Appendix A

APPENDIX A

DESIGN CRITERIA FOR STRUCTURES AND EQUIPMENT

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1.0 DEFINITION OF SEISMIC DESIGN CLASSIFICATIONS

All equipment and structures are classified as Class I, and Class II, or Class III as recommended in:

a) TID-7024, "Nuclear Reactors and Earthquakes" August, 1963 and,

b) G.W. Housner, "Design of Nuclear Power Reactors Against Earthquakes", Proceedings of the Second World Conference on Earthquake Engineering, Vol. I, Japan 1960, Pg. 133, 134 and 137.

Class I

Those structures and components including instruments and controls whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of excessive amounts of radioactivity. Also, those structures and components vital to safe shutdown and isolation of the reactor.

Class II

Those structures and components which are important to reactor operation but not essential to safe shutdown and isolation of the reactor and whose failure could not result in the release of substantial amounts of radioactivity.

Class III

Those structures and components which are not directly related to reactor operation or containment.

All components, systems and structures classified as Class I are designed in accordance with the following criteria:

1. Primary steady state stresses, when combined with the seismic stress resulting from the response to a ground acceleration of 0.05g acting in the vertical and 0.1g acting in the horizontal planes simultaneously, are maintained within the allowable stress limits accepted as good practice and, where applicable set forth in the appropriate design standards, e.g., ASME Boiler and Pressure Vessel Code, USAS B31.1 Code for Pressure Piping, ACI 318 Building Code Requirements for Reinforced Concrete, and AISC Specifications for the Design and Erection of Structural Steel for Buildings.

2. Primary steady state stresses when combined with the seismic stress resulting from the response to a ground acceleration of 0.10g acting in the vertical and 0.15g acting in the horizontal planes simultaneously, are limited so that the function of the component, system or structure shall not be impaired as to prevent a safe and orderly shutdown of the plant.

All Class II structures and components are designed on the basis of a static analysis for a ground acceleration of 0.05g acting in the vertical and 0.1g acting in the horizontal directions simultaneously.

The structural design of all Class II structures meets the requirements of the applicable building code which is the "State Building Construction Code" State of New York, 1961. This code does not reference the Uniform Building Code.

Table A.1-1 gives the damping factors used in the design of components and structures.

The design of Class I structures and components utilizes the "response spectrum" approach in the analysis of the dynamic loads imparted by the earthquake. The analysis is based upon the response spectra shown on Figures A.1-1 and A.1-2.

The following method of analysis is applied to Class I structures and components, including instrumentation:

1. The natural period of vibration of the structure or component is determined.

2. The response acceleration of the component to the seismic motion is taken from the response spectrum curve at the appropriate period.

3. Stresses and deflections resulting from the combined influence of normal loads and the seismic load due to the design earthquake (0.05g acting in the vertical and 0.1g acting in the horizontal planes simultaneously) are calculated and checked against the limits imposed by the design standard.

4. Stresses and deflections resulting from the combined influence of normal loads and the seismic loads due to the maxium potential earthquake (0.1g acting in the vertical and 0.15g acting in the horizontal planes simultaneously) are calculated and checked to verify that deflections do not cause loss of function and that stresses do not produce rupture.

   Where the vibrator system is of a highly complex geometric shape, such as piping systems, the maximum response from the response curve with the appropriate damping factor is selected. By using this conservative value and demonstrating that the stresses are satisfactory, it becomes unnecessary to perform any further analysis to determine the natural periods of the system.

TABLE A.1-1

DAMPING FACTORS

Component	Per Cent of Critical Damping
Containment Structure	2.0
Concrete Support Structure of Reactor Vessel	2.0
Steel Assemblies:	
(a) Bolted or Riveted	2.5
(b) Welded	1.0
Vital Piping Systems	0.5
Concrete Structures above Ground:	
(a) Shear Wall	5.0
(b) Rigid Frame	5.0

2.0 CLASSIFICATION OF PARTICULAR STRUCTURES AND EQUIPMENT

Examples of particular structure and equipment classifications are given below. These classifications are not intended to be all-inclusive.

Item	Class
Buildings and Structures	
Containment (including all penetrations and air locks, the concrete shield, the liner and the interior structures)	I
Spent fuel pit	I
Control room	I
Diesel generator room	I
Intake structure (to the extent that water is always available to the service water pumps)	I
Service water screenwell	I
Auxiliary building (except for steel superstructure)	I
Turbine structure	III
Buildings containing conventional facilities and auxiliary building steel superstructure	III
Equipment, Piping and Supports*	
Reactor Control and Protection System	I
Radiation Monitoring System	I

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* Class I components (equipment, piping, instrumentation, etc.) located in or supported on a Class II structure will be protected from earthquake damage or will be backed up by other Class I components located in or supported by a Class I structure.

Process Instrumentation and Controls	I
Reactor	I
Vessel and its supports	
Vessel internals	
Fuel assemblies	
RCC assemblies and drive mechanisms	
Supporting and positioning members	
In-core instrumentation structure	
Reactor Coolant System	I
Piping and valves (including safety & relief valves)	
Steam generators	
Pressurizer	
Reactor coolant pumps	
Supporting and positioning members	
Engineered Safety Features	I
Safety Injection System (including safety injection and residual heat removal pumps, refueling water storage tank, accumulator tanks, boron injection tank, residual heat exchangers and connecting piping and valving)	
Containment Spray System (including spray pumps, spray headers, spray additive tank and connecting piping and valving)	
Containment Air Recirculation Cooling and Filtration System (including fans, coolers, ducts, valves, absolute filters and demisters)	

Auxiliary Building Ventilation System	I
Condensate storage tanks	I
Pressurizer relief tank	II
Residual heat removal loop	I
Containment Penetration and Weld Channel Pressurization System	I
Component cooling loop	I
Isolation Valve Seal Water System	I
Sampling System	II
Spent fuel pit cooling loop	II
Fuel transfer tube	I
Emergency Power Supply System	I
Diesel generators and fuel oil storage tank	
D-C power supply system	
Power distribution lines to equipment required for transformers and switchgear supplying the engineered safety features	
Control panel boards	
Motor control centers	
Control Equipment, facilities and lines necessary for the above Class I items	I

Waste Disposal System	I
Chemical drain tank	
Waste holdup tanks	
Sump tank	
Gas decay tanks	
Spent resin storage tank	
Reactor coolant drain tank	
Compressors	
Waste evaporator	
Waste evaporator feed pump	
Waste holdup tank pumps	
Sump tank pumps	
Interconnecting waste gas piping	
Waste Disposal System	III
All elements not listed as Class I	
Containment crane	I
Manipulator and other cranes	III
Conventional equipment, tanks and piping, other than I and II Classes	III
Emergency Boiler Feed, Service Water and Fire Protection Systems' pumps and piping	I

The Chemical and Volume Control System is considered Class I except for those items listed below,

Batching tank	II
Monitor tanks	II
Monitor tank pumps	II
Chemical mixing tank	II
Resin fill tank	III

3.0 CLASS I DESIGN CRITERIA FOR VESSELS AND PIPING

All components of the Reactor Coolant System and associated systems are designed to the standards of the applicable ASME Code or USAS Code. The loading combinations which are employed in the design of Class I components of these systems, i.e., vessels, piping, supports, vessel internals and other applicable components, are given in Table A.3-1.

This Table also indicates the stress limits which are used in the design of the listed equipment for the various loading combinations.

To be able to perform their function, i.e., allow core shutdown and cooling the reactor vessel internals must satisfy deformation limits which are more restrictive than the stress limits shown in Table A.3-1. For this reason the reactor vessel internals are treated separately.

Piping, Vessels and Supports

The reasoning for selection of the load combinations and stress limits given in Table A.3-1. is as follows. For the design earthquake, the nuclear steam supply system is designed to be capable of continued safe operation, i.e., for the combination of normal loads and design earthquake loading. Critical equipment and supports needed for this purpose are required to operate within normal design limits as shown in line 2 of Table A.3-1.

In the case of the maximum potential earthquake, it is only necessary to ensure that critical components do not lose their capability to perform their safety function, i.e., shut the plant down and maintain it in a safe condition. This capability is ensured by maintaining the stress limits as shown in line 3 of Table A.3-1. No rupture of a Class I pipe can be caused by the occurrence of the maximum potential earthquake.

Careful design and thorough quality control during manufacture and construction and periodic inspection during plant life, ensures that the independent occurrence of a reactor coolant pipe rupture is extremely remote. If it is assumed that a reactor coolant pipe ruptures, the stresses in the unbroken leg will be as noted in line 4 of Table A.3-1.

Reactor Vessel Internals

Design Criteria for Normal Operation

The internals and core are designed for normal operation conditions and subjected to loads of mechanical, hydraulic, and thermal origin. The response of the structure under the design earthquake is included in this category.

The stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Article 4, has been adopted as a guide for the design of the internals and core with exception of those fabrication techniques and materials which are not covered by the Code, such as the fuel rod cladding. Seismic stresses are combined in the most conservative way and are considered primary stresses.

The members are designed under the basic principles of: (1) maintaining distortions within acceptable limits, (2) keeping the stress levels within acceptable limits, and (3) prevention of fatigue failures.

Design Criteria for Abnormal Operation

The abnormal design condition assumes blowdown effects due to a reactor coolant pipe double-ended break.

For this condition the criteria for acceptability are that the reactor be capable of safe shutdown and that the engineered safety features are able to operate as designed. Consequently, the limitations established on the internals for these types of loads are concerned principally with the maximum allowable deflections. The deflection criteria for critical structures under abnormal operation are presented in Table A.3-2

TABLE A.3-2

INTERNALS DEFLECTIONS UNDER ABNORMAL OPERATION
(Inches)

	Allowable Limit	No Loss-of-Function Limit
Upper Barrel, expansion/compression (to assure sufficient inlet flow area/and to prevent the barrel from touching any guide tube to avoid disturbing the RCC guide structure).	3	6
Upper Package, axial deflection (to maintain the control rod guide structure geometry).	1	2
RCC Guide Tube, cross section distortion (to avoid interference between the RCC elements and the guides).	0.035	0.072
RCC Guide Tube, deflection as a beam (to be consistent with conditions under which ability to trip has been tested).	1.0	1.5
Fuel Assembly Thimbles, cross section distortion (to avoid interference between the control rods and the guides).	0.035	0.072

Reactor Vessel

The criteria for movement of the reactor vessel, under the worst combination of loads, i.e., normal plus the maximum potential earthquake or normal plus reactor coolant pipe rupture loads, assures that the radial movement of the reactor vessel will not exceed the clearance between the reactor coolant piping and the surrounding concrete.

The relative motions between Reactor Coolant System components are controlled by the structures which are used to support the reactor vessel, the steam generators, the pressurizer and the reactor coolant pumps.

The maximum movement of the reactor vessel under the worst combination of loads, i.e., normal plus maximum potential earthquake or normal plus reactor coolant pipe rupture loads comprises an end deflection on the safety injection piping which is small even in comparison with that resulting from thermal growth during plant heat up, and is well within the flexability of the design of the piping system.

The supports are designed to limit the stresses in the pipe to the stress limits given in Table A.3-1.

TABLE A.3-1 - LOADING COMBINATIONS AND STRESS LIMITS

LOADING COMBINATIONS	VESSELS	PIPING	SUPPORTS
1. Normal Loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq S$ $P_L + P_B \leq S$	Working Stresses or Applicable Factored Load Design Values
2 Normal + Design Earthquake Loads	$P_m \leq S_m$ $P_L + P_B \leq 1.5\ S_m$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ S$	1-1/3 Working Stresses or Applicable Factored Load Design Values
3. Normal + Maximum Potential Earthquake Loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ (1.5\ S)$	Deflections and Stresses of Supports Limited to Maintain Supported Equipment Within their Stress Limits
4. Normal + Pipe Rupture Loads	$P_m \leq 1.2\ S_m$ $P_L + P_B \leq 1.2\ (1.5\ S_m)$	$P_m \leq 1.2\ S$ $P_L + P_B \leq 1.2\ (1.5\ S)$	Deflections and Stresses of Supports Limited to Maintain Supported Equipment Within Their Stress Limits

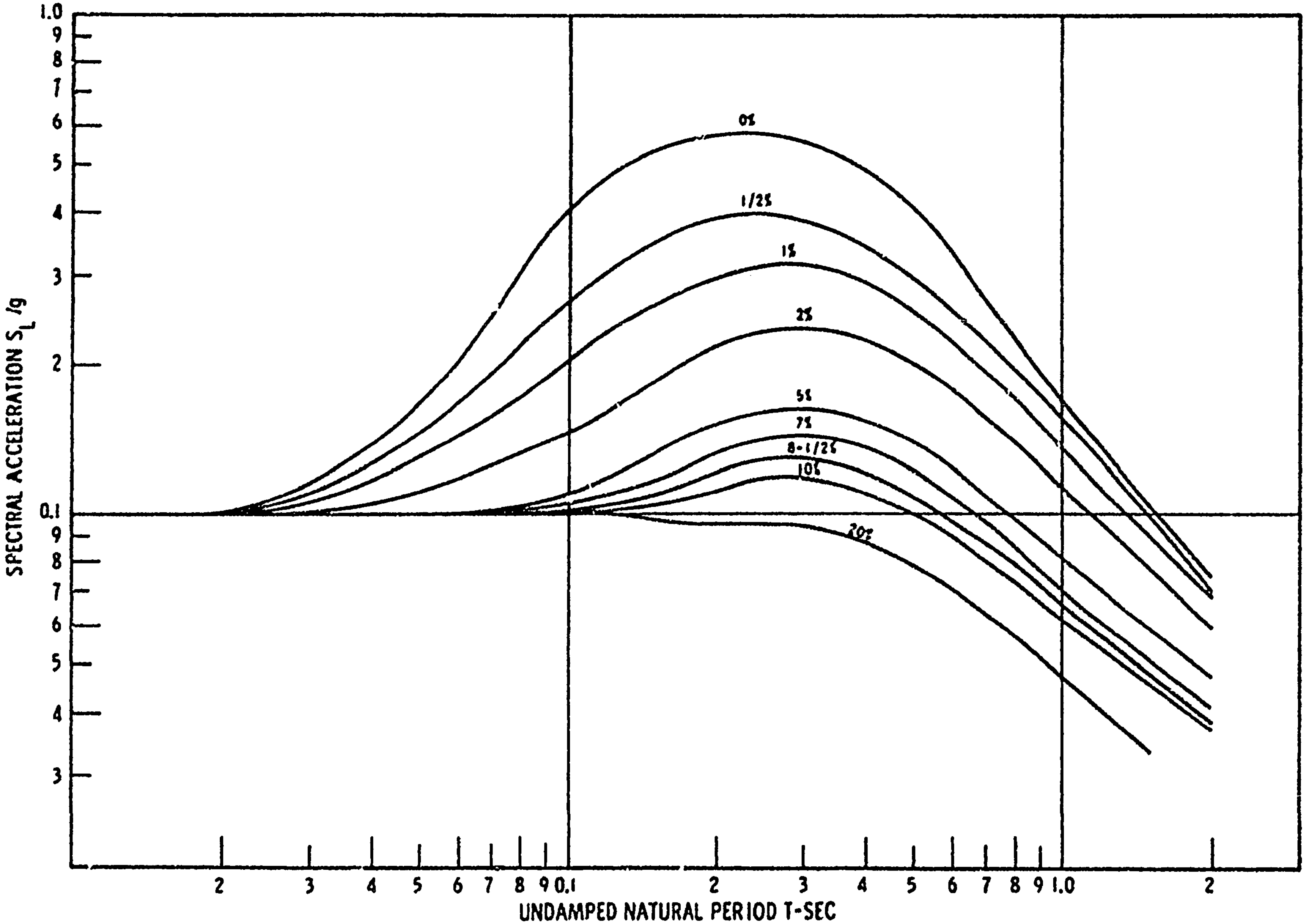
Where P_m = primary general membrane stress; or stress intensity

P_L = primary local membrane stress; or str[illegible]

P_B = primary bending stress; or stress intensity

S_m = stress intensity value from ASME B & PV Code, Section III

S = allowable stress from USAS B31.1 Code for Pressure Piping



10% OF GRAVITY RESPONSE SPECTRA

FIGURE A.1-1

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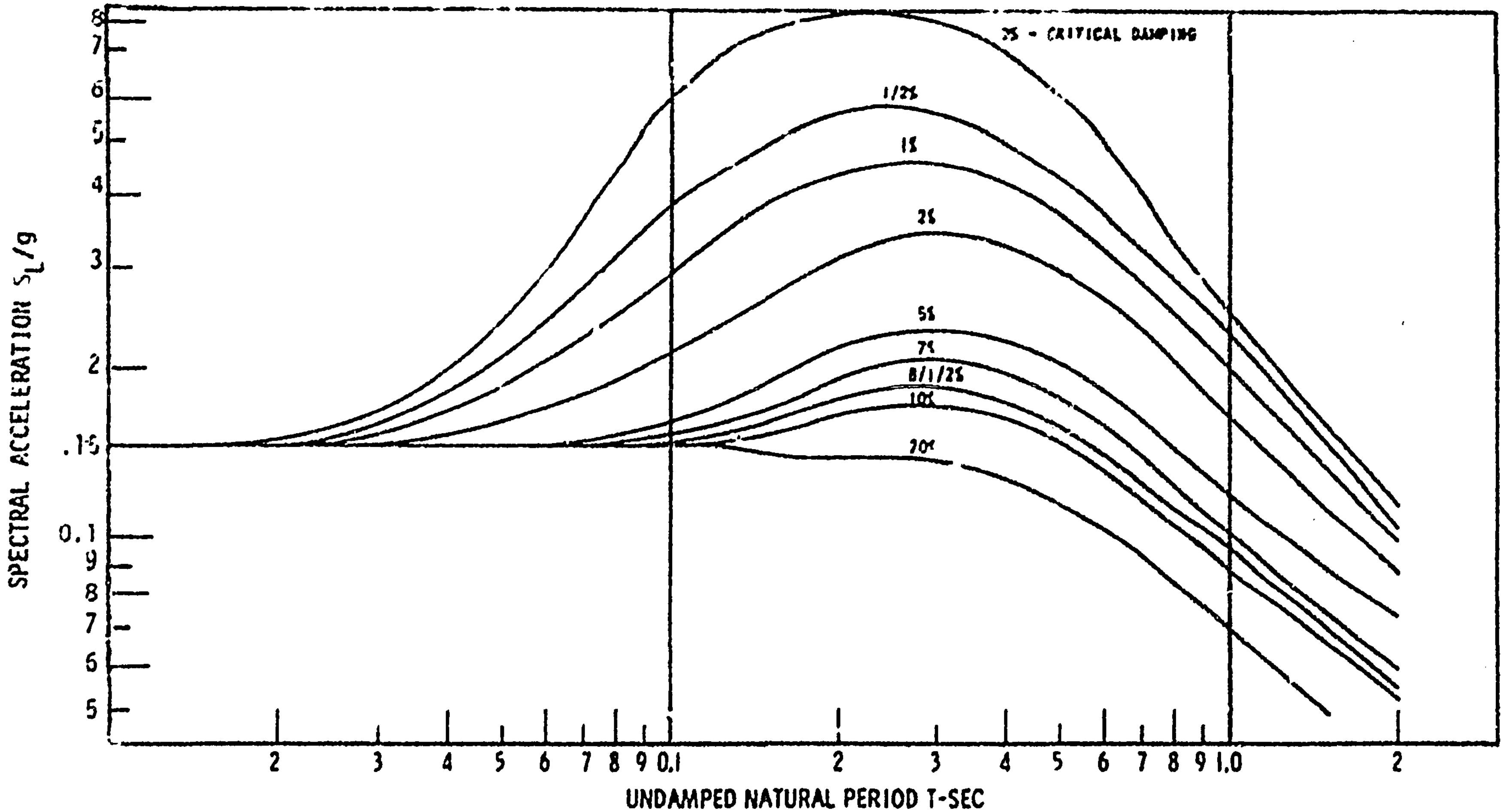

15% OF GRAVITY RESPONSE SPECTRA

FIGURE A.1-2

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Appendix B

APPENDIX B

QUALITY CONTROL PROGRAM

GENERAL

Herein is described in the applicant's quality assurance program for the IPP #2 project. This program is comprehensive and covers all phases of construction both off and on site. Included are all areas of activity which have influences on plant integrity, including design (drawings and specifications), manufacture and field erection and installation and all related activities such as cleanliness control, shipment and storage.

The program places special emphasis on the reactor coolant and safety systems, the containment and the other components necessary for the safety of the nuclear portion of the plant. The description which follows delineates the quality assurance organization and procedures but does not repeat the design and specification requirements set forth in the FSAR.

There are three principal organizations active in the area of Quality Assurance to insure the safety and integrity of the completed plant: Consolidated Edison, the applicant-owner; Westinghouse, the prime contractor; and United Engineers and Constructors, the Architect-Engineer-Constructor and a sub-contractor to Westinghouse. The direct activities of each organization are in turn supplemented by the quality control activities of other organizations, either in the form of specific subcontracts for this purpose or as a part of their responsibility as a supplier of material or equipment, conforming to specification. Where one of these organizations does not have the first line quality control responsibility, a surveillance (auditing and monitoring) function is performed by the contractor in any contractor-subcontractor relationship.

Westinghouse has the prime responsibility to provide all material and equipment for all construction. This responsibility is in turn dispatched through various channels. Westinghouse is the direct supplier of some equipment while in other cases it is supplied by way of purchase from other manufacturers. All of the construction materials and the remainder of the equipment are purchased by United Engineers. The relationship among these organizations is shown on Figure 1.

The scope and nature of the quality assurance functions of each of these three participants and their relationship to one another are discussed as follows.

CONSOLIDATED EDISON

While much of the quality assurance is carried out by the Prime Contractor and the Architect-Engineer, Consolidated Edison recognizes that it has ultimate responsibility in this area.

Con Edison's quality assurance function includes monitoring the Westinghouse and United Engineers efforts in critical areas through an independent detailed vendor surveillance by the United States Testing Company and company engineers during selected phases of in-shop manufacturing and on-site surveillance during the plant construction phase. In addition, there is continuous on-site surveillance by Con Edison personnel experienced in the construction field.

USTC keeps informed of the manufacturing status of critical items and makes visits from time to time to the various manufacturing facilities on behalf of Con Edison. Physical and chemical certifications are reviewed for compliance with applicable specifications. Test procedures involving radiograph, ultrasonics, etc. are reviewed against accepted standards and fabrication techniques are reviewed for general shop practice.

Selected tests and inspections are witnessed and test results including radiographic films are reviewed. Where procedural non-conformances or questionable test results of a minor nature are found, USTC requests an on-the-spot procedural revision or retest. Where procedural nonconformances or questionable test results of a major significant nature are found, USTC contacts the quality control engineering representative at Con Edison who assesses the situation. Westinghouse is contacted directly by Con Edison for necessary corrective action when required. Written reports of all USTC surveillance visits are forwarded to Con Edison and Westinghouse.

Con Edison maintains on-site a permanently assigned Superintendent of Construction and his full-time staff whose prime function is to ensure that the on-site work is accomplished in accordance with the contractual requirements, and to ensure that the on-site quality control programs are being properly implemented.

The superintendent has the authority to stop work in any area that he considers can affect the technical adequacy or safety of the plant.

Day to day progress is closely monitored by the superintendent. When major work is in progress USTC personnel supplement the Con Edison Personnel. Concrete pours in the containment structure are witnessed and concrete samples are tested for physical, chemical and mechanical properties. Rebar placement, cadwelding, liner welds and installation of critical systems components and piping are witnessed and the necessary inspection reports are reviewed. Written reports of all USTC surveillance visits and concrete sample tests are forwarded to Con Edison and Westinghouse.

WESTINGHOUSE

In the capacity of Prime Contractor, Westinghouse is responsible for the provisions of all material and equipment and for all construction. In discharging this responsibility, Westinghouse recognizes the importance of Quality Assurance throughout all stages of design, fabrication and construction, and accordingly maintains a comprehensive overall Quality Control Program. This program assures that design, engineering, materials and workmanship employed in the fabrication and construction of the facility meet safety, operability, maintainability and reliability objectives previously established. Particular emphasis is placed on the Nuclear Steam Supply System and other critical features.

Records and documentation pertinent to the required quality of materials, workmanship and testing are maintained by the Quality Control and Reliability Section.

Organization

The Organization Chart for Westinghouse relating to Quality Control is shown in Figure 2.

Component Design and Quality Control Groups have the responsibility for insuring off-site Quality Control, i.e., up to and including component fabrication and dispatching for shipment.

The Manager of Quality Control and Reliability is responsible for establishing and implementing the overall quality control program through three specialized Quality Control units each headed by its own manager. There are two mechanical Quality Control groups, one is responsible for vessels and tanks and the other is responsible for the balance of the mechanical equipment. There is also a Quality Control group for all electrical instrumentation and control equipment. There are senior Quality Control Engineers in each unit who have expertise in specific categories of equipment and who carry out the Quality Control Engineering planning. These engineers, supplemented by the efforts of Quality Control representatives, also do surveillance of the suppliers.

The Nuclear Power Service Group has the responsibility for insuring on-site Quality Control. i.e., from receiving all equipment and materials on-site, through erection, to plant start up.

All of the Groups are staffed by capable and experienced engineers who collectively provide the experience and effort to implement the overall Quality Control Plan to assure the quality of the finished plant.

Components Supplied By Westinghouse

All major components are supplied by Westinghouse, either directly through Westinghouse equipment manufacturing divisions or by way of purchase from other manufacturers.

There are four stages in the quality control program during component fabrication to ensure the required degree of quality of the finished product. They are:

A. Supplier Evaluation
B. Equipment Specifications
C. Purchase Order Review
D. Supplier Surveillance during Fabrication

A. Supplier Evaluation

An evaluation of prospective suppliers is conducted prior to award of a contract for important components. This evaluation establishes that the supplier has acceptable design, manufacturing and quality control capabilit Elements considered in conducting the evaluation include:

1. Previous experience with the supplier
2. Physical plant facilities
3. Quality control program
4. Personnel qualifications

5. Material control and inspection
6. In-process inspection
7. Assembly and test capability
8. Tool and gage control
9. Special processes required
10. Non-destructive testing
11. Inspection and test equipment

Responsibility for this evaluation is a function of the Quality Control Groups.

B. Equipment Specification

Individual equipment specifications cover all aspects of equipment design, manufacture, inspection and testing. For Class I components, such as those in the reactor coolant system, a specification which defines the supplier's quality control requirements is made a part of each purchase order. Specific requirements may include:

1. Drawings and change procedures.
2. Procedures and revisions covering welding, heat treating and other process control documents.
3. Inspection plans covering contracted and sub-contracted work.
4. Identification and disposal of non-conforming material or components.
5. Quality control records.
6. Material control.

These requirements are similar to those of Appendix IX to Section III of the ASME Boiler and Pressure Vessel Code.

Responsibility for assuring conformity with these requirements is a function of the Quality Control Group.

C. Purchase Order Review

Purchase orders, including the applicable drawings, welding specifications, non-destructive test procedures and other process control documents required to manufacture, inspect, and test the equipment are reviewed by Westinghouse Engineering and Material personnel to m : sure they include all contract requirements, meet applicable codes and quality control requirements and are compatible with the supplier's capabilities. In cases where existing inspection techniques are found to be inadequate to assure requisite quality, discussions are held with the engineers concerned, and the necessary adjustments are made. This procedure is applied to all types of material and components and provides the means for maintaining surveillance over the quality of procurements.

D. Supplier Surveillance

The Design Engineer/Quality Control Engineer team develops specific Quality Control Plans which details the inspections, surveillance, record verification, and surveillance which Westinghouse Quality Control personnel perform in suppliers' plants. They cover the entire cycle from purchase order placement through manufacture, inspection and test. These plans are developed as follows:

The many requirements of the equipment specification and its referenced specifications that establish the quality of equipment are listed in a Quality Requirements Summary for each category of equipment. This serves as a working tool from which the Westinghouse Quality Control Plan is developed. This Quality Control Plan is the action plan which details the inspection and surveillance done by Westinghouse Quality Control Engineers and field representatives in the suppliers' shops. It covers the auditing of the supplier's quality control organization, system, and procedures; surveillance of key shop operations such as welding, non-destructive testing and production testing, specific inspections such as radiographic, ultrasonic, material test reports, key dimensions, and other important requirements.

The design engineer also visits the supplier's plant periodically and monitors overall compliance with specification requirements. Particular emphasis is placed on inspections of the first units of fabricated new designs and from new suppliers.

Through the Westinghouse visit to the supplier's plant, the supplier's Quality Control system, in-process work, testing and records are evaluated to assure that he meets contract requirements. The extent of surveillance performed in a supplier's facility depends on the complexity of the component, the supplier's past performance, the observed effectiveness of the supplier's own quality control procedures, the relative importance with respect to safety of the component being fabricated, and whether the component is of an established or new design. The Quality Control Plan emphasizes the basic responsibility of the supplier to have systems and procedures to meet the requirements of the specifications and drawings and to control the quality of his product. The supplier is required to inspect and test his products and to furnish objective evidence that control requirements have been met. Direct evidence of performance is obtained by audit of critical areas by Westinghouse Quality Control representatives. For example, close attention is given to welder qualification to ASME Code, non-destructive testing records of both supplier and subvendor, and records of heat treating. The Westinghouse auditing of in-process control is aimed at minimizing or preventing defects or errors during fabrication.

In the case of the reactor vessel, Westinghouse has a full time resident inspector at the manufacturing facility.

Each supplier is required to inspect and test his products and to furnish evidence that all requirements of the contract have been met. This evidence includes proof of (1) conformance to specifications for raw materials and those manufacturing and purchased items, and (3) the accuracy of all test equipment used in evaluating product quality. Westinghouse Quality Control Engineers and Representatives review the supplier's quality control procedures and record keeping for conformance to Westinghouse requirements. Critical examination is made of the supplier's quality control records, reports, and inspection certificates gathered during production. Direct evidence of conformance is obtained by product auditing of samples of the suppliers inspection.

In this manner, Westinghouse determines whether or not a supplier has met the specified requirements through the use of objective evidence which is obtained by or from the supplier, and then verified and evaluated by Westinghouse inpsection engineers.

Shipment of Components

The detailed requirements for preparation of equipment for shipment are included in the "Equipment Specifications." These include sealing of all openings, protection of nozzle preparations, the use of desicants if required, etc. Where required, the suppliers submit detailed plans for review and approval.

The reactor vessel supplier provides a cover and seal system to protect all internal surfaces and external stainless steel and machined surfaces from exposure to ambient environments during shipment, storage at the site, and installation. The protective means comprises pressurized inert gas with covers.

For the reactor internals, the lower assembly is shipped on an up-ending skid, shock-mounted to limit loads transmitted to the assembly during shipment. Prior to installation onto the skid, the lower internals are wrapped in a plastic film and sealed. Internal bracing is used inside the assembly. The upper internal assembly is shipped in a shock mounted dual purpose shipping assembly stand in the vertical position. This package is also wrapped and sealed in a plastic film. Both the skid and the stand have a protective metal covering to provide weather protection and long-term storage protection at the site.

All other components have protection, as required, against mechanical or environmental damage during shipment and/or site storage.

Inspection and Installation of Equipment in the Field

For components and equipment supplied by Westinghouse or its subcontractors specifications are prepared not only for design manufacturing, cleanliness requirements and shipment, but also specifications and procedures are provided for on-site storage, erection, quality control and testing.

During component installation, Westinghouse Nuclear Power Services provides a capable and experienced group of specialists to monitor all construction related activities on the Nuclear Steam Supply System, Engineered Safeguards and Critical Structures. This group is staffed to provide coverage in all phases of construction such as welding, mechanical, electrical, systems, instrumentation and control, and startup. The primary responsibility of this staff is to insure proper erection of the Nuclear Steam Supply System, Engineered Safeguards and Critical Structures as outlined by Westinghouse specifications and procedures. This surveillance includes visits to selected shops of suppliers to the Architect Engineer-Constructor to insure that established procedures of inspection and documentation are properly followed. Secondary functions of this staff are to provide technical direction and assistance to the constructor during critical operations and to insure that adequate documentation is being maintained.

The Nuclear Power Service Staff is augmented by the assignment of a resident Quality Assurance Engineer. This engineer is responsible for quality and documentation of all construction activities on the Nuclear Steam Supply System. He will provide additional surveillance of critical operations, follow problems or deficiencies until disposition, aid staff specialists in performance of their duties when necessary, and monitor construction records for completeness.

Non-Conforming Components or Material

All non-conforming components or material, whether discovered at the supplier's facility or at the construction site, are documented, reviewed and disposed of as follows. All details pertinent to the non-conformity are shown on applicable forms. In all cases, the non-conforming component or material is positively identified and separated from acceptable items or items awaiting inspection. All cases of non-conforming components or material are reviewed by Westinghouse design and Quality Control engineers for resolution. Westinghouse's management is kept informed of all cases of major importance with recommendations for proper disposition.

Quality Control Records

A complete set of quality control records is maintained for each component by its manufacturer and/or purchaser and preserved for Westinghouse. These records include certified test reports, letters of compliance, product inspection reports, non-destructive test reports, and reports of non-conforming material. Records of personnel, procedures and equipment qualifications are also maintained as required by Westinghouse's specifications. Records for the components are transmitted periodically to Westinghouse so that all required data will be available for the completed plant.

UNITED ENGINEERS AND CONSTRUCTORS

In the capacity of Architect-Engineer, United Engineers and Constructors has the responsibility for the design of all systems and structures which are not designed by Westinghouse as a part of the Nuclear Steam Supply and associated Engineered Safeguard Systems. In addition, United Engineers specifies and purchases all equipment within their scope of design responsibility. Furthermore, they prepare all construction drawings and specifications and manage all construction work.

In its construction management capacity, UE&C will carry out much of the "first line" on-site quality control, including receipt inspection, identification, on-site storage, and initial inspection and testing during erection. Receipt inspection is carried out to determine whether the particular item is ready for installation including checking for damage, sealing, completeness, and cleanliness. Equipment is labeled or segregated, where appropriate, to assure that proper identification is maintained.

If erection cannot proceed immediately, the small items are placed in a permanent warehouse, and the very large items are stored outdoors, off the ground, and covered. Openings remain sealed until erection except when further inspection or pre-erection work may be required; afterwards, they are resealed until installed.

Dessicants are used and periodically monitored in components which are susceptible to damage by moisture. Heaters installed in equipment for moisture control are kept energized when required. Special precautions are taken to assure that the dessicant has been removed prior to system operation.

The requirements for the highest grade commercial cleanliness which can be obtained practically are observed during construction. Cleanliness specifications have been prepared with full awareness of the constraints imposed by the field conditions. The necessity of removing foreign material which could cause difficulties during operation is stressed. Gross dirt and debris are removed continually from the building area during erection. Equipment is protected as required and kept reasonably clean. Systems that will contain main coolant or are connected to the main coolant system are cleaned and rinse with demineralized water as the final cleaning operation.

As a part of the final cleaning procedures, for the engineered safety features components, a visual inspection of each system is performed following a solvent wash. All systems are flushed using demineralized water during which time temporary screens are installed in the pump suction lines as required.

The equipment and materials are installed in accordance with prescribed erection procedures. These procedures include such items as sequence of installation when required and specifications for welding, paying particular attention to methods that are not standard to the construction industry. Included in the welding specifications are non-destructive tests, such as dye penetrant and radiography.

The work is done by craftsmen skilled in their respective trades. Welders are given the necessary qualification tests as required by the applicable codes.

UE&C maintains an on-site quality control group which is independent of construction management and which monitors the construction activity at the site. This is described in more detail hereinafter.

Organization

The Organization Chart for UE&C relating to Quality Control is shown in Figure 3. The Quality Control Section is staffed by capable and experienced personnel and is divided into two principal operations: Home Office and Field.

The Quality Control Operation is headed by the Manager of Reliability and Quality Assurance in the Home Office who reports directly to the Vice President, Administration.

A Home Office Quality Control Engineer is assigned to the project reporting to the Supervising Engineer in matters relating to project operating control and reporting directly to the Manager of Reliability and Quality Assurance in matters relating to policy and technical control. His responsibility is to establish the quality assurance plan for United Engineers scope of project responsibility, assure adequacy of specifications, audit suppliers facilities and procedures and perform liaison duties between Home Office and Field Quality Control groups.

The Field Quality Control group is supervised by the Field Supervisor - Quality Control who reports to the Construction Manager in matters relating to project operating control and is responsible to the Manager of Reliability and Quality Assurance in matters relating to policy and technical control.

Quality Control Engineers assigned primarily to one basic project phase, i.e., (1) structural, (2) mechanical, (3) electrical and instrumentation and controls, and (4) piping, will report to the Field Supervisor - Quality Control.

Quality Control inspectors are assigned to various phases of the project as required and are supervised by the appropriate Quality Control Engineer.

Records

Records consist of items such as:

1. receipt inspection and storage inspection reports
2. installation procedures
3. non-destructive testing reports and radiographs
4. deficiency reports

The Field Quality Control group is responsible for maintaining records pertinent to and necessary for the quality control of materials and workmanship for UE&C activities on the project. These records are maintained throughout construction and final testing. At the completion of the project these records will be turned over to the Owner. Throughout the course of the project, these records are available for review by the Owner, regulatory agencies and/or their authorized representatives.

The Home Office Quality Control Engineer and the Field Supervisor - Quality Control develops the "filing system."

The filing system is maintained in a numerical order. Numbers are the same as the applicable specification which is also the purchase order number for the equipment or service applicable. As the total system is completed, all records pertinent thereto will be assembled in a "system package" and so maintained for submittal to the Owner.

Piping welding records are by system and a system package contains, but is not necessarily limited to material certifications, shop inspection and test documentation, performance test records, field welding data sheets, non-destructive testing reports, radiographs and isometrics. Again these records are maintained as the work progresses and data assembled into the final package when the system is completed. Where an item is used "across the project" (such as reinforcing steel), separate records are maintained in a "General" category.

Details of Quality Control and Inspection Work

A. UE&C Purchased Items - Vendor Surveillance

Where vendor plant surveillance is necessary, a program is organized and implemented according to the following plan:

1. Review purchase order to determine that quality control requirements are contained or referenced therein.

2. Review vendor's Quality Control Procedures for the subject item(s). Approve these or request changes.

3. Make a Manufacturer's Facilities Survey to assure vendor's capability and to determine his understanding regarding Quality Control documentation and test requirements. Complete the "Manufacturer's Facilities Survey" report.

4. Determine extent of shop inspection necessary to assure compliance with terms and intent of the order. Complete "Work Sheet for Testing Requirements and Reports".

5. During vendor inspections, in addition to the specific purpose of the inspection (e.g., witnessing shop tests or reviewing radiographs), the Quality Control representative shall:

   a. Audit applicable mill test reports and certifications for full compliance to specification requirements.

   b. Audit welding procedures and special techniques.

   c. Review individual welder qualifications for conformance with applicable ASME, AWS or UE&C codes.

   d. Review techniques and procedures for required inspection methods and tests.

   e. Determine that vendor quality control procedures are being carried out as approved.

6. Prior to any vendor surveillance, the Field Supervisor Quality Control or the Home Office Quality Control Engineer prepares an "Inspection Check List" for the Quality Control representative making the visit. This list contains specific details to be checked during the visit, including dimensional tolerances, test procedures, test pressures, applicable codes, etc.

   The Quality Control Engineer ensures that the Quality Control representative reviews the necessary codes, specifications, and drawings pertaining to the inspection.

B. Scheduling

Scheduling of quality control work is done with complete coordination with the construction, scheduling, engineering and purchasing groups connected with the project. Use of the CPM print-outs is made in this regard for long range scheduling, using separate sortings where helpful.

Copies of shipping schedule distribution and piping progress sheets for items are reviewed thoroughly as updated so that quality control schedules are kept on a current and correct basis.

C. Site Quality Control - Structural

1. Concrete Materials Approvals and Mix Design - Materials entering into production of structural concrete for the project are sampled and tested in full accordance with project specification requirements and applicable ASTM and ACI standards prior to any placement or concrete in the permanent structures. Using approved materials, mix proportions are determined in accordance with project requirements and ACI 613.

2. Mill test certificates are required and reviewed for cement used on the project. Separate storage facilities are maintained at the batch plant so that no mixing of cement types or brands takes place.

3. Concrete aggregates are periodically sampled and tested. The batch plant is inspected to ascertain that only previously approved aggregates are used.

4. The concrete batch plant is inspected for conformance to ASTM C-94. During concreting operations, the batch plant is checked to determine that approved mixes are being batched and all batching is being done in accordance with project specifications.

Items of non-conformance noted at the batch plant are noted by the Quality Control batch plant inspector and the pour discontinued until the discrepancy is corrected.

5. All concreting operations are inspected in the field including forms, reinforcing, placing, curing and stripping. Slump tests, air content checks, and concrete cylinders are made as required.

One set of six (6) concrete cylinders for compression testing are made, representative of each class of concrete per 100 cubic yards (or portion thereof) placed per day. Three cylinders are tested at seven (7) days and three at twenty-eight (28) days.

A slump test is made on each truck load of concrete delivered. <u>These tests are based on good quality control practice and past experience.</u>

The results of the compression testing (28-day results) are evaluated in accordance with ACI 318-63, Section 504.

Individual loads of plastic concrete which do not conform to the specification requirements are rejected by the Quality Control inspector responsible for field concrete inspection. Rejected loads are removed from the site.

When there is a question as to the quality of the concrete in the structure because of strength test failures due to laboratory-cured specimens falling below the specified requirements at 28 days, alternate strength tests in accordance with ASTM C42 or actual load tests in accordance with Building Code requirements ACI-318, may be allowed. Acceptance of the quality of the concrete by these alternate tests are based on engineering evaluation and fully documented. Concrete which does not meet the specification requirements or which cannot be tolerated after the engineering evaluation is removed.

6. All concrete materials testing, design mixes, batch plant inspection, site inspection, and cylinder testing are performed by an independent testing laboratory hired by the contractor. The Field Quality Control Organization oversees these areas to assure compliance with specifications.

7. Reinforcing Steel - Mill test reports are required and reviewed covering each heat of material supplied. Material is inspected upon receipt at the site for identification marks and condition. Fabricated high strength bars are checked for cracking at bends when received at the site.

   Reinforcing bars received at the project site are checked for grade, size, and heat number. Metal tags on individual bundles and mill markings indicate these identifications and these are checked against results of tests certified on the appropriate mill test reports and results of user testing where applicable.

   Only one type of reinforcing is specified and ordered for use on this site.

There is no specification requirement for user testing of reinforcing steel on the project. However, as an added quality control measure, one sample per size per heat received is selected at random for physical testing for the #18S, #14S, and #11 bars (those sizes used in the Containment Building and subject to Cadweld splicing). Random sampling continues up to six samples per size.

Bars are received in bundl egregated by size and heat number. Each bundle is individually tagged with metal tags bearing the heat number. Certified mill test reports covering the size and heat are checked against material received prior to use in the structure.

Heats/sizes represented by user samples which fail to meet specification requirements are held for resampling and retesting.

Rejected heats are removed from the project site.

Heats are rejected if they fail to meet the minimum specified yield, ultimate strength, or elongation requirements on the following basis:

If the original sample fails to meet requirements, the sample size is doubled and tested. If any one of the second sample fails to meet requirements, the sample is again doubled and tested. If any one of the third sample fails to meet requirements, the whole heat is rejected.

The above tests are based on good quality control practice and past experience.

All reinforcing bars are high-strength ASTM A432 bars and are not strength-welded or tack-welded except for Cadweld splicing as noted below.

8. Cadweld Splices - The Cadweld splices shall be capable of developing in tension at least 125 per cent of the specified yield strength of the reinforcing bar, in accordance with the requirements of ACI 318-63, Section 805-d.

Individual splices which do not meet 125% of yield are rejected.

The strength requirements for in-place structures by grade of re-bar, are as follows:

For high strength bar to ASTM A432, minimum yield strength 60,000 psi, minimum ultimate strength 90,000 psi.

The mean value of the ultimate strength of splices made during any time period shall be equal (as a minimum) to 75,000 psi, plus the standard deviation in strength from the mean ultimate strength. In addition, the mean value of the ultimate strength and the standard deviation shall show, by statistical analysis, that at least 99.0% of all of the splices will have an ultimate strength of 60,000 psi or greater.

Mill test reports covering sleeves and cartridges are required. These items in the storage areas are checked to assure that no deterioration has taken place.

Each splice is visually inspected by the following procedure. Any splice which, in the judgement of the inspector, does not pass visual inspection is cut out and replaced.

a. Bar ends shall be approximately square. They may be torch-cut, sawed or sheared. The cut faces of both re-bar, when inserted into the sleeve, shall be entirely within the specified limits for the size of bar.

b. Bar ends shall be cleaned of dirt, oil, moisture, concrete, or heavy rust, to a degree of cleanliness as represented by heating the end of the bar uniformly to a surface temperature of 200°F to 300°F, power wire brushing to bare metal, reheating to the same temperature range, and hand wire brushing to remove any resulting dust and/or loose material.

c. The re-bars shall be assembled with their sleeve immediately after cleaning and properly aligned.

d. Preheating is not generally required; however, if the air temperature is below 40°F, and/or the humidity is above 80%, the bar ends and sleeve shall be preheated to 100°F in order to remove moisture.

e. If it is necessary to remove a portion of the longitudinal rib on the re-bar in order to fit into the splicing sleeve, the metal shall be removed by grinding only. In no case shall the entire rib be removed nor shall there be any under-cutting of the rib into the stock material of the re-bar.

f. Properly made splices will have filler metal visible at both ends of the sleeve and at the tap hole in the center of the sleeve.

g. Filler metal will not flow to the very edge of the sleeve due to the gasket action of the asbestos wicking used to seal in the molten filler metal. A recess less than 1/2" will not be cause for rejection.

h. As a result of the Cadweld process, a shrinkage bubble may be visible at the tap hole where the molten metal is introduced and shrinkage fissures and pinholes may be visible at the top of a vertical splice. These casting flaws do not adversely effect the physical performance of the splice and, therefore, do not constitute cause for rejection.

Bars or splices which do not meet the requirements above are rejected and removed from the structure.

Joining of structural plate and rebar in the dome is by use of cadweld splices which are welded to the plate. These joints are tested in compliance with normal cadwelding procedure. All cadweld splices are 100 percent visually inspected. The joint between the structural plate and the cadweld sleeve is made in accordance with AWS welding procedure D1.0.

Before any Cadweld splicing crew is assigned to production work, they must demonstrate their ability to produce splices meeting the specification requirements.

Each crew is qualified using approved materials and procedures by making five splices of each type which are tested to destruction.

Each crew is qualified to do specific work only to the extent of having performed satisfactory qualification splices for each category of work; i.e., when qualified for horizontal bar, straight size splices, the crew can make only this type of splice. Any crew having prior qualification for the four types of splices (horizontal-straight, horizontal-reducing, vertical-straight, vertical-reducing) is deemed capable of making any type of splice required by the project.

Each crew is assigned an identification number and this number is not re-issued during the life of the project.

For purposes of this work a crew is defined as an operator who has been qualified in accordance with the above procedure and a competent helper.

For each crew starting work, the following splices are cut out of the work on a random basis and tension tested to destruction. The yield strength (if possible), the ultimate strength, and type of failure is recorded.

a. For the first fifty splices made, remove and test one out of each five made.

b. For the second fifty splices made, remove and test one out of each ten made.

c. For the next one hundred splices made, remove and test one out of each twenty made.

d. After the first two hundred splices are made, remove and test one out of each one hundred made.

e. Should the splice not meet the strength requirements, then the splice made by that same crew immediately preceding or following the sub-standard splice is cut out and likewise tested. If this second test splice meets the requirements, then these two joints may be replaced. These replacement joints may be included in the next grouping from which a random sample may be selected.

   If this second test splice does not meet the requirements, all work by this crew is stopped. The five previous splices by the crew are cut out and tested. Should any of these five splices not meet the requirements, the crew is required to requalify and the Structural Engineer must be able to justify statistically that this crew is capable of doing satisfactory work before executing any further splices. Should the five splices meet the requirements, the reinforcing is re-spliced. Any one of the re-splices may be selected for a test.

It has been noted that axial misalignment does not affect the integrity of strength of the splice. However, centering of the bars within the sleeve does not affect the strength of the splice and in this regard each bar is center-punched 12" from the end and dimensions checked to assure that ends to be spliced are gapped 1/6" to 1/4" in the sleeve and centered to within 1/4".

All operator qualification records are kept. A plot plan of all splices is maintained showing splice location and number, operator number, position, site, date made and date inspected. A log of all test results is maintained and results are charted. The Structural Engineer is informed of all test results, and results outside the specification are "red flagged.

9. Structural Steel - Mill test reports are required and reviewed. Material is inspected when received at the site for proper identification and condition.

Where shop fabrication is required, a vendor surveillance program is set up as described previously.

Erection of structural steel is subject to inspection as required. Connections are particularly checked for soundness and tightness.

10. Field Welding of Structural Steel - Welding procedure test and operator certifications are checked for conformance to applicable ASME, AWS or UEAC specifications. Only certified welders are allowed to perform structural welding on the permanent portion of the structure. Previous qualification of welders may be accepted as certification for the project if allowed by the governing code or specification.

Field welds are inspected and where required, non-destructive tests are performed, witnessed or reviewed by a representative of the Quality Control group.

Radiographs are reviewed and a report of the tests is maintained in the Quality Control file.

11. Containment Liner - Particular attention is placed on the fabrication of the vapor containment liner. Mill test certificates for liner plates covering certified Charpy impact properties, ductility, weldability, tensile and chemical properties are required and reviewed for all liner plate. The certificates become a part of the permanent quality control file.

As liner plates are received, the edges of all plates are checked at four points with vernier calipers to measure liner plate thickness.

Welding procedures and welder qualifications are checked and certifications become a part of the permanent quality control file.

Weldings and non-destructive testing are audited and non-destructive test reports are reviewed and become a part of the permanent quality control file. These tests include a leak testing of the pressure channels.

Final assurance as to the integrity of the seam welds under design accident conditions is provided by the containment integrated leak test.

Dimensional checks are made to assure that the liner in completed structure meets the requirements specified in Section 5.1.2.1.

These checks are based on measuring each 9 foot ring in 10° arcs from a reference point at the knuckle.

Any dimensions found outside of the limits are cause for rejection and any repair procedures are approved by the Structural Engineer and subject to necessary quality control.

Procedures for the final containment pressurization and test are reviewed and this test is witnessed. Records pertinent to the test become a part of the permanent quality control file.

D. Site Quality Control - Mechanical Equipment

1. Receiving Inspection - Items are checked when received at the site for shipping damage, sealing and generally to determine if the item is ready for installation.

2. The Field Supervisor - Quality Control prepares a check list of inspection requirements, limiting factors, and acceptance criteria. These inspections must be completed and results satisfactory prior to acceptance of an item into the structure.

3. Any non-destructive tests (including pressure and leak rate) required by the specifications are performed, witnessed, or reviewed by a member of the Quality Control group. Radiographs are reviewed and a report of the tests is maintained in the Quality Control file.

4. Field welding on mechanical equipment is subject to the same criteria as noted under "Field Welding of Structural Steel." Testing requirements of these welds is considerably more extensive for the mechanical portion than for the structural portion.

E. Site Quality Control - Mechanical - Piping

1. Receiving Inspection - Piping is checked when received at the site for damage, sealing, and cleanliness (where required).

2. Welders and welding procedures are certified in accordance with ASME Code requirements. For those welders or procedures requiring field certifications, tests for certification are witnessed by a representative of the Quality Control Group. Qualification records beome a part of the Quality Control file.

3. Field welds in the critical systems and others where required are numbered and these numbers will be the permanent identification of the joint. Numbering of these joints is the responsibility of the Welding Supervisor and he provides the Quality Control Engineer with a list of these numbers and the location of the joints to which they apply.

4. Non-destructive tests (including pressure and leak tests) are performed, witnessed, or reviewed by a member of the Quality Control group. Radiographs are reviewed and the "Piping System Inspection Record" completed. This record and all radiographs become a part of the Quality Control file when the system is completed.

5. The cleanliness of piping systems is carefully checked in full accordance with project specifications.

6. During the course of field welding on the critical piping systems, a "Record and Schedule of Field Welds" is maintained and available for checking progress. It is the responsibility of the Welding Supervisor to maintain this record.

F. Site Quality Control - Electrical, Instrumentation and Controls

1. Receiving Inspection - Items are inspected when received at the site for damage to the crating or equipment and for proper identification.

2. The inspection of installation is primarily a check upon the workmanship in mounting and wiring the equipment to code and specification requirements.

3. Pre-operational checks - After installation, components are tested for their proper electrical and mechanical functions without energizing the equipment. These tests are conducted to applicable USAS, IEEE, and NEMA standards and the "Start-up Check List for Electrical Equipment" as prepared by the Electrical Engineer and approved. On critical or other major items, the tests are witnessed by a representative of the Quality Control group.

4. Operational Checks - After successful pre-operational tests, equipment is further checked for proper mechanical and/or electrical functions with the equipment energized. Tests are conducted in accordance with applicable USAS, IEEE, and NEMA standards and the "Start-up Check List for Electrical Equipment" as prepared by the Electrical Engineer and approved. On critical or other major items, the tests are witnessed by a representative of the Quality Control Group.

PARTICIPATING ORGANIZATIONS IN QUALITY ASSURANCE

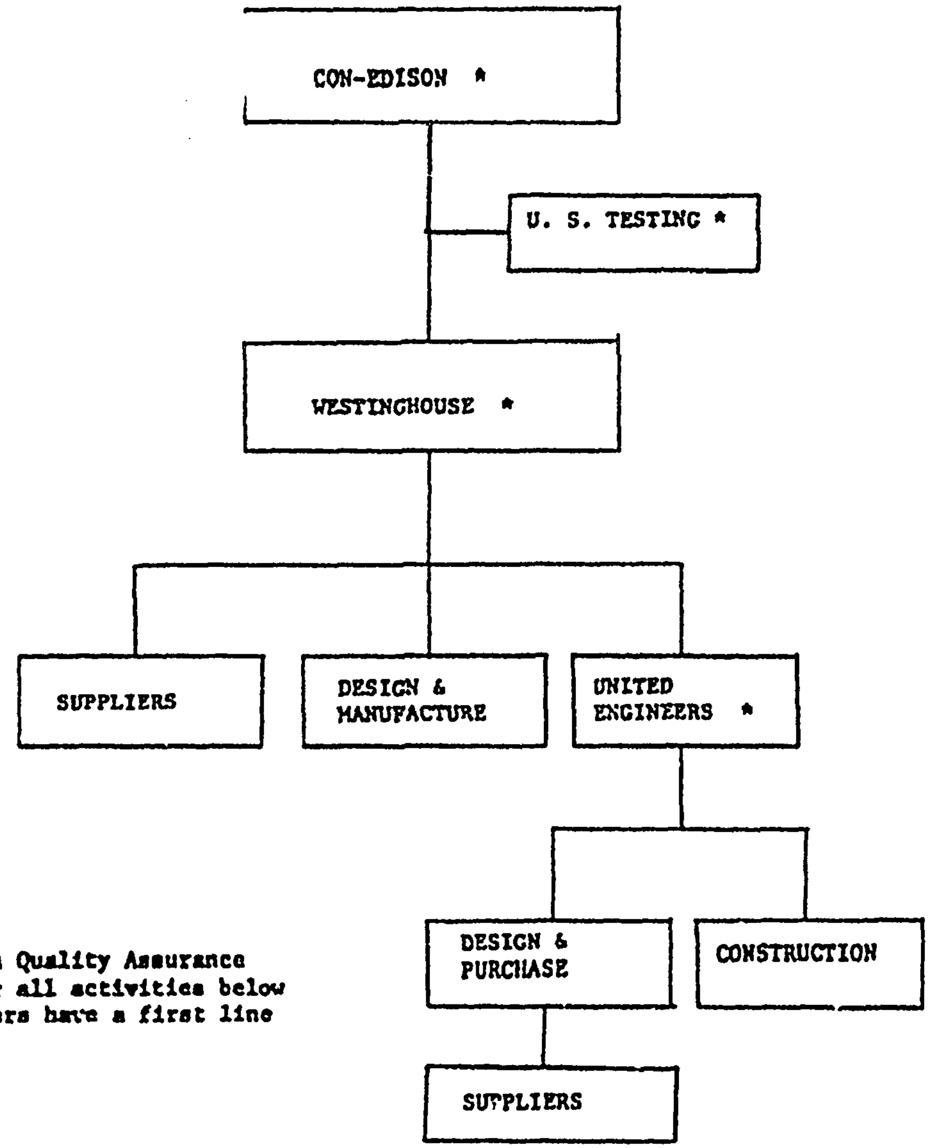
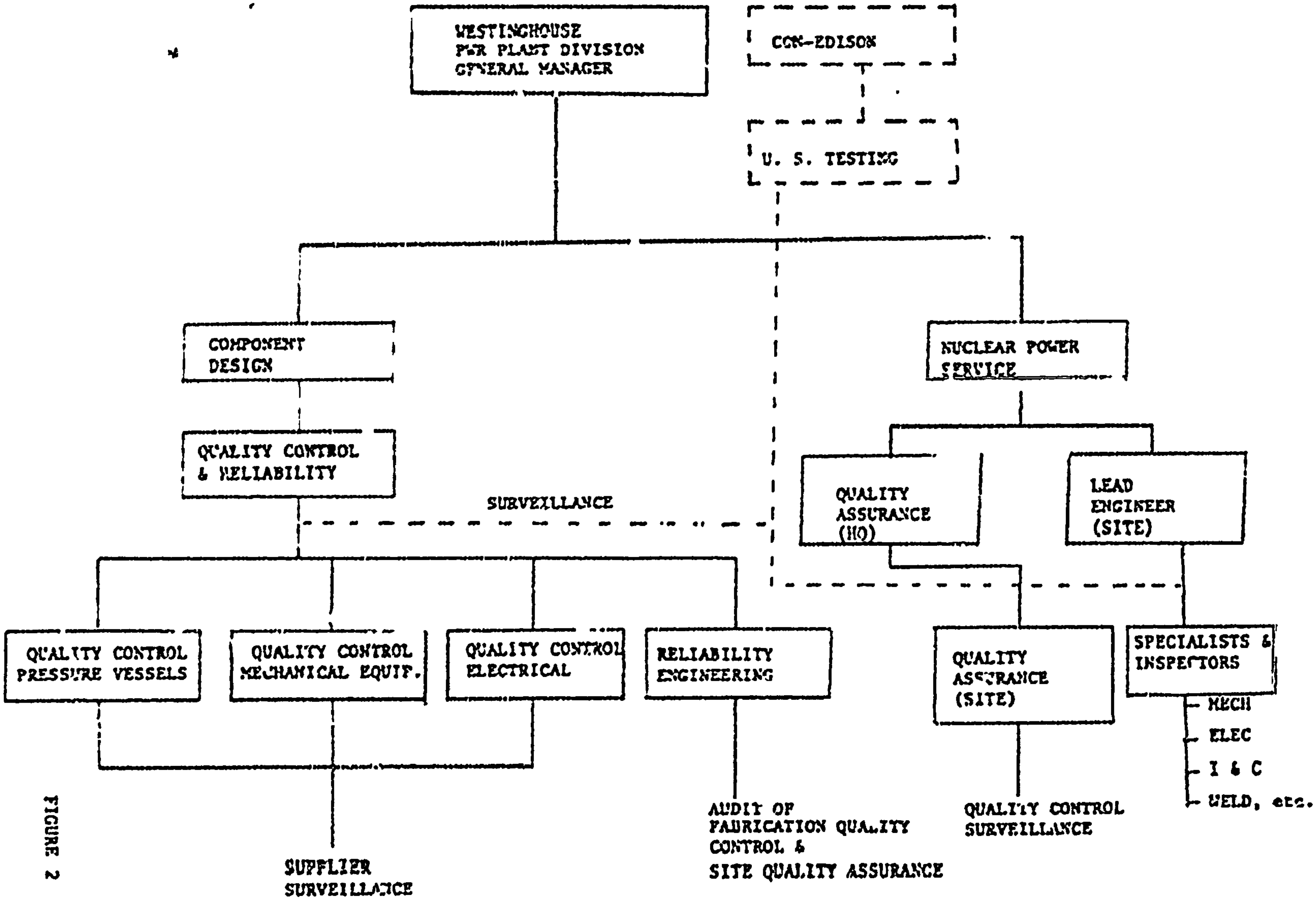


FIGURE 1

* Indicates performance of a Quality Assurance surveillance function over all activities below this level of chart. Others have a first line responsibility.



WESTINGHOUSE
PWR PLANT DIVISION
GENERAL MANAGER
CON-EDISON
U. S. TESTING
COMPONENT
DESIGN
NUCLEAR POWER
SERVICE
QUALITY CONTROL
& RELIABILITY
SURVEILLANCE
QUALITY
ASSURANCE
(HQ)
LEAD
ENGINEER
(SITE)
QUALITY CONTROL
PRESSURE VESSELS
QUALITY CONTROL
MECHANICAL EQUIP.
QUALITY CONTROL
ELECTRICAL
RELIABILITY
ENGINEERING
QUALITY
ASSURANCE
(SITE)
SPECIALISTS &
INSPECTORS
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WELD, etc.
AUDIT OF
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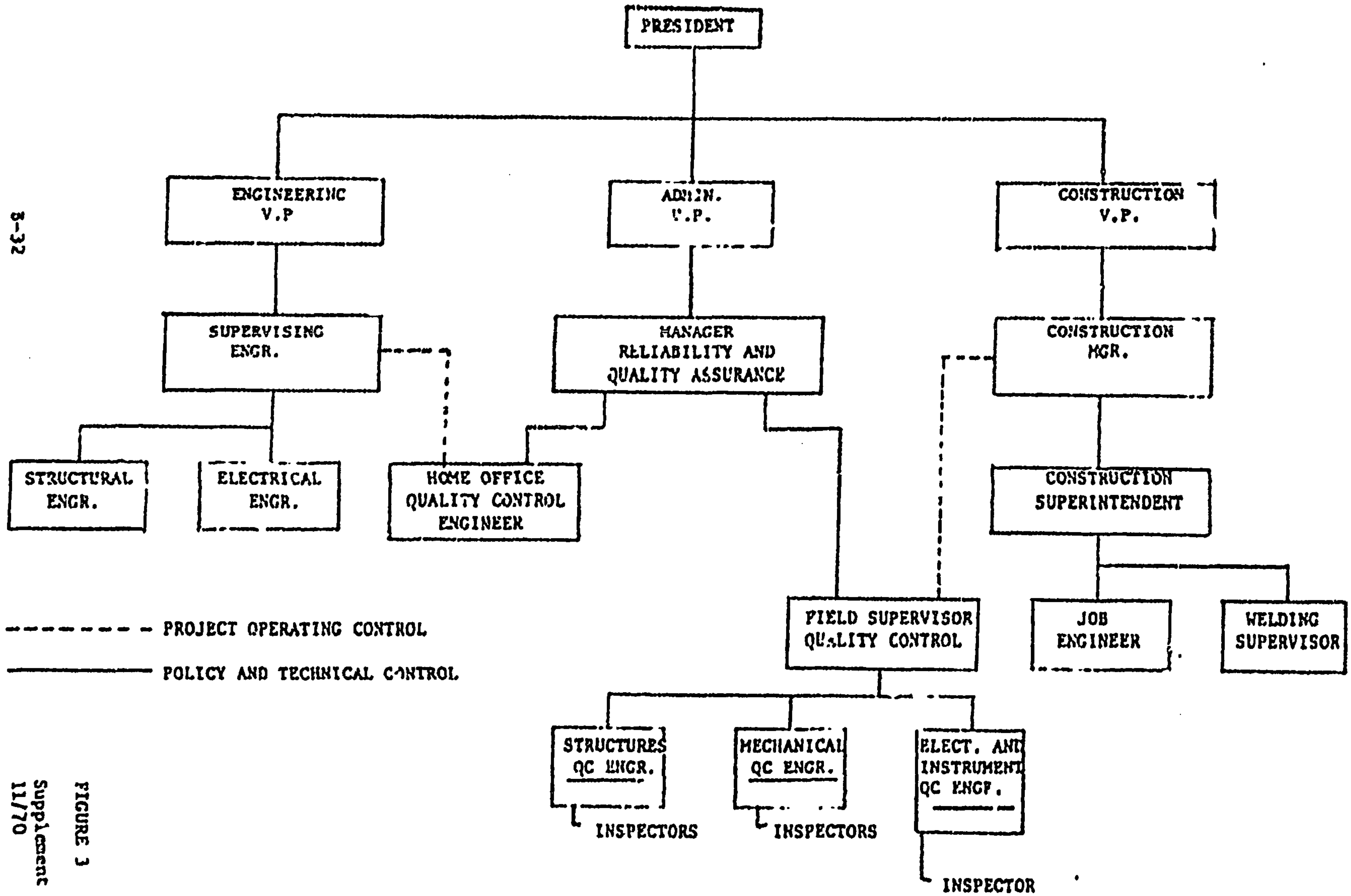
CONTROL &
SITE QUALITY ASSURANCE
QUALITY CONTROL
SURVEILLANCE
SUPPLIER
SURVEILLANCE

FIGURE 2



UNITED ENGINEERS AND CONSTRUCTORS
PROJECT ORGANIZATION
QUALITY CONTROL RELATIONSHIPS
1968

FIGURE 3

Appendix C

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

CONTAINMENT LINER STRESS ANALYSIS REPORT

July, 1968

TABLE OF CONTENTS

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LIST OF FIGURES

I. INTRODUCTION

The object of this report is to illustrate the design adequacy of the containment liner for the Indian Point Nuclear Generating Unit No. 2. To this end, this documentary report describes the design of the liner, as well as the fabrication and erection procedures, to demonstrate fulfillment of the design criteria.

The only element of the containment system which is being evaluated in this report is the steel liner. However, since the liner and surrounding concrete shell form an integral unit, by virtue of the attachment of the liner to the concrete, it is not possible to completely divide these two elements. Therefore, in enumerating design criteria and methods of stress analysis employed both components are considered as a unit where applicable. This, in turn, helps to clarify their relationship and show the effect of the concrete on the steel liner.

The following sections of this report enumerate the basic criteria that were used, the analyses that was developed to satisfy these criteria, the various loading combinations under normal and postulated accident conditions (including seismic effects), and the fabrication, erection and testing procedures that were employed to ultimately construct the liner at the site.

II. CRITERIA USED FOR DESIGN

1.0 PURPOSE OF LINER

The purpose of the steel liner, which is attached to the inside face of the concrete shell, is to ensure a high degree of leak tightness in the event of an accident resulting in the loss of reactor coolant and potential release of radioactive material. The liner is attached to the concrete by means of stud anchors so that it forms an integral part of the entire composite structure under all loadings.

2.0 DESIGN LOAD CRITERIA

The loads considered in the design of the containment structure, which can create stresses within the component parts such as the liner, are enumerated below.

2.1 DEAD LOAD

Dead load includes the weight of the concrete walls, dome, base and internals, steel liner and insulation.

2.2 LIVE LOADS

The live loads consist of snow on the dome and major components or equipment on the base. Snow loads are uniformly applied to the top surface of the dome at an estimated value of 20 pounds per square foot of horizontal projection.

2.3 INTERNAL PRESSURE TRANSIENT

The internal pressure transient used for the containment design indicated on the pressure-temperature transient curve is shown on Figure II-2.3-A.

The incident pressure load is based on the reference incident pressure of 47 psig as defined in the Final Safety Analysis Report, Section 5. This reference incident pressure contains a 7.5 psig margin above the pressure peak calculated for the double-ended reactor coolant pipe break with engineering safeguards operating at reduced effectiveness on emergency diesel power.

2.4 THERMAL EXPANSION LOADS

Thermal expansion loads, due to an internal temperature increase caused by a loss-of-coolant accident, have been considered. This temperature increase versus time is shown on Figure II-2.3-A which indicates that the maximum temperature at the uninsulated section of the liner under accident conditions is 247°F. For loading conditions equivalent to 125% and 150% of design pressure loading conditions, as shown on Figure II-2.4-A and II-2.4-B and as subsequently described in Section II-3.0, the corresponding liner temperatures will be 285°F and 306°F, respectively. The increased temperature design conditions defined above are calculated from the loss-of-coolant accident under the assumption that the maximum operating temperature at the start of the accident is 120°F. The minimum external ambient design temperature, averaged over a 24 hour period is -5°F.

2.5 SEISMIC LOADS

The design of the containment which is a Class I structure utilizes the "response spectrum" approach in the analysis of the dynamic loads imparted by earthquake. The seismic design is based on the acceleration response spectrum curves, Figures III-2.4.7.2B and C developed by G. Housner. Structural seismic accelerations have been computed as outlined in references 2 and 3.

The ground acceleration for the design earthquake has been determined to be 0.1g applied horizontally and 0.05g applied vertically. These values have been resolved as conservative numbers based upon recommendations from Dr. Lynch, Director of Seismic Observatory, Fordham University.

Additionally, an even greater hypothetical ground acceleration of 0.15g horizontal and 0.10g vertical has been used to design for the no-loss-of-function, as defined in Section II-4.0.

2.6 WIND LOADS

The American Standards Association "American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures" (A 58.1-1955) designates the site as being in a 25 psf zone at ground level. In this code, for height zones between 100 and 499 feet, the recommended wind pressure on a flat surface is 40 psf. Correcting for the shape of containment by using a shape factor of 0.60, the recommended pressure becomes 24 psf. However, the State Building and Construction Code for the State of New York stipulates a wind pressure up to 30 psf on a flat surface for heights up to 300 feet. Therefore, in order to be conservative for design, a 30 psf basic wind load is used for ground level up.

2.7 TEST LOADS

An internal pressure of 54 psig (115 per cent of design pressure) will be applied to test the structural integrity of the vessel. This pressure will be held for a minimum of one hour.

2.8 CONSTRUCTION LOADS

A load equivalent to the weight of wet concrete, placed in sections during construction of the concrete dome, is used on the design of the stiffened dome liner plate.

A construction live load of 50 psf is used on the design of the concrete dome but not concurrently with the snow load.

2.9 PENETRATION LOADS

The effects of lateral loads due to thermal expansion of pipes at the liner penetrations have been investigated and considered in the design of the penetration sleeves and reinforcements discussed later.

3.0 LOADING COMBINATION CRITERIA

The design of the containment system, including the liner, is based upon the limiting load factor concept as defined in Ref. 4. These load factors are the values by which loads will be multiplied under various combined loading conditions to provide assurance of the overall elastic behavior of the structure. The use of the load factor concept in the design and analysis of containment permits making a rational evaluation of the various individual types of loadings so as to provide a conservative method of combining such loads while assuring an adequate safety margin for the structure.

This approach, with its use of proper coefficients, permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. In the case of the containment structure maximum emphasis is placed on accident and earthquake or wind loads. However, the structure is also designed to accommodate the normal dead and live loads (including construction loads) as required by the ACI Building Code.

3.1 LOADING COMBINATION EQUATIONS

The load factor philosophy employed in Ref. 4 is used to define three loading combinations, as represented by the following equations; these equations define the loading requirement of any structural element in the containment structure.

a. $C = 1.0D \pm 0.05D + 1.5\,P + 1.0\,[T + TL]$

b. $C = 1.0D \pm 0.05D + 1.25\,P + 1.0\,[T' + TL'] + 1.25\,E$

c. $C = 1.0D \pm 0.05D + 1.0\,P + 1.0\,[T'' + TL''] + 1.0\,E'$

Symbols used in these equations are defined as follows:

C: Required load capacity of section.

D: Dead load of structure.

P: Incident pressure load

T: Load due to maximum temperature gradient through the concrete shell and mat based upon temperature associated with 1.5 times incident pressure.

TL: Load exerted by the exposed liner based upon temperature associated with 1.5 times incident pressure.

T': Load due to maximum temperature gradient through the concrete shell and mat based upon temperatures associated with 1.25 times incident pressure.

TL': Load exerted by the exposed liner based upon temperatures associated with 1.25 times incident pressure.

E: Load due to acceleration from the design earthquake and includes the combination of horizontal and vertical components of acceleration.

T": Temperature gradient load associated with design basis accident.

TL": Load due to exposed liner at temperature associated with design basis accident.

E': Seismic load associated with the hypothetical earthquake.

The magnitude of the various loads described above vary at different elevations of the containment liner. Tables II-3.1-A, B, C, D, E and F show the maximum composite loading values considered in the liner design for loading combination equations (a), (b) and (c), respectively, for eight positions in the liner which represent typical or limiting points for loading and stress calculation as shown in Figure II-3.1-A.

With respect to Tables II-3.1-A to F the dead weight, the particular temperature and the accident pressure which when combined give the maximum structural loads for the factored load equation considered are shown in columns 3, 4 and 5 respectively. The structural loads resulting from these inputs are shown in columns 6-10. The limiting points of interest where liner loads and stresses are evaluated are shown in Figure II-3.1-A.

The containment design does not required the liner to carry any structural loads. Therefore, in evaluating the adequacy of the liner, it is necessary only to investigate those loads imposed on the liner as a result of compatible behavior with the effect of the reinforced concrete containment shell.

3.2 ANALYTICAL ASSUMPTIONS

In using the above equations to define the design loading conditions, the following assumptions have been made:

1. The effects of wind pressure on the structure were investigated and compared to those resulting from earthquake in loading condition (b). The results of the analysis show that the seismic loads govern, being in all cases at least 20 times greater than wind load effects, therefore, seismic loads are used in the final analysis.

2. The temperature gradient, T, through the complete containment wall prior to an accident is essentially linear and is a function of the internal operating temperature and the average external ambient temperature. Accident conditions mainly affect the uninsulated liner since it is more closely exposed to the internal temperature than the bulk of the concrete. By the time the temperature of the concrete adjacent to the liner begins to rise significantly, the internal pressure and temperature in the containment have decayed somewhat from their maximum values. The resultant maximum temperature gradient is shown on Figure II-3.2-A.

The temperature effects described above induce stresses in the structure which are internal in nature; tension on the outside surface and compression on the inside surface of the shell (Figure II-3.2-B.2); the resultant force is zero. Loading combinations (Figure II-3.2-B.3) concurrent with these temperature effects may cause local stresses in the outside horizontal and vertical rows of bars to reach yield (Figure II-3.2-B.4); however, as local yielding is reached, any further load is transferred to the unyielding elements (Figure II-3.2-B.5).

At the full yield condition, the magnitude of the final load resisted across a horizontal and vertical section remains identical to that which would be carried if the temperature effects were not considered. Thus, the overall carrying capacity of the structure and the factor of safety of the structural elements are not affected by the temperature gradient.

The ability to satisfy load condition (a) indicates that the containment has the capacity to withstand loadings at least 50 per cent greater than those calculated for the postulated loss-of-coolant accident alone.

The ability to satisfy load condition (b) indicates that the containment has the capacity to withstand loadings at least 25 per cent greater than those calculated for the postulated loss-of-coolant accident with a coincident design earthquake.

The ability to satisfy load condition (c) indicates that containment will accommodate seismic loads of at least equal to those corresponding to the response of 0.15g horizontal and 0.10 vertical ground accelerations occurring simultaneously.

4.0 STRESS CRITERIA

4.1 General Philosophy

The design stress criteria for the liner is based on the philosophy that no gross deformation beyond the elastic limit occurs for all loading conditions defined previously.

In this reinforced concrete structure, the design limits for tension members (i.e., the capacity required for the design loads) are based upon ASTM specified minimum allowable stresses for reinforcing steel.

This reinforcement has also been designed so that it is not subject to average stresses beyond the yield point across any section due to the factored loads.

4.2 Additional Safety Provisions Regarding Stresses

As an additional safety factor, the allowable stress under any given load is reduced from the values referred to above by a capacity reduction factor, denoted as "Ø". This reduction provides for the possibility that small adverse variations in material strength, workmanship, dimensions, control and degree of supervision, while individually within required tolerances and the limits of good practice, occasionally may combine to result in undercapacity.

The values of "Ø" (Reference 4, page 66) used for reinforced concrete are as follows:

Tension Members	0.95
Flexure	0.90
Diagonal Tension, Bond and Anchorage	0.85

For principle compression and tension, the liner stresses are maintained below 0.95 specified minimum yield at normal operating temperature, i.e., 0.95 x 32,000 = 30,400 psi. For shear, the liner stresses are maintained below 0.6 specified minimum yield at normal temperature. (The actual shear stresses are well below this limit). The actual proportioning of seismic shear between the liner and the concrete shell is dependent upon the relative stiffness of the two elements. Conservatively assuming only the relative stiffness of the steel reinforcement in the concrete shell versus the stiffness of the liner approximately 30 percent of the seismic shear could be transmitted into the liner.

The limiting case governing the contribution of shear stress to direct stress in the liner to determine the maximum principal compressive stress shows the liner capable of carrying 40 percent of the seismic shear before principal yield compression would be reached. The liner plate material is ASTM A-442, Grade 60.

In order to ensure elastic stability of the liner, anchorage to the concrete is provided (Reference 5). Detailed methods of anchorage are discussed in Section III-1.0.

For the Structural Proof Test (Described in II-2.7) membrane stresses are maintained within elastic limits.

5.0 MISSILE PROTECTION

High pressure reactor coolant system equipment which could be the source of missiles is suitably shielded from impacting on the liner either by the concrete shield wall enclosing the reactor coolant loops and pressurizer or by the concrete operating floor to block any passage of missiles to the containment walls. A structure is provided for the control rod drive mechanism to block any missiles generated from fracture of the mechanisms.

LOAD TABLE

POSITION (1)	COORDINATES (FT) HOR. (2)	VERT.	DEAD LOAD (K/FT) (3)	TEMP. WHEN LINER STRESS IS A MAXIMUM (°F) (4)	PRESSURE WHEN LINER STRESS IS A MAXIMUM (PSI) (5)	PRESSURE LOAD (K/FT) (6)	SEISMIC SHEAR LOAD (K/IN) (7)	SEISMIC OVERTURNING MOMENT (K·IN) (8)	WIND SHEAR LOAD (K/IN) (9)	WIND MOMENT LOAD (K-IN) (10)
1	53.0	43.0	0	---	---	0	0	0	0	0
2	67.0	43.0	0	---	70.5	157	0	0	0	0
3	67.5	45.7	127	120	70.5	354	0	0	0	0
4	67.5	64.0	126	278	35	176	0	0	0	0
5	67.5	117.0	94	252	16	80.5	0	0	0	0
6	67.5	191.0(-)	37	252	16	80.5	0	0	0	0
7	67.5	191.0(+)	37	264	23	116	0	0	0	0
8	57.8	225.8	24	252	16	80	0	0	0	0

$$C = 1.0D \pm 0.05D + 1.5P + 1.0\,(T + TL)$$

VERTICAL

TABLE II-3.1-A

POSITION (1)	COORDINATES (FT) HOR. VERT. (2)	DEAD LOAD (K/FT) (3)	TEMP. WHEN LINER STRESS IS A MAXIMUM (°F) (4)	PRESSURE WHEN LINER STRESS IS A MAXIMUM (PSI) (5)	PRESSURE LOAD (K/FT) (6)	SEISMIC SHEAR LOAD (K/IN) (7)	SEISMIC OVERTURNING MOMENT (K·IN) (8)	WIND SHEAR LOAD (K/IN) (9)	WIND MOMENT LOAD (K-IN) (10)
1	53.0 43.0	0	---	---	0	0	0	0	0
2	67.0 43.0	0	---	---	0	0	0	0	0
3	67.5 45.7	0	120	70.5	10.6	0	0	0	0
4	67.5 64.0	0	264	23	231	0	0	0	0
5	67.5 117.0	0	252	16	161	0	0	0	0
6	67.5 191.0(-)	35	252	16	161	0	0	0	0
7	67.5 191.0(+)	35	264	23	116	0	0	0	0
8	57.8 225.8	6	252	16	80.5	0	0	0	0

$$C = 1.0D \pm 0.05D + 1.5P + 1.0\ (T + TL)$$

HORIZONTAL

TABLE II-3.1-B

LOAD TABLE

POSITION (1)	COORDINATES (FT) HOR. VERT. (2)	DEAD LOAD (K/FT) (3)	TEMP. WHEN LINER STRESS IS A MAXIMUM (°F) (4)	PRESSURE WHEN LINER STRESS IS A MAXIMUM (PSI) (5)	PRESSURE LOAD (K/FT) (6)	SEISMIC SHEAR LOAD (K/IN) (7)	SEISMIC OVERTURNING MOMENT (K·IN) (8)	WIND SHEAR LOAD (K/IN) (9)	WIND MOMENT LOAD (K-IN) (10)
1	53.0 43.0	0	---	---	0	0	0	0	0
2	67.0 43.0	0	---	59	132	0	0	0	0
3	67.5 45.7	127	120	59	298	12.9	27.9×10^6	0.32	10.8×10^5
4	67.5 64.0	126	269	42.0	212	12.[illegible]	24.2×10^6	0.30	9.4×10^5
5	67.5 117.0	94	252	27.8	140	10.6	14.1×10^6	0.21	4.86×10^5
6	67.5 191.0(-)	37	252	27.8	140	5.24	3.61×10^6	0.09	1.05×10^5
7	67.5 191.0(+)	37	237	18.2	92	5.24	3.61×10^6	0.09	1.05×10^5
8	57.8 225.8	24	237	18.2	91	3.14	1.01×10^6	0.07	$.37 \times 10^5$

$$C = 1.0D \pm 0.05D + 1.25P + 1.0\ (T' + TL') + 1.25E$$

VERTICAL

TABLE II-3.1-C

LOAD TABLE

POSITION (1)	COORDINATES (FT) HOR. VERT. (2)	DEAD LOAD (K/FT) (3)	TEMP. WHEN LINER STRESS IS A MAXIMUM (°F) (4)	PRESSURE WHEN LINER STRESS IS A MAXIMUM (PSI) (5)	PRESSURE LOAD (K/FT) (6)	SEISMIC SHEAR LOAD (K/IN) (7)	SEISMIC OVERTURNING MOMENT (K·IN) (8)	WIND SHEAR LOAD (K/IN) (9)	WIND MOMENT LOAD (K-IN) (10)
1	53.0 43.0	0	---	---	0	0	0	0	0
2	67.0 43.0	0	---	---	0	0	0	0	0
3	67.5 45.7	0	120	59	9	0	0	0	0
4	67.5 64.0	0	252	27.8	278	0	0	0	0
5	67.5 117.0	0	252	27.8	278	0	0	0	0
6	67.5 191.0(-)	35	252	27.8	278	0	0	0	0
7	67.5 191.0(+)	35	237	18.2	91.4	0	0	0	0
8	57.8 225.8	6	237	18.2	91.4	0	0	0	0

$$C = 1.0D \pm 0.05D + 1.25P + 1.0\ (T' + TL') + 1.25E$$

HORIZONTAL

TABLE II-3.1-D

LOAD TABLE

POSITION (1)	COORDINATES (FT) HOR. VERT. (2)	DEAD LOAD (K/FT) (3)	TEMP. WHEN LINER STRESS IS A MAXIMUM (°F) (4)	PRESSURE WHEN LINER STRESS IS A MAXIMUM (PSI) (5)	PRESSURE LOAD (K/FT) (5)	SEISMIC SHEAR LOAD (K/IN) (7)	SEISMIC OVERTURNING MOMENT (K·IN) (8)	WIND SHEAR LOAD (K/IN) (9)	WIND MOMENT LOAD (K-IN) (10)
1	53.0 43.0	0	---	---	0	0	0	0	0
2	67.0 43.0	0	---	47	105		0	0	0
3	67.5 45.7	127	120	47	237	15.4[illegible]	33.5×10^6	0.32	10.8 x 10
4	67.5 64.0	126	238	33.6	169	14.73	29.2×10^6	0.30	9.4 x 10
5	67.5 117.0	94	209	14.6	73.3	12.7	16.92×10^6	0.21	4.86 x 10
6	67.5 191.[illegible](-)	37	209	14.6	73.3	6.28	4.33×10^6	0.09	1.05 x 10
7	67.5 191.0(+)	37	209	14.6	73.3	6.28	4.33×10^6	0.09	1.05 x 10
8	57.8 225.8	24	209	14.6	73	3.77	5.20×10^6	0.07	.37 x 10

$$C = 1.0D \pm 0.05D + 1.0P + 1.0\,(T'' + TL'') + 1.0E'$$

VERTICAL

TABLE II-3.1-E

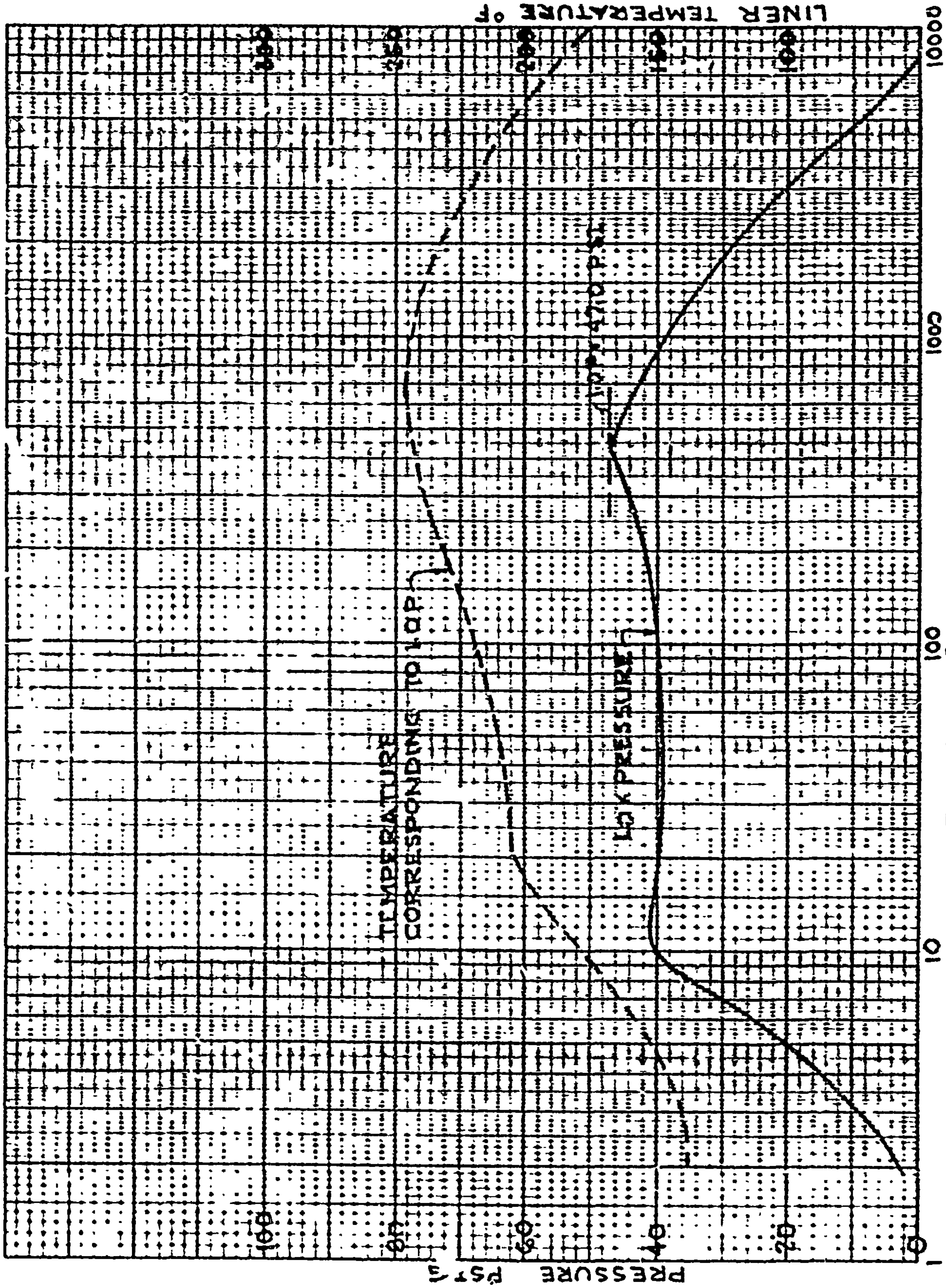
LOAD TABLE

POSITION (1)	COORDINATES (FT) HOR. (2)	VERT.	DEAD LOAD (K/FT) (3)	TEMP. WHEN LINER STRESS IS A MAXIMUM (°F) (4)	PRESSURE WHEN LINER STRESS IS A MAXIMUM (PSI) (5)	PRESSURE LOAD (K/FT) (6)	SEISMIC SHEAR LOAD (K/IN) (7)	SEISMIC OVERTURNING MOMENT (K-IN) (8)	WIND SHEAR LOAD (K/IN) (9)	WIND MOMENT LOAD (K-IN) (10)
1	53.0	43.0	0	---	---	0	0	0	0	0
2	67.0	43.0	0	---	---	0	0	0	0	0
3	67.5	45.7	0	120	47	7	0	0	0	0
4	67.5	64.0	0	238	33.6	332	0	0	0	0
5	67.5	117.0	0	209	14.6	151	0	0	0	0
6	67.5	191.0(-)	35	209	14.6	151	0	0	0	0
7	67.5	191.0(+)	35	209	14.6	75.3	0	0	0	0
8	57.8	225.8	6	209	14.6	75.3	0	0	0	0

$$C = 1.0D \pm 0.05D + 1.0P + 1.0\ (T'' + TL'') + 1.0E'$$

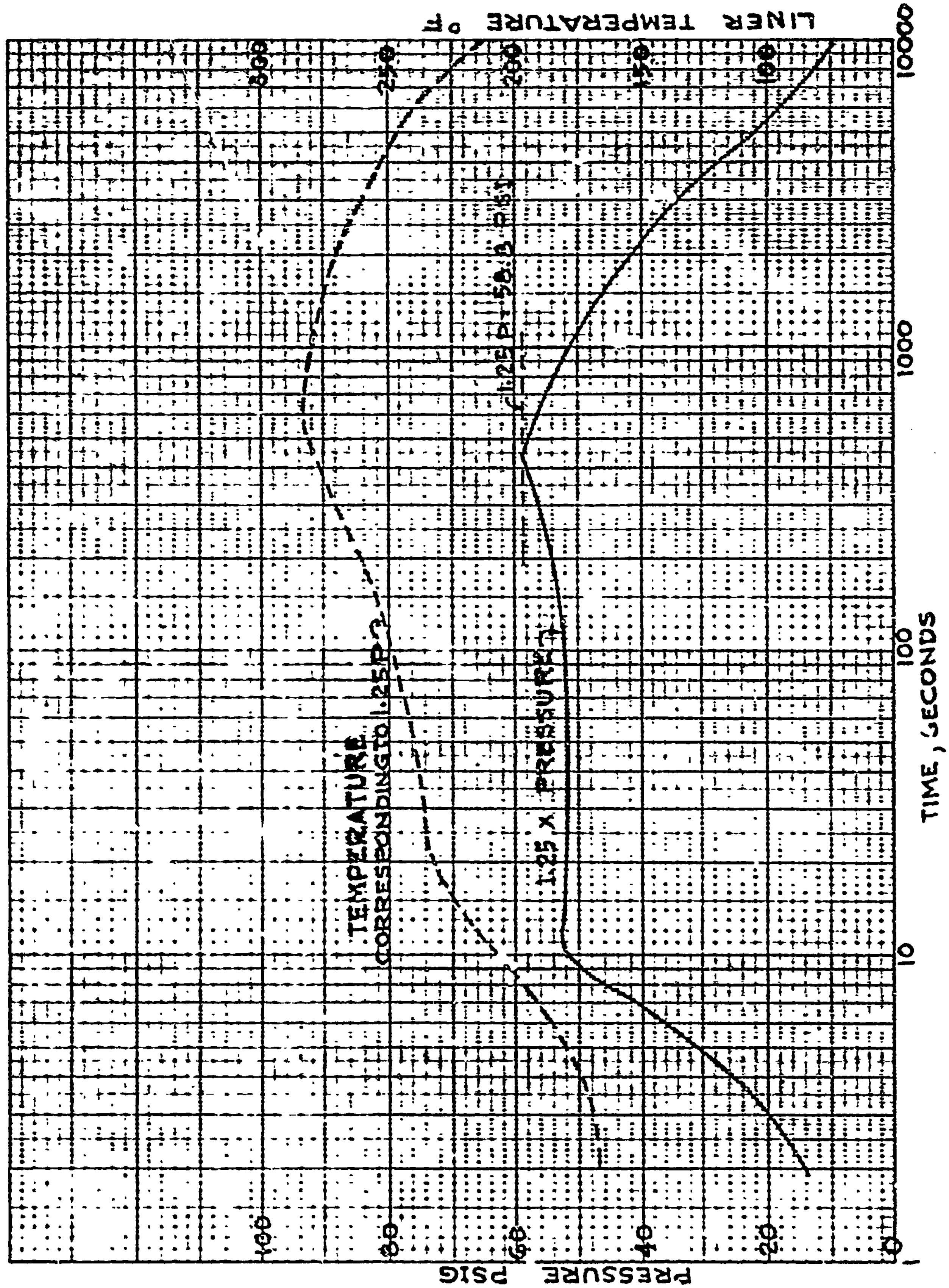
HORIZONTAL

TABLE II-3.1-F



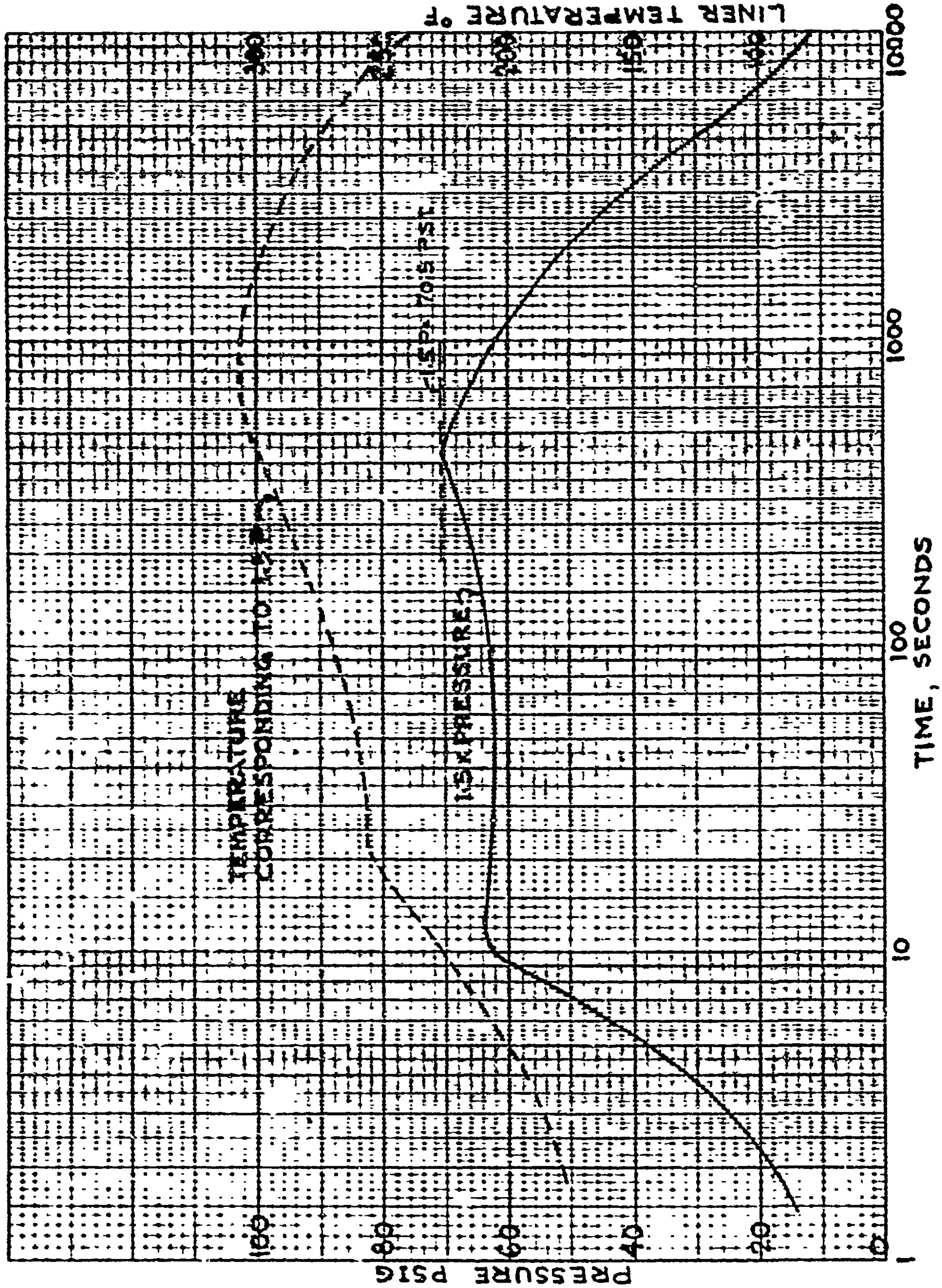
DESIGN PRESSURE-TEMPERATURE TRANSIENT

FIGURE II - 2.3-A



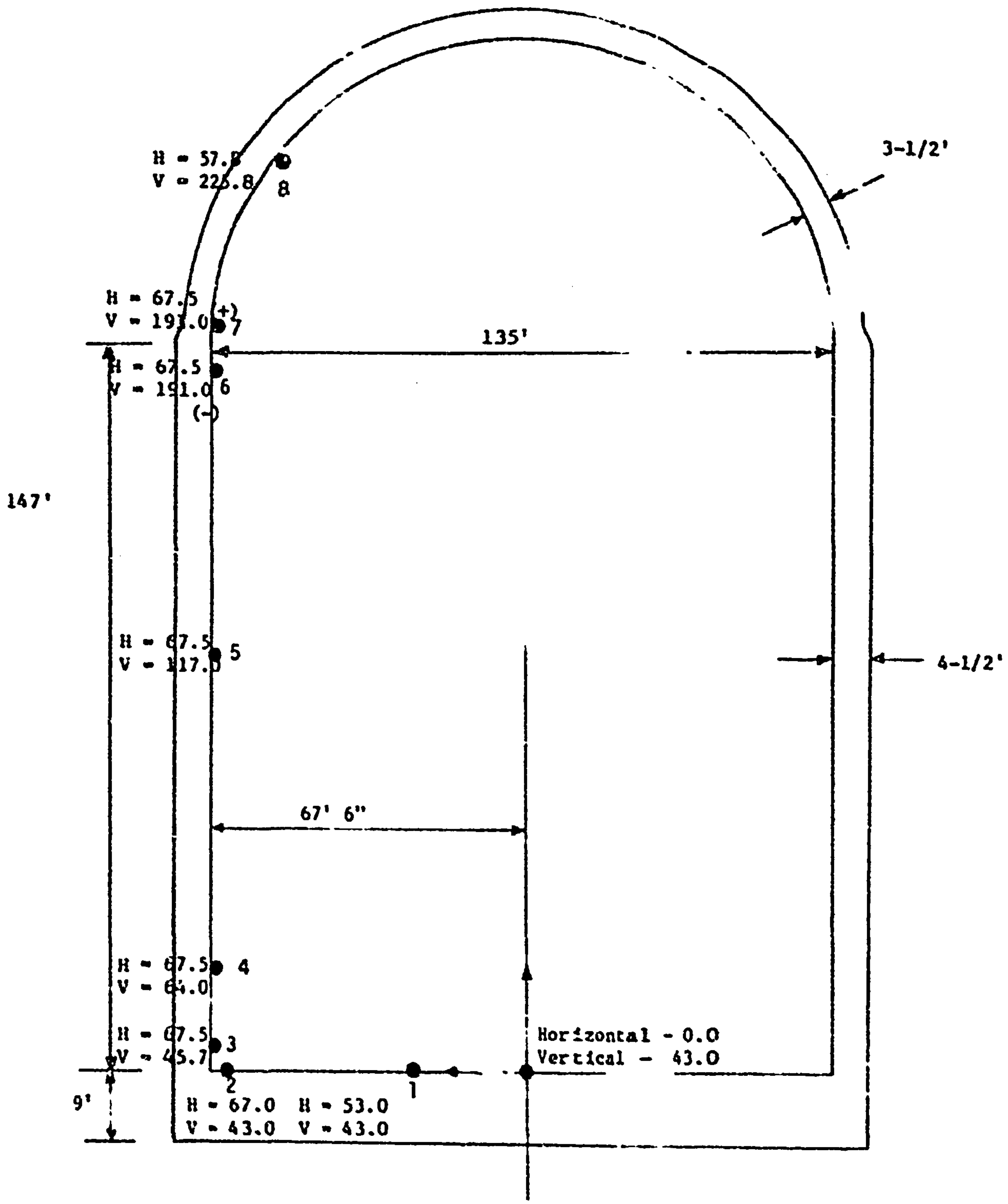
1.25 TIMES DESIGN PRESSURE-TEMPERATURE TRANSIENT

FIGURE II - 2.4-A



1.50 TIMES DESIGN PRESSU.E-TEMPERATURE TRANSIENT

FIGURE II - 2.4-8



COORDINATES OF LINER POINTS CONSIDERED IN REPORT

FIGURE II - 3.1-A

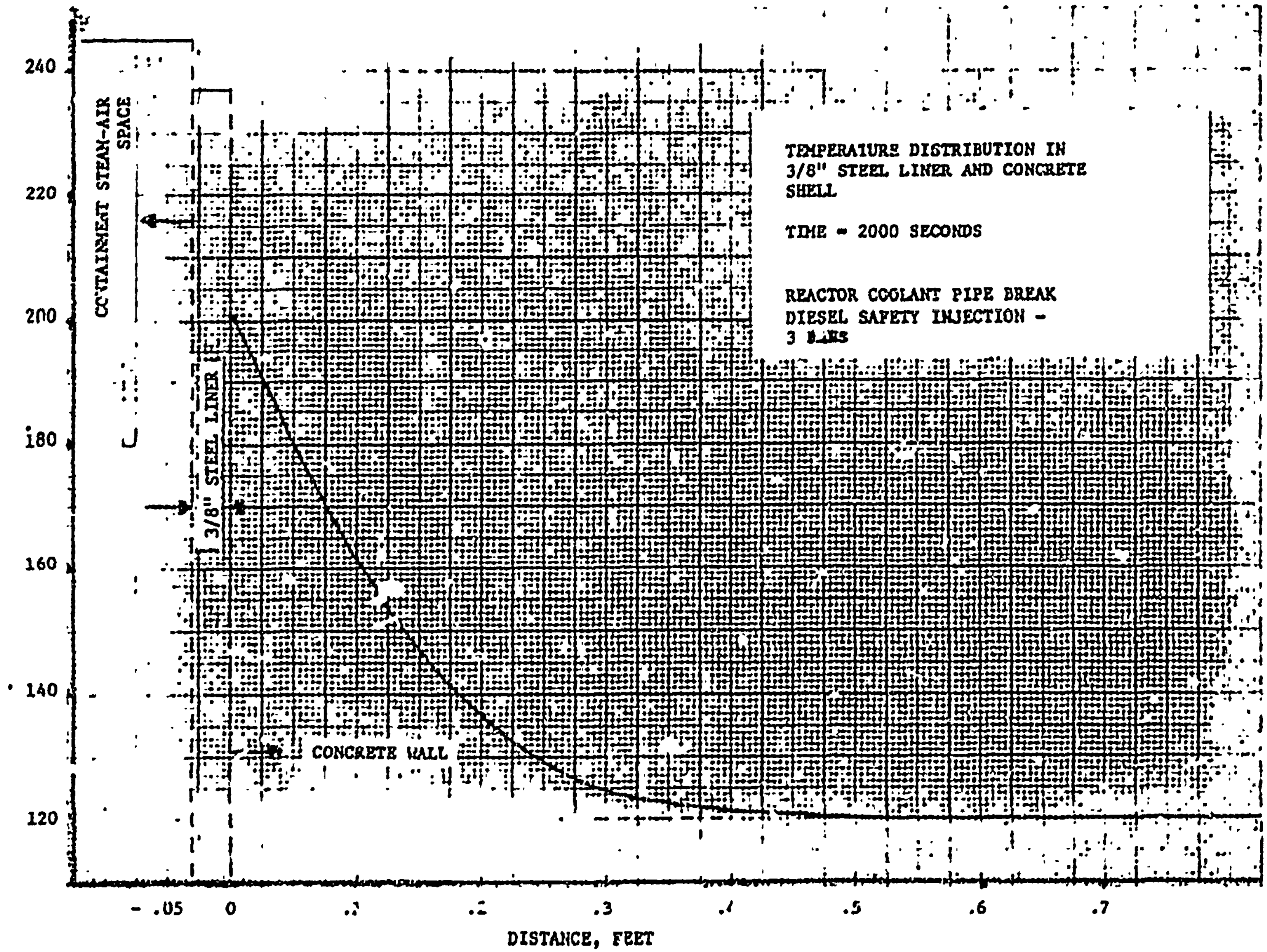


FIGURE II - 3.2-A

STRESS REDISTRIBUTION

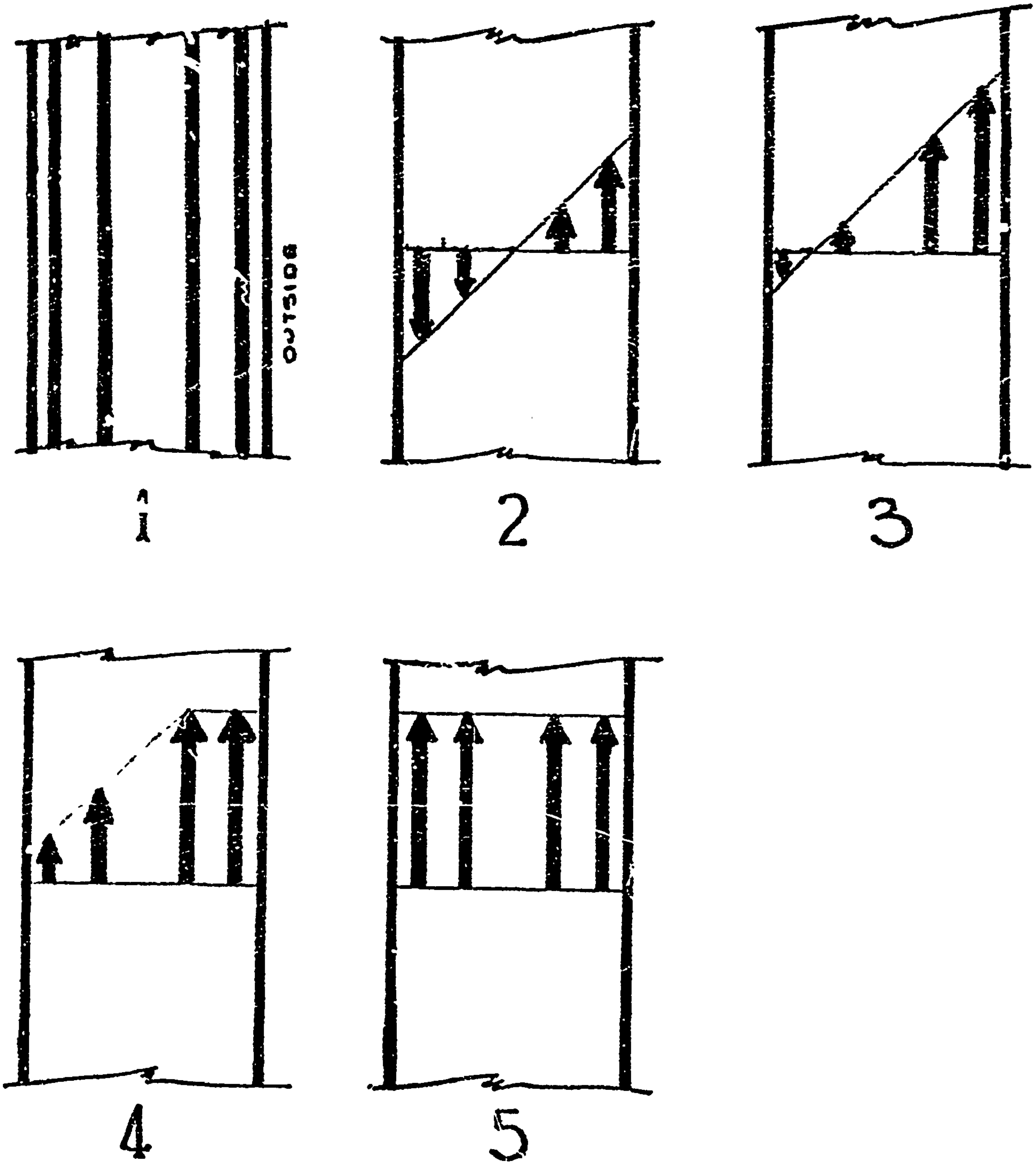


FIGURE II - 3.2-B

III. DESIGN AND STRESS ANALYSIS

1.0 DESIGN DESCRIPTION

The reactor containment is a reinforced concrete shell in the form of a vertical right cylinder with a hemispherical dome and a generally flat base, supported on rock. The inside surface of the structural concrete is lined with steel plate anchored in the concrete shell.

The containment, which is shown on Figure III-1.0-A, has cylindrical side walls which are 148 feet high measured from the horizontal liner plate on the base to the spring line of the dome and has an inside diameter of 135 feet. The dome above the side walls has an inside radius of 67'-6". The thickness of the reinforced concrete base is 9 feet, the side walls are 4'-6" and the dome thickness is 3'-6". The change in wall thickness of the dome and the cylinder will be accomplished above the spring line such that the inside radius of the dome and cylinder will be equal. The bottom horizontal liner plate will be covered with 3 feet of concrete, the top of which will form the floor of the containment. The reactor cavity has, as a minimum, 24 inches of concrete cover.

Anchorage of the liner to the concrete shell is effected as shown on Figure III-1.0-B and described below.

Attachment of the dome liner to the concrete is made by a combination of structural steel tee sections welded to the exterior face of the dome plate in two directions at approximately five foot intervals and Nelson Studs which are provided between the tees. The liner for the cylindrical portion of the concrete shell is anchored by means of Nelson Studs welded to the plate at 14 inches vertical spacing and 24 inches horizontal spacing on the 3/8 inch thick plate and 28 inches by 24 inches on the 1/2 inch thick plate. The first course of studs is approximately 18 inches above the base slab. Results of the analysis performed for the base slab preclude the need for anchorage of the bottom horizontal liner plate to the concrete base. This is discussed further in Section III-2.4.3.

The basic design concept for the liner utilizing stud anchorage ductility assures that the studs fail due to shear, tension or bending stress without the stud connection causing failure or tearing of the liner plate.

The design has also taken into consideration the possibility of daily stress reversals due to ambient temperature changes for the life of the plant. Fatigue limit of the studs, verified by extensive testing of the fatigue life of plates with stud shear connectors (Reference 6), will exceed the design requirements. Moreover, to accommodate possible fatigue failure in the plate-to-stud weldment, the depth of weld to the liner plate is controlled (Reference 7) to avoid impairment of liner integrity.

2.0 METHODS OF DESIGN ANALYSIS

2.1 General

The analytical model used in the design of the containment structure is based on the bottom mat, side walls and dome acting as one continuous structure under all possible loading conditions. The loads considered in the design result from gravity, internal pressure and temperature due to a loss-of-coolant accident, external earth pressure, earthquake, and wind as outlined previously in Section II and in the discussion that follows:

Basically three separate structural components are analyzed, each in equilibrium with loads applied to it and with constraints occurring at the juncture of the structures. The three components shown on Figure III-2.1-A are:

a. The 135 ft. ID hemispherical dome
b. The 135 ft. ID cylinder
c. The base slab

Mathematically, the dome and cylinder are treated as thin-walled shell structures which permits a membrane analysis (Reference 8). Since the thickness of the dome and cylinder is small in comparison with the radius of curvature (1/15) and there are no discontinuities such as sharp bends in the meridional curves, the stresses due to pressure and wind or earthquake are calculated by assuming that they are uniformly distributed across the thickness. The base slab is treated as a flat circular plate supported on a rigid non-yielding foundation.

2.2 Assumptions

In analyzing the dome and cylindrical portion of the containment structure the following basic assumptions are made:

1. All membrane tensile stresses are assumed to be carried by the concrete reinforcement and none by the concrete. This statement reiterates a basic ground rule; namely, that the concrete will not be counted upon to resist stresses other than compression, bond and shear. However, to carry excessive shear loads radial shear reinforcing is provided in the lower portion of the wall in the form of stirrups and bent up bars as shown on Figure III-2.2-A. There is also further need for diagonal shear reinforcing in the circumferential direction over the full height of the wall to resist earthquake shears. This diagonal shear reinforcing is also used to accommodate the loads resulting from the 115% over pressure test as discussed in Section III-2.4.8.

2. Internal pressures cause circumferential and meridional tensile membrane stresses in the dome and cylinder as well as secondary radial moments and shears caused by discontinuities.

3. Dead load results in compressive meridional stresses and compressive and tensile circumferential membrane forces in the dome. The cylinder walls will be in compression vertically, with no circumferential forces except for small forces at the spring line.

4. Earthquake and wind will result in circumferential shear forces with the maximum force/foot parallel to the direction of motion. The overturning moment due to earthquake or wind forces will cause vertical forces in the wall and the maximum tensile and compressive forces 180° apart in the direction of motion.

2.3 Summary Results

The stress values at different points in the liner due to the three loading conditions (a), (b) and (c) on the containment structure, described in Section II, are summarized in Tables, III-2.3-A, B, C, D, E and F, respectively. The results indicate that the calculated maximum liner stresses are in conformance with the criteria. In determining the final stress state both poisson ratio effects and elastic deformation of the concrete are considered.

Column 3, through 9 of Tables III-2.3-A to F show the stresses resulting from individual load components of the factored load equations. In column 10 is found the total resultant liner stress considering the containment wall ridge, that is neglecting the deformation of the liner due to elastic straining of the concrete shell, and no poisson ratio effects. The stresses corrected for the interaction between liner and concrete shell are presented in Column 11 and final liner stress intensities including poisson ratio effects are shown in column 12.

2.4 Detailed Analysis

The above results are based upon detailed analysis of the containment components which are described in the following paragraphs.

2.4.1 Dome Analysis

Two loading conditions were considered in the design of the dome, construction loads and operating loads which take into account loading conditions (a), (b) and (c) previously discussed.

2.4.1.1 Construction Loads

Erection of the concrete dome requires that the steel liner carry the load imposed by the wet concrete placed in sections.

The method chosen is to stiffen the dome liner by use of structural tees used as ribs on the exterior of the plate. The stiffening ribs are arranged and proportioned to provide sufficient strength for the liner plate to carry the entire wet concrete load for a full thickness pour, without any internal supporting falsework.

A complete set of design calculations (Reference 9) for this loading condition indicating reinforcements, maximum stresses, methods of construction, etc., prepared by General Analytics, Inc. are on file at Westinghouse in Pittsburgh, Pennsylvania. Their analysis indicates that the dome liner, reinforced with the stiffening ribs, is capable of carrying the load of wet concrete during construction and the resulting stresses are within the limits established by the criteria.

The above resulting stresses are even more conservative than calculations indicate because the method of pouring the concrete will utilize an external formwork system capable of supporting the load of the wet concrete, placed in a full thickness lift. Loading on the liner plate is transferred back through a grid system of form ties to the external formwork, thus relieving the loading on the liner.

2.4.1.2 Operating Loads

The analysis of the hemispherical dome is performed by the superposition of membrane forces resulting from gravity, accident pressure and accident thermal loads. In addition, earthquake or wind loading create both direct and shear stresses in the dome and the operating temperature of the liner creates tension and compression. In the upper area of the dome (beyond about 30° above the spring line) where the seismic shears are small, seismic shears are carried by dome reinforcing steel lying in the plane of principal tension.

The dome reinforcing steel is spliced to the vertical steel in the cylindrical concrete wall, so that a continuity between the dome and the cylinder is realized.

The various loadings on the concrete dome were investigated for their effect on the liner and the magnitude of the maximum resulting combined stress on the dome liner is 27.2 ksi tension as a result of test pressure load. The maximum compressive stress is -20.7 as determined by the factored load equation (a).

2.4.2 Cylinder Analysis

The analysis of the cylinder is accomplished by the superposition of membrane forces resulting from gravity, pressure and thermal loads; overturning due to earthquake or wind; and shears due to earthquake or wind. The concrete is reinforced circumferentially using steel hoops and vertically by straight bars. Diagonal bars are placed to resist the horizontal and vertical shears due to earthquake or wind. The required capacity of the diagonal bars is determined such that the horizontal component per foot of the diagonals is equal to the maximum value of shear load and internal pressure load. Although, in the cylinder the liner has some capacity available to resist the seismic shears in the cylinder, no credit is taken for the capacity. For all of the cylinder and the lower areas of the dome, the diagonal reinforcing is designed to accommodate all seismic shears. No credit is taken for the dowel action of the vertical and horizontal bars in resisting seismic shear.

The loading on the concrete is analyzed for its effect on the liner. Loads are included in the liner by the relative displacements or constraints imposed on the liner by the reinforced concrete. In addition to the constraint imposed by the concrete, the composite action between concrete and steel liner does transmit some loading to the liner, but only to the extent of the load transfer capacity of the studs welded to the liner.

An investigation of the liner indicates that a maximum compressive stress of -30.4 ksi is induced by the factored load equation (a). The maximum tensile stress in the cylinder is generated under test pressure conditions and is equal to 30.3 ksi.

2.4.3 Base Slab Analysis

As stated previously, the base slab is treated as a flat circular plate supported on a rigid non-yielding foundation. For loads applied uniformly around the slab, the analysis considered only a single wedge which because of symmetry is representative of the entire base slab. For the case of concentrated loads, on the slab, the wedge upon which it rests is assumed to carry the load. Earthquake and wind loading create a non-uniform loading on the slab.

For earthquake loads in combination with pressure loads, the outer perimeter of the base slab tends to lift off its rigid foundation. The analytical model which simulates this condition is a circular flat plate with a rigid central region acted upon by earthquake, pressure and dead loads.

Figure III-2.4.3-A displays the condition at the corner where the vertical wall ties into the horizontal mat. The deformation of the rock under load is very small compared to the deflection required for the reinforced concrete mat to develop significant flexure. The friction provided by the rough rock surface and the keying action of pit under the reactor vessel prevent relative horizontal motion between the mat and the rock. A steel area equal to 0.25% of area of concrete in the mat has been provided as extensions of the steel required in the area near the cylindrical wall (illustrated by Figure III-2.2-A), where there are shears and moments caused by end restraints and by tension forces in the wall. All overburden and loose or broken rock is removed to solid foundation material and backfilled with concrete.

Since the final grades of backfill material outside of the containment will be as shown in Figure III-2.1-A, there will be a differential earth pressure from one exterior face of the containment to another. This differential is considered as a static load on the structure and its effects are included in the analysis results shown in Tables III-2.3-A, B, C, D, E and F. Seismic effects of the backfill against the structure were considered in the design.

The above method of analysis used for the base slab discounts the possibility of the base liner plate being stressed to any significant degree in its major portion. This can be seen from an examination of the maximum stresses in the base liner which shows 5.8 ksi as its highest value. This is within the criteria.

2.4.4 Discontinuity Stresses

2.4.4.1 Points of Discontinuity

Discontinuity stresses occur at changes in section or direction of the containment shell (i.e., Figure III-2.4.4.1-A).

The junction of the cylinder to the dome is a point of discontinuity since, under the internal pressure and temperature design conditions, the cylinder will tend to increase in diameter somewhat differently than the dome. To compute the unrestrained dimensional changes the dome and cylinder are considered as steel membranes equivalent to the reinforcing steel. The moments and shears are computed by equating the deformations of the cylinder and hemispherical dome at the point of juncture and solving for deflections, moment, and shears at the cylinder and the dome. The moments and shears at this discontinuity are very small in magnitude.

The juncture of the cylindrical wall and the base mat is another point of discontinuity. The analysis of this discontinuity is made on the assumption that the reinforcing hoops and liner provide the circumferential restraint and the flexural rigidity is based on a cracked concrete section. In determining discontinuity moments and shears, the mat is considered as offering complete fixity. If the entire concrete section of the wall is used in the evaluation of the flexural rigidity, a conservatively high value for moments and shears are obtained. As cracking occurs and the reinforcement takes up the load, a redistribution of stress occurs, and the stiffness of the wall is greatly reduced thereby reducing the discontinuity moments and shears. As a conservatism in the design, however, the reinforced concrete wall is designed to accommodate the moments and shears determined on a cracked section analysis.

2.4.4.2 Discontinuity Equations

The differential equation (Reference 8, page 468)

$$\frac{D\,d^4w}{dx^4} + \frac{E_s h_s}{a^2} w = Z$$

represents the basis of solution for all problems of symmetrical deformation of circular cylindrical shells of constant wall thickness. The solution of this differential equation is,

$$w = \frac{e^{-\beta x}}{2\beta^3 D} [\beta M_o(\sin \beta x - \cos \beta x) - (Q_o \cos \beta x)] + \frac{a^2 Z}{E_s h_s}$$

For the fixed end condition,

$$(w)_{x = o} = \frac{-1}{2\beta^3 D} (\beta M_o + Q_o) + \frac{a^2 Z}{E_s h_s} = 0$$

and

$$\left(\frac{dw}{dx}\right)_{x = o} = \frac{1}{2\beta^2 D} (2\beta M_o + Q_o) = 0.$$

The nomenclature for the above equations referred to Figure III-2.4.4.1-A and are defined as follows:

D is the flexural rigidity

w is the radial deflection of the wall

x is the distance from the intersection of the wall and base

E_s is the modulus of elasticity

a is the mean radius of the wall

h_s is the area of circumferential membrane steel and liner per unit height of cylinder

Z is the load intensity

$$\beta = \sqrt[4]{\frac{E_s h}{4 a^2 D}}$$

M_o is the moment at the base

Q_o is the shear at the base

The moment and shear distribution at the base and above are evaluated from the preceding equations. This, in turn, permits an analysis of the structural sections and dictates the need for reinforcements in the concrete.

The calculation of the magnitude of the moments and shears, with their resulting stresses, permits an evaluation of the liner stresses at the juncture of mat and shell. The shear capacity of the studs in the vicinity of the juncture points is less than 10 per cent of the shear capacity required to transfer total bending stresses into the liner. For this reason stresses induced in the liner by bending of the concrete shell have been neglected.

2.4.5 Buckling Considerations

The design of the liner takes into consideration buckling of the plate under loading. In order to determine the critical buckling stress the plate is assumed to be hinged along EFGH as shown in Figure III-2.4.5. This assumption corresponds to buckling mode type III as identified in reference 10. The critical buckling stress for the case of equal bi-axial compression of the assumed hinged plate EFGH is 38.1 ksi. The maximum calculated stresses as shown in Tables III-2.3-A, B, C, D, E and F are -30.4 ksi vertically and -25.0 ksi horizontally and from a Mohr's circle consideration, the normal stress on the assumed hinged plate is -29.0 ksi and the shear stress 2.34 ksi. The shear stresses on the assumed hinged plate is of such low magnitude that no reduction of normal critical buckling stress results. Since the maximum applied stress of 29.0 ksi is less than the critical buckling stress of 38.1 ksi, the plate will not buckle.

2.4.6 Penetration Analysis

2.4.6.1 General Criteria

The adequacy of penetrations in retaining strength and ductility while preventing leakage is ensured by the following measures:

1. The materials for all components are selected primarily because of their high ductility.

2. By design, all penetrations can withstand all stresses imposed on them as a result of normal plant operation and the hypothetical loss-of-coolant accident. Specifically, the joint between the penetration sleeve and the building liner plate is reinforced with a thickened plate. The sleeve is anchored to the concrete by means of stud anchors welded to a steel ring which is, in turn, welded to the sleeve. The penetration end plates through which the pipes or electric cable pass are designed to withstand the penetration's internal air pressure during normal operation and also the containment internal pressure during the hypothetical loss-of-coolant accident.

3. Load transfer around penetrations is based on maintaining continuity of main reinforcing bars which is accomplished by bending and by the addition of diagonal reinforcing to ensure the transfer of tensions, bending moments and shears. At the equipment access opening, a reinforced concrete boss is provided to carry stresses around the opening and to resist bending and torsional moments created by the load transfer. Again, main reinforcement is bent to maintain continuity of stress and additional diagonals are provided to ensure load transfer.

4. The liner is basically not a load-carrying member and because of its integral relationship with the reinforced concrete is subjected to the strains which the reinforced concrete imposes upon it. Therefore, the criterion at penetrations is one of consistent deformations rather than transfer of load. Nevertheless, the liner is reinforced at each penetration according to the rules set forth in the ASME Unfired Pressure Vessel Code, Section VII UG-36. An additional conservatism is that the reinforcing requirements set forth in the ASME Code are based on unequal bi-axial stresses, whereas the liner principal stresses, being dependent on reinforcing bar strains, are essentially equal. For the penetrations the maximum stress at the opening is essentially the same as the average nominal stress of the liner.

5. The weldments of liner to penetration sleeve are of sufficient strength to accommodate the stress raisers around the openings. These welds shall adhere strictly to ASME Section VIII requirements for both type and strength. In addition, each weld has a channel placed over it (for pressurization and ultimate leak testing) which adds strength and stiffness to the welded area and assists in reducing stress in the weld and liner plate.

2.4.u.1 Penetration Loading

The penetration sleeves and end plates are designed to accommodate all loads imposed on them. These loads described in Section II-2.0 include the following:

1. Internal pressure

2. Concentrated loads imposed by the sleeve anchors to the concrete as the anchors strain in conjunction with wall movement under both operating and accident conditions.

3. Thermal effects due to both gradient and thermal reactions of the particular item passing through the sleeve.

4. Shear, bending and compression due to accident end pressures.

5. Shear and bending due to seismic movements of the particular item passing through the penetration.

The sleeve and expansion joint are designed to remain within the stress limitations imposed by ASME Code Section VIII.

2.4.6.2 Stress Concentration Considerations at Large Penetrations

Recent experiments (Reference 11) indicate that concrete under tension is insensitive to stress concentration effects. Should concentrations develop, however, the concrete would crack and the liner would have to span over the cracks. Although the liner has sufficient ductility to span relatively large cracks, reinforcing steel is used to control cracking and to accommodate stress concentrations should they arise. Making an analogy to a homogeneous, isotropic elastic material, subjected to equal bi-axial strains (since strain is a function of reinforcing bar stresses), the classical stress concentration factors as set forth by Timoshenki in Reference 12 give a magnification factor of 2 as the maximum stress occurring at the penetration. This concentration effect dampens quickly such that within a diameter of the edge of the opening, the stress is virtually the same as the unmagnified stress.

To accommodate these wall stresses, the main reinforcing bars are uninterrupted by bending the reinforcing bars around the penetration. In addition, reinforcing steel, equal in area to the area of reinforcing bent around the penetration, is supplied. This approach is conservative because of the following:

1. Experiments indicate very small stress raisers in concrete.

2. Additional reinforcing is provided to resist stress raiser factors of two although this factor occurs only at the edge of the opening.

3. The liner has sufficient strength and ductility to span any cracks in the concrete.

The pressure forces on the penetration result in moments and shears on the wall at the edge of the opening. This effect is damped out by the restraint of the horizontal bars on local deformations. Bars radial to the opening are provided to resist these effects.

2.4.7 Seismic Analysis

2.4.7.1 General

As indicated in Section II, the design of the containment, which is a Class I structure, utilizes the "response spectrum" approach in the analysis of the dynamic loads imparted by earthquake. The seismic design is based on the acceleration response spectrum curves developed by G. Housner for the ground acceleration at El Centro. Seismic accelerations have been computed as outlined in References 2 and 3. Seismic loads induced in the liner are based on the liner acting compositly with reinforced concrete shell. The liner is not considered as a separate load carrying member in the seismic design of the containment shell. However, stresses induced in the liner by its composite action with the concrete are considered evaluating resultant stress levels in the liner.

Damping factors used in containment design are as indicated in Table III-2.4.7.

TABLE III-2.4.7

	Component	Per Cent of Critical Damping
1.	Containment Structure	2.0
2.	Concrete Support Structure of Reactor Vessel	2.0
3.	Steel Assemblies:	
	a. Bolted or Riveted	2.5
	b. Welded	1.0
4.	Vital Piping ystems	0.5
5.	Concrete Structures Above Ground:	
	a. Shear Wall	5.0
	b. Rigid Frame	5.0

The ground acceleration has been determined to be 0.1g applied horizontally, and 0.05g applied vertically. These values have been resolved as conservative numbers based upon the recommendations from Dr. Lynch, Director of Seismic Observatory, Fordham University.

2.4.7.2 Stress Analysis

The natural period of vibration is computed by the Rayleigh method; (Reference 2, page 263) in this method the containment structure is analyzed as a simple cantilever intimately associated with the rock base and with broad base sections of adequate strength to assure full and continued elastic response during seismic motions. urther, both bending and shear deformations are considered (see Figure III-2.4.7.2-A).

The structure is divided into sections of equal length and loaded laterally by dead weight of the section and any equipment and live load occurring at the section. Deflections caused by shear and moments are then determined and the end deflection is given the value $\phi' = 1.0$ with corresponding values determined for other sections. The natural period of vibration (T) for the structure is then determined by the relation:

$$T = 2\pi \left[\frac{Y_o \Sigma \phi'^2 \ dm}{g\Sigma \ \phi' \ dm}\right]^{1/2}$$

This expression is derived by setting potential energy equal to kinetic energy and solving for T, wherein terms are defined as follows:

Y_o = Maximum Actual Deflection

ϕ' = $\dfrac{\text{Deflection of Section Under Consideration}}{\text{Maximum Actual Deflection}}$

g = Acceleration Due to Gravity

dm = Weight of Section Under Consideration

Using the derived Period, T, and entering the average acceleration spectral curves, Figure III-2.4.7.2-B and III-2.4.7.2-C and applying 2% critical damping, a spectral acceleration for the containment is selected. Since this average curve is based upon a ground acceleration 0.33g, the average spectral acceleration is multiplied by 0.1/0.33 for the containment structure acceleration with the seismic loading to be used for this plant. This value is derived to determine the base shear. The distribution of base shear will be upon a triangular loading assumption based upon the formula (see Figure III-2.4.7.2-A).

$$F_x = \frac{Vw_x h_x}{\Sigma \ w \ h}$$

where

F_x = lateral force applied to a level designated as x

V = total lateral load or shear at the base

w_x = that portion of total load which is located at the level designated as x

h_x = height in feet above the base to the level designated as x

Σwh = summation of the products of all $w_x h_x$ for the structure.

This formula, taken from Reference 13, yields a load distribution pattern with zero loading at the base to a maximum loading at the spring line of the dome. Above this line the loading will somewhat decrease due to a change in section and consequently change in weight. This load distribution allows the determination of shears and moments at any critical section through the containment from which the appropriate unit stresses are obtained.

A summary of the stresses computed for earthquake loading is included in Section III-2.3.

2.4.8 Stresses During Overpressure Testing

It will be assumed that during the 115% pressure test of the containment at 54 psig, the liner will contribute to the net overall cross-sectional strength of the structure to resist membrane forces. Since the liner will be anchored to the shell by Nelson Studs at appropriate intervals, elastic stability will be assured and the liner will not be loaded beyond a 95% yield. Results of the calculations for the overpressure test indicate maximum stresses of 30.3 ksi in the liner which are within the allowance of 95% of yield. Tables III-2.3-G and H show the liner stresses resulting from test pressure load at representative points in the structure as indicated in Figure II-3.1-A.

2.4.9 Thermal Expansion During Accident

The liner will make only a small contribution to the structural capability of the total containment under an accident loading condition. It will tend to expand faster than the concrete at increased temperature, and therefore, will be stressed first in tension, due to pressure build up, and then in compression, as a result of temperature rise. Insulation material will be applied to the lower 20 on the inside of the liner cylinder to maintain stresses within the design criteria and to ensure elastic stability.

The maximum liner stresses, computed for this condition, is 30.4 ksi, which is within the design criteria.

LINER STRESS TABLE

POSITION (1)	COORDINATES (FT) HOR. VERT. (2)	DEAD LOAD STRESS (KSI) (3)	THERMAL LOAD STRESS (KSI) (4)	PRESSURE LOAD STRESS (KSI) (5)	SEISMIC SHEAR STRESS (6)	SEISMIC OVERTURNING MOMENT STRESS (KSI) (7)	WIND SHEAR STRESS (8)	WIND MOMENT STRESS (9)	TOTAL* STRESS (KSI) (10)	LINER† STRESS (KSI) (11)	FINAL LINER• STRESS (KSI) (12)
1	SEE TABLE II-3.1-A	0	0	0	0	0	0	0	0	0	0
-		0	0	5.8	0	0	0	0	5.8	5.8	5.8
3		-4.80	insulated	13.36	0	0	0	0	8.56	8.56	9.37
4		-4.98	-36.0	6.97	0	0	0	0	-34.01	-24.4	-30.4
5		-5.97	-30.2	5.11	0	0	0	0	-31.06	-20.4	-26.4
6		-2.35	-30.2	5.11	0	0	0	0	-27.44	-17.9	-23.6
7		-2.14	-32.1	6.73	0	0	0	0	-27.51	-16.5	-20.7
8		-1.44	-30.4	4.80	0	0	0	0	-27.04	-16.0	-19.7

$$C = 1.0D \pm 0.05D + 1.5P + 1.0\,(T + TL)$$

VERTICAL

* Assuming Wall Rigid
† Considering Interaction of Wall and Liner
• Considering Poisson's Ratio Effects

TABLE III-2.3-A

LINER STRESS TABLE

POSITION (1)	COORDINATES (FT) HOR. VERT. (2)	DEAD LOAD STRESS (KSI) (3)	THERMAL LOAD STRESS (KSI) (4)	PRESSURE LOAD STRESS (KSI) (5)	SEISMIC SHEAR STRESS (6)	SEISMIC OVERTURNING MOMENT STRESS (KSI) (7)	WIND SHEAR STRESS (8)	WIND MOMENT STRESS (9)	TOTAL* STRESS (KSI) (10)	LINER† STRESS (KSI) (11)	FINAL LINER• STRESS (KSI) (12)
1	SEE TABLE II-3.1-A	0	0	0	0	0	0	0	0	0	0
2		0	0	0	0	0	0	0	0	0	0
3		0	insulated	.92	0	0	0	0	.92	.92	3.26
4		0	-33.4	7.08	0	0	0	0	-26.32	-17.1	-25.0
5		0	-31.1	7.5	0	0	0	0	-23.6	-17.5	-24.0
6		1.64	-31.1	7.5	0	0	0	0	- 9.56	-16.2	-22.0
7		2.19	-31.8	7.27	0	0	0	0	-22.34	-11.6	-16.8
8		.47	-29.2	6.31	0	0	0	0	-22.42	-9.9	-15.6

$$C = 1.0D \pm 0.05D + 1\ 5P + 1.0\ (T + TL)$$

HORIZONTAL

* Assuming Wall Rigid
† Considering Interaction of Wall and Liner
• Considering Poisson's Ratio Effects

TABLE III-2.3-B

LINER STRESS TABLE

POSITION (1)	COORDINATES (FT) HOR VERT. (2)	DEAD LOAD STRESS (KSI) (3)	THERMAL LOAD STRESS (KSI) (4)	PRESSURE LOAD STRESS (KSI) (5)	SEISMIC SHEAR STRESS (KSI) (6)	SEISMIC OVERTURNING MOMENT STRESS (KSI) (7)	WIND SHEAR STRESS (8)	WIND MOMENT STRESS (9)	TOTAL* STRESS (KSI) (10)	LINER† STRESS (KSI) (11)	FINAL LINER' STRESS (KSI) (12)
1	SEE TABLE II-3.1-A	0	0	0	0	0	NOT CALCULATED SINCE WIND LOADS ARE LESS THAN SEISMIC LOADS		0	0	0
2		0	0	4.85	0	0			4.85	4.85	4.85
3		-4.80	insulated	11.25	7.35	4.45			10.9	10.9	11.8
4		-4.98	-34.3	8.38	7.15	-3.96			-34.86	-24.8	-30.2
5		-5.97	-30.2	8.89	6.40	-3.49			-30.77	-20.6	-25.5
6		-2.35	-30.2	8.90	2.85	-0.83			-24.48	-16.1	-20.8
7		-2.14	-27	5.33	2.85	-0.75			-24.56	-14.7	-18.3
8		-1.44	-27.5	5.46	.92	-3.00			-23.78	-14.1	-17.1

$$C = 1.0D \pm 0.05D + 1.25P + 1.0\ (T' + TL') + 1.25E$$

VERTICAL

* Assuming Wall Rigid
† Considering Interaction of Wall and Liner
' Considering Poisson's Ratio Effects

TABLE III-2.3-C

LINER STRESS TABLE

POSITION (1)	COORDINATES (FT) HOR. VERT. (2)	DEAD LOAD STRESS (KSI) (3)	THERMAL LOAD STRESS (KSI) (4)	PRESSURE LOAD STRESS (KSI) (5)	SEISMIC SHEAR STRESS* (KSI) (6)	SEISMIC OVERTURNING MOMENT STRESS (KSI) (7)	WIND SHEAR STRESS (8)	WIND MOMENT STRESS (9)	TOTAL* STRESS (KSI) (10)	LINER† STRESS (KSI) (11)	FINAL LINER˙ STRESS (KSI) (12)
1	SEE TABLE II-3.1-A	0	0	0	0	0	NOT CALCULATED SINCE WIND LOADS ARE LESS THAN SEISMIC LOADS		0	0	0
2		0	0	0	0	0			0	0	0
3		0	insulated	.77	7.35	0			0.77	0.77	3.73
4		0	-31.1	8.5	7.15	0			-22.6	-14.5	-22.1
5		0	-31.1	12.92	6.40	0			-18.18	-13.3	-19.5
6		1.64	-31.1	12.92	2.85	0			-16.54	-12.2	-17.3
7		2.19	-26.8	5.69	2.85	0			-18.92	-9.9	-14.5
8		0.47	-26.4	7.16	.92	0			-18.77	-7.6	-11.8

$$C = 1.0D \pm 0.05D + 1.25P + 1.0\,(T' + TL') + 1.25E$$

HORIZONTAL

* Assuming Wall Rigid
† Considering Interaction of Wall and Liner
˙ Considering Poisson's Ratio Effects

TABLE III-2.3-D

LINER STRESS TABLE

POSITION (1)	COORDINATES (FT) HOR. VERT. (2)	DEAD LOAD STRESS (KSI) (3)	THERMAL LOAD STRESS (KSI) (4)	PRESSURE LOAD STRESS (KSI) (5)	SEISMIC SHEAR STRESS (KSI) (6)	SEISMIC OVERTURNING MOMENT STRESS (KSI) (7)	WIND SHEAR STRESS (8)	WIND MOMENT STRESS (9)	TOTAL* STRESS (KSI) (10)	LINER† STRESS (KSI) (11)	FINAL LINER• STRESS (KSI) (12)
1	SEE TABLE II-3.1-A	0	0	0	0	0	NOT CALCULATED SINCE WIND LOADS ARE LESS THAN SEISMIC LOADS		0	0	0
2		0	0	3.87	0	0			3.87	3.87	3.87
3		-4.80	insulated	8.95	8.75	5.36			9.5	9.5	10.3
4		-4.98	-28.4	6.47	8.60	-4.74			-31.45	-22.8	-27.8
5		-5.96	-22.1	4.65	7.70	-4.19			-27.60	-18.7	-23.0
6		-2.35	-22.1	4.65	3.42	-1.02			-20.82	-13.7	-17.8
7		-2.14	-21.7	4.25	3.42	- .93			-20.52	-12.4	-15.4
8		-1.44	-22.2	4.38	1.10	- .36			-19.62	-11.6	-14.2

$$C = 1.0D \pm 0.05D + 1.0P + 1.0\ (T'' + TL'') + 1.0E'$$

VERTICAL

* Assuming Wall Rigid
† Considering Interaction of Wall and Liner
• Considering Poisson's Ratio Effects

TABLE III-2.3-E

LINER STRESS TABLE

POSITION (1)	COORDINATES (FT) HOR. VERT. (2)	DEAD LOAD STRESS (KSI) (3)	THERMAL LOAD STRESS (KSI) (4)	PRESSURE LOAD STRESS (KSI) (5)	SEISMIC SHEAR STRESS (KSI) (6)	SEISMIC OVERTURNING MOMENT STRESS (KSI) (7)	WIND SHEAR STRESS (8)	WIND MOMENT STRESS (9)	TOTAL* STRESS (KSI) (10)	LINER† STRESS (KSI) (11)	FINAL LINER' STRESS (KSI) (12)
1	SEE TABLE II-3.1-A	0	0	0	0	0	NOT CALCULATED SINCE WIND LOADS ARE LESS THAN SEISMIC LOADS		0	0	0
2		0	0	0	0	0			0	0	0
3		0	insulated	.64	8.75	0			.64	.64	3.21
4		0	-28.5	[illegible]	8.60	0			-18.3	-13.1	-20.0
5		0	-23	7.03	[illegible]	0			-15.97	-11.6	-17.4
6		1.64	-23	7.03	3.42	0			-14.33	-10.15	-14.8
7		2.19	-21.5	4.71	3.42	0			-14.6	- 7.8	-11.7
8		0.47	-21.1	5.9	1.10	0			-14.73	- 7.0	-10.5

$$C = 1.0D \pm 0.05D + 1.0P + 1.0\,(T'' + TL'') + 1.0E'$$

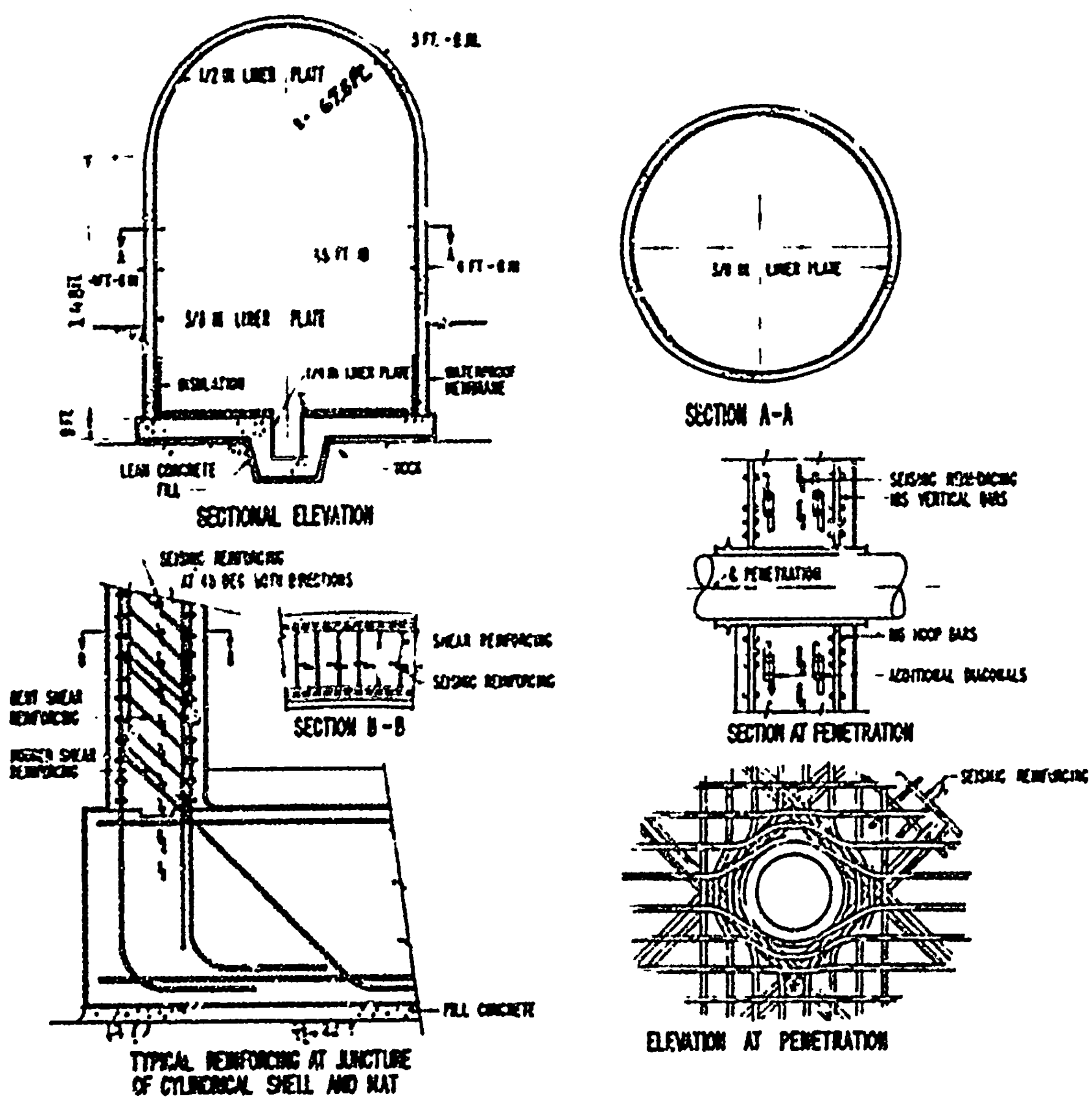
HORIZONTAL

* Assuming Wall Rigid
† Considering Interaction of Wall and Liner
' Considering Poisson's Ratio Effects

C.3

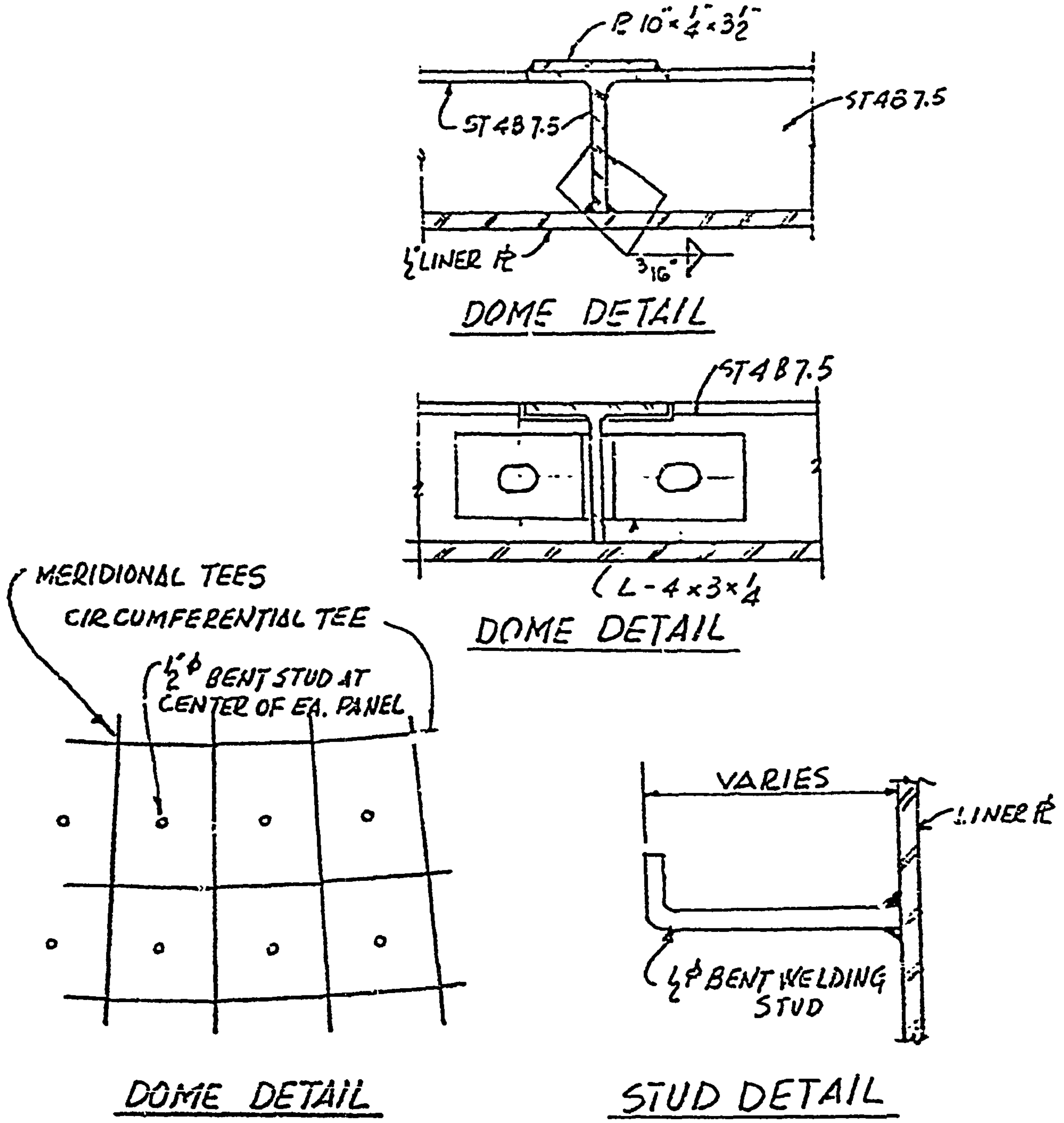
TABLE III-2.3-F

CONTAINMENT STRUCTURE



CONTAINMENT STRUCTURE

FIGURE III - 1.0-A



LINER ANCHORAGE DETAILS

FIGURE III-1.0-B

INDIAN POINT UNIT NO. 2, CONTAINMENT

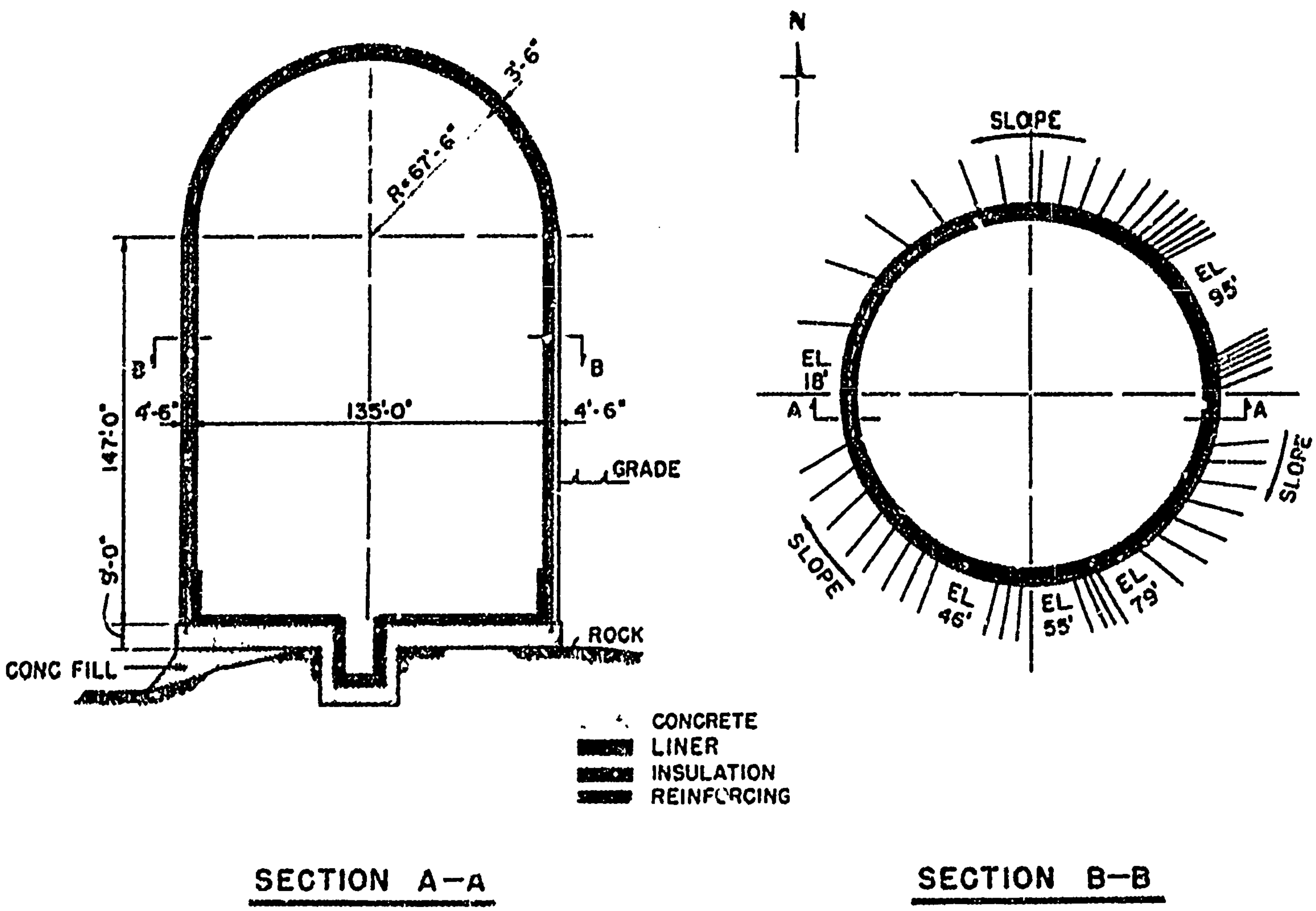
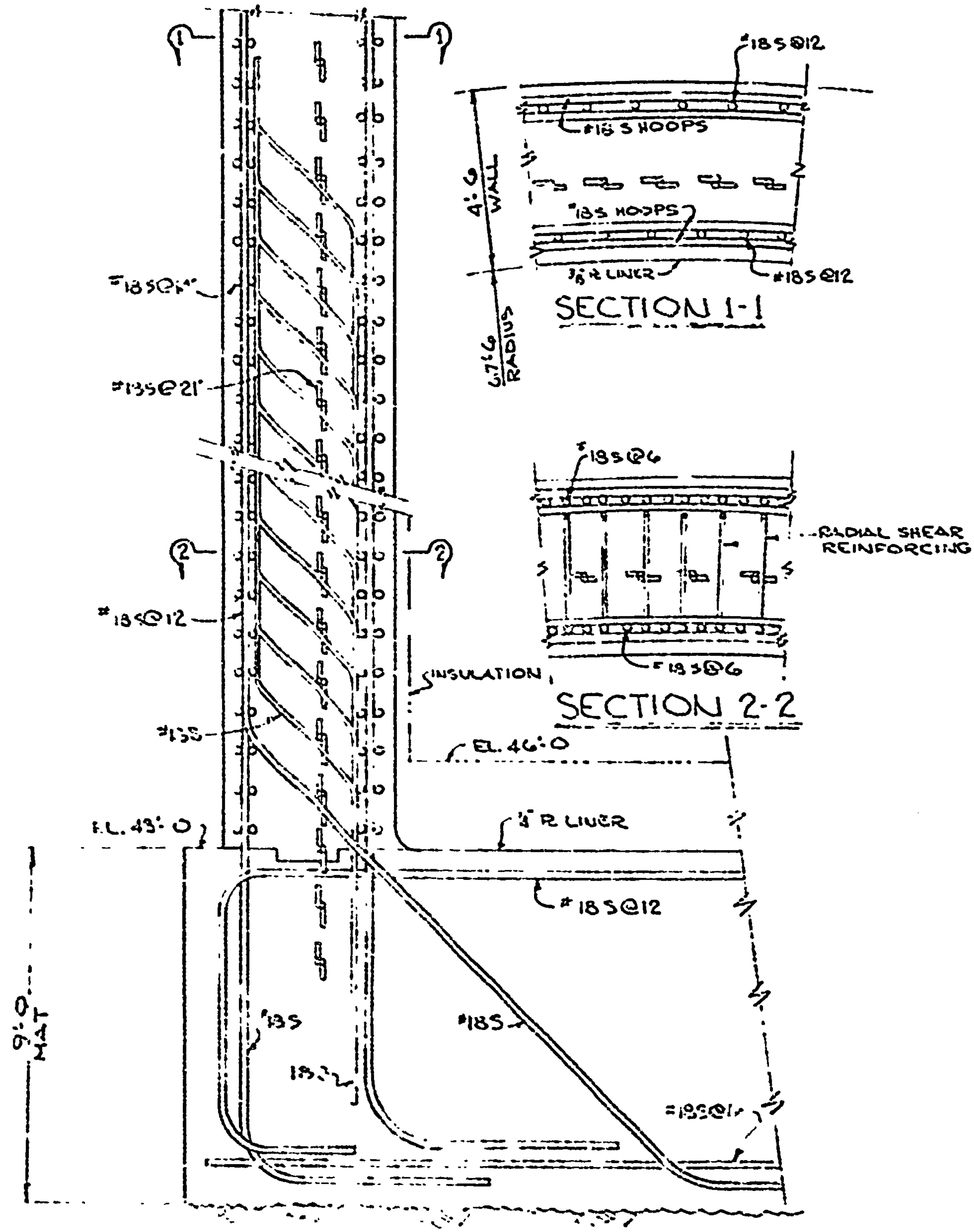


FIGURE III - 2.1-A



NOMINAL SHELL REINFORCING

NO SCALE

FIGURE III-2.2-A

PRESSURE DEFORMATION TENDENCY

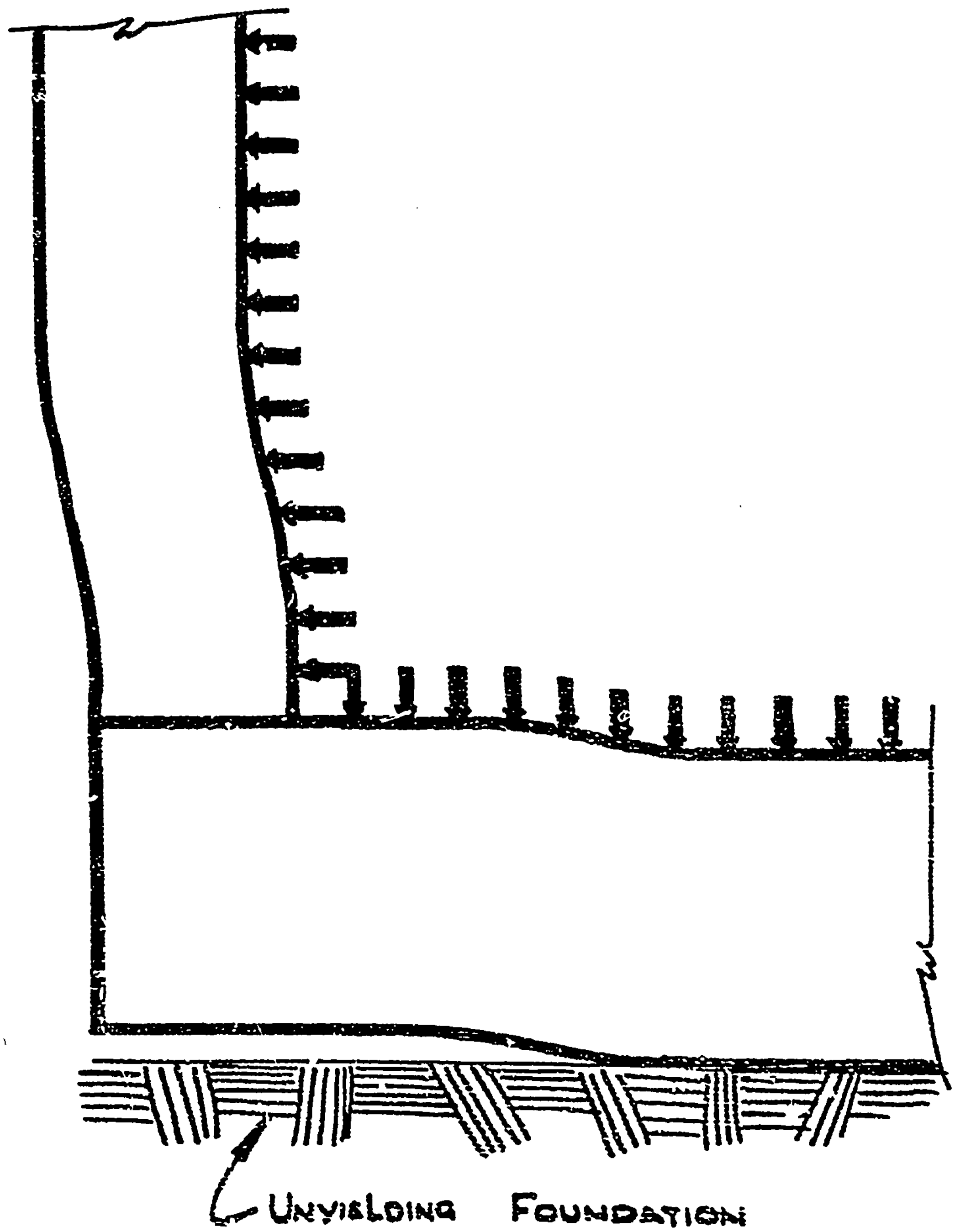
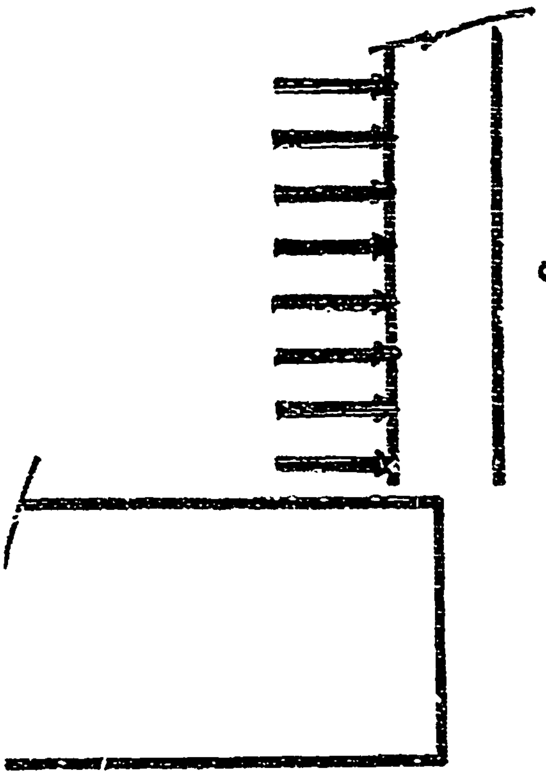


FIGURE III - 2.4.3-A

DISCONTINUITY EFFECTS



$$D\frac{d^4w}{dx^4} + \frac{E_s h}{a^2}w = Z$$

$$w = \frac{e^{-\beta x}}{2\beta^3 D}\left[\beta M_o(\sin\beta x - \cos\beta x) - Q_o \cos\beta x\right] + \frac{a^2 Z}{E_s h}$$

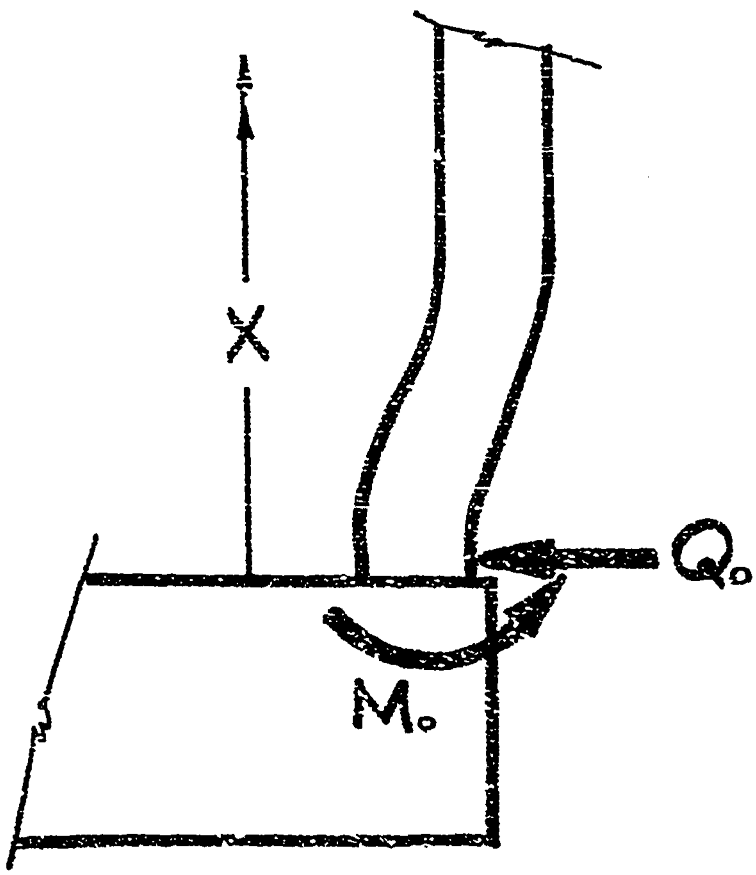
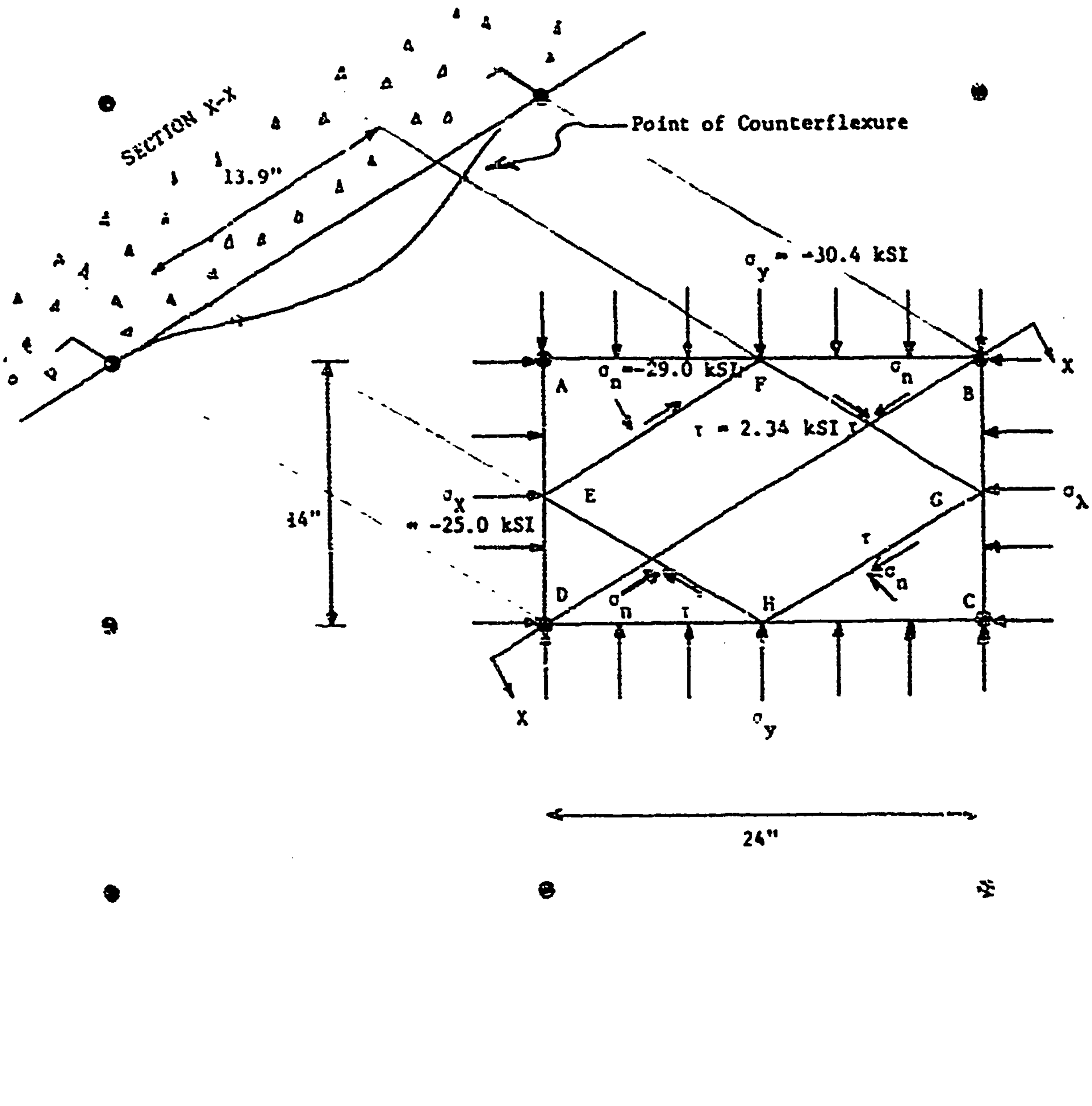


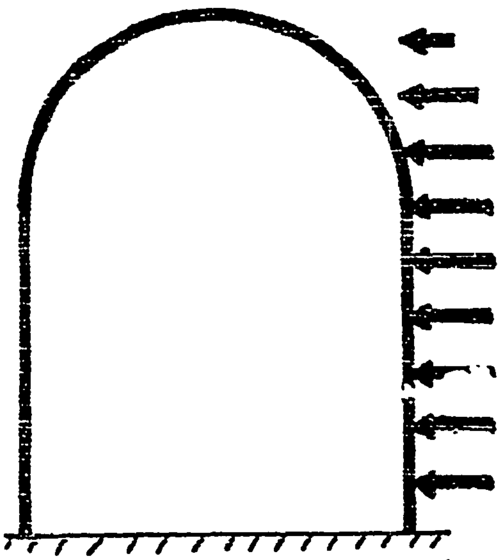
FIGURE III - 2.4.4.1-A



LINER PLATE BUCKLING MODEL

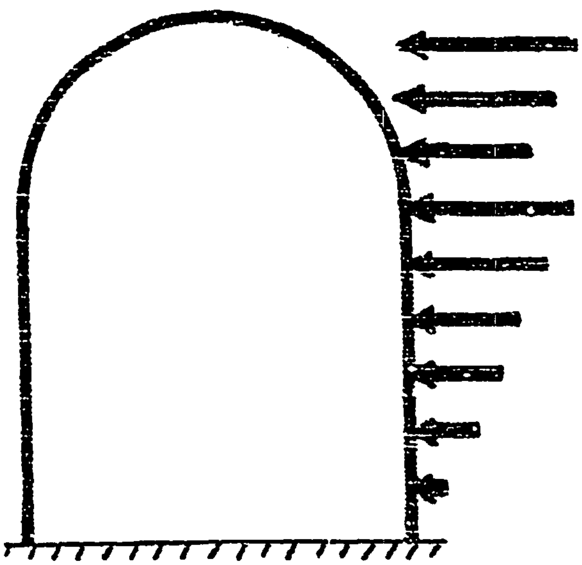
FIGURE III-2.4.5

SEISMIC ANALYSIS



$$T = 2\pi \left[\frac{\gamma_o \Sigma \phi^2 dm}{g \, \Sigma \phi \, dm} \right]^{1/2}$$

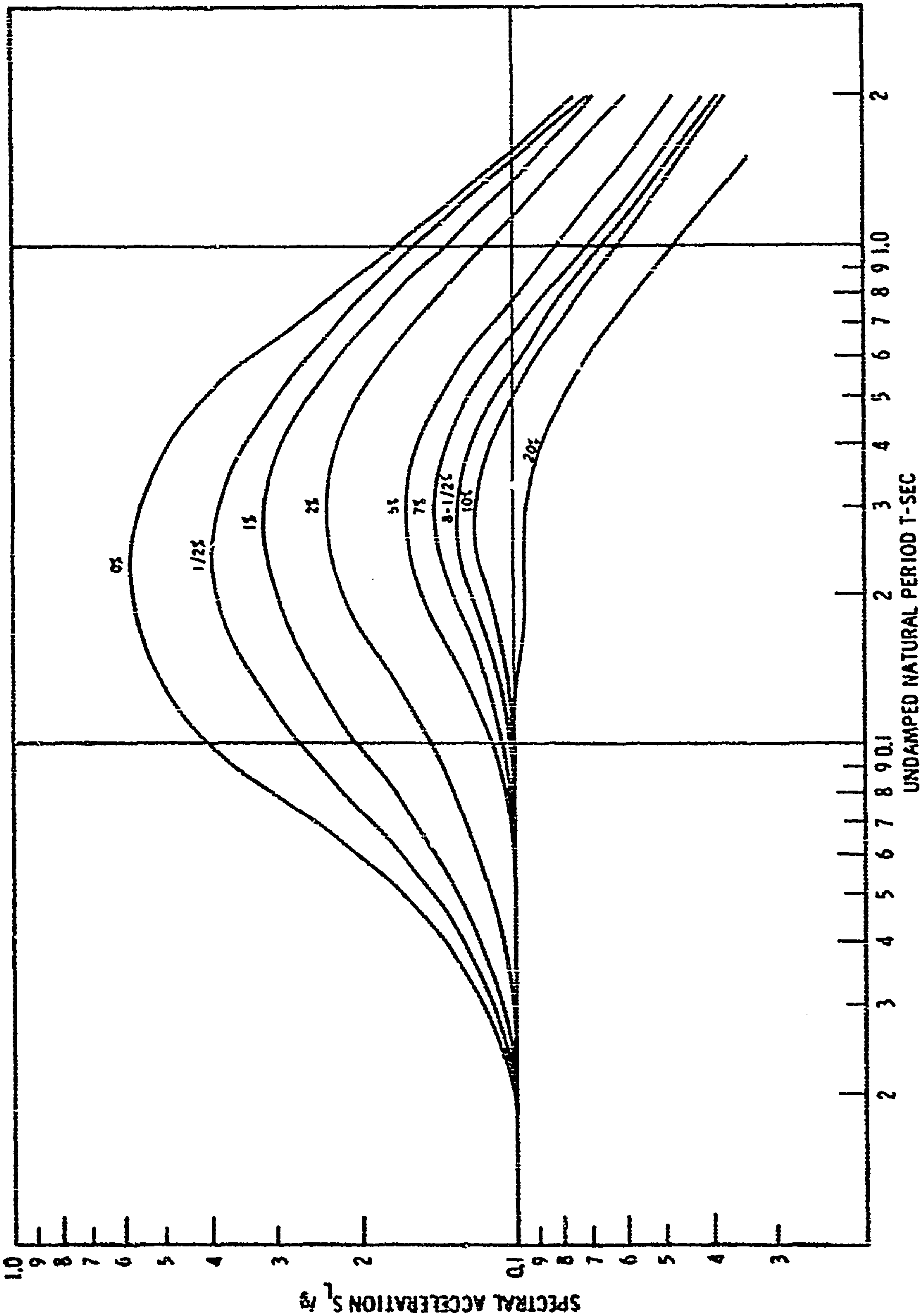
RAYLEIGH'S METHOD



$$F_x = \frac{V w_x h_x}{\Sigma w h}$$

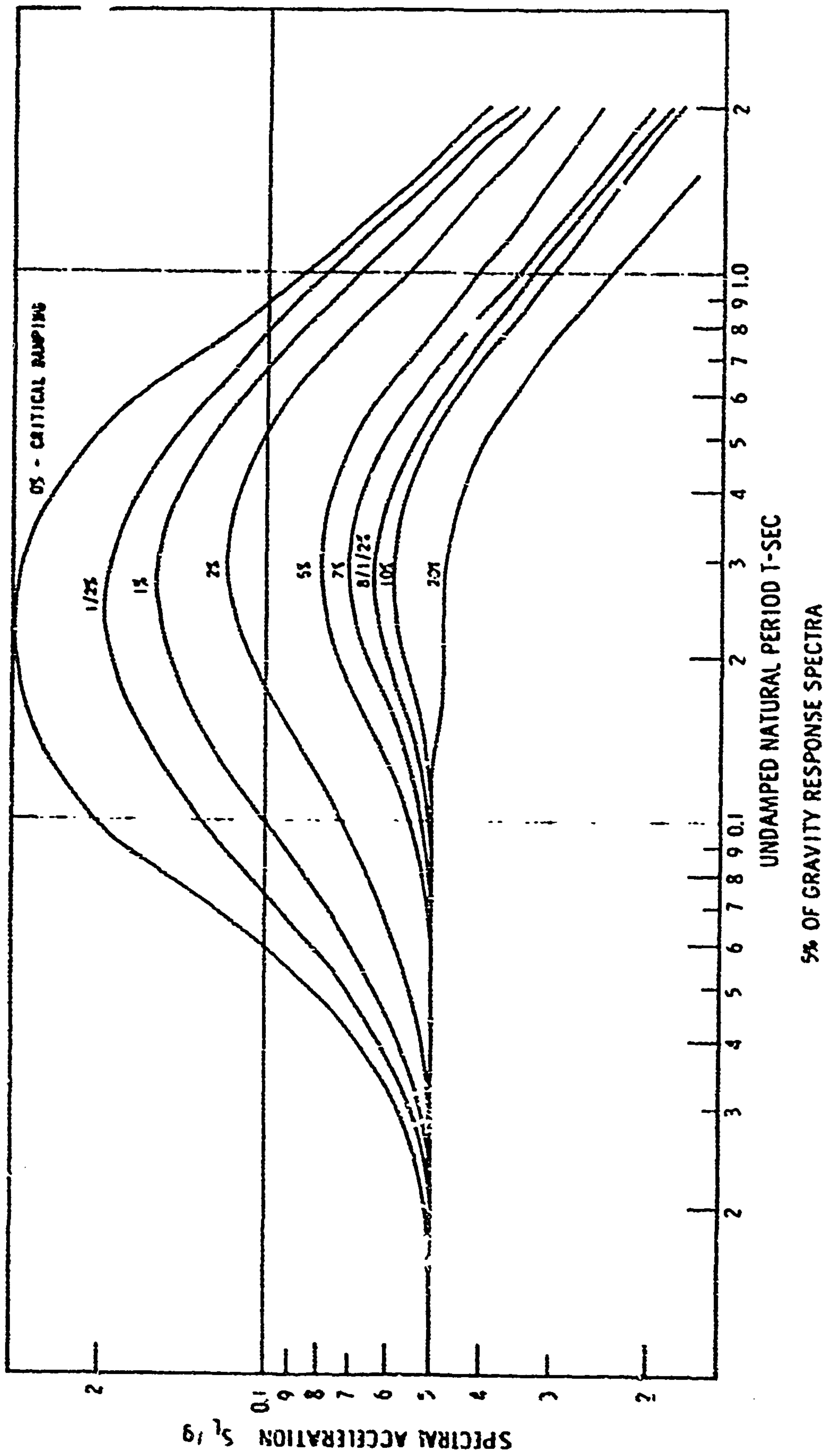
DISTRIBUTION of BASE SHEAR

FIGURE III - 2.4.7.2-A



10% OF GRAVITY RESPONSE SPECTRA

FIGURE III - 2.4 7.2-4



5% OF GRAVITY RESPONSE SPECTRA

FIGURE III - 2.4.7.2-G

IV. PENETRATIONS

1.0 GENERAL

In general, a penetration consists of a sleeve embedded in the concrete wall and welded to the containment liner. Piping penetrations pass through an embedded sleeve and the ends of the resulting annulus are closed off, either by welded end plates, bolted flanges or a combination of these.* Provision is made for differential expansion and misalignment between pipe or cartridge, and sleeve. The cartridges, however, have no expansion provisions as they are only connected at one end.

Penetrations are designed with double seals so as to permit continuous pressurization during plant operation to prevent outleakage in the event of a loss-of-coolant accident. In addition, small steel channels are welded over all joints in the containment vessel liner to form chambers which also permit continuous pressurization to demonstrate the integrity of the penetration-to-liner weld joint. Pressurizing connections are provided to continuously demonstrate the integrity of the penetration assemblies. Pressure in the penetrations and liner joint channels is maintained as 50 psig which is above containment incident pressure but less than the maximum overpressure. This is accomplished by the Containment Penetration Pressurization System. This system also allows introduction of Freon or a similar tracer gas for leak detection as may be required should consumption of pressurizing air be excessive. These provisions, in addition to the Isolation Valve Seal Water System, effectively block all containment leakage paths.

2.0 TYPES

2.1 Electrical Penetrations

"Cartridge" type penetrations are used for all electrical conductors passing through the containment. The penetrations are provided with a pressure connection to allow continuous pressurization. Ceramic type seals are used

* Electrical penetrations and the equipment hatch pass through an embedded sleeve but the ends are not closed off outside the containment building.

to provide a pressure barrier for the conductors. Typical electrical penetrations are shown in Figure IV-2.1-A.

2.2 Piping Penetrations

Double barrier piping penetrations are provided for all piping passing through the containment. The pipe is centered in the embedded sleeve which is welded to the liner. End plates are welded to the pipe at both ends of the sleeve. Several pipes may pass through the same embedded sleeve to minimize the number of penetrations required. In this case, each pipe is welded to both end plates. A connection to the penetration sleeve is provided to allow continuous pressurization of the compartment formed between the piping and the embedded sleeve. In the case of piping carrying hot fluid, the pipe is insulated and cooling is provided to maintain the concrete temperature adjoining the embedded sleeve at or below 150°F. Typical piping penetrations are shown in Figure IV-2.2-A.

2.3 Equipment and Personnel Access Hatches

An equipment hatch is provided which is fabricated from welded steel and furnished with a double-gasketed flange and bolted dished door. The hatch barrel is embedded in the containment wall and welded to the liner. Provision is made to continuously pressurize the space between the double gaskets of the door flanges and the weld seam channels at the liner joint, hatch flanges and dished door. Pressure is relieved from the double gasket spaces prior to opening the joints. The personnel hatch is a double door, mechanically-latched, welded steel assembly. A quick-acting type, equalizing valve connects the personnel hatch with the interior of the containment vessel for the purposes of equalizing pressure in the two systems when entering or to prevent both being opened simultaneously and to ensure that one door is completely closed before the opposite door can be opened. Remote indicating lights and annunciators situated in the control room indicate the door operational status. An emergency lighting and communication system operating from an external emergency supply is provided in the lock interior. Emergency access to either the inner door, from the containment interior; or to th outer door, from outside is possible by the use of special door unlatching tools.

2.4 Fuel Transfer Penetration

A fuel transfer penetration is provided for fuel movement between the refueling transfer canal in the reactor containment and the spent fuel pit. The penetration consists of a 20 inch stainless steel pipe installed inside a 24 inch pipe. The inner pipe acts as the transfer tube and is fitted with a pressurized double-gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pit. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment liner and provision is made, by use of a special seal ring, for pressurizing all welds essential to the integrity or the penetration during plant operation. Bellows expansion joints are provided on the pipes to compensate for any differential movement between the two pipes or other structures. The fuel transfer penetration is shown in Figure IV-2.4-A.

2.5 Containment Supply and Exhaust Purge Ducts

The ventilation system purge ducts are each equipped with two quick-acting tight-sealing valves (one inside and one outside of the containment) to be used for isolation purposes. The valves are manually remotely opened for containment purging but are automatically closed upon a signal of high containment pressure or high containment radiation level. The space between the valves is pressurized above design pressure while the valves are normally closed during plant operation.

2.6 Sump Penetrations

The piping penetration in the containment sump area is welded directly to the base liner. The weld to the liner is shrouded by a test channel which is used to demonstrate the integrity of the liner. This sump pipe has a normally open butterfly valve at its inlet end. This valve can be remotely closed in the event of a leak in the sump pipe or first valve.

2.7 Dome Penetration

An opening is located in the dome at the top of the vessel. This opening is for construction ventilation and will be permanently closed at the conclusion of the construction work.

2.8 Temporary Construction Openings

Temporary construction openings are provided in positions most suitable for facilitating equipment access. Reinforcement of the liner is necessary in these areas. The opening will be permanently closed after the equipment is in place and before the containment is completed and finally tested.

3.0 DESIGN OF PENETRATIONS

All personnel locks and any portion of the equipment access door extending beyond the concrete shell conform in all respects to the requirements of ASME Section VIII Nuclear Vessels Code. The weldments of liner to penetration sleeve are of sufficient strength to accommodate stress concentrations and adhere strictly to ASME Code Section VIII requirements for both type and strength. Liner reinforcements are designed to support penetrations in the appropriate portion of the liner plate during shop testing, shipping and field erection.

4.0 LEAK TESTING OF PENETRATIONS

A proof test will be applied to each penetration by pressurizing the necessary areas to 54 psig. This pressure will be maintained for a sufficient time to allow soap bubble and Freon sniff tests of all welds and mating surfaces. Any leaks found are to be repaired and retested; this procedure to be repeated until no lea ; exist.

All penetrations, the personnel air lock and the equipment hatches will be desigr-d with double seals which will be normally pressurized at 50 psig. Individual testing at 115% of containment design pressure is also possible.

5.0 STRESS ANALYSIS OF PENETRATIONS

Sufficient reinforcement is being provided around penetrations in the concrete cylinder wall to maintain strains within the elastic range. Since deformation of the liner is limited by deformation of the supporting concrete cylinder wall strains in the liner due to the discontinuity at the opening are maintained at or below yield.

NO.	[illegible]	QTY	DESCRIPTION
1	[illegible]	1	STAINLESS STEEL
2	FLANGE	1	CARBON STEEL
3	[illegible]	1	STAINLESS STEEL
4	TUBE	2	STAINLESS STEEL
5	POTTING		SILICONE [illegible]
6	INSULATION		SILICONE SLEEVING
7	POTTING		EPOXY
8	[illegible]	2	PHENOLIC
9	[illegible]		SILICONE RUBBER
10	[illegible] TUBE	1	STAINLESS STEEL
11	[illegible] TAG		

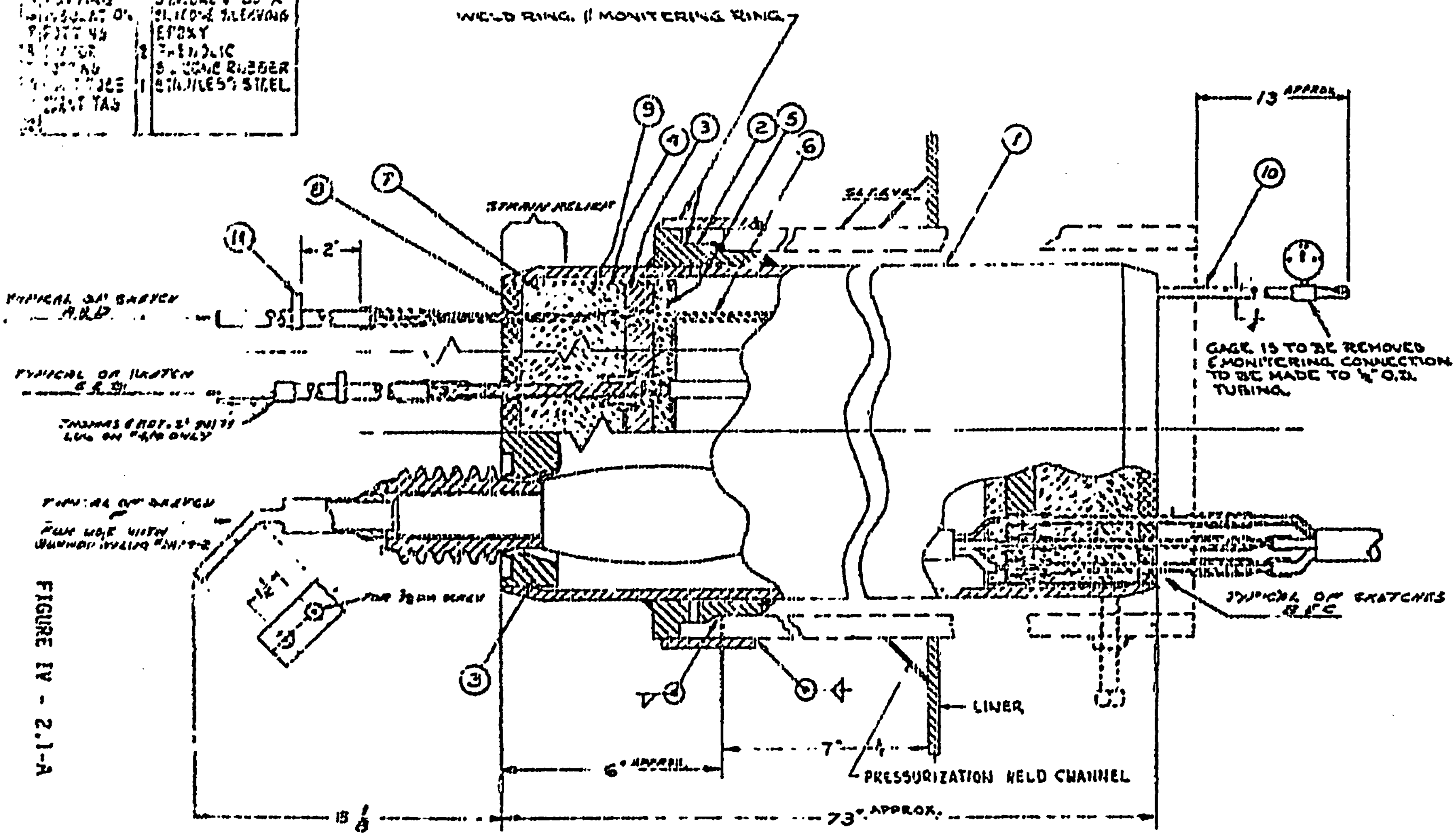
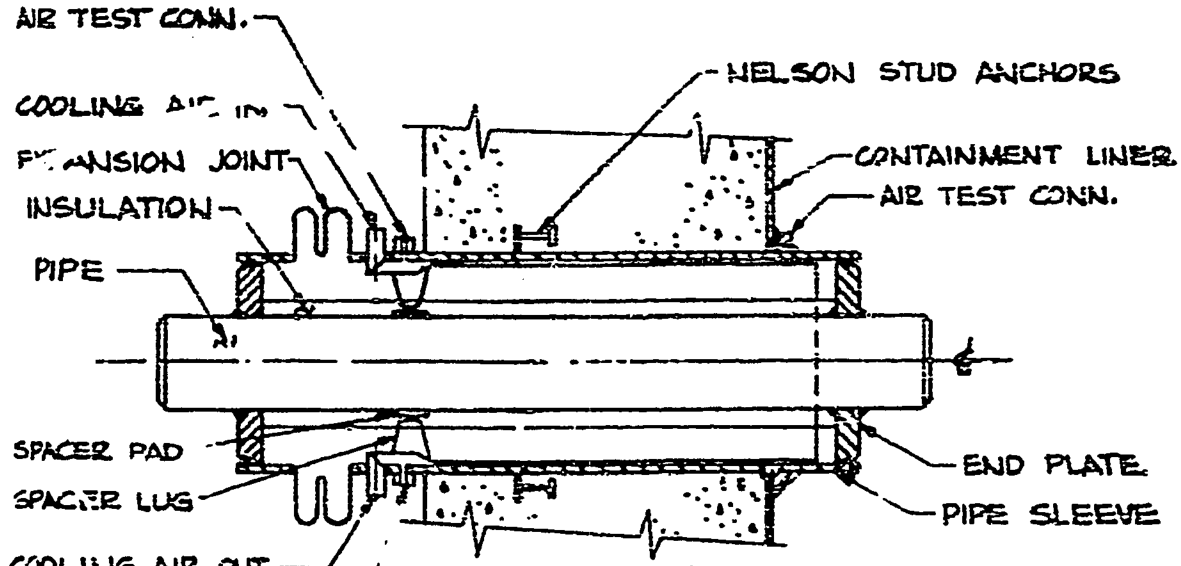
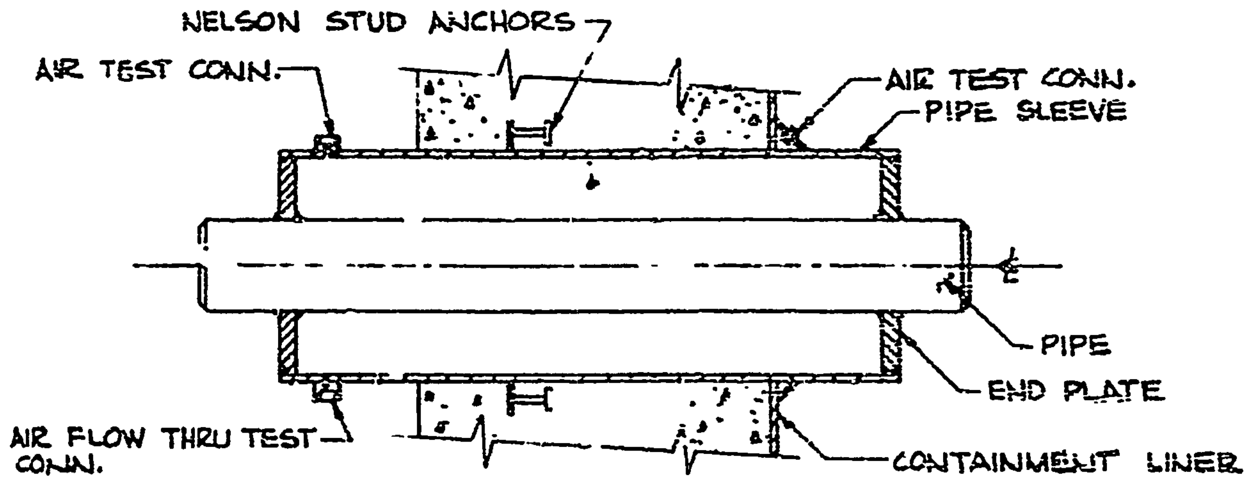


FIGURE IV - 2.1-A

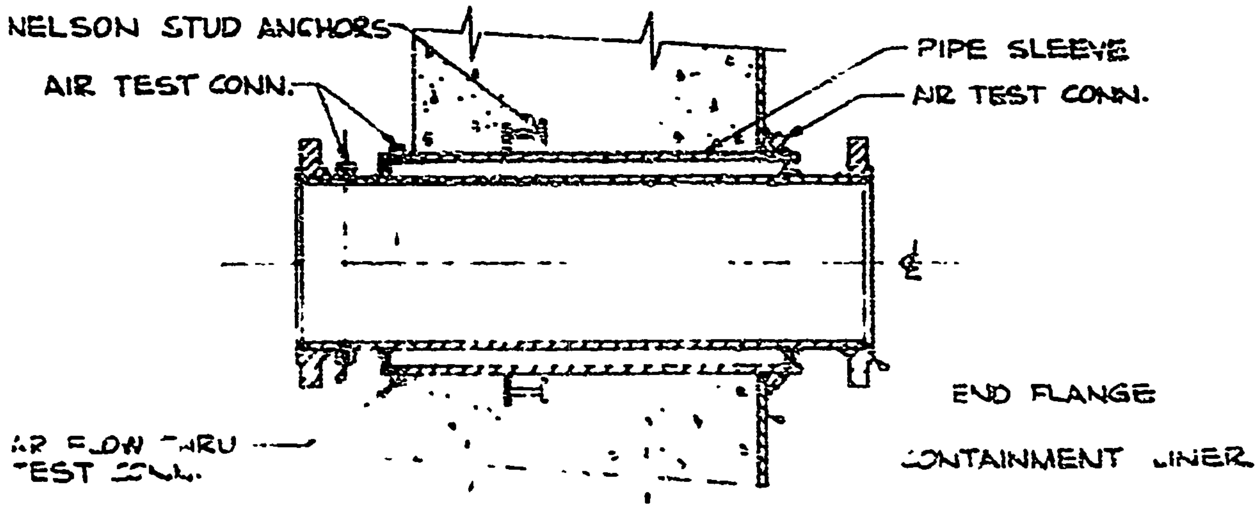
TYPICAL ELECTRICAL PENETRATIONS



PIPING PENETRATION
HOT PIPE



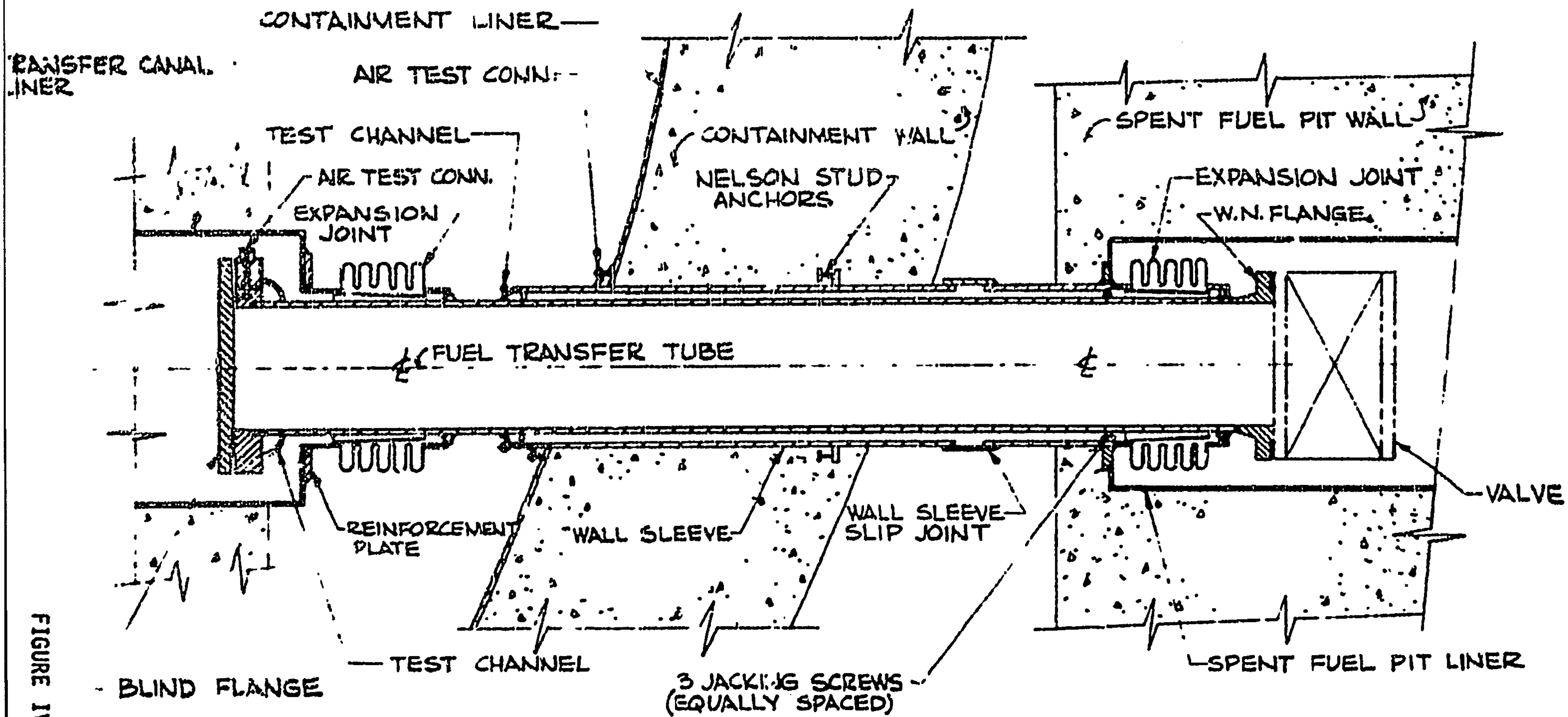
PIPING PENETRATION
COLD PIPE



PIPE PENETRATION
PURGE PIPING

E 37292

FIGURE IV - 2.2-A



FUEL TRANSFER TUBE PENETRATION
(CONCEPTUAL DRAWING)

FIGURE IV - 2.4-A

V. FABRICATION, ERECTION AND TESTING

1.0 GENERAL

A specification package for the containment building liner was prepared by United Engineers & Constructors, Inc., subcontractor to Westinghouse to cover the furnishing of all tools, equipment, material, labor, and supervision necessary to detail, fabricate, deliver, unload, test and erect all material required for the leak proof liner complete with all appurtenances, for the containment building in conformance with the applicable engineering drawings, including design, fabrication and erection of structural supports for placing concrete on the dome.

The specifications define the scope of work, codes, materials, fabrication, erection, inspection and testing procedures as well as the sequence of erection to be followed. They also delineate the responsibilities of the Chicago Bridge & Iron Co., subcontractor to U.E. for this portion of the work.

2.0 CHICAGO BRIDGE & IRON CO. SHOP FABRICATION

Complete facilities for shop fabricating all components were provided by CB&I's eastern manufacturing plant located at Greenville, Mercer County, Pennsylvania and New Castle, New Castle County, Delaware. All portions of the liner were fabricated in accordance with materials as outlined in the specifications. Mill Test Reports of the materials which are on file at U.E. and Westinghouse were reviewed by U.E. and Westinghouse, and a representative from the AEC, to assure their compliance with the specifications. All shop fabrication was done in accordance with the best shop practices as regards edge treatment, alignment and general workmanship. All vertical, flat and dome plates were single vee grooved for erection welding in accordance with the details shown on the drawings. All pieces were fabricated in sections as large as practical for field erection to reduce the number of field welds.

All penetration sleeves to which attachments are made were prepared for welding and fabricated in accordance with the drawing details. The equipment hatch was completely shop fabricated, assembled and tested before shipment. The personnel lock was fabricated, tested and shipped in accordance with the drawing details and specifications. The surface of all shell and dome plates were pickled. No painting was required on exterior surface of plates. No heating or hammering of plates was allowed for straightening purposes; only pressing or other methods, which would prevent damage to the plates, was permitted.

3.0 FIELD ERECTION

3.1 General

The various components of the liner were erected in accordance with the specifications. First, the knuckle was installed, then the bottom plates and walls were started simultaneously. The bottom liner including the sumps were completed first then the walls and dome were completed. Channels to permit pressurization and leak checking were installed and the entire bottom plate assembly (excepting sumps) was pressurized and tested. After the installation of the knuckle plate ring, which makes the transition from the bottom plate to the cylindrical portion of the liner, the three foot concrete floor was poured over the bottom liner plate. The balance of the cylindrical portion was erected in rings nine feet in height to the spring line level. Erection continued on the dome until completion.

All penetration sleeves were installed with their associated pressure channels and they, too, were then tested. Brackets for lights, crane conductors, etc., were also installed.

All lugs attached to the liner plates for erection purposes, were removed by chipping and projections were ground smooth.

Placement of channels for pressurizing and testing of the cylinder and dome welds proceeded by zones with each zone being individually tested for leaks.

During erection, adequate temporary bracing such as struts and similar items were provided in order to assure that the proper round shape of the structure was maintained. Out of roundness and plumbness tolerances of the structure were closely controlled in order to keep deviations within specified allowances.

Welding of the liner was done in accordance with the welding requirements of the ASME Boiler and Pressure Vessel Code. At the present time, the erection of the liner has been completed excepting for two openings left in place and required for further construction and equipment installation.

3.2 Liner Bulge

During erection an inward bulge, somewhat beyond the tolerances allowed by the specifications, occurred in a portion of the liner in the refueling transfer pipe. The bulge was corrected by the use of hydraulic jacks which were placed at various points on the plate. By this mechanism, the bulge was reduced to within allowable tolerances. More details are presented in Appendix A.

4.0 INSPECTION AND TESTING

4.1 Weld Radiography

For the liner complete radiograph is made of the first 10 feet of full penetration weld made by each welder or welding operation. A minimum of a 12" "spot" radiograph is made on every 50 feet of weld thereafter on the side walls and dome except where back-up plates are used. Any defects shown by "spot" radiographs require radiographing of adjacent "spots" and any resulting defects require that all of the welding performed by the welder be 100% radiographed to determine the end of defect. In accordance with ASME Code, the radiograph films are given to the contractor for his review.

All defects shown by radiographing are removed, repaired and inspected in accordance with the requirements for the original welds. In locations where radiography is not possible, such as the lower courses of shell plates where back-up plates are used, the liner fabricator welds on a 2" long overrun coupon. The overrun is chipped off, marked for location and given to the contractor for testing.

4.2 Construction Tests

During erection of the liner the following inspection and tests were performed.

4.2.1 On Bottom Liner Plate

1. Coupon Testing
2. Vacuum Box Test
3. Strength Test
4. Leak Test

4.2.2 On Cylindrical Walls and Dome

1. Spot Radiograph
2. Vacuum Box Test
3. Strength Test
4. Leak Test

4.2.3 On Penetrations

1. Leak Test
2. Other Leak and Strength Tests of Penetration. Internals to be performed by others in the future.

4.3

After completion of the entire containment structure when locks, hatches and all electrical and piping penetrations are in place, a post-construction leak rate test will be conducted as well as a 115% overpressure test.

VI. SUMMARY AND CONCLUSION

A review of the preceeding Sections I through V indicates the following:

a) Containment criteria applicable to the liner was developed and is reported in this document.

b) A Favorable review of the adequacy of the criteria has been made by N. M. Newmark and W. J. Hall and has been reported in Reference 14.

c) This report delineates the design analysis that was prepared to demonstrate the compliance with this criteria.

d) This report further shows the completeness of the analysis as proven by the loading diagrams (Section II), stress analysis and results (Section III), design of penetrations (Sections IV) and fabrication, erection and testing procedures that were followed.

Thus, in the foregoing it is shown that the liner meets the criterion of maintaining stresses below 0.95 specified minimum yield at normal operating temperature and the criterion for buckling.

Therefore, it can be concluded that the IPP-2 Containment Liner fulfills the requirements necessary to provide a safe vapor barrier for the plant.

REFERENCES

1. Preliminary Safety Analysis Report - Consolidated Edison Company of New York, Inc. - Indian Point Nuclear Generating Unit No. 2.

2. Nuclear Reactors and Earthquakes - TID-7024 - United States Atomic Energy Commission - 1963.

3. Design of Multistory Reinforced Concrete Building for Earthquake Motions - J. Blume, N. Newmark, L. Corning - Portland Cement Association - 1961.

4. Structural Analysis and Proportioning of Members-Ultimate Strength Design - ACI 318-63 - Part IV-B.

5. Theory of Elastic Stability - S. Tomoshenko & Gere.

6. Fatigue Tests of Plates and Beams with Stud Shear Connectors - J. E. Stellmeyer, W. H. Munse, E. A. Selby - Highway Research Record No. 76.

7. The Growth of Stud Welding - R. C. Singleton - Welding Engineer - July, 1963.

8. Theory of Plates and Shells - S. Timoshenko, S. Warnowsky - Krieger 2nd Ed. McGraw-Hill Book Co. 1959.

9. Preliminary Design of a Stiffened Liner Plate Dome Indian Point No. 2 - Project Calculation Brief - General Analytics, Inc., 1960.

10. Preliminary Safety Analysis Report - Pacific Gas and Electric Co., Nuclear Plant Diablo Canyon Site, Supplement No. 4.

11. Stress Concentrations in Concrete - Nature Volume 203 - W. Wright, and J. G. Byrne - September 26, 1964.

12. Theory of Elasticity - S. Timoshenko and Godier, 2nd Ed. McGraw Hill Book Co., 1961.

13. Structural Engineers Association of California (SEAOC) Code.

14. Report to AEC Regulatory Staff on Adequacy of Structural Criteria of IPP-2 - Appendix E - N. M. Newmark and W. J. Hall.

APPENDIX A

REPORT ON THE

CONTAINMENT BUILDING LINER

PLATE BUCKLE

IN THE VICINITY OF THE

FUEL TRANSFER CANAL

INDIAN POINT GENERATING STATION

UNIT NO. 2

JANUARY, 1968

PREFACE

This summary has been prepared in order to accumulate in one place the background and corrective measures taken as related to the containment building liner bulge in the vicinity of the fuel transfer canal penetration at the Indian Point Generating Station, Unit No. 2.

Incorporated into this summary is a brief description of the containment building, the tests and test procedures used during construction and after repairs were made, conclusions, the measurements made and their evaluation, and a description of the corrective actions taken.

INTRODUCTION

During construction and erection of the welded steel liner on the Indian Point No. 2 containment structure a buckle or liner deformation was observed in the vicinity of the fuel transfer tube canal penetration. This report is aimed at identifying the resulting deformations, assessment of the problem as related to liner integrity and the remedial action taken to assure functional adequacy.

DESCRIPTION OF STRUCTURE

The reactor containment structure is a reinforced concrete vertical right cylinder with a flat base and a hemispherical dome. A welded steel liner with a minimum thickness of 1/4" is attached to the inside face of the concrete shell to assure a high degree of leak tightness. The liner is anchored to the concrete shell by means of anchors so that it forms an integral part of the entire composite structure under all loadings.

The plate steel liner is carbon steel conforming to ASTM Designation A442-65 "Standard Specification for Carbon Steel Plates with Improved Transition Properties," Grade 60. This steel has a minimum yield strength of 32,000 psi and a minimum tensile strength of 60,000 psi with an elongation of 22 per cent in an 8-in. gauge length at failure. The liner is 1/4-in. thick at the bottom, 1/2-in. thick in the first three courses except 3/8-in. thick for remaining portion of the cylindrical walls and 1/2-in. thick in the dome. The liner material was tested to assure an NDT temperature more than 30° F lower than the minimum operating temperature of the liner material.

Impact testing was done in accordance with Section N331 of Section III of the ASME Boiler & Pressure Vessel Code. A 100 per cent visual inspection of liner anchors was made prior to pouring concrete.

TESTS AND PROCEDURES

Qualification of welding procedures and welders on the containment liner is in accordance with Section IX, "Welding Qualifications of the ASME Boiler & Pressure Vessel Code." All welded joints in the liner have steel channels welded over them from the inside of the vessel. During construction, the channel welds were tested by means of pressurizing sections with Freon gas and checking for leaks by means of a Freon snifter. A strength test was performed on the liner plate weld and the channel weld by pressurizing the channel with air at 54 psig for 15 minutes. In addition, each zone of channel cover weld will be leak tested, using Freon-air mixture at 47 psig.

Following repair of the liner bulge, leak tests were performed on the channels in the affected area in accordance with the above test procedures. Magnetic particle inspection was made of all weld channel fillet welds in the same area. The magnetic particle test is not required by the specification but was utilized in this special situation as an additional feature to insure that the integrity of the liner was preserved. The weld channel system passed the leak and magnetic particle tests.

CONCLUSION

As shown by the preceding discussion and the following details, it is concluded that the integrity of the liner has not been violated.

Since the liner material (A-442) is highly ductile, and the liner is deflection limited by the reinforced concrete wall, strains in the liner will remain elastic and the leakproof integrity of the liner would be maintained under all anticipated conditions.

Technical data reports and quality control records referred to in the remainder [illegible]is report are available and on file in the UE&C field offices for revie[illegible] [illegible]ired.

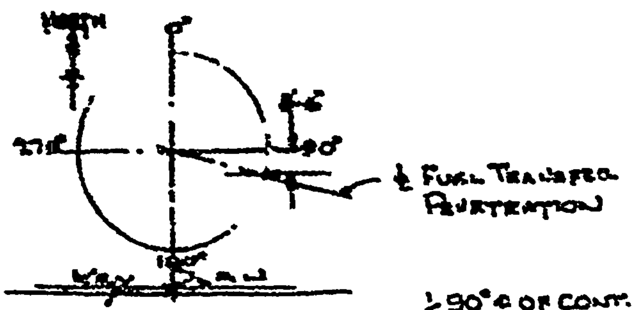
MEASUREMENTS AND EVALUATION

During a routine inspection of the Vapor Containment liner by Quality Control personnel on August 4, 1967, a buckle was noted in the liner plate near the fuel transfer tube between elevations 56'-7" and 59'-7". Chicago Bridge & Iron's drawing No. 5 designates these plates as 3H and 3J. Figure 1 shows the location of the plates.

Field measurements were taken at the buckled zone (elevations 56'-7" and 59'-7"). The measurements show the distance the liner had buckled from a normal position at various stations. This information is shown on the attached Figures 2 and 3. Figure 4 is a plot of this data in the form of a contour map of the buckled zone.

Specification 9321-01-225-3, Containment Building Liner, allows 2" tolerance for local buckling. From the above information, it is noted that points C and D at elevation 56'-7" were beyond the acceptable limits by 1/8" and 5/16", respectively.

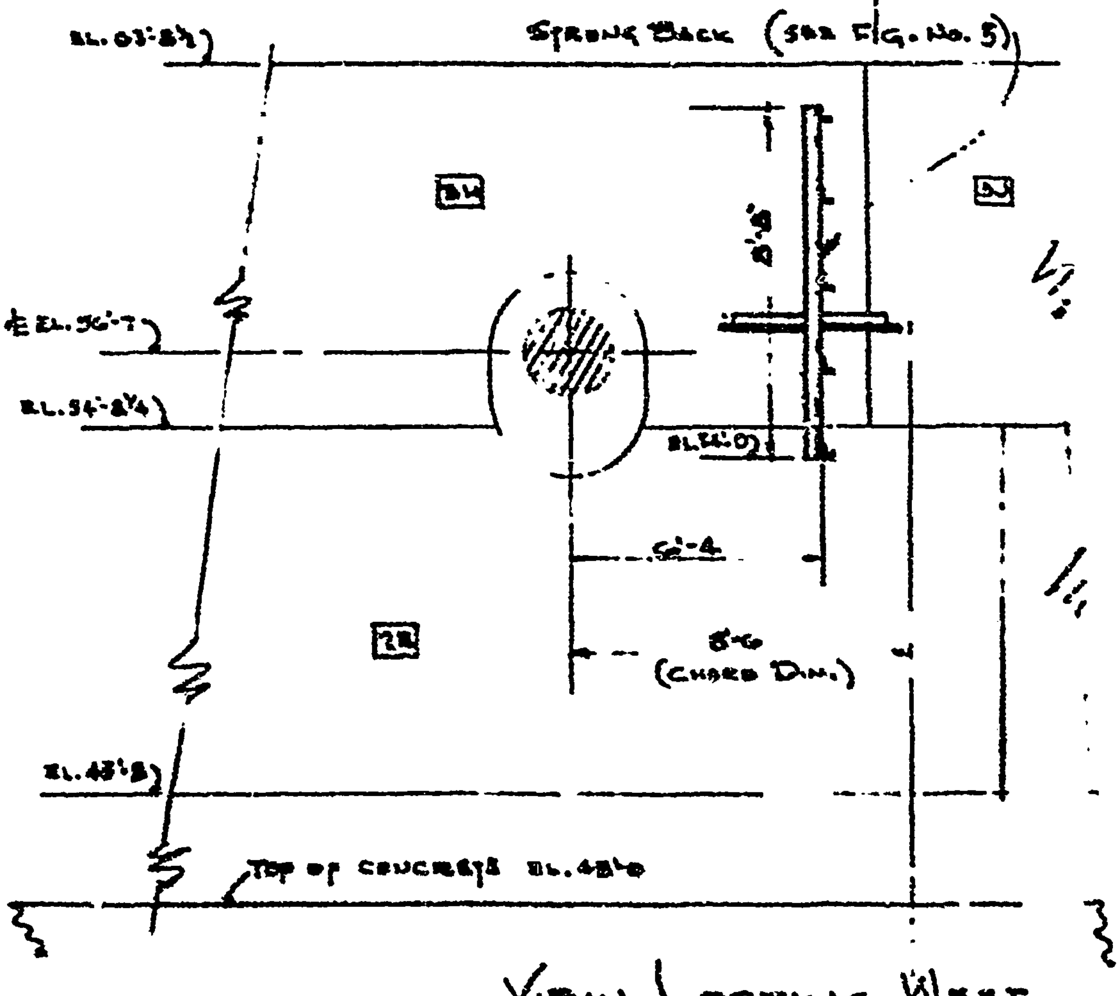
FIGURE 1



Notes;

a) [3H] etc. denotes CB&I Pl. No. per CB&I Dwg. No. 5.

b) For location of seven (7) additional studs see Fig. No 6.

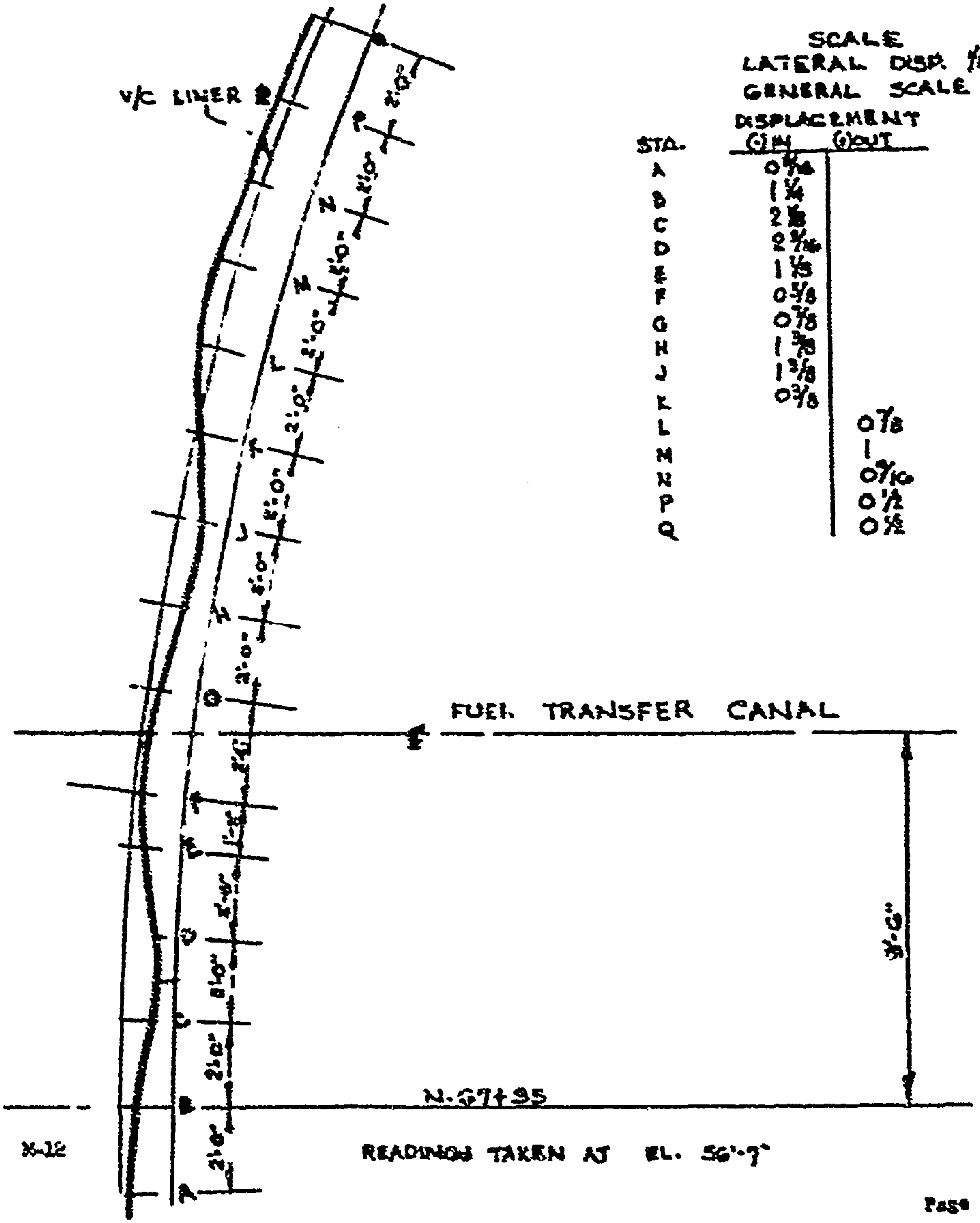


View Looking West

Scale 1/4" = 1'-0"

H-11

FIGURE 2



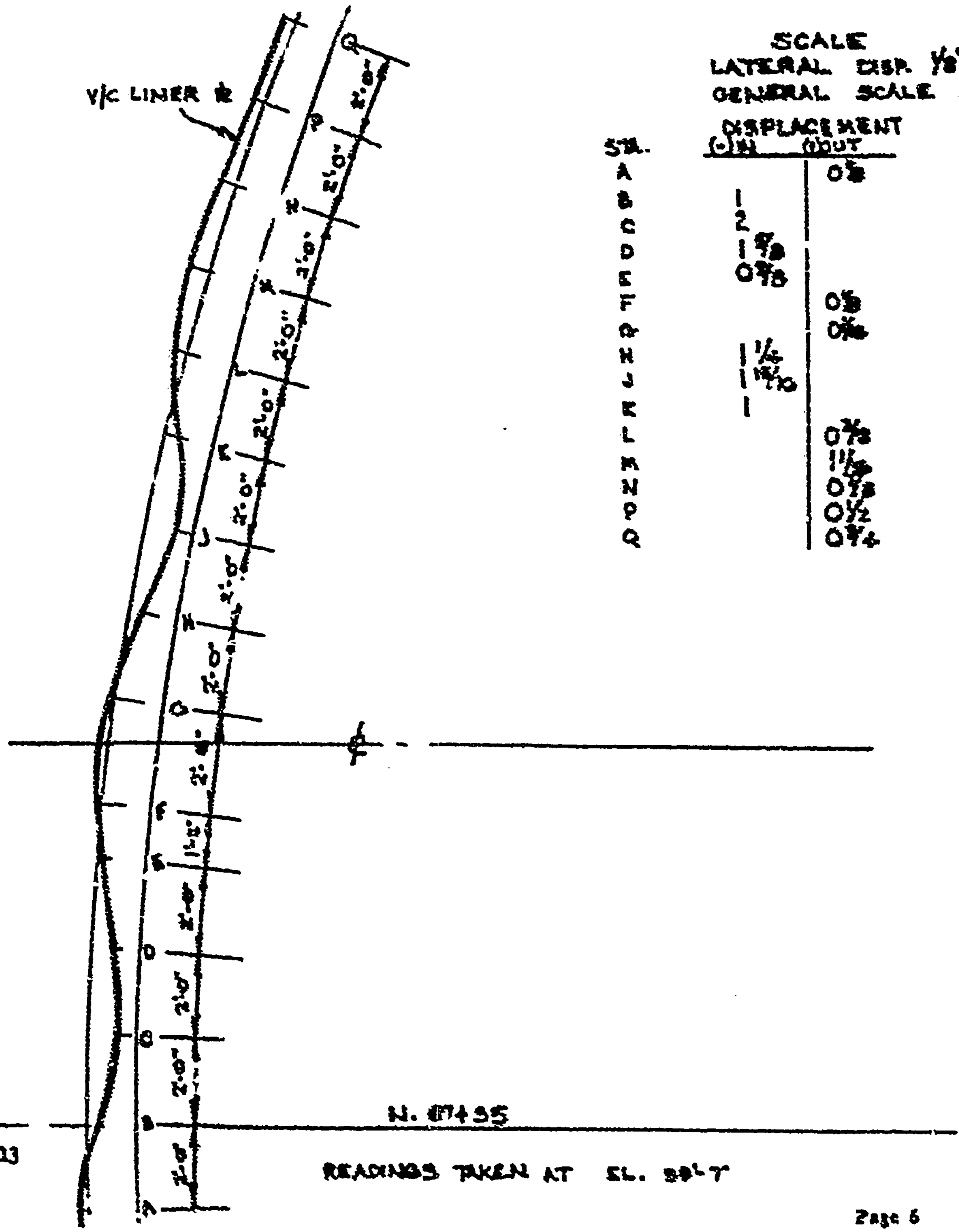
SCALE
LATERAL DISP. 1/8" = 1"
GENERAL SCALE 1" = 5'

STA.	DISPLACEMENT (-) IN	DISPLACEMENT (+) OUT
A	0 3/16	
B	1 1/4	
C	2 1/8	
D	2 5/16	
E	1 1/8	
F	0 5/8	
G	0 7/8	
H	1 3/8	
J	1 3/8	
K	0 3/8	
L		0 7/8
M		1
N		0 9/16
P		0 1/2
Q		0 1/2

X-12

READINGS TAKEN AT EL. 56'-7"

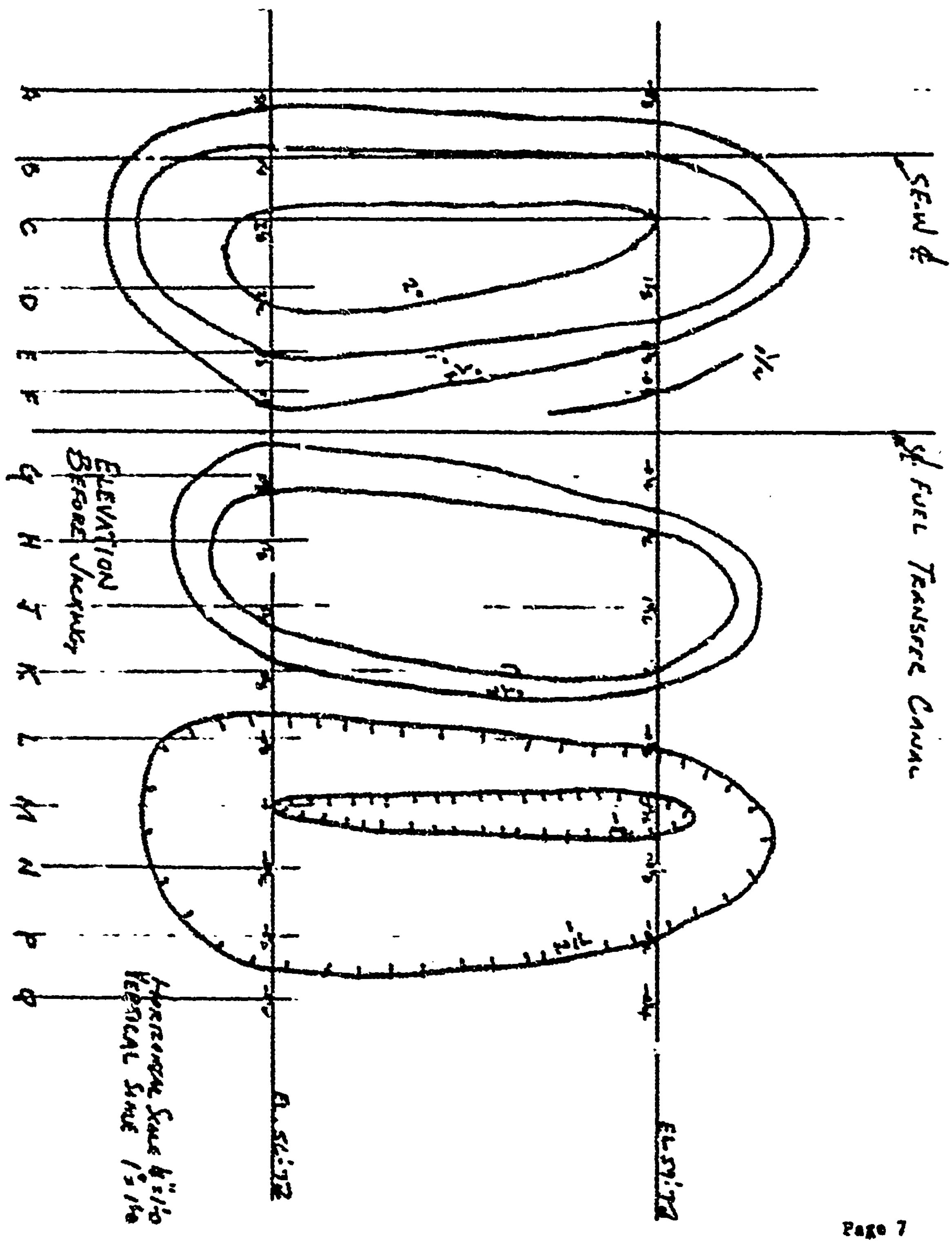
FIGURE 3



SCALE
LATERAL DISP. 1/8" = 1"
GENERAL SCALE 1" = 3'

STA.	DISPLACEMENT (-) IN	DISPLACEMENT (+) OUT
A		0 3/8
B	1	
C	2	
D	1 7/8	
E	0 7/8	
F		0 3/8
G		0 5/16
H	1 1/4	
J	1 15/16	
K	1	
L		0 7/8
M		1 1/16
N		0 7/8
P		0 1/2
Q		0 3/4

FIGURE 4



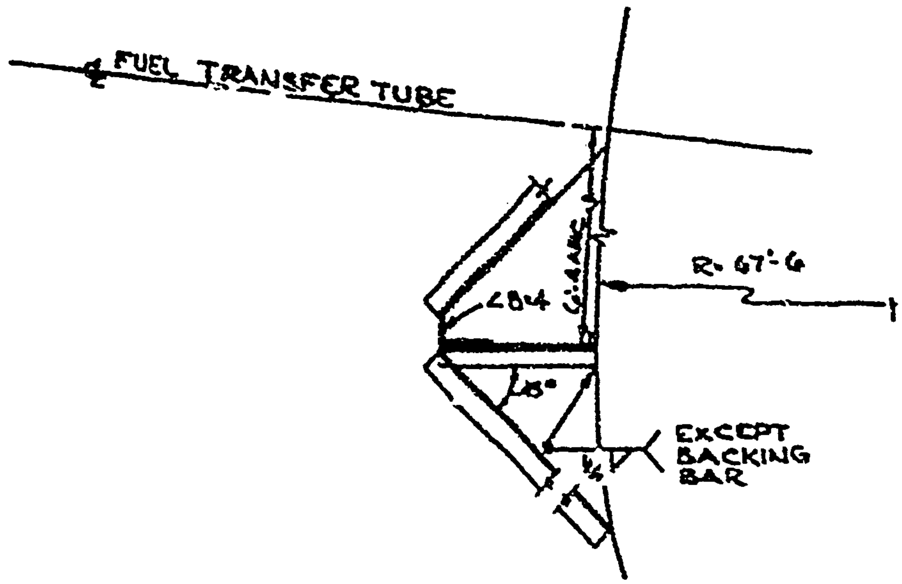
CORRECTIVE ACTION

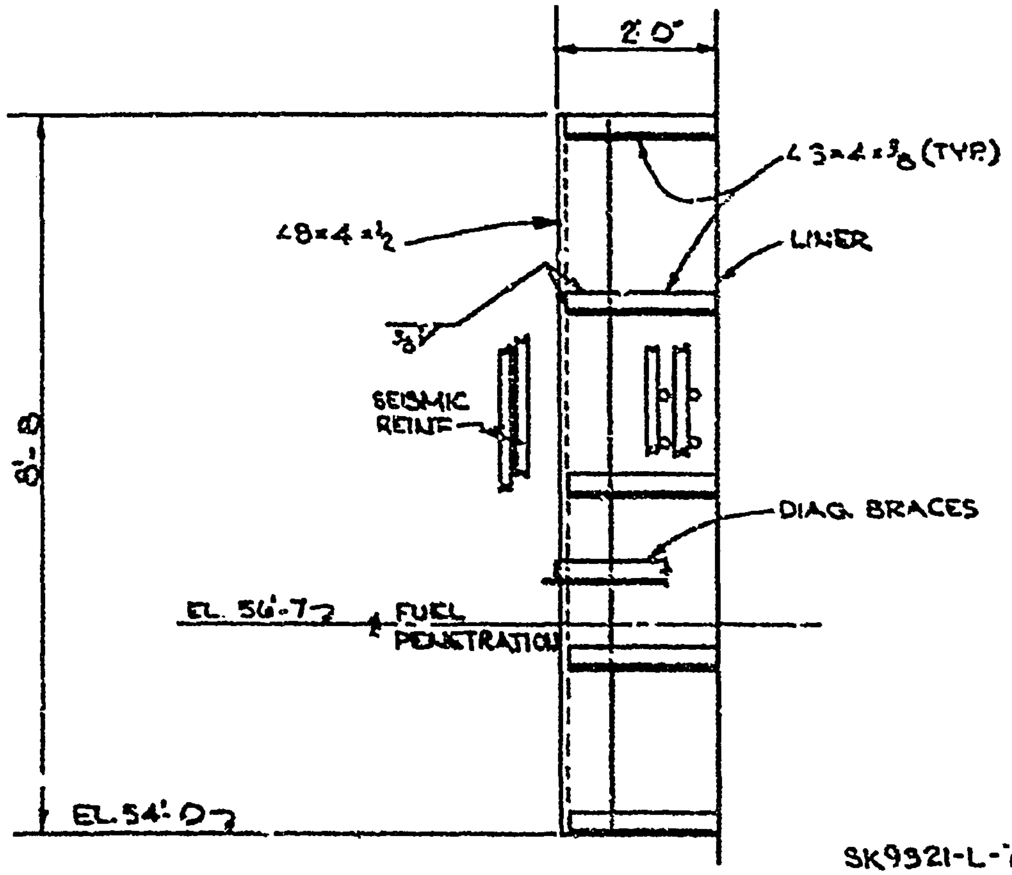
Various approaches, independent of support from exterior concrete, were examined. The adopted solution was a system consisting of a combination of a strongback and stud anchors, as illustrated in Figures 5 and 6. The liner was jacked to within tolerances prior to strongback installation.

The stud anchors were installed at the high points of the buckles on a judgment basis, in order to prevent additional buckling of these points under compressive loadings by holding the liner to the concrete.

Figures 7, 8 and 9 show the liner configuration after corrective action was taken. Figure 10 is a contour map of the buckled zone after corrections and in addition, all of the studs including the original installation and the additional studs are shown. At all points the liner is within the allowable tolerances.

FIGURE 5





SK9321-L-7677

FIGURE 6

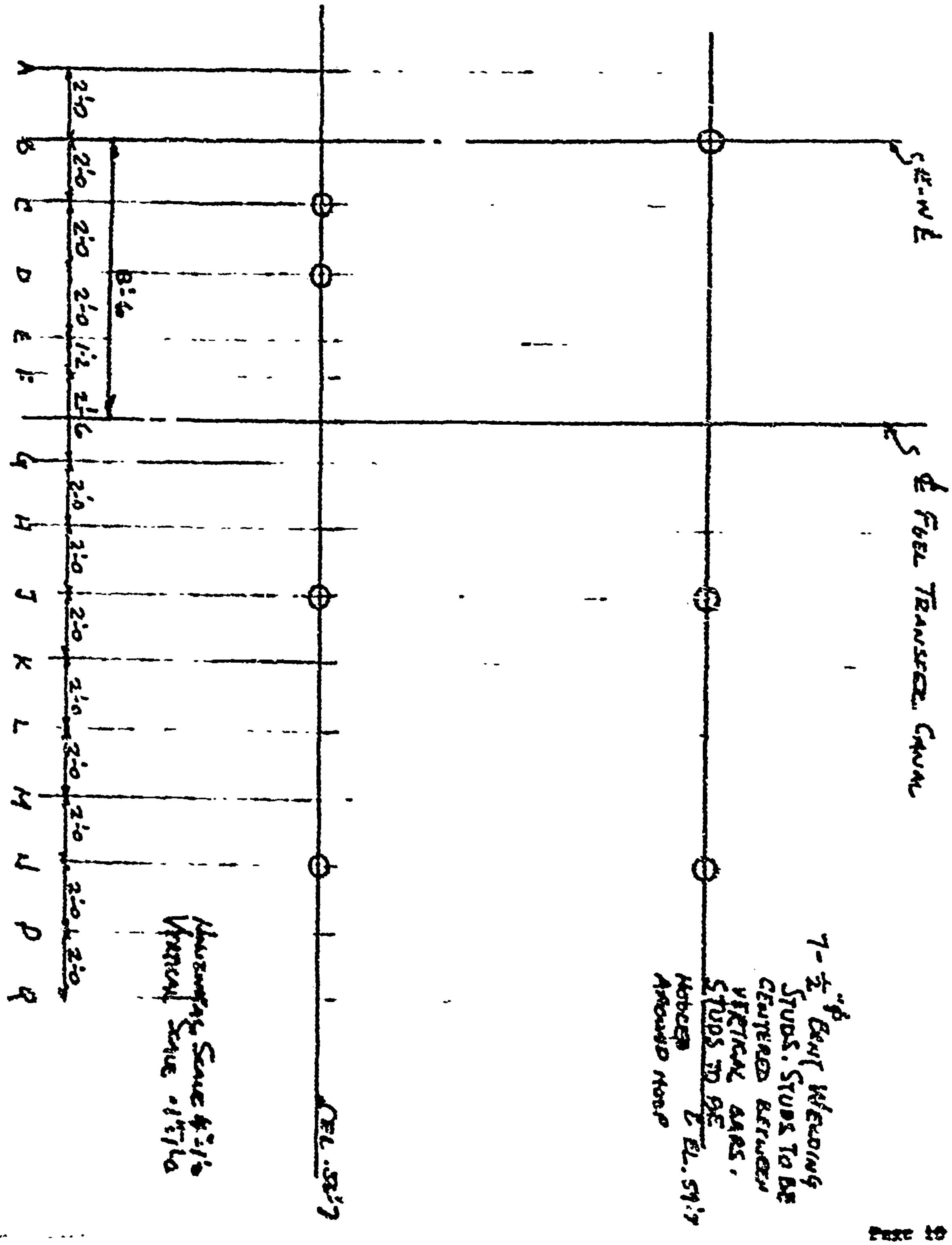
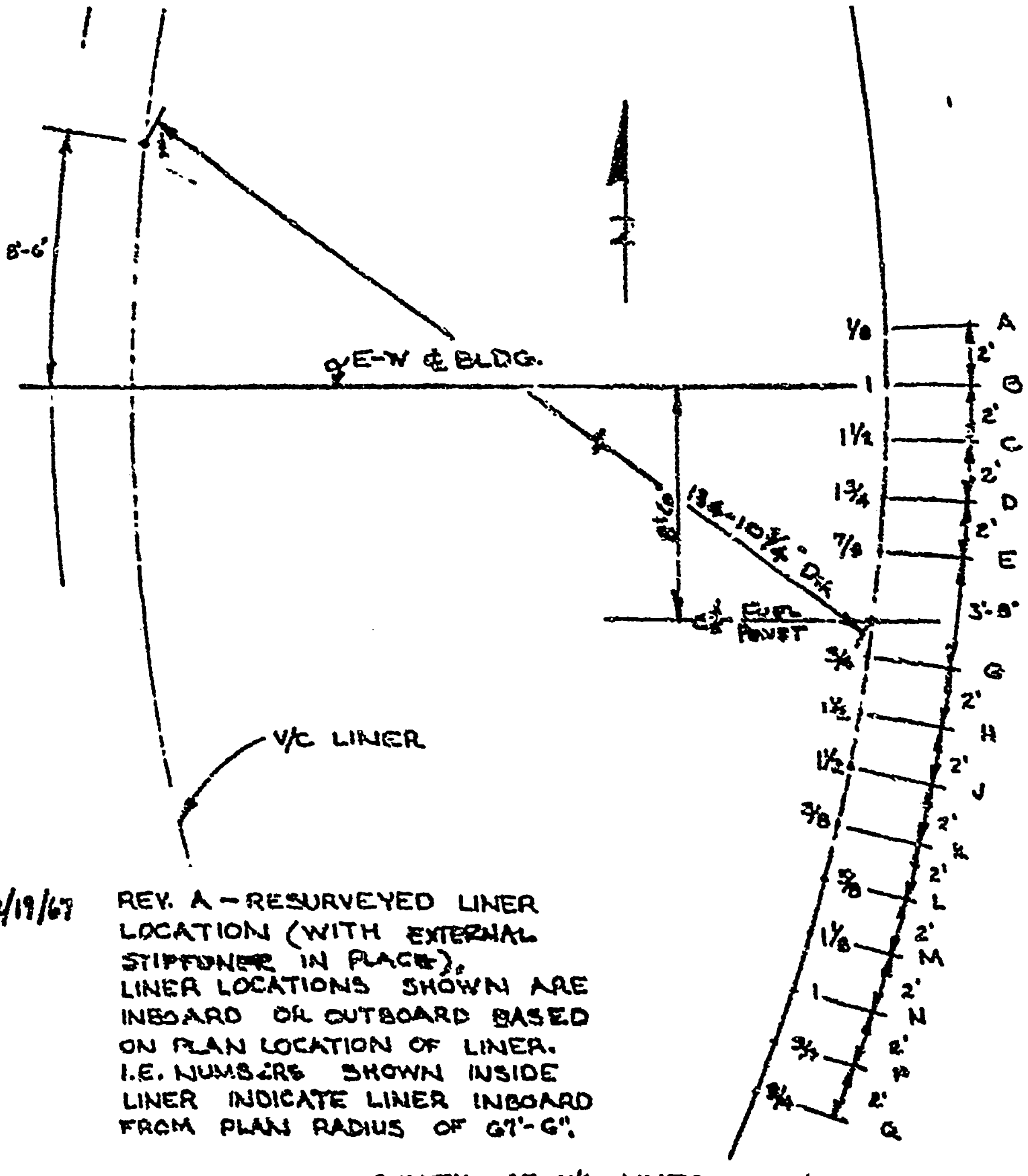

A
B
C
D
E
F
G
H
J
K
L
M
N
P
Q
2'0
8'6
℄ FUEL TRANSFER CANAL
7-½" ϕ BENT WELDING STUDS. STUDS TO BE CENTERED BETWEEN VERTICAL BARS. STUDS TO BE HOOKED AROUND HOOP
℄ EL. 59'7
℄ EL. 53'7

FIGURE 7



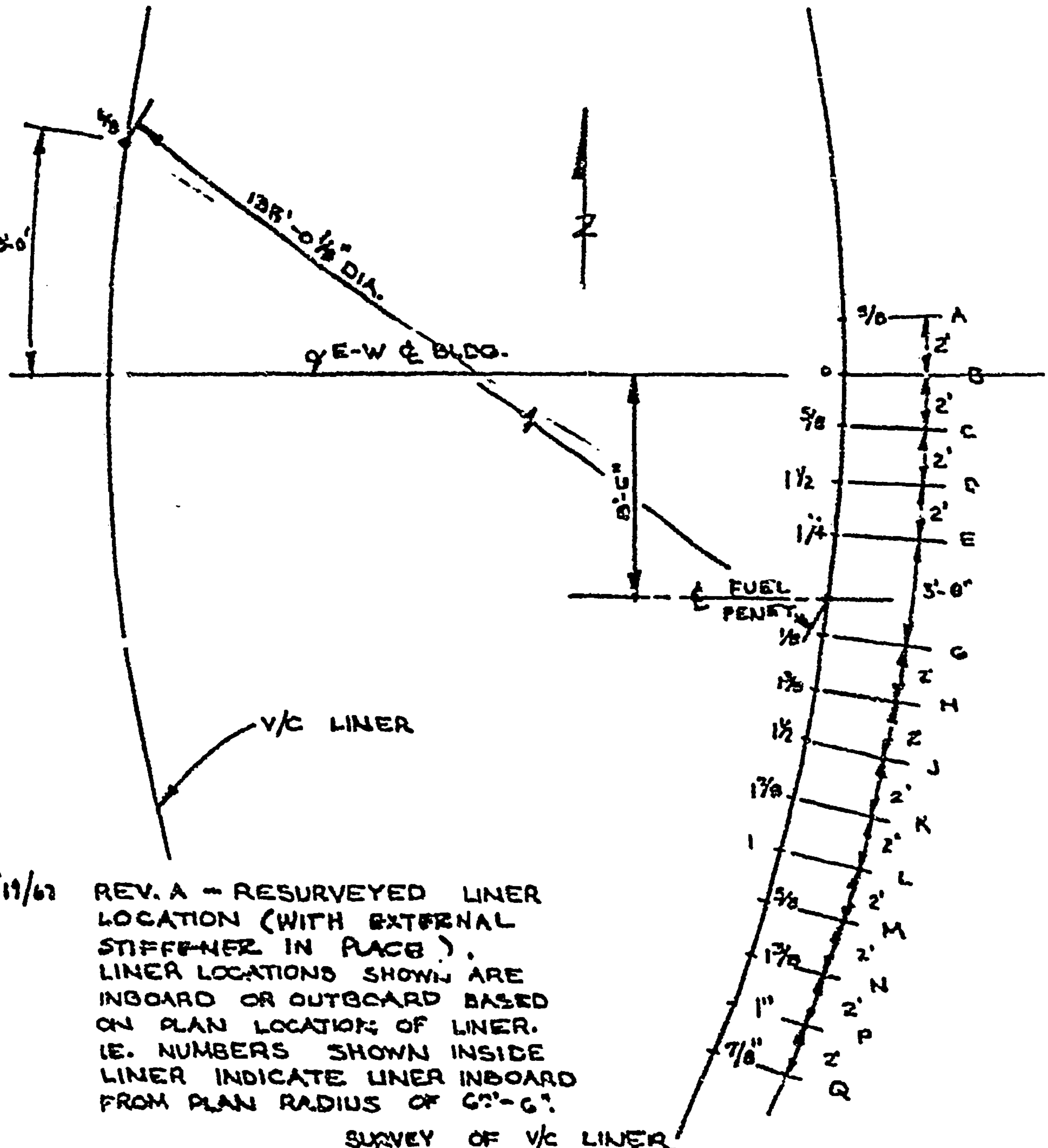
12/19/67 REV. A – RESURVEYED LINER LOCATION (WITH EXTERNAL STIFFENER IN PLACE). LINER LOCATIONS SHOWN ARE INBOARD OR OUTBOARD BASED ON PLAN LOCATION OF LINER. I.E. NUMBERS SHOWN INSIDE LINER INDICATE LINER INBOARD FROM PLAN RADIUS OF 67'-6".

SURVEY OF V/C LINER

EL. 56'-7"

H-18

FIGURE 8

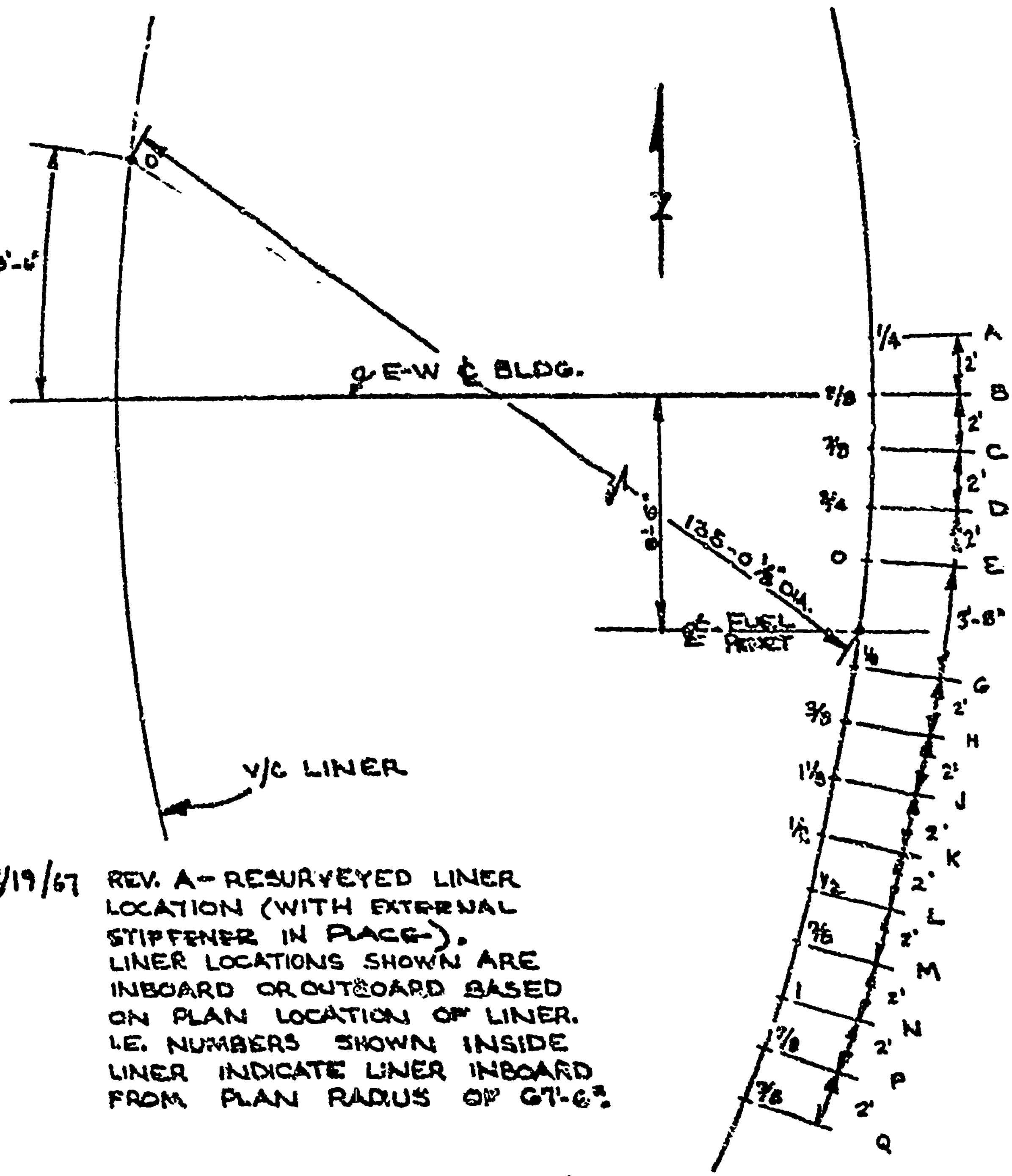


12/19/67 REV. A – RESURVEYED LINER LOCATION (WITH EXTERNAL STIFFENER IN PLACE). LINER LOCATIONS SHOWN ARE INBOARD OR OUTBOARD BASED ON PLAN LOCATION OF LINER. IE. NUMBERS SHOWN INSIDE LINER INDICATE LINER INBOARD FROM PLAN RADIUS OF 69'-6".

SURVEY OF V/C LINER

EL. 59'-7"

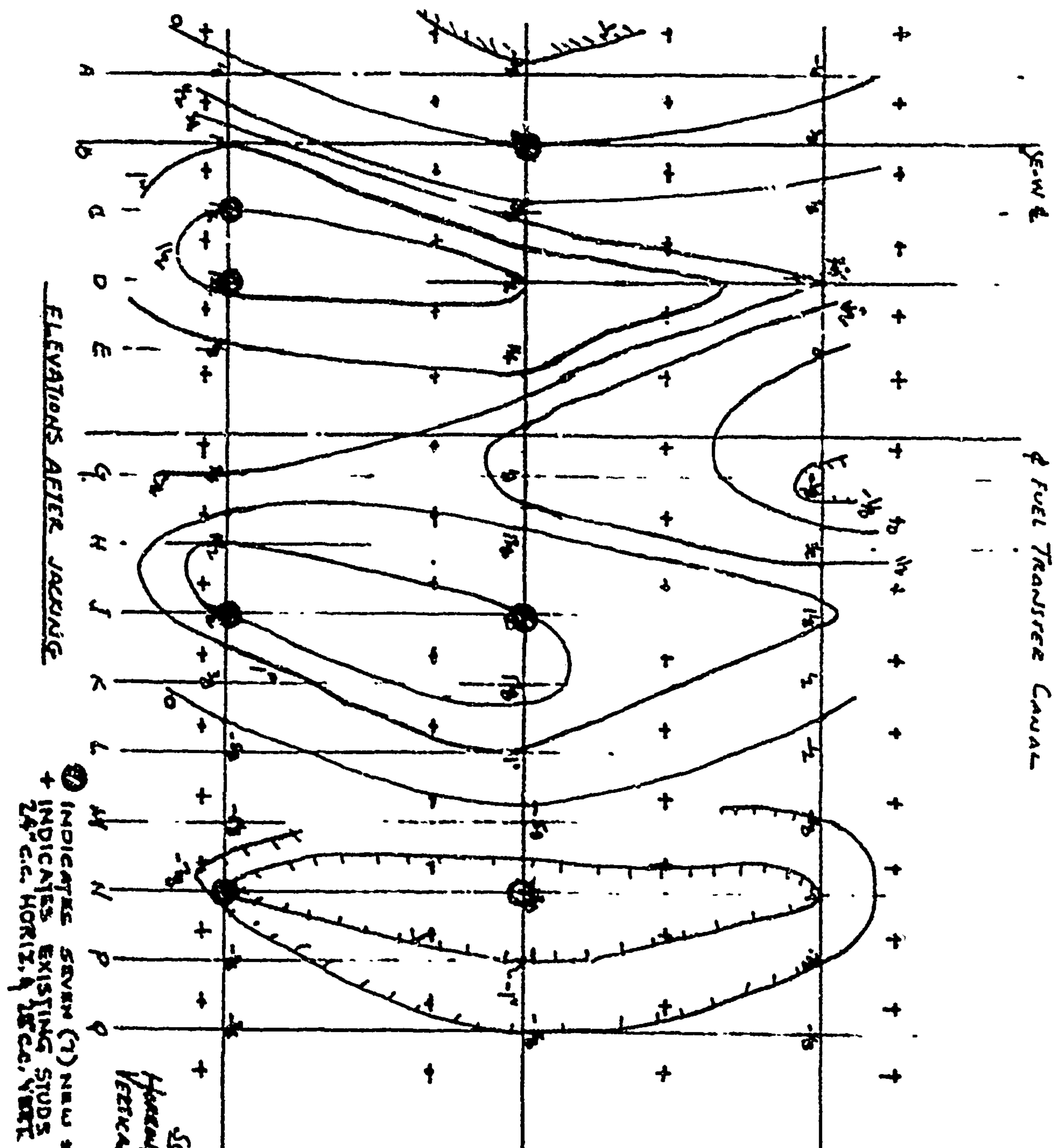
FIGURE 9



SURVEY OF V/C LINER

EL. 62'-7"

FIGURE 10


ELEVATIONS AFTER JACKING
℄ FUEL TRANSFER CANAL
INDICATES SEVEN (7) NEW STUDS
+ INDICATES EXISTING STUDS ON 24" C.C. HORIZ. & 18" C.C. VERT.
SCALE
HORIZONTAL 1/4"=1'0
VERTICAL 1/2"=1'0
H-21

9

M, Q1.1-1 to Q6.3-32

10623

10623

Received w/Ltr Dated 9-12-69

U.S. Atomic Energy Commission

Docket No. 50-247
Exhibit B-8 Volume 5

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

FINAL FACILITY DESCRIPTION AND SAFETY ANALYSIS REPORT

3247

Acre 1-7-70

Regulatory File Cy.

Received w. Ltr Dated 8-28-70

PREFACE TO SUPPLEMENT NO. 14

Supplement 14 to the Indian Point Unit No. 2 Final Safety Analysis Report consists of corrections and additional information for the report in the form of page changes. The revised pages should be added to the Report as listed below. This preface page should be provided in Volume 5 of the Report immediately following the preface to Supplement 13.

Two figures included with this Supplement were inadvertently omitted from the original Report. They are being added at this time to enhance the understanding of the evolution of the Westinghouse organization. The present Westinghouse Nuclear Energy System organization including the WEDCO organization is described in Volume 6 of the Report following Tab III.

INSTRUCTIONS FOR SUPPLEMENT 14 PAGE CHANGES

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
5.1.4-7/5.1.4-8	5.1.4-7/5.1.4-8
6.2-25b/6.2-26	6.2-25b/6.2-26
6.6-9/6.6-10	6.6-9/6.6-10
6.6-11/Table 6.6-1	6.6-11/Table 6.6-1
9.2-21/9.2-22	9.2-21/9.2-22
9.3-11/9.3-12	9.3-11/9.3-12
9.6-5/Table 9.6-1	9.6-5/Table 9.6-1
11.2-21/11.2-22	11.2-21/11.2-22
Appendix B (Vol. 4)	Appendix B (Vol. 4)
B-30/B-31	-----/-----
Q 1.3-13/Q 1.3-14	Q 1.3-13/Table 1.3-1
Table 1.3-1/Tables 1.3-2, 3, 4	-----/-----
Q 1.9-17/Q 1.9-18	Q 1.9-17/blank
Q 1.9-19/blank	-----/-----
Q 8.1(a)-1/blank	Q 8.1(a)-1/blank
Q 9.5-1/Figure Q 9.5-1	Q 9.5-1/Figure Q 9.5-1

Remove	Insert
Q 12.5, radiation contingency plan	Q 12.5, radiation contingency plan
p 8/p 9	p 8/p 9
p 32/p 33	p 32/p 33
Figure 3/blank	Figure 3/blank
Figure 13.4.1-3/blank	Figure 13.4.1-3/blank
Q 13.4(2)-61/Q 13.4(2)-62	Q 13.4(2)-61/Q 13.4(2)-62
Q 13.4(2)-91/Q 13.4(2)-92	Q 13.4(2)-91/Q 13.4(2)-92
Q 13.4(2)-93/Q 13.4(2)-94	Q 13.4(2)-93/Q 13.4(2)-94
Q 13.4(2)-95/Q 13.4(2)-96	Q 13.4(2)-95/Q 13.4(2)-96
Q 13.4(2)-97/Q 13.4(2)-98	Q 13.4(2)-97/Q 13.4(2)-98
Q 13.4(2)-99/blank	Q 13.4(2)-99/Q 13.4(2)-100
-----/-----	Q 13.4(2)-101/Q 13.4(2)-102
-----/-----	Q 13.4(2)-103/Q 13.4(2)-104
-----/-----	Q 13.4(2)-105/blank

Supplement 14
8/70

dorn 6-22-70

Regulatory File Cy.

PREFACE TO SUPPLEMENT NO. 9

Received w/Ltr Dated 6-3-70

Supplement 9 to the Indian Point Unit No. 2 Final Safety Analysis Report consists of corrections and additional information for the Report in the form of page changes. The revised pages should be added to the Report as listed below. This preface page should be inserted in Volume 5 of the Report immediately following the preface to Supplement 8.

INSTRUCTIONS FOR SUPPLEMENT NINE PAGE CHANGES

Insert Revised Sheet Front/Back	Discard Old Sheet Front/Back
1.5-3/1.5-4	1.5-3/Blank
2.4-1/2.4-2	2.4-1/Blank
3.2.3-15/3.2.3-16	3.2.3-15/3.2.3-16
3.3-1/3.3-2	3.3-1/3.3-2
N-3/N-4	N-3/N-4
Table 7.2-1/Table 7.2-1 cont'd	Table 7.2-1/Table 7.2-1 cont'd
Table 7.2-2/Table 7.2-3	Table 7.2-2/Table 7.2-3
8.2-11/8.2-12	8.2-11/8.2-12
8.2-13/8.2-14	8.2-13/8.2-14
Table 9.5-1/Blank	Table 9.5-1/Blank
Fig. A.1-1/Fig. A.1-2	
Q 1.1-1/Blank	Q 1.1-1/Blank
Q 1.3-1/Q 1.3-2	Q 1.3-1/Q 1.3-2
Q 1.9-15/Q 1.9-16	Q 1.9-15/Blank
Q 5.11(b)-1/Blank	
Q 5.16.1-1/Fig. 9.16 Sheet 1	Q 5.16.1-1/Fig. 9.16 Sheet 1
Q 6.3-17/Q 6.3-18	Q 6.3-17/ Q 6.3-18
Q 6.3-19/Q 6.3-20	Q 6.3-19/Q 6.3-20
Q 6.3-21/Q 6.3-22	Q 6.3-21/Q 6.3-22
Q 6.3-23/Q 6.3-24	Q 6.3-23/Q 6.3-24
Q 6.3-25/Q 6.3-26	Q 6.3-25/Q 6.3-26
Q 6.3-27/Q 6.3-28	Q 6.3-27/Blank
Q 6.3-29/Q 6.3-30	
Q 6.3-31/Q 6.3-32	
Q 6.3-33/Q 6.3-34	
Fig. Q 6.3-9/Fig. Q 6.3-10	Fig. Q 6.3-9/Blank
Fig. Q 6.3-11/Blank	
Q 7.12-1/Blank	Q 7.12-1/Blank
Q 9.5-1/Fig. 9.5-1	Q 9.5-1/Fig. 9.5-1
Q 14.6-1/Q 14.6-2	Q 14.6-1/Q 14.6-2
Q 14.6-3/Q 14.6-4	
Q 14.6-5/Q 14.6-6	
Q 14.6-7/Q 14.6-8	
Q 14.6-9/Q 14.6-10	
Q 14.6-11/ Q 14.6-12	

Insert Revised Sheet	Discard Old Sheet
2.0-2/2.0-3	2.0-2/2.0-3
3.0-1/3.0-2	3.0-1/3.0-2
3.0-3/3.0-4	3.0-3/3.0-4
3.0-5/3.0-6	3.0-5/3.0-6
3.0-7/3.0-8	3.0-7/3.0-8
3.0-9/3.0-10	3.0-9/3.0-10
3.0-11/3.0-12	3.0-11/3.0-12
3.0-13/3.0-14	3.0-13/3.0-14
3.0-19/3.0-20	3.0-19/3.0-20
3.0-21/3.0-22	3.0-21/3.0-22
3.0-23/3.0-24	3.0-23/3.0-24
3.0-25/3.0-26	3.0-25/3.0-26
3.0-29/3.0-30	3.0-29/3.0-30
3.0-31/3.0-32	3.0-31/3.0-32
3.0-33/3.0-34	3.0-33/3.0-34
3.0-35/3.0-36	3.0-35/3.0-36
3.3.10-7/Fig. 3.1	3.3.10-7/Fig. 3.1
R-3/R-4	R-3/R-4
R-15/R-16	R-15/R-16
R-17/R-18	R-17/R-18
R-19/R-20	R-19/R-20
R-83/R-84	R-83/R-84
R-85/R-86	R-85/R-86
R-87/R-88	R-87/R-88
R-89/R-90	R-89/R-90
R-91/R-92	R-91/R-92
R-93/R-94	R-93/R-94
R-95/R-96	R-95/R-96
R-97/R-98	R-97/R-98
R-99/R-100	R-99/R-100
R-101/R-102	R-101/R-102
R-103/R-104	R-103/R-104
R-105/R-106	R-105/R-106
R-107/R-108	R-107/R-108
R-109/R-110	R-109/R-110
R-111/R-112	R-111/R-112
R-113/R-114	R-113/R-114
R-115/R-116	R-115/R-116
R-117/R-118	R-117/R-118
R-119/R-120	R-119/R-120
R-121/R-122	R-121/R-122
R-123/R-124	R-123/R-124
R-125/R-126	R-125/R-126
R-127/R-128	R-127/R-128
R-128a/Blank	
R-131/R-132	R-131/R-132
4.0-9/4.0-10	4.0-9/4.0-10
4.0-11/Blank	4.0-11/Blank
5.0-1/5.0-2	5.0-1/5.0-2
5.0-3/5.0-4	5.0-3/5.0-4
5.0-7/Blank	5.0-7/Blank
6.0-5/6.0-6	6.0-5/6.0-6

dore 4-24-70

DOCKET NO. 50-247

Regulatory File Cy.

Received w/Ltr Dated 4-17-70

PREFACE TO SUPPLEMENT NO. 8

Supplement 8 to the Indian Point Unit No. 2 Final Safety Analysis Report consists of responses to questions from the Atomic Energy Commission and page changes to the Report. The questions were contained in a letter to Arthur N. Anderson, of Consolidated Edison Company of New York, from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Consolidated Edison Company of New York, from Peter A. Morris, dated November 13, 1969. The responses consist of questions and answers to be added to Volume 5 of the Report in the proper order behind Tab II.

The supplement responds to questions concerning Chapters 4, 6, 7, and 13 of the Report. This preface page should be inserted in Volume 5 of the Report immediately following the preface to Supplement No. 5.

Also included with this Supplement are Volume 6 binders. Tab III and Tab IV material should be transferred to Volume 6.

INSTRUCTIONS FOR SUPPLEMENT EIGHT PAGE CHANGES

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
8-i/8-ii	8-i/8-ii
Tbl. 4.5-1/Tbl. 4.5-1	Tbl. 4.5-1/Tbl. 4.5-1
Tbl. 4.5-1/Tbl. 4.5-2	Tbl. 4.5-1/Tbl. 4.5-2
5.1.7-2/5.1.8-1	5.1.7-2/5.1.8-1
5.1.8-4/5.1.8-5	5.1.8-4/5.1.8-5
5.1.8-6/5.1.8-7	5.1.8-6/5.1.8-7
6.2-13/6.2-14	6.2-13/6.2-14
6.2-15/6.2-16	6.2-15/6.2-16
6.2-21/6.2-22	6.2-21/6.2-22
6.2-27/6.2-28	6.2-27/6.2-28
6.2-31/6.2-32	6.2-31/6.2-32
6.2-33/6.2-34	6.2-33/6.2-34
N-5/N-6	N-5/N-6
N-11/N-12	N-11/N-12
N-19/N-20	N-19/N-20
N-21/N-22	N-21/N-22

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Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
6.4-5/6.4-6	6.4-5/6.4-6
6.4-7/6.4-8	6.4-7/6.4-8
6.4-27/6.4-28	6.4-27/6.4-28
6.6-1/6.6-2	6.6-1/6.6-2
6.6-9/6.6-10	6.6-9/6.6-10
6.6-11/Tbl. 6.6-1	6.6-11/Tbl. 6.6-1
7.2-33/7.2-34	7.2-33/7.2-34
Tbl. 7.2-1/Tbl. 7.2-1 (cont'd)	Tbl. 7.2-1/Tbl. 7.2-1 (cont'd)
Tbl. 7.2-1 (cont'd)/ Tbl. 7.2-1 (cont'd)	Tbl. 7.2-1 (cont'd)/ Tbl. 7.2-1 (cont'd)
Fig. 8.2-5 (Foldout)/Blank	Fig. 8.2-5 (Foldout)/Blank
D-6/D-7	D-6/D-7
Fig. 9.6-1 (Foldout)/Blank	Fig. 9.6-1 (Foldout)/Blank
13.3-1/13.3-2	13.3-1/13.3-2
13.3-3/13.3-4	13.3-3/13.3-4
13.3-5/13.3-6	13.3-5/13.3-6
13.3-9/Blank	13.3-9/Blank
M-2 (Foldout)/Blank	M-2 (Foldout)/Blank
M-3 (Foldout)/Blank	M-3 (Foldout)/Blank
M-4 (Foldout)/Blank	M-4 (Foldout)/Blank
13.4-1/13.4-2	13.4-1/13.4-2
14-1/14-2	14-1/14-2
Q 1.9-1/Q 1.9-2	Q 1.9-1/Q 1.9-2
Q 1.9-3/Q 1.9-4	Q 1.9-3/Q 1.9-4
Q 1.9-5/Q 1.9-6	Q 1.9-5/Q 1.9-6
Q 1.9-7/Q 1.9-8	Q 1.9-7/Blank
Q 1.9-9/Q 1.9-10	-----/-----
Q 1.9-11/Q 1.9-12	-----/-----
Q 1.9-13/Q 1.9-14	-----/-----
Q 1.9-15/Blank	-----/-----
Q 5.6(a)-1/Blank	Q 5.6(a)-1/Blank
Q 5.6(f)-1/Q 5.6(f)-2	Q 5.6(f)-1/Q 5.6(f)-2
Q 5.6(f)-3/Q 5.6(f)-4	Q 5.6(f)-3/Q 5.6(f)-4
Q 5.16.1-1/Blank	Q 5.16.1-1/Blank
Q 7.8-1/Q 7.8-2	Q 7.8-1/Q 7.8-2
Q 7.8-3/Q 7.8-4	Q 7.8-3/Blank
Q 7.8-5/Q 7.8-6	-----/-----
Q 7.8-7/Q 7.8-8	-----/-----
Q 7.8-9/Q 7.8-10	-----/-----
Q 7.8-11/Q 7.8-12	-----/-----
Q 7.8-13/Q 7.8-14	-----/-----
Fig. Q 7.8-1/Fig. Q 7.8-2	-----/-----
Fig. Q 7.8-3/Fig. Q 7.8-4	-----/-----
Q 13.1-5/Q 13.1-6	Q 13.1-5/Q 13.1-6
Q 13.1-9/Tbl. Q 13.1-1	Q 13.1-9/Tbl. Q 13.1-1
Q 14.2-1/Q 14.2-2	Q 14.2-1/Q 14.2-2
Q 14.2-3/Blank	Q 14.2-3/Blank
III-7/III-8 (Volume 6)	-----/-----
Fig. 5/Blank (Volume 6)	-----/-----

Done 4-24-70
50-247

Regulatory File Cy.

Received w/Ltr Dated 3-30-70

PREFACE TO SUPPLEMENT NO. 7

Supplement 7 to the Indian Point Unit No. 2 Final Safety Analysis Report consists of responses to questions from the Atomic Energey Commission and page changes to the Report. The questions were contained in a letter to Arthur N. Anderson, of Consolidated Edison Company of New York, from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969, and a letter to William J. Cahill, Jr., of Consolidated Edison Company of New York, from Peter A. Morris, dated November 13, 1969. The responses consist of questions and answers to be added to Volume 5 of the Report in the proper order behind Tab II.

The supplement responds to several questions concerning Chapters 4, 5, 6, 9, 13 and 14 of the Report. Later Supplements will complete the responses to the questions contained in the August 4, and November 13, 1969, letters. This preface page should be inserted in Volume 5 of the Report immediately following the preface to Supplement No. 5.

INSTRUCTIONS FOR SUPPLEMENT SEVEN PAGE CHANGES

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
3.1.3-1/3.1.3-2	3.1.3-1/3.1.3-2
Table 4.2-1/(Continued)/Table 4.2-2	Table 4.2-1 (Continued)/ Table 4.2-2
6.2-17/6.2-18	6.2-17/6.2-18
Table 6.2-1/6.2-2	Table 6.2-1/6.2-2
Fig. 6.2-1/Blank	Fig. 6.2-1/Blank
Fig. 6.2-7/Blank	Fig. 6.2-7/Blank
Fig. 6.2-8/Blank	Fig. 6.2-8/Blank
Fig. 6.2-9/Blank	Fig. 6.2-9/Blank
6.3-1/6.3-2	6.3-1/6.3-2
6.3-3/6.3-4	6.3-3/6.3-4
6.3-5/6.3-6	6.3-5/6.3-6
6.3-7/6.3-8	6.3-7-6.3-8
6.3-9/6.3-10	6.3.9/6.3-10
6.3-11/6.3-12	6.3-11/6.3-12

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
6.3-13/6.3-14	6.3-13/6.3-14
6.3-15/6.3-16	6.3-15/6.3-16
6.3-17/6.3-18	6.3-17/6.3-18
6.3-19/Blank	6.3-19/Blank
Table 6.3-1/Table 6.3-2	Table 6.3-1/Table 6.3-2
Table 6.3-3/Table 6.3-4	Table 6.3-3/Table 6.3-4
Table 6.3-4 (Continued)/Table 6.3-4 (Continued)	Table 6.3-4 (Continued)/ Table 6.3-4 (Continued)
Table 6.3-5/Blank	Table 6.3-5/Blank
Fig. 6.3-1/Blank	Fig. 6.3-1/Blank
6.4-1/6.4-2	6.4-1/6.4-2
6.4-3/6.4-4	6.4-3/6.4-4
6.4-5/6.4-6	6.4-5/6.4-6
6.4-7/6.4-8	6.4-7/6.4-8
6.4-9/6.4-10	6.4-9/6.4-10
6.4-11/6.4-12	6.4-11/6.4-12
6.4-13/6.4-14	6.4-13/6.4-14
6.4-15/6.4-16	6.4-15/6.4-16
6.4-17/6.4-18	6.4-17/6.4-18
6.4-19/6.4-20	6.4-19/6.4-20
6.4-21/6.4-22	6.4-21/6.4-22
6.4-23/6.4-24	6.4-23/6.4-24
6.4-25/6.4-26	6.4-25/6.4-26
6.4-27/6.4-28	6.4-27/6.4-28
Table 6.4-1/Table 6.4-2	Table 6.4-1/Table 6.4-2
Fig. 6.4-1/Blank	Fig. 6.4-1/Blank
Fig. 6.4-2/Blank	Fig. 6.4-2/Blank
Fig. 6.4-3/Blank	Fig. 6.4-3/Blank
Fig. 6.4-4/Blank	Fig. 6.4-4/Blank
7.2-7/7.2-8	7.2-7/7.2-8
7.2-9/7.2-10	7.2-9/7.2-10
7.2-25/7.2-26	7.2-25/7.2-26
Fig. 8.2-1/Blank	Fig. 8.2-1/Blank
9.5-3/9.5-4	9.5-3/9.5-4
9.5-7/9.5-8	9.5-7/9.5-8
9.6-5/Table 9.6-2	9.6-5/Table 9.6-2
Fig. 9.2-1/Blank	Fig. 9.2-1/Blank
11.2-27/11.2-28	11.2-27/11.2-28
Fig. 14.3.4-1/Fig. 14.3-4-2	Fig. 14.3.4-1/Fig. 14.3.4-2
Q 4.11-1/Q 4.11-2	Q 4.11-1/Q 4.11-2
Q 4.11-3/Q 4.11-4	Q 4.11-3/Q 4.11-4

done 3-16-70

Docket No. 50-247
Regulatory File Cy.
Received w/Ltr Dated 3-3-70

PREFACE TO SUPPLEMENT NO. 6

Supplement 6 to the Indian Point Unit No. 2 Final Safety Analysis Report consists of responses to questions from the Atomic Energy Commission. The questions were contained in a letter to Arthur N. Anderson of Consolidated Edison Company of New York from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969 and a letter to William J. Cahill, Jr. of Consolidated Edison Company of New York from Peter A. Morris dated November 13, 1969. The responses consist of questions and answers to be added to Volume 5 of the Report in the proper order behind Tab II. Page changes for the report are included with this supplement.

The supplement responds to several questions concerning Chapers 1, 3, 4, 6, 9, and 14 of the Report. Later Supplements will complete the responses to the questions contained in the August 4, and November 13, 1969 letters. Also included with this supplement is the Indian Point Unit 2 Containment Design Report which is to be placed in Volume 5 of the FSAR immediately following the WEDCO Project reorganization report issued with Supplement 4. Table IV should be placed in front of the Containment Design Report. This preface page should be inserted in Volume 5 of the Report immediately following the preface to Supplement No. 5.

INSTRUCTIONS FOR SUPPLEMENT SIX PAGE CHANGES

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
1.1-5/blank ✓	1.1-5/blank ✓
1.3-3/1.3-4 ✓	1.3-3/1.3-4 ✓
1.3-13/1.3-14 ✓	1.3-13/1.3-14 ✓
1.3-27/1.3-28 ✓	1.3-27/1.3-28 ✓
1.5-1/1.5-2 ✓	1.5-1/1.5-2 ✓
i-1/blank ✓	i-1/blank Supp. 51 ✓
3-i/3-ii ✓	3-i/3-ii ✓
3-v/3-vi ✓	3-v/3-vi ✓
3.1.2-3/3.1.2-4 ✓	3.1.2-3/3.1.2-4 ✓
3.2.1-5/3.2.1-6 ✓	3.2.1-5/3.2.1-6 ✓

INSTRUCTIONS FOR SUPPLEMENT FIVE PAGE CHANGES (Cont'd)

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front Back)
0-3/0-4	0-3/0-4
0-5/0-6	0-5/0-6
Figure 3.2.1-1/Figure 3.2.1-2	Figure 3.2.1-1/
	Figure 3.2.1-2
3.3-3/3.3-4	3.3-3/3.3-4
Appendix 3B	________
7i/7ii	7i/7ii
7.3-3/7.3-4	7.3-3/7.3-4
7.3-7/blank	7.3-7/blank
9.2-21/9.2-22	9.2-21/9.2-22
14iii/14iv	14iii/14iv
14.2.6-7/14.2.6-8	14.2.6-7/14.2.6-8
14.2.6-9/14.2.6-10	14.2.6-9/14.2.6-10
Z-i/blank	14.2.6-11/Z-1
Q1.9-7 (Volume 5, Tab II)	Q 1.9-7 (Volume 5, Tab II)

date 2-26-70

PREFACE TO SUPPLEMENT NO. 5

Supplement 5 to the Indian Point Unit No. 2 final Safety Analysis Report consists of responses to questions from the Atomic Energy Commission. The questions were contained in a letter to Arthur N. Anderson of Consolidated Edison Company of New York from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969 and a letter to William J. Cahill, Jr. of Consolidated Edison Company of New York from Peter A. Morris dated November 13, 1969. The responses consist of questions and answers to be added to Volume 5 of the Report in the proper order behind Tab II. Page changes for the Report are included with this supplement.

The supplement responds to several questions concerning Chapters 1, 4, 6, 11, 12 and 14 of the Report. Later Supplements will complete the responses to the questions contained in the August 4, and November 13, 1969 letters. Also included with this supplement is a Tab III to be placed in Volume 5 of the FSAR immediately preceeding the WEDCO Project reorganization report issued with Supplement 4. This preface page should be inserted in Volume 5 of the Report immediately following the preface to Supplement No. 4.

INSTRUCTIONS FOR SUPPLEMENT FIVE PAGE CHANGES

Insert Revised Sheet (Front/Back	Discard Old Sheet (Front/Back)
7.2-19/7.2-20	7.2-19/7.2-20
8.1-1/8.1-2	8.1-1/8.1-2
8.1-3/Blank	8.1-3/Blank
8.2-3/8.2-4	8.2-3/8.2-4
8.2-11/8.2-12	8.2-11/8.2-12

Done 1-7-70

PREFACE TO SUPPLEMENT NO. 4

Supplement 4 to the Indian Point Unit No. 2 Final Safety Analysis Report consists of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Consolidated Edison Company of New York from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969. The responses consist of questions and answers to be added to Volume 5 of the Report in the proper order behind Tab II. Page changes for the Report are included with Supplement No. 4. Also included with this supplement is a description of the Project reorganization within Westinghouse. This section should be added to Volume 5 of the report immediately following the response to Question 14.5. In a later supplement, a separate Tab III will be provided for identif'cation

This supplement responds to several questions concerning Chapters 4, 5, 7, 11 and 14 of the Report. Later Supplements will complete the responses to the questions contained in the August 4, 1969 letter. This preface page should be inserted in Volume 5 of the Report immediately following the preface to Supplement No. 3. The following list indicates the disposition of revised pages of the FSAR:

INSTRUCTIONS FOR SUPPLEMENT TWO PAGE CHANGES

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
9.9.2-1/9.9.2-2	9.9.2-1/9.9.2-2
14.2.6-9/14.2.6-10	14.2.6-9/14.2 6-10
14.2.6-11/Z-1	14.2.6-11/Z-1

PREFACE TO SUPPLEMENT NO. 3

Supplement 3 to the Indian Point Unit No. 2 Final Safety Analysis Report consists of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Consolidated Edison Company of New York from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969. The responses consist of questions and answers to be added to Volume 5 of the Report in the proper order behind Tab II. Page changes for the Report are included with Supplement No. 3.

This supplement responds to several questions concerning Chapters 1, 4, 5, 7, 8 and 11 of the Report. Later Supplements will complete the responses to the questions contained in the August 4, 1969 letter. This preface page should be inserted in Volume 5 of the Report immediately following the preface to Supplement No. 2. The following list indicates the disposition of revised pages of the FSAR:

INSTRUCTIONS FOR SUPPLEMENT TWO PAGE CHANGES

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
8.2-13/8.2-14	8.2-13/8.2-14
8.2-15/8.2-16	8.2-15/8.2-16

fire 1-5-64 SS.

PREFACE TO SUPPLEMENT NO. 2

Supplement 2 to the Indian Point Unit No. 2 Final Safety Analysis Report consists of responses to questions from the Atomic Energy Commission and page changes to the report. The questions were contained in a letter to Arthur N. Anderson of Consolidated Edison Company of New York from Peter A. Morris, Director of the Division of Reactor Licensing, dated August 4, 1969. The responses consist of questions and answers to be added to Volume 5 of the Report in the proper order behind Tab II. Page changes for the Report are included with Supplement No. 2.

This supplement responds to seve .l questions concerning Chapters 1, 4, 5, 6, 7, 12 and 14 of the Report. Later Supplements will complete the responses to the questions contained in the August 4, 1969 letter. This preface page should be inserted in Volume 5 of the Report immediately following the preface to Supplement No. 1. The following list indicates the disposition of revised pages of the FSAR:

INSTRUCTIONS FOR SUPPLEMENT TWO PAGE CHANGES

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
6.2-25/6.2-25a	6.2-25/6.2-26
6.2-25b/6.2-26	-----/-----
Fig 11A-1/blank	-----/-----
Fig 11A-2/blank	-----/-----
Q 1.11-1/blank (Volume 5, Tab II)	Q 1.11-1/bla r

PREFACE TO SUPPLEMENT NO. 1

Supplement 1 to the Indian Point Unit No. 2 Final Safety Analysis Report consists of responses to questions from the Atomic Energy Commission contained in two letters. The first letter from Peter A. Morris, Director of the Division of Reactor Licensing, on March 5, 1969 to Mr. Donham Crawford of Consolidated Edison Company of New York, Inc. requested additional information on the medical plans and facilities at Indian Point. The questions and responses are found following Tab I of Volume 5 of the FSAR. The responses to these questions have been incorporated into Section 11.2 of the FSAR as page changes. The responses to the questions in Volume 5 indicate where the specific answer may be found in the page change.

The second letter to Arthur N. Anderson of Consolidated Edison Company of New York from Peter A. Morris, dated August 4, 1969, requested additional information on Chapters 1, 2, 3, 4, 5, 6, 7, 8, 11, 12 and 14 of the FSAR. Supplement 1 responds to several of the questions in the second letter found behind Tab II of Volume 5 of the FSAR. The responses consist of questions and answers given in Volume 5 of the FSAR and also of page changes to the original text of the FSAR in some instances. The page changes are referenced when used as a response to the question. Later supplements will complete the responses to the questions contained in the August 4, 1969 letter. The following list indicates the dispostion of revised pages of the FSAR:

INSTRUCTIONS FOR SUPPLEMENT ONE PAGE CHANGES

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
7.2-14/Blank	7.2-14/Blank
7.2-27/7.2-28	7.2-27/7.2-28
11.2-27/11.2-28	~~11.2-27/11~~.2-28

Insert Revised Sheet (Front/Back)	Discard Old Sheet (Front/Back)
11.2-29/11.2-30	11.2-29/11.2-30
11.2-31/11.2-32	11.2.31/11.2-32
11.2-33/11.2-34	11.2-33/11.2-34
11.2-35/11.2-36	11.2-35/11.2-36
11.2-37/11.2-38	11.2-37/Tbl 11.2-I
11.2-39/11.2-40	-----/-----
11.2-41/11.2-42	-----/-----
11.2-43/11.2-45	-----/-----
Tbl 11.2-1/Blank	-----/-----
Tbl 11.2-10/Blank	Tbl 11.2-10/H-1
Tbl 11.2-11/Blank	-----/-----
Tbl 11.2-11 (Cont'd)/Blank	-----/-----
Tbl 11.2-12/Blank	-----/-----
Tbl 11.2-13/Blank	-----/-----
Tbl 11.2-13 (Cont'd)/Blank	-----/-----
Fig 11.2-5/Blank	-----/-----
11A-1/11A-2	-----/-----
11A-3/Blank	-----/-----
Fig 11A-1/Blank	-----/-----
Fig 11A-2/Blank	-----/-----
11B-1/11B-2	-----/-----
11B-3/Blank	-----/-----
11C-1/Blank	-----/-----
Fig 11C-1/Blank	-----/-----
Fig 11C-2/Blank	-----/-----
11C-2/Blank	-----/-----
14.3.1-13/14.3.1-14	14.3.1-13/14.3.1-14
14.3.1-27/14.3.1-28	14.3.1-27/14.3.1-28
Fig 14.3.2-42/Blank	-----/-----
Quirk, Lawler & Matusky Flooding Evaluation-insert at end of FSAR Section 2.5	-----/-----

ITEM 1

A description of onsite facilities for decontamination and immediate emergency treatment of injured personnel, including details concerning the following:

a. decontamination space including size, location in reference to plant population and operations and/or hazards relating to radioactivity, shielding provided, shower and water availability, waste control and disposal;

b. existing emergency medical care facility, including location, equipment, supplies, plumbing, waste disposal and hours of availability on round-the-clock basis;

c. equipment and supplies available for immediate gross decontamination of personnel, including injured; and

d. equipment, supplies, written procedures and standing orders for immediate control and emergency treatment of injured personnel.

ANSWER

The response to Item 1 parts a, b, c and d is found on revised pages 11.2-28 through 11.2-37, including Figures 11.2-4 and 11.2-5, Tables 11.2-11 and 11.2-12, and Appendix 11A.

ITEM 2

A description of qualifications, professional education and special training (e.g., training in supervision and care of injured radioactivity contaminated persons and other occupational health hazards) of:

a. resident professional personnel, such as physicians, nurses, industrial hygienists, and health physicists;

b. resident semi-professional personnel, such as nursing assistants, and health physics technicians;

c. readily available offsite professional medical personnel;

d. provisions for meeting cost of services rendered by each of the above persons; and

e. hours of onsite duty, offsite availability, location and distance from site, and hospital staff appointments currently held for personnel listed in (a) through (c) above.

ANSWER

The response to Item 2, parts (a) through (e) is found on page 11.2-37 and Table 11.2-13 as revised by Supplement 1.

ITEM 3

A description of arrangements for transport of injured personnel, including:

a. equipment and supplies for in-transit emergency treatment;
b. standing orders for emergency procedures kept in vehicles;
c. location of vehicles, including distance from site and average time to respond to call; and
d. availability on round-the-clock basis.

ANSWER

The response to Item 3 is found on pages 11.2-37 and 11.2-38 as revised by Supplement 1.

ITEM 4

Identification and location (including distance from site) of hospital agreeable to accepting:

a. patients for further decontamination; and
b. contaminated injured personnel, for treatment including:

   (1) description and location of special facilities designated for contaminated patients;
   (2) description and location of special facilities for treating radiation injuries; and
   (3) equipment, including surgical facilities and supplies for handling radiation or contamination victims.

ANSWER

The response to Item 4 is found on pages 11.2-38 through 11.2-40 and Appendix 11B and 11C as revised by Supplement 1.

ITEM 5

Qualifications of professional medical personnel at the support facility (hospital or clinic) to treat radiation and contamination victims, including number and types of physicians and a description of any specialized training related to contamination or radiation injuries.

ANSWER

The response to Item 5 is found on page 11.2-37, third paragraph, as revised by Supplement 1.

ITEM 6

A description of any limitations that exist regarding availability of offsite medical facilites and support, with particular regard to:

a. time of admission of accident casualties;
b. length of stay for contaminated patients;
c. extent of contamination or direct radiation levels associated with injuries;
d. types of injuries or illnesses; and
e. any special limitations on admission or treatment, such as indemnification of the medical facility by the licensee.

ANSWER

The response to Item 6, parts (a) through (e) is found on page 11.2-40 of the FSAR as revised by Supplement 1.

ITEM 7

Presence of written plan and standing orders in receiving area of hospital detailing actions to be taken and procedures to be followed when contaminated person with or without injury is brought to hospital.

If plan and orders are not posted, is hospital willing to:

a. have such plan and orders readily available for emergency use; and
b. instruct professional and administrative staff about plan and orders?

ANSWER

The response to Item 7, parts (a) and (b) is found on page 11.2-41 and in Appendix 11B of the FSAR as revised by Supplement 1.

QUESTION 1.1

In order to determine whether an extensive survey should be made for post-earthquake damage prior to continuing operation, information should be available as to the loadings experienced by the structures and Class I equipment. Indicate whether, and how, strong motion seismographs will be installed, and how determination will be made that the response of structures and Class I equipment is within allowable design limits.

ANSWER

A strong-motion accelerograph will be installed on a concrete slab directly on bedrock in the yard area of the plant.

The yard was selected since bedrock is readily accessible in this area. Since large quantities of rock had to be removed to form the yard area and build the structures surrounding the yard, the dynamics of the bedrock system will not be significantly changed by the added weight of the new structures. Therefore, during any seismic activity at the site, the amount of feedback from these structures into the rock which could influence the recordings of the seismograph will be very small.

This model will be equipped with three precision leveling screws and provisions for permanent mounting. It will be an automatically actuated completely self-contained three component seismograph. Three traces will be recorded on 70-mm photographic film measuring ground accelerations versus time for two horizontal and one vertical axis.

In the event of an earthquake, a visual inspection will be made of the external structures and all accessible internal areas and equipment for signs of deformation or damage. The developed traces of the ground acceleration will be available for further analysis if any damage is found.

Question 1.2

Define the relative terms "excessive" and "substantial" as used in definitions of Class I and Class II components on page A2 of the FSAR, in terms of radiation doses specified in 10 CFR Part 20 or 100.

Answer

1.0 DEFINITION OF SEISMIC DESIGN CLASSIFICATIONS

All equipment and structures are classified as Class I, and Class II, or Class III as recommended in:

a) TID-7024, "Nuclear Reactors and Earthquakes" August, 1963 and,

b) G. W. Housner, "Design of Nuclear Power Reactors Against Earthquakes", Proceedings of the Second World Conference on Earthquake Engineering, Vol. 1, Japan 1960, Pg. 133, 134 and 137.

Class I

Those structures and components including instruments and controls whose failure might cause or increase the severity of a loss-of-coolant accident or result in an uncontrolled release of radioactivity causing more than 10 rem. to the thyroid or 10 rem. whole body to the average adult beyond the nearest site boundary. Also, those structures and components vital to safe shutdown and isolation of the reactor.

Class II

Those structures and components which are important to reactor operation but not essential to safe shutdown and isolation of the reactor and whose failure could result in the release of radioactivity causing more than

1.0 rem to the thyroid or .5 rem whole body dose to the average adult beyond the nearest site boundary.

Class III

Those structures and components which are not directly related to reactor operation and containment.

In Indian Point Unit No. 2, the only portions of the plant which might carry substantial radioactivity which are not Class I because of required safeguards operation or requirements for safe shutdown and isolation of the reactor are portions of the chemical and Volume Control System and Waste Disposal System.

The specific components in the Chemical and Volume Control System are the volume control tank, holdup tank, and the concentrates holding tank with associated piping, valves and supports. These components, are all Class I. In addition, the design of the system tanks and their location was based upon the commitment that a vessel rupture would not cause doses in excess of 10CFR20 limits at the exclusion radius.

The specific components in the Waste Disposal System are the gas decay tanks with the associated piping, valves and supports. These components are all Class I. In addition, the gas decay tanks of the Waste Disposal System have been designed such that the failure of any tank will not exceed 10CFR20 doses at the exclusion radius.

The Analysis showing that the rupture of the volume control tank or a gas decay tank does not exceed the special dose limits selected for Indian Point Unit No. 2 is found in Section 14.2.3 of the FSAR. See also the Technical Specifications, Section 3.

Those components of the Chemical and Volume Control System that are not Class I are listed in Appendix A to the FSAR, page A-10. In addition, the Boric Acid evaporator and the condensate demineralizer are not Class I.

Those components of the Waste Disposal System which are not Class I are as follows: regenerant tank, waste condensate tank and pumps, baler, reactor coolant drain tank, and the waste evaporator.

Failure of these components will not result in off site doses in excess of 10CFR20 limits at the site exclusion radius.

QUESTION 1.3

Indicate Class I structures and components which are so located that they could be endangered by failure of Class II or Class III structures, and what protection has been provided these components. Also, can the failure of the Class III cranes endanger any Class I function?

ANSWER

The only Class I structures and components which are so located that they could be endangered by failure of Class III structures are the Control Building, main steam piping and feedwater piping which could be endangered by the Class III Turbine Building. No special provisions have been provided except in the case of the main steam and feedwater lines up to the isolation valves which are protected by the shield wall and the structural frame at the north end of the shield wall. Since these are located near the braced end of the Turbine Hall, we would not anticipate any structural failure in this area.

The only Class III crane whose failure could endanger any Class I function is the Fuel Storage Building crane. Failure of this crane will not impair a safe and orderly shutdown.

The wheels of the bridge and the trolley are shaped such that sliding perpendicular to the rail would not be possible. The lateral load from an earthquake on the trolley crane rail is about 50% greater than the lateral loads from impact which the AISC Code specifies for design within working stress limits. The stresses on the crane rail are low due to the earthquake load. For this reason no failure of the crane rail is anticipated.

The manipulator crane in the Containment Building, a Class III crane, is restrained from overturning and will not endanger Class I structures.

The turbine building, and fuel handling building of Unit #2 are functionally Class III structures. However, these structures have been analyzed using a multidegree of freedom modal dynamic analysis method to insure that there is no potential for gross structural collapse of these structures as a result of the maximum hypothetical earthquake as defined in Figure A.1-2 of Appendix A of the FSAR. The results of the analyses are given below. A value of 7 percent structural damping was assumed in the analysis. Total response of the structure was determined on the basis of the square root sum of square basis of each mode contribution. A similar dynamic analysis was also performed to insure that no potential gross failure of the Unit #1 stack or super heater building could occur for the maximum hypothetical earthquake and the design basis tornado for Unit #3. The resultant dead, live and seismic design stresses in the basic building structure will be limited to 0.9 yield of the steel. Modification of building structures will be made as required to insure this criterion is met.

Results of Analyses

A. Seismic Analysis of the Indian Point Unit #2 Turbine Building

A spectrum response analysis was performed for the Turbine Building considering the Design Basis Earthquake (DBE) which has a peak horizontal ground acceleration of 0.1⁻ The associated earthquake response spectra is shown in Figure A.1-2 of the FSAR normalized for 0.15g zero period ground acceleration for 7 percent damping.

The foundation was considered rigid since the footings for the structural frames of the building are underlain by either rock or a lean concrete which bears on rock. Also, in the analysis, interaction between the turbine and the structural frame for the building was neglected. The analysis, as performed, represents a linear elastic system.

The analysis of the Turbine Building was performed under the assumption that the north-south motions, east-west motions and vertical motions will be uncoupled. The dynamic analysis effort was limited only to hor' 'tal motions in the east-west and north-south directions. However, vertical components of the earthquake were considered by adding a 0.13g component to dead loads. Each of the models was simulated for the computer program called STARDYNE. a description of the modeling capabilities of STARDYNE are contained in "STARDYNE Structural Analyses System Users' Manual" prepared by Mehcanics Research, Inc., for Control Data Corporation.

The STARDYNE Structural Analysis Program was used in three ways. First, the portal frames were analyzed for a static unit force at each portal to determine their resistance to horizontal motions resulting from the turbine bay crane. This information was incorporated into the model for the analysis of the crane girder to determine the distribution of horizontal turbine bay crane loads to the various E-W portal frames. Secondly, the program was used to determine the forces induced in the frames as a result of gravity forces, and, thirdly, the STARDYNE p.ogram was used to determine the fundamental frequencies of each of the models and the characteristic shapes. In addition, the STARDYNE program is also capable of determining the modal member forces for each of the fundamental frequencies. This information for each model and mode was stored on tape along with the gravity forces for each model and later used in a earthquake analysis program to determine the maximum probable deflection, acceleration, member forces, member stresses and the combined gravity plus earthquake member stress responses. Dynamic characteristics of the turbine building are shown in Table 1.3-1.

Results of the analysis indicated that the 0.9fg combined load allowable stress was not violated except locally in the flange of columns where cross bracing framed in eccentric to other jo'nt members. These stresses will be reduced to allowable values by the addition of flange cover plates.

While allowable stresses in the cross bracing did not exceed the 0.9 yield stress allowable it was determined that most of the x cross bracing would buckle at very low compressure stress due to high ℓ/r ratios. In order to assure the lateral stiffness of the bents and load carrying capacity as determined in the analysis, cover plates will be attached to the bracing equal to the original area of the x cross bracing. This assures design adequacy with only x cross bracing in tension assumed active in carrying lateral load.

B. Seismic Evaluation of the Fuel Storage Building Structure Above the Fuel Storage Pit

The fuel storage building for Indian Point Unit #2 consists of the fuel storage pit constructed of reinforced concrete and founded on rock. The fundamental frequency of the pit is approximately 22 cps and therefore can be considered rigid. The steel superstructure above the pit encloses the pit and supports the fuel cask handling crane. This superstructure was designed as a Class III structure.

The seismic loads used in the analysis of the steel superstructure were as follows:

1. Zero Period Ground Acceleration
   0.15 g horizontal
   0.10 g vertical
2. 7 percent damping
3. Response Spectrum curve as defined in Figure A.1-2 of the Indian Point Unit #2 FSAR.
4. Inertial forces for each mass point are determined on the basis of the square root sum of the squares.

A dynamic multidegree of freedom modal analysis of the structure was constructed as shown in Figures 1.3-1 and 1.3-2. The stiffness properties of the elements were determined by the combined stiffness of the frame bents in the N-S and E-W directions taken separately. The stiffness of each bent was determined by the computer program STRUDL. The total inertial forces determined by the dynamic analysis were distributed to each individual bent and resultant member stresses determined. The crane was assumed fully loaded. Evaluation of these seismic stresses show maximum stresses occurring in diagonal bracing. The maximum stress thus determined in the cross bracing was 18.5 ksi.

The maximum combined dead and seismic column load stress determined by the analysis was 12.8 psi compression.

On the basis of these results it was determined that the fuel storage building superstructure was adequately designed without modification to carry the seismic load defined for the site.

In addition to the analysis of the building structure the fuel crane bridge was evaluated to determine the potential for the crane bridge to lift off its track support in the event of a seismic disturbance. The vertical mode fundamental frequency of the fuel storage building is approximately 9 cps.

The crane bridge has also been analyzed dynamically both loaded and unloaded and for various positions of the trolley. It was determined that the crane with the trolley at the end of the span and unloaded would have a fundamental frequency of approximately 9 cps. Considering potential resonance with the fundamental vertical mode of the building at 9 cps the resulting g loading was 1.05 g. The only potential for crane lift off will be in the unloaded condition with the trolley parked near the support since the unloaded crane will not be parked over the pool no potential hazard exists and vertical restraints are not required.

C. Seismic and Wind Analysis of the Super Heater Stack of Indian Point Unit #1

SHORTENED STACK

The Indian Point Unit 1 Super Heater Stack has been analyzed for seismic, tornado and vortex shedding wind load effects. The results of this analysis are summarized below. As a result of this analysis on the existing stack it is concluded:

1. The stack can withstand a tornado wind load of approximately 300 mph prior to buckling failure of the stack steel shell.

2. The maximum stress in the stack at the critical vortex shedding frequency wind velocity is 7660 psi which provides a 3.64 factor of safety against stack failure by this mode.

3. The maximum combined dead and seismic stress for the earthquake parameters defined for the site is 19,140 psi which provides a 1.46 factor of safety against stack failure by this mode.

Load Case 1 - Tornado

I. Load Criteria

a) Wind = 300 mph
$L = D + W'$

where:

L = Total Load
D = Dead Load
W' = Tornado Load

II. Method of Load Analysis

As prescribed in ASCE Paper 3269 for uniform wind velocity with height; no gust factor.

III. Allowable Stress Criteria[1]

$$\sigma_a = \frac{0.72Et}{\pi\,(1-\nu^2)r} = 27{,}900 \text{ psi}$$

where:

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(1) Roark, R. J. Formulas for Stress and Strain, 4th Ed. McGraw-Hill Book Co., New York, 1965 (p. 389)

σ_a = allowable stress (psi)

E = modulus of elasticity (psi)

t = shell thickness (in)

ν = Poisson's ratio

r = radius of stack (in)

IV. Stress Determination

$$\sigma = \frac{D}{A} + \frac{W'\bar{y}r}{I} = 1.54 + 25.75 = 27.29 \text{ ksi}$$

where:

$\bar{y}$ = centroidal height of stack (in)

I = moment of inertia of stack (in^4)

A = crossectional area of stack (sq. in.)

$$\text{Factor of Safety} = \frac{\sigma a}{\sigma} = \frac{27.9}{27.29} = 1.02$$

Load Case 2 - Seismic

I. Load Criteria

a) Zero Period Ground Acceleration
   0.15 g horizontal
   0.10 g vertical
b) Damping 7 percent
c) Ground Response Curve - Figure A.1-2 of Indian Point #2 FSAR.

$$L = D + E_h' + E_v'$$

where:

E'_h = load resulting from horizontal earthquake component

E'_v = load resulting from vertical earthquake component

II. Method of Load Analysis

Multidegree of freedom modal analysis of the superheater building and stack as shown in Figure 1.3-3. The square root sum of squares of seismic inertia forces at mass points are used to determine resultant shear and moments in he stack.

III. Allowable Stress Criteri:

See Load Case 1, item I' .

IV. Stress Determination

$$\sigma = \frac{D}{A} + \frac{E'_v}{A} + \frac{E'_h \bar{x} r}{I}$$

$$\sigma = 1.54 + 0.20 + 17.4 = 19.14$$

$$\text{Factor of Safety} = \frac{\sigma a}{\sigma} = \frac{27.9}{19.14}$$

$$= 1.46$$

where:

$\bar{x}$ = lever arm of node inertia force

Load Case 3 - Vortex Shedding

I. Expression for maximum uniformly distributed force due to Vortex Shedding[2,3]

$$P = (MF)\ (1/2\ \rho v^2 \cdot C_L \cdot D \cdot L \frac{\pi}{\delta}$$

(2) Smith, J. O. and McCarthy, "Dynamic Response of Tall Stacks to Wind Excitation," ASME Paper No. 63-WA-248, 1963.

(3) Jones, G. W., "Unsteady Lift Forces Generated by Vortex Shedding About a Large, Stationary, and Oscillating Cylinder at High Reynolds Number," ASME Paper No. 68-FE-36, 196.

C_L = Lift Coefficient for a stationary circular cylinder

MF = A multiplying factor applied to the lift coefficient to account for a vibrating cylinder

D = Average stack diameter (ft)

L = Length of Stack (ft)

δ = Logarithmic decrement

ρ = Air density (0.0023385 # - sec^2/ft^4)

v = F 1 · Vc

Vc = Critical Vortex shedding velocity (fps)

F1 = A correction factor which accounts for fact that stack oscillations have occurred as high as 30% above shedding velocity

$$Vc = \frac{f \cdot D}{S}$$

S = Stronhal number

f = Fundamental frequency (cps)

II. Pertinent parameters

CL = 0.1
MF = 4.0
D = 20'
L = 334.5'

$\delta = 0.04\pi$ (2% critical damping)

$V_c = 42.7$ fps

$F_l = 1.2$

$S = 0.27$

$f = 0.576$ cps

III. Stress criteria

$$f = \frac{D}{A} + \frac{Phr}{2I} = 1.54 + 6.12 = 7.66 \text{ ksi}$$

$$\text{Factor of Safety} = \frac{\sigma a}{\sigma} = \frac{27.9}{7.66} = 3.64$$

In addition to the analysis performed for the existing stack it was determined that the stack with 80 feet removed from the top would have the capacity to resist a 360 mph wind for the criteria as defined in Load Case I, the seismic as defined in Load Case II and the Vortex Shedding as defined in Load Case III.

D. Seismic and Tornado Evaluation of the Super Heater Building Indian Point Unit #1

A spectrum response analysis was performed for the Super Heater Building considering the Design Basis Earthquake (DBE) which has a maximum horizontal ground motion of .15g. A dampening coefficient equal to seven percent was assumed for all modes. The earthquake response spectra used is shown in Figure A.1-2 of the Indian Point Unit #2 FSAR normalized to 0.15g zero period ground acceleration. In the analysis no interaction with the foundation was considered since the footings for the structural frame for the building are underlain by rock. Also, in the analysis, the stiffness interaction between the turbine building and the structural frame for the Super Heater Building was neglected but the mass of the turbine building was included in the dynamic analysis. The analysis, as performed, represents a linear elastic system.

The analysis of the Super Heater Building was performed under the assumption that the north-south motions, east-west motions and vertical motions were uncoupled. The analysis effort was limited only to horizontal motions in the east-west and north-south directions, and no attempt was made to model vertical motions or to combine vertical and horizontal motions. However, vertical seismic motions have been considered in the results by increasing the dead load stress in building members by a factor equal to two thirds the combined mode horizontal inertial g load as determined in either the east-west or north-south direction.

In each direction, north-south and east-west, the column lines were modeled in detail. These structural models were developed for elastic-static analyses obtained from the computer program STRUDL. They were uses for two purposes: to develop the master stiffness matrices associated with the two directions, east-west and north-south, used in the dynamic analyses and to determine resultant member stresses using the equivalent static seismic forces determined from the dynamic analyses.

The dynamic characteristics, frequencies and mode shapes, of the Superheater Building were determined using the Westinghouse computer program SAND. The equivalent static forces resulting from the dynamic response were developed using a response spectrum seismic analysis performed by the Westinghouse computer program SPECTA.

The equivalent static force associated with a particular mass resulting from a dynamic response is defined as the square root of the sum of the squares of the equivalent static forces associated with that mass for each mode. The equivalent static force associated with a mode and a mass point is defined as the value of the mass times the maximum acceleration associated with the mass point for that particular mode. The maximum acceleration associated with a mode and mass point is defined as follows:

$$(\ddot{U}rn)\ Max = (\ddot{A}n)\ Max\ \phi rn \qquad (1)$$

$$(\ddot{A}n)\ Max = \Gamma_n\ San \qquad (2)$$

$$\Gamma_n = \frac{\sum_r Mr\ \phi' rn}{\sum_r Mr\ \phi^2\ rn} \qquad (3)$$

Where:

n = Refers to mode n

r = Refers to mass r

ϕ'_{rn} = Component of ϕ_{rn} in the direction of the earthquake

ϕ_{rn} = Component of mode shape n for mass r

M_r = Mass lumped at point r

$(\ddot{A}_n)_{Max}$ = Maximum modal acceleration for mode n

S_{an} = Spectral acceleration for mode n from response curve for 7 percent damping

$(\ddot{U}_{rn})_{Max}$ = Maximum acceleration in mode n for mass point r

Γ_n = Modal participation factor for mode n

Sectional views in the N-S and E-W directions are shown in Figures 1.3-3 and 1.3-4. A typical column line modeled for STRUDL to determine overall column line stiffness and permit determination of resultant seismic stresses is shown in Figure 1.3-5. In Figure 1.3-6 is presented the dynamic modal used to determine inertial forces.

Results of the analysis showed several column lines contained diagonal bracing with stresses which exceeded the allowable stress value of 0.9 fy. In addition several of the cross bracing showed compressive stress levels which exceeded the expected buckling stress as determined by the ℓ/r ratio for the member. In these cases it is proposed to strengthen the approximately 200 members which were found to be overstressed by attaching cover plates to the angle bracing. In a few instances columns were found to be locally overstressed due to eccentric poritioning of cross bracing. These areas will be reinforced by flange cover plates. Approximately 30 tons of additional plate will be required to strengthen the structure.

With respect to tornado resistance of the strucutre, total lateral load in the N-S direction is approximately 10 percent and in the E-W direction 20 percent less than the seismic induced lateral load on the structure.

Tornado loads were based on a 360 mph wind using the shape factors for rectangular building as defined in ASCE Paper 3269. It was assumed 20 percent of the wall area of th building was still intact as a reaction surface for the wind in addition to the total surface area of major equipment and the stack at its existing height. On the basis of this analysis, the building has approximately the same resistance capacity to a 360 mph tornado wind as it does for the 0.15g earthquake.

<u>Evaluation of Structural Modifications</u>

In the analysis of the Super Heater and Turbine Building under lateral loads, the following connections will be examined:

1. Gusset Plates
2. Check of connections between beams and columns to determine their adequacy to transfer horizontal shear load.
3. Check of connections at column bases in the foundation to determine their ability to transfer the given horizontal shear load. For those column base connections subjected to a net uplift load an analysis will be made to be certain that they are adequate for these loads.

If it is found that a connection is inadequate to support the given load it will be redesigned.

It is not necessary to reanalyze the Turbine Building after it has been redesigned because the building stiffness characteristics are essentially the same as those assumed in the initial analysis. This is because the significant fixes involve the cross bracing system which is made up of pairs of cross bracing members. In the initial analysis, both sets of cross bracing were assumed active. However, the bracing system was such that cross members will buckle under a very small compressive load. Therefore, lateral building load must be carried in Tension by the bracing system.

The fix used in redesign has been to double the area of cross bracing. The bracing in compression, due to buckling, is not active in resisting lateral building load. Therefore, only half of the cross bracing assumed in the initial analysis, which is in tension, resists this load. However, since the area of cross bracing has been doubled, the resultant effective lateral resistance is the same as that assumed in the original analysis.

An initial analysis was made of the Super Heater Building using the existing design parameters. After completion of the analysis, the over-stressed members were strengthened and a dynamic reanalysis made.

In Tables 1.3-2, 1.3-3 and 1.3-4 are given the relative comparisons in stiffness, horizontal inertial load, and frequency between the initial analysis and the reanalysis.

As indicated in Table 1.3-3, a maximum increase in seismic acceleration of less than 10 percent is indicated. As determined in the original analysis, the stack maximum stress is 19.14 ksi or 1.10 x 19.14 = 21.05 < 27.9 ksi.

Table 1.3-1

Dynamic Characteristic of Turbine Building

Mode No.	Freq. CPS	Valves
1	.5042	.08
2	1.6141	.12
3	2.2849	.19
4	4.3292	.2
5	5.2813	.2
6	8.2814	.18
7	12.1704	.15
9	15.1274	.15
10	20.754	.15
11	22.4809	.15
12	23.8001	.15
13	27.3040	.15
14	33.9678	.15

TABLE 1.3-2 RELATIVE STIFFNESS PERCENTAGES

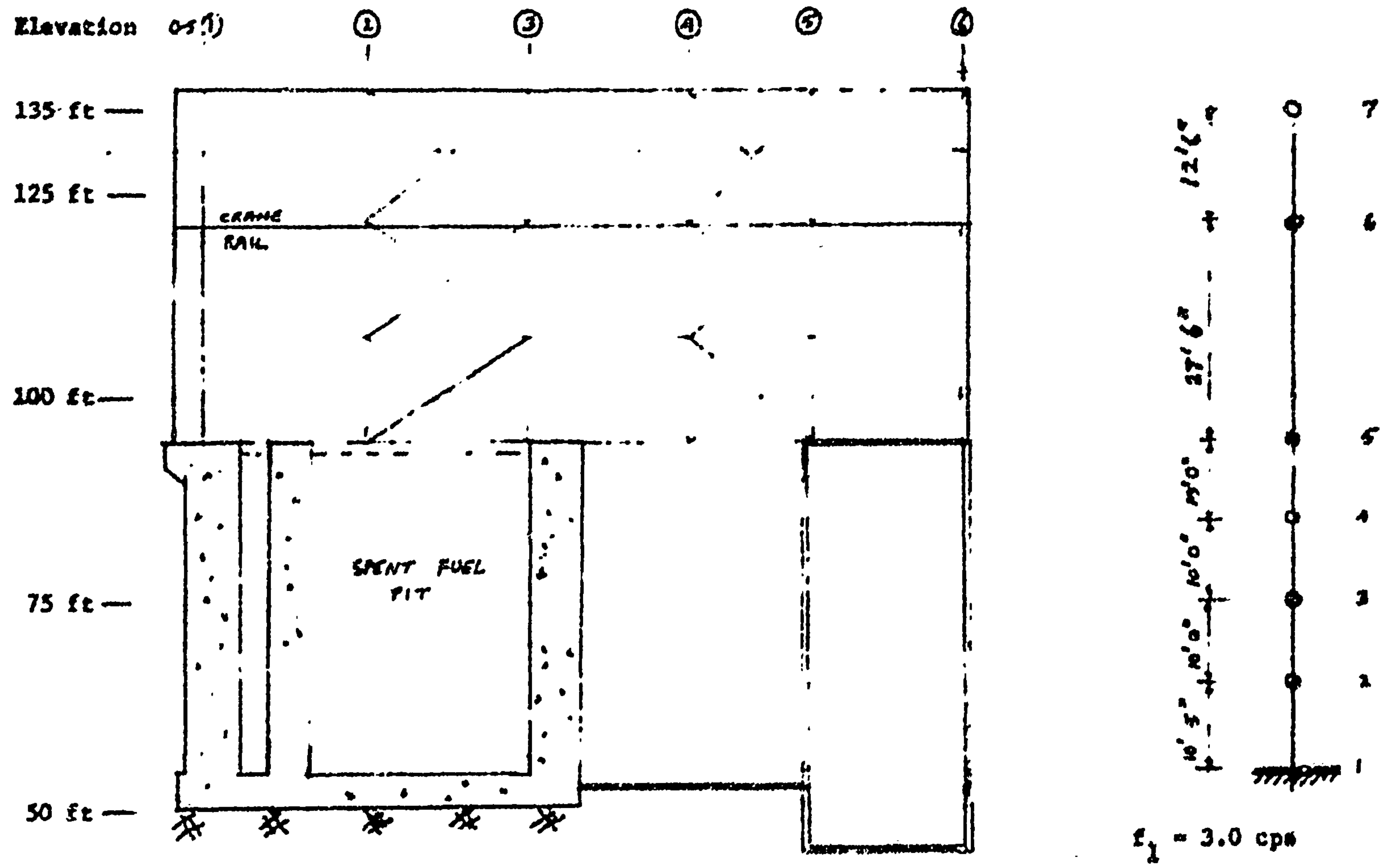
Relative loaction in Super Heater Building	Percentage increase in stiffness between first and second analysis	
	East-West Direction	North-South Direction
Bottom	8%	56.7%
Middle	18.3%	41.4%
Top	19.9%	10.4%

TABLE 1.3-3 INERTIAL LOADS

Relative location in Super Heater Building	Inertial loads for first and second analysis (Units: Kips)			
	East-West Direction		North-South Direction	
	Original	Reanalysis	Original	Reanalysis
Bottom	908	908	1091	1102
Middle	1888	1914	1687	1803
Top	1242	1271	1082	1181

TABLE 1.3-4 FREQUENCIES

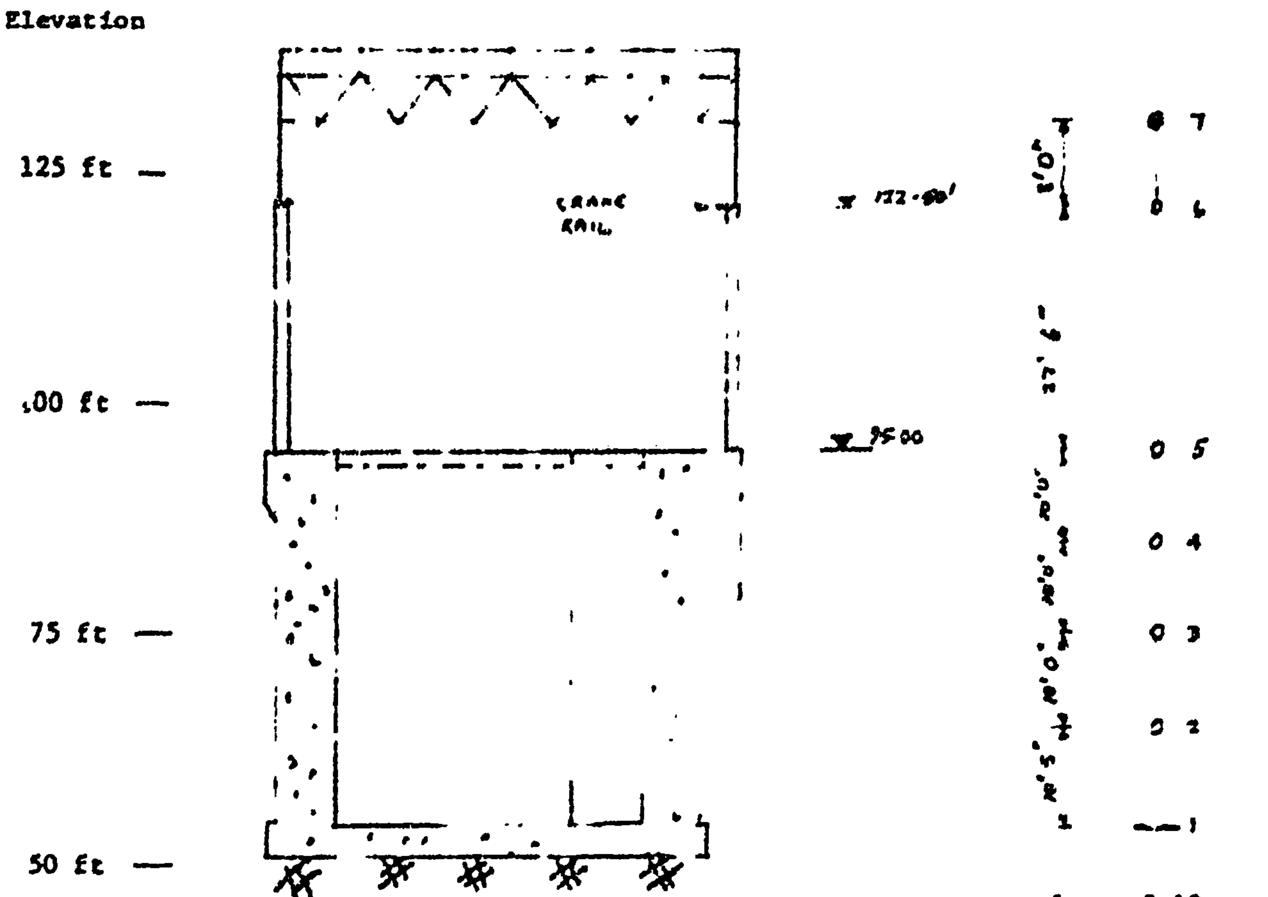
Mode	Frequencies for first and second analysis			
	East-West Direction		North-South Direction	
	Original	Reanalysis	Original	Reanalysis
1	0.94 cps	1.0 cps	0.72 cps	0.88 cps
2	2.07 cps	2.15 cps	1.58 cps	2.13 cps
3	4.08 cps	4.19 cps	3.47 cps	4.12 cps



f_1 = 3.0 cps

f_2 = 9.57 cps

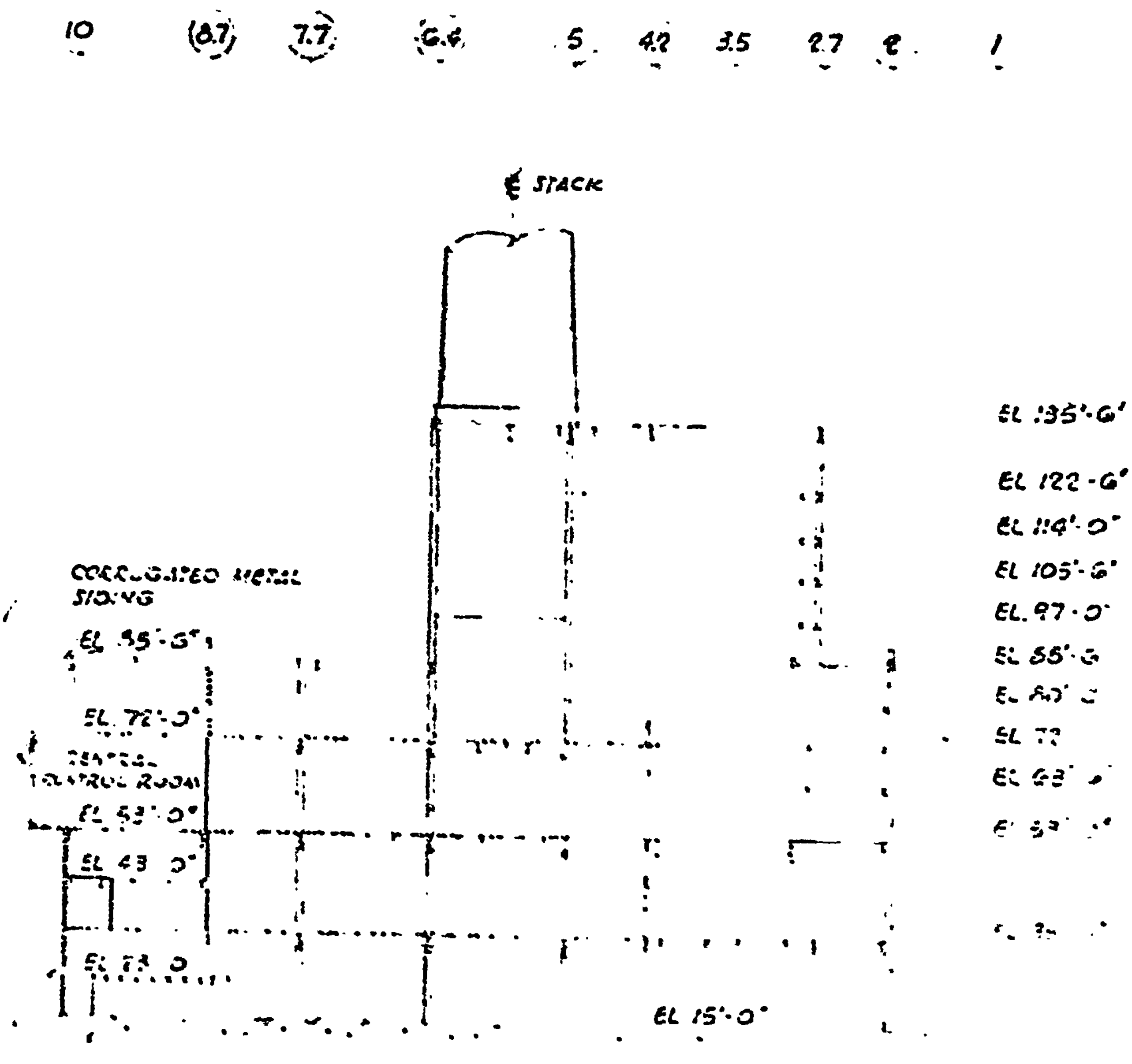
FIGURE 1.3-1 - FUEL STORAGE BUILDING NORTH-SOUTH MODEL



f_1 = 2.10 cps

f_2 = 12.97 cps

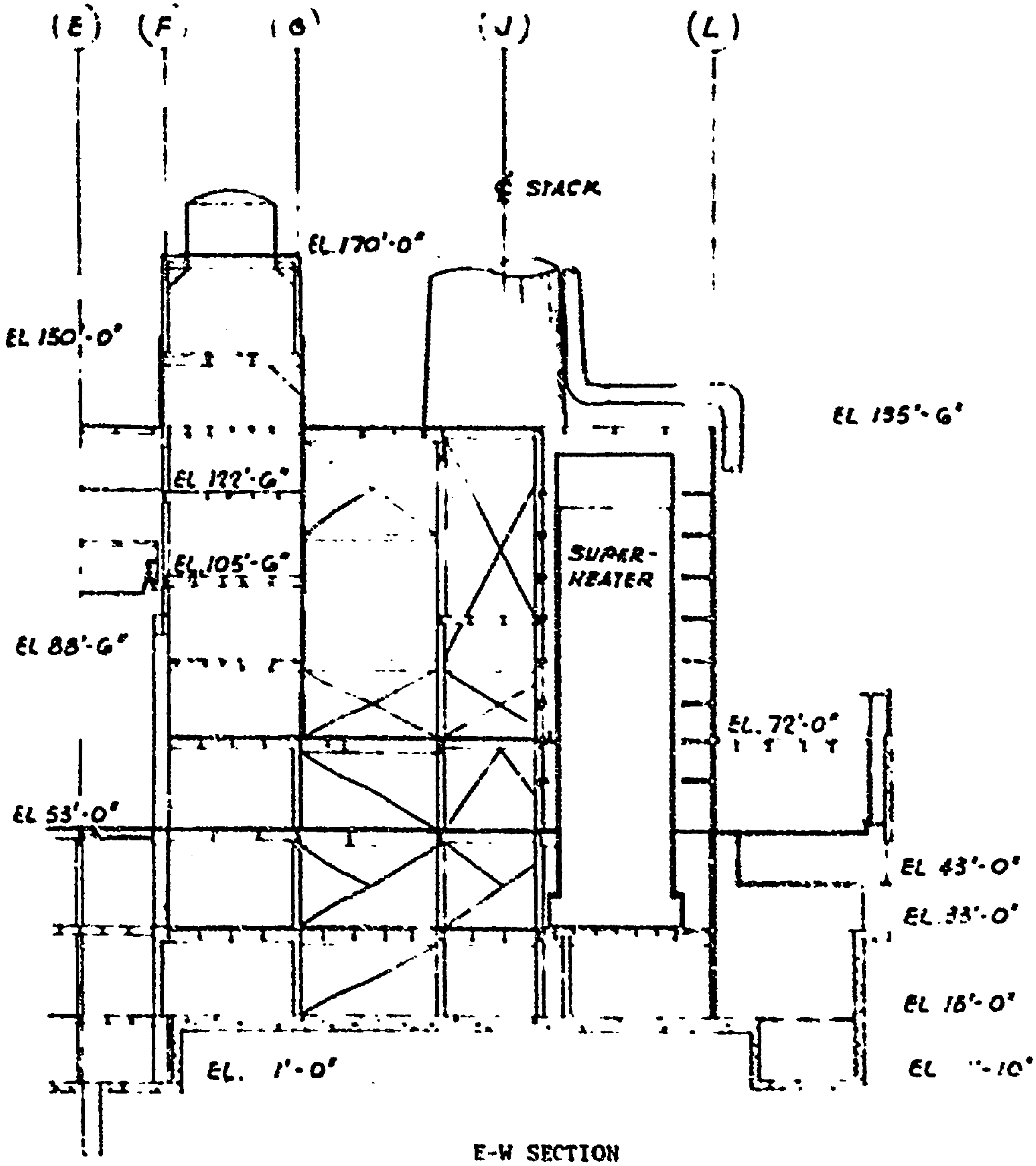
FIGURE 1.3-2 - FUEL STORATE BUILDING EAST-WEST MODEL



N-S SECTION

INDIAN POINT UNIT NO. 1
SUPERHEATER BUILDING

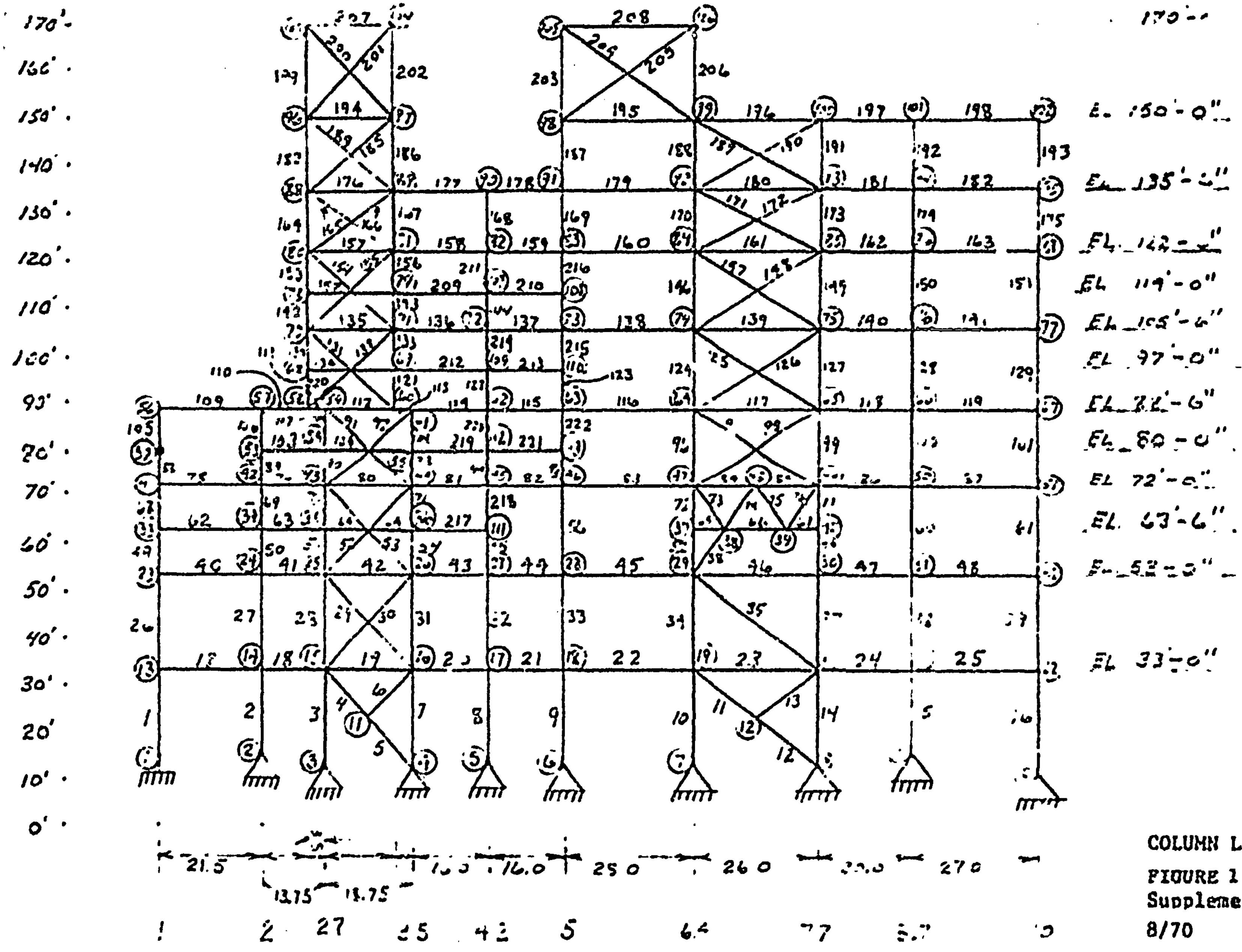
FIGURE 1.3-3
Supplement 13
8/70



E-W SECTION

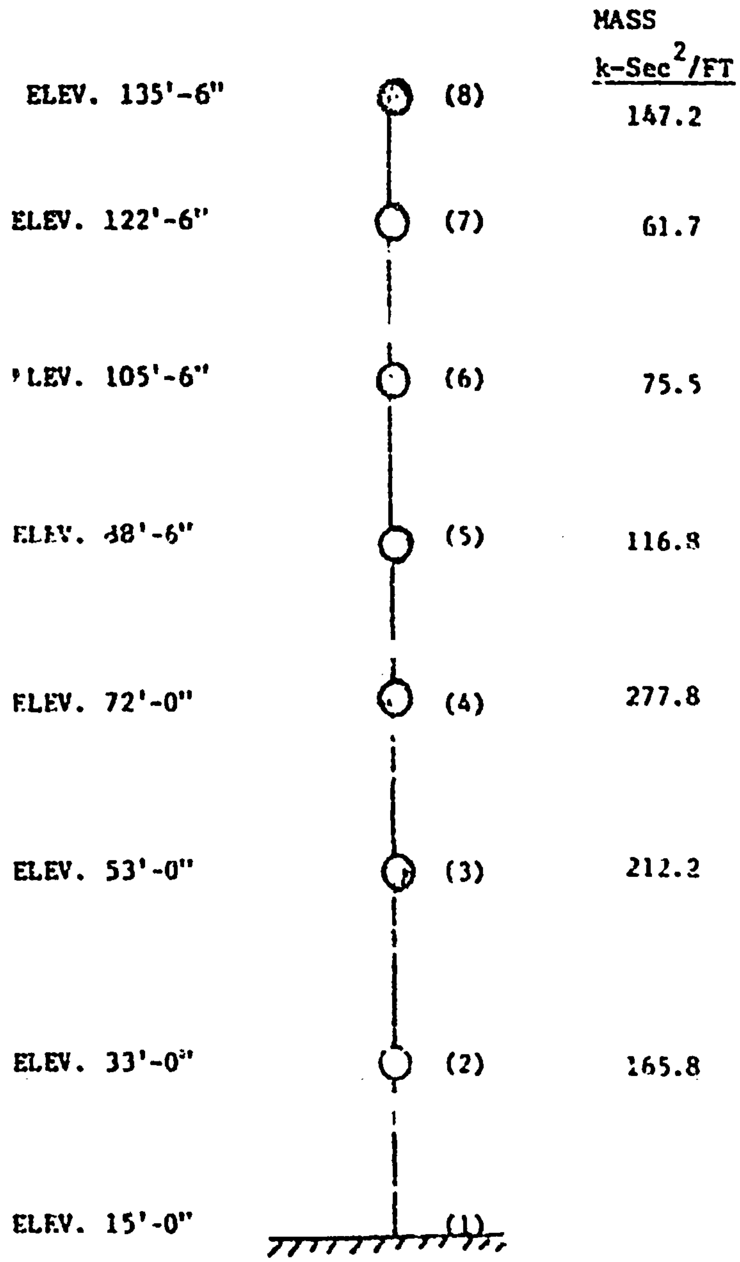
INDIAN POINT UNIT NO. 1
SUPERHEATER BUILDING

FIGURE 1.3-4
Supplement 11
8/70



COLUMN LINE "G"

FIGURE 1.3-5
Supplement 13
8/70



NOTE: Stack Lumped at Mass Point 8

Mode	CPS N-S Freq.	CPS E-W Freq.
1	0.72	0.95
2	1.58	2.07
3	3.48	4.07
4	4.65	5.18
5	6.0	7.0
6	7.15	8.0
7	8.25	9.7

NOTE: STIFFNESS MATRICES USED TO DEFINE THE STIFFNESS RELATIONSHIP BETWEEN MASS POINTS IN THE E-W AND N-S DIRECTION

FIGURE 1.3-6 - REPRESENTATION OF LUMPED MASS MODEL OF SUPER HEATER BUILDING USED IN DYNAMIC ANALYSIS.

QUESTION 1.4

Clarify whether backfill has, or has not, been placed against the Containment walls. If backfill is present, describe the bases for the seismic loading assumed to be generated by the backfill and used in the design of the containment. How has the non-symmetrical seismic loading of the backfill been considered?

ANSWER

Portions of the Containment structure are subjected to the effects of backfill bearing against the Containment Wall. The effects on the structure are:

a) Shear and overturning effects due to seismic response and interaction between the soil and structure and,

b) Discontinuity effects caused by the soil restraining deformation of the structure under accident pressures.

For case a) two limiting cases were investigated. The first was the case where the structure and soil are out of phase. Here it was assumed that the structure was subjected to the passive pressure of the soil with the mass of soil within the shear failure envelop accelerated against the structure with ground acceleration. In the second case the soil and structure move in phase. Here it was assumed that the structure was subjected to the active pressure of the soil within the shear failure envelope accelerated with the structure at ground acceleration.

These loads were then treated as external loads on the structure. (See Section 3.1.5 of Containment Design Report)

For case b) the structure was analyzed for the passive pressure case. The restraint of the deformation of the structure due to the soil was calculated. Vertical and circumferential bending moments due to this restraint were then determined. Reinforcing bar stresses were calculated and found to be minor. This analysis was then verified by a finite element analysis.

In this analysis, full contribution of the backfill was assumed. During the course of construction it became necessary, to facilitate construction, to build a retaining wall in a substantial area of the backfill. This retaining wall extends over fifty feet in plan and includes all of the high fill points assumed in the analysis and design. It can therefore be concluded that the analysis was conservative in that the backfill effects on the completed structure would be only a fraction of that assumed in the original design.

QUESTION 1.5

The FSAR discussion on primary system supports (pp 5.1.5-1 through 9) does not indicate whether these were designed for a combination of seismic and accident loads. Provide resultant stresses and deformations under such a combination, including jet forces, and furnish sketches showing the support details for the equipment discussed.

ANSWER

The primary system supports were not designed to resist combined seismic and accident loads. They were designed as statically uncoupled component supports as described in the FSAR.

A complete reactor coolant system loop, including the steam generator and the reactor coolant pump supports, has been analyzed for combined dead, seismic and blowdown loads. Stresses were determined by means of the three dimensional frame computer program, Strudl. The dead load assumed is the flooded weight of the component. The seismic load considered is 0.6g horizontal acceleration times the flooded mass of the component at the center of gravity of the component acting in the N-S, E-W, NW-SE and SW-NE directions analyzed separately. The horizontal earthquake component acting on the steam generator is assumed to be carried by the upper steam generator lateral support located at the center of gravity of the steam generator. The vertical component of earthquake is assumed equal to 0.4g acting simultaneously with the horizontal load at the c.g. of both the pump and steam generator. The system was analyzed for each separate accident or pipe rupture resulting in a jet load equal to 1500 kips as shown in Figure 5-1.1.

The combined dead plus seismic plus accident maximum resultant member axial stress and axial plus bending stress (in parenthesis) for the steam generator and pump supports are shown in Figures 5-1.4 through 5-1.11 (stressed are expressed in ksi.) The section views of the support shown can be identified by the isometric views of the pump and steam generator supports shown in Figures 5-1.2 and 5-1.3. Negative values indicate compression and positive values tensile stress. Since response of the primary systems is elastic, deformations are very small and were not considered as design parameters required to verify the design adequacy of the supports.

It should be noted the stresses shown are not for a particular combined blowdown or seismic load case but rather the worst combination for a given member hence the values shown are upper limits for each member and could not in fact actually occur in the combination shown. It should also be noted that the primary support structures are designed as trusses rather than frames hence the bending stresses indicated are secondary in nature.

QUESTION 1.6

As stated in Appendix A of the FSAR Class I structures and equipment are designed such that for a ground acceleration of 0.15g a safe shutdown can be achieved. Please indicate in detail the criteria for functional adequacy in this case for structures, equipment, piping, instrumentation and controls, and the manner in which these criteria are met.

ANSWER

No loss of function implies that rotating equipment will not freeze, pressure vessels will not rupture, supports will not collapse under the load, systems required to be leak tight will remain leak tight and components required to respond actively (such as valves and relays) will respond actively.

The criteria, for functional adequacy of the structures, states stresses will not exceed yield when subjected to a 0.15g ground acceleration.

The manner in which this criteria was met was by limiting stresses in Class I structures to meet the above criteria.

For all Class 1 piping and its supports, the criteria for functional adequacy and the manner in which the criteria are met is the following: With a ground acceleration of 0.15g horizontal, the spectral acceleration corresponding to the maximum point on the 0.5% critical damping response curve was used to calculate an equivalent static force imparted to the pipe at its support points. This resulted in a seismic design load approximately equal to 0.6W horizontally and 0.40W vertically taken simultaneously. The sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code allowable. The stresses in the pipe supports and hangers were likewise limited to 1.2 times code B31.1 allowable.

Since all the buildings containing Class I piping are essentially rigid structures, no amplification is expected.

For Class I equipment and tanks the same method was used to arrive at an equivalent static force. In each case, the total of seismic and normal stresses was limited to the applicable code allowable.

QUESTION 1.7

As stated in Appendix A of the FSAR Class I structures and equipment are designed such that for the combination of normal loads plus a 0.10g earthquake the stresses are within code allowable. Please list the codes used for structures, equipment, piping, instrumentation and controls, and identify those elements for which a stress increase has been used, as permitted by the codes.

ANSWER

Structures, other than the Containment Structure, were designed only for the combination of normal loads plus response to a 0.15g earthquake. Allowable stresses in this case were limited to the yield point of the material.

For the containment design see the Containment Design Report.

All Class I piping was designed in accordance with the USAS Code for Pressure Piping B31.1.0 for response to 0.15g earthquake. The Refueling Water Storage Tank and Condensate Storage Tank were designed in accordance with the stress limitations of American Water Works Association Standard D 100.

Class I equipment, associated with the primary reactor coolant loop, were designed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels for response to a 0.15g earthquake.

QUESTION 1.8

Show that seismic design considerations have been adequately included in the design of the battery racks.

ANSWER

Seismic design considerations have been adequately included in the design of the battery racks as shown in the design calculations given below.

Summary of Stress Analysis of Battery Racks S44372 and S44374

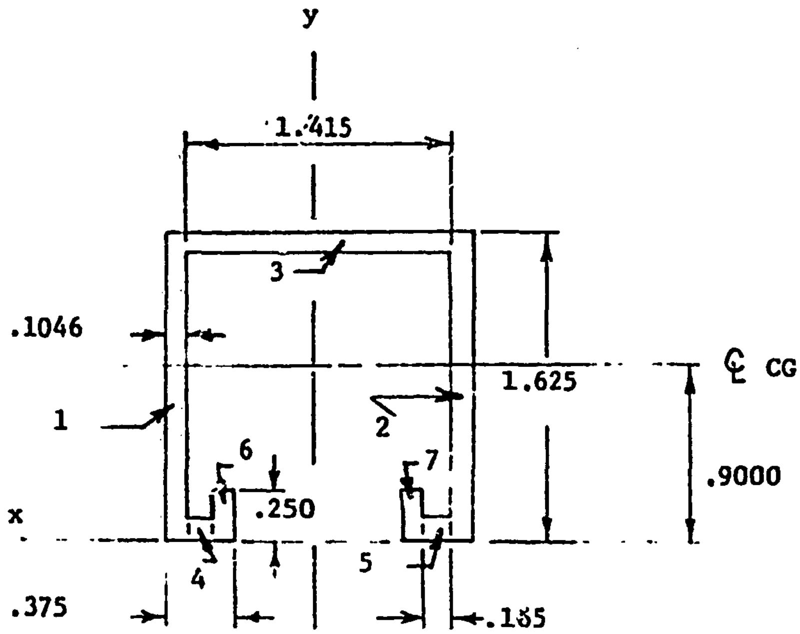
The stresses due to static loads alone are sufficiently low. The highest value is 7840 lb/in^2 at supports 2 and 4 on rack S44374 (Page 1.8-7). The maximum deflection is at the center of the spans between supports 1 and 2, 4 and 5. (.0187 in) (Page 1.8-7)

For response in the vertical direction, the natural frequency of the stringers on both racks fall in a range giving acceleration factors of 1.47 (S44372) and 1.57 (S44374). The maximum stress at supports 2 and 4 on rack S44374 then becomes 12,300 lbs/in^2 and the deflection between supports 1 and 2, 4 and 5 becomes .0294 in. (Page 1.8-11).

For vibration in the horizontal transverse direction it was assumed that the full acceleration factor of .86g acted on the rack rather than to try to calculate the natural frequency of the rack in this direction. The maximum stress in the bracing is 18,017 lbs/in^2 (Page 1.8-12).

For vibration in the horizontal longitudinal direction it was again assumed that the full .86g acted on the rack. The maximum stress is in the cross braces and has a maximum value of 8350 lbs/in^2 (Page 1.8-14).

In all three directions, both static and dynamic loads produce stresses below an allowable (20,000 psi). Assumptions made in all cases were those that would give the worst condition of loading.



						Area
1.	(.1046)	(1.625)	(.8125)	=	.1381	.1700
2.	(.1046)	(1.625)	(.8125)	=	.1381	.1700
3.	(.1046)	(1.415)	(1.573)	=	.2328	.1480
4.	(.165)	(.1046)	(.0523)	=	.0009	.0173
5.	(.165)	(.1046)	(.0523)	=	.0009	.0173
6.	(.1046)	(.250)	(.125)	=	.0033	.0262
7.	(.1046)	(.250)	(.125)	=	.0033	.0262
					.5174 in^3	.5750 in^2

Center of Gravity $= \frac{.5174}{.5750} = .8998 \approx \underline{\underline{.9000}}$

Moment of Inertia about CG

$$\int_{x_1}^{x_2} wx^2 dx = \frac{1}{3} wx^3 \Big/_{x_1}^{x_2} \qquad w = \text{width of section}$$

SECTION 1-2 UPPER

$$I = \frac{1}{3} (.2092)\ (.725)^3 = .0265 in^4$$

SECTION 1-2 LOWER

$$I = \frac{1}{3}(.2092)\ (.9)^3 = .0508 in^4$$

SECTION 3

$$I = \frac{1}{3}(1.415)\ [\underbrace{(.725)^3 - (.621)^3}_{(.142)}] = .0670 in^4$$

SECTION 4-5

$$I = \frac{1}{3}(.315)\ [\underbrace{(.9)^3 - (.796)^3}_{(.225)}] = .0237 in^4$$

SECTION 6-7

$$I = \frac{1}{3}(.209)\ [\underbrace{(.9)^3 - (.65)^3}_{(.454)}] = .0317 in^4$$

Total Moment of Inertia about CG

$$I_{GG} = \underline{\underline{.200 in^4}}$$

Check using formula for hollow rectangle

$$I = \frac{D^4 - d^4}{12} = \frac{(1.625)^4 - (1.416)^4}{12} = .247\ in^4$$

Subtract void between Section 6 and 7

$$I = \frac{1}{3}(.875)\ [\underbrace{(.9)^3 - (.796)^3}_{.225}] = .0657\ in^4$$

Add turned up portion of 6 and 7

$$I = \frac{1}{3}(.209)\ [\underbrace{(.796)^3 - (.65)^3}_{(.230)}] = .0160\ in^4$$

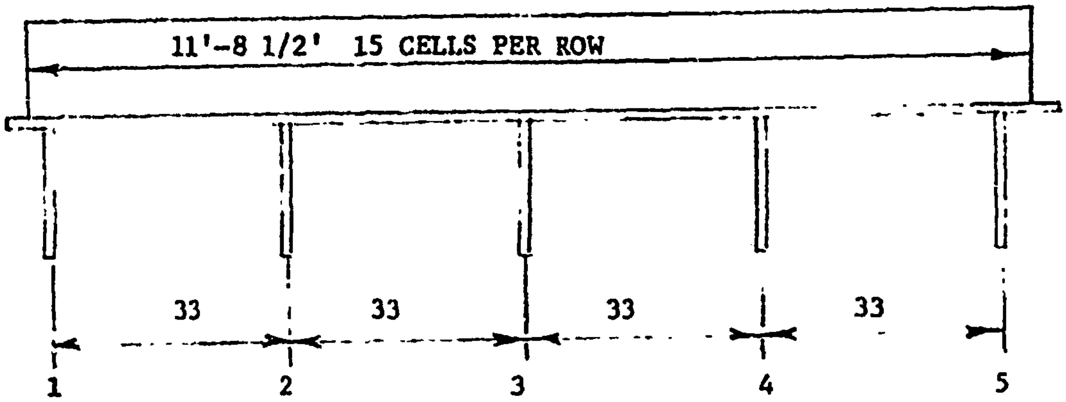
Total Moment of Inertia

$$I_{CG} = .247 + .016 - .065 = \underline{\underline{.200}}\ in^4$$

WT = 2 lbs/ft (3 stringers) = $\underline{\underline{.5\ lbs/in}}$ To be added to battery wt.

A. BATTERY #22 FTA-17 60 Cells 227 lb. each

Rack No. S-44372



TOTAL LOAD 15(227) = 3405 lbs.

$$\frac{3405 \text{ lbs}}{140.5 \text{ in}} = 24.2 \text{ lbs/in}$$

Since the load is uniform the moments at supports 2,3 and 4 are equal and there is no moment at supports 1 and 5. Each span 2-3 and 3-4 may be considered a beam uniformly loaded and fixed at each end. Each span 1-2 and 4-5 may be considered a beam uniformly loaded fixed at one end free at one end. The moment of inertia of the beam is 3(.200) = .600 since there are 3 stringers.

Beams 2-3 and 3-4

$$M_{max} = \frac{w\ell^2}{12} = \frac{24.7\ (33)^2}{12} = 2240 \text{ lbs. in. at supports}$$

Flexural Stress at Extreme Fiber

$$f_{ends} = \frac{MC}{I} = \frac{2240(.9)}{.6} = \underline{\underline{3360 \text{ lbs/in}^2}} \text{ at supports}$$

$M_{center} = \frac{w\ell^2}{24} = 1120$ lbs in

$f_{center} = \underline{\underline{1680 \text{ lbs/in}^2}}$

Deflection at Center

$$\Delta y = \frac{w\ell^4}{384EI} = \frac{24.70\ (33)^4}{384\ (30\text{x}10^6).6} = \underline{\underline{.0042 \text{ in}}}$$

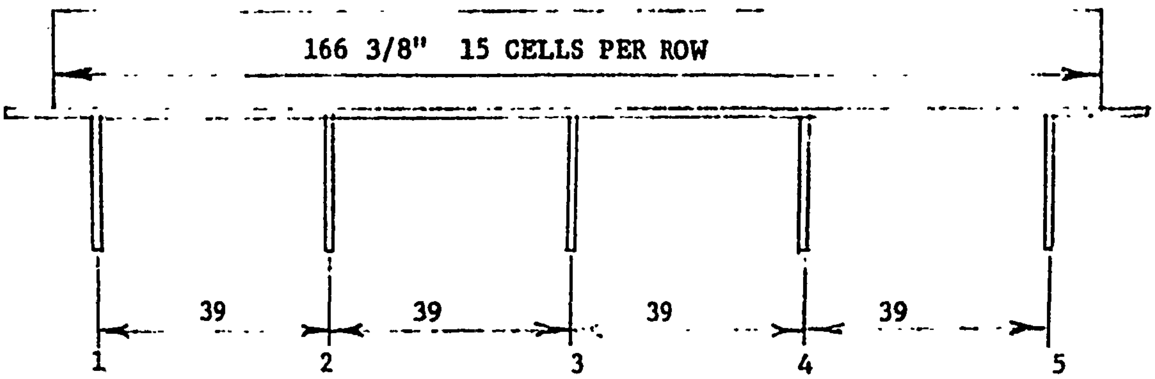
Beams 1-2 and 4-5

$$M_{max} = \frac{w\ell^2}{8} = \frac{24.7\ (33)^2}{8} = 3380 \text{ lb in at fixed end}$$

$$f_{max} = \frac{MC}{I} = \frac{3380\ (.9)}{.6} = 5050 \text{ lbs/in}^2$$

$$\Delta y = \frac{w\ell^4}{185EI} = \frac{24.7\ (33)^4}{185\ (30\text{x}10^6)(.6)} = \underline{\underline{.0088 \text{ in}}}$$

B. Battery #21 FTA-23 60 Cells 299 lbs each
Rack No. S 44374



The same assumptions apply to rack S 44374 as applied to Rack S 44372.

Total Load 15(299) = 4485 lbs.

$$\frac{4485}{166.375} = 26.9 \text{ lbs/in}$$

Beams 2-3 and 3-4

$$M_{max} = \frac{w\ell^2}{12} = \frac{27.4(39)^2}{12} = 3475 \text{ lb. in. at supports}$$

$$f_{max} = \frac{MC}{I} = \frac{3475(.9)}{.6} = \underline{\underline{5220 \text{ lbs/in}^2}} \text{ at supports}$$

$$M_{center} = \frac{w\ell^2}{24} = 1737 \text{ lb. in.}$$

$$f_{center} = \frac{MC}{I} \quad \frac{1737(.9)}{.6} = \underline{\underline{2610 \text{ lbs/in}^2}}$$

Deflection at Center

$$\Delta y = \frac{w\ell^4}{384\ EI} = \frac{27.4(39)^4}{384(30\times10^6).6} = \underline{\underline{.009 \text{ in}}}$$

Beams 1-2 and 4-5

$$M_{max} = \frac{w\ell^2}{8} = \frac{27.4(39)^2}{8} = 5220 \text{ lb in at fixed end}$$

$$f_{max} = \frac{MC}{I} = \frac{5220(.9)}{.6} = \underline{\underline{7840 \text{ lbs/in}^2}}$$

$$\Delta y = \frac{w\ell^4}{185EI} = \frac{26.9(39)^4}{185(30x10^6)(.6)} = \underline{\underline{.0187 \text{ in}}}$$

C. TRANSVERSE VIBRATIONAL STRESS

The rack may be considered a single uniformly loaded beam simply supported at 1 and 5. Supports 2, 3, and 4 limits the modes of vibration to multiples of 4.

RACK S 44374

n = 4

n = 8

$$w = A_n \sqrt{\frac{EI}{\mu \ell^4}}$$

$$w = 158 \sqrt{\frac{30 \times 10^6 \ (.6)}{.855 \ (156)^4}}$$

$$w = 158 \sqrt{\frac{18 \times 10^6}{5.02 \times 10^8}}$$

$$w = 158 \sqrt{3.58 \times 10^{-2}}$$

$$w = 158 \ (.189)$$

$$w = 29.9$$

$$f_4 = \frac{29.9}{2\pi} = 4.75/\text{sec}$$

$$T_4 = 1/f_4 = \frac{1}{4.75} = \underline{\underline{.2105 \text{ sec}}}$$

$$f_8 = 4.75 \left(\frac{630}{158}\right) = 18.95/\text{sec}$$

$$T_8 = 1/18.95 = \underline{\underline{.0528 \text{ sec}}}$$

$A_8 = 64\pi^2 = 630$

$A_4 = 16\pi^2 = 158$

w = angular frequency

a = mode of vibration

$\mu = \frac{w}{g \ell}$ = mass/unit length

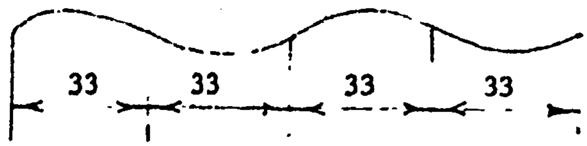
ℓ = length

$\mu = \frac{27.4 \text{ lb/in}}{32 \text{ ft/sec}^2} = .855$

$f = \frac{w}{2\pi}$ = frequency

T = period of vibration

RACK S 44372



$$w = A_4 \sqrt{\frac{EI}{\mu \ell^4}} \qquad \mu = \frac{24.7}{32 \text{ ft/sec}} = .775$$

$$w = 158 \sqrt{\frac{30\text{x}10^6(.6)}{.755(132)^4}} \qquad A_4 = 158$$

$$w = 158 \sqrt{\frac{18\text{x}10^6}{2.36\text{x}10^8}}$$

$$w = 158 \sqrt{7.63\text{x}10^{-2}}$$

$$w = 158\ (.276) = 43.6$$

$$f_4 = \frac{w}{2\pi} = \frac{43.6}{2\pi} = 6.95/\text{sec}$$

$$T_4 = \frac{1}{f4} = \frac{1}{6.95} = \underline{\underline{.144 \text{ sec}}}$$

$$f_8 = 6.95\ \left(\frac{630}{158}\right) = 27.7/\text{sec}$$

$$T_8 = 1/27.7 = \underline{\underline{.0361 \text{ sec}}}$$

RACK S 44372 0% Critical Damping

T_4 = .144 a = .47 g a = acceleration

T_8 = .036 a = .13 g

Consider the increase on static loads due to worst condition where a = .47 g at 0% critical damping

$W' = 24.7 \text{ lbs/in} + .47(24.7 = 36.3 \text{ lbs/in}$

$\frac{w'}{w} = \frac{36.3}{24.7} = 147\%$

Beams 2-3 and 3-4

$f_{ends} = 1.47 (3360) = \underline{\underline{4950 \text{ lb/in}^2}}$

$\Delta y = 1.47 (.0042) = \underline{\underline{.0062 \text{ in}}}$

Beams 1-2 and 4-5

$f_{max} = 1.47 (5050) = \underline{\underline{7093 \text{ lbs/in}^2}}$ Highest stress due to vertical vibration

$\Delta y = 1.47 (.0088) = \underline{\underline{.0130 \text{ in}}}$

Rack S 44374

T_4 = .2105 a = .57g

T_8 = .0528 a = .17g

$w' = 27.4 \text{ lb/in} + .57(27.4) \text{ lb/in} = 4.30 \text{ lbs/in}$

$\frac{w'}{w} = \frac{43}{27.4} = 157\%$

BEAMS 2-3 and 3-4

$f_{max} = 1.57(5220) = \underline{\underline{8200\ lb/in^2}}$

$\Delta y = 1.57(.009) = \underline{\underline{.014\ in.}}$

BEAM 1-2 and 4-5

$f_{max} = 1.57(7840) = \underline{\underline{12,300\ lb/in^2}}$ HIGH STRESS DUE TO VERTICAL VIBRATION

$\Delta y = 1.57(.0187) = \underline{\underline{.0294\ in}}$

MOMENT OF INERTIA OF STRINGER ABOUT Y AXIS

$$I = \frac{1}{3}(.1046)(.7075)^3 = .0123\ in^4$$

$$I = \frac{1}{3}(1.625)[(.8125)^3 - (.7075)^3] = .0986\ in^4$$

$$I = \frac{1}{3}(.1046)[(.7075)^3 - (.5425)^3] = .0067\ in^4$$

$$I = \frac{1}{3}(.250)[(.5425)^3 - (.4379)^3] = .0063\ in^4$$

$$.1239\ in^4$$

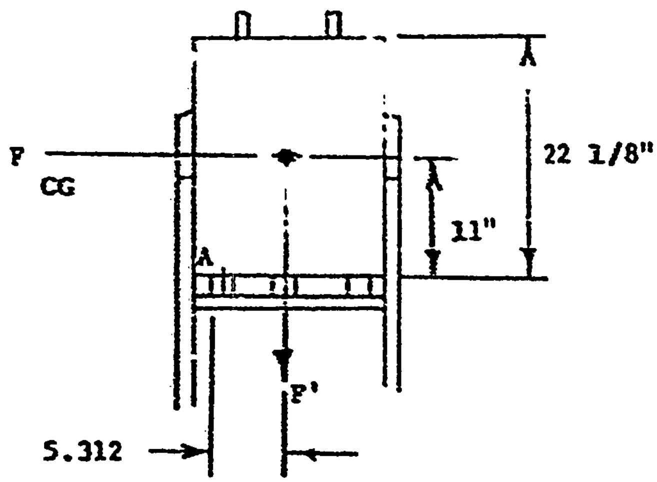
$$I_{y_{Total}} = 2(.1239) = \underline{\underline{.2478\ in^4}}$$

Assume that each stringer acts independently in restraining 1/3 of a total transverse horizontal force applied under the worst condition of 0% critical damping and .86g acceleration shown on the horizontal response spectra graft. This is a very conservative assumption as the moment of inertia of all three stringers acting together would be greater than 3 times their individual moments of inertia. Also their natural frequencies are being assumed to be those that give the highest value of acceleration computing only for beam 1-2 and 4-5 on rack S44374 which will have the highest flexural stress.

$$LOAD = .86\left(\frac{27.4\ lb/in}{3}\right) = \underline{\underline{7.85\ lbs/in}}$$

$$M_{max} = \frac{w\ell^2}{8} = \frac{7.85(39)^2}{8} = 1475\ lb/in$$

$$f_{max} = \frac{MC}{I} = \frac{1475(.8125)}{.2478} = \underline{\underline{4840\ lb/in^2}}$$



Assume the max. g load of .86 acts in the horizontal transverse direction

$F = 4485(.86) = 3850$ lbs.

Moment about point A

$M_A = F(11) - F'(5.312)$

$= .3850(11) - 4485(5.312)$

$= 42,350 - 23,825$

$= 18,525$ in. lbs.

The moment being restrained by each brace is then:

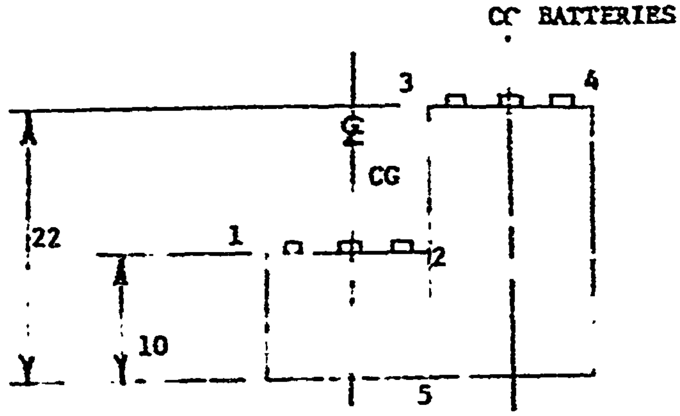
$M = \frac{18,525}{10} = 1,852$ in lbs.

Each brace is 2 x 1 x 3/16 moment of inertia = .07 in^4 distance from CG to extreme fiber = .68 in. the maximum stress in each brace is then

$f = \frac{MC}{I}$

$= \frac{1852(.68)}{.07} = \underline{\underline{18,017 \ lb/in^2}}$

Rack 544374



Total load acting at each ℄ along the length of the rack with 1g = 4485.

The horizontal load at points 1, 2, 3 and 4 is then:

$$\frac{4485}{2} = \underline{2243 \text{ lbs.}}$$

The max. moment will be on the center supports

$M = 2243(22) + 2243(10) =$

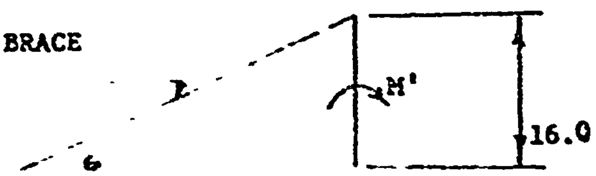
$= 49346 + 22430$

$= 71,776$ in. lbs.

Moment on each of 5 supports

$$M' = \frac{71,776}{5} = 14380 \text{ in. lbs.}$$

CROSS BRACE



$$\text{TAN } \theta = \frac{16}{33} = .485$$

$$\theta = 26°$$

Force restrained by cross brace

$$F = \frac{14,380}{16} = 900 \text{ lbs.}$$

Tension in cross brace

$T \cos \theta = 900$

$T = \frac{900}{.898} = 1000$ lbs.

Area of cross brace = 1.5 x .125 = .12 in^2

Stress in brace

$S = \frac{1000}{12} = 8350$ lbs/in^2

QUESTION 1.9

Describe the mathematical models and methods used for the seismic design of Class I structures, equipment, piping and instrumentation and controls. Explain how the elasticity of the structures, and the damping have been evaluated. If modal analysis has been used, indicate for every important structure, piping system, or equipment how many modes have been considered and describe how the damping was evaluated for each mode. By how much does the use of smooth response spectra underestimate the true response of Class I structures and equipment?

Discuss how closely these mathematical models represent the actual conditions, especially the effect of the following: Non-linear behavior of the actual sturctures, piping and equipment; effect of appendages (small masses elastically attached to large masses), clearances (gaps) at equipment restraints, and supports; and variable friction.

ANSWER

The variety of design problems associated with the seismic analysis of all Class I structures, systems, and equipment were approached by various methods. The design method for Class I structures is presented in Appendix A and is supplemented below.

With the exception of the containment, PAB, and Electric Cable Tunnel no dynamic analyses were performed on Unit No. 2 structures, hence no mathematical models were developed. The following methods were used in the seismic design of Class I structures.

A. Containment Building

See Sections 2.0 and 3.0 of the Containment Design Report for Indian Point Unit No. 2.

1. Steel

In the design of the steel 100% of the dead load and 50% of the live load was considered. The peak of the response curve for .15g ground acceleration and 1.0% critical damping was used to obtain the seismic forces which were distributed by the method described in the Containment Design Report for Indian Point Unit No. 2, and resisted by the bracing. The 1.0% critical damping is conservative since the structure is shop welded and field bolted to the columns. The actual critical damping value would be between 1.0% (welded) and 2.5% (bolted). A. 1/3 increase over working stress was allowed in the design of the bracing.

2. Concrete

In the design of the concrete 100% of the dead load and 50% of the live load was considered. The Modified Rayleigh Method was used to calculate the natural period and the base shear was distributed by the same method described in the Containment Design Report for Indian Point Unit No. 2. The forces determined from the response curve for a .15g ground acceleration with 5% critical damping were applied at the node points, where the masses were lumped for the Rayleigh approach. These loads were resisted by the vertical walls, which acted as shear walls, and horizontal reinforcing which resisted the moment. A 1/3 increase on working strength design allowable stresses was conservatively assumed.

C. Control Building

The dead load and equipment loads were considered. The period was determined from the formula T = 0.1 n where n = no. of stories (Design of Multistory Reinforced Concrete Building for Earthquake Motions by N. M. Newmark et. al.). The response curve for .15g ground acceleration with 2 1/2% critical damping was used to determine the base shear. This base shear was distributed at the floor levels by the same method described in the Containment Design Report for Indian Point Unit No. 2 and resisted by a rigid frame structure with a 1/3 increase on allowable working stresses. The design was controlled by a deflection limitation due to the adjacent Unit No. 1 Control Building.

D. Diesel Generator Building

Due to the light weight of the structure, the wind load controlled the design.

E. Fan House

100% of the dead load and 50% of the live load was considered. The peak of the response curve for .5g ground acceleration with 5% critical damping was used. The loads were distributed as described in the Containment Design Report for Indian Point Unit No. 2. A 1/3 increase in allowable working stresses was allowed.

F. Boric Acid Evaporator Building

100% of the dead load was considered. For method of design see Fan House. Without allowing a 1/3 stress increase for seismic design the controlling factor for reinforcing design was the minimum temperature steel requirements of the ACI-318 Building Code.

G. Intake Structure

100% of the live and dead load were considered. The peak of the response curve for .15g ground acceleration with 5% critical damping was used to obtain the seismic loads. The effect of water sloshing was considered in the earthquake analysis (per TID 7024 "Nuclear Reactors and Earthquakes" Section 6.5). The controlling factor in the design of the Intake Structure was the service load with the worst combination being one chamber empty and the adjacent chamber filled with water.

H. Waste Hold-Up Tank Pit

100% of the dead load and 50% of the live load was considered (including the tank dead weight on the roof). The peak of the response curve for .15g ground acceleration with 5% damping was used to determine the base shear. Using working strength limits for the seismic design, service loads controlled the design of the top slab. The bottom slab and wall of the Pit were designed for earthquake loads with stresses limited to yield times Ø factors recommended in Section IV-B of the ACI-318-63 "Building Code". Consideration was given to the tanks in the Pit when designing the base slab.

I. Spent Fuel Pit

See Answer to Question 1.10 in Supplement to the Indian Point Unit No. 2 FSAR for the amplification factors on the .15g ground acceleration. The seismic loads, as determined in TID-7024 "Nuclear Reactors and Earthquakes" Section 6.5, were resisted by the reinforced concrete walls and base slab. Working stresses were used except for the moment at the base of the walls where ultimate strength design was considered with stresses limited to ϕfy.

J. Electrical Penetration Tunnel

The peak of the response curve for .15g ground acceleration with 5% critical damping was considered using working strength design limits. The load was considered to act at 2/3 L where L = the height of the tunnel. Temperature Steel considerations controlled the design of the concrete while service loads controlled the structural steel.

K. Pipe Penetration Tunnel

100% of the dead load, plus 50% of the live load was considered. The peak of the response curve for .15g ground acceleration with 5% damping was used to find the shear which was considered as a concentrated load applied at the top slab of the tunnel. A 1/3 increase on working stress allowables was used in the design.

L. Electrical Cable Tunnel

100% of the dead load, 50% of the surcharge and 50% of live load in the tunnel was considered. The Modified Rayleigh Method was used to determine the natural period and the loads were distributed as described in the Containment Design Report for Indian Point Unit No. 2. The response curve for .15g ground acceleration with 5% critical damping was used. A 1/3 stress increase was permitted on working stress allowables when considering the effect of seismic loads.

M. Shield Wall

The peak of the response curve for .15g ground acceleration with 5% critical damping was used. The pipe break loads controlled the design.

N. Retaining Wall At Equipment Entrance

Soil pressures controlled the design of the wall when the effect of surcharge while loading the reactor was considered.

O. Primary Water Storage Tank and Refueling Water Tank Foundation

The seismic loads on the circular wall and center pier were those supplied by the tank manufacturer. The shear force from the earthquake on the water in the tank was applied at 3/4 L above the top slab. The shear force from the earthquake on the tank was applied at L/2 above the top slab, where L = the height of the tank. The horizontal shear force from the earthquake effect on the dead weight of the foundation was determined by using the peak of the response curve for .15g ground acceleration with 5% critical damping. A triangular distribution was used. The earthquake effect of the backfill was also considered. The load was applied to the walls as the resultant of a triangular pressure distribution. The stresses were limited to working strength design limits. The temperature steel considerations controlled the design of the walls and center pier.

P. Condensate Water Storage Tank Foundation

The seismic loads on the spread footing foundation were those supplied by the tank manufacturer. The shear forces from the earthquake on the water in the tank was applied at 3/4 L above the footing. The shear force from the earthquake on the tank was applied at L/2 above the top of the footing, where L = the height of the tank. The stresses were limited to working strength design limits.

A multi degree-of-freedom modal analysis was performed on all Class I structures for Indian Point Unit No. 3. The results indicated that all except the containment structure were rigid. The only significant differences between the structural design of Units 2 and 3 Class I buildings are the Control Building and the steel structural portion of the Primary Auxiliary Building for Indian Point Unit No. 2 which are flexible steel structures. On Unit No. 3 they are rigid concrete st tures. All Class I structures on Indian Point Unit No. 2 except the Control Building and the Containment shell are rigid and move with zero period ground acceleration. However, the design of all Class I structures on Unit No. 2 were standardized and based on the peak acceleration of the ground response spectrum which is extremely conservative for rigid structures.

In the preceeding designs, limits have been placed on stresses to assure that all structures will respond elastically to the earthquake. If for some reason inelastic response were to occur, the period of the structure would be expected to increase. Since the majority of the structures were designed for the peak of the response curve the effect of any change in period would be to decrease the coefficient of spectral acceleration and thus lower all shears and moments.

For a discussion of the evaluation of damping factors, see the Answer to Question 9a in Supplement No. 2 to the Indian Point Unit No. 2 PSAR.

For validity of the response spectra in connection with the site, see the Answer to Question 2.7e in Supplement No. 2 to the Indian Point Unit No. 3 PSAR.

For Unit No. 2 there were no mathematical models used for structures, except as indicated by the use of the Rayleigh Method.

The method used for the design of Class 1 equipment and piping is described in the answer to Question 1.6.

Mathematical models are not used for seismic design of instrumentation. Ability to withstand the seismic condition is determined by actual vibration type testing of typical instrumentation equipment under simulated seismic accelerations to demonstrate its ability to perform its functions. The seismic testing is reported in Westinghouse proprietary report WCAP-7397-L titled "Topical Report Seismic Testing of Electrical and Control Equipment", E. L. Vogeding, dated January, 1970. The following is a summary:

> In a nuclear power plant, electrical and control equipment which initiates reactor trips, actuates safeguards systems and/or monitors radioactive releases from the plant must be capable of performing their functions during and after an earthquake that has occured at the plant site. To demonstrate the ability of this equipment to perform under earthquake conditions, selected types of this essential equipment representative of all protection and safeguards circuits and equipment were subjected to vibration tests which simulated the seismic conditions for the "low seismic" class of plants.* During the tests, equipment operation was monitored to prove proper performance of functions. The results show that there were no electrical malfunctions. Based on these results, it is concluded that the equipment will perform their design functions during as well as following a "low seismic" earthquake.

The locations of this protection and safeguards control and electrical equipment in Indian Point Unit 2 have been identified. The most adverse location, seismically, is the control building floor at elevation 53', which supports the nuclear instrumentation, radiation monitoring, process instrumentation and safeguards logic racks. Dynamic analyses of this building for the plant design basis earthquake of 0.15g show that the significant horizontal and vertical accelerations of this floor are within the specified low seismic test envelopes given in WCAP-3797-L.

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*Those having a Design Basis Earthquake horizontal acceleration less than or equal to 0.2g.

Seismic analysis of Class I equipment including heat exchangers, pumps, tanks, valves, motors, and electrical equipment components are analyzed using one of four methods: (1) Equipment which is rigid and rigidly attached to its support structure is analyzed for a g loading equal to the peak acceleration of the supporting structure at the appropriate elevation. (2) Equipment which is not rigid and therefore potential for response to the support motion exists, is analyzed for the peak of the floor response curve for appropriate damping values. (3) In some instances non-rigid equipment is analyzed using a multidegree of freedom modal analysis. All contributing modes are considered. In addition, it should be pointed out that a sufficient number of masses were included in the mathematical models to insure that coupling effects of members within the component are properly considered. The results of these analyses indicate that the models contain more masses than necessary, and that future analyses of comparable equipment could be considerably simplified by considering fewer masses. The method of dynamic analysis uses a proprietary computer code called WESTDYN. This code uses as input, inertia values, member sectional properties, elastic characteristics, support and restrain data characteristics, and the appropriate seismic response spectrum. Both horizontal and vertical components of the seismic response spectrum are applied simultaneously. The modal participation factors are combined with the mode shapes and the appropriate seismic response spectra to give the structural response for each mode. The internal forces and moments are computed for each mode from which the modal stresses are determined. The stresses are then summed using the root mean square method. (4) Type testing of selected electrical equipment has been conducted to demonstrate seismic design adequacy as described in WCAP 7397-L.

For the analysis of equipment to resist the vertical seismic component, 2/3 of the horizontal response spectrum curves are used to determine the acceleration appropriate to the vertical frequency.

Engineered Safeguards tanks, e.g. Boric Acid, Accumulator Spray Additive and Surge, were analyzed using method (3) above, for combined horizontal and vertical seismic excitation occuring simultaneously, and in conjuction with normal loads, without exceeding allowable s esses. Hydrodynamic analyses of these tanks have been performed using the methods described in Chapter 6 of the "U.S. Atomic Energy Commission - TID 7024". The stresses for these components due to the aboved mentioned load combination were found to be within allowable limits.

Heat exchangers associated with the Engineered Safeguards Systems, e.g., Component Cooling and Residual Heat Removal, were analyzed using method (3) above, and the results show that stresses and deflections are within allowable limits.

Selected critical Engineered Safeguards valves have been analyzed using method (3) above and the results indicate that their fundamental natural frequency is sufficiently separated from the building frequency. The results further indicate that the total stress, considering all modes, is far below the allowable stress limit.

Damping values used for each item of equipment are in conformity with Table A.1-1 of Appendix A.

Non-linearities such as gaps, frictional forces, etc., have not been considered in the model. It is felt that these non-linearities would increase the percentage of critical damping thus decreasing the response.

Appendages, such as motors attached to motor operated valves, have been included in the mathematical models.

Class I piping systems for Indian Point Unit No. 2 have been designed and analyzed as described in the succeeding paragraphs. However, in an attempt to correlate the simplified method of analysis suggested by the AEC for the H. B. Robinson Nuclear Generating Station the following discussion is presented:

> If no dynamic analysis is performed on Class I piping systems, these systems, for H. B. Robinson plant, were to be checked to determine whether the results conform to the folllowing formula:

$$1.3^{*}\ K\ S_s + S_n \leq 1.8\ S_a$$

where:

S_s - represents seismic stress including effects of valve motors, from design calculations

S_n - represents normal primary and bending stresses for loadings other than siesmic, from design calculations

S_a - equals 1.8 times the allowable stress or yield stress, whichever is higher for code listed materials.

K - ratio of peak acceleration of floor response spectra to acceleration used in the piping design

The piping design criteria limited the deadweight and seismic stresses to $0.2\ S_a$. The longitudinal pressure stress is $0.5\ S_a$. Therefore, the above mentioned formula becomes

$$1.3\ K\ (0.2\ S_a) + 0.5\ S_a \leq 1.8\ S_a$$

Solving the K-factor becomes:

$$K = 5$$

This factor when combined with the 1.3 modal contribution factor gives a combined factor of 6.5 which is more than double the original suggested multiplier of 3.

It therefore appears that the Indian Point Unit #2 conservatively meets the criteria suggested for application on the H. B. Robinson Plant for seismic Class I piping.

However, a different and more detailed method of analysis was actually undertaken to illustrate the conservatism of design approach used for the Indian Point Plant. This approach is described in detail below:

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*The 1.3 factor was recommended by the AEC to represent the contributions of higher modes above the fundamental mode. Detailed dynamic analyses performed on Indian Point Unit No. 2, and described later, indicate that where significant stresses exist in piping systems, a more realistic modal contribution factor would be 1.1. However, for the present discussion we will adhere to the 1.3 factor for additional conservatism.

It is obviously necessary to use simplifying assumptions when performing initial design of piping systems, including restraints, rather than a dynamic analysis involving a trial and error procedure. Simplified design procedures are not uncommon and often suggested in codes, i.e. USAS B31.1 - Power Piping Code.

A complete flexibility analysis involving detailed modelling of all Class I piping systems is unnecessary if the conservatism of the simplifying assumptions used in the initial design can be demonstrated.

A "third party" review was conducted to establish the adequacy and conservatism of the original design criteria for Class I piping system as performed by the Architect/Engineer (United Engineers & Constructors, Inc.) and the seismic restraint supplier (Bergen-Paterson Pipesupport Corp.). The review invloved the following steps:

(1) Representatives from Westinghouse and United Engineers & Constructors, Inc., visited the Indian Point Generating Station - Unit No. 2 site and inspected the Class I piping systems.

(2) Based upon their best engineering judgment, representative worst-case lines were selected for detailed dynamic analyses.

(3) In exercising their engineering judgement these representatives looked for the following characteristics which would indicate possible sources of problems.

(a) Amplification due to the location and elevation in building
(b) Large concentrated masses such as overhung motor operated valves, particularly in what appear to be flexibly sections of the pipe.
(c) Complexity of configuration of the piping system itself such that application of the original design criteria would be difficult.

(d) Manual excitation of the pipe by pushing or kicking indicated excessive flexibility either in the pipe excited or the piping attached to it.

(4) The results of the dynamic analyses were compared with original design values to determine whether the design approach was conservative.

Besides analysis of the reactor coolant loop, portions of the following systems were analyzed:

(a) Safety Injection
(b) Main Steam
(c) Residual Heat Removal
(d) Service Water
(e) Accumulator Discharge
(f) Containment Spray
(g) Component Cooling

DESIGN APPROACH

The design and placement of seismic restraints was predicted on the principle of containing the seismic stresses without restricting the free thermal expansion of the piping system. The systems were designed to have sufficient flexibility to prevent the movements from causing failure of piping or anchors from overstress.

Two fundamental principles underly the design approach, namely:

(1) The system be designed such that its fundamental natural frequency does not coincide with the exciting frequency.

(2) The maximum seismic stresses in piping be less than the USAS B31.1 code allowable value. The seismic stresses were limited to 0.2 S allowable (3000 psi). This is extremely conservative since

the longitudinal pressure stress accounts for approximately 0.5 S allowable leaving a margin of safety of 0.5 S allowable which is unused. (Note this is based on a maximum allowable of 1.2 S_a.)

These fundamental principles should insure that stresses will be within code allowable stress limits, and that the piping will not go into resonance with the exciting frequency.

The design procedure was developed for Bergen-Paterson Pipesupport Corp. by Professor J. R. Curreri of the Polytechnic Institute of Brooklyn. Tables of recommended maximum spacing of supports, for straight runs of pipe, were developed. The recommended spacing of supports was modified near bends and concentrated masses (i.e. valves) to account for additional weight and flexibility.

Analysis Approach

In order to determine whether the design procedure resulted in an acceptable system, selected worst case Class I piping systems were modeled and a dynamic flexibility analysis performed. A detailed description of the method of analysis is given below.

The analysis was performed using a proprietary computer code called WESTDYN. The code uses as input, system geometry, inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and the appropriate Indian Point seismic floor response spectrum for 0.5% critical damping. Both horizontal and vertical components of the seismic response spectrum are applied simultaneously.

With this input data, the overall stiffness matrix of the three dimensional piping system is generated (including translational and rotational stiffnesses). The modal participation factors are computed and combined with the mode shapes and the appropriate seismic response spectra to give the structural response for each mode.

Each piping run is modeled as a three dimensional system which consists of straight segments, curved segments, and restraints. Straight segments are distinguished from curved segments during data output.

The computer code requires that the piping be represented by a discrete mass model. Each mass includes the contribution of both the steel encasement and conveyed fluid. Where valves or other concentrated masses exist in the piping system, these are included in the model.

Restraints are included in the model at their proper location. The directionality of the restraints is also considered.

The detailed dynamic analyses of selected worst cases Class I piping indicates that the method used to design the seismic restraints is conservative. Based on this critical review of the selected worst case systems and the consistent application of the same design procedure to all completely engineered Class I seismic systems, the seismic design of other Class I systems, not analyzed, is deemed adequate.

The maximum stresses imposed by the normal loads plus loads associated with the Design Basis Earthquake (DBE) are below 1.2 S, where S is the allowable stress limit obtained from the Power Piping Code - USAS B31.1.0 - 1955.

Some of the items of conservatism employed in the seismic design of Class I piping systems for Indian Point Unit No. 2 are: (1) The maximum longitudinal stress due to seismic excitation is limited to 0.2 S rather than the usual 0.7 S. (2) The maximum allowable stress is limited to 1.2 S. If the combination of normal and DBE loads is considered as a faulted condition, the allowable membrane and bending stresses could be chosen as those corresponding to 20% and 40% of the material's uniform strain at temperature, respectively. This would give more than a factor of 2 margin between the allowable and the maximum actual stresses. (3) A low value of the fraction of critical damping was adopted (0.5%). Dr. N. M. Newmark recommends a value of 2% for vital piping at or just below the yield point. This would reduce the maximum amplification of the ground acceleration. (4) The maximum longitudinal stresses due to pressure, deadweight, and seismic loads are presumed to occur at the same cross-section and some point-in the cross-section.

Some averaging of the response spectra is performed to smooth out the erratic response of the earthquake's random behavior. At the high frequency end of the spectra, the acceleration levels of the smoothed spectra converge to the values of the unsmoothed spectra.

It is therefore concluded that design procedure, used to design Class I seismic restraints for Indian Point Generating Station - Unit No. 2, is conservative.

RCS Analysis for Combination Loading of
Design Basis Earthquake and Design Basis Accident

The Indian Point Unit No. 2 Reactor Coolant System was not committed to be designed for the combination of the seismic and blowdown loads. However, an analysis fo this combination of loadings has been performed for the Indian Point Unit No. 3 Reactor Coolant System, which is identical to Unit. No. 2. The analysis has been performed as outlined below:

(1) A lumped mass dynamic mathematical model of the primary coolant loop and support system was developed.

(2) This dynamic model was subjected to multiple simultaneous time history hydraulic forcing functions for the blowdown analysis. The double-ended ruptures were located at places of large change in flexibility. Time history response of the total structure to these conditions was computed and reduced to time history stresses.

(3) The dynamic model was then subjected to a floor response spectra earthquake analysis.

(4) The loads as determined above were used for an evaluation of the stresses along the piping system.

(5) The stresses as determined from the basis described above are lower than the allowable stresses calculated by using the approach described in WCAP 5890, Rev. 1 and the following parameters:

(i) 20% of the uniform strain on the allowable membrane and average strain; and

(ii) 23,100 psi as the at-temperature yield in the axial direction. This value is based on the minimum value of the at-temperature yield in the loop direction as measured with samples from the Unit No. 2 piping increased by 10% for the increase in strength in-going from the loop to the axial direction. The tensile tests on the Unit No. 2 piping material at-temperature yielded at a minimum value of 20,900 psi, a maximum of 29,700 psi and an average over eleven samples of 23,300 psi.

Based on the above analysis, it is concluded that the Unit No. 2 Reactor Coolant System can stand the combination of blowdown and seismic loads within acceptable stress limits.

Service Water Lines

The Service Water lines consist of two (2) 24" diameter carbon steel pipes. They run in a common trench which is backfilled. Assuming that the ends of a pipe are free to displace vertically but not rotate and that the maximum permissible stress is restricted to 30,000 psi, a parametric study shows that the following maximum allowable relative displacements may occur during a seismic disturbance without overstressing the pipe:

Length (ft.)	1	10	25	50	75	100
Displacement (inches)	0.002	0.20	1.25	5.01	11.27	20.04

It is therefore concluded that the Service Water lines could withstand, without being overstressed, relative bedrock displacements associated with the earthquakes defined for the Indian Point site.

Seismic Evaluation of the Fan Cooler and Hydrogen Recombiner Systems

The seismic analysis of the fan cooler system has been completed in two parts.

a) Analysis of the structural steel enclosure of the fan cooler units to include the effect of supported equipment.

The structural analysis considering simultaneous incident pressures and earthquake forces has been conducted on particular members, plates and connections which are considered critical to the structural performance of the RCFC. The casing frame has been considered as an inverted L-shaped frame with a 14 WF 61 or 14 WF 111 as a vertical leg and a 14 WF 30 or a combination of two 8 WF 58 as the horizontal beam.

The connection of the beam to the concrete shell and that of the column to the floor have been considered as transmitting shear only. The connection of the beam to the column has been assumed to transmit moment and shear; the effect of this assumption has been checked by also considering a pinned connection.

Earthquake loadings have been treated using the response spectra techniques. A horizontal force of .6W and a vertical force of .4W, where W is the weight of the member including static forces, were concentrated at the center of gravity of each member. Differential pressures of 5, 9, and 20 psig were used.

All loading were assumed to act simultaneously and comparison was made with the allowable stresses permitted by the AISC spec's for

A 36 steel and A 325 high strength bolts. A one-third increase in allowable stresses under dynamic loadings was not considered. Where applicable, allowable concrete stresses were taken from ACI Specifications.

Results of the analysis on the enclosure demonstrate that the design is adequate.

b) Evaluation of the fan-motor system and their foundation.

The fan motor and supporting structural system have been evaluated using acceleration values for a maximum hypothetical earthquake. These values are 0.6g for the horizontal direction and 0.4g for the vertical direction. These accelerations were assumed to act simultaneously.

The failure modes considered for the motor unit are excess deflection of the rotor shaft which results in rubbing against the housing or by bearing failure. The failure modes for the fan are by failure of the fan shaft support bearings or by deflection of the fan housing and fan wheel to cause binding. In addition, the potential for shear and overturning failure of the motor-fan assembly at the foundation anchorage was evaluated.

Based on analyses made on similar fan-motor systems, it is to be concluded the fan cooler units in the containment are adequately designed to resist the seismic loading defined for the site and supporting building structure.

The two (2) hydrogen recombiners are located in the containment at the 95 ft. elevation. The hydrogen recombiners are as shown on Figure 6.8(a)-1. The hydrogen recombiners are rigidly bolted to the structural steel frame work at elevation 95 ft. The recombiners were analyzed in accordance with method number (1) as outlined on page Q 1.9-8 which states "Equipment which is rigid and rigidly attached to its support structure is analyzed for a g loading equal to the peak acceleration of the supporting structure at the appropriate elevation.

The analysis indicated that the frequency of the recombiners is $\sim$ 39 cycles per second. The maximum hypothetical earthquake seismic load stresses were determined on the basis of 0.6g horizontal acceleration and 0.4g vertical acceleration taken simultaneously. The analysis included the evaluation of the following areas:

a. overturning of the unit

b. compressive stress on support columns

c. bending stress in support columns

d. check of the precast insulating liner of the burner

The results of this analysis shows that the resulting stresses in the structural members are well within the allowed stress as published by AISC. The resulting stress in the pre-cast liner is well below that published by the liner supplier.

QUESTION 1.10

List the amplification factors with respect to ground motion as determined from seismic analysis for the reactor, recirculating pumps, Class I piping, and spent fuel pool.

ANSWER

For the design of the reactor, recirculating pumps and Class I piping an amplification factor of 4.0 was used with respect to ground motion of 0.15g. This amplification factor was based on the maximum for a one half percent damping of the ground response spectrum. The fundamental frequency of the reactor building internal structure is approximately 17 cps. As can be seen from Figure A.1-2 of the FSAR for this frequency level no significant building amplification of the ground response is encountered.

Procedures outlined in Section 6.5 of TID-7024-Nuclear Reactors and Earthquakes were used for the seismic design of the spent fuel pool. The effects of water in the pool is accounted for in this design approach.

QUESTION 1.11

With respect to as-built tornado protection for the facility, (1) determine the wind loading and pressure drop that structures and equipment comprising the facility are capable of sustaining and provide the basis for the determination in each case; (2) discuss the ability of the structures and exposed equipment to sustain tornado-originated missiles.

Structures which should be considered include the containment building, the primary auxiliary building, the control room building, the refueling building including the spent fuel storage pool, and the intake structure.

ANSWER

(1) Containment Structure - See answer to Questions 2.1 (1) and 2.4 (d) in Supplement 4 to the Indian Point Unit No. 3 PSAR.

PAB; Control Building; Fuel Storage Building; - Based on information from the siding manufacturer the siding panel will blow out at 170 psf which is equivalent to a 1.18 psi negative pressure. Panels fail at 60 psf external pressure which is equivalent to 162 mph external wind load.

The girts will fail at 90 psf which is equivalent to .62 psi negative pressure. The 60 psf mentioned above controls the external loading condition.

(2) Containment Structure - It will not be penetrated by the following missiles
a) 4" x 12" x 12' wood plank at 300 mph.
b) 4000# auto at 50 mph less than 25'-0 above the ground.

PAB; Control Building; Fuel Storage Building; - The 3 1/4" thick siding panels are not capable of resisting any tornado generated missiles.

The intake structure is capable of resisting any wind or missil[illegible] generated by a tornado. This is true only for the structure an[illegible] not necessarily include equipment. The spent fuel pool tornado protection is discussed in proprietory WCAP 7313-L. A topical report will be filed with the AEC in the fourth quarter of 1969 which will consider the effect of tornado missiles on the Indian Point Spent Fuel Pool.

QUESTION 1.12

Provide an analysis of the as-built resistance to tornado winds for the Indian Point No. 1 superheater stack. Discuss the effect of a collapse of this stack on Indian Point 2 Class I structures and equipment.

ANSWER

The Indian Point No. 1 stack is constructed of steel plates varying in thickness from 7/16" at the top to 11/16" at the bases. The diameter of the stack at its base elevation 135'-6" is 30'-0", and at the top elevation 470'-0" is 9'4-". The entire stack is lined with 2-1/2" of gunite for protection against corrosion.

A 48" diameter ventilation duct is supported from the east side of the stack, and the wind load on this duct contributes to the loading on the stack.

Conservatively neglecting the strength of the gunite lining, the critical buckling stress for the stack is 18.75 ksi. The wind velocity required to produce this stress is 242 mph. (An analysis of this is attached).

It must [illegible]ever, that the stack is supported from the roof of the Indian Poi[illegible]uperheater Building.

The Superheater Building is a steel frame structure with brick and metal siding. The building is located directly between Indian Point Units 2 and 3 which are both presently under construction. Both these complexes, plus the natural topography north and south of Indian Point No. 1, effectively shield to a large degree the Superheater Building from direct wind loads from the north or south direction.

The arrangement of the diagonal bracing in the Superheater Building affords the structure a much greater resistance to wind shear from the north than from the south. A preliminary investigation of the bracing system in the

ANSWER (Con't)

building was made assuming the tornado strikes from the south. The compression members were loaded to the design allowable stress and the tension members were loaded to the yield point.

Neglecting the fact that the building is shielded to a large extent as described above, and assuming that the entire south wall surface area of the structure is exposed to the tornado, the Superheater Building is capable of resisting winds with a velocity of 170 mph.

The effect of the stack falling on Indian Point No. 2 structures will be presented in an analysis prior to February 1, 1970.

SUMMARY OF CALCULATIONS FOR STACK ANALYSIS

$\bar{y} = \frac{h}{3}\left(\frac{2a+b}{a+b}\right) = \frac{334}{3}\left(\frac{2\times14+34}{14+34}\right) = 144'$

Total surface area = [(3.14) (30) + (3.14)(10)] (334) (1/2)

= 21,000 ☐'

wt. of stack = (21,000)(0.0204 k/☐') = 429^k

wt. of duct = π (4) (334) 10.2 = 44

Add 10% for ribs, lddrs = 473 (.1) = 47

520^k

Wt. of gunite = 21,000 (32) = 675^k

Total DL = 520 + 675 = $\underline{1195^k}$

Area of metal = $776 in^2$ at base

$I = \Pi r^3 t = (3.14)(180)^3 (0.687) = 12.6 \times 10^6$ in

Surface area exposed to wind (including 48 duct etc) = 8544☐'

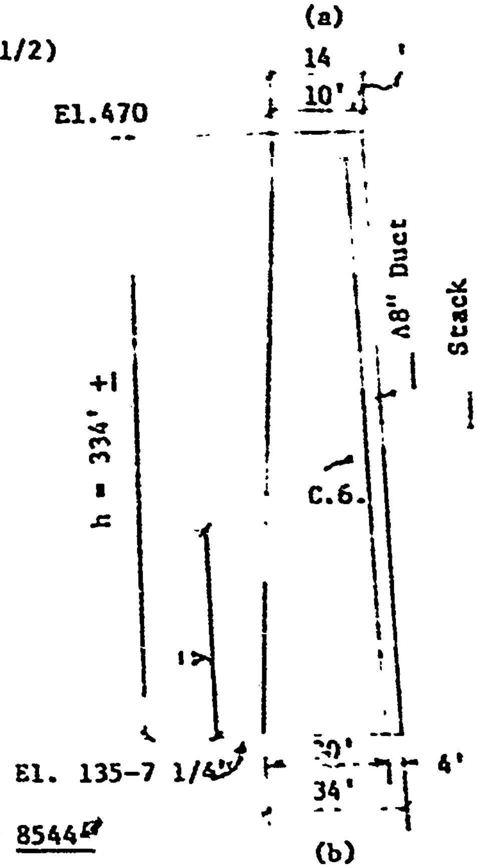
Assume 230 psf wind pressure (That is 500 mph wind)

Moment @ base = 0.23 ksf x 8544☐' x 0.6 x 144' = $169,786^{K-1}$ Say 170,000

$$f = \frac{P}{A} \pm \frac{MC}{I} = \frac{1195}{776} \pm \frac{(170,000)(12)(180)}{12.60 \times 10^6} = 1.54 \pm 28.2$$

$= 29.74^{ksi}$ comp.

$= 26.66^{ksi}$ Tens.



Check Buckling of Shell (Stack)

$$\sigma_{critical}(\text{compression}) = \frac{0.2E}{\sqrt{3(1-\nu^2)}} \times \frac{t}{R}$$

$$= \frac{(0.2)\ (29)\ (10)^3}{\sqrt{3(1-0.3^2)}} \times \frac{0.687}{180} = \underline{13.4 \text{ KSI}}$$

Assume bending stress is 1.4 times buckling stress in compression

$$\therefore \sigma_{critical}(\text{bending}) = 13.4 \times 1.4 = \underline{18.75} \text{ KSI}$$

18.75 KSI < 30.66 KSI That means the shell of the stack will buckle before the stress reaches its optimum value.

$$f = \frac{P}{A} \pm \frac{MC}{I}$$

$$18.75 = 1.54 + \frac{(12M)\ (180)}{12.60 \times 10^6} \quad \text{or} \quad M = \frac{(17.21)\ (12.60)\ (10)^6}{(12)\ (180)} = \underline{100{,}500^{K-1}}$$

Equivalent concentrated load to produce the same moment is:

$$\frac{100{,}500^{K-1}}{144} = \underline{698^k}$$

$$\text{Press./}\# = \frac{698{,}000\#}{8{,}544} = 81.8 \text{ psf.}$$

$$\text{Press. before reduction by shape factor of } 0.6 = \frac{8.18}{0.53} = \underline{154} \text{ psf.}$$

$$\text{Equivalent wind velocity} = \sqrt{\frac{154.0}{0.00256}} = \underline{242 \text{ mph}}$$

QUESTION 1.13

With reference to the seismic design of the piping supports:

(a) Stress the magnitude and effects of out-of-phase seismic loads, and

(b) Verify that the piping systems have been analyzed based on "as built" support locations.

ANSWER

a) Relative displacement between anchor points has been considered in the seismic analysis of the main steam lines, where largest relative displacements are expected. Analysis indicates that the stress at the highest seimically stressed point is affected by less than 10% when relative anchor displacements are considered.

b) Those seismic supports installed to date have been verified to agree with the design location and therefore the locations used in the analyses. Each of the remaining seismic supports to be installed will also be so verified. If for any reason, such as installed piping interferences, the restraint cannot be located as designed and significant deviations exist between design and field installation, the affected piping and supports will be reanalyzed to assure as-built adequacy.

QUESTION 2.1

Based upon calculations provided for the Indian Point 3 facility, the Maximum Probable Hurricane would result in the highest potential flood levels at the Indian Point site. We understand that this flood has been reanalyzed. Provide this reanalysis with all assumptions, calculations, and justifications. Compare the flood protection for all Class I structures, components, and equipment with this calculated hurricane flood. Will a model study be conducted to evaluate the potential wave run-up? If not, justify the run-up assumptions being made.

ANSWER

A reanalysis of the flooding potential at the Indian Point site due to the Maximum Probable Hurricane has been made by Quirk, Lawler and Matusky Engineers. Their report, submitted with this supplement to be inserted at the end of Section 2.5 of the FSAR, is entitled "Evaluation of Flooding Conditions at Indian Point Nuclear Generating Unit No. 3 - February 1969". The report contains all assumptions, calculations and justifications used to determine the maximum hurricane surge height and the potential wave run-up at the site.

The report indicates that the combination of the hurricane surge, spring high tide and wave run-up will cause the water level at Indian Point to reach 14.5 feet above Mean Sea Level. Since all Class I structures, components and equipment are located at Elevation +15.0 and above, the Maximum Probable Hurricane presents no threat to the safe operation of Indian Point Unit No. 2.

QUESTION 2.2

Analysis of fresh water runoff flood levels performed for the Indian Point 3 facility indicated that fresh water flood levels above the 15 ft. design protection level for Indian Point 2 are possible. We understand that this flood has been reanalyzed. Provide this reanalysis with all assumptions, calculations, and justifications.

ANSWER

A reanalysis of the maximum fresh water runoff flood level at Indian Point is contained in the same report referenced in the answer to Question 2.1. The report indicates that the combination of the maximum probable rainfall, upstream dam failures and the maximum ebb tide in the Hudson River will cause the water level at Indian Point to reach 11.7 feet above Mean Sea Level.

Question 3.1

There is insufficient information in the FSAR to indicate how the reactor will be maintained within thermal limits (design, nuclear hot channel factors) in the presence of power distribution anomalies caused by xenon or misplaced control rods. Please provide the power distributions peaking factors, and DNBR's for diametral and axial xenon oscillations and various potential control rod errors including those involving part-length and x-y control rods.

Answer

A discussion of the means provided to monitor and control power distributions anomalies caused by misplaced control rods and xenon oscillations is given in the Indian Point No. 2 FSAR, Appendix 3-B.

A description of the protective function in the event of axial xenon oscillations, including calculated peaking factors and DNBR's, and the automatic trip set point reduction is given in WCAP-7208.[1] Additional information on the response to out-of-core ion chambers, including comparision with experimental information, is given in WCAP-9010[2] and in WCAP-7407-L.[3] X-Y control rods are not required nor are they employed in the Indian Point-2 reactor.

A discussion of the consequences of control rod malposition is given in the Indian Point Unit No. 2 FSAR, Appendix 3-B, and in WCAP-7407-L (Section 3.2).[3]

No automatic protective function is necessary, since even the complete misalignment of a control rod in the most limiting case (see WCAP-7407-L,[3] Section 3.2.2) cannot lead to a DNBR less than 1.3 at operating conditions.

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[1] Westinghouse Electric Corp., Power Distribution Control of Westinghouse Pressurized Water Reactors, Report WCAP-7208 (APD Proprietary Class 2), September 1968.

[2] R. J. Johnson, Connecticut-Yankee Tests on Detection of Power Maldistribution, Report WCAP-9010 (NES Proprietary Class 2), Westinghouse Electric Corp., February 1969.

[3] R. F. Barry et al., Power Maldistribution Investigations, Report WCAP-7407-L (Proprietary Class 2), Westinghouse Electric Corp.

Furthermore: (1) rod position indicators are provided; (2) the existence of an asymmetric control rod misalignment would be revealed by the out-of-core instrumentation; and (3) both asymmetric and symmetric control rod misalignments can readily be detected by the in-core thermocouple system as indicated in WCAP-7407-L[3] and WCAP-9010.[2]

QUESTION 3.2

Discuss the bases for predictions of xenon stability indices, indicating how conservatisms are achieved in predicting possible oscillation modes.

ANSWER

In assessing potential power distribution instabilities arising from spatial xenon redistribution, and in determining stability indices, primary reliance has been placed on time-dependent digital calculations in three dimensions representing feedback reactivity effects by means of semi-empirically fitted expressions whose coefficients were determined from other calculations using standard design analytical techniques and computer codes (e.g., LEOPARD code).

To assess the level of credibility and range of uncertainty attached to xenon stability analyses, conservative values of the reactivity feedback parameters were used to arrive at a reasonable upper limit for the stability index. This technique gives reasonable assurance that the reactor will in fact be stable toward diametral xenon oscillations. Even if the unexpected were to occur, and xenon instability were observed on startup tests, means are available to increase the moderator temperature feedback term in order to stabilize the reactor response to diametral xenon oscillations (see WCAP-7407-L, Section 3.1.1[1]). The reference three-dimensional time dependent digital calculations indicate that the Indian Point-2 reactor will be stable against diametral xenon transients.

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[1] R. F. Barry et al., Power Maldistribution Investigations, Report WCAP-7407-L (Proprietary Class 2), Westinghouse Electric Corp.

QUESTION 3.3

Provide analyses of second harmonic axial and cross coupled xenon oscillations including power distributions. Show how such oscillations produce identifiable X-Y and axial offsets using out-of-core ion chambers. Show that in-core neutron detectors are not required for protection and control.

ANSWER

Cross coupled transients are discussed in reference 1 on the basis of full three dimensional analyses of xenon transients.

"Second overtone" xenon transients from quadrant to quadrant (X-Y transient) are also discussed in reference 1 in the form of radial transients. That is to say that a power-xenon perturbation is introduced by moving the center control rod. "Second overtone" xenon transients from top-to-bottom (axial transient) have been analyzed with results presented in reference 2. For clarity, such transients are presented here specifically for the Indian Point #2 reactor.

Figures 3.3-1 through 3.3-3 show the transient axial power peaking factor and axial offset (percent power in top half of core minus percent power in bottom half of core) at three points in core life caused by the prompt removal of the part length rods from an equilibrium core. The resulting oscillation is divergent late in core life, but the transient is in the "first overtone" even though the perturbation was to the "second overtone". This is because of the dominance of the moderator coefficient acting on the vertical coolant flow.

Figure 3.3-4 shows a cross-plot of the axial peaking factor and axial offset for the three transients. This plot is completely in agreement with such plots shown in reference 2. In short, the excore detector-based protection system is capable of detecting (by means of axial offset) transients which result from perturbations to the "second axial overtone" of the power distribution.

Reference 1: WCAP-7407-L, Power Maldistribution Investigations, Westinghouse Proprietary

Reference 2: WCAP-7208, Power Distribution Control of Westinghouse Pressurized Water Reactors, Westinghouse Proprietary

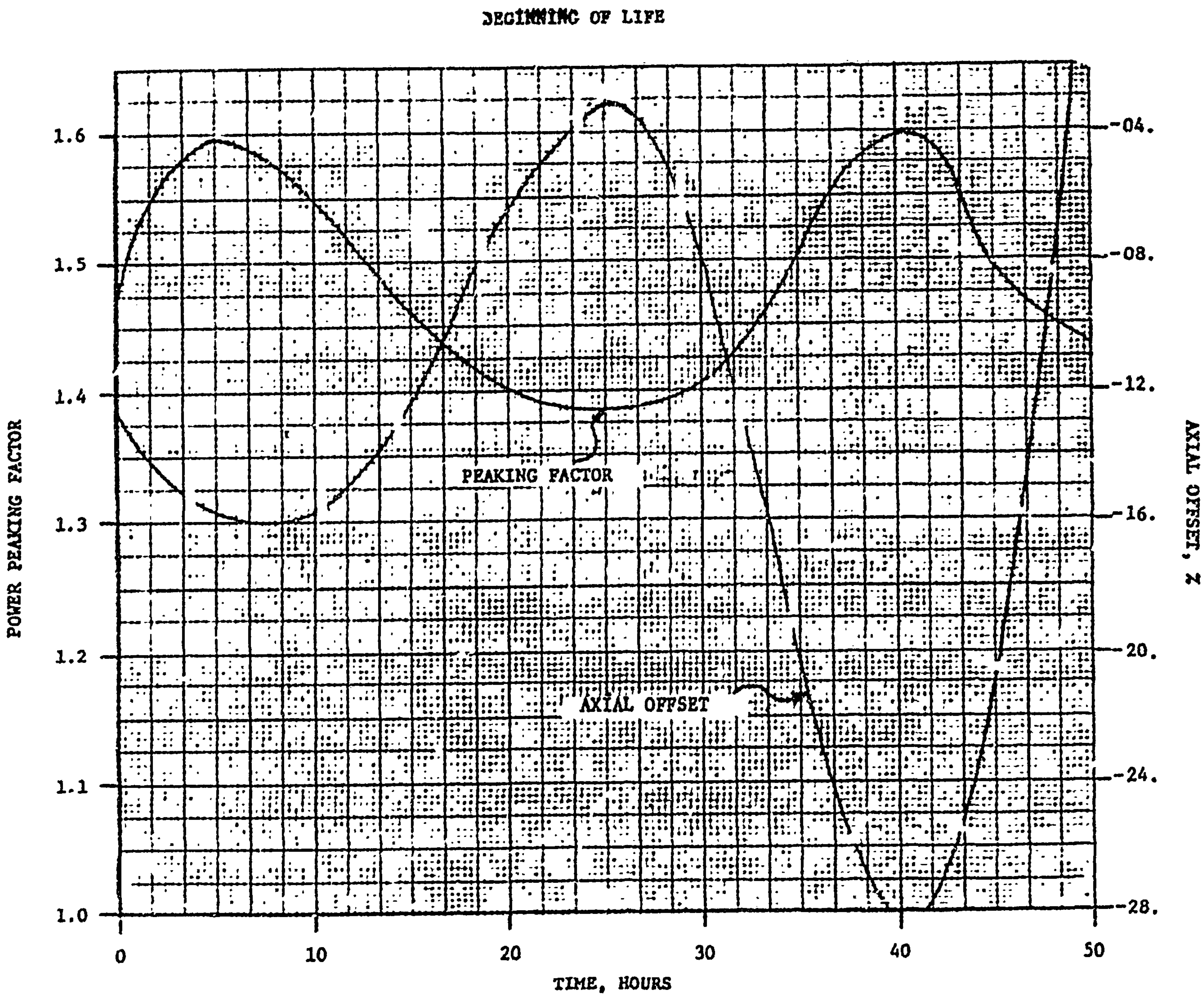
BEGINNING OF LIFE
POWER PEAKING FACTOR
1.6
1.5
1.4
1.3
1.2
1.1
1.0
PEAKING FACTOR
AXIAL OFFSET
0
10
20
30
40
50
TIME, HOURS
-04.
-08.
-12.
-16.
-20.
-24.
-28.
AXIAL OFFSET, %

FIGURE 3.3-1

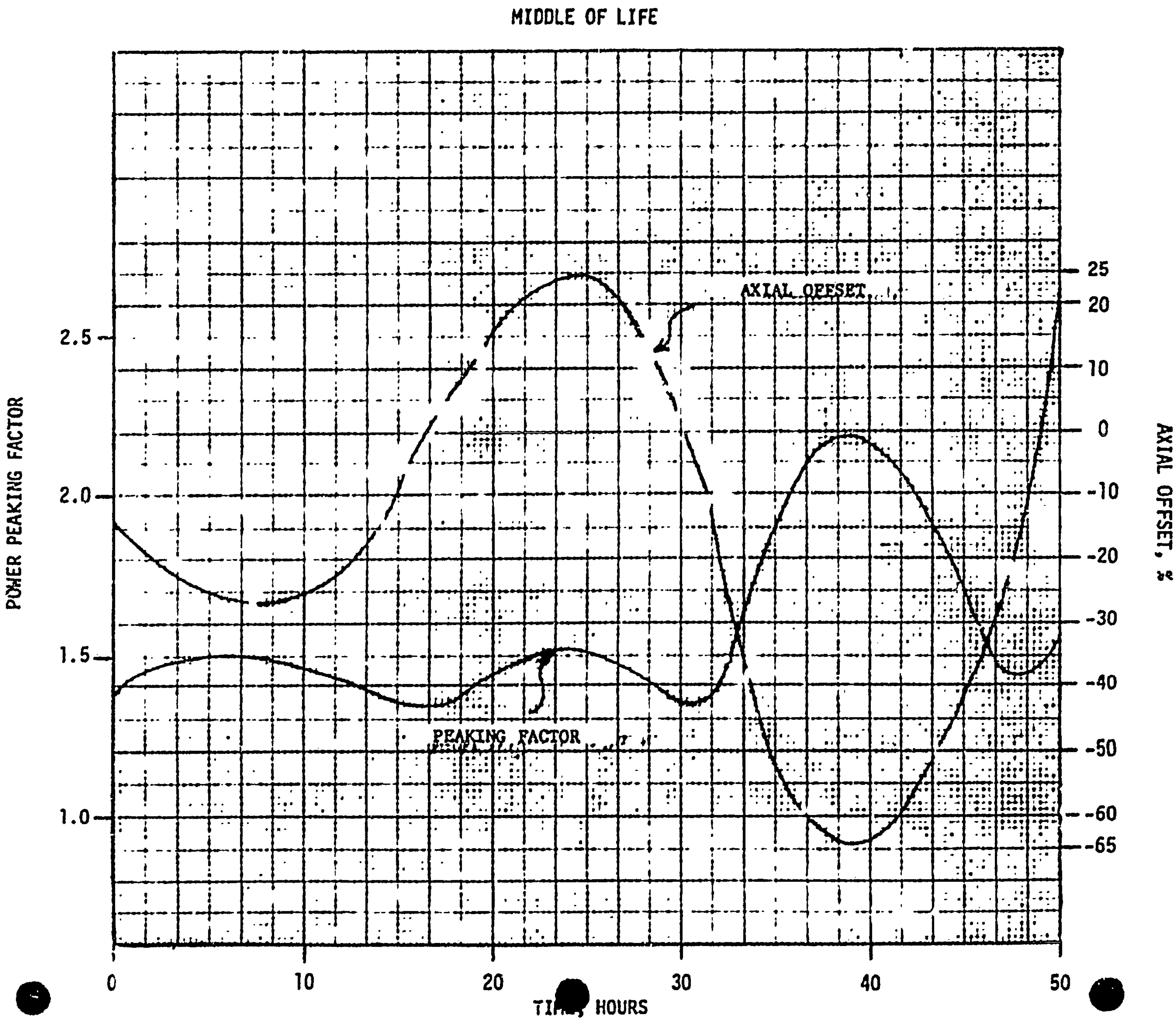

FIGURE 3.3-2

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END [illegible] LIFE

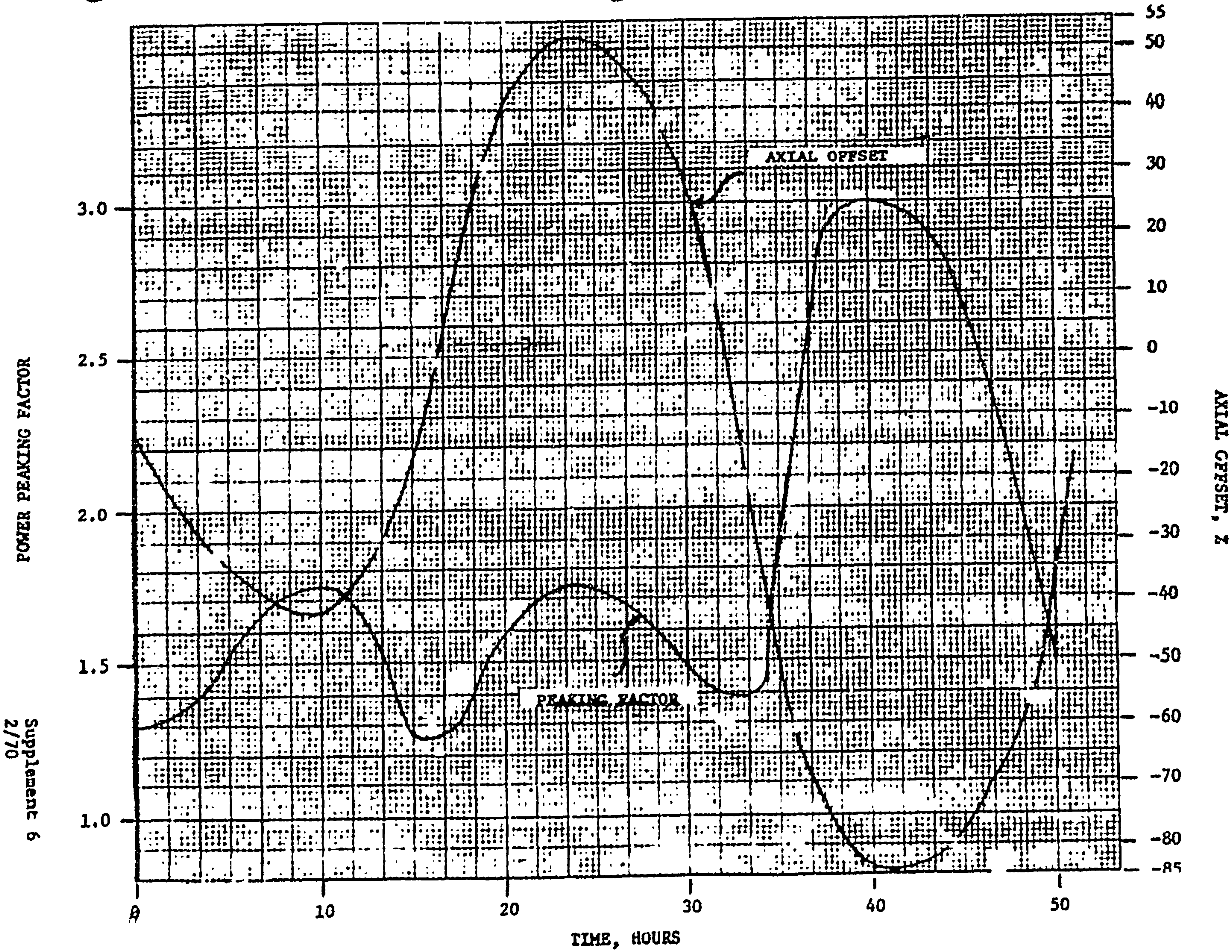

FIGURE 3.3-3

POWER PEAKING FACTOR VS. AXIAL OFFSET

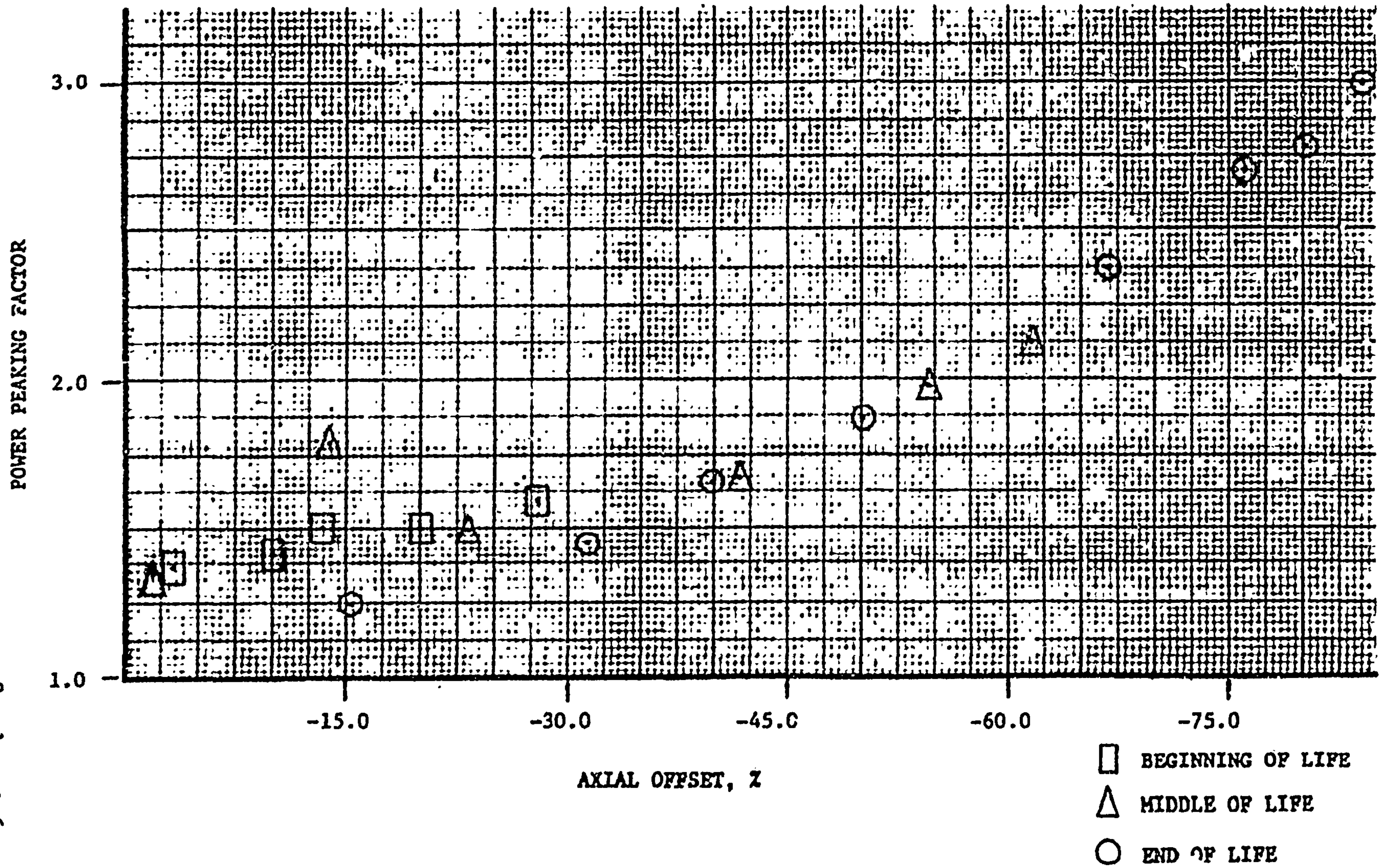

FIGURE 3.3-4

Question 3.4

What is the basis for separation of axial and diametral xenon instabilities? What problems arise if separability is not complete?

Answer

The separation of dimensions is a conceptual artifice which greatly facilitates analysis of xenon transients. Full three-dimensional transient analyses have been performed, as reported in WCAP-3680-22[1] and WCAP-7407-L.[2] Conclusions relating to power distribution stability against spatial xenon redistribution are based on results of these analyses. Cross-coupling between axial and diametral xenon osicllations are inherently accounted for in the three-dimensional time dependent calculations. Results of these calculations do not reveal any unique problems arising from cross-coupling. See Indian Point No. 2 FSAR, Appendix 3-B, for additional discussion.

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[1] F. B. Skogen and A. F. McFarlane, Xenon-Induced Spatial Instabilities in Three-Dimensicns, USAEC Report WCAP-3680-22, Westinghouse Electric Corp., September 1969.

[2] R. F. Barry et al., Power Maldistribution Investigations, Report WCAP-7407-L (Proprietary Class 2), Westinghouse Electric Corp.

Question 3.5

If the reactor is operated to xenon equilibrium, is shut down, and restarted at the time of maximum xenon buildup, how is the radial power shape affected by nonlinearities in the xenon poisoning? Show the effect as a function of time, and how it is accounted for in the plant for power distribution control.

Answer

If the core was originally operating with a symmetric quadrant-to-quadrant power distribution, the effect of excess xenon poisoning would be to flatten the power distribution because the xenon excess would be greatest where the equilibrium power has been greatest. While this additional power flattening would tend to decrease the stability of the reactor, the analytical evaluation described in WCAP-3680-22[1] has already assumed a power distribution flatter than expected in the actual reactors; thus, an allowance has already been made in the WCAP-3680-22[1] analysis to account for this effect. Furthermore, the higher xenon inventory present under the postulated conditions (i.e., maximum xenon buildup) would decrease the required boron concentration and lead to a more stable reactor response from the enhanced negative moderator reactivity feedback effect.

As burnup progresses, the required boron concentration further decreases resulting in an increasingly more stable reactor response to diametral xenon oscillations. This effect is greater than the effect of burnup on radial power flattening.

If the equilibrium power was not balanced from quadrant to quadrant, the effect of the excess xenon poisoning may be to cause the quadrant power to reverse and perhaps to increase in magnitude. If the quadrant power tilt were to reach the limit given in the Technical Specifications, the operator would take action to maintain core thermal margins.

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[1] F. B. Skogen and A. F. McFarlane, Xenon-Induced Spatial Instabilities in Three-Dimensions, USAEC Report WCAP-3680-22, Westinghouse Electric Corp., September 1969.

In any event, the excess xenon which might be present under the conditions postulated would decay naturally and cannot be regarded as a continuing source of power distribution anomalies. Similarly, a top-to-bottom power imbalance could be temporarily increased by the excess xenon poisoning. Such an imbalance cannot be a safety problem because the Reactor Protection System is cognizant of the axial power imbalance and if necessary will reduce trip setpoints accordingly.

QUESTION 3.6

We understand that fixed in-core neutron instrumentation is to be installed in the Indian Point 2 reactor. Provide a detailed description of this instrumentation and how it will be employed in reactor operation.

ANSWER

The Indian Point 2 plant will have a system of fixed in-core neutron detectors. These detectors are presently undergoing life and parametric tests at Brookhaven and at Tuxedo Park test reactors. In the Brookhaven reactor, the detectors are being subjected to neutron and gamma fluxes which are three times that expected in a PWR, and have been efficiently operating for ten months, which is equivalent to 30 months in a PWR. They have a design limit of 3×10^{21} nvt.

The intent of providing this system is to determine if it will enhance operations. The detectors will provide input to the computer only and will in no way be used in or relate to the safety instrumentation. Adequate information from the safety instrumentation is provided to the operator so that he does not have to rely upon the fixed in-core detectors and the computer information.

The fixed in-core detector system consists of eight flux thimbles located symmetrically (radially and axially) throughout the core as shown in Figure 1. Each flux thimble will have four miniature detectors (small argon filled, highly enriched U^{235} fission chambers) with a sensitive length of about one inch and will be about 0.15-inch diameter, maximum. These locations are tentative and preliminary.

Within a thimble, the four detectors are axially spaced at fixed locations with the four signal cables brought out at the trailing end of the thimble. Each thimble assembly is stainless steel of 0.385" OD with perforations to allow coolant flow to circulate inside the thimble to cool the detectors A high pressure seal is provided in the assembly to provide a pressure barrier between the reactor coolant and the atmosphere.

The detectors are not movable when the primary system is at operating pressure. Each detector must be moved manually with a special tool through conduits identical to those used for the movable detector thimbles. The assemblies are extracted downward from the core during refueling. The seal table used for the movable detector assemblies is also utilized for the fixed in-core detector assemblies.

The detectors have individual adjustable d-c power supplies (0 to 250 vdc). The individual output signals for the detectors are fed to the plant computer through a resistor/potentiometer network in each power supply to provide a detector current signal in the form of a 1.0 volt full scale signal.

Each detector is scanned at a normal interval of 8 seconds and are compared to fixed limits. The 8-second values from each detector are then time averaged for one minute and the time averaged value used for all calculations performed by the computer. These calculations include the following:

a) Mean power level seen by each detector string
b) Axial offset seen by each string
c) Core mean power level
d) Core mean axial offset
e) Radial quadrant tilt factors for the eight quadrants which describes the tilted power distribution curve for each detector string.

The computer prints out alarm messages for the operator information whenever:

a) Any of the 8 mean power levels exceeds its limit
b) Any of the 8 radial tilting factors exceeds its limit
c) Any of the 8 axial offsets exceeds its limit
d) The core mean axial offset exceeds its limit.

This information is provided for operator's convenience and is not required for the safe operation of the plant.

The installation of the fixed in-core system is expected to cause no significant reactivity effect. If a fixed detector were to fail, the expulsion of reactor coolant would be accommodated by the charging pumps (see Diablo Canyon Unit No. 2 PSAR, page 12.82h and i, Docket No. 50-323) which is common for ruptures of very small cross section. This would enable the operator to execute an orderly shutdown. If a seal were to fail at the seal table, flow from the reactor coolant system would be through the annulus defined by OD of the flux thimble and ID of the conduit (approximately 0.13 sq. in.). The fixed in-core system will be designed to Class I standards such that the likelihood of a failure causing a loss of coolant will be extremely remote.

The computer system being provided for the Indian Point Unit 2 is a Westinghouse Prodac 250. It is being provided as an adjunct to the normal control room instrumentation to assist the operator in the operation of the reactor by monitoring reactor performance and displaying it in a consistent, well ordered, useable form. This system is not required for safety, and operation of the reactor is in no way dependent upon the availability of the computer.

The computer system performs functions such as scanning, signal converting, calculating, indicating, recording and alarm annunciating. These functions will provide the operator with thermal outputs, unit net efficiency, control rod cluster position deviation, tilting factor, operational reactivity balance and xenon calculations which will greatly enhance the evaluation and operability of the plant.

Briefly, the analog scanning includes reading all inputs in a pre-established manner, checking the readings, converting values to engineering units, storing them for future use, updating information, and checking alarm conditions. Some inputs are scanned once every second and status placed in memory; other inputs are not scanned periodically but are given immediate attention. The alarm program compares the values of the inputs against the fixed or variable alarm limits and announces when off-normal conditions exist by use of an audible alarm and a printout alarm message, generally in red. A typical alarm printout format would be as follows:

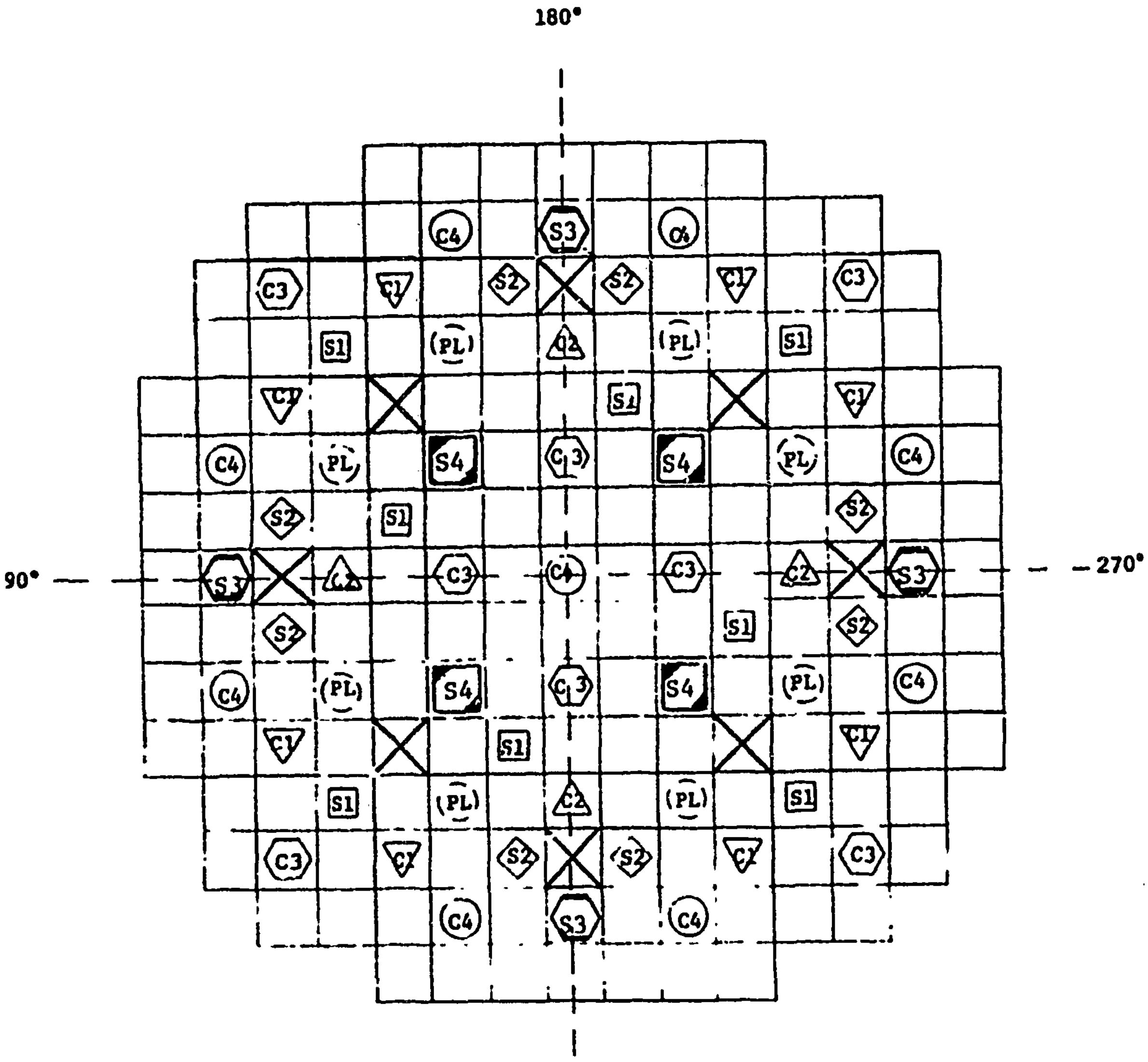
Time	Reason for printout	Input No.		Value	Exceeded limit	Units
1203	Alarm*	P169	Up. to. 32. Char. Descr.	xxxxxx	FHxxxxxx	xxxxxx
1216	Normal	P169	Up. to. 32. Char. Descr.	xxxxxx	FHxxxxxx	xxxxxx
1216	Alarm	D0161	Up. to. 32. Char. Descr.	Disconnected		
1217	Normal	D0161	Up. To. 32. Char. Descr.	Connected		
1221	Alarm	FC02	Up. to. 32. Char. Descr.	xxxxxx	IHxxxxxx	xxxxxx

FH = fixed high limit was exceeded

IH = incremental high limit was exceeded

* = printout in red

Recording includes typewriters which will print out Pre-trip Data, Post-Trip Data, Sequence of Events, and normal periodic demand and summary logs.



ROD CLUSTER CONTROL BANKS

BANK	SYMBOL
S1	□
S2	◇
S3	⬡
S4	◪
C1	▽
C2	△
C3	⬡
C4	○
PL (Part Length Rod)	◌
Fixed Incore	×

FIGURE 3.6-1

QUESTION 4.1.1

Section 4.5.1, Reactor Coolant System Inspection, of the FSAR is addressed to your plans for inservice inspection of the primary coolant system. The above section discusses components and areas available for inspection but provides only very limited information as to methods or frequency of planned inspections. Using Table I of Section ISI-250 of the N-45 Code Committee Draft Standard* as a guide, establish the areas and extent of examination, degree of examination sampling, and examination methods which will be used. Summarize in tabular form the inservice inspection which within existing accessibility limitations, if any, will be performed based on the application of direct or remote examination methods using currently available (or to be specially developed) equipment and techniques. The table should include the components and areas subject to examination, the sampling selection, the extent of examination, the examination method, and the inspection frequency.

ANSWER

The in-service inspection program for the Indian Point Unit No. 2 is described in Section 4.2 of the proposed Technical Specifications and Bases.

QUESTION 4.1.2

In addition to inservice inspection of the primary system pressure boundary as defined in the N-45 Code Committee Draft Standard referred to above, plans for inservice inspection of other systems or components should be formulated at this time. Systems or components which should be considered include:

Supports for primary system components and piping.
Engineered safety features.
Primary pump flywheels.
Other Class I mechanical and fluid handling systems.

Please present details of items to be inspected, methods to be used, and inspection schedule for the above.

ANSWER

The extent of the plants inservice inspection program will be that which is described in the proposed Technical Specifications and Bases.

QUESTION 4.2.1

Please ... de a design description and results of design analysis for the prima ... flywheel to supplement that information presented i.. paragraph 5.1.2.. o. the FSAR. The additional information should include:

a - A description and dimensions of flywheel.
b - Complete material specification.
c - Methods of fabrication.
d - Methods of quality assurance.
e - Normal operating speed of the flywheel (pump).
f - Bursting speed of the flywheel.

ANSWER

(a) A description and dimension of the flywheel. The primary coolant pump flywheels are shown in Figure 4.2.1-1.

They are fabricated from two rolled, vacuum-degassed, ASTM A-533 steel plates. The plates are bolted together with bolts aligned perpendicular to plane of the plates. Thus the bolts carry no stress during operation.

(b) Complete Material Specification.

The complete material specification is as follows:

ASTM A-533 Grade B Class I, plus supplementary MT requirements and charpy tests as detailed in (d), Methods of Quality Assurance.

(c) Methods of Fabrication.

The flywheel blanks are flame-cut from the plates, with allowance for exclusion of flame-affected metal. They are then machined to the specified dimensions, and the bolt holes are drilled.

Two plates are then bolted together and the finished flywheel is attached to he motor shaft and the whole unit is balanced to yield vibration levels at operating speed less than 0.001 inches double amplitude.

(d) Methods of Quality Assurance.

An NDTT less than +10°F is specified. A minimum of three charpy tests are made from each plate parallel and normal to the rolling direction, to determine that each blank satisifies design requirements.

The finished flywheels are subjected to 100% volumetric ultrasonic inspection.

The finished machined bores are also subjected to magnetic particle, or liquid penetrant examination.

These design-fabrication techniques yield flywheels with primary stress at operating speed (shown in Figure 4.2.1-2) less than 50% of the minimum specified material yield strength at room temperature (100 to 150°F).

A fracture mechanics evaluation was made on the reactor coolant pump flywheel. This evaluation considered the following assumptions:

a. Maximum tangential stress at an assumed overspeed of 125% compared with a maximum expected overspeed of 109%.

b. A through crack through the thickness of the flywheel at the bore.

c. 400 cycles of start up operation in 40 years.

Using critical stress intensity factor and crack growth data attained on fly-wheel material, the critical crack size for failure was greater than 17 inches radially and the crack growth data was 0.030" to 0.060" per 1000 cycles.

An ultrasonic inspection capable of detecting at least 1/2" deep cracks from the ends of the flywheel and a dye penetrant or magnetic particle test of the bore for one pump at each refueling shutdown will be more than adequate as part of the plant surveillance program.

(e) Normal operating speed of the flywheel (pump).

The primary coolant pumps run at 1189 rpm, and may operate briefly at overspeeds up to 109% (1295 rpm) during loss of outside load. For conservatism, however, 125% of operating speed was selected as the design speed for the primary coolant pumps. For the overspeed condition, which would not persist more than 30 seconds, pump operating temperatures would remain at about the design value.

(f) Bursting speed of the flywheel.

Bursting speed of the flywheels have been calculated on the basis of Robinson's results[1], to be 3900 rpm, more than three times the operating speed. This is confirmed using Griffith-Irwin theory as detailed in Reference 2.

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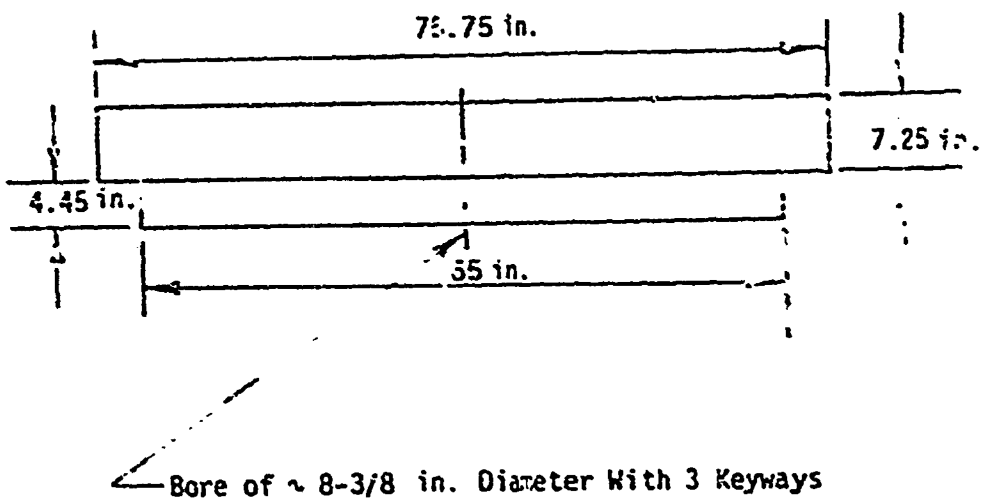
(1) Ernest L. Robinson, "Bursting Tests of Steam-Turbine Disk Wheels," Transactions of the A.S.M.E. July 1944.

(2) Application of the Griffity-Irwin Theory of Crack Propagation to the Bursting behavior of discs, including analytical and experimental studies - by D. H. Winne and B. M. Wundt.

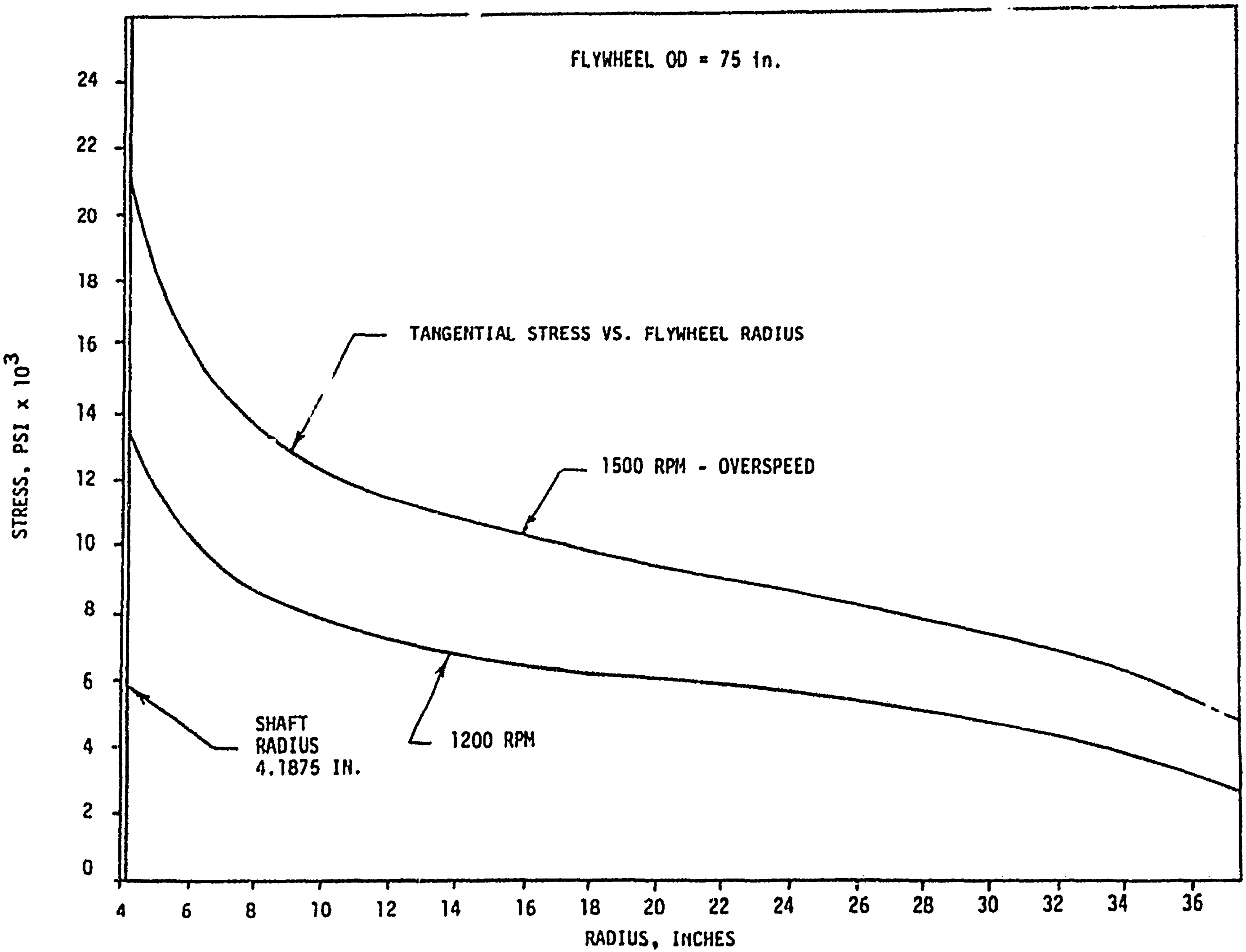
ASME - Paper No. 57-A-249. Dec. 1, 1957.

FLYWHEEL



NOTE: The plates are bolted together with the bolts aligned perpendicular to the planes of the plates.

Figure 4.2.1-1



FLYWHEEL OD = 75 in.
TANGENTIAL STRESS VS. FLYWHEEL RADIUS
1500 RPM - OVERSPEED
1200 RPM
SHAFT
RADIUS
4.1875 IN.
STRESS, PSI x 10^3
RADIUS, INCHES
0
2
4
6
8
10
12
14
16
18
20
22
24
4
6
8
10
12
14
16
18
20
22
24
26
28
30
32
34
36

Figure 4.2.1-2

QUESTION 4.2.2

Please provide a description of the protection provided to prevent pump motor overspeed. What inherent design features, if any, limit overspeed of the pump-motor-flywheel shaft?

ANSWER

During normal operation, the reactor coolant pumps are supplied from the unit auxiliary bus and are therefore tied to the turbine generator frequency (speed). On occurrence of unit (turbine) trip the pump electrical busses are tripped from the auxiliary transformer without any intentional delay.

On most electrical and mehanical events which cause the turbine to be tripped, the reactor coolant pump busses and the unit are tripped simultaneously and the pumps will therefore not exceed their normal or pretrip running speed. If for some unlikely reason the only plant trip is a turbine overspeed trip (mechanical - hydraulic trip), then the pump trip will be initiated by the turbine hydraulic system and the trip point will be between 106 and 110 percent of the turbine generator synchronous speed. The turbine overspeed trip point will be set at 105.5 percent of synchronous speed (1900 rpm).

QUESTION 4.2.3

Failure of the bearings on the primary pump motor shaft or of the shaft itself could lead to creation of a missile consisting of the flywheel and part of the motor shaft. Either failure could conceivably lead to creation of missiles through breakup of the flywheel. Provide the results of an analysis of the effects of applicable load combinations, including seismic loads, on the pump motor unit, and indicate the margins against failure of the bearings, the shaft, and other critical components.

ANSWER

The Reactor Coolant Pump motor bearings are of conventional design the radial bearings are the segmented pad type, and the thrust bearings are tilting pad Kingsbury bearings. All are oil lubricated - the lower radial bearing and thrust bearings are submerged in oil, and the upper radial bearing is oil fed from an impeller integral with the thrust runner. Low oil levels would signal an alarm in the control room and require shutting down of the pump. Each motor bearing contains embedded temperature detectors, and so initiation of failure, separate from loss of oil, would be indicated and alarmed in the control room as a high bearing temperature. This, again, would require pump shut down. Even if these indications were ignored, and the bearing proceeded to failure, the low melting point Babbitt metal on the pad surfaces would ensure that no sudden seizure of the bearing would occur. In this event the motor would continue to drive, as it has sufficient reserve capacity to operate, even under such conditions. However, it would demand excessive currents and at some stage would be shut down because of high current demand.

It may be hpothesized that the pump impeller might severely rub on a stationary member and then seize. Analysis has shown that under such conditions, assuming instantaneous seizure of the impeller, the pump shaft would fail in torsion just below the coupling to the motor. This would constitute a loss of coolant flow in the one loop, the effect of which is analyzed in Section 14.

Following the seizure, the motor would continue to run without any overspeed, and the flywheel would maintain its integrity, as it would still be supported on a shaft with two bearings.

There are no other credible sources of shaft seizure other than impeller rubs. Any seizure of the pump bearing would be precluded by shearing of the graphitar in the bearing. Any seizure in the seals would result in a shearing of the anti-rotation pin in the seal ring. The motor has adequate power to continue pump operation even after the above occurrences. Indications of pump malfunction in these conditions would be initially by high temperature signals from the bearing water temperature detector, and excessive No. 1 seal leakoff indications, respectively. Following these signals, pump vibration levels would be checked. These would show excessive levels, indicating some mechanical trouble. Again, the pump would be shut down for investigation.

The design specifications for the reactor coolant pumps include as a design condition the stresses generated by a maximum hypothetical earthquake ground acceleration of 0.2g. Besides examining the externally produces loads from the nozzles and support lugs, an analysis is made of the effect of gyroscopic reaction on the flywheel and bearings and in the shaft, due to rotational movements of the pump about a horizontal axis, during the maximum seismic disturbance.

The pump would continue to run unaffected by such conditions. In no case does any bearing stress in the pump or motor exceed or even approach a value which the bearing could not carry.

The design requirements of the bearings are primarily aimed at ensuring a long life with negligible wear, so as to give accurate alignment and smooth operation over long periods of time. To this end, the surface bearing stresses are held at a very low value, and even under the most severe seismic transients or other accidents, do not begin to approach loads which cannot be adequately carried for short periods of time.

because there are no established criteria for short-time stress-related failures in such bearings, it is not possible to make a meaningful quantification of such parameters as margins to failure, safety factors, etc. A qualitative analysis of the bearing design, embodying such considerations, gives assurance of the adequacy of the bearing to operate without failure.

As is generally the case with machines of the size, the shaft dimensions are predicated on avoidance of shaft critical speed conditions, rather than actual levels of stress.

There are many machines as large as, and larger than these, that are designed to run at speeds in excess of first shaft critical. However, it is considered desirable in a superior product to operate below first critical speed, and the Reactor Coolant Pumps are designed in accordance with this philosophy. This results in a shaft design which, even under the severest postulated transient, gives very low values of actual stress. While it would be possible to present quantitative data of imposed operational stress relative to maximum tolerable levels, if the mode of postulated failure were clearly defined, such figures would have little significance in a meaningful assessment of the adequacy of the shaft to maintain its integrity under operational transients. However, a qualitative assessment of such factors give assurance of the conservative stress levels experienced during these transients.

So in each of these cases, where it is the functional requirements of the component that controls its dimensions, it can be seen that if these are met, the stress-related failure cases are more than adequately satisfied.

It is thus considered to be out of the bounds of reasonable credibility that any bearing or shaft failure could occur that would endanger the integrity of the pump flywheel.

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QUESTION 4.3.1

How much higher will the neutron fluence exposure of the specimens be than the fluence at the vessel wall?

ANSWER

The factor by which the maximum specimen exposure exceeds that at the vessel wall (at the location of maximum vessel wall exposure) has a maximum value of 2.6. 4 of the 8 irradiation specimens will lead the vessel wall maximum exposure by this factor.

QUESTION 4.3.2

Explain why Indian Point Unit 2 has a predicted fluence at the vessel wall of 2.4×10^{19} n/cm^2 and the prediction for Indian Point Unit 3 is 1.4×10^{19} n/cm^2.

ANSWER

For Indian Point Unit 3 the maximum integrated fast neutron ($E > 1$ Mev) exposure of the vessel is computed to be 2.6×10^{19} n/cm^2 for 40 years operation at 80 per cent load factor, at a power level of 3025 MWt, as reported in Supplement 1 to the Preliminary Facility Description and Safety Analysis Report.

This calculated neutron exposure exceeds the value of 1.4×10^{19} n/cm^2 ($E > 1$ Mev) reported in the Preliminary Facility Description and Safety Analysis Report. The reasons for the increase are:

a) Core design considerations leading to the adoption of a radial power distribution which includes an increased energy generation in the peripheral assemblies of the core; and

b) The associated effect on the azimuthal variation of fast neutron fluxes at the reactor vessel inner surface.

The difference between the maximum exposure of the vessel for Units 2 and 3 is due to the lower power level of Unit 2.

Question 4.4.1

The maximum leak rate from an unidentified source will be permitted during operation, and the predi ted crack size corresponding to this leak rate. Include a discussion of the bases for the selected leak rate, and a description of the analytical methods used to establish the relationship between the leak rate and the crack size.

Answer 4.4.1

The maximum leak rate from an unidentified source that will be permitted during normal operation is 5 gpm.

Leakage from the Reactor Coolant System is collected in the containment or by the other closed systems. These closed systems are: the Steam and Feedwater System, the Waste Disposal System and the Component Cooling System, Assuming the existence of the maximum allowable activity in the reactor coolant, the rate of 5 gpm unidentified leakage is a conservative limit on what is allowable before the guidelines of 10CFR20 would be exceeded. This is shown as follows: If the reactor coolant activity is $60/\bar{E}$ μ Ci/cc ($\bar{E}$ = average beta plus gamma energy per disintegration in Mev) and 5 gpm of leakage is assumed to be discharged through the air ejector, or through the Component Cooling system vent line, the yearly whole body dose resulting from this activity at the site boundary, using an annual average $\chi/Q - 1.2 \times 10^{-6}$ sec/m^3 is 0.20 R/yr, compared with the 10CFR20 guidelines of 0.5 R/yr.

With the limiting reactor coolant activity and assuming initiation of 5 gpm leak from the Reactor Coolant System to the Component Cooling System, the radiation monitor in the component cooling pump inlet header would annunciate in the control room and initiate closure of the vent line from the surge line in the Component Cooling system, within less than one minute. In the case of failure of the closure of the vent li... and resulting continuous discharge in the atmosphere via the component cooling surge tank vent, the resultant dose at the site boundary would be 0.20 R/yr, as given above.

Leakage directly into the containment indicates the possibility of a breach in the coolant envelope. The limitation of 5 gpm for a source of leakage not identified is sufficiently above the minimum detectable leakage rate to provide a reliable indication of leakage. The 5 gpm limit is well below the capacity of one coolant charging pump (98 gpm).

A meaningful relationship between leak rate and crack size cannot be found because for a given crack size the leak rate could be anything depending upon the assumed crack configuration. Therefore, leak rate detection is not relied upon for assuring the integrity of the primary system pressure boundary during operation. The conservative approach which is utilized in the design and fabrication of the components which constitute the primary system pressure boundary together with the operating restrictions which are imposed for system heat-up and cooldown give adequate assurance that the integrity of the primary system pressure boundary is maintained throughout plant life. The periodic examination of the primary pressure boundary via the proposed in-service inspection program (specified in the Technical Specification to be submitted) will physically demonstrate that the operating environment will have no deleterious effect on the primary pressure boundary integrity.

No attempt is made, nor is it required, to relate crack size to leak rate.

The maximum unidentified leak rate of 5 gpm which is permitted during normal operation is well within the sensitivity of the leak detection systems incorporated within the containment, and it reflects good operating practice based on operating experience gained at other PWR plants. Detection of leakage from the primary system directs the operators attention to potential sources of leakage such as valves, and permits timely evaluation to ensure that any associated activity release does not constitute a public hazard, that the reactor coolant inventory is not significantly affected and that the leakage is well within the capability of the containment drainage system.

QUESTION 4.4.2

Describe the leak detection systems provided for other Class 1 fluid systems, and list those Class 1 fluid systems for which no special leak detection systems is provided.

ANSWER

For Class 1 systems located outside the containment, leakage is determined by one or more of the following methods:

a) For Systems containing radioactive fluids, leakage to the atmosphere would result in an increase in local atmospheric activity levels and would be detected by either the plant vent monitor or by one of the area radiation monitors. Similarly leakage to other systems which do not normally contain radioactive fluids would result in an increase in the activity level in that system.

b) For closed systems such as the component cooling system, leakage would result in a reduction in fluid inventory.

c) All leakage would collect in specific areas of the building for subsequent handling by the building drainage systems, e.g. leakage in the vicinity of the residual heat removal pumps would collect in the sumps provided, and would result in operation, or increased operation, of the associated sump pumps.

Details of how these methods are utilized to detect leakage from Class 1 systems located outside of the containment are given in Section 6.7.

Refer to Table 1.

TABLE 1

CLASS 1 FLUID SYSTEMS FOR WHICH

NO SPECIAL LEAK DETECTION IS PROVIDED

	System	Remarks on Leakage Detection
1.	Residual Heat Removal (RHR)	Refer to item a,b,c, and Section 6.7.
2.	Component Cooling	Refer to a,b,c, and Section 6.7.
3.	Service Water	Refer to item c and Section 6.7.
4.	Auxiliary Feedwater	Visual
5.	Waste Disposal	Auxiliary building sump pump operation and refer to item a.

QUESTION 4.4.3

Identify and list normal leakage paths from the primary system. Estimate the extent of leakage from each source and indicate whether the leakage is to be collected or not. How will an increase in leakage from these paths be detected?

ANSWER

In considering potential leakage from the reactor coolant system containing primary coolant at high pressure, four categories should be considered:

I Leakage to the reactor coolant drain tank.

II Leakage to the pressurizer relief tank.

III Leakage to the containment environment.

IV Leakage to the interconnecting systems.

For clarity, each of these paths are discussed in turn.

I - Paths Directed to the Reactor Coolant Drain Tank

The routes directed to the Reactor Coolant Drain Tank may be summarized as follows:

a. RCS Loop Drains

b. Accumulator Drains

c. Auxiliary System Equipment Drains

d. Excess Let-down

e. Valve Leakoffs

f. Reactor Coolant Pump Seal Leakage

g. Reactor Flange Leak-off

Of these paths, (a) through (d) do not present a leakage load on the RCD tank during normal operation; leakage from the high pressure systems is not expected because of the use of double isolation valves. Paths (e) through (g) merit some discussion.

e. <u>Valve Leak-offs</u>

<u>Source</u> - there are some twenty-three valves provided with leak-offs in the containment. Of these valves, only four valves on the reactor coolant system, one valve in the Chemical and Volume Control System and four valves in the safety injection system will normally have their valve stem packings subjected to pressure.

PCV-455A PCV-455B	Pressurizer spray valves which are assumed to be modulating
535 536	Pressurizer relief isolation valves which are normally fully open are provided with backseats which limit leakage
LCV 459	First isolation valve in the letdown line which is normally fully open and which is provided with a backseat to limit leakage
894A 894B 894C 894D	Accumulator isolation valves which are normally open are provided with backseats. Only leakage would be of borated non-radioactive water.

Estimated Leakage - Total leakage of reactor coolant fluid during normal power operation is conservatively estimated to be 15 cc/hr. This is made up as follows. Since the back-seats for valves 535, 536 and LCV 459 are capable of limiting leakage to less than 10 cc per hour per inch of stem diameter, assuming no credit for valve packing, it is assumed that the valves will leak at this rate. Hence for these three valves, a total leakage of 3 cc/hr is assumed.

For the modulating valves PCV, 455A and B, leakage is assumed to be 6 cc/hr/valve even though the packing of all modulating valves is adjusted to give negligible leakage during the preoperational system testing and periodically throughout plant life thereafter. Hence for these two valves, a total leakage of 12 cc/hr is assumed.

Indication to operator - The operator is alerted to abnormal conditions by an increase of the drain tank water temperature and eventually the change in tank level. Drain tank temperature, pressure, and level are continuously indicated on the "waste disposal/boron recycle" panel in the auxiliary building. High pressure, high temperature, high level and low level are annunciated on that panel. Any alarm on the WDS/BR panel causes annunciation of a single window on the main control board.

f. Reactor Coolant Pump Seals

Source - Charging flow is directed to the reactor coolant pumps via a seal-water-injection filter. It enters the pumps at a point between the labyrinth seals and the No. 1 face seal. Here the flow splits and a portion (normally about 5 gpm) enters the reactor coolant system via the labyrinth seals and thermal-barrier-cooler cavity. The remainder of the flow (normally about 3 gpm) flows up the pump shaft (cooling the lower bearing) and leaves the pump via the No. 1 seal where its pressure is reduced to about 25-30 psig and its temperature is increased from 130°F to about 136°F. The labyrinth flows (20 gpm total for four reactor coolant pumps) are removed from the system as a portion of the letdown flow. The No. 1 seal discharges (12 gpm total for four reactor coolant pumps) flow to a common manifold and then via a filter (seal-water filter) through the seal-water heat exchanger (where the temperature is reduced to about 130°F) to the volume control tank.

The leak-off system between the No. 2 and No. 3 seals is considered to be part of the Reactor Coolant System. The leak-off system collects leakage passed by the No. 2 seal, provides a constant backpressure on the No. 2 seal and constant pressure on the No. 3 seal. A standpipe is provided to give a constant backpressure during normal operation. The first outlet from the standpipe is orificed to permit normal No. 2 seal leakage to flow to the reactor coolant drain tank; excessive No. 2 seal leakage will result in a rise in the standpipe level and eventual overflow to the reactor coolant drain tank via a second overflow connection.

Leakage - The normal No. 2 seal flow will be 3 ghp/pump. This is the value specified in the Reactor Coolant Pump Equipment Specification.

Indication to Operator - Level instrumentation on the standpipes is provided to alert the operator to abnormal conditions. The standpipe consists of a pipe with an orificed overflow at the mid point, a normally closed drain (for service) at the bottom, and a free-flowing overflow at the top. Normal No. 2 seal leakage will flow freely out the mid-point overflow. Excessive leakage will "back-up" in the standpipe until it overflows out the top. A level switch in the upper standpipe actuates an annunciator indicating excessive flow. A level switch in the lower standpipe causes annunciation of the opposite condition which could result in undesirable dry operation of the No. 3 seal.

g. Reactor Vessel Flange Leak-off

Source - The reactor vessel flange and head are sealed by two metallic O rings. To facilitate leakage detection, a leak-off connection is placed between the two O-rings and a leak-off connection is placed beyond the outer O-ring. Piping and associated valving is provided to direct any leakage to the reactor coolant drain tank.

Leakage - During normal operation, it is anticipated that the leakage will be negligible since it is specified in the Reactor Vessel Equipment Specification that there is to be zero leakage past the outer O-ring under normal operating and transient conditions.

Indication to Operator - A temperature detector will indicate leakage by a high temperature alarm. The operator is further alerted by the associated increase in drain tank water temperature and eventually the change in tank level.

II - Paths Directed to Pressurizer Relief Tank

Source - The pressurizer R.T. condenses and cools the discharge from the pressurizer safety and relief valves. Discharge from smaller relief valves located inside the containment is also piped to the relief tank. During normal operation, leakage could possibly occur from either the pressurizer safety valves, pressurizer relief valves or the CVCS let-down station relief valve.

Leakage - During normal operation, the leakge to the pressurizer relief tank is expected to be negligible since the valves are designed for essentially zero leakage at the normal system operating pressure, as specified in the respective valve equipment specifications.

Indication to Operator - For each valve, temperature detectors are provided in the discharge piping to alert the operator to possible leakage.

The rate of increase of the water temperature in the pressurizer relief tank and the level change will indicate to the operator the magnitude of the leakage. In the event of excessive leakage into an interconnecting system causing lifting of the local relief valves, the operator would again be alerted to the situation by a rising tank water temperature (ref. disc· sion for category IV).

III - Releases to the Containment Environment

Source - The main contributors of leakage to the containment environment may be listed as follows:

a. Valve stem leakages

b. Reactor Coolant pump No. 3 seal leakage

c. Weld leakages

d. Flange leakages

a. Valve stem leakage

The modulating valves within the containment are provided with leak-off connections which in turn are piped to the reactor coolant drain tank. Of the remaining valves which serve lines and components containing reactor coolant, only two are not normally fully open or fully closed; i.e. the continuous spray by-pass needle valves around the main spray valves. The remaining valves are of the back seated type which prevent the valve stem packings from being subjected to high pressures when in the open position.

b. Reactor coolant pump No. 3 seal leakage

A small continuous leakage is anticipated past the No. 3 seal to the containment environment; this fluid will be charging water. The No. 3 seal leak-off is diverted to the local open drains and is thus released to the containment environment.

c. Weld flanges

The welded joints throughout the system are subjected to extensive non-destructive testing, leakage through metal surfaces and welded joints is very unlikely.

d. Flange joints

There are a number of flanged joints in the system all of which will be subjected to leak testing before power operation. Experience has shown that hydrostatic testing is successful in locating leaks in a pressure containing system.

Methods of leak location which can be used during plant shutdown include visual observation for escaping steam or water or for the presence of boric acid crystals near the leak. The boric acid crystals are transported outside the Reactor Coolant System in the leaking fluid and deposited by the evaporation process.

Leakages - The main contributors to leakage to the containment environment are considered to be (a) and (b); experience with operating reactors has shown that following the normal pre-operational testing, leakage from these sources are negligible.

Valve stems

Normally open valves have backseats which limit leakage to less than one cubic centimeter per hour per inch of stem diameter assuming no credit for packing in the valve. Normally closed glove valves are installed with recirculation flow under the seat to prevent stem leakage from the more radioactive fluid side of the seat.

On the basis of these pessimistic assumptions, the leakage from valves is estimated to be approximately 50 cc/hr.

Reactor coolant pump No. 3 seal leakage

The fluid will be charging water and is anticipated to be of the order 100 cc/hr. per pump. This is the valve specified in the Reactor Coolant Pump Equipment Specification.

Conclusion

On the basis of the above, the analysis of the situation indicates a total leak rate to the containment environment of the order 450 cc/hr. For design purposes, 50 lb/day (i.e. 1000 cc/hr. is assumed.

Indication to Operator - The methods of indication of leakage to the containment environment are discussed in detail in Section 6 7 of the FSAR.

IV - Leakage to Interconnecting Systems

Each of the interconnecting systems are dealt with in turn.

System	Discussion
CVCS	This is a normally operating interconnecting system with redundancy for isolating purposes if required.
SS	In the event of sample valves failing to close or seat, adequate redundancy is provided by containment isolation valves; the piping between the sets of valves is designed for RCS pressure.
RHR Hot Leg Connection	Two isolation valves are provided; in the unlikely event of leakage past the two valves, interconnecting piping is provided to enable pressure relief via the RHR loop relief valve to the pressurizer relief tank.
RHR Cold Leg	In the unlikely event of leakage past the accumulator check valves, RHR loop check valves and the motorized isolation valves; pressure relief will take place via the RHR loop relief valves to the pressurizer relief tank.

SIS High Head Pump Injection Lines	In the event of leakage past a check valve and motorized gate valve in any one of the four lines, pressure relief will take place to the PRT via the relief valve in the SIS test line.
SIS Accumulator Connections	Provisions have been made to check the leak tightness of the accumulator check valves. The implications of leakage past these valves are discussed in Section 6.2 of the FSAR.

Although leakage of primary fluid to the secondary system via the steam generator primary/secondary boundary is not expected during normal operation because of the conservative design of the U-tubes in the steam generator, any such leakage would result in an increase of activity level in the secondary system and would be detected by the condensor air ejector gas monitor or by the steam generator liquid sample monitor. (Reference is made to Section 11.2 of the FSAR.)

QUESTION 4.4.4

Since initial operation will be substantially without benefit of the containment air particulate monitor and radioactive gas monitor to indicate leaks, describe the preoperational test program to verify performance of the humidity detector and the condensate measuring system.

ANSWER

During the period of plant hot functional testing, a reactor coolant leak of known magnitude will be simulated inside the containment vessel, and the performance of the humidity detector and condensate measuring system will be observed. The leak will be simulated by introducing steam into one of the loop compartments during a period when containment atmospheric conditions are stable and the fan-cooler units are operating. The increase in containment atmosphere moisture content, as indicated by the humidity detectors, will be recorded as a function of time following initiation of the simulated leak. As a check, the same information will be determined independently using different instrumentation. Elapsed time until condensation on the fan-cooler unit cooling coils begins, as indicated by the condensate measuring devices, will be recorded and compared with the calculated valve based on the initial containment humidity. Steam flow will continue, and the performance of the condensate measuring devices in indicating the magnitude of steady cooling coil runoff will be observed.

QUESTION 4.5

Although it is stated that the reactor coolant system valves, fitti.gs and piping are designed, fabricated, inspected and tested in conformance with the USAS B31.1 Code for Power Piping, we wish to evaluate the degree to which they meet the requirements of USAS B31.7 Code. Accordingly, discuss the following in detail:

2. The design and stress analysi' criteria employed in comparison with those of USAS B31.7. For example, how do these criteria assure the absence of a continuing cycle of plastic deformation, i.e., shakedown; how are thermal stresses and thermal discontinuity stresses due to radial and longitudinal temperature gradients considered?

ANSWER

2. The design and stress criteria specified in USAS B31.7 are not directly comparable to that of USAS B31.1 (1955). The following describes how USAS B31.1 (1955) was used in Indian Point Unit No. 2 design. A thermal expansion flexibility stress analysis was performed on the main primary coolant piping and pressurizer surge line in accordance with the criteria set for.n in USAS B31.1 (1955) to assure that the stress range and number of thermal cycles (usage factor) are safely within the limits prescribed in B31.1. In addition, seismic analyses were performed on the composite piping which included the combined stress effects of all the sustained (pressure and weight) loadings plus seismic vertical/horizontal loading components. The resultant reactions of the piping due to the separate and combined effects of thermal, sustained, and seismic loadings were factored into the checking of the final design of the equipment nozzles to which the piping is interconnected. In turn, the equipment supporting structures are checked for adequate design including the added effects of these same loadings. Thus the total design analyses including pipe, equipment, and structures considers the effects of thermal expansion, sustained and seismic loadings with a normal usage factor.

Thermally-induced stresses arising from temperature gradients are limited to a safe and low order of magnitude in assigning a maximum permissible time rate of temperature change on plant heat up, cool down, and incremental loadings in the plant operation procedure.

An added margin of conservatism is obtained through the use of thermal sleeves in nozzles wherein a cold fluid is introduced into a pipe conveying a significantly hotter fluid or visa versa. Typical examples are the charging line, pressurizer surge, and residual heat return nozzle connections to the primary coolant loop piping.

The use of thermal sleeves is not a specific requirement in B31.7.

QUESTION 4.5

Although it is stated that the reactor coolant system valves, fittings and piping are designed, fabricated, inspected and tested in conformance with the USAS B31.1 Code for Power Piping, we wish to evaluate the degree to which they meet the requirements of USAS B31.7 Code. Accordingly, discuss the following in detail.

4. Indicate whether this same comparison holds for the balance of the plant; if not, please discuss the remainder of the Class I piping in the same manner.

ANSWER 4.5.4

The Quality Assurance criteria specified in the response to questions 4.5.1 and 4.5.3 apply to all nuclear Class I piping and fittings.

The analysis criteria specified in the response to question 4.5.2 applies to all the main primary coolant piping and the pressurizer surge line.

Additional thermal flexibility plus seismic stress analysis are being performed on nuclear Class I Auxiliary systems as discussed in the response in the Question 1.9.

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QUESTION 4.5

Although it is stated that the reactor coolant system valves, fittings, and piping are designed, fabricated, inspected and tested in conformance with the USAS B31.1 Code for Power Piping, we wish to evaluate the degree to which they meet the requirements of USAS B31.7 Code. Accordingly, discuss the following in detail:

3. The fabrication requirements specified as compared with those of USAS B31.7.

ANSWER

3. Shop and field fabrication requirements, documentation, and quality assurance examinations all comply with those found in USAS B31.7 for Class I Nuclear Piping.

QUESTION 4.6

Table 4.1-8 of the FSAR lists the number of design cycles for certain transient conditions which are stated to be estimates for equipment design purposes (40 year life) and not intended to accurately represent the actual number of transients expected, or to reflect actual operating experience. Discuss the margin between the number of design cycles and the number of operating cycles the plant is expected to experience during its lifetime. In addition, please discuss the number of stress cycles used in the fatigue evaluation as compared with the expected number of operating cycles.

ANSWER

To provide the necessary high degree of integrity for the equipment in the reactor coolant system, the transient conditions selected for equipment fatigue evaluation are based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients. To a large extent, the specific transient operating conditions considered for equipment fatigue analyses are based upon engineering judgment and experience. Those transients are chosen which are representative of transients to be expected during plant operation and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior.

Clearly it is difficult to discuss in absolute terms the transients that the plant will actually experience during the 40 years operating life. For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients.

1. Heatup and Cooldown

The heatup or cooldown cases are conservatively represented by a continuous operation performed at a uniform temperature rate of 100°F per hour.

For these cases, the heatup occurs from ambient to the no-load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100°F per hour will not be attained because of other limitations such as:

a. Material NDT considerations which establish maximum permissible temperature rate of change, as a function of plant pressure and temperature, which are below the design rate of 100°F per hour.

b. Slower initial heatup rates when using pumping energy only.

c. Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

The number of such complete heatup and cooldown operations is specified at 200 times each which corresponds to five such occurrences per year for the 40-year plant design life. For the ideal plant only one heatup and one cooldown would occur per 100% full power year, i.e., the period between refueling. In practice experience to date indicates theat during the first year or so of operation additional unscheduled plant cooldowns may be necessary for plant maintenance; the frequency of maintenance shutdowns reduce as the plant matures. As experience was gained with Yankee-Rowe, the number of shutdowns decreased; for example Core II ran for a year from 1962 to 1963 with no cooldowns. Attached for reference is a summary of the Yankee-Rowe plant outage for the period 1964 through to 1969.

2. Unit Loading and Unloading

The unit loading and unloading cases are conservatively represented by a continuous and uniform ramp power change of 5% per minute between no load and full load. The reactor coolant temperature will vary with load as prescribed by the temperature control system. The number of each operation is specified at 14,500 times or once per day for the 40-year plant design life.

In practice the plant will be operated either at base load or on a load follow program less severe than between no load and full load at 5%

per minute. Figure 4.6-1 represents what may be regarded as a typical load follow program for Indian Point.

3. Step Increase and Decrease of 10%

The ±10% step change in load demand is a control transient which is assumed to be a change in turbine control valve opening which might be occasioned by disturbances in the electrical network into which the plant output is tied. The reactor control system is designed to restore plant equilibrium without reactor trip following a ±10% step change in turbine load demand initiated from nuclear plant equilibrium conditions in the range between 15% and 100% full load, the power range for automatic reactor control. In effect, during load change conditions, the reactor control system attempts to match turbine and reactor outputs in such a manner that peak reactor coolant temperature is minimized and reactor coolant temperature is restored to its programmed set point at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step load decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the reactor coolant system average temperature and pressurizer pressure also initially increase. Because of the power mismatch between the turbine and reactor and the increase in reactor coolant temperature will be ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature set point change is made as a function of turbine-generator load as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the reactor coolant decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash which reduces the rate of pressure decrease. Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step load increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature will be raised to a value above its initial equilibrium value at the beginning of the transient.

The number of each operation is specified at 2000 times or 50 per year for the 40-year plant design life.

4. Large Step Decreases in Load

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature will automatically initiate a secondary side steam dump system that will prevent a reactor shutdown or lifting of steam generator safety valves.

The number of occurrences of this transient is specified at 200 times or 5 per year for the 40-year plant design life. Reference to the Yankee-Rowe record indicates that this basis is adequately conservative.

5. Reactor Trip From Full Power

A reactor trip from full power may occur for a variety of causes resulting in temperature and pressure transients in the reactor coolant system and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the reactor coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued supply of feedwater and controlled dumping of secondary steam remove the core residual heat and prevent the steam generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the reactor protection system causes the control rods to move into the core.

The number of occurrences of this transient is specified at 400 times or 10 per year for the 40-year plant design. Reference to the Yankee-Rowe record indicates that this basis is indeed conservative.

6. Hydrostatic Test Conditions

The pressure tests outlined below are based upon information contained in reference (b) and apply to field hydrostatic tests conducted on the erected reactor coolant system. The number of tests given below does not include any allowance for pressure tests conducted on a specific component in the manufacturer's shop in accordance with vessel code requirements.

a. Primary Side Hydrostatic Test Before Initial Startup at 3110 psig

This hydro test is performed at a minimum water temperature of 100°F imposed by reactor vessel material DTT value of 100°F at beginning of life, and a maximum test pressure of 3110 psig. In this test, the primary side of the steam generator will be pressurized to 3110 psig coincident with the secondary side pressure of 0 psig. The reactor coolant system will experience 5 cycles of this hydro test.

7. Primary Side Post Operation Hydrostatic Test at 2485 psig

This type of hydro test will be performed to test the integrity of the reactor coolant system after a maintenance procedure has been completed in which the reactor coolant system boundary has been opened. To account for the shift in DTT on the reactor vessel due to irradiation effects later in life, this hydro test is performed at a minimum water temperature of 400°F and a test pressure of 2485 psig. The design heatup rate is limited to 100°F per hour. Since pumping power will be used to heat the water, the actual heatup rate will be considerably below 100°F per hour. The number of these tests is specified at 40 or one per year over the 40-year plant design life. The normal requirement will be that following a refueling operation, i.e., probably less than an average of once per calendar year over a 40 year life.

Refer also to Question 4.8.4 and answer.

All components in the Reactor Coolant System are designed to withstand the effects of transients that result in system temperature and pressure changes.

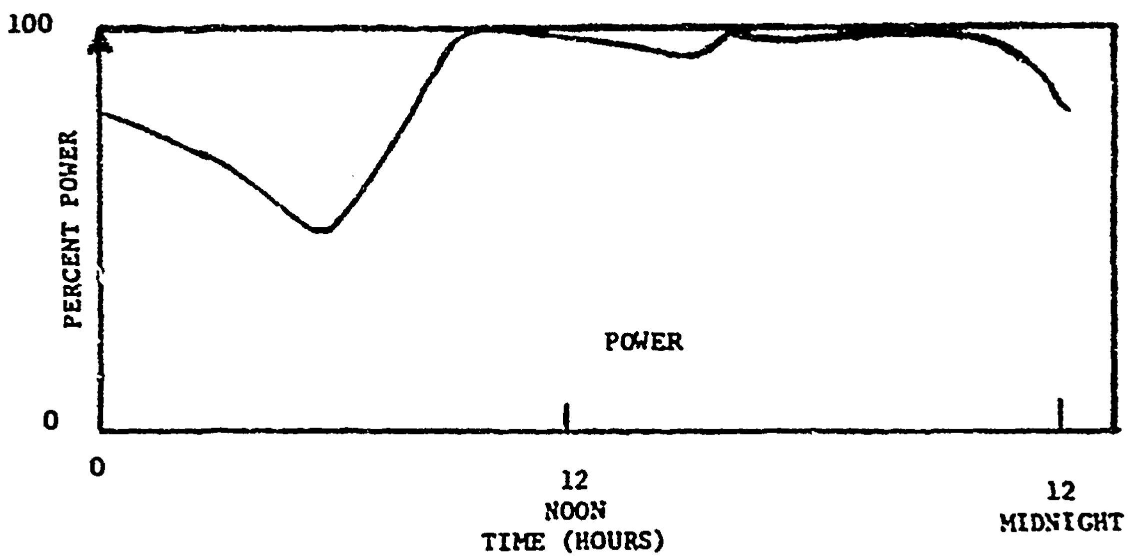
Stress intensity values at all critical points in the reactor vessel due to these excursions of pressure and temperature are determined for each of these transients, through systematic analytical procedures. These stress intensity values S_{ij} (i,j = 1,2,3) are plotted against time interval for each cycle. This plot may represent one or more stress cycles. For each cycle extreme values of S_{max} and S_{min} are determined. From these values, the largest S_{alt} (alternating stress intensity) is found.

For this value of S_{alt}, an allowable number of cycles (N) is determined through design fatigue curves established for different materials. The ratio of design cycles (n) to allowable cycles (N) gives the usage factor u_i. (i = 1, 2, 3 -------etc). Usage factor is determined in this manner for all transients. The cumulative usage factor is determined by summing the individual usage factors. The cumulative usage factor $(U = u_1 + u_2 + u_3 \ldots)$ is never allowed to exceed a value of 1.0.

SUMMARY OF PLANT OUTAGE FOR YANKEE-ROWE (1964 to 1969)

STARTING DATE	DURATION DAYS	DURATION HOURS	OUTAGE TYPE	CASE EQUIPMENT/SYSTEM
1/17/64	-	3.1	Forced	Turbine Trip
2/12/64	-	21.8	Scheduled	Control Rod Drop Testing
3/11/64	-	4.5	Forced	Moisture separator level switch tripped due to vibration
3/26/64	-	4	Forced	Control Valves Sticking
5/18/64	-	5.4	Forced	Low condensate pump discharge pressure
8/2/64	35	-	Scheduled	Refueling and general maintenance
8/9/64	-	2.4	Scheduled	Check of Overspeed Trip
8/11/64	-	14.7	Forced	Spurious Reactor Trip
10/18/64	-	12.2	Forced	Condenser Noise
10/22/64	-	22.4	Forced	Neutron Counter Gain Control
2/12/65	-	15.2	Forced	Switchyard Electric
3/5/65	-		Scheduled	Switchyear Electric
8/9/65	93	6	Scheduled	Refueling
11/26/65	2	20	Scheduled	Turbine Repair-Physics Testing
2/4/66	-	3.12	Forced	Reactor Scram
4/4/66	-	89.5	Schedule	Leaking Pressurizer Safety Valves
7/10/66	-	3.68	Forced	Reactor Scram
8/25/66	-	2.40	Forced	Reactor Scram
10/4/66	34	10.23	Scheduled	Refueling
12/24/66	-	2.88	Forced	Reactor Scram
12/28/66	-	2.12	Forced	Reactor Scram

STARTING DATE	DURATION DAYS/HOURS	OUTAGE TYPE	CAUSE EQUIPMENT/SYSTEM
3/8/67	11 21	Scheduled	Steam Generator Leak Repair
5/12/67	- 16.87	Scheduled	Condensor Cleaning
7/9/67	17 1.5	Scheduled	Steam Generator Leak Repairs
10/28/67	9	Scheduled	AEC Operator Examinations
10/13/67	2.6	Forced	Reactor Scram
3/23/68	38 days	Scheduled	Core VI-VII Refueling and maintenance
7/20/68	1/10	Scheduled	Repair Leak from No. 1 M.C. Pump Stator Cap
11/8/68	6 days-16.42 hours	Scheduled	Repair No. 4 Main Coolant Pump Thermal Barrier Leak and other Maintenance
1/18/69	1/2 .1	Scheduled	Operator Training
2/15/69	1/1.8	Scheduled	Operator Training
3/1/69	0/11	Scheduled	AEC Operator Examination
4/11/69	4/18	Forced	Repair Reactor Instrument Leak
7/17/69	0/4.8	Forced	Reactor Scram
8/2/69	53/18.5	Scheduled	Refueling maintenance
10/16/69	0/6.1	Forced	Reactor Scram
10/29/69	0/12	Scheduled	Turbine Valve Flange Steam Leak Repair



TYPICAL LOAD FOLLOWS PROGRAM

Figure 4.6-1

QUESTION 4.7

The FSAR states that vibration loads are considered in the design of the primary system. State the extent, methods and findings of the analyses which have been made. In this statement please include the following:

1. Discuss the normal and emergency modes of operation that have been considered.

2. State the design limits, amplitude and frequency that apply to these conditions.

ANSWER

Vibration loads have been considered in the design for the reactor internals, steam generator tube bundles, and the reactor coolant piping. Reactor coolant pump vibration is insignificant. Instrumentation is provided to check the vibration level of these pumps if an abnormal condition is suspected.

Reactor Internals

Model tests of the Indian Point Unit No. 2 reactor internals have been performed for normal operating and transient conditions. Results of the combined analytic and experimental work have been factored into the design.

Predicted stresses and deflections are in agreement with tests on reactors having similar internals design. Most recently, the results of the vibration tests performed on the Ginna reactor (reported in WCAP-7408, Westinghouse proprietary report) confirm that the tests agree very closely with the predicted performance and margins. A more extensive testing program will be performed during preoperational testing for Indian Point Unit No. 2 as described in the response to Question 13.1.

Allowable stress amplitude for flow induced vibration is extablished on the basis of the material fatigue properties (endurance limit of 20,000 psi for 10^{10} cycles). Since infinite cycle fatigue is a criterion, no limits are then necessary for frequency. Displacement amplitudes for reactor internals vibration are not governing. Stress limits are more restrictive.

An analysis of the dynamic response of the Indian Point Unit No. 2 internals under seismic and blowdown loads was made. Allowable criteria have been established and stresses and deflections determined to assure that seismic and blowdown loads will not prevent core shutdown or will not interfere with the effectiveness of the emergency core cooling system (reported in detail in WCAP 7332-L, Westinghouse proprietary report).

Steam Generators

a) Tube Vibration Analysis

In the design of Westinghouse 44 Series steam generators used on Indian Point Unit No. 2, consideration has been given the possibility of vibratory failure of tubes due to mechanical or flow induced excitation. This consideration includes detailed analysis of the tube supporting system as well as an extensive research program with tube vibration model tests at the Westinghouse R&D Laboratories.

The major cause of tube vibratory failure in heat exchanger components has been that due to hydrodynamic excitation by the fluid outside the tube. Consideration has been given by Westinghouse to (3) regions where the possibility of flow induced vibration may exist:

1. At the entrance of downcomer feed to the tube bundle (cross flow)
2. Along the straight sections of the tube (parallel flow)
3. In the curved tube section of the U-bend (cross flow)

From the description of these regions, it is noted that two types of flow exist, namely, cross flow and parallel flow. For the case of parallel flow, analysis is done to determine the vibratory deflections according to methods of Burgreen, et al (1) and Paidoussis (2) which give empirical data on tube vibration related to the hydrodynamic and elastic system properties. Analysis of the steam generator tubes indicates the flow velocities to be sufficiently below that required for damaging (fatigue or impacting) vibratory amplitudes. The support system, therefore, is deemed adequate to preclude parallel flow excitation.

For the case of cross flow excitation, it is noted in the literature that several techniques for the analysis of tube vibration exist. The earliest technique was that by Sebald and Nobles (3) which has been used widely by the industry and has resulted in safe tube support design. At present, the most popular techniques for vortex shedding analysis in tube arrays has been that due to Chen (4). Here the design pro lem is to ascertain that the tube natural frequency is well above the vortex shedding frequency in order that resonant vibration will not occur. The design of the Westinghouse steam generator is such that adequate tube supports are provided to assure tube natural frequencies above the vortex shedding frequency as computed by Chen. In addition, analyses according to methods of Sebald and Nobles (3) are performed.

Since the problem of cross-flow induced vibration was of major concern to the design of shell-tube heat exchangers, Westinghouse has given serious consideration to the experimental evaluation of the behavior of tube arrays under cross flow. While consideration was given to instrumentation of actual units in service, the hostile environment would limit the amount and quality of information obtained therefrom. As a result, it was deemed prudent to undertake a research program which would allow the study of fluid elastic vibrator behavior of tubes in arrays. A wind tunnel was built specifically for this purpose at the R&D Laboratories and Westinghouse has invested approximately 3 years of research into the study of this problem. In addition, the research facilities for tube vibration study have expanded with the construction of a water tunnel facility.

The results of this research have been to study confirmation of the Chen vortex shedding mechanism. Another study for vibration has been the evaluation of a fluidelastic mechanism unassociated with vortex shedding not commonly understood from the literature which could be a source of vibration failure. Westinghouse steam generators are evaluated on this basis in addition to the aforementioned techniques and found adequately designed. General results of this research are reported in a thesis by Conners (5). Testing has also been conducted using specific parameters of the 44 Series steam generator and the results show the support system to be adequate.

Summarizing the results of analysis and tests of 44 Series steam generator tubes for flow induced vibration, it can be stated that a check of support adequacy has been made using all published techniques believed appropriate to heat exchanger tube support design. In addition, the tube support system is consistent with accepted standards of heat exchanger design utilized throughout the industry (spacing, clearance, etc.). Furthermore, the design techniques have been supplemented with a continuing R&D program to understand the complex mechanism of concern here.

Further consideration has been given the possibility of mechanically excited vibration, in which resonance of external forces with tube natural frequencies must be avoided. It is believed that the transmissibility of external forces either through the structure of from fluid within the tubes is negligible and should cause little concern.

Finally, it should be noted that successful operational experience with several steam generator designs has given confidence in the overall approach to the tube support design problem.

b) <u>Tube Sheet Analysis</u>

The evaluation of Indian Point No. 2 steam generator tubesheets is performed according to rules of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels, Section III, Article 4 - Design. The design criteria considered encompass consideration of both steady state, transient and emergency operations specified in the Equipment Specification for the particular plant.

The evaluation of the tubesheet involves the heat conduction and stress analysis of the tubesheet, channel head, secondary shell structure for particular steady design conditions for which Code stress limitations are to be satisfied and for discrete points during transient operation for which the temperature/pressure conditions must be known to evaluate stress maxima and minima for fatigue life usage. In addition, limit analyses are performed to determine tubesheet capability to sustain emergency operating conditions for which elastic analysis does not suffice. The analytic techniques utilized are computerized and significant stress problems are verified experimentally to justify the techniques where possible.

The stress analysis of the tubesheet complex consists of performing an interaction analysis between tubesheet and attached shells to determine the interaction forces and moments between parts of the structure. For the perforated region of the tubesheet the flexural rigidity is based on studies of behavior of plates with square hole arrays utilizing techniques such as those reported by O'Donnell[6], Mahoney[7], Lemcoe[8], and others. Similarly, stress intensity factors are determined for square hole arrays using the combined equivalent plate interaction forces and moments applied to results of photoelastic tests of model coupons of such arrays as well as verification using computer analysis techniques such as "Point Matching" or "Collocation". The stress analysis considers stress due to symmetric temperature and pressure distribution as well as assymetric temperature distribution due to temperature drop across the tubesheet divider lane.

The fatigue analysis of the complex is performed at potentially critical regions in the complex such as the junction between tubesheet and channel head or secondary shell as well as at many locations throughout the perforated region of the tubesheet. For the holes for which fatigue evaluation is done, several points around the hole periphery are considered to assure that the maximum stress excursion has been considered. The fatigue evaluation is computerized to include stress maxima-minima excursions considered on an intra-transient basis.

In all cases evaluated, the Indian Point No. 2 steam generator tubesheet complex meets the stress limitations and fatigue criteria specified in Article 4 of the Code as well as emergency condition limitations.

c) <u>Tube-Tube Sheet Analysis</u>

Examination of the introductory paragraph 1-900 of ASME Boiler and Pressure Vessel Code, Section III - Nuclear Vessels (1968), reveals a precise explanation that consideration may be given to the stiffening effect of tubes in perforations, and staying action of the tubes if applicable, effect of stiffening on the plate stress levels, etc. Furthermore, it is noted that the stress analysis methods in Appendix I of Section III are described as accepted techniques for obtaining solutions to problems for which these procedures are applicable. It allows and requires use of other valid analytical or experimental techniques where necessary.

The Westinghouse analysis of the steam generator tubesheets is based on the stress and fatigue limitations outlined in Article-4 Design of Section III. The stress analysis techniques utilized include all factors considered appropriate to conservative determination of the stress levels utilized in evaluation of the tubesheet complex. The analysis of the tubesheet complex includes the effect of all appurtenances attached to the perforated region of the tubesheet considered appropriate to conservative analysis of stress for evaluation of basis of Section III stress limitations.

Due to the complex nature of the tube-tubesheet-shell-head structure, the analysis of the tubesheet requires the application of results of related research programs (such as the design data on perforated plates resulting from PVRC programs) and the utilization of current techniques in computer analysis, the application of which is verified by comparison of analytical and experimental results for related equipment.

Generally, the analytic treatment of the tube-tubesheet complex includes determination of elastic equivalent plate stress within the perforated region from an interaction analysis utilizing effective elastic constants appropriate to the nature of the perforation array. Tubes are considered to have negligible stiffening effect on the system and stress state due to welding or rolling not significant in the primary stress or fatigue evaluation of the tubesheet. Separate consideration is given the problem of interaction between tube and perforation and the stress levels consequent therefrom. Also considered in a conservative manner is the partition stress level due to boundary displacements of the tubesheet and hemispherical head.

A wide range of computational tools are utilized in these solutions including finite element, heat conduction and thin shell computer solutions. In addition, analysis techniques have been verified by photoelastic model tests and strain gaging of prototype models of an actu   ›am generator tubesheet.

Finally, in order to evaluate the ultimate safety of structural complex, a computer program for determining a lower-bound pressure limit for the complex based on elastic-plastic analysis has been developed and applied to the structure. This was verified by a strain gage steel model of the complex tested to failure.

d) Tube-Tubesheet Juncture Analysis

The evaluation of the tube-to-tubesheet juncture of Westinghouse PWR System steam generators is based on a stress analysis of the interaction between tube and tubesheet hole for the significant thermal and pressure transients that are applied to the steam generator in its predicted histogram of cyclic operation. The evaluation is based on the numerical limits specified in the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

Of importance in the analysis of the interaction system is the behavior of the tube hole, where it is recognized that the hole behavior is a function of the behavior of the entire tubesheet complex with attached head and shell. Hence, the output of the tubesheet analysis giving equivalent plate stress in the perforated region is utilized in determining the free boundary displacements of the perforation to which the tube is attached.

Analysis of the juncture for the fillet-type weld utilized in the Westinghouse steam generator design has been made with consideration of the effect of the rolled-in joint in the weld region as well as with the conservative assumption that the tube flexure relative to the perforation is not inhibited with the rolled-in effect.

The major concern in fatigue evaluation of the tube weld is the fatigue strength reduction factor to be assigned to the weld root notch. For this reason, Westinghouse has conducted low-cycle fatigue tests of tube material samples to determine the fatigue strength reduction factor and applied them to the analytic interaction analysis results in accordance with the accepted techniques in the Nuclear Pressure Vessel Code for Experimental Stress Analysis. The fatigue strength reduction factors determined therefrom are not different from that reported in the well known paper on the subject by O'Donnell and Purdy.(9) An actual tubesheet joint contained in a tubesheet has been successfully tested experimentally under thermal transient conditions much more severe than that to achieved in anticipated power plant operation.

In all cases, the tube-to-tubesheet juncture fatigue usage factor falls well within the tolerance accepted by the ASME Boiler and Pressure Vessel Code for Nuclear Vessels.

Reactor Coolant Pumps

The Reactor Coolant Pumps are checked during full flow tests for flow induced vibration. Each of the reactor coolant pumps is equipped with two International Research & Development Model 544 Vibration Pickups mounted at the top of the motor support stand. They are mounted in a horizontal plane, and so pick up radial vibrations of the pump. One is aligned parallel to the pump discharge; the other is aligned perpendicular to the pump discharge. Their signals are taken to a multi-point selector switch mounted outside the reactor containment. The signals from both reactor coolant pumps are taken to this selector switch. Also supplied is an I.R.D. Model 306W Vibration Meter. This is a portable device that may be plugged into the selector switch, and so the signal from any one pickup may be monitored at one time. The normal vibration level shown from the pickups is less than 0.001 inches. A maximum level of 0.002 inches is specified. The vibration levels of the reactor coolant pump can be checked if an abnormal condition is suspected.

There is no need for continous monitoring or recording (see response to 2.3.2). No analysis for vibration loading is performed for the Reactor Coolant Pumps.

Primary System Piping

The Reactor Coolant System piping has been designed for normal and emergency conditions. For emergency conditions, the piping is designed and analyzed for seismic loads and blowdown forces due to a loss-of-coolant accident By design, the main piping of the reactor coolant loop is not subjected to induced vibration due to pressure pulses from the reactor coolant pump impeller or the pistons of the charging pump.

The positive displacement charging pump has a damping device (surge pot) which effectively removes the pressure surges that might induce vibration in the piping system. Therefore, the mechanism for insuring satisfactory operation is the hydraulic damper.

By comparison, the perturbating frequency of the reactor coolant pump is quite high when compared to the piping natural frequency. Frequency separation, therefore, insures a very small probability of self-excited or sympathetic vibration. This is borne out of satisfactory operation of several representative coolant loops.

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4. Y. N. Chen, "Flow Induced Vibration and Noise in Tube Bank Heat Exchangers due to Von Karman Vortex Streets", ASME 67-vibr.-48.

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QUESTION 4.8

With regard to the reactor vessel:

1. Provide a summary description of the reactor vessel stress analysis which includes simple sketches showing the location and geometry of the areas of discontinuity or stress concentration as well as any other areas which were subjected to a detailed analysis.

ANSWER

The following components of reactor pressure vessel are analyzed in detail through systematic analytical procedures.

a. Control Rod Housings
b. Closure Head Flange and Shell
c. Main Closure Studs
d. Inlet Nozzle (and Vessel Support)
e. Outlet Nozzle (and Vessel Support)
f. Vessel Wall Transition
g. Core-Barrel Support Pads
h. Bottom Head to Shell Juncture
i. Bottom Head Instrument Penetrations etc.

Method of Analysis

Item (a). An interaction analysis is performed on the CRDM housing. The flange is assumed to be a ring and the tube a long cylinder. The different values of Young's Modulus and coefficients of thermal expansion of the tubes are taken into account in the analysis. The local flexibility is considered at appropriate locations. The closure head is treated as a perforated spherical shell with modified elastic constants. The effects of redundants on the closure head are assumed to be local only. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation is made for the J weld.

Item (b). The closure head, closure head flange, vessel flange, vessel shell and closure studs are all evaluated in the same analysis. An analytical model is developed by dividing the actual structure into different elements such as long sphere, ring, long cylinder and contilever beam, etc. An interaction analysis is performed to determine the stresses due to mechanical and thermal loads. These stresses are evaluated in light of the strength and fatigue requirements of the ASME Boiler and Pressure Vessel Code Section III.

A similar analysis is performed for the vessel flange to vessel shell juncture; and main closure studs. (Item c)

Item (d). For the analysis of nozzle and nozzle to shell juncture, the loads considered are internal pressure, operating transients, thermally induced and seismic pipe reactions, static weight of vessel, earthquake loading and expansion and contraction, etc. A combination of methods is used to evaluate the stresses due to mechanical and thermal loads and external loads resulting from seismic pipe reactions, earthquake and pipe break, etc.

For fatigue evaluation, peak stresses resulting from external loads and thermal transients are determined by concentrating the stresses as calculated by the above described methods. Combining these stresses enables the fatigue evaluation to be performed. Method of analysis for outlet nozzle and vessel supports (Item e) is the same as described above.

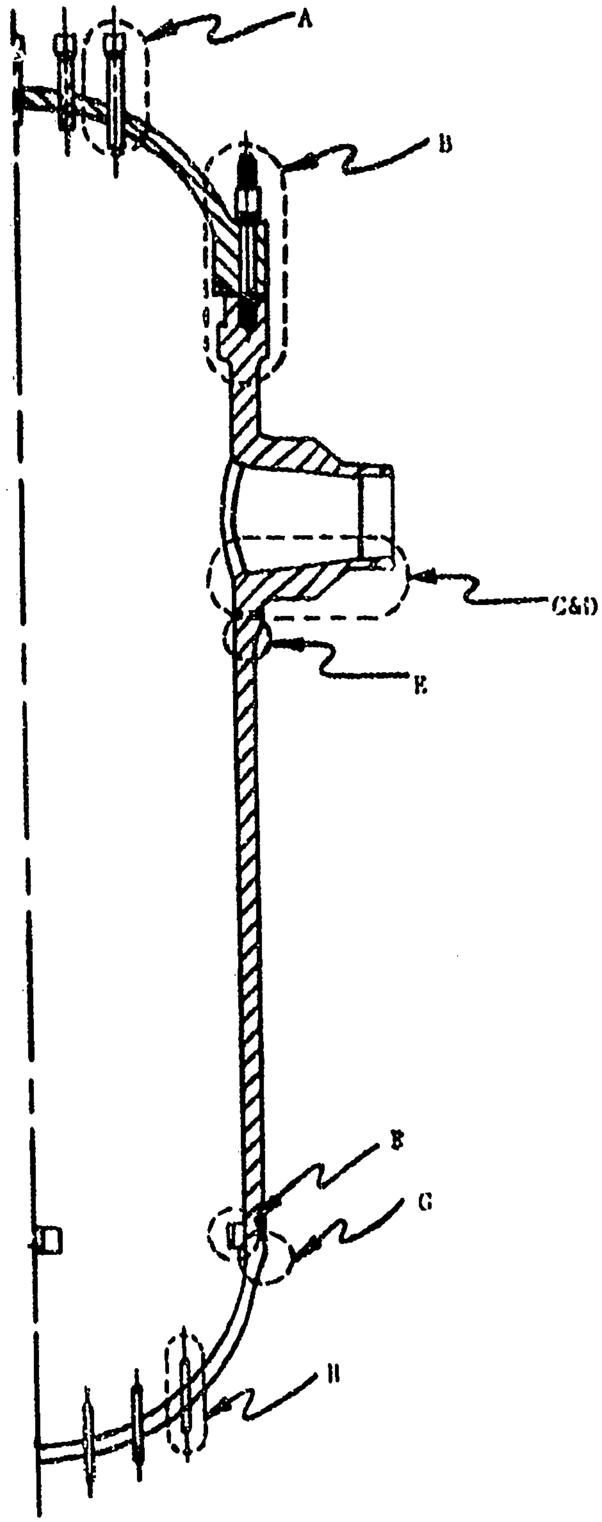
Item (f). Vessel wall transition is analyzed by means of a standard interaction analysis. The thermal stresses are determined by the skin stress method where it is assumed that the inside surface of the vessel is at the same temperature as the reactor coolant and the mean temperature of the shell remains at the steady state temperature. This method is considered conservative.

Item (g). Thermal, mechanical and pressure stresses are calculated at various locations on the pad and at the vessel wall. Mechanical stresses are calculated by the flexure formula for bending stress in a beam, pressure stresses are taken from the analysis of the vessel to bottom head juncture and thermal stresses are determined by the conservative method of skin stresses. The stresses due to the cyclic loads are multiplied by a stress concentration factor where applicable and used in the fatigue evaluation.

Item (h). The standard interaction analysis and skin methods are employed to evaluate the stresses due to mechanical and thermal stresses respectively. The fatigue evaluation is made on a cumulative basis where superposition of all transients is taken into consideration.

Item (i). An interaction analysis is performed by dividing the actual structure into an analytical model composed of different structural elements. The effects of the redundants on the bottom head are assumed to be local only. It is also assumed that for any condition where there is interference between the tube and the head no bending at the weld can exist. Using the mechanical and thermal stresses from this analysis a fatigue evaluation is made for the J weld.

The location and geometry of the areas of discontinuity and/or stress concentration are shown in the sketches attached, Figures 4.8-1, 4.8-2 and 4.8-3.



REACTOR VESSEL LONGITUDINAL SECTION

Figure 4.8-1
Supplement 5-
1/70

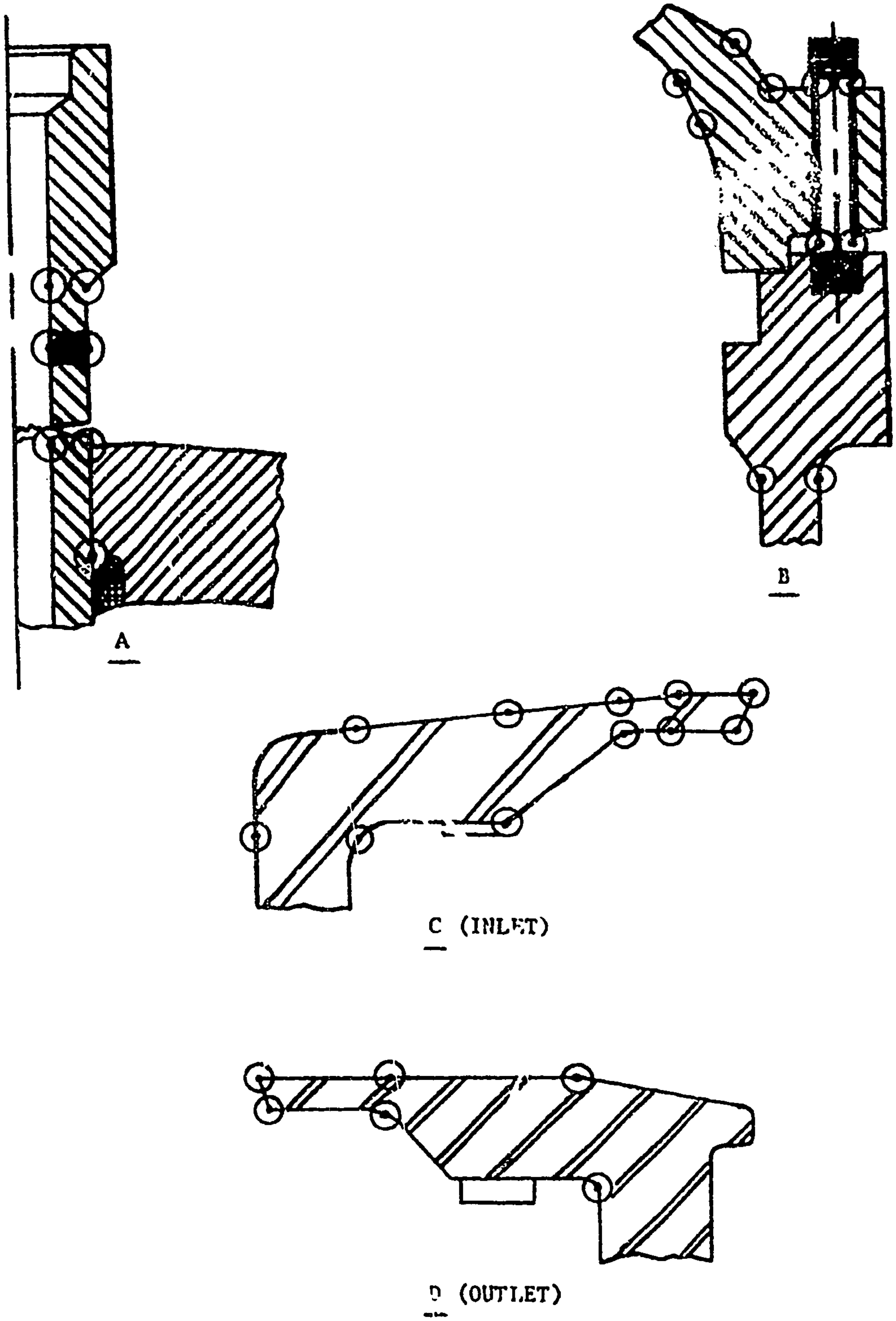


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Figure 4.8-2
Supplement 5
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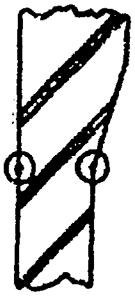

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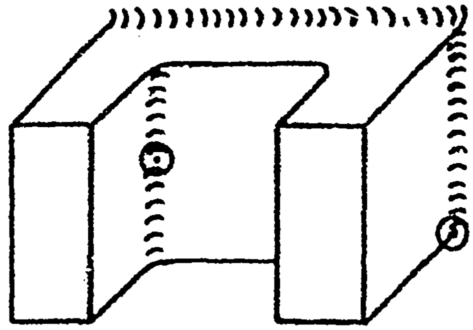

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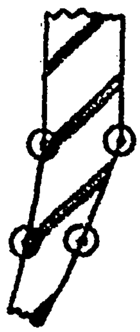

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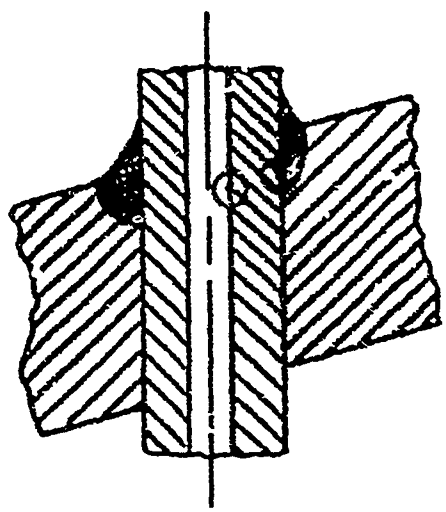

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NOTE:

THE POINTS CIRCLED IN THE SKETCHES REPRESENT THE GENERAL LOCATION AND GEOMETRY OF THE AREAS OF DISCONTINUITY AND/OR STRESS CONCENTRATION.

Figure 4.8-3

QUESTION 4.8.

With regard to the reactor vessel:

2. Describe any special requirements in addition to those specified in Section III of the ASME Code which are imposed on the Indian Point 2 reactor vessel designs by local state regulations.

ANSWER

The state of New York has adopted ASME Code Section III, Nuclear Vessels, and imposes no additional design requirements beyond those listed in this code.

QUESTION 4.8

With regard to the reactor vessel:

3. Have ring forgings been used for reactor vessel shell sections other than the closure flanges? If so, please provide a list giving the location of these forgings.

ANSWER

No.

QUESTION 4.8

With regard to the reactor vessel:

4. Discuss those transients, such as loss of flow and loss of load, that cause temperature and pressure excursions influencing the cumulative fatigue of the reactor vessel in a significant manner.

ANSWER

Loss of Load Transient. This is the most severe anticipated transient on the reactor coolant system. This transient applies to a step decrease in turbine load from full power occasioned by the loss of turbine load without immediately initiating a reactor trip. The reactor and turbine eventually are assumed to trip as a consequence of a high pressurizer level trip initiated by the reactor protection system. Figure Q4.5.4-1 gives the pressure-temperature transient assumed in the analysis for usage factor. This analysis is more severe than that reported in Section 14.1.8 of the FSAR.

Loss of Flow Transient. This transient applies to a partial loss of flow accident from full power in which a reactor coolant pump is tripped out of service as a result of a loss of power to that pump. The consequences of such an accident are a reactor and turbine trip, followed by automatic opening of the steam dump system and flow reversal in the affected loop. The net result of the flow reversal is a sizeable reduction in the hot leg coolant temperature of the affected loop. Figure Q4.8.4-2 gives the temperature transient assumed in the analysis for usage factor. This analysis is more severe than that reported in Section 14.1.6 of the FSAR.

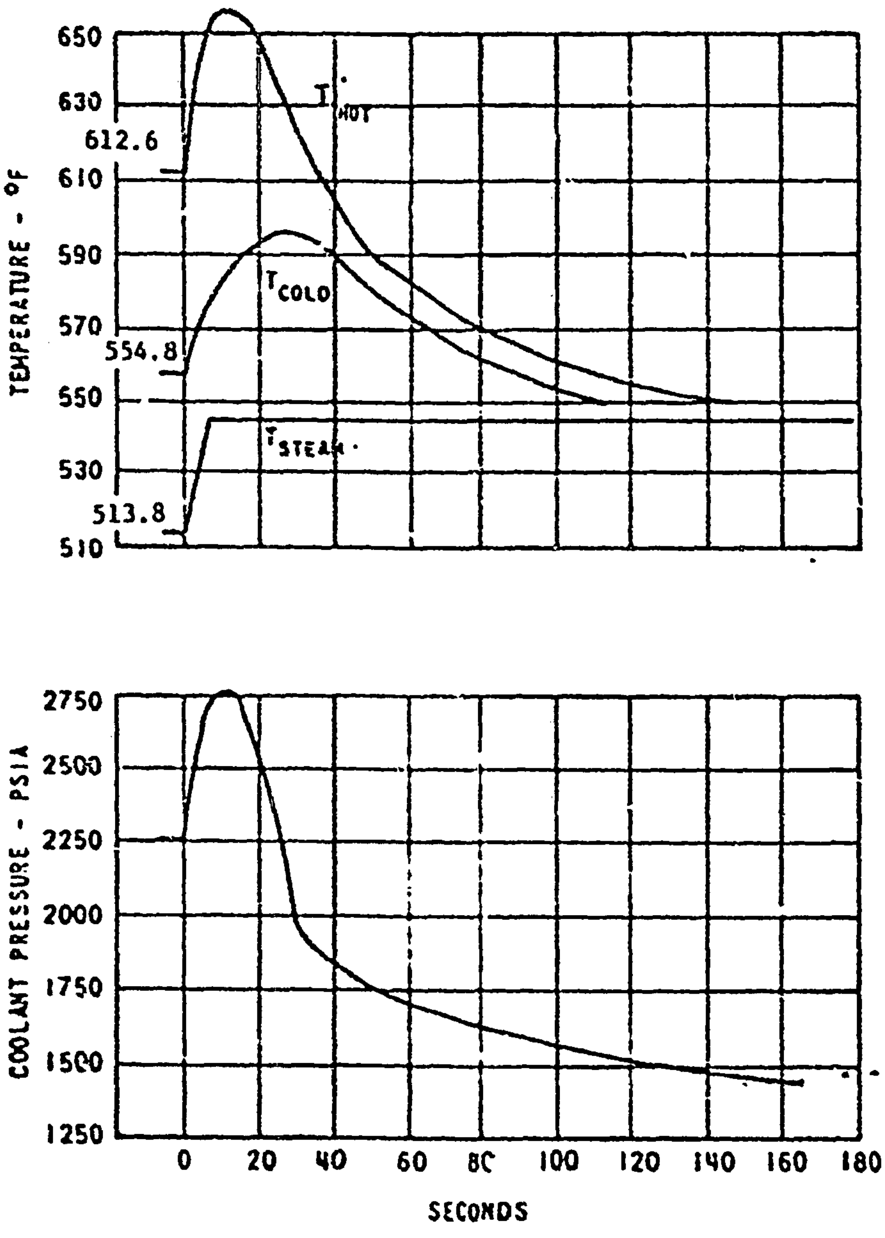
The number of occurrences of each of the above transients is generally specified at 2 per year of plant design life.

All components in the Reactor Coolant System are designed to withstand the effects of these and other transients that result in system temperature and pressure changes.

Stress intensity values at all critical points in the reactor vessel due to these excursions of pressure and temperature are determined for each of these transients, through systematic analytical procedures. These stress intensity values Sij (i, j = 1, 2, 3) are plotted against time interval for each cycle. This plot may represent one or more stress cycles. For each cycle extreme values of Smax and Smin are determined. From these values, the largest S_{alt} (alternating stress intensity) is found.

For this value of S_{alt}, an allowable number of cycles (N) is determined through design fatigue curves established for different materials. The ratio of design cycles (n) to allowable cycles (N) gives the usage factor (u). The usage factor is determined in this manner for all transients. The cumulative usage factor is determined by summing the individual usage factors. The cumulative usage factor ($U = u + u_2 + u_3$ ....) is never allowed to exceed a value of 1.0.

Subsection N415.2 of the 1965 Edition of the ASME Code was used for calculating the usage factors.



LOSS OF LOAD

FIGURE Q4.8.4-1
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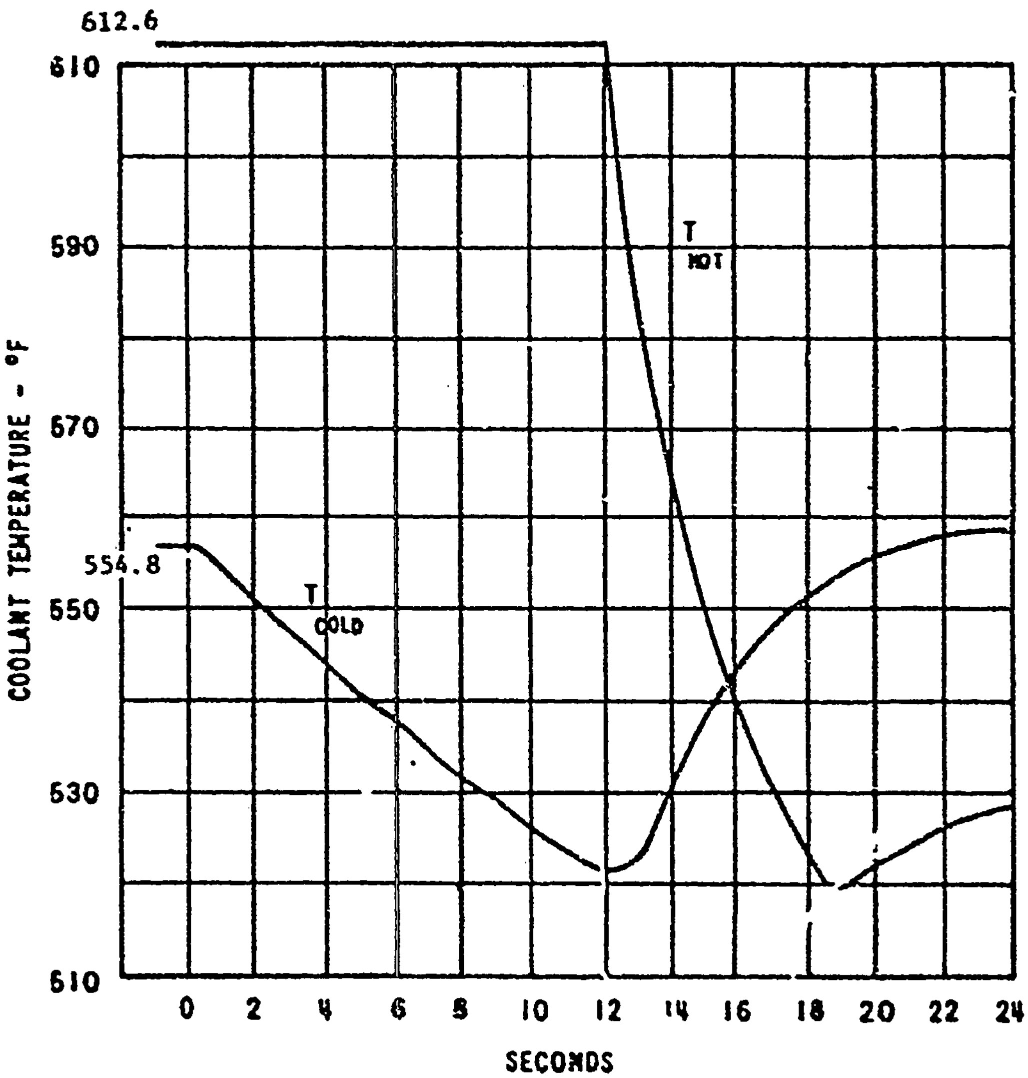

LOSS OF FLOW, ONE PUMP

FIGURE Q4.8.4-2
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QUESTION 4.8

With regard to the reactor vessel:

5. Discuss the magnitude of the stress in the reactor vessel membrane induced by gamma-ray heating.

ANSWER

For Indian Point Unit No. 2 reactor vessel the maximum thermal stress due to gamma-ray heating occurs in the cylindrical portion of the vessel adjacent to the core and its value is about 2500 psi. This additional thermal stress does not augment the stress intensity values considerably. The maximum stress intensity values under steady state and transient operating conditions are still far below the allowable limits of N-414 of ASME Boiler and Pressure Vessel Code Section III. The effect of gamma-ray heating on the cumulative usage factor is negligible.

Refer to Table 4.3-1 and 4.3-2 in the FSAR.

QUESTION 4.8

With regard to the reactor vessel:

6. Provide a list of pressure or strength bearing stainless steel component parts in the reactor vessel and associated reactor coolant systems that have become furnace sensitized during the fabrication sequence.

ANSWER

INDIAN POINT UNIT NO. 2 SYSTEM

1. Reactor Vessel

   (a) Eight primary nozzle safe ends (forgings), (these have been overlayed in the field with stainless steel weld metal).

2. Steam Generators

   (a) Two primary nozzle safe ends per generator - weld metal buttering.

3. Pressurizer

   (a) All nozzle safe ends (forgings) in top and bottom heads.

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QUESTION 4.8

With regard to the reactor vessel:

7. Provide a summary to results of Charpy V-Notch and drop weight tests for the reactor vessel plates and forgings.

ANSWER

See Table 4.8.7-1.

Table 4.8.7-1

COMPONENT	GRADE	30 FT-LB FIX (°F)	DROP WEIGHT NDT (°F)
Head Dome	A533B CL1	-2	10
Head Peel Segment	"	-10	10
Head Peel Segment	"	12	0
Upper Shell Plate	"	33	-10
Upper Shell Plate	"	31	-10
Upper Shell Plate	"	9	-10
Intermediate Shell Plate	"	14	-20
Intermediate Shell Plate	"	-11	-30
Intermediate Shell Plate	"	18	-10
Lower Shell Plate	"	-5	-20
Lower Shell Plate	"	-32	-20
Bottom Peel Segment	"	-12	-20
Bottom Peel Segment	"	-9	-10
Bottom Dome	"	8	-30
Head Flange	A508 CL2	10	-
Vessel Flange	"	-18	-
Inlet Nozzle	"	-102	-
Inlet Nozzle	"	-84	-
Inlet Nozzle	"	-95	-
Inlet Nozzle	"	-51	-
Outlet Nozzle	"	-32	-
Outlet Nozzle	"	<10	-
Outlet Nozzle	"	-45	-
Outlet Nozzle	"	<10	-

QUESTION 4.9

With regard to reactor internals:

1. Discuss the extent, methods and results of the analysis of the thermal stresses in the core barrel and support structure due to the occurance of loss-of-coolant and subsequent operation of emergency core cooling equipment.

ANSWER

Details of the analysis are presented in WCAP 7332 (Westinghouse Proprietary). "Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Excitation", as summarized below:

Maximum thermal stresses in the core barrel would occur if cold water were injected from the accumulators due to the occurance of a loss-of-coolant accident. The barrel is exposed to cold water in the downcomer annulus and somewhat hotter water in the compartments between barrel and baffle, producing a thermal gradient across barrel wall. The lower support structure is cooled more uniformly because of the large and numerous flow holes and consequently thermal stresses are lower.

The method used to obtain the maximum barrel stresses is as follows:

1) Temperature distribution across the barrel wall is computed as a function of time taken into consideration water temperatures and film coefficients.

2) Assuming that the obtained thermal gradients are axisymetrically distributed, which is conservative for stresses, maximum thermal stresses are computed in the barrel considered as an infinite cylinder.

3) Thermal stresses are added to primary stresses including seismic in order to obtain the maximum stress state of the barrel.

Results of analyses show that maximum thermal stresses in the barrel wall are well below the allowable criteria given for design by Section III of the ASME Code.

QUESTION 4.9

With regard to reactor internals:

2. Discuss the methods by which the seismic stresses were determined for the reactor internals. Give sufficient detail to show the development of the seismic loadings from the ground motion inputs to the final input used for the analysis of the internals structural members. Identify the methods of analysis employed and their interfaces, e.g., dynamic to static, elastic to plastic.

ANSWER

Details of the analysis are presented in WCAP 7332 (Westinghouse Proprietary). "Indian Point Unit No. 2 Reactor Internals Mechanical Analysis for Blowdown Exitation", as summarized below:

Seismic Analysis of Reactor Internal

The maximum stresses are obtained by combining the contributions from the horizontal and vertical earthquakes in the most conservative manner. The following paragraphs describe the horizontal and vertical contributions.

Horizontal Earthquake Model and Procedure

The reactor building with the reactor vessel support, the reactor vessel, and the reactor internals are included in this analysis. The mathematical model of the building, attached to ground, is identical to that used to evaluate the building structure. The reactor internals are mathematically modeled by beams, concentrated masses, and linear springs.

All masses, water, and metal are included in the mathematical model. All beam elements have the component weight or mass distributed uniformly, e.g., the fuel assembly mass and barrel mass. Additionally, wherever components are attached somewhat uniformly their mass is included as an additional uniform mass, e.g., baffles and formers acting on the core barrel. The water near and about the beam elements is included as a distributed mass.

Horizontal components are considered as concentrated masses acting on the barrel. These concentrated masses also include components attached to the horizontal members since this is the media through which the reaction is transmitted. The water near and about these separated components is considered as being additive at these concentrated mass points.

The concentrated masses attached to the barrel represent the following: a) the upper core support structure, including the upper vessel head and one-half the upper internals; b) the upper core plate, including one-half the thermal shield and the other half of the upper internals; c) the lower core plate, including one-half of the lower core support columns; d) the lower one-half of the thermal shield; and e) the lower core support, including the lower instrumentation and the remaining half of the lower core support columns.

The modulus of elasticity is chosen at its hot value for the three major materials found in the vessel, internals, and fuel assemblies. In considering shear deformation, the appropriate cross-sectional areas are selected along with a value for Poisson's ratio. The fuel assembly moment of inertia is derived from experimental results by static and dynamic tests performed on fuel assembly models. These tests provide stiffness values for use in this analysis.

The fuel assemblies are assumed to act together and are represented by a single beam. The following assumptions are made in regards to connection restraints. The vessel is pinned to the vessel support which is the surrounding concrete structure and part of the containment building. The barrel is clamped to the vessel at the barrel flange and spring connected to the vessel at the lower core barrel radial support. This spring corresponds to the radial support stiffness for two opposite supports acting together. The beam representing the fuel assemblies is pinned to the barrel at the locations of the upper and lower core plates.

Modal analysis, plus the response spectrum method[1] is used in this analysis. The modal analysis is studied by the use of a transfer matrix method.

---

[1] Shock and Vibration Handbook, edited by Harris and Crede, Volume 3, Chapter 50: "Vibration of Structures Induced by Seismic Waves" by George W. Housner.

The maximum deflection, acceleration, etc., is determined at each particular point by summing the absolute values obtained for all modes. With the shear forces and bending moments determined, the earthquake stresses are then calculated.

Figure Q4.9.2-1 shows the mathematical model studied.

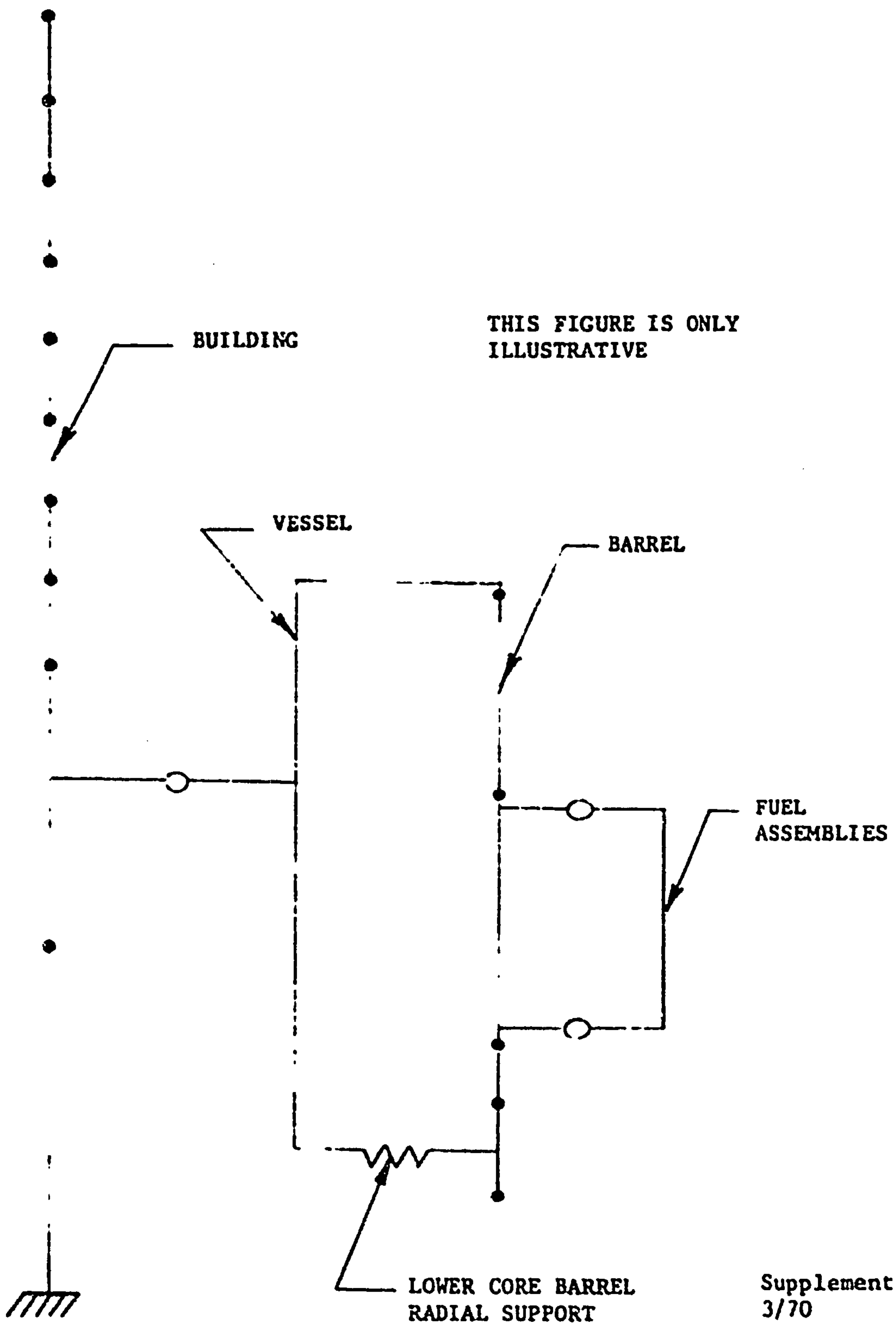
Analytical Model for Vertical Earthquake Model and Procedure

The reactor internals is modeled as single degree of freedom system for vertical earthquake analysis. The maximum acceleration at the vessel support is increased by the amplification due to the building soil interaction.

There are no interfaces in the analyses (e.g., dynamic to static, elastic to plastic).

MATHEMATICAL MODEL FOR REACTOR VESSEL
INTERNALS ANALYSIS-HORIZONTAL EXCITATION

THIS FIGURE IS ONLY
ILLUSTRATIVE



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NOT TO SCALE

FIGURE Q4.9.2-1

QUESTION 4.10

To which edition of the ASME Code, Section III and addenda, are applicable Class I components designed and fabricated.

ANSWER

The edition of the ASME Code, Section III and addenda to which the jor components in the Reactor Coolant System will be designed and fabricated are:

Equipment	Code Edition	Applicable Addenda
Reactor Vessel	1965	Summer 1965 and Code ases 1332, 1335, 1339, 1359
Pressurizer	1965	Summer 1966
Steam Generators	1965	Summer 1966
Control Rod Drive Mechanism	1965	Summer 1966
Control Rod Drive Mechanism (Part Length)	1965	Summer 1967
Reactor Coolant Pumps	*	

In addition the reactor coolant pipe was designed to B31.1-1955.

*The reactor coolant pump, though not a coded vessel, was designed to Section III of the ASME Boiler and Pressure Vessel Code.

QUESTION 4.11

Discuss the extent to which electroslag welding was used in the fabrication of Class I components. If electroslag welding was used, describe the process, its variables, and the quality control procedures employed.

ANSWER

The Indian Point Unit No. 2 90° elbows were electroslag welded. The following efforts were performed for quality assurance of these components.

1. The electroslag welding procedure employing one wire technique was qualified in accordance with the requirements of ASME B&PV Code Section IX and Code Case 1355 plus supplementary evaluations as requested by WNES-PWRSD. The following test specimens were removed from a 5 inch thick weldment and successfully tested. They are:

    a. 6 Transverse Tensile Bars - as welded

    b. 6 Transverse Tensile Bars - 2050°F, H_2O Quench

    c. 6 Transverse Tensile Bars - 2050°F, H_2O Quench + 750° stress relief heat treatment

    d. 6 Transverse Tensile Bars - 2050°F, H_2O Quench, tested at 650°F

    e. 12 Guided Side Bend Test Bars

2. The casting segments were surface conditioned for 100% radiographic and penetrant inspections. The acceptance standards were ASTM E-186 severity level 2 (except no category D or E defectiveness was permitted) and ASME Section III, Paragraph N-627, respectively.

3. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME Section III, Paragraph N-627.

4. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME Section III, Paragraph N-627.

5. Weld metal and base metal chemical and physical analysis were determined and certified.

6. Heat treatment furnace charts were recorded and certified.

Two of the Indian Point Unit No. 2 reactor coolant pump casings were electroslag welded. The following efforts were performed for quality assurance of these two components.

1. The electroslag welding procedure employing two and three wire technique was qualified in accordance with the requirements of the ASME B&PV Code Section IX and Code Case 1355 plus supplementary evaluations as requested by WNES-PWRSD. The following test specimens were removed from an 8 inch thick and from a 12 inch thick weldment and successfully tested for both the 2 wire and the 3 wire techniques, respectfully. They are:

A. Two wire electroslag process - 8" thick weldment.

   1. 6 Transverse Tensile Bars - 750°F post weld stress relief

   2. 12 Guided Side Bend Test Bars

B. Three wire electroslag process - 12" thick weldment

   1. 6 Transverse Tensile Bars - 750°F post weld stress relief

   2. 17 Guided Side Bend Test Bars

   3. 21 Charpy Vee Notch Specimens

4. Full section macroexamination of weld and heat affected zone

5. Numerous microscopic examinations of specimens removed from the weld and heat affected zone regions.

6. Hardness survey across weld and heat affected zone.

C. A separate weld test was made using the 2 wire electroslag technique to evaluate the effects of a stop and restart of welding by this process. This evaluation was performed to establish proper procedures and techniques as such an occurrence was anticipated during production applications due to equipment malfunction, power outages, etc. The following test specimens were removed from an 8 inch thick weldment in the stop-restart-repaired region and successfully tested. They are:

1. 2 Transverse Tensile Bars - as welded

2. 4 Guided Side Bend Test Bars

3. Full section macroexamination of weld and heat affected zone.

D. All of the weld test blocks in (A), (B) and (C) above were radiographed using a 24 Mev Betatron. The radiographic quality level obtained was between one-half of 1% to 1%. There were no discontinuities evident in any of the electroslag welds.

1. The casting segments were surface conditioned for 100% radiograpic and penetrant inspections. The radiographic acceptance standards were ASTM E-186 severity level 2 except no category D or E defectiveness was permitted for section thickenss up to 4-1/2 inches and ASTM E-280 severity level 2 for section thicknesses greater than 4-1/2 inches. The penetrant acceptance standards were ASME B&PV Code Section III, paragraph N-627.

2. The edges of the electroslag weld preparations were machined. These surfaces were penetrant inspected prior to welding. The acceptance standards were ASME B&PV Code Section III, paragraph N-627.

3. The completed electroslag weld surfaces were ground flush with the casting surface. Then, the electroslag weld and adjacent base material were 100% radiographed in accordance with ASME Code Case 1355. Also, the electroslag weld surfaces and adjacent base material were penetrant inspected in accordance with ASME B&PV Code Section III, paragraph N-627.

4. Weld metal and base metal chemical and physical analyses were determined and certified.

5. Heat treatment furnace charts were recorded and certified.

The two remaining Indian Point Unit No. 2 reactor coolant pump casings were submerged arc welded. Quality Assurance procedures and Quality Assurance inspections equivalent to the above were also exercised on these casings.

QUESTION 4.12

Document the design and operating requirements for the equipment that will be used to detect failed fuel during plant operation.

ANSWER

Refer to page 11.2-21 and 22 of Section 11 of the FSAR as revised by Supplement 13.

QUESTION 5.1

For the containment structure, provide:

a) Drawings of the containment presenting details of the base slab and the dome-cylinder junction, showing reinforcing and liner features, including liner anchors. Fig. 5.1-1.

ANSWER

See Figs. 4.6 of the Containment Design Report for Indian Point Unit No. 2.

See also Figs. 5.1a-1, 5.1a-2, 5.1a-3, and 5.1a-4.

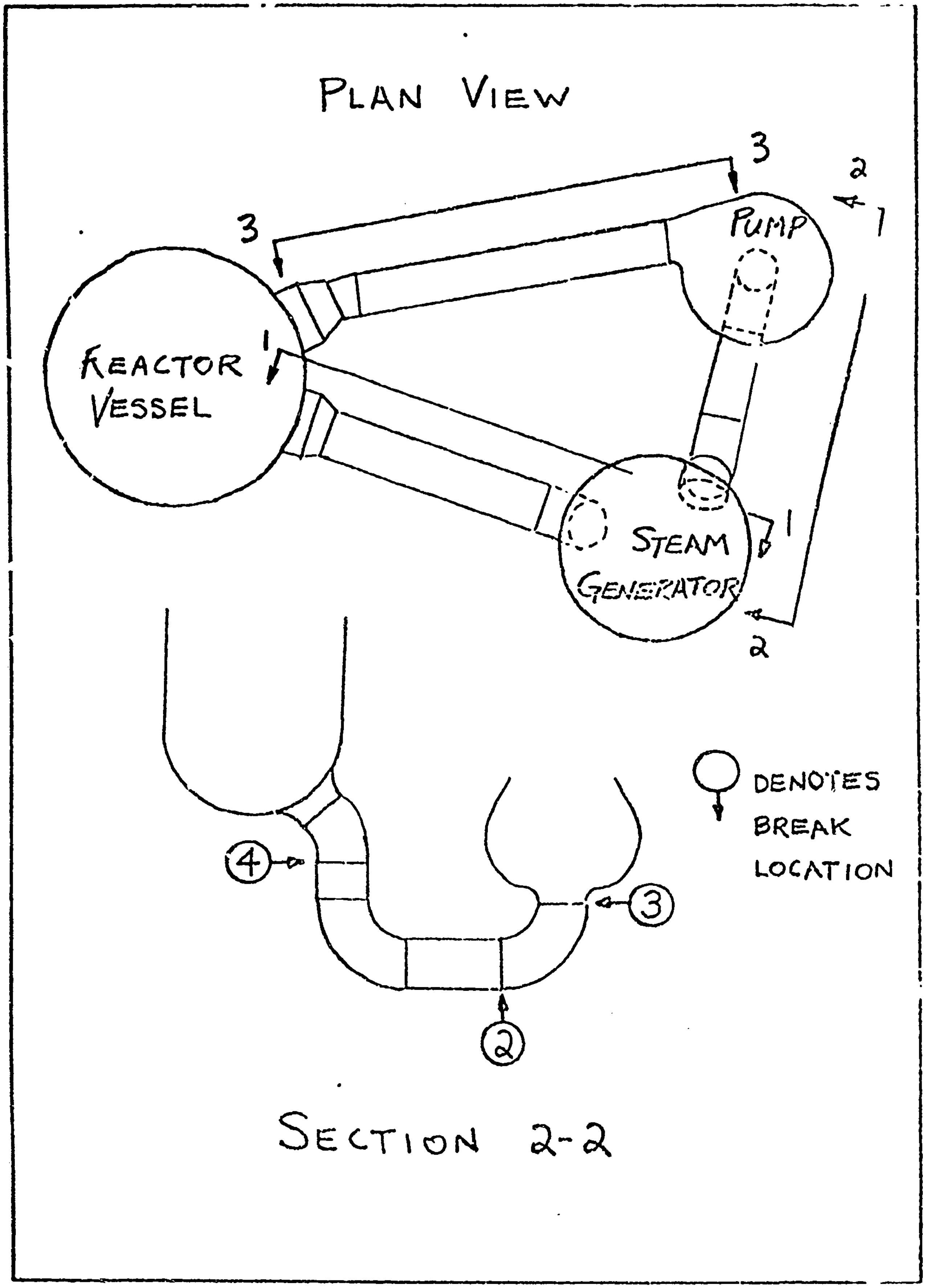
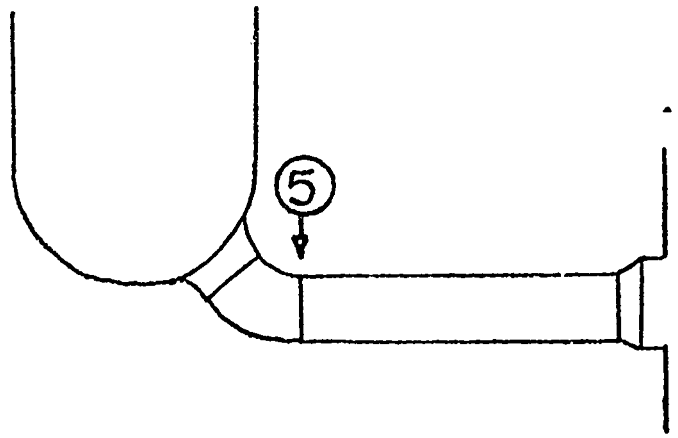
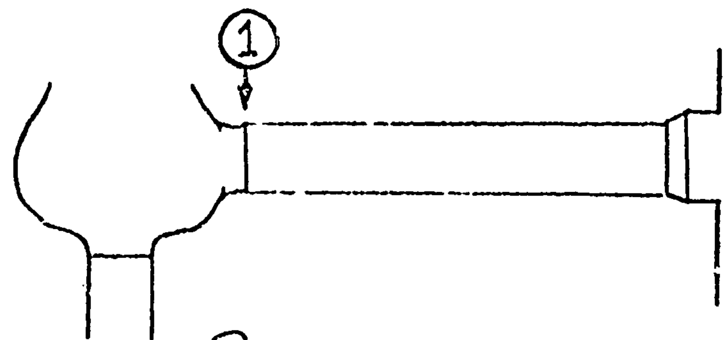


FIGURE 5.1-1 ASSUME PIPE RUPTURE ACCIDENT BREAK LOCATIONS



SECTION 1-1



SECTION 3-3

(1) HORIZONTAL & VERTICAL SLOT BREAK, AXIAL BREAK

(2) HORIZONTAL & VERTICAL SLOT BREAK

(3) AXIAL PIPE BREAK

(4) AXIAL PIPE BREAK

(5) HORIZONTAL & VERTICAL SLOT BREAK, AXIAL BREAK

FIGURE 5.1-1 CONTINUED

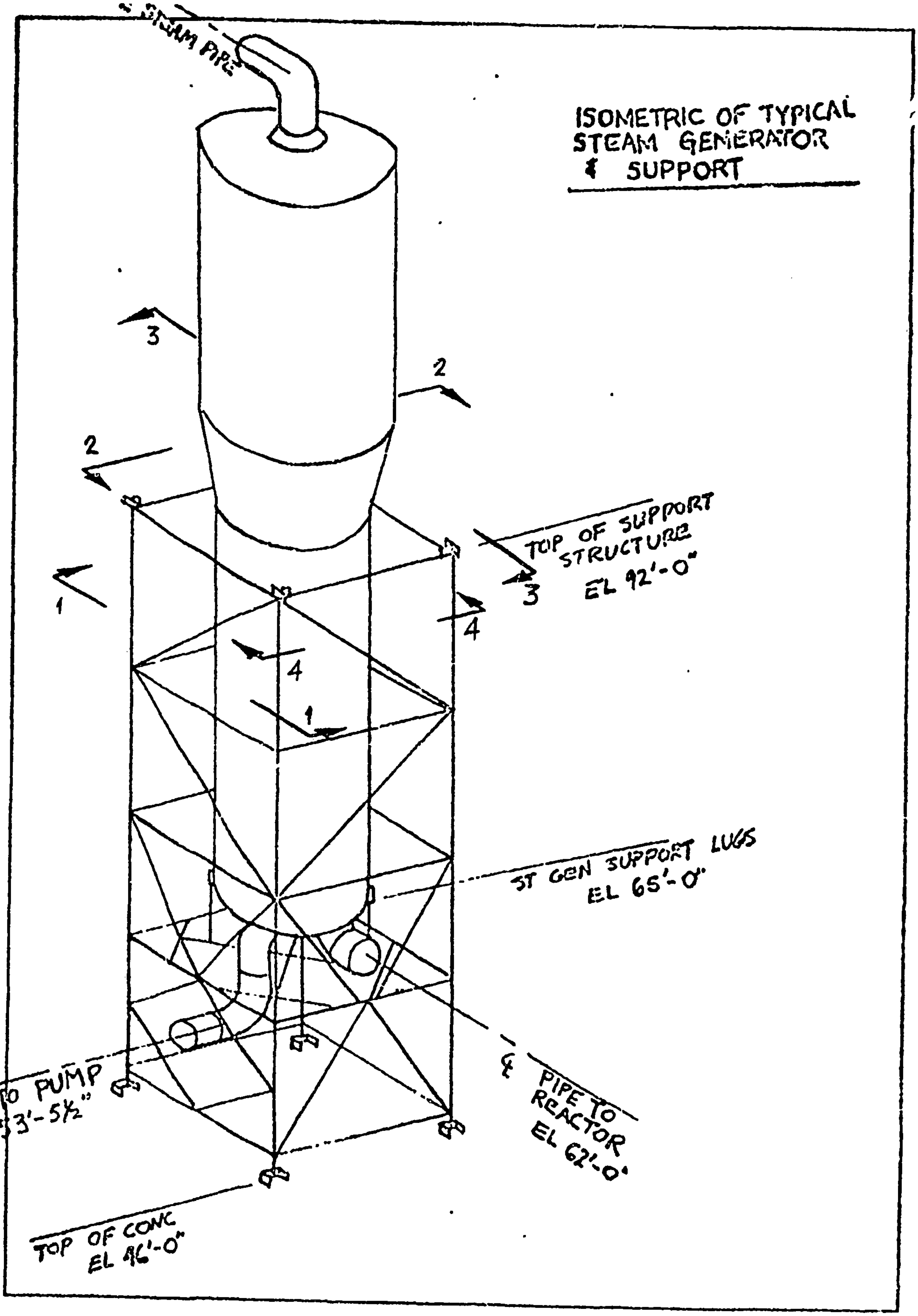


FIGURE 5.1-2 ISOMETRIC VIEW OF STEAM GENERATOR SUPPORT

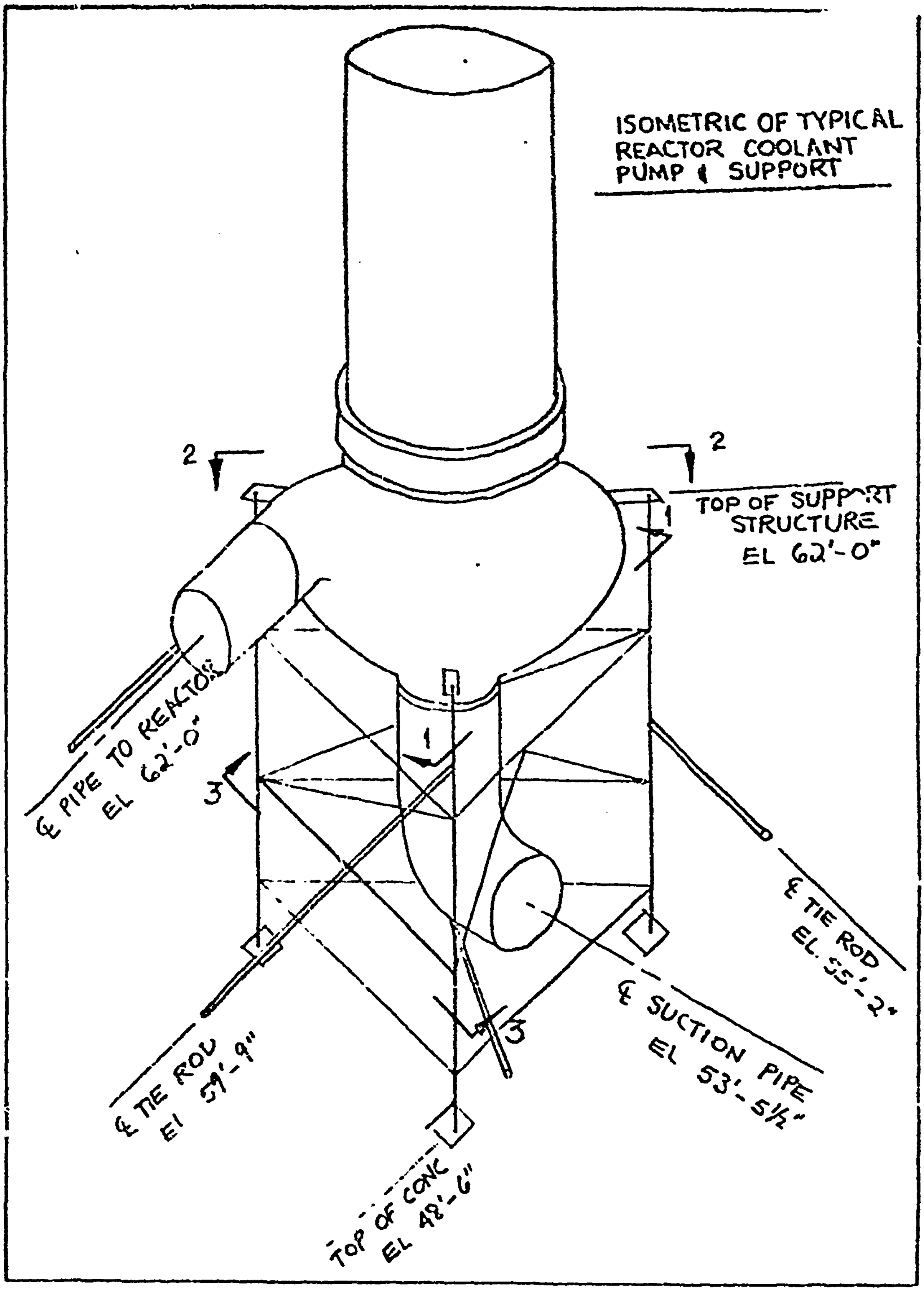


FIGURE 5.1-3 ISOMETRIC VIEW OF REACTOR COOLANT PUMP SUPPORT

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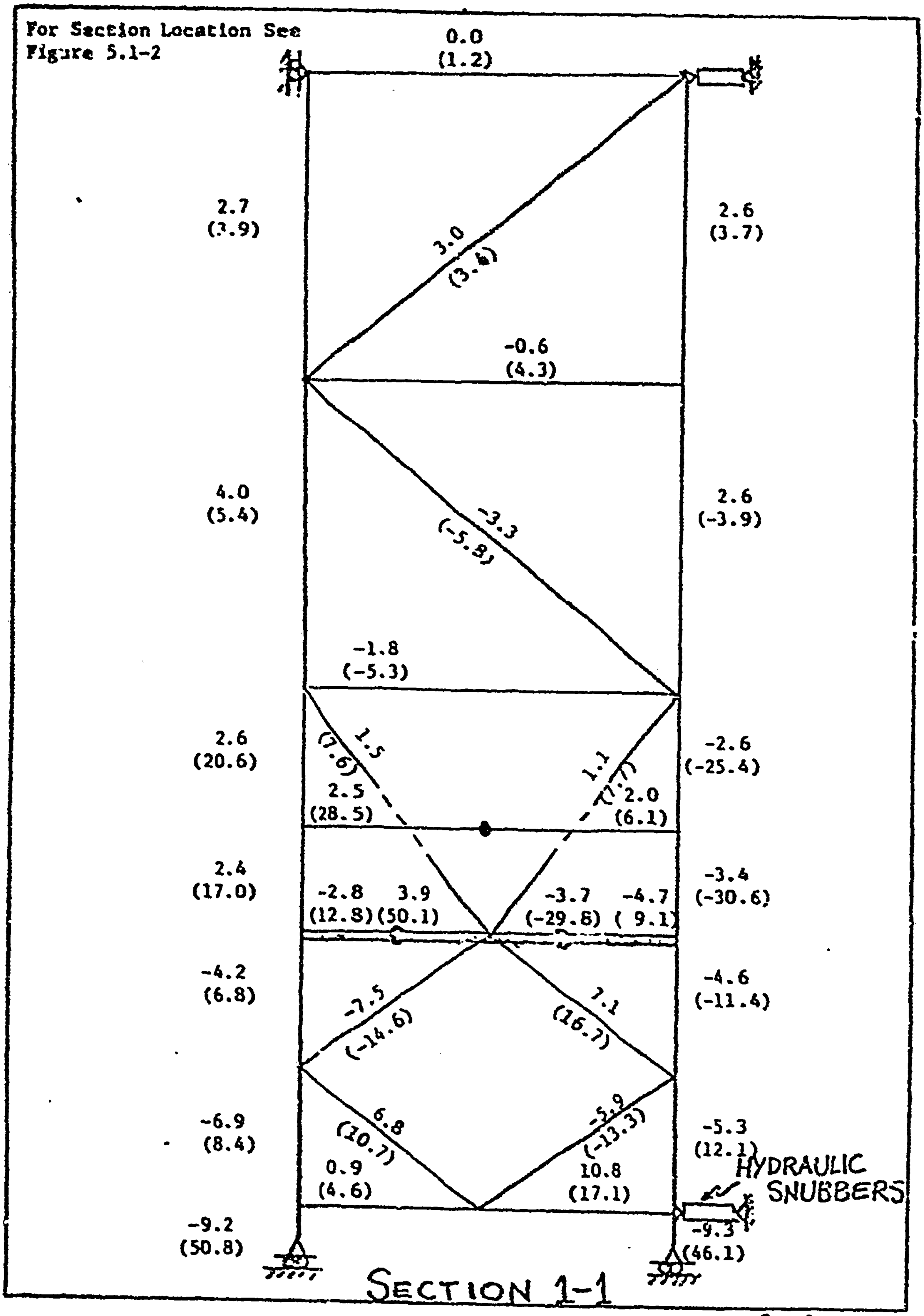


FIGURE 5.1-4 SECTION 1-1 STEAM GENERATOR SUPPORT

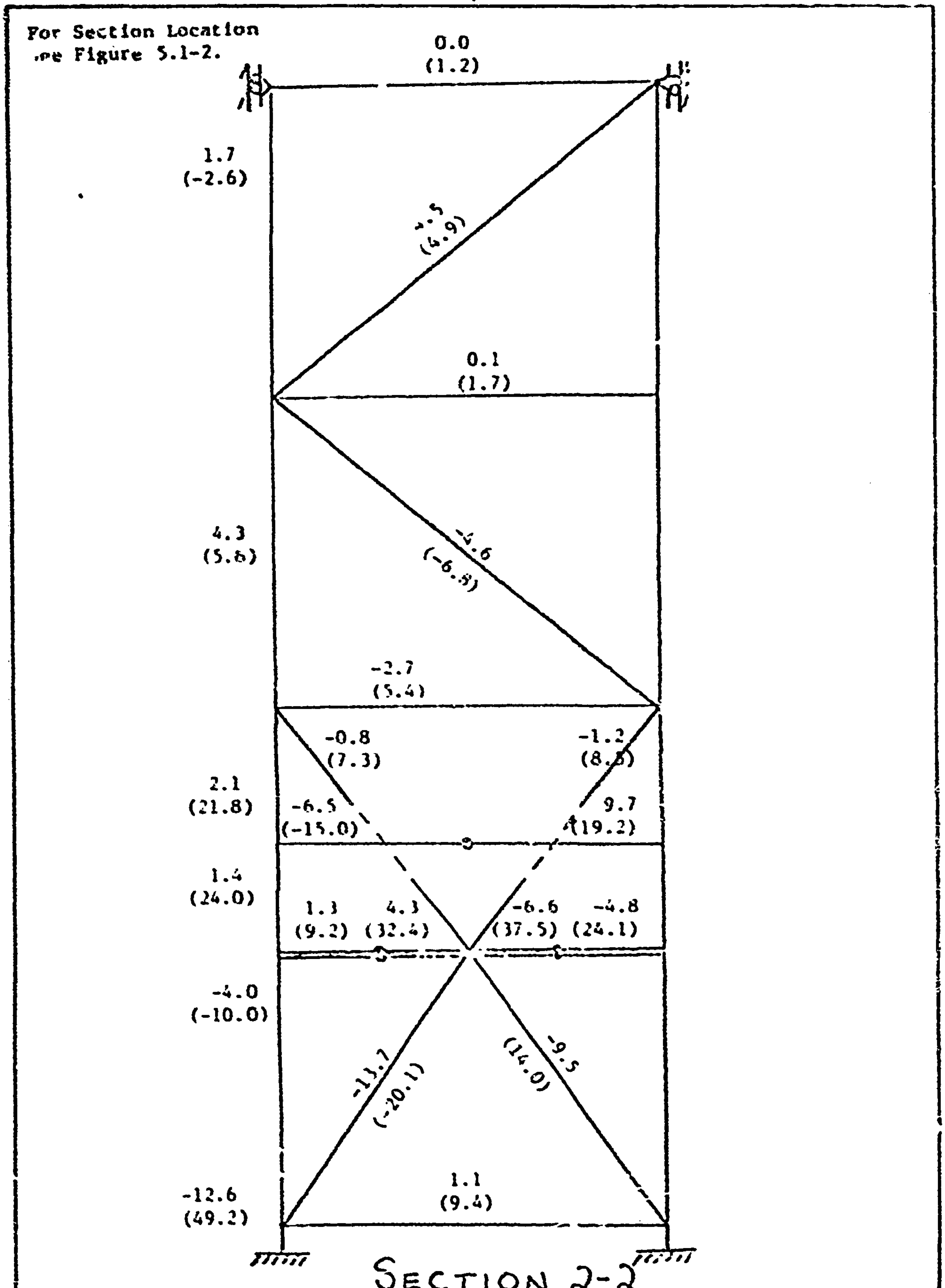


FIGURE 5.1-5 SECTION 2-2 STEAM GENERATOR SUPPORT

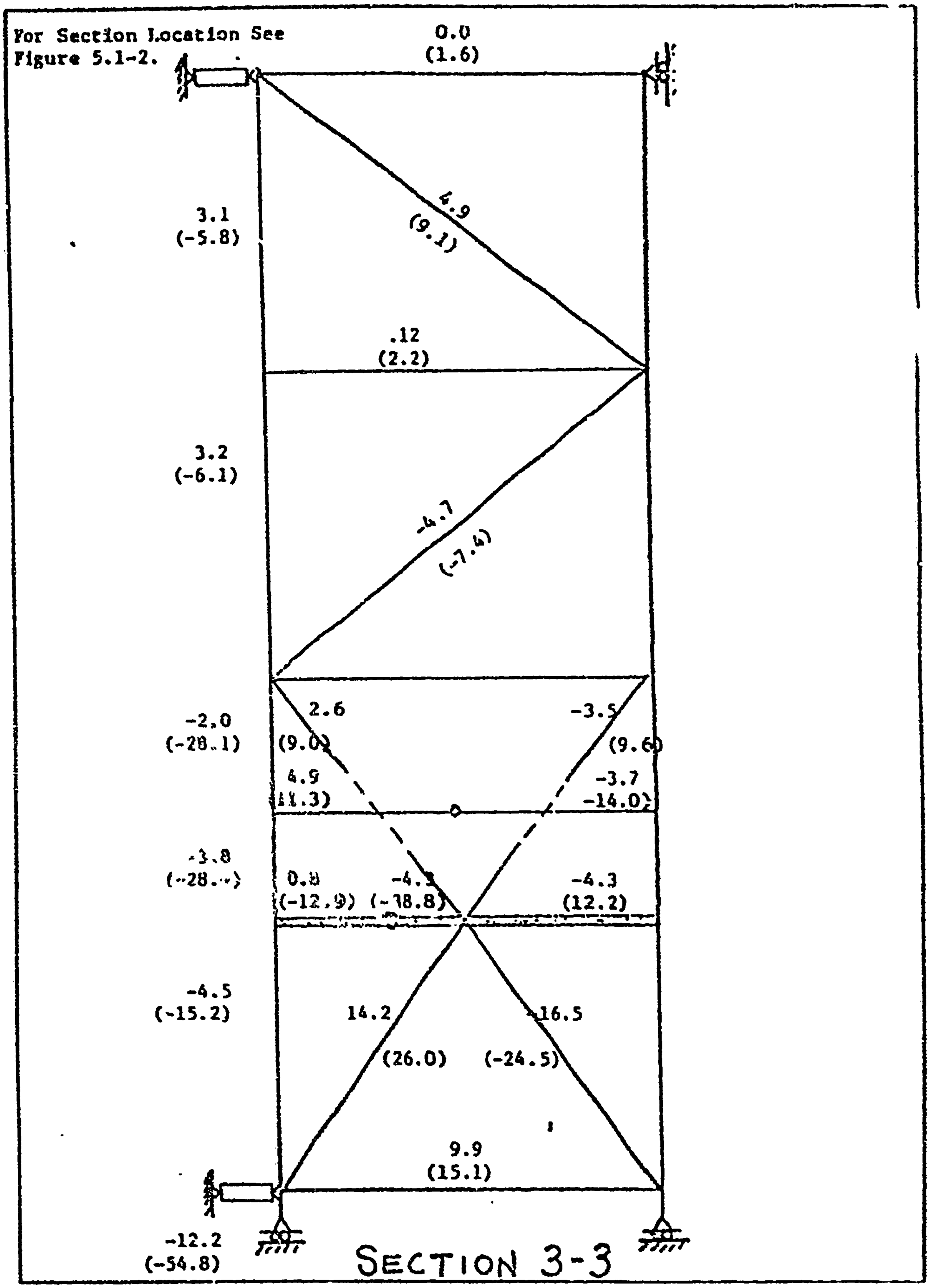


FIGURE 5.1-6 SECTION 3-3 STEAM GENERATOR SUPPORT

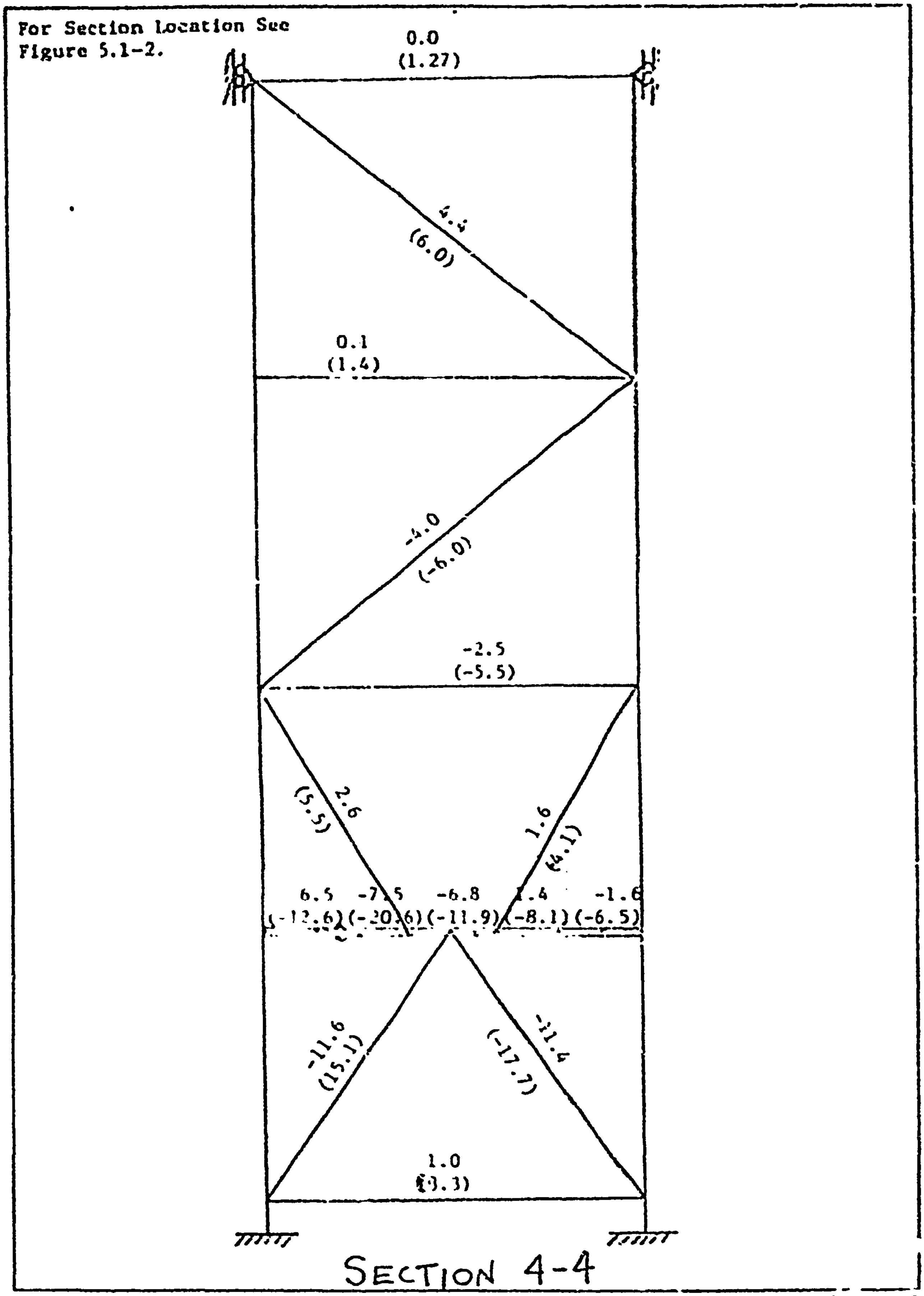
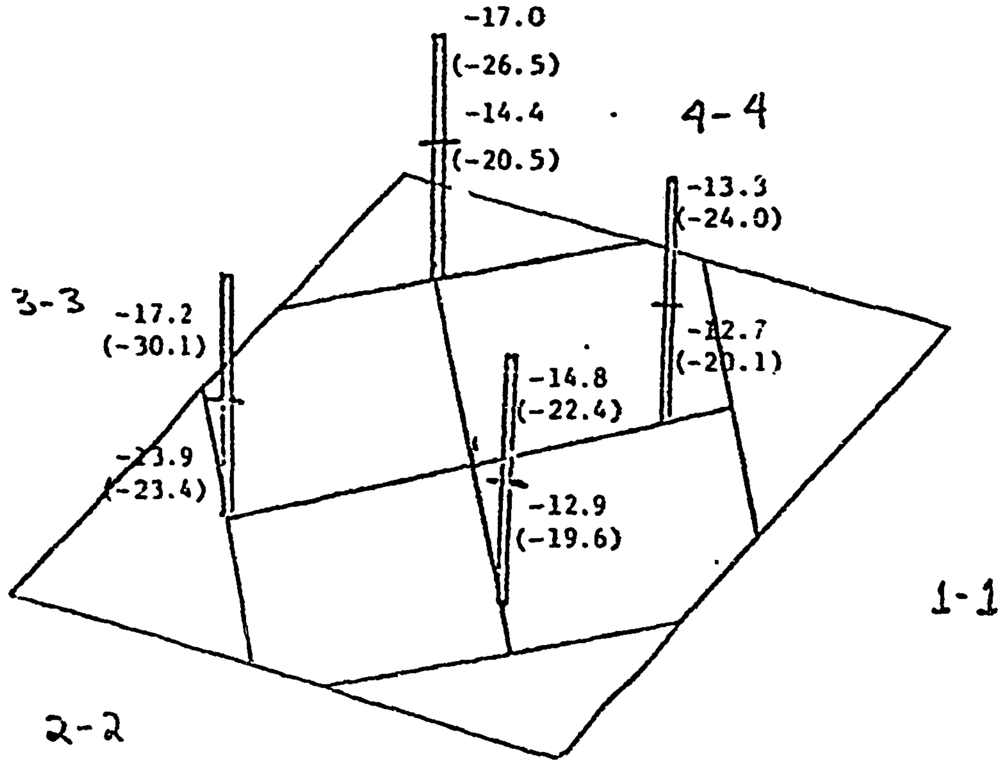
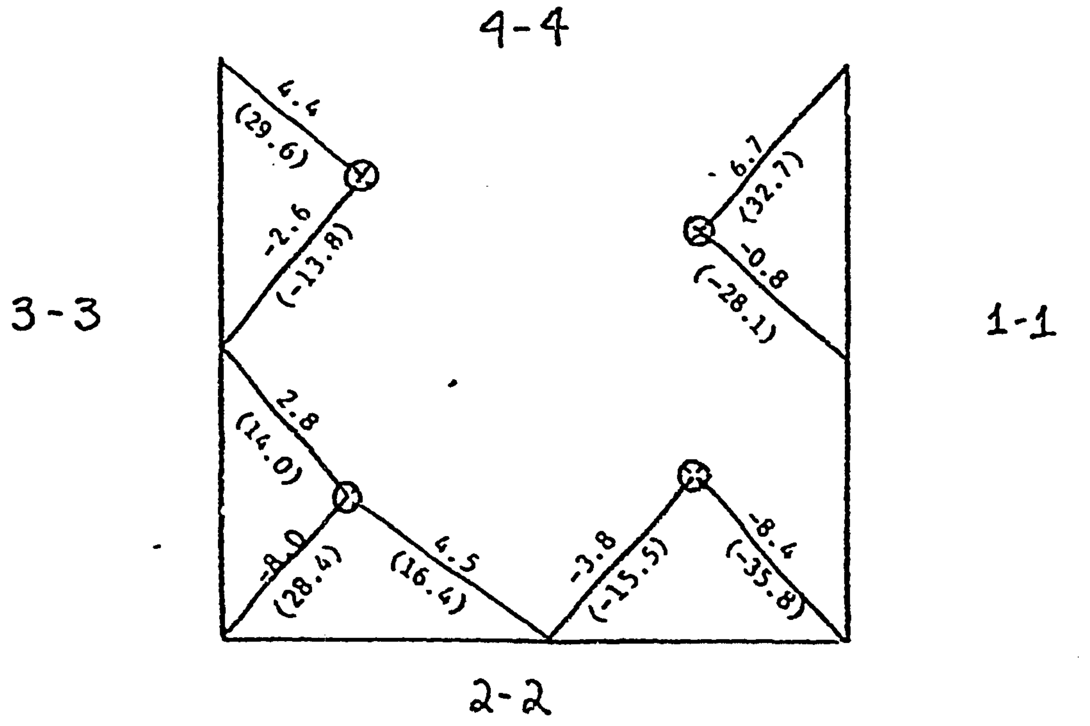


FIGURE 5.1-7 SECTION 4-4 STEAM GENERATOR SUPPORT

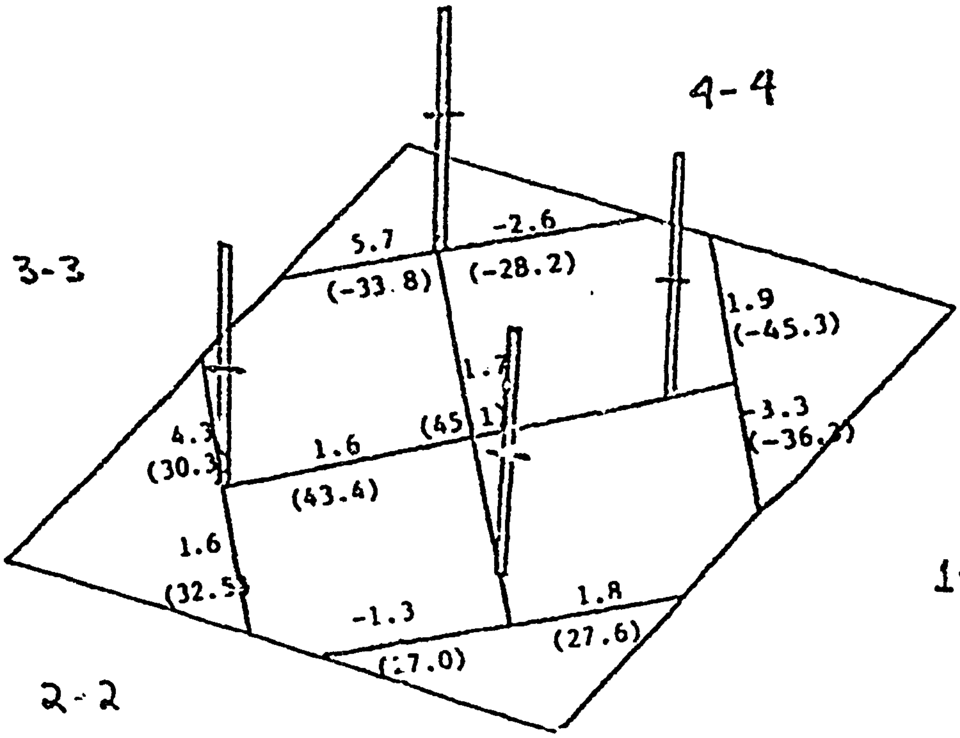


PLAN AT EL. 60'-0"

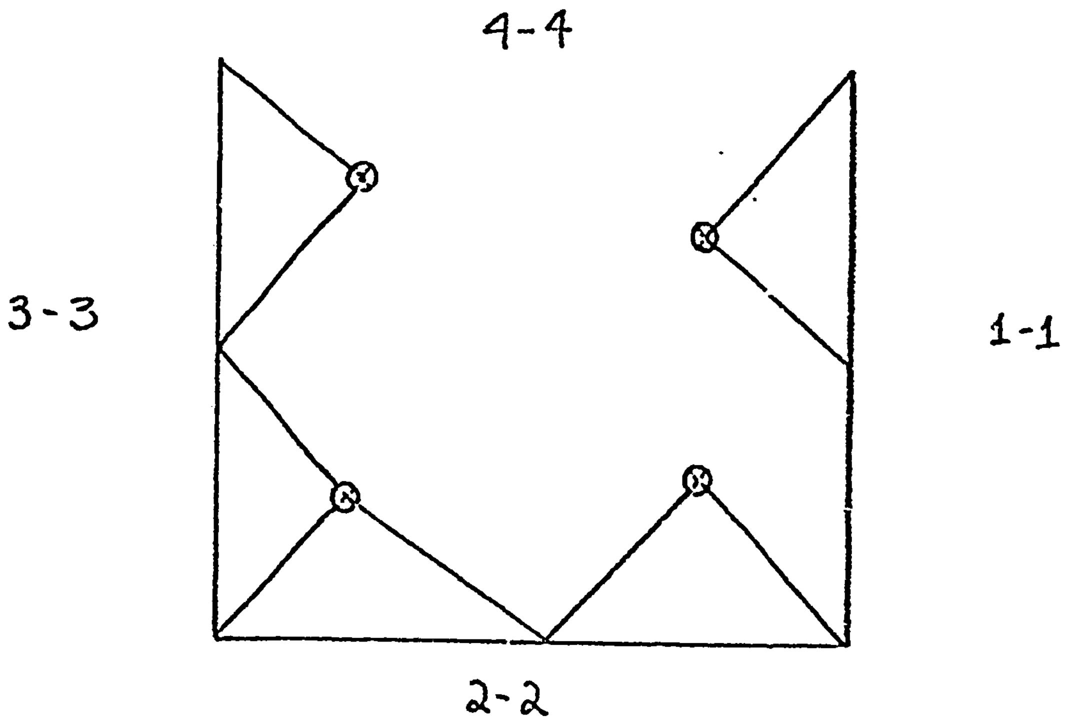


PLAN AT EL. 63'-0"

FIGURE 5.1-8 PLAN LOCATION El. 60 and 63 of STEAM GENERATOR SUPPORT

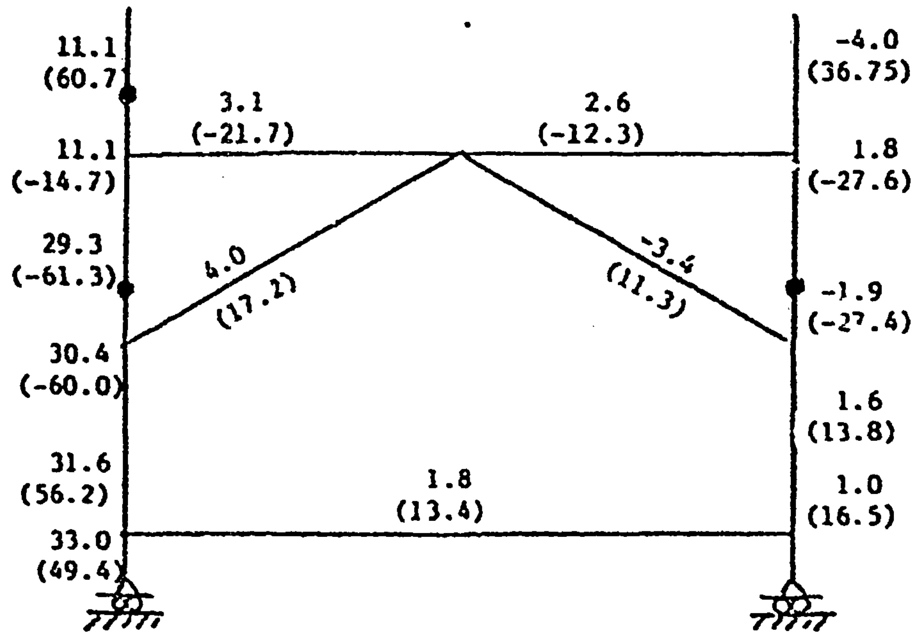


PLAN AT EL. 60'-0"



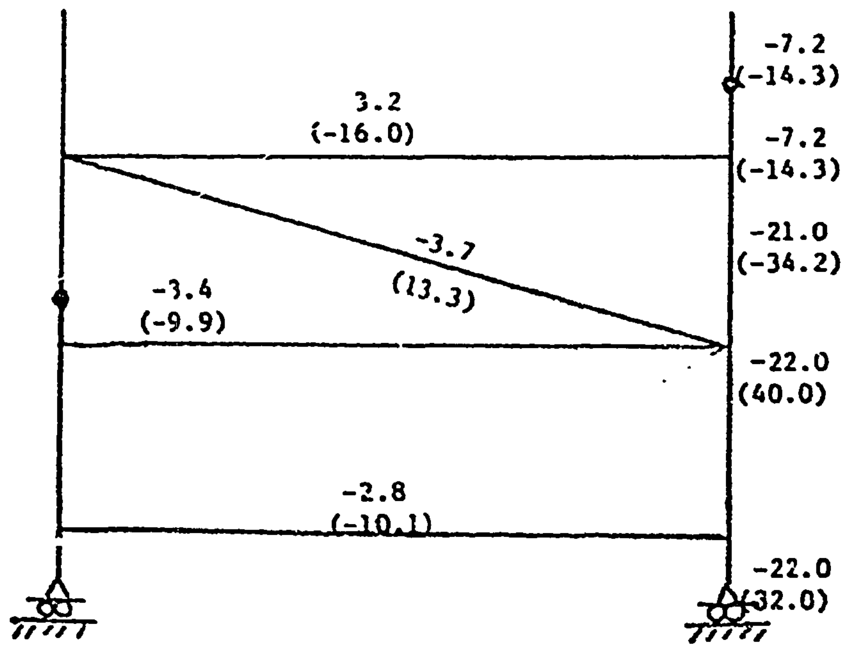
PLAN AT EL. 63'-0"

FIGURE 5.1-9 PLAN LOCATION El. 60 and 63 OF STEAM GENERATOR SUPPORT



SECTION 1-1

For Section Location See
Figure 5.1-3.



SECTION 2-2

FIGURE 5.1-10 SECTION 2-2 AND 3-3 PUMP SUPPORT

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For Section Location See Fig. 5.1-3

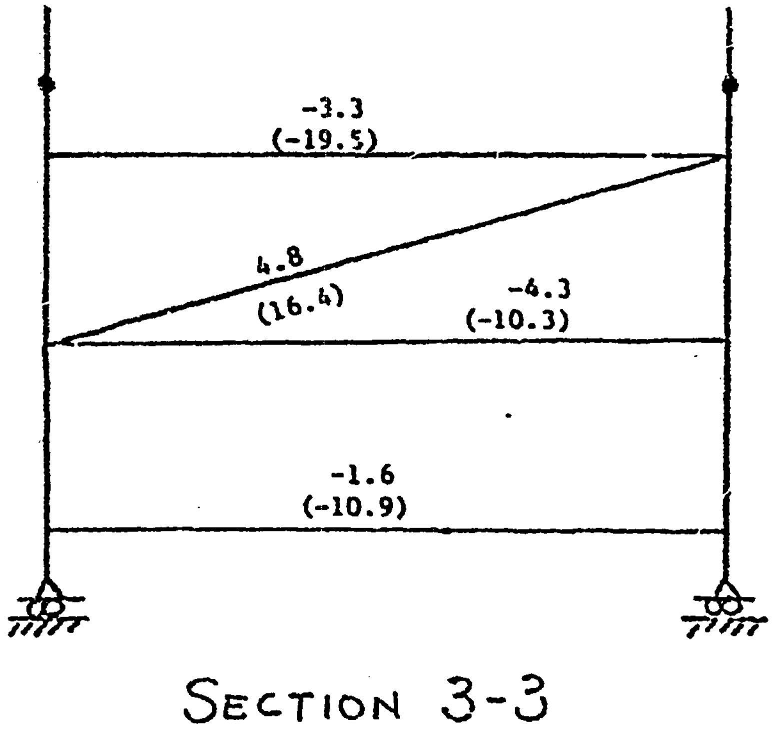

FIGURE 5.1-11 SECTION 3-3 PUMP SUPPORT

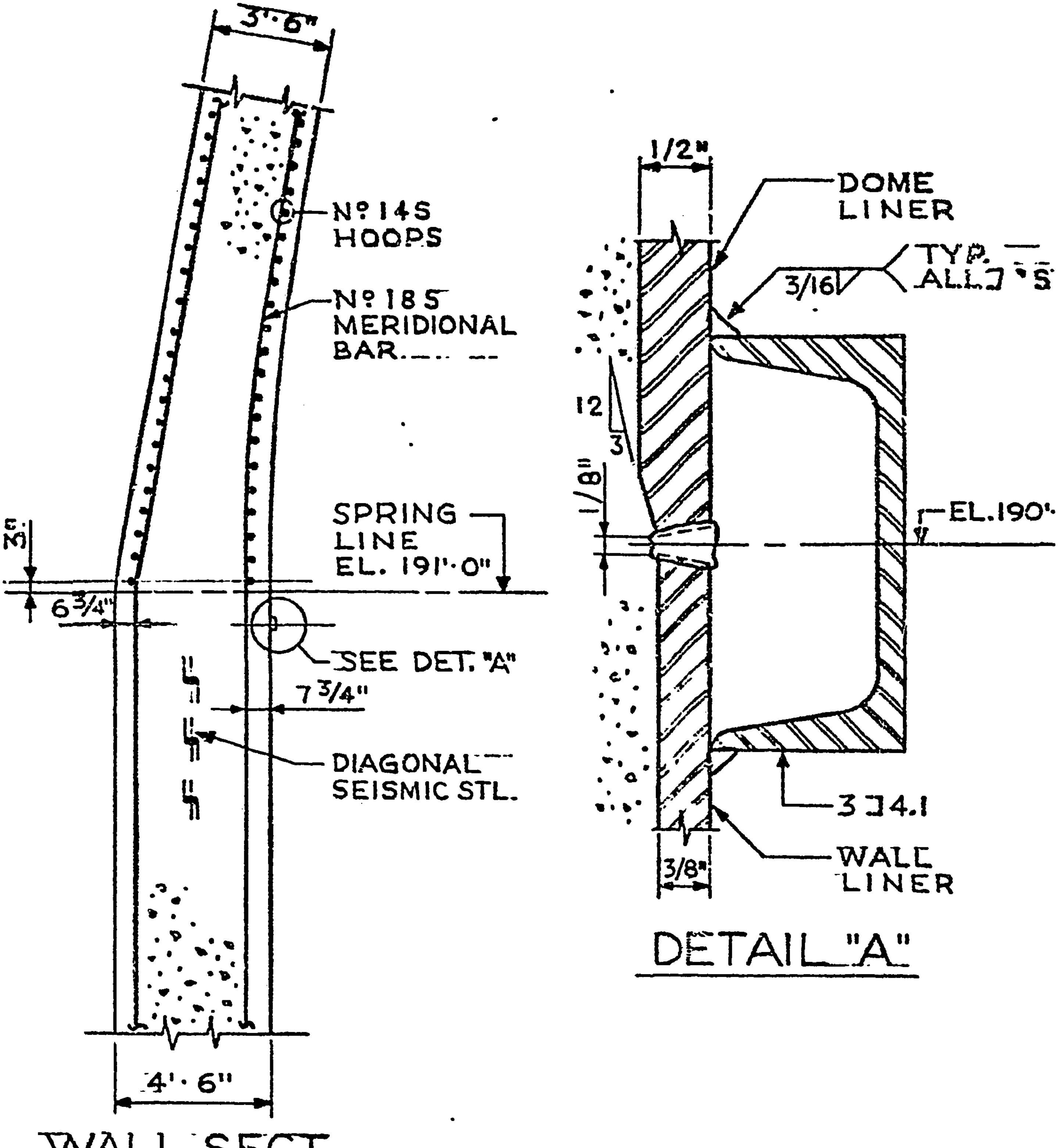

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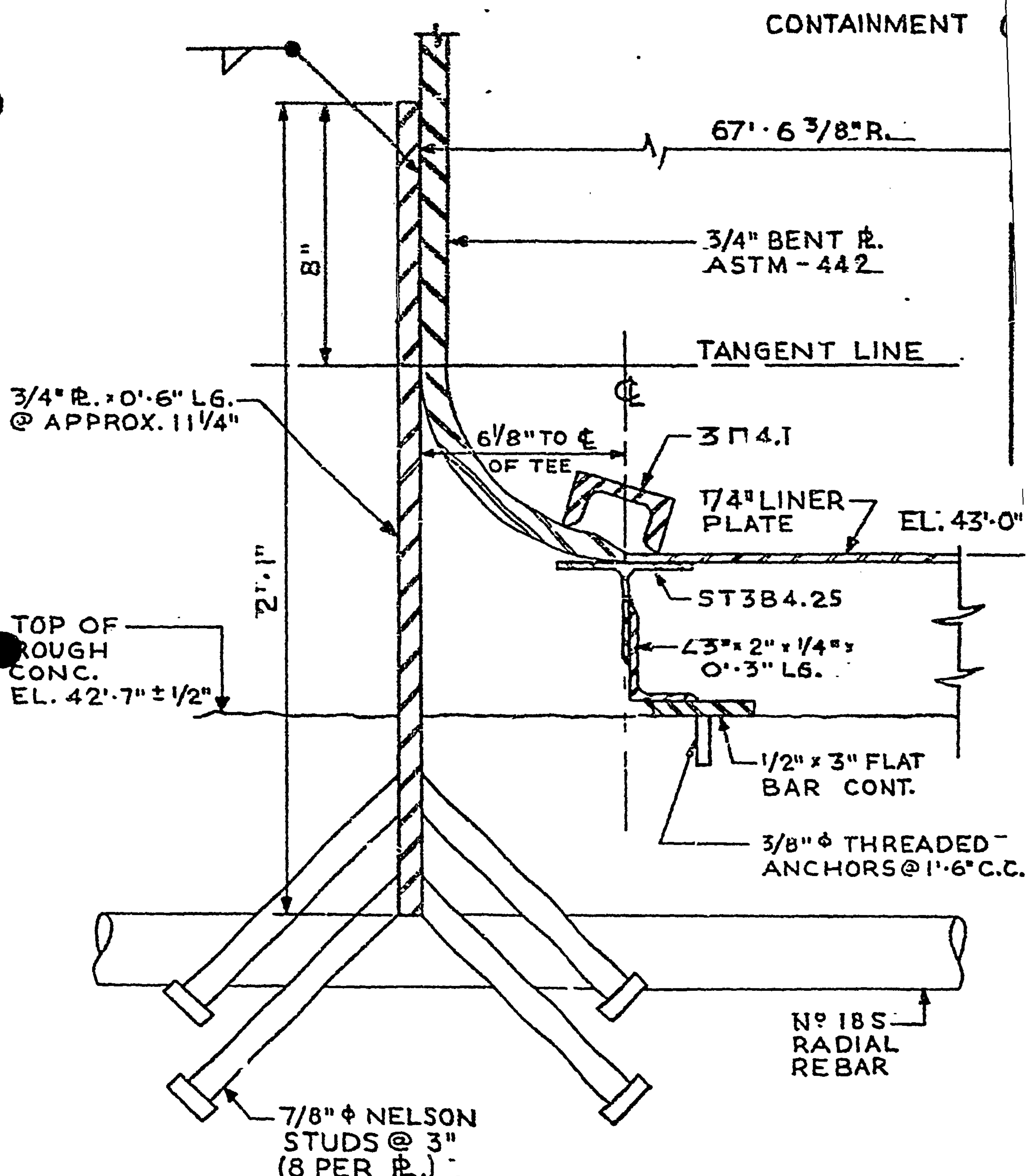
FIG. 5.1(a)-1



CYLINDER · BASE SLAB
LINER JUNCTURE

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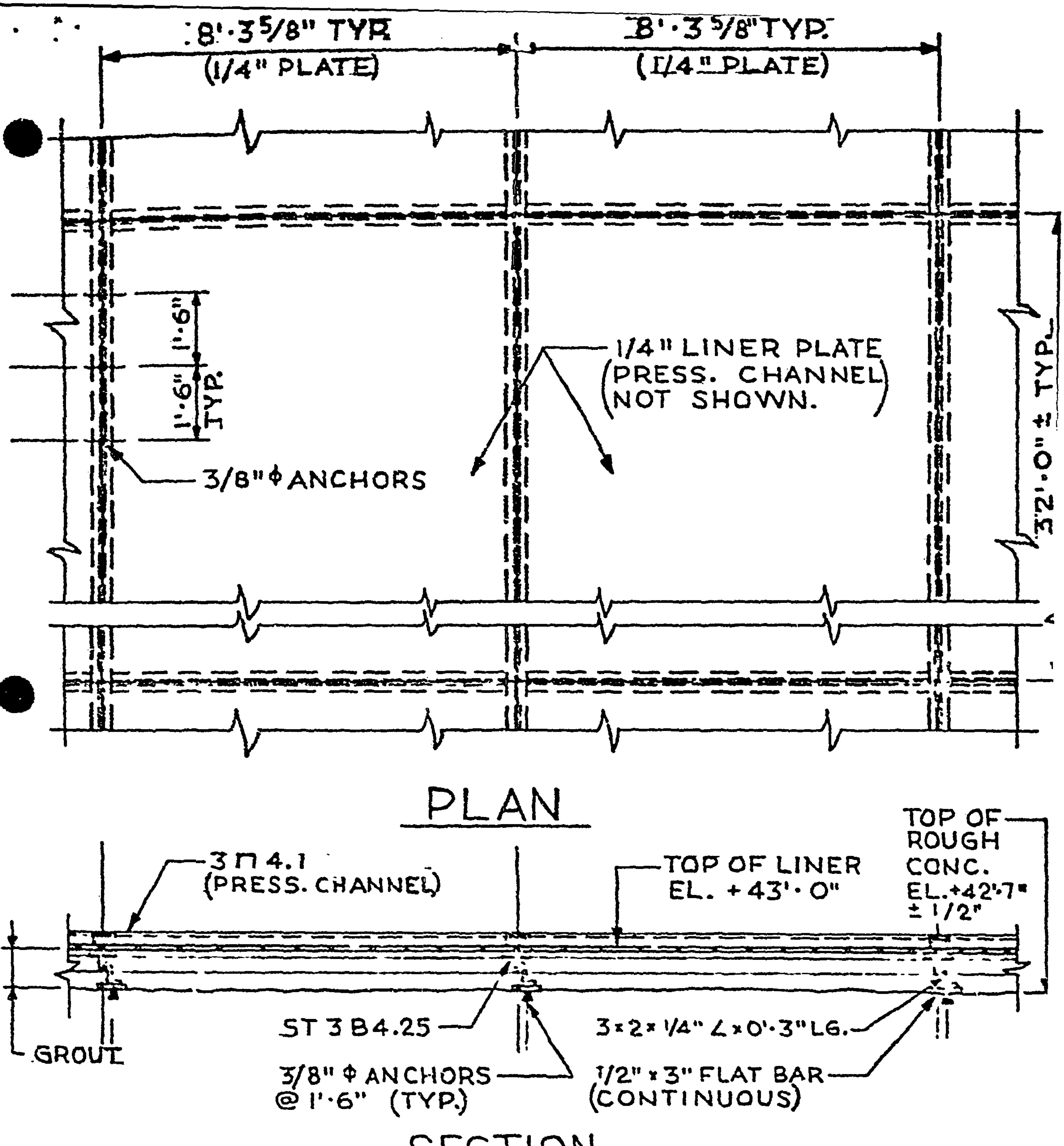
FIG. 5.1 (a)-2



TYPICAL BASE MAT
LINER DETAIL

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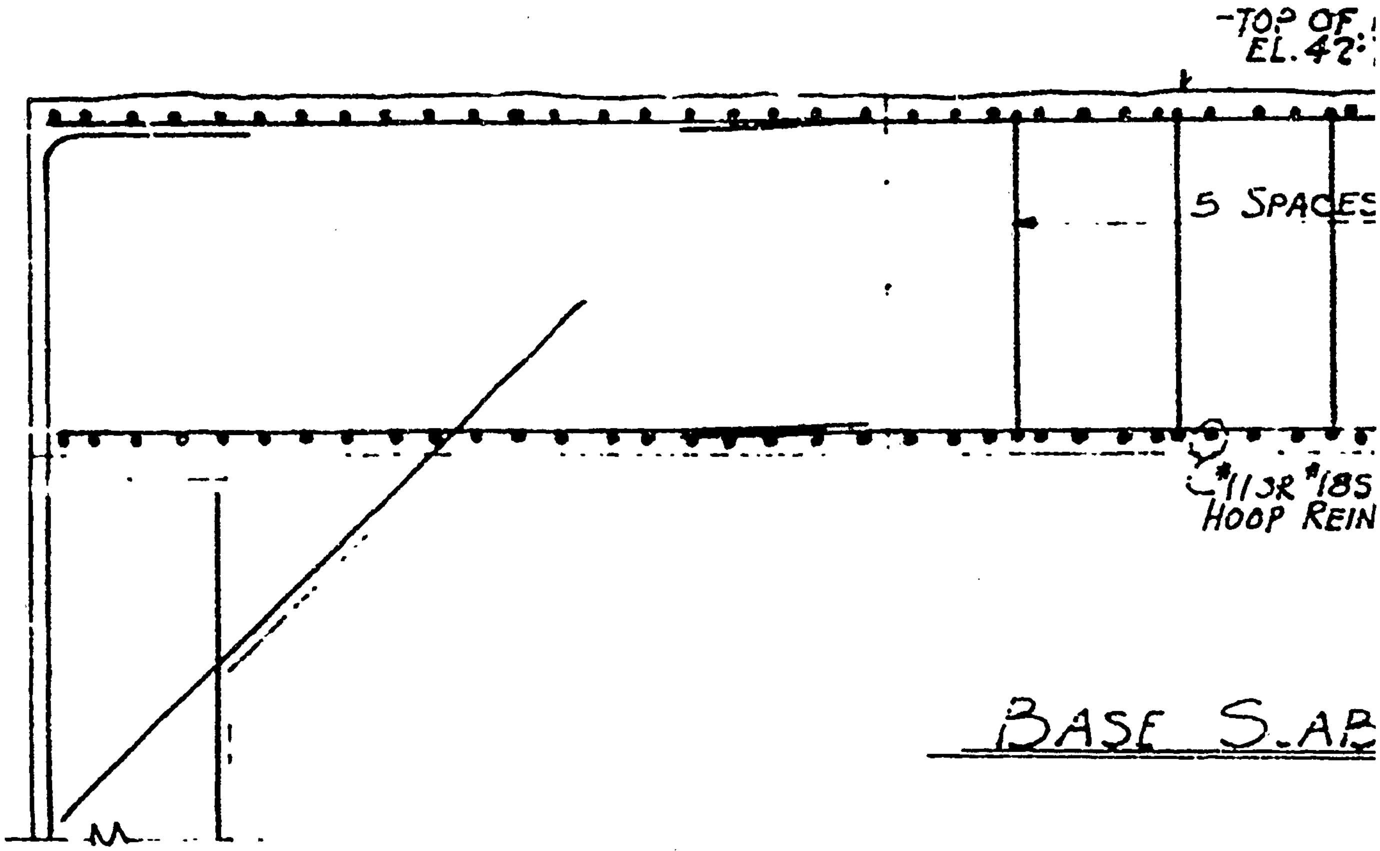
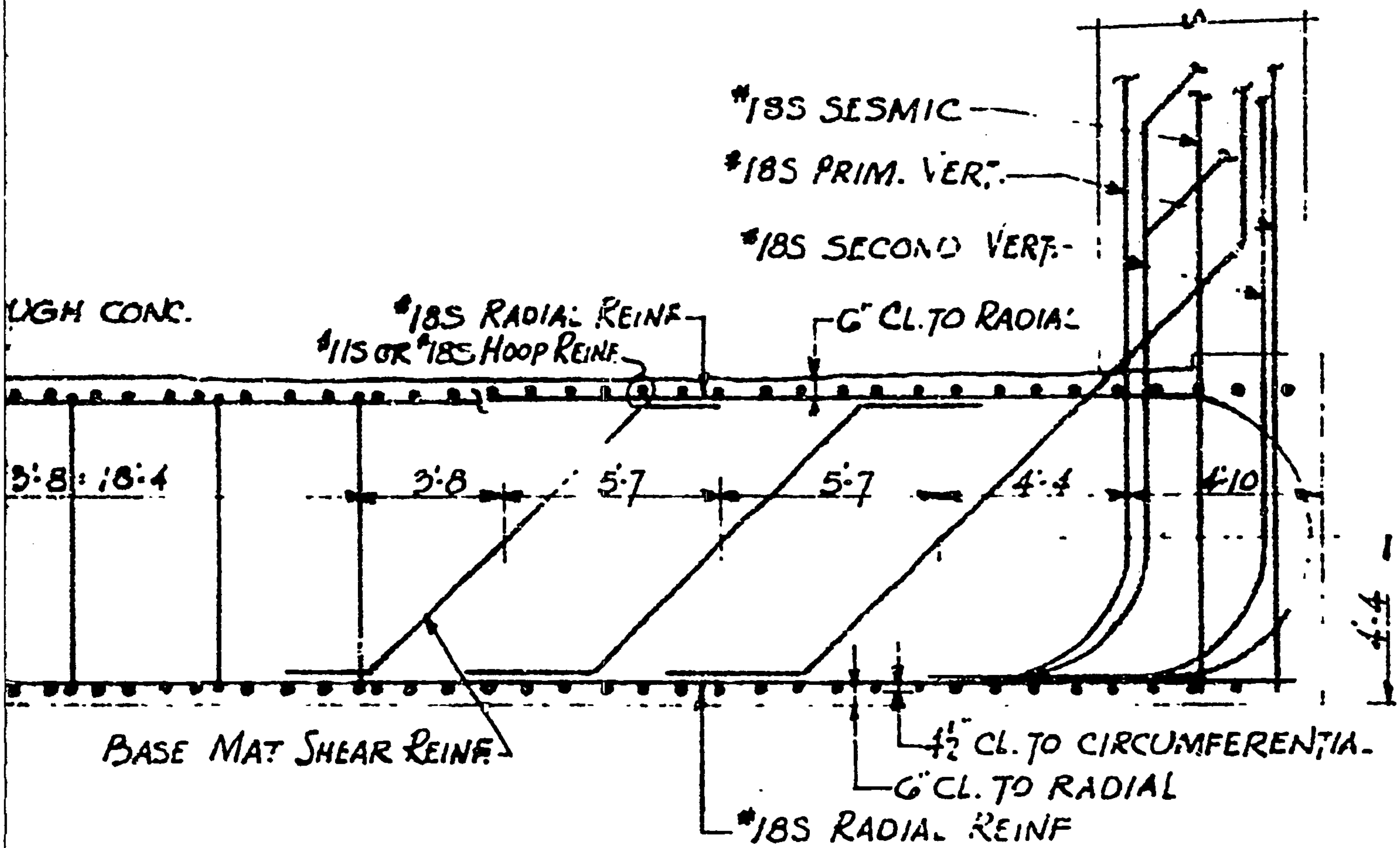
FIG. 5.1 (a)-3



BASE SLAB



REINFORCING DETAIL

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FIGURE 5.1a-4

QUESTION 5.1

For the containment structure, provide:

b) A description of how torsional loads have been handled. These torsional loads may be generated by an earthquake or tornado.

ANSWER

Due to symmetry of the containment structure torsional loads generated by an earthquake are insignificant and have not been considered.

Tornado loads have not been considered in the design of the Unit No. 2 Containment; however, the answer to question 2.4(d) of Supplement 4 to the PSAR for Indian Point Unit No. 3 indicates that the seismic bars provide a more than adequate mechanism to withstand this torsional effect if it were to occur.

QUESTION 5.1

For the containment structure, provide:

c) The analytical procedures used for arriving at the forces, shears, and moments in the structural shell, and the analytical procedures used for determining discontinuity stresses at the base and dome. State the assumptions, with regard to structural stiffness, that form the basis for these stress determinations and indicate the extent to which variations of E_c and u_c are considered.

ANSWER

A description of all analytical procedures is found in the Containment Design Report, Indian Point Unit No. 2.

For the extent to which variations in E_c and u_c are considered, see Question 2.4e in Supplement 2 as revised by Amendment No. 4 of the PSAR for Indian Point Unit No. 3.

QUESTION 5.1

For the containment structure, provide:

d) The values of E_c and u_c for cracked and uncracked reinforced concrete structure for different elevations and explain their use in the design of the concrete shell and in thermal liner loading computations. Include the effect of shrinkage. State whether the design takes into account these variations of E_c and u_c, and also the effect of axisymetric loads.

ANSWER

See answer to Question 2.4f, of Supplement 4 to the Indian Point No. 3 PSAR.

QUESTION 5.1

For the containment structure, provide:

e) An explanation of the manner in which shears are transferred through a concrete section which will be cracked under test and incident conditions.

ANSWER

See answer to Question 2.4h, of Supplement 4 to Indian Point No. 3 PSAR.

QUESTION 5.1

For the containment structure, provide:

f) Information on how the reinforcing bars are anchored at certain critical points, such as at the center of the dome, and the intermediate terminal points of radial bars in the dome, at locations where bars are provided to take discontinuity stresses, and diagonal bars, etc. Because of cracking of concrete due to shrinkage, testing, thermal stresses, and during an accident, the problem of adequate bar anchorage is of special concern.

ANSWER

See answer to Question 2.4j of Supplement 4 to the PSAR Indian Point No. 3.

QUESTION 5.2

Describe the "splicing" of inclined bars, or horizontal stirrups provided to take the radial shears in the base of the walls with the vertical bars. If done by lapping the diagonal bar with a vertical bar or by bending the stirrup around a vertical bar, demonstrate that, despite biaxial tensile stresses in concrete and vertical and horizontal crack patterns, the load in the diagonal bars or stirrups can be transmitted safely to the vertical bars.

ANSWER

Approximately 67% of the inclined bars provided to resist radial shear at the base of the containment wall are secondary vertical bars which are inside the primary vertical bars on the outside face and inside face and inside face of the wall. These bars are continuous and are bent across the wall where reinforcing is required to resist the radial shear. The remaining 33% of the required steel area is provided by stirrups which are hooked around the vertical bars by means of a 90° hook. Only 1/3 of the shear reinforcing at a particular elevation is made up of these hooked bars which occur at four elevations up the wall. See Fig. 4.16 of the Containment Design Report for Indian Point Unit No. 2. The lowest elevation at which these hooked bars are used is at a point where only 65% of the maximum shear at the base is present. The remaining three levels are in regions where the shear is less than 25% of maximum base shear. Since the large majority of the shear is resisted by continuous vertical bars a minimal amount of load must be transmitted to the vertical bars. The hooked stirrups will mechanically transmit the small amount of shear which they carry. The main function of the stirrups is to contain the formation of the diagonal tension crack. The mechanical anchorage of the stirrups is sufficient for this purpose.

QUESTION 5.3

The reinforcing steel may be stressed to the yield point when subject to the consequences of the DBA. Justify the use of this stress when under certain conditions, this would allow the liner to be stressed beyond the yield point Identify those areas of the liner which are stressed beyond the yield point.

ANSWER

The liner is stressed beyond the yield point in very local areas adjacent to the transition from the thickened equipment hatch boss to the cylinder wall. The maximum stress is equal to 39.28 ksi for the 1.5P loading condition. The strain corresponding to this stress (.17%) is below the limits stated in section 2.0-9 of the Containment Design Report (.5%). The average liner stress in the cylinder for the 1.5P load combination is approximately -15 ksi in the vertical direction and -2.0 ksi in the horizontal direction.

For a complete discussion of liner stresses, see Containment Liner Report for Indian Point Unit No. 2. For a detailed discussion of liner stresses in the Equipment Hatch area and further justification of the stress noted above, see Section 3.4.4 of the Containment Design Report for Indian Point Unit No. 2.

QUESTION 5.4

Indicate whether, in the design of the containment foundation mat, consideration was given to:

a) The influence of the elasticity of the ground on the stress distribution in the foundation mat.

ANSWER

See answer to Question 2.6a, Supplement 3 to the PSAR for Indian Point Unit No. 3.

QUESTION 5.4

Indicate whether, in the design of the containment foundation mat, consideration was given to:

b) The fact that the mat is under radial tension.

ANSWER

See answer to Question 2.5b of Supplement 2 to Indian Point Unit No. 3, PSAR.

The maximum rebar stress associated with the 1.5P load combination is approximately 6 ksi.

QUESTION 5.4

Indicate whether, in the design of the containment foundation mat, consideration was given to:

c) The lack of symmetry of seismic or tornado loads acting on the mat.

ANSWER

A Finite Element Analysis was performed on the base mat utilizing loads determined for the three basic loading conditions specified in the Containment Design Report. Maximum hoop moment caused by lack of symmetry of the seismic loading was found to be 454 in-k/in. This compares with a capacity of 690 in-k/in. for the in-place hoop reinforcing.

Although tornado loads were not considered in the design of the Unit No. 2 Containment, tornado loadings discussed in answer to Question 2.1(1) of Supplement No. 4 to the Indian Point Unit No. 3 PSAR are small compared to seismic loadings.

QUESTION 5.4

Indicate whether, in the design of the containment foundation mat, consideration was given to:

d) Thermal stresses.

ANSWER

See answer to Question 2.6a of Supplement No. 3 to the Indian Point Unit No. 3 PSAR.

QUESTION 5.5

With respect to liner design, describe:

a) The design approach that was used where loadings must be transferred through the liner, such as at crane brackets or machinery equipment mounts. Provide design details and computation results.

ANSWER

See answer to Question 2.8j of Supplement No. 4 to the Indian Point Unit No. 3 PSAR.

QUESTION 5.5

With respect to liner design, describe:

b) How the shears, especially those due to thermal expansion and earthquake, are accommodated. The bottom liner is not accessible for inspection during the life of the plant. It is therefore very important to avoid any unnecessary stresses and strains in the bottom liner. The arrangement for load transfer through the liner under the bottom of the interior structure should have provided for transfer of shears parallel to the liner.

ANSWER

See answer to Question 2.8k of Supplement 2 to Indian Point Unit No. 3 PSAR.

QUESTION 5.5

With respect to liner design, describe:

c) The liner buckling characteristics if a stud anchor should fail or be missing.

ANSWER

See answers to Questions 2.8b, 2.9a, 2.9b, and 2.9c of Supplement 4 to Indian Point Unit No. 3, PSAR.

QUESTION 5.5

With respect to liner design, describe:

d) The stud sizes and stud/liner weld sizes.

ANSWER

See answer to Question 2.9d of Supplement 4 to the Indian Point Unit No. 3 PSAR.

The arc stud welding process produces a circular weld around the 1/2" Ø stud with a diameter (outside to outside of weld) equal to .687 inches and a height equal to .157 inches.

QUESTION 5.5

With respect to liner design, describe:

e) The maximum stress in concrete at liner anchors.

ANSWER

Since the stud anchors are hooked around reinforcing bars concrete stresses for pull out loads are negligible. For high shear loads, which would be caused if a stud anchor should fail or be missing, local crushing of the concrete occurs; however, integrity of the anchor and liner plate is not impared.*

See answer to Question 2.9a of Supplement 4 to Indian Point Unit No. 3 PSAR.

*Design data - Nelson Concrete Anchor - printed 8/1/61

QUESTION 5.6

For penetration design, provide:

(a) a discussion of whether the stress limits of the ASME Code, Section VIII can be met, including the effects of jet forces.

ANSWER

All piping penetrations are designed for normal loads within the stress limits of the ASME Code, Section VIII.

All piping penetrations except main steam and feedwater are designed as anchors for the pipes passing through them and will transmit piping loads to the reinforced concrete wall. The anchorage strength exceeds the maximum combined forces imposed by the effects on the piping penetration of dead load, loads induced from a loss of coolant accident, thermal expansion of the pipe, penetration air pressure, and earthquake loads. The piping penetrations are designed to transmit the above combined loadings to the concrete structure without exceeding the yield strength of penetration steel.

In addition, each piping penetration is designed to withstand, within emergency load criteria, the effect of the rupture of a pipe passing through that penetration at or near the penetration.

The main steam and feedwater penetrations are designed so that the pipes themselves are effectively enclosed for blowdown just inside and just outside the wall. These anchors are designed to prevent a main steam or feedwater pipe rupture from causing a breach of containment at the penetrations. The anchors are designed to 90% of yield strength.

All piping penetrating the containment is designed to meet the requirements of USAS B31.1 (1955) Power Piping Code.

QUESTION 5.6

For penetration design, provide:

b) The extent to which the penetrations and the applicable surrounding linear regions will be subjected to vibratory loading from equipment attached to the piping systems. Indicate how these loads will be treated in design.

ANSWER

Pipes which penetrate the containment building wall and which are subject to machinery originated vibratory loadings, such as the reactor coolant pumps, will have their supports spaced in such a manner that the natural frequency of the piping system immediately adjacent to the penetrations will be greater than the dominant frequencies of the pump. Pipe line vibration will be checked during preliminary plant operation; and where necessary, vibration dampers will be fitted. This checking and fitting will effectively eliminate vibrating loads as a design consideration.

QUESTION 5.6

For penetration design, provide:

c) The capability of the penetration design to absorb liner strain without severe distress at the opening.

ANSWER

See answer to Question 2.10d of the PSAR for Indian Point No. 3.

QUESTION 5.6

For penetration design, provide:

d) For all penetrations, the criteria that have been used for the bending of reinforcing bars which have to clear the opening. Criteria defining maximum slopes and minimum bending radii to avoid local crushing of concrete and a discussion of the anchorage of bars added to typical reinforcing or interrupted at the opening should be included.

ANSWER

See answer to Question 2.10f of Supplement 4 to the Indian Point Unit No. 3 PSAR. All bars except stirrups and facing bars which are not counted on to carry any load, are continuous around the openings.

QUESTION 5.6

For penetration design, provide:

e) For penetrations between approximately 9 inches and 4 feet in diameter, explain how normal, shear, bending, and torsional stresses are covered by the reinforcing bars.

ANSWER

See answer to Question 2.10g of Supplement 4 to the Indian Point Unit No. 3 PSAR.

Question 5.6

For penetration design, provide:

f) Sample drawings, sketches, and design computations for penetrations and shell adjacent to them.

Answer

For sample drawings and sketches of the penetrations and the shell adjacent to them, see Figure 2.10a-2 of Supplement 4 to the PSAR for Indian Point Unit No. 3.

For design computations of penetrations and the shell adjacent to them, see pages Q 5.6(f)-6 and Q 5.6(f)-7. On page Q 5.6(f)-2 the formula for radial deformation of a hole in a plate subjected to biaxial stresses is determined by performing an integration of the tangential strains around the periphery of the hole.

On page Q 5.6(f)-3 the relationship between the deflection determined from above to the final plate and penetration sleeve deformations is developed and the formulas for stress in the liner and the stress in the penetration sleeve are developed.

Pages Q 5.6(f)-4 and Q 5.6(f)-5 show a summary of the liner and penetration stresses and state the assumptions made in the analysis.

In addition, thermal loads have been investigated for their effect on the shell adjacent to the penetration sleeve and found to be insignificant (38 psi bearing stress on the concrete is the maximum stress on the concrete shell).

For further discussion of penetration design, see answer to Question 2b of Supplement No. 3 to the Indian Point Unit No. 2 PSAR.

From Timoshenko, page 81, "Theory of Elasticity"

$\sigma o = S - 2S \cos 2\theta + [S' - 2S'\cos(2\theta - \pi)]$ S = Horizontal Stress in Liner
S' = Vertical Stress in Liner

$$\delta D = \frac{1}{E} \int_0^{\pi} (S - 2S\cos 2\theta + [S' - 2S'\cos(2\theta - \pi)])\, r\sin\theta d\theta$$

$$\delta D = \frac{r}{E}\left[\int_0^{\pi} S \sin\theta d\theta - 2S\int_0^{\pi}\cos 2\theta \sin\theta d\theta + S'\int_0^{\pi}\sin\theta d\theta - 2S'\int_0^{\pi}\cos(2\theta - \pi)\sin\theta d\theta\right]$$

$$\int \cos(2\theta - \pi)\sin\theta d\theta = -\int \cos 2\theta \sin\theta d\theta$$
$$= -\int (1 - 2\sin^2\theta)(\sin\theta) d\theta$$
$$= -\int(\sin\theta - 2\sin^3\theta) d\theta$$
$$= -[(-\cos\theta) - 2(-\frac{\sin^2\theta\cos\theta}{3} + \frac{2}{3}\int \sin\theta d\theta)]$$
$$= -(-\cos\theta + 2/3\sin^2\theta\cos\theta + \frac{4}{3}\cos\theta)]$$

$$\int\cos(2\theta - \pi)\sin\theta = \frac{-\cos\theta}{3} - \frac{2}{3}\sin^2\theta\cos\theta$$

$$\delta = \frac{r}{E}\left[-S\cos\theta - 2S(\frac{\cos\theta}{3} + \frac{2}{3}\sin^2\theta\cos\theta) - S'\cos\theta - 2S'(\frac{-\cos\theta}{3} - \frac{2}{3}\sin^2\theta\cos\theta)\right]_0^{\pi}$$

$$\delta = \frac{r}{E}[(S + \frac{2}{3}S + S' - \frac{2}{3}S') - (-S - 2/3S - S' + 2/3S')]$$

$$\delta = \frac{r}{E}[2S + \frac{4}{3}S + 2S' - \frac{4}{3}S']$$

$$\frac{r}{E}[\frac{10}{3}S + \frac{2}{3}S']$$

$\frac{2}{3}\frac{r}{E}[5S + S']$ (For Stresses in the Same Direction)

$\frac{2}{3}\frac{r}{E}(5S - S')$ (For Stresses in Opposite Directions)

$$\Delta_{UN} = \Delta_{\text{℄}}\ \text{Res.} + \Delta_{\text{Sleeve}}\ ^{*}$$

$$\Delta_{UN} = \frac{S_1}{E}(1-V)\,R + \frac{S_1(t_{\text{℄}})\,R^2\lambda}{2Et_{\text{sleeve}}}\ ^{**}$$

$$\Delta_{UN} = \frac{S_1}{E}\left[\,R(1-V) + \frac{t_{\text{℄}}\,R^2}{2\,t_{\text{sleeve}}}\lambda\right]$$

$$S_1 = \frac{\Delta_{UN}E}{R\left[(1-V)+\left(\frac{t_{\text{℄}}\,R\lambda}{2t_{\text{sleeve}}}\right)\right]}$$

$$S_{\text{sleeve}} = \frac{S_1 t_{\text{℄}}\ R\lambda}{2t_{\text{sleeve}}}$$

$$S_{\text{sleeve}} = \frac{\Delta_{UN}E\ t_{\text{℄}}\ R\ \lambda}{R\left[(1-V)+\left(\frac{t_{\text{℄}}\ R\lambda}{2t_{\text{sleeve}}}\right)\right]\ 2\ t_{\text{sleeve}}}$$

$$\text{where } \lambda = \sqrt[4]{\frac{3(1-V^2)}{R^2 t^2_{\text{sleeve}}}}$$

* (See Answer to Question 5.6c)

** (S_1 = Stress in Liner)

Summary

Penetration	Stress in Sleeve (ksi)	Stress in Liner (ksi)
Air Purge	-23.8	-19.5
Main Steam	-33.4	-27.94
Typical Mech. Penetration	-31.0	-31.1
Electrical Penetration		
A) C & T*	-22.5	-29.5
B) T & T*	+18.2	+19.7
Fuel Transfer		
A) C & T*	-25.7	-25.6
B) T & T*	+20.8	+16.6

A) ignores effects of insulation in the vertical direction B) considers effects of insulation

C = Compression in Liner T = Tension in Liner

Conservative Assumptions

1. The weld pressurization channel stiffens the area.

2. The liner alone was designed for stress concentration effects while the cracked concrete was ignored.

3. The unrestrained growth is based on maximum growth from a stress concentration consideration.

* first letter represents direction of vertical liner stress
second letter represents direction of horizontal liner stress

4. The main steam and mechanical penetrations have been considered in a non-insulated zone when they are just inside the insulated zone. The compression in the hoop direction will be greatly reduced or perhaps go into tension, thus reducing the stresses.

5. The allowable stress in the sleeve = 56,700 psi except for the stainless steel fuel transfer penetration = 49,500 psi. These values come from Table N-421 and Figure N-414 ot the ASME Nuclear Vessel Code Section III.

(Sample calculation)

GENERAL COMPUTATION SHEET

UNITED ENGINEERS & CONSTRUCTORS INC.

FORM 361

J. O. NO. 9321-06

NAME OF COMPANY: CON EDISON - INDIAN POINT No 2 — SHEET NO. 6 OF ____

DATE ____

SUBJECT: CONTAINMENT - STRESSES ON PENETRATIONS & LINER — COMP. BY L.B.S. — C'K'D BY [illegible]

ELECTRICAL PENETRATIONS

EL. 57'-0 D.A=10.75
SLEEVE t=.594

A) COMPRESSION N AND TENSION H

$$S' = \frac{(70.5)(.937)(12)}{(1000)\,24.96}(.43)\ \text{RESTRAINT}$$

= 12.2 k/in² TENSION

LINER A_s/FT = (.75)(12) = 9.00
SEISMIC 4 8($\frac{1}{10}$)(1.414) = 2.26
HOOPS +16($\frac{1}{12}$) = 13.70
= 24.96 □"/FT

S ≈ 20.2 k/in² COMPRESSION

$$\Delta_{UN} = \frac{2}{3}\cdot\frac{5.375}{29\times10^3}\left[5(20.2) - 12.2\right] = 11.00\times10^{-3} = .011\times .5 = 0.0055$$

(88.8)

(radial deformation)

$$\lambda = \sqrt[4]{\frac{3(1-.25)}{(5.375^2)(.594)^2}} = .724$$

$$S_{SLEEVE} = \frac{(.0055)(29\times10^3)(.750)(.724)}{\left[(1-.25) + \underbrace{\frac{(.750)(5.375)(.724)}{2(.594)}}_{3.21}\right]2(.594)} = .0225\times10^3 = \underline{\underline{22.5\ \text{ksi}}}$$

$$S_{LINER} = \frac{(.0055)(29\times10^3)}{5.375\left[(1-.25) + \underbrace{\frac{(.750)(5.375)(.724)}{2(.594)}}_{3.21}\right]} = .00938\times10^3 = \underline{\underline{9.3\ \text{ksi}}}$$

Supplement 4
12/69

FORM 501

(Sample calculation)
GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY: CON EDISON - INDIAN POINT #2 — J. O. NO. 9321-06 — SHEET NO. 7 OF
SUBJECT: CONTAINMENT - STRESSES ON PENETRATIONS & LINER — DATE — COMP BY B.B.S. CK'D BY

ELECTRICAL PENETRATIONS

8) TENSION V & TENSION H

$S = 12.2^{K}/in^2$

$S' \; \frac{297}{27.26} = 10.9^{K}/in^2$

As/FT (V)
LINER .75(12) = 9
VERTICAL BARS 4×4 = 16
SEISMIC 8(13/30)(1.414) 2.26
27.26"/f

$\Delta_{UN} = \frac{2}{3} \cdot \frac{5.375}{29 \times 10^3} \left[\overset{71.9}{5(12.2) + 10.9}\right]$

$\Delta_{UN} = \frac{71.9}{88.8} \times .011 = .0089$

$S_{SLEEVE} = \frac{71.9}{88.8} \times 22.5 = \underline{\underline{18.2}}^{\,ksi}$

$S_{LINER} = \frac{71.9}{88.8} \times 93 = \underline{\underline{7.52}}^{\,ksi}$

QUESTION 5.7

For large openings, describe:

a) The number and size of the large openings for the containment which require special design consideration.

ANSWER

The 16'-0 diameter equipment hatch opening and the 8'-6 diameter personnel are the only openings which require special design consideration.

QUESTION 5.7

For large openings, describe:

b) The primary, secondary, and thermal loads that have been considered in design of the openings, including jet, seismic and tornado loads.

ANSWER

See Section 3.4 of the Containment Design Report for Indian Point Unit No. 2. No jet forces or tornado loads have been considered.

QUESTION 5.7

For large openings, describe:

c) The stress analysis procedures that were used in design.

ANSWER

See Indian Point Unit No. 2 Containment Design Report, Section 3.4.0.

QUESTION 5.7

For large openings, describe:

d) The method that was followed for the design (working stress design method, ultimate strength design method, or both). If ultimate strength was used, the factored load combinations should be given together with corresponding capacity reduction factors.

ANSWER

See Indian Point Unit No. 2 Containment Design Report Section 3.4.0.

QUESTION 5.7

For large openings, describe:

e) How the existence of biaxial tension in concrete (cracking) has been taken care of in the design and how the normal and shear stresses due to axial load, two-directional bending, two-directional shear, and torsion are combined. Also, state the criteria for the design of the thickened part of the wall around the opening.

ANSWER

See Indian Point Unit No. 2 Containment Design Report Section 3.4.0.

QUZSTION 5.7

For large openinge, describe:

f) The method used to check the design of the thickened stiff part of the shell around large openings and its affect on the shell. Include the manner of considering shrinkage. Decribe how torsional stresses have been checked.

ANSWER

See Indian Point No. 2 Containment Design Report, Section 3.4.0.

QUESTION 5.7

For large openings, describe:

g) Additional information on reinforcing patterns used around large openings. How are deformations and forces handled around the large opening and in the transition zone into the main portion of the structure?

ANSWER

See Indian Point No. 2 Containment Design Report, Section 3.4.0.

QUESTION 5.7

For large openings, describe:

h) The safety factor provided in design at large openings. Sample computations should be provided, listing all the criteria and analyzing the effect of all pertinent factors, such as cracking.

ANSWER

See Indian Point No. 2 Containment Design Report, Section 3.4.0.

QUESTION 5.8

For the insulation, provide:

a) The design safety factor provided on insulation performance, including specified (80°F) vs. tolerable temperature rise in liner.

ANSWER

The manufactures analog transient analysis indicates only a 5° rise in liner temperature 1000 seconds after an exposure to 310°F for the entire duration of the analysis.

This provides a factor of safety of approximately 15.0 on specified tolerable temperature rise in the liner.

A factor of safety of 2.0 is provided on specified insulation performance vs. tolerable temperature rise in liner.

QUESTION 5.8

For the insulation, provide:

b) An analysis of the consequences of one or more insulation panels being displaced from the liner during, or as a consequence of, an accident situation. Consider whether jet forces may remove panels.

ANSWER

See answer to Question 2.12c of Supplement 4 to the Indian Point Unit No. 3 PSAR.

Jet forces cannot remove the panels since the forces will be compressing the insulation panels against the liner and exterior wall. The panels are anchored to the liner with 3/16 Ø stainless steel studs.

QUESTION 5.8

For the insulation, provide:

c) The consideration given to increased conductivity due to humidity and compression during accident pressure transients and pre-compression from structural and leakage testing.

ANSWER

See the answer to Question 2.12d of Supplement 4 to Indian Point Unit No. 3 PSAR.

QUESTION 5.8

For the insulation, provide:

d) An evaluation of the compatibility of the insulation materials and steel liner relative to chemical reaction, etc.

ANSWER

See answer to Question 2.12e of Supplement 4 to the PSAR for Indian Point No. 3.

QUESTION 5.9

Discuss the extent of consideration given to the need for cathodic protection. What protection is provided? If soil resistivity surveys have been conducted, report the results.

ANSWER

A complete survey and tests to determine the need for cathodic protection on Unit 2 was made by the A. V. Smith Engineering Company of Narberth, Pennsylvania. Electrical resistivity measurements and a visual inspection of the area away from the river, where the Turbine Generator Building, Reactor Building, P. A. Building and associated facilities are located indicated that the environment is mostly rock with areas of dry sandy clay. The electrical resistivity of the soil ranged from 3,500 to 30,000 ohm-centimeters with the majority of the readings being above 10,000 ohm-centimeters. On this basis, it was determined that cathodic protection was not required on underground facilities in areas away from the river or the containment building liner, although a protective coating on pipes was recommended to eliminate any random localized corrosion attack.

An analysis of Hudson River water data, obtained from the Consolidated Edison plant chemist, showed the electrical resistivity of the water to vary over an extremely wide range due to salt intrusion from the ocean. The range of resistivity has been from 59 to 10,000 ohm-centimeter with a large number of readings in the 300 ohm-centimeter area. This value was considered to be extremely corrosive and the following structures in the area near the river were placed under cathodic protection:

1. Circulating water lines
2. De-icing lines
3. Service water lines
4. Bearing piles
5. Sheet piling (earth and water side) and wing wall anchorage system.
6. Metallic structures inside intake structure (traveling screens, bar racks, circulating water pump suction, service water pump suction).

QUESTION 5.10

Discuss the extent to which protective coatings have been applied to the liner. State the effect of post-accident environment including temperature, pressure, and reagent spray on the coatings.

ANSWER

One 3 mil shop coat of Carbozinc #11 primer and one 4 mil minimum finish coat of Phenoline #305 as manufactured by the Carbozinc Company have been applied to the liner in accordance with the manufacturer's recommendations.

The effect of the post-accident environment on protective coating was conservatively evaluated for Indian Point Unit No. 2. The coatings showed no deterioration after a number of cycles. A more thorough discussion is presented in WCAP-7198-L entitled "Evaluation of Protective Coatings for Use in Reactor Containment" - April 1968 which has been submitted to the AEC.

QUESTION 5.11

With regard to the design of the interior structure of the containment used to support and enclose the primary system and other equipment provide the following:

a) Explain generally the methods used for design of the interior structure. Indicate the method of stress analysis and give a list of critical stresses, their nature and location.

ANSWER

The interior structure may be separated into five main structural components. They are:

1. 3' thick fill slab
2. 3' thick crane wall
3. 4' to 6' thick refueling canal
4. 2' - 0 thick operating floor slab
5. Primary Shield Wall

The method of design, stress analysis, critical stresses and locations are as follows:

1. 3' Thick Fill Slab - The controlling loads on the 3' fill slab where the reactions from the primary equipment supports due to various postulated pipe breaks. The slab was designed as a series of radial beams running under the equipment supports and spanning between the reactor support wall and the crane wall. Stresses in reinforcing were limited to 0.9 fy. Maximum stresses occur immediately below the primary equipment supports.

2. 3' Thick Crane Wall - The crane wall was designed for a 7 psi differential pressure occurring immediately after a primary pipe break and prior to pressure equalization.

   Although the stress levels associated with this pressure differential were sufficiently low to establish that the concrete could resist the pressure loading, sufficient reinforcing was provided to resist all membrane forces without any contribution from the concrete. Stresses were limited to 0.9 fy. The membrane hoop stress was 33 ksi and the axial vertical rebar stress was 14.3 ksi.

A two dimensional Finite Element Analysis was performed to determine the area which would be affected by the Jet Force. The analysis indicated that in local areas (at the application of the force) yielding of the crane wall rebar will occur. The load was assumed to act at the mid-height of the wall, thus causing maximum bending moment. The ability of the wall to support the dead load of the crane was checked, considering the yielded area indicated by the Computer analysis as unable to carry load. A beam 12'-0 long and 5'-0 deep (the underside of the operating floor to the top of the potential yield portion of the crane wall) was found to provide more than twice the ultimate capacity required. This analysis was very conservative for three (3) reasons:

1. A Jet Force load at this location would cause little yielding since it is not located at mid span.

2. The haunch at the underside of the operating floor was not considered.

3. The membrane effect of the circular crane wall was not taken into account.

Further stability of the crane wall was demonstrated by determining the ultimate failure load by means of a yield line analysis. This analysis indicated that the structure has the capacity, through strain energy of structural response, to resist the uniform Jet Force load of 1500 k or 975 k with the 7 psi pressure differential without failure.

The containment internal concrete is essentially rigid; (fundamental frequency ~ 17 cps) therefore, seismic loads were calculated using the maximum ground acceleration (.15g).

The crane wall was considered as a cantilever beam and the base shear determined by the response spectrum approach. The base shear was distributed to the individual nodes by the formula

$$F_x = \frac{W_x h_x}{\Sigma W_x h_x} V$$

where V = Base Shear

W_x = Weight of note under consideration

h_x = Distance from base to section under consideration.

The moment at the base was determined and the uplift calculated by considering a circular ring of thickness equal to the area of steel per inch. This maximum uplift which occurs at one point at the base of the structure stresses the rebar to 5.2 ksi.

The crane wall was also designed to resist steam and feed water pipe break reactions of 340 k and 200 k where supports are connected to the wall. This extra steel provided for pipe break loads is available in the form of steel buttresses to resist pressure, Jet Force and seismic loads; however, it was not considered in the analysis.

3. <u>4' to 6' Thick Refueling Canal</u>

The refueling canal was designed for the 7 psi pressure differential. The wall resists the pressure by spanning vertically between the refueling floor and the operating floor. Stresses were limited to 0.9 fy.

A Finite Element Analysis was also performed to check the effects of the Jet Force load. Some local yielding was indicated; however, the cross section was sufficient to provide stability since the moment capacity is more than four times greater than that of the crane wall. A yield line analysis was performed and provided the basis for the above.

The seismic load was determined by the same procedure used for the crane wall. The average load in kips/ft was distributed over the wall and the vertical span was conservatively assumed to carry the entire load. The resulting bending moment produces a stress of approximately 3 ksi in the rebar.

4. <u>2' to 0 Thick Operating Floor Slab</u>

Because of the many openings in the floor for equipment, the floor was designed as a series of beams. Principal loadings were D.L. + 500 psf live load and 7 psi pressure differential + D.L. The first loading (D.L. + 500 psf live load) was designed in accordance with Part IV-B of ACI 318. Stresses for the pressure differential case were limited to 0.9 fy.

The operating floor was investigated. There appears to be very little area of the operating floor which could be reached by the expanding jet of water from a break in the Reactor Coolant System. The jet will be greatly dispersed in the distance between the primary coolant piping and the underside of the operating floor. The only area of the floor which could be struck by a jet will span between areas of the floor heavily reinforced as beams. The span cross section will consist of a T-beam with the 2'-0 thick floor acting as the flange and the 7'-0 high biological shielding wall as the web. This section can resist the jet force load within 0.9 fy stress limit on the rebar.

5. <u>Primary Shield Wall</u> was designed for two loading conditions due to a split in the reactor. The stress in the reinforcing was limited to the tensile strength of the bars. The first load considered was a 1'-0 wide longitudinal split along the length of the reactor. The vessel is assumed accelerated through a six inch distance against the support wall by the Jet Force caused by a 2200 psi pressure which results in a live load of 625 k/ft. This load is imposed by considering an impact factor of two. The maximum rebar stress is 69.5 $^{k}/in^{2}$. The second load considered a pressure build up of 1000 psi inside the pit due to release of reactor contents. This produces a rebar stress of 86 ksi.

To protect the containment base liner an average of 2' of concrete, above the containment liner, plus a 1" liner plate embedded on top of the concrete was provided at the bottom of the Containment Reactor Cavity Pit. Below the containment liner plate is 4-1/2 ft. of structural concrete poured on rock.

QUESTION 5.11

(b) With regard to the design of the interior structure of the containment used to support and enclose the primary system and other equipment provide the following:

Indicate the design pressures and the design temperature differentials that were used for different chambers within the structure. Provide the basis or method of determination of the above. Provide a sketch summarizing the above information.

ANSWER

The evaluation of containment internal structures appears on page 14.3.4-22. The design conditions for internal containment walls is explained in he response to Question 5.11a. Temperature differential conditions as a result of a LOCA are considered to be of such short duration that the effects were not used in the design of interior structures for stress analysis. A sketch of the design conditions is given in Figure 5.11(b)-1.

During normal operations, the only significant transient temperature gradients occur during startup. The minimum containment internal temperature is limited to 50°F. The maximum operating containment internal temperature is 120°F. Forced movement of containment air is used to limit the concrete temperature surrounding the reactor vessel. This forced air movement of the containment air as well as normal convection and radiation is expected to limit the concrete temperature differentials in the range of 5F° to 10F°. To demonstrate the large margin available in the concrete crane wall and the primary shield wall, a conservative assumption of a 30F° temperature gradient has been evaluated. The evaluation included the gradient effect through the crane wall, the 6' thick portion of the primary shield wall below the reactor coolant pipe nozzle, the 5' thick portion of the primary shield wall where the nozzles penetrate the wall, and the 4' thick wall above the shield wall.

The maximum rebar stress was found to be 4500 psi and occurs in the vertical rebar in the crane wall. The maximum compressive concrete stress was found to be 226 psi and occurs in the hoop direction in the 5' portion of the primary shield wall. These stresses are approximately 20% of the allowable working stress values and will have no significant effect on the design adequacy of the structures analyzed.

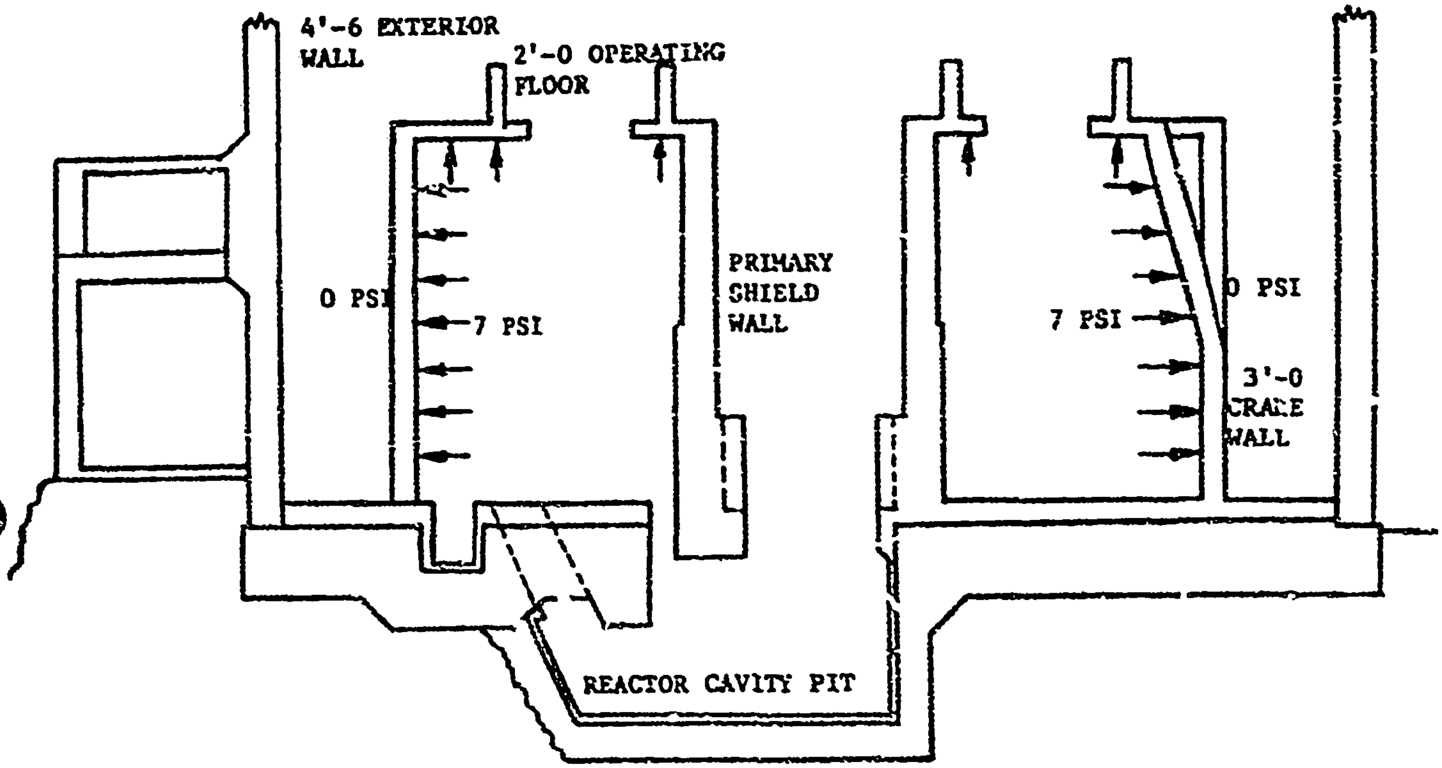


FIGURE 5.11(b)-1

QUESTION 5.11

With regard to the design of the interior structure of the containment used to support and enclose the primary system and other equipment provide the following:

(c) Explain the method or computational model used to determine jet forces on interior structure resulting from pipe fracture.

ANSWER

The jet force associated with a pipe break has been based on the static force PA where P is the primary system operating pressure and A is the cross sectional area of the coolant pipe. See response to 5.11(a) for method of applying the jet force.

QUESTION 5.12

Show how thermal stresses in the walls of the spent fuel pool resulting from temperature gradients were evaluated and state the provisions that have been made to limit cracking and prevent leakage.

ANSWER

The thermal stresses in the walls of the spent fuel pool resulting from temperature gradients were evaluated by the procedure outlined in the ACI 505-54 "Specifications for Design and Construction of Reinforced Concrete Chimneys". For the portion of the pool below grade a linear gradient with a 120°F water temperature and a 50°F outside temperature was assumed for the analysis. A gradient of 120°F water temperature to 0°F outside temperature was used for the structure above grade. A maximum liner stress of -2.7 ksi and a maximum rebar stress of 19 ksi were determined by the above analysis. These stresses are both well below the yield of the liner and the reinforcing steel (32 ksi and 60 ksi respectively).

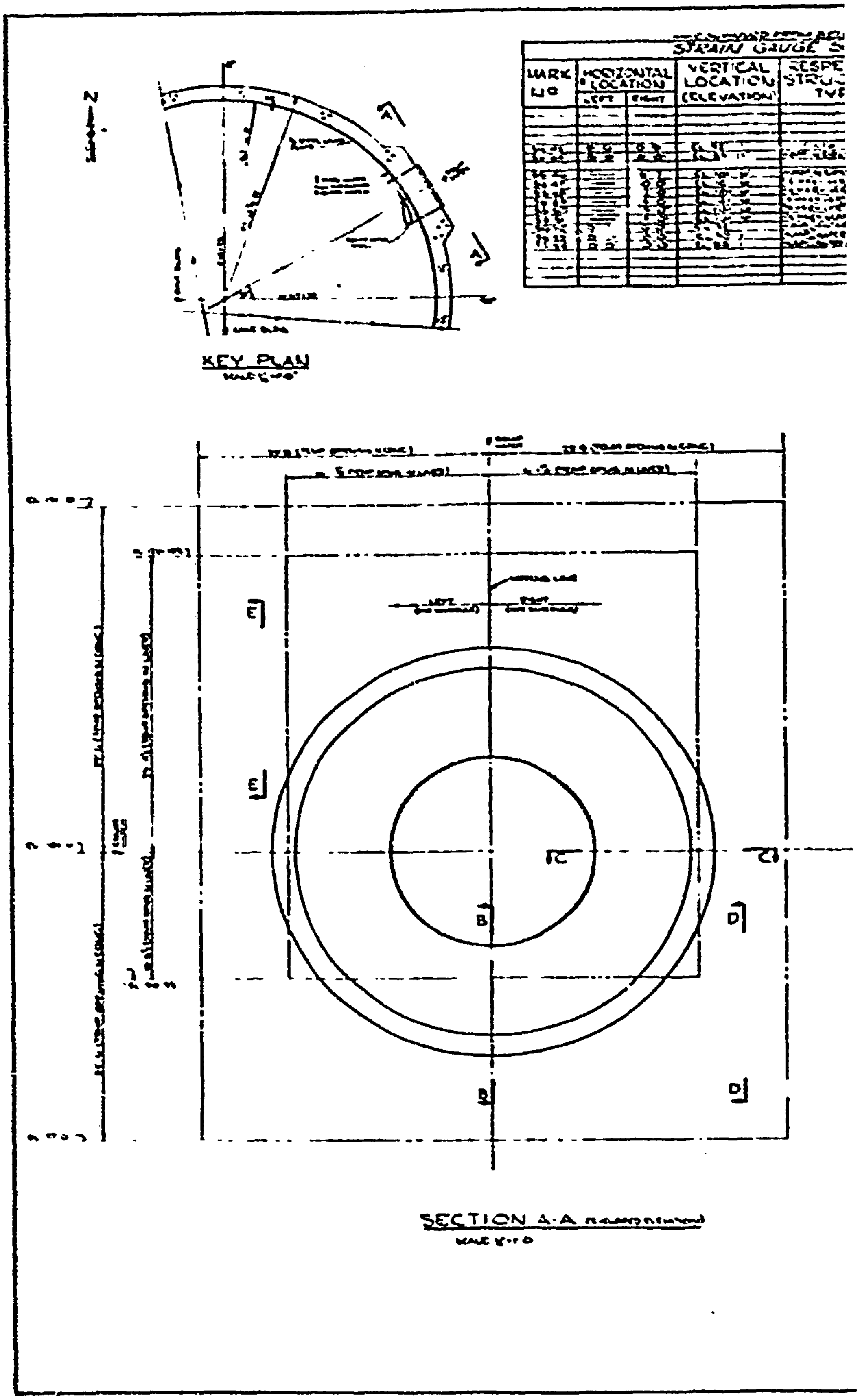
No special provisions were made to limit cracking and prevent leakage since the pit is lined with a leak proof stainless steel liner. All welds were vacuum-box tested during construction to assure a leak tight membrane. The effect of a thermal gradient would be to compress the liner, thereby, preventing any leakage.

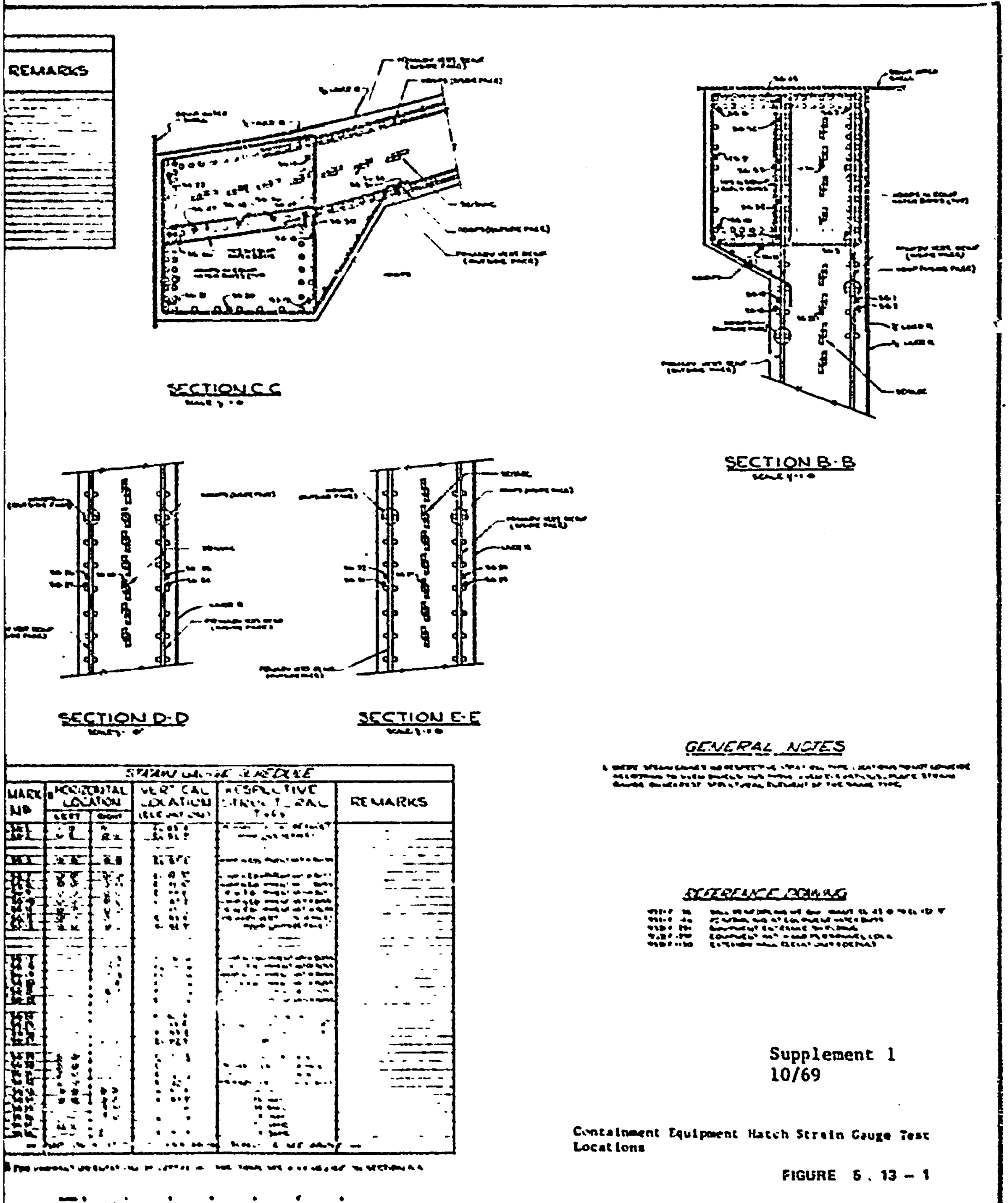
QUESTION 5.13

Indicate the areas of the containment and liner which are to be instrumented for the strength test.

ANSWER

For the areas of the containment and liner which are to be instrumented for the strength test see Figures 5.13-1; 5.13-2; 5.13-3; and 5.13-4.


STRAIN GAUGE S
MARK NO
HORIZONTAL LOCATION
LEFT
RIGHT
VERTICAL LOCATION (ELEVATION)
RESPE STRUC TYP
KEY PLAN
LEFT
RIGHT
SECTION A-A



Supplement 1
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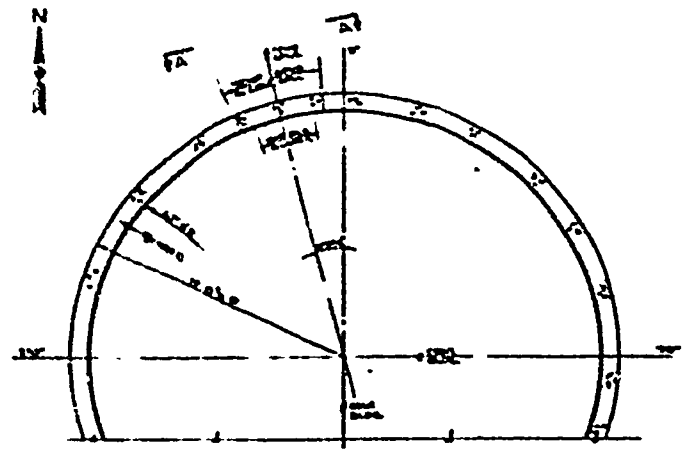
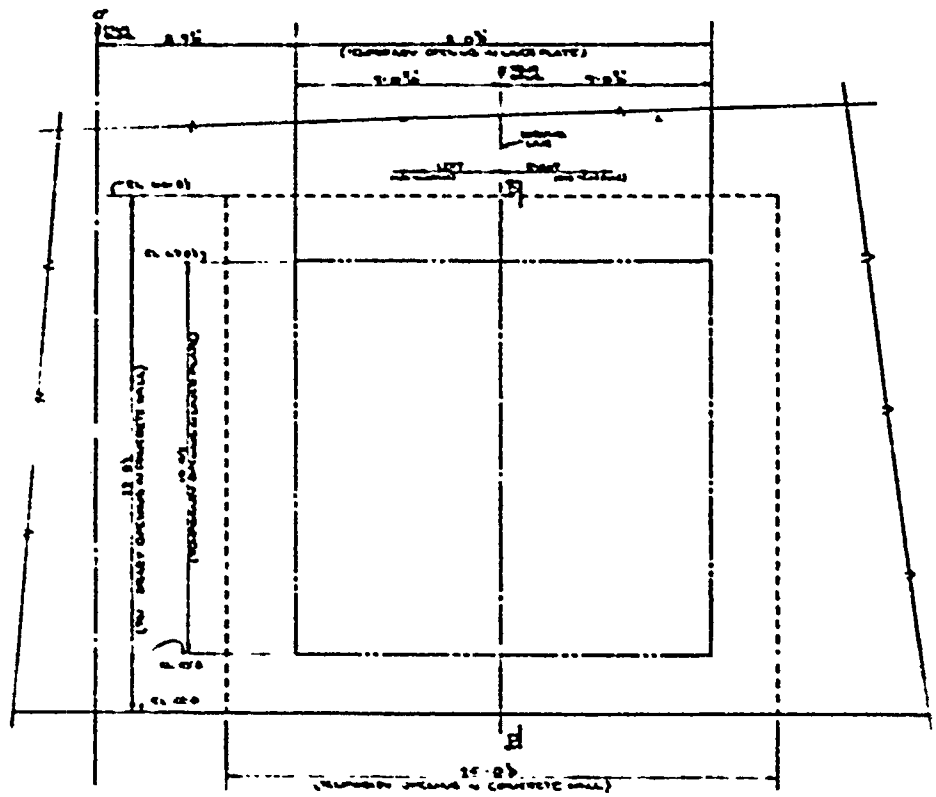
Containment Equipment Hatch Strain Gauge Test Locations

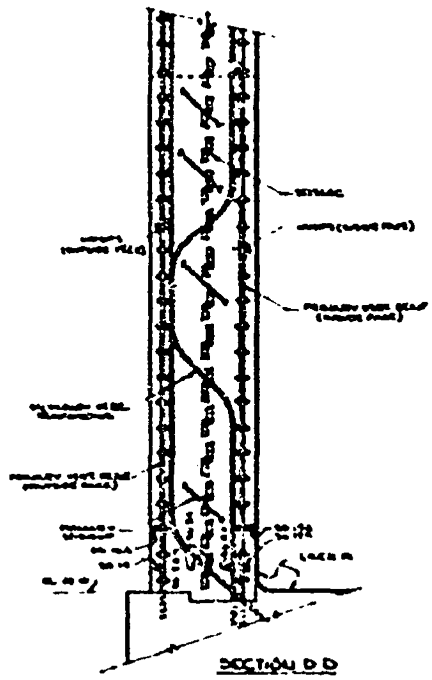
FIGURE 5.13-1



KEY PLAN



SECTION A A



[illegible]					
MARK NO	HORIZONTAL ℄ LOCATION		VERTICAL LOCATION	... STRUCTURAL TYPE	REMARKS
	[illegible]	[illegible]	[illegible]		
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[illegible]

GENERAL NOTES

[illegible]

REFERENCE DRAWINGS

[illegible]

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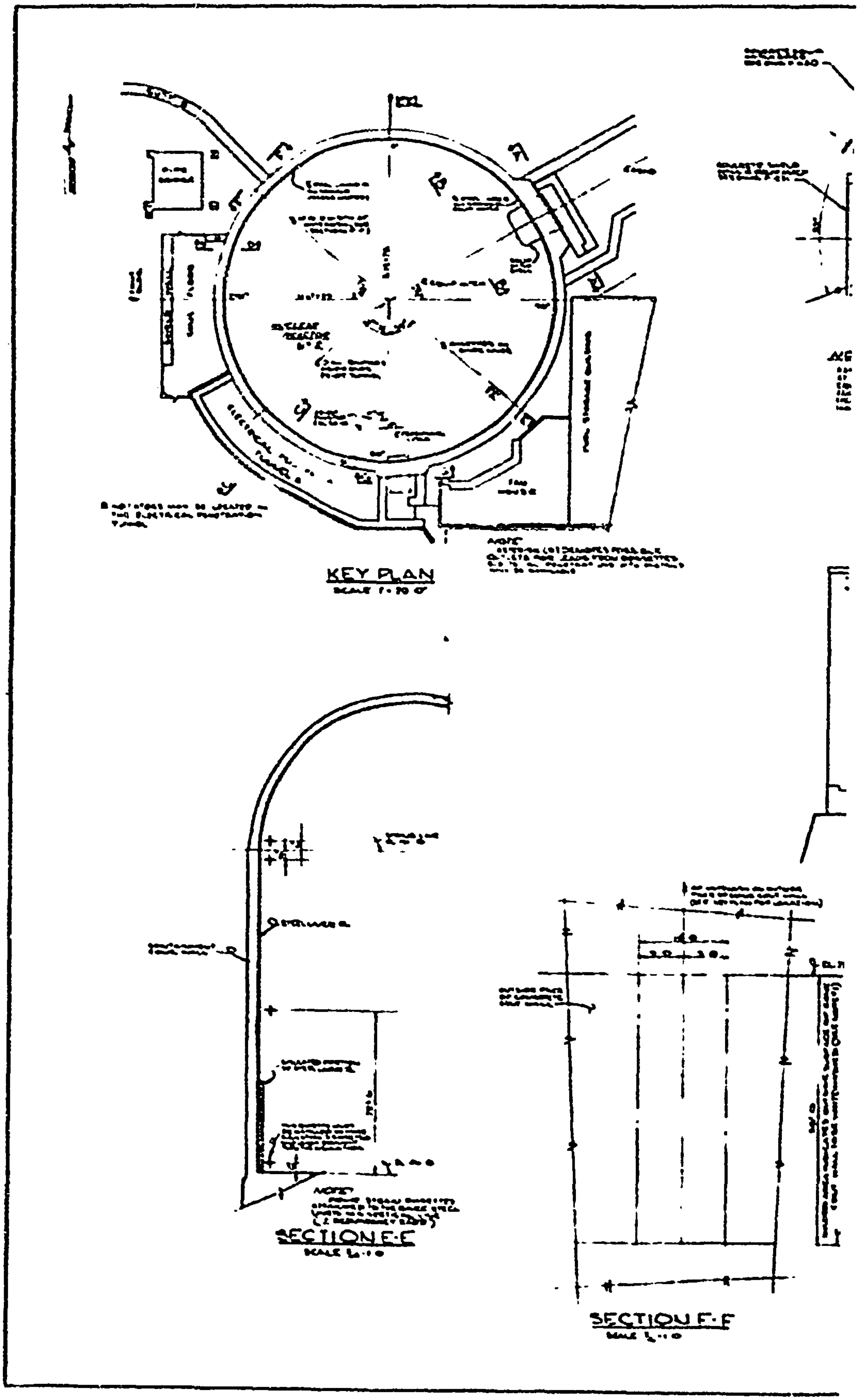
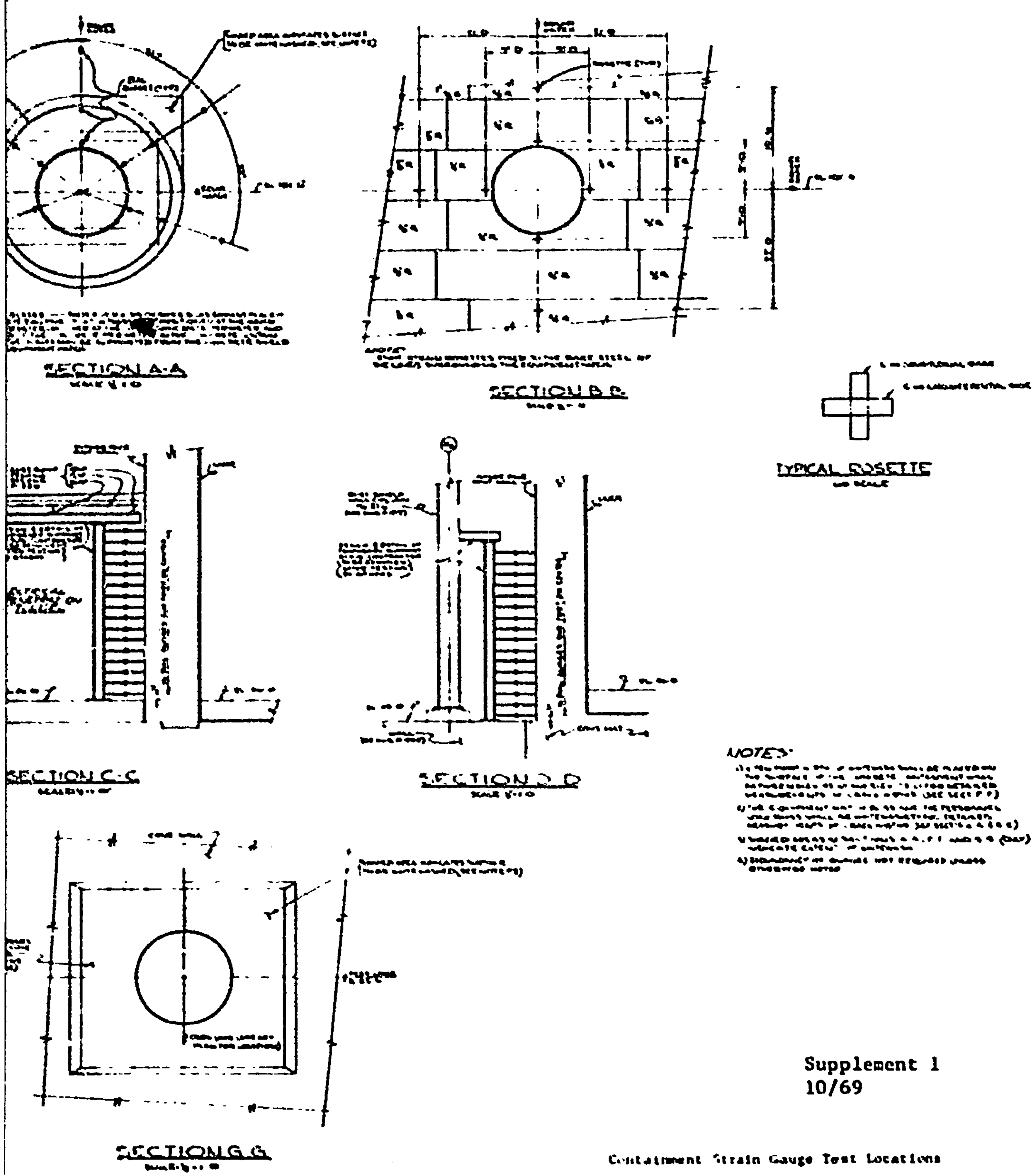
Containment Temporary Opening in N. W. Quadrant
Strain Gauge Test Locations

FIGURE 5.13 – 2


KEY PLAN
SECTION F·F



Supplement 1
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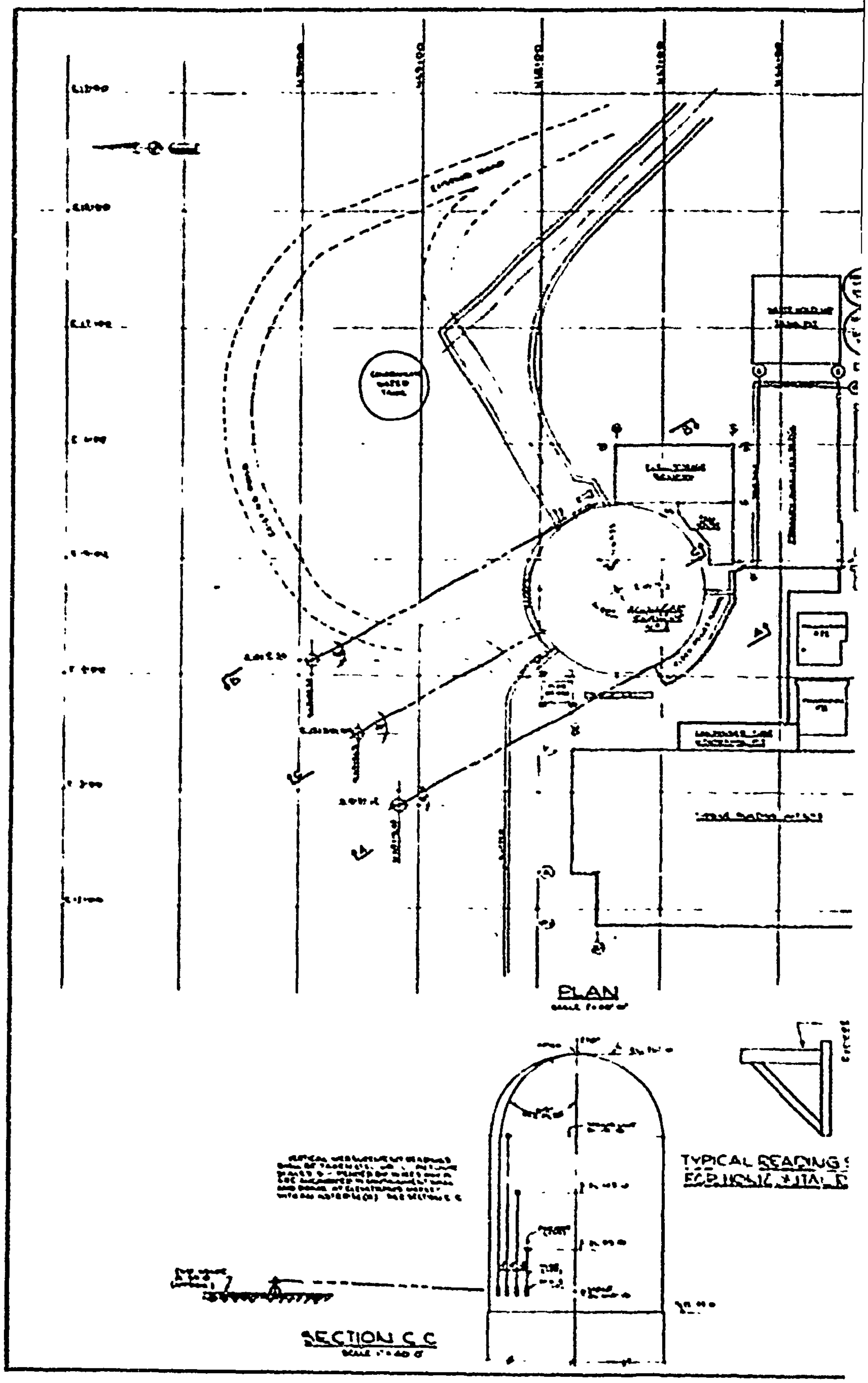
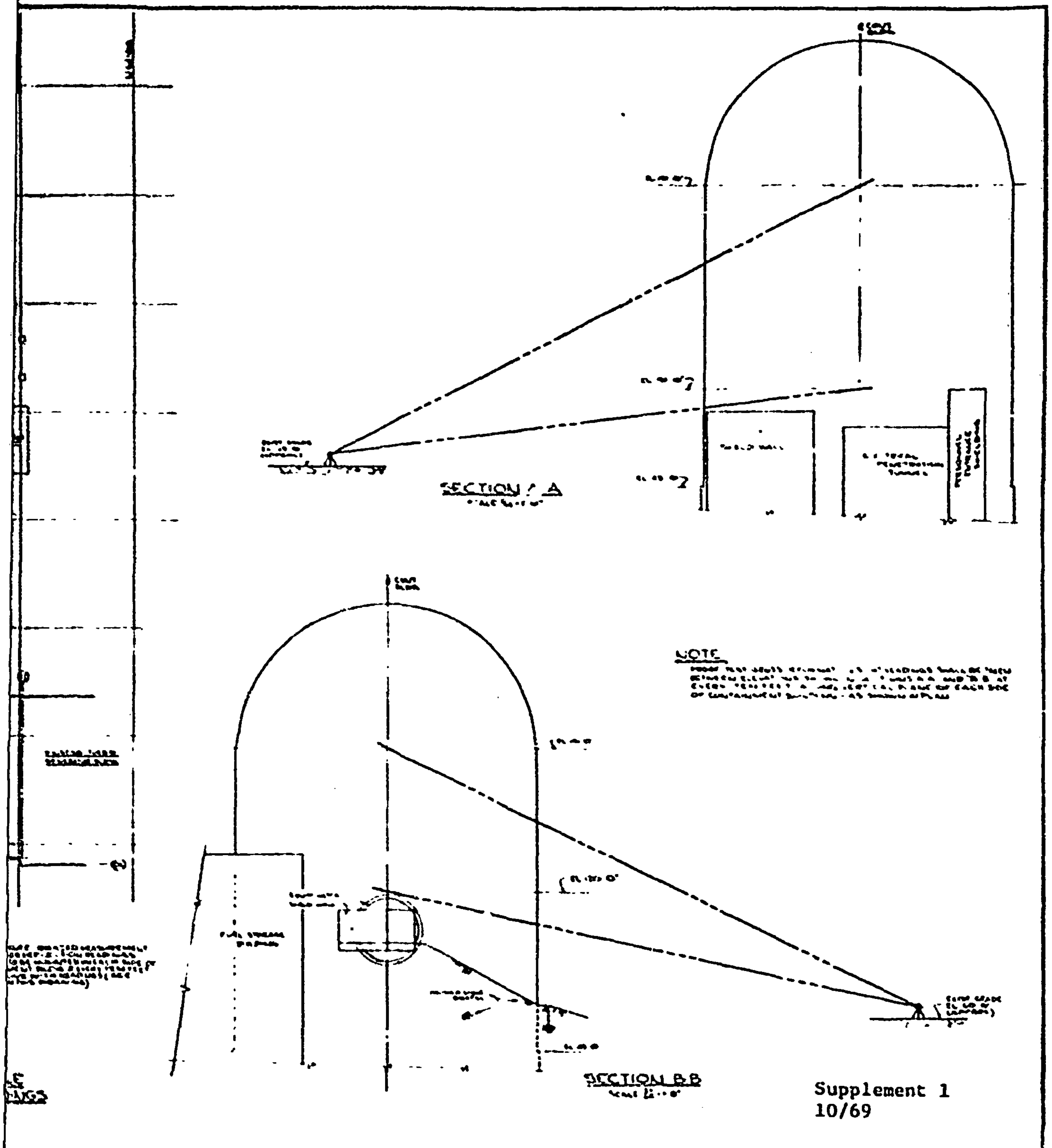
Containment Strain Gauge Test Locations

FIGURE 5.13-3


PLAN
TYPICAL READING
SECTION C C



Supplement 1
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Containment Proof Test Gross Deformation Measurements

FIGURE 5.13 – 4

QUESTION 5.14

Show how the test pressure for the containment proof test demonstrates structural adequacy of the containment for design basis accident loads and for concurrent design basis accident with the design basis earthquake loads. Include the following:

a) Thermal stresses at large openings, evaluation of temperature gradients, stress computations for concrete and reinforcing steel, methods of combining stresses due to normal, tangential, bending, and torsional load, assumptions on cracking, stresses in stirrups, etc.

ANSWER

See answer to Question 7.3a of Supplement 4 to the PSAR for Indian Point No. 3 and Section 3.4 of the Containment Design Report.

QUESTION 5.14

Show how the test pressure for the containment proof test demonstrates structural adequacy of the containment for design basis accident loads and for concurrent design basis accident with the design basis earthquake loads. Include the following:

b) Influence of shrinkage.

ANSWER

See answer to Question 7.3b of Supplement 4 to the PSAR for Indian Point No. 3.

QUESTION 5.14

Show how the test pressure for the containment proof test demonstrates structural adequacy of the containment for design basis accident loads and for concurrent design basis accident with the design basis earthquake loads. Include the following:

c) Influence of liner elastic and plastic deformations.

ANSWER

A finite element analysis of the equipment hatch area indicated local liner plastic deformations during the pressure test. For the order of magnitude and location of these stresses, see Section 3.4.0 of the Containment Design Report for Indian Point No. 2. These deformations will have no influence on the structure during the pressure test due to the ductility of the studs and liner plate.

The limiting elastic liner deformations during test pressure will be from tensile stresses. During an accident loading they will be from compressive stresses. Therefore, a relationship between the pressure and accident loads cannot be determined directly. However, the test pressure will demonstrate the ductile behavior of the liner.

QUESTION 5.14

Show how the test pressure for the containment proof test demonstrates structural adequacy of the containment for design basis accident loads and for concurrent design basis accident with the design basis earthquake loads. Including the following:

d) Liner stresses before cracking of concrete occurs.

ANSWER

See answer to Question 7.3e of Supplement 4 to the PSAR for Indian Point No. 3.

QUESTION 5.14

Show how the test pressure for the containment proof test demonstrates stru ctural adequacy of the containment for design basis accident loads and fo: concurrent design basis accident with the design basis earthquake loads. Include the following:

e) Influence of transient thermal gradients.

ANSWER

See answer to Question 7.3f of Supplement 4 to the PSAR for Indian Point No. 3.

QUESTION 5.15

Describe the surveillance capabilities provided by the containment design with reference to both periodic inspection of the steel liner and periodic structural testing of the containment. If the leak rate testing is intended to be performed at reduced pressure, provide an evaluation of the minimum level of such tests that would also serve to verify continued structural integrity. Consider in the evaluation structural response and surveillance instrumentation requirements.

ANSWER

See answer to Question 7.4 in Supplement 4 of Unit No. 3, PSAR.

QUESTION 5.16

The reactor pressure vessel is enclosed by the vessel cavity. This cavity incorporates the structural support for the vessel and provides missile shielding against the highly unlikely failure of the reactor vessel.

5.16.1 Present and discuss the structural design provisions for the cavity as they relate to potential pressure vessel failure.

ANSWER

The reactor pressure vessel is enclosed by a 6'-0 thick circular reinforced concrete Shield Wall which is designed to sustain the internal pressure and provide missile protection for the Containment and Liner in the highly unlikely failure of the reactor vessel due to a longitudinal split. All stresses will be maintained within specified minimum ultimate rebar tensile stress.

In the event of a circumferential reactor break the 1/4" base mat liner plate at the bottom of the Containment Reactor Cavity Pit directly under the reactor vessel is protected by 2'-0 of concrete with a 1" steel liner plate embedded on top of the concrete. Below the containment base mat liner plate is 4 1/2 feet of concrete poured on rock. (See Figure 5.16, sheets 1 through 6).

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QUESTION 5.16

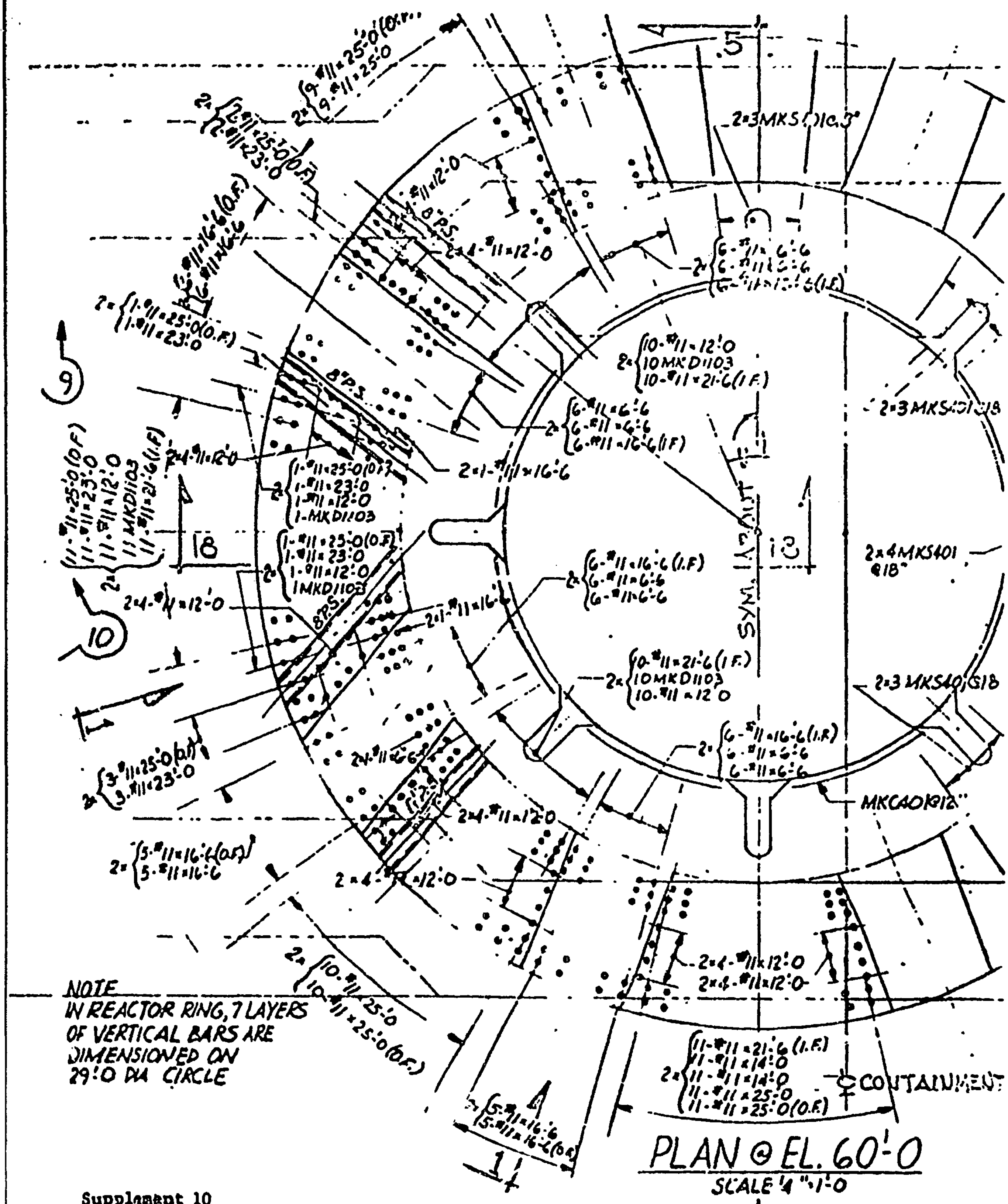
The reactor pressure vessel is enclosed by the vessel cavity. This cavity incorporates the structural support for the vessel and provides missile shielding against the highly unlikely failure of the reactor vessel.

5.16.1 Present and discuss the structural design provisions for the cavity as they relate to potential pressure vessel failure.

ANSWER

The reactor pressure vessel is enclosed by a 6'-0 thick circular reinforced concrete Shield Wall which is designed to sustain the internal pressure and provide missile protection for the Containment and Liner in the highly unlikely failure of the reactor vessel due to a longitudinal split. All stresses will be maintained within specified minimum ultimate rebar tensile stress. 10

In the event of a circumferential reactor break the 1/4" base mat liner plate at the bottom of the Containment Reactor Cavity Pit directly under the reactor vessel is protected by 2'-0 of concrete with a 1" steel liner plate embedded on top of the concrete. Below the containment base mat liner plate is 4 1/2 feet of concrete poured on rock. (See Figure 5.16, sheets 1 through 6).



NOTE
IN REACTOR RING, 7 LAYERS OF VERTICAL BARS ARE DIMENSIONED ON 29'-0 DIA CIRCLE

PLAN @ EL. 60'-0
SCALE ¼"=1'-0

FIG. 5.16
SHEET-1

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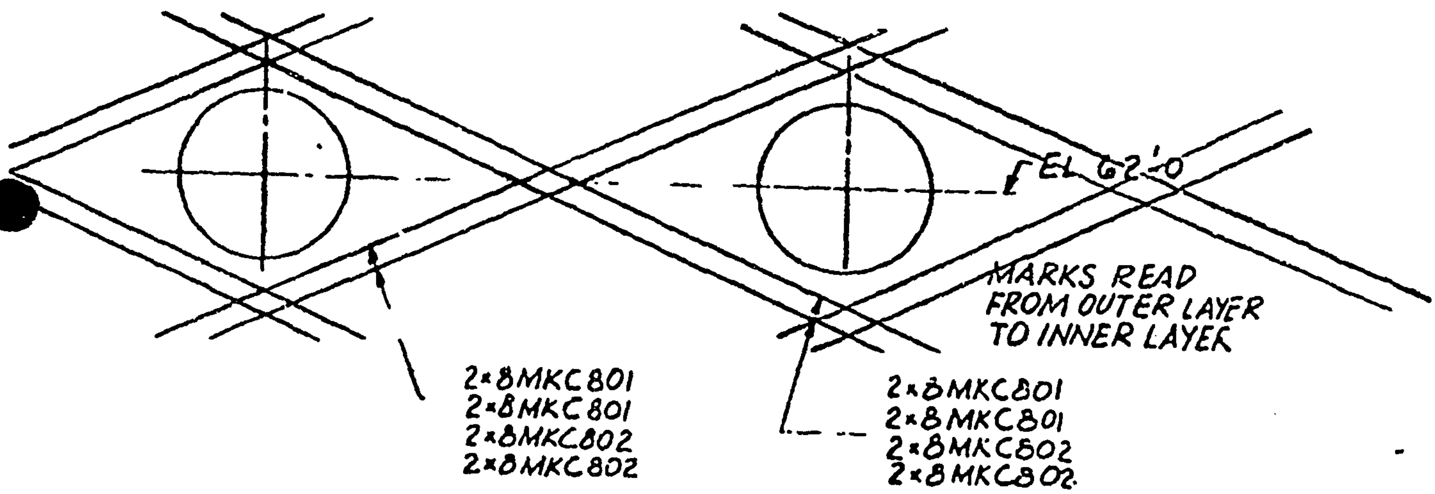

TYP. LAYER @ REACTOR RING

SCALE ¼" = 1'-0

SECTION 1-1

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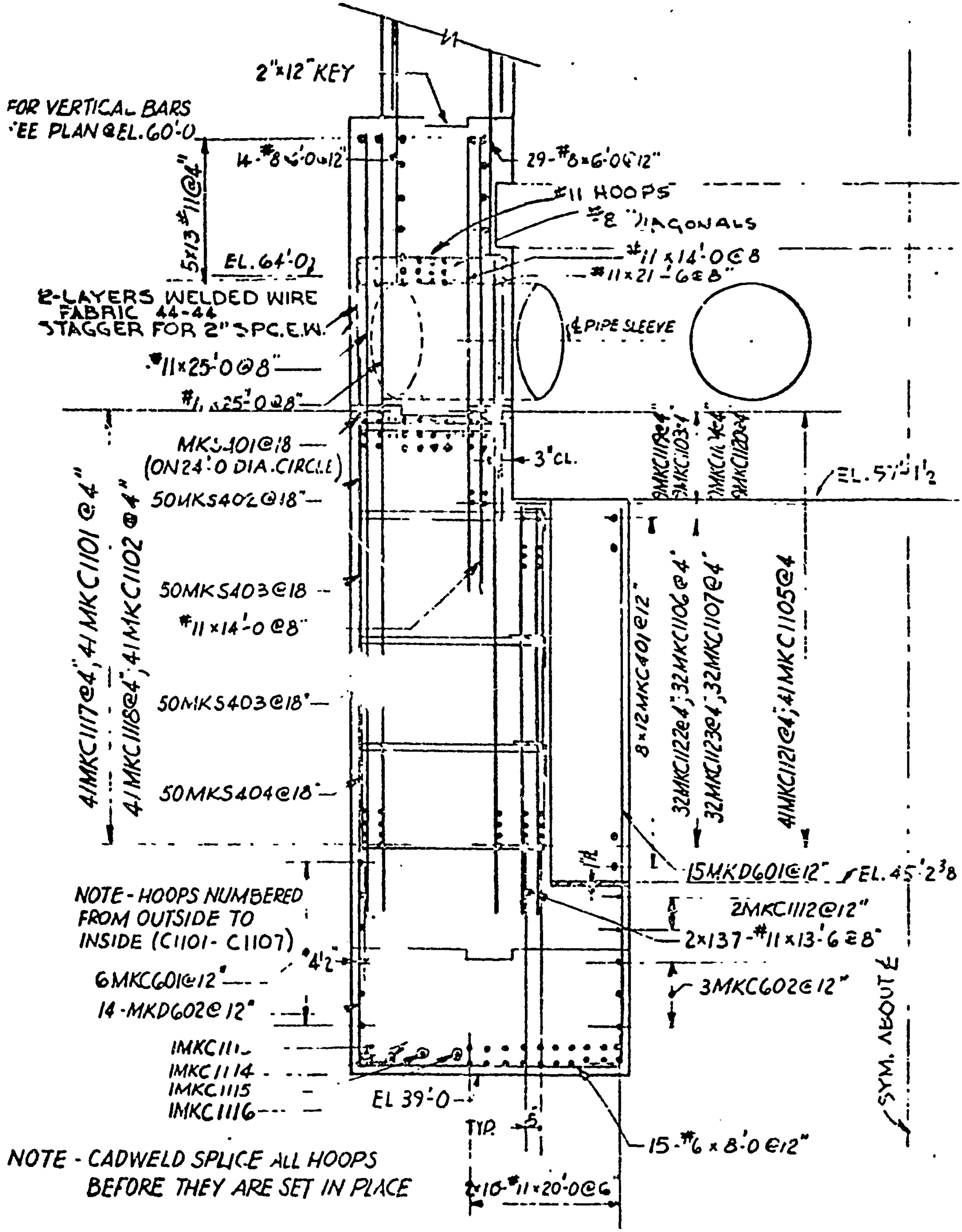
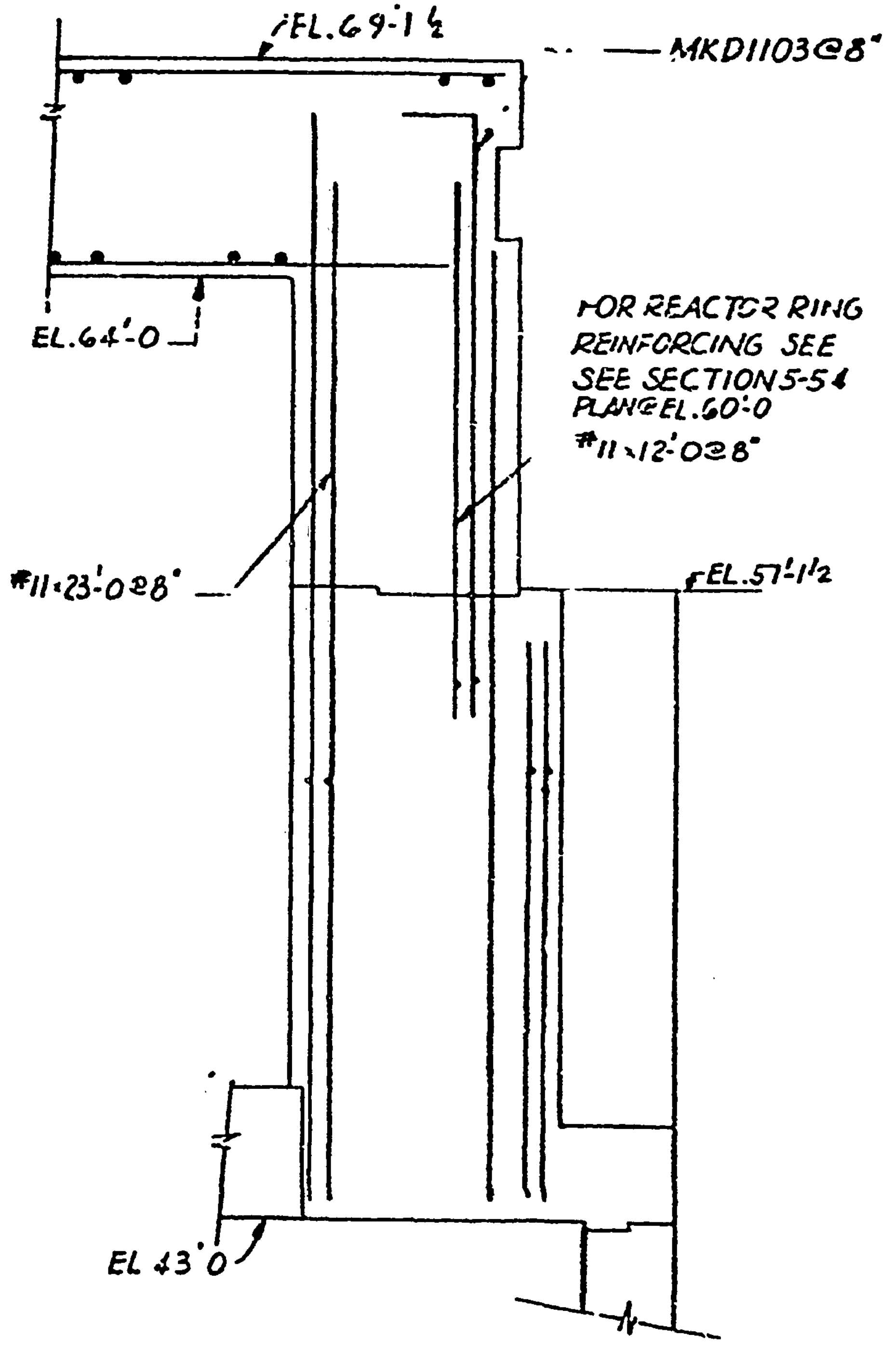
FIG. 5.1(
SHEET -



SECTION 5-5

FIG. 5.16
SHEET #3



SECTION 18-18

Supplement 7

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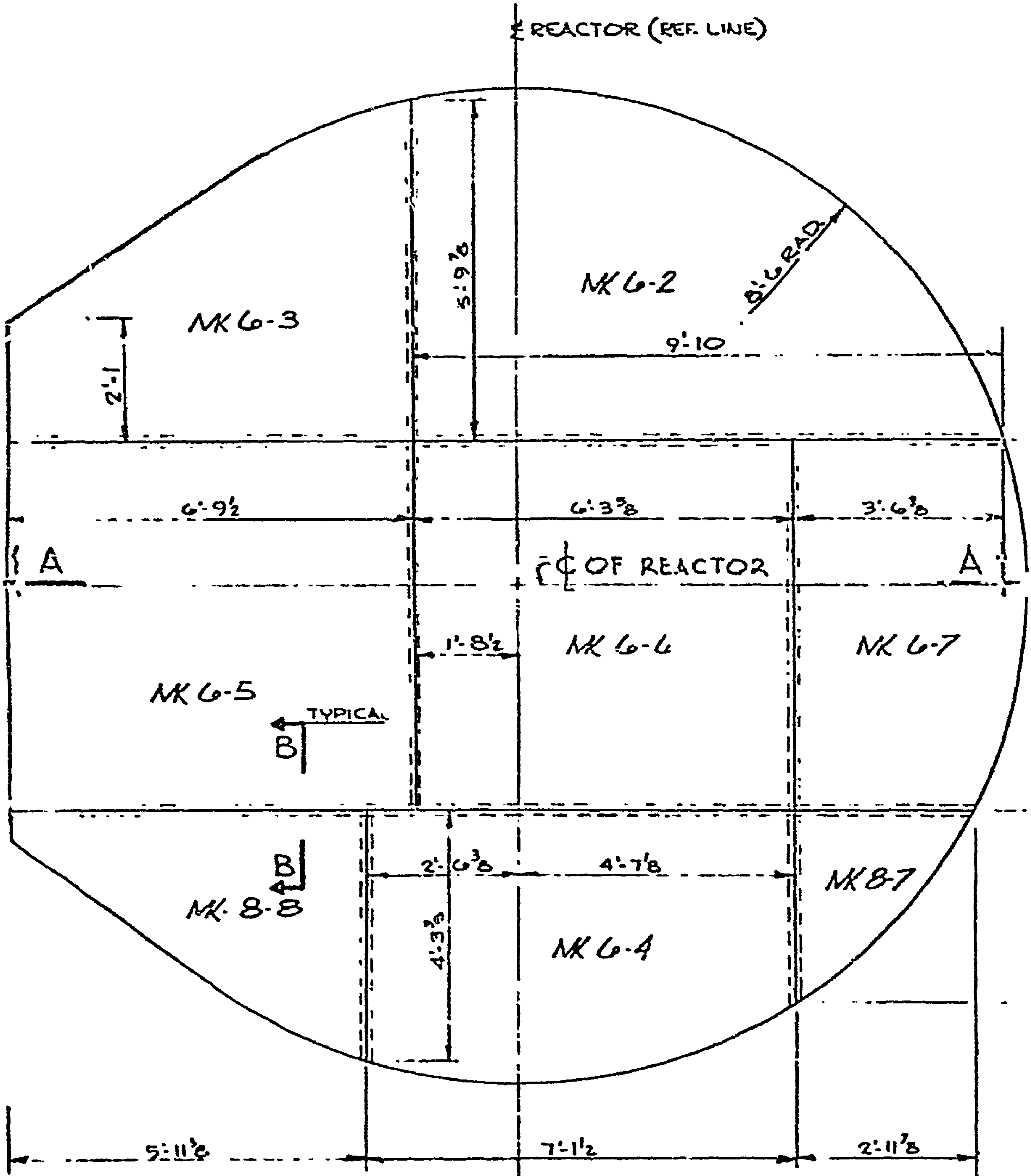
FIG. 5.16
SHEET #4



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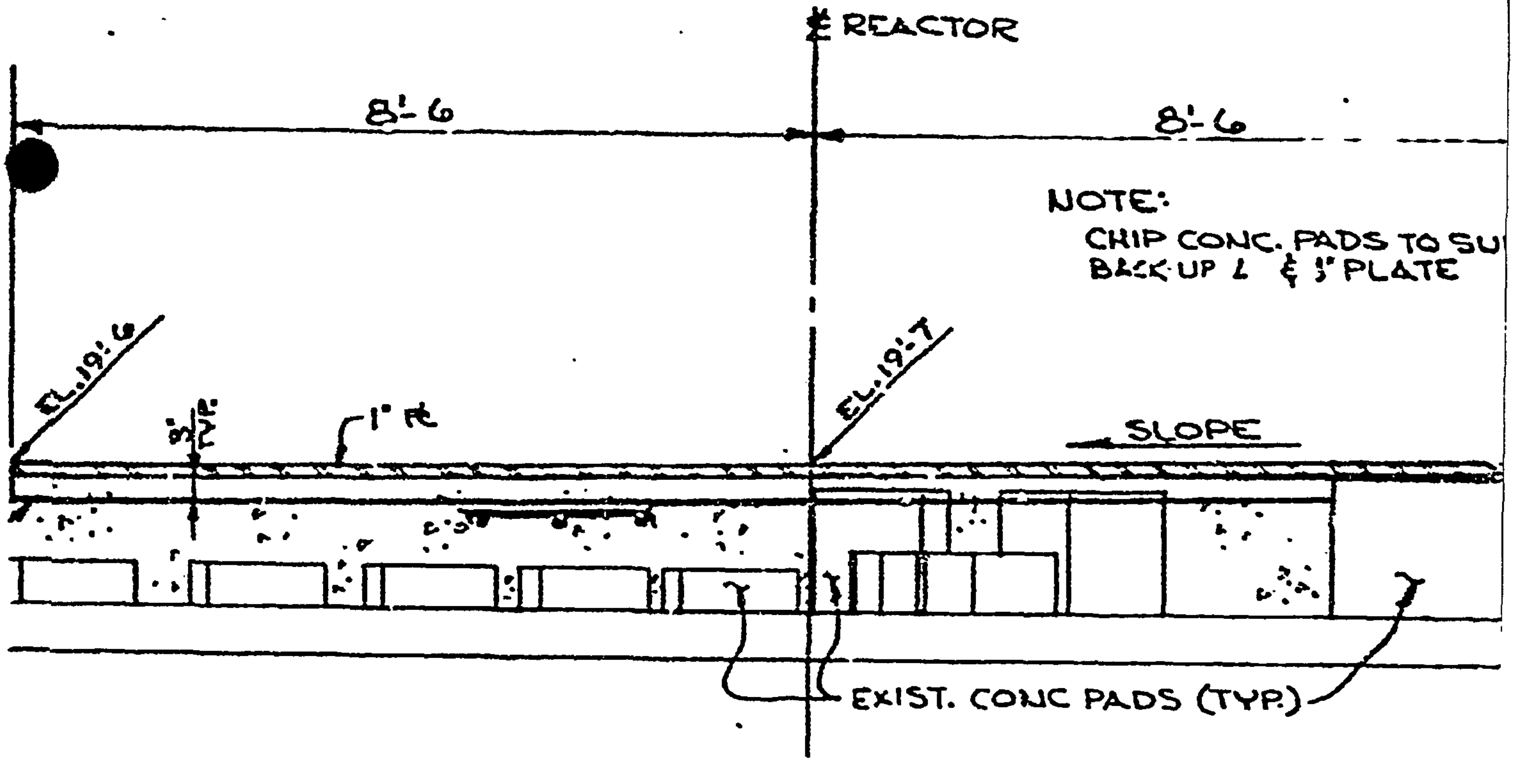
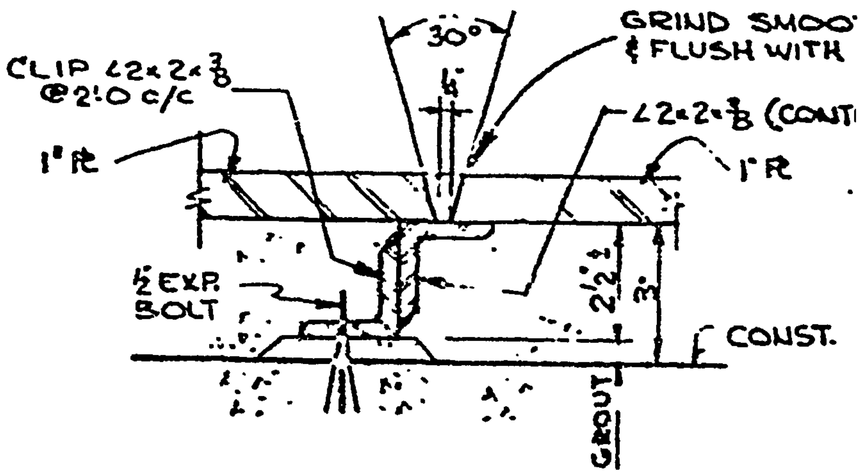
PLAN AT EL. 19'-7

FIG. 5.16
SHEET 5



SECTION A-A



SECTION B-B

Supplement 7
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FIG. 5.16
SHEET 6

QUESTION 5.16

The reactor pressure vessel is enclosed by the vessel cavity. This cavity incorporates the structural support for the vessel and provides missile shielding against the highly unlikely failure of the reactor vessel.

5.16.2 Discuss the ability of the cavity to provide missile protection for the containment structure and liner in the event of reactor vessel failure by longitudinal splitting or various modes of circumferential cracking.

ANSWER

Circumferential Cracking

The worst circumferential crack location from the standpoint of downward missiles is just below the RCS piping nozzles. As the following calculations show, this missile will not violate the containment structure and liner integrity.

As a consequence of this circumferential crack, the downward missile represented by bottom vessel head has the following characteristics at the time of impact on the cavity floor:

1. Weight: 381,000 lb.
2. Cross sectional area of crater: 63 ft^2.
3. Downward velocity: 213 ft/sec.
4. Concrete crushing strength: 4,000 psi.

The depth of penetration has been calculated by using the Petri formula for penetration into an infinite thick concrete slab, as reported in Nav. Docket P-51:

$$D = k \frac{W}{A} \log_{10} \left(1 + \frac{V^2}{215{,}000}\right)$$

where:

D = depth of penetration, ft

K = penetration coefficient for 4,000 psi concrete

W = missile weight, lb.

A = missile area, ft^2

V = missile velocity, lb/sec

The following parameters have been used:

K = 2.8×10^{-3}

W = 381,000 lb.

A = 63 ft^2

V = 213 ft/sec.

The result is a depth of penetration of 1.4 ft.

As mentioned in the answer to Q 5.16.1, the 1/4" base mat liner is covered by 2'-0" of concrete with, on top, a 1" steel plate. As it can be readily seen, even neglecting the 1" steel plate in the penetration calculations the liner will not be reached.

Longitudinal Splitting

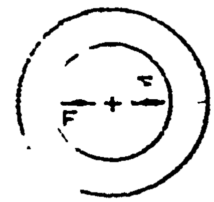
The cavity wall is designed to withstand the forces and internal pressurization associated with a longitudinal split without gross damage.

The assumed accident condition is a longitudinal split of the cylindrical part of the reactor vessel (i.e. 24.4 ft long) having an average width of 1.0 ft.

As the result of this assumed accident, the following two loading cases have been considered in the analysis.

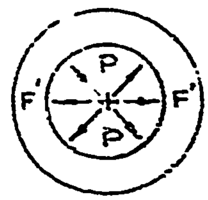
Load Condition 1:

Load on cavity walls at the instant of vessel rupture.



F = 650 k/ft equivalent static line load at the instant of vessel rupture applied as shown in sketch based on a dynamic load factor of 2 applied to the subcooled pressure of 2250 psi times the average width of the break.

Load Condition 2:



Load on cavity walls as shown in sketch.

F = 190 k/ft equivalent static line load
p = 600 psi equivalent static pressure

Line load based on saturated pressure of 1300 psi times the average width of the break and pressure load based on energy release and vent area available. The maximum stress level in the rebar under these loading conditions is to be limited to the ultimate strength of the rebar. For load condition I and load condition II maximum rebar stresses, assuming the concrete to be cracked, are 69.5 ksi and 75.5 ksi, respectively. The rebar used is ASTM A 432 with specified yield of 60 ksi and ultimate tensile strength of 90 ksi.

QUESTION 5.16

The reactor pressure vessel is enclosed by the vessel cavity. This cavity incorporates the structural support for the vessel and provides missile shielding against the highly unlikely failure of the reactor vessel.

5.16.3 Discuss the ability of the cavity to sustain the internal pressure in the event of reactor vessel failure without jeopardizing the integrity of the vessel support.

ANSWER

The criterion on which this question is believed to be based was not a criterion when the design was made final. Nevertheless, in the event of reactor vessel failure a pressure build up of 600 psi inside the pit due to release of reactor contents is assumed. The Shield Wall Analysis shows rebar stresses of 52 ksi when assuming all concrete is cracked. Since the integrity of the wall is not jeopardized the integrity of the vessel support which is supported on the wall will not be jeopardized. Deflection of the Shield Wall will not cause large stresses in the vessel support since a lubricated surface is provided on the shoes, allowing the vessel support to slide.

QUESTION 6.1

Provide a detailed description of the chemical additive spray system as constructed, including system parameters essential for performance evaluation. Describe the method of addition of sodium hydroxide. Describe the possibility for iodine re-evolution and the ultimate effect on the iodine reduction factor.

ANSWER

Section 6.3 of the FSAR has been revised to reflect the eduction method of getting the sodium hydroxide solution into the spray. Re-evolution of iodine from the containment sump is dependent on the iodine concentration gradient between the sump liquid and the containment atmosphere. The relationship between these concentrations is described by the iodine partition coefficient $\left(\frac{\text{concentr. in liquid}}{\text{concentr. in gas}}\right)$.

For the range of interest, the partition coefficient is a function of the iodine concentration and the pH of the solution.*

Figure Q 6.1-1 shows the variation of the sump pH with time. If it is assumed that the iodine concentration remains constant, the partition coefficient will be a function of the sump pH. The variation of the iodine partition coefficient, calculated with the iodine concentration held constant at its peak value, is indicated in Figure Q 6.1-2, which shows the iodine partition coefficient vs. time. No credit is taken for the iodate formation in this calculation.

It can be seen from Figure Q 6.1-2, that the effect of re-evolution of iodine from the sump would have a negligible effect on the iodine removal constant.

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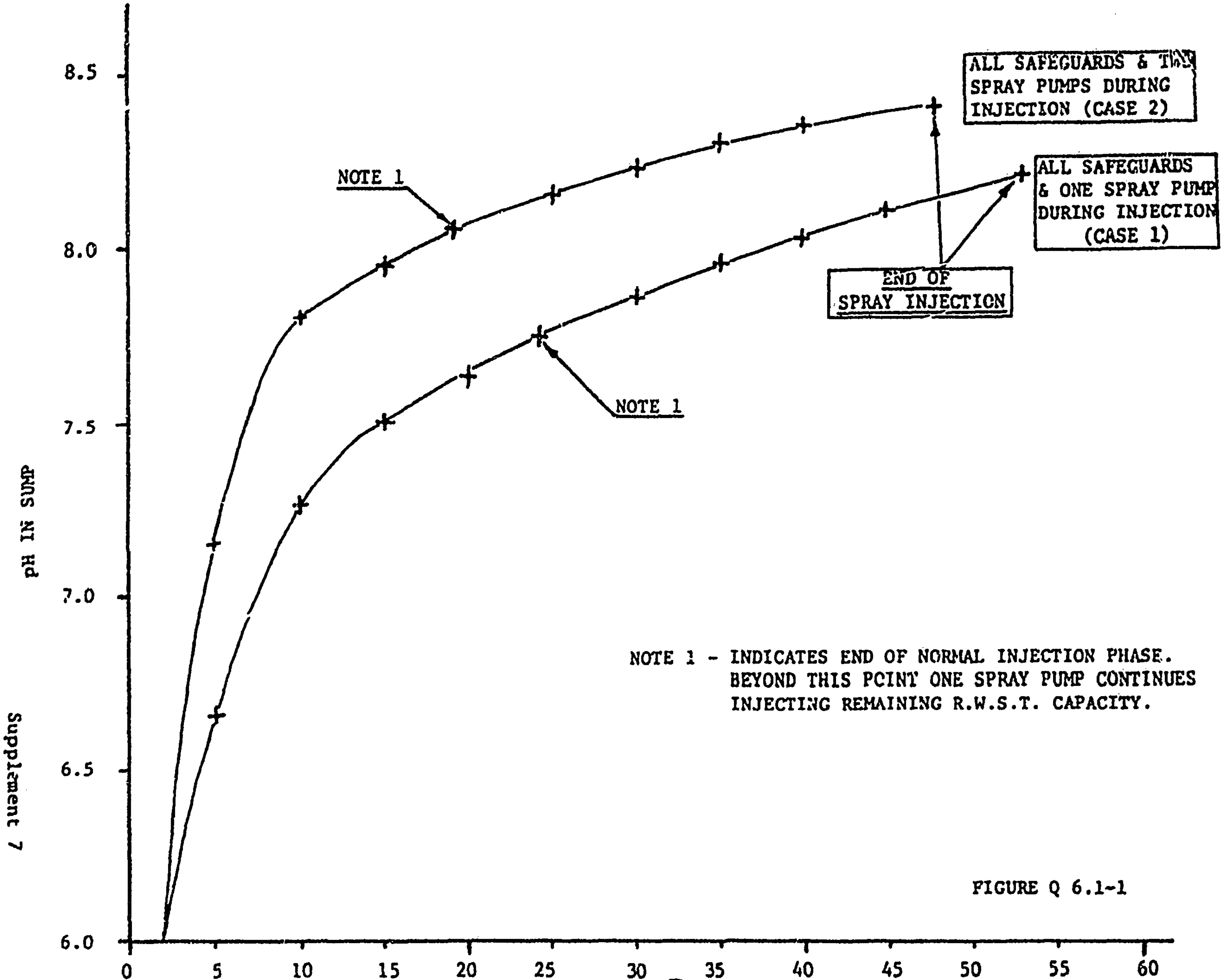
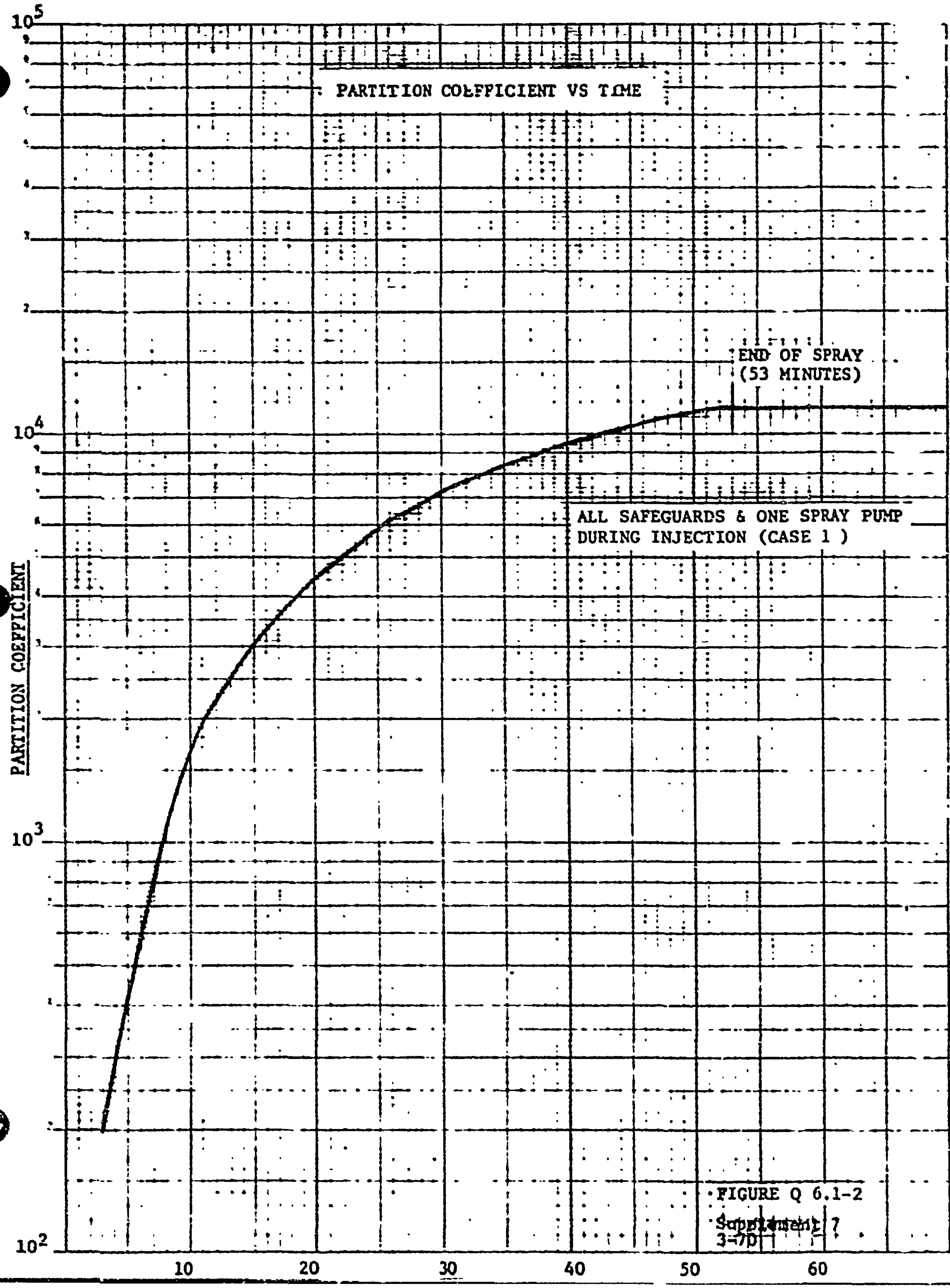
* Data for the iodine partition coefficient are taken from Eggleton, AERE-R 4887, at 212°F.

SUMP pH VS TIME AFTER LOCA



NOTE 1 - INDICATES END OF NORMAL INJECTION PHASE. BEYOND THIS POINT ONE SPRAY PUMP CONTINUES INJECTING REMAINING R.W.S.T. CAPACITY.

FIGURE Q 6.1-1



PARTITION COEFFICIENT VS TIME
END OF SPRAY
(53 MINUTES)
ALL SAFEGUARDS & ONE SPRAY PUMP
DURING INJECTION (CASE 1)
PARTITION COEFFICIENT
10^5
10^4
10^3
10^2
10
20
30
40
50
60

FIGURE Q 6.1-2

QUESTION 6.2

Evaluate the error limits in the calculational model for predicting the effectiveness of the chemical additive spray system for iodine removal. Specify conservative values for the average drop size, deposition velocity, building volume covered by sprays and mixing velocity of residual volume, minimum fall distance, and effective residence time.

ANSWER

The effectiveness of the spray system for iodine removal in the Indian Point Unit #2 plant is calculated by the single drop model described in the Appendix of Chapter 6 of the FSAR.

This model is considered a conservative one [1], [2]. More detailed investigations [3], as well as experimental data [1], [4] have shown that values calculated with this model underpredict the iodine removal constants with a considerable margin.

The principal parameters in the model are:

Average drop diameter ($\bar{d}$)
Drop fall velocity (V_T)
Spray flow rate (F)
Containment Volume (V_C)
Mass transfer deposition velocity (V_G)

In order to evaluate the error limits of the calculational model employed the uncertainty in each of the above parameters will be investigated:

1. Average Drop Diameter

The largest uncertainty of the single drop model lies in the average drop diameter. This is primarily due to two factors: The drop size distribution, and the change in distribution due to coalescence.

Other parameters influencing the drop size are the nozzle pressure, containment temperature and pressure, and condensation on the drops. However, these effects can be predicted accurately by the theory or from existing data.

a) Drop Size Distribution

In our model the spectrum of drop sizes emerging from the spray nozzles is represented by a single droplet diameter, which, when extrapolated to the total flow rate, must give the same result as that obtained from the complete spectrum.

Several averaging methods may be used to obtain a drop size representative of the entire distribution:

the number mean diameter
the mass (or volume) mean diameter
the surface mean diameter
the Sauter (surface per unit volume) mean diameter

Since the absorption of iodine by the spary droplets is a surface dependent process, only the surface mean, and Sauter mean diameters were considered. Of these the Sauter mean gives the more conservative results.

Measurements of the drop size distribution of the Sprayco 1713 Nozzle with various nozzle pressures have shown that the drop size varies between 200 and 2400 microns, with the maximum number of droplets in the 500 to 800 micron range. The surface per unit volume average diameters for these distributions (Sauter mean) are reported in the FSAR. To further analyze the effect of the actual distribution, the drops were divided into 10 groups; and the iodine removal constant λ was calculated by application of equation (4) to each group.

The resulting λ of 46.3 $hrs.^{-1}$ is higher than that calculated by considering only one group because the effect of the smaller droplets, having a higher surface to volume ratio, improved mass transfer characteristics, and longer residence time. If this calculated removal constant is substituted into the single drop size model, the equivalent drop diameter would be 950 microns. (Not considering the effects of coalescence and condensation described below)

b) Coalescence

Collisions between droplets may effectively change the droplet size distribution (i.e. mean diameter). A conservative estimate of the change in the size distribution due to coalescence was obtained by considering collisions between the groups of droplets. Results calculated in this way have confirmed those reported by Parsley [1], resulting in a reduction in λ due to this effect by less than 10 to 20%.

c) Effect of Nozzle Pressure

The drop size distribution varies with the nozzle pressure drop such that the maxima of the distribution are shifted towards the smaller diameters for increased pressure drops. This results in a smaller mean diameter, and hence in an improved λ.

The pressure drop across the nozzles will exceed a minimum of 40 psi during all phases of operation of the spray system. This conservative value of 40 psi has been used throughout the calculation.

d) Condensation

Since the spray droplets enter the containment at temperatures far below that of the initial temperature of the containment atmosphere, condensation of steam from the containment air-steam mixture will increase the initial size of the drops, until they are in thermal equilibrium with the ambient.

From an energy balance on the drop:

$$m\,h + m_{(cond.)}h_g = m'h_f \qquad (1)$$

where m = mass of drop before condensation

m' = mass of drop after condensation

h = enthalpy at inlet temperature

h_f = saturation liquid enthalpy at containment condition

h_g = saturation vapor enthalpy at containment condition

therefore, the mass of the new drop is

$$m' = \frac{4\pi r^3}{3v} + \left(\frac{h_f - h}{h_{fg}}\right)\frac{4\pi r^3}{3v}$$

where v = specific volume at inlet conditions

and the increase in the drop diameter due to condensation is:

$$\frac{d'}{d} = \sqrt[3]{\left(\frac{v}{v_f}\right)\left(\frac{h_g - h}{h_{fg}}\right)} \qquad (2)$$

$\frac{d'}{d}$ = 1.08, or an 8% increase in the mean diameter.

Taking the effects of coalescense and condensation into consideration, the value for the mean droplet diameter for the single drop size model is

d = 1184 microns

2. Deposition Velocity

A basic assumption of the FSAR model is that the mass transfer process is gas-film controlled. This condition has been predicted by theoretical considerations [6], and is supported by substantial experimental evidence [4], [7].

As an additional measure of conservatism in the model, let us assume that there is a detectable liquid phase mass transfer resistance.

Then the total deposition velocity is calculated from the well known equation

$$\frac{1}{V_T} = \frac{1}{V_G} + \frac{1}{HV_L} \qquad (3)$$

From reference [3], the gas-film deposition velocity, Vg, for the containment conditions of 47 psig and 271°F, is:

$$V_G = 7.1 \text{ cm/sec}$$

The iodine partition coefficient, from reference [5], (for a minimum recirculation phase pH of 8.2, and an I_2 concentration of 5×10^{-3} gr./ℓ) is:

$$H = 1 \times 10^5$$

For a liquid film resistance of 5×10^{-3}, the total deposition velocity would be:

$$V_T = \frac{1}{7.1} + \frac{1}{(10^5)\ (5 \times 10^{-3})}$$

$$V_T = 7.0$$

3. Containment Volume

The spray system nozzle and header arrangement is des..ned to cover a maximum area in the upper containment. Four headers, arranged in concentric circles are located in the containment dome at elevations of 213.5, 2.8.6, 223.6, and 228.6 ft.

Even though the spray drops may fall to a lower elevation in some areas, 95 ft. (i.e. the operating deck) is used as the lower limit of the spray fall height, giving a total volume covered of $1.41 \times 10^6 ft^3$, or 54% of the containment free volume.

In the "minimum safeguards" case only two of the four headers would be operating (headers 1 and 3 or headers 2 and 4). Headers 2 and 3 (the 2 inner rings) are provided with nozzles at a 45° angle, to cover the are. of the faulty headers in the minimum safeguards case. Thus only a 1% reduction in the sprayed volume occurs in this case.

4. Mixing Within the Containment

In the post accident mode, a minimum of three of the five 70,000 CFM fan-coolers are in operation. The post accident ventilation rate, therefore, varies between 210,000 and 350,000 CFM. The ventilation system is designed to provide effective ventilation to all parts of the containment. For this purpose, all fan-cooler units discharge into a ring header, to assure ventilation of all parts of the containment, independent of the number of units operating.

Suction to the fan-cooler units is taken from the upper (sprayed) portion of the containment. With exception of 20,000 CFM ducted to the containment dome, the discharge from the ring header is distributed to the various compartments below the operating deck, giving a minimum mixing velocity of 190,000 CFM between sprayed and unsprayed portions of the containment.

Mixing within the containment is further aided by the heat removal of the spray system. Because of the cooling effect of the spray the pressure in the upper part of the containment will be reduced faster than that below the operating deck, thereby causing make-up air-steam mixture to flow towards the upper (sprayed) portion of the containment.

5. Drop Fall Distance and Drop Residence Time

As discussed under item 3, the spray headers are located on 4 levels in the containment dome, resulting in a range of minimum fall distances of 118.5, 123.5, 128.5 and 133.5 ft.

When averaged over the total flow from each header, the average fall distance for the two halves of the spray system (header 1-3 and 2-4) is:

$$h_{1,3} = 121.1 \text{ ft}$$

$$h_{2,4} = 124.1 \text{ ft}$$

One of the simplifying assumptions of the model is that the residence time of the drop in the containment atmosphere may be approximated by $h_{1,3}/U_t$, where U_t is the terminal velocity of the drop.

The actual residence time of the drop, however, is considerably longer, since the drops do not leave the nozzle with only a vertical velocity component, but with an additional horizontal component, which causes the droplets to fall along a trajectory which increases the residence time of the droplet. (This fact is further amplified by the 45° nozzles). Thus the use of $h_{1,3}/U_t$, which gives a minimum residence time of 11.9 sec for a 1130 μ drop, adds another measure of conservatism to the model.

Substituting the specified conservative estimate of each of the variables of the single-dropsize model discussed above into the equation for the iodine removal constant λ_s:

$$\lambda_s = 1470 \left(\frac{V_G}{U_t}\right) \left(\frac{Fh}{V_T d}\right) \tag{4}$$

where V_T is the total mass transfer deposition velocity

we obtain

$$\lambda_s = 60.1 \text{ hrs}^{-1}$$

where the flow rate F = 2,500 gpm corresponds to the minimum flow rate at the initial containment pressure peak of 47 psig, and the volume V_T is the actually sprayed volume.

If we use the entire containment volume, instead of the sprayed volume only, the conservative value of λ_s for the Indian Point containment is

$$\lambda_s = 32.6 \text{ hrs}^{-1}$$

In order to evaluate the sensitivity of λ_s to the uncertainties in the variables discussed above, let us assume that each of these variables may be varied independently.

Then the uncertainty interval, ε_{λ_s}, for λ_s is found from the equation:

$$\varepsilon_{\lambda_s}^{\ 2} = \left(\frac{\partial \lambda_s}{\partial V_T}\right)^2 \varepsilon_{V_G}^{\ 2} + \left(\frac{\partial \lambda_s}{\partial \mu_t}\right)^2 \varepsilon_{U_t}^{\ 2} + \left(\frac{\partial \lambda_s}{\partial d}\right)^2 \varepsilon_d^2 \tag{5}$$

where

ε_{V_G} = uncertainty interval in V_G

ε_{U_t} = uncertainty interval in U_t

ε_d = uncertainty interval in d or

$$\left(\frac{\varepsilon_{\lambda_s}}{\lambda_s}\right)^2 = \frac{\varepsilon_{V_T}^{\ 2}}{V_T^{\ 2}} + \frac{\varepsilon_{U_t}^{\ 2}}{U_t^{\ 2}} + \frac{\varepsilon_d^{\ 2}}{d^2} \tag{6}$$

where

$$\varepsilon_{V_T}^{\ 2} = \left(\frac{\partial V_T}{\partial V_G}\right)^2 \varepsilon V_G^{\ 2} + \left(\frac{\partial V_T}{\partial H}\right)^2 \varepsilon_H^{\ 2} + \left(\frac{\partial V_T}{\partial V_L}\right)^2 \varepsilon_{V_L}^{\ 2} \tag{7}$$

From equation (3):

$$\frac{\partial V_T}{\partial V_G} = \frac{V_T^{\ 2}}{V_G^{\ 2}} \tag{8}$$

$$\frac{\partial V_T}{\partial H} = \frac{V_T^{\ 2}}{H^2 V_L} \tag{9}$$

$$\frac{\partial V_T}{\partial V_L} = \frac{V_T^{\ 2}}{H\ V_L^{\ 2}} \tag{10}$$

Then

$$\varepsilon_{V_T}^{\ 2} = \left(\frac{V_T^{\ 2}}{V_G^{\ 2}}\right)^2 \varepsilon_{V_G}^{\ 2} + \left(\frac{V_T^{\ 2}}{H^2 V_L}\right)^2 \varepsilon_H^{\ 2} + \left(\frac{V_T^{\ 2}}{H V_L^{\ 2}}\right)^2 \varepsilon_{V_L}^{\ 2} \tag{11}$$

and

$$\frac{\varepsilon_{\lambda_s}}{\lambda_s} = \left[\left(\frac{V_T}{V_G^{\ 2}}\right)^2 \varepsilon_{V_G}^{\ 2} + \left(\frac{V_T}{H^2 V_L}\right)^2 \varepsilon_H^2 + \left(\frac{V_T}{H V_L^{\ 2}}\right)^2 \varepsilon_{V_L}^{\ 2} + \left(\frac{\varepsilon_{U_t}}{U_t}\right)^2 + \left(\frac{\varepsilon_d}{d}\right)^2\right]^{-1} \tag{12}$$

Equation (12) gives the error in λ_s due to uncertainties in each of the parameters discussed above.

A summary of the results of substituting uncertainty intervals for each of the variables is presented in Table I.

TABLE I

Summary of Uncertainties of Equation (12) and Their Effect on λ_s

Parameter	Symbol	Range	Conservative Value Used	% Variation in λ_s
Drop Diameter	d	950 μ - 1230 μ	1130 μ	+16% -9%
Gas-film deposition velocity	ν_g	$7.1 - 7.0 \frac{cm}{sec}$	$7.0 \frac{cm}{sec}$	+1.4%
Iodine partition coefficient	H	$1.0x10^4-\infty$	$1.0x10^5$	neglegible
Liquid-Film deposition velocity	ν_L	$.005-\infty \frac{cm}{sec}$	$.005 \frac{cm}{sec}$	neglegible
Sprayed containment Volume	V_s	$1.39-1.41\ 10^6 ft^3$	$1.41x10^6 ft^3$	+1%
Total Containment Volume	V_c	$1.39-2.6\ 10^6 ft^3$	$2.6x10^6 ft^3$	+47%
Fall Distance	h	118.5 - 133.5 ft	121.1 ft	+10.2%, -2.2%

References

[1] Row, Parsly, Zittel, Design Considerations of Containment Spray Systems - Part I, ORNL-TM-2412.

[2] Parsly, L. F., Gas Absorption Theory Applied to Containment Sprays, ORNL-TM-2002.

[3] Parsly, L. F., Removal of Elemental Iodine from Steam-Air Atmospheres by Reactive Sprays, ORNL-TM-1911.

[4] Parsly, Franzreb, Removal of Iodine Vapor from Air and Steam-Air Atmospheres in the NSPP by Use of Sprays, ORNL-4253.

[5] Eggleton, A. E. J., A Theroretical Examination of Iodine-Water Partition Coefficients, AERE-R 4887.

[6] Griffith, V., The Removal of Iodine from the Atmosphere, UKEA report AHSB(s) R 45.

[7] Row, T. H., Spray and Pool Absorbtion Technology Program ORNL-4360.

QUESTION 6.3

Provide a detailed listing of all major construction materials which will be exposed to the spray solution, and the corrosion or deterioration rates for each of these under maximum exposure conditions. For each material which will be adversely affected, state approximate quantity and exposed surface areas. Include surface coatings, electrical wiring, air filtration components, etc. Analyze the potential consequences of corrosion and/or deterioration and on all materials with regard to post-accident operation of engineered safety features and potential additional hazards.

ANSWER

1.0 Definition of Post-Accident Containment Environmental Conditions

An evaluation of the suitability of materials of construction for use in the reactor containment system must be performed considering the following:

a. The integrity of the materials of construction of engineered safeguards equipment when exposed to post design basis accident (DBA) conditions, and

b. The effects of corrosion and deterioration products from both engineered safeguards (vital equipment) and other (non-vital) equipment, on the integrity and operability of the engineered safeguards equipment.

Reference post DBA environment conditions of temperature, pressure, radiation and chemical composition are described in the following sections. The time-temperature-pressure cycle used in the materials evaluation is most conservative, since it considers only partial safeguards operation during the DBA. The spray and core cooling solutions, considered herein, include both the design chemical compositions and the design chemical compositions contaminated with deterioration products and fission products, which may conceivably be transferred to the solution during recirculation through the various containment safeguards systems.

1.1 Design Basis Accident Temperature-Pressure Cycle

Figure 14.3.4-2 of the FSAR and Figure 6.3-1 present the temperature-pressure-time relationship following the design basis accident. This figure represents the containment condition for the following safety feature operation. One of the two spray pumps is considered to inject 2500 gpm into the containment. When the refueling water storage tank is empty, the recirculation pumps supply a flow of 2400 gpm to the spray headers. Recirculation flow through one recirculation pump is cooled in the residual heat exchanger.

Figures 6.3-2 and 6.3-3 present materials evaluation test conditions for the containment and core environment, respectively.

Materials evaluations, to be described, were performed, in general, for conditions either simulating the time-temperature conditions of Figure 6.3-2 or conservatively considering higher temperatures for longer periods. The basis for each material evaluation is described with the discussion of its particular suitability.

1.2 Design Basis Accident Radiation Environment

Evaluation of materials for use in containment includes a consideration of the radiation stability requirements for the particular materials application. Figures 6.3-4 and 6.3-5 present the post DBA containment atmosphere direct gamma dose rate and the integrated direct gamma dose respectively. These data were calculated on the basis of a core melt-down and assuming the following fission product fractional releases, consistent with the TID-14844 model:

Noble Gases	Fractional Release	1.0
Halogens	Fractional Release	0.5
Other Isotopes	Fractional Release	0.01

1.3 Design Chemical Composition of the Emergency Core Cooling Solution

The system designs provide for use of alkaline adjusted boric acid solution as the spray and core cooling fluid.

1.3.1 Alkaline Sodium Borate

Plant designs which utilize the spray solution for fission product iodine removal, as well as containment cooling, include provisions for injection of chemical additive (sodium hydroxide) to the emergency core cooling system. Boric acid solution, containing approximately 2,000 ppm B, is pumped from the refueling water storage tank to the containment system by means of the safety injection system pumps, residual heat removal pumps and the spray pumps.

The chemical additive tank contains sufficient sodium hydroxide solution, such that, when its contents, the refueling water storage tank contents, the ice melt, and the reactor coolant system fluid are mixed, the resulting pH will be between 8.5 and 10.0. During the initial 30 to 60 minutes of spraying, the spray solution may be at a pH about 10.

Figure 6.3-6 shows a plot of sodium hydroxide concentration versus pH for a 2500 ppm boron solution. Tentative limits of pH 8.5-10 for the mixed spray solution are indicated on this figure.

For the purpose of materials evaluation in the design chemistry solution, the following concentration/time relationship was considered:

0	→	1 hour	pH	10.0	Boron 2500 ppm
1 hour	→	12 months	pH	9.0	Boron 2500 ppm

The solutions are considered aerated through the entire exposure period.

1.4 Trace Composition of Emergency Core Cooling Solution

During spraying and recirculation, the emergency core cooling solution will wash over virtually all the exposed components and structures in the reactor containment. The ECC solution is recirculated through a common sump and hence, any contamination deposited in or leached by the solution from the exposed components and structures will be uniformly mixed in the solution.

The materials compatibility discussion includes consideration of the effects of trace elements which are identified as conceivably being present in the ECC solution during recirculation.

To identify the trace elements in containment which may have a deleterious effect on engineered safeguards equipment, one must first establish which elements are potentially harmful to the materials of construction of the safeguards equipment and second ascertain the presence of these elements in forms which can be released to the ECC solution following a design basis accident. Table 6.3-1 presents a listing of the major periodic groups of elements. Elements which are known to be harmful to various metals are noted and potential sources of these elements are identified. A discussion of the effects of these elements is presented in latter sections.

Table 6.3-1

Review of Sources of Various Elements in Containment and Their Effects on Materials of Construction

Group	Representative Elements	Corrosivity of Elements	Sources of Elements
0	He, Ne, Kr, Xe	No effect on any materials of construction.	Fission product release.
I a	Li, Na, K	Generally corrosion inhibitive properties for steels, and copper alloys - harmful to aluminum	Li - coolant pH adjusting agent Na - spray additive solutio[n] concrete leach product K - concrete leach product
II a	Mg, Ca, Sr, Ba	Generally not harmful to steel or copper base alloys.	Concrete leach products - deteriorated insulation.
III a	Y, La, Ac	Not considered harmful in low concentrations.	Fission product release.
IV a	Ti, Zr, Hf	Not considered harmful to any materials.	Fuel rod cladding, control rod material, alloying constituent.
V a	V, Nb, Ta	Not considered harmful to any materials.	Alloying constituents in low concentration.
VI a	Cr, Mo, W	Not considered harmful to any materials.	Alloying constituents in equipment.
VII a	Mn, Tc, Re	Not considered harmful.	Mn - alloy constituent.
VIII	Fe, Ni, Cr, Os,	Fe, Ni, Cr - not harmful to any materials.	Fe, Ni, Cr - alloying constituents. Others have no identifiable sources.

(Continued)

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Table 6.3-1 (Continued)

Group	Representative Elements	Corrosivity of Elements	Sources of Elements
I b	Cu, Ag, Au	Not harmful to any materials.	Cu present as material of construction and alloying constituent.
II b	Zn, Cd, Hg	Hg - harmful to stainless steel, Cu alloys, aluminum Zn - unknown Cd - unknown	Hg has been entirely excluded from use in the containment. Cd finish plating on components. Zn galvanizing and alloying constituent.
III b	B, Al, Ga, In	Not harmful to material.	B - neutron poison additive Al - materials of construction
IV b	C, Si, Sn, Pb	C, Si, Sn not harmful to materials Pb considered harmful to nickel alloys	Si - concrete leach product Pb - alloy constituent in some brazes
V b	N, P, As, Sb, Bi	No effect from N unless ammonia is formed. Others unknown.	N - containment air. Others not identified in significant materials.
VI b	O, S, Se, Te	S possibly harmful to nickel alloys	Te - fission product S - oils, greases, insulating materials
VII b	F, Cl, Br, I	F, considered potentially harmful to Zircaloy. Cl, potentially harmful to stainless steel. Br and I, not generally harmful	Cl - concrete leach product, general contamination F - organic materials I and Br - fission products, low concentration.

2.0 Materials of Construction in Containment

All materials in containment are reviewed from the standpoint of insuring the integrity of equipment of which they are constructed and to insure that deterioration products of some materials do not aggravate the accident condition. In essence, therefore, all materials of construction in containment must exhibit resistance to the post-accident environment or, at worst, contribute only insignificant quantities of trace contaminants which have been identified as potentially harmful to vital safeguards equipment.

Table 6.3-2 lists typical materials of construction used in the reactor containment system. Examples of equipment containing these materials are included in the table.

Corrosion testing, described in Section 3.0, showed that of all the metals tested only aluminum alloys were found incompatible with the alkaline sodium borate solutions. Aluminum was observed to corrode at a significant rate, with the generation of hydrogen gas. Since hydrogen generation can be hazar s to containment integrity a detailed survey was conducted to identify all aluminum components in containment.

Table 6.3-3 lists the nuclear steam suppl tem aluminum inventory which is considered present in the reactor containment. Included in the table is the mass of metal and exposed surface area of each component. The 1100 and the 6000 series aluminum alloys are generally the major types found in containment. This inventory reflects the determination to exclude as much as practicable the use of aluminum in the containment.

Table 6.3-2

Materials of Construction in Reactor Containment

Material	Equipment Application
300 Series Stainless Steel	Reactor coolant system, residual heat removal loop, spray system
400 Series Stainless Steel	Valve materials
Inconel (600, 718)	Steam generator tubing, reactor vessel nozzles, core supports, and fuel rod grids
Galvanized Steel	Ventilation duct work, CRDM shroud material, I & C conduit
Aluminum	Nuclear detectors, I & C equipment, CRDM connectors, paints
Copper	Service water piping, fan cooler material
70-30 Cu Ni	Fan cooler material
90-10 Cu Ni	Fan cooler material
Carbon Steel	Component cooling loop, structural steel, main steam piping, ect.
Monel	Possibly instrument housings
Brass	Possibly instrument housings
Polyvinyl chloride	Conduit sheating, electrical insulation, containment liner insulation
Protective Coatings Inorganic Zincs Epoxy Modified Phenolics	General use on carbon steel structures and equipment, concrete
Silicones - neoprene	Ventilation duct work gasketing, sealants

Table 6.3-3

Inventory of Aluminum in Containment

	Item	Mass (lbs.)	Surface Area (ft^2)
1.	CRDM Connectors	122	42
2.	Reactor Vessel Insulation Foil	269	Very High
3.	Area Monitors	6	4
4.	Source, Intermediate, and Power Range Detectors	140	40
5.	Process I & C	420	84
6.	Lighting Fixtures and Equipment	1061	380
7.	Paint on Steam Generator, Pressurizer and Reactor Vessel	140	Very High
8.	Contingency	250	85

3.0 Corrosion of Metals of Constructions in Design Basis ECC Solution

Emergency core cooling components are austenitic stainless steel and, hence, are quite corrosion resistant to the alkaline sodium borate solution, as demonstrated by corrosion tests performed at Westinghouse and ORNL.[1] The general corrosion rate, for Type 304 and 316 stainless steels was found to be 0.01 mils/month in pH 10 solution at 200°F. Data on corrosion rates of these materials in the alkaline sodium borate solution have also been reported by ORNL[2,3] to confirm the low values.

Extensive testing was also performed on other metals of construction which are found in the reactor containment. Testing was performed on these materials to ascertain their compatibility with the spray solution at design post accident conditions and to evaluate the extent of deterioration product formation, if any, from these materials.

Metals tested included Zircaloy, Inconel, aluminum alloys, cupronickel alloys, carbon steel, galvanized carbon steel and copper. The results of the corrosion testing of these materials are reported in detail in reference 1. Of the materials tested, only aluminum was found to be incompatible with the alkaline sodium borate solution. Aluminum corrosion is discussed in Section 5.0. The following is a summary of the corrosion data obtained on various materials of construction exposed for several weeks in aerated alkaline (pH 9.3 - 10.0) sodium borate solution at 200°F. The exposure condition is considered conservative since the test temperature (200°F) is considerably higher than the long term design basis accident temperature.

Material	Maximum Observed Corrosion Rate mil/month
Carbon Steel	0.003
Zr-4	0.004
Inconel 718	0.003
Copper	0.015
90 - 10 Cu-Ni	0.020
70 - 30 Cu-Ni	0.006
Galvanized Carbon Steel	0.051
Brass	0.010

Tests conducted at ORNL[2,3] also have verified the compatibility of various materials of construction with alkaline sodium borate solution. In tests conducted at 284°F, 212°F, and 130°F stainless steels, Inconel, cupronickels, Monel, and Zircaloy-2 experienced negligible changes in appearance and negligible weight loss.

Corrosion tests at both PWRD and ORNL have shown copper suffers only slight attack when exposed to the alkaline sodium borate solution at DBA conditions. The corrosion rate of copper, for example, in alkaline sodium borate solution at 200°F is ~ 0.015 mil/month.[1] The corrosion of copper in an alkaline sodium borate environment under spray conditions at 284 and 212°F have been reported by ORNL. Corrosion penetrations of less than 0.02 mil was observed after 24 hour exposure at 284°F (see reference 3, Table 3.13) and a corrosion rate of less than 0.3 mil per month was ovserved at 212°C (see reference 2, Table 3.6).

It can be seen therefore that the corrosion of copper in the post accident environment will have a negligible effect on the integrity of the material. Further the corrosion product formed during exposure to the solution appears tightly bound to the metal surface and hence will not be released to the ECC solution.

The corrosion rate of galvanized carbon steel in alkaline sodium borate (3000 ppm B, pH 9.3) is also low. Tests conducted in aerated solutions showed the corrosion rate to be 0.003 mil/month (0.046 mg/dm^2/hr) and 0.002 mil/month (0.036 mg/dm^2/hr) for temperatures of 200°F and 150°F respectively. It can be seen therefore that the corrosion of zinc (galvanized) in alkaline borate solution is minimal and will not contribute significantly to the post accident hydrogen buildup.

Consideration was given to possible caustic corrosion of austenitic steels by the alkaline solution. Data presented by Swandby[4] (Figure 6.3-7) shows that these steels are not subject to caustic stress cracking at the temperature (285°F and below) and caustic concentrations (less than 1 weight percent) of interest. It can be seen from Figure 6.3-7 that the stress cracking boundary minimum temperature as defined by Swandby coincides with a high free caustic concentration (~40%) and is considerably above (~80°F) the long term post accident design temperature. Further, from Figure 6.3-7, a temperature in excess of 500°F is required to produce stress corrosion cracking at sodium hydroxide concentration greater than 85%.

It should be noted when considering the possibility of caustic cracking of stainless steel that the sodium hydroxide - boric acid solution is a buffer mixture wherein no free caustic exists at the temperatures of interest - even should the solution be concentrated locally through evaporation of water and hence the above consideration is somewhat hypothetical with regard to the post accident environment.

4.0 Corrosion of Metals of Construction by Trace Contaminants in ECC Solution

Of the various trace elements which could occur in the emergency core cooling solution in significant quantities, only chlorine (as chloride) and mercury are adjudged potentially harmful to the materials of construction of the safeguards equipment.

4.1 The use of mercury or mercury bearing items, however, is prohibited in containment. This includes mercury vapor lamps, fluorescent lighting and instruments which employ mercury for pressure and temperature measurements and for electrical equipment. Potential sources of mercury therefore, are excluded from containment and hence no hazard from this element is recognized.

4.2 The possibility of chloride stress corr sion of austenitic stainless steels has also been considered. It is believed that corrosion by this mechanism will not be significant during the post accident period for the following reasons:

1. Low Temperature of ECC Solution

The temperature of the ECC solution is reduced after a relatively short period of time (i.e. a few hours) to about 150°F. While the influence of temperature on stress corrosion cracking of stainless steel has not been unequivocally defined, significant laboratory work and field experience indicates that lowering the temperature of the solution decreases the probability of failure. Hoar and Hines[5] observed this trend with austenitic stainless steel in 42 weight percent solutions of $MgCl_2$ with temperature decrease from 310 to 272°F. Staehle and Latanision[6] present data which also shows the decreasing probability of failure with decreasing solution temperature from about 392°F to 302°F. Staehle and Latanision[6] also report the data of Warren[7] which showed the significant change with decrease in temperature from 212°F to 104°F. The work of Warren while pertinent to the present consideration in that it shows the general relationship of temperature to time to failure is not directly applicable in that the chloride concentration (1800 ppm Cl) believed to have effected the failure was far in excess of reasonable chloride contamination which may occur in the ECC solution.

2. <u>Low Chloride Concentration of ECC Solution</u>

It is anticipated that the chloride concentration of the ECC solution during the post accident period will be low. Throughout plant construction, surveillance is maintained to insure that the chloride inventory in containment would be maintained at a minimum. Controls on use of chloride bearing substances in containment include the following:

a. restriction in chloride content of water used in concrete

b. prohibition of use of chloride in cleaning agents for stainless steel components and surfaces

c. prohibition of use of chloride in concrete etching for surface preparation

d. use of non chloride bearing protective coatings in containment

e. restriction of chloride concentration in safety injection solution, 0.15 ppm chloride maximum

The effect of decreasing chloride concentration on decreasing the probability of failure of stressed austenitic stainless steel has been shown by many experimenters. Staehle and Latanision[6] present data of Staehle which shows the decrease in probability of failure with decrease in chloride concentration at 500°F. Edeleanu[8] shows the same trend at chloride concentrations from 40 to 20% as $MgCl_2$ and reported no failures in this experiment at less than about 5% $MgCl_2$.

Instances of chloride cracking at representative ECC solution temperatures and at low solution chloride concentration have generally been on surfaces on which concentration of the chloride occurred. In the ECC system, concentration of chlorides is not anticipated since the solution will operate subcooled with respect to the containment pressure and further the containment atmosphere will be 100% relative humidity.

3. Alkaline Nature of the ECC Solution

The ECC solution will have a solution pH of between 8.5 and 10.0 after the addition of spray additive (NaOH). Numerous investigators have shown that increasing the solution pH decreases the probability of failure. Thomas et al[9] showed that the failure probability decreases with increasing pH of boiling solutions of $MgCl_2$. More directly applicable, Scharfstein and Brindley[10] showed that increasing the solution pH to 8.8 by the addition of NaOH prevented the occurrence of chloride stress corrosion cracking in a 10 ppm Cl (as NaCl) solution at 165°F. 30 stressed stainless steel specimens including 304 as received, 347 as received and 304 sensitized were tested. No failures were observed.

Other test runs by Scharfstein and Brindley showed the influence of solution pH on higher chloride concentrations, up to 550 ppm Cl; however, in these tests the pH adjusting agents were either sodium phosphate or potassium chromate. The authors express the opinion, however, that in the case of the chromate solution, chloride cracking inhibition was simply due to the hydrolysis yielding pH 8.S and not to an influence of the chromate anion. A similar hydrolysis will occur in the borate solution.

Studies conducted at Oak Ridge National Laboratory by Griess and Bocarella[11] on type 304 and type 316 stainless steel U-bend stress specimens exposed to an alkaline borate solution (0.15M NaOH - 0.28M H_3BO_3) containing 100 ppm chloride (as NaCl) showed no evidence of cracking after 1 day at 140°C, 7 days at 100°C, 29 days at 55°C. These extreme test conditions, combined with the fact that some parts of the test specimens were subjected to severe plastic deformation and intergranular attack before exposure, show that the probability of chloride induced stress corrosion cracking in a post accident environment are very low indeed.

In summary, therefore, it is concluded that exposure of the stainless steel engineered safety feature components to the ECC solution during the post accident period will not impair its operability from the standpoint of chloride stress corrosion cracking. The environment ' low temperature, low chlorides and high pH which will be experienced during the post accident period will not be conducive to chloride cracking.

5.0 Corrosion of Aluminum Alloys

Corrosion testing has shown that aluminum alloys are not compatible with alkaline borate solution. The alloys generally corrode fairly rapidly, at the post accident condition temperatures, with the liberation of hydrogen gas. A number of corrosion tests were conducted in the PWRD laboratories and at ORNL facilities. A review of applicable aluminum corrosion data is given in Table 6.3-4 and on Figure 6.3-8.

5.1 Aluminum Corrosion Products in Alkaline Solution

The corrosion of aluminum in alkaline solution, expected following a design basis accident (DBA) has been shown to proceed with the formation of aluminum hydroxide(14), (15), (16) and the aluminate ion, as well as with the production of hydrogen gas.

The DBA conditions expected for the Indian Point Unit No. 2 plant include the establishment of an alkaline ECC solution having a total volume of liquid of 4.47×10^5 gallons after actuation of the engineered safety features.

As mentioned above, aluminum is known to corrode in alkaline solutions to give a precipitate of $Al(OH)_3$ which in turn can redissolve in an excess of alkali to form a complex aluminate. Van Horn(14) noted that the precipitation of $Al(OH)_3$ begins about pH 4 and is essentially complete at pH 7. A further increase in pH to about 9 causes dissolution of the hydroxide with the formation of the aluminate.

It can be seen, therefore, that the solubility of aluminum corrosion product is a function of the pH of the environment. Consistent with this, the corrosion of aluminum is also strongly dependent on the solution pH since when the corrosion products are dissolved from the metal surface corrosion of the base metal can proceed more freely.

Figure Q 6.3-9 presents a plot of aluminum corrosion rate as a function of solution pH.[1] The corrosion rate of aluminum is seen to decrease by a factor of 21 (1/.048) as the pH decreases from 9.3 to 8.3 and by a factor of 83 (1/.012) as the pH decreases from 9.3 to 7.0.

Therefore, one must consider both corrosion and the dissolution of the corrosion products at specific reference conditions since the two are directly related.

The corrosion reactions that are of interest in the DBA condition here would include the reaction of aluminum in alkaline solution to form aluminum hydroxide: i.e.,

$$2\ Al + 6\ H_2O \rightleftharpoons 2\ Al\ (OH)_3 + 3\ H_2 \qquad (1)$$

and dissolution of the hydroxide to form the aluminate, i.e.,

$$Al\ (OH)_3 \rightleftharpoons AlO_2^- + H^+ + H_2O \qquad (2)$$

A knowledge of the solubility product of the aluminum hydroxide in an alkaline solution allows the determination of the solubility expected for the hydroxide in the DBA environment.

Deltombe and Pourbaix[17] have determined the solubility product of aluminum hydroxide. Using the value of 2.28×10^{-11} for K_{sp}, as reported by Deltombe and Pourbaix, the following calculation can be made.

The solubility of $Al(OH)_3$ is determined from Equation 2

$$Al(OH)_3 \rightleftharpoons AlO_2^- + H^+ + H_2O$$

$$K_{sp} = [AlO_2^-]\ [H^+]$$

$$2.28 \times 10^{-11} = [AlO_2^-]\ [H^+]$$

at pH = 9.3

$$[AlO_2^-] = \frac{2.28 \times 10^{-11}}{5 \times 10^{-10}} = 4.6 \times 10^{-2} \text{ moles/liter}$$

therefore, the solubility of $Al(OH)_3$ in a pH 9.3 solution at 25°C (77°F) is 4.6×10^{-2} moles/liter or 3.0×10^{-2} lbs/gal. Expressed as aluminum, the solubility at these conditions is 1.05×10^{-2} lbs/gal.

The solubility of the aluminum corrosion products in the post accident environment is a function of both solution pH and temperature. Figure Q 6.3-10 presents plots of the corrosion product solubility, expressed in terms of aluminum, versus solution pH for temperatures of 77°F and 150°F. The change in solubility with temperature is found utilizing the relationship of the free energy of formation, temperature, and the solubility product.

With the data available from Figures Q 6.3-9 and 10 and a knowledge of the reference aluminum corrosion behavior for any specific plant, one can calculate the expected solubility limits for the corrosion reaction.

For Unit No. 2, there are 4.47×10^5 gallons of ECC solution after actuation of the safety features. The total amount of aluminum present in the Unit No. 2 containment is given in Table Q 6.3-2. Table Q 6.3-5 shows the corrosion of aluminum with time for the design basis pH 9.3 post accident environment.

Table Q 6.3-6 presents a summary of the applicable solubility and corrosion parameters for various conditions. The table lists the applicable solubility products (K_{sp}) and solubilities at the various temperatures and solution pH's together with the soluble aluminum limit for the Unit No. 2 system at the specific conditions. The last values in the table give the aluminum solubility margin after 100 days corrosion; that is, the soluble Al limit divided by the aluminum corroded. It can be seen that in all cases, including the very conservative low temperature and low pH conditions, the ECC solution is not expected to be saturated with aluminum corrosion products. Further, within the expected design conditions for temperature and pH, the aluminum solubility margin ranges from approximately 20 to 106.

It is concluded therefore, that the corrosion products of aluminum will be in the soluble form during the post accident period considered and hence, there is no potential for deposition on flow orifices, spray nozzles or other equipment.

Behavior of Circulating Aluminum Corrosion Products

The solubility of aluminum corrosion products has shown that for Unit No. 2, the entire inventory produced after 100 days exposure to the post DBA condition would remain in solution. The review also indicates that the ECC solution is only approximately 17% saturated at 77°F and less than 1% saturated at 150°F.

It is of interest, however, to review the experience of facilities which have operated with insoluble aluminum corrosion products and to relate their conditions with those expected in the post accident environment.

The most significant experience available to date is that of Griess(18) who operated a recirculating test facility to measure the corrosion resistance of a variety of materials in alkaline sodium borate spray solution.

Tests were conducted on 1100, 3003, 5052 and 6061 aluminum alloys exposed at 100°C in pH 9.3 sodium borate solution (0.15 M NaOH - 0.28 M H_3BO_3). It was reported that even though the solution contained copious amounts of flocculent aluminum hydroxide, it has no effect on flow through the spray nozzle (0.093 inch orifice). The pH of the solution did not change because of the increase in the corrosion products.

Griess* in describing his observations with regards to aluminum corrosion product deposition potential stated that:

a) no significant deposition was observed on the cooling coil installed in the solution.

b) no significant deposition was observed on the heated surfaces of the facility.

c) no significant deposition was observed on isothermal facility surfaces.

The amounts of aluminum corroded to the solution in the tests conducted by Griess at 55°C and 100°C were approximately 4.0 and 18.6 grams respectively. The concentration of aluminum present in the recirculation stream, therefore, was approximately 0.2 and 1 gram/liter respectively. This value is about a factor of about 5 above the aluminum concentration expected in the post accident ECC solution at Unit No. 2 in a pH 9.3 solution after 100 days.

Hatcher and Rae[19] describe the appearance of turbidity in the NRU reactor and "propose" that deposition of aluminum corrosion products may have occurred on heat exchanger surfaces, although they do not report any specific examination results. Moreover, Hatcher and Rae report no operations problems associated with the presence of aluminum corrosion product turbidity in the NRU Reactor. The overall heat transfer coefficient for each NRU reactor heat exchanger was measured after 2 years of full power operation on several occasions and within the limit of accuracy of the measurements, reported at approximately 5%, no change in the thermal resistance had been observed.

It is concluded, therefore, both from the work of Griess and Hatcher and Rae, that the deposition of aluminum corrosion products on heat exchangers surfaces will not be significant in the post accident environments even for the curcumstances of insoluble product formation.

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* Private Communciation.

Table 6.3-4

Corrosion of Aluminum Alloys in Alkaline Sodium Borate Solution

Data Point	Temperature °F	Alloy Type	Test Duration	Corrosion Rate $mg/dm^2/hr$	pH	Exposure Condition	Reference
1	275	5053	3 hrs.	96.2	9	Solution	WCAP-7153, Table 9
2	275	5005	3 hrs.	840	9	Solution	WCAP-7153, Table 9
3	200	6061	320 hrs.	15.4	9.3	Solution	WCAP-7153, Table 8 WCAP-7153, Figure 9
4	210	5052	7 days	53.0	9	Solution	WCAP-7153, Table 7 WCAP-7153, Figure 8
5	210	5052	2 days	14.0	9	Solution	WCAP-7153, Table 5
6	210	5005	2 days	27.1	9	Solution	WCAP-7153, Table 5
7	284	5052	1 day	54	9.3	Spray	ORNL-TM-2425, Table 3.13
8	284	5052	1 day	31.5	9.3	Solution	ORNL-TM-2425, Table 3.13
9	212	6061	3 days	126	9.3	Spray	ORNL-TM-2368, Table 3.6
10	212	6061	3 days	110	9.3	Solution	ORNL-TM-2368, Table 3.6
11	150	6061	7 days	2.9	9.3	Solution	PWRD recent data
12	150	5052	7 days	4.2	9.3	Solution	PWRD recent data

Table 6.3-5

Corrosion Products of Aluminum Following DBA

Indian Point Unit No. 2

Time After Reactor Trip Days	Mass of Aluminum Corroded lb x 10^{-2}	Hydrogen Produced SCF x 10^{-3}	Mass of Al $(OH)_3$ Formed lb x 10^{-2}
1	1.71	3.41	4.94
5	4.31	8.60	12.4
10	4.50	8.98	13.0
20	4.88	9.75	14.1
30	5.26	10.5	15.2
40	5.66	11.3	16.4
50	6.06	12.1	17.5
60	6.41	12.8	18.5
70	6.81	13.6	19.7
80	7.21	14.4	20.9
90	7.61	15.2	22.0
100	7.97	15.9	23.0

Table Q6.3-6

Summary of Unit 2 Aluminum Corrosion Product Solubility Data

Parameter	Solution Temperature			
	[illegible]°F		150°F	
	pH 9.3	pH 8.3	pH 9.3	pH 8.3
Solubility Product K_{sp}	2.28×10^{-11}	2.28×10^{-11}	6.15×10^{-10}	4.16×10^{-10}
Al Solubility (# Al/gal)	1.05×10^{-2}	1.05×10^{-3}	1.9×10^{-1}	1.9×10^{-2}
Soluble Al Limit/a for ECCS (#)	4.69×10^{3}	4.69×10^{2}	8.49×10^{4}	8.49×10^{3}
Al Corrosion Rate (Normalized)	(Not Used)	(Not Used)	1	0.048
Al Corroded After 100 Days (#)	(Not Used)	(Not Used)	795	439/b
Al Solubility Margin at 100 Days	5.9/c	1.1/c	106	19

/a Indian Point Unit No. 2 solution volume 4.47×10^{5} gal.

/b Value assumes rapid corrosion of all Al paint and reactor vessel foil insulation.

/c Note corrosion rate at 150°F was used for "al corroded" value; hence, value is very conservative.

6.0 Compatibility of Protective Coatings with Post Accident Environment

The investigation of materials compatibility in the post accident design basis environment also included an evaluation of protective coatings for use in containment.

The results of the protective coatings evaluation presented in WCAP-7198(12), showed that several inorganic zincs, modified phenolics and epoxy coatings are resistant to an environment of high temperature (320°F maximum test temperature) and alkaline sodium borate. Long term tests included exposure to spray solution at 150 - 175°F for 60 days, after initially being subjected to the conservative DBA cycle shown in Figure 6.3-3. The protective coatings, which were found to be resistant to the test conditions, that is, exhibited no significant loss of adhesion to the substrate nor formation of deterioration products, comprise virtually all of the protective coatings recommended for use in containment. Hence, the protective coatings will not add deleterious products to the core cooling solution.

It should be pointed out that several test panels of the recommended types of protective coatings were exposed for two design basis accident cycles and showed no deterioration or loss of adhesion with the substrate.

7.0 Evaluation of the Compatibility of Concrete-ECC Solution in the Post Accident Environment

Concrete specimens were tested in boric acid and alkaline sodium borate solutions at conditions conservatively (320°F maximum and 200°F steady state) simulating the post DBA environment.

The purpose of this study was to establish:

a) the extent of debris formation by solution attack of the concrete surfaces.

b) the extent and rate of boron removal from the ECC solution through boron - concrete reaction.

Tests were conducted in an atmospheric pressure, reflux apparatus to simulate long term exposure conditions and in a high pressure autoclave facility to simulate the DBA short term, high temperature transient.

For these tests the total surface area of concrete in the design containment which may be exposed to the ECC solution following a DBA was estimated at 6.3×10^4 square feet. This value includes both coated and uncoated surfaces. The ECC solution volume for a reference plant was considered at approximately 313,000 gallons and the surface to volume ratio from these values is $\sim$ 29 in^2/gallon. The surface to volume ratios for the concrete - boron tests used were between 28 and 78 in^2/gallon of solution. Table 6.3-7 presents a summary of the data obtained from the concrete - boron test series.

Testing of uncoated concrete specimens in the post accident environment showed that attack by both boric acid and the alkaline boric acid solution is negligible and the amount of deterioration product formation is insignificant. Other specimens covered with modified phenolic and epoxy protective coatings, showed no deterioration product formation. These observations are in agreement with Orchard[13] who lists the following resistances of Portland Cement concrete to attack by various compounds:

boric acid	- little or no attack
alkali hydroxide solution under 10%	- little or no attack
sodium borate	- mild attack
sodium hydroxide over 10%	- very little attack

Exposure of uncoated concrete to spray solution between 320°F and 210°F has shown a tendency to remove boron very slowly, presumably precipitating an insoluble calcium salt. The rate of change of boron in solution was measured at about 130 ppm per month with pH 9 solution at 210°F for an exposed surface of about 36 square inches per gallon of solution (much greater than any potential exposure in the containment). The boron loss during the high temperature transient test (320°F maximum) was about 200 ppm. Figure 6.3-11 shows a representation of the boron loss from the ECC solution versus time, by a boron - concrete reaction following a DBA. The time period from 0 - 6 hours shows the loss during a conservative high temperature transient test, ambient to 320°F to 285°F. The data from 6 hours to 30 days is based on 210°F data.

A depletion of boron at this rate poses no threat to the safety of the reactor because of the large shut down margin and the feasibility of adding more boron solution should sample analysis show a need for such action.

Table 6.3-7

Concrete Specimen Test Data

Concrete - Boron Test #	Total Exposure Period (Days)	Surface/Volume (in^2/gal)	Exposed Weight Change (Grams)	Initial Specimen Weight (Grams)	Visual Examination
1	24	28	-22.4	560.0	No apparent change
3	28	20	+21.5	404.0	Light, yellowish, deposit on specimen
4[(a)]	72	38	0	641.2	No apparent change - coating adhesion excellent
5	72	43	-0.2	769.5	Light, hard deposit on specimen
6	∿ 4[(b)]	54	-	601.4	No apparent change - small amount of sand particles in test can
7	175	23	+11.0	457.0	No apparent change
8[(a)]	175	38	+26.5	751.0	No apparent change - coating adhesion excellent
9[(a)]	∿ 5[(b)]	78	+4.0	732.0	No apparent change - coating adhesion excellent

(a) These specimens coated with Phenoline 305. All others were uncoated.

(b) These tests were at high temperature DBA transient conditions. All others at 195 - 205°F.

8.0 Miscellaneous Materials of Construction

8.1 Sealants

Candidate sealant materials for use in the reactor containment system were evaluated in simulated DBA environments. Cured samples of various sealants were exposed in alkaline sodium borate solution, pH 10.0, 3000 ppm to a maximum temperature of 320°F.

Table 6.3-8 presents a summary of the sealant materials tested together with a description of the panels appearance after testing. Three generic types of sealants were tested: butyl rubber, silicone, and polyurethane. Each of the materials was the "one package" type, that is no mixing of components was necessary prior to application. The materials were applied on stainless steel and allowed to cure well prior to testing.

The test results showed that the silicone sealants tested were chemically resistant to the DBA environment and are acceptable for use in containment.

Sealant 780 by Dow Corning Corporation would be acceptable for use in the containment. Major applications of this sealant could be as concrete expansion joint sealant on the liner insulation panels. Sealant 780 will contribute no deterioration products to the ECC solution during the post DBA period and will maintain its structural integrity and elastic properties.

8.2 PVC Protective Coating

Tests were conducted to determine the stability of the polyvinyl chloride protective coating, of the type which might be used on conduit in the DBA environment. Samples of the PVC exposed to alkaline sodium borate solutions at DBA conditions showed no loss in structural rigidity and no change in weight or appearance.

A sample of PVC coated aluminum conduit (1" O.D. x 8" length) was irradiated by means of a Co-60 source, at an average dose rate of 3.2×10^6 rads/hr to a total accumulated dose of 9.1×10^7 rads. The specimen was immersed in alkaline sodium borate solution (pH 10, B = 3000 ppm) at 70°F. Visual examination of the coating after the test showed no evidence of cracking, blistering or peeling and the specimen appeared completely unaffected by the gamma exposure. Chemical analysis of the test solution indicated that some bond breakage had occurred in the PVC coating as evidenced by an increase in the chloride concentration. The gamma exposure of $\sim 10^8$ rad resulted in a release to the solution of 26 mg of chloride per square foot of exposed PVC surface. Considering a total surface area of PVC coating present in containment ($\sim$ 500 ft^2) and the ECC solution volume of 313,000 gallons, the chloride concenrtation increase in the ECC solution due to irradiation of the coating, would be $\sim$ 0.01 ppm.

It is concluded, therefore, that PVC protective coating will be stable in the DBA environment.

8.3 Fan Cooler Materials

Samples of the following air handling system materials were exposed in an autoclave facility to the DBA temperature - pressure cycle:

a) mositure separator pad

b) high efficiency particulate filter media

c) asbestos separator pads

d) adhesive for joining separator pads and HEPA filter media corners

e) neoprene gasketing material

The materials were exposed in both the steam phase and liquid phase of a solution of sodium tetraborate (15 ppm B) to simulate the concentrations expected down stream of the fan cooler cooling coils. Examination of the specimens after exposure showed the following:

a) moisture separator pads were somewhat bleached in color but maintained their structural form and showed good resiliency as removed in both liquid and steam phase exposure.

b) high efficiency particulate filter media maintained its structural integrity in both the liquid and steam phase. No apparent change.

c) asbestos separator pads showed some slight color bleaching. however, both steam and liquid phase samples maintained their structural integrity with no significant loss in rigidity.

d) adhesive material for the HEPA/separator pad edges showed no deterioration or embrittlement and maintained its adhesive property.

e) neoprene gasketing material is also satisfactory in both the steam and liquid phase. The material showed only weight gain and a shrinkage of 15 to 30 percent based on a superficial, one flat side area. The gasket thickness decreased about 10 percent. The gasket material was unrestrained during the exposure and hence the dimensional changes experienced, are greater than those which would result in the fan cooler unit.

Table 6.3-8

Evaluation of Sealant Materials for Use in Containment

Sealant Type	Manufacturer	Post-Test Appearance
Butyl rubber	A	Unchanged, flexible
Silicone	B	Unchanged, flexible
Silicone	B	Unchanged, flexible
Polyurethane	C	Sealant bubbled and became very soft. Solution permeated into bubbles.
Polyurethane	C	Sealant swelled and became soft, solution permeated into material.
Polyurethane	C	Sealant swelled, very soft and tacky, solution permeated into material.

10

Q 6.3-33 to

Q 12.1-59

10624

10624

LIST OF REFERENCES

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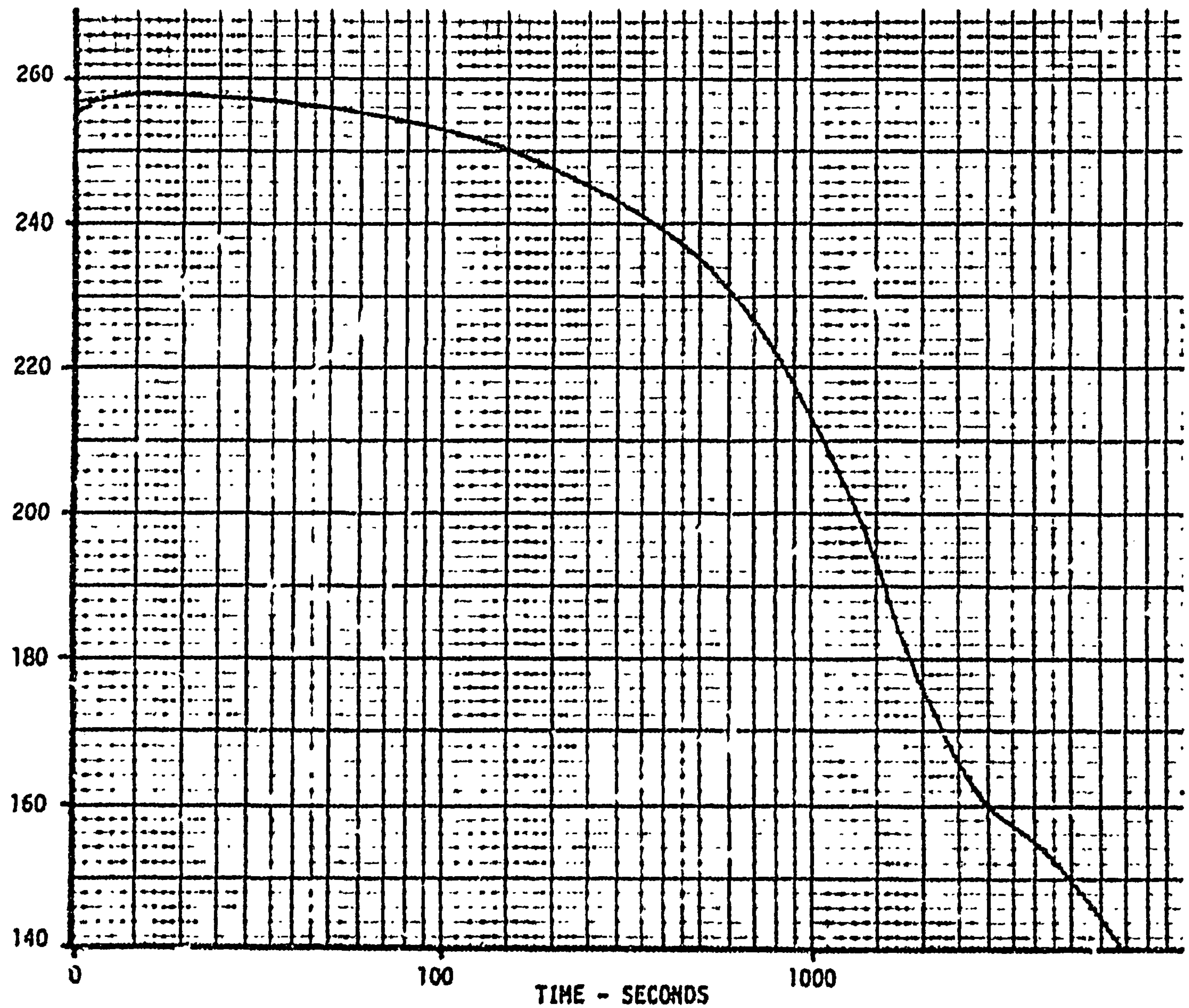
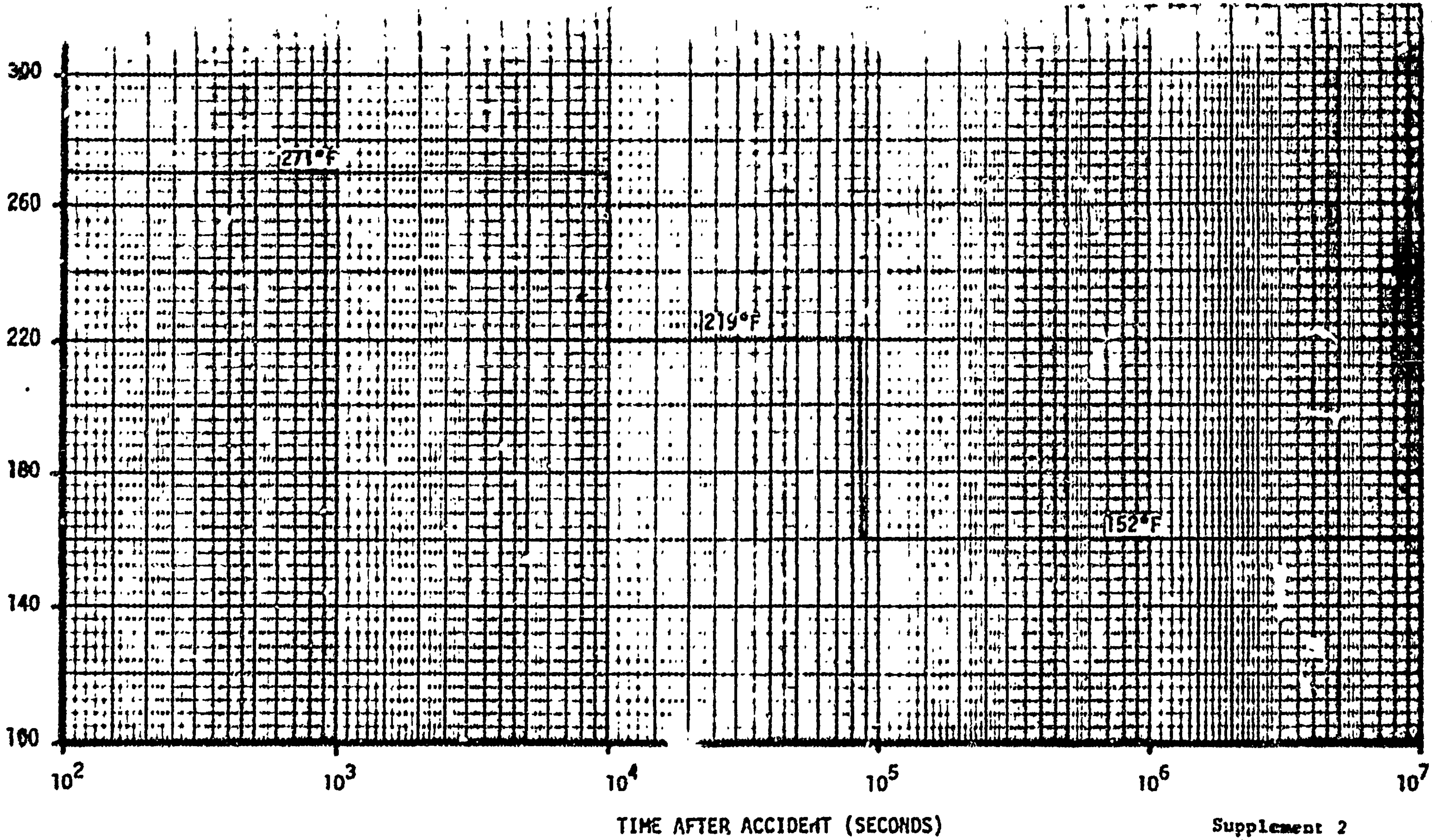
FIGURE 6.3-1 CONTAINMENT ATMOSPHERE TEMPERATURE
DESIGN BASES SAFETY INJECTION



INDIAN POINT UNIT NO. 2

FIGURE 6.3-2 POST-ACCIDENT CONTAINMENT MATERIALS DESIGN CONDITIONS



Supplement 2
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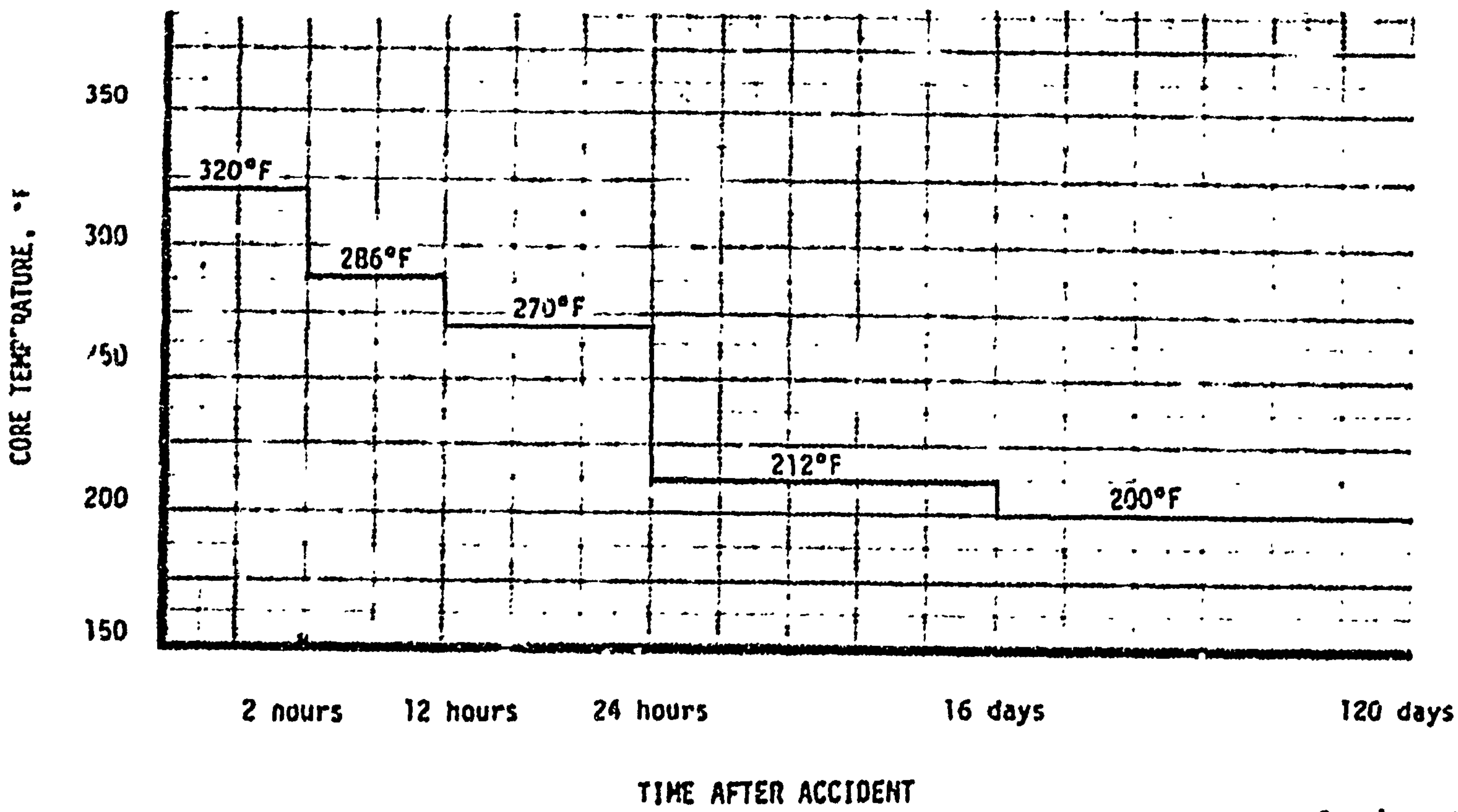
FIGURE 6.3-3 POST-ACCIDENT CORE MATERIALS DESIGN CONDITIONS



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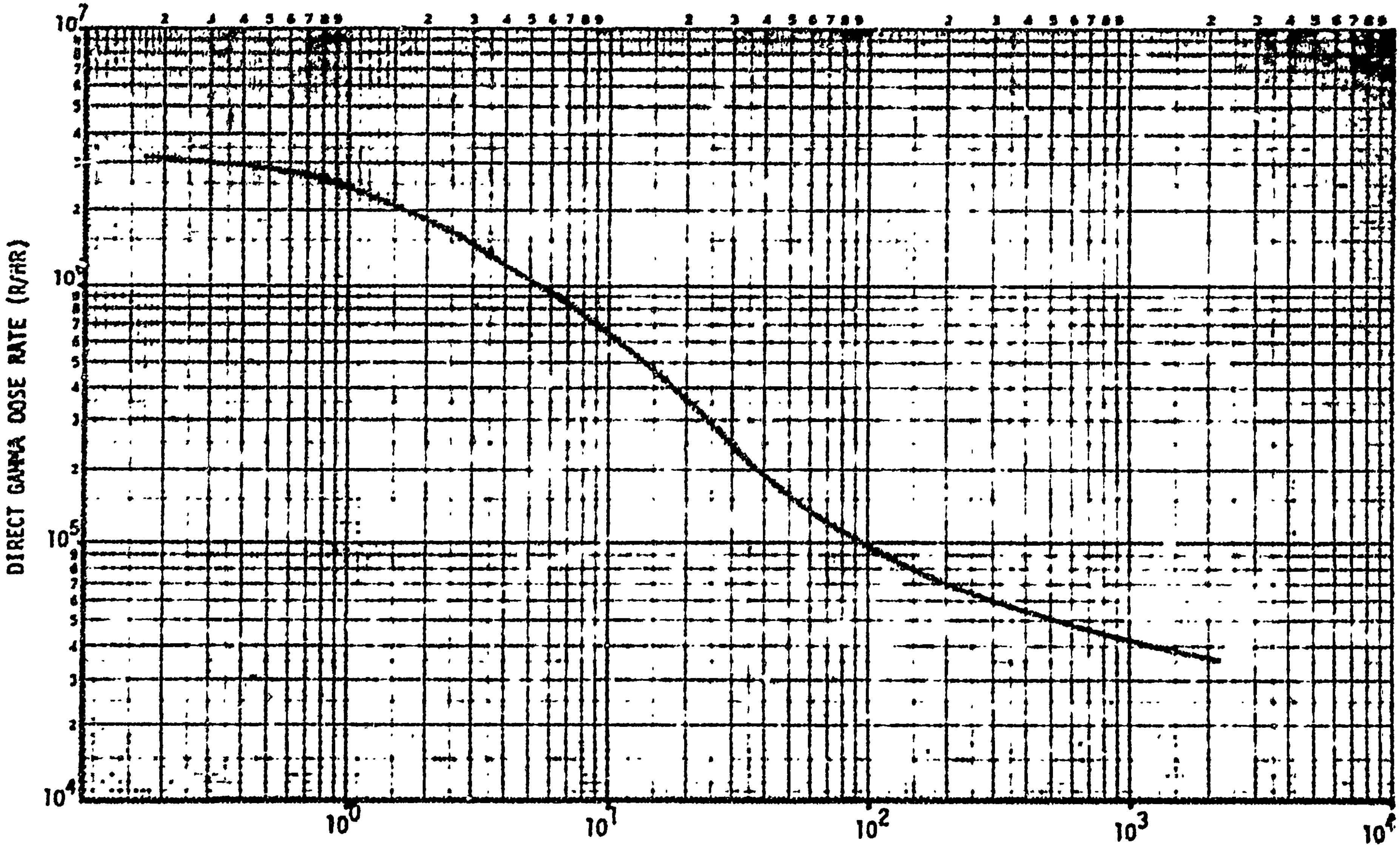
INDIAN POINT UNIT NO. 2

FIGURE 6.3-4 CONTAINMENT ATMOSPHERE DIRECT GAMMA DOSE RATE



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INDIAN POINT UNIT NO. 2

FIGURE 6.3-5 CONTAINMENT ATMOSPHERE INTEGRATED GAMMA DOSE LEVEL

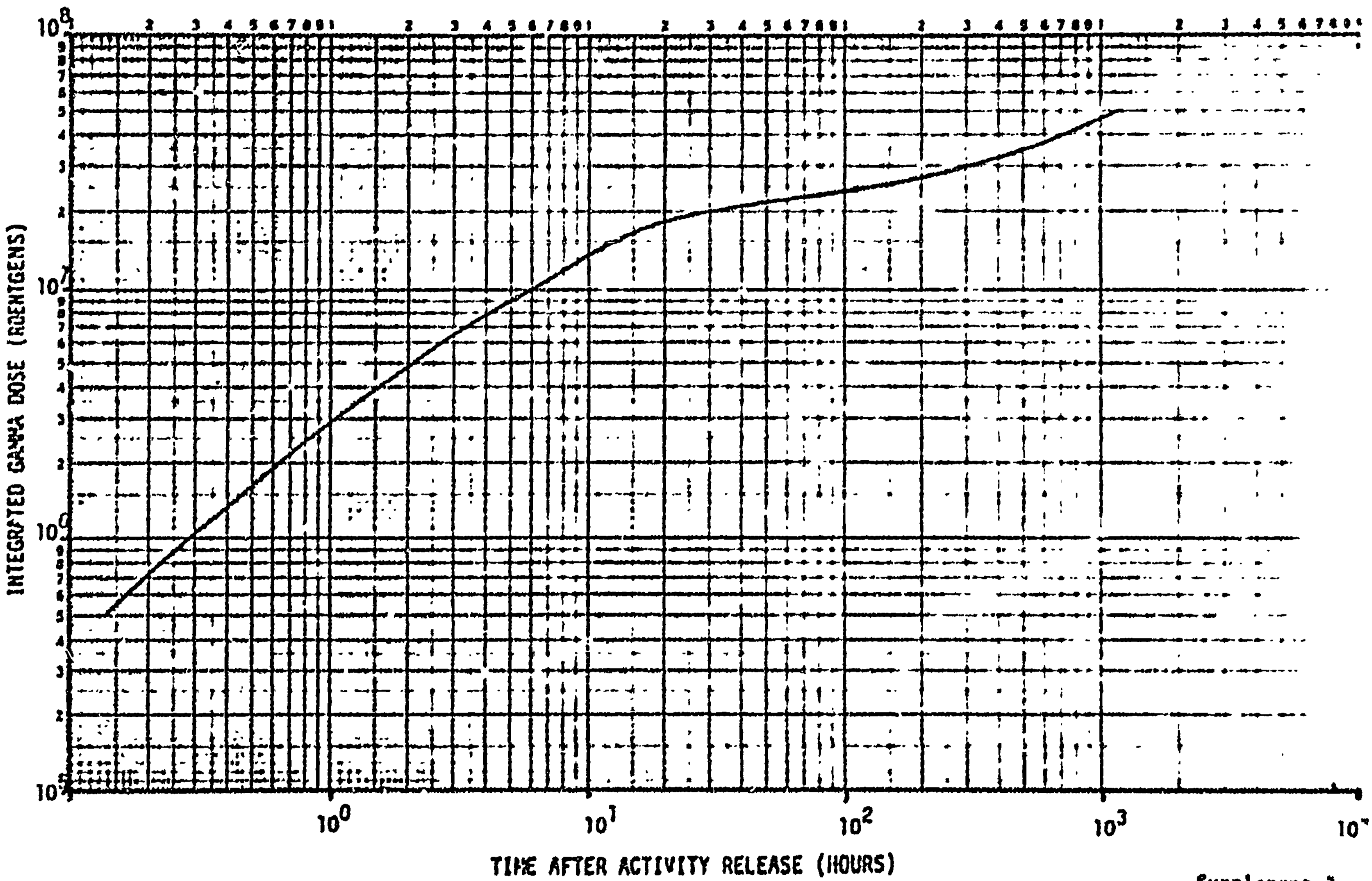

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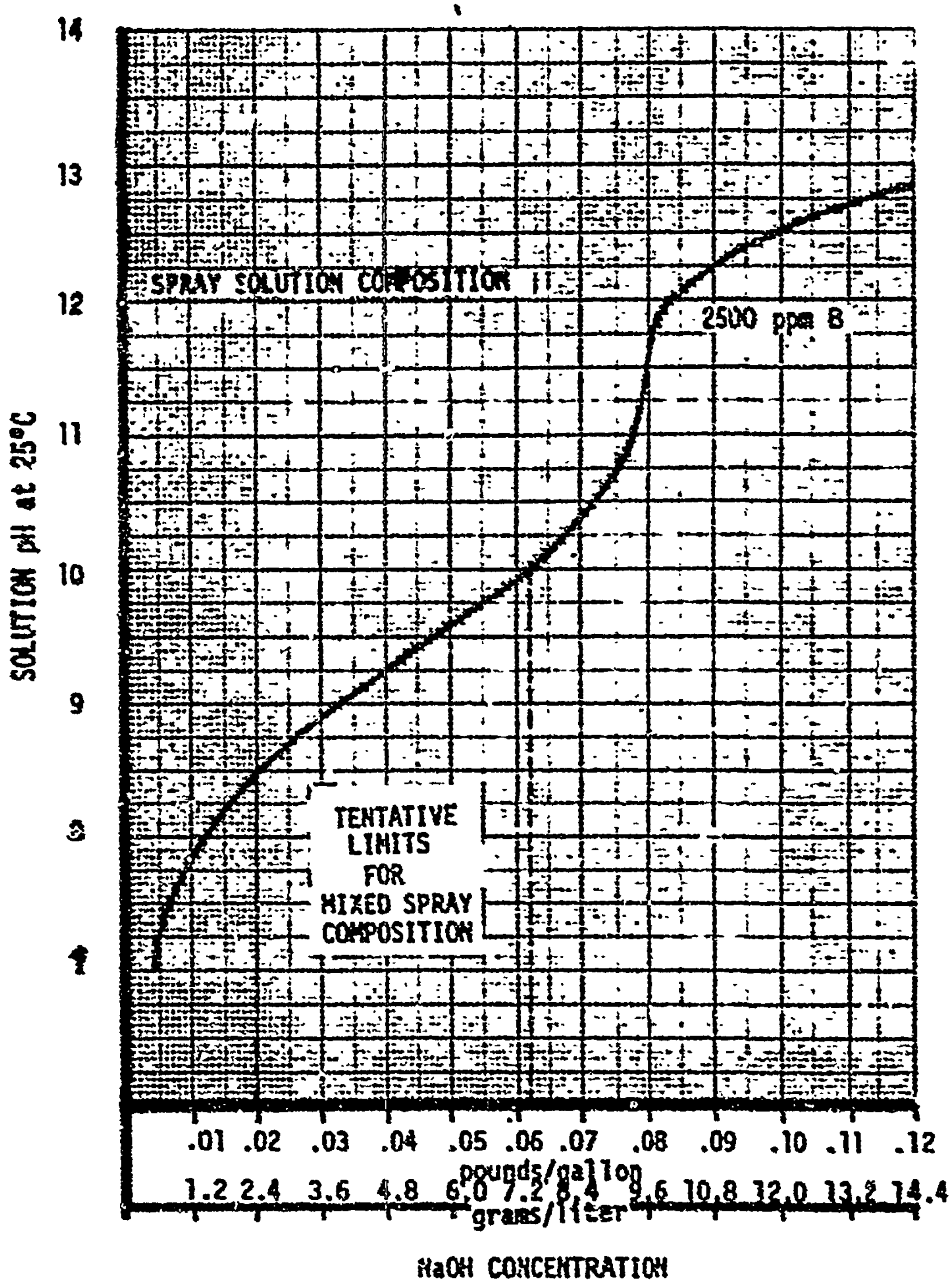
FIGURE 6.3-6 TITRATION CURVE FOR BORIC ACID WITH SODIUM HYDROXIDE



TEMPERATURE - CONCENTRATION RELATION FOR CAUSTIC
CORROSION OF AUSTENITIC STAINLESS STEEL

(AFTER SWANDBY, R.K. CHEM. ENG. 69, 186 NOV. 12, 1962)

FIGURE 6.3-7

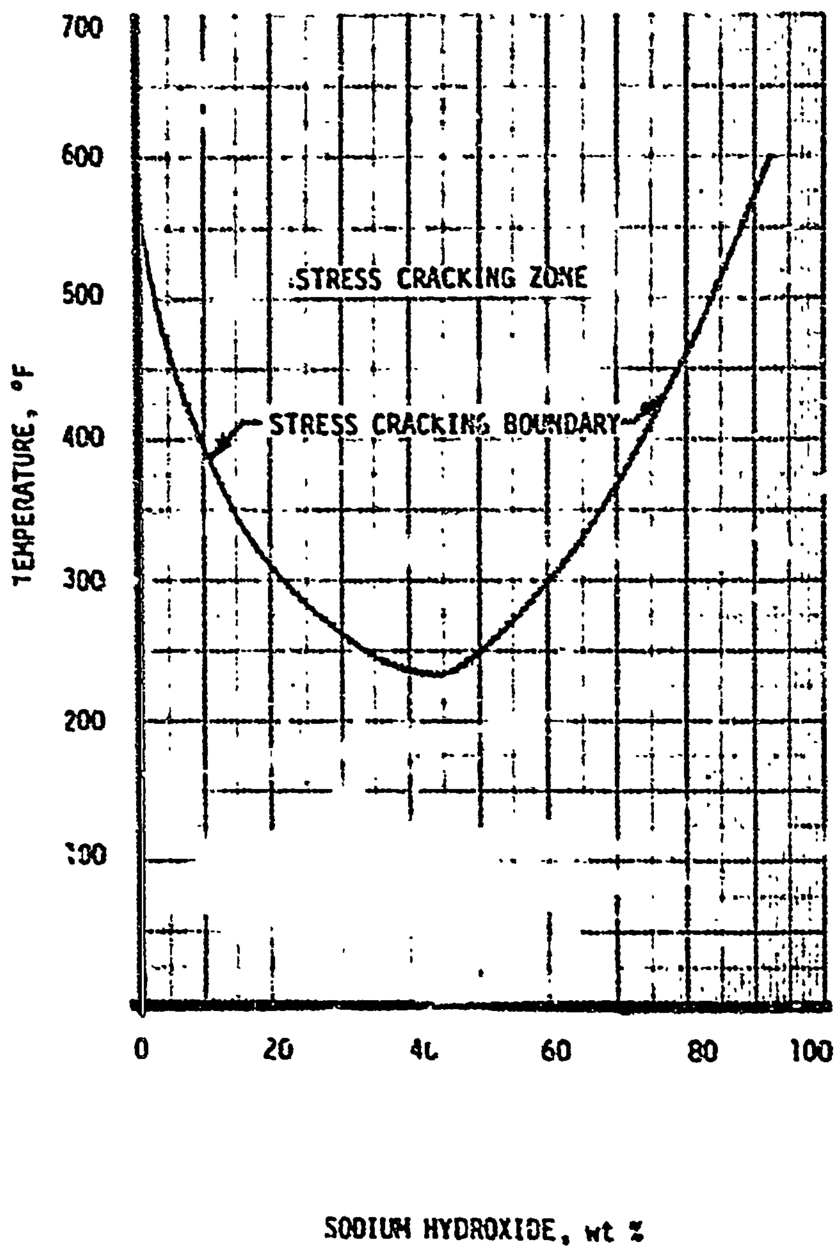
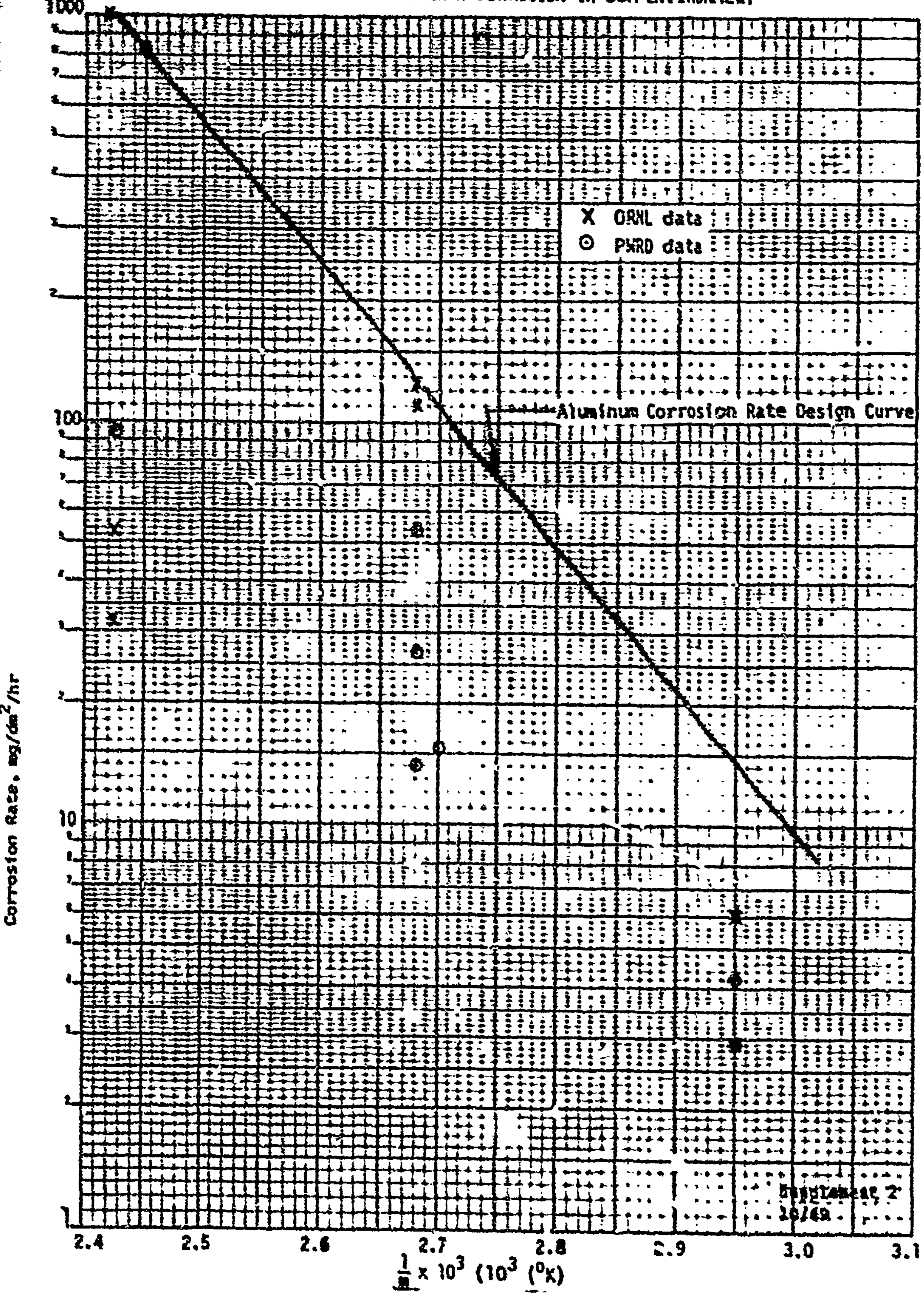
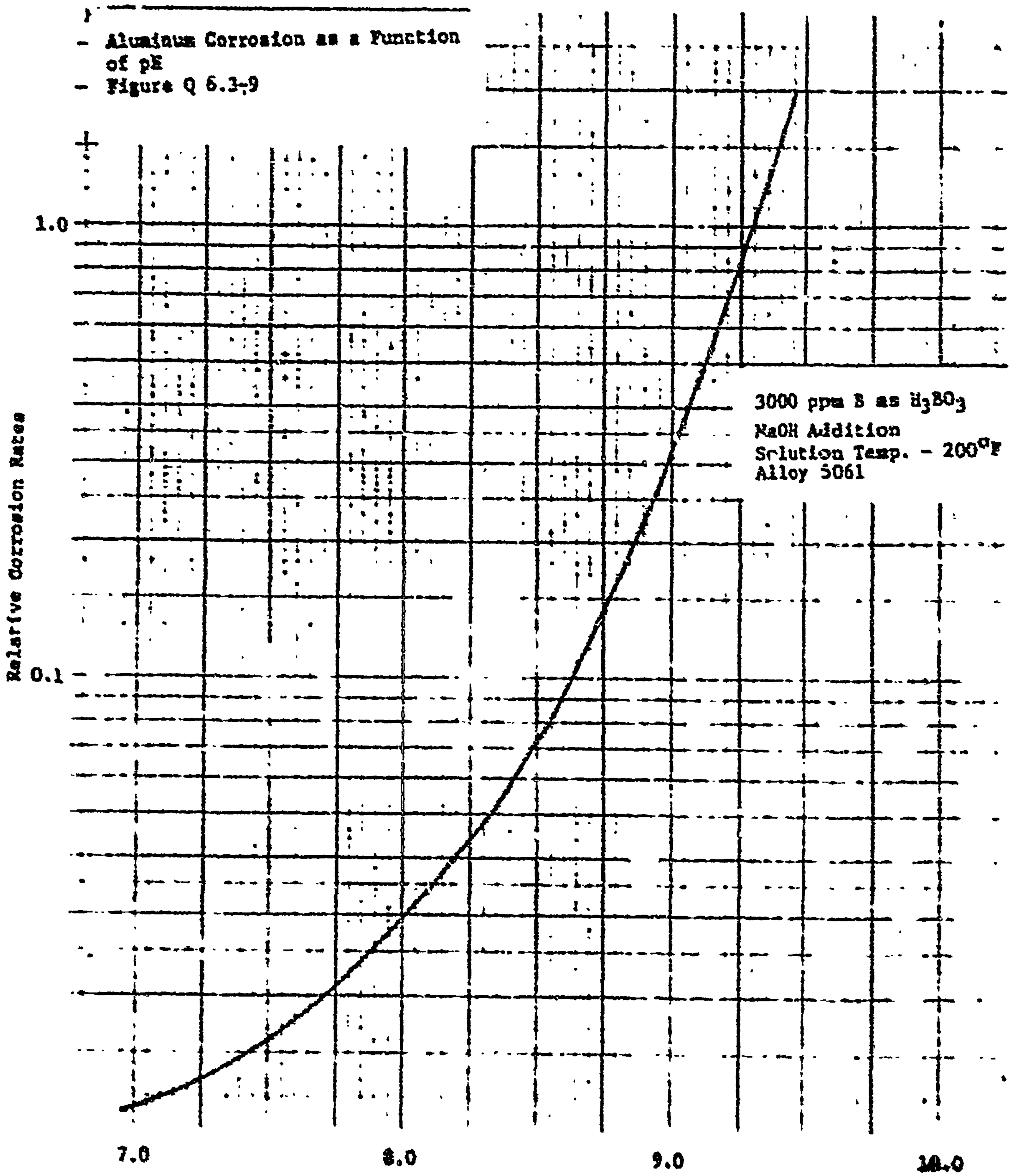


FIGURE 6.8-8 ALUMINUM CORROSION IN DBA ENVIRONMENT




Aluminum Corrosion as a Function of pH
Figure Q 6.3-9
3000 ppm B as H_3BO_3
NaOH Addition
Solution Temp. - 200°F
Alloy 6061
Relative Corrosion Rates
1.0
0.1
7.0
8.0
9.0
10.0

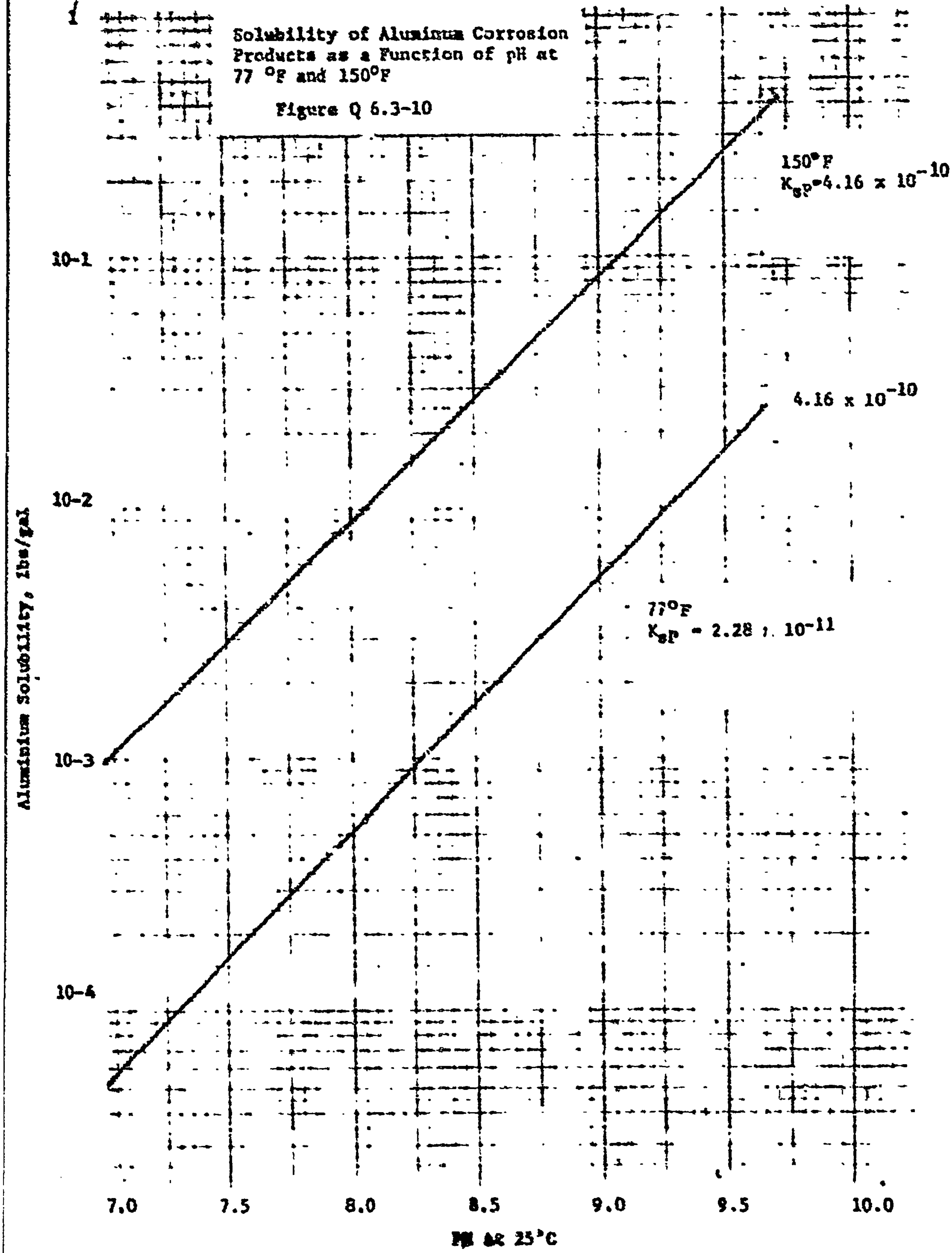
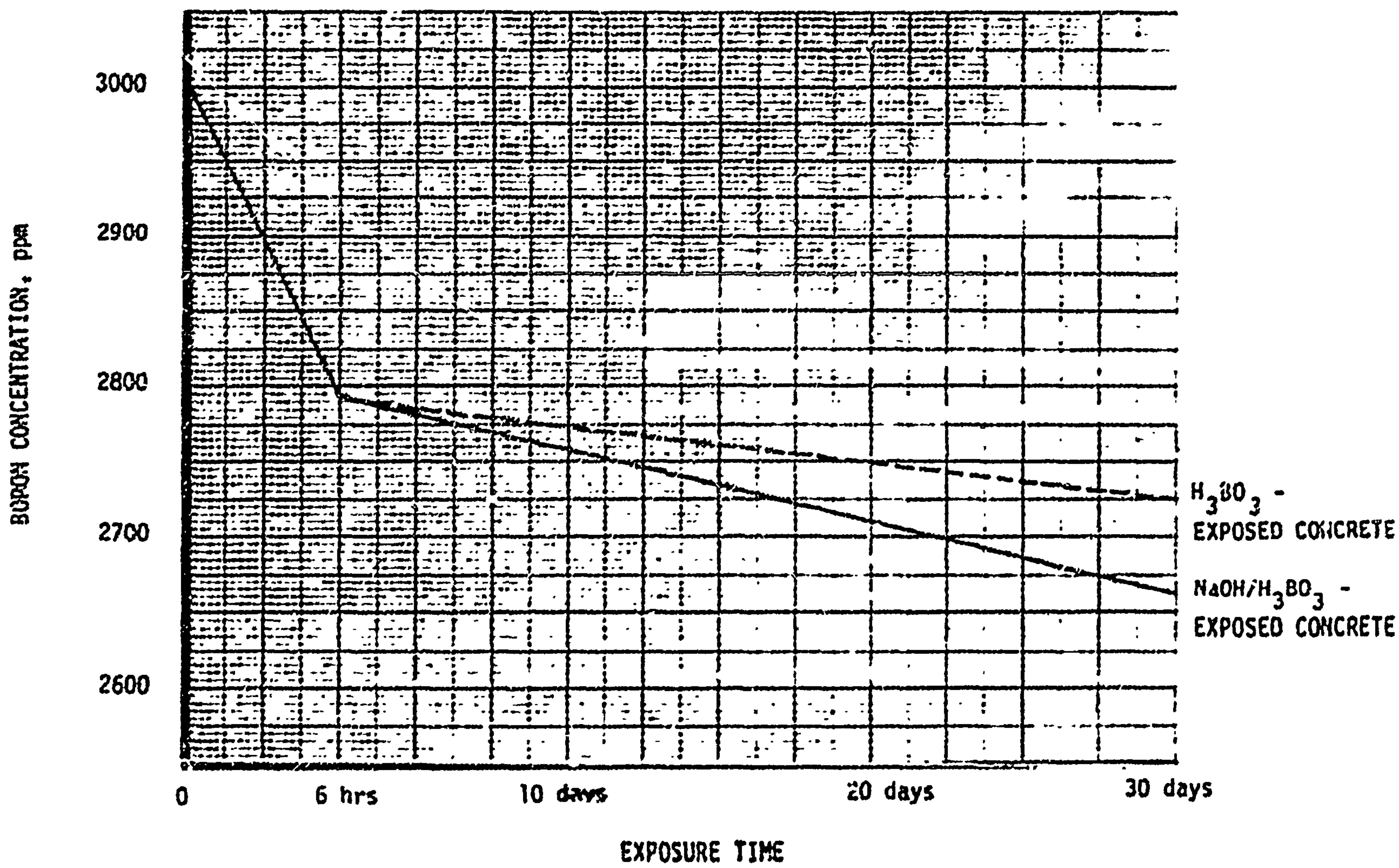

Solubility of Aluminum Corrosion
Products as a Function of pH at
77 °F and 150°F
Figure Q 6.3-10
150°F
Ksp=4.16 x 10-10
4.16 x 10-10
77°F
Ksp = 2.28 x 10-11
10-1
10-2
10-3
10-4
Aluminum Solubility, lbs/gal
7.0
7.5
8.0
8.5
9.0
9.5
10.0
PH At 25°C

FIGURE [illegible]13-11 BORON LOSS FROM BORON - CONCRETE REACTION FOLLOWING A DBA



QUESTION 6.4

Describe the long term storage conditions for the concentrated sodium hydroxide. Consider tank corrosion, air containmination, valve galling, etc. Analyze the potential for clogging of delivery lines by deposited solids due to chemical corrosion and/or temperature extremes.

ANSWER

A materials compatibility review for the spray additive tank and associated equipment during long term storage of sodium hydroxide is presented below. The exposure conditions are shown in Table 1. The materials for the various components are shown in Table 2. The corrosion rates for the various materials at or near the long term exposure conditions with air contamination are shown in Table 3. The immunity of most of the materials in Table 2 to caustic cracking at the exposure conditions listed in Table 1 has been reported by Logan (Ref. 6) (See Figure 6.4-1). No caustic cracking of 17-4 PH (Ref. 7) or Stellite has been reported.

The effect of carbon dioxide from air exposure on corrosion of iron is shown in Figure 6.4-2 (Ref. 8). At pH 14, no additional corrosion is observed over that observed in carbon dioxide free solution. In the Indian Point Unit 2 system, a nitrogen blanket is continuously maintained over the sodium hydroxide solution in the spray additive tank thus essentially eliminating any carbon dioxide contamination of the solution.

The Nordel[/a] rubber diaphragm material, used in the tank valves were exposed in 33 w/o sodium hydroxide solution at 110°F for 6 months and found to be unaffected by the simulated spray additive tank solution. Westinghouse is continuously testing the Nordel rubber at the above conditions. The completely unchanged appearance of Nordel rubber after 6 months exposure in sodium hydroxide solution would indicate that integrity of the Nordel rubber diaphragm in the spray additive tank valves would not be affected by long term exposure to the spray additive solution.

/a Nordel is a product of Dupont De Nemours and Company.

The integrity of the structural materials in the spray additive tank system would not be adversely affected even using the corrosion rates presented in Table 3 where air contaimination is present. In the Indian Point Unit 2 system, where nitrogen blanketing of the spray additive tank would prevent air contamination, the corrosion rates would be even lower with even less effect on the material integrity.

Diamond Shamrock Company (Ref. 10) reported no galling of steel valves occurred after exposure to 50% sodium hydroxide at [illegible] to 140°F for greater than [illegible] years. Stainless steel valves, exhibiting lower corrosion rates, would have an even lower propensity toward galling than steel. Therefore, no galling should occur on the valves exposed to the long term storage conditions.

The total corrosion product released to the spray additive tank as oxide would be less than 1000 grams per year with aerated solution and would be much less with the air free solution, i.e., the Indian Point Unit 2 solution.

This small quantity of corrosion product should not present any problems with clogging of delivery lines.

No sodium hydroxide precipitation will occur for a 30 w/o solution if the temperature of the tank and liners are maintained above 35°F. Since this system is located in an area of the auxiliary building which is continuously heated to maintain a 50°F minimum temperature, no solid sodium hydroxide would be present and therefore no clogging of the lines could occur.

Table 1

Exposure Conditions

Temperature, °F	110
Nitrogen Overpressure	slight positive pressure
Sodium Hydroxide Concentration, w/o	30
Oxygen Concentration - Normal	nitrogen blanketed
Carbon Dioxide Concentration - Normal	nitrogen blanketed

Table 2

Component Materials

Component	Material
Spray Additive Tank	304 stainless steel cladding on steel A-516 GR-70
Piping	304 stainless steel
Valve Bodies	304 and 316 stainless steel
Valve Seats	austenitic stainless steel or Stellite
Valve Stems	17-4 PH and 410 stainless steel
Valve Diaphragm	Ethylene-Propylene Dipolymer (Nordel Rubber by Dupont)

Table 3

Corrosion Rates

Material	Temperature, °F	NaOH Concentration, ppm	Aeration	Corrosion Rates, mils/yr	Reference No.
304 S/S	136	22 to 50	Yes	< 0.1	1
316 S/S	125	30	Yes	< 2	2
Steel	179	30 to 50	Yes	< 20	2
410 S/S	125	30	Yes	< 2	2
17-4 PH	176	30	Yes	3 to 6	7
Stellite	150	50	Yes	< 0.6	4
Nordel Rubber	110	33	Yes	< 0.004	5

REFERENCES

Reference 1 - A Guide to Corrosion Resistance, J. P. Polar (Climax Molybdenum).

Reference 2 - Corrosion Data Survey (1960 Edition) (Shell Development Company).

Reference 3 - Resistance of Huntington Alloys to Corrosion (Huntington Alloy Products Division of International Nickle Company, Inc.), page 28.

Reference 4 - Metals Handbook, 8th Edition, Vol. 1, Properties and Selection of Metals, page 670 (American Society for Metals).

References 5 - From unreported work performed at W NES laboratories.

Reference 6 - The Stress Corrosion of Metals by H. L. Logan, John Wiley & Sons, Inc., N.Y., 304 and 316 Stainless Steel, page 138, 410 Stainless Steel, page 101, A-516 - GR-70, page 44.

Reference 7 - Letter from R. R. Gaugh, Armco Steel on Date from an Armco Internal Report. Dated September 26, 1969, to D. D. Whyte.

Reference 8 - Corrosion Causes and Prevention by F. N. Speller, McGraw Hill Book Company, Inc., New York, 1951, page 195.

Reference 9 - The Corrosion and Oxidation of Metals by V. R. Evans, Edward Arnold Publishers, Ltd., London, 1960, page 454.

Reference 10 - Personal communication with Robert Sheppard, Assistant Plant Manager, Divisional Technical Center of Diamond Shamrock Company, Painsville, Ohio.

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TEMPERATURE - CONCENTRATION RELATION FOR CAUSTIC CORROSION OF AUSTENITIC STAINLESS STEEL

(AFTER SWANDBY, R.K. CHEM. ENG. 69, 186 NOV. 12, 1962)

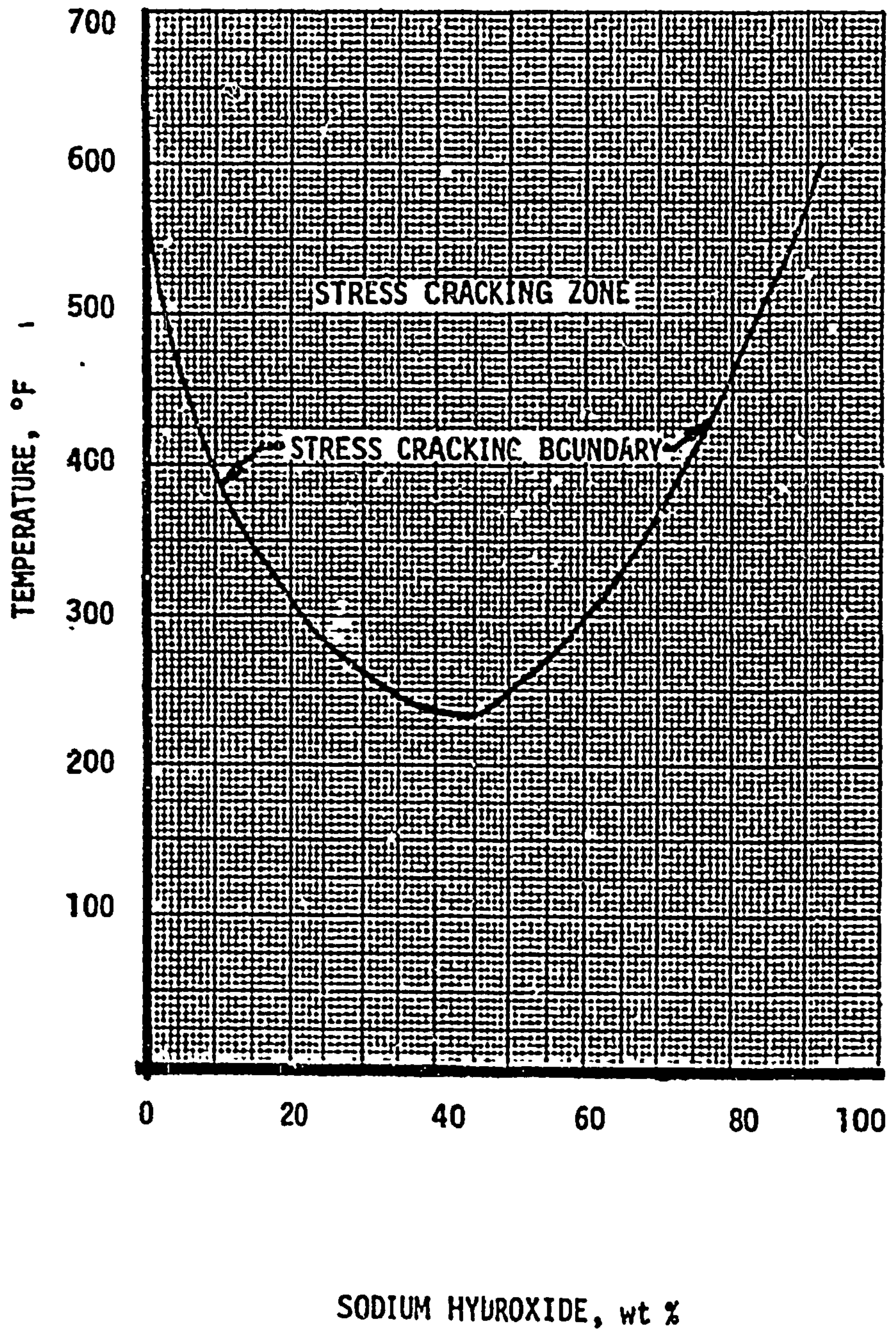

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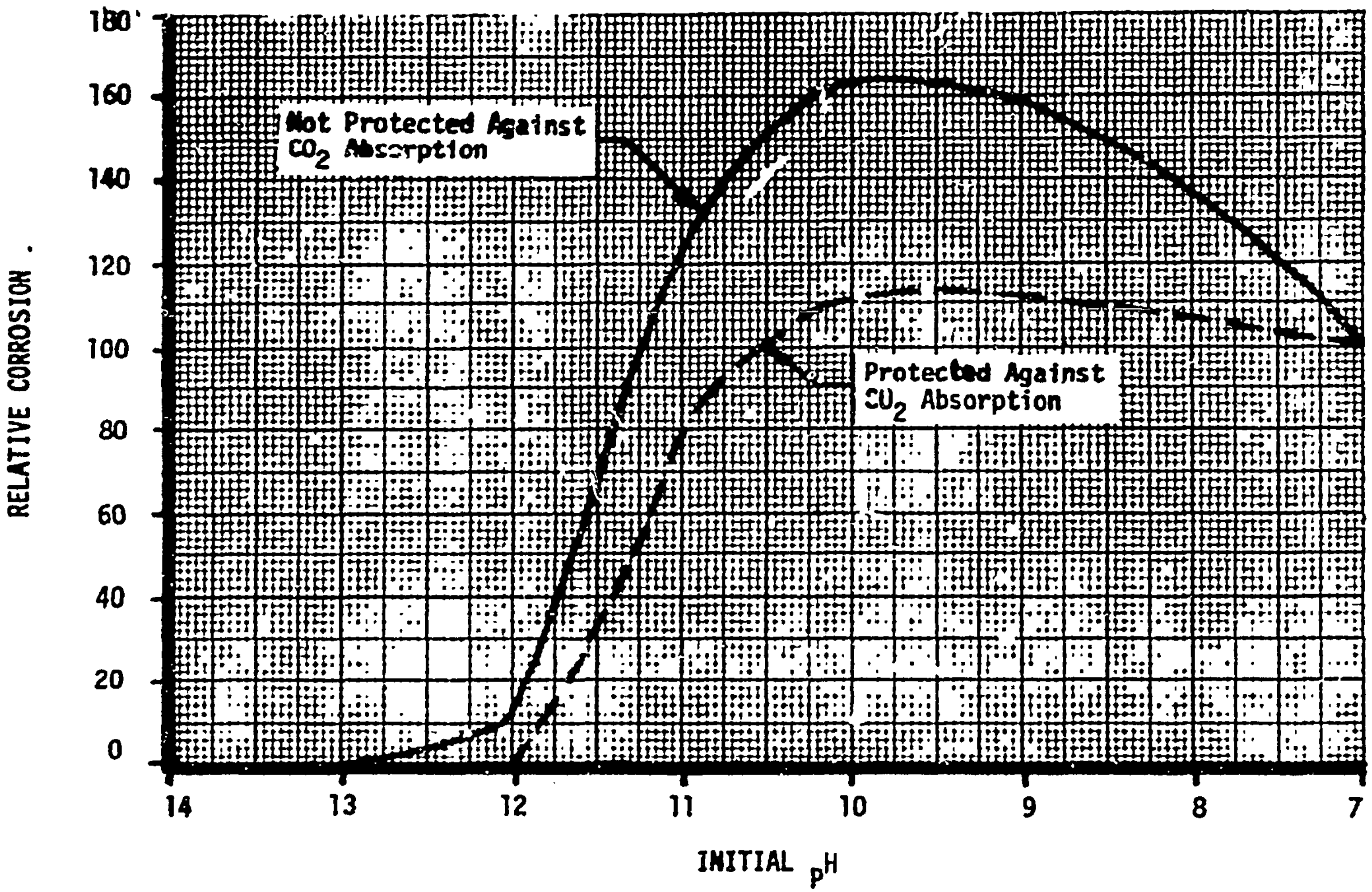
FIGURE 6.4-1

FIGURE 6.4-2 EFFECT OF CARBON DIOXIDE ON CORROSION OF IRON IN NaOH SOLUTION



Question 6.5

Describe the pre-operational and in service test programs and schedules for the chemical additive spray system.

Answer

Pre-operational

The principal components of the Containment Spray System are two pumps, one spray additive tank with associated eductors, spray ring headers and nozzles, and the necessary piping and valves.

In discussing pre-operational testing and generally proving that the system will meet the design specification it is necessary to consider both individual component testing and on site testing.

Off-site work

Three components in the system are subjected to off-site test work:

a. the spray pumps
b. the spray nozzles
c. the eductors.

a. The spray pumps have been subjected to conventional acceptance tests and the performance characteristic plotted to illustrate that the pumps meet the design specification.

b. Spray nozzles - as part of the development work in support of Westinghouse plants a nozzle of the type used in the spray system has been subjected to a performance test to demonstrate and prove the nozzle characteristic, e.g. flow/pressure drop, droplet size spread of spray etc.

As part of the quality assurance program, a random 25% of the nozzles to be installed at the Indian Point Unit 2 site will be given a general performance test.

c. Eductors - the eductors are produced and tested in two stages.

1. A prototype is made to check nozzle calculations prior to manufacture of the stainless steel units.

2. A performance test on one of the finished stainless steel units is made by the manufacturer to confirm the capacity at the specified conditions. A sugar-water solution is used to simulate the 30% Sodium Hydroxide suction fluid.

On-site test work

The aim of on-site testing is to:

a. Demonstrate and prove that the system is adequate to meet the design pressure conditions; outside the containment this involves part radiographic inspection and part hydro-testing; inside the containment the spray headers will be subjected to 100% radiographic inspection.

b. Demonstrate that the spray nozzles in the containment spray header are clear of obstructions by passing air through the test connections.

c. To verify that the proper sequencing of valves and pumps occurs in initiation of the containment spray signal and demonstrate the proper operation of all remotely operated valves.

d. To verify the operation of the spray pumps; each pump will be run at shut-off and the mini-flow directed through the normal path back to the refueling water storage tank. During this time, the mini-flow will be adjusted to that required for routine testing.

e. Demonstrate the operation of the spray eductors. The eductor and spray additive system is checked by running, in turn, each spray pump on miniflow with the spray additive tank filled with water and open to the spray eductor suction. During drain down of the spray additive tank, the tank level and corresponding eductor suction flow is recorded via the system instrumentation. Finally, the system performance with water will be extrapolated to that with sodium hydroxide and the adequacy of the system thus verified.

In order to establish a reference eductor suction tests flow for routine testing of the system the above test will be made with the spray additive tank isolated and the eductor drawing water through the RWST/eductor suction test line.

Routine in Service Testing

The aim of the periodic test is:

a. To verify that the proper sequencing of valves and pumps occurs on initiation of the containment spray signal and demonstrate the proper operation of all remotely operated valves.

b. To verify the operation of the spray pumps; each pump will be run at shut-off and the mini-flow directed through the normal path back to the refueling water storage tank.

c. Demonstrate the operation of the spray eductors. With the spray injection valves and sodium hydroxide tank valves closed, each spray pump will be operated in the shut-off condition but with the mini-flow line open. The test line from the refueling water storage tank will be opened to permit water to be drawn through the sodium hydroxide injection line to the eductor suction. A flow rate meter in the sodium hydroxide injection line will indicate the test flow established during preoperational testing.

QUESTION 6.6

Describe the preoperational and inservice test programs and schedules for the internal recirculation filter system. Include a discussion of both system tightness and component efficiency.

ANSWER

The in place testing of HEPA filters is performed to demonstrate gasket and media integrity, and overall bank efficiency, rather than an investigation of individual pinhole leaks in the filter media.

Testing will be carried out using "cold" generated DOP generated by Laskin type atomizer nozzles.

The test procedure will be as outlined in the proposed standard Efficiency Testing of Air Cleaning System Containing Devices for Removal of Particulates which was drafted by USASI Task Group N5.2.11. This procedure is currently designated N101.7.3.

Large filter installations will be tested at approximately 20% of the full rated flow. Besides limiting the quantity of DOP to be introduced into the ventilation system and containment, this is the flow rate at which filter imperfections would most readily be noticable. At higher flow rates, the turbulant flow through pinhole leaks and other imperfections becomes proportionally less than the laminar flow through the media. Filters therefore increase in efficiency with increasing air flow rates. When an in-place test carried out in accordance with N101.7.3, shows an unacceptable efficiency, leakage paths can be detected by passing the aerosol through the system, and probing the downstream side of the bank of filters and mounting frame with a probe connected directly to the photometer.

Charcoal filters will not be contaminated with DOP, and will be removed from the system before any testing takes place.

For small charcoal filter installations, filter bank efficiencies would be determined using Freon 112, in accordance with the procedures described in

DP1082 "Standarized Nondestructive test of carbon beds for Reactor Confinement Applications." For large installations, the use of this procedure would necessitate the release of excessive amounts of Freon 112 within the containment. Due to problems of possible Fluoride formation, it is desirable to keep Freon contamination to a minimum.

Consequently, instead of introducing Freon into a fully operating ventilation system, charcoal filter installations will be tested a few cells at a time. The procedure will be to use a small temporary portable blower and duct on the inlet side, while checking for leakage on the downstream side of the installation with a Halogen leak detector. Any Freon pick up which may occur in the section of the filter under test will be released following the completion of the test and will have no effect on filter performance.

All of these procedures have been effectively used at the Robert Emmett Ginna site during recent filter installation tests.

The proposed inservice filter testing schedule is given in the Technical Specification.

QUESTION 6.7

Identify electrically operated equipment located in the containment that should be operable following a loss-of-coolant accident. For each item of equipment describe the anticipated operating cycle and the length of time the equipment must be operable.

ANSWER

In addition to the electrical equipment in the containment together with the environmental design criteria defined in the answers to Questions 7.8, 7.9 and 6.8, the following equipment is required to be operable for the period specified.

a) Fan Coolers (one year)
b) Recirculation Pumps (one year)

The fan coolers will operate continuously after the LOCA while the recirculation pumps will only be required to operate during the recirculation phase.

QUESTION 6.8

We understand that installation of the Westinghouse flame recombiner system is being considered for the Indian Point Unit No. 2 plant as an engineered safety feature in order to control hydrogen evolved within the containment following a loss-of-coolant accident.

a. Please clarify your intentions in regard to the above and provide information relating to the detailed design arrangement of the recombiner system in the Indian Point Unit No. 2 plant.

ANSWER

The flame recombiner system will be installed in the Indian Point No. 2 as an engineered safety feature in order to control the hydrogen evolved in the containment following a loss-of-coolant accident.

Inside the containment are two (2) full-rated flame recombiner systems. Each is capable of maintaining the ambient H_2 concentration at or below two volume percent (v/o). Each system consists of an air supply blower, a combustion chamber complete with main hydrogen burner, two ignitors(one a spare), pilot hydrogen burner and a diluent chamber and associated monitoring and control instrumentation.

An air duct from the main ventilation ring header directs a[illegible]eas 1000 CFM of air from the main header to the recombiner blower suctio[illegible] remaining portion of the 5500 CFM total fan capacity is sup[illegible]ed by local ambient air.) This arrangement assures a moving, well mixed air stream at all times to the recombiner suction. It delivers containment air to the combustor. From the combustor, air passes on to the diluent chamber which serves to reduce the unit exhaust temperature. The combustion chamber is fueled by an externally-supplied fuel gas, employing containment air as the oxident. Hydrogen in the containment air is oxidized in passing through the combustion chamber. H_2 gas is used as the externally supplied fuel in order that non-condensible combustion products are avoided which would cause a progressive rise in containment pressure. Oxygen gas is added to the containment atmosphere through a separate containment feed (not coupled to the combustor) to prevent depletion of oxygen in the air below the concentration required for stable operation of the combustor (12%). The oxygen injection point is in an air recirculation system duct which discharges into the reactor coolant pump and steam generator area. The connection to the duct is outside the crane wall. To maintain the ambient hydrogen concentration at or below 2 v/o at the maximum production rate, it is necessary to operate the main burner. Reduced main H_2 burner fuel flow or the pilot H_2 burner may be sufficient when the containment H_2 concentration is below 2 v/o. The ignitors provided for each system are redundant. The dilution chamber is located downstream of the flame zone where products of combustion are mixed with a large excess of containment air to reduce the temperature of gas leaving the system below 700°F. The air discharge ducts from the recombiner will be positioned upward. With a total air flow rate through the recombiner of approximately 5500 SCFM, the upward air velocity out of each duct will be approximately 46 feet per second. The recombiner air discharge ducts are located a minimum of 28 feet laterally and 13 feet above the nearest air intake to a fan cooler. This separation plus the buoyancy and jet effect of the hot air from the recombiner directed upward away from the fan cooler intakes will preclude bypassing of the recombiner exhaust into the fan cooler intakes.

The flame recombiner systems are located on the operating floor in the southeast and southwest quadrants approxi-mately 90° apart. The systems are designed to operate at ambient steam overpressures corresponding to 0-5 psig in the containment and to withstand the design basis transient environment prior to operation. The combustor units are rated at 350 SCFM of containment air input at 5 psig at 33-100% hydrogen recombination efficiency over the range of process variables which includes combustor outlet temperature, percent oxygen, steam and hydrogen in containment air, entrained water content and operation under main hydrogen burner or pilot burner.[1][2] Figure 6.8(a)-1 is an arrangement drawing of one of the hydrogen flame recombiner units showing the assembly of the combustion unit including the burners, ignitors and diluent chamber.

Figure 6.8(a)-2 is a layout arrangement drawing showing recombiner system components located inside the containment.

The oxygen and hydrogen facilities, including bulk gas supply, metering, piping and penetrations are separated to prevent any accidental interaction. Gaseous hydrogen and oxygen are supplied from tubes mounted on trailers parked in widely separated areas outside the Primary Auxiliary Building. Figure 6.8(a)-3 shows the oxygen truck supply station outside of the northeast corner of the PAB building. The hydrogen trailer is attached to the high pressure side of the waste disposal system hydrogen supply manifold through a standard commercial connector located near the outside of the northwest wall of the PAB building. Pressure is reduced immediately by the manifold pressure regulators and the low-pressure gas is then piped through the Fan House in double walled pipe to the recombiner control station in the Fan House, as shown in Figure 6.8(a)-4. Two control stations are provided, one for each flame recombiner unit. Each control station is connected to independent power supply buses. From the Fan House, the gas flows directly to the recombiners as indicated in Figure 6.8(a)-2. The oxygen trailer is attached to a standard commercial connector located outside the northeast wall of the PAB building. Oxygen flows through a pressure reducing station to the recombiner control station. Oxygen is bled into the containment vessel through a separate penetration to be mixed with containment gases by the main containment ventilating blowers.

Hydrogen fuel flow is controlled to produce a predetermined temperature at the outlet of the combustion chamber. The nominal outlet temperature for complete combustion of the ambient hydrogen passing through the chamber is 1400 to 1600°F with 1400°F being the nominal combustor adjustable operating setpoint.

Oxygen makeup to the containment is proportioned to hydrogen fuel flow so that net depletion of oxygen in the containment will not occur. However, it should be noted that the combustor will operator quite efficiently and with good stability down to 1[illegible]% oxygen in the inlet air. This will allow considerable initial operation before oxygen is required to be added to the containment. The recommended minimum oxygen concentration is based upon tests which indicate that although flame stability was maintained to about 8% oxygen by volume the efficiency of the unit (in either pilot or burner operating modes) declined under 12% O_2 by volume.

The decision to start or throttle the combustion system to the pilot hydrogen burner or to change the makeup oxygen ratio, is based on intelligence from containment air samples analyzed for hydrogen and oxygen. It is intended that the combustor will be ignited when the hydrogen in the containment atmosphere reaches about 2 v/o. It may be run full throttle until the hydrogen is reduced to about 1.5 v/o and then it may be cut back by reducing the amount of hydrogen fuel to the combustor or by putting the unit on pilot burner only.

Combustor ignition is provided by a capacitance-type system equipped with two surface gap plugs designed for operation in a wet environment. The capacitors, which are located inside the containment near the combustor, are designed to withstand the accident and then operate in the containment atmosphere. The plugs are located on the combustion chamber. Ignition leads into the combustor are completely housed in a conduit with the wire connections at the ignitor fields brazed at its pressure tight end. This provides absolute continuity from the capacitor to either of the plugs.

The combustor flow rate is governed by a damper valve. The position of the damper is selected at the initial checkout of the unit and the damper position is fixed in place to prevent change. The same flow rate is maintained during throttled-back operation. The blower motor is single speed; therefore, air flow through the combustor is fixed regardless of operation.

Design of the thermocouple system which indicates pilot flame and main burner ignition parallels that of the ignition system. Each combustor system contains two thermocouples (one a spare). Thermocouple leads are fabricated by a procedure similar to that used for ignitor system leads.

The unit is protected against deleterious effects of high temperature by automatic tripping of the fuel valve at a combustor temperature of 1525°F. Although test data showed satisfactory performance up to 1600°F, 1525°F provides ample margin for planned operation at 1400°F, above which no significant improvement in combustion efficiency was observed. (Detailed instructions call for a nominal combustor operating temperature of 1400°F, although the test report identified the range 1400 - 1600°F as satisfactory.)

Figure 6.8(a)-5 is a flow diagram of the two hydrogen combustor systems. It will be noted that the systems incorporate the following features for operational safety and reliability:

1. Two complete control systems for fuel gas.

2. Isolation provisions for each fuel gas line consisting of a check valve inside the containment and at least two series normally closed valves outside the containment.

3. A block-and-bleed provision for each fuel gas line to prevent inleakage when not in use.

4. Provisions for purging fuel lines with nitrogen from the Waste Disposal System cylinder manifold before introducing combustible gas.

5. Alarm functions to alert the operator in case of loss of blower pressure, low-combustor temperature (flameout) and low fuel gas or oxygen manifold pressure. If the combustor temperature falls 200°F below (less than) the adjustable operating temperature (normally 1400°F), the fuel supply is shutdown outside the containment. The purpose is to prevent introducing unrecombined hydrogen into the containment in the event of a flameout.

6. Capability to test the complete control systems at any time by carrying out a complete dry-run startup using artifically generated thermocouple signals to simulate lightoff.

7. Capability to test the system at any time by a complete test and verification of ignition. This test can be conducted from operating stations outside the containment.

8. Piping design in conformance with ASA-B-31.1 Code for Pressure Piping with double-walled hydrogen piping for seismic and leak protection.

9. Pneumatic control valves and a means to supply emergency air in the event of loss-of-instrument air.

10. System redundancy such that no single component failure can disable both combustor systems.

11. Instruments located at the hydrogen valve stands to detect hydrogen leakage.

12. Protection during startup for ignition failure. If, during startup of the unit, a combustor temperature of 450°F is not reached within 60 seconds after starting ignition the fuel supply valve is automatically closed. The setpoint prevents any significant quantity of unburned hydrogen from entering the containment.

Instruments, controls and suitable panels provided to perform all these functions in a safe and reliable manner. An operator will be stationed at the recombiner control area whenever that system is in operation or under test. All the controls are located in the Fan House at the southeast side of the containment. Radiation dose rates at the control panel nine days after the postulated accidents are less than .05 rem/hr. to the thyroid and 2.0×10^{-5} rem/hr whole body (· 0.4 rem thyroid dose for one 8 hour watch with no portable breathing apparatus). Assumptions in the calculations are the same as described in response to Question 14.2 for the design basis accident with a reduction factor of 10^{-2} in the whole body dose rate due to shielding effect of the building housing the control panel (for the hypothetical model, TID-14844 release model, the corresponding dose rates are 1.5 rem/hr thyroid and 2.0×10^{-5} rem/hr. whole body). All leakage is assumed terminated at 30 days at which time the dose rate at the station will be zero.

The following features are incorporated in the control system and panel design to ensure operational safety and reliability:

1. A separate panel for each recombiner system.

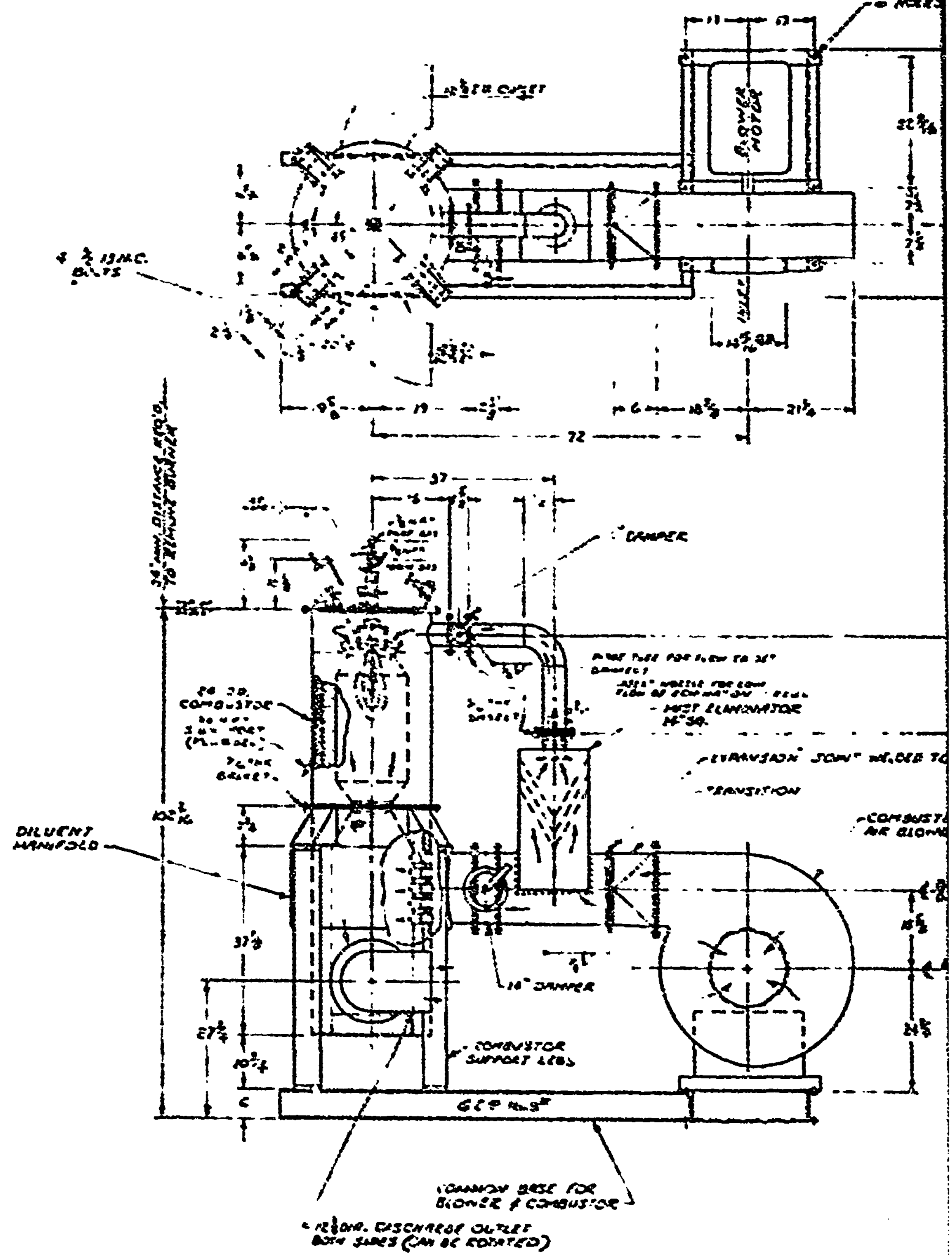
2. Physical and functional separation of redundant features such that no single feature can invalidate both features.

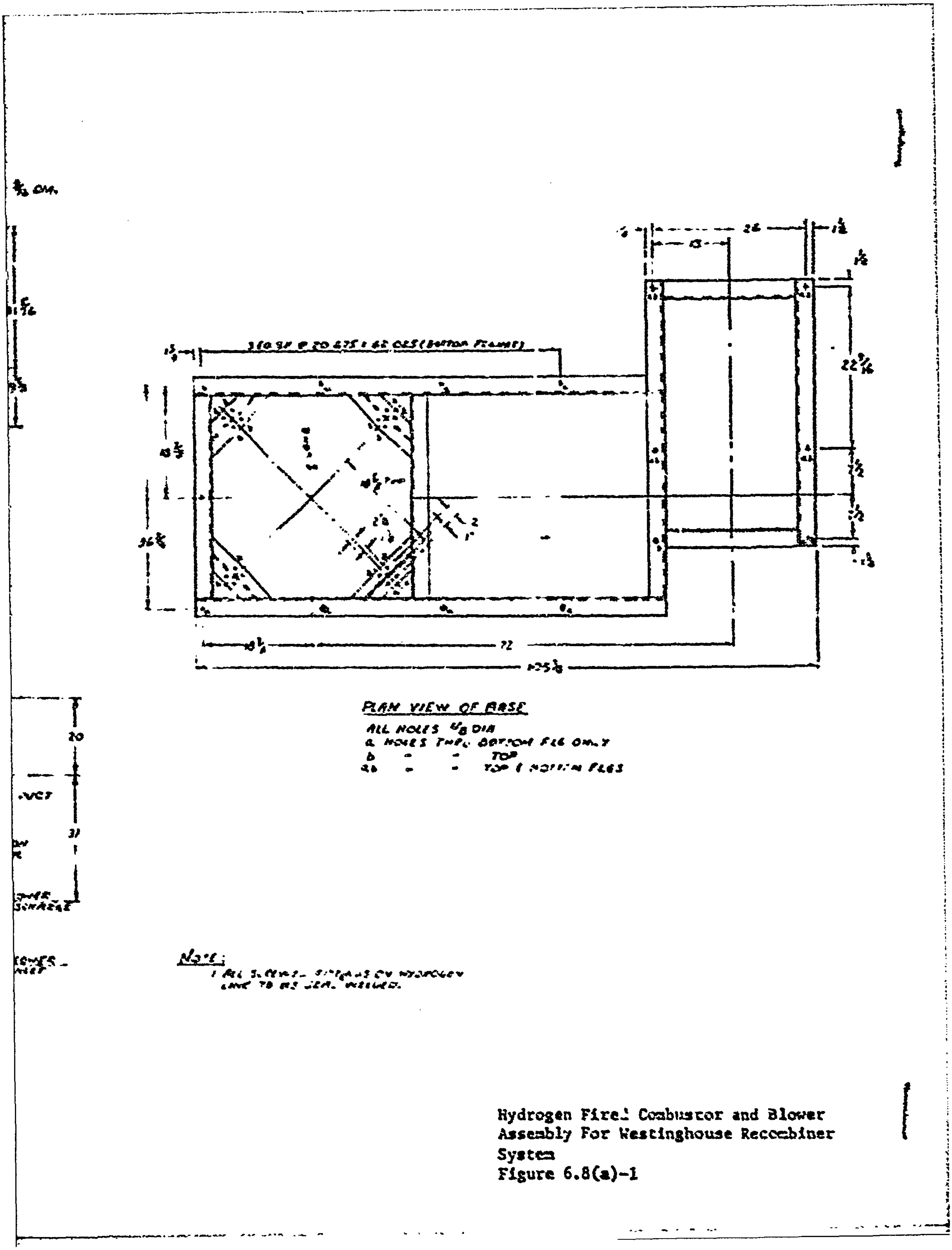
3. Provision to switch off and lock out power to the control system when the panel is unattended.

<u>References</u>

1. WCAP-9001, A Controlled Combustion System to Prevent Hydrogen Accumulation Following a Loss-of-Coolant Accident, February 1969, Proprietary Westinghouse Report.

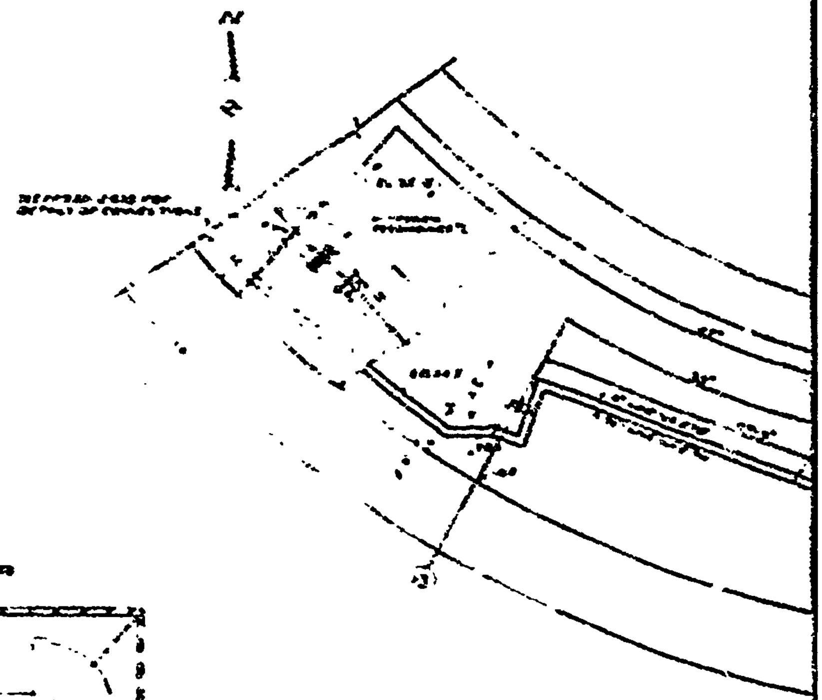
2. WCAP-7301-L, Report of Test Results on Hydrogen Flame Recombiner, submitted March 26, 1969, Proprietary Westinghouse Report.

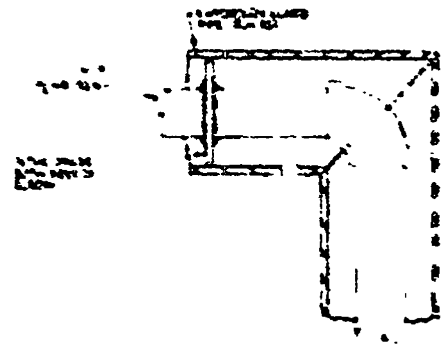

BLOWER MOTOR
DILUENT MANIFOLD
MIST ELIMINATOR
TRANSITION
DAMPER
14" DAMPER
COMBUSTOR SUPPORT LEGS
COMMON BASE FOR BLOWER & COMBUSTOR
BOTH SIDES (CAN BE ROTATED)
72
37



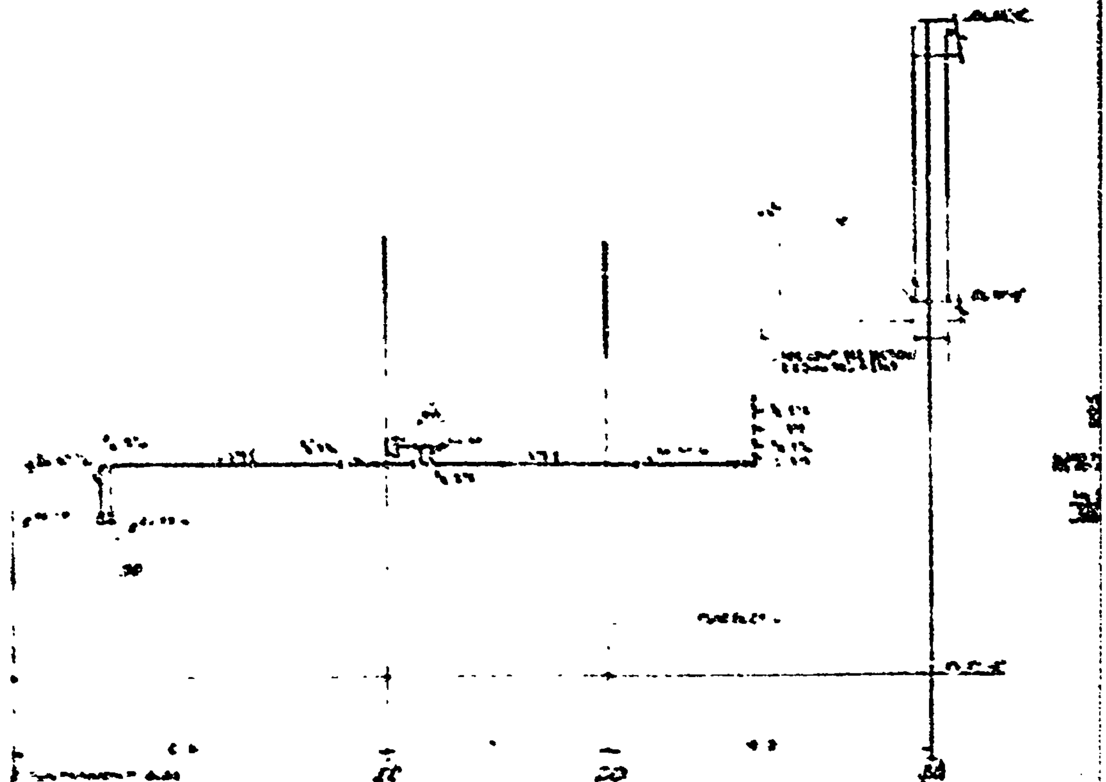
Hydrogen Fired Combustor and Blower Assembly For Westinghouse Recombiner System

Figure 6.8(a)-1

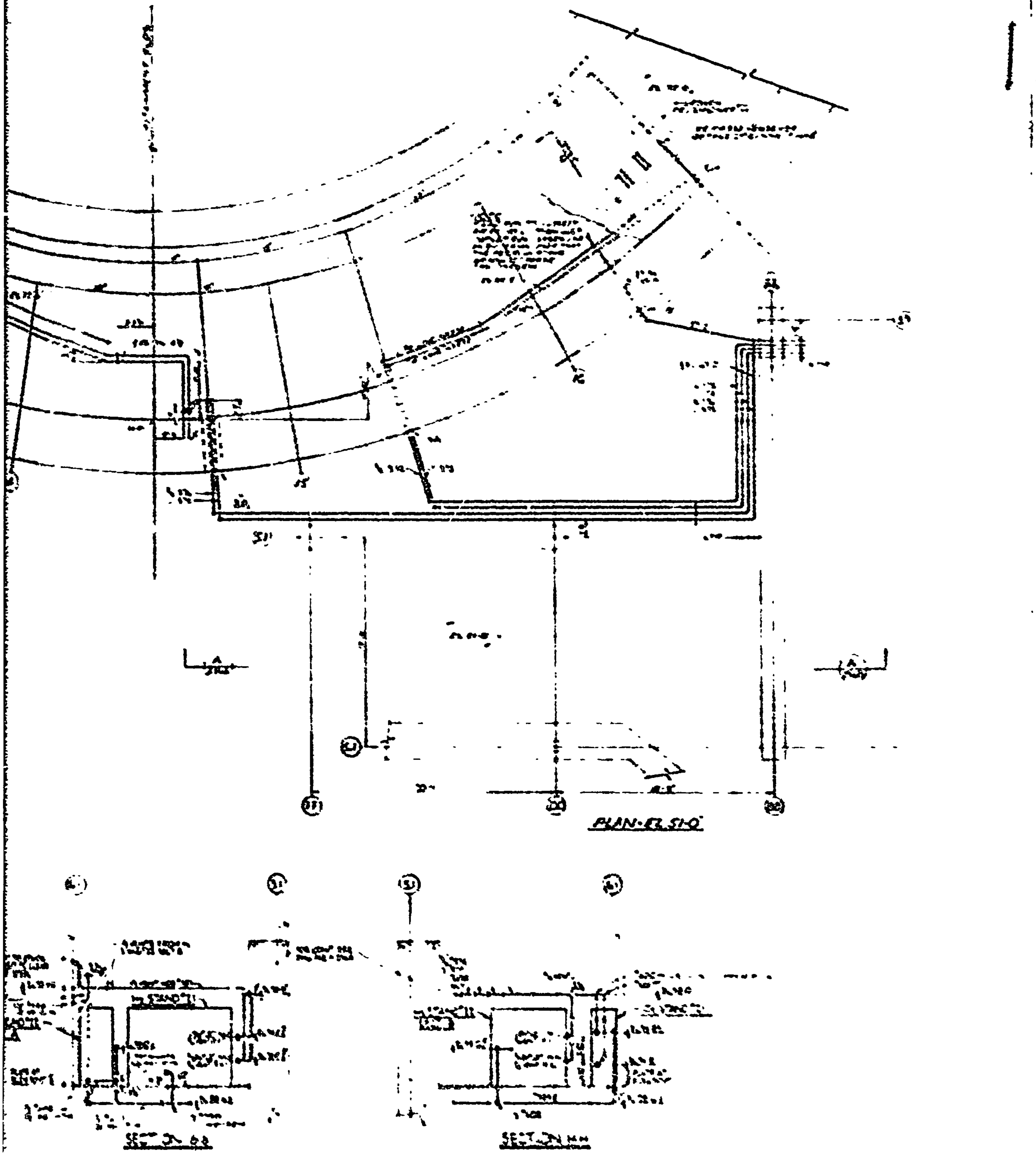




DETAIL A

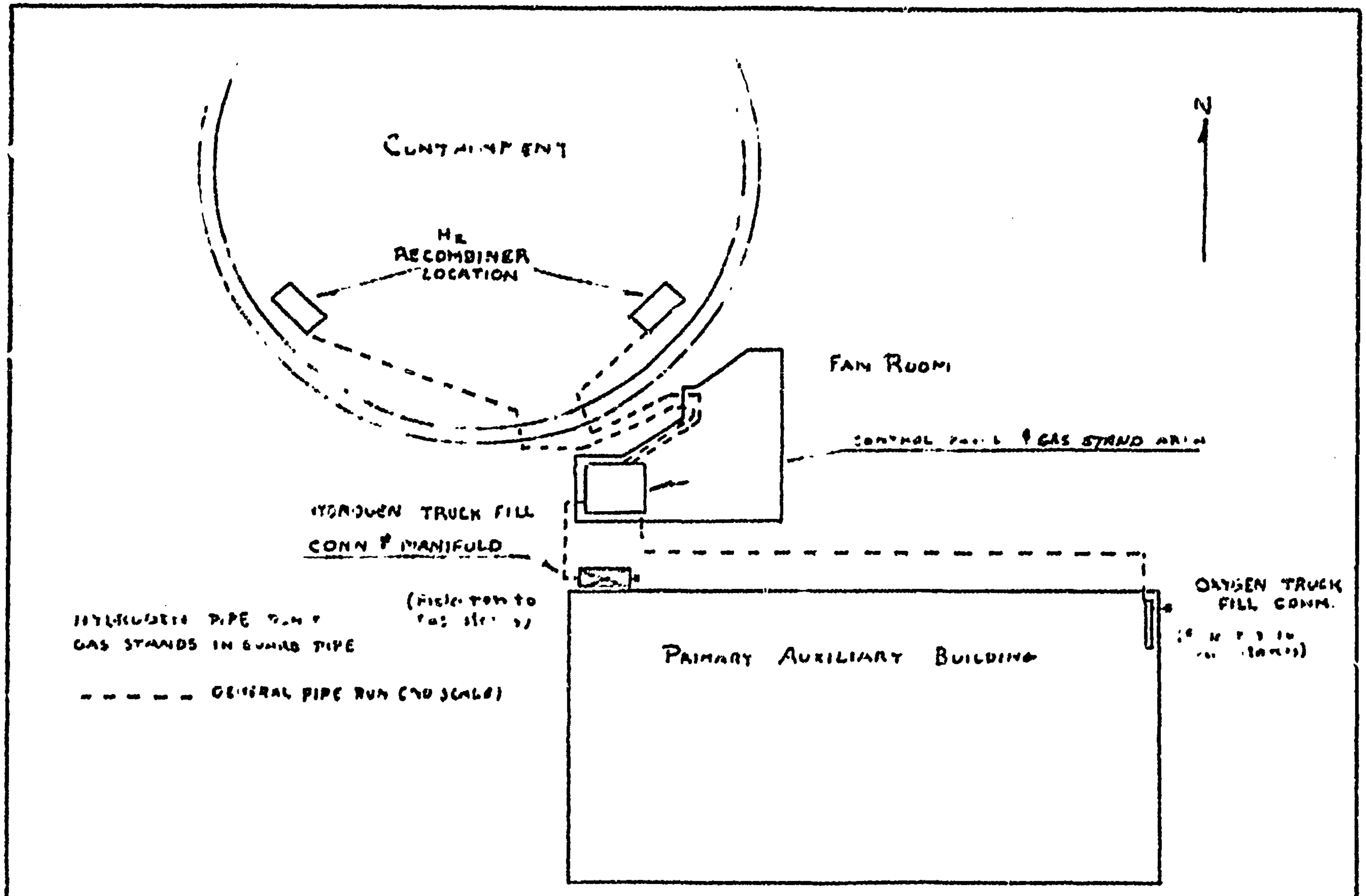


SECTION A-A



HYDROGEN RECOMBINER
PIPING SHEET NO. 1

FIGURE 6.8(a)-2



1" = 40' 0"

Figure 6.8(a)-3
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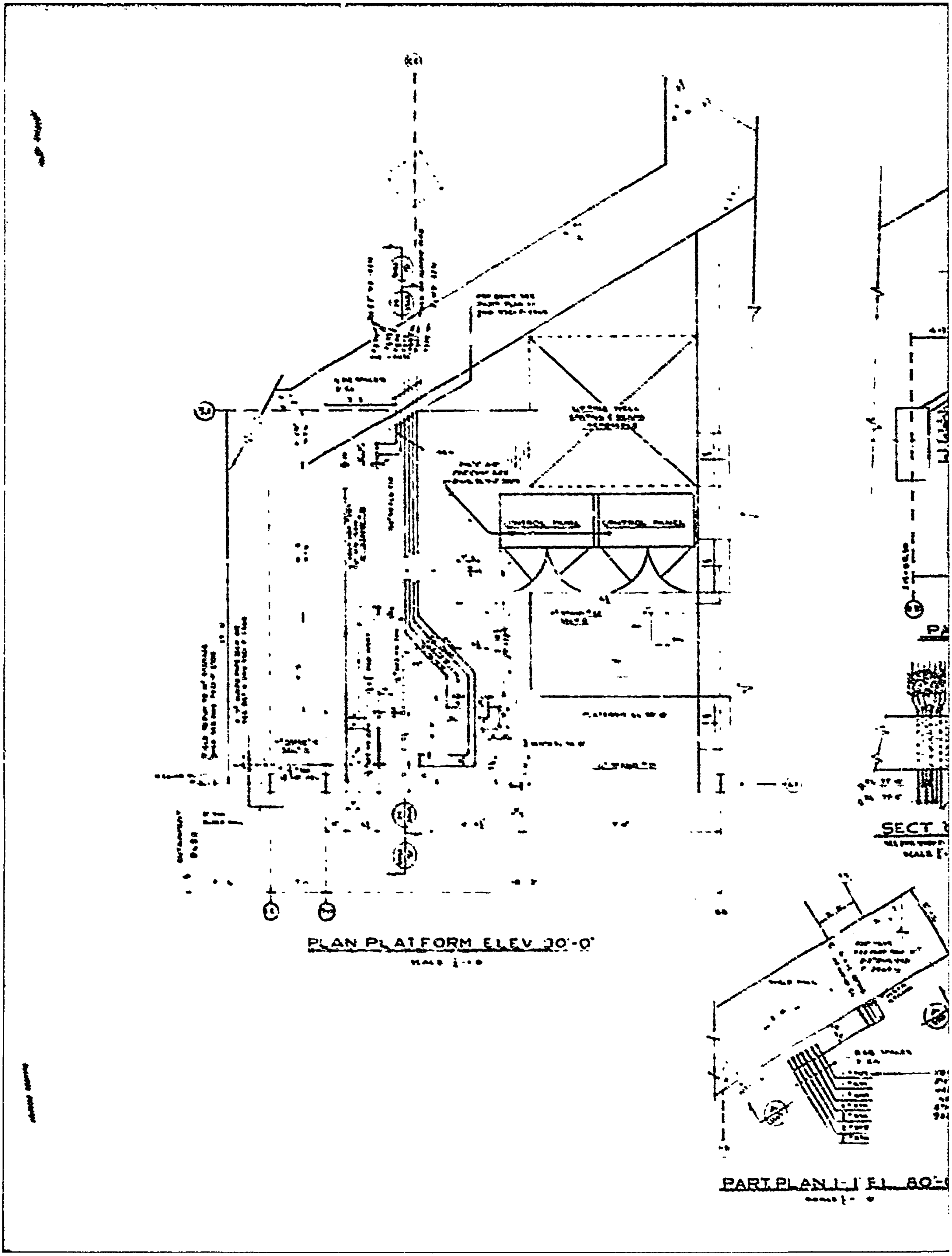
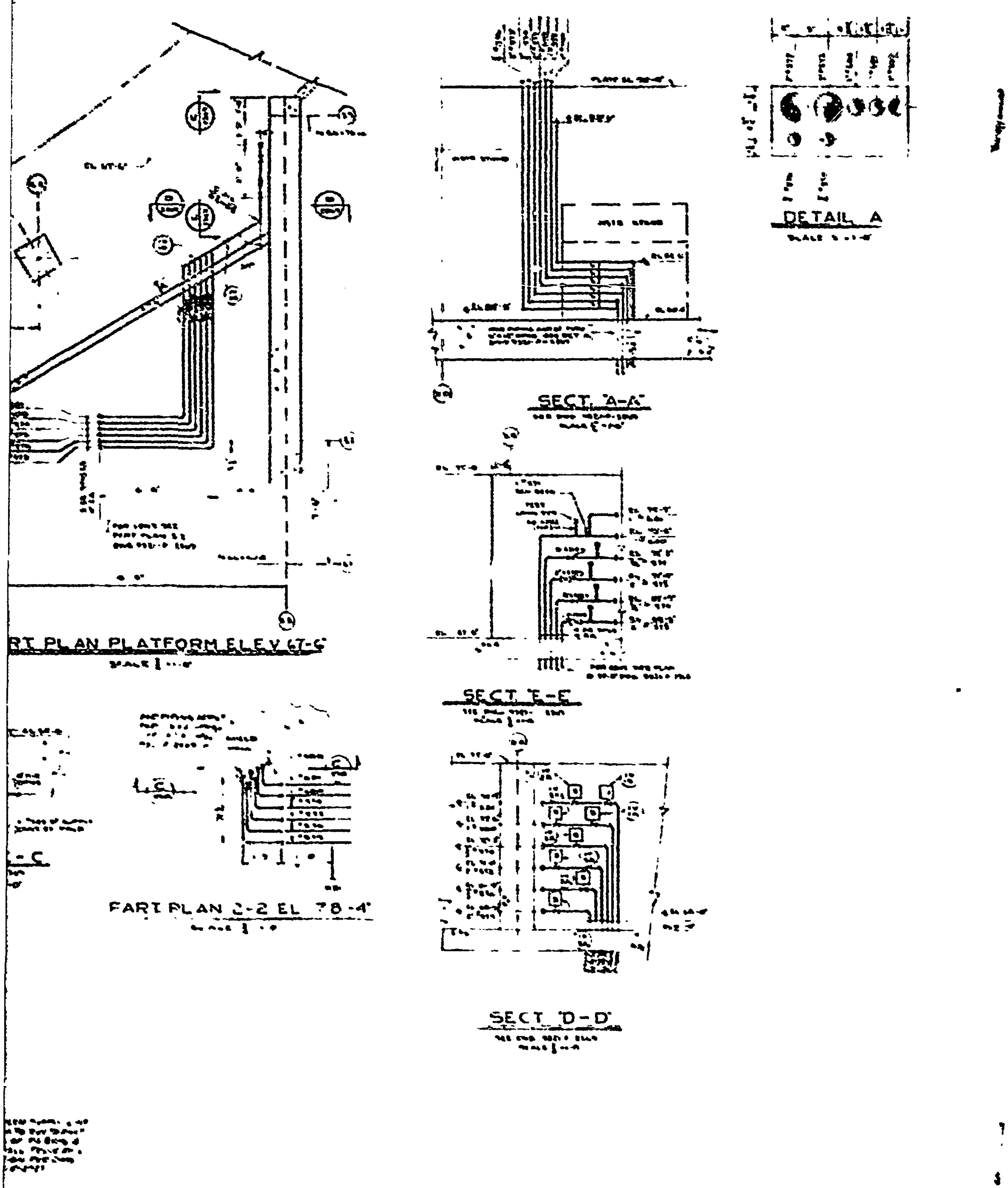
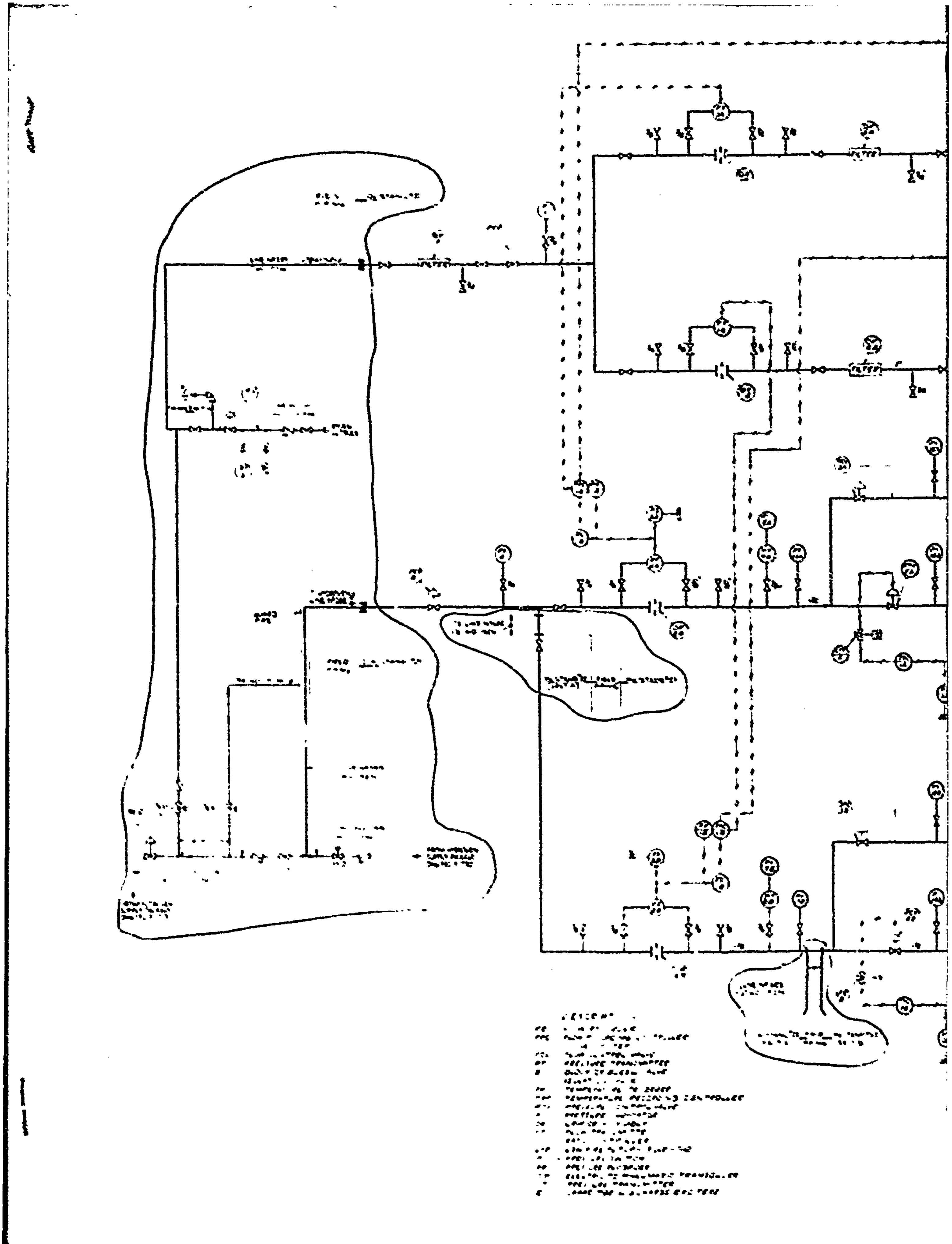
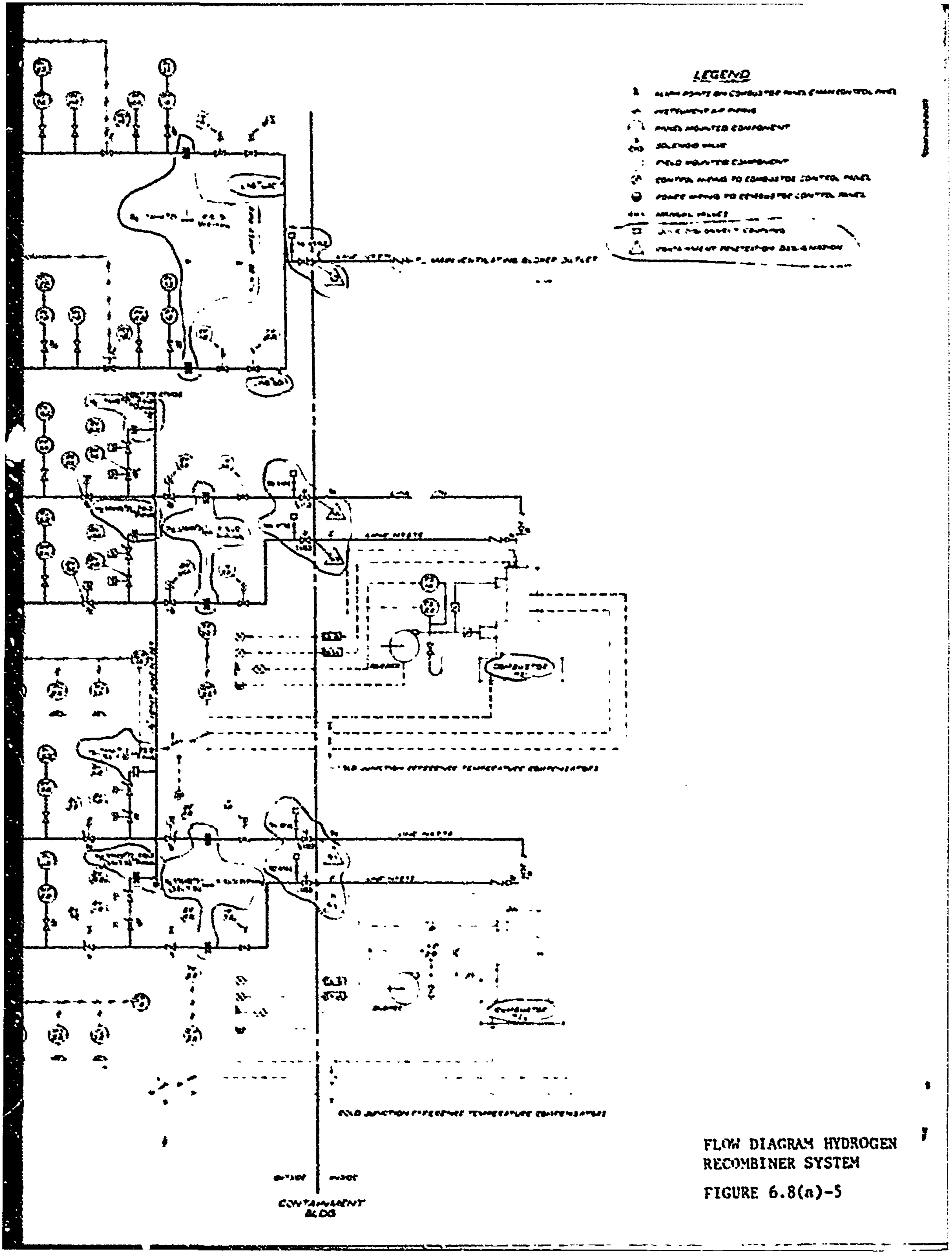

CONTROL PANEL
CONTROL PANEL
PLAN PLATFORM ELEV 30'-0"
SECT
PART PLAN 1-1 EL 80'-



HYDROGEN RECOMBINER
PIPING SHEET NO. 2

FIGURE 6.8(a)-4





FLOW DIAGRAM HYDROGEN RECOMBINER SYSTEM

FIGURE 6.8(a)-5

QUESTION 6.8

We understand that installation of the Westinghouse flame recombiner system is being considered for the Indian Point II plant as an engineered safety feature in order to control hydrogen evolved within the containment following a loss-of-coolant accident.

b. Provide suitable discussion and analyses to support the adequacy of the design bases for operation of the recombiner system. This should include, but not be limited to the following:

(1) Sampling procedures (liquid, gas), time to sample, location where measurements are taken, sampling errors, and stratification considerations.

ANSWER

The following is a description of the post accident containment sampling system planned for Indian Point Unit #2:

A sample line will originate in each of the reactor containment fan cooler units at a location downstream of the fan but upstream from the charcoal filter as shown in Figure 6.8(b).1-1. A pump will be used to provide sufficient head to bring containment air in a loop from the fan cooler through a containment penetration to a sample vessel outside the containment, and then through a second penetration to the sample line termination inside the containment. Before a sample is taken, the line will be purged by allowing containment gas to circulate through the loop for a few minutes. Samples will be taken in a closed, pressure tight vessel which can be removed from the line for transfer to the laboratory. Present plans are to provide an independent sample line from each reactor containment fan cooler unit up to and including the sample vessel location. Return lines into the containment may be headered together. The local sample station will be located as near to the containment penetrations as pratical.

Sampling should begin 24 hours and 48 hours post accident depending on other, more critical, demands for operator attention. The actual sampling operation should require only fifteen to thirty minutes per sample taken.

Sampling errors will be minimized with the planned arrangement because

1. several sample locations are provided, one at each fan cooler;

2. each sample location is in a continuously circulating, well mixed containment air stream;

3. the linear velocity of air flowing in the sample line is high enough to permit thorough purging of the line in only a minute or two;

4. there is adequate turbulence in the sample line to provide a well mixed sample at the sample vessel location.

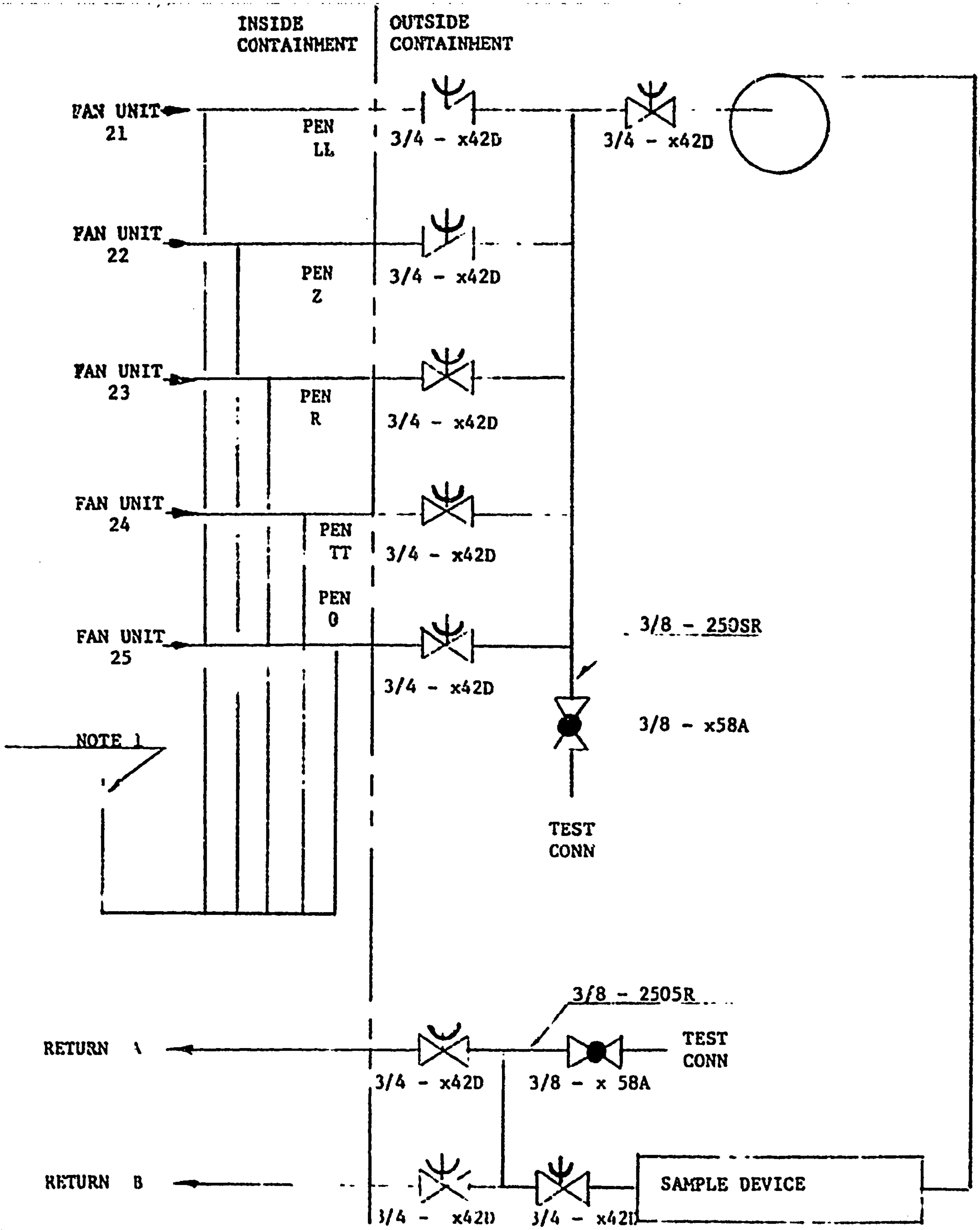
Analytical error is small in the concentration range of interest. Reproducibility is about ± 5% of the measured value when comparing the sample with a known gas by the gas partitioner method.

To assure that stratification effects or sample errors would not permit all or parts of the containment to hold hydrogen in excess of the lower flammable limit (4.1 v/o) when the measured concentration is 2.0 v/o, the following checks were made. First, it was determined that the minimum reliable air circulation rate by three of the main ventilating blowers within the containment had the capacity to recirculate the entire containment air volume on the average 4.8 times an hour (or 210,000 cfm). But the calculated hydrogen generation rate during the first day post accident is 16,300 scf yielding a ratio of air circulation to hydrogen generation in excess of 18,500:1. Due to the decreased rate of hydrogen generation with time, the ratio increases to an even greater value before the hydrogen concentration in the containment reaches two per cent. At the conservatively predicted generation rate, 24 hours are required to produce hydrogen in the amount of one per cent of containment volume. During this same period, the entire atmosphere of the containment has been recirculated on the average 115 times. Furthermore, the air handling system is designed to promote the interchange of air in all regions of the containment to avoid the possibility of accumulation of hydrogen in stagnant pockets or strata. For example, in the highest part of the containment dome (above the top spray ring), minimum air recirculation provides one air change approximately every 61 seconds. For these three reasons it is concluded that the stratification error is negligible.

Unless the spatial variation of the H_2 concentration is high or it changes rapidly with time, the sample taken from the recirculation air cooler units will be very representative of the local concentration near the recombiner. Neither exception can be considered reasonable as long as forced circulation of air is maintained and the production rate of H_2 is of the magnitude for which the recombiner is designed. Note that in the event the fuel mixture fails to ignite, the automatic control system provides a delay period for sweeping away any pocket of unburned fuel H_2 before the cycle is repeated.

Based on the fore going discussion, it is concluded that the two volume percent design concentration for operating the recombiner provides more than adequate margin for error associated with sampling the containment atmosphere. The calculated containment hydrogen concentration does not reach two volume precent until 13 days post accident, so it is highly unlikely that any significant concentration gradient will exist in the containment when the recombiner is started. Furthermore, since tests have been run with a full scale recombiner system at hydrogen concentrations up to and including 3.5 volume precent hydrogen, a hydrogen concentration between 2 and 3.5 volume percent at the recombiner suction would have no adverse effect on the recombiner operation.[1]

1. WCAP-7310-L, Report of Test Results on Hydrogen Flame Recombiner, submitted March 26, 1969, proprietary Westinghouse report.



NOTES:

1. Stand pipe at least 3 ft. high
2. All Tube 3/4 in. class 153R unless marked otherwise

INDIAN POINT UNIT POST ACCIDENT CONTAINMENT SAMPLING SYSTEM
Figure 6.8(b) 1-1

QUESTION 6.8

We understand that installation of the Westinghouse flame recombiner system is being considered for the Indian Point II plant as an engineered safety feature in order to control hydrogen evolved within the containment following a loss-of-coolant accident.

b. Provide suitable discussion and analyses to support the adequacy of the design bases for operation of the recombiner system. This should include, but not be limited to the following:

   (2) Systems failure mode analyses including the built-in protective and failure mitigating devices.

ANSWER

There will be two (2) full rated hydrogen recombination systems installed in the Indian Point Unit No. 2 plant. Either system is capable of preventing the hydrogen concentration from exceeding 2% by volume within the containment. Each of the hydrogen recombination systems is separate from the other.

Each system consists of the following major items.

a. Combustor unit located inside of the containment.
b. Control panel located outside of the containment.
c. Hydrogen gas stand located outside of the containment.

Each combustor system includes two (2) ignitors (one is a spare) with an excitor for each ignitor. Each combustor has two (2) temperature detectors (one is a spare) to monitor combustor temperature. Each combustor has two (2) differential pressure measuring instruments to detect blower operation, only one instrument is required for operation. Power supplies for the blowers and ignitors are separate, so that loss of one power supply will not affect the remaining system.

The control system for the hydrogen recombiner has the following built-in protective devices:

A. Shutdown Devices

1. Flame failure system will shut off hydrogen flow under the following conditions:

   a. If combustion temperature does not reach 450°F within one minute after pilot is lit,

   b. If combustion temperature falls 200°F below set point,

   c. If combustion temperature exceeds 1525°F.

2. Blower detection system will shut off hydrogen flow if the blower differential pressure drops below set point.

3. Loss of A-C voltage will shut off hydrogen flow due to loss of blower differential pressure and the fail safe position of the solenoid block and bleed valves.

B. Alarm Devices on hydrogen recombiner control panel (These alarms will also indicate on the main control board)

1. A. C. Power Failure
2. Instrument Air Pressure Failure
3. Oxygen Pressure Drop
4. Hydrogen Pressure Drop
5. Panel Access Door Open
6. Oxygen Pressure Low
7. Hydrogen Pressure Low
8. Low Pilot Temperature
9. Low Combustor Temperature
10. High Combustor Temperature
11. Low Flow Combustor Air
12. Hydrogen Leak Detection Outside Containment.

QUESTION 6.8

We understand that installation of the Westinghouse flame recombiner system is being considered for the Indian Point II plant as an engineered safety feature in order to control hydrogen evolved within the containment following a loss-of-coolant accident.

b. Provide suitable discussion and analyses to support the adequacy of the design bases for operation of the recombiner system. This should include, but not be limited to the following:

(3) Fuel system supply; the handling arrangements, logistics, and availability requirements.

ANSWER

Hydrogen fuel for the recombiners is stored in small quantities (standard gas cylinders) sufficient for periodic testing. These cylinders are not connected to the fuel supply headers except during the test and in accordance with test procedures.

In the event of its need following an accident, hydrogen fuel would be brought to the site to permit starting the process by the thirteenth day. This schedule would prevent H_2 from exceeding 2 v/o in the containment. Subsequent operation will require an average fuel consumption of 15,000 SCF per day over the next 90 days, and about half as much oxygen, to maintain the ambient H_2 at less than 2 v/o. Bulk gas would be delivered in trailer mounted tubes at 60-100,000 SCF per load, requiring about fifteen such deliveries of H_2 during that 90 day period, and seven similar deliveries of O_2. Sufficient H_2 could be brought to the site in approximately 5 days following a DBA to begin combustor operation if required.

QUESTION 6.8

We understand that installation of the Westinghouse flame recombiner system is being considered for the Indian Point II plant as an engineered safety feature in order to control hydrogen evolved within the containment following a loss-of-coolant accident.

b. Provide suitable discussion and analyses to support the adequacy of the design beses for operation of the recombiner system. This should include, but not be limitied to the following:

(4) Post-installation checkout and evaluation of the recombiner system, including the planned processing setpoints.

(5) System testing procedures and frequency.

ANSWER

After installation the system will be checked out by operating the combustor. At this time the damper position and blower differential pressure set point will be established, combustor temperature confirmed, ignitor operation confirmed and control panel and gas valve stand operation confirmed. After initial check out, an operating check of the system will be performed periodically, as specified in the Technical Specifications.

Operating procedures for the system for under accident conditions are outlined below:

I. Piping System Purge

A. Purpose

This instruction describes procedures for purging air from the recombiner systems hydrogen piping.

B. Initial conditions

1. The Waste Disposal System nitrogen supply manifold is operational.

2. Containment pressure is 5 psig or less.

3. Containment temperature is 155°F or less.

C. Instructions

1. Check all valves and switches to be sure they are in the normal startup postion.

2. Insert the spool piece between the Waste Disposal hydrogen manifold (low pressure side) and the hydrogen recombiner fuel line.

3. Open the manual shut-off valve and regulation to admit nitrogen from the Waste Disposal nitrogen manifold (low pressure side) to the hydrogen recombiner fuel line.

4. Switch Power on the control circuits.

5. Adjust pressure control valves so that nitrogen gas pressure is 5 psi above containment pressure.

6. Place Temperature Recording Controller on manual control and open flow control valve to recombiner.

7. Clear the containment trip signal from isolation valves and obtain permission from the main control room to open the valves.

8. Close bleed valves and open block valves in the pilot line.

9. Open isolation valve in one header.

10. Purge sufficiently well to reduce the oxygen level in the hydrogen piping.

11. Close isolation valve and return block and bleed valves to "Auto".

12. Close bleed valves and open block valves in the main fuel line.

13. Open isolation valve in main fuel line and purge.

14. Adjust flow control valve to set pressure at 5 psi above containment pressure. Maintain during purge.

15. Close isolation valve and return the block and bleed valves to "Auto".

16. Return temperature recording controller to "Auto".

17. Switch off power to recombiner control circuits.

II. Recombiner Operation

A. Purpose

This instruction describes procedures for operating the containment hydrogen recombiner system to maintain the hydrogen concentration at a safe level.

B. Initial conditions

1. Containment pressure is 5 psig or less.

2. Containment temperature is 155°F or less.

3. The hydrogen content of the containment atmosphere has been sampled and anlayzed such that analytical results consistantly demonstrate that there is no discernable trend for the hydrogen level to reach or exceed a true value of 4.1% by volume before the system can be placed in service.

4. A supply of hydrogen has been delivered to the site and has been attached to deliver gas to the recombiner system.

5. The containment oxygen concentration is 12% by volume or greater.

6. At least two containment circulating fans are operating and have been operating continuously from the time sampling of containment atmosphere started.

7. The Waste Disposal hydrogen and nitrogen supply manifolds are operational.

8. If oxygen is to be made up to the containment, a supply must have been delivered to the site and attached to deliver gas to the system.

9. The hydrogen lines have been purged, and the pressure in the piping has continuously exceeded atmospheric pressure since purging. Otherwise, purging must be repeated.

C. Instructions

1. Isolate the recombiner fuel line from Waste Disposal nitrogen supply.

2. Open the shut-off valve between the recombiner fuel line and the Waste Disposal hydrogen supply.

3. Select one recombiner for operation and close valves in branch lines leading to the other unit.

4. Switch power onto the recombiner control circuits.

5. Confirm that fuel line pressure exceeds containment pressure; confirm block and bleed valves in "Auto".

6. If oxygen is to be added at this time, the following steps are followed.

   a. Open the shut-off valve in the oxygen line branch associated with the proper recombiner. Co ...rm that the valve in the other branch is closed.

   b. Set ratio control at two volumes hydrogen to one volume oxygen. (Reset as needed based on containment atmosphere analysis.)

   c. Open valving from truck to pressurize the o· ygen line.

   d. Confirm pressure in pipe is greater than containment pressure.

   e. Obtain permission from main control room to open isolation valves.

7. Select one of the two igniter circuits for operation and put the igniter in "Auto."

8. Select the control thermocouple to be used.

9. Open isolation valves.

10. Start the blower.

11. Set temperatue recording controller to control at 800°F.

12. When the "end of purge" light comes on, press the "Flame-Start" button.

13. When steady main burner lightoff is proved as indicated by a steady thermocouple readout, bring the combuster temperature up to the desired operating temperature over a 10 minute period by adjusting temperature recording controller in 100°F increments until 1400°F is reached.

14. If oxygen is being added at this time, confirm pressure in header is above containment pressure and open isolation valves.

15. When subsequent samples of containment atmosphere indicate that the hydrogen concentration has been reduced to the desired level and the system is to be shutdown,* close isolation valves.

16. Press the "flame-stop" button.

17. Confirm that the valves are in the proper position.

18. Turn off the blower when combustor temperature reaches 500°F.

19. Switch power off to panel contro circuits.

20. Set the adjustable alarms for hydrogen and oxygen pressure at the cylinders (rack) at a position 50 psi below the pressure in the downstream header.

21. Close the manual shut-off valves around the spoolpiece connecting the hydrogen supply and fuel line, and remove the spoolpiece.

* It is recommended, in order to reduce thermal shock, that the combustor temperature be reduced step by step by adjustment at set point down to 500-600 of in 10-15 minute period.

QUESTION 6.8

We understand that installation of the Westinghouse flame recombiner system is b' ɿ considered for the Indian Point II pl‐.‿t as an engineered safety feat ⁾n order to control hydrogen evolved wi. .in the containment following a loss-of-coolant accident.

c. Discuss the capability of the various components of the recombiner system, e.g., motors, valves, ignitors, and instrumentation, to withstand the post-accident environment (i.e., pressure, temperature, moisture, radioactivity, and corrosive chemical conditions) and remain functional. Identify the various system components subject to such environmental conditions which must remain functional for satisfactory recombiner initiation and operation. Indicate for these components, the test data or other applicable evidence to support the expected functional capability. Discuss the expected operating lifetime requirements and the design lifetime of the recombiner system.

ANSWER

The components of the hydrogen recombiner system whic.. will be exposed to the post accident environment are as follows:

a. Combustor which includes the motor, blower, ignitors and temperature detectors.

b. Ignitor excitors

c. Pressure differential measuring instruments

1. A 2 hp motor (constructed as a recombiner motor) has been irradiated to an integrated dose of 2×10^8 rads In addition, the motor was thermally aged to an equivalent of 7 years of operation

The motor with a shaft mounted blower unit to provide loading was tested to the environmental test conditions shown in Figure Q 6.8(c)-1 and Figure Q 6.8(c)-2. The test was conducted over a three-week period. During this period, the motor was operated on a daily schedule of 8 hours on, 16 hours off. One week into the test, it was discovered that the motor had been partially flooded due to a blockage in the chamber drain valve (this condition discovered when it was found that the motor was drawing ~ 3400 watts). The chamber was then drained and the te . continued for the required total of three weeks.

The results of this test indicate that the recombiner motor will perform its required function following the loss-of-coolant accident.

2. The ignitor exciter and pressure differential measuring instruments have been tested in simulated post accident environment. Following supplier pressure (90 psig) testing of an exciter igniter unit, the unit was exposed to an integrated radiation dose of 1.7×10^8 rads. The results of the irradiation testing indicate that although the unit has been degraded by irradiation (decrease in pulsing rate from 8 pulses/sec down to 7 pulses/sec) the unit continued to operate satisfactorily.

   Following irradiation testing, the unit was exposed to the environmental test conditions shown in Figure Q 6.8(c)-1 and Figure Q 6.8(c)-2 and operated intermittently during a 3-week test period. The results of these tests indicate that the unit will perform its required function following the loss-of-coolant accident.

3. The blower surfaces exposed to containment spray are painted with Carboline Company s Phenoline 355 Oven CZ 11 primer to provide resistance against any corrosive environment. This is true also for the exposed surfaces of the combustor unit.

4. The ignitors and temperature detectors were tested with the combustor through a wide range of operating conditions.(1) The ignitors have also been tested under water.

(1) WCAP 7301-L "Report of Test Results on Hydrogen Flame Recombiner - W Proprietary Class 2.

The maximum expected required operating time for the recombiner system is 250 hours with 2% by volume of hydrogen in the containment. If the combustor is operated only on pilot, the expected operating time would be 2160 h urs with 2% by volume of hydrogen in the containment. (Above hour values are exclusive of testing) The expected motor life would far exceed either of the above numbers. The ignitor life time is 5000 to 10000 starts which is considerably more then the number of starts that could be expected for testing and operation of the recombiner system.

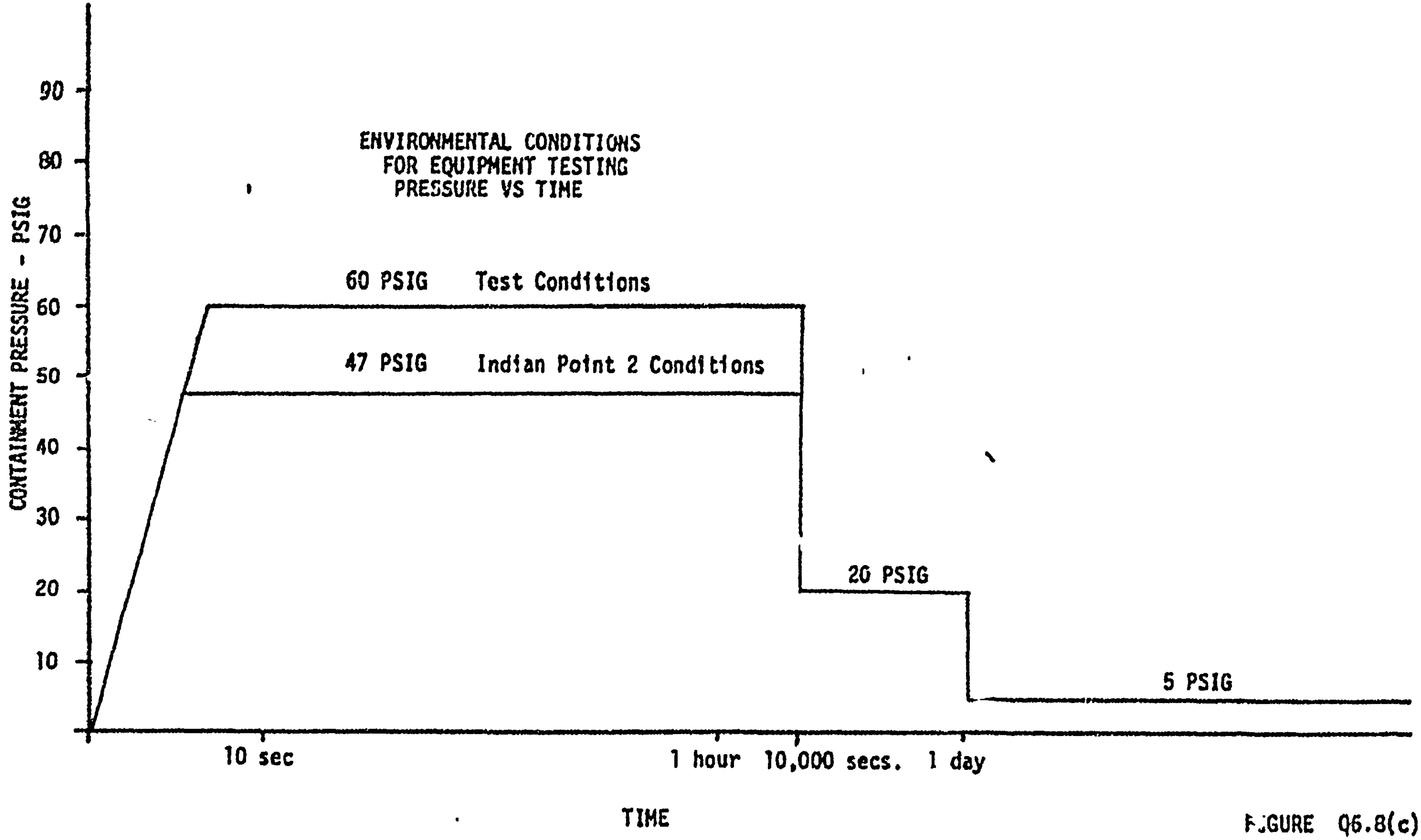


FIGURE Q6.8(c)-1

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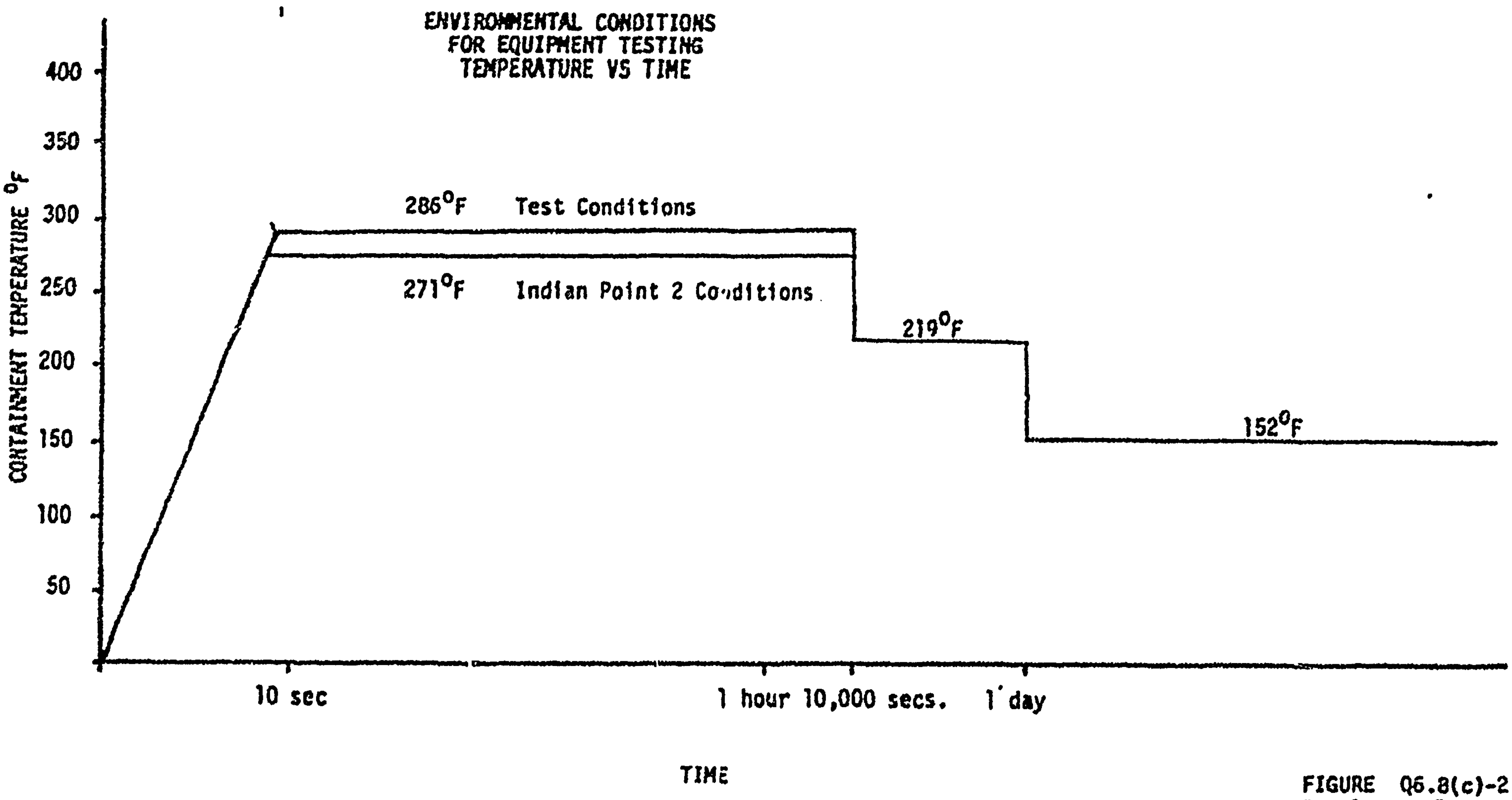

FIGURE Q6.8(c)-2
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QUESTION 6.8

We understand that installation of the Westinghouse flame recombiner system is being considered for the Indian Point II plant as an engineered safety feature in order to control hydrogen evolved within the containment following a loss-of-coolant accident.

d. How soon following the loss-of-coolant accident might the flame recombiner system be capable of being initiated, given the unlikely occurrence of greater than predicted hydrogen levels in the containment? Discuss those features and/or operating characteristics of the recombiner system which form the limiting time-to-initiate considerations. This discussion should include considerations of time to sample and measure, time to acquire and connect fuel supplies, time and exposure restriction regarding control station manning, and the restrictions imposed by recombiner design (e.g., blower rating, or processing setpoints).

ANSWER

The only limitation to the operation of the recombiner system would be the blower motor rating. The motor is sited for continuous operation in an environment of 5 psig. This should not be of concern as the pressure in the containment following an accident should be down to 5 psig within < 5 days, and with conservative calculations the hydrogen content within the containment will not reach 2% by volume until 13 days after an accident.

QUESTION 6.9

Prior to and in conjunction with the long-term operation of the flame recombiner system, it may be necessary or desirable to continue operation of certain other engineered safety features such as the containment spray systems and/or the fan recirculation systems. This may be desirable from the viewpoint of providing good mixing of containment gases in order to minimize the potential for stratification and pocketing. Given the design basis loss-of-coolant accident, please provide a discussion of the expected long-term operating modes of such other engineered safety features. Relate the period of operation of these systems to the various time phases of the accident, i.e., fission product reduction phase, heat removal phase, and mixing and circulation phase, in order that the integrated functional requirements over the long-term period may be more completely understood.

ANSWER

Following a loss-of-coolant accident both the containment spray system and reactor containment fan cooler systems are placed in operation for fission product reduction, heat removal and containment air recirculation.

During the injection phase of the accident a minimum of one spray pump and three of five fan coolers are in operation.

The heat removal requirement for the design basis accident are met with these minimum requirements during both the injection and recirculation phase. Section 14.3.4 discusses the pressure transient and heat removal capability of using minimum safeguards.

During the injection phase sodium hydroxide is added to the containment spray and along with the filtration system of the fan cooler units the two hour dose is reduced to acceptable levels as described in Chapter 14.

Since the spray is effective in removal of inorganic iodine which is removed during the first two hour period following the accident, the spray pump operation could be terminated after the containment pressure is reduced and stabilized.

The fan cooler units would continue in operation alone during the long-term recirculation phase during which the containment fission products, primarily organic iodide, and pressure are continually reduced. In addition, effective recirculation is provided to all parts of the containment Suction t.. the fan cooler units is taken from the upper portion of the containment and discharged from the fan coolers through a ring header to various compartments below the operating deck.

QUESTION 6.10

We understand that in selecting a proposed combustible gas control system, alternative measures were studied for feasibility. Provide a discussion of those alternatives studied and the favorable/unfavorable features and technical considerations which led to the final selection of the flame recombiner system.

ANSWER

Several methods have been investigated for controlling hydrogen in the containment following a loss of coolant accident, where site considerations preclude safe disposal by venting to the atmosphere.

The flame type recombiner design has been demonstrated to be a satisfactory solution to controlling hydrogen. This unit has undergone extensive testing which has proved that this design can control hydrogen inside the containment in the environment existing after the DBA.

In addition to the flame recombiner, a catalytic recombination system has been considered. The obvious problem with the catalytic unit is poisoning of the catalyst. Because of the uncertainty concerning catalyst poisoning and the excellent reliability of the flame recombiner, installation of a catalytic recombiner inside the containment is not recommended.

More recently, an investigation was started to evaluate the feasibility of recirculating containment gases through a small catalytic recombiner outside the containment to remove hydrogen gas. No conclusion is available on this concept at present because the study is still at a very preliminary stage.

A feasibility study has been completed based on cryogenics. Since the equipment used to control hydrogen must be designed as engineered safeguards, the equipment complexity and large quantities of energy required for this method makes it impractical for a nuclear installation.

The flame recombiner method of controlling hydrogen has been clearly demonstrated to be a safe and practical means of coping with hydrogen following a DBA such that alternate methods must be capable of having the same high reliability and operational confidence level. To date investigation of these alternatives has failed to show an advantage over the flame unit in terms of overall reliability.

QUESTION 6.11

Since controlled containment purging could provide a backup to the flame recombiner system, please provide an evaluation of controlled purging for the Indian Point II plant.

ANSWER

A flame recombiner system has been provided to control the post accident hydrogen concentration in the Indian Point Unit 2 containment. Design and operating criteria have been defined for that system, and the capability of that system has been demonstrated by full scale proof tests. In March, 1969, a comprehensive report on this work was submitted to USAEC (See WCAP-7301-L, Report of Test Results on Hydrogen Flame Recombiner, Westinghouse Proprietary Class II). Adequate backup for the flame recombiner system has already been provided in the form of a second, completely independent flame recombiner system. However a controlled containment purging system will be designed using existing spare penetrations and will be provided in a manner satisfactory to the regulatory staff during the first two years of power operation. 15

QUESTION 6.12

We note that the refueling water storage tank has been designed and fabricated to the code requirements of AWWA D100-65 (Table 6.2-1). However, quality standards for this component were not included in Table 6.2-13 "Quality Standards of Safety Injection System Components". State what inspections, non-destructive testing, and special quality control procedures were used in tank fabrication.

ANSWER

The following quality standards were applied to the fabrication and erection of the Refueling Water Storage Tank:

A. Tests and Inspections

1. Vacuum box test of tank bottom seams.
2. Hydrostatic test of tank.
3. Hydrostatic test of tank heater coil.
4. Spot radiography of longitudinal and girth welds.

B. Special Manufacturing Process and Material Control

1. Weld, fabrication, NDT and inspection procedure review.
2. Surveillance of suppliers quality control and product.
3. Material chemical and physical properties certification.

QUESTION 6.13

Specify which other Class I systems' components, valves, and piping are designed to the same quality standards as described for the safety injection system in Table 6.2-13. Identify the exceptions and discuss the bases for the differences.

ANSWER

The other Class I heat exchangers meet the same quality standards as the residual heat exchanger in Table 6.2-13.

The other Class I pumps meet the same quality standards as the safety injection, recirculation and residual heat removal pumps in Table 6.2-13.

Most Class I tanks meet the same quality standards as the accumulator. The notable exceptions are the boric acid and component cooling surge tanks which contain only atmospheric pressure (vented) and for which radiography of longitudinal and girth welds is not required. A standing water test is substituted for the hydrostatic test. An atmospheric tank is not exposed to pressure transients and only has to withstand the weight of the fluid it contains. A vented tank wall thickness will generally be set by the structural stability requirements during fabrication instead of the minimum wall thickness set by pressure of the contained fluid.

The quality standards for valves in Table 6.2-13 were set for Class I system valves and are the same for all valves to prevent installation of a lower quality valve in a Class I system.

Question 7.1

In regard to the protection systems which actuate reactor trip and safety feature action, the following information is requested:

a. For those systems designed and built by Westinghouse, identify which are identical to systems used in the Ginna Station. Discuss any design differences in systems which are not identical to those used in the Ginna Station.

b. Where systems are designed and/or built by other than Westinghouse, identify the supplier of the system. Also, identify any features of the design which differ from the criteria of IEEE 279 and the appropriate General Design Criteria. Explain the reasons for such differences.

Answer to Item a.

1. Reactor Logic Protection System

The reactor protection system, as designed and built for Indian Point Unit 2, is functionally identical to R. E. Ginna except as noted below.

1) Turbine Trip/Reactor Trip

a. On R. Ginna, a turbine trip below ~50% power (P-9) will not actuate a reactor trip if the condenser is available for steam dump; below ~10% power (P-7), a turbine trip will not actuate a reactor trip. On Indian Point Unit 2, above ~10% power (P-7) a turbine trip will always actuate a reactor trip, below P-7 a turbine trip will not actuate a reactor trip. The net effect of this difference is a slightly more flexible design for R. Ginna. As shown in section 14.1.8 of the Indian Point No. 2 FSAR, an immediate reactor trip on turbine trip is not required for reactor protection, thus, this design need not satisfy IEEE 279.

b. On R. Ginna a turbine trip signal is generated when both main feedwater pumps are tripped, whereas no such signal exists for Indian Point. This signal is generated in anticipation of the resultant decrease in steam generator water level, however, and is not required for plant safety since the reactor trips on low feedwater flow and low-low water level will adequately protect the plant. With regard to these trips, the designs for R. Ginna and Indian Point are basically identical, though the nomenclature is not. For Indian Point the low feedwater flow trip is called the steam/feedwater flow mismatch trip and the low-low water level trip is called the low level trip.

c. On R. Ginna a turbine trip signal is generated when both circulating water pumps are tripped, whereas no such signal exists for Indian Point. Again, this is an anticipatory signal which is not required for plant safety. The absence of the signal would result in the turbine being tripped by other signals, e.g. high condenser pressure. It should be noted that, except for the turbine trip following reactor trip, no turbine trip signals are required for reactor protection.

d. For R. Ginna a turbine trip signal dumps the auto-stop oil and closes all turbine stop valves simultaneously whereas, for Indian Point a turbine trip signal redundantly dumps the auto-stop oil which, in turn, closes all turbine stop valves. Conversely, for R. Ginna, a reactor trip or turbine trip is generated by sensing the loss of auto-stop oil or the closure of all turbine stop valves, whereas, for Indian Point a reactor trip on turbine trip is generated by redundantly sensing the loss of auto-stop oil. As shown in section 14.1.8 of the Indian Point No. 2 FSAR, an immediate reactor trip on turbine trip is not required for reactor protection, thus, this design need not satisfy IEEE 279.

2) Turbine Runbacks

A turbine runback is employed following a rod drop accident. For R. Ginna, which utilizes E-H turbine governor control, this is achieved by reducing both the load limit and load reference signals as required to achieve the required load reduction; for Indian Point, which utilizes mechanical hydraulic turbine governor control, this is achieved by reducing the signal to each of two load limit values as required to achieve the required load reduction. The turbine runback is not required for reactor protection and, thus, this design need not satisfy IEEE 279.

3) Low Flow Protection

a. For R. Ginna the reactor trip on underfrequency is generated on a signal indicating an underfrequency condition on both reactor coolant pump buses, with two signals per bus; for Indian Point this trip function is generated on a signal indicating an underfrequency condition on two of four reactor coolant pump buses, with one signal per bus. The purpose of this trip is to provide protection following a major network frequency disturbance. Either design satisfies IEEE 279.

b. The undervoltage trip for R. Ginna is also generated on a signal indicating an undervoltage condition on both reactor coolant pump buses, with two signals per bus, whereas for Indian Point this trip function is generated on a signal indicating an undervoltage condition on two of four buses, with one signal per bus. This trip is also provided for protection following a complete loss of power. Both designs satisfy IEEE 279.

c. For R. Ginna, on Reactor Coolant Pump bus underfrequency, the reactor is tripped directly and both reactor coolant pumps are tripped; for Indian Point all reactor coolant

Pumps are tripped on underfrequency with this signal generating a reactor trip. In the event of a frequency disturbance, the primary requirement is to release the Reactor Coolant Pumps from the network to preserve their kinetic energy. In this respect the Indian Point and R. Ginna designs are identical. For both R. Ginna and Indian Point, the minimum DNBR during the accident is limited to a value above 1.30.

4) Low Feedwater Flow Trip
(Steam/Feedwater Flow Mismatch Trip)

Figure 7.1-1 shows the implementation of the low feedwater flow trip for both R. Ginna and Indian Point. Though these are implemented differently, they perform essentially identical functions. Both satisfy IEEE 279.

5) Steam Break Protection

a. Safety Injection System Actuation

For R. Ginna safety injection is actuated by low pressurizer pressure coincident with low pressurizer water level, or high containment pressure or low steam line pressure (not compensated) in either steam line; for Indian Point safety injection is actuated identically for the first two items but Indian Point does not use low steam line pressure for safety injection actuation. In lieu of steam line pressure, the following signals actuate safety injection for Indian Point; 2/3 high differential pressure between steam lines or 2/4 high steam flow in coincidence with 2/4 low T_{avg} or 2/4 low steam line pressure. The reason for the difference in design is that the Indian Point design, though suitable for a four loop plant, is not desirable for a two loop plant; conversely, the R. Ginna design is not completely desirable for a four loop plant. Both designs satisfy IEEE 279.

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b. Steam Line Isolation

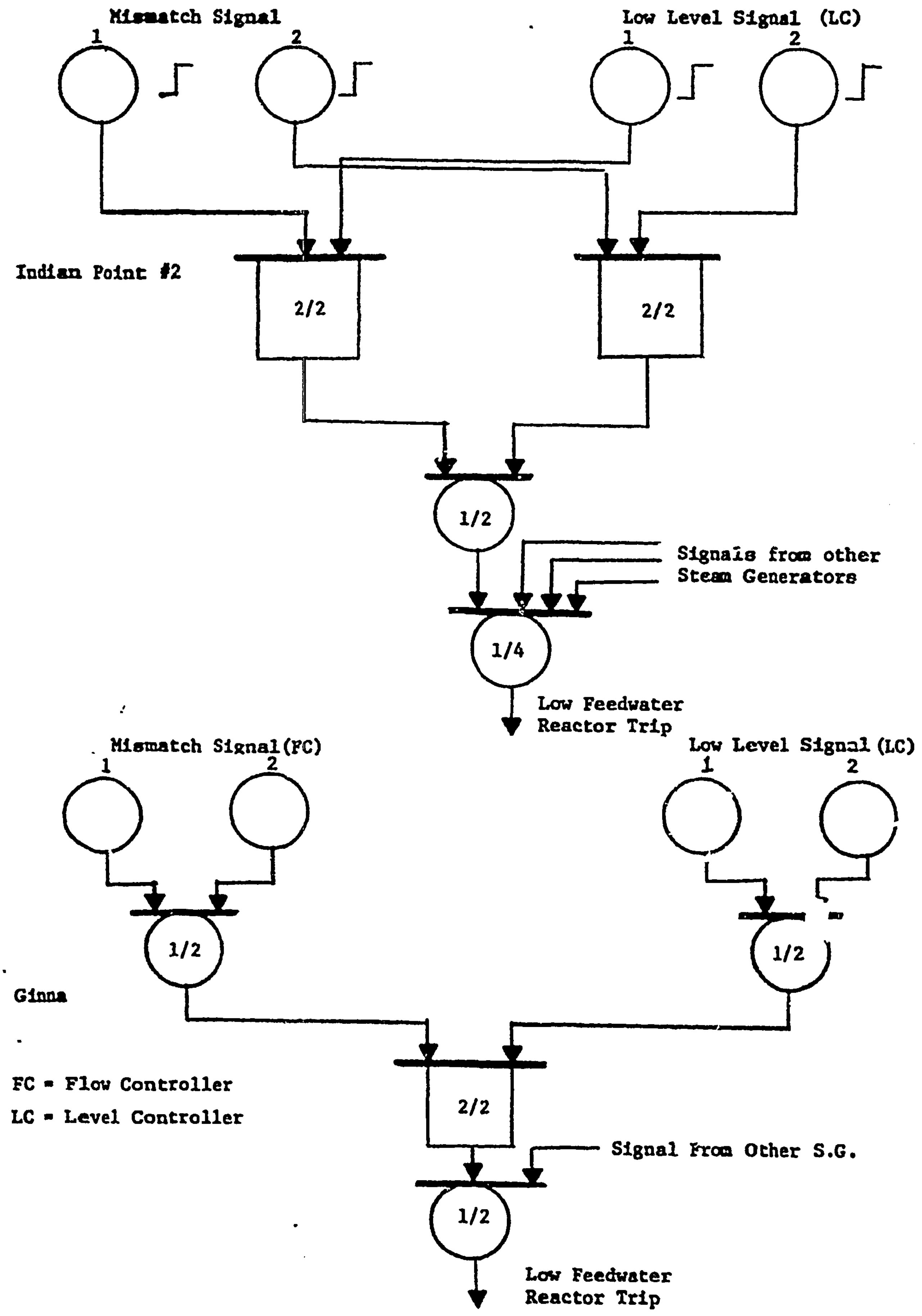
For R. Ginna a steam line isolation signal is generated as follows; manually, high-high containment pressure, 1/2 high-high steam flow with safety injection, and 1/2 high steam flow with 2/4 low T_{avg} and safety injection. For Indian Point a steam line isolation signal is generated similarly as R. Ginna for the first two items. Steam line isolation is also caused by 2/4 high steam flow in coincidence with 2/4 low T_{avg} or 2/4 low steam pressure. Again, the difference in design can be attributed to the difference in reactor coolant loops. Either design satisfies IEEE 279.

2. Nuclear Instrumentation System

The Nuclear Instrumentation Systems for both Indian Point Unit 2 and R. Ginna are the same.

Answer to Item b.

Westinghouse designed and procured all systems which actuate reactor trip and safety feature actions for the Indian Point Unit 2. The design of protective grade instrumentation and logic systems are in accordance with IEEE 279. The functional design is originated at Westinghouse with equipment procurement through vendor supplies. Equipment compatibility and integration of component hardware is factored into the design by Westinghouse or under the direct supervision of Westinghouse.



COMPARISON OF INDIAN POINT #2 AND GINNA
LOW FEEDWATER FLOW REACTOR TRIP
Figure 7.1-1

QUESTION 7.5

Please describe the quality control procedures which apply to the equipment in the reactor protection system, engineered safety feature and containment isolation systems, and the associated emergency power systems. This description should include, but not necessarily be limited to, (a) quality control procedures used during equipment fabrication, shipment, field storage, field installation, component checkout, system checkout; and (b) records pertaining to (a) above.

ANSWER

The electric equipment and the instrumentation in the reactor protection system, engineered safety feature and containment isolation systems, and their associated emergency power systems are all Class I components. Therefore, they fall automatically in the category of equipment to which the quality control procedures of Appendix B of the FSAR apply.

The quality control program during all stages of fabrication, shipment, construction and checkout is described in Appendix B at the following places:

a) Component fabrication on pages B-5 through B-9 under items A. Supplier Evaluation, B. Equipment Specification, C. Purchase Order, and D. Supplier Surveillance.

b) Component shipment on pages B-9 and B-10 under "Shipment of Components".

c) Component storage and construction on page B-10 under "Inspection and Installation of Equipment in the Field", on page B-11 under "Non-Conforming Components or Material," on page B-11 under "United Engineers and Constructors", and on page B-29 under "Site Quality Control" items 1 and 2.

d) Component Checkout on page B-29 under "Site Quality Control" items 3 and 4.

Quality Control Records are described on page B-10, last paragraph, under "Inspection and Installation of Equipment in the Field", on page B-11 under "Quality Control Records", and on pages B-14 and B-15 under "Records".

QUESTION 7.11

Figure 7.2-8 of the FSAR shows two bypass breakers (AB-1 and AB-2) in the scram circuit. During the Ginna review that applicant decided to utilize one bypass breaker as a safeguard against inadvertently having both bypass breakers closed at the same time. What is your intent with regard to this problem? What information is provided to the control room operator which allows him to evaluate that the breaker(s) is installed properly?

ANSWER

The intention is to leave the bypass breakers housed in their respective switchgear units locked in the withdrawn (non-operating) position. The positioning of these breakers to the operating position for logic system testing will be under the administrative control of the operator. The closing of the breaker is controlled from its respective logic test panel in the control room. The status of the breaker is indicated in the control room by indicating lights. An interlock will be provided that will trip both bypass breakers open if a second bypass breaker is closed.

QUESTION 7.15

Page 7.2-1 of the FSAR discusses the Control Room. Please provide the following additional information:

b. What facilities for emergency lighting are provided?

ANSWER

CONTROL ROOM EMERGENCY LIGHTING

The emergency lighting in the control room consist of 15 - 150 watt incandescent floodlights. These lights are strategically located to illuminate the instrument panels, flight panel, supervisory control panels and the operator's desk. The normal voltage supply for these lights is the lighting switchgear. When the supply voltage drops to 70 percent or less of rated voltage, the load is automatically transferred to the emergency d.c. supply. The emergency supply source is the d.c. batteries.

Question 7.16

For the process instrumentation which provides signals to the reactor protection and engineered safety feature actuation circuitry, please provide a table which lists the following information:

a. The parameter being sensed.

b. The type of sensor (e.g., Foxboro pressure), manufacturers specified accuracy and repeatability and expected failure mode.

c. The type of readout (e.g., indicating, blind, etc.)

d. The type of power required (e.g., external, self).

e. Use of channel (to provide information to reactor protection or engineered safety feature circuitry).

f. Number of redundant sensors on common sensing line.

Answer

See attached table.

Parameter	Transmitter/ Sensors	Read-Out	Power	Prot/Safeguards Use	Taps
Reactor Coolant Temperature	8 RTD's	C.B. Meter	Ext.	ΔT trips T_{avg} permissives	1 each
Pressurizer Pressure	4 Transmitters	C.B. Meter	Ext.	Hi/Lo Pressure Trips, SIS	3 (Top Level) One Shared 3 Pairs
Pressurizer Level	3 ΔP Transmitters	C.B. Meter	Ext.	SIS	3 (Top Level) One Shared 3 Pairs
Steam Flow	8 ΔP Transmitters	C.B. Meter	Ext.	Mismatch Trip SIS	1 Pair Each
Feedwater Flow	8 ΔP Transmitters	C.B. Meter	Ext.	Mismatch Trip	1 Pair Each
Steam Pressurizer	12 Transmitters	C.B. Meter	Ext.	SIS	1 Each
Steam Generator Level	12 ΔP Transmitters	C.B. Meter	Ext.	Mismatch Trip Low Level Trip	1 Pair Each
Reactor Coolant Flow	12 ΔP Transmitters	C.B. Meter	Ext.	Low Flow Trip	1 High Pressurizer Each 1 Low Pressurizer Shared/Loop
Containment Pressure	6 Transmitters	C.B. Meter	Ext.	SIS (3) Spray (3+3)	3 Shared
Turbine 1st Stage Pressurizer	2 Transmitters	Blind	Ext.	Set Point Programs and Turbine Power Permissives	1 Each

* C.B. is Control Board

Question 7.2

In regard to the Westinghouse designed control systems, the following information is requested:

a. Identify the control systems which are identical to systems used in the Ginna Station.

b. List and discuss any design differences in systems not identical to those used in the Ginna Station. This discussion should include an evaluation of the safety significance of each system change.

Answer to Item a.

All major process control systems at Indian Point Unit 2 are functionally identical to those of the Robert E. Ginna Station. These systems include Reactor Temperature Control, Feedwater Control, and Pressurizer Pressure and Level Control. Both units use the same solid state rod control system design.

Answer to Item b.

Process Control System

There are no system design changes applicable to the major process control system as stated above. In minor process systems, being defined as those systems having no effect on the Nuclear Steam Supply System, differences do exist. These are due to different daily plant operations or individual plant layout considerations.

QUESTION 7.3

Where reactor protection and engineered safety feature signals feed annunciators and/or a data logging computer, the design criterion used to assure isolation should be described and evaluated.

ANSWER

The design criterion used to assure electrical isolation exists between protective and control grade signals. Where protection signal intelligence is required for other than protection functions an isolation amplifier (part of the protection set) is used to transmit the intelligence. The isolation amplifier prevents perturbation of the protection channel signal (input) due to any disturbance of the isolated signal (output) which could occur near any termination of the output wiring external to the protection racks. A detailed discussion of the isolation amplifiers that are used in Indian Point Plant Unit 2 is given in Test Report of Isolation Amplifier, J. Lipchak, R. Bartholomew, WCAP 9011 (Proprietary).

Isolation of the reactor protection and engineered safety feature signals in the reactor protection logic racks is achieved by physical separation. There are three decks containing the poles on the type relay used. The rear deck is used for reactor protection and engineered safety feature signals. The center and front decks are used for annunciator and computer signals respectively. This provides the necessary isolation between the safety signals and the annunciator and/or computer signals. The separation is maintained by using separate wireways for safety signals, annunciator signals and computer signals.

QUESTION 7.4

What are the design bases for the reactor protection system, the instrumentation and controls for engineered safety features, and the electric power with regard to earthquakes? Will the system be capable of actuating reactor trip or engineered safety feature action during the maximum peak acceleration? If a seismic disturbance occurred after a major accident, would emergency core cooling be interrupted? What tests and analysis were performed to assure that the seismic design bases are met? What seismic specifications are employed in the instrumentation and controls purchase order(s)?

ANSWER

Westinghouse is using the following design basis for the protection and safeguards systems equipment:

For either earthquake (operation basis or design basis) the equipment will be designed to ensure that it does not lose its capability to perform its function; i.e. shut the plant down and maintain it in a safe shutdown condition. For the maximum potential earthquake, there may be permanent deformation of equipment provided that the capability of the equipment to perform its function is maintained.

The instrumentation and electrical equipment, associated with emergency core cooling, will not cause an interruption of this function during the earthquake. Results appropriate to Indian Point #2 of the seismic simulation tests on instrumentation and control equipment will be presented to the AEC in January 1970. See also response to Question 1.9.

Instrumentation and control specification include only static "g" level requirements. As a result of recently established seismic criteria, the static requirements are considered inappropriate and Westinghouse has adopted the position of type testing equipment to demonstrate design adequacy (See also the response to Question 1.9).

QUESTION 7.6

Please submit your cable installation design criteria for preserving the independence of redundant reactor protection system and engineered safety feature circuits (instrumentation, control, and power). For the purpose of cable installation, the protection system circuits should be interpreted in their broadest sense to include sensors, instrument cables, control cables, and power cables (both a.c. and d.c.), and the actuated devices (e.g., breakers, valves, pumps, etc.):

a. Cable separation should be considered in terms of space and/or physical barriers between redundant cables. Please address (1) the separation of power cables from those used for control and instrumentation, (2) the intermixing of control and instrument cables within a tray (or conduit, ladder, etc.), (3) the intermixing within a tray (or conduit, ladder, etc.) of cables for different protection channels, and (4) the intermixing of non-vital cabling with protection system cabling.

ANSWER

The reactor protection and engineered safety system cable circuits are divided into as many channels as is required to preserve the basic redundancy and independence of the systems. Channel separation is maintained as indicated below and is continuous from the sensors to the entrance to the receiver racks to logic cabinets to actuation devices in such a manner that failure within a single channel is not likely to cause the loss of the basic protection system or cause a failure which would prevent actuation of the minimum safeguards devices when called for.

In general this requires the use of four (4) instrument channels. three (3) power channels and three (3) control channels. In addition to such channels of separation, cables are also assigned to individual routing systems in accordance with voltage level, size and function category. Six (6) independent conduit and/or tray systems are used for such purposes and establish separation of:

(1) 6900 volt power

(2) Heavy 125 V D-C power cables and heavy 480 V A-C (over 100 HP) power cables

(3) Lighting panel feeders and medium power (greater than No. 12 AWG wire size) 480 volt A-C cables.

(4) Control and small power cables.

(5) Instrument cables.

(6) Rod control cables.

There is no mixing of the above categories within a tray or conduit. There is some mixing of non-vital cables with vital cables in channelized trays, but cables of different voltage classes are not mixed within a tray.

Separation of channels is established wherever practical by the use of separate trays and conduits. In the cable spreading, tunnel and other congested areas multiple channels are run in a single ladder tray, but separation is maintained by the use of 16 gauge sheet metal barriers 4" high within the tray. Where such barriers are used in heavy power cable trays a double sheet metal barrier with approximately one (1) inch of space between is utilized. In addition, whenever a power tray is located beneath an instrument or control channel tray, or a different channel of heavy power cables, a 1/4" thick transite barrier is installed between the trays. Such barriers are considered to be redundant as the power cable insulation being used is fire retardant and will not support combusion without excitation.

The electrical tunnel consists of a square concrete conduit having an inside dimension of approximately ten feet wide and eight feet high. Arrayed on either side of a three foot aisle are seven (7) thirty-six inch ladder trays on one side and four (4) thirty-six inch and one (1) one foot tray on the other side. Channel separation is maintained in the tunnel as outlined above. The minimum vertical channel separation for instrument and control circuits is approximately seven and one half (7-1/2") inches with the majority having a separation of over nineteen inches. Horizontal separation varies from a 16 gauge thick 4" high metal barrier as previously outlined to the width of the aisle (three foot).

The power channels have a vertical separation of seven and one half inches with a 1/4" transite barrier between trays as previously described. Power channels are separated horizontally by two 16 gauge sheet metal barriers with one inch space between them. The four channels of nuclear instrumentation sensor cables are in individual conduits which are supported at the end of 4 trays and have a separation of approximately twelve inches.

Figure 7.6-1 shows a section view of the tunnel and identifies the channeling used within the area.

QUESTION 7.6

Cable separation should be considered in terms of space and/or physical barriers between redundant cables. Please address (1) the separation of power cables from those used for control and instrumentation, (2) the intermixing of control and instrument cables within a tray (or conduit, ladder, etc.), (3) the intermixing within a tray (or conduit, ladder, etc.) of cables for different protection channels, and (4) the intermixing of non-vital cabling with protection system cabling.

b. What are your criteria with respect to (1) the separation of penetration areas, (2) the grouping of penetrations in each area, and (3) the separation of penetrations which are mutually redundant.

ANSWER

The electrical penetrations are in a single area, comprised of some sixty assemblies arrayed in a group of low voltage power, control and instrument wire assemblies and a separate group of 6900 volt assemblies. The 6900 volt assemblies are separated from the rest of the units by a distance of approximately six feet.

The main group of assemblies (penetration canisters) are arranged in four rows high, with each row separated from another row by three (3) feet. Each assembly in a row is spaced on approximately three (3) foot centers. Each assembly has only one category of circuit within it. The various penetration canisters consist of units of #12 AWG, shield twisted pairs, shield twisted quads. #10 AWG, #4 AWG, 250 MCM #4/0 AWG and triax. Channel separation is maintained to and from the penetrations and no two channels are run through a single penetration. Large and small power assemblies are placed in only two rows.

In general the separation between redundant or channelized circuits is expected to be greater than the spacing between two adjacent assemblies. However, some channels are in adjacent units and free air spacing can be expeced to be twenty-eight inches or more at the face of the penetration. The control instrument and small power assemblies are furnished with factory installed pigtails and field splices are therefore well away from the canister face. The cable spreading and penetration areas are in a concrete vault, dead ended at one end so that no traffic is expected in this area.

QUESTION 7.6

Please submit your cable installation design criteria for preserving the independence of redundant reactor protection system and engineered safety feature circuits (instrumentation, control, and power). For the purpose of cable installation, the protection system circuits should be interpreted in their broadest sense to include sensors, instrument cables, control cables, and power cables (both a.c. and d.c.), and the actuated devices (e.g., breakers, valves, pumps, etc.):

c. Please discuss cable tray loading, insulation, derating, and overload protection for the various categories of cable.

ANSWER

All cables outside the containment, with the exception of 8 Kv, are PVC insulated with a variety of shields and jackets as required for a specific use. Excluding 8 Kv, cables used inside the containment are silicone rubber or Kerite insulated to provide greater radiation resistance. The 8 Kv cables are insulated with XLPE and are run in separate trays with maintained spacing.

Physical loading of cable trays is controlled by means of the Conduit and Cable Schedule. Trays containing instrumentation and control cables are regulated to have a maximum fill of 70 percent of the tray area, while those containing power cables are limited to two or four layers depending on the size and use of the cables.

Cables in trays with no maintained spacing are derated according to their temperature rating, the number of cables in the tray, and size variation of these cables. The base rating and foundation for all derating calculations is taken from IPCEA Publication P-46-426, "Power Cable Ampacities - Copper Conductors", using the proper conductor temperature and ambient conditions for our application. A derating factor is then applied to the base rating according to the number of conductors in the tray. Cable on opposite sides of the barriers in power trays are considered to be in different trays for this calculation. The derating factor used is based on standard load diversity. Lastly, the cables are derated to eliminate any hot spots that might occur due to the presence of larger than average size conductors in the tray. For the pressurizer heater cables which have no diversity, a thermal study was made using actual load conditions to determine that the internal temperature of the cables was within safe limits.

All cables serving 6900 volt motors or station service transformers, 480 volt motors 100 hp and over (with a few exceptions), and 480 V motor control centers are protected aginst overloads by circuit breakers. The 480 volt circuits for motors under 100 hp are protected by fuses. In some instances, these fuses are backed up by circuit breakers in the local starters for these motors. Instrumentation and d.c. circuit. are protected by circuit breakers.

Since it is necessary to provide forced air circulation to maintain cable conductor temperatures within acceptable limits, two separate fans either of which is capable of removing all heat necessary to prevent excessive cable temperatures during operation of the safeguard equipment have been provided. These fans will be supplied from separate diesel generator buses.

QUESTION 7.6

Please submit your cable installation design criteria for preserving the independence of redundant reactor protection system and engineered safety feature circuits (instrumentation, control, and power). For the purpose of cable installation, the protection system circuits should be interpreted in their broadest sense to include sensors, instrument cables, control cables, and power cables (both a.c. and d.c.), and the actuated devices (e.g., breakers, valves, pumps, etc.):

d. Please discuss your criteria with respect to fire stops, protection of cables in hostile environments, temperature monitoring of cables, fire detection, cable and wireway markings, and the administrative responsibility for, and control over, all of the foregoing (a-d) during design and installation.

ANSWER

Firestops are provided where cable trays pass through walls and floors, and enter switchgear or other equipment. Three types of firestops are used according to the function of the cable in the tray (control, power, etc.) and ventilation requirements of the areas involved. The first type of firesto. is used in trays containing control cables passing through walls, floors, or into equipment where an air seal is not required. It is composed of two aluminasilica ceramic fiber blankets, 36 inches long, laid in the tray and compressed around the cables using a cable tray cover. An ignited cable is extinguished by this firestop because the ceramic fiber blanket cuts off the oxygen supply. The blanket has a low thermal conductivity and can be used at temperatures up to 2300°F, without showing any physical change. Even beyond that temperature, it still retains its fire preventive characteristics.

Because of its low thermal conductivity and the fact that it covers three feet of cable surface area, this blanket cannot be used with power cables, which dissipate considerable amounts of heat. In addition, it cannot easily be installed in control trays, where an air seal of the wall or floor opening is required for ventilation purposes. The firestop used for these configurations consist basically of (1) a transite sheet to substantially close the opening, (2) Flamemastic 71A Mastic sprayed on the cables 6" on either side of this sheet, and (3) Flamemastic 71A Mastic trowelled into the cable tray on top of the cables to seal any remaining air passage between rooms. Flamemastic 71A has recently been accepted and used by a number of utilities for this purpose. It is non-toxic, not damaging to cable insulation, and requires no derating of cables when applied over a one-foot section. Tests by various power companies and cable manufacturers have shown that a 1/16" coating of Flamemastic 71A will not burn through after 15 minutes exposure to a propane torch at 2050°F.

The third type of firestop is used only for openings in the floor where control or power cables enter switchgear, motor control centers, supervisory cabinets, or other equipment from the tray below. This configuration combines packed fiberglass with a 1/4" coating of Flamemastic 71A sprayed on either side of the closure. It provides both protection and separation of cables as they pass through the floor. <u>This type of fire stop is used where control cables enter the panel in the control room.</u>

The control cables to the safeguards control panels (SB-1 and Sb-2) will have protective barriers installed at the back of the panel to prevent accidental damage to the switches and cabling by personnel in the service aisle. These devices will consist of barriers over horizontal terminal strips, boots over the top of vulnerable switches, 12" high kick plates in back of panels and in front of the auxiliary panels across the service asile from SB-1 and SB-2, and a rubberized expanded metal cover over the back or the panels and over the front of the auxiliary panels.

Cables are protected in hostile environments by a number of devices. Running the cable in rigid, galvanized conduit is the most frequently used method of protection. For underground runs, PVC heavy wall conduit encased in a concrete envelope provides maximum protection. When cable is run in tray, peaked covers are used in areas where physical damage to cables may result from falling objects or liquids. In addition, covers are provided on horizontal cable trays which are exposed to the sun.

Fire detection is provided for areas where there are large grouping of cables in stacked cable trays. These areas have been divided into six zones as follows:

1.	Control Building	-	Elevation 15
2.	Control Building	-	Elevation 33
3.	Electrical Tunnel	-	
4.	Primary Auxiliary Building	-	Piping and Electrical Tunnel
5.	Containment Building	-	Piping Penetration Area
6.	Containment Building	-	Electrical Penetration Area

The fire detection system consists of (1) ionization type detectors capable of sensing the products of combustion before visible smoke, heat or flame are initiated, (2) a zone indicating unit giving the location of the alarm and (3) a fire-indicating unit, which designates loss of a.c. supply, supervisory circuit trouble, or alarm. An independant temperature rate of rise detection system is provided in the electrical tunnel to automaticolly actuate the water fogging system. Because of the presence of these systems, and the conservative rating of power cables, temperature monitoring of cables is not required.

Cables and wireways are marked by means of zinc or aluminum tags attached at each end. These tags are embossed to conform with the identification given in the Conduit and Cable Schedule. At each multiple conductor cable termination, a card in a transparent plastic covering is attached and marked to indicate color code, terminal designation, and service of each conductor.

The control over and administrative responsibility for all of the above during design and installation rests with United Engineers & Constructors Inc. as the Architect Engineer and with WEDCO as the construction contractor.

An automatically operated water fire spray (fog) system is installed within the electrical tunnel. Provisions are included to prevent water from entering or draining into the switchgear area in the control building. The diesel bus duct is completely insulated and will be protected from direct spray impingement.

Redundancy and separation requirements are initiated by the cognizant electrical or mechanical design engineer. These are then received by the designers of the electrical system installation, thus providing a check. The work of the designer, who prepares the applicable circuit schedule sheet, (which designates the cable routing and termination), is spot checked by the cognizant electrical engineer.

The construction group installs the cable as directed by the circuit schedule sheet. The installations are followed by Westinghouse field engineers and spot checks of circuit installations are made to further insure that the installation is in accordance with the design. Consolidated Edison will spot check the installation.

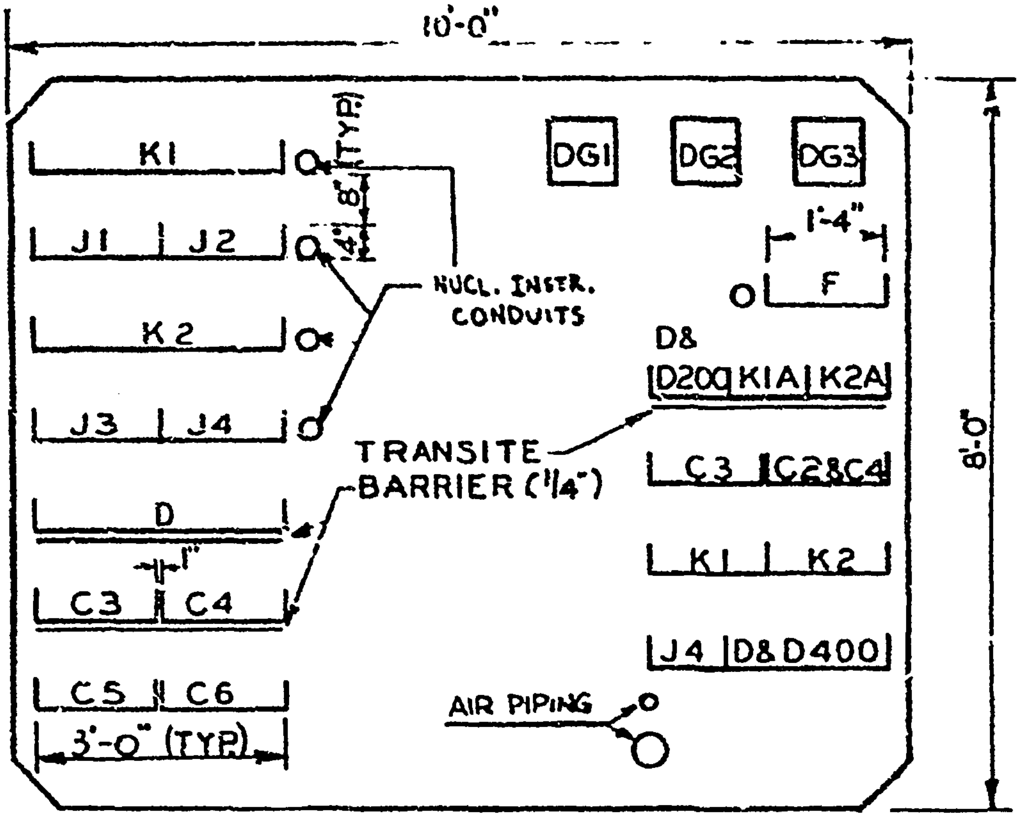
In order to protect switchgear from potential failures in the compressed instrument air system leading to possibility of missile formation or pipe whip, the following provisions have been incorporated.

1. The compressed instrument air line runs in the vicinity of the switchgear are supported at the piping bends to resist a step loading of pA, which could occur in the event of instantaneous circumferential rupture, without occurrence of a plastic hinge. This eliminates the possibility of pipe whip.

2. A guard cover is supplied around the air compressor flywheel designed to absorb the translational kinetic energy associated with a compressor flywheel missile.

3. A guard barrier is supplied adjacent to the compression chamber of the air compressor designed to absorb the kinetic energy associated with a compression chamber segment.

These provisions ensure that no missile or whipping pipe originating from postulated failures in the compressed instrument air system will strike switchgear.



TYPICAL SECTION CABLE TUNNEL

K1	CONTROL TRAY ONE CHANNEL (FOR VALVE CONTROL) ETC
J1-J2	2 INSTR TRAYS WITH BARRIERS - 2 CHANNELS
K2	CONTROL TRAY ONE CHANNEL (FOR VALVE CONTROL) ETC
J3 & J4	2 INSTR TRAYS WITH BARRIERS - 2 CHANNELS
D	MAX #12 WIRE FOR SMALL POWER & ONE CONTROL CHANNEL
C4	(480V HEAVY POWER) CONT RECIRC FAN 21 SAF INJ PUMP 21 RECIRC PUMP 21
C3	(480 V HEAVY POWER) CONT RECIRC FAN 25 RES HT REM PUMP 22 SAF INJ PUMP 23 CONT RECIRC FAN 23
C6	(480 V HEAVY POWER) CONT RECIRC FAN 22 SAF INJ PUMP 22
C5	(480 V HEAVY POWER) RECIRC PUMP 22 RES HT REM PUMP 21 CONT RECIRC FAN 24
F	2ND CHANNELS FOR "F" CABLES D C SUPPLY (NORMAL) DIESEL GEN CONT PNL 22
F	(1ST CHANNEL) D C SUPPLY (EMERG) DIESEL GEN CONT PNL 23 LTG & MOTOR FEEDERS
D & D200 - K1A-K2A	DIESEL GEN CONTROL 3 CHANNELS
C3	(480V HEAVY POWER) COMP CLG PUMP 23 MCC27 FEEDER MCC 26B FEEDER LTG SWGR TRANSF 22 CHARGING PUMP 22 CHARGING PUMP 23 CONT SPRAY PUMP 22
C2	(480V HEAVY POWER) D C EMERG LTG FEEDER
C4	CONT SPRAY PUMP 21 COMP COOLING PUMP 21 COMP COOLING PUMP 22 MCC 26A FEEDER LTG SWGR TRANSF 23 CHARGING PUMP 21
K1-K2	CONTROL, 2 CHANNELS
J4	INSTR 1 CHANNEL
D & D400	MISC SMALL POWER & CONTROL 1 CHANNEL

FIGURE 7.6-1

QUESTION 7.7

Figures 1.2-3 and 5.1-4 of the FSAR show that there is a single electrical penetration area. Figures 1.2-3 and 1.2-6 show a single electrical tunnel. A description of these two areas is requested which provides sufficient detail to show the installation of electrical cables associated with protection systems and the manner in which the electrical cables would be protected from a single casualty, such as fire. The answer to this question might be combined with Question 7.6.

ANSWER

A description of the penetration and tunnel areas is included in the discussion of question 7.6a thru 7.6.d.

In the cable tunnel the damage from a fire is limited by the water spray system which operates automatically to extinguish the fire before extensive damage can result. The transite barriers above power cables are designed to prevent damage to cables located above in the event of a high capacity fault. The sheet metal barriers between adjacent channels are designed to prevent damage to more than one channel from a single incident.

In the penetration area the distance between the penetration of adjacent channels is at least 28". It is considered unlikely that any incident could affect more than one such penetration.

QUESTION 7.7

Figures 1.2-3 and 5.1-4 of the FSAR show that there is a single electrical penetration area. Figures 1.2-3 and 1.2-6 show a single electrical tunnel. A description of these two areas is requested which provides sufficient detail to show the installation of electrical cables associated with protection systems and the manner in which the electrical cables would be protected from a single casualty, such as fire. The answer to this question might be combined with Question 7.6.

ANSWER

A description of the penetration and tunnel areas is included in the discussion of question 7.6a thru 7.6.d.

In the cable tunnel the damage from a fire is limited by the water spray system which operates automatically to extinguish the fire before extensive damage can result. The transite barriers above power cables are designed to prevent damage to cables located above in the event of a high capacity fault. The sheet metal barriers between adjacent channels are designed to prevent damage to more than one channel from a single incident.

In the penetration area the distance between the penetration of adjacent channels is at least 28". It is considered unlikely that any incident could affect more than one such penetration. However, transite barriers above the power cables at the penetrations inside and outside of containment are designed to give added protection against damage to cables located above in the event of a high capacity fault.

QUESTION 7.8

Various pages of the FSAR discuss the design of electrical equipment inside containment (e.g., Page 6.1-6). Other pages discuss the proposed testing of some components (e.g., Page 6.2-35a). Please identify all equipment and components (e.g., motors, cable, etc.) located in the primary containment which are required to be operable during and subsequent to a loss-of-coolant or a steam-line-break accident, and describe the qualification tests which have been or will be performed on each of these items to insure their availability in a combined high temperature, pressure and humidity environment.

ANSWER

Table Q 7.8-3 provides a listing of the equipment located within the primary containment (reactor containment building) which are required to be operable during or following a loss-of-coolant or a steam-line-break accident. In addition, Table Q7.8-3 also lists the equipment operational and environmental testing requirements.

Figure Q 7.8-1 and Figure Q 7.8-2 present the environmental conditions of pressure and temperature, respectively, for both, the Table Q 7.8-3 required equipment test conditions, and for the containment design post loss-of-coolant accident conditions. Figure Q 7.8-3 and Figure Q 7.8-4 present the maximum calculated instantaneous and integrated radiation dose levels inside the containment as a function of time following a TID-14844 model loss-of-coolant accident.

Category 1 - Instrumentation

Except for sump level channels LT-938, 939, 940 and 941, the supplier has completed preliminary qualification tests on pressure and differential pressure transmitters. These are reported in "Topical Report Supplier Post-Accident Testing and Process Instrumentation", J. Nay (Proprietary) WCAP-7354-L.

Additional instrumentation tests have been performed by Westinghouse on equipment obtained from the Indian Point Unit No. 2 Plant equipment supplier. The results of these tests confirmed that the equipment will provide the required signals in the post loss-of-coolant accident environment.

The test conditions of the Westinghouse test were: Steam environment - a five second period rise to 280°F and 60 psig pressure; and the maintenance of these conditions for 2 hours. All equipment, listed below, continued to operate throughout the test and are typical of transmitter ranges used in the containment.

Static Pressure Transmitters	Differential Pressure Transmitters
0-2500 psig	0-240 Inches of Water
1700-2500 psig	0-300 psid

Containment sump and recirculation sump level channels consist of hermetically sealed magnetic switches in a stainless steel housing. The instrumentation is designed for submerged service in borated water at 295°F at a pressure of 69 psig. Since instruments of this design have seen considerable actual service in applications more severe than the post LOCA design conditions, environmental testing for these instruments is not required.

Category 2 - Valves

Currently the Indian Point Unit No. 2 valve operator supplier has conducted loss-of-coolant environmental tests on a Class H unit similar to those used in this plant. Preliminary reports of results indicate that the unit operated satisfactorily at test conditions more severe than those expected in the Indian Point Unit No. 2 loss-of-coolant or steam-break environment.

In addition, Westinghouse has performed environmental tests on a unit similar to that being used in the Indian Point Unit No. 2 plant. The results of the Westinghouse tests indicate that the equipment will perform its required function in the post-LOCA environment.

Tests performed on valve operators, both Class H and Class B, include:

1. Preliminary heat tests (dry heat 16 hrs @ 375°F) on limit and torque switches. All parts operated freely.

2. Preliminary heat tests on actuator. A complete operator assembled and baked at 325°F for 12 hours. Unit operated every half-hour for 2 minutes full open to full close. All operations satisfactory.

3. Preliminary live steam test. Live steam injected into switch compartment. Unit operated every half-hour for 2 minutes over a period of nine hours. All operations satisfactory.

4. Heat aging of motor. Heat aging at 180°C for 100 hours (equivalent to 40-year life) was performed. Comparison of insulation resistance between new and aged motor indicated no significant insulation degradation.

5. Life cycle test. 150 life cycle test under loaded conditions (valve operator produced ∿ 16,500 pounds of thrust). No noticeable change in operator following test.

6. Environmental test. Valve operators subjected to environmental conditions shown in Figures Q 7.8-1 and Q 7.8-2 and sprayed continuously for a period of 3 hours with a solution of boric acid and sodium hydroxide.

   Results of test:

   Class H operator (actual peak test conditions 320°F at 90 psig), operator survived 1st day of exposure during which 12 complete reversing cycles were accomplished. Following 1 week exposure to 247°F and 14.7 psig the unit was operated for 2 complete reversing cycles. The unit operated satisfactorily.

   Class B operator, operator survived the 1st day of exposure with 12 complete reversing cycles. However, after 5 days of exposure, operator failed (failure found to be a short in the motor winding).

As a result of the above tests, Class H operators are being supplied where long term operation required. Class B operators supplied where short term (< 12 hours) operation required.

A production line valve motor has been irradiated to a level of 2 x 10^8 rads using a cobalt-60 irradiation source. The irradiated motor and an identical unirradiated motor have undergone series of reversing tests at room temperature, followed by a series of reversing tests at 275°F. The room temperature test was repeated while vibrating the motors at a frequency of 30 cycles per second. Both motors operated satisfactorily during all of the tests. No significant difference was evident in the comparison of the data for the two units throughout the test period.

Category 3 - Miscellaneous Items

A. Fan Cooler and Internal Recirculation Pump Motors

The motors used on the Indian Point fan cooler units and recirculation pump motors are the same. As a result of the tests described below, minor field modifications have been made to the Indian Point internal recirculation pump motors to improve their integrity (e.g., cork gaskets were changed to neoprene and gasket sealing compound was used in some joints). The fan cooler motors were likewise modified in the suppliers shop.

1. Motor Unit Environmental Tests

A fan cooler motor has undergone a series of nine test runs in conditions more severe than expected in the Loss-of-Coolant or Steam-Break accidents. Approximately 100 hours of testing under load conditions have been completed on the motor at pressure and temperature of 78 psig and 290°F respectively. Chemical spray consisting of 1.7 w/o boric acid adjusted to a pH of 9.5 using sodium hydroxide was injected during approximately 30% of the tests. The motor performed satisfactorily during the tests. It was then partially disassembled and visually inspected. The motor windings were found to be in virtually the same condition as in the pre-test. This testing program is completed. The results of these tests may be found in a WCAP report, "Fan Cooler Motor Unit Development and Test", C.V. Fields, WCAP-9003 (Proprietary), which is referenced in Section 6.4 of the FSAR.

2. Motor Insulation Irradiation Testing

The testing program has been completed on the effects of radiation on the WF-8AC "Thermalastic"* Epoxy insulation system used in the reactor containment fan cooler motor. Tests description and results are presented in the Westinghouse Proprietary report, WCAP 7343-L, "Topical Report - Reactor Containment Fan Cooler Motor Insulation Irradiation Testing", July, 1969.

Irradiation of form wound motor coil sections was accomplished up to exposure levels exceeding that calculated for the design basis loss-of-coolant accident. Three coil samples received the following treatment sequence: Irradiation, high-potential test, vibration test, high-potential test and breakdown voltage test. Nine coil samples received an alternate treatment sequence: Thermal aging, high-potential test, irradiation, high-potential test, vibration test. (Six of nine coil samples), high-potential test and breakdown voltage test.

All coil samples passed the high potential tests. The breakdown voltage levels of all coils were well in excess of those required by the design, and clearly indicate that the reactor containment fan cooler motor isulation system will perform satisfactorily following exposure to the radiation levels calculated for the design basis accident.

3. Motor Lubricant Irradiation Testing

This report summarizes the results of tests performed on samples of unirradiated and irradiated Cheyron BRB-2 lubricant, which is used in the RCFC fan bearing as well as the motor bearing. The results of these tests indicate that the shear stability, or

* Westinghouse Electric Corporation Trademark

consistency, of the grease is increased by irradiation to levels anticipated in the containment following a design basis accident. The consistency of the grease following irradiation remained within the most common recommended consistency for ball bearing application (NLGI #2).

The purpose of this test program was to establish the effect of irradiation on the bearing lubricant used on both the RCFC motor and fan bearing. The maximum calculated 1 year integrated dose on the bearing lubricant, using the design basis accident (TID-14844) with no credit for fission product removal from the containment atmosphere other than by natural decay, is 1.5×10^8 rads and would be experienced by the fan bearings. The motor bearings would receive a lesser exposure due to self shielding effects of the motor housings.

Samples of the lubricant were placed in a vented 1.5 inch x 12 inch aluminum tube. The tube was then placed adjacent to a 34 kilo-curie cobalt 60 source and irradiated for a period of 79 hours. Dosimetry measurements were made at various locations in the tube using Dupont light blue calibration paper 300 MS-C, #CB-91639.

Following exposures to average levels of 1.2×10^8 rads, 1.5×10^8 rads, and 1.8×10^8 rads, the irradiated grease along with unirradiated grease taken from the same supply were subjected to the Micro Cone Penetration Test using standard apparatus conforming to ASTM D1403-56T.

The results of the penetration test are presented in Table Q 7.8-2. In general, it was found that as exposure was increased the grease underwent a change in thickness function to the point that at 1.8×10^8 rads, sufficient change had taken place to cause the grease to increase in consistency to a NLGI #2 rating as the grease was "worked" or sheared rather than decrease as in the unirradiated grease. The most commonly used greases, for ball bearing applications such as those in the RCFC, have consistencies ranging between NLGI #1 and #3.

Understanding of the data listed in Table Q 7.8-2 may be afforded by listing the industry standard for lubricating greases in Table Q 7.8-1 below:

Table Q 7.8-1

NLGI Lubricating Grease Consistency Classification

Consistency Number	ASTM Worked Penetration at 77°F
0	355 to 385
1	310 to 340
2	265 to 295
3	220 to 250
4	175 to 205
5	130 to 160
6	85 to 115

A consistency of #0 implies a very soft semifluid grease, with numbers 1, 2, 3 etc., indicating progressively stiffer grease up to #6 which indicates a stiff, tacky water pump lubricant type material.

Table Q 7.8-2

Motor and Fan Bearing Lubricant Irradiation Testing

Sample	Unworked	Micro-Cone Penetration 60 Strokes	500 Strokes	1000 Strokes	50,000 Strokes
Unirradiated Chevron BRB-2	308	320	368	370	>400
Irradiated BRB-2 $1.2 \times 10^8 R$	300	300	308	324	400
Irradiated BRB-2 $1.5 \times 10^8 R$	308	288	292	298	364
Irradiated BRB-2 $1.8 \times 10^8 R$	340	320	304	296	280

Based on the test results from irradiation and ASTM Micro-Cone penetration measurements, the RCFC bearing lubricant, Chevron BRB-2, undergoes no significant change in properties, as measured in terms of consistency.

B. Hydrogen Combustion System

Full scale proof tests have been performed and have demonstrated the operability of the flame recombiner system as discussed in response to Indian Point Unit No. 2 Question 6.11 (see WCAP-7301-L, Report of Test Results on Hydrogen Flame Recombiner, Westinghouse Proprietary).

1. Exciter Igniter Tests

Following supplier pressure (90 psig) testing of an exciter igniter unit, the unit was exposed to an integrated radiation dose of 1.7×10^8 rads. The results of the irradiation testing indicate that although the unit has been degraded by irradiation (decrease in pulsing rate from 8 pulses/sec down to 7 pulses/sec) the unit continued to operate satisfactorily.

Following irradiation testing, the unit was exposed to the environmental test conditions shown in Figure Q 7.8-1 and Figure Q 7.8-2 and operated 15 minutes a day during a 3-week test period. The results of these tests indicate that the unit will perform its required function following the loss-of-coolant accident.

2. Recombiner Motor Testing

A 2 hp motor (constructed as a recombiner motor) has been irradiated to an integrated dose of 2×10^8 rads. In addition, the motor was thermally aged to an equivalent of 7 years of operation.

The motor with a shaft mounted blower unit to provide loading was tested to the environmental test conditions shown in Figure Q 7.8-1 and Figure Q 7.8-2. The test was conducted over a three-week period. During this period, the motor was operated on a daily schedule of 8 hours on, 16 hours off. One week into the test, it was discovered that the motor had been partially flooded due to a blockage in the chamber drain valve (this condition discovered when it was found that the motor was drawing ~ 3400 watts). The chamber was then drained and the test continued for the required total of three weeks.

The results of this test indicate that the recombiner motor will perform its required function following the loss-of-coolant accident.

C. Cable and Splice Tests

Cabling of the type that is installed in the Indian Point Unit No. 2 plant has undergone tests simulating conditions during a loss-of-coolant accident. These tests were conducted by the Cable Manufacturer and Westinghouse, and consisted of the following:

1. A test was performed by the Cable Manufacturer in a steam environment of 214°F for 436 hours. During this test some cable was energized and was carrying current. A visual inspection following this test showed the cables to be in excellent condition. High voltage, tensile, elongation and stretch showed insignificant changes in their characteristics.

2. A test was performed by the Cable Manufacturer where the specimens were exposed to a gamma radiation field of 2.8×10^7 rads followed by exposure in a steam atmosphere of 85 psig for two - thirty minute cycles. Following these tests the physical appearance of the cables were excellent. Changes in electrical characteristics were as follows:

   a) Insulation Resistance - 90 percent of original value.

   b) Specific Inductance (SIC) No change.

c) Dissipation Factor - Change from 2.2 to 2.1 percent.

d) AC Breakdown - 82 percent of original value.

These percentages represent an average of seven samples.

Westinghouse performed cable testing in a post accident steam and chemical environment of 80 psig (maximum) and a temperature in excess of 300°F. The duration of these tests were in excess of 200 hours in the post-accident steam and chemical environment, 68 hours of which was at a steam pressure higher than containment design pressure. The general appearance of the cables following these tests were good. Some loosening of the jacket from the insulation at the cable ends occurred, generally believed to be due to the rapid decrease in pressure during the tests. Had the cable ends been properly made up, this separation could have been prevented.

Westinghouse has performed additional testing on 18 cable and cable splice test specimens. The testing consisted of the following:

1. Thermal aging to an equivalent of 40 years of operation. (Kerite cable - 150°C for 192 hours, Silicone cable 210°C for 30 days)

2. Irradiating to levels up to 2×10^8 rads.

3. Exposing the cables for three weeks to the environmental test conditions shown in Figure Q 7.8-1 and Figure Q 7.8-2.

4. Apply a potential of 480 volts (with respect to ground) to the cables and conduct rated current through the cables on a daily schedule of 8 hours on, 16 hours off.

Prior to the admission of steam to the test chamber, it was found that 4 of the test specimens had open conductors. These four specimens were removed from the chamber and subsequent examination indicated that the conductors had not been crimped properly.

Of the 14 specimens tested in the environment, thirteen survived. The fourteenth was found to have shorted against the test grounding pipe surrounding the cable. The short appeared to have occurred because of the whipping of the cable caused by the steam injection against the cable.

Based on the above tests, the safeguards cable and splices used on the Indian Point Unit No. 2 plant will maintain their required integrity under the post loss-of-coolant accident conditions.

Table Q 7.8-3

INDIAN POINT POST-ACCIDENT EQUIPMENT (INSIDE CONTAINMENT)
OPERATIONAL AND TESTING REQUIREMENTS

Equipment Name and Tag Number	Operating Mode	Duration of Operation	Environmental Testing
CATEGORY 1 - INSTRUMENTATION			
Pressurizer pressure channels: P-455, 456, 457, 474	Continuous	1/2 hr (S.I. initiation)	Required
Pressurizer level channels: LT-406, 459, 461, 462	Continuous	1/2 hr (S.I. initiation)	Required
High-head flow channels: FT-924, 925, 926, 927	Continuous	1 hr (minimum)	Required
Accumulator pressure channels PT-936A,B,C,D, 937A,B,C,D	During injection phase	1 hr	Required
Recirculation spray flow channels: FT-945A,B	Intermittent	Available to 1 year	Required
Recirculation sump level channels: LT-938, 939	Continuous	Available to 1 year	Not required
Containment sump level channels: LT-940, 941	Continuous	Available to 1 year	Not required
Residual heat loop flow channels: FT-640, 946A,B,C,D	Continuous	Available to 1 year	Required
CATEGORY 2 - VALVES			
High-head injection line valves: MOV-856A,B,C,D	Continuous	1 hour minimum	Required
Charcoal filter dousing valves: MOV 880 A thru K	Open on demand	Available to 1 year	Required
Isolation valves: MOV 894A,B,C,D	Open on S.I.signal	1/2 hr minimum	Required
Recirculation spray valves: MOV-889A,B	Open or close on demand	Available to 1 year	Required
Recirculation pump discharge valves: MOV-1802A,B	Open after injection phase, close on demand	Available to 1 year	Required

Table Q 7.8-3 (Continued)

Equipment Name and Tag Number	Operating Mode	Duration of Operation	Environmental Testing
Containment sump isolation valve MOV-1805	Continuous	Available to 1 year	Required
Residual heat exchanger cooling water supply valves: MOV-822A,B	Open on demand	Available for 1 year	Required
Residual heat exchanger isolation valves: MOV-745A,B 746, 747	Open or close on demand	Available for 1 year	Required
Residual heat loop flow control valves: MOV-638, 640	Inter-mittent	Available for 1 year	Required
Air operated isolation valves	Close on demand	10 seconds after signal	Not required*
CATEGORY 3 - MISCELLANEOUS ITEMS			
Fan cooler motors: 21, 22, 23, 24, 25	Continous	Available for 1 year	Required
Internal recirculation pump motors:	Start after injection phase and continue operating	Available for 1 year	Required
Hydrogen - ystem	Operate on demand	Available for 1 year	Required
Safeguard eq power, control and instrument cable	Continuous	Available for 1 year	Required

* All air operated valves fail in 2 closed conditions.

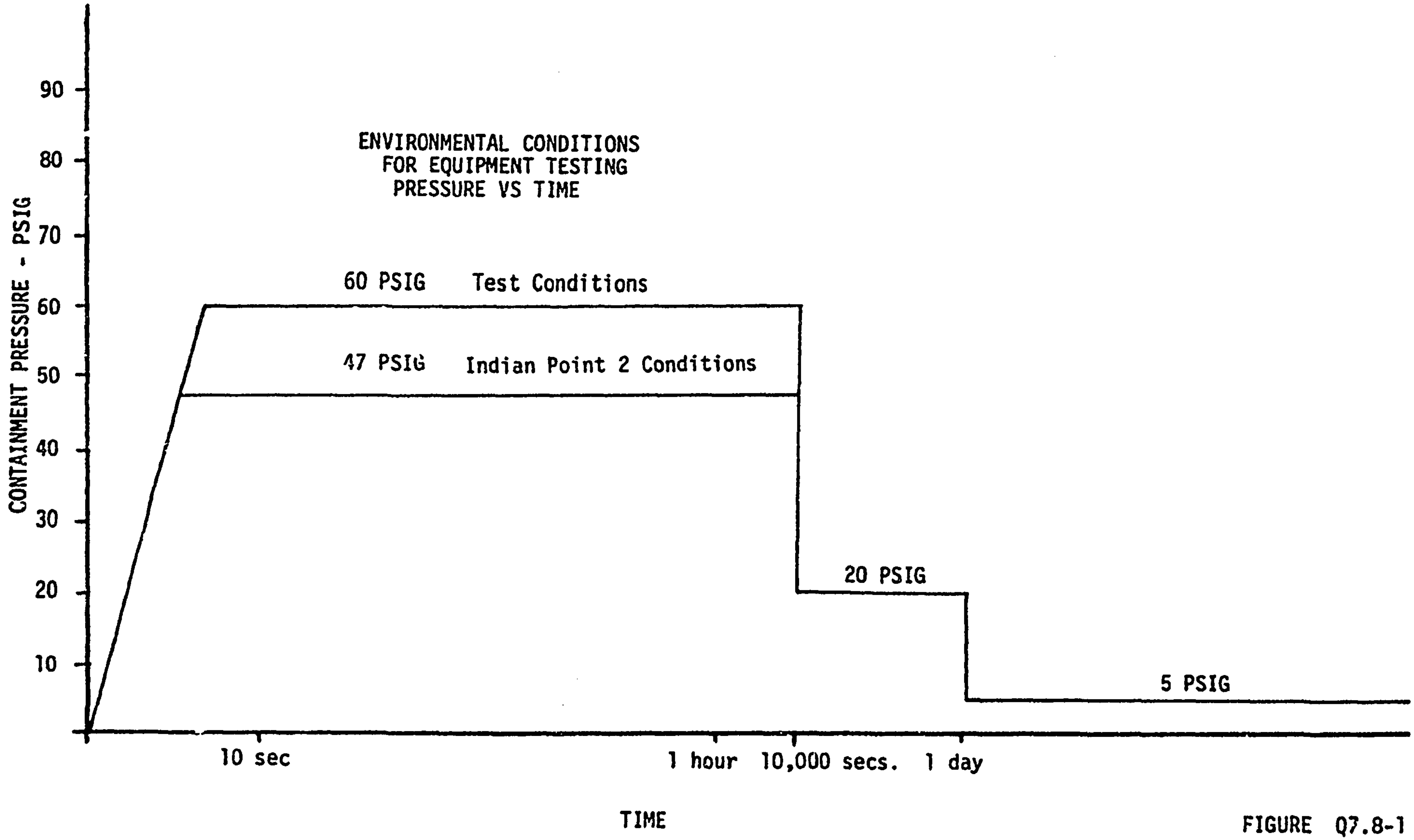


FIGURE Q7.8-1

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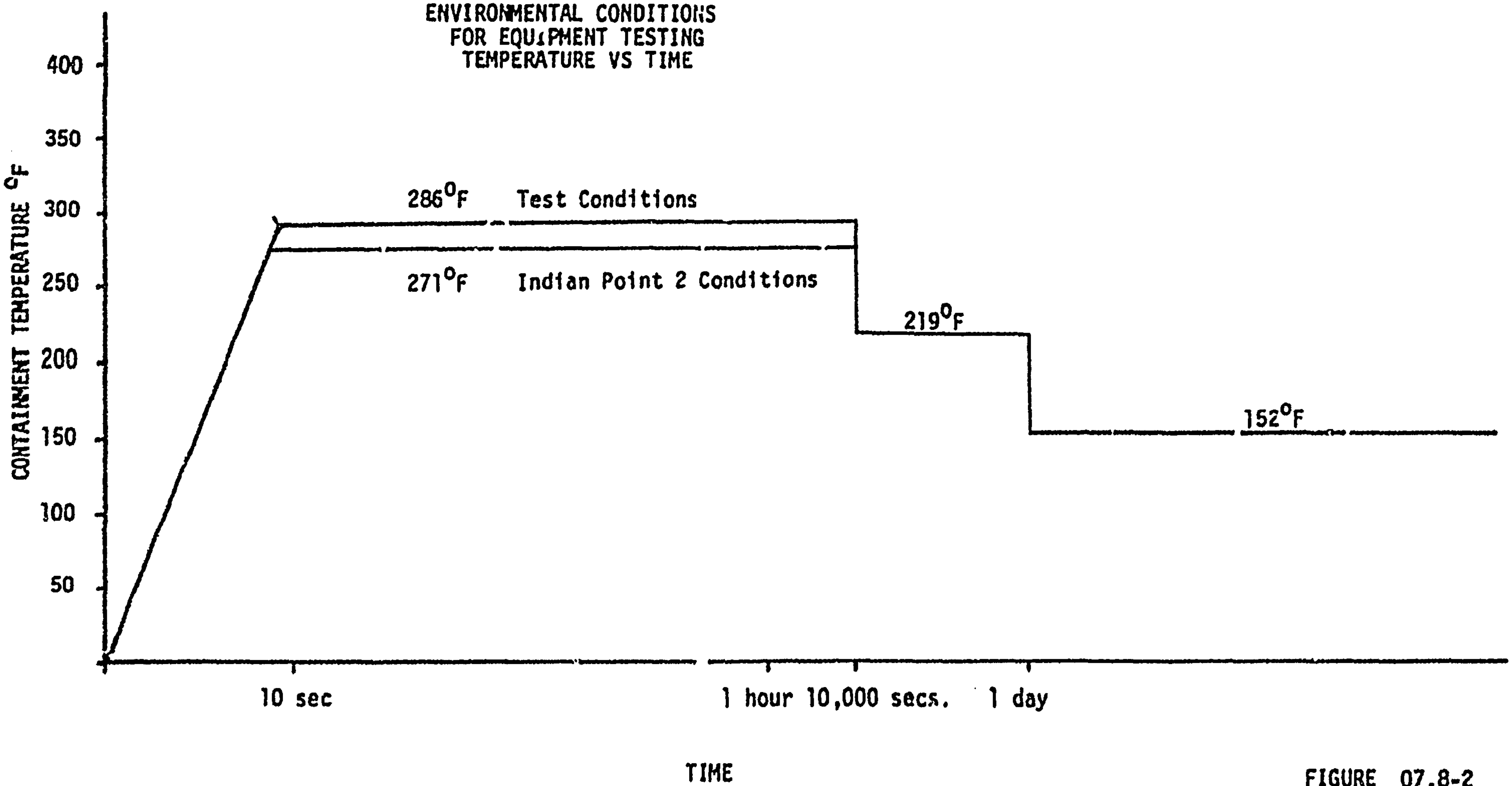

FIGURE Q7.8-2
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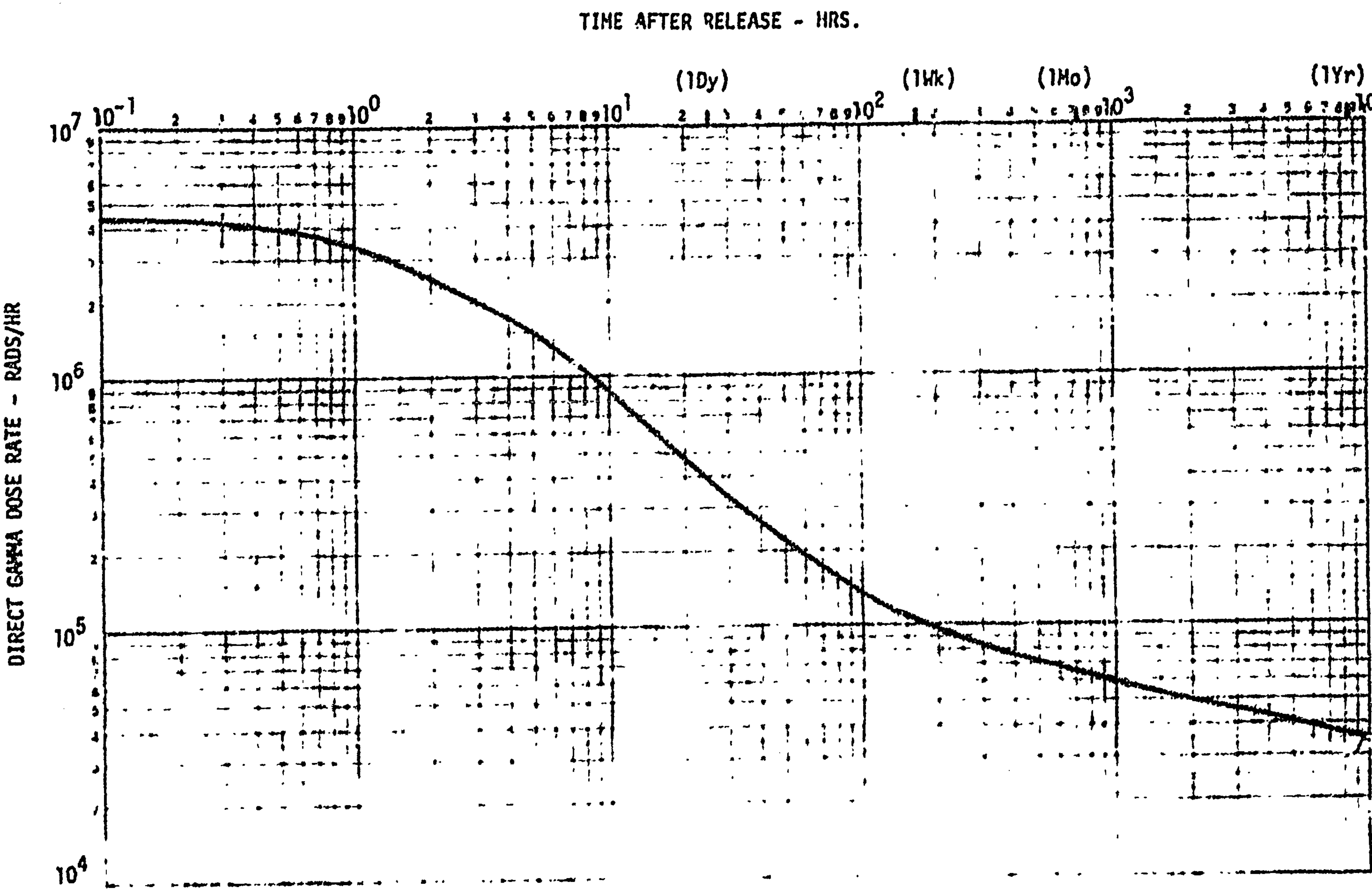

INSTANTANEOUS GAMMA DOSE RATE INSIDE THE
CONTAINMENT AS A FUNCTION OF TIME AFTER RELEASE

FIGURE 07.A-3
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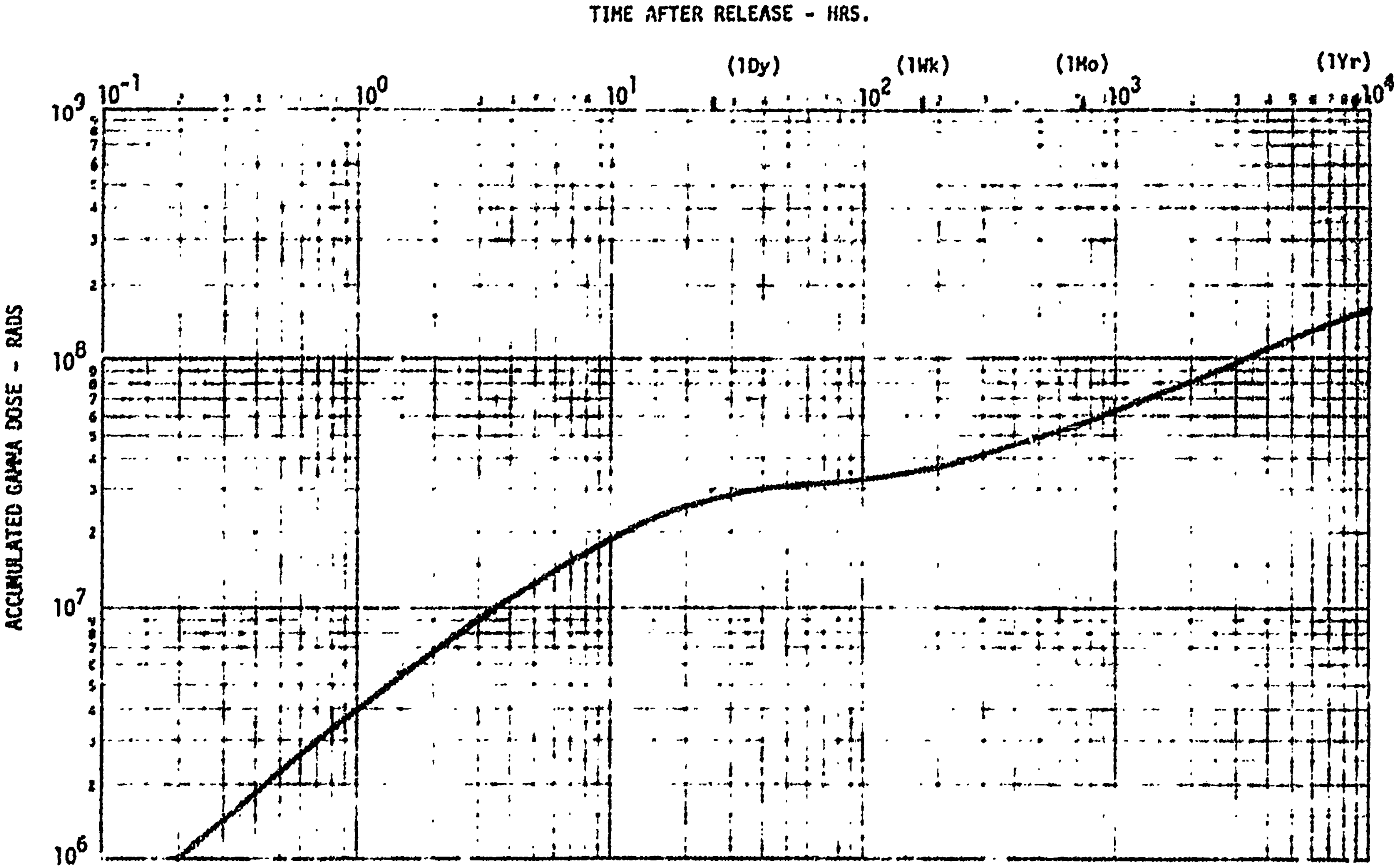

INTEGRATED GAMMA DOSE LEVEL INSIDE THE
CONTAINMENT AS A FUNCTION OF TIME AFTER RELEASE

FIGURE Q7.8-4

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QUESTION 7.9

What are your design bases for reactor protection system and engineered safety feature equipment (e.g., sensors, cabling, pumps) located in the containment or elsewhere in the plant which take into account the effect of radiation on the component? Please describe the analysis or testing performed to determine the long term effect and the design basis accident condition superimposed on the long term effect.

ANSWER

Design basis for the reactor protection system and engineered safety feature equipment radiation exposure are that the equipment must function after exposure associated with the TID-14844 model accident. The maximum anticipated exposure for components located within the containment is 1.6×10^8 rods which is accumulated during one year following the accident. (Note that the integrated exposure for safeguards equipment during 40 years of operation is less than 5×10^5 rads). In the determination of exposure no credit was taken for containment clean-up nor other removal mechanism other than isotope decay. The expected integrated exposure on the outside of the containment building again assuming TID-14844 releases and no credit for clean-up will be less than 10^2 rads integrated over a year at the containment outside surface.

Cross-reference is made to the Answer to Question 7.8 regarding high temperature, pressure and humidity environment.

To establish the combined effect of long term operation followed by exposure to accident conditions inside the containment, selected components were subjected to thermal aging followed by irradiation. In addition components were first irradiated and then subjected to thermal aging. Results of the tests indicate that the components would perform satisfactorily following a Design Basis Accident.

Currently Indian Point Unit No. 2 cables are being tested using the same approach as described above, i.e., irradiation, thermal aging followed by steam exposure and thermal age, irradiation followed by steam exposure. During exposure to steam the cables carry nominal voltage and current.

QUESTION 7.10

Describe how reactor protection system and engineered safety equipment will be physically identified as safety equipment in the plant.

ANSWER

A color code of red, white, blue and yellow is established for analog protection channel sections I, II, III and IV respectively. Large identification plates with the appropriate background color are attached at the front and back surface of each analog rack for identification of analog protection channel racks. Protection and safeguards relay racks are identified similarly on the input side of the racks where protection signals from the various protection channels are received.

Cable trays and cables have numbered tags for identification which, in conjunction with plant drawings can be related to specific functions. The identification tags do not themselves differentiate between protection and non-protection cables and trays.

QUESTION 7.12

Describe what information is available to the control room operator which would allow him to recognize that the door to the reactor protection system panels have been opened improperly (e.g., two doors are open at the same time). Page 7.2.2.3-3 of the Beaver Valley PSAR indicates each panel has an associated annunciator.

ANSWER

Because of the control room arrangement, access to the protection racks is under the administrative control of the plant operator. Opening of a rack door is not annunciated; however, opening of any of the test panel covers which give access to the switches and signal injection points are annunciated on a protection set basis. (i.e., four windows, one for each protection set.) Access to these switches and signal injection points permits the channel to be defeated.

QUESTION 7.13

The following logic diagrams are requested:

a. Activation logic diagram for Safety Injection. Containment Spray and Isolation (similar to Figure 7.2-6 of Ginna FSAR).

b. Analog channel testing arrangement (similar to Figure 7.2-7 of Ginna FSAR).

c. Actuation circuits of engineered safety features circuitry (similar to Figure 7.2-15 of Ginna FSAR).

ANSWER

a) The logic diagrams which illustrate the logic train for Safety Injection, Containment Spray and Containment Isolation may be found in Figure 7.13-1. Numerical coincident requirements are listed in Table 7.2-1 of the FSAR.

b) Refer to Analog Channel Testing, Pg. 7.2-18 and Figure 7.2-7 of the FSAR for required information.

Engineered Safety Features Actuation Instrumentation Description

The Engineered Safety Features actuation circuitry is designed to maintain channel isolation up to and including the bistable operated logic relay, similar to that of the reactor protection circuitry as discussed in Section 7.2. The general arrangement of this layout is shown in Figure 7.13-2 with supplemental details in Figures 7.13-3 and 7.13-4. Although a four channel system is illustrated in Figure 7.13-2 circuitry and hardware layout discussion is sufficiently general to apply to an "n" channel system. Channel separation is maintained by providing separate racks for each analog protection channel and separate relay rack compartments for each logic train. Channel identity is lost in the relay wiring required for matrix logic make-up. It should be noted that although channel individualization is lost, twin matrix logic trains are developed thus assuring a redundant actuation system.

The engineered safety feature bistable drive the logic relay coils "C" and "D" as shown in Figures 7.13-2 and 7.13-4. These logic relay coils are de-energized by their bistables when an abnormal condition exists; exceptions to this de-energized to operate principle are initiation of containment spray and main steam line isolation. The two contacts of each relay are so arranged as to develop the logic matrix required to initiate action. In Figure 7.13-2 these relay contacts are shown directly below the relay coil. Since these coils would normally be energized, their contacts would remain open and thus an open circuit between the voltage source and master actuating relay would exist. De-energizing any of the two logic relay coils would cause their corresponding contacts to close, which would complete the circuit and energize the master actuating relays. Although the illustration here is for a two-out-of-four (2/4) matrix, the design and sequence of operation for any of the logic matrices is the same. The master actuating realy (M) is a latch type relay having two coils; and operate (M/O) and a reset (M/R) coil. Once the logic matrix is made up, as described above, the circuit which energizes the master actuating relay is complete. Figure 7.13-2 illustrates the master actuating relay (M); an enlarged view may be found in Figure 7.13-3. With potential applied to the relay, the operate coil (M/O) is energized thus closing the (M) contacts which energizes the slaves relays (SR's), as shown in Figure 7.13-2. The master relay is latched in this position until the reset coil (M/R) is energized.

Slave relay output from do "Train A" logic system actuate the "A" Safeguards components. Slave relay outputs from "Train B" logic system actuate the Train B Safeguards components. Components not identified with a specific train .. actuated by either logic system.

After a time delay to assure completion of the actuation sequence, the master actuating relay may be manually reset by operating the reset switch (see Figures 7.13-2 and 7.13-3). With the reset coil (M/R) energized, all of the (M) contacts are returned to their de-energized positions as shown in Figure 7.13-2. It should be noted that once reset action is taken, the master relay operation is blocked by the reset relay (R) until the safeguards initiating signal clears, at which time it is automatically unblocked, and restored to service. Resetting the master relay dees not interfere with the operation status of the engineered safety features equipment.

Engineered Safety Features Logic Testing

Figures 7.13-2 and 7.13-3 and 7.13-4 illustrate the basic logic test scheme. Test switches are located in the associated relay racks rather than in a single test panel. The following procedures indicate the method of testing the logic matrixes:

(a) Test of either train A or train B is made at one time; this is under administrative control.

(b) A selection of the function to be tested is made. Figure 7.13-3 for example, illustrates some of these functional matrixes.

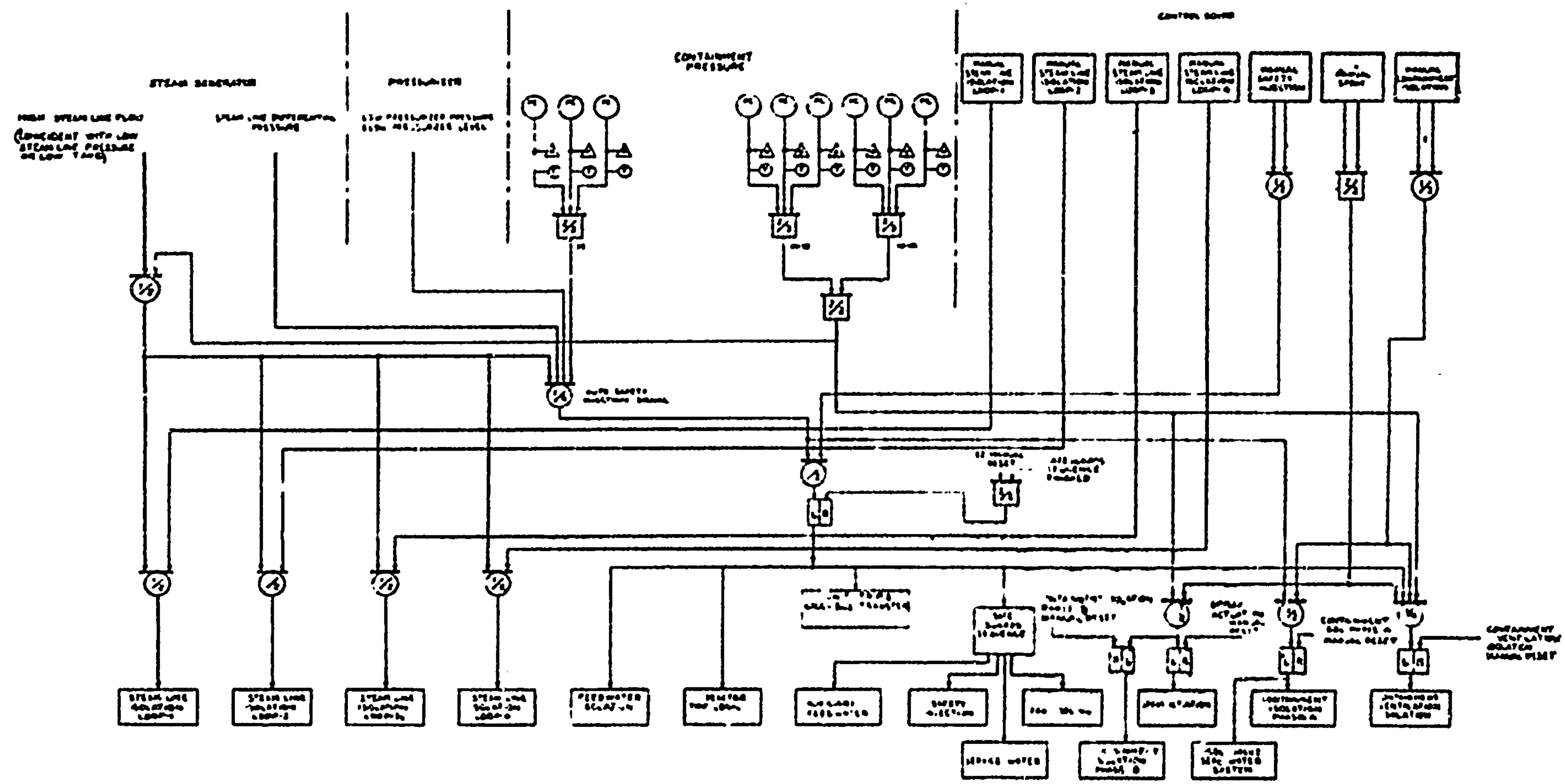
(c) The logic test switch is a dual function switch which is first turned to operate one series of contacts and then depressed to operate other contacts. Turning the logic test switch to the test position opens the circuit to the master actuating relay by opening all logic test switch contacts shown in Figure 7.13-3, or represented by the opening of switch contacts position 1 as shown in Figures 7.13-2 or 7.13-4. Turning the logic test switch to the test position also energizes the "on-test" lamp (L_4 or L_5) by closing contacts 3 of the switch. The master actuating relay is removed from this part of the test in order to avoid unintentional starting of the engineered safety features equipment. Intentional start is available through the other train that has operational status and the other functional matrices not under test.

(d) Depressing the logic test switch, the circuit which energizes the logic relay coil is de-energized, thus closing logic relay contacts of that coil, i.e., depressing of logic test switch (C_1) closes contacts 2 as shown in Figure 7-22 or 7-24. By performing the above sequence for C_2, C_3, or C_4, it is possible to simulate all actuating logic combinations required to develop the matrix.

(e) Proper test development of a logic mstrix would be indicated by the lighting of the matrix test lamps, as shown in Figure 7.13-4 as L_2 or L_3.

(f) When the testing of the logic matrix is complete, the equipment is returned to operational status by turning all test switches to the operate position. The control board annunciatro warns the operator of any test switch left in the test position; thus return to operational status by action of the individual doing the test is verified by the operator at the control board. Testing procedures for the logic matrix of train B is identical to that described above for train A.

(g) Verification of the integrity of the master actuating coil and slave relay coils is made by connecting an ohmmeter across the coil terminals.



1. SPRAY ACTUATION BI-STABLES ARE ENERGIZED-TO-ACTUATE

Figure 7.13-1

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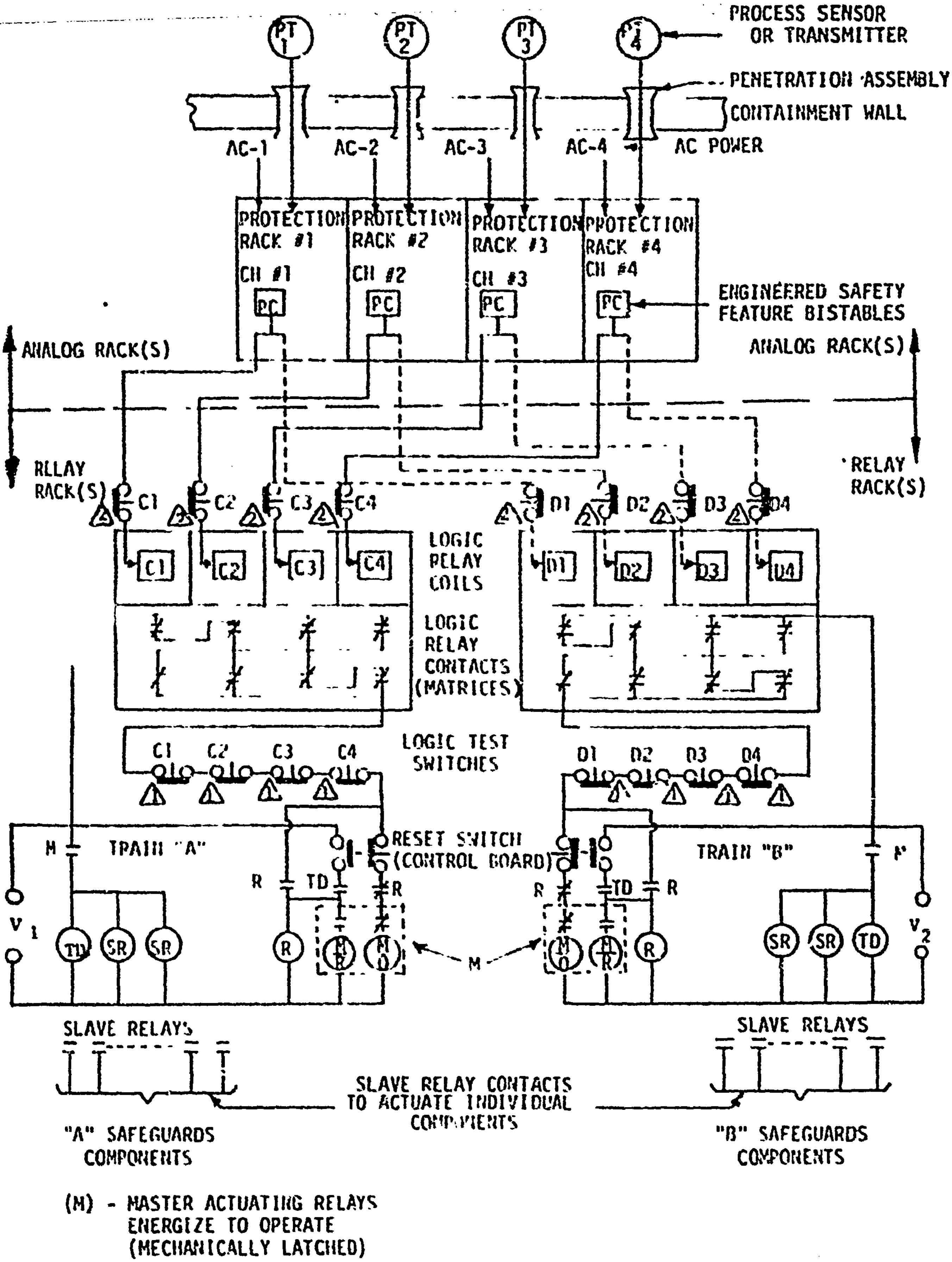
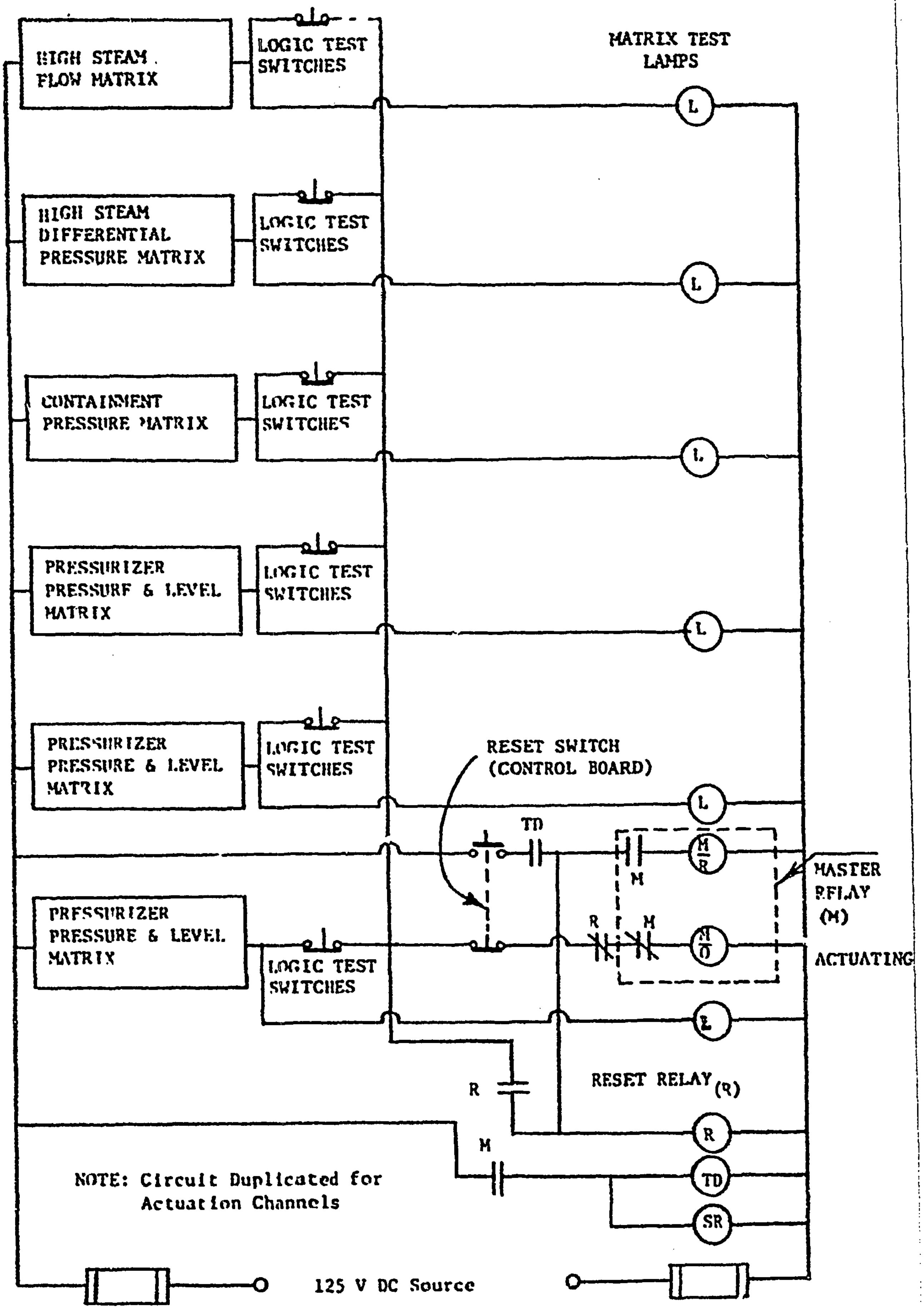


ACTUATION CIRCUITS OF ENGINEERED SAFEGUARDS

FIGURE 7.13-2



SIMPLIFIED DIAGRAM FOR OVERALL LOGIC RELAY TEST SCHEME

FIGURE 7.13-3

C_1, D_1: Relay coil, channels A and B, resp.

L_2, L_3: Matrix test lamp channels A and B, resp.

L_4, L_5: "On test" labeled lamp, channel A and B resp.

L_1 : Bistable proving lamp.

M_A, M_B: Master relay, channel A and B, resp.

S_1 : Bistable trip switch, two positions, normal position shown.

C_1 : Relay logic test switch channel C_1: Operation is as follows:

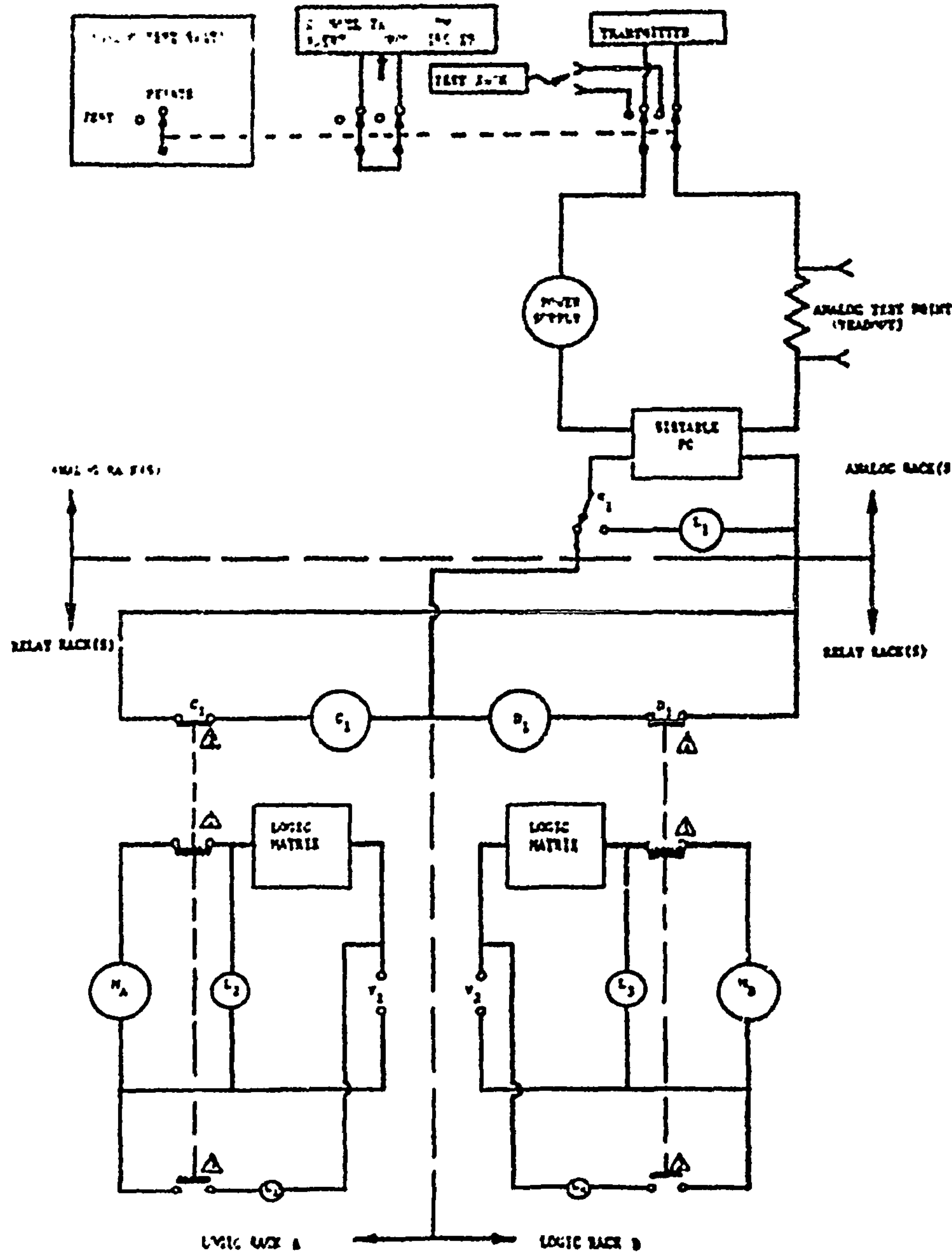
△1 - Opens when switch is turned to test position

△2 - Opens when switch is depressed in test position

△3 - Closes when switch is turned to test position

D_1 : Relay logic test switch channel D_1: Operation same as for S_2.

V_1, V_2: 125 V D.C., sources 1 and 2, resp.



SIMPLIFIED DIAGRAM RELAY LOGIC CHANNEL TESTING FIGURE 7.13-4

Question 7.14

Page 7.2-26 of the FSAR indicates that the Low Reactor Coolant Flow Trip has been modified from the Ginna design (a two-loop plant) to provide a direct reactor trip on undervoltage on 2/4 reactor coolant pump buses. This would indicate that three-loop operation is permissible. Are any manual adjustments required to the reactor protection system for the three-loop mode of operation? If so, please identify the adjustments and show that unsafe operation cannot result from their misadjustment.

Answer

In order to operate with a reactor coolant loop out of service (three loop operation), the overtemperature ΔT trip set-point calculation would have to be modified. This set-point is calculated for each reactor coolant loop as follows:

$$\Delta T_{sp} = \Delta T_o \left[K_1 - K_2 (T_{avg} - 570) + K_3 (P - 2235) - f(\Delta I)\right]$$

where T_{avg} = reactor coolant loop average temperature, °F

P = pressurizer pressure, psig

ΔI = difference between upper and lower long ion chamber current readings

Sustained operation with a reactor coolant loop out of service is a rare event. When this mode of operation is chosen, the variables K_1, K_2 and K_3 must be adjusted and the overtemperature ΔT trip channels must be recalibrated. These adjustments and calibrations must be made in the protection system racks and are performed as is done for four loop operation before initial startup and during normal calibration procedures. The set-point adjustments are made based on limits as set forth in the Technical Specifications for three loop operation and sufficient safety margin is maintained for three loop operation.

To provide assurance that the reactor is in a safe condition and would not exceed design limits in the event of an anticipated transient during three loop operation, the P-8 permissive set point during normal four loop operation will be set at approximately 60%* power. This set point for P-8, in conjunction with the overtemperature ΔT trip set points for four loop operation, will prevent the DNB ratio from going below 1.30. For the plant to return to normal three loop power levels, the overtemperature ΔT trip set points will be adjusted only by members of the technical service bureau of Con Edison under the direct supervision of a member or members of the Operations Staff of the Con Edison Nuclear Power Generation Department. Following the completion of the required instrumentation adjustments, the P-8 permissive set point will be reset to 75% power. P-8 will be placed at the approximately 60%* level when the loop out of service is restored to operation.

* Precise set point limit to be specified in Technical Specifications.

Question 7.14

Page 7.2-26 of the FSAR indicates that the Low Reactor Coolant Flow Trip has been modified from the Ginna design (a two-loop plant) to provide a direct reactor trip on undervoltage on 2/4 reactor coolant pump buses. This would indicate that three-loop operation is permissible. Are any manual adjustments required to the reactor protection system for the three-loop mode of operation? If so, please identify the adjustments and show that unsafe operation cannot result from their misadjustment.

Answer

In order to operate with a reactor coolant loop out of service (three loop operation), the overtemperature ΔT trip set-point calculation would have to be modified. This set-point is calculated for each reactor coolant loop as follows:

$$\Delta T_{sp} = \Delta T_o \, [K_1 - K_2 \, (T_{avg} - 570) + K_3 \, (P - 2235) - f \, (\Delta I)]$$

where T_{avg} = reactor coolant loop average temperature, °F

P = pressurizer pressure, psig

ΔI = difference between upper and lower long ion chamber current readings

Sustained operation with a reactor coolant loop out of service is a rare event. When this mode of operation is chosen, the variables K_1, K_2 and K_3 must be adjusted and the overtemperature ΔT trip channels must be re-calibrated. These adjustments and calibrations must be made in the protection system racks and are performed as is done for four loop operation before initial startup and during normal calibration procedures. The set-point adjustments are made based on limits as set forth in the Technical Specifications for three loop operation and sufficient safety margin is maintained for three loop operation.

QUESTION 7.15

Page 7.2-1 of the FSAR discusses the Control Room. Please provide the following additional information:

a. What communication systems are available to the Control Room for:

(1) Operational control (e.g., page party).

(2) Administrative control (e.g., Bell system).

(3) Special purpose (e.g., sound powered phones)

(4) Emergency (e.g., radio)

ANSWER

1. PLANT COMMUNICATIONS SYSTEM

Central Control Room No. 2

The public address system for the Indian Point No. 2 plant consists of three channels, namely, "Page No. 2", "Party No. 3", and "Party No. 4". The "Page No. 2" and "Party No. 3" channels are common to both the primary (nuclear) and secondary (conventional) portions of the plant. They operate independent of or can be merged with the existing "Page No. 1" and "Party No. 1" channels of the Indian Point No. 1 plant as desired by the Unit No. 2 operator in the central control room. "Party No. 4" provides an additional channel in the primary portion of the Unit No. 2 plant only and cannot be merged with any other channels. Speakers for monitoring each of these lines are located in the central control room.

Three handsets are located on the No. 2 plant operator desk in the central control room. The "Page" handset is used for page purposes only and calls orginating from same will be heard on all loudspeakers in the primary and secondary portions of the No. 2 plant. The remaining two "Page-Party" handsets are used for loudspeaker paging and party-line conversations throughout the No. 2 plant, as selected by the central control room operator. The No. 2 plant operator has the means on his central control room desk to isolate or merge the "Page No. 2" and "Party No. 3" channels on an intraplant basis.

1. PLANT COMMUNICATIONS SYSTEM (Continued)

An emergency alarm switch is provided on the No. 2 plant central control room desk, which will actuate the existing alarm oscillator and connect same to the Page No. 2 channel. Merging of the page channels on a intra-plant basis will also merge the alarm system, should it be actuated under this merged condition.

A switch is provided on the central control room desk, which will allow all outdoor speakers of the No. 2 plant to be turned off at night.

All calls initiated from the Indian Point No. 2 handsets in the primary and secondary plant will light a lamp, sound a buzzer, and are heard over the party-line speaker in the central control room. A "Page" call from a plant area handset will only go to the control room. Removal of either "Page-Party" handset from its desk edge carriage will silence the buzzer and speaker, but the lamp will remain lighted until the conversant handsets are returned to their respective hook switches. It is possible to carry on two independent party-line conversations simultaneously between the central control room and the primary (nuclear) plant of Indian Point No. 2.

Provisions for merging the page and party channels of the Unit No. 2 system with the future page and party channels of Unit No. 3 have been incorporated into the control switches on the control room desk.

The plant communications system is normally powered from MCC 26B which is connected to 480 V bus 6A. In the event of loss of all normal a-c power, 480 V bus 6A and MCC 26B are automatically connected to the emergency diesel generators.

2. BELL TELEPHONE SYSTEM

There are two (2) Bell Telephones located on the Unit No. 2 control room desk. One phone has a direct Auxiliary Line and the operator can dial outside the plant directly to all areas, local and long distant.

The other phone has a restricted line and the operator can dial within the plant and Con-Ed system plus dial into a limited local area by dialing "9", then the number.

The inplant telephone system power supply is battery-backed in case of plant blackout conditions.

3. SPECIAL PURPOSE

There are no sound powered phone facilities available for Indian Point Unit No. 2.

4. EMERGENCY

Wired communications will be provided between the Indian Point Unit No. 2 control room and Indian Point Unit No. 1 Base Radio Station. This will provide emergency radio communications between Indian Point Unit No. 2 and the System Operator. In the event of loss of all normal a-c power, the Station Radio Transmitter can be connected to one of the two Unit No. 1 battery-backed special a-c load boards.

QUESTION 7.17

Page 6.2-12 of the FSAR states that during safety injection, the first low level alarm ($\sim$15 minutes) on the refueling water storage tank sounds and the operator should take appropriate action to assure that only a certain number of pumps are operating. On a second alarm ($\sim$22 minutes) the operator switches to the recirculation mode. Figure 6.2-1 shows one level indicator and one level alarm on the refueling water storage tank. Please describe the instrumentation provided to the operator in sufficient detail to show that a single failure cannot lead to improper operation or an interruption of cooling to the core.

ANSWER

Refueling water storage tank level is measured by two independent sensors.

a) a level indicator at the tank also contains an alarm switch to actuate the low level alarm

b) a transmitting channel provides level indication in the control room and a low-low level alarm in addition to back up actuation of the low level alarm.

In the case of a DBA and full operation of all safeguards and spray pumps, the low level alarm will annunciate after approximately 20 minutes. At this time the operator's attention is directed to the tank level indicator, and when a predetermined level is reached (after a further 2 minutes), he is instructed to proceed with the changeover sequence. At this same level, the low-low level alarm annunciates. Information on the level of water in both the recirculation and containment sumps is also available to the operator during this period via the sump level instrumentation.

In view of the information provided to the operator, together with the procedure which he is required to follow, no single instrument failure would cause him to follow a course of action which could in any way jeopardize core cooling.

QUESTION 7.18

Page 7.3-10 of the FSAR states that the a-c power supply for the rod drive system uses a single overhead run of enclosed duct which is bolted to, and therefore comprises part of, the power cabinet arrangement. What is the length of this run for Indian Point 2?

ANSWER

The length of the run between the top of the reactor trip breaker switchgear cabinet and the leading edge of the rod control system cabinets, including vertical sections, is 23'-6". The length of the run across the top of the rod control system cabinets is 21'-0". Therefore the total length is 44'-6".

QUESTION 7.19

What are your design bases for the safety related electrical equipment (control room or other equipment rooms) which take into account the loss of the air conditioning and/or ventilation system(s)? Please describe the analyses performed to identify the worst cast environment (e.g., temperature, humidity) for the instrumentation, control, and electrical equipment. What is the limiting condition with regard to temperature that would require reactor shutdown? What is your basis? Describe any testing (factory and onsite) which has been or will be performed to determine the equipment characteristics for this environment.

ANSWER

The safety related electrical equipment is designed to operate and perform its design function within specified safe limits without degradation of performance (accuracy, repeatability, time response) under the expected normal and abnormal ambient conditions associated with its location. The normal ambient design temperature range is 75°F plus or minus 10°F for control room located equipment. The abnormal ambient condition associated with the design of control room located safety equipment is 120°F for short term operation associated with a loss of air conditioning. Safety related electrical equipment in other than the control room is designed to operate under the worst case environment for which it is required to perform its function. For example, in the containment the out-of-core neutron detectors and cables are designed to operate in an environment of 175°F, 90 percent relative humidity and 100 psig for a short time duration (8 hours). Safety related process transmitters and sensors throughout the plant will function normally in an environment up to 140°F at 100 percent relative humidity.

The control room contains the majority of the safety related equipment, therefore represents the limiting condition for temperature that would require reactor shutdown. The ventilation systems of concern outside the control room are designed to take a single active failure; thus, analysis of loss of ventilation is not considered necessary. The control room ventilation system is designed to accommodate certain active or passive failure. Operator action is not required to prevent unacceptable temperatures in safety related equipment located in the control room.

The Indian Point Unit 1 and Indian Point Unit 2 central control rooms are common, but the Indian Point Unit 1 and Indian Point Unit 2 control room air conditioning systems are independent and interties are not provided. The Indian Point Unit 2 air conditioning system consists of one 9,200 cfm air conditioning unit, one 1,840 cfm HEPA-charcoal filter unit, two (for redundancy) 1,840 cfm booster fans for filter operation; one 2,000 cfm emergency by-pass fan located in the supervisory control panel exhaust system, and associated duct work and dampers. All active components, except the by-pass fan, and the filter unit are located in the Indian Point Unit 1 heating, ventilation and air conditioning equipment room which is located above the Indian Point Unit 1 control room. They by-pass fan is located in the Indian Point Unit 2 control room area. The Indian Point Unit 2 air conditioning unit is supplied with water from the Indian Point Unit 2 service water system. The Indian Point Unit 2 air conditioning and ventilating system, including the air conditioner, erergency by-pass fan and filter fans, will be powered from one of the busses serviced by the emergency diesel generators and will start automatically following a blackout.

It is design policy that the functional capacity of the control room shall be maintained at all times inclusive of accident conditions, such as, an MCA or a fire. . . .(cf. p. 7.7-3). Hence, to specify the limiting conditions, two cases must be considered, namely, failure of the air conditioning system during normal operation and failure subsequent to or coincident with an MCA. Considering first the case where failure occurs during normal operation. The objective here is to insure that temperatures do not exceed levels where reactor protection sytem, and safeguards system set points are altered appreciably, and to insure that remote hot shutdown capability is not compromised. The maximum tolerable upper limit is 120°F.

On a loss of the Indian Point Unit 2 air conditioning system, the control room temperature under operating conditions and outside design temperatures of 93 DB and 75 DB will rise to a level where the heat released to the room by the equipment and lights will balance the transmission lossed through the walls, floor and ceiling. This temperature has been calculated for the following two conditions:

Case No. 1

Unit No. 1 air conditioning system operating with 10% outside air.

Case No. 2

Unit No. 1 air conditioning system operating with 100% recirculated air (no outside air).

In both Case No. 1 and No. 2, when the loss of Unit 2 air conditioning occurs the following action will take place:

A. All lights except emergency lights will be turned off.

B. The emergency by-pass fan on the supervisory control panel exhaust system will start automatically.

Under these conditions, the maximum room temperature will be:

Case No. 1	109 F
Case No. 2	104.6 F

The room supply air to the supervisory panel will be at a temperature approximately two (2) degrees lower than these room temperatures due to stratification. Therefore, the supervisory panel tempe tures will be approximately:

Case No. 1

109F - 2F + 4.7F = 111.7F

Case No. 2

104.6F - 2F + 4.7F = 107.3F

(NOTE: 4.7F = T.R. DUE TO HEAT PICKUP IN SUPERVISORY CONTROL PANEL based on 3kw load)

There is no latent heat released to the room from equipment and an insignificant amount from the operators. Therefore, the humidity will remain 50% or lower and will decrease as the temperature increases.

The design basis is that the safety related analog type electrical equipment will perform its required functions within the required accuracies for ambient conditions of 120°F. If the control room temperature reaches 110°F, steps will be taken to bring the reactor to a safe and orderly shutdown. If the temperature should reach 120°F before action is taken, the reactor will be tripped (manually). Control room annunciation is not provided for high ambient temperatures or loss of air conditioning.

As noted under Case No. 2, the outside make-up air to Indian Point Unit 1 control room is assumed as being cut off. The outside damper control will be modified to close on an SI signal, in order to accomplish this.

During the post accident period the Indian Point Unit 2 fan/filter system is required to pressurize the Indian Point Unit 1/Indian Point Unit 2 control room area to prevent inleakage of fission products and to remove halogens from the control room atmosphere. Since the Indian Point Unit 2 system ductwork is not designed to recirculate the Indian Point Unit 1 control room atmosphere, portable fans may be required to insure adequate circulation through the charcoal filter system.

Factory testing has been performed on various safety related systems such as: process control, nuclear instrumentation and logic relay racks. This testing involved, demonstrating operation of proper safety functions as increased ambient temperatures to at least 120°F for process control and nuclear instrumentation. The logic relay racks were tested to determine temperature rise of the cabinet under full load conditions. From this test it was determined that the relays would perform their function in an ambient temperature of 130°F.

QUESTION 7.20

For some relatively small breaks in the primary system the only signal available to initiate scram would be the low pressurizer pressure signal. As we have discussed with you previously, it would be desirable to add a diverse scram signal which would assure timely reactor scram in the event that the pressurizer signal fails. Please state your intent in this regard and provide a supporting analysis to demonstrate that the core will be protected over the range of break sizes which would require this scram signal to shutdown the reactor.

ANSWER

The signal selected to provide a diverse reactor trip to assure timely reactor trip in the event that the pressurizer signal fails is the high containment pressure signal. Initially, the high containment pressure signal was set at 5.0 psig to provide a diverse signal for actuating the Emergency Core Cooling System. This signal is now set at 2.0 psig to provide a diverse reactor trip. An analysis has been performed to demonstrate that the core will be protected over the range of break sizes where this diverse signal is required.

In order for the signal to provide the necessary diversity, it should be actuated before the pressurizer empties. Figure 7.20-1 presents a plot of time for the pressurizer to empty as a function of break size for a range of small loss of coolant accidents. A plot of time to reach the high containment pressure set point is also presented on this figure. In the calculations, to determine the containment pressure transient conservatively, high heat removal from the fans and condensing surfaces were assumed to minimize the containment pressure increase.

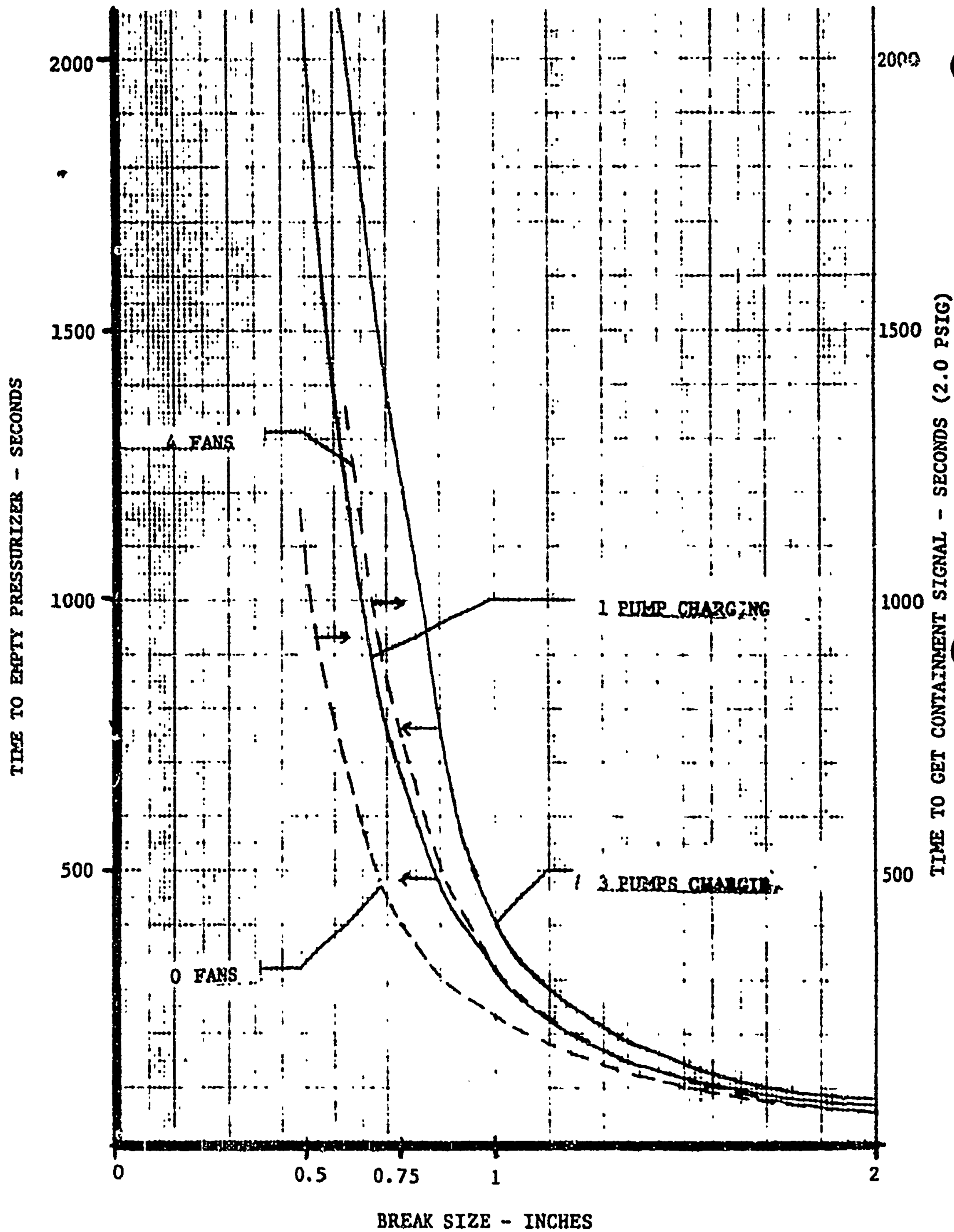


FIGURE 7.20-1

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QUESTION 7.21

Provide a description of the instrumentation that will be available to the plant operator and the procedures he will use to assess the course of each of the postulated accident conditions.

ANSWER

The postulated accidents which require operator diagnostic action are a loss of reactor coolant, loss of secondary coolant, and a steam generator tube rupture. A detailed series of emergency instructions are provided for the operator which include symptons, immediate action, and recovery. The response to Question 7.8 lists those instruments inside the containment which are designed to withstand the post design basis accident conditions.

In many respects the behavior of the plant parameters displayed in Control Room will be similar following any one of the above listed incidents. For example, the symptoms of all three accidents which should immediately become apparent to the operator are falling pressurizer pressure and level, and in the case of slower accidents, increased charging pump speed prior to trip. A brief description of each accident and the objects of the recovery procedure are given below.

1. Loss of Reactor Coolant

This emergency results in a breach of the primary pressure boundary such that maximum charging flow and reactor coolant pump seal injection flow can no longer maintain pressurizer level. Safety injection and reactor trip will be initiated by the falling pressurizer pressure and level on a time scale dependent upon the magintude of the break and injection flow will increase with decreasing reactor pressure. The accumulators will automatically discharge their fluid inventory when the reactor coolant system pressure drops below the accumulator set point and in case of rapid depressurization leading to very low reactor pressure, the residual heat removal pumps will commence injection and refill of the reactor vessel. Long-term control and cooldown of the reactor coolant system is by recirculation of spilled reactor coolant from the containment sump. This is carried out by the internal recirculation pumps or by a high-head pump or pumps taking suction from the outlet of the residual heat exchangers. Containment pressure increases due to the release of energy from the reactor coolant system to the containment; containment Phase A isolation will result from the safety injection signal. Spray actuation will occur at approximately 50 percent of containment design pressure; spray actuation is accompanied by containment Phase B isolation.

The main function of the operator in this type of accident is to:

a. carry out the change over from the injection phase to the recirculation phase and to check for possible existence of a leak in an injection line and carry out the relevant isolation procedure.

b. eventually set up a low head path recirculation once the reactor coolant pressure has been reduced, if initial recirculation is via the high-head pump path.

c. with the completion of containment spray injection the operator aligns the system for combined core cooling and containment circulation spray for a period of 24 hours after the accident.

2. Loss of Secondary Coolant

This emergency is the result of a break in a main steam line or feed line and it will result in a reduction in reactor coolant temperature and pressure at a rate which is dependent upon the size and location of the break. The reactor automatic protection system is designed to shut the plant down safely and this system, together with the action of the safety injection system in pumping boric acid to the reactor coolant, will ensure continued shutdown.

In the case of a full blowdown of one steam generator, reactor coolant temperatures and pressures will have fallen to the region of 400°F and 600 psi in two or three minutes and safety injection flow will be underway. The continued action of safety injection will repressurize the reactor system to the shut-off head of the pump with little change in reactor coolant temperature. At this state the water level will have returned in the pressurizer, and in fact, the pressurizer may be full of subcooled fluid. The subsequent course of the accident will be for reactor temperatures and pressure to increase under the influence of residual heat and a steady-state condition will be attained when core temperatures are high enough to enable residual heat to be removed by the intact steam generators by steam dump, the reactor system will be repressurized to the pressurizer safety valve setting (2485 psi) with discharge of reactor fluid to the containment. Operator action should therefore be to eliminate or at least minimize this effect by dumping steam from the intact steam generators. At an early stage in the accident the operator should also isolate auxiliary feedwater flow to the faulty steam generator to minimize thermal shock.

3. Steam Generator Tube Rupture

This accident results in leakage of reactor coolant into the plant steam system and as a result pressurizer pressure and level will decrease leading to a low pressurizer pressure reactor trip. The automatic reactor coolant cooldown following plant trip will drain the pressurizer of any remaining water, actuating safety injection. After passing through a minimum, dependent upon the size of the leak, the reactor pressure will be increased to a stable value of about 1400 psi by the continued action of safety injection. In the absence of action by the operator, continued leakage into the steam generator, together with auxiliary feedwater flow, would result in a rise of the steam generator water level into the steam lines of the faulty steam generator as it becomes water solid.

Operator action should be to minimize the contamination of the steam system by prompt isolation of the faulty steam generator. To achieve this, steam dump must be carried out using initially all four steam generators. Auxiliary feedwater flow to the faulty steam generator should be cut off when it has been identified.

SYMPTOMS

The following symptoms may arise in a plant which has undergone, or is undergoing, one of the above accidents.

1.

1. Pressurizer low-pressure alarm
2. Pressurizer low-level alarm
3. High containment pressure alarm
4. Containment high-radiation monitor alarm
5. Containment high sump levels; i.e., either or both containment sump and circulation sump.
6. Steam generator low-pressure alarm
7. Rapidly decreasing reactor coolant average temperature
8. Air ejector radiation monitor alarm
9. Low steam generator water level alarm
10. High steam flow in one or more steam generators and/or steam flow/feedwater flow mismatch.
11. Steam pressure low in one line with respect to other three.
12. Increased charging pump speed
13. Steam generator blowdown high-radiation alarm

MANUAL ACTIONS

In the event that the safeguards equipment has not been properly actuated, the operator should take action to place the equipment in operation manually.

1. Verify that reactor trip and safety injection initiation have both taken place following the "S" signal.

2. Verify that containment spray has taken place following the "P" signal.

3. Verify that the safety injection system is pumping in boric acid from the boric acid tank, or from the refueling water storage tank when the boron injection tank level reaches the low level set points.

4. Verify with the monitor lights that the valves associated with safety injection are in the proper position after "S" initiation.

5. Verify that the main feedwater pumps have tripped and that the auxiliary feedwater pumps are actuated.

6. Check for proper operation of the diesel generators.

IDENTIFICATION OF ACCIDENT TYPE

The above manual actions are those which require the operator's attention in the event of a large loss of reactor coolant or secondary fluid leading to rapid reactor trip and actuation of safety injection. The operator will determine the accident type subsequently. However, in the event of a slow transient in which reactor trip is delayed for a few minutes, the operator may be able to decide which type of fault has occurred prior to reactor trip and initiate safety injection. Therefore, two cases arise and are dealt with below and in Figure Q 7.21-1:

a. After Reactor Trip and Safety Injection Actuation.

1. Observe the steam pressure in all steam generators.

2. If the pressure is rising or normal in all steam generators together with low pressurizer pressure and level, the accident is either a loss-of-coolant accident or a steam generator tube rupture. These can be distinguished as follows:

(a) If there is an increase in containment pressure, a containment high-radiation alarm, rising sump water level, or any combination of these symptoms, the situation is uniquely defined as a loss-of-coolant accident.

3. If the pressure is abnormally low in one steam generator, coincident with low pressurizer pressure and level, the accident is a secondary side loss-of-coolant accident.

4. In the case of a secondary side break, the approximate break location may be determined as follows:

   (a) Steam stop valves closed and similar behavior in the steam generators indicates that the break is downstream of these stop valves.

   (b) Pressure in one steam generator substantially lower than the others, together with rising containment pressure, indicates that the break is in the lower pressure steam line or feedwater line and inside the containment.

   (c) As in (b) above, but with no increase in containment pressure, no rise in containment sump level, and with all steam stop valves closed indicates a break in the lower pressure steam line outside the containment and upstream of the stop valve.

   (d) As in (c) above, with only one stop valve closed invokes two possibilities. If the lower pressure steam line is the one with the closed stop valve, the break is outside the containment and upstream of the closed stop valve. However, if the higher pressure steam line contains the closed valve, the break is outside the containment, but not upstream of the closed stop valve.

b. Before Reactor Trip and Safety Injection Actuation (See Figure Q 7.21-2.

In a slow loss-of-coolant accident or a steam generator tube rupture, T_{av} will vary very little before reactor trip, whereas a definite and continuous decrease T_{av} will be observed in a steam side break accident. The loss-of-coolant accident and the tube rupture accident can then be differentiated as described above.

As described in the above procedures for recovery, the instrumentation used by the operator will consist of flow indicators, level indicators, pressure indicators, valve position monitors and equipment operational status lights. The same instrumentation, along with a radiation monitor and sampling provisions, will be used in post accident flow of the ECCS performance.

QUESTION 7.22

Discuss how both the control and power circuits for the boron injection tank valves meet the IEEE 279 criteria for both opening and closing actions.

ANSWER

The boron injection tank discharge line is arranged with two normally closed valves in parallel through two normally open valves in series. One parallel and one series valve receive power for the valve motors from motor control center 26A and control from Channel #1. The other parallel series valve combination receives power from motor control center 26B and control from Channel #2. The redundant control channels receive signals from 3 level instruments with logic derived from 2/3 signals.

QUESTION 7.23

Submit the procedures for the testing of the initiating and control instrumentation for Engineered Safety Features.

ANSWER

SAFEGUARDS INITIATING CIRCUITRY

The safeguards actuation circuitry and hardware layout are designed to maintain circuit isolation through the bistable operated logic relays. The channelized design follow through is shown on the Figure Q 7.23-1 block diagram.

The safeguards bistables, mounted in the analog protection racks, drive both "A" and "B" logic matrix relays. Each matrix contains its own test light and test circuitry. The "A" and "B" logic matrices operate master relays for actuating channels A and B respectively as shown in Figure Q 7.23-2. Control power for logic channels A and B is supplied from DC sources 1 and 2 respectively. These redundant actuating channels operate the various safeguards components required, with the large loads sequenced as necessary.

Manual reset of the safeguards actuation relays may be accomplished 2 minutes following their operation. Once reset action is taken, the master relay is reset and its operation blocked until the safeguards initiating signal clears at which time it is automatically unblocked and restored to service.

Protection channel identity is lost in the intermixing of the relay matrix wiring. Separation of A and B logic channels is maintained by the separate logic racks.

Analog Channel Testing

The basic elements comprising an analog protection channel are shown in Figure Q 7.23-3. This system consists of a transmitter, power supply, bistable, bistable trip switch and proving lamp, test signal injection switch, test signal injection jack and test point.

Each protection rack will include a test panel containing those switches test jacks and related equipment needed to test the channels contained in the rack. A hinged cover encloses the signal injection switch and signal injection jack of the test panel.

Opening the cover or placing the test operate switch in the "Test" position will initiate an alarm identifying the rack under test. These alarms are arranged on a rack basis to preclude entry to more than one redundant protection rack (or channel) at any time. The test panel cover is designed such that it cannot be closed (and the alarm cleared) unless the test device plugs (described below) are removed. Closing the test panel cover will mechanically return the test switches to the "normal" position.

Administrative procedures will require that the bistable in the channel under test be placed in the tripped mode prior to test. This places a proving lamp across the bistable output so that the bistable trip setting can be checked during channel calibration. The bistable trip switches must be manually reset after completion of a test. Closing the test panel cover will not restore these switches to the untripped mode. To prevent safety injection trip, procedures limit bistable testing to one circuit at a time.

Actual channel calibration will consist of producing a test signal using the transmitter power supply external calibration device which plugs into the signal injection jack. In this application, where specified, the channel power supply will serve as a power source forthe calibration device to permit verifying the output load capacity of the power supply. Test points are located in the analog channel and provide an independent means of measuring and/or monitoring the calibration signal level.

Logic Channel Testing

Figures Q 7.23-2 and Q 7.23-3 show the basic logic test scheme. Test switches will be located in the associated relay racks rather than in a single test panel. The following procedures will be used for testing the logic matrices:

1. Following administrative procedure, test Channel A or B, one at a time.

2. Select a matrix and turn the test switches to "test" then depress the push button. Test lights will indicate upon actuation of the matrix being tested. Release pushbutton and return test switch to "operate." "On test" lights glow anytime any switch is in a test position. Test lights can be tested by depressing the lens.

3. Verify master actuating relay coil integrity by connecting ohmmeter across coil terminals.

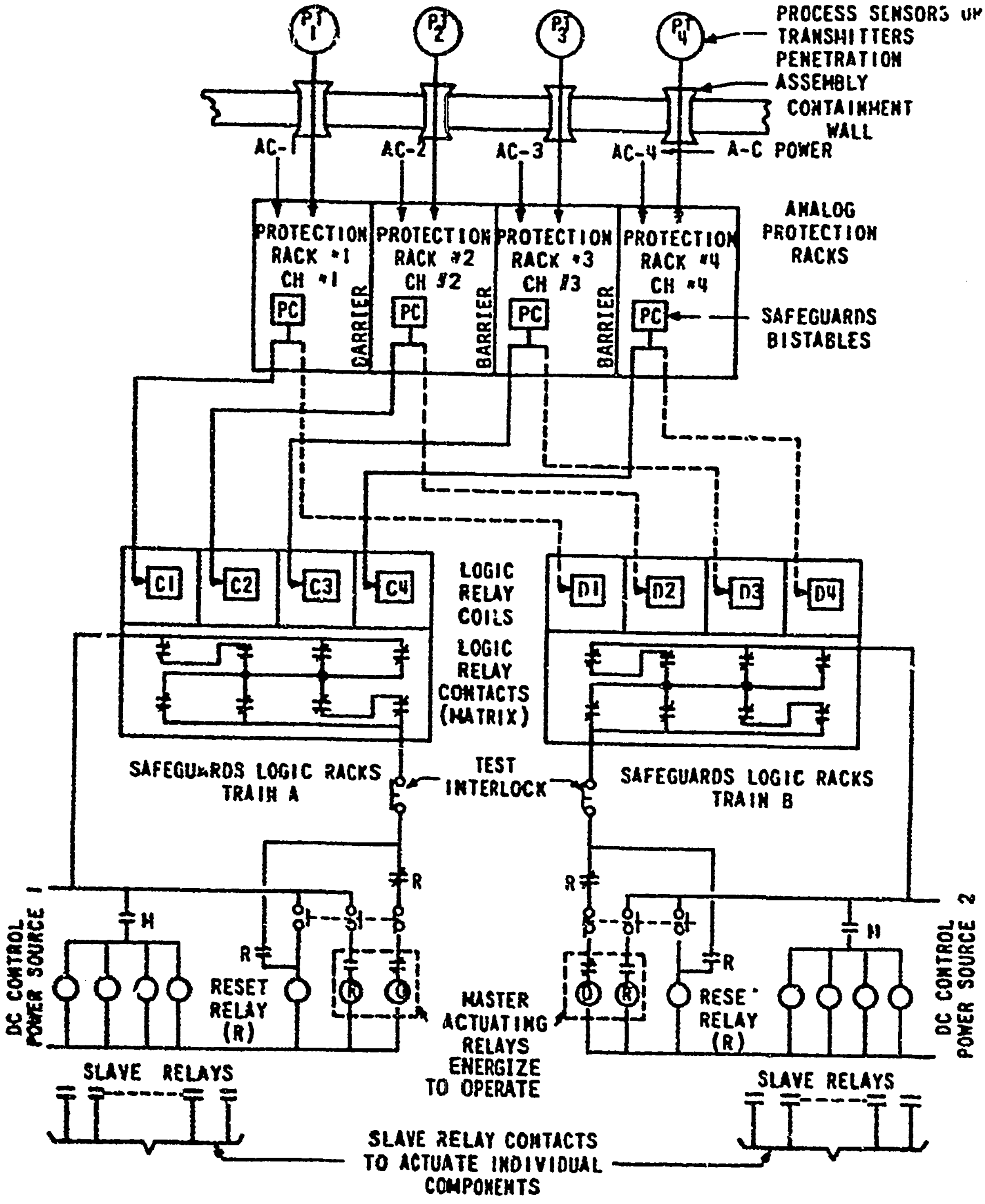

FIGURE Q7.23-1

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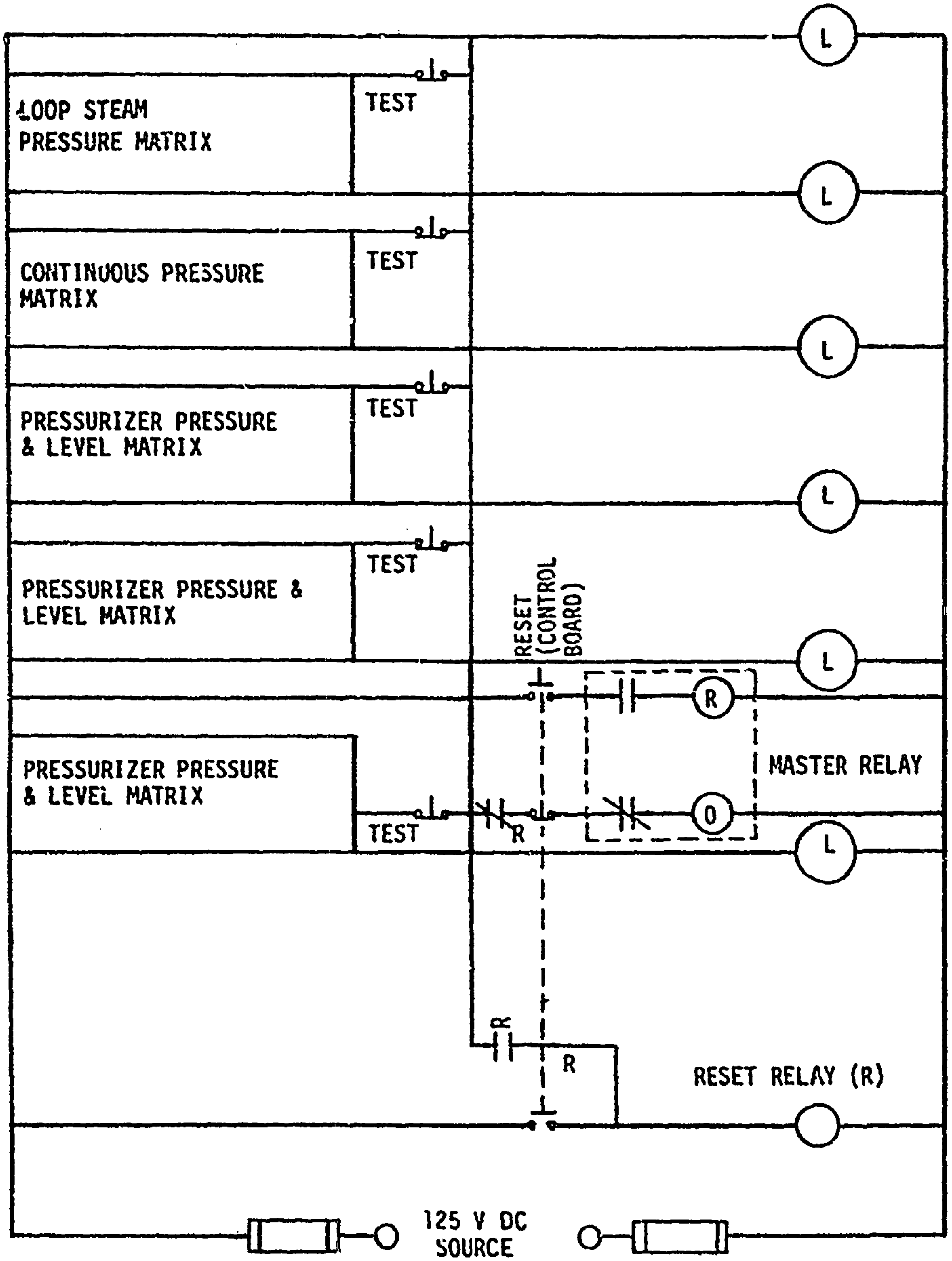
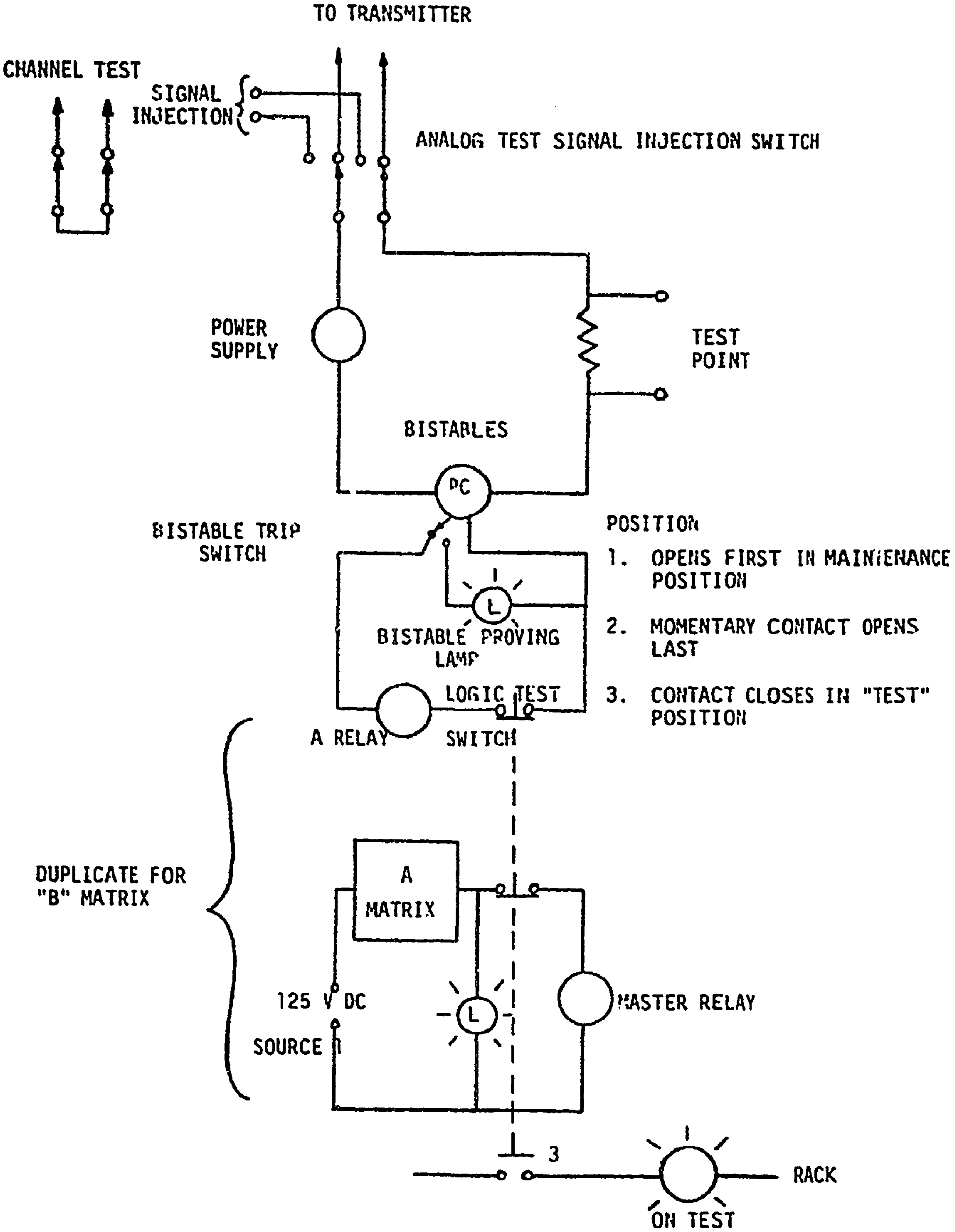
(Safety injection shown as typical)



Note: Above circuit duplicated for actuation channel B

SAFEGUARDS INITIATION

FIGURE Q7.23-2



ANALOG AND LOGIC CHANNEL TESTING
FIGURE Q7.23-3

QUESTION 7.24

Document that scram breaker "position" lights will be added in the control room to alert the operator as to the position of the scram breakers.

ANSWER

Reactor trip breaker "position" lights for both the main and bypass breakers (4) are in the control room on the reactor protection test panels.

QUESTION 7.25

The response to Question 7.1 indicates that the reactor trip on turbine trip, and the turbine runback circuits meet IEEE 279. It was subsequently determined that these circuits do not meet IEEE 279 and that they need not meet it since they are anticipatory signals and are not required for reactor safety. Change the response to Question 7.1, accordingly.

ANSWER

Refer to the response to Question 7.1, pages Q 7.1-1, Q 7.1-2, and Q 7.1-3 as revised by Supplement 12.

QUESTION 7.26

Correct Page 8.2-14 of the FSAR to delete the mention of automatic switching of the bus tie breakers between the vital buses.

ANSWER

Refer to page 8.2-14 of Section 8 of the FSAR as revised by Supplement 12.

QUESTION 7.27

Complete your documentation on the seismic testing of protection system equipment. WCAP7397-L, "Seismic Testing of Electrical and Control Equipment" does not include all electrical equipment necessary to the operation of the protection system.

ANSWER

A safeguards signal may be initiated by an instrument or transmitter which has the ability to withstand seismic forces as demonstrated in WCAP-7397-L (Westinghouse proprietary), Sec. 4.8. This signal is carried in conduit and cable trays whose supports have been studied for resistance to seismic forces. The signal passes to the process control racks proven as described in WCAP-7397-L, Sec. 4.2. The signal is sent next to the safeguards actuation racks proven as described in WCAP-7397-L, Sec. 4.3. The actuation signal proceeds through a switch on the control board to the appropriate switchgear. The control boards were specified to "be designed such that the maximum stresses including simultaneous seismic accelerations of 0.52g in the horizontal and vertical directions shall not dislodge or cause relative movement between components such as to impair the functional integrity of circuits or equipment." This acceleration exceeds that calculated as input to the boards from the floor of the control room. In shipment, boards of typical manufacturer and construction have recorded shocks of 8-10g, and when wired, the switches have operated without repair.

The switchgear equipment has been specified to withstand accelerations in excess of 0.15g horizontally and 0.10g vertically. This capability was a matter of procurement specification of Westinghouse and its design agents and design action of the vendors. The safeguards circuits employ Westinghouse Series W motor control centers, DB and DH circuit breakers and associated metal-enclosed or metal-clad switchgear. Review of these switchgear for proof of adequacy of the seismic resistant design determined that the Series W motor control centers and DB breakers, mounted in the metal enclosures, have been shock tested and proven to remain fully operable for shocks of at least 3g in any direction. Proof of resistance of the DH metal-clad switchgear to a seismic response spectrum established for Point Beach has been demonstrated by vibration testing of typical, equivalent metal-clad switchgear, incorporating the DHP circuit breaker. The DH circuit breakers installed in Point Beach are an earlier design than the DHP. However, the general configuration, weight distribution, and vibration resistant design approach of the DH are essentially identical to the DHP. When subjected to a spectrum equivalent to or greater than Figure B-2, there was no loss of function of the DHP metal-clad switchgear.

The power supply leaving the switchgear operates the safeguards equipment completing the actuation train. The seismic design of this equipment is described in the answer to Question 7.4-1, 1.9-6 and 1.9-7. The D.C. power supply may be considered as a branch to this main train of actuation. The source of D.C. power is the station batteries. The batteries and battery racks present a simple structural problem which was analyzed and found adequate for the forces imparted by the floor upon which they are located. The conduit and cable trays carrying the D.C. power to the main station train received the same study for seismic support as described above.

QUESTION 7.28

We understnad that you intend to make the manual actuation of the containment spray independent of the automatic portion of the circuit. Please describe your intent in this regard.

ANSWER

Manual (2/2 spray push buttons) initiation will position valves and start the spray pumps anytime before or after automatic safety injections initiated pump sequencing is in progress. Safety injection reset will block automatic spray initiation.

Question 8.1

Please perform an analysis to demonstrate the ability of IPP #2 to meet the single failure criterion recognizing that a single 138 kv line connects the station to the Buchanan substation, a distance of approximately 3/4 mile. This analysis should include:

a) The ability of the grid to provide offsite power while losing the IPP #2 generating capacity or the next largest unit on the grid.

Answer

The 138 Kv supply to Indian Point No. 2 is obtained from the Buchanan 138 Kv station. This station has two connections to Millwood 138 Kv station and two to the Orange and Rockland Company's Lovett Station. There is also a connection to Indian Point Unit No. 1. The Indian Point No. 2 345 Kv connection to the system goes to Buchanan 345 Kv substation (which is not connected directly to the 138 Kv station) and then directly to Millwood 345 Kv station. System stability studies have been made that show that the system is stable for the loss of any generating unit including Indian Point No. 2.

There is no direct connection between Buchanan 345 and 138 Kv station and the 138 Kv supply to Indian Point No. 2 will not be directly affected by the loss of Indian Point No. 2 345 Kv output.

The 138 Kv connection, however, supplies one station auxiliary transformer at Indian Point No. 2 and the loss of this transformer would interrupt the 138 Kv supply to the station. For this reason, an alternate supply at 6.9 Kv is provided. This supply is manually connected in the event that 138 Kv supply is lost.

Question 8.1

Please perform an analysis to demonstrate the ability of IPP #2 to meet the single failure criterion recognizing that a single 138 kv line connects the station to the Buchanan substation, a distance of approximately 3/4 mile. This analysis should include:

b) The degree of backup being supplied by the 13 kv line to include:

(1) Capacity of 13 kv line with regard to engineered safety feature and safe shutdown loads; to include line and 13/6.9 kv transformer.

(2) Starting characteristics of the gas turbine; what signals are provided to start the generator? What are onsite fuel storage details and fuel resupply for the generator?

(3) What are siting details and protection provided for the 13 kv line and the gas turbine?

(4) Under what conditions is power from this source required for IPP #1?

Answer

(b) (1) The 13 Kv line is rate 19.8 MVA at 13 Kv. The 13/6.9 Kv transformer is rated 20 MVA. The maximum engineered safety feature and safe shutdown loads are 9.2 MVA.

(2) The gas turbine will be started manually from Indian Point Nos. 1 and 2 control room when required. There are no automatic starting signals. The onsite fuel supply for the gas turbine consists of two (2) - 30,000 gallon fuel oil tanks sufficient for operation at full load for 30 hours or at 1/2 load for 48 hours. Additional fuel can be delivered to the station within 12 hours. The starting time for the gas turbine is 10 minutes to full load.

(3) The 13 Kv line will be underground from Buchanan Substation to the gas turbine switchgear. The gas turbine is located in a rock cut on the sourth side of Indian Point No 1.

(4) Power from the gas turbine can supply auxiliary load for Indian Point No. 1 if no other supply is available. Normally Indian Point No. 1 auxiliaries are supplied by 3 separate underground feeders from Buchanan Substation and the gas turbine is not used for this purpose. The gas turbine is not required to provide safe shutdown power to Indian Point No. 1.

Question 8.2

The three diesel generators are located in a common enclosure which does not appear to provide tornado protection, also there is no protection between the machines in case of missiles. Please explain the rationale for placing the diesel generators in this common enclosure. Also discuss the rationale for providing fuel oil for 54 hours of operation for two diesels.

Answer

The diesel building is by virtue of its location protected from tornados and major missiles generated by them. The building is situated between large buildings as shown in Figure 8.2-1 attached.

It is considered the diesel installation is redundant to other lines of power supply. As described in Section 8.1, Page 8.1-2 of the FSAR, there are alternate power supplies. Reliance in the case of a tornado is placed on power supply redundancy, not solely on the diesel installation.

Missile protection between machines is not considered necessary on the basis of the engine manufacturers case histories of engine failures. Many of the Alco model 251 engines are in service in railroad locomotives. Field case histories disclose a complete absence of damage to the engine environs as a result of engine component failure. Engine failures, usually the result of extreme operating conditions, can be classed as follows:

a. Stuck Valve

A valve sticks open and is struck by the piston. The damaged valve, and possibly part of the piston, enters the exhaust manifold, damages the turbo charger, and passes harmlessly up the stack. There is no record of a damaged piston generating a missile external to the engine.

b. <u>Piston Seizure</u>

A piston seizes and causes bending and eventual fracture of the connecting rod. All damaged parts remain inside the engine block.

c. <u>Turbo Charger Failure</u>

A turbo charger turbine wheel fouls the casing as a result of over-speed or overheat. The robust double walled casing contains all parts.

d. <u>Engine Overspeed</u>

The engine's normal operating speed is 900 rpm. Protection against overspeed is provided by two trips, one set at 990 rpm the other at 1035 rpm. These trips shut off the fuel at each individual fuel injection pump. No cast iron is used in the engine block and base so that even if the overspeed trip failed, the engine structure, which is not brittle by nature, would contain any fracture parts. Isolation cases of crank shaft fractures have resulted in no flying missiles.

e. <u>Cylinder Head Failure</u>

Cylinder heads are secured to the block by high tensile studs. No cap gaskets are used between the head and cylinder liners. This pre-stressed design, with no possibility of slackness developing, has resulted in an assembly which has had no incidents of heads flying off, even when failed pistons have pounded the heads. Cases also are on record of improperly timed engines resulting in excessively high firing pressures, over 2,000 psi (normal pressure 1600-1700 psi) and the heads have always remained intact.

Operating experience with the Alco engine indicates that internal missiles do not escape from the engine. Alco does not have any evidence of blades coming through the turbo casing. Valves from the engine have broken and been exhausted through the turbo and caused damage to turbo, but are contained within the casing. There is no evidence of connecting rods escaping from the engine.

In order to generate any flying parts, the generator would have to be in an overspeed condition beyond what is normally possible with a diesel engine. The construction of the stator windings and stator barrel frame would have to be penetrated by a rotor part in order to escape. The rugged construction of each complements their ability to contain flying objects.

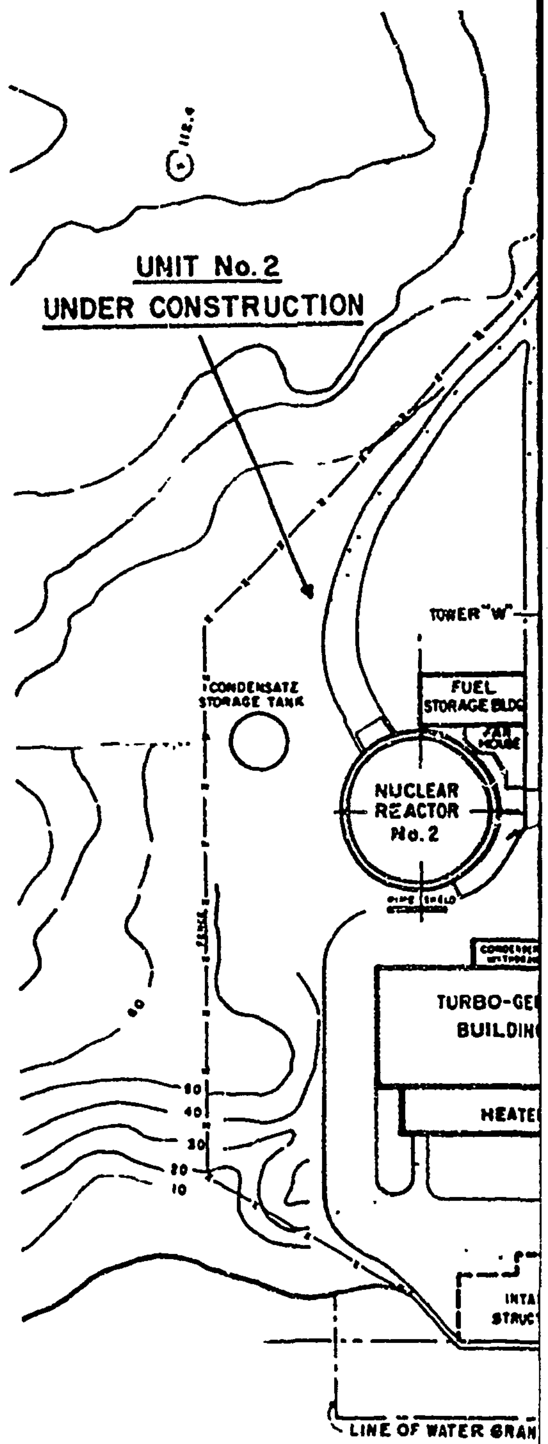
Since the engine has overspeed trips and would not operate much beyond this speed because the valves would hang up, it is felt that the generator would never reach any critical speeds.

Each diesel generator is provided with a head tank and bulk storage tank. A minimum fuel storage of 19,000 gallons will be maintained in the three bulk storage tanks. The 54 hours is based on two tanks being at the minimum level 6,340 gallons each, with No. 2 diesel fuel oil at the lowest density 6.87 lbs/gallon. The engine consumption rate is 128 gallons/hour based on the engine tests with the specified fuel. It is assumed the third tank is unavailable. Should all three tanks be available at the minimum storage level, i.e., 19,000 gallons of fuel at 6.87 lbs/gallon, then 80 hours operation is possible for two diesels with recirculation load.

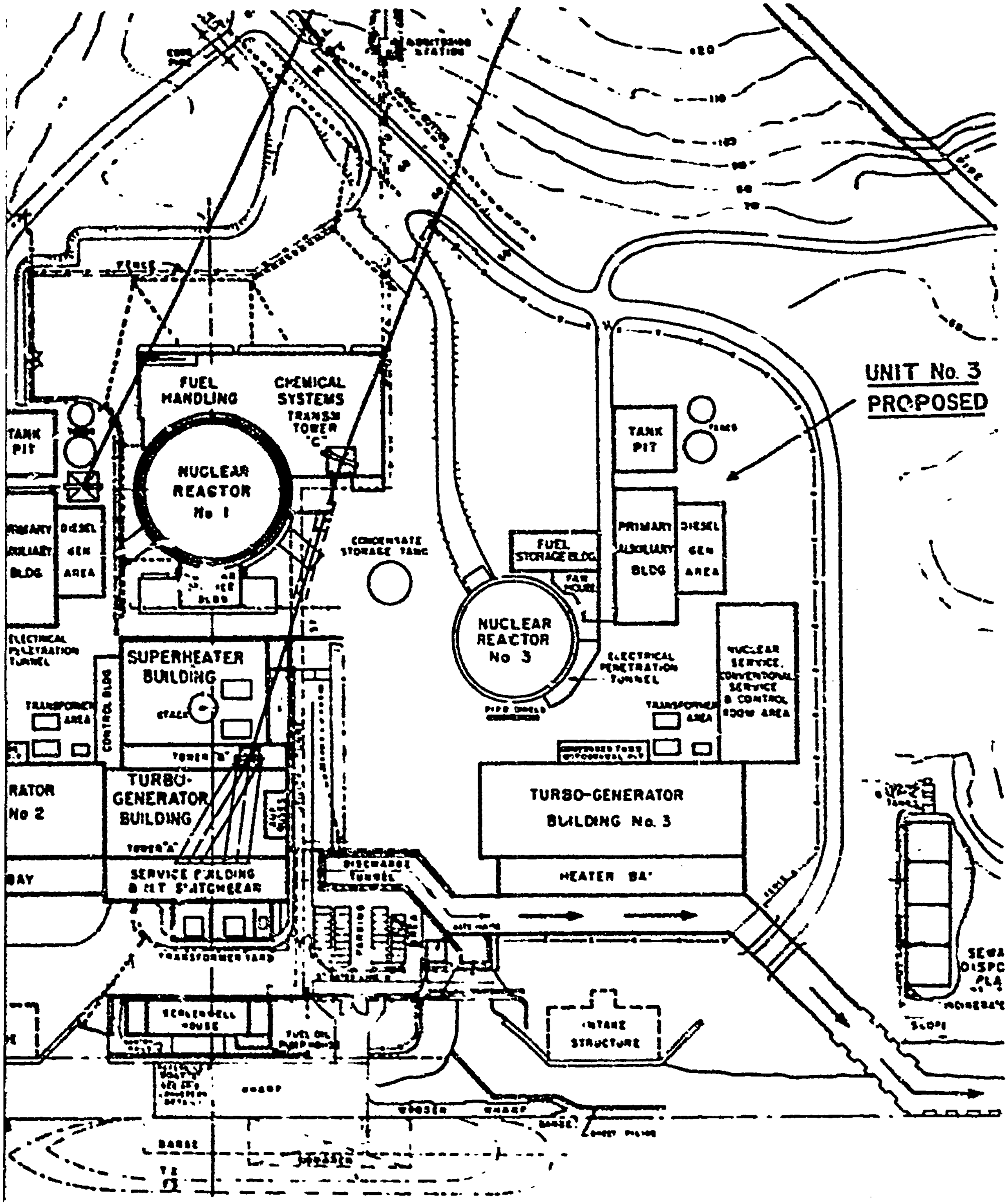
Additional fuel oil suitable for the diesel engines is stored on site for the gas turbines. Two tanks each of 30,000 gallons nominal capacity are located in a protected area. Should one of these tanks be at a low level of 20,000 gallons and made available to the diesel engines, a further 85 hours running time is assured. This storage time is based on the lightest oil. In the case of a heavier oil in the tanks the time would be increased proportionally to the ratio of 6.87 lbs/gallon and the actual fuel density. An upper limit of 7.39 lbs/gallon is common for No. 2 diesel oil.

Additional supplies of diesel oil are available locally. Under normal conditions, 25,000 gallons can be delivered on a one or two day notice. Additional supplies are also maintained in the New Rochelle - Mount Vernon area (about 40 miles from the plant) and are available for use during emergencies, subject to extreme cold weather conditions (increased domestic heating usage) and available transportation.

The basis for fuel oil storage is an absolute minimum of 12,680 gallons (54 hours) in the engine bulk stop tank plus a minimum of 20,000 gallons (85 hours) in one gas turbine tank which is sufficient for 139 hours of operation (5-1/2 days). Fuel supply facilities enable 25,000 gallons (106 hours - 4-1/4 days operation) to be delivered within the time available from the site stored capacity. Continuous supplies can be established within the delivery capacity of 4-1/2 days operation.



UNIT No. 2
UNDER CONSTRUCTION
TOWER "W"
FUEL STORAGE BLDG
CONDENSATE STORAGE TANK
NUCLEAR REACTOR No. 2
TURBO-GE
BUILDIN
HEATE
INTA
STRUC
LINE OF WATER GRAN
50
40
30
20
10



Supplement 3
11/69

SITE PLOT PLAN
FIG. 8.2-1

QUESTION 8.3

From the information contained on Page 8.2-14 of the FSAR, it has been determined that the IPP #2 onsite power system is designed identical to that originally proposed for IPP #3 and commented on in the ACRS letter. If the proposed design is to be retained, please perform an analysis to show that the independence of the onsite power is not compromised by the use of automatic breakers between essential buses.

ANSWER

The proposed design will not be retained and no automatic closure of tie breaker will be used. The system is being redesigned to incorporate the same modification being made to the Indian Point Station - Unit 3 system.

QUESTION 8.4

Please provide an analysis to show that a single failure in the d-c power to the station switchyard will not prevent the grid from supplying offsite power.

ANSWER

The criteria of dc power supply redundancy is satisfied by the presence of two independent battery supplies and two dc load supply boards.

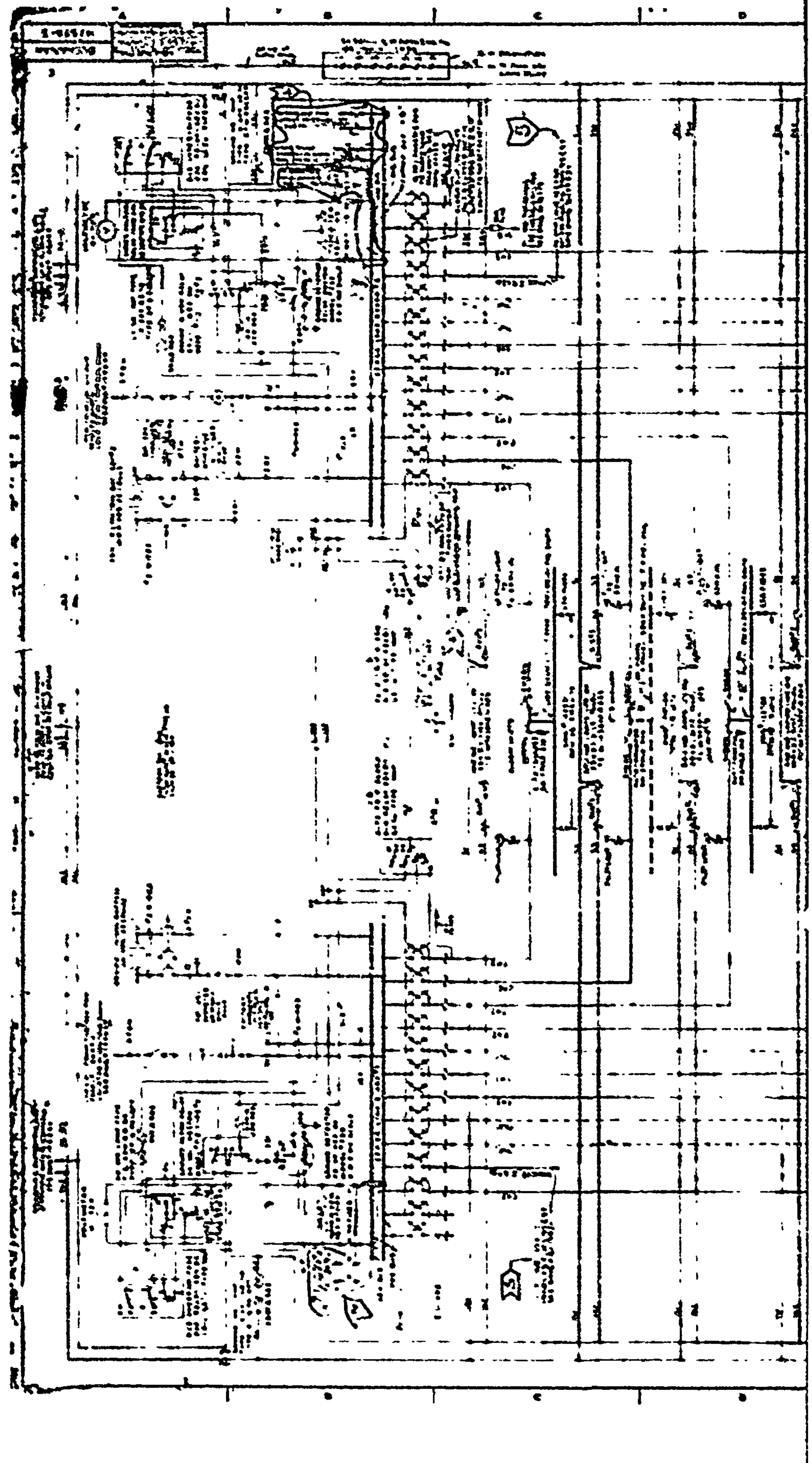
In accordance with Consolidated Edison Company Drawing No. 142595, the Buchanan Substation is equipped with two indep dent 125 V dc, 100 ampere-hour batteries. Each battery is fed from a separat charger and each charger is fed from a separate ac power panel. Each ry bus is equipped with a sensitive-type undervoltage relay (GE C .9), which provides alarm/indication of an undervoltage cond: . Ground alarms are also provided on each board.

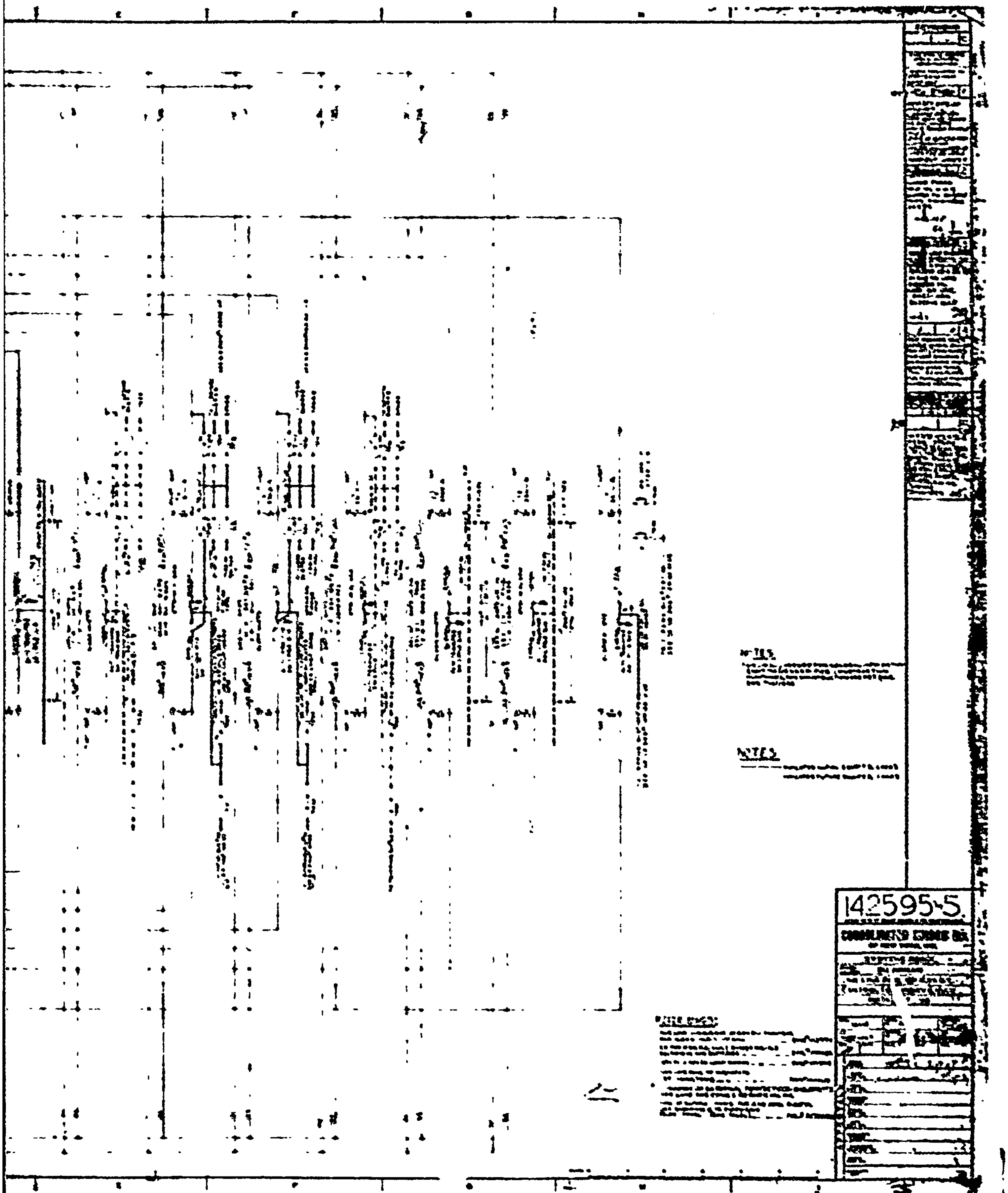
Loss of any individual dc feed to any of the distribution panels, all of which are fed from both load boards through individual circuit breakers, will result in the automatic transfer from one dc supply to the alternate supply for that distribution panel. An alarm will indicate the occurrance of a transfer. The automatic transfer of supply will provide continuity of dc supply to the switchgear in the event of a loss of one dc feed.

Total loss of feed to the switchgear and associated equipment will not cause a loss of offsite power through inadvertant tripping of Indian Point light and power supply circuit breakers, as the presence of dc is required to trip a breaker. Loss of dc feed to protective relaying will cause an alarm condition rather than initiation of protective action. If necessary, the light and power circuit breakers in the Buchanan Substation may be tripped manually at the breaker mechanisms.

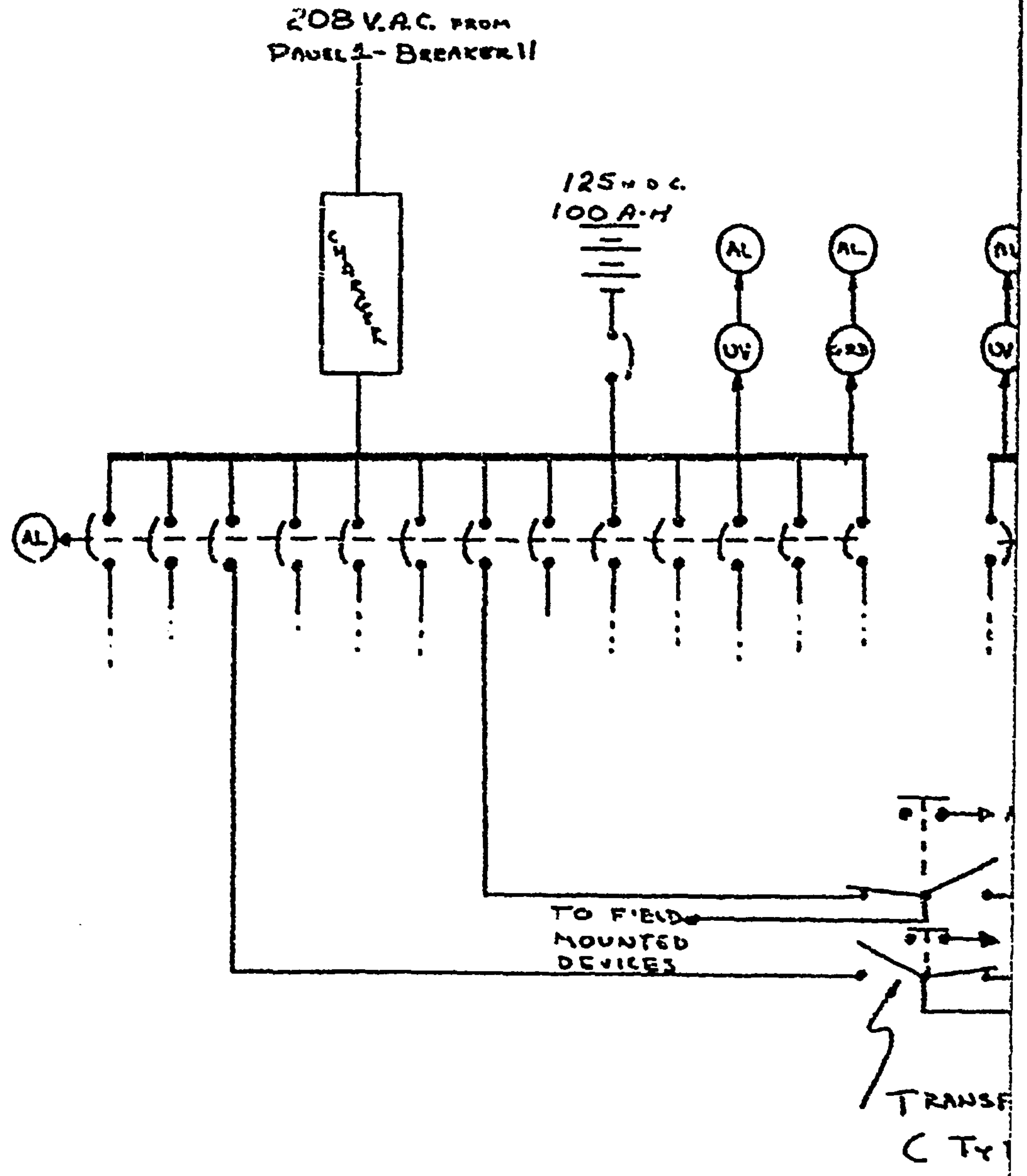
Alarms are indicated locally at the Buchanan Substation and a category alarm is also indicated at the Energy Control Center.

Reference: Consolidated Edison Company Drawing No. 142595
Attached sketch No. C-SK-IPB-1448

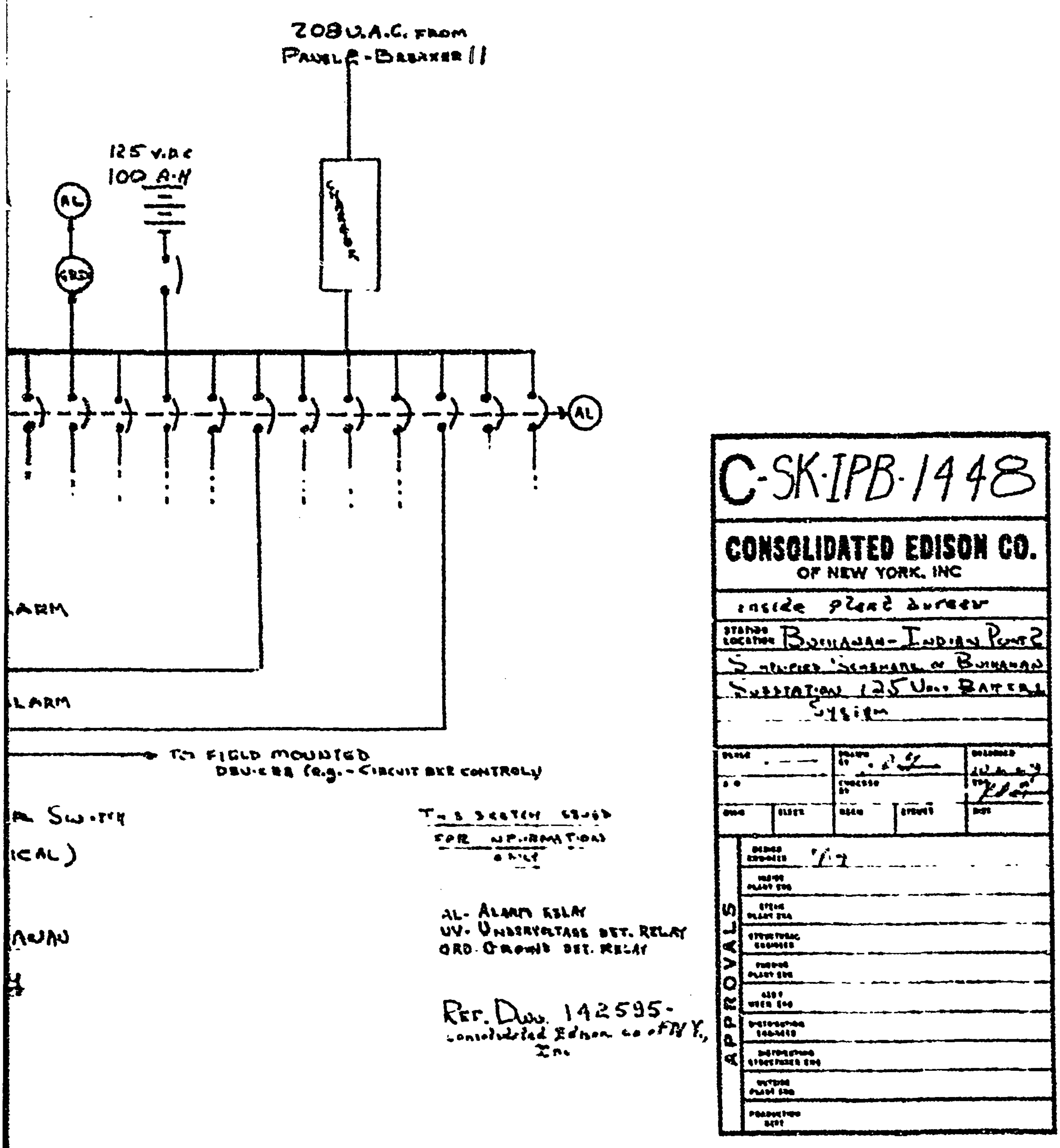




One Line Diagram of 125 V.D.C. Control
for 138 KV & 13 KV Inst...lations
Drawing No. 162595-5



SIMPLIFIED SCHEMATIC OF BU
SUBSTATION 125-VOLT BATTE
SYSTEM



Simplified Schematic of Buchanan
Substation 125 Volt Battery System
Drawing No. C-SK-IPB-1448

QUESTION 8.5

Discuss the analysis performed to determine that additional restraints are required for the instrument air line which passes near the 480 volt essential switchgear. Further, describe the barrier which will be installed to shield the switchgear and cables from potential missiles that could originate from the air compressor.

ANSWER

Refer to the response to Question 7.6(d), page Q 7.6(d)-4, as revised by Supplement 12.

QUESTION 8.6

Describe the additional work being performed in the electrical penetration area to provide added assurance of cable protection.

ANSWER

Refer to the response to Question 7.7, page Q 7.7-1, as revised by Supplement 12.

QUESTION 8.7

Describe the concrete wall which will be installed to shield the diesel generator control panel from potential missiles that could originate from the diesel generators.

ANSWER

Refer to page 8.2-13 of Section 8 of the FSAR as revised by Supplement 9.

QUESTION 8.8

Describe the equipment which would sense undervoltage of the essential buses, and signal that the diesel generators be started.

ANSWER

Two out of 3 undervoltage relays on busses 5A and 6A or one undervoltage relay on bus 2A or 3A will start all three diesels.

QUESTION 9.1

Describe the temperature detectors, alarm and control systems, and electrical power requirements and sources for heat tracing used to keep the boric acid piping and tanks at temperatures well above the precipitation point. Evaluate single failures in the controller/alarm units as well as loss of one source of power to instrumentation, tank heaters and electrical heat tracing.

ANSWER

Heat Tracing

All boric acid piping is provided with double circuit (one circuit redundant) electrical tracing in conjunction with insulation to maintain the concentrated solution within a temperature range of 163°F to 167°F, when subjected continuously to an ambient temperature of 40°F in still air.

The normal source of power for the tracing on the boric acid piping totaling 60 KW, is 480 volt Motor Control Center 26A. This motor control center is automatically switched to Diesel Generator 21 on loss of power. In addition, all circuits can be manually switched to Motor Control Center 26B, which is supplied from Diesel Generator 23 under the same condition. The electric tank heaters, totaling 39 KW, are supplied from Motor Control Center 27, which can be manually switched to Diesel Generator 23 on power loss.

Each individual pipe tracing circuit has a local control cabinet containing operating and alarm devices as follows:

1. Operating Thermostat - A line thermostat with remote bulb temperature sensor. The bulb is strapped on the pipe underneath the insulation. This thermostat energizes the tracing, when the pipe temperature falls below 161°F, and opens the circuit on a temperature rise to 165°F (the setpoints are lower than the desired temperature range because of the difference in temperature between the pipe exterior and the fl ' inside the pipe).

2. Alarm Thermostat -- A two-stage thermostat with remote bulb sensing device strapped on the pipe in the same area as the operating thermostat bulb. It is used to monitor the pipe temperature. A high (175°F) or low (155°F) temperature condition on any tracing circuit is indicated on a local annuciator panel in the Primary Auxiliary Building. In addition, this condition is alarmed on the main annuciator in the Control Room, as is loss of power to the local annuciator.

3. Test Circuit - A manually operated circuit consisting of test switch, current relay, and indicating light. It is used to monitor and insure the integrity of the de-energized redundant circuit, and to check the status of the operating circuit. Power for this circuit is supplied from the same source as the heat tracing circuit.

Failure of the operating circuit will result in decrease in pipe temperature, and will alarm in the Control Room. Test and connection of the redundant circuit can be accomplished within 15 minutes. Likewise, failure of any operating device in the local control cabinet will result in alarm. Spares are available so that any defective device can be replaced within one hour.

Boric Acid Tank Heaters

Each boric acid tank has two 100% capacity electric heaters which are connected in parallel and controlled from a single controller and a single temperature sensing controller and a single temperature sensing device (TIC-107 in Tank 1 and TIC-103 in Tank 2) and are powered by a single source. The heaters maintain the boric acid solution at 165°F (temperature range of 160°F to 170°F).

TIC-107 (and TIC-103) are "filled system" temperature devices. The instrument mechanism is connected to the thermal bulb in the tank by a capillary. Thermal expansion of the full fluid is converted into a motion which:

a) controls the local indicating pointer directly

b) controls an electronic transmitter

c) controls the contacts used for controlling the tank heaters.

The local indicating pointer operates independently of any power source.

The electronic transmitter provides a signal to a control board indicator and to an alarm unit which provides audible and visual low alarm in the control room.

The contacts which control the heaters operate through an internal relay. Loss of instrument power will cause a low alarm and turn the heaters on and will cause the remote indicator to give minimum temperature readings. Since the meter is calibrated from 50 to 200°F, the erroneous reading is obvious to the operator.

QUESTION 9.2

Describe the instrumentation and/or methods used :. monitor the concentration and level in the concentrated boric acid storage tanks.

ANSWER

The tanks of concern are:

1. The two boric acid storage tanks of the Chemical and Volume Control System.

2. The boron injection tank of the Safety Injection System.

Boron Concentration

Local samples may be taken from each of these tanks for boron concentration analysis. The Technical Specifications for the plant require that:

1. Boric acid tanks be sampled at weekly intervals.

2. Boron injection tank be sampled at monthly intervals.

Makeup batches of boric acid to either of these tanks are checked for boron concentration before transfer.

Tank Levels

Boric Acid Tanks:

For cold shutdown purposes, there must be a minimum of 4400 gallons 11-1/2% to 13% solution of boric acid available in the boric acid tanks. The operator is alerted to an approach to the cold shutdown level in either tank by a low level alarm in each tank corresponding to 4550 gallons. It is, however, optional whether the operator chooses to operate normally above the low level alarm in both tanks. Each tank is instrumented for level by use of a bubbler system and differential pressure cells. Two purge rotameter assemblies (per tank) maintain an air or nitrogen purge through an upper and lower bubbler tube. The differential pressure cell measures and transmits the difference in pressure due to submergence of the lower bubbler tube. Indication is on the Chemical and Volume Control System supervisory panel in the control room.

The low level condition is audibly annunciated in the control room with the annunciator drop located on the same panel.

The level transmitter factory accuracy is ±0.5% of span. The installed accuracy depends on the final field adjustments, but the repeatability does not. Since the span is approximately 105 inches of pure water (102 linear inches of boric acid solution), the transmitter, when properly calibrated, will repeat any particular level measurement within 0.1 inch or better.

Boron Injection Tank:

The level in the boron injection tank is instrumented for level by duplicate sealed systems, bellows type, with differential pressure cells. Indication is in the control room. The setting of the alarms are as follows:

Low Alarm - alerts the operator to an approach to the Technical Specification minimum volume.

Low-Low Alarm - trips the tank heaters.

Very-Low Alarm - isolates the tank by closing the discharge valves.

Accuracy of level instrumentation for the boron injection tank is comparable to that for the boric acid tank level instrumentation.

QUESTION 9.3

Describe the instrumentation used to verify injection of concentrated boric acid flow into the primary system.

ANSWER

Concentrate boric acid is injected into the reactor coolant system by means of the charging pumps which take suction from the boric acid tanks via the boric acid transfer pumps. Each operation is considered in turn:

I. Concentrate boric acid can be delivered to the suction of the charging pumps using the following paths:(Refer to Fig. 9.2-1 as revised by Suppl. 7)

   a. Through the blender and valve FCV-110B; for this operation the operator may read flow meter FT-110 and the boric acid tank levels LT-106 for tank No. 1 and LT-102 for tank No. 2.

   b. Through manual valve path 293 - for this operation the operator may read flow meter FT-110 and the boric acid tank levels LT-106 for tank No. 1 and LT-.02 for tank No. 2.

   c. In the event that neither flow paths (a) and (b) are available, the operator would use the emergency boration path through valve MOV-333 - for this emergency operation the operator may read the boric acid tank levels LT-106 for tank No. 1 and LT-102 for tank No. 2.

II. The charging pumps can deliver boric acid into the Reactor Coolant System via the following paths:

   a. Normal charging line via flow meter FT-128.

   b. Seal water supply line to the reactor coolant pumps while by-passing the seal injection filters. If this path had to be used local flow indicators FI-115, FI-116, FI-143 and FI-144 would indicate flow.

FT-110 is a magnetic flowmeter operating on the "Hall Effect" principle. Instrument power is necessary for it's operation. Its signal is both indicator and recorded in the control room.

LT-102 and LT-106 are electronic transmitters sensing tank level by means of the ΔP generator by bubbling air through a tank extending to the bottom of the tank. Instrument air and instrument power are required for operation of these instruments. However, loss of instrument air does not immediately "kill" the signal since the air trapped in the bubblor tube will be sufficient to provide an accurate signal for several hours to several days depending of the amount of cycling of the level in the tank. Each of these channels is indicated in the control room.

FT-128 is an electronic transmitter sensing flow by means of the differential pressure generated across an orifice. Instrument power is required for it's operation. Two indicators are provided, one in the control room and one near the charging pumps.

FI-115, FI-116, FI-143, and FI-144 are local differential pressure gages sensing flow by means of the differential pressure generated across an orifice. These indicators are located outside the containment and are self actuated. (i.e. no power required).

QUESTION 9.4

Describe the provisions for analyzing the primary coolant for boron concentration. What is the normal frequency of sampling and analysis for boron content? Is this capability available at the plant site at all times?

ANSWER

Facilities are provided to enable primary coolant samples to be taken from the following points:

Pressurizer liquid space
Loop 1 hot leg, reactor coolant system
Loop 3 hot leg, reactor coolant system
Upstream of demineralizers (Chemical and Volume Control System)
Downstream of demineralizers (Chemical and Volume Control System)

The Technical Specifications for the plant require a boron content analysis at specified intervals. Samples would normally be taken from either the Loop 1 hot leg or the Loop 3 hot leg for routine analysis; the sample will be analyzed by the standard acid base titration method. This analysis can be made at all times provided the necessary trained personnel are on site. It is however, important to note that the main indicator to the operator during power operation as to the requirement for boration or dilution is control rod position. (see Section 14.1.5-1 in FSAR)

Following a boration or dilution, the operator normally will sample both the reactor coolant system hot legs and the pressurizer liquid space for record purposes and to check that homogenization of thepressurizer liquid with the recirculating reactor coolant has been completed.

During startup and refueling, the main indicator to the operator of abnormal conditions are the BF_3 detectors. Abnormal dilution conditions are discussed in Section 14.1.5-1 of the FSAR. As for power operation, it is considered that frequent boron analysis of the primary coolant is not essential for safe operation.

For a cold shut-down, the operator borates the system prior to the start of cooldown. Boration is indicated by the flow indicators in the boric acid transfer pump discharge line. The prime indicator that sufficient boron has been added to the system is inventory from the boric acid storage tanks.

QUESTION 9.5

The FSAR on page 14.2.1-3 states that "Crane facilities do not permit the handling of heavy objects, such as a spent fuel shipping container, above the fuel racks." Please describe how this objective is implemented in the facility layout.

ANSWER

Figure Q 9.5-1 shows the plan arrangement of the Fuel Storage Building.

During normal operation when the spent fuel cask is being placed in or removed from its position in the spent fuel pit, mechanical stops will be incorporated on the bridge rails which will make it impossible for the bridge of the crane to travel further north than a point directly over the spot reserved for the cask in the pit.

During normal reactor operation, the two southernmost spent fuel racks, each holding 25 fuel assemblies, and the southern half of the rack holding 32 fuel assemblies in the southeast corner of the pit will be covered with removable stainless steel plates, to prevent the normal storage of fuel assemblies in those 66 positions closest to the south wall of the spent fuel pit. These covered storage locations would be utilized only in the event that the total core fuel assemblies are removed and 1/3 of a core from a previous refueling is present.

Thus it will be possible to handle the spent fuel cask with the 40 ton hook and to move new fuel to the new fuel elevator with the 5 ton hook, but under normal conditions it will be impossible to carry any object over the spent fuel storage area with either the 40 or 5 ton hook of the fuel storage building crane.

Indian Point
Final Facility Description and Safety Analysis Report
Dated: October 15, 1968

Figure Q 9.5-1, Titled "Plan Arrangement of Fuel Storage Building"

The above was **redacted** by NRC staff as **sensitive information** to allow release of this document to the public.

QUESTION 9.6

Discuss the provisions that will be made to prevent dropping the spent fuel element cask into the spent fuel storage pool. If the spent fuel element cask must be moved over the spent fuel storage pool, analyze the consequences of dropping the cask into the pool. Consider the possibility of (1) loss of pool water and ability to continue cooling the spent fuel, and (2) damage to other equipment by flooding if the integrity of the pool liner is lost.

ANSWER

It is extremely improbable that the cask would be inadvertently or otherwise dropped during the process of transfer. This is due to the following provisions:

- Conservative design margins used for the cask related handling equipment (crane, rigging, hooks, etc.)

- Periodic non-destructive equipment tests and inspection procedures

- Use of qualified crane operator and riggers and

- Use of approved operating and administrative procedures

These provisions will be rigorously met so that the inadvertent drop of the cask into the pool is highly improbable. However, should such a highly unlikely accident occur, the basic assumptions for analysis are as follows:

The drop would be from the cask's highest position which is 5 feet above the water surface and 43 feet above the bottom of the pool.

- The cask is fully loaded and weighs 40 tons.

The results of the analysis indicate that the cask would hit the bottom of the pit with a velocity of approximately 40 ft/sec., assuming a conservative drag coefficient of 0.5. In comparison, the cask would have reached a velocity of 52 ft/sec. if dropped through 43 feet in air.

Using the Ballistic Research Laboratories formula for the penetration of missiles in steel, the depth of penetration of the cask into the 1-inch wear plate covering the 1/4-inch pit liner plate would be 0.32-inch, assuming the cask struck the wear plate while in a perfectly vertical position. In the event that the cask falls through the water at an angle, its terminal velocity would be somewhat less because of the increased drag. However, the cask would strike the wear plate with an initial line contact and would penetrate the wear plate and the pit liner plate, causing some cracking of the concrete below. This reinforced concrete is a minimum of 3'-7" thick and rests on solid rock.

Water would initially flow through the punctured liner plate and fill the cracks in the concrete. Since the pit is founded on solid rock and since the bottom of the pit is approximately 24 feet below the surrounding grade, very little water can be lost from the pit. The capacity of the make-up demineralized water supply to the pit is 150 gpm. In addition, the spent fuel pit cooling system piping has a 4" blind flange connection for temporary cooling and/or make-up water.

Since the bottom of the spent fuel pit is 24 feet below grade and since no equipment areas are in the vicinity, there can be no flooding of other areas and subsequent damage to equipment.

QUESTION 9.7

List the seismic design classification of the various components of the fire protection system. Indicate to what extent this system can function with any single failure. To facilitate understanding, provide a diagram of the system. Identify those portions of the fire protection systems designed to Class 2 seismic standards whose failure could damage Class 1 structures and components. Would failure of a Class 2 portion of the system prevent fire protection to any Class 1 structures or components?

ANSWER

The fire protection system for Indian Point Unit No. 2 plant is an extension of Unit No. 1 of the same station. The system is outlined in the Final Safety Analysis Report, Exhibit B-8, Volume 3, under Section 9.6.2 and shown on piping diagrams Figure 9.6.2-1 and Figure 9.6.2-2.

Unit No. 1 fire protection system, including fire pumps was designed and installed in accordance with design criteria that did not include seismic requirements. This system should be classified as Class III. Unit No. 2 fire protection system, being an extension of Unit No. 1 system, was also designed and installed to satisfy Class III standards.

The materials used on Unit No. 2 fire protection system are as follows:

1. Pipe and Fittings

   Underground: Schedule 40 steel coated and wrapped, cement lined, welded joints; welding fittings

   Aboveground: Schedule 40 steel, cement lined, welded joints, welding fittings

2. Valves

   Underground: FM approved, 175 lbs working pressure, ductile iron, bronze mounted, flanges

   Aboveground: Gate, screwed, 150 lbs working pressure

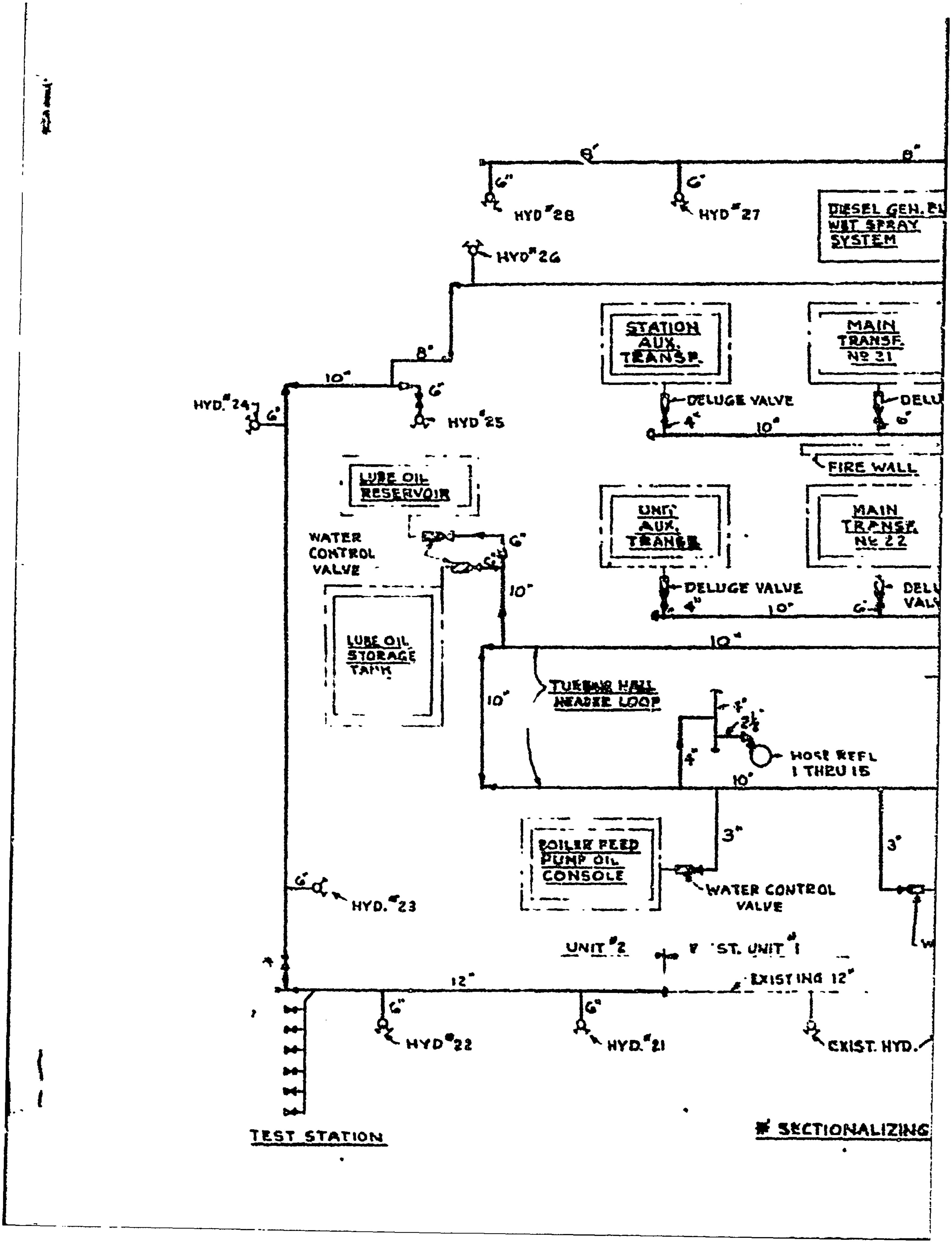
3. Hydrants: FM approved, 175 lbs working pressure, ductile iron

4. Fittings: Diesel Generator System: Malleable iron

The fire line headers are designed as a loop system to permit water flow in either direction. Sectionalizing valves throughout the system are located to permit isolating damaged sections of header, as can be seen in Figure 9.7-1 Diagram of Plant Fire Protection Unit No. 2.

The diesel generator building (Class I structure) fire protection spray system is supplied with water from Unit No. 1 (Class III yard header system). A failure in certain sections of the headers that cannot be isolated by sectionalizing valves could prevent fire protection to the diesel generator building by the spray system. However, there are secondary methods for fire protection in the form of portable extinguishers and nearby hydrants. Fire pumps, though not Class I, are redundant.

There is no Class III portion of the fire protection system whose failure could damage any Class I structure or component.


8"
8"
6"
HYD #28
6"
HYD #27
DIESEL GEN. BL
WET SPRAY
SYSTEM
HYD #26
STATION
AUX.
TRANSF.
MAIN
TRANSF.
№ 21
8"
10"
6"
HYD. #24
6"
HYD #25
DELUGE VALVE
4"
10"
6"
FIRE WALL
LUBE OIL
RESERVOIR
WATER
CONTROL
VALVE
6"
UNIT
AUX.
TRANSF.
MAIN
TRANSF.
№ 22
10"
DELUGE VALVE
4"
10"
6"
LUBE OIL
STORAGE
TANK
10"
10"
TURBINE HALL
HEADER LOOP
4"
2½"
4"
HOSE REEL
1 THRU 15
10"
3"
3"
BOILER FEED
PUMP OIL
CONSOLE
WATER CONTROL
VALVE
6"
HYD. #23
UNIT #2
ST. UNIT #1
12"
EXISTING 12"
6"
HYD #22
6"
HYD. #21
EXIST. HYD.
TEST STATION
SECTIONALIZING

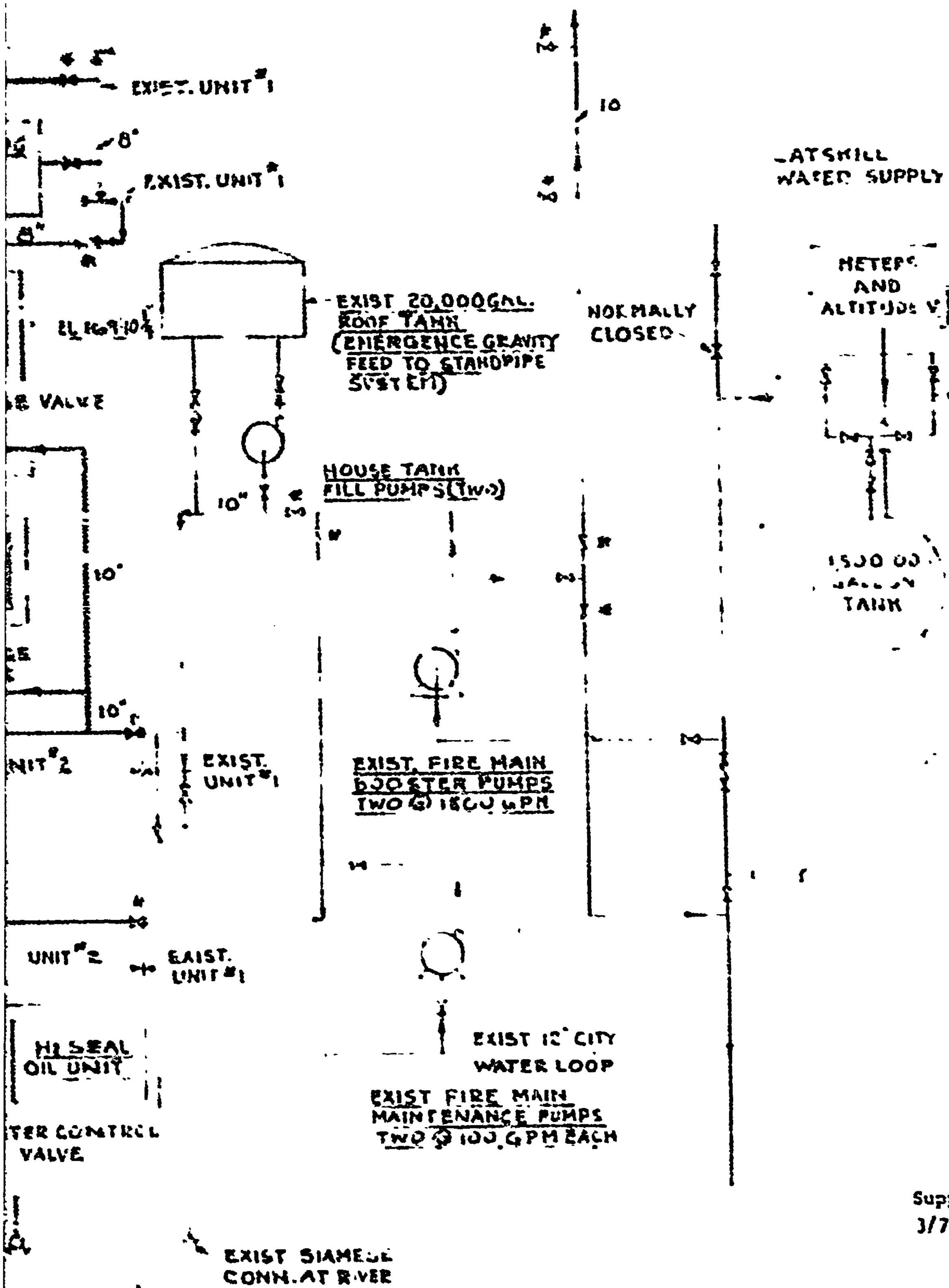


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DIAGRAM OF PLANT FIRE PROTECTION SYSTEM-UNIT 2

VALVES

FIGURE 9.7-1

Question 11.1

Provide a complete description and analyses where required to show how the provisions of 10 CFR 20 will be met for the Indian Point site, taking into account the combined releases from all three of the facilities at this site for both liquid and gaseous effluents on an isotopic basis. Based on the above, provide the exposure dose from effluent releases versus number of people of the exposed population in the vicinity of the site assuming release at expected levels from the three plants and also at the 10 CFR 20 limits. Relate releases at these levels to the amount of failed fuel in the core.

Answer

Indian Point Units Nos. 1, 2 and 3 will be treated as a single facility as far as the release of liquid and gaseous effluents is concerned. The Indian Point site will release liquid and gaseous effluents in a manner to insure compliance with the requirements of 10CFR20.

Compliance to 10CFR20 regulations will be accomplished by administrative procedures that will control the manner in which gaseous and liquid effluents are released, and based on a knowledge of the isotopic content of the effluents.

Control will be exercised at the source of the effluent to insure that the requirements of 10CFR20 are satisfied, Experience to date with operation of Indian Point Unit No. 1 indicates that this is the most efficient and most reliable way of insuring compliance with 10CFR20 regulations.

Compliance to the commitment to operate the site within the guidelines of 10CFR20 is and will be evaluated in three additional ways, two of which are independent of Con Edison, viz:

1. The Consolidated Edison Company's Environmental Monitoring Program.

2. The New York State Radiological Survey Program.

3. The New York University Medical Center Research Program on Hudson River ecology.

These programs which are discussed briefly below provide the secondary control over the release of gaseous and liquid effluents.

A survey to determine the radioactivity in the environment in the vicinity of the Indian Point Station was instituted in 1958, four years prior to the initial operation of Indian Point Unit No. 1. The purpose of this survey was to determine the natural background radioactivity and to show the variations in the activities that may be expected from natural sources, fallout from bomb tests, and other sources in the vicinity. This program has been continued to the present so that changes in the environment resulting from operation of the Station could be accounted for. The results of these surveys are reported semi-annually to the AEC under Docket No. 50-3.

In addition, the New York State Department of Health has conducted surveys throughout the State of New York since 1955, including extensive surveys in the vicinity of Indian Point since 1958. In 1965 and 1966, they reported the findings in the vicinity of the Indian Point Station in two special reports. Since that time, their reporting has been on a statewide basis in quarterly bulletins and in annual reports.

In 1965, the New York University Medical Center began a research program on the ecology of the Hudson River. The New York University studies include the biology of the Hudson River, the distribution and abundance of fish in the river, pesticides and radioecological studies. This program has been supported by the United States Public Health Service, the New York State Department of Health, and the Consolidated Edison Company, and progress is reported in periodic progress reports.

The various studies mentioned above include measurements of radioactivity in fresh water, river water, river sediments, fish, aquatic vegetation, vegetation, soil, and air in the vicinity of the Indian Point Station. The results of these monitoring programs have shown that the Indian Point Station has had no deleterious effect on the environment.

The environmental monitoring program will continue throughout operation of the plant as Indian Point Units Nos. 2 and 3 become operational. New York University, in part supported by Con Edison, will continue their research program to supplement the Company's own environmental program. These data will be made available at regular intervals to the scientific literature and to the regulatory bodies having jurisdiction over these matters. The New York State Department of Health is expected to also continue their statewide monitoring program.

The environmental monitoring program conducted by Consolidated Edison presently supplies and will continue to supply data as a supplement to the primary control at the source of the effluents to insure compliance of Indian Point Units Nos. 1, 2, and 3 with the requirements of 10CFR20.

The results of the environmental program are and will continue to be reported to the AEC and New York State, as summarized below.

Results of the environmental monitoring program are presently and will continue to be reported on a semi-annual basis to the Atomic Energy Commission who are thereby advised of the short and long-term trends in the environment. In addition, discharges of radioactive liquids and gases are presently and will continue to be reported semi-annually to the AEC and the New York State Department of Health. In addition, it should be noted that 10CFR20 makes provision for rapid reporting of any unusually high releases.

In the present Unit No. 1 program, if the monthly release in liquid discharges, exclusive of tritium, were greater than five curies, it has been agreed that a report shall be sent to the New York State Department of Health, Bureau of Radiological Health. Accordingly, the New York State Department of Health is made aware of any significant increase in liquid discharges and their monitoring programs can be accordingly adjusted as required to provide greater intensity of monitoring. This procedure will be incorporated in the program for Units 1, 2 and 3.

In the event that Con Edison detects any Iodine 131 in the air, vegetation, or soil on the plant site, the Company, by agreement, would notify the New York State Department of Health, Bureau of Radiological Health. The Department of Health plans call for an increase in its surveillance program in the vicinity of the Indian Point Station as the need requires, and Con Edison would increase the frequency of collections at the existing sample locations.(1) This procedure will be incorporated in the program for Units 1, 2 and 3.

It should also be noted that the results of the research program on the Hudson River of the Institute of Environmental Medicine of New York University's Medical Center which is concerned both with evaluation of the total ecology and of radionuclides relating to ecology are presently and will continue to be reported on an annual basis to the New York State Department of Health, the Department of the Interior, the U.S. Public Health Service, the Consolidated Edison Company, the Hudson River Valley Commission, the New York State Conservation Department, the New York City Environmental Protection Agency and the New York State Power Development Association, and other interested organizations. Although this is not a monitoring program, the results of this research are germane as they bear upon the manner in which radionuclides behave in the river ecosystem.

To summarize the above, although the design of the Indian Point Units and administrative controls will be such that gaseous and liquid effluents will be released in accordance with the requirements of 10CFR20, the environmental monitoring program of the Consolidated Edison Company provides a redundant means of evaluating that the operation of the Units will not pose any undue risk to the health and safety of the public. The New York State and New York University programs provide an independent means of verifying the proposed facilities compliance with 10CFR20.

The paragraphs that follow discuss in detail the releases of gaseous and liquid effluents to the environment. The criteria required for compliance to 10CFR20 are stated. Analyses are presented to show that these criteria are valid, practical and easily implemented by administrative procedures.

Gaseous Releases

Releases of gaseous activity from the Indian Point site will be in accordance with the requirements of 10CFR20. The release of gaseous effluents from all three units or any combination thereof will be such that the requirements of 10CFR20 are satisfied. These requirements are satisfied if the following criteria are met:

Criterion 1 As individual standing at any point whole body on the site boundary will not receive an annual dose greater than 0.5 rem.

Criterion 2 The concentrations of the released isotopes will not exceed the restrictions of 10CFR20 at ground level at any point on or beyond the site boundary.

The analyses discussed in the paragraphs that follow describe in detail how these criteria can be satisfied.

The analytical approach was as follows:

1. The meteorology was studied in detail in order to determine the site's meteorological dilution factors for multi-unit operation.

2. The site's geometrical and meteorological properties were studied to determine the controlling, with regard to gaseous effluents, sector.

3. The results of steps 1 and 2 were used to determine the releases from each unit that would satisfy Criterion 2.

4. Downwind cloud concentrations from step 3 were used to calculate the whole body immersion dose to insure that Criterion 1 was satisfied.

The analytical studies indicate that for Unit 1, 2 and 3 operation only Criterion 2 governs; that is if Criterion 2 is fulfilled, Criterion 1 is also satisfied.

Figure 1 shows the arrangement of the three units on the site, viz., Indian Point Unit 1 (operating), Indian Point Unit 2 (operation in 1971) and Indian Point Unit 3 (under construction). Indian Point Unit 1 gaseous effluents are to be released from the superheater building stack and Units 2 and 3 will each release from a vent located on top of their respective containment buildings. Hence, from an analytical point of view, Unit 1 releases are represented by a point elevated source, and Units 2 and 3 are each represented by a ground source with wake dilution, i.e., by a virtual point source.

Mathematically for average meteorological conditions, the elevated point source was represented by

$$\chi/Q = \frac{2}{\beta\sqrt{\pi}} \sum_i \frac{F_i \exp[(-h^2/(C_{zi}^2 \; x^{(2-n_i)})]}{\bar{u}_i \cdot C_{zi} x^{(2-n_i/2)}} \qquad (1)$$

where χ = downwind groundlevel concentration at x meters from the stack in Ci/m^3

Q = continuous release rate in C/sec.

β = 2 tan (θ/2)

θ = Angle denoting sector of interest

F_i = fraction of time the wind is blowing into the sector while the meteorological category exists

h = effective height of stack in meters (113m)

x = distance downwind of stack in meters

n_i = dimensionless meteorological stability parameter for i^{th} weather category

C_{zi} = diffusion parameter for i^{th} weather category in $(\text{meters})^{n_i/2}$

$\bar{u}_i$ = average wind velocity in meters/sec. for each weather category

The ground source with wake dilution equation for average meteorology is given by

$$(\chi/Q) = \frac{2}{\beta\sqrt{\pi}} \sum_i \frac{F_i}{\bar{u}_i\, C_i\, x\, [x_i]^{(1 - n_i/2)}} \qquad (2)$$

where the above parameters have been just defined and

(x_i) is given by

$$x_i = x + \left(\frac{A}{\beta C_{yi}\, C_{zi}}\right)^{(1-2/n_i)}$$ (See Q. 11.10 for more detail;

where: A = cross sectional area of containment building in (meters) = 2000 m^2

C_{yi} = diffusion parameter for i^{th} weather category in $(\text{meters})^{n_i/2}$

The meteorological parameters describing the climatology of the Indian Point Site used were obtained from a meteorological program conducted by the Research Division of New York University during 1955-1957. (See section 2.0 of FSAR-IP-2.)

The models presented above were used to determine the dilution factors (χ/Q) as a function of downwind distance for each unit and for each meteorological sectors. Using the shortest site boundary distance within each section, it was determined that the worst annual dilution factors accurred in sector 025-040 at a point located 468 in from IP1, 576 in from IP-2 and 363 in from IP-3. The parameters for the 025-040 meteorological sector are as follows:

Weather Category	C_{yi}	C_{zi}	n_i	$\bar{u}_i$	F_i
Lapse - L_1	0.6	0.48	0.2	1.739	0.0097
Lapse - L_2	0.53	0.43	0.3	5.236	0.0019
Neutral - N	0.47	0.39	0.4	2.793	0.0198
Inversion - I	0.40	0.07	0.5	2.028	0.0574

Constants

$\beta = 2 \tan (10^0) = 0.353$

The dilution factors (χ/Q) obtained for each plant at the site boundary point are tabulated below:

Unit 1 (site boundary distance 468 m) = 1.5×10^{-7} sec/m^3
Unit 2 (site boundary distance 576 m) = 1.38×10^{-5} sec/m^3
Unit 3 (site boundary distance 363 m) = 2.6×10^{-5} sec/m^3

Figure 2 presents the actual site boundary point on Sector (025-040)
Figure 3 presents the annual dilution factor, χ/Q vs normalized dominal distance from IP-2 for sector 025-040
(Note that 576 m from IP-2 corresponds to 468 m from IP-1, and 363 m from IP-3.)

The data shown in Figure 3 in conjunction with Table II of Appendix B to 10CFR20 is all that is required to show how Criterion 2 can be satisfied on an isotopic basis. Since all isotopes can be related to each other through the ratio of their MPC values specified in 10CFR20, it is sufficient to demonstrate compl ance for one isotope and show how compliance for all isotopes is achieved. The isotope Xe-133 is representative of gaseous effluents, hence, it will be used in the following discussions.

The MPC value for Xe-133 is 3×10^{-7} C/m^3. Based on this and the computed χ/Q values at the site boundary, the maximum permissible release rates of Xe-133 based on Criterion 2 for Units 1, 2 and 3 assuming that each is the only unit operating are:

Unit 1 maximum permissible release rate (MPQ_1) = 2.0 Ci(Xe-133)/sec
Unit 2 maximum permissible release rate (MPQ_2) = 2.18×10^{-2} Ci(Xe-133)/sec
Unit 3 maximum ·ermissible release rate (MPQ_3) = 1.155×10^{-2} Ci(Xe-133)/sec

That is, assuming that any plant is releasing alone Xe-133 for a whole year at a release rate of MPQ quoted jus above, a concentration of 3.0×10^{-7} Ci/m^3 will be achieved at a point located 576 m from IP-2, 363 m from IP-3, and 548 m from IP-1. It must be emphasized that a technical specification value of $\chi/Q = 5.88 \times 10^{-7}$ sec/m^3 is presently used as a basis for IP-1 releases so that the maximum permissible release rate for IP-1 becomes 0.51 Ci/sec which is about 25% of that quoted above; with this specified maximum release rate, the maximum concentration of Xe-133 due to IP-1 releasing alone is about 1/4 of the MPC value for Xe-133 on $\underset{\sim}{\sim}$ 0.75 $\times 10^{-3}$ Ci/m^3.

The cloud-dose calculations required to demonstrate compliance with Criterion 1 consider the effects of both β and γ radiation. Since the effect of various isotopes with regard to doses are related through their MPC values, only one isotope, viz., Xe-133 was analyzed. This isotope emits a 0.35 MeV β and a 0.081 MeV γ.

Beta radiation has a limited range (1 to 10 meters in air), hence, a finite cloud of β emitters with an extent greater than or equal to that of the range of the β particles can be considered to be an infinite cloud. For a cloud of this type equilibrium conditions can be assumed which yields the following expression for the β dose (Cf. pp 328 ff, TID-24190)

$$D_\beta(x,y,z) = 0.457\ \bar{E}_\beta\ \psi(x,y,z) \tag{3}$$

where

$D_\beta(x,y,z)$ = total direct β dose at point X, Y, Z, in the cloud in rads.

$\bar{E}_\beta$ = average β energy per disintegration in MeV/dis.

$\psi(x,y,z)$ = concentration time integral at point X, Y, Z in curie-sec/m^3

Equation 3 describes the dose rate from an infinite cloud. In the actual case the dose rate will be about one half that calculated from the infinite cloud (Cf. p 330, TID-24190) because of the receptor's perturbation to the field. An additional perturbation that reduces the dose rate by a factor of about 0.75 is caused by the surface of the earth. Accounting for these and by expressing $\psi(x,y,z)$ in familiar terms, equation 3 becomes:

$$D_\beta(x,y,z) = 0.17Qt\bar{\bar{E}}_\beta \qquad (4)$$

where Q is redefined as C/m^3

t = time of exposure to cloud in seconds

Equation 4 was used to calculate the cloud beta dose.

The cloud gamma dose was obtained from a complex numerical integration technique. Basically the procedure was as follows:

1. The concentration at any point in space, i.e., $\chi(x', y', z')$ was represented by Sutton's well known point source equation.

2. The integral equation representing the dose at any point in space x, y, z due to an infinitesimal volume source at x', y', z' was determined.

3. Integration in the y and z plane was performed using an Hermite approximation, and in the x direction by a Gaussian approximation.

A 20th degree polynomial was used initially for the numerical fits. For distances greater than 10^4 meters this technique produced doses that fell off too rapidly with distance. To correct this situation a 32nd degree polynomial has been employed. Results for distances greater than 10^4 meters were not required for this analysis.

Using the above technique the dose at a point x, y, z in space is represented analytically by

$$D_\gamma (x,y,z) = \int_0^\infty \int_{-\infty}^{\infty} \int_{-\infty}^{\infty} G(r-r') \phi (x', y', z')\, dx'\, dy'\, dz' \qquad (5)$$

where

$D_\gamma(x,y,z)$ = total γ dose at point x, y, z for a single isotope

ϕ (x',y', z') = gamma source strength at point x', y', z'

$G(r-r')$ = γ dose at r due to a unit source at r'

The numerical representation is as follows:

$$D_\gamma(x,y,z) = \xi \sum_{nn=1}^{\infty} L_{nn} \sum_{n=1}^{K} H_{n\phi} \sum_{m=1}^{1} H_{m\theta}\, G(r-r') \quad (6)$$

where

ξ = constant

L_{nn} = Legendre weighting factor

$H_{n\phi}, H_{m\theta}$ = Hermite weighting factors

Equation 6 was used to calculate cloud gamma doses.

Figures 4, 5, and 6 show the annual dose vs nomalized distance from IP-2, for each plant as if each were releasing alone using the MPQ release rates mentioned before and the β and γ dose models just described. It is shown that for any plant, Criterion 1.0 is met at 576 m from IP-2, 363 m from IP-3 and 448 m from IP-1 stack even using the MPQ calculated from the computed χ/Q's tabulated on page Q 11.1-8. Thus, if Criterion 2.0 is satisfied, Criterion 1.0 is also satisfied.

Figure 7 shows the expected annual doses (Xe-133) vs normalized distance from IP-2 for the Indian Point Site. These are based on Unit 1 operating experience and Units 2 and 3 with 1% failed fuel and no primary to secondary leakage. The expected annual dose at the site boundary is shown to be about 0.008 rem.

Since the site will be operated in a manner that satisfies Criteria 1 and 2, a person residing at any point immediately beyond the site boundary is assured of receiving a dose less than 0.5 rem per year due to permissible gaseous release rates. The expected dose as shown will be well below this value. However, the actual residential pattern (cf, Figure 1) in the Indian Point vicinity is such that the nearest resident would receive a dose significantly below 0.5 rem annually even if the plant were discharging at Part 20 limits. This dose reduction is due to the fact that the nearest residents are approximately 700 meters east of the Unit No. 1 stack and well out of the meteorological sector (025-040). Figure 11 shows the nearest private dwellings to be in the vicinity of Broadway and Bleakley Avenues and to be located at a point in a meteorological sector where the yearly average χ/Q indicates approximately three times greater atmospheric dispersion that at the site boundary in the 025-040 sector.

A good estimate of the dose incurred by the most approximate resident is approximately 0.1 rem per year if any of the plants were to discharge gaseous effluents alone continuously at Part 20 limits.

From Figure 11, it is also evident that no significant population density exists within 1100 meters of the plants. The initial high-density region is east of the site, beyond 1100 meters, where χ/Q values are about an ord r of magnitude below those in the 025-040 sector at the site boundary. Therefore, if any of the facilities are releasing alone at Part 20 limits, the initial high population region will experience dose rates below 0.05 rem per year. As the distance from the plant increases even further, dose rates rapidly decrease.

It has been shown that if any plant releases gaseous effeuents alone for one year at its corresponding maximum permissible release rate, Criterion 2.0 is satisfied. It has also been shown that if Criterion 2.0 is fulfilled, Criterion 1.0 is also satisfied.

Now, Criterion 2.0 may be expressed mathematically by:

$$\frac{(\chi/Q)_j\, Q_{ij}}{MPC_i} \leq 1.0$$

where

$(\chi/Q)_j$ = dilution factor for the j^{th} plant at the site boundary (sec/m^3)

Q_{ij} = the release rate for the i^{th} isotope in the j^{th} plant (Ci/sec)

MPC_i = maximum permissible concentration specified by 10CFR20, Appendix B, Table II for the i^{th} radioisotope

(Note that when the above expression equals 1.0, Q_{ij} becomes MPQ_{ij}.)

Thus it can be seen that if Units 1, 2, and 3 are operated as a single facility with the restriction that

$$\sum_{ij} (\chi/Q)_j \frac{Q_{ij}}{MPC_i} \leq 1.0$$

Then Criterion 2.0 is fulfilled for the site and consequently, Criterion 1.0 is also satisified for the site.

Thus, compliance to 10CFR20 regulations is assured by the knowledge of the isotopic content of the effluents and the administrative procedures which control the mannual in which the effluents are released.

LIQUID RELEASES

Release of liquid activity from the Indian Point site will be in accordance with the requirements of 10CFR20, i.e., the release of liquid activity from all three units will be such that the requirements of 10CFR20 are satisfied. This requirement is satisfied if the following criterion is met:

Criterion 3

The concentration of released isotopes in the outfall of the plant prior to entrance in the river will not exceed the limits set forth in 10CFR20.

Units 1, 2 and 3 will discharge liquid effluents to the river via a common condenser circulating water discharge canal. The circulating water when discharged to the river will mix with the river, thereby diluting the concentrations of contaminated effluent; hence, the concentrations of liquid effluents in the river will never reach 10CFR20 limits.

Analytical models have been developed to study the transport of liquid effluents in the river. These models are discussed in detail in Section 2.5 of the Indian Point No. 2 FSAR.

Considerable work has been completed on the dispersion of contaminants in the river. Some of this work is discussed below in order to provide a complete description of the behavior of liquid wastes in the river, and to show that these effluents in no way adversely affect the existing potential sources of drinking water at Chelsea and Castle Point Veterans Hospital. The former is located 22.0 miles upstream and the latter 20.5 miles upstream of the site. Studies indicate that the concentrations of effluents at Chelsea and Castle Point are essentially the same: hence, only one need be discussed here.

The river's salinity is a very significant parameter in the analyses because it is an excellent indicator of the upstream movement of any pollutant. Salinity provides the following information on the upstream transport of Indian Point effluents:

1. If salt is not present at Chelsea, then neither will any pollutant discharged from Indian Point Station.

2. When salt is present at Chelsea, the ratio between the salt concentrations at Indian Point and Chelsea is a measure of the "mechanical dilution" i.e., dilution due to the river's flow and dispersion characteristics. Since all isotopes decay and salt does not, this dilution factor will yield the minimum reduction in isotope concentrations which will occur between Indian Point and Chelsea.

   The upstream movement of salt is the result of a rather delicate balance which is struck between the salinity-induced density currents, which tend to drive the salt itself up the estuary and fresh water flow, which tends to hold back the salt movement. The river's dispersion characteristics are strongly influenced by this phenomenon.

There are four factors governing the concentrations at Chelsea, via:

1) The salinity at Chelsea.
2) The half life of the isotope.
3) The distance to Chelsea.
4) The time for the isotope to reach equilibrium.

The effect of each of these factors is shown on Figures 9, 10 and 11, for a normalized release of 1 c/day for each isotope. Equilibrium values are 80% of the value at time equals infinity.

Two values of salinity at Chelsea are considered in these figures. The 2000 ppm value corresponds to a salinity level that would represent a realistic upper level for New York City use. Chelsea would not be operated at a salinity level above 1000 ppm. The 2000 ppm value at Chelsea occurred during the 1964 drought and is the highest level of salinity recorded at Chelsea.

The figures show that:

1) Isotopes with half lives less than 1 day will never reach any significant level at Chelsea.

2) Isotopes with half lives 1.0 day $< t_{1/2} <$ 10.0 days will reach equilibrium in less than 33 days for the drought condition and the equilibrium concentration is strongly dependent on the half life.

3) Isotopes with half lives greater than ten days have equilibrium concentrations that are almost independent of the half life.

4) The time to reach equilibrium for isotopes with half lives greater than 100 days is strongly dependent on the salinity at Chelsea.

The occurrence of 2000 ppm of salt at Chelsea has only been observed once, the latter part of 1964 after six months of low flow. Hence, realistically, only isotopes with half lives 1 day < $t_{1/2}$ < ten days will ever approach equilibrium concentrations, i.e., assuming that the plant is continuously discharging these isotopes during the drought period.

The normalized curves can be used to determine the concentration at Chelsea for any isotope. For example, assume that I-131 ($t_{1/2}$ = 8.05 d) is the only isotope released, then 1,170 curies could be released during a year (3.2 curies/day) with all three units operating. This would result in a concentration at Chelsea of 3×10^{-9} μc/ml, which is 1.0% of MPC.

The preceding shows that if Criterion 3 is satisfied, the concentration in the river will always be below MPC and at Chelsea the concentration will not be significant.

OPERATION WITH FAILED FUEL

It is theoretically possible, but practically not very feasible, to achieve the high plant release rates associated with operation of the plant at 10CFR20 levels.

If the plant is operating with no primary to secondary leakage, a 60% clad defect level would be required to obtain gaseous release rates of 0.02 c/sec of equivalent Xe-133. Due to large hold-up times (45 days) for noble gas activity almost all of the activity released under these conditions would be Xe-133 and Kr-85. With 60% defective fuel and no primary to secondary leakage 15.0×10^{-2} curies of I-131 would be discharged into the river annually. Based on a typical liquid release with failed fuel approximately 380 curies could be discharged annually to the river, therefore, it can be seen that in this mode of operation, noble gas releases would be the limiting factor on plant operation.

When primary to secondary leakage is introduced into the analysis, many combinations of leakage and percent defective fuel exist which would result in the plant discharging at 10CFR20 limits. For a 0.02 c/sec release rate of equivalent Xe-133 a tabulation of typical combinations of leak rate and failed fuel is possible:

Steam Generator Leak GPM	Allowed Defect Leak %
60.0	0.052
30.0	0.061
15.0	0.080
1.5	0.32
0.15	2.8

A typical mixture of noble gases for a 4 gpm leak and 0.16% fuel defect level is presented immediately below:

Isotope	Release Rate Curies/sec
Kr-85	1.9×10^{-5}
Kr-85m	2.4×10^{-4}
Kr-87	12.0×10^{-5}
Kr-88	3.8×10^{-4}
Xe-133m	18.4×10^{-5}
Xe-133	10.8×10^{-3}
Xe-135m	17.6×10^{-6}
Xe-135	11.0×10^{-4}

If the release rates for these eight isotopes are weighted by their MPC's and added, the result will indicate that the plant is discharging approximately 0.02 c/sec equivalent Xe-133.

When steam generator leaks are included in an analysis such as this, liquid releases via the blowdown system will be significantly more limiting than the noble gas effluents just discussed. A good "feel" for the situation can be obtained by considering the I-131 limit in the discharge canal during operation with failed fuel and boiler leakage. When the condenser flow rate is 840,000 gpm (Unit No. 2 alone) the facility can discharge 1.04 c/day of I-131. This combination of flowrate, isotopes released, and release rate (see response to Question 11.9) results in 1MPC in the condenser outfall which is the limit specified in Criterion 3.

A specific case can be used to illustrate the fact that liquid discharges will be limiting insofar as releasing effluents to the environment with failed fuel and steam generator leakage. The postulated situation will be the design failed fuel level where 1% of the fuel elements are defective and the steam generator leakage is such that 1.04 c/day of I-131 are being released to the Condenser outfall with a primary to secondary leak rate of 0.11 gpm. The tabulation which follows presents the annual releases of the most significant isotopes under these operational conditions:

"LIQUID DISCHARGES"

Isotope	Annual Release Curies
I-131	3.8×10^2
I-132	1.9×10^1
I-133	3.4×10^2
I-134	5.0
I-135	1.1×10^2

"GASEOUS DISCHARGES"

Isotope	Annual Release Curies
Kr-85	4.0×10^3
Kr-85m	3.8×10^2
Kr-87	1.8×10^2
Kr-88	5.7×10^2
Xe-133	5.0×10^4
Xe-133m	4.9×10^2
Xe-135	2.0×10^3
Xe-135m	2.4×10^1
Xe-138	7.3×10^1

It can immediately be discerned from these listings that the 1.04 c/day I-131 equivalent limit is reached while the 0.02 c/sec equivalent Xe-133 limit has not been attained (the above listing of gas releases represents a 0.003 c/sec equivalent Xe-133 release rate).

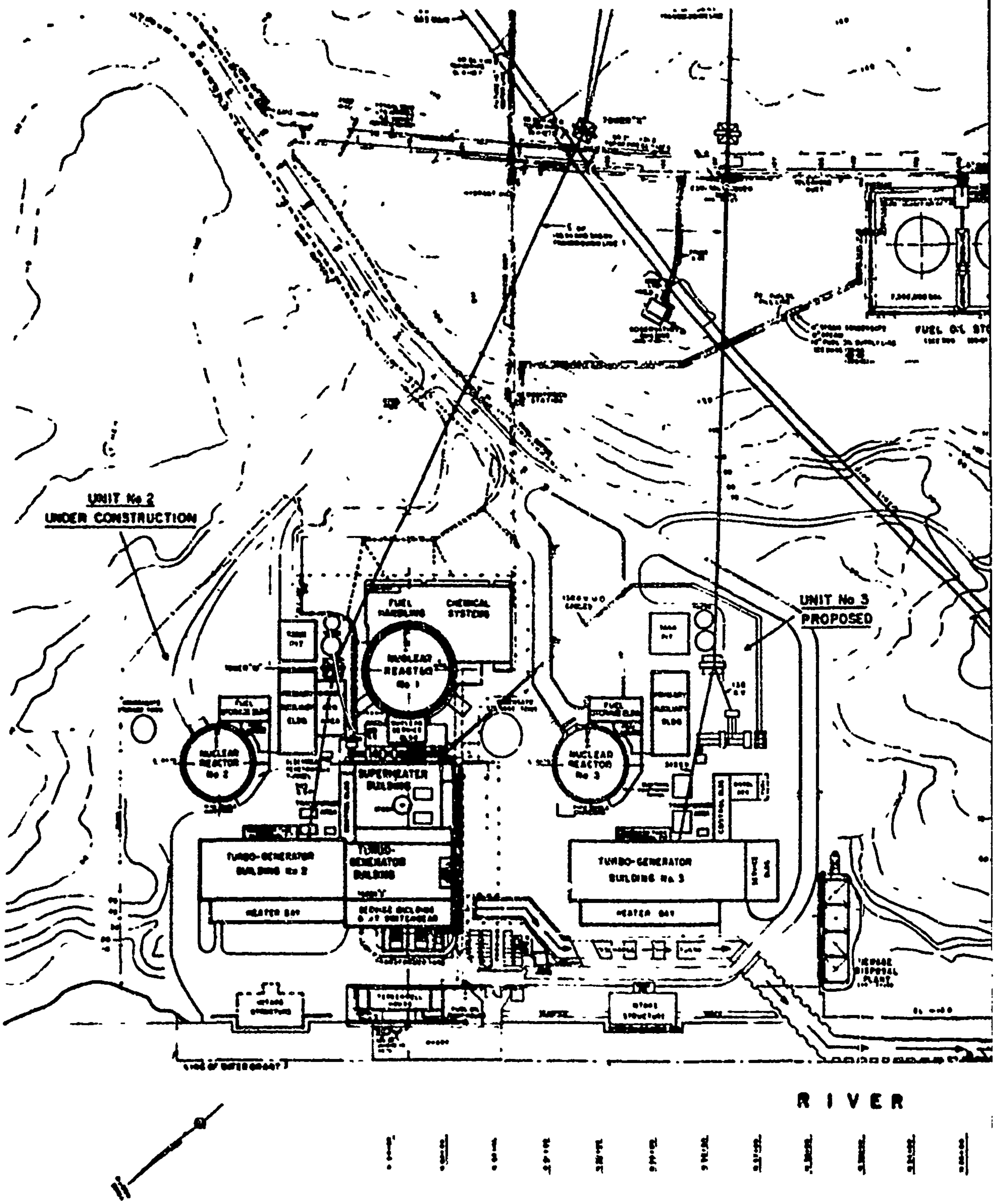
The following assumptions and parameters were used:

Core thermal power	3216 MWt
Reactor coolant volume	12,600 ft^3
Purification flow rate	75 gpm
Mixed bed demineralizer decontamination factors:	
a. Noble gases	1.0
b. Y-90, 91, Mo-99, Cs-134, 136, 137	1.0
c. Other isotopes	10.0
Cation demineralizer	
a. Effective flow	7 gpm
b. Y-90, 91, Mo-99, Cs-134, 136, 137 decontamination factor	10
Waste disposal design	
a. Decay time provided for normal gas release	45 days
b. Decay time for liquid process	500 minutes
c. Waste evaporator decontamination factor	10^6
Dilution Flow rate in discharge canal	840,000 gpm
Volumes reactor coolant wastes processed per year	4

As discharges to the river approach the I-131 limit of 380 Curies annually, releases via the blowdown tank vent warrant consideration. When liquid enters the tanks and flashes, some fraction of the Iodine activity will partition between the steam and water phases (the partition factor will be 0.1 as defined in the response to 11.2). The state of the blowdown is such that approximately one-third of it will flash giving a steam to water weight ratio of one-half. Under these circumstances, and with the blowdown rate being such that 380 c/year of I-131 would be discharged to the plant outfall if no partition occurred, than with partition no more than 19 c/year of I-131 would be vented directly to the atmosphere from the blowdown tank. The release of 19 Curies per year of I-131 via the blowdown tank vent released uniformly throughout the year will not exceed Criteria 1 and 2. Since higher partition factors are expected, less than 19 Curies per year of I-131 will be released. The release rate of I-131 via the blowdown vent may be approximated by analytical techniques.

References

(1) Technical Specification and Bases - Specification 4.10.



UNIT No 2
UNDER CONSTRUCTION
UNIT No 3
PROPOSED
FUEL HANDLING
CHEMICAL SYSTEMS
NUCLEAR REACTOR No 1
NUCLEAR REACTOR No 2
NUCLEAR REACTOR No 3
SUPERHEATER BUILDING
TURBO-GENERATOR BUILDING No 2
TURBO-GENERATOR BUILDING
TURBO-GENERATOR BUILDING No 3
HEATER BAY
RIVER

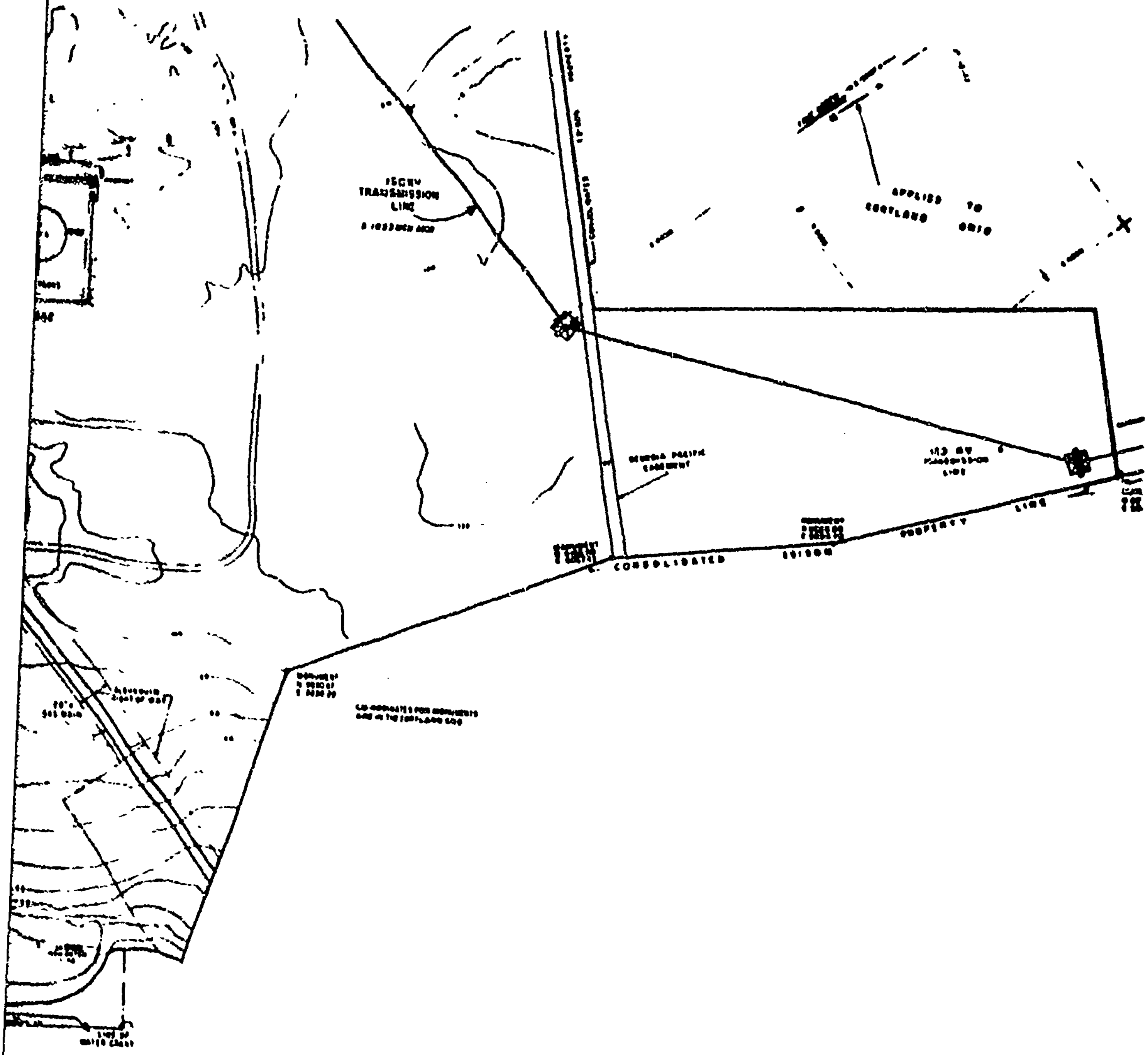


FIGURE 1

ACTUAL DIRECTION OF
METEROLOGICAL SECTOR (025-040)

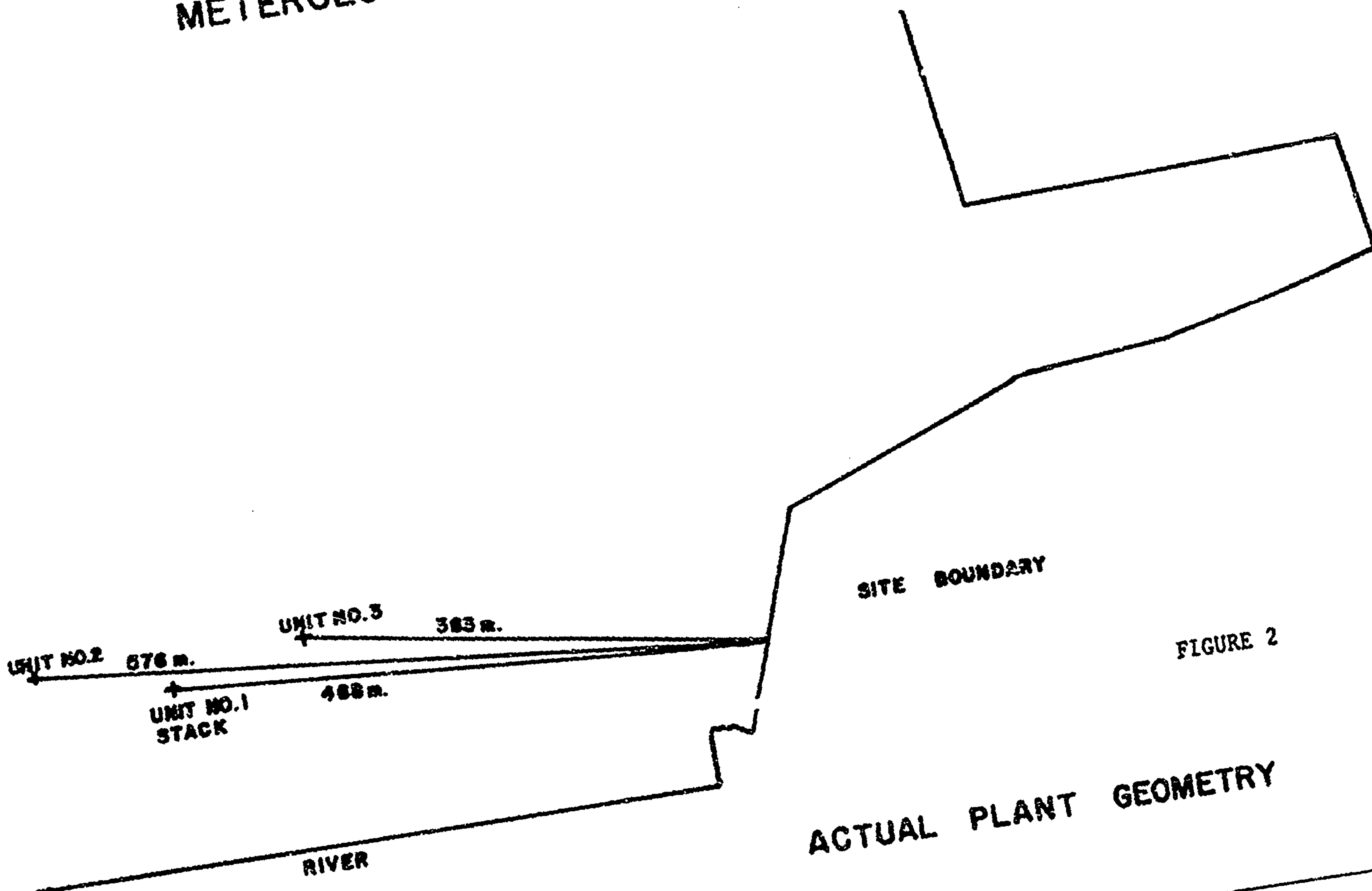


FIGURE 2

ACTUAL PLANT GEOMETRY

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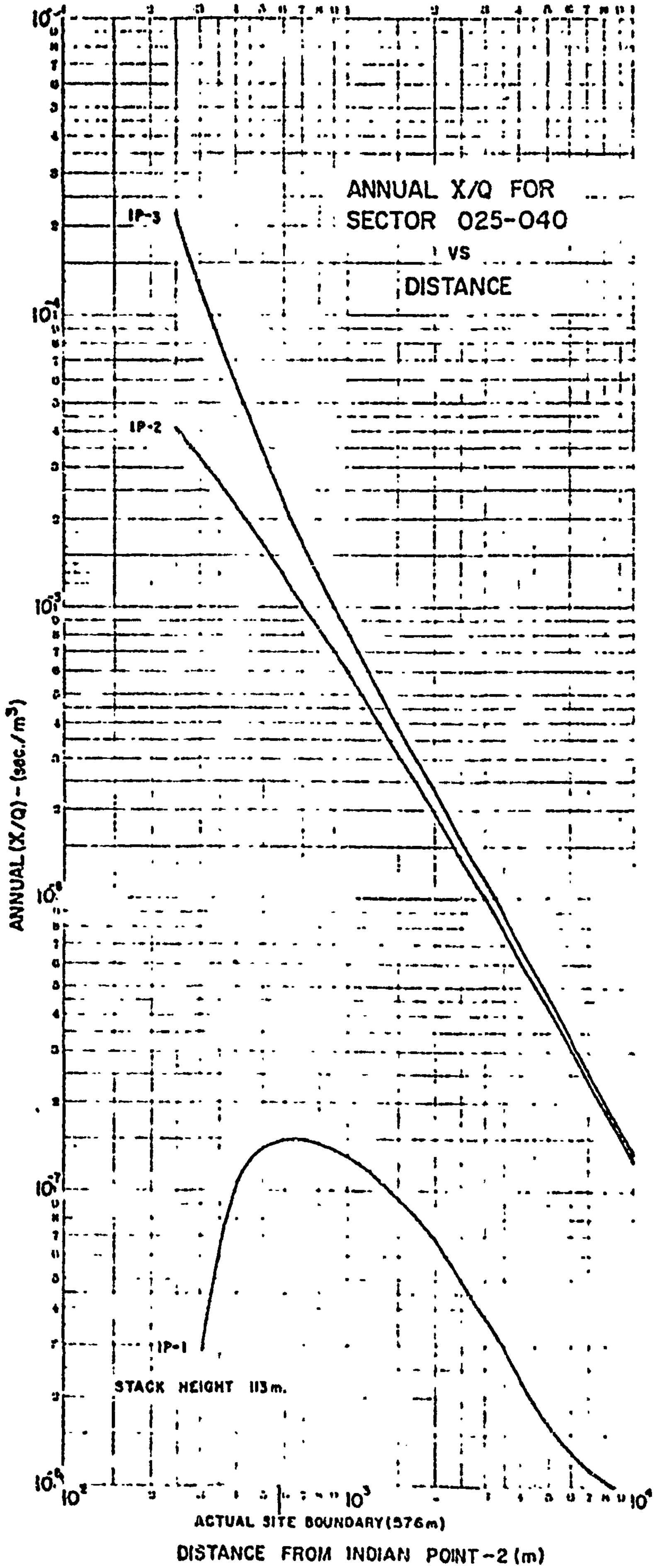

FIGURE 3

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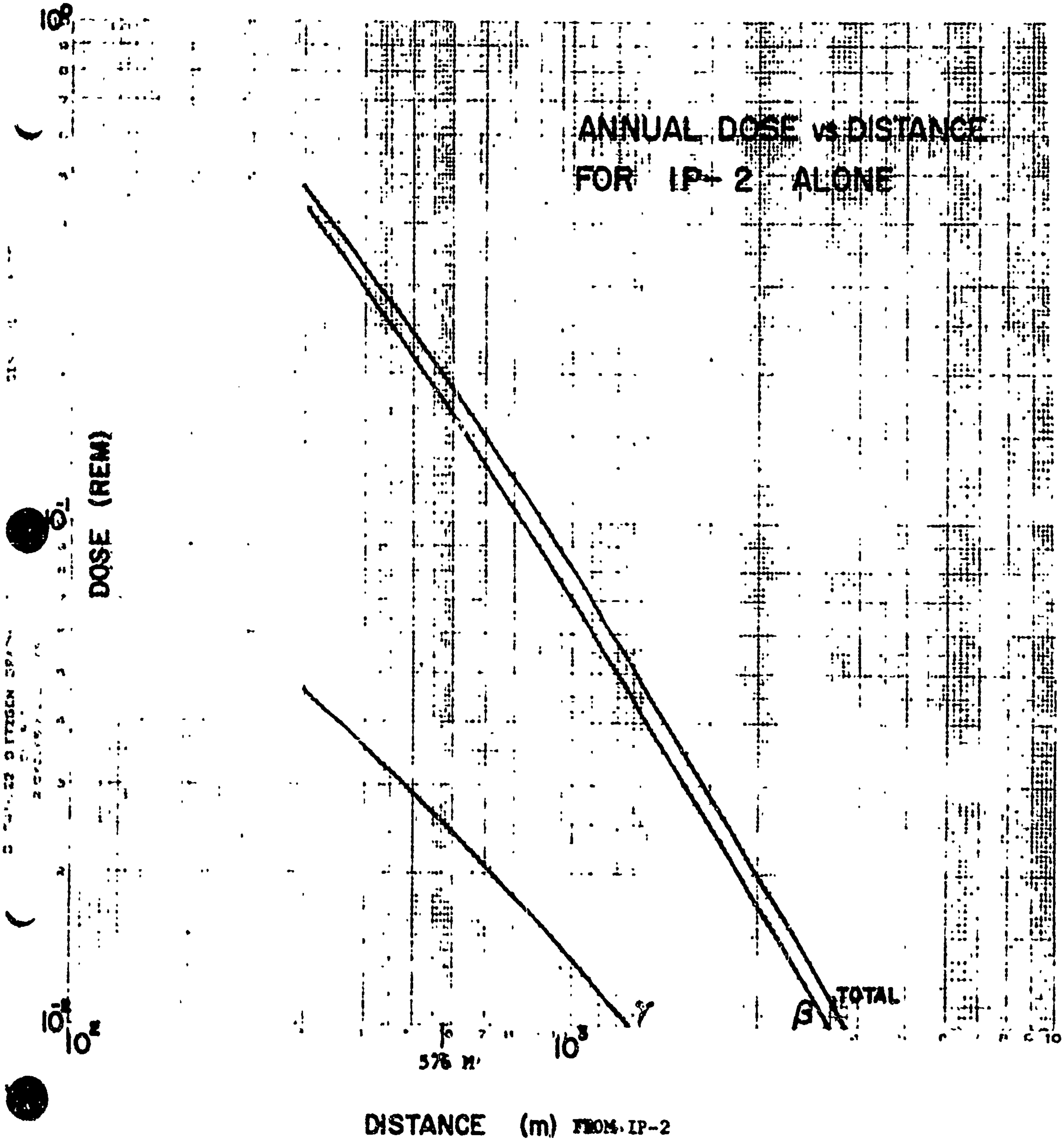
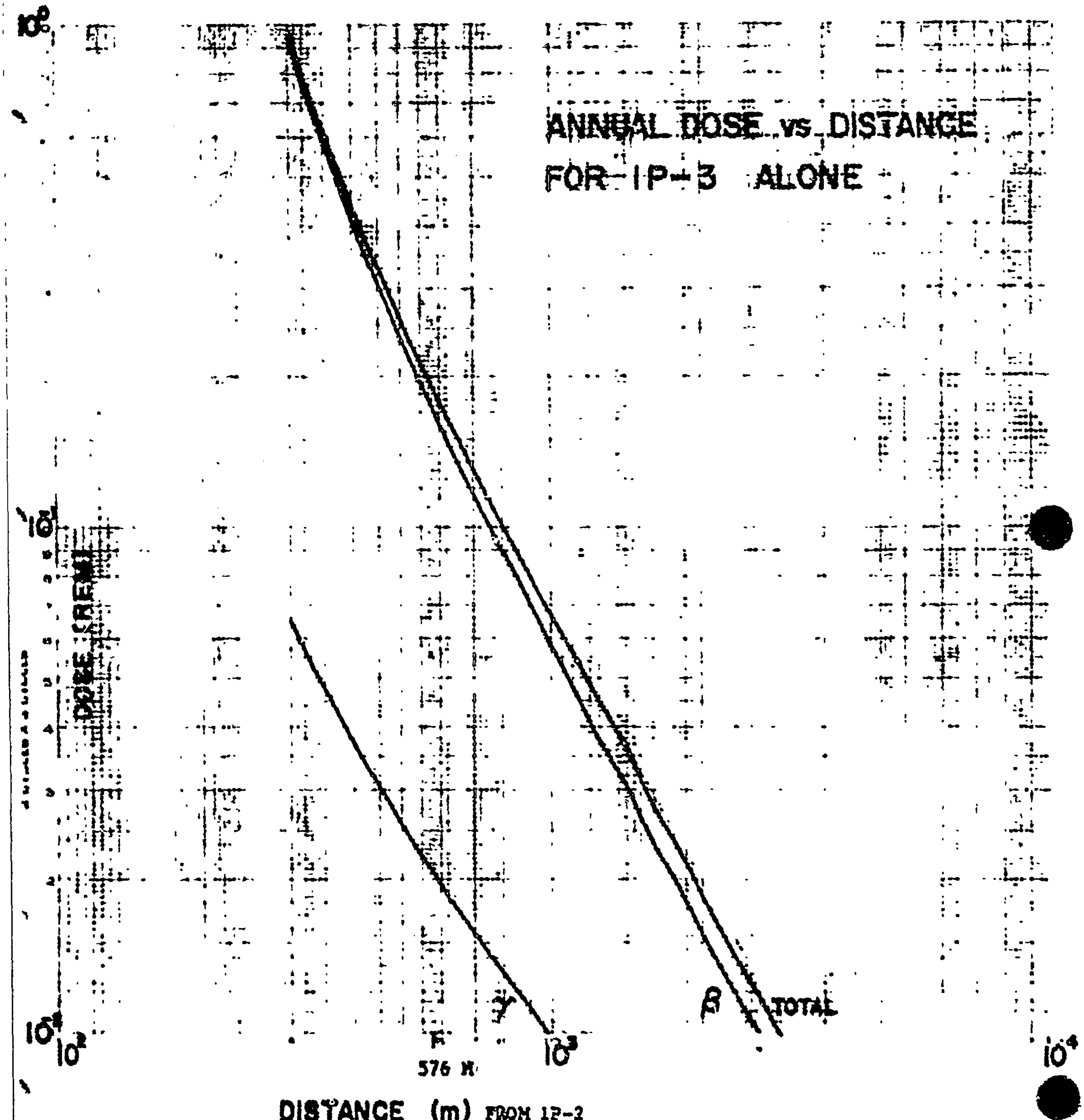


ANNUAL DOSE vs DISTANCE
FOR IP-2 ALONE
DOSE (REM)
TOTAL
576 M
DISTANCE (m) FROM IP-2

FIGURE 6



ANNUAL DOSE vs DISTANCE
FOR IP-3 ALONE
DOSE (REM)
10^2
10^1
10^0
10^2
10^3
10^4
576 M
γ
β
TOTAL
DISTANCE (m) FROM IP-2

FIGURE 5

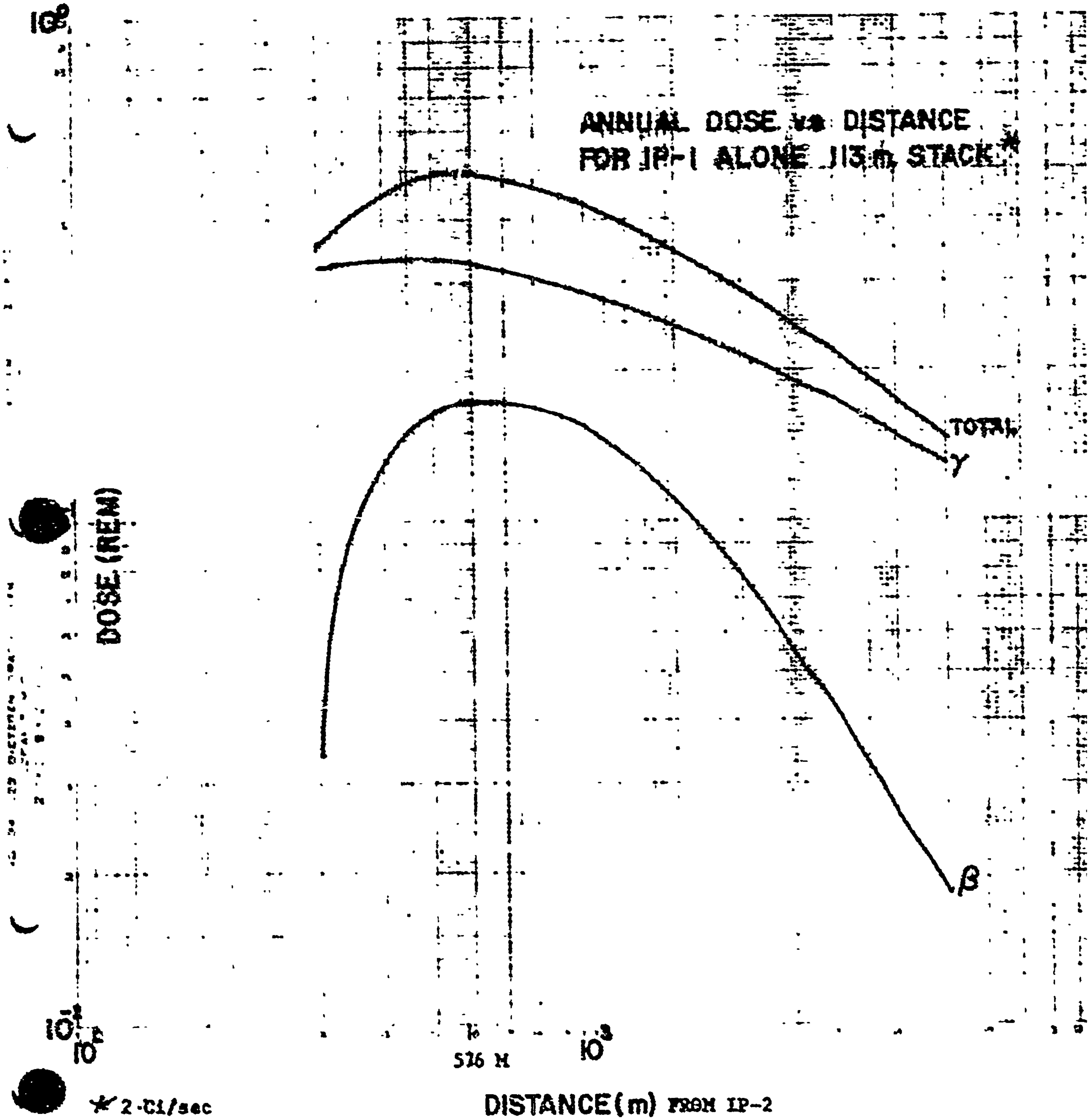


FIGURE 6

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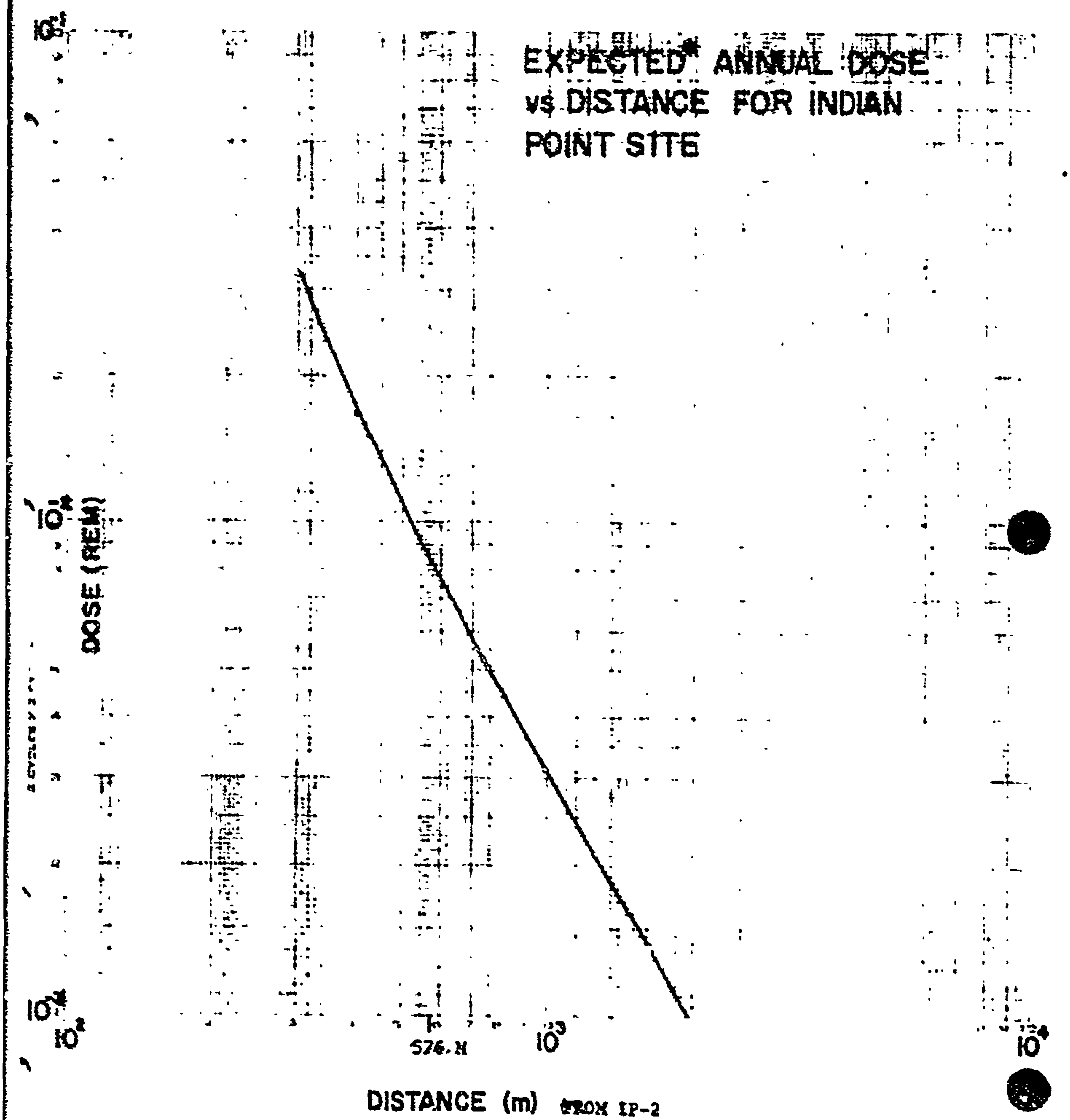

*—FOR UNITS NO. 2 and 3 WITH 1% FAILED FUEL

FIGURE 7

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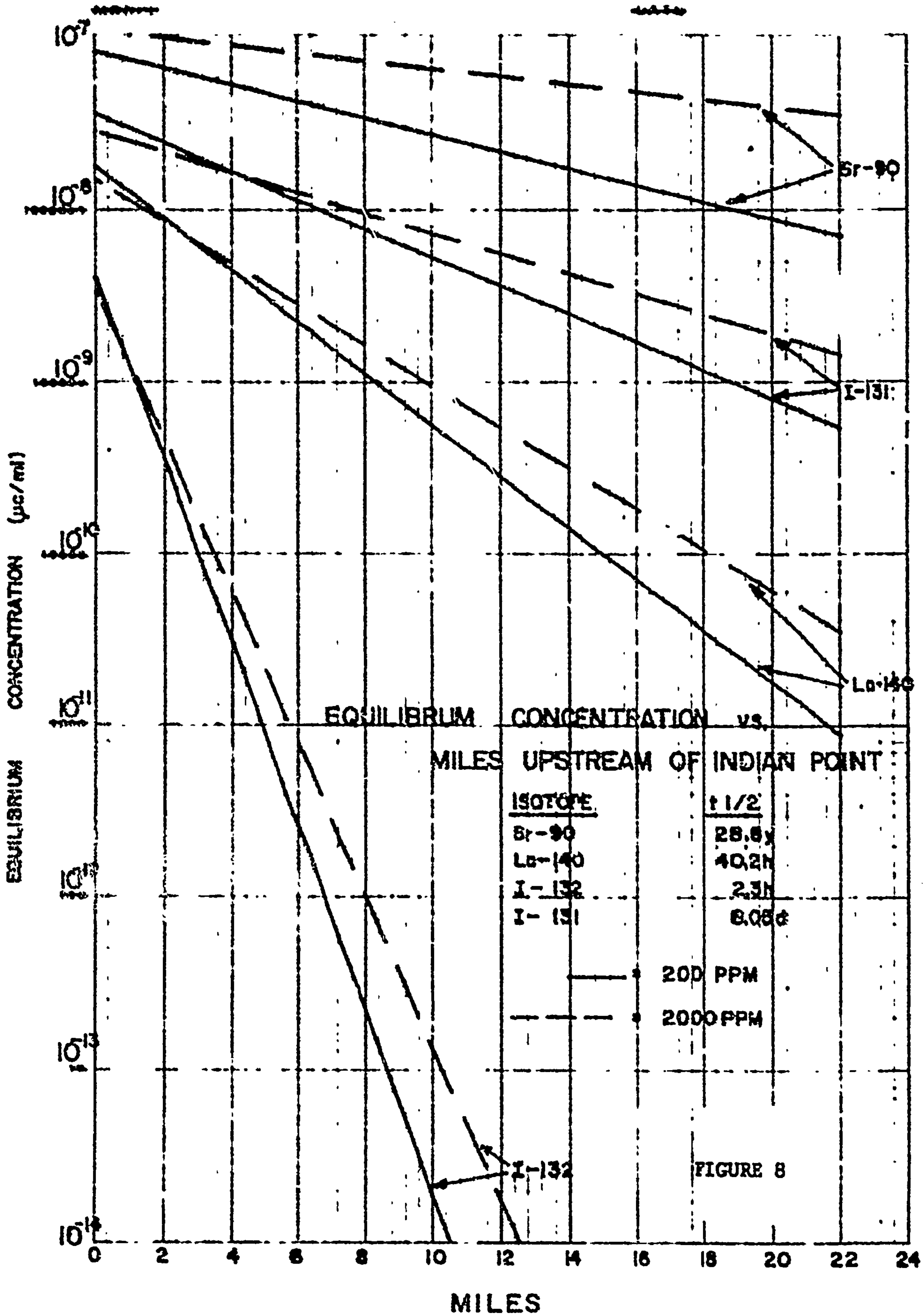

FIGURE 8

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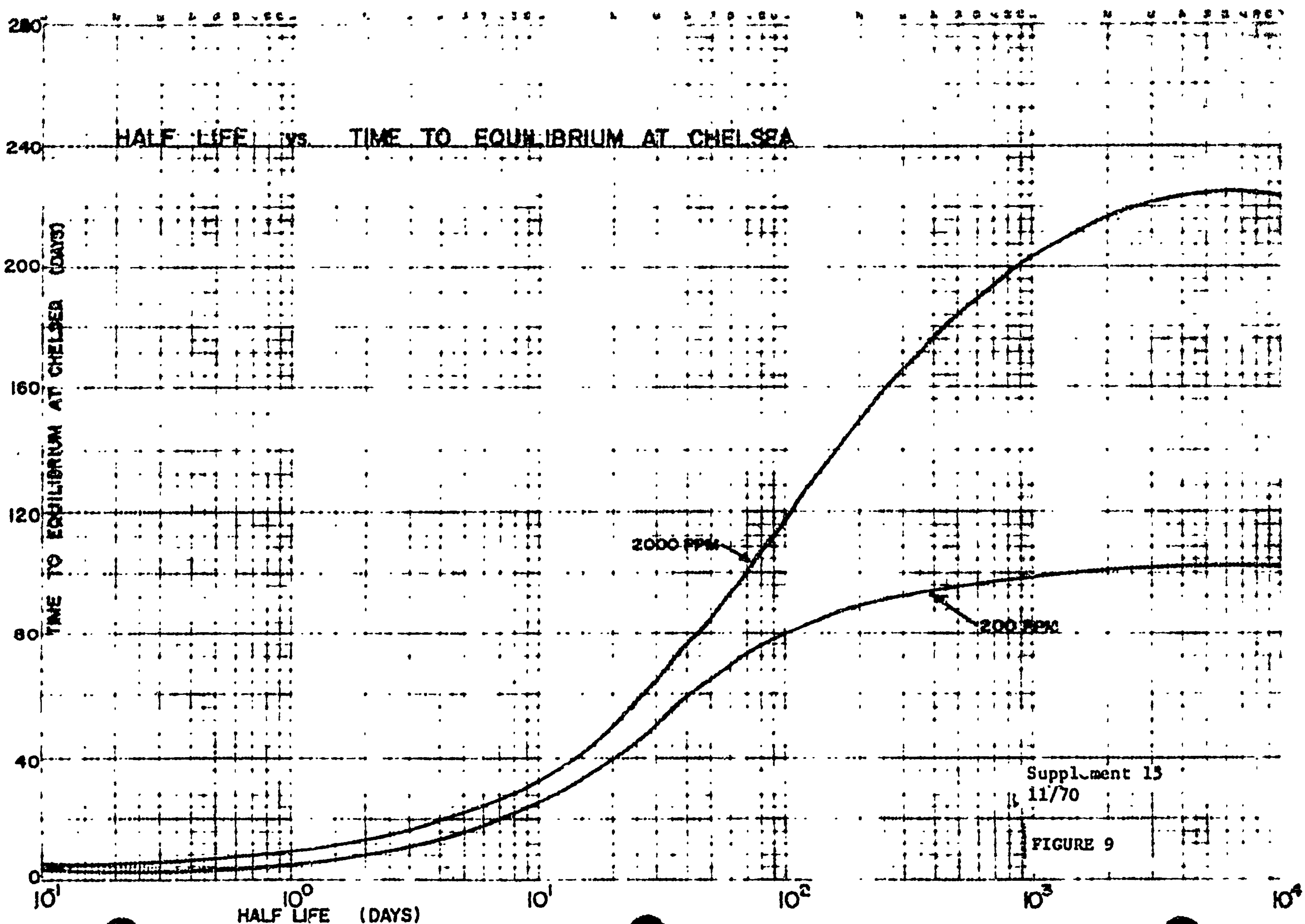

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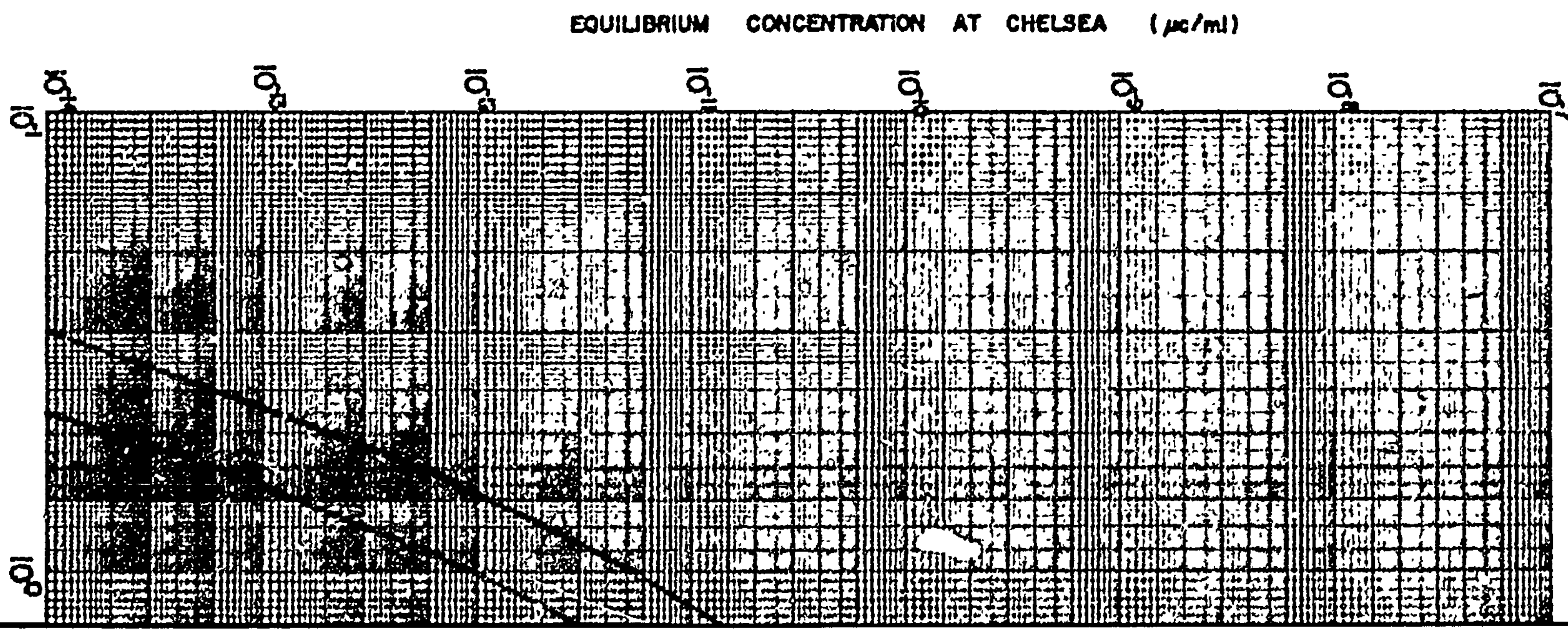
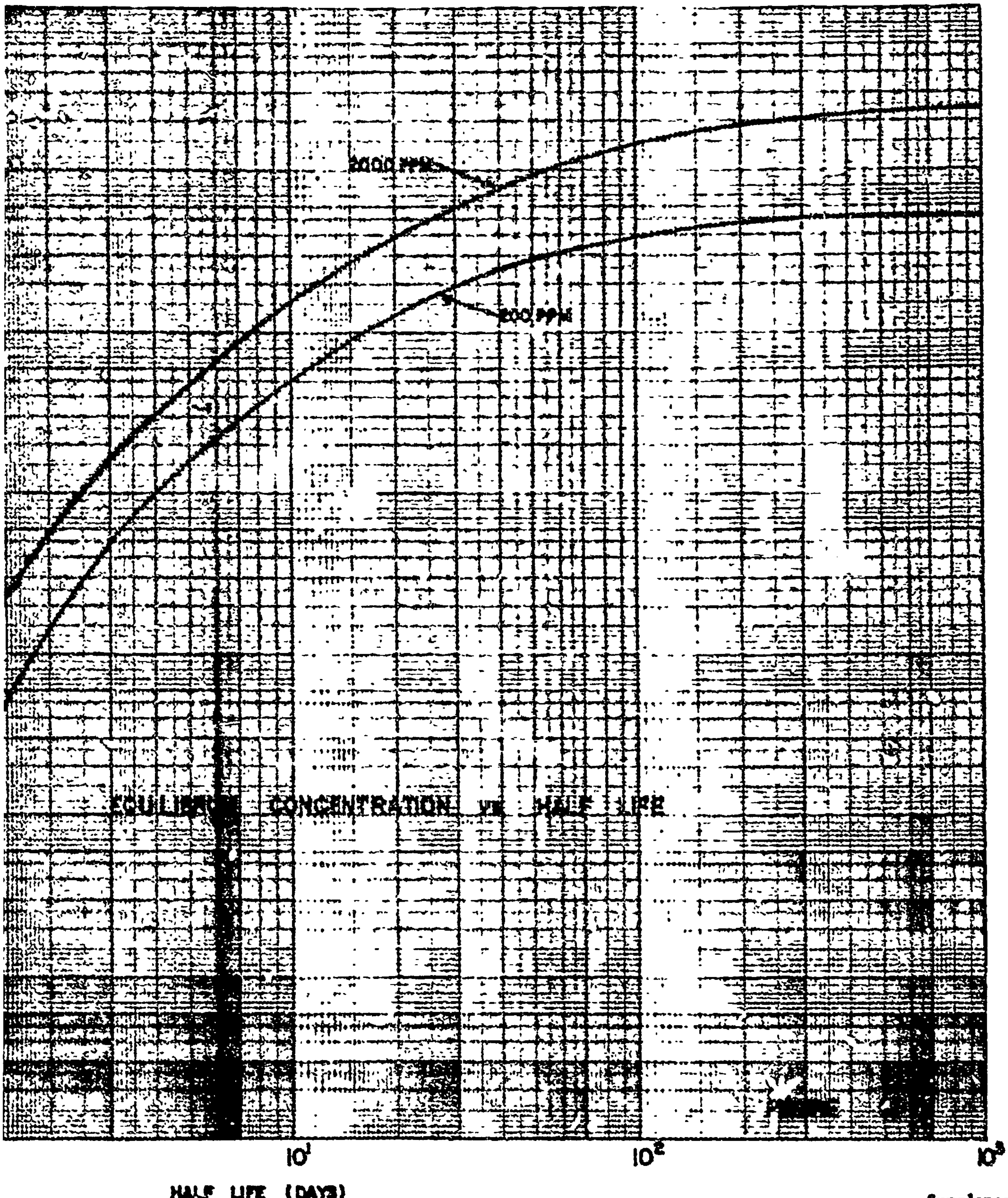
FIGURE 9


EQUILIBRIUM CONCENTRATION AT CHELSEA (μc/ml)


2000 PPM
200 PPM
CONCENTRATION vs HALF LIFE
10^1
10^2
10^3
HALF LIFE (DAYS)

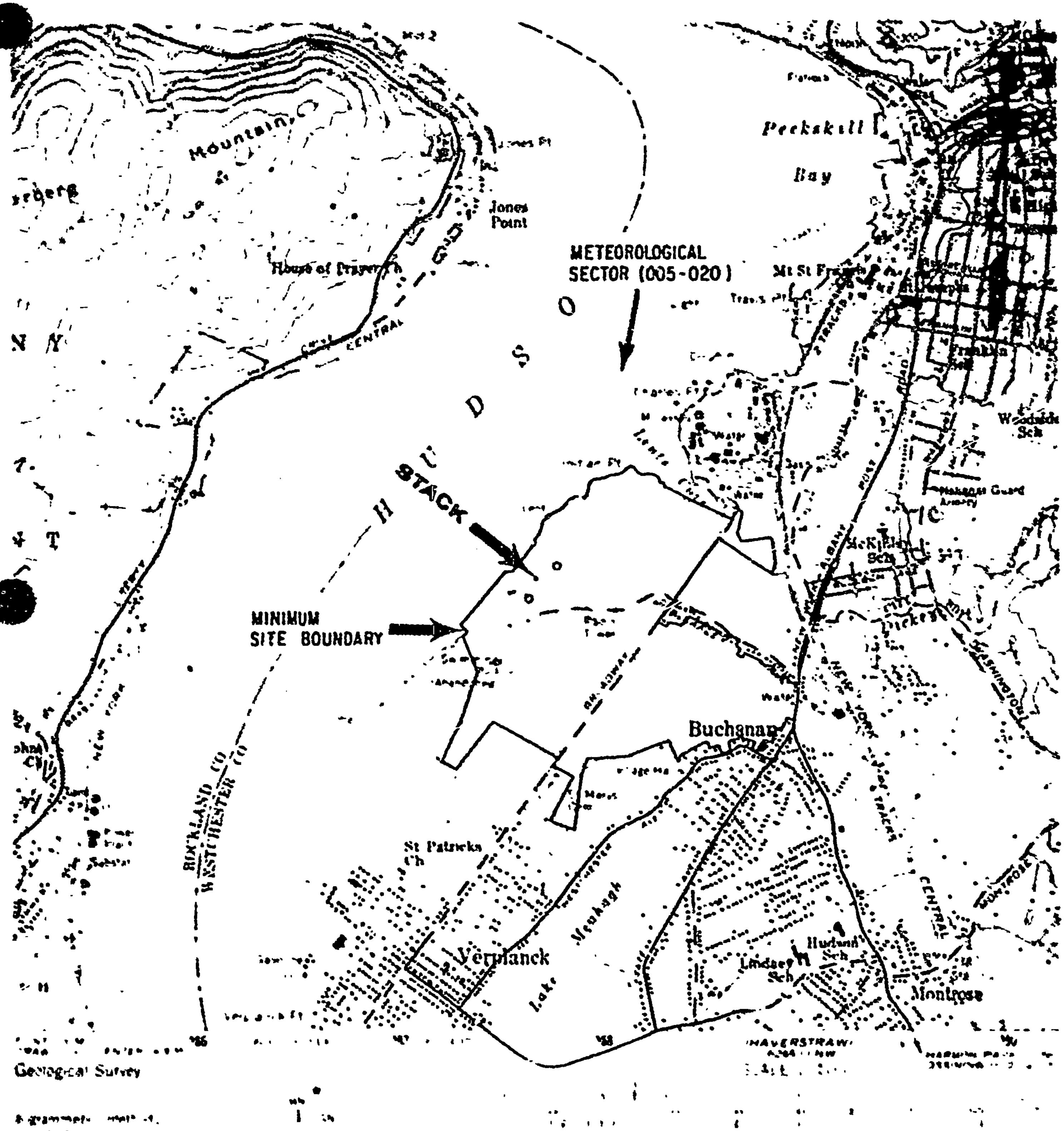


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FIGURE 11

QUESTION 11.2

Based upon the proposed technical specification for primary-to-secondary coolant leak rate and primary coolant activity based upon 1% failed fuel, provide the equilibrium secondary coolant activity on an isotopic basis. Provide all assumptions including secondary coolant cleanup assumptions and include justification for these assumptions.

ANSWER

The activities of significant isotopes are given for the major components of the secondary system in Tables I and II. These values are established by the maximum permissible concentration of I-131 in the discharge canal. These maximum allowable activities can result from various combinations of primary coolant activity and primary to secondary leakage. The maximum value of failed fuel is considered to be 1%. The assumptions used in the calculations of the concentrations and the processes taken into account, with parameters, are given below.

I. Assumptions

1. Iodine partition factor in the steam generators and the blowdown tank is 0.1 $\frac{\text{Amount I/Unit Mass Steam}}{\text{Amount I/Unit Mass Water}}$.

   The iodine partition factor in the condenser is 1×10^{-4}

   $$\frac{\text{Amount I/Unit Vol. Gas}}{\text{Amount I/Unit Vol. Water}}.$$

   REFERENCE:

   "Transfer of iodine from aqueous solutions to saturated vapor" UDC 621.039.562.5.

   M. A. Styrikovich, O. J. Martynova.

   Translated from Atomnaya Energiya. Vol. 17, No. 1, pp 45-49. July 1964.

2. Leakage occurs when the fission product concentrations in the primary coolant are at equilibrium condition.

3. Demineralizer removal effect is constant.

4. Leakage is evenly distributed in all steam generators.

II. Processes that affect the fission products in the primary and secondary systems.

1. Leakage between the primary and the secondary systems.

2. Air ejector release.

3. Steam generator blowdown.

4. Blowdown tank venting.

5. Demineralizer clean up.

6. Gas Stripping.

7. Iodine decontamination (partition).

8. Radioactive decay.

9. Make up water dilution.

10. Boron reduction.

III. Parameters used to calculate the Equil brium Activities in the secondary system.

Water Vol. of Four Steam Generators	V_w = 6,452 Ft^3
Steam Vol. of Four Steam Generators	V_s = 11,864 Ft^3
Water Vol. of Three Condensers	V_L = 15,242 Ft^3
Vol. of Steam and Non-condensibles in Three Condensers	V_A = 120,300 Ft^3
Primary Coolant Vol.	V_p = 10,286 Ft^3
Vol. of Condensate in Blowdown Tank	V_B = 46.3 Ft^3
Weight of Water in Four Steam Generators	W_w = 311,691 Lbs.
Weight of Steam in Four Steam Generators	W_s = 19,475 Lbs.
Weight of Water in Three Condensers	W_v = 944,350 Lbs.
Weight of Condensate in Blowdown Tank	W_B = 2770 Lbs.
Main Steam Flow Rate	q = 1.326×10^7 Lbs/Hr.
Air Ejector Flow Rate (Total)	J = 60 SCFM
Steam Generator Blowdown Rate (Total)	B = 20,000 Lbs/Hr.
Blowdown Tank Venting Rate	P = 7,278 Lbs/Hr.
Blowdown Tank Discharge Rate	D = 12,722 Lbs/Hr.

Boron Concentration in Primary Coolant

(Equilibrium Cycle) B_o = 825 ppm

Boron Dilution Rate (ppm/Full power Day) B' = 1.75 ppm/24hr.

TABLE I - EQUILIBRIUM CONCENTRATIONS CORRESPONDING TO MAXIMUM PERMISSIBLE I-131 CONCENTRATION IN DISCHARGE CANAL

Nuclide	Concentrations - µc/cc				
	C_W	C_S	C_L	C_A	C_B
I131	4.60×10^{-3}	1.55×10^{-5}	6.00×10^{-4}	5.90×10^{-8}	8.80×10^{-3}
I132	2.50×10^{-4}	8.60×10^{-7}	3.30×10^{-5}	3.10×10^{-9}	4.40×10^{-4}
I133	4.50×10^{-3}	1.50×10^{-5}	5.90×10^{-4}	5.80×10^{-8}	8.70×10^{-3}
I134	6.60×10^{-5}	2.30×10^{-7}	8.00×10^{-6}	8.00×10^{-10}	1.05×10^{-4}
I135	1.30×10^{-3}	4.40×10^{-6}	1.60×10^{-4}	1.65×10^{-8}	2.30×10^{-3}
Kr85	negligible	6.60×10^{-7}	negligible	1.50×10^{-3}	negligible
Kr85M	negligible	2.50×10^{-7}	negligible	9.03×10^{-5}	negligible
Kr87	negligible	1.25×10^{-7}	negligible	1.50×10^{-5}	negligible
Kr88	negligible	3.90×10^{-7}	negligible	9.42×10^{-5}	negligible
Xe133	negligible	3.00×10^{-5}	negligible	5.92×10^{-2}	negligible
Xe135	negligible	1.33×10^{-6}	negligible	8.26×10^{-4}	negligible
Xe138	negligible	1.70×10^{-8}	negligible	4.66×10^{-7}	negligible

NOMENCLATURE

C_W = Concentration in steam generator water

C_S = Concentration in main steam

C_L = Concentration in condenser water

C_A = Concentration in condenser gas

C_B = Concentration in the water of blowdown tank

TABLE II - EQUILIBRIUM ACTIVITIES CORRESPONDING TO MAXIMUM PERMISSIBLE I-131 CONCENTRATION IN DISCHARGE CANAL

Nuclide	Activities - Curies				
	A_W	A_S	A_I	A_A	A_B
I131	8.50×10^{-1}	5.40×10^{-3}	2.70×10^{-1}	2.10×10^{-4}	1.10×10^{-2}
I132	4.60×10^{-2}	2.85×10^{-4}	1.35×10^{-2}	1.05×10^{-5}	5.60×10^{-4}
I133	8.40×10^{-1}	5.30×10^{-3}	2.60×10^{-1}	2.00×10^{-4}	1.08 x [illegible]
I134	1.20×10^{-2}	7.50×10^{-5}	3.50×10^{-3}	2.70×10^{-6}	1.35×10^{-4}
I135	2.35×10^{-1}	1.50×10^{-3}	7.20×10^{-2}	5.50×10^{-5}	3.00×10^{-3}
Kr85	negligible	2.25×10^{-4}	negligible	4.89×10^{0}	negligible
Kr85M	negligible	8.40×10^{-5}	negligible	3.05×10^{-1}	negligible
Kr87	negligible	4.10×10^{-5}	negligible	5.06×10^{-2}	negligible
Kr88	negligible	1.30×10^{-4}	negligible	3.23×10^{-1}	negligible
Xe133	negligible	1.00×10^{-2}	negligible	2.00×10^{-2}	negligible
Xe135	negligible	4.40×10^{-4}	negligible	2.80×10^{0}	negligible
Xe138	negligible	5.80×10^{-6}	negligible	1.57×10^{-3}	negligible

NOMENCLATURE

A_W = Activities in the water of four steam generators

A_S = Activities in the steam of four steam generators

A_L = Activities in the water of three condensers

A_A = Activities in the gas of three condensers

A_B = Activities in the water of blowdown tank

QUESTION 11.3

Describe procedures used to properly account for activity released to the environment via the steam generator blowdown. What methods are used to analyze for activity in the secondary (steam) system and what is the frequency and sensitivity of this analysis?

ANSWER

The steam generator blowdown system is operated intermittently and only when the boiler water begins to deviate from chemical specifications. At that time, a blowdown rate is set and the steam generators are blown down long enough to return the system to normal chemistry specifications.

Paralleling the blowdown system is the steam generator sample system. This system has its own isolation valves as it emerges from the Containment Building and its own isolation seal water supply. The sample line can branch into three streams:

a. A stream to the sample sink in the Primary Auxiliary Building.

b. A stream through a continuously running conductivity element.

c. A composite stream through a continuously running radiation monitor.

The conductivity analyzers continuously monitor the boiler water to determine if it is within the specifications. The radiation monitor determines that there is no in-leakage of radioactivity through a leaking stream generator tube. In the event of a high radiation alarm, all of the automatic valves at the penetrations for both the blowdown leads and the sample leads are closed. This prevents any outleakage of radioactive water into the river.

The blowdown leads are designed such that the time to carry a sample through the continuous radiation monitor is shorter than that required for the blowdown water to enter the blowdown tank. For this reason, if there is a high-radiation alarm, the isolation valve on the blowdown and sampling systems will close prior to entry of radioactivity into the blowdown tank.

A control switch is provided in the sample room to override the radiation trip for the sample leads. Samples can be taken in the sample room on a selective basis to determine which steam generator is leaking.

The steam generators can also be drained through the blowdown leads, since each individual lead is free draining to the blowdown tank, after which it can be discharged to the river or to the waste disposal system. Should a steam generator become contaminated, it can be drained into the blowdown tank and from there into the waste disposal system by way of a line that connects the blowdown tank to the sump tank in the Primary Auxiliary Building. From the sump tank, the contaminated blowdown is pumped to the waste holdup tank and from there it can be processed through the waste disposal system.

During blowdown operations with activity in the steam generators, but at levels below the set point of the radiation monitor in the sampling system, low levels of activity will be released directly to the river. The specific composition of the activity can be determined by analysis. Instrumentation is not provided to measure the blowdown rate directly. Releases will comply with Criterion 3 in the response to Question 11.1.

Radio-analysis for activity in the steam generator blowdown system is done on a weekly basis. Samples are collected and evaporated in planchets for counting. A spectral and chemical analysis is performed on composite samples at the end of each month.

The sampling frequency is dependent on the magnitude of the primary to secondary leak rate and the variations in the leak rate. A 400-channel analyzer is used in conjunction with a three-by-three sodium iodine crystal. Instrument sensitivity is 5×10^{-8} micro-curies per milliliter for gaseous and liquid samples.

QUESTION 11.4

Provide a diagram of the liquid and gaseous radwaste system which indicates the locations where grab samples are taken and continuous monitoring is done in this system. Provide the grap sampling frequencies as related to radwaste releases, and describe the analyses to be performed on these samples.

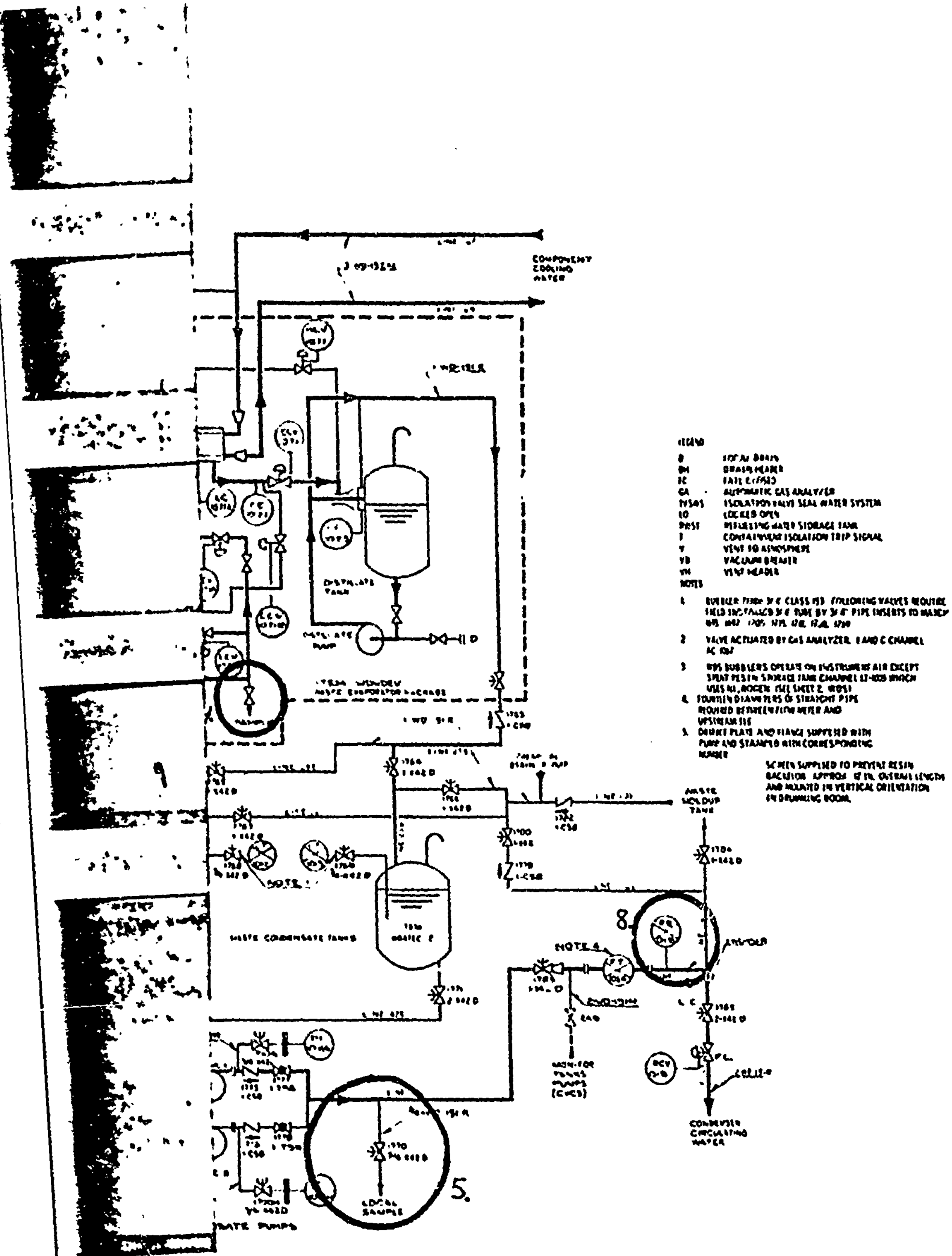
ANSWER

Figures 11.4-1 and 11.4-2 (liquid and gaseous waste disposal system) indicate the locations of the continuous monitoring and grab sample points. Table 11.4-1 provides the grab sample frequency, types of analysis performed, and the purpose of the analysis.

TABLE 11.4-1

	Sample Point	Frequency	Analyses Performed and Purpose
1.	Regenerant Tank	As needed, before resin regeneration	NaOH concentration - to ensure proper concentration for resin regeneration
2.	Chemical Drain Tank	As needed, before emptying (only when being pumped directly to the waste condensate tanks)	Total solids concentration and activity - to guide operation of the WDS
3.	Waste Holdup Tank	As needed, before evaporator processing	Total solids concentration and activity - to guide operation of WDS
4.	Waste Evaporator	As needed	Total solids concentration and activity - to evaluate evaporator performance and to guide operation of WDS
5.	Waste Condensate Tanks	As needed, before release of waste liquids	Total solids concentration and activity, to determine if tank contents are suitable for release and to ensure compliance with Tech. Spec.
6.	Gas Decay Tanks	Automatic, part of normal gas analyzer cycle	H_2, O_2, activity level - to supply information to allow isolation of tank if either O_2 or activity level are above specified limits and to ensure compliance with Tech. Spec.
7.	Gas Decay Tanks	As needed, before release of waste gas	Activity level - to determine if tank contents are suitable for release and to ensure compliance with Tech. Spec. Isotopic content measurement.
8.	Waste Liquid Overboard Line	Continuous	Gross activity - automatic isolation if activity level exceeds specified limit
9.	Waste Gas Overboard Line	Continuous	Gross activity - automatic isolation if activity level exceeds specified limit
10.	Demineralizer Regeneration Water Return	Continuous, whenever regenerating	Conductivity - to follow progress of regeneration

See Section 3.9 of Tech. Spec. for limits



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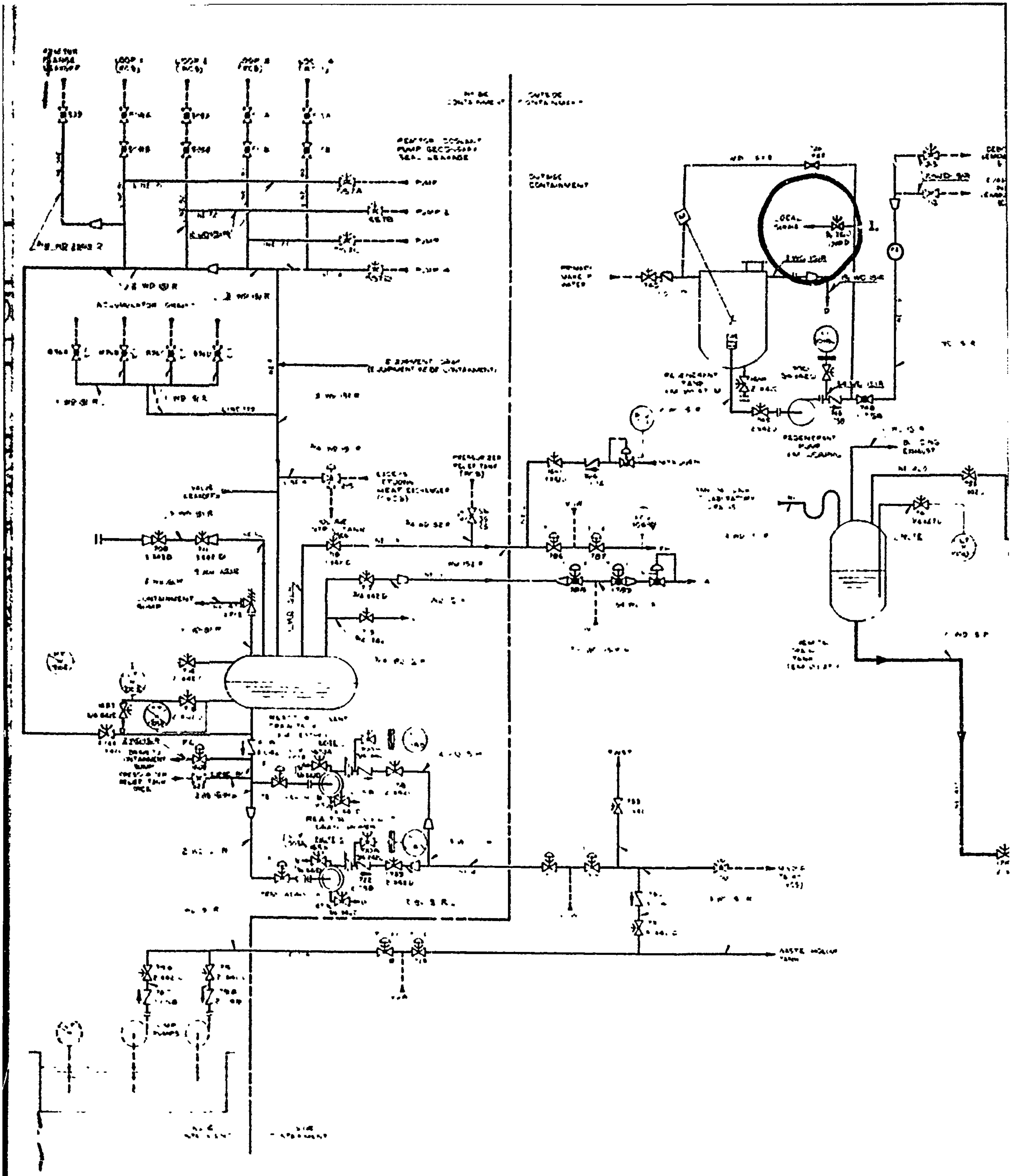
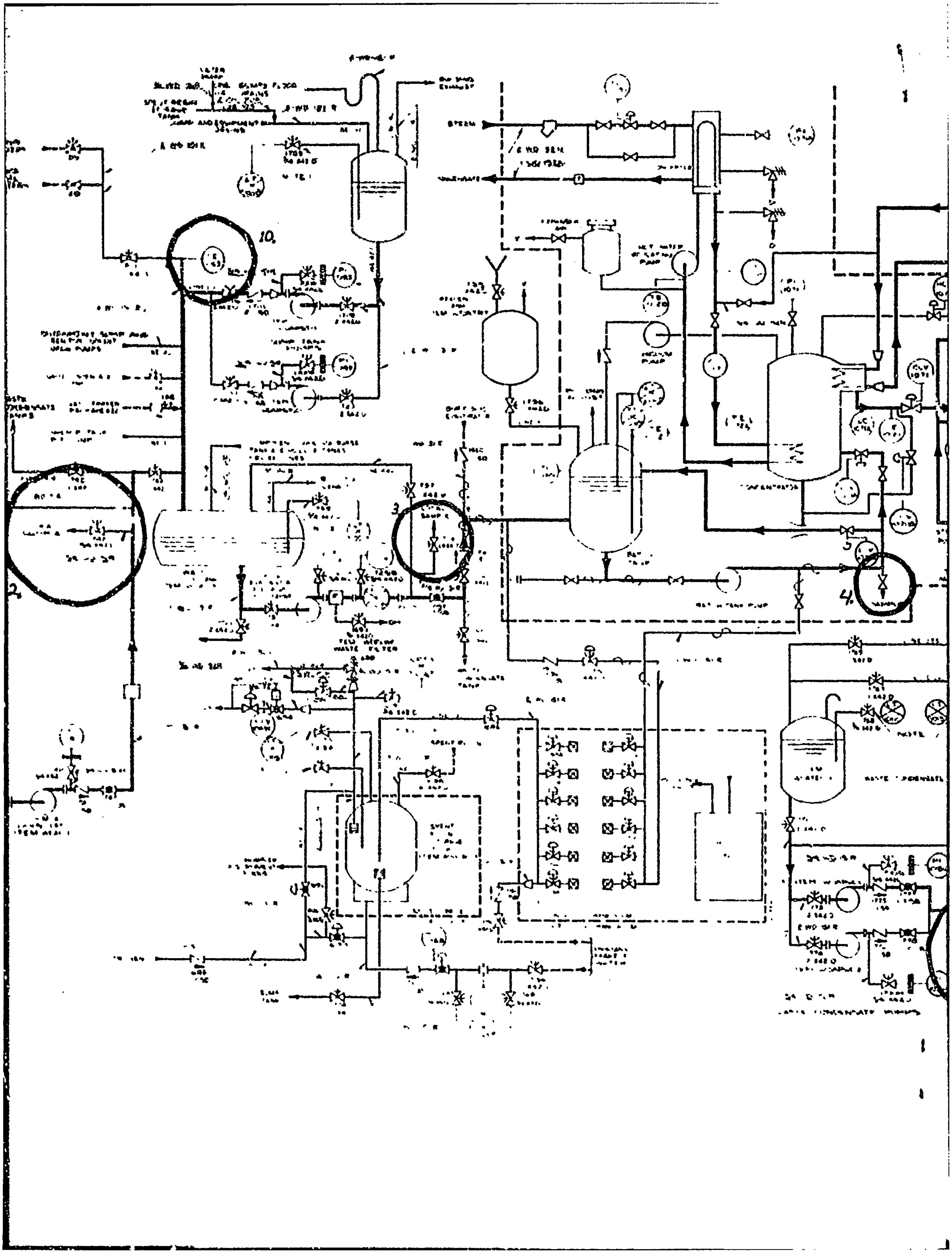
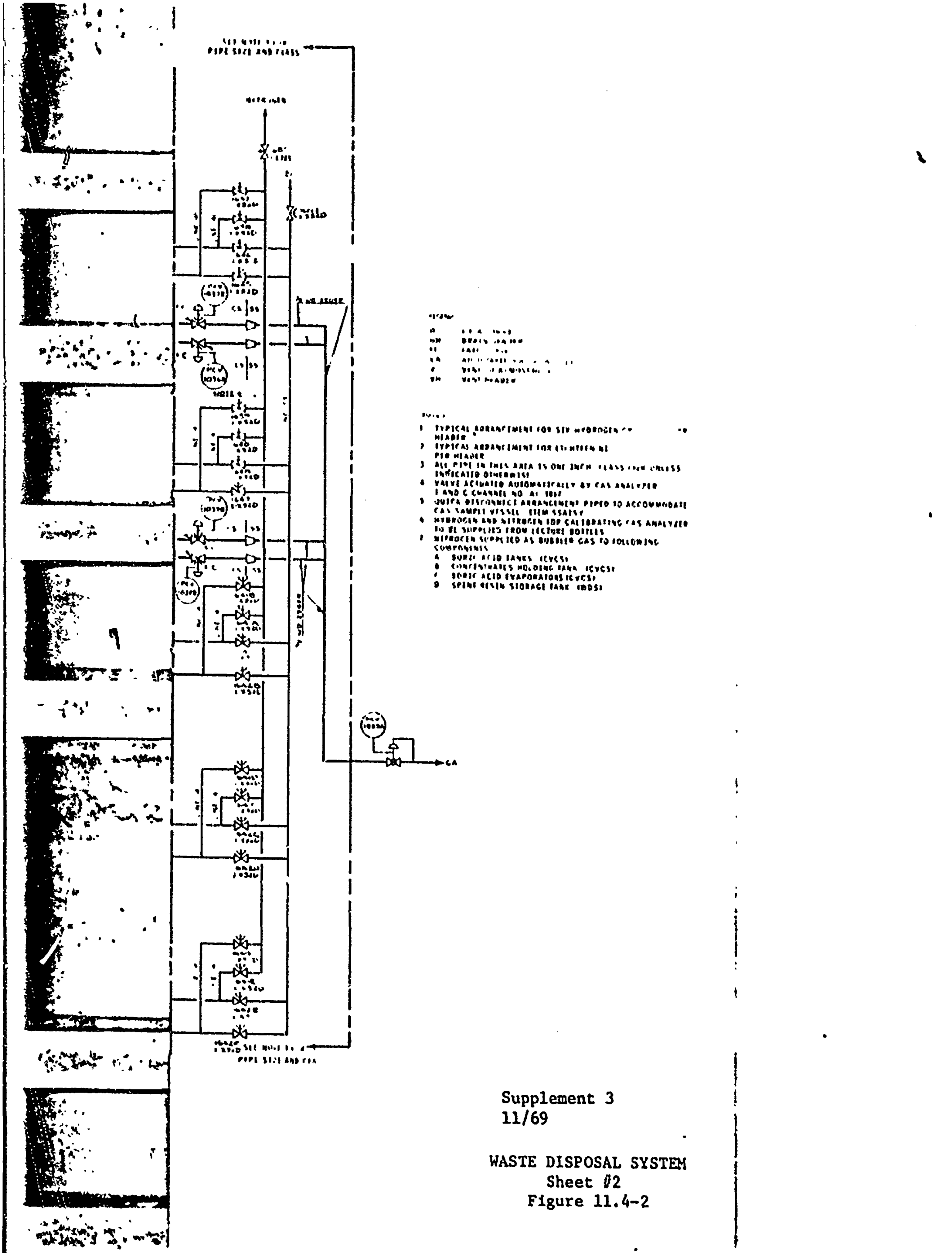
WASTE DISPOSAL SYSTEM
Sheet #1
Figure 11.4-1




STEAM
10.
2.
3
4.
5



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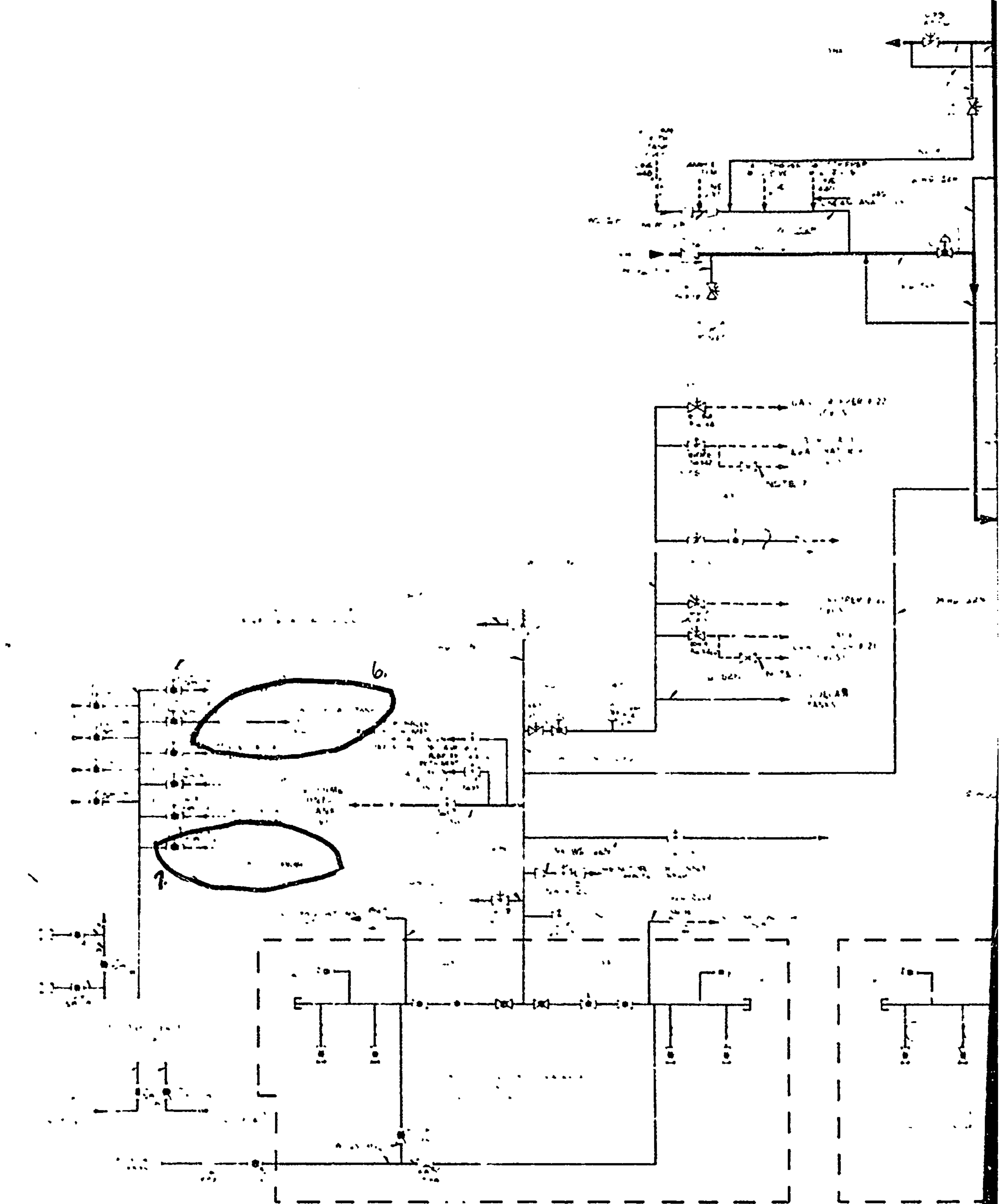
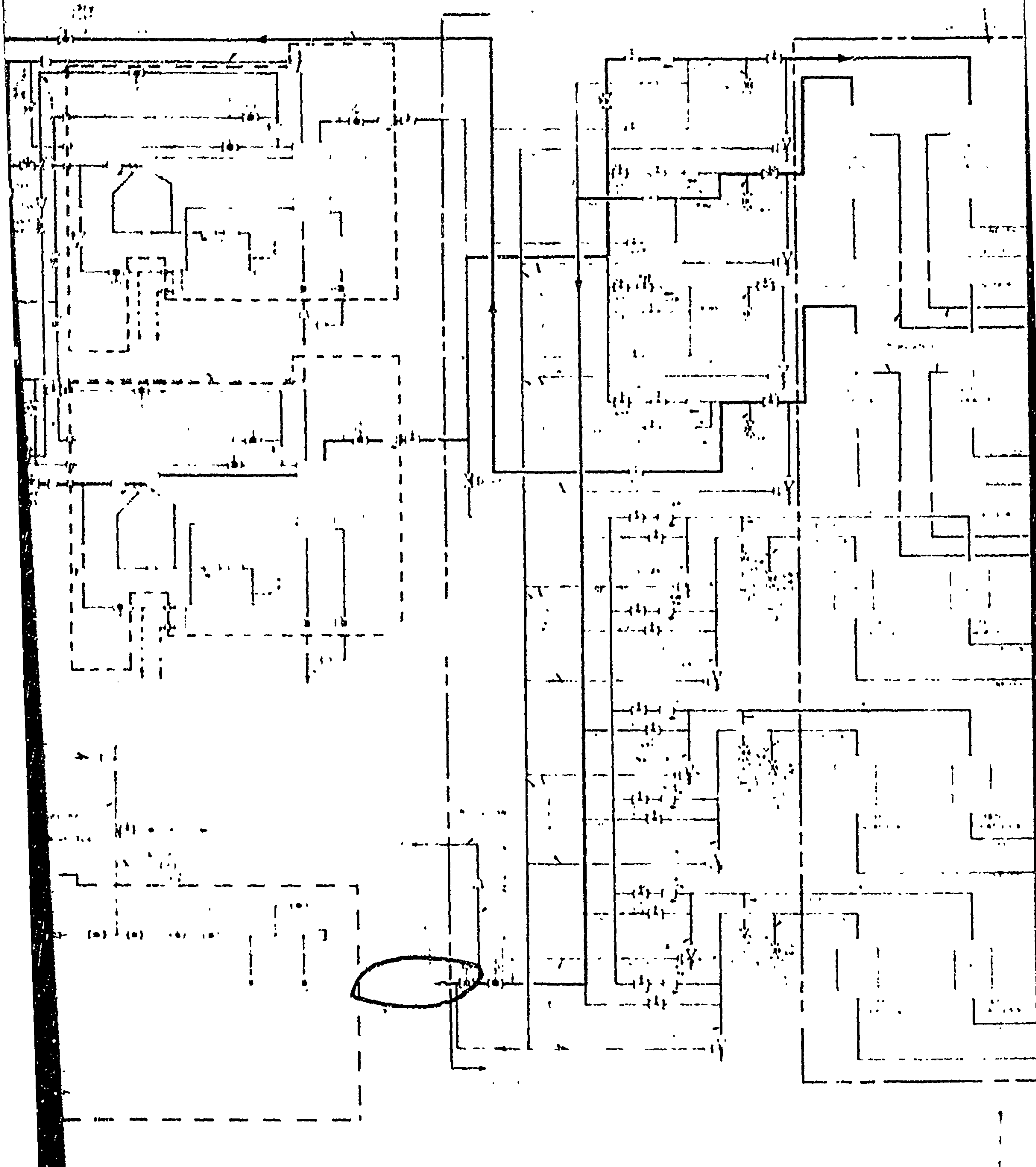
WASTE DISPOSAL SYSTEM
Sheet #2
Figure 11.4-2





QUESTION 11.5

The technical specifications for this facility will have to separate liquid discharge limits - one for tritium and one for all the other radioactivity in the liquid wastes. We understand that it is not your intent to sample for tritium in the liquid radwaste system on a batch by batch basis prior to release from this facility. If this is so, provide the basis by which compliance with the technical specifications will be demonstrated.

ANSWER

Analysis for tritium concentrations prior to each batch release is not necessary providing the following criterion can be met. Note 5 of 10CFR20, Appendix B, stipulates "A radionuclide may be considered as not present if a. the ratio of the concentration of that radionuclide in the mixture (C_A) to the concentration - limit for that radionuclide specified in Table II of Appendix B does not exceed 1/10, (i.e., $C_A/MPC_A \leq 1/10$)...

The MPC for tritium from 10CFR20, Table II, Apendix B for unrestricted areas is 3×10^{-3} microcuries per milliliter. To consider the radionuclide as not present, the tritium concentration shall not exceed 1/10 MPC for tritium or 3×10^{-4} microcuries per milliliter.

The Technical Specifications for Unit No. 2 maintain at least one circulating water pump will be operating during any liquid radwaste discharge. The total flow for one month with one circulating water pump operating is 2.3×10^{13} milliliters. The total amount of tritium which may be released during one month and considered as not present is 6,900 curies, which is greater than the yearly tritium inventory of the primary coolant. Since 6,900 curies of tritium could not be released during any predetermined sampling period within the year, sampling each batch for tritium concentration prior to discharge is not necessary.

The tritium concentration in a composite sample taken away from every batch discharged to the river will be determined periodically and used to establish the quantity of tritium released during that period.

QUESTION 11.6

We understand that the discharge canal is sampled for radioactivity. Provide the details of the method of sampling, frequency of sampling, analyses performed on these samples, and the sensitivity of the analyses performed.

ANSWER

Sampling of the condenser inlet water and discharge water system is done continuously.

The condenser inlet sampling system consists of two lines (1" diameter), each with a flow rate of approximately 5 gpm. The lines enter a common mixing header from which a portion is continuously extracted to a 50 gallon sample container. The sample flow extraction is regulated to fill the 50 gallon container in seven days. Upon completion of the sampling period of seven days, the mixture in the sample container is stirred to obtain uniformity of distribution. Three quarts of sample are removed and used for analysis. Aliquots of the sample are prepared for gross beta-gamma analysis. Tritium concentrations are determined from monthly composite samples.

The condenser discharge sampling system consists of four lines (1 1/4" diameter), each with a flow rate of approximately 5 gpm. The four lines are positioned at different locations and levels in the discharge canal. This sampling system is identical to the inlet sampling system with the above exceptions. Analysis for activity in the discharge water is identical to that of the inlet system.

The sampling program will be conducted in accordance with the following schedule:

a. If the gross beta-gamma activity of the station releases to the river is less than 1% of MPC during the month just ended, the environmental survey will be conducted in accordance with Program I for the subsequent month.

b. If the gross beta-gamma activity of the station releases to the river is greater than 1% of MPC but less than 10% of MPC during the month just ended, the environmental survey will be conducted in accordance with Program 2 for the subsequent month. If the samples taken under Program 2 do not indicate any significant increase in environmental radioactivity, the survey will revert to Program 1.

c. If the gross beta-gamma activity of the station releases to the river is greater than 10% of MPC during the month just ended, the environmental survey will be conducted in accordance with Program 3 for the subsequent month. If the samples taken under Program 3 do not indicate any significant increase in environmental radioactivity, the survey will revert to Program 2.

PROGRAMS

	1		2		3	
Media	Sample Frequency	Analysis	Sample Frequency	Analysis	Sample Frequency	Analysis
Hudson River Water (Condenser Inlet and Discharge)	W	GBG	TW	GBG	D	GBG
	MC	T		GSA		GSA
			MC	T	MC	RA
						T

Nomenclature for Sample Frequency

W - Weekly
TW - Twice Weekly
D - Daily
MC - Monthly

Nomenclature for Analysis

GBG - Gross Beta-Gamma
GSA - Gamma Spectrometer Analysis
RA - Radiochemical Analysis to Determine Biologically Important Isotopes
T - Tritium

Minimum sensitivity is 1 picocurie per liter for gross beta-gamma
Minimum sensitivity is 3000 picocuries per liter for tritium

QUESTION 11.7

The technical specifications for this facility will contain two separate limits for gaseous effluents, one for iodine and particulates with half-lives longer than eight days and one for all the other airborne radioactivity releases. We understand that it is not your intent to continuously monitor for iodines and particulates on the plant vent. Describe how compliance with the halogen and particulate technical specification gaseous release limits will be demonstrated without this monitoring capability.

ANSWER

For normal operation

The Technical Specifications for Indian Point Unit No. 2 do not have two separate limits for gaseous effluents, one for iodine and particulate with half-lives longer than eight days, and one for all the other airborne activity. The combined containment air particulate and radiogas monitor (R-11 and R-12) is used continuously for monitoring the containment atmosphere. These monitors can be switched to the plant vent stack to monitor for particulate activity. The system design incorporates a continuous on-line plant vent gas monitor (channel R-14). The stack gas monitor is used to insure compliance with the allowable release rates of gaseous activity including halogens and particulates. The particulate fission product activity is related to the noble gas activity which indicates when the air particulate monitor should be switched to sampling the vent for particulate activity. In practice, if the stack radiogas monitor were to approach 10% of the alarm set point, the air particulate monitor will be transferred to sampling the vent. Grab samples will be taken and analyzed if the particulate monitor indicates that the fission product particulate activity is approaching the maximum allowable concentrations.

(See Section 11.2 of the FSAR for further details of the referenced radiation monitors)

QUESTION 11.8

We understand that the monitor tanks are Class II and that if they were to fail their contents would spill uncontained on the ground and then into the river. What is the maximum amount of radioactivity that could be contained within these tanks? Provide an analysis of the river water concentrations that would result from the failure of these tanks. State all assumptions and justify them.

ANSWER

In dealing with the question concerning the possible activity in the Monitor Tanks, it is necessary to consider the processing to which the reactor coolant is subjected to before it passes into the Monitor Tanks.

The purpose of the recycle portion of the CVCS is to accept and process all effluents which can be readily reused as make-up to the reactor coolant system. Effluents are initially collected in the CVCS hold-up tanks. Prior to the hold-up tanks, particularly if the reactor is operating with defect fuel, the let-down from the reactor coolant system is passed through the mixed bed demineralizers. Both forms of resin remove fission products and corrosion products. As fluid enters the hold-up tanks, released gases (hydrogen and fission gases) mix with the nitrogen and are eventually drawn off to the waste gas system.

Three gas-stripper-feed pumps take suction from the holdup tanks and pump the fluid to the gas-stripper units. Initially, the fluid passes through the evaporator feed ion exchangers where lithium and fission products (primarily cesium isotopes) are removed. The resin is a hydrogen form cation resin, with a total capacity sufficient for one equilibrium core cycle assuming load-follow operation and one percent fuel defects. Two ion exchangers are employed in series for each evaporator train. Series operation is recommended to ensure prevention of breakthrough of cesium in the event of operation with one percent defects. If defects are small, one exchanger can be employed per train.

From the feed ion exchangers the fluid flows through the ion-exchange filter and a flow control valve into the gas-stripper package, where hydrogen, nitrogen and residual fission gases are removed.

The fluid from the gas stripper flows to the boric acid evaporator. Concentrated boric acid is produced in the evaporator bottoms while vapor condensate is fed to the evaporator condensate demineralizers. Before the contents in the boric acid evaporator feed tank are transferred, a sample is taken and analysis made for boron and activity level.

Distillate from the evaporator flows continuously to the monitor tanks via the evaporator condensate demineralizers (employed only in the event of excessive carryover of boron in the distillate) and the condensate filter. The contents of the monitor tanks are sampled and evaluated before being pumped by the monitor tank pumps to the primary storage tank. If the water is outside the primary water specification, the fluid is either recycled to the holdup tanks or the condensate demineralizers or disposed of via the waste disposal discharge to the circulating water outfall.

During operation of the recycle process, samples may be taken at various positions through the system in order to assess the performance of the individual system components. Local samples may be obtained from:

(a) Before and after the Evaporator feed ion exchangers.

(b) Gas samples may be taken from the gas strippers via the gas analyzer.

(c) Before and after the Evaporator condensate demineralizers.

(d) At each monitor tank.

(e) At the monitor tank pump discharge header.

(f) The evaporator bottoms are checked for activity before the concentrated boric acid is transferred to the concentrates holding tank.

The design constants of the system are as follows:

Mixed be demineralizers	Min. DF 10
Feed ion exchangers (two in series)	Min. DF 10
Feed ion exchange filters	25 microns
Gas stripper	DF 10^5
Boric acid evaporator	DF 10^6
Boric acid evaporator condensate demineralizer	Min. DF 10
Condensate filter	25 microns

Assuming only that the boric acid evaporator DF of 10^6 and the basic acid evaporator condensate demineralizer DF of 10 for iodine is achieved, the maximum specific activities during operation with 1% failed fuel are as follows:

(DF = 10^6 across evaporator)

Cs-134 = 2.8×10^{-6} μc/cc

Cs-137 = 1.65×10^{-5} μc/cc

(DF = 10^6 across evaporator and DF = 10 across demineralizer)

Mo-99 = 1.4×10^{-5} μc/cc

I-131 = 7.7×10^{-6} μc/cc

I-133 = 3.83×10^{-6} μc/cc

I-135 = 5.5×10^{-7} μc/cc

The isotopes listed above represent those isotopes which contribute significently to the anticipated liquid releases. Other isotopes are at least a factor of 10 lower then those considered above.

The total volume of a monitor tank is 7,500 gallons.

Using this data, the total curies released for each isotope are determined.

Cs - 134	79uc
Cs - 137	468uc
I - 131	219uc
I - 133	109uc
I - 135	16uc
MO - 99	398uc
Total	1289uc

The dispersion in the river of burst releases has been discussed in detail in the response to question 11.9. The peak concentration for each of these isotopes at any point in the river upstram of Indian Point can be determined from Figure 3 of the response to question 11.9. The peak concentration expressed as a fraction of MPC at the point of interest, viz., Chelsea are presented below for the drought conditions stated in 11.9 (3500CFS)

Isotope	Fraction of MPC @ Chelsea
Cs - 134	2.2×10^{-9}
Cs - 137	5.7×10^{-9}
I - 131	6.0×10^{-8}
I - 133	1.8×10^{-10}
I - 135	5.0×10^{-14}
MO - 99	3.4×10^{-11}

Question 11.9

What minimum river water dilution factors are available between the Indian Point site and the nearest public drinking water intakes on the Hudson River (at the Veterans Administration Hospital and the Chelsea pumping station) for both continuous routine release and accident slug releases? State all assumptions and justify them.

Answer

The analytical techniques used to analyze the dispersion of continuous and burst releases of liquids is discussed in detail in "Transport of Contaminants in the Hudson River above Indian Point Station," which is appended to Section 2.5.

There are two potential sources of drinking water in the Hudson River, namely, New York City's Chelsea Pumping Station and the Castle Point Veteran's Hospital. The city of New York's Chelsea Pumping Station is located about one mile north of Chelsea, New York, on the east bank of the Hudson River. The pumping station is 22 miles upriver from Indian Point measured along the centerline of the river. The Castle Point Veteran's Hospital is a relatively small intake located approximately 21 miles upriver from the proposed site.

Analyses have been conducted to determine the difference in concentration at Chelsea and Castle Point Veteran's Hospital. The difference in concentration is small; hence, the discussion of one potential intake, viz, Chelsea, is sufficient. This is shown for continuous releases in Figure 9 of the response to Question 11.1 and for burst releases in Figure 1 of the response to this questio

The River drought conditions analyzed have been characterized in terms o salinity because the operation of the Chelsea Station is dependent on t level of salt at the station. Consider the following five drought conditions, ie., salinities at Chelsea:

Salt Concentration in ppm		Runoff	Dispersion Coefficient
At Chelsea	At Indian Point	(cfs)	(Sq. Miles/day)
200	2300	5000	5.24
300	2800	4600	5.28
500	4000	4400	5.43
1000	5500	4000	6.00
2000	7000	3500	7.16

The first two drought conditions correspond to concentrations of salinity at Chelsea, at which the New York City Department of Water Resources would begin to be concerned about using Chelsea for New York City's water supply.

The third condition, a salinity of 550 ppm, corresponds to the "mid-thousand" level, which might constitute the maximum level at which Chelsea operation would be stopped. This also corresponds to the Public Health Service drinking water standard for total dissolved solids.

The fourth condition, a salinity of 1,000 ppm represents the maximum level at which Chelsea operation would be stopped.

The fifth condition, a salinity of 2,000 ppm, corresponds to the highest levels of salinity known to have occurred at Chelsea and represents the most conservative River conditions used in this analyses. This concentration of salinity at Chelsea was reached in late November 1964 at the end of six months of Hudson River low flows. Support that the 1964 drought was the worst of record after regulation of the Hudson River is given in a recent report concerning the potential of the Hudson River supplementing New York City's water supply system.*

The upstream movement of salt is the result of a rather delicate balance which is struck between the salinity-induced density currents, which tend to drive the salt itself up the estuary and fresh water flow, which tends to hold back the salt movement. The River's dispersion characteristics are strongly influenced by this phenomenon, so that salinity profiles become the chief means of estimating the longitudinal dispersion coefficient in the River.

Calculation of dispersion coefficients requires a knowledge of the salinity changes between two fixed points and the River's flow. The essential point, however, is that the behavior of a conservative substance is identical to the salt behavior which is well-defined; hence, the salinity at Chelsea is an excellent indicator of the upstream movement of any pollutant introduced to the River below the station. This is explained as follows:

a. If salt is not present at Chelsea, then neither will any other pollutant, discharged many miles below Chelsea, be present at Chelsea.

b. When salt is present at Chelsea, the ratio between the salt concentrations at Indian Point and Chelsea is a measure of the "mechanical dilution", i.e., dilution due to the River's flow and dispersion characteristics for non-decaying pollutants.

* "Comprehensive Public Water Supply Study for the New York City of New York and County of Westchester" - Report CPWS-17 submitted by Metcalf & Eddy, Hazen & Sawyer, and Malcolm Pirnie Engineers to the New York State Department of Health, August 1967.

Hence, for the five drought conditions cited above, the mechanical dilution factors between Indian Point Station and Chelsea may be obtained directly from the ratio of salinity at these two points and are as follows:

Runoff (cfs)	Mechanical Dilution
5000	11.5
4600	9.4
4400	8.0
4000	5.5
3500	3.5

To obtain the concentrations of decaying radionuclides at Chelsea, simple ratios of the salt concentrations at Indian Point and Chelsea are not used. Rather, a material balance on each isotope is struck over any segment of the River by considering the transport mechanisms of net flow and longitudinal dispersion, and the radioactive decay mechanism. The longitudinal dispersion coefficient is obtained from salt profiles. The approach is described in the reference cited above in Section 2.5.

To show how the significant parameters, viz, the salinity and the half life affect the River's ability to reduce concentration of introduced pollutants, a study was made assuming a normalized continuous release rate for each isotope of one curie/day and a normalized burst release for each isotope for one curie. Since the concentrations at Chelsea are directly proportional to the source term, the normalized curves can be used to quickly determine the concentration at Chelsea due to a known burst or continuous release from Indian Point, or to determine dilution factors.

Continuous Release

The results for the normalized continuous release study are shown in Figures 9, 10 and 11 of the response to Question 11.1. From these figures, the minimum dilution factors for the severe drought condition (2,000 ppm) and for various isotopes can be readily determined, e.g.,

Isotope	Half Life	Mechanical Dilution	River Dilution Factor
Sr-90	28.8 y	3.5	3.5
I-131	8.05 d	3.5	23
La-140	40.2 h	3.5	384
I-132	2.3 h	3.5	7×10^9

Where the River Dilution Factor is the ratio of isotopic concentrations at Indian Point to that at Chelsea. From this it is seen that for long-lived isotopes, the mechanical dilution well represents the river dilution; for very short-lived isotopes the isotopes do not reach Chelsea, and for intermediate-lived isotopes, the River dilution factors are one to two orders of magnitude greater than the mechanical dilution factor.

A hypothetical case where primary coolant with 1% failed fuel being release directly to the discharge canal was considered so that the behavior of all isotopes of possible concern in the River could be presented. The activity is released at a constant rate, the value of which is set so that the MPC of the mix will not be exceeded in the discharge water.

The most severe drought conditions have been utilized; for the continuous release, these consist of a long-term steady upstream runoff of 3,500 cfs, which causes the salt concentration at Chelsea to reach 2,000 ppm.

Other pertinent River parameters used in the analysis are:

Longitudinal dispersion coefficient, "E" = 7.16 square mile/day.

Average cross-sectional area, "A" = 140,000 square feet.

The results of this analysis are presented in Table 1 and the computational procedure follows:

Column 1 - Unit No. 3 PSAR, Column 2, Part B, Table 16 (E-3.1).

Column 2 - 0.693 divided by half life in days.

Column 3 - Allowable release rate based on MPC of mix in discharge canal.

Column 4 through 7 - Computation procedure for continous release, QL&M report to Con Edison on Chelsea concentrations (May 1966), and included in both Unit Nos. 2 and 3 submittals. (Analyses appended to Section 2.5 of Indian Point Unit 2 FSAR).

Column 8 - Concentration at Chelsea divided by concentration at Indian Point.

The minimum dilution factors for all isotopes of concern are given in Column 8 of Table 1.

For the effect of all three units at Indian Point releasing radioactivity to the River under the conditions described above, the corresponding Chelsea and Indian Point concentrations can be computed by multiplying the concentrations in these tables by 1,960,000/840,000 or 2.34, the ratio of the total condenser flow to the Unit Nos. 2 or 3 condenser flow. This assumes that the mix distribution from each unit is the same.

Burst Release

The results of the normalized burst release studies are presented in Figures 1, 2, 3, 4 and 5. They are based on a one curie burst release of each isotope. The following conclusions can be reached from these figures.

1. Referring to Figure 1, the peak concentrations at Chelsea and Castle Point are for the purpose of this discussion essentially the same.

2. Referring to Figure 2, variations in drought conditions, i.e., changes in low runoff values do not appreciably affect the peak concentrations at Chelsea.

3. Referring to Figure 5, the runoff does not appreciably affect the time for an isotope to reach a peak concentration at Chelsea; the time to the peak is a weak function of half life for isotopes with half lives less than 100 days, and the time to the peak is not sensitive to half-life for isotopes with half lives greater than 100 days.

4. Referring to Figures 3 and 4, short-lived (less than one day) isotopes will not reach Chelsea; peak concentrations of intermediate isotopes (one day to 100 days) are strongly dependent on the half life.

The River dilution factor between Indian Point and Chelsea for the burst release is a non-applicable concept. When the maximum radioactivity effect of each isotope occurs at Chelsea, the corresponding concentration of that isotope at Indian Point will be very low. Furthermore, Chelsea will not see the maximum concentration of each isotope at the same time.

For these reasons, for the burst release, the concentration in the Hudson River are considered for Indian Point one-half day after the release and at Chelsea at the time when the concentration of the given isotope is maximum at that point. Zero time cannot be used at Indian Point because the equations used will yield infinity for the concentration at $x = 0$, $t = 0$. One-half day later was used because this corresponds to one tidal cycle, the minimum time necessary to provide the River mixing which these equations presume.

Based on the above definition of dilution factor for the burst release, the minimum dilution factors for the burst release were determined for the drought condition resulting in 2,000 ppm of salt at Chelsea. The hypothetical case where the entire primary coolant with fission product inventory due to operation with 1% failed fuel was dumped into the River was used to arrive at the dilution factors for all isotopes of concern. The results of this analysis are given in Table 2 and the computational procedure is as follows:

> Columns 1 and 2 - Taken from Table 9.2-5 (Unit No. 3 PSAR) entitled "Reactor Coolant System Equilibrium Activities: and computed using a primary coolant volume of 3.56×10^8 ml. Tritium activity of 890 curies added later.

Columns 3 through 7 - Computation procedure for accidental release, QL&M report to Edison on Chelsea, May 1966, and included in Unit Nos. 2 and 3 submittals. (As appended to Section 2.5 of Indian Point Unit No. 2 FSAR.)

Column 8 - Based on burst release dilution factor definition cited above.

TABLE 1

CONCENTRATIONS OF PRIMARY COOLANT ISOTOPES IN THE HUDSON RIVER AT INDIAN POINT AND CHELSEA.

Hypothetical Continuous Release, 1% Failed Fuel
MPC In Discharge Canal

(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)
			Behavior At				
			Indian Point		Chelsea		
Isotope	Decay Rate Day-1	Discharge Rate μc/day	Conc. μc/ml	Fract. of mpc	Conc. μc/ml	Fract. of mpc	River Dilution Between I.P. & C.
Mn 54	2.3×10^{-3}	1.54×10^{2}	15.25×10^{-12}	1.5×10^{-7}	3.99×10^{-12}	3.99×10^{-8}	3.82
Mn 56	6.3	3.33×10^{4}	118.5×10^{-12}	1.2×10^{-6}	5.5×10^{-20}	5.5×10^{-16}	2.16×10^{9}
Co 58	0.97×10^{-2}	4.62×10^{3}	332×10^{-12}	3.3×10^{-6}	6.35×10^{-11}	6.35×10^{-7}	5.22
Fe 59	1.5×10^{-2}	1.07×10^{2}	6.77×10^{-12}	1.1×10^{-7}	1.05×10^{-12}	1.75×10^{-8}	6.45
Co 69	3.6×10^{-4}	5.45×10^{2}	61.8×10^{-12}	1.2×10^{-6}	1.73×10^{-11}	3.45×10^{-7}	3.58
Br 84	3.15×10^{-3}	1.62×10^{4}	1530×10^{-12}	-	-	-	-
Rb 88	5.6×10^{-3}	1.54×10^{6}	1.28×10^{-7}	-	-	-	-
Rb 89	6.48×10^{-3}	3.56×10^{4}	2870×10^{-12}	-	-	-	-
Sr 89	1.37×10^{-2}	1.20×10^{3}	76.4×10^{-12}	2.5×10^{-5}	1.25×10^{-11}	4.28×10^{-6}	6.11
Sr 90	0.69×10^{-4}	0.81×10^{2}	9.35×10^{-12}	3.1×10^{-5}	2.68×10^{-12}	8.92×10^{-6}	3.49
Y 90	2.6×10^{-1}	1.66×10^{2}	2.88×10^{-12}	1.4×10^{-7}	2.24×10^{-14}	1.12×10^{-9}	352
Sr 91	1.73	7.82×10^{2}	5.32×10^{-12}	0.8×10^{-7}	6.1×10^{-17}	8.70×10^{-13}	8.72×10^{4}
Y 91	1.2×10^{-2}	3.56×10^{2}	23.9×10^{-12}	8×10^{-7}	4.27×10^{-12}	1.34×10^{-7}	5.60

(Continued)

TABLE 1
(Continued)

Isotope	Decay Rate Day-1	Discharge Rate μc/day	Behavior At Indian Point: Conc μc/ml	Behavior At Indian Point: Fract. of mpc	Behavior At Chelsea: Conc. μc/ml	Behavior At Chelsea: Fract. of mpc	River Dilution Between I.P. & C.
Mo 99	2.5×10^{-1}	1.96×10^{6}	3.47×10^{-8}	1.7×10^{-4}	2.84×10^{-10}	1.42×10^{-6}	122
I 131	8.62×10^{-2}	1.04×10^{6}	3.07×10^{-8}	1×10^{-1}	1.35×10^{-9}	4.5×10^{-3}	22.7
Te 132	0.9×10^{-2}	1.10×10^{5}	8.08×10^{-9}	2.7×10^{-4}	2.38×10^{-12}	7.94×10^{-7}	3400
I 132	7.2	3.56×10^{5}	1.18×10^{-9}	1.5×10^{-4}	1.63×10^{-19}	2.03×10^{-14}	7.25×10^{9}
I 133	0.81	8.05×10^{5}	7.97×10^{-9}	8×10^{-3}	2.82×10^{-12}	2.82×10^{-6}	2830
Te 134	23	1.16×10^{4}	21.6×10^{-12}	-	-	-	-
I 134	19	2.12×10^{5}	4.34×10^{-10}	2.2×10^{-5}	7.70×10^{-26}	3.85×10^{-21}	5.64×10^{15}
Cs 134	0.93×10^{-3}	1.36×10^{5}	1.47×10^{-8}	1.6×10^{-3}	4.01×10^{-9}	4.46×10^{-4}	3.67
I 135	2.39	8.05×10^{5}	4.58×10^{-9}	1.1×10^{-3}	5.88×10^{-15}	1.47×10^{-9}	7.8×10^{5}
Cs 136	5.14×10^{-2}	1.32×10^{4}	4.95×10^{-10}	6×10^{-6}	3.49×10^{-11}	3.88×10^{-7}	14.2
Cs 137	6.3×10^{-4}	5.76×10^{5}	6.34×10^{-8}	3.2×10^{-3}	1.91×10^{-8}	9.55×10^{-4}	3.32
Cs 138	32	2.62×10^{4}	41.8×10^{-12}	-	-	-	-
Ba 140	5.4×10^{-2}	3.56×10^{2}	12.1×10^{-12}	$4.\times10^{-7}$	9.09×10^{-13}	3.03×10^{-8}	13.3
La 140	0.415	3.70×10^{2}	5.1×10^{-12}	2.5×10^{-7}	1.33×10^{-14}	6.65×10^{-10}	384
Ce 144	2.44×10^{-3}	1.25×10^{3}	122.5×10^{-12}	1.2×10^{-5}	3.05×10^{-11}	3.05×10^{-6}	4.02
Pr 144	5.13×10^{-2}	1.37×10^{6}	5.13×10^{-8}	-	-	-	-
Tritium	∞	1.49×10^{6}	1.74×10^{-7}	5.8×10^{-5}	4.75×10^{-8}	1.59×10^{-5}	3.66
Total		9.15×10^{6}					

TABLE 2

CONCENTRATIONS OF RADIOISOTOPES IN THE HUDSON RIVER AT
INDIAN POINT AND CHELSEA

Accidental Loss of Entire Primary Coolant
(1% Failed Fuel) in a Burst Release

(1) Isotope	(2) Equilibrium Activity in the Primary Coolant (curies)	(3) River Conc. at I. P. 1/2 Day After Release μc/ml	(4) fractions of mpc	(5) Time for Max.Conc. to reach Chelsea (days)	(6) Maximum River Conc. at Chelsea μc/ml	(7) fractions of mpc	(8) River Dilution Between I P & C
Mn 54	0.092	5.83×10^{-9}	5.83×10^{-5}	20.4	2.22×10^{-11}	2×10^{-7}	2.9×10^{2}
Mn 56	19.9	2.26×10^{-10}	2.26×10^{-6}	1.4	2.68×10^{-16}	3×10^{-12}	7.5×10^{5}
Co 58	2.78	1.76×10^{-10}	1.76×10^{-6}	17.6	4.75×10^{-12}	5×10^{-8}	3.7×10^{1}
Fe 59	0.064	4.05×10^{-10}	6.75×10^{-6}	16.2	1.21×10^{-11}	2×10^{-7}	3.4×10^{1}
Co 60	0.29	1.84×10^{-9}	3.68×10^{-5}	21.4	8.18×10^{-11}	2×10^{-6}	1.8×10^{1}
Br 84	9.65	6.1×10^{-8}	-	0.7	2.87×10^{-26}	-	2.1×10^{18}
Rb 88	920	5.81×10^{-6}	-	0.5	2.3×10^{-30}	-	2.5×10^{24}
Rb 89	1.95	2.39×10^{-8}	-	0.5	3.89×10^{-34}	-	6.2×10^{25}
Sr 89	0.91	5.73×10^{-9}	1.91×10^{-3}	16.5	1.94×10^{-10}	6×10^{-5}	3.2×10^{1}
Sr 90	0.049	3.1×10^{-10}	1.0×10^{-3}	21.6	1.2×10^{-11}	4×10^{-5}	2.5×10^{1}
Y 90	0.099	4.84×10^{-10}	2.42×10^{-5}	6.3	2.11×10^{-12}	1×10^{-7}	2.4×10^{2}

Q 11.9-12

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(continued)

TABLE 2
(Continued)

(1) Isotope	(2) Equilibrium Activity in the Primary Coolant (curies)	(3) River Conc. at I. P. 1/2 Day After Release µc/ml	(4) fractions of mpc	(5) Time for Max.Conc. to reach Chelsea (days)	(6) Maximum River Conc. at Chelsea µc/ml	(7) fractions of mpc	(8) River Dilution Between I P & C
Sr 91	0.469	$1.25x10^{-9}$	$1.79x10^{-5}$	2.7	$4.25x10^{-14}$	$6x10^{-10}$	3×10^{4}
Y 91	19.9	$1.20x10^{-7}$	$4.0x10^{-3}$	17.0	$4.01x10^{-9}$	$1x10^{-4}$	4×10^{1}
Mo 99	1170	$6.56x10^{-6}$	$3.28x10^{-2}$	6.4	$2.61x10^{-8}$	$1x10^{-4}$	3.3×10^{2}
I 131	622	$3.8x10^{-6}$	12.2	9.8	$4.99x10^{-8}$	$1.7x10^{-1}$	7.2×10^{1}
Te 132	65.7	$4.14x10^{-7}$	$1.38x10^{-2}$	18±	$1.3x10^{-8}$	$4x10^{-4}$	3.5×10^{1}
I 132	195	$3.35x10^{-8}$	$4.18x10^{-3}$	1.3	$9.7x10^{-16}$	$1x10^{-10}$	4.2×10^{7}
I 133	485	$2.06x10^{-6}$	2.06	3.8	$8.03x10^{-10}$	$8x10^{-4}$	2.6×10^{3}
Te 134	6.94	$6.73x10^{-13}$	-	0.8	$5.6x10^{-24}$	-	1.2×10^{11}
I 134	127	$6.04x10^{-11}$	$3.02x10^{-6}$	0.8	$3.45x10^{-21}$	$2x10^{-16}$	1.5×10^{10}
Cs 134	81.5	$5.17x10^{-7}$	574	21.1	$2.01x10^{-8}$	$2.2x10^{-3}$	2.6×10^{6}
I 135	485	$9.3x10^{-7}$	$2.3x10^{-1}$	2.2	$6.62x10^{-12}$	$1.6x10^{-6}$	1.4×10^{5}
Cs 136	7.9	$5.0x10^{-8}$	$5.55x10^{-4}$	11.5	$8.98x10^{-10}$	$1x10^{-5}$	5.6×10^{1}

(continued)

Q 11.9-13

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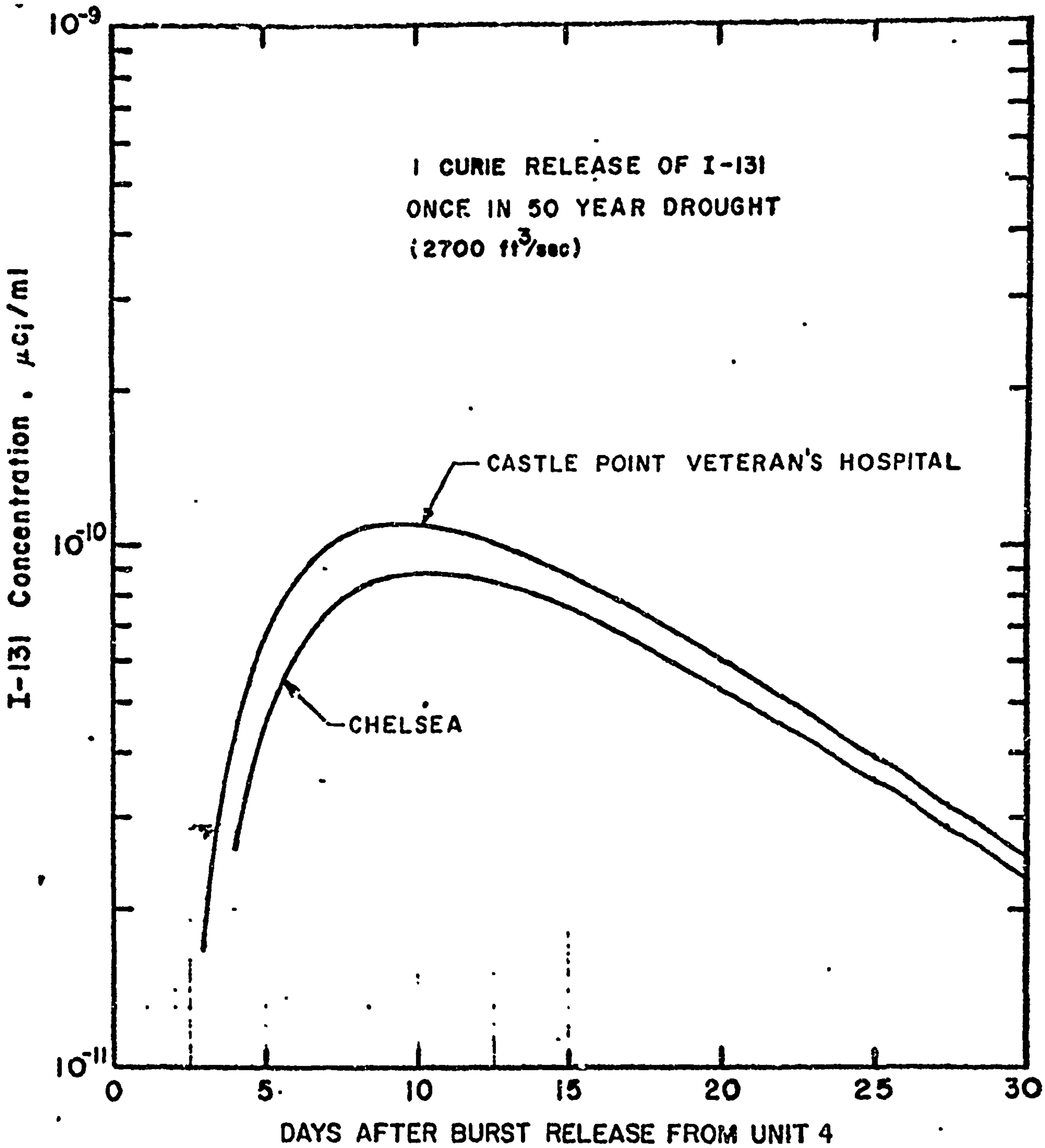
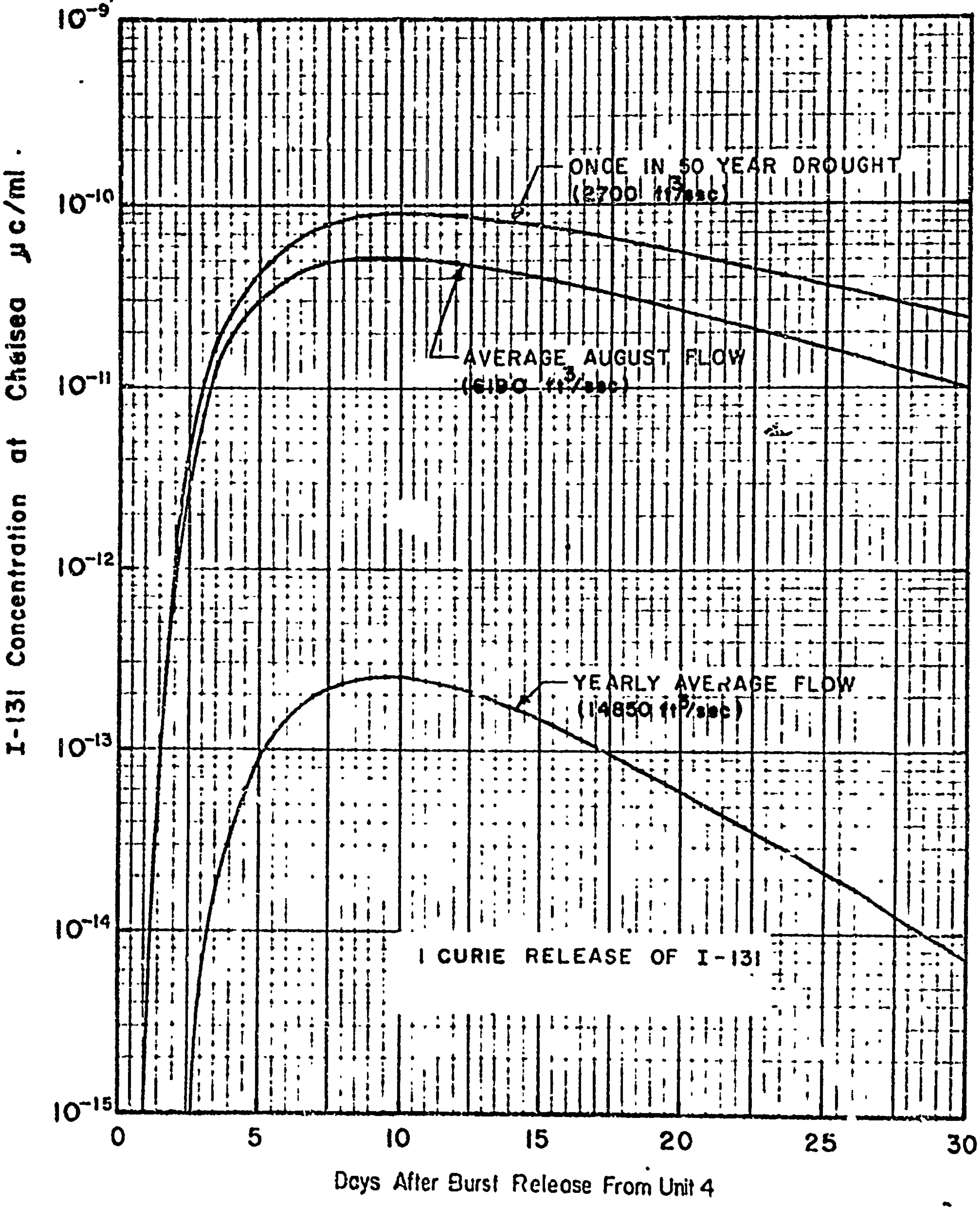
TABLE 2
(Continued)

(1) Isotope	(2) Equilibrium Activity in the Primary Coolant (curies)	(3) River Conc. at I. P. 1/2 Day After Release μc/ml	(4) fractions of mpc	(5) Time for Max.Conc. to reach Chelsea (days)	(6) Maximum River Conc. at Chelsea μc/ml	(7) fractions of mpc	(8) River Dilution Between I P & C
Cs 137	348	2.20×10^{-6}	1.10×10^{-1}	21.6	8.73×10^{-8}	4.4×10^{-3}	2.5×10^{1}
Cs 138	15.7	1.09×10^{-15}	-	0.7	5.23×10^{-26}	-	2.1×10^{10}
Ba 140	0.212	1.35×10^{-9}	4.50×10^{-5}	11.5	2.3×10^{-11}	8×10^{-8}	5.6×10^{2}
La 140	0.22	1.15×10^{-9}	5.75×10^{-5}	5.2	1.95×10^{-12}	1×10^{-7}	5.8×10^{2}
Ce 144	0.075	4.74×10^{-10}	4.75×10^{-5}	20.3	1.78×10^{-11}	2×10^{-7}	2.4×10^{2}
Pr 144	0.082	5.19×10^{-10}	-	11.7	9.65×10^{-12}	-	5.4×10^{1}
Tritium	890	5.36×10^{-6}	1.79×10^{-3}	21.8	2.22×10^{-7}	8×10^{-4}	2.2×10^{0}



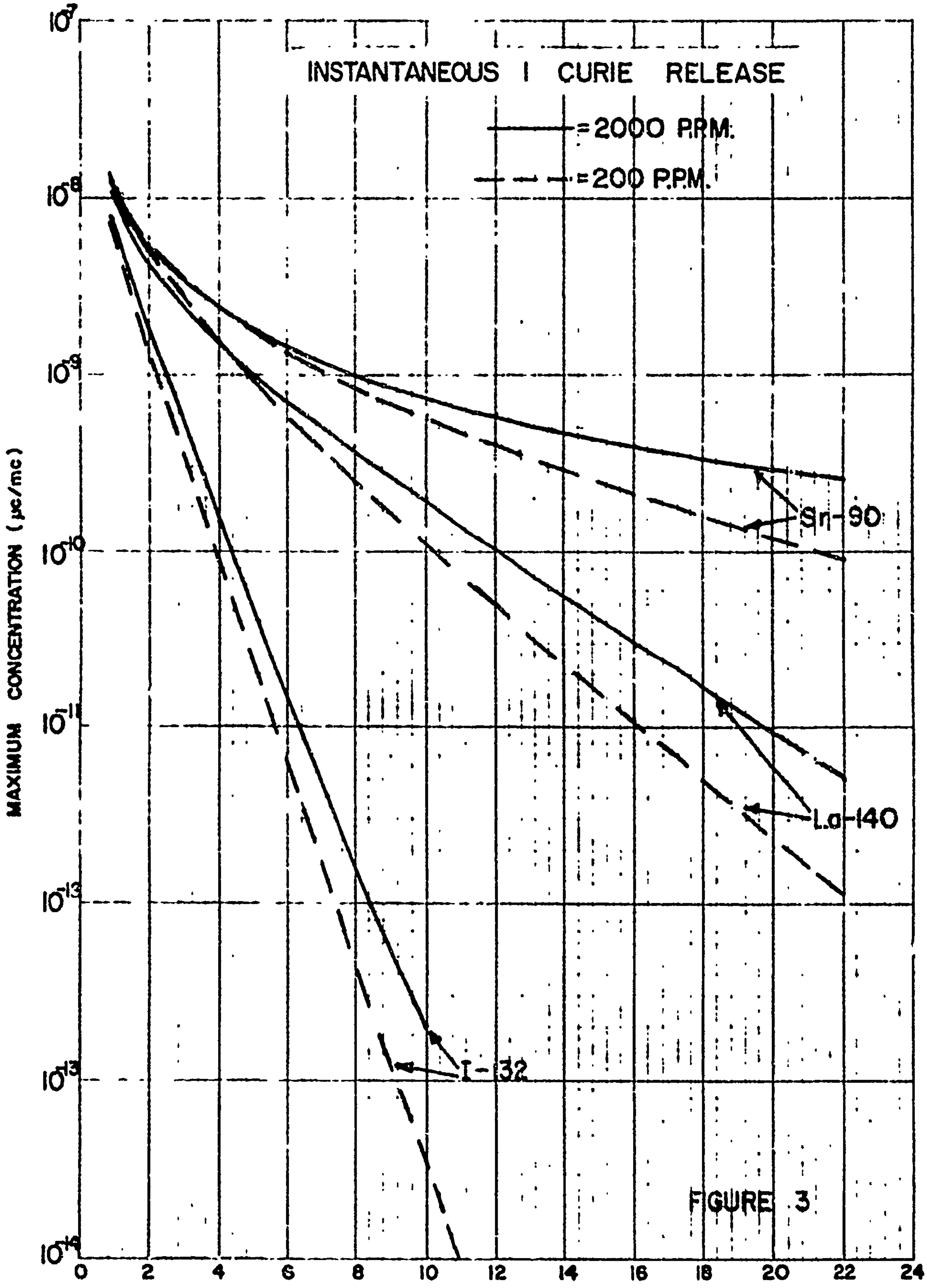
1 CURIE RELEASE OF I-131
ONCE IN 50 YEAR DROUGHT
(2700 ft³/sec)
CASTLE POINT VETERAN'S HOSPITAL
CHELSEA
I-131 Concentration, μci/ml
10⁻⁹
10⁻¹⁰
10⁻¹¹
0
5
10
15
20
25
30
DAYS AFTER BURST RELEASE FROM UNIT 4

FIGURE 1

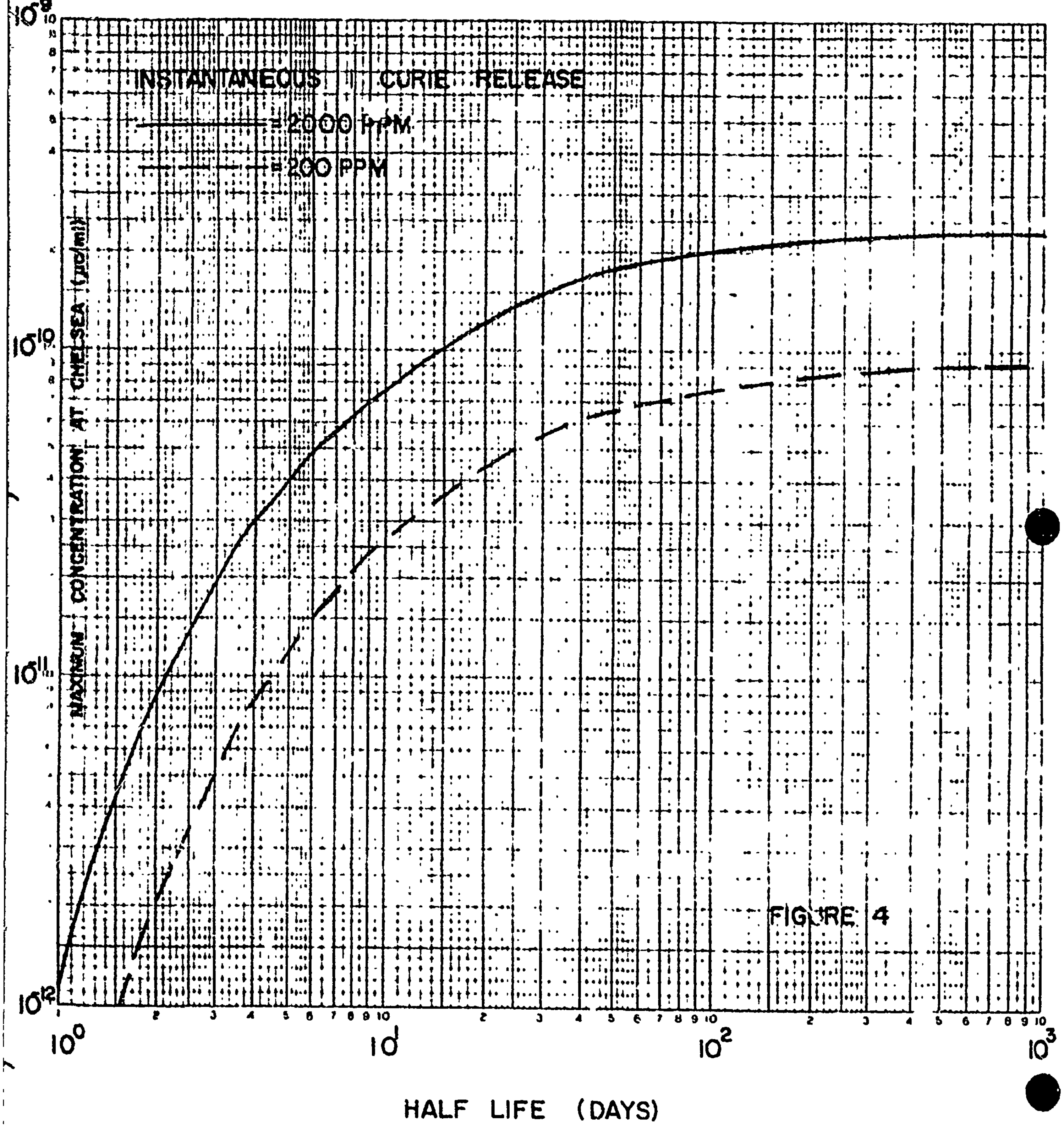


ONCE IN 50 YEAR DROUGHT
(2700 ft³/sec)
AVERAGE AUGUST FLOW
(6100 ft³/sec)
YEARLY AVERAGE FLOW
(14850 ft³/sec)
1 CURIE RELEASE OF I-131
I-131 Concentration at Chelsea μc/ml
Days After Burst Release From Unit 4

FIGURE 2



INSTANTANEOUS 1 CURIE RELEASE
= 2000 PPM.
= 200 PPM.
Sr-90
La-140
I-132
FIGURE 3
MAXIMUM CONCENTRATION (μc/mc)
DISTANCE UPSTREAM (MILES)



Q11.9-18

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Full Logarithmic, 3 × 3 Cycles

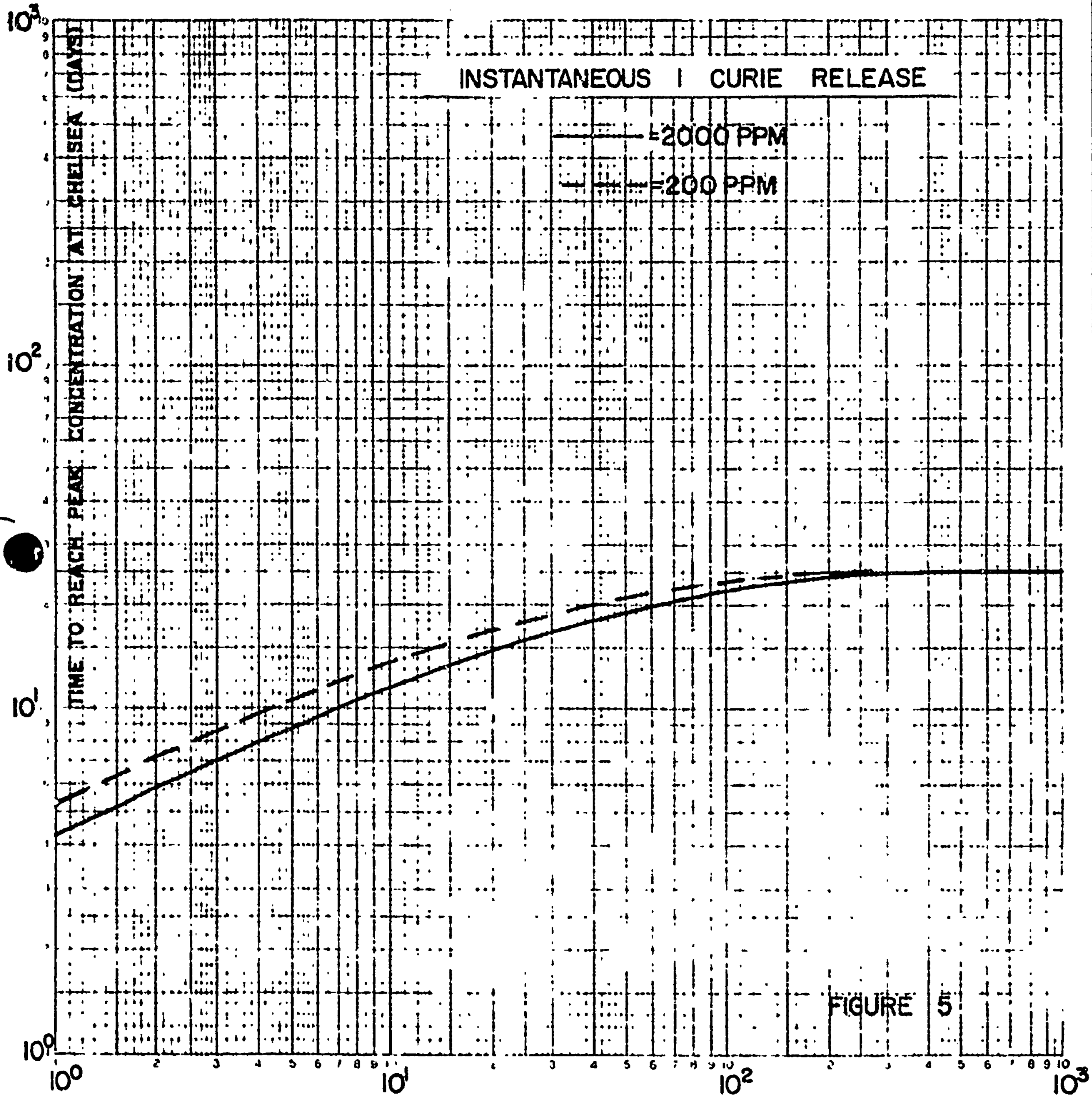

FIGURE 5

Q11.9-19

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Full Logarithmic, 3 × 3 Cycles

QUESTION 11.10

Provide the analysis used in obtaining the average annual dilution factor of 1.2×10^{-5} sec/m^3 stated as applicable for the setting of the routine release limits.

ANSWER

A study was performed using the meteorological data presented in FSAR Section 2 and the actual site boundary distance in each sector to determine the limiting sector for routine releases. This limiting sector was determined to be the 025-040 sector. The annual average dispersion factor of 1.38×10^{-5} sec/m^3 has been calculated using the sector 025-040 data presented in Section 2.0 and the shortest distance in that sector of 576 m. The appropriate mathematical model for this calculation is given below:

$$(\chi/Q) = \frac{2}{\beta\sqrt{\pi}} \sum_i \frac{F_i}{\bar{u}_i \; C_{zi} \; X[X_i]^{(1-m_i/2)}}$$

F_i = Fraction of time wind is blowing into sector while the ith metereological category exists

β = 2 tangent $(\theta/2)$

θ = Angle denoting sector of interest

$\bar{u}_i$ = Wind speed (average) for each metereological category

C_{zi} = Dispersion parameter

m_i = Stability factor

X_i = $X + X_{oi}$ = Downwind distance including wake effect for the various wind categories

X_{oi} = Building wake distance factor

The virtual source displacement X_{oi} is given by:

$$X_{oi} = \left(\frac{A}{8C_{yi}C_{zi}}\right)^{\frac{1}{2-m_i}}$$

and is applied in all stability categories. Values of X_{oi} are tabulated below:

L_1 X_{oi} = 43 meters

L_2 X_{oi} = 61 meters

N X_{oi} = 91 meters

I X_{oi} = 430 meters

The use of a virtual source displacement in all stability conditions is predicated on the physical reality that gas released at the surface of a building in a wind stream will spread over the lee surface of the building in any stability condition prior to being transported downwind. It may be noted that values of X_{oi} calculated by the above equation are those which would be obtained using a wake area correction factor c = 1/5 in the Gaussian formulation. This is shown below. Compare the two atmospheric dispersion formulations:

<u>Gaussian</u> $\chi/Q = \dfrac{1}{(\pi\theta_y\theta_z + CA)u}$

<u>Sutton</u> $\chi/Q = \dfrac{2}{\pi u C_y C_z (X+X_o)^{2-m}}$

Set these two forms equal at X=0, $\theta_y = \theta_z = 0$, and neglecting second order effects, the above terms can be equated, yielding:

$$2CA = \frac{\pi}{8} A$$

thus,

$$C = \frac{\pi}{16} \approx \frac{1}{5}$$

This is more conservative in comparison with the commonly used value of C = 1/2.

The data used in this model is presented in Section 2 and repeated here:

Sector 025-040

Category i	Fraction F_i	$i/\bar{u}_i$	C_{zi}	C_{yi}	m_i
L_1	0.0097	0.575	0.48	0.6	0.2
L_2	0.0019	0.191	0.43	0.53	0.3
N	0.0198	0.358	0.39	0.47	0.4
I	0.0574	0.493	0.07	0.40	0.5

$\beta = 0.353$

QUESTION 11.11

Submit details of the proposed operational environmental radiation monitoring program including, a map delineating sampling locations, types of sampling done, numbers of samples taken, frequency of sampling, types of analyses done, monitoring equipment, laboratory equipment to be used, the logic used in determining food chains to man, anticipated reconcentrating agents and isotopes reconcentrated biota collected.

ANSWER

The env... ..ntal monitoring program for Indian Point Station is described in the at...ed chart. A map delineating sample locations is also included.

For the list of laboratory equipment refer to the Final Safety Analysis Report, Volume 3, Section II.2.

The question whether radionuclides released at low level from nuclear reactors or fallout can be reconcentrated biologically to appear in significant amounts in food sources, has been under active study in the Hudson River ecosystem by New York University's Institute of Environmental Medicine over the past five years.[1, 2, 3]

The Hudson River is not a major source of food for man. The major contributors to the food chain for man are shad and striped bass, both migratory fish, which spend part of their lives in the river. Although there are fish which are indigenous to the local area, they are not used primarily for food. They have little economic value but, of course, fish such as yellow perch may be consumed by fishermen. Edible shellfish are not taken in the stretch of the river twenty miles above and below Indian Point. The nearest shellfish are in Raritan Bay at the mouth of the Hudson, an area presently closed to shellfishing because of pollution from sewage.[4, 5] At this end of the Hudson, more than forty miles south of Indian Point, a tremendous dilution of the river flow takes place due to the tidal motion. Although crabs were seen in the vicinity of Indian Point some ten years ago, they have been relatively scarce in recent years. Accordingly, fish and water represent the main sources of possible intake for man. Both will be monitored.

In the Hudson River system in the vicinity of Indian Point, presently, only fish represent a signficant food source. About 22 grams/day or less than one ounce of fish and sea food is consumed per capita in the United States, so that the possible intake of radionuclides from Hudson River fish (should this make up the whole daily consumption) would be very small.[6]

The longer lived nuclides present in low level releases include isotopes of manganese, cobalt and cesium. While some species of rooted vegetation and filter feeding molluscs, concentrate some of the radioactive components of reactor effluent in the Hudson, none of these species are used for human or animal consumption. Fish, on the other hand, while possible sources of food do not demonstrate accumulation of the nuclides in question. For both manganese and cobalt there is a natural barrier to absorption in the gut of fish which restricts their uptake of these elements. In fact, the major part of these radioelements would be expected to be located in undigested gut residues rather than in the fish flesh which may be consumed. Hence, the potential contamination of diet from this source is miniscule.[7, 8, 9]

New York University has conducted studies of the Hudson River biota. They have identified several species of rooted aquatic vegetation which concentrate stable mangenese of the genus Potomogeton. These vegetations will be collected in our sampling program during the growing season and will be used as biological indicators. The seasonal behavior of these aquatic vegetations, with respect to manganese uptake is under active investigation at New York University.

1. Radioecological Survey at the Hudson River

   Progress Report No. 1, submitted to the Division of Radiological Health, USPHS, Contract PH86-64-Neg. 141, March 15, 1965, by New York University Medical Center, Institute of Environmental Medicine.

2. Ecological Survey at the Hudson River

   Progress Report No. 2, submitted to the Division of Radiological Health, USPHS, Contract PH86-95, Neg. 141, December 1966, by New York University Medical Center, Institute of Environmental Medicine.

3. Ecological Survey of the Hudson River

   Progress Report No. 3, submitted to the National Center for Radiological Health, Contract PH86-65-102 and the New York Department of State, Contract No. C-19560, September 30, 1968, by New York University Medical Center, Institute of Environmental Medicine.

4. The Hudson, Fish and Wildlife

   A report on fish and wildlife resources in the Hudson River Valley prepared for the Hudson River Valley Commission by the Division of Fish and Game of the New York State Conservation Department, State of New York, undated.

5. Environmental Radioactivity in New York State - 1968

   Dr. Thompson, Report to the Commissioner, Department of Health, State of New York, July 14, 1969.

6. Food Consumption of Households in the United States

   Spring 1965, U. S. Department of Agriculture, ARS 62-16, 1967.

7. A Survey of the Invertebrates from Selected Sites at the Lower Hudson River

   H. I. Hirshfield, J. W. Rachlin and E. Leff in Hudson River Ecology, Symposium Proceedings, p. 220ff, October 1966.

8. The Elementary Chemical Composition of Marine Organisms

   A. P. Vinagrador, Memor, Sears Foundation No. 2, New Haven, Conn. U.S.A., 1953.

9. Development of a Biological Monitoring System and a Survey of Trace Metals, Radionuclides and Pesticide Residues in the Lower Hudson River

   Final Report to New York State Department of Health, Institute of Environmental Medicine, October 10, 1969.

TABLE 1:

INDIAN POINT STATION

No.	Media	Type	Sampling Frequency	Method of Collection	Locatior
1	Fallout	Continuous	Monthly	Open pot type collector	Point 1 miles sc site of
2	Air Particulate and Organic Iodide	Continuous at 1 CFM	Weekly	Two fixed membrane filters (0.8 micron size) preceding a charcoal filter	Points 4 and 5 additio at poin Peekski and Ver one wee consecu

ENVIRONMENTAL SURVEY

	Analysis	Minimum Sensitivities	Measurement Instrumentation	Remarks
nd 15 th of Eastview	Gross beta and tritium	1 picocurie per liter for gross beta	Gas flow, windowless proportional counter for gross beta	Measurements made 48 hours after collection to allow for decay of randonthoron daughters
			Nuclear Measurement Corporation Type PC 3A Type PC 11A Type PC 11T	
		3000 picocuries per liter for tritium		
, 2, 3, and in offsite s in 1, Buchanan, lanck for periods ively	Gross beta and gamma spectrum	0.1 picocuries per cubic meter for gross beta	Same as 1 for gross beta	Measurements made soon after collection and 48 hours later to allow for decay of randon-thoron daughters
		2 picocuries per cubic meter for I-131	Gamma spectrum with 3" x 3" NaI crystal with 400 channel analyzer	
			Radiation Instruments Development Laboratories Model 3612 Gamma Spectrometer	

TABLE 11.

INDIAN POINT STA

No.	Media	Type	Sampling Frequency	Method of Collection	Lo
3	Drinking Water Supplies	Grab	Monthly		Po an
4	Hudson River Water	Continuous	Weekly	Continuous flow regulated to fill 50 gallon drums. Representative sample taken once a week and drums emptied	Hu in i an di ca ar
5	Lake Water	Grab	Monthly	1 liter sample off-shore	Pc ar
6	Well Water	Grab	Monthly	From deep-well pumps	Pc ar
7	Lake Aquatic Vegetation	Grab	Once each in Spring, Summer and Fall	Along the lake shore	S
8	Hudson River vegetation	Grab	Same as 7	Along river shore	P 1 m c P T a V
9	Hudson River Bottom Sediment	Grab	Same as 7	Same as 8	S
10	Hudson River Fish	Catch	Monthly	Same as 8	W a

-1 (Continued)

ON ENVIRONMENTAL SURVEY

tions	Analysis	Minimum Sensitivities	Measurement Instrumentation	Remarks
nts 6, 7 8	Same as 1	Same as 1	Same as 1	Same as 1
son River t pipe o the plant, at plant charge al. Points 9 10	Same as 1 and tritium on monthly composite	Same as 1	Same as 1	Same as 1
nts 11, 12, 13	Same as 1	Same as 1	Same as 1	Same as 1
nts 6, 14 Verplanck	Same as 4	Same as 1	Same as 1	Same as 2
e as 5	Same as 2	1 picocurie per gram for gross beta 2 picocuries per gram for I-131	Same as 2	Dry weight for spectrum soon after collection. Sample asked and counted 48 hours after collection for gross beta
nts 10, 15, and 17. At th of dis- rge canal, kskill Bay, pkins Cove off planck	Same as 7	Same as 7	Same as 2	Same as 7
e as 8	Same as 7	Same as 7	Same as 2	Mud dried for both measurements
re avail- e near site	Same as 7	Same as 7	Same as 2	Sample ashed and counted 48 hours after collection for gross beta and gamma spectrum taken

TA

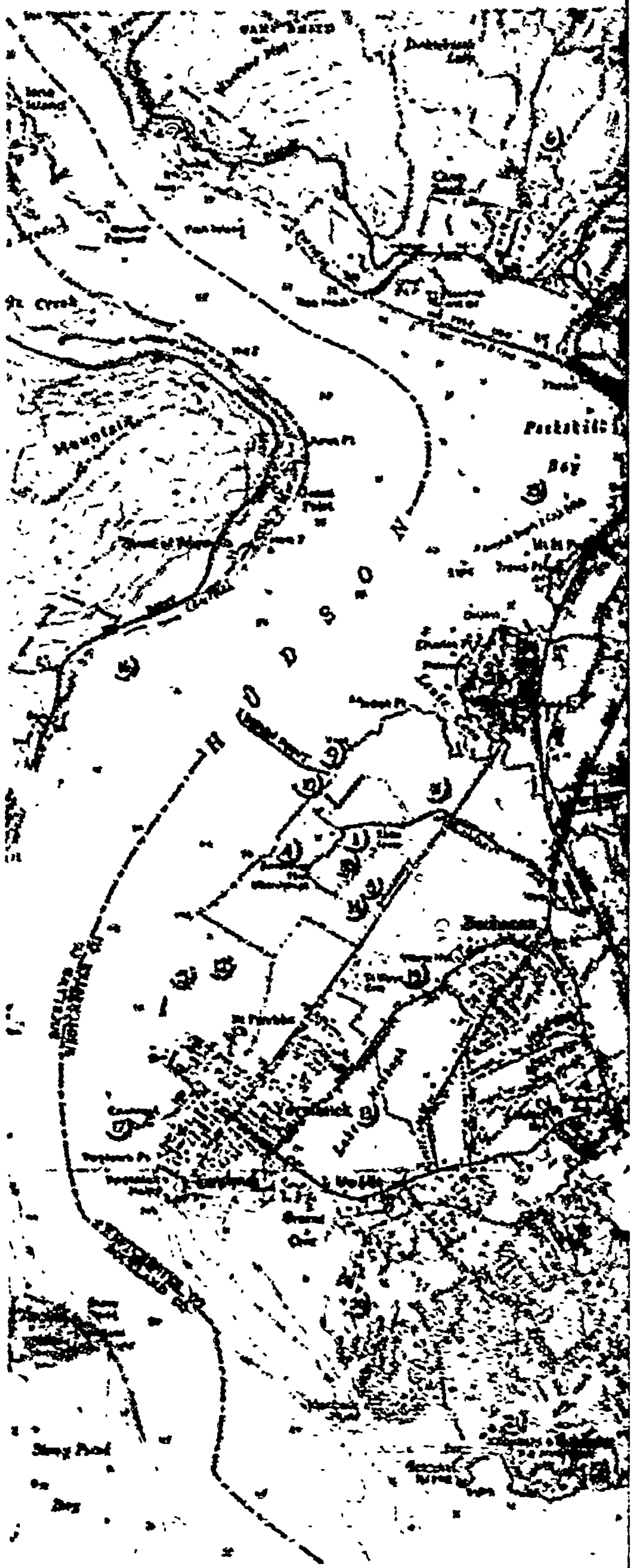
INDIAN PO

No.	Media	Type	Sampling Frequency	Method of Collection	Lo
11	Vegetation	Grab	Same as 7	Grab samples with 100 ft^2 area	Po 19
12	Soil	Grab	1 per year	Grab samples 2" in diameter by 2" deep	Sa
13	Direct Gamma	Spot Readings	Once a year		Al re 5 of
15 14	Direct Gamma	Continuous	Monthly		Se ti Bu Ve Mo Pe an nu po at pe

ΛΕ 11.11-1 (Continued)

NT STATION ENVIRONMENTAL SURVEY

ations	Analysis	Minimum Sensitivities	Measurement Instrumentation	Remarks
nts 6, 18, 20 and 21	Same as 7	Same as 7	Same as 2	
e as 11	Same as 7	Same as 7	Same as 2	Soil dried for spectrum and measured soon after collection. Gross be'ı of dried soil made 48 hours after collection
ng principal ds within a ile radius plant	Gross gamma background	2.2×10^6 counts per minute in a Cesium-137 field of 1 mr/hr	Franklin Systems Inc. Model 15-2	Instrument readings in counts per minute measured at approximately 1/10 mile intervals. Readings converted to microrem per hour
ected loca-ns in hanan, planck, trose, ukfill, at a ber of nts on-site the plant imeter	Same as 13	1 mr	Victoreen Ionization Chamber Model 239 0-10 mr or Film badges or TLD-Thermo-luminescent dosimeters	



HUDSON
Peekskill
Bay
Stony Point
Bay

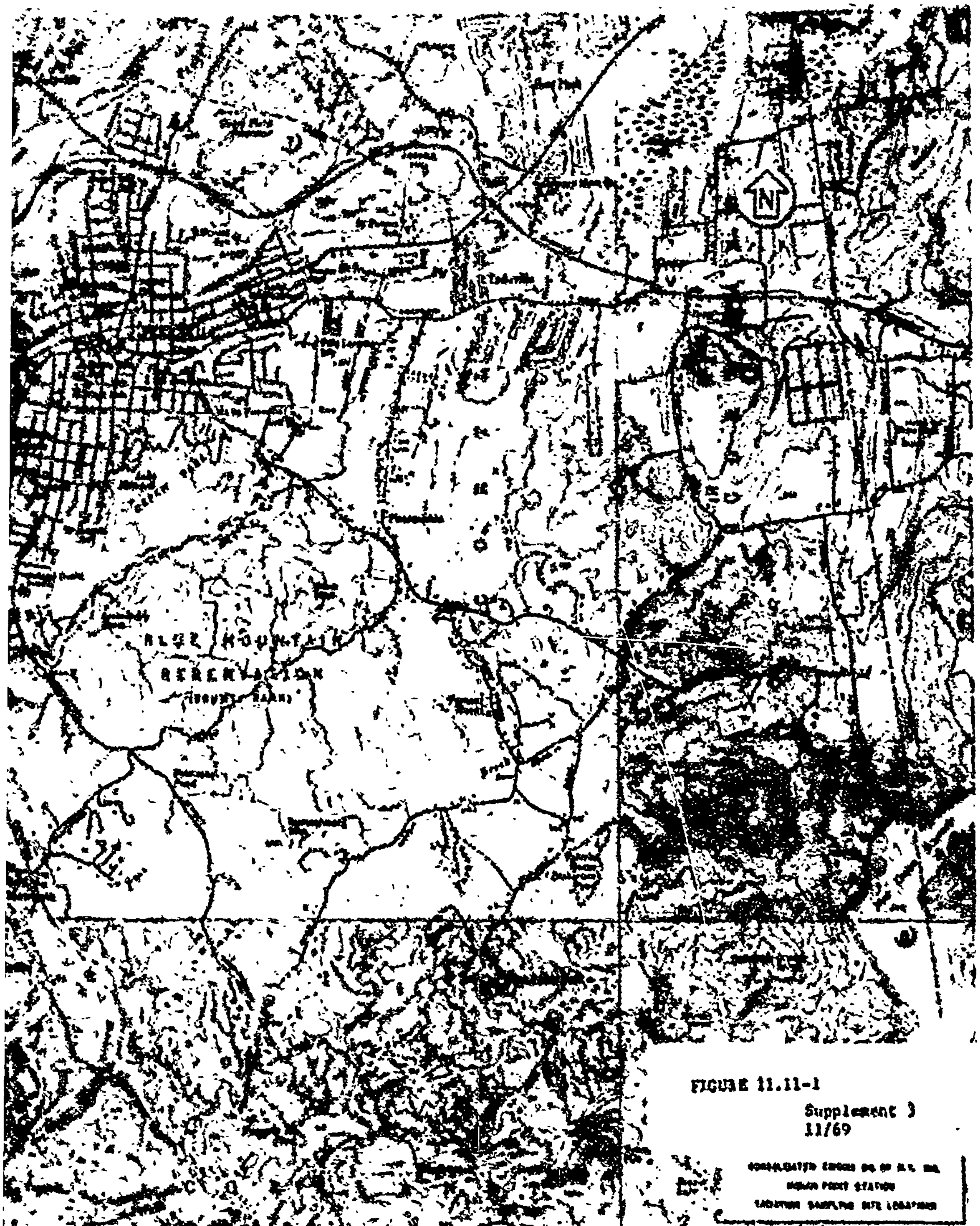


FIGURE 11.11-1

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QUESTION 11.12

We understnad that it is not proposed that milk and river bottom sediment sampling be done as part of the proposed operational radiation environmental monitoring program for this facility. Since these are two prime modes of reconcentrating radioactivity released to the environment, explain the bases for your position that their collection is not necessary.

ANSWER

Hudson River bottom sediment has been sampled on a periodic basis since the environmental monitoring program was revised in 1966.(1) As part of the expanded environmental monitoring program for Indian Point Station, sediment samples will now be taken at the discharge canal and at three locations where river aquatic vegetation is collected. (Refer to plate attached for 11.11).

In addition to a continuing comprehensive program being carried out by the New York State Health Department which is able to compare results to those 15
from other areas of the site,(2) milk will be sampled on a monthly basis under program 3 for monitoring gaseous releases as presented in Technical Specification 4.10.

(1) - Results of sediment measurements are reported bi-annually in the "Survey of Environmental Radioactivity in the Vicinity of Indian Point Station" Series USAEC Docket No. 50-3.

(2) - The results of these surveillance activities are reported in quarterly Radioactivity Bulletins issued by the Department of Health at the State of New York. An annual summary report is also issued.

QUESTION 11.13

Please describe any "as-installed" calibrations which will be performed on the continuous radiation monitoring system. This description should include but not be limited to:

1. Are liquid and gaseous sources similar to those expected during normal operation discharged through the monitors to determine that they are properly installed and operating correctly?

2. How often during the life of the plant will this "primary" calibration be performed?

3. For routine calibrations, how are secondary standard calibrations performed?

4. Describe the maintenance procedures which will be utilized to guard against radioactive contamination of the pipe or line being monitored.

ANSWER

Liquid and gaseous sources, similar to those expected during normal plant operations, will not be used to verify proper installation and operating capability of the detectors. A check source, installed in the sampler, will be used to verify that the detectors are operating and properly installed.

A "Primary" calibration is performed on a one time basis in the vendor's "Design Verification Test." Further primary calibrations are not required since the geometry cannot be significantly altered within the sampler. The "Design Verification Test" utilizes typical isotopes of interest to determine proper detector response.

Secondary standard calibrations are performed with a radiation source of known activity. These single point calibrations are intended to verify the original vendor calibration. Cesium sources will be used for both gaseous and liquid effluent monitors. The secondary standard calibration will be performed by removing the detector and placing the source on the sensitive area of the detector. The secondary standard calibration will be performed at each refueling outage.

An additional secondary calibration will be performed periodically by manually sampling the system involved and analyzing for composition and activity using gamma spectrometry, gross beta or gamma counting and radiochemistry. The knowledge of the isotopes present will then be used for proper instrument calibration.

There are no specific routine maintenance procedures for the Radiation Monitoring System monitors. If background buildup is observed decontamination procedures will be performed.

QUESTION 11.14

If there is no "as-installed" system calibration capability, explain the bases for your position that the installed monitoring systems are adequate to identify with reasonable accuracy the nature and amount of radioactivity contained within the systems being monitored.

ANSWER

The installed monitoring systems are not designed to determine the nature and amount of radioactivity in the systems being monitored, but are designed to detect the minimum concentrations of the isotopes of interest, as indicated in Section 11, Table 11.2-7. These systems monitor gross activity and are designed to generate an alarm under abnormal conditions and in most cases generate automatic responses. Isotopic identification and concentrations are determined by grab sample analysis.

QUESTION 12.1

Staffing, Training and Experience.

1. Provide a chart of the proposed organization for Indian Point No. 2. At the shift level, indicate the shift composition for both dual-unit operation and for Unit No. 2 when it is operating as a single unit facility, such as when Unit No. 1 is shut down for refueling.

2. Describe the responsibilities of all facility personnel from the foreman level up to and including the General Superintendent. Indicate those positions for which the individual will exercise responsibility at Unit No. 1 in addition to his responsibilities at Unit No. 2.

3. Provide the following for facility personnel that have been added to the staff for Unit No. 2.

    a. The training program they will participate in, including courses taken, general course content, and number of hours of each course.

    b. Resumes of the qualifications of personnel from the supervisory level up through superintendent level.

4. Provide the following for current facility personnel who will assume responsibilities relative to Unit No. 2.

    a. A resume of their experience that qualifies them for the duties they will assume with respect to Unit No. 2.

    b. The specific training they will receive relative to Unit No. 2.

ANSWER

Staffing, Training and Experience

1. As stated in Section 12 of the Unit No. 2 Final Safety Analysis report, the existing Unit No. 1 organization will be expanded to provide the administration and technical needs for both Unit No. 1 and Unit No. 2. The General Superintendent who is now responsible for the overall operation of Unit No. 1 will also be responsible for the overall operation of Unit No. 2. The attached Figure 1 shows the proposed organization for Indian Point dual unit facility. For simplicity sake, the operating organization at the shift level is not shown on this figure but is shown in detail on the attached Figure 2.

At the shift level, we would expect that the shift composition for Unit No. 2 when it is operating as a single unit facility, such as when Unit No. 1 is shut down for refueling, would remain essentially as shown on Figure 2. Under conditions such as this, the shift composition for Unit No. 2 would in fact be strengthened in that the Control Room Operator B would be able to devote a substantially larger portion of his time to the operations of Unit No. 2. Whenever Unit No. 1 is shutdown for maintenance or refueling and its controls are not being manipulated or in the case of refueling, no changes in core geometry are being made which would result in added reactivity to the core, the monitoring functions for Unit No. 1 may be assumed by the Unit No. 2 licensed control room operator provided that he holds a valid reactor operators license for Unit No. 1.

2. Facility personnel responsibilities:

General Superintendent -

The General Superintendent is in charge of administering all phases of operation, training and maintenance of the facility.

Superintendents (Operations) -

The Operations Superintendents are responsible for operation of the individual generating facilities (electrical, mechanical and nuclear) and for all associated maintenance and janitorial activities.

Superintendent (Performance) -

The Performance Superintendent supervises and directs the activities of station technical personnel and supervises the training of Reactor Operator and Senior Reactor Operator license candidates. Responsible for the maintenance of all records and reports relating to inventory control of facility SNM.

Reactor Engineer -

The Reactor Engineer is responsible for supervision of all phases of reactor physics testing, core reactivity follow, nuclear instrumentation calibration, facility operations reporting, and all matters relating to reactor operations.

Production Engineers -

The Production Engineers provide staff support for the Operations Superintendents in the areas of training, development and implementation of test procedures, evaluation of test results, etc. Since December, 1967 they have devoted essentially full time to Unit No. 2 startup activities participating in all phases of the training programs, preparation of system descriptions and operation procedures, review of pre-startup test procedures, etc. It is expected that they will provide technical support on a shift basis during Unit No. 2 startup and initial operation.

Assistant Superintendent (Maintenance)

The Assistant Maintenance Superintendent is responsible for all facility maintenance work both electrical and mechanical. In addition, he is responsible for the ordering and inventory control of all equipment spare parts.

Assistant Superintendents (Operations)

The Assistant Superintendents are responsible for the day by day coordination of all activities associated with Unit operations including those activities relating to waste disposal, refueling, spent fuel shipping, etc. In addition, the Assistant Superintendents will maintain close surveillance on all operating log books and recorder charts with the aim of detecting early trends which may lead to unsafe conditions.

Supervising Engineer (Health Physicist)

The Supervising Engineer is responsible for the testing, servicing and repair of all station instrumentation and relays and for the health physics functions within the station.

Asst. Supv. Engr. (Health Physics)

The Assistant Supervising Engineer for Health Physics is responsible for the implementation of the health physics program established for the station.

Asst. Supv. Engr. (Nuclear Plant Instruments)

The Assistant Supervising Engineer for nuclear plant instruments is responsible for the testing, servicing and repair of all relays and instruments associated with the nuclear plant.

Asst. Supv. Engr. (Conventional Plant Instruments)

The Assistant Supervising Engineer for conventional plant instruments is responsible for the testing, servicing and repair of all non-nuclear station relays and instruments.

Production Engineer (Training)

The Production Engineer (Training) is responsible for the training of Reactor Operator and Senior Reactor Operator license candidates.

General Watch Foreman -

The General Watch Foreman is in charge of the station and its operating personnel during his work shift. He is responsible for assuring that all operations are conducted in accordance with approved procedures and the Production Departments rules and regulations and are within the limitations set forth in the facility Technical Specifications.

Watch Foreman -

As shown on Figure 2, Unit No. 1 and Unit No. 2 will each have a shift Watch Foreman who will have the direct responsibility for operation of his respective Unit. The Watch Foreman will hold a Senior Reactor Operators License for his particular Unit and as such will assist the General Watch Foreman in the discharge of his duties.

Station Chemist -

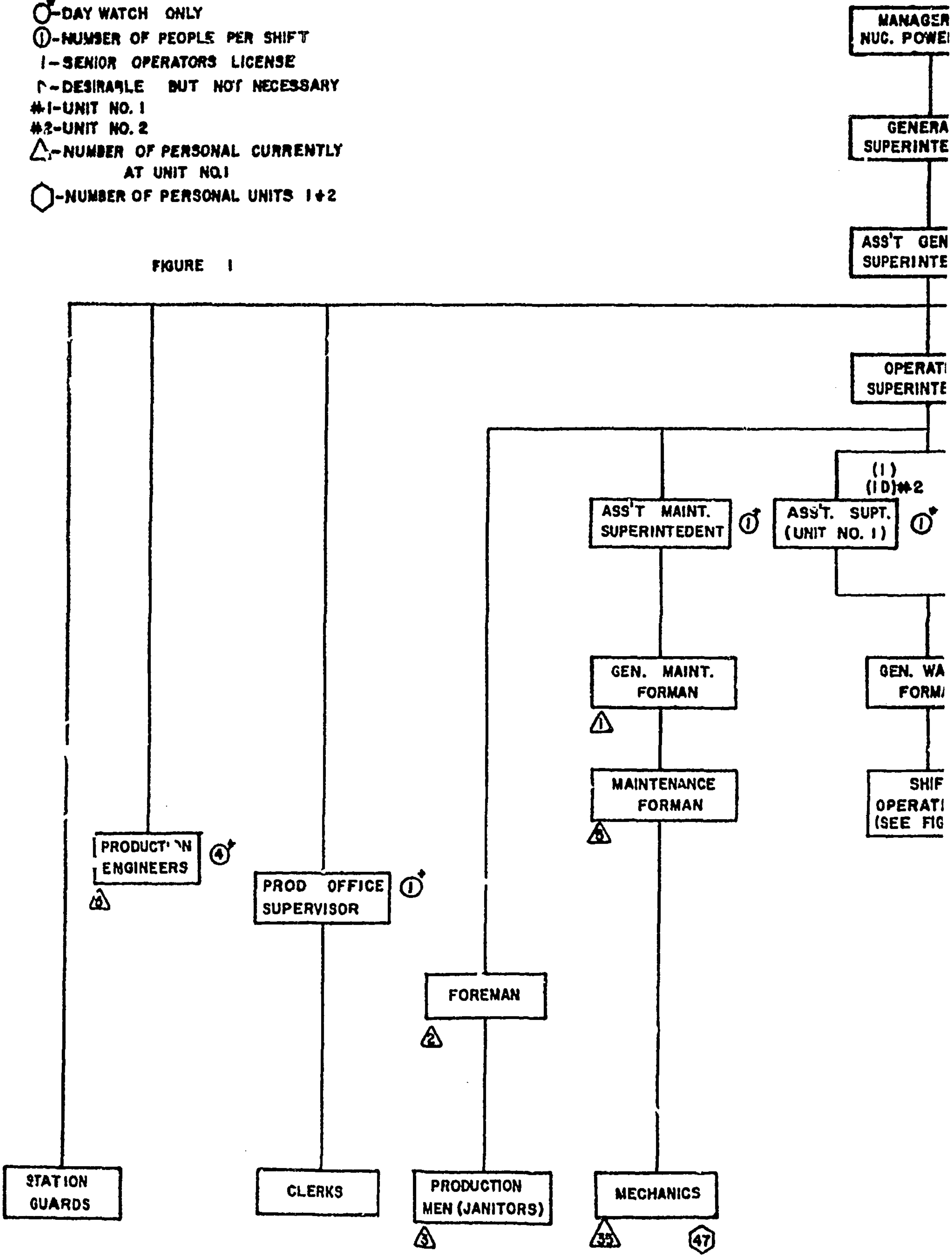
The Station Chemist is responsible for maintaining surveillance and control of the cold and hot chemistry of the various systems within the station.

Technical Assistant (Performance) -

The Performance Technical Assistant is responsible for maintaining surveillance of station performance and the preparation of all related records and reports.

NOTES-
- DAY WATCH ONLY
(1)-NUMBER OF PEOPLE PER SHIFT
I-SENIOR OPERATORS LICENSE
-DESIRABLE BUT NOT NECESSARY
#1-UNIT NO. 1
#2-UNIT NO. 2
△-NUMBER OF PERSONAL CURRENTLY AT UNIT NO.1
⬡-NUMBER OF PERSONAL UNITS 1+2

FIGURE 1



INDIAN POINT STATION
ORGANIZATION CHART

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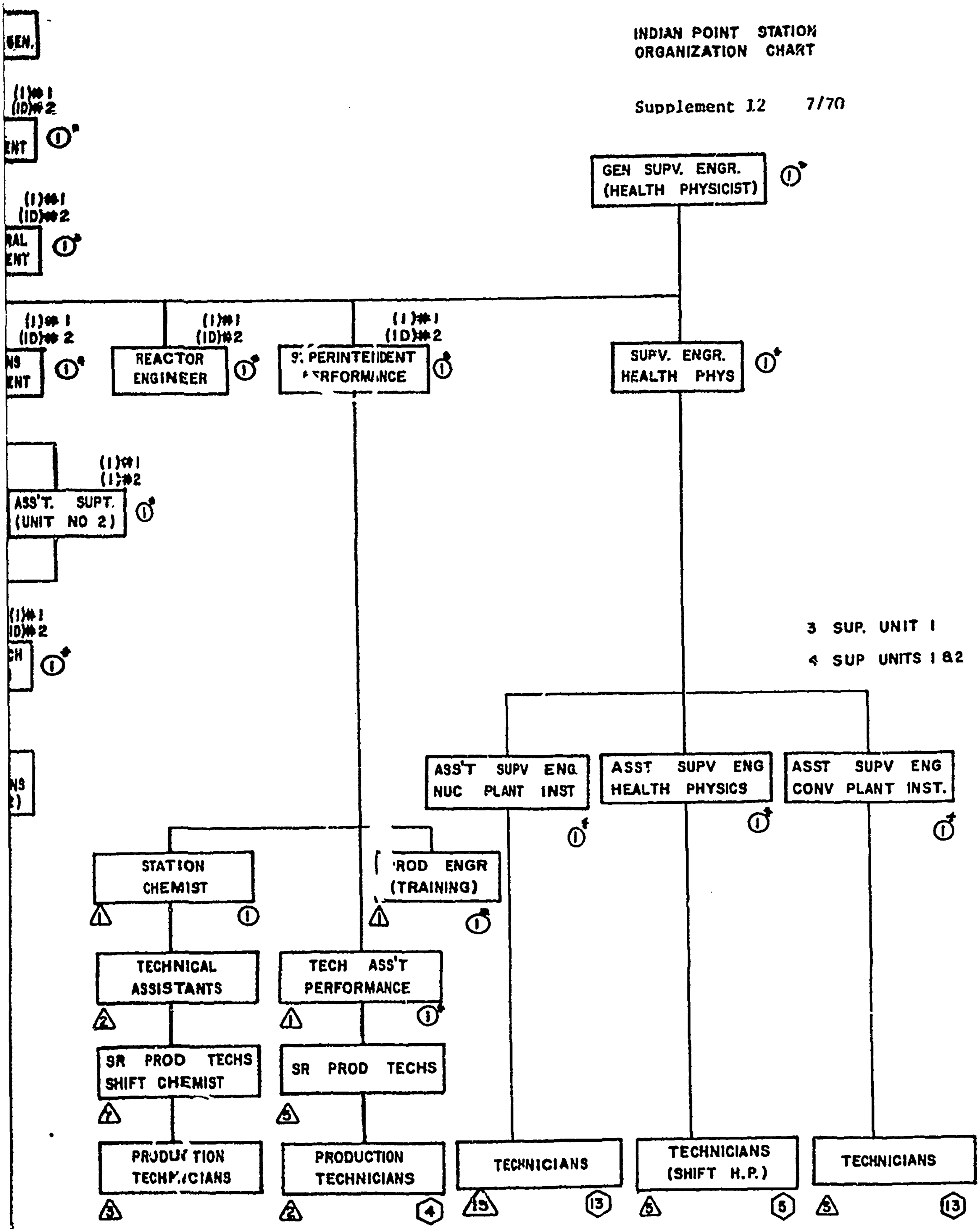
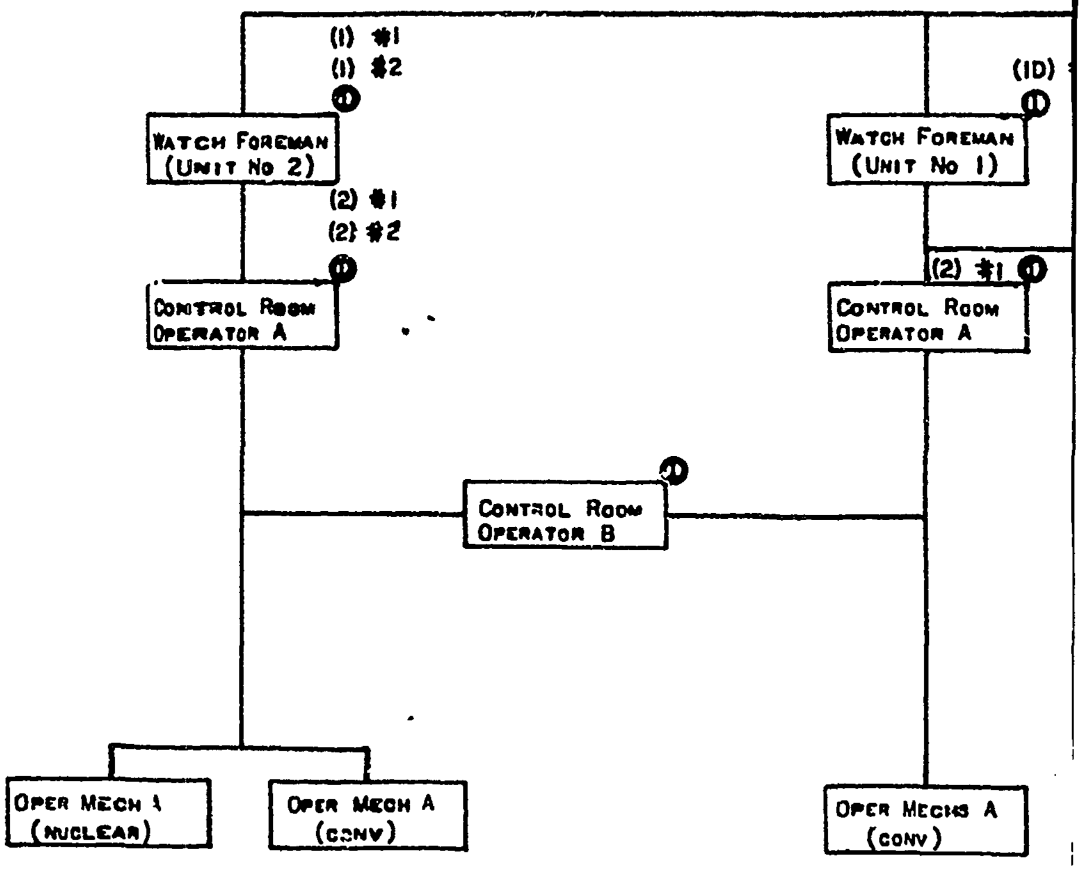
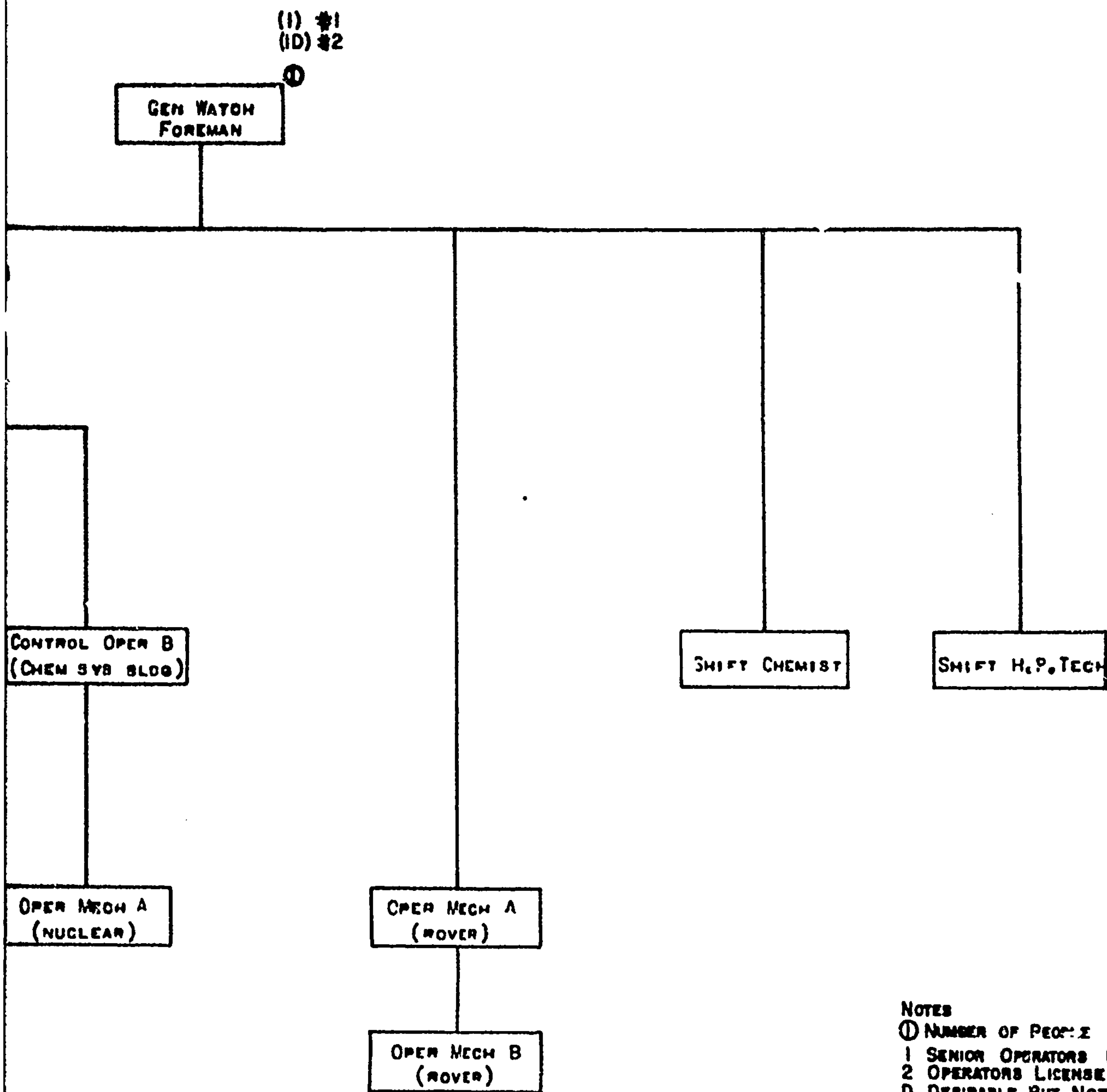


Consolidated Edison Co of N.Y. Inc.



INDIAN POINT STATION
SHIFT ORGANIZATION



NOTES
① NUMBER OF PEOPLE PER SHIFT
I SENIOR OPERATORS LICENSE
2 OPERATORS LICENSE
D DESIRABLE BUT NOT NECESSARY
#1 No. 1 UNIT
#2 No. 2 UNIT

FIGURE 2

3. (a) A training program was initiated in December, 1967 for certain key supervisory personnel assigned responsibilities in connection with the startup and operation of Unit No. 2. Included in the group were the General Superintendent, Unit No. 2 Operations Superintendent, Reactor Engineer, Assistant Operations Superintendent, four staff Production Engineers and four candidates for the position of Unit No. 2 Watch Foreman.

The basic objective of the training program was to train shift supervisory, and ultimately control room personnel, to qualify for Unit No. 2 AEC examinations for Senior Reactor Operator and Reactor Operator. The fou. staff Production Engineers are expected to provide technical support on a shift basis during startup and initial operation.

The first four and one-half months of the program was spent in familiari[illegible]n with Unit No. 2 and the Preliminary Safety Analysis Report p[illegible]o attending an off-site six week training course sponsored by the reactor supplier (Westinghouse). This course covered all facets of plant design and operation.

The principal purpose of the off-site training course was to familiarize Unit No. 2 personnel with plant design bases and characteristics. Formal classroom lectures were supplemented by field trips to nearby Westinghouse manufacturing plants where several pieces of Unit No. 2 hardware such as the reactor coolant pumps, fuel assemblies, rod cluster control elements, main generator, control rod drive mechanisms, etc. were viewed in various stages of manufacture. Subjects covered during the lecture portion of the course were as follows:

a. Reactor coolant system
b. Reactor coolant piping
c. Steam generators
d. Reactor coolant pumps
e. Reactor vessel internals
f. Fuel assemblies

g. Rod cluster control elements

h. Control rod drive mechanisms

i. Fuel handling equipment

j. Auxiliary coolant system

k. Chemical and volume control system

l. Sampling system

m. Radioactive waste disposal system

n. Heat exchangers, evaporators, gas strippers and demineralizers

o. Electrical systems

p. Control boards

q. Plant control aspects

r. Plant protection aspects

s. Nuclear plant tripping functions

t. Nuclear instrumentation

u. Overpower protection

v. Tavg. control system

w. Steam dump system

x. Solid state rod control system

y. Rod position indicating system

z. Pressurizer level and pressure control

aa. Control hardware

Misc. process control channels

cc. In-core instrumentation

dd. Radiation sources and shielding

ee. Plant radiation monitoring system

ff. Prodac 250 computer

gg. Pre-operational testing

hh. Nuclear design

ii. Nuclear operations

jj. Plant operations

Following the off-site training course, five control room operators were added to the startup group. After an initial indoctrination program for these five operators, a formal on-site training program was started in which all members of the staff participate. The program covers in detail the design and operation of Unit No. 2 with frequent field trips to enhance the classroom lectures. In addition, the training program includes a six week refresher course in reactor theory.

It is expected that supervisory personnel will receive a total of 2160 hours of training and control room operators 1240 hours before the Unit is placed in service. The following is a breakdown of the training time:

	Supervisory Personnel	Control Room Operators
Off-site Training Program	240 hrs.	-
Conventional Plant Training Sessions	680 hrs.	320 hrs.
Nuclear Plant Training Sessions	1600 hrs.	680 hrs.
Reactor theory	240 hrs.	240 hrs.
Total	2160 hrs.	1240 hrs.

(b) Resumes of the qualifications of personnel from the supervisory level up through the superintendent level are attached.

4. In addition to those personnel whose responsibilities and training relative to Unit No. 2 startup and operations are described in 3 (a), certain other key supervisory personnel such as the Performance Superintendent and Supervising Engineer (Health Physicist) will expand their present Unit No. 1 responsibilities to include those associated with the operation of Unit No. 2. The Supervising Engineer and the Assistant Supervising Engineer for nuclear plant instrumentation attended portions of the off-site training program relating to nuclear instrumentation and plant control and protection. In addition, they have participated in pertinent on-site training sessions, test procedure reviews and protective relay adjustments on new equipment being placed in service.

The Performance Superintendent is in charge of all license training activities. He has personally administered the Unit #2 training program.

The performance staff, while not being exposed to a formal training program, is undertaking a program of self instruction. This program includes the reading of system descriptions, field trips and the compilation of the necessary drawings and data to support their objectives on Unit #2. In addition, they attend operator training lectures as time permits and the need arises.

One member of the chemical staff, a Technical Assistant, has been assigned full time to Unit #2. His duties include the training of the remaining supervisory and eligible employees of the chemical staff.

No formal training course is planned for the Chemist or Performance Technical Assistant; however, special indoctrination programs will be given as required to acquaint these personnel with any unique features of Unit No. 2 systems or equipment.

<u>TECHNICAL SERVICES</u> - UNIT NO. 2 TRAINING

A. General Supervising Engineer & Assistant Supervising Engineer (Nuclear Plant) attended I&C, Nuclear Instrumentation and Control Rod Drive portion of Westinghouse sponsored Design Course on Unit 2 (As specified in Q12.1)

B. One Assistant Supervising Engineer (Gov. Plant) and Four Technicians attended the Foxboro sponsored course on Instrumentation, the course was of one week's duration.

C. The Supervising Engineer and Three Technicians attended a 3-day course on Radiation Monitoring Instruments, the course was given by Tracerlab.

D. An Associate Project Engineer assigned to the Technical Services Bureau, but not in residence at Indian Point, attended a two week course given by the Public Health Service on "Gamma Spectroscopy".

TECHNICAL SERVICES
COURSE CONTENTS

A - Foxboro School -
Classroom and Laboratory work in the theory, operation, and maintenance on Foxboro Instrumentation installed in Unit No. 2

B - Tracerlab School -
Circuit operation, calibrations and panel operations of Unit No. 2 RMS Instrumentation

C - PHS - "Gamma Spectroscopy" -
Course outline attached

TECHNICAL SERVICES - UNIT NO. 2 MANPOWER

Unit No. 1
- 1[illegible] Testers (Technicians)
- 5 Shift H.P.'s
- 3 Supervisors

Unit No. 2 and Unit No. 1 Requirements
- 4 Supervisors
- 5 Shift H.P.'s
- 26 Technicians (some do H.P. work from time to time)

Presently 6 men are assigned to Unit 2 to follow instrumentation and control testing.

5. Summary of Unit 2 Operating Manpower Requirements

Assistant Superintendent - One required - Two others are available for back-up

Watch Foreman - Four required - Four additional are available for back-up

Control Room Operators - Four required - Three additional are available for back-up

The number of personnel required to start-up and operate Unit 2 include:

1-Assistant Superintendent
4-Watch Foreman
4-Control Room Operators
9-Auxiliary Operators

A back-up crew of current Unit 2 Management Operations personnel totaling 13 members and a back-up crew of 3 control room operators are available to be deployed as required for start-up and initial plant operations.

The following individuals may prepare for hot license following initial operation of Unit 2. The General Superintendent, Superintendent of Unit 1, Assistant Superintendent and Unit 1 and the General Watch Foreman.

SUMMARY OF UNIT 2 PO

NAME	PRESENT TITLE	PRESENT LICENSE HELD UNIT 1	LICENSE SOUGHT ON UNIT 2	LICENSE OPTION MANDATORY	DESIRABLE	HOT	C
A.A. Karosza	Assistant General Superintendent	SRO	SRO		D		
J.M. Makepeace	Reactor Engineer	SRO	SRO		D		
S.H. Cantone	Superintendent Performance	SRO	SRO		D		
A.A. Nespoli	Assistant Superintendent	SRO	SRO	M			
W.A. Monti	Assistant Superintendent	SRO	SRO		D		
M.F. Shatkovski	Production Engineer	SRO	SRO		D		
C.C. Limoges	Production Engineer	SRO	SRO		D		
B.T. Moroney	Production Engineer	-	SRO		D		
C. Agan	Production Engineer	-	SRO		D		
G. Case	Production Engineer	-	SRO		D	H	
M. Hughes	Watch Foreman	SRO	SRO	M			
M. Anderson	Technical Assistant	SRO	SRO	M			
C. Powell	Technical Assistant	SRO	SRO	M			
Seven Control Room Operators	Control Room Operator "A"	SRO	SRO	4M	3D		
Nine Auxiliary Operators	Operating Mechanic "A"	-	-	-	-	-	
2 Operating Foreman and 2 Engineers	-	-	-	-	-	-	

[illegible]TION ASSIGNMENTS

D	CANDIDATE FOR UNIT 2 POSITION	BACK UP FOR UNIT 2 POSITION	REMARKS
	Acting Unit 2 Operations Superintendent		Presently administrating over all start-up and training program of Unit 2
	Reactor Engineer		Actively engaged as Reactor Engineer of Unit 1 and aiding in start-up functions of Unit 2
	-	Assistant Superintendent	Directly responsible for training operators for Unit 2 and aiding in start-up functions of Unit 2
	Assistant Superintendent	-	This group is actively engaged in training and start-up functions of Unit 2; writing system descriptions, operating procedures and other plant documents; preparing for plant functions such as lubrication, spare parts, charts, etc.; participating in equipment acceptance and testing; and participating in other plant commissioning requirements.
	-	Assistant Superintendent; Watch Foreman	
	-	Watch Foreman; Control Room Operator	
	-	Watch Foreman; Control Room Operator	
	-	Watch Foreman; Control Room Operator	(Will assume future [illegible] responsibilities.)
	-	Watch Foreman; Control Room Operator	
	-	-	(Responsible for preservice and [illegible] service inspections, quality control and [illegible] for site operations.)
	Watch Foreman	Control Room Operator	These candidates for Watch Foreman and Control Room Operator positions are receiving formal class room and field training and [illegible] contributing to and participating in start-up functions.
	Watch Foreman	Control Room Operator	
	Watch Foreman	Control Room Operator	
	Control Room Operators	-	
	Nuclear and Conventional Auxiliary Operating Posts	-	These nine Operating Mechanics and others from a manpower pool from Unit 1 as required are performing all the operating requirements during construction and start-up testing Unit 2.
	-	-	Four of the people presently supervising and directing the operation of construction and start-up testing of Unit 2 may be added for Watch Foreman back-up during plant start-up and initial operations if found necessary.

Steven H. Cantone - Superintendent Performance

Education:

Dumont High School, Dumont, New Jersey
from 1955 to 1959, Graduate.

Stevens Institute of Technology, Hoboken, New Jersey
from 1959 to 1963, Graduate.

Experience:

June 1963 to date - Consolidated Edison Company of New York, Inc.

June 1963 - Employed as Cadet Engineer. Assigned to Mechanical Engineering Department.

December 1963 - Transferred to Ravenswood Generating Station. Assigned to Production Department staff.

June 1964 - Transferred to Mechanical Construction Department at Ravenswood Generating Station. Assigned to Unit No. 3 Turbine Construction.

December 1964 - Transferred to Production Department at Ravenswood Generating Station. Assigned to Production staff.

June 1965 - Assigned to Production staff at Ravenswood Station.

March 1966 - Transferred to Indian Point Station. Under the supervision of a licensed facility senior operator received training on controls of the nuclear and conventional plants of the Indian Point Unit No. 1 Facility.

December 1966 - Assisted in the administration of the reactor operator training program on Unit No. 1.

October 1967 - Obtained Senior Reactor Operators License for Unit No. 1.

October 1967 - Participated in Unit No. 1 refueling program as watch supervisor.

December 1967 - Participated as an assistant in the administration of the reactor operator training program on Unit No. 1.

April 1968 - Attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of the Indian Point Unit No. 2 Facility.

June 1968 - Since this date, has actively participated in the Indian Point Unit No. 2 Facility startup program. This has included the preparation of written system descriptions and procedures, involvement in plant preoperational testing and attending the Indian Point Unit No. 2 facility formal training program. Has been responsible for the overall coordination of the training program and administration of the reactor theory lectures as well as lectures dealing with specific systems related to the Indian Point Unit No. 2 Facility.

October 1969 - Superintendent Performance - Indian Point.

Donald J. McCormick - General Superintendent

Education:

Regis H.S. New York City, 1944 to 1947. St. Simon Stock H.S. New York City, 1947 to 1948, graduate. State University of New York, White Plains, New York, 1948 to 1950 ABS Degree. Fordham University New York City, 1950 to 1953, B.S. Degree.

Experience:

1945 to 1950 - Dennison Mfg. Co., 5th Ave. New York City, Stock Clerk.

September 5, 1950 to date - Consolidated Edison Co. of N. Y. Inc.

1950 to 1953 - Technician - Astoria Laboratory. Product analysis and experimental work in water demineralization.

1953 to 1958 - Cadet Engineer and Production Engineer at Waterside Generating Station. Concerned with fuel and ash handling system, boiler water chemistry and water demineralization plant.

1958 to 1960 - Production Engineer in Nuclear Engineering Division of Mechanical Engineering Department. Aided in the design of Indian Point Station particularly the purification systems and waste disposal facilities.

1958 - Attended Nuclear Engineering Course given by Vitro Engineering Company equivalent to approximately 9 hours of college credit.

1960 - Spent six months training and operating the Engineering Test Reactor and the Materials Test Reactor at National Reactor Testing Station operated for the Commission by the Phillips Petroleum Co. In addition to operating these facilities, participated in refueling operations and various test loop programs.

1960 - Attended the four month Shippingport Atomic Power Station training school for the purpose of receiving basic indoctrination in the theory, design and operation of a pressurized water thermal reactor power plant. During this training and in addition to considerable classroom time on the theory and design of that station, personally executed pre-critical checkoff procedure and raised reactor power from full shutdown to operating load, operated at power for a period of time, and shut station down.

November 1960 - January 1965 - Production Engineer assisting in the startup of Indian Point Station. Duties included writing of operating and test procedures, supervision of installation of equipment and test and operation of various pieces of equipment and systems. Also acted as General Watch Foreman during this period.

January 1965 - April 1969 - Superintendent - Conventional Plant. Responsible for the operation of the conventional equipment associated with the reactor plant.

April 1969 - October 1969 - Superintendent - Unit No. 1. Responsible for the operation and maintenance of both the nuclear and conventional portions of Unit No. 1.

October 1969 - General Superintendent - Indian Point.

Anthony A. Karkosza - Assistant General Superintendent
(Acting Unit No. 2 Operations Superintendent)

Education:

Brooklyn Tech. H.S., DeKalb Ave., Brooklyn, N.Y., from 1942 to 1946, graduate. U.S.M.M.C.C. Academy, Kings Point, N.Y., 1946 to 1950, B.S. degree. Columbia University, Morningside Heights, N.Y., M.S. degree in Industrial and Management Engineering earned in February 1969.

Experience:

1946 to 1950 - U.S. Merchant Marine Cadet Corp. Spent time at sea aboard U.S. cargo ships, assuming the duties of a Jr. Officer which at times involved being in charge of the watch.

September 1950 to date - Consolidated Edison Co. of N.Y. Inc.

September 1950 to 1956 - Production Engineer, Production Department.

a. Assigned to the Technical Staff to study and improve the efficiency of power plant operation at various generating stations.

b. Training employees in more efficient operation.

c. Has acted as Maintenance Foreman, boiler room and turbine room, at one of the larger electric generating stations.

1957 - Followed design of Indian Point Station for Production Department; reviewing and commenting on specifications, drawings and proposals.

1958 - Attended Nuclear Engineering Course given by Vitro Engineering Company equivalent to approximately 9 hours of college credit.

March 1958 to January 1960 - On loan to Mechanical Engineering Department, Nuclear Division. Specified and purchased maintenance and handling equipment including fuel handling and associated core handling equipment. Associated with many facets of Indian Point Plant design.

1959 - Attended five week Vallecitos Boiling Water Reactor Training School. Participated in test program involving repeated approaches to criticality and measuring of control rod scram times.

1960 - Spent six months training and operating the Engineering Test Reactor and the Materials Test Reactor at National Reactor Testing Station operated for the Commission by the Phillips Petroleum Co. In addition to operating these facilities, participated in refueling operations and various test loop programs.

1960 - Attended the four month Shippingport Atomic Power Station training school for the purpose of receiving basic indoctrination in the theory, design and operation of a pressurized water thermal reactor power plant.

November 1960 to October 1962 - Production Engineer assisting in the startup of Indian Point facility. Duties included assisting in writing of operating and test procedures and supervision of installation of equipment. From the time of initial fuel loading until October 1962, functioned as Shift Supervisor in charge of the watch. As such, supervised all facility operations, and during off hours, was in full charge of the facility.

October 1962 to November 1963 – Conventional Plant Superintendent, Indian Point facility, directly responsible for operation and maintenance of conventional portion of facility.

September 1963 – Received Senior Reactor Operators License for Unit No. 1 SOP-154 which is maintained to date (September 1969 SOP-154-3).

November 1963 to September 1966 – Reactor Engineer, Indian Point Facility, a counter part line title of Conventional Plant Superintendent except being directly responsible for operation and maintenance of the nuclear portion of the facility.

September 1966 to April 1969 – Superintendent Unit No. 1 Nuclear Plant.

April, 1968 – Attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of the Indian Point Unit No. 2 Facility.

April 1969 – Duties refined to slowly phase out responsibilities for Unit No. 1 Superintendent functions and devote more time on Unit No. 2.

June 1968 to date – Has actively participated and directed the Indian Point, Unit No. 2 Facility start-up program. This has included preparation and review of written system descriptions and procedures, involvement in plant pre-operational testing and attending the Indian Point Unit No. 2 Facility training program.

October 1969 – Assistant General Superintendent – Indian Point

John M. Makepeace - Reactor Engineer

Education:

Sanford High School, Sanford, North Carolina, from 1936 - 1941, Graduate.

Purdue University, West Lafayette, Indiana, from 1941 - 1943, Graduate. B.S.M.E. degree

Purdue University, West Lafayette, Indiana, from 1946 to 1948, Graduate. M.S.M.E. degree

Experience:

1945 to 1946 - U.S. Navy, Ensign, Assistant Ship Superintendent for yard and District Craft at N.Y. Naval Shipyard.

1946 - U.S. Navy, Ensign, Assistant "M" Division Officer aboard U.S.S. Missouri. Honorable Discharge.

1948 to 1965 - Employed by Babcock and Wilcox Company, New York, New York.

1948 to 1955 - Field Service Engineer in the Babcock and Wilcox Stationary Service Department, responsible for the startup and servicing of the Company's products. These products ran the gamut from small fuel burning assemblies to large central station steam generators rated at a million pounds of steam per hour or more. This work consisted of equipment startup and coordination of all Company activity on the job, with the customer, his consulting engineer, sub-contractors to Babcock and Wilcox, and with other contractors and equipment suppliers involved in the startup. It included the training of customer personnel in the proper and safe operation of the equipment, conducting all necessary testing to see that the equipment supplied met the design requirements and guarantees.

1955 to 1960 - Field Service Engineer in the Field Operations Department of Babcock and Wilcox Atomic Division. Supervised start up of nuclear components, checking equipment to determine it met guarantees and correcting manufacturing and design problems for the following:

Mine Safety Appliance Co., Callery, Pa. SIR Mark (B) test facility.

Knolls Atomic Power Laboratory, West Milton, N.Y. "Seawolf" prototype steam generator.

Atomic Power Development Associates, Detroit, Michigan. Liquid metal heater.

Pratt and Whitney Aircraft Corporation, East Hartford, Conn. Liquid metal heater.

KAPL, Groton, Conn. Start of Seawolf Steam Generators.

Southern California Edison Company, Santa Susana, California. Sodium heated steam generator for SRE.

Babcock and Wilcox Critical Experiment Laboratory, Lynchburg, Va. MARAD Critical Experiments.

Arco, Idaho. S-1W reactor vessel.

U.S. Naval Shipyard, Portsmouth, N.H. U.S.S. Seadragon and U.S. S. Thresher steam generators.

Electric Boat Co., Groton Conn., U.S.S. Skipjack steam generators.

1960 to 1962 - Field Service Engineer assigned as Babcock and Wilcox representative at N.Y. Shipbuilding Corporation shipyard at Camden, N.Y. Participated in program to prepare test procedures for N.S. Savannah power plant, coordinated Babcock and Wilcox personnel efforts and reviewed work being performed. Also coordinated various Babcock and Wilcox activities relating to the testing and operation of Babcock and Wilcox furnished equipment aboard the N.S. Savannah. Participated in the N.S. Savannah crew training program.

1962 - 1965 - Field Service Engineer assigned to Con Edison Indian Point nuclear power plant as Site Representative. Acted as liaison engineer with Babcock and Wilcox Atomic Energy Division, Lynchburg, Va., and provided consulting services to Con Edison in various matters relating to the nuclear power plant.

1965 to date - Reactor Engineer, Indian Point Station, responsible for supervision of all phases of reactor physics testing, core reactivity follow, nuclear instrumentation calibration, facility operations reporting, and all matters relating to reactor operations.

March, 1967 obtained Senior Reactor Operators License for Unit No. 1.

January 1968 to date - Member of the ASME Ad Hoc Committee on Inservice Inspection of Nuclear Reactor Coolant Systems.

April, 1968 attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of the Indian Point Unit No. 2 Facility.

June, 1968 to date - Has actively participated in the Indian Point, Unit No. 2 Facility startup program. This has included preparation and review of written system descriptions and procedures, involvement in plant pre-operational testing and attending the Indian Point Unit No. 2 Facility formal training program.

Anthony A. Napoli - Assistant Superintendent

Education:

Erasmus High School, Flatbush Ave., Brooklyn, from 1938 to 1943, grad. U.S. Navy, 1943 to 1945, Motor Machinist Mate 1/c attended Richmond Diesel School and Camp Lejune Diesel School. Honorable Discharge. Attended City College of New York, New York City from 1947 to 1949, night classes, Mechanical Engineering. Attended Pratt Institute, Brooklyn, New York from 1951 to 1955, night classes, Mathematics, Power Generation, etc.

Experience:

February 1946 to date - Consolidated Edison Co. of New York, Inc.

February 1946 - Started in the Company in the Production Department as a Boiler Cleaner B assigned to boiler cleaning at the Hudson Avenue Gold Street Station.

September 1946 - Promoted to Auxiliary Operator assigned to the operation of turbine auxiliaries.

April 1951 - Title changed to Mechanic B assigned to maintenance of regulators and controls for the boilers and auxiliaries.

May 1953 - Transferred to Astoria Station as a Mechanic B and assigned to maintenance of electric equipment.

July 1956 - Promoted to Operator B and assigned to the No. 1-2 Unit Control Room operating the controls for the turbo-generators, boilers and associated auxiliary equipment.

September 1958 - Promoted to Operator A and assigned in charge of the No. 1-2 Unit Control Board.

December 1958 - Promoted to Watch Foreman in charge of the operations and maintenance of No. 1-2-3 main units.

October 1960 to date - Assigned to Indian Point Station as a Watch Foreman. In 1961 aided in preliminary testing of equipment and attended training school for reactor operators. In 1962 aided in systems testing, fuel assembly, fuel loading, and initial facility startup, and trained as Watch Foreman on the operation and maintenance of the nuclear plant under the supervision of a licensed facility operator. In 1963 has continued training for the operation and maintenance of the nuclear plant under the supervision of a licensed operator. Has participated in preparations for reactor start, in reactor starts, operation, shutdowns, and maintenance.

July 1963 - Obtained a Senior Reactor Operator's License for Unit No. 1.

January 1965 - Promoted to General Watch Foreman in charge of shift on operation and maintenance of No. 1 unit at Indian Point Station.

December 1967 - Promoted to Assistant Superintendent to supervise Unit No. 1 operation, maintenance and coordination of maintenance outages.

April 1968 - Attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of Indian Point, Unit No. 2 facility.

June 1968 - Since this date resummed duties as Assistant Superintendent and has actively participated in the Indian Point Unit No. 2 facility startup program. This involved plant pre-operation testing and attending the Indian Point, Unit No. 2 facility formal training program.

Michael F. Shatkouski - Production Engineer

Education:

Brooklyn Technical High School, Brooklyn, New York from 1954 to 1959, Graduate.

City College of New York, Manhattan, New York from 1959 to 1965, Graduate.

Experience:

July 1964 to September 1964 Consolidated Edison Company of New York, Inc. Junior Production Technician 59th St. Station Performance Division.

February 1965 to date - Consolidated Edison Company of New York, Inc.

February 1965 - Employed as Cadet Engineer and assigned to Waterside Station Production Staff.

February 1966 - Transferred to Station Construction and Shops Department at East River Station.

August 1966 - Transferred to Production Department at Indian Point Station with the title of Production Engineer. Under the supervision of a licensed facility senior operator received training on controls of the nuclear and conventional plants of the Indian Point Unit No. 1 facility.

October 1967 - Obtained Senior Reactor Operator License for Unit No. 1.

April 1968 - Attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of the Indian Point, Unit No. 2 facility.

June 1968 - Since this date has actively participated in the Indian Point, Unit No. 2 facility start-up program. This has included preparation of written system descriptions and procedures, involvement in plant pre-operational testing and attending the Indian Point, Unit No. 2 facility formal training program.

Charles Clifford Limoges - Production Engineer

Education:

Woodrow Wilson Vocational High School, Queens, N.Y.
from 1953 to 1957, Graduate.

Delahanty High School, Jamacia, New York
from 1958 to 1959, High School.

Jamacia High School, Jamacia, New York
from Jan. 1960 to June 1960.

City College of New York, Manhattan, New York
from 1960 to 1965, Graduate.

Experience:

June 1957 to September 1960 - Apprentice joiner, N.Y. Naval Shipyard, Brooklyn, New York.

June 1961 to September 1961 - Cabinet makers helper, laminated veneers, Richmond Hill, New York.

August 1962 to September 1962 - Frazer operator, S.:. Frank and Co. Richmond Hill, New York.

August 1963 to September 1963 - Electric wirer (refrigerators) Traulsen Refrigerator Co., Flushing, New York.

August 1964 to date - Consolidated Edison Company of New York, Inc.

August 1964 - Employed as Jr. Production Technician and assigned to Astoria Station Performance Division.

June 1965 - Assigned to Production staff duties at Astoria Station.

June 1966 - Transferred to Mechanical Engineering Bureau. Assigned to Nuclear Division.

December 1966 - Transferred to Indian Point Station with the title of Production Engineer. Under the supervision of a licensed facility senior operator received training on controls of the nuclear and conventional plants of the Indian Point, Unit No. 1 facility.

October 1967 - Obtained senior operator license No. SOP-953, issued by the Atomic Energy Commission.

April 1968 - Attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of the Indian Point, Unit No. 2 facility.

June 1968 - Since this date has actively participated in the Indian Point, Unit No. 2 facility startup program. This has included preparation of written system descriptions and procedures, envolvement in plant pre-operational testing and attending the Indian Point, Unit No. 2 facility formal training program.

May 1970 - Attended a seven week course sponsored by Westinghouse Electric Corp. covering programming operations for the PRODAC-250 computer to be installed at the Indian Point, Unit No. 2 Facility.

William A. Monti - Assistant Superintendent

Education:

DeWitt Clinton High School, Bronx, New York
from 1954 to 1957, Graduate.

State University of New York, Maritime College
Fort Schuyler, Bronx, New York
from 1957 to 1962, Graduate.

Experience:

June 1962 to August 1962 - Third Assistant Engineer, U.S. Naval Service - MSTS, Brooklyn, New York.

September 1962 to December 1964 - Assistant Project Engineer (Nuclear), Foster Wheeler Corporation, Livingston, New Jersey.

December 1964 to April 1967 - Project Engineer (SSW Nuclear Project), General Dynamics/Electric Boat Division, Groton, Connecticut.

May 1967 to date - Production Engineer

Employed as a Production Engineer at Indian Point Station under the supervision of a licensed facility senior operator received training on controls of the nuclear and conventional plants of the Indian Point Facility Unit No. 1.

October 1967 received a Senior Reactor Operator License for Indian Point Unit No. 1.

October 1967 to April 1968 acted as a Staff Engineer at Indian Point Station.

April 1968 to June 1968 attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of Indian Point Unit No. 2.

June 1968 to date - Participating in Unit No. 2 startup program. This has included preparation of written system descriptions and procedures, involvement in plant preoperational testing and attending on site Unit No. 2 formal training program. Acting as co-ordinator of startup test program for Unit No. 2.

January 1968 - Participated in refueling of Indian Point Unit No. 1. Acted as part of a shift refueling crew on refueling elevation of containment.

July 1969 - Appointed as a member to Company Quality Assurance Task Force. The group functions to support quality requirements on Nuclear plants.

January 1970 - Assistant Superintendent - Indian Point.

William J. Smith - Watch Foreman

Education:

Lynbrook High School, Lynbrook, Long Island, New York from 1942 to 1943.

Experience:

July 1943 to December 1943 - Grumann Aircraft Co. Bethpage, Long Island., New York, Riveter.

December 1943 to April 1946, U.S. Navy, Sonarman 3/c, Honorable Discharge.

May 1946 to June 1948 - R. and N. Combustion Co., 371 E. 135 Street, Bronx, New York, Steamfitter helper.

July 1948 to date - Consolidated Edison Company of New York, Inc.

July 1948 - Started in the Company in the Production Department as a Boiler cleaner assigned to boiler cleaning at the Hudson Avenue Station.

August 1950 - Title changed to Electric Mechanic B and assigned to maintenance of Station equipment at Gold Street Station.

April 1952 - Transferred to Hudson Avenue Station and assigned to maintenance of station equipment.

March 1953 - Promoted to Electric Operator A assigned as a galleryman to the Low Tension Control Board, holding off low tension equipment, equipment inspection and cleaning.

January 1956 - Promoted to Sr. Production Operator A assigned as assistant operator on the Low Tension Control Board regulating voltages on auxiliary equipment, starting and stopping frequency changer sets, motor generation sets and house turbines.

March 1960 - Promoted to Sr. Electric Operator B assigned as assistant operator on the High Tension Control Board regulating bus voltages, synchronizing generators to the bus, holding off high tension equipment, and conducting thyratron and kenetron tests on high tension feeders.

August 1961 - Transferred to Indian Point Station with the title of Operator B. In 1961 aided in preliminary testing of equipment and attended training school for reactor operators. In 1962 aided in systems testing, fuel assembly, fuel loading, and initial facility startup, and trained as operator on the controls of the nuclear plant under the supervision of a licensed facility operator. In 1963 he continued training for reactor operation under supervision of a licensed operator. Has participated in preparations for reactor start, in reactor starts, operation and shutdown.

August 1963 - Attended a 72 hour training course conducted by the Consolidated Edison Company of New York Inc. The course consisted of 20 hours of instruction on radiological safety practices and 52 hours of instruction on nuclear reactor theory and design.

December 1963 - Obtained operator license No. OP-1638, issued by the Commission.

July 1964 - Assumed the duties of Control Operator A and functioned as a licensed operator in the manipulation of the controls of the facility.

During 1965, received an additional 100 hours of individual instruction in those areas of knowledge required by Paragraph 55.22 of the Atomic Energy Commission's regulations.

April 1966 - Received a Senior Reactor Operator License for Unit No. 1.

April 1968 - Promoted to Watch Foreman.

April 1968 - Attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of Indian Point Unit No. 2.

June 1968 to date - Has actively participated in Indian Point Unit No. 2 startup program. This participation includes preparation of written system descriptions and procedures, involvement in plant pre-operational testing and attending the Unit No. 2 formal training program.

June 1968 to date - Periodically supervise the manipulation and controls of the Unit No. 1 facility.

Max E. Hughes - Watch Foreman

Education:

Shopville High School, Shopville, Kentucky - from 1936 to 1940, Graduate.

Experience:

December 1940 to January 1947 - U.S. Navy, Signalman 1/c, Honorable Discharge.

January 1947 to date - Consolidated Edison Company of New York, Inc.

January 1947 - Employed as utility mechanic and assigned to Sherman Creek Station in the coal handling crew.

March 1948 - Promoted to Electric Operator B and assigned as generator brushman and wiper.

February 1950 - Promoted to Electric Operator A assigned to assist the high tension control board operators in switching operations

October 1953 - Title changed to Sr. Electric Operator C and assigned to maintenance and operation of the H.P. Station auxiliaries. Title for this assignment changed April, 1954 to Sr. Production Operator A.

July 1961 - Transferred to Indian Point Station with the title of Operator B. In 1961 aided in preliminary testing of equipment and attended training school for reactor operators.

In 1962 aided in systems testing, fuel assembly, fuel loading, and initial facility startup, and trained as operator on the controls of the nuclear plant under the supervision of a licensed facility operator.

In 1963 continued training for reactor operation under supervision of a licensed operator. Participated in preparations for reactor start, in reactor starts, operation and shutdown.

July 1963 - Obtained operator's license No. OP-1570, issued by the Commission.

April 1966 - Obtained Senior Reactor Operators License for Unit No. 1.

February 1964 - Assumed duties of Operator A functioning as a licensed operator, in manipulation of controls of the Indian Point Unit No. 1 Facility.

April 1968 - Promoted to Watch Foreman.

April 1968 - Attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of Indian Point Unit No. 2.

June 1968 to date - Has actively participated in Indian Point Unit No. 2 sta   up program, this includes preparation of written system descriptions and procedures, involvement in plant pre-operational testing and attending the Indian Point Unit No. 2 formal training program, periodically supervising the m   ation and controls of Indian Point Unit No. 1 Facility.

Charles R. Powell, Sr. - Technical Assistant

Education:

Niagara Falls High School, Niagara Falls, New York from 1940 to 1943 Graduate with Equivalency Diploma 1948.

Experience:

November 1942 to July 1962 - Active duty U.S. Navy

Completed following Military courses:

Machinist Mate - 4 months
Oil Burning - 2 months
Electric Tech. - 1 year
Nuclear Power - 1 year

August 1962 to date - Consolidated Edison Company of New York, Inc.

August 1962 - Employed as a Technical Assistant and assigned to Indian Point Station Chemical Division.

December 1966 - Under the supervision of a licensed facility Senior Operator received training on the controls of the nuclear and conventional plants of the Indian Point Unit No. 1 Facility.

October 1967 - Obtained Senior Reactor Operator License for Unit No. 1.

April 19[illegible] - Attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of the Indian Point, Unit No. 2. Facility.

June 1968 - Since this date has actively participated in the Indian Point, Unit No. 2 Facility startup program. This has included preparation of written system descriptions and procedures, involvement in plant pre-operational testing and attending the Indian Point, Unit No. 2 Facility Formal Training Program.

Michael J. Anderson - Technical Assistant

Education:

Harverstraw High School, Haverstraw, New York
from 1953 to 1957, Graduate.

United States Navy
from August 1957 to August 1962, Honorable Discharge.

United States Navy Nuclear Power Training School,
New London, Conn.

Experience:

Prototype training at West Milton, New York on S3G.

Qualified as High Pressure Stainless Steel Welder at West Milton.

Boiler and radio chemistry on USS Seawolf. Attended Health Physics courses on USS Fulton.

September 1962 to date - Consolidated Edison Company of New York, Inc.

September 1962 - Employed as a Technical Assistant and assigned to Indian Point Station Chemical Division.

February 1967 - Under the supervision of a licensed facility senior operator received training on the controls of the nuclear and conventional plants of the Indian Point, Unit No. 1 Facility.

October 1967, obtained Operator License No. OP-2317, issued by the Atomic Energy Commission.

April 1968 - Obtained Senior Reactor Operator License for Unit No. 1.

April 1968 - Attended a six week course sponsored by Westinghouse Electric Corp. covering the design and operation of the Indian Point, Unit No. 2 Facility.

June 1968 - Since this date has actively participated in the Indian Point, Unit No. 2 Facility startup program. This has included preparation of written system descriptions and procedures, involvement in plant pre-operational testing and attending the Indian Point, Unit No. 2 Facility formal training program.

Charles E. Agan - Production Engineer (Training Supervisor)

Education:

Harold G. Hoffman High School, South Amboy, New Jersey
from 1954 to 1958, graduate.

Pennsylvania State University, University Park, Pennsylvania
from 1958 to 1963, B.S. M.E.

Experience:

June 1963 to March 1969 - Jersey Central Power and Light Company

June 1963 - Assigned to the Sayreville Generating Station.

May 1965 - Assigned to the Yards Creek Pumped Storage Project as a Shift Supervisor during the initial plant start-up and operation.

June 1966 - University of Michigan Seminar - "Elements of Nuclear Power Reactor Engineering".

July 1966 - Assigned to the Oyster Creek Nuclear Generating Station as Assistant Technical Engineer - Nuclear

September 1966 - Attended three month course sponsored by the General Electric Company covering nuclear fuel management and reactor start-up testing.

1967 - 1968 - Participated in courses at Oyster Creek covering reactor physics, gamma spectroscopy, BWR technology and radiation protection.

1968 - 1969 - Assumed duties of Operating Shift Supervisor at Oyster Creek during pre-operational testing phases.

1969 - Assisted in writing Oyster Creek core calculation procedures, surveillance testing procedures and plant operating instructions.

April 1969 - Supervised initial fuel inspection and fuel loading of Oyster Creek Unit No. 1.

May 1969 - Attended two week course sponsored by the General Electric Company at the BWR simulator, Dresden Station, for license applicants.

October 1969 - Received Senior Reactor Operators License SOP-1237 for Oyster Creek Unit No. 1.

1969 - 1970 - Participated in the start-up and commercial power operation of Oyster Creek Unit No. 1.

May 1970 to date - Consolidated Edison Company of New York, Inc.

May 1970 - Employed as a Production Engineer at the Indian Point Station as director of training for licensed reactor operators. Presently assigned to Unit No. 2 activities.

Philip J. Gaudio - Assistant Supervising Engineer

Education:

Samuel Gompers H.S., Bronx, N. Y. Graduated 1947.

Completed 16 month Radar Repair Course. The U.S. Army Signal School, Fort Monmouth, N. J. 1951-1952.

Completed 2 year evening course at Manhattan College, "Radiological Health and Science" - Sponsored by U.S. Public Health Service 1966-1967.

Experience:

Con Edison -

1947 to 1960 - Technician - testing of underground electric, gas and steam transmission facilities to prevent corrosion.

1961 to 1967 - Technician at Indian Point Construction and startup testing on conventional and nuclear control systems, rod drive system, nuclear instrument and safety system. Periodically assigned to health physics program.

1967 - Technician - Health physics, primarily assigned to special projects.

Thomas M. Law - Unit No. 1 Operations Superintendent

Education:

Cardinal Hayes High School Bronx, N. Y. from 1949 to 1953, graduate. United States Merchant Marine Academy, Kings Point, N. Y., 1953 to 1957, B. S. degree. Commissioned U.S. Naval Officer.

Experience:

September 1957 to March 1958 - Ensign, U. S. Naval Reserve, Active Duty. Assigned to U. S. S. Essex, CVA9, as Jr. Boiler Division Officer.

April 1958 to date - Consolidated Edison Company of New York, Inc.

April 1958 - started in the Company in the Production Department at East River Station as a Cadet Engineer. Assigned to observe boiler operating procedures and to make recommendations to improve performance. Also assigned, under a General Maintenance Foreman, to supervise a group of plant maintenance men. Later assigned to follow the conversion of high pressure [illegible] from coal to oil firing and to assist the [illegible] ation of this work with the Engineering and Construction Departments as the Production Department representative.

October 1959 - Assigned as a Cadet Engineer to the Station Construction and Shops Department, Mechanical Construction Bureau assisting in the supervision of re-building of a 50MW turbine at Sherman Creek Station.

February 1960 - Assigned to the Electric Construction Bureau of the Station Construction and Shops Department as a Cadet Engineer to follow electric construction in connection with the installation of #4 Unit at Astoria Station.

April 1960 - Returned to the Production Department at East River Station and resumed with increased responsibility the task of following the conversion of the boilers to oil firing.

November 1960 - Assigned with the title of Production Engineer to Indian Point Station to assume the duties of a Watch Foreman. In 1961 aided in preliminary testing of equipment and attended training school for reactor operators. In 1962 aided in systems testing, fuel assembly, fuel loading, and initial facility startup, and trained as Watch Foreman on the operation and maintenance of the nuclear plant under the supervision of a licensed facility operator. In 1963 continued training for the operation and maintenance of the nuclear plant under the supervision of a licensed operator. Participated in preparations for reactor start, in reactor starts, operation, shutdowns, and maintenance.

July 1963 - Obtained a Senior Reactor Operator's License for Unit No. 1 and assumed responsibilities of General Watch Foreman in charge of shift operations and maintenance on Unit No. 1.

December 1964 - Assigned to station staff as Production Engineer responsible for training of reactor operators.

January 1967 - Promoted to Superintendent - Performance- responsible for supervising and directing the activities of station technical personnel and supervising the training of Reactor Operator and Senior Reactor Operator license candidates. Responsible for the maintenance of all records and reports relating to inventory control of facility SNM.

October 1969 to date - Assumed the duties of Unit No. 1 Superintendent - Operations, responsible for the operation of the individual generating facilities (electrical, mechanical, and nuclear) and for all associated maintenance and janitorial activities.

Authur S. Darden - Assistant Maintenance Superintendent

Education:

University of Vermont - Graduated 1935, BSEE

Experience:

1936-1959

Employed by the Interborough Rapid Transit Co. (subsequently the NYC Board of Transportation and the NYC Transit Authority) in various power stations. Assigned to Operations Department until 1945. In 1945 was promoted to General Maintenance Foreman at 59th Street Station.

1959 - present

Con Edison of N. Y.

1959-1962 General Maintenance Foreman at 59th Street station.

1962-1967 Assigned to Indian Point plant as General Maintenance Foreman in charge of all nuclear and conventional plant maintenance activities.

1967

Promoted to Assistant Superintendent responsible for all maintenance ctivities at Indian Point plant.

Robert L. Simms - Assistant Superintendent Unit 1

Education:

Bryant High School, Bridge Plaza, Long Island City, New York, from 1931 to 1935, graduate.

U. S. Navy, February 1941 to October 1945 Chief Machinist Mate, U.S.S. Baham and Floyd Bennet Field; attended Diesel School, Internal Combustion School, Goodyear School and others.

Experience:

September 1935 to May 1936 - Pask and Walbridge, 1 Wall Street, N. Y. C., Runner and Bookkeeper Asst.

June 1936 to October 1936 - Daly's Chevrolet, Flushing, New York, Painter's Assistant.

October 1936 to date - Consolidated Edison Company of New York, Inc.

October 1936 - Started in the Company as a Messenger in the General Office Service Department.

January 1937 - Transferred as a Jr. Clerk to the Accounting Department.

July 1938 - Promoted to Clerk assigned to the Commercial Relations Department.

February 1940 - Promoted to Collection Clerk.

March 1940 - Transferred to Production Department, Hell Gate station as a janitor.

February 1941 to October 1945 - U. S. Navy, Chief Machinist Mate, Honorably Discharged.

October 1945 - Returned from Military Service and assigned to Hell Gate Station as an Auxiliary Operator B for vacation relief and assigned to the operation of main turbines and auxiliaries.

April 1946 - Promoted to Auxiliary Operator B with same assignments.

November 1949 - Temporarily assigned as a Watch Foreman. At various times was in charge of the operation of the turbine room, pump room and boiler room.

May 1950 - Transferred to Sherman Creek for vacation relief as a Watch Foreman assigned to the Turbine Room.

November 1950 - Promoted to Watch Foreman assigned to the Pump Room and Boiler Room.

February 1956 to March 1958 - Assigned as a Maintenance Foreman supervising maintenance of station equipment.

In 1960 assigned for approximately 12 weeks as a General Watch Foreman relief in charge of the operation of the steam station on the various watches.

October 1960 - Assigned to Indian Point Station as a Watch Foreman. In 1961 aided in preliminary testing of equipment and attended training school for reactor operators. In 1962 aided in systems testing, fuel assembly, fuel loading, and initial facility startup, and trained as Watch Foreman on the operation and maintenance of the nuclear plant under the supervision of a licensed facility operator. In 1963 has continued training for the operation and maintenance of the nuclear plant under the supervision of a licensed operator. Has participated in preparations for reactor start, in reactor starts, operation, shutdowns, and maintenance.

July 30, 1963-Received Operator's License No. OP1593 issued by the Commission.

August 1963 - Attended a 72 hour training course conducted by the Consolidated Edison Company of New York, Inc. The course consisted of 20 hours of instruction on radiological safety practices and 52 hours of instruction on nuclear reactor theory and design.

December 24, 1963 - Received Senior Operator's License No. SOP-359 issued by the Commission

January 1965 - Promoted to General Watch Foreman in charge of shift operation and maintenance of Unit 1 at Indian Point Station.

January 1966 - Promoted to Assistant Superintendent of Unit 1 at Indian Point Station which includes maintenance and operations of the nuclear and conventional plants.

Dominic Joseph Sarc - General Watch Foreman

Education:

Brooklyn Automotive High School, Nassau Street, Brooklyn, New York from 1941 to 1945, graduate.

Pierce Technical School, Astor Place, New York City from 1947 to 1948, radio and television, graduate.

Experience:

April 1945 to November 1946 - U.S. Army, 11th Armored Infantry Battalion, Rifleman - PFC, Honorable Discharge.

November 1946 to May 1947 - Machinist, Rex Slide Fastener Company New York City, New York

May 1947 to November 1947 - Assembly line inspector, Garod Radio Corporation, Washington Street, Brooklyn, New York.

March 1948 to April 1949 - Television repairman, Friendly Frost, Myrtle Avenue, Brooklyn, New York

November 1949 to date - Consolidated Edison Company of New York, Inc.

November 1949 - started in the Company in the Gas Production Department at Astoria Gas Plant as a Workman C assigned as an oiler on purifier exhausters, tar processing, and other by-product equipment.

January 1952 - transferred on loan to Mechanical Construction Bureau, Station Construction and Shops Department for fire watching and safety watch on new construction of No. 4 Unit, Astoria Station.

December 1952 - returned to Gas Production Department, Astoria Gas Plant, as an oiler.

April 1953 - transferred to Hudson Avenue Station, Production Department as Electric Mechanic B and assigned to maintenance of station electric equipment.

September 1955 - promoted to Production Operator B assigned as a galleryman to the Low Tension Control Board, holding off low tension equipment, equipment inspection and cleaning.

June 1956 to July 1959 - worked as a Senior Production Operator from time to time for vacation relief working as an assistant operator on the Low Tension Control Board.

November 1959 - promoted to Senior Production Operator A and assigned as assistant operator on the Low Tension Control Board regulating voltages on auxiliary equipment, starting and stopping frequency changer sets, motor generator sets and house turbines.

November 1960 transferred to Indian Point Station with the title of Operator A. In 1961 aided in preliminary testing of equipment and attended training school for reactor operators. In 1962 and 1963 aided in systems testing, fuel assembly, fuel loading, and initial facility startup, and trained as operator on the controls of the nuclear plant under the supervision of a licensed facility operator. Since obtaining operator license on July 9, 1963 has been functioning as a licensed operator in the manipulation of the controls of the facility.

July 9, 1963 received an Operator's License for Indian Point Unit No. 1.

May 19, 1964 received a Senior Reactor Operator License for Indian Point Unit No. 1.

November 7, 1965 promoted to General Watch Foreman.

Alexander Bobik - General Watch Foreman

Education:

Maynard High School, Maynard, Mass.
from 1936 to 1940, graduate.

Experience:

June 1940 to December 1940 - Maynard Woolen Mills, Card Room Machine Operator.

December 1940 to September 1942 - Maintenance Man in various hotels in Florida.

September 1942 to November 1945 - Seaman, U.S. Coast Guard, Naval Ammunition Depot, Sandy Hook, New Jersey.

November 1945 to March 1946 - American Railway Express Company, Terminal, Jersey City, New Jersey, package sorter.

March 1946 to date - Consolidated Edison Company of New York, Inc.

March 1946 - started in the Company in the Production Department as a Boiler Cleaner B assigned to boiler cleaning at the Hudson Avenue Station.

September 1947 - promoted to Boiler Operator assigned to the operation of stoker fire boilers, coal lorries and the forced and induced draft fans.

January 1951 - promoted to Sr. Boiler Operator B assigned as a watertender or in charge of the operation of four coal and oil-fired boilers.

October 1953 - title changed to Sr. Boiler Operator B-HP and assigned as a control room operator responsible for the efficient operation of No. 10 turbo-generator and boiler.

July 1956 - promoted to Watch Foreman, directing the operation of 16 boiler feed pumps, 8 condensers and turbine auxiliaries of 8 turbo-generators.

March 1958 - assigned as Watch Foreman in charge of the turbine room with 8 turbo-generators having a capacity of 880 MW.

January 1960 - assigned as Watch Foreman responsible for the operation of No. 10 Unit.

October 1960 to date - transferred to Indian Point Station as a Watch Foreman. In 1961 aided in preliminary testing of equipment and attended training school for reactor operators. In 1962 aided in systems testing, fuel assembly, fuel loading, and initial facility startup, and trained as watch foreman on the operation and maintenance of the nuclear plant under the supervision of a licensed facility operator. In 1963 continued training for the operation and maintenance of the nuclear plant under supervision of a licensed operator. Participated in preparations for reactor start, in reactor starts, operation, shutdowns, and maintenance.

January 7, 1964 received an Operator's License for Indian Point Unit No. 1.

February 1, 1967 received a Senior Reactor Operator License for Indian Point Unit No. 1.

Promoted to General Watch Foreman on April 30, 1967.

Jerome Augustine Walden - General Watch Foreman

Education:

DeWitt Clinton High School, Bronx, New York
from 1940 to 1943 - Non Graduate.

U. S. Navy
from January 1943 to February 1946

U. S. Navy Ship Fitters School, Pearborh Mich.,
from June 1943 to October 1943, Graduate.

Experience:

January 1942 to January 1943 - Plumbers Apprentice.

March 1946 to June 1947 - Plumbing and Steam Fitting Business

June 1947 to date - Consolidated Edison Company of New York, Inc.

June 1947 - Employed as Boiler Cleaner B and assigned to Hudson Avenue Station boiler cleaning crew.

May 1948 - Transferred to Sherman Creek Station and assigned to boiler cleaning crew.

September 1950 - Assigned to Water Treatment Plant, HP and LP Boiler Feed Pumps, HP and LP auxiliaries, and to the operation of the main turbo-generators.

August 1951 to August 1952 recalled to U.S. Navy.

February 1959 - Assigned to Boiler Control Board for No. 90 and No. 100 HP Boilers.

September 1961 - Title changed to Watch Foreman. At various times was in charge of the operation of the turbine room, pump room and boiler room.

June 1963 - Transferred to Indian Point Station as a Watch Foreman. Trained as Watch Foreman on the operation and maintenance of the nuclear and conventional plant under the supervision of a licensed senior operator.

April 3, 1968 received a Senior Reactor Operator License for Indian Point Unit No. 1.

Promoted to General Watch Foreman on November 1, 1969.

John Isaac Pruyn - General Watch Foreman

Education:

Gorton High School, Yonkers, New York from 1929 to 1933, Graduate.

Textile High School, W. 18 Street, New York City from 1938 to 1939, Electrical Course, Graduate.

Experience:

June 1934 to date - Consolidated Edison Company of New York, Inc.

June 1934 - Started in the Company in the Commercial Department as a Jr. Clerk.

October 1938 - Transferred to Sub-station operation Department as a Sub-Station Assistant Operator.

August 1948 - Transferred Glenwood Station as an Electric Operator A assigned to switching operations and maintenance of electric equipment.

September 1951 - Promoted to Sr. Electric Operator C assigned to the High Tension Control Room assisting the Operator in regulating voltages, controlling the operation of turbo-generators and performing switching operations.

April 1954 - Transferred to Hell Gate Station with a title change to Sr. Production Operator A and assigned to the maintenance of high and low tension equipment.

June 1954 - Title changed to Electric Mechanic A assigned to the maintenance of high and low tension equipment.

From August 1958 to June 1959 - Temporarily assigned to Glenwood Station as a Sr. Electric Operator A in the High Tension Control Room, in charge of regulating bus voltage turbo-generator operation and switching operations.

June 1959 - Returned to Hell Gate Station as an Electric Mechanic A when Glenwood Station was shut down.

January 1960 to April 1960 - Assigned to Glenwood Station as Sr. Electric Operator B assisting the operator in the High Tension Control Room.

April 1960-Returned to Hell Gate Station as an Electric Mechanic A.

May 1961 to July 1961 - Assigned as Sr. Electric Operator B in the Hell Gate Station High Tension Control Room, assisting the control room operator.

July 1961 - Transferred to Indian Point Station with the title of Operator B. In 1961 aided in preliminary testing of equipment and attended training school for reactor operators. In 1962 aided in systems testing, fuel assembly, fuel loading, and initial facility startup, and trained as operator on the controls of the nuclear plant under the supervision of a licensed facility operator. In 1963 has continued training for reactor operation under supervision of a licensed operator. Has participated in preparations for reactor start, in reactor starts, operation and shutdowns.

May 1963 - promoted to Acting Operator A Indian Point Station. Working as Central Control Room Operator on the controls of the nuclear and conventional plants under the direct supervision of a licensed facility operator.

October 1964 received an Operator License for Indian Point Unit No. 1.

April 6, 1966 received a Senior Reactor Operator License for Indian Point Unit No. 1.

Promoted to Watch Foreman on April 30, 1967.

Promoted to General Watch Foreman on November 1, 1969.

Thomas William Keith - General Watch Foreman

Education:

Roosevelt High School, Tuckahoe Road and Central Ave. Yonkers, New York from 1932 to 1936.

ICS Course, Power House Engineering, Home Study from 1950 to 1952, Diploma

High School Equivalency Diploma issued in 1953 by New York State Education Department.

Experience:

1936 to 1937 - worked as a carpenters helper for a private contractor.

1938 to May 1939, L. Oppenheimer and Co., 133st. and Broadway New York City, warehouseman.

May 1939 to March 1941, Dugan Bros. Mount Vernon, Route Salesman.

March 1941 to June 1945 - U.S. Army, 4th. Medical Battalion, Honorable Discharge.

July 1945 to date - Consolidated Edison Company of New York, Inc.

July 1945 started in the company in the Production Dept. at Sherman Creek Station as a Mechanic B assigned to maintenance of electric equipment.