtained from AEC Licensing Staff, Site Analysis Branch, Directorate of Licensing (Reference 27). For the 0-2 hour time interval, the program ranked, in descending order, all X/Q values for each wind direction sector. A long-probability plot of the resulting order list of X/Q values was prepared for each sector. The 50 percent value was selected from these plots.

For the EAB and LPZ, the 50 percentile X/Q values for each averaging time, given in Table 2.3-31, are the highest of the 16 values (one for each wind direction sector). The highest X/Q values occurred in the west-northwest to northwest sectors. Consequently, the X/Q values in Table 2.3-31 occurred with either east-southeast or southeast wind directions, whichever provided the maximum X/Q values.

For time intervals of 0-8 hours, 8-24 hours, 1-4 days and 4-30 days, the technique of overlapping moving averages was used to compute X/Q values. the resulting averages were ordered and plotted for each wind direction. For example, the 8-hour X/Q values were determined after all hourly X/Q values were calculated. Those X/Q values in the first 8 hours corresponding to the north wind direction were summed and divided by eight. This procedure treated all X/Q values not associated with that wind direction as zero. Average X/Q values were determined for each 8-hour period during the year, these averages were ranked in descending order, and the procedure at this point became identical to the 0-2 hour case.

These steps were then repeated for each of the remaining 15 wind direction sectors. Time intervals of 8-24 hours, 1-4 days and 4-30 days were treated in the same manner, except that averaging times of 16 hours, 72 hours and 624 hours, respectively, were used.

Calculation of hourly atmospheric dilution factors was based on Gaussian diffusion equations derived from R.G. 1.4 for centerline ground level concentrations from a continuously emitting ground-level, point source:

$$\chi/Q = \frac{1}{3\pi \bar{u}\, \sigma_y\, \sigma_z} \qquad (1)$$

and:

$$\chi/Q = \frac{1}{\bar{u}\,(\pi\, \sigma_y \sigma_z + A/2)} \qquad (2)$$

All parameters are as defined previously. Both equations include building wake corrections as specified in R.G. 1.4.

For all stabilities and wind speeds, the computer program calculated X/Q values from equations (1) and (2) and picked the larger of the two values.

The equation for calculating hourly atmospheric dilution factors for postulated release times greater than 8 hours is given in R.G. 1.4 and here includes a terrain correction factor:

$$\frac{\chi}{Q} = \frac{2.03\ T}{\sigma_z\ u\ x} \qquad (3)$$

Where

x is the distance downwind,

T is an open terrain correction factor (1 to 4) depending on distance downwind, as specified in Figure 2, R.G. 1.111, Revision 0.

This equation assumes that the plume meanders uniformly over a 22.5 degree sector.

For all downwind distances, stabilities, and wind speeds, the effects of the turbulent wake were taken into account by adding to the dispersion parameter a wake effect based on the maximum allowed under NRC Regulatory Guide 1.4 or the height of the building as suggested by Sagendorf (Reference 28). In practice, Sagendorf increases $\sigma_z$ by the square root of three or substitutes $(\sigma_z^2 + \frac{CD}{\pi})^{\frac{1}{2}}$ in Equation (3). In this case, C is the wake factor equal to 0.5 and D the building height, taken as 51.5 meters. Equation (3) is evaluated for both changes in $\sigma z$, the results are compared, and the larger values used.

The open terrain correction factor (T) was used to simulate the differences between Equation (3), a constant mean wind direction X/Q equation, and a time dependent mean wind direction X/Q equation. This correction factor was taken from R.G. 1.111, Rev. 0, 1976. Because of the additional dilution expected from the increased mechanical turbulence from the complex terrain, it is believed that use of the open terrain correction factor will generally result in overestimation of concentrations. This is supported by Van der Hoven's evaluations of the results of field studies, including one conducted at the Site in the mid 1970's (Reference 23).

### 2.3.5 Long Term Average Diffusion Estimates

Hourly average X/Q values were calculated using Equation (3), with the building wake factor, for the same year of record, for downwind distances up to 50 miles, using the 33-foot level wind data (wind speed and wind direction) and the 33-to 200-foot stability data. All X/Q values corresponding to a given wind direction sector for the entire year were summed and divided by the total number of X/Q values for all wind direction sectors.

This procedure was applied to all 16 wind direction sectors, yielding an annual average $\chi$/Q value for each sector and given downwind distance. Results are listed in Table 2.3-32.

Least dilution is found in the sector to the northwest of the plant which is consistent with the relatively high percentage of type F and G stability conditions associated with light winds that blow from the southeast.

1. U.S., Department of Commerce, NOAA, Local Climatological Data with Comparative Data, 1964, Oak Ridge, Tennessee, Area Station (X-10), National Weather Records Center, Asheville, North Carolina.

2. U.S., Department of Commerce, NOAA, Daily, Monthly and Annual Climatological Data for Oak Ridge, Tennessee, Townsite and Area Stations - January 1951 through December 1971, ATDL, Oak Ridge, Tennessee, July 1972.

3. U.S., Department of Commerce, NOAA, Local Climatological Data With Comparative Data through October 1974, Knoxville, Airport, National Weather Records Center, Asheville, North Carolina.

4. U.S., Department of Commerce, Weather Service Office, Local Climatological Data with Comparative Data through October 1974, Oak Ridge, Tennessee, NOAA Environmental Research Laboratories, ATDL, Oak Ridge, Tennessee.

5. U.S., Department of Commerce, Weather Bureau, Climates of the States, Tennessee, Climatography of the United States No. 60-40, Washington, D. C., February 1960.

6. U.S., Department of Commerce, Climatic Atlas of the United States, ESSA, Environmental Data Service, Washington, D. C., June 1968.

7. Janick, J., Schery, R. W. Woods, F. W. and Ruttan, V. W., Plant Science an Introduction to World Crops, W. H. Freeman and Company, San Francisco, 1969.

8. U.S., Department of Commerce, Weather Bureau, Maximum Recorded United States Point Rainfall for 5 minutes to 24 hours at 296 First Order Stations, Technical Paper No. 2, Revised 1963, p 28.

9. U.S., Department of Commerce, Weather Bureau, Rainfall Frequency Atlas of the United States, Technical Paper No. 40, 1963.

10. U.S. Army, Quartermaster Research and Engineering Command, Glaze - Its Meteorology and Climatology, Geographical Distribution, and Economic Effects, Technical Report - EP-105, Natick, Massachusetts, March 1959, pp 60, 62 and 63.

11. Changnon, S. A., Jr., Examples of Economic Losses from Hail in the United States, Journal of Applied Meteorology, Vol. 11, No. 7, 1972, pp 1128-37.

12. Vaiksnoras, J.V., Tornado Occurrences in Tennessee, 1916-1970, National Weather Service Office, Nashville, Tennessee, 15 April 1971, memgraphed data.

13. Vaiksnoras, J.V., Tornadoes in Tennessee (1916-1970) With Reference to Notable Tornado Disasters in the United States (1880-1970), University of Tennessee, Institute for Public Service, Knoxville, Tennessee, revised October 1972.

14. Pautz, M. E., Severe Local Storm Occurrences, 1955-1967, ESSA Technical Memorandum WBTM-FCST 12, ESSA, Silver Springs, Md., 1969.

15. Court, A., Tornado Incidence Maps, ESSA Technical Memorandum ERLTM-NSSL 49, National Severe Storms Laboratory Norman, Oklahoma, 1970.

16. Thom, H. C. S., Tornado Probabilities, Monthly Weather Review, Vol 91, No. 10-12, October-December 1963, pp 730-736.

17. Vaiksnoras, J. V., Tennessee Tornadoes - 1971 and 1972, Data Reproduced in Storm Data, U.S., Department of Commerce, National Oceanic an Atmospheric Administration, Environmental Data Service.

18. Thom, H.C.S., New Distributions of Extreme Winds in the United States, Journal of the Structural Division, Proceedings of the American Society of Civil Engineers, July 1968, pp 1787-1799.

19. U.S., Department of Commerce, NOAA, Local Climatological Data With Comparative Data, 1973, Chattanooga, Tennessee, National Weather Record Center, Asheville, North Carolina.

20. U.S., Department of Commerce, NOAA, Some Devastating North Atlantic Hurricanes of the 20th Century, NOAA/PA 70024, 1971.

21. Holzworth, G. C., Mixing Heights, Wind Speed and Potential for Urban Air Pollution Throughout the Contiguous United States, Environmental Protection Agency, January 1972.

22. Tennessee Valley Authority, Hydraulic Data Branch, Fog Occurrence and Fog Station Description at Melton Hill Reservoir at the Dam, Melton Hill Reservoir - Clinch River at Bull Run Creek and Melton Hill Reservoir at Clinton, Tennessee, Jan. 1964-Oct. 1970, memographed data.

23. Van der Hoven, Isaac, "A Survey of Field Measurements of Atmospheric Diffusion Under Low-Wind Speed Inversion Conditions," Nuclear Safety, Vol 17, No. 2, March-April 1976, pp. 223-230.

24. U.S. Nuclear Regulatory Commission, Regulatory Guide, 1.145, Atmospheric Dispersion Models For Potential Consequence Assessments at Nuclear Power Plants, Proposed, August 1979.

25. U.S. Nuclear Regulatory Commission, REgulatory Quide 1.70, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 3, November 1978.

26. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.4, Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Pressurized Water Reactors, June 1974.

27. Letter, Kornasiewicz, R., Meteorologist, USAEC, to Van Vleck, L. D., WESD, September 1973.

28. Sagendorf, J. F., A Program for Evaluating Atomspheric Dispersion from a Nuclear Power Station, NOAA Technical Memorandum GRL-ARL-42, May, 1974.

TABLE 2.3-1

MAXIMUM RECORDED POINT RAINFALL [8]

KNOXVILLE, TENNESSEE AIRPORT
(1899-1961)

Rainfall in Indicated Periods (inches)

| Minutes | | | | | Hours | | | | |
|---|---|---|---|---|---|---|---|---|---|
| 5 | 10 | 15 | 30 | 60 | 2 | 3 | 6 | 12 | 24 |
| 0.58 | 0.99 | 1.37 | 2.57 | 3.52 | 3.57 | 3.97 | 4.88 | 5.60 | 6.20 |

Maximum monthly: 11.74

Maximum annual: 61.49

(8) - Reference 8

TABLE 2.3-2

CALCULATED MAXIMUM RAINFALL FOR VARIOUS TIME PERIODS [9]
RECURRENCE INTERVAL 100 YEARS
CRBRP SITE AREA

| Time Period (hours) | Rainfall (inches) |
|---|---|
| 0.5 | 2.50 |
| 1.0 | 3.00 |
| 2.0 | 3.75 |
| 3.0 | 4.00 |
| 6.0 | 4.80 |
| 12.0 | 5.80 |
| 24.0 | 6.50 |

(9) - Reference 9



Amend. 65
Feb. 1982

TABLE 2.3-3

MEAN NUMBER OF DAYS WITH SNOW AND/OR ICE WITH THUNDERSTORMS
OAK RIDGE CITY OFFICE

| | Jan. | Feb. | Mar. | Apr. | May | June | July | Aug. | Sept. | Oct. | Nov. | Dec. |
|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Snow, Ice Pellets+ 1.0 inch or more | 1 | 1 | ** | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 0 | 1 |
| Thunderstorms++ | 1 | 2 | 3 | 5 | 8 | 9 | 11 | 9 | 3 | 1 | 1 | 1 |

*Mean number of days
**Less than one-half day
+1953-1973
++1949-1964

TABLE 2.3-4

MONTHLY TEMPERATURE SUMMARY
OAK RIDGE AREA STATION, X-10[1]
1945-1964

| Month | Climatological Standard Normals 1931-1960 Mean Monthly (°F) | Daily Maximum (°F) | Daily Minimum (°F) | Highest Temp. (°F) | Lowest Temp. (°F) |
|---|---|---|---|---|---|
| December | 40.4 | 49.4 | 31.3 | 76 | -5 |
| January | 40.1 | 48.9 | 31.2 | 77 | -8 |
| February | 41.7 | 51.6 | 31.8 | 77 | 0 |
| Winter | 40.7 | 50.0 | 31.4 | 77 | -8 |
| March | 48.0 | 58.9 | 37.0 | 87 | 4 |
| April | 58.2 | 70.0 | 46.3 | 89 | 24 |
| May | 66.9 | 79.0 | 54.8 | 94 | 32 |
| Spring | 57.7 | 69.3 | 46.0 | 94 | 4 |
| June | 74.7 | 86.1 | 63.3 | 99 | 41 |
| July | 77.4 | 88.0 | 66.7 | 103 | 49 |
| August | 76.5 | 87.4 | 65.6 | 99 | 44 |
| Summer | 76.2 | 87.2 | 65.2 | 103 | 41 |
| September | 71.1 | 83.0 | 59.2 | 103 | 33 |
| October | 60.0 | 72.2 | 47.7 | 91 | 21 |
| November | 47.6 | 58.6 | 36.5 | 83 | 4 |
| Fall | 59.6 | 71.3 | 47.6 | 103 | 4 |
| Annual | 58.5 | 69.4 | 47.6 | 103 | -8 |
| Oak Ridge City Office[4] Climatological Standard Normals 1941-1970 | | | | | |
| Annual | 57.8 | 68.6 | 47.0 | 105* | -9* |
| Knoxville Vicinity[3] Climatological Standard Normals 1941-1970 | | | | | |
| Annual | 59.7 | 69.8 | 49.5 | 104** | -16 |

*May 1947 - October 1974
** 1874 - October 1974

(4) - Reference 4
(3) - Reference 3

(1) - Reference 1



Amend. 65
Feb. 1982

TABLE 2.3-5

ANNUAL JOINT FREQUENCY OF WIND DIRECTIONS AND WIND SPEEDS FOR

STABILITY CLASS A

CRBRP PERMANENT TOWER, 33-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| NNE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| NE | .000000 | .000000 | .000000 | .000117 | .000117 | .000000 | .000000 | .000000 | 2 | .000235 | 6.8 |
| ENE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| E | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| ESE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| SE | .000000 | .000000 | .000000 | .000117 | .000000 | .000000 | .000000 | .000000 | 1 | .000117 | 6.0 |
| SSE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| S | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| SSW | .000000 | .000000 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 1 | .000117 | 4.8 |
| SW | .000000 | .000000 | .000117 | .000235 | .000586 | .000117 | .000000 | .000000 | 9 | .001055 | 7.4 |
| WSW | .000000 | .000000 | .000000 | .000235 | .000704 | .000352 | .000000 | .000000 | 11 | .001290 | 8.6 |
| W | .000000 | .000000 | .000000 | .000235 | .000469 | .000235 | .000000 | .000000 | 8 | .000938 | 8.4 |
| WNW | .000000 | .000000 | .000000 | .000000 | .002597 | .000152 | .000000 | .000000 | 26 | .003049 | 8.4 |
| NW | .000000 | .000000 | .000000 | .000000 | .001407 | .000000 | .000000 | .000000 | 12 | .001407 | 8.4 |
| NNW | .000000 | .000000 | .000000 | .000000 | .000235 | .000000 | .000000 | .000000 | 2 | .000235 | 7.9 |
| HRS | | 0 | 2 | 8 | 53 | 9 | 0 | 0 | 72 | | |
| FREQ | .000000 | .000000 | .000235 | .000938 | .006216 | .001055 | .000000 | .000000 | | .008444 | |
| AVGSPD | .0 | .0 | 4.2 | 5.7 | 7.9 | 12.7 | .0 | .0 | | | 8.2 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-6

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

STABILITY CLASS B

CRBRP PERMANENT TOWER, 33-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .000000 | .000117 | .000000 | .000117 | .000000 | .000000 | .000000 | 2 | .000235 | 5.2 |
| NNE | .000000 | .000000 | .000000 | .000235 | .000000 | .000000 | .000000 | .000000 | 2 | .000235 | 5.4 |
| NE | .000000 | .000000 | .000117 | .000117 | .000352 | .000000 | .000000 | .000000 | 5 | .000586 | 6.9 |
| ENE | .000000 | .000000 | .000352 | .000352 | .000235 | .000000 | .000000 | .000000 | 8 | .000938 | 5.8 |
| E | .000000 | .000000 | .000352 | .000117 | .000000 | .000000 | .000000 | .000000 | 4 | .000469 | 4.5 |
| ESE | .000000 | .000000 | .000117 | .000235 | .000117 | .000000 | .000000 | .000000 | 4 | .000469 | 6.1 |
| SE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| SSE | .000000 | .000000 | .000000 | .000117 | .000117 | .000000 | .000000 | .000000 | 2 | .000235 | 7.3 |
| S | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| SSW | .000000 | .000000 | .000235 | .000235 | .000000 | .000000 | .000000 | .000000 | 4 | .000469 | 4.4 |
| SW | .000000 | .000000 | .000586 | .001407 | .001055 | .000117 | .000000 | .000000 | 27 | .003166 | 6.3 |
| WSW | .000000 | .000000 | .000821 | .000938 | .000821 | .000821 | .000000 | .000000 | 29 | .003401 | 7.6 |
| W | .000000 | .000000 | .000117 | .000352 | .001055 | .000352 | .000000 | .000000 | 16 | .001876 | 8.1 |
| WNW | .000000 | .000000 | .000000 | .000469 | .002345 | .000469 | .000000 | .000000 | 28 | .003284 | 7.9 |
| NW | .000000 | .000000 | .000000 | .000469 | .002228 | .000000 | .000000 | .000000 | 23 | .002697 | 7.6 |
| NNW | .000000 | .000000 | .000000 | .000117 | .000821 | .000000 | .000000 | .000000 | 8 | .000938 | 7.3 |
| HRS | 0 | 0 | 24 | 44 | 79 | 15 | 0 | 0 | 162 | | |
| FREQ | .000000 | .000000 | .002815 | .005160 | .009265 | .001759 | .000000 | .000000 | | .018998 | |
| AVGSPD | .0 | .0 | 4.2 | 5.7 | 7.8 | 12.2 | .0 | .0 | | | 7.1 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

## TABLE 2.3-7

## ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR STABILITY CLASS C

## CRBRP PERMANENT TOWER, 33-FOOT LEVEL

## FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

Wind Speed (Knots*)

| Wind Direction | .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .000352 | .001173 | .000235 | .000352 | .000000 | .000000 | .000000 | 18 | .002111 | 4.3 |
| NNE | .000000 | .000821 | .001407 | .000938 | .000117 | .000000 | .000000 | .000000 | 28 | .003284 | 4.0 |
| NE | .000000 | .000117 | .002580 | .001642 | .000117 | .000000 | .000000 | .000000 | 38 | .004456 | 4.5 |
| ENE | .000000 | .000117 | .001994 | .000821 | .000352 | .000000 | .000000 | .000000 | 28 | .003284 | 4.6 |
| E | .000000 | .000235 | .001642 | .000821 | .000117 | .000000 | .000000 | .000000 | 24 | .002815 | 4.5 |
| ESE | .000000 | .000000 | .000704 | .000117 | .000000 | .000000 | .000000 | .000000 | 7 | .000821 | 4.2 |
| SE | .000000 | .000117 | .000235 | .000117 | .000000 | .000000 | .000000 | .000000 | 4 | .000469 | 3.9 |
| SSE | .000000 | .000000 | .001525 | .000821 | .000000 | .000235 | .000000 | .000000 | 22 | .002580 | 5.2 |
| S | .000000 | .000000 | .000117 | .000235 | .000000 | .000000 | .000000 | .000000 | 3 | .000352 | 5.5 |
| SSW | .000000 | .000000 | .001525 | .000586 | .000000 | .000000 | .000000 | .000000 | 18 | .002111 | 4.3 |
| SW | .000000 | .000235 | .004456 | .002463 | .002228 | .000117 | .000000 | .000000 | 81 | .009499 | 5.5 |
| WSW | .000000 | .000235 | .002463 | .001759 | .000704 | .000235 | .000000 | .000000 | 46 | .005395 | 5.3 |
| W | .000000 | .000000 | .000704 | .000821 | .000704 | .000352 | .000000 | .000000 | 22 | .002580 | 6.7 |
| WNW | .000000 | .000000 | .000469 | .002345 | .002463 | .000352 | .000000 | .000000 | 48 | .005629 | 6.7 |
| NW | .000000 | .000117 | .000352 | .000704 | .002463 | .000000 | .000000 | .000000 | 31 | .003636 | 6.8 |
| NNW | .000000 | .000117 | .000235 | .00469 | .00704 | .000000 | .000000 | .000000 | 13 | .001525 | 5.9 |
| HRS | 0 | 21 | 184 | 127 | 88 | 11 | 0 | 0 | 431 | | |
| FREQ | .000000 | .002463 | .021579 | .014894 | .010320 | .001290 | .000000 | .000000 | | .050545 | |
| AVGSPD | .0 | 2.5 | 4.0 | 5.5 | 7.7 | 12.1 | .0 | .0 | | | 5.3 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-8

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

STABILITY CLASS D

CRBRP PERMANENT TOWER, 33-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .008209 | .003049 | .000235 | .000000 | .000000 | .000000 | .000000 | 98 | .011493 | 2.5 |
| NNE | .000000 | .007623 | .004808 | .000704 | .000000 | .000000 | .000000 | .000000 | 112 | .013135 | 2.9 |
| NE | .000000 | .008913 | .010907 | .002815 | .000235 | .000000 | .000000 | .000000 | 195 | .022869 | 3.4 |
| ENE | .000000 | .014190 | .016301 | .004926 | .000117 | .000000 | .000000 | .000000 | 303 | .035534 | 3.4 |
| E | .000117 | .012666 | .008444 | .000821 | .000352 | .000000 | .000000 | .000000 | 191 | .022399 | 2.9 |
| ESE | .000117 | .006216 | .004691 | .000235 | .000000 | .000000 | .000000 | .000000 | 96 | .011258 | 2.8 |
| SE | .000000 | .004691 | .002815 | .00117 | .000000 | .000000 | .000000 | .000000 | 65 | .007623 | 2.7 |
| SSE | .000000 | .008209 | .011376 | .004222 | .002697 | .001173 | .000000 | .000000 | 236 | .027677 | 4.3 |
| S | .000000 | .005277 | .005864 | .000938 | .000938 | .000117 | .000000 | .000000 | 112 | .013135 | 3.8 |
| SSW | .000000 | .004808 | .004456 | .001407 | .000000 | .000000 | .000000 | .000000 | 91 | .010672 | 3.3 |
| SW | .000000 | .008913 | .016301 | .008678 | .005746 | .001407 | .000117 | .000000 | 351 | .041163 | 4.7 |
| WSW | .000000 | .011962 | .018647 | .008326 | .007154 | .002697 | .000117 | .000000 | 417 | .048903 | 4.8 |
| W | .000000 | .008092 | .008326 | .002111 | .002463 | .001173 | .000000 | .000000 | 189 | .022165 | 4.3 |
| WNW | .000000 | .007623 | .007857 | .010672 | .010555 | .002815 | .000000 | .000000 | 337 | .039522 | 5.6 |
| NW | .000000 | .004222 | .005043 | .004105 | .005864 | .001173 | .000000 | .000000 | 174 | .020406 | 5.3 |
| NNW | .000000 | .006450 | .002345 | .003049 | .000938 | .000000 | .000000 | .000000 | 109 | .012783 | 3.5 |
| HRS | 2 | 1092 | 1119 | 455 | 316 | 90 | 2 | 0 | 3076 | | |
| FREQ | .000235 | .128064 | .131230 | .053360 | .037059 | .010555 | .000235 | .000000 | | .360736 | |
| AVGSPD | .7 | 2.2 | 3.8 | 5.5 | 7.7 | 11.5 | 16.7 | .0 | | | 4.1 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-9

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

STABILITY CLASS E

CRBRP PERMANENT METEOROLOGICAL TOWER, 33-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| | Wind Speed (Knots*) | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| Wind Direction | .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
| N | .000469 | .011610 | .000352 | .000000 | .000117 | .000000 | .000000 | .000000 | 107 | .012548 | 1.5 |
| NNE | .000352 | .010437 | .000938 | .000000 | .000000 | .000000 | .000000 | .000000 | 100 | .011727 | 1.8 |
| NE | .000352 | .009147 | .002932 | .000586 | .000235 | .000000 | .000000 | .000000 | 113 | .013252 | 2.5 |
| ENE | .000352 | .008913 | .004105 | .000704 | .000000 | .000000 | .000000 | .000000 | 120 | .014073 | 2.5 |
| E | .000586 | .015715 | .001525 | .000586 | .000235 | .000000 | .000000 | .000000 | 159 | .018647 | 2.0 |
| ESE | .000000 | .007623 | .001055 | .000117 | .000469 | .000000 | .000000 | .000000 | 79 | .009265 | 2.0 |
| SE | .000235 | .012900 | .000821 | .000235 | .000000 | .000000 | .000000 | .000000 | 121 | .014190 | 1.7 |
| SSE | .000117 | .012548 | .003166 | .000938 | .001642 | .000586 | .000000 | .000000 | 162 | .018998 | 2.9 |
| S | .000235 | .007857 | .002580 | .000938 | .000235 | .000000 | .000000 | .000000 | 101 | .011845 | 2.6 |
| SSW | .000117 | .003166 | .001407 | .001407 | .000704 | .000000 | .000000 | .000000 | 58 | .006802 | 3.5 |
| SW | .000469 | .008561 | .003166 | .000938 | .001642 | .000117 | .000000 | .000000 | 127 | .014894 | 3.1 |
| WSW | .000235 | .014190 | .006685 | .002345 | .001759 | .000235 | .000000 | .000000 | 217 | .025449 | 3.2 |
| W | .000586 | .018529 | .007857 | .001759 | .000704 | .000000 | .000000 | .000000 | 251 | .029436 | 2.7 |
| WNW | .000117 | .018178 | .005160 | .002932 | .001759 | .000352 | .000000 | .000000 | 243 | .028498 | 2.9 |
| NW | .000352 | .014073 | .003166 | .003049 | .000586 | .000117 | .000000 | .000000 | 182 | .021344 | 2.6 |
| NNW | .000117 | .011376 | .001525 | .000586 | .000117 | .000000 | .000000 | .000000 | 117 | .013721 | 1.9 |
| HRS | 40 | 1576 | 396 | 146 | 87 | 12 | 0 | 0 | 2257 | | |
| FREQ | .004691 | .184825 | .046441 | .017122 | .010203 | .001407 | .000000 | .000000 | | .264689 | |
| AVGSPD | .7 | 1.6 | 3.7 | 5.5 | 7.8 | 11.4 | .0 | .0 | | | 2.5 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-10

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

STABILITY CLASS F

CRBRP PERMANENT TOWER, 33-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000352 | .004808 | .000117 | .000117 | .000000 | .000000 | .000000 | .000000 | 46 | .005395 | 1.2 |
| NNE | .000469 | .004574 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 44 | .005160 | 1.1 |
| NE | .001407 | .005395 | .000352 | .000000 | .000000 | .000000 | .000000 | .000000 | 61 | .007154 | 1.2 |
| ENE | .000469 | .009851 | .000235 | .000000 | .000000 | .000000 | .000000 | .000000 | 90 | .010555 | 1.2 |
| E | .000000 | .010672 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 91 | .010672 | 1.2 |
| ESE | .001055 | .010437 | .000235 | .000000 | .000000 | .000000 | .000000 | .000000 | 100 | .011727 | 1.1 |
| SE | .002228 | .020406 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 193 | .022634 | 1.2 |
| SSE | .000938 | .014307 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 131 | .015363 | 1.2 |
| S | .000000 | .002580 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 23 | .002697 | 1.3 |
| SSW | .000000 | .003166 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 28 | .003284 | 1.2 |
| SW | .001055 | .003284 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 37 | .004339 | 1.0 |
| WSW | .000469 | .006450 | .001173 | .000000 | .000000 | .000000 | .000000 | .000000 | 69 | .008092 | 1.5 |
| W | .000821 | .011258 | .000469 | .000000 | .000000 | .000000 | .000000 | .000000 | 107 | .012548 | 1.4 |
| WNW | .000352 | .013487 | .001173 | .000000 | .000000 | .000000 | .000000 | .000000 | 128 | .015011 | 1.4 |
| NW | 001876 | .009382 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 96 | .011258 | 1.1 |
| NNW | .000469 | .007036 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 65 | .007623 | 1.1 |
| HRS | 102 | 1169 | 37 | 1 | 0 | 0 | 0 | 0 | 1309 | | |
| FREQ | .011962 | .137094 | .004339 | .000117 | .000000 | .000000 | .000000 | .000000 | | .153512 | |
| AVGSPD | .7 | 1.2 | 3.5 | 6.3 | .0 | .0 | .0 | .0 | | | 1.2 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

# TABLE 2.3-11

## ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR STABILITY CLASS G

## CRBRP PERMANENT TOWER, 33-FOOT LEVEL

## FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .001759 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 15 | .001759 | 1.0 |
| NNE | .000586 | .003636 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 36 | .004222 | 1.0 |
| NE | .000704 | .004456 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 44 | .005160 | 1.0 |
| ENE | .001642 | .007740 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 81 | .009499 | 1.1 |
| E | .000352 | .016301 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 143 | .016770 | 1.1 |
| ESE | .001290 | .010672 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 102 | .011962 | 1.0 |
| SE | .003870 | .024745 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 244 | .028615 | 1.0 |
| SSE | .001642 | .016301 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 154 | .018060 | 1.1 |
| S | .000352 | .003636 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 34 | .003987 | 1.1 |
| SSW | .000704 | .002815 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 30 | .003518 | .9 |
| SW | .000352 | .003753 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 35 | .004105 | .9 |
| WSW | .000000 | .006098 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 53 | .006216 | 1.2 |
| W | .000352 | .010320 | .000117 | .000117 | .000000 | .000000 | .000000 | .000000 | 93 | .010907 | 1.4 |
| WNW | .002463 | .008678 | .000117 | .000000 | .000000 | .000000 | .000000 | .000000 | 96 | .011258 | 1.2 |
| NW | .000352 | .003401 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 32 | .003753 | 1.0 |
| NNW | .000352 | .002932 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 28 | .003284 | 1.0 |
| HRS | 128 | 1085 | 6 | 1 | 0 | 0 | 0 | 0 | 1220 | | |
| FREQ | .015011 | .127243 | .000704 | .000117 | .000000 | .000000 | .000000 | .000000 | | .143075 | |
| AVGSPD | .7 | 1.1 | 3.5 | 6.1 | .0 | .0 | .0 | .0 | | | 1.1 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-12

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

ALL STABILITY CLASSES

CRBRP PERMANENT TOWER, 33-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
| N | .000821 | .026739 | .004808 | .000586 | .000586 | .000000 | .000000 | .000000 | 286 | .033541 | 2.0 |
| NNE | .001407 | .027090 | .007271 | .001876 | .000117 | .000000 | .000000 | .000000 | 322 | .037762 | 2.2 |
| NE | .002463 | .028029 | .016888 | .005277 | .001055 | .000000 | .000000 | .000000 | 458 | .053712 | 2.8 |
| ENE | .002463 | .040812 | .023103 | .006802 | .000704 | .000000 | .000000 | .000000 | 630 | .073883 | 2.7 |
| E | .001055 | .055588 | .012079 | .002345 | .000704 | .000000 | .000000 | .000000 | 612 | .071772 | 2.1 |
| ESE | .002463 | .034948 | .006802 | .000704 | .000586 | .000000 | .000000 | .000000 | 388 | .045503 | 1.8 |
| SE | .006333 | .062859 | .003870 | .000586 | .000000 | .000000 | .000000 | .000000 | 628 | .073648 | 1.4 |
| SSE | .002697 | .051366 | .016301 | .006098 | .004456 | .001994 | .000000 | .000000 | 707 | .082913 | 2.7 |
| S | .000586 | .019350 | .008678 | .002111 | .001173 | .000117 | .000000 | .000000 | 273 | .032016 | 2.8 |
| SSW | .000821 | .013956 | .007857 | .003636 | .000704 | .000000 | .000000 | .000000 | 230 | .026973 | 2.9 |
| SW | .001876 | .024745 | .024628 | .013721 | .011258 | .001876 | .000117 | .000000 | 667 | .078222 | 4.2 |
| WSW | .000704 | .038935 | .029905 | .013604 | .011141 | .004339 | .000117 | .000000 | 842 | .098745 | 4.0 |
| W | .001759 | .048200 | .017591 | .005395 | .005395 | .002111 | .000000 | .000000 | 686 | .080450 | 3.1 |
| WNW | .002932 | .047965 | .014777 | .016418 | .019819 | .004339 | .000000 | .000000 | 906 | .106251 | 4.0 |
| NW | .002580 | .031195 | .008561 | .008326 | .012548 | .001290 | .000000 | .000000 | 550 | .064501 | 3.7 |
| NNW | .000938 | .027911 | .004222 | .004222 | .002815 | .000000 | .000000 | .000000 | 342 | .040108 | 2.5 |
| HRS | 272 | 4943 | 1768 | 782 | 623 | 137 | 2 | 0 | 8527 | | |
| FREQ | .031899 | .579688 | .207341 | .091709 | .073062 | .016067 | .000235 | .000000 | | 1.000000 | |
| AVGSPD | .7 | 1.5 | 3.8 | 5.5 | 7.8 | 11.7 | 16.7 | .0 | | | 3.0 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-13

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

STABILITY CLASS A

CRBRP PERMANENT TOWER, 200-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| NNE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| NE | .000000 | .000000 | .000000 | .000000 | .000118 | .000118 | .000000 | .000000 | 2 | .000235 | 10.6 |
| ENE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| E | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| ESE | .000000 | .000000 | .000000 | .000000 | .000118 | .000000 | .000000 | .000000 | 1 | .000118 | 7.4 |
| SE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| SSE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| S | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| SSW | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| SW | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| WSW | .000000 | .000000 | .000118 | .000235 | .000823 | .000706 | .000235 | .000353 | 21 | .002469 | 12.0 |
| W | .000000 | .000000 | .000000 | .000118 | .000118 | .000588 | .000470 | .000000 | 11 | .001294 | 12.9 |
| WNW | .000000 | .000000 | .000000 | .000000 | .000588 | .001999 | .000000 | .000000 | 22 | .002587 | 10.7 |
| NW | .000000 | .000000 | .000000 | .000000 | .000470 | .001294 | .000000 | .000000 | 15 | .001764 | 10.6 |
| NNW | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| HRS | 0 | 0 | 1 | 3 | 19 | 40 | 6 | 3 | 72 | | |
| FREQ | .000000 | .000000 | .000118 | .000353 | .002234 | .004704 | .000706 | .000353 | | .008467 | |
| AVGSPD | .0 | .0 | 4.7 | 6.1 | 8.7 | 11.2 | 18.7 | 23.0 | | | 11.4 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7 The .7 knots is the stall threshold speed of the wind direction sensor.

## TABLE 2.3-14

## ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR STABILITY CLASS B

## CRBRP PERMANENT TOWER, 200-FOOT LEVEL

## FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

Wind Speed (Knots*)

| Wind Direction | .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .000000 | .000000 | .000118 | .000000 | .000000 | .000000 | .000000 | 1 | .000118 | 5.9 |
| NNE | .000000 | .000000 | .000118 | .000000 | .000000 | .000000 | .000000 | .000000 | 1 | .000118 | 4.7 |
| NE | .000000 | .000000 | .000000 | .000353 | .000823 | .000470 | .000000 | .000000 | 14 | .001646 | 9.1 |
| ENE | .000000 | .000000 | .000000 | .000353 | .000235 | .000000 | .000000 | .000000 | 5 | .000588 | 6.5 |
| E | .000000 | .000000 | .000118 | .000118 | .000235 | .000000 | .000000 | .000000 | 4 | .000470 | 6.3 |
| ESE | .000000 | .000000 | .000000 | .000000 | .000118 | .000000 | .000000 | .000000 | 1 | .000118 | 9.4 |
| SE | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 0 | .000000 | .0 |
| SSE | .000000 | .000000 | .000000 | .000000 | .000118 | .000118 | .000000 | .000000 | 2 | .000235 | 9.3 |
| S | .000000 | .000000 | .000000 | .000000 | .000118 | .000000 | .000000 | .000000 | 1 | .000118 | 7.6 |
| SSW | .000000 | .000000 | .000245 | .000000 | .000000 | .000000 | .000000 | .000000 | 2 | .000235 | 3.3 |
| SW | .000000 | .000000 | .000000 | .000118 | .000706 | .000000 | .000000 | .000000 | 7 | .000823 | 7.4 |
| WSW | .000000 | .000000 | .000235 | .001411 | .001764 | .001058 | .000353 | .000470 | 45 | .005292 | 10.0 |
| W | .000000 | .000000 | .000000 | .000118 | .000823 | .001058 | .000235 | .000235 | 21 | .002469 | 12.6 |
| WNW | .000000 | .000000 | .000000 | .000000 | .000588 | .001646 | .000235 | .000000 | 21 | .002469 | 11.5 |
| NW | .000000 | .000000 | .000000 | .000118 | .002117 | .001529 | .000000 | .000000 | 32 | .003763 | 9.8 |
| NNW | .000000 | .000000 | .000000 | .000118 | .000235 | .000235 | .000000 | .000000 | 5 | .000588 | 8.6 |
| HRS | 0 | 0 | 6 | 24 | 67 | 52 | 7 | 6 | 162 | | |
| FREQ | .000000 | .000000 | .000706 | .002822 | .007879 | .006115 | .000823 | .000706 | | .019050 | |
| AVGSPD | .0 | .0 | 4.1 | 5.7 | 8.3 | 12.0 | 18.2 | 22.6 | | | 9.9 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-15

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

STABILITY CLASS C

CRBRP PERMANENT TOWER, 200-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .000235 | .000235 | .000118 | .000118 | .000000 | .000000 | .000000 | 6 | .000706 | 4.3 |
| NNE | .000000 | .000235 | .001294 | .000353 | .000235 | .000353 | .000000 | .000000 | 21 | .002469 | 5.2 |
| NE | .000000 | .000118 | .002234 | .002469 | .003293 | .000470 | .000000 | .000000 | 73 | .008584 | 6.2 |
| ENE | .000000 | .000118 | .000941 | .001294 | .000706 | .000000 | .000000 | .000000 | 26 | .003057 | 5.3 |
| E | .000000 | .000000 | .000353 | .000706 | .000470 | .000000 | .000000 | .000000 | 13 | .001529 | 5.9 |
| ESE | .000000 | .000000 | .000588 | .000353 | .000118 | .000000 | .000000 | .000000 | 9 | .001058 | 4.7 |
| SE | .000000 | .000000 | .000118 | .000118 | .000118 | .000000 | .000000 | .000000 | 3 | .000353 | 5.7 |
| SSE | .000000 | .000000 | .000118 | .000470 | .000588 | .000000 | .000118 | .000000 | 11 | .001294 | 7.5 |
| S | .000000 | .000000 | .000235 | .000353 | .000235 | .000118 | .000000 | .000000 | 8 | .000941 | 6.8 |
| SSW | .000000 | .000000 | .000235 | .000353 | .000353 | .000000 | .000000 | .000000 | 8 | .000941 | 5.8 |
| SW | .000000 | .000235 | .000941 | .001646 | .001881 | .000706 | .000118 | .000000 | 47 | .005527 | 7.1 |
| WSW | .000000 | .000000 | .002587 | .002234 | .002469 | .001529 | .000235 | .000000 | 77 | .009055 | 7.3 |
| W | .000000 | .000118 | .000706 | .001058 | .000941 | .000706 | .000118 | .000235 | 33 | .003881 | 8.5 |
| WNW | .000000 | .000000 | .000235 | .000588 | .002940 | .001646 | .000235 | .000000 | 48 | .005644 | 9.0 |
| NW | .000000 | .000000 | .000118 | .000706 | .001999 | .001294 | .000000 | .000000 | 35 | .004116 | 9.0 |
| NNW | .000000 | .000000 | .000353 | .000118 | .000823 | .000235 | .000000 | .000000 | 13 | .001529 | 7.4 |
| HRS | 0 | 9 | 96 | 110 | 147 | 60 | 7 | 2 | 431 | | |
| FREQ | .000000 | .001058 | .011289 | .012935 | .017286 | .007056 | .000823 | .000235 | | .050682 | |
| AVGSPD | .0 | 2.1 | 4.0 | 5.6 | 7.8 | 12.0 | 18.8 | 22.7 | | | 7.1 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-16

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

STABILITY CLASS D

CRBRP PERMANENT TOWER, 200-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .002469 | .001176 | .000588 | .001058 | .000000 | .000000 | .000000 | 45 | .005292 | 4.1 |
| NNE | .000000 | .004821 | .004468 | .001294 | .000941 | .000118 | .000000 | .000000 | 99 | .011642 | 3.6 |
| NE | .000000 | .008819 | .016228 | .007291 | .009290 | .001529 | .000000 | .000000 | 367 | .043156 | 5.0 |
| ENE | .000000 | .007173 | .010936 | .009525 | .007173 | .000470 | .000000 | .000000 | 300 | .035278 | 4.9 |
| E | .000000 | .008937 | .006115 | .001881 | .001058 | .000118 | .000000 | .000000 | 154 | .018109 | 3.5 |
| ESE | .000000 | .002587 | .006468 | .001411 | .000235 | .000000 | .000000 | .000000 | 91 | .010701 | 3.7 |
| SE | .000000 | .001646 | .002352 | .000353 | .000118 | .000000 | .000000 | .000000 | 38 | .004468 | 3.5 |
| SSE | .000000 | .003057 | .004821 | .002469 | .001294 | .000470 | .000353 | .000000 | 106 | .012465 | 5.1 |
| S | .000000 | .001646 | .005174 | .005056 | .005762 | .002822 | .000353 | .000235 | 179 | .021049 | 6.9 |
| SSW | .000000 | .002822 | .005409 | .004116 | .002822 | .000823 | .000000 | .000000 | 136 | .015992 | 5.1 |
| SW | .000000 | .003881 | .012582 | .010583 | .013993 | .003881 | .000470 | .000000 | 386 | .045390 | 6.4 |
| WSW | .000000 | .005762 | .015287 | .008114 | .008349 | .006115 | .001999 | .000823 | 395 | .046449 | 6.9 |
| W | .000000 | .005527 | .007879 | .002940 | .003293 | .006585 | .001058 | .000823 | 239 | .028104 | 7.3 |
| WNW | .000000 | .002822 | .003175 | .003645 | .011524 | .010230 | .001999 | .000000 | 284 | .033396 | 8.8 |
| NW | .000000 | .002940 | .004116 | .001999 | .006938 | .004468 | .000823 | .000000 | 181 | .021284 | 7.5 |
| NNW | .000000 | .002117 | .001646 | .001764 | .002705 | .000235 | .000000 | .000000 | 72 | .008467 | 5.3 |
| HRS | 0 | 570 | 917 | 536 | 651 | 322 | 60 | 16 | 3072 | | |
| FREQ | .000000 | .067027 | .107832 | .063029 | .076552 | .037865 | .007056 | .001881 | | .361242 | |
| AVGSPD | .0 | 2.2 | 3.9 | 5.6 | 8.0 | 12.3 | 17.8 | 23.4 | | | 6.0 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-17

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

STABILITY CLASS E

CRBRP PERMANENT TOWER, 200-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000000 | .004233 | .001646 | .000588 | .001294 | .000235 | .000000 | .000000 | 68 | .007996 | 3.9 |
| NNE | .000118 | .007879 | .002117 | .001294 | .000588 | .000000 | .000000 | .000000 | 102 | .011994 | 2.9 |
| NE | .000000 | .008114 | .006232 | .003410 | .006468 | .000941 | .000000 | .000000 | 214 | .025165 | 4.7 |
| ENE | .000118 | .009643 | .005997 | .005527 | .005292 | .000118 | .000000 | .000000 | 227 | .026693 | 4.4 |
| E | .000118 | .013523 | .005409 | .001176 | .000706 | .000235 | .000000 | .000000 | 180 | .021167 | 3.0 |
| ESE | .000118 | .006232 | .002587 | .001058 | .000706 | .000353 | .000000 | .000000 | 94 | .011054 | 3.4 |
| SE | .000000 | .002822 | .001881 | .000118 | .000118 | .000118 | .000000 | .000000 | 43 | .005056 | 3.0 |
| SSE | .000000 | .002469 | .000823 | .000706 | .000706 | .000353 | .000118 | .000000 | 44 | .005174 | 4.7 |
| S | .000000 | .003998 | .002469 | .002469 | .003175 | .001881 | .000706 | .000000 | 125 | .014699 | 6.2 |
| SSW | .000000 | .003410 | .003528 | .002940 | .001058 | .001411 | .000000 | .000000 | 105 | .012347 | 5.0 |
| SW | .000000 | .004704 | .006232 | .004821 | .004939 | .002940 | .000118 | .000000 | 202 | .023754 | 5.8 |
| WSW | .000000 | .008467 | .009878 | .005056 | .006115 | .002940 | .000353 | .000000 | 279 | .032808 | 5.3 |
| W | .000000 | .008231 | .005409 | .004351 | .005997 | .001764 | .000118 | .000235 | 222 | .026105 | 5.2 |
| WNW | .000000 | .004821 | .002705 | .001999 | .005880 | .001999 | .000470 | .000000 | 152 | .017874 | 6.2 |
| NW | .000000 | .005527 | .002234 | .002469 | .003881 | .001411 | .000118 | .000000 | 133 | .015640 | 5.4 |
| NNW | .000000 | .004468 | .001176 | .000353 | .000706 | .000235 | .000000 | .000000 | 59 | .006938 | 3.2 |
| HRS | 4 | 838 | 513 | 326 | 405 | 144 | 17 | 2 | 2249 | | |
| FREQ | .000470 | .098542 | .060325 | .038335 | .047625 | .016933 | .001999 | .000235 | | .264464 | |
| AVGSPD | .7 | 2.0 | 3.9 | 5.6 | 7.9 | 11.9 | 17.4 | 23.5 | | | 4.8 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

## TABLE 2.3-18

## ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR STABILITY CLASS F

## CRBRP PERMANENT TOWER, 200-FOOT LEVEL

## FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000118 | .004351 | .000470 | .000470 | .000235 | .000118 | .000000 | .000000 | 49 | .005762 | 2.5 |
| NNE | .000118 | .006468 | .000706 | .000000 | .000118 | .000000 | .000000 | .000000 | 63 | .007408 | 2.0 |
| NE | .000000 | .008819 | .002352 | .001411 | .000235 | .000118 | .000000 | .000000 | 110 | .012935 | 2.8 |
| ENE | .000235 | .007291 | .003645 | .001764 | .001058 | .000118 | .000000 | .000000 | 120 | .014111 | 3.4 |
| E | .000000 | .011054 | .001881 | .000706 | .000118 | .000000 | .000000 | .000000 | 117 | .013758 | 2.3 |
| ESE | .000000 | .006585 | .000235 | .000000 | .000000 | .000000 | .000000 | .000000 | 58 | .006820 | 1.9 |
| SE | .000000 | .006115 | .000235 | .000118 | .000000 | .000000 | .000000 | .000000 | 55 | .006468 | 1.7 |
| SSE | .000235 | .006938 | .000706 | .000235 | .000000 | .000000 | .000000 | .000000 | 69 | .008114 | 1.9 |
| S | .000353 | .007173 | .001058 | .000235 | .000353 | .000000 | .000000 | .000000 | 78 | .009172 | 2.3 |
| SSW | .000000 | .003881 | .001999 | .000470 | .000706 | .000000 | .000000 | .000000 | 60 | .007056 | 3.3 |
| SW | .000118 | .003645 | .001881 | .001176 | .000823 | .000000 | .000000 | .000000 | 65 | .007643 | 3.4 |
| WSW | .000353 | .010230 | .005174 | .001999 | .001764 | .000000 | .000000 | .000000 | 166 | .019520 | 3.2 |
| W | .000000 | .009525 | .002469 | .001176 | .000706 | .000000 | .000000 | .000000 | 118 | .0138766 | 2.8 |
| WNW | .000000 | .006468 | .000941 | .000118 | .000823 | .000000 | .000000 | .000000 | 71 | .008349 | 2.7 |
| NW | .000118 | .006115 | .001294 | .000706 | .000235 | .000000 | .000000 | .000000 | 72 | .008467 | 2.7 |
| NNW | .000000 | .002822 | .000353 | .000118 | .000235 | .000000 | .000000 | .000000 | 30 | .003528 | 2.2 |
| HRS | 14 | 914 | 216 | 91 | 63 | 3 | 0 | 0 | 1301 | | |
| FREQ | .001646 | .107479 | .025400 | .010701 | .007408 | .000353 | .000000 | .000000 | | .152987 | |
| AVGSPD | .7 | 1.8 | 3.8 | 5.5 | 7.6 | 11.2 | .0 | .0 | | | 2.7 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-19

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

STABILITY CLASS G

CRBRP PERMANENT TOWER, 200-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000118 | .001999 | .000000 | .000000 | .000000 | .000000 | .000000 | .000000 | 18 | .002117 | 1.4 |
| NNE | .000353 | .005762 | .000823 | .000000 | .000000 | .000000 | .000000 | .000000 | 59 | .006938 | 2.1 |
| NE | .000470 | .009995 | .002234 | .001176 | .000823 | .000118 | .000000 | .000000 | 126 | .014817 | 2.8 |
| ENE | .000470 | .005997 | .002940 | .002822 | .000706 | .000118 | .000000 | .000000 | 111 | .013053 | 3.4 |
| E | .000235 | .010583 | .002117 | .000823 | .000118 | .000000 | .000000 | .000000 | 118 | .013876 | 2.3 |
| ESE | .000118 | .004704 | .000470 | .000000 | .000118 | .000000 | .000000 | .000000 | 46 | .005409 | 2.0 |
| SE | .000000 | .003410 | .000235 | .000000 | .000118 | .000000 | .000000 | .000000 | 32 | .003763 | 2.0 |
| SSE | .000118 | .004233 | .000235 | .000000 | .000118 | .000000 | .000000 | .000000 | 40 | .004704 | 2.0 |
| S | .000235 | .005527 | .001176 | .000118 | .000353 | .000000 | .000000 | .000000 | 63 | .007408 | 2.6 |
| SSW | .000000 | .004116 | .003293 | .001529 | .000470 | .000000 | .000000 | .000000 | 80 | .009407 | 3.5 |
| SW | .000000 | .003410 | .003057 | .001764 | .001294 | .000000 | .000000 | .000000 | 81 | .009525 | 3.9 |
| WSW | .000118 | .011289 | .005997 | .003410 | .001764 | .000000 | .000000 | .000000 | 192 | .022578 | 3.4 |
| W | .000118 | .008114 | .003645 | .001646 | .001176 | .000118 | .000000 | .000000 | 126 | .014817 | 3.3 |
| WNW | .000118 | .004116 | .001529 | .000823 | .000118 | .000000 | .000000 | .000000 | 57 | .006703 | 2.8 |
| NW | .000118 | .004116 | .000823 | .000235 | .000118 | .000000 | .000000 | .000000 | 46 | .005409 | 2.3 |
| NNW | .000000 | .001881 | .000588 | .000118 | .000000 | .000000 | .000000 | .000000 | 22 | .002587 | 2.5 |
| HRS | 22 | 759 | 248 | 123 | 62 | 3 | 0 | 0 | 1217 | | |
| FREQ | .002587 | .089252 | .029163 | .014464 | .007291 | .000353 | .000000 | .000000 | | .143109 | |
| AVGSPD | .7 | 1.8 | 3.8 | 5.5 | 7.6 | 11.2 | .0 | .0 | | | 2.9 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7 The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-20

ANNUAL JOINT FREQUENCY OF WIND DIRECTION AND WIND SPEED FOR

ALL STABILITY CLASSES

CRBRP PERMANENT TOWER, 200-FOOT LEVEL

FEBRUARY 17, 1977 THROUGH FEBRUARY 16, 1978

| Wind Direction | Wind Speed (Knots*) .0-.7 | .8-3.0 | 3.1-4.8 | 4.9-6.5 | 6.6-10.0 | 10.1-16.1 | 16.2-21.1 | 21.2-99.9 | HRS | FREQ | AVGSPD |
|---|---|---|---|---|---|---|---|---|---|---|---|
| N | .000235 | .013288 | .003528 | .001881 | .002705 | .000353 | .000000 | .000000 | 187 | .021990 | 3.4 |
| NNE | .000588 | .025165 | .009525 | .002940 | .001881 | .000470 | .000000 | .000000 | 345 | .040569 | 2.9 |
| NE | .000470 | .035865 | .029280 | .016110 | .021049 | .003763 | .000000 | .000000 | 906 | .106538 | 4.5 |
| ENE | .000823 | .030221 | .024459 | .021284 | .015169 | .000823 | .000000 | .000000 | 789 | .092780 | 4.3 |
| E | .000353 | .044097 | .015992 | .005409 | .002705 | .000353 | .000000 | .000000 | 586 | .068909 | 2.9 |
| ESE | .000235 | .020108 | .010348 | .002822 | .001411 | .000353 | .000000 | .000000 | 300 | .035278 | 3.1 |
| SE | .000000 | .013993 | .004821 | .000706 | .000470 | .000118 | .000000 | .000000 | 171 | .020108 | 2.6 |
| SSE | .000353 | .016698 | .006703 | .003881 | .002822 | .000941 | .000588 | .000000 | 272 | .031985 | 3.9 |
| S | .000588 | .018344 | .010113 | .008231 | .009995 | .004821 | .001058 | .000235 | 454 | .053387 | 5.3 |
| SSW | .000000 | .014229 | .014699 | .009407 | .005409 | .002234 | .000000 | .000000 | 391 | .045978 | 4.5 |
| SW | .000118 | .015875 | .024694 | .020108 | .023636 | .007526 | .000706 | .000000 | 788 | .092662 | 5.8 |
| WSW | .000470 | .035748 | .039276 | .022460 | .023048 | .012347 | .003175 | .001646 | 1175 | .138170 | 5.6 |
| W | .000118 | .031515 | .020108 | .011406 | .013053 | .010818 | .001999 | .001529 | 770 | .090546 | 5.6 |
| WNW | .000118 | .018227 | .008584 | .007173 | .022460 | .017521 | .002940 | .000000 | 655 | .077023 | 7.2 |
| NW | .000235 | .018697 | .008584 | .006232 | .015757 | .009995 | .000941 | .000000 | 514 | .060442 | 6.1 |
| NNW | .000000 | .011289 | .004116 | .002587 | .004704 | .000941 | .000000 | .000000 | 201 | .023636 | 4.1 |
| HRS | 40 | 3090 | 1997 | 1213 | 1414 | 624 | 97 | 29 | 8504 | | |
| FREQ | .004704 | .363358 | .234831 | .142639 | .166275 | .073377 | .011406 | .003410 | | 1.000000 | |
| AVGSPD | .7 | 1.9 | 3.9 | 5.6 | 8.0 | 12.1 | 17.9 | 23.2 | | | 4.9 |

* 1 knot = 0.515 m/sec; 1 knot = 1.16 mph

Note: The frequencies of calms winds are given in the first wind speed column, 0.0-0.7
The .7 knots is the stall threshold speed of the wind direction sensor.

TABLE 2.3-21

MONTHLY WIND DATA SUMMARIES

| Month | Oak Ridge City Office* Average Speed (mph) | Oak Ridge City Office* Prevailing Direction | Knoxville Airport** Average Speed (mph) | Knoxville Airport** Prevailing Direction | Area Station X-10+ Average Speed (mph) | Area Station X-10+ Prevailing Direction | CRBRP Meteorological Tower ++ 33 Foot Level Average Speed (mph) | CRBRP Meteorological Tower ++ 33 Foot Level Prevailing Direction | CRBRP Meteorological Tower ++ 200 Foot Level Average Speed (mph) | CRBRP Meteorological Tower ++ 200 Foot Level Prevailing Direction |
|---|---|---|---|---|---|---|---|---|---|---|
| January | 4.8 | SW | 8.2 | NE | 5.3 | SSW | 4.5 | WNW | 7.5 | WNW |
| February | 5.0 | ENE | 8.7 | NE | 6.0 | SSW | 4.4 | WNW | 7.5 | NE |
| March | 5.3 | SW | 9.2 | NE | 6.8 | WSW | 4.3 | SSE | 7.2 | WSW |
| April | 5.7 | SW | 9.3 | WSW | 7.0 | SSW | 3.7 | WSW | 5.9 | WSW |
| May | 4.5 | SW | 7.4 | SW | 6.2 | NE | 2.8 | ENE | 4.4 | NE |
| June | 4.2 | SW | 6.7 | SW | 6.2 | WSW | 3.3 | SW | 5.3 | WSW |
| July | 3.9 | SW | 6.3 | WSW | 4.2 | SSW | 2.8 | WSW | 4.1 | WSW |
| August | 3.7 | E | 5.7 | NE | 1.5 | SSW | 2.6 | SW | 4.1 | WSW |
| September | 3.8 | E | 5.9 | NE | 2.9 | NNE | 2.4 | E | 4.1 | NE |
| October | 3.6 | E | 5.9 | NE | 2.9 | NNE | 3.0 | WNW | 4.8 | WSW |
| November | 4.1 | E | 7.2 | NE | 3.2 | N | 3.3 | WNW | 6.0 | ENE |
| December | 4.5 | SW | 7.6 | NE | 4.3 | NNE | 4.0 | WNW | 6.8 | WSW |
| Annual | 4.4 | SW | 7.3 | NE | 4.7 | SSW | 3.5 | WNW | 5.6 | WSW |

* 16-year record on wind speed, 13-year record on prevailing direction[4]

** 31-year record on wind speed, 14-year record on prevailing direction[3]

\+ 1-year record[26] (102 feet, sensor elevation)

++ 1-year record, February 17, 1977 - February 16, 1978

TABLE 2.3-22

MONTHLY AVERAGE RELATIVE HUMIDITY VALUES FOR THE CRBRP SITE[1]

FEBRUARY 17, 1977 - FEBRUARY 16, 1978

| Month | Relative Humidity In Percent (%) |
|---|---|
| February | 60 |
| March | 64 |
| April | 69 |
| May | 77 |
| June | 77 |
| July | 77 |
| August | 81 |
| September | 86 |
| October | 80 |
| November | 80 |
| December | 74 |
| January | 75 |
| Annual Average | 75 |

1. Derived from dew point and ambient temperature data collected at the 370 foot permanent tower site.



Amend. 65
Feb. 1982

TABLE 2.3-23

MONTHLY AVERAGE RELATEIVE HUMIDITY VALUES(3)
FOR KNOXVILLE AIRPORT
1961-1973

| Month | Relative Humidity at Indicated Time (E.S.T.) 0100 | 0700 | 1300 | 1900 | Average |
|---|---|---|---|---|---|
| January | 76 | 79 | 63 | 64 | 71 |
| February | 71 | 77 | 60 | 59 | 67 |
| March | 69 | 78 | 54 | 54 | 64 |
| April | 70 | 78 | 51 | 52 | 63 |
| May | 77 | 83 | 54 | 56 | 68 |
| June | 84 | 88 | 59 | 62 | 73 |
| July | 86 | 90 | 62 | 66 | 76 |
| August | 87 | 92 | 61 | 66 | 77 |
| September | 86 | 91 | 58 | 66 | 75 |
| October | 83 | 88 | 55 | 62 | 72 |
| November | 78 | 83 | 59 | 65 | 71 |
| December | 76 | 80 | 64 | 67 | 72 |
| Year | 79 | 84 | 58 | 62 | 71 |

(3) Reference 3



Amend. 65
Feb. 1982

TABLE 2.3-24

PRECIPITATION DATA SUMMARY
OAK RIDGE AREA STATION, X-10 (1)
1944 - 1964

| Month | Monthly Average* (Inches) | Monthly Maximum (Inches) | Monthly Minimum (Inches) | Maximum In 24 Hours (Inches) |
|---|---|---|---|---|
| December | 5.22 | 10.28 | 1.98 | 4.38 |
| January | 5.24 | 12.37 | 1.11 | 3.96 |
| February | 5.39 | 10.01 | 1.89 | 3.23 |
| Winter | 15.85 | | | |
| March | 5.44 | 9.69 | 2.06 | 3.84 |
| April | 4.14 | 8.54 | 1.25 | 2.39 |
| May | 3.48 | 7.01 | 0.90 | 2.09 |
| Spring | 13.06 | | | |
| June | 3.38 | 7.55 | 1.18 | 3.08 |
| July | 5.31 | 10.19 | 2.14 | 3.74 |
| August | 4.02 | 10.31 | 0.50 | 3.31 |
| Summer | 12.71 | | | |
| September | 3.59 | 12.84 | 0.21 | 7.75 |
| October | 2.82 | 6.43 | 0.00 | 2.32 |
| November | 3.49 | 12.00 | 1.01 | 3.20 |
| Fall | 9.90 | | | |
| Annual | 51.52 | 12.84 | 0.00 | 7.75 |

* Standard climatological normals (1931-1960)

(1) - reference 1

TABLE 2.3-25

PRECIPITATION SUMMARY FOR THE CRBRP SITE[1]
FEBRUARY 17, 1977 - FEBRUARY 16, 1978

| Month | Precipitation In Inches |
|---|---|
| February | 1.44 |
| March | 4.81 |
| April | 6.95 |
| May | 1.36 |
| June | 3.55 |
| July | 1.01 |
| August | 4.22 |
| September | 8.96 |
| October | 4.36 |
| November | 6.55 |
| December | 3.37 |
| January | 5.21 |
| Annual total | 51.79 |

1. Data collected at permanent 370 foot tower site.

TABLE 2.3-26

SNOW OR ICE PELLET SUMMARY FOR
OAK RIDGE CITY OFFICE(4)
1948 - OCTOBER 1974

| | Snow or Ice Pellets (Inches | | |
|---|---|---|---|
| Month | Mean* Total | Maximum Monthly | Maximum In 24 Hours |
| January | 3.2 | 9.6 | 8.3 |
| February | 2.9 | 11.3 | 9.1 |
| March | 1.5 | 21.0 | 12.0 |
| April | T | 0.3 | 0.3 |
| May | 0.0 | 0.0 | 0.0 |
| June | 0.0 | 0.0 | 0.0 |
| July | 0.0 | 0.0 | 0.0 |
| August | 0.0 | 0.0 | 0.0 |
| September | 0.0 | 0.0 | 0.0 |
| October | T | T | T |
| November | 0.5 | 6.5 | 6.5 |
| December | 2.2 | 14.8 | 10.8 |
| Year | 10.3 | 21.0 | 12.0 |

Maximum Annual 41.4 Inches (1959 - 1960 snowfall season)

*1949 - 1973

T = Trace

(4) - Reference 4

TABLE 2.3-27

MONTHLY MEAN NUMBER OF HEAVY FOG DAYS FOR KNOXVILLE**
AND OAK RIDGE CITY OFFICE+

| Month | Fog Days (mean number) Knoxville | Oak Ridge |
|---|---|---|
| January | 3 | 1 |
| February | 2 | 1 |
| March | 1 | 1 |
| April | 1 | 1 |
| May | 2 | 2 |
| June | 2 | 2 |
| July | 2 | 3 |
| August | 3 | 4 |
| September | 4 | 4 |
| October | 5 | 8 |
| November | 3 | 6 |
| December | 2 | 2 |
| Annual | 31 | 34 |

* Visibility less than 1/4 mile
** 31-year record (1943-1973) (Reference 3)
+ 14-year record (1951-1964) (Reference 4)

TABLE 2.3-28

MEAN NUMBER OF DAYS WITH FOG
JANUARY 1964 THROUGH OCTOBER 1970[22]
(Visibility Less Than Stated Value)

| Month | Melton Hill Lake at Bull Run Creek, Clinch River Mile 46.4 | | Melton Hill Lake at Dam, Clinch River Mile 23.1 | |
|---|---|---|---|---|
| | <1100 yards | <550 yards | <1100 yards | <550 yards |
| January | 3 | 2 | 4 | 4 |
| February | 3 | 2 | 5 | 4 |
| March | 2 | 1 | 5 | 4 |
| April | 2 | 1 | 6 | 5 |
| May | 6 | 4 | 8 | 7 |
| June | 7 | 3 | 12 | 10 |
| July | 12 | 7 | 13 | 10 |
| August | 15 | 9 | 14 | 12 |
| September | 13 | 8 | 16 | 15 |
| October | 10 | 8 | 15 | 14 |
| November | 11 | 6 | 13 | 12 |
| December | 6 | 3 | 8 | 8 |
| Annual* | 91 | 54 | 119 | 106 |

(22) - Reference 22

* Annual values may differs slightly from the sums of the respective months because of rounding procedures.

TABLE 2.3-29

NUMBER AND PERCENT OCCURRENCE

PASQUILL STABILITY CLASSES

CRBRP PERMANENT TOWER

| | | Stability Classes | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | A | B | C | D | E | F | G |
| March 1977 | Number | 2 | 9 | 37 | 249 | 165 | 69 | 208 |
| | Percent | 0.27 | 1.22 | 5.01 | 33.69 | 22.33 | 9.34 | 28.15 |
| April 1977 | Number | 19 | 20 | 24 | 235 | 156 | 67 | 194 |
| | Percent | 2.66 | 2.80 | 3.36 | 32.87 | 21.82 | 9.37 | 27.13 |
| May 1977 | Number | 1 | 13 | 44 | 253 | 158 | 143 | 128 |
| | Percent | 0.14 | 1.76 | 5.95 | 34.19 | 21.35 | 19.32 | 17.30 |
| June 1977 | Number | 17 | 21 | 51 | 196 | 153 | 82 | 95 |
| | Percent | 2.76 | 3.41 | 8.29 | 31.87 | 24.88 | 13.33 | 15.45 |
| July 1977 | Number | 3 | 14 | 82 | 241 | 114 | 175 | 67 |
| | Percent | 0.43 | 2.01 | 11.78 | 34.60 | 16.38 | 25.14 | 9.63 |
| August 1977 | Number | 1 | 11 | 40 | 245 | 185 | 175 | 76 |
| | Percent | 0.14 | 1.50 | 5.46 | 33.42 | 25.24 | 23.87 | 10.37 |
| September 1977 | Number | 0 | 2 | 17 | 239 | 241 | 176 | 41 |
| | Percent | 0.0 | 0.28 | 2.37 | 33.38 | 33.66 | 24.58 | 5.73 |
| October 1977 | Number | 6 | 14 | 37 | 205 | 227 | 148 | 101 |
| | Percent | 0.81 | 1.90 | 5.01 | 27.78 | 30.76 | 20.05 | 13.68 |
| November 1977 | Number | 3 | 14 | 19 | 276 | 218 | 84 | 106 |
| | Percent | 0.42 | 1.94 | 2.64 | 38.33 | 30.28 | 11.67 | 14.72 |
| December 1977 | Number | 3 | 11 | 26 | 287 | 232 | 91 | 78 |
| | Percent | 0.41 | 1.51 | 3.57 | 39.42 | 31.87 | 12.50 | 10.71 |
| January 1978 | Number | 14 | 23 | 24 | 364 | 239 | 44 | 36 |
| | Percent | 1.88 | 3.09 | 3.22 | 48.92 | 32.12 | 5.91 | 4.83 |
| February 1978 | Number | 3 | 10 | 30 | 283 | 161 | 47 | 87 |
| | Percent | 0.48 | 1.61 | 4.83 | 45.57 | 25.93 | 7.57 | 14.01 |
| Annual* | Number | 72 | 162 | 431 | 3072 | 2249 | 1301 | 1217 |
| | Percent | 0.85 | 1.90 | 5.07 | 36.12 | 26.46 | 15.30 | 14.31 |

* February 17, 1977 - February 16, 1978

TABLE 2.3-30

# DESIGN BASIS ACCIDENT χ/Q VALUES FOR THE EXCLUSION AREA BOUNDARY (EAB) AND LOW POPULATION ZONE (LPZ) DISTANCES[1]
## 33 FT. WIND SPEED AND DIRECTION; 200-FT TO 33-FT DELTA T
## PERMANENT TOWER DATA
## FEBRUARY 17, 1977 - FEBRUARY 16, 1978

| | EAB | LPZ | | | | | |
|---|---|---|---|---|---|---|---|
| | 0-2 Hr | 0-2 Hr | 0-8 Hr | 8-24 Hr | 1-4 Day | 4-30 Day | Annual |
| Design Accident /Q Value[2]: | 1.1E-3 | 2.6E-4 | 1.2E-4 | 8.4E-5 | 3.7E-5 | 1.2E-5 | 2.8E-6 |
| Maximum Sector 0.5 Percentile /Q Value: | 1.1E-3 | 2.6E-4 | 1.2E-4 | 8.4E-5 | 3.7E-5 | 1.2E-5 | 2.8E-6 |
| Overall Site 5th Percentile /Q Value: | 8.7E-4 | 2.3E-4 | 1.1E-4 | 7.7E-5 | 3.5E-5 | 1.1E-5 | 2.8E-6 |

Sector Dependent 0.5 Percentile /Q Values

| Exclusion Area Boundary (2 hr only) | | | Low Population Zone[4] | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| Direction[5] | Value | Distance (m) | Direction[5] | 0-2 Hr | 0-8 Hr | 8-24 Hr | 1-4 Day | 4-30 Day | Annual |
| N | 3.5E-4 | 681 | N | 8.8E-5 | 3.9E-5 | 2.6E-5 | 1.1E-5 | 3.2E-6 | 7.0E-7 |
| NNE | 8.2E-4 | 671 | NNE | 1.2E-4 | 5.2E-5 | 3.4E-5 | 1.4E-5 | 3.7E-6 | 7.7E-7 |
| NE | 1.0E-3 | 671 | NE | 1.5E-4 | 6.5E-5 | 4.3E-5 | 1.7E-5 | 4.7E-6 | 9.7E-7 |
| ENE | 1.1E-3 | 671 | ENE | 1.9E-4 | 8.4E-5 | 5.6E-5 | 2.3E-5 | 6.5E-6 | 1.4E-6 |
| E | 9.1E-4 | 718 | E | 1.9E-4 | 8.9E-5 | 6.0E-5 | 2.6E-5 | 8.0E-6 | 1.9E-6 |
| ESE | 7.8E-4 | 832 | ESE | 2.1E-4 | 9.3E-5 | 6.1E-5 | 2.5E-5 | 7.0E-6 | 1.5E-6 |
| SE | 9.8E-4 | 832 | SE | 2.6E-4 | 1.2E-4 | 8.4E-5 | 3.7E-5 | 1.2E-5 | 2.8E-6 |
| SSE | 8.9E-4 | 870 | SSE | 2.3E-4 | 1.0E-4 | 7.1E-5 | 3.0E-5 | 8.8E-6 | 2.0E-6 |
| S | 1.7E-4 | 1966 | S | 8.9E-5 | 3.9E-5 | 2.5E-5 | 1.0E-5 | 2.8E-6 | 5.8E-7 |
| SSW | 2.9E-4 | 1134 | SSW | 9.1E-5 | 3.8E-5 | 2.5E-5 | 9.8E-6 | 2.5E-6 | 5.0E-7 |
| SW | 6.0E-4 | 832 | SW | 1.5E-4 | 6.4E-5 | 4.2E-5 | 1.7E-5 | 4.4E-6 | 8.9E-7 |
| WSW | 5.4E-4 | 839 | WSW | 1.4E-4 | 6.4E-5 | 4.3E-5 | 1.8E-5 | 5.3E-6 | 1.2E-6 |
| W | 6.2E-4 | 839 | W | 1.6E-4 | 7.4E-5 | 5.0E-5 | 2.2E-5 | 6.5E-6 | 1.5E-6 |
| WNW | 6.4E-4 | 882 | WNW | 1.7E-4 | 7.9E-5 | 5.4E-5 | 2.4E-5 | 7.2E-6 | 1.7E-6 |
| NW | 5.3E-4 | 1008 | NW | 1.4E-4 | 6.4E-5 | 4.3E-5 | 1.8E-5 | 5.3E-6 | 1.2E-6 |
| NNW | 6.9E-4 | 756 | NNW | 1.3E-4 | 5.6E-5 | 3.7E-5 | 1.5E-5 | 4.1E-6 | 8.5E-7 |

1. Computed according to R.G. 1.145
2. Greater of maximum sector 0.5 percentile value and 5th percentile overall
3. Computed according to R.G. 1.111
4. Distance of approximately 2.5 miles
5. Direction from which wind is blowing

TABLE 2.3-31

FIFTIETH PERCENTILE X/Q VALUES FOR EAB AND LPZ DISTANCES[1]
33-FT WIND SPEED AND DIRECTION; 200-FT TO 33-FT DELTA T
PERMANENT TOWER DATA
FEBRUARY 17, 1977 - FEBRUARY 16, 1978

| | Distance (miles) | 50th Percentile X/Q Values (sec/m$^3$) | | | | |
|---|---|---|---|---|---|---|
| | | 0-2 Hr | 0-8 Hr | 8-24 Hr | 1-4 Day | 4-30 Day |
| Minimum Exclusion Area Boundary | 0.42 | 1.01E-3 | 1.55E-4 | 1.23E-4 | 7.69E-5 | 9.06E-5 |
| Low Population Zone Distance | 2.5 | 1.59E-4 | 2.30E-5 | 3.58E-6 | 2.29E-6 | 2.60E-6 |

1. Values computed according to R.G. 1.4 (6/74). These maximum sector values occurred in the west-northwest or northwest sectors (east-southeast or southeast wind direction).

TABLE 2.3-32

ANNUAL AVERAGE χ/Q's (In Sec/m³) AT VARIOUS DOWNWIND DISTANCES
FOR EACH WIND SECTOR BASED ON PERMANENT TOWER DATA
FEBRUARY 17, 1977 - FEBRUARY 16, 1978
(33-Foot Wind and 200-Foot to 33-Foot Delta T)

| Downwind Distance (Miles) | Annual Average χ/Q Value (sec/m³)* Wind Direction** | | | | | | | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | N | NNE | NE | ENE | E | ESE | SE | SSE | S | SSW | SW | WSW | W | WNW | NW | NNW |
| 0.1 | 2.95E-4 | 3.40E-4 | 4.20E-4 | 6.23E-4 | 8.05E-4 | 6.60E-4 | 1.29E-3 | 9.29E-4 | 2.51E-4 | 2.21E-4 | 3.91E-4 | 5.13E-4 | 6.57E-4 | 7.58E-4 | 4.88E-4 | 3.69E-4 |
| 0.2 | 8.44E-5 | 9.78E-5 | 1.21E-4 | 1.80E-4 | 2.32E-4 | 1.91E-4 | 3.74E-4 | 2.68E-4 | 7.22E-5 | 6.38E-5 | 1.12E-4 | 1.47E-4 | 1.89E-4 | 2.18E-4 | 1.40E-4 | 1.06E-4 |
| 0.3 | 4.18E-5 | 4.84E-5 | 5.99E-5 | 8.88E-5 | 1.15E-4 | 9.34E-5 | 1.83E-4 | 1.32E-4 | 3.58E-5 | 3.15E-5 | 5.61E-5 | 7.34E-5 | 9.31E-5 | 1.07E-4 | 6.89E-5 | 5.21E-5 |
| 0.34 | 3.35E-5 | 3.86E-5 | 4.78E-5 | 7.08E-5 | 9.08E-5 | 7.39E-5 | 1.44E-4 | 1.04E-4 | 2.85E-5 | 2.51E-5 | 4.50E-5 | 5.88E-5 | 7.40E-5 | 8.55E-5 | 5.49E-5 | 4.16E-5 |
| 0.42 | 2.43E-5 | 2.77E-5 | 3.42E-5 | 5.06E-5 | 6.47E-5 | 5.24E-5 | 1.02E-4 | 7.41E-5 | 2.05E-5 | 1.79E-5 | 3.25E-5 | 4.25E-5 | 5.30E-5 | 6.12E-5 | 3.95E-5 | 2.99E-5 |
| 0.5 | 1.86E-5 | 2.10E-5 | 2.59E-5 | 3.81E-5 | 4.88E-5 | 3.92E-5 | 7.59E-5 | 5.57E-5 | 1.56E-5 | 1.35E-5 | 2.47E-5 | 3.24E-5 | 4.03E-5 | 4.63E-5 | 2.99E-5 | 2.27E-5 |
| 0.6 | 1.41E-5 | 1.58E-5 | 1.94E-5 | 2.85E-5 | 3.64E-5 | 2.92E-5 | 5.62E-5 | 4.14E-5 | 1.18E-5 | 1.01E-5 | 1.86E-5 | 2.44E-5 | 3.02E-5 | 3.47E-5 | 2.25E-5 | 1.71E-5 |
| 0.7 | 1.12E-5 | 1.25E-5 | 1.53E-5 | 2.24E-5 | 2.86E-5 | 2.89E-5 | 4.39E-5 | 3.25E-5 | 9.28E-6 | 7.92E-6 | 1.47E-5 | 1.93E-5 | 2.39E-5 | 2.74E-5 | 1.79E-5 | 1.36E-5 |
| 1.0 | 5.49E-6 | 6.03E-6 | 7.41E-6 | 1.08E-5 | 1.37E-5 | 1.10E-5 | 2.11E-5 | 1.56E-5 | 4.46E-6 | 3.80E-6 | 7.04E-6 | 9.31E-6 | 1.16E-5 | 1.33E-5 | 8.79E-6 | 6.65E-6 |
| 1.5 | 2.32E-6 | 2.45E-6 | 2.97E-6 | 4.24E-6 | 5.39E-6 | 4.33E-6 | 8.20E-6 | 6.19E-6 | 1.81E-6 | 1.49E-6 | 2.81E-6 | 3.77E-6 | 4.73E-6 | 5.37E-6 | 3.66E-6 | 2.76E-6 |
| 2.0 | 1.13E-6 | 1.24E-6 | 1.52E-6 | 2.22E-6 | 2.81E-6 | 2.30E-6 | 4.41E-6 | 3.23E-6 | 9.07E-7 | 7.77E-7 | 1.42E-6 | 1.89E-6 | 2.39E-6 | 2.74E-6 | 1.84E-6 | 1.38E-6 |
| 2.5 | 7.07E-7 | 7.77E-7 | 9.56E-7 | 1.39E-6 | 1.78E-6 | 1.46E-6 | 2.82E-6 | 2.05E-6 | 5.70E-7 | 4.89E-7 | 8.82E-7 | 1.18E-6 | 1.51E-6 | 1.73E-6 | 1.16E-6 | 8.70E-7 |
| 3.0 | 4.97E-7 | 5.52E-7 | 6.80E-7 | 9.96E-7 | 1.27E-6 | 1.05E-6 | 2.03E-6 | 1.47E-6 | 4.06E-7 | 3.50E-7 | 6.27E-7 | 8.36E-7 | 1.07E-6 | 1.23E-6 | 8.18E-7 | 6.15E-7 |
| 3.5 | 3.70E-7 | 4.13E-7 | 5.09E-7 | 7.47E-7 | 9.56E-7 | 7.89E-7 | 1.53E-6 | 1.11E-6 | 3.04E-7 | 2.62E-7 | 4.67E-7 | 6.23E-7 | 8.04E-7 | 9.21E-7 | 6.12E-7 | 4.60E-7 |
| 4.0 | 2.91E-7 | 3.26E-7 | 4.02E-7 | 5.91E-7 | 7.39E-7 | 6.27E-7 | 1.22E-6 | 8.79E-7 | 2.40E-7 | 2.08E-7 | 3.68E-7 | 4.90E-7 | 6.35E-7 | 7.28E-7 | 4.82E-7 | 3.62E-7 |
| 4.5 | 2.36E-7 | 2.65E-7 | 3.26E-7 | 4.80E-7 | 6.18E-7 | 5.11E-7 | 9.97E-7 | 7.16E-7 | 1.95E-7 | 1.69E-7 | 2.97E-7 | 3.97E-7 | 5.17E-7 | 5.92E-7 | 3.92E-7 | 2.94E-7 |
| 5.0 | 1.97E-7 | 2.22E-7 | 2.73E-7 | 4.03E-7 | 5.20E-7 | 4.31E-7 | 8.42E-7 | 6.03E-7 | 1.63E-7 | 1.42E-7 | 2.49E-7 | 3.32E-7 | 4.34E-7 | 4.97E-7 | 3.28E-7 | 2.46E-7 |
| 7.0 | 1.08E-7 | 1.23E-7 | 1.52E-7 | 2.25E-7 | 2.93E-7 | 2.43E-7 | 4.79E-7 | 3.40E-7 | 9.10E-8 | 7.95E-8 | 1.37E-7 | 1.83E-7 | 2.42E-7 | 2.77E-7 | 1.82E-7 | 1.36E-7 |
| 7.5 | 9.66E-8 | 1.11E-7 | 1.37E-7 | 2.03E-7 | 2.63E-7 | 2.19E-7 | 4.32E-7 | 3.07E-7 | 8.18E-8 | 7.15E-8 | 1.23E-7 | 1.64E-7 | 2.17E-7 | 2.49E-7 | 1.63E-7 | 1.22E-7 |
| 9.0 | 7.59E-8 | 8.76E-8 | 1.08E-7 | 1.61E-7 | 2.10E-7 | 1.75E-7 | 3.46E-7 | 2.45E-7 | 6.48E-8 | 5.68E-8 | 9.67E-8 | 1.29E-7 | 1.72E-7 | 1.98E-7 | 1.29E-7 | 9.67E-8 |
| 10.0 | 6.60E-8 | 7.64E-8 | 9.42E-8 | 1.40E-7 | 1.84E-7 | 1.53E-7 | 3.04E-7 | 2.14E-7 | 5.65E-8 | 4.97E-8 | 8.41E-8 | 1.12E-7 | 1.50E-7 | 1.72E-7 | 1.12E-7 | 8.45E-8 |
| 15.0 | 3.81E-8 | 4.47E-8 | 5.53E-8 | 8.27E-8 | 1.09E-7 | 9.14E-8 | 1.82E-7 | 1.27E-7 | 3.31E-8 | 2.93E-8 | 4.88E-8 | 6.52E-8 | 8.83E-8 | 1.01E-7 | 6.54E-8 | 4.92E-8 |
| 20.0 | 2.61E-8 | 3.10E-8 | 3.83E-8 | 5.75E-8 | 7.61E-8 | 6.40E-8 | 1.28E-7 | 8.92E-8 | 2.30E-8 | 2.04E-8 | 3.37E-8 | 4.49E-8 | 6.14E-8 | 7.04E-8 | 4.52E-8 | 3.40E-8 |
| 21.0 | 2.45E-8 | 2.91E-8 | 3.59E-8 | 5.40E-8 | 7.15E-8 | 6.02E-8 | 1.20E-7 | 8.38E-8 | 2.16E-8 | 1.92E-8 | 3.15E-8 | 4.21E-8 | 5.76E-8 | 6.61E-8 | 4.25E-8 | 3.19E-8 |
| 25.0 | 1.95E-8 | 2.33E-8 | 2.88E-8 | 4.34E-8 | 5.76E-8 | 4.85E-8 | 9.74E-8 | 6.76E-8 | 1.73E-8 | 1.54E-8 | 2.52E-8 | 3.37E-8 | 4.63E-8 | 5.31E-8 | 3.40E-8 | 2.56E-8 |
| 35.0 | 1.27E-8 | 1.54E-8 | 1.90E-8 | 2.88E-8 | 3.86E-8 | 3.25E-8 | 6.57E-8 | 4.54E-8 | 1.15E-8 | 1.03E-8 | 1.66E-8 | 2.21E-8 | 3.06E-8 | 3.52E-8 | 2.22E-8 | 1.68E-8 |
| 45.0 | 9.31E-9 | 1.14E-8 | 1.41E-8 | 2.14E-8 | 2.89E-8 | 2.43E-8 | 4.93E-8 | 3.39E-8 | 8.54E-9 | 7.67E-9 | 1.22E-8 | 1.63E-8 | 2.28E-8 | 2.61E-8 | 1.64E-8 | 1.24E-8 |
| 50.0 | 8.23E-9 | 1.02E-8 | 1.25E-8 | 1.91E-8 | 2.58E-8 | 2.17E-8 | 4.41E-8 | 3.03E-8 | 7.61E-9 | 6.84E-9 | 1.08E-8 | 1.44E-8 | 2.02E-8 | 2.32E-8 | 1.45E-8 | 1.10E-8 |

* Computed according to R.G. 1.111
** Direction from which wind is blowing

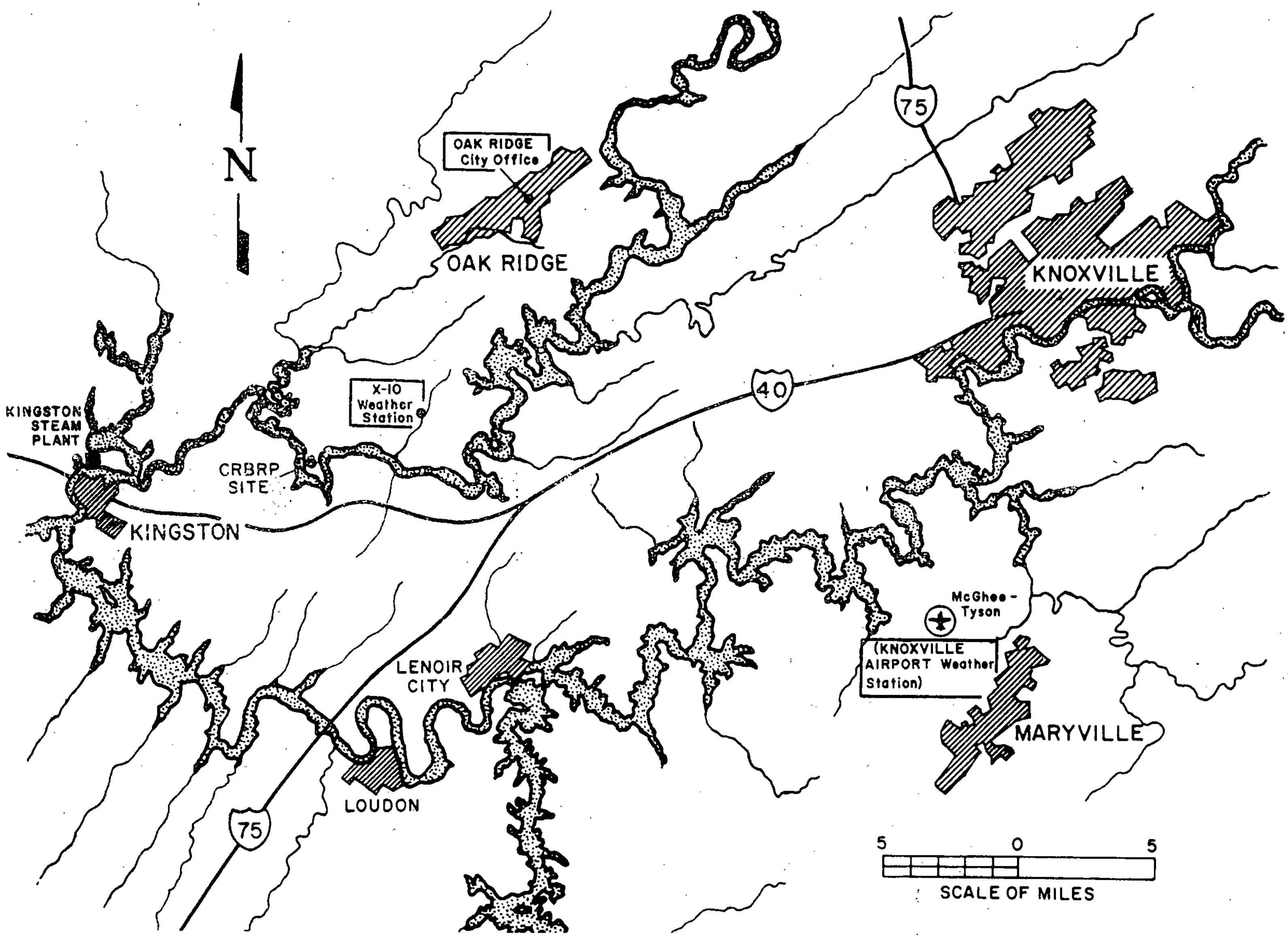



Figure 2.3-1 LOCATIONS OF WEATHER STATIONS NEAR SITE

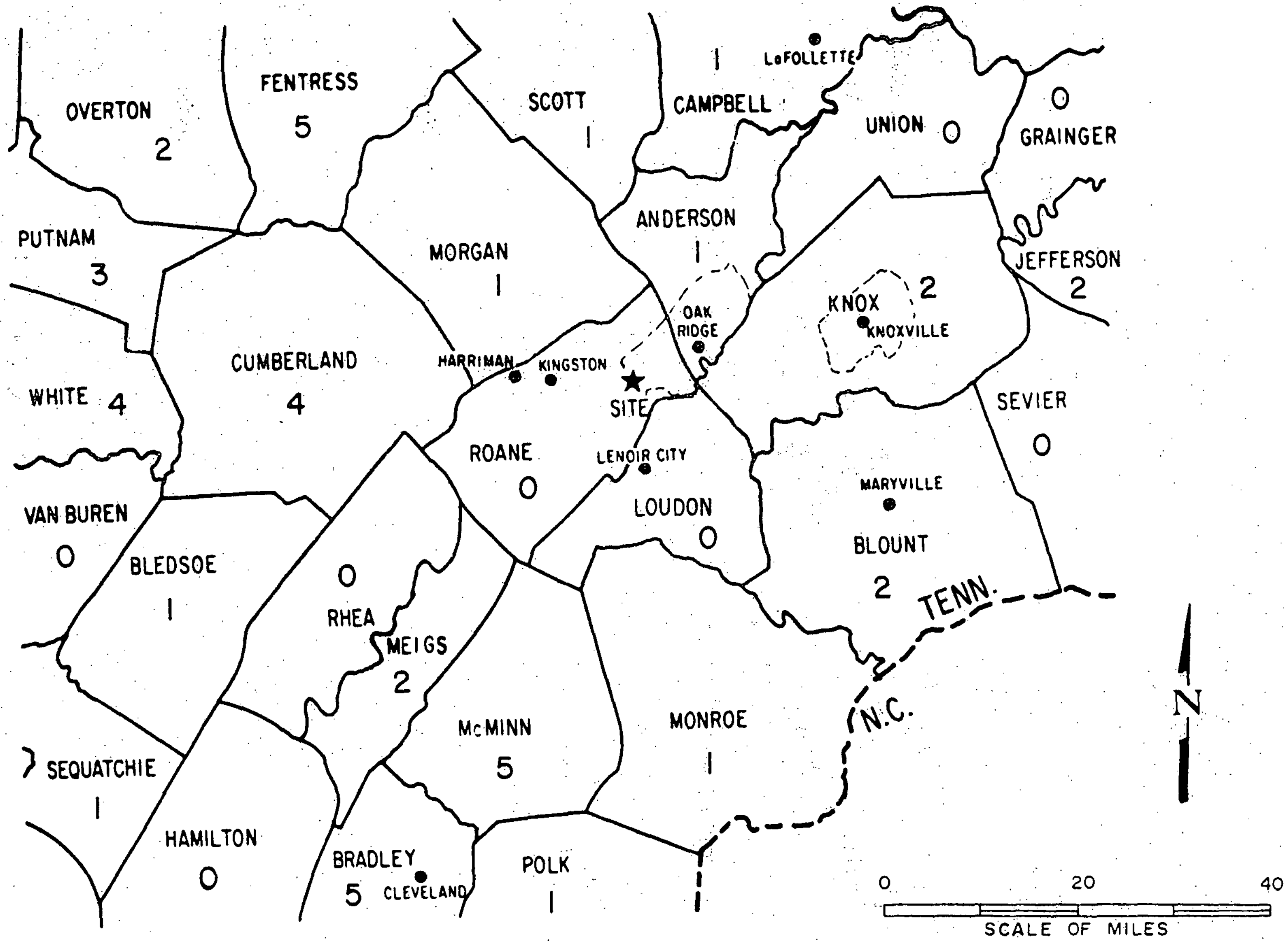
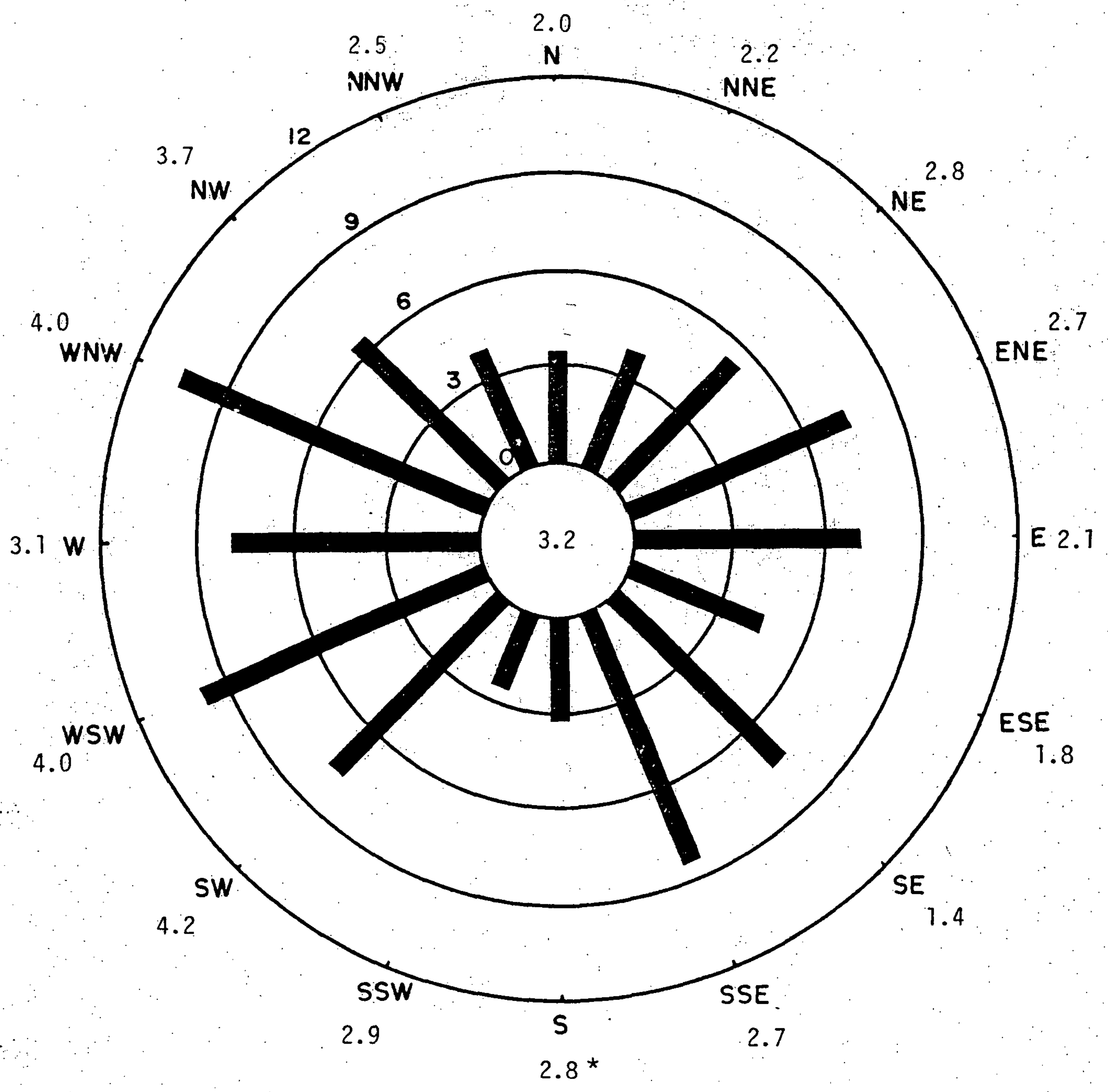


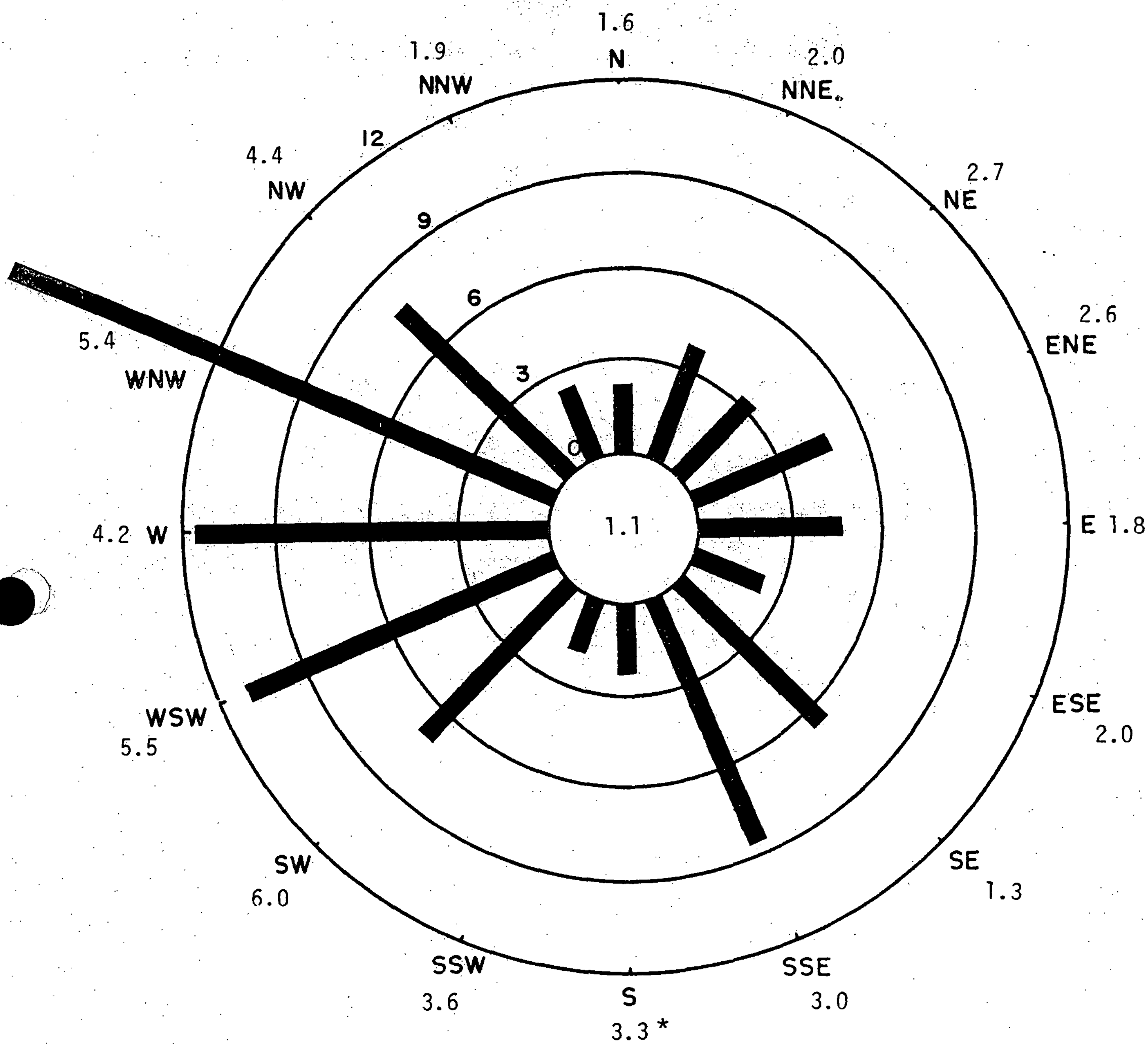

Figure 2.3-2 RAW COUNT DATA ON TORNADO OCCURENCE FOR COUNTIES NEAR SITE, 1916 - 1972




*Value denotes average wind speed for each sector.

Figure 2.3-3 ANNUAL WIND ROSE FOR THE 33 FOOT LEVEL FROM CRBRP PERMANENT TOWER METEOROLOGICAL DATA FOR FEB. 17, 1977 THROUGH FEB. 16, 1978

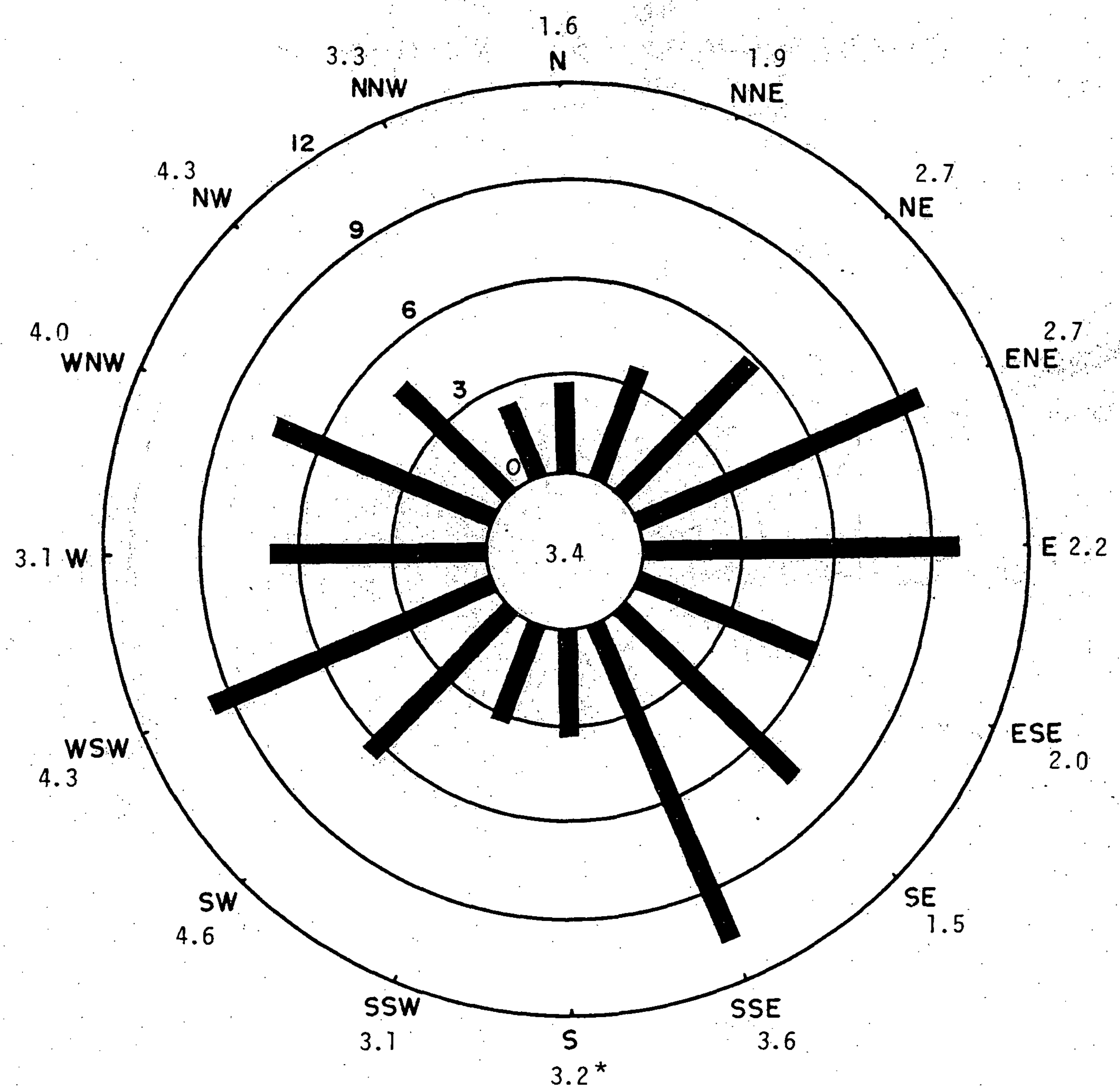
Amend. 65
Feb. 1982




*Value denotes average wind speed for each sector.

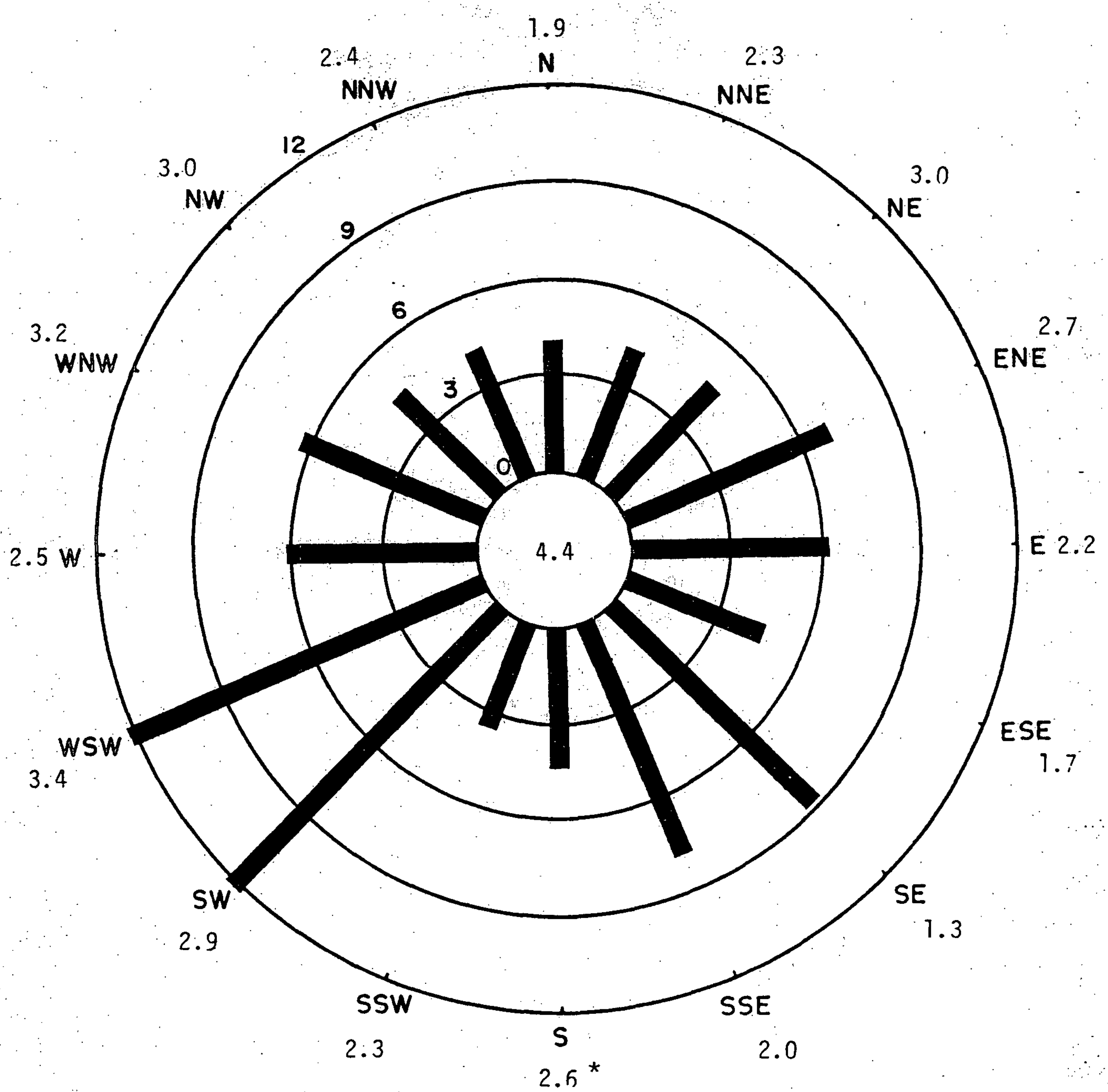
Figure 2.3-4 WINTER WIND ROSE FOR THE 33-FOOT LEVEL FROM CRBRP PERMANENT TOWER METEOROLOGICAL DATA

Amend. 65
Feb. 1982



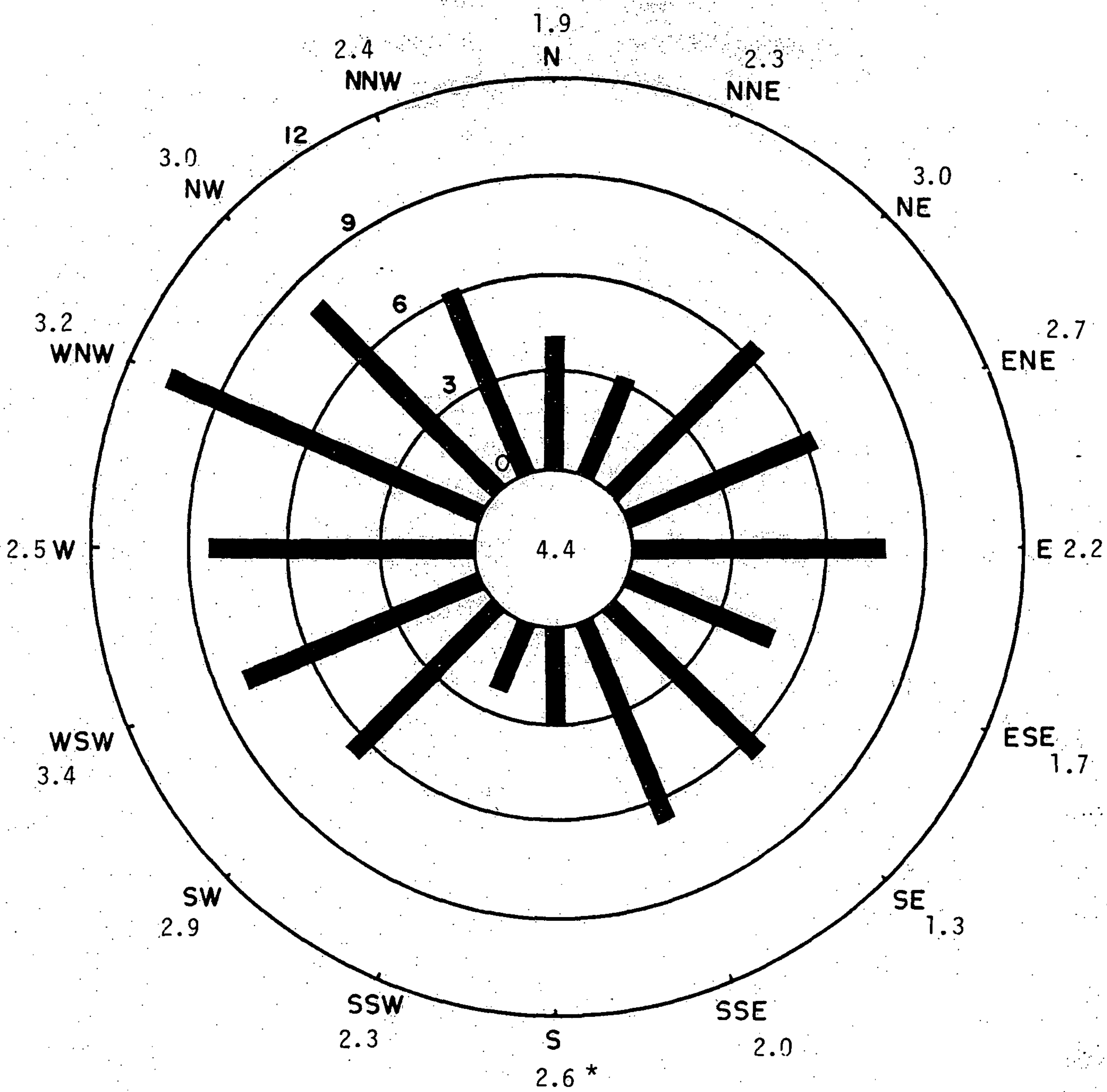

*Value denotes average wind speed for each sector.

Figure 2.3-5 SPRING WIND ROSE FOR THE 33-FOOT LEVEL FROM CRBRP PERMANENT TOWER METEOROLOGICAL DATA




*Value denotes average wind speed for each sector.

Figure 2.3-6 SUMMER WIND ROSE FOR THE 33-FOOT LEVEL FROM CRBRP PERMANENT TOWER METEOROLOGICAL DATA

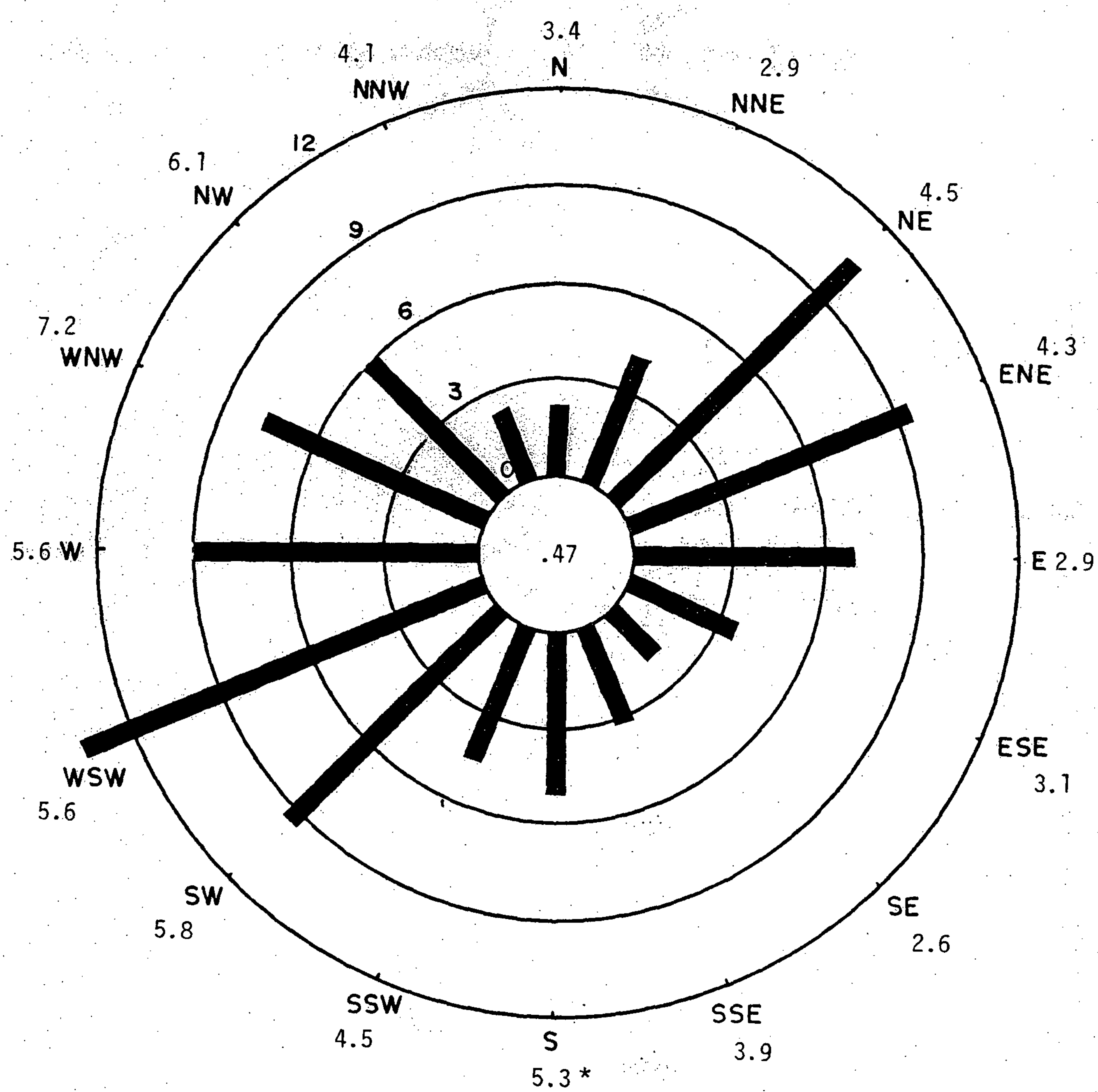



*Value denotes average wind speed for each sector.

Figure 2.3-7 FALL WIND ROSE FOR THE 33-FOOT LEVEL FROM CRBRP PERMANENT TOWER METEOROLOGICAL DATA

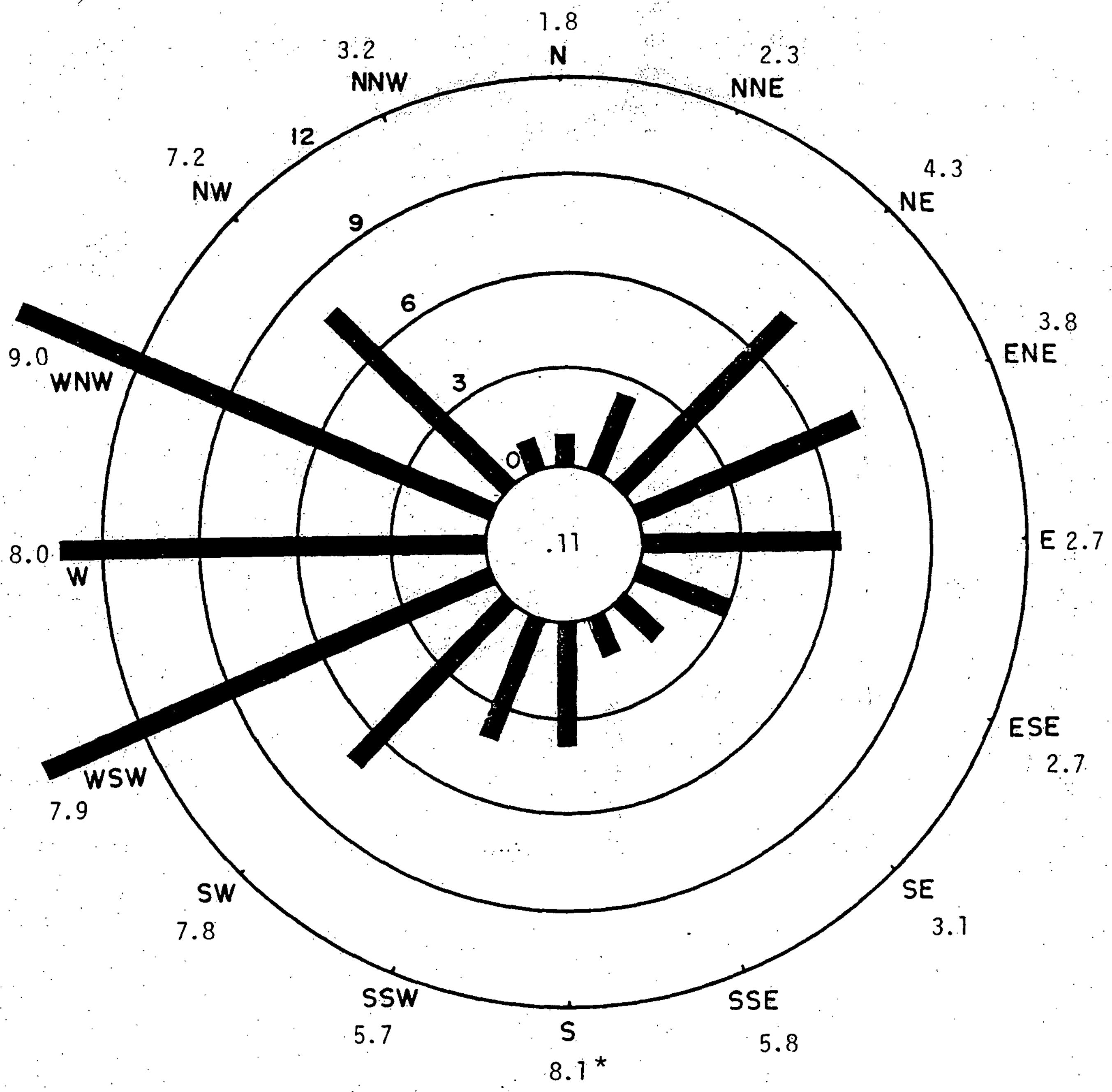


Amend. 65
Feb. 1982



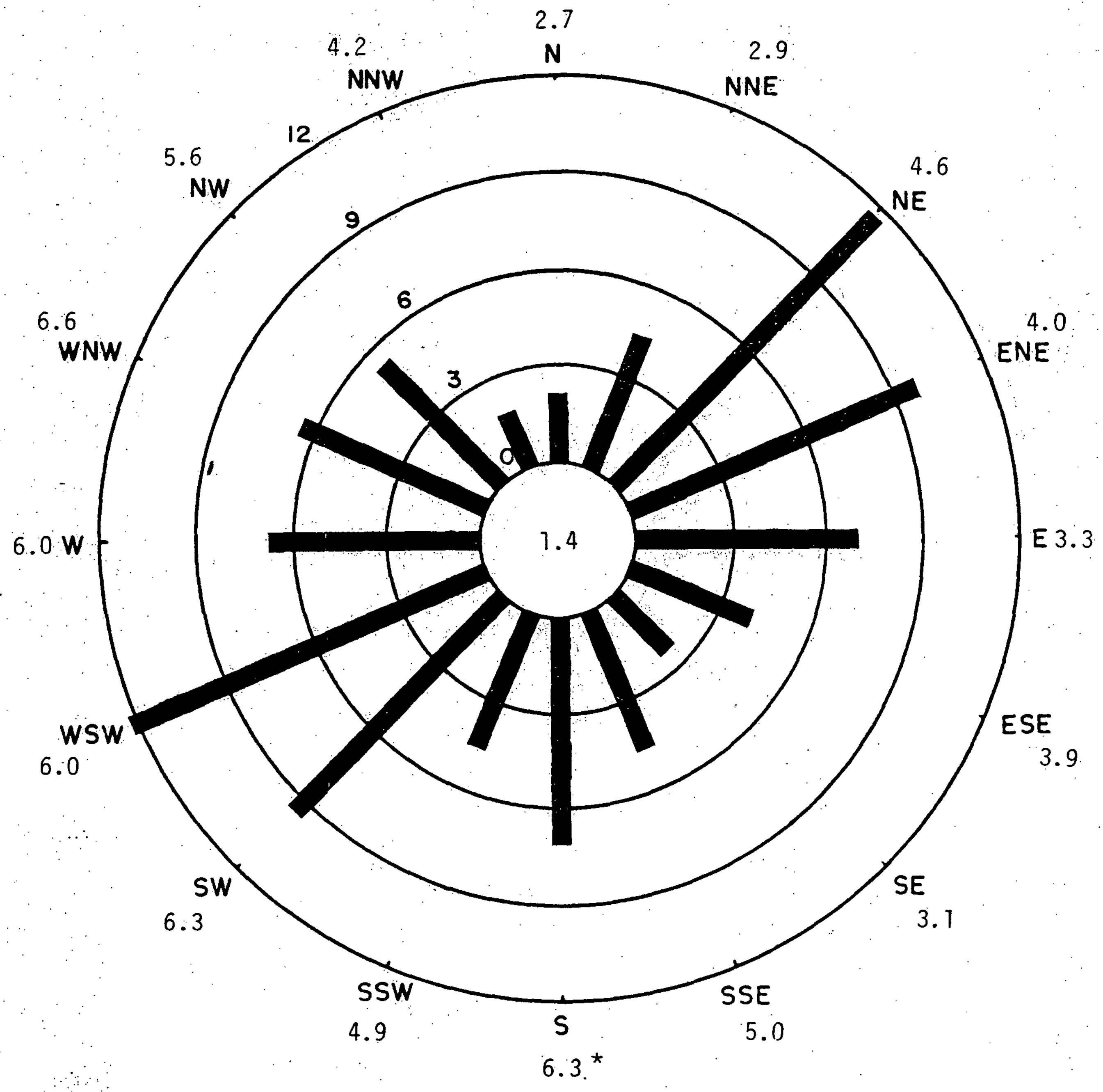

*Value denotes average wind speed for each sector.

Figure 2.3-8 ANNUAL WIND ROSE FOR THE 200-FOOT LEVEL FROM THE CRBRP PERMANENT TOWER METEOROLOGICAL DATA FOR FEB. 17, 1977 THROUGH FEB. 16, 1978




*Value denotes average wind speed for each sector.

Figure 2.3-9 WINTER WIND ROSE FOR THE 200-FOOT LEVEL FROM CRBRP PERMANENT TOWER METEOROLOGICAL DATA

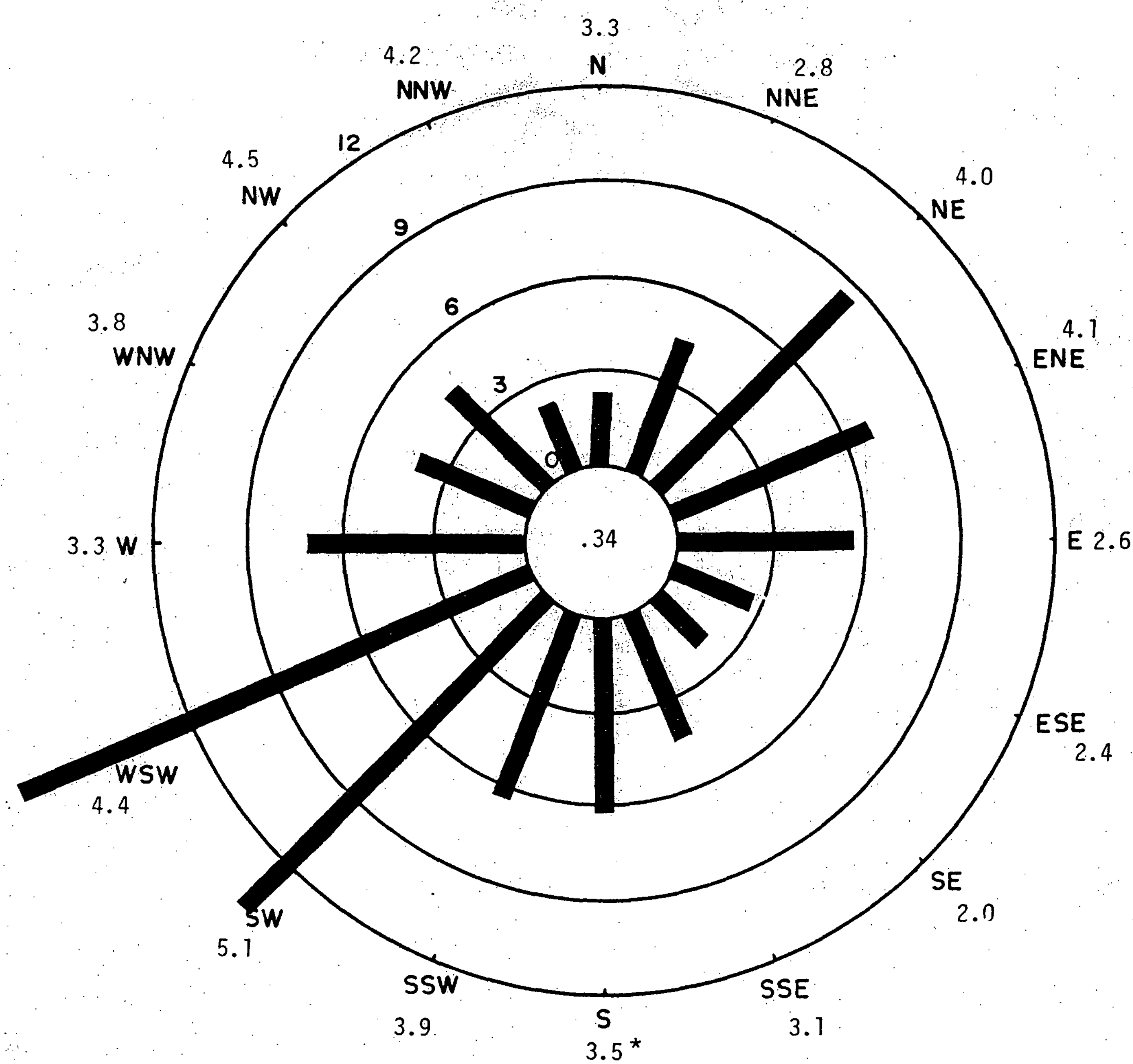



*Value denotes average wind speed for each sector.

Figure 2.3-10 SPRING WIND ROSE FOR THE 200-FOOT LEVEL FROM CRBRP PERMANENT TOWER METEOROLOGICAL DATA

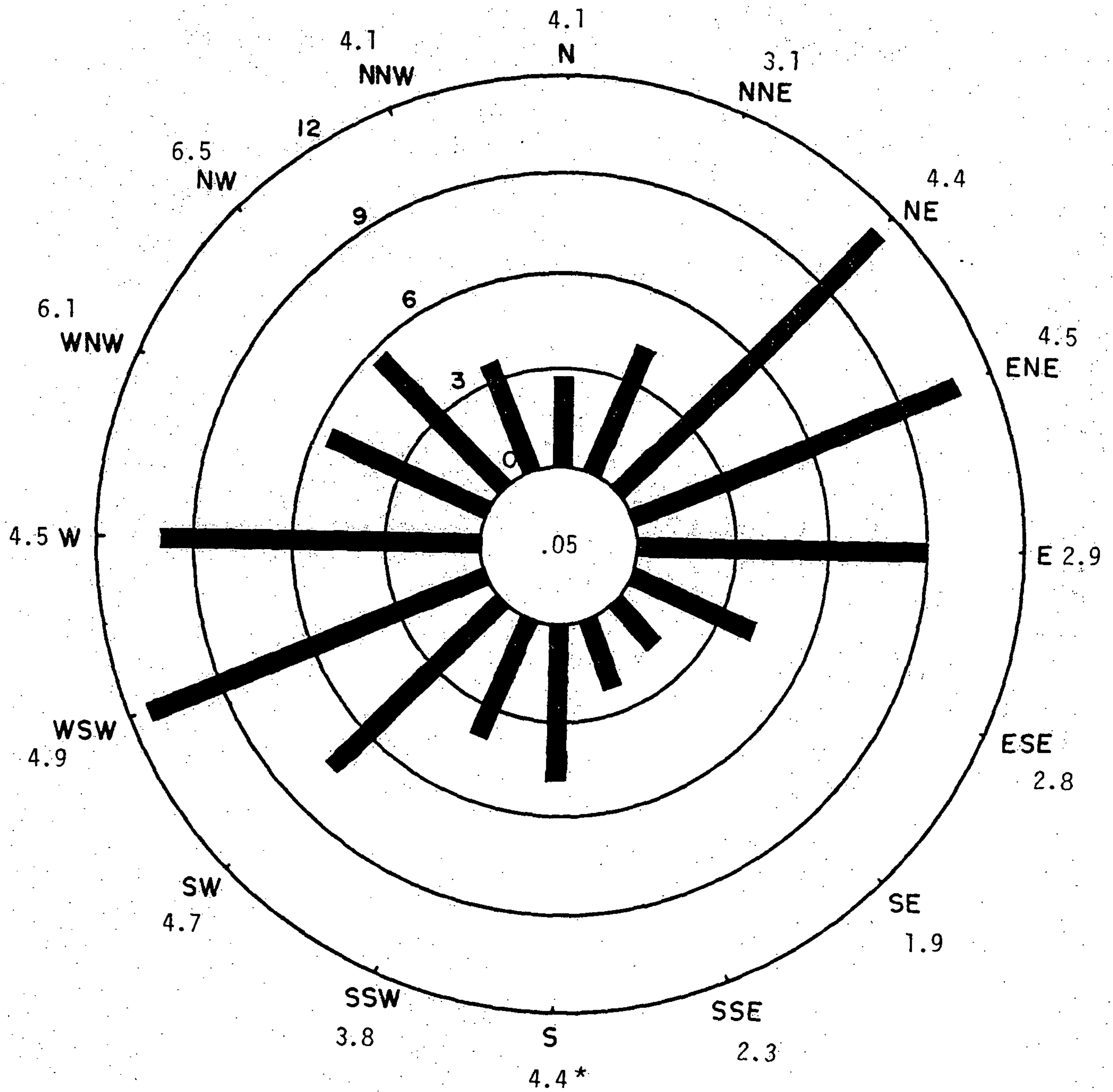


Amend. 65
Feb. 1982




*Value denotes average wind speed for each sector.

Figure 2.3-11 SUMMER WIND ROSE FOR THE 200-FOOT LEVEL FROM CRBRP PERMANENT TOWER METEOROLOGICAL DATA




*Value denotes average wind speed for each sector.

Figure 2.3-12 FALL WIND ROSE FOR THE 200-FOOT LEVEL FROM CRBRP PERMANENT TOWER METEOROLOGICAL DATA

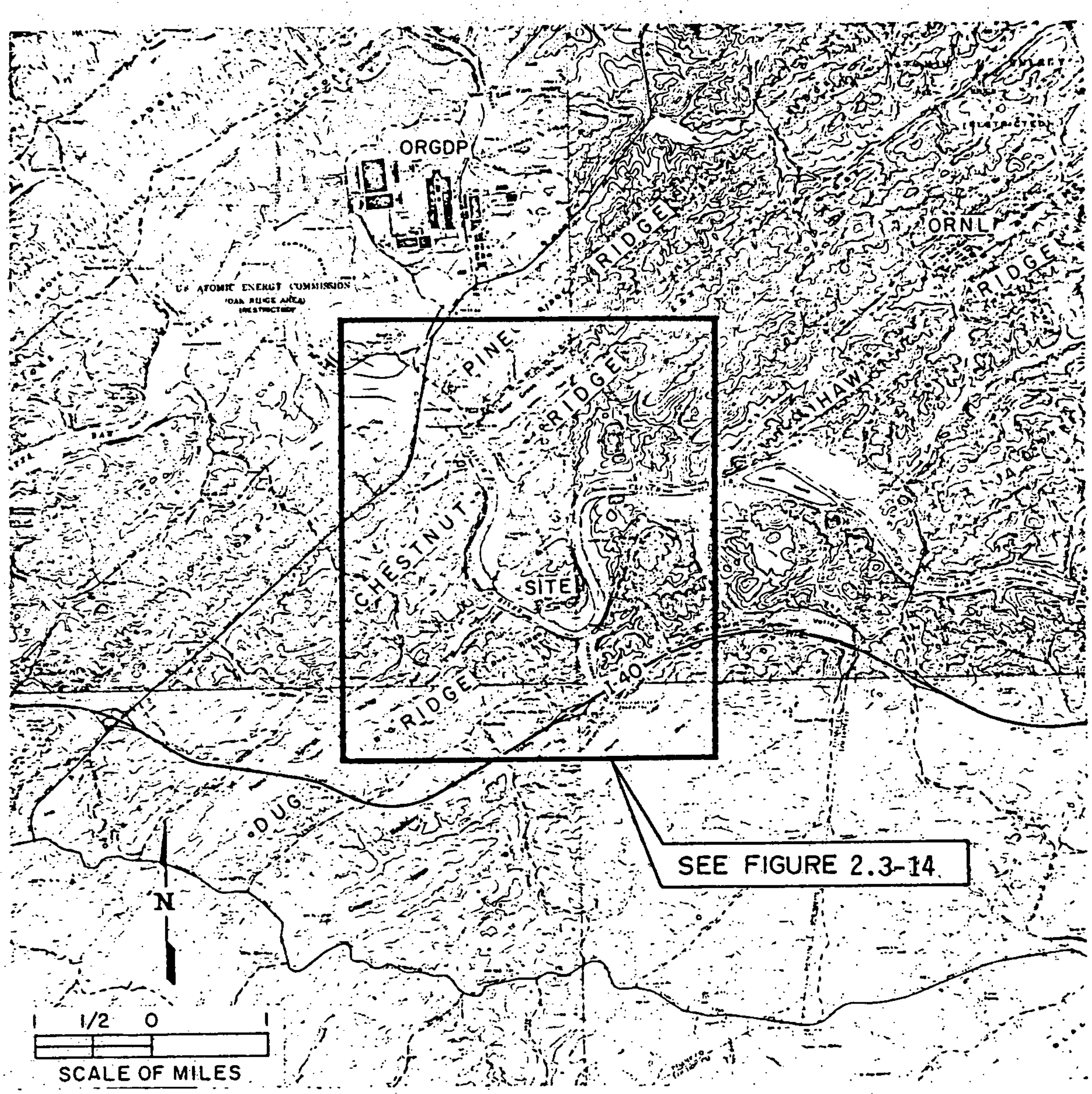



Figure 2.3-13 TOPOGRAPHY SURROUNDING CLINCH RIVER SITE



Amend. 65
Feb. 1982

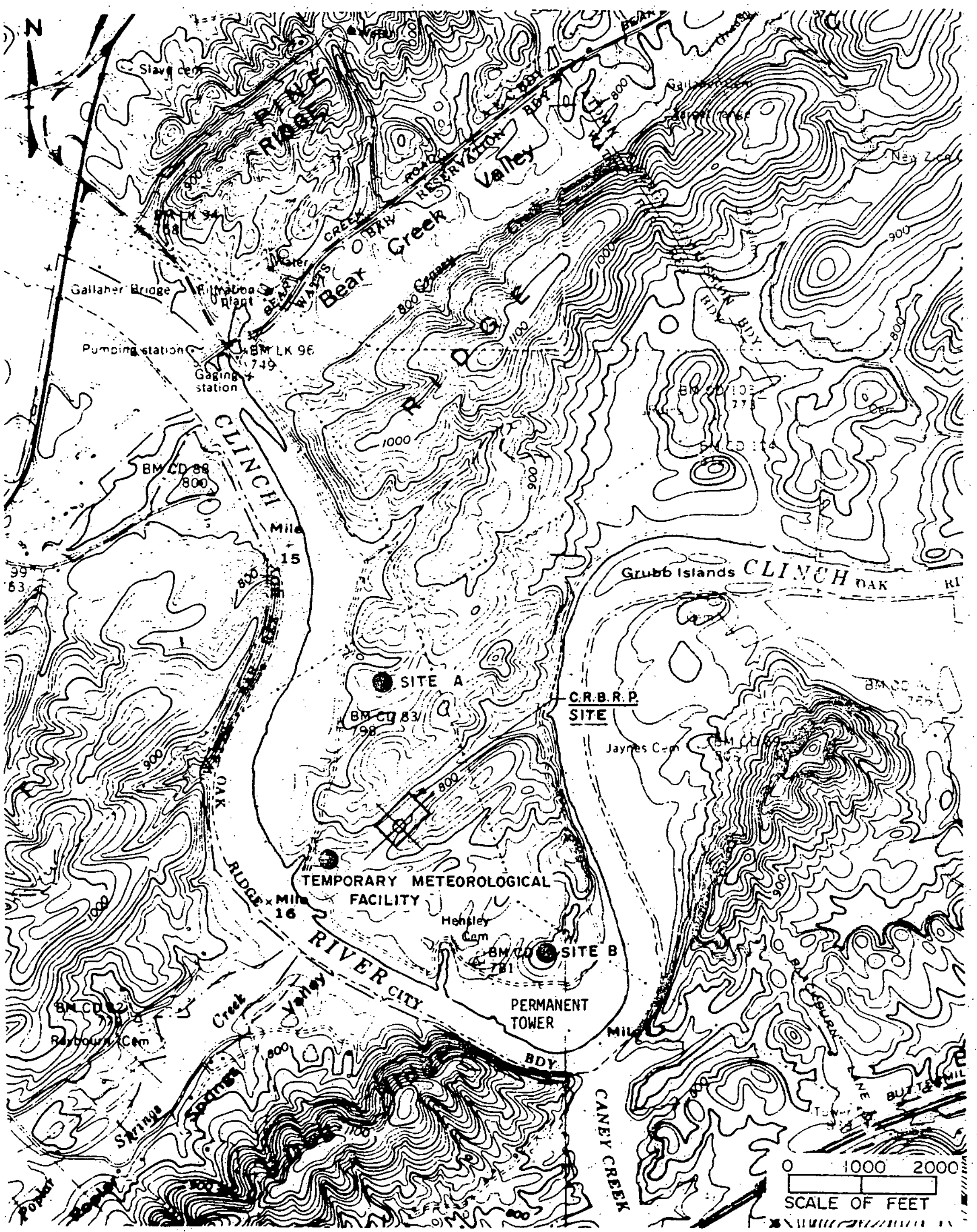



Figure 2.3-14 SITE TOPOGRAPHIC MAP

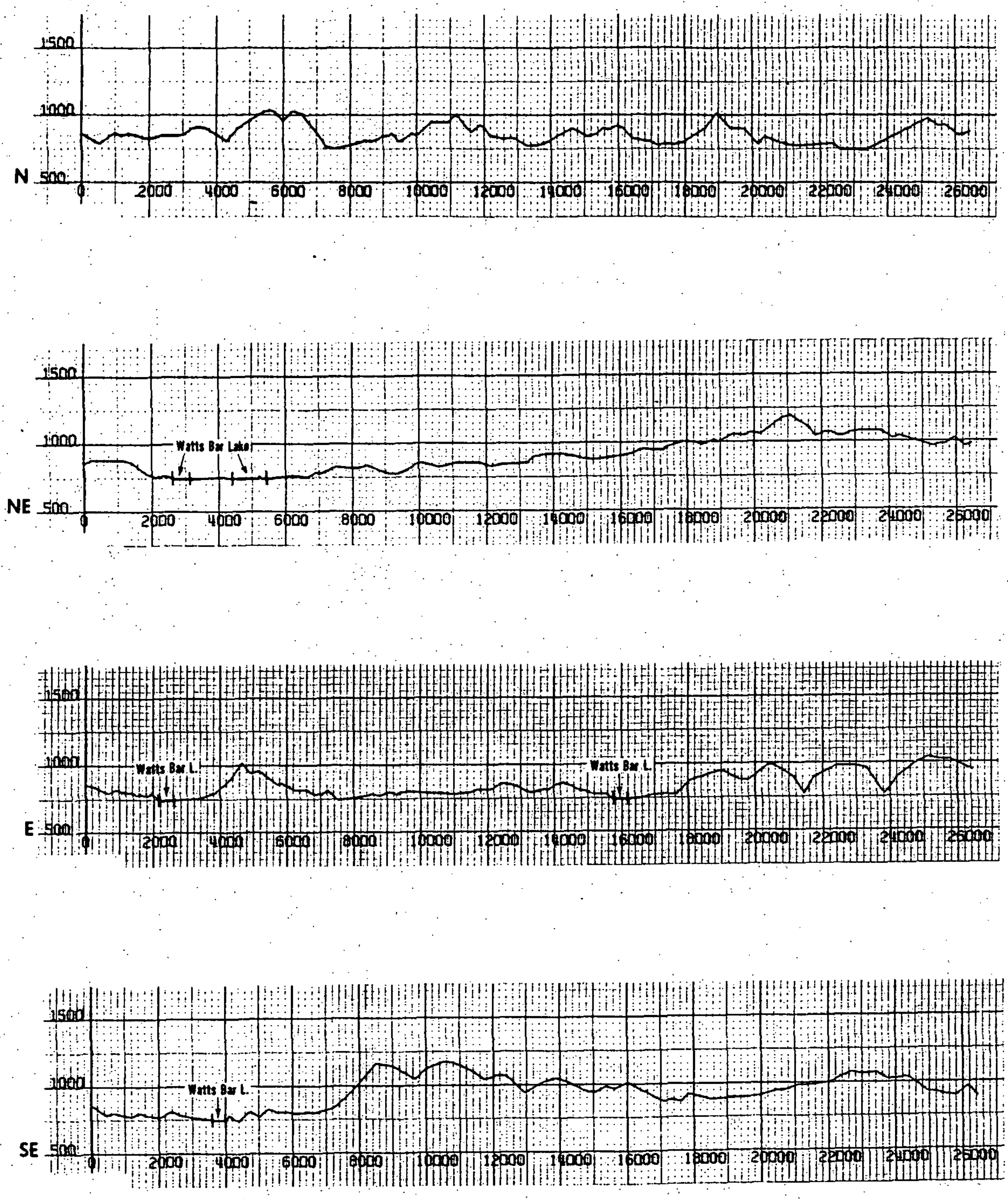



Figure 2.3-15 TOPOGRAPHIC PROFILE CROSS SECTIONS FROM SITE (sheet 1 of 2)

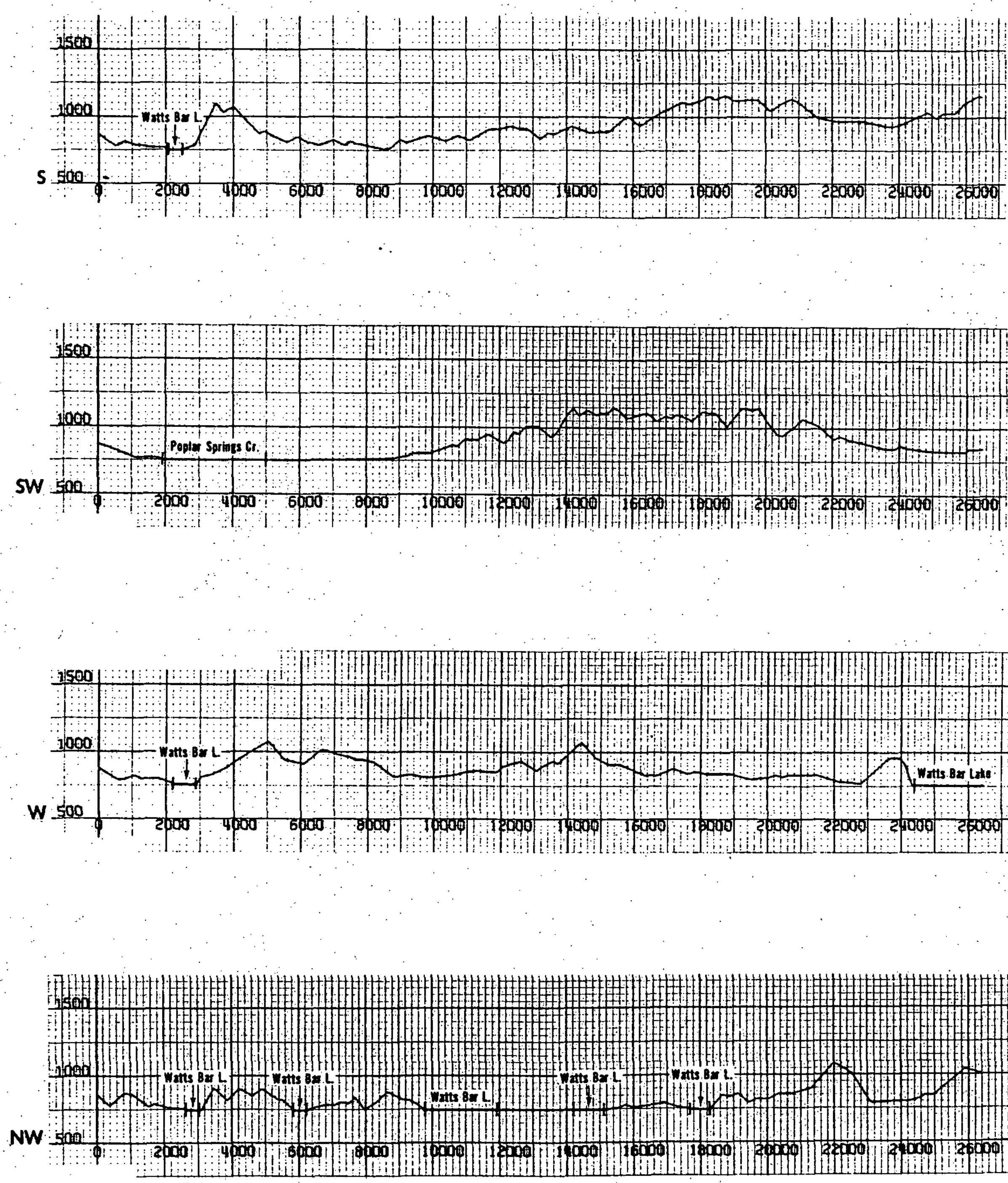



Figure 2.3-15 TOPOGRAPHIC PROFILE CROSS SECTIONS FROM SITE (sheet 2 of 2)

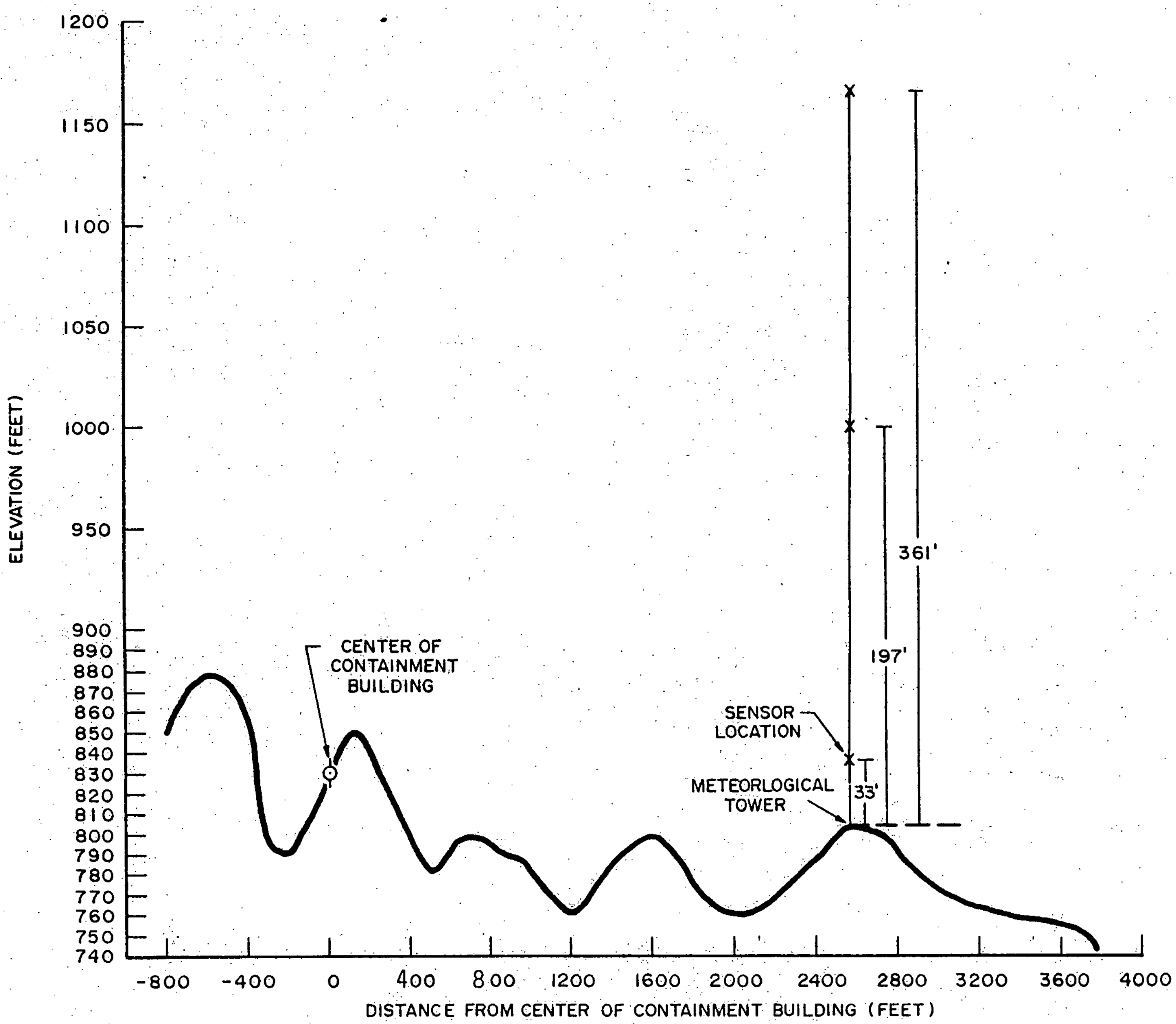



Figure 2.3-16 TOPOGRAPHICAL CROSS SECTION INCLUDING METEOROLOGICAL TOWER AND CENTER OF CONTAINMENT BUILDING