THE CLEVELAND ELECTRIC ILLUMINATING COMPANY
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Dalwyn R. Davidson
VICE PRESIDENT
SYSTEM ENGINEERING AND CONSTRUCTION

September 8, 1981

RECEIVED
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U.S. NUCLEAR REGULATORY COMMISSION

Mr. Robert L. Tedesco
Assistant Director for Licensing
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Perry Nuclear Power Plant
Docket Nos. 50-440; 50-441
Response to Request for
Additional Information -
Radiation Protection

Dear Mr. Tedesco:

This letter and its attachment is submitted to provide final responses to your letter dated June 11, 1981 concerning radiation protection and radiological assessment. It is our intention to incorporate these responses in a subsequent amendment to our Final Safety Analysis Report.

Very truly yours,

Dalwyn R. Davidson

Dalwyn R. Davidson
Vice President
System Engineering and Construction

DRD:dlp

Attachment

cc: M. D. Houston
G. Charnoff
NRC Resident Inspector

B001
S.1/1

471.06
(13.1)

In the Perry Nuclear Power Plant (PNPP) organization, Figure 13.1-3 the Supervisor, Health Physics Unit (Radiation Protection Manager (RPM) in Regulatory Guide 8.8), reports to the Radiation Protection Section General Supervising Engineer, who reports to the Plant Manager.

The RPM should have direct access to responsible management personnel and be independent of operating divisions as specified in Regulatory Guide 8.8, Section C.1.b(3) and in NUREG-0731 "Criteria for Utility Management and Technical Competence." Section 13.1 or Section 12.5 of the FSAR should be revised to show that in health physics matters, the Health Physics Supervisor has direct access to the Plant Manager.

Response

The Supervisor, Health Physics Unit, is the Radiation Protection Manager (RPM) in Regulatory Guide 8.8. See revised Section 13.1.2.2.

General Supervising Engineer, Technical Section

The General Supervising Engineer, Technical Section is responsible for directing all activities associated with providing technical support and services related to monitoring plant performance, equipment and system testing, instrument maintenance, calibration and repair and reactor technology. He is also responsible for the programming, operation and maintenance of the process computer and related software development. The General Supervising Engineer, Technical Section is a member of the Plant Operations Review Committee and reports to the Superintendent, Plant Operations.

General Supervising Engineer, Radiation Protection Section

The General Supervising Engineer, Radiation Protection Section is responsible for directing all activities associated with the chemical, radiochemical, radwaste and other radiological control services required to support plant operation and maintenance activities. This includes conducting laboratory and plant survey activities required to ensure that personnel exposure to radiation and radioactive materials is within regulatory guidelines and that such exposure is kept as low as reasonably achievable (ALARA). The General Supervising Engineer Radiation Protection Section is a member of the Plant Operations Review Committee, ALARA Committee, and reports to the Plant Manager.

Supervisor, Health Physics Unit

The Supervisor, Health Physics Unit is designated the Radiation Protection Manager (RPM) and is responsible for development and implementation of the radiation protection program for the plant. This includes supervising all health physics activities, monitoring plant radiation health and safety practices reviewing all health physics instructions, establishing the ALARA program and supervising preparation of reports and manuals. The Supervisor, Health Physics Unit, is a member of the Plant Operations Review Committee and reports to the General Supervising Engineer, Radiation Protection Section. The RPM has direct recourse to the Plant Manager in order to resolve questions related to the conduct of the radiation protection program. 471.06, 471.09, 471.10

471.07 (13.1) Based on information contained in NUREG-0731 "Criteria for Utility Management and Technical Competence," it is our position that your organization chain contain a qualified health physicist to provide backup in the event of the absence of the Supervisor, Health Physics. The December 1979 revision of ANSI 3.1 specifies that individuals temporarily filling the RPM position should have a B.S. degree in science or engineering, two years experience in radiation protection, one year of which should be nuclear power plant experience, six months of which should be onsite. It is our position that such experience be professional experience. Identify and provide an outline of the qualifications of the individual who will act as the backup for the RPM in his absence.

Response

The Supervisor, Health Physics Unit, is designated as the Radiation Protection Manager (RPM). The Radiation Protection Section General Supervising Engineer will act as the backup for the RPM in his absence. The following are their resumes of experience and training:

James E. Bontempo; Supervisor, Health Physics Unit

Education

Miami Dade Community College, A.A., PreEngineering, 1977
U.S. Army Engineer Reactors Group, Health Physics Speciality, 1965.

Experience

January, 1980 to Present: Health Physics Supervisor, Cleveland Electric Illuminating Co., Perry Nuclear Power Plant (2, 1205 MWe BWRs). Assist the General Supervising Engineer, Radiation Protection Section in the development of the Radiation Protection Program for the Perry Nuclear Power Plant. This includes facility and equipment reviews, procedure preparation, and developing Health Physics Technician staffing and training. Assigned to the Edison Electric Institute's Health Physics Committee, 1981.

July 1977 - January 1979: Station Health Physicist, Houston Lighting & Power Company, South Texas Project (2-1250 MWe PWRs) and Allen's Creek Project (1-1200 MWe BWR). Assist in the development of a Radiation Protection Program for the company nuclear projects. This included writing of health physics procedures, developing health physics technician training courses, and performing ALARA reviews of facility and equipment design.

June, 1976 - June, 1977: Health Physics Shift Supervisor, Florida Power & Light Company, Turkey Point Plant (2-760 MWe PWRs). Responsible for Radiation Protection Men activities for radiological surveys. Initiate, review, and authorize Radiation Work Permits; ship and receive radioactive materials; control of personnel exposure; direct health physics coverage for the following functions: refueling operations, steam generator eddy current testing and tube plugging, equipment maintenance, calibration and repair of systems instrumentation, radioactive laundry and waste processing; write and review health physics operating procedures.

May, 1974 - June, 1976: Health Physics Administration Assistant, Florida Power & Light Company, Turkey Point Plant. Responsible for all phases of personnel exposure records; write and maintain computer programs and associated files (Fortran IV); research and order health physics supplies and equipment; draft various health physics reports; conduct basic and refresher training in health physics fundamentals for vendor, maintenance, and operations personnel.

October, 1968 - May, 1974: Radiation Safety Office, The Ohio State University. Provide radiological health services to operations staff of the Nuclear Reactor Laboratory. Perform analysis on water and air samples, conduct area and irradiated materials surveys. Initiate and maintain environmental surveillance p gram for the

Nuclear Reactor Laboratory. Review and recommend radiological practices for medical and reasearch applications for use of radioisotopes. Generally oversee Broad License requirements and ensure compliance with Nuclear Regulatory Commission regulations at The Ohio State University.

December, 1967 - October, 1968: Health Physics - Plant Chemistry Specialists, Plant Operator, SM-1 Nuclear Power Plant, Fort Belvoir, Virginia. Provide health physics services and advice to operating crew, plant management, and maintence personnel. Perform water analysis and treatment on potable, reactor associated, and secondary water systems. Operate equipment and plant components, participate in core loading, control rod maintenance, and reactor physics testing.

June, 1966 - December, 1967: Health Physics - Plant Chemistry Specialist, Plant Operator, PM-3A Naval Nuclear Power Plant, McMurdo Sound, Antarctica. Provide health physics services and advice to operating crews, plant management, and maintenance personnel. Perform water analyis and treatment on potable, reactor disposal system for liquid and gaseous wastes. Operate plant systems and components. Operate desalinization unit for producing fresh water from sea water.

February, 1965 - June, 1966: Health Physics - Plant Chemistry Instructor, Nuclear Power Plant Operator School, Fort Belvoir, Virginia. Give instruction in health physics theory and practices; counting and monitoring equipment theory and operation; radiochemistry, inorganic, qualitative, and quantitative chemistry; and gamma spectroscopy. Write lesson plans and student handouts for portions of these courses.

Steven F. Kensicki, General Supervising Engineer - Radiation Protection Section

Education

Bachelor of Science Degree in Chemical Engineering, University of Detroit, 1968.

Registered Professional Engineer, State of Ohio, Serial Number E-42618

Experience

January 1978 to present: General Supervising Engineer, Radiation Protection Section, Cleveland Electric Illuminating Company, Perry Nuclear Power Plant. Responsible for directing all activities of the Health Physics, Chemistry, and Radwaste Units. Developing the Radiation Protection, Chemistry, and Radwaste Management Programs for the Perry Plant. This includes facility and equipment reviews, procedure preparation and approval, and developing supervisory and technician training and staffing requirements. Assigned to the Electric Power Research Institute, Nuclear Engineering and Operations Task Force, Chemistry, Radiation, and Monitoring Subcommittee, 1981.

June 1976 - January 1978: Engineer, Cleveland Electric Illuminating Company. Nuclear Engineering Department. Participated in the design review of radwaste and water treatment systems. Responsible for the design of the preoperational chemical cleaning systems and the layout of the laboratory and health physics facilities for the Perry Plant.

October 1972 - June 1976: Engineer, Cleveland Electric Illuminating Company, assigned to Toledo Edison Company, Davis Besse Nuclear Station, Operations Section. Responsibilities included writing startup, operating, periodic, and surviellance test procedures. Procedures were prepared for all phases of plant operation including NSSS, Radwaste, and Secondary Systems. Served as a test leader during station startup. Completed all aspects of Health Physics training for station employees.

May, 1968 - October, 1972: Junior Engineer, Associate Engineer, Cleveland Electric Illuminating Company, Chemical Engineering Unit. Participated in the startup of two 650 megawatt supercritical fossil fired electrical generating units. Responsible for the preoperational cleaning, checkout, initial operation, and operator training on all water treatment and analytical sampling equipment. The equipment included high flow rate deep bed demineralizers with external regeneration, condensate filtration equipment, and sodium, oxygen, silica, conductivity, and PH continuous flow analytical equipment.

Formal Training

1973-74 Nuclear Technology Course for Power Plant Engineers, General Physics Corporation, 152 hours.

Major areas of study included; general physics review, BWR and PWR primary cycles, nuclear reactions, radioactivity, interaction of radiation with matter, principles of radiation detection, reactor core physics, reactor operations, water chemistry, health physics requirements, and PWR technology.

1976 Research Reactor Operation Training, University of Wisconsin Nuclear Reactor, 80 hours.

Class included 46 hours of operation observation plus 28 hours of laboratory exercises concerned with radiation, instrumentation, and radiation safety. Reactor operation consisted of 15 startups and shut downs with 9.25 console operating hours. Laboratory exercises included work with health physics survey instruments, N-16 and AR-41 production demonstration, shielding calculations, radiochemistry instruments, and decontamination procedures.

1976 Perry Plant Design and Fundamentals Course, Perry Plant Department, 120 hours.

Course consisted of classroom lectures on the Perry nuclear steam supply system, auxiliary systems, and typical operating evolutions. The lectures addressed design principles, system components, system operation, logic, typical control setpoints, integrated plant operation, and processing and monitoring of radioactivity.

1977 Nuclear Science Heidelberg College, 3 credit hours. Major areas of study included; nuclear structure, radiation detection, radiation safety, radiochemical separation and sampling techniques, nuclear spectroscopy, and neutron activation. Laboratory experiments included:

- Introduction of Geiger counting - half life
- Introduction to sample preparation and statistics
- Beta energy measurements
- Sample preparation - backscatter and self absorption
- Gamma Spectroscopy - single and multichannel analyzer
- Liquid Scintillation Counting.

1978 BWR Chemistry, General Electric 12 weeks.

Intensive instruction and practical experience in both radiochemical and chemical analysis for process control, waste disposal, effluent monitoring, process and laboratory instrument calibration and evaluations. Course covered material on compliance with and interpretation of the chemical and radiochemical aspects of BWR plant technical specifications. Laboratory work consisted of working with u Ci quantities of most fission and activation products including a 65 u Ci Sr-90, 10 m Ci Co-60, 45 m Ci Yb-169, and a 68 u Ci Mo-99-Tc 99 liquid sources.

1979 Radiation Protection, Engineering Technology, Inc., 40 hours.

One week course to provide an overview of the basic scientific and engineering principles underlying radiation protection. Topics included; radiation physics, biological effects of radiation, measurement of radiation, applied radiation control, legal basis of radiation exposure, and environmental monitoring. Course text included "Radiation Safety Technician Training Course" of Argonne National Laboratory.

1980 Radiological Engineering, General Electric, 8 weeks.

Course consisted of classroom lectures, field exercises and projects. Heavy emphasis was placed on a vigorous approach to calculational methods and the direct application of health physics to the day-to-day operating plant problems. Topics covered include; external gamma and beta dosimetry, biological effects of radiation, internal contamination and dosimetry (ICRP 2&10), health physics instrumentation, contamination control, radiation exposure control, BWR health physics operation, and BWR accident analysis. See attached instructor evaluation.

471.08 (13.1.3.2) Regulatory Guide 1.8, states "The RPM should have a bachelor's degree or the equivalent in a science of engineering subject including some formal training in radiation protection" and at least 5 years of essional experience in applied radiation protection. It is our position that equivalent as used in Regulatory Guide 1.8 for the bachelor's degree means (a) four years of formal schooling in science or engineering (b) four years of applied radiation protection experience at a nuclear facility, (c) four years of operational or technical experience or training in nuclear power, or (d) any combination of the above totaling four years.

From the information submitted in Resume Number 27, we are unable to determine that the Supervisor, Health Physics Unit, has training and experience equivalent to that specified in Regulatory Guide 1.8. Therefore, justify the selection of the individual delineated for this postion based on his training and experience and specify, as required, how he will achieve the aforementioned experience or training, prior to the plant being licensed, to qualify as the RPM.

<u>Response</u>

The Supervisor, Health Physics Unit (James Bont..po), will attend formal training that covers the material listed below to qualify as the RPM. The scope of subjects will be as listed or equivalent. The training will be completed six months prior to fuel loading.

a. Internal Dosimetry for Fixed Nuclear Facilities
   This training provides instruction in techniques for calculating the radiation dose from internally deposited radionuclides.

b. Applied Health Physics

This training will emphasize the fundamentals of Health Physics, problems and practices in providing Radiation Protection. The mechanism of radiation damage, and methods and procedures for evaluating radiation hazards.

c. Systems Training

Systems training program developed for plant personnel that require a working knowledge of systems and their interface. The course material and presentations will be developed and conducted by Perry Simulator SRO certified instructors.

471.09
(13.1.2.2)

Provide an outline of the function, responsibility, and authority of the Perry Nuclear Power Plant's Health Physics Supervisor (RPM). (See Regulatory Guide 8.8, Section C.1.b(3) for examples of some duties).

Response

The Supervisor, Health Physics Unit, is the Radiation Protection Manager. An outline of his function, responsibilities, and authority is provided in revised Sections 12.1.1, 12.5.1 and 13.1.2.2.

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 supervising all health physics activities, monitoring plant radiation health and safety practices, reviewing all health physics instructions, and supervising the preparation of reports and manuals required by the Company and regulatory agencies.

471.09

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

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

The Radiation Protection Section General Supervising Engineer (RPSGSE) is responsible for directing the implementation of the plant (site) radiation protection program, which encompasses the handling and monitoring of radioactive materials, including special nuclear, source, and by-product materials. He is also responsible for assuring that plant operation meets the radiation protection requirements of federal and state regulations which are applicable to the radiation protection program. He reports to the Plant Manager, Perry Nuclear Power Plant (See Figure 13.1-3 and Section 13.1).

471.09, 471.10

Reporting to the RPSGSE are the Health Physics, Radwaste and Chemistry Unit Supervisors. The Health Physics Unit Supervisor is responsible for developement and implementation of the radiation protection and ALARA program for the plant.

The Health Physics Technicians who report to the Health Physics Unit Supervisor, perform the various surveys and analyses for health physics protection. One health physics personnel is provided for each shift.

For a more detailed discussion of the responsibilities and authority of the supervisory positions mentioned above, and of the training and qualifications of the personnel holding these positions, refer to Sections 13.1.2.2 and 13.2.3 and Appendix 13A.

12.5.2 EQUIPMENT, INSTRUMENTATION, AND FACILITIES

12.5.2.1 Facilities

All health physics and radiochemistry facilities are located on elevation 599'0" (B1) of the Control Complex. These facilities include the following rooms and areas:

a. Personnel Decontamination Room
b. Medical Aid Room

General Supervising Engineer, Technical Section

The General Supervising Engineer, Technical Section is responsible for directing all activities associated with providing technical support and services related to monitoring plant performance, equipment and system testing, instrument maintenance, calibration and repair and reactor technology. He is also responsible for the programming, operation and maintenance of the process computer and related software development. The General Supervising Engineer, Technical Section is a member of the Plant Operations Review Committee and reports to the Superintendent, Plant Operations.

General Supervising Engineer, Radiation Protection Section

The General Supervising Engineer, Radiation Protection Section is responsible for directing all activities associated with the chemical, radiochemical, radwaste and other radiological control services required to support plant operation and maintenance activities. This includes conducting laboratory and plant survey activities required to ensure that personnel exposure to radiation and radioactive materials is within regulatory guidelines and that such exposure is kept as low as reasonably achievable (ALARA). The General Supervising Engineer Radiation Protection Section is a member of the Plant Operations Review Committee, ALARA Committee, and reports to the Plant Manager.

Supervisor, Health Physics Unit

The Supervisor, Health Physics Unit is designated as the Radiation Protection Manager (RPM) and is responsible for development and implementation of the radiation protection program for the plant. This includes supervising all health physics activities, monitoring plant radiation health and safety practices reviewing all health physics instructions, establishing the ALARA program and supervising preparation of reports and manuals. The Supervisor, Health Physics Unit, is a member of the Plant Operations Review Committee and reports to the General Supervising Engineer, Radiation Protection Section. The RPM has direct recourse to the Plant Manager in order to resolve questions related to the conduct of the radiation protection program.

471.06, 471.09, 471.10

471.10 As recommended in Regulatory Guide 8.8, Section C.1.b(3),
(12.1.2) the responsibility and authority for implementing the plant's program for maintaining occupational radiation exposures ALARA should be assigned to an individual (or committee) with organizational freedom to ensure development and implementation. Identfy by title the individual(s) responsible for the ALARA program coordination and describe how he (they) are placed in the organization, particularly the mechanism for communication with plant management.

Response

The response to this question is provided in revised Sections 12.5.1 and 13.1.2.2.

12.5 HEALTH PHYSICS PROGRAM

12.5.1 ORGANIZATION

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471.09, 471.10

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General Supervising Engineer, Radiation Protection Section

The General Supervising Engineer, Radiation Protection Section is responsible for directing all activities associated with the chemical, radiochemical, radwaste and other radiological control services required to support plant operation and maintenance activities. This includes conducting laboratory and plant survey activities required to ensure that personnel exposure to radiation and radioactive materials is within regulatory guidelines and that such exposure is kept as low as reasonably achievable (ALARA). The General Supervising Engineer Radiation Protection Section is a member of the Plant Operations Review Committee, ALARA Committee, and reports to the Plant Manager.

Supervisor, Health Physics Unit

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471.06, 471.09, 471.10

471.11 (12.1.2) Based on information contained in Regulatory Guide 8.10, Section C.1.b. It is our position that the plant's management staff should periodically review operating procedures and exposure information to determine major changes in problem areas, and areas in which worker groups are accumulating the highest exposures. The staff at Perry Nuclear Power Plant should use the information obtained by management review to recommend equipment modification or changes in plant procedures. Outline your methods for implementing this position.

Response

The response to this question is provided in revised Section 12.1.3.

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

The ALARA Committee is comprised of senior plant staff including the Plant Manager, Superintendent, Plant Operations, General Supervising Engineer, Radiation Protection and the Supervisor, Health Physics Unit. The Committee shall meet twice a year and within 30 days prior to a refueling and/or major planned maintenance shutdown (greater than 21 day duration).

Responsibilities of the Committee shall include:

a. Review outage plans relative to ALARA, to assess the potential for radiation exposure,

b. Provide technical guidance and assistance in programs to keep exposure ALARA,

c. Review radiological controls inspection reports from the NRC and evaluate corrective action plans or implementation,

d. Review changes in regulation in the area of radiological controls and evaluate or effect appropriate implementation,

e. Review all ALARA Reviews generated by the ALARA Coordinator since the last meeting,

f. Review the Annual Operating Report Annual Personnel Occupational Radiation Exposure Tabulated by Work and Job Function to evaluate the effectiveness of the ALARA program in maintaining exposures ALARA, and 471.11

g. Review implementing procedures for the ALARA program.

12.1.3.1 <u>ALARA Training Program</u>

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, recommendation can be made to change procedures or equipment that will result in lower radiation exposures in the future.

471.12 (12.2) As requested in Regulatory Guide 1.70, Section 12.2.1, provide maximum neutron and gamma dose equivalent levels, in routinely visited areas in the containment, in the vicinity of major drywell shield penetrations. Areas of interest are, i.e., Reactor Water Cleanup and Standby Liquid Control System (drywell purge penetrations), TIP Station (personnel and equipment lock drywell penetrations), etc. Describe location, dimensions and shielding for the drywell shield penetrations. Provide average neutron and gamma exposure levels at the CRD hydraulic control units, at the CRD master control and at the containment personnel lock area and provide an estimate of average daily personnel exposure time in these areas.

Response

The response to this question is provided in revised Section 12.3.2.2.1. and Table 12.4-7.

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 Rem/hr at full power operation.

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.

Areas in containment routinely visited during power operation include the following systems: SLC(C41), RWCU(G33), CRD(C11), and TIP(C51). The expected occupancy requirements to these areas and the average personnel exposures is provided in Table 12.4-7. Their respective locations are shown on Figures 12.3-2, 12.3-3, 12.3-4 and 12.3-6. Routine surveillance and control functions are all accomplished from Zone II areas where the design basis gamma dose rate is $\leq$ 2.5 mr/hr. The neutron dose rate in these areas is negligible due to shielding provided by the five foot thick concrete drywell shield wall.

471.12

There are several major penetrations through the drywell shield wall. These as well as all other plant penetrations are located and/or designed to preclude the possibility of streaming from high to low radiation areas, or otherwise will be adequately shielded. Details of the personnel access lock and shield door and the equipment hatch are shown on Figures 3.8-31 and 3.8-32, respectively. Figure 12.3-2 provides their orientation in the plant layout. 471.12

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.

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	0.5	1	1/week	2.6
Sample stations in reactor building	5.0	0.5	1	1/shift	2.7

471.12

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	0.1
Containment personnel lock	1.0	0.05	3	1/shift	0.16
Total					43

471.12

471.13 (12.2.2) Because the steam dryer and steam separator must be transferred partially out of water during refueling, there is potential for high concentrations of airborne radioactive material during the transfer. You should outline our proposed methods (other than maintaining wet) to reduce airborne radioactive material during these transfers. Provide an estimate of expected airborne concentration, on the separator transfer from the reactor vessel to the storage area. Consider equipment contamination buildup after at least 10 years of operation and address particulates as well as iodines. (See Regulatory Guide 1.70, Section 12.2.2).

Response

The primary method of minimizing exposure to airborne radioactive material during the transfer of the steam dryer and steam separator is through administrative controls. These include direct Health Physics surveillance, the use of respiratory protection equipment and excluding the entry to containment of all personnel except those directly involved in the transfer operations. Predicted radiation levels after ten (10) years operation are estimated to be 7R/hr for the steam dryer and 25R/hr for the steam separator. Radiation levels caused by contamination will be predominantly due to the following nuclides: Mn-54, Zn-65, Fe-59, Co-58, Co-60, Cr-51, I-131 and I-133. Dated based on other operating plant experience estimates radioactive particulate ranges as high as 2×10^{-8} μCi/cc and the I-131 as high as 4×10^{-8} μCi/cc.

After removal and storage of the reactor pressure vessel (RPV) head, the steam dryer assembly is transferred dry to its storage space.

The steam line plugs are installed to permit flooding of the RPV and simultaneous removal of the safety/relief valves for testing.

The steam separator bolts are untensioned and unlatched from the RPV flange.

The upper pools are flooded and the steam separator is transferred with the shroud head under water to its storage space.

471.14 (12.3) Describe the Perry Nuclear Power Plant's accident radiation monitoring system. Include the location of installed instruments that have emergency power supplies (including LOCA) and the location of portable instruments placed to be readily accessible to personnel responding to an emergency. Regulatory Guide 1.97 (Revision 2) specifies the area radiation monitors in areas requiring access after an accident and portable survey meters should have a range up to 10^4 R/hr.

Response

Accident conditions are to be monitored by redundant high range gamma monitors to be located in the reactor building and in the drywell. The four monitors are to have a range of 1 R/hr. to 10^7 R/hr. and will be powered by the diesel backed 120 AC bus. These monitors shall be procured in accordance with Regulatory Guide 1.97 and NUREG 0737 Table II.F.1-3.

Revised Table 12.5-2 and Table 7-5 of the Emergency Plan (Appendix 13A) list the location and number of portable monitoring equipment available to personnel responding to an emergency.

High range noble gas monitors are to be added to the effluent flow paths, i.e.:

a) main plant unit vent
b) heater bay/turbine building vent
c) off-gas vent

These monitors will provide range extension to include the high level noble gas concentration in accordance with Regulatory Guide 1.97 and NUREG 0737. Power is to be derived from the diesel backed 120 VAC bus.

TABLE 12.5-2

PORTABLE SURVEY INSTRUMENTS

Type	Quantity	Range	Sensitivity/Accuracy	Remarks
G-M Pancake	6	0-50K CPM	5,000 CPM/mR/hr	-
G-M Hand	6	0-50K CPM	2,500 CPM/mR/hr	mR/hr and CPM
G-M mR/hr	20	0.1-2,000 mR/hr	±10% of Full Scale	Wide Range
Ion Chamber mR/hr	15	1-1,000 mR/hr	±10% From 60 KeV to 1.3 MeV	-
mR/hr Telescoping	7	0.01 R/hr - 999 R/hr	±15% From 70 KeV to 1.3 MEV	-
Ion Chamber R/hr	4	1 mR/hr - 19.99 KR/hr	±15% of Full Scale	-
mR/hr Neutron	3	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	-

471.19, .14

TABLE 7-5

PORTABLE RADIATION MONITORING EQUIPMENT

A. PORTABLE RADIATION SURVEY INSTRUMENTS

TYPE	RANGE	DETECTOR	QUANTITY	LOCATION
High Range Survey Instrument	1 mR/hr-19.99 KR/hr	Ion Chamber	2	OSC
High Range Survey Instrument	1 mR/hr-50 R/hr	Ion Chamber	4	EOF
			1	TSC
			4	OSC
Low Range Survey Instrument	0-200 mR/hr	G.M.	4	EOF
			1	TSC
			4	OSC
Alpha Survey Instrument	$0-10^5$	Proportional	1	RCA
Neutron	$0-10^5$ CPM	BF3	1	RCA
Survey Instrument	1mRem/hr-5Rem/hr	BF3	1	RCA

B. PERSONNEL MONITORING DEVICES

TYPE	RANGE	QUANTITY	LOCATION
Direct Reading Pocket Dosimeter	0-100R	10	EOF
		5	TSC
		10	RCA
	0-5R	10	EOF
		5	TSC
		10	RCA
	0-500 mR	20	EOF
		10	TSC
		70	RCA
Thermoluminescent Dosimeter (TLD)	0-100R	20	EOF
		50	RCA

471.15
(12.3.1.1)

Outline the Perry Nuclear Power Plant's program for implementing a chemistry control program to reduce radiocobalt production and crud buildup in normally radioactive systems (See Regulatory Guide 8.8, Section C.2.e(3)).

Response

The response to this question is provided in revised Section 12.3.1.1.

To further eliminate corrosion and crud tra[illegible] the General Electric oxygen control program will be evaluated to control oxygen in the condensate and feedwater to 50 ± 20 PPB. The condensate cleanup system is further described in Section 10.4.6.

The Company has participated in the BWR Radiation Assessment and Control (BRAC) Program since its inception in 1973. BRAC is a joint venture between utilities, General Electric and EPRI formed to investigate and research the problem of radiocobalt production and associated radiation field buildup on primary system piping. The Perry Plant design and operation of cleanup systems is based on results and recommendations that the BRAC program has developed through its research.

In addition to the installed water purification systems, consideration has been given to reduce radiocobalt production and crud buildup in normally radioactive systems through material selection and equipment design as discussed in Section 12.3.1.1.

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 primary defense against corrosion product buildup and associated neutron activation in the reactor vessel followed by crud transport is to minimize the input of impurities (i.e. iron, cobalt) in the feedwater. The Perry Plant design includes both full flow condensate filters and deep bed demineralizers. This design provides maximum removal of both suspended and dissolved impurities. In addition an extensive condenser sampling and analyses system is provided to ensure prompt detection of small condenser leaks. The condensate demineralizers are provided with conductivity cells to measure water quality in the bed effluent and at two thirds resin depth. The second conductivity cell will provide indication of resin depletion and allow for regeneration prior to total bed breakthrough.

471.15

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

471.16 (12.4.3) The doses to plant personnel in the reactor building as they exist after a type 2 safety/relief valve isolation scram is estimated in Table 12.4-14 of the FSAR. However, it appears that the doses provided are average values and may not reflect actual doses to workers. Accordingly, explain all of the assumptions used to calculate the whole body, skin, and thyroid dose to plant personnel following a safety/relief valve discharge listed in Table 12.4-14.

Provide an estimate of personnel exposure (similar to Table 12.4-14), resulting from actuation of safety relief valves, based on the following considerations:

1. Use design basis radiation sources:

   a) Noble gas concentrations corresponding to an off-gas release rate of 0.1 Ci/sec after 30 minutes decay:

   b) Halogen concentration in reactor water FSAR, Table 11.1-3,

2. Operator working at TIP drive floor at a location closest to the low-set safety relief valve discharge;

3. Assume that all safety relief valves open; low set relief valves remain open following closure of others (TYPE 2 occurrence);

4. Operator exposure 4 minutes;

5. Normal ventilation in containment (do not assume homogeneous mixing of airborne contaminants in the entire containment volume within the first four minutes);

6. Dose reduction factor can be applied if a clear-air shower is provided in the vicinity of the containment personnel lock;

7. Containment airborne concentrations should not be corrected for plate out on walls, as it will be negligible in the first few minutes.

Specify the noble gas and halogen pool retention factors, the average radiohalogen (reactor water to steam) carry-over factor, and other significant, dose-effecting factors employed in the calculations.

Response

The response to this question is provided in revised Sections 12.2.2.1, 12.4.3, and Table 12.4-14.

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 2.5 times per year. Reference 2 provides the source terms used to determine the containment airborne concentrations following a Type 2 event and the methodology used to determine operational doses following the event. Section 12.4.3 presents the anticipated operator exposures per event. 471.16

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 1.1×10^6 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 $1x10^{-6}$ μCi/cc based on information given in Reference 3. The building free volume is assumed to be $1.5x10^6$ 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. General Electric Co., "Mark III Containment Dose Reduction Study", 22A5718 Rev. 2, Jan. 29, 1980.

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

TABLE 12.2-13

(Deleted)

TABLE 12.2-13 (Continued)

(Deleted)

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 at the initiation of the isolation scram, an operator is located at the TIP drive floor. The operator egress is at the personnel air lock 180° from the TIP

471.16

floor area at the same elevation. Operator egress is conservatively assumed to take four minutes.

The dose rates used are those calculated for the immediate area above the suppression pool. Normal ventilation is assumed and airborne concentrations are not corrected for plate out on walls. The dose assessment methodology including pool retention factors and average radiohalogen carry-over factors are provided in Reference 2 of Section 12.2.3. 471.16

No dose calculations regarding Type 1 events are presented since resulting personnel doses would be negligible.

TABLE 12.4-14

SAFETY/RELIEF VALVE DISCHARGERS

DOSE FOR TYPE 2 EVENT

Organ	Dose
Whole body + eye (γ)	150 mRem/event
Skin (β)	440 mRem/event
Thyroid	.87 mRem/event

471.16

471.17 (12.5.2) Verify that your portable radiation detection instruments calibration program meets Regulatory Guide 8.25 or describes an equivalent alternative.

Response

The response to this question is provided in revised Sections 12.5.2.3.4 and 12.5.3.2.8.

Personnel contamination survey instruments shall include Geiger-Mueller friskers, portable monitors, and hand and foot counters. These instruments will be calibrated according to Health Physics Instructions semiannually when in use, or 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 in accordance with Regulatory Guide 8.25 and Health Physics Instructions. Typical surveys will be performed for particulate and radioiodine airborne concentrations. 471.17

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 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 Instructions. These instructions are prepared in conformance with Reg Guide 8.15.

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 Clothing Storage Area, and at any plant controlled access point necessary to support plant

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 minimize the possible spread of 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 radiation detection instruments are calibrated in accordance with manufacturers recommendations and Health Physics instructions. 471.17

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.

12.5.3.2.9 Plant Cleanliness

Plant cleanliness is maintained in accordance with Regulatory Guide 1.39 as discussed in Sections 1.8 and 17.2. 471.05

471.18 (12.5.2.3.1) Identify the iodine counter and gamma spectrometer marked "Later" in Table 12.5-1 or indicate when this information will be provided.

Response

The iodine counter and gamma spectrometer information will be provided in an amendment submitted on or about January, 1982.

471.19 (12.5.2.3.2) Table 12.5-2, portable survey instruments show the number of radiation detection instruments which will be available for both units. It is our position that the number of portable radiation survey instruments (especially those which are most frequently used by radiation protection personnel, 0 - 5000 mr/hr) be increased to reflect the following considerations (a) both units can be shut down for repair at the same time, (b) a number of instruments out of service (in need of calibration or repairs), and (c) a number of spare, operational instruments, should be always available for use in unusual occurrences. You should evaluate the number of portable survey instruments required by the plant using the above considerations and amend the FSAR accordingly. (See Regulatory Guide 8.8, Section C.4)

Response

The response to this question is provided in revised Section 12.5.2.3.2 and Table 12.5-2.

Each portable instrument will be calibrated according to Health Physics Instructions when in use, or prior to use after repair. Sufficient quantities of portable instrumentation will be available to permit calibration, maintenance, or repair to instruments without causing a shortage of operational equipment. Typical portable equipment is listed in Table 12.5-2. An additional 30 percent of each type of instrument listed in Table 12.5-2 is available for replacement of instruments that are out of service for repair or calibration.

471.19

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.

12.5.2.3.3 Personnel Monitoring Instruments

Personnel monitoring shall be provided by use of the thermoluminescent dosimeters (TLDs), direct reading pocket ionization dosimeters, and some electronic instruments. All persons entering the Radiological Control Area will be issued a TLD which will be used to measure exposure to beta-gamma radiation. This badge will contain thermoluminescent chips with suitable energy filters. TDLs will be analyzed at least monthly for personnel records, and anytime personnel doses need to be ascertained as prescribed by Health Physics Instructions. The exposure history established by the TLD readings shall constitute the official record of personnel exposure at PNPP.

It is not expected that an individual's Neutron dose will exceed 300 mRem per quarter; therefore, a calculated Neutron dose equivalent measured with portable monitoring instruments and known occupancy times will be issued in place of Neutron dosimeters. In the event that such operating data indicates the Neutron dose equivalent does not exceed 300 mRem per quarter, the calculated Neutron dose equivalent will be assumed equal to zero.

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

TABLE 12.5-2

PORTABLE SURVEY INSTRUMENTS

Type	Quantity	Range	Sensitivity/Accuracy	Remarks
G-M Pancake	6	0-50K CPM	5,000 CPM/mR/hr	-
G-M Hand	6	0-50K CPM	2,500 CPM/mR/hr	mR/hr and CPM
G-M mR/hr	20	0.1-2,000 mR/hr	±10% of Full Scale	Wide Range
Ion Chamber mR/hr	15	1-1,000 mR/hr	±10% From 60 KeV to 1.3 MeV	-
mR/hr Telescoping	7	0.01 R/hr - 999 R/hr	±15% From 70 KeV to 1.3 MEV	-
Ion Chamber R/hr	4	1 mR/hr - 19.99 KR/hr	±15% of Full Scale	-
mR/hr Neutron	3	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	-

471.19, 14

TABLE 7-5

PORTABLE RADIATION MONITORING EQUIPMENT

A. PORTABLE RADIATION SURVEY INSTRUMENTS

TYPE	RANGE	DETECTOR	QUANTITY	LOCATION
High Range Survey Instrument	1 mR/hr-19.99 KR/hr	Ion Chamber	2	OSC
High Range Survey Instrument	1 mR/hr-50 R/hr	Ion Chamber	4 1 4	EOF TSC OSC
Low Range Survey Instrument	0-200 mR/hr	G.M.	4 1 4	EOF TSC OSC
Alpha Survey Instrument	$0\text{-}10^5$	Proportional	1	RCA
Neutron Survey Instrument	$0\text{-}10^5$ CPM 1mRem/hr-5Rem/hr	BF3 BF3	1 1	RCA RCA

B. PERSONNEL MONITORING DEVICES

TYPE	RANGE	QUANTITY	LOCATION
Direct Reading Pocket Dosimeter	0-100R	10 5 10	EOF TSC RCA
	0-5R	10 5 10	EOF TSC RCA
	0-500 mR	20 10 70	EOF TSC RCA
Thermoluminescent Dosimeter (TLD)	0-100R	20 50	EOF RCA

471.20 (12.5.2.3.3) Based on information contained in Regulatory Guide 8.8, Section C.3.b(2), it is our position that all personnel assigned TLD or film badgss also wear direct reading pocket dosimeters when entering controlled areas. The readings from these dosimeters should be used to keep a running total of the dose to individuals prior to TLD or film badge processing and to allow analysis of doses by tasks. Outline your methods for implementing this position.

Response

The response to this question is provided in revised Section 12.5.2.3.3.

extremities. The dosimeters will be calibrated semi-annually, or anytime damage is suspected.

The individual records the pocket dosimeter reading on the RWP Exposure Record Card while exiting the Radiological Control Area. Daily, the readings are transferred to the individual's occupational exposure records by Health Physics Personnel. This provides a running total of the dose to the individual.

The TLD results are compared to the pocket dosimeter readings upon receipt. Discrepancies between the pocket dosimeter readings and TLD results of greater than 25 percent are investigated by the Health Physics Unit to determine the probable cause. 471.20

Annually or as required by the Health Physics Supervisor, an analysis of dose by task is performed to determine which operations or tasks can be modified to reduce exposure.

471.21 (12.5.2.3.4) In Table 12.5.4 you indicate that 50 full-face masks will be available at the plant. This number appears low for routine (nonemergency) use compared to the needs of currently operating nuclear power plants. You should revise this inventory or justify this number by evaluating the number of full-face masks expected to be needed during normal and expected operational occurrences, during emergency situations and number of masks not available to plant personnel due to routine maintenance. (See Regulatory Guide 8.8, Section C.4).

<u>Response</u>

The response to this question is provided in revised Table 12.5-4.

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
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)
Full Face Masks	200	<2000 PF	CF and PD Mode
Full Face Filters	300	< 50 PF	-
Respirator Hoses	20	N/A	-
Respirator Junction Box	3	N/A	To 6 Respirators Per Junction Box
Waterproof Suits	30	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.
Laundry Monitor	1	160-7000 CPM	-

471.21

471.22 (12.5.3.6.2) Section 12.5.3.6.2, states all personnel who wear respirators will be whole body counted or have a bioassay at least once per year. "Personnel who freqently use respirators, or are suspected of having an accidental exposure to airborne radioactivity may be bioassayed or whole body counted more often." You should specify that your bioassay program will be implemented in accordance with Regulatory Guide 8.26 "Applications of Bioassay for Fission and Activation Products," or you should describe your equivalent bioassay program.

Response

The response to this question is provded in revised Section 12.5.3.6.2.

Exposure data for all personnel will be recorded on Form NRC-5, or the equivalent. Occupational exposures incurred by individuals who are expected to exceed 1.25 Rem in any quarter 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 and individual, as required to assist in keeping individual radiation exposures ALARA. Each worker shall receive exposure reports in accordance with 10 CFR 19.13.

Personnel monitoring, and equipment for personnel monitoring and surveys, will be in conformance with the requirements of 10 CFR 20 and Regulatory Guides 8.4 and 8.9, as discussed in Section 1.8.

12.5.3.6.2 Internal Exposures

The bioassay program is implemented in accordance with Regulatory Guide 8.26. 471.22

All personnel who take part in the Respiratory Protection Program (wear or may 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 <u>Evaluation and Control of Potential Airborne Radioactivity</u>

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.

471.23 (12.5.3.8)

Section 12.5.3.8 of the PNPP FSAR "sealed radionuclides having activities greater than the amounts listed in Appendix C of 10 CFR Part 20 will be subject to controls for radiological protection." Since the radionuclides and activities listed in Appendix C are associated with allowable sewerage release limits authorized in 10 CFR 20.303 and not intended as de mininus quantities, you should revise your procedures to require all licensed sources to be subject to material controls. However, pursuant to Section 30.18 of 10 CFR Part 30 sealed sources obtained from a manufacturer licensed to distribute exempt quantities in accordance with Section 32.18 of 10 CFR Part 32 may be exempted from your material control system.

Response

The response to this question is provided in revised Section 12.5.3.8.

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.

All radioactive sources 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, and other personnel in the work vicinity. 471.23

Special Nuclear Material is maintained, handled, and stored in accordance with 10 CFR 70.

471.28 (Appendix 1A) Please provide the information requested in II.B.2, II.F.1(3) and III.D.3 of NUREG-0737, "Clarification of TMI Action Plan Requirements."

Response

Item II.B.2 addresses the requirements for post-LOCA vital area access.

Vital Area Access

A review of the plant identified systems which were likely to contain highly radioactive fluid following a design basis LOCA. The radioactive material was assumed to be instantaneously mixed in those systems, connected either to the reactor coolant system or to the containment atmosphere, that are not isolated at the start of the accident. Non-essential systems that are isolated and have no post-accident function were not considered in the review.

After determining the systems and post-accident source distribution to be used for the shielding review, the SDC point kernel shielding code was used to calculate the associated post-accident radiation doses.

Areas which may require occupancy to permit an operator to aid in the mitigation of an accident are vital areas. The evaluation to determine the necessary vital areas included the control room, technical support center, post-LOCA hydrogen control system, containment isolation system, sampling and sample analysis areas, remote shutdown panel, ECCS alignment functions, motor control center, instrument panels, emergency power supplies, security center and radwaste control panels. Of these it was determined that for the Perry plant, the control room and technical support center will require continuous occupancy and the post-accident sample area will require infrequent occupancy. The remote shutdown panel is available for frequent occupancy if required.

For further information see "Shielding Report" (attached). This report will be added to the FSAR as new Section 12.6 with a future amendment.

II.F.1-3

High range gamma monitors are to be added to the reactor building and to the drywell to provide conformance with NUREG 0737 Table II.F.1-3 with a range of 1 R/hr to 10^7 R/hr and to respond to the requirements of Regulatory Guide 1.97, Revision 2.

As yet the specific monitors and manufacturer have not been selected but the data will be provided when available. However, the monitors will be installed and operable prior to plant initial criticality. They will be powered from independent 120 volt a-c, diesel-backed buses and will be provided with continuous readout and multipoint recorders in the control room. Although the calibration procedure for the monitors will vary from model to model, it will generally be by calibration source below 10 R/hr., and by electronic signal input for ranges above 10 R/hr. Detailed calibration procedures will be provided at a later date.

Planned location of the monitors is as follows: two monitors shall be located in the drywell at approximately core midplane spread approximately 32° apart centered approximately at 225° azimuth for Unit 1, and 125° azimuth for Unit 2. Two monitors shall be located in the Reactor Building at approximately the 689' level, and the same degree spread and azimuth as those in the drywell. Plant layout drawings will be provided showing specific locations of the monitors when available.

III.D.3

CEI is currently evaluating two methods to provide iodine monitoring during accident conditions.

The first uses a Silver Zeolite cartridge evaluated with a single channel analyzer set for the 0.364 Mev gamma of Iodine 131.

The second method, published by C. Distenfeld and J. Klemisa, Brookhaven National Laboratory, uses a silver impregnated silica gel canister which has a low affinity for noble gases while retaining a high affinity for iodines. The silica gel is then measured with a GM Tube inserted into the canister and the reading converted to an iodine concentration.

Evaluation of the methods to provide iodine monitoring during accident condition will be completed in October, 1982. Appropriate detailed information on the number and type of samplers, sample media, flushing methods and sample analysis system will be provided at that time. Procedures and training will be provided to personnel for post-accident iodine sampling.

DESIGN REVIEW OF PLANT SHIELDING FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS OUTSIDE CONTAINMENT AT PERRY NUCLEAR POWER PLANT

1.0 Introduction

An accident equivalent to that described in Regulatory Guide 1.3 would release large fractions of the nuclear core fission product inventory. Once released these fission products could be transferred to various areas in the plant creating high radiation areas and limiting personnel access. This review determines if these post-accident radiation fields unduly limit personnel access to areas necessary for mitigation of an accident. Corrective actions for problems identified are also determined. This design review of plant shielding for spaces and systems required for post-accident operations outside containment is in accordance with "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (NUREG-0578) Section II.B.2 as clarified by NUREG-0660 and NUREG-0737.

The review is based on the following guidelines:

a. The post accident dose rate in areas requiring continuous occupancy should not exceed 15 mr/hr. (Control room and onsite technical support center.)

b. The post accident dose rate in areas which do not require continuous occupancy should be such that the dose to an individual during a required access period is less than 5 Rem whole body or its equivalent. (Sample stations, panels, motor control centers, etc.)

c. The integrated dose to safety equipment as a result of the accident should be less than the dose for which the equipment has been qualified to ensure that the capability of the equipment to perform its safety function has not been degraded.

d. The minimum radioactive source term used in the evaluation should be equivalent to the source term recommended in Regulatory Guide 1.3.

The occupancy and radiation design objectives for this review are given in Table 2.0-1.

2.0 Radioactive Source Release

The initial core inventory is shown in Table 2.0-2. The percents of radioactive fission product core inventory assumed to be released from the fuel rods are:

noble gases (Kr, Xe)	100%
halogens (I, Br)	50%
alkali metals (Cs, Rb)	50%
others	1%

This release is assumed to occur and be distributed instantaneously at the start of the accident.

3.0 Radioactive Source Distribution

The following radioactive source distributions were used to determine the shielding review concentrations:

Source A - Liquid in the suppression pool and other systems not isolated from the core at the start of the accident and containing only liquid from a depressurized source is assumed to contain the following percents of core inventory of radioactive fission products:

noble gases (Kr, Xe)	0%
halogens (I, Br)	50%
alkali metals (Cs, Rb)	50%
others	1%

These fission products are assumed to be uniformly mixed in a volume of 117,105 ft3 which is the low level water volume of the suppression pool.

Source B - For evaluating vital areas, the drywell atmosphere is assumed to contain the following percents of radioactive fission product core inventory:

noble gases (Kr, Xe)	100%
halogens (I, Br)	25%
others	0%

These fission products are assumed to be uniformly mixed in the volume of the drywell air space of 277,685 cubic feet.

Source C - For evaluating vital areas, the primary containment atmosphere is assumed to contain the following percents of radioactive fission product core inventory:

noble gases (Kr, Xe)	100%
halogens (I, Br)	25%
others	0%

These fission products are assumed to be uniformly mixed in the free volume of the containment of 1,141,014 cubic feet.

Source D - Until the reactor vessel is depressurized, gases in the steam lines and any other vapor containing lines not isolated from the reactor coolant system are assumed to contain the following percents of radioactive fission product core inventory:

noble gases (Kr, Xe)	100%
halogens (I, Br)	25%
others	0%

These fission products are assumed to be uniformly mixed in the volume of the reactor coolant system steam space (9,189 cubic feet). This is a highly conservative source estimate because steam usage would deplete this source shortly after the start of the accident.

Source E - For equipment qualification inside containment, the larger of the two following source terms are used:

noble gases (Kr, Xe)	100%
halogens (I, Br)	50%
alkali metals (Cs, Rb)	50%
others	1%

These fission products are assumed to be released to the containment atmosphere, or

noble gases (Kr, Xe)	100%
halogens (I, Br)	50%
alkali metals (Cs, Rb)	50%
others	1%

These fission products are assumed to be uniformly mixed in a volume no greater than the liquid volume of the reactor coolant system (11,838 cubic feet).

Source F - For qualification of equipment inside containment for non-LOCA events which do not depressurize the primary system, the following source terms are used:

noble gases (Kr, Xe)	10%
halogens (I, Br)	10%
others	0%

These fission products are assumed to be uniformly mixed in the volume of the reactor coolant system (11,838 cubic feet).

The initial radioactive source terms used are given in Table 3.0-1.

4.0 Systems Containing Radioactive Sources

A review of Perry Nuclear Power Plant identified systems which were likely to contain highly radioactive fluid following a design basis LOCA. The radioactive material is assumed to be instantaneously mixed in those systems connected either to the reactor coolant system or to the containment atmosphere, that are not isolated at the start of the accident. Non-essential systems that are isolated and have no post accident function are not considered in this review.

4.1 LPCS, HPCS, RHR (LPCI mode), RCIC Systems

Following an accident, the LPCS, HPCS, RHR (LPCI mode), and RCIC (water side) systems draw water from the suppression pool and inject it into the reactor vessel for emergency core cooling. The suppression pool water is assumed to be the only injection water source, although the HPCS and RCIC systems would initially draw water from the condensate storage tank. The RCIC (steam side) system draws main steam from the reactor vessel until the reactor is depressurized.

4.2 RHR (Shutdown Cooling Mode)

After reactor depressurization, the RHR system cools recirculating reactor water. Essentially all noble gas would be released from the reactor water upon depressurization. Therefore, it is assumed there are no noble gases in the RHR system while it is performing its shutdown cooling function. Also, no reactor water dilution by the condensate storage tank or suppression pool water is assumed.

4.3 RHR (Suppression Pool Cooling Mode)

The suppression pool cooling function of the RHR system maintains the correct temperature of the suppression pool water by circulating it through the RHR heat exchanger and returning it to the suppression pool.

4.4 RWCU

The RWCU system is designed to automatically isolate following a low reactor water level signal caused by a LOCA. Since the system is not designated for post-LOCA recovery cleanup operations it is not included in the LOCA review.

Since the RWCU system would not receive a low reactor water level signal following a high energy line break accident (HELBA) that did not depressurize the reactor coolant system, it would not automatically isolate. Therefore, it is assumed to contain the non-LOCA equipment qualification HELBA source (Source F).

4.5 Liquid Radwaste System

The liquid radwaste system is automatically isolated from parts of the systen that may contain highly contaminated post-accident water. Therefore, the liquid radwaste systen is not included in this review.

4.6 Sampling System

Following a accident it is necessary to obtain and perform radioisotopic and chemical analyses of reactor coolant and containment atmosphere samples. The samples are taken at the 599' elevation of the Intermediate Building and analyzed in the Control Complex Count Room as shown on Figure 7.0-2. The post accident sampling system and Count Room are shielded such that an operator may collect and analyze samples 'under degraded core conditions without excessive radiation exposures.

5.0 Shielding Review

After determining the systems and post-accident source distribution to be used for the shielding review, the SDC (Ref.1) and SPOT 1 (Ref.2) shielding codes were used to calculate the associated post accident radiation doses.

Each compartment radiation level is calculated at what is judged to be the maximum radiation dose point. This point is on the surface of the major system component and includes contributions from piping and other simultaneously operating components in that compartment.

Calculated radiation levels in corridors include contributions from:

1) unattenuated radiation penetrating adjacent compartments shield walls.

2) direct radiation from piping or equipment.

3) radiation scattered over or around shield walls.

Potential dose contributions not considered in this review are:

1) normal operating sources which may exist at the time of the accident.

2) airborne sources from equipment leakage.

6.0 Areas Requiring Personnel Access

Areas which may require occupancy to permit an operator to aid in the mitigation of an accident are vital areas. The evaluation to determine the necessary vital areas included the control room, technical support center, post-LOCA hydrogen control system, containment isolation system, sampling and sample analysis areas, remote shutdown panel, ECCS alignment functions, motor control centers, instrument panels, emergency power supplies, security center and radwaste control panels.

A study of Perry Nuclear Power Plant determined that several of the areas considered were not vital areas. The post-LOCA hydrogen control system is located inside the reactor building and is remotely operated from the control room. The containment isolation reset controls and ECCS alignment controls are initiated from the control room. The safety related motor control centers, are located in the control complex. They require no local actions but are accessible for infrequent occupancy, if required. The radwaste control panel is excluded because the radwaste system is automatically isolated in the event of an accident and is not designed to process accident wastes. The emergency power supply areas (Standby Diesel Generators) are not vital areas because the required functions are initiated from the Control Room and require no local action, however, they are accessible for infrequent occupancy, if required.

The security center is not a vital area because after an accident security functions will be inititated from the secondary security center in the control room.

The vital : _as for the Perry Nuclear Power Plant assumed to require post-accident personnel access to bring the plant to cold shutdown and to implement the emergency plan include:

1. For continuous occupancy - the Control Room and Technical Support Center.

2. For frequent occupancy - The Remote Shutdown Panel, if required.

3. For infrequent occupancy - The Sampling Station and the Sample Analysis Area.

These areas meet the occupancy and radiation design objectives given in Table 2.0-1.

7.0 Post-Accident Radiation Zone Drawings & Summary

Post-accident radiation zone drawings are given in Figures 7.0-1 through 7.0-7. These radiation zones represent the maximum expected radiation dose rate for each area at the start of an accident in either unit. These levels would not exist simultaneously in both units. Normal operating dose levels which may exist at the time of the accident are not shown on these drawings. A summary of major locations and accident dose rates is given in Table 7.0-1.

7.1 Radiation Dose Rates as a Function of Time Following an Accident

The decay curves for radiation dose rates as a function of time following an accident for the Auxiliary and Intermediate Buildings are given in Figures 7.1-1 and 7.1-2. These curves may be used along with the radiation zone drawings and summary table to predict radiation dose rates at a given time after the accident.

References

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

2) Kamphouse, J. L., "SPOT1 Shield Code", Gilbert Associates, Inc., October 1979.

3) A. Tobias, Data for the Calculation of Gamma Radiation Spectra and Beta Heating from Fission Products, Revision 3, Central Electricity Generating Board, RD/B/M2666, CNDC (73) P4, June 1973.

TABLE 2.0-1

OCCUPANCY AND RADITAION DESIGN OBJECTIVES

Required Occupancy	Dose Rate Limit	Integrated Dose Objective
Continuous	15 mR/hr	5 Rem for duration
Frequent	100 mR/hr	5 Rem for all activities
Infrequent	500 mR/hr	5 Rem per activity
Accessway	5000 mR/hr	Included in above doses

TABLE 2.0-2

INITIAL CORE INVENTORY

ISOTOPE	UCI/WATT	ISOTOPE	UCI/WATT	ISOTOPE	UCI/WATT	ISOTOPE	UCI/WATT
I-131	2.7 + 4	SR-89	2.4 + 4	RH-105M	5.1 + 3	CS-139	4.9 + 4
I-132	3.8 + 4	SR-90	2.3 + 3	RU-106	1.6 + 4	BA-139	5.0 + 4
I-133	5.5 + 4	Y-90	2.4 + 3	RH-106	1.7 + 4	BA-140	4.8 + 4
I-134	5.9 + 4	SR-91	3.0 + 4	RH-107	1.6 + 4	LA-140	5.0 + 4
I-135	5.1 + 4	Y-91	3.1 + 4	SB-127	2.8 + 3	BA-141	4.6 + 4
KR-83M	-	SR-92	3.3 + 4	TE-127	3.6 + 3	LA-141	4.6 + 4
KR-85M	7.2 + 3	Y-92	3.3 + 4	SB-128	4.1 + 3	CE-141	4.6 + 4
KR-85	2.9 + 2	SR-93	3.8 + 4	SN-128	2.6 + 3	PR-142	3.5 + 3
KR-87	1.2 + 4	Y-93	3.9 + 4	SB-129	9.0 + 3	BA-142	3.7 + 4
KR-88	1.8 + 4	Y-94	4.1 + 4	TE-129M	1.6 + 3	LA-142	4.2 + 4

TABLE 2.0-2

INITIAL CORE INVENTORY

ISOTOPE	UCI/WATT	ISOTOPE	UCI/WATT	ISOTOPE	UCI/WATT	ISOTOPE	UCI/WATT
XE-131M	-	Y-95	4.4 + 4	TE-129	8.6 + 3	CE-143	4.1 + 4
XE-133M	1.5 + 3	ZR-95	4.5 + 4	SB-130	1.3 + 4	PR-143	4.0 + 4
XE-133	5.5 + 4	NB-95	4.5 + 4	SB-131	2.2 + 4	CE-144	3.5 + 4
XE-135M	9.7 + 3	ZR-97	4.6 + 4	TE-131	2.4 + 4	PR-144	3.5 + 4
XE-135	7.4 + 3	NB-97	4.6 + 4	TE-131M	4.3 + 3	PR-145	2.3 + 4
XE-138	4.7 + 4	MO-99	5.1 + 4	TE-132	3.8 + 4	PR-146	2.3 + 4
BR-83	2.7 + 3	TC-99M	4.5 + 4	TE-133	2.4 + 4	PR-147	1.7 + 4
BR-84	5.4 + 3	MO-101	4.6 + 4	TE-133M	3.2 + 4	ND-147	1.8 + 4
BR-88	1.8 + 4	RU-103	4.4 + 4	TE-134	4.9 + 4	PM-148	3.8 + 3
RB-88	2.3 + 4	RH-103M	4.3 + 4	CS-134	2.0 + 3	ND-149	1.0 + 4
BR-89	2.3 + 4	TC-104	3.5 + 4	CS-137	3.3 + 3	PM-149	1.5 + 4
BR-89	2.3 + 4	TC-104	3.5 + 4	CS-137	3.3 + 3	PM-149	1.5 + 4
		RU-105	2.4 + 4	BA-137M	3.0 + 3	PM-151	5.4 + 3
		RH-105	2.4 + 4	CS-138	5.0 + 4	EU-156	4.8 + 3

TABLE 3.0-1 - INITIAL RADIOACTIVE SOURCE TERMS (GAMMAS/CC-SEC)

	SOURCE	SUPPRESSION POOL	RWCU SYSTEM	STEAM DOME	REACTOR BLDG ATMOSPHERE	REACTOR COOLANT SYSTEM
	%CORE NOBLE GAS	0	10	100	100	100
GAMMA-ENERGY	%CORE HALOGENS	50	10	25	25	50
GROUP (MEV) (REF.3)	%CORE SOLIDS	1/50(CS-RB)	0	0	0	1/50 (CS-RB)
0.1 - 0.5		1.86 +9	5.46 +9	2.06 +10	1.34 +8	1.84 +10
0.5 - 1.0		6.84 +9	1.33 +10	3.32 +10	2.15 +8	6.77 +10
1.0 - 1.5		3.26 +9	3.26 +9	4.41 +10	2.86 +8	3.23 +10
1.5 - 2.0		6.63 +8	1.48 +9	1.18 +10	7.69 +7	6.56 +9
2.0 - 2.5		4.57 +8	1.16 +9	8.50 +9	5.52 +7	4.52 +9
2.5 - 3.0		9.31 +7	1.94 +8	1.29 +10	8.36 +7	9.22 +8
3.0 - 3.5		1.49 +7	2.39 +6	2.28 +9	1.48 +7	1.48 +8
3.5 +4.0		2.76 +7	3.87 +7	1.14 +8	7.41 +5	2.73 +8
4.0 - 5.0		7.74 +5	5.80 +5	1.80 +6	1.16 +4	7.66 +6

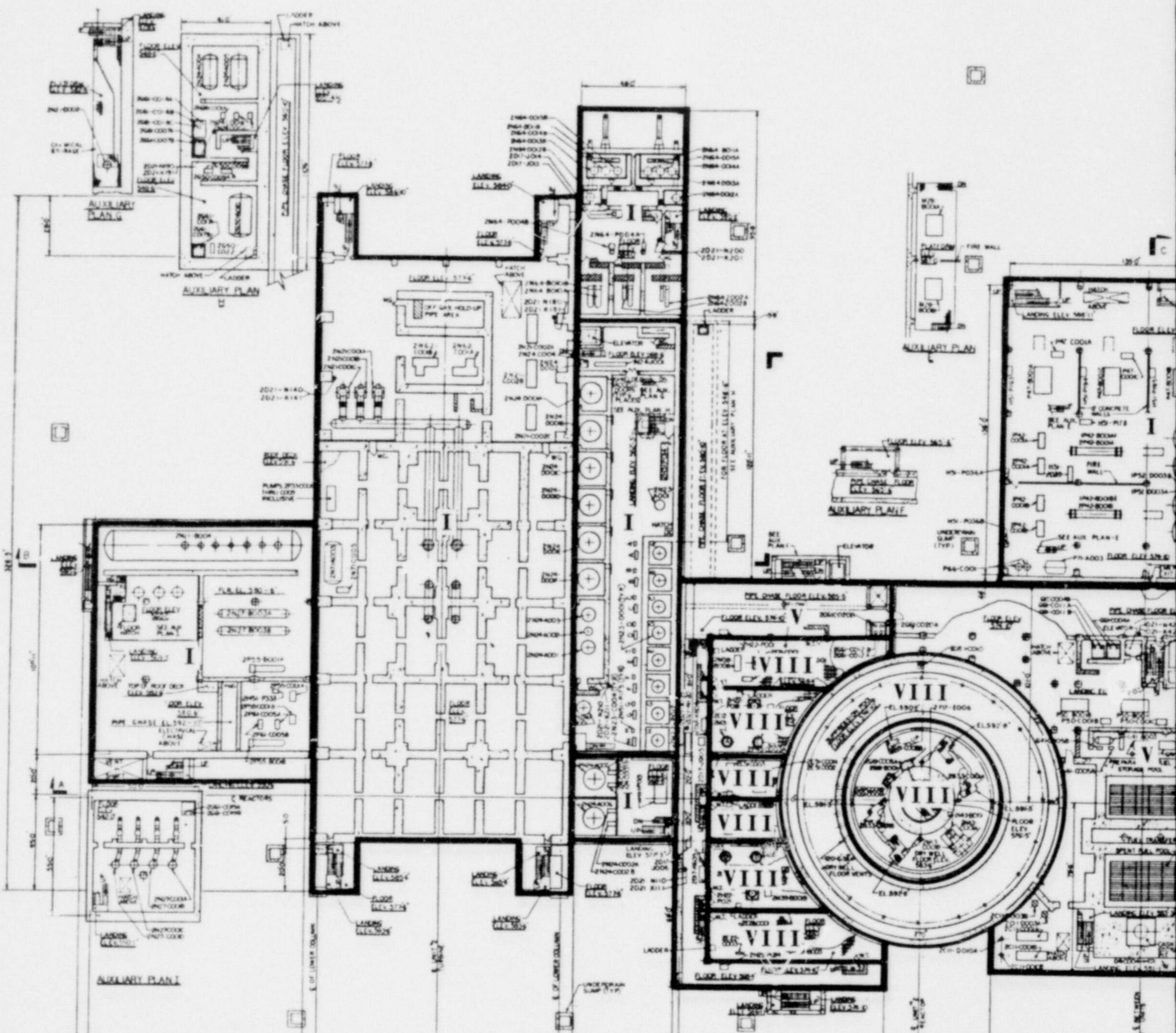
TABLE 7.0-1

DOSE RATES

LOCATION	TIME 0 DOSE RATE (mr/hr)
AUXILIARY BUILDING	
Steam Tunnel	2.0 +9
RWCU Pump Cubicle	4.6 +8
LPCS Pump Cubicle	3.7 +8
RCIC Pump Cubicle	1.3 +9
RHR HX Cubicle	5.3 +8
Corridor	
Outside RCIC Pump	3.3 +4
Outside RHR HX	
@ Elev. 599'-0"	2.7 +4
@ Elev. 568'-4"	2.7 +4
Outside RWCU Pump	1.6 +2
Outside LPCS Pump	4.2 +3
INTERMEDIATE BUILDING	
Above Elev. 646'-0"	3.0 +4
Elev. 599'-0" & 620"-6"	1.6 +4
Elev. 574'-10"	1.9 +4
DIESEL GENERATOR BUILDING	4.8 +0

CONTROL COMPLEX	<15(1)
Control Room	<15(1)
Count Room	Later
SECURITY OFFICE	3.1 +2(1)
GUARD HOUSE	1.1 +2(1)
TECHNICAL SUPPORT CENTER	<15(1)

(1) Represents average dose rate for 30 days.



LEGEND:

RADIATION ZONE	MAX.DOSE RATE	RADIATION ZONE	MAX.DOSE RATE
I	<0.015 R/HR	V	<50 R/HR
II	<0.100 R/HR	VI	<500 R/HR
III	<0.500 R/HR	VII	<5000 R/HR
IV	<5 R/HR	VIII	>5000 R/HR

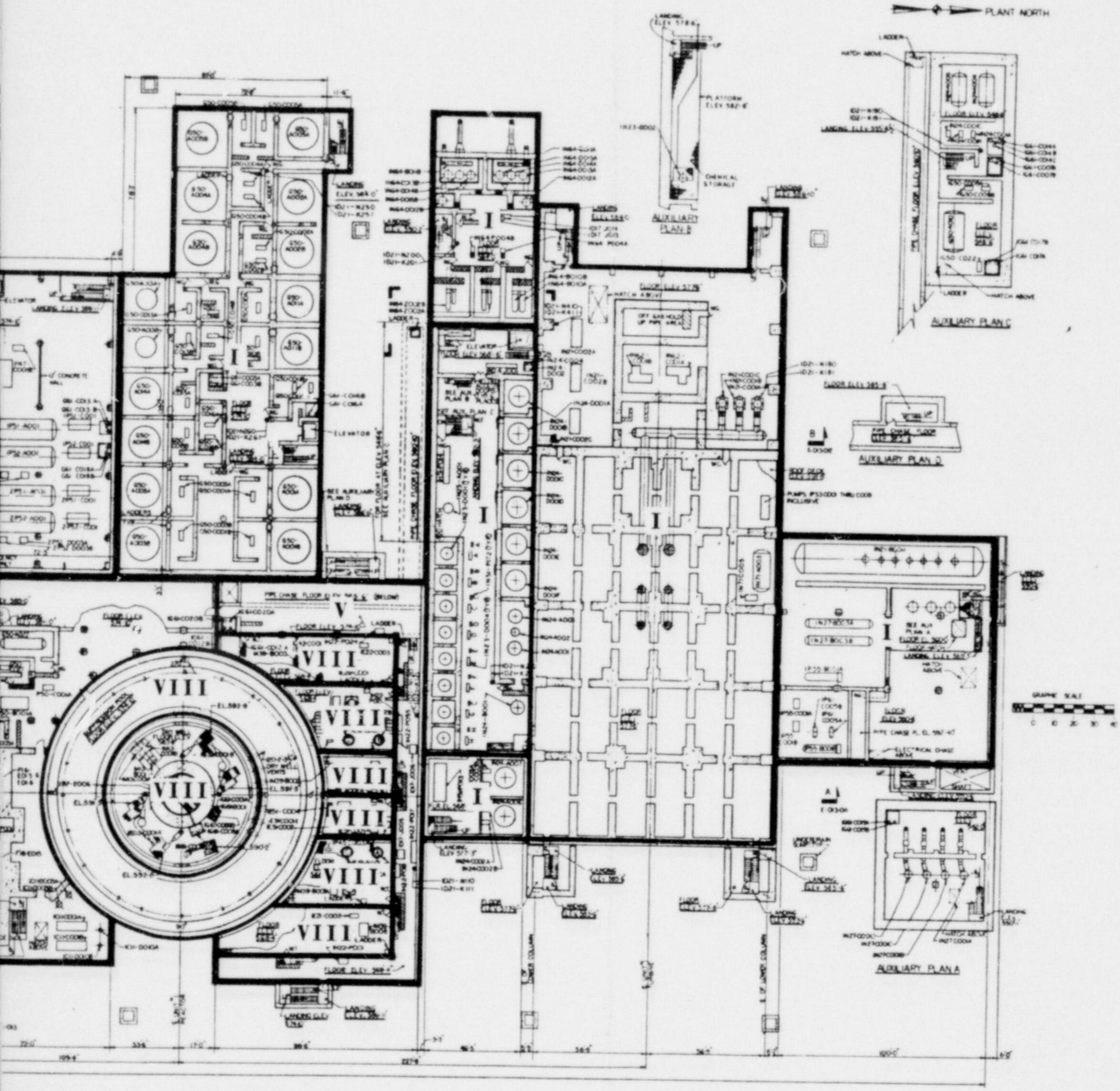
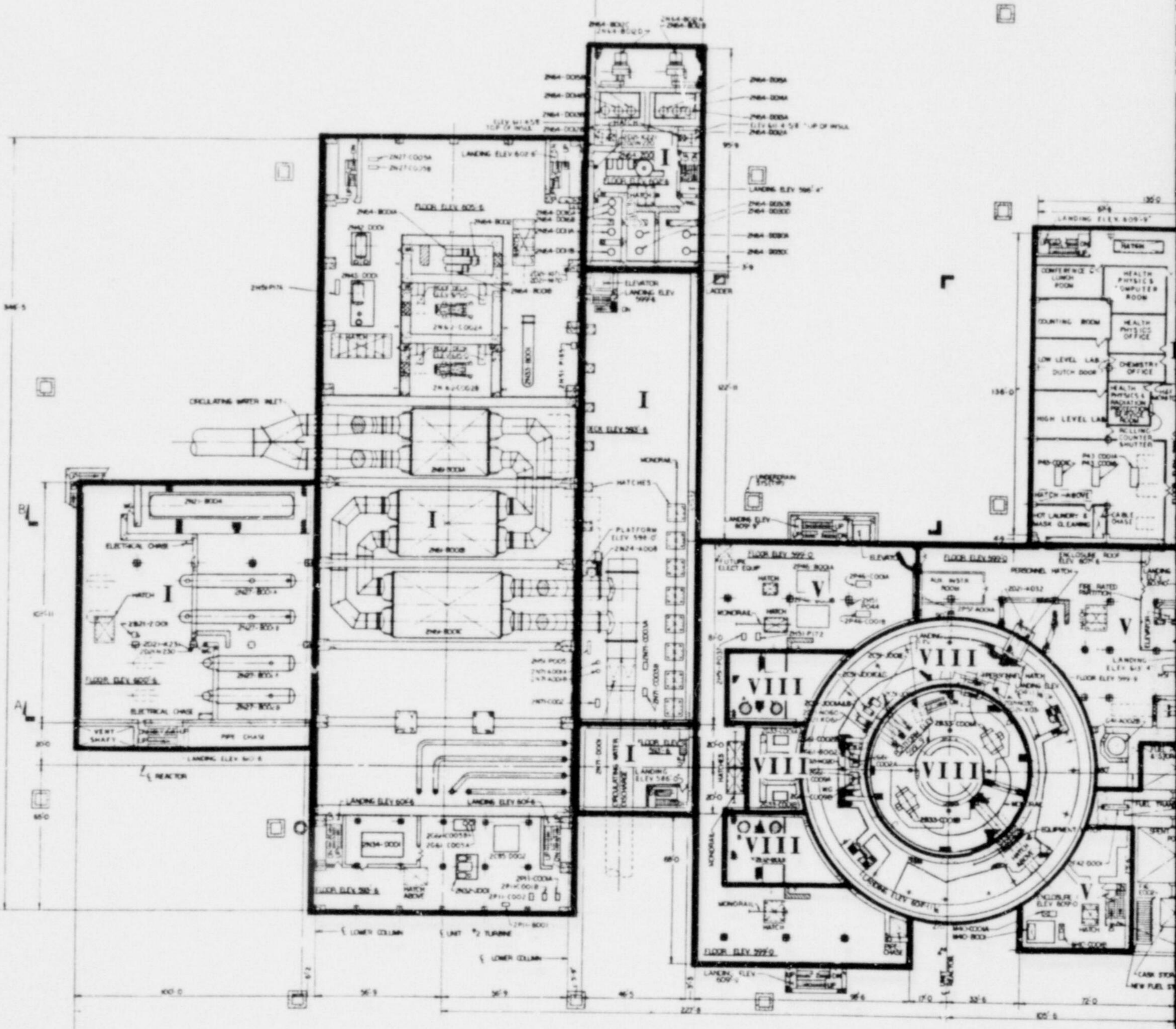

PLANT NORTH
AUXILIARY PLAN A
AUXILIARY PLAN B
AUXILIARY PLAN C
AUXILIARY PLAN D
CHEMICAL STORAGE
ELEVATOR
GRAPHIC SCALE

Figure 7.0-1



LEGEND:

RADIATION ZONE	MAX.DOSE RATE	RADIATION ZONE	MAX.DOSE RATE
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II	<0.100 R/HR	VI	<500 R/HR
III	<0.500 R/HR	VII	<5000 R/HR
IV	<5 R/HR	VIII	>5000 R/HR

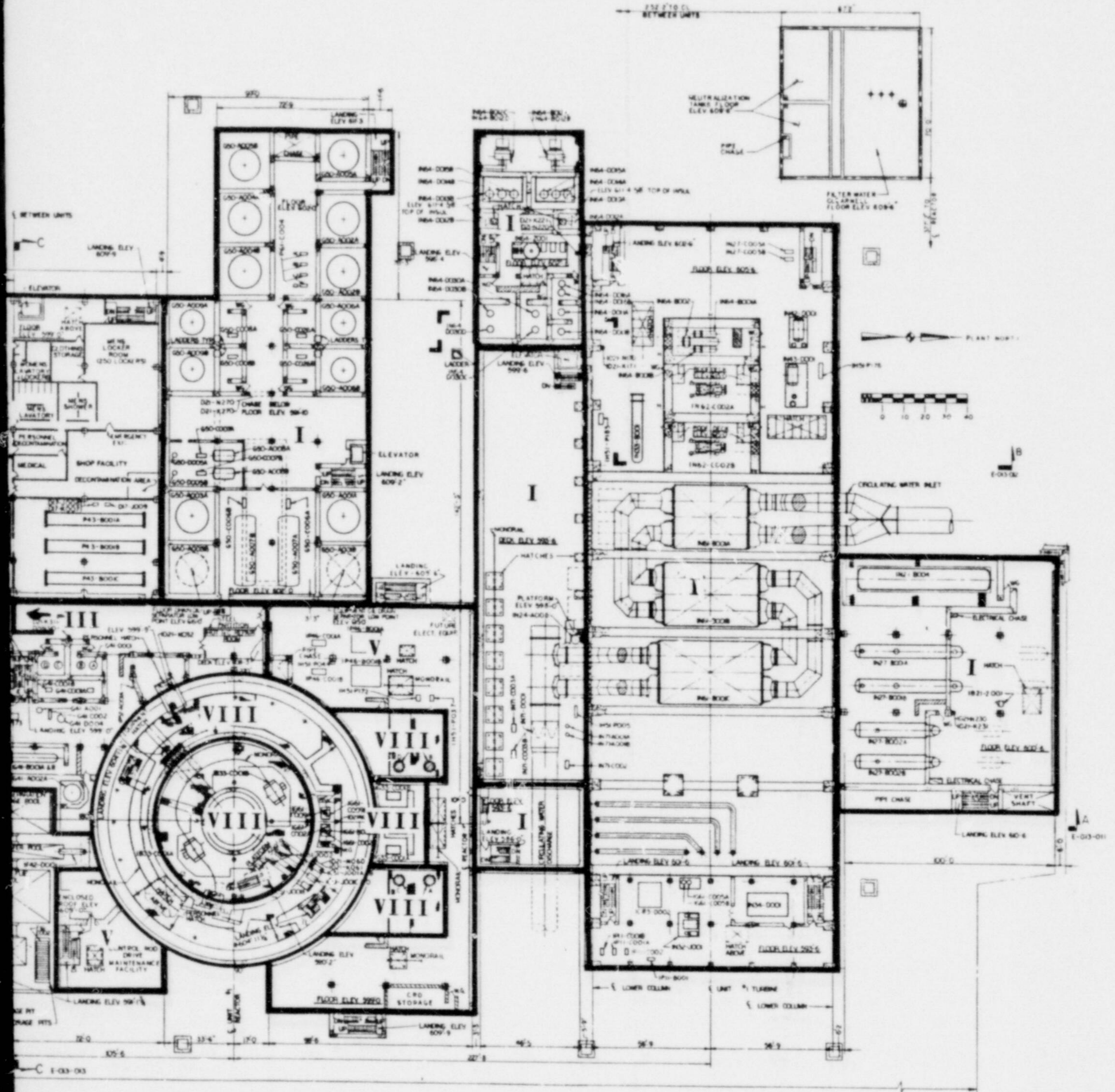
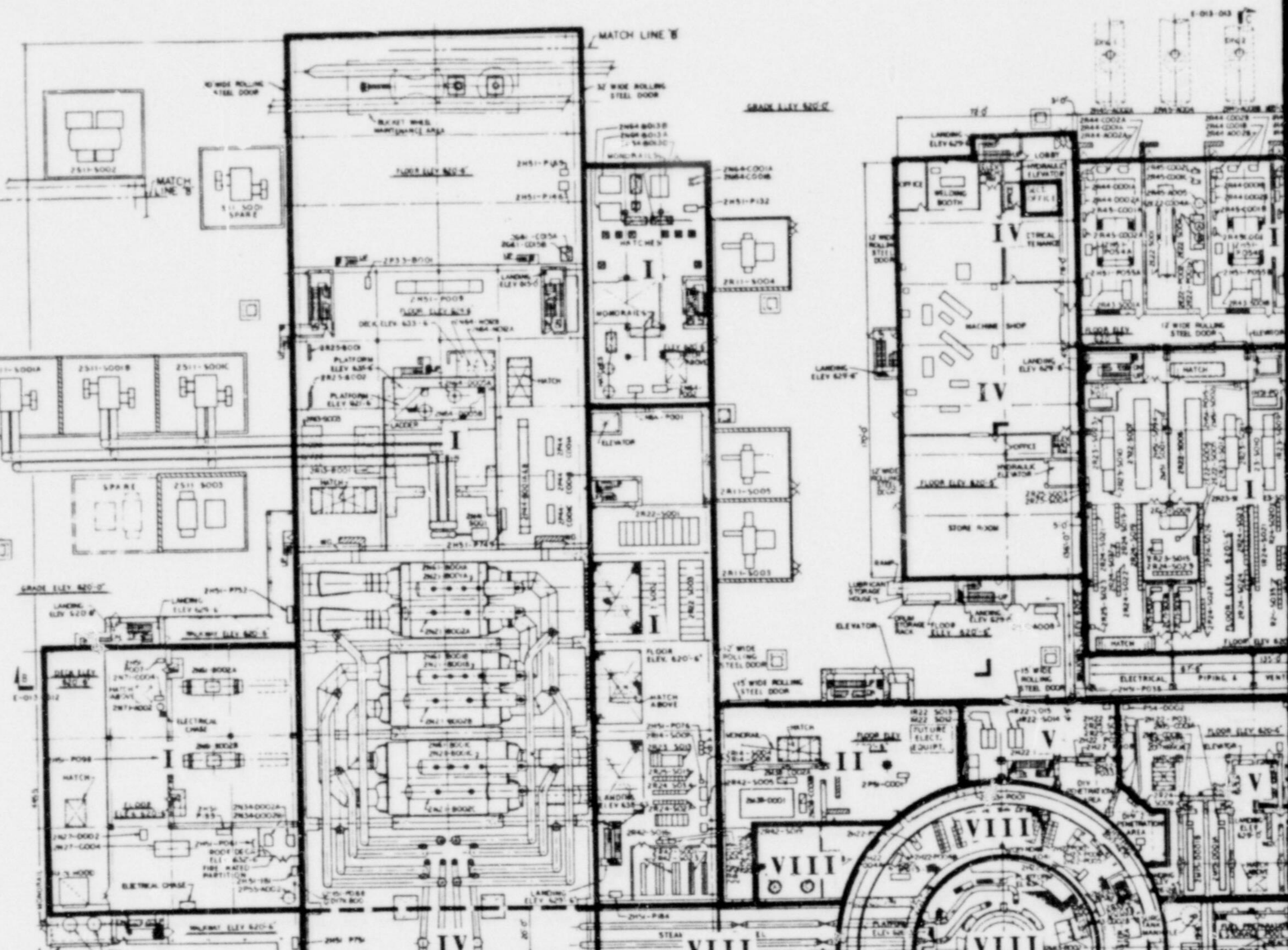


Figure 7.0-2



LEGEND:

RADIATION ZONE	MAX. DOSE RATE	RADIATION ZONE	MAX. DOSE RATE
I	<0.015 R/HR	V	<50 R/HR
II	<0.100 R/HR	VI	<500 R/HR
III	<0.500 R/HR	VII	<5000 R/HR
IV	<5 R/HR	VIII	>5000 R/HR

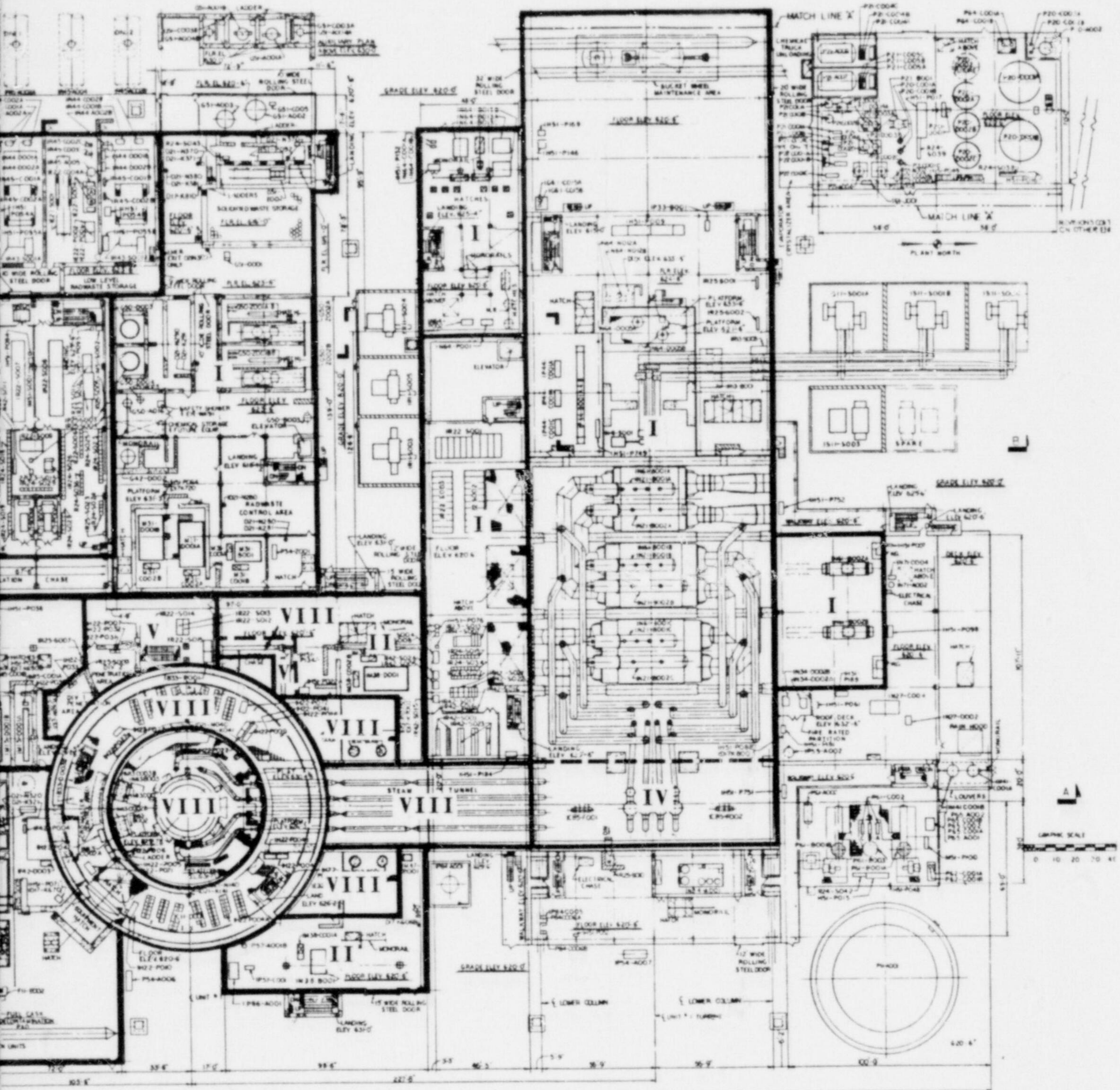
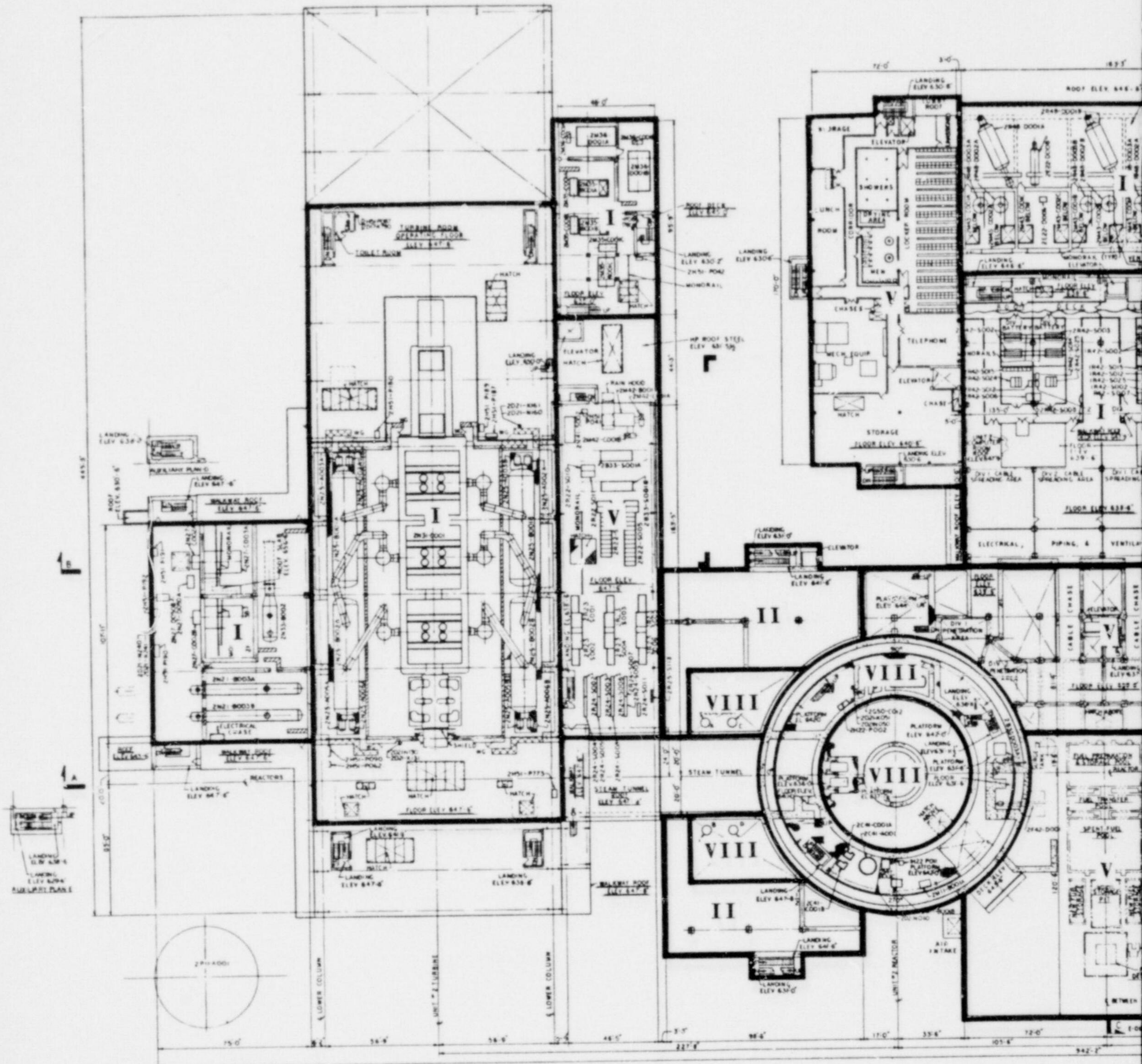


Figure 7.0-3



LEGEND:

RADIATION ZONE	MAX.DOSE RATE	RADIATION ZONE	MAX.DOSE RATE
I	<0.015 R/HR	V	<50 R/HR
II	<0.100 R/HR	VI	<500 R/HR
III	<0.500 R/HR	VII	<5000 R/HR
IV	<5 R/HR	VIII	>5000 R/HR

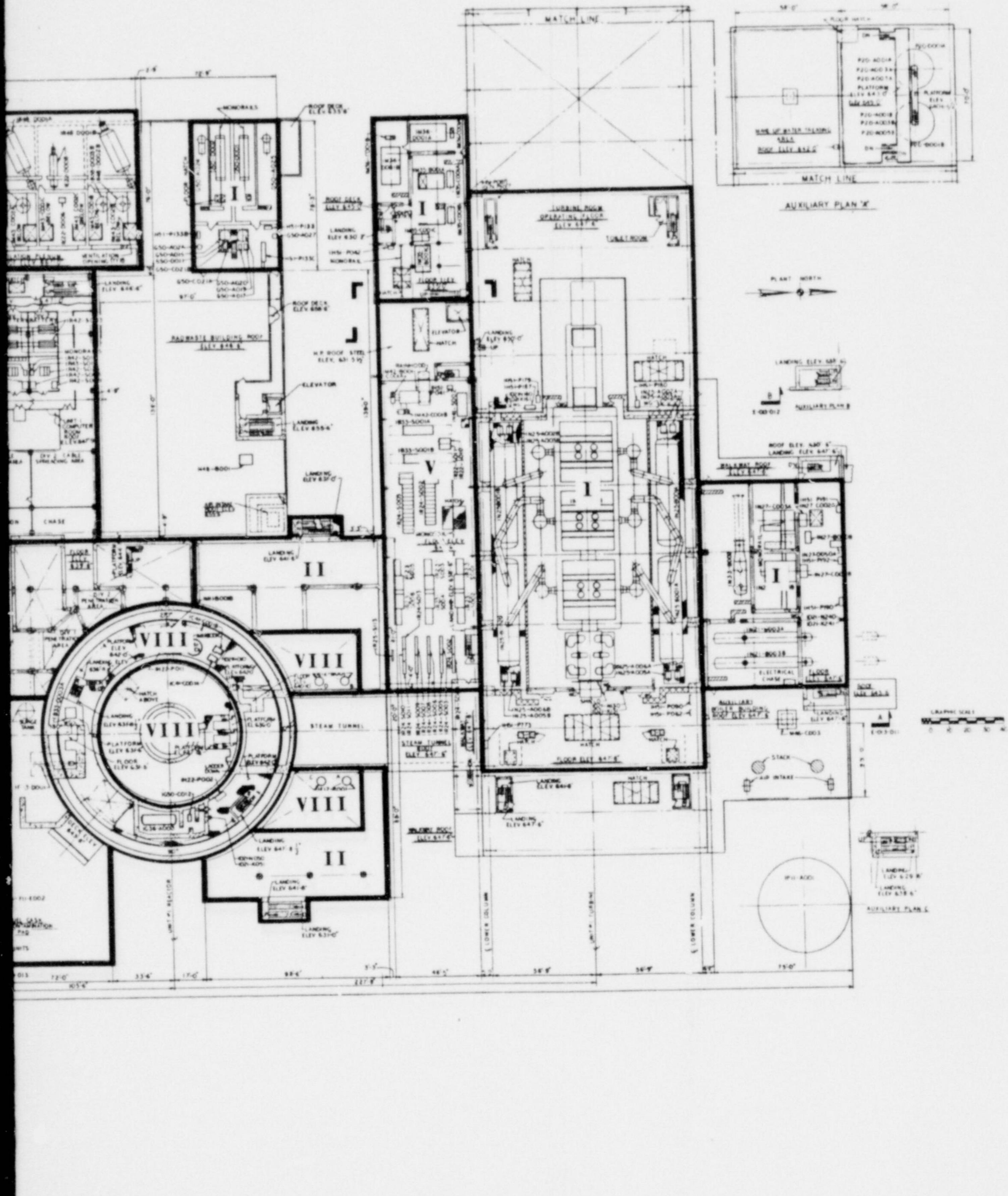
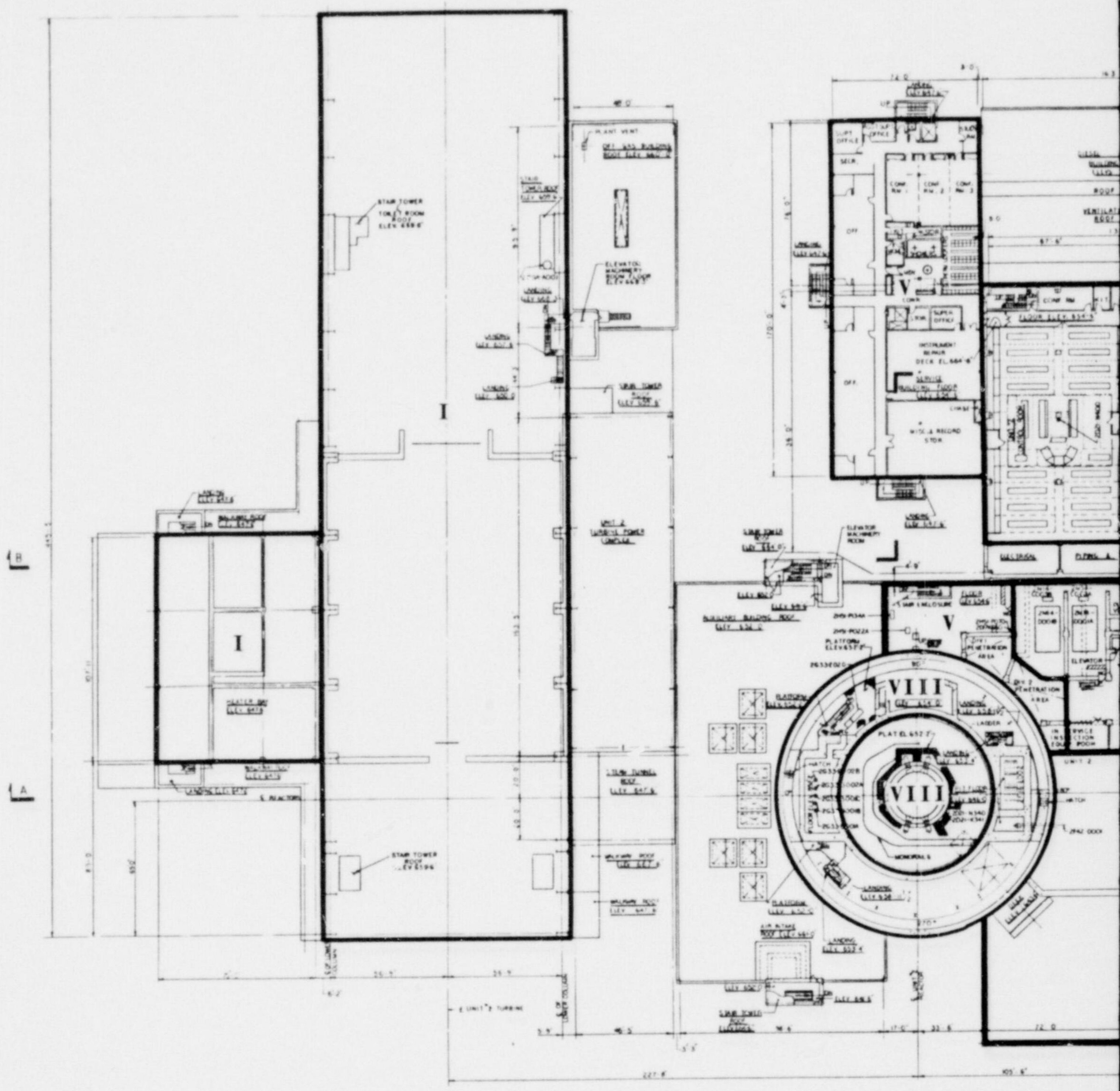

MATCH LINE
AUXILIARY PLAN 'A'
TURBINE ROOM OPERATING FLOOR
RADWASTE BUILDING ROOF
ELEVATOR
STEAM TUNNEL
PLANT NORTH
STACK
AIR INTAKE
AUXILIARY PLAN C
I
II
V
VIII

Figure 7.0-4



LEGEND:

RADIATION ZONE	MAX.DOSE RATE	RADIATION ZONE	MAX.DOSE RATE
I	<0.015 R/HR	V	<50 R/HR
II	<0.100 R/HR	VI	<500 R/HR
III	<0.500 R/HR	VII	<5000 R/HR
IV	<5 R/HR	VIII	>5000 R/HR

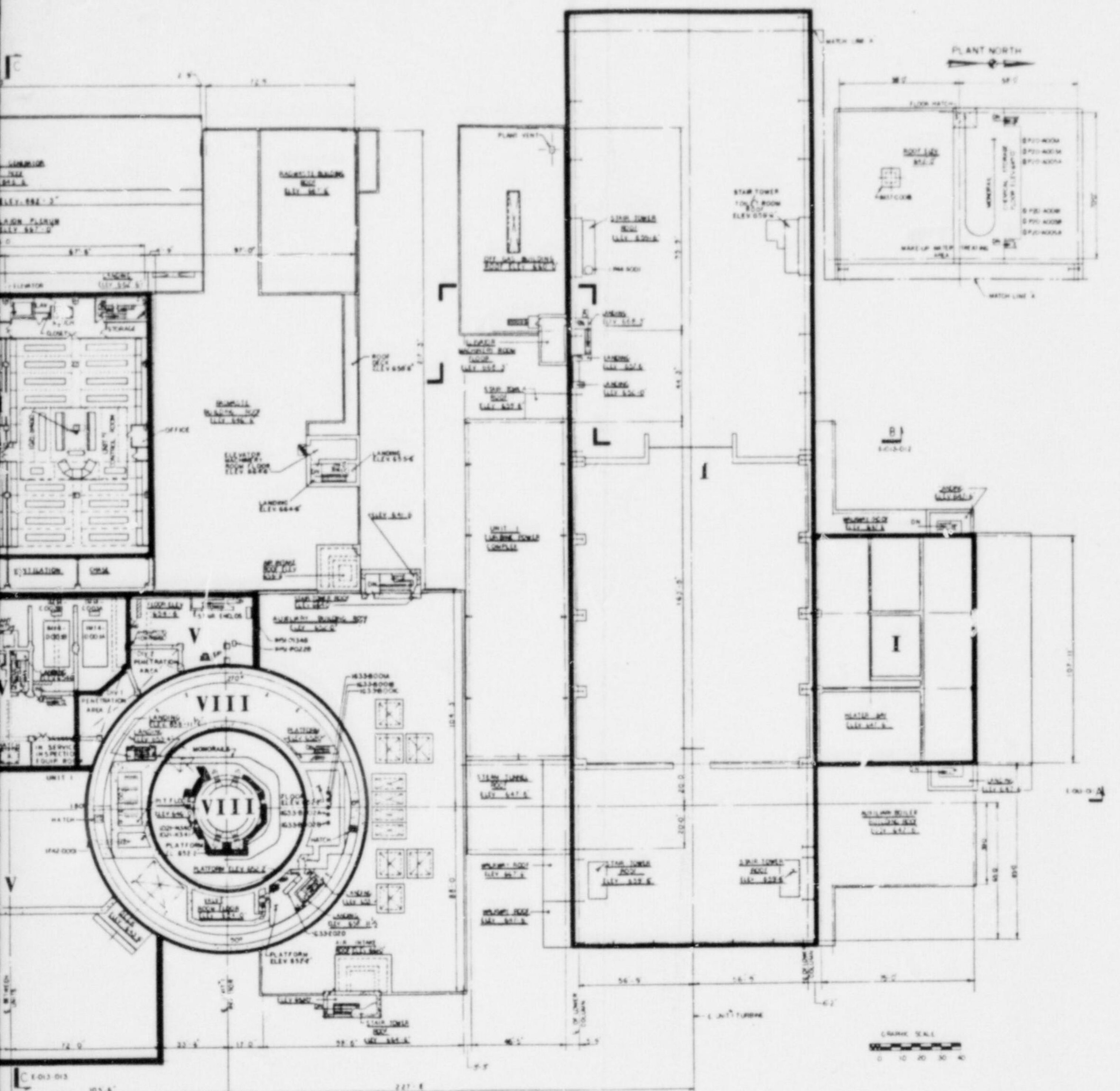
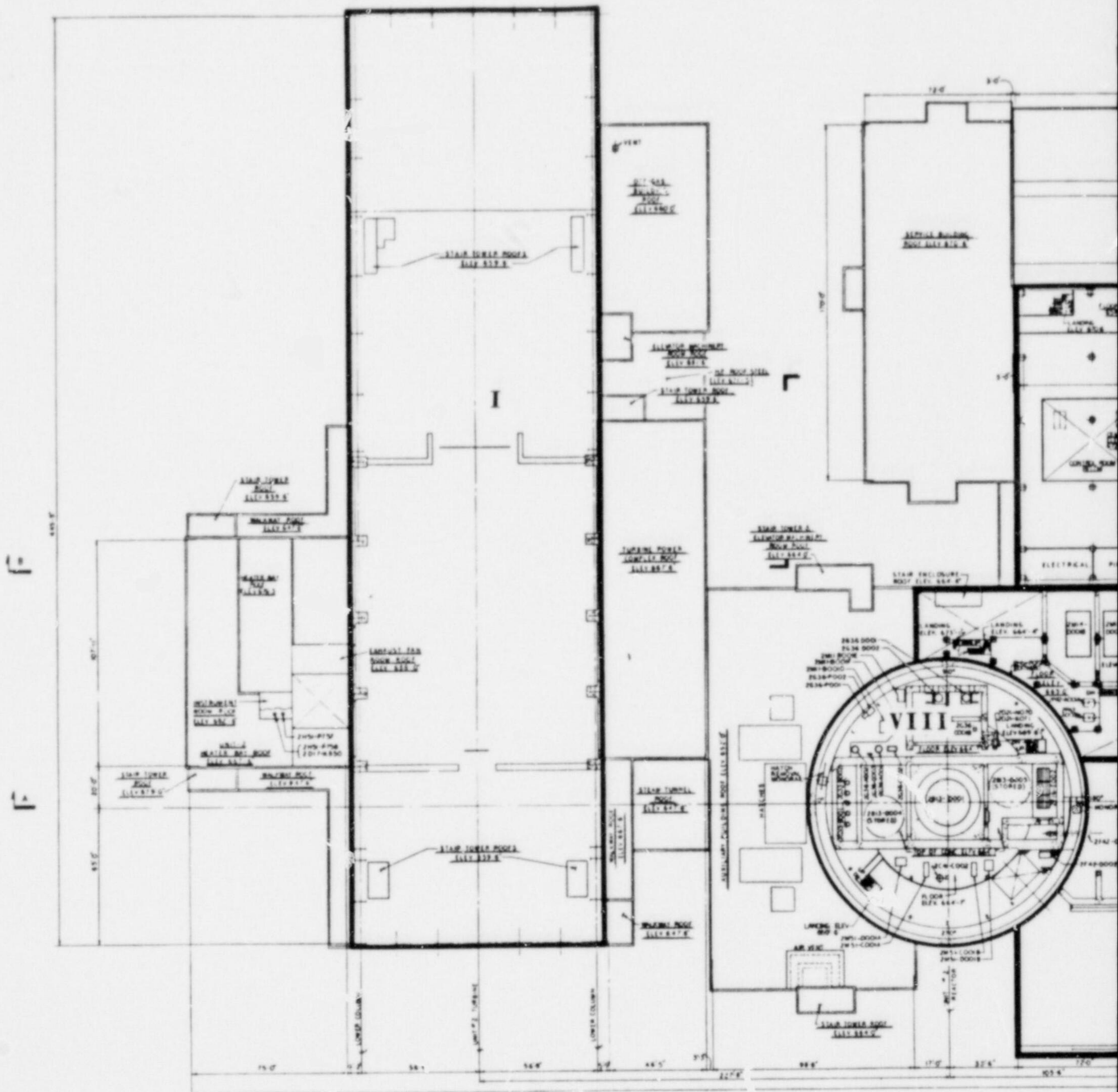


Figure 7.0-5



LEGEND:

RADIATION ZONE	MAX.DOSE RATE	RADIATION ZONE	MAX.DOSE RATE
I	<0.015 R/HR	V	<50 R/HR
II	<0.100 R/HR	VI	<500 R/HR
III	<0.500 R/HR	VII	< 5000 R/HR
IV	<5 R/HR	VIII	> 5000 R/HR

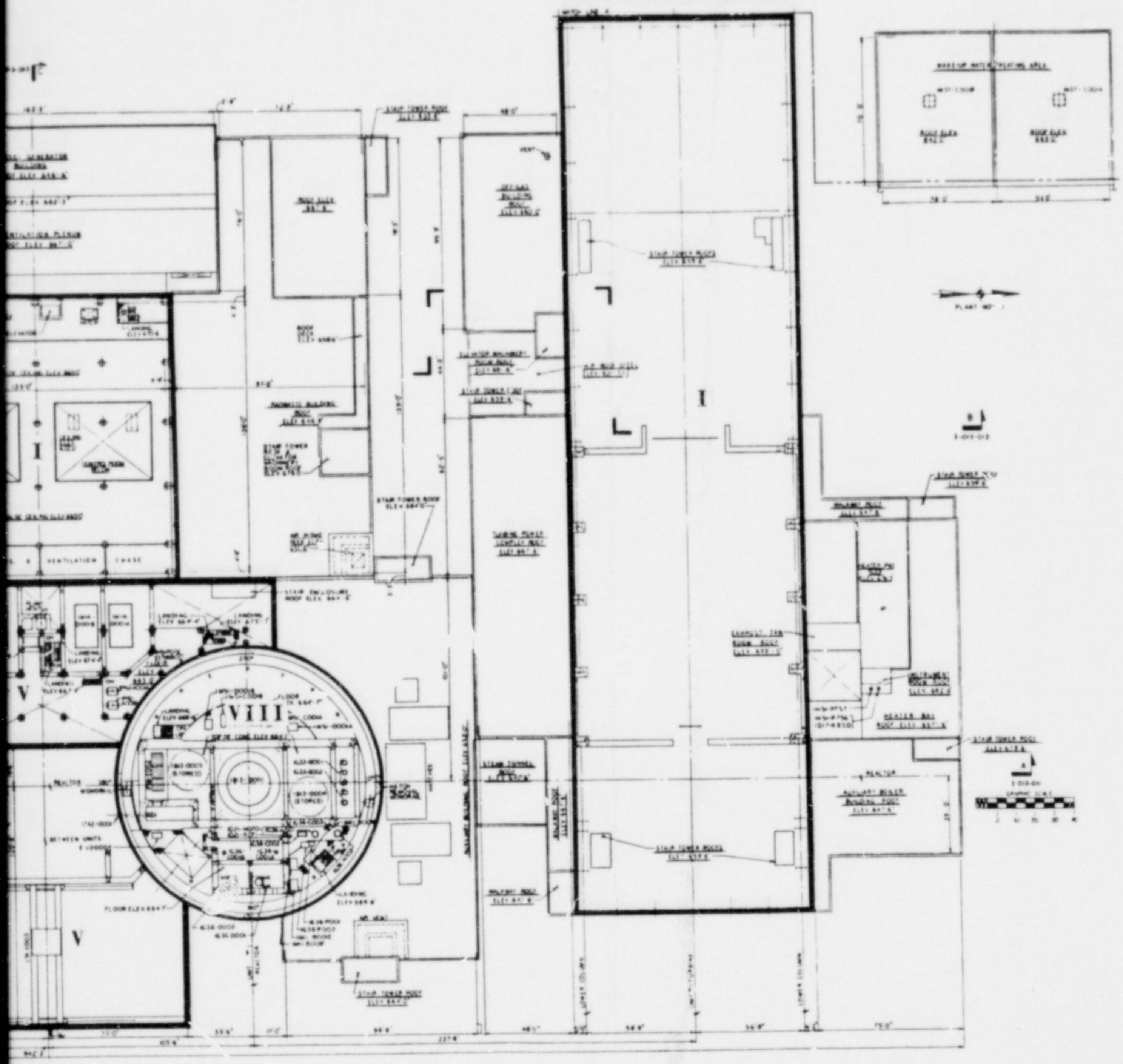
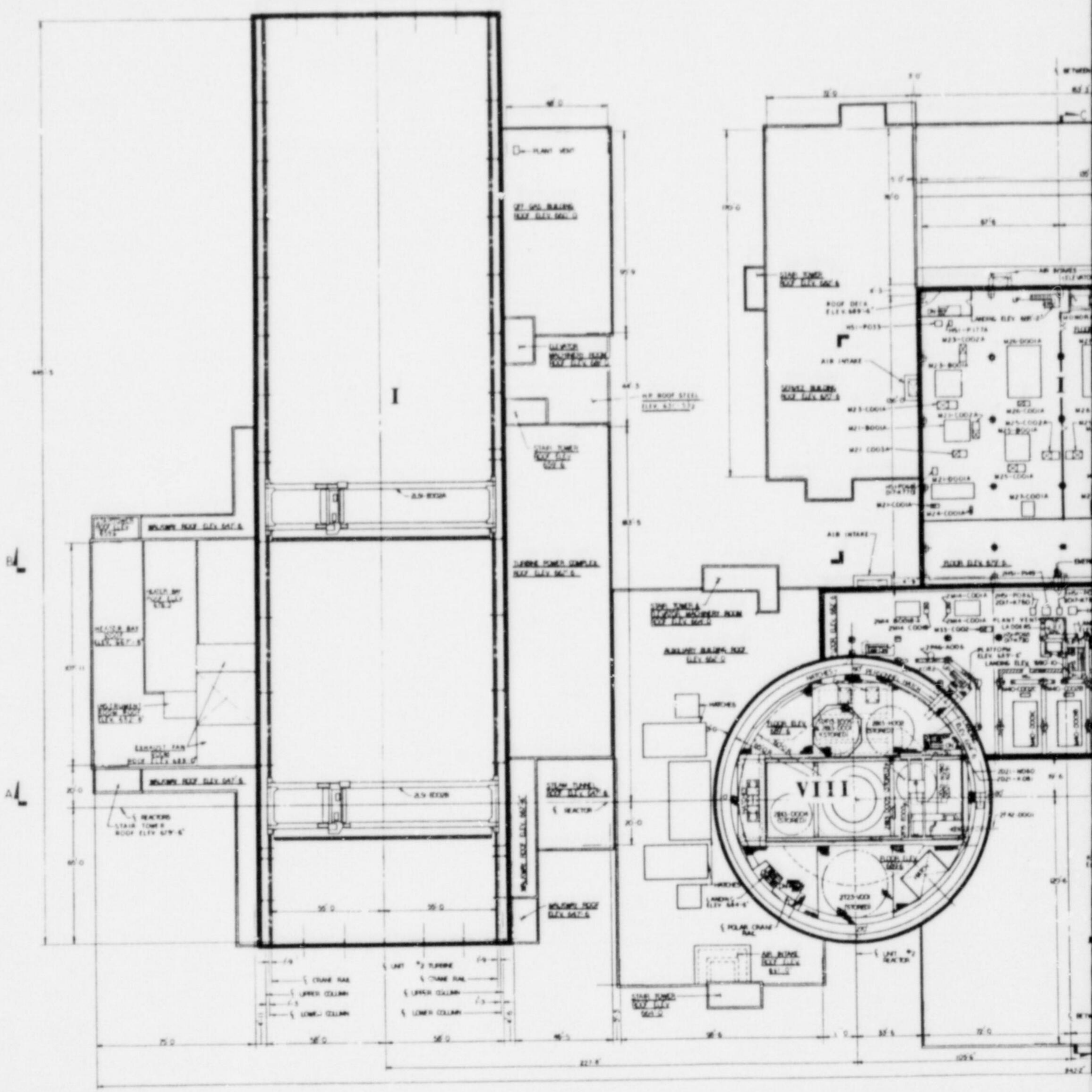


Figure 7.0-6



LEGEND:

RADIATION ZONE	MAX.DOSE RATE	RADIATION ZONE	MAX.DOSE RATE
I	<0.015 R/HR	V	<50 R/HR
II	<0.100 R/HR	VI	<500 R/HR
III	<0.500 R/HR	VII	<5000 R/HR
IV	<5 R/HR	VIII	>5000 R/HR

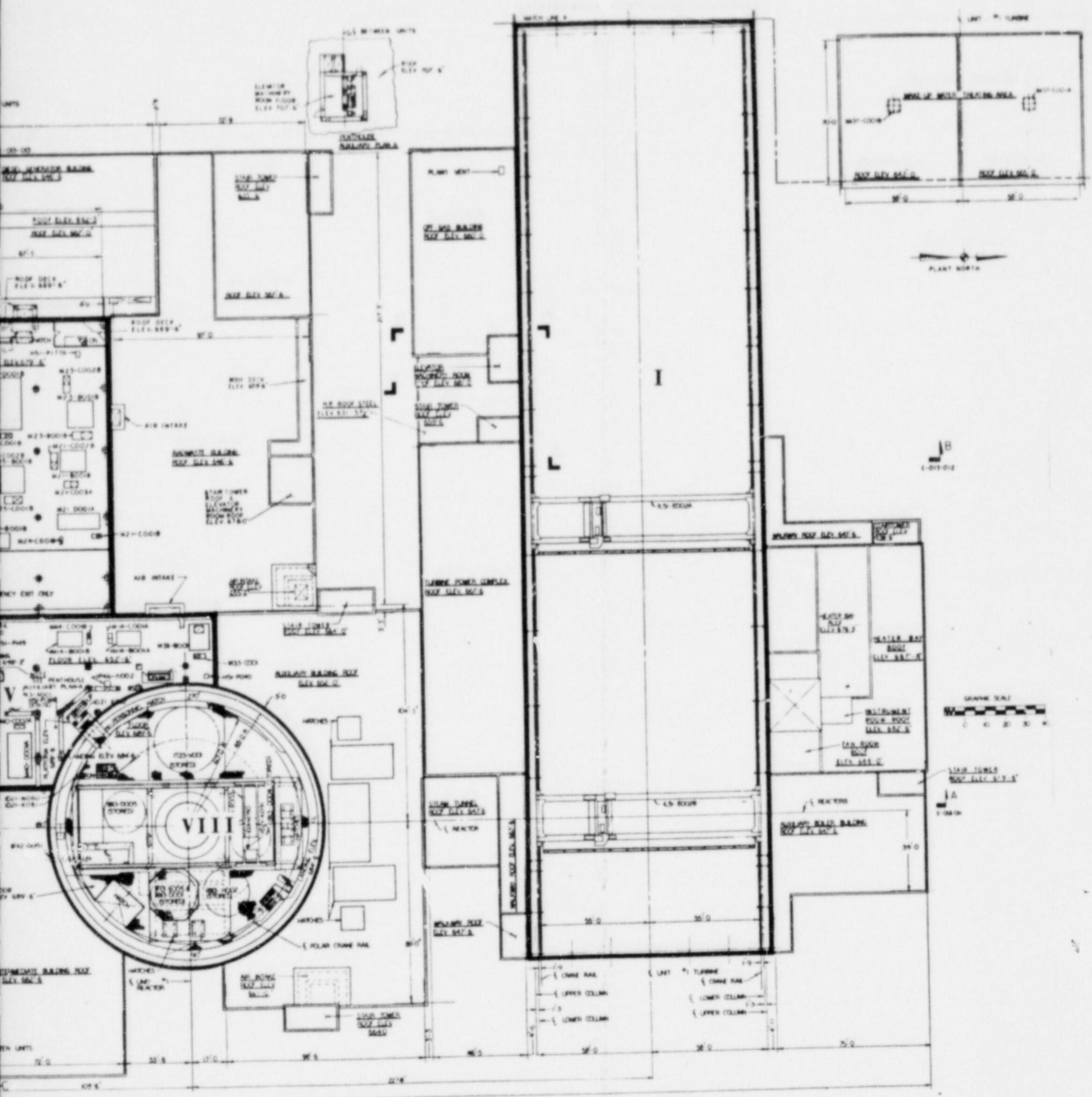
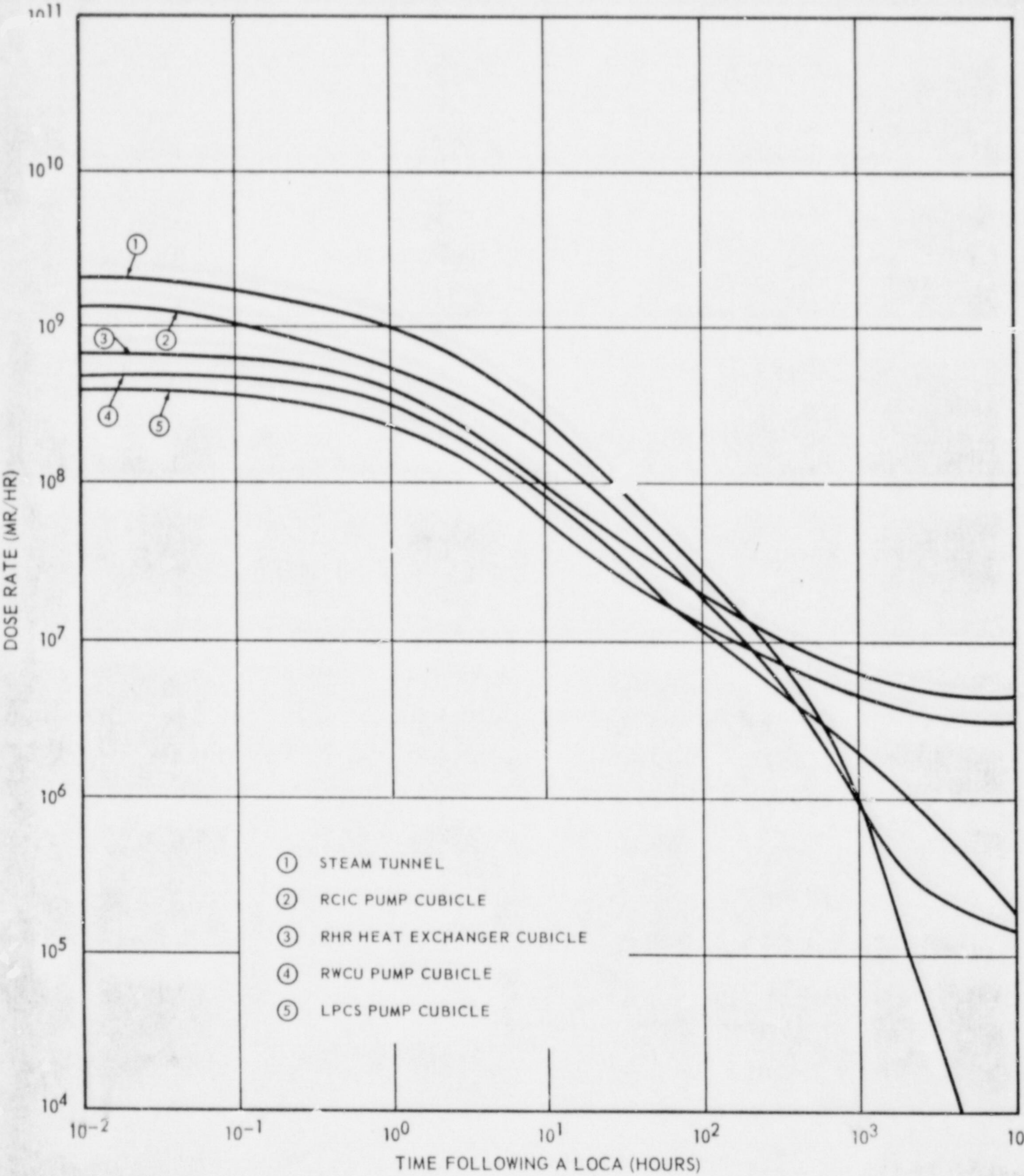
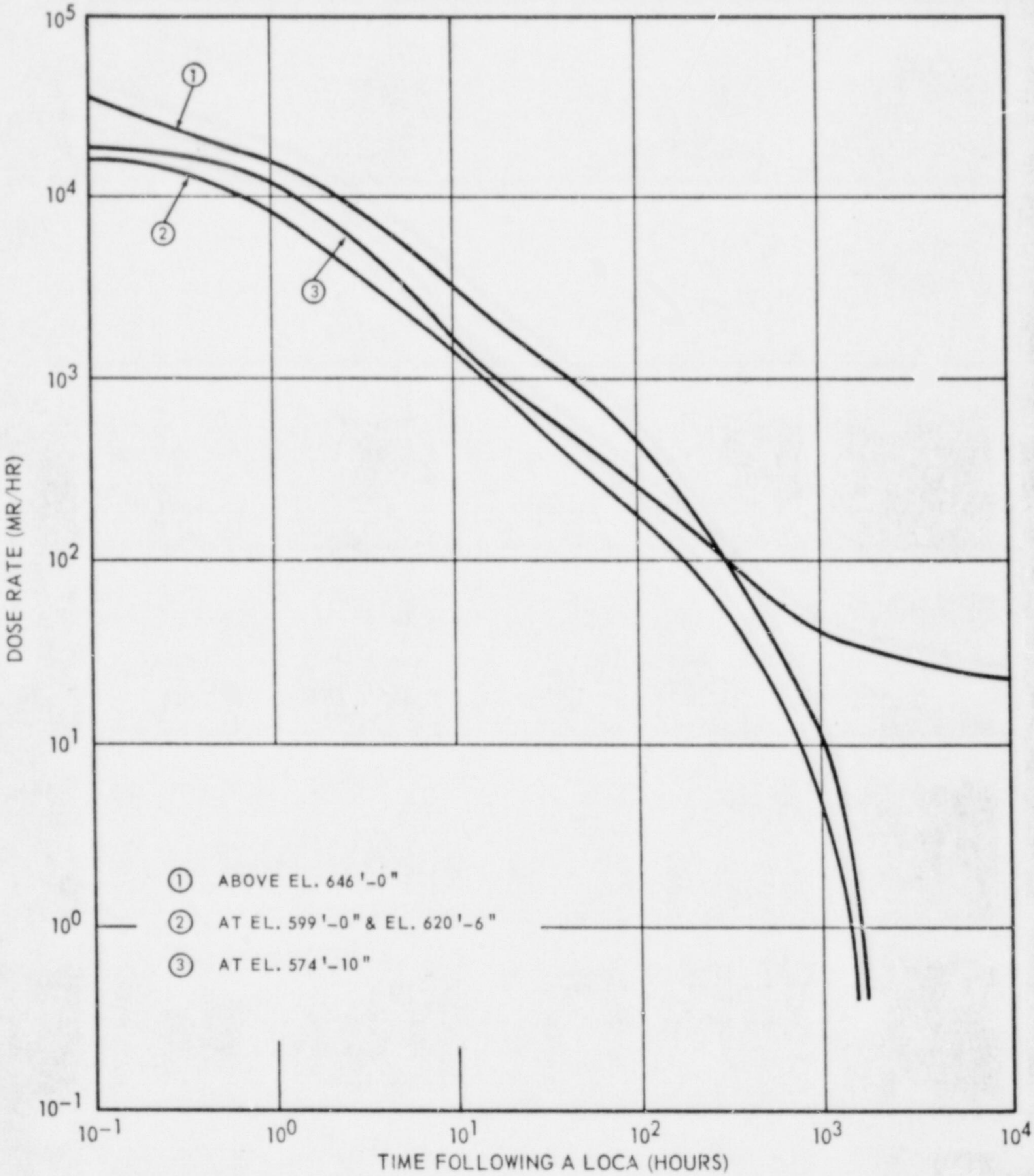


Figure 7.0-7



10^{11}
10^{10}
10^{9}
10^{8}
10^{7}
10^{6}
10^{5}
10^{4}
DOSE RATE (MR/HR)
10^{-2}
10^{-1}
10^{0}
10^{1}
10^{2}
10^{3}
10^{4}
TIME FOLLOWING A LOCA (HOURS)
1 STEAM TUNNEL
2 RCIC PUMP CUBICLE
3 RHR HEAT EXCHANGER CUBICLE
4 RWCU PUMP CUBICLE
5 LPCS PUMP CUBICLE

Figure 7.1-1



DOSE RATE (MR/HR)
TIME FOLLOWING A LOCA (HOURS)
10^5
10^4
10^3
10^2
10^1
10^0
10^{-1}
10^{-1}
10^0
10^1
10^2
10^3
10^4
1
2
3
1 ABOVE EL. 646'-0"
2 AT EL. 599'-0" & EL. 620'-6"
3 AT EL. 574'-10"

Figure 7.1-2

ADDITIONAL CONCERNS

1. Control of Access to Spent Fuel Transfer Tube Areas

Response

The response to this concern is provided in revised Section 12.3.2.2.1.

2. Location of whole body counter.

Response

The response to this concern is provided in revised Section 12.5.2.1 and Figure 12.5-1.

There are three Maintenance Access Areas. All three areas are posted with a placard stating that potentially lethal radiation fields are possible during fuel transfer. Temporary, local, audible and visible alarming radiation monitors are installed during refueling to alert personnel in the event of a significant increase in the radiation fields in these areas. Two of the areas have concrete plugs with lock bars across the plug to prevent access. The third area is the Valve Room Maintenance Access Area. This area is controlled by a locked door. The key for S90 is used to unlock these areas. If access is attempted during IFTS operation, an interlock secures the power to the IFTS and annunciates, IFTS ACCESS ROOM OCCUPIED, on P680-7A in the Control Room. Each of the areas has a panic button. The Valve Access Room's panic button is incorporated into a crash bar on the door. Pushing any panic button will de-energize the system and alarm in the Control Room, IFTS ACCESS ROOMS OCCUPIED. When the IFTS is energized, the door to the Valve Access Room is electrically locked, a rotating beacon inside the Valve Access Room is energized, and a warning light external to the room is illuminated. In addition, if entry to the Valve Access Room is attempted, a loud alarm will sound locally.

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.

c. Health Physics Office

d. Chemistry Office

e. Health Physics Service Room

f. Counting Room

g. Low Level Chemistry Laboratory

h. High Level Chemistry Laboratory

i. Laundry and Mask Cleaning Area

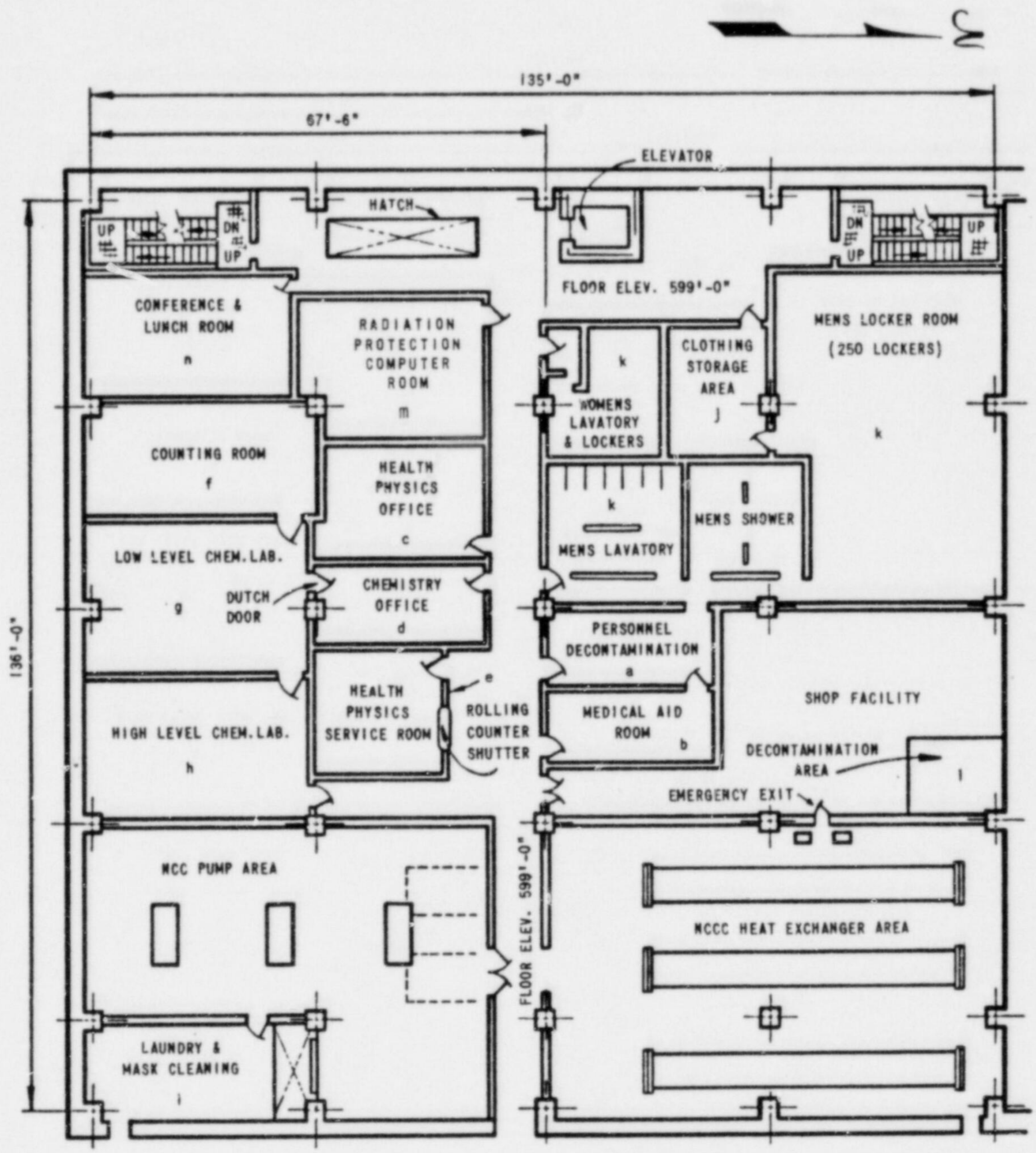
j. Clothing Storage Area

k. Mens and Women Locker Rooms and Lavoratories

l. Shop Facility and Decontamination Area

m. Radiation Protection Computer Room

n. Conference and Lunch Room



PERRY NUCLEAR POWER PLANT
THE CLEVELAND ELECTRIC ILLUMINATING COMPANY

Facility Locations

Figure 12.5-1