TABLE OF CONTENTS

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TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF FIGURES

LIST OF FIGURES (Continued)

12.0 RADIATION PROTECTION

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE(ALARA)

12.1.1 POLICY CONSIDERATIONS

The management of the Cleveland Electric Illuminating Company recognizes its responsibility and authority to operate and maintain the Perry Nuclear Power Plant in a manner that provides for the safety of plant personnel and the public. In accordance with this philosophy, the company endorses the policy of keeping radiation exposure as low as reasonably achievable (ALARA).

It is the intent of the ALARA Program to demonstrate that reasonable measures have been taken to maintain the radiation exposure of plant personnel and members of the public as far below the regulatory limits as reasonably obtainable.

The ALARA policy will be implemented at the PNPP. The Supervisor of the Health Physics Unit provides technical support in the health physics aspects of plant operation. This includes all health physics activities, monitoring plant radiation health and safety practices and participating in the preparation of health physics procedures, and reports and manuals required by the Company and regulatory agencies.

The Radiation Protection Engineer is responsible for ensuring that plant practices are in full compliance with the radiation health and safety requirements of a nuclear installation. This includes supervision of activities associated with radiation safety, preparing radiation safety procedures, and reports and manuals required by the Company and regulatory agencies.

The Superintendent, Plant Operations is responsible for directing the supervision of all personnel engaged in the operation, maintenance, technical and radiological activities concerned with the equipment, structures and properties comprising the plant.

The Manager of the Perry Nuclear Power Plant has complete responsibility for all onsite activities in connection with the safe and efficient operation and maintenance of the PNPP. He is responsible for managing the affairs of the plant so as to ensure reliable, efficient and safe operation.

The Plant Manager, Superintendent, Plant Operations, Radiation Protection Engineer, and the Supervisor of the Health Physics Unit will work in concert to implement the Company's ALARA policy. However, it is the responsibility of all plant supervision to enforce the requirements for keeping radiation exposures ALARA, and the responsibility of each individual to comply with these requirements.

12.1.2 DESIGN CONSIDERATIONS

12.1.2.1 Design Features

Ensuring that occupational radiation doses are ALARA was begun during the early design of PNPP. By familiarizing design engineers with ALARA concepts, and by providing review of the design by radiation protection personnel, the operation of the PNPP will result in personnel doses that are ALARA and will fulfill the intent of Regulatory Guides 8.8 and 8.10.

The following design criteria have been, and will continue to be, adhered to:

a. Access labyrinths are provided for rooms housing equipment that contains high radiation sources to preclude a direct radiation path from the equipment to accessible areas.

b. Piping penetrations, ducts and voids in radiation shield walls are located to preclude the possibility of streaming from a high to low radiation area or otherwise will be adequately shielded.

c. Shielding discontinuities caused by shield plugs, concrete hatch covers and shield doors to high radiation areas are provided with offsets to reduce radiation streaming.

d. Radioactive piping is routed through high radiation areas where practicable, or in shielded pipe chases in low radiation areas.

e. Sufficient work area and clearance space is provided around equipment to permit ease of servicing.

f. Instruments requiring in-situ calibration will not normally be located in high radiation areas.

g. Non-radioactive equipment which requires servicing will not normally be located in proximity with potentially radioactive equipment.

h. Spread of contamination from radioactive spillage is minimized by:

   1. Providing steps at doorways to contain the spills.

   2. Providing a floor drain system which collects and routes the liquid to the liquid waste processing system for proper handling.

   3. Using step-off pads between clean and contaminated areas.

   Decontamination of an area is facilitated by use of materials and coatings which lend themselves to cleaning by standard methods.

i. Natural traps which could be potential pockets of corrosion product activity are minimized in pipe and ducts by avoiding sharp bends, rough finishes and cracks.

j. Shielding is provided for all equipment which is anticipated to be normally radioactive. The dose levels are designed not to exceed 10 CFR 20 requirements under the worst operating conditions of the plant.

k. Movable shielding, such as lead bricks and steel plates, will be available on the site in case it is ever needed.

l. Remote handling of radioactive materials is provided wherever it is needed and practicable.

m. All process piping that may contain radioactive fluids is routed and dimensioned on piping system drawings, thereby eliminating any field run radioactive piping.

n. Redundant components for the radwaste processing system have been supplied to increase flexibility in plant operations and decrease radiation doses during maintenance.

o. Radioactive components can be flushed to decrease radiation levels and subsequent personnel doses from maintenance.

p. The radwaste handling system has been designed to be a completely remote operation with remote manual overrides in case a failure occurs.

12.1.2.2 Utilization of Experience Gained from Operating Facilities to Ensure that Radiation Doses are ALARA

An important aspect in the design of a nuclear facility is the feedback information obtained from plants currently operating. For this design feedback, information has been obtained directly from operating facilities and from governmental and industrial publications. The following areas of information represent the type of feedback useful in plant design:

a. Operational radiation levels.

b. Trends in radiation levels associated with years of operation plant type, plant size, power levels and plant design.

c. Radiation zones as determined by occupancy requirements and actual radiation levels.

d. Location of components with respect to plant operability.

e. Reliability of components.

f. Adequancy of plant layout in terms of traffic patterns, and space allocation such as around radioactive components requiring maintenance and inspections, and pipe routing.

g. Number of plant employees associated with different tasks and the resulting man-rem doses.

Feedback information is used in the design of the plant to identify potential problem areas. Such problems are reviewed with regard to current design and, where applicable, modifications made to the design to eliminate or decrease the potential for such problems to arise.

12.1.2.3 Design Guidance Given to Individual Designers

The following design guidance has been given to individual designers of systems associated with radiation protection:

a. Follow guidance given in Regulatory Guides 8.8 and 8.10.

b. Use maximum realistic source strengths in all calculations.

c. Consider all sources that significantly contribute to the dose at a particular point.

d. Place instrument readouts and items needing routine surveillance or attention in as low as practicable radiation zone.

e. Make the design sensitive to the expected procedures of plant personnel under normal operation and anticipated occurances.

12.1.2.4 Design Review Procedures

Independent design reviews are performed by competent specialists to assess the implementation of the design criteria and further ensure that the final design

is compatible with maintaining occupational radiation doses ALARA. All designs influencing radiological control in the plant are reviewed by competent professionals in the area of radiation protection.

As indicated in Section 12.1.2.3, all personnel associated with design aspects influencing radiation protection have been given the basic ALARA principles. After the preliminary design and layout of the system, a shielding engineer analyzes the various radiation sources and specifies shielding as necessary to conform to the appropriate radiation zone requirements. Radiation protection personnel review the system in terms of the total plant operation and specify necessary changes to keep occupational radiation doses ALARA. Both shielding engineers and radiation protection personnel are part of the overall design team and report their findings directly to the design project management for the PNPP. These findings are considered in conjunction with design requirements from disciplines not directly associated with radiation control to determine what modifications, if any, should be made to promote ALARA radiation doses.

12.1.3 OPERATIONAL CONSIDERATIONS

12.1.3.1 ALARA Training Program

The Radiological Training Program at PNPP will help implement the Company's ALARA policy. The Training Program will assure that workers understand how radiation protection relates to their jobs and all workers will have frequent opportunities to discuss radiation safety with the Health Physics Unit personnel when the need arises.

All work at PNPP involving systems that contain, collect, store or transport radioactive materials and may cause radiation exposure will require a Radiation Work Permit (RWP). The RWP will help implement the Company's ALARA policy by defining the radiological hazards and requiring specific radiological precautions. The RWP also becomes a record of how various jobs were performed and the radiological problems associated with specific jobs. By reviewing expired RWP's, recommendations can be made to change procedures or equipment that will result in lower radiation exposures in the future.

Training and RWP requirements will ensure that the Company's ALARA policy is fulfilled. Some techniques that may be used are:

a. Temporary shielding, such as lead sheets or blankets may be used. Temporary shielding will be used only if total exposure, which includes installation and removal of the shielding, will be effectively reduced.

b. Prior to performing maintenance work, consideration will be given to flushing and/or chemically decontaminating in order to reduce CRUD levels and personnel exposure.

c. Dry run training will be used for jobs with exceptional radiological problems to familiarize personnel with the work they must perform at the job site. These techniques will assist in improving efficiency and minimize the amount of time spent in radiation areas. These efforts will be documented to improve future efforts.

d. As much as practicable, work will be performed outside radiation areas. This includes items such as reading instruction manuals or procedures, adjusting tools or jigs, repairing valve internals and prefabricating components.

e. For long term repair jobs, consideration will be given to establishing remote observation stations to assist supervising personnel in monitoring work progress from a lower radiation area.

f. On some jobs, special tools or jigs will be used so that work will be performed more efficiently to reduce errors, thus minimizing the time spent in a radiation area. Special tools may be used to increase the distance from a radiation source to the worker, thereby reducing the exposure.

g. Entry and exit control points will be established in areas with low levels of radiation to limit the exposure of personnel donning protective equipment or generally preparing to work in such areas. The access control points will be designed to minimize the spread of contamination from the work areas.

h. Protective clothing and respiratory protection equipment will be selected to minimize the discomfort of workers. Efficiency will increase and less time will be spent in radiation areas.

i. Personnel will be assigned self-reading dosimeters to estimate exposure during a work assignment.

j. On jobs where general area radiation levels are high, health physics coverage may be required during the period of work.

k. On intricate jobs, especially those which involve high radiation levels, preplanning will include estimation of the man-rem needed to complete the job. At the completion of the work, a debriefing session will be held with the personnel that performed the work (when practical) in an effort to determine how the work could have been completed more efficiently and with less radiation exposure.

12.1.3.2 Radiation Zoning and Access Control

During normal pull power operation, the maximum whole body dose rates within the plant that might be received by operating personnel, contractors and authorized visitors will depend upon the following zone designations:

Zone	Designation	Dose Rate (mrem/hr)
I	Unlimited Occupancy	≤ 0.5
II	Normal Continuous Occupancy	≤ 2.5
III	Controlled, Limited Access (4hr/wk)	≤ 25
IV	Controlled, Limited Access (1hr/wk)	≤ 100
V	Restricted Areas	>100

12.1.3.2.1 Zone I

Zone I areas can be occupied by station personnel or visitors on an unlimited time basis. Health hazards due to ionizing radiation in Zone I are absolutely minimal. Access control to Zone I areas is due to security considerations.

12.1.3.2.2 Zone II

Zone II areas are not posted areas but will have higher radiation levels than Zone I. Entry to all Zone II areas begins by passing through the main plant access control point on the B1 level of the Control Complex. Access to Zone II areas requires Radiation Work Permit authorization. If work is planned requiring an individual to spend greater than 40 hours in any week in a Zone II area, the area must be surveyed and posted by the Health Physics unit prior to starting work.

12.1.3.2.3 Zone III

These areas shall be posted with "Caution - Radiation Area" signs affixed conspicuously at all entry points. Dose rates of significance within a Zone II area shall be noted at the point of entry. Access requires Radiation Work Permit authorization prior to entry. If work by an individual might exceed the 4 hour limit in any week, the area must be comprehensively surveyed by the Health Physics unit prior to starting work to identify problem areas and potential high dose areas.

12.1.3.2.4 Zone IV

Access shall be controlled such that routine work shall be limited to short periods of time (F one hour/week). The point of entry shall have "Caution - Radiation Area" signs conspicuously posted. Dose rates of significance shall be noted at the point of entry. Access requires Radiation Work Permit authorization prior to entry. Any localized spot within the Zone IV area that exceeds the 100 mrem/hr criterion shall be conspicuously posted with a "Caution - Hot Spot" sign that displays the dose rate of the hot spot.

12.1.3.2.5 Zone V

Access shall be restricted so that entry might not occur unless absolutely necessary. The area shall be comprehensively surveyed by the Health Physics Unit prior to entry. Access requires Radiation Work Permit authorization. All access points shall be conspicuously posted with a sign or signs bearing the words, "Caution - High Radiation Area." In areas where dose rates range between 100 and 1000 mrem/hr, the access point shall be posted and barricaded in accordance with paragraph 20.203(c)(2) of 10CFR20. For areas where dose rates generally exceed 1000 mrem/hr, access shall be precluded by locked doors. Access keys will be maintained by the Shift Supervisor and/or the Health Physics Unit.

12.1.3.2.6 General Guidelines for Maintenance in High Radiation Areas

a. As much work as possible will be done outside high radiation areas.

b. Provisions shall be taken to drain, flush, and/or decontaminate equipment prior to performing maintenance, if practicable.

c. Pre-job planning will be undertaken for work in the High Radiation Areas to minimize man-rem exposure and reduce the potential for erroneous actions.

d. When possible, job debriefings shall be performed to improve procedures and limit future radiation exposure should the job need to be done again.

e. Records of exposure data, contamination problems, airborne hazards, and internal exposure will be maintained and reviewed to provide guidance for similar future situations.

f. Special tools or materials that may significantly reduce exposures should be utilized.

g. All efforts shall be made to uniformly distribute radiation doses to personnel within related grades or classifications so that specific personnel are not unduly limited for operation or maintenance late in the calendar quarter.

h. During work in a High Radiation Area, Health Physics unit cognizance is mandatory. Portable survey instruments shall be used to keep a continual account of dose rates while the area is occupied.

12.2 RADIATION SOURCES

12.2.1 CONTAINED SOURCES

12.2.1.1 Source Terms

With the exception of the vessel and drywell shields, shielding designs are based on fission product and activation product sources consistent with Section 11.1. For shielding, it is conservative to design for fission product sources at peak values rather than an annual average, even though experience supports a lower annual average than the design average (Reference 1). It should be noted that activation products, principally nitrogen 16, control shielding calculations in most of the primary system. In areas where fission products are significant, conservative allowance is made for transit decay, while at the same time providing for transient increase of the noble gas source, daughter product formation and energy level of emission. Areas where fission products are significant relative to nitrogen-16 include: the condenser off-gas system downstream of the steam jet air ejector, liquid and solid radwaste equipment, portions of the reactor water clean up system, and portions of the feedwater system downstream of the hotwell including condensate treatment equipment.

For application, the design sources are grouped first by location and then by equipment type (e.g., reactor building, core sources). The following paragraphs represent the source data in various pieces of equipment throughout the plant. General locations of equipment are shown in the general plant arrangement drawings of Sections 1.2 and 12.3.

12.2.1.2 Reactor Building

12.2.1.2.1 Radiation Sources from the Reactor Core

The information in this section defines a reactor vessel model and the associated gamma and neutron radiation sources. This section is designed to

provide the data required for calculations beyond the vessel. The data selected were not chosen for any given program, but were chosen to provide information for any of several shield program types. In addition to the source data, calculated radiation dose levels are provided at locations surrounding the vessel. This data is given as a potential check point for calculations by shield designers.

12.2.1.2.1.1 Physical Data

Table 12.2-1 presents the physical data required to form the model in Figure 12.2-1. This model was selected to contain as few separate regions as possible to adequately portray the reactor. Table 12.2-1 provides nominal dimensions and material volume fractions for each boundary and region in the reactor model. To describe the reactor core, Table 12.2-1 provides thermal power, power density, core dimensions, core average material volume fractions and reactor power distributions. The reactor power distributions are given for both radial and axial distributions. This data contains uncertainties in the volume regions near the edge of the core. The level of uncertainties for these regions is estimated at 20 percent.

12.2.1.2.1.2 Core Boundary Neutron Fluxes

Table 12.2-2 presents peak axial neutron multigroup fluxes at the core equivalent radius. The core equivalent radius is a hypothetical boundary enclosing an area equal to the area of the fuel bundles and the coolant space between them. The peak axial flux occurs adjacent to the portion of the core with the greatest power. As shown by the data in Table 12.2-1, this point is below the core mid-plane. Since this data is calculated with a core equivalent radius, the flux represents a mean flux in the azimuthal angle around the core. While the flux within any given energy group is not known within a factor of 2, the total calculated core boundary flux is estimated to be within ±50 percent.

12.2.1.2.1.3 Gamma Ray Source Energy Spectra

Reactor core gamma ray source energy spectra are as follows:

a. Core Spectra

Table 12.2-3 presents average gamma ray energy spectra per watt of reactor power in both core and non-core regions. In Table 12.2-3 - item A, the energy spectra in the core is presented. The energy spectra in the core respresents the average gamma ray energy released by energy group per watt of core thermal power. The energy spectra in Mev per sec per watt can be used with the total core power and power distributions to obtain the source in any part of the core.

The gamma ray energy spectra includes the fission gamma rays, the fission product gamma ray and the gamma rays resulting from inelastic neutron scattering and thermal neutron capture. The total gamma ray energy released in the core is estimated to be accurate to within ±10 percent. The energy release rate above 6 Mev may be in error by as much as a factor of ±2.

b. Post Operation Gamma Ray Energy Spectra

Table 12.2-3, item B, gives a gamma ray energy spectrum in Mev per sec per watt in spent fuel as a function of time after operation. The data was prepared from tables of fission product decay gamma fitted to integral measurements for operation times of 10^8 seconds or approximately 3.2 years. To obtain shutdown sources in the core, the gamma ray energy spectra is combined with the core thermal power and power distributions. Shutdown sources in a single fuel element can be obtained by using the gamma ray energy spectra and the thermal power that the element contained during operation.

c. Non-Core Gamma Ray Energy Spectra

In Table 12.2-3, item C, the gamma ray energy spectra in the cylindrical regions of the reactor from the core through the vessel are given. The energy spectra is given in terms of MeV per sec-watt-m^3 at the inside surface and outside surfaces of the region. This energy spectrum multiplied by the core thermal power is the gamma ray source. The point on the inside surface of the region is the maximum point within the region. In the radial direction, the variation in source intensity may be approximated by an exponential fit to the data on the inside and outside surfaces of the region. The axial variation in a region can be estimated by using the core axial variation. The uncertainty in the gamma ray energy spectra is due primarily to the uncertainty in the neutron flux in these regions. The uncertainty in the neutron flux is estimated to vary from approximately ±50 percent at the core boundary to a factor of ±3 at the outside of the vessel. The calculations were carried out with voids beyond the vessel. The presence of shield materials beyond the vessel will cause an increase in the gamma source on the outside of the vessel.

12.2.1.2.1.4 Gamma Ray and Neutron Fluxes Outside Vessel

Table 12.2-4 presents the maximum axial neutron and gamma ray fluxes outside the vessel. The maximum axial flux occurs on the vessel opposite the elevation of the core with the maximum power level. This elevation can be located using the data from Table 12.2-1. The fluxes at this elevation are based on a mean radius core and do not show azimuths angle variations. The calculational model for these fluxes assumed no shield materials beyond the vessel wall. The presence of shield materials will significantly alter the neutron fluxes in the lower end of the neutron energy spectrum. The gamma ray calculations include gamma ray sources from all of the cylindrical regions between the center of the core and the edge of the vessel. While the uncertainties in a given energy group flux may be a factor of ±3, the uncertainties in the total integral flux are estimated to be within a factor of 2.

12.2.1.2.2 Radioactive Sources in the Reactor Water, Steam and Off-Gas

The radioactive sources in the reactor water, steam and off-gas are discussed in Sections 11.1. This material provides the concentrations during normal operation of the radioisotopes in the reactor vessel or leaving the reactor vessel.

12.2.1.2.3 Reactor Water Cleanup System

The radioactive sources in the cleanup system are the result of the activity in the reactor water in transit through the system or accumulation of radioisotopes removed from the water. Components for this system include regenerative and non-regenerative heat exchangers, pumps, valves, filter demineralizers, and the backwash receiving tank. The system is described in Section 5.4.8. The accumulated sources in the filter-demineralizers, backwash receiving tanks and heat exchangers are given in Table 12.2-5.

The source is present in the filters and receiving tank during all modes of operation. Therefore, backwashing capability is provided to remove the residual activity for effective radwaste handling.

12.2.1.2.4 Main Steam System

All radioactive materials in the main steam system result from radioactive sources carried over from the reactor during plant operation. In most of the components carrying live steam, the source is dominated by N-16. In components where the N-16 has decayed, the other activites carried by the steam become significant. During plant shutdown, there is a residual activity resulting from prior plant operations.

12.2.1.2.5 Radioactive Sources in the Spent Fuel

The radiation source for spent fuel is given in Table 12.2-3 in terms of MeV per sec per watt. The design calculation is carried out for a mean element for an appropriate time after shutdown.

12.2.1.2.6 Other Radioactive Sources

Additional radioactive sources in the reactor building are:

a. TIP

The material composition and radiation sources for the traversing incore probe system (TIP) are provided in Tables 12.2-6, 12.2-7, 12.2-8 and 12.2-9. The radiation source is based upon location within the core and residence time. As indicated in the tables, the TIP system consists of three components for shielding calculations: the fissionable material, nonfissionable material and the cable. Sources are provided for each component for 100 seconds irradiation and zero seconds decay.

b. Reactor Startup Source

The reactor startup source is shipped to the site in a special cask designed for shielding. The source is transferred under water while in the cask and loaded into Berylium containers. This is then loaded into the reactor while remaining under water. The source remains within the reactor for its lifetime. Thus, no unique shielding requirements after reactor operation are required.

12.2.1.3 Auxiliary Building

12.2.1.3.1 Radioactive Sources in the RHR and RCIC sytems

The basic sources in the safeguard systems are the result of the radioactive materials in the reactor water or steam being transported to the system. The design basis sources for this equipment assume the total activity is the concentration of reactor water or steam decayed for the appropriate time interval times the total volume of steam or water in the equipment.

The gamma source strengths in the residual heat removal (RHR) and reactor core isolation cooling (RCIC) systems were evaluated for both the shutdown and steam condensing modes of operation.

In the steam condensing mode of operation these systems condense reactor steam and recirculate the reactor coolant to remove heat during hot standby operation. The sources in the RHR and RCIC systems are the nitrogen-16 activities in the volumes of reactor steam contained in these systems. These sources are provided in Table 12.2-5.

The design gamma source strengths from fission products in the engineered safeguard systems following shutdown is typified by the source strength and fission product inventories for the system given in Table 12.2-5. In the shutdown mode the RHR system recirculates reactor coolant to remove decay heat. The system is operated from approximately 2-4 hours after shutdown until the end of the refueling period. The source in the RHR system is the activity in the volume of reactor water contained in the system. This includes the increase of activity as a result of depressurization.

The system includes three RHR pumps and two heat exchangers. The highest radiation levels during reactor shutdown occur at the heat exchangers during the cool down period. At other times or in other modes of operation, except hot standby, the sources are considerably decreased.

Source strengths of equivalent concentration in downstream piping are conservative for use in layout and shielding design of pipe chases.

12.2.1.4 Intermediate Building

12.2.1.4.1 Radioactive Sources in the Spent Fuel

The radiation source for spent fuel is given in Table 12.2-3 in terms of MeV per sec per watt.

12.2.1.4.2 Fuel Pool Cooling and Cleanup System

Sources in the fuel pool cooling and cleanup (FPCC) system are a result of transfer of radioactive isotopes from the reactor coolant into the spent fuel pool during refueling operations.

The FPCC system removes fission products and activated corrosion products as well as decay heat from the spent fuel pool. This system consists of surge tanks, pumps, heat exchangers and precoat filter demineralizers. The activities in the FPCC components are given in Table 12.2-5.

12.2.1.5 Turbine Building

12.2.1.5.1 Turbine Building Sources

The radioactive sources in the turbine building are the result of the carryover of radioisotopes in the steam. These isotopes are distributed throughout the equipment in the turbine building. During power operation of the equipment, the most significant radioisotope source is N-16. Both the concentration and the energy per disintegration of N-16 contribute to the importance of this isotope during operation. Fission product sources are not important except in equipment where the steam or condensate passage time is sufficiently long to permit the N-16 to decay or where the physical processes preferentially separate the fission product isotopes.

Turbine component N-16 inventories are listed in Table 12.2-10.

12.2.1.5.2 Turbine Power Complex Sources

The radioactive sources in the turbine power complex are primarily fission products and activated corrosion products carried over from the steam. These sources are contained in condensate lines, filters and demineralizers, and associated tanks. Activities in the condensate system are listed in Table 12.2-5.

12.2.1.6 Radwaste Building

The source of activity which enters the radwaste system is the activity in the reactor coolant including activation and fission products. The sources used for shielding radwaste system components are listed in Table 12.2-5.

In general, the maximum activity possible, even though remote, has been used in most pieces of equipment. For example, in the liquid radwaste system reactor coolant has been used in each tank of the collector subsystems because of the interconnections of these tanks. In normal practice, undiluted reactor water would not be present in the subsystem.

12.2.1.7 Off-gas Building

Radioactive sources in the off-gas treatment system result from the decay of the noble gases and radioisotopes carried with the non-condensible gases. Radioactive sources entering the system have been defined at the design basis of an annual average release of 100,000 μ/Ci/sec after a 30 minute holdup. An expected noble gas delay time of one minute was used from the vessel nozzle to the exit of the steam jet air ejector. It was assumed that the solid fission product isotopes were washed out in the main condenser. The N-16 activity at this point is primarily the result of the driving steam to the air ejector. Partitioning of other coolant activites is assumed to be like the N-16 with the distribution in the main condenser equivalent to the mass flows of the steam.

The radioactive sources in the gas treatment system are a function of the sources entering the system, the operational mode of the equipment piece and the residence time of the gas in the equipment. As a guide to the sources (other than N-16) in the gas treatment system, the licensing topical report NEDO-10734 "A General Justification for Classification of Effluent Treatment System Equipment as Group D" was used. Since this report was prepared as a conservative justification for a more conventional classification of the equipment, the conservative sources used in the report represent source levels which would be reached only after long term operation with design basis release rates.

The sources contained in the components of the off-gas system are listed in Table 12.2-5.

12.2.1.8 Sources Resulting From Design Basis Accidents

The radiation sources from design basis accidents are discussed in Chapter 15.0.

12.2.2 AIRBORNE RADIOACTIVE MATERIAL SOURCES

The airborne radioactivity sources that contribute to the plant effluent releases through the radioactive waste management system and the plant ventilation system are described in Section 11.3. The primary sources of airborne radionuclides are leakage of reactor coolant and main steam. Contributions to the airborne radionuclide concentrations in plant buildings due to leakage from the radioactive waste management system are small compared to the contribution from primary coolant.

12.2.2.1 Reactor Building (Outside of Drywell)

The main sources of activity leakage to the reactor building atmosphere during normal operation are conservatively estimated to be 2,000 lbs/hr of steam leakage from the safety relief valves and 34,000 lb/month of steam from the RCIC turbine exhaust. The path for the resultant activity to reach the containment atmosphere is through the suppression pool. Other sources of leakage can be identified in the RWCU equipment vaults but these are normally inaccessible areas. In addition, the HVAC system is designed to provide a high degree of assurance that there will be no exfiltration of activity from these rooms to the normally accessible area of containment. The halogen and noble gas concentrations in the steam are taken from Section 11.1. The reactor building free volume is assumed to be 1.168×10^6 ft^3. Table 12.2-11 lists other parameters used in the analysis.

Table 12.2-12 presents the calculated airborne concentrations in the reactor building during normal power operation, assuming a continuous low purge flow rate of 5,000 cfm.

During refueling it is anticipated that the only major contribution to airborne activity in the reactor building will be from radioiodides. The fuel pool cooling and cleanup system is designed to clean and purify the water in the spent fuel pool and the upper fuel pools in the containment. The iodine activity in the pools will be reduced by passing the water through a 1,000 gpm filter/demineralizer. The resultant airborne concentrations of iodine in the reactor building are expected to be less than 2 percent of the equilibrium values during normal operation. For the purposes of calculating operating exposures in Section 12.4, a value of 2 percent of the normal operation thyroid dose rate is assumed.

Another source of potential airborne contamination in the reactor building is the activity release through relief valve discharge to the suppression pool. These are classified as Type 1 and Type 2 events. Type 1 events are of minor consequences because of the relatively short duration of the blowdown ($\leq$15 seconds). Type 2 events are of more concern because they involve isolation and depressurization of the system. The expected frequency of the Type 2 events is once per year. Table 12.2-13 presents a summary of the activity in the reactor building with time after the event; Section 12.4 presents the anticipated operator exposures per event.

12.2.2.2 Radwaste Building

Leakage to the radwaste building is assumed to be 2,000 gallons per day at 10 percent of the primary coolant iodine activity. The airborne noble gas activity in the radwaste building is negligible. A partition factor of .001 is assumed for iodine. The radwaste building free volume is 1.1 x 10^6 ft^3 and the purge rate is 30,000 cfm.

Table 12.2-14 presents the calculated airborne concentrations in the radwaste building.

12.2.2.3 Turbine Building

Leakage to the turbine building atmosphere is assumed to be 1,700 lb/hr of steam at main steam activity. A partition factor of 1 is assumed for both noble gases and halogens. The turbine building free volume is assumed to be 3.2×10^6 ft^3 and the purge rate is 1.8×10^5 cfm.

Table 12.2-15 presents the calculated airborne concentrations in the turbine building.

12.2.2.4 Fuel Handling Area of the Intermediate Building

Leakage to the fuel handling area atmosphere is based on evaporation from the spent fuel pool. The evaporation rate is calculated to be 320 lb/hr assuming the buiding is at 90°F and 50 percent relative humidity, and the pool is at 120°F. The equilibrium I-131 concentration in the pool was conservatively taken at 1×10^{-6} µCi/cc based on information given in Reference 2. The building free volume is assumed to be 1.5×10^6 ft^3 and the purge rate is 30,000 cfm.

Table 12.2-16 presents the calculated airborne concentration for I-131 and proportional values for I-133 and I-135.

12.2.2.5 Other Buildings

Other plant buildings are expected to have negligible noble gas and iodine airborne activity concentrations.

12.2.3 REFERENCES FOR SECTION 12.2

1. Smith, J.M., "Noble Gas Experience in Boiling Water Reactors," Paper No. A-54, presented at Noble Gases Symposium, Las Vegas, Nevada, September 24, 1974.

2. Johnson, A.B., "Behavior of Spent Nuclear Fuel in Water Pool Storage," BNWL-2256, Batelle, Pacific Northwest Laboratories, September, 1977.

TABLE 12.2-1

BASIC REACTOR DATA

A.	Reactor thermal power, MW	3,579
B.	Average power density, watts/cm^3	54.07
C.	Physical dimensions[(1)]	

		Radii (in)
1.	Core equivalent	92.58
2	Inside shroud	99.90
3.	Outside shroud	101.90
4.	Inside vessel (nominal)	119.0
5.	Outside vessel (nominal)	125.0
6.	Outside vessel (reinforced - nominal)	125.75
7.	Shroud head inside	192.0
8.	Vessel top head inside	119.0
9.	Vessel bottom head inside	130.19

		Distance (in)
10.	Outside of vessel bottom head	-7.75
11.	Inside of vessel bottom head	-0.25
12.	Vessel bottom head tangent	129.94
13.	Bottom of core support plate	202.56
14.	Top of core support plate	204.56
15.	Bottom of active fuel	213.50
16.	Top of reinforced vessel wall	210.00
17.	Top of active fuel	363.5
18.	Bottom of top guide	371.31
19.	Top of fuel channel	377.87
20.	Shroud head tangent	424.23
21.	Inside of shroud head	452.27

	Distance (in)
22. Outside of shroud head	454.27
23. Normal vessel water level	566.6
24. Top of steam dryer	720.63
25. Vessel top head tangent	727.0
26. Inside of vessel top head	846.0
27. Outside of vessel top head	849.0

D. Material Densities, gm/cm^3

Region(2)	Coolant	UO_2	Zircaloy	340 Stainless
A	0.740	0.0	0.0	0.178
B	0.338	0.0	0.0	4.349
C	0.318	2.3	0.978	0.056
C-1	0.597	0.0	0.166	1.697
C-2	0.234	0.0	1.099	0.255
D	0.240	0.0	1.004	1.209
E	0.390	0.0	0.0	0.0
F	0.669	0.0	0.0	0.200
G	0.036	0.0	0.0	0.0
H	0.740	0.0	0.0	0.0
I	0.740	0.0	0.0	0.26

E. Typical Core Power Distributions

Radial Power Distribution		Axial Power Distribution (Typical end-of-life)	
Percent of Equivalent Radius	Relative Power	Distance(3) (in)	Relative Power
0	1.2	-75	0.343
20	1.2	-68	0.755
35	1.19	-60	1.055
50	1.17	-48	1.190

TABLE 12.2-1 (Cont'd)

E. Typical Core Power Distributions (Cont'd)

Radial Power Distribution		Axial Power Distribution (Typical end-of-life)	
Percent of Equivalent Radius	Relative Power	Distance[3] (in)	Relative Power
60	1.15	-36	1.200
70	1.12	-24	1.190
80	1.05	-12	1.170
85	0.995	0	1.155
90	0.778	12	1.140
92.5	0.590	24	1.105
95.0	0.430	36	1.055
97.0	0.375	48	0.945
98.0	0.395	60	0.715
99.0	0.432	68	0.462
100.0	0.518	75	0.212

NOTES:

1. Relative locations of dimensions are shown in Figure 12.2-1.

2. Region locations are shown in Figure 12.2-1.

3. Distances are measured from the mid-plane of the core.

TABLE 12.2-2

CORE BOUNDARY NEUTRON FLUXES

Energy Bounds		Neutron (neutrons/cm^2 - sec)
16.5	Mev	3.9 E+10
10.00	Mev	5.5 E+11
6.065	Mev	2.0 E+12
3.679	Mev	3.8 E+12
2.231	Mev	4.4 E+12
1.353	Mev	3.9 E+12
820.8	Kev	3.8 E+12
497.9	Kev	2.6 E+12
302.0	Kev	2.3 E+12
183.2	Kev	3.2 E+12
67.38	Kev	2.2 E+12
24.79	Kev	2.2 E+12
9.119	Kev	2.0 E+12
3.355	Kev	2.0 E+12
1.234	Kev	1.9 E+12
454.0	ev	2.0 E+12
167.0	ev	1.9 E+12
61.44	ev	1.8 E+12
22.60	ev	8.8 E+12
13.71	ev	8.8 E+12
8.315	ev	8.2 E+12
5.043	ev	8.4 E+11
3.059	ev	8.3 E+11
1.855	ev	8.2 E+11
1.125	ev	8.8 E+11
0.616	ev	3.2 E+13
0.00	ev	

TABLE 12.2-3

GAMMA RAY SOURCE ENERGY SPECTRA

A. Gamma ray sources in the core during operation

Energy Bounds (Mev)	Gamma Ray Source (Mev/Sec-watt)
16.5	7.8 E+8
8.0	7.3 E+9
6.0	5.9 E+10
4.0	5.8 E+10
3.0	5.2 E+10
2.6	6.7 E+10
2.2	7.2 E+10
1.8	8.3 E+10
1.4	9.1 E+10
1.0	7.5 E+10
0.75	6.8 E+10
0.5	6.1 E+10
0.25	9.8 E+10
0.003	

B. Post operation gamma sources in core(1) (Mev/sec - watt)

Energy	Time After Shutdown			
Bounds (Mev)	0 Sec.	1 Day	1 Week	1 Month
6.0	8.2 E+9	<1.0 E+6	<1.0 E+7	<1.0 E+6
4.0	1.8 E+10	7.0 E+6	7.0 E+6	<1.0 E+6
3.0	1.1 E+10	5.7 E+6	3.7 E+6	<1.0 E+6
2.6	1.7 E+10	5.7 E+6	2.7 E+6	<1.0 E+6
2.2	2.1 E+10	4.5 E+8	4.0 E+7	5.2 E+6
1.8	3.3 E+10	3.1 E+9	2.1 E+9	6.4 E+8
1.4	3.7 E+10	2.3 E+9	1.6 E+9	1.1 E+9
0.9	5.1 E+10	7.5 E+9	3.8 E+9	2.1 E+9
0.4	1.2 E+10	1.8 E+9	8.7 E+8	3.6 E+8
0.1				

TABLE 12.2-3 (Continued)

C. Gamma ray sources in non-core regions during operation (Mev/cm^2-sec-watt)

Energy Bounds(Mev)	Region H Inside	Region H Outside	Shroud Inside	Shroud Outside
16.5	3.0 E-1	5.1 E-2	3.9 E+2	6.0 E+1
8.0	2.7	4.2 E-1	1.2 E+3	1.9 E+2
6.0	5.1 E-3	1.0 E-3	3.6 E+2	5.6 E+1
4.0	2.5 E-2	5.6 E-3	1.6 E+2	2.7 E+1
3.0	1.1 E-3	1.8 E-4	6.5 E+1	1.1 E+1
2.6	2.6 E+2	6.5 E+1	3.6 E+1	6.5
2.2	6.1 E-3	1.4 E-3	4.1 E+1	7.7
1.8	6.7 E-5	1.1 E-5	1.1 E+1	1.8
1.4	2.9 E-3	6.8 E-4	5.1 E+1	9.3
1.0	8.4 E-3	1.9 E-3	6.0 E+1	1.4 E+1
0.75	5.0 E-4	8.0 E-5	4.5 E+1	6.8
0.5	-	-	1.8 E+2	1.8 E+1
0.25	-	-	1.4 E+2	2.1 E+1
0.003				

Energy Bounds (Mev)	Region I (Jet Pumps) Inside	Region I (Jet Pumps) Outside	Vessel Inside	Vessel Outside
16.5	1.4	2.2 E-2	2.2 E-1	2.1 E-4
8.0	4.6	6.9 E-2	2.1	1.6 E-3
6.0	1.3	2.0 E-2	5.6 E-1	1.6 E-3
4.0	6.3 E-1	9.3 E-3	2.8 E-1	2.0 E-3
3.0	2.6 E-1	3.7 E-3	1.0 E-1	1.2 E-3
2.6	6.1	4.7 E-2	4.7 E-2	1.1 E-3
2.2	1.8 E-1	2.4 E-3	5.3 E-2	1.4 E-3
1.8	4.2 E-1	6.0 E-3	1.8 E-1	1.1 E-3
1.4	2.1 E-1	2.9 E-3	7.5 E-2	9.3 E-4
1.0	3.2 E-1	3.6 E-3	9.1 E-2	5.5 E-3
0.75	1.6 E-1	2.4 E-3	6.4 E-2	4.2 E-5
0.50	6.6 E-1	9.9 E-3	2.6 E-1	2.0 E-4
0.25	5.1 E-1	7.6 E-3	1.9 E-2	1.8 E-4
0.003				

NOTE:

1. Operating history of 3.2 years.

TABLE 12.2-4

GAMMA RAY AND NEUTRON FLUXES OUTSIDE THE VESSEL WALL

A. Neutron fluxes

Energy Bounds		Neutrons/cm^2-sec.
16.5	Mev	5.8 E+6
10.00	Mev	2.9 E+7
6.065	Mev	2.2 E+7
3.679	Mev	4.5 E+7
2.231	Mev	7.5 E+7
1.353	Mev	1.1 E+8
820.8	Kev	1.6 E+8
497.9	Kev	1.5 E+8
302.0	Kev	9.1 E+7
183.2	Kev	1.1 E+8
67.38	Kev	1.2 E+7
24.79	Kev	6.7 E+7
9.119	Kev	1.4 E+7
3.355	Kev	8.6 E+6
1.234	Kev	6.4 E+6
454.0	ev	2.9 E+6
167.0	ev	4.2 E+6
61.44	ev	3.9 E+6
22.60	ev	1.9 E+6
13.71	ev	2.0 E+6
8.315	ev	1.8 E+6
5.043	ev	1.6 E+6
3.059	ev	1.5 E+6
1.855	ev	1.4 E+6
1.125	ev	7.9 E+5
0.616	ev	6.0 E+5
0.000	ev	

TABLE 12.2-4 (Continued)

B. Gamma ray energy fluxes

Energy Bounds (Mev)	Mev/cm^2-sec^2
16.5	1.0 E+9
8.0	3.4 E+9
6.0	3.3 E+9
4.0	1.7 E+9
3.0	7.0 E+8
2.6	1.0 E+8
2.2	6.9 E+8
1.8	6.1 E+8
1.4	5.3 E+8
1.0	3.2 E+8
0.75	4.2 E+8
0.50	4.0 E+8
0.25	1.5 E+7
0.003	

TABLE 12.2-5

RADIATION SHIELDING SOURCE TERMS

Equipment Identification		Source Volume (cc)	Sources Used for Shielding Calculations (γ/cc-sec) 0.2 Mev	0.6 Mev	1.0 Mev	1.6 Mev	2.4 Mev	3.4 Mev	3.9 Mev	4.7 Mev	6.1 Mev	7.1 Mev
A. AUXILIARY BUILDING												
E12B001	RHR hx shutdown mode	8.45+6	8.01+4	4.60+4	5.91+4	4.55+4	5.70+3	2.44+2				
B12B001	RHR hx stm. condensing mode	8.45+6	8.01+4								6.62+5	4.81+4
B12C002	RHR pump	1.36+5	8.01+4	4.60+4	5.91+4	4.55+4	5.70+3	2.44+2				
B12C001	LPCS pump	1.36+5	8.30+4	6.30+4	4.42+4	4.37+4	2.11+4	7.68+2				
B12C001	HPCS pump	1.80+5	8.30+4	6.30+4	4.42+4	4.37+4	2.11+4	7.68+2				
B51C001	RCIC pump	2.84+5	8.30+4	6.30+4	4.42+4	4.37+4	2.11+4	7.68+2				
G33C001	RWCU pump	1.11+4	7.76+4	3.55+4	4.42+4	3.83+4	2.11+4	7.68+2			1.28+6	9.27+4
B. REACTOR BUILDING												
G33B001	RWCU hx (regen.)	2.47+5	7.76+4	3.55+4	4.42+4	3.83+4	2.11+4	7.68+2			1.28+6	9.27+4
G33B002	RWCU hx (non-regen.)	2.47+5	7.76+4	3.55+4	4.42+4	3.83+4	2.11+4	7.68+2			7.41+5	5.34+4
G36A003	RWCU F/D bkwsh. rec. tk	1.18+7	1.07+7	2.84+6	1.31+6	1.02+6	4.45+4	1.82+1				
G36C001	RWCU F/D holding pump	1.11+4	7.76+4	3.55+4	4.42+4	3.83+4	2.11+4	7.68+2				
G36D001	RWCU F/D	1.80+6	3.53+7	9.39+6	4.32+6	3.38+6	1.47+5	6.00+1			6.50+4	4.70+3
G50C012	RWCU bkwsh. trans. pump	1.25+4	1.07+7	2.84+6	1.31+6	1.02+6	4.45+4	1.82+1				
C. INTERMEDIATE BUILDING												
G41A002	Fuel pool surge tk.	2.28+7	1.25+4	4.18+3	1.73+1	3.04+0	3.04+0	1.84-1				
G41A003	Fuel trans. tube drn. tk.	3.24+6	1.25+4	4.18+3	1.73+1	3.04+0	3.04+0	1.84-1				
G41B001	Fuel pool HX	1.90+6	1.25+4	4.18+3	1.73+1	3.04+0	3.04+0	1.84-1				
G41C001	Fuel pool F/D holding pump	6.95+3	1.25+4	4.18+3	1.73+1	3.04+0	3.04+0	1.84-1				
G41C003	Fuel pool circ. pump	7.72+4	1.25+4	4.18+3	1.73+1	3.04+0	3.04+0	1.84-1				
G41C004	Cask pool drn. pump	1.31+4	1.25+4	4.18+3	1.73+1	3.04+0	3.04+0	1.84-1				
G41C005	Fuel trans. tube drn. pump	6.95+3	1.25+4	4.18+3	1.73+1	3.04+0	3.04+0	1.84-1				
G41D001	Fuel pool F/D	2.14+6	1.41+5	4.70+4	1.71+5	3.20+4	3.02+4	1.99+3				
G50A022	Fuel pool F/D bkwsh. rec. tk.	3.79+7	1.41+5	4.70+4	1.71+5	3.20+4	3.20+4	1.99+3				
G50C027	Fuel pool F/D bkwsh. trans. pump	2.84+4	1.41+5	4.70+4	1.71+5	3.20+4	3.20+4	1.99+3				
D. RADWASTE BUILDING												
G50A001	Liquid waste coll. tk.	1.33+8	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50A002	Liquid waste sample tk.	1.33+8	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50A003	Floor drns. coll. tk.	1.33+8	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50A004	Floor drns. sample tk.	1.33+8	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50A005	Chemical waste tk.	8.76+7	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50A006	Concentrated waste tk.	1.89+7	1.17+7	4.82+6	4.04+5	8.35+5	2.20+4	1.65+2				
G50A007	Chem. waste dist. tk.	7.50+7	3.62+0	2.90+0	1.36+0	2.02+0	9.80-2	3.35-3				

TABLE 12.2-5 (Continued)

Equipment Identification		Source Volume (cc)	Sources Used for Shielding Calculations (γ/cc-sec) 0.2 Mev	0.6 Mev	1.0 Mev	1.6 Mev	2.4 Mev	3.4 Mev	3.9 Mev	4.7 Mev	6.1 Mev	7.1 Mev
D. RADWASTE BUILDING (Cont'd)												
G50A009	Spent resin tk.	3.78+7	1.03+6	4.36+6	1.75+5	1.21+6	1.40+4	9.48+1				
G50A011	Cnds. F/D settling tk.	7.95+7	3.77+5	1.45+5	4.38+4	4.35+4	6.16+2	3.28+1				
G50A013	RWCU settling tk.	1.70+7	2.87+6	1.65+6	7.54+5	6.87+5	2.09+4	2.72+0				
G50A014	Waste sludge settling tk.	7.95+7	1.03+4	1.06+5	3.93+5	6.28+4	8.22+1					
G50A024	Waste coll. filtrate tk.	1.89+6	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50A025	Floor drains, filtrate tk.	1.89+6	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50Z001	Waste evap. condenser	1.10+6	3.22+6	1.85+6	2.40+6	1.84+6	2.36+5	9.80+3				
G50C001	Waste collector trans. pump	1.25+4	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50C002	Waste sample pump	7.27+3	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50C003	Floor drns. coll. pump	1.25+4	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50C004	Floor drns. sample pump	7.27+3	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50C005	Chemical waste pump	6.95+5	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50C006	Chemical waste dist. pump	1.25+4	3.65+0	2.90+0	1.36+0	2.02+0	9.80-2	3.25-3				
G50C008	Spent resin pump	1.25+4	1.03+6	4.36+6	1.75+5	1.21+6	1.40+4	9.48+1				
G50C010	Cond. sludge disch. mix pump	1.25+4	3.77+5	1.45+5	4.38+4	4.35+4	6.16+2	3.28+1				
G50C011	Cond. sludge decant. pump	1.25+4	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50C012	RWCU sludge disc # mix pump	1.25+4	2.87+6	1.65+6	7.54+5	6.87+5	2.09+4	2.72+0				
G50C014	RWCU sludge decant pump	2.90+3	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50C015	Waste sludge disc. mix pump	1.25+4	1.03+4	1.06+5	3.93+5	6.28+4	8.22+1					
G50C016	Waste sludge decant pump	1.25+4	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50C017	Waste coll. filtrate pump	4.70+3	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
C50C018	Floor drns. filtrate pump	4.70+3	3.31+3	5.25+4	1.16+3	1.42+3	1.21+3	1.66+2				
G50C026	Conc. waste trans. pump	1.25+4	1.17+7	4.82+6	4.04+5	8.35+5	2.20+4	1.65+2				
G50D001	Waste collector filter	7.08+5	4.14+6	1.44+6	2.26+6	3.32+5	3.08+4					
G50D002	Floor drains filter	7.08+5	4.14+6	1.44+6	2.26+6	3.32+5	3.08+4					
G50D003	Waste demin.	2.34+6	1.17+7	5.10+7	2.99+6	1.69+7	2.00+5	1.42+3				
G50D004	Floor drns. demin.	2.34+6	1.17+7	5.10+7	2.99+6	1.69+7	2.00+5	1.42+3				
E. TURBINE POWER COMPLEX												
N23D001	Condensate filter	1.05+7	6.39+4	6.73+4	2.56+4	2.16+4	1.68+2	2.74+1				
N24A001	Conds. demin. cation regen. tk.	1.61+7	5.87+5	6.21+6	2.32+5	2.00+5	1.26+5	2.53+2				
N24A002	Conds. demin. anion regen. tk.	6.61+6	5.87+5	6.21+6	2.32+5	2.00+5	1.26+5	2.53+2				
N24A0	Conds. demin. mix & hold tk.	9.81+6	5.87+5	6.21+6	2.32+5	2.00+5	1.26+5	2.53+2				
N24A004	Conds. demin. bkwsh. rec. tk.	1.87+7	5.87+5	6.21+6	2.32+5	2.00+5	1.26+5	2.53+2				

TABLE 12.2-5 (Continued)

Equipment Identification		Source Volume (cc)	0.2 Mev	0.6 Mev	1.0 Mev	1.6 Mev	2.4 Mev	3.4 Mev	3.9 Mev	4.7 Mev	6.1 Mev	7.1 Mev
			Sources Used for Shielding Calculations (γ/cc-sec)									
E. TURBINE POWER COMPLFX (Cont'd)												
N24A005	Conds. demin. regen. chem. rec. tk.	1.87+7	5.87+5	6.21+6	2.32+5	2.00+5	1.26+5	2.53+2				
N24C001	Waste transfer pump	7.27+3	5.87+5	6.21+6	2.32+5	2.00+5	1.26+5	2.53+2				
N24D001	Condensate demin.	2.23+7	5.87+5	6.21+6	2.32+5	2.00+5	1.26+5	2.53+2				
G50A010	Cnds. F/D bkwsh. rec. tk.	1.87+7	5.05+5	5.33+5	2.00+5	1.71+5	1.08+5	2.17+2				
G50C009	Cond. bkwsh. trans. pump	2.84+4	5.05+5	5.33+5	2.00+5	1.71+5	1.08+5	2.17+2				
F. TURBINE BUILDING												
N64B001	Offgas preheater	1.80+6							5.54+4		1.27+5	
N64B002	Offgas condenser	5.00+6							5.81+6	7.20+2	1.34+7	
N64D005	Offgas catalytic recombiner	8.00+5							3.62+5	1.06+1	8.25+5	
G. OFF GAS BUILDING												
N64B010	Offgas cooler condenser	7.10+5	1.90+5	1.39+5	4.10+4	9.71+4	7.99+4	4.84+3		4.71+2		
N64B011	Offgas cooler	4.56+5	1.90+5	1.39+5	4.10+4	9.71+4	7.99+4	4.84+3		4.71+2		
N64D011	Offgas prefilter	3.10+5	1.76+6	4.00+6	9.45+6	8.12+6	3.43+6	5.12+3		2.95+4		
N64D012	Offgas charcoal absorber	6.80+6	5.14+7	1.94+6	1.31+6	1.68+6	2.27+6	4.34+3		1.17-3		
N64D016	Offgas after filter	3.10+5	4.75+2	1.24+1	9.19-1	8.94-1	1.41+0	-				
N64D030	Offgas dessicant dryer	6.00+5	3.32+6	2.74+6	1.30+6	2.36+6	1.78+4	6.00+4		4.17-1		
N64Z003	Gas dryer	7.10+5	1.90+5	1.39+5	4.10+4	9.71+7	7.99+4	4.84+3		4.71+2		

NOTE:

1. Source model geometry for all calculations is cylindrical in shape.

TABLE 12.2-6

MATERIAL COMPOSITION OF THE TIP SYSTEM COMPONENTS USED IN ACTIVATION CALCULATIONS

A. Detector Region

Material	Quantity
AISI 304 stainless steel	4g
Commercially pure titanium	3g
Fosterite ceramic	0.5g
Nichrome	0.02g
Uranium - 235	0.001g

B. Cable Region

AISI 3041 stainless steel	0.12g/inch
AISI C1070 carbon steel	2.1g/inch
Magnesium oxide	0.12g/inch

TABLE 12.2-7

TRAVERSING INCORE PROBE DETECTOR

DECAY GAMMA ACTIVITIES

Decay Time = 0 Seconds

Activation Time = 10^2 Seconds

Energy (Mev)	Activities[1] (Mev/sec)
0.1 - 0.4	3.4 + 9
0.4 - 0.9	1.5 + 10
0.9 - 1.35	1.2 + 10
1.35 - 1.8	1.1 + 10
1.8 - 2.2	8.0 + 9
2.2 - 2.6	6.4 + 9
2.6 - 3.0	5.6 + 9
3.0 - 3.5	5.1 + 9
3.5 - 4.0	4.3 + 9
4.0 - 4.5	2.5 + 9
4.5 - 5.0	1.5 + 9
5.0 - 5.5	7.9 + 8

NOTE:

1. .001g of U-235.

TABLE 12.2-8

TRAVERSING INCORE PROBE DETECTOR

DECAY GAMMA ACTIVITIES OF MATERIALS IN THE DETECTOR[1]

Decay Time = 0 Seconds

Activation Time = 10^2 Seconds

Activated (Isotope)	Activity (μCi)
Fe-59	1.1 + 1
Mn-56	1.7 + 5
Cr-51	7.0 + 1
Mn-54	2.1 + 0
Co-58M	3.5 + 3
Co-58	2.2 - 2
Ni-57	1.1 - 1
Co-57	6.0 - 7
Ni-65	4.0 + 2
Co-60M	7.6 + 3
Co-60	1.8 - 3
Co-61	9.6 + 0
Si-31	2.9 + 1

NOTE:

1. Excluding U-235.

TABLE 12.2-9

DECAY GAMMA ACTIVITIES OF MATERIALS IN THE CABLE

Decay Time = 0 Seconds

Activation Time = 10^2 Seconds

Activated Isotope	Activity (μCi/in)
Fe-59	8.2 + 0
Mn-56	7.4 + 4
Cr-51	3.7 + 0
Mn-54	1.6 + 0
Co-58M	1.0 + 2
Co-58	6.5 - 4
Ni-57	3.3 - 3
Co-57	1.8 - 8
Ni-65	1.2 + 1
Co-60M	2.2 + 2
Co-60	5.1 - 5
Co-61	2.8 - 1
Si-31	8.7 - 1

TABLE 12.2-10

TYPICAL TURBINE COMPONENT N-16 INVENTORIES

	System/Components	Inventory (Curies)
1.	Main steam line and header system	263
2.	High pressure turbine	6.4
3.	Low pressure turbines (6 flow machine)	9.8
4.	Moisture separator shell-side steam	53
5.	Moisture separator shell-side liquid	41
6.	Moisture separator drain system	56
7.	First stage reheat system(1)	33
8.	Second stage reheat system(1)	32
9.	First stage reheat drain system(2)	1.4
10.	Second stage reheat drain system(2)	1.1
11.	Crossover pipe system	59
12.	Crossaround pipe system	17
13.	Feedwater heater and extraction system	
	First Stage(3)	26
	Second Stage(3)	23
	Third Stage(3)	27
	Fourth Stage(3)	15
	Fifth Stage(4)	0.6
	Sixth Stage(5)	42
	Condenser(6)	286
14.	Hotwell(7)	18
15.	SJAE first stage system(8)	0.6
16.	Recombiner system	0.4
17.	Separate steam system(9)	0.9
18.	Feedwater turbine system(9)	8.8
	Total	1021

TABLE 12.2-10 (Continued)

NOTES:

1. Includes inventory in liquid and steam in reheat tubes and in steam supply line.

2. Includes total inventory beyond reheater outlet.

3. Includes total inventory beyond extraction point. Distribution of this will depend on equipment arrangement and sizing.

4. Excludes moisture separator drain system activity listed in item 6.

5. Excludes first and second stage reheat drain system activities listed in items 7, 8.

6. Excludes residual activity returned from feedwater turbine.

7. Excludes residual activity returned from feedwater turbine.

8. Includes inventory in steam supply system.

9. Includes total inventory beyond inlet at steam supply line.

TABLE 12.2-11

PARAMETERS AND ASSUMPTIONS USED IN CALCULATING REACTOR BUILDING AIRBORNE ACTIVITY

Initial iodine partition factor in suppression pool	.00086
Iodine halving time in suppression pool	1,000 hours
Initial noble gas partition factor in suppression pool	0.5
Noble gas halving time in suppression pool	96 hours
Suppression pool cleanup demineralizer decontamination factor foı iodine	100

TABLE 12.2-12

REACTOR BUILDING AIRBORNE ACTIVITY

Nuclide	Concentration (μCi/cc)
Kr-83m	4.1-8
Kr-85m	1.2-7
Kr-85	1.1-9
Kr-87	1.9-7
Kr-88	3.1-7
Kr-89	7.1-8
Xe-131m	1.4-9
Xe-133m	1.0-8
Xe-133	3.5-7
Xe-135m	6.5-8
Xe-135	5.5-7
Xe-137	9.8-8
Xe-138	2.0-7
I-131	3.7-10
I-132	6.1-10
I-133	1.9-9
I-134	3.7-10
I-135	1.5-9

TABLE 12.2-13

SUMMARY OF ACTIVITY IN CONTAINMENT BUILDING WITH TIME AFTER ISOLATION SCRAM

Nuclide	Activity (Curies) Time After Isolation Scram(1) 4 Min.	4 Hr.	8 Hr.	16 Hr.
Kr-83m	6.9 + 0	5.9 + 1	1.4 + 1	7.3 - 1
Kr-85m	1.5 + 1	1.4 + 2	7.7 + 1	2.3 + 1
Kr-85	2.7 - 2	5.3 - 1	5.4 - 1	5.7 - 1
Kr-87	2.5 + 1	5.0 + 1	5.7 + 0	7.5 - 2
Kr-88	4.2 + 1	2.7 + 2	1.0 + 2	1.5 + 1
Rb-88	2.3 + 0	2.8 + 2	1.1 + 2	1.6 + 1
Kr-89	5.0 + 0	-	-	-
Rb-89	4.2 - 1	-	-	-
Kr-90	1.7 - 2	-	-	-
Rb-90	3.2 - 2	-	-	-
I-131	4.8 - 5	7.0 - 2	7.3 - 2	7.1 - 2
Xe-131m	2.1 - 1	3.7 + 0	3.7 + 0	3.8 + 0
I-132	8.8 - 5	3.7 - 1	1.9 - 1	1.1 - 1
I-133	8.7 - 5	7.2 - 2	6.5 - 2	5.0 - 2
Xe-133m	1.8 + 0	3.1 + 1	3.0 + 1	2.8 + 1
Xe-133	8.1 + 1	1.4 + 3	1.4 + 3	1.4 + 3
I-134	4.5 - 4	4.1 - 2	1.8 - 3	3.3 - 6
I-135	1.5 - 4	5.0 - 2	3.4 - 2	1.5 - 2
Xe-135m	1.3 + 1	3.2 + 1	2.3 - 2	9.7 - 3
Xe-135	3.2 + 1	6.4 + 2	4.8 + 2	2.8 + 2
Xe-137	9.0 + 0	-	-	-
Xe-138	3.3 + 1	5.4 - 3	-	-

TABLE 12.2-13 (Continued)

Nuclide	Activity (Curies) Time After Isolation Scram[1] 4 Min.	4 Hr.	8 Hr.	16 Hr.
Cs-138	1.1 + 0	7.5 - 1	4.3 - 3	-
Xe-139	5.8 - 2	-	-	-
Total	2.7 + 2	2.9 + 3	2.2 + 3	1.8 + 3

NOTE:

1. The pressure relief transient is described as follows:

Time	Event
0 sec.	Turbine trip with 10 percent bypass following extended operation at fuel failure corresponding to 100,000 μCi/sec at 30 minutes reference time
	Relief of reactor pressure to main condenser by 10 percent bypass and to the suppression pool by relief valves.
45 sec.	Main steam isolation valves close.
	Relief of reactor pre-sure to suppression pool by relief valves only.
	Reactor in hot standby mode.
1 hr.	Begin reactor cooldown and depressurization by relief valve release to suppression pool.
4 hr.	Reactor depressurization complete.

TABLE 12.2-14

RADWASTE BUILDING AIRBORNE ACTIVITY

Nuclide	Concentration (μCi/cc)
I-131	1.4-11
I-132	1.5-10
I-133	1.0-10
I-134	2.2-10
I-135	1.5-10

TABLE 12.2-15

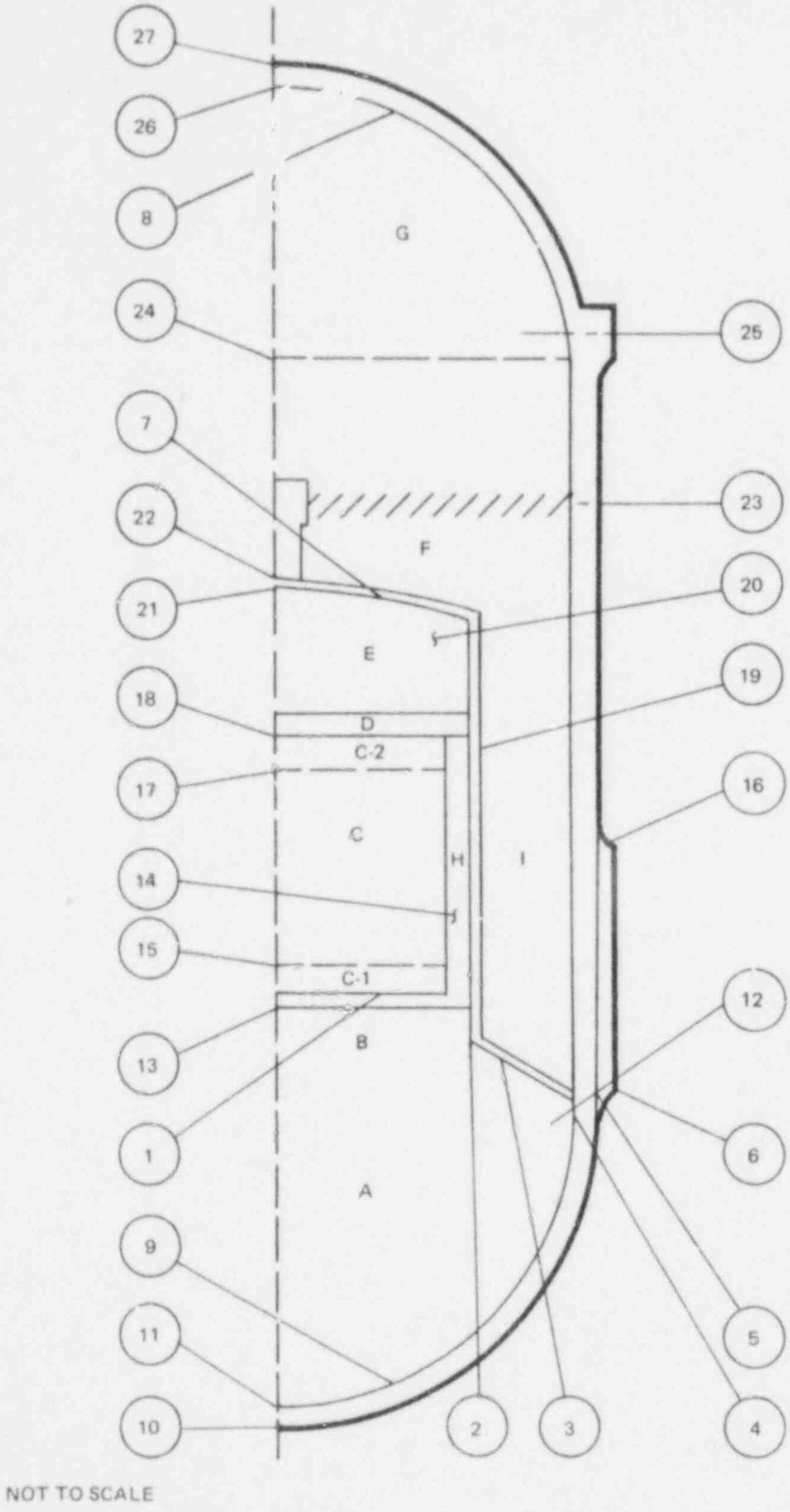
TURBINE BUILDING AIRBORNE ACTIVITY

Nuclide	Concentration (μCi/cc)
Kr-83m	3.9-9
Kr-85m	7.6-9
Kr-85	2.8-11
Kr-87	2.2-8
Kr-88	2.4-8
Kr-89	3.4-8
Xe-131m	1.9-13
Xe-133m	3.5-10
Xe-133	1.1-8
Xe-135m	1.9-8
Xe-135	2.8-8
Xe-137	4.7-8
Xe-138	6.7-8
I-131	1.0-11
I-132	1.2-10
I-133	7.5-11
I-134	2.0-11
I-135	1.2-10

TABLE 12.2-16

FUEL HANDLING AREA AIRBORNE ACTIVITY

Nuclide	Concentration (μCi/cc)
I-131	3.2-12
I-133	1.4-11
I-135	1.1-11



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Source Model

Figure 12.2-1

Note: Nodes are as shown on Table 12.2-1.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 FACILITY DESIGN FEATURES

The PNPP has been designed to attain as low as is reasonably achievable radiation doses to plant personnel as well as personnel located around the facility. The guidance of Regulatory Guide 8.8 has been used in designing the facility to result in radiation doses that are only a small fraction of the limits given in 10 CFR 20.

12.3.1.1 Equipment and Facility Design Features

The plant is separated into five controlled access zones to aid in the design of radiation protection features and plant operation. An access zone designation is assigned to each area of the plant for each of two operating conditions: normal operation and shutdown/refueling. Table 12.3-1 defines each of the five zone designations. Figure 1.2-1 lists the Perry equipment names and numbers. Figures 12.3-1 through 12.3-11 show the plant layout including:

a. Locations of the sources described in Section 12.2 and Chapter 11.

b. Shield wall thicknesses.

c. Radiation zones for normal operation and refueling.

d. Personnel and equipment decontamination areas.

e. Contamination control areas and the locations of health physics facilities.

f. Control panels for radioactive waste treatment equipment.

g. Onsite laboratories for analysis of chemical and radioactive samples.

h. Counting room.

Figures 12.3-1 through 12.3-11 also illustrate traffic patterns during normal operation and locations of airborne radioactivity and area monitors.

The counting room is located so that the background radiation levels will be low enough to allow for continuous occupancy and to provide an accurate analytical environment under normal operating conditions and anticipated operational occurrences. The counting room is sized to provide adequate space for the required instrumentation. See Section 12.5 for a discussion of instrumentation in the counting room.

Nonradioactive equipment that may require maintenance is located, when possible, in either Zone I or Zone II. Adjacent areas containing potentially radioactive systems are designed to maintain a radiation level less than the Zone IV maximum (100 mRem/hr) during required maintenance.

Equipment located in Zone IV or Zone V is designed to minimize required maintenance and to be operated remotely. Shield wall penetrations for remote operating devices, electrical equipment, pipes and ventilation ducts are designed and located at positions that prevent a direct line of sight to any significant source, thereby minimizing radiation streaming.

The following design consideration has been given to reduce radiocobalt production and crud buildup in normally radioactive systems:

a. System materials are specified for low corrosion and erosion rates and for low neutron activation source characteristics. Hardfacing materials which have high cobalt content, such as Stellite, are used only where substitute materials cannot satisfy performance requirements.

b. Packless valves are specified for systems which normally handle radioactive fluids. Where packed valves are specified, they are provided

with positive backseats, lantern ring leakoffs to the liquid waste management system, and special close tolerance graphoil packing in lieu of conventional packing.

c. System design considers decontamination of components. Isolation, vent and drain valves are provided in suitable locations to facilitate local decontamination of system components.

d. Piping systems are of all welded construction with minimum use of flanged and socket weld connections.

Design practices will allow, whenever practicable, the separation of radioactive piping from nonradioactive piping, electrical equipment and personnel passageways.

The following examples illustrate specific design features that aid in minimizing exposure levels:

a. Components containing radioactive materials will be separated, when practicable, to reduce radiation doses associated with maintenance.

b. Cubicles are sized to provide adequate space around components for anticipated maintenance operations and for ease of entry and exit.

c. The service mode of operation for the waste filters proceeds automatically after operator initiation from the radwaste building control room. Backwashing and precoating are done from a local control panel outside the filter cubicles.

d. Both regenerable and nonregenerable demineralizers have provisions for remote removal of radioactive contents (spent resins or regenerative solutions) to the waste management systems.

e. Activated carbon adsorber media is removed from the filter plenums by a portable vacuum removal system. Adsorber media can be removed without entering the filter plenum.

f. Particulate filters are bagged during removal and the bags are sealed before the filters are removed from the plenums. Filters are covered with plastic during the entire change.

g. Tanks containing potentially radioactive fluids are vented and can be drained to the waste management systems. Tanks containing fluids at atmospheric pressure are designed to withstand a pressure equivalent to a full tank of water. Static heads will be somewhat less due to overflow lines near the top of tanks.

h. Evaporators have provisions for removal of noncondensables to the waste management systems. Flush and rinse features permit decontamination before maintenance.

12.3.1.2 Illustrative Examples of Plant Design Features to Minimize Occupational Doses

Plant design features represent a comprehensive effort to achieve minimization of radiation exposure. These features include:

a. Radiation shielding of individual items of equipment.

b. Accumulation of associated items of non-safety related equipment within contiguous areas of plant structures.

c. Shielded chases for pipe runs between equipment cells and elsewhere about the plant.

d. Other structural design relative to minimizing radiation exposure to operating and maintenance personnel.

Individual shielding means that the person approaching the location of a radioactive component is shielded from direct and most scattered radiation from both the item of equipment he is approaching (until he enters the equipment area) and all other items of radioactive equipment in the path to, and in the near vicinity of, the equipment being approached.

A semi-automated solid radwaste packaging and handling system, the United Nuclear System, has been included in the PNPP design to minimize the radiation doses associated with this routine operation. This system is discussed thoroughly in Section 11.4. In general, the automated system consists of an overhead indexing crane which is remotely attached to, and detached from, various sized shipping containers. Personnel at the solid radwaste control panel are completely shielded from liners in the fill process and in the storage area. Individuals should not have to enter the waste storage area or the solidification area during its operation. All liners can be smear tested for contamination through portals with remote tooling. If a failure of the overhead crane occurs, a manual-remote override can be used to move the crane to a lower radiation area for maintenance. The transfer cart is moved to the crane pickup point where it can be moved to a low radiation area for maintenance. As mentioned in Section 12.4, the anticipated man-Rem dose from waste processing for PNPP is only a small fraction of that currently being experienced in the industry.

The Mark III containment design includes a water filled suppression pool that provides the following functions:

a. A heat sink for safety relief valve (SRV) discharges

b. A heat sink for hot standby operation

c. A means to condense steam released to the drywell during a LOCA

d. A continuing long term source of water for the emergency core cooling system

The surface of the suppression pool is open to the containment so that some fractions of radionuclides discharged to the suppression pool from safety relief valve operation and other sources can evolve into the containment atmosphere. Previous studies of the radiological consequences have concluded that the expected exposures of operation personnel are within the limits of 10 CFR 20; however, there is a need to achieve exposures as low as reasonably achievable (ALARA) during normal operation and following normally expected transients.

There are occasional plant upset events that result in steam release from the SRV's. One such event is a complete depressurization of the reactor to the suppression pool following a power isolation transient. During such a transient there may be sufficient increase in radioactivity within the containment to require egress of all personnel. In an effort to reduce these operator exposures at the PNPP, the following two changes were made during the design process:

a. Addition of a suppression pool cleanup system

This system uses an anion resin deep bed demineralizer (1,000 gpm). The main benefit of the system is to significantly reduce operator exposure to radioiodides which would evolve relatively slowly from the pool after the transient. In addition, the system will improve plant availability by allowing earlier operator re-entry without the use of respiratory equipment following a power isolation event.

b. Reroute RHR heat exchanger vent from suppression pool to drywell

When using the RHR heat exchangers for steam condensing during reactor depressurizations, noncondensable gases will collect in the exchanger shells. These gases must be vented to avoid interference with the condensation process. These gases consist mainly of radiolytic hydrogen and oxygen plus small quantities of radioactive noble gases. In addition, the vent gas will contain some water vapor and traces of iodine. The original design was to discharge vent gases in the

suppression pool where they can evolve into the containment. An alternative method was chosen to vent the gases to the portion of the suppression pool within the drywell. The drywell volume is sufficient to contain the vent gas volume without significant pressure rise and the drywell walls provide adequate shielding. The main benefit of the system will be to reduce containment re-entry skin and whole body dose rates following depressurizations.

A further change has been made as a result of the dose assessment for PNPP in the area of evaporator maintenance. Initially the evaporators and associated equipment were mounted on a single skid. To decrease radiation doses during maintenance, all non-radioactive components have been separated from radioactive components.

12.3.2 SHIELDING

12.3.2.1 Design Objectives

The design objectives of the plant radiation shielding are:

1. To ensure that during normal operation, including anticipated operational occurrences, the radiation dose to plant personnel and authorized site visitors is as low as reasonably achievable and within the limits set forth in 10 CFR 20.

2. To provide the necessary protection for plant operating personnel following a reactor accident to maintain habitability of the control room as specified in 10 CFR 50, Appendix A, Criterion 19.

3. To limit offsite exposures to the general public to meet the dose requirements of 10 CFR 100 for all postulated accident conditions and to maintain exposures as low as reasonably achievable, a small fraction of the 10 CFR 20 limits during normal operation.

4. To protect certain components from excessive radiation damage or activation.

12.3.2.2 Design Description

12.3.2.2.1 Plant Shielding Description

Detailed layout drawings showing all plant structures are shown in Figures 1.2-3 through 1.2-17. A general description of the major shielding in the buildings housing radioactive process equipment and fluids is outlined as follows:

a. Reactor building complex

   The reactor building complex shielding includes the biological shield wall, drywell shield walls and the shield building wall.

   The purpose of the biological shield is to minimize gamma heating in the drywell shield wall, to provide access to the drywell during shutdown and to reduce activation of drywell equipment and materials. The design dose rate used in sizing this shield is to maintain a radiation level in the drywell below 100 mRem/hr at full power operation. (Zone V).

   The drywell shield wall maintains the area outside the drywell at Zone II level except for some individual cubicles housing radioactive process equipment and piping, such as cubicles for the reactor water cleanup system and the chase for the main steamline pipes. The shielding for these is sized to maintain a Zone II level outside of each respective cubicle.

   The shield building completely surrounds the steel containment vessel and ensures that levels outside the building are less than 0.5 mRem/hr (Zone I) during normal plant operation. In addition, the building serves to attenuate radiation to plant personnel and the general public in the event of an accident.

b. Turbine room and heater bay

The major source of radiation in the turbine room and heater bay are the N-16 gammas. Shielding is provided around the radioactive equipment in the following systems to ensure that the dose levels are consistent with the access requirements:

1. Condensate and feedwater

2. Condensate filtration and demineralizer

3. High pressure heater drains and vents

4. Turbine

5. Steam seal

6. Condenser and auxiliaries

7. Off-gas

Areas within the shield enclosures will normally be restricted access.

c. Turbine skyshine

The O^{16} (n, p) N^{16} reaction is of interest in the boiling water reactor because of the coolant activation induced by the high energy end of the fission neutron spectrum. The N^{16} present in the steam of a direct cycle boiling water reactor is carried with the steam into the turbine, moisture separators and associated equipment of the secondary cycle. Although the N^{16} decays with a half life of 7.35 seconds, the gamma emission can present a radiation dose problem to the site boundry as a result of the high energy gamma scatter from structures and atmosphere. Relative to this, turbine building gamma ray air scattering (skyshine) analyses with the Morse[1] Monte-Carlo code and the G-33[2] code were

made to evaluate the site boundary dose contribution from the N^{16} radiation. The Morse Code results, as presented in Reference 3, were normalized in the G-33 code. The G-33 point kernel procedure which was then used to perform the site boundary dose rate calculations is based on application of the Klein Nishina scattering formula to the uncollided flux in a predetermined scattering grid. Normal scattering in air was approximated by use of water buildup factors on the scattered leg from each point in the scattered grid. This approach as well as previous work cited in literature has shown this method yields results in very close agreement with Monte-Carlo (for example, above cited Morse Code) air scattering calculations and with measurements.

The turbine component N-16 inventories are presented in Table 12.2-4 and the layout of the turbine building walls and floors is presented in the general arrangement drawings, Figure 1.2-6 through 1.2-12. The major shields are:

1. A three foot thick outer turbine building wall extending above the moisture separator-reheaters.

2. A steel shield plate at the ends of the high and low pressure turbines with a labyrinth entry into the turbine area.

3. A 6-inch steel shield plate inboard of the moisture separator reheaters with labyrinth entry into the moisture separator reheater area.

4. A one foot thick concrete floor slab extending between the turbine and the 6-inch steel shield plate.

For distances beyond 300 feet, a single lumped source was assumed in the turbine building; for distances less than 300 feet, all major sources were considered separately. A curve of the dose rate (mRem/hr) versus distance from the turbine building is presented in Figure 12.3-12.

d. Off-gas building, auxiliary building, radwaste building, and intermediate building.

The shielding provided in these buildings is designed to maintain the dose levels consistent with the plant zone designations given in Section 12.3.1. As far as practicable, separation of radioactive equipment is provided in order that maintenance and repairs may be accomplished without the necessity of shutdown and decontamination of equipment in adjacent cubicles.

The charcoal adsorber system, located in the off-gas building, is comprised of two parallel trains with four vessels in each train. The shielding for the system is accomplished by providing refrigerated vaults (refer to Figure 1.2-4). The first vessel in each train is isolated in a separate compartment and the remaining two compartments have three vessels each.

The first vessel in a train is separately shielded since over 80 percent of the total activity in the system is associated with this vessel.

e. Control room

The control room shielding is designed to maintain the dose requirement of 5 Rem whole body or its equivalent to any organ for the duration of the accident as specified in 10 CFR 50, Appendix A, Criterion 19. The analysis of the operator dose is presented in Section 15.6.5.

The control room emergency filter system used under accident conditions is described in Section 6.4.

f. General yard area

Areas outside of the plant structures are maintained as Zone I areas, i.e., less than 0.5 mRem/hr. Zone I level adjacent to the unshielded condensate storage tanks will be maintained by fencing and/or controlling the level of radioactivity in the stored condensate.

12.3.2.2.2 General Design Criteria

The following design criteria are applied to the plant shielding to minimize personnel exposures and to fulfill the intent of Regulatory Guide 8.8:

a. Access labyrinths are provided for rooms housing equipment that contains high radiation sources to preclude a direction radiation path from the equipment to accessible areas.

b. Piping penetrations, ducts and voids in radiation shield walls are located to preclude the possibility of streaming from a high to low radiation area, or otherwise will be adequately shielded.

c. Shielding discontinuities caused by shield plugs, concrete hatch covers and shield doors to high radiation areas are provided with offsets to reduce radiation streaming.

d. Radioactive piping is routed through high radiation areas where practicable, or in shielded pipe chases in low radiation areas.

e. Sufficient work area and clearance space is provided around equipment to permit ease of servicing.

f. Instruments requiring in-situ calibration will not normally be located in high radiation areas.

g. Non-radioactive equipment which requires servicing will not normally be located in proximity with potentially radioactive equipment.

h. Spread of contamination from radioactive spillage is minimized by:

    1. Providing steps at doorways to contain the spills.

    2. Providing a floor drain system which collects and routes the liquid to the liquid waste processing system for proper handling.

3. Using step-off pads between clean and contaminated areas.

Decontamination of an area if facilitated by use of materials and coatings which lend themselves to standard cleaning methods.

i. Natural traps which could be potential pockets of corrosion product activity are minimized in pipe and ducts by avoiding sharp bends, rough finishes, cracks, etc.

j. Shielding is provided for all equipment which is anticipated to be normally radioactive. The dose levels are designed not to exceed 10 CFR 20 requirements under the worst operating conditions of the plant.

k. Movable shielding will be available on the site in the event it is ever needed. This will include lead bricks, steel plates, etc.

l. Remote handling of radioactive materials is provided wherever it is needed and practicable.

12.3.2.2.3 Calculation Methods

Shielding calculations are performed using the following computer codes:

a. SDC(4)

This code is used for calculations involving relatively simple source configurations such as cylinders, spheres, slabs, disks or rod clusters.

b. QAD6G(5)

This code is used for performing shielding calculations with complex geometries.

c. G-33B[2]

This code is used to calculate scattering dose rates.

d. ANISIN[6]

This code is used to calculate the reactor neutron and gamma flux spectra through shield walls.

The source terms used in these computer codes are described in Section 12.2.1. In general, the maximum activity possible, even though remote, has been used in most pieces of equipment.

Beyond a conservative choice of source strengths, several simplifying assumptions have been used in the shielding analysis. Normally energy degradation (softening) of the radiation spectrum by the shielding is not considered. Thus, the calculated dose averaged energy is higher than would actually be the case with the given unshielded source.

Effects of structural steel (e.g., rebar) are neglected in shield walls. Instead, a uniform density of ordinary concrete of 2.35 grams/cc is assumed.

Dose rates are generally calculated on the centerline of the piece of equipment and 6 inches from the outside of the shield wall. In the case of a floor slab, the dose rate is calculated 2 feet above the floor on the equipment centerline. Further attenuation as the receiver moves farther from the shield is neglected.

The resultant radiation shielding thicknesses provided to maintain Zone II levels are given in Table 12.3-2 for the radioactive components.

12.3.3 VENTILATION

12.3.3.1 Design Bases

The plant ventilation systems are designed to accomplish the following:

a. Maintain the required ambient air temperature to prevent extreme thermal environmental conditions for operating personnel and equipment.

b. Protect the operating personnel against possible airborne radioactive contamination in areas where this may occur.

c. Ensure that maximum airborne radioactivity levels for normal and emergency operations, including anticipated operational occurrences, are within the limits of 10 CFR 20, Appendix B, for areas within plant structures and the restricted areas on the plant site where construction workers and visitors are permitted. The maximum levels correspond to design bases reactor coolant inventory.

d. Provide a suitable environment for continuous personnel occupancy in the control room under normal and post-accident conditions in accordance with 10 CFR 50, Appendix A, Criterion 19.

Those aspects of the plant design that transfer airborne radioactivity into the effluent control systems from equipment cubicles, corridors and operating areas normally occupied by operating personnel are described in Section 11.3.

The guidelines used to meet the intended design objectives for the plant ventilation systems are as follows:

a. Air movement patterns are provided from areas of lesser radioactive contamination to areas of progressively greater radioactive contamination prior to final exhaust.

b. Slightly negative pressures are maintained in specific areas such as the RWCU equipment rooms and the annulus between the containment and shield building to prevent uncontrolled exfiltration of contamination. The control room is maintained at slightly positive pressure during normal operation to prevent infiltration of potential contaminants.

c. Valves and equipment are as leak tight as practicable to prevent leakage of radioactive fluids and subsequent airborne contamination.

d. Individual air supplies are provided for each building to keep potentially contaminated air flows separate from noncontaminated air.

e. Potentially radioactive air is exhausted through filter trains consisting of roughing, HEPA, and charcoal filters to reduce onsite and offsite radiation levels. Filtered and monitored exhausts are provided in all buildings that could potentially contain radioactive airborne contamination, with the exception of the turbine building. In the turbine building, air exhausted from the heater bay and turbine operating floor is monitored for radioactivity and then directly discharged to the atmosphere through the turbine vent stack.

f. Roughing, HEPA, and charcoal filters are used for filtration of the recirculated air of the control room and associated offices during accident and other abnormal conditions.

g. Radiation exposures will be kept as low as practicable while servicing ventilation equipment by the following provisions (incorporated in the plant design):

   1. The ventilation equipment is not located in normally high radioactive areas.

   2. Suitable access doors and service aisles are provided to permit ease in servicing and maintenance.

3. The roughing and HEPA filters in the ESF filter trains and the HEPA filters in the non-ESF filter trains are serviced on the downstream side of the filter to minimize personnel exposure.

4. The activated charcoal adsorber is bulk loaded into the permanently installed, seal welded and gasketless adsorber section. Spent charcoal adsorber material is vacuumed from the bottom of the plenums and loaded directly into drums for shipment offsite, with new charcoal adsorber material being added at the top of the adsorber section. Thus, personnel are not directly exposed to potentially contaminated charcoal during the changing operation.

h. Control of airborne contaminants during maintenance operations is accomplished by the ventilation system as follows:

   1. Equipment redundancy is provided where practicable and idle equipment is isolated by dampers so that these components can be serviced without disrupting the operation of the system.

   2. Ventilation is accomplished in potentially contaminated areas by supplying air to the clean areas (corridors) and drawing it into the rooms through doorways and/or wall openings so that the air flow is always from clean to contaminated areas. This precludes any air flow reversal or air flow pattern disruption when doors of access hatches are opened for maintenance operations.

i. Ventilation ducts enter potentially radioactive areas at least 7 feet above the floor where practical. This minimizes direct exposure to personnel from radiation streaming. Penetration criteria are discussed in Section 12.3.2.

j. Ventilation equipment and ductwork are located in low exposure areas where practical. Therefore, automatic dampers, manual dampers and fire dampers can be maintained with minimum exposure to radiation. In potentially high radioactive areas, only manual balancing dampers on the inlet and outlet registers require attention. These are adjusted and set

in position prior to plant operation and require little further maintenance or adjustment.

k. Interior surfaces of ducts are designed to minimize the buildup of dust. Shop made duct joints are welded and field joints are gasketed with bolted connections. Ductwork joints, therefore, do not have gaps where dust could settle.

   The air flow velocity in the ductwork is generally high enough to keep dust suspended in the air stream. For the type of dust expected (light particles from clean surfaces in most auxiliary building areas), an air flow velocity of 1,500 to 2,000 fpm is sufficient.

   Grinding in the hot machine shop and changing of activated carbon adsorber material are dusty operations for which special filtering and ventilation are provided.

l. Access panels allow cleaning and inspection of ductwork. When ductwork needs cleaning, vacuum cleaning should usually be adequate. Where practical, ductwork is accessible for service and maintenance.

m. Exhaust air from vacuum cleaning or special ventilation can be ducted to an inlet connection in the building exhaust air system in most cases. In radioactive waste areas and the charcoal cleaning plenum areas, inlet connections are close to the expected cleaning equipment location. This minimizes extensive lengths of temporary ducting.

Figures 12.3-13 through 12.3-20 illustrate typical activated carbon adsorber plenum designs, showing filter mountings, access doors, aisle space, service galleries and provisions for testing, isolation and decontamination.

The criteria established for changing of ESF air filters and adsorbers in the charcoal cleaning systems, and a discussion of compliance with Regulatory Guide 1.52 is included in Section 6.5.1. Roughing and HEPA filters in non-ESF activated carbon adsorber plenums are replaced when the pressure loss across the filter exceeds twice the initial (clean filter) pressure loss. The pressure

loss is measured by permanently installed differential pressure indicators. Activated carbon adsorber b'.is are changed when laboratory tests of representative samples show that the adsorber fails to satisfy the testing requirements of Table 2 of Regulatory Guide 1.52. Design features of non-ESF filters are compared to the recommendations of Regulatory Guide 1.140 in Table 12.3-3.

Design bases and methods of operation for the plant ventilation system are discussed in Section 6.4, 6.5, and 9.4. For many plant ventilation systems protection of operating personnel from airborne radioactivity is not a limiting design consideration. These ventilation systems maintain a temperature suitable for continuous equipment operation . ..y.

The assumptions and analysis regarding the sources and amount of radioactivity that surround and leak into the control room (to adequately meet the radiation control requirements of 10 CFR 50, Appendix A, Criterion 19) are discussed in Section 15.6.5.

Assumptions used in the analysis of the plant ventilation systems are given in Table 11.3-(8) and Section 12.2.2. Maximum expected airborne radioactivity levels in the plant structures and building free volumes are also discussed in Section 12.2.2.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVTY MONITORING INSTRUMENTATION

Area monitoring instrumentation aids in minimizing personnel exposure to radiation. In addition, area monitors located in selected plant areas can provide useful information to the reactor operator following an incident, thereby enhancing the operators' ability to determine the nature and extent of the incident. Area monitors have a range of 0.1 to 10,000 mR/hr. This instrumentation also aids the operator in making correct decisions in directing personnel in the event of an incident involving high radiation. The system is used to monitor and demonstrate conformance with the guidelines of Regulatory Guide 8.8 and 10 CFR 20.

Airborne radiation monitoring instrumentation monitors for airborne radioactivity with sufficient sensitivity to provide information to the reactor operator to permit assessment of the radiological conditions within the plant. The system also aids in maintaining personnel exposure as low as reasonably achievable (ALARA). The system is used to monitor and demonstrate conformance with the guidelines of 10 CFR 20, and Regulatory Guide 8.8.

To ensure system availability, the non-safety radiation monitoring channels, which are non-movable, receive electrical power from an emergency diesel backed instrument bus. This supply is not available immediately following a LOCA.

Portable instrumentation and laboratory analysis of manual samples are used in assisting the determination of a course of action for major plant incidents.

12.3.4.1 Area Radiation Monitoring

The objective of the area radiation monitoring system is to indicate and record gamma radiation levels in areas where radioactive material may be present, stored, handled or inadvertently introduced. These monitors will alarm when administrative limits are close to being exceeded.

12.3.4.1.1 Design Bases

The area radiation monitoring system is designed to:

a. Provide plant personnel with a system that will indicate that the radiation levels are below those requiring special monitoring equipment.

b. Provide a system which can aid in minimizing personnel exposure to radiation and maintain occupational radiation exposure ALARA.

c. Provide instrumentation for the reactor operator to monitor selected plant area gamma radiation levels following an incident, thereby enhancing his ability to determine the nature and extent of the incident.

d. Augment and supplement other monitoring systems, such as the leak detection system and the airborne radiation monitoring system, in the detection of incidents involving release of radioactive material.

e. Provide alarms (alert and high radiation) to warn personnel when the gamma radiation level of selected areas increases substantially.

f. Provide a record of radiation levels as a function of time at key strategic areas within the plant.

g. Provide the reactor operator with alert, high radiation, and circuit failure alarms for each channel.

h. Aid the operator in personnel deployment decisions following an incident involving high radiation.

i. Assist in the detection of unauthorized or inadvertent movement of radioactive material in the plant.

j. Warn of excessive gamma radiation levels in areas where special nuclear material is stored or handled.

k. Warn personnel of high radiation in areas prior to entry by personnel.

The criticality accident alarm system (in accordance with Regulatory Guide 8.12 and ANSI N16.2-1969) is not provided, based on the analysis detailed in Section 9.1.2.

12.3.4.1.2 System Description

The area radiation monitoring system provides continuous detection, measurement and indication of the ambient gamma radiation level through the use of gamma sensitive detectors located in selected areas of the plant. This system supplements radiological protection for plant personnel, helps to minimize personnel exposure to radiation, and aids the reactor operator by providing instrumentation which may be used for monitoring radiation levels

throughout the plant during normal operation and following an incident. Data derived from these monitors in support of the plant radiation level surveillance is as recommended by ANSI N 13.2 and Regulatory Guide 8.2 and is used in demonstrating compliance with 10 CFR 20.

The system consists of 58 independent channels strategically located throughout the plant in areas where radioactive material may be present or inadvertently introduced, in areas where high radiation levels may develop, or in areas where the operator may gain information regarding the nature and extent of an incident. Fifty-one of the instrument channels use centralized control from rack mounted ratemeters (readout modules) located in the control room. Seven instrument channels are local self-contained units that have no interface with the control room.

Channels associated with Unit 1 (Table 12.3-4) have instrumentation readout modules located in the Unit No. 1 control room. Channels associated with Unit 2 (Table 12.3-5) have instrumentation readout modules located in the Unit 2 control room. Channels associated with the common areas (Table 12.3-6) of the plant have readout modules located on a common area radiation monitoring panel located in the Unit No. 1 control room. Channels associated with the waste processing and the radwaste building have remote alarms and meter indication on the radwaste control panel located in the radwaste building.

Most channels are operated from the control room panels. Each of these channels consist of three basic components: detector, alarm indicator unit, and a control room readout module. Channels for the turbine floor, personnel air lock and the upper pool area have additional remote alarm-indicator units. The control room channel is provided with a detector and readout module and has no alarm-indicator unit.

Local instrument channels (Table 12.3-7) consist of the detector, and an enclosure containing a single channel ratemeter, alarm light, and horn. The portable drywell area monitor is an enclosure with a single channel ratemeter This monitor will actuate the drywell evacuation alarm in a high radiation condition.

The detectors are wall mounted gamma sensitive devices located in the specific area of concern. The alarm and indicator unit is located nearby to provide plant personnel in the area with radiation dose rate level indication, visual alarms, and an audible alarm. Where necessary, remote warning units are provided in addition to the local alarm and indicator units.

Each channel has two warning functions at the local alarm and indicator unit: an amber warning light corresponds to an alert radiation level and a red warning light, with an associated audible alarm, corresponds to a high radiation level. Each channel has visual alarm indication of alert, high radiation and channel failure on the readout module. In addition, all channels (except local channels) are recorded on multipoint recorders located in the control room.

Components of the area radiation monitoring system are classified as non-safety related. Because of the relative importance of this system regarding plant personnel safety and information available to the reactor operator, the components of this system are high quality, reliable, stable and capable of operating in the expected environments at the installed location.

During refueling operations in the containment, a postulated fuel bundle drop in the upper pool would cause gamma radiation dose levels in the drywell (at elevation 655'-0") to reach approximately 1 R/hr. Since personnel will be in the drywell for maintenance and test operations during the refueling period, a portable area radiation monitoring system is provided during this time (Table 12.3-7).

The instrument channel consists of a detector, local alarm and indicator unit, and a connector for the drywell evacuation alarm. These components are portable, and are placed in the drywell prior to refueling. After refueling is completed, they are removed from the drywell. Electrical power, interconnecting cabling and connection boxes are permanently installed in the drywell. To install the portable components, personnel must place the detector on a wall bracket provided on the side of the primary reactor shield wall at elevation 655'-0", and place the alarm and indicator unit at elevation

629'-0". Connection boxes are mounted near the mounting brackets. Electrical power for each channel is provided by the local 120-volt a-c supply.

Channels 1D21N030, 1D21N080, 2D21N030, and 2D21N080 have an additional alarm and indicator unit remotely located and in full view just outside the respective containment personnel locks. This additional alarm provides plant personnel with containment radiation level status at the point of entry.

Channels D21N250, D21N260, D21N270, and D21N290 have an additional remote meter and an alarm window in the radwaste control room in order to provide the radwaste control room operator with radiation level information relative to certain areas of the radwaste building.

Channels D21N370 and D21N380 are associated only with the radwaste control panel.

Area radiation monitoring for personnel on the refueling bridges and fuel service handling platform will be provided with a permanently mounted local instrument channel. Each channel monitors the pool area below the platforms and has a local alarm and indicator unit for alert and high radiation warnings. The following are the channels on the platforms:

a. Channel 1D[illegible]0 - Refueling Bridge, Unit 1 Containment
b. Channel [illegible]0 - Refueling Bridge, Unit 2 Containment
c. Channel [illegible] - Fuel Service Handling Platform, Intermediate Building

Each channel consists of a detector mounted below the platform and a platform mounted alarm and indicator unit. This alarm and indicator unit has an alert alarm which activates an amber warning light, and a high radiation alarm which activates a red warning light and klaxon horn. Power to the channel and horn is from the platform's 120 volt a-c supply.

Area radiation monitoring for the solid radwaste drumming and storage area (elevation 623' 6" of the radwaste building) is provided by using two permanently mounted local channels. These channels indicate and alarm locally, and on the radwaste control panel, to warn personnel when gamma

radiation levels exceed predetermined limits. Each channel consists of a wall mounted detector, local alarm-indicator unit, and a meter and alarm on the radwaste control panel. Each alarm-indicator unit provides circuitry to activate an amber alert warning light, a red high radiation warning light and a single klaxon horn providing an audible alarm. The channels and horn are powered by local 120-volt a-c supply.

A detailed description of area radiation monitoring equipment is as follows:

a. Detectors

The detector is a gamma sensitive device housed in a sealed container separated from the local alarm and indicator unit. A halogen quenched Geiger-Mueller tube is coupled with a preamplifier to convert the incident gamma radiation into an electrical signal which is transmitted to the readout module. Design information for the detector is listed in Table 12.3-8. The detector assembly is wall mounted and strategically located so as to effectively survey the area for gamma radiation. The unit can be easily removed from its mounting for calibration or repair. The unit contains a radioactive check source which can be activated from the associated readout module to provide a verification of channel response. Radiation monitor detectors with Geiger Mueller tubes use circuitry which, upon saturation of the detector, maintain a continuous upscale reading.

b. Local Alarm and Indicator Unit

All channels, except the refueling bridge channels (D21N390, 1D21N090, and 2D21N090), the control room channels (1D21N400 and 2D21N400), and the local channels (1D21N340, 2D21N340, D21N380, and D21N370), are equipped with an alarm-indicator unit located near the detector assembly.

The local alarm-indicating unit consists of a wall mounted NEMA enclosure containing the following:

1. A readout meter.

2. A high radiation alarm light visible on a 180 degree horizontal azimuth from the wall and protected by a watertight red glass cover and a metal cage. The light receives 120-volt a-c, 60-Hertz, electrical power from the readout module.

3. An alert alarm light visible on a 180 degree horizontal azimuth from the wall and protected by a watertight amber glass cover and a metal cage. The light receives 120-volt a-c, 60-Hertz, electrical power from the readout module.

4. A klaxon horn providing an audible alarm in conjunction with the high radiation alarm. The horn receives 120-volt a-c, 60-Hertz, electrical power from the readout module. The audible alarm is capable of being silenced at the associated readout module in the control room.

c. Remote Warning Unit

Channels 1D21N030, 1D21N080, 2D21N030, and 2D21N080 have remote warning units in addition to alarm-indicator units. The remote warning units are identical to the alarm-indicator unit except that no audible alarm is required.

A remote warning light (required for channels 1D21N160 and 2D21N160 in addition to alarm and indicator units) is provided. The light is an incandescent wall-mounted, 120-volt a-c, 60-Hertz, 75 watt device powered from the associated alarm-indicator unit. The light is mounted on a NEMA 12 enclosure. The remote warning light (high radiation alarm) is visible on a 180 degree horizontal azimuth from the wall and is protected by a watertight red glass cover and a metal cage.

d. Readout Module

The readout module contains most of the electronic circuitry for system operation. The module consists of compact, solid state circuitry, a modular design which allows up to 3 modules to be arranged side-by-side in a rack mounted chassis located in the area radiation monitoring panel. The module contains alarm circuitry, a functional control switch and signal processing amplifiers for dose rate indication. Each module contains an independently fused regulated power supply. The modular compact design allows removal of the module from the chassis for replacement or repair. Circuit alignment can be accomplished while the system is energized.

The front panel of the readout module has the following features:

1. Three and one-half inch meter with meter range of 0.1 mR/hr to 10^4 mR/hr.

2. High level alarm lamp.

3. Alert alarm lamps.

4. Failure alarm lamp.

5. Switching capability with the following functions:

   (a) Function switch with off, operate and alarm positions.

   (b) Alarm trip test.

   (c) Check source actuate.

   (d) Horn silence.

(e) Alarm acknowledge which changes module alert or high level alarm lamp from "flashing" to "on" state, and returns module alarm output contacts to normal.

Design information for the readout module is listed in Table 12.3-9.

e. Recorders

Multi-point strip chart recorders, located on the area radiation monitoring recorder panels, provide a permanent record of the radiation levels at the selected locations throughout the plant. Each centralized radiation monitoring channel provides an input to one of the recorders. (Tables 12.3-4 and 12.3-5)

f. Power Supply

The area radiation monitoring system channels utilizing control from the centralized control room panels receive electrical power from two sources. The 120-volt a-c power to the readout module electronics is supplied from non-class 1E a-c instrument bus which is emergency diesel generator backed through a transfer switch (except during a LOCA). The second power source is used to supply 120-volt a-c power for all horns and alarm lamps. This power is supplied from a miscellaneous 120-volt a-c distribution panel and is not diesel backed.

g. System Setpoints

The alarm setpoints are adjustable and will be placed at or below the levels dictated by the anticipated or demonstrated levels of radiation, as indicated in the zones illustrated by Figures (12.3-1 to 12.3-11) with nominal setpoints of twice background for alert, three times background for the high alarm, but not more than 2.5 mR/hr in areas of very low radiation levels.

h. Calibration

Area radiation monitors are to be calibrated on a routine basis and after any major maintenance work is performed on the detector or its associated ratemeter. Detector calibration is obtained by exposure of the detector to a standard radioactive source with its activity traceable to a National Bureau of Standards Calibration. Frequency of routine calibration will be established on a yearly basis or as listed in the technical specification (Chapter 16).

12.3.4.2 Airborne Radioactivity Monitoring

One of the design objectives for plant ventilation systems is to minimize the accumulation of airborne radioactivity in areas within the plant by maintaining proper air movement patterns. This design objective aids in providing protection for plant operating personnel against airborne radioactive exposure, and aids in maintaining personnel occupational radiation exposure ALARA. The airborne radiation monitoring system supplements the design objective of the plant ventilation systems by monitoring for airborne radioactive contamination in the gaseous, iodine, or particulate form, in plant ventilation system exhaust paths and in the atmosphere of certain areas of the plant, in order to demonstrate compliance with 10 CFR 20, Appendix B, Table I. The airborne radiation monitoring system also provides information to the reactor operator that permits an assessment of the radiological conditions to be encountered within the many areas of the plant.

The system complies with Regulatory Guide 8.8 to assure that personnel exposures are maintained ALARA. The system aids the reactor operator in determining the nature and extent of incidents involving the release of radioactivity by augmenting and supplementing the reactor coolant pressure boundary leak detection system and the area radiation monitoring system. Data is recorded to provide a permanent record of plant airborne radioactivity levels. The instrument channels of this system provide control room indication, alarm, and control functions as required to limit the dispersal of radioactive material within ventilation systems.

Where required, movable radiation monitor subsystems are used for sampling selected work areas. These units indicate local airborne radioactivity levels and alarm locally to warn personnel within these areas that a significant increase in airborne radioactivity exists.

12.3.4.2.1 Design Basis

The airborne radiation monitoring system is designed to:

a. Furnish quantitative information (based on representative sampling) to the reactor operator and to operations personnel on the level of airborne radioactivity in plant ventilation systems and selected areas of the plant.

b. Provide a system which can aid in minimizing personnel exposure to airborne radioactivity and maintain occupational radiation exposure ALARA.

c. Provide an analysis sample medium from which the quantity of each principal radionuclide collected can be determined in conformance with Regulatory Guide 1.21.

d. Furnish information to substantiate radiation surveys as required by 10 CFR 20, and provide supporting documentation of working environments.

e. Provide instrumentation for the reactor operator to monitor plant ventilation systems, and selected areas of the plant for level of radioactivity during and following an incident, thereby enhancing the ability to determine the nature and extent of the incident.

f. Supplement the leak detection system in detecting leakage from the reactor coolant pressure boundary.

g. Provide overall plant monitoring of airborne radioactivity and reasonable assurance that the ambient airborne radiation levels are below those requiring special monitoring equipment.

h. Supplement other monitoring systems, such as the area radiation monitoring system, in the detection of incidents involving release of radioactive material.

i. Aid in the protection of the plant personnel from exposure to airborne radioactive materials in excess of the levels allowed by 10 CFR 20, Appendix B, Table I.

j. Provide the reactor operator with alarms for each channel (warning, high radiation, or channel failure) and alarms for each subsystem (sample flow low).

k. Provide the operations staff with a hard copy record of radiation levels in the monitored systems.

l. Continuously monitor the plant ventilation systems for airborne radioactivity in order to perm t an assessment to be made of the radiological hazards to be encountered within various regions of plant buildings, and to call attention to equipment malfunction or component failure resulting in the release of radioactivity.

m. Provide instrumentation for use as the basis for initiating actions related to the plant radiation emergency plan.

12.3.4.2.2 System Description

The airborne radiation monitor typically consists of a particulate measuring channel, an iodine measuring channel, and a gas measuring channel. These monitors provide supporting data for the surveillance of plant radioactive levels as recommended by ANSI N13.2 and Regulatory Guides 8.2 and 8.8 and documentation for demonstrating compliance with the requirements of 10 CFR 20. Instrumentation is provided to monitor the locations listed in Table 12.3-10.

A typical airborne radiation monitor subsystem is as follows:

A representative sample of air from a ventilation duct is drawn through a sample line to the airborne monitor unit by means of a Roots-type air blower. Sampling of the ducts is achieved by the use of an isokinetic sample probe placed in the air stream. The area of the probe tip is sized so that the velocity of the sample at the probe tip equals the velocity of the air at the design flow rates in the duct. The line is a 1-inch stainless steel pipe with a minimum number of bends and kept short to minimize loss of particulates due to gravity deposition. Sampling guidelines, as outlined in ANSI N13.1 (1969), are used where applicable. Table 12.3-11 indicates channels which use isokinetic probes.

Sampling points on ventilation ducts are taken, whenever possible, a minimum of five duct diameters downstream from abrupt changes in flow direction or flow entry, and a minimum of three duct diameters upstream of abrupt changes in flow direction; points are chosen such that the ventilation flow is fully developed and mixing is complete.

Sample line flowrate is such that particulate losses are limited due to gravity settling or turbulent flow. Bends in sample lines are of large radius (approximately 10 times line diameter).

The sample passes through a particulate, iodine, and gas channel in series. Each channel is independent. In the particulate channel, the sample air passes through a fixed or moving filter which collects particulates and is monitored by a beta scintillation detector, the output of which is preamplified and transmitted to a ratemeter located in the control room. The detector and filter are enclosed in a 4-Pi lead shield to reduce the background radiation effects. In the iodine channel, the sample passes through an activated charcoal cartridge which traps the radioactive iodine. A 4-Pi shielded gamma scintillation detector monitors the cartridge. The output signal is preamplified and transmitted to a ratemeter located in the control room. In the gas channel, the sample enters a 4-Pi shielded volume monitored by a beta sensitive scintillation detector. The output signal is preamplified and transmitted to a ratemeter in the control room.

The gas is exhausted back to the ventilation duct. Differential pressure switches across the filter and charcoal cartridges are provided to give a low flow alarm at the unit and in the control room. Flow regulation is used to maintain a constant flow through the filters of approximately 1-CFM. A flow indicator, flow alarms, and log ratemeter indication and alarms are also provided on the unit enclosure.

General system design requirements are as follows:

a. Equipment located outside the control room is housed in NEMA type 12 ventilated enclosures.

b. Setpoint adjustment devices are protected to prevent inadvertent operation.

c. Readout modules are accessible for test, alignment, changing setpoints, and calibration or inspection without interrupting power to the module.

d. The detectors used in the airborne radiation monitoring system are scintillation detectors. The entire detector assembly is built into a housing which serves as a shield against changes due to light photons or electrostatic or magnetic fields. The housing extends entirely over the base of the assembly, making a single unit. 4-Pi shielding covers the detector assembly. The design is such that the detector and source geometry are reproducible. The beta sensitive scintillation detector employs a phosphor with low sensitivity to gamma radiation, and the gamma scintillation subsystem resolution does not exceed 10 percent at full width at half maximum of the 0.661 MeV photopeak.

e. The detector preamplifier is contained within the detector housing. Preamplifiers are drift-free, linear and ensure a high signal-to-noise ratio.

f. High and alert radiation level alarm trip setpoints are adjustable over the entire range at the readout module.

g. Equipment is designed to be capable of withstanding an integrated gamma dose of 10^4 rads.

h. Remote actuated check sources provided with the detector assembly are Cesium -137 for beta scintillation detectors, and Barium -133 for gamma scintillation detectors.

Instrumentation channel stability is as follows:

a. The system operates on 120-volts a-c, 60 Hertz within voltage or frequency changes of ±15 percent.

b. At the fixed environmental design center, the channel drift will not exceed ±0.5 percent of linear full scale per day and ±3 percent of linear full scale in a period of 30 days.

c. For the operating temperature range, the shift due to temperature is less than 0.1 percent of linear full scale per °C.

d. The channel will operate within the design performance specifications within a time interval no greater than 5 minutes after the channel is energized from the cold condition.

e. At the environmental design center, the repeatability is ±10 percent at the 95 percent confidence level of any single measurement level.

The overall accuracy of the system is as follows:

The instrument error shall not exceed ±20 percent of reading over the upper 80 percent of its range, with the error defined as:

$$\% \text{ error} = \frac{R_T - R_R}{R_T} \times 100\%$$

where:

R_T = true quantity based on the signal leaving a pulse rate signal generator.

R_R = indicated quantity based on the linear recorder output signal.

Particulate filter media, similar to Hollingsworth and Vose type H-70 Gauze Back Filter, has a particulate collection efficiency of 99 percent for aerosol particulates of 0.3 micron and larger.

Activated charcoal cartridges or filters provide a minimum of 95 percent efficiency for field iodine retention (elemental and organic).

Sensitivity Requirements:

a. Sensitivity of particulate channels equipped with fixed filter:

The approximate sensitivity is 2.7×10^{-11} μCi/cc for Cs^{137} and is understood as being the concentration of airborne radioactivity which will produce, after 8 hours sampling, a count rate twice the count rate caused by a background radiation of 1×10^{-10} μCi/cc Radon in equilibrium with daughters plus an ambient field of 2.5 mR/hr (Cs^{137}, gamma) incident on the 4-Pi shielded sampler subassembly.

b. Sensitivity of particulate channels equipped with moving filter:

The approximate sensitivity is 4.1×10^{-10} μCi/cc for Cs^{137} and is understood as being the concentration of airborne radioactivity which will produce, at equilibrium, a count rate twice the count rate caused by a background radiation of 1×10^{-10} μCi/cc Radon in equilibrium with daughters plus an ambient field of 2.5 mR/hr (Cs^{137}, gamma) incident on the 4-Pi shielded sampler subassembly.

c. Sensitivity of iodine channels:

The approximate sensitivity is 1.6×10^{-11} μCi/cc for I^{131} and is understood as being the concentration of I^{131} (elemental or methyl iodide) which will produce, after 8 hours sampling, a count rate (with analyzer on differential) equal to the count rate due to background radiation caused by an ambient field of 2.5 mR/hr (Cs^{137}, gamma) incident on the 4-Pi shielded sampler subassembly.

d. Sensitivity of gas channels:

The approximate sensitivity is 4.7×10^{-7} μCi/cc for Kr^{85} and is understood as being the concentration of airborne radioactivity which will produce a count rate equal to the count rate caused by a background radiation of an ambient field of 2.5 mR/hr (Cs^{137}, gamma) incident on the 4-Pi shielded sampler subassembly.

Detectors:

a. Particulate monitoring channel:

The detector is a photomultiplier tube coupled to a beta sensitive plastic scintillator.

b. Iodine monitoring channel:

The detector is a photomultiplier tube coupled to a gamma sensitive NaI (Tl) scintillator.

c. Gas monitoring channel:

The detector is a photomultiplier tube coupled to a beta sensitive plastic scintillator.

Readout Module:

a. The readout module contains most of the electronic circuitry for system operation. The module consists of compact, solid state circuitry and modular design which provides for modules to be arranged side by side in a standard 19-inch rack mounted chassis. The readout modules are located in the control room airborne radiation monitoring instrument panels, except for readout modules associated with the "movable" subsystems. These readout modules are located on the movable equipment enclosure.

b. The readout module has a time constant which is inversely proportional to the count rate with the probable statistical error "E" less than 15 percent:

$$E = 0.67 \quad \frac{1}{\sqrt{2RC\ (CPM)}}$$

where RC = instrument time constant, min.

c. Each readout module contains its own independent fused regulated power supply suitable for 120-volt a-c, 60-Hertz power input.

d. Each readout module has provision for determining voltages essential for proper channel operation.

e. The readout modules have the following features (front panel):

1. Range: 10 to 10^6 CPM logarithmic

2. Meter Size: 4-1/2"; ±2% full scale accuracy

3. Meter Scales: 10 to 10^6 CPM
   High voltage 500 to 2500 volts d-c;
   Calibration check point.

4. Alarm Lamps:

   (a) High level.

   (b) Alert.

   (c) Failure.

5. Switching capability with the following functions:

   (a) Function switch: OFF-Calibrate-High Voltage-Operate.

   (b) Check source actuate.

   (c) Alarm trip test.

   (d) Alarm acknowledge.

6. Iodine channel readout modules have a single channel analyzer circuit to provide energy discrimination for selectively monitoring the 364 KeV I^{131} photo-peak. These modules are used in conjunction with the iodine sampling subassemblies. Switching is provided on these modules for "Integral" or "differential" mode. The differential and integral modes differ as follows: The differential mode selects the I^{131} photo peak; the integral mode provides for gross counting of all photopeaks present. The window and baseline of the analyzer is adjustable to allow differential measurement over the complete photo peak.

f. The readout module has the following features:

1. Independent regulated power supply.

2. High level alarm contacts (nonlatching).

3. Alert alarm contacts (nonlatching).

4. Failure alarm contacts (nonlatching).

5. Buffered output for computer (0 to 10 volts dc isolated, positive signal).

6. Output for recorder (0 to 10 mV d-c).

7. Output for remote readout meter(s).

8. Fixed circuit failure alarm setpoint.

9. Alert level alarm setpoint adjustment (variable over full range).

10. High level alarm setpoint adjustment (variable over full range).

11. Test points (scaler MCA and signal generator jacks).

g. Special features:

1. Both the high level and the alert level alarm lamps "flash" in an alarm condition. These lamps change from the "flashing" state to a "steady on" state when the module alarm acknowledge button is depressed. The lamps extinguish when the alarm condition has cleared.

2. Switching capability for alarm setpoint trip adjustment is provided such that, when operated, the meter will indicate the alarm setpoint and the alarm trip circuitry will actuate. Alarm point adjustment can also be made at this position.

3. Switching capability for check source operation is provided such that, when the pushbutton is depressed, a radioactive source is actuated at the detector assembly to provide a response check of the channel. In addition, the circuitry incorporates an alarm defeat

provision such that when the check source is actuated, the alert and high level alarm will not actuate. The check source returns to the retracted position upon loss of power.

4. The failure alarm will actuate upon loss of detector voltage, loss of line voltage, loss of amplifier output signal, or loss of detector input signal.

5. The readout module alert level alarm circuit, high level alarm circuit, and the failure alarm circuit will automatically reset when the initiating signal returns to normal.

6. Alarm outputs:

   (a) Alarm outputs have a DPDT contact for use with external control and annunciator circuitry.

   (b) Alarm relays are fail safe. The alarm relay remains energized until an alarm signal causes the relay to deenergize.

7. An accessible test point or connector is provided for input of a signal generator to the circuit for establishing alarm trip setpoints. Capability is provided for disconnection of the detector input signal during pulse generator test. In addition, a test point or connector is provided for scalar readout at a point after the discriminator circuit.

8. Readout modules for subsystems 1D17K670, 2D17K670, 1D17K660, and 2D17K660 have two separate alert and two separate high level setpoint adjustment circuits, either setpoint of which may be inserted into the alarm circuit. This provides two levels of alert and two levels of high alarm capability.

9. The readout module is provided with four connectors for termination of wiring on the module. One is provided for termination of electrical power supplied to the module, one provided for

termination of field wiring (excluding high voltage), one for termination of high voltage and one is used for termination of module output circuits (which includes alarm outputs, recorder and computer signals).

10. Power supply:

The airborne radiation monitoring channels utilizing control room readout modules receive electrical power from two sources. The 120-volt a-c power to the readout module electronics is supplied from non-Class 1E a-c instrument bus which is emergency diesel generator backed through a transfer switch (except during a LOCA). The second power source is used to supply 120-volt a-c non-Class 1E power for local horns and alarm lamps. This power is fed from miscellaneous 120-volt a-c distribution panels and is not diesel backed. Air sample pumps are fed from the 480-volt a-c 3 phase bus.

11. Calibration:

Each channel is calibrated routinely by exposure to an N.B.S. traceable source for verification of its initial calibration. Calibration of the monitors is also performed following any major required maintenance of the detectors.

Table 12.3-10 contains a summary of plant airborne radioactivity monitoring. Airborne monitors used as effluent monitors are also described in Section 11.5. A general description of each unit follows:

a. Reactor Building:

1. Drywell Atmospheric Radiation Monitor (1D17K670 and 2D17K670)

The purpose of the drywell atmospheric radiation monitor is to provide airborne radiological monitoring during periods of drywell entry as well as indicate drywell activity to the operator during reactor operation. This unit monitors for particulate, iodine, and

gaseous airborne radioactivity and utilizes a moving particulate filter. Alarms and level indication are provided in the control room and at the monitor which is located outside of the containment. No trips are associated with this monitor, but a high radiation alarm will actuate the drywell evacuation alarm system.

This monitor complies with Regulatory Guide 1.45 for reactor coolant pressure boundary leak detection systems whereas the channel components are qualified to function during and after a safe shutdown earthquake. Therefore, the monitor is classified as Seismic Category I.

The sample pump motor is powered from the diesel backed bus F1D08 to ensure system availability.

Isolation valves are provided at all containment penetrations as required for drywell and containment isolation. The intake valves are motor operated ball valves and the discharge valves are solenoid operated. These valves automatically close on a LOCA isolation signal. Valve control switches and indication lights are located in the control room.

2. Containment Atmospheric Radiation Monitor (1D17K680 and 2D17K680)

The purpose of the containment atmospheric radiation monitor is to alert the operator and personnel entering the containment of the airborne activity levels in the containment. This unit monitors the containment recirculated air for particulates, iodine and gaseous activity (Figure 9.4-16) and is located outside the containment. Local and control room alarms and level indication are associated with this unit. High radiation alarm will actuate the containment and drywell evacuation alarm system. In the event of reactor building isolation, this unit will provide iodine, particulate, and gaseous monitoring capability when it is feasable to re-open the containment isolation valves.

Isolation valves are provided at all containment penetrations as required for drywell and containment isolation. These valves automatically close on a LOCA isolation signal. Sample line intake valves are motor operated

ball valves. Valve control switches and indication lights are located in the control room.

3. Refueling Operation Atmospheric Radiation Monitor (D17K650)

During a refueling operation, this moveable airborne radiation monitor will be located inside containment to provide local information for use in protection of personnel. Particulate, iodine and gaseous monitoring is provided with local alarms and level indication. High radiation alarm will actuate the containment and drywell evacuation alarm system by use of a local interlock jack connection.

4. Containment Vessel and Drywell Purge Exhaust Radiation Monitor (1D17K660 and 2D17K660)

This unit provides particulate, iodine and gaseous monitoring of the containment exhaust airstream (Figure 9.4-17).

An isokinetic probe at a point outside the containment upstream of the exhaust filter trains is used to obtain the monitored air sample. Local and control room alarms and level indication are available. High radiation will alarm actuate the containment and drywell evacuation alarm.

b. Radwaste Building:

1. Radwaste Building Ventilation Exhaust Radiation Monitor (D17K720)

The unit provides monitoring of the exhaust airstream for iodine, particulate and gaseous activity at a point upstream of the building exhaust filter trains (Figure 9.4-7).

A common plenum discharges to either of two filter/exhaust fan trains. One isokinetic probe is located in each suction duct to a filter/exhaust fan train. Each probe line terminates at a common sample delivery line to provide a sample flow of approximately 1-CFM

to the monitor unit. Motor operated ball valves (one per line) are installed in each probe line and are interlocked to open when the corresponding exhaust fan train is operated.

Interlocks stop the radwaste building ventilation supply fans (M31C001A & B) upon a high alarm from the gaseous monitoring channel. Alarms and indications are provided locally, in the control room, and on the radwaste control panel.

2. Solid Waste Drumming Area Atmospheric Radiation Monitor (D17K810)

This monitor is used to provide information for protection of personnel in the solid waste drumming area. The monitor will continuously monitor the room air for particulate and gaseous radioactivity. Local alarms and indication are provided.

c. Auxiliary Building:

Auxiliary Building Ventilation Exhaust Radiation Monitor (1D17K700 and 2D17K700)

This unit provides monitoring of the building exhaust airstream for iodine, particulate and gaseous activity. An isokinetic probe at a point upstream of the building exhaust filter trains (Figure 9.4-5) is used to obtain a 1-CFM air sample. Interlocks stop the auxiliary building ventilation supply fans (M38C001A,B) upon a high alarm from the gaseous monitoring channel. Alarms and level indicators are provided locally and in the control room. A high level warning light is provided on the local ventilation control panel.

d. Control Complex:

Control Room Airborne Radiation Monitor (D17K770)

This unit is used to monitor the control room atmosphere for particulates, iodine, and gaseous radioactivity in order to maintain

control room habitability as required by 10 CFR 50, Appendix A, Criterion 19. An isokinetic probe is used to obtain a 1-CFM air sample from a point downstream of the common supply plenum (Figure 6.4-1). High gaseous radioactivity, initiates a signal to isolate the control room from the outside environment, and place the control room ventilation system into the emergency recirculation mode. The system also monitors the recirculated air. Alarms and indication are provided locally and in the control room.

A postulated design basis accident (LOCA) could result in radiation entering the control room through the ventilation system. Since a LOCA signal will also place the control room ventilation system into the emergency recirculation mode, and isolate the control room from the environment; and considering the fact that a LOCA signal itself incorporates sufficient redundancy, the airborne radiation monitor signal is considered a "diverse" signal, and does not require redundancy. Electrical isolation is utilized to disassociate safety class and non-safety class circuits.

e. Intermediate Building:

Intermediate Building Ventilation Exhaust Radiation Monitor (D17K730)

This unit is used to provide monitoring for gaseous activity in the building exhaust airstream. An isokinetic probe is used at a point downstream of the building exhaust fan (Figure 9.4-18) to obtain a 1-CFM air sample. Iodine and particulate filters are available for sampling purposes and laboratory analysis. Interlocks are provided to stop the intermediate building ventilation supply fan (M33C001) upon a high alarm. Alarms and level indication are provided locally and in the control room. A high radiation alarm will also energize a radiation trouble light on the local intermediate building ventilation control panel.

f. Fuel Handling Building:

Fuel Handling Area Ventilation Exhaust Radiation Monitor (D17K710)

This unit is used to provide monitoring for iodine, particulate and gaseous activity in the fuel handling area ventilation exhaust airstream. An isokinetic probe upstream of the filter exhaust plenums (Figure 9.4-4) is used to obtain a 1-CFM air sample. Interlocks are provided to stop the fuel handling building ventilation supply fans (M40C001A,B) upon a high alarm from the gaseous monitoring channel. Alarms and level indication are provided locally and in the control room. High radiation actuates the fuel handling area evacuation alarm system.

g. Heater Bay/Turbine Building Area:

1. Heater Bay Atmospheric Radiation Monitor (1D17K800 and 2D17K800)

   A moveable monitor is placed in the heater bay for continuous monitoring of gaseous activity and for collection of particulate and iodine samples for laboratory analysis. Alarms and level indications are local.

2. Turbine Building Atmospheric Radiation Monitor (1D17K790 and 2D17K790)

   Moveable units identical to the heater bay atmospheric radiation monitors are placed on the turbine room operating floor and provide monitoring of gaseous activity and collect particulate and iodine samples. Alarms and level indications are local.

h. Off-Gas Building:

Off-Gas Building Ventilation Exhaust Radiation Monitor (1D17K760 and 2D17K760)

This unit is used to provide monitoring of the building exhaust for iodine, particulate and gaseous activity. An isokinetic probe is used at a point upstream of the exhaust filter trains (Figure 9.4-10) to provide a 1-CFM air sample. Alarms and level indication are provided locally and in the control room.

12.3.4.3 Detection of MPCa Levels of Airborne Radiation

The function of the airborne radioactivity monitoring system is to monitor the air within a particular enclosure, or the exhaust air from an enclosure, for airborne radioactivity, and to indicate to the operator the level of airborne radioactivity. In performing this function, the system will assist plant operating personnel in maintaining the essentials of personnel industrial hygiene and in maintaining airborne radioactive contamination levels ALARA. The adequacy of the system and the necessity for the particular location of airborne radioactivity monitoring units is based on the following analysis:

The particular radioisotopes considered to be representative of typical airborne activities associated with BWR operation are Cs^{137}, I^{131}, Xe^{135}, Xe^{133}, Kr^{87}, Kr^{85m}, I^{132}, I^{133}, and I^{135}. Of these isotopes, Cs^{137} can be considered to be representative of the particulate group, I^{131} as representative of the iodine (halogen) group and Kr^{85} can be considered to be representative of the noble gas group for the purpose of the calculation which follows. To determine the adequacy of the radiation monitoring system, the dilution of the airborne radioactivity, as it is mixed with the building ventilation system, must be considered. A one MPC concentration (10 CFR 20, Appendix B) of the above mentioned radioisotopes was postulated separately for each sub-compartment throughout the plant during normal operation (Tables 12.3-4 to 12.3-11, inclusive), and the capability of detecting this

radioactivity was determined. The following data was used in the analysis (Tables 12.3-12 to 12.3-19, inclusive):

a. Assume a constant one MPC level of either Cs^{137}, I^{131}, or Kr^{85} in any sub-compartment during normal operation.

b. Exhaust flowrate from the sub-compartment.

c. Air dilution factor relative to the airborne radiation monitor sampling point is:

$$D = \frac{F_i}{F_t}$$

where: F_i = sub-compartment exhaust flowrate (CFM)

F_t = total flowrate at sample point (CFM)

d. Typical sensitivity of airborne radiation monitor as referenced in Section 12.3.4.2.2.

e. Airborne radioactivity concentration at radiation monitor sampling point.

$$C = C_i D$$

where: C_i = concentration of airborne radioactivity in sub-compartment

D = air dilution factor

As a result of the analysis, the following conclusions were reached:

a. Reactor Building:

1. The containment exhaust radiation monitor can detect 1 MPC level of I^{131} or Cs^{137} in any reactor building sub-compartment listed in Table 12.3-12.

2. The containment vessel and drywell purge monitor can detect a noble gas concentration of 1 MPC in the drywell (drywell at purge); containment pool area (refueling operation); containment free space, and the RWCU HX area (normal operation).

3. In the following locations, a noble gas concentration of 10 MPC can be detected by the containment vessel and drywell purge exhaust radiation monitor:

   (a) RWCU fill and drain backwash receiver tank area

   (b) 654'-0", east room

   (c) RWCU fill and drain holding pump room

   (d) RWCU fill and drain room

   These areas are radiation Zone 5 areas and as such are not normally entered. Entry is discussed in Sections 12.1.5 and 12.4.2.

4. With the use of the containment atmospheric radiation monitor, sufficient radiological surveillance is available for the information of personnel in the reactor building.

5. During refueling operations when the reactor is opened, radioactive substances from the reactor coolant may locally contaminate the air and not be detectable for some time on the exhaust monitor. The common refueling portable airborne radiation monitor (D17K650) is

provided to detect these local airborne radioactivity levels and to provide the local workers with alarms and indication. The moveable monitor's detection limits are below 1 MPC for I^{131}, Cs^{137} and Kr^{85}.

b. Radwaste Building:

1. The radwaste building ventilation exhaust radiation monitor can detect a 1 MPC level of I^{131} or Cs^{137} in any radwaste building sub-compartment listed in Table 12.3-13.

2. The radwaste building ventilation exhaust radiation monitor can detect a 10 MPC level of noble gas in any radwaste building sub-compartment listed in Table 12.3-13 except in the RWCU sludge decant pump room A where a 20 MPC level could be detected.

To assess the relative potential for airborne radioactivity in the areas where there are potential sources, it was assumed, that in a small sub-compartment with a low exhaust flowrate, the following conditions exist:

(a) Cold primary coolant in the system.

(b) A concentration in the sub-compartment of 1 MPC for any of the following isotopes: I^{131}, Cs^{137}, noble gas Kr^{85}.

(c) Partition factor: 10^{-3} for iodine, 10^{-4} for particulates, 1.0 for noble gases.

Using these assumptions, it was determined by calculation that the amount of leakage from equipment and components required to yield a concentration of 1 MPC for Cs^{137} or noble gas Kr^{85} was unlikely. However, the leak rate required to yield a 1 MPC level of I^{131} was found to be approximately 31.6 gal/hr. This leak rate is considered abnormal yet in any event, a 1 MPC level of I^{131} in any sub-compartment listed in Table 12.3-13 can be detected by the exhaust radiation monitor. Indication of abnormal leakage from

equipment is provided through the use of floor drains and sumps. Sump level alarms are provided to alert the operator when abnormal leakage conditions exist. Abnormal leakage conditions will also be monitored by periodic patrolling of the building by shift personnel.

3. The moveable solid waste drumming area atmospheric radiation monitor (D17K810) is provided for personnel protection with detection limits below 1 MPC for I^{131}, Cs^{137}, and Kr^{85}. The unit samples the local atmosphere.

c. Auxiliary Building:

1. The auxiliary building ventilation exhaust radiation monitor can detect a 1 MPC level of I^{131} or Cs^{137} in any building sub-compartment listed in Table 12.3-14.

2. The auxiliary building ventilation exhaust radiation monitor can detect a 1 MPC level of noble gas in building sub-compartments as listed in Table 12.3-14, except for some of the pump rooms where less than 10 MPC could be detected.

d. Intermediate Building:

The radiation Zone 5 areas in the intermediate building consist of spent fuel pool cooling and cleanup system equipment areas. These areas are exhausted through the fuel handling area ventilation system. The equipment located in the sub-compartments listed in Table 12.3-15 present no airborne radiological hazard to personnel occupying the intermediate building. Administrative controls for radiation Zone 5 areas, ventilation flow patterns, and absence of potential sources where open access is permitted, will limit personnel exposure to airborne radioactivity.

The intermediate building ventilation exhaust monitor is considered adequate for personnel protection.

e. Fuel Handling Area:

1. The fuel handling area ventilation exhaust radiation monitoring system can detect a 1 MPC level of I^{131} or Cs^{137} in any sub-compartment listed in Table 12.3-16

2. The fuel handling area ventilation exhaust radiation monitors can detect a 1 MPC level of noble gas in the cask storage pool, spent fuel pool, fuel transfer pool area and other areas as listed in Table 12.3-16.

f. Heater Bay:

1. The heater bay ventilation exhaust radiation monitors can detect a 1 MPC level of I^{131} or Cs^{137} on any floor level in the building (Table 12.3-17).

2. A noble gas concentration of 1 MPC on the following floors can be detected.

   (a) Elevation 580'6"

   (b) Elevation 600'6"

3. A 1 MPC level of noble gas at elevation 620'-6" can be detected in summer and less than 15 MPC in winter operation due to differences in the dilution factors. Less than 15 MPC can be detected in the FDW Lube Oil purifier room in summer or winter operation.

4. A 25 MPC level of noble gas at elevation 647'-6" hallway can be detected in winter operation and less than 2 MPC can be detected in summer operation.

To protect personnel at elevation 647'-6" and 620'-6" of the heater bay, local moveable monitors (1D17K800 and 2D17K800) are provided. These units can detect less than 1 MPC of I^{131}, Cs^{137} and Kr^{85} in the local atmosphere.

g. Turbine Building:

1. The turbine building ventilation exhaust radiation monitor can detect a 1 MPC level of I^{131} or Cs^{137} in all three areas of interest in Table 12.3-18.

2. A noble gas concentration of 1 MPC can be detected by the exhaust radiation monitors in all sub-compartments except in the condenser vacuum pump and sample extraction areas where less than 25 MPC can be detected.

Because of the size of the turbine building and the dose limits of detectable gas concentrations to the maximum permissible concentrations, moveable airborne radiation monitors (1D17K780 and 2D17K780) are provided for local sampling and gaseous monitoring with local alarms and indicators. The principal use of these monitors is for surveillance of specific localized areas in the turbine building to augment and supplement analyses provided by the turbine building fixed monitor.

h. Off-Gas Building:

1. The off-gas building ventilation exhaust radiation monitor can detect a 1 MPC level of I^{131} or Cs^{137} in any sub-compartment listed in Table 12.3-19.

2. A noble gas concentration of 1 MPC can be detected in areas listed in Table 12.3-19 by the off-gas building ventilation exhaust radiation monitor with the exception of the following areas where 10 MPC can be detected:

   (a) Elevation 584'-0",

   (b) Elevation 568'-0", Filter and demineralizer cubicles

   (c) 548'-6", condensate demineralizer backwash receiving tank area and backwash receiving tank area

   (d) 602'-6", dessicant dryer area after-filter, pre-filter room

   (e) 624'-0", hydrogen analyzer area

   These areas are either radiation Zone 5 areas, which are normally not entered, or Zone 2 areas.

In summary, the analysis of the adequacy of the airborne radioactivity monitoring system resulted in the following conclusions. A one MPC level (equivalent to less than 10 MPC hours) of Cs^{137} or I^{131} in any sub-compartment can be detected by the airborne radioactivity monitoring system. These radioisotopes are representative of the iodine and particulate groups of concern in considering airborne radiation monitoring. Kr^{85} was considered to be representative of the noble gas group of radioisotopes, although noble gases are considered to be an external airborne hazard. A one MPC level of Kr^{85} can not be detected in certain areas by the airborne radioactivity monitoring system. These areas in most cases are either radiation Zone 2 or radiation Zone 5 areas. Areas throughout the plant were considered in detail for their potential in presenting airborne radioactive hazards. It should be emphasized that radiation Zone 2 areas (normal continuous occupancy areas) contain no equipment, components, or piping capable of presenting a credible fixed or airborne radiation hazard.

Radiation Zone 5 areas (restricted access) are high radiation areas and, as such, are locked. Entry is under administrative control. Air samples of the

area will be taken prior to and during occupancy of the area by the health physics group.

Radiation Zone 4 areas (controlled, limited access) in the radwaste and off-gas building ventilation systems were analyzed for their potential for presenting an airborne radioactivity hazard. It was determined that an MPC level of Cs^{137} or Kr^{85} was unlikely. However, an abnormal leak could result in an MPC level of I^{131}, yet the radiation monitoring system and sump systems in the building would provide indication to alert the operator of this abnormality.

Moveable atmospheric radiation monitors are placed in selected locations throughout the plant for additional, localized surveillance of airborne radioactive concentrations.

Table 12.3-10 lists the ventilation systems, by identification number and name and the corresponding airborne monitors and sample points associated with each system. Table 12.3-11 gives calculated system flow rates which can vary as much as ±15 percent with no major effects upon the detectability of radioactive contaminants in the air.

On the basis of the above discussion, the airborne radioactivity monitoring system is adequate to ensure conservatively sufficient surveillance of airborne radioactive concentrations. The system will provide indication to the operator that an airborne hazard exists, should that hazard be manifest. The relative location of the airborne hazard can then be determined by using portable air samplers that can be manually connected to various selected sample points in the building ventilation ducts. Use of these sample points will direct health physics personnel to the particular area of concern. The exact location of the airborne radioactivity can then be found by sampling the particular subcompartments in that area.

The combination of the airborne radioactivity monitoring system in conjunction with administrative controls restricting and limiting personnel access, standard health physics practices, ventilation flow patterns throughout the plant, plant equipment layout, lack of sources in radiation Zone 2 areas, and

restricted and locked radiation Zone 5 areas, is sufficient to ensure that airborne radioactivity levels will be conservatively acceptable in terms of the required duration of personnel access through each area of the plant. A general review of these concepts follows:

a. High radiation areas (radiation Zone 5 areas), where whole body dose levels may exceed 100 mrem/hr, are kept restricted, locked, and conspicuously posted in accordance with 10 CFR 20. These areas are not normally entered and authorization must be obtained before entry. Prior to entry, a high volume portable air sampler will be used by the health physics group to collect a representative air sample. Gaseous and particulate activity of the air sample will be analyzed before entry.

b. Air flow patterns are consistent with the basic ventilation design criteria of the plant. Clean, filtered outside air is supplied to Zone 2 areas (corridors, clean areas); these areas are exhausted into rooms and areas of successively higher potential for airborne contamination. Air flow is such that reversal or exfiltration from potentially contaminated areas is precluded. This ventilation arrangement essentially eliminates the possibility of personnel exposure to airborne radioactivity in continuous occupancy areas such as radiation Zone 2 areas. (See Section 12.3.3.)

c. All credibly potential sources of hazardous levels of airborne radioactivity from piping and equipment are located in Zone 5 radiation areas, which are locked or guarded and Zone 4 radiation areas, where entry is administratively limited. Radiation Zone 2 areas do not contain piping and components that would be potential sources of airborne radioactivity. This reduces the potential for above specification airborne radioactivity exposure to occupants of radiation Zone 2 areas where general entry is permitted. (See Sections 12.1.2.1 and 12.1.2.2)

d. Maintenance, in significantly radioactive areas (radiation Zone 4 areas), that requires a time interval longer than would be desirable on a normal building surveillance tour will require a radiation work permit. Particulate and gaseous samples of the area will be taken during

occupancy. These samples will be analyzed as required for good personnel radiological hygiene.

e. Healtn physics programs are discussed in Section 12.5.

Additional comments are as follows:

a. Auxiliary Building Ventilation Exhaust:

A major hazard has not been identified in this building; however, a monitor is provided for prompt detection of any unusual releases. With the sensitivity as discussed in Section 12.3.4.2.2, the monitor will provide direct indication of excessive release. Analysis of the particulate and halogen filters will allow evaluation of the potential airborne hazards and indicate the need for respiratory equipment, if warranted. With a weekly analysis of the activity build-up on the sampling filters, a gross sensitivity of approximately 1.5×10^{-11} μCi/cc can be interpreted. This level is well within 10 CFR 20 limits.

b. Radwaste Building Ventilation Exhaust:

The monitor sensitivity is as discussed in Section 12.3.4.2.2. The hazard condition referenced in Section 15.7.2 will release 6.40×10^{-3} curies of I^{131}. With an approximate room volume of 4,000 ft^3, a ventilation turn-over rate of 6/hour and a total building exhaust of 30,000 CFM, the concentration of I^{131} in the building exhaust is expected to be within the detection range of the monitor. At this level, respiratory equipment may be required to enter the hazardous area.

c. Reactor Building Purge Exhaust:

The failure of an instrument line has been identified as a possible hazard as indicated in Section 15.6.2. With a purge air flow, an I^{131} activity of 2×10^{-2} μCi/g in the reactor coolant, an instantaneous mixing of the released activity in the whole containment volume and a monitor sensitivity for iodine of approximately 2×10^{-9} μCi/cc (for five

minute sampling) a high radiation alarm can be obtained in the control room following the leakage of less than 10 gallons. A plate cut factor of 2 is taken into consideration for this evaluation.

With a reactor coolant pressure of 1,000 psi, and a flow restrictor of 1/4 inch, the initial flow rate through the break assuming 100 percent flashing will be in excess of 10 gpm, thus a monitor response to the potential hazard of less than 10 minutes is expected.

d. Fuel Handling Building Exhaust:

The drop of a channeled spent fuel bundle has been identified in Section 15.7.4 as a hazard for personnel in this building.

Assuming that 7.3×10^2 Ci of Kr^{85} (Table 15.7-18)* are released and mixed instantaneously into the whole volume of the fuel handling building, the resulting concentration is expected to be 1.7×10^{-2} μCi/cc. This activity level is well within the range capability of the monitor.

The response time of the monitor is inversely proportional to the activity level and is expected to be negligible at the high anticipated levels which may be reached during this calculable fuel handling accident.

The particulate and iodine filters are removable for laboratory analysis to verify and identify activity levels and to provide a backup to the continual monitoring of the areas of surveillance.

Portable air samplers with appropriate filters will be used to determine localized exposure levels and to permit the proper selection of respiratory protective equipment for the occupied portions of these buildings.

*CEI to provide fuel handling accident data (from PAR before releases can be finalized.

Radiation monitors located in each of the auxiliary, radwaste, reactor and fuel handling buildings take representative samples of the building exhaust air. Iodine and particulate filters are used to collect samples, and will accommodate the identification of specific types of contamination which can be localized and cleaned up. Health physics personnel will maintain a regular sampling routine in these areas to complement the radiation monitors.

12.3.4.4 System Setpoints

Alarm setpoints are adjustable. The alert setpoint is usually selected on the basis of operational considerations and may be varied at the discretion of the operator to provide flexibility for alerting personnel of conditions that might approach high alarm status or significant variations from normal operating levels. This setting is always at a point less than the high alarm setpoint. The high alarm level is established as a setpoint consistent with the background levels and the maximum concentrations desired for the applicable system or situation.

The high level setpoints take into consideration the following:

a. Efficiency of the filter or cartridge (where used for particulate or iodine).

b. Maximum operational concentrations in each individual ventilation system.

c. Background radiation effects which may include Radon for particulate Beta channels.

d. Meter and instrument error.

e. Calibrated sensitivity of the instrument channel.

f. Sample period and flow rate for fixed filter particulate and iodine channels.

g. Filter advance rate and sample flow rate for moving filter particulate channels.

The minimum level for the high alarm setpoint will be at a countrate equal to three times background. This setpoint may be increased to a level consistent with the maximum concentration allowed for the applicable area or ventilation system and may be expressed as follows:

$$[(CKT) + (B)] \geq \text{setpoint} \geq [3B]$$

where:

B = countrate resulting from background radiation (may include effects of Radon for particulate Beta channels).

C = concentration in μCi/cc of the ventilation system or area established as the alarm condition.

K = constant for instrumentation channel based on sensitivity expressed as counts per minute per μCi/cc (CPM/μCi/cc/minute for filter channels) and including the effects of instrument error and filter efficiency where applicable.

T = time of sample period expressed as 480 minutes for fixed filter channels. Expressed as unity (1) for gas channels.

Typical maximum high level setpoint calculations based on MPC levels are expressed as follows:*

a. Gas Channels

assume: C 1×10^{-5} μCi/cc
K = 3.9×10^{7} CPM/μCi/cc and instrument error of 25 percent
B = 25 CPM

*Will modify based on current Victoreen data.

then:

$$(1 \times 10^{-5}) \times (3.9 \times 10^{7}) \times (1 - .25) + (25) \cong 300 \text{ CPM}$$

b. Particulate Channel (fixed filter)

assume: $C = 1 \times 10^{-9}$ μCi/cc

$K = 1.3 \times 10^{10}$ CPM/m/μCi/cc and instrument error of 25 percent assuming 99 percent filter efficiency

B = 52 CPM plus 150 CPM (Radon)

T = 480 minutes

then:

$$(200) + (1 \times 10^{-9}) \times (1.3 \times 10^{10}) \times (1 - .25) \times (.99) \times (480) \cong 4{,}800 \text{ CPM.}$$

Since 4,800 is not an easily readable point on the logarithmic scale meter, the setpoint would be reduced to the next readable graduation which is approximately 4,000 CPM.

c. Iodine Channel

assume: $C = 9 \times 10^{-9}$ μCi/cc

K = 11 CPM/min for 9×10^{-9} μCi

instrument error is 25 percent

cartridge efficiency is 77 percent

iodine available for collection 97 percent

B = 20 CPM

T = 480 minutes

then:

$$(20) + (9 \times 10^{-9}) \times (11/9 \times 10^{-9}) \times (1 - .25) \times (.97) \times (.77) \times (480) \cong 2,970 \text{ CPM}$$

The setpoint would then be set at the closest graduation on the meter, which is 3,000 CPM.

12.3.5 REFERENCE FOR SECTION 12.3

1. E.A. Straker, et al., "The Morse Code, A Multigroup Neutron and Gamma Rate Monte Carlo Transport Code", ORNL-4585, Oak Ridge National Laboratory (1970).

2. "G-33B, Multigroup Gamma Ray Scattering Code", NASA Lewis Research Center, Cleveland, Ohio.

3. W. A. Woolson, et al., "Calculation of the Dose at Site Boundaries from N-16 Radiation in Plant Components", JRB 72 5076J, December 8, 1972.

4. Arnold, E. D. and Maskewitz, B. F., "SDC - A Shield Design Calculation Code for Fuel Handling Facilities", ORNL-3041, March 1966.

5. Malenfont, R. E., "QAD, A Series of Point-Kernel General Purpose Shielding Programs", Los Alamos Scientific Laboratory Report No. 3573, April 5, 1967.

6. Boling, M. A. and W. A. Rhoads, "ANISIN/DT FII, A One-Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering." AL-66-MEMD-171.

TABLE 12.3-1

RADIATION ZONE DESIGNATIONS AND CODE

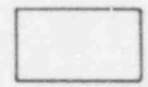
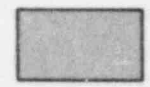
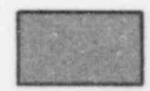
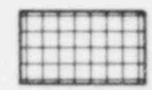
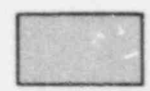
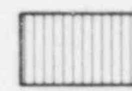
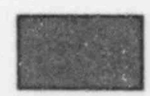
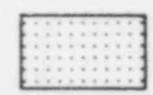
Zone Designation	Maximum Exposure (mRem/hr)	Normal Access	Normal Operation	Shutdown/ Refueling
I	0.5	Continuous		
II	2.5	40 hours/week		
III	25.0	4 hours/week		
IV	100.0	1 hour/week		
V	> 100.0	Established by Health Physics Staff (less than 1 hour/week)		

TABLE 12.3-2

RADIATION SHIELD THICKNESSES

Equipment Identification		Shield Thickness (ft. of concrete)
a. AUXILIARY BUILDING		
E12B001	RHR heat exchanger	2
E12C002	RHR pump	2
E21C001	LPCS pump	2
E22C001	HPCS pump	2
E51C001	RCIC pump	2
G33C001	RWCU pump	3
b. REACTOR BUILDING		
G33B001	RWCU heat exchanger (regen.)	4
G33B002	RWCU heat exchanger (non-regen.)	4
G36A003	RWCU fill and drain backwash receiving tank	3
G36C001	RWCU fill and drain holding pump	2
G36D001	RWCU fill and drain	3-1/2
G50C012	RWCU backwash transfer pump	2
c. INTERMEDIATE BUILDING		
G41A002	Fuel pool surge tank	1
G41A003	Fuel transfer tube drain tank	2
G41B001	Fuel pool heat exchanger	2
G41C003	Fuel pool circulating pump	2
G41C005	Fuel transfer tube drain pump	2
G41D001	Fuel pool fill and drain	2-1/2
G50A022	Fuel pool fill and drain backwash receiving tank	2-1/2
G50C027	Fuel pool fill and drain transfer pump	2-1/2
d. RADWASTE BUILDING		
G50A001	Liquid waste collection tank	2
G50A002	Liquid waste sample tank	2
G50A003	Floor drains collection tank	2
G50A004	Floor drains sample tank	2
G50A005	Chemical waste tank	2
G50A006	Concentrated waste tank	2-1/2
G50A007	Chemical waste distribution tank	1
G50A009	Spent resin tank	3
G50A011	Condensate fill and drain settling tank	3
G50A013	RWCU settling tank	3
G50A014	Waste sludge settling tank	2
G50A024	Waste collection filtrate tank	2
G50A025	Floor drains filtrate tank	2
G50C001	Waste collector transfer pump	1
G50C002	Waste sample pump	1

TABLE 12.3-2 (Continued)

	Equipment Identification		Shield Thickness (ft. of concrete)
d.	RADWASTE BUILDING (Cont'd)		
	G50C003	Floor drains collector pump	1
	G50C004	Floor drains sample pump	1
	G50C005	Chemical waste pump	1
	G50C006	Chemical waste distribution pump	1
	G50C008	Spent resin pump	2
	G50C010	Condensate sludge discharge mix pump	1
	G50C011	Condensate sludge decant pump	1
	G50C013	RWCU sludge discharge mix pump	2
	G50C014	RWCU sludge decant pump	1
	G50C015	Waste sludge discharge mix pump	1
	G50C016	Waste sludge decant pump	1
	G50C017	Waste collection filtrate pump	1
	G50C018	Floor drains filtrate pump	1
	G50C026	Conc. waste transfer pump	2
	G50D001	Waste collector filter	2
	G50D002	Floor drains filter	2
	G50D003	Waste demineralizer	2-1/2
	G50D004	Floor drains demineralizer	2-1/2
	G50Z001	Waste evaporator cond.	2-1/2
e.	TURBINE POWER COMPLEX		
	N23D001	Condensate filter	2-1/2
	N24A001	Condensate demineralizer cation regen. tank	2-1/2
	N24A002	Condensate demineralizer anion regen. tank	2-1/2
	N24A003	Condensate demineralizer mix and hold tank	2-1/2
	N24D001	Condensate demineralizer	2-1/2
f.	OFF-GAS BUILDING		
	N64B010	Off-gas cooler condenser	3
	N64D011	Off-gas prefilter	3
	N64D012	Off-gas charcoal absorber	3
	N64D016	Off-gas after filter	1
	N64D030	Off-gas dessicant dryer	3
g.	TURBINE ROOM		
	N25B001	Moisture separator reheater	3
	N31C001	Main turbine	3
	N64B001	Off-gas preheater	4
	N64B002	Off-gas condenser	4
	N64D005	Off-gas catalytic recombiner	4

TABLE 12.3-2 (Continued)

	Equipment Identification		Shield Thickness (ft. of concrete)
h.	HEATER BAY		
	N21B003	Compressure feedwater heater no. 3	2
	N21B004	Direct contact heater	2-1/2
	N27B001	Intermediate pressure feedwater heater no. 5	2
	N27B002	High pressure feedwater heater no. 6	2
	N27B003	Heater no. 5 drain cooler	1-1/2
	N27C003	Main feedwater pump drive turbine	2
	N33B002	Steam seal evaporator	2

TABLE 12.3-3

COMPARISON OF NON-ESF CHARCOAL FILTER SYSTEMS TO REGULATORY GUIDE 1.140 CRITERIA

Regulatory Position	System Design Feature
1.a	The design conforms with this position.
1.b	The design conforms with this position.
1.c	The design conforms with this position.
1.d	The design conforms with this position.
2.a	All non-ESF charcoal plenums are Seismic Cateogry I and consist of the following components in sequence: prefilters, HEPA charcoal, and HEPA. Fans, ducts, dampers and related instrumentation are also provided.
2.b	The design conforms with this position except that HEPA filter arrangement for the radwaste and auxiliary buildings is 4 wide by 5 high.
2.c	The design conforms with the intent of Section 5.6 of ERDA 76-21.
2.d	The design conforms with this position.
2.e	The design conforms with this position.
2.f	The design conforms with this position.
3.a	The relative humidity of exhaust air for the non-ESF charcoal filter systems is not expected to exceed 70 percent.
3.b	The design conforms with this position.
3.c	The design conforms with this position.
3.d	The design conforms with the intent of Section 4.4 of ERDA 76-21.
3.e	The design conforms with this position.
3.f	The design conforms with this position.
3.g	The design conforms with this position.
3.h	The design conforms with this position.
3.i	The design conforms with this position.

TABLE 12.3-3 (Continued)

Regulatory Position	System Design Feature
3.j	The design conforms with this position.
3.k	The design conforms with this position.
3.l	The design conforms with this position.
4.a	The design conforms with the intent of the recommendations of Section 2.3.8 of ERDA 76-21 and Section 4.7 of ANSI N509-1976.
4.b	The maximum length of component plus 2ft. 6inches is provided due to space limitations imposed by equipment room size. It was determined that this is adequate for the replacement of the prefilters and HEPA filters, and is consistent with the manufacturer's recommendation.
4.c	The design conforms with this position.
4.d	The design conforms with this position.
5.a	The testing procedure will conform with this position.
5.b	The testing procedure will conform with this position.
5.c	The testing procedure will conform with this position.
5.d	The testing procedure will conform with this position.
6.a	The testing procedure will conform with this position.
6.b	The design and testing procedures will conform with this position.

TABLE 12.3-4

DETECTORS ASSOCIATED WITH UNIT 1[(1)]

Detector[(4)]	Channel[(2)]	Location[(5)]
1D21N030[(3)]	Personnel air lock	Containment at el. 600'-6"
1D21N040	CRD HCU west	Containment at el. 620'-6"
1D21N050	RWCU fill and drain receiver tank area	Containment at el. 642'-0" east
1D21N060	TIP drive area	Containment at el. 600'-6"
1D21N070	RWCU fill and drain area	Containment at el. 664'-7"
1D21N080[(3)]	Upper pool area	Containment at el. 689'-6"
1D21N110	Auxiliary building, 574' east	Auxiliary building at el. 574'-10" east
1D21N120	Auxiliary building, 574' west	Auxiliary building at el. 574'-10" west
1D21N130	Turbine room east	Turbine room at el. 647'-6" east
1D21N140	CRD HCU east	Containment at el. 620'-6"
1D21N160[(3)]	Turbine room west	Turbine room at el. 647'-6" west
1D21N170	Turbine building, 605'	Turbine building at el. 605'-6"
1D21N180	Hotwell pump area	Turbine building at el. 577'-6"
1D21N190	Turbine building sump area	Turbine power complex at el. 548'-6"
1D21N200	Off-gas building, 584'	Off-gas building at el. 584'-0"
1D21N210	Condensate filter pump area	Turbine power complex at el. 568'-6"
1D21N220	Off-gas after filter area	Off-gas building at el. 602'-6"
1D21N230	High pressure feedwater heater area	Heater bay at el. 600'-6"
1D21N240	Feedpump area	Heater bay at el. 647'-6"

TABLE 12.3-4 (Continued)

Detector(4)	Channel(2)	Location(5)
1D21N400	Control room	Control complex at el. 654'-6"
1D21N410	Off-gas holdup area	Turbine building at el. 577'-6"

NOTES:

1. Readout modules located on panel 1H13-P803 and recorded on panel 1H13-P600.

2. All channels have a local alarm and indicating unit except 1D21N400.

3. In addition to a local alarm and indicating unit, channels 1D21N030 and 1D21N080 have remote warning units, and channel 1D21N160 has a remote warning light.

4. Range of detectors is 0.1 to 10^4 mr/hr.

5. Detector and local alarm and indicating units located inside containment are housed in unpainted aluminum NEMA Type 4 enclosures. All other equipment is housed in NEMA Type 12 enclosures.

TABLE 12.3-5

DETECTORS ASSOCIATED WITH UNIT 2[1]

Detector[2]	Channel	Location[3]
2D21N030	Personnel air lock	Containment at el. 600'-6"
2D21N040	CRD HCU west	Containment at el. 620'-6"
2D21N050	RWCU fill and drain receiver tank area	Containment at el. 642'-0" east
2D21N060	TIP drive area	Containment at el. 600'-6"
2D21N070	RWCU fill and drain area	Containment at el. 664'-7"
2D21N080	Upper pool area	Containment at el. 689'-6"
2D21N110	Auxiliary building, 574' east	Auxiliary building at el. 574'-10" east
2D21N120	Auxiliary building, 574' west	Auxiliary building at el. 574'-10" west
2D21N130	Turbine room east	Turbine room at el. 647'-6" east
2D21N140	CRD HCU east	Containment at el. 620'-6"
2D21N160	Turbine room west	Turbine room at el. 647'-6" west
2D21N170	Turbine building, 605'	Turbine building at el. 605'-6"
2D21N180	Hotwell pump area	Turbine building at el. 577'-6"
2D21N190	Turbine building sump area	Turbine power complex at el. 548'-6"
2D21N200	Off-gas building, 584'	Off-gas building at el. 584'-0"
2D21N210	Condensate filter pump area	Turbine power complex at el. 568'-6"
2D21N220	Off-gas after filter area	Off-gas building at el. 602'-6"
2D21N230	High pressure feedwater heater area	Heater bay at el. 600'-6"
2D21N240	Feedpump area	Heater bay at el. 647'-6"
2D21N400	Control room	Control complex at el. 654'-6"
2D21N410	Off-gas holdup area	Turbine building at el. 577'-6"

TABLE 12.3-5 (Continued)

NOTES:

1. Readout modules located on panel 2H13-P083 and recorded on panel 2H13-P600.

2. Range of detectors is 0.1 to 10^4 mr/hr.

3. Detector and local alarm and indicating units located inside containment are housed in unpainted aluminum NEMA Type 4 enclosures. All other equipment is housed in NEMA Type 12 enclosures.

TABLE 12.3-6

DETECTORS ASSOCIATED WITH COMMON AREAS[1]

Detector[4]	Channel[2]	Location
D21N250[3]	Radwaste 574' west	Radwaste building at el. 574'-10" west
D21N260[3]	Radwaste 574' east	Radwaste building at el. 574'-10" east
D21N270[3]	Radwaste 602'	Radwaste building at el. 602'-0"
D21N280	Process sample room	Radwaste building at el. 623'-6"
D21N290[3]	Radwaste evaporator area	Radwaste building at el. 623'-6"
D21N310	Fuel pool cleanup fill and drain area	Intermediate building at el. 599'-0"
D21N320	Fuel preparation pool	Intermediate building at el. 620'-6"
D21N330	Spent fuel storage pool	Intermediate building at el. 620'-6"
D21N420	Fuel pool cooling circulating pump area	Intermediate building at el. 574'-10"

NOTES:

1. Readout modules located on panel H13-P906 and recorded on panel H13-P907.
2. All channels have an alarm and indicating unit.
3. In addition to an alarm and indicating unit, channels D21N250, D21N260, D21N270 and D21N290 have a remote meter and remote alarm contacts for annunciation on panel H51-P031.
4. Range of detectors is 0.1 to 10^4 mr/hr.

TABLE 12.3-7

DETECTORS ASSOCIATED WITH LOCAL CHANNELS

Detector(3)	Channel	Location(4)
1D21N340	Unit No. 1 drywell (portable)	Containment at el. 655'-0" (Azimuth approximately 160°)
2D21N340	Unit No. 2 drywell (portable)	Containment at el. 655'-0" (Azimuth approximately 200°)
D21N380(1)	Waste compactor area	Radwaste building at el. 623'-6"
D21N370(1)	Solid radwaste drumming area	Radwaste building at el. 623'-6"
1D21N090(2)	Refueling bridge	Containment at el. 689'-6"
2D21N090(2)	Refueling bridge	Containment at el. 689'-6"
D21N390(2)	Fuel service handling platform	Intermediate building at el. 620'-6"

NOTES:

1. Channels D21N380 and D21N370 are locally mounted channels and have a remote meter on panel H51-P031 and a remote alarm window.

2. Channels 1D21N090, 2D21N090, and D21N390 are locally mounted channels mounted to refueling bridges.

3. Range of detectors is 0.1 to 10^4 mr/hr.

4. Detector and alarm and indicator units located inside containment are housed in unpainted aluminum NEMA Type 4 enclosures. All other equipment is housed in NEMA Type 12 enclosures.

TABLE 12.3-8

DETECTOR DESIGN REQUIREMENTS

Type	G-M tube
Range, mR/hr	0.1 to 10^4
Energy dependence	±20% (100 KeV to 2.5 MeV)
Circuitry	Solid state preamplifier
Mounting	Wall bracket
Remote Capability	Up to 1500 feet
Exposure	Capability of withstanding a total integrated dose of 10^5 rads
Enclosure	NEMA 12 (NEMA 4 in containment)
Energy Range	Responsive to gamma radiation over an energy range of 80 KeV to 7 MeV
Dead Time	20µ sec.

TABLE 12.3-9

READOUT MODULE DESIGN REQUIREMENTS

Response Time	Meter response is approximately 2.5 seconds for full scale deflection. Time constant of 60, 6, .06 seconds at 0.01, 0.1, 1 mr/hr respectively.
Susceptibility	The input signal is shielded to prevent gross fluctuations and false trips due to normal electromagnetic interference caused by electric motors, circuit breaker closure, welding.
Stability	Drift is less than ±3% of the measured point over a period of 30 days at environmental design center.
	The system is capable of operation on 120 volt ac, 60 hertz and will operate within specifications under voltage or frequency changes of ±10%.
	For the operating temperature range the shift due to temperature will be less than 0.5% per °C.
Accuracy	The overall accuracy of the system will be the actual reading relative to the true reading within ±25% of any decade at a reference energy in the range of 0.1 to 2.5 MeV.
Precision	The precision will be ±10% of any single measurement level at the environmental design center.

TABLE 12.3-10

AIRBORNE RADIATION MONITOR SUBGROUP
UNITS 1 AND 2

Radiation Monitor Subsystem	Sample Point	Instrument Channels[1]	Function of Subsystem	Location
D17K720 Radwaste Building Ventilation Exhaust Radiation Monitor	Ventilation ductwork upstream of filter trains	GSP HSP PSP	Local, control room and radwaste control room indication and alarms. Ventilation supply fan trip on high radiation.	Radwaste Bldg. 623'-6" East
1D17K700, 2D17K700 Auxiliary Building Ventilation Exhaust Radiation Monitor	Ventilation ductwork upstream of filter trains	GSP HSP PSP	Local and control room alarms and indication. Ventilation supply fan trip on high radiation.	Aux. Bldg. 620'-6" West
D17K730 Intermediate Building Ventilation Exhaust Radiation Monitor	Ventilation ductwork downstream of exhaust fan	GSP H&P Filters	Local and control room alarms and indication. Supply fan trip on high radiation.	Intermediate Bldg. 682'-6" S.W.
D17K710 Fuel Handling Area Ventilation Exhaust Radiation Monitor	Ventilation ductwork upstream of the exhaust filters	GSP HSP PSP	Local and control alarms and indication. Supply fan trip on high radiation. Fuel handling area evac. alarm on high rad.	Intermediate Bldg. 682'-6" N.W.
1D17K760, 2D17K760 Off-Gas Building Ventilation Exhaust Radiation Monitor	Ventilation ductwork upstream of exhaust filter trains	GSP HSP PSP	Local and control room alarms and indication.	Off-Gas Bldg. 635'-0"
1D17K660, 2D17K660 Containment Vessel and Drywell Purge Exhaust Radiation Monitor	Ventilation ductwork outside containment upstream of exhaust filter trains	GSP HSP PSP	Local and control room alarms and indication. Drywell and containment evac. alarm.	Intermediate Bldg. 654'-6"
D17K770 Control Room Airborne Radiation Monitor	Ventilation supply duct downstream of common supply plenum	GSP HSP PSP	Local and control room alarms and indication. High radiation on gas channel shifts ventilation into emergency recirculation mode.	Control Complex 679'-6"

TABLE 12.3-10 (Continued)

Radiation Monitor Subsystem	Sample Point	Instrument Channels(1)	Function of Subsystem	Location
1D17K670, 2D17K670 Drywell Atmospheric Monitor	Drywell, 617'-3" El.	GSP HSP PSP	Local and control room indication and alarms and drywell evacuation alarm.	Fuel Handling Bldg. 620'-6"
1D17K680, 2D17K680 Containment Atmospheric Monitor	Recirculated containment air. 674' El.	GSP HSP PSP	Local and control room indication and alarms and containment and drywell evac. alarm.	Intermediate Bldg. 665'-0"
D17K810 Solid Waste Drumming Area Atmospheric Radiation Monitor	Drumming area (movable)	GSP PSP	Local alarms and indication.	Radwaste Bldg. 623'-6" West
D17K650 Refueling Operation Atmospheric Radiation Monitor	Containment service floor (movable)	GSP HSP PSP	Local alarms and indication and containment and drywell evac. alarm.	Containment Service Floor 689'-6"
1D17K800, 2D17K800 Heater Bay Atmospheric Radiation Monitor	Heater Bay 620'-6" El. (movable)	GSP H&P Filters	Local alarms and indication.	Heater Bay 620'-6"
1D17K790, 2D17K790 Turbine Building Atmospheric Radiation Monitor	Turbine Room - West 647'-6" El. (movable)	GSP H&P Filters	Local alarms and indication.	Turbine Bldg. 647'-6"

Note: Tag Numbers prefixed by 1D17 are components associated with Unit No. 1.
Tag Numbers prefixed by 2D17 are components associated with Unit No. 2.
Tag Numbers prefixed by D17 are components associated with the common areas of the plant.

(1) Analog signals are recorded.

GSP = Gas chamber scintillator-photomultiplier
HSP = Halogen cartridge scintillator-photomultiplier
PSP = Particulate filter scintillator-photomultiplier
H = Halogen
P = Particulate

TABLE 12.3-11

ISOKINETIC PROBES

UNITS 1 & 2 Ventl. System	(1) Monitor No.	(1) Isokinetic Probe No.	Duct (inches)	Design Flow CFM
M14 Containment Vessel and Drywell Purge	1D17K660	1D17N661A 1D17N661B	48 x 48 48 x 48	5,000 30,000
M15 Reactor Bldg. Annulus Exhaust Gas Treatment	1D17K690A 1D17K690B	1D17N691A 1D17N691B	14 x 16 14 x 16	400(2) 400(2)
M36 Off-Gas Bldg. Vent. System	1D17K760	1D17N761	30 x 46	15,000
M38 Auxiliary Bldg. Vent. System	1D17K700	1D17N701	30 x 90	30,500
COMMON TO BOTH UNITS				
M25 Control Room HVAC and Emerg. Recirc. System	D17K770	D17N771	38 x 32	11,540
M31 Radwaste Bldg. Vent. System	D17K720	D17K721A D17K721B	32 x 90 32 x 90	30,000 30,000
M33 Intermediate Bldg. Vent. System	D17K730	D17N731	46 x 46	27,400
M40 Fuel Handling Area Vent. System	D17K710	D17N711	40 x 60	30,000

NOTES:

1. Unit 1 has 1 preceeding the number, i.e., 1D17K---.
   Unit 2 has 2 preceeding the number.

2. With no recycle to the annulus space, 2,000 CFM is possible.

TABLE 12.3-12

REACTOR BUILDING SUBCOMPARTMENT VENTILATION DATA
M14 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Reactor Operation:	5,000	-
642' level, RWCU F/D backwash rec. tank area (radiation Zone 5 area)	250	0.05
652' level north, RWCU HX (radiation Zone 5 area)	1,700	0.34
654' level east RWC4 Valve Area (radiation Zone 5 area)	450	0.09
664' level, RWCU F/D valve & holding pump room (radiation Zone 5 area)	450	0.09
664' level, RWCU F/D room (radiation Zone 5 area)	250 (each)	0.05
Reactor Shutdown Mode of Operation:	30,000	-
Drywell Area	20,000	0.660
Containment Free Space	5,000	0.167
642' level, RWCU F/D backwash rec. tank area (radiation Zone 5 area)	250	0.0083
652' level north, RWCU HX (radiation Zone 5 area)	1,700	0.0567
654' level east RWCU F/D valve nest area (radiation Zone 5 area)	450	0.015
664' level, RWCU F/D valve & holding pump room (radiation Zone 5 area)	450	0.015
664' level, RWCU F/D room (radiation Zone 5 area)	250 (each)	0.0083

TABLE 12.3-13

RADWASTE BUILDING SUBCOMPARTMENT VENTILATION DATA
M31 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Operation	30,000	-
574'-0" Level:		
Floor drain collection pump room A and B	400 (each)	0.013
Fuel pool sludge decant pump room	200	0.0066
Equipment drain sump pump	200	0.0066
RWCU sludge decant pump room A	100	0.0033
RWCU sludge decant pump room B	200	0.0066
Floor drain sump room	200	0.0066
Condensate sludge decant pump room	400	0.013
Waste sample pump room	600	0.02
Floor drain sample pump room	600	0.02
Chemical waste pump room A and B	600 (each)	0.02
Corridor	1,700	0.0567
Waste collector transfer pump room A and B	400 (each)	0.03
Radiation Zone 5 Areas:		
Fuel pool sludge discharge mixing pump area A and B	400 (each)	0.013
RWCU sludge discharge mixing pump area A and B	400 (each)	0.013
Condensate sludge discharge mix pump area A and B	400 (each)	0.013
Fuel pool F/D backwash settling tank room area A and B	400 (each)	0.013
RWCU F/D backwash settling tank A and B	400 (each)	0.013
Condensate filter backwash settling tank A and B	400 (each)	0.013
602' Level:		
Chemical waste distillate tank area	900	0.03
Detergent drain tank & pump area	300	0.01
Corridor	1,800	0.06

TABLE 12.3-13 (Continued)

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Radiation Zone 5 Areas:		
Spent resin pump area A and B	400 (each)	0.013
Concentrated waste transfer pump area A and B	400 (each)	0.013
Floor drain collection tank area A and B	600 (each)	0.02
Waste collection tank area A and B	700 (each)	0.023
Spent resin tank area A and B	400 (each)	0.013
Concentrated waste tank area A and B	400 (each)	0.013
Chemical waste tank area A and B	700 (each)	0.023
Floor drain sample tank area A	300	0.01
Floor drain sample tank area B	600	0.02
Waste sample tank area A	600	0.02
Waste sample tank area B	300	0.01
623' Level:		
Process sample room	1,900	0.063
Radwaste control panel area	Air Supplied Only	-
Empty drum storage area	1,000	0.033
Corridor	5,400	0.180
Loading and drum storage area	3,600	0.12
Reverse osmosis unit room	400	0.013
Chemical treatment room	300	0.01
Ventilation supply equipment area	350	0.01
Ventilation exhaust equipment room	1,200	0.04
Radiation Zone 5 Area:		
Chemical waste evaporator rooms A and B	900 (each)	0.0333
Waste demineralizer area	300	0.01
Floor drain demineralizer area	300	0.01
Waste cement mixing pump room A and B	500 (each)	0.0167
Waste mixing dewatering tank room A and B	500 (each)	0.0167
646' Level:		
Filter, precoat pump and tank room	2,300	0.0767
Waste collector filtrate pump room	300	0.01
Waste collector filtrate room	850	0.0283
Floor drain filtrate pump room	300	0.01
Floor drain filtrate room	850	0.0283
616' Level:		
Full drum storage area	3,200	0.10
Decontamination area	1,600	0.05

TABLE 12.3-14

AUXILIARY BUILDING SUBCOMPARTMENT VENTILATION DATA
M38 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normai Operation	30,500	-
574' level - corridor	2,000	0.0656
Radiation Zone 5 Areas		
568' level: LPCS pump room	2,500	0.081
RHR-A pump room	6,000	0.196
RCIC pump room	1,500	0.049
RHR-C pump room	1,500	0.049
RHR-B pump room	6,000	0.196
HPCS pump room	2,500	0.081
599' level - corridor and containment vessel/turbine building water chiller area	11,000	0.361
599' level: RWCU pump rooms A and B	1,000 (each)	0.033
620' level - northwest corridor	3,500	0.145
Northeast corridor and		
ventilation supply equipment area	8,500	0.278
- pipe chase access room	1,000	0.033

TABLE 12.3-15

INTERMEDIATE BUILDING SUBCOMPARTMENT VENTILATION DATA
M33 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Operation	27,400	-
574' level general area	4,500	0.164
fuel pool cooling & cleanup circulating pump	400	0.014
fuel pool cooling & cleanup F/D transfer pump room	200	0.007
fuel pool cooling & cleanup F/D backwash Rec. tank room	300	0.011
intermediate building floor & equipment drain sump pump room	300	0.011
599' level general area	4,800	0.175
fuel pool cooling & clean-up heat exchanger	400	0.014
fuel pool cooling & clean-up F/D room A & B	200	0.007
fuel pool cooling & clean-up F/D room C & D	200	0.007
control complex controlled access entry	100	0.004
electrical equipment room	1,200	0.044
hot I&C repair room	600	0.022
620' level	8,300	0.303
annulus exhaust gas treatment rooms (4)	500 (each)	0.018
654' level general area & recombiner area	4,800	0.175
containment vessel & drywell purge	2 @ 1,000	0.036
exhaust plenum area (4)	2 @ 1,200	0.044
in-service inspection room	Air supplied only	-
682' level general area	6,000	0.218
fuel handling area exhaust filter rooms (3)	800 (each)	0.029

TABLE 12.3-16

FUEL HANDLING AREA SUBCOMPARTMENT VENTILATION DATA
M40 SYSTEM

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Operation	30,000	-
574' level		
corridor, north and south	1,000 (each)	0.054
Radiation Zone 5 Areas:		
control rod drive pump room north and south	2,500 (each)	0.083
599' level		
corridor, north and south	1,000 (each)	0.054
control rod drive maintenance area	8,100	0.27
Radiation Zone 5 Areas:		
Pool Area:		
(Cask storage pool, spent fuel pool, fuel transfer pool)	15,300	0.51

TABLE 12.3-17

HEATER BAY SUBCOMPARTMENT VENTILATION DATA
M41 SYSTEM

Subcompartment	SUMMER OPERATION[1] Exhaust Flow Rate (CFM)	SUMMER OPERATION[1] Dilution Factor	WINTER OPERATION[2] Exhaust Flow Rate (CFM)	WINTER OPERATION[2] Dilution Factor
Normal Operation	360,000	-	180,000	-
580' level (Hot Water Heating Equipment)	97,000	0.269	48,500	0.269
Rad. Zone 5 areas: DC feedwater heaters and auxiliary boiler evaporator	120,000	0.333	60,000	0.333
600' level (Hallway)	28,500	0.0792	14,250	0.0792
Rad. Zone 5 areas: intermediate feedwater heater, HP feedwater heater	158,500	0.440	79,250	0.440
620' level (Hallway)	21,500	0.0597	750	0.004
FDW Lube Oil Purifier Room	1,500	0.004	750	0.004
Rad. Zone 5 area: Auxiliary Condenser	166,000	0.461	79,600 CFM	0.44
647' level (Hallway)	14,000	0.0389	375	0.002
Rad. Zone 5 areas:				
steam seal evaporator	83,000	0.23	39,800	0.22
feedwater pump area (2)	41,500 (each)	0.11	19,900 (each)	0.11
Feedwater heater area	60,000	0.17	22,400	0.12

NOTES:

1. With louvers open.

2. With louvers closed.

TABLE 12.3-18

TURBINE BUILDING SUBCOMPARTMENT VENTILATION DATA
M35 SYSTEM

Subcompartment	SUMMER OPERATION[1] Exhaust Flow Rate (CFM)	SUMMER OPERATION[1] Dilution Factor	WINTER OPERATION[2] Exhaust Flow Rate (CFM)	WINTER OPERATION[2] Dilution Factor
Normal Operation	360,000	-	180,000	-
Condenser bay area (radiation Zone 5)	96,000	0.27	96,000	0.53
Turbine building (west end)	27,200	0.076	27,200	0.15
Turbine operating floor	180,000	0.50	118,100	0.66
Hot well pump & L.P. heater area	25,500	0.07	25,500	0.14
Condenser vacuum pump area	300	0.001	300	0.002
Sample extraction area	600	0.002	600	0.003
Steam seal exhaust area	24,700	0.07	24,700	0.14

NOTES:

1. With louvers open
2. With louvers closed

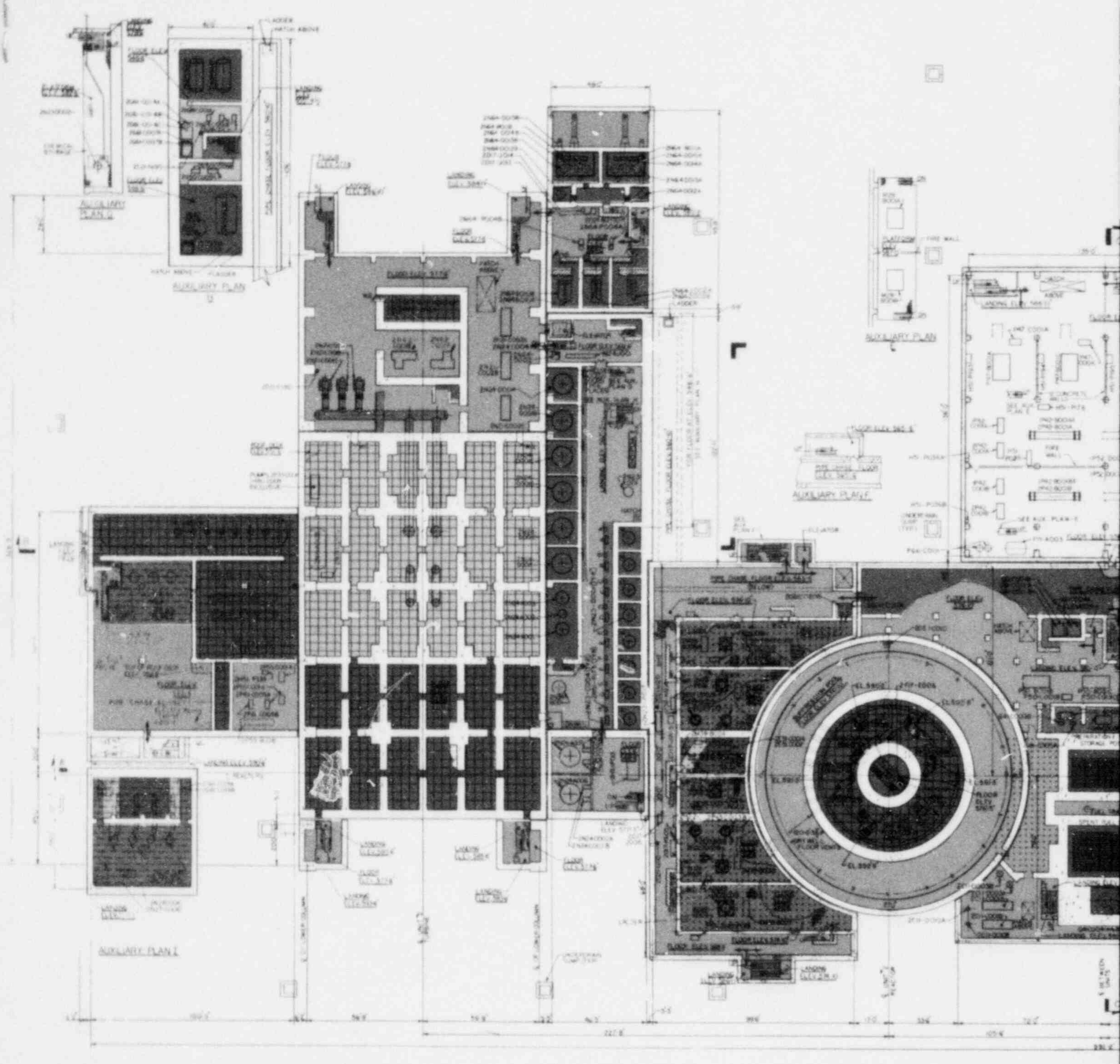
TABLE 12.3-19

OFF-GAS BUILDING VENTILATION EXHAUST SUBCOMPARTMENT VENTILATION DATA
M36 SYSTEM

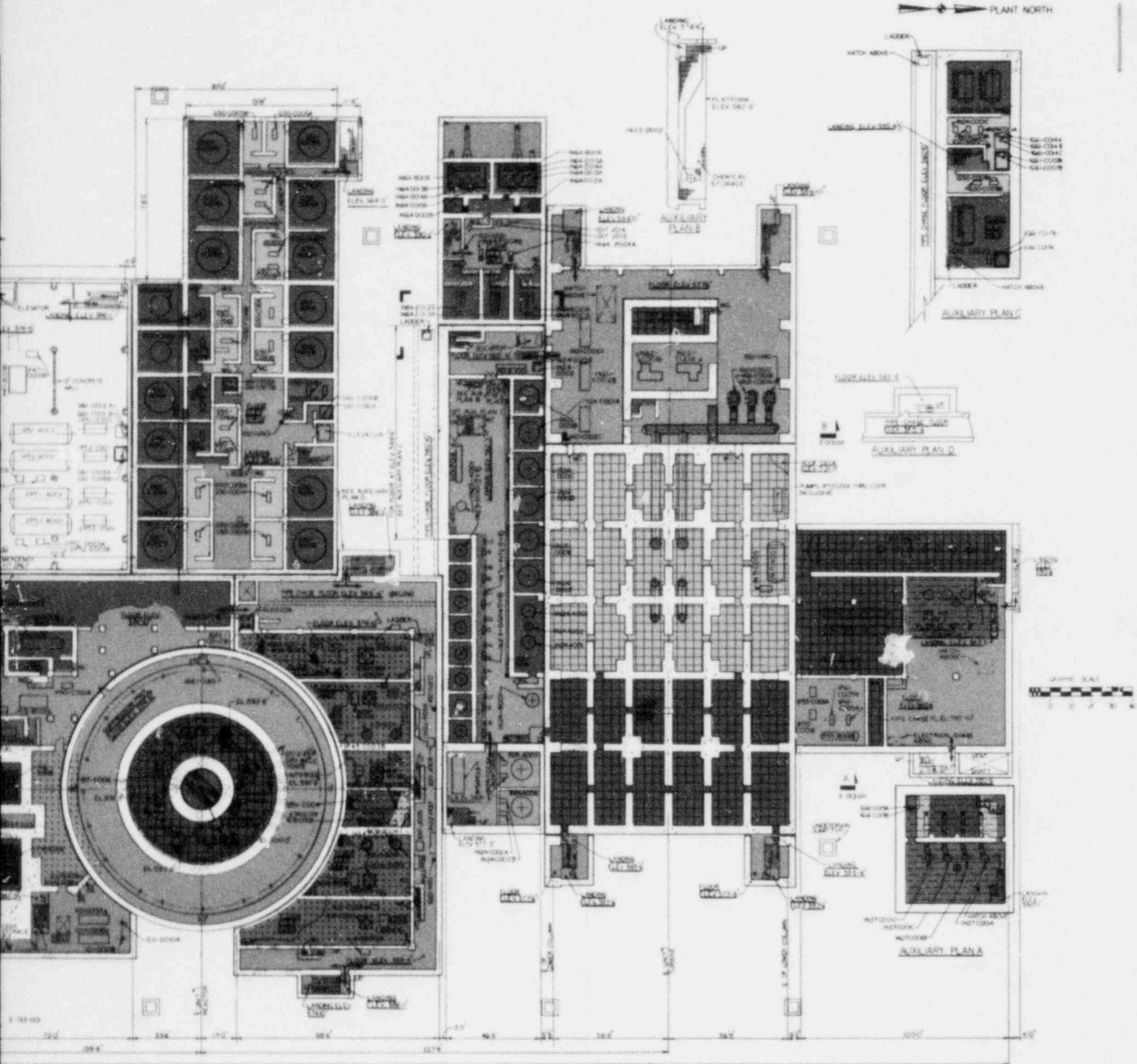
Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Normal Operation	15,000	-
548' level		
Turbine power complex, condensate demineralizer waste transfer pump area	1,300	0.086
Radiation Zone 5 areas:		
Turbine power complex		
a. Condensate demineralizer backwash rec. tank	600	0.04
b. Condensate filter backwash rec. tank area	700	0.046
568' level		
Turbine power complex, corridor	2,100	0.14
Turbine power complex, condensate filter pump area	1,500	0.1
Condensate Demin. Cubicles (6)	500 (total)	0.033
Condensate filter cubicles (8)	500 (total)	0.033
Caustic & acid storage tank room	2,000	0.133
Off-gas exhaust plenum area	600	0.04
Radiation Zone 5 area:		
Turbine power complex, Cation regen. tank area	500	0.033
577' level		
(Radiation Zone 5 area) Turbine building, holdup pipe area	1,000	0.066
584' level		
Off-Gas building, corridor	700	0.046

TABLE 12.3-19 (Continued)

Subcompartment	Exhaust Flow Rate (CFM)	Dilution Factor
Radiation Zone 5 areas:		
Off-Gas cooler condenser room	200	0.013
Off-Gas regenerator room A & B	250 (each)	0.016
Charcoal absorber rooms	1,000 (Total)	0.066
602' level		
Radiation Zone 5 areas:		
Off-Gas building, corridor	1,500	0.1
Desiccant dryer area	450	0.030
After filter and prefilter rooms	200 (each)	0.013
605' level		
Radiation Zone 5 areas:		
Steam jet air ejector room A and B	800 (each)	0.053
Preheater area	800	0.053
620' level		
Off-Gas building, floor area	2,900	0.19
624' level		
Turbine building, lab. hoods	2,400	0.16
Turbine building, hydrogen analyzer area (A and B)	300 (each)	0.02



NOTE: POST-LOCA, CONTAINMENT BUILDING AND RHR AREA IS ZONE 5.

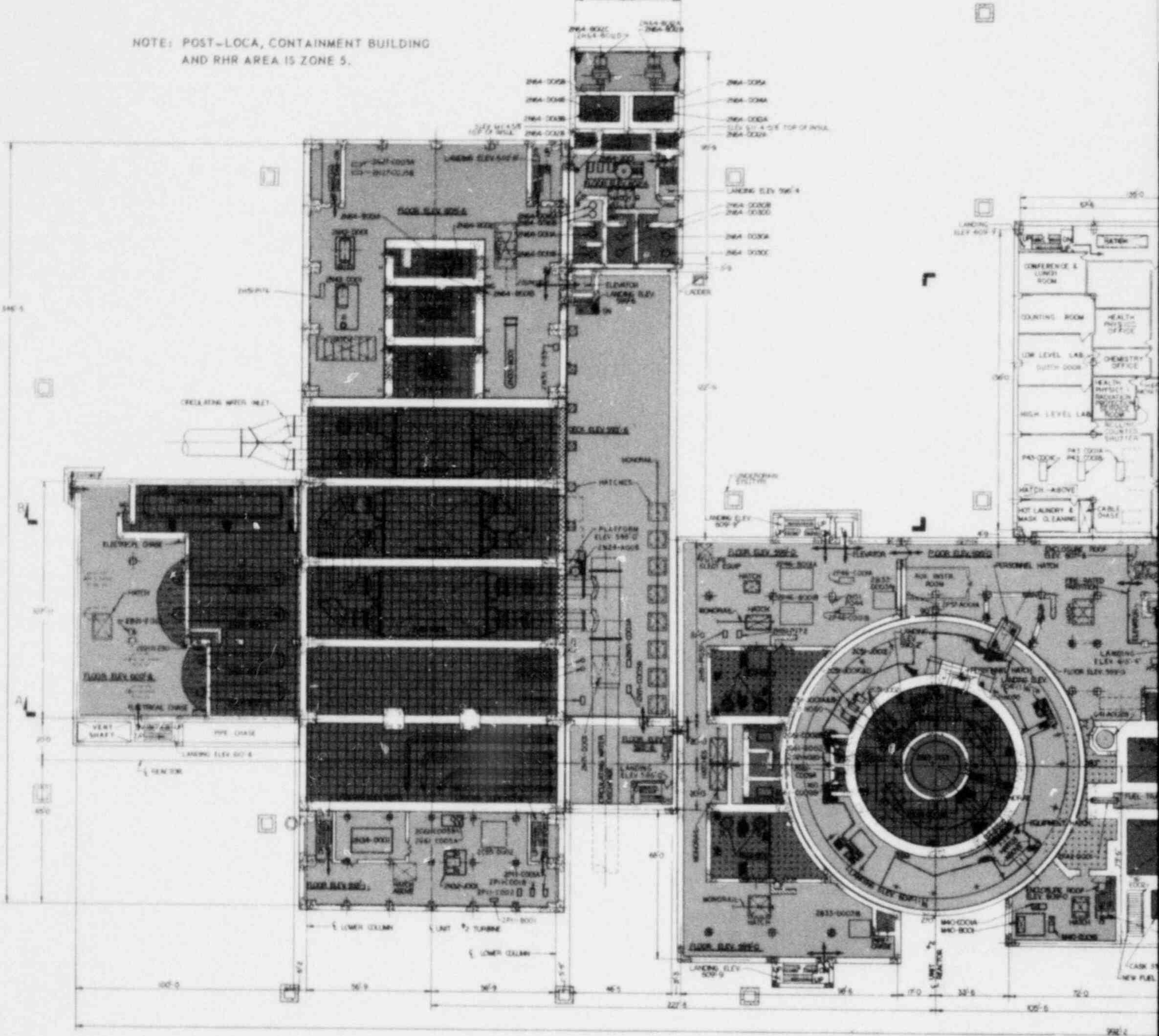


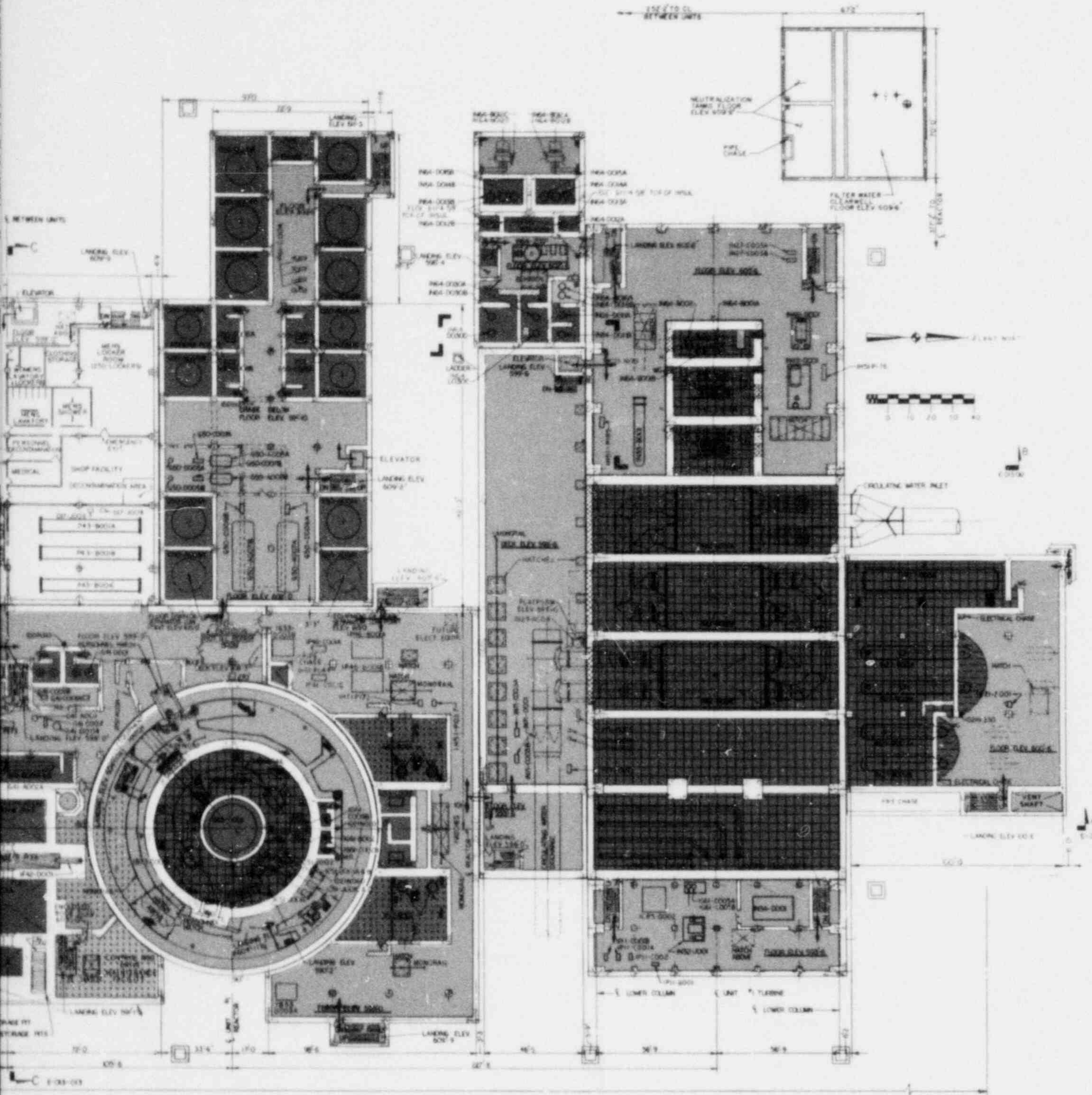
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Zones, Plan A above
Elevs. 568'-6", 574'-10",
577'-6" and 580'-6"

Figure 12.3-1

NOTE: POST-LOCA, CONTAINMENT BUILDING AND RHR AREA IS ZONE 5.

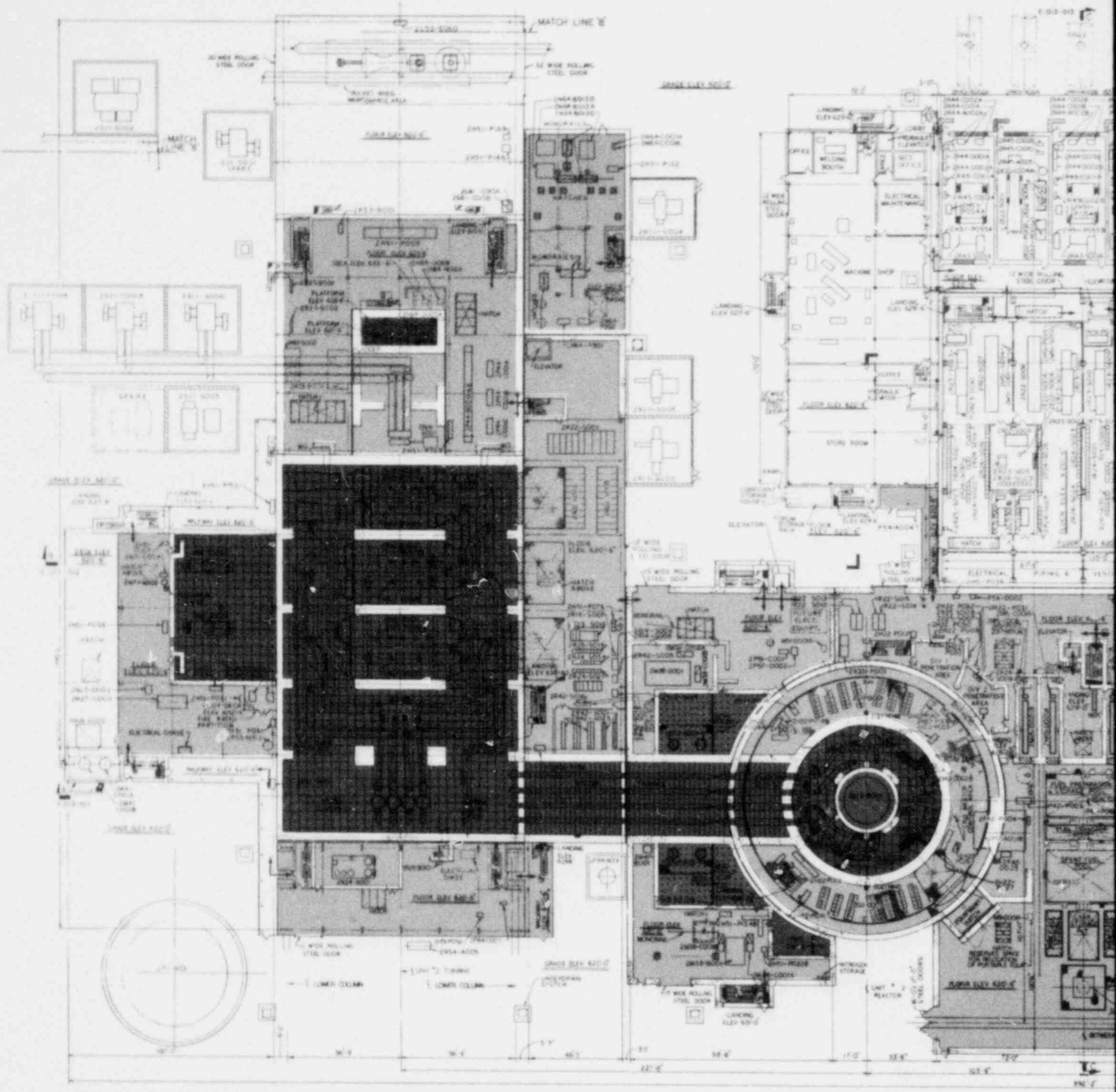




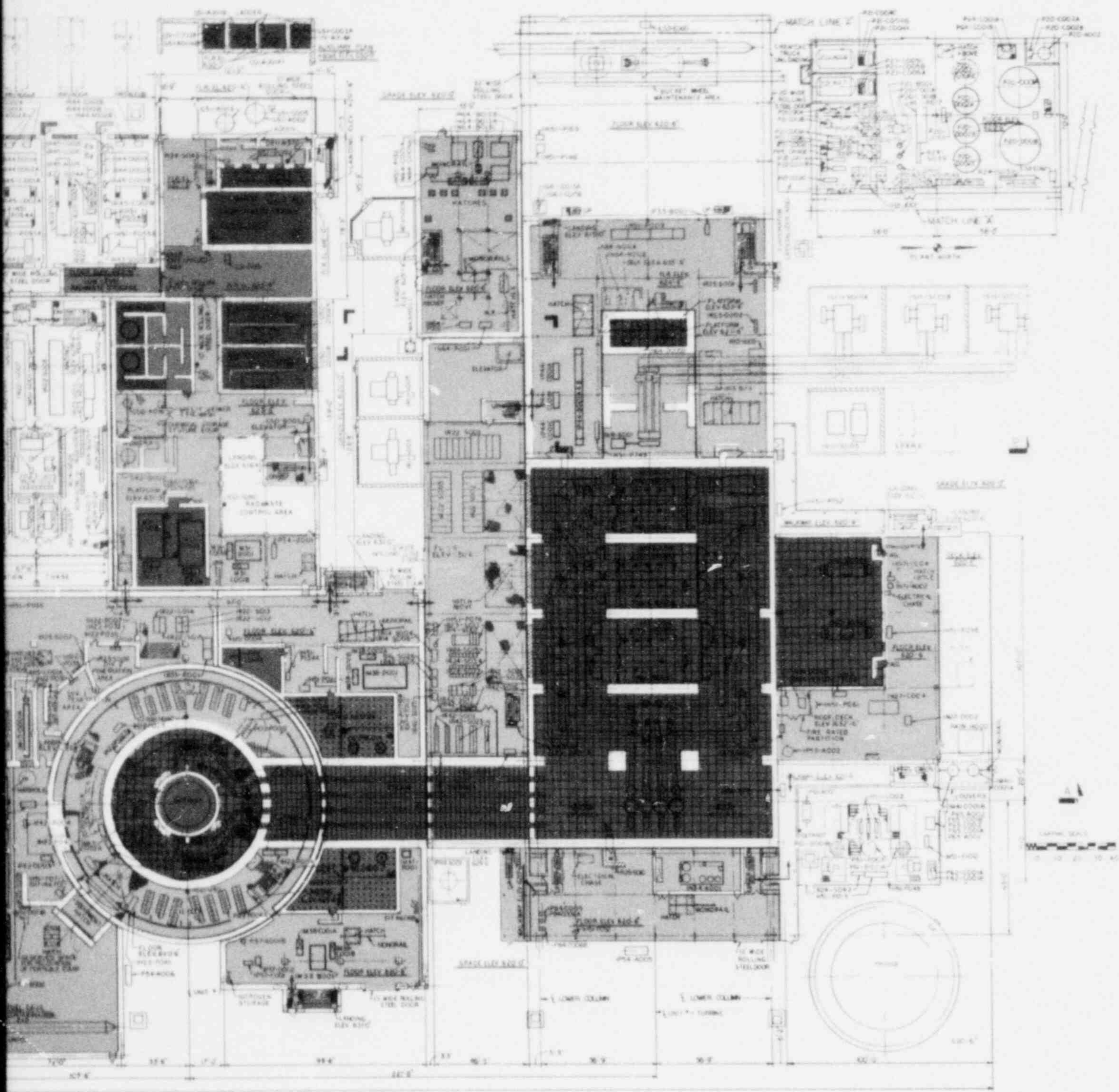
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Radiation Zones, Plan B above Elevs. 593'-6", 599'-0", 600'-6", 602'-6" and 605'-6"

Figure 12.3-2



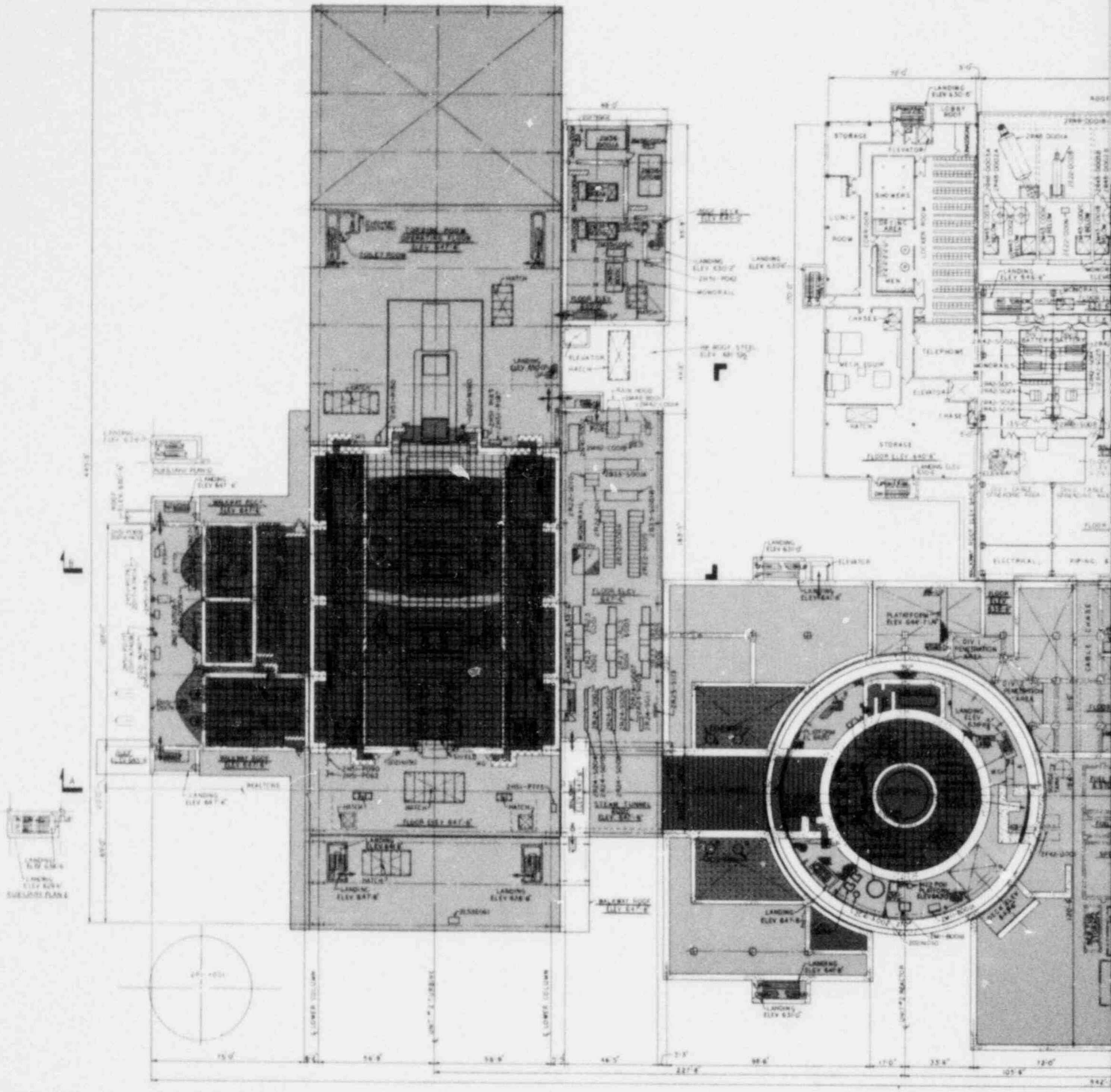
NOTE: POST-LOCA CONTAINMENT, RHR, PENETRATION AREAS IN INTERMEDIATE BUILDING IS ZONE 5.



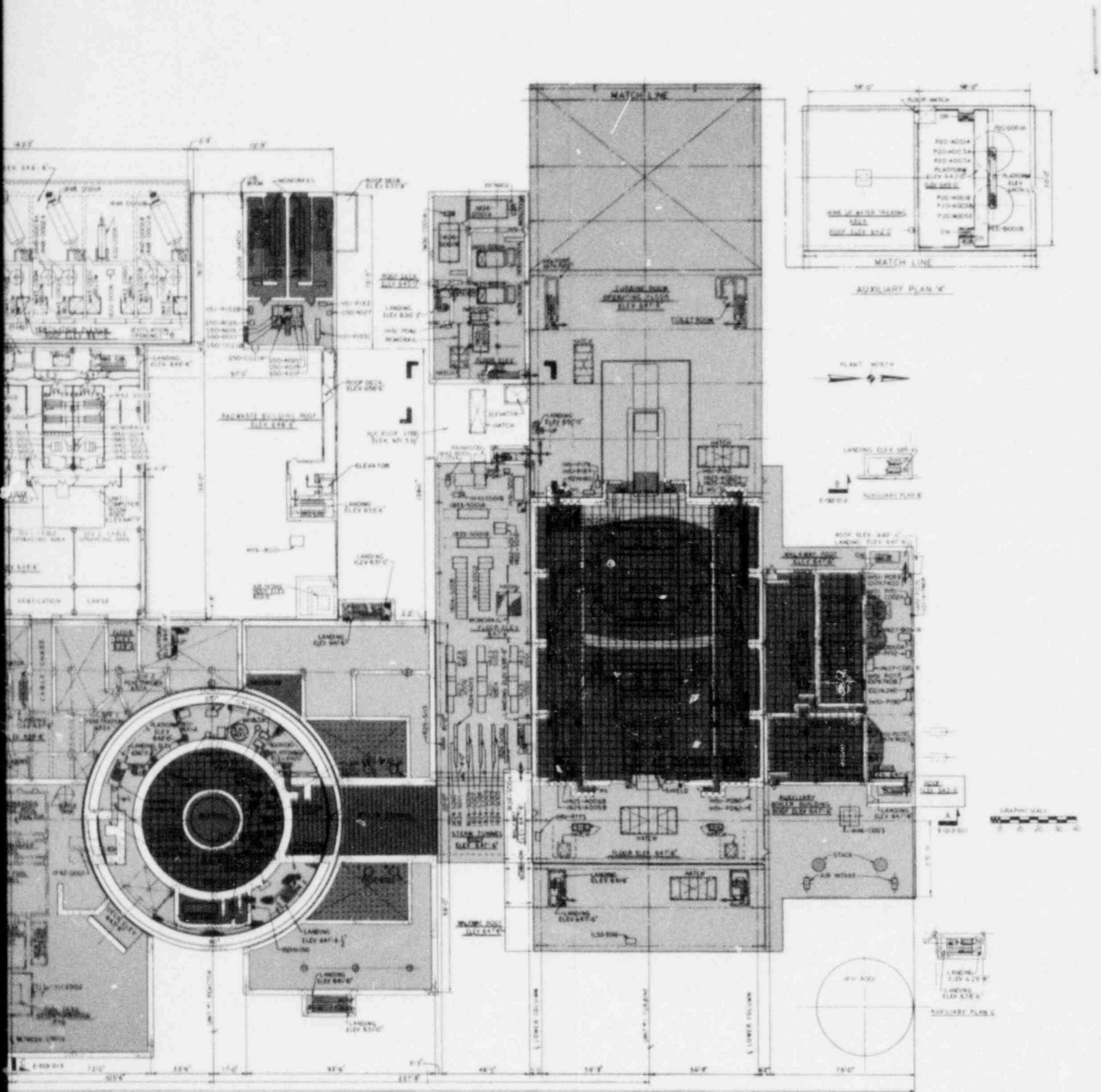
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Zones, Plan C above
Elevs. 620'-6", 623'-6"
and 624'-6"

Figure 12.3-3



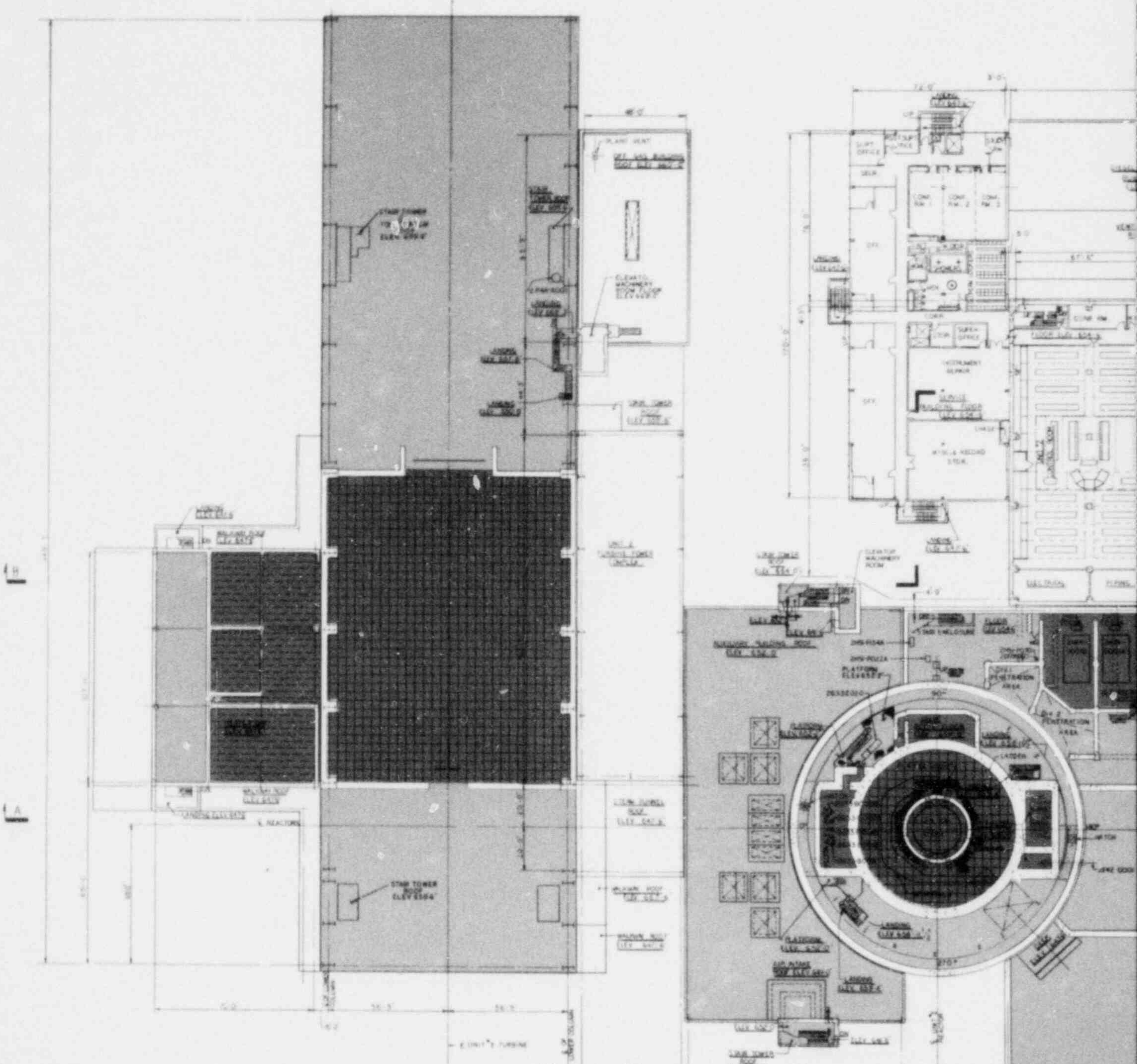
NOTE: POST-LOCA CONTAINMENT BUILDING, RHR
AND PENETRATION AREAS ARE ZONE 5.



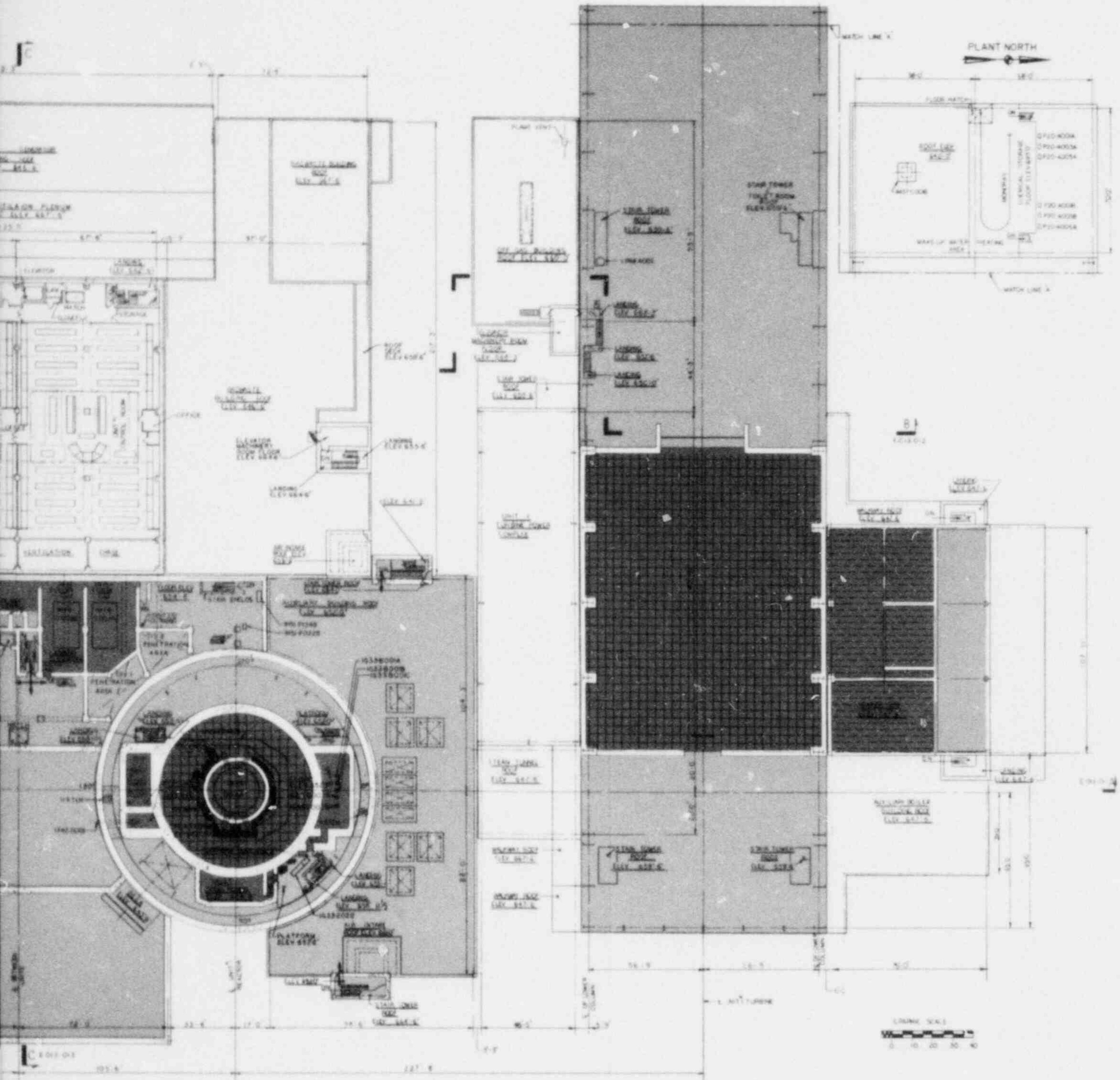
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Zones, Plan D above
Elevs. 538'-6", 642'-0",
and 647'-6"

Figure 12.3-4



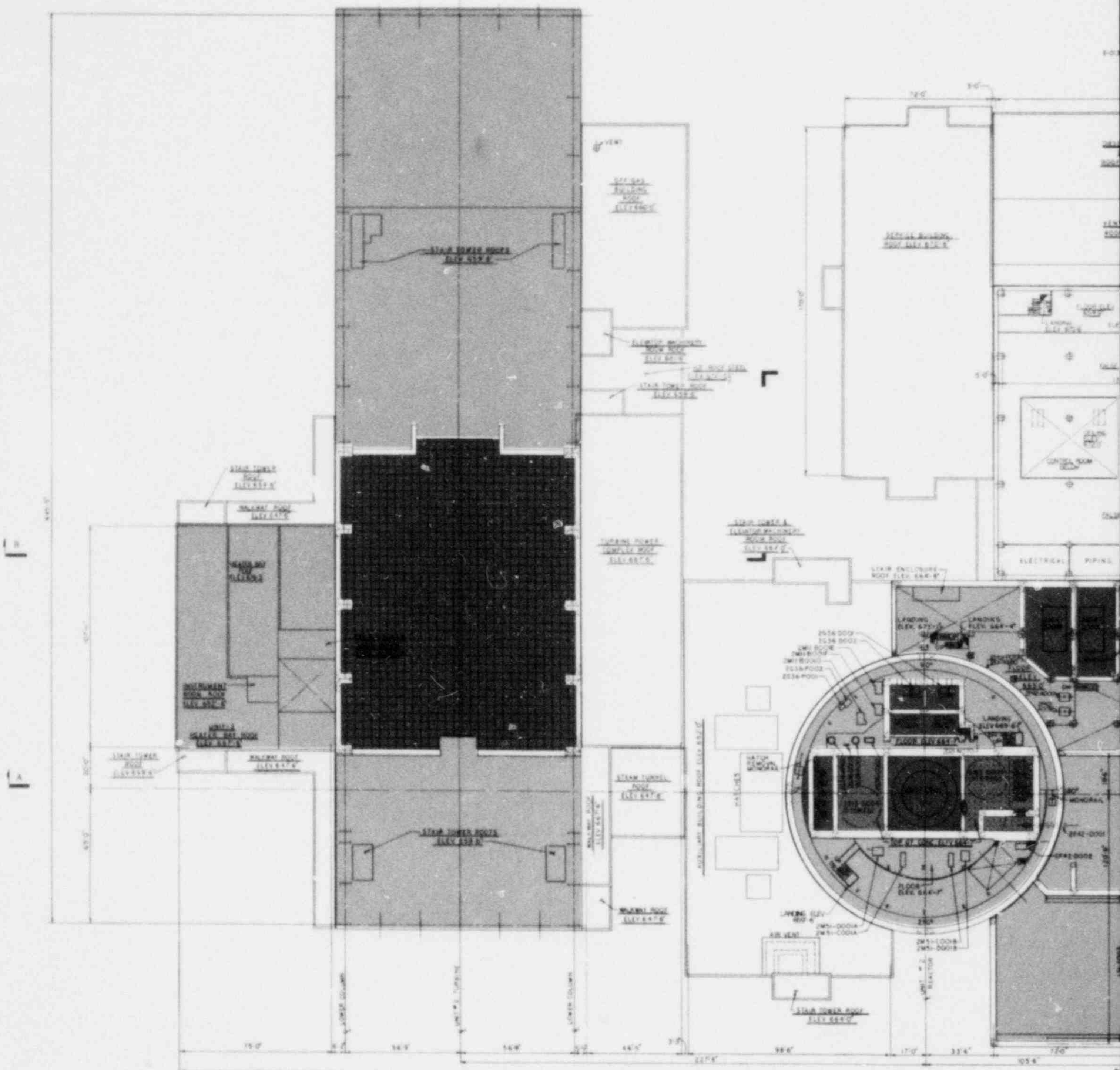
NOTE: POST-LOCA CONTAINMENT BUILDING AND PENETRATION AND HYDROGEN RECOMBINER AREA OF INTERMEDIATE BUILDING IS ZONE 5.



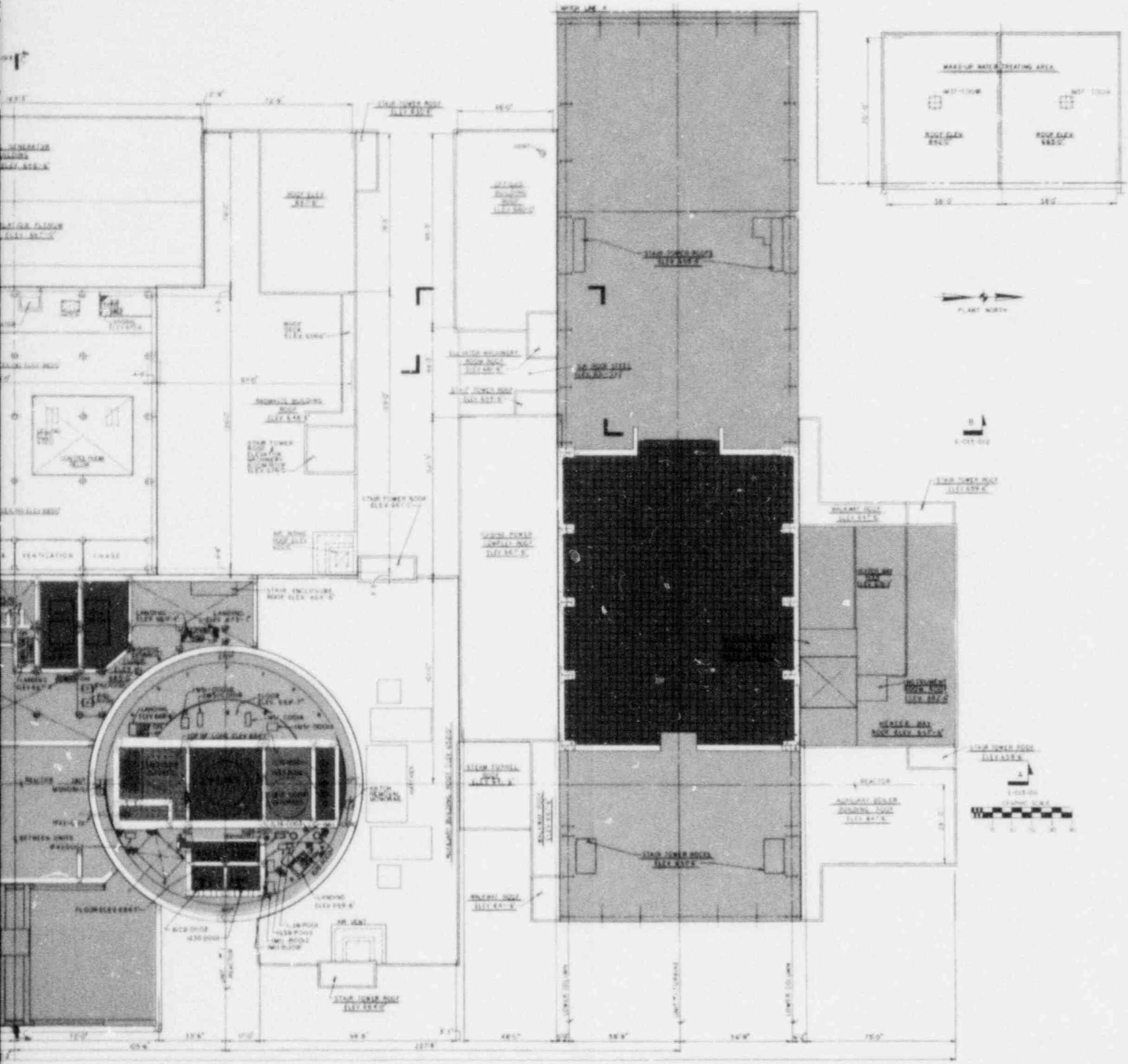
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Zones, Plan E above
Elevs. 652'-0" and 654'-6"

Figure 12.3-5



NOTE: POST-LOCA CONTAINMENT
BUILDING IS ZONE 5.

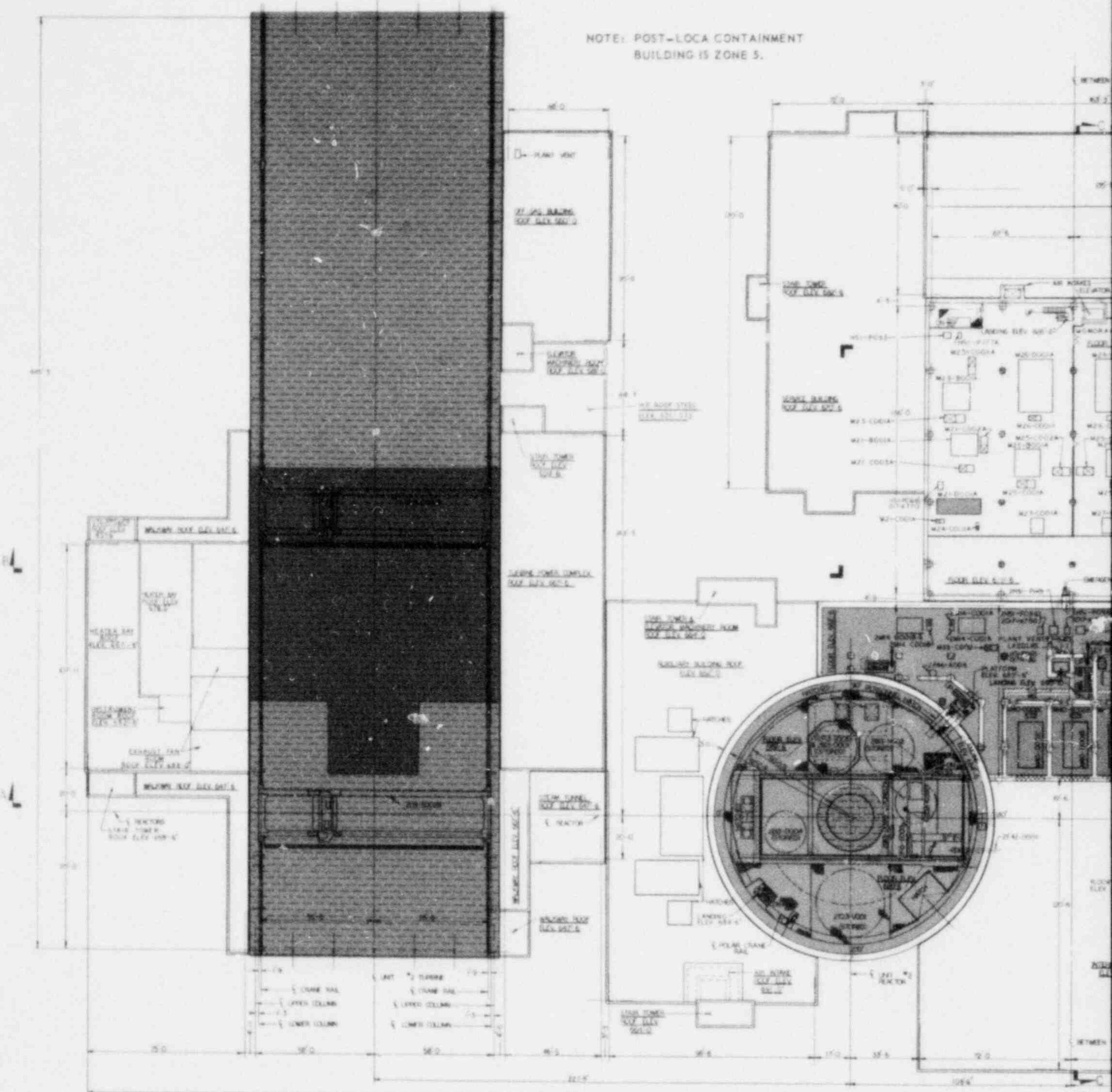


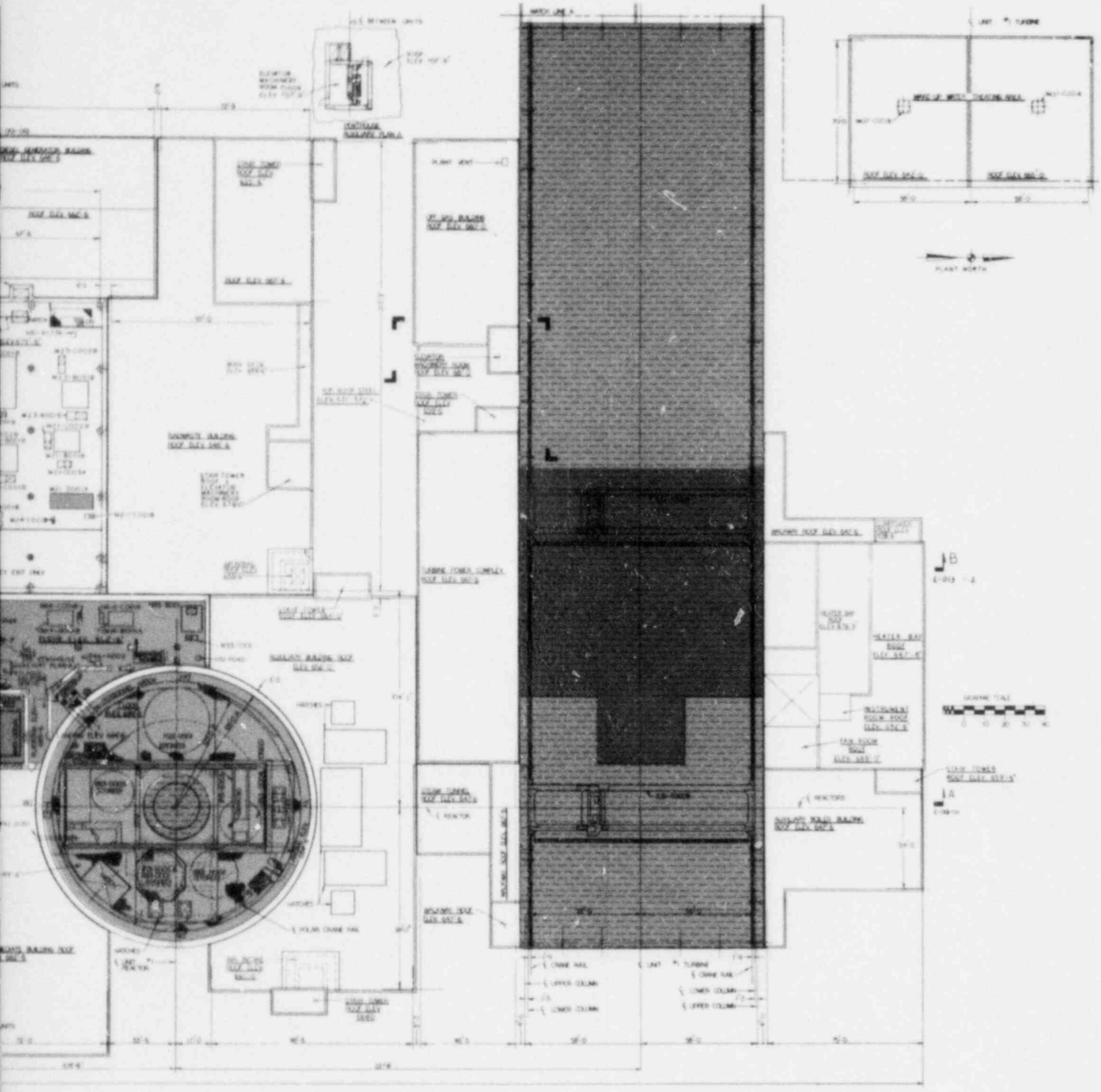
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Zones, Plan F above
Elevs. 664'-7", 665'-0"
and 670'-6"

Figure 12.3-6

NOTE: POST-LOCA CONTAINMENT BUILDING IS ZONE 5.

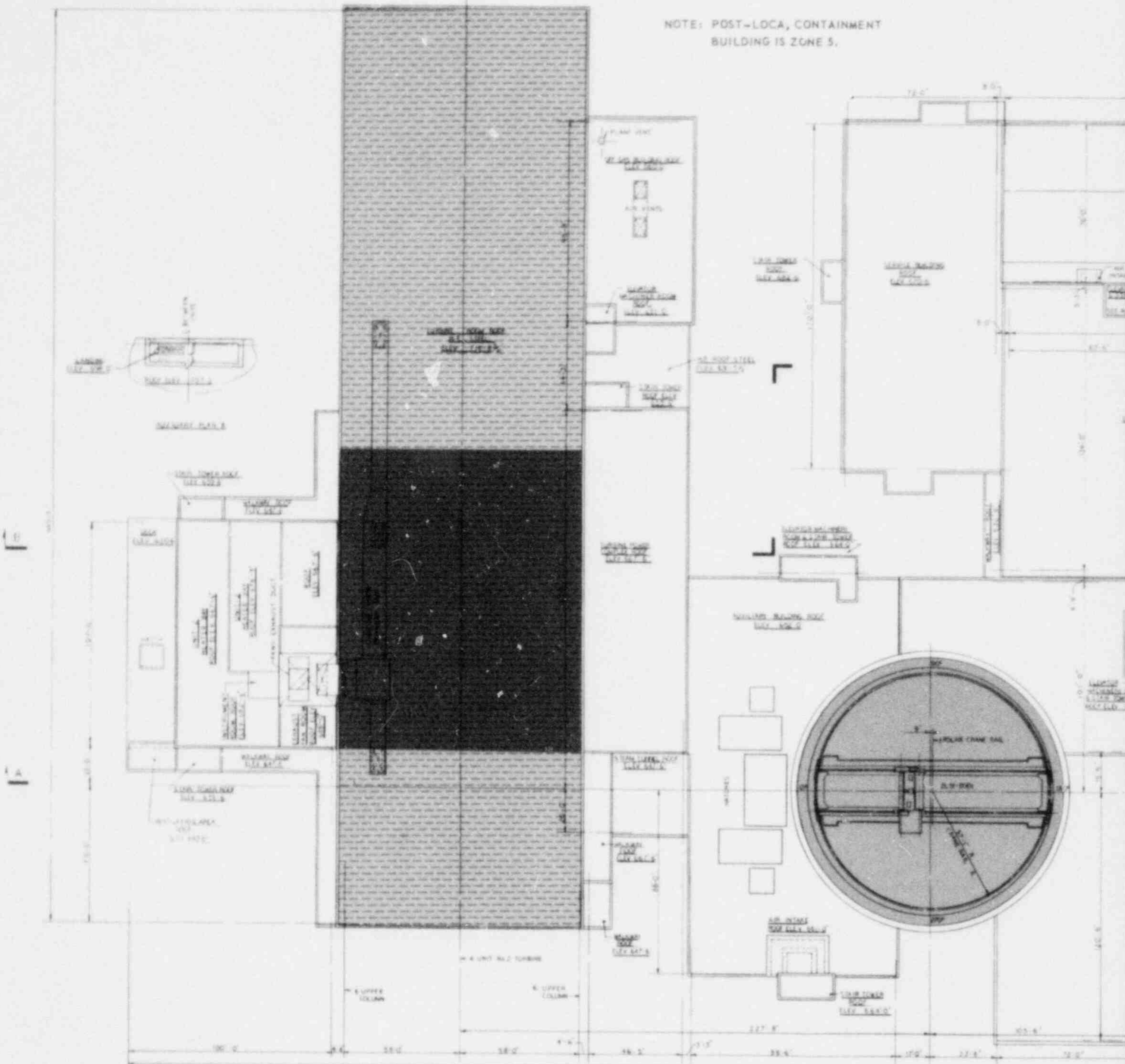


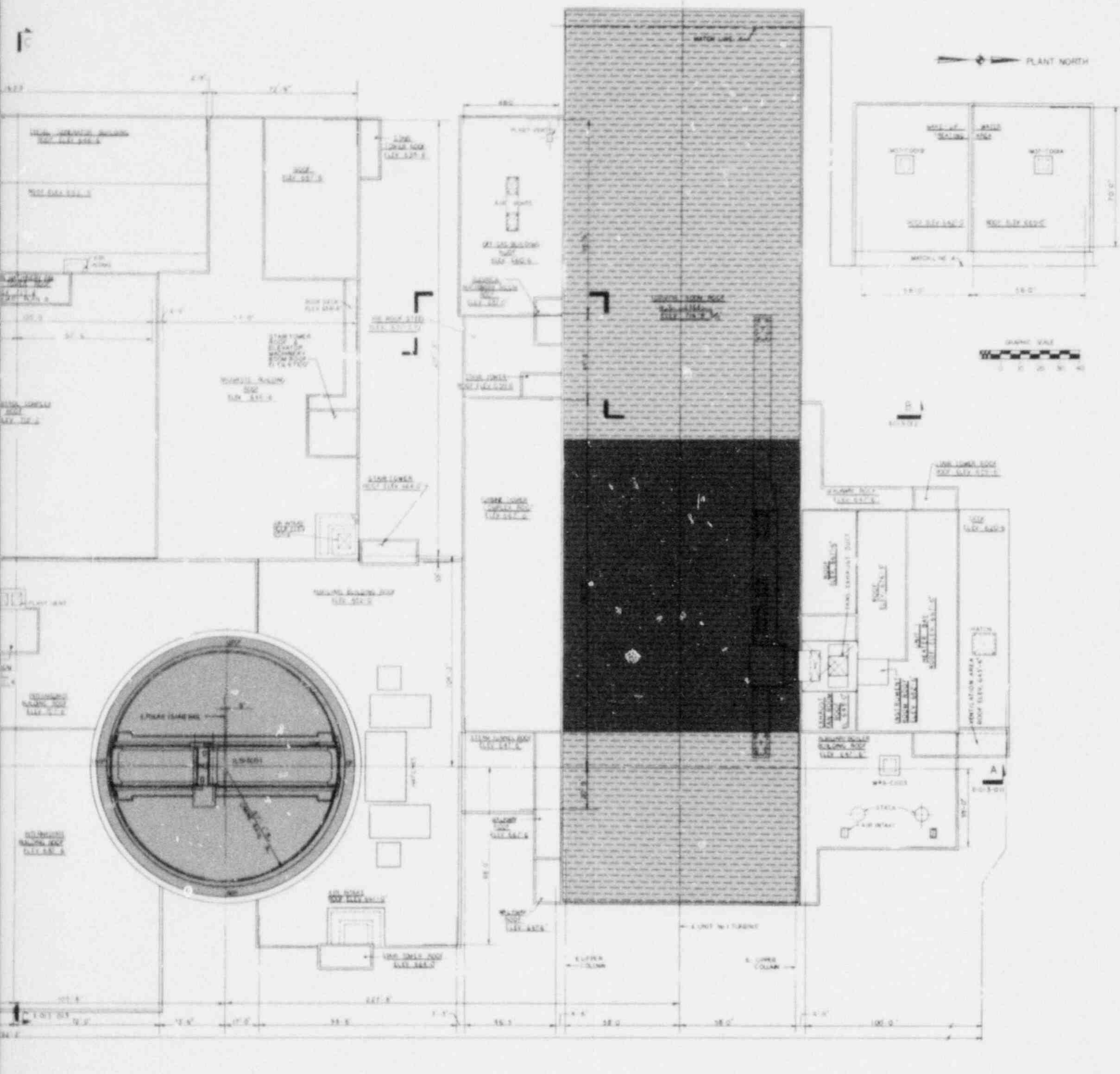


PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Zones, Plan G
above Elev. 689'-6"

Figure 12.3-7

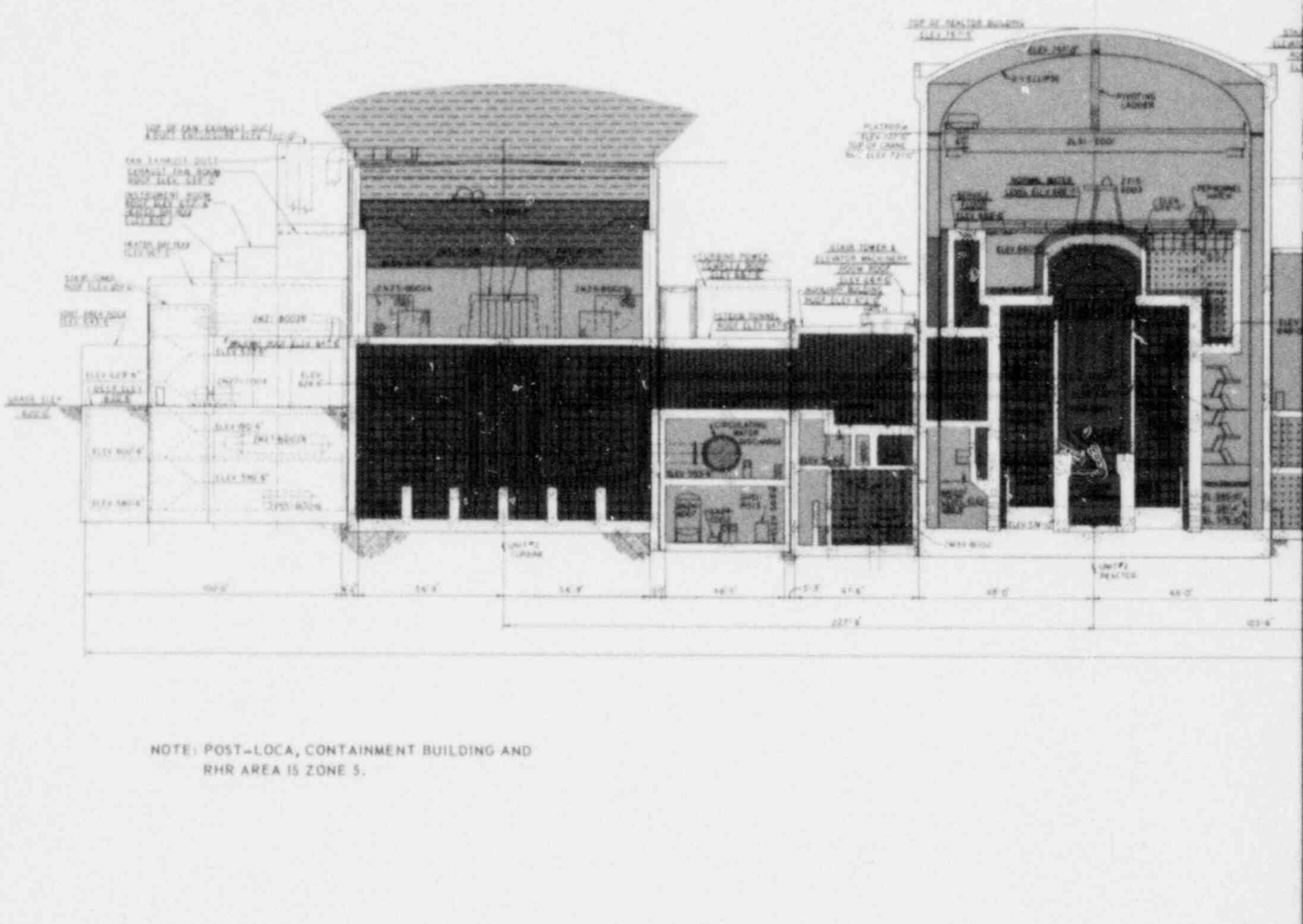

NOTE: POST-LOCA, CONTAINMENT
BUILDING IS ZONE 5.



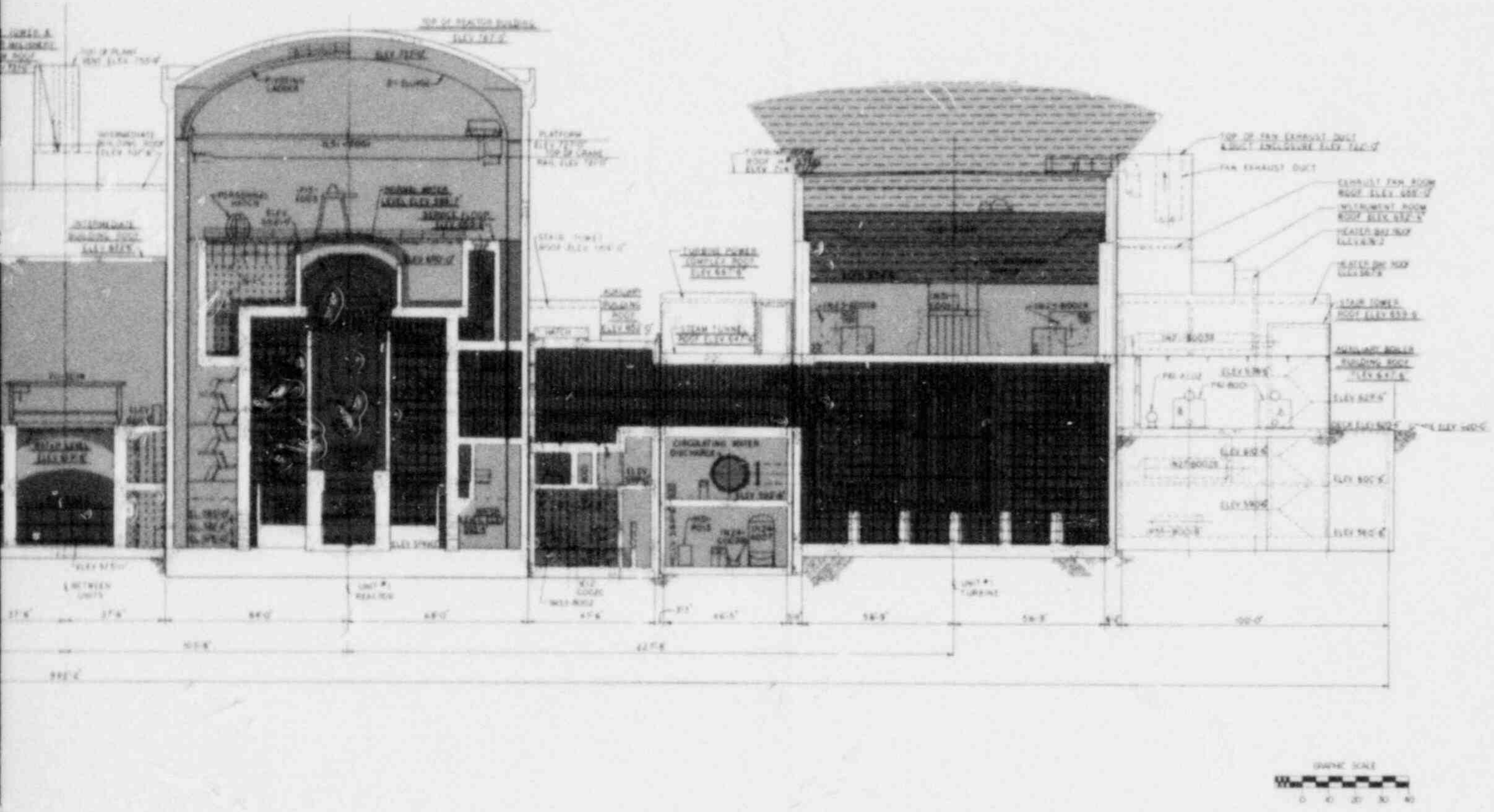
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Zone, Plan H
Roof Plan

Figure 12.3-8



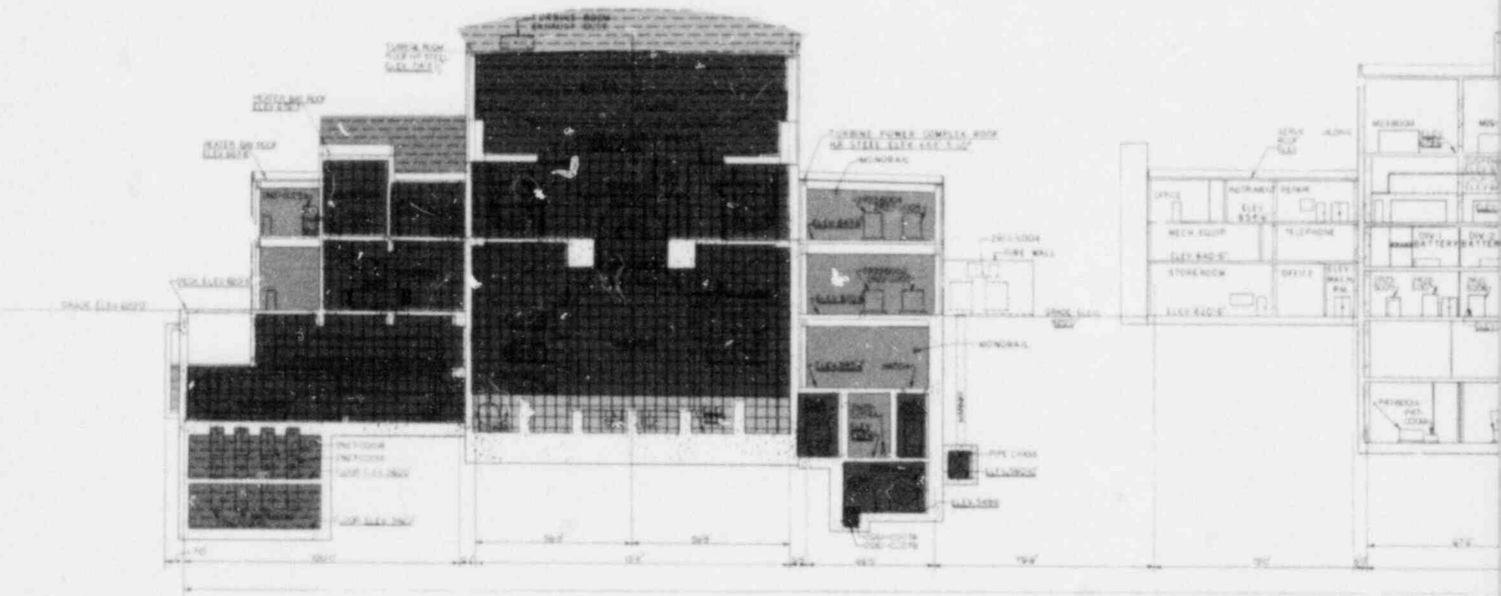
NOTE: POST-LOCA, CONTAINMENT BUILDING AND
RHR AREA IS ZONE 5.

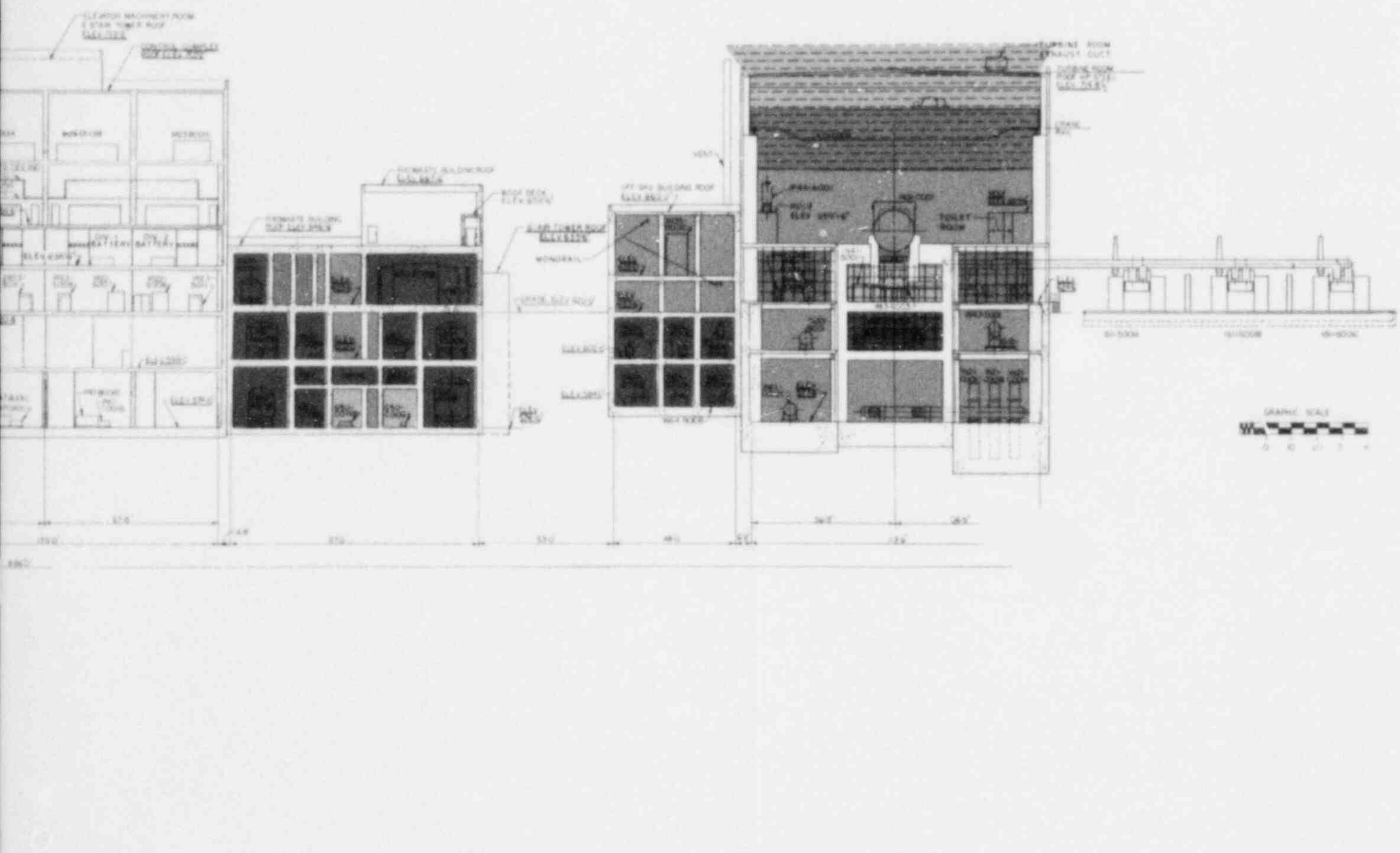


PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Zone, Section A-A

Figure 12.3-9


TURBINE POWER COMPLEX ROOF
MONORAIL
FIRE WALL
OFFICE
INSTRUMENT REPAIR
MECH. EQUIP
TELEPHONE
STOREROOM
OFFICE
MONORAIL

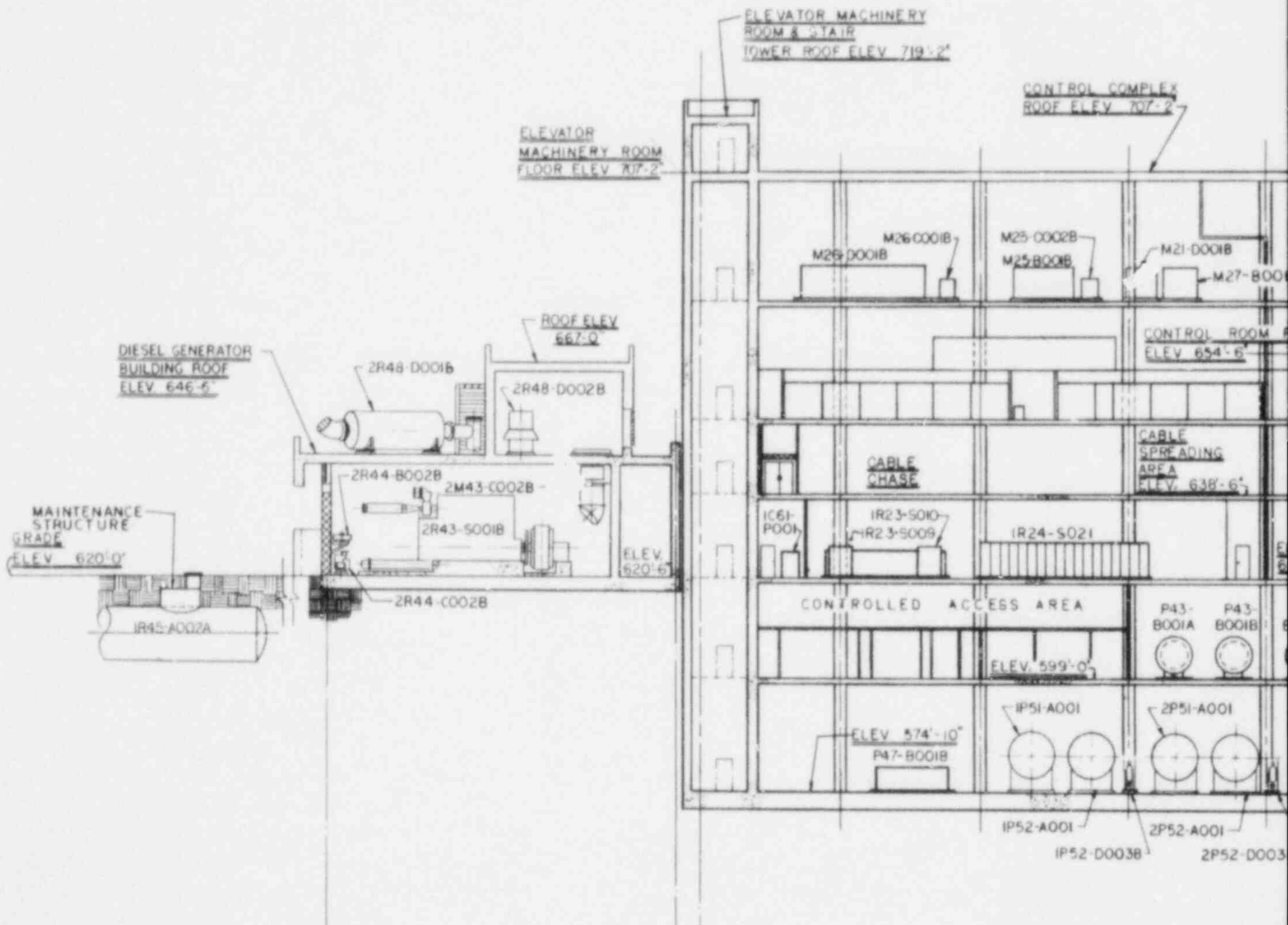


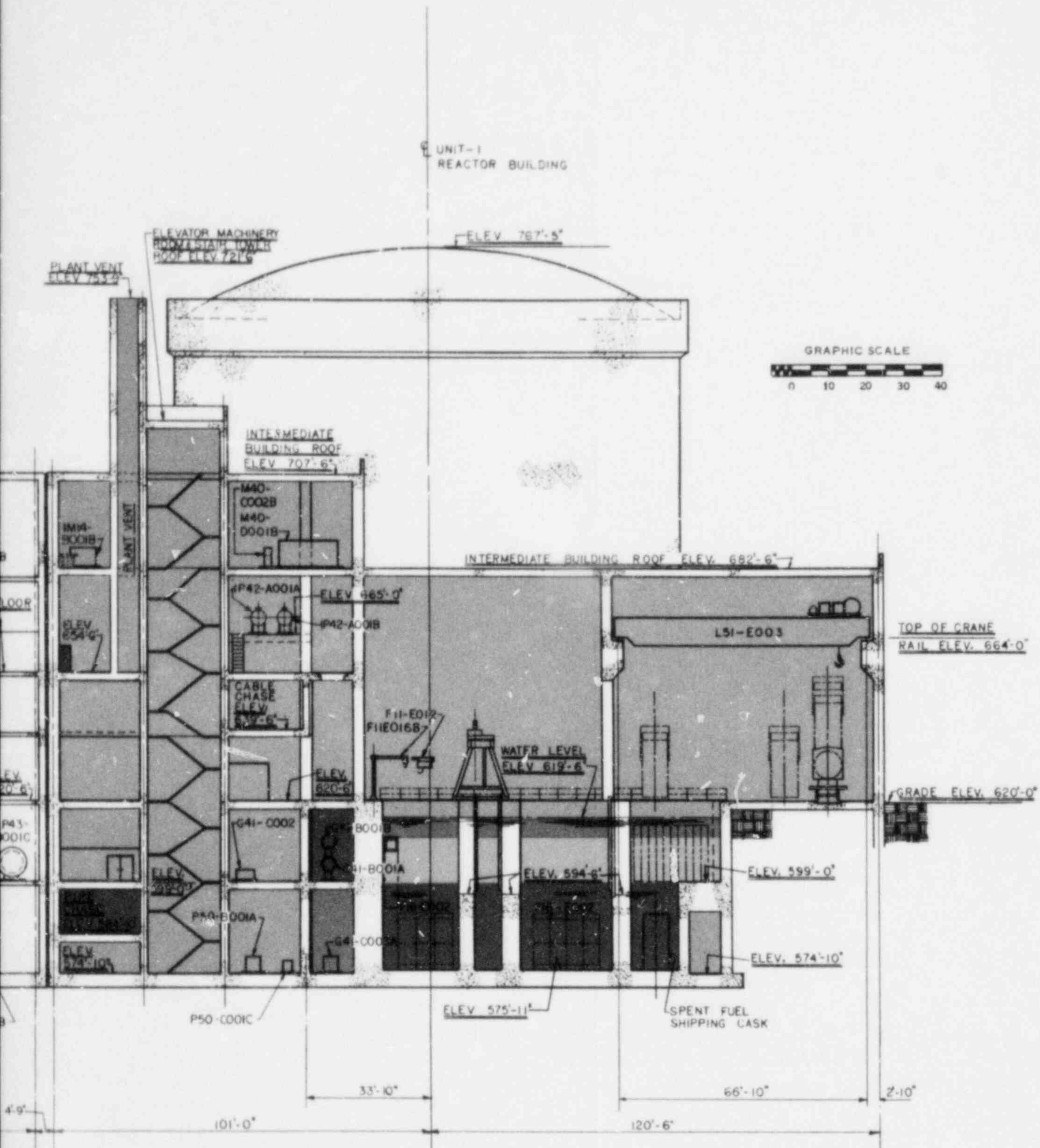
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Radiation Zone, Section B-B

Figure 12.3-10

NOTE: POST-LOCA, PENETRATION AND HYDROGEN RECOMBINER AREA OF INTERMEDIATE BUILDING IS ZONE 5.

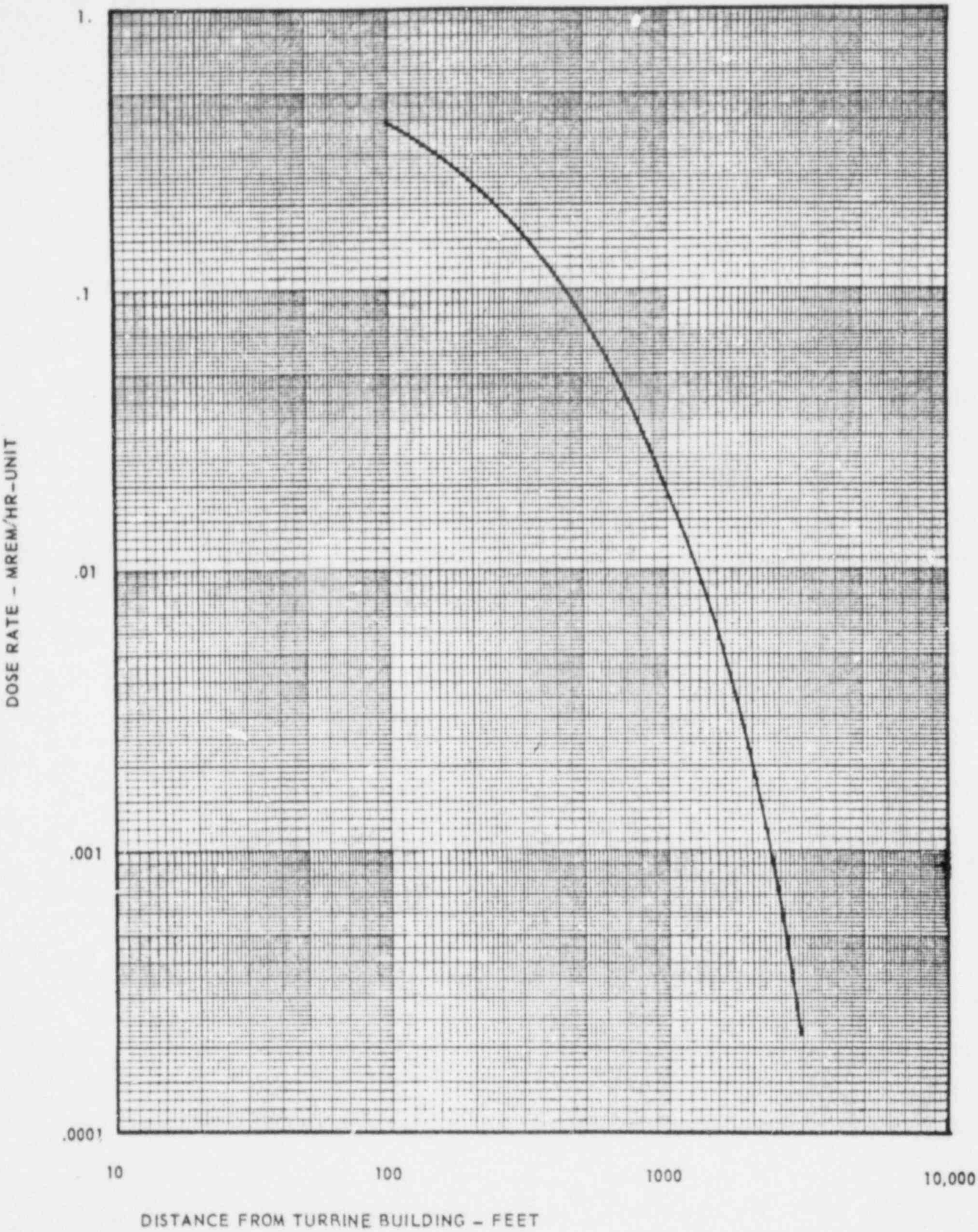




PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Radiation Zone, Section C-C

Figure 12.3-11

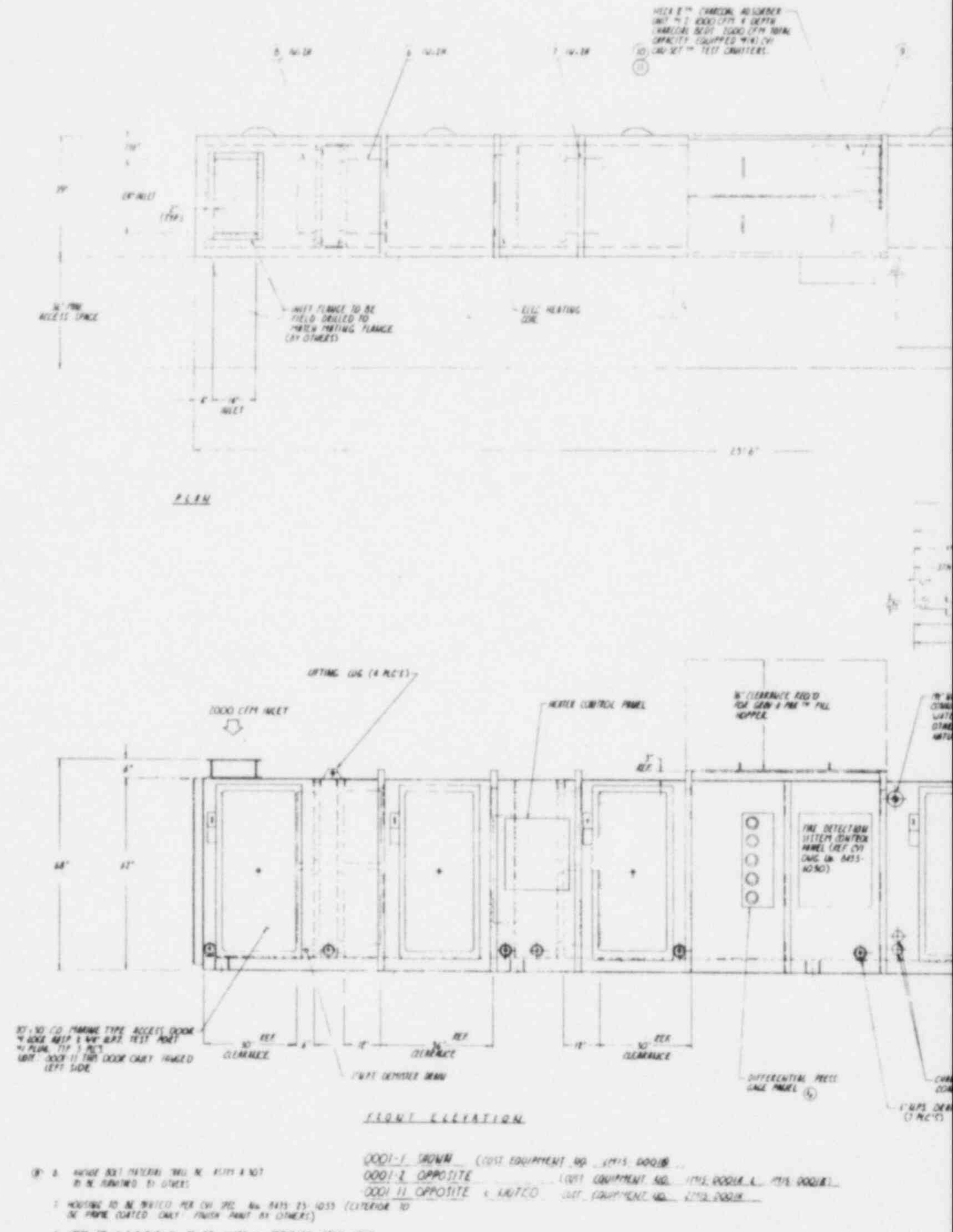


PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Skyshine Dose Rate (Mr/Hr)
vs. Distance (Ft.)

Figure 12.3-12

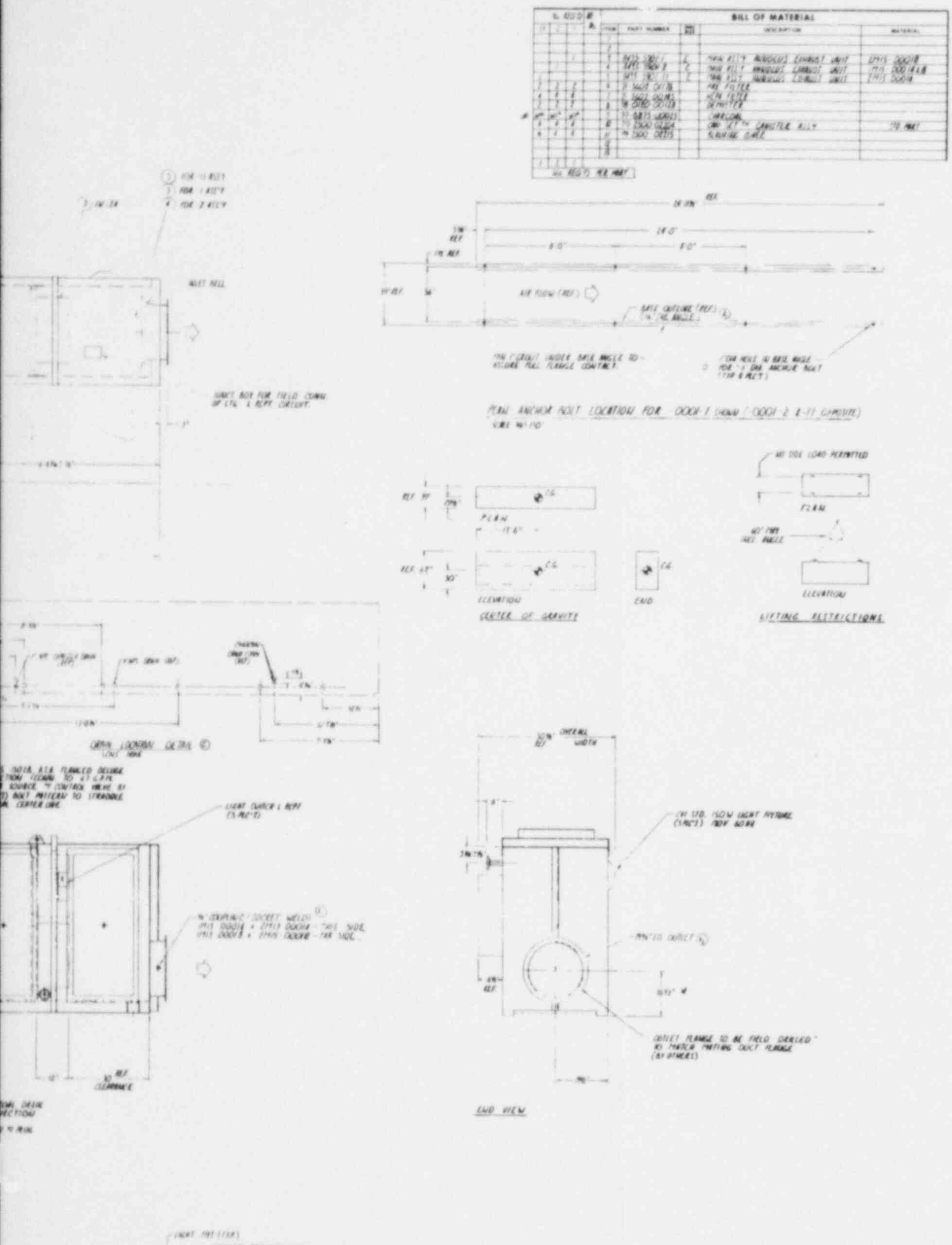
REVISIONS



0001-1 SHOWN (CUST EQUIPMENT NO. 1M15 D001B)
0001-2 OPPOSITE (CUST EQUIPMENT NO. 1M15 D001A & 1M15 D001B)
0001-11 OPPOSITE & KAITCO CUST EQUIPMENT NO. 2M15 D001A

8. ANCHOR BOLT MATERIAL THRU BE ... & NOT TO BE FURNISHED BY OTHERS
7. HOUSING TO BE PAINTED PER CVI SPEC. No. 8455-25-1033 (EXTERIOR TO BE PRIME COATED ONLY. FINISH PAINT BY OTHERS)
6. ITEM NO. 4, 7, 8, 9, 10 & 11 TO BE SHIPPED SEPERATE FROM UNIT
5. TOTAL OPERATING WT. 9450 LBS(E). SHIPPING WT. 8200 LBS(EST)
4. SEE CVI O&M MANUAL FOR INSTALLATION OF FILTERS & COMPONENTS
3. HOUSING DESIGN PRESSURE -1 PSIG TO +1 PSIG
2. DESIGN AIR FLOW 2000 CFM
1. CUSTOMER TO MAKE FINAL INSPECTION TO THIS DRAWING

GENERAL NOTES

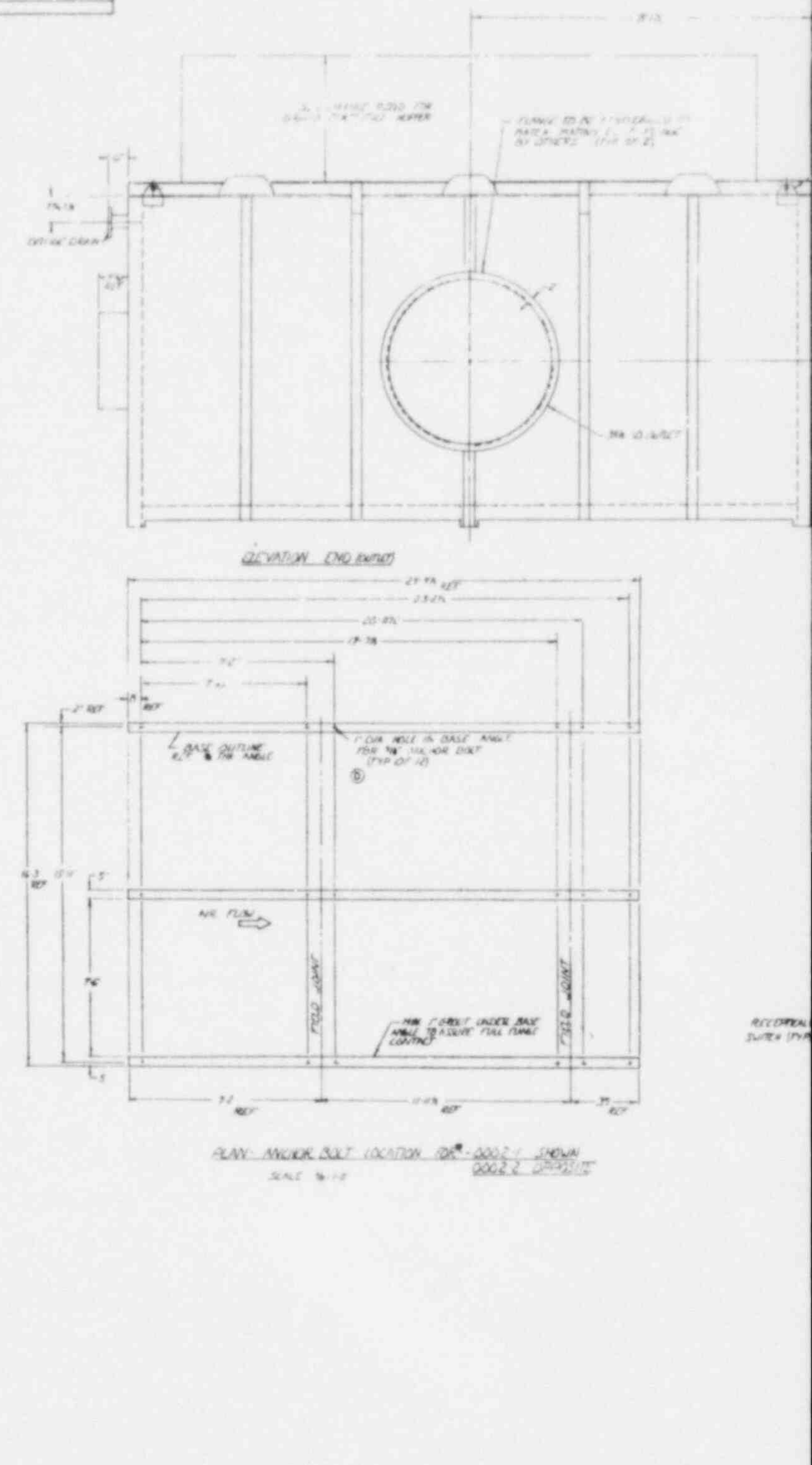


PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

General Arrangement -
Annulus Exhaust Unit

Figure 12.3-13

REVISIONS						
CHG	DATE	DWN	CHKD	APPS	REVS	DESCRIPTION
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible] REDRAWN
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	CHANGED ANCHOR BOLT [illegible]



8 ANCHOR BOLT MATERIAL SHALL BE [illegible] TO BE FURNISHED BY OTHERS

7 HOUSING TO BE PAINTED PER [illegible] BY OTHERS

6 ITEM No. [illegible] TO BE SHIPPED SEPARATE FROM UNIT

5 [illegible] OPERATING WT [illegible] LB EST. SHIPPING WT OF HEAVIEST ITEM [illegible]

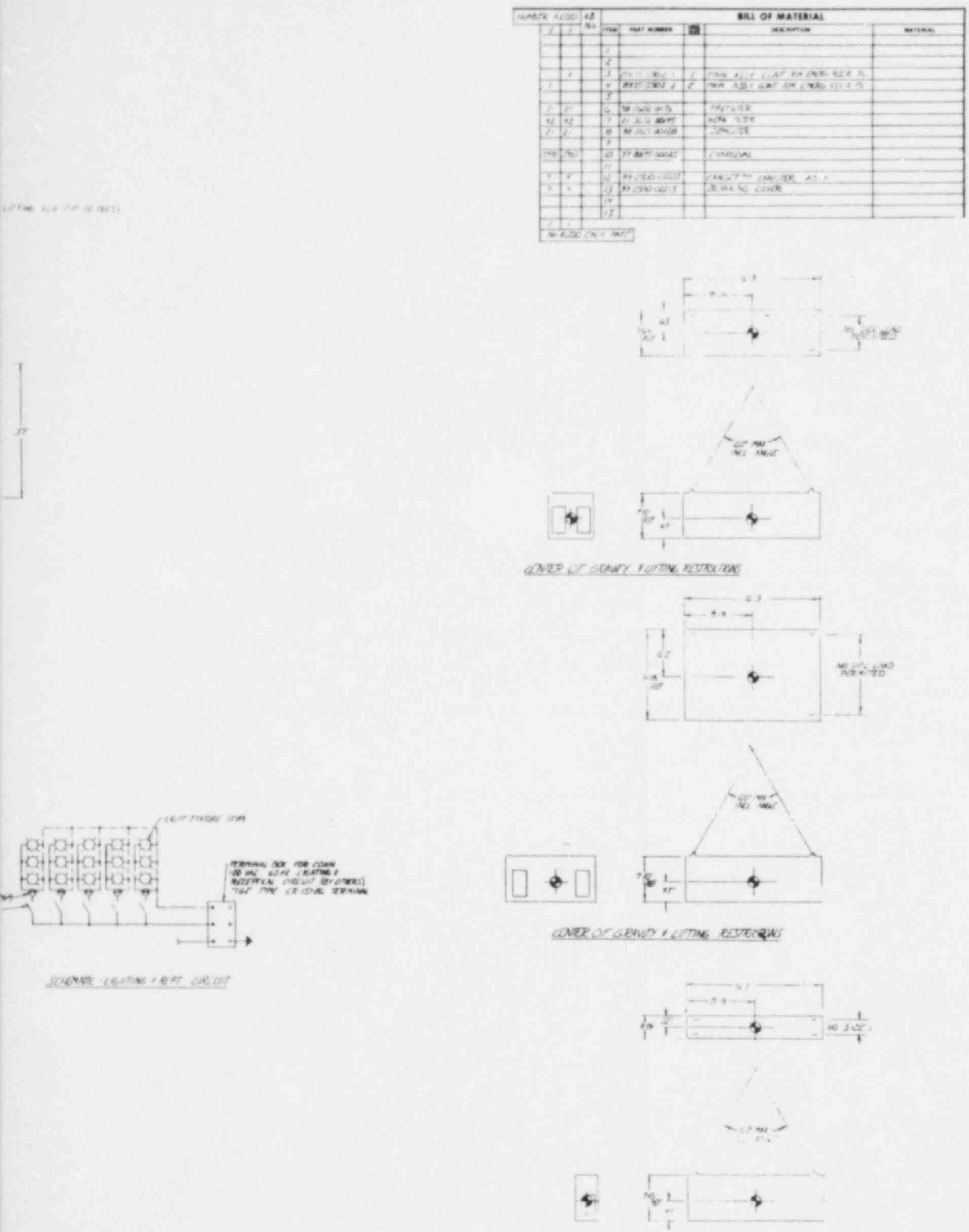
4 [illegible] FOR INSTALLATION OF [illegible] COMPONENTS

3 HOUSING DESIGN PRESSURE [illegible]

2 DESIGN AIR FLOW [illegible] CFM

1 CUSTOMER TO MAKE FINAL INSPECTION TO THIS DRAWING

GENERAL NOTES

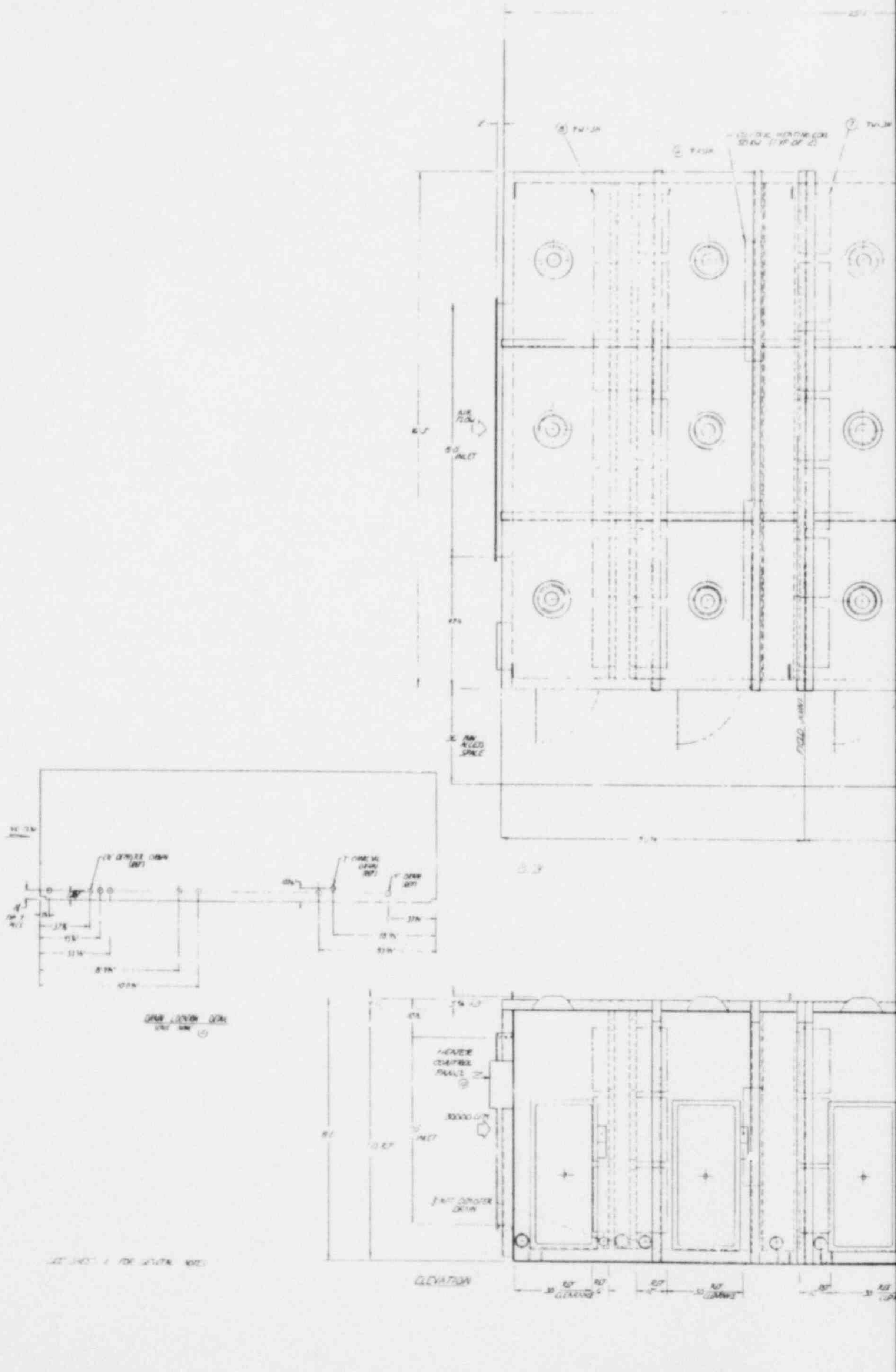


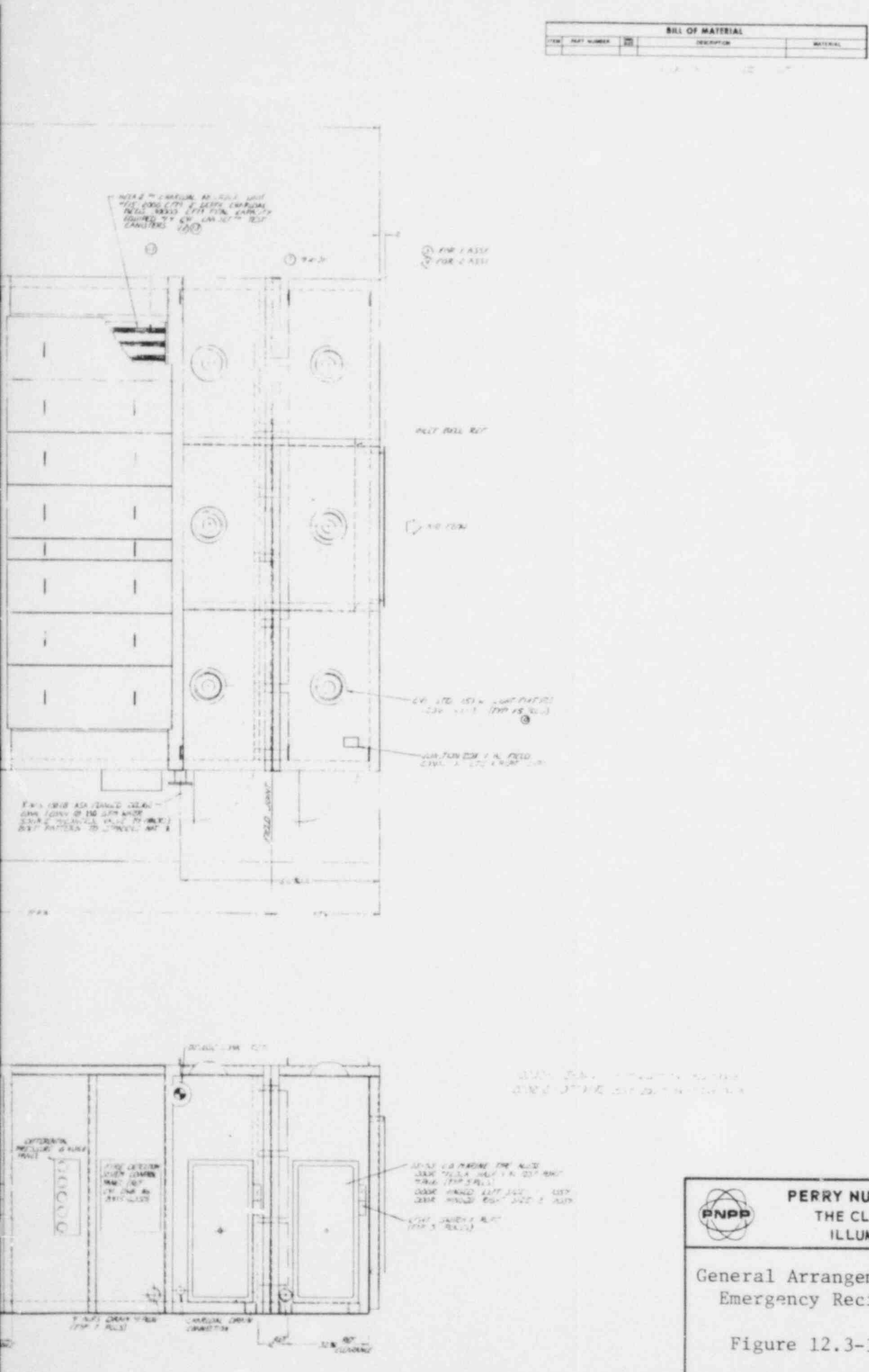
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

General Arrangement - Control Room
Emergency Recirculation Plenum

Figure 12.3-14 (Sheet 1 of 2)

REVISIONS



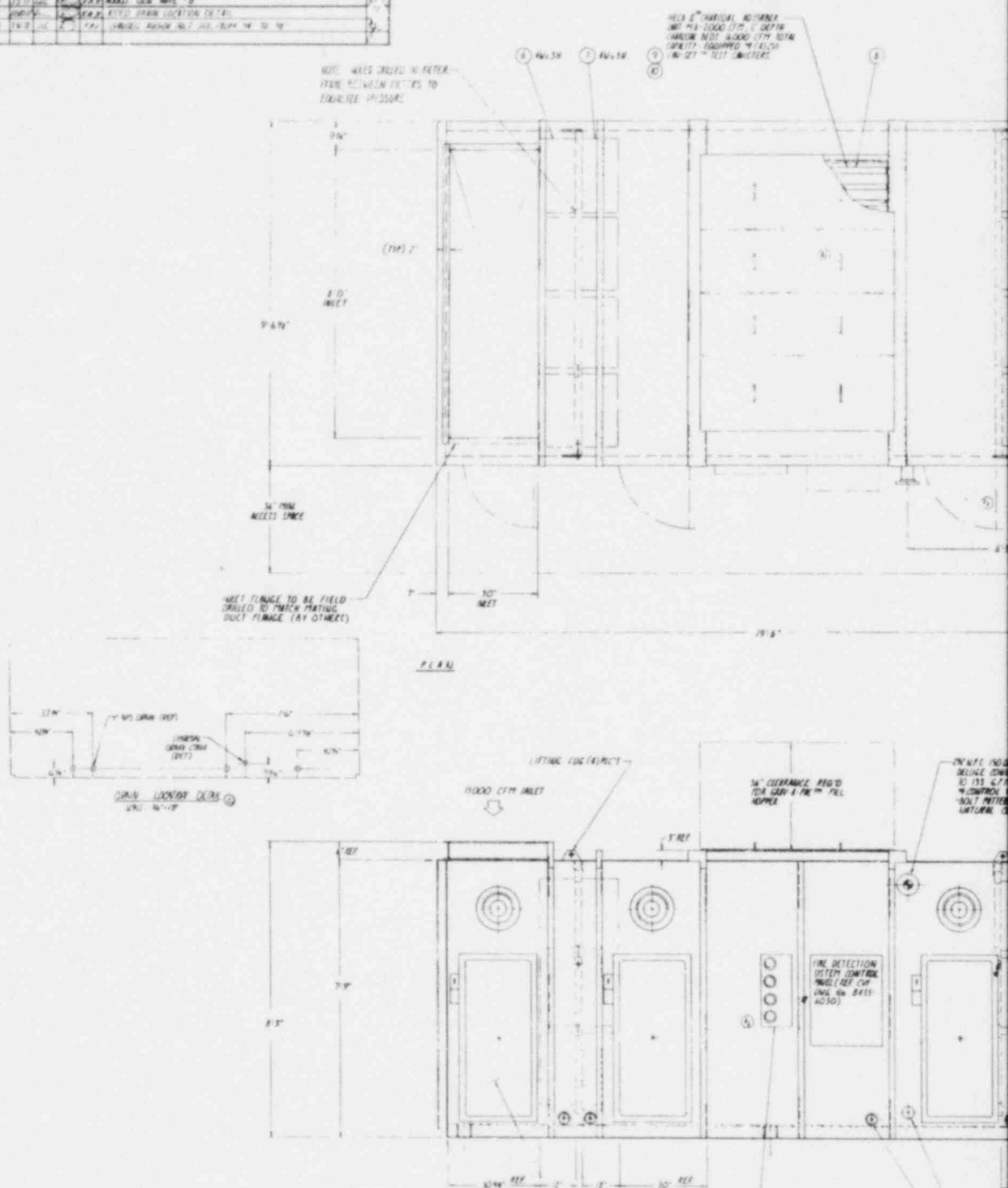


PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

General Arrangement - Control Room Emergency Recirculation Plenum

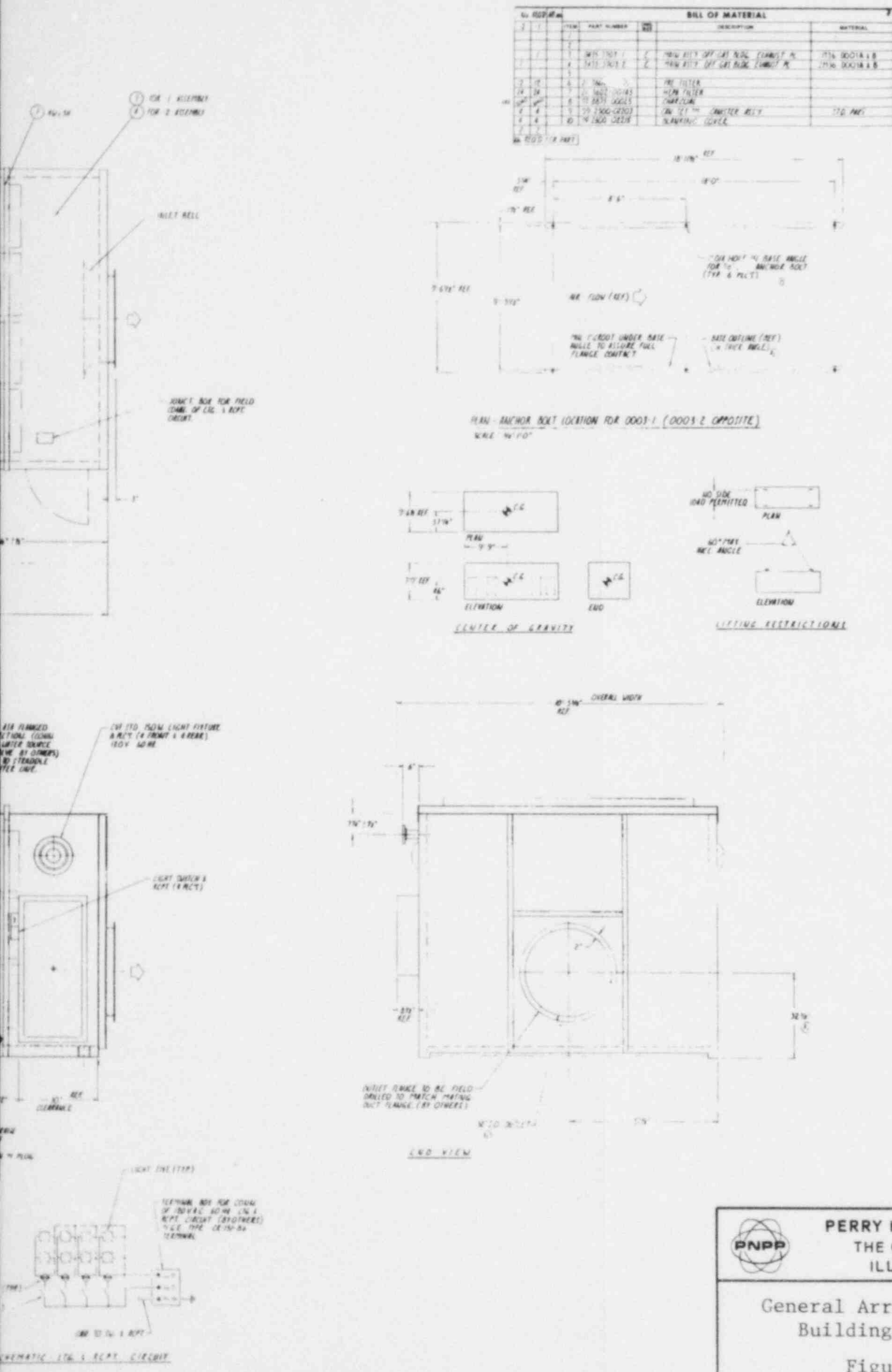
Figure 12.3-14 (Sheet 2 of 2)

REVISIONS					
CHG.	DATE	DWN.	CHKD.	APPD.	DESCRIPTION
A	[illegible]	[illegible]	[illegible]	[illegible]	1 REV OUTLET ... 55 1/2 x 56 3/4 HIGH. 2 ADDED ANGLE THICKNESS 3 REV CAGE PANEL 4 REV SIDE SHEET IN PLAN 5 REV NECK COVERS
B	[illegible]	[illegible]	[illegible]	[illegible]	ADDED GEN NOTE # 8
C	[illegible]	[illegible]	[illegible]	[illegible]	ADDED DRAIN LOCATION DETAIL
D	[illegible]	[illegible]	[illegible]	[illegible]	CHANGED ANCHOR BOLT SIZE FROM [illegible] TO [illegible]



8. ANCHOR BOLT MATERIAL TO BE ASTM A307 TO BE FURNISHED BY OTHERS.
7. HOUSING TO BE PAINTED PER CVI SPEC. ... (EXTERIOR TO BE PRIME COATED ONLY - FINISH PAINT BY OTHERS).
6. ITEM No. 6,7,8,9 & 10 TO BE SHIPPED SEPERATE FROM UNIT.
5. TOTAL OPERATING WT. 19,500 LB (EST). SHIPPING WT. 15,000 LBS (EST)
4. SEE CVI O&M MANUAL FOR INSTALLATION OF FILTERS & COMPONENTS.
3. HOUSING DESIGN PRESSURE -1 PSIG TO +1 PSIG.
2. DESIGN AIR FLOW 15,000 CFM
1. CUSTOMER TO MAKE FINAL INSPECTION TO THIS DRAWING.

GENERAL NOTES

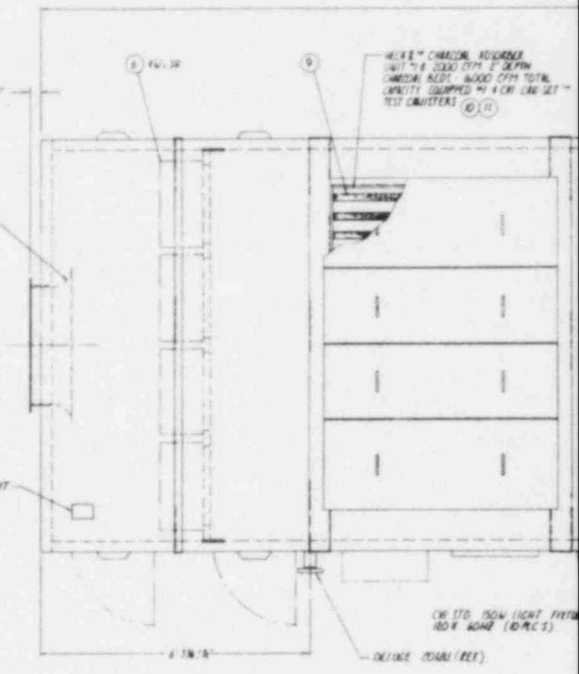


PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

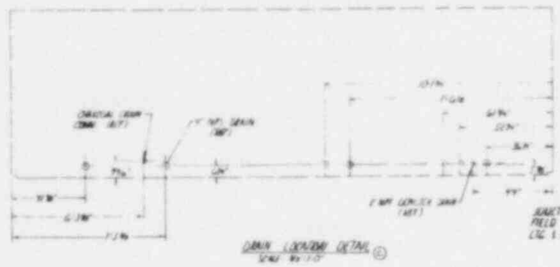
General Arrangement - Off-Gas
Building Exhaust Plenum

Figure 12.3-15

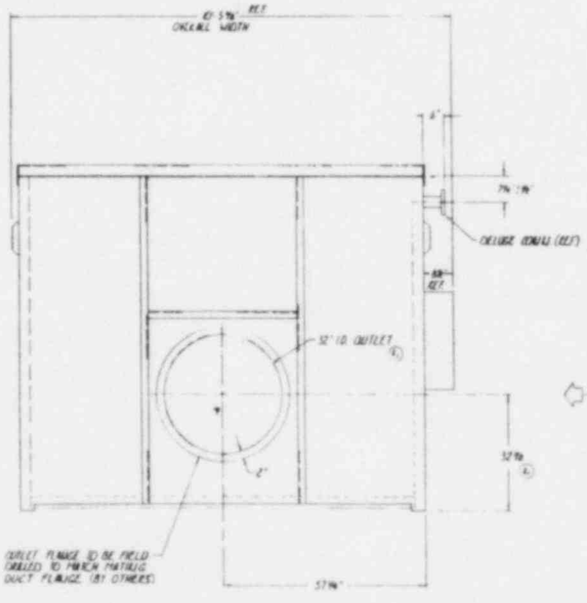
REVISIONS						
CHG.	DATE	DWN.	CHKD.	APPD.	DESCRIPTION	
A					1. REV OUTLET WAS 32¼" HGH 2. ADDED ANGLE THICKNESS	
					3. REV GAGE PANEL	
B					ADDED GEN NOTE 8	
C					ADDED DRAIN LOCATION DETAIL	
D					REVISED ANCHOR BOLT SIZE FROM ⅝ TO ¾	



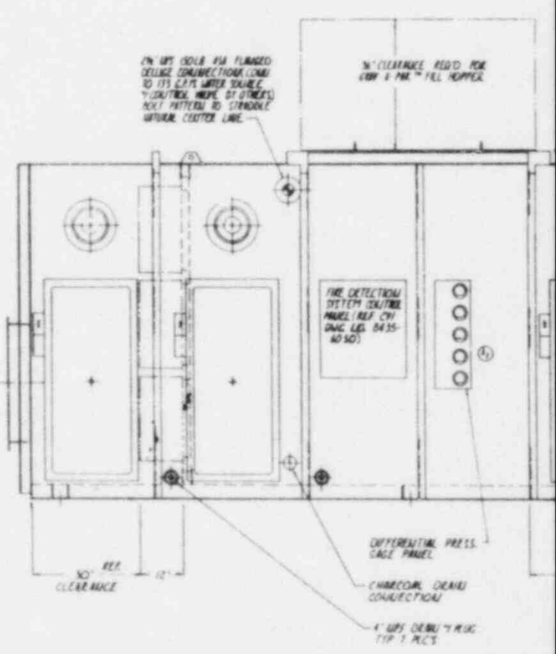
PLAN



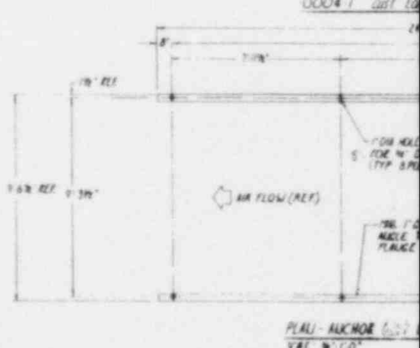
DRAIN LOCATION DETAIL



END ELEVATION



FRONT ELEVATION



PLAN - ANCHOR

8.

7. HOUSING TO BE PAINTED PER CVI SPEC. No. 8435-23-1055 (EXTERIOR TO BE PRIME COATED ONLY. FINISH PAINT BY OTHERS)

6. ITEM NO. 3, 6, 7, 9, 10 & 11 TO BE SHIPPED SEPERATE FROM UNIT.

5. TOTAL OPERATING WT. 28000 LB (EST) SHIPPING WT. 21000 LB (EST)

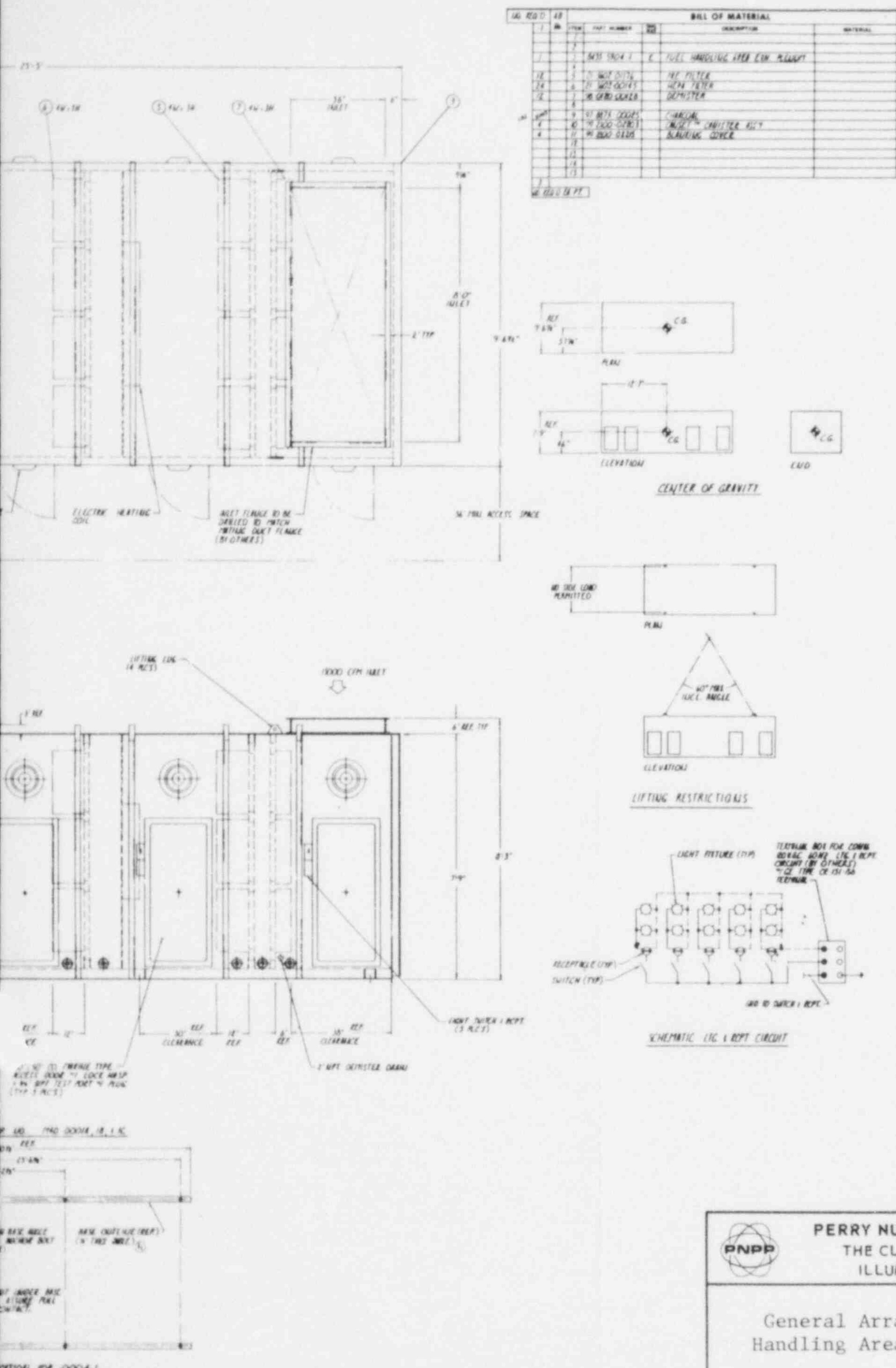
4. SEE CVI O&M MANUAL FOR INSTALLATION OF FILTERS & COMPONENTS.

3. HOUSING DESIGN PRESSURE -1 PSIG TO +1 PSIG

2. DESIGN AIR FLOW 15000 CFM

1. CUSTOMER TO MAKE FINAL INSPECTION TO THIS DRAWING.

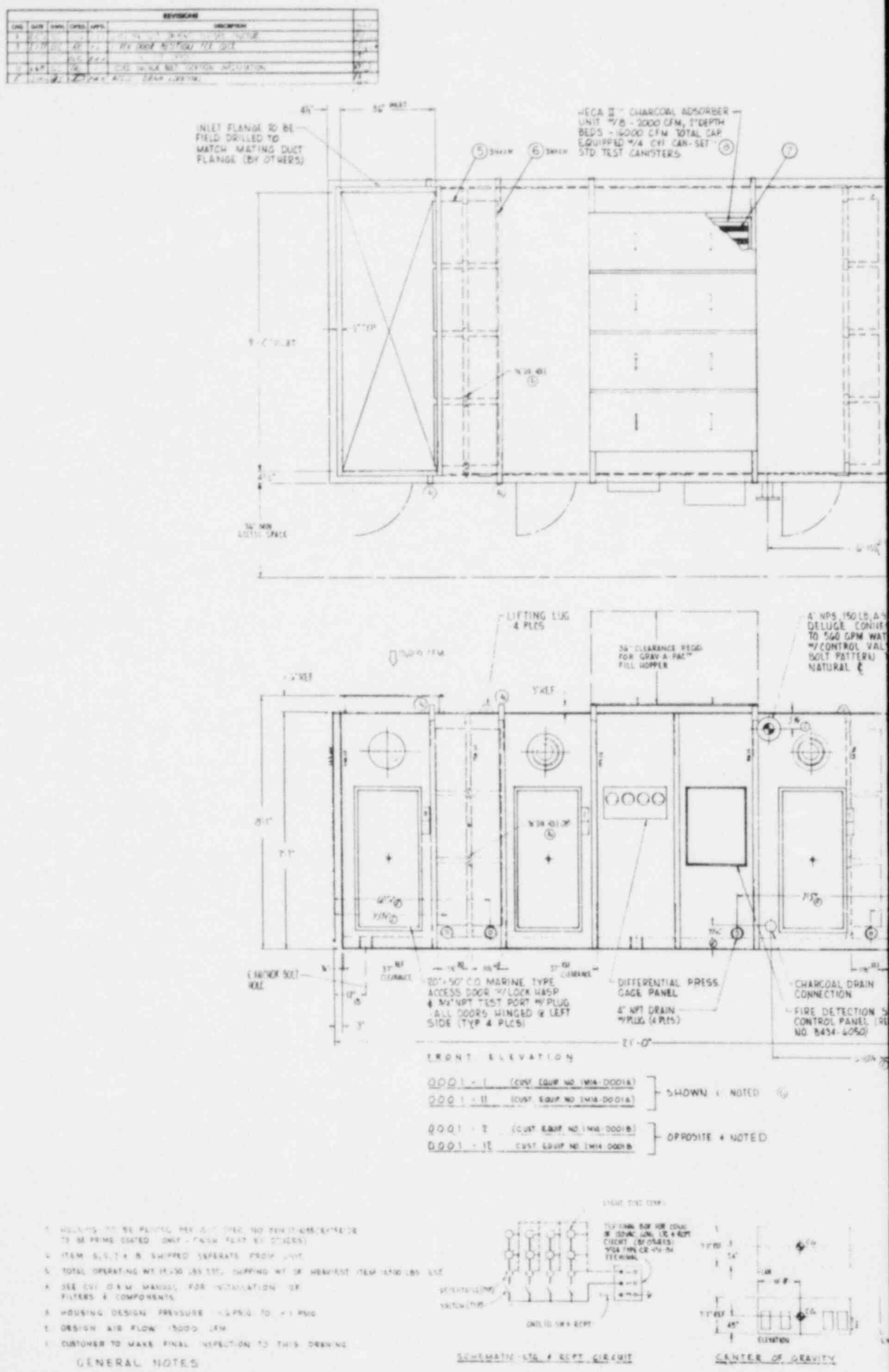
GENERAL NOTES



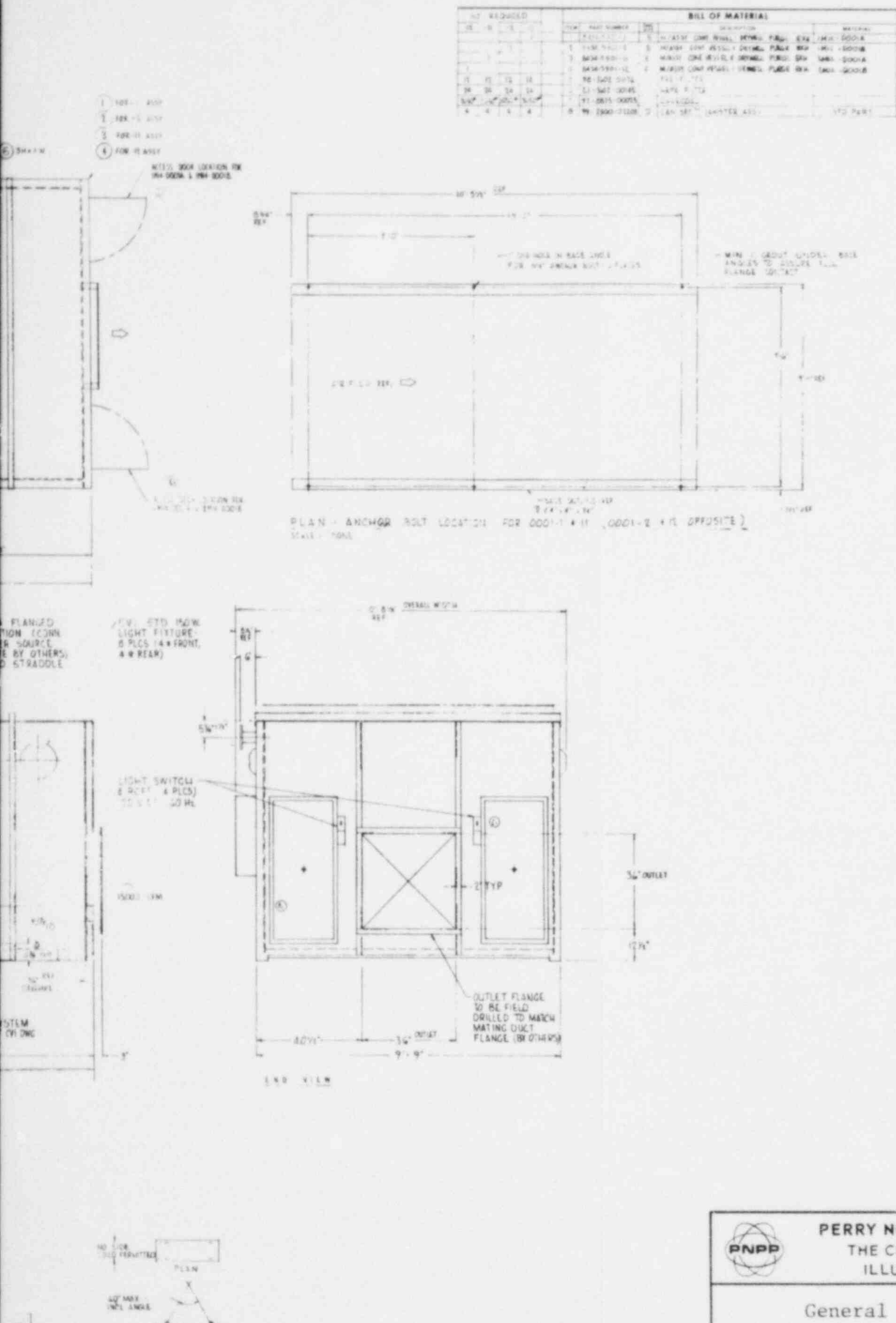
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

General Arrangement - Fuel Handling Area Exhaust Plenum

Figure 12.3-16



REVISIONS
INLET FLANGE TO BE FIELD DRILLED TO MATCH MATING DUCT FLANGE (BY OTHERS)
HECA II CHARCOAL ADSORBER UNIT W/8 - 2000 CFM, 2" DEPTH BEDS - 16000 CFM TOTAL CAP. EQUIPPED W/4 CVI CAN-SET STD TEST CANISTERS
LIFTING LUG 4 PLCS
36" CLEARANCE REQD FOR GRAV-A-PAC FILL HOPPER
NATURAL ℄
20"x50" C.O. MARINE TYPE ACCESS DOOR W/LOCK HASP & 3/4" NPT TEST PORT W/PLUG (ALL DOORS HINGED @ LEFT SIDE (TYP 4 PLCS)
DIFFERENTIAL PRESS GAGE PANEL
4" NPT DRAIN W/PLUG (4 PLCS)
CHARCOAL DRAIN CONNECTION
FRONT ELEVATION
0001-1 (CUST EQUIP NO 1M14-0001A)
0001-11 (CUST EQUIP NO 1M14-0001A)
SHOWN & NOTED
0001-2 (CUST EQUIP NO 1M14-0001B)
0001-12 (CUST EQUIP NO 1M14-0001B)
OPPOSITE & NOTED
4. ITEM 5,5,7 & 8 SHIPPED SEPERATE FROM UNIT
6. SEE CVI O&M MANUAL FOR INSTALLATION OF FILTERS & COMPONENTS
3. HOUSING DESIGN PRESSURE
2. DESIGN AIR FLOW
1. CUSTOMER TO MAKE FINAL INSPECTION TO THIS DRAWING
GENERAL NOTES
SCHEMATIC LTG & RCPT CIRCUIT
CENTER OF GRAVITY

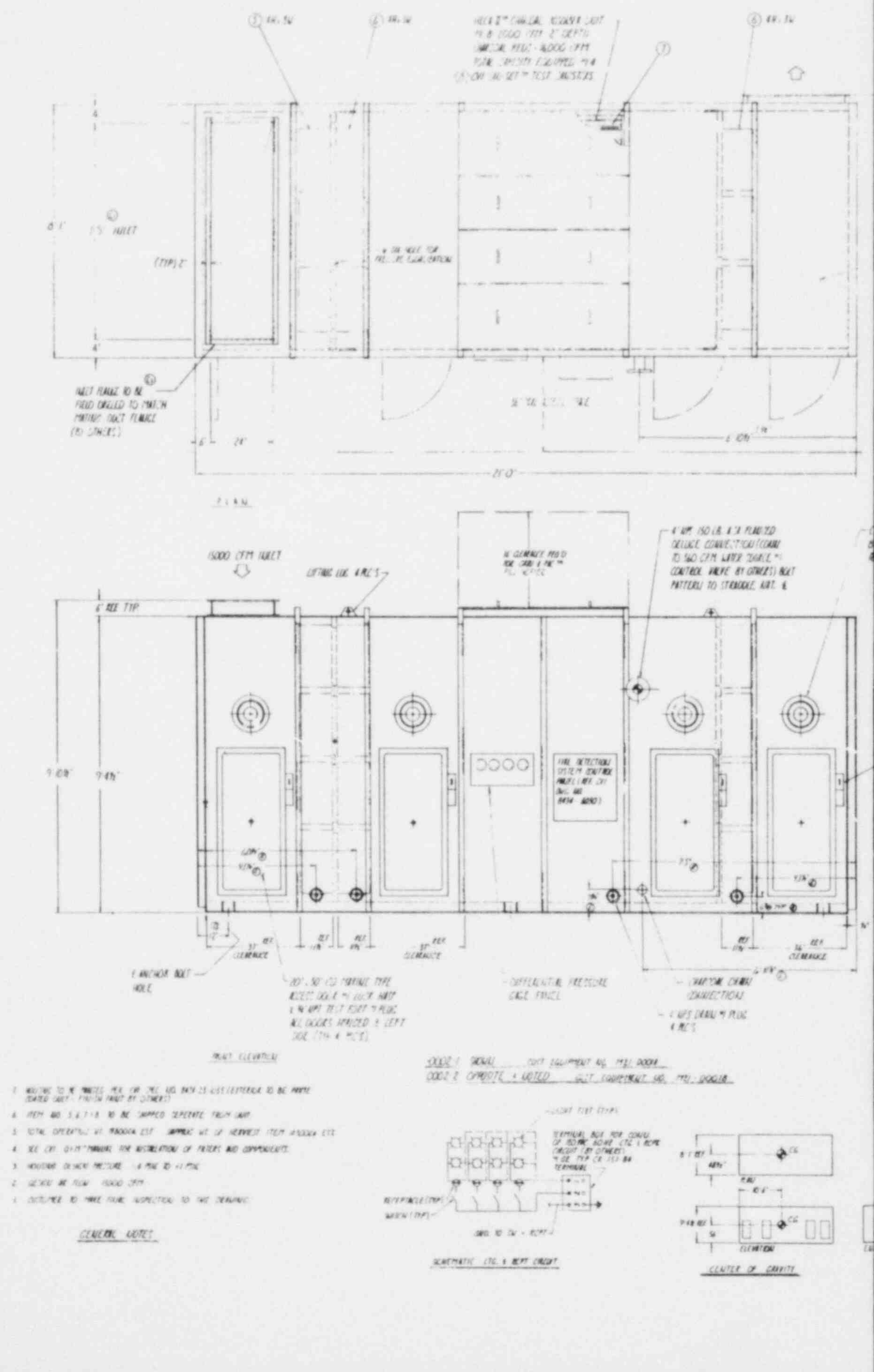


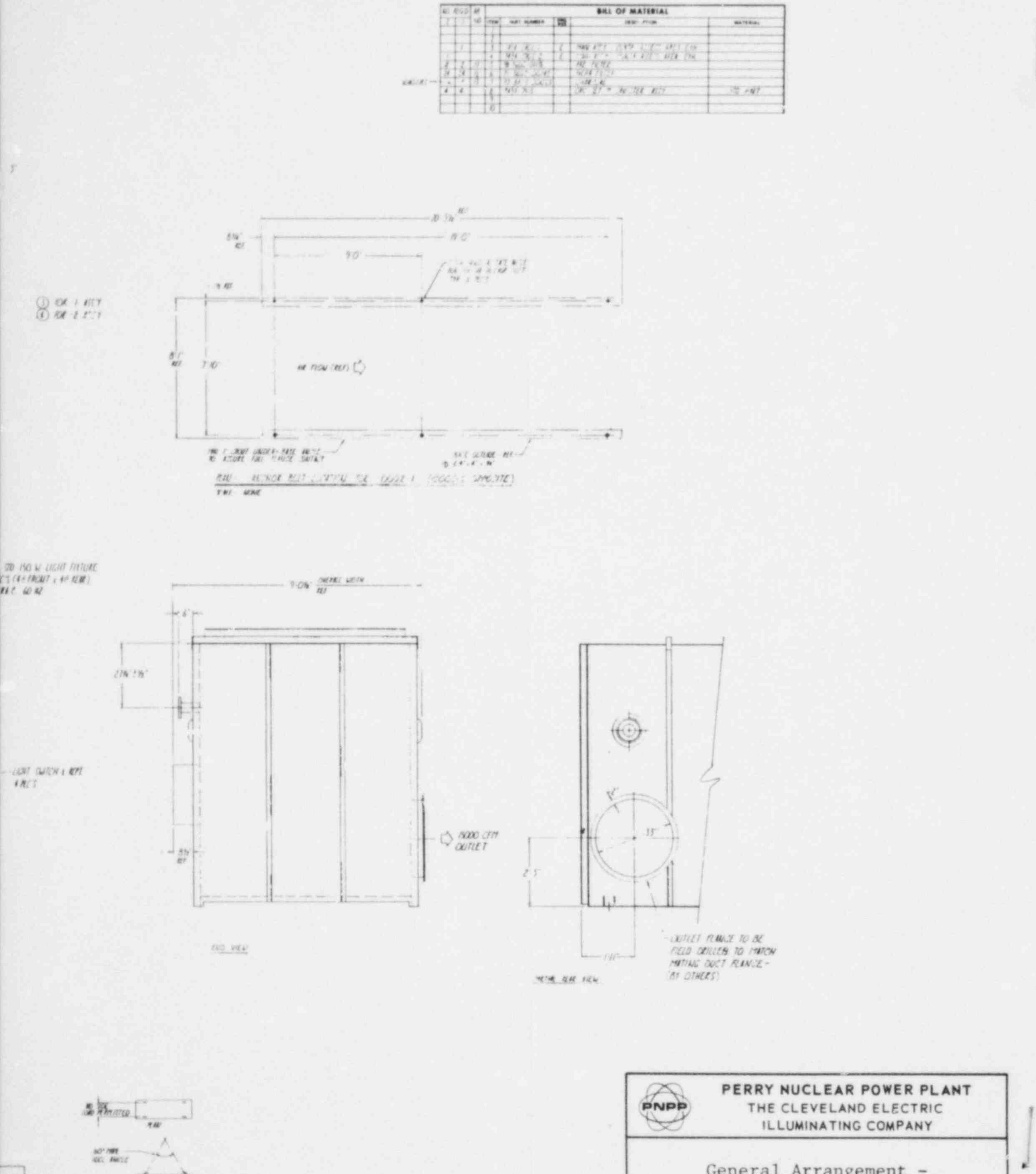
PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

General Arrangement - Containment Vessel & Drywell Purge Exhaust

Figure 12.3-17

REVISIONS						
CHG.	DATE	DWN.	CHKD.	APPD.	DESCRIPTION	
B	[illegible]	[illegible]	[illegible]	[illegible]	REV TO 15000 CFM & REDRAWN	[illegible]
C	[illegible]	[illegible]	[illegible]	[illegible]	REV PER CUST COMMENTS & ADD INLET NOTE	[illegible]
D	[illegible]	[illegible]	[illegible]	[illegible]	ADDED ANCHOR BOLT INFORMATION	[illegible]
E	[illegible]	[illegible]	[illegible]	[illegible]	ADDED DRAIN LOCATIONS	[illegible]

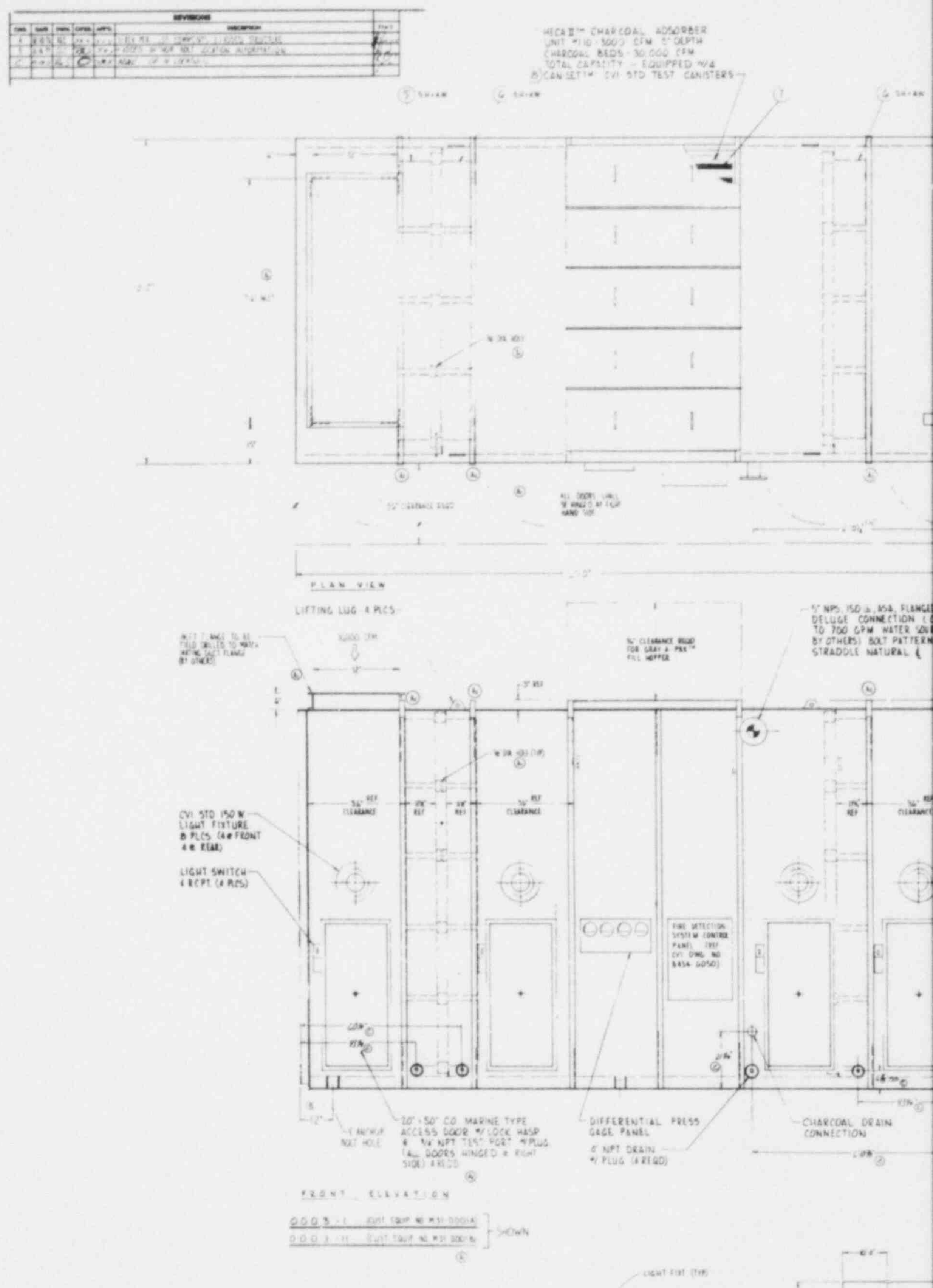




PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

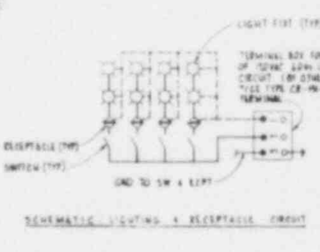
General Arrangement -
Controlled Access Area
Exhaust Plenum

Figure 12.3-18

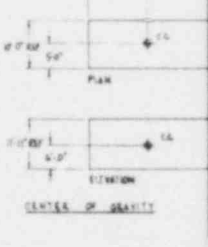


7. HOUSING TO BE PAINTED PER CVI SPEC (EXTERIOR PRIMED ONLY FINISH COAT BY OTHERS)
6. ITEM NO. 5, 6, 7 & 8 TO BE SHIPPED SEPERATE FROM UNIT
5. TOTAL OPER WT 17,800 LBS EST. SHIPPING WT. OF HEAVIEST ITEM 14,500 LBS EST.
4. SEE CVI OEM MANUAL FOR INSTALLATION OF FILTERS & COMPONENTS
3. HOUSING DESIGN PRESSURE -6 PSIG TO +1 PSIG
2. DESIGN AIR FLOW 30,000 CFM
1. CUSTOMER TO MAKE FINAL INSPECTION TO THIS DRAWING

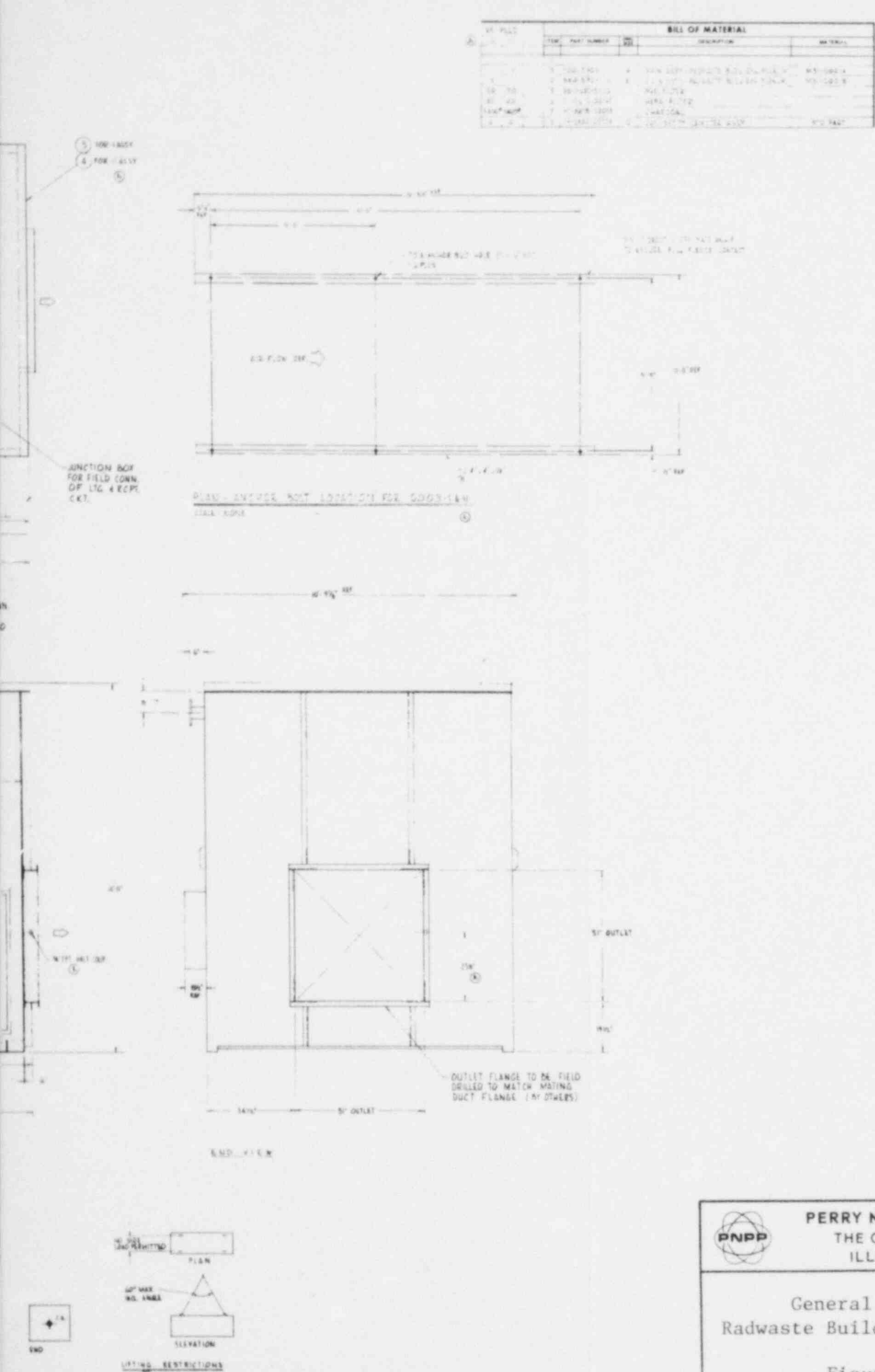
GENERAL NOTES



SCHEMATIC LIGHTING & RECEPTACLE CIRCUIT



CENTER OF GRAVITY

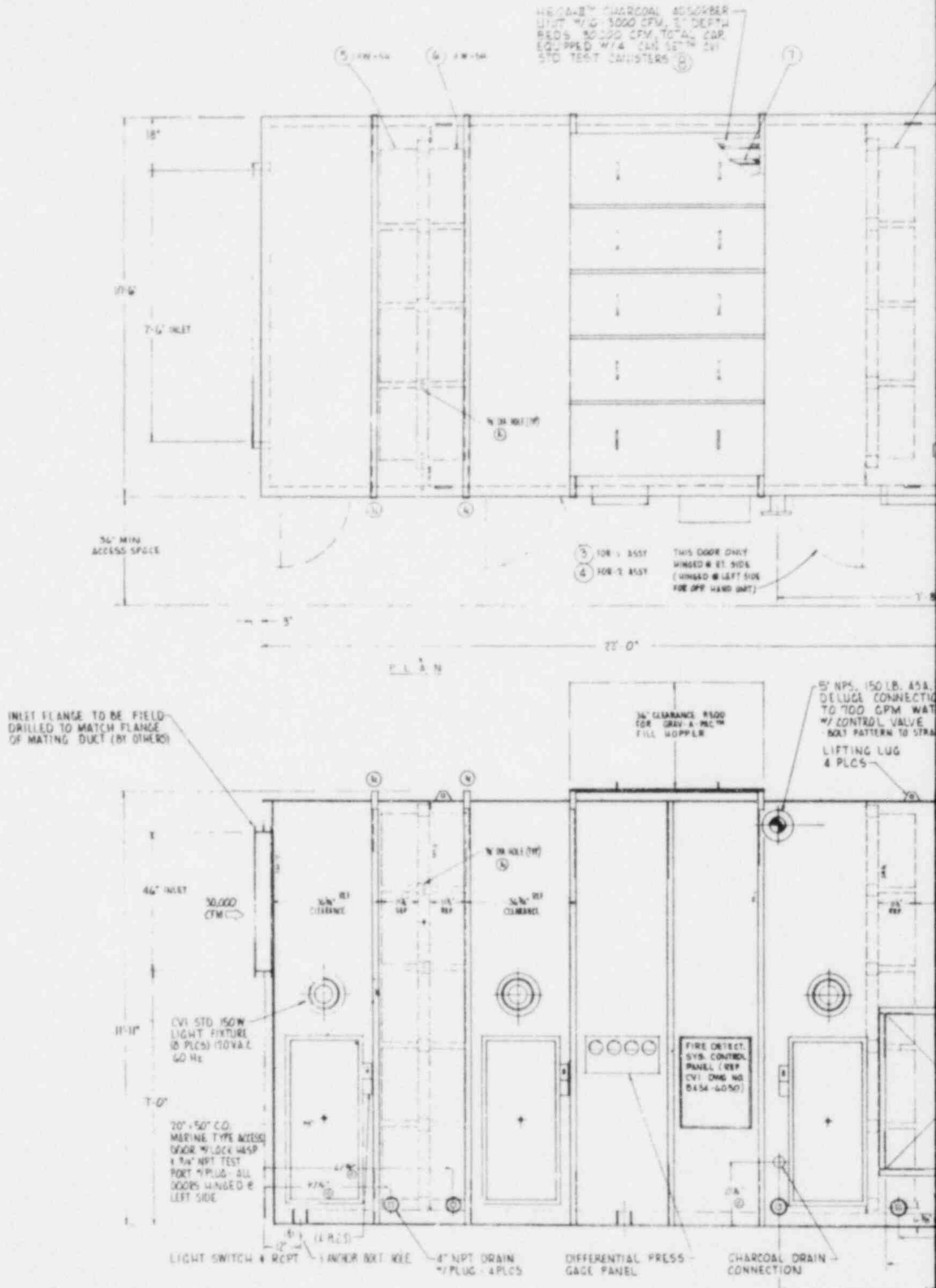


PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

General Arrangement - Radwaste Building Exhaust Plenum

Figure 12.3-19

REVISIONS						
CHG.	DATE	DWN.	CHK'D.	APP'D.	DESCRIPTION	
A	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]
B	[illegible]	[illegible]	[illegible]	[illegible]	ADDED ANCHOR BOLT LOCATION INFORMATION	[illegible]
C	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]



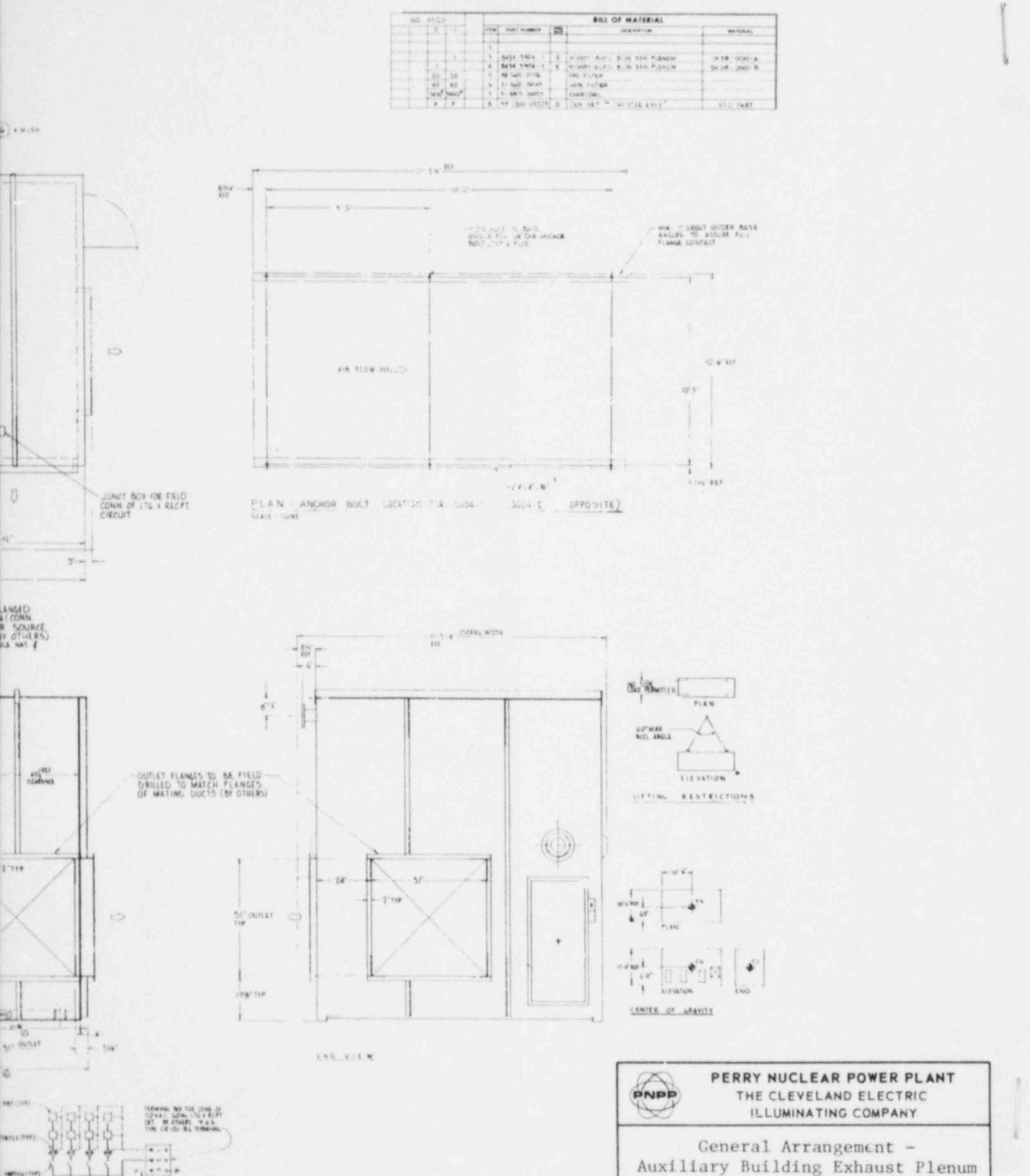
FRONT ELEVATION

0004-1 SHOWN (CUST EQUIP NO. 1M38-D001)

0004-2 OPPOSITE & NOTED (CUST EQUIP NO. 1M38-D002)

7. HOUSING TO BE PAINTED PER CVI SPEC 8434-23-1055 (EXTERIOR TO BE PRIME COATED ONLY - FINISH PAINT BY OTHERS)
6. ITEM NO. 5,6,7 & 8 SHIPPED SEPERATE FROM UNIT
5. TOTAL OPERATING WT 15300 LBS EST; SHIPPING WT OF HEAVIEST ITEM 14000 LBS EST.
4. SEE CVI O & M MANUAL FOR INSTALLATION OF FILTERS & COMPONENTS
3. HOUSING DESIGN PRESSURE -6 PSIG TO +1 PSIG
2. DESIGN AIR FLOW 30,000 CFM
1. CUSTOMER TO MAKE FINAL INSPECTION TO THIS DRAWING

GENERAL NOTES



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

General Arrangement - Auxiliary Building Exhaust Plenum

Figure 12.3-20

12.4 DOSE ASSESSMENT

The estimates of exposure use both design radiation zones and the associated occupancy times described in Section 12.3.1, along with operational data gathered from similar BWR plants. Design radiation levels used have been developed from conservative assumptions which indicate maximum radiation levels and not those anticipated for normal plant operation.

12.4.1 ESTIMATES OF PERSONNEL OCCUPANCY REQUIREMENTS

Occupancy requirements throughout the plant are based on operating experience and project manpower needs, and are considered in establishing the radiation zones described in Section 12.3.1. To estimate occupancy requirements, plant personnel are categorized into five groups according to work function. Feedback information from operating facilities indicates that contract workers will be called upon to do some tasks as indicated in Table 12.4-3. Essentially all of the dose received by contract personnel will be from maintenance activities. Table 12.4-1 lists the estimated size of each group and the estimated occupancy requirements for each group in Zones I, II and III. Personnel requirements are for both Units 1 and 2. Estimates were not made for Zones IV and V since routine occupancy is not anticipated in these zones during normal operation or during anticipated operational occurrences.

12.4.2 ESTIMATES OF ANNUAL MAN-REM DOSES

Annual doses to plant personnel are estimated based on the assumption of 2,000 hours per year work for each employee. It is anticipated that the general radiation levels for each zone will be less than the design values stated for the zone, although isolated higher levels will exist in certain areas within the zone.

The estimated man-Rem doses for the various categories of plant personnel discussed in Section 12.4.1 are listed in Table 12.4-2. Tables 12.4-3 and 12.4-4 list the percentages of exposures by job function and work function,

respectively, as reported by operating nuclear power plants for the year 1976.[1] The total annual dose for plant operation is conservatively estimated to be 404 man-Rem/unit of which 175 man-Rem/unit will be received by contract personnel. Table 12.4-5 lists several of the larger operating BWR facilities and the associated man-Rem dose for the years 1974, 1975 and 1976.[2] This indicates that the man-Rem dose does vary considerably but is generally higher than the estimated man-Rem dose for PNPP operation.

A further estimate of the man-Rem doses has been made by identifying specific tasks anticipated to occur at the plant. Various data from operating plants and in current publications (References 1 through 8) were used to identify these tasks, the manpower effort required to complete each task and the radiation levels associated with performing each task. The guidance of Regulatory Guide 8.19 has been followed to complete these dose estimates. Tables 12.4-6 through 12.4-12 summarize the man-Rem doses per tasks. These tables give man-Rem estimates on a per unit basis. Table 12.4-6 indicates that special maintenance tasks represent a large percentage of the annual man-Rem dose. Comparison of these results with Table 12.4-4 indicates that routine and special maintenance make up over 70 percent of the annual man-Rem dose at operating facilities. Disagreement in the estimated values for PNPP and industry averages can be partially accounted for by design changes at PNPP (see Section 12.2.1), and in the way various tasks have been categorized.

Particular mention should be made regarding feedwater sparger repair. This task is a very special maintenance task and may not occur during the projected life of the plant. However, a man-Rem dose has been added for this task under the assumption that it will occur once during the 40 year life of the plant. For other special maintenance tasks it has been assumed that frequency of occurrance will coincide with refueling outages. In actuality, the frequency or special maintenance will be quite variable.

In addition to the facility design aspects that will promote ALARA, both the ALARA-oriented health physics program (see Section 12.5) and the utilization of applicable operational data and guidance will be very important in ensuring that personnel doses are ALARA.

12.4.3 ESTIMATED INHALATION DOSES

Radiation doses associated with airborne radioactivity have not been analyzed in terms of tasks due to the lack of sufficient data. Inhalation doses were estimated using the airborne sources of radioactivity described in Section 12.2.2. The ventilation system has been designed to move air from areas of unlimited occupancy (no radiation sources) to areas of limited occupancy (potential airborne radioactivity sources). Appropriate health physics procedures will be established to measure the radiological conditions in areas with potential airborne contamination, ensuring that radiation doses are maintained ALARA. Where required, respirators will be used to further reduce inhalation doses.

Table 12.4-13 lists specific areas of the plant where airborne activity may be present in quantities that would result in a measurable dose to the whole body and, as a result of gaseous iodines, a dose to the thyroid. The areas included are the reactor building (during normal operation and during refueling), the fuel handling building, the radwaste building and the turbine building. For each area man-hours per work function have been estimated using References 4 and 6. The exposure rates have been determined from specific activities listed in Section 12.2.2. Sources of airborne activity will be primarily from valve and pump leakage. During refueling operations the refueling and spent fuel pools will release small amounts of airborne activity to the reactor building and the fuel handling building, respectively. However, as a result of pool cleanup systems, it is anticipated that contributions from these sources will be minimal. The resulting annual man-Rem doses per unit are listed in Table 12.4-13.

An additional airborne activity source will come from the actuation of the safety/relief valves after an isolation scram (Type 2 event). Type 1 and Type 2 events of steam discharges to the suppression pool are discussed in Section 12.2. Table 12.4-14 gives the resulting doses to personnel in the reactor building as they exit after a Type 2 event. It has been assumed that one Type 2 event will occur per year and that exit time for individuals in the reactor building will be four minutes. No dose calculations regarding Type 1 events are presented since resulting personnel doses would be negligible.

12.4.4 ESTIMATED ANNUAL DOSE OUTSIDE THE NUCLEAR FACILITY AT THE BOUNDARY OF THE RESTRICTED AREA

Potential direct radiation doses to individuals outside the nuclear facility will arise from the following:

a. Skyshine and direct dose from turbines

b. Direct dose from stored radwaste

c. Direct dose from the external surfaces of buildings

d. Dose from the gaseous radioactive plume

12.4.4.1 Skyshine and Direct Dose from Turbines

The dose analyses for normal operation of both units were based on an 80 percent load factor, and 50 and 24 percent occupancy factors for offsite and onsite exposures, respectively. For distances beyond 300 feet, a single lumped source was assumed in the turbine building; for distances less than 300 feet, all major sources were considered separately. The resultant doses at selected locations are given in Table 12.4-15 and a curve of the dose rate (mRem/hr) versus distance from the turbine building is presented in Figure 12.3-13.

It is anticipated that non-plant personnel could receive a radiation dose when Unit 1 is in commercial operation and Unit 2 is still being completed. A recent study (Reference 9) has analyzed several two unit stations where construction personnel are completing a unit next to an operating one. This study indicated that construction personnel are likely to receive radiation doses that are not distinguishable from background radiation levels. For the PNPP, however, a conservative dose analysis for non-plant personnel has been performed based on the following assumptions:

a. The activities scheduled during this period on Unit 2 can be divided into three basic categories: completion of construction, start-up/testing,

and site engineering. The total duration of these activities is scheduled to be 41 months. A listing of the significant dates used in the analysis is given as follows:

1. October, 1982 - End of Unit #1 construction
2. November, 1982 - Unit #1 fuel load
3. May, 1984 - Commercial operation of Unit #1
4. October, 1986 - End of Unit #2 construction
5. November, 1986 - Unit #2 fuel load
6. May, 1988 - Commercial operation of Unit #2

b. The relative orientation of the structures is as shown on Figure 1.2-1.

c. The start-up/testing activities on Unit 2 consist of such items as fuel loading, pre-operational testing and startup and power testing. The bulk of this work will be accomplished in the last 12 of the 41 month period between commercial operation of Unit 1 and Unit 2.

d. The occupational composition and size for the start-up/testing work for a single shift is conservatively assumed as follows:

1. Engineers 12 required
2. Electricians 20 required
3. Pipe fitters 20 required
4. Crane operators - 4 required
5. Operating engineers 4 required
6. Utility personnel - 10 required
7. Laborers - 10 required

e. There will be three 9-hour shifts per day.

f. During any shift, the following time schedules are assumed:

1. A portion of the work force (25 percent) spends 5 hours on the turbine building operating floor, 2 hours outside all plant structures and 2 hours inside plant structures. It is conservatively assumed that while inside a plant structure, the

individuals are shielded by a minimum of two feet of concrete and are located at a distance of 300 feet from the Unit 1 turbine building.

2. The remainder of the work force spends 2 hours outside all plant structures and 7 hours inside plant structures.

g. The completion of construction activities on Unit 2 consists of installation of equipment and piping, electrical wiring, closing of concrete construction openings, completion of painting and architectural items and completion of final roadways and landscaping in the area around Unit 2.

h. The size of the work force needed for the completion of Unit 2 construction, based on present schedule resource loading, is given on Figure 12.4-1.

i. The work force is assumed to be composed of the occupational crafts listed in Table 12.4-16.

j. During a working day, the following time schedules are assumed:

1. A portion of the work force (40 percent) spends 2 hours outside all plant structures, and 7 hours inside plant structures. It is conservatively assumed that while inside a plant structure, the individuals are shielded by a minimum of two feet of concrete and are located at a distance of 300 feet from Unit 1 turbine building.

2. The remaining 60 percent of the work force is conservatively assumed to spend all 9 hours of the work shift outside plant structures.

k. The site organization is a field engineering group housed on-site to complete portions of the final design and handle field construction problems. The estimate of the work force does not include the Perry Plant Department (operations, Rad-Chem, maintenance and I&C) since their occupational exposures are discussed in Section 12.4.2.

l. The size of the site organization work force is shown on Figure 12.4-2.

m. During the working day, the following schedule is assumed: the average worker spends 5 hours inside the warehouse/office, 2 hours outside all plant structures and 2 hours inside plant structures.

The radiation dose to the startup/test personnel, construction personnel, and site organization personnel will arise from radiation levels at the external surfaces of buildings (Section 12.4.4.3), from the gaseous radioactive plume (Section 12.4.4.4) and from the predominant source (skyshine). A summary of direct doses from these sources is given in Table 12.4-17. The resulting man-Rem information is summarized in Table 12.4-18.

12.4.4.2 Direct Doses From Stored Radwaste

All radwaste will be stored in the radwaste storage area which is a part of the total building complex. All radwaste storage areas have outside walls that are 4 feet thick. These walls are designed for an outside radiation zone of 0.5 mRem/hr continuous occupancy as indicated in Table 12.3-1. The radiation from this area is equivalent to that calculated in Section 12.4.4.3.

12.4.4.3 Direct Doses From the External Surfaces of Buildings

All external walls of buildings have been designed to attenuate radiation sources from within to comply with Zone I conditions (0.5 mRem/hr). For calculational purpose, an expected radiation dose of 0.25 mRem/hr has been used. It has been assumed that the dose rate at the surface of the building decreases inversely proportional to the distance from the wall. In addition to distance, air attentuation also will decrease the doses. Table 12.4-18 lists distances from the building and associated direct dose rates and annual doses out to the exclusion boundary. Assumed occupancies are also given in Table 12.4-19.

12.4.4.4 Doses From Gaseous Radioactive Plume

A small direct dose will result from noble gases released from Units 1 and 2. The methodology used to calculate this dose has been taken from Regulatory Guide 1.109. The meteorological data has been taken from Section 2.3 and the quantity of nuclides released from Section 11.3.3. An 80 percent load factor and 50 and 24 percent occupancy factors for offsite and onsite exposures, respectively, were again assumed in calculating the expected dose rate and annual dose from this source (see Table 12.4-20).

12.4.5 REFERENCES FOR SECTION 12.4

1. U.S. Nuclear Regulatory Commission, NUREG-0323, "Occupational Radiation Exposure at Light Water Cooled Power Reactors, 1976," March 1978.

2. U.S. Nuclear Regulatory Commission, NUREG-0322, "Ninth Annual Occupational Radiation Exposure Report, 1976," October 1977.

3. U.S. Nuclear Regulatory Commission, "Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants, Design Stage Man-Rem Estimates," Regulatory Guide 8.19, May 1978.

4. Vance, J., Weaver, C. L., Lepper, E. M., "A Preliminary Assessment of the Potential Impacts on Operating Nuclear Power Plants of a 500 mRem/hr Occupational Exposure Limit," Report to the Nuclear Regulatory Staff by the Atomic Industrial Forum, April, 1978.

5. Verna, B. J., "Radioactive Maintenance, Parts 1-4," Nuclear News, September 1973, November 1975, January 1976, March 1976.

6. General Electric letter (PY-GEN/GAI-17) to P. B. Gudikunst from L. E. Wood, August 9, 1972.

7. Murphy, T. D. et al., "Occupational Exposure at Light Water Cooled Power Reactors, 1969-1975," U.S. Nuclear Regulatory Commission, NUREG-0109, August, 1976.

8. Pelletier, C. A., et al., "Compilation and Analysis of Data on Occupational Radiation Exposure Experienced at Operating Nuclear Power Plant," September, 1974.

9. Endres, G. W. R., Garcia, W. T., and Shipler, D. B., BNWL-SA-6103, "Dose to Construction Workers at Operating Reactor Sites," presented at ACRS Meeting on Radiological and Site Evaluation, Washington, D.C., July 11, 1978.

TABLE 12.4-1

ESTIMATED MANPOWER NEEDS AND OCCUPANCY REQUIREMENTS, UNITS 1 AND 2

Group	Number	Zone I (%)	Zone II (%)	Zone III (%)
Administrative	11	90	10	0
Radiation protection	21	60	38	2
Technical section	28	85	14	1
Operations	54	75	23	2
Maintenance	44	60	38	2
Contractors	100	60	38	2
Total	258			

TABLE 12.4-2

MAN-REM ESTIMATES FOR NORMAL PLANT OPERATIONS, ANTICIPATED OPERATIONAL OCCURENCES AND ROUTINE MAINTENANCE, UNITS 1 & 2

Group	Zone I (0.5 mrem/hr)	Zone II (2.5 mrem/hr)	Zone III (25 mrem/hr)
Administrative	9.9	5.5	0
Radiation protection	12.6	39.9	21.0
Technical section	23.8	19.6	14.0
Operations	40.5	62.1	54.0
Maintenance	26.4	83.6	44.0
Contractors	60.0	140.0	100.0

Subtotals: 457 man-Rems (station personnel)
350 man-Rems (contract personnel)

Total: 807 man-Rems

TABLE 12.4-3

PERCENTAGES OF EXPOSURE BY JOB FUNCTION

Job Function	Utility	Contractor	Total
Maintenance	30.0	43.7	73.7
Operations	10.3	0.3	10.6
Health Physics	4.0	1.8	5.8
Supervisory	2.6	1.1	3.7
Engineering	3.6	2.6	6.2
Totals	50.5	49.5	100

Note: See Reference 1.

TABLE 12.4-4

PERCENTAGES OF EXPOSURE BY WORK FUNCTION

Work Function	Utility	Contractor	Total
Reactor Operations	9.7	1.2	10.3
Routine Maintenance	16.3	14.8	31.1
Inservice Inspection	1.1	4.9	6.0
Special Maintenance	16.0	23.9	39.9
Waste Processing	3.7	1.2	4.9
Refueling	5.3	2.5	7.8
Totals	51.5	48.5	100

Note: See Reference 1.

TABLE 12.4-5

YEARLY OPERATIONAL MAN-REM FOR SELECTED BWR PLANTS

		1974	1975	1976
1.	Dresden 1, 2 & 3	1,662	3,423	1,680
2.	Monticello	349	1,353	263
3.	Nine Mile Point	824	681	428
4.	Peach Bottom 2 & 3	--	228	840
5.	Quad Cities 1 & 2	482	1,618	1,651
6.	Vermont Yankee	216	153	411
	Ave. Man-Rem/Unit	442	745	527

Note: See Reference 2.

TABLE 12.4-6

SUMMARY OF TOTAL OCCUPATIONAL RADIATION EXPOSURE ESTIMATES BY TASK

Function	Dose (man-Rems/yr-unit)	Percentage of Total man-Rem dose
Routine operation	43	11
Routine maintenance	34	9
Waste processing	3	1
Refueling	45	12
Inservice inspection	39	10
Special maintenance	212	57
Total man-Rems/yr-unit	376	

TABLE 12.4-7

OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE OPERATIONS AND SURVEILLANCE

Activity	Average Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rems/ year)
Control room	0.1	6,000	2	-	1.2
Walking and checking:					
Turbine and feedwater	100	0.1		1/shift	10.0
heat exchanger	1.0	1.0	1	1/shift	1.0
Containment cooling system	1.0	1.0	1	1/day	0.36
Standby liquid control system	1.0	1.0	1	1/day	0.36
ECCS and process equip	1.0	1.0	1	1/shift	1.0
C&I panels and equip in containment	1.0	1.0	1	1/shift	1.0
Fuel pool system	1.0	0.4	1	1/day	0.1
RWCU	1.0	0.5	1	1/shift	0.6
CRD system	1.0	0.5	1	1/shift	0.6
Recirc flow control	1.0	0.6	1	1/day	0.22
Misc auxiliary building	1.0	1.0	1	1/shift	1.0
	100	0.1	1	1/shift	10.0
Traversing incore probe system	10	0.1	1	1/shift	1.2
Misc. in containment	1.0	1.0	1	1/day	0.4
Instrument calibration in containment	1.0	0.6	1	1/week	0.03
Radiochemistry	1.0	1.0	2	1/day	0.73
Health physics surveys	1.0	4.0	1	1/day	1.46
	15	1.0	1	1/day	5.48
	100	5.0	1	1/week	26.0
Sample stations in reactor building	5.0	0.5	1	1/shift	2.7

TABLE 12.4-7 (Continued)

Activity	Average Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rems/ year)
Other local samples	5.0	0.1	1	1/day	0.15
Remote sampling	1.0	0.3	1	1/day	1.0
Total					43

TABLE 12.4-8

OCCUPATIONAL DOSE ESTIMATES DURING ROUTINE MAINTENANCE

Activity	Average Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rems/ year)
RWCU filter precoat	1.0	2.0	1	1/week	0.1
RWCU pump and valve	150	1.5	2	1/week	18.7
TIP system	10	2.0	2	1/month	0.5
CRD	2.5	10.0	1	1/week	1.5
HVAC systems	1.0	8.0	1	1/week	0.4
Radwaste evaporator	60	50	2	1/year	5.0
Sample stations	5.0	1.5	1	1/day	2.6
Misc. auxiliary building pumps (LPCS, HPCS, RCIC, etc.)	2.0 25.0	4.0 0.2	2 2	1/week 1/week	0.8 0.5
Feedwater and condensate pumps valves and heat exchangers	25.0 1.0	0.5 1.5	2 2	1/week 2/weeks	1.3 0.3
Condensate demineralizer and heat exchangers filters	1.0	1.5	2	1/day	1.0
Total					34

TABLE 12.4-9

OCCUPATIONAL DOSE ESTIMATES DURING WASTE PROCESSING

Activity	Average Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rems/ year)
Operation of liquid waste processing system	0.2	112	1	1/week	1.1
Operation of solid waste processing system	1.0	30	1	1/week	1.5
Total					3

TABLE 12.4-10

OCCUPATIONAL DOSE ESTIMATES DURING REFUELING

Activity	Average Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rems/ year)
Reactor pressure vessels head and internals - removal and installation	60	40	10	1/year	27
Refueling operations	10	100	15	1/year	18.0
Fuel sipping	2.0	100	2	1/year	0.4
Total					45

TABLE 12.4-11

OCCUPATIONAL DOSE ESTIMATES DURING INSERVICE INSPECTION

Activity	Average Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rems/ year)
Providing access: installation of platforms, ladders, etc., removal of thermal insulation	200	10	4	1/year	8
Drywell	100	50	6	1/year	30
Reactor building	5	50	2	2/year	1
Total					39

TABLE 12.4-12

OCCUPATIONAL DOSE ESTIMATES DURING SPECIAL MAINTENANCE

Activity	Average Dose Rate (mRem/hr)	Exposure Time (hr)	Number of Workers	Frequency	Dose (man-Rems/ year)
CRD replacement	260	35	5	1/year	54
CRD repair	15	200	6	1/year	20
Low power range monitor	95	20	4	1/year	9
MSIV repair	75	100	6	1/year	52
Feedwater sparger	800	60	244	1/40 years	6
RHR system	50	27	8	1/year	11
RWCU pump	[illegible]0	35	3	1/year	22
RWCU valve and heat exchanger	110	45	6	1/year	36
Turbine inspection	3	80	10	1/3 years	0.8
Turbine overhaul	3	250	20	1/20 years	0.75
Total					212

TABLE 12.4-13

PERSONNEL RADIATION DOSES FROM AIRBORNE ACTIVITY

	Routine Surveillance	Routine Operation	Instrument Calibration	Maintenance	Refueling	ISI	Total man-hrs/hr	Dose Rates (Rem/hr) WB	Dose Rates (Rem/hr) Thyroid	Yearly Man-Rem Doses (man-Rem/yr-unit) WB	Yearly Man-Rem Doses (man-Rem/yr-unit) Thyroid
Reactor building (during operation)	3,800	800	20	1,650	- - -	- -	6,270	1.2×10^{-3}	1.9×10^{-3}	7.5	11.9
Reactor building (refueling)	- - -	- - -	40	2,600	1,500	440	4,580	Neg[(1)]	3.8×10^{-5}	Neg	1.7×10^{-1}
Fuel handling building	150	- - -	20	300	1,500	- -	1,970	Neg	1.5×10^{-5}	Neg	3.0×10^{-2}
Radwaste building	150	7,400	20	300	- - -	- -	7,870	Neg	1.2×10^{-4}	Neg	9.4×10^{-1}
Turbine building	1,200	- - -	40	5,000	- - -	- -	6,240	2.5×10^{-4}	9.0×10^{-5}	1.6	5.6×10^{-1}
Total								1.5×10^{-3}	2.2×10^{-3}	9.1	13.6

NOTE:

1. Negligible.

TABLE 12.4-14

SAFETY/RELIEF VALVE DISCHARGERS DOSE FOR TYPE 2 EVENT[1]

Organ	Dose
Whole body (γ)	297 mRem/event
Skin (β)	214 mRem/event
Thyroid	.35 mRem/event

NOTE:

1. Four minute egress.

TABLE 12.4-15

ESTIMATED SKYSHINE DOSES[1]

Distance (ft)	Occupancy Factor[2]	Dose (mRem/yr)
500	0.19	258
1,500	0.19 (40 hrs/wk)	19
2,900 (Exclusion boundary)	.4 (50% occupancy)	1.6

NOTES:

1. Two units operating.

2. Plant factor (80 percent) times percentage occupancy.

TABLE 12.4-16

WORK FORCE BY OCCUPATIONAL CRAFTS

Occupational Craft	Approximate Percentage of Work Force (%)
Boilermakers	7
Electricians	18
Ironworkers	9
Pipe fitters	15
Painters	7
Laborers	11
Operating engineers	6
Carpenters	9
All other crafts	18
Total	100

TABLE 12.4-17

SUMMARY OF DIRECT DOSES[1]

Source	Distance (ft) 500	1500	2,900 Exclusion Boundary
Skyshine	258	19	1.6
Surfaces of buildings	7.2×10^{-1} (Note 2)	6.3×10^{-2} (Note 2)	2.6×10^{-3}
Radioactive plume	13.6	7.8	1.1

NOTES:

1. Doses in mRem/yr.
2. Estimated from Table 12.4-18.

TABLE 12.4-18

DOSE TO NON-PLANT PERSONNEL

Function/Location	Direct Dose Buildings	Radioactive Plume	Skyshine	Daily Dose	Dose Per Job Duration
A. Startup/test (80 workers - 7.5 mo)	(mRem/hr)	(mRem/hr)	(mRem/hr)	(mRem/day)	(man-Rem)
1. 25% of work force					
5 hrs/day in turbine building[1]	Negligible	Negligible	0.046	0.23	1.0
2 hrs/day inside plant[2]	Negligible	Negligible	0.0016	0.0032	0.015
2 hrs/day outside plant[3]	0.008	0.025	0.16	0.37	1.7
2. 75% of work force					
2 hrs/day outside plant[3]	0.0008	0.025	0.16	0.37	5.1
7 hrs/day inside plant[2]	Negligible	Negligible	0.0016	0.0112	0.15
				Total	8.0 man-Rem
B. Construction workers[5]					
1. 40% of work force					
2 hrs/day outside plant[4]	0.00026	0.008	0.06	0.14	13.6
7 hrs/day inside plant[2]	Negligible	Negligible	0.0016	0.0112	1.1
2. 60% of work force 9 hrs/day outside[4]	0.00026	0.008	0.06	0.61	88.9
				Total	103.6 man-Rem

TABLE 12.4-18 (Continued)

	Function/Location	Direct Dose Buildings (mRem/hr)	Radioactive Plume (mRem/hr)	Skyshine (mRem/hr)	Daily Dose (mRem/day)	Dose Per Job Duration (man-Rem)
C.	Site Organization[5]					
	2 hrs/day inside plant[2]	Negligible	Negligible	0.0016	0.0032	1.1
	2 hrs/day outside plant[3]	0.0008	.025	.16	.37	132.7
	5 hrs/day at warehouse office[6]	0.00026	0.008	.02	.14	50.2
					Total	184.0 man-Rem

NOTES:

1. Assume Unit 2 turbine is 666 feet from Unit 1 turbine.
2. Assume 300 feet from any radiation source plus 2 feet concrete shielding.
3. Assume 300 feet from any radiation source.
4. Assume 600 feet from any radiation source.
5. Valves based on one 9 hour shift per day, 5 days per week.
6. Assume 600 feet from radiation source for direct dose and plenum, and 1,000 feet for skyshine.

TABLE 12.4-19

DIRECT DOSE FROM THE EXTERNAL SURFACES OF BUILDINGS

Distance (ft)	Dose Rate (mRem/hr)	Occupancy Factor[1]	Annual Dose (mRem/yr)
100	2.0×10^{-3}	0.19	3.3
300	7.8×10^{-4}	0.19	1.3
600	2.6×10^{-4}	0.19	4.3×10^{-1}
1,200	5.6×10^{-5}	0.19	9.3×10^{-2}
2,000	8.0×10^{-6}	0.19	1.3×10^{-2}
2,900 (Exclusion Boundary)	7.3×10^{-7}	0.4	2.6×10^{-3}

NOTE:

1. Plant factor (80%) times percentage occupancy.

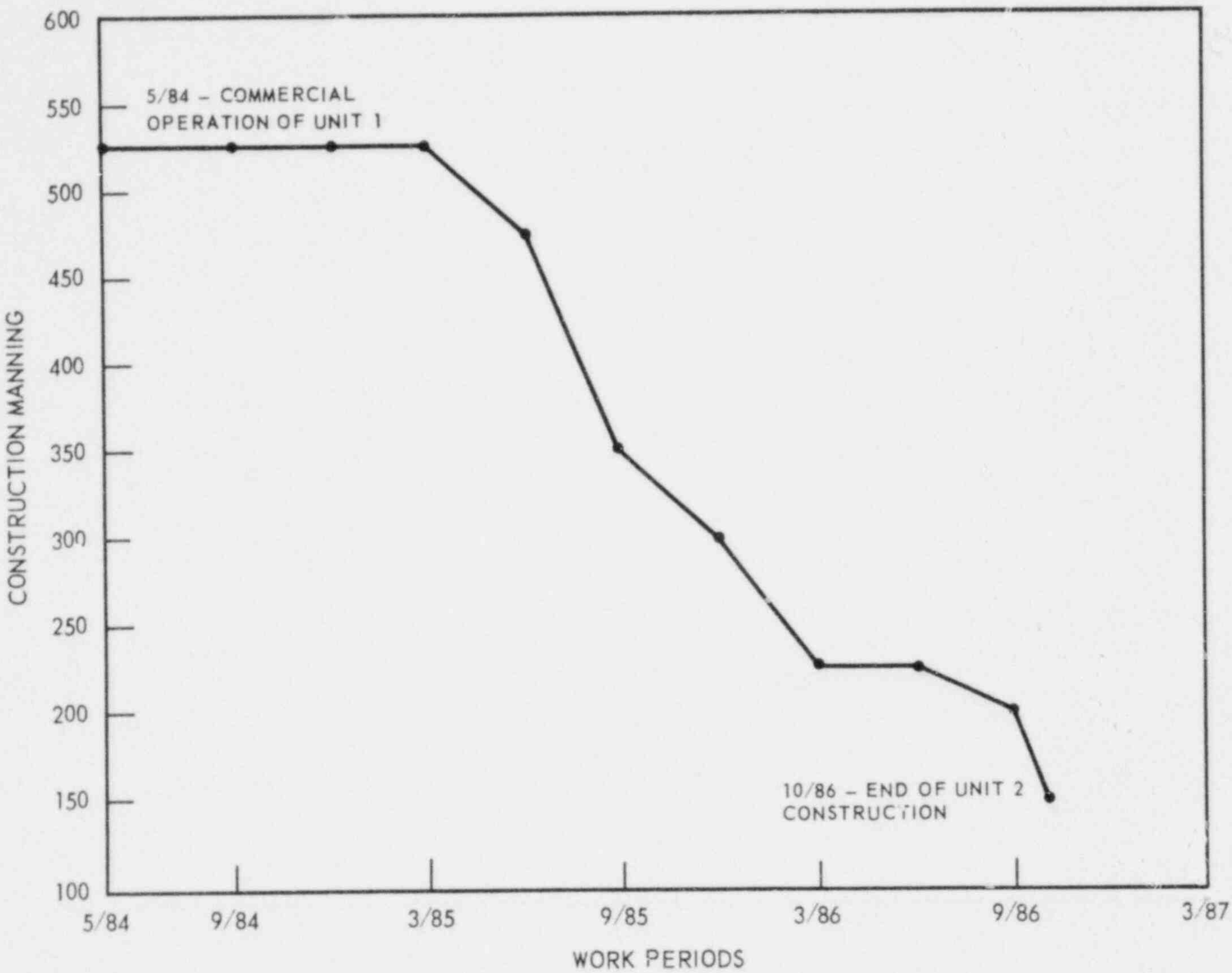
TABLE 12.4-20

DOSE FROM GASEOUS RADIOACTIVE PLUME

	Distance (ft)		
	500	1,500	2,900
Expected dose rate (mRem/hr)	8.0×10^{-3} (Note 1)	2.2×10^{-3} (Note 2)	7.0×10^{-4} (Note 3)
Expected annual dose (mRem)	13.6	7.8	1.1

NOTES:

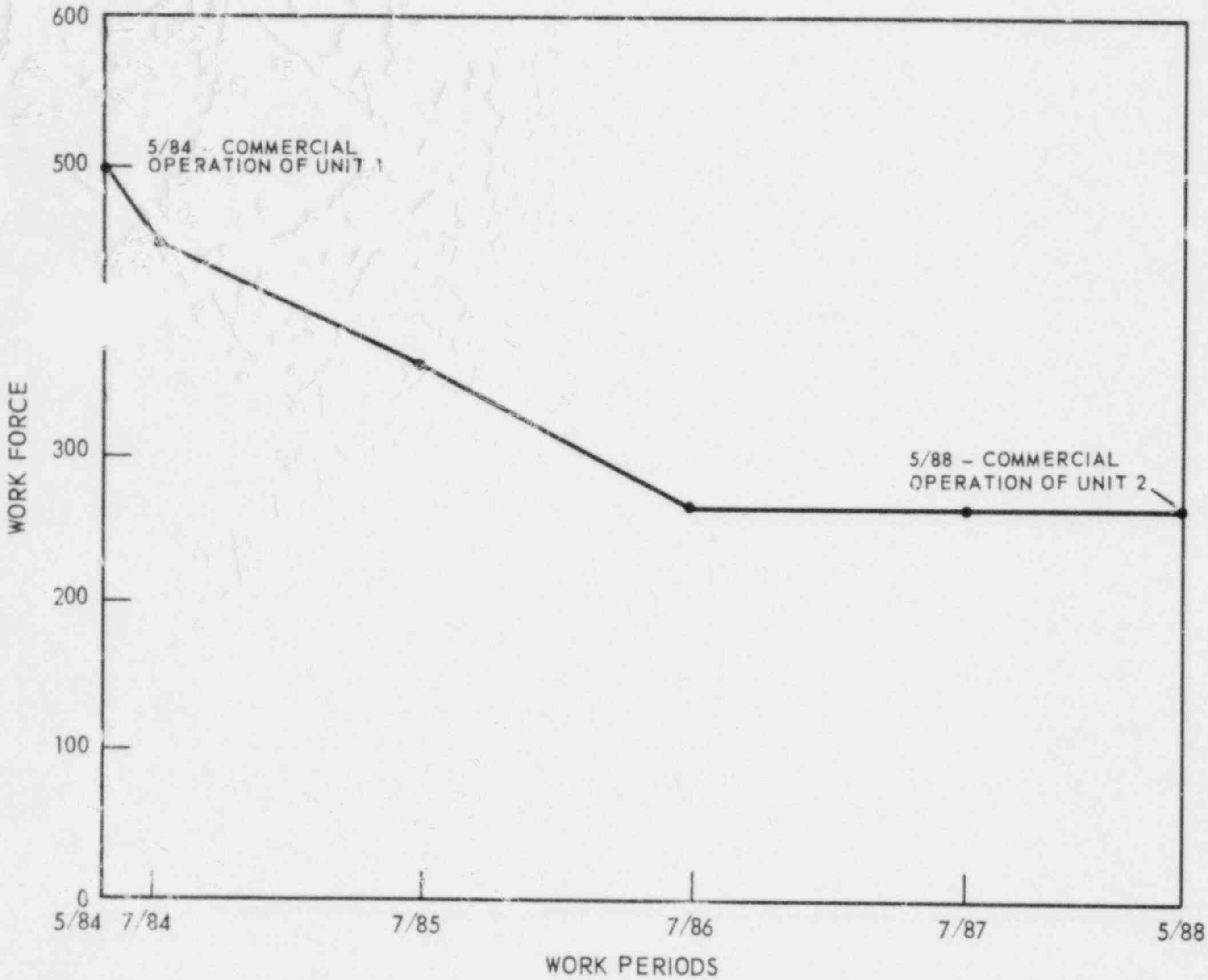
1. A χ/Q at 200 meters is used.
2. A χ/Q at 400 meters is used.
3. A χ/Q at 800 meters is used.



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC
ILLUMINATING COMPANY

Engineering and Construcion
Schedule

Figure 12.4-1



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Site Organization Manning (Without Perry Plant Department)

Figure 12.4-2

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

The personnel responsible for the Health Physics program and for handling and monitoring radioactive materials, including special nuclear material, source and byproduct materials meet the experience and qualification requirements set forth in Regulatory Guide 1.8. Detailed descriptions of the qualifications and experience is given in Section 13.1.1.

12.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

12.5.2.1 Facilities

All health physics and radiochemistry facilities are located on elevation 599'0" of the control complex. These facilities include the following rooms and areas:

a. Personnel Decontamination Room
b. Medical Aid Room
c. Health Physics Office
d. Chemistry Office
e. Health Physics and Radiation Protection Service Room
f. Counting Room
g. Low Level Chemistry Laboratory
h. High Level Chemistry Laboratory
i. Laundry and Mask Cleaning Area
j. Clothing Storage Area
k. Mens and Women Locker Rooms and Lavoratories
l. Shop Facility and Decontamination Area
m. Whole Body Counting Room
n. Conference Room

The floor plan of elevation 599'0" is depicted on Figure 12.5-1; the numbers representing each room are referenced to the above listing. Portable radiation survey equipment, airborne radiation monitoring equipment,

dosimeters, and other miscellaneous equipment will be stored in the health physics and radiation protection service room (No. 5). Respiratory protective equipment, protective clothing, and other miscellaneous anti-contamination material for personnel, will be stored in the clothing storage area (No. 10).

Locker and lavoratory facilities (No. 11) for men and women shall contain personal effect lockers for workers as well as toilet, washroom and shower acommodations. Protective clothing may be issued from the clothing storage area and/or any local access control point, as may be warranted. The protective clothing donning areas shall be provided with benches and clothes racks.

Health physics sample counting equipment is located in the health physics and radiation protection service room (No. 5). Equipment includes Geiger-Mueller beta counters, an alpha counter, an iodine counting system and calibration sources and equipment.

The radio-chemistry low level and high level laboratories (Nos. 7 and 8) are adjacent to the counting room. Analyses capabilities will include gamma isotopic analysis, beta and alpha low level analysis, and other miscellaneous special sample counting analyses.

Personnel decontamination shall be performed in the personnel decontamination room (No. 1) standup showers, a special accident victim shower for stretcher-borne personnel, and other necessary decontamination equipment shall be located in the personnel decontamination room.

Large or highly contaminated equipment will be decontaminated in the shop facility decontamination area (No. 12). Shop equipment for decontamination will include ultrasonic cleaners, wash basins, and other equipment necessary for proper decontamination. Ventilation systems are so designed as to maintain a habitable environment for personnel performing decontamination. Drainage from the decontamination areas is directed to radwaste.

12.5.2.2 Access and Exit to Health Physics Facilities

Access to and from the plant shall, usually, be through the double doors at the east end of the main corridor between the NCC heat exchanger and NCC pump areas. All contaminated materials and radioactive samples are administratively required to be brought into the access control area through these double doors. Personnel traffic, to and from the access control facility, may travel via two stairways and one elevator at the west end of the facility. Friskers will be strategically located for personnel contamination surveys. Also, a hand and foot monitor will be provided for the surveying of all personnel who enter the facility from the plant.

12.5.2.3 Health Physics Instrumentation

12.5.2.3.1 Laboratory Instruments

Instrumentation located in the counting room, health physics and radiation protection service room, low level chemistry laboratory, and high level chemistry laboratory, will allow technical personnel to ascertain the radioactive material concentrations in survey samples. Typical samples would be: contamination survey smears, airborne survey particulate filters, and charcoal halogen cartridges. Typical laboratory instruments are listed on Table 12.5-1. Each counting system will be checked and calibrated at regular intervals with radioactive sources traceable to a National Bureau of Standards (NBS) sources in accordance with health physics procedures. Counting efficiency, background count rates, and detector voltage settings will be checked periodically. Instrumentation will also be calibrated after repair.

12.5.2.3.2 Portable Survey Instruments

Portable instruments shall be stored in the health physics and radiation protection service room and, as required, at any inplant access control point for plant maintenance and/or repair. These instruments allow personnel to perform alpha, beta, gamma, and neutron surveys, for area radiation, airborne, and surface contamination monitoring.

Each portable instrument will be calibrated according to health physics procedures when in use, or after repair, prior to use. Sufficient quantities of portable instrumentation will be available to permit calibration maintenance, or repair to some of these instruments without causing a shortage of operational equipment. Typical portable equipment is listed in Table 12.5-2.

A large, heavily shielded, self-contained, multi-source calibrator will be provided for calibrating gamma dose rate instrumentation. Other sources will be provided as required. Instruments may also be calibrated by a qualified consultant. All sources used for calibration will be traceable to NBS sources.

12.5.2.3.3 Personnel Monitoring Instruments

Personnel monitoring shall be provided by use of the thermoluminescent dosimeters (TLDs), and/or film badges (FBs), direct reading pocket ionization dosimeters, neutron FBs/TLDs, and some electronic instruments. All persons entering radiation control areas will be issued TLD/FB which will be used to measure exposure to beta-gamma radiation. These badges will contain, at least, two thermoluminescent chips with suitable energy filters. TDLs/FBs will be analyzed at least monthly for personnel records, and anytime personnel doses need to be ascertained as necessitated by accident or emergency related exposures. The exposure history established by the TLD/FB readings shall constitute the official record of personnel exposure at PNPP.

When a potential for neutron exposure exists, affected personnel will be issued neutron-sensitive TLDs/FBs. These badges will provide the official record of personnel neutron exposure.

Direct reading dosimeters shall be issued to personnel as necessary for indication of exposure. These dosimeters provide "up-to-the-minute" indication of radiation exposure. These dosimeters may also be used to monitor the extremities. The dosimeters will be calibrated semi-annually, anytime damage is suspected, or if repairs are performed.

Personnel contamination survey instruments shall include Geiger-Mueller friskers, portable monitors, and hand and foot counters. These instruments will be calibrated semiannually when in use, and prior to use after repair. Personnel internal exposures will be evaluated by whole body counting as described in Section 12.5.3.6.2.

Typical personnel monitoring instruments are listed on Table 12.5-3.

12.5.2.3.4 Health Physics Equipment

Portable air samples are used to determine airborne radioactive material concentrations. Air samplers will be calibrated for flow annually. Surveys will be performed for particulate and radioiodine airborne concentrations.

Portable continuous air monitors will normally be located in the common refueling area, solid waste drumming area (radwaste), the turbine operating floor, and in the heater bay. Local information and trend indication is provided. Alarm setpoints are variable in accordance with health physics procedures. Audible and visual alarms are provided to warn local personnel of airborne radioactive concentrations in excess of specified limits.

Respiratory equipment will be provided and stored in the clothing storage area, service area, or any remote controlled access point in the plant, as required. Emergency respiratory equipment shall be stored at strategic locations within the plant. The equipment will be maintained and used in accordance with applicable health physics procedures.

Protective clothing will be provided for personnel working in radiologically controlled areas. Specific requirements for clothing will be prescribed by health physics personnel based on actual or anticipated radiological conditions. An adequate inventory of protective clothing will be maintained in the health physics facility, and at any plant controlled access point necessary to support plant maintenance or operations activities. This clothing includes: coveralls, booties, hoods, rubbers, rubber gloves, cotton glove liners, waterproof suits, disposable jump suits, and rain gear.

Contamination control supplies will include: cleaning cloths, drum liners, plastic bags, wet/dry vacuum cleaners, step-off pads, masking tape, radiation tape, radiation rope, various signs, mops, sponges, paper towels, absorbent bench top paper, smears, and other supplies necessary for good radiological housekeeping.

A typical listing of health physics equipment is provided in Table 12.5-4.

12.5.2.3.5 Other Health Physics Instruments

The area radiation monitoring system (ARM) is installed in areas where it is desirable to have constant dose rate indication. Area monitors display the dose rates of strategic plant locations locally and on control room panel modules. The area monitors provide audible and visual alarm indication when a preset dose rate is exceeded. Fixed process monitors (PRM) are provided to monitor the radioactive contamination levels in selected plant process systems. Audible and visual alarms actuate when present radioactive concentrations are exceeded. Some process monitors initiate automatic control functions and/or trips of selected process systems. Fixed airborne radiation monitoring (ABRM) is provided for strategic plant locations when personnel exposure to airborne gaseous and particulate radionuclides and radioiodine may be anticipated.

12.5.2.3.6 Emergency Equipment

Sufficient health physics equipment will be available to personnel responding to an accident. The equipment includes, dosimeters, portable survey meters, respiratory protection devices, protective clothing and air samplers. The equipment will be stored in strategic locations at the plant (control room, main guard house and service water building) and will be stored at, or readily transportable to, the offsite emergency control center. This equipment will be inventoried and checked for operability at regular, scheduled intervals.

12.5.3 PROCEDURES

Adherence to PNPP health physics procedures will help assure that personnel radiation exposures are kept ALARA, and within the limits of 10 CFR 20.

12.5.3.1 Radiation and Contamination Surveys

Health Physics personnel will normally perform radiation and contamination surveys of all accessible areas in the plant. The surveys will be performed on an appropriate frequency, depending on the probability of radiation and contamination levels changing, and the frequency with which the areas are visited. Surveys related to specific operations and maintenance activities may be performed prior to, during, and/or after the activity, based on information required to keep radiation exposures ALARA.

12.5.3.2 Procedures and Methods Assuring ALARA

Section 12.1.3 described the operational considerations to keep radiation exposures ALARA.

Methods to maintain exposures ALARA will not only be included in radiation work permits, but will also be contained in applicable operating and maintenance procedures. Some examples of methods that will be used to maintain exposures ALARA are discussed for the following operations:

12.5.3.2.1 Refueling

After the reactor coolant system is depressurized, it will be degassed and sampled to verify that the gaseous activity is low prior to removing the reactor head. After flooding the reactor well and the refueling pool, purification of the pool water shall be continued to maintain minimal radioactivity in the water and, therefore, radiation exposures ALARA.

Movement of spent fuel assemblies will be done with the assemblies under a sufficient depth of water to provide shielding to keep radiation exposures ALARA.

12.5.3.2.2 Inservice Inspection

Review of system drains; maintenance inspection histories, pictures and radiological survey reports may be required as preparation for entering radiological control areas. An RWP will be issued to specify the radiological protection required to keep personnel exposures ALARA.

12.5.3.2.3 Radwaste Handling

Radiation exposure to personnel handling radwaste will be ALARA due to plant design. Liquids, spent filter media, and resins are remotely processed in shielded cubicles. Filled drums of solidified waste can be remotely surveyed for contamination and, if necessary, decontaminated. The drums will be loaded on trucks for shipment using an indexed crane and TV camera system.

12.5.3.2.4 Spent Fuel Handling, Loading, and Shipping

All movements of unshielded spent fuel will be done under at least eight feet of water to provide shielding and cooling. The water is purified to reduce the concentration of radioactive materials. After the fuel is loaded into the shipping cask, the cask will be decontaminated using a pressurized water spray.

12.5.3.2.5 Normal Operations

Major radiation hazards in the plant are minimized by plant design. Equipment containing large quantities of radioactive materials are housed in shielded cubicles and are operated remotely to keep operator radiation exposures ALARA.

An area radiation monitoring (ARM) system is provided so personnel can move about and work in the plant with reasonable assurance that radiation levels are below those requiring special monitoring precautions. The ARM system indicates, alarms, and records abnormal gamma radiation levels in areas where radioactive materials may be present, stored, handled, or inadvertently introduced.

The plant ventilation system is designed to keep areas of possible radioactive contamination at a negative air pressure to prevent spreading contamination. Airflow from most areas is monitored for particulate, iodine, and gaseous radioactivity. The airborne radiation monitoring system indicates, alarms, and records abnormal airborne radioactivity levels, allowing personnel to move about and work in the plant with reasonable assurance that airborne radioactivity concentrations are below those requiring special monitoring.

12.5.3.2.6 Routine Maintenance

All maintenance work at PNPP that involves systems that contain, collect, store, or transport radioactive materials and may cause radiation exposure will require a radiation work permit (RWP) as per health physics procedures. The RWP will specify radiological hazards associated with a job and the safety precautions required for performing the job.

When applicable, procedures will specify portions of radioactive systems or components which are to be isolated, flushed, and/or drained. This will reduce the radiation levels in the maintenance area.

Where applicable, special tools will be used for remote handling of components. This will help keep radiation exposures ALARA by increasing the distance between the workers and the sources of radiation, and by decreasing the workers exposure time. Temporary shielding will be used as deemed necessary by the health physics unit.

12.5.3.2.7 Sampling

Periodic sampling of process streams will help keep radiation exposures ALARA. Analysis of samples will help verify that process stream monitors are accurate and are providing reliable information.

Most of the sampling of radioactive systems will be performed in chemical fume hoods. The fume hoods provide negative air pressure and prevent the spread of possible contamination. Appropriate protective clothing and equipment will be specified in the sampling procedures. Where applicable, survey meters will be used to monitor radiation levels at the fume hood and on the sample container. The possibility of radioactive spills and radiation exposure will be maintained ALARA during sample transport by the use of special shielded or remote handling and transportation devices.

12.5.3.2.8 Calibration

Periodic calibration of radiation detection instruments will help keep radiation exposures ALARA by assuring that the instruments are accurate and are providing reliable information.

Portable survey meters will be calibrated using an enclosed and shielded calibrator. Although the calibrator can calibrate instruments up to approximately 500 R/hr, it is shielded so that the external radiation level is about 0.1 mR/hr at one meter. Portable sources used to calibrate fixed instruments like the area radiation monitoring system are in shielded containers that slip over the detectors, keeping radiation exposure to personnel ALARA.

12.5.3.3 Controlling Access and Stay Time

Personnel who have not satisfactorily demonstrated comprehension of the information presented in general employee training will not be allowed to enter radiological control areas unless escorted by someone who has completed the General Employee Training Program.

All maintenance work performed on systems that contain, collect, store or transport radioactive materials and may cause radiation exposure will require a radiation work permit (RWP). The RWP will specify the radiological hazards associated with a job, including the radiation exposure rate. If necessary, full time health physics coverage will be provided and appropriate "stay time" will be used to maintain radiation exposures ALARA.

For additional details, see Section 12.5.2.2.

12.5.3.4 Contamination Control

Contamination surveys will be performed as described in Section 12.5.3.1. Contamination limits for areas, tools, and clothing are described in the PNPP Radiation Protection Manual.

Plant design will also help prevent the spread of contamination. The ventilation systems will keep a negative air pressure in areas of possible contamination. By keeping air flow in these areas, the possibility of contamination leaving the areas will be minimized.

Personnel friskers will be placed at strategic locations throughout the plant. Step-off pads will be used to differentiate between a potentially contaminated area and a clean area. After removing protective clothing, workers will frisk themselves. If contamination is found, the personnel will be decontaminated as quickly as possible to minimize the radiation exposure and prevent the spread of the contamination.

Specially marked tools will normally be used in contaminated areas. All tools used in contaminated areas will not be taken from controlled areas unless surveyed and released by health physics personnel.

12.5.3.5 Training Programs

All individuals working in or frequenting any portion of a radiation control area shall be kept informed of the storage, transfer, or use of radioactive materials and/or of radiation levels in the areas, and will be instructed in:

a. The health protection problems associated with radiation or radioactive materials

b. The precautions or procedures to minimize exposure

c. The purpose and use of protective devices

d. The appropriate response to warning signals made in the event of an unusuai occurrence or malfunction at the plant

As a minimum, the above listed information will be present in general employee training. Personnel who have not satisfactorily demonstrated comprehension of the information presented in the general employee training will not be allowed to enter radiation control areas unless escorted by someone who has satisfactorily completed the training.

12.5.3.6 Personnel Dosimetry

12.5.3.6.1 Personnel External Exposures

Personnel who are regularly assigned to the PNPP site are required to wear dosimeters any time they are within the protected areas of the plant. Visitors and support personnel will be issued dosimeters if their work requires entrance into radiation control areas as per 10 CFR 20.202.

Thermoluminescent dosimeters (TLDs) and/or film badges (FBs) are the primary and official method of measuring an individuals occupational radiation exposure to the whole body.

Exposure data for all personnel will be recorded on Form NRC-5, or the equivalent. Occupational exposures incurred by individuals prior to working at PNPP will be summarized on Form NRC-4, or the equivalent. These records will be maintained by CEI and will be preserved indefinitely or until the NRC authorizes their disposal. Current exposure status will be made available to each supervisor to assist in keeping individual radiation exposures ALARA. Each worker will receive an update of his exposure at least quarterly.

12.5.3.6.2 Internal Exposures

All personnel who take part in the respiratory protection program (wear respirators) will be whole body counted or have a bioassay at least once per year. Personnel who frequently use respirators, or are suspected of having an accidental exposure to airborne radioactivity, may be bioassayed or whole body counted more often. The results of the whole body counting or bioassays will be maintained with, and become part of, an individual's dosimetry file.

12.5.3.7 Evaluation and Control of Potential Airborne Radioactivity

Portable air samplers, air monitors, and fixed air monitors are used to determine the concentrations of airborne radioactivity in the plant. Particulate filters and charcoal cartridges from the samplers and monitors will be analyzed in the health physics service room or the counting room using equipment described in Sections 12.5.2.1 and 12.5.2.3.1. Samples from the continuous air monitors will be changed and analyzed at least weekly. Where applicable, air samples will be taken as a routine part of radiological surveys as described in Section 12.5.3.1.

For additional details, see Sections 12.5.2.3.4 and 12.5.2.3.5.

Training in the use and care of respiratory protection devices will be given by a qualified and experienced instructor. The instructor will develop a training program based on the hazards to be encountered and the types of respirators to be worn. Training will be given to personnel who will use

respirators and to the users' supervisors. All personnel who are expected to continue using respirators will be retrained at least annually to retain a high degree of proficiency and help maintain radiation exposures ALARA.

12.5.3.8 Radioactive Material Handling and Storage Methods

Handling of radioactive samples is described in Section 12.5.3.2.7. Various other types and quantities of radioactive sources are used to calibrate equipment. Recognized methods for the safe handling of radioactive materials, such as those recommended by the National Council on Radiation Protection and Measurements, will be used to maintain potential external and internal radiation exposures ALARA.

Sealed radionuclides having activities greater than the amounts listed in Appendix C of 10 CFR 20 will be subject to controls for radiological protection. Radioactive sources that are subject to the controls will be used or handled by, or under the direction of, health physics personnel. Individuals using these sources will be familiar with the radiological restrictions, regulations, and limitations placed on their use. These limitations will help protect both the user and the source integrity.

TABLE 12.5-1

LABORATORY EQUIPMENT

Type	Quantity	Range	Sensitivity/Accuracy	Remarks
Alpha Counter	1	$1\text{-}10^6$ CPM	80% of 2π (Pu-239)	ZnS (Ag) Phosphor
Beta Counter	2	$1\text{-}10^6$ CPM	12k CPM per mR/hr CO^{60}	G-M tube
Iodine Counter	1	(Later)	(Later)	(Later)
Gamma Spectrometer	1	(Later)	(Later)	(Later)

TABLE 12.5-2

PORTABLE SURVEY INSTRUMENTS

Type	Quantity	Range	Sensitivity/Accuracy	Remarks
G-M Pancake	5	0-70K CPM	5,000 CPM/mR/hr	-
G-M Hand	5	0-70K CPM	2,500 CPM/mR/hr	mR/hr and CPM
G-M mR/hr	5	0.1-2,000 mR/hr	±10% of Full Scale	Wide Range
Ion Chamber mR/hr	5	1-1,000 mR/hr	±10% From 60 KEV to 1.3 MEV	-
mR/hr Telescoping	2	0.01 R/hr - 999 R/hr	±15% From 70 KEV to 1.3 MEV	-
Cutie Pie (Type)	4	0-1,000 R/hr	±15% of Full Scale	-
mR/hr Neutron	2	0-5K mR/hr	.025 EV (Thermal) to approx. 10 MEV	-
R-Chamber	1	0-200R	-	For calibration
Alpha-Proportional Counter	2	0-50,000 CPM	46% of 2 π	-

TABLE 12.5-3

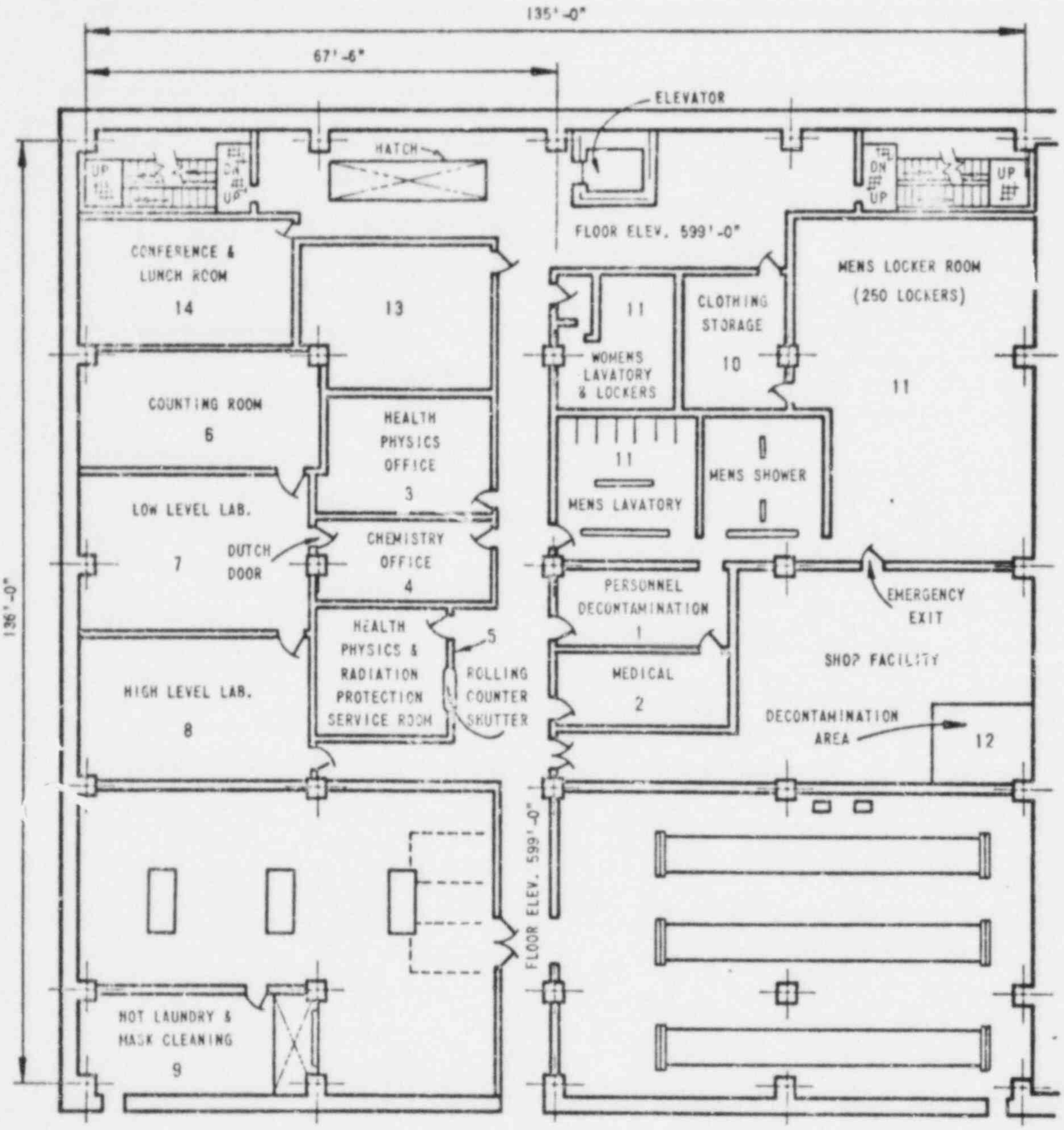
PERSONNEL MONITORING INSTRUMENTS

Type	Quantity	Range	Sensitivity/Accuracy
TLD	(Later)	10mR-3000R	±10%
TLD (Neutron)	(Later)	10mR-3000R	±30%
Dosimeter, Pocket	500	0-200 mRem	±20%
Dosimeter, Pocket	20	0-1 Rem	±20%
Dosimeter, Pocket	10	0-5 Rem	±20%
Dosimeter, Pocket	10	0-20 Rem	±20%
Dosimeter Charger	4	N/A	N/A
Portal Monitors	3	160-7000 CPM	(Later)
Hand and Foot Monitors	1	0-50,000 CPM	(Later)
Laundry Monitors	1	160-7000 CPM	(Later)
Personnel Friskers	20	0-500K CPM	±10%

TABLE 12.5-4

HEALTH PHYSICS EQUIPMENT

Type	Quantity	Range	Remarks
Large Calibrator	1	75mR/hr-500R/hr	50 curies; CS-137 traceable to NBS standard
MSA Combustible Gas and Indicator	2	(Later)	
High Vol. Air Sampler	3	Approx. 30 CFM	
Low Vol. Air Sampler	3	Approx. 3 CFM	
Self-Contained Breathing Apparatus	10	<50 MPC(D)	10,000(PD)
Half Face Masks	75	<5 MPC (Particulate)	-
Half Face Filters	600	N/A	
Full Face Masks	50	<10 MPC	<2000 (CF and PD)
Full Face Filters	300	N/A	-
Respirator Hoses	20	N/A	-
Respirator Junction Box	3	N/A	To 6 Respirators Per Junction Box
Waterproof Suits	30	N/A	-
Disposal Jump Suits	150	N/A	-
Disposal Rain Suits	150	N/A	-
Clothing Sets (Coveralls, Hoods, and Booties)	3,000	N/A	-
Rubbers (Pr.)	3,000	N/A	-
Rubber Gloves (Pr.)	5,000	N/A	-
Cotton Gloves (Pr.)	2,000 Doz.	N/A	-
Vacuum Cleaners	2	N/A	Wet/Dry-55 Gal.



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Facility Locations

Figure 12.5-1