ILLINOIS POWER

Clinton Power Station Individual Plant Examination For External Events Final Report

September 1995

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CLINTON POWER STATION

INDIVIDUAL PLANT EXAMINATION

FOR EXTERNAL EVENTS

FINAL REPORT

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COVER/IPEEE95

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TABLE OF ACRONYMS

ACRONYM	MEANING
AC	Alternating Current (supply or system)
ACRS	Advisory Committee on Reactor Safeguards
ADS	Automatic Depressurization System
ARI	Alternate Rod Insertion
ARSAP	Advanced Reactor Severe Accident Program
ASEP	Accident Sequence Evaluation Program
ATWS	Anticipated Transient Without Scram
BOP	Balance of Plant (non-NSSS systems)
BWR	Boiling Water Reactor
BWROG	Boiling Water Reactor Owners' Group
CA	Condenser Vacuum System
CAFTA	Computer-Aided Fault Tree Analysis Software
CB	Condensate Booster System
CCDP	Conditional Core Damage Probability
CCF	Common Cause Failure
CCI	Core-Concrete Interaction
CD	Condensate System
CDF	Core Damage Frequency
CET	Containment Event Tree
CM	Corrective Maintenance
CPS	Clinton Power Station
CRD	Control Rod Drive System
CS	Containment Spray (mode of RHR)
CSF	Critical Safety Function
CW	Circulating Water System
CWSH	Circulating Water Screen House
CY	Cycled Condensate System
DC	Direct Current (supply or system)
DCH	Direct Containment Heating
DDT	Deflagration to Detonation Transition
DG	Diesel Generator
ECCS	Emergency Core Cooling System(s)
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guidelines
EPRI	Electric Power Research Institute
EQE	Earthquake Engineering International
ERAT	Emergency Reserve Auxiliary Transformer
ESAP	Equipment Seismic Assessment Program
ESW	Extremely Severe Weather (>125 mph)
ET	Event Tree
FEDB	Fire Events Data Base (EPRI-NSAC-178L)
FP	Fire Protection System
FPRA	Fire Probabilistic Risk Assessment
FT	Fault Tree
FTR	Fail to Run
FTS	Fail to Start
FW	Feedwater System
GE	General Electric Company
GESSAR	General Electric Standard Safety Analysis Report
GG	Grand Gulf (Nuclear Station)
GIP	Generic Implementation Plan (Seismic Issue A-46 Resolution)

TABLE OF ACRONYMS

ACRONYM	MEANING
GS	Main Turbine Gland Seal System
GSI	Generic Safety Issue
HCLPF	High Confidence (95%) of Low Probability (5%) of Seismic Fail
HCU	Hydraulic Control Unit for Control Rod Drives
HEP	Human Error Probability
HGL	Hot Gas Layer
HP	High Pressure Core Spray
HPCS	High Pressure Core Spray
HRA	Human Reliability Analysis
HRR	Heat Release Rate
HVAC	Heating, Ventilation, and Air Conditioning
IA	Instrument Air
IAP	Interaction Analysis Program
IDCOR	Industry Degraded Core Rulemaking Effort
IE	Initiating Event
IEEE	Institute of Electrical and Electronic Engineers
IIRT	IPE Independent Review Team
IN	ECCS/RCIC/ARI/DG Initiation Logic
IORV	Inadvertent Open Relief Valve
IPE	Individual Plant Examination
IPEEE	IPE FOR External Events
IPEM	Individual Plant Evaluation Methodology (by IDCOR)
ISLOCA	Interfacing System LOCA
IST	Independent Sub-Tree
LCO	Limiting Conditions for Operation (Technical Specifications)
LLOCA	Large LOCA
LLRT	Local Leak Rate Test
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-Site Power
LP	Low Pressure Core Spray
LPCI	Low Pressure Coolant Injection (Mode of RHR)
LPCS	Low Pressure Core Spray
LTSB	Long-Term Station Blackout
MAAP	Modular Accident Analysis Program (developer - IDCOR, EPRI)
MCR	Main Control Room
MEL	Master Equipment List
MGL	Multiple Greek Letter Common Cause Probability Model
MLOCA	Medium LOCA
MOV	Motor-Operated Valve
MS	Main Steam System
MSCWL	Minimum Steam Cooling Water Level
MSIV	Main Steam Isolation Valve
MSL	Feet above Mean Sea Level
MWth	Mega-Watts, Thermal
NB	Nuclear Boiler System
NPSH	Net Positive Suction Head
NSAC	Nuclear Safety Analysis Center
NSED	Nuclear Station Engineering Department
NSPS	Nuclear System Protection System
NSSS	Nuclear Steam Supply System
OG	Off Gas System
OS	Operational Schematic Drawings

TABLE OF ACRONYMS

ACRONYM	MEANING
OSP	Off-Site Power
PCS	Power Conversion System (BOP)
PDS	Plant Damage State
PM	Preventive Maintenance
PRA	Probabilistic Risk Assessment
PSF	Performance Shaping Factor(s)
PTA	Penetration Thermal Attack
PWR	Pressurized Water Reactor
RAT	Reserve Auxiliary Transformer
RC	Reactor Coolant
RCA	Radiologically Controlled Area
RCIC	Reactor Core Isolation Cooling System
RD	Control Rod Drive System
RFP	Recovery Failure Probability
RH	Residual Heat Removal System
RHR	Residual Heat Removal System
RI	Reactor Core Isolation Cooling System
RPS	Reactor Protection System
RPT	Reactor Recirculation Pump Trip
RPV	Reactor Pressure Vessel
RSP	Remote Shut-Down Panel
RT	Reactor Water Cleanup System
RWCU	Reactor Water Cleanup System
S&L	Sargent & Lundy (Plant Designer)
SA	Service Air System
SAIC	Science Applications International Corporation
SBO	Station Black Out
SC	Standby Liquid Control System
SCRAM	Safety Control Rod Axe Man (Rapid Reactor Shut Down)
SE	System Engineer
SETS	Set Equation Transform System Software for Solving PRAs
SEWS	Screening and Evaluation Work Sheet (Seismic Walkdown)
SJAE	Steam Jet Air Ejector
SLC	Standby Liquid Control System
SLOCA	Small LOCA
SMA	Seismic Margins Assessment
SME	Seismic Margin Earthquake
SMRT	Senior Management Review Team
SPC	Suppression Pool Cooling (mode of RHR)
SPMU	Suppression Pool Make-Up
SQUG	Seismic Qualification Utility Group
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRT	Seismic Review Team
SRV	Main Steam Safety Relief Valve
SSE	Safe Shutdown Earthquake
SSI	Soil-Structure Interaction
SSPR	Safety System Performance Review
STA	Shift Technical Advisor
STSB	Short-Term Station Blackout
SX	Shutdown Service Water System
TBCCW	Turbine Building Closed Cooling Water

TABLE OF ACRONYMS

ACRONYM	MEANING
TOAF	Top of Active Fuel (Reactor Water Level)
TRS	Test Response Spectrum
UAT	Unit Auxiliary Transformer
UHC	Ultimate Hydrogen Concentration
UHS	Ultimate Heat Sink
USAR	Updated Safety Analysis Report
USI	Unresolved Safety Issue
VC	Main Control Room Ventilation
VD	Diesel-Generator Room Cooling
VX	Essential Switchgear Cooling
VY	Emergency Core Cooling System Room Cooling
WS	Plant Service Water

1. EXECUTIVE SUMMARY

This report describes the Individual Plant Examination for External Events (IPEEE) performed for Clinton Power Station (CPS). External events are those that could potentially lead to core damaging accidents but typically originate outside the systems of a nuclear power plant, such as fires, earthquakes, floods, high winds and transportation accidents. The IPEEE is an adjunct to the original Individual Plant Examination (IPE) which examined the risks associated with internal events (typically initiated by equipment failures).

1.1 Background and Objectives

The purposes of the IPEEE are to develop an understanding of how external events can contribute to the risk of a core damaging accident and to determine cost effective safety improvements, if appropriate, to reduce this risk.

The internal events IPE was requested by NRC Generic Letter 88-20 issued in November of 1988 (reference 1-1). The results of the CPS Individual Plant Examination were submitted to the NRC in September 1992 (reference 1-2).

Generic Letter 88-20, Supplement 4, (reference 1-3), issued in June of 1991, requested performance of the IPEEE. This report documents the results of the IPEEE study for CPS.

1.2 Plant Familiarization

Clinton Power Station is a 2894 Mw-thermal General Electric BWR 6 with a Mark III containment. It is located in Central Illinois on Clinton Lake, which provides the cooling water supply for plant equipment under both normal and emergency conditions. CPS received its operating license in 1986. Because CPS is of relatively recent vintage for US nuclear plants, it received extensive reviews during design and construction. The CPS design included seismic qualification of safety related equipment to a safe shutdown earthquake level of 0.25g. Good divisional separation of plant components makes it unlikely that problems in one area (e.g. fire or floods) can cause loss of multiple divisions of equipment. More details regarding the CPS design are contained in section 2.4.1.

1.3 Overall Methodology

The IPEEE program for the Clinton Power Station was conducted with methods that were approved by the NRC for responding to Generic Letter 88-20, Supplement 4, as follows.

- Seismic Events - Evaluated with the Electric Power Reseach Institute (EPRI) Seismic Margins Assessment (SMA) method as described in EPRI report NP-6041 (reference 1-4). In this method two groups of plant equipment, each capable of bringing the plant to safe shutdown conditions, are selected. These safe shutdown paths are then evaluated using screening methods and other analyses as required to demonstrate that they are capable of functioning after the review level earthquake specified by the NRC for the Clinton site (0.3g).

- Fire Events - Evaluated with the Fire Probabilistic Risk Assessment (PRA) method described in EPRI report 3385-01 (reference 1-5). This method calculates a core damage frequency associated with fire events. The Fire PRA takes into account the likelihood of fire in individual locations of the plant along with the potential risk from damage to equipment by a fire in that location.

- Other Hazards - Tornadoes, high winds, external flooding, nearby facility accidents and transportation accidents were re-evaluated against the Standard Review Plan requirements regarding these hazards.

1.4 Summary of Major Findings

The overall conclusion reached from this study is that CPS has strong capability to withstand the effects of external events. This capability is due in large part to the conservative design and operating philosophy used at CPS.

The Seismic Margins Assessment demonstrated that CPS is capable of attaining safe shutdown conditions after the review level earthquake anchored at 0.3g. The SMA did not find any potential vulnerabilities in the safe shutdown components, systems and structures in the two safe shutdown paths selected. No plant improvements in this area are needed.

The estimated core damage frequency from internal fires is 3.3E-6 events per year, which is less than the core damage frequency due to internal events of 6.0E-6 events per year. This internal events core damage frequency is the current best estimate based on the CPS "PRA Update Report" (reference 1-6), which includes improvements to the plant model from the original IPE submittal. The core damage frequency estimated for internal fires is based upon plant improvements that will be made as a result of the CPS response to the Thermo-Lag 330-1 fire barrier issue. These improvements will be made in Refueling Outage 6 currently scheduled for the fall of 1996. Section 7.2 contains more detailed information. No additional improvements are being proposed as a result of the fire PRA.

The review of other IPEEE hazards, including high winds, tornadoes, external flooding, nearby facility accidents and transportation accidents, results in the conclusion that CPS meets the Standard Review Plan requirements regarding these hazards and therefore the risk from such events is acceptably low.

1.5 References For Chapter One

1-1. NRC Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities", November 23, 1988.

1-2. Illinois Power Company Letter Y-602040, "Clinton Power Station Individual Plant Examination Final Report", September 23, 1992.

1-3. NRC Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", June 28, 1991.

1-4. Electric Power Research Institute report NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1)", August, 1991.

1-5. Electric Power Research Institute Report Project 3385-01, "Fire Risk Analysis Implementation Guide", Draft Report, January 1994.

1-6. Illinois Power Letter Y-104709, "PRA Update Report", January 31, 1995.

2. EXAMINATION DESCRIPTION

2.1 Introduction

This section describes how the IPEEE analysis was performed in order to ensure that the objectives of NRC Generic Letter 88-20, Supplement 4, were met. In addition to compliance with the Generic Letter, the IPEEE was developed to provide a decision optimization tool that can be used to aid in achieving corporate goals related to the continuation and enhancement of the safe, reliable, and efficient operation of the plant.

2.2 Conformance with Generic Letter and Supporting Material

The program objectives for the CPS IPEEE were as follows:

1) Develop an overall appreciation of severe accident behavior,

2) Understand the most likely severe accident sequences that could occur at the Clinton Power Station,

3) Gain a qualitative understanding of the overall likelihood of core damage and radioactive material releases, and

4) If indicated, reduce the overall likelihood of core damage and radioactive material release by appropriate modification to hardware and procedures.

The knowledge gained during the course of the IPEEE study can be factored into future risk studies and the Severe Accident Management Program.

To accomplish the IPEEE program objectives, the analysis methods discussed in section 2.3 were employed in accordance with Generic Letter 88-20, Supplement 4. The evaluation was performed and controlled by a team of Illinois Power Company engineers who are intimately familiar with CPS and were the primary authors of the CPS IPE. An independent, in-house review was performed at several key stages of the process. Review and technical advice were supplied, as necessary, by consultants. Specific information on the team makeup, structure, and experience level and the review processes is included in sections 6.1 and 6.2.

This submittal is formatted in accordance with the guidance of NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities".

2.3 General Methodology

The methods employed for the CPS IPEEE conform to the guidance provided in Generic Letter 88-20 and NUREG-1407.

The following paragraphs highlight the main topics of the methodology used to perform the CPS evaluation.

2.3.1 Seismic Margins Assessment

The seismic methodology applied to Clinton Power Station (CPS) is the EPRI-developed seismic margin methodology, with the enhancements specified in Generic Letter 88-20, Supplement 4, for a plant that was binned at the focused-scope level of review. Although subsequent changes were promulgated by the NRC for both binning and the review process, CPS chose to remain with the original focused-scope requirements. This decision was primarily based on timing and is considered conservative with respect to the later published requirements for the Seismic Margins Assessment (SMA) approach.

The SMA methodology is detailed in EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" (Reference 2-1). This approach is deterministic and applies success path modeling rather than fault tree and event tree modeling.

Two safe shutdown success paths, designated as the primary and alternate, were identified along with the components required for their operation. Each success path is a group of plant systems that is capable of bringing the reactor to a stable condition, either hot or cold shutdown, and maintaining that condition for at least 72 hours after the earthquake. The structures, systems and components used in the success paths were selected in accordance with guidance provided by EPRI. The SMA demonstrates the operability and survivability of the structures and the components within the two success paths.

The success paths were evaluated by the Seismic Review Team (SRT), who performed walkdowns of the structures and components. The SRT is described in section 3.1.1.d. The seismic margin walkdowns were used to search for weak links and to determine where more detailed evaluations needed to be performed. The SRT either screened-out structures and components from further evaluation based upon their known ruggedness or evaluated them to assure adequate seismic margin was present. Screening and evaluation guidelines and walkdown documentation forms contained in EPRI-6041-SL were used during plant walkdowns.

Post-walkdown analyses and documentation of the evaluations were then completed, along with peer review and comment resolution. The peer reviewers were the IPEEE Independent Review Team (IIRT) and Individual Plant Evaluation Partnership (IPEP) which included representatives from EQE International. These review teams are more fully discussed in section 6.2. The CPS SMA was controlled and conducted by Illinois Power Company engineers who are located at the plant site and are involved in the day-to-day activities of the plant. Consultants were used in selected areas of the evaluation to provide specific expertise and to transfer technology to the IP team members. Throughout the project, emphasis was placed on enhancing the knowledge base of the utility staff members such that their internal capabilities could enable them to deal with potential future evaluations and updates.

2.3.2 Fire PRA

The internal fire risk analysis was performed using the Fire Probabilistic Risk Assessment (FPRA) methodology detailed in EPRI Report Project 3385-01, January 31, 1994 (reference 2-2). This methodology was based on the EPRI Fire Induced Vulnerability Evaluation (FIVE) methodology.

The first task in the FPRA methodology was to perform a screening analysis for every firezone in the plant. This screening analysis had two major steps. The first step required the identification of equipment and supporting electrical cables located within each firezone. The equipment identified was that which was previously modeled in the CPS internal events PRA and included both safety and non-safety related components. Using this equipment list, the basic events in the PRA model for a particular firezone were identified. The impact on the plant of the loss of noted equipment was determined and initiating events from the PRA model that could occur as a result of this loss were also identified. Using the resulting list of initiating and basic events, the PRA model was solved and a conditional core damage probability (CCDP) was determined for each firezone. This CCDP is the probability of core damage occurring as a result of a fire that disables all equipment and cables in a particular firezone.

The second step of the screening analysis determined the frequency at which fires occur in each firezone. This task required the identification of all equipment in the plant with the potential to start a fire. The FPRA methodology detailed what type of components to consider as well as the fire frequency on both a component type and firezone type basis. This ignition frequency was multiplied by the CCDP to determine a screening core damage frequency (CDF) for each zone. Firezones with screening values below the screening threshold were eliminated from further analysis.

Firezones which were not eliminated by the screening process underwent fire modeling for both fixed and transient ignition sources. Fixed sources, such as a pump, are permanently installed, therefore their locations are well defined for fire modeling. Transient sources, such as a welding torch, can be located essentially anywhere within a firezone, and therefore, they are analyzed in all possible locations. Fire modeling develops a set of ignition source-target set combinations for each ignition source in a firezone. Calculations based on heat release rate, heat loss fraction, firezone ambient temperature, radiant energy flux and target distance are performed to determine if a target can be damaged for a particular ignition source. The damaged equipment in a target set is used to develop PRA model inputs and generate a CCDP. The CCDP is multiplied by the ignition source ignition frequency to determine a CDF. The summation of all source-target set CDFs is the CDF for that particular firezone. A modifier for automatic or manual suppression of the fire can also be applied in the modeling process.

The analysis of the control room differs from the fire modeling analysis just described. Fire modeling typically addresses damage to overhead cables from a plume or ceiling jet or room heatup by a hot gas layer (HGL). Control rooms, however, typically have cables enclosed in cabinets. Additionally, the people manning the controls are important "targets".

In control room fire analysis, fires begin in electrical cabinets and their effects on controls within the cabinet and in adjacent cabinets need to be evaluated. People manning controls must see them, so smoke rather than temperature in the hot gas layer is evaluated to determine if and when evacuation could occur.

While fire modeling tools were different, the analysis of the control room nevertheless followed a process similar to the rest of the FPRA. Boundaries for fire spread in cabinets and the equipment within those boundaries were identified. CCDPs and ignition frequencies were calculated. The resulting cabinet CDFs were ranked and more detailed analysis of fire spread within the cabinets was performed for the most significant cabinets. Finally, smoke effects were analyzed for their potential to cause evacuation of the control room.

The final step of the FPRA was to perform a multi-compartment analysis to determine the potential for a fire in one firezone propagating to a second zone. This analysis was performed for all firezones in the plant. Since propagation between firezones requires the formation of an HGL, all firezones were examined to determine if the potential for HGL formation existed. Firezones without HGL formation potential were eliminated from further consideration, as were firezones that could form an HGL but could not generate enough heat to sustain one when combined with adjacent firezones. The remaining firezones were analyzed using a method similar to the screening analysis; however, in this case, model inputs were generated using the equipment lists for both an exposing and an adjacent firezone. Modifiers were applied both for the use of fire suppression systems and probability of the barrier between firezones failing.

The results from the fire modeling analysis, main control room analysis and multi-compartment analyses are combined to determine the CDF from internal fires. A more detailed description of this analysis is provided in chapter 4.

2.3.3 Other Hazards Analysis

The Other Hazards Analysis examined high winds, tornadoes, flooding, transportation and nearby facility accidents to confirm that CPS continues to meet the 1975 (or later) Standard Review Plan (reference 2-3) requirements with regard to these hazards. The approach used in these studies was to review the original licensing basis against current conditions. Differences in the nature of the hazard or in the ability of the plant to cope with the hazard from the licensing basis were noted. These differences were evaluated in light of the Standard Review Plan criteria. Walkdowns were used to augment this study. The method used for evaluating the Other Hazards is in accordance with NUREG 1407.

For the high winds, tornadoes, and flooding analyses, the Standard Review Plan was reviewed to determine the current design requirements. The plant's existing design was compared to these more recent standards to determine whether the plant has adequate protection from these hazards. Walkdowns were conducted to observe the material condition of the wind and flooding barriers.

For the transportation and nearby facility accidents analyses, recent data regarding the types and quantities of hazardous materials stored at facilities in the area were reviewed. Hazardous materials stored at facilities existing within five miles of CPS were evaluated for facility accidents. Hazardous materials stored at facilities in DeWitt County were screened for the potential to be a transportation hazard using the approach discussed in section 5.3.2. Materials which were not eliminated by the above screening were subject to further analysis, which typically involved determining whether their shipping routes passed near the site. Probability arguments were used in some instances to show a low risk significance from these hazards.

Aircraft hazards were evaluated by obtaining recent air traffic data for nearby airports and flyways. This data was used, with the methods described in the USAR, to calculate a new frequency for airplane crashes into CPS.

2.4 Information Assembly

The following sections discuss some of the sources of information which were factored into the IPEEE analysis. The combination of plant design & operational information, reference documents, walkdowns, second tier documentation, comment resolutions and the collective experience of the IPEEE team form the basis for the IPEEE.

2.4.1 Plant Layout

Clinton is a Boiling Water Reactor (BWR) rated at 2894 megawatts-thermal (MWt). It is a BWR 6 with a Mark III containment. Some of the major plant features include the following:

- Inventory Make-up Systems

  - 4 motor driven low pressure ECCS trains (LPCS & LPCI) rated at approximately 5000 gpm each.

  - 1 motor driven high pressure ECCS train (HPCS) rated at approximately 5000 gpm.

  - 1 steam driven high pressure system (RCIC) rated at approximately 600 gpm.

    (The above listed systems are each located in their own separate room which provides protection from flooding sources external to the room. Their support systems (power and cooling) are separated into three safety-related divisions. This arrangement makes it unlikely that problems in one area of the plant can affect many other systems.)

  - Feedwater delivery system consisting of 2 turbine driven and 1 motor driven feedwater pumps with 4 motor driven condensate pumps and 4 motor driven condensate booster pumps.

- Main Steam System

  - 16 Safety Relief valves, 7 of which are Automatic Depressurization System (ADS) Valves. The ADS system is separated into two safety-related divisions.

  - 35% turbine bypass capability.

  - Two main steam isolation valves (MSIVs) on each of the four main steam lines.

- Electric Power Systems

  - 4 off-site power circuits (3 lines at 345 kv through the switchyard and 1 line at 138 kv bypassing the switchyard).

  - 3 emergency, safety-related AC buses with good divisional separation.

  - 3 divisional standby diesel generators.

  - 4 divisions of safety-related batteries.

- 2 non-safety-related batteries.

- 4 hour battery life for the safety-related batteries (with load shedding).

° Cooling Water Systems

- The Plant Service Water system supplies cooling water from Clinton Lake to balance of plant equipment and is the normal supply for safety-related equipment. It has 3 pumps supplied by balance of plant power supplies.

- The Shutdown Service Water system provides cooling water from Clinton Lake to safety-related equipment under abnormal conditions. This system has three pumps which correspond to the three divisions of safety-related equipment. These pumps are powered from the corresponding divisional power supplies. Each pump is located in a separate room which serves as a flood barrier.

° CPS Mark III Containment

- Steel-lined reinforced concrete containment, with a volume of 1,550,000 ft^3.

- Drywell structure with a volume of 246,500 ft^3 enclosed by the containment.

- Suppression pool with a volume of 135,700 ft^3, which communicates between the drywell and containment.

- 2 trains of containment spray, suppression pool cooling or shutdown heat removal.

- A reinforced concrete basemat approximately 10 feet in depth.

2.4.2 Reference Documentation

Documents used during the course of the IPEEE study are listed in the reference section at the end of each chapter. Because CPS is a relatively new U.S. nuclear plant, good information regarding the CPS design is available. The information sources listed in Table 2.1, regarding CPS design and operation, were used throughout the IPEEE study.

TABLE 2.1

CPS DESIGN AND OPERATING INFORMATION USED DURING

THE IPEEE STUDY

DOCUMENT	INFORMATION
Clinton Drawings Piping and Instrument Drawings Electrical Drawings Mechanical Drawings Structural Drawings Vendor Drawings	System Components System Interconnections and Plant Layout
Master Equipment List	Instrument and Equipment Hardware Characteristics
Updated Safety Analysis Report	Previous Analysis Regarding External Hazards
Procedures Normal Off-Normal Emergency Maintenance	System Operations, Maintenance Activities, Operator Actions, and Plant Information
Licensee Event Reports	Operating History
PRA Update Report	Plant PRA Model Including Fault Trees and Event Trees

2.4.3 Walkdowns

Plant walkdowns were performed for the IPEEE to verify system information accuracy, identify special or unusual characteristics of individual components or their locations, identify potential recovery actions and examine the condition of plant features that protect against external hazards. The fire walkdowns were primarily concerned with fire ignition sources, combustibles, fire barriers and potential fire damage targets. The seismic walkdowns examined components on the safe-shutdown equipment list (SSEL) to identify equipment configurations or potential interactions that require further analysis to confirm their seismic adequacy. Walkdowns were performed to confirm that plant features which protect against floods and tornado missiles are in place.

Observations and insights obtained during the walkdowns were documented. The IPEEE team, located at the plant site, performed additional walkdowns as necessary to answer specific questions as they arose. Details concerning the IPEEE walkdowns for Seismic, Fire and Other Hazards analyses are discussed in sections 3, 4 and 5 respectively.

2.4.4 Documentation

In order to capture the thought processes, methods and results as the study progressed, reports were developed during the different stages of the study. These reports are referred to as supporting documentation and include the following:

1) Selection of Safe Shutdown Paths,
2) Seismic Walkdown Report,
3) Review of Soils Issues in Support of IPEEE Studies for the Clinton Power Station (prepared by EQE),
4) Clinton Power Station Seismic Margin Assessment for IPEEE Seismic Response Motion Comparison (prepared by EQE),
5) Clinton Power Station IPEEE Internal Fire Analysis Supporting Documentation, and
6) Clinton Power Station IPEEE Other Hazards Analysis Supporting Documentation.

All reports have been reviewed for accuracy and completeness. Reports prepared by Illinois Power have been reviewed by the IPEEE review teams as described in section 6. The above listed reports form part of the second tier of documentation and could be used for future applications and updates.

Information from these reports has been directly used in development of this submittal.

2.5 References for Chapter 2

2-1. EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin", Revision 1, August 1991.

2-2. Electric Power Research Institute Report Project 3385-01 "Fire Risk Analysis Implementation Guide", Draft Report, January 1994.

2-3. NUREG-0800 "Standard Review Plan", various dates.

3. SEISMIC ANALYSIS

3.0 Seismic Methodology Selection

The Seismic Margin Assessment (SMA) was performed in response to the guidelines contained in Generic Letter 88-20 Supplement 4 and NUREG 1407. The SMA methodology, as defined in EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin" [Ref. 3-1], enabled development of a practical method of assessing nuclear plant seismic margins.

For CPS, the Seismic Margin Earthquake (SME) assigned by the NRC is the median NUREG/CR-0098 [Ref. 3-16] spectrum anchored at 0.3g. The CPS design Safe Shutdown Earthquake (SSE) is based on Regulatory Guide 1.60 [Ref. 3-17] response spectra anchored at 0.25g. The difference between the SSE and the SME is the "margin" being assessed in the SMA program. It is not the intent of this program to determine the absolute largest earthquake the site could withstand, only to demonstrate with a high degree of confidence that the site could withstand an SME.

The following sections detail the methodology and results of the analysis.

3.1 Seismic Margins Method

The SMA consisted of several steps:

- assemble and train a Seismic Review Team (SRT) for performing the analysis and walkdown,
- select safe shutdown paths and component lists,
- evaluate relay chatter,
- perform soil-structure interaction (SSI) analysis,
- assess seismic margin of soils,
- prepare walkdown information,
- perform seismic capability walkdowns,

- evaluate equipment and structural capacity, and
- document the evaluation.

The seismic margin analysis guidelines help focus on the identified success paths instead of reanalyzing the entire plant. These success paths can achieve and maintain a safe shutdown condition for a minimum of a 72 hour period following an SME. Components, equipment, and structures in the systems involved are reviewed and evaluated. The safe shutdown paths and component list were compiled along EPRI developed guidelines and formed the basis for the plant walkdowns. The reactor vessel internals were not evaluated in this analysis.

The safe shutdown path walkdown methodology used at CPS has been adapted from industry work regarding Unresolved Safety Issue (USI) A-46, "Seismic Qualification of Equipment in Operating Plants". Technical research by the Seismic Qualification Utility Group (SQUG) and the NRC resulted in an approach for A-46 resolution called the Generic Implementation Procedure (GIP) [Ref. 3-2].

CPS has not been required to perform an A-46 review since the plant was constructed after implementation of the IEEE 344-1975 standard for qualification of Class I electrical equipment. However, the GIP provides the detailed technical approach, generic procedures, and documentation guidance which can be used to verify the seismic adequacy of mechanical and electrical equipment used in the safe shutdown paths of the plant.

Relay chatter was evaluated using the guidance from NUREG-1407 to locate and evaluate low-seismic ruggedness relays.

Soils issues, including soil-structure interaction were analyzed for instability, settlement, and liquefaction using the guidance in EPRI NP-6041.

All aspects of the IPEEE study were performed under the direction and control of utility personnel. Outside specialists were used for expertise in certain areas.

The following sections detail the process further.

3.1.1 Review of Plant Information, Screening, and Walkdown

A safe shutdown path is a string of systems used to accomplish each of the four following safe shutdown functions:

1) reactor reactivity control,
2) reactor coolant pressure control,
3) reactor coolant inventory control, and
4) decay heat removal.

Because of the redundancy and diversity in the design of nuclear power plants, there may be several paths which could be used to accomplish the four functions. Two paths are chosen for the purpose of this analysis, a preferred and an alternate path. Only the equipment in a preferred path and alternate path need be identified as "success paths" for SMA.

The alternate path is selected to include equipment or a backup train of equipment so that the plant can be shut down in the event of an active failure or unavailability of a single item of equipment in the preferred path.

The SMA methodology defines those systems required to bring the plant to a stable condition (either hot or cold shutdown) and maintain that condition for at least 72 hours. There is no requirement on which path leads to hot or cold shutdown condition. Table 3.1 lists the front-line and support systems used in the success paths.

Table 3.1

FRONT LINE SYSTEMS FOR SAFE SHUTDOWN OF THE REACTOR

Function	Preferred Path	Alternate Path
Reactivity Control	CRD	CRD
Pressure Control	SRV div 1	ADS (SRV) div 2
Inventory Control	RCIC	LPCI C
Decay Heat Removal	RHR A in SPC (Hot Shutdown)	RHR B in SDC (Cold Shutdown)

SUPPORT SYSTEMS FOR SAFE SHUTDOWN OF THE REACTOR

MCR HVAC (VC)	Division 1	Division 2
ECCS Room Cooling (VY)	Division 1	Division 2
SX room HVAC (VH)	Division 1	Division 2
DG room HVAC (VD)	Division 1	Division 2
SWGR Heat Removal (VX)	Division 1	Division 2
Shutdown Service Water (SX)	Division 1	Division 2
Diesel (DG/DO)	Division 1	Division 2
AC Power	Division 1	Division 2
DC Power	Division 1	Division 2

The definition of shutdown is the reactor mode switch in "Shutdown" position, with the reactor subcritical. An average reactor coolant .RC) temperature greater than 200 degrees Fahrenheit is "hot"; average RC temperature equal to or less than 200 degrees Fahrenheit is "cold".

The first step in the selection of the shutdown paths was to identify the various paths that could be used to achieve the four previously defined safety functions. The front-line systems that were identified in the Individual Plant Examination (IPE) Final Report [Ref. 3-3] were used in determining the systems that make up the paths. Figure 3.1 shows the success path logic diagram (SPLD) utilizing these front-line systems. As is evident from the SPLD there are several paths that could be used to achieve the four required safety functions.

In selecting the preferred and alternate shutdown paths, a balance was maintained among the following criteria:

- system availability
- operator training
- plant procedures
- previous analyses
- walkdown accessibility

The balance means that no one factor dominated the selection process, all inputs were considered to the extent practicable.

The two paths chosen (Figures 3.2 and 3.3) closely follow methods one and two of the Updated Safety Analysis Report (USAR) Appendix F, "Fire Protection Safe Shutdown Analysis" [Ref. 3-4]. Use of this previous study and verifying operator knowledge on the systems provides a high confidence that the best success paths have been chosen. Operators use existing procedures to operate all of the systems in both of the success paths and are trained extensively on the use of these procedures in an on-going operator training program. The lead member of the seismic review team (SRT) is actively involved with this training and therefore has direct knowledge and understanding of these procedures.

An additional requirement from EPRI NP-6041-SL is that both success paths need to be capable of accommodating a loss of off-site power (LOOP) and one of the success paths needs to be capable of mitigating a small break loss of coolant accident (SBLOCA) in conjunction with the SME. Both success paths are capable of mitigating the effects of a LOOP. The alternate path in this analysis is capable of mitigating the effects of a SBLOCA.

SUCCESS PATH LOGIC DIAGRAM

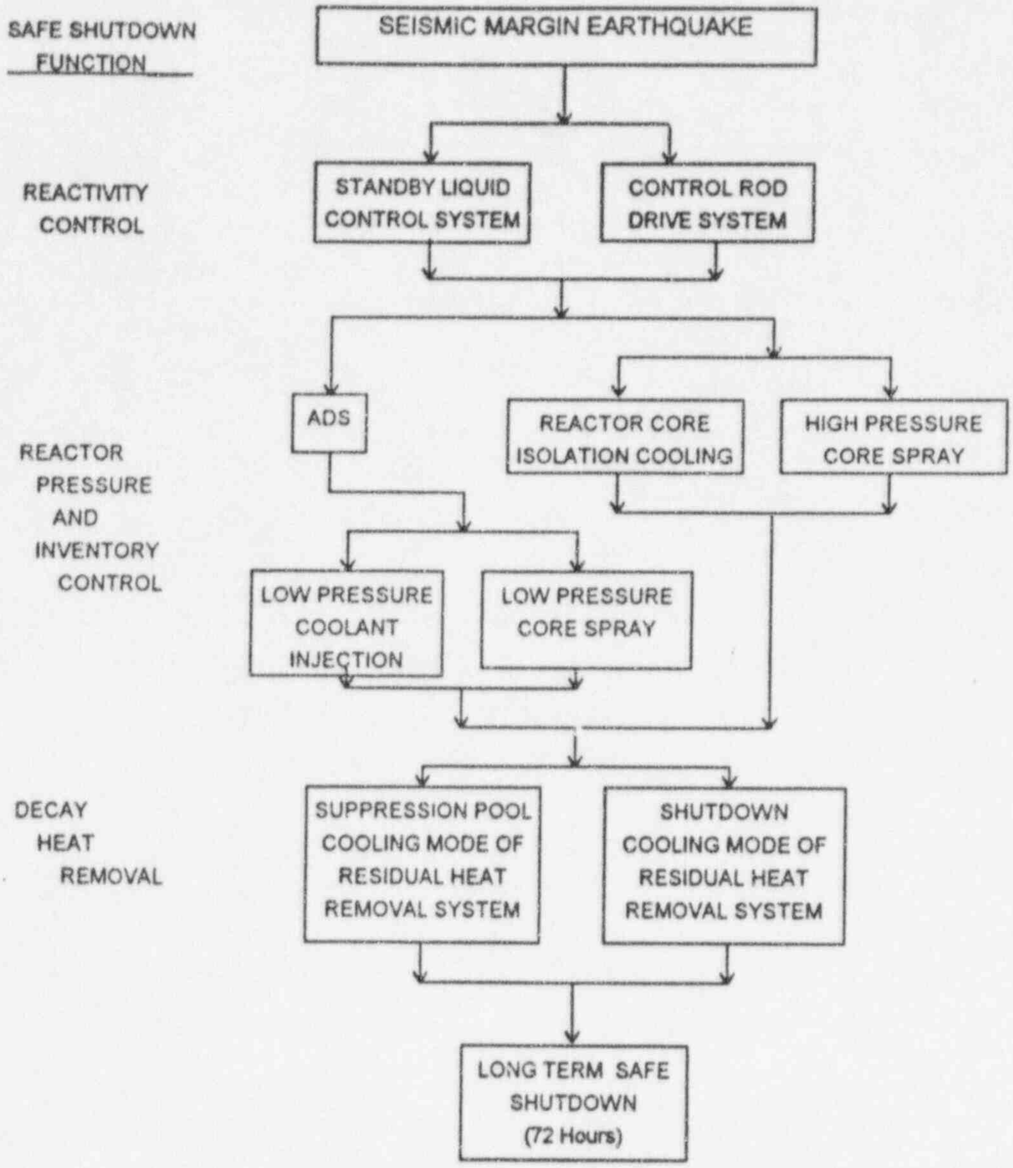


Figure 3.1

PREFERRED SHUTDOWN PATH

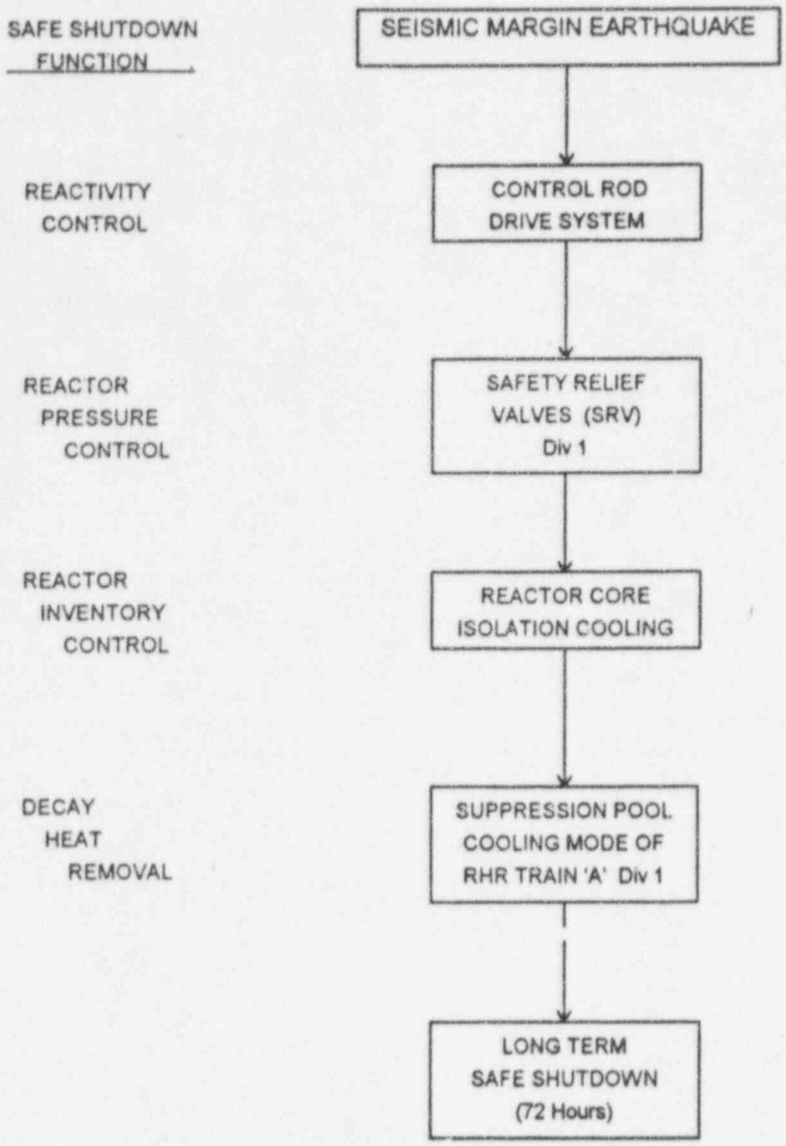


Figure 3.2

ALTERNATE SHUTDOWN PATH

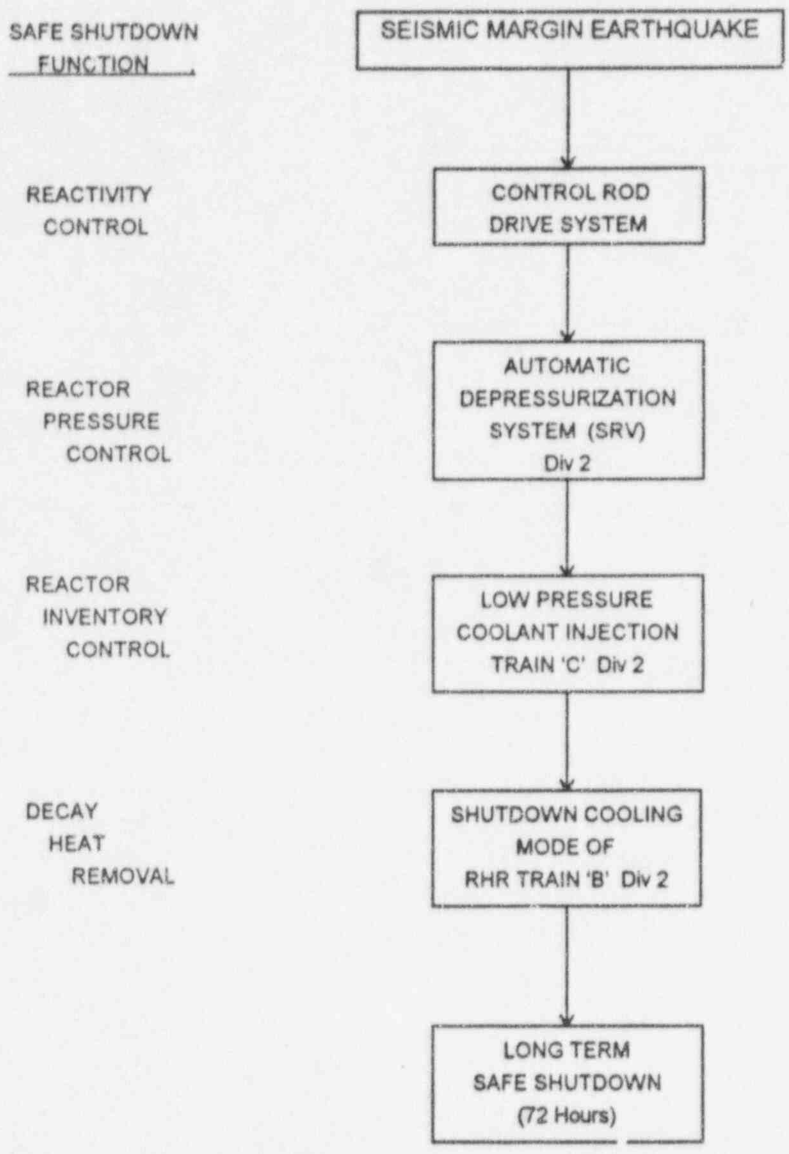


Figure 3.3

3.1.1.a Preferred Path Selection

The preferred path consists of the control rod drive system, division 1 safety relief valves (SRV), reactor core isolation cooling (RCIC), and residual heat removal (RHR) train 'A' in suppression pool cooling. The only difference between this path and the one described as method 1 in USAR Appendix F is the truncation at hot shutdown, i.e., for the SMA analysis, RHR 'A' is not used in the shutdown cooling mode to achieve cold shutdown conditions. Figure 3.2 shows the preferred safe shutdown path.

In this case, if an SME occurs, the reactor protection system (RPS) will provide a SCRAM signal. If an automatic SCRAM does not occur, a manual SCRAM can be performed, if necessary. The control room operators then verify that the reactor is shut down using the light emitting diode (LED) representing "all rods in" on the Rod Action Control Cabinet, 1H13P651.

The RCIC system automatically starts on a low reactor water level signal (level 2), if not already manually started. According to procedure, operators verify that the RCIC system is operating and providing inventory makeup. Reactor vessel water level control will be manual in accordance with the operating procedure. If manual control is not taken, RCIC will stop injecting on a high water level (level 8) signal and no manual actions are required for restarting injection. The system automatically restarts injection on a low water (level 2) signal.

RCIC will be used to maintain inventory while the SRVs maintain reactor pressure, either automatically at the set relief pressure or manually in a lower pressure range.

The suppression pool cooling mode of RHR 'A' will be used when the pool temperature increases because of discharge from RCIC turbine exhaust and the SRVs.

This path places the plant in hot shutdown utilizing division 1 front-line and support systems.

3.1.1.b Alternate Path Selection

The alternate path consists of the control rod drive (CRD) system, automatic depressurization system (ADS) division 2, RHR 'C' for vessel water level control, and RHR 'B' in shutdown cooling mode. This is essentially method 2 of Appendix F of the CPS USAR which takes the reactor to cold shutdown. A deviation from method 2 is that RHR 'B' is not used in suppression pool cooling mode before shutdown cooling mode. Figure 3.3 shows the alternate shutdown path.

A reactor SCRAM will be initiated as described for the preferred path. Operators verify the reactor is shut down using the "all rods in" LED on the Rod Action Control Cabinet 1H13P652. Since this path does not contain the high pressure makeup systems following the SCRAM, the reactor inventory will be reduced when the reactor is depressurized by manual actuation of SRVs through the ADS logic. When pressure drops within the operating range of the low pressure coolant injection (LPCI) system, RHR 'C' pump is started, if not already running in minimum flow mode, and vessel injection commences.

After water level is restored and stabilized, and pressure is below the shutdown cooling (SDC) setpoint, operators procedurally place RHR 'B' in SDC mode. This allows the reactor to be brought to cold shutdown conditions.

The alternate path functions can be accomplished using division 2 equipment except for SDC mode suction valve 1E12F008 which would be manually opened by the operators in accordance with the current operating procedure.

3.1.1.c Systems and Components Used in Success Paths

The CRD system, common to both success paths, is single failure-proof. For the purposes of this analysis, the non-seismically qualified CRD pumps are conservatively assumed to be unavailable during the event. The seismically qualified and rugged hydraulic control units (HCU), with accumulators, provide the driving force to the control rods for the SCRAM.

Table 3.1 shows a matrix of the front-line and support systems used in the preferred and alternate success paths.

The components used in the success paths were identified from EPRI NP-6041-SL using the following criteria.

1) Active mechanical and electrical equipment in front-line and support systems that operate or change state to accomplish a safe shutdown function.

2) Tanks, pumps, and heat exchangers used by or in the identified safe shutdown paths.

3) Passive electrical equipment.

4) All valves, except check valves, which change state during system actuation and operation are included. Check valves are excluded because they do not have exposed operators that could be affected by spatial interactions and inertial loads due to the SME. These valves are judged not to be a cause of damage.

5) Instrumentation needed to confirm that the four safe shutdown functions have been achieved and are being maintained.

6) Instrumentation and controls needed to operate the safe shutdown equipment.

The equipment list [Ref. 3-5] also includes important passive Motor Operated Valves (MOV) such as the suction and discharge MOVs, even though these valves do not change state.

The equipment and instrument lists [Ref. 3-5] include active and passive components in the shutdown paths. These items were initially developed using plant records and have been validated through actual plant walkdowns. Since both success paths are capable of accommodating a LOOP, the auxiliary transformers, bus duct, and switchyard components are not included in the safe shutdown paths or in the walkdowns.

3.1.1.d Seismic Review Team

A Seismic Review Team (SRT) was formed to conduct seismic walkdown evaluations and verifications. The SRT consisted of four onsite utility engineers who attended and successfully completed a "Walkdown Screening and Seismic Evaluation Training Course" and a "Seismic IPE Training Course" which were developed and sponsored by the Seismic Qualification User's Group (SQUG) and the Electric Power Research Institute (EPRI). These two courses provided thorough training in the performance of seismic evaluation methodology to prepare the novice seismic analyst for performance of a seismic margins assessment. During the courses, individuals received detailed training in the methods from the SQUG-developed Generic Implementation Procedure (GIP) and NP-6041-SL. The techniques contained in these references help determine the seismic adequacy of Safe Shutdown Equipment List (SSEL) components. Actual walkdowns in industrial facilities provided hands-on opportunities to more fully develop relevant expertise for evaluating plant equipment capabilities.

One of the four utility team members is an active off-shift STA. All SRT members have a bachelor degree in engineering; two in mechanical and two in civil engineering. Additionally, one has a masters degree in civil engineering. Assigning individuals with a strong formal education background is indicative of the high level of commitment given to the SMA. The SRT experience level is also very high, ranging from 11 to 17 years with an average of over 14 years of direct work experience. Additionally, all four are registered professional engineers.

The composite background and experience of the SRT members make them fully qualified to perform the following tasks:

1) Seismic capability walkdowns of elements in the preferred and alternate success paths,

2) Screening of elements from the SMA which have sufficient ruggedness to assure a high confidence of low probability of failure (HCLPF) equal to or greater than the SME level,

3) Specifying possible component failure modes that were investigated and the type of review required for elements not screened-out, and

4) Documenting the walkdowns.

External contractors were utilized in specialized areas of the SMA. The analysis of structure response and soils issues were two areas where the outside expertise was used to develop a sound analysis and evaluation. The special expertise of the contractors was also utilized on some walkdowns to provide an independent review of the walkdown methodology.

3.1.1.e Product Review Process and Qualifications

Illinois Power's role in the seismic IPEEE included the implementation of SMA methodology, careful review of interim products at each step of the process, critical analysis and evaluation of all results, and overall project management.

The product review process that was used on the seismic portion of the project, along with relevant information on the independent review group members, is detailed in Section 6.2.

3.1.2 System Analysis - Execution of the Walkdowns

The seismic walkdowns were a major portion of the system analyses and accounted for most of the man hours expended. They were planned to make the most effective use of resources.

The walkdowns were initiated on July 19, 1993 by the Illinois Power Company SRT with assistance from consultants from TENERA and EQE International. TENERA and EQE provided knowledgeable and experienced consultants who were invaluable in assisting the utility SRT in their assessments. This external expertise gave added assurance that walkdown techniques and methods were being performed in accordance with the intent of Generic Letter 88-20 Supplement 4 [Ref. 3-7], and provided the basis for effective technology transfer.

Walkdowns continued throughout the summer of 1993 by the utility SRT members in areas that were accessible during power operations. Early walkdowns in low radiological dose areas allowed SRT members to gain experience with evaluating components. Later walkdowns included higher radiological dose areas. All entries into the radiological controlled area (RCA) conformed to the As Low As Reasonably Achievable (ALARA) program for occupational radiation doses.

Areas that were not accessible during power operation were walked-down during refueling outage 4 (RF-4) in September of 1993; these included the drywell (upper elevations) and steam tunnel areas.

To maximize efficiency, the walkdowns were planned and performed area-by-area rather than on a system-by-system basis. They were structured to facilitate a review of components, piping and supports, cable tray and supports, block walls and structures, and seismic interactions in an area during one trip rather than conduct repeated trips to the same area for different reviews. Table 3.2 lists the areas which were walked-down by the SRT for this review. The components associated with the safe shutdown paths plus the containment isolation valves constituted the equipment evaluated during the walkdowns. All buildings and structures included in the safe shutdown paths are seismic category 1.

Table 3.2

AREAS WALKED-DOWN BY THE SRT

LPCS pump room (isolation valves and water leg pump)

RCIC pump room

RHR A pump room and heat exchanger bay (all elevations)

RHR B pump room and heat exchanger bay (all elevations)

RHR C pump room

Diesel Generator Building (all elevations)

MSIV blower room and anterooms (737')

Containment (755' in detail and walk-bys on all other elevations)

Drywell (upper elevations)

Fuel building (all elevations for containment penetration evaluation)

Aux building (781' for electrical equipment and all elevations for containment penetration evaluation)

Table 3.2 (Cont'd)

Control building (781' for electrical equipment, 800' for MCR, 825' for HVAC equipment)

Aux building steam tunnel

Screenhouse (699' Shutdown Service Water Rooms)

3.1.3 Analysis of Structure Response

This section summarizes analyses performed to develop comparisons between the Clinton Power Station SSE design spectra and the SME review spectra for soil and structure response. Soil liquefaction is also discussed in detail.

Analyses were performed to develop comparative CPS SSE design and SME review spectra. Multiple analysis methods were utilized including direct comparisons of free-field input motions, comparisons of deconvolved motions to the structure foundation level, and simplified soil-structure interaction (SSI) analyses.

3.1.3.a Earthquake Spectra Comparison

For seismic design bases, CPS first used a finite element method for SSI analysis and then an elastic half-space approach which included a variation of soil properties. This design analysis procedure was typical of plants designed during the mid- to late 1970s.

The comparison of SSE and SME input motions required the evaluation of analysis methodologies and parameter values to determine if significant margins exist between the design basis methodology and the SME median centered methodology. The SSE and SME are defined for different regulatory purposes and with inherently different levels of conservatism. The seismic response of the containment, power block, and circulating water screenhouse is discussed.

The seismic response of massive structures, such as nuclear power plant buildings, is affected by soils-related phenomena in three fundamental ways:

1) The motions at the free soil surface (excluding the effect of the massive structure) are influenced by the characteristics of the site soil profile. This effect is one of amplification/de-amplification of the surface motion from the rock input at depth. These amplifications are dependent on the dynamic properties of the underlying soil layers,

2) When the site includes a stiff building foundation embedded into the soil media, waves traveling horizontally or vertically will be filtered, giving rise to rotational components of motion. This effect is often referred to as scattering or kinematics interaction,

3) When the building is massive, the structure dynamic motions can cause additional deformation of the soil that affect the foundation motions. This effect is commonly called inertial or Soil Structure Interaction (SSI).

SSI analysis methodology, called the substructure approach, as applied to structures subjected to earthquake excitations consist of the following:

1) Specifying the free-field ground motion,

2) Defining the soil profile and soil property variations,

3) Performing site response analysis,

4) Calculating the foundation input motion,

5) Calculating the foundation impedances; determining the dynamic characteristics of the structure, and

6) Performing the SSI analysis, (i.e., combining the previous steps to calculate the response of the coupled soil-structure system)

The foundation input motion differs from the free-field ground motion for two reasons. First, the free-field motion varies with soil depth. Second, the soil-foundation interface scatters waves because motion of the foundation is constrained according to its geometry and stiffness.

Deep embedment of building foundations is one of the most significant parameters on structure response, and modeling this embedment is essential. The primary nonlinearities of the soil are considered by the development of equivalent linear soil properties. Variations in soil properties are also modeled.

The dynamic characteristics of the structures to be analyzed are described by their fixed-base eigensystem and modal damping factors. Modal damping factors are the viscous damping factors for the fixed-base structure expressed as a fraction of critical damping. The structures' dynamic characteristics are then projected to a point on the foundation at which the total motion of the foundation, including SSI effects, is determined.

The final step in the substructure approach is the actual SSI analysis. The results of foundation input motion, foundation impedances, and structure model are combined to solve the equations of motion for the coupled soil-structure system.

Comparisons of the Clinton seismic input motions were performed utilizing several methods. The comparisons included:

1) Free field input - comparisons of free field SSE design spectra with SME motions were performed.

2) Foundation level input - deconvolution of free field motions to the foundation level were performed and compared. These analyses included quantification of soil dynamic properties and amplification/deamplification effects of the site.

3) Instructure SSI input and responses - simplified SSI models were constructed and analyses performed to provide comparable foundation and instructure spectra results. These analyses included quantification of both kinematics and inertial interaction effects.

A comparative analysis shows the SSE bounds the SME spectrum in the frequency range of 2 to 13 Hertz (Hz), corresponding to the primary modes for the power block, the containment structure, and the first North-South mode of the circulating water screenhouse (CWSH). To ascertain the impact of these exceedances on the building responses, the response spectra of both input motions at the foundation level were computed and compared.

To determine the motion at the foundation level, the ground motion as defined at the free field is propagated downward through the soil layers to rock. The motion at rock may then be propagated back upward to the foundation level. This analytic process is called deconvolution and incorporates the effects of the soil amplification/de-amplification on the input spectra. A site response analysis of the soil profile was conducted to obtain the input motions at the foundation level of the power block foundation.

The foundation level SSE and SME spectra both show the effect of the soil column at frequencies around 5 Hz, which amplifies and de-amplifies the input motions. The free field surface motions are de-amplified in the 5 Hz frequencies by the deconvolution process to rock. The rock motions are amplified by the soil column in the propagation of the motion to the foundation level. The net effect at the foundation level is lower spectral accelerations in the 5 Hertz range than those at the free field surface.

After the deconvolution analysis, the SSE spectra enveloped the corresponding SME spectra in all but a small range of frequencies. The frequency range where the SSE does not exceed the SME was very near the frequency of predominant modes for the power block and the containment structure. The CWSH, however, had predominant frequencies of 10.94 and 17.88 Hz (Table 3.3). The SSE spectra enveloped the SME spectra in this range, therefore the CWSH SSE response enveloped the SME response.

Table 3.3

CWSH MODEL MASS PARTICIPATION

Mode	Freqncy	Participation	%Total Mass	Cum.% Mass	Participation	%Total Mass		
1	10.94	0.02	0.0	0.0	48.50	89.8	89.8	*
2	17.88	50.53	97.5	97.5	-0.03	0.0	89.8	
3	20.41	-0.03	0.0	97.5	8.85	3.0	92.8	
4	24.24	-0.11	0.0	97.5	-11.13	4.7	97.6	
5	31.88	-7.90	2.4	99.9	0.43	0.0	97.6	
6	36.91	0.68	0.0	99.9	4.38	0.7	98.3	
7	38.72	0.19	0.0	99.9	-3.94	0.6	98.9	
8	40.36	1.45	0.1	100.0	1.14	0.0	98.9	
9	56.19	0.14	0.0	100.0	-5.24	1.0	100.0	

* Frequency range where Design Basis input bounds the margin level input(includes SSI effects)

Although this analysis was more refined than the direct free field spectra comparison, the foundation level spectra comparison neglected kinematics and inertial soil-structure interaction effects. For massive structures such as the power block and the containment structure, which share a common foundation, SSI can significantly alter the building response. To include the effects of SSI, a simplified structural and foundation model of the power block/containment structure was constructed and analyzed. This approach offered the most realistic comparison of SSE and SME responses of these buildings.

A simplified idealized mathematical model of the power block/containment structure was constructed and is shown in Figure 3.4. The model utilized geometric, mass, and stress properties derived from the seismic design basis calculations. To capture the responses of the power block and the containment structures, the structure was modeled as two separate shear beams. The containment and the drywell structures were modeled separately on the containment beam.

POWER BLOCK STRUCTURE SIMPLIFIED SEISMIC MODEL

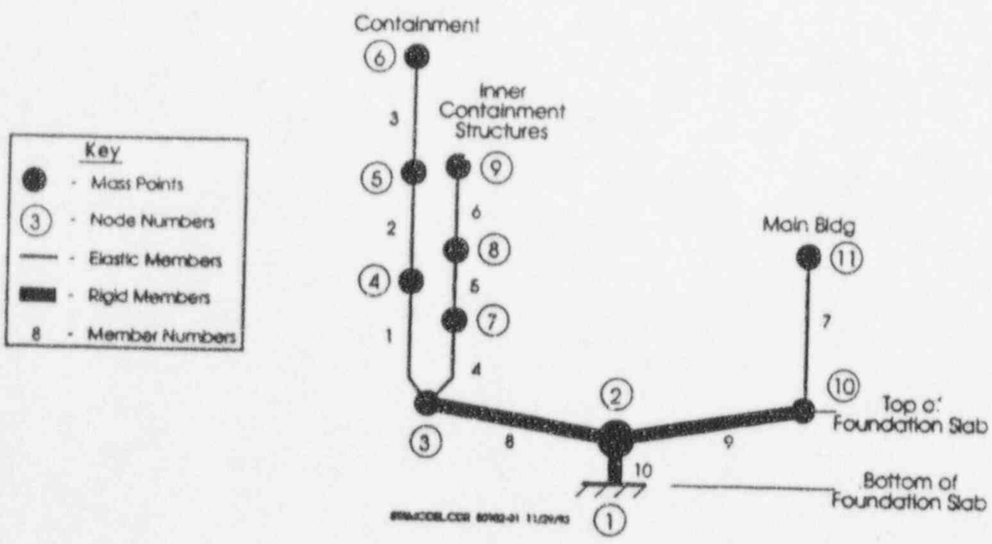


Figure 3.4

The modal masses of the containment were determined from the mass-normalized participation factors summarized in Table 3.4. In the design basis analyses, for both the East-West and North-South directions, there was a significant mode at approximately 4.86 Hz and another at 4.98 Hz. Since the lower frequency mode had the smaller participation, it was the inner containment structure. In addition, there were two frequencies in each direction (between 11.23 and 16.96 Hz) which represented the second modes of the two structures. The first two were the inner containment response, and the others were assigned to the containment structure.

Table 3.4

CONTAINMENT MODEL MASS PARTICIPATION

		E - W			N - S		
Mode	Freqncy	Participation	%Total Mass	Cum.% Mass	Participation	%Total Mass	Cum.% Mass
1	4.86	26.93	22.0	22.0	0.00	0.0	0.0 *
2	4.87	0.00	0.0	22.0	-26.91	28.4	28.4
3	4.98	40.98	51.0	73.1	-1.55	0.1	28.4
4	4.98	1.81	0.1	73.2	35.04	48.1	76.5
5	5.10	-4.75	0.7	73.9	-0.01	0.0	76.5
6	7.68	5.88	1.0	74.9	0.00	0.0	76.5
7	7.80	0.00	0.0	74.9	-6.21	1.5	78.0
8	8.28	-2.77	0.2	75.1	0.00	0.0	78.0
9	8.35	0.00	0.0	75.1	-2.56	0.3	78.3
10	11.02	2.67	0.3	75.4	-0.02	0.0	78.3
11	11.23	-0.83	0.0	75.4	15.23	9.1	87.4
12	13.46	15.24	7.1	82.5	0.00	0.0	87.4
13	14.77	0.99	0.0	82.5	-14.89	8.7	98.0
14	16.96	21.51	14.1	96.5	-0.04	0.0	96.0
15	21.60	0.72	0.0	96.5	0.61	0.0	96.1
16	21.79	0.00	0.0	96.5	0.60	0.0	96.1
17	24.41	-6.88	1.4	98.0	0.00	0.0	96.1
18	24.41	0.00	0.0	98.0	-6.88	1.9	97.9
19	26.77	-5.98	1.1	99.1	0.00	0.0	97.9
20	26.98	0.00	0.0	99.1	6.02	1.4	99.3
21	27.55	0.00	0.0	99.1	-4.10	0.7	100.0
22	27.99	5.56	0.9	100.0	0.00	0.0	100.0

* Frequency range where Design Basis input bounds the margin level input

Both structures were modeled as three node shear beams with lengths equal to the actual building heights. The equivalent beam shear areas were calculated from standard expressions for shear beam frequencies and the remaining properties were assumed rigid. This representation captured the primary building modes and adequately modeled the second responses at the correct frequencies (close to 13 Hz).

The power block had one dominant mode in each direction, as shown in Table 3.5, and was therefore modeled with a single mass shear beam. The modal mass was placed at the approximate height of the dynamic shear resultant, as calculated from standard shear beam solutions. All remaining modal masses were lumped at the base of the structural model. The shear areas were calculated such that the first frequencies of the model matched those of the building.

Table 3.5

POWER BLOCK MODEL PARTICIPATION FACTORS

		-------- E - W --------			-------- N - S --------		
Mode	Freqncy	Participation	%Total Mass	Cum.% Mass	Participation	%Total Mass	Cum.% Mass
1	1.95	-9.00	0.5	0.5	0.03	0.0	0.0
2	2.47	-0.06	0.0	0.5	-9.16	0.5	0.5
3	4.97	0.85	0.0	0.5	-0.20	0.0	0.5
4	6.80	-103.00	69.8	70.3	8.50	0.5	1.0
5	7.58	-15.74	1.6	71.9	-97.69	62.2	63.2
6	8.75	-24.06	3.8	75.7	28.23	5.2	68.4
7	12.82	-9.25	0.6	76.3	41.82	11.4	79.8
8	14.72	-37.76	9.4	85.7	6.50	0.3	80.1
9	17.57	28.40	5.3	91.0	12.21	1.0	81.1
10	18.16	-20.15	2.7	93.6	0.92	0.0	81.1
11	20.52	4.10	0.1	93.7	-46.64	14.2	95.3
12	22.43	-28.08	5.2	98.9	0.59	0.0	95.3
13	23.84	-2.02	0.0	99.0	3.66	0.1	95.4
14	27.36	-6.00	0.2	99.2	-15.65	1.6	97.0
15	29.42	9.01	0.5	99.7	8.83	0.5	97.5
16	30.13	6.32	0.3	100.0	-19.69	2.5	100.0
17	33.71	1.02	0.0	100.0	1.24	0.0	100.0

* Frequency range where Design Basis input bounds the margin level input

For the final analysis, the three simplified building models were attached to a common rigid foundation. The global model included the translational and rotational masses of the foundation in order to better represent the overall system behavior.

Soil impedances were also developed and included in the analysis. Two different approaches were taken to calculate the three dimensional foundation impedances for the combined building model.

The first approximated the rectangular foundation as an embedded disk. The second method used the CLASSI computer programs and assumed the structure was surface founded which modeled the true foundation dimensions.

Analyses for models utilizing the two soil impedance derivations described similar response spectra, demonstrating that the analysis was insensitive to the impedance approximations.

Spectral shapes and peaks for the SSE and SME were generally similar and occurred at approximately the same frequencies. The frequencies where responses were amplified most significantly correspond to dominant building response modes. Because of simplifications of the structural models used in this study, an exact match was not anticipated; however, the responses bore sufficient resemblance to validate the simplified analysis. The design basis earthquake spectra developed in this study exhibited a similarity to the CPS design basis spectra for comparative purposes.

The spectra plots which correspond to the top of the slab at both the containment (Node 3) and power block (Node 10) closely resemble the foundation level input spectra. These spectra all show similar acceleration input in the 2.5 Hertz frequencies of about 0.75g and in the 10 Hertz frequencies of about 0.5g. The comparisons indicate that little amplification occurs from the foundation level to the top of the foundation. Nodes are shown in Figure 3.4.

The spectra plots which correspond to the top of containment (Node 6) closely resemble one another. Each shows significant building amplification at the containment fundamental frequency of about 5 Hertz in both the North-South and East-West directions. Similar patterns of amplified response for the power block also occur at building fundamental frequencies.

Comparisons of SSE versus SME responses for both soil impedance cases demonstrate that the SSE design spectra envelopes the SME response. In some of the spectral comparisons, the SME was slightly greater than the corresponding SSE response. This typically occurred in frequencies below 2 Hz. The exceedances were always small and were characterized by very narrow frequency bands. In this study, only the raw SSE responses were developed for the best estimate soil profile. The actual SSE design spectra developed by Sargent & Lundy broadened the SSE responses by 15% to account for uncertainties in the structural modeling. Thus, the broadened SSE spectra bounds the SME spectra for all frequencies.

Based upon these evaluations, the broadened SSE spectra for the CWSH, containment, and the power block envelope the SME requirement. Structures, systems, and components which were designed using the broadened SSE design spectra input and the technical methods of regulatory guidelines will be conservative relative to the IPEEE seismic margin earthquake.

3.1.3.b Soils Issues

An evaluation of soil-related issues was performed as part of the SMA before Supplement 5 to G.L. 88-20 was issued. For CPS, the scope of soils issues could have been reduced by Supplement 5. This section provides a summary of the evaluation and a discussion on the changes.

The scope of the evaluation focused on the technical methods used to perform the seismic soils assessment. No investigation was warranted in cases where the methodology used in the design and construction stage was consistent with or conservative compared to current practice. Topics reviewed include the following:

1) bearing capacity,
2) seismic wave propagation,
3) liquefaction of structural fills,
4) liquefaction of natural slopes, and
5) slope stability.

The result of the evaluation is that the technical approaches used to evaluate the above topics have been determined to be conservative and consistent with current practice. The soils issues have adequate margin against failure when evaluated for the SME.

As discussed in section 3.1.3.a, the SSE design spectra envelopes the SME spectra. This adds to the technical evaluation that soil-related failures will not adversely affect safe shutdown equipment.

3.1.4 Evaluation of Seismic Capacities of Components and the Plant

3.1.4.a Walkdown Results

The walkdowns were an integral part of this evaluation, and appropriately received considerable resources in the form of time and effort.

Walkdown findings were recorded on Seismic Evaluation Walkdown Sheets (SEWS) from Appendix F of EPRI NP-6041-SL. The SEWS are organized to prompt the user of the various issues that should be considered rather than provide a narrowly focused checklist. Adequacy of anchorage, spatial interactions, ruggedness of other mounted equipment, and unusual configurations were some of the issues that were noted during the walkdowns.

When components are screened-out, it means that those items are considered to be seismically adequate for the SME evaluation. The screened-out components would have an expected high confidence of low probability of failure (HCLPF) number that is at or above the SME of 0.3g. The screening process makes it unnecessary to calculate HCLPF numbers for screened-out components. The HCLPF capacity is intended to represent an earthquake level in which there is approximately 95% confidence of less than about a 5% failure probability. There is no further review required for screened-out components. The seismic margin capability (expressed in terms of HCLPF) for any success path is then assessed to be equal to the seismic margin capability of the weakest component in that success path.

A summary of various equipment groups that were evaluated during the walkdowns follows.

Motor-operated valves (MOV) were evaluated using cantilever (height-and-weight) screening criteria and allowable stress limits. Some of the motor-operated valves (MOV) on the safe shutdown walkdown and the containment walkdown did not pass the first screen, the SQUG cantilever (height-and-weight) criteria. These were cases that involved large motor operators on valves in small lines. The second screening for these valves involved checking allowable stress levels in the valves with expected demand levels. All MOVs passed the first and second levels of screening.

Instrument racks were walked-down for adequacy of anchorage, instrument mountings, and spatial interactions.

In the main control room (MCR), there were several areas that were the focus of the review. The floor section details were reviewed to determine the adequacy of the anchorage to the embedments in the floor and connections to the termination panels and control panels. Relay mounting details were inspected and relay chatter is discussed further in section 3.1.4.b. Non-seismically mounted light fixtures and ceiling tiles in the back panel area were reviewed for II/I interactions with the panels. This was not considered to be a problem since the light fixtures and the 12 inch square tiles are not of sufficient weight to cause impact problems with the panels. The panels are not "soft" targets, meaning that no damage would occur if hit by falling lights and ceiling tiles. In the front panel area, arranged in a horseshoe layout, the seismically qualified ceiling was inspected for adequacy of bracing. The unistrut hangers and vertical rod holders use bolts and stiff metal angles to support and contain the diffuser gratings. This arrangement is adequate to withstand seismic motion of the SME. All non-seismically mounted furniture (file cabinets, table, chairs, drawing racks), were reviewed for impact with panels and determined to not pose a seismic hazard for safe shutdown of the reactor.

Pumps, as expected from earthquake experience, were found to be rugged components. The exception to this was the SX pumps with long 42 foot shafts extending down to the screenhouse inlet area. A review of design documents, as documented on the SEWS walkdown sheets, showed that the shafts have two supports, one at 27 feet and one at 35 feet, each which provide lateral restraint near the lower end. This configuration was shown to be sufficient to meet the requirements of NP-6041-SL for long shaft pumps.

Air handlers and fans were found in all areas of the plant from the emergency core cooling system (ECCS) pump rooms to the main control room HVAC (VC) trains on the 825' elevation. Vibration isolators were targeted in seismic training courses as possible weak links in fan anchorage details. The ones at CPS were reviewed and found to possess the lateral seismic supports that are desirable for surviving earthquake motion. Ductwork and air handlers also appeared, in the judgment of the SRT, to have been built rugged enough to function properly after an earthquake.

In the containment, the hydraulic control units (HCU) for the control rod drive (CRD) mechanisms were found to be exceptionally well braced for lateral motion as well as vertical motion. HCUs were bolted onto common supports that effectively made each line of from four to seven units a single seismic unit.

The ADS air bottle racks were walked-down as part of the back-up air supply for the SRVs. All of the equipment and anchorages were shown to be seismically rugged.

The diesel generators were walked-down to assess the adequacy of anchorage and attachment of peripheral equipment. All of the equipment and anchorages were shown to be seismically rugged.

In areas where masonry block walls were present, the review centered around whether or not the walls had reinforcing. All block walls were evaluated and passed this review.

The reactor pressure vessel (RPV), pedestal, nuclear steam supply system (NSSS) piping, CRD mechanisms, containment structure, and containment internal structures were not required to be walked-down for CPS per the guidelines for the seismic margins assessment.

There were no flat bottom metal fluid storage tanks included in the safe shutdown paths. Existing flat bottom tanks were reviewed for problems to safe shutdown equipment caused if rupture were to occur during an earthquake. There are three such tanks around the power block. The closest to the building is the Reactor Core Isolation Cooling (RCIC) tank containing 125,000 gallons of water. This tank is surrounded by an earthen berm to contain the contents of the tank in case of rupture. For the SMA, the rupture of the RCIC tank has no impact on safe shutdown equipment. The two other flat bottom tanks are also for the storage of water. The Makeup Condensate Water (MC) tank and the Cycled Condensate (CY) tank contain 400,000 gallons of water each. The CY tank has an earthen berm around it which is designed to contain the contents of the tank in case of a rupture. The MC tank does not have a berm, but is located approximately 300 feet from the closest building. The distance is greater than 600 feet to safe shutdown equipment. The amount of water from this tank can be handled by drainage ditches that were built for normal precipitation and therefore, is not a concern.

All areas of the plant that contain safe shutdown equipment were checked for seismically induced flooding during the walkdowns with no problems noted.

The SMA walkdowns covered all areas that contain equipment that is procedurally used by Operations to bring the plant to a stable condition and maintain that condition for at least 72 hours following an SME.

As mentioned in section 3.0, it is not the intent of this program to determine the largest earthquake the site could withstand, so HCLPF numbers are not calculated. The screening results show that expected component HCLPFs would be equal to or greater than the SME of 0.3g.

The plant walkdowns have met the intent of the request in Generic Letter 88-20, Supplement 4, using the methodology outlined in EPRI NP-6041-SL with no "weak-link" components or areas that need further seismic evaluation. All of the items on the SSEL component list and containment isolation valves were walked-down by the SRT by November of 1993.

3.1.4.b Relay Chatter Evaluation

The relay evaluation for the focused scope bin that CPS is classified in consists of locating and evaluating low seismic ruggedness relays (bad actors).

Appendix E of EPRI NP-7148-SL [Ref. 3-11] identifies the relays that have been determined to be bad actors. This list was compared with the plant relay data base maintained by the Electrical Design group of Nuclear Station Engineering Department (NSED). A computer search was performed for each relay model on the list. When relays were found to match the relay model number, a computer print-out was made which contained the relay device number, drawing number, panel number, safety or non-safety classification, comments, and quantity.

The computer print-out was cross-referenced to the safe shutdown equipment list (SSEL) (section 3.1.1) to eliminate those relays from the evaluation which are not in either of the safe shutdown systems paths. A circuit analysis was then performed by a well qualified and experienced licensed senior reactor operator (SRO) to determine the function of the circuits the relays were in, and what functions the relays performed. The results are tabulated in Table 3.6.

Table 3.6

LIST OF LOW RUGGEDNESS RELAYS AT CPS

INCLUDED IN SUCCESS PATH SYSTEMS

RELAY MODEL #	SYSTEM	PANEL	REMARKS
GE CEH	DG A	1PL12JA	Annunciator only, DG A low oil temp
GE CEH	DG B	1PL12JB	Annunciator only, DG B low oil temp
GE HGA	DG A	1PL12JA	DG A fuel priming pump start
GE HGA	DG A	1PL12JA	DG A fuel priming pump start
GE HGA	DG A	1PL12JA	DG A overspeed/overcrank lockout
GE HGA	DG A	1PL12JA	DG A lube oil high temp lockout
GE HGA	DG A	1PL12JA	DG A overcurrent lockout
GE HGA	DG B	1PL12JB	DG B fuel priming pump start
GE HGA	DG B	1PL12JB	DG B fuel priming pump start
GE HGA	DG B	1PL12JB	DG B overspeed/overcrank lockout
GE HGA	DG B	1PL12JB	DG B lube oil high temp lockout
GE HGA	DG B	1PL12JB	DG B overcurrent lockout
W SSC	AP	1AP07EJ	Bus overcurrent trip
W SSC	AP	1AP07EJ	Bus overcurrent trip
W SSC	AP	1AP09EB	Bus overcurrent trip
W SSC	AP	1AP09EB	Bus overcurrent trip
W SSC	LPCS	1AP07EE	LPCS pump overcurrent trip
W SSC	RHR A	1AP07EG	RHR A pump overcurrent trip
W SSC	RHR B	1AP09ED	RHR B pump overcurrent trip
W SSC	RHR C	1AP09EF	RHR C pump overcurrent trip
W SSC	SX A	1AP07ED	SX A pump overcurrent trip
W SSC	SX B	1AP07EG	SX B pump overcurrent trip

Walkdowns of the panels containing all of the above relays verified that the relays were actually mounted in the panels listed in the database and that the relay mountings were seismically adequate.

These relays have been tested, by Wyle Laboratories, to withstand, without compromise of structures or electrical function, the CPS SSE, as documented by the seismic qualification packages that are part of the design basis of CPS. Since the CPS broadened SSE response spectrum envelopes the SME response spectrum [Ref. 3-9], no further evaluation of these relays is required.

3.1.4.c Seismic Spatial Interaction

In addition to all safe shutdown components, those valves and piping which had anchor points on different buildings were evaluated for interaction. An example of this appeared in piping which crossed the seismic gap between the containment building and fuel or auxiliary buildings. Resolution of these concerns utilized design calculations which had been performed to show that seismically induced stresses did not exceed allowable stresses on the component or the pipe. For all components, actual clearance distances were compared to the calculated building and/or piping displacements based on the SSE at the particular point to ensure that actual distances were greater than calculated motions.

Another program that had been performed at CPS during construction is discussed below. Seismic spatial interactions were evaluated during the Interaction Analysis Program (IAP). This program was intended to identify and correct potential seismic interactions. The IAP walkdowns generated 8331 potential interaction reports (PIRs) which were resolved by engineers. IP, Burns and Roe, and Sargent and Lundy were all involved in the resolution of the PIRs. For dispositions requiring rework, a surveillance walkdown team verified the adequacy of redesigns by walkdown after correction. As a result of this major construction effort, no spatial interaction concerns were identified in the walkdown that needed modification.

Previous confirmatory reviews of the CPS seismic design have also been performed including the Equipment Seismic Assessment Program (ESAP) [Ref. 3-8]. ESAP was developed in response to a request made as a result of the Advisory Committee on Reactor Safeguards (ACRS) in March of 1982. Phase I was a review of piping designed by the CPS Architect Engineer to ensure that piping was designed to withstand SSE loads by checking design calculations. Phase II examined the as-built equipment configurations of the decay heat removal and emergency power supply systems for seismic concerns.

Phase III consisted of evaluation of equipment stress levels for loadings based upon the revised response spectra using the elastic half-space approach. The ESAP work was completed in October of 1985 and submitted to the NRC.

There were no unusual configurations identified during the walkdowns and all identified concerns were resolved after checking stress and displacement calculations.

3.1.5 Analysis of Containment Performance

The CPS IPE for Internal Events report [Ref. 3-3] was referenced in determining the extent of the containment analysis. The IPE results show that CPS has a stronger containment for internal pressure loadings than other domestic BWR-6 plants because of more concrete reinforcement. The internal IPE confirmed there are no plant vulnerabilities and the containment is robust. The IPEEE walkdowns included containment penetrations to look for unique penetration configurations and spatial interaction concerns. The containment walkdown was integrated into the systems walkdown on an area-by-area basis. The hydrogen ignitors were not included in the containment walkdowns. From the IPE Internal Events report, the ignitors were not critical in prevention of early containment failure.

The containment equipment hatch was inspected closely because of a small clearance with the topping slab of the floor of the fuel building. There is a possibility of induced loads into the floor of the fuel building if there is differential motion between the containment and fuel buildings. The building differential motion, as shown in design calculations, at this location is indeed negligible, almost zero, so no further evaluation is necessary.

No other concerns were generated as a result of the containment walkdowns. The containment performance evaluation did not identify any vulnerabilities that involve early failure of containment functions because of an SME. There were no situations identified in the containment walkdown that needed modification to eliminate seismic concerns. An insight of this External Events analysis is that the findings of the IPE Internal Events analysis are validated on containment integrity.

Following the guidance in NUREG-1407 and a review of the CPS IPE, no additional systems were added to the containment performance portion of the IPEEE analysis. There are no inflatable seals used on containment hatches and there are no containment penetrations that require post-accident cooling at CPS. There are no masonry walls inside the containment. The containment walkdown and analysis concluded that the containment and components are seismically rugged when analyzed for the SME spectra.

3.2 Other Seismic Safety Issues

CPS is not required to perform an A-46 review since the plant was constructed after implementation of the IEEE 344-1975 standard for qualification of Class I electrical equipment. There were no A-46 related concerns which developed from the application of the seismic methodology at CPS.

Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements", was subsumed into the IPE. The residual heat removal system was included in the safe shutdown analysis portion of the IPEEE. The previously completed ESAP program [Ref. 3-8], mentioned in section 3.1.2, specifically targeted the seismic capability of the decay heat removal system. No potential vulnerabilities were found that would prevent this system from functioning in the design heat removal role. There were no additional USI A-45 related concerns which developed from the application of the seismic methodology at CPS.

3.3 References for Chapter 3

3-1. EPRI NP-6041-SL, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin", Revision 1, August 1991.

3-2. Seismic Qualification Utility Group, "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment", Revision 2, February 1992.

3-3. Clinton Power Station "Individual Plant Examination", Final Report, September 1992, letter number U-602040.

3-4. Clinton Power Station Updated Safety Analysis Report, Appendix F, "Fire Protection Safe Shutdown Analysis".

3-5. Clinton Power Station, "Selection of Safe Shutdown Paths for the Seismic Margins Assessment", July 1994, letter number Y-104037.

3-6. EPRI NP-7217-SL, "Seismic Margin Assessment of the Edwin I. Hatch Nuclear Plant, Unit 1", Final Report, June 1991.

3-7. NRC Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", June 1991.

3-8. Clinton Power Station, "Equipment Seismic Assessment Program (ESAP) Report", October 1985, letter number U-600281.

3-9. EQE Report "CPS Seismic Margin Assessment for IPEEE Seismic Response Motion Comparison", March 1994.

3-10. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", Final Report June 1991.

3-11. EPRI NP-7148-SL, "Procedure for Evaluating Nuclear Power Plant Relay Seismic Functionality", December 1990.

3-12. NUREG-0853, "Safety Evaluation Report for CPS Unit 1", February 1982.

3-13. Clinton Power Station Technical Specifications, Docket No. 50-461, April 17, 1987.

3-14. Sargent & Lundy, "Clinton Project Unique Procedures and Design Basis", EMD 035307, February 1982.

3-15. "Equipment Seismic Qualification Design Basis Document for Clinton Power Station", Sargent & Lundy Engineers, Component and Materials Division, January 1993.

3-16. NUREG/CR-0098, "Development of Criteria for Seismic Review of Selected Nuclear Power Plants", May 1978.

3-17. Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants", December 1973.

3-18. EPRI NP-6359, "Seismic Margin Assessment of the Catawba Nuclear Station", Final Report, April 1989.

3-19. EPRI TR-101055 Research Project RP2722-30 "Comparison of the Guidance in GIP and NP-6041", draft, May 1992.

3-20. CPS 3310.01, "Reactor Core Isolation Cooling (RI)", Revision 10, April 1992.

3-21. CPS 3312.01, "Residual Heat Removal (RHR)", Revision 19, July 1992.

3-22. CPS 4100.01, "Reactor Scram", Revision 11, May 1992.

3-23. CPS 4401.01, "EOP-1 RPV Control", Revision 21, April 1992.

3-24. CPS 4407.01, "EOP-3 Emergency RPV Depressurization (Blowdown)", Revision 21, April 1992.

4. INTERNAL FIRE ANALYSIS

4.0 Methodology Selection

In accordance with Section 4.1 of NUREG-1407, a fire Probabilistic Risk Analysis (FPRA) methodology was selected for the CPS IPEEE. The FPRA methodology selected has recently been developed by the Electric Power Research Institute (EPRI) and is described in EPRI Project Report 3385-01, "Fire Risk Analysis Implementation Guide" dated January 1994.

This FPRA methodology consists of the following major tasks:

1) Develop Fire-Induced Sequences,
2) Develop Fire Scenarios,
3) Evaluate Fire Damage Sequences, and
4) Document and Verify Analysis.

These tasks are further defined by a series of 10 subtasks or steps that

1) Define fire areas and zones,
2) Develop spatial information,
3) Determine the ignition sources and firezone ignition frequencies,
4) Define fire Scenarios,
5) Evaluate fire propagation and damage,
6) Evaluate fire detection and suppression,
7) Evaluate recovery actions,
8) Evaluate multi-compartment scenarios,
9) Evaluate fire risk scoping study (FRSS) issues and uncertainties, and
10) Verify, evaluate and document results.

The methodology described by the steps listed above is considered equivalent to that outlined in Section 4.1, NUREG-1407.

Additionally, the CPS IPEEE has closely followed the methods outlined in the EPRI FPRA methodology.

The existing CPS PRA model developed for the internal IPE was used as the basis for evaluating fire risk in the FPRA.

4.1 Fire Hazard Analysis

Fire hazard analysis is composed of the first two listed FPRA tasks (first four "steps") identified in Section 4.0 above. The first steps of the analysis are intended to develop inputs to the existing PRA model to allow screening those plant firezones with minimal impact on core damage frequency (CDF). The firezones used to begin the analysis are almost exclusively based on the Fire Hazards Analysis of the CPS Appendix R study.

The firezones evaluated in the CPS FPRA are identified in Table 4.1. These correspond to figures 4.1 through 4.23 with one exception. Firezone CB-1i (825 ft elevation of the control building) on figure 4.12 has a 3 hour barrier that bisects the zone. In the FPRA this zone was split into two firezones. A special walkdown was performed to identify the location of cables and equipment in relation to the 3 hour barrier. The only firezones not included in the evaluation were the containment and drywell. These firezones were specifically excluded in the FPRA methodology based on the physical structure and fire history of these locations.

The first screening step identified and located all of the equipment modeled in the CPS PRA as well as all cables required for that equipment to operate. This information allowed the identification of fire-induced initiators and equipment failures to be incorporated in the PRA model and quantified for each firezone. The value obtained from the quantification process is called the conditional core damage probability (CCDP) and represents the probability of core damage occurring given that a fire damages all equipment and cables in a given firezone.

The CCDP for each firezone was calculated by identifying all basic events (failures) in the PRA model as well as initiators that could occur as a result of a fire damaging all equipment and cables located in a firezone. All equipment in the zone and equipment dependent on the cables traversing the zone are assumed failed and the PRA model solved. Those firezones determined to have no significant impact on core damage were screened from further analysis.

The remaining firezones were examined for ignition sources, and fire ignition frequencies were calculated for each zone. Fire ignition sources and the determination of the ignition frequencies were compiled from domestic nuclear plant data. This nuclear plant fire data is contained in the EPRI fire events database (NSAC-178L). Estimation of the number and type of potential ignition sources within each firezone was accomplished by using a combination of the following: a plant equipment location database, drawings, plant walkdowns, and the CPS master equipment list (MEL).

To compute the fire ignition frequency for each firezone, the firezones were recategorized by location type such as main control room, diesel generator room, switchgear room, etc. The ignition frequencies were then computed from the domestic nuclear plant fire reports based on the location and component types on a plant wide basis. The frequencies were then apportioned to individual firezones by evaluating the potential ignition sources and applying the guidelines in the FPRA methodology. Switchgear rooms were treated somewhat differently due to the large variance in the number of electrical cabinets located in some firezones. The FPRA methodology directs that the plant switchgear room ignition frequency be divided by the number of switchgear locations and assigned equally to each location. At CPS, the number of electrical cabinets in switchgear locations ranged from 229 to 28. Using the same ignition frequency for all zones would significantly understate the risk of locations with higher concentrations of cabinets. To avoid this situation, the plantwide switchgear risk was apportioned to individual switchgear locations using the ratio of the number of electrical cabinets in a zone over the total population of electrical cabinets in all switchgear locations. Column 4 of Table 4.2 provides the results of this initiating frequency determination.

The second screening step involved comparing the CDF for each previously unscreened firezone with a significance threshold. The CDF was calculated by multiplying the firezone CCDP and ignition frequency. The criteria for screening firezones not considered to be a significant fire risk was established in the FPRA methodology at a firezone CDF of less than 1.0×10^{-7}/yr. Additionally, firezones with core damage frequency between 1.0×10^{-6} and 1.0×10^{-7}/yr with no potential to cause failure of containment isolation functions were also screened. The firezones not screened constituted the set of fire hazards subjected to detailed fire modeling to establish the actual fire risk.

The results of the screening analysis are provided in the last column of Table 4.2. Twenty-one firezones were identified as requiring detailed fire modeling. These firezones, with one exception, are located within the intake structure or the auxiliary and control buildings. Since cabling to the safety systems and other important systems run through these buildings, the high screening CDFs were an expected intermediate result. The single firezone located outside of the noted buildings is in the radwaste building. This zone is located directly adjacent to the control building and acts as an intersection point for balance of plant cables. This concentration of electrical cables made the inclusion of this firezone in the list requiring fire modeling an expected intermediate result.

4.1.1 Equipment Identification

Several databases were developed from plant specific data to track the location and cross reference information for equipment modeled in the CPS PRA. One database was used to provide the firezone location information of the modeled PRA equipment. A second database tracked the equipment modeled in the PRA by the associated basic event(s). A third database was developed to compile the location information for plant ignition sources.

Table 4.1

Firezone Descriptions

A-1a - 707'6" Auxiliary Bldg, Hallway

A-1b - 737' Auxiliary Bldg, General Access Area

A-1c - 737' Auxiliary Bldg, Instrument Storage Room

A-1d - 737' Auxiliary Bldg, 737' Aux Bldg, PC Storage Room

A-1e - 737' Auxiliary Bldg, West Airlock and Ramp Area

A-2a - 707'6" Auxiliary Bldg, RCIC Room

A-2b - 707'6" Auxiliary Bldg, RHR "A" Room

A-2c - 707'6" Auxiliary Bldg, LPCS Room

A-2d - 737' Auxiliary Bldg, Containment Airlock Area

A-2e - 737' Auxiliary Bldg, MSIV Leakage Control Room

A-2f - 737' & 762' Auxiliary Bldg, Pipe Tunnel

A-2g - 737' Auxiliary Bldg, RT Pump Room "A"

A-2h - 737' Auxiliary Bldg, RT Pump Room "B"

A-2i - 737' Auxiliary Bldg, RT Pump Room "C"

A-2j - 750'6" Auxiliary Bldg, RT Pipe Mezzanine

A-2k - 762' Auxiliary Bldg, Div 1 Non-safety Switchgear Room

A-2m - 762' Auxiliary Bldg, East Gas Control Boundary Room

A-2n - 781' Auxiliary Bldg, Div 1 Safety Switchgear Room

A-2o - 781' Auxiliary Bldg, East Gas Control Boundary Room

A-3a - 707'6" Auxiliary Bldg, RHR "B" Room

A-3b - 707'6" Auxiliary Bldg, RHR "C" Room

A-3c - 707'6" Auxiliary Bldg, Aux Bldg Floor Drain Tank/Pump Area

A-3d - 762' Auxiliary Bldg, Div 2 Non-safety Switchgear Room

A-3e - 762' Auxiliary Bldg, West Gas Control Boundary Room

A-3f - 781' Auxiliary Bldg, Div 2 Safety Switchgear Room

A-3g - 781' Auxiliary Bldg, West Gas Control Boundary Room

A-4 - 781' Auxiliary Bldg, Battery 1A Room

A-5 - 781' Auxiliary Bldg, Battery 1B Room

CB-1a - 712' & 719' Control, Unit 2 DG Tank Rooms

CB-1b - 702' Control, Entire Level Excluding Stairwells

CB-1c - 719' Control, Entire Level Excluding Stairwells

Table 4.1 (Cont'd)

Firezone Descriptions

CB-1d - 737' Control, Chemistry Lab Areas
CB-1e - 737' & 751' Control, General Access and Lab HVAC Areas
CB-1f - 762' Control, Component Cooling Water Equipment Area
CB-1g - 781' Control, Storage Area
CB-1h - 702' through 847' Control, East Stairwell
CB1i-E - 825' Control, Control Room HVAC Area, East of Barrier
CB1i-W - 825' Control, Control Room HVAC Area, West of Barrier
CB-2 - 781' Control, Div. 2 Cable Spreading Area
CB-3a - 781' & 790' Control, DC/UPS Equipment Area
CB-3b - 781' Control, Div. 4 Inverter Room
CB-3c - 781' Control, Battery 1E Room
CB-3d - 781' Control, Div. 4 Battery Room
CB-3e - 781' Control, 1E Inverter Room
CB-3f - 781' Control, 1F Inverter Room
CB-3g - 781' Control, 1F Battery Room
CB-4 - 781' Control, Div. 1 Cable Spreading Room
CB-5a - 781' Control, Div. 3 Switchgear Area
CB-5b - 781' Control, Div. 3 Battery Room
CB-5c - 781' Control, Vertical Cable Chase Area
CB-6a - 800' Control, Main Control Room
CB-6b - 800' Control, Operations Support Center
CB-6c - 800' Control, Technical Support Center
CB-6d - 800' Control, Ops Kitchen/Restroom/Storage Areas
CB-7 - 702' through 847'2" Control, West Stairwell and Hall to 800' TB
D-1 - 712' & 719' DG Bldg, Div. 3 Diesel Fuel Tank Room
D-2 - 712' & 719' DG Bldg, Div. 1 Diesel Fuel Tank Room
D-3 - 712' & 719' DG Bldg, Div. 2 Diesel Fuel Tank Room
D-4a - 737' DG Bldg, Div. 3 DG Room
D-4b - 737' DG Bldg, Div. 3 DG Day Tank Room
D-5a - 737' DG Bldg, Div. 1 DG Room

Table 4.1 (Cont'd)

Firezone Descriptions

D-5b - 737' DG Bldg, Div. 1 DG Day Tank Room
D-6a - 737' DG Bldg, Div. 2 DG Room
D-6b - 737' DG Bldg, Div. 2 DG Day Tank Room
D-7 - 762' DG Bldg, Div. 3 DG Ventilation Area
D-8 - 762' DG Bldg, Div. 1 DG Ventilation Area
D-9 - 762' DG Bldg, Div. 2 DG Ventilation Area
D-10 - 762' DG Bldg, HVAC Area
F-1a - 712' Fuel Bldg, General Access Area
F-1b - 712' Fuel Bldg, HPCS Room
F-1c - 712' Fuel Bldg, Fuel Bldg Floor Drain Tank Room
F-1d - 712' Fuel Bldg, Fuel Bldg Floor Drain Pump Room
F-1e - 712' Fuel Bldg, Fuel Bldg Equipment Drain Tank Room
F-1f - 712' Fuel Bldg, Fuel Bldg Equipment Drain Pump Room
F-1g - 712' Fuel Bldg, Fuel Cask Area Pump Room
F-1h - 712' Fuel Bldg, Fuel Pool Cooling and Cleanup (FC) Valve Room
F-1i - 712' Fuel Bldg, FC Pump Room
F-1j - 737' Fuel Bldg, Cask Washdown Change Room
F-1k - 737' Fuel Bldg, CRD Rebuild Room
F-1m - 737' Fuel Bldg, General Access Area
F-1n - 737' Fuel Bldg, FC Heat Exchanger Room
F-1o - 748'6" Fuel Bldg, Tunnel/Gamma Scan Area
F-1p - 755' & 781' Fuel Bldg, Entire Area of Both Elevations
M-1 - 699' Screenhouse, Div. 1 SX Pump Room
M-2a - 699' Screenhouse, Div. 3 SX Pump Room
M-2b - 699' Screenhouse, Div. 2 SX Pump Room
M-2c - 678' & 699' Screenhouse, General Access and Pipe Tunnel Areas
M-3 - 699' Screenhouse, Diesel Driven Fire Pump "B" Room
M-4 - 699' Screenhouse, Diesel Driven Fire Pump "A" Room
MUWPH - Make-up Water Pump House, All Areas

Table 4.1 (Cont'd)

Firezone Descriptions

R-1a - 702' Radwaste, Makeup Demineralizer Area
R-1b - 702' Radwaste, Charcoal Adsorber Room
R-1c - 702' Radwaste, General Access Area
R-1d - 702' Radwaste, Floor Drain Tank Area
R-1e - 702' Radwaste, Phase Separator Tank Room
R-1f - 702' Radwaste, Waste Tank Room
R-1g - 702' Radwaste, Chem Waste Tank Room
R-1h - 720'6" Radwaste, General Access Area
R-1i - 737' & 750' Radwaste, Maintenance Shop Areas
R-1j - 737' Radwaste, Waste Solidification Area
R-1k - 737' Radwaste, Oil Tank Room
R-1m - 737' Radwaste, Storeroom
R-1n - 737' Radwaste, Paint and Oil Storage Room
R-1o - 737' Radwaste, Radwaste Operation Center
R-1p - 762' Radwaste, Maintenance Office Area
R-1q - 762' Radwaste, Service Air Compressor Area
R-1r - 762' Radwaste, 480V Unit Substation Area
R-1s - 762' Radwaste, HVAC Area
R-1t - 781' Radwaste, General Access Area
R-1u - 781' Radwaste, Calibration Lab
RITANK - RCIC Storage Tank Room
SERVICE - Service Building, All Areas
T-1a - 712' Turbine, General Access Area
T-1b - 712' Turbine, Condensate Booster Pump Room
T-1c - 712' Turbine, Condensate Pump Room
T-1d - 712' Turbine, Condenser Pit
T-1e - 737' & 762' Turbine, Heater Bay & Pipe Tunnel Areas
T-1f - 737' Turbine, General Access Area
T-1g - 762' Turbine, Heater/MSR Bays
T-1h - 762' Turbine, General Access Area
T-1i - 762' Turbine, Turbine Oil Reservoir Room

Table 4.1 (Cont'd)

Firezone Descriptions

T-1j - 781' Turbine, SJAE Rooms

T-1k - 781' Turbine, General Access Area

T-1m - 800' Turbine, Turbine Deck

T-1n - 800' Turbine, Hydrogen Analyzer Room

Figure 4.1

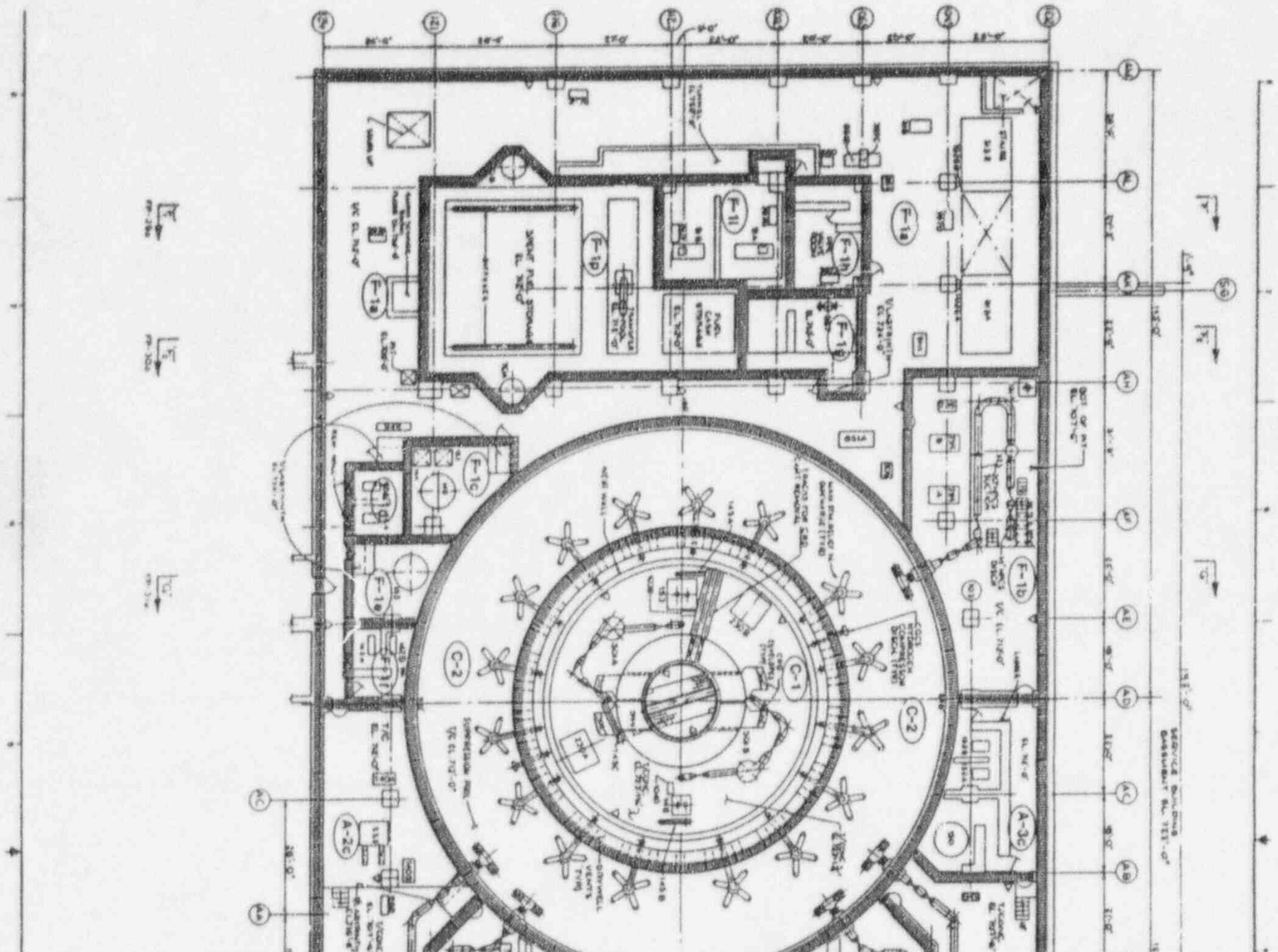

CLINTON POWER STATION
FIRE PROTECTION
FIGURE FP-2c
FIRE ZONE BOUNDARIES
AUXILIARY FUEL BUILDING AND CONTAINMENT
BASEMENT FLOOR PLAN - EL 707'-6" & 712'-0"

Figure 4.2

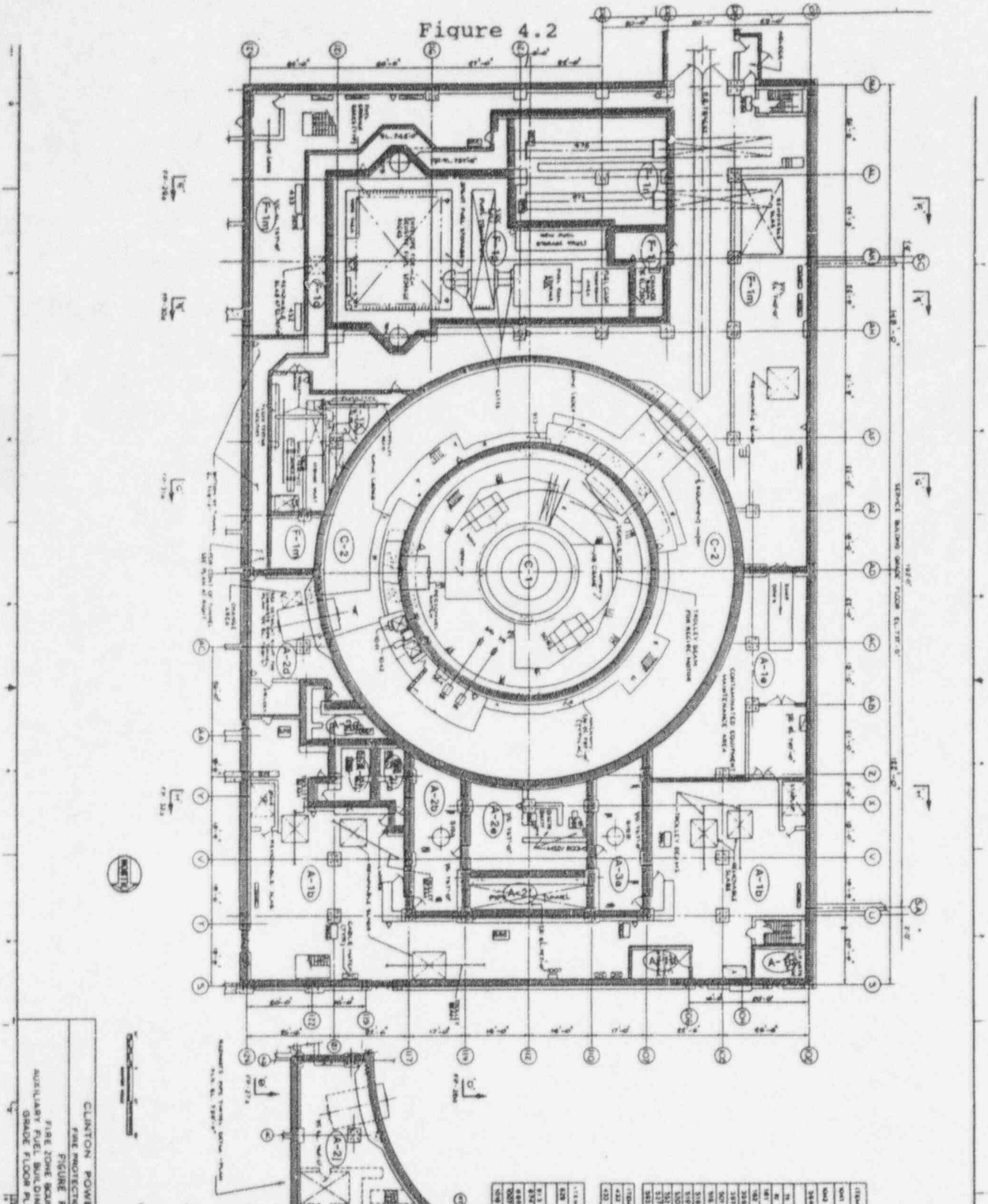


Figure 4.3

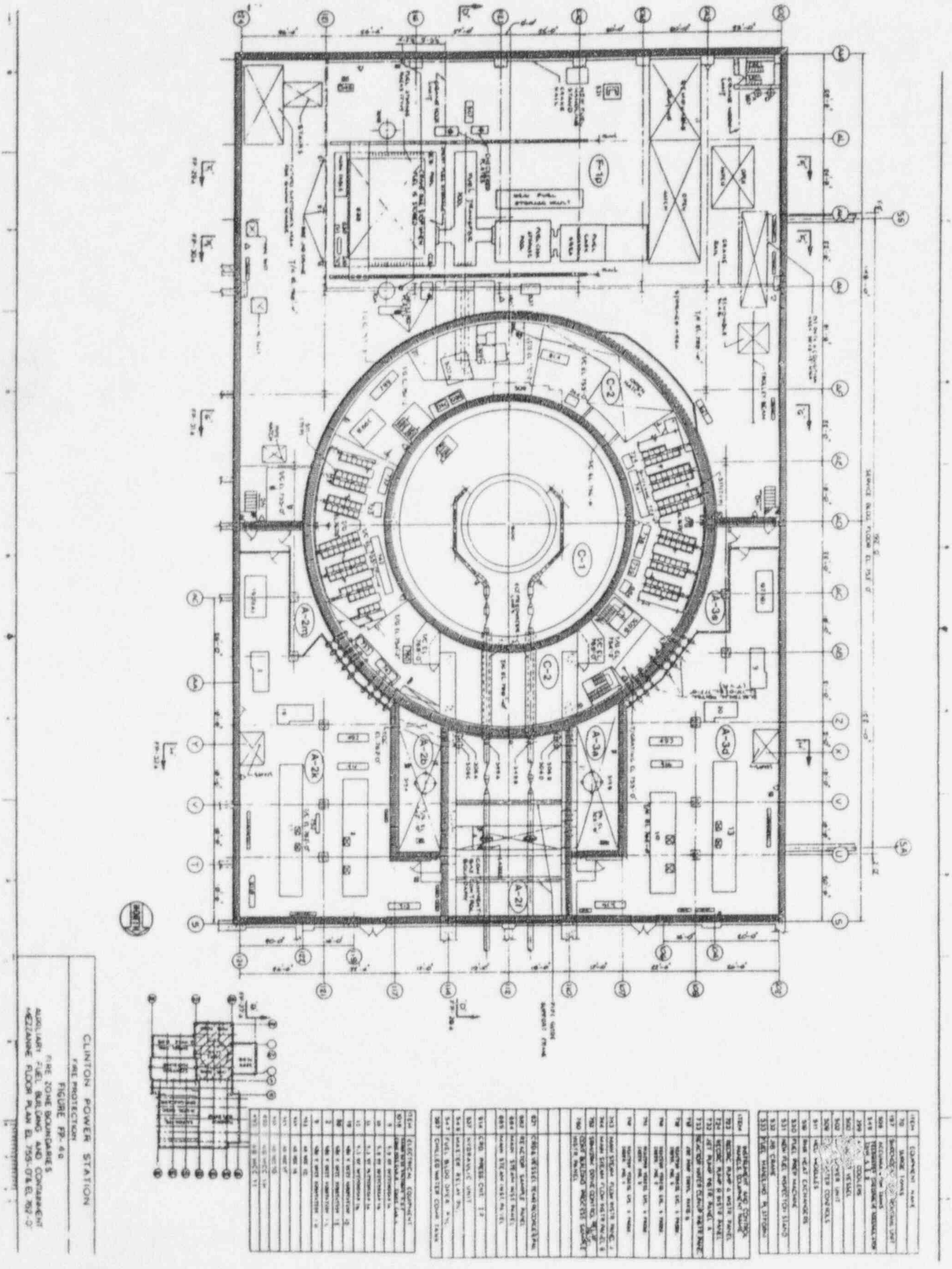


Figure 4.4

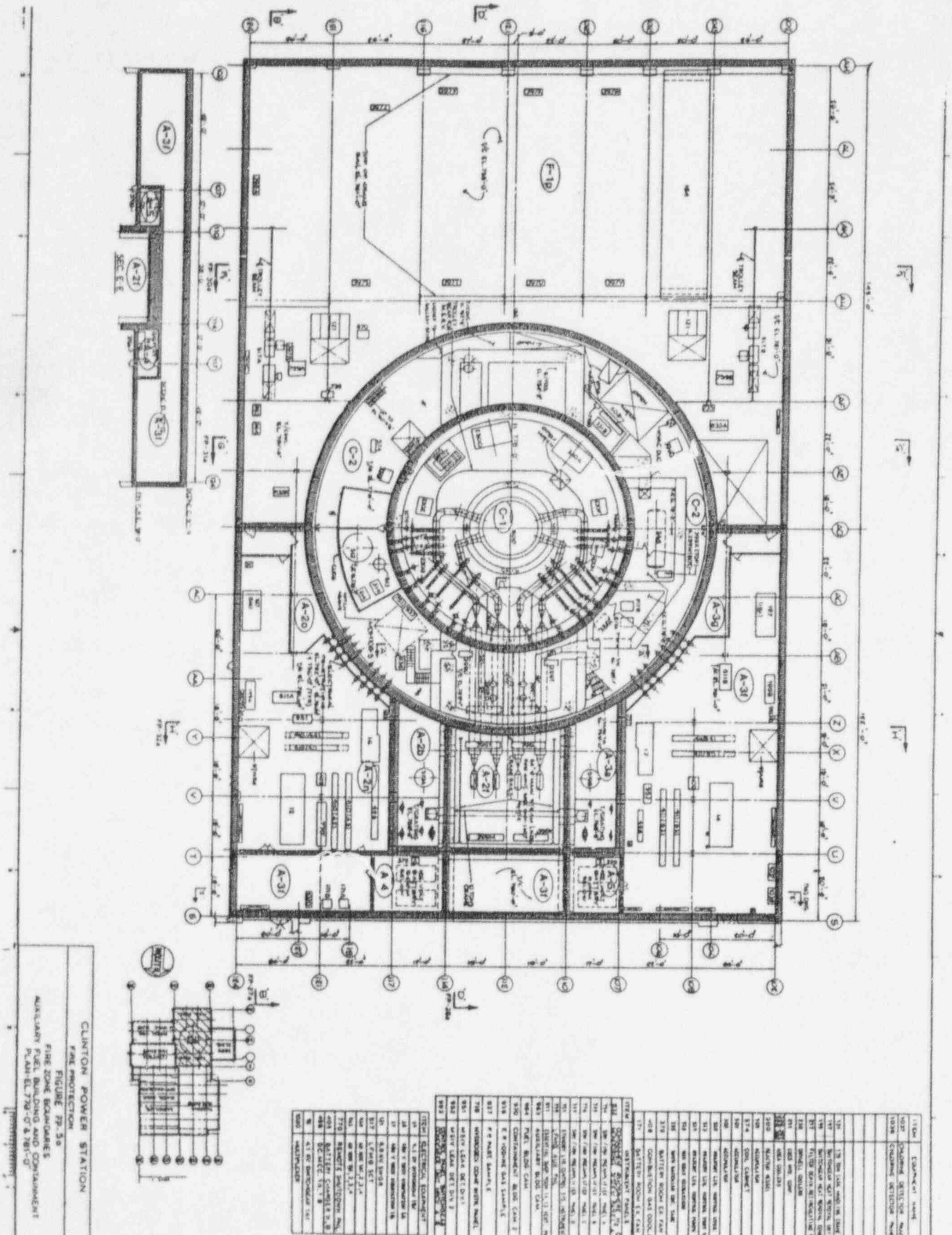


Figure 4.5

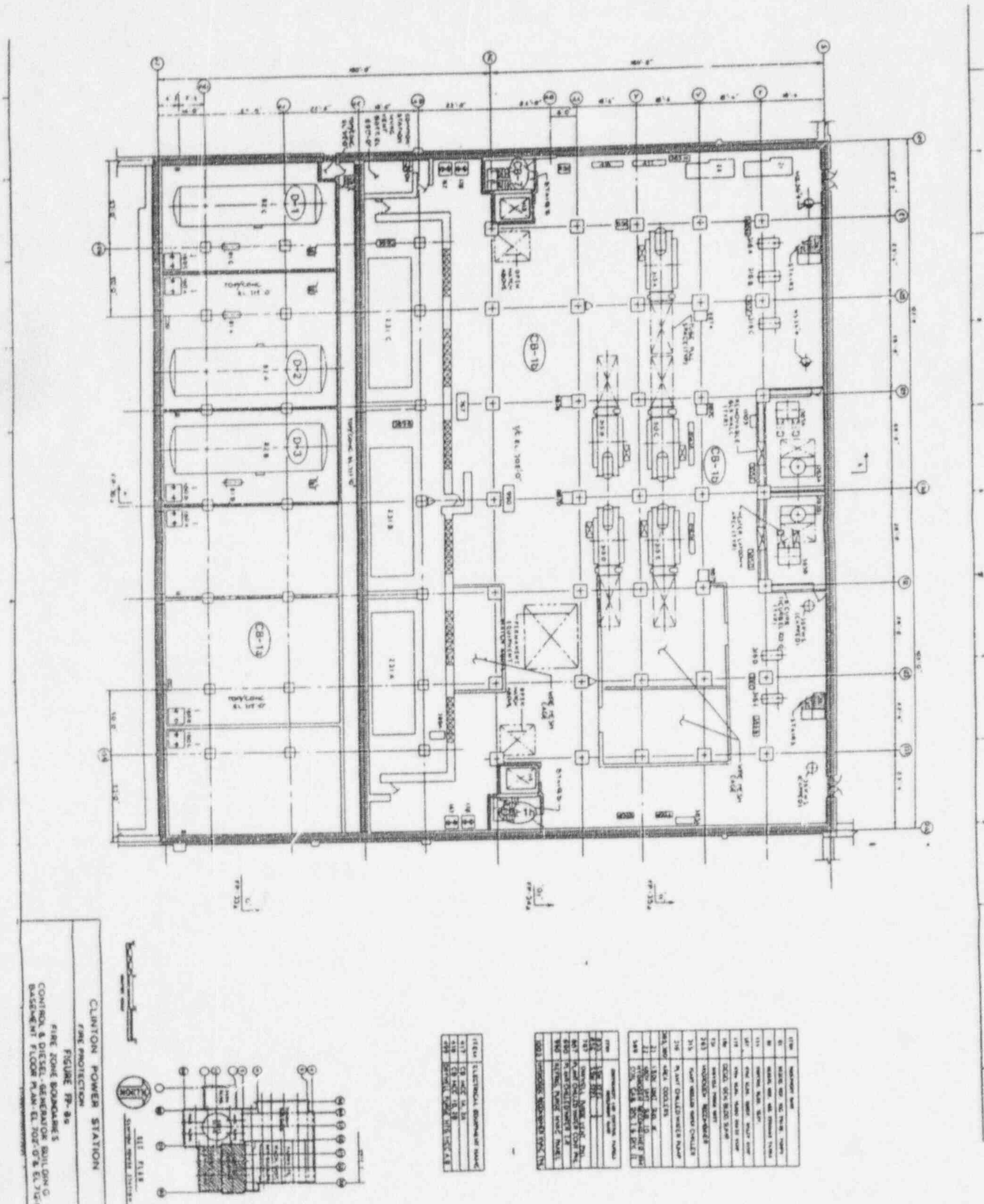

CB-1b
CB-1a
D-1
D-2
D-3
CLINTON POWER STATION
FIRE PROTECTION
FIGURE FP-8a
FIRE ZONE BOUNDARIES
CONTROL & DIESEL-GENERATOR BUILDING
BASEMENT FLOOR PLAN-EL 702'-0" & EL 712'-0"

Figure 4.6

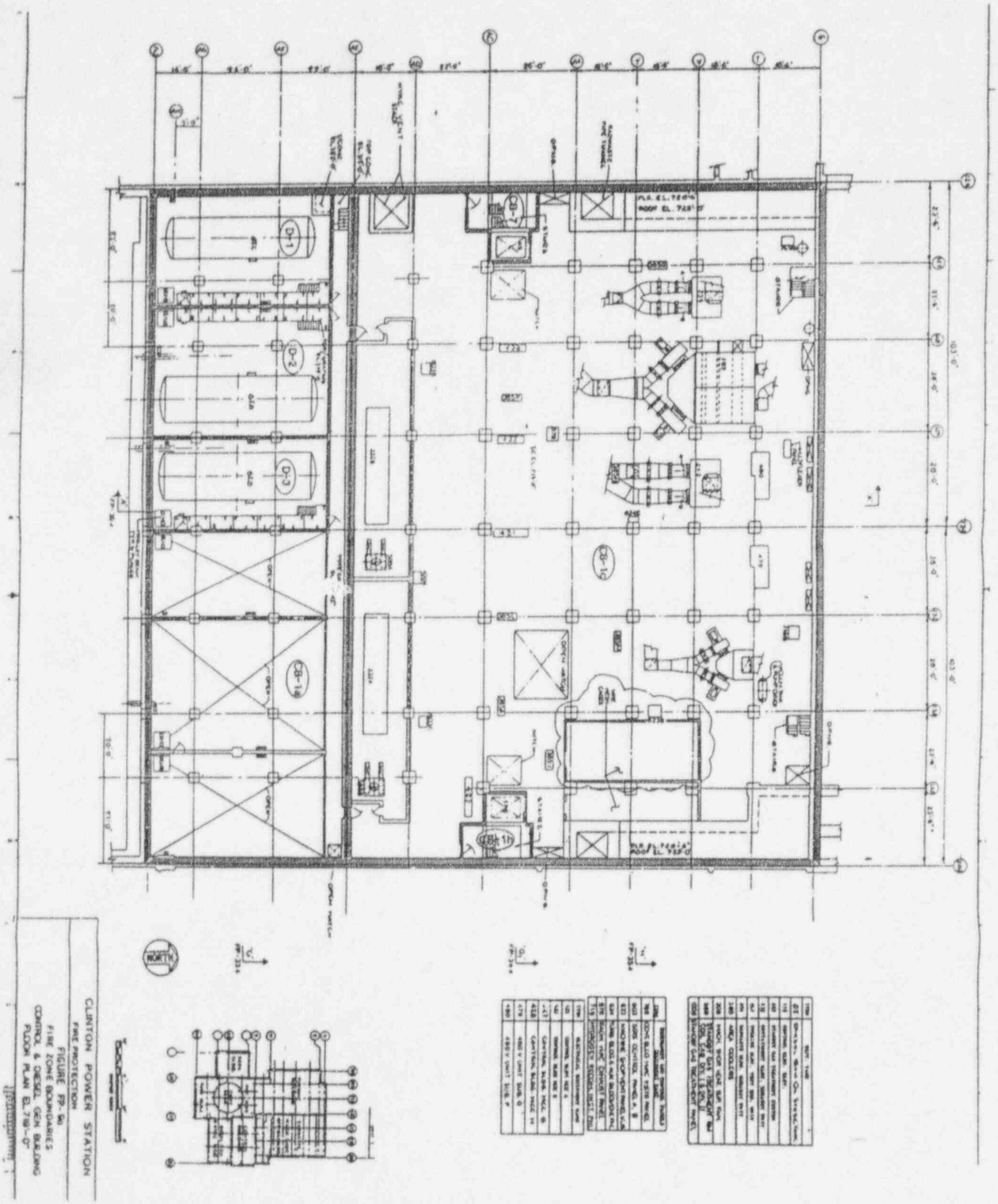

CLINTON POWER STATION
FIRE PROTECTION
FIGURE FP-56
FIRE ZONE BOUNDARIES
CONTROL & DIESEL GEN BUILDING
FLOOR PLAN EL 719'-0"
D-1
D-2
D-3
CB-1g
CB-1h
CB-1i
CB-7
NORTH

Figure 4.7

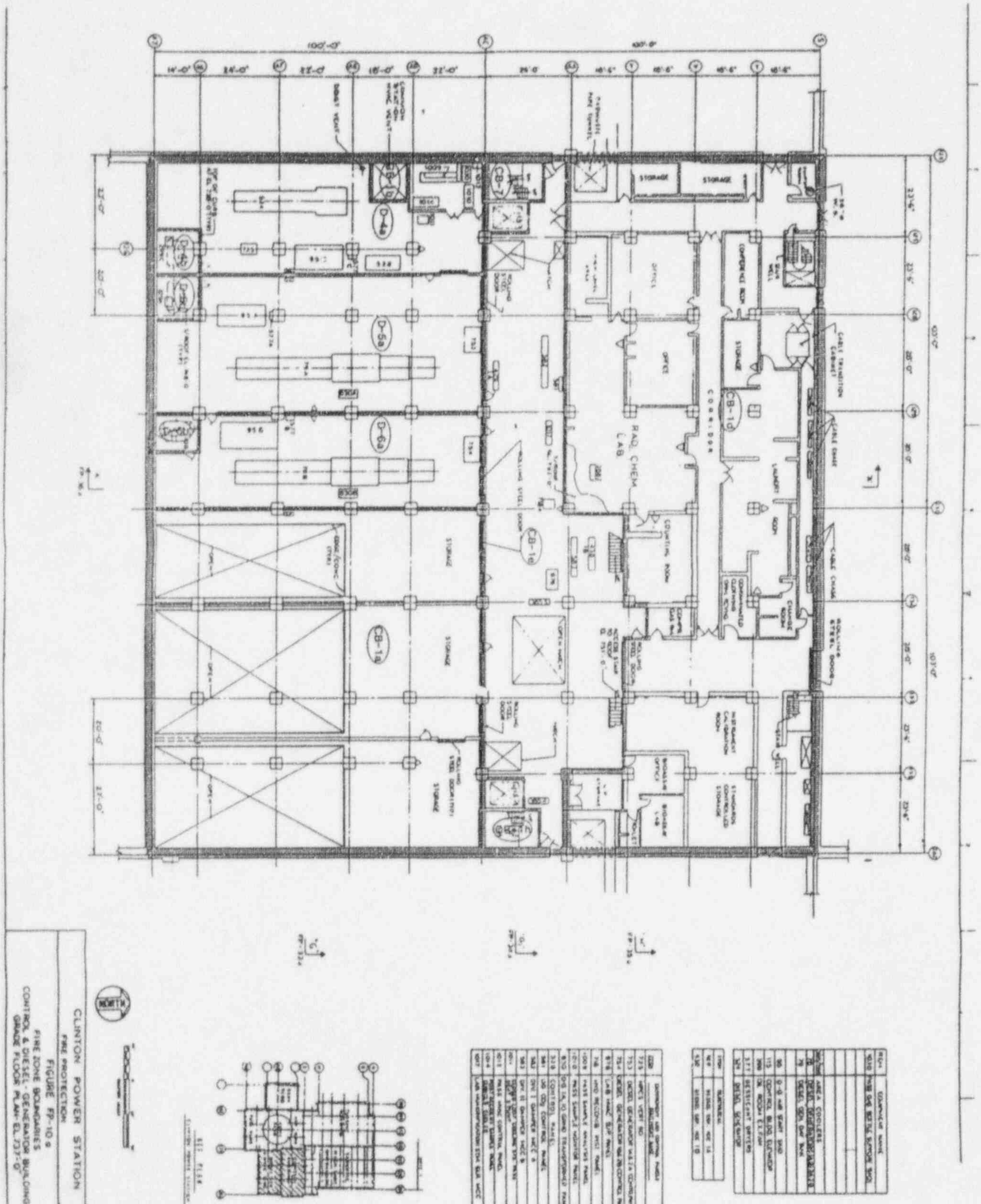


Figure 4.8

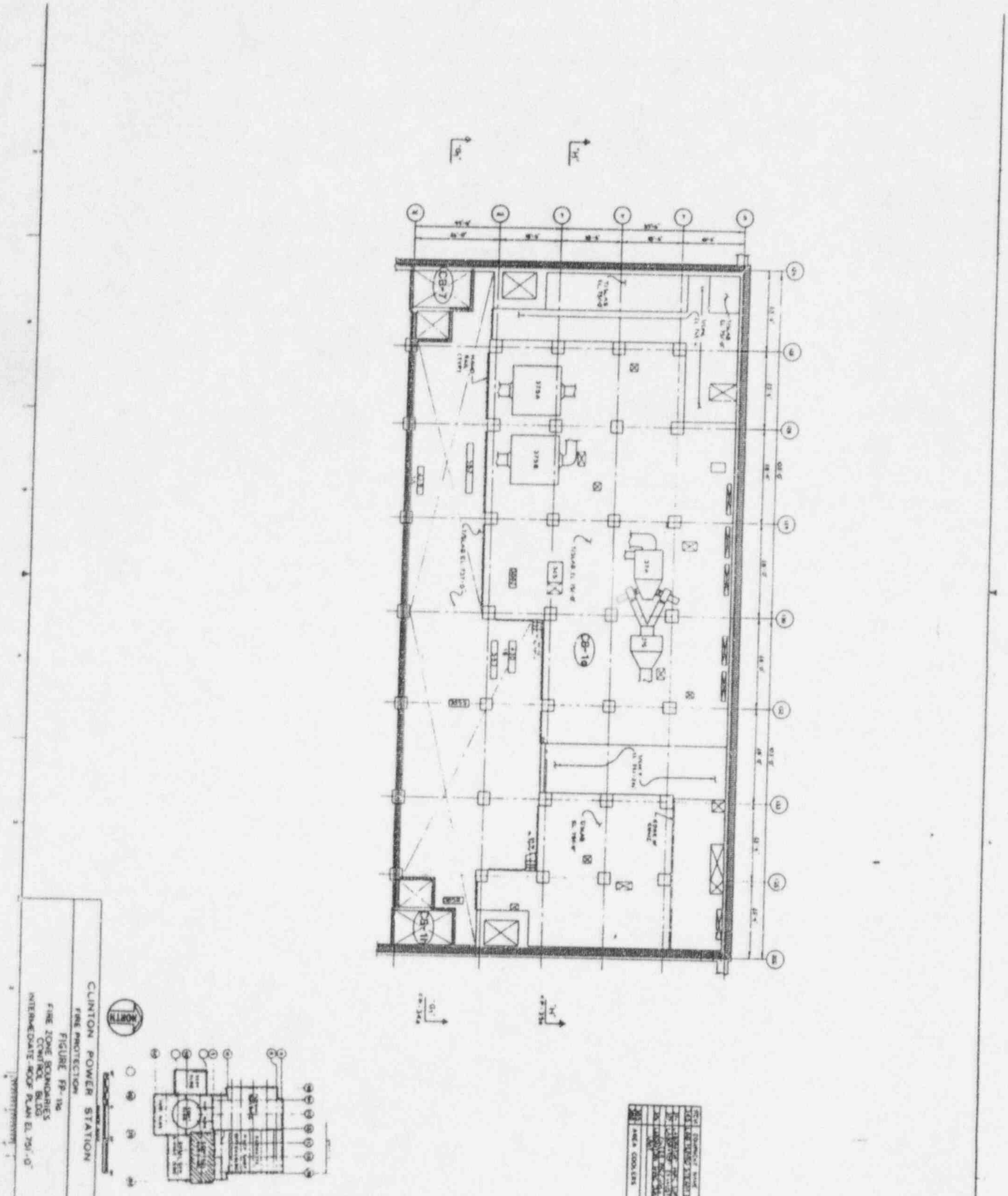


Figure 4.9

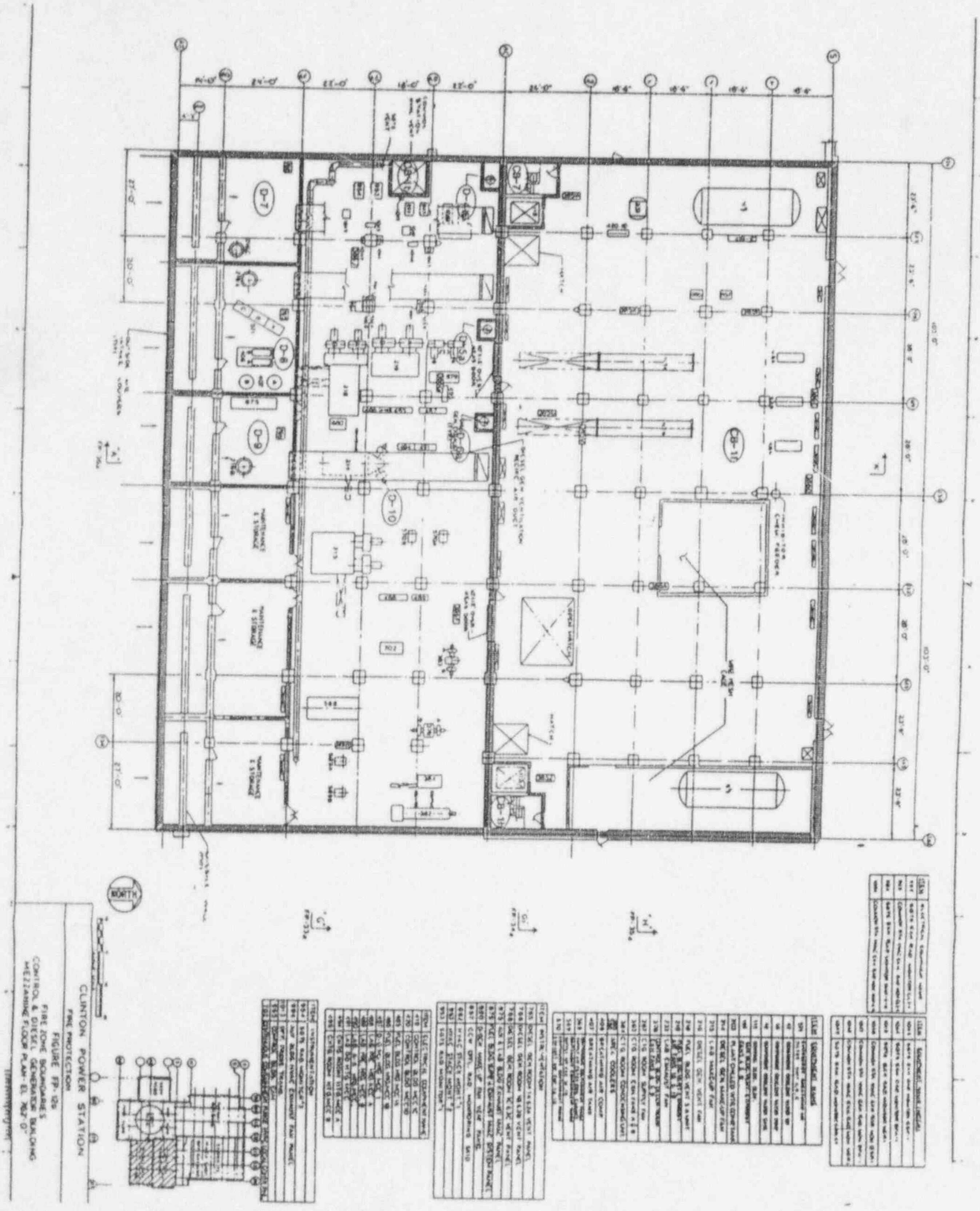


Figure 4.10

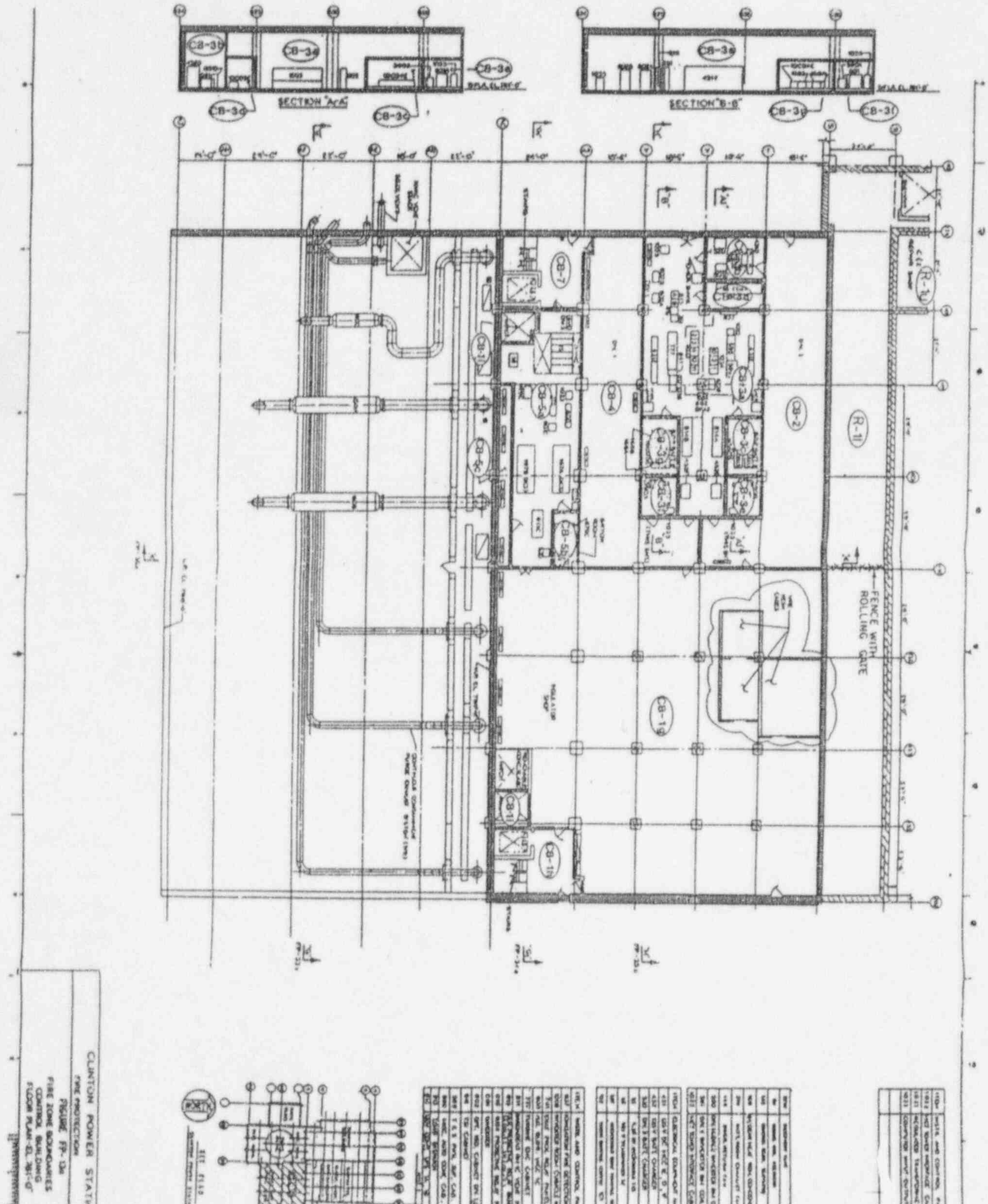

SECTION "A-A"
SECTION "B-B"
FENCE WITH ROLLING GATE
CLINTON POWER STATION

Figure 4.11

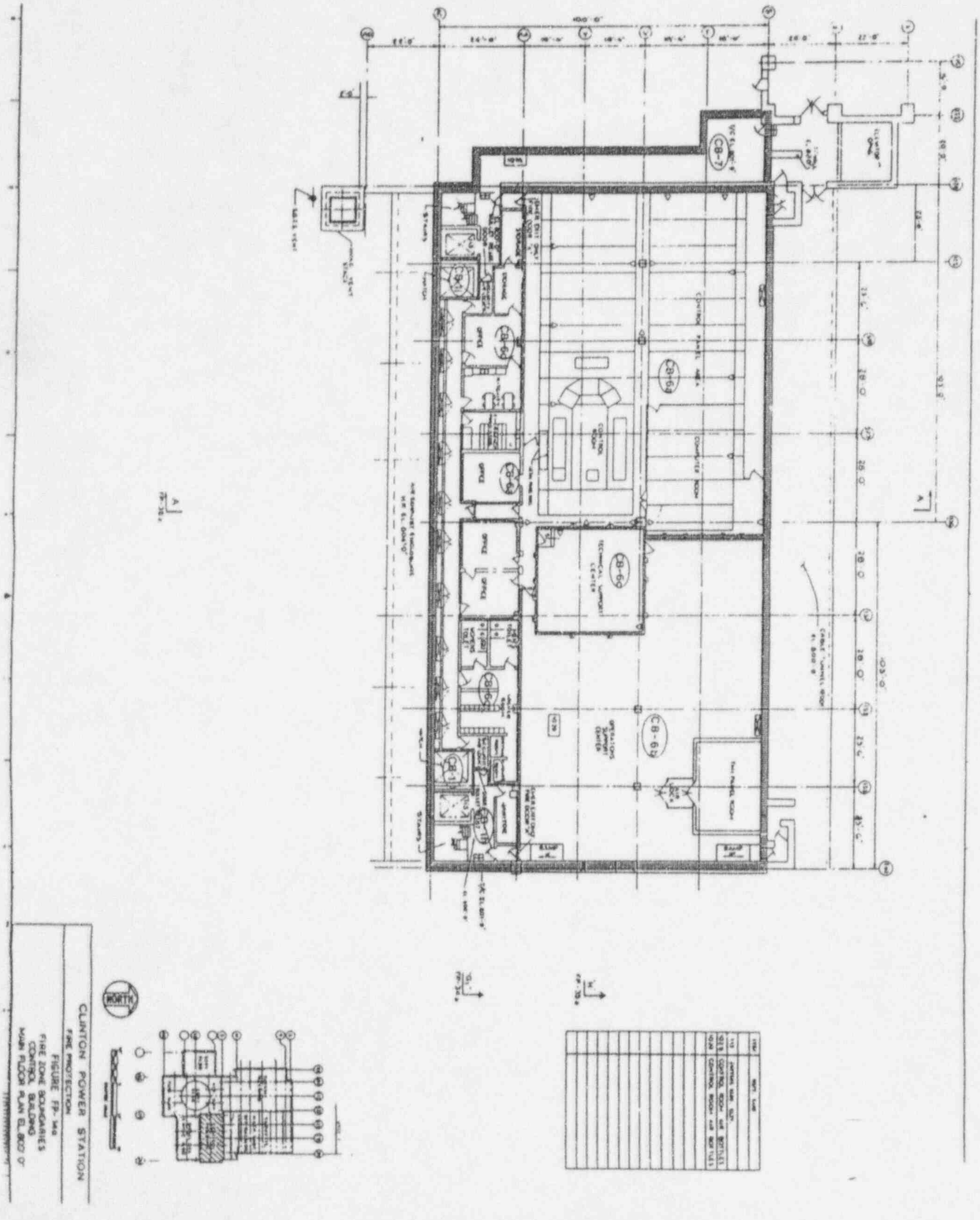


Figure 4.12

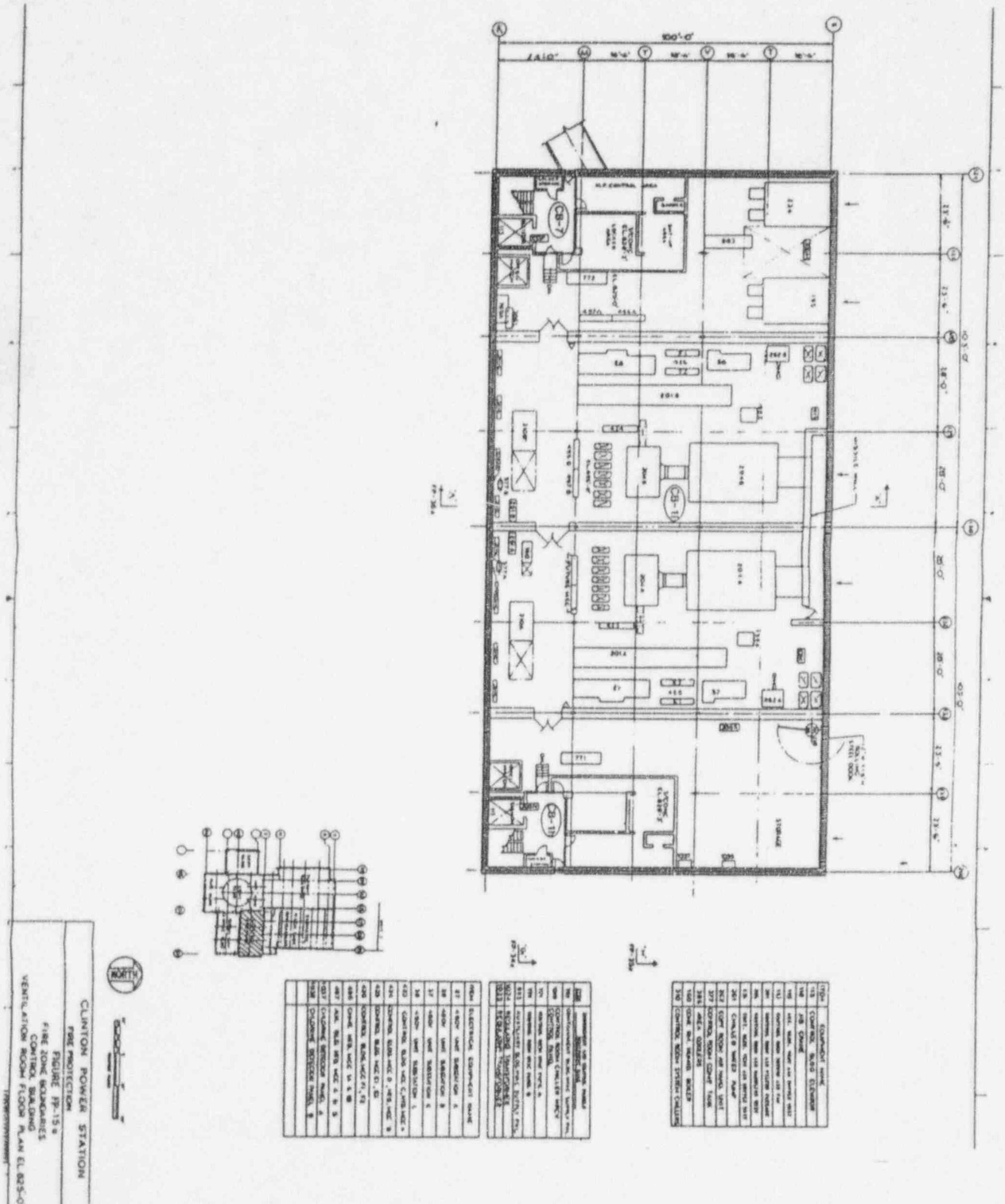

CLINTON POWER STATION
FIRE PROTECTION
FIGURE FP-15a
FIRE ZONE BOUNDARIES
CONTROL BUILDING
VENTILATION ROOM FLOOR PLAN EL.825'-0"

Figure 4.13

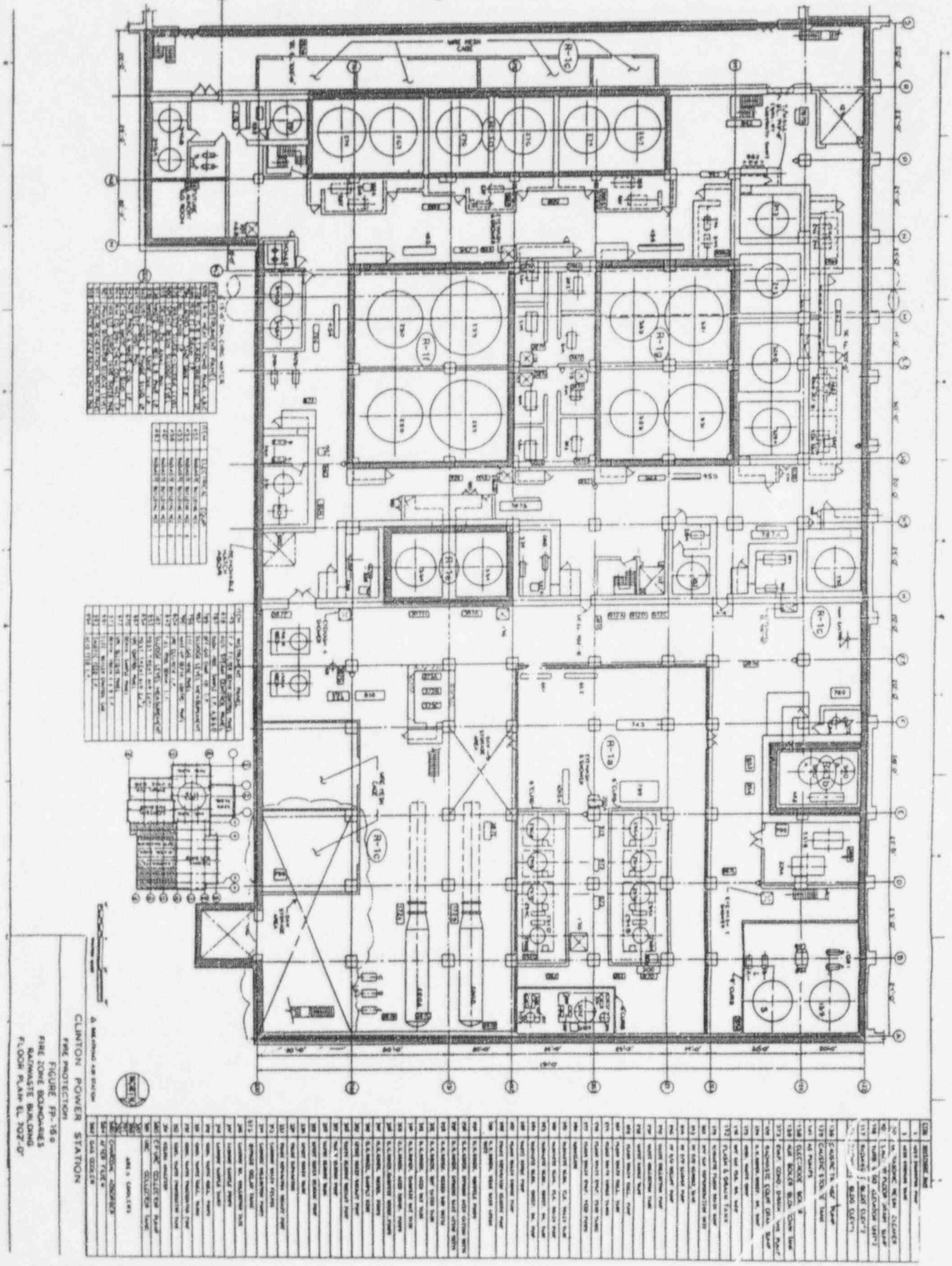

CLINTON POWER STATION
FIRE PROTECTION
FIGURE FP-16a
FIRE ZONE BOUNDARIES
RADWASTE BUILDING
FLOOR PLAN EL 702'-0"

Figure 4.14

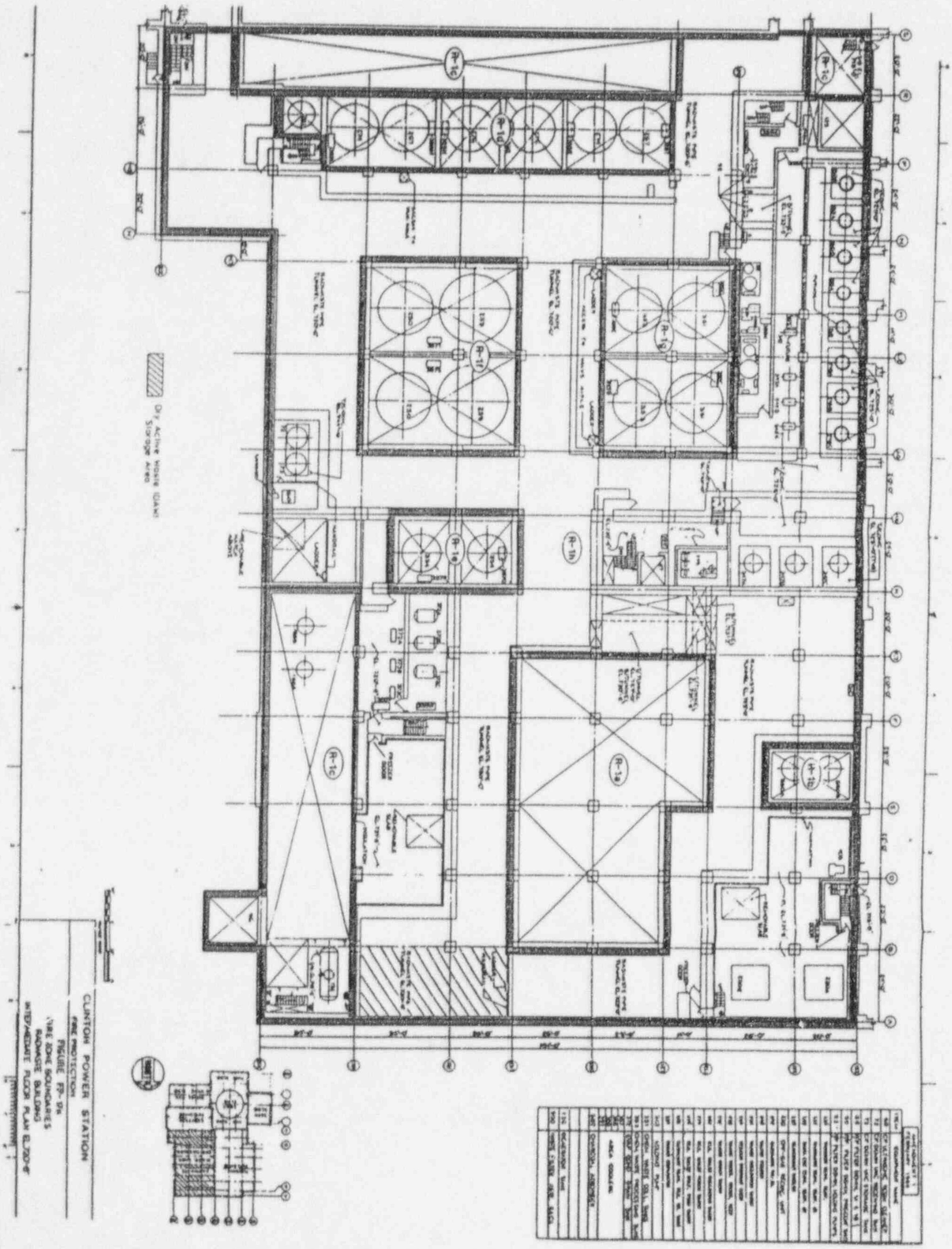


Figure 4.15

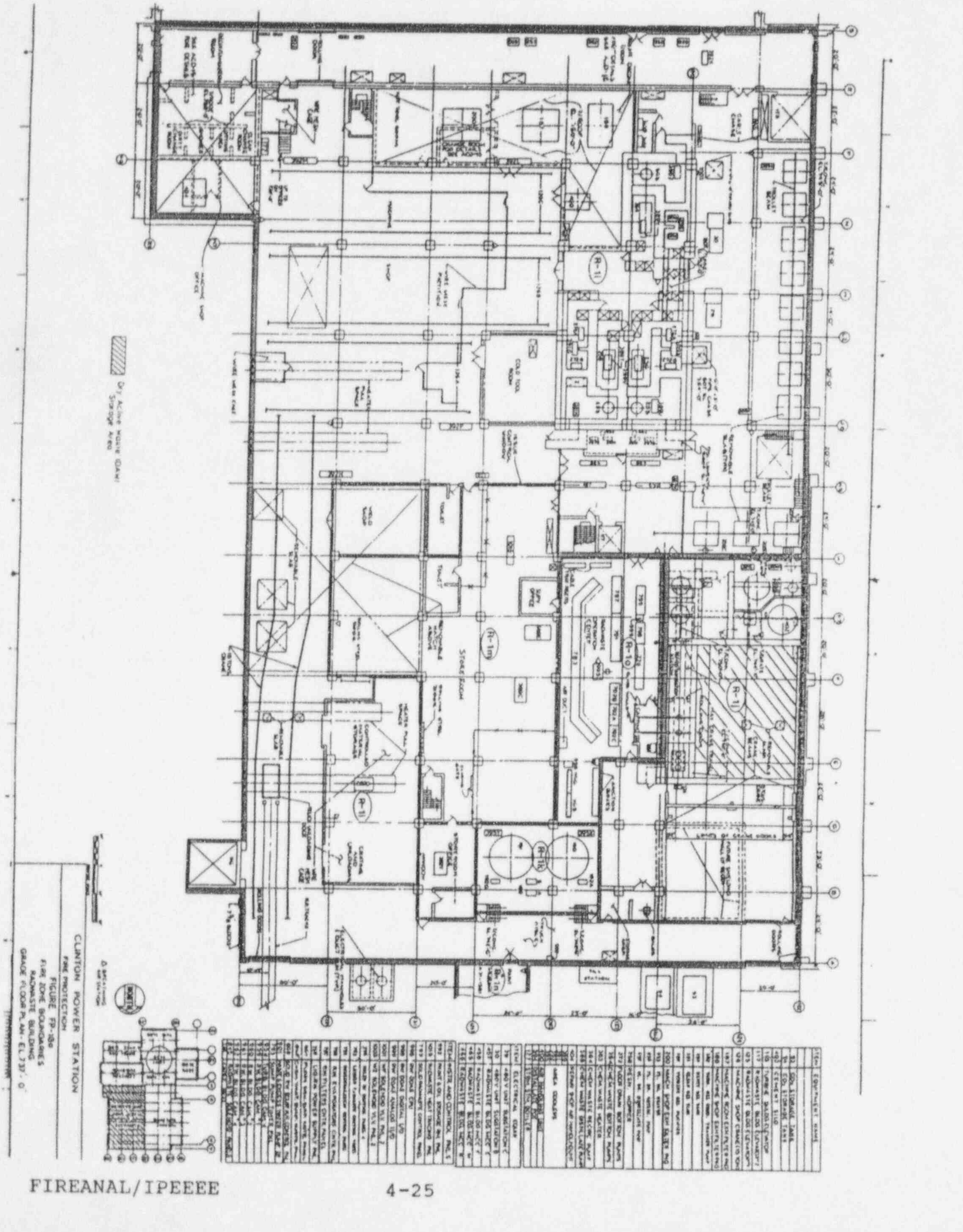


Figure 4.16

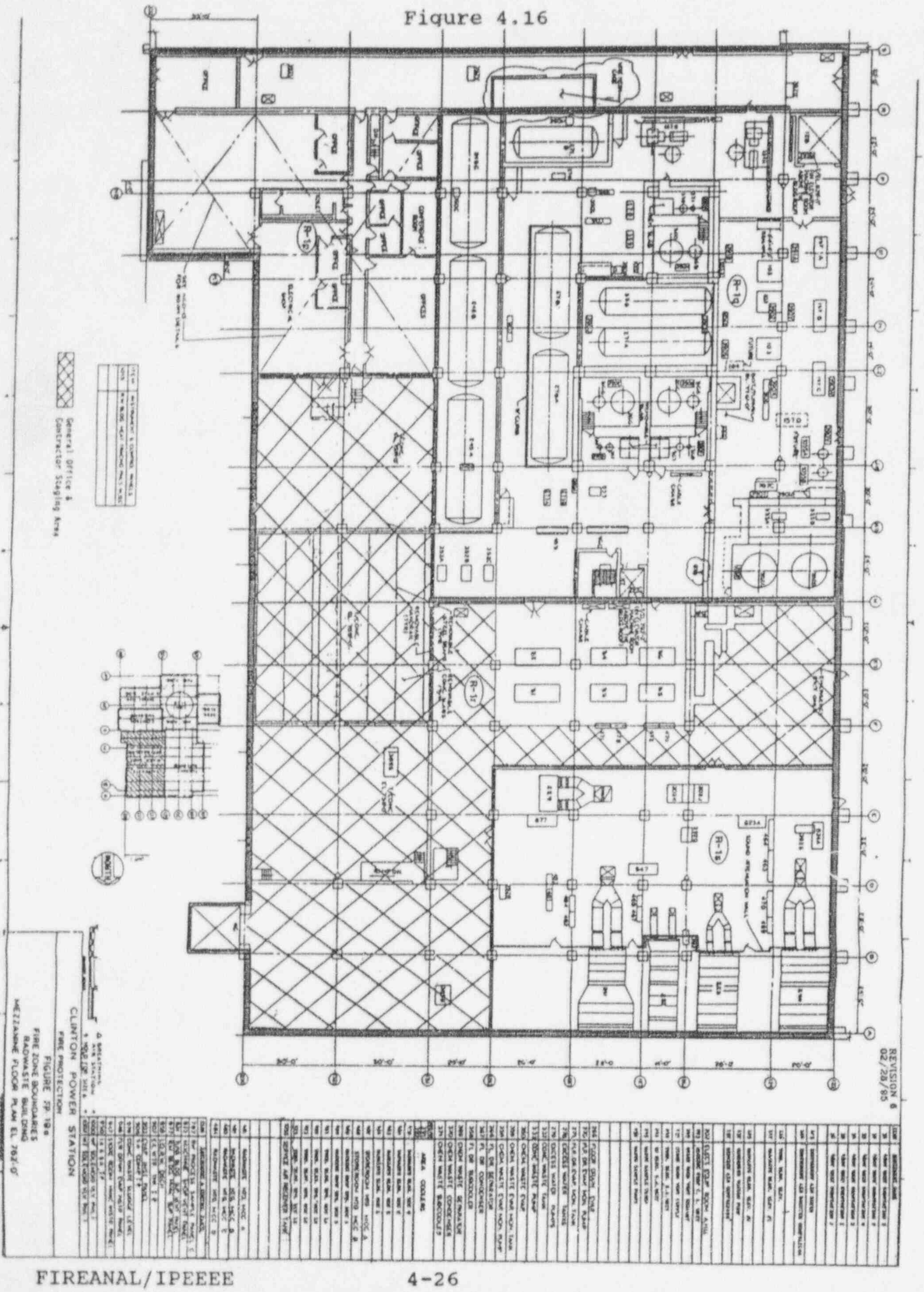

General Office & Contractor Staging Area
REVISION 6
02/28/85
CLINTON POWER STATION
FIRE PROTECTION
FIGURE FP-10a
FIRE ZONE BOUNDARIES
RADWASTE BUILDING
MEZZANINE FLOOR PLAN EL 762'-0"

Figure 4.17

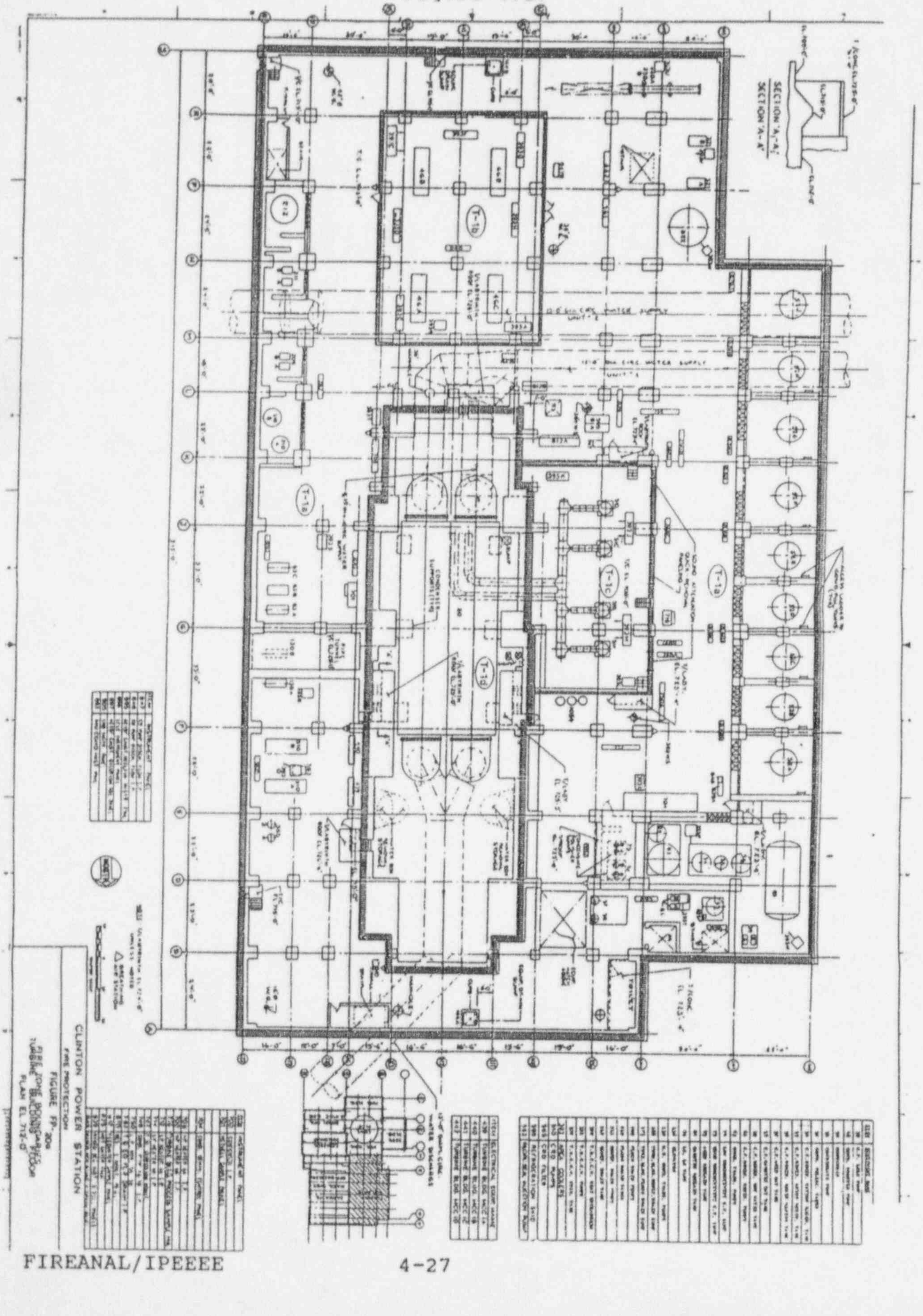


Figure 4.18

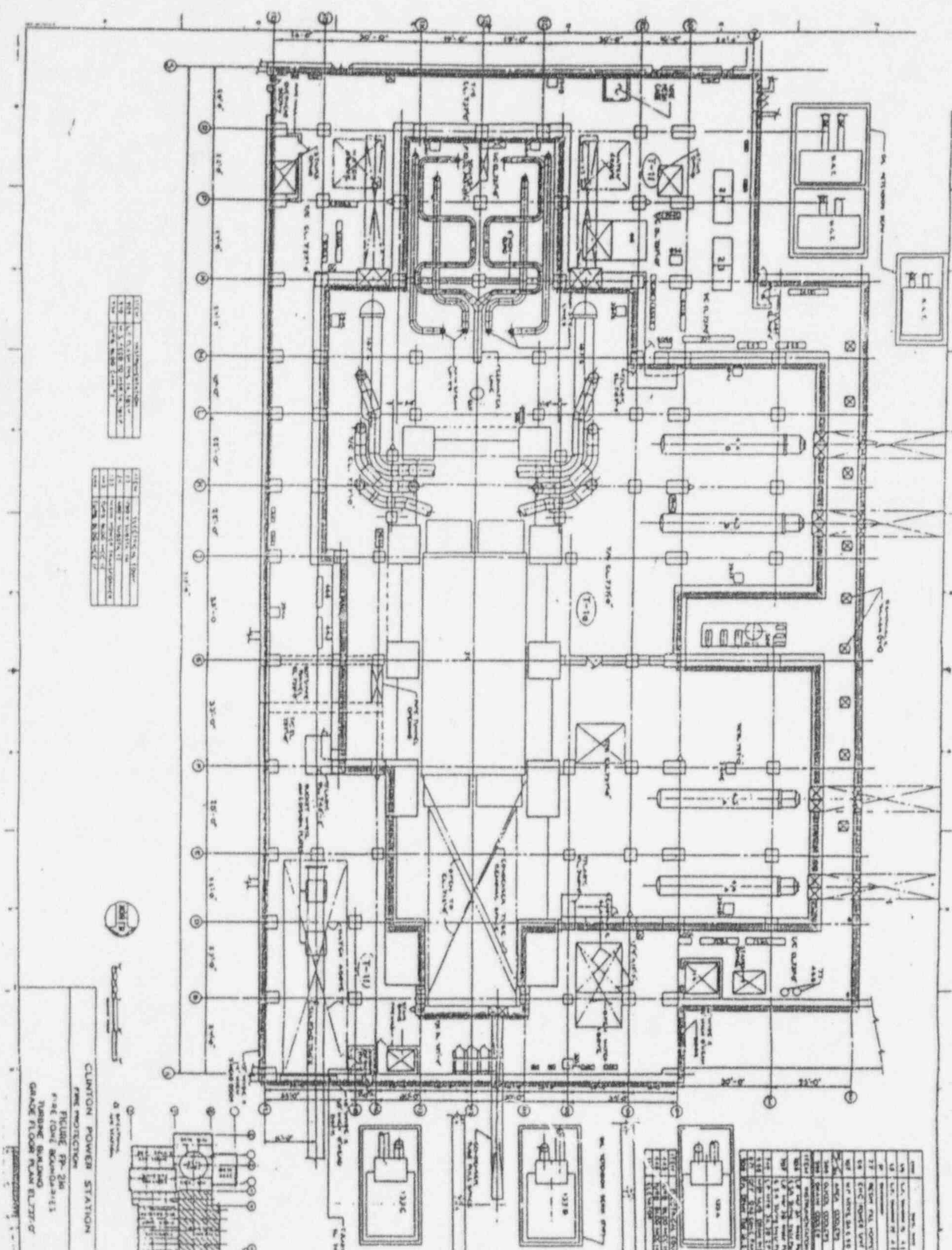


Figure 4.19

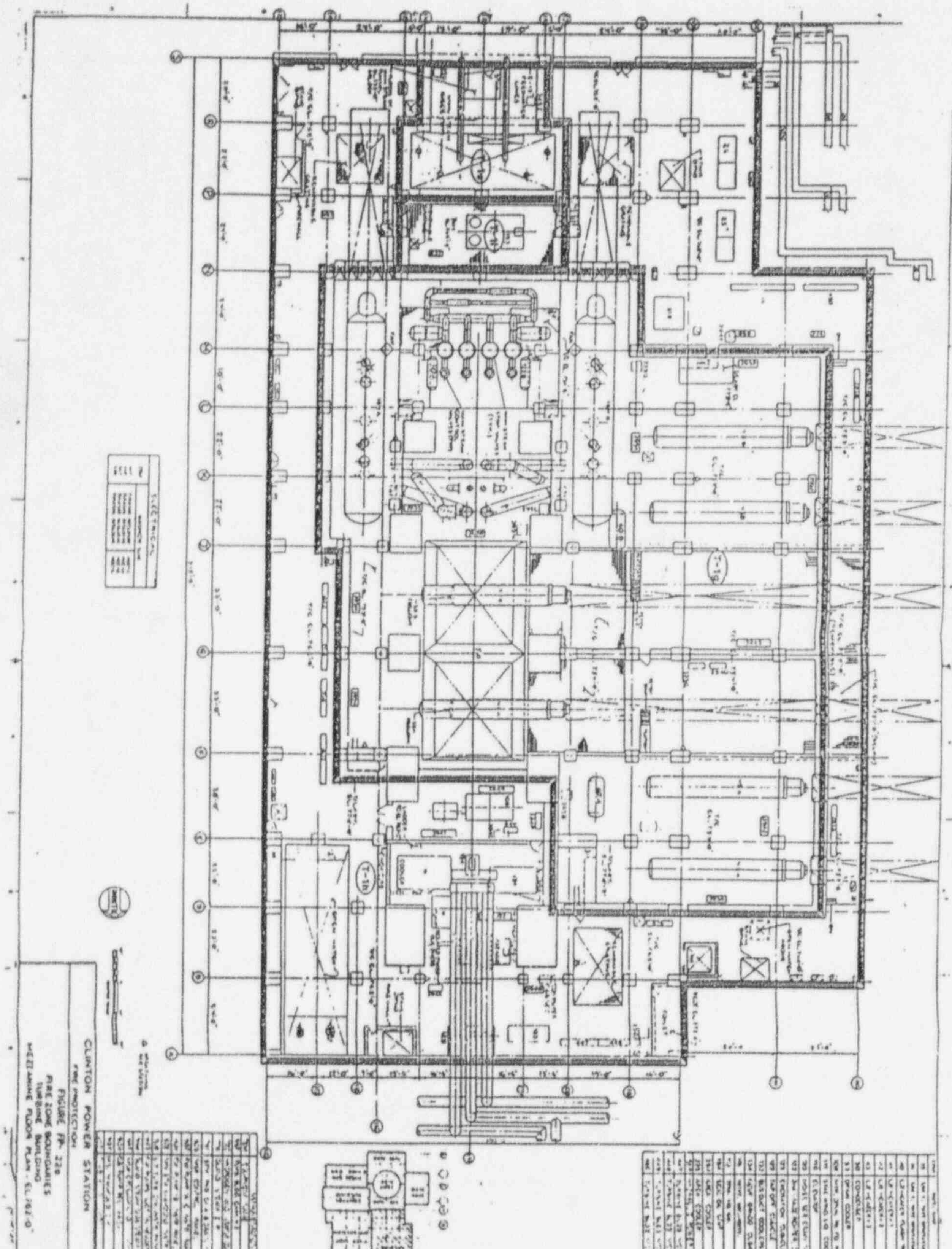


Figure 4.20

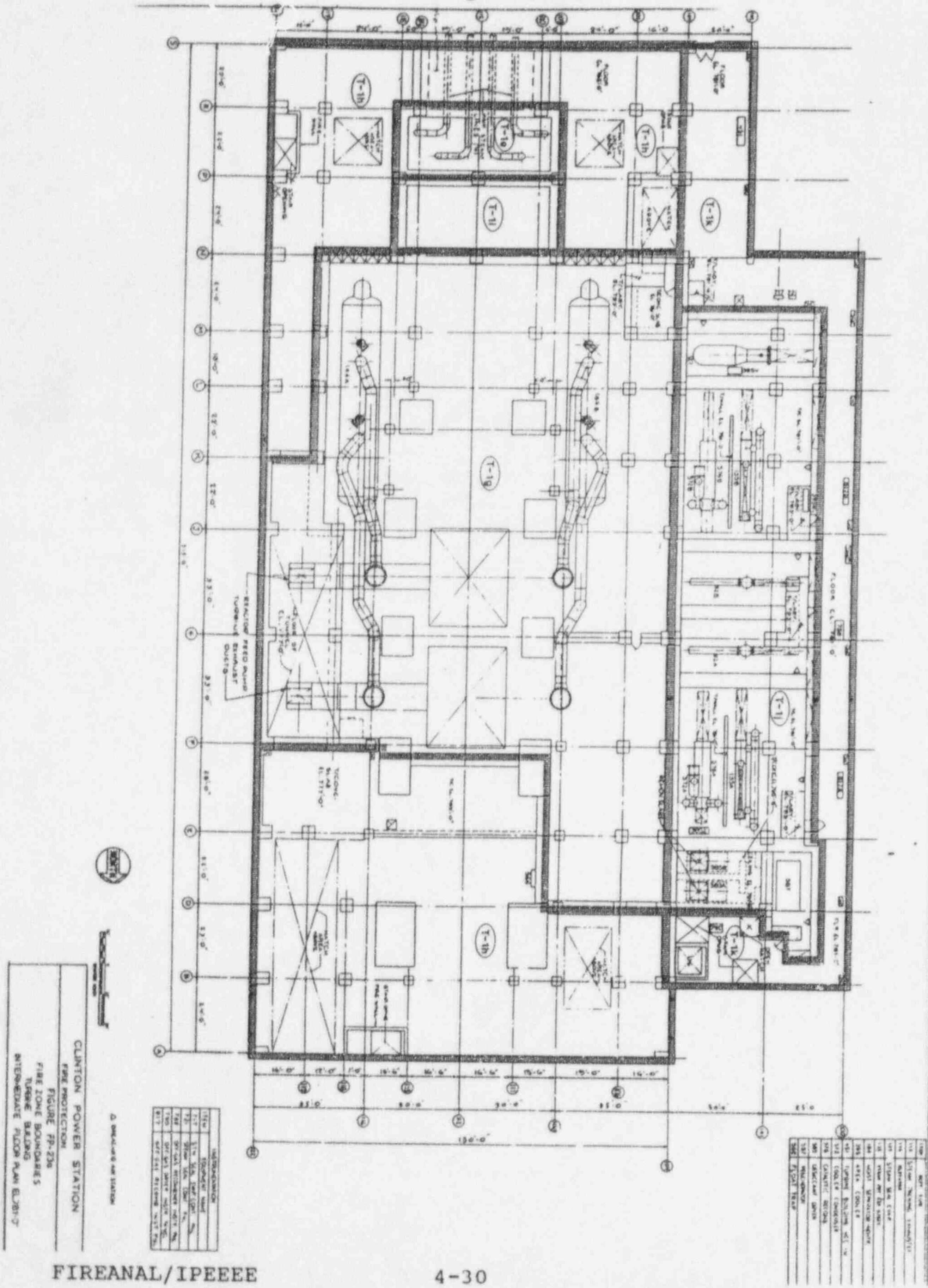


Figure 4.21

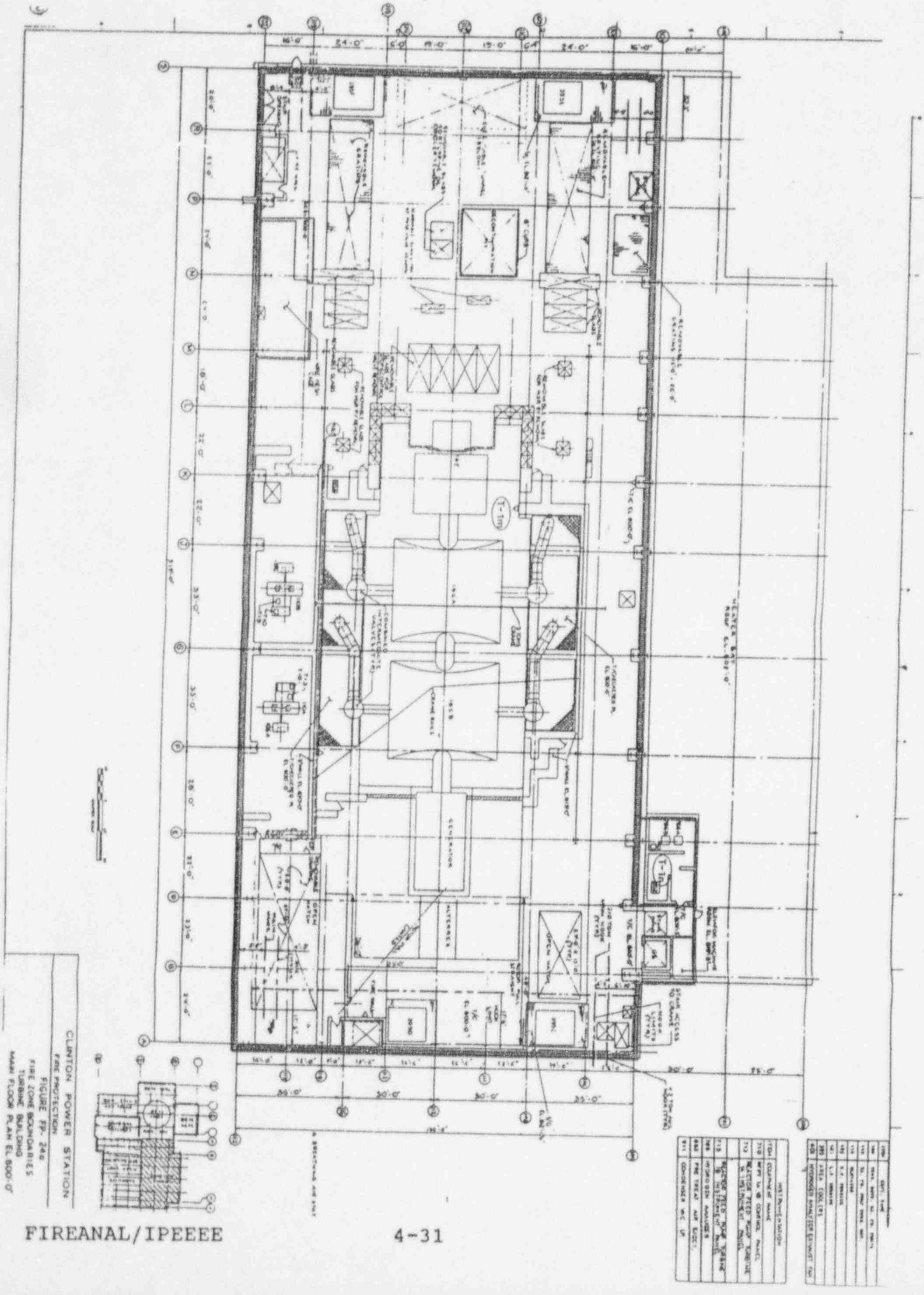

CLINTON POWER STATION
FIRE PROTECTION
FIGURE FP- 24a
FIRE ZONE BOUNDARIES
TURBINE BUILDING
MAIN FLOOR PLAN EL 800-0"

Figure 4.22

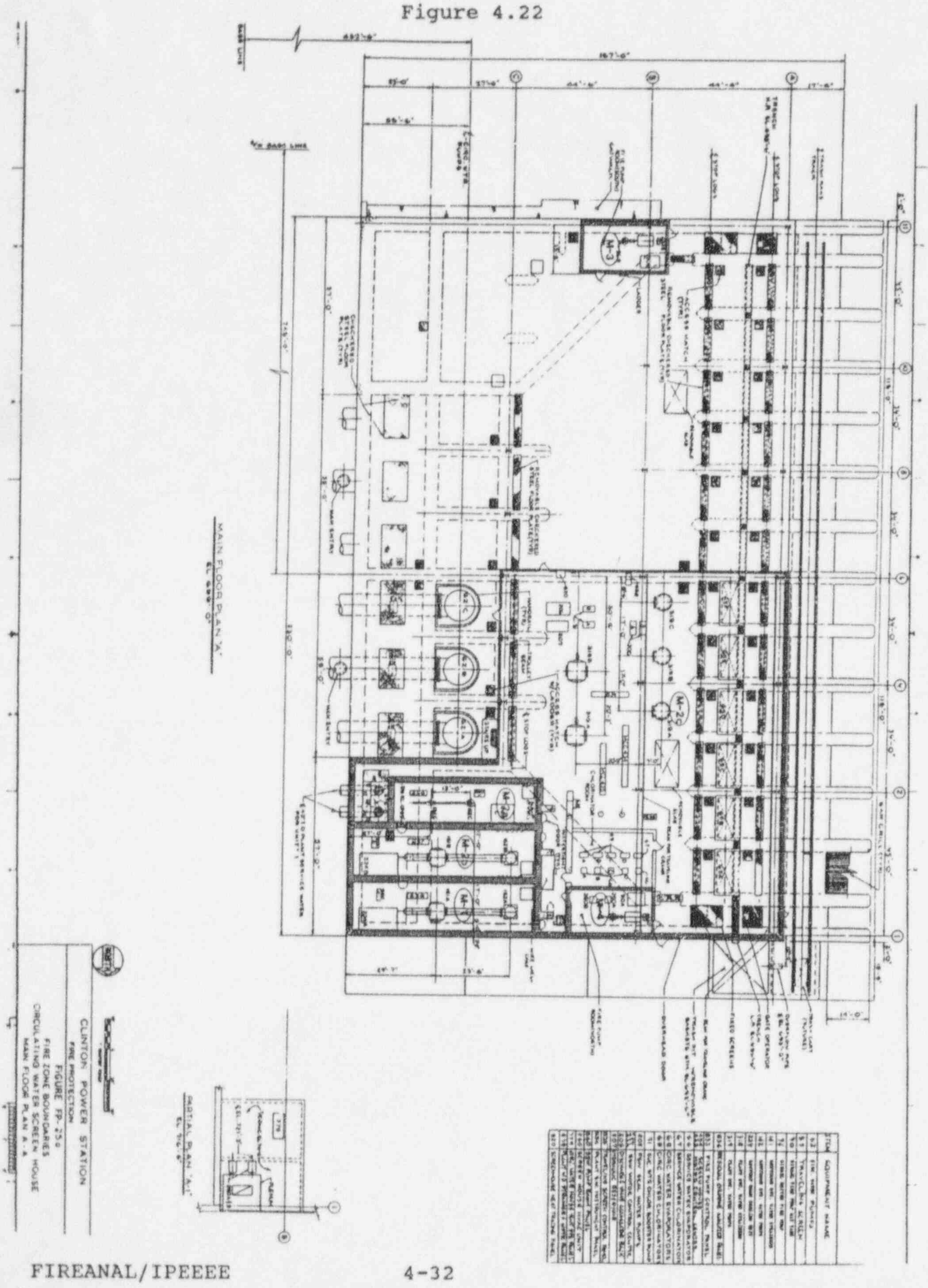
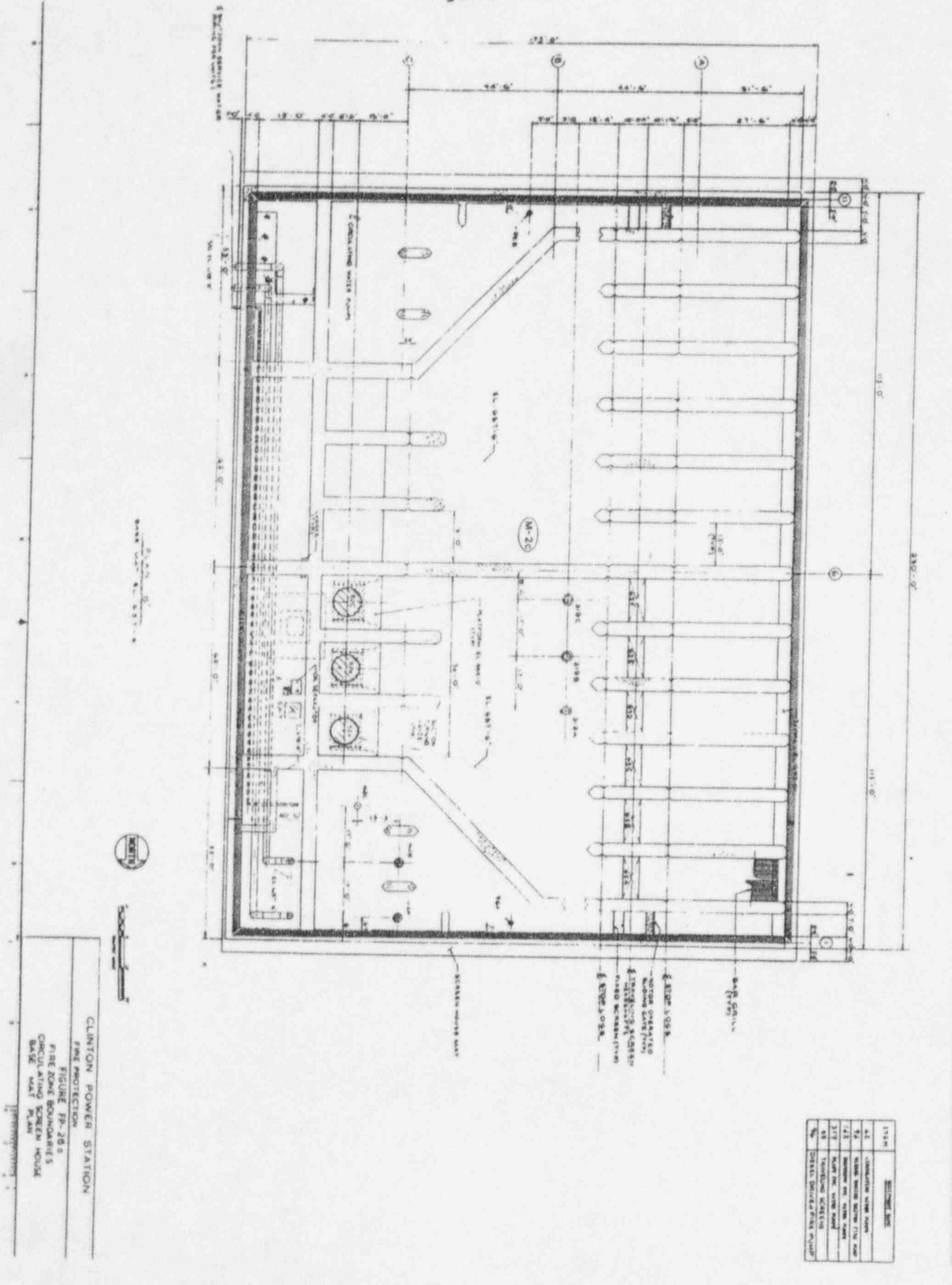

CLINTON POWER STATION
FIRE PROTECTION
FIGURE FP-2c
FIRE ZONE BOUNDARIES
CIRCULATING WATER SCREEN HOUSE
MAIN FLOOR PLAN A-A
MAIN FLOOR PLAN 'A'

Figure 4.23


M-2c
CLINTON POWER STATION
FIRE PROTECTION
FIGURE FP-26c
FIRE ZONE BOUNDARIES
CIRCULATING SCREEN HOUSE
BASE MAT PLAN

4.2 Review of Plant Information and Walkdowns

For firezones that were not screened from further analysis, the effects of specific fires within the zones required detailed evaluation. The impact of a fire on a particular piece of equipment or cable is a function of the ignition source, the potential energy releasable, cable/equipment characteristics and damage criteria, and the firezone geometry. The ignition sources can be either fixed or transient (can be located anywhere in the firezone). The majority of this information could only be obtained or verified by walkdowns of the affected firezones.

Prior to performing walkdowns, the IP FPRA analyst successfully completed a course on the overall FPRA methodology and detailed fire modeling presented by Scientific Applications International Corporation (SAIC). These courses lasted seven days and covered all aspects of fire screening analysis, walkdowns, fire modeling and CDF determination.

As was stated earlier, the FPRA was performed as part of an EPRI tailored collaboration. This arrangement allowed a greater degree of technology transfer to IP personnel than a typical utility - contractor arrangement by providing insights from all plants participating in the collaboration. The participating plants all had different start dates for their analyses which allowed following plants to gain additional benefit from the leading plants. CPS started significantly after the two lead plants in the collaboration (River Bend and Comanche Peak stations) which helped to identify potential difficulties in the screening, walkdown and modeling portions of the FPRA.

Walkdown sheets were developed prior to performing the walkdowns. These sheets contained all of the potential ignition sources previously identified from plant documentation. The sheets also contained fire damage range calculations for the different types of fire sources likely to be encountered in the firezone. These calculations required the determination of target damage temperatures and radiant heat fluxes, scenario heat loss factors, firezone ambient temperatures and the heat release rate (HRR) for different types of ignition sources.

The heat loss factor accounts for the fraction of heat energy absorbed into the floor, ceiling, and walls of the subject firezone. A value of 0.85 was utilized based on FPRA guidance.

The target damage and ignition temperatures selected from the FPRA methodology for the EPR-Hypalon IEEE-383 rated cables installed at CPS were a damage temperature of 700°F and an ignition temperature of 932°F. For Thermo-Lag 330-1, an ignition temperature of 1000°F was selected based on NSED Standard ME-08.00, Rev. 0. Note: No protection was credited for Thermo-Lag installations in the FPRA analysis.

A damage radiant heat flux of 1.0 BTU/s/ft2 was selected for IEEE-383 rated cable. For Thermo-Lag 330-1, an ignition heat flux of 2.2 BTU/s/ft2 was used.

The FPRA Implementation Guide provides experimentally determined HRRs for a variety of different ignition sources. Fire damage range calculations were performed for all ignition source types anticipated to be encountered in the walkdowns.

Preparations were made for walkdowns of the unscreened firezones by developing a thorough understanding of the evaluations and calculations that were to be performed based on the walkdown information. Also familiarizing the analyst with key issues of the FPRA methodology guidance for performing the walkdowns was an essential part of the walkdown preparations.

The walkdowns were initially performed by the IP fire PRA analyst and an FPRA and fire expert from the developers of the FPRA methodology (SAIC). After gaining experience in collecting fire relevant information with the guidance of the SAIC personnel, the IP FPRA analyst conducted numerous walkdowns using the equipment location/scenario information and ignition source screening forms. As a check to ensure the adequacy of the walkdowns, an IP fire protection engineer independently walked down four firezones and compared walkdown results with the FPRA analyst. The fire protection engineer had a bachelor of science degree in mechanical engineering and five years of experience in the fire protection area as well as having completed the FPRA methodology training class. The independent walkdowns found the FPRA walkdowns highly effective in collecting the location specific information required for the fire modeling process.

The FPRA analyst collected equipment type and location information on sources and targets during the walkdowns and compared these lists to equipment databases and drawings. This approach verified the as-built conditions of the firezones and provided a cross check to assure completeness of the equipment and ignition source evaluation. Separate walkdowns were conducted for the evaluation of fixed ignition sources and transient ignition sources as well as multi-compartment analysis. Thus, each firezone was walked down several times to collect information, verify information, and conduct scoping evaluations of ignition sources and targets.

This multiple walkdown approach was used for two main reasons. First, none of the firezones where walkdowns were performed were contamination areas and only 1 firezone was even posted as a radiation area. For the single firezone posted as a radiation area, walkdowns were performed to minimize dose to personnel. These factors allowed multiple walkdowns without violating ALARA principles. The second reason for multiple walkdowns was the tedious nature of the walkdowns themselves. For example, transient walkdowns required examining the entire floor space of each room and measuring the distance from each cable, conduit cable tray and piece of equipment to determine floor areas where a transient combustible could cause damage. This process had to be performed for items as high as 13 feet above the floor. This painstaking process was facilitated by performing the larger firezone walkdowns in sections at different times.

The evaluations for ignition sources that were accomplished as part of the walkdowns, included in the following activities:

1) determining the physical characteristics of the subject firezones including geometry and unique features that could affect heat transfer analysis,

2) identifying all fixed ignition sources and their spatial locations within the subject firezone,

3) determining targets, target damage and ignition criteria,

4) evaluating the impact of the plume and radiant heat on potential targets, and

5) collecting additional information for any potential fire propagation.

Prior to the walkdowns for transient ignition sources, procedures were reviewed, including combustible material control and housekeeping procedures, to identify limits that apply to various storage conditions and allowable amounts of combustible material. Also combustible load calculations were reviewed to identify what transient and stored combustibles could be expected. Interviews with personnel from plant fire protection were conducted to determine whether additional considerations and conditions should be included in the walkdowns. Layouts of the firezones and adjacent zones were examined to determine what combustibles might traverse a firezone.

Walkdowns for transient ignition sources were conducted in accordance with guidelines provided in the EPRI FPRA methodology. The information collected during the walkdowns included potential transient combustible types, storage method, quantities present, handling methods, and physical location. Emphasis was placed on the presence of oil within the firezones either as a transient or fixed combustible.

Table 4.2

Firezone Screening Results

ZONE	CCDP	ISOLATION FAILURE POTENTIAL	IGNITION FREQUENCY	SCREENING CDF	MODELING REQUIRED
A-1a	1.23E-02	N/A	1.6E-03	1.97E-05	YES
A-1b	1.14E-01	N/A	8.3E-04	9.46E-05	YES
A-1c	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
A-1d	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
A-1e	<2.0E-07	N/A	7.4E-04	1.48E-10	NO
A-2a	<2.0E-07	N/A	2.0E-03	4.0E-10	NO
A-2b	4.64E-05	N/A	1.6E-03	7.42E-08	NO
A-2c	<2.0E-07	N/A	2.5E-03	5.0E-10	NO
A-2d	<2.0E-07	N/A	6.3E-04	1.26E-10	NO
A-2e	<2.0E-07	N/A	9.5E-04	1.90E-10	NO
A-2f	4.89E-05	N/A	6.5E-04	3.18E-08	NO
A-2g	<2.0E-07	N/A	7.9E-04	1.58E-10	NO
A-2h	<2.0E-07	N/A	7.9E-04	1.58E-10	NO
A-2i	<2.0E-07	N/A	7.9E-04	1.58E-10	NO
A-2j	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
A-2k	1.52E-03	N/A	3.3E-03	5.02E-06	YES
A-2m	9.25E-06	N/A	4.1E-04	3.79E-09	NO
A-2n	1.66E-01	N/A	5.8E-03	9.63E-04	YES
A-2o	6.25E-07	N/A	4.1E-04	2.56E-10	NO
A-3a	3.06E-06	N/A	1.7E-03	5.20E-09	NO
A-3b	<2.0E-07	N/A	1.5E-03	3.0E-10	NO
A-3c	<2.0E-07	N/A	5.7E-04	1.14E-10	NO
A-3d	4.01E-03	N/A	3.3E-03	1.32E-05	YES
A-3e	5.36E-07	YES	5.8E-04	3.11E-10	NO
A-3f	1.12E-03	N/A	5.2E-03	5.82E-06	YES
A-3g	<2.0E-07	N/A	4.1E-04	8.2E-11	NO
A-4	<2.0E-07	N/A	9.4E-04	1.88E-10	NO

Table 4.2 (Cont'd)

Firezone Screening Results

ZONE	CCDP	ISOLATION FAILURE POTENTIAL	IGNITION FREQUENCY	SCREENING CDF	MODELING REQUIRED
A-5	<2.0E-07	N/A	9.3E-04	1.86E-10	NO
CB-1a	<2.0E-07	N/A	2.4E-03	4.8E-10	NO
CB-1b	5.50E-06	NO	5.5E-02	3.03E-07	NO
CB-1c	5.10E-04	N/A	7.6E-03	3.88E-06	YES
CB-1d	5.10E-04	N/A	1.0E-02	5.10E-06	YES
CB-1e	8.83E-02	N/A	9.6E-03	8.48E-04	YES
CB-1f	3.39E-01	N/A	6.4E-03	2.17E-03	YES
CB-1g	<2.0E-07	N/A	4.9E-04	9.8E-11	NO
CB-1h	<2.0E-07	N/A	1.5E-03	3.0E-10	NO
CB1i-E	4.08E-06	NO	7.1E-03	2.90E-08	NO
CB1i-W	3.86E-05	NO	8.6E-03	3.32E-07	NO
CB-2	1.61E-01	N/A	4.0E-04	6.44E-05	YES
CB-3a	1.11E-01	N/A	4.5E-03	5.00E-04	YES
CB-3b	6.94E-05	N/A	1.3E-03	9.02E-08	NO
CB-3c	1.53E-06	N/A	4.0E-04	6.12E-10	NO
CB-3d	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
CB-3e	<2.0E-07	N/A	5.2E-04	1.04E-10	NO
CB-3f	1.26E-06	N/A	5.2E-04	6.55E-10	NO
CB-3g	1.53E-06	N/A	4.3E-04	6.58E-10	NO
CB-4	3.04E-01	N/A	4.0E-04	1.22E-04	YES
CB-5a	2.27E-02	N/A	2.3E-03	5.22E-05	YES
CB-5b	<2.0E-07	N/A	4.1E-04	8.2E-11	NO
CB-5c	1.96E-03	NO	4.0E-04	7.84E-07	NO
CB-6a	1.00E-00	YES	1.0E-02	1.0E-02	YES
CB-6b	<2.0E-07	N/A	8.7E-04	1.74E-10	NO
CB-6c	<2.0E-07	N/A	4.6E-04	9.20E-11	NO
CB-6d	6.44E-04	YES	4.1E-04	2.64E-07	YES

Table 4.2 (Cont'd)

Firezone Screening Results

ZONE	CCDP	ISOLATION FAILURE POTENTIAL	IGNITION FREQUENCY	SCREENING CDF	MODELING REQUIRED
CB-7	1.80E-05	N/A	1.9E-03	3.42E-08	NO
D-1	<2.0E-07	N/A	1.6E-03	3.2E-10	NO
D-10	<2.0E-07	N/A	5.4E-03	1.08E-09	NO
D-2	<2.0E-07	N/A	1.6E-03	3.2E-10	NO
D-3	<2.0E-07	N/A	1.6E-03	3.2E-10	NO
D-4a	6.25E-07	N/A	9.9E-03	6.19E-09	NO
D-4b	<2.0E-07	N/A	4.1E-04	8.2E-11	NO
D-5a	<2.0E-07	N/A	9.9E-03	1.98E-09	NO
D-5b	<2.0E-07	N/A	4.1E-03	8.20E-10	NO
D-6a	<2.0E-07	N/A	9.9E-03	1.98E-09	NO
D-6b	<2.0E-07	N/A	4.3E-04	8.60E-11	NO
D-7	<2.0E-07	N/A	4.3E-04	8.60E-11	NO
D-8	<2.0E-07	N/A	1.4E-03	2.80E-10	NO
D-9	<2.0E-07	N/A	4.3E-04	8.60E-11	NO
F-1a	3.33E-04	YES	2.6E-03	8.66E-07	YES
F-1b	6.25E-07	N/A	1.6E-03	1.0E-09	NO
F-1c	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
F-1d	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
F-1e	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
F-1f	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
F-1g	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
F-1h	<2.0E-07	N/A	4.1E-04	8.2E-11	NO
F-1i	<2.0E-07	N/A	1.2E-03	2.4E-10	NO
F-1j	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
F-1k	<2.0E-07	N/A	4.1E-04	8.2E-11	NO
F-1m	4.25E-02	N/A	3.0E-03	1.28E-04	YES
F-1n	<2.0E-07	N/A	4.3E-04	8.6E-11	NO

Table 4.2 (Cont'd)

Firezone Screening Results

ZONE	CCDP	ISOLATION FAILURE POTENTIAL	IGNITION FREQUENCY	SCREENING CDF	MODELING REQUIRED
F-1o	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
F-1p	1.57E-01	N/A	6.7E-03	1.05E-03	YES
M-1	<2.0E-07	N/A	9.0E-04	1.8E-10	NO
M-2a	6.25E-07	N/A	8.4E-04	5.25E-10	NO
M-2b	<2.0E-07	N/A	8.9E-04	1.78E-10	NO
M-2c	1.00E-00	N/A	5.2E-03	5.2E-03	YES
M-3	<2.0E-07	N/A	2.5E-03	5.0E-10	NO
M-4	<2.0E-07	N/A	2.5E-03	5.0E-10	NO
MUWPH	<2.0E-07	NO	N/A	2.0E-07	NO
R-1a	<2.0E-07	N/A	9.4E-04	1.88E-10	NO
R-1b	<2.0E-07	N/A	8.3E-04	1.66E-10	NO
R-1c	3.34E-06	NO	4.6E-02	1.54E-07	NO
R-1d	<2.0E-07	N/A	9.2E-04	1.84E-10	NO
R-1e	<2.0E-07	N/A	8.3E-04	1.66E-10	NO
R-1f	<2.0E-07	N/A	9.1E-04	1.82E-10	NO
R-1g	<2.0E-07	N/A	8.9E-04	1.78E-10	NO
R-1h	<2.0E-07	N/A	1.1E-03	2.2E-10	NO.
R-1i	4.44E-05	NO	2.0E-03	8.88E-08	NO
R-1j	<2.0E-07	N/A	1.1E-03	2.2E-10	NO
R-1k	<2.0E-07	N/A	8.8E-04	1.76E-10	NO
R-1m	<2.0E-07	N/A	9.9E-04	1.98E-10	NO
R-1n	<2.0E-07	N/A	8.5E-04	1.7E-10	NO
R-1o	<2.0E-07	N/A	9.5E-04	1.9E-10	NO
R-1p	3.34E-06	N/A	1.0E-03	3.34E-09	NO
R-1q	3.34E-06	N/A	4.8E-03	1.60E-08	NO
R-1r	<2.0E-07	N/A	2.6E-03	5.2E-10	NO
R-1s	<2.0E-07	N/A	1.5E-03	3.0E-10	NO

Table 4.2 (Cont'd)

Firezone Screening Results

ZONE	CCDP	ISOLATION FAILURE POTENTIAL	IGNITION FREQUENCY	SCREENING CDF	MODELING REQUIRED
R-1t	4.00E-03	NO	8.3E-04	3.32E-06	YES
R-1u	<2.0E-07	N/A	8.5E-04	1.7E-10	NO
RITANK	6.25E-07	NO	N/A	6.25E-07	NO
SERV.	<2.0E-07	NO	N/A	2.0E-07	NO
T-1a	5.71E-06	N/A	7.6E-03	4.34E-08	NO
T-1b	<2.0E-07	N/A	1.2E-03	2.4E-10	NO
T-1c	2.45E-07	N/A	1.1E-03	2.70E-10	NO
T-1d	<2.0E-07	N/A	4.0E-04	8.0E-11	NO
T-1e	2.08E-07	N/A	6.4E-04	1.33E-10	NO
T-1f	4.46E-05	NO	6.3E-03	2.81E-07	NO
T-1g	2.08E-07	N/A	6.8E-04	1.41E-10	NO
T-1h	9.72E-06	NO	1.3E-02	1.26E-07	NO
T-1i	<2.0E-07	N/A	1.4E-02	2.8E-09	NO
T-1j	4.38E-06	N/A	4.5E-04	1.97E-09	NO
T-1k	4.38E-06	N/A	1.2E-03	5.26E-09	NO
T-1m	<2.0E-07	N/A	1.2E-02	2.4E-09	NO
T-1n	<2.0E-07	NO	4.1E-04	8.2E-11	NO

4.3 Fire Growth and Propagation

For the screening evaluation of a firezone, a fire was assumed to disable all equipment and cables within the zone. Fire modeling identifies different scenarios in which only equipment or cables within the damage/ignition range of an ignition source are assumed disabled.

Fire modeling is performed on an ignition source-by-source basis for both fixed and transient sources. Fixed ignition sources are those sources that are either installed such that they are immobile (a pump, for example) or by procedure or practice are always located in the same place with a firezone. Transient ignition sources are defined as sources that have the potential to be located essentially anywhere within a firezone.

Fire modeling of fixed ignition sources is performed in a stepwise fashion. This modeling procedure is detailed in the FPRA methodology guide. The detailed steps involved in fire modeling are described in the following sections.

4.3.1 Fire Damage from Fixed Ignition Sources

First, the physical characteristics of the firezone were determined from design documents. This information included the floor area, ceiling height, room volume, maximum ambient temperature, and identification of any unique features that could affect the analysis.

Next, all fixed ignition sources and their spatial locations within the firezone were identified. During the walkdowns, any existing material storage sites were identified as potential fixed ignition sources.

Each fixed ignition source was categorized to determine the appropriate heat release rate (HRR). Cases where HRRs were not explicitly specified in the FPRA methodology were resolved by soliciting expert judgment from Scientific Applications International Corporation (SAIC), the developer of the FPRA Methodology for EPRI.

Using walkdown information, a location factor was assigned for each fixed ignition source in the firezone. The location factor accounts for the higher fire plume temperatures occurring near walls and in corners.

Targets for each fixed ignition source were determined by identifying all equipment, cables, conduits, and cable trays within the damage heights and ranges calculated on the walkdown sheets. This information was obtained in the walkdowns and verified by review of design documents. Targets located outside of the radiant damage range and are not located in the fire plume were also evaluated for potential damage from the ceiling jet.

If a target was found to be within the ignition range of a fixed ignition source, the target was in turn considered an ignition source and additional targets were identified. This models the propagation of a fire from a source to a series of combustible targets.

The potential for the formation of a hot gas layer (HGL) is also examined for each fixed ignition source. A HGL is defined as a layer of gas of temperature equal to the target damage temperature, extending from the firezone ceiling to the virtual surface of the fire. This analysis considered the BTU loading of the ignition source and all ignited targets. Since HGL formation scenarios are longer duration scenarios (>5 minutes), a heat loss factor of 0.94 was used. This heat loss factor was recommended in the FPRA implementation guide.

Once the potential targets of a fixed ignition source have been identified, the consequences of the loss of the source and targets can be evaluated. The set of cables and equipment associated with each target are determined. This list of impaired equipment was then used to extract a list of associated basic events (BEs) from the CPS PRA model.

Using the list of BEs, a Set Equation Transformation System (SETS) code input deck for each target was created. This input deck, along with appropriate PRA model accident initiators, was used in the quantification of the PRA model and a CCDP determined. The resulting CCDP is the probability of core damage occurring given that the ignition source and target equipment are disabled.

An ignition frequency for the individual ignition source is calculated based on FPRA methodology guidance. This ignition frequency is multiplied by the CCDP to generate a core damage frequency (CDF) associated with the particular ignition source. For scenarios where suppression was credited, the fire non-suppression probability was multiplied by the CCDP and ignition frequency to determine the CDF.

4.3.2 Fire Damage from Transient Ignition Sources

Modeling of transient ignition sources and combustibles is similar to fixed ignition sources with some additional steps to account for the uncertainty in location and amounts of the combustibles. These additional steps are described in the following paragraphs.

First, the type of transient combustibles that are likely to be present or traverse the subject firezone must be determined. This evaluation is based on the preventive maintenance and equipment history data, the physical layout of the firezone and adjacent firezones as well as the CPS fire load calculation. The procedures that dictate how transient combustible materials are controlled were also reviewed at this point.

Using the previously determined data, a transient ignition frequency is calculated for each modeled firezone. This number is based on the number and types of FPRA methodology specified activities that are not prohibited within the firezone.

Next, an appropriate HRR was selected for the transient combustibles likely to be found in the firezone. This value was chosen from the Sandia National Laboratory (SNL) tests detailed in the FPRA implementation guide. Typically, a bag of miscellaneous maintenance waste or canvas cart loaded with protective clothing (PC) was selected. The selection of the canvas cart of PCs is based on both the likelihood of a cart being present and the accessibility to a cart of a particular firezone or portion of a firezone.

Target sets were identified for specific floor locations using damage range calculations for transient combustibles in the same way that fixed ignition source targets were identified. The floor area that the transient combustible fuel package could be located within and damage each target set was measured. The ratio of the damage area over the firezone floor area is then calculated. This ratio represents the probability that a transient combustible is located in the area affecting a particular target set.

Following the calculation of the area ratio, the transient sets were evaluated using the same techniques as fixed ignition sources. The only difference being that the CDF for each transient ignition source - target scenario was multiplied by the appropriate area ratio.

The final step in the analysis was to sum all CDFs for both fixed and transient ignition sources. This summation represents the core damage frequency for all fires occurring in a particular firezone. Certain assumptions were applied during the course of the fire modeling analysis. These assumptions are detailed in the following section.

4.3.3 Fire Modeling Assumptions

The following assumptions were used in the fire modeling process:

1) No energy-shielding credit is taken for any equipment (concrete, piping, conduits, HVAC ductwork, hangers, etc.) located between any target (cables, equipment, etc.) and the ignition source unless explicitly identified in individual fire modeling scenarios.

2) No adjustment to the room volume was made for equipment located in the firezone. The floor area calculation accounted for building columns and interior walls within the firezone. Any further reduction in room volume due to concrete or equipment inside the room is at least balanced by the capacity of such equipment to absorb heat; therefore, no deduction or credit was taken for either effect.

3) Cables within the calculated damage ranges of an ignition source are considered damaged without taking credit for the heat absorption or dissipation ability of their raceway (tray, riser or conduit). Additionally, no credit is taken for the solid bottoms on cable trays or tray covers on risers which would reduce temperatures at the cables by acting as heat sinks. The only exception to this is for the cable risers near the Component Cooling Water pumps for which a special calculation was performed.

4) No credit is taken for installed Thermo-Lag material on cable raceways, which would reduce temperatures at the cable targets. Instead, the cable is modeled as if exposed with the cable damage temperature as the failure criteria.

5) The ambient temperature in each firezone was conservatively assumed to be the design basis limiting temperature accounting for summertime temperature/humidity and surrounding rooms at

elevated temperatures. No credit is taken for the extra capacity in the HVAC system as installed, which normally maintains temperatures well below the design basis level.

6) The fraction of violations of the CPS transient combustible control program was set at 10%. This is the screening value specified in the FPRA methodology. This value is conservative because the CPS program exceeds the minimum requirements for a program as specified in the FPRA methodology.

7) The analysis in firezone CB-5a took credit for a modification that will be performed during the next refueling outage, currently scheduled for October, 1996. This modification will reroute the Division 2 safety related cables outside the firezone CB-5a and is being performed as part of the CPS program to address problems with Thermo-Lag 330-1.

4.3.4 Main Control Room Analysis

The analysis of the control room differs from other portions of fire propagation analysis. Fire propagation analysis typically addresses damage to overhead cables from a plume or ceiling jet or room heatup by a hot gas layer. Control rooms typically however have cable enclosed in cabinets. Additionally, the impact on the operators manning the controls is an important fire effect.

In control room fire analysis, fires begin in electrical cabinets and their effects on controls within the cabinet and in adjacent cabinets need to be evaluated. People manning controls must see them, so smoke rather than temperature in the hot gas layer is evaluated to determine if and when evacuation could occur.

Just as the natures of the evaluations are different, the tools to perform them are also different. Practical fire models that exist do not have the capability to predict fire growth within a cabinet nor descending smoke layers in a room. However, a reasonable body of evidence is available in the Sandia cabinet fire tests (NUREG/CR-4527) and in EPRI's Fire Events Database, NSAC-178L, (FEDB) to characterize the potential that cabinet controls are damaged and smoke obscures control panels. The EPRI Fire PRA Implementation Guide describes the interpretation of that data and its application to control room fire analysis. This approach is used in the analysis.

While fire modeling tools were different, the analysis of the CPS control room nevertheless followed a process similar to the rest of the fire PRA. Boundaries for fire spread in cabinets and the equipment within those boundaries were identified. Conditional core damage probabilities and ignition frequencies were estimated. The resulting cabinet core damage frequencies were ranked and more detailed analysis of fire spread within the cabinets was performed for the most significant cabinets. Finally, smoke effects were analyzed for their potential to require evacuation of the control room.

4.3.4.a Assumptions

The major assumptions in the control room analysis are as follows:

1. The underfloor area poses no fire risk due to its design (i.e., all qualified cable and protection by automatic halon system).

2. The loss of a cabinet containing divisional equipment or BOP equipment was assumed to include the loss of all such equipment unless specifically analyzed.

3. Evacuation of the control room was assumed to occur at the time smoke visibly obscured the control panels (which was then shown not to be likely).

4. Human detection was credited to be as effective as an in-cabinet smoke detector in cabinet P680.

4.3.4.b Control Room Analysis Methodology

The analysis of the control room is based upon two levels of fire severity. In the most severe case, a fire can develop and can fail to be suppressed for a long time. In this case, smoke can accumulate in the control room until a layer begins to descend and become dense enough to obscure the main control board. These scenarios will be referred to as "evacuation scenarios". Fires in the CPS control room causing evacuation have been evaluated as an insignificant source of risk.

Fires less severe than this extreme case can be significant, however. While the fire is suppressed before evacuation becomes necessary, it has the potential to damage the controls of one or more cabinets and safe shut down capability can be compromised. Operators are assumed to attempt to shut down the plant using the remaining capability available from the control room. If that capability fails, e.g., equipment fails to operate, operators are expected to use the remote shutdown capability. These scenarios will be referred to as "critical cabinet scenarios".

The analysis of each of these scenarios followed the guidelines of Appendix J of the Fire Risk Analysis Implementation Guide. The first effort, identifying the boundaries for fire spread in cabinets and the equipment within those boundaries, involved the following steps:

Step 1 - Determine the logical boundaries for the spread of fires initiated in a cabinet (or cabinet section). The logical boundaries of fire spread in a control room cabinet were based on the principal that a double wall with an air gap will contain the spread of a fire. After a period of time, solid state circuitry in adjacent cabinets can still be affected; however, it was assumed that manual suppression would reliably occur in the intervening time.

Step 2 - Evaluate the plant capability available for responding to control room fires. This step involves the review of the existing plant safe shutdown analysis to identify what equipment can be operated from outside of the control room. Additionally, plant procedures were reviewed to determine the conditions under which this equipment can be operated.

Step 3 - Determine which divisions are contained in each cabinet. Completing this effort determined the number and type of cabinets for ignition frequency calculations. It also determined the combinations of divisions which could be affected and the design and procedural capability for remote shutdown alternatives.

Step 4 - Calculate the CCDP for each cabinet. Due to the complexity in determining exactly which pieces of equipment are affected by the loss of some of the cabinets, the CCDPs were calculated based on the loss of an entire division. BOP cabinets were modeled as a simultaneous loss of service water, feedwater and instrument air. This approach allowed a matrix to be developed which contained all the two division/BOP cabinet combination CCDPs. Some three division cases were also evaluated based on some cabinet configurations.

As described earlier, CPS has remote shutdown capability for both Division I and Division II. This capability can significantly impact the cabinet fire scenarios in which one or both of these divisions are affected by a fire and random equipment failures prevent the use of other safe shutdown paths.

To determine the impact of plant remote shutdown capability on CCDPs, an analysis was performed. CCDPs with and without these divisions were compared for each CCDP group. By ratioing the two values, a conditional probability was obtained which indicated the potential value of Remote Shutdown Panel (RSP) recovery. However, the two CCDP values compared assumed operator event probabilities corresponding to in-control room actions. Therefore, it was important to check the values to determine that they reasonably reflected the additional constraints on the operator.

The following assumptions were used in developing the final RSP recovery values:

No value less than 0.01 shall be used, and

Use of Division II will be no better than 0.1 when used alone and 0.25 when used after Division I has been tried remotely and failed.

These assumptions reflect the belief that despite the capability of the design and the quality of the procedures and training at CPS, that 0.01 was a reasonable bound to place on RSP capability. Similarly, the fact that Division I is the preferred path with more capability available from its panel required that Division II be given a higher failure probability. It was also felt that if Division II were attempted after Division I had failed when remotely actuated that the reduction in time available and the potential increase in stress would result in a more significant bound on the credit that should be given to Division II operation. Finally, Division II was not credited in the fire analysis after a loss of offsite power (LOOP). No credit was taken under SBO conditions since most of Division II does not have permanently installed lighting with LOOP capability.

Step 5 - Determine nominal cabinet ignition frequency. This effort was based on gross assumptions, namely that all divisions in a cabinet would be failed both in their entirety and immediately and that ignition sources were distributed rather evenly among the 113 cabinets.

The resulting cabinet core damage frequencies were ranked and more detailed analysis of fire spread within the cabinets was performed for the most significant cabinets. This process involved three steps:

Step 6 - Rank cabinets by CDF and select cabinets for detailed analysis.

Cabinet CCDPs and ignition frequencies from Step 4 and Step 5 respectively were multiplied to obtain a screening value for cabinet core damage frequency. The screening values were ranked by CDF. The top 24 cabinets were selected for detailed analysis. The cabinets selected represent different types of cabinets, e.g., termination cabinets, back panels and control panels. The list included 15 cabinets which each represented only a small fraction of the total screening CDF, namely between 1 and 0.5 percent of the total. These 15 cabinets, while not significant contributors in the screening analysis, might prove significant if they contained a substantial number of ignition sources, i.e., relays and circuit cards. Therefore, a more accurate estimate of cabinet ignition frequency is determined in the next step.

Step 7 - Determine the ignition frequency for each critical cabinet.

Indications from other tests of the EPRI methodology indicated that both the overall risk estimate and the risk profile of various cabinets varied significantly when based on the distribution of relays and cards. The insights from these test applications indicated the following:

The numbers of relays and circuit cards varied from cabinet to cabinet (although it generally fell in a range of about ten around the nominal value),

Sources were often concentrated in non-safety, balance-of-plant cabinets,

Cabinets with redundant safety equipment, e.g., the main control boards, often contained few or no ignition sources, and

Occasionally a cabinet with important controls contained a large fraction of ignition sources.

Consequently, relay and circuit card counts were obtained.

Relay counts by cabinet were obtained in electronic form from a database. Information on individual bays was not available from the list. Visual counts were not found to be accurate, but did provide a general indication. For these reasons, the count for bays was estimated based on the assumption that safety related relays would be in Division I through IV bays and non-safety relays would be in non-divisional bays. This assumption appeared consistent with visual observations.

Circuit card counts were not available in electronic form. A cognizant engineer provided an estimate of a total count, namely between five and nine thousand circuit cards. A preliminary walkdown found that counts for individual cabinets and bays could be performed by observation. The final count was obtained using the following assumptions:

The total number of cards in the control room was 8000,

The number of cards in computer room cabinets was 1/2 of the control room total (4000 cards), and

The number of cards in all control room cabinets not in the computer room was 4000.

Visual observations obtained counts for 34 cabinets totaling about 2100 cards. Termination cabinets were known to contain no circuit cards (or relays). Apportioning the remaining cards equally among applicable cabinets yielded a result similar to visual observations. The 31 computer cabinets should each contain about 129 cards and the other 30 cabinets should each contain about 63 cards. Since a screening analysis based on risk importance was used to select the cabinets for visual observation, these estimates were found to be reasonable for estimating control room risk.

The ignition frequency is based on the fraction of relay and circuit card fire events (6/11 and 3/11 respectively from the FEDB) and the fraction of relays and circuit cards in the individual cabinet. All cases have a minimum of two events corresponding to the "other" category. The ignition frequency for all cabinets must be apportioned among all the cabinets in the control room. The ignition frequency (IF) is apportioned as follows:

IF = 9.5E-3*(2/11*1/113+3/11*C/8000+6/11*R/1261) where:

C= the number of circuit cards in the cabinet, and

R= the number of relays in the cabinet

The results of the detailed ignition frequency calculation range from 7.5E-4/yr to 1.5E-5/yr. The results reflect the concentration of ignition sources, relays in particular, in a few cabinets. Three cabinets, P839, P851/852, and P861/862, represent more than twenty percent of the total ignition frequency and eleven cabinets represent more than fifty percent.

Step 8 - Determine final CDF for critical cabinets.

As visual examinations of critical cabinets were done, it became apparent that there was significant conservatism in the initial CCDP analysis. The assumption that an entire division was failed if any of its circuits were located in the cabinet was found to be conservative because often only a few division components were in the cabinet. For example, P870 is shown in drawings to contain Division I and II as well as BOP equipment. Upon inspection of the cabinet, it was identified that the only divisional equipment on this BOP control board were containment isolation valves. Upon confirmation, the CCDP was changed to reflect the actual equipment on the board.

The results of Step 6 had indicated that a significant fraction of control room risk (i.e., greater than 10%) was attributable to cabinets solely with BOP circuits. The principal reason was that selected BOP circuits, e.g., plant service water, were significant to plant risk. However, as the visual examination was being done, it became apparent that a number of BOP only cabinets could be reclassified as plant trip only cabinets. Upon confirmation, the cabinet CCDPs were adjusted. An example of this situation was the 1H13-P870 panel. This panel was noted on drawings as containing Division I, Division II and BOP equipment. Examination of the panel found the divisional equipment limited to some BOP system containment isolation valves. When this panel was requantified with only the isolation valves failed, the CCDP was significantly lower than the loss of the entire division.

Step 9 - Evaluate fire spread within cabinets for critical scenarios

The CCDP ranking facilitated by step 8 provided a strong basis for determining which cabinets required detailed evaluation. The subsequent evaluation involved visual examination of 24 cabinets. Ignition sources were counted to develop cabinet specific ignition frequencies. Internal cabinet boundaries were evaluated to determine the probability of fire spread within cabinet sections. And, in some cases, specific components in the divisions effected were evaluated to obtain a precise CCDP for the cabinet.

The FPRA implementation guide indicates that only 20% of CR fires have the potential for spread to multiple divisions. This severity factor was used (i.e., 0.2) to estimate the significance of fire spread between cabinet sections. For the cases that were risk significant after this analysis, the specific boundary was examined and the probability of suppression calculated.

The specific suppression probabilities applied are provided in Table 4.3.

Table 4.3

Main Control Room Suppression Probabilities

Cabinet	Bays Affected	Suppression Failure Probability	Barrier Type
P663	B & C,A2,A3	0.1	Internal double wall with small opening and cable
P671	A & B	0.03	Internal double wall with optical isolator connections
P671	B & A,C	0.03	Internal double wall with optical isolator connections
P821/822	B & A,C	0.1	Single wall with optical isolator connections
P821/822	A & B	0.1	Single wall with optical isolator connections
P663	A3 & A2,A1	0.1	Internal double wall with small opening and cable
P664	D3 & D2,D4	0.1	Internal double wall with small opening and cable
P664	D2 & C,D3,D4	0.1	Internal double wall with small opening and cable
P664	D4 & C,D2,D3	0.1	Internal double wall with small opening and cable
P664	C & B,D2,D4	0.1	Internal double wall with small opening and cable
P670	B & A,C	0.1	Internal double wall with small opening and cable
P669	B & A,C	0.1	Internal double wall with small opening and cable

The suppression probabilities reflect very conservative interpretations based on insights from the Sandia cabinet fire tests. Application of these probabilities instead of the 0.2 severity factor resulted in a reduction in total control room CDF of about 3E-7 or about 25%. Generally, even when the 0.2 factor was applied significant barriers existed to fire spread. Hence, this analysis is conservative.

Given that individual cabinet damage had been evaluated, fire effects were analyzed for their potential to cause more widespread damage, e.g., smoke causing evacuation of the control room and damage of components in adjacent cabinets. This analysis included two steps:

Step 10 - Determine the probability of suppression failure prior to control room evacuation.

Normally, the probability of cabinet fire not being suppressed before the need for control room evacuation is 3.4E-3 based on the Implementation Guide. This value assumes detection at or prior to about the time that visible smoke appears in the cabinet. Automatic detection will occur at about this time if in-cabinet smoke detectors are present. It also assumes 15 minutes from the time of detection to the time of smoke obscuration of the control panels.

At CPS, plant-specific factors, such as room volume and room ventilation and the use of Tefzel cable, will prevent or substantially delay smoke obscuration.

The CPS control room volume is substantially larger than the Sandia test facility (188,100 ft^3 versus 48,000 ft^3) and has high ventilation rates. During smoke purge mode, the ventilation system provides roughly ten room changes per hour (the maximum available in the Sandia tests). For these reasons, we would expect a substantial delay in smoke accumulation in the CPS control room if all other factors were the same.

Additionally, NEDO-10466 indicates there are low smoke generation properties associated with the Tefzel cable used at CPS. Since the Implementation Guide test times are based on non-qualified cable, it is reasonable to assume that a CPS control room fire would not require evacuation, or at the very least, would be very unlikely to do so.

For these reasons, evacuation scenarios were considered insignificant contributors to risk.

Step 11 - Evaluate fire effects on adjacent cabinets

Only one combination of adjacent cabinets could result in significantly different CCDPs from the single cabinet case. This case was further analyzed and determined to be an insignificant contributor (~1%) due to the long time between initial detection and damage to the adjacent cabinet.

The suppression probability is based on the conclusion that temperatures in an adjacent cabinet reach 150 degrees F somewhat after smoke obscures the panel in the Sandia cabinet tests. This finding, together with the findings from Step 10, indicated that the probability of non-suppression before damage is less than 3.4E-3, the probability of non-suppression at 15 minutes.

4.3.4.c Control Room Results

The total core damage frequency for the CPS control room from internal fire is 1.2E-06 per year. The low CDF reflects the CPS control room design which is particularly effective in preventing fire spread to important components in multiple divisions. The low CDF also reflects the quality of the plant's control room fire procedures as well as the plant's remote shutdown capability. The conditional core damage probabilities indicate the strength of the above two factors. Except for one cabinet (P680), the chance of core damage given a control room fire is at most one in one thousand. Such a number implies redundant safe shutdown paths are almost always available.

The fire risk also reflects the insignificant contribution of cabinet fires requiring evacuation. Had such a scenario been as probable as a typical control room, an additional increase of 25% would result. The limited significance of even such a worst case scenario also reflects the factors mentioned above. That is, even in the unlikely event evacuation were required, the plant capability makes the conditional probability of core damage still only about one in one hundred.

Besides the overall insight offered by the low total CDF and the low CCDPs, insight can be drawn from examining the cabinets which contribute most significantly. As the results described below suggest, sources of core damage generally come from a wide variety of sources.

Roughly one-third of the risk is in three cabinets, P680, P663 and P839. These three cabinets have ignition frequencies and CCDPs that are low/high, moderate/moderate and high/low respectively. That is, the three are each contributors for different reasons.

Roughly 75% of the risk is in the top 12 cabinets (out of 113). Except for P680, each of these cabinets has a higher than average ignition frequency. All were analyzed in detail for fire spread, except for P839, P612 and P855. These three cabinets had very high or relatively high ignition frequencies but effectively impact only BOP circuits. Of the remaining 9 cabinets, two can impact Division III and are important because there is no RSP recovery if Divisions I and II randomly fail. Four can impact multiple divisions. Three cabinets impact only recoverable divisions (I or II), but generally fit the category of high ignition frequency (-4%). In four of the nine cabinets, the cabinet CDF was reduced only slightly (<2%) by the evaluation of fire spread. Four of the other five were reduced by about an order of magnitude while one was reduced by a factor of thirty.

Roughly 90% of the risk is in the top 33 cabinets. Of those, only P664 was analyzed in detail for fire spread (Step 9). The analysis of fire spread tended to change the numerical results more than importance of the cabinet. At the 90% cumulative "break point", cabinets below this threshold individually contribute less than 0.5% to the total risk.

In conclusion, the control room results indicate that, despite its significance to the total fire risk at CPS, the control room represents no significant vulnerability. The total CDF reflects a design and operations philosophy that provides redundancy against threats from internal fires. Also, the contributions from various risk sources are diverse and point to no single common element. The control room's relative significance is more a reflection of low fire risk at CPS rather than significant fire risk in the CPS control room.

4.3.5 Multi-Compartment Analysis

Multi-compartment analysis was performed to determine whether fires within a zone could propagate to adjacent zones. Since the formation of an HGL is a prerequisite for fire propagation between firezones, the potential for the formation of an HGL was evaluated for all firezones in the plant. A screening was performed by examining the room volume and design maximum ambient temperature to determine the heat required to form an HGL and by determining the heat content of the combustibles within the zone. Firezones without the potential for HGL formation were screened from further analysis.

Firezones that were identified as having the potential to form an HGL were then combined with individual adjacent firezones to determine if a fire in the exposing zone had the potential to form an HGL in the combined firezone. Fire scenarios where the exposing zone does not have the potential to form an HGL in the combined firezone were screened from further analysis.

Some fire scenarios were found in which the list of damaged equipment in the exposing firezone was identical to, or encompassed, the list of equipment for the adjacent zone. In these situations, if the exposing compartment screened in the single compartment analysis, the multi-compartment scenario also screened. Similarly, some exposing-adjacent firezone combinations did not contain any modeled equipment or cables and were screened from further analysis.

Multi-compartment scenarios that were not screened by other means had CDF calculations performed and compared with the screening threshold of 1.0E-06/yr. Determination of the scenario CDF required that the CCDP for both firezones be calculated. The set of failed equipment in each zone was combined for this calculation. The scenario CCDP was multiplied by the exposing zone ignition frequency, the barrier failure probability and any automatic suppression failure probability. Scenarios where automatic suppression systems were located in both firezones had credit taken for only one system due to potential dependencies between zone suppression systems.

544 separate multi-compartment scenarios were identified that required evaluation. 114 scenarios were screened on the basis of one of the firezones not being susceptible to HGL formation. 207 scenarios were screened due to no potential for HGL formation to occur in the combined zone. 89 scenarios either had no modeled equipment associated with both zones or had common equipment and an exposing compartment that screened alone. 134 scenarios required CDF calculation or a specific analysis for evaluation. Analysis found that none of the evaluated scenarios exceeded the 1.0E-06/yr CDF screening criteria.

4.4 Evaluation of Component Fragility and Failure Modes

Equipment was assumed to fail by loss of electrical function or spurious signal if the damage criteria was reached (700°F) regardless of the length of time the cable was at this temperature. Equipment was assumed to fail if any cable providing an essential service to the equipment was predicted to reach the damage criteria. Fires were not assumed to cause plugging failures, tank failures, or check valve opening and closing failures or piping system failures.

4.5 Fire Detection and Suppression

Fire detection and suppression varies among firezones depending on the importance of the equipment and the combustible loading within the zone. Fire detection is provided in any of the zones meeting the criteria for equipment importance or containing moderate to high quantities of combustibles. The detection systems used are either ionization, thermal, or flame detector types. Suppression systems are wet pipe, preaction, Halon, and Carbon Dioxide systems. All zones not having automatic suppression systems contain manual portable fire extinguishers.

Firezones containing wet pipe systems with whole zone coverage were credited in the fire modeling and multi-compartment analyses. Since wet pipe suppression systems do not rely on detectors for actuation, no detector analysis was performed. The generic wet pipe suppression system unavailability probability contained in the FPRA methodology was used as the suppression failure probability in the firezones where credit was taken.

The only firezone where manual suppression was credited was the main control room. Analysis of the manual suppression probabilities is detailed in section 4.3.4.

4.6 Analysis of Plant Systems, Sequences, and Plant Response

The analysis of plant systems' response to fire initiators and the corresponding evaluation of fire initiated accident sequences was performed for the screening of firezones as described in section 4.1 above. Of the 121 firezones evaluated, all but 21 zones showed no significant risk of core damage due to fire scenarios within the zones.

As described above, the analysis of each fixed and transient ignition source has associated targets that include equipment and cables that are modeled in the CPS PRA. Solution decks for each set of equipment were developed for quantification with the SETS code. The solution to each of these target sets generated a CCDP for the specific ignition source. The ignition frequency of each ignition source was calculated based on the guidance of the EPRI FPRA methodology. The core damage frequency due to each ignition source was determined by the product of the ignition frequency and the CCDP result from the SETS analysis. In addition to the evaluation of each ignition source within a firezone, the impact of the fire in one zone spreading to adjacent zones was determined.

The CCDP results establish the accident sequences that could result from the set of equipment and cables predicted to be disabled by a particular ignition source. The CCDPs are calculated by the modification of the fault trees and the identification of the basic events that are assumed to be failed. SETS is used to recompute the modified sequences. The fault trees are modified to account for the failed components in the fault trees that are included in the set of equipment or are dependent on the failed cables. The failed basic events and the CAFTA output (linked fault tree models) are input to SETS to perform the CCDP quantification. The unmodified portions of the fault trees are allowed to fail at their normal frequency.

The total core damage frequency due to internal fires was calculated by the summation of the CDFs from individual ignition sources within each firezone. No contribution from the screened firezones was included in the total CDF calculation. Fire modeling found that CDFs typically drop 1 to 3 orders of magnitude from the screening value following the fire modeling process. Inclusion of screened firezones in the overall CDF would tend to obscure firezones with the greatest CDF risk by including unrealistically large contributions from screening evaluations.

These detailed determinations of core damage frequency were developed conservatively. In modeling the effect of fires or establishing the amount of transient combustibles in each zone the conservative assumptions made included the following:

1. Credit was generally not taken for the energy shielding effect of target equipment or the ignition source unless clearly established,

2. No credit was taken for installed Thermo-Lag material on cable raceways that would reduce temperatures or impede fire damage in any way,

3. No credit is taken for the heat absorption capability of much of the installed equipment,

4. The initial ambient temperature in each zone was assumed to be the highest design value,

5. The fraction of violations of the CPS transient combustible control program was taken at 10 percent (screening value from the FPRA methodology), substantially above the value estimated by plant fire protection,

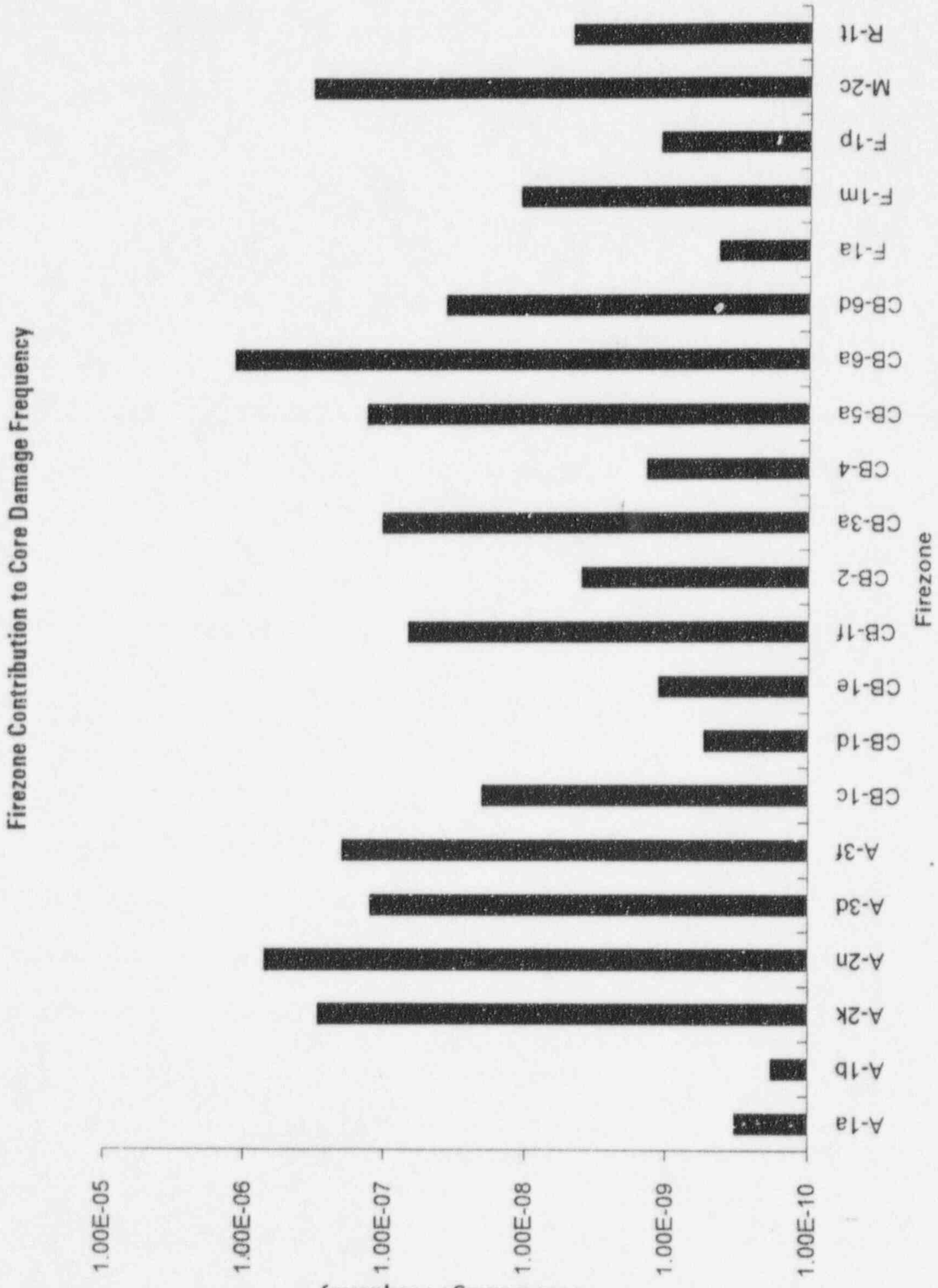
6. The transient combustible fuel package used for all locations accessible to a cart was a canvas cart of protective clothing (This cart has the highest recommended transient combustible heat release rate at 333 BTU/sec.), and

7. The air-cooled transformers located in motor control centers were binned with a classification that resulted in a higher ignition frequency.

4.6.1 Core Damage Frequency by FireZones

Figure 4.24 displays the results of the FPRA by firezone for the significant firezones.

Figure 4.24
Fire Zone Contribution to Core Damage Frequency



4.6.2 Summary of Results and Conclusions

The overall result of the FPRA has components from the fixed and transient ignition source analysis, the main control room analysis and the multi-compartment analysis. The summation of these components is the CDF from internal fires at CPS. The contribution from each analysis is as follows:

Fixed and Transient Ignition Sources	=	2.06E-06/yr
Main Control Room	=	1.20E-06/yr
Multi-compartment	=	0.0
Total	=	3.26E-06/yr

Analysis of the CDF contribution per firezone found that the main control room accounted for 36.8 % of the total risk from internal fires. A high importance for the control room makes intuitive sense since most of the divisional and important balance of plant equipment is controlled from this location.

The second major contributor to internal fire CDF was the group of switchgear rooms. These locations collectively accounted for 44.6% of the internal fire risk. Again, the importance of these firezones was anticipated since power and control feeds for divisional and balance of plant equipment are concentrated in the switchgear compartments.

One switchgear room, CB-5a, was identified from fire modeling as having the potential for formation of an HGL in a particular fire scenario. This scenario was of particular significance since CB-5a contains cables for Divisions 1,2 and 3. While the Division 1 cables were protected by Thermo-Lag 330-1, no credit was taken for this material due to combustibility and protection concerns. As part of the CPS response to these concerns, the Division 2 cables will be rerouted outside of this zone during the next refueling outage. In the interim, hourly firewatch surveillance of this firezone is maintained. Credit was taken for this modification in the analysis which resulted in a 75.5% reduction in plant internal fire CDF.

Firezone M-2c, the general access area of the screenhouse, was the only other zone to account for a significant portion (10.4%) of the internal fire CDF. This firezone contains the plant service water pumps, cables for the circulating water pumps, fire protection pump cables as well as some divisional cables associated with the shutdown service water pumps. Because of this concentration of equipment and cables, it was expected that firezone M-2c would have a relatively high importance.

In order for a firezone to be identified as a significant contributor to the internal fire CDF, the zone must contain important cables or equipment and these components must be in relatively close proximity to an ignition source. Prior to performing fire modeling, it was anticipated that that the two cable spreading areas, firezones CB-2 and CB-4, would account for a significant portion of the internal fire risk since these locations have the potential to disable entire safety-related divisions of equipment as well as a large number of balance of plant components. Fire modeling found that due in a large part to the lack of fixed ignition sources, use of solid metal bottom cable trays and the existence of a whole zone coverage wet pipe sprinkler system, the fire risk in the two cable spreading area accounted for less than 1% of the internal fire CDF.

One fact noted from the analysis was the relatively small portion of the internal fire risk that transient ignition sources accounted for in comparison to fixed ignition sources. Transient sources accounted for only 5.2% of the internal fire CDF. This result is attributed in part to the fact that many firezones at CPS have relatively high ceilings. This allowed many conduits and cable trays to be mounted high enough to be unaffected by fires ignited by transient ignition sources.

Fire suppression uncertainty is an issue that was examined for impact on CDF. At CPS, credit for fire suppression was applied in only 4 firezones. Three of these zones credited a whole zone, wet pipe, automatic suppression system. The fourth firezone (the main control room) credited manual suppression. The suppression failure probability used for the wet pipe system was the generic value of 2.0E-02 from the FPRA implementation guide. Without taking credit for automatic suppression in the three firezones the plant CDF would have been a factor of 266 higher.

As was stated earlier, manual fire suppression was only credited in the main control room. Manual suppression was only applied to six main control room electrical cabinets and resulted in a 25% reduction in the main control room CDF. This reduction corresponds to an 11% reduction in the overall CDF from internal fires.

4.7 Analysis of Containment Performance

The impact of fires within the containment is specifically excluded from analysis as part of the FPRA methodology. This exclusion is based on the generic assumptions 1) that an HGL is unlikely to form that could damage cables and 2) the low historical frequency and consequences of fires in containment buildings. The accuracy of these assumptions regarding the CPS containment building was examined to ensure no potentially significant risk existed.

The first assumption was that an HGL was unlikely to form in the containment. CPS has a GE Mark III containment structure with an internal volume of approximately 1.8 million cubic feet. A review of potential combustion sources did not identify any source with the capability to generate anything more than a small portion of the heat required to form an HGL in such a large structure. Correspondingly, the first assumption was viewed as being applicable to the CPS containment.

The second assumption is based partially on data in the EPRI Fire Events Database which found that the frequency of fires occurring inside the containment was very low. Review of CPS fire reports did not identify any situation that contradicted the low frequency assumption. Another aspect of the second assumption is that containment fires do not have the potential to damage redundant equipment with the same fire plume. In the CPS containment, redundant equipment is typically installed with large spatial separation. In most cases, such equipment has either the drywell or the steam tunnel between redundant equipment. No situations were identified in which redundant equipment could be disabled by a single fire.

The impact of a fire on the ability to isolate the containment is another aspect of containment performance. During the screening process, any firezone with a CDF of greater than 1.0E-07 was either subjected to fire modeling or evaluated for potential impact on containment isolation. The cables and equipment in each subject firezone were reviewed to determine if the loss of these components would either prevent any containment isolation valves from closing or cause an isolation valve to erroneously open. This review identified only 2 locations with the potential to impact containment isolation. The situation involves the same two containment isolation valves located in the RCIC system steam supply line. These two locations are deemed not to provide a significant hazard for the following reasons:

- The ignition frequency for the larger of the two locations is very small (6.6E-07/yr).

- Failure to isolate would require a sustained hot short in a normally shut MOV.

- Both fire locations are readily accessible and have automatic detection. These features enhance manual fire suppression.

- No credit was taken for the metal sides and face of the cable riser in these locations.

- The RCIC steam supply line exhausts back into containment. Any release would also require a piping/seal rupture.

4.8 Treatment of Fire Risk Scoping Study Issues

Six fire risk related issues were identified in the Sandia/NRC Fire Risk Scoping Study. In performing the CPS IPEEE, these issues were given special attention. None of these issues were found to play a significant role in the ability of CPS to respond to a fire or to contribute significantly to core damage initiated by a fire. The assessment of these issues as they apply to the CPS is provided below.

4.8.1 Seismic-Fire Interactions

This issue encompasses the following three concerns associated with seismic events:

1. The event induces a fire or fires;
2. The event actuates the fire suppression systems; and/or
3. The event degrades the fire suppression systems.

The first of these issues is principally concerned with the failure of flammable gas or liquid containers that have the potential to rupture and ignite. The CPS IPEEE seismic walkdowns conducted by a team of analysts including seismic experts from EQE International, IPC risk analysts and an IPC fire protection engineer, examined the safety related areas of the plant for potential failures of anchorages of diesel and lube oil containers and lines.

A limited number of portable containers of flammable gases are permitted within the buildings and storage locations of these gases in quantity are external to safety related structures. Hydrogen storage tanks are located in the open yard and the hydrogen is piped into the radwaste building and then into the turbine building. The hydrogen supply line is contained in a guardpipe from the point it enters the ground in the tank area until it enters the Hydrogen control station at the 762 ft. elevation of the turbine building. This routing of the hydrogen piping limits the potential for a seismic induced hydrogen fire to 3 firezones, none of which contain any safe shutdown equipment. These zones were also screened by the FPRA as providing no significant risk of core damage. Therefore, no significant risk of core damage exists from seismic induced hydrogen fires. Transient sources of flammable gases, such as welding gases and small hydrogen bottles used for calibration, are allowed within the plant but are strictly controlled by the CPS fire protection program and do not present an unevaluated threat.

Actuation of the fire suppression systems could occur during a seismic event. The effects of flooding from inadvertent actuation have been previously evaluated as part of the internal PRA flooding study and as part of a response to an NRC information notice. There are no impacts from flooding. Other fire suppression methods such as Halon or CO_2 could be actuated. All suppression systems in areas designed for seismic loadings have been designed to the same seismic capacity. Actuation of the CO_2 system in the diesel-generator room would have no impact since the air used for combustion is taken external to the room. Therefore, there is no impact from seismic induced release of the suppression systems.

The third concern is degradation of the fire suppression systems caused by the seismic event. The CPS fire protection systems located in plant areas designed for seismic loads are provided with supports that preclude damage to other safety related equipment. This protection was verified for the safe shutdown paths during the seismic walkdowns.

4.8.2 Fire Barrier Qualifications

Inspection of fire barriers at CPS is performed per procedure CPS 9601.01, Fire Rated Assemblies and Penetration Sealing Devices. This procedure directs that fire barriers be verified operable with the following periodicity:

Fire barrier assemblies (this includes walls, floors, ceilings, fire dampers, imbedded appurtenances and grouted penetrating items) - 100% once per 36 months.

Cable tray wrap, fire barrier intersections - 100% once per 18 months.

Penetration seals - 10% once per 18 months.

Fire doors - 100% verified shut every 24 hours, 100% door hardware inspected every 6 months.

CPS has Thermo-Lag 330-1 cable wrap material installed in several different locations. These installations have been analyzed as part of the CPS program to address concerns relating to the fire rating of Thermo-Lag cable wrap material. Installations that were determined to present a significant risk to plant safety will be modified in the next two refueling outages by either rerouting cables or construction of a different barrier.

NRC Information Notice 89-52, Fire Dampers, Potential Fire Damper Operational Problems, was addressed in internal document Y-211409. The resolution of this issue was completed by a combination of the replacement of several dampers, the implementation of administrative controls to shutdown specified ventilation systems and field testing of installed fire dampers.

NRC Information Notice 83-69, Fire Dampers, Improperly Installed Fire Dampers at Nuclear Power Plants, was addressed in internal document Y-37774. Review of fire damper installation and inspection procedures found that the established program precluded problems of the sort described in the notice from occurring at CPS.

NRC Information Notice 88-04 and 88-04, Supplement 1, Fire Penetration Seal Assemblies, Inadequate Qualification and Documentation of Fire Barrier Penetration Seals, was addressed in internal document Y-209506. Penetrations with potential exposure to temperatures in excess of 500°F were monitored monthly for signs of degradation. No degradation of seal material was observed at CPS.

NRC Information Notice 88-56, Fire Penetration Seal Assemblies, Potential Problems with Silicon Foam Fire Barrier Penetration Seals, was addressed in internal document Y-209911. Installation procedures at CPS differed from the notice and provided better control over the final product. The penetration seal performance inspection performed in 1988 examined 1273 seals and identified no material defects in any seal.

4.8.3 Manual Fire Fighting Effectiveness

The CPS plant has a very effective fire protection awareness program. Training is conducted annually for all plant personnel and at least quarterly for all fire brigade members. All zones have manual fire extinguishing equipment and automatic fire detection systems. Automatic fire suppression systems are installed where considered necessary. Personnel are trained in the use of the portable extinguishers and on the means of reporting fires.

Each shift has a manned fire brigade consisting of a minimum of 5 trained personnel. The brigade supervisors are also trained and are knowledgeable in plant systems and operations. The brigade members are fully equipped and trained to deal with fires and the resulting emergency situations.

Fire brigade training is comprehensive, covering all aspects of the fire fighting plan. Members are familiarized with fire hazards including toxic, explosive and radiological hazards. Plant layouts and location of equipment is covered, and proper use of available equipment is taught. Training includes hands on fire fighting practice as well as participation in periodic drills. Training records are kept on all personnel receiving training.

The CPS fire brigade staffing, fire plans and training were developed in accordance with NRC guidance and National Fire Protection Standards and is sufficient to meet the needs of CPS.

4.8.4 Total Environment Equipment Survival

Smoke generation from a fire is a recognized hazard to both equipment and personnel. The effect of short-term smoke exposure on safety-related equipment has not been adequately studied to quantify the impact. However, the detrimental effects of short term smoke exposure on equipment is not believed to be significant.

Removal of smoke is an important part of fire fighting. CPS procedure 1893.04, Fire Fighting, provides detailed instructions on the appropriate method to accomplish smoke removal from all firezones in the plant. These plans include the use of portable smoke removal equipment as well as installed HVAC systems. This strong capability for effective smoke removal would prevent accumulated smoke from remaining in firezones for extended periods of time.

The impact of spurious actuation on the CPS safe shutdown capability was analyzed in the internal review of Information Notice 83-41, "Actuation of Fire Suppression System Causing Inoperability of Safety-Related Equipment". This evaluation concluded that the initiation of either manual or automatic fire protection measures will not preclude safe shutdown of the plant. Additionally, inadvertent initiation of any automatic sprinkler system will not place safe shutdown of the plant in jeopardy.

Operator training at CPS includes annual training on shutting down the plant from the remote shutdown panel (RSP). The RSP is located on a lower elevation in a different fire area than the control room and would not be affected by heat or smoke from a main control room fire. Access to the RSP area is through firezones that would also be unaffected by a control room fire. Permanently installed emergency lighting is provided on the route from the control room to the RSP. CPS has the capability for remote shutdown using either Division 1 or Division 2 equipment. The Division 2 shutdown method is also performed in areas that would not be affected by heat or smoke from a control room fire. While permanently installed emergency lighting is not provided in all areas necessary for Division 2 shutdown, portable lighting is staged for operator use.

4.8.5 Control Systems Interaction

The RSP has controls for Division 1 equipment necessary for safe shutdown. It also has a limited number of Division 2 motor operated valves. When the transfer switches on the RSP are activated, the main control room controls are electrically isolated from the associated equipment. In addition to the Division 1 safe shutdown method, CPS also has Division 2 safe shutdown capability. The Division 2 safe shutdown equipment have a remote shutdown circuit breaker control switch which electrically isolates the equipment from the main control room when closed. These features ensure a fire in the main control room would not prevent safe shutdown of the plant.

4.8.6 Adequacy of Analytical Tools

The COMPBRN IIIe correlations as contained in the FIVE methodology were used in the CPS FPRA. No additional evaluation is required for this issue.

4.9 References for Chapter 4

4-1. EPRI Project 3385-01, "Fire Risk Analysis Implementation Guide", Draft Report, January 1994.

4-2. EPRI TR-100370, "Fire-Induced Vulnerability Evaluation (FIVE)", Final Report, Revision 1, September 1993.

4-3. NSED Standard ME-08.00, "Thermo-Lag 330-1 Combustibility Evaluation Methodology Plant Screening Guide", Rev. 0.

4-4. NSED Calculation IP-M-0177, "Fire Loads for CPS Firezones", Rev. 3.

4-5. NSED Calculation IP-M-0178, "Firezone Boundaries and Floor Areas", Rev. 1.

4-6. Sargent & Lundy Interactive Cable Engineering (SLICE) system databases.

4-7. Holman, J.P., "Heat Transfer", 1976, fourth edition, McGraw-Hill.

4-8. CRC Handbook of Chemistry and Physics, 59th Edition, 1978-1979.

4-9. NSAC-178L, Fire Events Database For U.S. Nuclear Power Plants, Final Draft Report, December 30, 1991.

5. HIGH WINDS, FLOODS, TRANSPORTATION AND INDUSTRIAL HAZARDS

NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities", identifies acceptable methods to evaluate potential vulnerabilities due to high winds, floods, and transportation and nearby facility accidents. If a plant can be shown to be in compliance with the Standard Review Plan (SRP) applicable for these hazards, it is considered to be at low risk from these hazards. NUREG-1407 cites the 1975 Standard Review Plan as the basis for the review, but recognizing that more recent studies tend to represent better understanding of the phenomena involved, the CPS study considered the most recent standards. Alternately, the hazards can be evaluated directly (i.e., without regard to the SRP) and be shown to be of low risk significance.

Clinton Power Station received its operating license in 1986 and the licensing reviews were typically performed using the applicable portions of the Standard Review Plan from 1975 or later. As a result, the IPEEE review for these other hazards largely focused on demonstrating Standard Review Plan compliance.

The analysis for the IPEEE High Winds, Floods, Transportation and Industrial Hazards (Other Hazards) generally consisted of the following steps:

1) The CPS licensing basis with regard to the hazard was reviewed to determine what was originally analyzed and accepted. The licensing basis for purposes of this analysis is described in the CPS Updated Safety Analysis Report (USAR).

2) The nature of the hazard was reviewed to determine if it has adversely changed or our understanding of the phenomena has changed since the licensing of the plant such that it is believed to be more severe than originally anticipated.

3) The plant features that protect against these hazards were reviewed to determine whether they are in place, providing adequate protection for the plant (especially those that were taken credit for in the USAR analysis). Walkdowns of the plant facilities were conducted as a part of this review.

For situations where the existing plant capability to resist a particular hazard exceeded that of the potential demand, the direct threat from this hazard was considered acceptably small. Indirect threats such as loss of offsite power due to severe weather were often already accounted for by the initiating frequencies for internal events (e.g. the Loss of Offsite Power initiator) in the original Individual Plant Examination. The analysis outlined above meets the intent of the first 3 steps of the flowchart from figure 5.1 of NUREG-1407. This flowchart shows the recommended IPEEE approach for winds, floods and other external events.

5.1 High Winds and Tornado Analysis

The regulatory requirements regarding high winds and tornadoes have not changed since original issuance of the "Clinton Safety Evaluation Report (SER)", NUREG-0853. They are specified in Standard Review Plan (SRP) section 3.3.1, "Wind Loadings", revision 2 July 1981; SRP section 3.3.2, "Tornado Loadings", revision 2 July 1981; SRP section 3.5.1.4, "Missiles Generated by Natural Phenomena", revision 2, July 1981; and Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants", April 1974. Regulatory Guide 1.117, "Tornado Design Classification", revision 1, dated April 1978, describes those structures, systems and components that should be protected from the effects of the Design Basis Tornado.

Clinton Power Station meets the current revision of SRP 3.3.1 (rev. 2, July 1981) regarding wind loadings. These requirements were met by using a design wind of 85 mph and by using the procedures contained in ANSI A58.1-1972, "Minimum Design Loads for Buildings and Other Structures", and ASCE Paper No. 3269, "Wind Forces on Structures", to transform the wind velocity into pressure loadings for the design of Category I structures. This is discussed in CPS USAR section 3.3.1 and CPS SER section 3.3.1.

The USAR analysis for tornadoes examines two aspects:

1) The pressure loadings a tornado could place on safety-related structures. Included in the loading combination are the tornado wind pressure, tornado differential pressure, and tornado missile loads. This is discussed in USAR section 3.3.2.

2) Tornado missiles are discussed in section 3.5.2, in which the depth of penetration and deflection of walls are considered.

The tornado used for analysis in the USAR meets the Regulatory Guide 1.76 (April 1974) definition of a Design Basis Tornado for Nuclear Power Plants. This is the current revision of this regulatory guide. Standard Review Plan Section 3.3.2, "Tornado Loadings", has not been revised since revision 2, dated July 1981. The methods used for the loading combinations analysis from USAR section 3.3.2 produce numerical values that are identical to those calculated using the methodology found in SRP section 3.3.2. These loads are combined with other non-wind loads in the design analysis of the structural capability for Seismic Category 1 structures. Therefore, the design of Category 1 structures meets the current design basis tornado requirements.

This issue is discussed in CPS USAR section 3.3.2 and CPS SER section 3.3.2.

The tornado missile penetration and displacement discussion is provided in USAR section 3.5.2. The suite cf missiles used in the USAR (Table 3.5-3) is generally the suite of missiles from SRP 3.5.1.4, rev. 2 (July 1981), which repeats those from the November 24, 1975, revision of this SRP. The NRC staff concluded that the spectrum of missiles used in the CPS missile analysis is representative of missiles at the site and is therefore acceptable (SER section 3.5.1.3). Based on this discussion, the Seismic Category 1 structures are capable of protecting safety-related equipment from these limiting tornado missiles.

5.1.1 Identification of Plant Changes

In an effort to identify changes that have occurred to the plant that could affect the ability of the plant to withstand tornadoes and high winds the following tasks were performed:

1. The summaries of safety evaluations reported to the NRC from the time of the receipt of the operating license through December 1993, were reviewed for changes to the plant wind and tornado protection design. No changes to the high winds, tornadoes or tornado missiles protection scheme were noted.

2. The walkdowns described in the next section were made, in part, in an attempt to identify differences in the high winds and tornado missiles protection design from that described in the USAR. No differences were found.

5.1.2 Examination of Area Surrounding the Plant

A walkdown of the plant protected area (inside the security fence) was performed to identify any previously unidentified hazards. The potential missiles noted in this area were no more limiting than the missiles already considered in the USAR. Some of the items noted in this area were unattached railroad ties and a garbage dumpster. These items would be less limiting than some of the items considered in the USAR analysis because of their mass or head-on contact area. Objects that can have a small head-on contact area with a relatively large mass tend to be the most potentially damaging missiles, because of their ability to deliver a large amount of kinetic energy to a small impact area. A utility pole colliding on end with a safety-related structure is an example of a tornado missile from the USAR analysis. It would be more limiting than a railroad tie because it has a much larger mass. It is also more limiting than a dumpster because the dumpster would have a large contact area when colliding with safety-related structures.

The portion of the Diesel Generator Building and Control Building wall adjacent to the unit 2 hole was examined to determine whether there are any gaps in the missile protection scheme because of penetrations into what was supposed to be unit 2. See figure 5.1. No deficiencies were noted.

5.1.3 Conclusions Regarding High Winds and Tornadoes

Based on the foregoing discussion, it was concluded that Clinton Power Station meets or exceeds the applicable Standard Review Plan sections and Regulatory Guides for high winds and tornadoes. Therefore, the risk of wind or tornado induced core damage is acceptably low.

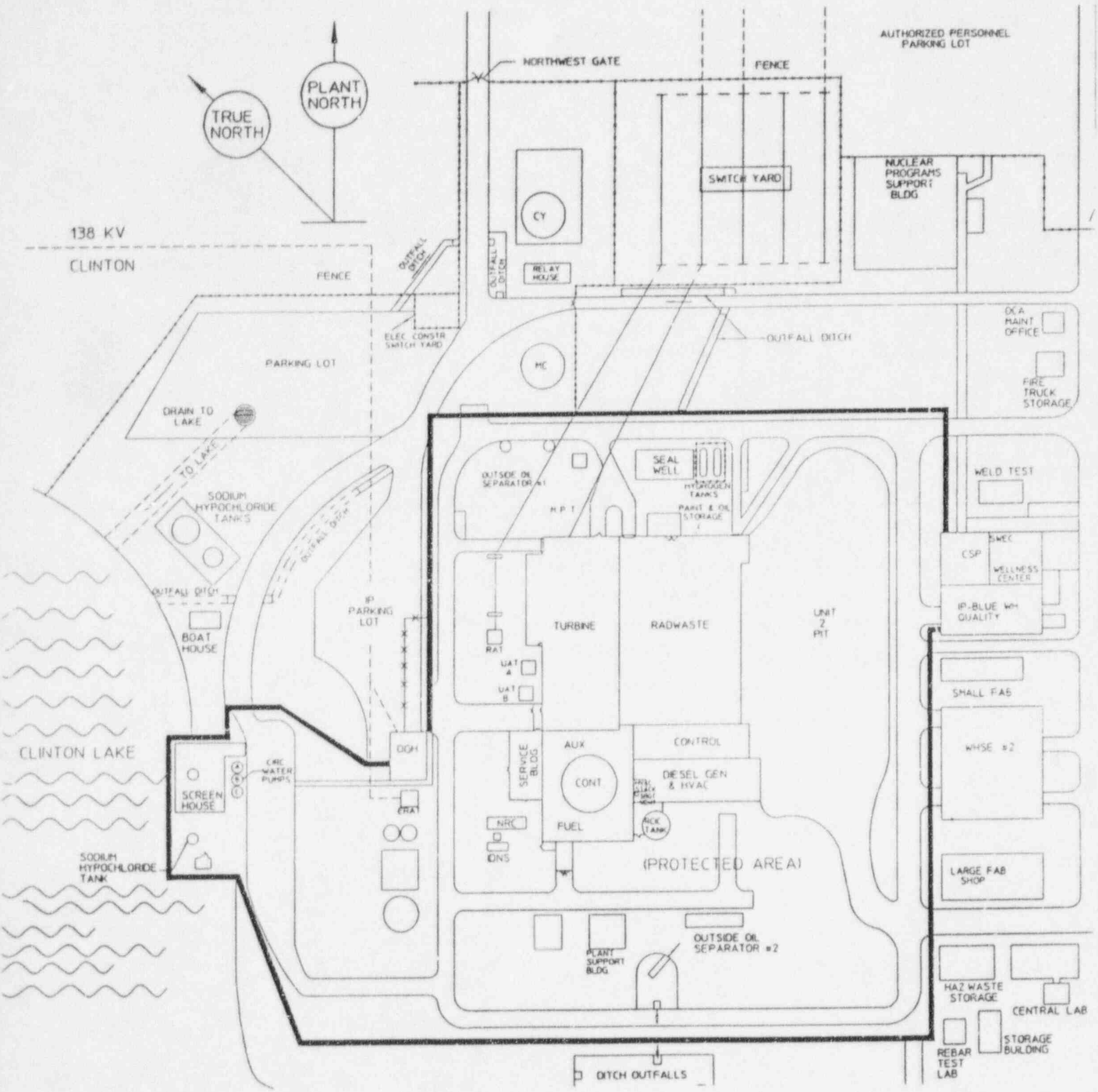


Figure 5.1

SITE MAP FOR CLINTON POWER STATION

5.2 Flooding Analysis

The regulatory requirements regarding the design basis flood have not changed dramatically since the issuance of the original Clinton Power Station Safety Evaluation Report (SER). They are specified in Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants", revision 2, dated August 1977, and Standard Review Plan (SRP) sections 2.4.2, "Floods", and 2.4.3, "Probable Maximum Flood", both revision 3, dated April 1989. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants", revision 1, dated September 1976, describes types of flood protection acceptable for the safety-related structures. SRP sections 2.4.2 and 2.4.3 were the only above-listed regulatory references modified since the time of licensing. The SRP updates reflect new criteria contained in more recent National Weather Service (NWS) Hydrometeorological Reports. In general, these latest NWS criteria call for higher rainfall intensities over shorter time intervals and smaller areas than have been previously considered.

CPS USAR section 2.4.3 discusses the Probable Maximum Flood (PMF) for CPS. This section shows the peak flood flow for Clinton Lake (USAR Figure 2.4-9) which was determined using a precipitation profile consisting of an antecedent storm followed by the Probable Maximum Precipitation (PMP) in the watershed flowing into Clinton Lake. The PMP used in the USAR was based on U.S. Weather Bureau Hydrometeorological Report No. 33 (April 1956). The maximum lake inflow determined, using the USAR method, was 173,615 cfs (USAR section 2.4.3.4) and the lake maximum discharge rate is stated to be 135,900 cfs. Appendix B of Regulatory Guide 1.59 lists what it considers to be conservative values of PMF peak discharge flows. For the Clinton Project the stated PMF peak discharge flow is 99,500 cfs which is considerably less than the peak discharge flow from the USAR. Because the lake discharge flow rate is controlled by the lake level at the dam, the USAR methodology yields results that are even more conservative than if the Regulatory Guide 1.59 PMF peak discharge flow was used.

The probable maximum flood level at the Station Site from the USAR analysis is calculated to be 708.9 ft. (USAR section 2.4.1.1). The grade elevation for the main part of the plant is located at the 737 ft. elevation. USAR section 2.4.10 describes waterproofing methods used in the circulating water screenhouse up to elevation 714 ft. for Shutdown Service Water Pump rooms. Even with coincident wind wave activity occurring with the maximum lake water level, the maximum wave runup is 713.8 ft. (USAR 2.4.3.6) which is below the top of the waterproofed portion of the Shutdown Service Water Pump rooms. The spillway rating curves shown in USAR Figure 2.4-8 show a steep increase in the dam discharge flow with increasing lake level related to the flow capacity of the auxiliary spillway. Although the curves only go up to lake elevation 710 ft. (corresponding to a total discharge flow of 161,103 cfs) the increase in discharge flow exceeds 20,000 cfs for each additional foot rise in lake level. The combination of a large dam discharge capacity coupled with the height of the flood barrier for the Screen House indicates that the CPS site could withstand a flood greater than the Probable Maximum Flood described in the USAR. Thus even with the new NWS data, with higher precipitation levels for short periods and small areas, the Shutdown Service Water Pump rooms would not be expected to flood. This conclusion is supported by the fact that the lake water inflow is determined by precipitation on the lake and the watershed flowing into it. The average precipitation values for larger areas such as these should not have changed significantly as a result of the newer NWS data.

CPS USAR section 2.4.2.3 discusses the effects of local intense precipitation. The ponding loads on the roofs of safety-related structures, even assuming the roof drains are plugged, are limited by the height of the parapet walls which surround the roofs (16 inches) plus the hydraulic head necessary for the water to flow over the parapet wall. The roofs of the safety-related buildings are designed to withstand the above loads assuming the winter probable maximum precipitation. Once the roof water level exceeds the level of the parapet, the roofs are able to accommodate a very large increase in rainfall with a relatively small increase in the roof water depth. Therefore the roof loadings due to more intense precipitation from the more recent NWS data would not increase the roof loadings significantly from the USAR analysis and it can be concluded that ponding loads would not cause roof failure.

The immediate areas surrounding the power block (including the Containment, Auxiliary, Fuel, Diesel Generator, Control, Turbine and Radwaste buildings) are graded to direct surface runoff away from the plant. As noted in the USAR, some ponding is expected in the areas enclosed by the roads and tracks near the plant. The estimated maximum water surface elevation around the power block is lower than the plant floor elevation of 737.0 feet except on the plant north side of the power block where the water elevation would be about 737.2 feet. The USAR recognizes that ponding in these areas could cause entry of water into the north side of the power block. Because there is no safety-related equipment located in the north side of the power block (the Radwaste and Turbine buildings), the safety significance of this event is expected to be minimal. Safety-related equipment required for emergency core cooling, including support systems, is located at the southern part of the power block in the Containment, Auxiliary, Fuel, Control and Diesel Generator Buildings. Water entering the northern part of the power block could run into the floor drain system or could flow down stairs or other openings into the basements of the radwaste or turbine buildings. Because of the large volume of these areas that would fill first, it is not expected that safety-related core cooling equipment would be adversely affected. More intense precipitation over a short period of time as reflected in the newer NWS data does not change this conclusion. Emergency core cooling systems (ECCS) located in the lowest elevation of the power block are surrounded by their own floodproof barriers. Support equipment for these ECCS is located in areas that are not susceptible to flooding (e.g. at higher plant elevations).

5.2.1 Identification of Plant Changes

In an effort to identify changes that have occurred to the plant that could affect the ability of the plant to respond to an external flood the following tasks were performed:

1. The summaries of safety evaluations reported to the NRC from the time of the receipt of the operating license through December, 1993, were reviewed for changes to the plant flood protection design. Few changes to the flood protection scheme were identified, and none of these changes were determined to have an adverse result on the flooding analysis.

2. The walkdowns described in the next section were made, in part, to identify differences in the flood protection design from that described in the USAR. No differences were noted.

5.2.2 Examination of Flood Protection Features

Walkdowns were performed to determine the general condition of the plant features which protect against external flooding. The focus of these walkdowns was on the Screenhouse, which, because of its low elevation and proximity to the lake, is exposed to potential flooding conditions. The following observations were made during the walkdowns:

1. The Shutdown Service Water (SX) pump rooms were examined to determine the condition of the walls and floors. The pump room walls are part of the flood barrier for the rooms. The SX system provides cooling water for safety-related cooling

loads in the plant. The condition of the walls and floors was good with no deficiencies noted.

2. The flood-proof doors leading into the A SX pump room, between the A and B SX pump rooms and between the B and C SX pump rooms were examined. The condition of the door seals appeared adequate.

3. The pipe tunnel that lies underneath the Shutdown Service Water pump rooms was observed. The tunnel is a flood barrier and communicates with the division 2 Shutdown Service Water pump room through a hatch in the floor of the pump room. The majority of the pipe tunnel lies below the normal level of Clinton Lake. Although there were locations where water was seeping into the tunnel at a slow rate, the overall flood integrity of the tunnel appeared adequate. Other than some small puddles, the floor of the tunnel was observed to be dry. The hatches leading to the pipe tunnel from the incomplete unit 2 side of the Screen House were observed to be in place. These hatches, located on the ceiling of the tunnel, protect the tunnel from the elements and are a flood barrier if the lake level rises above the level of the deck of the screenhouse (elevation 699'). These hatches were observed during a rain as a check of their leak integrity. They were found to be leaking, and a maintenance request was written to correct this condition. This maintenance work has been completed and the post maintenance leak check was satisfactory.

4. The SX pump room sump pump discharge piping was observed to pass through the SX pump room wall. These penetrations are located well above the floor of the SX pump rooms (drawing M05-1059 shows these penetrations to be at elevation 710 ft. while the floor elevation is 699 ft.) and have seals between the pipe and the penetration. The elevation of this penetration is higher than that for the maximum probable

flood, but is lower than the maximum wave runup used in the analysis. Backflow of water through the discharge line would be prevented by check valves downstream of the sump pumps. External to the SX pump rooms these discharge lines are vented which would prevent siphons from being formed. These sump pumps are supplied with power from non-safety-related motor control centers (MCCs) located in the Circulating Water Screenhouse in an area that does not have flood protection. Thus, for severe floods for which the lake level rises to the point that these MCCs fail, the sump pumps would be unable to pump out any water leaking into the SX pump rooms. However, this condition can be mitigated by operator action. CPS procedure 4303.02, "Abnormal Lake Level", section 4.2.2 currently directs that the screenhouse top hatch be prepared for use when lake level increases to the 696' msl elevation. This allows a means of access to the SX pump rooms to monitor conditions. So, although the sump pumps in the room may be non-functional, plant personnel could compensate for leaks into the SX pump rooms by bringing in portable pumping equipment.

5. The roof of the Screenhouse was examined to determine its general condition and to examine the roof drains. The roof is surrounded by a parapet which at its highest is approximately 16 inches taller than the roof. The roof drains have raised grates that could potentially become clogged (e.g. with leaves). As previously noted in Section 5.2, the roof has been analyzed and shown to be acceptable for this ponding load.

5.2.3 Conclusions Regarding External Flooding

Based upon the foregoing discussion, it was concluded that CPS has protection against external flooding which meets the Standard Review Plan criteria. Therefore, the risk of core damage from external flooding is acceptably small.

5.3 Transportation and Nearby Facility Accidents

This section discusses hazards to CPS as a result of transportation accidents; including highway and aircraft hazards. Industrial hazards to CPS involving nearby facilities that store hazardous materials are also discussed. Chemical hazards existing at the Clinton site are part of this discussion. Railroad hazards were not reviewed as part of the IPEEE review because of a separate commitment by Illinois Power to periodically review the rail traffic near CPS (see section 5.4).

5.3.1 Nearby Facility Accidents

SRP section 2.2.1-2.2.2, "Identification of Potential Hazards in Site Vicinity", revision 2 July 1981; and SRP 2.2.3, "Evaluation of Potential Accidents", revision 2 July 1981, discuss the reviews to be performed to evaluate external risks to the plant from external accidents. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release", June 1974, provides guidance for evaluating chemical hazards. Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants", revision 1 February 1978, covers explosive hazards. As a whole they are intended to identify nearby facilities and transportation routes that could pose a hazard to the safe operation of the nuclear power plant and to evaluate these hazards.

To identify the facilities near CPS that could pose a hazard to the safe operation of the plant due to accidents at these facilities, the "Annex V Hazardous Materials Contingency Plan to the DeWitt County/Clinton Emergency Operations Plan" (Reference 5-6) was reviewed. This plan was developed for responding to hazardous material incidents in DeWitt County by the DeWitt County Emergency Services and Disaster Agency and others. This document identifies the facilities in DeWitt County that store significant amounts of hazardous materials.

Regulatory Guide 1.78 states, "Chemicals stored or situated at distances greater than five miles from the facility need not be considered because, if a release occurs at such a distance, dispersion will dilute and disperse the incoming plume to such a degree that there should be sufficient time for the control room operators to take appropriate action". Figure 1 from Regulatory Guide 1.91 shows that facilities located at five miles or greater from the nuclear power plant site would have to have an explosive potential greater than 10,000,000 lbs of TNT to be considered an explosion hazard to the site, thus effectively precluding consideration of hazards in excess of 5 miles from the site. The Annex V Hazardous Materials Contingency Plan lists three facilities within a 5 mile radius of Clinton Power Station that have hazardous materials (Clinton Power Station is one of the three).

5.3.1.a Offsite Hazards

Terra International (formerly Shields Soil Service) located in DeWitt, Illinois, stores anhydrous ammonia for use as a fertilizer. USAR section 2.2.3.1.3 analyzed anhydrous ammonia releases from Shields Soil Service. The USAR analysis was based on a 40 ton tank of anhydrous ammonia existing at the Shields Soil Service. With conservatively assumed meteorological conditions, the control room operators could be incapacitated by a complete release of ammonia from this tank. This same USAR section shows a probability analysis for such an incapacitating event with the result being that such events are estimated to occur with a frequency of 5E-8 events per year. This frequency is sufficiently low to justify not considering this event as a design basis event. For this single tank rupture case there was only one wind direction and stability class that could cause incapacitation of the control room operators. This was a stability class G wind coming from the ENE direction which has a probability of occurrence of 0.005.

Since the time of the USAR analysis Shields Soil Service has added an additional 18,000 gallon (nominally 40 ton) tank to their ammonia storage capacity. The effect of this change would be to double the likelihood of a rupture of a single ammonia tank at this facility. It also creates the possibility of having an accident where both ammonia tanks could be released (e.g. by rupturing a manifold line common to both tanks). The estimated frequency of control room incapacitation events from this facility from single tank releases is now 1E-7 events per year, twice the previous case. Although the probability of a release of the contents of both tanks is less likely than for a single tank, there may be more wind stability classes that could cause incapacitation of the control room operators because of the larger quantity of material involved. A very conservative estimate of the rate of control room incapacitation from the release of both tanks was determined to be 2.7E-7 events per year. This estimate is the sum of the probabilities of control room incapacitation from several two tank failure mechanisms (e.g. failure of a common manifold line and rupture of one tank causing failure of the second). This estimate is conservative because it is based on using the probability of all wind stability classes from the ENE and uses conservative estimates of the probability of occurrence of two tank failure mechanisms. When the single and double tank failure cases are added together the frequency of a control room incapacitation event is 3.7E-7 events per year. This frequency is still acceptably low. Although Shields Soil Service has been purchased by Terra International, this facility continues to store the same quantities of materials.

The second facility is the Corn Belt F.S. also located in DeWitt, Illinois. Per the Hazardous Materials Contingency Plan this facility stores propane in the quantity range 100,000-999,999 lbs. Propane would not cause incapacitation of the control room operators at this distance because of its low toxicity. Propane is also an explosive hazard. A bounding calculation is shown here to demonstrate that it does not pose an explosive hazard to CPS. Assuming that 1,000,000 lbm of propane exists at this facility and using the methodology from regulatory guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants", produces the following results. Multiplying 1,000,000 lbm of propane by 2.4 yields an equivalent mass of TNT of 2,400,000 lbm. Using the formula R>=45(W^(1/3)), where R is the distance in feet from an explosion of W lbm of TNT, the safe distance is calculated to be 6,025 feet. Because the Village of DeWitt is approximately 2 miles from Clinton Power Station, this facility poses no hazard to CPS per the criteria contained in Regulatory Guide 1.91.

5.3.1.b Onsite Hazards

Clinton Power Station has a number of materials on hand that are potentially hazardous. Diesel fuel, gasoline, soda ash, hydrated lime, sodium hypochlorite, sodium hydroxide, sodium aluminate and sulfuric acid are listed in the Annex 5 Hazardous Materials Contingency Plan.

Diesel fuel is stored in the Diesel Generator Building for use with the emergency diesel generators. Each of the diesel generator fuel storage tanks is located in its own room with fire rated walls and ceiling. Under the conditions it is stored it does not present an explosion hazard. Fire hazards associated with it are covered under the Fire portion of the IPEEE.

Gasoline is stored well outside the protected area fence in underground tanks. These storage conditions make the formation of a large explosive vapor cloud of gasoline very unlikely. Therefore it is not a credible explosive hazard for CPS. Gasoline also has relatively low toxicity and is not a credible toxic hazard to the control room operators under these storage conditions.

Sulfuric acid has a low vapor pressure (well under 10 torr at 100°F) that can eliminate it from consideration as an airborne toxic hazard to the control room operators. With this low vapor pressure, only an insignificant amount could be transported as a vapor from a spill site. This conclusion is consistent with Regulatory Guide 1.78, which states, "For chemicals that are not gases at 100°F and normal atmospheric pressure but are liquids with vapor pressures in excess of 10 torr, consideration should be given to the rate of flashing and boiloff to determine the rate of release to the atmosphere and the appropriate time duration of the release."

Soda ash, hydrated lime, sodium hydroxide and sodium aluminate are materials with low vapor pressures that would not present a toxic hazard to the control room for that reason. Transportation of solids (as some of these materials are) by airborne mechanisms into the control room would be precluded by the normal alignment of the Control Room Heating Ventilating and Cooling system which has filters capable of removing powders and dusts.

Sodium hypochlorite is used for water treatment to control the biofouling of piping systems at CPS. This task was previously handled using gaseous chlorine, which was analyzed in the CPS USAR. Sodium hypochlorite which appears in liquid form is much less hazardous and would not present a hazard to the control room operators because of its low volatility.

In addition to the materials listed in the Annex V Hazardous Materials Contingency Plan, CPS has a bulk hydrogen storage facility located on the plant north side of the Radwaste Building. The hydrogen is used for purging and makeup of the main generator hydrogen cooling system. The storage facility is discussed in CPS USAR section 10.2.2.2.1. It is located in its own fenced in area, approximately 432 feet from the nearest building containing safety-related or Class 1E components. This separation distance, together with the open space location, precludes adverse safety effects resulting from the unlikely event of any explosion or fire.

Based upon the foregoing discussion, the materials stored onsite do not pose a hazard to control room habitability or an explosive hazard that can affect safety-related equipment.

5.3.2 Highway Transportation Accidents

The evaluation of hazardous material accidents is not limited to the fixed facilities discussed in the previous sections. Transportation facilities that transport hazardous materials in the vicinity of the nuclear plant site also present a potential for hazardous material accidents. Regulatory Guides 1.78 and 1.91 also cover the evaluation of transportation accidents. In general, transportation routes that do not pass within five miles of the nuclear plant do not need to be considered for their accident potential to the nuclear plant, for the reasons discussed in the previous section.

Illinois Highways 54, 10 and 48 are the only state or federal highways that pass within five miles of the site. Highway 48 passes just within the 5 mile radius for CPS. Highway 54 has the closest approach to the plant site, passing within approximately 3/4 mile. Specific information regarding the types of hazardous materials passing by the site on these highways is not readily available. Therefore this analysis is based on a review of local facilities to determine the types of hazardous materials that are likely to be transported by highway near CPS. None of these highways are logical routes for traffic between major cities. The major cities in the area are directly connected by interstate or U.S. highways that do not pass within 5 miles of the site. The logical routes between these cities are shown in table 5.1.

Table 5.1
LOGICAL HIGHWAY TRAFFIC ROUTES BETWEEN
MAJOR CITIES IN CENTRAL ILLINOIS

	Decatur	Springfield	Bloomington	Champaign
Decatur	NA	I-72	US 51	I-72
Springfield	I-72	NA	I-55	I-72
Bloomington	US 51	I-55	NA	I-74
Champaign	I-72	I-72	I-74	NA

Traffic between communities outside the perimeter of highways defined by I-72, I-74 and I-55 probably account for a small percentage of the traffic on highways 10, 54 and 48. Thus it is judged that most of the traffic passing by CPS on these highways is local in origin or destination. To obtain an estimate of the hazardous material traffic passing near CPS the "Annex V Hazardous Materials Contingency Plan to the DeWitt County/Clinton Emergency Operations Plan" was reviewed. This document was prepared by the DeWitt County Emergency Services and Disaster Agency (ESDA). It lists facilities in DeWitt County that store significant quantities of hazardous materials, the materials that they store, and the associated quantity range. This facility list is dated March 1993, which was the latest available at the time of the review. To pare down the list, the following methods were used to eliminate materials from further consideration as a hazard to Clinton Power Station:

1. Chemicals stored in quantities no greater than 100 lbm were eliminated. This was based on the assumption that materials would be transported in quantities no greater than would be stored at a single facility. While this may not be true in all cases, small quantities of these materials stored at a few facilities in the county implies that the quantity per shipment is small or that bigger shipments are made

infrequently. Regulatory Guide 1.78 states, "In the evaluation of control room habitability during normal operation, the release of any hazardous chemical to be stored on the nuclear plant site in a quantity greater than 100 lb should be considered". Elimination from review of chemicals stored in quantities less than this would be consistent with the Regulatory Guide.

2. Chemicals with a vapor pressure known to be less than 10 torr (1 torr=1 mm Hg) were eliminated. With a volatility this low it is reasonable to assume that a toxic concentration in the control room could not be formed from a spill occurring on the nearest highway for anything but the most extremely toxic substances (e.g. nerve gases). Elimination of chemicals with a low vapor pressure is consistent with Regulatory Guide 1.78.

3. Some chemicals were judged as not being toxic enough to cause incapacitation of the control room operators through airborne mechanisms in the quantities that are found locally. Chemicals falling in this category include methanol, ethyl alcohol, gasoline, diesel fuel, propane, lubricating oil, transformer oil, parafinnic oil, mineral spirits and hydrofluosilicic acid. Some of these commodities are flammable hazards; however, they are not shipped in quantities capable of causing a damaging overpressure event on safety-related structures. This conclusion is based on the methodology found in Regulatory Guide 1.91. At the closest approach of highway 54 to CPS, an explosion involving approximately 90 tons of hydrocarbon fuel under optimal conditions would be required to produce a 1 psi overpressure at the plant site. The safety-related structures are capable of withstanding well in excess of this pressure. Standard tank trucks do not carry this large of a quantity of material.

4. DeWitt County, like much of central Illinois, has a significant agricultural economy. As a result, a large percentage of the hazardous material traffic in DeWitt County is composed of agricultural chemicals. Agricultural chemicals that are normally liquid or solid in form would be expected to have low enough toxicity or volatility to allow safe handling by farmers. Generally these chemicals are poured through open air from their original containers into the containers from which they will be applied. If these chemicals were highly dangerous through a combination of their toxicity or volatility they would not be able to be handled in this manner.

The list of agricultural chemicals appearing in the DeWitt County Annex V Hazardous Materials Contingency Plan were pared down using the following criteria:

A. Solid agricultural chemicals were eliminated from consideration altogether. Solids typically have a vapor pressure less than 10 torr. The exceptions to this rule are unlikely to be used as constituents of agricultural chemicals.

   Also eliminated from consideration in this category are agricultural chemicals that are suspensions or solutions of chemicals that are normally solids. An example of these would be atrazine which is a solid. For ease of handling this chemical is often sold in liquid form. Such mixtures typically would result in a vapor pressure for the hazardous constituent no greater than it would be in the pure state.

B. All liquid or solid agricultural chemicals stored in quantities of less than 1000 lb were eliminated from the highway transportation accidents survey. This is consistent with Table C-2 of Regulatory Guide 1.78 which, for type C control rooms, materials with toxicity limits of 50 mg/m^3

and distances of 0.7 to 1 mile from the control room, shows that a 1000 lbm quantity would need to be considered. This tabular value was chosen because the nearest highway to the CPS control room is state highway 54 which passes within approximately 3/4 mile of the plant. The CPS control room under normal mode of operation has an air exchange rate of 0.7404 volumes per hour, less than a type C control room. Agricultural chemicals used locally would be expected to have a fairly high toxicity limit or have a very low volatility that would offset their toxicity. This assumption again is based on the need for these chemicals to be formulated such that they can be safely handled by farmers while pouring them through the open air.

After paring down the list of hazardous materials from the DeWitt County Hazardous Materials Contingency Plan using the preceding criteria, two commodities remained for further evaluation as transportation hazards, Chlorine and Anhydrous Ammonia.

Chlorine is used by a number of (mostly municipal) facilities in the county for water treatment. The facilities in DeWitt County that are listed in the Plan as having chlorine are the City of Clinton Water Treatment Plant, the City of Clinton Sanitary District, the Weldon Water Department, the Farmer City Light, Power & Water Utilities and the Woodlawn Country Club (in Farmer City). The first four facilities were contacted and it was determined that they do not receive chlorine shipments by a route that passes within 5 miles of CPS. The Woodlawn Country Club is listed as having 100 lb cylinders of Chlorine each of which is considered to be too small a quantity to be a hazard to CPS per the Regulatory Guide 1.78 criteria. Thus highway chlorine shipments are not believed to be a transportation hazard for Clinton Power Station.

Six facilities in DeWitt County are listed as having anhydrous ammonia in the DeWitt County Hazardous Materials Contingency

Plan. They are Marco Chemicals (in Clinton), Corn Belt F.S. (in Kenney), Corn Belt F.S. (in Wapella), Maroa Ag. (in Maroa), Vigoro Industries (in Kenney) and Shields Soil Service (in DeWitt). In conversations with personnel at these facilities, it was determined that none of these facilities receive their ammonia from routes that would pass within five miles of Clinton Power Station with the exception of Shields Soil Service. Shields Soil Service, as discussed in section 5.3.1 "Nearby Facility Accidents", is located within a five mile radius of CPS and receives anhydrous ammonia shipments from Eastern Illinois. Thus, Shields Soil Service is the likely closest point of travel for these shipments. Personnel at Shields estimated that approximately 50 shipments of 22 tons each were delivered to their facility in the past year.

The risk of incapacitating the control room operators from an accident involving a truck bringing anhydrous ammonia to Shields Soil Service is estimated as follows.

1. An estimate of the accident rate for trucks carrying hazardous materials was obtained from a document entitled, "A Modal Economic and Safety Analysis of the Transportation of Hazardous Substances in Bulk," prepared by Arthur D. Little Incorporated, dated May 1974. This study estimated a tank truck loss-of-lading rate of 27 accidents per billion miles of truck travel for the reporting period 1968-1972. This study was based on the number of loaded tank trucks involved in a spill during that period. This number was divided by an estimate of the number of loaded tank truck miles for this period. Interestingly, this study stated that fewer than 2%, or one in 50, of the reported accidents involving loaded tank trucks resulted in a spill.

   The data from the previous study is old. However it appears likely that the loss-of-lading rate for trucks has declined since 1972. This is borne out by US Department of

Transportation statistics which show that the fatality rate for large trucks went from 1.2 fatalities per 100,000,000 miles traveled in 1975 to 0.4 fatalities per 100,000,000 miles traveled in 1991. These fatalities were for the truck occupant. From the same source, the total number of trucks involved in fatal crashes has also declined. Trucks were involved in fatal crashes 5.4 times per every 100,000,000 miles traveled in 1977. In 1991 the rate was 2.9 times per 100,000,000 miles. Both of these accident rates steadily declined during the 1980's, indicating real improvements in traffic safety.

Based on the improvement in traffic safety since the 1970's, it appears the loss-of-lading statistic from the 1974 Arthur D. Little study is a conservative estimate of the truck loss-of-lading rate.

2. This loss-of-lading rate was used to calculate the frequency of a release from a truck delivering anhydrous ammonia to Shields Soil Service. There are approximately 3 miles of this truck route within a five mile radius of the plant (i.e. the 3 miles of highway 54 immediately to the east of DeWitt).

   To estimate a release frequency, the number of loaded truck miles per year over this segment is multiplied by the loss of lading rate.

   Prel = 50(trucks/yr) X 3(miles) X 27E-9(releases/truck-mi)
   = 4.05E-6 releases/yr

3. To potentially affect control room operators, the wind would have to be blowing from the spill site toward the plant with sufficient stability to keep the ammonia cloud from being diffused. The 3 mile stretch of road is all located in the ENE direction from the site, the same direction as Shields

Soil Service. The control room habitability study done for Shields Soil Service, in support of the USAR analysis, determined there was only one wind direction and stability class that could cause incapacitation of the control room operators. The tank assumed to rupture at Shields was larger than the 22 ton tank trucks supplying Shields, therefore the tank trucks would be no more limiting. The wind of concern was a wind from the ENE with a stability class G. These conditions have a probability of occurrence of 0.005 (0.5 %). Combining this with the release frequency from above yields a rate of potential control room incapacitation of 4.05E-6 X 0.005= 2.03E-8 events/yr. This is a rate low enough that transportation of ammonia in the area surrounding CPS does not need to be considered in the design of CPS. Even when combined with the ammonia release hazard from Shields Soil Service itself, the risk of incapacitating the control room operators remains acceptably small (3.9E-7 events/yr).

5.3.3 Aircraft Hazards

The methods for reviewing aircraft hazards are delineated in SRP section 3.5.1.6, "Aircraft Hazards", revision 2, July 1981. This SRP section describes an aircraft crash probability threshold below which aircraft crash hazards do not need to be considered in the design of the plant. This requirement is met if the probability of aircraft accidents resulting in radiological consequences greater than 10 CFR part 100 exposure guidelines is less than about 1E-7 per year.

The analysis of aircraft hazards in the USAR (section 3.5.1.6) looked at traffic on nearby federal low altitude airways and nearby private airstrips. In addition, it considered the number of operations from nearby commercial airports, Decatur and Bloomington-Normal. The number of operations from these two commercial airports was less than the SRP threshold of consideration, 1000 X D^2, where D is the distance from the site to the airport in miles. The USAR analysis determined a probability of an airplane crash onto the safety-related structures of the site from the two private airstrips within five miles of the site (the Martin and Thorp airstrips) and from the low altitude federal airways near the site. Each of these crash probabilities was below the SRP 3.5.1.6 threshold for consideration in the design of the plant.

In the more recent survey done for the IPEEE analysis the aircraft traffic volumes for each of the categories of flight hazards considered in the USAR had declined. Below is a comparison of the existing USAR data compared to the 1994 survey results.

Table 5.2
COMPARISON OF AIR TRAFFIC DATA
FROM USAR TO IPEEE RESULTS

CATEGORY	USAR DATA	IPEEE (1994) SURVEY DATA
Traffic over federal low altitude airways near CPS	80 flights/day estimate	40 flights/day estimate
Traffic from Martin airstrip	4-6 operations/wk	12 operations/yr estimate
Traffic from Thorp airstrip	4-5 operations/wk	2-4 operations/wk in warm months estimate
Traffic from Decatur airport	91,826 operation/yr	58,101 operations/yr
Traffic from Bloomington-Normal airport	86,970 operations/yr	28,995 operations/yr

To prove analytically that SRP 3.5.1.6 has been met, the analysis performed in support of the USAR discussion was examined to determine what the effect of the new air traffic survey data would be. This calculation considered the probability of an airplane crash into CPS from the following sources: the nearby federal low altitude airways and traffic from the airstrips within five miles of the site.

Using the 1994 low altitude airway data with the numbers for the worst case airway from the calculation. For the airway.

Paw =(3E-9)(40)(365)(0.00842)/9.21
= 4.0E-8 crashes/yr
where
3E-9 is the crash rate per aircraft-mile
40 is the number of flights per day
365 is the number of days per year
0.00842 is the effective crash area of the site in square miles
9.21 is the width of a low altitude flight path in miles

For Martin airstrip.

Pm =(1.2E-8)(12)(0.00842)
= 1.0E-9 crashes/yr
where
1.2E-8 is the crash rate per (operation mile squared)
12 is the number of operations per year
0.00842 is the effective crash area of the site in square miles

For Thorp airstrip.

Pt =(1.2E-8)(100)(0.00842)
= 1.0E-8 crashes/yr
where
1.2E-8 is the crash rate per (operation mile squared)
100 is the number of operations per year
0.00842 is the effective crash area of the site in square miles

Summing these three crash sources, as is done in the USAR calculation, yields a crash probability 5.1E-8 crashes/yr. Because the calculation was based upon having the crash area associated with a two unit site, the value for unit 1 alone would be less. In any case it is less than the acceptance criteria from SRP 3.5.1.6 which states "Aircraft accidents which could lead to radiological consequences in excess of the exposure guidelines of 10 CFR Part 100 with a probability of occurrence greater than about 1E-7 per year should be considered in the design of the plant."

5.4 Issues Not Included In the Other Hazards Analysis

A number of issues have not been included in the Other Hazards analysis of the IPEEE. These are briefly discussed in this section.

Railroad transportation accidents were not evaluated in this study because of Illinois Power's commitment to review the traffic of hazardous materials shipped by rail near Clinton Power Station every three years. The last railroad hazards study, documented in Illinois Power letter U-602412, dated February 9, 1995, covered shipments in the 1994 calendar year. The hazardous materials identified during this survey (88 carloads of phosphoric acid) were determined not to present a hazard to the CPS control room per the Regulatory Guide 1.78 criteria.

Clinton Power Station is located in a region of the country that has no active volcanoes. Therefore volcanoes are not considered to be a hazard for Clinton Power Station, and no further review was performed.

The design of Clinton Power Station includes features to protect against lightning. Lightning protection is provided for the station buildings and the HVAC stack. The lightning protection system is bonded to the station ground mat to provide an adequate low impedance path to lightning surges to ensure that the potential rise during lightning strikes is limited to reasonable values that equipment and personnel can safely withstand. A review of CPS Licensee Event Reports (LERs) issued since issuance of the CPS operating license identified no instances where lightning either caused or contributed to the event. Based upon the above factors, lightning was not reviewed as part of the IPEEE.

Severe temperature transients (extreme heat, extreme cold), severe weather storms (ice storm, hailstorm, snowstorm, dust storm, sandstorm), external fires (forest fires, grass fires) and extraterrestrial activity (meteorite strikes, satellite falls) need not be considered in the IPEEE per the guidance contained in NUREG 1407.

5.5 Conclusions of the Other Hazards Analysis

Based upon the reviews performed for the Other Hazards analysis, Illinois Power believes that CPS meets the Standard Review Plan requirements regarding these hazards, and therefore has an acceptably low risk from these hazards.

5.6 References for Chapter 5

5-1. NUREG-0853, "Safety Evaluation Report Related to the Operation of Clinton Power Station", February 1982.

5-2. NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, Final Report", June 1991.

5-3. NUREG-0800, "Standard Review Plan", various dates as discussed in text.

5-4. Hydrometeorological Report No. 33, "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1000 Square Miles and Durations of 6, 12, 24 and 48 Hours", U.S. Weather Bureau and U.S. Army Corp of Engineers, April 1956.

5-5. Clinton Power Station Procedure 4303.02, "Abnormal Lake Level", rev. 4, April 18, 1994.

5-6. "Annex V Hazardous Materials Contingency Plan to the DeWitt County/Clinton Emergency Operations Plan", March 1993.

5-7. "1992 Motor Vehicle Crash Data from FARS and GES", U.S. Department of Transportation.

5-8. COM-74-11271, "A Modal Economic and Safety Analysis of the Transportation of Hazardous Substances in Bulk", Arthur D. Little, Incorporated, May 1994.

5-9. Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants", April 1974.

5-10. Regulatory Guide 1.117, "Tornado Design Classification", rev. 1, April 1978.

5-11. ANSI A58.1-1972, "Minimum Design Loads for Buildings and Other Structures", 1972.

5-12. ASCE Paper No. 3269, "Wind Forces on Structures", 1961.

5-13. Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants", rev. 2, August 1977.

5-14. Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants", rev. 1, September 1976.

5-15. M05-1059 Sheet 3, "P&ID Floor & Equipment Drains Screen House", rev. J, February 3, 1988.

5-16. Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release", June 1974.

5-17. Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants", rev. 1, February 1978.

5-18. Illinois Power Letter U-602412, "Illinois Power Evaluation of Potential Transportation Accidents", February 9, 1995.

6. LICENSEE PARTICIPATION AND INTERNAL REVIEW TEAM

6.1 IPEEE Program Organization

The Clinton Power Station IPEEE program was performed and managed by Illinois Power Company (IP). The IPEEE team is part of the Nuclear Station Engineering Department (NSED), located at the plant site and its members have been involved in all aspects of Clinton activities. Two team members maintain qualifications for performing shift duties in the main control room, one a Senior Reactor Operator (SRO) and the other a Shift Technical Advisor (STA). This involvement enhances the ability of the IPEEE team to remain well informed of actual plant conditions and assures that the IPEEE study accurately represents the plant. All IPEEE team members participated in the development of the IPE for Internal Events.

A second team composed of senior IP personnel performed an independent review of the IPEEE products. This team, referred to as the IPEEE Independent Review Team (IIRT) was composed of personnel from various on-site departments. Most of the review team have held SRO licenses at CPS. Similar to the IPEEE team, all members of the review team are located at the plant site.

Consultants were used to augment technical expertise and provide technical advice, training, and review of the methods, results and documentation. The consultants used were provided through the Individual Plant Evaluation Partnership (IPEP) which is composed of Tenera, L.P., Fauske and Associates, and Westinghouse Electric Corporation. These organizations were the primary contractors to the Industry Degraded Core Rulemaking (IDCOR) program and have had extensive experience in risk assessment as well as perspectives that come only from experience with analysis of many plants. IPEP has a business agreement with EQE International for seismic support. EQE is one of the predominant seismic consultants in the country. The IPEP provided individuals that were technical experts in specific aspects of risk analysis and also provided a Senior Management Support Team (SMST) to provide technical review of the IPEEE program products. Science Applications International Corporation (SAIC) provided assistance with the fire analysis through an Electric Power Research Institute (EPRI) tailored collaboration arrangement.

Technology transfer from the consultant to IP employees was considered an important part of the IPEEE program. The major work tasks were performed by the CPS IPEEE team members. Technology was transferred and experience gained throughout the IPEEE program. This approach has improved Illinois Power's in-house risk assessment capability.

The relationship of the various organizations that participated in the IPEEE is shown in figure 6.1.

6.1.1 IPEEE Team Description

The IPEEE team is a task oriented, self-directed work team within the Nuclear Station Engineering Department (NSED). The primary IPEEE team members have been at CPS since construction and start-up testing. They are listed below along with a brief description of their applicable experience:

P. E. Walberg, P.E., Senior Engineer and IPEEE team coach, Bachelor of Science degree in Mechanical Engineering, 29 years experience in nuclear power in the following areas; nuclear navy, engineering, licensing and safety, and probabilistic risk assessment.

E. E. Tiedemann, P.E., Engineer II, Bachelor of Science degree in Mechanical Engineering, active STA certification, 17 years experience in nuclear power in the following areas; construction, system engineering, operations and probabilistic risk assessment.

C. H. Mathews, Project Engineer, Bachelor of Science degree in Nuclear Engineering, SRO license for CPS, 15 years experience in nuclear power in the following areas; reactor engineering, plant operations, plant startup testing, control room simulation and probabilistic risk assessment.

M. E. O'Flaherty, Engineer II, Master degree in Business Administration, Bachelor of Science degree in Nuclear Engineering, 14 years experience in nuclear power in the following areas; naval prototype operations, system engineering, reactor engineering and probabilistic risk assessment.

A. J. Hable, P.E., Engineer II, Bachelor of Science degree in Mechanical Engineering, 13 years experience in nuclear power in the following areas; technical assessment of licensing issues, independent safety engineering group and probabilistic risk assessment.

6.1.2 Seismic Review Team

The Seismic Review Team (SRT) for the Clinton Seismic Margins Assessment was made up of four Illinois Power engineers; two from the IPEEE team and two with Civil/Structural engineering backgrounds. The SRT is described in section 3.1.1.d.

6.2 Composition of Independent Review Teams

The following sections discuss the various review groups used in support of the IPEEE project along with relevant information on the members. The review teams collectively have extensive knowledge of plant design, operations and risk assessment practices.

6.2.1 IPEEE Independent Review Team (IIRT)

The IPEEE Independent Review Team (IIRT) is an internal group of experienced IP personnel at the supervisory level and is located at the CPS site. The purpose of the team was to review the IPE supporting documentation as well as this report, to assure accurate representation of CPS design, operating history, operator response, maintenance practices and recovery actions in the IPEEE study. In order to assure independence, none of the IIRT members were involved with producing any of the products reviewed.

The IIRT was composed of six members. The chairman was formerly the Director of Nuclear Safety & Analysis, four of the other members have CPS SRO licenses, while the sixth member has extensive maintenance experience.

The IIRT members have diverse backgrounds and represent the following departments: operations, engineering, maintenance, and nuclear training. The position titles of the members are listed below along with a short summary of their experience.

Business Process Coordinator (previously Director of Nuclear Safety & Analysis (L&S) and Director of Engineering Projects (NSED)), Bachelor of Science degree in Mechanical Engineering, Professional Engineer, 23 years experience in nuclear power in the following areas; nuclear navy prototypes, construction, start-up, field engineering, engineering projects, and licensing and safety.

Operations Task Coordinator (OPS), licensed SRO for CPS, 24 years nuclear navy and operations experience, including shift supervisor.

Senior Instructor-Training (NTD), licensed SRO for CPS, 25 years experience in nuclear navy, operations, and nuclear training.

Senior Engineer (previously Supervisor of NSSS Systems) (NSED), Bachelor of Science degree in Nuclear Engineering, licensed SRO for CPS, 17 years nuclear navy, operations, and engineering experience.

Senior Engineer (previously Supervisor of Nuclear Engineering) (NSED), Master of Science degree in Nuclear Engineering, licensed SRO for CPS, 21 years nuclear fuels and reactor engineering experience.

Project Manager Maintenance Rule (Maint), 21 years nuclear navy, CPS start-up, field engineering, and maintenance experience.

While the IIRT was most knowledgeable about plant design, operation and maintenance because of their backgrounds, they also gained an appreciation of risk assessment methods because of their reviews of the IPE and IPEEE. This has improved the site wide understanding of plant risk and is an asset in risk assessment applications.

The diverse background and extensive experience of this review group provided many substantive technical, editorial, and program enhancing comments during the course of the IPEEE evaluation.

6.2.2 Consultant Involvement

The prime consultant for the Clinton IPEEE was the Individual Plant Evaluation Partnership (IPEP).

The IPEP had several major responsibilities.

1) Assist in correct and consistent implementation and interpretation of IPEEE guidance as applied to Clinton.

2) Provide an IPEP Senior Management Support Team (SMST) consisting of senior IDCOR people to provide a quasi-independent review of the CPS IPEEE. This role helps to provide the IPEEE with an industry overview perspective.

EQE International produced the CPS seismic studies discussed in section 3. One study compared the Review Level Earthquake to the Safe Shutdown Earthquake utilized at CPS. The other study examined CPS soils issues. EQE also assisted the Seismic Review Team with the conduct of the Seismic Margins Assessment walkdowns. Their involvement ensured consistency with industry accepted methods for conducting the walkdowns.

The role that the IPEP performed helped to ensure the program was conducted and managed in a manner that fully satisfies the intent of the IPEEE program.

Science Applications International Corporation (SAIC) provided support for completion of the fire PRA, through an Electric Power Research Institute (EPRI) tailored collaboration project. SAIC provided training in fire modeling techniques and performed the main control room analysis. SAIC participated in the development of the fire PRA methods and has considerable experience in this area.

6.3 Areas of Review and Major Comments

The areas of review team expertise were previously discussed under the respective review teams in section 6.2. The IIRT reviews focused more on the accuracy of the IPEEE products as they relate to design and operation, while the contractor reviews were intended to address the adequacy of the IPEEE methods. The following types of comments were received during the review process:

1. Thoroughness: The review teams noted instances where pertinent aspects had not been considered in the analysis, or where it wasn't obvious that they had been considered.

2. Reasonableness: The review teams noted instances where assumptions made or methods used appeared questionable. Usually the response to this type of comment involved providing additional supporting information.

3. Clarity: The review teams provided editorial comments to make the text more readable.

The comments produced as a result of the IIRT review along with their resolutions have been retained as part of the IPEEE supporting documentation.

6.4. Resolution of Comments

Comment resolutions were incorporated into the supporting documentation at each stage of the project, before approval of each respective document. The final reports were more readable and more complete after inclusion of review teams' comments. This will assist the ongoing effort of the IPEEE as it will be easier for additional IP personnel to use the results of the IPEEE study.

To summarize, the independent review teams concluded that the IPEEE study is thorough and meets the intent of G.L. 88-20, Supplement 4.

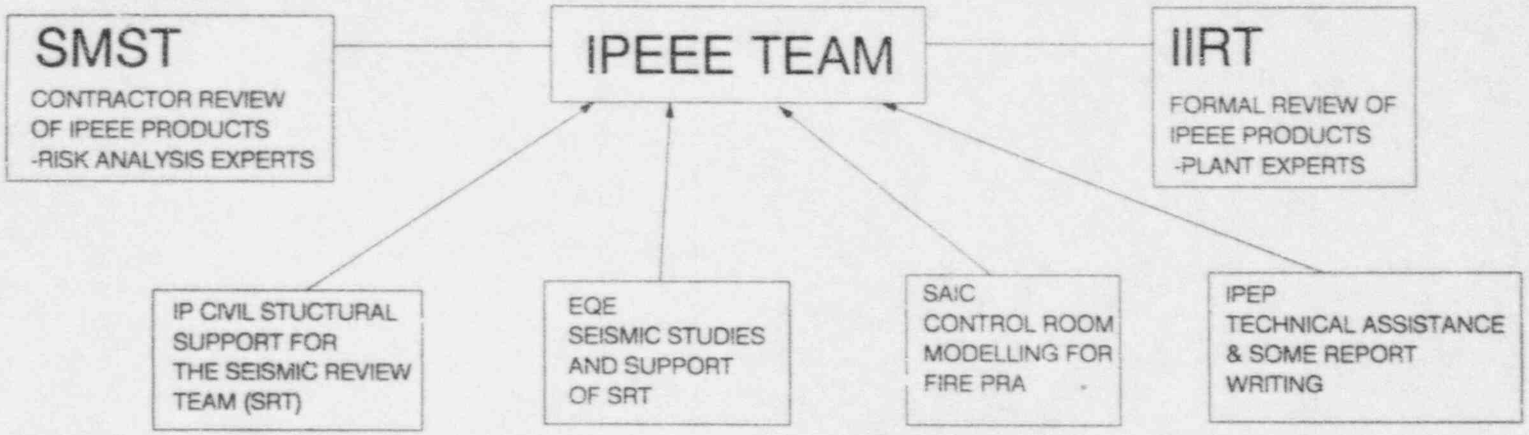


FIGURE 6.1

ORGANIZATIONS PARTICIPATING IN THE IPEEE

7. PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

This section discusses improvements made to the plant as a result of the IPEEE. Also discussed in this section are plant improvements that will be made as a result of the Thermo-Lag issue, because this work affected the Fire PRA results.

7.1 Plant Changes as a Result of the Seismic Margins Assessment

No improvements to the plant were identified as a result of the Seismic Margins Assessment because the plant was determined to be fully capable of attaining safe shutdown conditions after the Review Level Earthquake (RLE).

7.2 Plant Changes as a Result of the Fire PRA

No changes were identified directly as a result of the Fire PRA. However plant changes will be made as a result of the Illinois Power response to generic letter 92-08, "Thermo-Lag 330-1 Fire Barriers" (Reference 7-1). Specifically, cables that were previously routed from the division 2 inverter through the division 1 cable spreading room and then through the division 3 switchgear room will be rerouted during Refueling Outage 6, currently scheduled for the fall of 1996. This rerouting will remove them from the division 3 switchgear room altogether. The results of the Fire PRA discussed in section 6 are based on the plant configuration after the modification. Without the modification the core damage frequency from fires is estimated to be 1.3E-5 events/yr. The post modification core damage frequency (as reported in section 4) is 3.3E-6 events/yr, a 76% decrease. This 76% reduction is overstated in that it does not take credit for the enhanced detection and suppression provided by the hourly firewatch of this firezone. Additionally, no credit was taken for the Thermo-Lag installed on the division 1 cables which would provide some level of protection.

7.3 Plant Changes as a Result of the Other Hazards Analysis

No plant modifications were made as a result of the Other Hazards Analysis. As a result of an IPEEE flood barrier walkdown, repair of a leak path existing on a hatch over the shutdown service water pipe tunnel was made. Given the large volume existing in the tunnel below the level of any active components it is not expected that this leakage source would have prevented the operation of any safety related equipment during a probable maximum flood had one occurred.

7.4 Unique Safety Features

The conservative CPS design contributes to the low risk from the hazards evaluated in the IPEEE. The seismic margins assessment had a favorable outcome largely because the original CPS design included consideration of an SSE ground acceleration anchored at 0.25g. As discussed in section 3, the broadened SSE used for design actually envelopes the review level earthquake (anchored at 0.3g) at the plant floor levels. In addition, because the CPS design is of a more recent vintage it underwent extensive seismic reviews during the original design process, including interaction analysis walkdowns. CPS has a relatively low risk from internal fires because the original plant design generally had good physical separation of the different electrical divisions. For the Other External Hazards, such as flooding and high winds, the CPS design continues to meet the Standard Review Plan requirements. The 0.25g SSE is a design earthquake larger than most US nuclear plants. However, none of the aforementioned aspects of the CPS design can be considered to be unique, rather they are standard features of many recent US nuclear plant designs.

7.5 References for Chapter 7

7-1. NRC Generic Letter 92-08, "Thermo-Lag 330-1 Fire Barriers", December 17, 1992.

8. SUMMARY AND CONCLUSIONS

Illinois Power has performed an Individual Plant Examination for External Events for Clinton Power Station. The methodology used in this report follows the guidelines contained in NUREG 1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities". The bulk of the work was completed by in-house Illinois Power Company employees with contractors utilized for limited technical and review support.

A seismic margins assessment approach was used to demonstrate that the plant can be brought to safe shutdown conditions after the CPS review level earthquake (ground motion anchored at 0.3g).

PRA methods were used to analyze the risk from internal fires. The core damage frequency estimated from this analysis is 3.26E-6 events/yr, which is less than that estimated for internal events (current value is 6.0E-6 events/yr).

The analysis of other IPEEE hazards, including high winds, tornadoes, external floods, transportation accidents, and nearby facility accidents, was accomplished by reviewing the plant and environs against current regulatory requirements regarding these hazards. Based upon the reviews for these other hazards, it is concluded that CPS meets the applicable Standard Review Plan requirements and therefore has an acceptably low risk.

The overall conclusion of the IPEEE study is that CPS has a low risk of core damage from external hazards. As a result of the IPEEE, Illinois Power has gained an understanding of severe accident behavior resulting from external hazards, that can be used in accident management strategies and future risk assessment studies.

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