(17)

Q 12.1-60 to

Q 14.14 (a)-4

10625

10625

May 1946 promoted to Electric Operator B and assigned to the Electric Division as a brushman and wiper.

November 1947 promoted to Electric Operator A and assigned as a galleryman to assist the high tension control board operators on high tension switching.

February 1950 promoted to Sr. Electric Operator C and assigned to electric maintenance of No. 9 and 10 high pressure unit equipment.

September 1951 promoted to Electric Mechanic A and assigned to maintenance of all station electric equipment.

May 1953 transferred to Astoria Station in training for the operation of this new plant.

October 1953 title changed to Operating Mechanic and assigned to the operation and maintenance of steam and electric auxiliary equipment.

December 1955 promoted to Operator B and assigned to No. 1 and 2 Unit Control Room, operating the controls for the turbo-generator, boilers and associated auxiliaries.

September 1958 promoted to Operator A assigned in charge of the No. 1 and 2 unit control board.

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 opertor. Since obtaining operator license on July 5, 1963 has been functioning as a licensed operator in the manipulation of the controls of the facility.

July 5, 1963 received an Operator's License for Indian Point Unit No. 1.

October 2, 1964 received a Senior Reactor Operator License for Indian Point Unit No. 1.

Promoted to Watch Foreman on November 22, 1964.

Promoted to General Watch Foreman on January 1, 1966.

George S. Case, Jr. - Production Engineer

Education:

Iona Prep., New Rochelle, New York from 1950-1954

St. Bonaventure University, St. Bonaventure, N. Y. from 1954-1959

Experience:

I June 1967 to Date - Consolidated Edison Company of New York, Inc.

June 1967 - Employed as Production Engineer. Assigned to Major Maintenance Bureau. Placed in charge of 100 maintenance mechanics at East River Station for Boiler, Turbine overhaul and Capital Work.

November 1967 - Transferred to Hudson Avenue Station. Assumed charge of 125 man maintenance force - Boiler & Turbine overhaul work.

January 1968 - April 1968 - Assigned to various repair jobs at Arthur Kill, Ravenswood and Hudson Avenue.

April 1968 - Transferred to Major Maintenance Bureau Office. Assigned as Assistant to General Superintendent of Bureau of Critical Path Planning, Scheduling and Material Procurement for Turbine and Boiler overhauls. Conducted Bureau Liaison with

General Electric, Westinghouse and Combustion Engineering. Aided with Manpower Allocation (550 man bureau) and personnel relations. Represented General Superintendent at Bureau Head's meetings in his absence. Worked up annual budget forecasts for Bureau. Assembled proposals for Bureau streamlining, modernization and cost reduction.

As collateral duty handled all repair, dry-docking and maintenance of Con Edison oil barge fleet in accordance with Coast Guard requirements.

January 1970 - Transferred to Indian Point Station. Assigned to Unit No. 1 for Inst[illegible]ion and Licensing.

March 1970 - Assigned to Unit No. 2 at Indian Point. Present assignment includes adaptation of Quality Assurance requirements to the station's needs in areas of in-service inspection, maintenance work, material procurement (spare parts) and record keeping. In addition has been assigned to develop test procedures for test committments noted in FSAR and attend classes of instruction for Operator Licensing on Unit No. 2.

II March 1963 - June 1967 - General Dynamics/Electric Boat, Groton, Conn.

March 1963 - March 1964 - Assigned to S5W Project as Design Liaison and Reactor System Test Engineer. Performed Liaison work on 4 reactor systems under construction and performed hydrostatic tests on 2 Reactor plants undergoing overhaul. Completed training course on S5W Reactor Plant (Evening session).

March 1964 - February 1966 - Senior Engineer and Electric Boat member and Chairman of Joint Test Group for Reactor System Repairs and Tests on 4 consecutive availabilities. Coordinated all reactor system work through critical path planning and scheduling. Wrote required test procedures and supervised shift test engineers. Arranged for written procedures where required for Reactor System Quality Assurance.

February 1966 - June 1967 - Project Officer for Major overhauls (included primary system de-contamination and refueling) of S5W Reactor Plants. Responsible for screening all U.S. Navy Work Specifications and promulgating all authorizing paperwork required to accomplish overhaul work on

Reactor Systems. This included initiation, and authorization of material purchase orders, quality assurance procedures, special blueprints. Assisted Planning Department in maintaining critical paths. Supervised design liaison work.

III September 1959 - September 1962 - Boston Naval Shipyard, Boston, Massachusetts

Assigned to Production Department as a Ship Superintendent. Was responsible for co-ordination of shipyard trades to effect timely completion of thirty-two naval ship availabilities. Availabilities ranged through conversions, overhauls, repairs and outfittings covering main propulsion and auxiliary machinery, drydock work, sonar installations,guided missile and electronic guidance system modifications, gun mount and fire control systems overhaul.

Brendan T. Moroney - Production Engineer

Education:

Bishop Loughlin H. S., Brooklyn, N. Y. 1953-1957, Graduate.

Holy Cross College, Worcester, Mass., 1957-1961, Graduate, B. A. Mathematics.

Experience:

June 1961 - August 1969, Officer, United States Navy

1961-1963 - Division Officer, Communications

1963-1964 - Staff Communications and Personnel Officer, Destroyer Squadron

1964-1965 - USN, Nuclear Power School at Mare Island and SIC Prototype at Windsor. Qualified S5W & SIC Reactor Operator.

1965 - USN, Submarine School at New London

1966-1969 - USS Haddock (S5W), Reactor Control and Electrical Division Officer

August 1969 to present - Participating in Indian Point Unit No. 2 Start-Up Program. Including preparation of written system descriptions, plant operating procedures and active involvement in all phases of the pre-operational test program. Attendance at formal facility educational and training lectures.

QUESTION 12.2 WRITTEN PROCEDURES

1. Designate those safety related operations for which written procedures will be required.

2. Describe the review and approval procedure for all written procedures, including the means for assuring that these procedures have been distributed, read, and understood by the appropriate operating personnel.

ANSWER

1. In addition to detailed written procedures covering overall plant operations and individual systems operations; written procedures will be developed for certain operating conditions of an emergency or infrequent nature which may occur. These procedures will establish as nearly as possible the course of action to be followed in these situations. Operations will be in substantial accordance with the procedures and will always be in accordance with the basic design parameters and limitations to which the affected and associated systems were designed. Specific procedures that will be written to cover these emergency or infrequent operating conditions are as follows:

    1. Loss of Reactor Coolant
    2. Steam Feedwater Line Break
    3. Steam Generator Tube Failure
    4. Station Blackout Operation
    5. Control Room Inaccessibility
    6. Loss of Component Cooling Loop Flow
    7. Drop of a Full Length RCCA
    8. Loss of Reactor Coolant Flow
    9. Component Cooling Loop Pipe Break
    10. Malfunctioning Rod Position Indicator
    11. Continuous Rod Withdrawal
    12. Malfunctioning Rod Cluster Control
    13. Nuclear Instrumentation Malfunction
    14. High Reactor Coolant Activity
    15. Malfunction of Pressurizer Power Operated Relief or Safety Valves

16. High Activity - Radiation Monitoring System
17. Malfunction of Residual Heat Removal Loop
18. Failure of Spent Fuel Pit Cooling
19. Malfunction of Reactor Control System
20. Emergency Boration
21. Malfunction of Reactor Control
22. Letdown and Charging Line Out of Service
23. Malfunction of Reactor Coolant Pump Seal
24. Plant Flooding
25. Emergency Shutdown
26. Isolation Valve Seal Water System
27. Steam Generators - Blowdown System
28. Auxiliary Feedwater System
29. Condenser Air Removal System
30. Condensate Systems
31. Service Water and Cooling Water Systems

2. Written procedures are being developed by the Unit No. 2 startup group as part of the overall training effort. Included in this group, all of which hold Senior Reactor Operators licenses for Unit No. 1, are certain key staff personnel and those who will assume Unit No. 2 Watch Foreman and control room operator responsibilities.

Once a procedure has been developed, it will be given a thorough review within the group for accuracy and completeness before submittal to the General Superintendent and Nuclear Facilities Safety Committee for final approval.

Approved procedures will be distributed to all licensed Reactor and Senior Reactor Operators and will also be maintained at various key operating locations within the facility such as the General Watch Foremens Office, the Central Control Room, etc. Distribution of procedures will be effected by means of a formal distribution list which will be kept up to date by the Station office force.

It will be the prime responsibility of each Unit No. 2 Watch Foreman to assure himself that all watch personnel directly responsible to him have read the procedures and are thoroughly cognizant of their contents.

From time to time, it is expected that specific operating instructions will be needed to accomplish certain operating needs. Such instructions will be considered as supplemental to the basic operating procedures, and will be issued to the operating personnel after review and approval by the General Superintendent and appropriate members of his operating staff. Deviations from or modifications to the basic operating procedures, however, will only be made following review and approval by the General Superintendent and the Nuclear Facilities Safety Committee.

QUESTION 12.3

Provide the following information relative to the Nuclear Facility Safety Committee:

1. Define the required quorum for the committee to conduct its business.

ANSWER

Quorum

A majority of the full committee members which shall include the chairman or the Vice Chairman and of which a minority are from the Nuclear Power Generation Department shall constitute a quorum for meetings of the full committee.

QUESTION 12.3

Provide the following information relative to the Nuclear Facility Safety Committee:

2. Define the criteria to be used for committee membership.

ANSWER

Membership

The Committee shall have a membership of at least 12 persons of which a majority are independent of the Nuclear Power Generation Department and shall include technically competent persons from all departments of Consolidated Edison having a direct interest in nuclear plant design, operation or in nuclear safety. The Chairman and Vice Chairman will be Senior Officials of the Company experienced in the field of nuclear energy.

The Committee shall consist of:

The Chairman who shall be appointed by the Chairman of the Board or the President of the Company.

The Vice Chairman who shall be appointed by the Chairman of the Board or the President of the Company.

The Secretary who shall be appointed by the Chairmen of the Committee.

The following Committee Members shall be designated by the Vice President of the Company who is responsible for the functioning of the department or position stated below with the approval of the Chairman:

The Radiation Safety Officer of the Company.

A medical doctor from the Medical Department having experience in nuclear medicine.

A representative from the Mechanical Engineering Department having experience in nuclear engineering with special emphasis on reactor physics.

A representative from the Nuclear Power Generation Department having experience in nuclear chemistry.

An engineer from the Fuel Department having experience with nuclear fuel.

An engineer from the Electrical Engineering Department having experience in electrical engineering related to nuclear power plants with special emphasis on instrumentation and control.

An engineer from the Mechanical Engineering Department having experience in mechanical engineering related to nuclear power plants with special emphasis on heat transfer.

A representative from the Civil Engineering Department having experience in environmental engineering.

A lawyer from the Law Department who shall be familiar with legal matters affecting nuclear power plants.

The Manager of the Nuclear Power Generation Department.

The Manager of the System Operation Department.

The Reactor Engineer at the Indian Point Station.

Outside consultants as required, appointed by the Chairman without the right to vote.

The Nuclear Plant General Superintendent as a participant without the right to vote.

Each member will designate a permanent alternate to serve in his absence. The name of the alternates will be filed with the Chairman. Only the permanent member, however, will have the right to vote.

QUESTION 12.3

Provide the following information relative to the Nuclear Facility Safety Committee:

3. What will be the minimum meeting frequency of the Committee?

ANSWER

Minimum Meeting Frequency

The Committee shall meet not less frequently than quarterly, and at more frequent intervals at the call of the Chairman or in his absence the Vice Chairman, as required.

QUESTION 12.3

Provide the following information relative to the Nuclear Facility Safety Committee:

4. Describe more fully the specific functions and responsibilities of the committee.

ANSWER

The Committee will:

Not less than once each year audit and report the adequacy of all procedures used in the operation, maintenance and environmental monitoring of each nuclear power plant. The audits will include onsite inspections and verifications that procedures are adhering to the Operating Licenses and Technical Specifications.

Review and report upon each emergency or infrequent condition relating to nuclear safety including as a minimum those abnormal occurrences defined in the facilities Technical Specifications.

Review and report upon the adequacy of all proposed changes in plant facilities or procedures pertaining to the operation, maintenance and environmental monitoring having safety-significance, or which may constitute an "unreviewed safety question" as defined in Part 50, Title 10, Code of Federal Regulations.

Review and report upon the adequacy of nuclear safety provisions for all tests and experiments and results thereof, when such tests or experiments may constitute an "unreviewed safety question" as defined in Part 50, Title 10, Code of Federal Regulations.

Conduct not less than quarterly unannounced spot inspections of plant and monitoring operations and report the results thereof.

Review and report upon any activity, the occurrence or lack of which may affect the safe operation of the nuclear plants.

Review and report upon all proposed changes to the Technical Specifications or licenses.

At the request of the Nuclear Power Generation Manager or a Nuclear Plant General Superintendent the Committee will be promptly convened to review and act upon those nuclear safety matters deemed essential to the safe operation of the facility.

The Nuclear Facilities Safety Committee is constituted to advise the Executive Vice President, Central Operations, concerning the safety aspects of the operation of the nuclear power facilities. The Committee shall report to the Executive Vice President, Central Operations.

The Executive Vice President, Central Operations is responsible for the design, construction, operation and maintenance of nuclear power generation plants.

The Vice President, Power Supply and, under him, the Assistant Vice President in charge of Power Generatior, the Nuclear Power Generation Manager and Nuclear Plant General Superintendent are responsible for the day-by-day operation and maintenance of the plant. The Nuclear Facilities Safety Committee herein established will advise the Executive Vice President and through him the President and the Chairman of the Board concerning the safety aspects of the nuclear plant operation. The Safety Committee is to be kept fully and currently informed by the Nuclear Power Generation Manager and Nuclear Plant General Superintendent of all matters bearing on the safe operation of the plant. The Chairman may establish subcommittees and designate members of the full committee for assignment to the subcommittees. The Chairman, Vice Chairman and Secretary of the Nuclear Facilities Safety Committee are ex-officio members of all subcommittees.

If the Nuclear Plant General Superintendent decides to make a change in the facility or operating procedures, or to conduct a test or experiment, and concludes that the proposed change, test or experiment does not involve a change in the Technical Specifications or an unreviewed safety question, he may order the change, test or experiment to be made, shall enter a description thereof in the operating records of the facility, and shall send a copy of the instructions pertinent thereto, to the Chairman of the Nuclear Facilities Safety Committee. If the Chairman of the Committee, upon reviewing such instructions, is of the opinion that the change, test or experiment is of such a nature as to warrant consideration by the Committee, he shall order such consideration. If the Nuclear Plant General Superintendent desires to make a change in the facility or operating procedures or to conduct a test or experiment which in his opinion might involve a change in the Technical Specifications, might involve an unreviewed safety question or might otherwise not be in accordance with said License, he shall not order such change, test or experiment until he has referred the matter to the Nuclear Facilities Safety Committee for review and report. If the Committee is of the opinion that the proposed change, test or experiment does not require approval by the Atomic Energy Commission under the terms of said License, it shall so report in writing to the Nuclear Plant General Superintendent, together with a statement of the reasons for the Committee decision and the Nuclear Plant General Superintendent may proceed with the change, test or experiment. If, on the other hand, the Committee is of the opinion that approval of the Atomic Energy Commission is required, the Committee shall prepare a request for such approval, including an appropriate safety analysis in support of the request, and forward its report and request to the Vice Presidents in charge of Engineering, Power Supply and the Executive Vice President, Central Operations for their review with a copy to the Nuclear Plant General Superintendent. One of said Company Officers shall thereupon forward the report and request to the Atomic Energy Commission for approval unless, after review, the Executive Vice President, Central Operations either (a) disagrees with the opinion of the Committee that approval of the Atomic Energy Commission is required, or (b) decides that the proposed change, test or experiment is not necessary from the standpoint of Company policy or operations.

QUESTION 12.3

Provide the following information relative to the Nuclear Facility Safety Committee:

5. What are the recording and reporting requirements for the Committee?

ANSWER

Records

Draft minutes of all meetings will be distributed promptly to each committee member for comment and any corrections. Amended and corrected minutes will be circulated to committee members for final approval. Copies of approved minutes will be promptly distributed to each committee member and to the Assistant Vice President, Power Generation, the Vice President, Power Supply, the Executive Vice President, Central Operations, the President, the Chairmen of the Board, and the Corporation Secretary.

Special reports shall be in writing and will be distributed to all members of the Committee, the Assistant Vice President, Power Generation, the Vice President, Power Supply, the Executive Vice President, Central Operations, the President, the Chairman of the Board, and the Corporation Secretary.

QUESTION 12.4

Records

1. What records are kept of abnormal occurrences, principal maintenance activities and fuel inventories and transfers?

ANSWER

Records

a) Abnormal occurrences

All abnormal occurrences which occur during the course of facility operations will be recorded in the General Watch Foreman's log book and where appropriate, in the log books maintained by the licensed operator in the Central Control Room, the shift chemist and the shift health physics technician. In addition, pertinent charts will be marked as to date and time with a brief notation relative to the circumstances of the occurrence.

b) Principal maintenance activities

All principal maintenance activities are to be recorded in a set of equipment history books which will be maintained by the station's Performance Group. Input data for these books will be taken from written "maintenance memoranda" issued by watch personnel, reviews of the General Watch Foreman's logbook, and frequent oral discussions with operating and maintenance supervisory personnel.

For maintenance activities relating to instrumentation and control equipment, instrument test reports will be made out whenever maintenance work is performed. These reports will be maintained on file by our Technical Services Bureau.

c) Fuel inventories and transfers

Detailed records of total uranium, U235, Pu239 and Pu241 for all fuel in use or in storage will be maintained by the Performance Group. Records of fuel transfers will be maintained via proper execution of Form AEC-388. Specific locations for all fuel assemblies in the reactor core or in the fuel handling building storage pool will be maintained on appropriate core or fuel storage pool arrangement drawings.

QUESTION 12.5

Emergency Plan.

Provide an emergency plan including the following material.

- Organization: The emergency plan should include a description of the normal and emergency operating organizations.
- Spectrum of Accidents: The emergency plan should identify and consider the consequences of radiological accidents ranging from onsite accidents affecting only local operating personnel to design basis accidents which could affect both site and neighboring environment.
- Protective Measures: Attention should be given in the plan to immediate protective measures that may be necessary for site personnel and the general public.
- Technical Bases for Protective Measures: technical information necessary to support the proposed protective measures should be included.
- Protective Action Levels: The levels at which protective action is to be initiated should be specified in terms of instrumentation available in the post-accident instrumentation.
- Communications: The plan should specify emergency communication networks.
- Multi-Unit Site Considerations: A reactor site which includes more than one reactor should have both individual facility plans and a compatible site emergency plan.
- Offsite Support Groups: Provisions for utilization of offsite support groups should be included in the emergency plan.
- Medical Support: A summary of the medical facilities available should be included.
- Training Drills: The frequency and extent of drills should be included.
- Public Notification of Accident Conditions: The criteria for and method of notification should be included.
- Emergency Procedures: The emergency plan should clearly provide that procedures will be prepared to detail implementation of the emergency plan.
- Recovery Plan: General measures for a recovery plan and associated criteria and technical bases should be included.
- Review and Updating: Specific provisions for periodic formal review should be made.

ANSWER

See following report - "Contingency Plans Consolidated Edison Company of New York Inc. Indian Point Nuclear Generating Facility".

Supplement 5
1/70

CONTINGENCY PLANS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.

INDIAN POINT NUCLEAR GENERATING FACILITY

CONTINGENCY PLANS

TABLE OF CONTENTS

EARTHQUAKE CONTINGENCY PLAN

TABLE OF CONTENTS

1.0 INTRODUCTION

The Earthquake Contingency Plan describes the procedure which will be followed in the event of an earthquake at the Indian Point Site.

If an earthquake shock is felt or reported to occur in the vicinity of the Indian Point Station, the Company's seismological consultant will be contacted by the General Superintendent or his designated alternate for an evaluation of the magnitude of the earthquake at the Indian Point Site.

2.0 RESPONSIVE ACTIONS

The immediate protective actions taken by the key plant personnel are as follows:

<u>Central Control Room Operator</u>

1. Check the control boards to determine the effect, if any, on instrumentation, controls, and plant operations.

2. Notify the General Watch Foreman (or the Operations Superintendent if the earthquake occurs during the normal working hours when he is at the station).

3. If the unit has been shut down by the automatic protection systems, do not restart the unit until an inspection of the plant has been performed.

4. If the unit has not shut down, maintain the unit in its present operating condition.

<u>General Watch Foreman (Operations Superintendent)</u>

1. Order an inspection of all instruments, equipment and structures irreupective of whether or not unit has been shut down.

2. Notify Station Management.

3. If the unit has shut down, maintain shut down conditions until a complete evaluation has been made in consultation with the Station Management.

4. If the unit has not shut down, determine if it is to be shut down.

Operating Personnel

1. Check all instruments, and inspect all equipment and structures for malfunctions or possible damage.

2. Report findings to General Watch Foreman (Operations Superintendent).

Shift Health Physics Technician

1. Make a radiation survey of plant and adjacent areas.

2. Report findings to General Watch Foreman (Operations Superintendent).

Station Management

1. Determine if results of earthquake have damaged plant equipment.

2. Notify Company Management.

3. If the unit has shut down, determine, in consultation with the General Watch Foreman, whether it should be restarted.

3.0 SUBSEQUENT ACTIONS

The General Superintendent will, with the assistance of the plant operating staff and the Company's engineering forces, evaluate the subsequent reports on any possible damages to equipment and structures caused by the earthquake and will determine if any of the plant safeguards are damaged.

FIRE CONTINGENCY PLAN

Table of Contents

1.0 INTRODUCTION

Employees assigned to the station are trained in the areas of fire prevention and protection. General rules and regulations pertaining to the prevention of fires at any of the Company's generating facilities are enumerated in a handbook that each employee receives upon his assignment to a station. Supervisory personnel are responsible for assuring that all personnel under their jurisdiction comply with these general rules as well as others that are applicable to a particular job location. The Safety Services Bureau of the Company maintains a "fire-school" where qualified instructors are available to train and periodically retrain station employees in fire protection techniques. Invaluable experience is gained at this school inasmuch as actual fires of the type encountered in generating stations are controlled and extinguished by _ trainees.

2.0 ORGANIZATION

2.1 The following designated employees on all watches, except as noted, are assigned to a permanent Fire Fighting Brigade to form a nucleus around which an efficient fire fighting force can be organized for a fire contingency.

Title	Brigade Assignment
General Watch Foreman	In charge
Health Physics Technician	Report to General Watch Foreman
OMA - Rover	Report with CO_2 type extinguisher
CMB - Rover	Report with Chemox Mask
OMA's (Conventional Areas)	Report with Chemox Mask (12-8 and 4-12 watches only)
OMA's (Controlled Areas)	Report with CO_2 type extinguisher (12-8 and 4-12 watches only)
Chemical Technician	Report with CO_2 type extinguisher in Controlled Area or dry chemical type in Conventional Area
Mechanics (OMA's, Mech. A's, OMB's)	Report to General Watch Foreman at location of fire. See Fire Brigade schedule sheet for assignment. (8-4 watch only)
Production Men	Report to General Watch Foreman at location of fire. See Fire Brigade schedule sheet for assignment. (8-4 watch only)

Notes: 1. The assignments for Fire Brigade personnel are designated on the following weekly posted schedules:

Fire Brigade – Watch Personnel
Fire Brigade – Maintenance and Production Men Groups

Particular assignments of any men in these groups are also designated on the above mentioned schedule sheets.

2. Fire extinguishers from the Controlled Areas are not to be taken into the Conventional Areas.

2.2 On the 8–4 watch, a reserve pool of maintenance and production men personnel will be available when needed. A designated supervisor from the Maintenance group and the Foreman in charge of the production men group will contact the General Watch Foreman at the fire location and will secure additional personnel from their groups as required and assign duties to their men in accordance with the instructions of the General Watch Foreman. Except for those specifically designated on the Fire Brigade Schedule Sheet, personnel from these two groups shall not respond to fire alarms unless they are directly ordered to do so by their supervisors.

3.0 PROCEDURE

3.1 Upon being advised of a fire, the Central Control Room Operator will sound the fire emergency alarm (two long blasts on the variable pitch "squawker") and announce the location of the fire over the Public Address System.

3.2 When the Fire Alarm has been announced over the Public Address System the General Watch Foreman and the members of the Fire Fighting Brigade will proceed to the location of the fire. The General Watch Foreman will take charge and will direct all fire fighting activities.

3.3 When a fire is reported in the Controlled Area, the Health Physics Technician will proceed to the fire with the Fire Brigade. On the way, he will pick up film badges and dosimeters in the Security Room and such protective clothing and equipment from the Emergency Locker as required. The fire will be fought under the supervision of the General Watch Foreman assisted by the Health Physics Technician, who will control and monitor the radiation and contamination exposure of all personnel.

4.0 OUTSIDE ASSISTANCE

Although the need for outside assistance is considered extremely unlikely, the Verplanck Fire Department (Phone No. 737-1643) shall be called by the Central Control Room Operator if in the opinion of the General Watch Foreman their services are needed.

All possible steps will be taken to avoid the use of outside fire fighting assistance in the Controlled Areas. If such becomes absolutely necessary and the Verplanck Fire Department is called, special precautions relating to Health Physics aspects and accessibility will be taken before admitting them to the Controlled Areas. All fire fighting activities in the Controlled Areas will be under the direct control and supervision of the General Watch Foreman.

The Station Guards are to be notified by the Central Control Room Operator when the Verplanck Fire Department is called to direct them to the fire location.

5.0 DRILLS

Adequate familiarization with fire fighting procedures by all members of the operating staff is demonstrated periodically through the use of fire drills. To provide a true measure of the degree of readiness, these drills will generally be initiated without prior announcement.

6.0 EQUIPMENT

A detailed description of the Fire Protection System for the Indian Point Facility is provided in subsection 9.6.2 of the Unit No. 2 FFD and SAR.

TORNADO CONTINGENCY PLAN

Supplement 5
1/70

TABLE OF CONTENTS

1.0 INTRODUCTION

The Tornado Contingency Plan describes the procedures which will be followed in the event of a tornado watch(1) or tornado warning(2) at the Indian Point Site.

Meteorological conditions that could result in a tornado would be determined by the Weather Bureau. Their services are available to the Company's System Operator at the Company's Energy Control Center located at 128 West End Avenue in New York City.

2.0 RESPONSIVE ACTIONS

The immediate protective actions taken by the key Company personnel are as follows:

<u>System Operator</u>

1. Notify the Central Control Room Operator that either a tornado watch or a tornado warning exists for the vicinity of the Indian Point Facility.

2. Maintain contact with the plant until the tornado condition has abated.

<u>Central Control Room Operator</u>

1. Notify the General Watch Foreman (or the Operations Superintendent if the tornado situation occurs during the normal working hours when he is at the station).

---

(1) Tornado watch - means meteorological conditions are favorable for the formation of a tornado.

(2) Tornado warning - means that a tornado has been sighted in the area of the plant.

2. If a tornado warning has been received, start the gas turbine generator and arrange with System Operation to line up the underwater tie with Orange and Rockland.

General Watch Foreman (Operations Superintendent)

1. Assign station personnel to look and listen for a tornado.

2. If a tornado warning has been received, order all fuel handling operations in the fuel handling building halted. If a fuel handling cask is suspended from the crane at this time, order that it be set down. Order all other non-essential plant operations halted.

3. Notify station management.

Operating Personnel

1. Maintain a watch to listen for and look for a tornado.

2. If a tornado is sighted, notify Central Control Room Operator immediately.

3.0 SUBSEQUENT ACTIONS

If a tornado is sighted in the area of the plant, the General Watch Foreman (Operations Superintendent) shall take whatever additional action he deems necessary.

RADIATION CONTINGENCY PLAN

PREFACE

The Indian Point Nuclear Generating Facility has been designed and cons ructed to exacting standards and is operated by a highly qualified and competent staff. Prior to issuance of a Construction Permit, the characteristics of the site, the facility design, and numerous analyses to demonstrate the effectiveness of safeguards incorporated in the facility are evaluated by the Atomic Energy Commission (AEC), the Advisory Committee on Reactor Safeguards (ACRS) and by the public at a public hearing presided over by the Atomic Safety and Licensing Board. The AEC and ACRS conduct an additional review prior to issuance of an Operating License. A Construction Permit and Operating License are not granted until it is demonstrated that the facility can be operated safely.

The design of the facility is such that a radiation incident will be confined to the site. However, it is recognized that some highly improbable chain of events could result in an incident that ould result in abnormal releases of radioactive material beyond the site boundary. The following Radiation Contingency Plan has been developed to insure that under any situation the f- 'lity's staff will be prepared to respond quickly and efficiently to te_minate the incident and mitigate its consequences.

Implementation of the General Contingency Plan involving off-site considerations is not anticipated. If implementation were required, it is anticipated that the resulting exposures would be within the guidelines set fort in Title 10, Part 100 of the Code of Federal Regulations.

RADIATION CONTINGENCY PLAN

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

FIGURES

1.0 INTRODUCTION

The Radiation Contingency Plan for the Indian Point Station prescribes immediate action to be taken by plant personnel to minimize exposure to radiation resulting from an abnormal operating situation. This plan prescribes the action which is to be taken in order of priority, the responsibilities of personnel for taking such action and summarizes personnel and material resources available for assistance in minimizing exposure and in restraining and mitigating the consequences of the abnormal situation.

There are three phases of responsive action. The first phase would include initial actions directed toward the protection of personnel and the elimination of the potential for further exposure. The second phase would include immediate and planned action directed toward termination of the incident, containmnet of the effluent, establishment of incident boundaries, establishment of control and channeling of information and protection of the facility and equipment. The third phase would be to restore the facility to its normal operating condition.

The first two phases will be conducted by a Contingency Team with or without the assistance of other on-site personnel. The Congingency Team will be composed of shift personnel to insure that the team will be trained and available at all times. A Command Group will provide technical information to the Contingency Team, and assure a smooth transition to the final phase which it will direct.

2.0 ORGANIZATION 15

2.1 NORMAL ORGANIZATION

The Indian Point Facility normal operating organization for an overall facility and shift basis is shown on Figures 1 and 2, respectively.

2.1.1 Responsibilities and Duties

The responsibilities and duties of all personnel assigned to the normal operating organization from the foreman level up are described in subsection 12.1 of the FFD and SAR.

2.2 CONTINGENCY ORGANIZATION

The Indian Point Facility Contingency Organization is shown on Figure 3. The nucleus of the organization will consist of those personnel assigned to the regular operating shift with the General Watch Foreman in charge. The General Watch Foreman will act as the initial Contingency Coordinator for all activities. Depending upon the status of the incident, the General Watch Foreman may be relieved of his duties as Contingency Coordinator by the Operations Superintendent or his designated alternate, thus freeing the General Watch Foreman for a more direct involvement in the contingency.

During the normal work day, when a full complement of plant personnel are present, the organization will be expanded as necessary to include certain key staff personnel such as the Health Physicist, the General Chemist, the Assistant Maintenance Superintendent, the on-site First Aid Attendent and other personnel as indicated on Figure 1.

On each of the operating shifts, certain personnel are assigned to the Contingency Team. Members of the Contingency Team will be indoctrinated with their individual duties so that the required procedures can be carried out in the most expeditious manner.

2.2.1 Responsibilities and Duties

The responsibilities and duties of plant personnel assigned to the emergency organization are as follows:

Contingency Coordinator

a. Makes determination whether or not a contingency exists.

b. If conditions warrant, declares a contingency to exist.

c. Obtains necessary information to properly evaluate the situation.

d. Coordinates all on-site activities during the contingency.

e. Notifies the designated plant management of the situation as per Paragraph 5.2.

Contingency Team

a. Upon notification, the Contingency Team reports to assigned stations unless otherwise directed by the Contingency Coordinator.

b. Carries out assigned duties to protect personnel, gain control of the incident and limit its consequences.

General Superintendent

a. Notifies and directs the Command Group.

b. Coordinates all Con Edison activities during the contingency.

c. Makes required notification of AEC, N.Y. State Department of Health, and other agencies.

d. Makes request for outside assistance.

e. Coordinates release of information to the general public through the Con Edison Public Information Department.

Command Group

a. Provides overall direction for all on-site activities during the contingency.

b. Furnishes technical advice to outside assistance groups.

Operations Superintendent

a. Assumes duties of Contingency Coordinator upon arrival at the facility.

b. Serves as a member of the Command Group.

Assistant Maintenance Superintendent

a. Serves as a member of the Command Group.

b. Responsible for all required maintenance related activities including the procurement of special materials and equipment.

Assistant Operations Superintendent

Serves as a member of the Command Group.

Reactor Engineer

a. Serves as a member of the Command Group.

b. Furnishes advice and makes recommendations on all aspects of post-incident reactor operations.

Performance Superintendent

a. Serves as a member of the Command Group.

Health Physicist

a. Serves as a member of the Command Group.

b. Directs activities of the Health Physics Monitoring Team.

Health Physics Monitoring Team

a. Conducts on-site and off-site radiation surveys as directed.

b. Establishes exclusion areas.

c. Makes recommendations to the Command Group.

d. Supervises the decontamination of personnel and equipment.

e. Determines personnel exposures and maintains exposure records.

f. Specifies and issues protective equipment and personnel gear.

g. Provides monitoring services for all post-incident re-entry activities.

h. Furnishes advice to medical personnel relative to injuries involving radiation or contamination considerations.

i. Cooperates with A.E.C. or N.Y. State Radiological Assistance Teams in the performance of off-site radiation surveys.

Production Engineers

a. Serve as members of the Command Group.

b. Provide laison services with non Con Edison support groups.

3.0 SCOPE OF PLAN

The Radiation Contingency Plan for abnormal operating conditions prescribes procedures to cover the range of anticipated and hypothetical equipment and plant troubles. The variety of credible radiation incidents, including the hypothetical major rupture of the primary coolant system, should not require the utilization of this plan in its entirety. The numerous possible modes of radiation release, meteorological conditions and desired actions is such that the procedures presented herein must, of necessity, be flexible. However, these procedures will be such that the responsive action can change smoothly and logically as the status of the incident changes.

The Contingency Coordinator will determine the degree of severity of all incidents which might occur at the facility. The Radiation Contingency Plan is divided into three separate categories:

3.1 LOCAL CONTINGENCY

Involves only areas within the plant. May involve evacuation of areas within a building, or may involve evacuation of the building itself.

3.2 SITE CONTINGENCY

Involves the total plant site, and may require evacuation of all non-essenti operating personnel from the plant site. It does not, however, involve areas external to the station boundary; nor does it require the assistance of off-site support groups.

3.3 GENERAL CONTINGENCY

Involves areas beyond the site boundary, and may require the assistance from one or more off-site support groups.

4.0 PLAN IMPLEMENTATION LEVELS

4.1 INTRODUCTION

Instrumentation is provided at selected points in and around the plant to detect and record the radiation levels. In the event these radiation levels rise above a specified value, an alarm is initiated in the Central Control Room. The automatic Radiation Monitoring System operates in conjunction with routing and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff.

4.2 IMPLEMENTATION LEVELS

For the three different categories considered in Paragraph 3.0, the conditions at which the contingency plans will be implemented are as follows:

4.2.1 Local Contingency Plan

One area radiation monitor reaches its alarm set point.

OR

The plant vent radiogas monitor reaches its alarm set point.

OR

An unexpected increase in the level of radiation or airborne activity in a work area as indicated by temporary monitors.

4.2.2 Site Contingency Plan

Two area radiation monitors, each one in a separate building, indicate radiation levels in excess of their alarm set points, and the plant vent radiogas monitor reaches its alarm set point.

OR

A design basis accident is indicated

4.2.3 General Contingency Plan

A Site Contingency has been declared

AND

Off-site survey by the Health Physics Monitoring Team indicates radiation levels equal to or in excess of those specified in paragraph 7.3.2.

15

5.0 GENERAL INFORMATION

5.1 NOTIFICATION CHANNELS

Notification of a contingency will be accomplished by one or more of the following ways:

a. Personnel working in or around the facility.
b. Automatic alarm system (Radiation Monitoring System).
c. Plant Public Address System.

5.2 NOTIFICATION ROSTER

Names, addresses and telephone numbers of the individuals and organizations listed in this Paragraph are given in the attached Appendix "A". As changes occur, revised lists will be issued with an up-to-date list maintained in the Central Control Room at all times.

5.2.1 Con Edison Site Personnel

The Contingency Coordinator is responsible for notifying, in the order shown, the first of the following site management personnel who can be contacted. The individual first contacted shall in turn notify the others on the list, and shall make the necessary notification of Con Edison off-site mangement personnel.

a. General Superintendent
b. Operations Superintendent
c. Reactor Engineer
d. Performance Superintendent
e. Assistant Maintenance Superintendent
f. Assistant Operations Superintendents
g. Health Physicist
h. Production Engineers (assigned to Command Group)

5.2.2 Con Edison Management

Notification shall be made, in the order shown, of the first of the following individuals who can be contacted. The first person contacted shall in turn notify the others on the list.

a. Assistant Manager (Nuclear) - Production Department
b. Vice President - Power Supply
c. Manager - Production Department
d. Executive Medical Director
e. Radiation Safety Officer
f. Chief Mechanical Engineer
g. Vice President - General Counsel
h. System Duty Representative
i. Director of Public Information

5.2.3 Medical

Notification of one or more of the following individuals shall be made by the Contingency Coordinator or General Superintendent as deemed necessary.

a. Con Edison Assistant Medical Director

b. District Doctors

c. Station First Aid Attendant

d. Medical Consultant (Radiation)

e. N. Y. University Hospital

f. Peekskill Community Volunteer Ambulance Corp.

g. Verplanck Fire Department (Ambulance)

5.2.4 Atomic Energy Commission

a. N.Y. Operations Office - Division of Compliance

b. N.Y. Operations Office - Radiation Duty Officer

5.2.5 State of New York

a. Department of Health, Bureau of Radiological Health Services

b. State Police

5.2.6 County of Westchester

a. Department of Health

5.2.7 U. S. Coast Guard

a. Captain of the Port of New York

5.2.8 Local Authorities

a. Peekskill Police Department

b. Buchanan Police Department

c. Peekskill Fire Department

d. Buchanan Fire Department

5.3 AUTHORIZATION

The following individuals are authorized to declare a contingency.

a. General Superintendent or designated representative

b. Operations Superintnedent

c. Health Physicist

d. General Watch Foreman

e. Central Control Room Operator

5.4 CONTROL CENTER

During a Local or Site Contingency, the center for coordinating all activities will be the Central Control Room (or any other area as may be designated by the General Superintendent). During a General Contingency, the Control Center will be located in the Central Control Room, in the Production Department Office, or in the mobile headquarters of the New York State Radiological Assistance Team, as determined by the General Superintendent.

5.5 EXPOSURES

Although highly unlikely it is possible that sitautions may arise which transcend the normal requirements for limiting exposure. In such cases, exposures above those set forth in 10 CFR 20 may be accepted if the net effect of the exposure will be to limit the total exposure
15 burdens associated with the incident. Once in a lifetime, accidental or emergency exposure to penetrating radiation may be accepted up to 25 rem to the whole body. In all situations, though, every reasonable effort will be made to minimize the exposure. Rescue personnel may receive up to 100 rem of penetrating radiation to the whole body if the saving of human life will result. As long as this

limit is not exceeded, no permanent disability will result. However, under no circumstances will this type of exposure be permitted unless rescue personnel are wearing monitoring devices capable of measuring these exposures.

5.6 TRAINING DRILLS

Internal practice exercises to demonstrate adequate familiarization with the procedures by members of the operating stall will be performed on a semi-annual basis or more frequently should the results of a particular exercise indicate the need. In order to provide a true measure of the degree of station personnel readiness, practice drills will generally be initiated without prior announcement.

In addition to the above mentioned drills, communication by radio contact, such as might be required in certain situations, will be made on a weekly basis between the licensed reactor operator in the Central Control Room and the System Operator at the Company's Energy Control Center.

5.7 REVIEW AND UPDATING

A continuing review of operations will be performed by the Station operating staff, the Production Department administrative staff, and the executive level for those departments with operating, design, and safety responsibility for the facility. There will be periodic reviews by the Nuclear Facility Safety Committee of operations, normal and abnormal procedures, and changes to the facility or Technical Specifications. The contingency plans and procedures will be reviewed and updated as required.

5.8 NOTIFICATION AND REPORTS

5.8.1 Notification

5.8.1.1 Immediate Notification

Section 20.403 of 10 CFR 20 requires immediate notification of the AEC Region I Compliance Office by telephone and telegraph of any incident involving by-product, source or special nuclear material possessed by Con Edison that may have caused or threatens to cause:

a. Exposure of the whole body of any individual to 25 rems or more of radiation; exposure of the skin of the whole body of any individual of 150 rems or more of radiation; or exposure of the feet, ankles, hands or forearms of any individual to 373 rems or more of radiation; or

b. The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 5000 times the limits specified for such materials in Appendix B, Table II of 10 CFR 20; or

c. Damage to property in excess of $100,000.

5.8.1.2 Twenty-four Hour Notification

Section 20.403 of 10 CFR 20 requires twenty - four hour notification of the AEC Region I Compliance Office by telephone and telegraph of a.y incider involving licensed materia' possessed by 'm that may have caused or threaten to cause:

a. Exposure of the whole body of any individual to 5 rems or more of radiation; exposure of the skin of the whole body of any individual to 30 rems or more of radiation; or exposure of the feet, ankles, hands or forearms to 75 rems or more of radiation; or

b. The release of radioactive material in concentrations which, if averaged over a period of 24 hours, would exceed 500 times the limits specified for such materials in Appendix B, Table II of CFR 20: or

c. A loss of one day or more of the operation of any facilities affected; or

d. Damage to property in excess of $1000.

5.8.1.3 Thirty Day Notification

Section 20.405 of 10 CFR 20 requires thirty - day notification, via a written report, of the Director, Division of Compliance, U. S. Atomic Energy Commission, Washington 25, D.C. (with a copy to the Director of Division of Compliance, Region I) of:

a. Each exposure of an individual to radiation or concentrations of radioactive material in excess of any applicable limit in 10 CFR 20 or as set forth in the Facility Operating License.

b. Any incident for which notification is required by Paragraph 5.8.1.1 or 5.8.1.2.

c. Levels of radiation or concentrations of radioactive material (not involving excessive exposure of any individual) in an unrestricted area in excess of ten times any appl[illegible]able limit set [illegible]rth in 10 CFR 20 or [illegible]e Facility [illegible]perating License.

5.8.1.4 New York State Notification

The New York State Department of Health shall be promptly informed of any incident which requires notification of the Atomic Energy Commission under Paragraph 5.8.1.1, 5.8.1.2 and 5.8.1.3.

5.8.2 Reports

Each written report required under Paragraph 5.8.1 shall describe the extent of exposure of persons to radiation or to radioactive material; levels of radiation and concentrations of radioactive material involved; the cause of the exposure, levels or concentrations; and corrective steps taken or planned to assure against a recurrence.

5.9 PUBLIC NOTIFICATION

In the highly unlikely event that the conditions for a General

Contingency exists, and depending upon the severity of the situation as judged by the coordinator, it may be deemed necessary to notify the general public that an abnormal operating condition exists at the Indian Point Station. The Contingency Coordinator will inform the Con Edison Public Information Department by contacting one of the following persons:

Name	Work Phone	Home Phone
W. O. Farley	460-4111	AC 201-797-2746
S. P. Stengren	460-4113	AC 516-931-2490
C. E. Lohrfink	592-9010	AC 914-963-3554

All information given to the press or radio stations, regardless of what category of contingency exists, shall be issued through the Public Information Department.

5.10 OFF-SITE SUPPORT GROUPS

As described in Paragraph 3.3, a General Contingency would be declared for any incident that may involve off-site consideration. Incidents of this severity would require close cooperation and possible assistance of a number of off-site agencies and authorities - in particular, the AEC New York Operations Office and the New York State Department of Health, Bureau of Radiological Health Services.

The Atomic Energy Commission's radiological emergency assistance team is available 24 hours a day through the Region I Office of the Commission in lower Manhattan. Direct discussions between Consolidated Edison and Region I personnel to insure the ready availability of an effective support organization have been held. In the unlikely event of need for such support, Atomic Energy Commission personnel would be dispatched any time of the day or night from home or office locations in the New York City area, or with appropriate survey instrumentation and under the direction of an Atomic Energy Commission group leader. Such support would be provided by Consolidated Edison and the New York State Department of Health, Bureau of Radiological Health Services.

Similarly, New York State and Westchester County Departments of Health would provide radiological survey supporting effort at the request of Consolidated Edison. If the findings yielded by off-site surveys of these agencies indicate the need for additional action affecting the public, that determination would be made by the State Department of Health which would in turn notify the Governor's Office that an emergency exists. Through the Governor's Office, any additional state organizations as might be required such as Civil Defense or the New York State Police Department would be sought and action implemented under the direction of the State Health Department.

In addition to the above mentioned agencies and authorities, the Company enjoys the services of Dr. Roy E. Albert, a recognized expert in radiation medicine, and Dr. McDonald E. Wrenn, a certified health physicist and environmental health consultant, both of whose services would be solicited as consultants during any contingency.

6.0 FACILITIES AND SERVICES

6.1 SPECIAL EQUIPMENT

Special Equipment is located at strategic areas in the plant and on the plant property. This equipment is located in the Health Physics Office and the Indian Point Service Building. A list of this equipment is given in Appendix "B".

6.2 MEDICAL

A First Aid Room is located off the lobby on El. 15 of the

Conventional Service Building. This room can be used for treatment of ambulatory, non-contaminated injured personnel, and visitors to the plant.

A complete medical facility, consisting of a decontamination room and an examination room is located on El. 72 of the Nuclear Service Building for Unit No. 1. This centralize facility will also serve all additional nuclear units at or near the Indian Point Site.

The decontamination room is specially designed for the care of personnel contaminated by radioactivity. It is designed to allow the admission of several mixed ambulatory and non-ambulatory patients. A conventional examining room is located in the dispensary area of this medical facility and can be used for first aid after decontamination of ambulatory and litter cases. It can also be used for general clinical service. Medical and first aid supplies are kept in the dispensary and are passed through as they are needed into the decontamination room.

An alternate site for medical supplies and an alternate treatment facility for possible radiation casualties is located in the regular Medical Bureau in the Indian Point Service Building.

A portable ra oisotope decontamination kit is provided in the Indian Point Service Building for personnel decontamination. A portable kit, similar to the radioisotope decontamination kit, is available for transfer of an injured radiation casualty to an outside medical facility.

Arrangements have been made with University Hospital, New York University Medical Center in New York City for handling contaminated injured personnel.

A detailed description of the medical facilities is presented in Subsection 11.2 of the FFD and SAR.

6.3 COMMUNICATIONS

The following methods for normal or emergency communications are available at the Indian Point Facility:

1. Variable Pitch "Squawker"
2. Public Address and Party Line System
3. Dial Telephone (Con Edison System)
4. N.Y. Telephone Company (Eight Outside Lines)
5. Short-Wave Radio
6. FM Radio "Handie-Talkie" (Three Units)
7. Direct Telephone Line to the System Operator at the Company's Energy Control Center.

Details relative to the above communication methods are provided below:

1. The variable pitch "squawker" will be used to alert station personnel of any contingency at the facility. The "squawker" can be connected to Unit No. 1 and/or Unit No. 2 public address systems by means of appropriate switches on the respective operators control desk in the Central Control Rooms. The following schedule has been established for sounding the contingency alarm.

    Station Contingency - One long blast
    Fire Contingency - Two long blasts
    Air Raid Alert - One long blast and one short blast
        Note: Long blast - 10 seconds or longer
              Short blast - 5 seconds or less

2. The public address and party line 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 independently 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 operator in the Central Control Room. "Party No. 4" line provides an additional channel in the primary portion of the No. 2 plant only and does not have any merging facilities. Speakers for monitoring each of these lines are located in the central Control Room.

3. Standard dial telephones incorporated into the Consolidated Edison system are located at strategic locations throughout the facilities controlled and conventional areas.

4. Eight outside lines (N.Y. Telephone Co.) are available. These phones are located as follows:

Centrao Control Room	(2)
Health Physics Office	(1)
Maintenance Office	(1)
General Watch Foreman's Office	(1)
General Superintendent's Office	(1)
Unit No. 1 Containment	(1) jack only
Unit No. 2 Containment	(1) jack only

5. Facilities are provided in the Central Control Room so that the operator can talk into and listen to the regular Company shortwave radio system through the Indian Point receiver - transmitter equipment, as required.

In the event of loss of all normal a-c power, provision has been made to connect the station radio transmitter to one of the two Unit No. 1 battery backed special a-c load boards.

6. Three Motorola HT Series "Handie-Talkie" FM radio units and a multiple-unit charger are maintained in the Production Department Office located in the Indian Point Facility Conventional Service Building. These units have a range of approximately one mile and are available at all times.

7. A direct-wire telephone connection (N.Y. Telephone Co.) to the System Operator at the Company's Energy Control Center is located on the Unit No. 1 operators desk in the Central Control Room.

If in the course of plant operations a contingency develops, it will be announced by one long blast (>10 seconds) on the variable pitch "squawker". This will be followed, if it is deemed necessary, by an announcement over the public address system that all communications in progress, other than those directly involved in support of the contingency will cease immediately and that all outside telephones are to be used only for the purpose of notifying necessary personnel and authorities of the existing situation.

7.0 PROTECTIVE MEASURES

7.1 LOCAL CONTINGENCY PLAN

7.1.1 Responsive Action

Central Control Room Operator

1. Announce over the Public Address System that a high radiation alarm has been received. Give the location of the alarm and advise all personnel to leave the affected area and assemble in the Nuclear Service Building laundry foyer for monitoring. Make the announcement at least two times.

2. Notify the General Watch Foreman of the alarm.

3. Take appropriate operating action to limit, contain, or correct the condition.

Personnel in The Area

1. Leave the area indicated by the Public Address System announcement, warning any other personnel in the area to leave.

2. If alarms are received from temporary monitors, leave the area and advise the Central Control Room operator of the condition.

3. Do not enter any area where a local alarm is sounding unless such entry is approved by the General Watch Foreman.

4. Report to the Nuclear Service Building laundry foyer for monitoring.

General Watch Foreman

1. Report to the Central Control Room.

2. Evaluate the alarm received with respect to other radiation monitors and process instrumentation readouts in the Central Control Room.

3. Send the shift Health Physics technician into the affected area to make observations and evaluate radiation levels.

4. Evaluate the results of the Health Physics survey.
   A. If the results of the survey show radiation levels and conditions to be normal, permit routine entry to the area when the cause of the alarm has been determined. Have the Central Control Room operator announce over the Public Address System that the high radiation level no longer exists and that the area is cleared for routine entry.

   B. If the results of the survey shows radiation levels above normal in a local area, the affected area is to be roped off and appropriately marked as to specific conditions.

5. Determine the cause of the high radiation levels and take appropriate steps to return conditions to normal.

6. Notify plant General Superintendent of the situation.

7. Have Health Physics technician monitor all personnel who have assembled in the Nuclear Service Building laundry foyer.

Health Physics Technician

1. Perform radiation surveys and personnel monitoring as requested by General Watch Foreman.

2. Supervise personnel decontamination.

7.2 SITE CONTINGENCY PLAN

7.2.1 Responsive Action

Central Control Room Operator

1. Sound the alert and announce over the Public Address System that high radiation and radiogas levels have been detected. Give the location of the alarms and advise all non-essential operating personnel to report to the Conference Room in the Conventional Service Building. Repeat the alarm and the announcement two times.

2. Notify the General Watch Foreman.

3. Take appropriate operating action in accordance with plant emergency operating procedures and facility Technical Specifications.

4. Establish operating mode "d" for the Central Control Room ventilation system.

5. Turn on the radio transmitter and establish contact with the System Operator at the Company's Energy Control Center.

Personnel in the Area

1. All personnel (including visitors) in the plant, except those assigned specific responsibilities or those with fixed post duties, report to the Conference Room in the Conventional Service Building for head count and further instructions.

2. Personnel reporting to the Conference Room will take the most direct route unless such route will take them through the announced high radiation or high radiogas areas. In such an event, alternate routes should be taken to avoid unnecessary exposures.

3. All personnel in the Controlled Area will first proceed to the Nuclear Service Building laundry foyer for monitoring before reporting to the Conference Room. Those personnel wearing protective clothing shall remove same and change into clean clothing before reporting to the Conference Room.

4. For those personnel leaving the Controlled Area who are found to be contaminated, appropriate precautions are to be taken to prevent the spread of contamination.

General Watch Foreman

1. Report immediately to the Central Control Room.

2. Assume duties of Contingency Coordinator until properly relieved by the Operations Superintendent or a member of the operating staff.

3. Evaluate plant conditions by observing readings on the Area and Process Radiation Monitors and pertinent Central Control Room instrumentation.

4. Open emergency lockers and storage rooms and make available any special equipment required, if necessary.

5. Have shift Health Physics technician conduct an area and plant radiation survey.

6. Evaluate results of radiation surveys.

Contingency Coordinator

The position of the Contingency Coordinator will be assumed by the General Watch Foreman until such times as he can be relieved by the Operations Superintendent when the latter is on duty.

1. During normal work hours, notify the General Superintendent of his designated alternate. Have the Central Control Room operator announce over the Public Address System that all members of the Command Group are to report immediately to the Central Control Room.

2. During off-hours, follow the plan outlined in Paragraph 5.2 for personnel notifiication.

3. Contact the WEDCO management staff and alert them that a Site Contingency has been declared and that evacuation of all construction personnel may be required.

4. Notify the Station Security Forces and request that road blocks be established at the plant entrances to prevent unauthorized entry to or egress from the plant.

5. Obtain meteorological data from Central Control Room weather instrumentation and determine evacuation route to be taken if needed.

6. Evaluate monitoring data from Health Physics Monitoring Teams and Area and Process Radiation Monitors.

7. Direct the activities of the Contingency Team.

8. Determine that all personnel in the plant, except those directly involved with the contingency or plant operations, have reported to the Conference Room.

9. Verify that all personnel on the operating shift have been accounted for.

10. Contact the Visitors Observation Building and request that all visitors standby for possible egress for the plant.

Contingency Team

1. Report to assigned stations unless otherwise directed by the Contingency Coordinator.

2. Perform assigned duties to protect personnel, gain control and limit the consequences of the incident.

3. Carry out special assignments as directed by the Contingency Coordinator.

Command Group

The Command Group shall consist of those personnel shown on Figure 3 with the General Superintendent or his designated alternate nominally in charge. The Command Group may be supplemented, if necessary, by other Con Edison management personnel as indicated on Figure 3.

1. Notify the AEC N.Y. Operations Office, the N.Y. State and Westchester County Departments of Health that a Site Contingency has been declared.

2. Call in additional plant personnel as deemed necessary.

3. Direct the activities of the Health Physics Monitoring Teams in obtaining sufficient information to determine the extent of the radiological implications. Order off-site radiation surveys if required.

4. Initiate site evacuation of non-essential personnel if the level of radiogas in the vent stack exceeds MPQ and two area radiation monitors, each in separate buildings, indicate radiation levels in excess of 1.0 R/Hr. Personnel to be evacuated are those assembled in the Conference Room, site construction forces and any visitors standing by in the Visitors Observation Building.

5. If required, send a Health Physics Monitoring Team to the site exit to monitor and survey personnel and cars.

6. Provide data and information to the Company's Public Information Department.

   Note: All information released to the public, regardless of what category of contingency exists, shall be issued only through the Company's Public Information Department.

7. Take required action to evaluate the contingency and to return the plant to normal conditions or mitigate the consequences of the incident.

8. Determine when the contingency plan is no longer required and what steps are to be taken to return the plant to its normal operating condition.

9. Furnish technical advice and information as necessary upon request from any outside agency such as the N.Y. State Department of Health, N.Y. State Police, etc.

10. Make any required notification as outlined in Paragraph 5.8.

Health Physics Monitoring Team

1. Make on-site and off-site radiological surveys as directed by the Command Group or Health Physicist.

2. Assist in personnel monitoring and decontamination as required.

3. Establish exclusion areas and determine necessary radiological controls.

4. Take necessary steps to limit spread of any contamination.

5. Determine exposure times and personnel exposures. Maintain exposure records.

6. Recommend and obtain necessary equipment and supplies to handle the situation in a safe and efficient manner.

7. Provide survey and monitoring services for personnel evacuated from the site.

8. Based on the results of on-site and off-site radiation surveys, provide estimates of the magnitude of the release to the Command Group.

9. Advise and assist personnell re-entering any high radiation and/or high airborne activity areas.

7.3 GENERAL CONTINGENCY PLAN

7.3.1 Responsive Actions

The actions to be taken by the Central Control Room Operator, the General Watch Foreman, the Contingency Coordinator, the Command Group, and the Health Physics Monitoring Team are as delineated for a Site Contingency. Additional actions are as follows.

Command Group

1. Send the Health Physics Monitoring Team downwind to the site boundary and beyond to perform a radiation survey.

2. Notify the following that a General Contingency has been declared giving the details of the situation.

   a. Con Edison Management
   b. State of New York
   c. Westchester County

d. Atomic Energy Commission
e. U. S. Coast Guard
f. Private Consultants

3. Determine the magnitude and extent of the incident by evaluating information from the Central Control Room and from the Health Physics Monitoring Team surveys. This information will consist of instrumentation readings and any survey results available, and will serve as a basis for requesting outside assistance.

4. If the results of the evaluation of the environmental conditions exceed the criteria set forth in Paragraph 7.3.2, notify the State of New York Dept. of Health, Bureau of Radiological Health Services, Westchester County Dept. of Health and the AEC that immediate assistance is requested.

5. Provide the State of New York and/or Westchester County with the following information:

   a. Radiological data
   b. Meteorological data
   c. Plant status
   d. Engineering assessment of operating conditions.

6. Give all possible assistance to the State of New York, and/or any other additional support groups as may be required by the State of New York.

Health Physics Monitoring Team

1. Perform radiation surveys downwind at the site boundary and beyond as requested by the Command Group.

2. Perform radiation survey for the State of New York and/or Westchester County if required.

3. Assist in personnel decontamination if required.

State of New York

1. Evaluate the data supplied by the Command Group and determine action required.

7.3.2 Bases for Requesting Off-Site Support Group Assistance

If either one of the two following conditions listed below is equaled or exceeded as determined by a Health Physics Monitoring Team survey, off-site assistance shall be requested.

1. The results of a survey by the Health Physics Monitoring Team downwind at the site boundary or beyond indicate dose rates that could result in a 2 hour exposure in excess of 250 mrem to the whole body.

2. The results of a survey by the Health Physics Monitoring Team downwind at the site boundary or beyond indicate radioactivity levels in the air equal to or greater than 8.1×10^{-7} μCi/cc based on I^{131}.

10 CFR 100 lists criteria for designating an Exclusion Area and a Low Population Zone based on exposure to personnel located at any point on these boundaries during a hypothetical accident. These exposure limits are given only for the purpose of evaluating the site and are not intended as emergency exposures during an accident. A value of 1/100 of 10CFR100 levels is used as the point at which off-site assistance shall be requested.

8.0 RETURN TO NORMAL OPERATIONS

In order for the recovery phase of the contingency to commence, the conditions which caused the incident to be declared (Paragraph 4.2) must no longer exist. It is the responsibility of the Command Group to determine that the Facility and/or the surroundings are safe, together with the mutual agreement of the AEC, the State of New York and Westchester County if these agencies assistance was required. The plant personnel will then return to the facility, and under the direction of the Command Group, will begin the task of making the necessary repairs and decontamination as required to bring the plant back into normal operation.

APPENDIX A

EMERGENCY NOTIFICATION ROSTER

A. **Con Edison Site Personnel**

	Title	Name	Address and Phone No.
	General Superintendent	D. J. McCormick	28 Besen Parkway Monsey, N.Y. - 10952 AC 914-EL-6-2118
	Assistant General Superintendent (Acting Operations Superintendent - Unit No. 2)	A. A. Karkosza	Montebel Garrison On Hudson, N.Y. 10524 AC 914-265-2552
	Operations Superintendent	T. M. Law	261 Cordial Road Yorktown Heights, N.Y. 10598 AC 914-245-7723
	Reactor Engineer	J. M. Makepeace	45 Tamarack Drive Peekskill, N. Y. AC 914-737-4170
15	Performance Superintendent	S. H. Cantone	John Alexander Drive RFD Peekskill, N.Y. 10566 (914) 739-4174
	Assistant Maintenance Superintendent	A. S. Darden	17 Franklin Avenue Croton On Hudson, N.Y. 10520 AC 914-CR-1-3574
	Assistant Operations Superintendents	R. L. Simms	15 Birch Parkway Sparta, N. J. 07871 AC 201-PA9-7884
		A. A. Nespoli	422 Elizabeth Road Yorktown Heights, N.Y. 12590 AC 914-YO2-4305
	Supervising Engineer (Health Physics)	G. H. Liebler	3 Ronsue Drive Wappingers Falls, N.Y. 12590 AC 914-896-8375
15	Assistant Superintendent	W. A. Monti	Bloomer Road RFD No. 2 Mahopac, N.Y. 10541 AC 914-628-6709
	Production Engineers	C. Limoges	Cortlandt Street Crugers, N.Y. 10521 AC 914-737-8908

	B. Moroney	1417 The Circle Peekskill, N.Y. 10566 AC 914-739-5152
	M. Shatkouski	6 Rolling Way Hillcrest Park East- Apt. 6F Peekskill, N.Y. 10566 AC 914-737-5818

A2. <u>Con Edison Management</u>

	Title	Name	Address and Phone No.
	Manager - Nuclear Power Generation Department	J. A. Prestele	42 Lynn Drive Englewood Cliffs, N.J. 07632 AC 201-LO-7-3843
	Vice President - Power Supply	T. A. Griffin, Jr.	21 Besen Parkway Monsey N.Y. 10952 AC 914-356-1819
	Executive Medical Director	S. C. Franco	8950 Colonial Road Brooklyn, N.Y. 12209 AC 212-TE-6-4153
	Radiation Safety Officer	W. F. Nelson	7 Toddville Lane Peekskill, N.Y. 10566 AC 914-737-3865
	Chief Mechanical Engineer	J. J. Grob, Jr.	19 Browning Drive Livingston, N.J. 07039 AC 201-WY-2-1501
15	Vice President & General Counsel	J. D. Block	920 Park Avenue New York, N.Y. 10028 AC-212-628-8335
	System Duty Representative	Variable	Posted Weekly
	Director of Public Information	W. O. Farley	17 Darrow Court Kedall Park, N.J. 08824 AC 201-297-2746
15	Assistant Vice President Power Generation Operation	R. H. Fregberg	40 Clark Street Pleasantville, N.Y. 10570 914-769-0084
	Health Physicist	R. J. Bozek	8 Richmond Place Peekskill, N.Y. 10566 914-739-4629

A3. Medical

Title	Name	Address and Phone No.
Assistant Medical Director	A. C. Hamilton	590 Flatbush Avenue Brooklyn, N.Y. 11225 AC 212-IN-9-9840
District Doctors	M. Menache	220 Tate Avenue Buchanan, N. Y. AC 914-739-1770
Station First Aid Attendant	G. Blizzard	313 Delancey Avenue Mamaroneck, N.Y. AC 914-OW-8-3666
Medical Consultant (Nuclear)	R. M. Albert, MD	20 Tintern Lane Scarsdale, N.Y. AC 914-GR2-0696
N.Y. University Medical Center		550 First Avenue New York, N.Y. AC 212-679-3200
Peekskill Volunteer Ambulance Corp.		1980 Crompound Road Peekskill, N.Y. AC 914-737-0044
Verplanck Fire Dept. (Ambulance)		8th Street Verplanck, N.Y. AC 914-737-9668

A4. Atomic Energy Commission

N.Y. Operations Office
376 Hudson St.
New York, N.Y.
AC 212-989-1000

Division of Compliance
Newark, N.J.
AC 201-645-3960

Supplement 13
8/70

A5. State of New York

Bureau of Radiological Health Services		845 Central Ave. Albany, N.Y. 12206 Hot Line AC 518-457-2055 Regular AC 518 457-2163
	T. J. Cashman	35 St. Stephens Lane Scotia, N.Y. AC 518-399-3121
	W. J. Kelleher	182 Roweland Ave. Delmar, N.Y. AC 518-439-6281
	H. Farbas	50 Rapple Drive Albany, N.Y. AC 518-859-0676
	K. Anderson	44 Scotch Pine Dr. Voorheesville, N.Y. AC 518-765-4077
State Police - Troop K		Hawthorne, N.Y. AC 914-769-2600

A6. County of Westchester

Department of Health		County Office Building White Plains, N.Y. AC 914-949-1300
	If no answer	AC 914-949-6430

A7. U. S. Coast Guard — Captain of the Port of New York, Governors Island, N.Y.

8:30 AM - 5:00 PM AC 212-264-8753
(Dangerous Cargo Officer)

5:00 PM - 8:30 AM AC 212-264-8770
(Operations Duty Officer)

A8. Local Authorities

Peekskill Police Dept.		926 Central Avenue Peekskill, N.Y. 10566 AC 914-737-0003
Buchanan Police Dept.		Westchester Avenue Buchanan, N.Y. 10511 AC 914-737-1033
	If no answer	AC 914-739-0802

Peekskill Fire Department		Peekskill, N.Y. 10566 AC 914-737-3323
Verplanck Fire Department		8th Street Verplanck, N.Y. AC 914-737-9668

A9. Consultants

Environmental Health	M. E. Wrenn		245 E. 72nd St. New York, N.Y. 10021 AC 212-628-8809
		Work:	Tuxedo, N.Y. AC 914-351-2368
Environmental Health	M. Eisenbud		Sterling Lake Tuxedo, N.Y. 10987 AC 914-351-2937

Superseded by amend.
#23

Peekskill Fire Department — kskill, N.Y. 10566
AC 914-737-3323

Verplanck Fire Department — 8th Street
Verplanck, N.Y.
AC 914-737-9668

A9. <u>Consultants</u>

Environmental Health — M. E. Wrenn — 245 E. 72nd St.
New York, N.Y. 10021
AC 212-628-8809

Work: Tuxedo, N.Y.
AC 914-351-2368

APPENDIX B

EMERGENCY EQUIPMENT

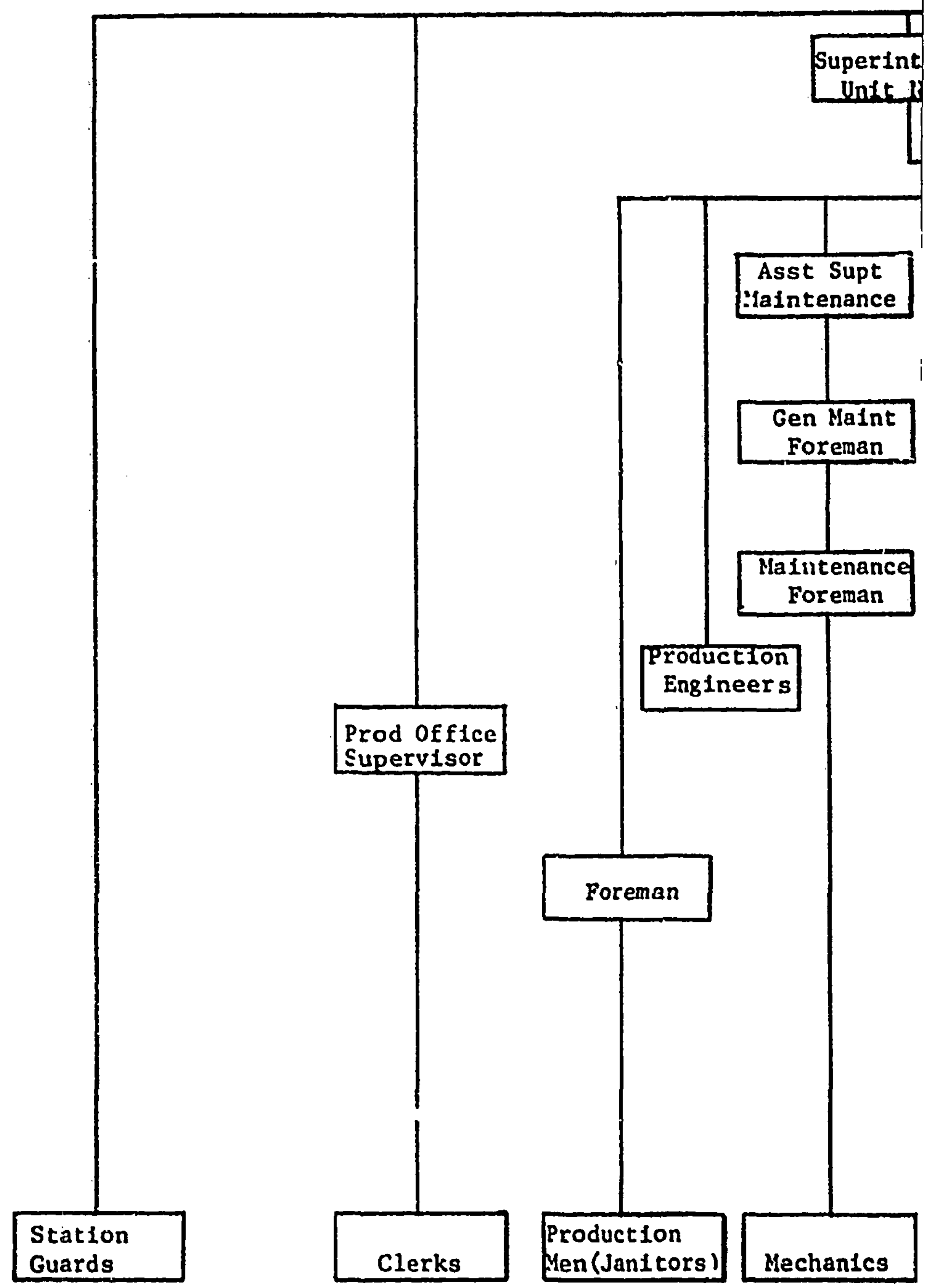
Health Physics Office

1. Emergency Procedures Book
2. Anti-Contamination Clothing
3. Respirators
4. Beta and Gamma survey instruments
5. High level pocket dosimeters
6. Dosimeter charger
7. Spare batteries
8. Film badges
9. High volume air sampler
10. Filter paper

Service Building

1. Emergency Procedures Book
2. Anti-Contamination Clothing
3. Respirators
4. Millipore pump and filter holder
5. Hurricane sampler and filter paper
6. Spare charcoal filters
7. High level pocket dosimeters
8. High level ratemeter

CONSOLIDATED EDISON CO OF N.Y. INC.



INDIAN POINT STATION
ORGANIZATION CHART

SUPPLEMENT 5
1/70

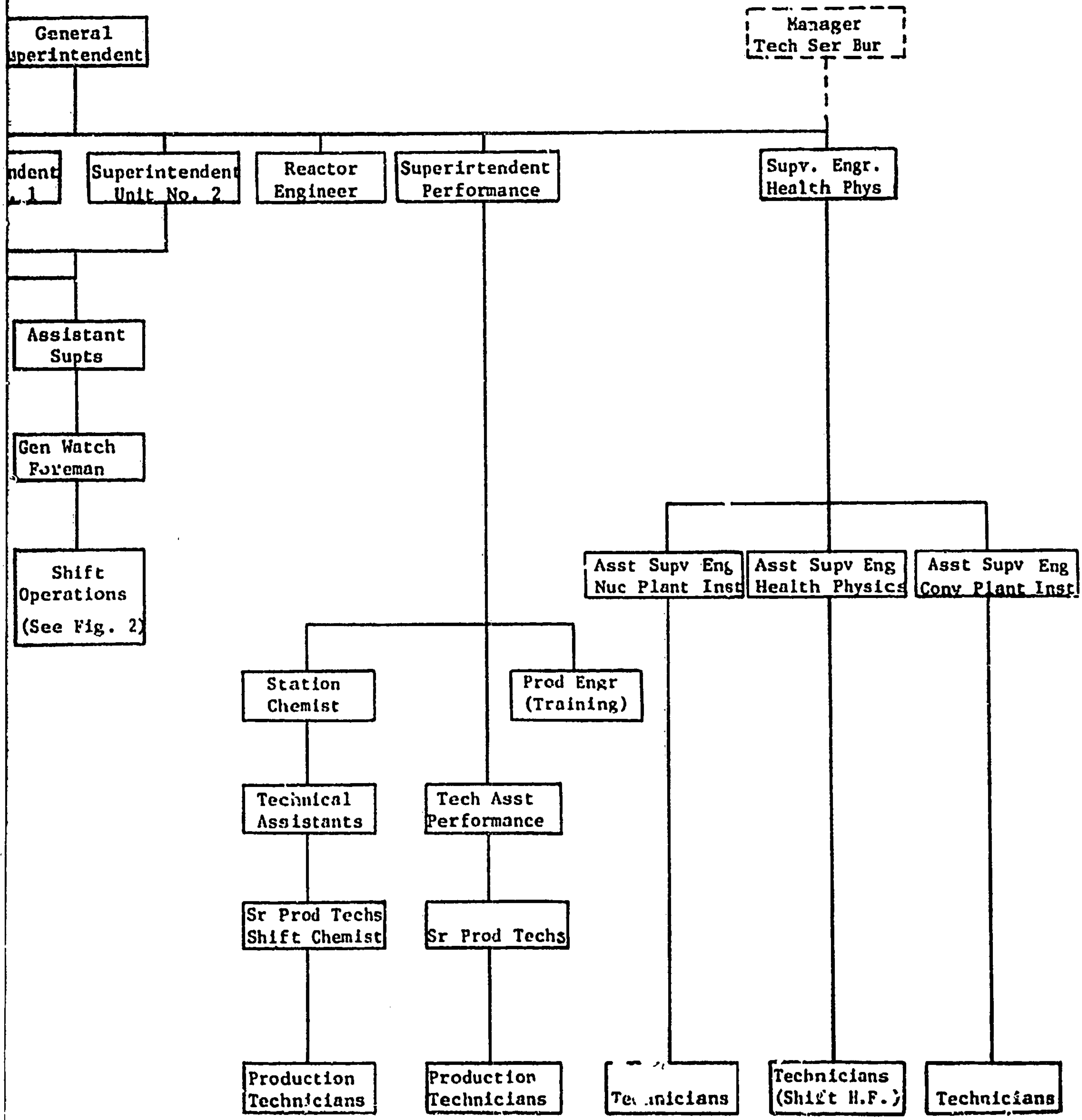
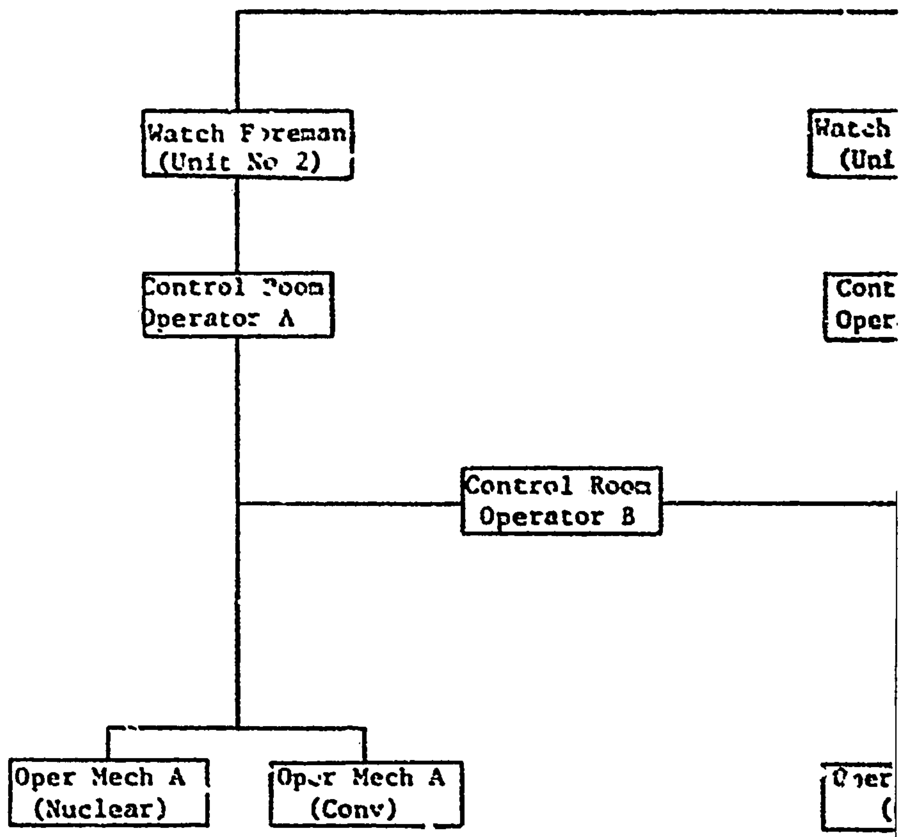


FIGURE 1

CONSOLIDATED EDISON CO OF N.Y. INC



INDIAN POINT STATION
SHIFT ORGANIZATION

Supplement 5
1/60

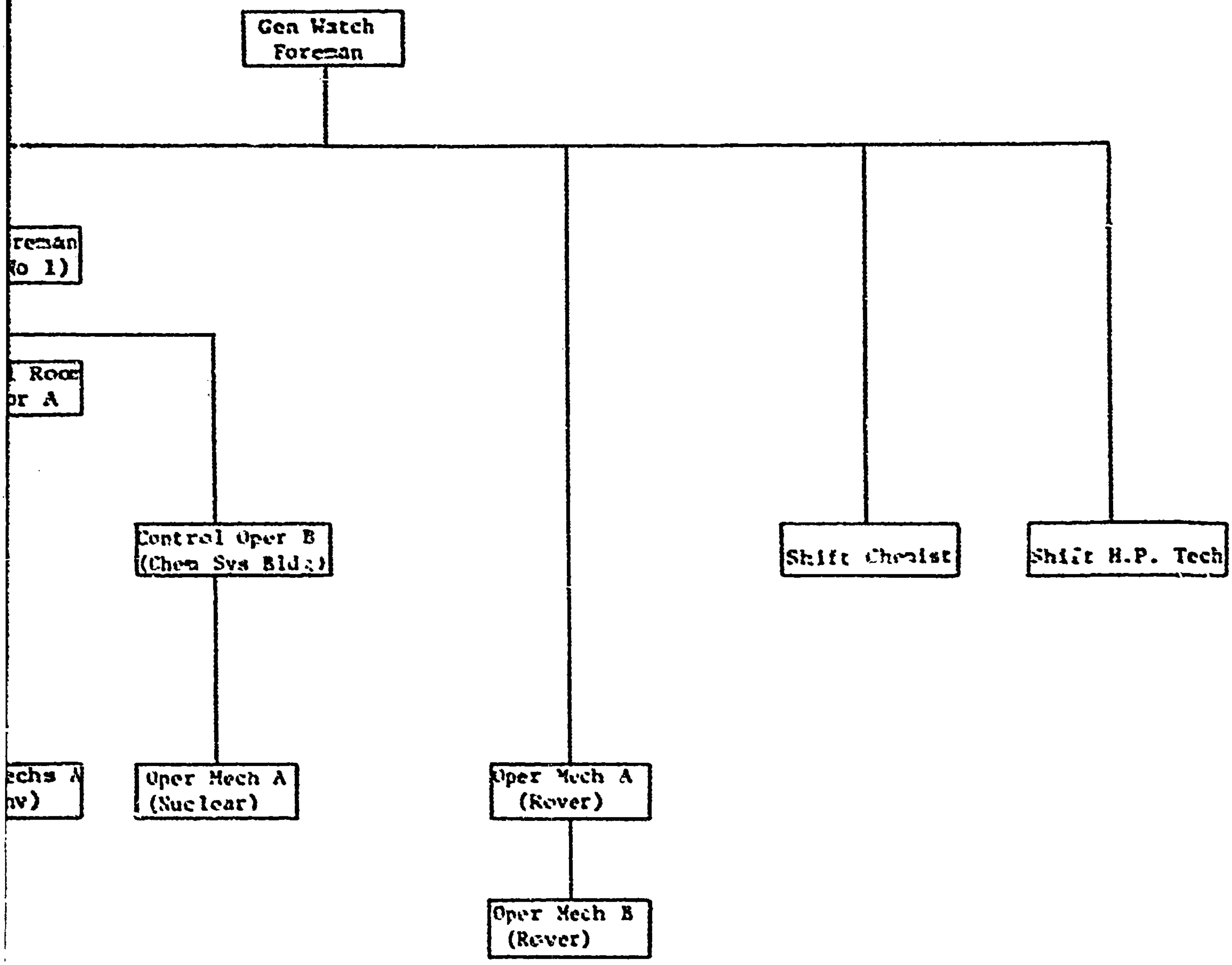
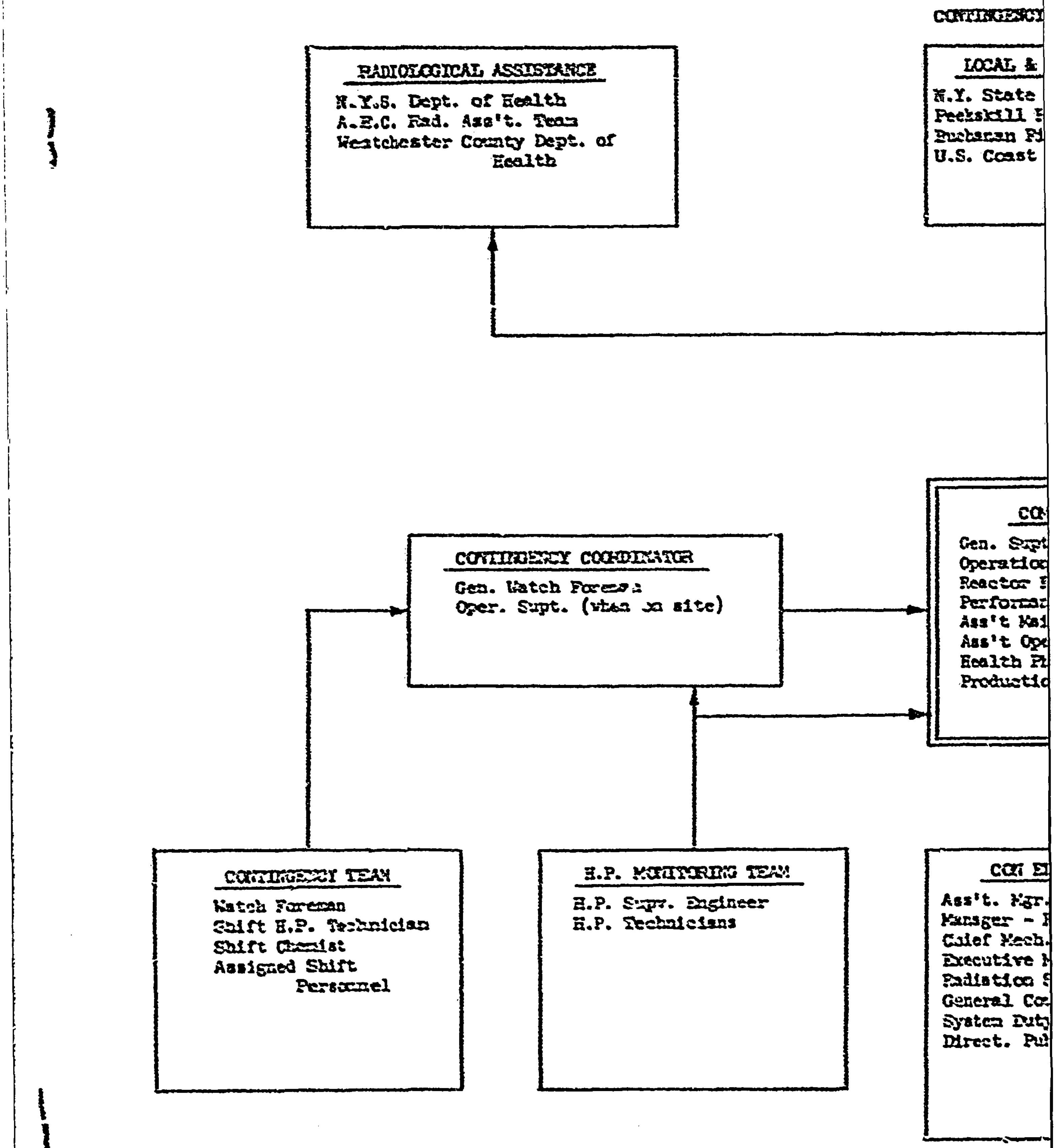


FIGURE 2


RADIOLOGICAL ASSISTANCE
N.Y.S. Dept. of Health
A.E.C. Rad. Ass't. Team
Westchester County Dept. of Health
LOCAL &
N.Y. State
Peekskill
Buchanan
U.S. Coast
CONTINGENCY COORDINATOR
Gen. Watch Foreman
Oper. Supt. (when on site)
Gen. Supt
Operatio
Reactor
Performa
Ass't Mai
Ass't Ope
Health Ph
Productio
CONTINGENCY TEAM
Watch Foreman
Shift H.P. Technician
Shift Chemist
Assigned Shift Personnel
H.P. MONITORING TEAM
H.P. Supv. Engineer
H.P. Technicians
Ass't. Mgr.
Manager -
Chief Mech.
Executive
Radiation
General Co
System Duty
Direct. Pu

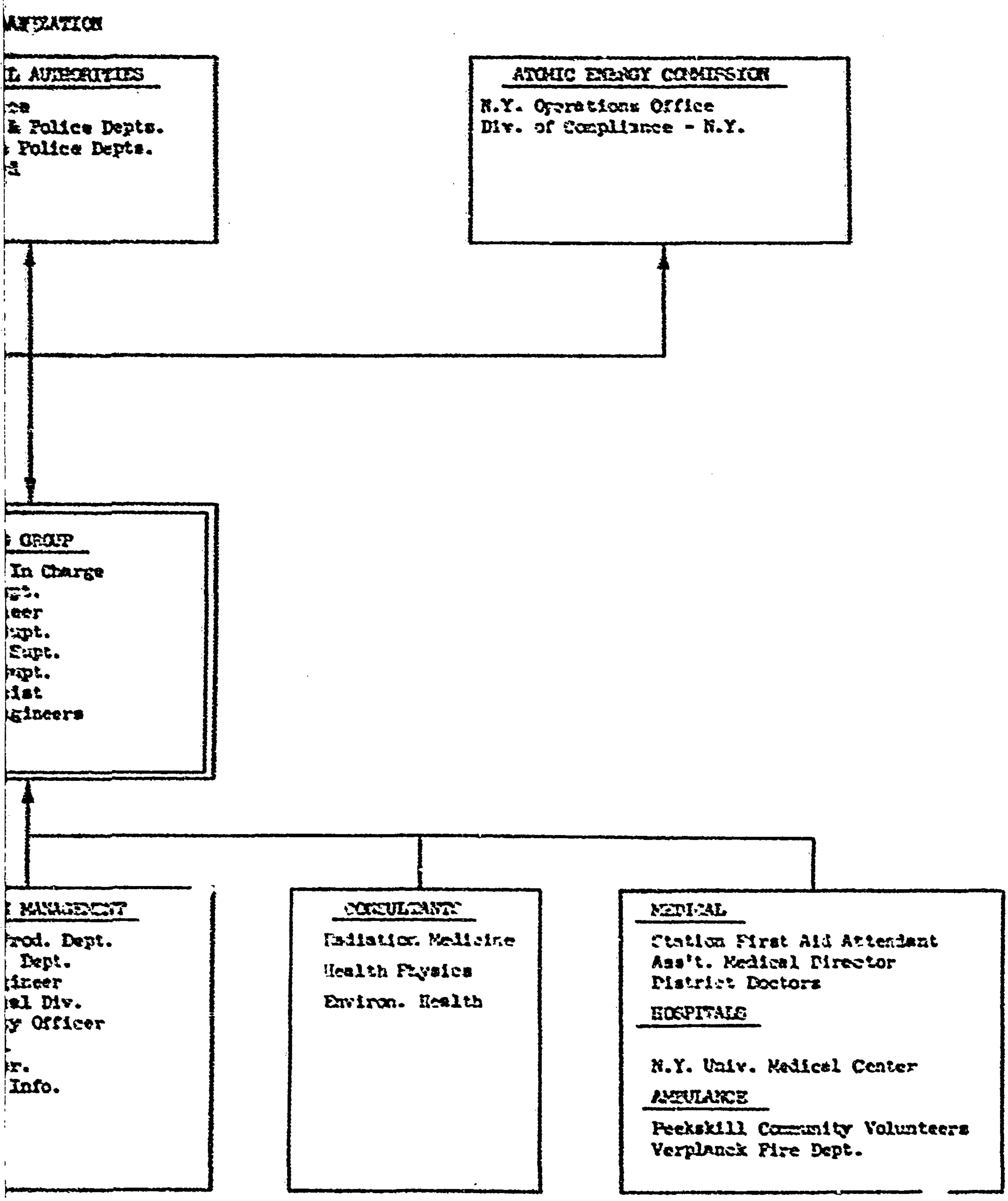

ANIZATION
L AUTHORITIES
& Police Depts.
Police Depts.
ATOMIC ENERGY COMMISSION
N.Y. Operations Office
Div. of Compliance - N.Y.
GROUP
In Charge
eer
Supt.
ist
gineers
MANAGEMENT
rod. Dept.
Dept.
ineer
al Div.
y Officer
r.
Info.
CONSULTANTS
Radiation Medicine
Health Physics
Environ. Health
MEDICAL
Station First Aid Attendant
Ass't. Medical Director
District Doctors
HOSPITALS
N.Y. Univ. Medical Center
AMBULANCE
Peekskill Community Volunteers
Verplanck Fire Dept.

FIGURE 3

QUESTION 12.6

Describe the security measures to guard against and to detect unauthorized access to:

- The reactor site.
- The control building and other principal buildings of the facility.
- The containment building.
- The fuel storage building.

ANSWER

The security measures employed to guard against and to detect unauthorized access to various portions of the facility will be a logical extension of those presently in effect for Unit No. 1 as described in Exhibit 0 to Docket 50-3. The fence enclosing the restricted area for Unit No. 1 will be extended to include the comparable area for Unit No. 2 and ultimately that for Unit No. 3. Access to the restricted area for all personnel will be through manned gate houses or locked gates which are under direct control of the Station security forces.

Station security forces are provided by the Burns International Detective Agency, Inc., and come under the direct supervision of the Production Department. In addition to controlling access to and from the restricted area, the Burns guards make routine patrols of all properties within the site boundary outside of the restricted area and are required to make hourly reports to the licensed Central Control Room Operator relative to the number of guards present and the condition of the Indian Point properties. All incidents are called in immediately to the Central Control Room Operator.

In addition to the protection provided by the Burns guards, it is recognized that the security of the station against accidental or malicious intrusion, and the potential consequences thereof, is one of the most important responsibilities of the watch operating personnel. Toward this end, the following security steps are presently employed to insure such protection:

1 Once a watch, the Watch Foreman, General Watch Foreman, or Assistant Superintendent shall tour all secured boundaries, i.e., the city water meter house, the fuel oil pump house, the restricted area fencing, etc., making certain that all doors and windows leading into those areas are properly secured.

2. If there is any evidence of unauthorized penetration of Company property involving jeopardy to life, equipment, or unit reliability, which, in the judgment of Watch Supervision involves the need for police protection, the emergency operator No. 9 shall be contacted and that protection requested.

3. In the event of unauthorized penetration of the plant, or any restricted portion thereof, an immediate review of all systems and equipment within that area of the plant is to be made by the operator responsible for that area, and any damage or impairment of function evaluated and corrected as quickly as possible.

4. Should any unauthorized entry be discovered by any member of the watch force, immediate notification of the General Watch Foreman shall be made by way of the Central Control Room and the page system, and an evaluation made by watch supervision as to whether police help is required to remove the intruding party.

All routine inspections and communications with the station security personnel shall be made a matter of Central Control Room log entry. Any unusual security occurrences shall be made a matter of General Watch Foreman's log entry and notification of the General Superintendent.

As is the case for Unit No. 1, the plant is considered as two separate sections as far as accessibility is concerned. These sections are the Conventional portion of the plant which includes the turbine hall, heater bay building, screenwell area, central control room etc. and the Controlled Area. The Controlled Area, as defined in Title 10 CFR 20 as a Radiation area, implies that the area is one that requires control of access, occupancy and working conditions for radiation protection purposes. The Controlled area for Unit No. 2 will include the containment building, the fuel storage building, the primary auxiliary building and the emergency diesel generating building.

Normal access to the Unit No. 2 Controlled area will be through the existing Security Room for Unit No. 1 located in the Unit No. 1 Nuclear Service Building. All other doors and hatches leading into the Controlled Area will be locked and will be supervised by means of door switches connected to the open-door alarm board in the Security Room, and the category alarm board in the Unit No. 1 Central Control Room.

Routine entry to the Controlled Area will be permitted only to personnel designated by the Production Department General Superintendent or his authorized representative and who meet the following qualifications:

1. Have received special radiological safety training.

2. Have Radiation Record on file in the Health Physics Office and have entry card filed in Security Room.

3. Have received required Medical Examination.

Infrequent entry into the Controlled Area for short periods of time may be permitted to other Company employees or to visitors properly authorized by the General Superintendent and escorted by a qualified guide.

QUESTION 12.7

Describe the extent of access to various portions of the facility (as indicated above) anticipated for:

- Members of the general public.

- Offsite Con Ed employees.

- Offsite non-Con Ed technicians and engineers.

Describe the procedures to be used in admitting the above groups.

ANSWER

The extent of access to various portions of the facility and admitting procedures anticipated for members of the general public, offsite Con Edison employees and offsite non-Con Edision technicians and engineers is as follows:

a) Members of the general public

The Company maintains an observation and exhibit building at the Indian Point site which is open to the general public during certain days of the week. As part of the program presented to the public, escorted tours are made to the plant during which visitors are shown the turbine hall and are permitted to view Central Control Room operations through a glass enclosed booth which projects into the west end of the control room. The routes taken by the visitors are clearly defined within the plant and individual groups are under a qualified escort at all times. Each visitor is furnished an identification badge before leaving the observation and exhibit building and is instructed to wear the badge in full view at all times when in the plant.

b) Offsite Con Edison employees

All offsite Con Edison employees who visit the facility are required to present proper identification to the guard on duty at main gate house to the restricted area and to sign in on a Report on Admission Sheet. Before the employee enters the restricted area however, confirmation of the visit is made by the guard on duty with the General Superintendent or his designated representative. The employee is instructed to wear his Company identification badge in full view at all times while in the station.

Where access to the Controlled Area is required by the employee during his visit, special authorization must be obtained from the General Superintendent or his designated representative. Upon authorization to enter the Controlled area, the employee is assigned a qualified guide who acts as an escort for the length of stay in the area.

c) Offsite Non-Con Edison Technicians and Engineers

The extent of access for offsite non-Con Edison technicians and engineers will essentially be dictated by a "need to know" criterion. Depending on the particular area of interest by an individual, access may be limited to the conventional portion of the plant or may be expanded to include the Controlled Area should the need arise.

The procedures for admitting personnel in this category will essentially be the same as that for offsite Con-Edison employees. After signing the Report on Admission Sheet and having been approved for entry by the General Superintendent or his designated representative, the individual will be issued an identification badge or button and instructed to wear same while in the plant.

As with offsite Con Edison personnel, access to the Controlled Area will only be granted upon special authorization by the General Superintendent or his designated representative. When access to the Controlled Area has been authorized, the individual will be assigned a qualified guide who will act as an escort for the length of stay in the area.

QUESTION 12.8

Provide personnel resumés for the Superintendent Performance, Supervisor Engineering (Health Physics), Assistant Superintendent (Maintenance), Assistant Supervisor Engineering (Nuclear Plant Instrumentation, Health Physics and Conventional Plant Instrumentation, and the remaining General Watch Foremen.

ANSWER

Refer to the response to Question 12.1, pages Q 12.1-12 through Q 12.1-66, as revised by Supplement 12, and the response to Question 13.4, pages Q 13.4 (3)-3 through Q 13.4 (3)-5, in Volume 5 to the FSAR.

Question 12.9

Indicate, relative to Figure 1, Section 12, Supplement No. 2, the anticipated number of individuals under the following job titles: Maintenance Mechanics, Technical Assistants (Chemist), Senior Production Technicians (Shift Chemist), Production Technicians (Chemist), Senior Production Technicians, Production Technicians (Performance), Technicians (Nuclear Plant Instruments), Technicians (Shift Health Physics) and Technicians (Conventional Plant Instruments).

Answer

Refer to Figure 1 of Question 12.1 in Volume 5 to the FSAR as revised by Supplement 12.

Question 12.10

Indicate on Figures 1 & 2, Section 12, Supplement No. 2, all positions for which you intend to license personnel on Unit No. 2; whether the licenses are Senior Operator Licenses or Operator Licenses; and whether these persons will be "cold" or "hot" licensed.

Answer

Refer to Figures 1 and 2 and page 12.1-11c of Question 12.1 in Volume 5 to the FSAR as revised by Supplement 12.

Question 12.11

Has the staff of the Superintendent Performance and/or the staff of the Supervisor Engineering (Health Physics) been expanded for Unit No. 2 operation and if so, describe the specific training received by the new personnel, including course content and number of hours. Describe the training to be received by the Superintendent Performance, Assistant Superintendent (Maintenance) and the Supervisor Engineering (Health Physics), including course content and the number of hours.

Answer

Refer to the response to Question 12.1, in Volume 5 to the FSAR, pages Q 12.1-11 to Q 12.1-11b, as revised by Supplement 12.

QUESTION 13.1

Provide a description of the primary coolant system vibration tests which will be performed for Indian Point 2. Include the number type and location of the instruments for each test and state flow conditions for each test. Discuss the need for any such instrumentation to remain available during plant operation.

ANSWER

REACTOR COOLANT SYSTEM VIBRATION TESTING PROGRAM

Two tests programs will be performed on the Indian Point Unit No. 2 Reactor Coolant System to measure the dynamic behavior. The two programs are I. Reactor Coolant System Impedance Test, and II., Reactor Internals and Reactor Coolant System Loop Vibration Test Under Steady State and Transient Conditions.

I. Reactor Coolant System Impedance Test

The purpose of the impedance test is to determine the natural frequencies, mode shapes and damping of the main components of the Reactor Coolant System. These tests will be performed with the Reactor Coolant System filled with water and will be performed prior to the installation of the core and control rods. The reactor coolant and charging pumps will not be in operation during the test.

Electromagnetic shakers will be attached at several points on one of the Reactor Coolant System loops so that normal modes of the structure can be excited. Accelerometers will be used to measure the response of the structure. The mode shape and damping at the natural frequencies will be deduced from acceleration measurements made at several points on the structure while vibrating at a natural frequency. The shaker will be attached at selected locations on the steam generator, reactor coolant pumps and loop piping, for example, the present plans are for the following locations:

a. Steam Generator No. 21, ~65' elevation, circumferentially (i.e., tangential to the wall of the vapor container).

b. Steam Generator No. 21, between the 100' elevation and the 120' elevation.

c. Main Coolant Pump No. 21, ~62' elevation, circumferentially.

d. Main Coolant Pump No. 21, ~83' elevation, circumferentially.

e. Main Coolant Pump No. 21, ~83' elevation, radially.

f. Intermediate Leg, Loop No. 21, ~54' elevation, radially.

Thirteen monitoring accelerometers will be attached to the structure at the locations specified in Table Q 13.1-1 under external transducers. In addition one or two hand held accelerometers will be moved from point to point to establish the exact mode shape. All shakers and accelerometer cables will be routed to a readout station from which the excitation will be controlled and response will be measured.

An initial impedance plot will be obtained by exciting the structure at a constant, low force level from a frequency not less than 1 Hz. to a frequency not greater than 300 Hz. This will be followed by additional sweeps at higher force levels to facilitate detection of natural frequencies which have relatively low response. A determination of the mode shape at each natural frequency of interest is made by measuring the amplitude and phase of the acceleration response at a large number of points relative to the drive point.

Data from which damping can be deduced will be obtained by suddenly opening the electrical input of the shaker while driving at a natural frequency and recording the resulting decrement.

II. Steady State and Transient Internals and Loop Vibration Measurements

The objectives of the instrumentation program for the second phase of testing are as follows:

1. To obtain data that provides increased confidence in the adequacy of the internals structures by establishing the design margins at key locations on the structure. The strain gage and maximum displacement indicators are used primarily for this purpose.

2. To obtain data that can be used to develop improved analytical tools for the prediction of internals vibrations. Comparison with the 1/7 scale model data and establishing model validity are part of this objective.

Instrumentation will be provided for the reactor coolant system major components, i.e. reactor vessel, reactor internals, reactor coolant piping, reactor coolant pumps, and steam generators.

Reactor Vessel and Loop Piping

Six accelerometer will be located on the vessel, three on the vessel head studs and three on the bottom of the vessel as in Part I (shown in Table Q 13.1-1). The six vessel tranducers are arranged so that the rigid body motion of the vessel can be measured. The loop piping will be instrumented with pressure tranducers which will be installed in temperature wells on an inlet and outlet leg. In addition, the data from the external transducers will be correlated with the internals data in an attempt to establish remote estimation of internals motions.

Steam Generators

One of the steam generators will be instrumented the same as described in Part I (see Table 13.1-1) of the testing program and will measure its gross motion. The dynamic analysis performed on the steam generator tube bundle is described in the response to Question 4.7.

Reactor Coolant Pumps

One of the reactor coolant pumps is equipped with three accelerometer mounted at the top of the motor support stand (See Table Q 13.1-1). They are mounted in a horizontal plane to pick up circumferential and radial vibrations of the pumps. Prior to vibration testing (during preoperational tests) the reactor coolant pumps are checked to ensure that they are within limits. The balance and alignment can be adjusted if they are not within limits initially as described in the response to Question 4.7.

Reactor Internals

The reactor internals will be monitored with strain gages, accelerometers, pressure tranducers, and maximum-displacement indicators. There will be 46 strain gages, 14 accelerometers, 5 pressure tranducers and 14 maximum-displacement indicators.

The instrumentation will be utilized as follows:

1. Guide Tubes - The instrumentation to be used on the guide tubes is the same as that used on the Zorita and Ginna reactors. Three guide tubes are to be instrumented with strain gages. The central guide tube was selected because it would have no set cross flow velocity during four pump operation; a guide tube near the outlet nozzle at $\sim 150°$ was selected because it is expected to have the highest cross flow velocity with the initial complement of guide tubes; and a guide tube near the opposite outlet nozzle at $\sim 330°$ was selected because it is expected to have the highest cross flow velocity for plutonium recycle operation. In the 1/7 scale model tests for Indian Point No. 2, the guide tube located at $\sim 150°$ was similarly instrumented. This data will be used to compare the scale model with the actual plant.

The response of the guide tubes over the expected range of vibration frequencies will be measured with strain gages and accelerometer to provide strain versus amplitude data and to assure that the proper location for the strain gages has been chosen prior to installation in the reactor vessel.

2. Core Barrel

Upper Core Barrel - Strain will be measured at two locations on the core barrel; 1) just below the core barrel flange and 2) at the upper to lower core barrel weldment which is a reduced cross section elevation (See Table Q 13.1-1).

In addition, an axial strain gage will be placed on the outside surface of the barrel, radially inward from the centerline of an inlet nozzle. This gage will be used to obtain an indication of the stress due to the ram effect of the inlet flow against the core barrel and to compare with previous data taken at this location on the 1/7 scale Indian Point No. 2 model, the 1/13 ENEL/SENA model and the Obrigheim plant.

Accelerometers are located on the upper core barrel to determine the vibration of the upper core barrel in its shell modes. This information should contribute significantly to understanding the upper barrel strain gage readings.

Accelerometers have also been placed on two thermal shield support blocks to obtain information on the vibration of the core barrel in its ring modes and beam modes. Data are available from the 1/7 scale model at similar locations.

3. Thermal Shield

Thermal Shield - The measurement of the maximum stress in the thermal shield with a reasonable number of strain gages is impossible because of the number and non-uniform spacing of supports and the flexibility of the core barrel. The most highly strained bolt that fastens the top of the shield to the core barrel will be instrumented with four strain gages. One of the four gages is redundant so that loss of one gage will not result in the loss of all information from this location. To measure the desired strains, the gages will lie in a vertical plane passing through the core centerline when the final torque on the bolt is reached (See Table Q 13.1-1).

Three flexures are to be instrumented. The locations of the gages are at 0°, 90° and 240°. These gages will provide the data needed to determine the forces in each of the instrumented flexures.

Three accelerometers are located at the mid-elevation of the shield and one near the bottom to provide data to assist in the interpretation of the strain gage results and to compare with 1/7 scale model data. Supporting data will be obtained from model and full-scale impedance tests.

Pressure measurements will be made at the inside and outside wall of the thermal shield. Four pressure transducers to measure the fluctuating static pressure have been located near the top (82 1/2°) and bottom (28°) of the thermal shield.

Fourteen maximum displacement indicators will be installed into the thermal shield snubber holes which are not occupied by pressure transducers (eleven at the upper end and three at the lower end).

The maximum decrease in the proximity of the thermal shield to the core barrel and the vibratory motion of the thermal shield relative to the core barrel is obtained from these indicators by interpretation of stylii scratches.

4. Upper Core Plate

Upper Core Plate - Four accelerometers on the upper core plate will be used to define the horizontal motion of the upper core plate. This information will be used to determine the degree to which base motion excites the guide tubes and support columns (Refer to Table Q 13.1-1).

5. Top Support Plate

Top Support Plate - A pressure transducer will be mounted on the top support plate so that it is sensitive to vertical pressure fluctuations in the upper plenum. In addition to providing pressures in the upper plenum it will be useful in relating the other pressure transducer signals to each other. A pressure transducer was placed in a similar location in the Obrigheim reactor.

Instrumentation Description

Trandducers measuring strain, acceleration and pressure as well as maximum displacement indicators will be used.

1. Strain gages - The strain gages will be integral lead gages similar to those used for the Zorita and Ginna experiments. The minimum sensitivity is greater than 3 μ in/in from .0 Hz to 1000 Hz.

2. <u>Accelerometers</u> - Piezoelectric accelerometers having a sensitivity of ∿ 200 pc/g are to be used with resolution expected to be greater than .005 g from 5 Hz (± 0.002 inches) to 1000 Hz.

3. <u>Pressure Transducers</u> - Piezoelictric pressure transducers will be used which have a resolution of 0.2 psi. The diaphrams of the pressure transducers will be flush with the surface where pressure is being measured.

4. <u>Maximum-Displacement Indicators</u> - The maximum-displacement indicators are similar to those used in the Zorita and Ginna experiments. The internal spring-loaded plunger within the displacement pin is designed to follow the relative cyclic motion between the thermal shield and core barrel, thus causing the two stationary spring-loaded stylii to leave small markings on the plunger. These marks provide a direct indication of the magnitude of the vibratory motion. The displacement indicators consist of a cylindrical pin held by means of a clamping fit within a housing block mounted on the thermal shield. The pin is assembled and adjusted within the block so that it is tight against the outer diameter of the core barrel. Sufficient clamping force is exerted on the pin to assure that the pin will move within the housing block only by a relative motion of the thermal shield toward the core barrel. This will create a gap between the end of the pin and the core barrel that can be measured during the post hot functional inspection. These measured gaps will provide an indication of the total relative motion between the thermal shield and core barrel resulting from thermal differential expansion, hydraulic forces and vibration.

<u>Test Conditions</u>

For Part II tests the following conditions will be required:

1. During cold hydrostatic testing, data is to be taken at one primary coolant temperature (less than 150°F). This temperature will be established by the temperature that exists when time for the testing occurs in the schedule. The temperature will be kept within ± 20°F during the testing.

2. During the hot functional tests, data is to be taken at a low temperature (less than 150°F) and at the maximum test temperature. Again, the main coolant temperature will be kept within ± 20° while data is being taken. During heat up, a selected number of instruments will be monitored continuously.

3. At the completion of hot functional testing, it is currently planned that all instruments be removed except six strain gages on two guide tubes, three strain gages on the core barrel, one pressure transducer on the top support plate and the thirteen accelerometers on the outside structure. These instruments will be monitored during pre-critical testing after the core has been loaded. The measurements will be made on these instruments for steady state and transient conditions. Data will be taken during control rod exercising, with and without moving the rods in the instrumented guide tube at the same temperature condition as specified in 1.) and 2.). For the above tests, data will be recorded during startup transients, shutdown transients and steady flow with several combinations of reactor coolant pumps running including each pump operating individually and all 4 pumps operating simultaneously. At the first refueling, the internal transducers will be removed.

The Reactor Coolant System testing program, outlined above, when coupled with experience from off-site testing, model testing and data from other recent testing programs on operating plants provide assurance that in-service vibration monitoring instrumentation is not required. This subject is discussed further in the response to Question 13.2.

TABLE Q 13.1-1

TRANSDUCER LOCATIONS FOR VIBRATION EXPERIMENTS

STRUCTURE	OUTER WALL	INNER WALL	ELEVATION	ANGLE	DIRECTION OF SENSITIVITY	ACCELEROMETER	PRESSURE TRANSDUCER	STRAIN GAGE
CORE BARREL	X	X	Upper Core Barrel	0°	A			2
	X	X	Below Flange	90°	A			2
	X	X	Weldment	270°	A			2
	X		Behind Inlet Nozzle	67 1/2°	A			1
	X	X	Weldment Upper-	0°	A			2
	X	X	Lower Core Barrel	0°	C			2
	X	X		90°	A			2
	X	X		90°	C			2
		X	Nozzle Elevation	0°	R	1		
	X			45°	R	1		
		X		90°	R	1		
		X		270°	R	1		
	X		On Thermal	22 1/2°	R	1		
	X		Shield Support Blocks	112 1/2°	R	1		

A = Axial

C = Circumferential

R = Radial

TABLE Q 13.1-1 (Cont'd)

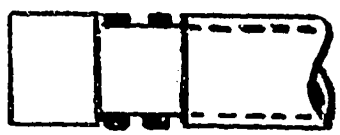
STRUCTURE	OUTER WALL	INNER WALL	ELEVATION	ANGLE	DIRECTION OF SENSITIVITY	ACCELEROMETER	PRESSURE TRANSDUCER	STRAIN GAGE
THERMAL SHIELD	X	X	Snubber Pin Holes — Top	82.5°	R		2	
	X	X	Snubber Pin Holes — Bottom	28°	R		2	
	X		Mid Elevation	0°	R	1		
				90°	R	1		
			Flexures	0°	R			6
				90°	R			6
				240°	R			6
			Top Support Bolt 	67 1/2°	R			4
	X		Mid. Elev.	270°	R	1		
	X		Near Bottom	90°	R	1		
UPPER CORE PLT.			Top Surface	0°	R	1		
				0°	C	1		
				180°	R	1		
				180°	C	1		
TOP SPT. PLT.			Bottom Surface		A		1	

TABLE Q 13.1-1 (Cont'd)

STRUCTURE	OUTER WALL	INNER WALL	ELEVATION	ANGLE	DIRECTION OF SENSITIVITY	ACCELEROMETER	PRESSURE TRANSDUCER	STRAIN GAGE
GUIDE TUBE			Near Top Supt. Plt.					
			Pos. D-14 (Plut. Recycle)					3
			H-8 (Center)					2
			K-2 (Max. Vel.)					4
EXTERNAL TRANSDUCERS								
VESSEL	X		Vessel Head Studs	0°	C	1*		
	X			0°	R	1*		
15	X			180°	C	1*		
	X		Bottom of Vessel	---	R	1*		
	X				C	1*		
	X				A	1*		
	X		Inlet Leg (21, 22 & 24)				3	
	X		Outlet Leg (21)				1	
Steam Generator No. 21			~65' (Support Pad Elev.)		C	1*		
			~120' (Near Top)		C	1*		
					R	1*		
Main Coolant Pump No. 21			~62' (Support Pad Elev.)		C	1*		
			~83' (Top Motor Flange)		C	1*		
					R	1*		
Intermediate Leg (Loop 21)			~54' (Center of Pipe)		R	1*		
Containment Floor			~46'		R	1		
					C	1		

*These instruments in addition to portable accelerometers will be used during the Impedance Test to determine mode shapes.

QUESTION 13.2

Discuss the possible means of in-service monitoring for vibration and the presence of loose parts in the reactor pressure vessel and other portions of the primary system and your plans to implement such means as are found practical and appropriate.

ANSWER

The design of Indian Point Unit No. 2 reactor and associated equipment is based on extensive analytical, test and operational information. Westinghouse PWR operating experience to date has been evaluated and the information derived has been incorporated where appropriate in this design. For example: (1) The Indian Point Unit No. 2 design uses a one piece thermal shield which is attached rigidly to the core barrel at one end and flexibly attached at the other. The early designs were multi-piece thermal shields that rested on vessel lugs and were not attached. (2) The Indian Point Unit No. 2 design uses a one piece core barrel. The early core barrel design had upper and lower cylinderical sections bolted together. An example of a design modification based upon operating reactor experience occured on the original San Onofre thermal shield. The thermal shield originally supplied for the San Onofre station was not rigidly supported. Evidence of vibrations found when the internals were inspected after hot functional testing before completion of plant construction indicated that modifications were required. As a result, the San Onofre thermal shield support was modified to the Connecticut Yankee design and hot functional testing was repeated. Inspection after this testing showed that vibration was reduced to insignificant levels.

'oint reactor internals are similar to those used in the Ginna, Co Yankee, and Zorita reactors. Tests performed in these reactors have med the design used does not have significant vibrations. The results of the Ginna tests, reported in Westinghouse proprietary report WCAP-7468-L, add additional assurance that the internals design is not subject to significant vibration levels. The testing performed on the Indian Point No. 2 reactor coolant system will include similar tests as described in the response to Question 13.1.

The reactor coolant pump vibration is monitored for alignment and balance prior to the loop tests. Excessive vibrations from the pumps can be eliminated by on-line adjustment. In addition, vibration monitors will be mounted on the pumps as described in the response to Question 4.7 and 13.1.

The steam generators have been extensively studied for tube bundle vibration as stated in the response to Question 4.7. The analysis and testing of similar steam generators indicate that tube bundle vibration is insignificant. During the vibration testing, described in the response to Question 13.1, accelerometers will be mounted on a steam generator to monitor system interaction.

One loop will be instrumented on Indian Point No. 2 to measure pressure fluctuations and vibration levels in the loop. These measurements will establish the frequencies that exist in the loop from the reactor coolant pump operation and verify that the levels of vibration are low.

The design of the Indian Point reactor coolant system reduces the probability of loose parts within the system to a very low level. The design of the reactor internals has been based upon eliminating sources of loose parts. Information from investigations of instances where loose parts have occurred is factored into the design. Furthermore, since loose parts are most commonly generated from excessive vibration in operating systems, the lack of significant vibration levels in the Indian Point reactor coolant system (confirmed by actual plants in operation and to be verified preoperational tests described in the response to Question 13.1) make inservice loose part monitoring unnecessary.

Some of the instrumentation in the vibration testing program will be left on the reactor coolant system (in particular the reactor vessel, two guide tubes, the core barrel, and the top support plate) following the hot functional tests to provide additional data during the low power testing.

It is thus considered unnecessary to provide for inservice vibration and loose parts monitoring on the Indian Point Unit No. 2 Reactor Coolant System. The experience, analysis, research and proposed full scale testing program combined are considered adequate to provide the necessary assurance that excessive vibration and loose parts are not concerns.

QUESTION 13.3

We understand that "preoperational" tests are to be performed to verify the stability of the reactor with respect to potential xenon oscillations in the X-Y plane. Please describe these tests in detail. What in-service testing, monitoring, or surveillance is to be performed to assure continued X-Y stability during reactor lifetime?

ANSWER

Since axial instability is anticipated and means for detection and control provided, only tests to demonstrate diametral stability will be necessary at startup.

The test sequence proposed would be approximately as follows:

a) Bring plant to rods-out (to withdrawal limit) xenon-equilibrium at a power level as near full power as reasonable without exceeding safety limits anticipated after the power anomaly is inserted. This will take about one day at most, depending on prior operating history.

b) Manually insert one off-center control cluster, holding power steady with boron dilution of approximately 10-15 ppm.

c) Wait one hour, then withdraw the cluster, holding power steady with boron injection, of about 10-15 ppm.

d) Hold power steady for about 120 hours (the longer the run, the clearer the results) periodically (4-hour period) logging ex-core currents and thermocouple readings. Partial moveable in-core power maps will also be of interest. The partial maps would be made on a 4-hour or 8-hour period.

If fixed in-core signals are available, those signals will also be collected.

e) Evaluation: The results will be satisfactory if the observed diametral oscillation converges. In the unexpected event that results are not satisfactory, one or more of the proposed backups will be employed, and the test will be repeated.

Although corrective action is not expected to be necessary, the backup options which are available are as follows: (For details see WCAP-7407-L, Section 3.1.3.[1]),

1) Insert control rods in order to reduce the required boron concentration and thereby increase the negative moderator coefficient feedback effect. At BOL, when the problem might arise, there is an excess shutdown margin under hot full power conditions (see IP-2 FSAR, Table 3.2.1-3); consequently, some of the control rods could be used to increase the reactor diametral xenon stability, if needed, without compromising the reactor safety (i.e., minimum required shutdown margin will be maintained).

2) Add poison shims (burnable or fixed) to accomplish an increase in reactor stability as indicated above. The number and location of these poison shims (to avoid power peaking problems) would have to be assessed after the first oscillation test reveals the magnitude of corrective action required.

3) Operate at a reduced power level to assure reactor stability and the existence of adequate safety margins. Operation at reduced power would continue until burnup has sufficiently reduced soluble boron concentration to result in a stable reactor response to a diametral oscillation.

---

[1] R.F. Barry et al., Power Maldistribution Investigations, Report WCAP-7407-L (Proprietary Class 2), Westinghouse Electric Corp.

In any event, the reactor will not be operated at full power until tests, as described above, have verified that the reactor is stable toward diametral xenon oscillations.

As described in the IP-2 FSAR, Appendix 3-B, instrumentation is provided to continuously monitor core performance for the existence of any diametral xenon oscillations (power tilts) and to alert the operator to any power anomalies. Further assurance is provided by the following instrument checks:

a) Control rod linear position indicators will be checked against the digital position counters when rods are moved.

b) Out-of-core detector response will be checked against the core thermocouples during power level change.

c) Out-of-core detector response will be checked against the moveable in-core detector response during power level change.

d) Out-of-core detectors will be tested in accordance with the technical specification requirements.

These provisions give assurance that throughout reactor lifetime the core can be maintained within thermal limits (designed nuclear hot channel factors).

QUESTION 13.4

With regard to the startup organization:

1. Identify chains of responsibility and authority for all groups participating in the initial tests and operation of the facility, including Westinghouse Support Groups.

ANSWER

The staffing, training and experience of the proposed operating organization for Indian Point No. 2 is described in detail in Section 12.1 of the Final Facility Description and Safety Analysis Report, and Con Edison's reply to Question 12.1 in Supplement No. 2. The chains of responsibility and authority for those Con Edison site personnel participating in the initial testing and operation of the facility will be the same as that described in Section 12.1. Additional support will be drawn, when required, from various technical support groups within the company. The six major technical support groups within the Company are the Inside Plant Bureau of the Electrical Engineering Department, the Nuclear Engineering Bureau and Mechanical Plant Engineering Bureau of the Mechanical Engineering Department, the Plant Sturctures Bureau of the Civil Engineering Department, the Design Engineering Bureau of the General Engineering Department, and the Technical Services Bureau of the Technical Services and Meter Department.

Due to the "turnkey" nature of the Indian Point Unit No. 2 contract, close cooperation between Con Edison and Westinghouse during all phases of startup testing and initial operation will be essential to insure that all procedures are executed in a safe and efficient manner. Toward this end, one of the Station's Production Engineers has been assigned to the position of Test Coordinator. The Test Coordinator, as a member of the Unit No. 2 startup group, will be responsible for reviewing all test procedures to determine that operations will be conducted in accordance with Company rules and regulations and the provisions of the facility Technical Specifications. In addition, the Test Coordinator will provide laison services between the Company and Westinghouse relative to all testing activities and will participate in the review and evaluation of all test results. The Test Coordinator will report directly to the Unit No. 2 Operations Superintendent.

The organization chart for the Con Edison start up group is given in Figure 13.4.1-1. The organization chart for the WEDCO operations is given in Charts I and II (Figures 13.4.1-2 and 13.4.1-3).

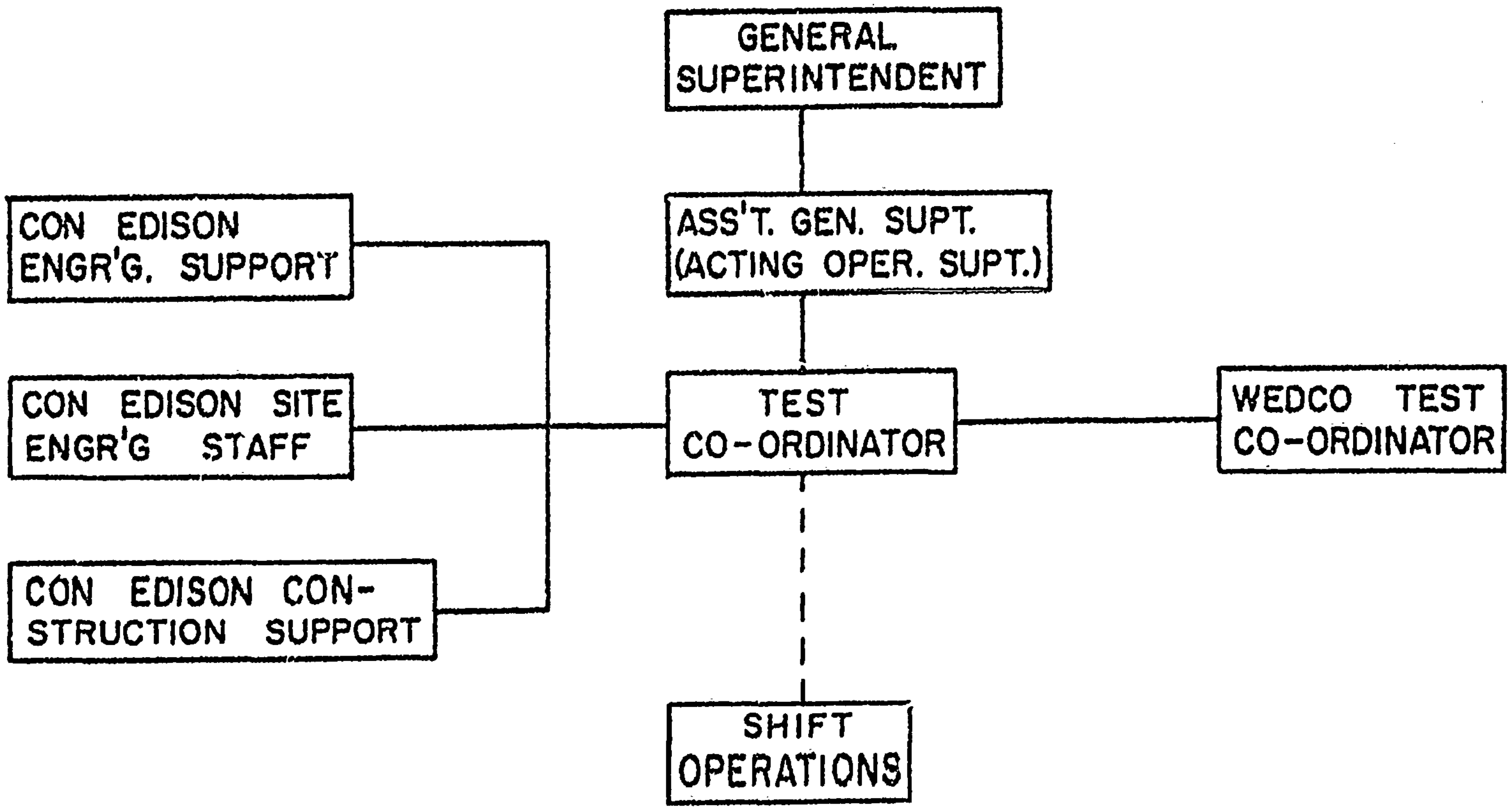
Chart I:

Staffing of the WEDCO Operations group has been fulfilled with the intent of satisfying two major objectives. The first objective was the planning and scheduling of test procedure issue and the necessary writing, review and approval of test procedures. Concurrent with this effort, flushing and hydrostatic tests are to be conducted as limited by construction completion. Chart No. I shows the organization created, and currently in existence, to satisfy this objective.

Chart II:

The second objective was to staff and modify the organization to perform a multi-shift test program when construction is essentially complete. This organization will technically direct the test program on shift through assignment of Startup Directors. The Startup Directors will be selected on the basis of proven competence and experience during the period of preliminary testing described above. They will report to the Startup Manager. In support of this on-shift test effort, the Test Program Manager will continue to supervise test procedure writing and revision, material coordination, technical support requirements to permit the shift testing organization to direct the test program at maximum productivity. Chart No. II shows the organization that will be created to meet these objectives.

The responsibilities, authority, and position requirements for each position are given in the individual position descriptions and resumes of Question 13.4.2.



CON EDISON ORGANIZATION
FOR
START-UP TEST PROGRAM
UNIT NO.-2

FIGURE 13.4.1-1

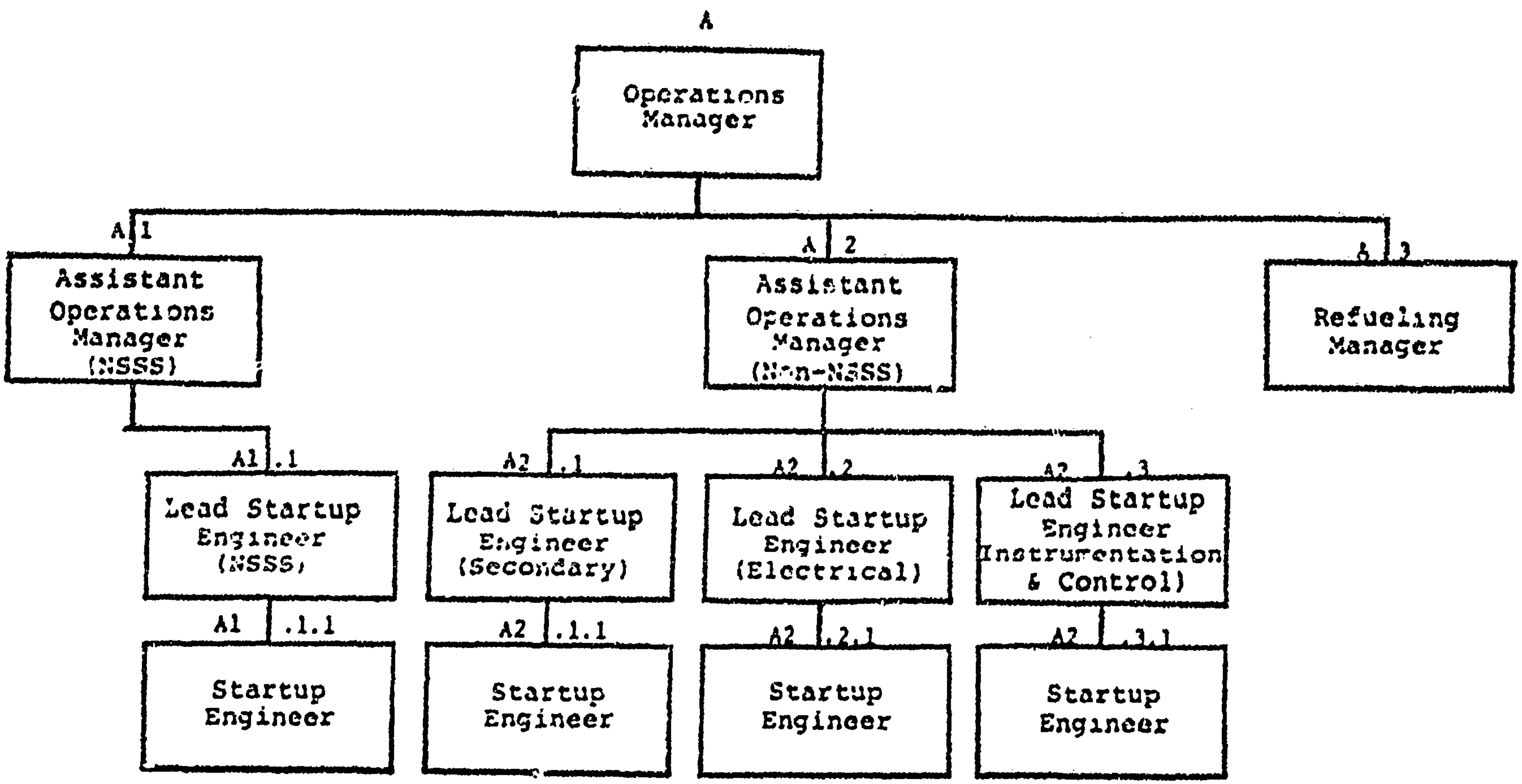


CHART I

Figure 13.4.1-2

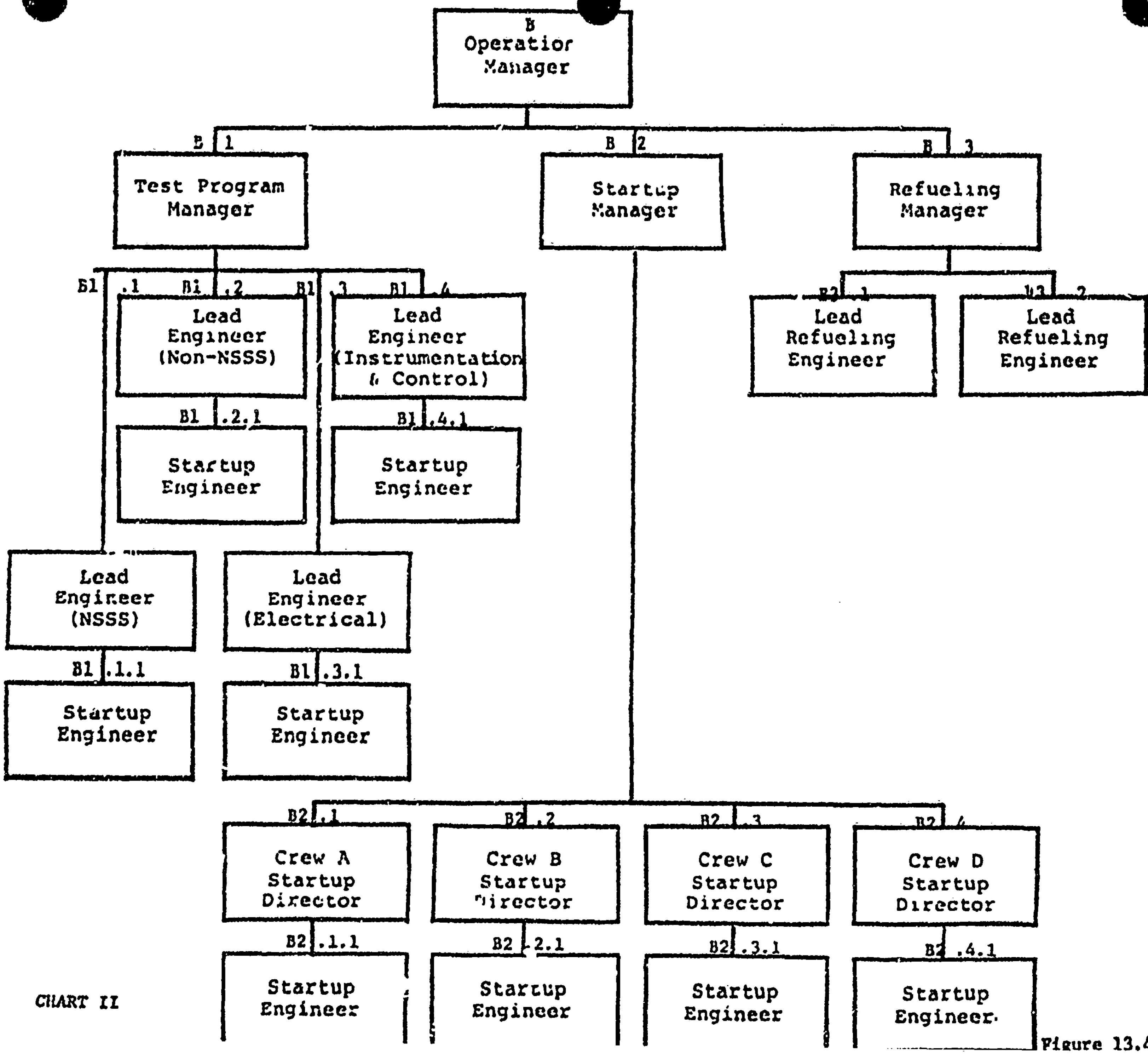


CHART II

Figure 13.4.1-3

QUESTION 13.4

With regard to the startup organization:

2. Submit personnel resumes for Westinghouse personnel participating in or acting as support during the initial tests and operation of the reactor.

ANSWER

Each position given in the charts for the response to Question 13.4.1 is described on the following pages of job descriptions and personnel resumes.

Position Title: Manager, WEDCO Operations (Chart I "A")

Primary Function:

Manage the activities of the operations section in conduct of plant startup, including core loading, from point of construction completion through commercial operation. This responsibility includes issue and approval of all test procedures and manuals required for control of above activities.

Duties and Responsibilities:

1. Plan and schedule work load, assign work; recommend budgets, control expenditures, select and train subordinates, review performance of subordinates, recommend wage or salary adjustments, report results and unusual situations to immediate supervisors.

2. Make Decisions and Take Action

   a. Establish the Operations organization chart, initiate job descriptions and duties, hire and assign personnel.

   b. Formulate policies and procedures to direct activities of subordinates. See that policies and procedures are correctly implemented.

   c. Direct the activities of the Startup Manager, Procedures Supervisor, Refueling Director and Training Coordinator.

   d. Determine limits of responsibilities for each department (Startup, Training, Procedures, Refueling).

   e. Establish working relationship with other WEDCO departments (construction, engineering, financial, etc.).

f. Supervise the immediate and follow-up action taken in the event of significant startup problems with equipment/components.

g. Superv se action, prior to plant/system acceptance, in event plant/systems be operated beyond approved limits. Evaluate need for repair/replacement and propose methods to prevent similar problems.

3. Maintain Relationship with other Departments and Agencies

a. Request services of chemistry, radiological control, quality assurance, etc. groups. Implement policies of these groups with respect to operations groups responsibilities.

b. Request services of vendors, supp.iers, other Westinghouse technical and field groups. Direct their activities at the site as necessary.

c. Maintain close relationship with utility management personnel in coordination of their needs and demands with WEDCO policies and schedules.

d. Provide technical and practical information on test program to engineering and design groups to continually improve method, procedures and schedules.

e. Communicate with other Turnkey projects to ensure "lessons learned" are exchanged and used to continually improve pace and progress at Turnkey sites.

4. Occasional Duties or Special Assignments

a. Serve as Chairman, Test Release Committee

(1) Provide final approval of equipment/system test conduct

(2) Determine adequacy of test procedure for safe & efficient conduct of test.

(3) Determine preparation of plant and personnel to safety conduct test.

b. Participate in "walk-through" and oral examination of utility trainees as requested to evaluate preparation for examination.

c. Participate in planning and scheduling meetings with other responsible section managers to determine short & long range commitments.

d. Provide recommendations for and supervise such other special tasks as directed by the Executive Vice President.

5. Problem Solving

a. Identify schedular delays before they occur and take action to eliminate them.

b. Evaluate schedular delays as they occur and take action to reduce their effect.

c. Revise procedures, manuals and directives to permit continued progress in meeting commitments.

d. Resolve utility requests and demands in satisfactory manner while maintaining targeted pace and goals.

e. Resolve critical problems identified by groups and individuals under my control. Assign responsibility as necessary, determine need for attention by higher authority.

f. Carry out testing and refueling program requirements of other Westinghouse NES design and engineering departments in coordination with Turnkey Programs and Schedules.

6. Decision Making

a. Determine adequacy of plant operating, testing, refueling, and training procedures and methods.

b. Determine preparation of plant and personnel to conduct testing, refueling assignments.

c. Determine and approve expenditures for materials & supplies to support activities.

d. Determine and approve expenditures for repair, modification and/or replacement of major components or systems or portions thereof.

e. Determine hiring, transferring and discharge of assigned personnel.

f. Determine need for rerates, reclassification of assigned personnel, and take action.

g. Determine necessity for above normal working hours and assign personnel, compensate as appropriate.

<u>Position Requirements:</u>

1. Education - High School, College (B.S. or Science degree)

2. Specialized or technical knowledge and skills - Must have substantial previous experience in operation, testing & maintenance of nuclear power plants. Must have completed formal technical training and qualification in nuclear plant operation including plant & system

design & construction, safeguards analysis, emergency procedures & environmental hazards. Should have previous core loading/defueling experience including criticality control and fuel handling procedures.

3. Types of work experience. Minimum number of years for each.

Plant Operation, Startup & Testing - non-supervisory - 6-8 years
Plant Operation, Startup & Testing - Supervisory - 4-7 years

Personnel Resume (Chart I "A")

Education:

High School Graduate - 1948

B.S. Marine Science - U.S. Merchant Marine Academy - 1953

Experience:

1953 - 1954 - Third Officer - Isthmian Steamship Lines - Watch Officer responsible for supervision of watch at sea and in port..

1954 - 1956 - U.S. Navy - Division Officer - U.S.S. Juneau, responsible for activities of gunnery division. Asst. Gunnery Officer and First Lieutenant U.S.S. Boston, responsible for coordination of activities of gunnery divisions and maintenance of all non-engineering spaces.

1957 - 1958 - Westinghouse Elect. Corp. - Lead Construction Follow Engineer - Responsible for group of six engineers in follow of construction and startup testing activities of A1W Prototype at the Naval Reactors Facility, Idaho Falls, Idaho.

1958 - 1960 - Qualified as Reactor Engineer at the A1W Prototype. Responsible for initial checkout and calibration of all reactor control and steam plant instrumentation on shift. Assigned to operations crew prior to initial criticality of first plant, remained as Reactor Engineer through full power testing and operation of first plant, initial criticality of second plant and initial dual plant power operation of both plants.

1960 - 1961 - Qualified as Chief Operator of S1W Prototype at the Naval Reactors Facility responsible for nuclear plant operations on shift. Qualified as Chief Reactor Technician on the S1W Prototype responsible for all instrumentation and control checkout, calibration and maintenance and protective system readiness for on shift. Served as Staff Asst. to S1W Plant Manager.

1961 - 1962 - Served as Technical Asst. to Operations Crew Supervisors. Assisted Crew Supervisor in direction of operations personnel on shift during power operation, reactor startup and shutdown, testing, refueling, shutdown for maintenance and modification and training of Naval Personnel.

1962 - 1963 - Served as Operations Crew Supervisor on shift with supervisory responsibilities for all shift activities in the S1W Plant.

1963 - 1965 - Served as Training Mgr. for the S1W Plant, responsible for the training program of Naval and civilian personnel assigned to S1W. Served as Oral Board Chairman during qualification of over two hundred Naval Officers and enlisted petty officers and civilians as Engineering Officers of the Watch.

1965 - 1968 - Served as Mgr. of S1W Operations, responsible for activities of four Operations Crew Supervisors in supervision of S1W Plant Operations. These duties included management of 35 Westinghouse employees, 120 Naval Officers and enlisted men in their duties during operation, testing, maintenance and training periods. These duties also included preparation of all personnel for periodic requalification by the Naval Branch of the AEC.

1968 - 1969 - Served as Shutdown Manager for the A1W Prototype. Responsible for management of the partial refueling of the A1W "A" plant, a first of its kind evolution. Also responsible for the A1W plant maintenance and modifications conducted concurrently with the Refueling Shutdown.

7-60 - 9-69 - Westinghouse Elect. Corp. - Turnkey Division - Reassigned to Turnkey Division as Operations Mgr. Turnkey Projects, Assisted General Mgr. Turnkey Division in coordination of Startup Activities of Turnkey Projects.

9-69 - Present -	Operations Mgr. - Indian Point Projects - Responsible for all operations activities for Indian Point Proj..ts - Units 2 and 3.

Position Title: Assistant Manager for Operations-NSSS (Chart I "A.1")

Primary Function:

Manage portions of the operational activities which are required to bring a new nuclear power plant to a licensible and acceptable operating condition.

Duties and Responsibilities:

1. Plan and schedule work load; assign work; recommend budgets, control expenditures, select and train subordinates, review performance of subordinates, recommend wage or salary adjustments, report results and unusual situations to immediate supervisors.

2. Plan the activities of operations section with regard to test procedure writing, review and issue, conduct of testing, manpower assignments, overtime compensation, control dates for start and completion of all above to properly carry out startup and acceptance of nuclear power plant.

3. Coordinate construction activities through establishing test starting dates and ensuring construction completion dates will support test and acceptance program.

4. Direct activities of construction personnel (foremen and craftsmen) during initial testing.

5. Determine engineering requirements as to system and equipment control parameters and report inadequacies of design.

6. Supervise preparation of operational information, test procedures, test results for issue to the customer and A.E.C. to document and prove acceptability.

7. Analyze, interpret and make recommendations on contracts with vendors. This includes attending contract negotiations meetings.

8. Supervise activities of Test Directors on shift in their direction of construction personnel, customer personnel, vendor representatives to prepare for, conduct and accept power plant systems and/or components. Resolve significant problems delaying test program, control costs, identify contractual conflicts or change, take action on all the above.

9. Direct personnel in completing test program expeditiously and safely while maintaining a good working relationship with and between construction bargaining unit crafts and customer personnel.

10. Basic duties involve supervision of the initial light-off of equipment, correction of deficiences, detailed operational testing, correcting deficiencies, integration of tested system/component into plant operation, documenting test results, concurrent training of customer personnel and acceptability of systems and plant by customer.

Problems solving in this accomplishment include review of design and engineering, supervising field correction of deficiencies and recommending corrections to technical groups that solve problems in line with schedules and without major modificaitons. Vendor field service personnel must be used in the field, work under our direction, correct problems to customer satisfaction within vendor contracts and warranties and on schedule. Test personnel under Asst. Mgrs. direction must be given liberty to satisfy test objectives using union personnel, customer personnel, technical engineering and design personnel, avoiding conflicts between groups, maintaining schedules, while supervision is imposed indirectly through Asst. Mgrs. instructions and policies. The length and extent of training balanced against schedules must be determined and agreement reached with the customer. Finally, the degree of documentation, meeting of operational objectives, deficiency corrections implemented to the satisfaction of the customer.

11. Decision Making

A. Make Final Decision

1. Determine adequacy of test procedures. Sufficient detail must exist to protect personnel and equipment. If tests are too involved, then a larger staff is required to write and conduct the tests requiring an increased budget.

2. Determine need for outside assistance from technical & vendor groups.

3. Commitments to management and the customer with regard to test schedules must be made

4. Determine need for temporary piping, wiring, power supplies, tank trucks, barges, chemicals, steam, water. Direct installation of temporary facilities to meet schedules.

B. Review with Supervisor

1. Determine staffing and types of personnel to support activity.

2. Contribute to overall testing schedule.

3. Recommend merit increases and overtime needs.

4. Establish commitments for other groups in support of test program and justify action to be taken by supervisor.

5. Recommend operations policies, administrative policies for approval.

6. Determine necessity of higher management attention in resolution of his operational activities.

Position Requirements:

1. Education - College (B.S. degree)

2. Specialized or technical knowledge and skills

   Must have previous experience in operation, testing & maintenance of nuclear power plants. Must have completed formal technical training & qualification in nuclear plant operation.

3. Types of work experience. Minimum number of ye rs for each

   Plant Operation - Startup & Testing - 3 - 5 years.

   Previous Supervisory Experience - 2 - 4 years.

   Design or Project Eng. experience is preferrable but not mandatory.

Personnel Resume (Chart I "A1")

Education

High School Graduate

B.S. Marine Steam and Electrical Engineering

California Maritime Academy, Vallejo, California - 1950

Experience

1950 - 1953 - Standard Oil Co. of California, Marine Dept. Third Assistant shipboard engineer in charge of engine room watch.

1953 - 1955 - U.S. Navy, Lieutenant Jr. Grade. Engineering Officer and/or Repair Officer on Landing Craft repair ship. Responsible for 30 enlisted men in engineering dept. operation and administration. As Repair Officer responsible for 40 enlisted men in the mechanical repair dept. which provided repair and maintenance service to other ships.

1955 - 1956 - Standard Oil Co. of California, Marine Dept. Second assistant licensed engineer in charge of engine room watch.

1956 - 1957 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory, Operation and Test Engineer. Qualified as Engineering Officer of the Watch (chief operator) prototype nuclear submarine installation. Responsible to plant operation crew supervisor for safe and efficient use of prototype plant during testing, training and maintenance periods.

1957 - 1959 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory. Senior Operations and Test Engineer.

During this period, was assigned as Engineering Officer of the Watch, previously described and part of original group responsible to compete and write power plant manual for operation, maintenance and training of S1W prototype nuclear plant.

1959 - 1960 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory. Staff Assistant to the Manager of operations of S1W prototype nuclear plant. Non-supervisory position responsible to carry out policy changes and over-see training and qualification of naval officers and enlisted personnel on S1W prototype nuclear plant. Administered final oral board qualificaiton examination to engineering of the watch trainees, both naval and civilian.

1960 - 1963 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory Plant Operations Crew Supervisor (POCS)

Responsible while on shift for the safety and operation of the S1W prototype nuclear plant, during training, testing, shutdown, depletion, refueling and maintenance periods. Direct supervision of thirty to fifty Naval and civilian personnel - indirect supervision of on-shift support personnel consisting of chemistry, industrial hygiene and craftsmen - recommend supervisory trainees for final written and oral board qualification examination.

1963 - 1969 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory Supervisor, Shift Operations, Expanded core Facility. Responsible for two engineers, twenty-five examniation technicians and nine support craftsmen conducting the examination of advance irradiated nuclear fuels and materials from prototype and test reactors, examination and preparation for disposal of spent fuel from all naval reactors, training and testing of personnel in nuclear fuel handling, radiation protection practices and quality control.

1969 - present - Westchester Engineering and Design Co. Assistant Manager for Operations.

Westinghouse Elect. Corp. Nuclear Energy System - Turnkey Projects - Assistant Manager for Operations.

Responsible for seven startup engineers, in writing, issuing test procedures for conducting of flushing, hydro, per-operational and acceptance tests of nuclear steam supply systems-ensuring the above are conducted in a safe and expeditious manner.

Position Title: Assistant Manager for Operations - non-NSSS (Chart I "A2")

Primary Function:

Manage portions of the operational activities which are required to bring a new nuclear power plant to a licensible and acceptable operating condition.

Duties and Responsibilities:

1. Plan and schedule work load; assign work; recommend budgets, control expenditures, select and train subordinates, review performance of subordinates, recommend wage or salary adjustments, report results and unusual situations to immediate supervisors.

2. Plan the activities of operations section with regard to test procedure writing, review and issue, conduct of testing, manpower assignments, overtime compensation, control dates for start and completion of all above to properly carry out startup and acceptance of nuclear power plant.

3. Coordinate construction activities through establishing test starting dates and ensuring construction completion dates will support test and acceptance program.

4. Direct activities of construction personnel (foremen and craftsmen) during initial testing.

5. Determine engineering requirements as to system and equipment control parameters and report inadequacies of design.

6. Supervise preparation of operational information, test procedures, test results for insue to the customer and A.E.C. to document and prove acceptability.

7. Analyze, interpret and make recommendations on contracts with vendors. This includes attending contract negotiations meetings.

8. Supervise activities of Test Directors on shift in their direction of construction personnel, customer personnel, vendor representatives to prepare for, conduct and accept power plant systems and/or components. Resolve significant problems delaying test program, control costs, identify contractual conflicts or change. take action on all the above.

9. Direct personnel in completing test program expenditiously and safely while maintaining a good working relationship with and between construction bargaining unit crafts and customer personnel.

10. Basic duties involve supervision of the initial light-off equipment, correction of deficiences, detailed operational testing, correcting deficiencies, integrating of tested system/component into plant operation, documenting test results, concurrent training of customer personnel and acceptability of systems and plant by customer.

Problems solving in this accomplishment include review of design and engineering, supervising field correction of deficiencies and recommending corrections to technical groups that solve problems in line with schedules and without major modifications. Vendor field service personnel must be used in the field, work under our direction, correct problems to customer satisfaction within vendor contracts and warranties and on schedule. Test personnel under Asst. Mgrs. direction must be given liberty to satisfy test objectives using union personnel, customer personnel, technical engineering and design personnel, avoiding conflicts between groups, maintaining schedules, while supervision is imposed indirectly through Asst. Mgrs. instructions and policies. The length and extent of training balanced against schedules must be determined and agreement reached with the customer. Finally, the degree of documentation, meeting of operational objectives, deficiency corrections implemented to the satisfaction of the customer.

11. Decision Making

A. Make Final Decision

1. Determine adequacy of test procedures. Sufficient detail must exist to protect personnel and equipment. If tests are too involved, then a larger staff is required to write and conduct the tests requiring an increased budget.

2. Determine need for outside assistance from technical and vendor groups.

3. Commitments to management and the customer with regard to test schedules must be made.

4. Determine need for temporary piping, wiring, power supplies, tank trucks, barges, chemicals, steam, water. Direct installation of temporary facilities to meet schedules.

B. Review with Supervisor

1. Determine staffing and types of personnel to support activity.

2. Contribute to overall testing schedule.

3. Recommend merit increases and overtime needs.

4. Establish commitments for other groups in support of test program and justify action to be taken by supervisor.

5. Recommend operations policies, administrative policies for approval.

6. Determine necessity of higher management attention in resolution of his operational activities.

Position Requirements:

1. Education - College (B.S. degree)

2. Specialized or technical knowledge and skills

   Must have previous experience in operation, testing and maintenance nuclear power plants. Must have completed formal technical training .d qualification in nuclear plant operation.

3. Types of work experience. Minimum number of years for each

   Plant Operation - Startup and Testing - 3-5 years.
   Previous Supervisory Experience - 2-4 years.
   Design or Project Eng. experience is preferable but not mandatory.

Personnel Resume (Chart I "A2")

Education:

High School Graduate

B.S. in Administrative Engineering, University of Denver, 1950

Experience:

1944 - 1946 - U. S. Navy, Phillipine Islands, Electricians Mate.

1946 - 1950 - University of Denver, Colorado.
B.S. in Administrative Engineering with a major in Chemical Engineering.

1950 - 1951 - Globe Chemical Company, Denver, Colorado
This company manufactured bleach and other household chemicals. I did chemical and analysis work and developed a napthalene purification process.

1951 - 1952 - U. S. Air Force, Wiesbaden, Germany. Special Detail.

1952 - 1956 - Army Chemical Corps (Civil Service), Rocky Mt. Arsenal, Denver, Colorado.

a. Shift Supervisor 1.5 years
This was supervisory position under the Shift Superintendent. The plant manufactured nerve gas. Consequently, there were extreme safety hazards due to the highly toxic raw materials and final product. In addition, there were many technical problems, since the process was new and the product had never been manufactured in the United States. There were thirty operators plus twenty craft-type employees per crew.

b. <u>Foreman</u> 1 year

I had a crew of six operators who worked with me to keep the various processing steps operating.

c. <u>Panel Board Operator</u> 1 year

I held this position during initial plant operations. Many process condition changes were at the discretion of the Panel Board Operator.

1956 - 1968 - Chief Operator Trainee 0.75 year - Westinghouse Elect. Corp.

Chief Operator (Watch Officer) 0.5 year

The chief operator on each shift is in immediate control of the reactor plant operation. All operating orders emanate from him. He must have a sound knowledge of the plant principles of operations, as well as an intimate knowledge of the mechanical, electrical and instrumentation systems and components comprising the reactor plant.

Staff Assistant to the Manager of Operations 1 year

This was non-supervisory position which encompassed activities such as writing the operating procedures required as a result of plant modifications, inoperative equipment and policy changes. One particularly interesting activity was being the cognizant operations engineer on the first decontamination of a Naval Reactor Plant. Three months after the decision to decontaminate was made, the test procedure and engineering changes were completed.

Plant Operations Crew Supervisor (POCS) 3 years

As a POCS, I was responsible while on shift, for the safety and operation of the plant. As the senior POCS on my shift at the Naval Reactor Facility, all inter-plant (three plants) problems occurring after regular hours were my responsibility. Specific duties included direct supervision of thirty to fifty Naval and civilian personnel, as well as indirect supervision of approximately twenty craftsmen, industrial hygiene, security and other auxiliary service personnel.

Training Supervisor 1 year

One of the functions of the S1W plant is to train Naval Officers and Enlisted Men to fill billets in the Nuclear Submarine Fleet. As training supervisor I was responsible for the formulation of a training guide that delineated the theoretical and practical knowledge required for qualifications as a watch officer (Chief Operator) and for enlisted men's in-rate qualification. While in this capacity, I certified the qualification of all military and civilian trainees. In addition, the instructors were trained under a program originated by myself. There were twenty-seven enlisted instructors, a training officer and three civilians to carry out the responsibilities of the training section.

Supervisor of Mechanical Engineering 1.6 years

Supervised eleven engineers in the accomplishment of the plant engineering, associated with the maintenance, testing, reactor refueling and modifications required to operate the S1W plant. In addition, I was custodian of all source and special materials for the plant, as well as being responsible for the nuclear safety of water pit and fuel storage areas. Various laboratory committees to which I belonged were: Fluid Systems, Control Rod Mechanisms, Suggestion and Incident Review. Communication with the customer (Naval Reactors) was accomplished verbally and through the submittal of technical proposals, procedures and letters.

The technical scope of the work the section accomplished, ranged from designing a portable boring machine for modifying the reactor vessel, to designating torque values for operating valves, from the design of a system to decontaminate the plant, to modifying the building steam heating system to prevent heater freezing and from writing a refueling procedure to writing a simple drain and flush procedure.

Westinghouse Test Representative 1.7 years
Bettis Atomic Power Laboratory Fuel development tests at the Engineering Test Reactor and the Materials Testing Reactor. As a test representative, I was responsible for issuing loop operating instructions to the reactor operator, the Phillips Petroleum Company. During loop start-up, operations, and shutdown, I monitored and evaluated the data obtained. Upon completion of a reactor cycle, the specimens were removed tested and measured in accordance with my instructions.

Test Car and Test Pad Engineer 2.0 years
Westinghouse Astronuclear Laboratory at the Nevada Test Site. For the NRX-A-6 Nerva Test my position was that of Test Car Engineer and Test Pad Engineer. As such, it was my responsibility to ensure that drawings and parts were correct in order to assemble the test article. As in most field endeavors, many field changes, substitutions and design modifications were required. In addition, all assembly procedures and systems functional test procedures were written under my cognizance.

Feb. '68 - Oct. '69 - General Electric - Nuclear Energy Division
Project Engineer 1.7 years
Project Engineer on the two Quad-Cities and the two Dresden turnkey reactor plants. The mechanical portions of these plants my responsibility. The discharge of these responsibilities entails establishing specifications and giving engineering direction to the Site personnel and the Architect Engineer. Customer and Vendor communications also constitute part of the work load; these interchanges concern different problems related to instrumentation, electrical systems, chemistry, operations, health physics and licensing.

Position Title: Refueling Manager (Chart I "A3")

Primary Function:

Direct preparations for and conduct of core loading for Indian Point Units No. 2 and 3.

Duties and Responsibilities:

1. Plan and schedule work load; assign work; recommend budgets, control expenditures, select and train subordinates, review performance of subordinates, recommend wage or salary adjustments, report results and unusual situations to immediate supervisors.

2. Determine scope of core loading task

   a. Plan and schedule entire task to meet critical date established by overall schedule.

   b. Review all evaluations coordinate procurement of all handling equipment, special and consumable materials.

   c. Direct the writing and review of all procedures for core loading, including approval by Con Edison

   d. Supervise the training of refueling personnel via lectures, study programs and use of tools, cranes, etc.

   e. Coordinate the efforts of WEDCO refueling personnel, craftsmen and Con Edison personnel in conduct of core loading.

   f. Determine the need for technical assistance to support the program, obtain their services and supervise correction of problems.

g. Maintain a daily log of preparatory and actual core loading activities, write a final report including lessons learned for the benefit of other core loading programs.

h. Keep Manager of Operations informed of progress, identify critical needs, make recommendations to safely and satisfactorily complete core loading on time.

3. Maintain liason with other facilities carrying out core loading activities; review their progress, lessons learned. Revise Indian Point Units No. 2 and 3 programs as required.

4. Problem Solving

a. Overcome problems in refueling schedule or those created by plant schedules that delay completion.

b. Prevent delays in receipt of handling equipment or materials from affecting schedule by expediting, field modifications, alternate materials.

c. Revise procedures as required by above problems or any others to maintain progress.

d. Resolve conflicts in craft jurisdiction or identify to crafts supervision.

e. Provide coordination of WEDCO with Con Edison activities in meeting schedules.

5. Decision Making

A. Make Final Decision

1. Determine scope and schedule based upon detailed knowledge of evolutions and equipment, manpower, skills of personnel, amount of required training.

2. Determine number and type of procedures required, detail of writing, safety and safeguards requirements, time required to write, review and approve.

3. Determine best solution for required changes, i.e. field modification, return to vendor, subcontract.

B. Review with Superior

1. Determine critical nature of problem, effect on schedule, financial loss, customer response.

Position Requirements:

1. Education - College (B.S. degree)

2. Specialized or technical knowledge and skills

Mechanical design, tool design, fuel design, core design. Knowledge of craftsmen skills in rigging.

3. Types of work experience. Minimum number of years for each

5 years experience in refueling preparations and conduct including 1-2 years in actual supervision of all refueling activities.

Personnel Resumes (Chart I "A3")

Education:

High School Graduate

B.S. Mech. Eng. - Indiana Institute of Technology - 1958.

Completed Nuclear Power Station Training Program at Shippingport Atomic Power Station.

Experience:

1958 - 1969 Duquesne Light Co. - Shippingport Atomic Power Station.

9/58 - 1/59 Test Engineer - Responsible for writing, review, conduct and final test evaluation of both primary and secondary systems assigned.

2/59 - 6/59 Completed Nuclear Power Training Program.

7/59 - 5/60 Assigned as quality control inspector for first refueling evolution.

6/60 -12/60 Assigned to Refueling Group with responsibility for writing and reviewing refueling procedure for second refueling.

1/61 -10/61 Served as Materials Supervisor with responsiblity for identifying and procuring consumable and special materials, tools, spares to support second refueling evolution.

11/61 -9/62 Served as maintenance Foreman. Responsible for supervising performance of welders, machinist and turbine repairman for plant maintenance.

10/62 -1/63 Assigned as Shift Foreman, reporting to Shift Supervisor in charge of refueling activities on shift for third refueling. Promoted to Shift Supervisor during this period.

2/63 - 5/63 Returned to Maintenance Foreman responsibilities as stated above.

6/63 - 2/64 Promoted to Mechanical Maintenance Engineer with responsibility to establish and issue preventive maintenance program for power plants.

3/64 - 5/65 Served as Technical Supervisor during this period of core change. Responsible for writing, review and issue of refueling and core loading procedures and revision of procedures controlling these evolutions. Concurrent with this effort, had responsibility for major turbine overhaul and writing of overhaul report.

5/65 -10/65 Returned to Mechanical Maintenance Engineer responsibilities as stated above.

10/65-11/65 Assigned as Canal Operations Supervisor with responsbility for all activities in canal during control rod modifications and partial refueling.

12/65-10/69 Promoted to Station Maintenance Supervisor with complete responsibility for the proper maintenance of all machinery and equipment, buildings and grounds associated with the Shippingport Atomic Power Station. These duties include that of Refueling Supervisor, fulfilled during the most recent Shippingport refueling, who has total responsibility for planning, preparation and execution of all refueling activities.

10/69-present WEDCO Operations - Assigned as Refueling Mgr. with responsibility for technical direction of core loading preparations, training and conduct. This effort is carried out in conjunction with Con Edison and Westinghouse Technical Support.

Position Title: Lead Start Up Engineer (Chart I "A1.1")

Primary Function:

Provide technical expertise and experience to a group of engineers writing and conducting the tests required to provide a licensible plant.

Duties and Responsibilities:

1. Helps establish methods and means of accomplishing test objective.

2. Establish and update the test logic required to issue PERT charts, Test schedules and running status.

3. Contacts and interfaces with vendors, architech - engineers, and Westinghouse.

4. Works with the Construction Group to ensure systems are ready for testing. Also that needed repairs or modifications, resulting from tests, are made.

5. Provide interface with the customer - in rendering technical assistance during the conduct of tests by customer operating personnel.

6. Responsible for detailed review of procedures to ensure personnel safety and equipment safety in the writing and carrying out of tests.

Position Requirements

1. Education - College degree, preferably in mechanical area or electrical engineering. Alternate to degree shall be high school graduate with formal Nuclear Operator training including operator qualification and license.

2. Experience - Minimum experience shall include five years in nuclear plant testing, operations, training, construction or direct support of these activities with some of this time preferably in a supervisory capacity.

Personnel Resume (Chart I "A1.1")

Education:

High School Graduate

B.S. Marine Steam and Electrical Engineering

California Maritime Academy, Vallejo, California - 1950

Experience:

1950 - 1953 - Standard Oil Co. of California, Marine Dept. Third Assistant shipboard engineer in charge of engine room watch.

1953 - 1955 - U.S. Navy, Lieutenant Jr. Grade. Engineering Officer and/or Repair Officer on Landing Craft repair ship. Responsible for 30 enlisted men in engineering dept. operation and administration. As Repair Officer responsible for 40 enlisted men in the mechanical repair dept. which provided repair and maintenance service to other ships.

1955 - 1956 - Standard Oil Co. of California, Marine Dept. Second assistant licensed engineer in charge of engine room watch.

1956 - 1957 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory, Operation and Test Engineer. Qualified as Engineering Officer of the Watch (chief operator) prototype nuclear submarine installation. Responsible to plant operation crew supervisor for safe and efficient use of prototype plant during testing, training and maintenance periods.

1957 - 1959 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory. Senior Operations and Test Engineer.

During this period, was assigned as Engineering Officer of the Watch, previously described and part of original group responsible for complete and write power plant manual for operation, maintenance and training of S1W prototype nuclear plant.

1959 - 1960 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory. Staff Assistant to the Manager of operations of S1W prototype nuclear plant. Non-supervisory position responsible to carry out policy changes and over-see training and qualification of naval officers and enlisted personnel on S1W prototype nuclear plant. Administered final oral board qualification examination to engineering of the watch trainees, both naval and civilian.

1960 - 1963 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory Plant Operations Crew Supervisor (POCS)

Responsible while on shift for the safety and operation of the S1W prototype nuclear plant, during training, testing, shutdown, depletion, refueling and maintenance periods. Direct supervision of thirty to fifty Naval and civilian personnel - indirect supervision of on-shift support personnel consisting of chemistry, industrial hygiene and craftsmen - recommend supervisory trainees for final written and oral board qualification examination.

1963 - 1969 - Westinghouse Elect. Corp. - Bettis Atomic Power Laboratory Supervisor, Shift Operations, Expanded core Facility. Responsible for two engineers, twenty-five examination technicians and nine support craftsmen conducting the examination of advance irradiated nuclear fuels and materials from prototype and test reactors, examination and preparation for disposal of spent fuel from all naval reactors, training and testing of personnel in nuclear fuel handling, radiation protection practices and quality control.

1969 – present – Westchester Engineering and Design Co.
Assistant Manager for Operations.

Westinghouse Elect. Corp. Nuclear Energy System – Turnkey Projects – Assistant Manager for Operations.

Responsible for seven startup engineers, in writing, issuing test procedures for conducting of flushing, hydro, pre-operational and acceptance tests of nuclear steam supply systems ensuring the above are conducted in a safe and expeditious manner.

Position Title: Startup Engineer (Chart I "A1.1.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of IPP/INT. Responsibilities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties and Responsibilities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives, personnel or plant safety to supervision.

Position Requirements:

Education and Experience Requirements - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Personnel Resume (Chart I "A1.1.1")

Education:

High School Graduate

B.S. Mechanical Eng. (Nuclear Option) - University of Wyoming - 1968.

Experience:

1968 - present - Westinghouse Elect. Corp. - Nuclear Energy Systems - Turnkey Projects - Startup Engineer assigned to Indian Point Projects. Responsible for writing and issue of flushing and hydrostatic test procedures for N.S.S. Systems, Supervise field personnel during initial flushing and hydrostatic testing of N.S.S. Systems.

Position Title: Lead Start Up Engineer (Chart I "A2.1")

Primary Function:

Provide technical expertise and experience to a group of engineers writing and conducting the tests required to provide a licensible plant.

Duties and Responsibilities:

1. Helps establish methods and means of accomplishing test objective.

2. Establish and update the test logic required to issue PERT charts, Test schedules and running status.

3. Contacts and interfaces with vendors, architect - engineers, and Westinghouse.

4. Works with the Construction Group to ensure systems are ready for testing. Also that needed repairs or modifications, resulting from tests, are made.

5. Provide interface with the customer in rendering technical assistance during the conduct of tests by customer operating personnel.

6. Responsible for detailed review of procedures to ensure personnel safety and equipment safety in the writing and carrying out of tests.

Position Requirements:

1. Education - College Degree, preferably in mechanical area or electrical engineering. Alternate to degree shall be high school graduate with formal Nuclear Operator training including operator qualification and license.

2. Experience - Minimum experience shall include five years in nuclear plant testing, operations, training, construction or direct support of these actitivities with some of this time preferably in a supervisory capacity.

Personnel Resume (Chart I "A2.1")

Education:

High School Graduate

Maryville College, Maryville, Tenn. - 1942

Aviation Cadet Training Program, which included 6 months college training at Lafayette College, Easton, Pa. - 1943

University of Chattanooga, Chattanooga, Tenn. - 1948

Tennessee Valley Authority Steam Plant Training. This program consisted of the following: safe operation and maintenance of power house auxiliary equipment, gas, air, water and coal handling systems; boiler construction, boiler operation, safety interlocks; first aid, fire fighting and automatic fire fighting systems; electrical distribution, relaying and relay protection; generators, exciters, motors, 250 V DC battery and charger systems; turbine construction and operation, maintenance and governing systems. - 1953

Experimental Gas Cooled Reactor, Oak Ridge, Tenn. Basic Nuclear Technology, which consisted of math, slide rule, basic nuclear physics, nuclear power engineering and health physics. Classroom work was conducted on reactor control, instrumentation, reactor operation, power supply, main coolant systems and reactor support systems. - 1966

Military - Army Air Force - Aviation Cadet Corps - Pilot

Experience:

May 51 - Aug. -68 - Tennessee Valley Authority. Responsible for safe and effecient operation of high pressure, high temperature pulverized fuel and gas fired boilers and turbogenerators, emergency power diesel generators, large treated water plants and large demineralizer plants.
Responsible for operator training for the above.

My work with T.V.A. in nuclear plants, includes six months operational training on the General Electric Test Reactors, Vallecitos Atomic Laboratory, Pleasanton, California. This work covered complete reactor operation; startups, shutdowns, new fuel loading, experiemental loop installation and operation; radiation monitoring isotope preparations and shipment, spent fuel shipment, preparation and storing of hot and warm radio-active waste. Classroom work was conducted at this reactor. At the end of the six months training, I successfully completed the examination for creditation for reactor operator.

Prepared and conducted lectures and classroom work on reactor systems for operations section at Experimental Gas Cooled Reactor. Prepared and conducted examination on reactor systems for operations section. These examinations were written, oral and "walk-through". Participated in reactor component shop tests at manufacturers plants.

Prepared operating procedures. Prepared and submitted design changes on reactor support systems. Revised prints to show "as built" revisions. Performed acceptance tests on reactor systems.

I am certified as reactor operator with Tennessee Valley Authority.

1967 - Prepared and presented six week lecture series on Basic Nuclear Engineering at Allen Steam Plant, Memphis, Tenn.

1968 - Startup Engineer - United Engineers and Constructors, Indian Point Generating Station. Responsible for coordinating craft supervisors to bring systems to completion, the check out of the systems and their components, the pre-operational flushing and testing, the correct operation and "running-in" of each piece of equipment.

My work, also, includes the writing of system descriptions, Operating procedures, periodic test and test procedures.

1969 - present - Westinghouse Elect. Corp. - Nuclear Energy System Turnkey Projects - Lead Startup Engineer, assigned to Indian Point Projects. Responsible for five startup engineers. Supervise engineers in writing and issue of test procedures for nuclear plant secondary systems, conduct of initial flushing and hydro of secondary systems, pre-operational and acceptance testing.

Position Title: Startup Engineer (Chart I "A2.1.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of Indian Point Units No. 2 and 3. Responsibilities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties and Responsibilities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives, personnel or plant safety to supervision.

Position Requirements:

Education and Experience Requirements - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Personnel Resume (Chart I "A2.1.1")

Education:

High School Graduate - Stavanger, Norway

Apprenticeship in Mech. Eng. in ship yard - 4 years

Stavanger Engineer School, Stavanger Norway

Marine Engineer - Chief-1st-2nd classes - Steam and Diesel

Bergen Institute of Engineering - Bergen, Norway
(Similar to Stevens Institute of Technology)

Graduate Mechanical Engineer - 1947.

Experience:

Twenty Years' experience as mechanical engineer in supervisory capacity in the Marine Industry, Power Plants, and other industrial situations.

Marine Engineer with various shipping companies in Norway and U.S.A. for a period of eight years.

1949 - 1952 - Bethlehem Steel Ore Navigation Corp. Sparrows Point, Md. - Chief Engineer of Ore Carriers, in charge of the Engine Dept. and responsible for the supervisor of 20 employees.

1953 - 1968 - Skinner Engine Co. Erie, Pa.
Manufacturers of Unaflow steam engines, turbines, centrifugal and reciprocating pumps, etc. for the Marine Industry, Power Plants, and various other industries.

As Field Supervisor for a period of 15 years, I was primarily responsible for the installation of new machinery, startup and trails, training of operators, as well as maintenance and repairs of old machinery.

Activities have involved:

Services at numerous ship yards in the U.S.A., Canada and the Caribbean Islands for the purpose of supervising in the installation of ships' main propulsion machinery, thrust bearings, generators, pumps, selfunloading conveyor machinery, etc. as well as converison and repair work of Merchant Marine - and U.S. Navy vessels.

Travelling throughout various parts of the U.S.A. for the purpose of supervising in the installation of new generators at industrial power plants, as well as trouble-shooting.

Acting as consultant to customers in the solving of various problems concerning our products. Recommend solutions to same problems. Break-in new crew on new machinery.

1968 - present - Westinghouse Elect. Corp. - NES - Turnkey Projects.

Assigned to IPP as Secondary Senior Startup Engineer.

Responsible for installation and aligning of Secondary components and systems.

Prepare test procedures for initial flushing, hydrostatic test, pre-operational and acceptance testing of Secondary Systems.

Review test procedures, and conduct testing of same systems. Responsible for operations and turnover of Secondary components and systems.

Position Title: Lead Start Up Engineer (Chart I "A2.2")

Primary Function:

Provide technical expertise and experience to a group of engineers writing and conducting the tests required to provide a licensible plant.

Duties and Responsibilities:

1. Helps establish methods and means of accomplishing test objective.

2. Establish and update the test logic required to issue PERT charts, Test schedules and running status.

3. Contacts and interfaces with vendors, architech - engineers, and Westinghouse.

4. Works with th construction Group to ensure systems are ready for testing. Also that needed repairs or modifications, resulting from tests, are made.

5. Provide interface with the customer in rendering technical assistance during the conduct of tests by customer operating personnel.

6. Responsible for detailed review of procedures to ensure personnel safety and equipment safety in the writing and carrying out of tests.

Position Requirements:

1. Education - College Degree, preferably in mechanical area or electrical engineering. Alternate to degree shall be high school graduate with formal Nuclear Operator training including operator qualification and license.

2. Experience - Minimum experience shall include five years in nuclear plant testing, operations, training, construction or direct support of these activities with some of this time preferably in a supervisory capacity.

Personnel Resume (Chart I "A2.2")

Education:

High School Graduate

U.S. Air Force Electronics and Radar Maintenance School

B.S.E.E. - Kansas State University

Western Elect. Co. - Digital and Gen. Purpose Computer School - 3 months

Westinghouse Elect. Corp. - Nuclear Power School - 6 months

Nuclear Plant Operator Training - 5 months

Experience:

1953 - 1957 - S/Sgt. USAF Radar Technician

1957 - 1960 - Student

1961 - 1962 - Western Elect. Co. - R & D Engineer anti missile project - responsible for installation, testing, design improvements and operational procedures for I.F. receiver of precision target tracking radar.

1962 - 1964 - Martin Marietta Corp. - Maintainability engineer - responsible for evaluation design specs., fabrication processes, prototype models, test and operating procedures for electronic aerospace ground and support equipment of the Titan Missile Systems to reduce maintenance actions and costs on final design equipment.

1964 - 1965 - Westinghouse Elect. Corp. - Bettis Atomic Power Div. - Associate design engineer. Responsible for upgrading radioactive waste disposal power and control system, assisted in updating and testing primary plant control system during refueling modification.

1965 - 1968 - Westinghouse Electric Corp. - Bettis Atomic Power Div. Design Engineer - Responsible for plant instrumentation and control systems, normal and emergency power systems and neutron detector test facility, supervised installation, checkout, modifications and plant testing, prepared test and operating procedures, reviewed design and safety analysis studies on above systems.

1968 - 1969 - Westinghouse Elect. Corp. - Bettis Atomic Power Div. - Senior Engineer - sponsible for maintenance and component testing of dual reactor plant. Co-responsible for six senior naval maintenance personnel. Reviewed and approved modifications, component maintenance and testing and evaluated system performance.

1969 - Present - Westinghouse Elect. Corp. - WEDCO Corp. - Lead Elect. Startup Engineer - Responsible for development and performance of all aspects of electrical Startup testing, including test identification, test sequencing, procedure writing test performance, system performance and data evaluation and data submitted to the customer.

Position Title: Startup Engineer (Chart I "A.2.2.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of Indian Point Units No. 2 and 3. Responsibilities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties and Responsiblities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives, personnel or plant safety to supervision.

Position Requirements:

Education and Experience Requirements - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Personnel Resume (Chart I "A2.2.1")

Education:

High School Graduate

Pre-engineering, Bowling Green State University

Nuclear Power Training USN

Electrician School, USN

Magnetic Amplifier and Sound Vibration School, USN

Experience:

1953 - 1958 - McMillen Feed Mills Plant operator. Operated chemical extraction equipment in major soybean oil processing plant.

1958 - 1964 - U.S. Navy. Enlisted in U.S. Navy and completed Electrician's School and Basic Submarine School.

Qualified in Submarine aboard the USS Pickerel (SS524) Duties included operation, maintenance and repair of electrical equipment associated with conventional submarine DC systems.

Completed Nuclear Power School at U.S. Naval Submarine Base. Continued Nuclear Power training at S3G reactor plant of West Milton, N. Y., and qualified as Reactor operator, Electric plant operator and Steam plant operator.

Assigned to [illegible]S Triton (SSN586). Qualified on Triton's twin pressurized water reactor plant. Duties included operation, maintenance and repair of main, auxiliary and DC electrical systems.

Followed yard personnel during 18 month yard overhaul period, including shop repair, re-installation and startup operations of individual systems.

1964 - 1967 - Westinghouse Astronuclear Laboratory, Plum Brook Test Operations. Senior Technician. Operated a water-cooled irradiation system as lead shift technician. Assisted in preparation of operating procedures, performed electrical hookups and checkout of experimental readout equipment, operated equipment and performed data acquisition.

Assisted in construction, checkout and initial startup of a cryogenic environmental irradiation system installed at Plum Brook, operated system lead shift technician and performed maintenance and repair of electrical and associated equipment.

1967 - 1969 - Westinghouse Atomic Power Division - Test Engineer Assisted with initial pr[illegible]e writing for initial equipment and system startup oper[illegible]for various reactor projects. Travelled to various re[illegible]sites to assist with startup operations.

1969 - present - WEDCO Corp. - Test Engineer.

Position Title: Lead Start Up Engineer (Chart I "A2.3")

Primary Function:

Provide technical expertise and experience to a group of engineers writing and conducting the tests required to provide a licensible plant.

Duties and Responsibilities:

1. Helps establish methods and means of accomplishing test objective.

2. Establish and update the test logic required to issue PERT charts, Test schedules and running status.

3. Contacts and interfaces with vendors, architech - engineers, and Westinghouse.

4. Works with the Construction Group to ensure systems are ready for testing. Also that needed repairs or modifications, resulting from tests, are made.

5. Provide interface with the customer in rendering technical assistance during the conduct of tests by customer operating personnel.

6. Responsible for detailed review of procedures to ensure personnel safety and equipment safety in the writing and carrying out of tests.

Position Requirements:

Education - College Degree, preferably in mechanical area or electrical engineering. Alternate to degree shall be high school graduate with formal Nuclear Operator training including operator qualification and license.

Experience - Minimum experience shall include five years in nuclear plant testing, operations, training, construction or direct support of these activities with some of this time preferably in a supervisory capacity.

Personnel Resumee (Chart I "A2.3")

Education:

High School Graduate - 1948

USN Radar School, Boston, Mass. - 1949

USN Submarine School, New London, Conn. - 1952

Armed Forces College Level General Education Development Examination - 1957

USN Nuclear Power School, New London, Conn. - 1959

S1W Prototype Training, Idaho Falls, Idaho - 1960

S5W Pressurized Water Reactor Training, Westinghouse Research Laboratories - Bettis - 1962

Experience:

1960 - 1962 - USS Patrick Henry (SSBN599) Chief Petty Officer in charge of Reactor Controls Division and Machinery Watch Supervisor through construction, test program and two Fleet Ballistic Missile Patrols.

Basic Functions: The Reactor Controls Division operates the reactor and is responsible for the maintenance, test, and repair of all reactor plant instrumentation including Steam Generator Water Level Control and Rod Control equipment.

The Machinery Watch Supervisor is responsible for the overall supervision of the propulsion plant including such operations as paralleling electric generating equipment, chemical analyses of primary plant and steam generator water, radiological control, and conducting tests on reactor control instrumentation.

A basic requirement for the Machinery Watch Supervisor is to be qualified on every watch station and pass a written and oral examination administered by the Engineer and Commanding Officer. This is in addition to a Naval Reactors Safeguards Examination prior to initial criticality and periodically thereafter.

1962 - 1964 - USS Lafayette (SSBN616) Chief Petty Officer in charge of Reactor Controls Division and Machinery Watch Supervisor through construction, test program and two Fleet Ballistic Missile Patrols.

1964 - 1967 - USS Benjamin Franklin (SSBN640) Chief Petty Officer in charge of Reactor Controls Division and Machinery Watch Supervisor through construction, test program and three Fleet Ballistic Missile Patrols. Collateral duty of Engineering Department Training Co-ordinator.

1968 - present - a. First Assignment - Westinghouse Electric Corp. - Nuclear Energy Systems - Senior Instrumentation and Controls Construction Engineer at Indian Point Projects. Responsible for two I & C construction engineers. Responsible for receipt inspection and testing of components, installation of Primary Process I & C, Auxiliary Process I & C and Reactor Control and Protection System. Prepare initial calibration procedures and data documentation forms and conduct initial calibration on instrumentation systems.

b. Second Assignment - WEDCO Corp. - Senior Instrumentation and Controls Startup Engineer. Responsible for three I & C Startup Engineers. Review data obtained by construction I & C engineers prior to Startup Acceptance of instrumentation systems. Prepare test procedures for final acceptance testing under actual dynamic conditions. Review fluid systems operational test procedures for adequate coverage of instrumentation checks. Conduct final acceptance testing and prepare documentation on Reactor Control, Protection and Safeguards Systems instrumentation.

Position Title: Startup Engineer (Chart I "A2.3.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of Indian Point Units 2 and 3. Responsiblities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties and Responsibilities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives, personnel or plant safety to supervision.

Position Requirements:

Education - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Personnel Resume (Chart I "A2.3.1")

Education:

High School Graduate

B.S. Chemical Engineering - Polytechnic Institute of Brooklyn - 1949

Experience:

1949 - 1950 - MADIGAN-HYLAND, Long Island City, New York

Concrete batch plant operator on construction of Brooklyn Expressway.

1950 - BRINKERHOFF, HALL, MAC DONALD ET AL, Mt. Holly, N. J.

Member of surveying crew - Rodman and Chainman on construction of New Jersey Turnpike.

1951 - AMERICAN ALKYD INDUSTRIES, Rutherford, New Jersey

Quality Control Chemist in Alkyd Resin Plant.

1951 - 1954 - BLAW KNOX COMPANY, Pittsburgh, Pennsylvania

Junior Process Engineer in Chemical Plants Division. Duties: Writing proposals, writing and checking equipment specifications, layout of process flow diagrams, writing operating instructions and startup of chemical plants.

1954 - 1964 - DE LAVAL SEPARATOR COMPANY, Poughkeepsie, New York

Field, Research and Process Engineer. Responsible for the design, installation and operation of De Laval Processing

Plants for the chemical process industries; e.g., vegetable oil, animal fat, tall oil and soap. Duties included preparation of process flow diagrams, writing of equipment specifications, ordering material, and supervision of on-site equipment installation, then plant startup to complete customer satisfaction.

1965 - 1969 - STANDARD BRANDS INCORPORATED, Peekskill, New York

Process development and design engineer with supervisory capacity for pilot scale industrial waste treatment plants as well as research and analytical laboratory facilities.

Duties included design and selection of equipment, supervision of operation and maintenance of pilot plant and laboratory, data evaluation, scale-up design, cost estimation, and finally, process selection. Physical, chemical and biological treatment studies covered unit processes and operations such as filtration, reverse osmosis, centrifugation, mixing fluid flow and heat and mass transfer.

1969 - Present - WEDCO Operations - Startup Engineer - Assigned to Instrument and Control Group with responsiblity for review of proper process instrumentation checkout and testing during installation and testing. Trouble shoot process instrumentation, problems, make design and engineering modification recommendations and follow problem corrections.

Position Title: Manager, WEDCO Operations (Chart II "B")

Job description and resume are the same as Chart I, Position "A".

Position Title: Test Program Manager (Chart II "B1")

Job description and resume are the same as Chart I, Position "A1".

Position Title: Startup Manager (Chart II "B2")

Job description and resume are the same as Chart I, Position "A2".

Position Title: Refueling Manager (Chart II "B3")

Job description and resume are the same as Chart I, Position "A3".

Position Title: Lead Engineer - NSSS (Chart II "B1.1")

Primary Function:

Provide technical expertise and experience to a group of engineers writing and conducting the tests required to provide a licensible plant.

Duties and Responsibilities:

1. Helps establish methods and means of accomplishing test objective.

2. Establish and update the test logic required to issue PERT charts, Test schedules and running status.

3. Contacts and interfaces with vendors, architech - engineers, and Westinghouse.

4 Works with the Construction Group to ensure systems are ready for testing. Also that needed repairs or modifications, resulting from tests, are made.

5. Provide interface with the customer in rendering technical assistance during the conduct of tests by customer operating personnel.

6. Responsible for detailed review of procedures to ensure personnel safety and equipment safety in the writing and carrying out of tests.

Position Requirements:

1. Education - College Degree, preferably in mechanical area or electrical engineering. Alternate to degree shall be high school graduate with formal Nuclear Operator training including operator qualification and license.

2. Experience - Minimum experience shall include five years in nuclear plant testing, operations, training, construction or direct support of these activities with some of this time preferably in a supervisory capacity.

Personnel Resume (Chart II "B1.1")

Resume same as Chart I, Position "A1.1.1".

Position Title: Startup Engineer (Chart II "B1.1.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of Indian Point Units No. 2 and 3. Responsibilities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties & Responsibilities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives, personnel or plant safety to supervision.

Position Requirements:

Education & Experience - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Personnel Resume (Chart II "B1.1.1")

Education:

High School Graduate
Navy Electronics School - Fundamentals & Shipboard Electronics
Deforest - Radio & TV Electronics.
Westinghouse Reactor Operator Training program covering reactor physics, reactor theory, reactor instrumentation and control, standard and emergency procedures, and radiation protection.
EAI - Analog Computer Operators Training.

Experience:

1957 - 1962 - Westinghouse Testing Reactor - Reactor Technician Maintained WTR Critical Facility reactor instrumentation. Successfully completed AEC examinations and received AEC Reactor Operator License. Licensed to operate 3 critical facility reactors. Responsible for setup, installation, irradiation, and removal of experiments.

1962 - 1968 - Westinghouse Astronuclear Lab. - Test Engineer Assisted in direction of EG&G technicians for maintenance and system checkout at Test Cell A, NRDS. Was assistant program operator during several NERVA runs. Responsible for maintenance and checkout of all control valves at Test Cell C, NRDS.

1968 - - Westinghouse Atomic Power Division, Nuclear Power Service Startup Services - Operations Engineer.

Assisted in the chemical cleaning of the Main Steam Feed piping at the Rochester Gas & Electric Nuclear Power Plant. Worked with Mr. H. Kordesh of Bechtels startup organization.

1969 - Supervisory Service Engineer Assigned to the Indian Point Unit 2 startup activities and support group at Pittsburgh, Pa. writing System Flushing and Hydro Procedures. Supplying information when requested by site personnel.

1969 - Present - Westinghouse Nuclear Energy Systems, Construction and Services. Supervisory Service Engineer.

Transferred to the Indian Point site to assist in carrying out the flushing and hydrostatic testing. Involved in writing startup procedures for various nuclear steam supply systems. Assisting in all phases of construction checkout and turnover to startup. Assisting in writing "Hot Functional" test series.

Position Title: Lead Engineer - non NSSS (Chart II "B1.2")

Job description and resume are the same as Chart I, position "A2.1".

Position Title: Startup Engineer (Chart II "B1.2.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of Indian Point Units No. 2 and 3. Responsibilities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties & Responsibilities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives, personnel or plant safety to supervision.

Position Requirements:

Education & Experience - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Personnel Resume (Chart II "B1.2.1")

Education:

High School Graduate - 1947

Navy Education:

1. Class "C" Engine Man School Great Lake, Ill. 1951
2. Air Conditioning and Refrigeration School, Va. 1952
3. U.S. Navy Submarine School, New London, Conn. 1954
4. U.S. Navy Nuclear Power School, Idaho. 1960
5. U.S. Navy Nuclear Welding School, Idaho. 1961
6. Newport News Shipbuilding & Dry Dock Company Nuclear Orientation School, Newport News, Va. 1967

Experience:

1947 - 1967 - U.S. Navy - serving on various ships and shore stations as an instructor and also maintaining diesel engines and auxiliary equipment.

1960 - 1967 - U.S. Navy - Pre-commissioning crew on two Nuclear Powered Submarines - one in Mare Island Navy Yard, California and one at Newport News, Va. I was the Senior enlisted man in charge of the mechanical equipment in the primary and secondary systems, worked closely with shipyard personnel on the installation, construction, flushing, hydrostatic testing, initial operation of primary and secondary systems, wrote operating instructions for many engine room components and systems. Became experienced in quality control and general construction procedures. I was qualified as a Machinery Watch Supervisor, requiring the ability to stand all watches in the plant. Retired from U.S. Navy in 1967 as a Chief Petty Officer.

1967 - 1969 - Newport News Shipbuilding and Dry Dock Company - Mechanic Test Engineer - Responsible for Testing Systems - Writing Repair Procedures and Coordinating tests between the trades and Navy personnel.

1969 - Present - WEDCO Corp. Operations Startup Engineer. Responsible for writing of phase I and phase II test procedu coordinating the safe startup and operation of components a the operation of these components as an itegrated system.

Position Title: Lead Engineer - Electrical (Chart II "B1.3)

Job description and resume are the same as Chart I, position "A2.2".

Position Title: Startup Engineer (Chart II "B1.3.1")

Job description and resume are the same as Chart I, position "A2.2.1").

Position Title: Lead Engineer - Instrumentation and Control (Chart II "B1.4")

Job description and resume are the same as Chart I, position "A2.3".

Position Title: Startup Engineer (Chart II "B1.4.1")

Job description and resume are the same as Chart I, position "A2.3.1".

Position Title: Startup Director (Chart II "B2.1")

Primary Function:

Provide technical supervision of testing evolutions carried out on shift.

Duties & Responsibilities:

1. Ensure testing is carried out in accordance with approved test procedures.

2. Evaluate, approve, or obtain higher level approval, the necessary revisions to test procedures.

3. Ensure systems/components are operated and maintained in accordance with good engineering practices to ensure personnel safety.

4. Coordinate the efforts of various participating groups (Operations, Con Edison, construction crafts or foremen, vendors, Westinghouse NES personnel etc.) o effectively and safely carry out assigned tests.

5. Maintain log of activities to ensure good communications between personnel on crew and between shifts.

6. Notify cognizant authorities of problems: significant delays, personnel or plant safety, satisfaction of test or design objectives, need for assistance.

Position Requirements:

1. Education & Experience - College Degree in engineering or science field or equivalent experience. Participate in IPP test procedures research, writing, review and issue. Participatte in IPP pre-operational flushing and hydro test program as cognizant systems test engineer. Participate in other operations, testing, startup programs such as those conducted at other turnkey sites, shipyards, Naval Reactor facilities, which involve nuclear reactors and steam equipment.

Personnel Resume (Chart II "B2.1")

Education:

High School Graduate

B.S. Marine Engineering - U.S. Merchant Marine Academy - 1963

M.S. Nuclear Engineering - University of Virginia - 1967

Experience:

8-63 - Third Asst. Engineer - MSTS Watch Engineer on turbo-electric troop ship.

9-63 - 12-63 - Third Asst. Engineer - United Fruit Co. - Watch Engineer on single and twin-screw refrigated ships.

1-64 - 8-65 - Nuclear Power Test Engineer - Mare Island Division, San Francisco Bay Naval Shipyard; Vellejo, Calif. Fully qualified as Shift Test Engineer of S5W plants. Developed procedures and equipment to accomplish objectives of nuclear plant test programs. Coordinated construction or repair work with test program and plant requirements, with particular emphasis on core protection and radiological safety. Experience with all phases of new construction and overhaul of nuclear plant (S5W) including refueling and decontamination (hydrostatic tests, initial core load and fill, initial criticality, steam tests, power operations, decay heat removal, and plant conditions for overhaul).

9-65 - 8-67 - University of Virginia - M.S.N.E. - Determined safety limits of test reactors at the University and U.S. Army Watertown Arsenal for use in application to AEC for increase in licensed power level.

9-67 - 12-67 - Third Asst. Engineer - American Export - Isbrandtsen Lines - Supervised engine room on conventional and semi-automated turbine driven ships.

1-68 - 10-69 - Reactor Division - Oak Ridge National Laboratory - Research Associate. Performed and analyzed experiments to determine the effect of hydrodynamic conditions on the performance of cellulose acetate desalination membranes. Responsible for ORNL contract covering municipal sewage treatment with dynamic membranes. Designed stationary and mobile test loops. Co-author of paper reporting experiments on the application of woven hose to cross-flow filtration and hyperfiltration with dynamic membranes.

10-69 - present-WEDCO Operations - Startup Engineer responsible for writing of startup test procedures and conduct of assigned tests. Responsible for full engineering of feedwater and steam piping chemical cleaning.

Position Title: Startup Engineer (Chart II "B2.1.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of Indian Point Units 2 and 3. Responsibilities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties and Responsibilities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives personnel or plant safety to supervision.

Position Requirements:

1. Education & Experience - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Personnel Resume (Chart II "B2.1.1")

Education:

High School Graduate

Navy Education: I Machinist's Mate "A" School (3 months)

Basic Turbine Theory, Theory of construction and operation of turbines, pumps, lubricating oil purifiers, reduction gears, governor systems, feed and condensate systems, distilling plants, air conditioning systems, steam system components and blueprints.

II Submarine School (2 months)

Theory of construction and operation of submarines and submarine system including hydraulics, fluid systems, electrical systems and various other associated systems.

III Nuclear Power School (6 months)

Academic instructions in the following subjects: Algebra, trigonometry, calculus, physics, nuclear reactor principals, nuclear reactor construction and design, metallurgy, thermodynamics, fluid dynamics, radiological control, radio chemistry and specialized mechanical training.

IV Operational Training at Combustion Engineering Reactor Site, SIC prototype, Windsor Locks, Conn. (6 months)

Classroom study and operational training in the theory and operation of pressurized water reactors, reactor primary systems, steam generation and distribution systems; feed and condensate systems, and specialized training in reactor and secondary systems. Upon successful completion of course and qualifications in the SIC reactor plant, was designated 3355 Nuclear Power Plant Mechanical Plant Operator.

Experience:

1964 - 1966 - U.S.S. John Adams SSBN 620 - qualified upper and lower level engine room watch. Responsible for operation and control of main propulsion and auxiliary systems.

1966 - 1968 - Precommissioning unit USS Greenling SSN 614 General Dynamic Quincy ship yard worked closely with shipyard personnel on the installation construction flushing hydrostatic testing initial operation of Primary and secondary systems. Wrote operating instructions for many engine room components and systems. Became experienced in quality control and general construction procedures.

Qualified for the following watch stations in the nuclear engineering plant:

Shutdown Electrical Operator. Responsible for the proper operation of the motor generators, distribution switchgear, and associated electrical auxiliary equipment throughout the Engineering Plant during shutdown conditions.

Auxiliary Electrician (Engineering). Responsible for the proper operation and maintenance of all electrical auxiliary equipment within the engineering plant, including the remote reading temperature and salinity indicating equipment motors, motor controllers and various other electrical equipment.

Steam Plant Control Panel. Responsible for the operation and control of the ship's main propulsion turbines, and systems associated with the main steam generating system.

Engineering Space Supervisor. Responsible for the safety and operation of the reactor, reactor primary systems, main propulsion, and auxiliary systems of the entire engineering plant during reactor operation.

1968 - 1969 - Honeywell Inc. Commercial Division - worked on installation and operation of temperature control systems.

1969 - Present - WEDOCO Corp. - Operations - Startup Engineer (Secondary side) - responsible for writing of phase I and phase II test procedures, coordinating the safe startup and operation of secondary components and the operation of these components as an integrated system.

Position Title: Startup Director (Chart II "B2.2")

Primary Function:

Provide technical supervision of testing evolutions carried out on shift.

Duties & Responsibilities:

1. Ensure testing is carried out in accordance with approved test procedures.

2. Evaluate, approve, or obtain higher level approval, the necessary revisions to test procedures.

3. Ensure syst[illegible] [illegible]onents are operated and maintained in accordance [illegible] practices to ensure personnel safety.

4. [illegible] the efforts of various participating groups (Operations, Con Edison, construction crafts or foremen, vendors, NES personnel etc.) to effectively and safety carry out assigned tests.

5. Maintain log of activities to ensure good communications between personnel on crew and between shifts.

6. Notify cognizant authorities of problems: significant delays, personnel or plant safety, satisfaction of test or design objectives, need for assistance.

Position Requirements:

Education & Experience - College Degree in engineering or science field or equivalent experience. Participate in Indian Point Unit No. 2 test procedure research, writing, review and issue. Participate in Indian Point Unit No. 2 pre-operational flushing and hydro test program as cognizant systems test engineer. Participate in other operations, testing, startup programs such as those conducted at other turnkey sites, shipyards, Naval Reactor facilities, which involve nuclear reactors and steam equipment.

Personnel Resume (Chart II "B2.2")

Resume same as Chart I, position "A1.1.1".

Position Title: Startup Engineer (Chart II "82.2.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of Indian Point Unit No. 2 and 3. Responsibilities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties & Responsibilities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives, personnel or plant safety to supervision.

Position Requirements:

Education & Experience - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Personnel Resume (Chart II "B2.2.1")

Education:

High School Graduate - 1946

2 yrs. Civil Engineering - Catholic University of America - 1948

B.S. Marine Engineering - U.S. Naval Academy - 1952

Nuclear Orientation course - Newport News Shipbuilding & Dry Dock Co. 1967.

Experience:

1952 - 1956 - United States Navy

1956 - 1963 - Assistant Cashier Delaware Trust Co.

1963 - 1965 - Data Processing Sales - IBM

1965 - 1967 - Assistant Vice President - The Bank of Virginia.

1967 - 1969 - Newport News Shipbuilding & Dry Dock Co. - Mechanical test engineer.

Performed technical review of reactor plant component technical manuals.

Prepared and performed technical review of instructions for installation and repair of reactor plant components & procedures for system installation.

Directed testing of S5W plant from initial flushing through power range testing.

Directed repairs & re-testing of S5W plant during post shakedown availability.

Position Title: Startup Director (Chart II "B2.3")

Primary Function: Provide technical supervision of testing evolutions carried out on shift.

Duties & Responsibilities:

1. Ensure testing is carried out in accordance with approved test procedures.

2. Evaluate, approve, or obtain higher level approval, the necessary revisions to test procedures.

3. Ensure systems/components are operated and maintained in accordance with good engineering practices to ensure personnel safety.

4. Coordinate the efforts of various participating groups (Operations, Con Edison, construction crafts or foremen, vendors, Westinghouse NES personnel etc.) to effectively and safely carry out assigned tests.

5. Maintain log of activities to ensure good communications between personnel on crew and between shifts.

6. Notify cognizant authorities of problems: significant delays, personnel or plant safety, satisfaction of test or design objectives, need for assistance.

Position Requirements:

Education & Experience - College Degree in engineering or science field or equivalent experience. Participate in Indian Point Unit No. 2 test procedure research, writing, review and issue. Participate in Indian Point Unit No. 2 pre-operational flushing and hydro test program as cognizant systems test engineer. Participate in other operations, testing, startup programs such as those conducted at other turnkey sites, shipyards, Naval Reactor facilities, which involve nuclear reactors and steam equipment.

Personnel Resume* (Chart II "B2.3")

Education:

High School Graduate

Pre-engineering - Wharton County Junior College, Texas. - 2 years.

Naval Nuclear Power School - Bainbridge, Md. 1964

Naval Prototype Training Center, Windsor Locks, Conn. - 1965

Management Principles for engineers, University of Wisconsin - 1969

Experience:

1965 - 1968	- United States Navy, Nuclear Submarine Service, Advanced to rate E-5, qualified as engineering space supervisor, U.S.S. Greenling, SSN614. Member of pre-commissioning crew, aboard Greenling during entire construction stage, and supervised in naval capacity, the entire startup of the engineering systems. Responsible for safe and efficient operation of all nuclear systems. Assisted in training programs.
1968 - Present	- Westinghouse Electric Corp. - Nuclear Energy Systems Turnkey Projects - supervisory service engineer, mechanical components engineer. Responsible for installation of major nuclear steam supply components and auxiliary pumps and equipment. Responsible for installation and evaluation of assigned equipment. Assisted systems engineers in nuclear piping systems evaluation.

*Position assignment is tentative.

Position Title: Startup Engineer (Chart II "B2.3.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of Indian Point Units No. 2 and 3. Responsibilities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties & Responsibilities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives, personnel or plant safety to supervision.

Position Requirements:

Education & Experience - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Personnel Resume (Chart II "B2.3.1")

Education:

High School Graduate

B.S. in Mathematics - Elizabeth City State University - 1966.

Completed Newport News Shipbuilding and Dry Dock Co. Nuclear Orientation Course.

Experience:

1966 - Present - Mechanical Test Engineer.

Technical Review of Reactor Plant Installation procedures.

Prepared technical instruction for service and repair of reactor plant components.

Technical review of reactor plant construction drawings.

Review of vendor technical manuals.

Technical review of reactor plant test procedures.

Conduct reactor plant systems tests.

Perform design calculations and analysis of test data.

Coordinate test program with cognizant trades

Position Title: Startup Director (Chart II "B2.4")

Primary Function:

Provide technical supervision of testing evolutions carried out on shift.

Duties & Responsibilities:

1. Ensure testing is carried out in accordance with approved test procedures.

2. Evaluate, approve, or obtain higher level approval, the necessary revisions to test procedures.

3. Ensure systems/components are operated and maintained in accordance with good engineering practices to ensure personnel safety.

4. Coordinate the efforts of various participating groups (Operations, Con Edison, construction crafts or foremen, vendors, Westinghouse NES personnel etc.) to effectively and safely carry out assigned tests.

5. Maintain log of activities to ensure good communications between personnel on crew and between shifts.

6. Notify cognizant authorities of problems: significant delays, personnel or plant safety, satisfaction of test or design objectives, need for assistance.

Position Requirement:

Education & Experience - College Degree in engineering or science field or equivalent experience. Participate in Indian Point Units No. 2 and 3 test procedure research, writing, review and issue. Participate in Indian Point Units No. 2 and 3 pre-operational flushing and hydro test program as cognizant systems test engineer. Participate in other operations, testing, startup programs such as those conducted at other turnkey sites, shipyards, Naval Reactor facilities, which involve nuclear reactors and steam equipment.

Personnel Resume (Chart II "B2.4")*

Education:

High School Graduate

Experience:

1959 - 1961 - Westinghouse Testing Reactor Technician

Operated and maintained steam plant and reactor auxiliary systems consisting of reactor primary cooling system, secondary cooling system, demineralizers and water softeners, chemical treatment of boiler and secondary water, reactor and plant ventilation systems and high and low pressure steam systems. Promoted to Reactor Technician and entered operator training program.

1961 - 1962 Westinghouse Testing Reactor, Reactor Technician

Participated in an extensive reactor operator training program covering reactor physics, reactor theory, reactor instrumentation and control, standard and emergency operations and radiation protection.

Successfully completed AEC examinations and received AEC Reactor Operator License OP-1044. Operated and maintained the reactor on a rotating shift basis, manipulated controls during startup, steady-state and shutdown operations. Responsible for the proper execution of reactor core changes and installation of customer experiments. Assisted in revising and changing reactor and process system operating procedures.

*Position assignment is tentative

1962 - 1963 - Westinghouse Astronuclear Laboratory, Technician

Reviewed and revised operating procedures for Test Cell A, NRDS. These procedures included all support systems for the NERVA reactor tests. Reviewed conceptual design of the cryogenic He gas cooled system proposed for PBRF.

Assisted in the design of the water cooled irradiation system for PBRF.

Assisted with the first Test Fixture design to be used for PBRF.

1963 - 1967 - Westinghouse Astronuclear Laboratory, Plum Brook Test Operations, Senior Technician.

Transferred to PBRF for the installation and operation of a water cooled inpile irradiation system consisting of an inpile water cooled capsule and associated process piping to supply primary, purge and emergency cooling water. Assisted in the checkout and testing of the system and assisted in preparing the Operations Manual for the system. Qualified as a system operator by NASA and WANL.

Assisted in the specification and design of a Charging Machine to be used with the water cooled irradiation system for automatic capsule insertion and withdrawal from the reactor. The machine and associated components operated under 25 feet of water.

Assisted in the assembly and checkout of the Charging Machine prior to shipment to PBRF for installation.

Assisted in the design and specification review of a cryogenic inpile test system. Monitored fabrication and erection of the system during construction phase. Assisted in all phases of initial plant pre-startup and startup. This included instrument checkout, hydrostatic testing, 700 hp He Compressor breakin and 16kw Expansion Engine breakin.

Assisted in the modifications and redesign of a new insertion machine to make it compatable with the cryogenic system and improve its reliability. Acted as Lead Mechanical Technician during the installation and testing of the new machine and as Lead Mechanical Technician for the post-testing disassembly and reassembly of the machine in the reactor area. Assisted in the final performance testing of the machine prior to NASA acceptance.

Assisted in the preparation of the Operations Manual for the cryogenic system and insertion machine. Operated on a rotating shift basis on both water cooled and cryogenic systems as Lead Technician.

1967 - 1969 - Westinghouse Atomic Power Division, Nuclear Power Service Startup Services. Operations Engineer.

Transferred to Nuclear Power Service from Westinghouse Astronuclear Laboratory, Plum Brook Reactor Facility, I was assigned as assistant Resident Engineer in the De-Fueling Operation of the French & Belgium "Centrale Nucleaire Des Ardennes" (S.E.M.A.) Pressurized Water Reactor. Set up a completely remote underwater inspection program of all reactor vessel internal parts and fuel assemblies.

Closely followed the disassembly, insepection and reassembly of the main reactor coolant pumps.

I spent 6 weeks at the "Centrale Electronucleaire E. Fermi" in Trino, Italy assisting with remote inspection of reactor internal parts and re-design of remote handling tools.

Completed my overseas assignment in July, 1968 and was then reassigned to assist in the chemical cleaning of the main Steam Feed Piping at the Rochester Gas & Electric Nuclear Power Plant. Worked with Mr. H. Kordesh of Bechtels startup organization. Completed this assignment in December 1968.

1969 - Supervisory Service Engineer. Assigned to the Indian Point Unit No. 2 startup activities and support group at Pittsburgh, Pa. writing System Flushing and Hydro Procedures. Supplying design information when requested by site personnel.

1969 - Present - Westinghouse Nuclear Energy Systems, Construction and Services. Supervisory Service Engineer.

Transferred to the Indian Point site to assist in carrying out the system Flush & Hydrostatic testing. Involved in writing startup procedrues for various nuclear steam supply systems. Assisting in all phases of construction checkout and turnover to startup. Assisting in writing "Hot Functional" test series.

Position Title: Startup Engineer (Chart II "B2.4.1")

Primary Function:

Act as cognizant engineer for assigned systems in startup of Indian Point Units No. 2 and 3. Responsibilities include test procedure research, writing, resolution of comments, final issue and test conduct.

Duties & Responsibilities:

1. Follow system through construction phase, report on status and remain familiar with field changes or other problems affecting testing.

2. Conduct research into design objectives, system parameters in sufficient detail to write test procedures.

3. Write test procedures.

4. Assist management in obtaining Con Edison approval of procedures.

5. Resolve comments and issue final approved test procedures.

6. Conduct system tests in accordance with approved procedures.

7. Resolve testing problems, coordinate activities of test personnel, identify significant problems of delay, inability to meet test objectives, personnel or plant safety to supervision.

Position Requirements:

Education & Experience - College Degree in Engineering, Physics or other Science. Alternate to degree shall be high school graduate with minimum of two years experience in nuclear plant testing, operations, training, construction or direct support of these nuclear activities.

Supplement 8
4/70

Personnel Resume (Chart II "B2.4.1")

Education:

B.S. in Electrical Engineering from Columbia University - 2/28/62.

2 years Law School at the University of Virginia
1 year Post-graduate study in Political Science at Columbia

Experience:

4-67 - 3-68 Senior Startup Engineer for Bechtel Corporation in San Francisco. Completed assignments at Great Canadian Oil Sands Project, Peach Bottom Nuclear Power Plant, and Tarapur Atomic Power Project (India).

Job Description

1. Preparation of test procedures.
2. Scheduling construction and testing.
3. Supervision of troubleshooting and repairs.
4. Responsibility for preparation and presentation of system modifications to the customer for approval.
5. Responsibility for presenting and selling completed systems to customer.
6. Liaison with customer and vendor personnel.
7. Training customer's operations and maintenance personnel.
8. Responsibility for plant safety and proper technical execution of tests.

Position Title: Lead Refueling Engineer (Chart II "B3.1")

Primary Function:

Ensure fuel is handled properly in accordance with approved procedures. Responsible for correct operation, positioning and monitoring of the in-core loading instrumentation. Concur in the insertion of each fuel assembly into the reactor.

Duties & Responsibilities:

1. Coordinate the efforts of WEDCO refueling personnel, craftsmen and Con Edison personnel in conduct of core loading.

2. Determine the need for technical assistance to support the program and assist in supervising correction of problems.

3. Assist in maintaining a daily log of actual core loading activities.

4. Observe and insure the proper operation of the fuel handling equipment sequencing, inspections and orientation of fuel assemblies from storage to installation of fuel into the reactor vessel.

5. Responsible for supervision and coordination of all fuel handling operations, necessary data acquisition and analysis relating to reactivity control.

6. Authorize the movement of each fuel assembly.

7. Observe and insure the proper sequencing of fuel handling and final material inspection of fuel assemblies prior to installation.

8. Designates the location and position of all fuel assemblies in the core.

9. Keep Manager of Refueling informed of progress, identify critical needs, make recommendations to safely and satisfactorily complete core loading on time.

Position Requirements:

Education & Experience - College Degree in Engineering, Physics or other science or equivalent experience. Participation in previous preparation and conduct of core loading.

Personnel Resume (Chart II "B3.1")

Education:

1953 B. A. Chemistry - Texas Technological College
1960 M.S. Physics - San Diego State College

Experience:

1958 - 1960	- General Atomic Division, General Dynamics Research assistant and licensed reactor operator conducting critical experiments at the Triga Reactor Facility.
1960 - 1967	- Westinghouse Reactor Evaluation Center. Senior Scientist and Senior Reactor Operator directing critical experiments related to PWR design and operations. Also directed and conducted training classes for Senior Reactor Operators.
1967 - Present	- Westinghouse Nuclear Energy Systems. Nuclear plant startup physicist and Plant Operations Specilist participating in plant startup and core loading at Saxton, San Onofre, Zorita, SENA, Indian Point Unit 1 and RGE. Also participated in the SENA and Selni restart programs.

Position Title: Lead Refueling Engineer (Chart II "B3.2")

Primary Function:

Ensure fuel is handled properly in accordance with approved procedures. Responsible for correct operation, positioning and monitoring of the in-core loading instrumentation. Concur in the insertion of each fuel assembly into the reactor.

Duties & Responsibilities:

1. Coordinate the efforts of WEDCO refueling personnel, craftsmen and Con Edison personnel in conduct of core loading.

2. Determine the need for technical assistance to support the program and assist in supervising correction of problems.

3. Assist in maintaining a daily log of actual core loading activities.

4. Observe and insure the proper operation of the fuel handling equipment sequencing, inspections and orientation of fuel assemblies from storage to installation of fuel into the reactor vessel.

5. Responsible for supervision and coordination of all fuel handling operations, necessary data acquisition and analysis relating to reactivity control.

6. Authorize the movement of each fuel assembly.

7. Observe and insure the proper sequencing of fuel handling and final material inspection of fuel assemblies prior to installation.

8. Designates the location and position of all fuel assemblies in the core.

9. Keep Manager of Refueling informed of progress, identify critical needs, make recommendations to safely and satisfactorily complete core loading on time.

Position Requirements:

Education & Experience - College Degree in Engineering, Physics or other Science or equivalent experience. Participate in previous preparation and conduct of core loading.

Personnel Resume (Chart II "B3.2")

Education:

High School Graduate

1 year U.S. Navy Electronics School

Graduate 1968 Capitol Radio Engineering Institute

Experience:

2 years	Union Switch and Signal - Electronics Technician, Flight Simulators
10 years	Westinghouse Reactor Evaluation Center - Test Engineer, Critical Experiments Senior Reactor Operator
2 years	Westinghouse Astronuclear - Test Engineer, Controls Group
9 months	Westinghouse Startup - Specialist, physcs measurements, all aspects of I&C computers. Participated in programs at Connecticut Yankee, San Onofre, and Indian Point. Primary experience is with reactor operations at power. Participated in core loading at Rochester Gas & Electric and Carolina Power & Light.
11-63 - 4-67	Employed by General Dynamics/Electric Boat Division in Groton, Connecticut. Spent 22 months as an Electrical Design Engineer at the Groton shipyard, followed by 20 months as a Nuclear Test Engineer at the National Reactor Testing Station in Idaho Falls, Idaho.

Job Description:

1. Design of submarine communications and weapons systems.

2. Resolution of construction and design deficiencies.

3. Responsibilities as a Nuclear Test Engineer were essentially the same as those of the Startup position described above.

4. Completed In-Plant Training Programs on the S5W and S5G Naval Nuclear Power Plants.

QUESTION 13.4

With regard to the startup Organization:

3. Submit personnel resumes for Con Edison personnel participating in the initial tests and operation of the facility such as Shift Supervisors and Assistant to the General Superintendent.

ANSWER

Con Edison site personnel participating in the initial tests and operation of the facility are as indicated on Figure 1 of the response to Question 12.1 in Supplement No. 2. Resumes of four members of Technical Services Bureau, who will actively participate in initial testing and operation of the facility, are attached.

Richard J. Bozek - General Supervising Engineer (Health Physicist)

Education:

Evander Childs H.S., New York City, 1942 to 1945, Graduate

The Cooper Union, New York City, 1952 to 1962 BSEE Degree.

Experience:

1946 to 1948 - William J Hirten Co., New York City, Stockman

April 1948 to date - Consolidated Edison Co of New York, Inc

April 1948 to December 1951 - Mechanic - Substation Maintenance Department - Maintenance of electrical equipment in A.C. and D.C. substations.

December 1951 to June 1960 - Technician - Technical Services Bureau - Testing and maintenance of recording, controlling and protective equipment in conventional generating stations.

June 1960 to July 1962 - Technician at Indian Point Unit No 1 involved in pre-operational testing of plant

July 1962 - Promoted to Assistant Engineer in Technical Services Bureau -

August 1962 to June 1966 - Westchester Operations of Technical Services Bureau - Control and protective equipment on transmission and distribution systems

February 1963 - Promoted to supervisor in Technical Services Bureau.

June 1966 - Assigned to Indian Point

April 1967 - Promoted to Supervising Engineer - (Health Physicist)

October 1967 - Completed course in "Basic Radiological Health" given by Department of Health, Education and Welfare

December 1969 - Promoted to General Supervising Engineer.

Irving M. King - Assistant Supervising Engineer

Educat' n

Manual Training H.S. Brooklyn, N.Y. - Graduated 1926

Pratt Institute, Brooklyn, N.Y. I.E.E. 1937

Experience

July 6, 1926 to date - Consolidated Edison Co. of N.Y. Inc.

1926 to 1935 - Tests and Repairs of watthour and demand meters, indicators and recorders. Two years of Technical Report Evaluation.

1935 to 1959 - Testing, new and reconditi ned distribution transformers and equipment. gas valves tests developing operational failures in high pressure gas regulators.

1956 - Completed course given by Westinghouse in Electronics

1960 to 1961 - Assisted with preoperational testing of Indian Point Station equipment

1961 to 1962 - Supervisor - Test Bureau at Indian Point Station.

1962 to 1965 - Health Physics Supervisor Test Bureau - Indian Point Station.

1963 - Completed course "Basic Radiological Health" by U.S. Public Health Service.

1965 to Date - Assistant Supervising Engineer for Technical Services Bureau at Indian Point Station.

George H. Liebler - Supervising Engineer

Education:

Cardinal Hayes H.S., Bronx, N.Y. - Graduated 1948.

Pratt Institute, Brooklyn, N.Y. - Two and one-half years electrical technology.

Experience:

1948 to 1959 - Technician in standards laboratory of Technical Services Bureau at 708 1st. Avenue. Certification of the standards for voltage and current used by Con Edison. Testing of all portable electric instruments used on system by the Bureau Started and worked on pre-operational environmental monitoring program for Indian Point.

1959 to 1961 - Technician in the Relay and Instrument Division of the Technical Services Bureau at Astoria and Hell Gate generating stations. Testing and repair of operating instrumentation.

1961 to 1965 - Health Physics Technician at the Indian Point Station.

1965 to 1969 - Assistant Supervising Supervising Engineer in charge of Health Physics work in the plant and the Radiological Environmental Program outside the plant.

1969 to Date - Supervising Engineer

James P. Mooney - Assistant Supervising Engineer

Education:

Saunders Tec. H.S., Yonkers, N.Y - Graduated 1952

Completed course on computers and semi conductors RCA Institute 1966.

Completed course on Basic Radiological Health - U.S. Public Health Service 1969. Attended Westinghouse training program Unit 2 Pittsburgh, Penn. 1968.

Experience:

1952 to 1960 - Technician Waterside generating Station. Instruments, Relays and control systems.

Assigned to System Operators Board 708 First Avenue.

1956 to 1959 - Electric, gas and steam telemetering and auto load frequency control.

1960 to 1968 - Technician at Indian Point - Assigned to Construction and startup testing on conventional and nuclear control systems, rod drive system, nuclear instrument and safety system.

1968 - Technician Foreman - In charge of nuclear control and instruments, rod drive, nuclear instrument and safety system

1969 - Assistant Supervising Engineer - In charge of nuclear instrument and safety system - rod drive gas turbine construction and startup testing - Unit No 2 Construction testing. Responsible for modifications to reactor auto control system, nuclear instrument and safety system and rod drive system

QUESTION 13.4

With regard to the startup Organization:

4. Who will analyze test results and give final approval as to the acceptability of plant components, systems and operating characteristics of the facility?

ANSWER

All test results will receive a preliminary review and evaluation by Con Edison site personnel and cognizant Westinghouse site startup engineers to determine the adequacy of test data for verification of design objectives. Detailed analyses of test results and issuance of final test reports will be performed by Westinghouse site startup and/or engineering and design personnel with input from Con Edison where appropriate. Con Edison will review all final test results to determine that design objectives and criteria have been met and will give final approval as to the acceptability of plant components, systems and operating characteristics of the facility.

QUESTION 14.1

Calculate the required iodine reduction factor necessary to meet the 10 CFR 100 guideline values for the available exclusion and low population zone radii using the following assumptions:

a. Power level of 3216 Mwt.

b. TID-14844 fission product release fractions (100% noble gases, 50% of the iodines, 1% of the solids).

c. An iodine plateout factor of 2.

d. 10% of the airborne iodines being methyliodide.

e. Containment leak rate of 0.1%/day for the first 24 hours, 0.045%/day thereafter.

f. With the following meteorology:

   1 Pasquill Type F, 1m/sec, non-varying wind direction, and volumetric building wake correction factor with C=1/2 and the cross-sectional area of the containment structure for the first 8 hours.

   2. From 8 to 24 hours, Pasquill Type F, 1m/sec with plume meander in a 22-1/2 degree sector.

   3. From 1 to 4 days, Pasquill Type F and 2m/sec with a frequency of 60% Pasquill Type D and 3m/sec with a frequency of 40%, with a meander in the same 22-1/2 degree sector.

   4. From 4 to 30 days, Pasquill Type C, D, and F each occurring 33-1/3% of the time with wind speeds of 3 m/sec, 3 m/sec, 2 m/sec, respectively, with meander in the same 22-1/2 degree sector 33-1/3 of the time.

g. A breathing rate of 3.47×10^{-4} m^3/sec for the first 8 hours, 1.75×10^{-4} m^3/sec from 8-24 hours, and 2.32×10^{-4} m^3/sec thereafter.

With the iodine dose reduction factors caluated as above and 10% methyliodide removal only by a system specifically designed to remove it, provide an analysis and discuss how the guideline values of 10 CFR 100 will be satisfied for the facility.

ANSWER

I. REQUIRED OVER-ALL DOSE REDUCTION FACTORS

The required analyses have been performed. The resulting unreduced doses for each time period are given below, and all input values are given in Table I. Dividing these unreduced doses by the 10 CFR 100

Time Period (hr)	Distance (M)	Unreduced Thyroid Dose (Rem)
0 - 2	520	1042
0 - 2	1100	544
2 - 8	1100	1464
8 - 24	1100	562
24 - 96	1100	358
96 - 720	1100	214
0 - 720	1100	3142

criteria level of 300 rems, the required over-all dose reduction factors are 3.5 for the two hour dose at the site boundary, and 10.5 for the 30 day dose at the low population zone. For these calculations of unreduced doses, the quantities of activity released from the containment were calculated with the PREL digital computer code, which solves the following first order linear differential equation for each isotope.

$$\frac{dC(I)}{dt} = -\lambda(I) \quad C(I)$$

where:

$C(I)$ = containment inventory of isotope I at any time, curies

$\lambda(I)$ = total removal rate of isotope I, hr^{-1}

t = time, hr

The total removal rate $\lambda(I)$ is the sum of the rates of reduction of the containment inventory due to natural decay and leakage.

The code uses values of $\lambda(I)$ which are constant for each of the several time periods in a computation. The total activity of each isotope released from the containment for each time period was computed with the relationship:

$$Q(I,T) = \lambda_L (I) \int_{t_1}^{t_2} C (I) \, dt$$

where:

$Q(I,T)$ = activity of isotope I released in time $t_2 - t_1$, curies
$\lambda_L(I)$ = containment leak rate, hr^{-1}

The resulting activity releases were used in the following standard relationship for the inhalation dose from each isotope for each time period:

$$D(I, T) = Q(I, T) \cdot DCF(I) \cdot B(T) \cdot \frac{\chi}{Q} (x, T)$$

where the undefined terms are:

$D(I,T)$ = inhalation dose from isotope I during period T, rems
$DCF(I)$ = inhalation dose conversion factor for isotope, I, rem/curie
$B(T)$ = breathing rate, m^3/sec
$\frac{\chi}{Q}(x,t)$ = atmospheric dilution factor, sec/m^3

II. DOSES FROM INORGANIC IODINES

The reduction of the inorganic iodines in the containment atmosphere has been discussed in Section 14.3.5 of the FSAR. For this evaluation it has been assumed that only one spray pump is operating, resulting in an iodine removal rate of 32 hr^{-1}. It is also assumed that after the first 24 hours, the spray system has no further effect. In addition, it is assumed that the inorganic fraction (90% of the 25% of core inventories initially available) is not affected by the charcoal filter system. Using these assumption, the inorganic doses below were determined.

Time Period (hr)	Distance (M)	Dose from Inorganic Iodines (Rem)
0-2	520	15
0-2	1100	8
2-720	1100	negligible

In determining the total doses, these values were added to the organic doses.

III. DOSES FROM ORGANIC IODINES

The design of the Indian Point Unit No. 2 charcoal filter syst.m has been modified to make it functionally identical to the system provided for Indian Point Unit No. 3. Section 6.4 of the FSAR has been modified to reflect this change. With the present Indian Point Unit No. 2 design, the sodium hydroxide NaOH spray system will be used to remove the elemental iodine and the charcoal filter system to remove the organic iodine.

The charcoal filter system in Unit No. 2 is similar to that in Unit No. 3, as described in supplement 7 to the Preliminary Safety Analysis Report for unit 3. The organic dose reduction factors for this analysis were calculated with the expression presented on page 5.4-3 of Supplement 7, and are listed for several values of the one pass exchange efficiency in Table II. This expression was developed from an activity balance in the containment, which includes the effects of possible long term "saturation" of the filters. The values of the dose reduction factor given by this model are considerably lower than those given by the normal exponential model. The values in Table II show that with the model used, the filters have little effect after 8 hours.

It can be concluded from test data presented in Supplement 7 of the Unit 3 PSAR, and an evaluation of other available data, that efficiencies greater than 70% would be expected for the removal of organic iodines under the conditions following a loss of coolant accident. Applying the dose reduction factors for 70% efficiency to the organic fraction (10%) of the unreduced doses, the organic contribution to the total doses is obtained. The results are listed for each time period below.

Time Period (hr)	Distance (M)	Dose from Organic Iodines (Rem)
0 - 2	520	76
0 - 2	1100	39
2 - 8	1100	55
8 - 24	1100	18
24 - 96	1100	11
96 - 720	1100	7
0 - 720	1100	130

The doses from organic iodines were also calculated assuming the filter efficiency was only 5%, and the results are tabulated below.

Time Period (hr)	Distance (M)	Dose from Organic Iodines (Rem)
0 - 2	520	101
0 - 2	1100	53
2 - 8	1100	128
8 - 24	1100	38
24 - 96	1100	14
96 - 720	1100	7
0 - 720	1100	240

IV. TOTAL INHALATION DOSES

Using the dose contributions from inorganic and organic forms calculated in the previous sections, the total doses are given below for both values of filter efficiencies.

Time Period (hr)	Distance (M)	Total Dose (Rem)	Filter Efficiency (%)
0 - 2	520	91	70
0 - 2	520	116	5
0 - 720	1100	138	70
0 - 720	1100	248	5

It can be seen from the results given in this table that the total doses are lower than the 10 CFR 100 levels, even for filter efficiencies lower than 5%.

TABLE I - INPUT VALUES FOR UNREDUCED DOSES

ISOTOPIC DATA *

Isotope	Initial Inventory (Curies/MW)	Dose Conversion Factor (Rem/Curie)	Decay Constant (Hr^{-1})
I-131	2.51×10^4	1.48×10^6	3.58×10^{-3}
I-132	3.81×10^4	5.35×10^4	2.97×10^{-1}
I-133	5.63×10^4	4.00×10^5	3.31×10^{-2}
I-134	6.58×10^4	2.50×10^4	7.92×10^{-1}
I-135	5.10×10^4	1.25×10^5	1.03×10^{-1}

ATMOSPHERIC DILUTION ***

Time Period (hr)	Atm. Dil. Fact. (520 M) (sec/m^3)	Atm. Dil. Fact. (1100 (sec/m^3)
0 - 8	6.7×10^{-4}	3.5×10^{-4}
8 - 24	-	1.2×10^{-4}
24 - 96	-	4.2×10^{-5}
96 - 720	-	5.3×10^{-6}

CONTAINMENT LEAKAGE **

Time Period (hr)	Leak Rate (percent/day)
0 - 24	0.1
24 - 720	0.045

BREATHING RATES ***

Time Period (hr)	Breathing Rate (M^3/sec)
0 - 8	3.47×10^{-4}
8 - 24	1.75×10^{-4}
24 - 720	2.32×10^{-4}

OTHER DATA

REACTOR POWER = 3216 MWth ***

FRACTION OF INITIAL INVENTORY AVAILABLE FOR LEAKAGE = 0.25 ***

REFERENCES

*	TID-14844
**	Indian Point Unit No. 2 FSAR
***	Specified AEC Assumptions

TABLE II - ORGANIC DOSE REDUCTION FACTORS

TIME PERIODS (HOURS)

Filter Efficiency	0 - 2	2 - 8	8 - 24	24 - 96	96 - 720
0.0	1.0	1.0	1.0	1.0	1.0
0.025	1.012	1.070	1.23	1.86	3.098
0.05	1.027	1.141	1.47	2.506	3.174
0.1	1.056	1.289	1.938	3.012	3.174
0.2	1.111	1.582	2.725	3.164	3.174
0.3	1.167	1.862	2.941	3.174	3.174
0.4	1.221	2.114	3.077	3.174	3.174
0.5	1.274	2.331	3.135	3.174	3.174
0.6	1.328	2.506	3.156	3.174	3.174
0.7	1.379	2.645	3.166	3.174	3.174
0.8	1.431	2.762	3.171	3.174	3.174
0.9	1.479	2.849	3.174	3.174	3.174
1.0	1.529	2.915	3.174	3.174	3.174

QUESTION 14.2

There appear to be ambiguities in the FSAR with regard to the dose design criteria for the control room. State the dose criteria for the control room.

Calculate the whole body and thyroid dose that would be received by an operator in continuous occupancy of the control room during the course of an accident using the assumptions indicated in 14.1 above. State all assumptions and justify them.

ANSWER

The dose criteria applicable to the control room are as follows:

> Personnel remaining in the control room for an 8 hour period of time following a loss of coolant accident must not receive doses greater than 5 rem to the whole body and 30 rem to the thyroid.

The design of the control room ventilation and air conditioning system is presented in section 9.9.2 of the FSAR. During normal operation, conditioned air is admitted to the control room through downward directed ceiling registers located 14'-9" above the control room floor. A perforated aluminum or egg crate ceiling is located 12 feet above the floor.

The damper in the makeup air supply duct is partially open during normal operation and under remote manual control. This damper will close automatically on a high activity signal from the area monitor (Radiation Monitoring System) in the control room. This signal also automatically starts the separate HEPA-charcoal filter unit fan and positions dampers to route flow through this unit.

The control building volume served by this system is 43,400 ft^3, and the air conditioning system flow is 9200 cfm. The flow rate through the charcoal filter unit (during accident mode operation only) is 1840 cfm. The filter efficiencies of the HEPA and charcoal filters are expected to exceed 95% and 90%, respectively.

The makeup flow from the outside is normally 920 cfm; however, as a result of a system change, this flow will also be terminated automatically by the ECCS demand signal. For this reason, no activity can enter the control room volume from makeup flow following a major accident. Activity is assumed to enter the control room through inleakage.

The thyroid and whole body doses for the Indian Point Unit No. 2 control room following a postulated loss of coolant accident have been calculated using the control room parameters. With the assumptions listed below the radiation doses in the control room are as follows:

	0-2 Hr	0-8 Hr	0-24 Hr	0-30 Day
Design Basis Accident				
Thyroid Dose (Rem)	1.29	9.46	14.65	26.9
Whole Body Dose (Rem)	0.0035	0.0087	0.014	0.017
Hypothetical Accident				
Thyroid Dose (Rem)	43.92	244.8	394	651
Whole Body Dose (Rem)	0.86	2.02	2.46	2.61

The assumptions used in the analysis are as follows:

1. Design Basis Accident

   Release to the containment atmosphere of 100% of the halogens (95% inorganic, 5% organic) and 100% of the noble gases in the fuel pin gap.

   Hypothetical Accident

   Release to the containment atmosphere of 50% of the halogens (47.5% inorganic, 2.5% organic) and 100% of the noble gases in the core.

2. Spray removal rate of 32 hr^{-1} for inorganic iodine and 70% filter efficiency for organic with 3 of 5 fan units operating in the containment

3. Containment leak rate of 0.1 % per day for the first day and 0.045% per day for the remainder of the period.

4. Leak rate of 200 CFM into control room two and 300 CFM into control room one from the atmosphere.

5. Leakage of 150 CFM from control room one to control room two.

6. Filter efficiency of 90% for inorganic iodine in control room two with a flow rate through the filters of 1800 CFM.

7. Finite cloud model used for whole body dose analysis.(1)

8. No beta contribution to the whole body dose from activity outside the control room.

The potential doses in the control room for the design basis accident assumption are within the specified criteria. As assurance that the Operators will not receive excessive doses even for the postulated hypothetical release, Scott air packs will be available in the control room. These self-contained breathing units will eliminate the inhalation dose for as long as the unit is worn. Area radiation monitors inside and outside the control room will alarm at low radiation levels to inform the Operators that a radiation hazard exists.

(1) Meteorology and Atomic Energy 1968, (TID-24190) U. S. Atomic Energy Commission.

Question 14.3.1

Provide plots of time-to-signal vs. break size for the following three safety injection initiation signals: coincidence of low pressurizer pressure and level, high containment pressure and high-high containment pressure. At what percentage of the containment design pressure will the two pressure signals be set?

Answer

The answer appears on revised pages 7.2-14, 7.2-27, 14.3.1-13, 14.3.1-28, and Figure 14.3.2-42. Figure 14.3.2-42 does not include a curve for high-high containment pressure since this is not a safety injection initiation signal as explained on page 7.2-27.

QUESTION 14.3.2

Describe the design and tests of the valve and valve operator which are added to the sump suction lines to limit the consequences of a passive failure in the sump suction line.

ANSWER

The design of the containment sump valves and operators is described on page 6.2-25 in the Motor Operated Butterfly Valves section. The environmental tests of the valve operators are given on page 6.2-35a and are identical to other essential valve operators in the containment. Related information on the compatability of the valve materials with the con ment environment appears in the response to Question 6.3 and 6.4.

In addition, Nordel material has been irradiated to levels from 1.6×10^6 to 1.1×10^8 rads. The exposed seals (fabricated from Nordel materials) operated satisfactorily through three hours test phases totaling 500 hours.

Question 14.3.3

Demonstrate that the 9-second difference in assumed starting time and design starting time does not significantly change the calculated performance of the ECCS pumps, both high head and low head, for representative large and small breaks.

Answer

The above question will be answered later in a WCAP report which describes the core cooling analysis for Indian Point Unit No. 2. This report will describe the codes which have been used for the Indian Point Unit No. 2 analysis and will present details of the computational models and experimental verification. A design evaluation of the ECCS and core will be presented including margins in ECCS design and a sensitivity study of the important parameters. This report will be issued about December 1, 1969.

Question 14.3.4

Provide a quantitative assessment of the analytical conservatism in the core heatup calculations for intermediate sized breaks (0.2 to 3.0 ft^2); include considerations of time to DNB, transition boiling, water level swell, fuel pin gap conductance.

Answer

For Answer, refer to Question 14.3.3.

Question 14.3.5

Provide the conditions and results of Westinghouse fuel rod perforation tests, cladding eutectic tests, and clad shatter tests in support of your LOCA analyses and conclusions. Provde details of the conditions for the perforation, eutectic, and shatter tests, including descriptions of the test rigs and geometries, steam flow, purity of steam and air content, fuel rod fabrication relative to commercial fabrication, fuel rod irradiation, clad heatup rate, and type of heater.

Answer

This question will be given in WCAP-7379-L which will be filed with the AEC.

QUESTION 14.3.6

Provide the details of the models used to simulate the reactor, loop, and reactor internals in the blowdown load calculations using the BLODWN program. Show how the code was used. Identify those components which must survive blowdown to ensure a shutdown and coolable core. Show the logic behind their identification. Provide the stresses or limited deformations predicted for these components.

ANSWER

For answer, refer to Question 14.3.3.

Question 14.4.1

Discuss the ejected rod worth and peaking factors used in your analyses and the relationship of these values to those corresponding to the following cases:

A. Control bank insertion to the Technical Specification limits;

B. Control bank insertion in excess of these bank limits;

C. Rods fully inserted long enough to result in xenon redistribution (assuming operations at both full power and low power following full power operation); and

D. X-Y xenon oscillations.

Answer

A. All analyses presented in Section 14.2.6 refer to rod bank insertion to at least the limits of the Technical Specifications. The cases considered in Section 14.2.6 are RCCA ejection from zero power and ejection from full power. No intermediate case is limiting. Transient hot channel factors are listed in Table 14.2.6-2. Initial, or pre-accident, hot channel factors for the full power cases are taken, conservatively, to be the design maximum, 3.23, even though the calculated values are lower.

B. Control bank insertion in excess of the bank limits has been considered (of page 14.2.6-8 and 9) even though visual and audio alarms are provided by the Reactor Control and Protection Systems at the bank limits. To summarize, ejection of a rod from a fully inserted bank at full power would not cause fuel or cladding to melt.

C. All analyses conservatively assume that the rod is ejected from an equilibrium (rodded) xenon distribution so that the local change in reactivity is a maximum.

D. The core will be operated within its design limits even in the presence of an X-Y xenon transient. As was stated in answer A. above, all analyses assume that the core was at design hot channel factors limits at the start of the accident.

Question 14.4.2

If the reactor is operating with one percent failed fuel, discuss the influence on the rod ejection analyses of any hi-burnup waterlogged fuel which may have lower threshold levels for prompt rupture.

Answer

This is an area where a relatively small amount of experience and data exists. We have, as a result. initiated an analytical study of the behavior of waterlogged fuel rods. However, there is no reason to expect waterlogging to occur in service except due to a random failure of a component such as an end plug weld. The rupture of such isolated fuel rods during a rod ejection accident would not, on the basis of the spert waterlogged fuel tests, cause coolant channel blockage, and the increase in coolant activity due to the failure of several waterlogged rods would not be significant.

Question 14.4.3

Your analyses of the control rod ejection accident indicate that limited fuel rod damage would be anticipated and that rapid fuel dispersals into the coolant would not occur. However, these analyses predict the fuel pin conditions for only the first few seconds following the hypothesized rod ejection. Since the events leading to the rod ejection accident also result in a loss of primary coolant accident, your analyses should be expanded to examine the fuel rod conditions over a longer time period.

The expanded analyses should consider the core lifetime and power conditions which lead to the reactivity insertion which is the most severe with respect to fuel rod conditions following the ejection accident. The analyses and discussion of the results should include the following considerations:

(a) The core cooling systems assumed to function and the sensitivity of the analytical results to changes in the assumed core coolant delivery.

(b) The transient primary coolant conditions within the core region and within the primary system.

(c) The energy input to the core due to the rod ejection, and the local energy input, including appropriate peaking factors. (These energy inputs should be compared with those customarily assumed for the analysis of the loss-of-coolant situation with the same break size but without the rod ejection.)

(d) The transient heat transfer assumptions employed for the rod ejection analyses and where these assumptions may differ from the loss-of-coolant analyses with the same break size.

(e) The transient fuel and clad conditions; i.e., temperature, perforations, oxidation, locally and extending over the core.

Based on the transient coolant conditions and the initially high enthalpy fuel conditions, discuss the potential for rod to rod propagating effects; e.g., heat transfer disturbances, perforation related disturbances, and oxidation related disturbances. How would the fuel and clad conditions predicted in (e) be influenced by such potential propagating effects? If the effects are adverse, then discuss the initial enthalpy conditions for which the fuel rod would remain in an intact and coolable status. Relate this to reactivity inserted from the rod ejection.

Answer

Since an extensive discussion of the loss-of-coolant accident is presented elsewhere in Section 14, and since the largest possible aperture for coolant loss which could occur in a rod-ejection accident is less than 2.5 square inches, the coolant loss has been ignored in the rod ejection analysis. The rod ejection accidents occurring at full power are essentially over in ten seconds with an excess energy insertion of less than 0.5 full power seconds (Figure 14.2.6-11). The change in primary coolant conditions during the first 10 seconds of the blowdown through the 2.5 square-inch aperture is insignificant and would act principally to reduce the slight pressure increase which accompanies the rod ejection accident (cf., page 14.2.6-10).

In response to particular questions:

a. Core cooling systems required are the same as for any small loss of coolant accident.

b. Analyses have been performed with and without DNB assumed. Coolant pressure has been assumed constant for the few seconds of interest. Consideration of a 10% pressure reduction would

   i. contribute negative reactivity and mitigate the power excursion

   ii. lead to a clad temperature not more than 100°F higher than those given.

c. Rate of energy addition is given in Figure 14.2.6-11 and 12. Energy addition peaking factors are given as "transient F_Q" in Table 14.2.6-2. The excess energy release in the full power cases is less than 0.5 full-power-seconds, and the hot spot temperatures have returned to near normal in less than 10 seconds, a short time interval compared to the time required to blow down the primary loop with a 2.5 square inch aperture.

d. Gap heat transfer coefficients are given in Table 14.2.6-2. Analyses have been performed with and without assumed DNB. Heat transfer models are consistent with those used in small-break loss of coolant accidents.

e. Local deformation due to the rod ejection is discussed on page 14.2.6-6 and in answer to question 14.4.2 above; in brief, lattice deformation is not expected. Deformation due to the coolant loss is discussed elsewhere in Section 14. The combination of rod-ejection and loss-of-coolant accident does not lead to worse results because the rod ejection accident is over, without lattice deformation, before the coolant loss becomes significant, and hot spot temperatures are no higher than design temperatures by the time the coolant loss becomes significant.

QUESTION 14.5

Provide an analysis of the consequences of total blockage of the inlet nozzles of fuel assemblies. Include the number of nozzles which must be blocked before fuel damage would occur, the nature of the fuel damage, and the means by which such blockage might be detected.

ANSWER

The blockage of fuel assembly inlet nozzles and the resulting effect on DNB has been considered. With the reactor operating at nominal conditions, partial blockage causing flow reductions of 90% in adjacent fuel assemblies (including the hot assembly) having a total cross-section area of approximately 210 in^2 can be tolerated before DNB is expected to occur. This cross-section area is equivalent to an open lattice array of 26 x 26 fuel rods. Partial blockage causing flow reductions of 80% at the inlet of adjacent assemblies (including the hot assembly) having a total cross section area of about 350 in^2 is also tolerable. This would correspond to an open lattice array of 33 x 33 fuel rods. The above blockage conditions result in a minimum DNBR of 1.11.

No fuel rod damage is expected for DNB ratios greater than this value. This is based on more than 100 rod bundle data points obtained by Westinghouse[1]. In these series of tests the ratio of measured power to predicted power using the W-3 DNB correlation was always equal to or greater than 0.9, the mean value being 1.08. Thus, the maximum value of DNBR at which DNB occurred was 1.11.

The answer provided above was calculated using the computer code THINC, which is the basic tool for the thermal and hydraulic analyses for the Westinghouse open lattice core. The question as stated asked for analysis for flow conditions (full blockage) that would result in an infinite enthalpy rise at the channel entrance, and thus, a meaningless response. We have provided results for several highly improbable flow blockage conditions which are still tolerable in that no fuel damage is expected to occur.

REFERENCES

(1) J. Weisman, A. H. Wenzel, L. S. Tong, D. Fitzsimmons, W. Thorne, and J. Batch, "Experimental Determination of the Departure from Nucleate Boiling in Large Rod Bundles at High Pressure," AIChE, Preprint 29, 9th National Heat Transfer Conference, 1967, Seattle, Washington.

QUESTION 14.6

Based on our evaluation of the fuel handling accident, we have concluded that the design bases for equipment in the fuel handling area should consider that all rods in an assembly could be perforated by dropping a fuel assembly during refueling. In calculating resulting offsite exposures we assume that 20% of the noble gas and 10% of the halogen would be released and that 90% of the halogens would be retained in the fuel storage pool. The resulting thyroid dose at the site boundary is in excess of 10 CFR 100 guidelines. Please state what corrective measures or design changes will be made to insure that off-site doses resulting from this accident will be less than the 10 CFR 100 guidelines.

ANSWER

Additional modifications will be made to the fuel handling building prior to the end of the first year of full power operation to give further assurance that the consequences of a fuel handling accident involving a spent fuel assembly will be acceptable. This system, prior to installation, will be subject to review by the AEC Division of Reactor Licensing The modifications, which include the addition of charcoal filters, will be designed to reduce the dose to a small fraciton of the guidelines of 10 CFR 100 at the site boundary. The charcoal filters will remove iodine with an efficiency of 90%, and the building and its ventilation system will be designed to give an overall dose reduction factor of 5 when compared to an uncontained release.

As further evidence that the consequences of the postulated accident in which all rods are breached are acceptable, an investigation was made to determine what the expected situation would be in terms of iodine available for release to the spent fuel pool water, the retention of iodine in the pool water, and the resulting doses at the site boundary. These analyses do not take credit for either building holdup of the iodine or removal by charcoal filters.

The consequences of an accident in which all rods in an assembly are breached under water in the spent fuel handling area were analyzed. Two cases were considered: an expected case, in which the expected characteristics of the highest rated assembly normally to be discharged at end-of-life are assumed, along with best-estimate transport behavior of fission products determined by tests; and a conservative design basis case, in which factors are introduced to allow for uncertainties.

The expected case is summarized in Table 14.6-1. The resulting maximum off-site doses are calculated to be 3.3 rem thyroid and 0.4 rem whole body, assuming continuous exposure during passage of the released radioactivity. These doses are well below 10 CFR 100 limits.

In order to impart a degree of bias to the calculation sufficient to assure conservative results, margins for uncertainty were then applied to experimental data and the assumptions regarding fission product inventories. In the design basis case thus obtained, summarized in Table 14.6-2, the maximum off-site doses are 14 rem thyroid and 1.1 rem whole body. Thus, even in the event that unexpected deviations from predicted behavior were to occur, the consequences of the postulated fuel handling accident are concluded to be within the guidelines of 10 CFR 100. In view of these results, no further provisions in the design of the fuel handling and storage facility are relied upon to assure conformance to regulations governing accidental release of radioactivity. All reasonable measures will be employed in the handling of irradiated fuel, however, to ensure against the occurrence of fuel damage and the associated radiological hazard.

Basis for Assumptions

In analyzing the conseuqneces of a fuel handling accident, we have made assumptions which differ from those set forth in Question 14.6. These differences are justified on the basis of specific design parameters and conditions of operation of the fuel of this reactor, and by means of tests conducted for this purpose. (See WCAP-7518-L, Westinghouse Proprietary, a summary of which follows). The major differences lies in the decontamination facto (DF) for iodine which is assumed for gas bubbles passing through the spent fuel pit liquid: whereas the DF stated in the question is 10 (i.e., 90% retained by the liquid) we have assumed 740 in the expected case and 500 in the design basis case. These values are based on tests conducted by Westinghouse to measure the absorption efficiency of bubbles under conditions closely simulating the accident.

An experimental arrangement, consisting of a 9 inch diameter vertical column containing boric acid solution at a depth of up to 7.5 feet, was utilized in the study. At the bottom of the column, a gas injection vessel was provided to permit the introduction of a fixed gas (nitrogen) containing iodine vapor at the design basis concentration for a Westinghouse fuel assembly. The gas mixture was injected in the solution and the stripping efficiency for the molecular iodine was determined by inventory of the fraction which excaped from the aqueous soltuion, compared to the quantity retained in the solution. The bubble size was controlled, as was the solution depth, so that the relationship between DF and these variables could be determined.

The results of these tests provide a relationship between DF and bubble size for given water depths (bubble residence time). To obtain a basis for selecting a conservative bubble size with which to model the release of gas from a broken fuel assembly, a full-scale mockup test was conducted. The mockup simulated in exact detail the cross section of a Westinghouse fuel assembly, fitted up in such a way as to permit the simultaneous release from all tubes of a quantity of gas at a pressure and volume corresponding to the void space of an actual fuel assembly. The gas was discharged in a large pool at a depth of 26 ft, approximately the depth of the top of the spent fuel racks. Effective bubble size was determined by measuring the absorption of a trace gas in the full scale release and comparing this with absorption of the same gas with controlled bubble diameter in the laboratory column.

Using the laboratory column data as a basis for the efficiency of absorption of iodine relative to the trace gas, the expected DF for iodine was then calculated for the full-scale release at the 26-ft depth. Best-fit of the experimental results gave a DF of 1200 when the release from a Westinghouse designed non-pressurized fuel assembly was simulated, and 740 when the gas pressure was elevated to simulate Westinghouse designed pressurized fuel. The thyroid dose in Table 14.6-1 was based on the lower DF, although the first core for Indian Point Unit 2 will not be pressurized; future cores in Unit 2 may employ pressurized fuel.

Evaluation of the experimental results was then repeated making allowanc for uncertainties in the measurements and modeling assumptions. The conclusion was that values of 900 and 500 would conservatively represent the DF for non-pressurized and pressurized fuel, respectively. Accordingly, a value of 500 was assumed for purposes of calculating the thyroid dose in Table 14.6-2.

Dose Calculation Mehtods

The dose calculations were performed for an instantaneous release of the fuel assemlby gap activity to the spent fuel pit water. The dispersion factor (χ/Q) used in the calculations is for the site boundary, short term dose.

A. Inhalation (thyroid) dose

Since practically all halogen isotopes except iodine-131 in the fuel assembly have decayed to insignificant levels after 100 hours decay time, the thyroid dose is calculated using I-131 activity only. The following equation is used to obtain an integrated dose:

$$D_{thy}\left(\frac{2\ hr}{S.B.}\right) = \left[A(131) \cdot e^{-\lambda_{131} t_d} \cdot \left(\frac{1}{DF}\right) \cdot DCF(131) \cdot \chi/Q^{(S.B.)} \cdot B \right]$$

where:

S.B. = site boundary

A(131) = activity of I-131 in fuel assembly gap - Ci

λ_{131} = time constant for I-131 decay

t_d = elapsed time before transport of fuel from containment

DF = decontamination factor for iodine in spent fuel pit water

DCF(131) = dose conversion factor for I-131 (1.48×10^6 rem/Ci)

$\chi/Q^{(S.B.)}$ = atmospheric dilution factor for site boundary distance

B = breathing rate

B. Whole body dose

The whole body dose is based upon both gamma and beta radiation exposure. On an isotopic basis, the equations for integrated doses are given by:

$$D_{WB\gamma_i}\left(\begin{smallmatrix}2\ hr\\ S.B.\end{smallmatrix}\right) = A_i \cdot {}^{X}/Q\ (S.B.) \cdot K \cdot E_{\gamma_i} \cdot e^{-\lambda_i t_d} \qquad \text{(Gamma)}$$

and

$$D_{WB\beta_i}\left(\begin{smallmatrix}2\ hr\\ S.B.\end{smallmatrix}\right) = A_i \cdot {}^{X}/Q\ (S.B.) \cdot K \cdot E_{\beta_i} \cdot e^{-\lambda_i t_d} \qquad \text{(Beta)}$$

where: S.B. = site boundary

A_i = activity of the i th isotope released to the atmosphe

K = combination of physical constants for gamma and beta radiation*

E_{γ_i} = gamma energy (Mev/disintegration)**

E_{β_i} = beta energy (Mev/disintegration)

λ_i = time constant for decay of i th isotope

t_d = elapsed time before transport of fuel from containment

${}^{X}/Q^{(S.B.)}$ = atmospheric dilution factor for site boundary distance

*K is a constant which relates a dose for a semi-infinite cloud in rem to equivalent energy of absorption, energy of decay, and activity.

$$K = 1/2\ (3.7\times10^{10}\ \text{disintegrations}/\text{curie-sec}) \cdot (1.6\times10^{-6}\ \text{ergs/Mev}) \cdot$$

$$\cdot\ (0.01\ \text{rad} - \text{gram}^{(air)}/\text{erg}) \cdot (\tfrac{1}{1200}\ \text{meters}^3(\text{air})/\text{gram(air)}) \cdot$$

$$\cdot\ (\text{RBE}\ {}^{rem}/\text{rad} = 1\ \text{for beta and gamma radiation})$$

$$K = 0.246\ \frac{\text{rem}}{\text{curie}\ (\text{sec}/\text{m}^3)\ (\frac{\text{Mev}}{\text{disintegration}})}$$

**See Table 14.6-3.

TABLE 14.6-1

FUEL HANDLING ACCIDENT-- EXPECTED CASE

Fuel Parameters[1]

Reactor Power (102%)	3280 MWt
No. of Assemblies	193
Fuel Rods per Assembly	204
Normalized Power, Highest Rated Discharged Ass'y	1.29
Thermal Power, Highest Rated Discharged Ass'y	21.93 MWt
Axial Peak/Avg., Highest Rated Discharged Ass'y	1.37
Peak Lineal Power, Highest Rated Discharged Ass'y	11.1 Kw/ft

Temp./Power Distribution[1]

Local Temperature	% Fuel Volume		Power (MWt) in Volume
>3700	0		0
3500 - 3700	0.03		0.008
3300 - 3500	1.13		0.333
3100 - 3300	1.89		0.553
2900 - 3100	3.05		0.856
2700 - 2900	3.49		0.967
2500 - 2700	4.91		1.276
<2500	85.49		17.933
	100.	TOTAL	21.93

TABLE 14.6-1 (Cont'd)

Fission Product Inventory[2]

Isotope	Curies MWt	Curies Assembly	Fraction in Void	Curies in Void	λ-Decay Sec^{-1}	Curies in Void at 100 hrs.
I-131	2.51×10^{4}	5.50×10^{5}	0.0165	9.07×10^{3}	9.96×10^{-7}	6.34×10^{3}
I-132	3.81×10^{4}	8.36×10^{5}	0.00183	1.52×10^{3}	8.02×10^{-5}	0
I-133	5.63×10^{4}	1.24×10^{6}	0.00599	7.42×10^{3}	9.20×10^{-6}	2.7×10^{2}
I-134	6.58×10^{4}	1.44×10^{6}	0.00115	1.65×10^{3}	2.20×10^{-4}	0
I-135	5.10×10^{4}	1.18×10^{6}	0.00338	3.99×10^{3}	2.86×10^{-4}	0
Kr-85m	1.11×10^{4}	2.43×10^{5}	0.00472	1.14×10^{3}	4.41×10^{-5}	0
Kr-85	3.62×10^{2}	7.92×10^{3}	0.242	1.92×10^{3}	2.14×10^{-9}	1.92×10^{3}
Kr-87	3.62×10^{2}	4.66×10^{5}	0.00137	6.38×10^{2}	1.48×10^{-4}	0
Kr-88	3.03×10^{4}	6.64×10^{5}	0.00472	3.14×10^{3}	6.95×10^{-5}	0
Xe-133m	1.44×10^{3}	3.16×10^{4}	0.0090	2.84×10^{2}	3.49×10^{-6}	8.08×10^{1}
Xe-133	5.70×10^{4}	1.24×10^{6}	0.0135	1.67×10^{4}	1.52×10^{-6}	9.66×10^{3}
Xe-135	1.55×10^{4}	3.40×10^{5}	0.0037	1.26×10^{3}	2.11×10^{-5}	6.33×10^{-1}

TABLE 14.6-1 (Cont'd)

Fission Product Release

D. F. for Retention by Pool Water	halogen	740.[3]
	noble gas	1.00
Curies released to environment		
I-131		8.57×10^{0}
Kr-85		1.92×10^{3}
Xe-133		9.66×10^{3}

Dispersion and Potential Exposure at Site Boundary

Atmospheric Dispersion Factor (χ/Q, sec/m^3)	$?.5 \times 10^{-4}$
Receptor Breathing Rate (m^3/sec)	3.47×10^{-4}
Calculated Adult Thyroid dose, rem	3.3
Calculated Whole Body Dose, rem	0.4

Footnotes:

(1) Conditions of the highest rated assembly are those calculated for the assembly with maximum power in the discharge region during the last 6-weeks prior to refueling shutdown.

(2) Fission products in the void space are determined according to the model described in Section 14.3.5 of the FSAR.

(3) Based on best-fit analysis of data from laboratory column and pool tests simulating iodine release from Westinghouse designed pressurized fuel rods. Indian Point Unit 2 Core will initially have unpressurized fuel rods. For unpressurized fuel, the expected DF would be 1200.

TABLE 14.6-2

FUEL HANDLING ACCIDENT DESIGN BASIS CASE

Fuel Parameters[1]

Reactor Power (102%)	3280 MWt
No. of Assemblies	193
Fuel Rods per Assembly	204
Normalized Power, Highest Rated Discharge Ass'y.	1.63
Thermal Power " " " "	27.6 MWt
Axial Peak /Avg " " " "	1.72
Peak Lineal Power " " " "	15.5 kw/ft

Temp./Power Distribution[1]

Local Temperature °F	% Fuel Volume		Power (MWt) in Volume
>3900	0		
3700 - 3900	1.04		0.372
3500 - 3700	2.29		0.813
3300 - 3900	2.97		1.045
3100 - 3300	3.43		1.155
2900 - 3100	5.37		1.655
2700 - 2900	8.55		2.449
2500 - 2700	9.35		2.620
<2500	66.93		17.521
	100.00	TOTAL	27.63 MWt

TABLE 14.6-2 (Cont'd)

Fission Product Inventory[2]

Isotope	Curies MWt	Curies Assembly	Fraction in Void	Curies in Void	λ-Decay Sec^{-1}	Curies in Void at 100 hrs.
I-131	2.51×10^4	6.99×10^5	0.0368	2.57×10^4	9.96×10^{-7}	1.80×10^4
I-132	3.81×10^4	1.04×10^6	.00426	4.43×10^3	8.02×10^{-5}	-
I-133	5.63×10^4	1.52×10^6	.0129	1.96×10^4	9.20×10^{-6}	7.13×10^2
I-134	6.58×10^4	1.79×10^6	.00264	4.72×10^3	2.20×10^{-4}	-
I-135	5.10×10^4	1.40×10^6	.0073	1.02×10^4	2.86×10^{-4}	-
Kr-85m	1.11×10^4	3.05×10^5	0.00464	1.42×10^3	4.41×10^{-5}	-
Kr-85	3.62×10^2	1.00×10^4	0.345	3.45×10^3	2.14×10^{-9}	3.45×10^3
Kr-87	2.13×10^4	5.99×10^5	0.0032	1.92×10^3	1.48×10^{-4}	-
Kr-88	3.03×10^4	8.38×10^5	.00464	3.89×10^3	6.95×10^{-5}	-
Xe-133m	1.44×10^3	3.96×10^4	.0203	8.04×10^2	3.49×10^{-6}	2.29×10^2
Xe-133	5.70×10^4	1.54×10^6	.0296	4.56×10^4	1.52×10^{-6}	2.64×10^4
Xe-135	1.55×10^4	4.27×10^5	.0086	3.67×10^3	2.11×10^{-5}	1.84×10^0

TABLE 14.6-2 (Cont'd)

Fission Product Release

D. F. for Retention by	halogen	500.(3)
Pool Water	noble gas	1.00
Curies released to environment		
I-131		3.60×10^{1}
Kr-85		3.45×10^{3}
Xe-133		2.64×10^{4}

Dispersion and Potential Exposure at Site Boundary

Atmospheric Dispersion factor (χ/Q sec/m^3)	7.5×10^{-4}
Receptor Breathing Rate (m^3/sec)	3.47×10^{-4}
Calculated Adult Thyroid dose, rem	14
Calculated Whole Body dose, rem	1.1

Footnotes:

(1) Conditions of a hypothetical assembly, with axial peaking arbitrarily increased 25% and peak Kw/ft arbitrarily increased 40% over the referenced nominal discharge assembly.

(2) Fission products in void space, calculated for conservative power/temperature distribution using same model as in Table 14.6-1.

(3) Based on conservative analysis of test data cited in Table 14.6-1, assuming adverse combination of measurement errors.

TABLE 14.6-3

ISOTOPIC DATA

(Taken From Perkins and King)

ISOTOPE	GAMMA ENERGY (Mev/Disintegration)	BETA ENERGY (Mev/Disintegration)
I-131	0.392	0.183
I-132	2.130	0.485
I-133	0.565	0.493
I-134	1.023	0.941
I-135	1.680	0.316
Xe-133m	0.026	0.207
Xe-133	0.027	0.155
Xe-135	0.261	0.304
Kr-85m	0.157	0.252
Kr-85	0.004	0.221
Kr-87	1.586	1.341
Kr-88	1.915	0.372

Supplement 9
5/70

QUESTION 14.7

With respect to reactor protection for anticipated plant transients please state the applicability of the report WCAP-7306 "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors" to Indian Point 2.

Provide an analysis of the loss-of-load transient for Indian Point 2 assuming the reactor does not scram on any of the successive trip signals.

Provide an analysis of the loss-of-flow transient for Indian Point 2 assuming that the reactor does not scram on any of the three levels of trip protection provided.

ANSWER

The analyses presented in WCAP-7306-L "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors," are, in general, applicable to all Westinghouse Pressurized Water Reactors.

As shown in this report, the reactor core is adequately protected against anticipated plant transients by the Reactor Protection System in the Westinghouse design which is both redundant and diverse. As a result, failure to trip is therefore not a credible situation. Nevertheless, in response to the second part of Question 14.7 anticipated plant transients (including loss of flow and loss of load accidents) have been analyzed for the Indian Point 2 plant with the assumption that all reactor trip signals generated by the Reactor Protection System are inactive.

A primary design requirement for the Reactor Protection Systems is to maintain the minimum core hot channel DNB ratio at or above 1.30 for anticipated transients. This ratio is defined as the ratio between the calculated DNB heat flux (calculated as a function of local fluid temperature, flow and pressure with the W-3 correlation) and fuel rod heat flux. This limit is conservatively maintained to ensure a large margin to possible DNBR and thus preclude fuel damage anywhere in the core.

The complete failure of the Reactor Protection System to provide this protection, particularly in view of the demonstrated diversity is an incredible event. Any analyses with such a postulation should be evaluated in light of more realistic limits. These analyses are performed on the basis of a more realistic DNBR limit of 1.0 on the hot assembly providing reasonable assures that gross fuel clad damage is prevented.

The analyses presented here indicate that the most severe anticipated transient conditions could not result in gross fuel clad damage. Since there is no consequential damage to the reactor coolant system, any fission products released through the clad will be retained and the safety of the public will not be affected by the postulated failure of all reactor trip signals.

No. 1 Loss of Load

A turbine trip results in the maximum load loss which can be suffered by the plant. The most severe plant conditions that could result from a loss of load occur if one postulates that the plant is initially operating at full power with the turbine trip being caused by high condenser pressure. Since the main feedwater pumps are turbine driven and exhaust to the condenser, this assumption also causes a loss of feedwater with the turbine trip. Such an accident, would, of course, be terminated almost immediately by the Reactor Protection System as discussed in detail in WCAP-7306. A summary of Reactor Trip Signals that would terminate the accident is given in Table 1.

Since these trip signals are both redundant and diverse, the assumption that the Reactor Protection System failed to trip the reactor is certainly not a credible situation. Nevertheless, the loss of load analysis was performed assuming that all reactor trip signals generated by the Reactor Protection System are inactive. Additional conservatism was introduced by the following assumptions.

1. The initial reactor power, coolant temperatures, and pressure are at extreme values consistent with steady state, full power operation, including errors for calibration and instrumentation. This results in the maximum power difference for the load loss and the minimum margin to DNB at the initiation of the accident.

2. No control due to rod insertion assumed.

3. A conservatively small beginning of life moderator coefficient and a conservatively large power coefficient were assumed. A large power coefficient opposes the effect of the moderator coefficient in reducing power when the reactor coolant temperature increases. Only the beginning of life results are presented since a large moderator coefficient at the end of life reduces the nuclear power and results in an increasing DNBR.

4. Reactor Coolant Pump cavitation is assumed to occur when the pump inlet is 6°F subcooled. It is conservatively represented by a flow coastdown, with a time constant of ~5 sec., at that time. This representation is conservative since its effect is to move the pump inlet temperature away from saturation, until the pump speed decreases to zero, whereas the pump cavitation effect would be to maintain the pump inlet temperature 6°F subcooled which would result in a greater pump flow rate.

Results are presented with and without primary pressure control (pressurizer spray and power operated relief valves). Neglecting pressure control results in the maximum primary pressure during the transient and including pressure control results in the minimum DNBR. Figures 14.7.1-1 and 14.7.1-2 show the transient response for a loss of load at the beginning of core life without primary pressure control. The steam pressure increases to the steam generator safety valve set point (1133 psia including 3% accumulation) and limits the initial reactor coolant system average temperature increase to 604°F. The pressurizer safety valves limit the initial pressure surge to 2500 psia. The slightly negative moderator reactivity coefficient causes the nuclear power to decrease to a pseudo-equilibrium value of 80 percent power. The pressurizer water volume increases 1680 ft^3 during this time. During this time period the hot channel DNBR decreases to a minimum value of 1.81. The hot assembly DNBR (not shown on the figures) decreases to a minimum value of 1.91.

The turbine trip signal is also assumed to trip the main feedwater pumps. Tripping of the main feedwater pumps will generate a signal to actuate the auxiliary feedwater pump ; the auxiliary feedwater pumps are conservatively assumed to start sixty seconds after receiving the auxiliary feedwater pump actuation signal. Because of their limited capacity, the auxiliary feedwater pumps are initially unable to maintain sufficient heat transfer from primary to secondary. This effect is seen as a rapid increase in temperature and pressure approximately 100 seconds after the turbine trip. The pressure increase is further accentuated by the filling of the pressurizer with water

(~120 seconds) resulting in water relief through the pressurizer safety valves. During this time period the nuclear power decreases continuously because of the increased moderator density reactivity feed back. The combined effect of the auxiliary feedwater system heat removal and the safety valve capacity limit the peak pressure to ~4000 psia at 135 seconds. The pressurizer pressure then decreases very rapidly until 160 seconds when the reactor coolant pumps begin to cavitate (simulated by rapid flow coastdown at that time). This effect decreases the rate of pressure decrease.

At ~170 seconds the heat removal capability of the auxiliary feedwater system exceeds the core power generation causing the Reactor Coolant System average temperature to decrease after this time. A steam bubble is reformed in the pressurizer at 210 seconds. At 260 seconds the pressurizer pressure returns to 2500 psia. During this time period the hot channel DNBR is always greater than 1.8.

Figures 14.7.1-3 and 14.7.1-4 show the transient response for a loss of load at the beginning of core life with primary pressure control. The response is similar to figures 14.7.1-1 and 14.7.1-2 with the following important exceptions;

1) The minimum hot channel (and hot assembly) DNBR is lower; a minimum value of 1.70 occurs during the transient.
2) The peak pressure is much lower; a maximum value of 3340 psia occurs as compared to 4000 psia without pressure control.

Figure 14.7.1-5 shows the effect of initial moderator reactivity coefficient on peak pressure assuming both pressure control and no pressure control. Also shown on that figure are the pressure corresponding to the Reactor Vessel Yield stress and the expected moderator reactivity coefficient at the completion of testing and the beginning of full power operation. The figure shows that the yield stress will never be exceeded at the time at which full power operation is expected. In fact, with pressure control, it will never be exceeded and without pressure control it will be exceeded only during a very limited period early in core life. Fifty percent of ultimate stress pressure for the reactor vessel, which is considered to be the allowable limit on pressure for this transient, is not exceeded at any time in core life.

In the analysis no credit is taken for the effect of void feedback on power distribution. However, the results show that a total loss of load presents no hazard to the Reactor Coolant System or Steam System. The minimum DNBR in the hot assembly for the beginning of core life with pressure control is well above the limit of 1.0. In fact, in this case the minimum DNBR at the hot spot is greater than the Reactor Protection System Design value of 1.30.

Table 1

List of Protective Trip Functions

1. Direct Reactor Trip (On Turbine Trip)

2. High Pressurizer Pressure Trip

3. High Pressurizer Level Trip

4. Steam/Feedwater Flow Mismatch Trip

5. Low Steam Generator Water Level Trip

No. 2 Excessive Load Increase

The most severe core conditions that could result from a credible load increase occur if one postulates a simultaneous accidental opening of all steam dump valves when the plant is operating at full power. Such an accident would, of course, be terminated almost immediately by the Reactor Protection System as discussed in detail in WCAP-7306 "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors". A summary of Reactor Trip signals that would terminate the accident is given in Table 1. Since these Reactor Trip signals are both redundant and diverse, the assumption that the Reactor Protection System failed to trip the reactor is certainly not a credible situation. Nevertheless, the excessive load increase analysis was performed assuming that all reactor trip signals generated by the Reactor Protection System are inactive.

Additional conservatism was introduced by the following assumptions:

1. The initial reactor power, coolant temperatures, and pressure are at extreme values consistent with steady state, full power operation, including errors for calibration and instrumentation. This results the minimum margin to DNB at the initiation of the accident.

2. A conservatively large negative moderator coefficient and a conservatively small power coefficient were assumed. This maximizes the power level reached following the accident.

3. The feedwater system flow capability was assumed to be limited to its initial value. This does not affect the initial transient response. However, since the feedwater flow is not able to match the steam flow, the steam generator tubes eventually begin to uncover causing a decrease in steam generator heat transfer capability, resulting in an increase in reactor coolant temperature and an insurge into the pressurizer.

4. The reactor control system is initially in operation and hence the primary system average temperature returns to its initial value in a short time. In practice rod withdrawal would be blocked by the nuclear overpower rod stop. When the reactor coolant system temperature increases due to steam generator tubes uncovering no credit is taken for control rod motion to decrease T_{avg}.

5. No credit is taken for auxiliary feedwater pump operation even though the auxiliary feedwater pumps would be actuated by the steam generator low water level signal which actuates reactor trip.

The transient was analyzed using a full plant digital simulation. The results are shown on figures 14.7.2-1 and 14.7.2-2. The increase in steam load results in a reduction in steam temperature and a corresponding decrease in reactor coolant system temperature. This reduction in reactor coolant system temperature causes an increase in nuclear power. The reactor control system responds to restore T_{avg}; T_{avg} initially settles out at a value of 572.2°F with the nuclear power stabilizing at 125% of nominal. During this time the pressurizer pressure decreases to 2180 psia and recovers to approximately 2230 psia. At approximately 300 seconds after initiation of the transient the steam generator tubes begin to uncover. This causes the reactor coolant system temperature and pressure to increase due to the decrease in steam generator heat transfer capability. No credit is taken for the automatic reactor control system action to reduce T_{avg} to its initial value at this time. The effect of the increase in T_{avg} is to reduce the nuclear power until an equilibrium condition exists between primary and secondary. The peak reactor coolant system temperature during the transient is 578.3°F and the peak pressure is 2340 psia. The DNBR hot channel never decreases below 1.50 throughout the transient.

It should be noted that the analysis assumed that the hot channel factors remain constant during the accident while actually the power feedback will result in considerable flattening of the power distribution. This effect by itself would be enough to maintain the DNB ratio above the 1.3 limit. Thus, we conclude that the worst credible excessive load increase accident with failure to initiate a reactor trip will not result in core damage.

Table 1

List of Protective Trip Functions

1. Overpower nuclear trip.

2. Overtemperature delta T trip.

3. Overpower delta T trip.

4. Low steam generator water level.

5. Steam/Feedwater Flow Mismatch Trip.

No. 3 - Loss of one Feedwater Pump

The most severe core conditions that could result from the loss of one feedwater pump occur if one postulates that the plant is initially operating at full power without any subsequent action from either the Reactor Protection System or the Reactor Control System. Such an accident would, of course, be terminated almost immediately by the Reactor Protection System as discussed in detail in WCAP-7306. A summary of Reactor Trip signals that would terminate the accident is given in Table 1.

Since these trip signals are both redundant and diverse, the assumption that the Reactor Protection System failed to trip the reactor is certainly not a credible situation. Nevertheless, the loss of one feedwater pump analysis was performed assuming that all reactor trip signals generated by the Reactor Protection System are inactive. Additional conservatism was introduced by performing the analysis at a much higher rating, a higher peak heat flux, and a higher primary average temperature, resulting in a lower initial DNB ratio. Thus resulting in less margin to the hot assembly DNBR limit of 1.0 used for this analysis. Table 2 summarizes the differences between the Indian Point 2 plant rating and the plant rating used for this very conservative analysis.

A general investigation of the feedwater system indicated that the loss of one feedwater pump results in a net feed flow deficiency of about 25% of nominal feedwater flow. For Indian Point 2 this feed flow deficiency would be considerably smaller since the feedwater pumps are sized for a power level of 3083 MWt whereas the license application is for 2758 MWt; this fact was not considered in the analysis. This feedwater flow deficiency forces the steam generator downcomer water level to fall with a corresponding increase in the riser exist quality, until finally the steam generator operates essentially on a once through basis.

A detailed steam generator simulation was performed to calculate the primary to secondary heat transfer coefficient (steam generator UA) under transient conditions. The analysis indicates that no reduction in UA will occur until the steam and water inventory falls to approximately 100,000 lbs. This would occur after approximately four minutes if the loss of one feedwater pump accident is initiated from the normal steam generator water level. Hence, in addition to the Reactor Protection System trips, there would be adequate time for the operator to manually trip the reactor. However, in the overall plant simulation of this transient, it was conservatively assumed that the accident is initiated from a much lower steam generator level than the normal operating level. As a result, the steam generator UA starts to decrease at about 30 sec. as shown in Figure 14.7.3-1.

As the steam generator heat transfer coefficient decreases, the primary temperature rises and the negative moderator coefficient results in a decrease in core power. At the same time steam generator pressure falls. The turbine governor attempts to maintain load, but at full power the steam valve soon be es fully open and steam flow falls with falling steam generator pressure. Eventually the system reaches equilibrium with steam flow equaling feed flow, and with higher primary temperatures. In order to maximize the primary temperature rise a large Doppler power coefficient and small moderator coefficient were assumed.

The results of the analysis are shown on the attached figures (figures 14.7.3-1 thru 14.7.3-5). The maximum value of primary average temperature was 630°F and the maximum primary pressure 2530 psia. The core hot channel DNB ratio did not fall below 1.30. The hot assembly DNB ratio (not shown on the figures) never fell below 1.40 during the transient.

It should be noted that the analysis assumed that the hot channel factors remain constant at the design values during the accident while in actuality the power feedback will result in considerable flattening of the power distribution. Even ignoring this effect, the DNBR limit of 1.0 in the hot assembly is not exceeded at any time during the accident. In fact, the minimum DNB ratio in the hot channel is even above the Reactor Protection System design value of 1.30. Considering this and the fact that the Indian Point 2 rating is well below the rating used in the analysis, we conclude that the loss of one feedwater pump accident with failure to initiate a reactor trip will not result in any gross core damage.

Table 1 List of Protective Trip Functions

1. Steam/Feedwater Flow Mismatch Trip.
2. Low water level in steam generator.
3. Overtemperature delta T.
4. High pressurizer pressure.
5. High pressurizer level.

Table 2

Parameter	Indian Point Initial Rating	Value used in study
Power (MWt)	2758	3411
Peak heat flux (BTU/hr-ft^2)	/,300	579,600
T_{AV} (°F)	571	586
Nominal DNBR	2.00	1.89

No. 4 - Loss of Flow (1 Pump)

The most severe core conditions that could result from a credible loss of coolant flow accident occur if one postulates that the plant is initially operating at full power without any subsequent action from the Reactor Protection System. The Reactor Protection System would provide the necessary protection against a loss of coolant flow incident. A summary of Reactor Trip signals that would terminate the accident is given in Table 1.

These Reactor Trip signals are both redundant and diverse (as described in WCAP-7306). The assumption that the Reactor Protection System fails to trip the reactor is, therefore, certainly not a credible situation. Nevertheless, the loss of reactor coolant flow analysis was performed assuming that all reactor trip signals are inactive.

Assumptions

The following conservative assumptions were made.

1. The plant is initially operating at 102 percent of 2758 MWt with the reactor coolant average temperature at 573.5°F (Nominal full power T_{avg} + 4°F).

2. A large value of the Doppler coefficient (-1.0×10^{-5} Δk/°F) and a small value of the moderator density coefficient (typical of beginning of core life conditions) are assumed.

3. Power is lost to one of the four reactor coolant pumps. Each pump is supplied by a separate bus and so loss of a single pump is the worst credible fault. A total loss of power would de-energize the rod drive mechanisms and produce a "de facto" reactor trip.

4. Steam flow to the turbine is assumed to vary linearly with the steam pressure.

Method of Analysis

A detailed digital simulation of the plant including heat transfer to the steam generator of the active and dead loops was used to compute the core power, hot spot heat flux, reactor coolant temperature and the system pressure transient responses. These data were then used in a detailed thermal-hydraulic computation by the THINC-III code to compute DNB ratio.

Results

Two cases were analyzed; one case without any action from either the Reactor Protection System or the Reactor Control System, and the other case with automatic control rod motion.

(1) One pump loss of flow without control rod motion

Reactor coolant flow coastdown curves are shown on Figure 14.7.4-1. Figure 14.7.4-2 shows the nuclear power and hot spot heat flux responses for the one pump loss of flow without any action by the Reactor Control and Protection System. The reactor coolant temperature and pressure responses are shown on Figures 14.7.4-3 and 14.7.4-4. The reduction in the reactor core flow increases the enthalpy rise in the core, causing the core water temperature to rise. At the same time, the decrease in the primary-side film heat transfer coefficient in the steam generators of the dead loop increases the core inlet temperature as shown in Figure 14.7.4-3. The increase in the core water temperature reduces reactor power through the negative reactivity feedback due to the slightly positive density coefficient of reactivity (negative temperature coefficient). Around 28 seconds after the incident, the reactor coolant flow in the dead loop is reversed, resulting in rapid decreases in the average reactor coolant temperature and pressure. The new equilibrium power level is about 96 percent of full power, which is less than the initial power level because the steam flow is reduced by the effect of reduced steam pressure.

Figure 14.7.4-5 shows the DNB ratio in the hot channel and at the hot assembly as a function of time. A minimum DNB ratio of 1.38 in the hot channel is reached 30 seconds after initiation of the incident. Since DNB does not occur, there is no core damage and no release of fission products into the reactor coolant.

(2) One pump loss of flow with automatic rod control

Figure 14.7.4-6 shows the nuclear power and the hot spot heat flux transient for one pump loss of flow with automatic rod control. Initially, control rods are driven into the core and then withdrawn to restore the reactor coolant average temperature as shown in Figure 14.7.4-7. It brings up nuclear power slightly above full power. The DNB ratio in the hot channel and the hot assembly are shown in Figure 14.7.4-8 as a function of time. A minimum DNB ratio of 1.04 is reached in the hot channel and a minimum DNB ratio of 1.26 is reached at the hot assembly about 50 seconds after the incident. Since a DNB ratio of 1.0 in the hot assembly is not exceeded, it is concluded that the one pump loss of flow with failure to initiate a reactor trip will not result in any gross core damage.

It should be noted that the analysis assumed that the hot channel factors remain constant during the accident while in actuality the power feedback will result in flattening of the power distribution in the direction of improving the DNB ratio. This effect would increase the margin to DNB conditions.

Table 1 List of Protective Trip Functions

1. Pump circuit breaker opening.
2. Low reactor coolant flow.
3. Overpower

No. 5 Rod Withdrawal Accident

A rod withdrawal accident may result from a control system malfunction which causes the rod speed programmer to demand a rod withdrawal in the absence of a temperature deviation of a power mismatch. The most severe credible plant conditions that could result from a rod withdrawal accident occur if one postulates that the plant is initially operating at full power without any subsequent action from either the Reactor Protection System or the Reactor Control System. Such an accident would, of course, be terminated almost immediately by the Reactor Protection System as discussed in detail in WCAP-7306, "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors". A summary of Reactor Trip signals that would teminate the accident is given in Table 1.

Since these trip signals are both redundant and diverse, the assumption that the Reactor Protection System failed to trip the reactor is certainly not a credible situation. Nevertheless, the rod withdrawal analysis was performed assuming that all reactor trip signals generated by the Reactor Protection System are inactive. Additional conservatism was introduced by the following assumptions:

1. The initial reactor power, coolant temperatures, and pressure were at extreme values consistent with steady-state, full power operation, including errors for calibration and instrumentation. This resulted in the minimum margin to DNB at the initiation of the accident.

2. The rod insertion began with control bank D at the rod insertion limit low-low limit alarm setpoint and ended with bank D fully withdrawn. Two values of reactivity insertion were examined, 0.3% and 0.8%.

3. The reactivity insertion rate due to the rod withdrawal corresponded to a bank of conservatively high rod worth being withdrawn at maximum rod speed.

4. The moderator coefficient of reactivity was assumed to be a function of moderator density. Val[illegible] ical of beginning of life operation were chosen in order to prov[illegible] a conservatively low amou[illegible]t o[illegible] reactivity feedback due to changes in moderator temperature.

5. The Doppler coefficient of reactivity was assumed to be a function of fuel temperature. Values were chosen so that the resultant amount of feedback was conservatively low.

6. Primary pressure control (pressurizer spray and power operated relief valves) was assumed since this resulted in the minimum reactor coolant system pressure and, thus, the minimum DNBR throughout the transient.

7. The feedwater system flow capability was assumed to be limited to its initial value. This does not affect the initial transient response. However, since the feedwater flow is not able to match the steam flow, the steam generator tubes eventually begin to uncover causing a decrease in steam generator heat transfer capability resulting in an increase in reactor coolant system temperature and an insurge into the pressurizer.

The transient was analyzed using a full plant digital simulation. The results are shown on the attached figures. Figures 14.7.5-1 through 14.7.5-3 show the results for a reactivity insertion 0.3% Δk while figures 14.7.5-4 through 14.7.5-6 show the results for an insertion of 0.8% Δk. For either case, the initial response is an almost immediate increase in nuclear power, reactor coolant system temperature and pressure. When the control rod assembly is fully withdrawn the negative feedback effect of the moderator and Doppler coefficients tends to decrease the power, temperature and pressure to constant values. When the steam generator tubes begin to uncover and heat transfer between primary and secondary is decrease, the reactor coolant system temperature tends to increase. The negative reactivity effect of the temperature rise causes a decrease in power until a new steady state is reached. For the reactivity insertion of 0.3% Δk the peak nuclear power of 143% of nominal is reached with a peak pressurizer pressure of 2350 psia. For this case the hot assembly DNB ratio is 1.1. For the reactivity insertion of 0.8% Δk the peak nuclear power is 158%. For this case the pressurizer fills with water at approximately 60 seconds. Since the pressurizer relief valves have considerably less relief capability when relieving water the pressurizer pressure increases to 2500 at that time. This same effect is seen when the steam generator tubes begin to uncover. For this case the hot assembly DNB ratio falls below 1.0 approximately forty seconds after the accident occurs.

Since the hot assembly DNBR value of 1.0 is approached for the 0.3% Δk insertion and exceeded for the 0.8% Δk insertion, the peak fuel rod clad and fuel temperatures were examined assuming stable film boiling after the DNBR decreases below 1.30. For the 0.3% Δk insertion the peak fuel centerline temperature reaches 5600°F with approximately 20% of the fuel melted in the lead rod pellet at the highest powered axial location which goes through DNB; the peak clad temperature is 1100°F. For the 0.8% Δk insertion the peak fuel centerline temperature reaches 6600°F with approximately 40% of the fuel melted in the lead rod pellet at the highest powered axial location which goes through DNB; the peak clad temperature is 1325°F. It is, therefore, concluded that the rod withdrawal accidents with failure to initiate a reactor trip will not result in any gross core damage.

Table 1

List of Protective Trip Functions

1. High Nuclear Flux Reactor Trip
2. Overpower delta T
3. Overtemperature delta T
4. High pressurizer pressure
5. High pressurizer level
6. Steam/Feedwater Flow Mismatch
7. Low Steam Generator Water Level

No. 6 Startup Accident

The most severe core conditions which could result from a startup accident occur if one postulates the complete withdrawal of a RCC bank, while the plant is at hot zero power, without any subsequent action from the Reactor Protection System. A startup accident from hot zero power would be more severe than one from cold zero power because the higher initial temperature results in a higher overall heat transfer coefficient, a less negative water temperature coefficient, and an increased fuel thermal capacity.

A startup accident would, of course, be terminated almost immediately by the Reactor Protection System as discussed in WCAP-7306, "Reactor Protection System Diversity in Westinghouse Pressurized Water Reactors". A summary of Reactor Trip signals that would terminate the accident are given in Table I. Since the trip signals are both redundant and diverse, the assumption that the Reactor Protection System failed to trip the reactor is certainly not a creditable situation. Nevertheless, the startup accident analysis was performed assuming that all Reactor Trip signals generated by the Reactor Protection System are inactive.

The withdrawal of an RCC bank results in a positive reactivity insertion and is characterized by a fast rise in neutron flux. The flux rise continues until it is terminated by the reactivity effect of the negative Doppler fuel coefficient. The neutron flux then falls until a steady-state value is reached where the reactivity insertion is balanced by the reactivity effects of fuel temperature and moderator density. The neutron flux rise is accompanied by a rise in the thermal flux, which lags the neutron flux, and, of course, by an increase in both reactor coolant system temperature and pressure. If the feedwater flowrate is insufficient to match the steam flow, the steam generator tubes will eventually start to uncover causing the steam generator heat transfer capability to decrease, causing an increase in the reactor coolant system temperature and an insurge into the pressurizer. This increase in temperature will cause the nuclear power to decrease to a new steady state value because of the negative moderator density feedback.

The following assumptions were made in the analysis:

1. The plant was initially at hot zero power with nominal reactor coolant flow.

2. A total reactivity insertion of 1% was assumed. This is representative of the complete withdrawal of one RCC bank.

3. Reactivity insertion rate corresponding to the bank being withdrawn at maximum speed with average bank worth was assumed.

4. The moderator coefficient of reactivity was assumed to be a function of moderator density. Values were chosen so that a conservatively low amount of feedback resulted as typical of beginning of core life conditions.

5. The Doppler coefficient of reactivity was assumed to be a function of fuel temperature. Values were chosen so that the resultant amount of feedback was conservatively low.

6. The feedwater control system was assumed to be in manual control. This does not affect the initial transient response. However, since the feedwater flow is not able to match steam flow, the steam generator tubes eventually begin to uncover causing a decrease in heat transfer capability resulting in an increase in reactor coolant system temperature and an insurge into the pressurizer. The auxiliary feedwater pumps are assumed to be actuated on steam generator water level with the time delay on actuation conservatively assumed to be 60 seconds

Figure 14.7.6-1 shows the neutron flux response and the thermal flux response. The peak value of neutron flux is 55%; the initial steady-state value is 54%. It can be seen that the thermal flux lags the neutron flux and that the peak value and steady-state value coincide at 54%. At approximately 380 seconds, the steam generator tubes begin to uncover. This causes a decrease in power to a new final steady state value of 11%.

Figures 14.7.6-2 and 14.7.6-3 show, respectively, the reactor coolant system temperature and pressure responses. Steam relief through the secondary system safety valves allows the reactor coolant system to reach an initial steady-state. As previously noted, the effect of steam generator tube uncoverage can be seen at 380 seconds as an additional increase in temperature and pressure.

Figure 14.7.6-4 shows the DNB ratio as a function of time during the transient. It should be noted that the minimum DNB ratio reached during the transient is much higher than 1.30. Thus, a startup accident without a reactor trip will not result in core damage.

Table 1

<u>List of Protective Trip Functions</u>

1. Source Range High Neutron Flux Reactor Trip

2. Intermediate Range High Neutron Flux Reactor Trip

3. Power Range High Neutron Flux Reactor Trip
   a. Low Setting
   b. High Setting

4. Pressurizer High Pressure Reactor Trip

5. Low Steam Generator Water Level Trip

6. Steam/Feedwater Flow Mismatch Trip

No. 7 - Station Blackout

The discussion of the Station Blackout as presented in WCAP-7306, section 5.12, is applicable to the Indian Point 2 Station. Limiting core conditions for this accident have already been submitted in section 14.1.12 of the Indian Point 2 FSAR.

No. 8 - Startup of an Inactive Loop

The analysis already submitted in the Indian Point 2 FSAR, section 14.1.7, indicates that the core is adequately protected against this accident without a reactor trip.

No. 9 Accidental Depressurization of the Reactor Coolant System

The most severe core conditions that could result from a credible accidental depressurization of the Reactor Coolant System occur if one postulates the inadvertent opening of a pressurizer safety valve. Such an accident would, of course, be terminated almost immediately by the Reactor Protection System. A summary of Reactor Trip signals that would terminate the accident is given in Table 1.

Since these trip signals are both redundant and diverse, the assumption that the Reactor Protection System failed to trip the reactor is certainly not a credible situation. Nevertheless, the accidental depressurization of the Reactor Coolant System was performed assuming that all reactor trip signals generated by the Reactor Protection System are inactive. Additional conservatism was introduced by the following assumptions:

1. The initial reactor power, coolant temperatures, and pressure were at extreme values consistent with steady-state, full power operation, including errors for calibration and instrumentaton. This resulted in the minimum margin to DNB at the initiation of the accident.

2. The moderator coefficient of reactivity was assumed to be a function of moderator density. Values typical of beginning of life operation were chosen in order to provide a conservatively low amount of reactivity feedback due to chang in moderator temperature.

3. The Doppler coefficient of reactivity was assumed to be a function of fuel temperature. Valves were chosen so that the resultant amount of feedback was conservatively high to retard any power decreased due to the moderator reactivity feedback.

4. Automatic reactor control was assumed since this results in the maximum power level and, thus, minimum DNBR throughout the transient.

5. Reactor coolant pump cavitation is assumed to occur when the pressurizer pressure decreases to 1050 psia. It is conservatively represented by a flow coastdown, with a time constant of ~5 sec., at that time. This representation is conservative since its effect is to move the pump inlet temperature away from saturation, until the pump speed decreases to zero, whereas the pump cavitation effect would maintain the pump at its NPSH limit which would result in a greater pump flow rate.

Initially the accidental depressurization results in a rapidly decreasing Reactor Coolant System pressure until the system pressure reaches a value corresponding to the hot leg saturation pressure. At that time the pressure decrease is slowed considerably. The pressure continues to decrease, however, throughout the duration of the transient. The effect of the pressure decrease would be to decrease the nuclear power via the moderator density feedback but the Reactor Control System functions to maintain power essentially constant throughout the initial portion of the transient. After the control rods are fully withdrawn the moderator density feedback causes the power to decrease.

The accidental depressurization of the Reactor Coolant System was analyzed by means of a detailed digital simulation of the plant. The results of this simulation are shown in the attached figures. Figure 14.7.9-1 shows the Reactor Coolant System Pressure response. Initially, the pressure decreases at a rate of approximately 10 psi per second. After saturation in the hot leg occurs this decrease is slowed to approximately 3 psi per second. Finally, as the nuclear power decrease slows the pressure decrease rate slows until it settles out at 850 psia. Figures 14.7.9-2 and 14.7.9-3 show the T_{avg} and nuclear power responses. Initially because of the Reactor Control System very little change in T_{avg} occurs, and the nuclear power remains constant at approximately its initial value. At 110 seconds the control rods are fully withdrawn and the power and temperature begin to decrease. At 280 seconds the pumps are assumed to cavitate further increasing the rate of power and temperature decrease until steady state values of 35% power and 450°F are reached. Figure 14.7.9-4 shows the pressurizer water volume response. Note that the pressurizer fills with water at approximately 180 seconds and remains full until a steam bubble is reformed at approximately 410 seconds. Figure 14.7.9-5 shows the hot channel DNBR response. It shows that the hot channel DNBR never decreases below 1.08 throughout the transient. The DNBR for the hot assembly (not shown on the figure) will behave in a similar manner and never decreases below 1.17.

It should be noted that the analysis assumed that the hot channel factors remained constant at the design values during the accident while actually the core feedback effects would result in considerable flattening of the power distribution. This effect by itself would significantly increase the minimum DNB ratio. However, even ignoring this effect, the limit of 1.0 in the hot assembly is not exceeded at any time during the accident. Therefore, it is concluded that the Accidental Depressurization of the Reactor Coolant System with failure to initiate reactor trip will not result in any gross core damage.

Table 1

List of Protective Trip Functions

1. Low Pressure Reactor Trip
2. Pressurizer High Level Reactor Trip

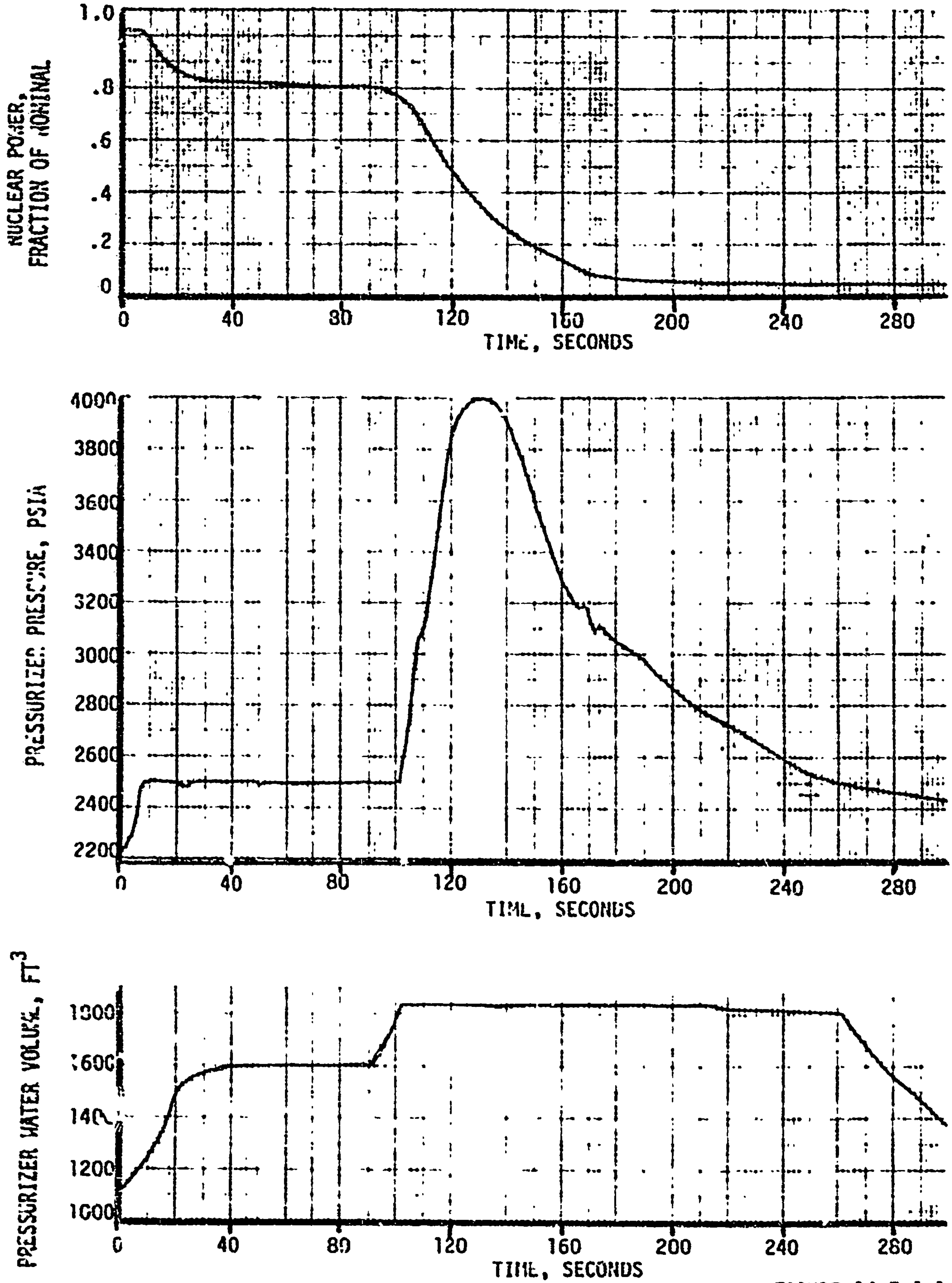


FIGURE 14.7.1-1
Supplement 12
7/70

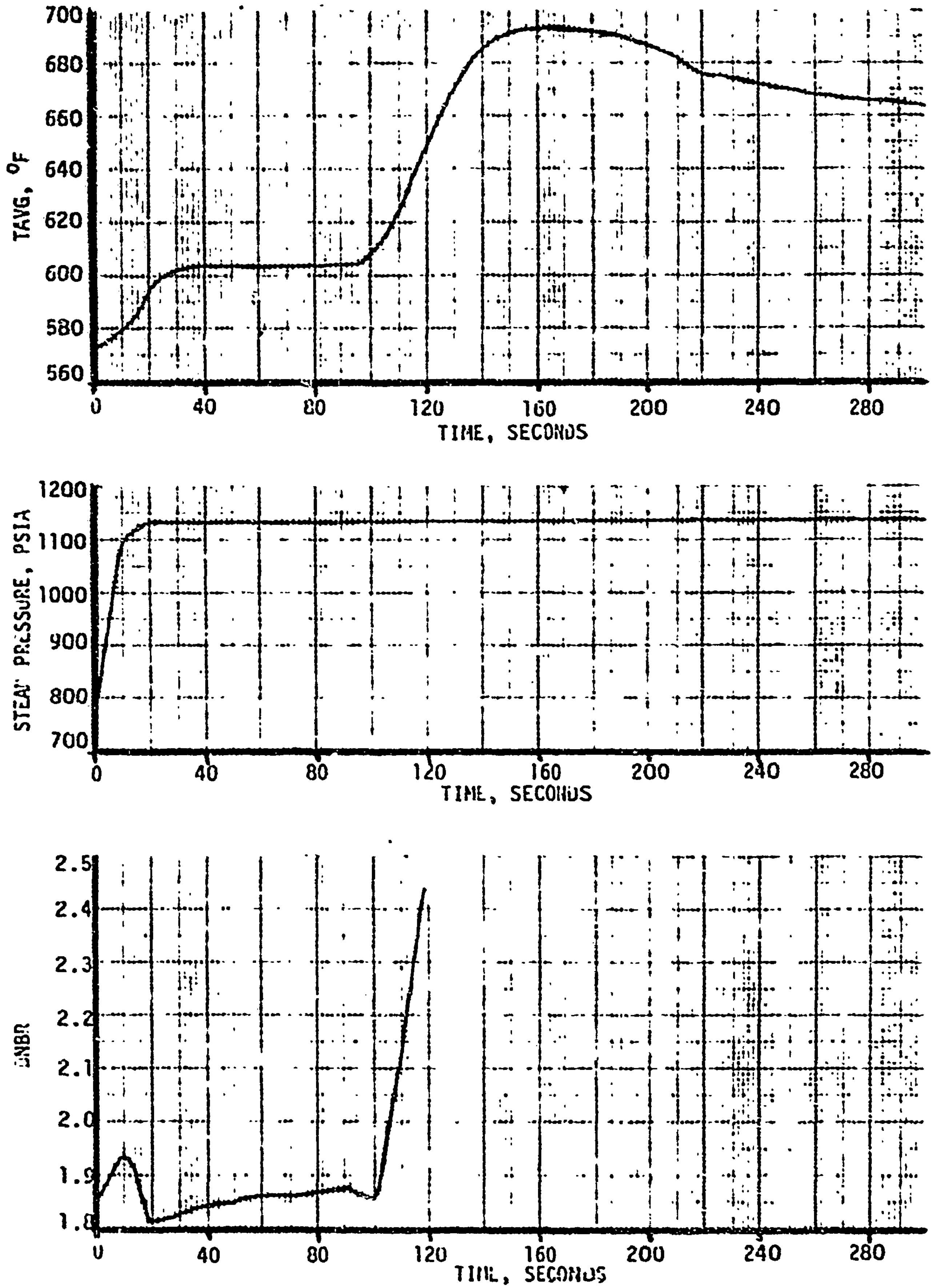


FIGURE 14.7.1-2
Supplement 1?
7/70

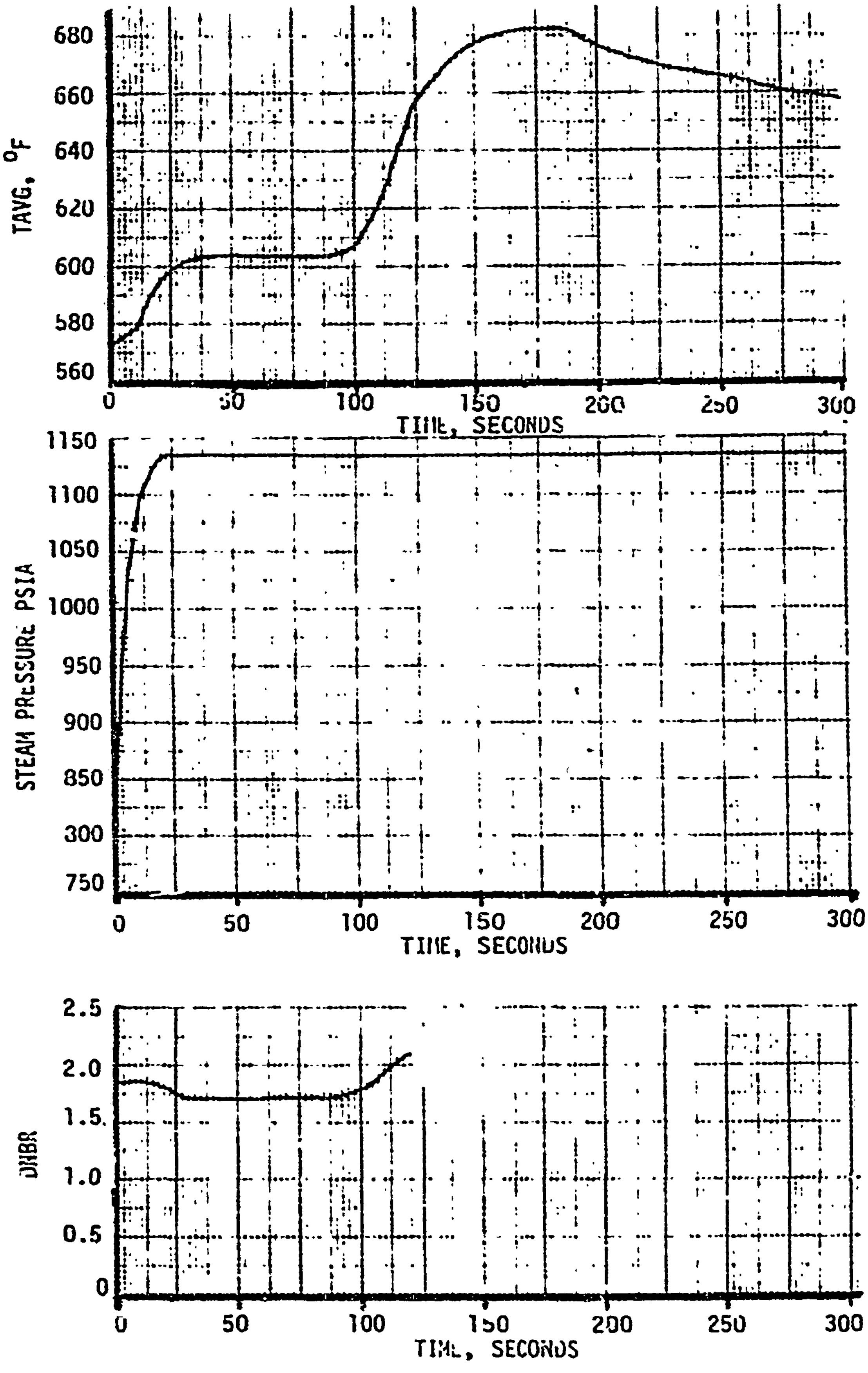


FIGURE 14.7.1-3
Supplement 12
7/70

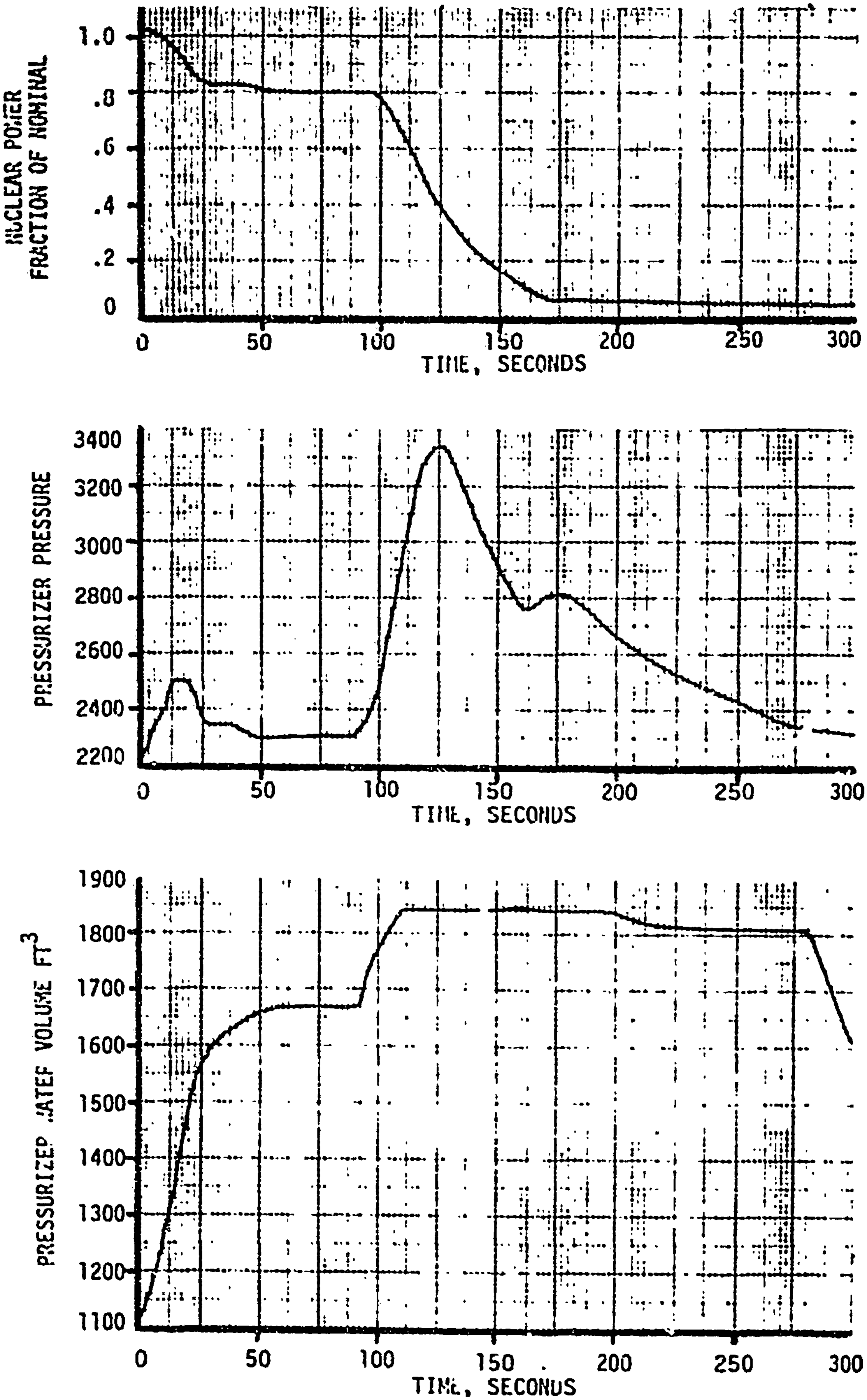


FIGURE 14.7.1-4
Supplement 12
7/70

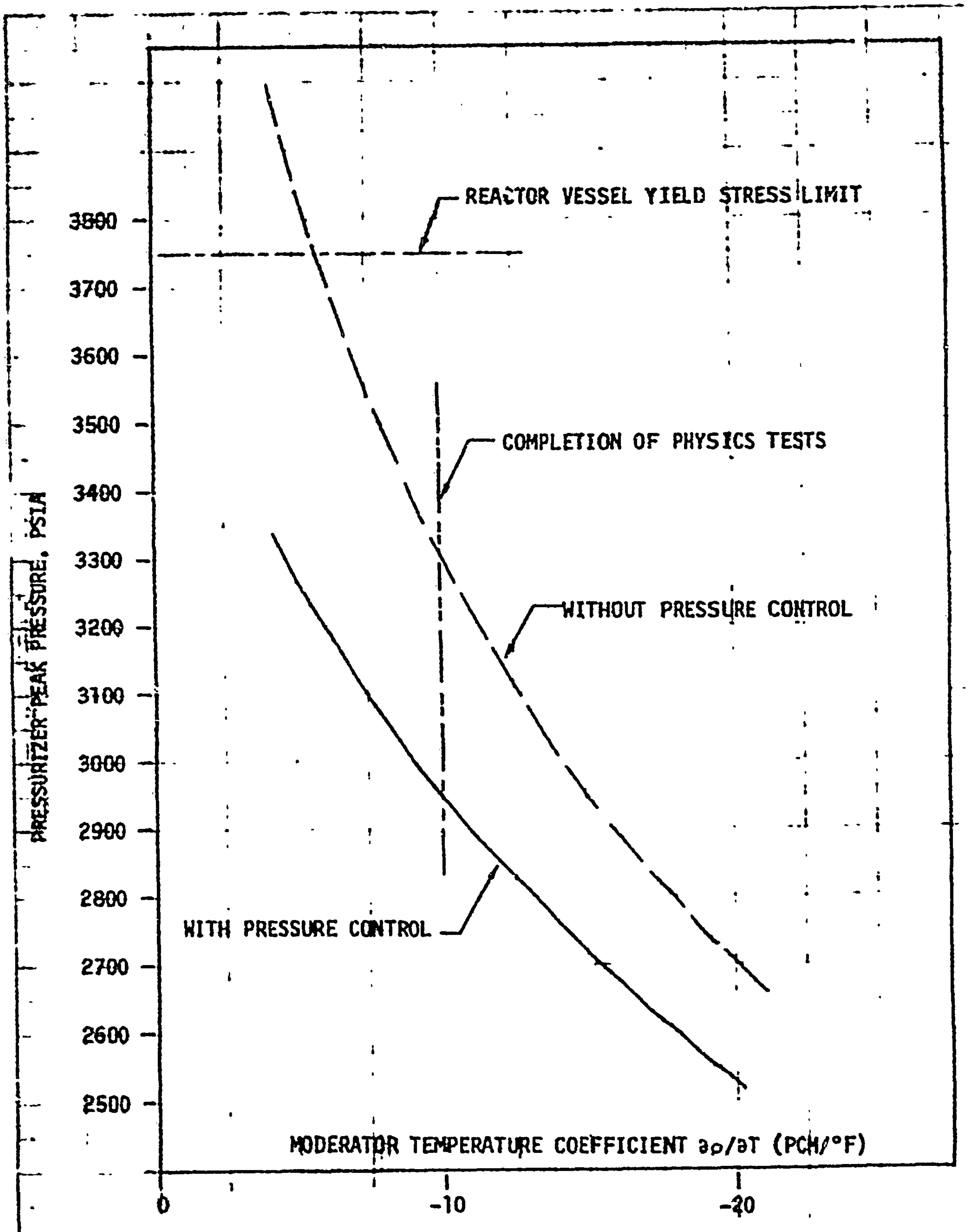


Figure 14.7.1-5
Supplement 12
7/70

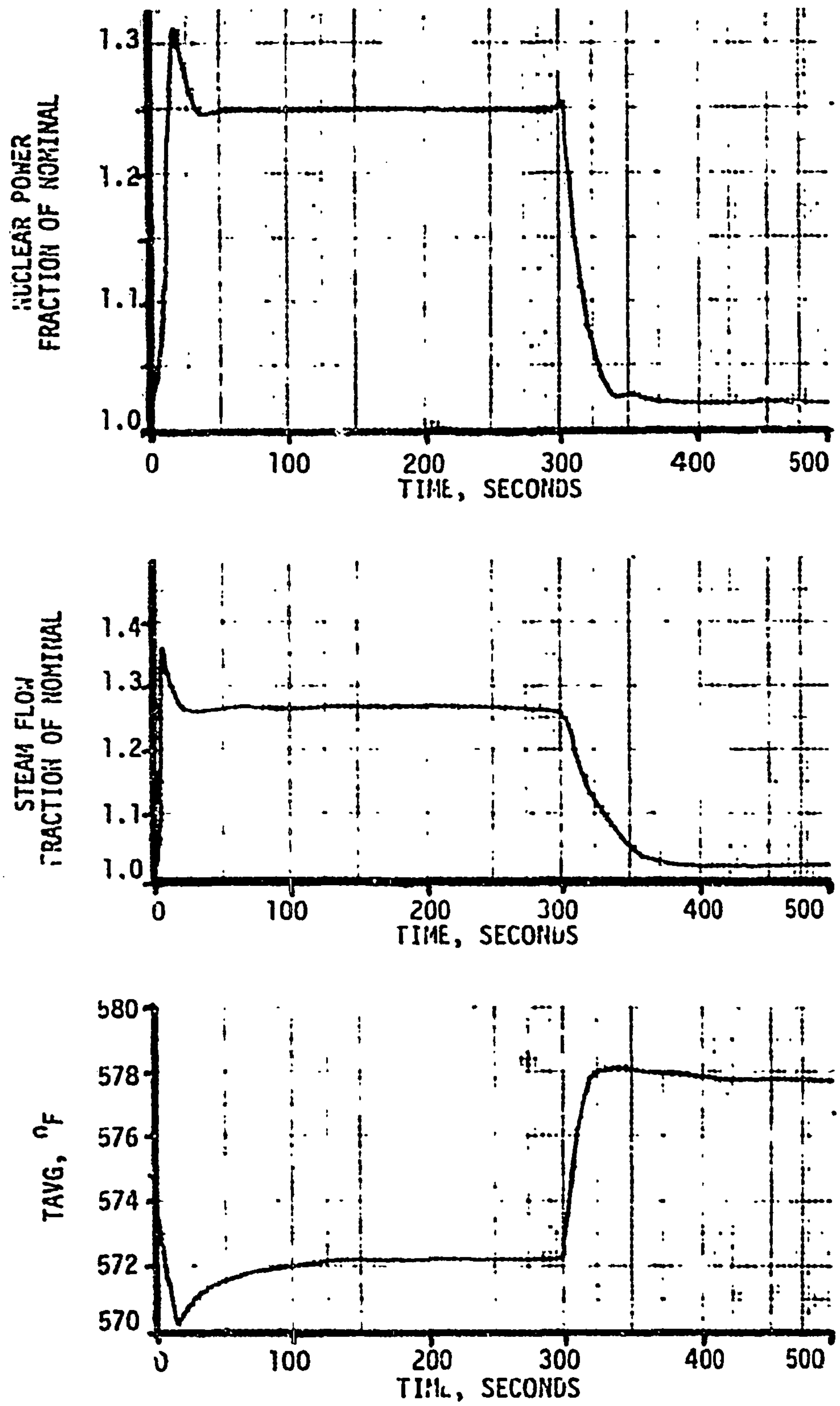


FIGURE 14.7.2-1
Supplement 12
7/70

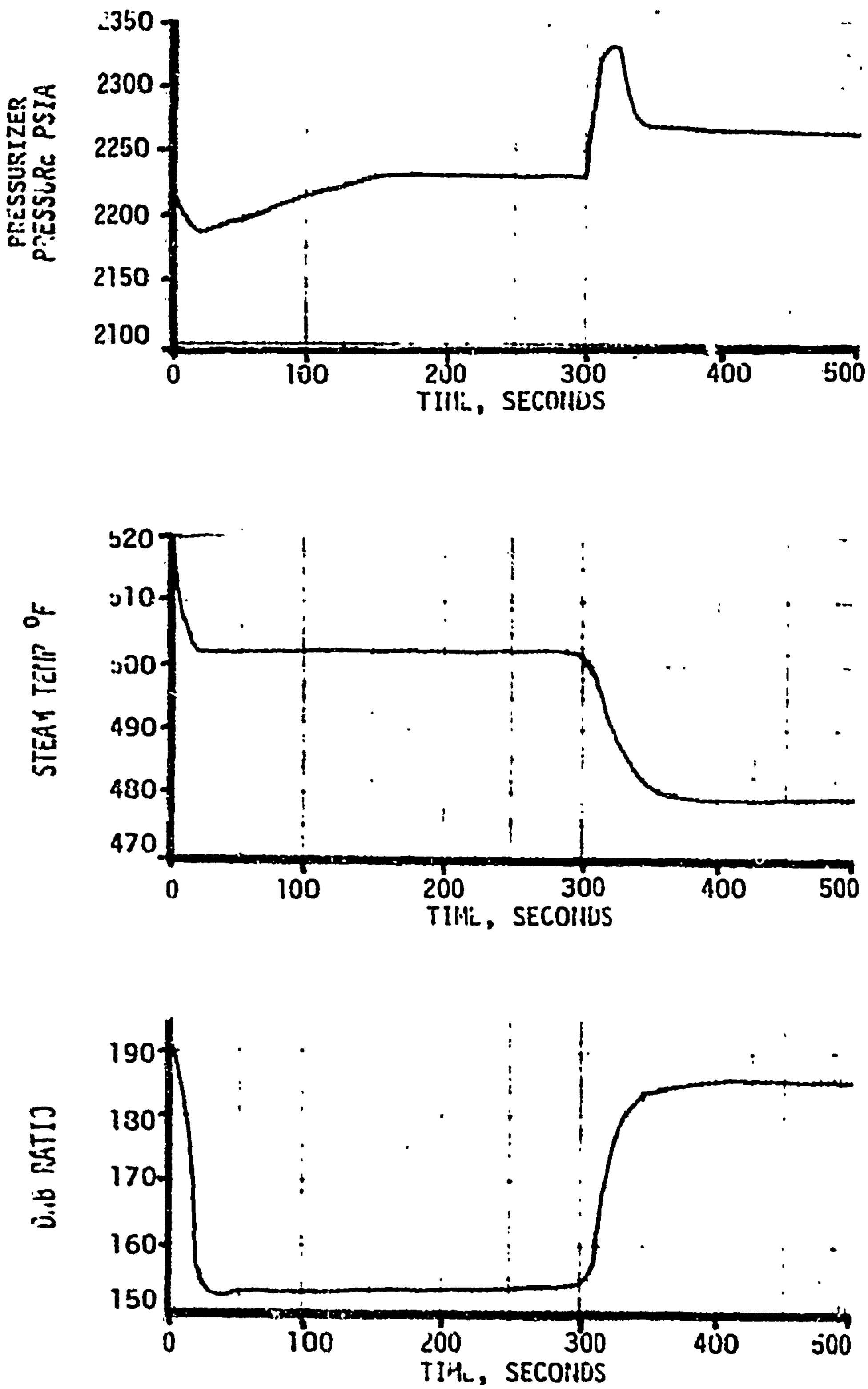
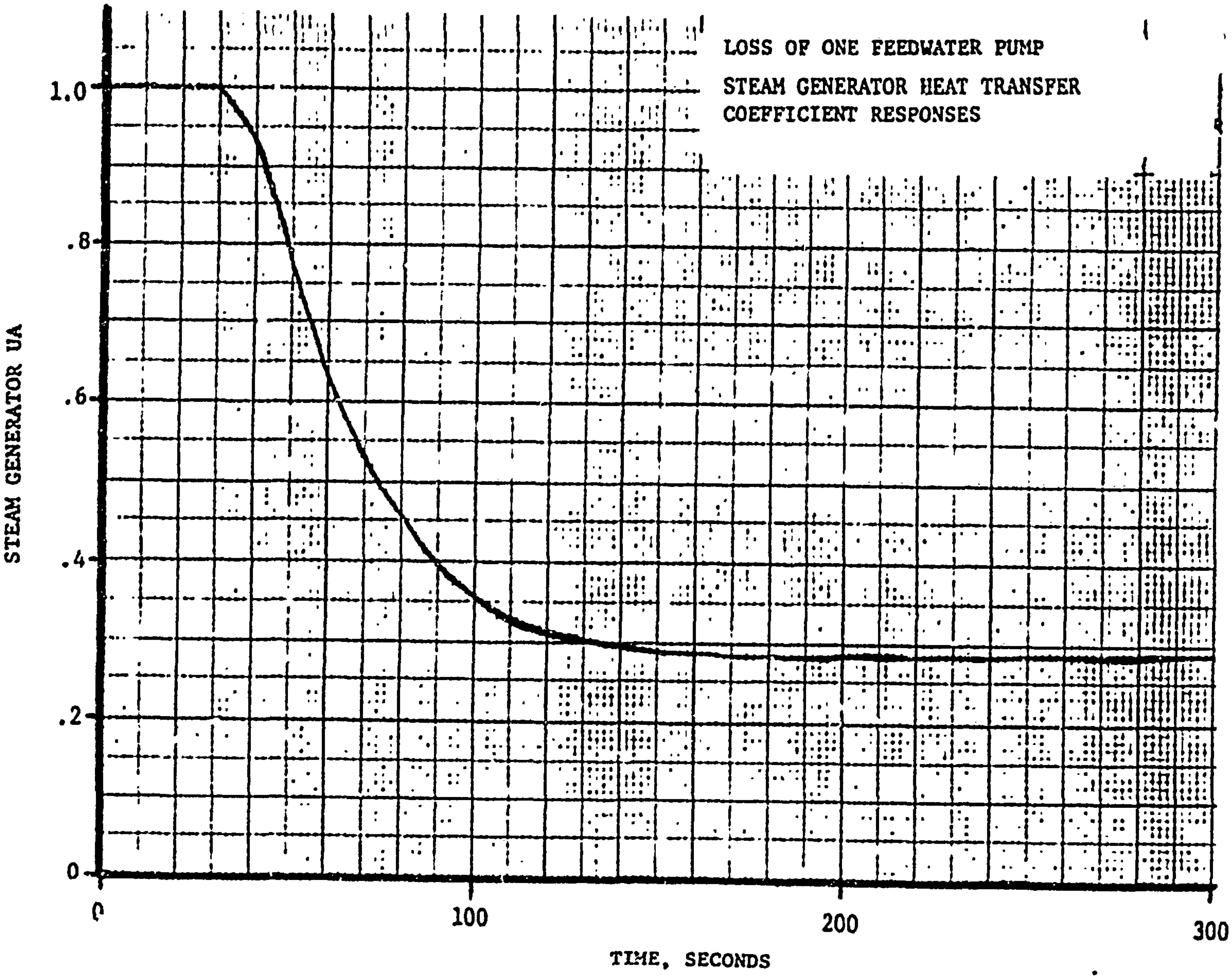


FIGURE 14.7.2-2
Supplement 12
7/70

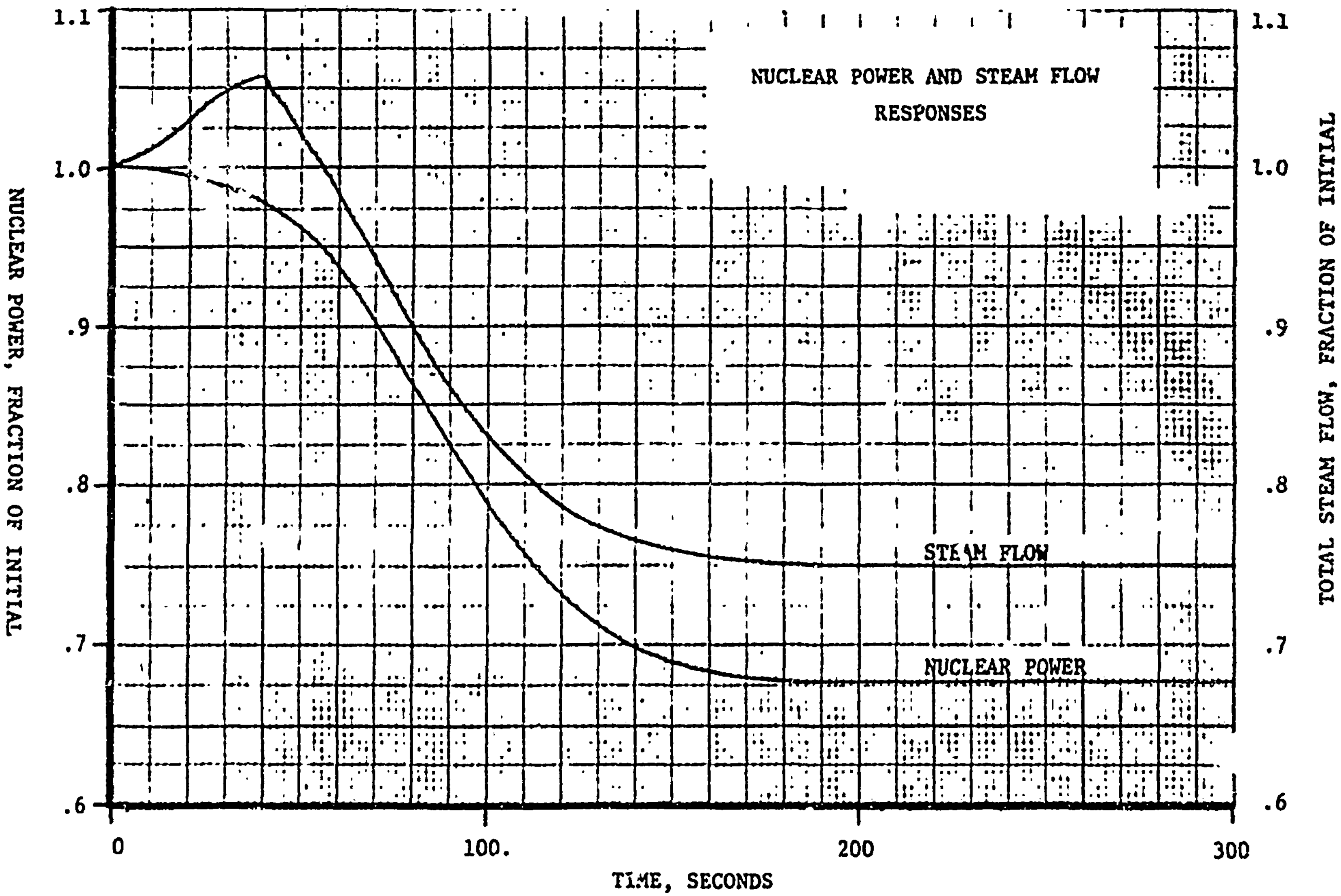
INDIAN POINT ACCIDENT ANALYSIS WITH FAILURE TO
INITIATE REACTOR TRIP



Supplement 7
3/70

FIGURE 14.7.3-1

INDIAN POINT ACCIDENT ANALYSIS WITH FAILURE TO
INITIATE REACTOR TRIP - LOSS OF ONE FEEDWATER PUMP



Supplement 7
3/70

FIGURE 14.7.3-2

INDIAN POINT ACCIDENT ANALYSIS WITH FAILURE TO INITIATE REACTOR TRIP - LOSS OF ONE FEEDWATER PUMP - PRIMARY AND SECONDARY SYSTEM PRESSURE RESPONSES

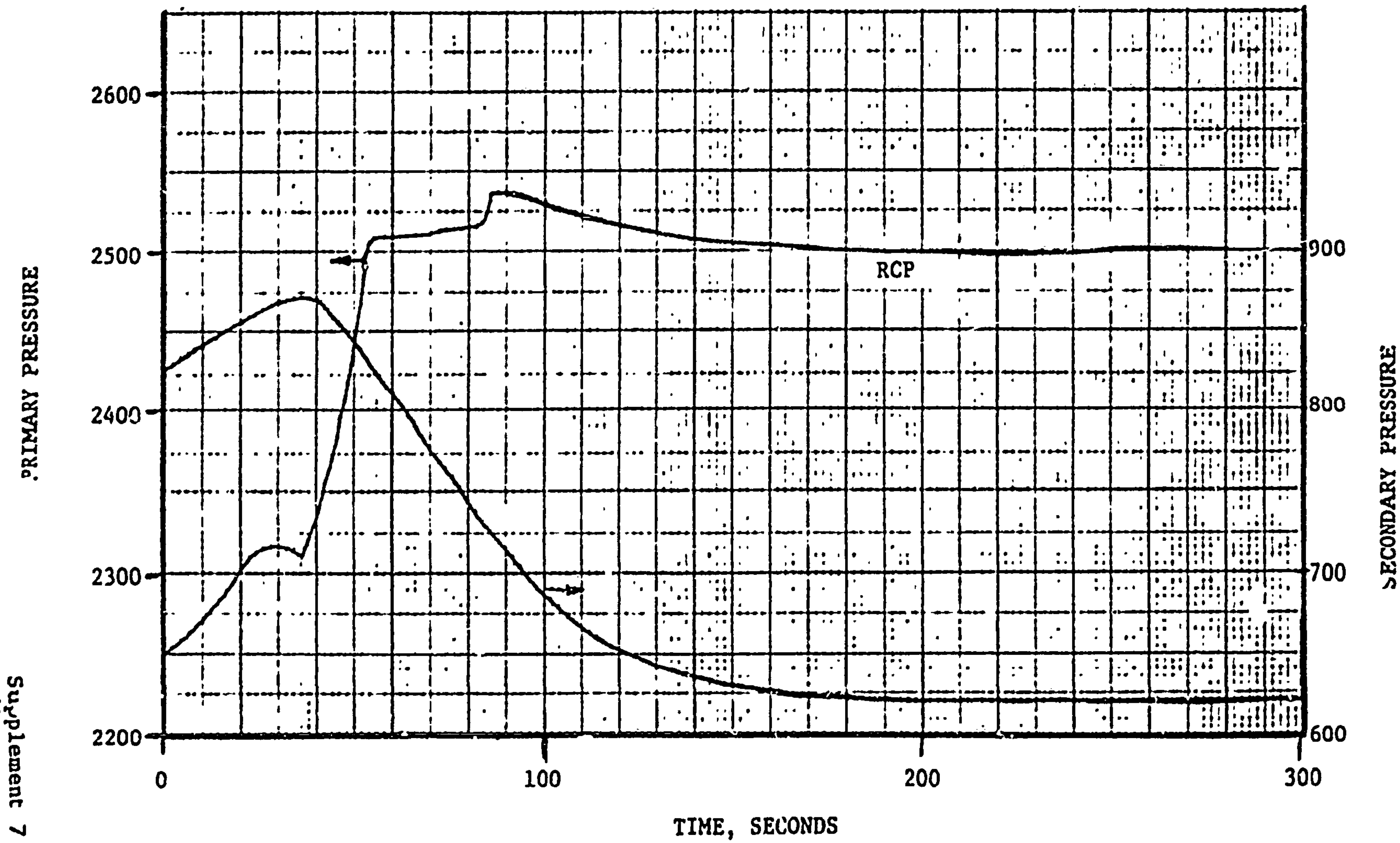


FIGURE 14.7.3-3

Supplement 7
3/70

INDIAN POINT ACCIDENT ANALYSIS WITH FAILURE TO
INITIATE REACTOR TRIP LOSS OF ONE FEEDWATER PUMP

T_{AVG} AND TIN (CORE INLET TEMPERATURE) RESPONSES

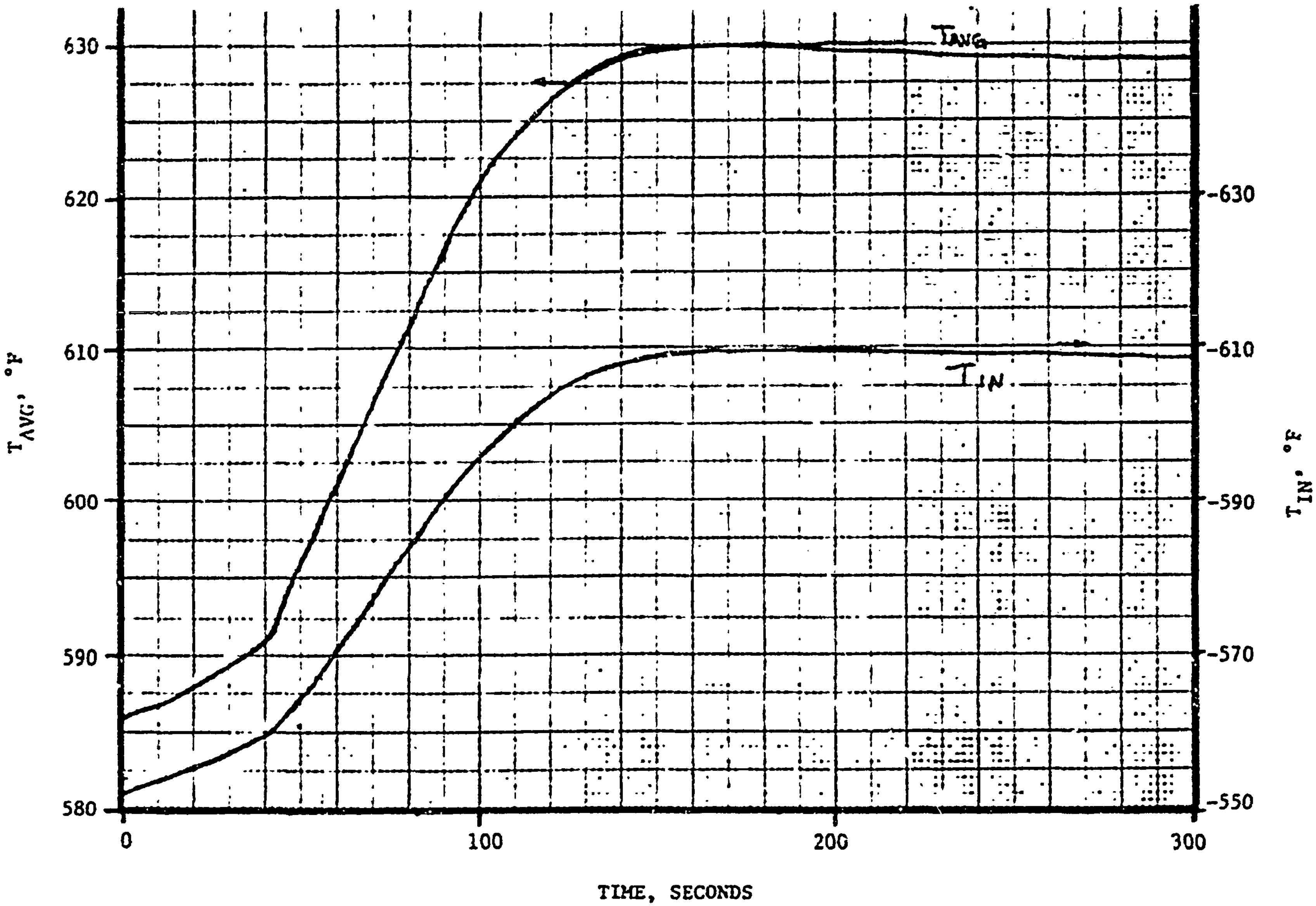
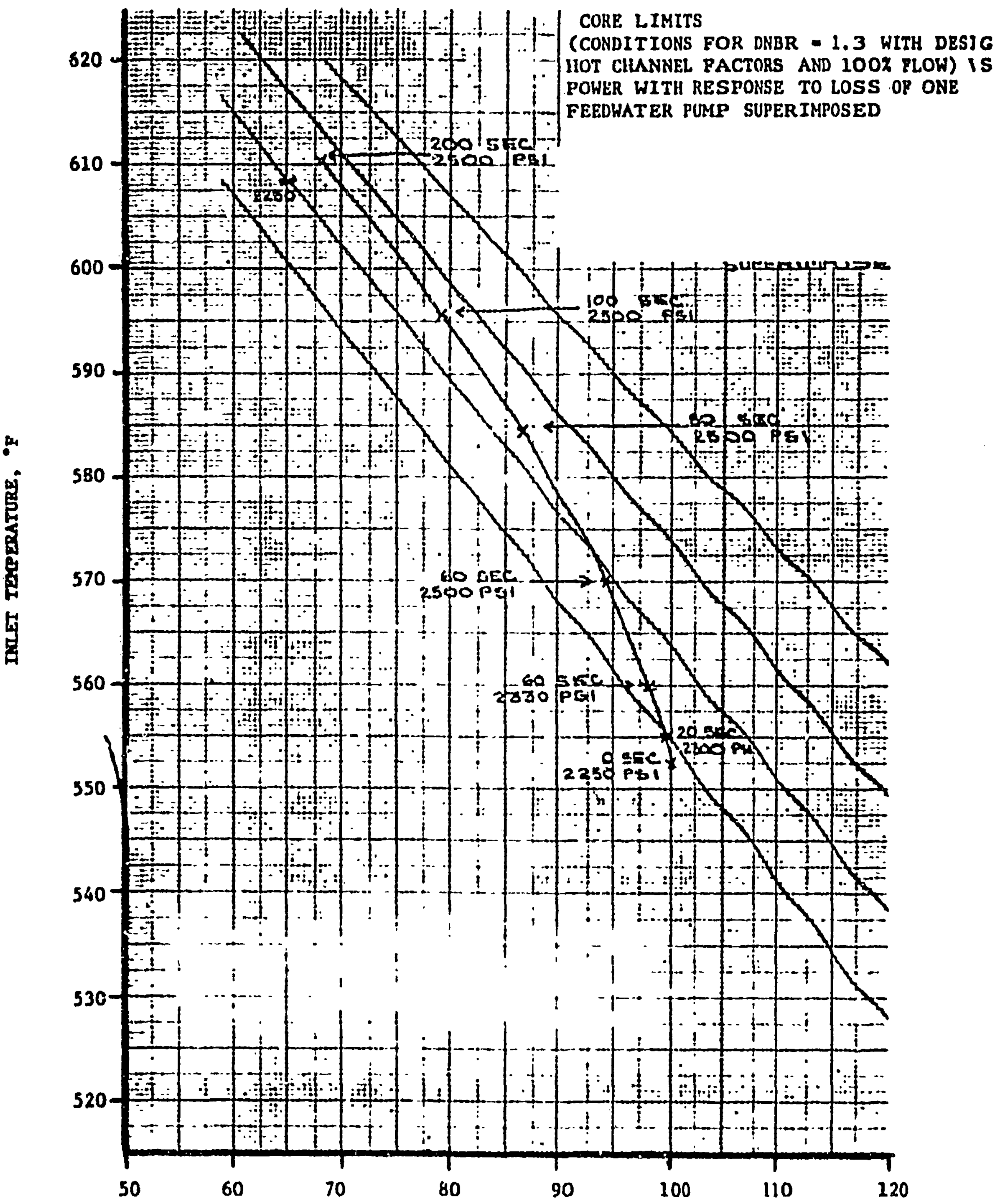


FIGURE 14.7.3-4

Supplement 7
3/70

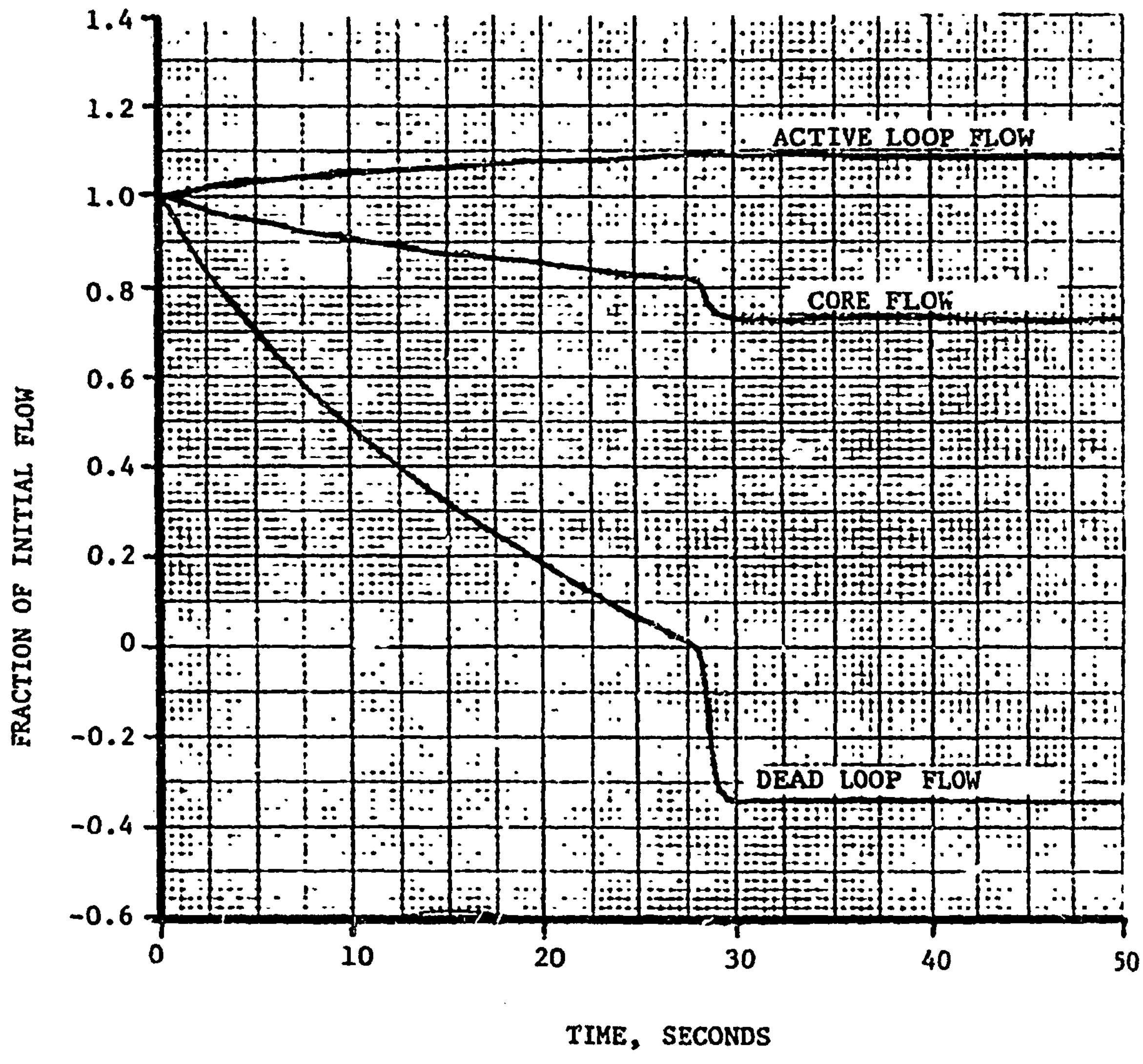
INDIAN POINT ACCIDENT ANALYSIS WITH FAILURE TO INITIATE REACTOR TRIP LOSS OF ONE FEEDWATER PUMP



Supplement 7
3/70

FIGURE 14.7.3-5

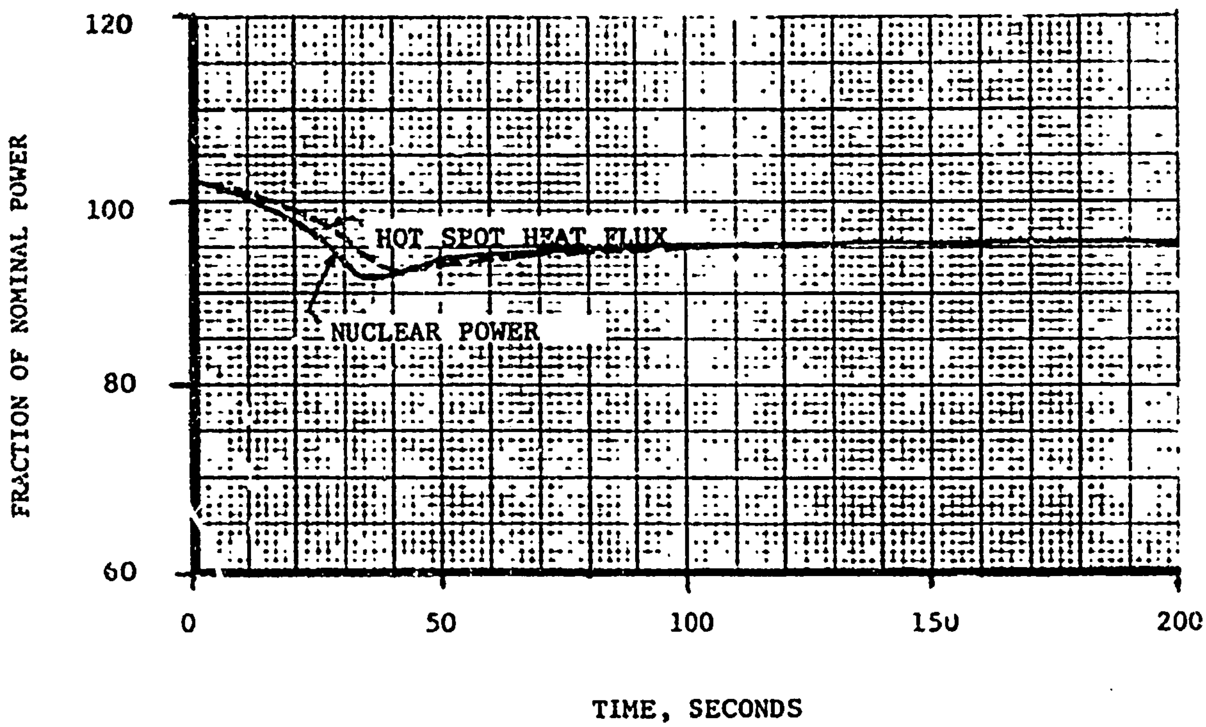
ONE PUMP LOSS OF FLOW COASTDOWN



Supplement 7
3/70

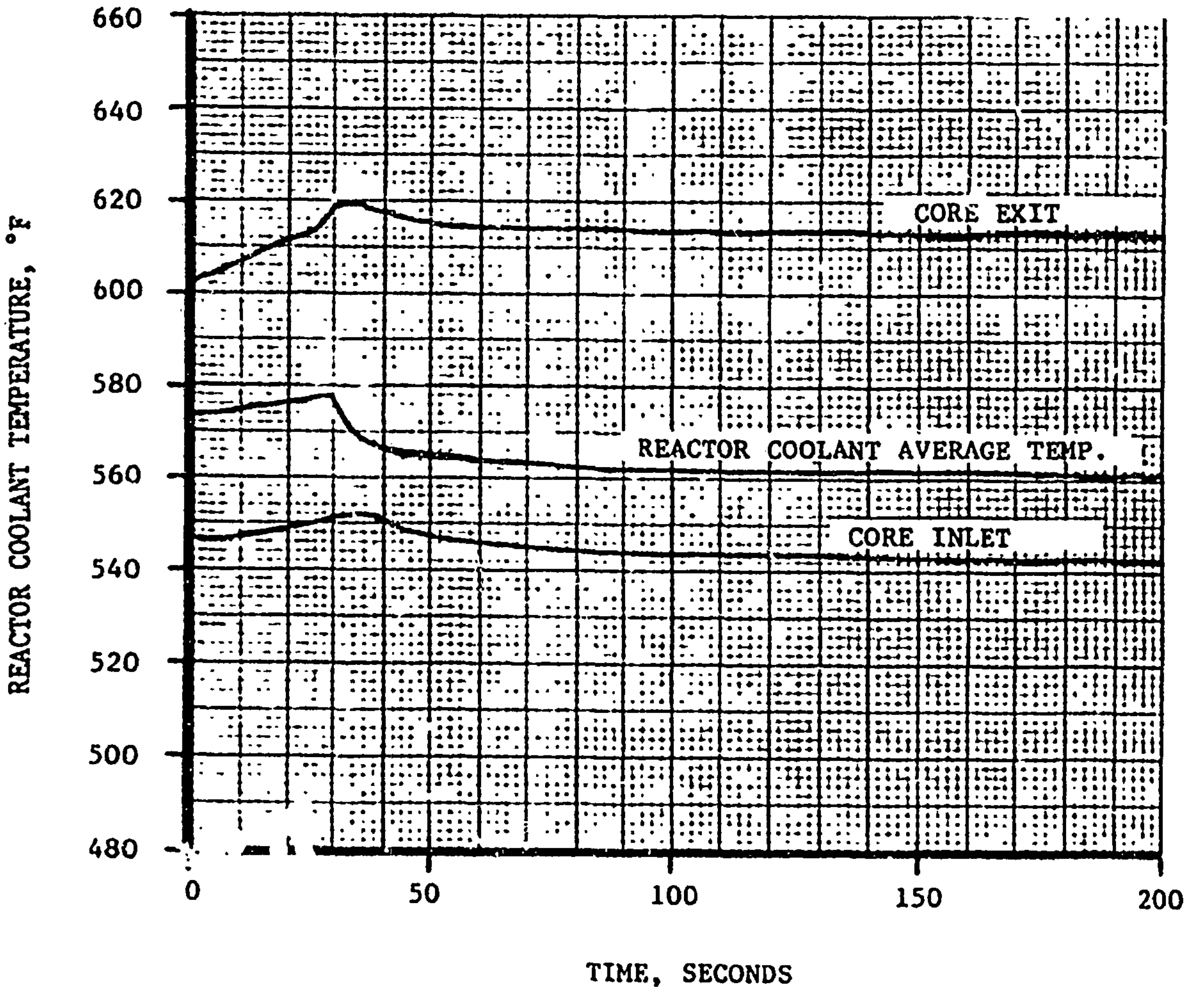
FIGURE 14.7.4-1

ONE PUMP LOSS OF FLOW WITHOUT ANY ACTION FROM EITHER REACTOR PROTECTION OR CONTROL SYSTEM



Supplement 7
3/70
FIGURE 14.7.4-2

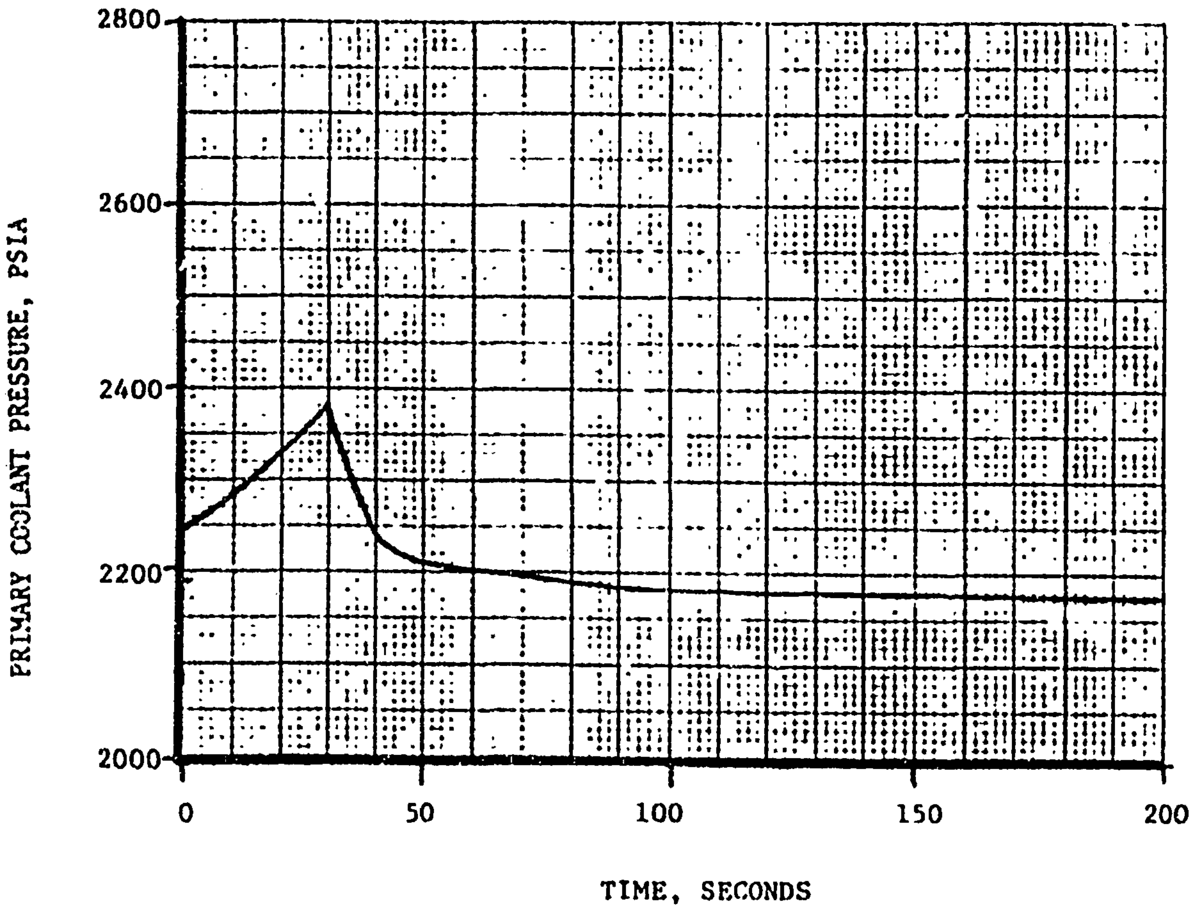
ONE PUMP LOSS OF FLOW WITHOUT ANY ACTION
FROM EITHER REACTOR PROTECTION OR CONTROL
SYSTEM



Supplement 7
3/70

FIGURE 14.7.4-3

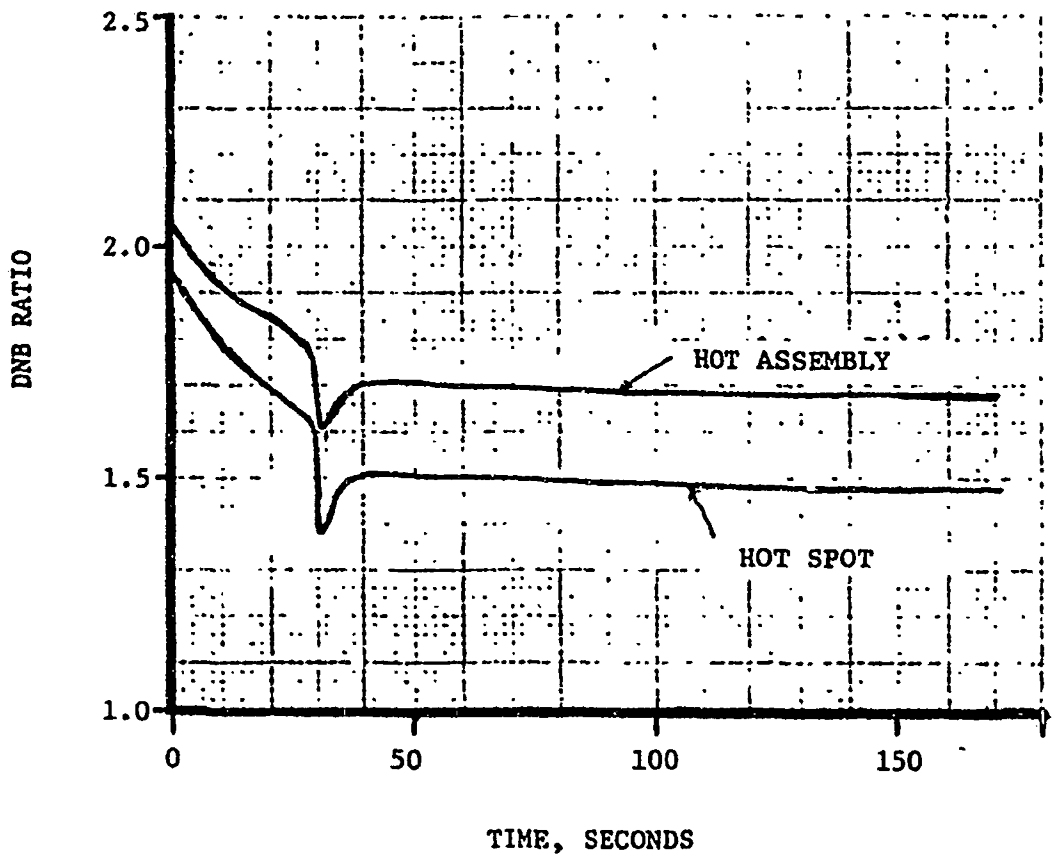
ONE PUMP LOSS OF FLOW WITHOUT ANY ACTION
FROM EITHER REACTOR PROTECTION OR CONTROL
SYSTEM



Supplement 7
3/70

FIGURE 14.7.4-4

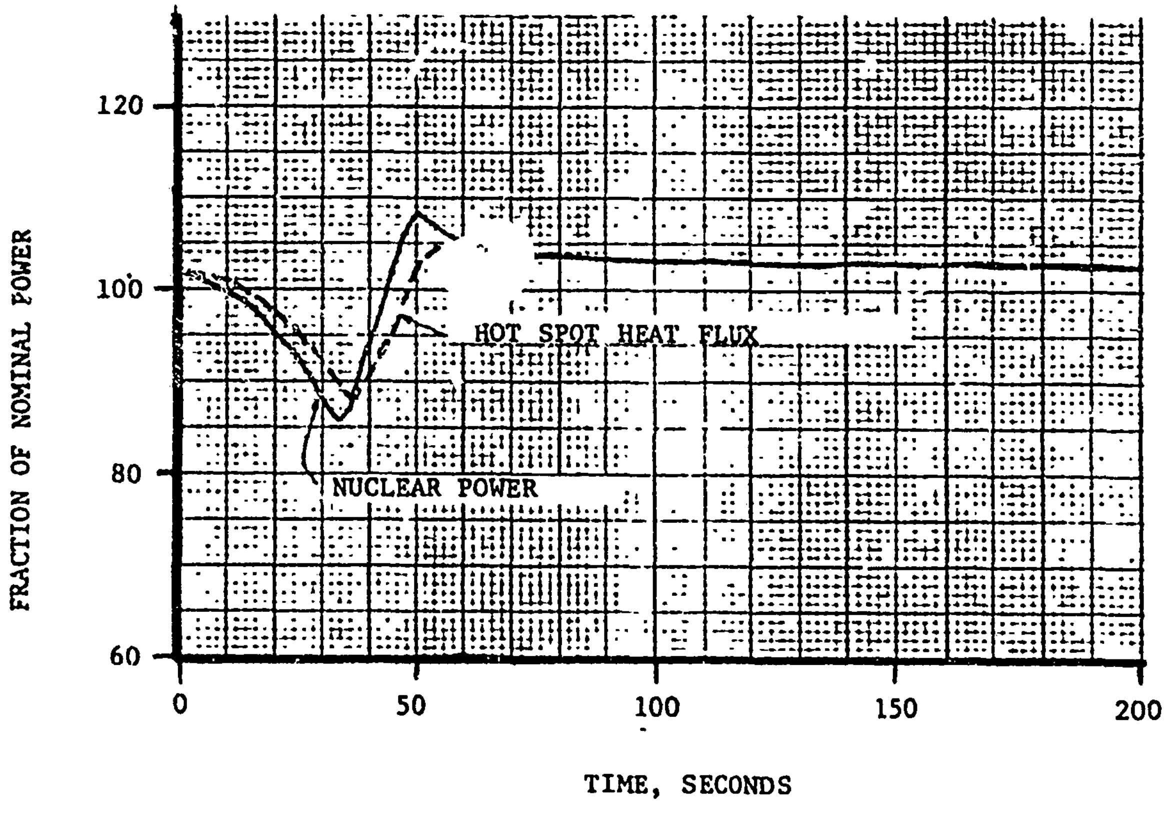
ONE PUMP LOSS OF FLOW WITHOUT ANY ACTION
FROM EITHER REACTOR PROTECTION OR CONTROL
SYSTEM



Supplement 7
3/70

FIGURE 14.7.4-5

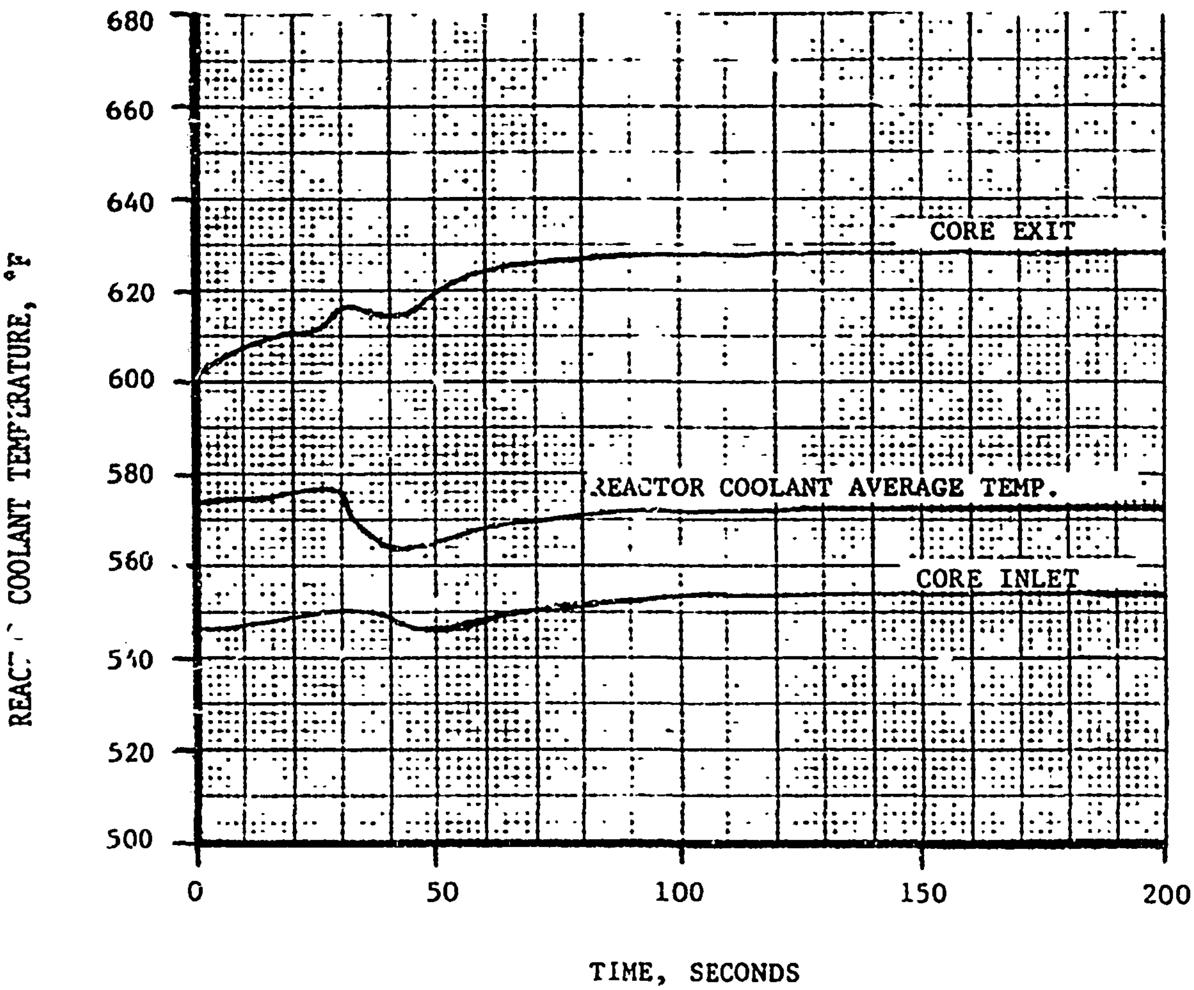
ONE PUMP LOSS OF FLOW WITH AUTOMATIC ROD CONTROL, WITH FAILURE TO TRIP REACTOR



Supplement 7
3/70

FIGURE 14.7.4-6

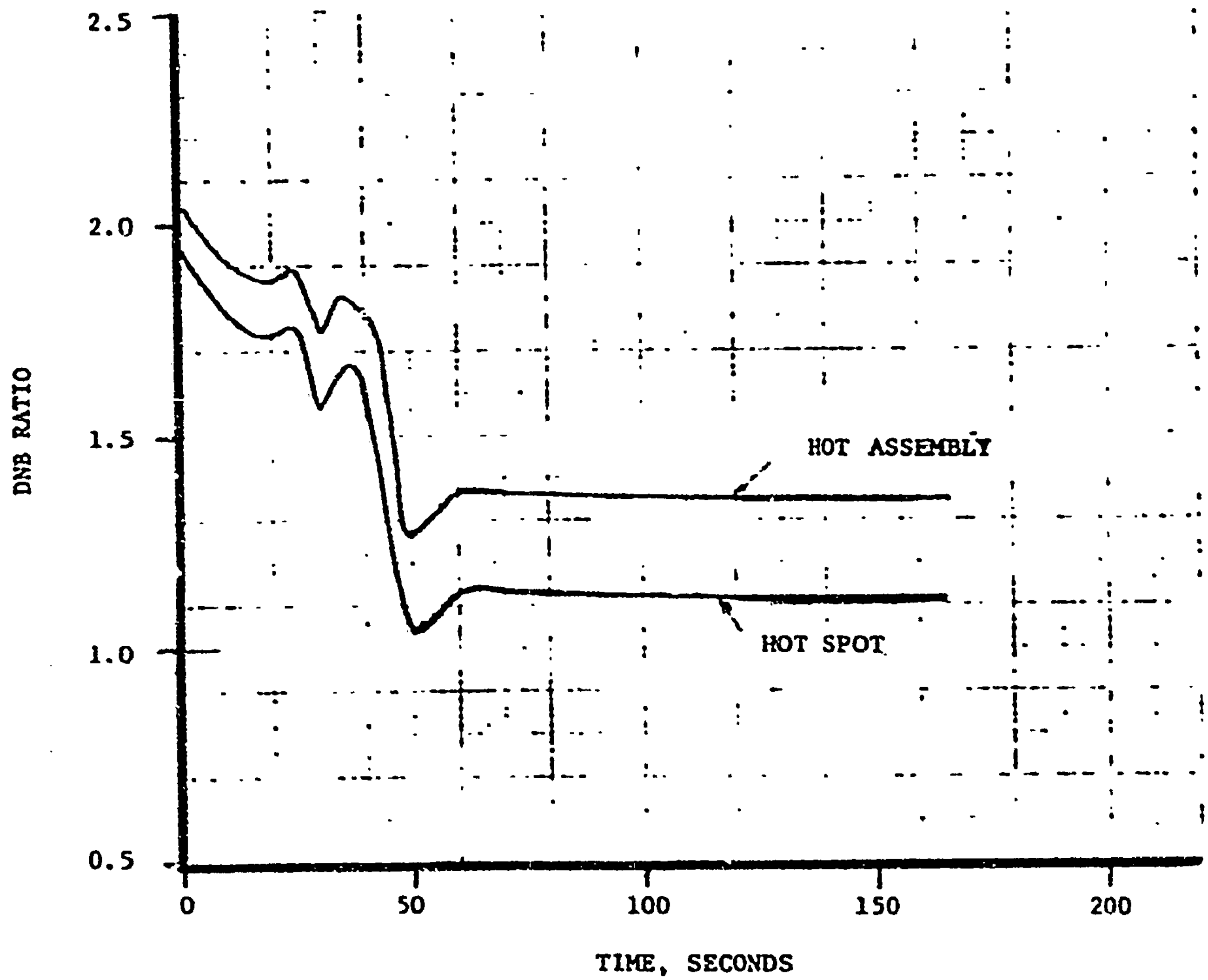
ONE PUMP LOSS OF FLOW WITH AUTOMATIC ROD CONTROL WITH FAILURE TO TRIP REACTOR



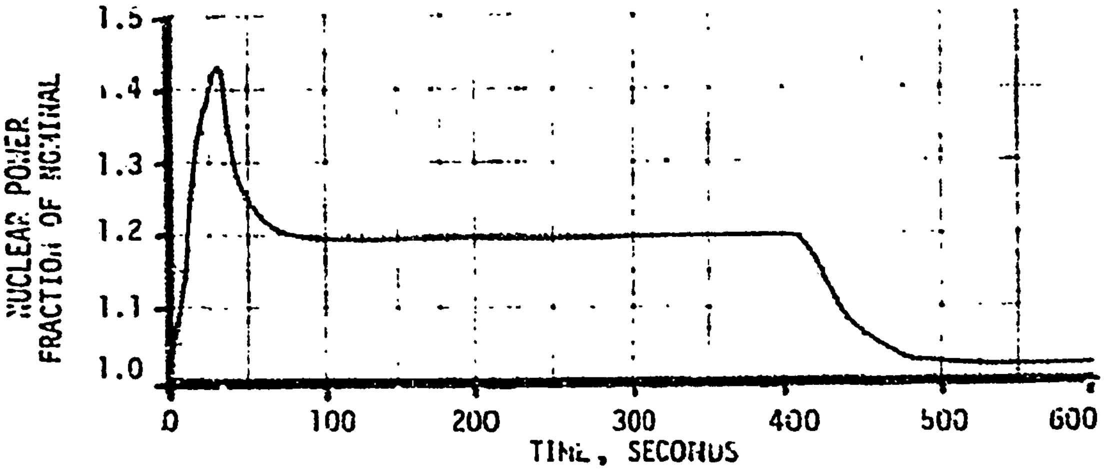
Supplement 7
3/70

FIGURE 14.7.4-7

ONE PUMP LOSS OF FLOW WITH AUTOMATIC ROD CONTROL, WITH FAILURE TO TRIP REACTOR



Supplement 7
3/70
FIGURE 14.7:4-8



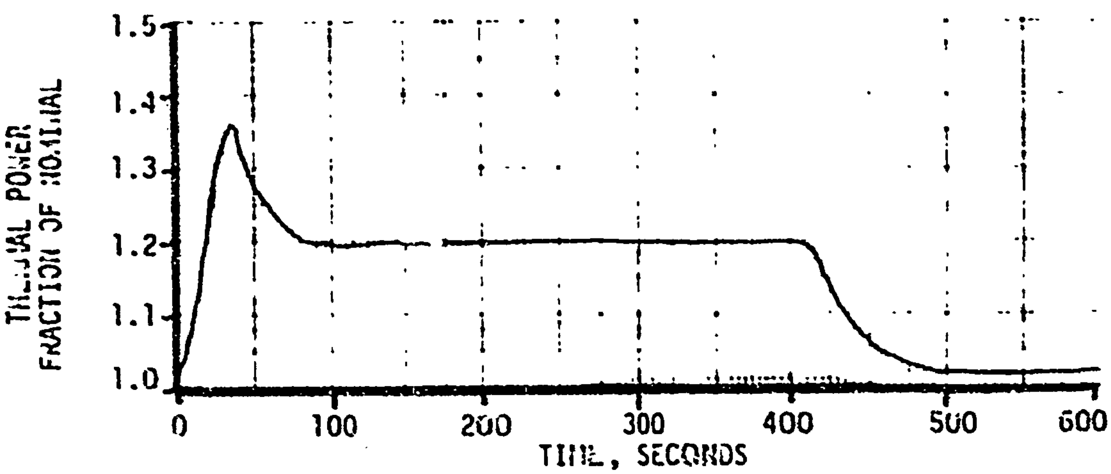


FIGURE 14.7.5-1
Supplement 12
7/70

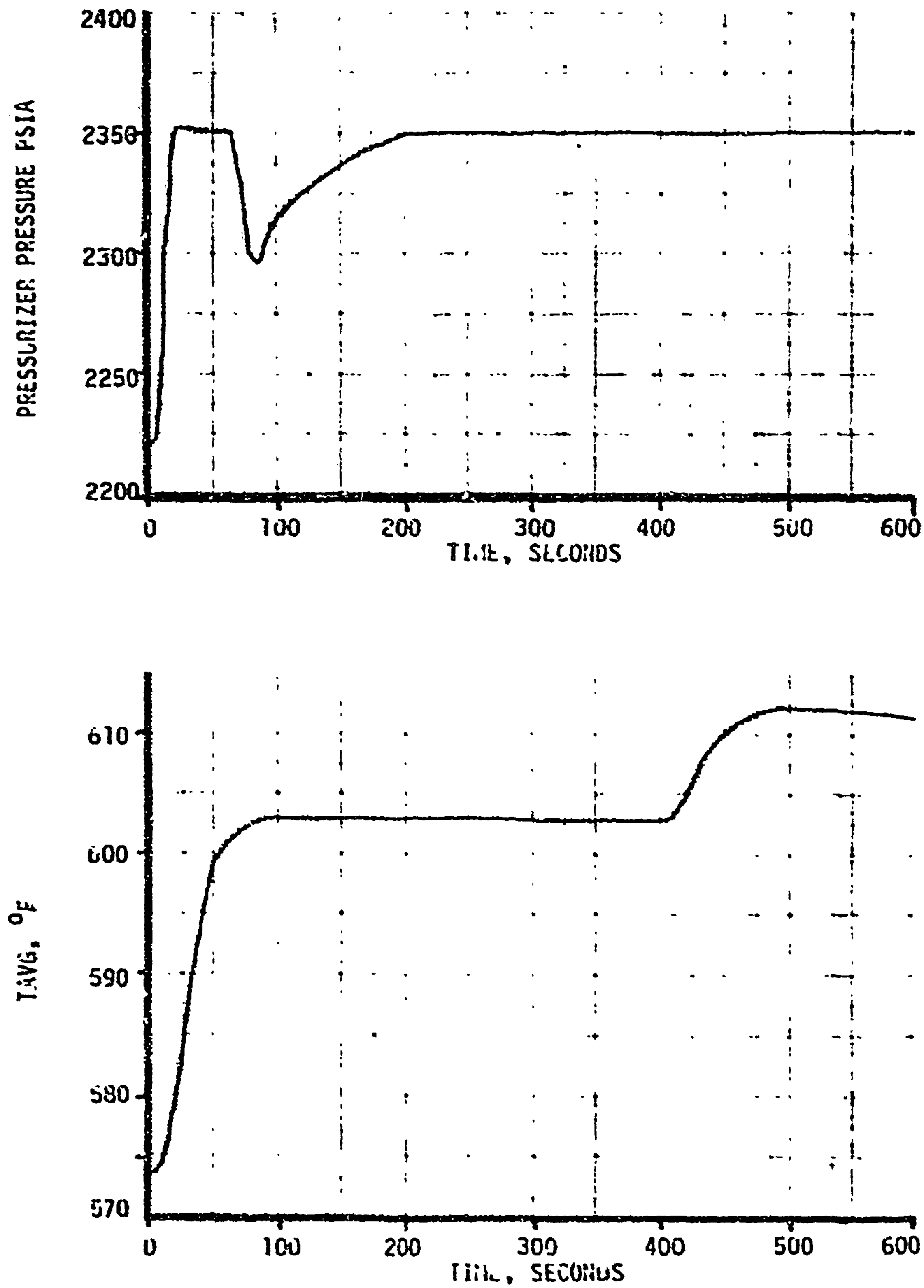


FIGURE 14.7.5-2
Supplement 12
7/70

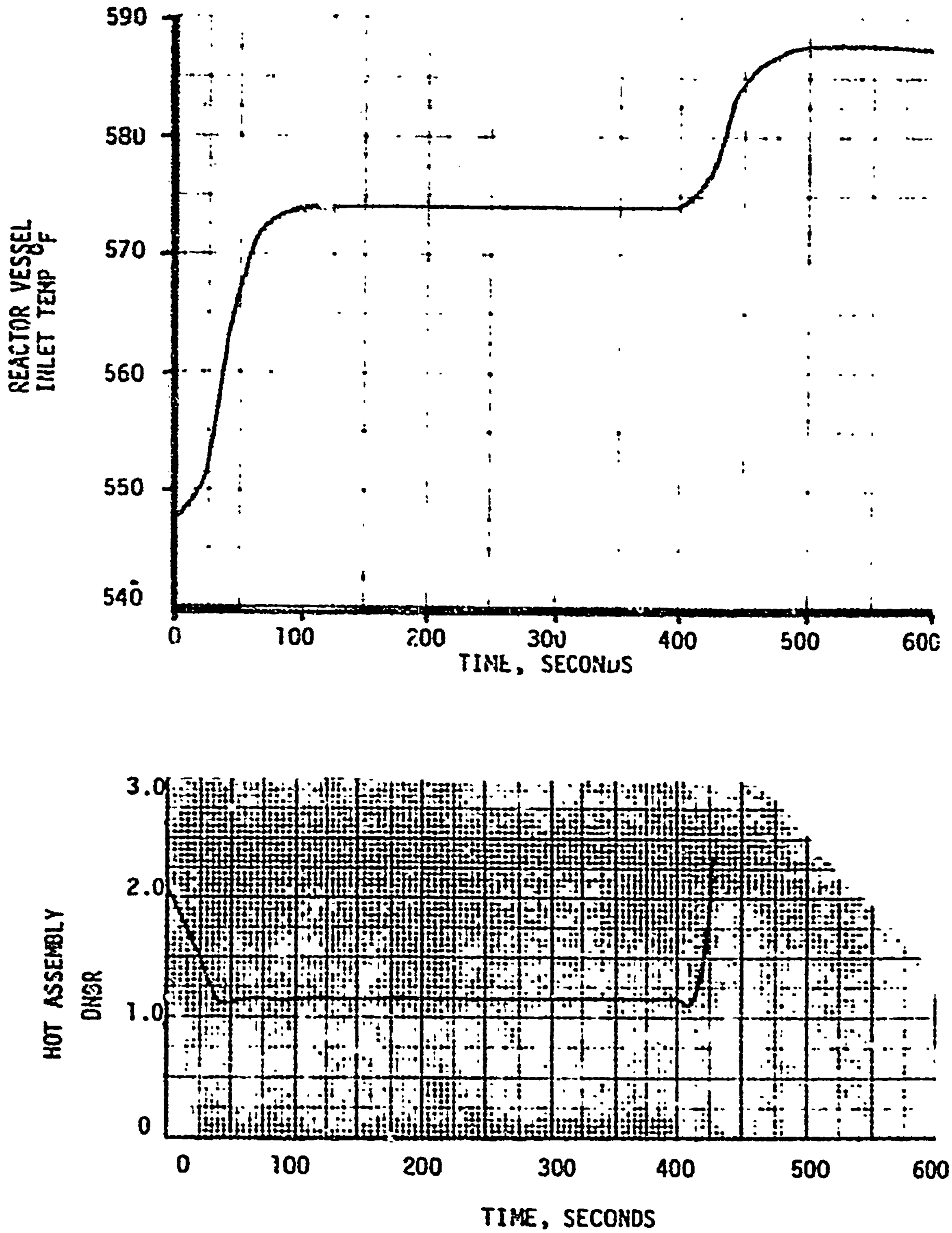
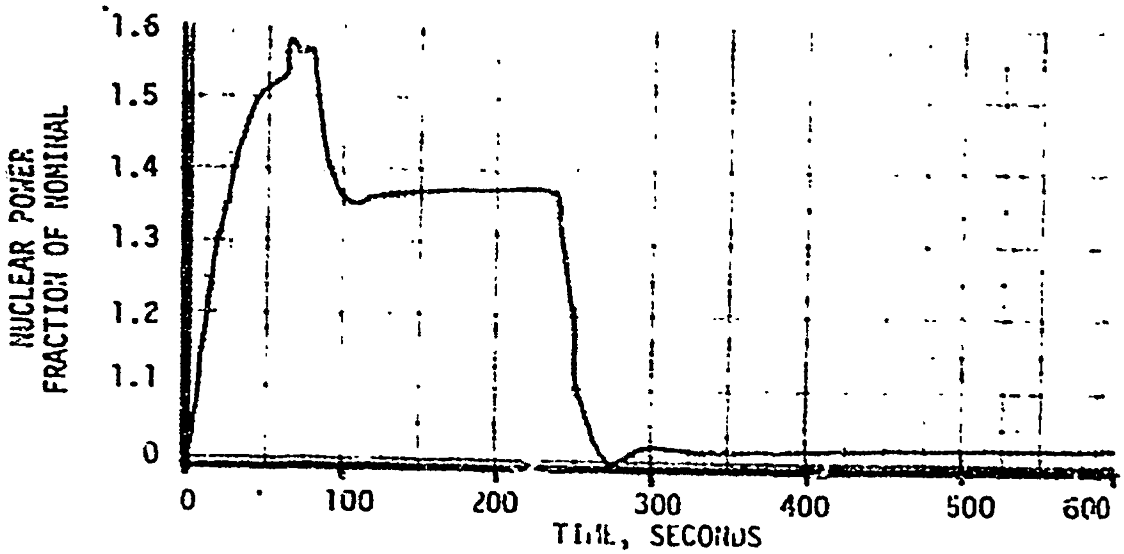


FIGURE 14.7.5-3

Supplement 12
7/70



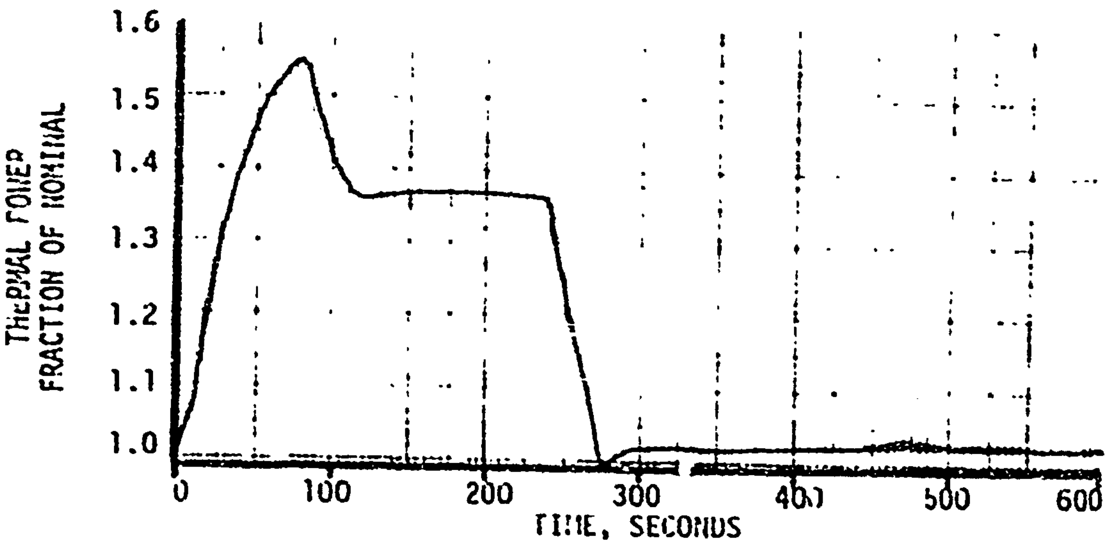


FIGURE 14.7.5-4
Supplement 12
7/70

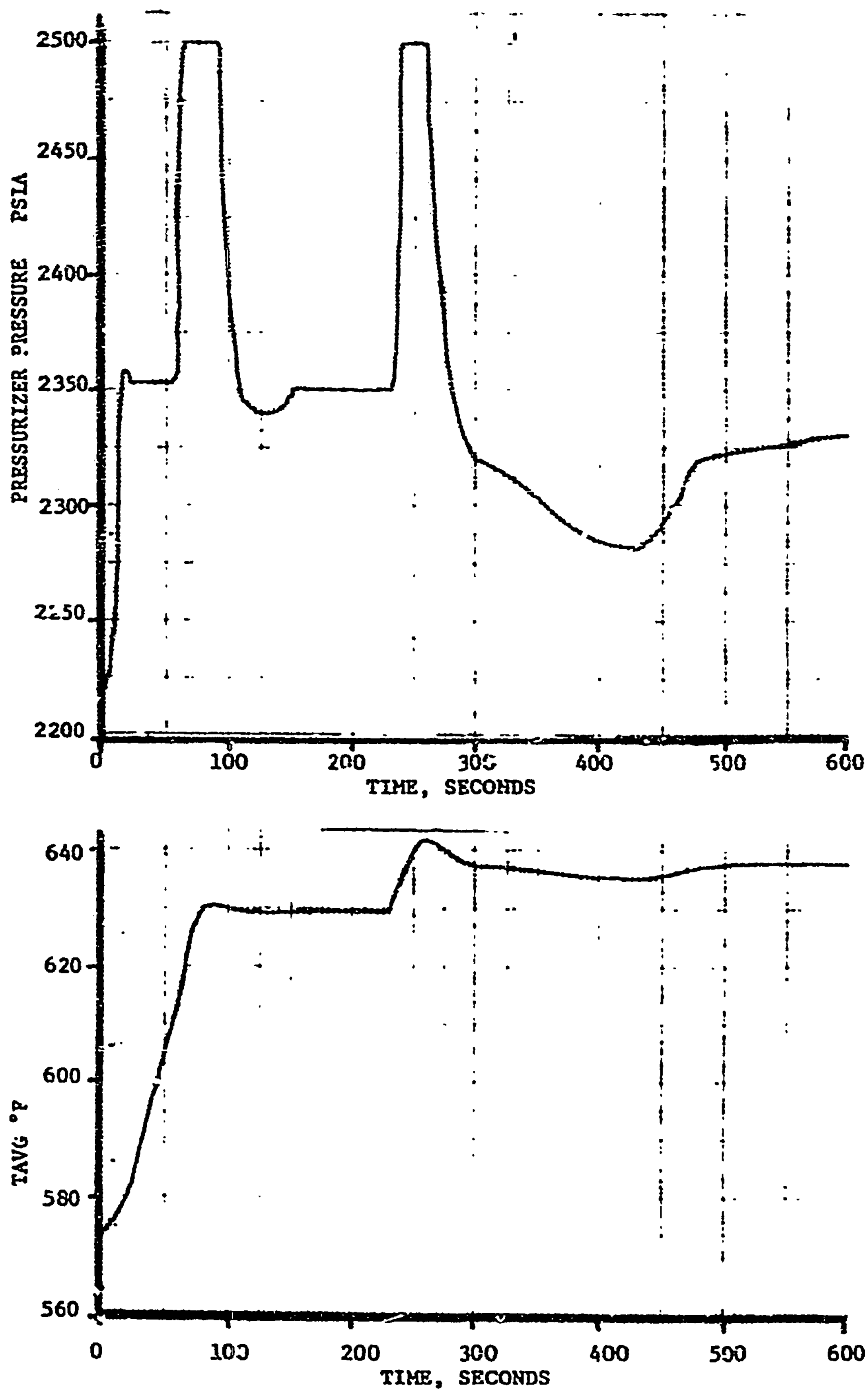


FIGURE 14.7.5-5
Supplement 12
7/70

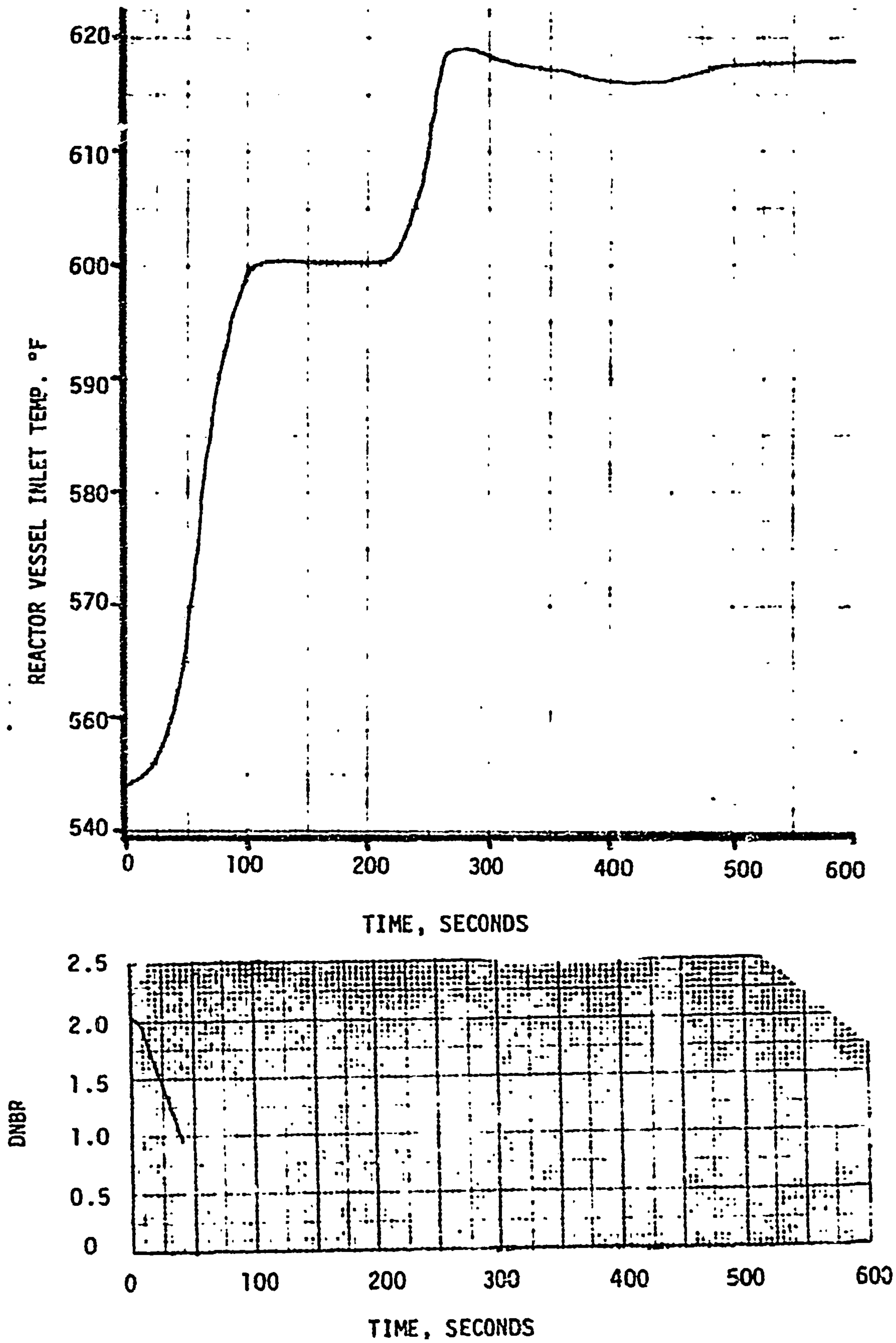


FIGURE 14.7.5-6

Supplement 12
7/70

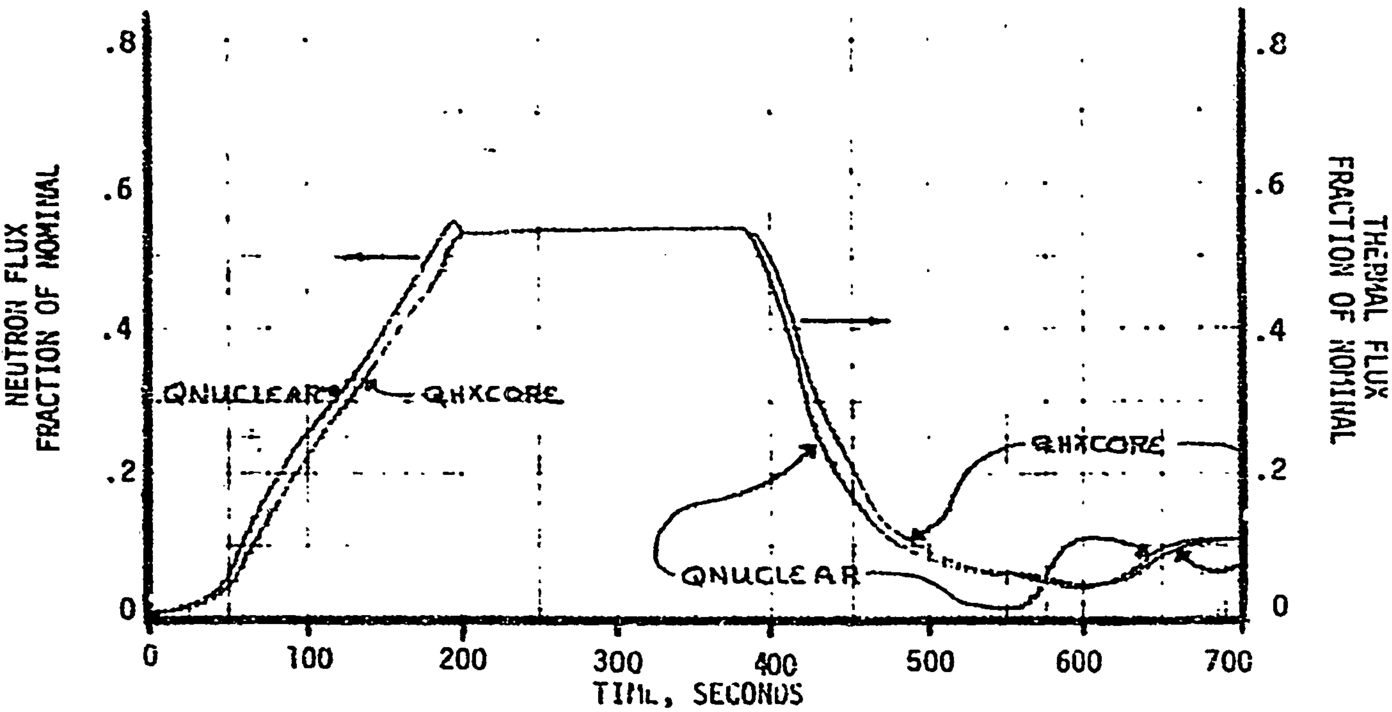


FIGURE 14.7.6-1
Supplement 12
7/70

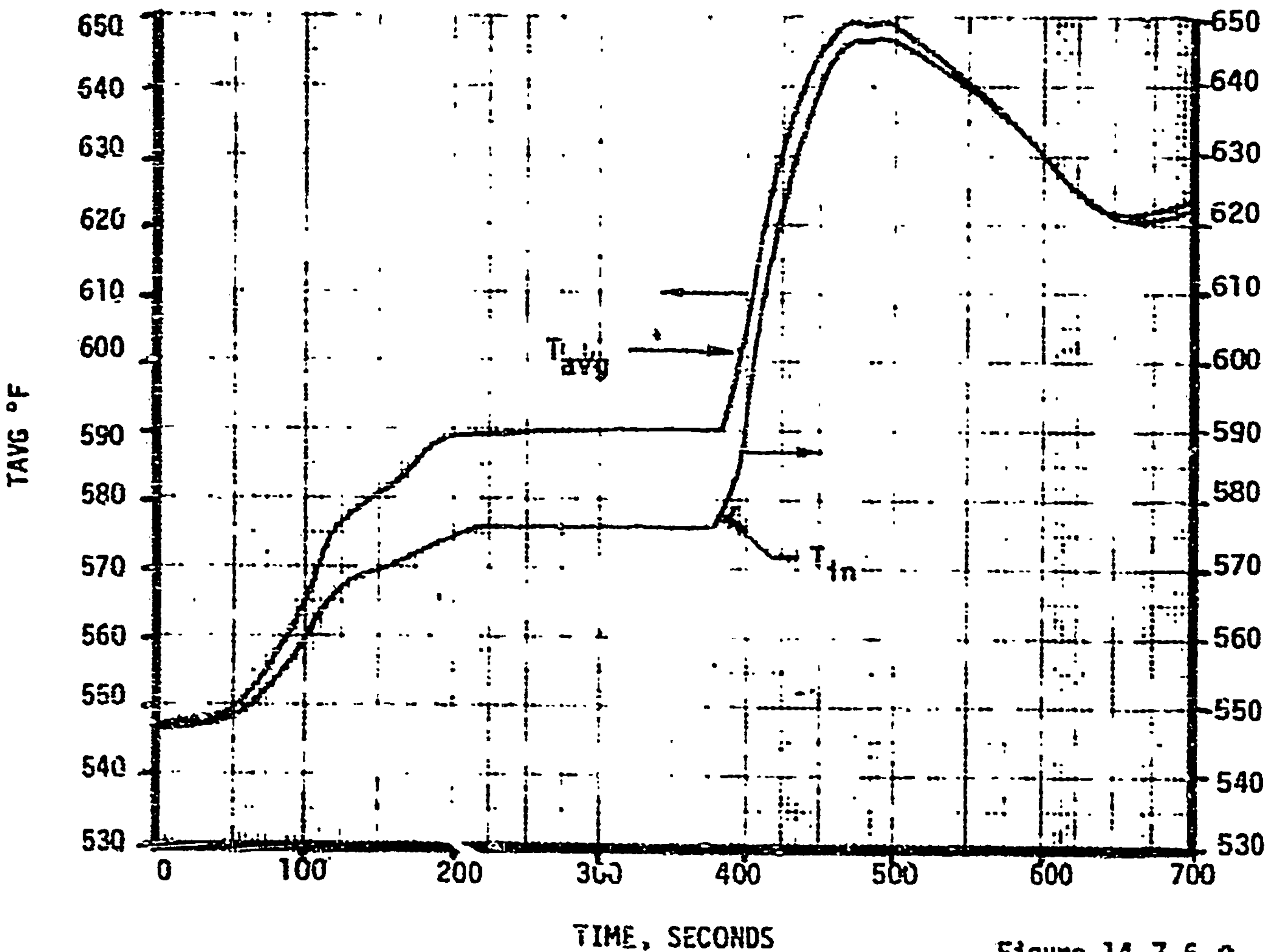


Figure 14.7.6-2

Supplement 12
7/70

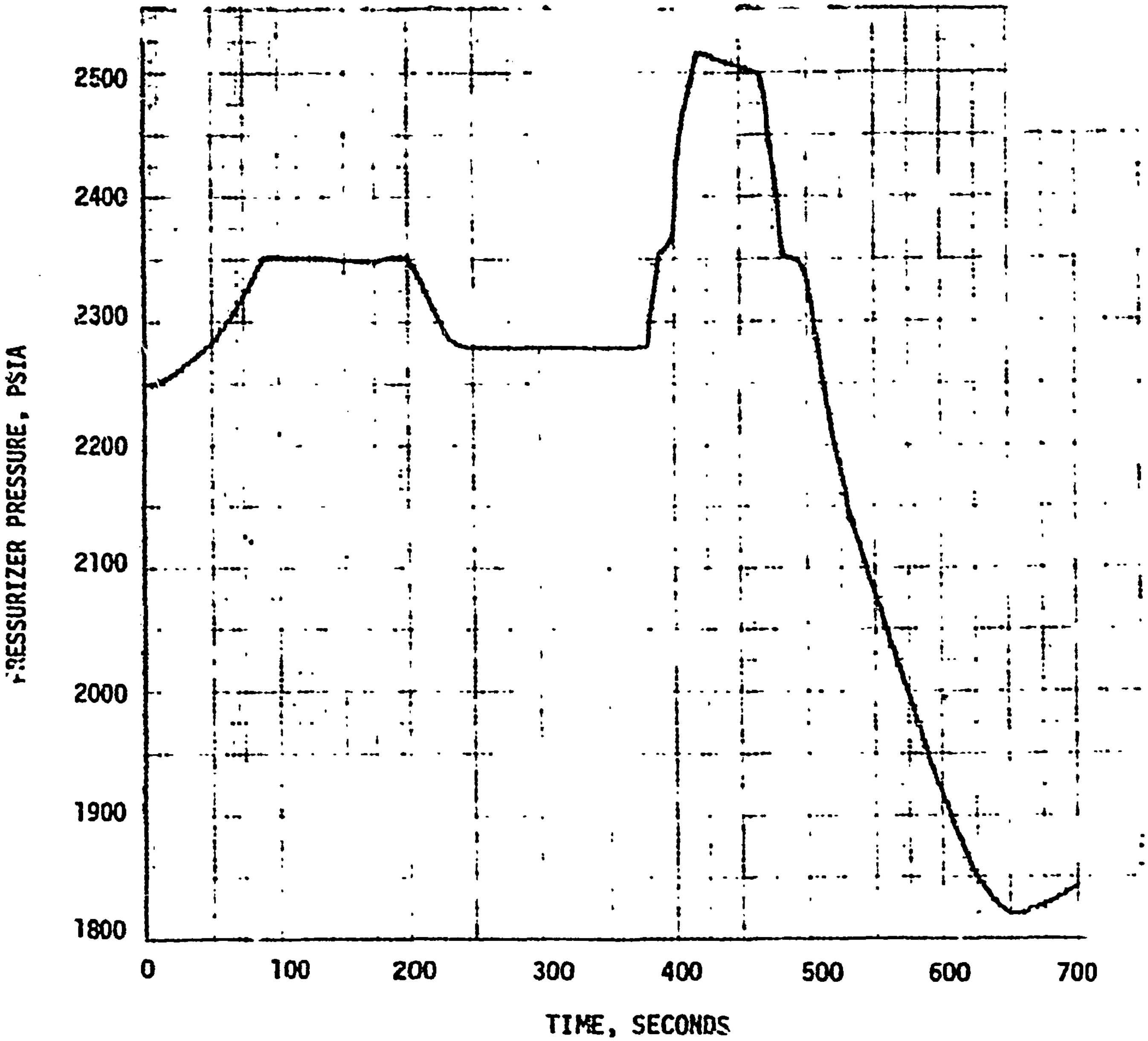


FIGURE 14.7.6-3

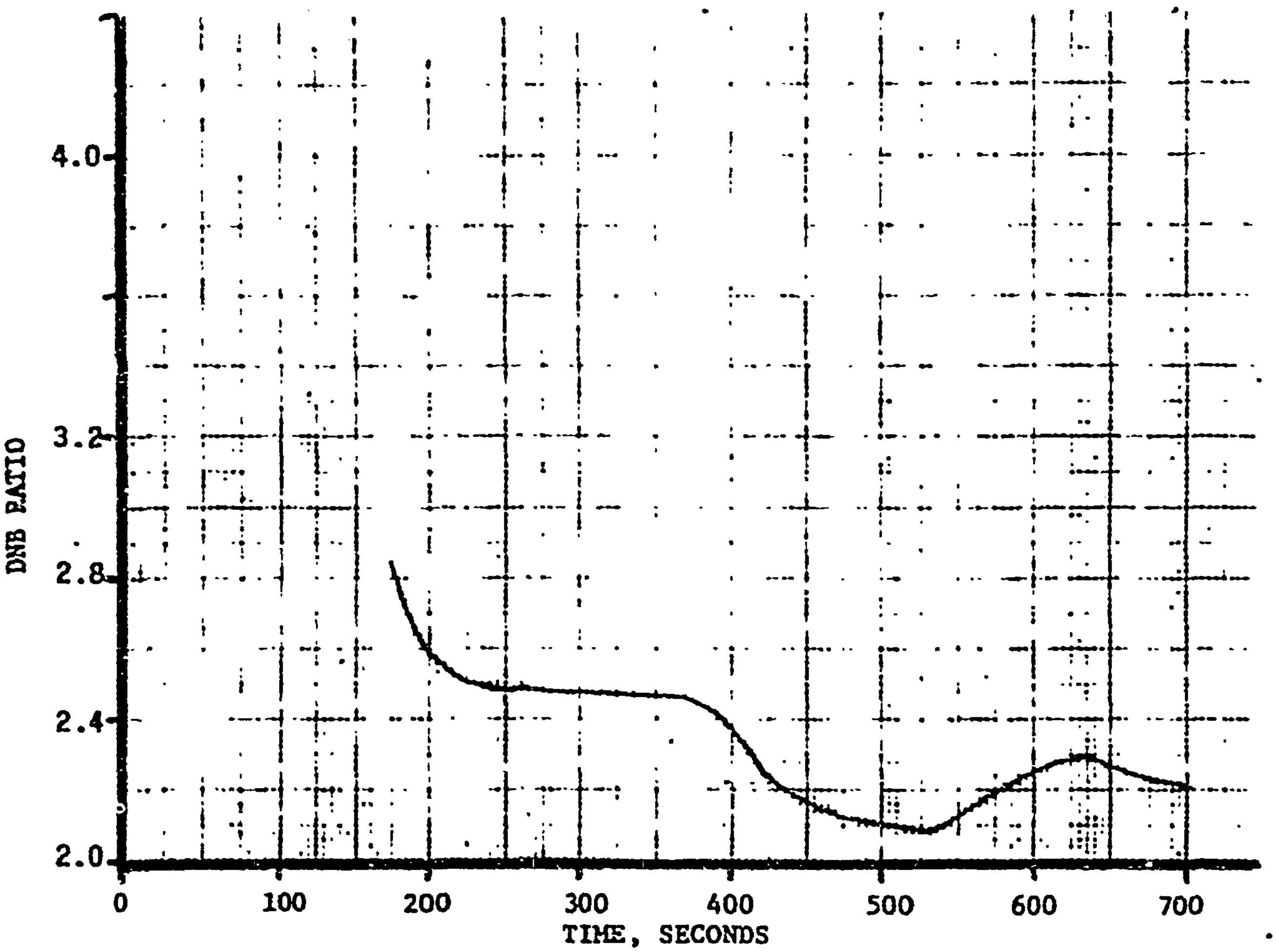


FIGURE 14.7.6-4
Supplement 12
7/70

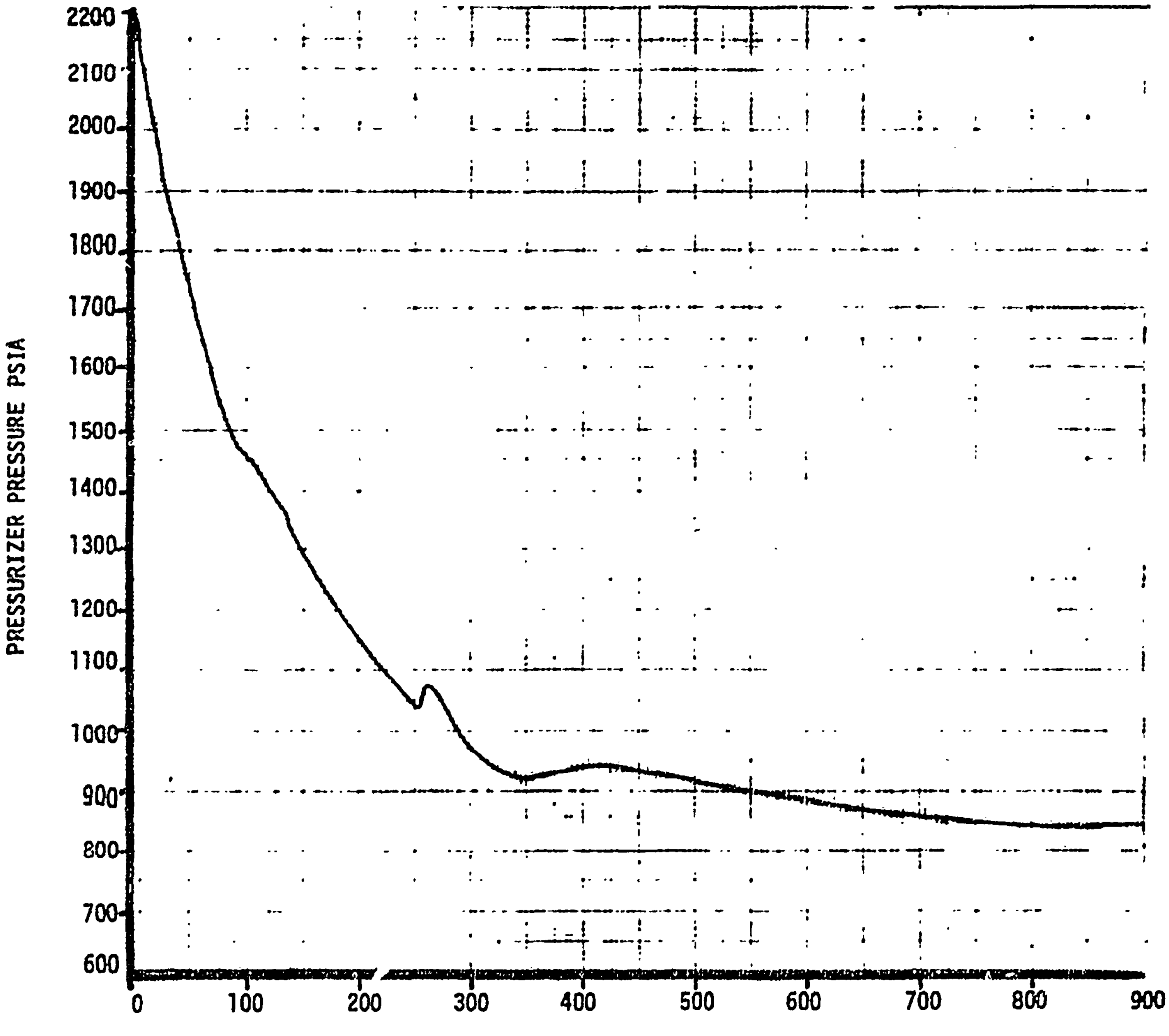


FIGURE 14.7.9-1
Supplement 12

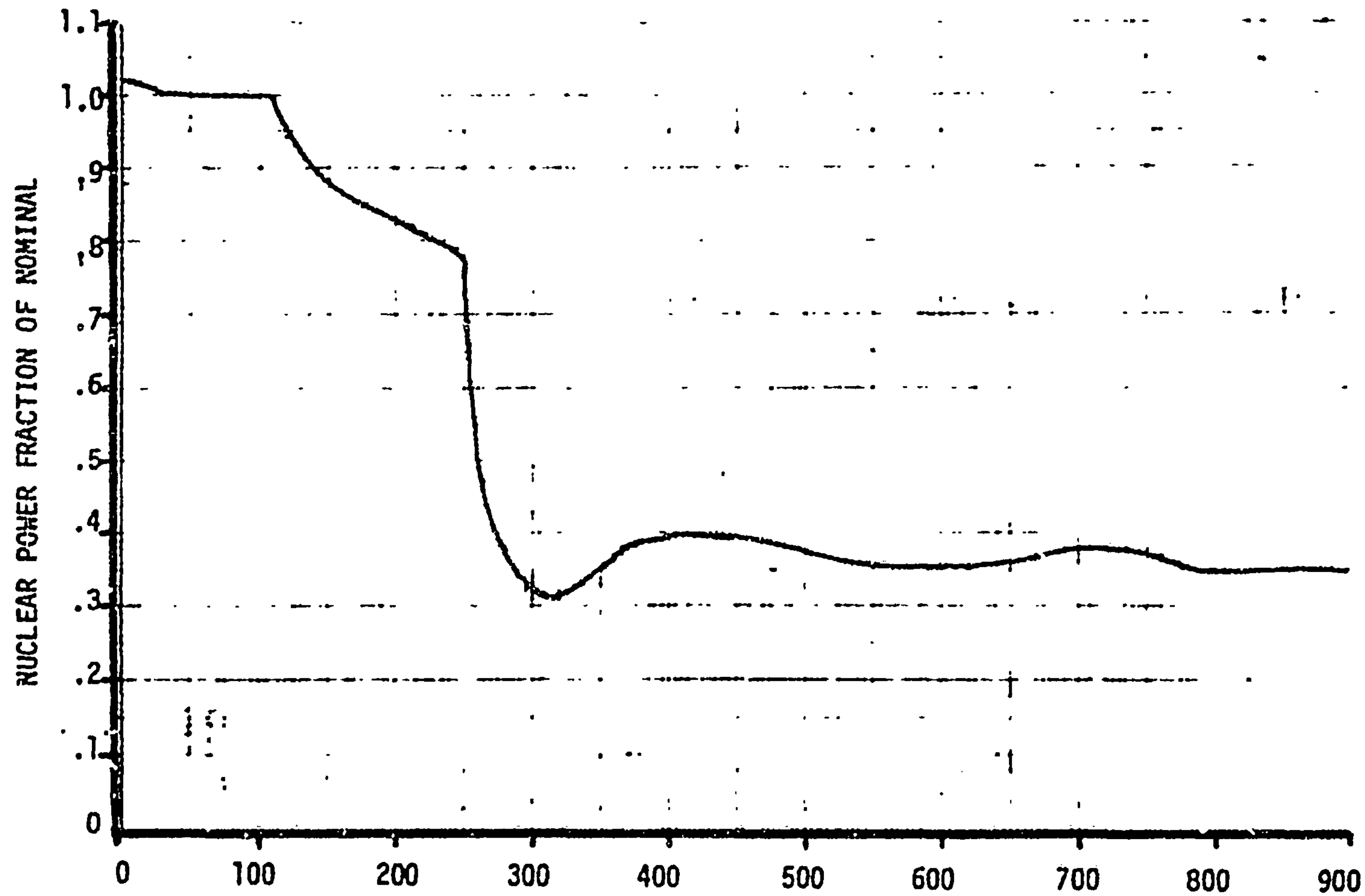


Figure 14.7.9-2
Supplement 12
7/70

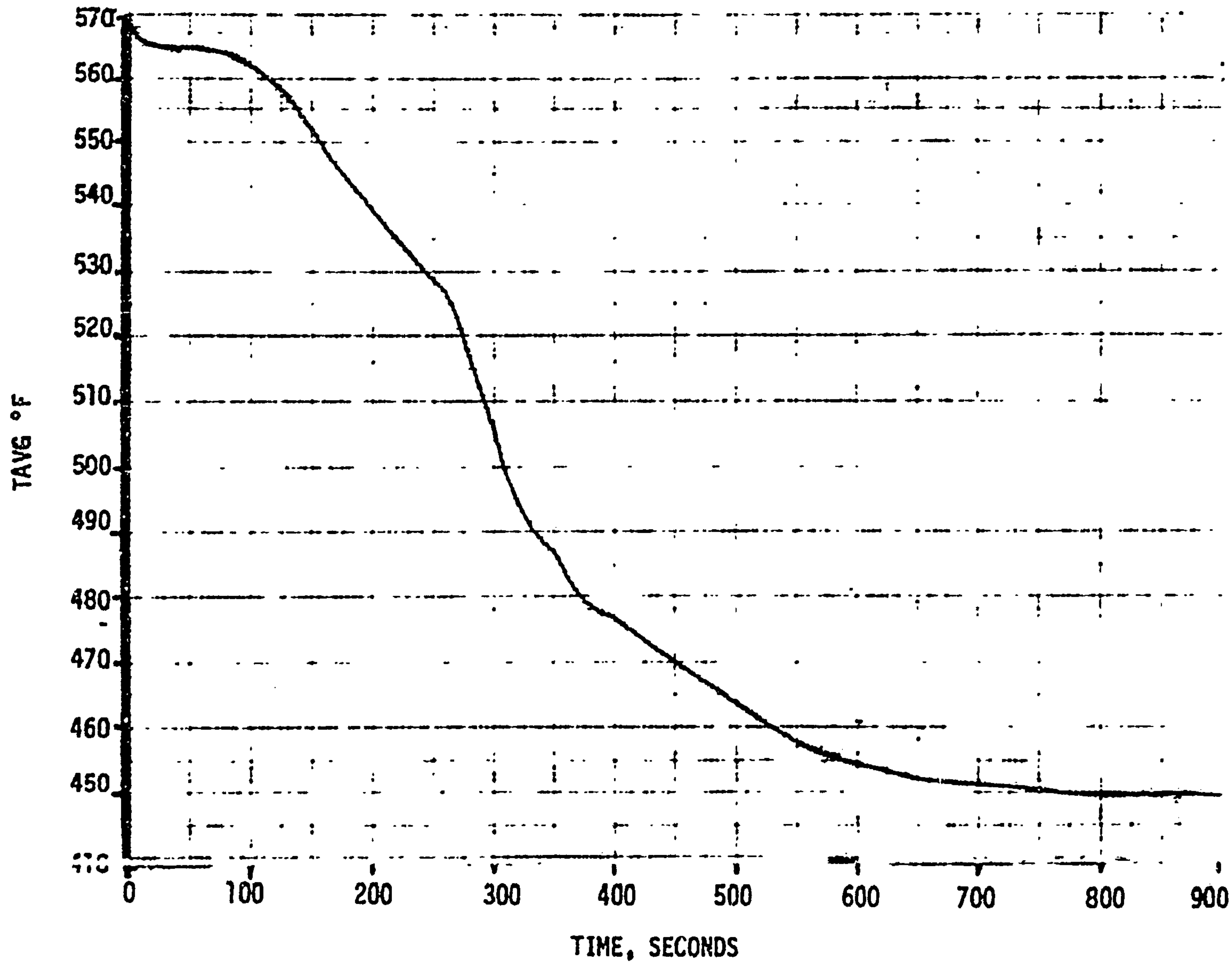


Figure 14.7.9-3

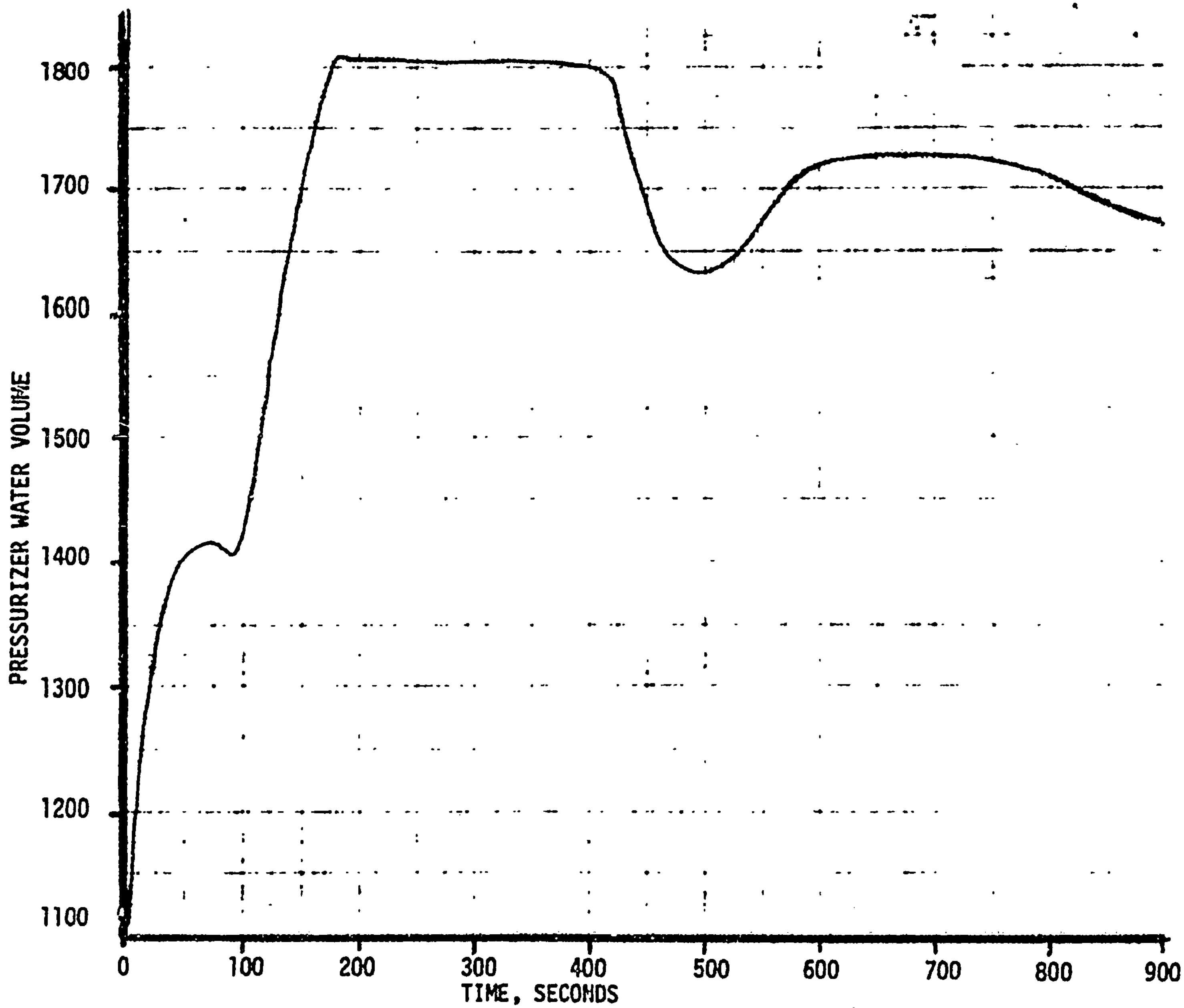


FIGURE 14.7.9-4

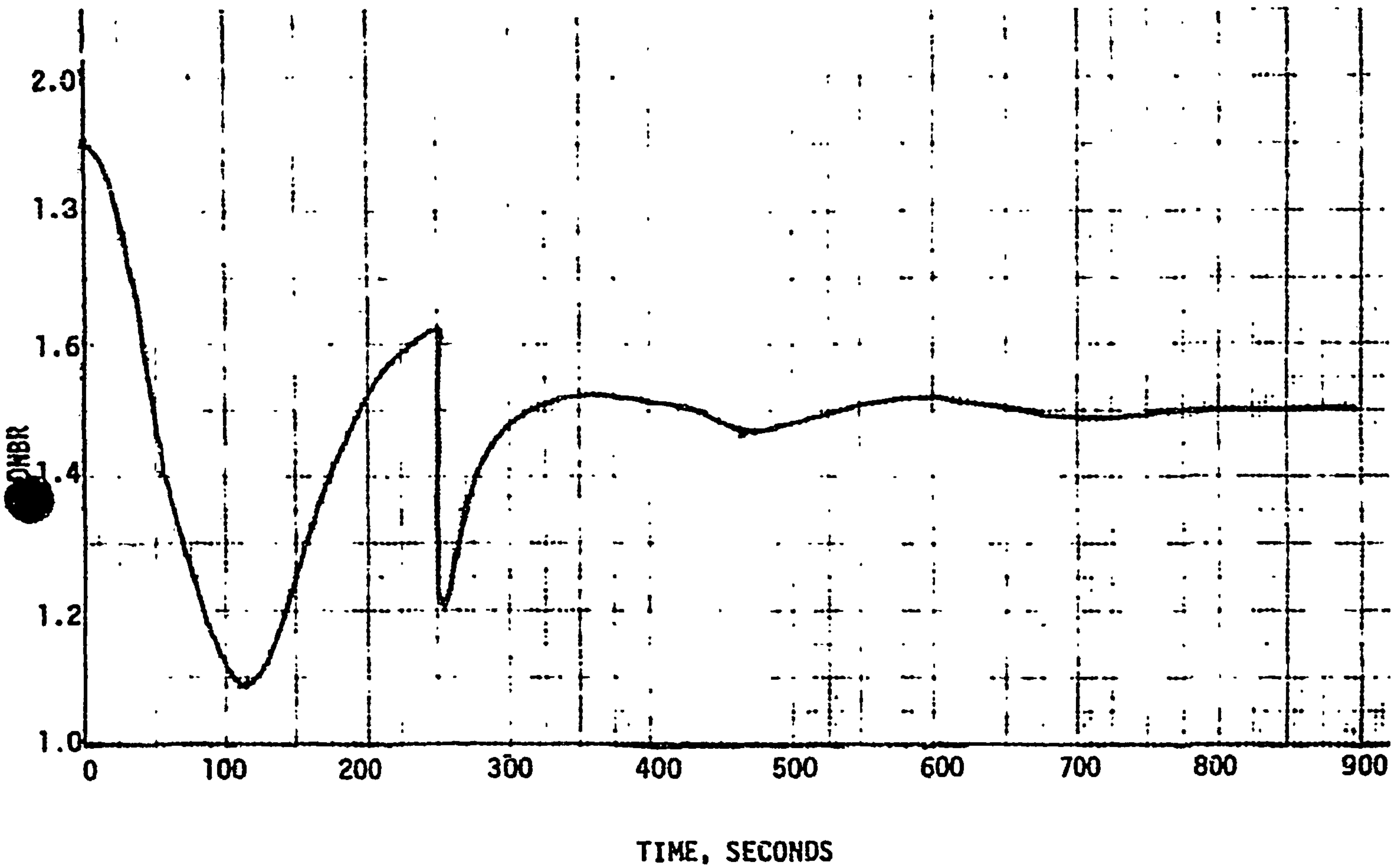


FIGURE 14.7.9-5

Supplement 12
7/70

QUESTION 14.8

Your analyses of the potential hydrogen evolution over the post loss-of-coolant period neglects certain potential hydrogen sources such as the clad-water reaction and the chemical reaction of materials subject to corrosive attack in the post-accident environment. In addition, we understand that more refined calculations regarding coolant energy deposition would indicate that the predicted evolution of hydrogen by coolant radiolysis, as shown in the FSAR, may be conservative. Please update your FSAR analyses to include all potential hydrogen sources and to factor in the more refined calculations for coolant radiolysis.

ANSWER

Attached Figure Q14.8-1 shows the quantity of hydrogen generated in the containment during the first 100 days post accident by the following sources:

1. Zirconium - water reaction
2. Chemical reaction of materials subject to corrosive attack
3. Radiolytic decomposition of coolant in the core
4. Radiolytic decomposition of coolant in the sump.

These results have been obtained on the following bases:

A. Zirconium - Water Reaction

1. 2% of the core cladding reacts immediately with core cooling solution according to the reaction

$$Zr + H_2O \longrightarrow ZrO_2 + H_2 + \text{heat}$$

2. 41,994 lbs. of zirconium cladding in the core.

B. Corrosion of Materials of Construction

1. Corrosion of aluminum according to the reaction

$$2\ Al + 3H_2O \rightleftharpoons Al_2O_3 + 3H_2$$

2. Corrosion rates vs. time post accident as shown on figure Q 14.8-2.

3. Aluminum available for reaction as follows:

Item	Mass lbs	Area ft^2
A. Control rod drive mechanism connectors	122	42
B. Reactor vessel insulation foil	269	10,000
C. Area monitors	6	4
D. Source, intermediate, and power range detectors	140	40
E. Process instrumentation and controls	420	84
F. Lighting fixtures and equipment	1061	380
G. Paint on steam generator, pressurizer, and reactor vessel	140	10,000
H. Contigency	250	85

C. Core Radiolysis

1. 50% of the halogens, 100% of the noble gases and 99% of all other fission products are retained in the core following the accident;

2. 0.44 molecules of hydrogen are generated per 100 ev of energy absorbed by water in the core;

3. 7.4 percent of the core fission product gamma energy is absorbed by the solution in the core (Calculations indicate that not more than 7.4% of the fission product gamma decay energy is absorbed in the core coolant. This value was obtained by an integration, over the entire core, of the volumetric deposition rate of fission product gamma energy and the determination of the ratio of deposited energy in the coolant to total deposited energy throughout the core. The energy deposition calculations were carried out using a computer program based on the point kernel attenuation model.);

4. Beta energy is absorbed by the fuel and cladding and does not contribute to hydrogen generation in the core;

5. 50% of the total core fission product decay energy is gamma radiation;

6. Plant operation for 830 days at 3216 megawatts thermal prior to the accident.

D. Sump Radiolysis

1. 50% of the halogens, none of the noble gases and 1% of all other fission products are released from the core to the sump during the accident;

2. 0.30 molecules of hydrogen are generated per 100 ev of energy absorbed by the sump solution (the radiolysis yield of hydrogen in solution has been studied extensively by Westinghouse. Results of static capsule tests illustrate that hydrogen yields much lower than 0.44 molecules per 100 ev can be expected where the gas to liquid volume ratio approaches zero as would be the case in the core. The sump solution will have considerable depth, which prevents the ready diffusion of hydrogen from solution. This retention of hydrogen in solution will have a significant effect to reduce the hydrogen yields to the containment atmosphere. The build-up of hydrogen concentration in solution will enhance the back-reaction reformation of water to lower the net hydrogen yield; therefore, a reduced hydrogen yield is a reasonable assumption to make for the case of sump radiolysis. While it can be expected the yield will be on the order or 0.1 or less, a conservative value of 0.30 molecules/100 ev has been used);

3. All beta and gamma energy emitted by fission products in the sump solution are absorbed and contribute to hydrogen generation;

4. Plant operation for 830 days at 3216 megawatts thermal prior to the accident.

The calculated volume percent of hydrogen in the containment is based on a containment free volume of 2,610,000 cubic feet assumed to be at 120°F and one atmosphere pressure immediately before the accident.

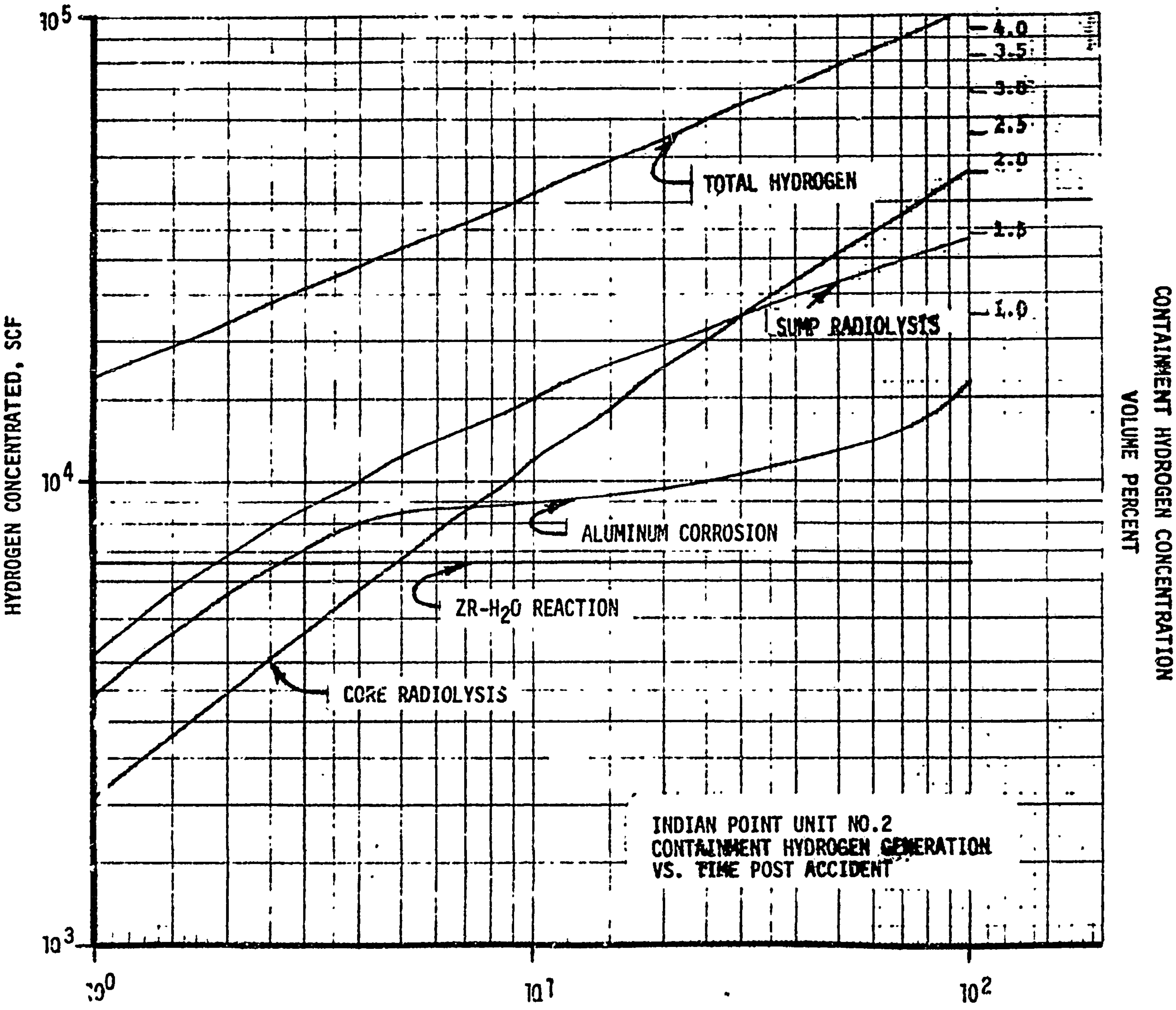


Figure 14.3-1
Supplement 6
2/70

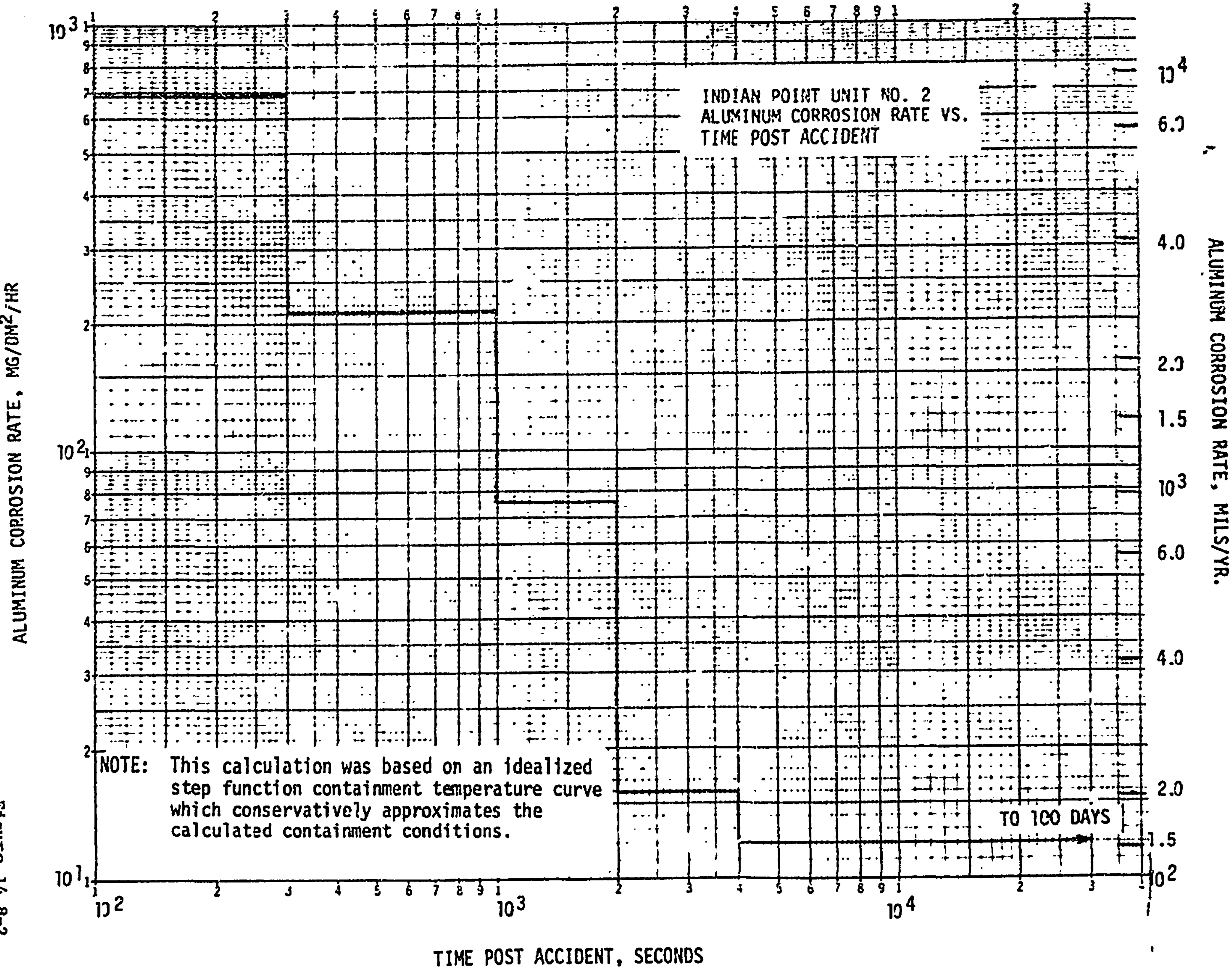


Figure 14.8-2
Supplement 6
2/70

QUESTION 14.9

We understand that Westinghouse has conducted dynamic loop testing in order to simulate and explore the influence that certain post-accident parameters (e.g., core coolant flow, coolant temperature, radiation doses) may have on coolant radiolysis. Provide a discussion of the results from these tests in order that we may acquire a better understanding of the degree of conservatism included in your analyses of the post-accident hydrogen evolution.

ANSWER

Westinghouse has conducted an extensive program into the evolution of radiolytic hydrogen following the design basis accident. In order that the results of this program would be directly applicable to the plant situation, tests were conducted in a system designed to duplicate as many of the actual plant conditions as possible. Plant parameters, such as, containment vessel volume, active core volume and core cooling flow rates were directly scaled down for the experiments. In addition, the system contained the necessary hardware to maintain and measure flow rates, as well as permit temperature variation up to 190°F.

Tests conducted in the dynamic loop system investigated the effect of solution flow rate, absorbed dose, and solution temperature on radiolytic hydrogen production. The solution flow rate and absorbed dose in the test system are directly related, since the dose rate was constant and the flow rate determined the -flux residence time. Test data showed that the rate of hydrogen production, $G(H_2)$, is directly related to core cooling solution flow rate (absorbed dose). As the solution flow was increased, the time in-flux was decreased and the hydrogen production rate approached the theoretical yield of 0.44 molecules/100 ev of energy absorbed.

It is clearly illustrated, therefore, that if a lower core radiolysis $G(H_2)$ value is applicable, as would be the case with reduced core coolant flow, the hydrogen buildup rate in containment would also decrease. Even at extremely high flow rates, about double full core cooling flow, the maximum hydrogen production rate observed was 0.44 molecules/100 ev.

The effect of solution flow rate was investigated at several temperatures, 70°F, 100°F, 140°F, and 190°F. At these temperatures, no effect on the production rate was observed. That is, the maximum radiolytic hydrogen production rate observed was 0.44 molecules/100 ev at the four temperatures investigated.

The results of these tests are yet to be published; however, some insight can be obtained from the preliminary ORNL results given in Nuclear Safety Research and Development Program Bi-Monthly, May-June, 1969.[1]

Until the above studies are completed, Westinghouse has chosen to perform its calculations of hydrogen yield from core radiolysis with the very conservative value of 0.44 molecules/100 ev. That this value is conservative and a maximum for this type of aqueous solution and gamma radiation is confirmed by many published works. The unpublished Westinghouse results from the dynamic studies show 0.44 to be a maximum at very high solution flow rates through the gamma radiation field. The referenced ORNL work also confirms this value as a maximum at high flow rates. A. O. Allen[2] presents a very comprehensive review of work performed to confirm the primary hydrogen yield to be a maximum of 0.44 - 0.45 molecules/100 ev.

---

1. ORNL Nuclear Safety Research and Development Program Bi-Monthly Report for May-June 1969.

2. Allen, A. O., "The Radiation Chemistry of Water and Aqueous Solutions".

QUESTION 14.10

We understand that additional testing regarding the effectiveness of methyl iodide absorption by impregnated charcoal filters in 100% humidity environment has been completed. Provide a summary of the results of these tests.

ANSWER

Summary of Results of ORNL Tests is as follows:

1. Decontamination efficiencies for radioactive methyl iodide transported in flowing steam air range from 98.6 to 88.9% at average relative humidities ranging from 94.0 to 99.6%.

2. At calculated relative humidities slightly in excess of 100%, a portion of the charcoal may be wetted, and cause lower decontamination efficiency. This behavior was illustrated wherein decontamination efficiencies of 30.7 & 87.3% were obsessed at calculated relative humidities of 101.5 and 100.2% respectively.

3. At 100% relative humidity, the decontamination efficiency for methyl iodide was found to be ∿88%.

The results of the ORNL Test series is given in ORNL TM 2728 dated Oct. 2, 1969.

QUESTION 14.11

Sta·e what primer and finishing coatings were used on the inner surfaces of the containment. Provide technical data and/or references which indicate the stability of the paint under loss-of-coolant accident conditions.

ANSWER

Essentially all of the painted surfaces in containment are coated with Carbozinc 11 and/or 300 series Phenalines, which are products of the Carboline Co. St. Louis Missouri. These coatings have been shown to be resistant to the post loss of coolant accident environment in tests performed at Westinghouse[1] and also at ORNL.

[1] WCAP 7198L "Evaluation of Protective Coatings for Use in Containment" Westinghouse Proprietary, April 1968.

QUESTION 14.12

Provide the results of your evaluation of the LOCA, using a multi-node analytical model (such as the SATAN code) for a 27 1/2 inch ID, double-ended, cold-leg pipe rupture. In addition to providing information on clad temperatures and system pressures, also provide the core and hot channel flow rates in sufficient detail to fully characterize the thermal and hydraulic performance during blowdown. These details should include:

(a) core pressure drop, quality, mass velocity;

(b) hot channel pressure drop, quality, mass velocity;

(c) heat flux distribution in hot channel;

(d) flow rates in upper and lower plenums;

(e) flow-rate in broken and intact cold-leg and hot-leg piping; and

(f) flow rate out the break.

Identify the heat transfer correlations used for the various phases of the blowdown and refill period and relate these correlations to the most recent experimental data available.

ANSWERS

(a) & (b) Figures 14.12-1, 14.12-2, and 14.12-3 present the core pressure drop, quality, and mass velocity as calculated by the SATAN Code.

(c) Figures 14.12-4 through 14.12-7 show the heat flux distribution in the hot channel for the double ended cold leg break loss of coolant accident for the Indian Point No. 2 plant. These heat fluxes were obtained from the LOCTA-R2 computer code calculations of the LOCA analysis presented in the Core Cooling Analysis Supplement 12, page 14B-11.

(d), (e) & (f) Figure 14.14-8 presents a sketch of the SATAN model used to analyze the 0.5 ft^2 break. A 70 control volume simulation was used for the analysis. Noted on the figure, are the control volumes that provide the most useful information for interpreting the transient. These are (control volume numbers in parenthesis):

(d), (e) & (f) cont'd

1. Core Region Flow
   a. Inlet (10)
   b. Midplane (1)
   c. Exit (2)

2. Upper Plenum Flow
   a. Into upper plenum (3)
   b. Out of upper plenum unbroken loops (40)
   c. Out of upper plenum broken loop (12)

3. Reactor Vessel Inlet Flow
   a. Flow broken loop (39)
   b. Flow unbroken loop (67)
   c. Downcomer region (5, 8)

4. Break Area
   a. Break Flow (37) (38)
   b. Upstream of break (36)
   c. Downstream of break (38)

Identify the heat transfer correlations used for the various phases of the blowdown and refill period and relate these correlations to the most recent experimental data available.

As discussed in Supplement 12 to the FSAR, parametric analysis were performed through a series of iterations was run between SATAN AND LOCTA until the total heat transferred from the coolant as a function of time were essentially identical in both cases. This was done to properly treat the feedback effect from core heat transfer on blowdown. Figures 14.12-9 and 14.12-10 present the heat transfer coefficient and heat flow rate for the two codes. This analysis was performed at the maximum calculated thermal rating (3216 MWt).

Heat Transfer Correlations

Table 14.12-2 shows the heat transfer correlations used in the core cooling analysis to calculate the clad temperature transient during a LOCA and specifies their application during the accident. The design basis analysis presented in Supplement #12, Page 14B-11, for the double-ended cold leg break utilized the transition to stable film boiling correlation from the onset of DNB (assumed at 0.5 seconds) until the uncovery or steam cooling period was reached at 7.4 seconds. A quality of 1.0 is not reached until this time because of the low reactor power rating (2758 MWt). The radiation and convection to steam heat transfer correlations were used until the end of blowdown which occurred at approximately 17 seconds. After this time no heat is transferred from the surface of the fuel rod cladding until 26 seconds when the recovery phase or the entrainment process is initiated by the flooding of the core with the accumulator water. The values of the heat transfer coefficients predicted by these correlations are conservative as indicated by the following experimental data.

a. Transition Boiling and Film Boiling

The comparison of this correlation with experimental data was presented in WCAP-9005 (Westinghouse Proprietary Report). Figure 5 shows a plot of the heat transfer coefficients predicted by this correlation versus the experimental data. It is readily apparent that the correlation is conservative with respect to the results of this t . since the measured value is greater than predicted for 98% of the data. Furthermore, it should be noted that the degree of conservatism contained in the correlation increases with increasing values of the heat transfer coefficient.

b. Steam Cooling

The correlations used to calculate the heat transfer coefficient during the steam cooling period were verified by the work sponsored by Westinghouse at the University of Michigan. This was part of the Flashing Heat Transfer Program. The results of this phase of the program have been documented in WCAP-7396-L (Westinghouse Proprietary Report). The primary objective of this tes . was to determine the behavior of radiation heat transfer to steam at elevated pressures (up to 5 atm.).

The results of the heat transfer test yield the following conclusions:

1. The McEligot et al. correlation realistically predicts the convective heat transfer coefficients in turbulent flow.

2. In turbulent flow the radiant heat transfer contribution to the total heat transfer coefficient is adequately predicted by Hottel's technique.

3. The total heat transfer coefficient to steam in turbulent flow may be calculated by adding the convective term determined by McEligot's correlation and radiative term determined by Hottel's technique.

   A comparison of predicted versus measured total turbulent heat transfer coefficient is shown in Figure 6 and excellent agreement can be seen.

4. In laminar flow the total heat transfer coefficient is conservatively predicted by using the correlation of Hausen and Kays for the convective contribution and the method of Hottel for the radiant contribution.

5. The prediction of laminar heat transfer coefficient can be improved by evaluating the steam properties at film conditions instead of bulk conditions.

6. The effect of a non-uniform heat flux on the heat transfer coefficient is negligible for the conditions which exist during a loss-of-coolant accident.

c. Bottom Flooding Heat Transfer

In a LOC thermal analysis, during reflooding the core is assumed to be cooled by a two-phase mixture which is present due to the entrainment leaving the flooded region of the core. This assumption was verified by high speed movies of the FLECHT (WCAP-7435) test bundles that show that water is entrained by the steam generated in the lower portion of the core and is carried along the channels. In addition, the conservatism of the correlations used to evaluate clad temperature, turn-around time, and heat transfer coefficient is indicated by the fact that the clad temperature rise after reflooding as predicted by the Westinghouse design model, is higher than that measured in the FLECHT test.

TABLE 14.12-1

INDIAN POINT UNIT NO. 2
DE-CL BREAK (2758 MWt)
FLOW RATE (LBS/SEC)

TIME (SEC)	CORE INLET	CORE MIDPLANE	CORE EXIT	UPPER PLENUM	DOWNCOMER	DOWNCOMER	UNBROKEN LOOP HOT LEG	UNBROKEN LOOP COLD LEG	BROKEN LOOP HOT LEG	BROKEN LOOP COLD LEG	DOWNSTREAM OF BREAK	UPSTREAM OF BREAK	BREAK FLOW	BREAK FLOW
	10	1	2	3	5	8	40	67	12	39	38	36	37	38
0	37311.0	37311.0	37311.0	37311.0	37311.0	37311.0	27983.0	27983.0	9327.7	9327.7	9327.7	9327.7	0	0
0.5	-20913.9	-5389.9	3430.3	1.5591.1	-23945.1	-23731.6	37062.7	33438.3	-2835.1	-57390.2	-5[illegible]	29051.5	30212.6	55987.5
1.0	-1956.9	1577.5	5502.8	25879.5	-22418.2	-22036.2	23563.7	31834.8	-166.0	-54210.9	-54218.2	29734.9	29993.2	[illegible]
1.6	-2044.3	-2966.0	-5495.9	41572.6	-13339.4	-13349.7	6247.8	3414.9	-2459.5	-4924.1	-98249.4	27932.4	28617.0	46007.9
2.0	-7699.6	-10.5	1243.1	7909	-15231.3	-14924.0	20708.1	29715.5	-1141.9	-44955.5	-44490.2	21097.3	14191.5	41728.5
3.0	9481.1	2793.3	2650.3	7765.9	-7918.7	618.2	1891.2	28789.8	3645.5	-[illegible].17.9	-36539.5	12865.5	12897.3	32540.8
4.0	4491.5	3714.9	[illegible]48.7	5240.5	6598.2	5392.9	2058.7	32086.2	8096.9	[illegible]47.1	-30922.0	12060.5	12283.9	30664.5
5.0	4412.2	3848.6	4948.7	4878.7	-1799.8	1336.1	-1179.9	20682.1	7707.3	-2599.1	-26160.9	9258.4	9256.7	26969.6
6.0	449.7	1639.0	2404.7	3267.3	-5738.1	-2858.9	-2905.3	14251.9	7547.7	-22876.1	22945.7	8168.9	8190.9	23120.3
7.0	534.9	91.7	639.8	1568.4	-6446A	-949.9	-4339.1	10243.2	6584.6	16800.8	-18411.2	7374.3	7879.1	19029.9
8.0	-939.4	-373.2	-151.1	8957.8	-6196.2	-4584.2	-5197.8	7671.9	6085.6	-15027.5	-15075.8	6766.2	6769.1	15718.2
9.0	-567.6	-99.9	-691.2	-865.7	-5315.2	-3516.8	-3994.5	5481.3	5385.7	-12638.9	-12685.5	6340.6	6244.9	12727.2
10.0	-1657.8	-1982.5	-2918.2	-2205.7	3714.8	-2897.4	-2793.6	4828.9	3762.8	-9047.9	-9078.5	5590.2	5598.2	4105.9
11.0	-1358.0	-1202.1	-890.2	104.4	-2697.6	-3748.2	455.7	1434.9	328.0	-5601.4	-5619.8	3417.[illegible]	3436.7	5627.1
12.0	-1108.9	-1040.9	-912.8	-971.1	-1494.3	-1775.2	99.5	919.2	991.4	-3844.8	-3854.0	1569.1	1572.4	3863.5
13.0	-872.4	-833.6	-740.8	-513.0	-1223.8	-639.5	275.3	626.5	524.9	-2651.9	2660.1	1015.0	1006.8	2663.0
14.0	-619.3	-536.1	-477.8	-343.9	33.6	1253.1	154.8	486.9	514.8	1253.1	-1264.1	741.0	[illegible]	[illegible]
14.4	-260.1	-211.1	-1281.3	-967.5	1[illegible]1.8	1363.0	302.5	435.9	385.3	-857.8	-861.7	706.1	711.4	865.7
15.0	-573.0	-577.7	-57.9	831.2	[illegible]3.3	564.1	164.7	403.5	290.6	-592.7	-592.6	291.4	294.8	592.1
16.0	-318.9	-110.9	-117.4	-724.6	174.1	-520.3	74.3	370.0	312.4	-550.3	-548.9	269.5	265.6	598.8

TABLE 14.12-2

HEAT TRANSFER CORRELATIONS USED FOR THE VARIOUS PHASES OF BLOWDOWN AND REFILL

Time of Application During Accident	Process	Correlation	References
Time of break till DNB	Nucleate boiling	$\Delta T_{SAT} = 1.9 e^{-P/900} q''^{1/4}$	Jens, W. H. and P. A. Lottes, USAEC Report, ANL-4627 (1951)
From DNB till time of uncovery or steam cooling period	Transition to film boiling	Proprietary	WCAP-9005 (Westinghouse Proprietary Report)
During uncovery or steam cooling period	Laminar forced convection to steam	$(\frac{hD}{k})_{ISO} = 3.66$ $h/h_{ISO} = (\frac{T_b}{T_w})^{0.25}$	H. Hausen VDI ZEIT No. 4 (1943) W. M. Kays TRANS ASME, Vol. 77, (1955)
	Turbulent forced convection to steam	$\frac{hD}{k} = .020 (Re)_b^{0.8} (Pr)_b^{0.4} (\frac{T_w}{T_b})^{-0.5}$	D. M. McEligot, et al. JOURNAL OF HEAT TRANSFER, Vol. 87, (1965) JOURNAL OF HEAT TRANSFER Vol. 88, (1966)
	Radiation to steam	$h = 0.1713 \frac{\epsilon[(\frac{Tw}{100})^4 - (\frac{Tb}{100})^4]}{Tw-Tb}$	H. C. Hottel, HEAT TRANSMISSION by McAdams, (1954)
Recovery	Entrainment	Curve of R. F. Davis	Proc. Inst. Mech. Engrs. Vol. 144, (1940-41) Mitsuishi, N., S. Sakata, Y. Yamamoto, and Y. Oyama, STUDIES ON LIQUID ENTRAINMENT, AEC-tr-4225, (1951)
Upon recovery with a steam velocity of at	Mist heat transfer in dispersed flow	$h = .023 \frac{k_v}{D} [\frac{\rho_v D_e}{\mu_v} (\frac{Q_L + Q_V}{\dots})]^{0.8} Pr_v^{0.4}$	Dougall, R. S. and W. M. Rohsenow, MIT REPORT 9079-26, (1963)

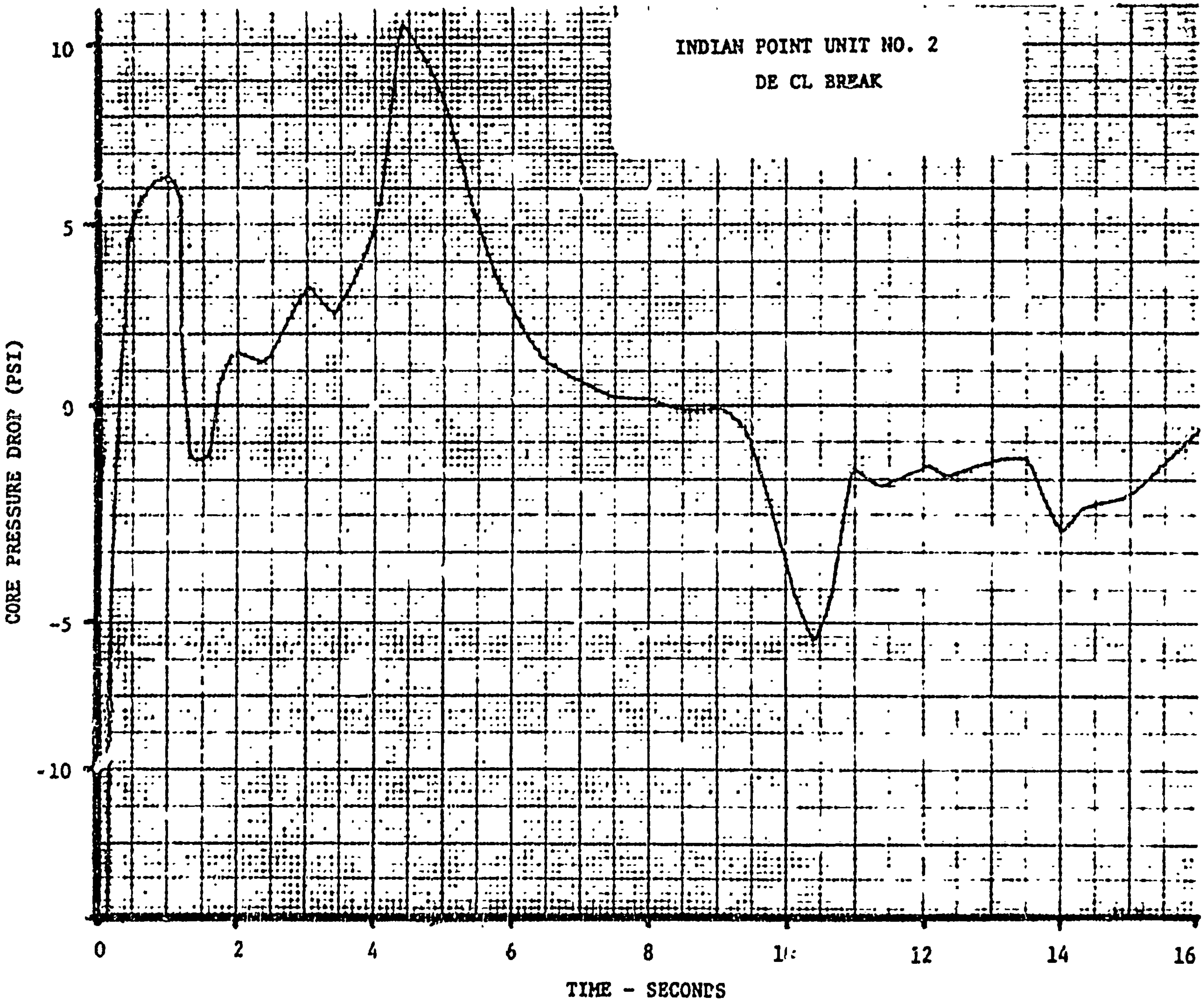
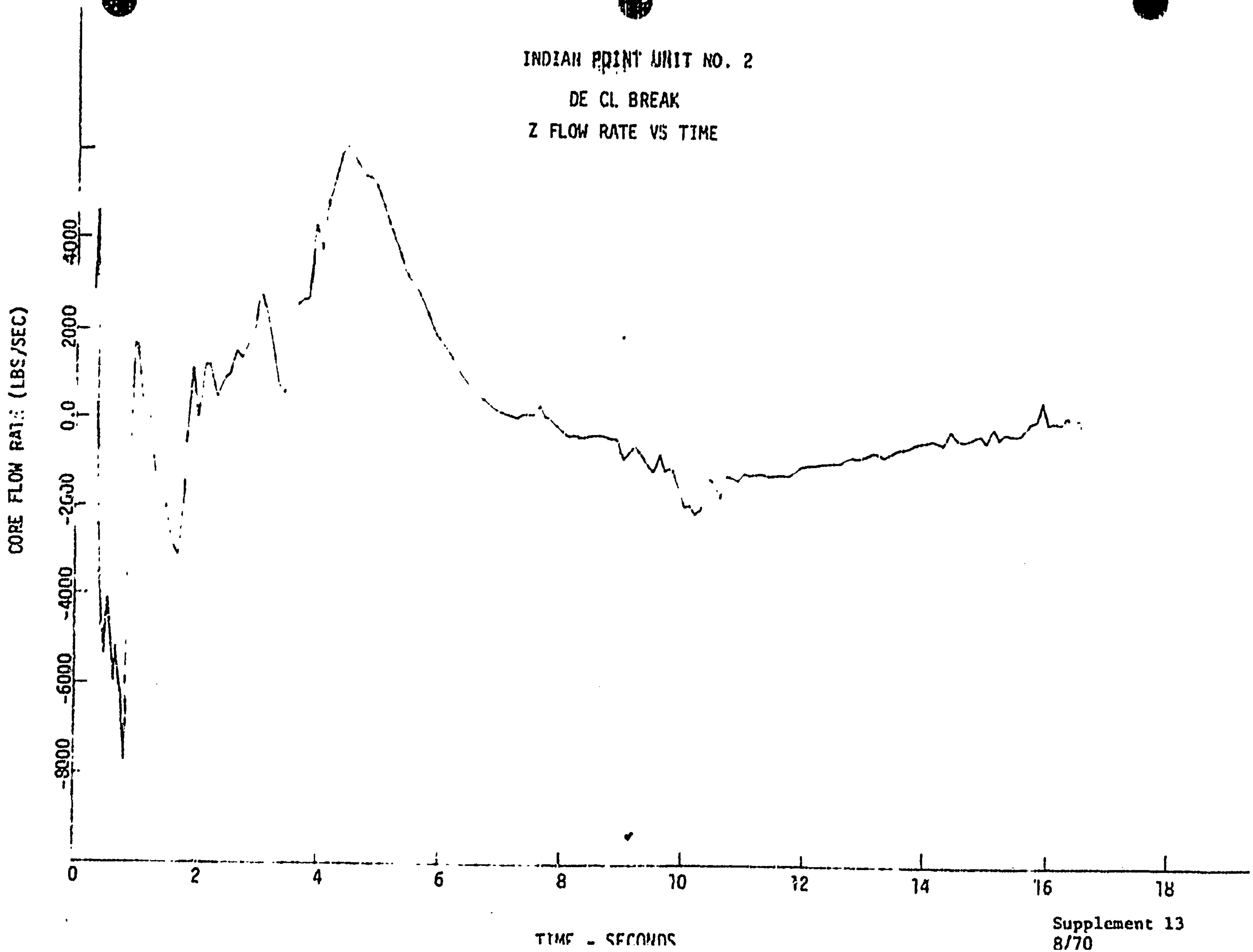


FIGURE 14.12-1
Supplement 13



INDIAN POINT UNIT NO. 2
DE CL BREAK
Z FLOW RATE VS TIME
CORE FLOW RATE (LBS/SEC)
4000
2000
0.0
-2000
-4000
-6000
-8000
0
2
4
6
8
10
12
14
16
18
TIME - SECONDS

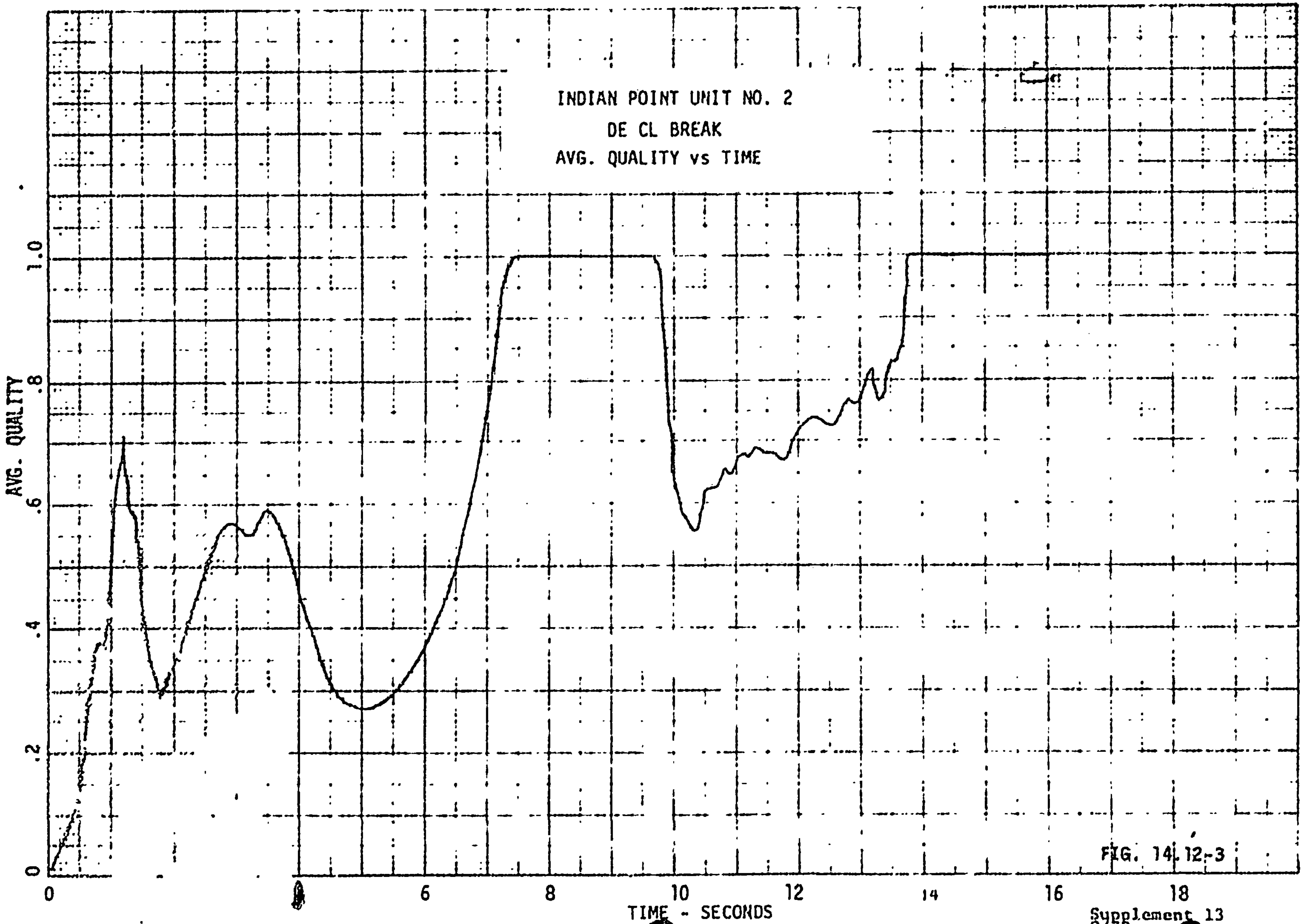

INDIAN POINT UNIT NO. 2
DE CL BREAK
AVG. QUALITY vs TIME
AVG. QUALITY
0
.2
.4
.6
.8
1.0
0
6
8
10
12
14
16
18
TIME - SECONDS

FIG. 14.12-3

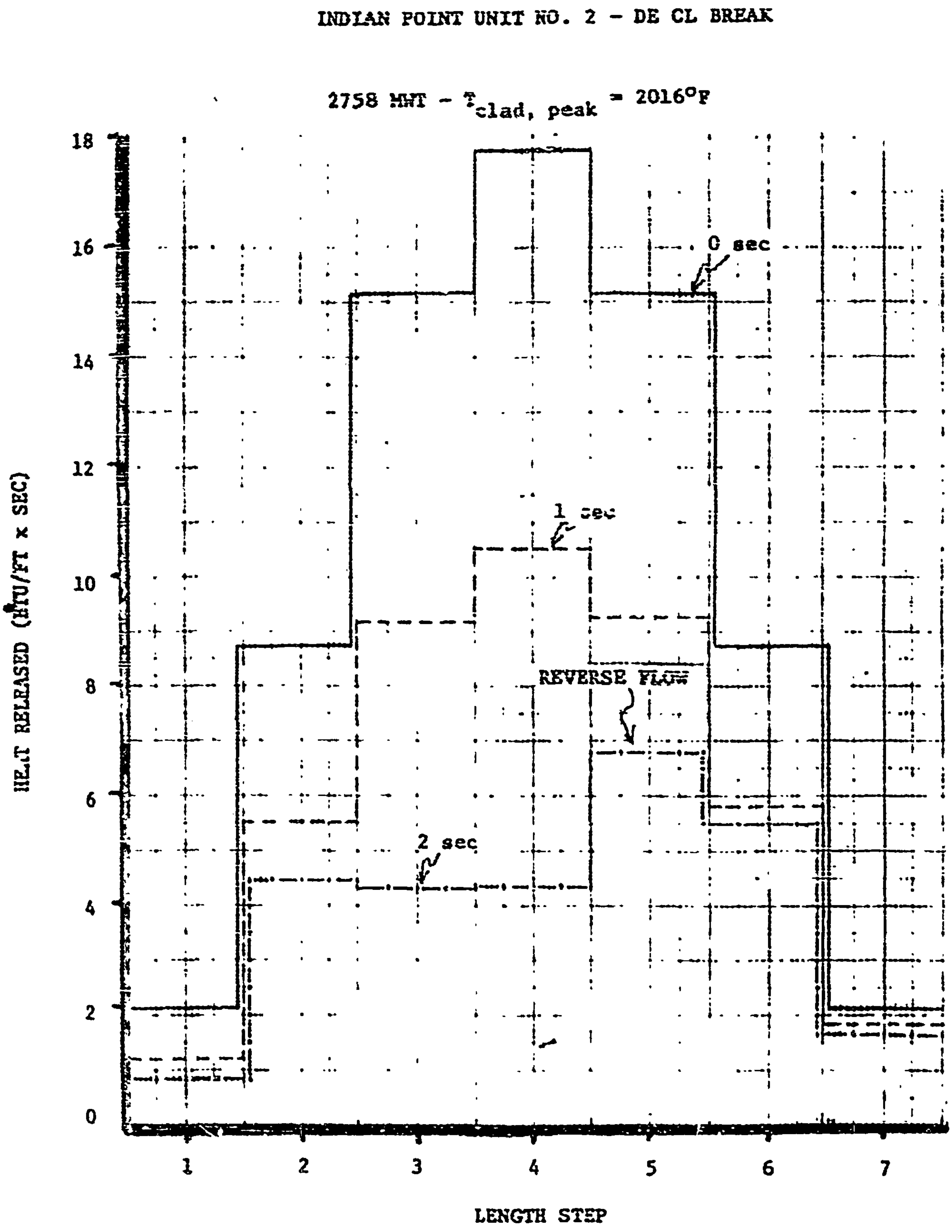


Figure 14.12-4

INDIAN POINT UNIT NO. 2

DE-CL BREAK

2758 MWT - T_{CLAD}, PEAK = 2016°F

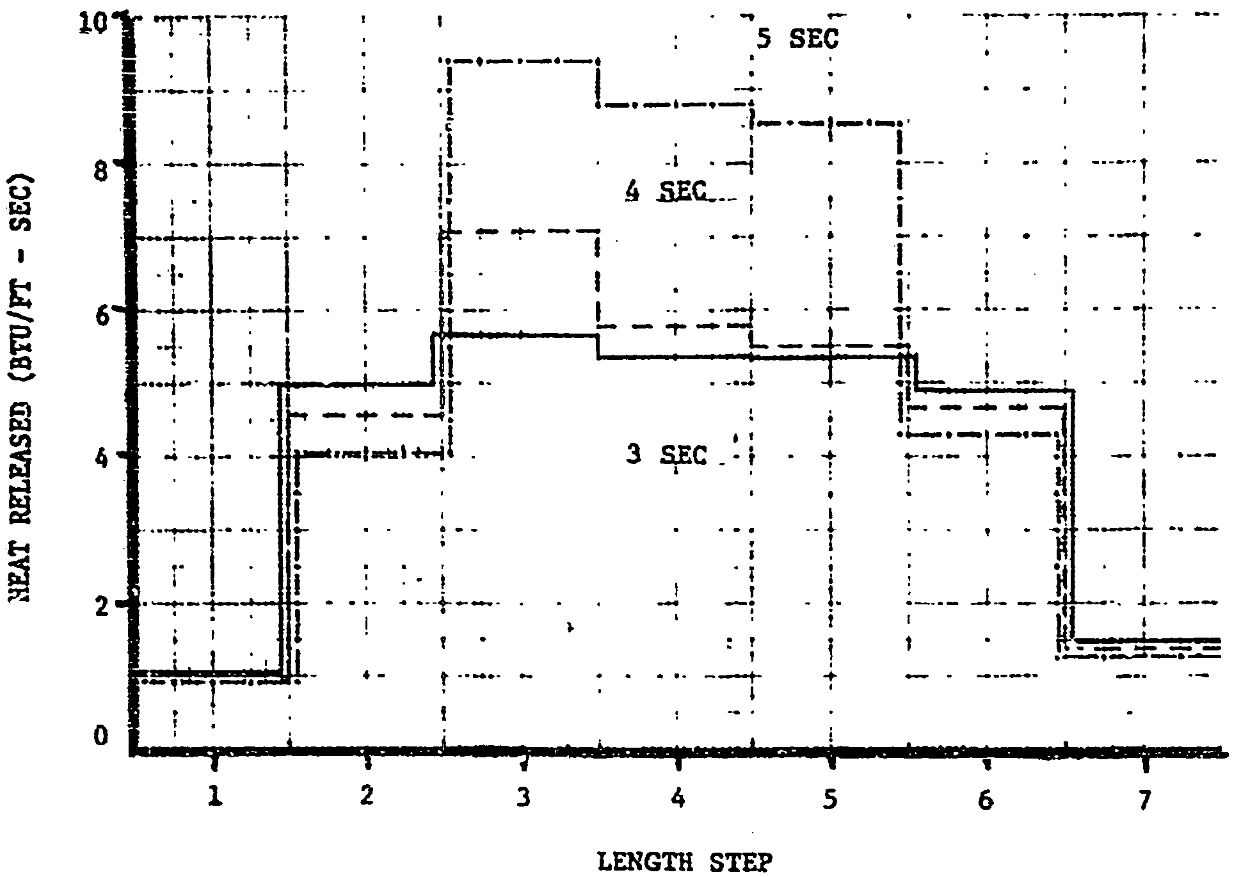


FIGURE 14.12-5

Supplement 13
8/70

INDIAN POINT UNIT NO. 2 - DE CL BREAK

2758 MWT - $T_{c, peak}$ = 2016°F

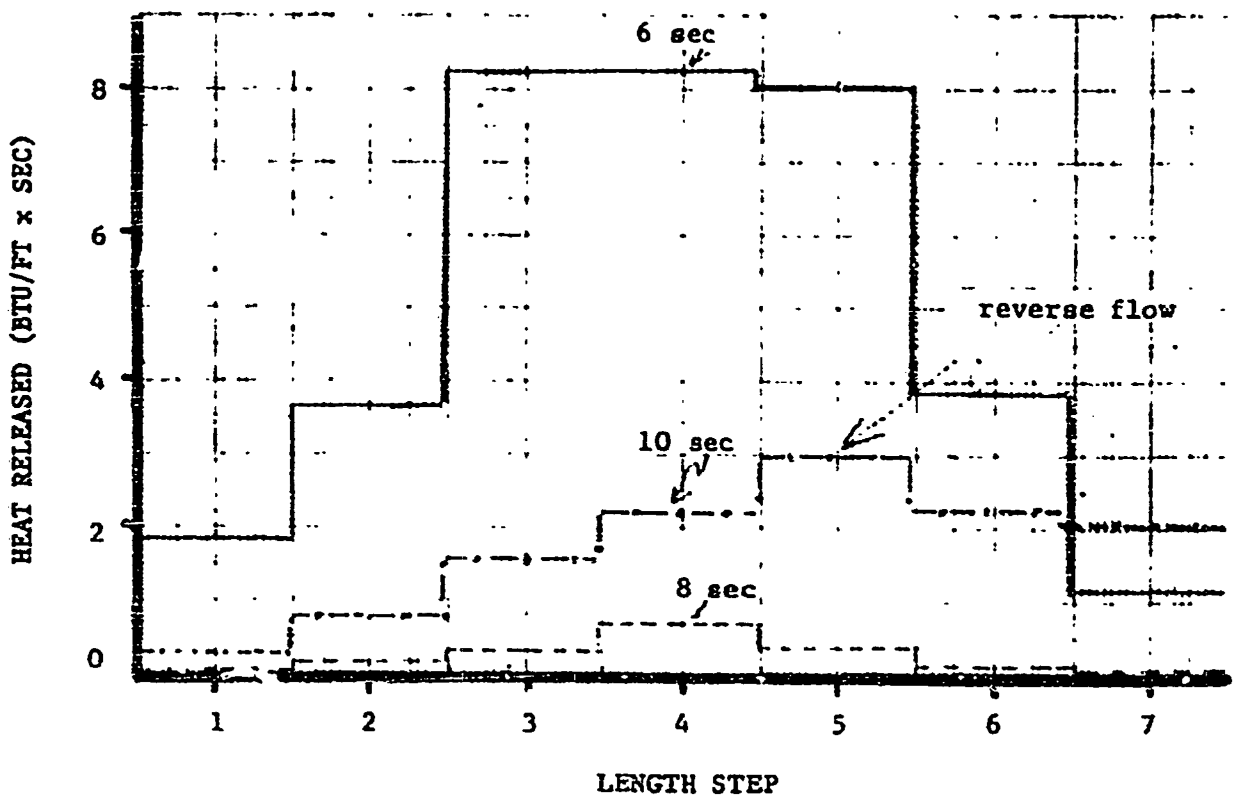


FIGURE 14.12-6

INDIAN POINT UNIT NO. 2

DE CL BREAK

2758 MWt - $T_{CLAD,\ PEAK}$ = 2016 °F

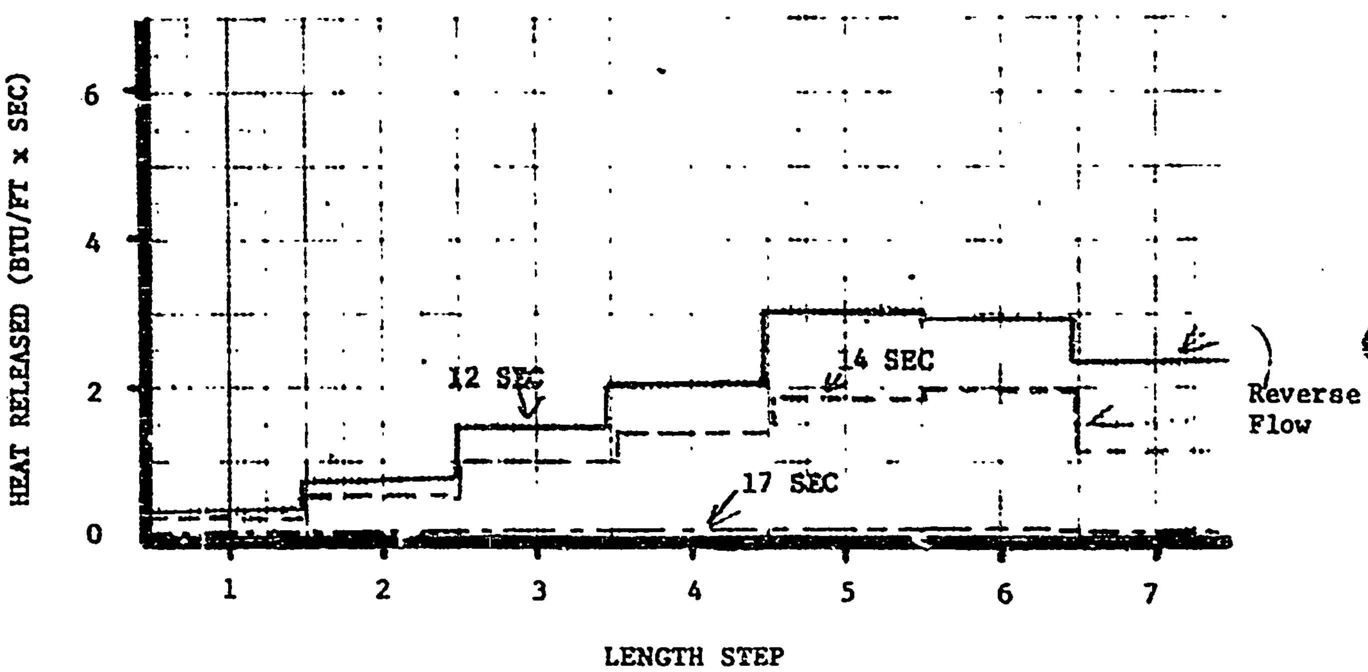


FIGURE 14.12-7

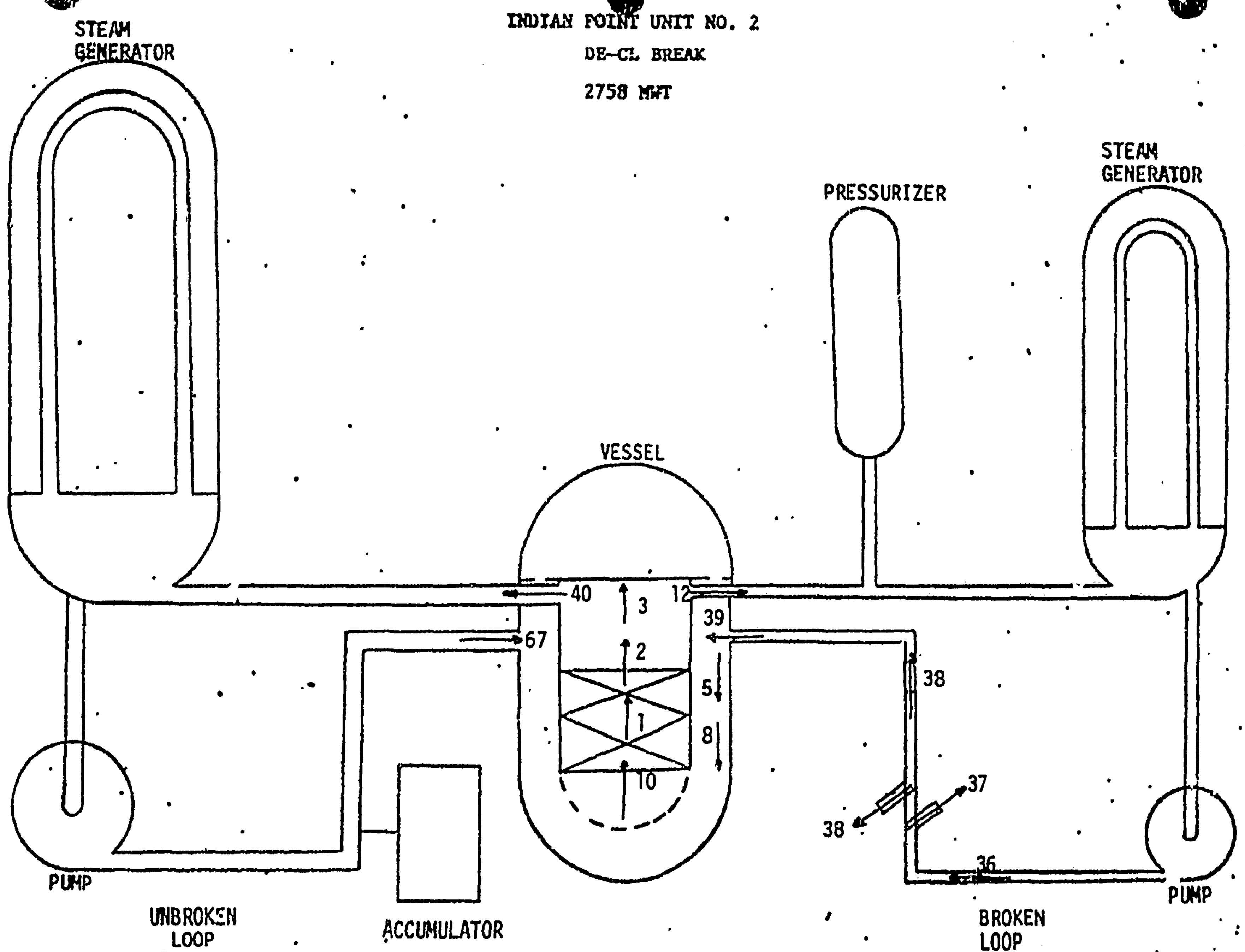

INDIAN POINT UNIT NO. 2
DE-CL BREAK
2758 MWT
STEAM
GENERATOR
PRESSURIZER
STEAM
GENERATOR
VESSEL
40
3
12
39
67
2
5
8
1
10
38
37
38
36
PUMP
UNBROKEN
LOOP
ACCUMULATOR
BROKEN
LOOP
PUMP

FIG. 14.12-8

INDIAN POINT UNIT NO. 2

DE-CL BREAK

h_{FILM} FROM LOCTA & SATAN

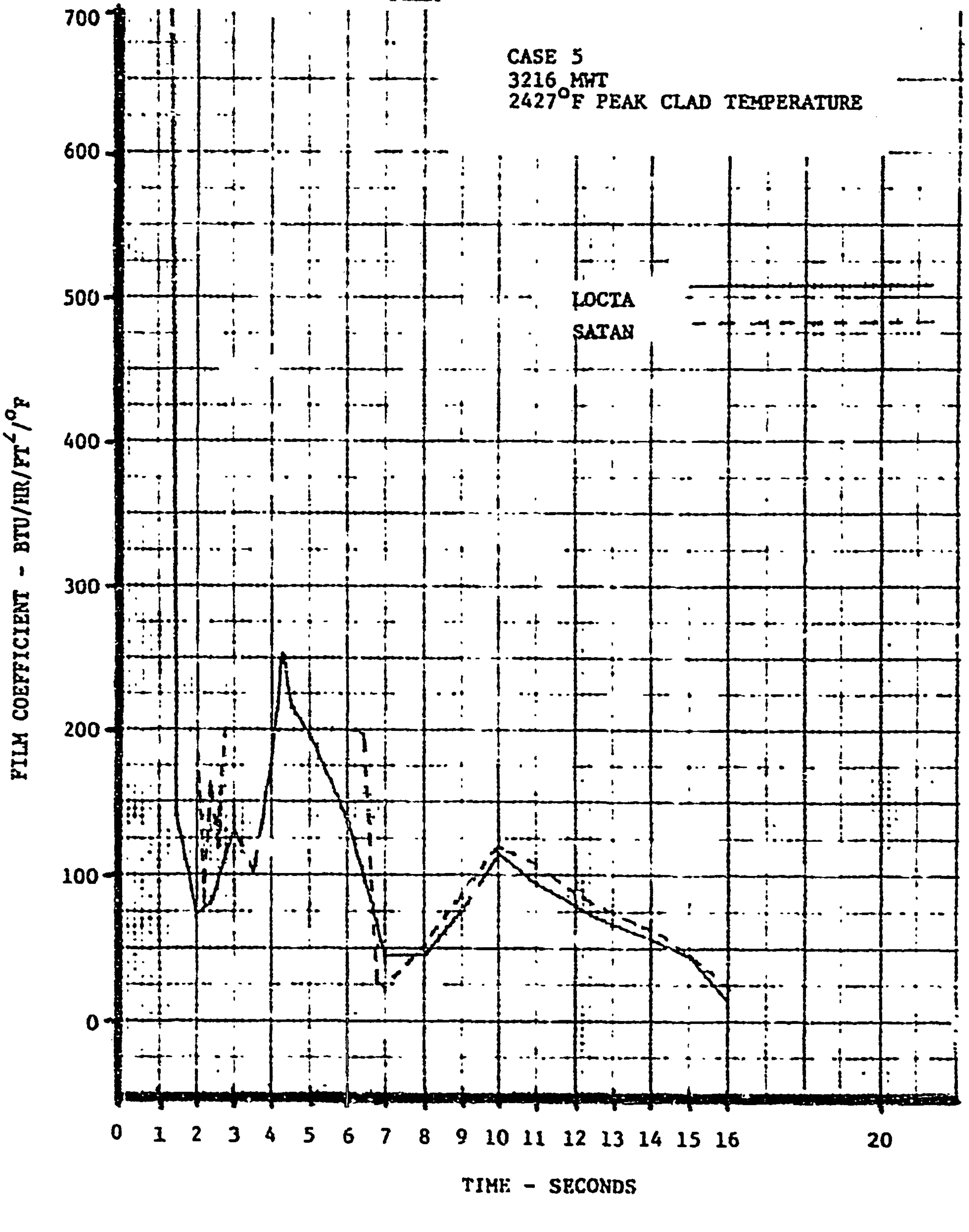


FIGURE 14.12-9

Supplement 13
8/70

INDIAN POINT UNIT NO. 2

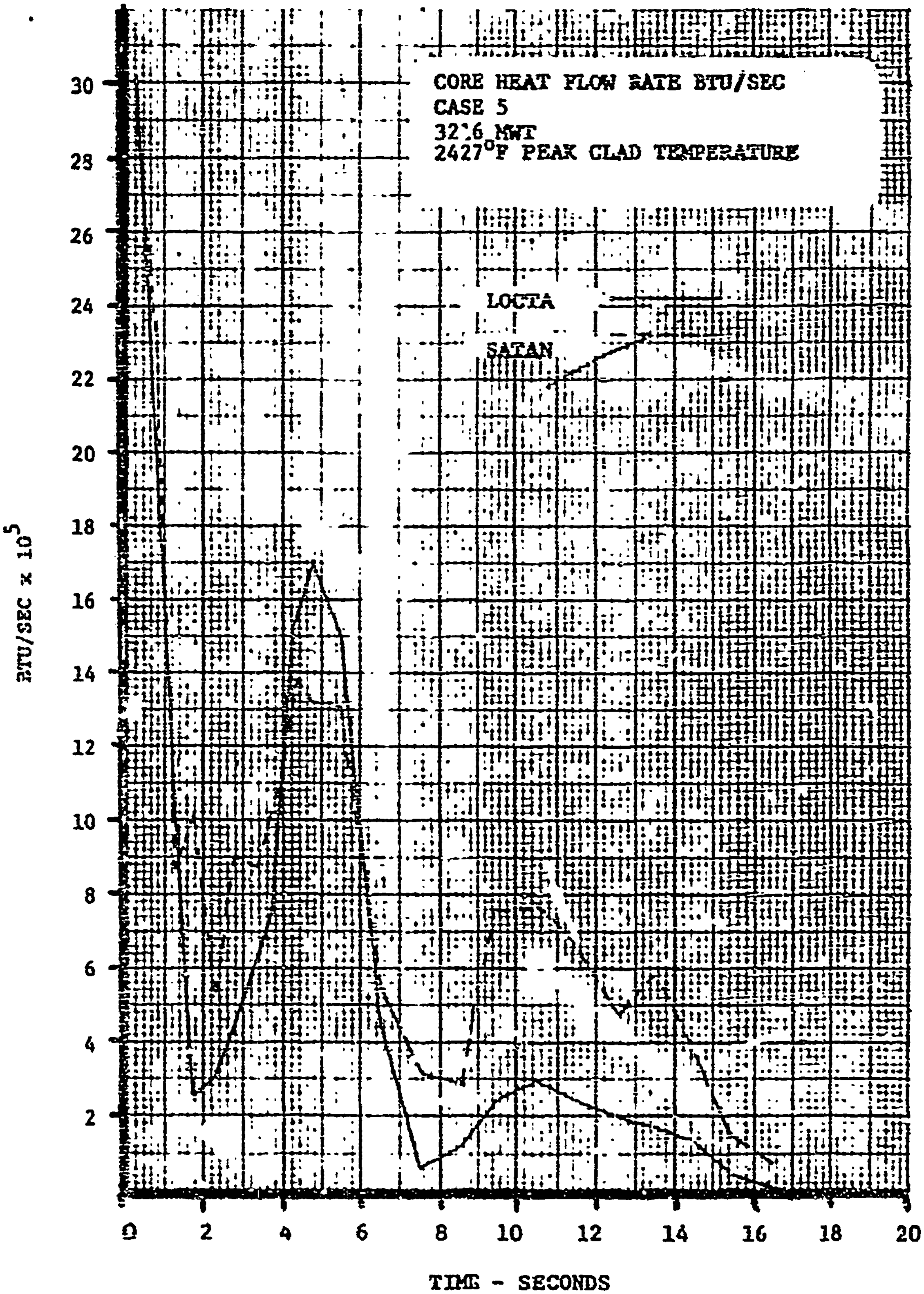
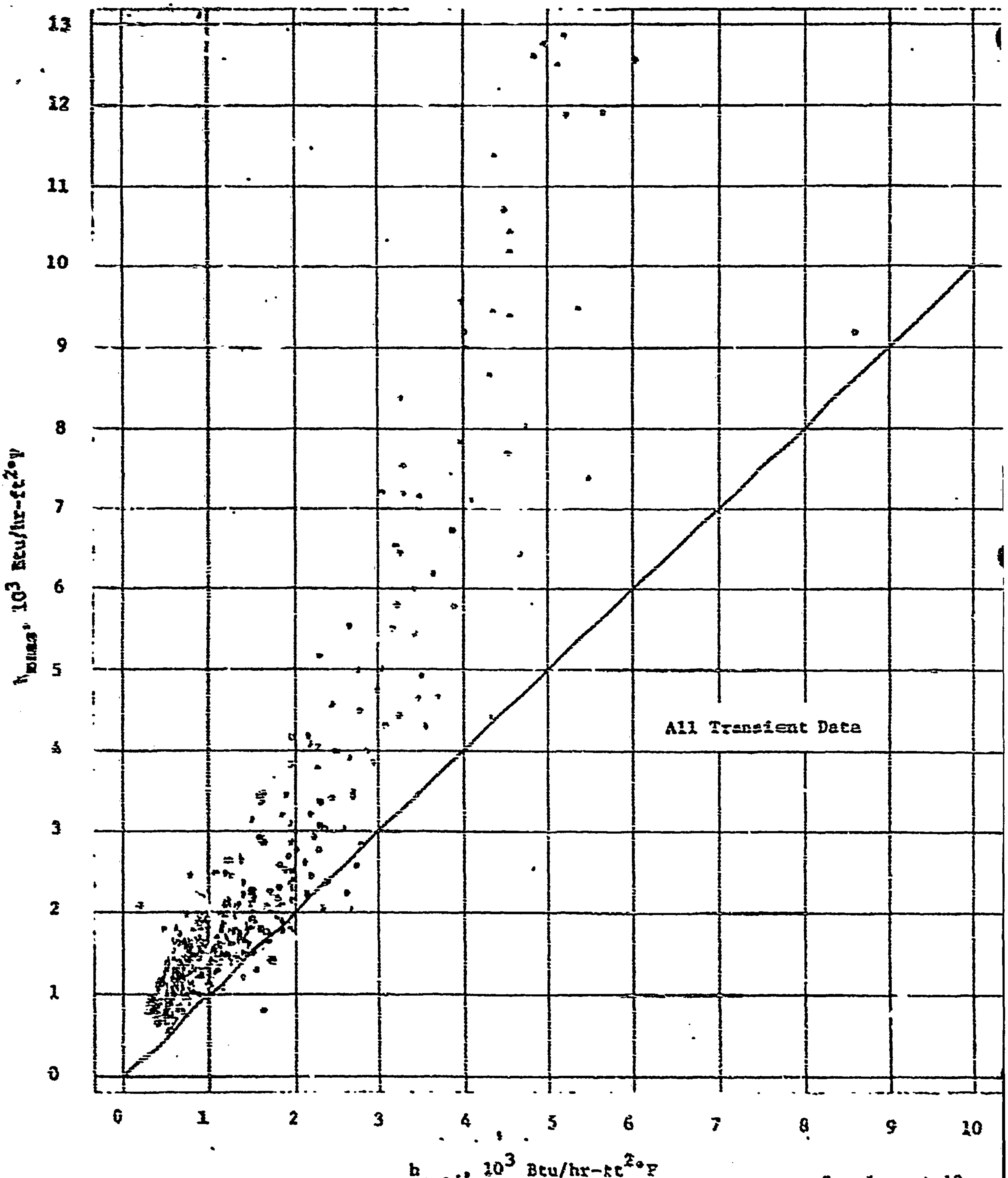


FIGURE 14.12-
Supplement 13
8/70

Figure 14.12-11



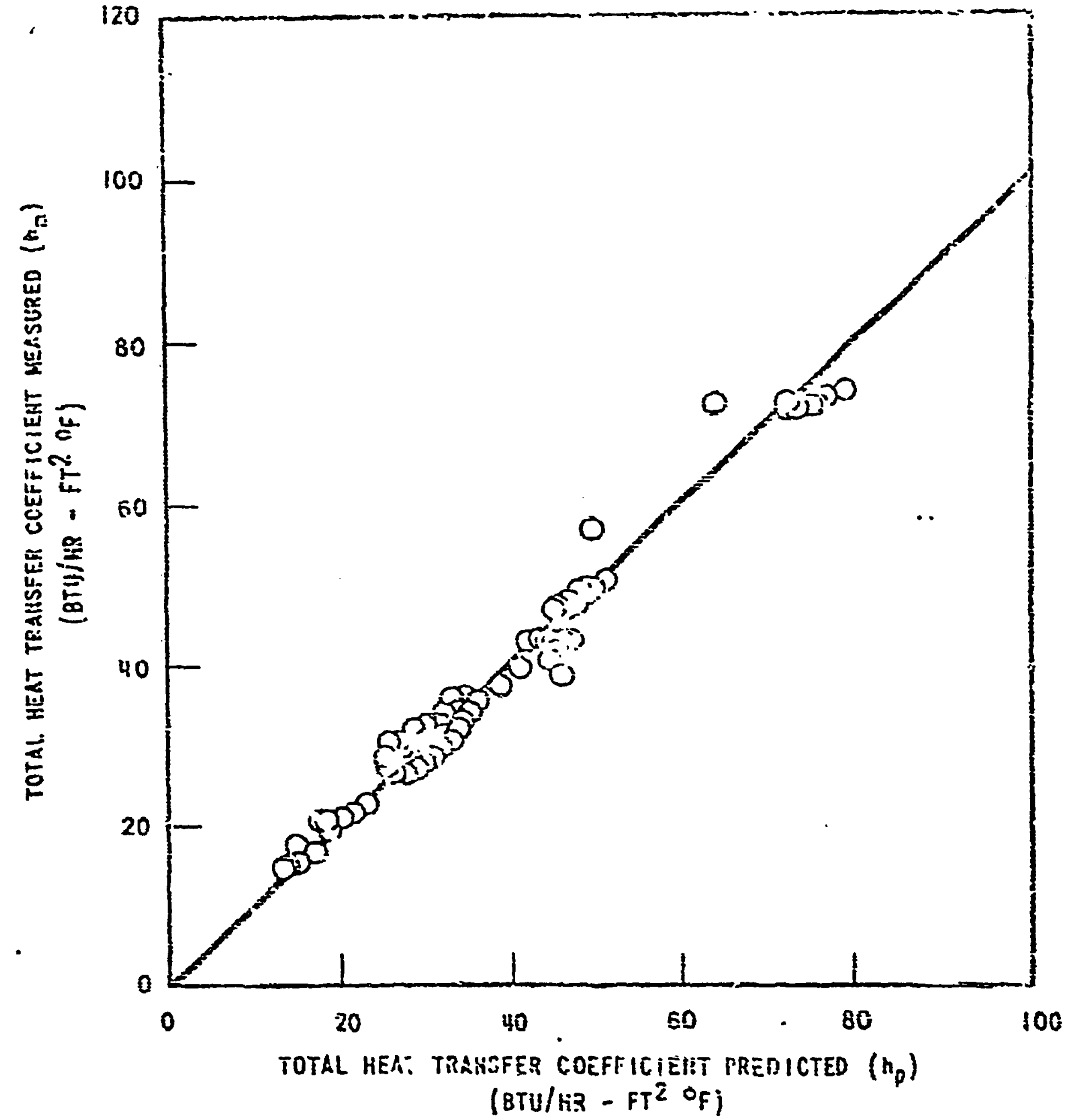


FIGURE 14.12-12 University of Michigan Heat Transfer Test - Turbulent Flow Data - Total Heat Transfer Coefficient - Low Pressure Heat Transfer Test

QUESTION 14.13

In the same manner, provide the results of your evaluation of a 29 inch ID, double-ended, hot leg pipe rupture.

ANSWER

The double ended hot leg break was analyzed with the multi-note SATAN analysis at a core power of 3216 MWt and the resulting peak clad temperature was less than 1500°F.

Figure 14.13-1 presents the SATAN model and the analysis while Table 14.13-1 presents the pertinent SATAN results.

TABLE 14.13-1

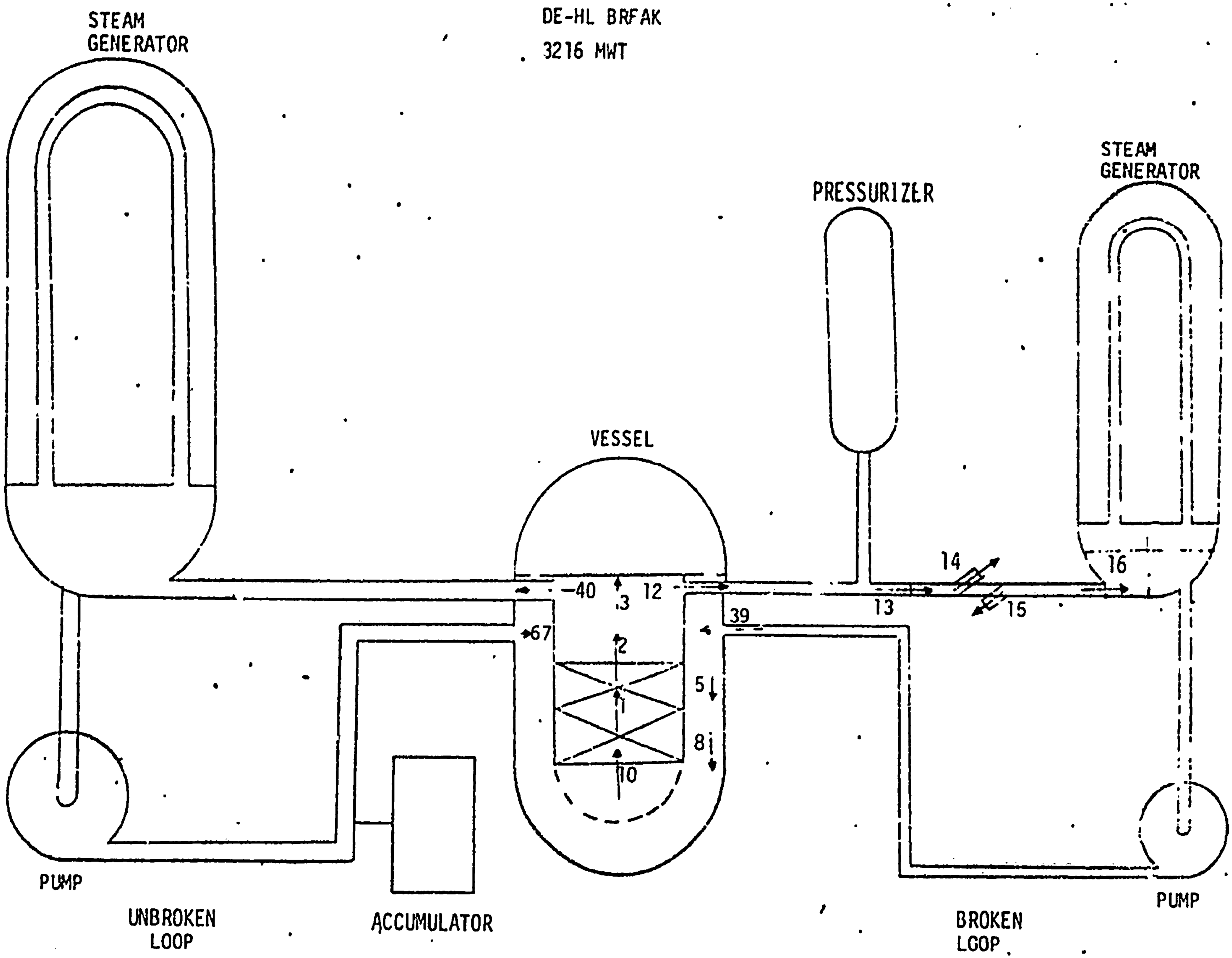
INDIAN POINT UNIT NO. 2
DE-HL BREAK (3216 MWt)
FLOW RATE (LBS/SEC)

TIME (SEC)	CORE INLET	CORE MIDPLANE	CORE EXIT	UPPER PLENUM	DOWNCOMER	DOWNCOMER	UNBROKEN LOOP HOT LEG	UNBROKEN LOOP COLD LEG	BROKEN LOOP HOT LEG	BROKEN LOOP COLD LEG	DOWNSTREAM OF BREAK	UPSTREAM OF BREAK	BREAK FLOW	BREAK FLOW
	10	1	2	3	5	81	12	39	40	67	16	13	19	15
0	37311.0	37311.0	37311.0	37311.0	37311.0	37311.0	9327.7	9327.7	27983.0	27983.0	9327.7	9327.7	0	0.
0.5	116001	12683.9	25217.4	50426.5	11427.1	1458.9	2[illegible]557.7	-15212.5	33972.5	29800.1	-24676.9	38167.9	38305.5	25173.5
1.0	7525.8	19901.8	22653.2	38155.5	7529.1	7508.8	26035.8	-19532.1	24670.8	27217.3	-23575.9	37213.3	37393.0	23204.3
2.0	11305.9	11374.0	12033.2	17409.7	11216.2	11246.2	22752.5	-17310.2	9151.7	29604.7	-22813.3	30974.2	31119.7	23130.8
3.0	12357.1	12963.4	13017.2	13904.2	12189.9	12266.3	14863.2	-13802.5	2954.9	20041.5	-19971.8	24816.6	24895.7	20229.7
4.0	9186.	9799.5	10277.1	11494.9	9036.2	9125.5	15467.9	-10925.6	-3042.3	19993.0	-16812.9	17279.1	17273.6	17027.3
5.0	10361.4	11464.9	11664.1	11216.9	8169.5	-3608.8	1693[illegible]	-8789.9	-6438.2	2145.0	-13727.4	17219.0	17209.6	13882.1
6.0	3741.7	9755.0	5627.2	7717.1	-6827.3	3965.7	17793.[illegible]	-7576.9	-9791.8	1039.6	-11109.4	18008.4	18006.8	11210.9
7.0	6538.6	6719.2	5775.4	4268.1	-1064.9	861.8	1[illegible]37.4	-6932.8	-4340.8	4041.2	-9413.2	17547.0	17519.0	9458.4
8.0	3120.4	4038.2	4011.9	4895.8	-2598.5	-1031.2	17239.8	-6490.5	-11506.2	2300.9	-8207.9	17511.9	17511.7	8246.9
9.0	3361.8	3510.3	3646.9	4247.3	-1420.7	44.6	15772.8	-5[illegible]8.9	-9[illegible]76.5	2750.5	-7158.9	16097.0	16[illegible]60.1	7188.4
10.0	3111.4	3281.9	3419.9	3790.4	-1086.6	195.6	13737.8	-5083.1	-7576.5	2574.5	-6251.2	14004.4	14023.6	6277.3
11.0	931.1	1864.9	2340.6	2954.2	-714.3	674.9	11523.5	5357.9	-6380.8	2918.6	-5321.8	11814.3	11824.9	5352.7
12.0	284.1	-11.7	56.9	344.9	463.6	1907.5	8348.7	-4246.6	-5227.7	3258.5	-4498.2	8911.2	8936.8	4570.[illegible]
13.0	956.7	515.7	569.6	1274.7	400.5	2973.2	4489.0	-2817.2	-1208.3	816.6	-3906.1	5026.9	5044.3	3922.8
14.0	408.1	358.2	360.3	725.1	-135.8	1396.9	2406.7	-515.9	-689.7	1342.5	2651.7	2911.3	2914.9	2687.6
15.0	-9.8	-96.9	84.3	115.8	1425.4	1769.1	1817.8	-229.3	-758.3	715.7	-1262.8	2073.7	2075.6	1284.9
16.0	-37.3	191.5	155.4	250.3	206.7	-1595.7	1313.9	-282.9	-445.5	382.2	-770.2	1444.7	1450.2	786.2
17.0	-44.8	9.4	51.2	207.4	267.8	5202.2	858.8	-286.5	-257.6	275.3	-313.4	941.2	941.7	334.2

INDIAN POINT UNIT NO. 2

DE-HL BREAK

3216 MWT



QUESTION 14.14

Provide evaluations, using your latest analytical techniques to determine:

(a) the limiting loss-of-coolant break size for the rupture of (i) a cold-leg pipe, and (ii) a hot-leg pipe, for which assured core cooling is predicted.

ANSWER

As discussed in Supplement 12 to the Indian Point Unit No. 2 FSAR and the answers to questions 14.12, 14.13, and 14.15 of this supplement, the multinode analysis to describe the blowdown process and the conservative core thermal analysis are sufficient to assure adequate core cooling for all break sizes up to and including the double ended rupture of the Reactor Coolant Piping. It has been suggested by the AEC staff that the flow reversals and low pressure drops through the core which occur early in the large cold leg break transient introduce uncertainties (Question 14.15). Although with the current multinode analysis, these uncertainties are treated conservatively, an analysis has been performed showing a break size where these flow reversals do not exist early in the transient.

The case analyzed was a 0.5 ft^2 cold leg break. (Analyzed at 3216 MWt and 18.8 Kw/ft) Figure 14.14-1 presents the core flow, pressure, quality and heat transfer coefficients for the case, while Figure 14.14-2 presents the peak clad temperature for this case. In this case, the Reactor Coolant pumps were tripped at reactor trip. It is appropriate to provide a general description of this transient followed by a quantitative justification for the core flows.

Early in the transient, core flows are high because the Reactor Coolant pumps (coasting down after trip) in the three unbroken loops dominate the effect of the break flow in the broken loop. At 5.2 seconds a peak clad temperature of 1480°F is reached, because the core was assumed to go through DNB at 0.5 secs. Reactor Trip is completed at 2.9 secs and the core powers rapidly decrease. The high flow through the core continues to cool the core until the fuel clad approaches the coolant temperature at about 20 seconds. The combined action of pump coastdown and fluid inertia provides positive flow through the core until the influence of the cold leg break flow begin to dominate (65 seconds). There is a near stagnation region until about 100 seconds until the break flow causes a steady reverse flow through the core which gradually increases until the clad temperature rise is terminated at 118 seconds. After blowdown is completed, the clad temperature again begins to rise until the accumulators reflood the core at 145 seconds. At no time did the clad temperature exceed the first peak of 1480°F.

It is appropriate to point out that in the following parts of the transient even a considerable reduction in heat transfer coefficient would not leak to unacceptable peak clad temperatures. During the period, the Reactor Coolant pump coast dominates the transient (0-65 second), the fuel and clad are reduced to near coolant temperatures before 20 seconds, therefore heat transfer coefficients much lower than those used in the analysis could be used and the clad temperature would be at the coolant temperature before the flow stagnation occurs. During the stagnation period, and until the break causes reverse flow through the core heat transfer coefficient of zero could have been used and the peak clad temperatures would not have reached an unacceptable value. It remains then to show the core flow predicted are justified.

Figure 14.14-3 presents a sketch of the SATAN model used to analyze the 0.5 ft^2 break. A 70 control volume simulation was used for the analysis. Noted on the figure, are the control volumes that provide the most useful information for interpreting the transient. These are (control volume numbers in parenthesis):

1. Core Region Flow
   a. Inlet (10)
   b. Midplane (1)
   c. Exit (2)

2. Upper Plenum Flow
   a. Into upper plenum (3)
   b. Out of upper plenum unbroken loops (40)
   c. Out of upper plenum broken loop (12)

3. Reactor Vessel Inlet Flow
   a. Flow broken loop (39)
   b. Flow unbroken loop (67)
   c. Downcomer region (5, 8)

4. Break Area
   a. Break Flow (37)
   b. Upstream of break (36)
   c. Downstream of break (37)

The arrows on the figure depict the normal flow direction for each control volume. Table 14.14-1 presents the flows in each control volume as a function of time after the accident. Entries with a minus sign indicate the flow is reversed from its normal direction.

At the time of the accident the system is in its normal condition with 37,311 lbs/sec flowing through the core equally distributed through the four loops. For the unbroken loops, the flows listed are the total flows for the three loops. One second after the break, the flow through the break is 12,740.9 lbs/secs, which is supplied by 9781.5 lbs/sec in the normal flow direction and 2952 lbs/sec which has reversed in the broken loop. The flow from the unbroken loops exceeds the demand for break flow and an undisturbed flow through the core is maintained. This trend continues well into the transient. The ability of the three unbroken loop Reactor Coolant pumps to overpower the demand for break flow in the broken loop continues during the time the RC pumps are coasting down to a very flow flow at about 65 seconds. From this point onward, the break begins to dominate and after 100 seconds a steady reverse flow through the core is developed. This continues until the end of blowdown.

The 0.5 ft^2 cold leg break is the minimum break size for which this well behaved response is expected. For hot leg breaks, an even better response is expected because there is no tendency to reverse the flow in this accident and all four accumulators are available for delivery. The largest connecting pipe to the cold leg is 0.39/ ft , and the largest connecting pipe to the hot leg is 0.682 ft^2. It is therefore concluded that the analysis for the break of the largest connecting pipe to the Reactor Coolant System is not subject to the uncertainties as the large area cold leg breaks.

TABLE 14.14-1
INDIAN POINT UNIT NO. 2
0.5 FT^2 COLD LEG BREAK (3216 MWt)
FLOW RATE (LBS/SEC)

TIME (SEC)	CORE INLET	CORE MIDPLANE	CORE EXIT	UPPER PLENUM	DOWNCOMER	DOWNCOMER	UNBROKEN LOOP HOT LEG	UNBROKEN LOOP COLD LEG	BROKEN LOOP HOT LEG	BROKEN LOOP COLD LEG	DOWNSTREAM OF BREAK	UPSTREAM OF BREAK	BREAK FLOW
	10	1	2	3	5	8	40	67	12	39	37	36	37
0	37311	37311	37311	37311	37311	37311	27983	27983	9327.7	9327.7	9327.7	9327.7	0
1.0	29420.9	29620.8	29924.4	30828.9	[illegible]	[illegible]	28348.4	[illegible]	[illegible]	[illegible]	-2928.8	-2952.0	12790.7
5.0	20651.2	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	19392.9	[illegible]	[illegible]	[illegible]	2124.3	6763.2	8901.5
7.0	19622.7	19632.5	9837.1	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	6183.7	6921.7
10.0	15332.6	[illegible]	15802.2	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	5821.0
15.2	[illegible]	12740.9	12962.6	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	5217.7
23.2	5245.4	5495.4	6304.9	6898.7	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	-2314.9	[illegible]	1967.3	4286.8
24.2	[illegible]	4919.2	5234.9	5467.6	3292.2	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1482.5	4274.2
30.0	[illegible]	[illegible]	3349.4	3282.7	2728.9	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1525.1	4211.5
40.0	[illegible]	2992.6	[illegible]	2771.9	[illegible]	[illegible]	2033.3	[illegible]	810.4	-2809.8	[illegible]	[illegible]	4102.2
50.0	1558.2	1417.9	1444.3	1685.8	[illegible]	1096.7	[illegible]	[illegible]	[illegible]	-2394.1	2943.1	934.7	3900.5
60.0	550.5	[illegible]	[illegible]	868.4	-118.6	82.4	695.3	[illegible]	438.8	-2739.4	2785.7	[illegible]	3526.6
70.0	214.2	217.9	237.7	324.3	-499.8	-571.9	472.1	[illegible]	149.3	[illegible]	-2421.6	555.8	[illegible]
80.0	[illegible]	221.2	166.9	225.7	-510.4	-16.8	186.4	1155.4	[illegible]	-2106.9	-2122.9	[illegible]	2552.0
110.0	[illegible]	-96.2	-102.3	-170.5	-810.1	[illegible]	93.8	257.3	-77.0	-1277.4	-1279.4	135.8	1415.5
120.0	-357.5	-411.8	-445.9	-247.8	-586.9	-412.4	15.7	114.8	-149.6	-863.2	-874.7	78.3	957.4
130.0	[illegible]	-392.9	-699.3	192.3	7.1	217.2	38.5	295.2	-200.9	-405.3	-411.5	96.5	512.2

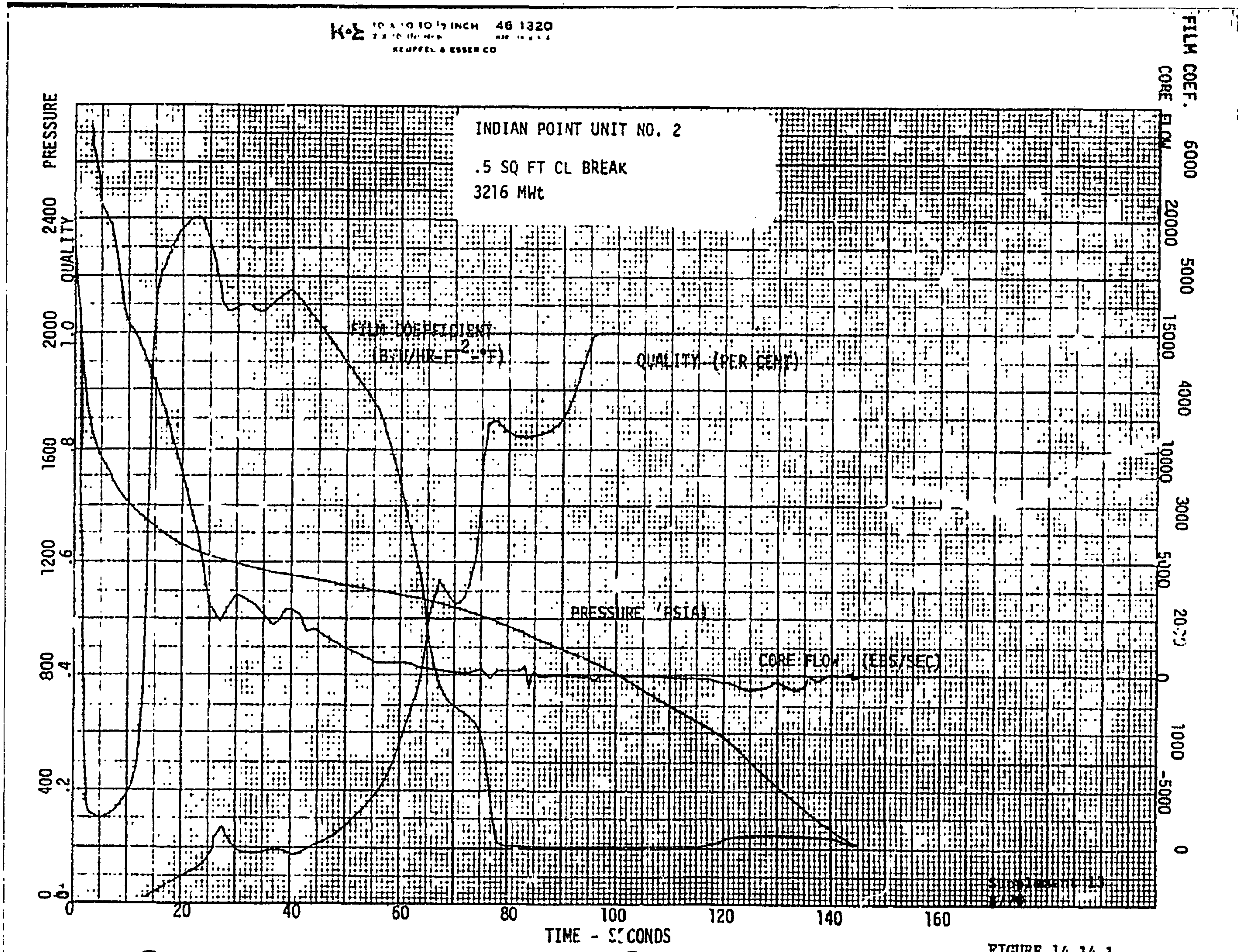
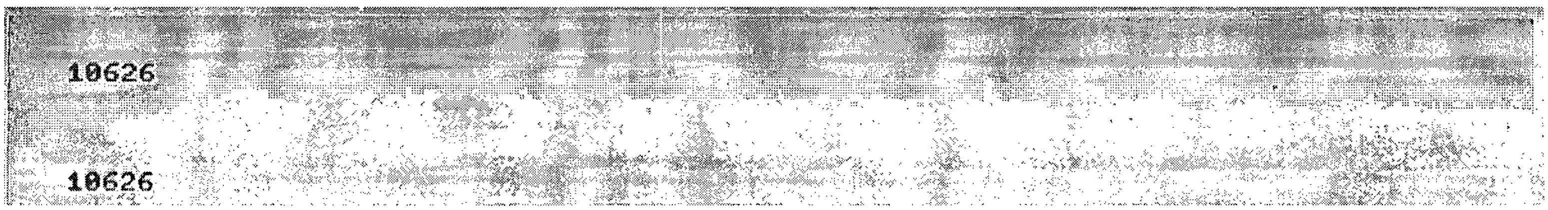


FIGURE 14.14-1

12

Q14.14(s)-1

to D-16



0.5 FT^2 BREAK CLAD TEMPERATURE TRANSIENT

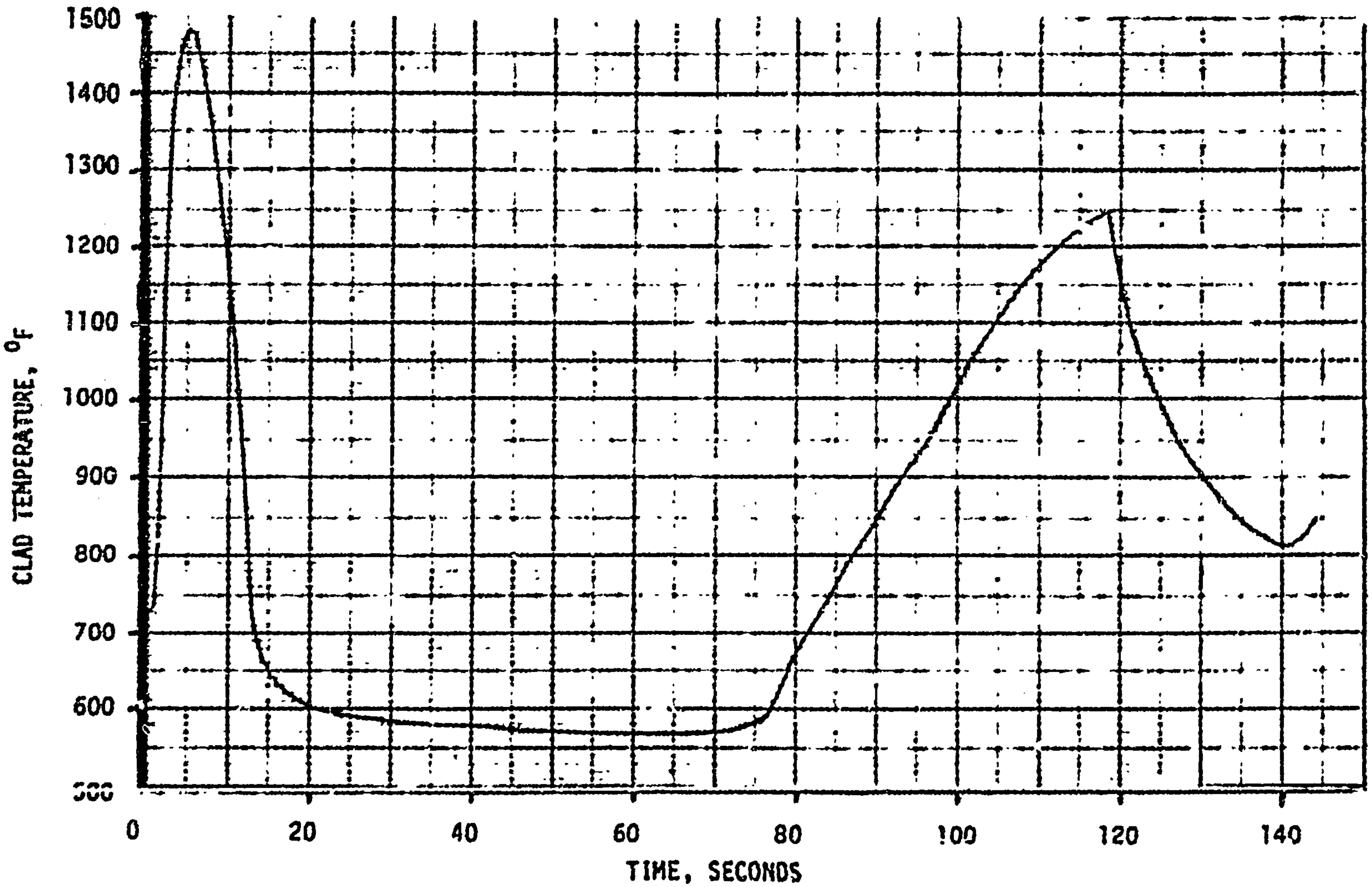


FIGURE 14.14-2

Supplement 13
8/70

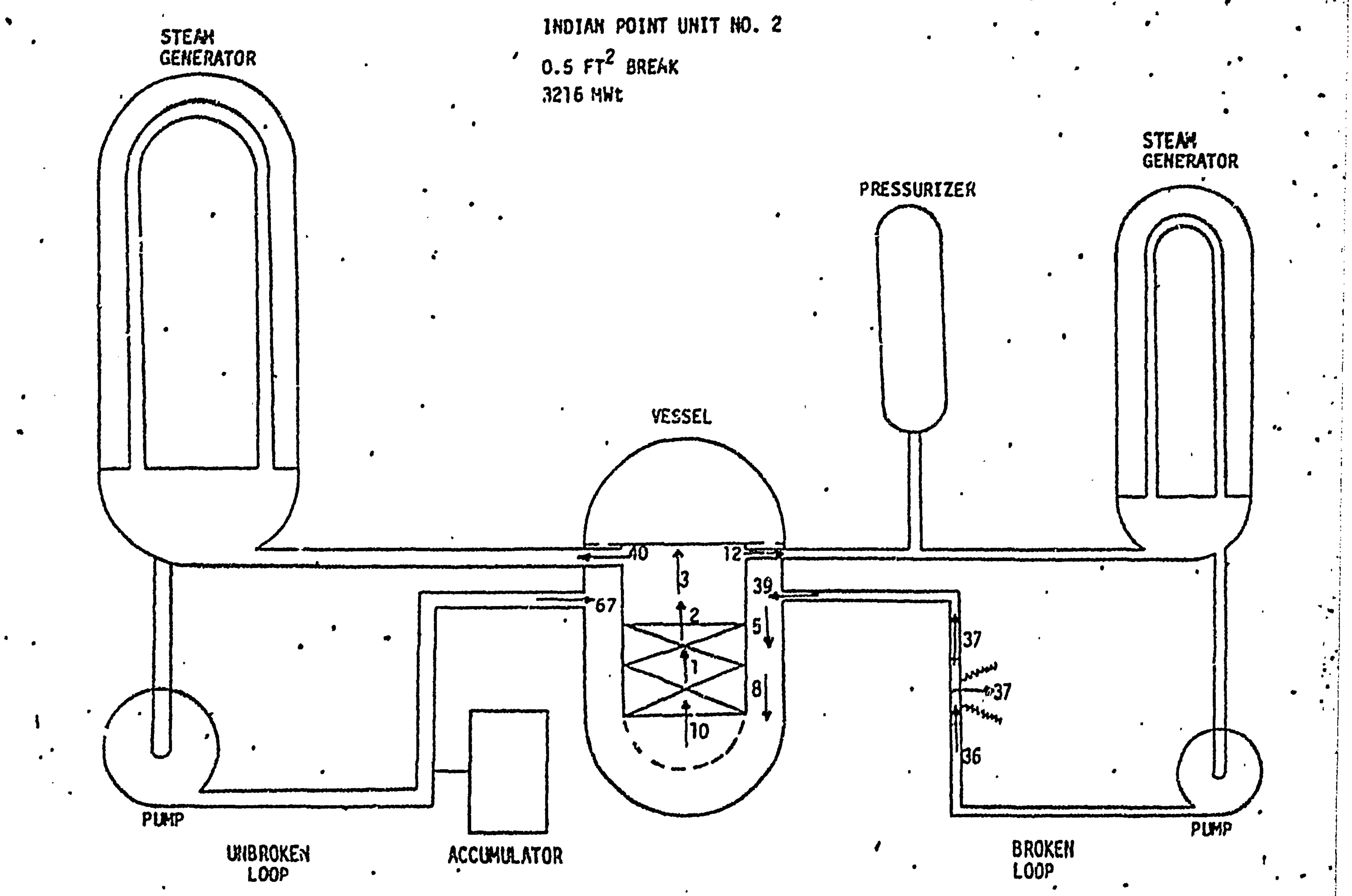


FIG. 14.14-3,
Supplement 13
8/70

QUESTION 14.14

Provide evaluations, using your latest analytical techniques to determine:

(b) the limiting reduced power level for which assured core cooling is predicted, for the double-ended rupture of (i) a cold-leg pipe, and (ii) a hot-leg pipe.

ANSWER

Core cooling can be assured for all break locations up to and including the double-ended breaks for powers up to and the maximum calculated thermal power of the reactor (32[illegible] MWt). This conclusion is supported by:

1. The multinode analysis presented in Supplement 12 of the IPP FSAR for a rating of 3216 MWt demonstrated with a conservative combination of assumptions that the peak clad temperatures are limited to 2457°F which is below the peak clad limits established by the Westinghouse Rod Burst program.

2. With this same analysis but taking credit for the expected accumulator performance (pg. 14 B-10 of Supplement 12) the peak clad temperature is below 2300°F.

3. The maximum clad temperature calculated (2015°F) at 102% of the application rating of 2758 MWt (18.8 Kw/ft) is well below the Westinghouse 2700°F maximum temperature criterion and the local hot spot metal water reaction (0.6%) is well below the 160% metal-water reaction criterion indicating considerable margin for flow blockage effects and calculational uncertainties.

QUESTION 14.14

Provide evaluations, using your latest analytical techniques to determine:

(c) Estimates of the volumes of core associated with local areas potential flow instability.

ANSWER

During the blowdown transient, the effect of lower heat flux, no inlet subcooling, and the increased resistance at the inlet due to two phase flow would all tend to decrease the likelihood of instability in a PWR core compared to the existing conditions during steady state operation.

Since fluid conditions are changing very rapidly with time, it is felt that the persistence of flow instability for any significant time period, is highly unlikely. Under steady state conditi[illegible] the reactor can have an 80% increase in the hot channel power [illegible] instability is calculated to occur. Therefore, we believe there are no volumes of the core having flow instability problems.

Question 14.15

Provide a summary discussion regarding your acceptance criteria for the ECCS functional performance. Your discussions should include:

(a) identification of any supporting information which has become available as a result of the Commission-sponsored emergency core cooling test programs.

Answer

ECCS FUNCTIONAL PERFORMANCE

To assure effective cooling of the core, the ECCS should perform in such a way to limit the clad temperature to below the melting temperature of the Zircaloy-4 and below the temperature at which core geometry distortion including clad fragmentation may occur. In addition the total core metal-water reaction is limited to less than 1% of the available zircaloy.

As specified in Supplement 12, to determine the limits on peak clad temperature and local metal water reactor above which effective cooling of the core may not be assumed, Westinghouse peformed the Simple Rod Burst Program[1]. The results of this test established that for conditions within the area of safe operation as shown in Figure 8 fuel rod integrity is maintained. Additional experimental data could further increase this area of safe operation.

Furthermore, Westinghouse performed the Multi-Rod-Burst Test program[2] to determine the geometry distortion caused by bursting and swelling of the fuel rods during a LOCA and the corresponding effect on the clad temperature transient. The test results indicated that the fuel rods burst in a staggered manner so that the maximum average assembly wise flow area blockage does not exceed 50%. The effect of the geometry distortion was calculated to increase the peak clad temperature during a LOCA less than 100°F. It follows that peak clad temperatures determined on the basis of no geometry distortion should be limited to 100°F below the limits presented in Figure 8.

The ability of the ECCS system to terminate the fuel rod temperature transient following a LOCA in a PWR core with and without blockage was established by the FLECHT test. The test results confirmed the conservatism of the Westinghouse loss of coolant accident analyses and yielded the following major conclusions.

a. The heat transfer coefficient increases with increasing initial clad temperature.

b. The heat transfer coefficient increases with increasing flooding rate.

c. The presence of the flow blockage plate causes a significant increase in the heat transfer coefficients at locations immediately downstream of the plate, whereas the plate did not affect heat transfer at the other locations in the bundle.

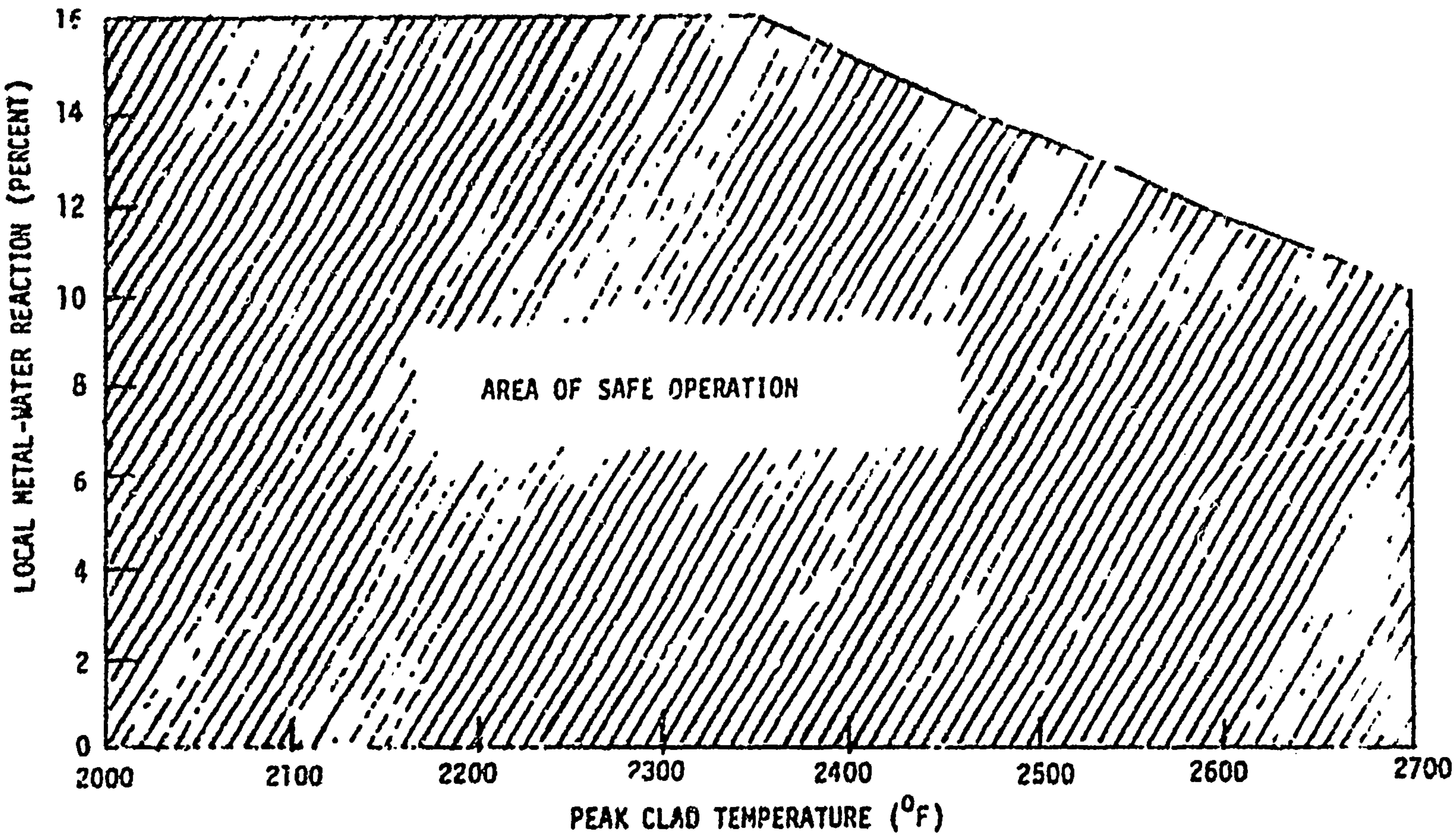


FIGURE 8 - PEAK CLAD TEMPERATURE AND METAL WATER REACTION LIMIT FOR A LOSS-OF-COOLANT ACCIDENT

Question 14.15

Provide a summary discussion regarding your acceptance criteria for the ECCS functional performance. Your discussions should include:

(b) an assessment of the adequacy of your analytical techniques, including the SATAN code, to accurately predict core behavior during the loss of coolant accident. A discussion should be presented on each area of uncertainity, with an estimate of the probability for more or less adverse consequences than those predicted.

Answer

Supplement 12 to the Indian Point Unit No. 2 FSAR presented a parametric survey that was performed to determine the most conservative combination of assumptions used in the SATAN Code. The use of the multi control volume SATAN analysis provided a unique opportunity to perform detailed parametric surveys of the important phenomena affecting the blowdown process.

In the SATAN analysis the control volumes were selected so that the system was divided into broken and unbroken loops to provide a better description of the blowdown flow distribution through the systems. In addition, the two control volumes used to describe the core region now provides a better description of the interaction between core heat transfer and blowdown. These capabilities were not available in the FLASH model previously used. The parametric analyses were:

1. Heat transfer from the core to coolant during blowdown.

2. Reactor Coolant Pump Characteristics

3. Steam Generator Heat Transfer

4. Loop Resistance and Break Location

5. Accumulator Performance

These studies were run in series, and each parameter was varied sufficiently in each direction to establish the most conservative combination of assumptions. This was intended to provide a conservative limit of credit for blowdown heat transfer. The hydraulic parameters calculated by the SATAN Code are then input to the LOCTA Code which conservatively calculated the rod heat transfer coefficient as discussed in the answer to question 14.12. On pages 14B-4 to 14B-6 of Supplement 12, a discussion of additional conservatism in the core cooling analysis is provided.

It is therefore concluded that because:

1. A parametric survey was performed to establish the most conservative application of the SATAN Code,

2 The SATAN results were applied conservatively in the LOCTA analysis where conservative heat transfer coefficients are calculated,

3. With more detailed core models to take credit for coolant motion and heat transfer due to non-uniform flux distributions greater blowdown heat transfer is expected,

4. The calculation was based on the design power distribution, (18.8Kw/ft) rather than the more reasonable peak level of 12 Kw ft, and since

5. The peak clad temperature for the application rating is 2070°f

the functional performance of the ECCS meets its acceptance criteria with substantial margin, and less adverse consequences than those predicted are expected.

QUESTION 14.16

Based on calculations supplied in response to Question 14.2 (Supplement No. 8) the radiation doses that would be received by personnel in the plant control room following a design basis loss-of-coolant accident do not meet the criterion currently required for approval of construction permit applications. This criterion requires that exposures be limited to 5 Rem whole body, or its equivalent, to any part of the body for the duration of the accident. Although we may not require absolute conformance to the dose criterion that we now apply to construction permit reviews, some modification of the design of the control room would be desirable so as to increase the assurance that the health and safety of the operating staff, and thus their efficiency and effectiveness in the event of an accident, would be protected in an acceptable manner. Accordingly, summarize those design modifications that could be made to reduce the radiation doses to approach those specified above using a spray inorganic removal coefficient of 4.5 hr^{-1} and a charcoal organic removal efficiency that is justified by present experimental data.

ANSWER

Steps have been taken to reduce the in-leakage to the Control Room area (combined Unit 1 and 2). However, because of the structural design of the Control Building, it is not possible to be guaranteed less than a total of 500 cfm leakage into the combined Control Room volume of 103,400 cubic feet. To further reduce the in-leakage would require complete redesign and rebuilding both Unit 1 and Unit 2 Control Buildings.

The Control Room ventilation and air conditioning system is equipped with a 1800 cfm recirculation system including treated charcoal filter beds. The charcoal filter and recirculation systems are as large as can physically be accommodated by the existing structures without major modification. However. additional charcoal filters would not significantly reduce the thyroid dose to the operators as the continuous assumed in-leakage accounts for most of this calculated dose.

Weld channel and penetration pressurization and seal water systems have been provided to eliminate or drastically reduce any leakage through the containment vessel after an accident.

The weld channel and penetration pressurization system, described in Section 6.6, provides part of a leaktight barrier to leakage from the containment by maintaining air pressure above the containment ambient temperature. The leakage criteria (specified for containment leak rate acceptance testing) of 0.1% of the containment atmosphere per day at the design basis accident condition is determined without this system in operation. Furthermore, to complete the leaktight barrier, piping penetrations are isolated from the outside atmosphere by the seal water system described in Section 6.5 of the FSAR. With these two systems in operation, leakage during the course of the pressure transient will be reduced significantly below the assumed 0.1% per day used in the design basis accident.

The weld channel and penetration pressurization system is continuously pressurized during operation and makeup requirements are monitored to determine the effectiveness of the boundary integrity. The seal water system is periodically tested during operation to assure operability and integrity of the system. Since these systems are not affected by loss of offsite power, the systems are assured to be available during the unlikely event of a design basis accident.

The exposure situation in the control room affects only a few operators, therefore protective clothing and full face respirators can and will be provided and will be readily available to the operators of both Unit 2 and Unit 1.

Exposure calculations for control room personnel were made using the following assumptions:

A. Containment

% of core/gap halogen inventory released (see discussion)

100% of core/gap noble gas activity released

λ_{spray} = 4.5 (DF = 100)

η_{filter} = 0% elemental

70% methyl

Flow Rate 24000 CFM

Leak Rate = 0.1% per day (0-24 HR)

= 0.045% per day (24-720 HR)

Free Volume = 2.6×10^6 Ft^3

B. Control Room

In-leakage Rate = 500 CFM

Free Volume = 103400 Ft^3

Flow Rate through filters = 1800 CFM

η_{filter} = 95% elemental

= 90% methyl

C. χ/Q

0-2 Hours	2.3×10^{-3} sec/m^3
2-24 Hours	1.15×10^{-3} sec/m^3
24-720 Hours	5.5×10^{-4} sec/m^3

D. Breathing Rates

0-8 Hours	3.47×10^{-4} m^3/sec
8-24 Hours	1.75×10^{-4} m^3/sec
24-720 Hours	2.32×10^{-4} m^3/sec

E. Gamma-Beta Energies

As per Table of Isotopes - 1968

F. Finite Cloud

1. Atmosphere

   At d = 30 meters, σy = 30 meters

2. Control Room

   σy = 17 meters

3. Reduction Factors per isotope from Meteorology and Atomic Energy - 1968 - pg 347

G. Shielding by Control Room

β Rays 100% Shielding

γ Rays 0% Shielding

Three separate dose analyses for the control room have been made. These analyses include the consideration of 50% of the fuel-clad gap inventory released to the containment. Using the most conservative λ_s of 4.5 HR^{-1} the total thyroid dose in 30 days is 11.1 Rem and the total whole body dose during this same period would be 1.8 Rem. Considering a 25% of the halogen inventory in the core available for release with a λ_s of 4.5 HR^{-1}, a value considered inappropriately low for this parameter (WCAP 7499-L), and assuming that these removal processes are only effective until a decontamination factor of 100 is attained, the thyroid dose in 30 days is 319 Rem and the total whole body dose is 52.5. The (10CFR100 site boundary thyroid) dose of 300 Rem is not exceeded until nearly 30 days giving sufficient time for the safety systems discussed previously to come into play, the use of airpacks or the replacements of operators. The seal water system is designed to terminate all leakage of radioactivity from the containment within one minute. The result of having this system operational in one minute would limit the control room dose based on the 25% release model (4.5 λ_s) to 5.6 Rem to the thyroid and a total whole body exposure of 0.4 Rem. 39 Rem of the whole body dose is contributed by beta radiation and may be readily shielded by protective clothing, leaving a dose of 13.5 Rem for the case where the seal water system is not assumed operational.

QUESTION 14.17

Document the type, manufacturer, and the flow characteristics of the chemical additive spray system nozzles used for post LOCA iodine removal.

ANSWER

Refer to page 6.3-10 of Section 6 of the FSAR as revised by Supplement 12.

QUESTION 14.18

Provide the design details of the charcoal adsorber system with respect to the type, weight, and distribution of charcoal in the filter units, and the arrangement of the filter units in each plenum.

ANSWER

Refer to page 6.4-18 of Section 6 of the FSAR as revised by Supplement 12.

QUESTION 14.19

With reference to the equipment to be installed in the containment vessel to handle post-accident radiolytic hydrogen, please provide the following information:

(a) Discuss more fully the suction arrangement of the air supply blower for the recombiner units and the connection with the ring distribution header from the recirculation cooling and filtration system. Provide a sketch of the arrangement, which includes indications of the local air circulation patterns to the recombiner suction point and the specific locations of the sampling lines.

ANSWER

Refer to the response to Question 6.8(a) of Volume 5 to the FSAR, page Q6.8(a)-2, as revised by Supplement 12. See also Figures Q14.19-1 and Q6.4-1 of Section 6 of the FSAR.

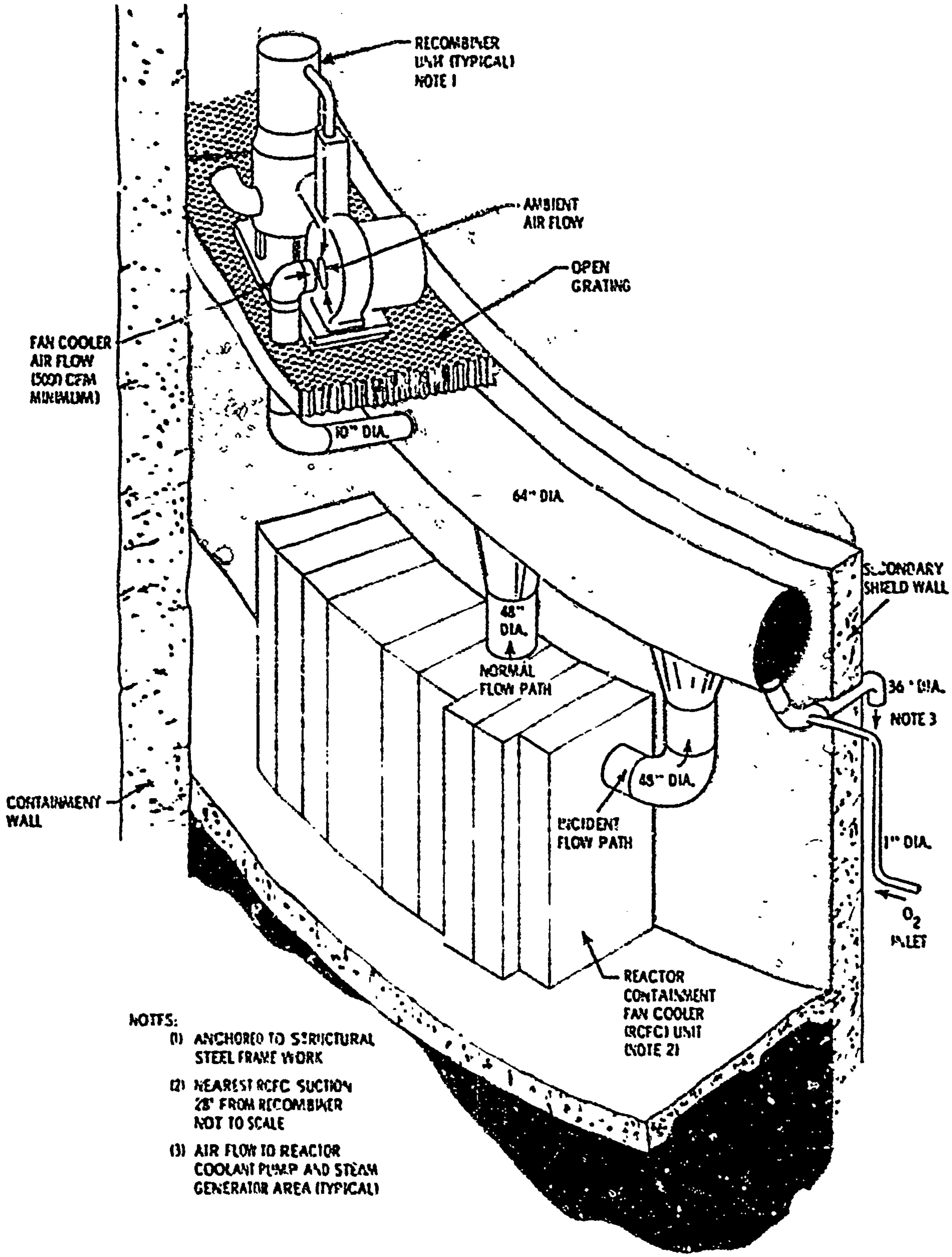


Figure 14.19 (a)-1. Supplement 13 8 70

QUESTION 14.19

With reference to the equipment to be installed in the containment vessel to handle post-accident radiolytic hydrogen, please provide the following information:

(b) Discuss the location selected for introducing oxygen makeup into the containment and the means whereby mixing of the oxygen is assured.

ANSWER

Refer to the response to Question 6.8(a) of Volume 5 to the FSAR, pages Q6.8(a)-1, 2, 3 and 4, as revised by Supplement 12.

QUESTION 14.19

With reference to the equipment to be installed in the containment vessel to handle post-accident radiolytic hydrogen, please provide the following information:

(c) Higher than ambient hydrogen concentrations (arising from unmixed gases evolved within the containment or from leakage of recombiner fuel lines) might exist near a recombiner unit prior to its startup. It is not clear from our review of the information submitted to-date that such local conditions could be detected prior to a recombiner startup or is precluded by the unit design. Our concern relates to potential flame propagation upon recombiner startup and to local and unit damage that might result from such propagation. It appears that the ability to sample locally, prior to recombiner startup and perhaps periodically during its operation, could be of advantage to the safe operation of the recombiner unit. Provide a discussion of this matter and state the design provisions, such as local air circulation rates, and sampling capability that currently exist in your design to preclude or to detect higher than ambient hydrogen concentrations in the regions near the recombiner units.

ANSWER

Refer to the responses to Questions 6.8(a) and 6.8b(1) of Volume 5 to the FSAR, pages Q6.8(a)-1 and 2 and 6.8b(1)-3, as revised by Supplement 12. Also see Figure Q14.19(c)-1 of Supplement 13 to the FSAR.

QUESTION 14.19

With reference to the equipment to be installed in the containment vessel to handle post-accident radiolytic hydrogen, please provide the following information:

(d) We understand that in contrast to your response to Question 6.8b(1) regarding the design of the post-accident containment sampling system, your plan is to employ a vacuum pump arrangement to provide sufficient flow in the sampling lines. Describe the sampling system design that will be installed. Support your description with suitable drawings.

ANSWER

Refer to the response to Question 6.8b(1) of Volume 5 to the FSAR, page Q6.8b(1)-1 and Figure 6.8(b)1-1, as revised by Supplement 12.

QUESTION 14.19

With reference to the equipment to be installed in the containment vessel to handle post-accident radiolytic hydrogen, please provide the following information:

(a) Your response to Question 6.8b(4) and (5) is not clear with respect to the post-installation testing and test frequency, and with respect to the processing setpoints you intend to establish for the recombiner units. Provide more specific information on the combustor and the diluent air flow settings and on the oxygen depletion range within which you would expect to operate.

ANSWER

Refer to the responses to Questions 6.8(a) and 6.8(b)4&5 of Volume 5 to the FSAR, pages Q6.8(a)-1, 3, 4, 5 and 6 and page Q6.8(b)4&5-1 to 6, as revised by Supplement 12.

QUESTION 14.19

With reference to the equipment to be installed in the containment vessel to handle post-accident radiolytic hydrogen, please provide the following information:

(f) We note that in order to permit "throttle-back" operation of the recombiner unit, bypasses will need to be used or certain adjustments in the protective devices for the flame failure system may need to be performed at the extern control station. Describe the actions that will be required to permit "throttle-back" operation including the types of bypass or adjustment actions that will need to be taken. State whether these operations will be simulated in the periodic testing programs.

ANSWER

Refer to the response to Question 6.8(a) of Volume 5 to the FSAR, page 6.8(a)-4, as revised by Supplement 12.

QUESTION 14.20

Provide a discussion of the potential for, and consequences of, missiles generated by failure of any of the main turbine-generator units planned to be operated at the site.

ANSWER

Refer to Appendix 14A of Section 14 of the FSAR as provided in Supplement 12.

PROJECT REORGANIZATION - DECEMBER 1969

This Section describes a reorganization in project management which is presently being implemented by Westinghouse. Reference is made to pages 1.3-1, 5.1.2-13, 5.1.2-14, and 5.1.2-15 and to Appendix B of the Final Safety Analysis Report for the information affected by this reorganization.

Westinghouse has formed a wholly owned subsidiary corporation, called WEDCO Corporation (WEDCO), to perform certain functions at the Indian Point site of Consolidated Edison. Westinghouse remains the prime contractor and will continue to exercise overall control and to have full responsibility for the Indian Point No. 2 project. WEDCO will perform, under Westinghouse, project management, engineering, quality assurance, construction and procurement functions for Indian Point No. 2. As more fully described herein, these functions were previously carried out by Westinghouse or United Engineers & Constructors (UE&C).

The entire Westinghouse senior management organization which, prior to the advent of WEDCO, was responsible for the Westinghouse effort at Indian Point No. 2, remains responsible. All other personnel within Westinghouse senior management who, prior to WEDCO, carried any responsiblity in any area for Indian Point Unit No. 2 continue to carry those responsibilities, regardless of the formation of WEDCO or changes in title or designation. Furthermore, WEDCO has behind it the full organization and strength of the Westinghouse Electric Corp, and Westinghouse engineering legal and other personnel continue to be available for the project.

The functional relationships among Consolidated Edison, Westinghouse, WEDCO, and UE&C are shown in Figure 1. The organization chart for WEDCO is shown in Figure 2, which depicts the combined Westinghouse/WEDCO organizational relationships.

Westinghouse WEDCO-United Engineers Relationship

Westinghouse retained UE&C as its architect-engineer-constructor to perform certain work and services in connection with the plant. Initially, UE&C performed services within its scope in the following areas:

(a) Design and Engineering
(b) Procurement
(c) Construction Management and Construction
(d) Quality Assurance (including Home Office Quality Control Engineering, Vendor Surveillance and On-Site Quality Control)

Westinghouse has removed items (b) and (c) from the scope of work to be performed by UE&C and has assigned these functions to WEDCO. In these areas, however, UE&C is providing qualified personnel to assist in effectuating the transition of work to Westinghouse and WEDCO.

UE&C continues to have responsibility for all of the design and engineering functions and all of the quality assurance functions, including home office quality control engineering, vendor surveillance and on-site quality control, for which it had responsibility prior to the advent of WEDCO. UE&C continues to have direct corporate responsibility to Westinghouse for all of the work within its present scope.

In its current organizational structure, WEDCO exercises a high level quality and engineering reliability function. This function includes the activities previously performed by the Nuclear Power Service Staff Resident Quality Assurance Engineer, and in addition includes the centralization and overall management for quality assurance activities previously performed by various organizations. This function will be carried out by a Reliability Manager who will be located at the site. The Reliability Manager will be responsible for surveillance visits to selected shops of suppliers. This function was previously delegated to the Westinghouse Nuclear Power Services Group. In addition, the Reliability Manager will continually audit the quality assurance efforts of UE&C. In effect, a new reliability management function over and obove those previously set forth has now been established while all existing organizational functions and responsibilities for quality assurance are being maintained.

The quality control functions previously performed at various Westinghouse organizational levels will continue to be performed. At the Westinghouse headquarters level, the staff quality assurance audit team will review periodically the quality control program for Indian Point No. 2 as it has done in the past. At the PWR Systems Division level, the quality control functions performed by that division for the nuclear steam supply system will continue as before.

Organization and Staffing of WEDCO

The management of WEDCO is as follows:

Directors

J. W. Simpson	-	Executive Vice President of Westinghouse and President of the Power Systems Company of Westinghouse.
J. C. Rengel	-	Vice President of Westinghouse and Executive Vice President of the Nuclear Energy Systems of Westinghouse.
T. STERN	-	General Manager, PWR Systems, Division of Westinghouse.
J. T. Stiefel	-	General Manager, Turnkey Projects of Westinghouse.
W. B. Lee	-	Formerly Project Manager Indian Points Projects - Westinghouse.

Principal Officers

J. T. Stiefel	-	President
W. B. Lee	-	Executive Vice President
A. A. Simmons	-	Vice President, Engineering
D. E. Anderson	-	Vice President, Construction
T. A. Gudiness	-	Treasure and Assistant Secretary
E. O. Pearson	-	Secretary

Figure 3 depcits the Westinghouse Project Organization for the Indian Point No. 2 project prior to the activation of WEDCO. Figure 4 shows the current Westinghouse Project Orgainzation.

As is apparent from the above list of directors and officers and from the charts, those persons responsible within the Westinghouse organization for various functions relating to Indian Point No. 2 remain responsible under the revised internal Westinghouse orgainzation. The inclusion of these detailed charts is intended to reflect the continuity of experienced personnel related to the Indian Point No. 2 Project. This does not, of course, constitute a representation that every single person identified on these charts will remain in the position specified for the duration of the project. Rather it is intended to show the management team which, in the main, will direct the project for its duration.

With respect to Westinghouse personnel, various persons employed by Westinghouse have been transferred to WEDCO, including all the Westinghouse project people assigned by Westinghouse to the site. In addition, all of the other project people (i.e., those not at the site) previously assigned to Indian Point No. 2 remain so assigned. Accordingly, the formation of WEDCO has not caused any Westinghouse person to be taken away from the project.

To staff WEDCO below the operating department heads, Westinghouse will in addition to its own personell utilize personnel drawn from other organizations such as UE&C, J. A. Jones Construction Company, Charlotte, North Carolina; Nuclear Services and Construction Company, Newport News, Virginia, a subsidiary of Newport News Shipbuilding & Drydock Company; Catalytic Construction Company, Philadelphia, Pennsylvania; and Mauchly Associates, Inc., Fort Washington, Pennsylvania.

As previously indicated, the entire staff of Westinghouse remains available to WEDCO for assistance and expertise as and when needed.

Consolidated Edison

The project reorganization described above does not in any way alter the ultimate responsibility of Consolidated Edison for the quality assurance program. There is no basic change in the Consolidated Edison program. However, the following minor procedural changes have been made in view of the existence of WEDCO:

(1) Consolidated Edison's monitoring function will include monitoring the activities of WEDCO.

(2) Consolidated Edison will forward the United States Testing Company quality assurance reports to Westinghouse and/or WEDCO;

(3) Consolidated Edison will contact Westinghouse and/or WEDCO for necessary corrective action.

PROJECT REORGANIZATION - MARCH 1970

This section describes a change which will be implemented by Westinghouse in the spring of 1970 in the project organization as described in various portions of the Final Safety Analysis Report and in the foregoing section entitled "Project Reorganization - December 1969".

The changes made in December 1969 involved the creation of WEDOC and the delegation to WEDCO by Westinghouse of certain functions at the Indian Point site previously carried out by Westinghouse or United Engineers & Constructors (UE&C). Following the December 1969 reorganization, UE&C retained the following functions within its scope of work as architect-engineer-constructor:

(a) Design and Engineering
(b) Quality Assurance (including Home Office Quality Control Engineering, Vendor Surveillance, and On-Site Quality Control).

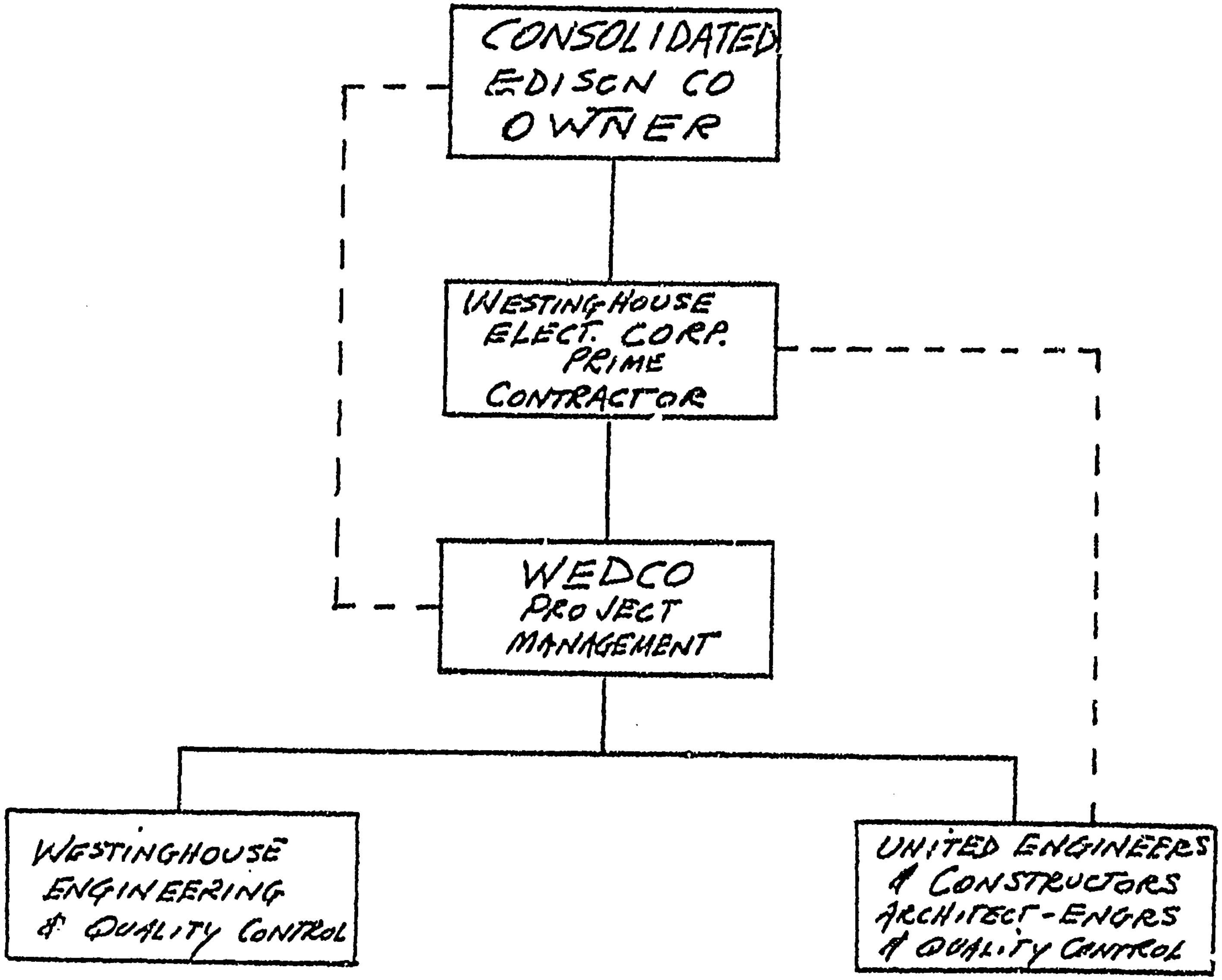
The current change consists of the removal of the vendor surveillance and on-site quality control portions of item (b) from the scope of work to be performed by UE&C, and the assigning of these functions to WEDCO.

It is anticipated that there will be little change of personnal involved in the transfer of the on-site quality control function. UE&C personnel who have been performing this function will be transferred from UE&C to WEDCO. With respect to the vendor surveillance function, the UE&C personnel presently discharging this function will not be so transferred since they are not used solely for WEDCO suppliers. Instead, WEDCO will employ a Manager of Vendor Surveillance and other personnel for this work. The transition in this respect will be gradual. New personnel will be phased in and the UE&C personnel will be utilized during the transition period to assure continuity of the surveillance program. The transfer will be made on a purchase-order-by-purchase-order basis, with UE&C personnel working with new personnel in performing the surveillance during the transition.

To assure that a level of quality assurance review will not be lost, the organization of the WEDCO reliability group is being structured to provide for an independent, internal audit of the two quality assurance functions currently being transferred to WEDCO. The Vendor Surveillance Group and On-Site Quality control Group each will report directly to the Reliability Manager. The activities of both the Vendor Surveillance Group and the On-Site Quality Control Group will be audited by a Systems Reliability Group. The Systems Reliability Group will report directly to the Reliability Manager to assure its functional independence. Figure 5, attached, shows these organizational relationships in chart form.

With the exception of the change described above, the roles of the various organizations involved in the construction of Indian Point 2 remain the same as previously described.

FIGURE №1
FUNCTIONAL RELATIONSHIPS



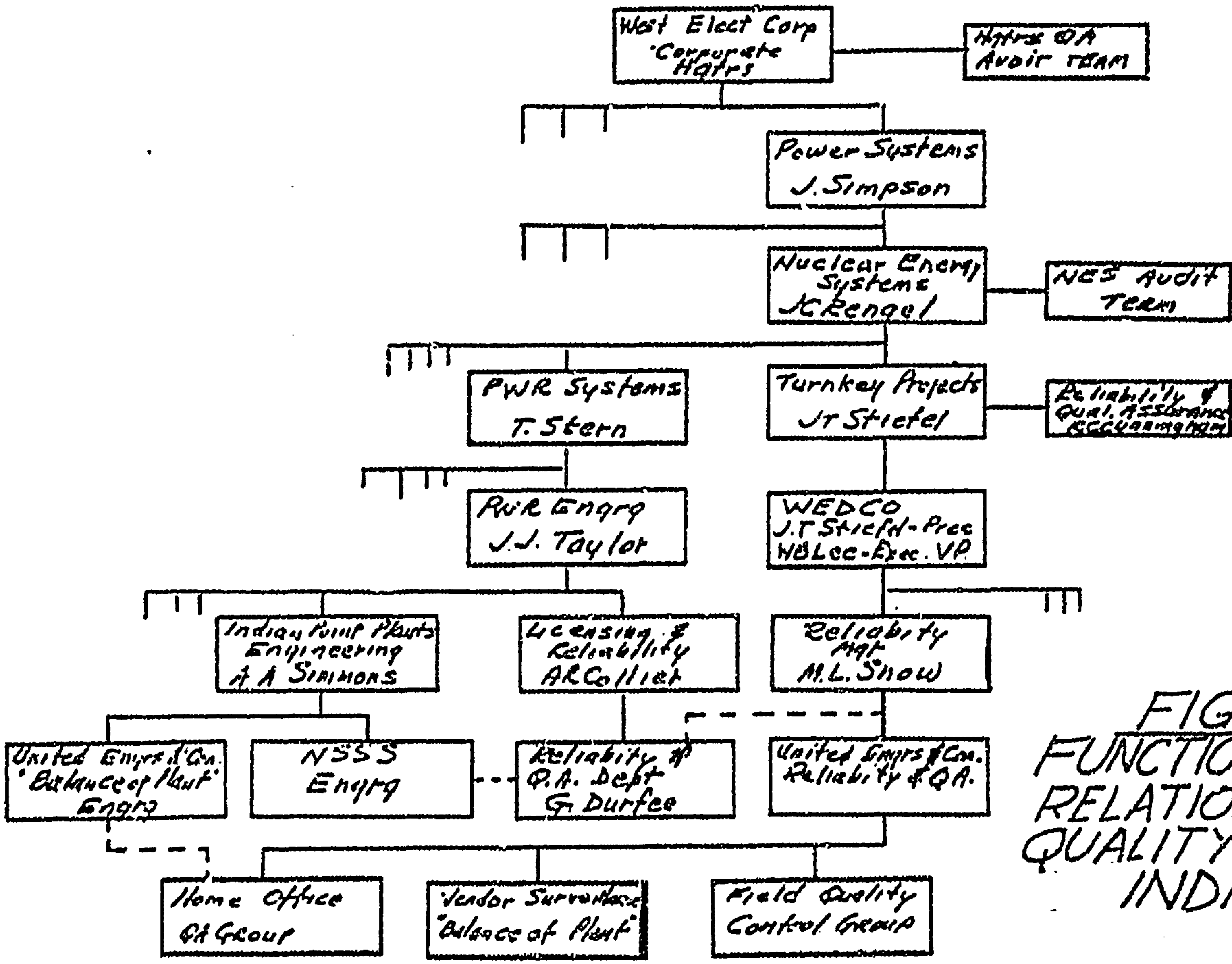
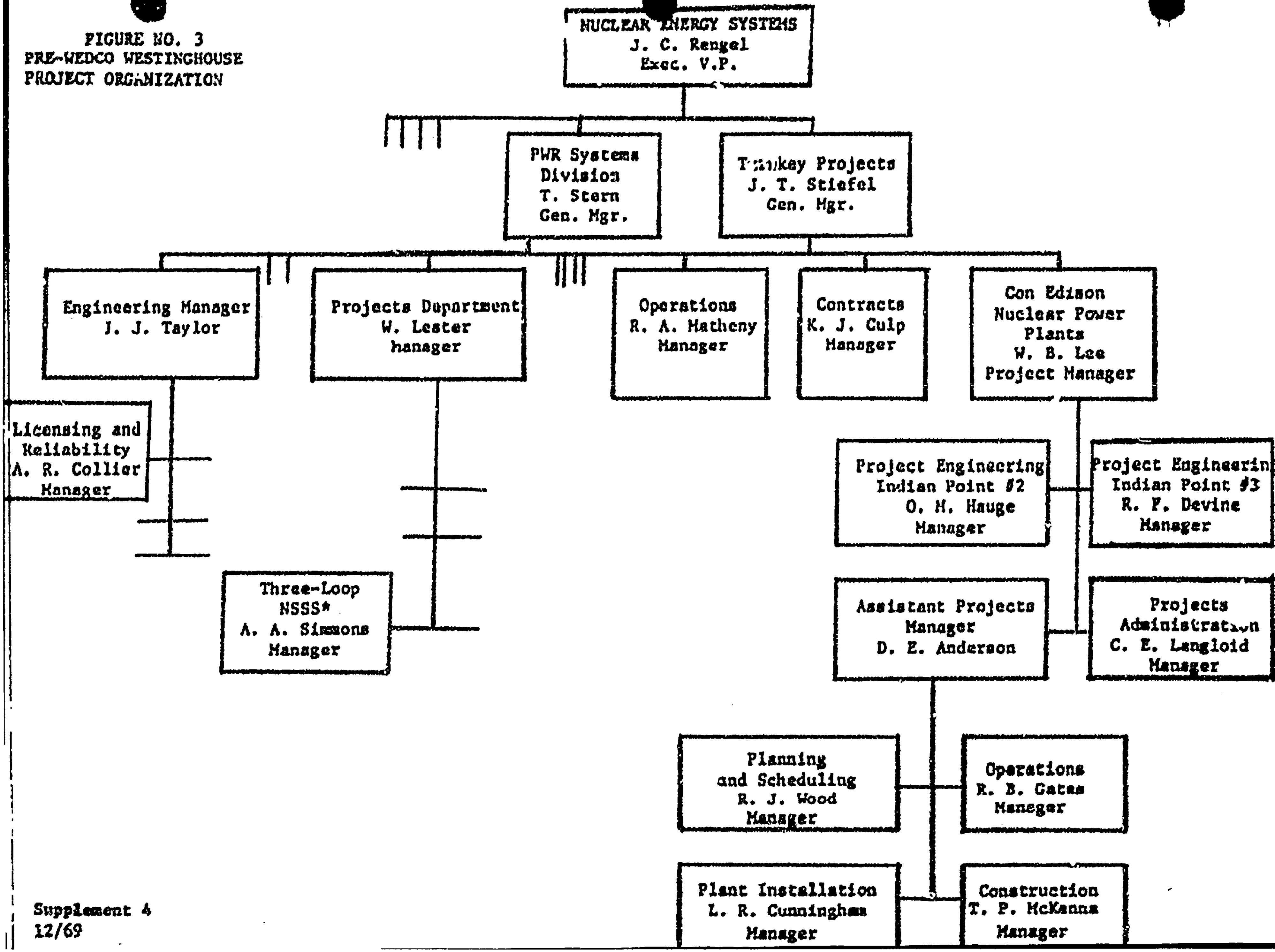


FIGURE Nº2
FUNCTIONAL RELATIONSHIPS FOR QUALITY-ASSURANCE INDIAN POINT

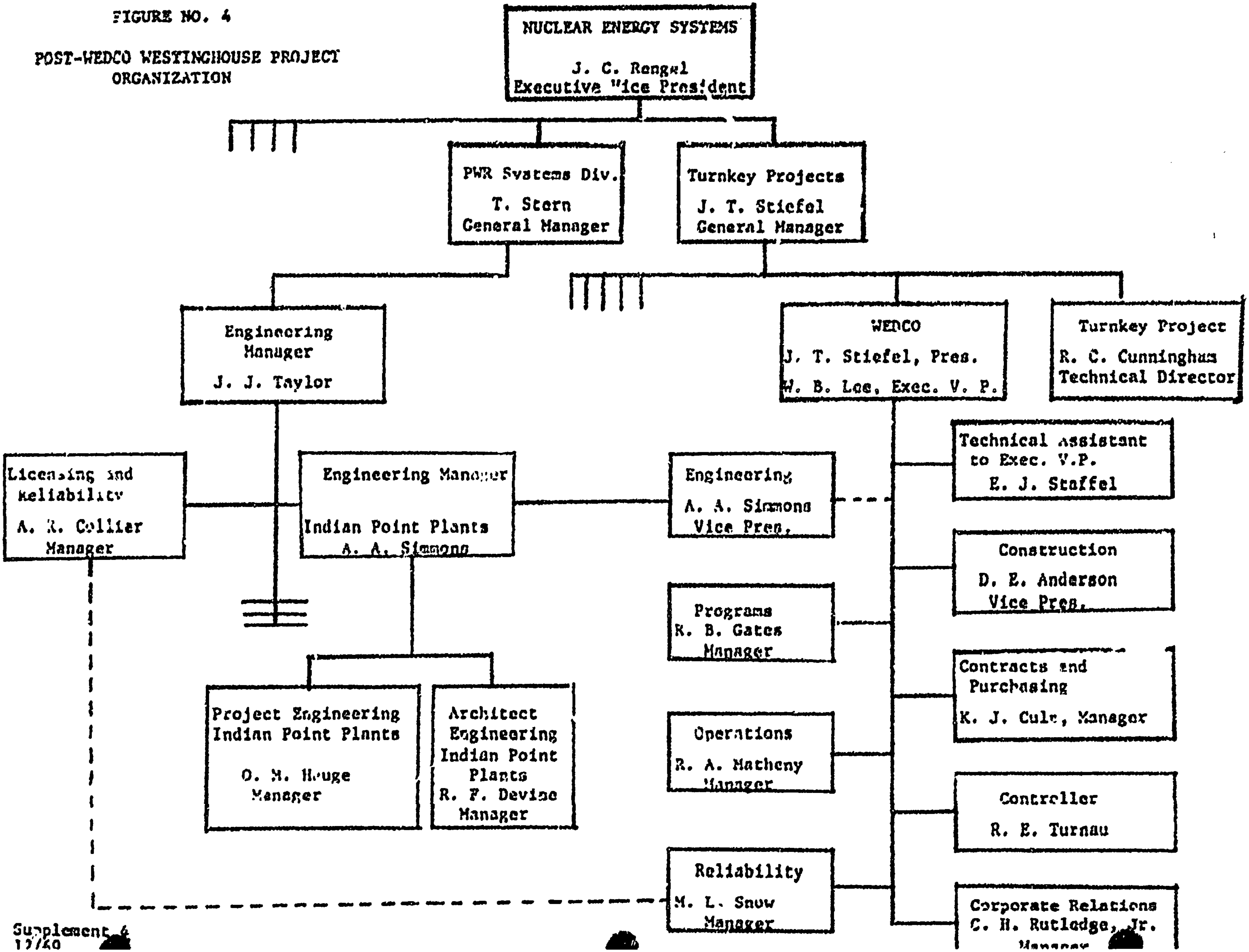
FIGURE NO. 3
PRE-WEDCO WESTINGHOUSE
PROJECT ORGANIZATION

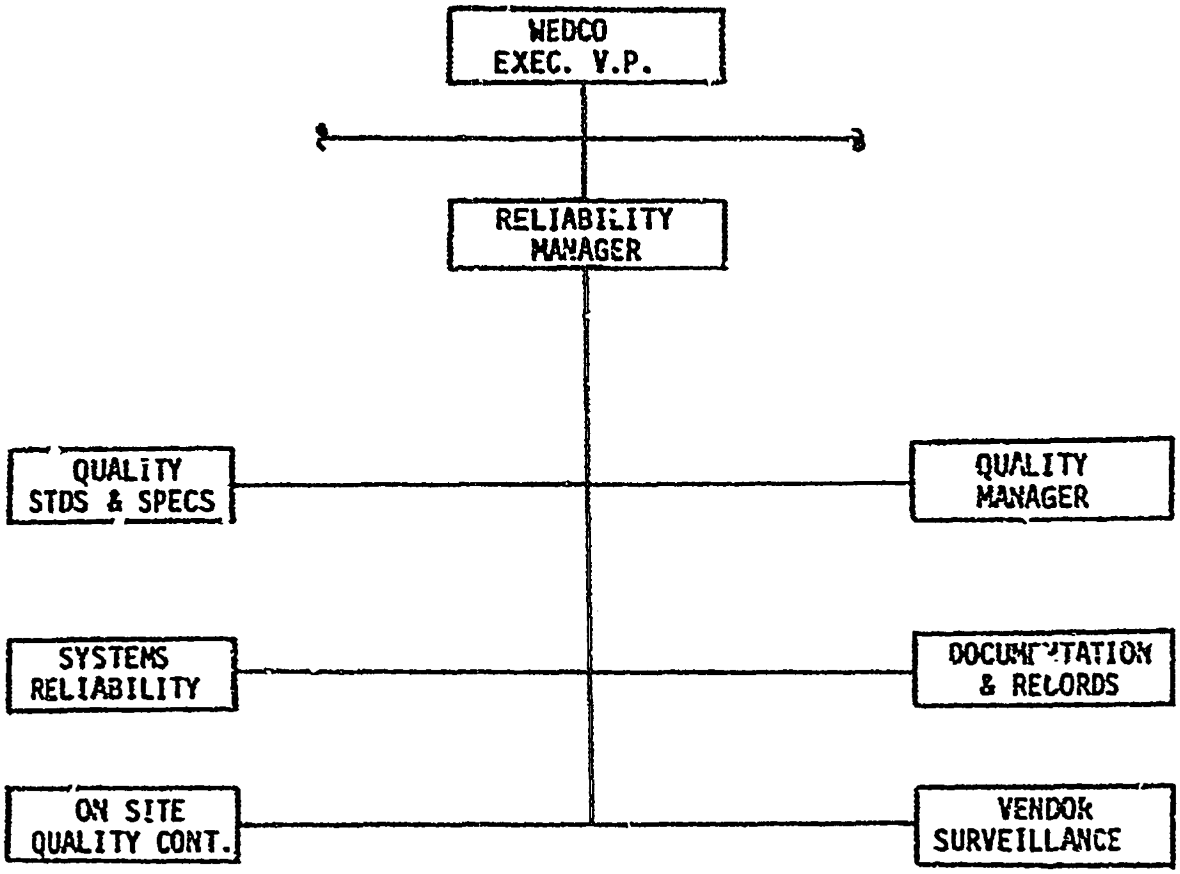


Supplement 4
12/69

FIGURE NO. 4

POST-WEDCO WESTINGHOUSE PROJECT ORGANIZATION





ORGANIZATION CHART
WEDCO
RELIABILITY GROUP

FIGURE 5
Supplement 8
4/70

WESTINGHOUSE NUCLEAR ENERGY SYSTEMS
UNITED ENGINEERS AND CONSTRUCTORS

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 2

CONTAINMENT DESIGN REPORT

March, 1969

S. Barnes
P. J. Gallagher
B. Scott
J. Slotterback
J. D. Stevenson

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

1.0 INTRODUCTION

1.1.0 PURPOSE & SCOPE OF REPORT

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

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 construction and testing procedures that were employed to ultimately construct the containment structure at the site.

1.2.0 FUNCTION OF CONTAINMENT STRUCTURE

The containment structure completely encloses the entire reactor and reactor coolant system and ensures that essentially no leakage of radioactive materials to the environment would result even if gross failure of the reactor coolant system were to occur. The structure will provide biological shielding for normal and accident situations.

The containment structure is designed to safely withstand several conditions of loading and their credible combinations. The limiting extreme conditions are:

a) Occurence of a gross failure of the reactor coolant system which creates a high pressure and temperature condition within the containment.

b) Coincident failure of the reactor coolant system with an earthquake or wind.

The design pressure and temperature of the containment will be, as a minimum, equal to the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any rupture of the reactor coolant system up to and including the hypothetical severance of a reactor coolant pipe. Energy contribution from the steam system is included in the calculation of the containment pressure transient due to reverse heat transfer through the steam generator tubes. The supports for the reactor coolant system will be designed to withstand the blowdown forces associated with the sudden severance of the reactor coolant piping so that the coincidental rupture of the steam system is not considered credible. In addition, the design pressure will not be exceeded during any subsequent long term pressure transient determined by the combined effects of heat sources such as residual heat and limited metal-water reactions, structural heat sinks and the operation of the engineered safeguards; the latter utilizing only the emergency electric power supply.

The design pressure and temperature on the containment structure will be those created by the hypothetical loss-of-coolant accident. The reactor coolant system will contain approximately 512,000 lbs. of coolant at a weighted average enthalpy of 595 Btu/lb. for a total energy of 304,000,000 Btu In a hypothetical accident, this water is released through a double-ended break in the largest reactor coolant pipe, causing a rapid pressure rise in the containment. The reactor coolant pipe used in the accident will be the 29-in. ID section because rupture of the 31-in. ID section requires that the blowdown go through both the 29-in. and the 27-1/2-in. ID pipes and would, therefore, result in a less severe transient.

Additional energy release was considered from the following sources:

a) Stored heat in the reactor core.

b) Stored heat in the reactor vessel piping and other reactor coolant system components.

c) Residual heat production.

d) Limited metal-water reaction energy and resulting hydrogen-oxygen reaction energy.

The following loadings will be considered in the design of the containment in addition to the pressure and temperature conditions described above:

a) Structure dead load.

b) Live loads.

c) Equipment loads.

d) Internal test pressure.

e) Earthquake.

f) Wind.

The containment structure is inherently safe with regard to common hazards such as fire, flood and electrical storm. The thick concrete walls are invulnerable to fire and only an insignificant amount of combustible material, such as lubricating oil in pump and motor bearings, is present in the containment.

Internal structures consist of equipment supports, shielding, reactor cavity and canal for fuel transfer, and miscellaneous concrete and steel for floors and stairs. All internal structures are supported on the mat with the exception of equipment supports secured to the intermediate floors.

A 3-ft. thick concrete ring wall serving as a partial radiation shield surrounds the reactor coolant system components and supports the polar-type reactor containment crane. A 2-ft. thick reinforced concrete floors covers the reactor coolant system with removable gratings in the floor provided for crane access to the reactor coolant pumps. The four steam generators, pressurizer and various piping penetrate the floor. Spiral stairs provide access to the areas below the floor.

The refueling canal connects the reactor cavity with the fuel transport tube to the spent fuel pool. The floor and walls of the canal are concrete. The floor is 5 ft. thick. The concrete walls and floor are lined with 1/4-inch thick stainless steel plate. The linings provide a leakproof membrane that is resistant to abrasion and damage during fuel handling operation.

1.3.0 CONTAINMENT DESCRIPTION

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-inch is attached to the inside face of the concrete shell to insure a high degree of leak-tightness. The design objective of the containment structure is to contain all radioactive material which might be released from the core following a loss-of coolant accident. The structure serves as both a biological shield and a pressure container.

The structure consists of side walls measuring 148 feet from the liner on the base to the springline of the dome, and has an inside diameter of 135-feet. The side walls of the cylinder and the dome is 4-ft. 6-in. and 3-ft. 6-in. thick respectively. The inside radius of the dome is equal to the inside radius of the cylinder so that the discontinuity at the springline due to the change in thickness is on the outer surface. The flat concrete base mat is 9-ft. thick with the bottom liner plate located on top of this mat. The bottom liner plate is covered with 3-ft. structural slab of concrete which serves to carry internal equipment loads and forms the floor of the containment. The internal pressure within the containment is self-contained in that the vector sum of the pressure forces is zero; therefore, there is no need for mechanical anchorage between the bottom mat and underlying rock. The base is supported directly on rock.

The basic structural elements considered in the design of the containment structure is the base slab, side walls and dome acting as one structure under all possible loading conditions. The liner is anchored to the concrete shell by means of stud anchors so that it forms an integral part of the entire composite structure under all membrane loadings. The reinforcing in the structure has an elastic response to all primary loads with limited maximum strains to insure the integrity of the steel liner. The lower 20 feet of the cylindrical liner is insulated to avoid deformation of the liner due to restricted radial growth when subjected to a rise in temperature.

2.0 CONTAINMENT STRUCTURAL DESIGN BASIS

2.1.0 DESIGN LOAD CRITERIA

The following loads were considered to act upon the containment structure creating stresses within the component parts.

2.1.1 DEAD LOADS

Dead load consists of the weight of the concrete wall, dome, liner, insulation, base slab and the internal concrete. Weights used for dead load calculations were as follows:

a) Reinforced Concrete : 150 lb/ft^3

b) Steel Lining : 490 lb/ft^3 using nominal cross-sectional area

d) Insulation : 6 lb/ft^3 including stainless steel jacket.

2.1.2 OPERATING LIVE LOADS

Operating live loads consist of the weight of major components of equipment in the containment. Equipment loads were those specified on the drawings supplied by the manufacturers of the various pieces of equipment.

All major pieces of equipment are supported on the 3'-0" base slab or on the interior concrete, which in turn bears directly on the 9'-0" mat.

Item		Flooded Operating Weight, lb.
Pressurizer	-1	346,000
Steam Generators	-4	3,746,000
Reactor	-1	
a) Vessel		868,000
b) Internals		420,000
c) Piping		1,000,000
Reactor Pumps	-4	824,000
Accumulator Tanks	-4	529,000
175 Ton Polar Crane	-1	650,000
Ventilation Fans	-4	656,000
Reactor Coolant Drain Tank	-1	20,000
Pressure Relief Tank	-1	100,000
Other Misc. Equipment		100,000
		9,259,000

2.1.3 SNOW LOADS

Snow and ice loads have been applied uniformly to the top surface of the dome at an estimated value of 20 pounds per square foot of horizontal projection of the dome. This loading represents approximately 2-ft. of snow, which was considered to be a conservative amount since the slope of the dome tends to cause much of the snow to slide off.

2.1.4 CONSTRUCTION LOADS

A construction live load of 50 pounds per square foot has been used on the dome, but was not considered to act concurrently with the snow load.

A load equivalent to the weight of wet concrete, placed in sections during construction of the concrete dome, was used for the design of the stiffened dome liner plate. During the pressure test of containment, the concrete will crack and thereby relieve the effe. . of shirnkage and creep.

2.1.5 WIND LOADS

The American Standards Association "American Standard Code Requirements for Minimum Design Loads in Buildings and Other Structures" (A58.1-1955) designates the site as being in a 25 psf zone. 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 the containment by using a shape factor of 0.60, the recommended pressure becomes 26 psf. 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. For design, a uniform 30 psf basic wind load has been used from ground level up.

While not an initial design consideration, the containment has been investigated for tornado load effects as described in Appendix B.

2.1.6 OPERATING TEMPERATURE

The operating temperature assumed in the design of the containment structure is 120°F, with a -5F° outside winter temperature. Thermal loads induced in the containment as a result of operating temperature effects are composed of a) the steady state temperature gradient through the wall as shown in Figure 1.1 for Winter conditions for both the insulated and uninsulated portions of the liner and b) the effective load induced in the concrete shell as the concrete acts to restrain the steel liner when the mean temperature of the concrete differs from that of the liner.

2.1.7 CREEP AND SHRINKAGE LOADS

The containment structure has been investigated for end of life creep and shrinkage factor as follows:

(a) $k_{(creep)} = 0.22 \times 10^{-6}$ in/in/psi

(b) $k_{(shrinkage)} = 70 \times 10^{-6}$ in/in

The maximum stress induced in the steel reinforcement by this maximum condition is less than 4000 psi. Since the limiting case for design is accident pressure load which effectively cracks the concrete and places the reinforcement into membrane tension creep and shrinkage induced stress are not a limiting factor in design. During the pressure test of containment, the concrete will crack and thereby relieve the effects of shrinkage and creep.

2.1.8 SEISMIC LOADS

The ground acceleration for the operational basis earthquake was determined to be 0.1g applied horizontally and 0.05 applied vertically. These values were resolved as conservative numbers based upon recommendations from Dr. Lynch, Director of Seismic Observatory, Fordham University. A dynamic analysis has been used to arrive at equivalent design loads. Additionally, a design basis earthquake acceleration of 0.15 horizontaly and 0.10 vertical has been used to analyze for the no-loss of function.

A damping factor of 2 percent was assumed for the reinforced concrete structure for both earthquakes. The response spectra used were based on the Spectrum curves presented in Figure 9.1 of the PSAR.

2.1.9 ACCIDENT PRESSURE LOADS

The design basis accident pressure load is shown in Figure 5.1-8 of the FSAR as a function of time. This design value is at least 5 percent in excess of maximum calculated containment pressure as shown in Figure 14.3.6-8 of the FSAR.

2.1.10 ACCIDENT TEMPERATURE LOADS

The design basis accident containment temperature assumed in the design of the containment is also shown in Figure 5.1-8 of the FSAR as a function of time. This containment temperature induces loads in the concrete shell as the concrete acts to restrain liner thermal expansion. This thermal load effect on the liner is combined with pressure load effects to develop design basis accident design load requirements as a function of time. Accident temperature induced thermal gradients through the wall are not a factor in concrete shell design since the accident temperature effect penetrates approximately 10 percent of the containment wall thickness during the significant overpressure phase of the accident and the cracking of the concrete shell due to containment pressurization acts to relieve secondary stresses induced by thermal gradient effect.

2.1.11 LOADS AT PENETRATIONS

The effect, of growth of the liner due to accident conditions, has been considered in the design of penetrations and sleeves together with the effects of lateral loads due to thermal expansion of pipes, seismic motion, pipe break loads and pressure loads. In addition, stress concentration effects on large penetrations have been considered.

2.1.12 COMBINED FACTORED LOAD EQUATIONS

The design was based upon limiting load factors which were used as the ratio by which loads were multiplied for design purposes to assure that the loading formation behavior of the structure was one of elastic, tolerable strain behavior. The load factor approach was used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach 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, therefore, this approach places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The loads utilized to determine the required limiting capacity of any structural element on the containment structure are computed as follows:

a) $C = 1.0D \pm 0.05D + 1.5P + 1.0\ (T + TL)$ (2.1.1)

b) $C = 1.0D \pm 0.05D + 1.25P + 1.0\ (T' + TL') + 1.25E$ (2.1.2)

c) $C = 1.0D \pm 0.05D + 1.0P + 1.0\ (T'' + TL'') + 1.0E'$ (2.1.3)

Symbols used in these formulae are defined as follows:

C: = Required load capacity section.

D: = Dead load of structure and equipment loads.

P: = Accident pressure load as shown on pressure-temperature transient curves.

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

TL: = Load exerted by the liner based upon temperatures associated with 1.5 times accident pressure.

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

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

E: = Load resulting from operational basis earthquake.

T": = Load due to maximum temperature gradient through the concrete shell, and mat based upon temperature associated with the accident pressure.

TL": = Load exerted by the liner based upon temperatures associated with the accident pressure.

E': = Load resulting from design basis earthquake.

Load conditions a) indicates the 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.

Load condition b) indicates that the containment has the capacity to withstand loadings at least 25 per cent greater than those calculated for the design basis accident with a coincident operational basis earthquake.

Load condition c) indicates the containment will withstand loads at least equal to those calculated for the design bases accident coincident with a design basis earthquake. The Indian Point Unit No. 2 cont. ment has the capacity withstand loadings associated with the design basis accident and a coincident earthquake with essentially elastic response.

The loads resulting from design wind on any portion of the structure did not exceed those resulting from earthquake.

All structural components have been designed to have a capacity required by the most severe loading combination. The loads resulting from the use of these equations will hereafter be termed "factored loads."

The load factors utilized in these equations are based upon the load factor concept employed in Part IV-B, "Structural Analysis nd Proportioning of Members Ult'mate Strength Design" of ACI 318-63. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors in the design primarily provide for a safety margin on the load assumptions.

2.1.13 TEST PRESSURE LOADS

Internal pressure will be applied to test the structural integrity of the vessel up to 115 per cent of the design pressure of 47 psi. The maximum calculated pressures was 42 psi. For this structure the test pressure will be 54 psig.

2.2.0 STRESS, STRAIN OR DEFORMATION CRITERIA

The containment is designed such that under all factored load conditions the behavior of the structure will be in the small deformation elastic range. This behavior range is defined by the stress limits contained in the ACI-318-63 Code to include additional margin as provided by the capacity reduction factor, Ø.

2.2.1 CAPACITY REDUCTION FACTOR Ø.

The theoretical member capacity is lowered by the reduction factor Ø to recognize variation in quality of materials and permissible tolerances in bar and plate areas and section dimensions, as well as approximations inherent in theoretical analysis. In theory the capacity reduction factor should be divided into the calculated load effect to determine actual design load requirements. Since Ø is less than one this always results in a design load requirement in excess of calculated requirements.

As a practical matter in the design of this containment the capacity reduction factor has been applied as a multiplier to the theoretical stress criteria. This has the result of reducing the allowable stress as a function of the type of load being carried. The following Ø factors for both concrete and steel are used in design:

Ø = .95 (tension)

Ø = .90 (flexure)

Ø = .85 (diagonal tension, bond and anchorage)

2.2.2 ('ETE STRESS CRITERIA

The stress criteria governing behavior are as specified in Part IV-B of the ACI-318-63 Code. Specifically the code limitations on concrete compression, tension, shear strength with and without web reinforcement, bond and anchorage are followed. These values are further reduced by applicable capacity reduction factors.

2.2.3 CONCRETE REINFORCING STEEL

The calculated structural capacity of reinforced concrete sections is based on the specified minimum yield strength of the reinforcement using the design methods specified in Part IV-B of the ACI-318-63 Code. This limiting stress value is further reduced by the applicable capacity reduction factor.

2.2.4 STEEL LINER PLATE

The maximum steel stress is limited to 0.95 yield under all primary loading conditions. In regions of local stress concentrations or stresses due to localized secondary load effects the maximum liner strain is limited to 0.5 per cent. (Detailed finite element computer analysis has identified regions of high localized liner stresses which would not have been detected using conventional analytical techniques).

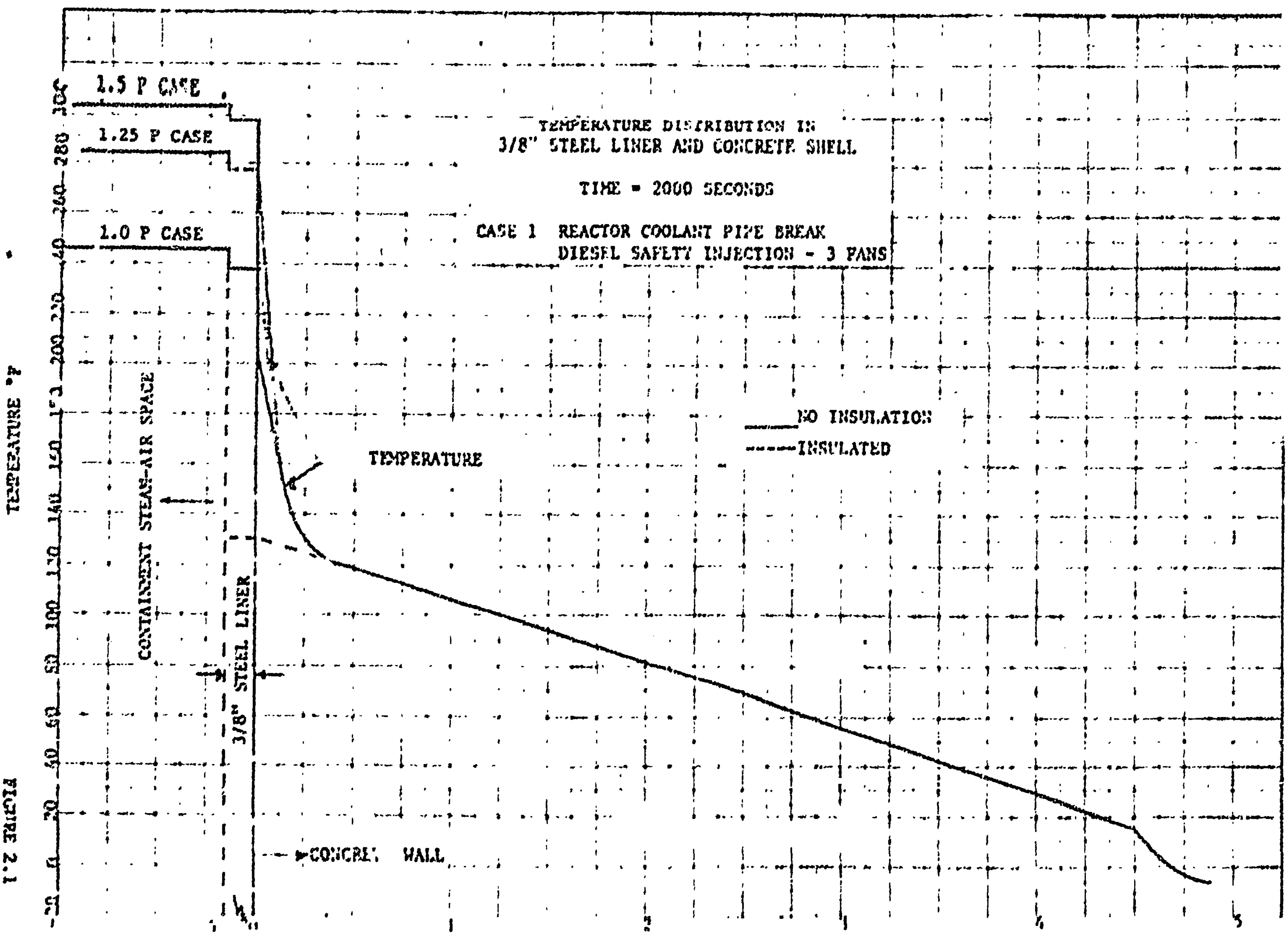
2.2.5 PENETRATIONS

The steel penetration elements not backed up by concrete are designed to carry design basis accident loads plus operational basis earthquake loads (unfactored) within the stress limitations of the ASME Section VIII Unfired Pressure Vessel Code stress limitations. It should be noted the ASME Code is a "working stress" design code and as such has safety margin contained in the reduced stress levels rather than in the factored load concept.

2.2.6 SUMMARY OF MATERIAL STRESS STRAIN PROPERTIES

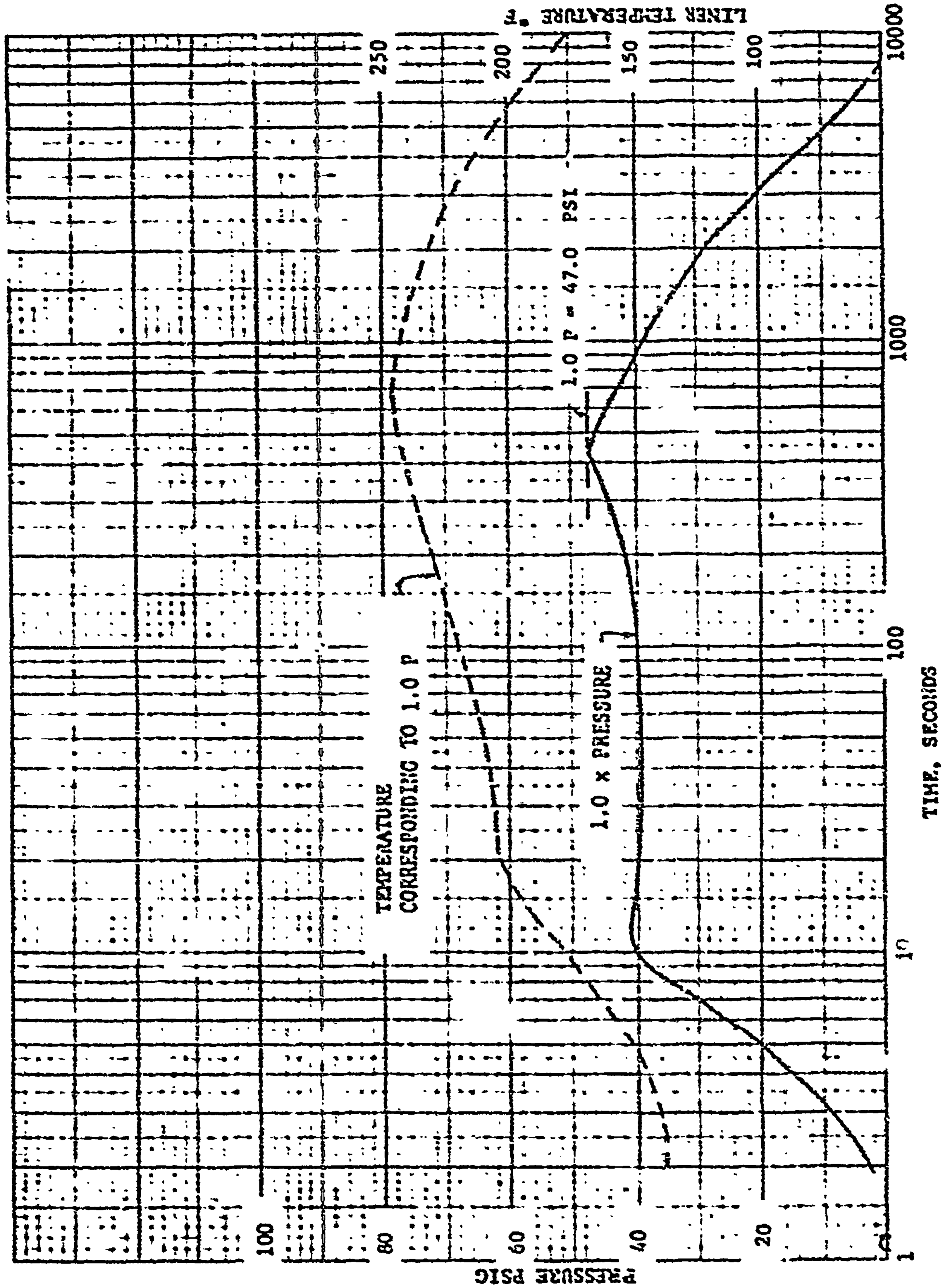
The materials used in containment conform to stress-strain limitations as follows:

	Item	Specification	Min. Yield Strength (PSI)	Min. Ultimate Strength (PSI)	Elongation
1.	Concrete	ACI-318	-	3,000	-
2.	Reinforcing Steel	ASTM A432 ACI-318	60,000	90,000	7% in 8"
3.	Liner Plate	ASTM A442 Gr. 60	32,000	60,000	22% in 8"
4.	Mech. Penetration Sleeve-12" Dia. and under	ASTM A333, Gr. I	30,000	55,000	35% in 2"
5.	Mech. - Over 12" Dia.	ASTM A201, Gr. B to A300	32,000	60,000	22% in 8"
6.	Rolled Shapes	ASTM A36, ASTM A131 Gr. C	36,000 32,000	58,000 58,000	20% in 8" 21% in 8"
7.	End Plates	a)ASTM A300 Cl.1 Firebox A 201, Gr.B	32,000	60,000	22% in 8"
		b)ASTM A240 Tp. 304L	25,000	70,000	40% in 2"
8.	Fuel Transfer Tube	ASTM A 240 Tp. 304L	25,000	70,000	40% in 2"
9.	Bellows	ASTM A312 tp. 304L	25,000	70,000	35% in 2"
10.	Elec. Penetrations	ASTM A333 Gr. 1	30,000	55,000	35% in 2"
11.	Equip. Hatch Insert	ASTM A300 Cl.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
12.	Equip. Hatch Flanges	ASTM A300 Cl.1 Firebox A201, Gr.B	32,000	60,000	22% in 8"
13.	Equip. Hatch Head	ASTM A 300 Cl.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"
14.	Personnel Hatch	ASTM A300 Cl.1 Firebox A201, Gr. B	32,000	60,000	22% in 8"



TEMPERATURE DISTRIBUTION IN
3/8" STEEL LINER AND CONCRETE SHELL
TIME = 2000 SECONDS
CASE 1 REACTOR COOLANT PIPE BREAK
DIESEL SAFETY INJECTION - 3 FANS
1.5 P CASE
1.25 P CASE
1.0 P CASE
NO INSULATION
INSULATED
TEMPERATURE
CONTAINMENT STEAM-AIR SPACE
3/8" STEEL LINER
CONCRETE WALL
TEMPERATURE °F

FIGURE 2.1



DESIGN PRESSURE-TEMPERATURE TRANSIENT

FIGURE 2.2

3.0 CONTAINMENT ANALYSIS METHODS AND COMPARISON WITH CRITERIA

3.1.0 GENERAL CONTAINMENT LOADS

3.1.1 DEAD LOAD

The weight of the concrete structure above the point under consideration based on a density of 150#/ft^3 which includes only the weight of the reinforced concrete structure. Since the maximum rebar stress occurs in tension it is conservative not to consider snow loads or any other load which will add to the dead load.

The formula for dead load in k/ft at any point is

$$T_{DL_i} = 150\ Vi/2\pi R \qquad (3.1.1)$$

where:

Vi = the volume of concrete in feet cubed above point i
R = mean radius in feet
T_{DL_i} = the dead load at any point in the structure (k/ft) of wall

The horizontal thrust from the dead weight of the dome is computed by considering

$$H = -T + wr \cos \phi_o \qquad (3.1.2)$$

and

$$T = \frac{W}{2\pi r \sin^2 \phi_o} \qquad (3.1.3)$$

where:

$$W = 2\pi r^2 w(1-\cos \phi_o) \qquad (3.1.4)$$

H = the horizontal or hoop thrust in the dome in k/ft of shell
r = mean radius of dome in feet
ϕ_o = the central angle measured from the top of the dome to the point under consideration
W = the total weight of the dome above the point defined by ϕ_o in kips
w = the dead load per unit surface area of shell in k/ft^2.
T = the vertical or meridianal thrust in the dome in k/ft of shell

3.1.2 DESIGN BASIS ACCIDENT PRESSURE LOAD

Membrane pressure loads in the vertical direction in the cylinder and either direction in the dome are determined by

$$P = \frac{pR}{2} \qquad (3.1.5)$$

For the horizontal or hoop direction in the cylinder

$$P = pR \qquad (3.1.6)$$

where:

P = pressure load in #/in of wall

p = internal design pressure in #/in^2

R = mean radius in inches

3.1.3 DISCONTINUITY MOMENT AND SHEAR LOAD

The bending moments, shears and deflections induced in the cylindrical shell by the restraint provided by the base are found by considering a cylindrical shell with a uniform internal pressure [1]. Using the general equations for deflection and slope for cylinder with end moment and shear, and substituting boundary conditions of $w = \delta$ and $\theta = 0$ at $\chi = 0$ (the built in end) where δ = the unrestrained growth of a cylinder under uniform internal pressure, one obtains formulae for the moment and shear at the built-in end to cause zero deflection and rotation

$$Mo = \frac{P}{2\beta^2} \quad \text{and } Q_o = \frac{\cdot P}{\beta} \qquad (3.1.7)$$

where:

P = the internal pressure in #/in^2

$$\beta^2 = \left[\frac{E_s h_s}{4a_c^2 D}\right]^2 \qquad (3.1.8)$$

$$D = \frac{E_c h_c^3}{12\ (1-\mu)} \quad \text{(flexural rigidity of the shell)} \qquad (3.1.9)$$

h_s = area of horizontal steel and liner in the cylinder which acts as a spring constant (in^2/in)

a_c = mean radius of the containment cylinder in inches

h_c = effective depth or thickness of the wall

μ = Poisson's ratio = 0 for cracked concrete

E_s = modulus of elasticity of steel = 29×10^6 psi

E_c = modulus of elasticity of concrete = 3.2×10^6 psi

M_o = moment at built in Section to cause 0 rotation

Q_o = shear at built in Section to cause 0 deflection

Substituting these values in the following expressions, values for bending moment, shear and deflection at any distance from the end can be found:

$$\Delta_x = w = \frac{-1}{2\beta^3 D} \left[\beta M_o \; \psi(\beta x) + Q_o \; \theta(\beta x)\right] \qquad (3.1.10)$$

$$\theta_x = \frac{dw}{dx} = \frac{1}{2\beta^2 D} \left[2\beta M_o \; \theta(\beta x) + Q_o \; \phi(\beta x)\right] \qquad (3.1.11)$$

$$M_x = D\frac{d^2 w}{dx^2} = \frac{-1}{2\beta D} \left[2\beta M_o \; \phi(\beta x) + 2Q_o \delta(\beta x)\right] D \qquad (3.1.12)$$

$$V_x = D\frac{d^3 w}{dx^3} = \frac{1}{D}\left[2\beta M_o \; \delta(\beta x) - Q_o \; \psi(\beta x)\right] D \qquad (3.1.13)$$

where:

$\phi(\beta x) = e^{-\beta x} (\cos \beta x + \sin \beta x)$

$\psi(\beta x) = e^{-\beta x} (\cos \beta x - \sin \beta x)$

$\theta(\beta x) = e^{-\beta x} \cos \beta x$

$\delta(\beta x) = e^{-\beta x} \sin \beta x$

Δ_x = the deflection of the shell at x

θ_x = the slope of the shell at x

M_x = the moment of the shell at x

V_x = the shear in the shell at x

From these values Figures 3.1 and 3.2 are plotted showing moment and shear vs. height of wall in inches. The effect of backfill on the restraint provided the containment cylinder was evaluated and the resultant shift in restrained moment and shear higher into the shell was determined not a design factor.

The problem of determining the discontinuity moment and shear at the springline is similar to that at the base. Discontinuity forces at the dome-cylinder junction are only a function of the relative deformation at this point, since the rotations of the cylinder and of the dome due to the internal pressure are zero and therefore present no discontinuity. The extension of the radius of the cylindrical shell due to the internal pressure is given by

$$\delta_C = \frac{Pa_D^2}{2E_{sh_s}^{D}} (1 - \mu) \qquad (3.1.14)$$

and the unrestrained extension of the dome (δ_D) is given by

$$\delta_D = \frac{Pa_D^2}{2E_{sh_s}^{D}} (1 - \mu) \qquad (3.1.15)$$

where:

a_D = mean radius of the containment dome in the inches

h_S = area of horizontal steel and liner of the dome which acts as a spring constant (in^2/in)

Since the area of the hoop steel per foot in the dome is approximately one half that of the cylinder, the values of δ_C and δ_D are nearly equal and therefore the relative deformation is insignificant.

In calculating the discontinuity effects, the bending is of a local character so that an approximate solution can be obtained by assuming that the bending is of importance only in the zone of the dome close to the springline and that this zone can be treated as a portion of a long cylindrical shell. Equations of continuity for deflection and rotation are written such that the values of M_o and Q_o at the springline may be found. The distribution of the moment and shear into the dome and the cylinder are then found by substituting M_o and Q_o into Eqs. 3.1.12 and 3.1.13.

3.1.4 BASE MAT LOADS

The beam shears and moments in the base mat can be calculated by considering the loads shown acting on a 1'-0 wide beam. The 1'-0 strip of mat to be considered is located at the point where the uplift from the overturning moment in the containment due to earthquake is maximum. This gives the maximum moments and shears in the strip.

The loads considered as shown in Figure 3.3 are

$$U_T = P + T_{EQ} + T_V - T_{DL} \quad \text{in k/ft} \qquad (3.1.16)$$

where:

P = design basis loss-of-coolant accident pressure effect load in the wall in k/ft of wall

T_{EQ} = the tensile load k/ft of wall developed by the earthquake overturning moment

T_V = the effective tensile load or reduction of dead load in k/ft of wall caused by response of the containment structure to vertical earthquake motion.

T_{DL} = the dead load in the wall in k/ft

M = base discontinuity moment defined in Section 3.1.3

V_u' = base discontinuity shear defined in Section 3.1.3

D = the dead weight of the base mat on the outside of containment cylindrical wall center line in k/ft

C = the reaction of the internal structural support columns which are based on the 3'-0 reinforced concrete fill mat; in all cases equal to 50 K spaced every 23'-0

E = average dead weight of the backfill on top of the 1'-0 ledge

w = $12\,p + \frac{\rho z}{12^3}$; the effective uniform load acting on a 1' wide segment of the base slab per inch of segment length

where p = the containment internal pressure in $kips/in^2$

ρ = the density of reinforced concrete in $\#/ft^3$ = $150\#/ft^3$

z = the total depth of section including the 3'-0 fill slab.

The crane wall reaction in k/ft is determined by

$$R = D_c + D_o + P_c \qquad (3.1.17)$$

where:

D_c = ρt_1 H; or the dead weight of the crane wall in k/ft

t_1 = the thickness of the crane wall = 3.0 ft.

H = the height of the crane wall = 50.0 ft.

$D_o = \frac{\pi R_1^2\, t_2 \rho}{2\pi\, R_2}$ or the approximate dead weight of the operating floor in k/ft.

R_1 = the outside radius of the operating floor = 53'-0

R_2 = the mean radius of the crane wall

= 51'-6

t_2 = the thickness of the operating floor = 2' - 0

P_2 = 12 $p t_1$ or the pressure load acting on the top of the crane wall with t_1 given in inches

Moments and shears are calculated by writing equations for moment and shear in terms of x as the origin, with x increasing toward the center of the containment building and x measured in inches.

The formulas are as follows:

For $0 \leq x \leq 201$

$$V_x = U_T - D - 2C^* - E - wx \qquad (3.1.18)$$

$*$ - $2C$ when $x \geq 20$

- C when $x < 20$

with V_x assumed constant and equal to the volume of V_x at 201 inches for the region under the crane wall $201 < x < 237$.

For $x \geq 237$

$$V_x = U_T - D - C - w(201) - R - C - w(x-237) - E \qquad (3.1.19)$$

or

$$V_x = U_T - 2C - D - R - wx + 36w - E \qquad (3.1.20)$$

where:

V_x = uplift shear at any point x (inches) in k/ft

Equation 3.1.20 is considered applicable until $V_x \leq 0$.

The design moment in the base slab is determined for $0 \leq x \leq 201$

$$M_x = M + V'_u e + D(x + 19.5) + \frac{wx^2}{2} + C(x - 27) - U_T x + E(x + 33) \qquad (3.1.21)$$

with M_x assumed constant and equal to the value of M_x at 201 inches for the region under the crane wall $201 \leq x \leq 237$.

For $x \geq 237$

$$M_x = M + V'_u e + D(x + 19.5) + w(201)(x - 105.5) + C(x-27) + E(x + 33)$$

$$+ C(x - 201) + R(x - 219) + \frac{w(x - 237)^2}{2} - U_T x \qquad (3.1.22)$$

or

$$M_x = M_u + V'_u e + D(x + 19.5) + w(1/2\, x^2 - 36x + 6850) + 2Cx - 228C \qquad (3.1.23)$$

$$+ R(x - 219) - U_T x$$

where:

e = the effective depth of the 9'-0 base mat divided by 2 and
M_x = the base moment at any point x (inches) in in-k/ft

At the point where $V_x \leq 0$ flexural beam action is no longer considered since uplift is 0 and the mat acts as a flat circular plate supported on a rigid non-yielding foundation.

Again it should be noted that these maximum values for shear and moment occur at only one point on the base slab circumference where the uplift from the horizontal earthquake is maximum and decreases to zero 90° from this point; therefore it is considered that the calculations shown are conservative.

A gradient with an operating temperature of 120°F inside the containment and a 50°F temperature at the mat-rock interface was considered and the stresses determined are negligible. Accident temperatures have no appreciable effect on the base slab.

3.1.5 SEISMIC LOAD

Horizontal Earthquake

The fundamental frequency and mode shape is found by use of the Modified Rayleigh Method as described in reference 2.

The containment structure is divided into segments and each is loaded at its centroid with a load equal to the weight of the segment (See Figure 3.4). Using these loads, shears and moments are calculated and the M/EI diagram used to load a conjugate beam. From the conjugate beam the deflection at each node including the shear deflection is calculated and normalized by setting the maximum deflection equal to 1.0. The fundamental period is calculated by

$$T = 2\pi \left(\frac{Yo \; \Sigma \; \phi^2 \; dm}{g \; \Sigma \; \phi \; dm} \right)^{1/2} \qquad (3.1.24)$$

where:

Yo = max. deflection at the top of containment in inches

ϕ = normalized deflection at a point

dm = equivalent load equal to weight of structure represented at node

g = 386 in/sec^2

T = period in seconds

Using this period and the response spectral acceleration curve with 2% critical damping a coefficent of spectral acceleration of 0.24g is obtained. Multiplying this coefficient by the total mass of the structure yields the base shear on the structure equal to 13.5 k/in^2.

This base shear is assumed distributed up the structure by

$$F_r = \frac{W_r h_r}{\overset{r}{\Sigma} \; W_r h_r} V \qquad (3.1.25)$$

where:

F_r = horizontal force at node r (See Figure 3.4 for locations)
W_r = weight concentrated at mass r
h_r = height of mass r above the base
V = total base shear

The shear at any section of the cylinder is determined by

$$V_i = \Sigma F_{ri} \qquad (3.1.26)$$

where:

V_i = the seismic shear at point i
F_{ri} = the horizontal inertial force at r nodes above elevation i

The loads at the nodes are used to calculate the moments and displacements at various points in the structure.

Figures 3.5 and 3.6 show the base seismic shear and overturning moment distribution as a function of height of wall.

In order to evaluate the maximum shear in the structure the effect of the backfill being accelerated against the structure was investigated:

Case I - Backfill forces acting opposite to inertial forces of structure

a) 1.25 (factored load) Passive pressure of soil. Mass of soil in envelope of shear failure accelerated at .10g x 1.25.

b) Passive pressure of Soil. Mass of soil in envelope of shear failure accelerated at .15g.

Case II - Backfill forces acting in same direction as inertial forces

a) 1.25 (factored load) Active pressure of soil. Mass of soil in envelope of shear failure accelerated at .10g x 1.25

b) Active pressure of soil. Mass of soil in envelope of shear failure accelerated at .15g.

By adding these forces in the region of backfill to the distributed base shears from above and calculating moments shears and displacements it was found that case Ia) governed in the base area of containment for shear considerations.

The shear flow is determined by consideration of a hollow ring with a total thickness of 2t.

$$S_f = \frac{V_s Q}{I} \qquad (3.1.27)$$

where:

S_f = Shear flow in the wall

V_s = Shear at the elevation under investigation as determined by (Eq3.1.14) or the backfill effects, whichever is larger, in k/ft.

$$Q = 2 \int_0^{\pi/2} y\,dA$$

where:

y = Distance from element under consideration to the neutral axis of the circular tube cross section or R sin θ

R = Radius of containment in inches

θ = angle from neutral axis to the element under consideration in radians

dA = The area of the element under consideration or Rtdθ

t = The thickness of the containment shell in inches

I = Moment of inertia of section about neutral axis in $in.^4$

Vertical Earthquake:

The period is calculated by

$$T = 2\pi\sqrt{\frac{M}{K}} \qquad (3.1.28)$$

where:

M = Total mass of structure $k\text{-}sec^2/ft$

K = Stiffness of structure k/in

Using this period and the response spectral acceleration curve with 2% critical damping a coefficient of spectral acceleration of 0.13g (1.0E) is obtained. Multiplying this coefficient by the total mass of the structure yields the vertical earthquake reaction in k/ft of wall.

$$T_{v_i} = \frac{k M_i}{2\pi R} \qquad (3.1.29)$$

where:

k = coefficient of seismic acceleration in the vertical direction; (.13g for 1.0E'), (.065g for 1.25E)

M_i = mass of containment shell above point i in kips/g of wall

R = radius of containment in feet

Uplift from the Horizontal Earthquake:

The horizontal inertial force on the containment structure produce overturning movements which in turn produce tension on one side of the containment and compression on the other side in the direction of the earthquake. These forces per foot of wall section are computed by dividing the overturning moment on the section, considering the containment a cantilever beam, by the moment of inertia of the containment as a hollow cylinder. Since the concrete shell is assumed cracked and in tension under the loss-of-coolant accident pressure condition, only the area of the containment vertical rebar and liner are considered in determing the moment of inertia.

The seismic overturning moment above a point i about point i is determined:

$$M_i = \quad F_{ir}\, h_{ir} \tag{3.1.30}$$

where:

h_{ir} = The distance from the location of forces F_{ir} to the point i
F_{ir} = The horizontal inertial forces on the r segments above point i

The moment of inertia is computed by

$$I = \pi\ t_1\ r^3 \text{ (A hollow circular ring)} \tag{3.1.31}$$

where:

t_1 = equivalent thickness of vertical reinforcing steel, including liner in sq. in. per inch of wall

r = mean radius of containment in inches

and $$T_{EQ} = \frac{M_i C t_1}{I} \tag{3.1.32}$$

where:

T_{EQ} = vertical force in k/in induced in the containment wall by the seismic overturning moment

C = distance from neutral axis to outermost fiber of containment cross section.

3.1.6 TEMPERATURE EFFECT LOADS

An increase in internal temperature caused by a loss-of-coolant accident has been considered. The maximum temperatures, which do not occur at the same time as the maximum pressures, related to the design (P), 1.25P and 1.5P cases are 247°F, 285°F and 306°F respectively. This increase in temperature causes compressive forces in the restrained liner which in turn induces tensile stresses into the re-bar. The equivalent force induced in the containment wall is determined;

$$F_C = A_L \epsilon_{TL} E_S \tag{3.1.33}$$

where:

F_C = the equivalent tensile load induced in concrete containment shell by the attempted expansion of the liner

ϵ_{TL} = final compressive strain in the liner after pressure and temperature conditions and elastic relaxation of the concrete shell have been considered

E_S = modulus of elasticity for the liner steel

In addition to the liner temperature effect on the containment shell the effect of operating thermal gradients through the wall have been considered in analysis of the containment as shown in Section 3.2.5.

The effect of accident thermal gradients has been investigated and found to penetrate less than 10 percent of the containment wall thickness during the maximum temperature-pressure transient following a loss-of-coolant accident. For this reason the accident temperature transient thermal gradient effect has not been considered in design analysis.

3.1.7 WIND LOAD

The wind load will be determinedd by considering a conservative wind pressure of 30 psf for ground level up as stipulated in the state building and constructio code for the state of New York.

The forces due to the wind loading are given by

$$V_i = P_1 A_i \tag{3.1.34}$$

where:

V_i = the wind shear at point i
P_1 = the wind pressure of 30 psf
A_i = the projection, perpendicular to the direction of the wind, of the area of containment above the point i

and

$$M_i = P_1 A_i L \tag{3.1.35}$$

where:

M_i = overturning moment about point i determined from the wind load
L = the moment arm from the centroid of the projected area above point i to point i

In all cases the magnitude of the design wind loads will be less than the seismic loads therefore no stresses will be calculated.

The magnitude of the shear load and overturning moment from design wind is shown in Containment Liner Report, Appendix C of the FSAR, Table II-3.1.

3.1.8 LOAD COMBINATIONS

The loads discussed above were combined to design the containment structure as given in Section 2.1.12.

3.2.0 GENERAL STRESS/STRAIN FORMULA

3.2.1 DEAD LOAD STRESS

$$\sigma_{T_i} = \frac{T_{DLi}}{A_S} \text{ when overall effect is tension} \quad (3.2.1)$$

$$\sigma_{c_i} = \frac{T_{DLi}}{A_c} \text{ when overall effect is compression} \quad (3.2.2)$$

where:

A_{s_i} = area of vertical steel or hoop including liner, per foot of wall

A_{c_i} = area of concrete per foot of wall

T_{DLi} = dead load as defined in Section 3.1.1

3.2.2 DESIGN BASIS ACCIDENT PRESSURE LOAD STRESS

$$\sigma = \frac{P}{A_s} \quad (3.2.3)$$

where:

A_s = area of vertical steel or hoops, including liner, per foot of wall
P = pressure induced membrane force per foot of wall

3.2.3 DISCONTINUITY MOMENTS AND SHEAR LOAD STRESSES

The stresses induced in the containment shell wall from the discontinuity moment is calculated by considering formula (16-1) of the ACI 318-63 Code Ultimate Strength Design.

$$M = A_{s_1} f_s \left(d - \frac{a}{2}\right) \tag{3.2.4}$$

$$f_s = \frac{M}{A_{s_1} (d - a/2)} \tag{3.2.5}$$

where:

$$a = \frac{A_{s_1} f_y}{.85 f'_c b} \tag{3.2.6}$$

and

A_{s_1}	= area of steel on the tension side of the containment wall in in^2/ft
f_s	= stress in the steel in k/in^2
f_y	= yield strength of the steel in k/in^2
f'_c	= 3000 psi 28 day design compressive stress of concrete in k/in^2
b	= width of cross section: in all cases assumed equal to 12"
d	= effective depth of cross section in inches = 45"
M	= resisting moment in inch-kips per foot. The basis for this number is Figure A-2. This is less than the ultimate moment since $f_s < f_y$.

The stress in the stirrups is computed from Equation (17-6) of the ACI 318-63 Code - Ultimate Strength Design.

$$A_v = \frac{V' s}{F_s d (\sin \alpha + \cos \alpha)} \quad (3.2.7)$$

$$f_s = \frac{V' s}{A_v d (\sin \alpha + \cos \alpha)} \quad (3.2.8)$$

where:

A_v = total area of web reinforcement in tension within a distance, s, measured in a direction parallel to the longitudinal reinforcement in in^2.

V' = total shear to be carried by web reinforcement in kips

s = spacing of stirrups or bent bars in a direction parallel to the longitudinal reinforcement in inches

f_s = stress in the stirrups in k/in^2

d = effective depth of cross section in inches = 45"

α = angle between inclined web bars and longitudinal axis of member = 45°

3.2.4 Base Mat Stress

Stress from the moment is calculated by considering formula (16-1) of the ACI-318-63 code ultimate strength design as shown in Eq. 3.2.4, 3.2.5 and 3.2.6.

where:

As = area of steel on the tension face of the containment base slab in in^2/ft

f_s = stress in the steel in k/in^2

f_c = 3000 psi 28 day design compressive stress of concrete in k/in^2

b = width of cross section - in all cases assumed equal to 12"

d = effective depth of cross section in inches = 100"

M = resisting moment in inch - kips per foot

Stress from the uplift shear is computed from Eq. (17-6) of the ACI-318-63 code as shown in Equations 3.2.7 and 3.2.8.

where

$$V^1 = V - V_c \tag{3.2.9}$$

where:

$$V = \text{total shear}$$
$$V_c = v_c bd \tag{3.2.10}$$

and

v_c = the allowable concrete shear stress or $2\phi\sqrt{f'_c} = 93\text{k/in}^2$

ϕ = capacity reduction factor = .85

f_s = stress in the stirrups in k/in^2

α = angle between inclined web bars and longitudinal axis of member = 45°

b = width of the section = 12 inches

d = effective depth of the cross section = 100"

Additional web reinforcement was also provided on the basis of a minimum spacing of s equal to 0.75d.

Bond stresses in the stirrups are computed by considering the formula

$$\mu = \frac{A_v f_s}{\Sigma o \quad L} \tag{3.2.11}$$

where:

μ = the bond stress in k/in^2

Σo = sum of perimeters of all effective bars crossing the section on the tension side.

L = the anchorage length above or below the mid height of the mat. No credit is taken for additional anchorage provided by the bend in the bar.

The allowable bond stress for tension bars with deformations conforming to ASTM A408 and other than top bars is

$$\mu_A = (.8)(6\sqrt{f'_c}) \qquad (3.2.12)$$

where

μ_A = the allowable bond stress in k/in^2
.8 is the factor allowed by the ACI-318-63 ultimate strength design code for anchorage bond.

3.2.5 SEISMIC LOAD STRESS

Horizontal or Vertical Earthquake Effects

$$\sigma = \frac{Load}{A_s} \quad \text{(For overall tension)} \qquad (3.2.13)$$

$$\sigma = \frac{Load}{A_c} \quad \text{(For overall compression)} \qquad (3.2.14)$$

where:

A_s = area of vertical steel, including liner, per foot of shell
A_c = area of concrete per foot of shell
Load = force per foot of shell resulting from dead load response to vertical earthquake acceleration or overturning moment induced vertical load.

The basic assumptions considered in the seismic analysis are:

1) Maximum stress in the seismic reinforcing occurs under the action of seismic shear at 90° points from the direction of seismic motion.

2) The liner does not participate in resisting seismic shear.

3) The stress limitations on intersection bars under the combination of pressure plus earthquake shear in one bar may reach 95% of yield and the opposing bar may relieve stress to 0 ksi. Under this consideration only half of the seismic diagonal steel is considered active in resisting earthquake shear at any given instant.

Thus the stress can be calculated by considering the shear flow in the wall being resisted by diagonal bars in a hollow ring.

$$A_{s_s} = \frac{S_f \ (1.414)}{2 f_s} \qquad (3.2.15)$$

$$f_s = \frac{S_f \ (1.414)}{2 A_{s_s}} \qquad (3.2.16)$$

where:

A_{s_s} = area of diagonal steel per foot, in one direction, measured along a horizontal plane

f_s = stress in the steel in k/in^2

S_f = the shear flow as determined from Equation 3.1.26

The 1.414 take the 45° angle of inclination of the diagonal bars into account.

3.2.6 TEMPERATURE EFFECT STRESSES

As discussed in Section 3.1.6 temperature considerations must involve both temperature gradient and the interaction effects of the liner on the containment shell. The following development for interaction takes both of these phenomena into account.

Temperature effects as shown in Figure 3.7 are combined with deadload, pressure, and earthquake uplifts in the following manner.

Due to the redistribution of stresses in the rebar, the reinforcing steel is considered to carry an equal amount of tension which must balance the compression in the liner to satisfy $\Sigma F_x = 0$.

To satisfy equilibrium conditions:

$$F_{Liner} = F_{Wall} \quad (3.2.17)$$

$$A_L \varepsilon_{TL} E = - A_s \varepsilon_{TL'} E$$

$$A_L \frac{\varepsilon_{TL_x} + \mu \varepsilon_{TLy}}{1 - \mu^2} E = - A_s \varepsilon_{TL'} E$$

$$\varepsilon_{TL'x} = - \frac{A_L}{A_S} \frac{\varepsilon_{TLx} + \mu \varepsilon_{TLy}}{1 - \mu^2} \quad (3.2.18)$$

The 2nd condition which must be satisfied is the deformation compatibility

$$\varepsilon_{TLx} + \varepsilon_{\Delta T} = \varepsilon_{TL'} + \varepsilon_T \quad (3.2.19)$$

$$\varepsilon_{TLx} + \varepsilon_{\Delta T} = \varepsilon_T - \frac{A_L}{A_S} \frac{\varepsilon_{TLx} + \mu \varepsilon_{TLy}}{1 - \mu^2}$$

Let $\varepsilon_x = \varepsilon_T - \varepsilon_{\Delta T}$

$$\varepsilon_{TLx}\left(1 + \frac{A_L}{AL \quad (1 - \mu^2)}\right) = \varepsilon_x - \frac{A_L}{A_S} \frac{\mu \varepsilon_{TLy}}{1 - \mu^2}$$

$$\varepsilon_{TLx} = \frac{\varepsilon_x}{1 + \frac{A_L}{A_S\,(1 - \mu^2)}} - \frac{\mu\,\varepsilon_{TLy}}{\frac{A_S}{A_L}\,(1 - \mu^2) + 1} \qquad (3.2.20)$$

Let $\mu = .25$

$$\varepsilon_{TLx} = \frac{\varepsilon_x}{1 + 1.067\,\frac{A_L}{A_S}} - \frac{.25\,\varepsilon_{TLy}}{.9375\,\frac{A_L}{A_S} + 1} \qquad (3.2.21)$$

$$\varepsilon_{TLy} = \frac{\varepsilon_y}{1 + 1.067\,\frac{A_L}{A_S}} - \frac{.25\,\varepsilon_{TLx}}{.9375\,\frac{A_S}{A_L} + 1} \qquad (3.2.22)$$

To solve Eq. 3.2.18 for the strain in the rebar induced by liner compression solve Eq. 3.2.21 and 3.2.22 simultaneously and insert values for ε_{TLx} and ε_{TLy} into Eq. 3.2.18.

The definitions of the terms used in the above derivations are:

ε_T = strain in the rebar induced by the dead load, pressure and uplift from horizontal and vertical earthquakes.

ε_{TL} = final strain in liner causing stress or the restrained portion of the potential strain of the liner due to the temperature increase (X OR Y direction)

ε_{TL}' = strain in rebar from stress induced by liner compression. (X OR Y direction)

μ = poissons Ratio = .25

A_L = area of liner in in^2/in

A_s = area of rebar in in^2/ft

E = the modulus of elasticity o. steel when the section is in tension ($\epsilon_T + \epsilon_{TL}, \geq 0$) and modulus of elasticity of concrete when the section is in compression ($\epsilon_T + \epsilon_{TL}, \leq 0$). All preceeding developments are for the section in tension since this will yield the maximum rebar stress.

$\epsilon_{\Delta T}$ = the strain in the liner if unrestrained growth were allowed or $\alpha\ \Delta T$

where:

α = coefficient of thermal expansion in inch/inch/degree F

ΔT = the difference in temperature between the accident temperature felt by the liner and the temperature of the neutral surface (or the point through the wall where no ... mal stress exists because of a thermal gradient through the wall).

The gradient is assumed linear with the inside temperature equal to the operating temperature of 120°F and the outside surface temperature of 0°F.

ΔT can be considered in two steps

$$\Delta T_{gradient} = 120° - T_{neutral\ surface}$$

$$\Delta T_{interaction} = T_{Max} - 120°$$

This shows the contribution of both the gradient and interaction effects

The effect of accident temperatures on thermal gradients has not been considered since analysis has shown only 10 percent of the wall located on the inner face of the containment sees any change of thermal gradient during the pressure phase of the accident. In actuality the stresses induced by thermal gradients in the concrete shell are secondary in nature and are largely relieved by the shell cracking under design accident pressure load conditions. For conservatism however the operating temperature gradient was included in the stress analysis.

The location and temperature at the neutral surface as shown in Figure 3.8 is found by equating tension on the outside of the neutral surface to compression on the inside assuming the concrete carries no tension. This development of thermal stresses in the rebar is based on the method presented in ACI chimney code(4).

The total compressive force is equal to

$$1/2 \; L \; k^2 \; t \; T_x \; E_c + L \; k \; T_x \frac{E_s A_L}{b} + L \; (k - z_9) \frac{T_x E_s A_s}{b} \qquad (3.2.23)$$

$$+ L \; (k - z_8) \frac{T_x E_s A_s}{b}$$

and the total tensile force is equal to

$$L \; T_x \; E_s \frac{A_s}{b} \; (z_7 + z_6 - 2 \; k) \qquad (3.2.24)$$

When equating total tension to total compression the result is the following

$$k^2 + \frac{2k \; nt_L}{t} + \sum \frac{2n \; A_s}{bt} \; (k - z) = \qquad (3.2.25)$$

where:

k = distance from the liner to the neutral surface divided by the total thickness of the wall

b = re-bar spacing in inches

n = $\frac{E_s}{E_c}$

t = total wall thickness

t_L = liner thickness in inches

z = distance from the liner to the rebar under consideration divided by the total thickness of the wall

The temperature at the neutral surface = $(1 - k)\ \Delta T_1$ (3.2.26)

where: $\Delta T_1 = 120° - 0° = 120°$

To get the final stress in the rebar due to temperature

$$\sigma = (\epsilon_T + \epsilon_{TL'})\ E_s \qquad (3.2.27)$$

3.3.0 DETAILED ANALYSIS OF CONTAINMENT AT REPRESENTATIVE LOCATIONS

In order to perform a specific comparison between actual stress-strain levels and limiting behavior criteria several representative points on the containment shell to include the base, cylinder and dome are selected for analysis. The selected points are shown in Figure 3.11 and described in Sections 3.3 through 3.3.8. Detailed tabulation of design loads for the eight points listed are found in Section 3.3.9 with the resultant stresses and allowable stress criteria presented in Section 3.3.10. The detailed determination of the loads and stresses shown in Sections 3.3.9 and 3.3.10 are based on the equations given in Sections 3.1 and 3.2. The actual calculations are in the files of United Engineers and Contructors, Inc., Philadelphia, Pennsylvania.

3.3.1 POINT 1

Point 1 is located in the base mat at a point adjacent to the outside face of the crane wall in a region of negligible uplift, where the mat begins to act a. flat circular plate supported on a rigid non-yielding foundation, and high positive moment, point 1 is located at coordinates H = 53 ft, V = 43 ft.

3.3.2 POINT 2

Point 2 is located in the base mat near the containment wall in a region of high uplift and negative moment adjacent to the knuckle of the liner. Point 2 is located at coordinates H = 57 ft & V = 43 ft.

3.3.3 POINT 3

Point 3 is located in the cylindrical portion of the containment shell in a region of very high negative discontinuity moment at a point adjacent to the knuckle at the cylinder-base mat junction which is insulated against any thermal effects. It is located at coordinates H = 67.5 ft V = 45.7 ft.

3.3.4 POINT 4

Point 4 is located in the cylindrical portion of the containment shell in a region of relatively high positive discontinuity moment adjacent to the cut off point for liner insulation at coordinates H = 67.5 ft and V = 64 ft.

3.3.5 POINT 5

Point 5 is located in the cylindrical portion of the containment shell, about half way between the base mat and the spring line, in a region of membrane stresses only at coordinates H = 67.5 ft; V = 117 ft.

3.3.6 POINT 6

Point 6 is located in the cylindrical portion of the containment shell at a point just below the spring line. It is an area of membrane stress only since the discontinuity effects at the spring line are insignificant because the deflection of the [illegible] and cylinder are essentially equal due to the changing steel areas. It is located at coordinates H = 67.5 ft and V = 191.0 ft.

3.3.7 POINT 7

Point 7 is located in the dome portion of the containment shell at a point just above the spring line. It is an area of membrane stress only since the discontinuity effects at the spring line are insignificant because the deflection of the dome and cylinder are essentially equal due to the changing steel areas. Point 7 is located at coordinates H = 67.5 ft and V = 191.0 + ft.

3.3.8 POINT 8

Point 8 is located in the dome portion of the containment shell at a point approximately defined by a 30° arc from the spring line in a region of membrane stresses only. The seismic bars are terminated at this point and seismic shear is resisted by a combination of liner, dowel action and aggregate interlock. Point 8 is located at coordinates H = 57.8 ft and V = 225.8 ft.

3.3.9 SUMMARY OF CONTAINMENT DESIGN LOADINGS

In this Section are presented two tables relative to the design Points 1 through 8 shown in Figure 3.9. In Table 3.1 is shown the material and section properties relative to the eight design points selected while Table 3.2 shows the resultant loads for the points selected which were developed from the equations given in Section 3.1 for the load factors and combinations presented in Section 2.1.12.

3.5.9 SUMMARY OF CONTAINMENT DESIGN LOADINGS

MATERIAL & SECTION PROPERTY TABLE 3.1

Position (1)	Coordinates (Ft) Hor. (2)	Coordinates (Ft) Vert. (2)	Density of Reinforced Concrete ρ (#/ft³) (3)	Total Depth of Mat Z (Ft) (4)	Thickness of Crane Wall t_1 (Ft) (5)	Height of Crane Wall H (Ft) (6)	Outside Radius of Operating Floor R_o (Ft) (7)	Thickness of Operating Floor t_2 (Ft) (8)	Mean Radius of Crane Wall R_1 (Ft) (9)	Distance from Origin X (in) (10)	Effective Depth of Mat Divided by 2 e (11)	Effective Depth of Mat d (in) (12)
1	53.5	43.0	150	12'-0	3'-0	50'-0	53'-0	2'-0	51'-6	201	50	100
2	87.0	43.0	150	12'-0	3'-0	50'-0	53'-0	2'-0	51'-6	33	50	100

CONTINUED

Position (1)	Area of Steel on Tension Face of Mat As (in²/ft) (13)	28 Day Compressive Concrete Strength f'_c (k/in²) (14)	Total Area of Stirrups Av (in²/ft) (15)	Sum of Perimeters of Stirrups Σ_o (in) (16)	Anchorage Length of Stirrups L (in) (17)	Spacing of Stirrups S (in) (18)	Effective Depth of Mat, d (in) (19)
1	5.44	3	2.12	6.85	84	65	100
2	4.30	3	4.30	7.63	108	63	100

4.3.9 SUMMARY OF CONTAINMENT DESIGN LOADINGS (Cont'd)

MATERIAL AND SECTION PROPERTY TABLE 3.1

Position	Coordinates (ft) Hor. Vert.	Mean Contain. Radius (in)	Area of liner A_L (in^2/ft)	Area of steel A_s (in^2/ft) H	V	Modulus of Elas. Steel E_s (k/in^2)	Modulus of Elas. Conc. E_c (k/in^2)	Poisson's Ratio (ν Liner)	Coef. of Thermal Expansion α (in/in/°F)	Liner Thickness t_L (in)	Total Wall Thickness t-(in)	$\frac{kt}{T}$ =K	Spring Constant-Steel h_s (in^2/in)
(1)	(2)	(3)	(4)	H (5)	V	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)
3	67.5 45.7	837	6.0	16.93	20.50	29×10^3	3.2×10^3	.25	6.5×10^{-6}	.50	54.5	-	1.95
4	67.5 64.0	837	6.0	16.93	19.23	29×10^3	3.2×10^3	.25	6.5×10^{-6}	.50	54.5	.273H .272V	1.95
5	67.5 117.0	837	4.5	16.93	11.23	29×10^3	3.2×10^3	.25	6.5×10^{-6}	.375	54.375	.273H .231V	1.95
6	67.5 191.0(-)	837	4.5	15.52	9.82	29×10^3	3.2×10^3	.25	6.5×10^{-6}	.375	54.375	.273H .231V	-
7	67.5 191.0(+)	837	6.0	8.57	9.82	29×10^3	3.2×10^3	.25	6.5×10^{-6}	.50	54.375	.207H .220V	-
8	57.8 225.8	831	6.0	6.75	[illegible]7.67	29×10^3	3.2×10^3	.25	6.5×10^{-6}	.50	42.375	.191H .239V	-

Flexular Rigidity D (k/in)	(δ)	Natural Period T (sec)	Coef. of Spectral Acceler. (C)	Equivalent Ring Thickness T_1 (in)	Moment of Inertia/ Dist. to Extreme Fiber I/C (in^3/ft)	Vertical Natural Period T_v (sec)	28 [illegible] Concrete Compressive Strength f'c (k/in^2)	Tension Steel A_s (in^2/ft)	Spacing of Stirrups δ (in)	Moment of Area/ Moment of Inertia Q/I (1/in)	Area of Stir-rups Av (in^2/ft)	*Area of Seismic Steel AS_s (in^2/ft)
(16)	(17)	(18)	(19)	(20)	(21)	(22)	(23)	(24)	(25)	(26)	(27)	(28)
2.44×10^{10}	5.39×10^{-3}	.241	.240	2.21	1.15×10^{8}	.081	3.[illegible]	9.27	35	7.6×10^{-4}	1.85	3.42
2.44×10^{10}	5.39×10^{-3}	.241	.240	2.10	1.15×10^{8}	.081	3.[illegible]	8.00	35	7.6×10^{-4}	.63	2.85
2.44×10^{10}	5.39×10^{-3}	.241	.240	1.20	1.15×10^{8}	.081	[illegible]	4.00	-	7.6×10^{-4}	0	2.29
-	-	.241	.240	1.20	1.15×10^{8}	.081	3.[illegible]	4.00	-	7.6×10^{-4}	0	1.29
-	-	.241	.240	1.32	1.15×10^{8}	.081	3.[illegible]	4.00	-	7.6×10^{-4}	0	1.29
-	-	.241	.240	1.39	0.74×10^{8}	.081	3[illegible]	4.00	-	9.3×10^{-4}	0	0

* In one direction only at section with maximum requirements due to backfill

LOAD TABLE 3.2

Posi-tion	Dead Load T_N (k/ft)	Liner Temp. when Rebar Stress is a Maximum T-(°F)	Temp. at Neutral Surface T' (°F)	Temp. change causing stress in Liner ΔT (°F)	Pressure when Rebar Stress is a Maximum P(#/in^2)	Pressure Load P (k/ft)	Disc. Moment at the Base Mo (k-in/ft)	Disc. Moment M (k-in/ft)	Disc. Shear at the Base Qo (k/ft)	Disc. Shear V'u (k/ft)	Seismic Overturning Moment (k.in)	Uplift from Overturning Moment Teq (k/ft)	Uplift from Vertical Earthquake T_v (k/ft)	Seismic Shear Vs (k)	Shear Flow SkY (k/in)
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
3	126	300	-	-	70.5	354	14,530	9,740	157	127.0	0	0	0	0	0
4	114	300	87	213	70.5	354	14,530	2790	157	4.2	0	0	0	0	0
5	85	300	92	208	70.5	354	14,530		157	0	0	0	0	0	0
6	33	300	92	208	70.5	354				0	0	0	0	0	0
7	33	300	94	206	70.5	354				0	0	0	0	0	0
8	25	300	91	209	70.5	350				0	0	0	0	0	0

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

Vertical

LOAD TABLE 3.2

Position	Uplift Pressure Load	Uplift from Overturning Moment	Uplift from Vertical Earthquake	Dead Load	Total Uplift	Disc. Moment	Disc. Shear	Dead Wt. of Outer Base Mat	Column Reaction	Pressure when Rebar Stress is a Maximum
	P (k/ft)	T_{TO} (k/ft)	T_V (k/ft)	T_{DL} (k/ft)	U_T (k/ft)	M (k-in/ft)	V'u (k/ft)	D (k/ft)	C (k/ft)	P (#/ft^2)
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)
1	237	145	18	128	27?	9700	105	6.38	2.17	47
2	237	145	18	128	272	9700	105	6.38	2.17	47

CONTINUED

Position	Uniform Load on Base Mat	Dead Wt. of the Crane Wall	Dead Wt. of the Operating Floor	Pressure Load on Top of Crane Wall	Total Crane Wall Reaction	Total Shear	Total Moment	Allowable Shear Stress in Conc.	Shear Taken by Conc.	Total Shear Carried by Stirrups	Dead Wt. from Backfill
	W (k/in/ft)	D (k/ft)	D (k/ft)	P (k/ft)	R (k/ft)	V (k/ft)	M (in-k/ft)	v (k/in^2)	V (k/ft)	V_s (k/ft)	E (k/ft)
(1)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	
1	.714	22.5	8.2	20.3	51	117	-23,193	.093	111.5	5.5	3
2	.714	22.5	8.2	20.3	51	238.95	6,799	.093	111.5	127.45	3

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

LOAD TABLE 3.2

Position (1)	Uplift Pressure Load P (k/ft) (2)	Uplift from Overturning Moment T_{E} (k/ft) (3)	Uplift from Vertical Earthquake T_v (k/ft) (4)	Dead Load T_{DL} (k/ft) (5)	Total Uplift U_T (k/ft) (6)	Disc. Moment M (k-in/ft) (7)	Disc. Shear V'u (k/ft) (8)	Dead Wt. of Outer Base Mat D (k/ft) (9)	Column Reaction C (k/ft) (10)	Pressure when Rebar Stress is a Maximum P (#/ft^2) (11)
1	298	121	11	128	302	12,180	131	4.38	2.17	59
2	298	121	11	128	302	12,180	131	4.38	2.17	59

CONTINUED

Position (1)	Uniform Load on Base Mat W (k/in/ft) (12)	Dead Wt. of the Crane Wall D (k/ft) (13)	Dead Wt. of the Operating Floor D (k/ft) (14)	Pressure Load on Top of Crane Wall P (k/ft) (15)	Total Crane Wall Reaction R (k/ft) (16)	Total Shear V (k/ft) (17)	Total Moment M (in-k/ft) (18)	Allowable Shear Stress in Conc. v (k/in^2) (19)	Shear Taken by Conc. V (k/ft) (20)	Total Shear Carried by Stirrups V_s (k/ft) (21)	Dead Wt. from Backfill E (k/ft)
1	.858	22.5	8.2	25.5	56.2	120	-22,573	.093	111.5	8.5	3
2	.858	22.5	8.2	25.5	56.2	264	9,678	.093	111.5	152.7	3

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

LOAD TABLE 3.2

Position	Uplift Pressure Load	Uplift from Overturning Moment	Uplift from Vertical Earthquake	Dead Load	Total Uplift	Disc. Moment	Disc. Shear	Dead Wt. of Outer Base Mat	Column Reaction	Pressure when Rebar Stress is a Maximum
	P (k/ft)	T_{om} (k/ft)	T_v (k/ft)	T_{DL} (k/ft)	U_T (k/ft)	M (k-in/ft)	V'u (k/ft)	D (k/ft)	C (k/ft)	P (#/in^2)
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)
1	354	0	0	128	226	14,530	157	4.38	2.17	70.5
2	354	0	0	128	226	14,530	157	4.38	2.17	70.5

CONTINUED

Position	Uniform Load on Base Mat	Dead Wt. of the Crane Wall	Dead Wt. of the Operating Floor	Pressure Load on Top of Crane Wall	Total Crane Wall Reaction	Total Shear	Total Moment	Allowable Shear Stress in Conc.	Shear Taken by Conc.	Total Shear Carried by Stirrups	Dead Wt. from Backfill
	W (k/in/ft)	D (k/ft)	D (k/ft)	P (k/ft)	R (k/ft)	V (k/ft)	M* (in-k/ft)	v_c (k/in^2)	V_c (k/ft)	V_s (k/ft)	E (k/ft)
(1)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)	(20)	(21)	
	.996	22.5	8.2	30.4	61.1	14.18	-924	.093	111.5	0	3
2	.996	22.5	8.2	30.4	61.1	183.56	16,103	.093	111.5	72	3

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

* indicates tension on the bottom

LOAD TABLE 3.2

Position (1)	Dead Load T_{DL} (k/ft) (2)	Liner Temp. when Rebar Stress is a Maximum T-(°F) (3)	Temp. at Neutral Surface T' (°F) (4)	Temp. change causing Stress in Liner ΔT (°F) (5)	Pressure when Rebar Stress is a Maximum P(#/in^2) (6)	Pressure Load P (k/ft) (7)	Disc. Moment at the Base Mo (k-in/ft) (8)	Disc. Moment M (k-in/ft) (9)	Disc. Shear at the Base Q_o (k/ft) (10)	Disc. Shear V'u (k/ft) (11)	Seismic Overturning Moment (k-in) (12)	Uplift from Overturning Moment Teq (k/ft) (13)	Uplift from Vertical Earthquake T_v (k/ft) (14)	Seismic Shear Vs (k) (15)	Shear Flow ShF (k/in) (16)
*3	-	300	-	-	70.5	-	0	0	0	0	0	0	0	0	0
4	0	300	87	213	70.5	708	0	0	0	0	0	0	0	0	0
5	0	300	87	213	70.5	708	0	0	0	0	0	0	0	0	0
6	35	300	87	213	70.5	708	0	0	0	0	0	0	0	0	0
7	35	300	95	205	70.5	354	0	0	0	0	0	0	0	0	0
8	6	300	97	203	70.5	350	0	0	0	0	0	0	0	0	0

* Point 3 is almost fully restrained and the horizontal forces will be insignificant

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

Horizontal

LOAD TABLE 3.2

Position	Dead Load T_{DL} (k/ft)	Liner Temp. when Rebar Stress is a Maximum T-(°F)	Temp. at Neutral Surface T' (°F)	Temp. change causing stress in Liner ΔT (°F)	Pressure when Rebar Stress is a Maximum P(#/in^2)	Pressure Load P (k/ft)	Disc. Moment at the Base Mo (k-in/ft)	Disc. Moment M (k-in/ft)	Disc. Shear at the Base Q_o (k/ft)	Disc. Shear V'u (k/ft)	Seismic Overturning Moment (k-in)	Uplift from Overturning Moment Teq (k/ft)	Uplift from Vertical Earthquake T_v (k/ft)	Seismic Shear Vs (k)	Shear Flow ShF (k/in)
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
3	126	278	-	-	59	298	12,180	8,150	131	106.0	27.85×10^6	[illegible]	11	2.70×10^4	20.50
4	114	278	87	191	59	298	12,180	2,340	131	3.5	24.00×10^6	[illegible]	10	1.80×10^4	13.69
5	85	278	92	186	59	298	12,180	0	131	0	14.10×10^6	33	7	1.40×10^4	10.62
6	33	278	92	186	59	298	-	0	-	0	3.67×10^6	13	3	$.70 \times 10^4$	5.34
7	33	278	94	184	59	298	-	0	-	0	3.67×10^6	13	3	$.70 \times 10^4$	5.34
8	22	278	91	187	59	293	-	0	-	0	1.01×10^6	5	2	$.30 \times 10^4$	2.83

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

Vertical

LOAD TABLE 3.2

Position	Dead Load T_m (k/ft)	Liner Temp. when Rebar Stress is a Maximum T-(°F)	Temp. at Neutral Surface T' (°F)	Temp. change causing Stress in Liner ΔT (°F)	Pressure when Rebar Stress is a Maximum P(#/in²)	Pressure Load P (k/ft)	Disc. Moment at the Base Mo (k-in/ft)	Disc. Moment M (k-in/ft)	Disc. Shear at the Base Q_o (k/ft)	Disc. Shear V'u (k/ft)	Seismic Overturning Moment (k-in)	Uplift from Overturning Moment Teq (k/ft)	Uplift from Vertical Earthquake T_v (k/ft)	Seismic Shear Vs (k)	Shear Flow ShF (k/in)
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
3	-	278	-	-	59	-	0	0	0	0	0	0	0	0	0
4	0	278	87	191	59	596	0	0	0	0	0	0	0	0	0
5	0	278	87	191	59	596	0	0	0	0	0	0	0	0	0
6	35	278	87	191	59	596	0	0	0	0	0	0	0	0	0
7	35	278	95	183	59	298	0	0	0	0	0	0	0	0	0
8	6	278	97	181	59	293	0	0	0	0	0	0	0	0	0

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

Horizontal

LOAD TABLE 3.2

Posi- tion	Dead Load T_{DL} (k/ft)	Liner Temp. when Rebar Stress is a Maximum T-(°F)	Temp. at Neutral Surface T' (°F)	Temp. change causing Stress in Liner ΔT (°F)	Pressure when Rebar Stress is a Maximum P(#/in^2)	Pressure Load P (k/ft)	Disc. Moment at the Base Mo (k-in/ft)	Disc. Moment M (k-in/ft)	Disc. Shear at the Base Q_o (k/ft)	Disc. Shear V'u (k/ft)	Seismic Overturning Moment (k-in)	Uplift from Overturning Moment Teq (k/ft)	Uplift from Vertical Earthquake T_v (k/ft)	Seismic Shear Vs (k)	Shear Flow ShF (k/in)
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)	(11)	(12)	(13)	(14)	(15)	(16)
3	126	239	-	-	47	237	9,700	6,500	105	84.0	$33.4x10^5$	143	18	$2.50x10^4$	19.00
4	114	239	87	152	47	237	9,700	1,860	105	2.8	$28.8x10^6$	120	16	$2.16x10^4$	16.40
5	85	239	92	147	47	237	9,700	0	105	0	$16.9x10^6$	66	11	$1.68x10^4$	12.78
6	33	239	92	147	47	237	-	0	-	0	$4.4x10^6$	16	5	$.84x10^4$	6.38
7	33	239	94	145	47	237	-	0	-	0	$4.4x10^6$	16	5	$.84x10^4$	6.38
8	22	239	91	148	47	235	-	0	-	0	$1.2x10^6$	6	3	$.37x10^4$	3.40

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

Vertical

LOAD TABLE 3.2

Position (1)	Dead Load T_{DL} (k/ft) (2)	Liner Temp. when Rebar Stress is a Maximum T-(°F) (3)	Temp. at Neutral Surface T' (°F) (4)	Temp. change causing Stress in Liner ΔT (°F) (5)	Pressure when Rebar Stress is a Maximum P(#/in^2) (6)	Pressure Load P (k/ft) (7)	Disc. Moment at the Base Mo (k-in/ft) (8)	Disc. Moment M (k-in/ft) (9)	Disc. Shear at the Base Q_o (k/ft) (10)	Disc. Shear V'u (k/ft) (11)	Seismic Overturning Moment (k-in) (12)	Uplift from Overturning Moment Teq (k/ft) (13)	Uplift from Vertical Earthquake T_v (k/ft) (14)	Seismic Shear Vs (k) (15)	Shear Flow ShF (k/in) (16)
3	-	239	-	-	47	-	0	0	0	0	0	0	0	0	0
4	0	239	87	152	47	474	0	0	0	0	0	0	0	0	0
5	0	239	87	152	47	474	0	0	0	0	0	0	0	0	0
6	35	239	87	152	47	474	0	0	0	0	0	0	0	0	0
7	35	239	95	144	47	237	0	0	0	0	0	0	0	0	0
8	6	239	97	142	47	235	0	0	0	0	0	0	0	0	0

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

Horizontal

3.3.10 SUMMARY OF CONTAINMENT DESIGN STRESSES COMPARED TO CRITICAL STRESS LEVELS

In Table 3.3 is presented the stress resultants for the loads given for selected points in Table 3.2 of Section 3.3.9. The Table also presents a comparison between resultant stress and allowable stress levels.

SUMMARY OF CONTAINMENT DESIGN STRESSES COMPARED TO CRITERIA STRESS LEVELS

REBAR STRESS TABLE 3.3

Position (1)	Dead Load Strain (2)	Pressure Load Strain (3)	Horizontal Earthquake Uplift Strain (4)	Vertical Earthquake Uplift Strain (5)	Total Strain from Uplift ϵ_T (6)	Strain from ΔT $-\epsilon\Delta T$ (7)	Total Strain $-\epsilon x$ (8)	Total Liner Strain Considering Effect of Horizontal Strain $-\epsilon_{TL'_x}$ (9)	Total Vertical Rebar Strain from Interaction Effects $\epsilon_{TL'_x}$ (10)
3	.164	.461	0	0	.297	* -	-	-	-
4	.155	.483	0	0	.328	1.386	1.058	.767	.288
5	.186	.777	0	0	.591	1.351	.760	.520	.241
6	.079	.854	0	0	.775	1.351	.576	.382	.196
7	.072	.772	0	0	.700	1.339	.639	.366	.279
8	.065	.725	0	0	.680	1.358	.688	.417	.271

Total Vertical Rebar Strain $\epsilon_T+\epsilon_{TL'_x}$	Total Vertical Rebar Stress from Uplift & Interaction (k/in^2) (12)	Vertical Rebar Stress from Discontinuity Moment (k/in^2) (13)	Total Vertical Rebar Stress (k/in^2) (14)	Allowable Rebar Stress (k/in^2) (15)	Stress in Stirrups (k/in^2) (16)	Allowable Stress in Stirrups (k/in^2) (17)	Stress in Diagonal Bars (k/in^2) (18)	Allowable Stress in Diagonal Bars (k/in^2) (19)
-	8.6	25.5	34.1	54	37.8	51	0	57
.616	17.8	8.3	26.1	54	3.7	51	0	57
.832	24.1	0	24.1	57	0	51	0	57
.971	28.1	0	28.1	57	0	51	0	57
.979	28.4	0	28.4	57	0	51	0	57
.951	27.[illegible]	0	27.6	57	0	51	0	57

Note: All Strains x 10^{-3}
To convert rebar strains to Stress multiply by 29×10^6
* No temperature effects need be considered since Point 3 is in an insulated Zone

$1.0\ D \pm 0.05\ D + 1.5\ P + 1.?\ (T + TL)$

Vertical

REBAR STRESS TABLE 3.3

Position (1)	Dead Load Strain (2)	Pressure Load Strain (3)	Horizontal Earthquake Uplift Strain (4)	Vertical Earthquake Uplift Strain (5)	Total Strain from Pressure ϵ_T (6)	Strain from ΔT $-\epsilon\Delta T$ (7)	Total Strain $-\epsilon_y$ (8)
1							
2							
*3	-	-	-	-	-	-	-
4	0	.744	0	0	.744	1.383	.639
5	0	1.131	C	0	1.131	1.383	.292
6	.060	1.219	0	0	1.284	1.383	.099
7	.083	.839	0	0	.922	1.337	.415
8	.016	.947	0	0	.963	1.320	.357

CONTINUED

Total Liner Strain Considering Effect of Vertical Strain $-\epsilon_{TLy}$ (9)	Total Horizontal Rebar Strain from Interaction Effects ϵ_{TLy} (10)	Total Horizontal Rebar Strain $\epsilon_T+\epsilon_{TLy}$ (11)	Total Horizontal Rebar Stress from Pressure and Interaction (k/in^2) (12)	Horizontal Rebar Stress from Discontinuing Moment (k/in^2) (13)	Total Horizontal Rebar Stress (k/in^2) (14)	Allowable Rebar Stress (k/in^2) (15)
-	-	-	-	-	-	-
.400	.224	.968	28.1	0	28.1	57
.130	.089	1.220	35.4	0	35.4	57
.070	.052	1.336	38.7	0	38.7	57
.240	.248	1.170	33.9	0	33.9	57
.140	.231	1.191	34.6	0	34.6	57

(Position-Lines 1 and 2 blank)

Note: All Strains x 10^{-3}

$1.0\ D \pm 0.05D' + 1.5\ P + 1.0\ (T+TL)$

Horizontal

*Point 3 is almost fully Restrained and the Horizontal Forces will be Insignificant

REBAR STRESS TABLE 3.3

Position	Dead Load Strain	Pressure Load Strain	Horizontal Earthquake Uplift Strain	Vertical Earthquake Uplift Strain	Total Strain from Uplift ϵ'_T	Strain from ΔT $-\alpha\Delta T$	Total Strain $-\epsilon x$	Total Liner Strain Considering Effect of Horizontal Strain $-\bar{\epsilon}_{TL_x}$	Total Vertical Rebar Strain from Interaction Effects $\epsilon_{TL'_x}$
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)	(9)	(10)
3	.164	.388	.156	.014	.394	-	-	-	-
4	.155	.406	.13	.014	.401	1.240	.839	.602	.234
5	.188	.654	.121	.015	.602	1.209	.607	.411	.197
6	.080	.718	.031	.007	.676	1.209	.533	.349	.183
7	.072	.650	.028	.007	.613	1.200	.587	.335	.251
8	.046	.607	.010	.004	.575	1.213	.638	.385	.254

CONTINUED

Position	Total Vertical Rebar Strain $\epsilon_T+\epsilon_{TL'_x}$	Total Vertical Rebar Stress from Uplift & Interaction (k/in^2)	Vertical Rebar Stress from Discontinuity Moment (k/in^2)	Total Vertical Rebar Stress (k/in^2)	Allowable Rebar Stress (k/in^2)	Stress in Stirrups (k/in^2)	Allowable Stress in Stirrups (k/in^2)	Stress in Diagonal Bars (k/in^2)	Allowable Stress in Diagonal Bars (k/in^2)
(1)	(11)	(12)	(13)	(14)	(15)	(16)	(17)	(18)	(19)
3	-	11.9	21.2	33.1	54	31.5	51	51.0	57
4	.635	18.4	6.9	25.3	54	3.04	51	40.6	57
5	.799	23.2	0	23.2	57	0	51	39.4	57
6	.859	24.9	0	24.9	57	0	51	35.0	57
7	.864	25.4	0	25.1	57	0	51	35.0	57
8	.829	24.1	0	24.1	57	0	51	0	57

Note: All Strains x 10^{-3}

$$1.0\ D \pm 0.05\ D + 1.25\ P + 1.0\ (T^t + TL^t) + 1.25\ E$$

Vertical

REBAR STRESS TABLE 3.3

Position (1)	Dead Load Strain (2)	Pressure Load Strain (3)	Horizontal Earthquake Uplift Strain (4)	Vertical Earthquake Uplift Strain (5)	Total Strain from Pressure ϵ_T (6)	Strain from ΔT $-c\Delta T$ (7)	Total Strain $-\epsilon y$ (8)
3	-	-	-	-	-	-	-
4	0	.623	0	0	.623	1.241	.618
5	0	.949	0	0	.949	1.241	.292
6	.060	1.026	0	0	1.086	1.241	.155
7	.083	.707	0	0	.790	1.190	.400
8	.016	.793	0	0	.809	1.178	.369

CONTINUED

Position (1)	Total Liner Strain Considering Effect of Vertical Strain ϵ_{TLy} (9)	Total Horizontal Rebar Strain from Interaction Effects $\epsilon_{TL'y}$ (10)	Total Horizontal Rebar Strain $\epsilon_T+\epsilon_{TL'y}$ (11)	Total Horizontal Rebar Stress from Pressure and Interaction (k/in²) (12)	Horizontal Rebar Stress from Discontinuing Moment (k/in²) (13)	Total Horizontal Rebar Stress (k/in²) (14)	Allowable Rebar Stress (k/in²) (15)
3	-	-	-	-	-	-	-
4	.410	.212	.835	24.2	0	24.2	57
5	.200	.087	1.036	30.0	0	30.0	57
6	.100	.058	1.144	33.1	0	33.1	57
7	.200	.212	1.002	29.1	0	29.1	57
8	.150	.234	1.043	30.0	0	30.0	57

Note: All Strains $\times 10^{-3}$

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

Horizontal

REBAR STRESS TABLE 3.3

Position	Dead Load Strain	Pressure Load Strain	Horizontal Earthquake Uplift Strain	Vertical Earthquake Uplift Strain	Total Strain from Pressure ϵ_T	Strain from ΔT $-\epsilon\Delta T$	Total Strain $-\epsilon_y$
(1)	(2)	(3)	(4)	(5)	(6)	(7)	(8)
	-	-	-	-	-	-	-
3	-	-	-	-	.495	.923	.427
4	0	.496	0	0	.756	.923	.167
5	0	.756	0	0	.876	.923	.047
6	.060	.816	0	0	.645	.870	.225
7	~~.023~~	.562	0	0	.651	.923	.272
8	.016	.635	0	0			

CONTINUED

Position	Total Liner Strain Considering Effect of Vertical Strain $-\epsilon'_{TLy}$	Total Horizontal Rebar Strain from Interaction Effects $\epsilon_{TL'y}$	Total Horizontal Rebar Strain $\epsilon_T+\epsilon_{TL'y}$	Total Horizontal Rebar Stress from Pressure and Interaction (k/in^2)	Horizontal Rebar Stress from Discontinuing Moment (k/in^2)	Total Horizontal Rebar Stress (k/in^2)	Allowable Rebar Stress (k/in^2)
(1)	(9)	(10)	(11)	(12)	(13)	(14)	(15)
	-	-	-	-	-	-	-
3	.290	.158	.654	18.9	0	[illegible]	37
4	.110	.054	.810	23.5	0	23.5	[illegible]
5	.020	.027	.903	26.2	0	26.2	57
6	.100	.115	.760	22.0	0	22.0	57
7	.110	.176	.827	24.0	0	19.1	57
8							

1.0 D ± 0.05D + 1.0 P+1.0(T"+TL") + E'

Horizontal

REBAR STRESS TABLE 3.3

Position (1)	Dead Load Strain (2)	Pressure Load Strain (3)	Horizontal Earthquake Uplift Strain (4)	Vertical Earthquake Uplift Strain (5)	Total Strain from Uplift ϵ_T (6)	Strain from ΔT $-\alpha\Delta T$ (7)	Total Strain $-\epsilon_x$ (8)	Total Linear Strain Considering Effect of Horizontal Strain $-\epsilon'_{TL_x}$ (9)	Total Vertical Rebar Strain from Interaction Effects ϵ'_{TL_x} (10)
3	.164	.307	.186	.023	.352	-	-	-	-
4	.155	.323	.165	.022	.353	.988	.635	.456	.176
5	.188	.520	.146	.024	.502	.956	.454	.310	.144
6	.080	.572	.039	.012	.543	.956	.413	.276	.137
7	.072	.517	.039	.011	.491	.943	.452	.263	.204
8	.046	.488	.012	.006	.460	.962	.502	.304	.199

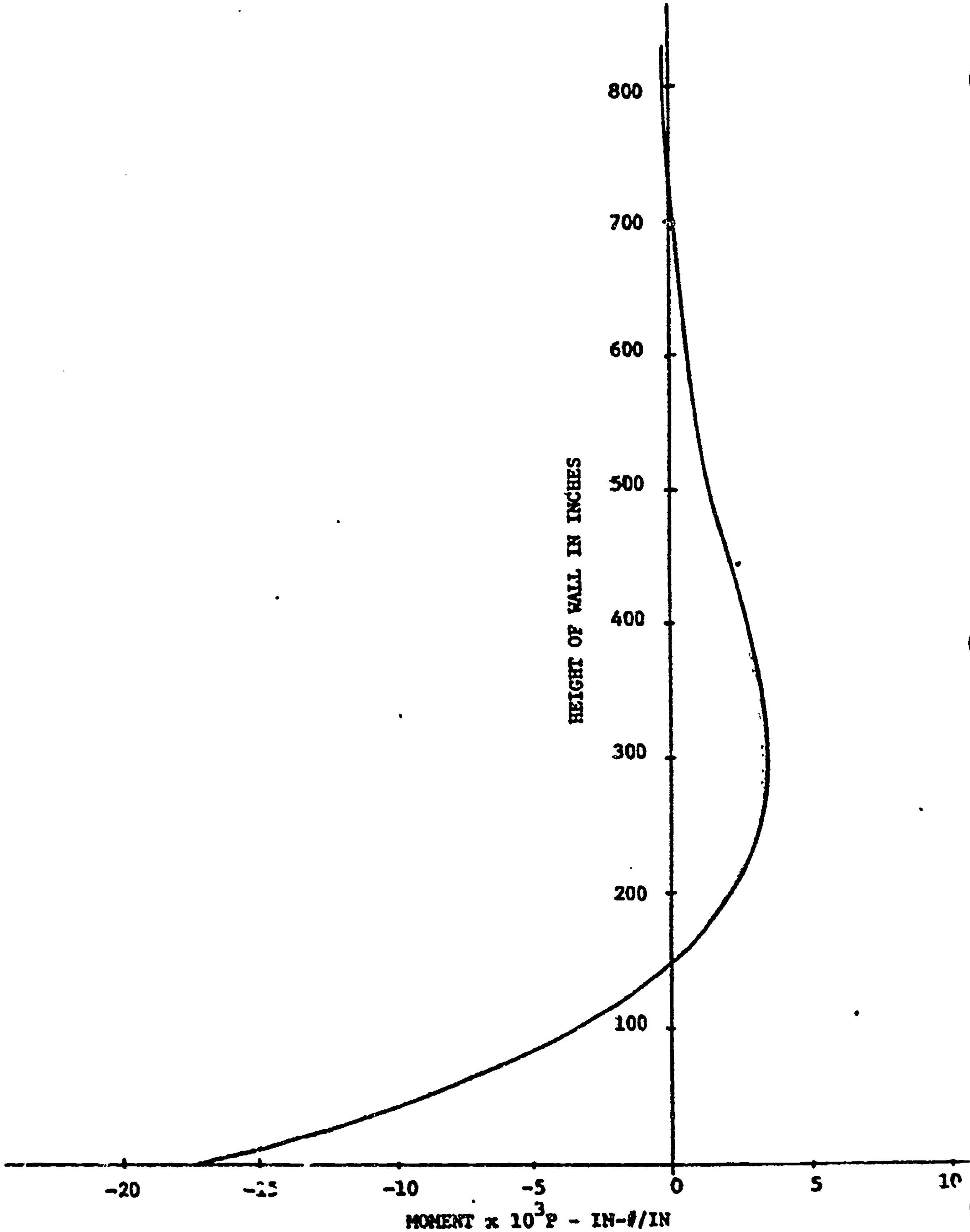
CONTINUED

Position (1)	Total Vertical Rebar Strain $\epsilon_T + \epsilon_{TLx}$ (11)	Total Vertical Rebar Stress from Uplift & Interaction (k/in^2) (12)	Vertical Rebar Stress from Discontinuity Moment (k/in^2) (13)	Total Vertical Rebar Stress (k/in^2) (14)	Allowable Rebar Stress (k/in^2) (15)	Stress in Stirrups (k/in^2) (16)	Allowable Stress in Stirrups (k/in^2) (17)	Stress in Diagonal Bars (k/in^2) (18)	Allowable Stress in Diagonal Bars (k/in^2) (19)
3	-	10.4	17.0	27.4	54	25.2	51	47.2	57
4	.529	15.3	5.5	20.8	54	2.4	51	48.8	57
5	.646	18.7	0	18.7	57	0	51	47.3	57
6	.680	19.7	0	19.7	57	0	51	42.0	57
7	.695	20.1	0	20.1	57	0	51	42.0	57
8	.659	19.1	0	19.1	57	0	51	0	57

Note: All Strains x 10^{-3}

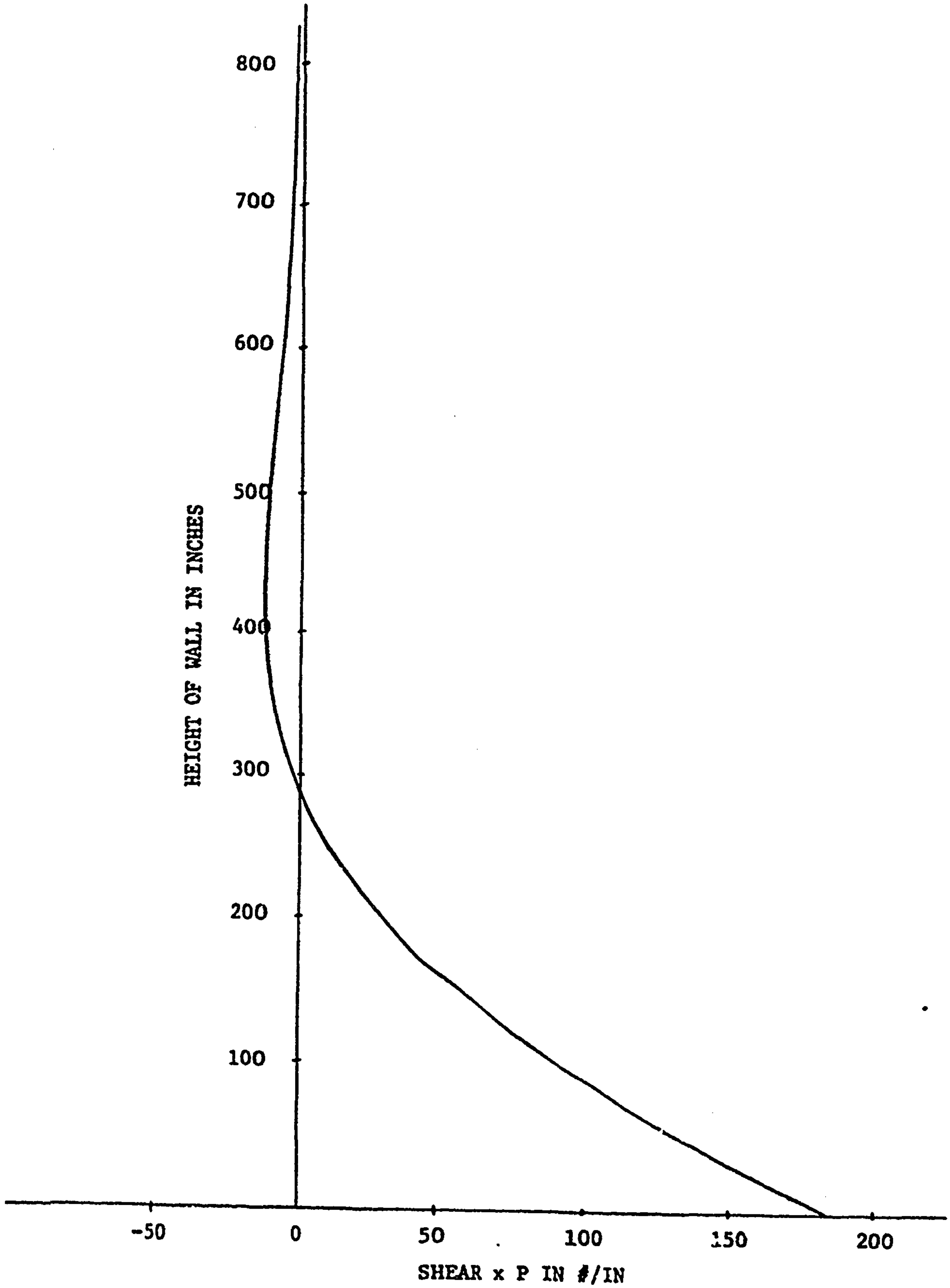
$1.0\ D \pm 0.05\ D + 1.0\ P + 1.0\ (T'' + TL'') + E'$

Vertical



DISCONTINUITY MOMENT VS. WALL HEIGHT

FIGURE 3.1



DISCONTINUITY SHEAR VS. WALL HEIGHT
(RADIALLY IN IS POSITIVE)

FIGURE 3.2

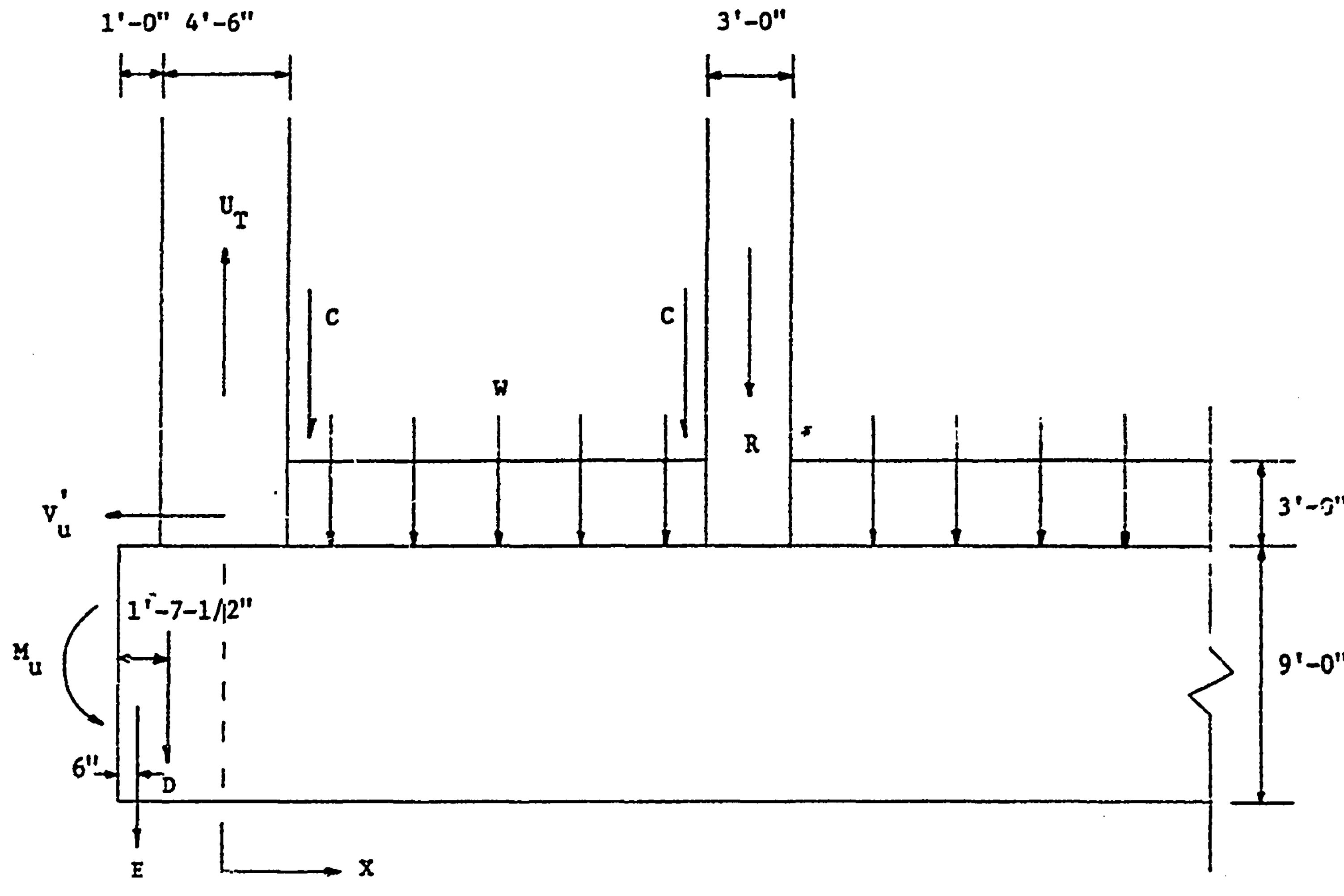
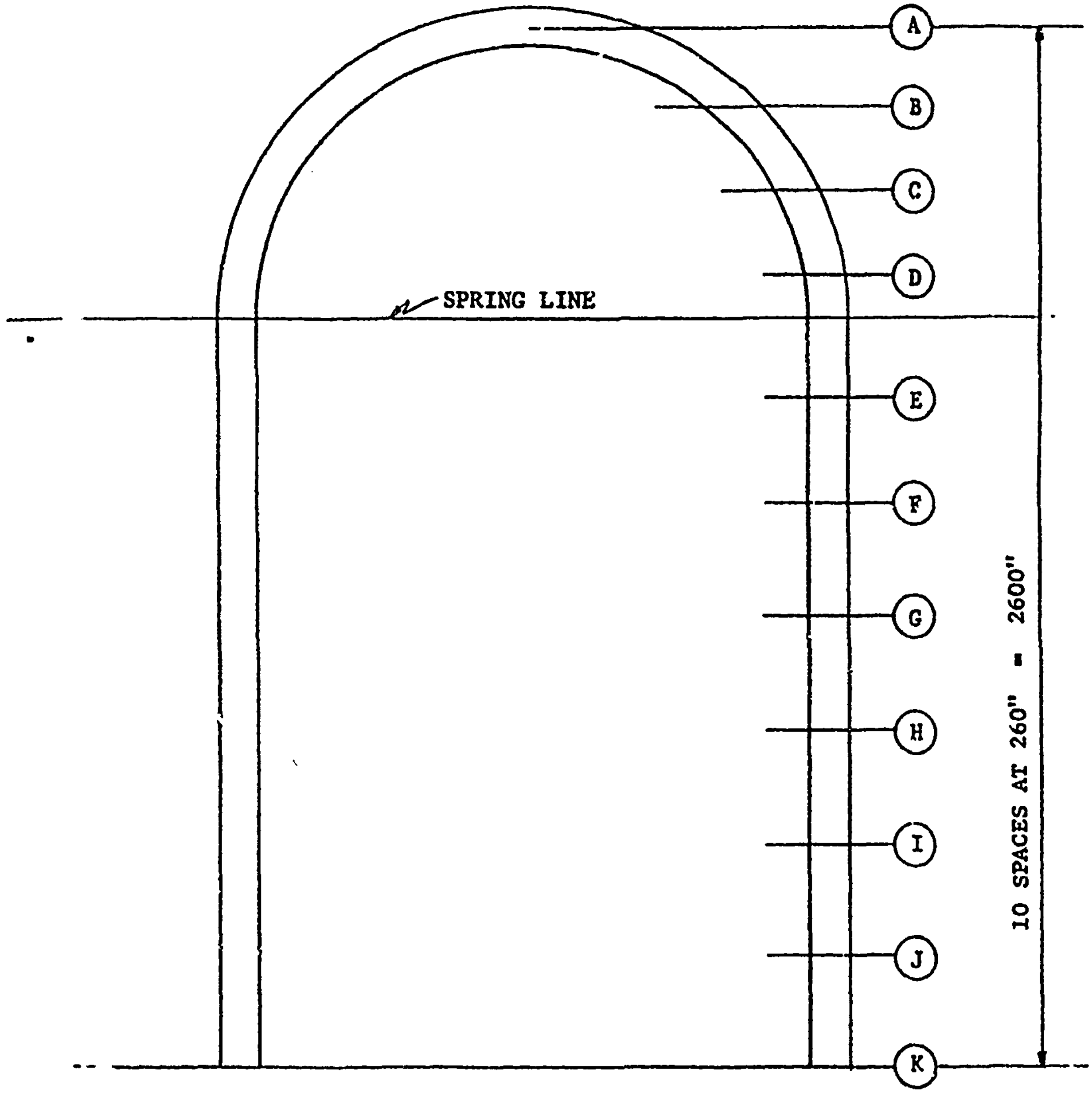


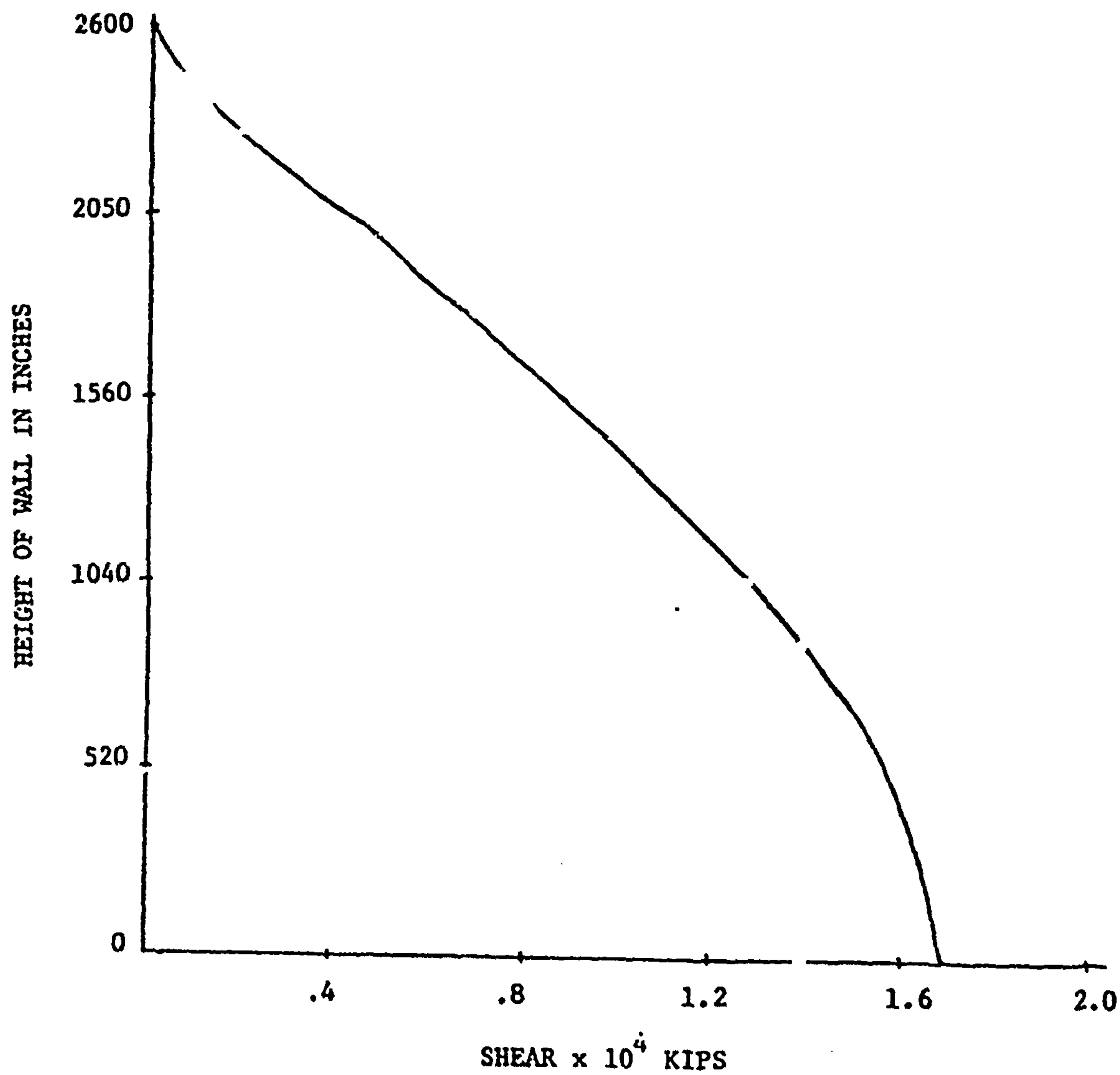
FIGURE 3.3

LOADS ON CONTAINMENT BASE MAT



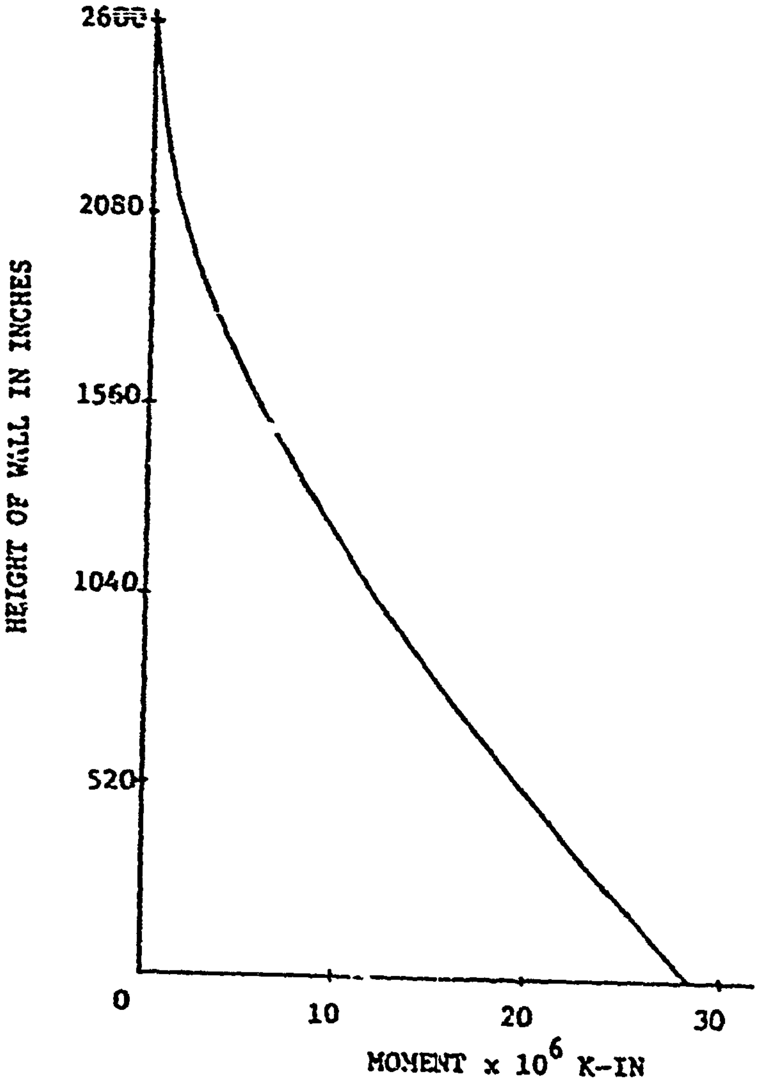
LOCATION OF NODE POINTS FOR SEISMIC ANALYSIS

FIGURE 3.4



SEISMIC SHEAR VS. WALL HEIGHT

FIGURE 3.5



SEISMIC MOMENT VS. WALL HEIGHT

FIGURE 3.6

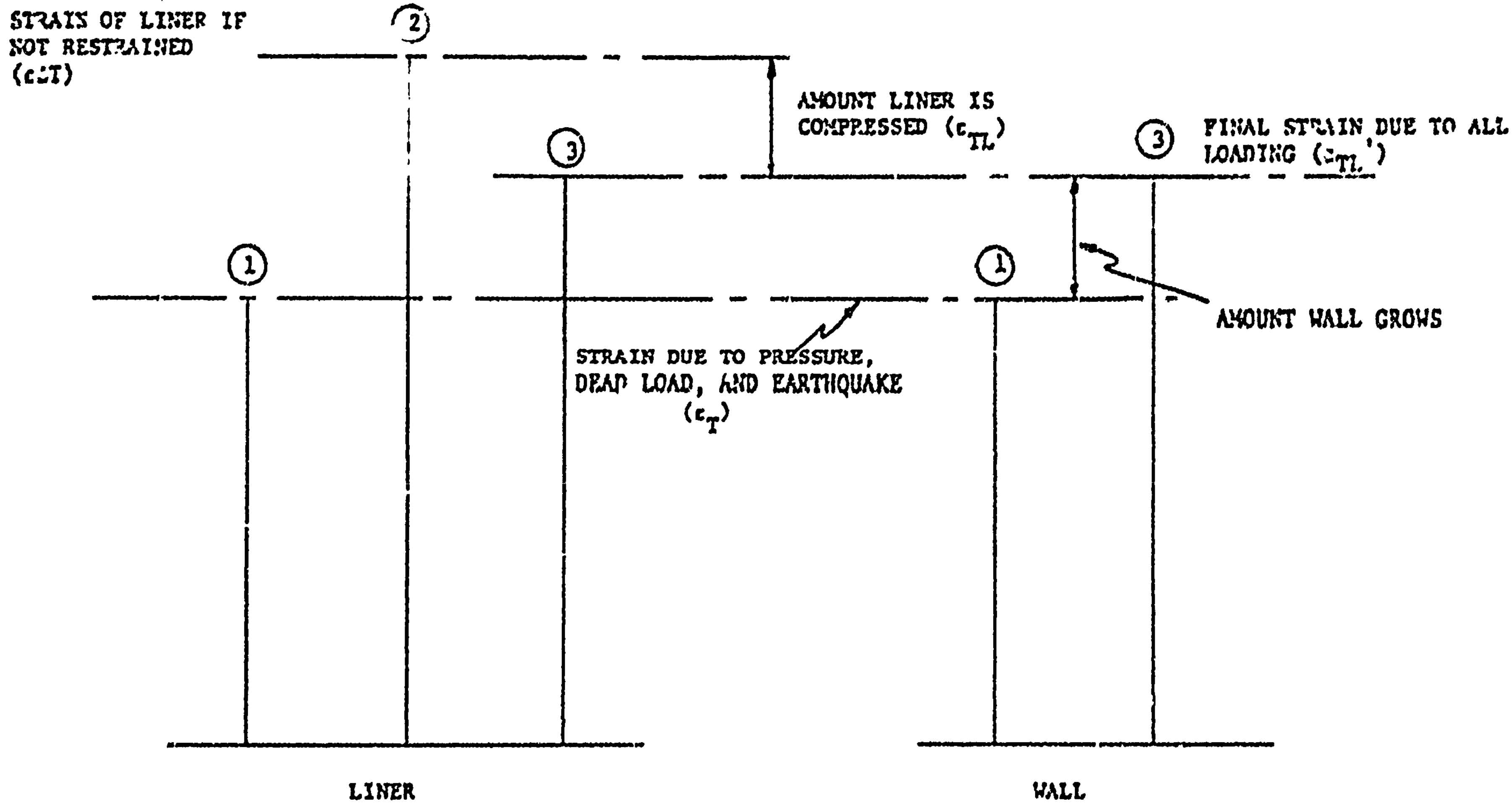
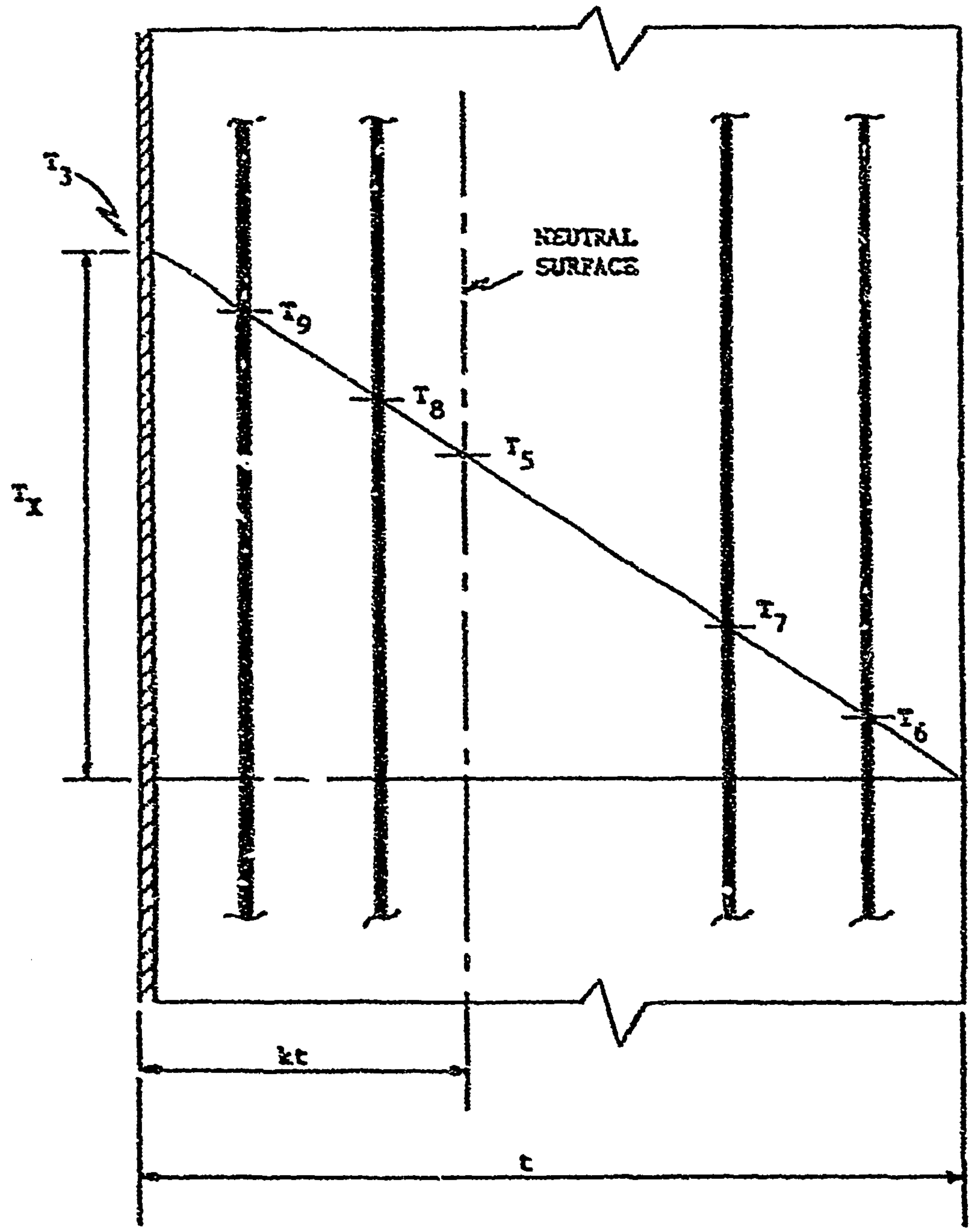


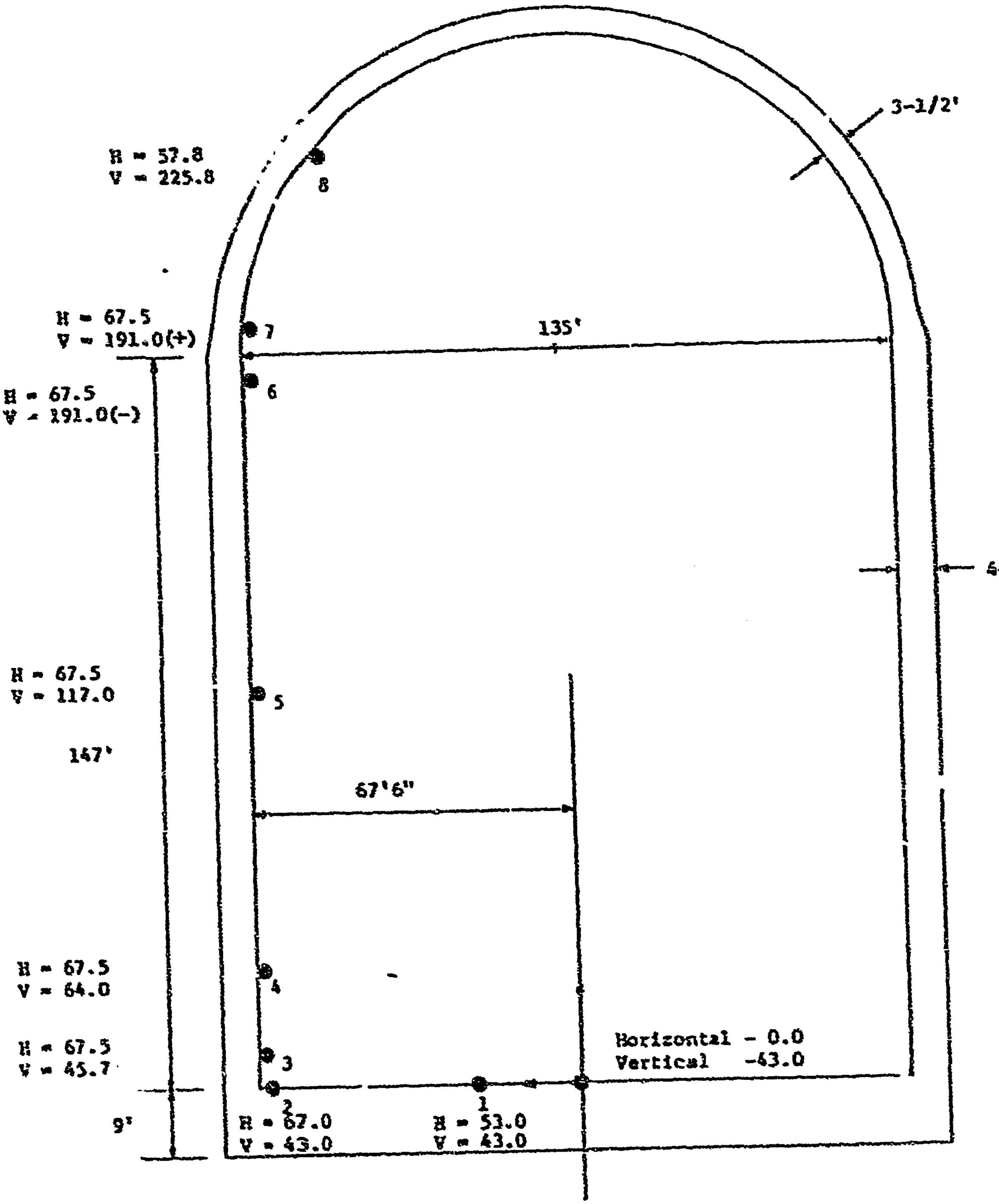
FIGURE 3.7

TEMPERATURE INTERACTION EFFECTS BETWEEN LINER AND WALL



OPERATING TEMPERATURE GRANDIENT THROUGH
THE CONTAINMENT WALL

FIGURE 3.8



COORDINATES OF CONTAINMENT POINTS CONSIDERED IN REPORT

FIGURE 3.9

CONTAINMENT DESIGN REPORT

3.6.0 Equipment Hatch and Personnel Lock - Boss Design

3.6.1 Introduction

There are two large openings in the Indian Point - Unit No. 2 Containment Structure. The Personnel Lock is located in the South East quadrant with a center line elevation of 83'-6 and an opening size of 8'-6 diameter. The Equipment Hatch is located in the North East quadrant of the Containment with a center line elevation of 101'-6 and opening size of 16'-0 diameter. Both of these openings along with their thickened reinforced concrete bosses are located a sufficient distance above the fixed base mat at El. 43'-0 that all moments and shears created at this discontinuity have substantially dissipated in the hatch area.

Both hatch and lock are constructed of ASTM 516 GR 60 (formerly A201 GRB) steel normalized to meet the requirements of ASTM A300. The material has been impact tested to meet the requirements of Section N331 of Section III of the ASME Boiler and Unfired Pressure Vessel Code.

All reinforcing steel in the cylindrical wall and the heavily reinforced hatch areas is high - strength deformed billet steel bars conforming to ASTM Designation A432-65 "Specification For Deformed Billet Steel Bars for Concrete Reinforcement With 60,000 psi Minimum Yield Strength". This steel has a minimum tensile strength of 90,000 psi and a minimum elongation of 7% in an 8-in. specimen. Bars No. 14S and 18S are spliced by the Cadweld process only. The splices used to join these bars are designed to develop at least 125% of the minimum yield point stress of the bar.

The plate steel liner inside the cylindrical wall including the hatch areas is carbon steel conforming to ASTM Designation A442-65 Grade 60 "Standard Specification for Carbon Steel Plates With Improved Transition Properties". 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% in an 8-in. gauge length at failure. The liner material is 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 and Pressure Vessel Code.

Internal forces and stresses in the concrete containment shell were determined for the factored load combinations listed in Section 3.4.3.1. In verifying the adequacy of resistance to these factored loads, capacity reduction factors recommended in ACI 318-63 Building Code Requirements for Reinforced Concrete were applied where applicable.

Under loadings which include incident pressure and temperature, some local yielding of the liner may occur; however, this has no adverse strength implications for the containment wall. Moreover the ductility of the liner fastening studs is sufficient to tolerate local inelastic buckling without stud failure.

Containment Design Report

3.4.1 Introduction (Continued)

Under load combinations 4, 5, and 6 on Page 5 from which the thermal effects have been deleted, and with the liner contribution to strength disregarded, calculated rebar stresses do not exceed ϕf_y (where ϕ is the capacity reduction factor). Under load combination 1 involving a factor not greater than 1.0 on reactor incident, and with the liner stress (and temperature) accounted for, calculated rebar stresses do not exceed ϕf_y. Under factored load combinations 2 and 3 involving a factor greater than 1.0 on reactor incident, and with the liner (and temperature loads) accounted for, a limited amount of local rebar yielding is permitted. These criteria guarantee not only assured resistance to the active loads but also minimize any local inelastic strains which may be associated with stress redistribution due to local rebar yielding.

For further discussion see Section 3.4.4.

The hatch and lock are anchored into reinforced concrete bosses by means of stud anchors. Along the Equipment Hatch there are 16 rows of 5/8" ϕ x 12" long studs with 100 per row around the hatch for a total of 1600 studs. Along the Personnel Lock there are 9 rows of 5/8" ϕ x 12" long studs with 44 per row around the lock for a total of 396 studs. In the areas adjacent to the penetrations, the liner is thickened to 3/4" and is anchored into the concrete by hooked L - anchors of ½" ϕ x 9" long (minimum including 2" hook).

The reinforced concrete bosses are thickened to 7'-6 at the Equipment Hatch and 5'-6 at the Personnel Lock. The bosses have flat outside faces and a smooth transition to the dimensions of the wall beyond the effects of the discontinuities (see Fig. 9321-L-1559).

The hatch and lock have been designed to withstand the internal Containment pressure plus operating and earthquake loads associated with the design accident in accordance with Section III Subsection B of the ASME Boiler & Pressure Vessel Code - Nuclear Vessels. The anchors have been designed to transmit these loads back into the reinforced concrete boss.

Both the Equipment Hatch and Personnel Lock penetrate the concrete shell. In the case of the 16' ϕ Equipment Hatch, a personnel lock is mounted in the head of the hatch and transmits all pressure loads thru the barrel to the concrete when the inside door is closed. Should the personnel door be left open on this lock, the temperature and pressure loads are transmitted to the lock but not into the concrete due to the space between the lock and hatch. Where the 8'-6" Personnel Lock is mounted in the concrete, the temperature and pressure loads inside the lock are transmitted to the concrete if the inside door is left open. (See figure 9321-L-1567).

3.4.1 Introductirn (Continued)

Rebar stresses were determined for internal forces and moment obtained from finite element analyses performed by FIRL (see Section 3.4.3). The reinforcing steel, as designed by UE&C was found to meet the criteria stated above.

3.4.2 Description of Opening Reinforcement

The thickened boss has been heavily reinforced in addition to the dense reinforcing which already exists in the 4'-6 thick Containment cylinder wall. The hoop, vertical and seismic wall reinforcing are bent around the openings to provide continuity of reinforcing and assure flow of membrane forces around the openings. All splices will be by the Cadweld process only. The splices are designed to assure that they will develop at least 125% of the minimum yield point stress of the rebar. Several secondary bars have been terminated by means of mechanical anchorage. At the continuous bar bends, hooked bars are provided to prohibit any local crushing of the concrete. In addition the radius of the bar bends is such that crushing of the concrete will not occur. Due to bending the main bars around the large openings,a void in reinforcing is created on the horizontal and vertical center lines. To prevent any cracking and spalling of concrete and to resist membrane tensions,these voids are filled with added rebar which are terminated by hooks at each end. See Sections 2, 3, 14, and 15 in the Acetate Overlays (Appendix A) and Fig. 9321-L-1560.

To accommodate stress concentrations and discontinuity effects of the opening hoop reinforcing is provided around the opening.

In addition to the membrane forces a moment on the ring is produced by the shear load from the pressure on the door of the hatch tending to cause the ring to rotate inside out. Since the ring is restrained from warping, bending moments occur in the cross section of the ring which are resisted by the additional hoops in the reinforced boss. These hoops are designed to resist the tensile loads in addition to bending mentioned above. Since the ring tends to rotate inside out and detach itself from the Containment shell about its outer boundary, a tensile load is induced on the inside surface of the ring and containment. This is resisted by the main vertical and horizontal reinforcing in the Containment cylinder continuous wall.

Since there is an eccentricity between the center of the wall and the center of the thickened ring, moments causing tension on the inside face of the ring develop. These moments are resisted in tension by the main vertical and horizontal bars which are continuous around the opening. In addition these bars assist in resisting membrane tensile loads.

In addition to the main vertical and horizontal reinforcing in the Containment cylinder wall, the two-way seismic reinforcing in the wall is continuous around the opening, thus increasing the steel area available to carry discontinuity forces and moments.

3.4.2 Description of Opening Reinforcement (Continued)

The reinforcing pattern in the hatch area can be seen in Figs. 9321-L-1559 and 9321-L-1560 and (Appendix A) which is an acetate overlay showing each layer of steel around the opening.

Transverse shears radial to the center of the containment and in plane shears are resisted by #8 stirrups placed radially to the opening at 6" centers around the opening (See Section I, Appendix A). Popout shears along the circumference of the opening caused by edge reactions from the pressure against the barrel head are resisted by 2-#9 bars @12" around the opening placed through the cross section perpendicular to the reference plane (See drawing No. 9321-L-1559). These bars are spaced at d/3 to insure that at least one bar will cross a potential diagonal crack through the cross section. One end of the bar will be hooked in order to develop adequate anchorage from the point of crack formation to the end of the bars. In addition to the above mentioned stirrups; concrete, extra stirrups at the voids created by the main horizontal and vertical rebar bending around the opening and inclined horizontal and vertical rebar are also available to resist shear loads. See Figs 9321-L-1559 and 9321-L-1560.

All of the loadings for the geometrical configuration with which we are concerned have been considered in the finite element analysis.

Revised: 4/2/70

Containment Design Report

3.4.3 Finite Element Model Analysis

3.4.3.1 Loads

Finite element analyses using the Displacement Method were performed in the area of the Equipment Hatch by the Franklin Institute Research Laboratory (FIRL). This hatch is 16'-0 in diameter and constitutes the largest penetration in the Containment structure. Pressure, temperature, dead and seismic loads were investigated separately and combined by the following factored load equations:

1. $C = (1.0 \pm .05)\ D + 1.0P + 1.0\ (T'' + TL'') + 1.0E'$

2. $C = (1.0 \pm .05)\ D + 1.25P + 1.0\ (T' + TL') + 1.25E$

3. $C = (1.0 \pm .05)\ D + 1.5P + 1.0\ (T + TL)$

*4. $C = (1.0 \pm .05)\ D + 1.5P$

*5. $C = (1.0 \pm .05)\ D + 1.25P + 1.25E$

*6. $C = (1.0 \pm .05)\ D + 1.0P + 1.0E'$

where terms are as defined on Page 2.0-6 of the Containment Design Report. Those loads most subject to variation and which m..t directly control the overall safety of the structure were subjected to the severest load factors. Each of the following loads was considered separately and combined as shown in the above equations.

1. Dead Load.

*2. 47 psi pressure load.

3. Seismic motion normal to the penetration causing maximum uplift or compression in the thickened boss. The magnitude of the ground acceleration was .10g.

4. Seismic motion parallel to the penetration causing maximum shear in the thickened boss. The magnitude of the ground acceleration was .10g.

5. Seismic motion in the vertical direction. The magnitude of ground acceleration was .05g.

6. Thermal Load associated with 1.5 pressure.

7. Thermal Load associated with 1.25P pressure.

8. Thermal Load associated with 1.0P pressure where P = 47 psi.

(Table 3.4.4.1 shows how these loads were combined in the computer solutio

*The total force is resisted by the R/C shell only (excluding the lir The force is determined by an integration of stresses throughout the entire cross section (including the liner).

*A conservative approach inherent to thin shell computer solutions is used whereby the pressure load is developed at the reference surface rather than the inside face of the wall. This accounts for about a 3% increase in pressure loads.

Containment Design Report

3.4.3.1 Loads (Continued)

The analysis for the pressure loading was performed by considering only the effect of the 47 psi pressure load. For other pressure loading conditions the results from this load were multiplied by the appropriate load factors.

The earthquake load was handled in a similar manner. The analysis was done for the .10g earthquake accounting for structural response and scaled for the 1.25 load factor and .15g ground acceleration. The vertical earthquake was handled in a similar manner. Earthquakes normal to the opening causing maximum moment, and parallel to the opening causing maximum shear were considered. Two seaprate runs were necessary since the boundary conditions on the model had to b varied to account for symmetry of the former and antisymmetry of the latter earthquake, with respect to the vertical axis of the opening.

Three thermal loads were considered. An equivalent linear thermal gradient was determined by FIRL's QUIKTEMP program which produced strains equal to those produced by the actual non-linear gradient. This gave deflections and forces equal to those produced by the actual gradient The mean temperature rise above the reference temperature and gradient obtained from this program were used as input to the Finite Element Progra A correction factor was added to the stresses to account for the variation between the linear gradient and the actual non-linear gradient. See Figs. 9321-L-1565 and 9321-L-1566.

The effect of insulation in the lower two (2) courses was considere in the Coarse Analysis. The panels in these two (2) courses were subjected to a linear gradient of approximately ½ of the normal operating gradient. The insulation was considered effective in preventing large temperature increases during reactor incident from reaching the liner surface.

3.4.2 Coarse Analysis

The finite element computer program (FELAP) utilizes quadrilateral elements. The analysis was performed in two stages, each using the FELAP program. First, the containment vessel as a whole with a square opening of 16'-0 x 16'-0 as a substitution for the actual hatch was analyzed. The substitution was mainly to facilitate the layout of grid points around the opening. The vessel loads are symmetrical for all loads except the earthquake parallel to the hatch and only half of the vessel is taken into account in the analysis. The model extended from the built-in base to the transition from the cylinder to the spherical dome (springline). The vertical edges of this 180^{o} segment of the containment structure were given symmetry boundary conditions. At the springline, forces obtained from FIRL's General Shell Program (GENSHL-2) were applied for the dead and Earthquake Loads.

In GENSHL-2 the structure was modeled from the fixed base to the top of the dome (a patch plate was used to reach the actual apex of the dome). These elements of revolution assume the structure is symmetric. The results were only used to obtain boundary conditions for the coarse model at the springline. Since the springline (89'-6 above the hatch and 107'-6 above thelock) is far removed from both the equipment

Containment Design Report

3.4.3.2 Coarse Analysis (Continued)

hatch and the personnel lock openings, it is in a membrane region and the results of GENSHL-2 are acceptable. See Fig. 9321-L-1563. For th pressure load boundary conditons hand calculated loads, based on membr theory, were used. This was necessary since the neutral surface of th section did not correspond to the reference surface used for the analy Since all strains are developed at the reference surface, small bendin moments were created to which the deformations of the coarse model wer very sensitive since the restraining effect of the dome was not includ in the model. Boundary conditions for Thermal Loads were developed by consideration of a Finite Element Strip in FIRL (FELAP) program. Although the hoop load in the dome is one-half that in the cylinder, t area of steel provided in the dome is one-half that in cylinder; there the stiffnesses are approximately equal and no appreciable discontinui moment is present. For this reason rotation about the springline was restrained.

A layout of the finite elements used for the coarse model is show in Fig. 9321-L-1558.

Each quadrilateral element was divided into 10 layers to represen the liner, concrete, and reinforcing. Layers representing reinforcing were given a thickness equal to the area of steel per inch. All plate representing rebar were given stiffness in only one direction (i.e. th direction of the rebar). For the plate representing the two layers of seismic steel, stiffness was given in two directions but Poisson's Ratio was set equal to zero to uncouple the two directions. The layer through the cross section are given in Table 3.4.3.2.1.

The coarse model was run twice. In the first run the concrete between the liner and the first layer of reinforcing (in the 4'-6 cyli wall) was considered uncracked. The remaining concrete was considered cracked. The results of this analysis showed nearly all the concrete stresses to be in excess of $3.5\sqrt{f'c}$ in tension which indicated that al the concrete in the cylinder was cracked. The problem was rerun considering all the concrete in the cylinder cracked. In the thicken portion of the wall around the equipment hatch the concrete was consid cracked a distance (d-kd). The principal purpose of the Coarse Analys was to determine the extent of the effect of the opening and provide boundary conditions for the fine model. Conservative boundaries were chosen to assure that the fine model extended to a region not affected by the opening; therefore, it was not considered necessary to iterate the cracking with respect to the thickened boss.

The liner was considered as a strength element for two (2) reason

1) As long as the stiffness is added uniformly throughout the region of interest, this should not appreciably affect the distribution of membrane forces.

2) A better representation of thermal effects will be achieved if the liner is considered.

3.4.3.2 Coarse Analysis (Continued)

By matching the boundary forces of the coarse and fine model it was determined that the boundary was essentially in a membrane region. This indicated that the opening had no effect at the boundaries of the fine model and the distance assumed for the extent of the fine analysis boundary was valid.

3.4.3.3 Fine Analysis

The displacements and rotations of the node points obtained from the Coarse Analysis were used as boundary conditions on the boundaries of the region used for the Fine Analysis. The quadrilateral elements used for this analysis were much smaller than those used previously and can be seen in Fig. 9321-LL-1564. The displacements and rotations of intermediate node points were obtained by interpolating the results of the Coarse Analysis.

A 180^{o} segment of the hatch and surrounding wall extending to the boundary determined by the Coarse Analysis was modeled about a vertical center line of the hatch opening. Symmetric and antisymmetric boundary conditions depending on the loading were used for the right-hand edge and prescribed displacements and rotations for the bottom edge, left-hand side and top edge.

Each quadrilateral element was divided into 10 layers to represent the liner, concrete and reinforcing. The liner was considered as a strength element for the model and also in QUIKTEMP to develop the equivalent linear gradient. Layers representing reinforcing were given a thickness equal to the area of steel per inch. All plate elements representing rebar were given stiffness in only one direction (i.e. the direction of the rebar). The layers through the cross section are given in Table 3.4.3.3.1. Material properties are given in Table 3.4.3.3.2.

FIRL modified their program so that the effect of the liner was not included in the stress resultants at the reference plane of the panel. When integrating the stresses through the wall cross section to obtain stress resultants, the first layer, which is the liner layer, was not included. The main reason for doing this is that thermal load stress resultants would greatly underestimate the force in the rebar since the net effect includes the compression in the liner. By virtue of this program revision all stress resultants will include only the effect on the cylindrical wall. When temperature effects were excluded; however, the total stress resultants (including the liner) were applied to the R/C shell (not including the liner).

There were two coordinate systems used in the analysis. Equilibrium coordinates are associated with node points and are used for concentrated loads and boundary conditions including restraints and specified displacements and rotations. Displacements and stress resultants are also outputted in equilibrium coordinates. The directions for this coordinate system are radial and tangential to the center of the cylindrical containment

3.4.3.3 Fine Analysis (continued)

The other system used is panel coordinates which are in the plane or perpendicular to the plane of the rectangular elements. All inputted pressure and other surface tractions (loads) are in panel coordinates.

Since two coordinate systems were used, a transformation is carried out internally when solving the problem. The stiffness matrix in panel coordinates was transformed into equilibrium coordinates and displacements were obtained in equilibrium coordinates.

The non-symmetry effects caused by the difference between the centroid of the 4'-6" wall and the 7'-6" boss were taken into account by the computer program since all coordinate points for the layers are given about the same cylindrical surface which coincides with the center line of the 4'-6" wall.

Shear loading caused by the pressure acting on the head of the hatch was inputted in equilibrium coordinates at the same time as the displacement and rotational boundary conditions.

The Equipment Hatch was not considered as a strength element and was not included when computing the individual plate element thicknesses. The restraint offered by the hatch at the edge of the opening was neglected in the analysis.

The concrete in the cylinder was considered completely cracked based on the results of the Coarse Analysis. The concrete in the thickened boss was initially considered cracked a distance (d-kd). The results of this analysis indicated that the concrete was completely cracked in the boss as well as the cylinder. To insure that all cases were included, it was decided to do the analysis with the boss concrete completely cracked in one run and completely uncracked in the other. In addition, each case included output where the stress resultants were obtained by integrating stresses through the cross section including the liner stress and not including the liner stress. For each integration, the stresses were identical; however, the stress resultants were different. The case with the boss concrete completely uncracked indicated concrete tensile stresses greater than $3.5\sqrt{f'c}$ throughout; therefore, the results of the completely cracked concrete run were used. Stress resultants from the completely cracked analysis including the liner were used for all load combinations not including thermal effects.

3.4.3.4 Program

The following is a brief outline of the procedures used by FIRL to perform the finite element analysis:

1. Using 10-layered elements the stiffness matrix was obtained as follows:

   a) A unit displacement was assumed

   b) Strains were obtained

   c) Using the material constants, stress-strain relationships were developed

   d) Using the theorem of minimum potential energy and the principle of virtual work the stiffness matrix for each element, at the reference surface, was developed.

2. Nodal rotations and displacements were obtained using the above stiffnesses and solving the simultaneous equations.

3. The nodal displacements and rotations were multiplied by the stiffness matrix to arrive at nodal forces in the equilibrium coordinate system.

4. From the nodal displacements and rotations strains were obtained and multiplied by the Material Property Matrix to get stresses at each layer.

5. The stresses at each layer are integrated through the section to get stress resultants (#/in. or in.#/in.) on the reference surface at the centroid of each panel. The following integrations are performed:

   $N_{11} = \int S_{11} dx3$ $\qquad M_{11} = \int S_{11}\, x3\, dx3$

   $N_{22} = \int S_{22} dx3$ $\qquad M_{22} = \int S_{22}\, x3\, dx3$

   $N_{12} = \int S_{12} dx3$ $\qquad M_{12} = \int S_{12}\, x3\, dx3$

   $Q_1 = \int S_{31} dx3$ $\qquad Q_2 = \int S_{23}\, x3\, dx3$

   Where

   N_{11} = vertical force (#/in.)

   S_{11} = vertical stress (#/in.^{2})

   M_{11} = moment about the hoop axis (in. - #/in.)

   N_{22} = horizontal force (#/in.)

   S_{22} = hoop stress (#/in.^{2})

   M_{22} = moment about the vertical axis (in.-#/in.)

3.4.3.4 Program (Continued)

N_{12} = in plane shear (#/in.)

S_{12} = in plane shear stress (#/in.^{2})

M_{12} = twisting moment (in.-#/in.)

Q_1 = radial shear force on 1 edge (#/in.)

S_{31} = radial shear stress on 1 edge (#/in.^{2})

Q_2 = radial shear force on 2 edge (#/in.)

S_{23} = radial shear stress on 2 edge (#/in.^{2})

The linear thermal gradient obtained from QUIKTEMP is used as input into the Finite Element Analysis. The element is considered fully restrained and the corresponding forces and stresses are obtained. The forces are imposed on the structure, in the opposition direction, with the real boundary conditions present and the corresponding displacements and stresses calculated. These stresses are then subtracted from the fully restrained stresses and the actual stresses caused by the linear thermal gradient are obtained.

Shear is a basic part of the program and the shear resultants are obtained by integrating through the section assuming a constant shear strain throughout. Shear deformations are accounted for in the program. Since deformations are used for calculating stresses, the shear stiffness of the section will influence the magnitude of the stress rsultants and deflections. It is felt a reasonable representation of the shear stiffness was obtained by using a single shear modulus for all concrete whether cracked or uncracked. All rebar layers except the seismic steel were given a shear modulus of zero since no dowel action was considered in resisting shear. FIRL revised their program to incorporate orthoganal shear moduli. This enabled the seismic steel to be given in plane shear strength while the transverse strength was eliminated. This change, along with the ability to represent the steel at 45°, enabled proper representation of the seismic rebar to be effected.

3.4.4 Results Of The Finite Element Analysis

Output from the Finite Element Analysis included the following:

1. Nodal displacements and rotations in the equilibrium coordinate system and nodal forces and moments in the equilibrium coordinate system.

2. Stress resultants at the centroid of the element consisting of vertical and horizontal axial forces, bending moments about the horizontal and vertical axis and in-plane and radial shear force: All results are given in "per inch" units to reflect continuity of the structure and are given in panel coordinates. Supplement

3.4.4 Results Of The Finite Element Analysis (Continued)

3. The in plane and principal stress components at the inside face, middle, and outside face of each layer of each element (See Figs. 9321-L-1568 and 9321-L-1569 for FELAP program sign conventions.)

The effect of temperature on the stresses has still been overestimated since the computer program considers only elastic behavior. No credit was taken for the redistribution of stress that would occur when the outer most rebar begins to yield.

On Sheet 49 a comparison between the computer solution and the pure membrane condition was checked considering the horizontal nodal forces on the two vertical boundaries. The results show a total of $33,804.2^k$ on the left boundary and $32,511.7^k$ on the right boundary. In addition, the classical formulas for membrane theory indicate a total load of $32,900^k$ along the left boundary which is in agreement with the computer results. A stress concentration factor of 2.88 was obtained by dividing the hoop stress resultant tangent to the opening at panel 113 by the value obtained remote from the opening (See Sheet 50.). This corresponds to a value of 3.0 which is an accepted value for the type of stress concentration which occurs when a horizontal load twice the magnitude of a vertical load acts on the structure and the two directions are not coupled by a Poisson's Ratio effect. This is the effect in the structure since the vertical and horizontal rebar were each given stiffness in only one direction.

These calculations along with the hand calculations mentioned later were used for an order of magnitude verification of the computer results.

The model of the hatch area is a good representation of the rebar present. As can be seen in Fig. 9321-L-1560, the rebar bending around the opening and the added hoop rebar are both very nearly tangent to the circular opening. The rebar in each element around the opening was given stiffness in only one direction, the direction tangent to the opening; and was distributed to three plates, one on the outside (Plate X), one on the inside (Plate Y), and one in the middle (Plate Z) of the cross section representing the structure (See Fig. 9321-L-1562). The seismic rebar in a direction tangent to the opening, was included at approximately the reference plane (27.375 inches from the inside face of the liner plate). The liner was also included in the model. Each 90° segment around the hatch was subdivided into (3) sectors (A, B, and C) and plate given properties such that B sector represents the hoop steel bending around the opening, A represents the vertical steel bending around the opening, and C or the middle sector represents a combination of hoop and vertical steel bending around the opening. (See Fig. 9321-L-1561) In addition, the added hoops were included in the plate which was physically closest to their location in the structure.

Since an accurate representation of the rebar was used in the model and the tensile stresses, are comparable to the hand calculations, all of the stresses for the individual rebar layers outputted by the FIRL Program can be used directly. A summary of stresses

3.4.4 Results Of The Finite Element Analysis (Continued)

for the fine model can be found on sheets 1 through 48. Representative elements in the upper quadrant of the thickened boss area (between 0^{o} and 90^{o}), the lower quadrant of the fine model in the cylinder region and the upper quadrant of the fine model in the cylinder region are summarized. Stresses in the middle of the rebar layer for the outside face and inside face for each of these areas are summarized for the vertical seismic load, dead load, internal pressure load, symmetric horizontal seismic load, thermal load 1, thermal load 2, thermal load 3 and the three load combinations including the vertical and horizontal earthquakes causing tension. The seismic rebar stresses fall between the outer and inner rebar stresses and therefore are not summarized in these tables. In addition, the liner stresses are summarized for the three load combinations including the vertical and horizontal earthquakes causing compression. A correction factor must be added to all liner stresses to take into account the difference between the linear gradient obtained from QUIKTEMP and the large temperature peak which the liner feels during a loss of coolant accident. This is applied to a fixed liner; therefore, compressive stresses equal to $\frac{E\alpha\Delta T}{1-\mu}$ are obtained. These calculations appear on Sheets 75 and 76.

An inspection of the stresses from computer printout shows that for the 1.0P and 1.25P load combinations including thermal load, no yielding of the rebar occurs. The highest stress in the outer rebar for the 1.0P case is 50.9 which is less than $\emptyset$ f. For the 1.25P case the highest outer rebar stress is equal to fy. Rebar stresses in the boss are considerably lower than those in the upper and lower quadrants of the cylinder.

For the 1.5P case including thermal load some local yielding of the outer rebar occurs. The maximum rebar stress occurs in several panel in the transition zone from the boss to the 4'-6 cylinder and is equal to 70^{ksi}. The boss does not indicate any outer rebar yielding. The highest stress for an element in the 4'-6 cylinder is 66.02 with the majorit of the elements showing stress in the $50\text{-}60^{ksi}$ range.

An interaction diagram was plotted for the horizontal and vertical directions in the 4'-6 cylinder considering the ultimate strength of the reinforced concrete section to find the ultimate moment capacity when no axial load is present and the ultimate axial load required to stress all the rebar to 60^{ksi} when no moment is present. (See Sheets 59 through 66). The liner was not included in calculating either the axial or bending load carrying capacity of the section. The axial load and moment stress resultants for load combination 12, for which the liner was not included when determining the stress resultants in order to preclude the net effect when integrating thermal loads across the section were plotted for each panel indicating yield of the outer rebar. The results on Sheets 60 and 64 show that the section provided is adequate to resist the load combinations leading to the (elastic) stress excess.

3.4.4 Results Of The Finite Element Analysis (Continued)

Investigation of the liner stresses show that for the 1.25P and 1.5P cases some local yielding of the liner in tension occurs in several elements in the 360° ring of the panels adjacent to the transition zone from the 4'-6 cylinder to the thickened boss. These occur in the boss at 45 degree to the vertical centerline of the equipment hatch opening. The liner stresses are local not only in location in the boss but adjacent elements show stresses considerably lower in magnitude indicating the presence of a stress riser in the transition from boss to cylinder. Local yielding of this nature (.17% strain) is well within the limits stated in section 2.0-9 of the Containment Design Report (.5% strain). In addition, ductility of the studs will prevent stud failure (See Page. 1). These liner stresses are considered higher than those the actual structure will be required to resist since the computer results indicate large moments at these sections. If the studs yield, the entire moment will not be transmitted to the liner, thus reducing liner stresses. The adequacy of the structure to resist loads without the liner is discussed on Page 14. No yielding of the liner in compression is indicated by the computer results.

In addition to the stresses summarized above, stresses were checked to insure that in any factored load combination from which the thermal effects have been deleted the total stress resultants were resisted by the R/C without assistance from the liner. This was done since the liner can experence substantial tensile stresses. In the event of local yielding of the liner the R/C would have to resist the loads. Prudence demands the provision of an adequate margin of safety in the rebars to resist test pressure loads without assistance from the liner. In order to check this, the stress resultants derived by integrating stresses through the entire cross section (including the liner) were imposed on the cross section (not including the liner) by hand calculations shown on Sheets 51 through 58. From the summary of stress resultants for the various load combinations it can be seen that the 1.5P + .95 DL case gives the largest stress resultants; therefore, only stresses for this condition have been investigated. A panel with maximum stress resultants in the membrane area was checked and the maximum stress in the vertical rebar was 31.4^{ksi} and the maximum horizontal rebar stress was 44.78^{ksi}. Panel 56, which is located in the boss adjacent to the transition from the boss to the 4'-6 cylinder was found to yield the largest stresses. The highest rebar stress was 35.4^{ksi}. These stresses show that the rebar meet the criteria of $\emptyset f_y$ specified in section 3.4.1. It should be noted that in computing the center of gravity of the section the liner has been included. The reason for this computation was to transfer the axis of the stress resultants from the reference surface (center of 4'-6 wall) to the centroid. When computing rebars stresses the liner was not included in the section. This procedure was used for the following reasons:

It is not feasible to determine the local stress resultants acting on the R/C or the distribution of the resultants, in the vicinity of local liner yielding. Obviously an analysis (by finite element) of the entire structure with liner ommitted is no more likely to give the correct local stress resultants than a corresponding analysis with the liner included. In some regions the liner would

R-14

3.4.4 Results Of The Finite Element Analysis (Continued)

contribute effectively to wall stiffness, in others not. Accordingly it appears at least as rational to determine stress resultants from a model in which the liner stiffness is everywhere included as from a model in which liner stiffness is everywhere excluded. In either case the stress resultants outputted at an arbitrary reference surface must, of course, be transformed to a centroid of the wall.

Since the criteria demands rebar stresses in the elastic range, for this load combination, and since some re-distribution in the vicinity of the yielded liner is inevitable, it is reasonable to apply to the rebar centroid the same set of stress resultants as were determined for the centroid of rebars and liner. This means that we are balancing the uncertainty in the distribution of stress resultants throughout the zone by a conservative rebar stress calculation, and by our conviction that there is a "limit" strength for the region which we have not determined, but which provides further margin of safety against actual rupture.

On Sheet 73 there appears a summary of the out of plane nodal forces around the perimeter of the Equipment Hatch. On Sheet 74 the pressure acting over the inside door of the Equipment Hatch was calculated and the reaction on the shell determined. In addition the pressure acting over 1/2 the width of the first row of panels from the edge of the opening was determined. This summation agreed with the nodal forces obtained thus adding confidence to the computer results since these nodal forces are a function of the stiffness matrix and the shear displacements. Since the displacements are correct the strains and thus the stresses are reasonable and the stress resultants obtained from these stresses are acceptable.

Radial and transverse shear stress resultants were used to check the adequacy of the radial stirrups. Stresses could not be used directly since the model did not include these stirrups. Since the concrete provided shear stiffness to the model thus creating higher shear forces, it could have been used to help resist the shear forces. For conservatism; however, the rebar was considered to resist the entire load.

The earthquake 90 degrees from the Equipment Hatch opening which causes maximum in plane shear stresses was investigated and included in the load combinations on Sheets 69 through 71. On Sheet 72 the total shear through the 90" wide boss was determined by considering the weighted average of the shear in the elements through the boss and the corresponding shear stress calculated. In the upper quadrant of the boss at about the 45^{o} point, the seismic rebar is available to resist shear. The maximum stress in the stirrups and rebar was found to be 40.8^{ksi}. In the region where only the stirrups resisted shear the stress was found to be 45.8^{ksi}. In all the above shear stress calculations the concrete, extra stirrups at the voids created by the main horizontal and vertical rebar bending around the opening, and inclined horizontal and vertical rebar have been conservatively ignored.

3.4.4 Results Of The Finite Element Analysis (Continued)

On Sheet 67 the maximum shear load in the elements around the hatch opening was found for each load combination by considering the square root of the sum of the squares for the two transverse shears. The maximum load was found to occur in Panel 107 for the 1.5P + .95 DL case. This is a small panel at the edge of the Equipment Hatch opening. This stress resultant (in kip/inch) was conservatively assumed to occur throughout the entire 90" boss even though very small transverse shear values were found in the 4'-6" cylinder wall adjacent to the thickened boss. The extra stirrups at the voids were also neglected. The maximum stress calculated on Sheet 68 was found to be 52 ksi.

In addition on Page 74 the maximum shear in the boss, obtained by taking the square root of the sum of the squares of the two transverse shears, is used to determine the quantity of radial bars required to resist pop out shear discussed on Page 4 and shown in Fig. 9321-L-1559.

In plane deformations were checked in order to see the ovaling which takes place at the juncture of the hatch and the reinforced concrete shell. Points 90° apart were investigated to find the variation in displacement along the axis of the hatch. Plots of this ovaling for the pressure load and computer load combination 11 are found on drawing No. 9321-L-1570 which shows a maximum horizontal in plane deformation of .0327 in. and a difference in vertical deflections between the top and bottom of the opening equal to .0587 in. for load combination 11. These are both very small displacements and will not present any difficulties at the juncture of the hatch and the reinforced concrete shell. This is conservative since the resistance offered by the 1" thick barrel and 7/8" studs were not considered in this analysis.

The FIKL plotting routine was utilized to aid in checking model geometry and interpreting results. By plotting the geometry any mistake in coordinates of nodal points or numbering of the points can be readily discovered by inspection of the geometry plot. Any discontinuity or crossing of lines indicates a mistake in input data. A plot of stresses greatly simplifies the interpretation of the output and provides a means for a quick check of the validity of the solution. It shows the variation in stress from one point to another, an important effect which is usually lost during inspection of volumes of computer printout. These plots can be found on Sheets 94 through 105.

They include:

a) Radial displacements on a developed section for pressure load

b) Inside and outside rebar stresses for the computer load combination 12.

c) Liner stresses for the computer load combination 12.

d) Shear stress resultants including transverse shear radial to the containment centerline and pop out shear along the circumference of the opening.

We know from analysis and observation of test results from similar containment structures that the combination of Dead Load and Pressure

3.4.4 Results Of The Finite Element Analysis (Continued)

experienced during the pressure test, will crack the structure and relieve shrinkage or creep stresses locked into the structure prior to pressurization. Since the overall structure will be subjected to this cracking, distribution of forces should not be substantially altered from that of the computer output.

3.4.5 Design Basis

3.4.5.1 Equipment Hatch

Before undertaking a comprehensive series of Finite Element Analyses already described, the adequacy of the rebar (pattern and quantity) in the vicinity of the Equipment Hatch was determined by hand calculations. These calculations, shown on Sheet 77 to 85, involved the determination of rebar stresses at points of the horizontal and vertical axis of the opening. The following loadings were considered:

1. The twisting moment on the ring.
2. The eccentricity of horizontal and vertical main wall reinforcing with the centerline of the boss.
3. The thermal effects in the liner.
4. Maximum tensile forces using a stress concentration of 3.
5. Effect of the moment on the inside face of boss from the rotation caused by pressure, ignoring restraint from the hatch.
6. Effect of this rotation on the main wall reinforcing.

The conservative characteristics of these stress calculations is indicated by a comparison of the stress level obtained at the corresponding locations determined by the Finite Element Analysis. The Finite Element Analysis indicated rebar stress 4 to 7 ksi smaller than the hand calculation

3,4.5.2 Personnel Lock

The adequacy of the rebar in the vicinity of the Personnel Lock is verified by the analysis similar to that described in 3.4.5.1. Calculations are shown on Sheets 86 to 93. The inner and outer rebar stresses for the Equipment Hatch and Personnel Lock are compared with the computer results for the Equipment Hatch Analysis. The results can be found in Table 3.4.5.2.1. The loading condition for the hand calculations consisted of 1.5 Pressure and Temperature Load 3. This was compared with Load Combination 12 of the computer analysis. The hand calculations are conservative since dead load was not considered.

It was judged unnecessary to further verify this area by Finite Element Analysis for the following reasons:

R-17

1. The Equipment Hatch area indicated that the Finite Element Analysis would lead to lower rebar stresses.

3.4.5.2 Personnel Lock (Continued)

2. The hand calculations for the Equipment Hatch and Personnel Lock indicated lower stresses in the Personnel Lock than in the Equipment Hatch.

This was to be expected because:

a) The diameter of the Personnel Lock opening (8'-6") is only about 1/2 the diameter of the Equipment Hatch opening.

b) The Personnel Lock Boss has been provided with a relatively higher quantity of rebar than the Equipment Hatch Boss (7.3% vs. 5.8%).

c) The thickened portion of the Personnel Lock boss is only 5'-6" thick vs. 7'-6" in the Equipment Hatch boss.

For a description of the rebar in the Personnel Lock boss see drawing 9321-L-1571 and 9321-L-1572. Except for bars which resist pop out shears identical patterns of rebar have been placed in this boss to resist the same forces described for the Equipment Hatch boss.

Pop out shears along the circumference of the opening, caused by edge reactions from the pressure against the closed inside lock door are resisted by two (2) #7 bars @12" placed in a manner identical to the two (2) #9 bars @12" in the Equipment Hatch and described on Page 4.

There is a slight possibility that the inside door of the lock might be open at the time of an accident. The effect of pressure and temperature within the lock barrel, in such an event, was considered in the analysis of the Personnel Lock area which shows rebar stresses below yield values. It should be noted that the analysis for the effects, on the R/C, of accident pressure and temperature within the barrel were conservative in that the barrel was assumed to develop a compressive stress in excess of 32 ksi yield and the thermal gradient was taken as the difference between the accident temperature and 0°.

3.4.6 Summary

Using the FIRL Finite Element Programs the cylinder and thickened portion of the wall around the Equipment Hatch were analyzed by the following procedure:

1. General Shell Analysis Program was used to obtain boundary conditions at the springline for Dead Load and Earthquake Loads

2. A strip of the containment was analyzed by the Finite Element Program (FELAP) to obtain boundary conditions at the springline for the Thermal Loads

Rev.: 9/8/69
Rev.: 4/2/70

3.4.6 Summary (Continued)

3. QUIKTEMP converted non-linear thermal gradients to equivalent linear gradients.

4. Pressure Load boundary conditions were developed by hand calculations

5. A coarse representation of 180^0 of the containment was analyzed by the FELAP Program to obtain boundary conditions for the fine model and determine the extent of the influence of the Equipment Hatch opening.

6. The fine grid model shown in Fig. 9321-LL-1564 was analyzed using the FELAP Program.

7. Various cracking patterns were investigated and it was determined that all concrete in the cylinder and boss was completely cracked.

8. Stresses and stress resultants obtained from the fine analysis were summarized and all the criteria on Page 2 were satisfied.

9. Hand calculations were performed to check the order of magnitude of the computer results.

10. Hand calculations were performed to check the stress in the Personnel Lock area.

Rev.: 9/8/69
Rev.: 4/2/70

TABLE 3.4.3.2.1

IDEALIZATION OF CYLINDER WALL SECTION FOR BOTH COARSE AND FINE MODEL

PANEL TYPE	LAYER #	THICK. (IN.)	MATERIAL	MAT'L CODE	THICK. CODE
All Cylinder Elements	1	.375	Steel PL	1	1
	2	7.75	Cr. Conc.	5	2
	3	.333	Ax. Rebar	3	3
	4	.572	Hoop Rebar	4	4
	5	19.085	Cr. Conc.	5	5
	6	.270	Seismic Rebar	6	6
	7	19.085	Cr. Conc.	5	5
	8	.572	Hoop Rebar	4	4
	9	.333	Ax. Rebar	3	3
	10	6.0	Cr. Conc.	5	7

TABLE 3.4.3.3.1

IDEALIZATION OF THICKENED WALL SECTION AROUND THE EQUIPMENT HATCH

*A 1

PANEL TYPE	LAYER	THICK.(IN.)	MATERIAL	MAT'L CODE	THICK. CO
	1	.750	Steel PL	1	8
58, 57, 56	2	7.750	Cr. Conc.	5	2
	3	2.797	Rebar	7, 8, 9, 10, 11, 12	9
111, 110, 109	4	12.745	Cr. Conc.	5	26
	5	.667	Seismic Rebar	7, 8, 9, 10, 11, 12	10
	6	20.982	Cr. Conc.	5	27
	7	3.867	Rebar	7, 8, 9, 10, 11, 12	11
	8	33.062	Cr. Conc.	5	28
	9	2.130	Rebar	7, 8, 9, 10, 11, 12	12
	10	6.000	Cr. Conc.	5	7

*A 2

PANEL TYPE	LAYER	THICK.(IN.)	MATERIAL	MAT'L CODE	THICK. CO
	1	.750	Steel PL	1	8
48, 47, 44	2	7.750	Cr. Conc.	5	2
	3	.667	Rebar	7, 8, 9, 10, 11, 12	10
101, 100, 97	4	14.875	Cr. Conc.	5	29
	5	.667	Seismic Rebar	[illegible] 8, 9, 10, 11, 12	10
	6	22.583	Cr. Conc.	5	30
	7	.667	Rebar	7, 8, 9, 10, 11, 12	10
	8	36.524	Cr. Conc.	5	31
	9	.267	Rebar	7, 8, 9, 10, 11, 12	13
	10	6.000	Cr. Conc.	5	7

*A 3

PANEL TYPE	LAYER	THICK.(IN.)	MATERIAL	MAT'L CODE	THICK. CO
	1	.750	Steel PL	1	8
13, 46, 43	2	7.750	Cr. Conc.	5	2
	3	.667	Rebar	7, 8, 9, 10, 11, 12	10
66, 99, 96	4	14.875	Cr. Conc.	5	29
	5	.667	Seismic Rebar	7, 8, 9, 10, 11, 12	10
	6	22.583	Cr. Conc.	5	30
	7	.667	Rebar	7, 8, 9, 10 11, 12	10
	8	36.391	Cr. Conc.	5	32
	9	.400	Rebar	7, 8, 9, 10, 11, 12	14
	10	6.000	Cr. Conc.	5	7

*See Drawing 9321-L-1561, 9321-L-1562 for location.

TABLE 3.4.3.3.1 (CONT'D)

*A 4

PANEL NO.	LAYER	THICK. (IN.)	MATERIAL	MAT'L. CODE	THICK. CODE
	1	.750	Steel PL	1	8
29, 45, 42	2	7.750	Cr. Conc.	5	2
	3	1.190	Rebar	7, 8, 9, 10, 11, 12	15
82, 98, 95	4	14.351	Cr. Conc.	5	33
	5	.667	Seismic Rebar	7, 8, 9, 10, 11, 12	10
	6	21.783	Cr. Conc.	5	34
	7	2.267	Rebar	7, 8, 9, 10, 11, 12	16
	8	34.792	Cr. Conc.	5	35
	9	1.200	Rebar	7, 8, 9, 10, 11, 12	17
	10	6.000	Cr. Conc.	5	7

*B 1

PANEL NO.	LAYER	THICK. (IN.)	MATERIAL	MAT'L. CODE	THICK. CODE
53, 52, 51	1	.750	Steel PL	1	8
	2	7.750	Cr. Conc.	5	2
106, 105, 104	3	2.930	Rebar	13, 14, 15, 16, 17, 18	19
	4	12.611	Cr. Conc.	5	36
	5	.667	Seismic Rebar	13, 14, 15, 16, 17, 18	10
	6	20.652	Cr. Conc.	5	37
	7	4.530	Rebar	13, 14, 15, 16, 17, 18	18
	8	32.730	Cr. Conc.	5	38
	9	2.130	Rebar	13, 14, 15, 16, 17, 18	12
	10	6.000	Cr. Conc.	5	7

*B 2

PANEL NO.	LAYER	THICK. (IN.)	MATERIAL	MAT'L. CODE	THICK. CODE
	1	.750	Steel PL	1	8
35, 32, 49	2	7.750	Cr. Conc.	5	2
	3	1.333	Rebar	13, 14, 15, 16, 17, 18	20
88, 85, 102	4	14.208	Cr. Conc.	5	39
	5	.667	Seismic Rebar	13, 14, 15, 16, 17, 18	10
	6	22.250	Cr. Conc.	5	40
	7	1.333	Rebar	13, 14, 15, 16, 17, 18	20
	8	36.192	Cr. Conc.	5	41
	9	.267	Rebar	13, 14, 15, 16, 17, 18	13
	10	6.000	Cr. Conc.	5	7

TABLE 3.4.3.3.1 (CONT'D.)

*B 3

PANEL NO.	LAYER	THICK.(IN.)	MATERIAL	MAT'L.CODE	THICK.CODE
	1	.750	Steel PL	1	8
34, 31, 50	2	7.750	Cr. Conc.	5	2
	3	1.333	Rebar	13, 14, 15, 16, 17, 18	20
87, 84, 103	4	14.208	Cr. Conc.	5	39
	5	.667	Seismic Rebar	13, 14, 15, 16, 17, 18	10
	6	22.250	Cr. Conc.	5	40
	7	1.333	Rebar	13, 14, 15, 16, 17, 18	20
	8	36.059	Cr. Conc.	5	42
	9	.400	Rebar	13, 14, 15, 16, 17, 18	14
	10	6.000	Cr. Conc.	5	7

*B 4

PANEL NO.	LAYER	THICK.(IN.)	MATERIAL	MAT'L.CODE	THICK.CODE
	1	.750	Steel PL	1	8
33, 30, 28	2	7.750	Cr. Conc.	5	2
	3	1.863	Rebar	13, 14, 15, 16, 17, 18	21
86, 83, 81	4	13.678	Cr. Conc.	5	43
	5	.667	Seismic Rebar	13, 14, 15, 16, 17, 18	10
	6	21.452	Cr. Conc.	5	44
	7	2.930	Rebar	13, 14, 15, 16, 17, 18	19
	8	34.460	Cr. Conc.	5	45
	9	1.200	Rebar	13, 14, 15, 16, 17, 18	17
	10	6.000	Cr. Conc.	5	7

*C 1

PANEL NO.	LAYER	THICK.(IN.)	MATERIAL	MAT'L.CODE	THICK.CODE
	1	.750	Steel PL	1	8
55, 54	2	7.750	Cr. Conc	5	2
	3	3.600	Rebar	19, 20, 21, 22	23
108, 107	4	11.941	Cr. Conc.	5	46
	5	.667	Seismic Rebar	19, 20, 21, 22	10
	6	20.319	Cr. Conc.	5	47
	7	5.197	Rebar	19, 20, 21, 22	22
	8	32.396	Cr. Conc.	5	48
	9	2.130	Rebar	19, 20, 21, 22	12
	10	6.000	Cr. Conc.	5	7

TABLE 3.4.3.3.1 (CONT'D)

PANEL NO.	LAYER	THICK. (IN.)	MATERIAL	MAT'L CODE	THICK CODE
			*C 2		
	1	.750	Steel PL	1	8
41, 38	2	7.750	Cr. Conc.	5	2
	3	2.000	Rebar	19, 20, 21, 22	24
94, 91	4	13.541	Cr. Conc.	5	49
	5	.667	Seismic Rebar	19, 20, 21, 22	10
	6	21.917	Cr. Conc.	5	50
	7	2.000	Rebar	19, 20, 21, 22	24
	8	35.858	Cr. Conc.	5	51
	9	.267	Rebar	19, 20, 21, 22	13
	10	6.000	Cr. Conc.	5	7
			*C 3		
40, 37	1	.750	Steel PL	1	8
	2	7.750	Cr. Conc.	5	2
	3	2.000	Rebar	19, 20, 21, 22	24
93, 90	4	13.541	Cr. Conc.	5	49
	5	.667	Seismic Rebar	19, 20, 21, 22	10
	6	21.917	Cr. Conc.	5	50
	7	2.000	Rebar	19, 20, 21, 22	24
	8	35.725	Cr. Conc.	5	52
	9	.400	Rebar	19, 20, 21, 22	14
	10	6.000	Cr. Conc.	5	7
			*C 4		
	1	.750	Steel PL	1	8
39, 36	2	7.750	Cr. Conc.	5	2
	3	2.533	Rebar	19, 20, 21, 22	25
92, 89	4	13.008	Cr. Conc.	5	53
	5	.667	Seismic Rebar	19, 20, 21, 22	10
	6	21.117	Cr. Conc.	5	54
	7	3.600	Rebar	19, 20, 21, 22	23
	8	34.125	Cr. Conc.	5	55
	9	1.200	Rebar	19, 20, 21, 22	17
	10	6.000	Cr. Conc.	5	7

TABLE 3.4.3.3.2

MATERIAL CODES

	E_1	E_2	V_1	G_1	G_3	ITEM
1	29×10^6	29×10^6	.3	0	0	Steel PL
2	3.2×10^6	3.2×10^6	.15	1.39×10^6	1.39×10^6	Conc.
3	29×10^6	0	0	0	0	Ax. Reba
4	0	29×10^6	0	0	0	Hoop Reb
5	0	0	0	1.39×10^6	1.39×10^6	Cr Conc.
6	29×10^6	29×10^6	0	11.5×10^6	0	Seismic
7 - 22	0	29×10^6	0	0	0	Boss Reb

E_1 = Modulus of Elasticity in Vertical Direction (#/in.^{2})

E_2 = Modulus of Elasticity in Horizontal Direction (#/in.^{2})

V_1 = Poisson's Ratio

G_1 = In Plane Shear Modulus (#/in.^{2})

G_3 = Transverse Shear Modulus (#/in.^{2})

TABLE 3.4.4.1

✱ COMPUTER LOAD COMBINATIONS

1. Vertical Earthquake Load 1.25 (.05g)

2. Dead Load

3. Pressure Load - 47 psi

4. Horizontal Earthquake Load (.10g - causing tension on boss)

5. Thermal Load 1

6. Thermal Load 2

7. Thermal Load

*8. $(1.0 \pm .05)$ D + 1.0P + 1.0E' + 1.0 (T" + TL")

9. $(1.0 \pm .05)$ D + 1.0P + 1.0E' + 1.0 (T" + TL")

*10. $(1.0 \pm .05)$ D + 1.25P + 1.25E + 1.0 (T' + TL')

11. $(1.0 \pm .05)$ D + 1.25P + 1.25E + 1.0 (T' + TL')

12. $(1.0 \pm .05)$ D + 1.5P + 1.0 (T + TL)

*Earthquake causing compression on boss.

✱The earthquake 90° from the hatch opening causing maximum shear at the opening was not included in this list since it was not included in the stress summary tables which are confined with tensile stresses only. Consideration of this load is given on sheets 69 through 72.

Revised 9/8/69

TABLE 3.4.5.2.1

COMPARISON OF EQUIPMENT HATCH BOSS AND PERSONNEL LOCK BOSS STRESSES

LOAD - 1.5P PRESSURE + TEMPERATURE

EQUIPMENT HATCH-HAND CALCULATIONS			EQUIPMENT HATCH-COMPUTER RESULTS			*PERSONNEL LOCK-HAND CALCULATIONS		
PANEL IN FINE GRID	INNER REBAR STRESS (KSI)	OUTER REBAR STRESS (KSI)	PANEL IN FINE GRID	INNER REBAR STRESS (KSI)	OUTER REBAR STRESS (KSI)	PANEL IN FINE GRID	INNER REBAR STRESS (KSI)	OUTE REBA STRE (KSI
113	46.25	13.75	113	23.56	10.48	113	42.15	23.[illegible]
106	28.3	8.3	106	16.43	6.85	106	21.05	13.[illegible]

*Personnel Lock Stresses do not include the effect of the possible pressure and temperature increases inside the Lock. These calculations are shown on Sheet 87.

VERTICAL SEISMIC LOAD (TENSION)

REBAR STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	INNER H	INNER V	OUTER H	OUTER V
113	104	-.06		-.05	
201	102	-.02		.05	
200	103	-.01		.13	
199	81	-.06		.18	
198	73	-.14	.51	-.04	.15
197	65	-.09	.48	-.06	.29
112	105	.03		0	
185	85	.01		.08	
184	84	.01		.09	
183	83	.02		.16	
182	74	-.17	.46	-.20	.39
181	64	-.10	.47	-.15	.35
110	107	.05		-.02	
159	91	.08		.07	
158	90	.1[illegible]		.04	
157	89	.13		.08	
156	76	-.13	.53	-.33	.64
155	62	-.14	.51	-.23	.62
108	109	.26		-.05	
140	97	.40		-.08	
139	96	.39		.19	
138	95	.44		.07	
137	78	-.04	.70	.24	.63
136	70	-.07	.66	.13	.61
106	111	.52		-.09	
122	101	.44		-.09	
121	66	.44		-.07	
120	82	.54		0	
119	80	-.03	.65	.13	.41
118	72	-.04	.64	.12	.69

Sheet 1

VERTICAL SEISMIC LOAD (TENSION)

REBAR STRESS - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
250	2	.71	-.04	.70	-.01
251	2	.70	-.04	.70	0
252	2	.70	-.04	.69	-.01
37	2	.70	-.04	.69	-.01
38	2	.70	-.04	.70	-.01
39	5	.70	-.03	.73	-.04
40	5	.71	-.07	.75	-.06
256	2	.75	-.03	.73	-.01
257	2	.75	-.03	.74	-.02
258	2	.75	-.03	.75	-.03
59	2	.73	-.03	.80	-.03
60	2	.59	-.04	.65	-.04
61	5	.74	-.05	.76	-.05
62	5	.72	-.06	.75	-.06
63	5	.71	-.06	.74	-.07
64	7	.65	-.06	.78	-.05
65	6	.66	-.04	.77	-.05
66	6	.67	-.04	.76	-.04
87	6	.68	-.04	.76	-.04
262	4	.79	-.03	.78	-.03
263	4	.79	-.02	.77	-.03
264	4	.78	-.02	.77	-.04
89	4	.77	-.03	.78	-.04
90	4	.76	-.03	.79	-.04
91	112	.74	-.03	.81	-.04
92	112	.72	-.03	.83	-.04
93	112	.70	-.03	.86	-.04
94	59	.68	-.03	.88	-.03
95	59	.67	-.03	.89	-.03
96	59	.67	-.04	.90	-.03
97	59	.67	-.04	.90	-.03

VERTICAL SEISMIC LOAD (TENSION)

REBAR STRESS - CYLINDER - UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
211	113	.57	-.02	.55	-.04
212	113	.56	-.02	.54	-.03
213	113	.55	-.02	.53	-.04
202	113	.54	-.03	.53	-.04
203	113	.52	-.03	.54	-.03
206	115	.47	-.04	.59	-.02
207	114	.45	-.04	.60	-.02
208	114	.45	-.04	.61	-.01
209	114	.44	-.05	.62	-.01
210	114	.44	-.05	.62	-.02
217	2	.58	-.03	.56	-.03
218	2	.58	-.03	.57	-.03
219	2	.58	-.03	.57	-.03
172	2	.57	-.03	.58	-.03
173	2	.57	-.04	.58	-.03
174	5	.56	-.05	.58	-.04
175	5	.54	-.06	.57	-.05
176	5	.52	-.06	.55	-.06
177	60	.47	-.06	.58	-.05
178	6	.48	-.04	.56	-.04
179	6	.48	-.04	.55	-.03
195	6	.49	-.04	.55	-.02
223	2	.60	-.04	.60	.01
224	2	.60	-.04	.60	0
225	2	.59	-.04	.59	-.01
150	2	.59	-.04	.59	-.01
151	2	.59	-.04	.60	-.01
152	5	.59	-.04	.62	-.04
153	5	.60	-.07	.64	-.06
154	5	.63	-.12	.63	-.12

VERTICAL SEISMIC LOAD (TENSION)

REBAR STRESS- CYLINDER - UPPER QUADRANT (KSI) (CONT'D)

PANEL NO.	PANEL TYPE	INNER		OUTER	
		V	H	V	H
229	1	.63	-.06	.55	.02
230	1	.62	-.05	.63	.01
133	1	.58	-.04	.55	.01
134	3	.59	-.04	.52	.02
135	69	.54	-.08	.59	.06
237	1	.62	-.06	.62	. 0
114	1	.60	-.05	.57	-.01
115	1	.57	-.05	.52	-.01
116	3	.55	-.04	.47	.01

DEAD LOAD

REBAR STRESS - BOSS

PANEL NO.	PANEL TYPE	INNER H	INNER V	OUTER H	OUTER V
113	104	.51	.	.39	
201	102	.14		-.39	
200	103	.08		-1.06	
199	81	.48		-1.41	
198	73	1.11	-3.99	.32	-1.16
197	65	.74	-3.74	.48	-2.27
112	105	.28		.03	
185	85	-.04		-.67	
184	84	-.10		-.75	
183	83	-.16		-1.27	
182	74	1.38	-3.61	1.65	-2.24
181	64	.81	-3.64	1.17	-2.71
110	107	-.40		.11	
159	91	-.63		-.59	
158	90	-.88		-.30	
157	89	-1.09		-.63	
156	76	1.11	-4.30	2.73	-5.16
155	62	1.17	-4.07	1.93	-4.97
108	109	-2.20		.39	
140	97	-3.32		.03	
139	96	-3.23		-1.56	
138	95	-3.57		-.60	
137	78	.37	-5.76	-1.99	-5.18
136	70	.59	-5.45	-1.03	-5.02

DEAD LOAD

REBAR STRESS - BOSS (Continued)

PANEL NO.	PANEL TYPE	INNER H	INNER V	OUTER H	OUTER V
106	111	-4.40		.74	
122	101	-3.67		.71	
121	66	-3.73		.60	
120	82	-4.50		-.02	
119	80	.21	-5.48	-1.06	-3.46
118	72	.32	-5.36	-.96	-4.06

DEAD LOAD

REBAR STRESS - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
250	2	-6.17	.37	-6.12	-.09
251	2	-6.13	.34	-6.07	.03
252	2	-6.10	.32	-6.03	.08
37	2	-6.11	.32	-6.02	.10
38	2	-6.09	.36	-6.12	.12
39	5	-6.06	.30	-6.34	.37
40	5	-6.17	.59	-6.55	.52
256	2	-6.67	.28	-6.52	.11
257	2	-6.66	.24	-6.58	.21
258	2	-6.66	.24	-6.63	.26
59	2	-6.64	.28	-6.70	.30
60	2	-6.61	.35	-6.76	.33
61	5	-6.56	.45	-6.77	.40
62	5	-6.42	.54	-6.73	.51
64	7	-5.31	.50	-6.96	.48
65	6	-5.92	.34	6.82	.48
66	6	-6.01	.37	-6.73	.35
82	6	-6.08	.36	-6.73	.33
262	4	-7.09	.25	-7.01	.22
263	4	-7.09	.22	-6.90	.28
264	4	-7.03	.22	-6.95	.32
89	4	-6.94	.24	-7.04	.34
90	4	-6.83	.27	-7.14	.35
91	112	-6.68	.29	-7.29	.35
92	112	-6.50	.29	-7.49	.35
93	112	-6.32	.29	-7.70	.34
94	59	-6.20	.30	-7.86	.30
95	59	-6.13	.31	-7.95	.28
96	59	-6.09	.34	-8.01	.28
97	59	-6.08	.37	-8.06	.29
63	5	-6.31	.54	-6.56	.58

DEAD LOAD

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
211	113	-4.30	.16	-4.19	.28
212	113	-4.23	.15	-4.09	.26
213	113	-4.16	.17	-4.05	.27
202	113	-4.06	.20	-4.06	.27
203	113	-3.94	.23	-4.11	.26
206	115	-3.46	.29	-4.48	.16
207	114	-3.36	.31	-4.60	.10
208	114	-3.30	.33	-4.69	.09
209	114	-3.27	.36	-4.75	.09
210	114	-3.26	.40	-4.79	.11
217	2	-4.56	.20	-4.40	.22
218	2	-4.56	.21	-4.45	.22
219	2	-4.52	.22	-4.49	.21
172	2	-4.48	.27	-4.54	.22
173	2	-4.43	.34	-4.52	.25
174	5	-4.35	.43	-4.55	.31
175	5	-4.20	.50	-4.47	.41
176	5	-4.04	.49	[illegible]	.47
177	60	-3.63	.44	-4.50	.38
178	6	-3.64	.31	-4.34	.32
179	6	-3.70	.32	-4.27	.20
195	6	-3.74	.31	-4.26	.17
223	2	-4.88	.32	-4.89	-.05
224	2	-4.82	.30	-4.83	.03
225	2	-4.79	.28	-4.78	.06
150	2	-4.79	.29	-4.76	.05
151	2	-4.78	.34	-4.85	.06
152	5	-4.76	.30	-5.03	.29
153	5	-4.84	.57	-5.21	.47
154	5	-5.10	.96	-5.08	1.06
229	1	-5.16	.49	-5.37	-.20

DEAD LOAD

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI) (CONT'D)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
230	1	-5.07	.44	-5.18	-.06
133	1	-4.01	.36	-4.50	-.08
134	3	-4.87	.32	-4.28	-.15
135	69	-4.48	.66	-4.84	-.53
237	1	-5.21	.49	-5.17	.05
114	1	-5.05	.46	-4.80	.11
115	1	-4.82	.40	-4.35	.06
116	3	-4.57	.31	-3.93	-.09

INTERNAL PRESSURE LOAD (47 psi)

REBAR STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	INNER H	INNER V	OUTER H	OUTER V
113	104	18.66		1.30	
201	102	14.70		2.05	
200	103	14.91		3.60	
199	81	18.38		7.00	
198	73	21.68	16.27	15.65	17.30
197	65	21.09	15 20	17.16	19.85
112	105	14.20		.80	
185	85	14.64		5.82	
184	84	15.02		10.57	
183	83	19.36		8.82	
182	74	22.74	13.87	17.19	26.22
181	64	20.19	15.29	18.51	19.99
110	107	8.13		.51	
159	91	9.51		5.45	
158	90	11.91		9.59	
157	89	16.26		8.85	
156	76	20.42	11.79	23.44	20.72
155	62	21.04	11.09	24.71	20.26
108	109	8.63		1.42	
140	97	10.01		17.51	
139	96	12.42		19.41	
138	95	15.58		13.77	
137	78	16.87	7.34	25.69	5.84
136	70	16.93	9.16	23.77	6.79
106	111	9.85		2.13	
122	101	8.08		10.58	
121	66	7.88		13.59	
120	82	9.75		14.72	
119	80	20.22	7.59	12.62	10.76
118	72	19.13	9.16	16.61	11.91

INTERNAL PRESSURE LOAD (47 PSI)

REBAR STRESS - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
250	2	11.48	21.03	14.51	22.26
251	2	11.22	20.93	14.25	22.47
252	2	10.89	20.95	13.98	22.54
37	2	10.55	21.09	13.64	22.63
38	2	9.92	21.46	13.64	22.80
39	5	9.04	22.60	14.06	22.74
40	5	9.04	22.68	14.69	24.54
256	2	10.00	20.66	17.43	21.65
257	2	9.90	20.75	17.41	21.64
258	2	9.89	20.90	17.37	21.73
59	2	9.91	21.14	17.50	21.87
60	2	10.09	21.39	17.83	22.15
61	5	10.61	21.56	18.40	22.46
62	5	11.44	21.53	19.19	22.58
63	5	12.91	21.26	19.47	22.20
64	7	11.08	20.80	22.98	21.85
65	6	12.25	21.30	22.76	19.85
66	6	12.57	20.94	22.86	19.63
82	6	12.43	20.98	23.30	19.09
262	4	9.12	18.31	21.07	19.05
263	4	9.46	18.12	21.05	18.92
264	4	9.81	18.07	21.18	18.67
89	4	10.33	18.06	21.28	18.48
90	4	10.90	18.09	21.40	18.37
91	112	11.42	18.16	21.56	18.79
92	112	11.73	18.32	21.80	18.25
93	112	11.72	18.61	22.08	18.19
94	59	11.70	18.80	22.19	18.31
95	59	11.88	18.85	22.16	18.07
96	59	12.12	18.57	22.18	17.56
97	59	12.31	18.36	22.27	17.09

INTERNAL PRESSURE LOAD (47 PSI)

REBAR STRESS - CYLINDER UPPER QUADRANT (PSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
211	113	12.57	20.49	12.17	21.28
212	113	12.67	20.54	12.19	21.28
213	113	12.87	20.63	12.12	21.20
202	113	13.20	20.71	12.02	21.10
203	113	13.62	20.77	11.96	20.97
206	115	14.42	21.00	12.19	20.23
207	114	14.53	20.98	12.21	20.10
208	114	14.75	21.02	12.12	19.83
209	114	14.96	20.86	12.01	19.60
210	114	15.07	20.76	11.98	19.41
217	2	12.38	20.23	11 96	21.34
218	2	12.31	20.39	11.78	21.34
219	2	12.22	20.61	11.62	21.40
172	2	12.17	20.89	11.63	21.53
173	2	12.29	21.15	11.82	21.78
174	5	12.72	21.30	12.22	22.04
175	5	13.44	21.23	12.87	22.04
176	5	14.90	20.91	12.94	21.57
177	60	12.72	20.35	16.70	21.31
178	6	14.12	20.99	16.12	19.26
179	6	14.54	20.55	16.15	19.16
195	6	14.46	20.57	16.59	18.66
223	2	12.51	19.86	12.57	21.32
224	2	12.28	19.83	12.27	21.38
225	2	11.98	19.87	11.91	21.39
150	2	11.64	20.01	11.49	21.43
151	2	10.98	20.35	11.43	21.56
152	5	10.06	21.51	11.81	21.40
153	5	10.02	21.50	12.35	23.17
154	5	11.84	21.28	14.17	25.17
229	1	12.59	19.79	13.14	21.51

INTERNAL PRESSURE LOAD (47 psi)

REBAR STRESS - CYLINDER UPPER QUADRANT (CONT'D)

PANEL NO.	PANEL TYPE	INNER		OUTER	
		V	H	V	H
230	1	12.42	19.51	12.92	21.36
133	1	11.47	18.54	11.44	20.94
134	3	11.24	18.57	10.12	20.47
135	69	8.10	17.34	11.40	22.85
237	1	12.23	19.14	12.85	21.61
114	1	12.01	18.80	12.38	21.19
115	1	11.61	18.62	11.75	20.40
116	3	11.15	18.67	11.27	19.38

SYMMETRIC HORIZONTAL SEISMIC LOAD (.10g)

REBAR STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	INNER H	INNER V	OUTER H	OUTER V
113	104	-.61		-.23	
201	102	-.34		.07	
200	103	-.33		.36	
199	81	-.65		.46	
198	73	-1.03	1.65	-.59	.07
197	65	-.80	1.49	-.71	.60
112	105	-.43		-.06	
185	85	-.26		.08	
184	84	-.28		.04	
183	83	-.39		.36	
182	74	-1.14	1.47	-1.35	.38
181	64	-.79	1.42	-1.10	.82
101	107	.0[illegible]		-.09	
159	91	.13		.06	
158	90	.18		-.21	
157	89	.10		.00	
156	76	-.91	1.86	-2.13	1.93
155	62	-.99	1.70	-1.67	1.95
108	109	.88		-.27	
140	97	1.48		-.94	
139	96	1.34		.21	
138	95	1.43		-.09	
137	78	-.60	2.86	.30	2.61
136	70	-.68	2.[illegible]1	-.06	2.48
106	111	2.08		-.49	
122	101	1.76		-.67	
121	66	1.77		-.66	
120	82	2.17		-.34	
119	80	-.52	2.72	.23	1.59
118	72	-.53	2.56	.11	1.86

SYMMETRIC HORIZONTAL SEISMIC LOAD (.10g)

REBAR STRESS - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER		OUTER	
		V	H	V	H
250	2	2.59	-.51	2.65	-.12
251	2	2.69	-.48	2.74	-.23
252	2	2.79	-.46	2.82	-.31
37	2	2.90	-.46	2.90	-.36
38	2	2.99	-.48	3.02	-.40
39	5	3.06	-.46	3.20	-.56
40	5	3.18	-.61	3.37	-.68
256	2	2.95	-.32	2.96	-.18
257	2	3.07	-.29	3.12	-.27
258	2	3.18	-.28	3.27	-.33
59	2	3.27	-.31	3.42	-.36
60	2	3.35	-.36	3.55	-.40
61	5	3.39	-.42	3.66	-.44
62	5	3.38	-.47	3.72	-.51
63	5	3.36	-.48	3.69	-.55
64	7	3.13	-.44	3.95	-.47
65	6	3.22	-.35	3.92	-.44
66	6	3.29	-.37	3.89	-.36
82	6	3.34	-.36	3.89	-.33
262	4	3.24	-.18	3.28	-.18
263	4	3.36	-.15	3.37	-.23
264	4	3.42	-.16	3.57	-.26
89	4	3.47	-.18	3.80	-.27
90	4	3.48	-.20	4.03	-.28
91	112	3.46	-.21	4.29	-.28
92	112	3.43	-.22	4.55	-.27
93	112	3.39	-.23	4.81	-.26
94	59	3.36	-.24	4.99	-.23
95	59	3.34	-.25	5.10	-.21
96	59	3.34	-.26	5.18	-.20
97	59	3.34	-.28	5.23	-.20

SYMMETRIC HORIZONTAL SEISMIC LOAD (.10g)

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
211	113	1.23	-.37	1.29	-.50
212	113	1.22	-.39	1.36	-.49
213	113	1.22	-.41	1.46	-.51
202	113	1.20	-.43	1.59	-.52
203	113	1.16	-.45	1.73	-.53
206	115	1.00	-.50	2.26	-.51
207	114	.96	-.51	2.40	-.48
208	114	.94	-.52	2.49	-.47
209	114	.92	-.53	2.55	-.47
210	114	.93	-.42	2.59	-.48
217	2	1.43	-.46	1.44	-.41
218	2	1.50	-.45	1.55	-.44
219	2	1.55	-.46	1.65	-.46
172	2	1.59	-.49	1.75	-.50
173	2	1.62	-.54	1.83	-.54
174	5	1.62	-.59	1.88	-.61
175	5	1.57	-.63	1.89	-.68
176	5	1.49	-.61	1.86	-.72
177	60	1.46	-.60	1.74	-.78
178	6	1.32	-.50	1.88	-.60
179	6	1.35	-.48	1.86	-.53
195	6	1.38	-.47	1.85	-.50
223	2	1.66	-.57	1.75	-.20
224	2	1.72	-.54	1.81	-.30
225	2	1.79	-.56	1.87	-.37
150	2	1.88	-.54	1.94	-.41
151	2	1.96	-.57	2.04	-.47
152	5	2.04	-.59	2.19	-.62
153	5	2.15	-.72	2.31	-.79
154	5	2.28	-.92	2.23	-1.18
229	1	1.85	-.67	2.05	-.10
230	1	1.91	-.63	2.06	-.24

SYMMETRIC HORIZONTAL SEISMIC LOAD (.10g)

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI) (CONT'D)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
133	1	2.05	-.56	1.97	-.38
134	3	2.14	-.55	1.94	-.37
135	69	2.04	-.70	2.20	-.28
237	1	2.14	-.65	2.20	-.34
114	1	2.15	-.63	2.09	-.42
115	1	2.12	-.59	1.93	-.44
116	3	2.06	-.55	1.77	-.39

THERMAL LOAD I

REBAR STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	INNER H	INNER V	OUTER H	OUTER V
113	104	-4.00		9.65	
201	102	-4.70		12.76	
200	103	-4.00		11.82	
199	81	-2.41		9.90	
198	73	-3.04	-4.55	14.24	18.40
197	65	-3.36	-4.27	15.41	19.93
112	105	-1.40		4.51	
185	85	-6.30		17.93	
184	84	-3.36		12.25	
183	83	-.63		9.43	
182	74	-6.59	-.41	17.03	19.55
181	64	-4.98	-4.16	16.18	22.65
110	107	-.20		4.41	
159	91	-5.44		10.87	
158	90	-3.70		12.97	
157	89	.11		10.97	
156	76	-11.49	-4.78	26.17	27.11
155	62	-7.87	.25	21.94	20.43
108	109	-.32		5.44	
140	97	-6.89		21.88	
139	96	-3.47		17.94	
138	95	.12		12.47	
137	78	-.71	-7.83	14.69	25.34
136	70	-.43	-8.78	13.51	26.19
106	111	-3.25		10.20	
122	101	-4.45		14.41	
121	66	-4.04		13.11	
120	82	-2.07		10.16	
119	80	-6.79	-2.96	14.34	16.06
118	72	-6.77	-1.94	14.72	17.76

THERMAL LOAD I

REBAR STRESSES - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
250	2	-6.78	-10.70	30.11	23.19
251	2	-6.95	- 9.66	29.84	21.67
252	2	-7.16	- 8.74	29.67	20.40
37	2	-7.56	-7.92	29.77	19.35
38	2	-7.32	- 7.42	29.34	18.93
39	5	-6.12	- 9.12	28.37	20.99
40	5	-5.43	- 8.54	28.37	20.45
256	2	-6.05	-12.35	27.79	19.19
257	2	-6.11	-12.08	28.24	18.32
258	2	-5.96	-11.72		
59	2	-5.82	-11.59	28.42	17.81
60	2	-5.72	-11.52	28.43	17.69
61	5	-5.56	-11.51	28.42	17.53
62	5	-5.27	-11.53	28.20	17.38
63	5	-6.39	-11.45	29.27	17.06
64	7	-1.03	-10.66	22.38	15.25
65	6	-3.73	-11.88	24.82	16.92
66	6	-5.25	-10.55	26.65	15.14
82	6	-5.59	-10.27	27.00	14.83
262	4	-1.84	-14.72	21.42	15.80
263	4	-1.99	-14.08	21.98	14.65
264	4	-3.54	-14.37	23.27	15.67
89	4	-3.62	-14.37	24.15	15.67
90	4	-3.73	-14.27	24.64	15.56
91	112	-3.84	-14.20	24.89	15.53
92	112	-3.85	-14.18	24.85	15.56
93	112	-3.68	-14.21	24.64	15.68
94	59	-3.71	-13.93	24.76	15.44
95	59	-4.05	-13.92	25.28	15.27
96	59	-4.41	-13.75	25.86	14.98
97	59	-4.62	-13.71	26.25	14.80

THERMAL LOAD I

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER		OUTER	
		V	H	V	H
211	113	-5.65	-8.34	27.19	21.39
212	113	-5.76	-8.41	27.70	21.57
213	113	-5.96	-8.38	28.13	21.59
202	113	-6.19	-8.32	28.53	21.58
203	113	-6.40	-8.26	28.82	21.57
206	115	-6.55	-8.22	28.78	21.66
207	114	-6.59	-7.93	28.86	21.37
208	114	-6.81	-7.99	29.19	21.30
209	114	-7.03	-7.87	29.55	21.07
210	114	-7.16	-7.86	29.79	20.94
217	2	-6.03	-8.57	26.64	22.82
218	2	-5.97	-8.31	26.84	22.21
219	2	-6.01	-8.11	27.00	21.85
172	2	-6.00	-7.96	27.07	21.64
173	2	-5.92	-7.84	27.02	21.46
174	5	-5.80	-7.80	26.90	21.31
175	5	-5.52	-7.80	26.61	21.19
176	5	-6.54	-7.71	27.55	20.90
177	60	-1.54	-6.82	21.43	19.21
178	6	-4.08	-7.96	23.68	20.85
179	6	-5.50	-6.72	25.43	19.12
195	6	-5.81	-6.46	25.79	18.81
223	2	-5.95	-9.26	27.23	24.50
224	2	-6.10	-8.37	27.03	23.04
225	2	-6.30	-7.54	26.90	21.77
150	2	-6.70	-6.78	27.01	20.73

THERMAL LOAD I

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI)(Cont'd)

PANEL NO.	PANEL TYPE	INNER		OUTER	
		V	H	V	H
151	2	-6.51	-6.33	26.59	20.29
152	5	-5.44	-7.95	25.63	22.19
154	5	-5.77	-7.58	26.99	22.29
229	1	-5.90	-9.73	28.19	25.52
230	1	-6.15	-8.66	27.63	23.65
133	1	-6.89	-5.35	25.88	18.52
134	3	-8.15	-4.93	27.05	17.63
135	69	-3.83	- .81	21.19	13.43
237	1	-6.43	-8.18	27.74	22.27
114	1	-6.27	-7.73	26.48	20.91
115	1	-5.68	-7.54	24.58	19.68
116	3	-4.73	-7.60	22.49	18.65
153	5	-4.85	-7.41	25.50	21.61

Sheet 15 (continued)

THERMAL LOAD I

*LINER STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
113	104	4.74	-5.20
201	102	.23	-6.74
200	103	-3.31	-7.07
199	81	-5.07	-5.88
★198	73	-12.38	-10.55
★197	65	-12.55	-11.24
112	105	9.61	-6.02
185	85	3.21	-11.51
184	84	-1.18	-7.17
183	83	1.44	-4.34
★182	74	-8.78	-14.41
★181	64	-13.72	-13.74
110	107	1.22	-3.83
159	91	-6.26	-12.67
158	90	-8.53	-8.78
157	89	-2.83	-4.79
★156	76	-18.87	-25.41
★155	62	-8.98	-17.10
108	109	-5.63	4.90
140	97	-14.65	-2.29
139	96	-6.98	-5.63
138	95	-2.86	-.07
★137	78	-17.30	-9.14
★136	70	-18.60	-9.07
106	111	-4.37	4.59
122	101	-6.61	-.41
121	66	-7.43	-4.65
120	82	-5.94	-7.29
★119	80	-11.27	-14.31
★118	72	-10.32	-14.58

*Correction factor of -2.60 ksi must be added to all liner stresses.

★For correction factor see Sheet 17

Sheet 16

THERMAL LOAD I

*LINER STRESSES CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
250	2	-21.65	-24.63
251	2	-21.29	-22.92
252	2	-21.07	-21.46
37	2	-21.21	-20.26
38	2	-20.57	-19.37
39	5	-19.60	-21.61
40	5	-18.43	-20.43
256	2	-20.56	-25.44
257	2	-20.55	-24.91
258	2	-20.22	-24.28
59	2	-19.99	-24.04
60	2	-19.81	-23.87
61	5	-19.59	-23.76
62	5	-19.15	-23.62
63	5	-20.81	-23.94
64	7	-11.79	-19.87
65	6	-16.47	-23.14
66	6	-18.22	-21.65
82	6	-18.61	-21.36
262	4	-14.26	-25.69
263	4	-14.26	-24.66
264	4	-16.78	-25.99
89	4	-17.09	-26.09
90	4	-17.29	-25.99
91	112	-17.47	-25.96
92	112	-17.45	-25.94
93	112	-17.21	-25.93
94	59	-17.14	-25.52
95	59	-17.70	-25.64
96	59	-18.22	-25.52
97	59	-18.55	-25.52

*Correction factor of -7.5 ksi must be added to all linear stresses.

THERMAL LOAD I

*LINER STRESS CYLINDER - UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
211	113	-18.44	-20.39
212	113	-18.74	-20.61
213	113	-19.09	-20.68
202	113	-19.46	-20.72
203	113	-19.78	-20.74
206	115	-19.95	-20.76
207	114	-19.89	-20.32
208	114	-20.27	-20.49
209	114	-20.58	-20.40
210	114	-20.79	-20.41
217	2	-19.01	-21.15
218	2	-18.83	-20.66
219	2	-18.81	-20.32
172	2	-18.74	-20.07
173	2	-18.56	-19.83
174	5	-18.34	-19.69
175	5	-17.91	-19.53
176	5	-19.40	-19.80
177	60	-10.95	-15.81
178	6	-15.38	-18.90
179	6	-17.03	-17.50
195	6	-17.40	-17.23
223	2	-19.43	-22.49
224	2	-19.13	-21.00
225	2	-18.94	-19.65
150	2	-19.10	-18.55
151	2	-18.56	-17.74
152	5	-17.71	-19.88
153	5	-16.64	-18.76
154	5	-18.31	-19.63
229	1	-19.84	-23.42
230	1	-19.4[illegible]	-21.59
133	1	-18.38	-16.09
134	3	-20.06	-15.90
135	69	-11.08	- 7.25
237	1	-19.58	-20.73
114	1	-18.81	-19.66
115	1	-17.45	-18.75
116	3	-15.68	-18.06

*Correction factor of -7.5 ksi must be added to all liner stresses.

THERMAL LOAD II

REBAR STRESS - BOSS(KSI)

PANEL NO.	PANEL TYPE	INNER		OUTER	
		H	V	H	V
113	104	-4.63		11.23	
201	102	-5.46		14.83	
200	103	-4.66		13.72	
199	81	-2.85		11.46	
198	73	-3.17	-5.53	16.44	21.47
197	65	-3.60	-5.24	17.68	23.43
112	105	-1.62		5.24	
185	85	-7.31		20.83	
184	84	-3.92		14.21	
183	83	- .78		10.92	
182	74	-7.29	- .70	19.59	22.90
181	64	-5.51	-5.14	18.53	26.62
110	107	- .24		5.13	
159	91	-6.34		12.62	
158	90	-4.33		15.06	
157	89	.09		12.72	
156	76	-13.05	-5.88	30.13	31.85
155	62	-8.83	- .15	25.16	24.16
108	109	- .44		6.34	
140	97	-8.09		25.44	
139	96	-4.09		20.84	
138	95	.06		14.49	
137	78	- .49	-9.84	16.73	30.12
136	70	- .22	-10.93	15.36	31.17
106	111	-3.89		11.87	
122	101	-5.26		16.76	
121	66	-4.77		15.24	
120	82	-2.45		11.80	
119	80	-7.76	-4.23	16.54	19.21
118	72	-7.69	-2.97	16.84	21.29

Sheet 19

THERMAL LOAD XI

REBAR STRESSES-CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
250	2	-8.32	-12.08	35.59	26.89
251	2	-8.51	-10.88	35.26	24.93
252	2	-8.77	-9.84	34.98	23.34
37	2	-9.30	-8.92	35.03	22.07
38	2	-9.06	-8.40	34.47	21.55
39	5	-7.65	-10.48	33.31	23.95
40	5	-6.82	-9.85	33.34	23.28
256	2	-7.15	-13.44	32.30	22.67
257	2	-7.15	-13.32	32.68	21.55
258	2	-7.18	-13.25	32.95	20.70
59	2	-7.10	-13.34	33.03	20.18
60	2	-6.93	-13.45	33.05	19.79
61	5	-6.70	-13.59	33.04	19.51
62	5	-6.30	-13.67	32.85	19.34
63	5	-7.52	-13.53	34.16	19.05
64	7	-1.24	-12.54	26.47	17.04
65	6	-4.31	-13.89	29.08	19 01
66	6	-6.06	-12.28	31.20	16.98
82	6	-6.46	-11.92	31.59	16.64
262	4	-3.83	-14.53	24.75	20.1[illegible]
263	4	-3.90	-15.10	25.60	18.67
264	4	-3.90	-16.00	26.96	18.00
89	4	-3.85	-16.6[illegible]	[illegible]	[illegible]
90	4	-3.84	-17.02	28.94	17.00
91	112	-3.89	-17.16	29.28	16.79
92	112	-3.94	-17.06	29.11	16.89
93	112	-3.92	-16.76	28.61	17.38
94	59	-4.15	-15.99	28.49	17.57
95	59	-4.65	-15.79	28.98	17.56
96	59	-5.08	-15.71	29.66	17.08
97	59	-5.30	-15.74	30.14	16.78

Sheet 20

THERMAL LOAD II

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
211	113	-6.[illegible]7	-9.60	31.83	24.65
212	113	-6.56	-9.54	31.88	24.64
213	113	-6.75	-9.48	32.25	24.64
202	113	-7.11	-9.42	32.81	24.65
203	113	-7.52	-9.37	33.38	24.66
206	115	-8.21	-9.36	34.15	24.85
207	114	-8.29	-9.02	34.31	24.50
208	114	[illegible]	-9.08	34.64	24.39
209	114	-8.70	-8.95	35.01	24.11
210	114	-8.86	-8.93	35.29	23.95
217	2	-7.51	-9.65	31.64	26.44
218	2	-7.41	-9.50	31.87	25.44
219	2	-7.40	-9.18	32.00	24.86
172	2	-7.36	-8.98	31.99	24.59
173	2	-7.27	-8.85	31.86	24.43
174	5	-7.15	-8.84	31.68	24.34
175	5	-6.86	-8.86	31.33	24.29
176	5	-8.05	-8.77	32.44	23.99
177	60	-2.26	-7.74	25.35	22.03
178	6	-5.19	-9.05	28.02	23.90
179	6	-6.83	-7.60	30.07	21.86
195	6	-7.18	-7.30	30.51	21.48
223	2	-7.43	-10.54	32.35	28.23
224	2	-7.62	-9.50	32.13	26.48
225		-7.67	-8.52	31.99	24.94
150	2	-8.35	-7.63	32.10	23.70
151	2	-8.13	-7.10	31.56	23.18
152	5	-6.85	-9.10	30.36	25.45
154	5	-7.23	-8.55	31.83	25.60
229	1	-7.36	-11.02	33.51	29.38
230	1	-7.68	-9.77	32.86	27.19
133	1	-8.64	-5.93	30.84	21.18
134	3	-10.14	-5.45	32.20	20.16
135	69	-5.04	-.61	25.26	15.22

THERMAL LOAD II

REBAR STRESS CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
237	1	-8.04	-9.18	33.03	2
114	1	-7.89	-8.68	31.57	2
115	1	-7.24	-8.49	29 36	2
116	3	-6.38	-8.58	26.91	2
153	5	-6.16	-8.36	30.14	3

THERMAL LOAD II

*LINER STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
113	104	5.50	-6.03
201	102	.24	-7.83
200	103	-3.92	-8.25
199	81	-6.03	-6.93
★198	73	-14.56	-11.83
★197	65	-14.87	-12.73
112	105	11.15	-6.96
185	85	3.67	-13.36
184	84	-1.47	-8.34
183	83	1.55	-5.12
★182	74	-10.38	-16.30
★181	64	-16.29	-15.67
110	107	1.33	-6.41
159	91	-7.42	-14.66
158	90	-10.06	-10.17
157	89	-3.43	-5.57
★156	76	-22.30	-29.22
★155	62	-10.96	-19.58
108	109	-6.70	5.74
140	97	-17.23	-2.60
139	96	-8.27	-6.51
138	95	-3.50	-.06
★137	78	-21.13	-10.60
★136	70	-22.60	-10.42
106	111	-5.20	5.35
122	101	-7.78	-.47
121	66	-6.72	-5.38
120	82	-6.95	-8.46
★119	80	-14.20	-17.35
★118	72	-12.98	-16.96

*Correction factor of -6.29 ksi must be added to all liner stresses.

★For correction factor see Sheet 23 .

THERMAL LOAD II

*LINER STRESSES - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
250	2	-25.74	-28.35
251	2	-25.29	-26.31
252	2	-25.04	-24.63
37	2	-25.29	-23.30
38	2	-24.60	-22.35
39	5	-23.50	-25.08
40	5	-22.10	-23.76
256	2	-23.70	-28.47
257	2	-23.65	-28.06
258	2	-23.66	-27.79
59	2	-23.58	-27.76
60	2	-23.38	-27.75
61	5	-23.11	-[illegible].79
62	5	-22.55	-27.67
63	5	-24.37	-27.99
64	7	-13.82	-23.18
65	6	-19.14	-26.84
66	6	-21.14	-25.04
82	6	-21.58	-24.66
262	4	-17.88	-27.50
263	4	-18.28	-27.[illegible]
264	4	-18.90	-29.13
89	4	-19.31	-29.91
90	4	-19.61	-30.36
91	112	-19.79	-30.54
92	112	-19.78	-30.44
93	112	-19.56	-30.12
94	59	-19.53	-29.21
95	59	-20.23	-29.18
96	59	-20.88	-29.17
97	59	-21.28	-29.26

*Correction factor of -10 ksi must be added to all liner stresses.

THERMAL LOAD II

*LINER STRESS - CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
211	113	-21.69	-23.62
212	113	-21.40	-23.46
213	113	-21.71	-23.47
202	113	-22.29	-23.58
203	113	-21.93	-23.71
206	115	-24.03	-24.07
207	114	-24.01	-23.58
208	114	-24.35	-23.73
209	114	-24.65	-23.59
210	114	-24.90	-23.61
217	2	-22.88	-24.75
218	2	-22.57	-23.93
219	2	-22.41	-23.37
172	2	-22.26	-23.02
173	2	-22.05	-22.77
174	5	-21.83	-22.67
175	5	-21.38	-22.54
176	5	-23.14	-22.90
177	60	-13.34	-18.27
178	6	-18.48	-21.82
179	6	-20.37	-20.17
195	6	-20.78	-19.84
223	2	-23.31	-26.04
224	2	-22.97	-24.28
225	2	-22.78	-22.69
150	2	-22.98	-21.39
151	2	-22.31	-20.43
152	5	-21.29	-22.95
153	5	-20.02	-21.64
154	5	-21.95	-22.62
229	1	-23.77	-27.01
230	1	-23.38	-24.89
133	1	-22.22	-18.54
134	3	-24.24	-18.34
135	69	-13.65	-8.17

THERMAL LOAD II

*LINER STRESS CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTA
237	1	-23.53	-23.86
114	1	-22.70	-22.65
115	1	-21.17	-21.64
116	3	-19.16	-20.90

*Correction factor of -10.0 ksi must be added to all liner stresses.

Sheet 24 (continued)

THERMAL LOAD III

REBAR STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	INNER H	INNER V	OUTER H	OUTER V
113	104	-4.92		12.06	
201	102	-5.83		15.94	
200	103	-4.59		14.76	
199	81	-3.07		12.32	
198	73	-5.94	-3.21	17.61	23.19
197	65	-3.68	-5.65	18.87	25.41
112	105	-1.69		5.63	
185	85	-7.82		22.41	
184	84	-4.19		15.28	
183	83	-.83		11.76	
182	74	-7.63	-.76	20.89	24.82
181	64	-5.75	-5.57	19.75	28.88
110	107	-.24		5.51	
159	91	-6.78		13.58	
158	90	-4.62		16.21	
157	89	.13		13.71	
156	76	-13.87	-6.36	32.23	34.64
155	62	-9.33	-.28	26.87	26.36
108	109	-.43		6.82	
140	97	-8.65		27.40	
139	96	-4.34		22.48	
138	95	.12		15.61	
137	78	-.41	-10.80	17.94	32.84
136	70	-.11	-11.99	16.44	33.95
106	111	-4.12		12.76	
122	101	-5.60		18.05	
121	66	-5.08		16.41	
120	82	-2.56		12.71	
119	80	-8.31	-4.80	17.76	21.00
118	72	-8.23	-3.40	18.02	23.27

THERMAL LOAD III

REBAR STRESSES - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
250	2	-9.14	-13.22	38.57	[illegible]
251	2	-9.36	-11.90	38.20	[illegible]
252	2	-9.61	-10.75	37.94	[illegible]
37	2	-10.13	-9.72	36.03	[illegible]
38	2	-9.83	-9.10	37.45	[illegible]
39	5	-8.28	-11.29	36.19	[illegible]
40	5	-7.39	-10.54	36.16	[illegible]
256	2	-7.84	-15.24	34.79	[illegible]
257	2	-7.88	-14.92	35.17	[illegible]
258	2	-7.87	-14.58	35.53	[illegible]
59	2	-7.74	-14.39	35.68	[illegible]
60	2	-7.54	-14.27	35.71	[illegible]
61	5	-7.31	-14.27	35.65	[illegible]
62	5	-6.91	-14.31	35.35	[illegible]
63	5	-8.27	-14.21	36.64	[illegible]
64	7	-1.61	-13.23	28.33	[illegible]
65	6	-4.90	-14.72	31.09	[illegible]
66	6	-6.74	-13.06	33.38	[illegible]
82	6	-7.14	-12.71	33.83	[illegible]
262	4	-3.97	-17.67	26.61	[illegible]
263	4	-3.99	-17.38	27.52	[illegible]
264	4	-4.28	-17.44	28.61	[illegible]
89	4	-4.54	-17.43	29.53	[illegible]
90	4	-4.76	-17.38	30.20	[illegible]
91	112	-4.93	-17.32	30.55	[illegible]
92	112	-4.93	-17.30	30.54	[illegible]
93	112	-4.72	-17.36	30.36	[illegible]
94	59	-4.76	-17.02	30.56	[illegible]
95	59	-5.19	-17.02	31.27	[illegible]
96	59	-5.64	-16.80	32.03	[illegible]
97	59	-5.91	-16.75	32.53	[illegible]

Sheet 26

THERMAL LOAD III

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
211	113	-7.78	-10.06	35.01	26.2
212	113	-7.92	-10.15	35.64	26.4
213	113	-8.17	-10.10	36.15	26.4
202	113	-8.46	-10.03	36.63	26.4
203	113	-8.72	-9.96	36.99	26.4
206	115	-8 90	-9.93	37.04	26.5
207	116	-8.96	-9.57	37.20	26.1
208	116	-9.24	-9.65	37.66	26.0
209	114	-9.52	-9.52	38.14	25.7
210	114	-9.69	-9.50	38.45	25.5
217	2	-8.28	-10.33	34.34	28.0
218	2	-8.21	-10.02	34.57	27.2
219	2	-8.25	-9.77	34.75	26.8
172	2	-8.24	-9.58	34.80	26.5
173	2	-8.13	-9.44	34.72	26.3
174	5	-7.96	-9.40	34.54	26.1
175	5	-7 60	-9.40	34.16	26.0
176	5	-8.83	-9.28	35.31	25.6
177	60	-2.58	-8.17	27.68	23,5
178	6	-5.70	-9.57	30.51	25.4
179	6	-7.43	-8.02	32.70	23.3
195	6	-7.80	-7.70	33.16	22.8
223	2	-8.19	-11.17	35.13	30.1
224	2	-8.40	-10.07	34.87	28.3
225	2	-8.66	-9.03	34.69	26.7
150	2	-9.17	-8.09	34.79	25.4
151	2	-8.93	-7.53	34.23	24.9
152	5	-7.56	-9.59	32.95	27.3
154	5	-7.93	-9.05	34.62	27.4
229	1	-8.13	-11.75	36.41	31.3
230	1	-8.47	-10.41	35.70	29.0
123	1	-9.50	-6.28	33.50	22.6
134	3	-11.13	-5.76	34.99	21.5
[illegible]	69	-5.61	-.52	27.50	16.2
237	1	-8.85	-9.82	35.90	27.2
114	1	-8.69	-9.29	34.35	25.6
115	1	-7.99	-9.08	31.97	24.1
116	3	-6.84	-9.19	29.34	22.8
153	5	-6.79	-8.88	32.75	26.6

THERMAL LOAD III

*LINER STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONT
113	104	5.91	-6.41
201	102	.27	-8.37
200	103	-4.18	-8.83
199	81	-6.43	-7.44
★193	73	-15.58	-12.45
★197	65	-15.98	-13.42
112	105	11.98	-7.41
185	85	3.96	-14.29
184	84	-1.54	-8.93
183	83	1.73	-5.48
★182	74	-11.12	-17.22
★181	64	-17.55	-16.60
110	107	1.46	-4.70
159	91	-7.94	-15.71
158	90	-10.75	-10.90
157	89	-3.58	-5.96
★156	76	-24.04	-31.20
★155	62	-11.94	-20.85
108	109	-7.12	6.17
140	97	-18.44	-2.78
139	96	-8.80	-6.96
138	95	-3.66	-.03
★137	78	-23.07	-11.36
★136	70	-24.66	-11.14
106	111	-5.52	5.75
122	101	-8.31	-.50
121	66	-9.32	-5.78
120	82	-7.40	-9.09
★119	80	-15.65	-18.72
★118	72	-14.30	-18.27

*Correction factor of -8.15 ksi must be added to all liner stresses.

★For correction factor see sheet 29.

THERMAL LOAD III

*LINER STRESSES CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
250	2	-28.05	-30.74
251	2	-27.58	-28.56
252	2	-27.28	-26.72
37	2	-27.48	-25.24
38	2	-26.67	-24.14
39	5	-25.41	-27.00
40	5	-23.87	-25.48
256	2	-25.93	-31.44
257	2	-25.87	-30.84
258	2	-25.76	-30.28
59	2	-25.53	-29.92
60	2	-25.21	-29.64
61	5	-24.88	-29.50
62	5	-24.29	-29.34
63	5	-26.31	-29.73
64	7	-15.09	-24.68
65	6	-20.80	-28.65
66	6	-22.92	-26.77
82	6	-23.37	-26.39
262	4	-19.68	-31.68
263	4	-19.76	-31.21
264	4	-20.42	-31.54
89	4	-20.97	-31.70
90	4	-21.39	-31.77
91	112	-21.60	-31.78
92	112	-21.66	-31.76
93	112	-21.37	-31.77
94	59	-21.31	-31.26
95	59	-22.01	-31.42
96	59	-22.67	-31.28
97	59	-23.10	-31.29

*Correction factor of-11.4ksi must be added to all liner stresses. .

Sheet 29

THERMAL LOAD III

*LINER STRESS CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTA
211	113	-24.04	-25.22
212	113	-24.41	-25.50
213	113	-24.84	-25.58
202	113	-25.30	-25.63
203	113	-25.70	-25.66
206	115	-25.94	-25.71
207	114	-25.89	-25.18
208	114	-26.38	-25.40
209	114	-26.78	-25.29
210	114	-27.06	-25.32
217	2	-24.78	-26.18
218	2	-24.56	-25.57
219	2	-24.52	-25.16
172	2	-24.43	-24.84
173	2	-24.20	-24.55
174	5	-23.90	-24.37
175	5	-23.33	-24.16
176	5	-25.13	-24.48
177	60	-14.57	-19.49
178	6	-20.04	-23.27
179	6	-22.03	-21.50
195	6	-22.46	-21.14
223	2	-25.34	-27.83
224	2	-24.97	-25.98
225	2	-24.75	-24.30
150	2	-24.98	-22.94
151	2	-24.27	-21.92
152	5	-23.17	-24.65
153	5	-21.78	-23.20
154	5	-23.83	-24.20
229	1	-25.86	-28.97
230	1	-25.45	-26.70
133	1	-24.19	-19.88
134	3	-26.40	-19.67
135	69	-14.94	-8.68
237	1	-25.64	-25.65
114	1	-24.73	-24.36
115	1	-23.09	-23.29
116	3	-20.94	-22.50

*Correction factor of -11.4 ksi must be added to all liner stresses.

LOAD COMBINATION 8

$(1.0 \pm .05)\ D + 1.0P + 1.0\ (T'' + TL'') + 1.0E'$

*LINER STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
113	104	4.48	19.51
201	102	2.18	13.23
200	103	1.65	13.30
199	81	4.23	20.52
★198	73	4.87	20.78
★197	65	3.18	17.59
112	105	7.75	20.65
185	85	1.58	15.59
184	84	1.71	19.87
183	83	8.02	29.72
★182	74	4.95	16.83
★181	64	2.04	13.48
110	107	.60	13.89
159	91	-11.50	14.26
158	90	-10.69	21.63
157	89	.62	34.04
★156	76	-9.05	-.85
★155	62	.97	8.51
108	109	-2.84	14.70
140	97	-16.74	13.40
139	96	-5.69	17.60
138	95	.14	32.72
★137	78	-15.43	8.56
★136	70	-13.45	10.21
106	111	-2.20	8.70
122	101	-5.07	6.18
121	66	-4.22	11.58
120	82	-2.00	18.58
★119	80	-8.14	9.60
★118	72	-5.3[illegible]	8.30

*Correction factor of -2.6 ksi must be added to all liner stresses.

★For correction factor see sheet 32

LOAD COMBINATION 8

(1.0 ± .05) D + 1.0P + 1.0 (T" + TL") + 1.0E'

*LINER STRESS - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTA[L]
250	2	-14.79	-.47
251	2	-14.99	.67
252	2	-15.42	1.84
37	2	-16.18	3.05
38	2	-16.31	4.25
39	5	-16.11	2.96
40	5	-15.13	4.41
256	2	-17.74	-3.43
257	2	-18.03	-3.01
258	2	-17.78	-2.24
59	2	-17.48	-1.60
60	2	-16.94	-.83
61	5	-15.81	0
62	5	-13.92	.78
63	5	-13.20	.95
64	7	-5.92	3.89
65	6	-8.70	1.69
66	6	-10.11	2.81
82	6	-10.80	3.06
262	4	-15.13	-7.32
263	4	-15.01	-6.56
264	4	-16.99	-7.73
89	4	-16.33	-7.45
90	4	-15.31	-6.87
91	112	-14.12	-6.29
92	112	-12.81	-5.65
93	112	-11.57	-5.01
94	59	-10.68	-4.13
95	59	-10.37	-3.89
96	59	-10.23	-3.76
97	59	-10.16	-3.77

*Correction factor of -7.5 ksi must be added to all liner stresses..

Sheet 32

LOAD COMBINATION 8

$(1.0 \pm .05)\ D + 1.0P + 1.0\ (T'' + TL'') + 1.0E'$

*LINER STRESS CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
211	113	-5.39	4.59
212	113	-5.49	4.31
213	113	-5.21	4.64
202	113	-4.66	5.11
203	113	-3.86	5.60
206	115	-.64	7.15
207	114	.20	7.83
208	114	.65	8.04
209	114	.93	8.19
210	114	1.01	8.22
217	2	-6.84	3.36
218	2	-6.78	3.96
219	2	-6.78	4.61
172	2	-6.56	5.37
173	2	-5.97	6.23
174	5	-4.85	7.01
175	5	-3.05	7.66
176	5	-2.07	7.79
177	60	3.55	10.12
178	6	1.72	8.64
179	6	.52	9.55
195	6	-.07	9.80
223	2	-8.49	1.44
224	2	-8.52	3.04
225	2	-8.48	4.31
150	2	-9.07	5.45
151	2	-9.31	6.51
152	5	-9.31	5.44
153	5	-8.45	6.62
154	5	-8.06	6.05
229	1	-8.27	1.70
230	1	-8.41	2.12
133	1	-9.90	5.76
134	3	-12.05	5.94
135	69	-6.82	12.40
237	1	-10.29	2.06
114	1	-10.02	2.55
115	1	-9.20	3.26
116	3	-7.96	4.14

*Correction factor of -7.5 ksi must be added to all liner stresses.

LOAD COMBINATION 9

(1.0+.05) D+1.0P+1.0 (T"+TL") +1.0E'

REBAR STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	INNER		OUTER	
		H	V	H	V
113	104	14.15		8.29	
201	102	9.60		14.61	
200	103	10.46		15.17	
199	81	15.35		16.53	
198	73	17.92	11.20	29.24	34.94
197	65	17.09	10.38	31.87	39.00
112	105	12.37		5.24	
185	85	7.93		23.37	
184	84	11.17		22.32	
183	83	18.03		17.83	
182	74	15.47	12.97	33.45	44.67
181	64	14.62	10.54	33.93	41.84
110	107	7.64		4.86	
159	91	3.78		15.96	
158	90	7.79		22.03	
157	89	15.71		19.34	
156	76	8.41	6.57	48.49	46.84
155	62	12.58	10.83	45.60	39.91
108	109	7.96		6.76	
140	97	2.82		38.45	
139	96	8.53		36.47	
138	95	15.08		25.65	
137	78	15.65	-.55	39.33	31.18
136	70	15.93	.18	36.39	32.91
106	111	6.38		12.16	
122	101	3.49		24.53	
121	66	3.70		26.16	
120	82	5.72		24.35	
119	80	12.81	4.54	26.49	26.58
118	72	11.80	6.99	30.76	29.37

Sheet 34

LOAD COMBINATION 9

(1.0+.05) D+1.0P+1.0 (T"+TL") +1.0E'

REBAR STRESSES - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER		OUTER	
		V	H	V	H
250	2	3.82	9.84	43.93	45.24
251	2	3.56	10.83	43.60	43.77
252	2	3.25	11.78	43.25	42.50
37	2	2.68	12.74	43.12	41.50
38	2	2.43	13.58	42.78	41.22
39	5	2.88	12.00	42.32	43.19
40	5	3.66	13.66	43.05	44.35
256	2	3.23	8.05	44.64	40.65
257	2	3.27	8.43	45.27	39.71
258	2	3.58	8.93	45.60	39.30
59	2	3.86	9.30	45.91	39.35
60	2	4.30	9.61	46.41	39.48
61	5	5.08	9.76	47.10	39.61
62	5	6.28	9.68	47.80	39.56
63	5	6.66	9.49	49.23	38.85
64	7	10.22	9.84	46.11	36.74
65	6	8.71	9.12	48.18	36.46
66	6	7.53	10.08	50.12	34.48
82	6	7.08	10.41	50.91	33.65
262	4	6.68	3.53	42.02	34.74
263	4	7.03	3.98	42.76	33.43
264	4	5.98	3.64	44.45	34.19
89	4	6.53	3.61	45.71	33.99
90	4	7.09	3.73	46.59	33.77
91	112	7.58	3.85	47.27	33.66
92	112	7.95	4.03	47.71	33.65
93	112	8.17	4.27	48.00	33.74
94	59	8.16	4.74	48.38	33.63
95	59	8.02	4.78	48.97	33.24
96	59	7.91	4.67	49.65	32.46
97	59	7.90	4.50	50.16	31.82

Sheet 35

LOAD COMBINATION 9

$(1.0 \pm .05)\ D + 1.0P + 1.0\ (T'' + TL'') + 1.0E'$

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
211	113	5.65	11.62	38.05	42
212	113	5.70	11.71	38.70	42
213	113	5.67	11.81	39.34	42
202	113	5.80	11.91	39.88	42
203	113	6.03	11.99	40.32	41
206	115	6.80	12.23	41.06	41
207	114	6.88	12.50	41.28	40
208	114	6.88	12.49	41.59	40
209	114	6.87	12.44	41.90	40
210	114	6.86	12.36	42.14	39
217	2	5.06	11.09	37.52	43
218	2	5.15	11.56	37.68	43
219	2	5.17	12.00	37.76	42
172	2	5.23	12.40	37.89	42
173	2	5.48	12.75	38.13	42
174	5	6.09	12.92	38.51	42
175	5	7.10	12.85	38.95	42
176	5	7.55	12.65	40.06	41
177	60	10.45	13.00	37.64	39
178	6	9.29	12.48	39.39	39
179	6	8.29	13.30	41.18	37
195	6	7.91	13.57	41.97	36
223	2	5.56	10.15	38.45	45
224	2	5.35	10.79	38.00	44
225	2	4.78	11.69	37.90	42
150	2	4.15	12.61	37.81	41
151	2	3.81	13.41	37.45	41
152	5	4.09	12.89	36.91	42
154	5	5.59	13.05	40.53	46
229	1	5.17	9.22	40.84	47
230	1	4.80	10.42	40.43	44
133	1	3.99	12.71	36.89	38
134	3	2.62	13.13	36.82	37
135	69	3.96	16.01	32.22	35
237	1	4.96	10.45	40.06	43
114	1	5.07	10.57	38.39	41
115	1	5.41	10.58	35.93	39
116	3	6.00	10.57	33.42	37
153	5	4.72	13.43	37.33	43

LOAD COMBINATION 10

$(1.0 \pm .05)D + 1.25P + 1.0\,(T' + TL') + 1.25E$

*LINER STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
113	104	5.32	24.22
201	102	3.28	16.84
200	103	3.93	17.22
199	81	8.08	25.88
*198	73	9.79	27.01
*197	65	7.30	23.26
112	105	9.22	25.53
185	85	2.94	20.10
184	84	4.14	24.88
183	83	12.63	36.78
*182	74	9.12	22.28
*181	64	5.77	18.34
110	107	2.30	16.80
159	91	-10.82	17.54
158	90	-9.07	26.49
157	89	4.96	41.37
*156	76	-7.33	1.61
*155	62	3.88	12.27
108	109	.35	16.81
140	97	-14.96	15.75
139	96	-2.17	21.77
138	95	5.42	40.05
*137	78	-14.75	11.92
*136	70	-12.47	13.94
106	111	1.01	9.98
122	101	-2.80	7.56
121	66	-1.72	14.52
120	82	1.54	23.52
*119	80	-6.16	13.81
*118	72	-2.87	12.16

*Correction factor of -6.29 ksi must be added to all liner stresses.

*For correction factor see Sheet 38.

LOAD COMBINATION 10

(1.0 ± .05) D + 1.25P + 1.0 (T' + TL') + 1.25E

*LINER STRESS CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
250	2	-13.02	2.58
251	2	-13.18	4.02
252	2	-13.64	5.42
37	2	-14.53	6.78
38	2	-14.78	8.07
39	5	-14.63	6.61
40	5	-13.48	8.05
256	2	-15.56	.16
257	2	-15.78	.51
258	2	-15.84	.96
59	2	-15.67	1.46
60	2	-15.04	2 12
61	5	-13.73	2.84
62	5	-11.55	3.60
63	5	-10.61	3.83
64	7	-2.80	7.11
65	6	-5.74	4.97
66	6	-7.33	6.33
82	6	-8.12	6.72
262	4	-13.99	-3.22
263	4	-14.15	-3.98
264	4	-14.12	-4.93
89	4	-13.42	-5.28
90	4	-12.36	-5.21
91	112	-11.06	-4.78
92	112	-9.75	-4.02
93	112	-8.61	-2.99
94	59	-7.84	-1.59
95	59	-7.66	-1.16
96	59	-7.63	-1.20
97	59	-7.61	-1.33

*Correction factor of -10 ksi must be added to all liner stresses.

LOAD COMBINATION 10

$(1.0 \pm .05)\ D + 1.25P + 1.0\ (T' + TL') + 1.25E$

*LINER STRESS CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
211	113	-2.77	8.09
212	113	-2.25	8.24
213	113	-1.90	8.66
202	113	-1.47	9.10
203	113	-.90	9.53
206	115	1.45	10.85
207	114	2.22	11.58
208	114	2.74	11.83
209	114	3.06	12.00
210	114	3.11	12.01
217	2	-4.88	6.37
218	2	-4.66	7.37
219	2	-4.52	8.30
172	2	-4.21	9.22
173	2	-3.55	10.14
174	5	-2.34	10.92
175	5	-.39	11.53
176	5	.71	11.64
177	60	6.59	14.14
178	6	4.62	12.72
179	6	3.27	13.79
195	6	2.61	14.13
223	2	-6.42	4.42
224	2	-6.47	6.20
225	2	-6.52	7.71
150	2	-7.21	9.08
151	2	-7.50	10.33
152	5	-7.52	9.18
153	5	-6.53	10.39
154	5	-5.99	9.62
229	1	-6.42	4.39
230	1	-6.57	5.14
133	1	-8.10	9.34
134	3	-10.52	9.52
135	69	-5.08	15.55
237	1	-8.40	5.15
114	1	-8.10	5.71
115	1	-7.19	6.48
116	3	-5.80	7.45

*Correction factor of -10 ksi must be added to all liner stresses.

LOAD COMBINATION 11

$(1.0 \pm .05)\ D + 1.25P + 1.0\ (T' + TL') + 1.25E$

REBAR STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	INNER H	INNER V	OUTER H	OUTER V
113	104	18.37		9.63	
201	102	12.61		17.14	
200	103	13.62		17.80	
199	81	19.70		19.62	
198	73	23.56	13.58	35.52	42.
197	65	22.38	12.55	38.64	47.1
112	105	15.83		6.19	
185	85	10.63		27.66	
184	84	14.42		26.85	
183	83	22.80		21.35	
182	74	20.85	15.50	40.76	54.3
181	64	19.41	12.74	41.27	50.3
110	107	9.60		5.74	
159	91	5.19		19.01	
158	90	10.04		26.55	
157	89	19.64		23.27	
156	76	12.26	7.63	59.03	55.8
155	62	17.20	12.47	55.57	47.8
108	109	9.63		8.10	
140	97	3.51		46.67	
139	96	10.44		44.06	
138	95	13.29		31.08	
137	78	20.16	-1.85	47.57	36.3
136	70	20.58	- .73	44.12	38.6
106	111	7.37		14.53	
122	101	4.00		29.75	
121	66	4.22		31.90	
120	82	6.46		29.76	
119	80	17.04	4.10	31.71	31.7
118	72	15.83	7.23	36.95	35.1

Sheet 40

LOAD COMBINATION 11

$(1.0 \pm .05)$ D + 1.25P + 1.0 (T' + TL') + 1.25E

REBAR STRESSES - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
250	2	4.04	13.86	51.91	54.51
251	2	3.67	14.97	51.43	52.70
252	2	3.19	16.04	50.90	51.16
37	2	2.38	17.13	50.60	49.97
38	2	1.94	18.10	50.09	49.63
39	5	2.36	17.42	49.49	51.96
40	5	3.27	18.20	50.35	53.47
256	2	3.42	12.21	52.28	49.60
257	2	3.45	12.46	52.79	48.44
258	2	3.55	12.72	53.14	47.67
59	2	3.78	12.93	53.52	47.30
60	2	4.30	13.14	54.07	47.23
61	5	5.29	13.21	54.89	47.34
62	5	6.84	13.10	55.81	47.32
63	5	7.54	12.90	57.59	46.56
64	7	11.68	13.31	54.30	44.14
65	6	10.12	12.55	56.75	43.64
66	6	8.80	13.69	59.03	41.34
82	6	8.25	14.10	59.99	40.34
262	4	5.66	8.35	49.28	43.92
263	4	6.16	7.54	50.28	42.28
264	4	6.73	6.57	52.01	41.28
89	4	7.55	5.92	53.52	40.46
90	4	8.38	5.56	54.68	39.89
91	112	9.10	5.50	55.43	39.58
92	112	9.56	5.79	55.73	39.62
93	112	9.69	6.44	55.71	40.05
94	59	9.53	7.44	55.82	40.37
95	59	9.32	7.68	56.33	40.07
96	59	9.23	7.42	57.07	38.97
97	59	9.26	7.12	57.68	38.08

Sheet 41

LOAD COMBINATION 11

(1.0 ± .05) D + 1.25P + 1.0 (T' + TL') + 1.25E

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
211	113	7.01	15.60	45.09	51.
212	113	7.40	15.81	45.25	50.
213	113	7.44	15.97	45.78	50.
202	113	7.55	16.11	46.41	50.
203	113	7.73	16.21	47.06	50.
206	115	8.29	16.47	48.49	49.
207	114	8.40	16.78	48.74	49.
208	114	8.52	16.77	48.98	48.
209	114	8.59	16.71	49.25	48.
210	114	8.58	16.61	49.50	47.
217	2	5.95	14.92	44.77	52.
218	2	6.04	15.60	44.83	51.
219	2	6.07	16.20	44.86	51.
172	2	6.14	16.74	44.89	51.
173	2	6.44	17.18	45.07	51.
174	5	7.17	17.39	45.48	51.
175	5	8.42	17.31	46.03	51.
176	5	9.11	17.02	47.36	50.
177	60	12.35	17.36	44.92	48.
178	6	11.18	16.78	46.98	47.
179	6	10.09	17.69	49.12	45.
195	6	9.66	18.01	50.11	44.
223	2	6.36	13.96	45.93	54.
224	2	6.03	14.78	45.38	53.
225	2	5.35	15.83	45.12	51.
150	2	4.53	16.93	44.89	50.
151	2	4.04	17.90	44.36	49.
152	5	4.28	17.38	4[illegible].63	51.
154	5	6.18	17.69	47.96	56
229	1	6.03	13.10	48.43	56.
230	1	5.56	14.33	47.84	53
133	1	4.17	16.91	43.83	46
134	3	2.44	17.39	43.67	45
135	69	3.83	20.76	38.17	42
237	1	5.45	14.39	47.58	52
114	1	5.48	14.47	45.65	49
115	1	5.77	14.44	42.76	47
116	3	6.35	14.39	39.82	45
153	5	4.99	18.08	44.06	53

Sheet 42

LOAD COMBINATION 12

$(1.0 \pm .05)\ D + 1.5P + 1.0\ (T + TL)$

REBAR STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	INNER H	INNER V	OUTER H	OUTER V
113	104	23.56		10.48	
201	102	16.35		18.64	
200	103	17.45		19.15	
199	81	24.95		21.48	
198	73	30.36	14.67	-.02	41.38
197	65	28.67	13.60	45.07	53.04
112	105	19.87		6.86	
185	85	14.10		30.51	
184	84	18.24		30.41	
183	83	28.06		23.77	
182	74	27.79	16.63	48.25	62.02
181	64	25.31	13.89	48.63	56.29
110	107	11.58		6.38	
159	91	6.89		21.19	
158	90	12.40		30.31	
157	89	23.48		26.39	
156	76	17.83	7.24	59.99	60.81
155	62	23.33	12.48	65.77	52.03
108	109	10.43		9.31	
140	97	3.20		54.27	
139	96	11.22		50.11	
138	95	20.01		35.68	
137	78	25.24	-5.26	54.59	36.67
136	70	25.84	-3.42	51.08	39.37
106	111	6.48		16.66	
122	101	3.04		34.59	
121	66	3.20		37.36	
120	82	5.09		34.76	
119	80	22.23	1.38	35.68	33.86
118	72	20.77	5.25	42.02	37.27

LOAD COMBINATION 12

(1.0 ± .05) D + 1.5P + 1.0 (T + TL)

REBAR STRESSES - CYLINDER LOWER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	INNER V	INNER H	OUTER V	OUTER H
250	2	2.00	18.62	54.35	61
251	2	1.42	19.76	53.62	60
252	2	.69	20.92	52.96	58
37	2	-.38	22.17	52.54	57
38	2	-1.03	23.38	51.83	57
39	5	-.75	22.86	50.97	59
40	5	.10	23.99	51.71	61
256	2	.67	15.95	54.53	55
257	2	.49	16.39	54.81	55
258	2	.49	16.96	55.01	54
59	2	.66	17.57	55.27	54
60	2	1.19	18.16	55.71	54
61	5	2.32	18.54	56.49	55
62	5	4.18	18.56	57.49	55
63	5	5.29	18.28	59.49	54
64	7	9.83	18.51	56.28	51
65	6	8.40	17.53	59.01	50
66	6	7.11	18.60	61.62	47
82	6	6.52	18.98	62.80	46
262	4	2.86	10.02	51.37	48
263	4	3.33	10.00	52.29	47
264	4	3.65	9.86	53.51	47
89	4	4.29	9.88	54.47	47
90	4	5.09	9.99	55.21	46.
91	112	5.97	10.16	55.68	46.
92	112	6.73	10.41	55.86	46.
93	112	7.27	10.76	55.88	46.
94	59	7.49	11.37	56.07	46.
95	59	7.53	11.42	56.61	45.
96	59	7.57	11.23	57.29	44.
97	59	7.66	10.97	57.85	43.

Sheet 44

LOAD COMBINATION 12

(1.0 ± .05) D + 1.5P + 1.0 (T + TL)

REBAR STRESS - CYLINDER UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	[illegible]		OUTER	
		[illegible]	H	V	H
211	113	6.41	20.81	49.15	58.44
212	113	6.99	20.79	49.84	58.67
213	113	7.10	20.99	50.26	58.59
202	113	7.45	21.21	50.55	58.40
203	113	7.98	21.38	50.76	58.16
206	115	9.78	21.75	50.74	56.93
207	114	10.09	22.06	50.77	56.26
208	114	10.30	22.04	50.97	55.70
209	114	10.43	21.94	51.20	55.04
210	114	10.48	21.83	51.41	54.59
217	2	5.83	20.15	47.93	60.24
218	2	5.78	20.73	47.81	59.49
217	2	5.83	20.15	47.93	60.24
218	2	5.78	20.73	47.81	59.49
219	2	5.63	21.34	47.68	59.15
172	2	5.63	22.00	47.66	59.10
173	2	5.97	22.62	47.81	59.27
174	5	6.92	23.00	48.27	59.53
175	5	8.59	22.98	48.98	59.45
176	5	9.85	22.61	50.50	58.39
177	60	13.30	22.81	48.49	55.82
178	6	12.45	22.17	50.73	54.61
179	6	11.41	23.02	53.11	52.24
195	6	10.93	23.33	54.30	51.09
223	2	5.74	18.88	49.18	62.01
224	2	5.22	19.92	48.49	60.36
225	2	4.51	21.01	47.80	58.81
150	2	3.48	22.19	47.27	57.58
151	2	2.7[illegible]	23.32	46 52	57.24
152	5	2.76	22.96	45.63	59.66
154	5	5.13	23.68	50.85	66.02
229	1	5.64	18.34	50.90	63.37
230	1	5.10	19.20	49.98	61.24
133	1	2.72	21.83	4[illegible].14	53.82
134	3	.64	22.36	45.86	51.93
135	69	1.81	26.08	39.67	49.71
237	1	4.24	19.22	50.06	59.57
114	1	4.13	19.22	48.07	57.23
115	1	4.30	19.14	45.03	54.43
116	3	4.78	19.10	41.92	51.46
153	5	3.46	23.91	46.11	61.66

LOAD COMBINATION 12

(1.0 ± .05) L + 1.5P + 1.0 (T + TL)

*LINER STRESS - BOSS (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
113	104	5.92	28.64
201	102	4.87	20.61
200	103	7.83	21.55
199	81	14.37	31.34
#198	73	17.89	33.32
#197	65	14.41	29.39
112	105	10.30	29.98
185	85	5.18	24.54
184	84	8.32	29.78
183	83	19.46	43.34
#182	74	15.82	28.05
#181	64	12.47	23.93
110	107	5.39	19.10
159	91	-6.95	20.38
158	90	-3.66	30.69
157	89	12.94	47.43
#156	76	-1.89	5.82
#155	62	9.52	16.88
108	109	6.97	17.71
140	97	-7.12	17.08
139	96	5.70	25.52
138	95	15.31	46.30
#137	78	-8.93	16.12
#136	70	-6.62	18.32
106	111	7.54	10.43
122	101	2.58	8.70
121	66	3.88	17.34
120	82	8.52	28.43
#119	80	.62	19.46
#118	72	3.94	17.30

*Correction factor of -8.15 ksi must be added to all liner stresses.

#For correction factor see sheet 47

LOAD COMBINATION 12

(1.0+.95) D+1.5P+1.0 (T+TL)

*LINER STRESSES (KSI) CYLINDER LOWER QUADRANT

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
250	2	-6.44	7.28
251	2	-6.45	8.98
252	2	-6.74	10.63
37	2	-7.46	12.21
38	2	-7.65	13.67
39	5	-7.50	12.47
40	5	-6.25	13.78
256	2	-8.90	4.56
257	2	-8.97	5.24
258	2	-8.76	6.08
59	2	-8.34	6.94
60	2	-7.52	7.86
61	5	-6.08	8.68
62	5	-3.84	9.36
63	[illegible]	-2.86	9.51
64	[illegible]	4.15	12.52
65	6	1.34	10.71
66	6	-.33	12.09
82	[illegible]	-1.14	12.56
262	4	-7.13	-.67
263	4	-6.67	-.20
264	4	-6.57	-.30
89	4	-5.91	-.04
90	4	-4.96	.42
91	112	-3.83	.99
92	112	-2.75	1.65
93	112	-1.90	2.34
94	59	-1.39	3.28
95	59	-1.39	3.49
96	59	-1.48	3.47
97	59	-1.53	3.35

*Correction factor of -11.4 must be added to all liner stresses.

LOAD COMBINATION 12

(1.0+.95) D+1.5P+1.0 (T+TL)

*LINER STRESS CYLINDER - UPPER QUADRANT (KSI)

PANEL NO.	PANEL TYPE	VERTICAL	HORIZONTAL
211	113	2.31	13.25
212	113	2.23	13.16
213	113	2.38	13.46
202	113	2.87	13.91
203	113	3.62	14.40
206	115	6.24	15.91
207	114	6.81	16.60
208	114	7.08	16.78
209	114	7.21	16.85
210	114	7.21	16.81
217	2	.76	11.59
218	2	1.00	12.53
219	2	1.12	13.39
172	2	1.38	14.28
173	2	2.04	15.20
174	5	3.33	15.98
175	5	5.37	16.59
176	5	6.62	16.75
177	60	11.90	19.14
178	6	10.10	18.08
179	6	8.71	19.22
195	6	8.02	19.65
223	2	-.27	9.34
224	2	-.27	10.97
225	2	-.50	12.57
150	2	1.19	14.08
151	2	-1.53	15.43
152	5	-1.61	14.43
153	5	-.57	15.46
154	5	.18	14.45
229	1	-1.11	8.13
230	1	-1.12	9.64
133	1	-1.81	14.27
134	3	-4.16	14.63
135	69	.43	21.03
237	1	-2.15	9.73
114	1	-1.67	10.38
115	1	-.60	11.22
116	3	.93	12.28

*Correction factor of -11.4 ksi must be added to all liner stresses.

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

J. O. NO. 9321.01
NAME OF COMPANY: CON. EDISON IPP #2
SHEET NO. 29 OF
DATE 8/7/69
SUBJECT: EQUIPMENT HATCH - ORDER OF MAGNITUDE CHECK
COMP BY [illegible] CH'D BY E.X.V.

FINE GRID MODEL PRESSURE LOAD (47 psi)

DATA TAKEN FROM COMPUTER OUTPUT

NODE #	LOAD KIPS	NODE #	LOAD KIPS
1	969.8	151	1492.8
14	1968.9	174	1497.9
27	2120.7	194	1496.2
40	2266.0	212	1926.3
53	2253.0	228	2357.0
66	1847.3	244	2329.0
79	1445.4	260	1990.6
94	1462.0	276	1539.3
111	1466.1	292	662.5
130	1484.4		

LEFT HAND SIDE OF MODEL

TOTAL = 32,511.7 K

$Pr = .047(837) = 39.4^{K/IN}$

HEIGHT OF FINE GRID = 836 IN

TOTAL = 39.4(836) = 32,900 K

NODE #	LOAD KIPS	NODE #	LOAD KIPS
13	-979.1	193	-1183.0
26	-1934.6	211	-1361.4
39	-1762.3	227	-1048.3
52	-1599.2	243	-2439.7
65	-1469.3	259	-2170.2
78	-1081.3	275	-1137.1
93	-2073.0	291	-1537.2
110	-2328.5	290	-1642.9
129	-1803.5	289	-1589.2
150	-1323.4	277	-1495.3
173	-1150.0	304	-695.7

RIGHT HAND SIDE OF MODEL

TOTAL = 33,804.2 K

32,511.7 K ~ 33,804.2 K

∴ ORDER OF MAGNITUDE CHECK O.K.

R-84

FORM 1095

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

J. O. NO. 9321.01

NAME OF COMPANY Con. Edison IPP #2

SHEET NO. 50 OF

DATE 8/7/69

SUBJECT Equipment Hatch - Stress Concentization Factor - Pressure Load (47 psi)

COMP BY EB C'K'D BY EB

From Computer Output:

Stress Resultant w/liner at Node #173 = 113.39 k/in

$Pr = 47(837) = 39.4\ ^{k}/in$

∴ Stress Concentization Factor = 2.88

$$\frac{113.39}{39.4} = 2.88$$

COMPARE 1.5P+.95DL WITH 1.0P+.95DL+1.0E' & 1.25P+.95DL+1.25E IN MEMBRANE & BOSS AREA TO DETERMINE IF 1.5P+.95 GOVERNS. APPLY GOVERNING STRESS RESULTANTS TO R/C ONLY DETERMINE STRESSES.

MEMBRANE AREA

PANEL #173 GIVES MAXIMUM RESULTS, OTHER PANELS WERE CHECKED TO VERIFY THIS

1.0P $N_{11} = 19.24^{k/in}$ $M_{11} = -202.33^{k\text{-}in/in}$ $N_{22} = 40.27^{k/in}$

$M_{22} = -243.55^{k\text{-}in/in}$

1.0 DL $N_{11} = -6.07^{k/in}$ $M_{11} = 43.04^{k\text{-}in/in}$ $N_{22} = -.10^{k/in}$

$M_{22} = 9.95^{in\text{-}k/in}$

1.0 SYM E.Q. $N_{11} = 2.23^{k/in}$ $M_{11} = -12.94^{k\text{-}in/in}$ $N_{22} = -.75^{k/in}$

$M_{22} = .18^{k\text{-}in/in}$

	N_{11}	M_{11}	N_{22}	M_{22}
1.0 P	19.24	-202.33	40.27	-243.55
1.25 P	24.05	-253.0	50.4	-304.0
1.5 P	28.85	-304.0	60.45	-365.[illegible]
.95 DL	-5.76	40.9	-.095	9.45
1.0 E'	3.34	-19.4	-1.125	.27
1.25 E	2.79	-16.20	-.937	.225

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY Con. Edison I.P.P. #2 | J. O. NO. 9321.c | SHEET NO. 52 | DATE 8/29/.. | COMP BY .. | C'K'D

SUBJECT Equipment Hatch. Stress Resultants

	N_{11}	M_{11}	N_{22}	M_{22}
1.5P + .95DL	23.09	-263.1	60.36	-355
1.0P + .95DL + 1.0E'	16.82	-180.83	39.05	-237
1.25P + .95DL + 1.25E	21.08	-228.3	48.34	-29

∴ IN MEMBRANE AREA 1.5P + .95DL GOVERN

BOSS AREA

PANEL #113 & #98 GIVE MAXIMUM STRESS RESULTAN[TS] PANELS WERE CHECKED TO VERIFY THIS. CHECK BOTH TO DETERMINE WHICH GOVERNS. REBAR IS ORTHOTRO[PIC] ∴ CONSIDER N_{22} & M_{22} ONLY.

PANEL # 113

1.0P	N_{22} = 113.39 K/IN	M_{22} = -857.84 INK/IN
1.0 DL	= 4.91 K/IN	= 44.62 INK/IN
1.0 Hyd. Eq.	= 6.48 K/IN	= -12.96 INK/IN

	N_{22}	M_{22}
1.0P	113.39	-857.84
1.25P	142.0	-1070.0
1.5P	170.0	-1287.0
.95DL	4.66	42.4
1.0E'	9.72	-19.41
1.25E	8.10	-16.20
1.5P + .95DL	174.66	-1244.6
1.0P + .95DL + 1.0E'	127.77	-834.85
1.25P + .95DL + 1.25E	154.76	-1043.

R-87

∴ 1.5P + .95DL GOVERN FOR PANEL #113 (BOSS

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY CON. EDISON I.P.P.#2 | J.O. NO. 9321.01 | SHEET NO. 53 OF
SUBJECT EQUIPMENT HATCH - STRESS RESULTANTS | DATE 8/19/69 | COMP BY | CH'K'D BY EX

PANEL #98

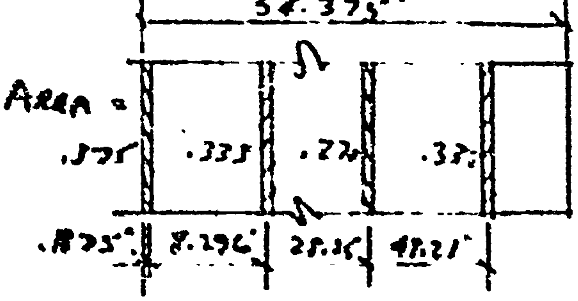
1.0P	N_{22} = 116.57 K/IN	M_{22} = -843.63 INK/IN
1.0 DL	= 5.15 K/IN	= 79.03 INK/IN
1.0 ASYM EQ.	= 6.62 K/IN	= -8.60 INK/IN

	N_{22}	M_{22}
1.0P	116.57	-843.63
1.25P	145.5	-1052.0
1.5P	175.0	-1264.0
.95 DL	4.89	75.0
1.0E'	9.92	-12.90
1.25E	8.26	-10.75
1.5P+.95 DL	179.89	-1189.0
1.0P+.95DL+1.0E'	131.38	-781.53
1.25P+.95DL+1.25E	158.65	-987.75

∴ 1.5P+.95 DL GOVERNS FOR PANEL # 98 (BOSS A

DETERMINING REBAR STRESSES IN MEMBRANE AREA

VERTICAL DIRECTION - CALCULATE C.G



A	$\bar{y}$	$A\bar{y}$
.375	.1875	.07
.333	8.296	2.76
.27	28.75	7.62
.33	48.21	16.02
1.311		26.47

$$\bar{y} = \frac{26.47}{1.311} = 20.2''$$

R-88

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

REVISED 4/2/7

NAME OF COMPANY CON. EDISON I.P.P #2

J. O. NO. 9321.01

SHEET NO 54 OF

SUBJECT EQUIPMENT HATCH - STRESS RESULTANTS

DATE 8/29/69

COMP BY ___ C'K'D BY E

VERTICAL REBAR STRESSES - MEMBRANE

REFERENCE SURFACE = 27.375"

∴ MOMENT ARM = 27.375 − 20.20 = 7.175"

N_{11} = 23.09 K/IN M_{11} = −263.1 IN-K/IN

∴ 23.09(7.175) = 165.0 IN-K/IN

∴ ACTUAL MOMENT ON SECTION = −263.1 + 165 = −98.1 IN

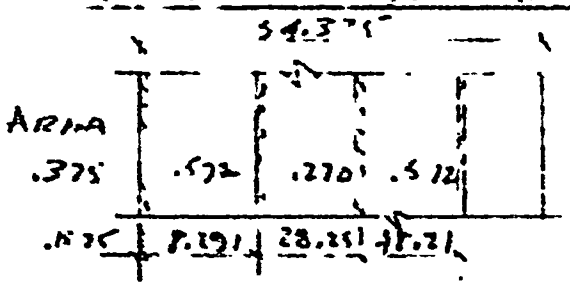
MOMENT ARM FOR VERTICAL STEEL = 98.21 − 8.291 = 39.91

$\frac{98.1}{39.919}$ = 2.46 K/IN   NET AREA - 1.311 − .375 = .936

σ = $\frac{23.09}{.936}$ ± $\frac{2.46}{.333}$ = 24.7 ± 7.4 = 32.1 & 17.3

σ_V - 32.1 KSI ϕf_y = 54.0 KSI   32.1 < 54.0 ∴ O.K.

HORIZONTAL REBAR STRESSES - MEMBRANE



A	$\bar{y}$	$A\bar{y}$
.375	.1875	.07
.572	8.291	4.74
.270	28.25	7.62
.572	48.21	27.50
1.789		39.93

$\bar{y}$ = $\frac{39.9}{.78}$ = 22.3

REFERENCE SURFACE = 27.375" ∴ M.A. = 27.375 − 22.30 = 5.0

N_{22} = 60.36 K/IN M_{22} = −355.55 IN-K/IN

∴ 60.36(5.075) = 306.0

∴ ACTUAL MOMENT ON SECTION = −355.55 + 306.0 = −49.5

$\frac{49.55}{39.919}$ = 1.241 K/IN   NET AREA - 1.789 − .375 = 1.414

R-89

σ = $\frac{60.36}{1.414}$ ± $\frac{1.241}{.572}$ = 42.6 ± 2.18 = 44.78 & 40.42

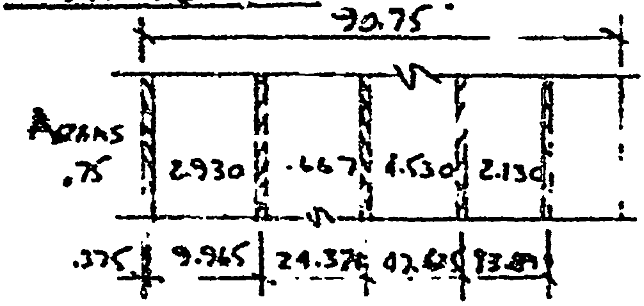
σ_H = 44.78 KSI ϕf_y = 54.0   44.78 < 54 ∴ O.K.

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY: CON. EDISON I.P.P. #2 | J. O. NO. 9321.01 | SHEET NO. 55 OF
SUBJECT: EQUIPMENT HATCH - STRESS RESULTANTS | DATE 8/29/69 | COMP BY RB | CH'K'D BY F.K.

DETERMINING REBAR STRESSES IN BOSS AREA

PANEL #113



A	$\bar{Y}$	$A\bar{Y}$
.75	.375	.281
2.930	9.965	29.200
.667	24.374	16.210
4.530	47.625	215.500
2.130	83.690	178.000
11.007		439.191

$$\bar{Y} = \frac{439.191}{11.007} = 39.9''$$

REF. SURFACE = 27.750"

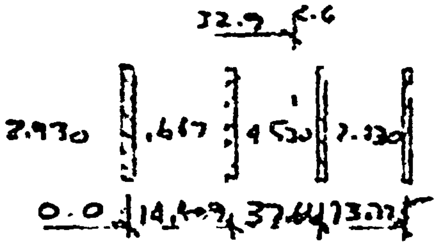
$\therefore$ M.A. = 39.9 − 27.750 = 12.150"

N_{22} = 174.66 ^{k}/IN M_{22} = −1244.6 $^{IN k}$/IN

$\therefore$ 174.66(12.150) = 2120. $^{IN k}$/IN

$\therefore$ ACTUAL MOM. ON SECTION = −2120. − 1244.6 = 3364.6

C.G. OF MOMENT RESISTING REBAR



A	$\bar{Y}$	$A\bar{Y}$
2.930	0	0
.667	14.409	9.62
4.530	37.66	170.50
2.130	73.725	157.00
10.257		337.12

$$\bar{Y} = \frac{337.}{10.2} = 32.9$$

$$\Sigma Ad^2 = 2.930(32.9)^2 + .667(18.491)^2 + 4.530(4.76)^2 + 2.130(40.575)^2$$
$$= 3160 + 227 + 102 + 3760$$
$$= 7249 \text{ IN}^4$$

$$\sigma_{B_T} = \frac{Mc}{Ad^2} = \frac{3364.6(32.9)}{7249} = 15.3 \text{ KSI}$$

$$\sigma_{B_C} = \frac{3364.6(40.575)}{7249} = -19.0 \text{ KSI}$$

NET AREA = 10.257

$$\sigma_m = \frac{174.66}{10.257} = 17.021$$

σ_{T_T} = 15.3 + 17.02 = <u>32.32 KSI</u> σ_{T_C} = −1.98 KSI ϕf_y = 54.0

32.32 < 54.0 $\therefore$ O.K.

FORM 301

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY Con. Edison I.P.P. #2 — J. O. NO. 9321.01 — SHEET NO. 56 OF — DATE 8/29/69

SUBJECT Equipment Hatch - Stress Resultants — COMP BY KLB — C'K'D BY F

CHECK PANEL #98 IN BOSS AREA (REBAR IS SAME AS PANEL — USE VALUES FROM #46

$N_{22} = 179.89$ K/IN $\quad M_{22} = -1189.0$ IN-K/IN

$\therefore 179.89(12.150) = -2180$ IN-K/IN

$\therefore$ TOTAL MOMENT $= -2180 - 1189 = -3369$ IN-K/IN

$\sigma_{B_T} = \dfrac{3369(32.9)}{7249} = 15.3$ KSI $\quad \sigma_{B_C} = -19.0$ KSI

$\sigma_m = \dfrac{179.89}{10.257} = 17.52$ KSI

$\sigma_{TT} = 15.3 + 17.52 = \underline{32.82 \text{ KSI} < 54.0 \text{ KSI}}$ O.K.

(BOSS AREA)

PANEL #56 STRESS RESULTANTS ARE SMALLER THEN PREVIOUS BUT REBAR AREAS ARE LESS THEN BEFORE.

1.0P	$N_{22} = 101.49$ K/IN	$M_{22} = -210.05$ IN-K/IN
1.0 DL	$= -4.45$	$= -106.58$
1.0 Asym. L.L.	$= 2.53$	$= 7.49$

	N_{22}	M_{22}
1.0P	101.49	−210.05
1.25P	127.0	−262.5
1.5P	152.2	−315.0
.95DL	−4.23	−101.0
1.0E'	3.8	11.22
1.25E	3.16	9.36
1.5P+.95 DL	147.83	−416.0
1.0P+.95DL+1.0E'	10[illegible]	−299.[illegible]
1.25P+.95DL+1.25E	125.[illegible]	−354.14

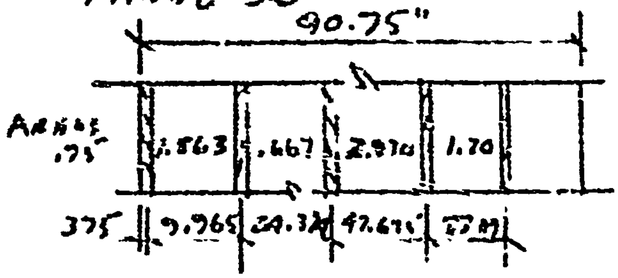
K-91

J. O. NO. 9321.01
SHEET NO. 57 OF
DATE 8/29/69
COMP BY AW C'K'D BY EJ

NAME OF COMPANY: CON. EDISON I.P.P. #2

SUBJECT: EQUIPMENT HATCH - STRESS RESULTANTS

PANEL #56



A	$\bar{y}$	$A\bar{y}$
.75	.375	.281
1.863	9.965	18.580
.667	24.374	16.210
2.930	47.625	139.500
1.200	83.500	100.200
7.410		274.771

REF. SURFACE = 27.750

$\bar{y} = \frac{274.771}{7.410} = 37.1"$

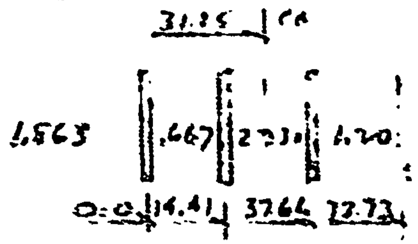
$\therefore$ N.A. = 37.1 - 27.750 = 9.350

$N_{22} = 147.2$ K/IN $M_{22} = -416.0$ IN-K/IN

$\therefore 147.2(9.350) = 1379.0$ IN-K/IN

$\therefore$ ACTUAL MOMENT ON SECTION = -1379.0 - 416 = -1795.0 IN-K

C.G. OF MOMENT RESISTING REBAR



A	$\bar{y}$	$A\bar{y}$
1.863	0	0.
.667	14.41	9.62
2.930	37.66	110.20
1.200	73.73	88.50
6.660		208.32

$\bar{y} = \frac{208.32}{6.66} = 31.25"$

$\Sigma Ad^2 = 1.863(\overline{31.25})^2 + .667(\overline{16.84})^2 + 2.93(\overline{6.41})^2 + 1.20(\overline{42.48})^2$

$= 1821 + 189 + 120 + 2160$

$= 4290$ IN⁴

$\sigma_{B_T} = \frac{Mc}{Ad^2} = \frac{1795(31.25)}{4290} = 13.1$ KSI NET AREA 6.60

$\sigma_{B_C} = \frac{1795(42.48)}{4290} = -17.75$ KSI $\sigma_m = \frac{147.2}{6.60} = 22.3$ KSI

$\sigma_{T_T} = 13.1 + 22.3 = \underline{35.4 \text{ KSI}}$ $\sigma_{T_C} = 4.55$ KSI $\phi f_y = 54.0$ K

$35.4 < 54.0$

FORM 801

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

J. O. NO. 7321.01

NAME OF COMPANY CON. EDISON IPP#2 SHEET NO. 58 OF

DATE 5/29/67

SUBJECT EQUIPMENT HATCH - STRESS RESULTANTS COMP BY [illegible] CH'K'D BY [illegible]

CONCLUSIONS:

PANEL # 56 IN THE LOWER QUADRANT OF THE BOSS ADJACENT TO THE TRANSITION SECTION WAS FOUND TO BE CRITICAL. STRESS RESULTANTS FROM THE GOVERNING 1.5P + .95 DL LOAD EQUATION WERE APPLIED TO THE R/C (NEGLECTING THE LINER) AND THE REBAR STRESS WAS 35.9 KSI WHICH IS BELOW THE ϕF_y CRITE[RIA].

OTHER PANELS ON THE VERTICAL ℄ OF THE HATCH WERE CHECKED AND THE STRESSES WERE BELOW ϕf_y. THE STRESS RESULTANTS OF THE REMAINING PANELS IN THE BOSS WERE COMPARED WITH THE STRESS RESULTANTS OF THE PANELS WHICH WERE CHECKED AND IN ALL CASES WERE LOWER.

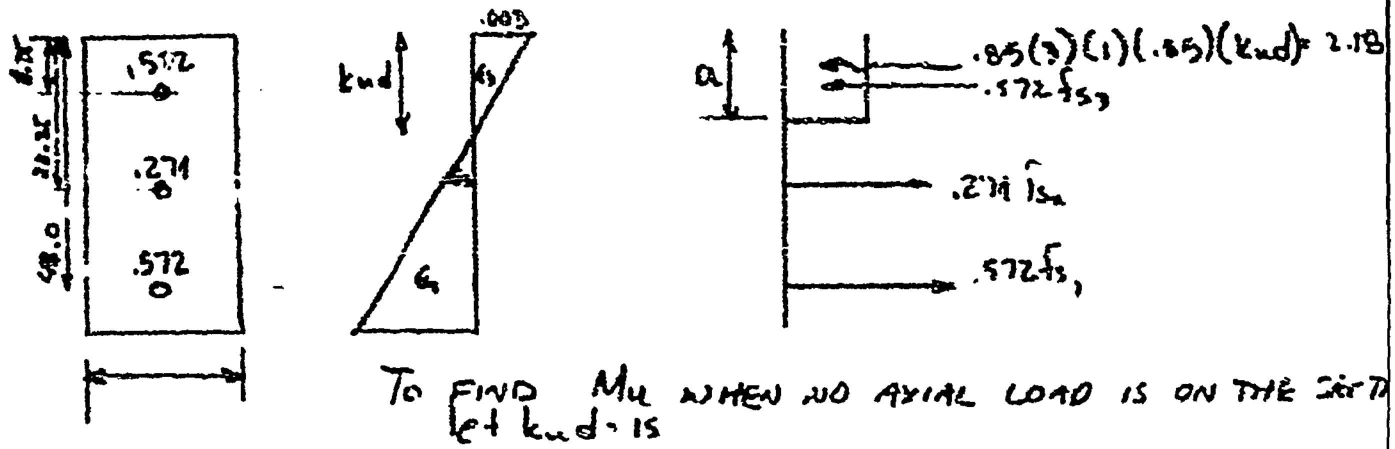
IN THE MEMBRANE AREA STRESS RESULTANTS WERE COMPARED AND THE MAXIMUM ONES USED. THEY OCCURRED IN PANEL # 173 LOCATED IN THE UPPER SECTION OF THE MODEL. THE CALCULATED STRESSES (NEGLECTING THE LIN[ER]) WERE 31.9 KSI VERTICAL AND 44.75 KSI HOOP. THESE [ARE] ALSO BELOW THE ϕF_y CRITERIA.

IT CAN BE CONCLUDED THAT UNDER 1.5P + .95 DL (NO TH[ERMAL] LOADING) AND NEGLECTING THE CONTRIBUTION OF THE LIN[ER] THE REBAR STRESS WILL NOT EXCEED ϕf_y.

R-93

Interaction Diagram

b) Horizontal



To find Mu when no axial load is on the sect.
let kud = 15

$\epsilon_1 = \frac{.144}{15} - .003 = .0096 - .003 = .0066 = 60\,ksi$

$\epsilon_2 = .00265 = 60\,ksi$

$\epsilon_3 = \frac{.003(15-8.75)}{15} = 36.2^{ksi}$

$\Sigma H = -2.18(15) - .572(36.2) + .271(60) + .572(60)$
$= -32.6 - 20.7 + 50.5$
$-53.3 + 50.5 = -2.8$

let kud = 12"
$f_{s1} = f_{s2} = 60\,KSI$

$\epsilon_3 = \frac{.003(12-8.75)}{12} = .00811 = 23.5^{ksi}$

$\Sigma = -2.18(12) - .572(23.5) + .271(60) + .572(60)$
$= -26 - 13.4 + 50.5$
$-39.4 + 50.5 = +11.1$

$\frac{X}{3} = \frac{2.8}{13.9} = .6$

kud = 14.4 in   a = 12.25 in

R-94

20004 R1

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

REVISED 9/2/6

J. O. NO. 9321-01

NAME OF COMPANY Con Ed - Indian Point #2

SHEET NO. 60 OF

DATE 8/3/69

SUBJECT Containment Structure

COMP BY BBS C'K'D BY F

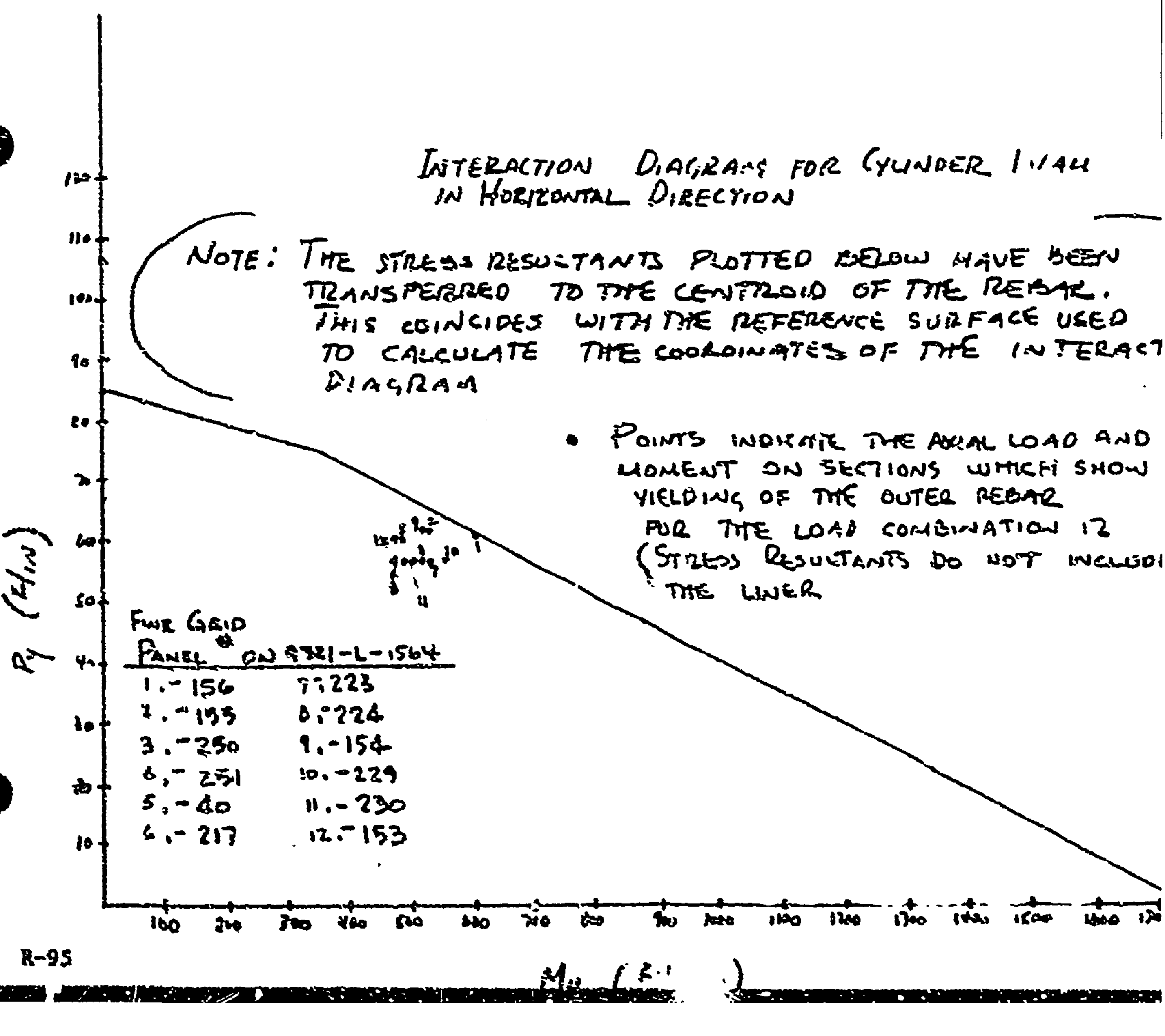
$$M_u = .572(60)(33.60) + .271(60)(13.85) + 2.18(14.4)(8.27) + .572(34.1)(5.65)$$

$$= 1152 + 225 + 359 + 110 = 17\ldots \text{ in-k/in}$$

A_s:

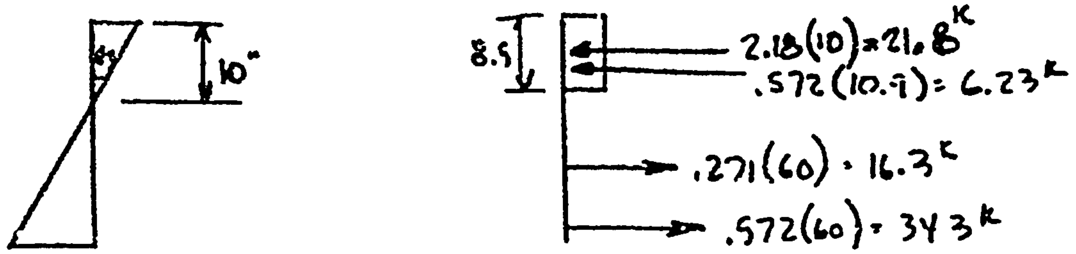
.572
12.70
.572
1.414 in^2

$P_y = 1.414\ in^2 \times 60\ k/in^2 = 85\ k/in$



FINE GRID PANEL # ON 9321-L-1564	
1. - 156	7. - 223
2. - 155	8. - 224
3. - 250	9. - 154
4. - 251	10. - 229
5. - 40	11. - 230
6. - 217	12. - 153

let $kud = 10''$

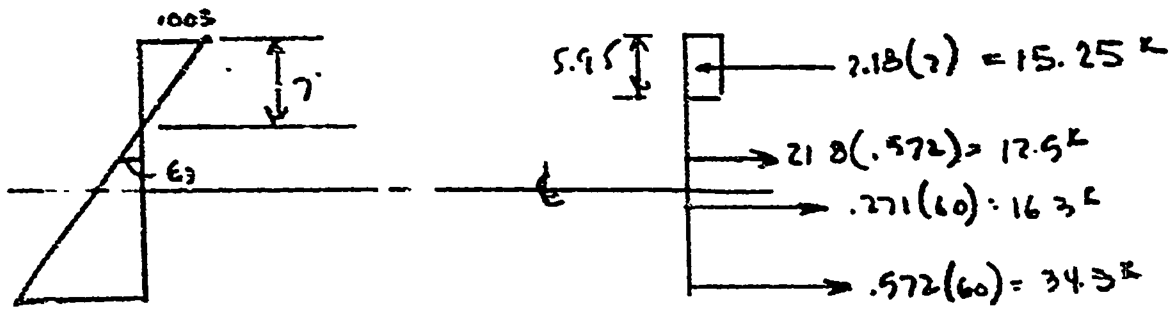


$$\epsilon_3 = \frac{.003(10-8.75)}{10} = .000375$$

$$P = 16.3+34.3-6.23-21.8$$
$$=22.6\ ^K/_{IN}$$

$$M_{℄} = 34.3(20.81)+16.3(1.06)+271(22.94)+6.23(18.44)$$
$$= 715+17.3+500+115$$
$$= 1347.3$$

let $kud = 7$



$$\frac{.003}{7} = \frac{\epsilon_3}{8.75-7} = .00075$$

$$P = 34.3+16.3+12.5-15.25 = 47.85^K$$

$$M_{℄} = 34.3(20.81)+16.3(1.06)+15.25(24.21)-12.5(18.44)$$
$$= 715+17.3+370-231$$
$$= 871\ ^{IN-K}/_{IN}$$

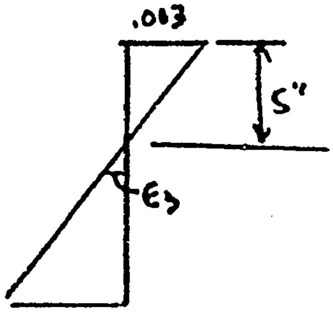
FORM 501

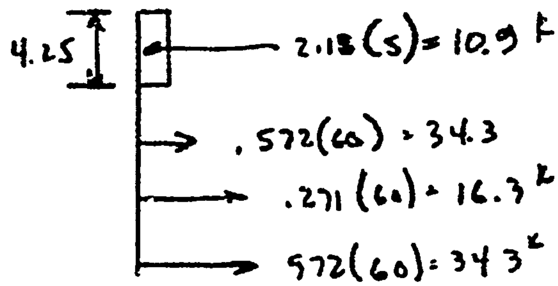
GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

REVISED 4/2/

J. O. NO. 9321-01
NAME OF COMPANY Con Ed - Indian Point #2
SHEET NO. 62 OF
DATE 8/8/69
SUBJECT Containment Structure
COMP BY BBS C'K'D BY F

let kud = 5





$$\frac{.003}{5} = \frac{\epsilon_3}{8.75-5} = .00225$$

P = 34.3 + 16.3 + 34.3 − 10.90 = 74.03 K

M_{CL} = 34.3(20.81) + 16.3(1.06) + 10.90(25.06) − 34.3(18.4

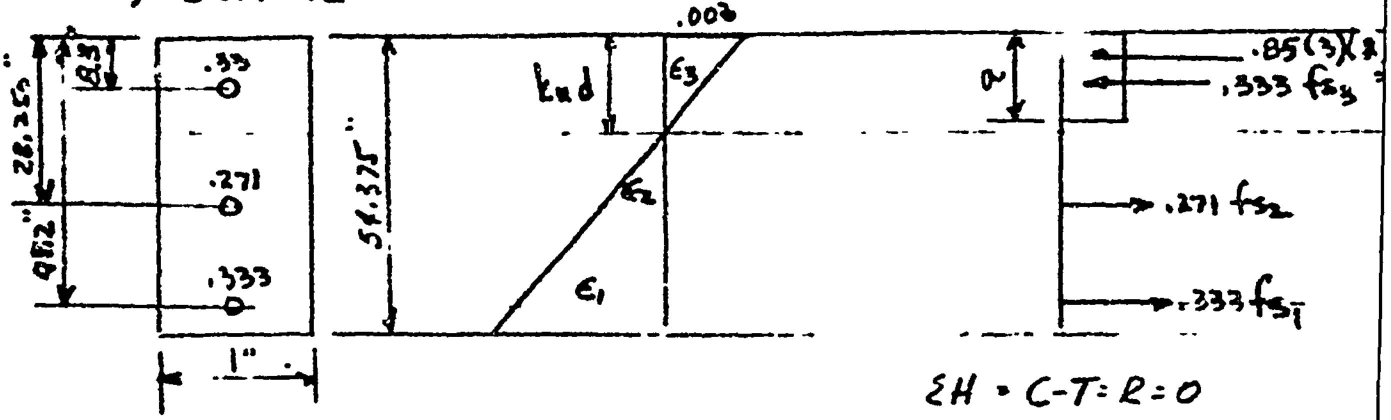
= 715 + 17.3 + 274 − 633

= 358.3 in-K/in

Interaction Diagrams

1. 4'-6 Cylinder $\quad \epsilon_y = \dfrac{60^{ksi}}{29 \times 10^3} = .00207$

a) Vertical



$\Sigma H = C - T = R = 0$

$$\frac{.003}{kud} = \frac{.003 + \epsilon_1}{d}$$

$$\epsilon_1 = \frac{.003d}{kud} - .003$$

$$\frac{.003}{kud} = \frac{\epsilon_2}{28.25 - kud}$$

$$\epsilon_2 = \frac{.003(28.25 - kud)}{kud}$$

$$\frac{.003}{kud} = \frac{\epsilon_3}{kud - 8.3}$$

$$\epsilon_3 = \frac{.003\,(kud - 8.3)}{kud}$$

let $kud = 15"$

$\epsilon_1 = \dfrac{.144}{15} - .003$

$.0096 - .003 = .0066$

$fs_1 = 60$ ksi

$\epsilon_2 = \dfrac{.003(28.25 - 15)}{15} = .002$

$fs_2 = 60^{ksi}$

$\epsilon_3 = \dfrac{.003(15 - 8.3)}{15} = .00134$

$fs_3 = 38.8^{ksi}$

ΣH

$2.17(15) + .333(38.8) = .271($

$32.6 + 12.9 = 36.2$

$-45.5 = 36.2$

$R = -\ 9.3$

try 12"

$fs_2 = fs_1 = 60^{ksi}$

$\epsilon_3 = \dfrac{.003(12 - 8.3)}{12} = .000925 = 26.8^{ksi}$

$R = (2.17)(12) + .333(26.8) - 36.2$

$26 + 8.9 - 36.2 = 1.3$

R-98

Assume Linear Interpolation $\quad \dfrac{X}{1.3} = \dfrac{3}{10.6} \quad X = .368$

$kud = 12.37"$ $\quad a = 10.5$

FORM 301

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

REVISED 9/8/69 4/2/70

J. O. NO. 1321-01
SHEET NO. 64 OF
DATE 8/5/69
COMP BY BBS C'K'D BY E

NAME OF COMPANY Con Ed - Indian Point #2

SUBJECT Containment Structure

$$M_u = .333(60)(35.83) + .271(60)(15.88) + 2.17(12.37)(7.12) + .333(28$$

$$M_u = 716 + 257 + 191 + 39 = 1203 \text{ IN-K/IN}$$

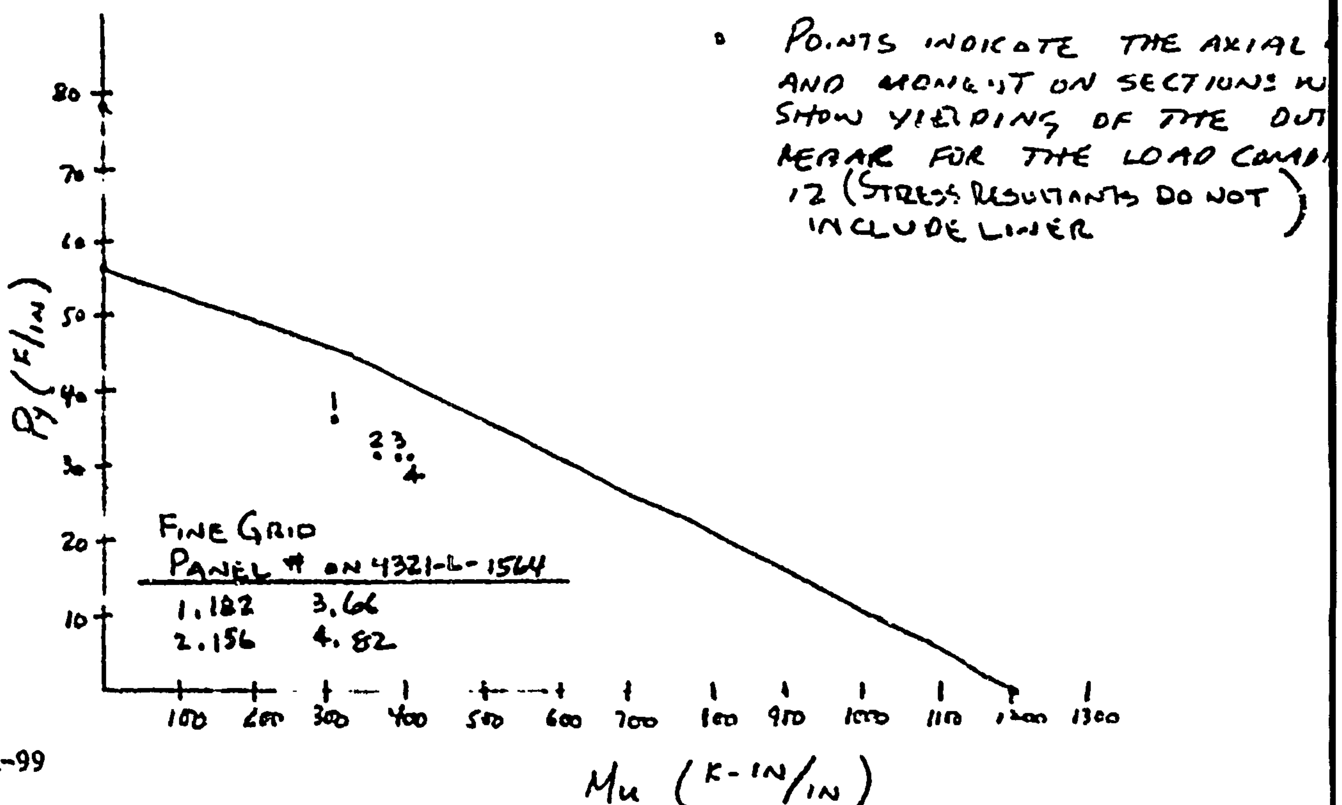
$A_s =$

$$\begin{array}{r} .333 \\ .270 \\ .333 \\ \hline .936 \text{ in}^2 \end{array}$$

$P_y = .936 \text{ in}^2 \times 60 \text{ K/in}^2 = 56.1$

NOTE: THE STRESS RESULTANTS PLOTTED BELOW HAV[E BEEN]
TRANSFERRED TO THE CENTROID OF THE REBA[R]
THIS COINCIDES WITH THE REFERENCE SURFACE U[SED]
TO CALCULATE THE COORDINATES OF THE INTERACTIO[N]
DIAGRAM

INTERACTION DIAGRAM FOR CYLINDER WALL
IN VERTICAL DIRECTION

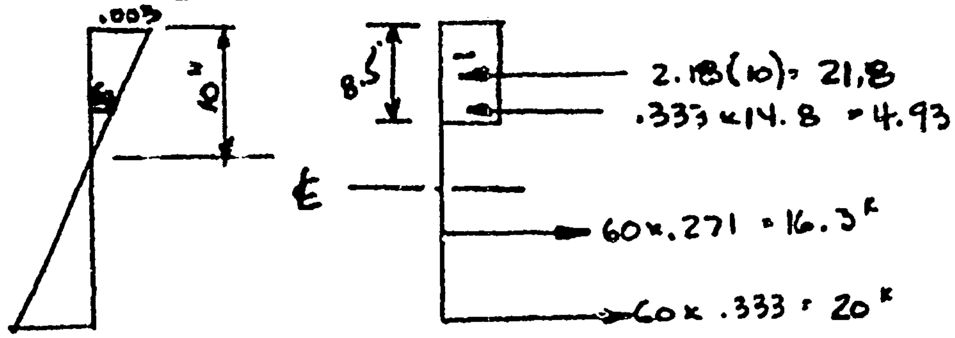


R-99

FORM 906 GENERAL COMPUTATION SHEET Revised 4/2/70
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY: ConEd - Indian Point #2
J. O. NO. 9321-01
SHEET NO. 65 OF
DATE 8/7/67
SUBJECT: Containment Structure
COMP. BY DBS C'K'D BY FK

Assume kud = 10"



$$\epsilon_3 = \frac{.003(10-8.3)}{10} = .00051$$

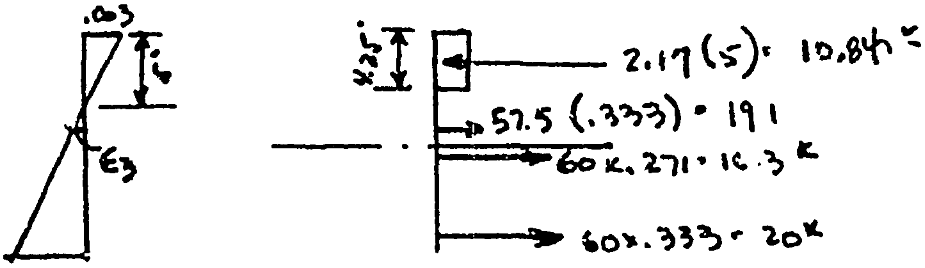
$P = 16.3 + 20 - 21.8 - 4.93 = 9.6^k$

$M_{℄} = 20(21.01) + 16.3(1.06) + 4.93(18.89) + 21.8(22.94)$

$= 420 + 17.3 + 93.2 + 498$

$= 1028.5$ in-k/in

Assume kud = 5"



$$\frac{.003}{5} = \frac{\epsilon_3}{(8.3-5)} = .00197 = 57.5$$

$P = 19.1 + 16.3 + 20 - 10.84 = 45.0$

$M_{℄} = 20(21.01) + 16.3(1.06) + 10.84(25.07) - 19.1(18.\ldots$

$= 420 + 17.3 + 272 - 361$

$= 348.3$ in-k/in

R-100

FORM 501

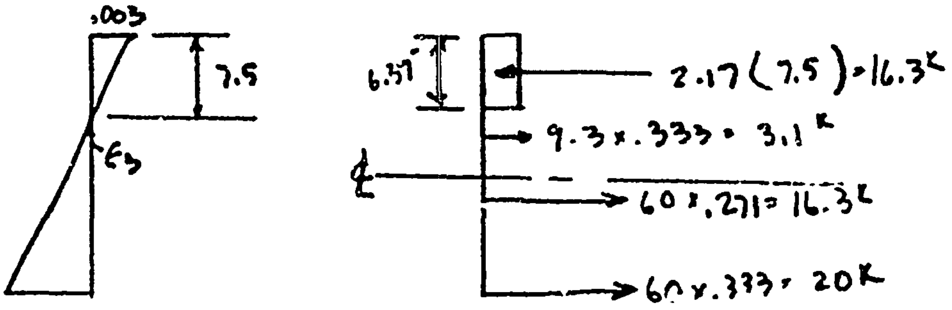
GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY Con Ed - Indian Point #2

SUBJECT Containment Structure

I. O. NO. 5321-01
SHEET NO. 66 OF
DATE 8/7/69
COMP. BY BBS C'K'D BY EK

Assume kud = 8.3 IN



$$\frac{.003}{7.5} = \frac{\epsilon_3}{8.3-7.5} = .00032$$

$P = 20 + 16.3 + 3.1 - 16.3$
$= 23.1$

$M_{℄} = 20(21.01) + 16.3(1.06) + 16.3(24.01) - 3.1(18.89)$
$= 420 + 17.3 + 392 - 58.5$
$= 770.8$ IN-K/IN

R-101

Check Load Equations to determine which governs (based on square root of sum of squares)

1. Load Combination 8

Panel #109 $\sqrt{(9.26)^2 + (-1.47)^2} = \underline{\underline{9.37}}$ K/in

2 Load Combination 9

Panel #106 $\sqrt{(6.18)^2 + (-.76)^2} = \underline{6.25}$ K/in

3. Load Combination 10

Panel # 106 $\sqrt{(-2.05)^2 + (10.02)^2} = \underline{\underline{10.23}}$ K/in

4 Load Combination 11

Panel # 106 $\sqrt{(-.76)^2 + (-7.81)^2} = \underline{\underline{7.84}}$ K/in

5. Load Combination 12

Panel # 106 $\sqrt{(1.90)^2 + (-10.00)^2} = \underline{10.05}$ K/in

6. 1.5 P + .95 D.L.

Panel # 107 $\overset{\text{Pressure}}{\sqrt{(6.12)^2 + (-5.67)^2}} = 8.35$ K/in

1.5 (8.35) - 12.50 K/in

D. L. = .95 (.77) = .73

13.23

$\underline{\underline{13.23}}$ K/in

Load Equation 6 governs

Check stirrups with 13.23 K/in over the entire boss

R-102

FORM #1

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

J. O. NO. 9321.01
NAME OF COMPANY: CON. EDISON IPP #2 SHEET NO. 68 OF
DATE 8/15/69
SUBJECT: EQUIPMENT HATCH - TRANSVERSE SHEAR STIRRUPS COMP BY RAB C'K'D BY E

FROM ACI 318-63 STIRRUPS #8 @ 6" c/c $A = .79$

$$V = \frac{A_v f_s d}{s} \qquad V = 13.23 \text{ k/in } (90 \text{ in}) = 1190 \text{ k}$$

$$d = 85$$

$$f_s = \frac{V s}{A_v d} = \frac{1190\,(6)}{2(.79)(90 - 3)} = 52 \text{ ksi}$$

(87)

$f_s = \underline{52 \text{ ksi}}$ $\quad f_y = 60 \text{ ksi}$

52 < 60 ∴ STIRRUPS WILL NOT YIELD DUE TO TRANSVERSE SHEAR

THIS SINGLE PANEL HAS BY FAR THE HIGHEST SHEAR STRESS

FORM 901

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

J. O. NO.

NAME OF COMPANY ______ SHEET NO. 69 OF ______

DATE

SUBJECT ______ COMP. BY ______ C'K'D BY EX

IN PLANE SHEAR STRESS RESULTANTS (INCLUDING AXISYM EARTHQUAKE (K/IN)

	1.0P	HORIZ. 1.0E'	.95D	VERT 1.0E'	THERMAL #1	TOTAL	AVE
113	-15.49	-6.97	-.01	0	-2.12	-24.59	
201	1.09	-7.00	1.68	-.34	2.08	-2.49	
200	-5.44	-6.95	1.47	-.30	3.73	-7.49	
199	-10.65	-7.75	.85	-.18	2.58	-15.15	-72.36/6 = -12.09 K/IN
112	-31.36	-9.10	1.18	-.24	3.22	-36.20	
185	-2.99	-4.99	3.04	-.62	5.84	.28	
184	-11.11	-6.92	2.79	-.56	9.36	-6.44	
183	-22.64	-8.13	2.78	-.58	2.22	-26.35	-101.50/6 = -16.92
111	-28.41	-9.40	1.37	-.27	-6.08	-42.79	
171	-5.36	-3.60	3.55	-.72	5.86	.27	
170	-10.40	-5.85	3.89	-.78	6.64	-6.50	
169	-27.03	-7.74	4.40	-.90	-3.53	-34.80	-125.12/6 = -20.85
110	-26.69	-9.44	1.80	-.35	-11.36	-46.04	
159	-9.51	-2.65	3.05	-.61	5.61	-4.11	
158	-10.44	-2.56	3.39	-.69	2.53	-7.77	
157	-29.83	-5.24	4.55	-.93	-10.78	-42.43	-150.55/6 = -25.09
109	-25.87	-9.25	2.05	-.40	-11.40	-44.87	
149	-14.51	-2.13	-.13	.03	5.89	-10.85	
148	-16.28	-1.96	-.08	.03	3.65	-14.64	
147	-34.92	-4.75	2.03	-.42	-10.04	-48.10	-181.20/6 = -30.20 K/IN
108	-24.88	-9.65	3.14	-.63	-6.35	-38.27	
140	-12.13	-2.96	-1.63	.34	6.42	-9.96	
139	-17.81	-5.71	-.37	.08	7.65	-16.16	
138	-31.97	-7.53	1.87	-.38	-2.35	-40.36	-161.27/6 = 26.88 K/IN
107	-24.78	-9.83	4.85	-.98	2.46	-28.28	
131	-8.46	-4.80	-1.79	.34	-6.32	-21.03	
130	-13.14	-7.15	.55	-.09	9.94	-9.89	
129	-21.62	-8.50	2.28	-.45	3.61	-24.68	-118.45/6 = -19.74 K/IN
106	-11.70	-7.42	2.42	-.48	-2.60	19.78	
122	-4.10	-7.35	-1.44	.29	2.52	-10.08	
121	-7.00	-7.36	.20	-.03	4.10	-10.09	
120	-7.67	-8.30	1.46	-.29	3.10	-11.60	-73.24/6 = 12.21 K/IN

FORM 501

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

REVISED 4/2/70

J. O. NO. ______

NAME OF COMPANY ______ SHEET NO. 70 OF ______

DATE ______

SUBJECT ______ COMP BY ______ C'K'D BY ______

IN PLANE SHEAR STRESS (INCLUDING AXISYMMETRIC LOAD (K/IN)

	1.25P	HOR 1.25E	.95D	VERT. 1.25E	THERMAL #2	TOTAL	AVE
113	-19.35	-5.82	-.01	0	-2.46	-27.64	
201	1.36	-5.83	1.68	-.21	2.46	-.54	
200	-6.80	-5.80	1.47	-.19	4.38	-6.44	
199	-13.30	-6.45	.85	-.11	3.06	-15.95	-73.96/6 = -12.33 k/in
112	-39.10	-7.58	1.18	-.15	3.76	-41.89	
185	-3.74	-4.16	3.04	-.39	6.87	+1.62	
184	-13.90	-5.78	2.79	-.35	10.98	-6.26	
183	-28.30	-6.77	2.78	-.36	2.72	-29.93	-112.65/6 = -18.78 k
111	-35.50	-7.84	1.37	-.17	-7.03	-49.17	
171	-6.70	-3.00	3.55	-.45	6.90	+.30	
170	-13.00	-4.88	3.69	-.49	7.84	-6.64	
169	-33.80	-6.45	4.40	-.56	-3.93	-40.34	-142.83/6 = -24.80 k
110	-33.40	-7.85	1.80	-.22	-13.16	-52.83	
159	-11.89	-2.21	3.05	-.38	6.60	-4.83	
158	-13.01	-2.14	3.39	-.43	3.02	-9.17	
157	-37.30	-4.37	4.55	-.58	-12.60	-50.30	-176.60/6 = -29.43 k/in
109	-32.40	-7.71	2.05	-.25	-13.20	-51.51	
149	-18.15	-1.77	-.13	.02	6.83	-13.20	
148	-20.30	-1.64	-.08	.02	4.22	-17.78	
147	-43.60	-3.96	2.03	-.26	-11.62	-57.41	-215.09/6 = -35.85 k/in
108	-31.10	-8.05	3.14	-.35	-7.17	-43.53	
140	-15.18	-2.46	-1.63	.21	7.41	-11.65	
139	-22.30	-4.76	-.37	.05	8.86	-18.52	
138	-40.00	-6.27	1.87	-.24	-2.70	-47.34	-186.90/6 = -31.20 k/in
107	-31.00	-8.18	4.85	-.61	3.00	-31.94	
131	-10.58	-4.00	-1.79	.23	7.30	-8.84	
130	-16.41	-5.89	.55	-.06	11.57	-10.24	
129	-27.00	-7.08	2.28	-.28	4.25	-27.83	-116.92/6 = -19.49 k/in
106	-14.60	-6.18	2.42	-.30	-2.93	-21.59	
122	-5.13	-6.13	-1.44	.18	2.92	-9.60	
121	-8.75	-6.16	.20	-.02	4.80	-9.93	
120	-9.59	-6.83	1.46	-.18	3.64	-11.50	-74.05/6 = -12.36 k/in

R-105

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC. REVISED 4/2/70

J. O. NO. ....

NAME OF COMPANY .... SHEET NO. 71 OF ....

DATE ....

SUBJECT .... COMP. BY .... C'K'D BY FKB

IN PLANE STRESS RESULTANTS (INCLUDING AXISYMMETRIC E.Q.)
(K/IN)

	1.5P	.95DL.	THERMAL 1E 3	TOTAL	AVE
113	-23.20	-.01	-2.70	-25.91	
201	1.63	1.68	2.64	5.95	
200	-8.15	1.47	4.68	-2.00	
199	-15.99	.85	3.29	-11.85	-47.66/6 = -7.94 K/IN
112	-46.80	1.18	3.94	-41.68	
185	-4.48	3.04	7.37	5.93	
184	-16.68	2.79	11.78	-2.11	
183	-34.00	2.78	2.90	-28.32	-96.61/6 = -16.10 K/IN
111	-42.60	1.37	-7.64	-48.87	
171	-8.04	3.55	7.39	2.90	
170	-15.60	3.89	8.41	-3.30	
169	-40.05	4.40	-4.27	-39.92	-132.41/6 = -22.07 K/IN
110	-40.00	1.80	-14.23	-52.43	
159	-14.30	3.05	7.07	-4.18	
158	-15.68	3.39	3.22	-9.07	
157	-44.80	4.55	-13.61	-53.86	-182.47/6 = -30.41 K/IN
109	-38.80	2.05	-14.28	-51.03	
149	-21.80	-.13	7.32	-14.61	
148	-24.20	-.08	4.53	-19.75	
147	-52.40	2.03	-12.54	-62.91	-230.96/6 = -38.49 K/IN
108	-37.30	3.14	-7.80	-41.96	
140	-18.20	-1.63	7.96	-11.87	
139	-26.70	-.37	9.50	-17.57	
138	-47.80	1.87	-2.96	-48.89	-186.75/6 = -31.13 K/IN
107	-37.10	4.85	3.12	-29.13	
131	-12.70	-1.79	7.85	-6.64	
130	-19.70	.55	12.40	-6.75	
129	-32.50	2.28	4.51	-25.71	-100.69/6 = -16.8 K/IN
106	-17.55	2.42	-3.22	-18.35	
122	-6.15	-1.44	3.12	-4.47	
121	-10.50	.20	5.14	-5.16	
120	-11.50	1.46	3.86	-6.18	-45.50/6 = -7.58 K/IN

IN PLANE SHEAR STRESSES

A) IN AREA WITH STIRRUPS RESISTING SHEAR

MAX V: 22.20 k/IN

A_s = #8 @ 6" STIRRUPS $= \frac{4(.79)}{6} = .525$ □"/IN

$$f_s = \frac{(22.65)(90)}{(.525)(85)} = \underline{\underline{45.8}} \text{ KSI} < 60 \text{ KSI} \therefore \text{OK}$$

B) IN AREA WITH STIRRUPS AND SEISMIC BARS RESISTING SHEAR

A_s = #8 @ 6" STIRRUPS = .525
#18 @ 6" @ 45° $= \frac{(4)(.707)}{6}$ = .471
.996 □"/IN

MAX V. = 38.49 k/IN

$$f_s = \frac{(38.49)(90)}{(.996)(85)} = \underline{40.8} \text{ KSI} < 60 \text{ KSI} \therefore \text{O}$$

SUMMARY OF NODAL TRANSVERSE FORCES (F3) AROUND THE EQUIPMENT HATCH (160°) FOR 47psi PRESSURE

		F3 KIPS
113	173	53.90
	172	-56.39
112	172	123.40
	171	-65.54
111	171	96.04
	170	-46.48
110	170	75.57
	169	- 6.97
109	169	38.48
	168	13.29
108	168	17.70
	167	9.75
107	167	24.53
	166	17.66
106	166	51.31
	165	56.07
105	165	51.67
	164	55.68
104	164	13.29
	187	29.19
103	187	5.09
	188	22.35
102	188	8.64
	189	41.91
101	189	-10.40
	190	77.94
100	190	-48.84
	191	98.29
99	191	-67.79
	192	126.50
98	192	-59.50
	193	53.90

800.25 KIPS x 2 = 1600.50

1600.5K ≈ 1634K

THERE IS GOOD CORRELATION BETWEEN THE COMPUTER RESULTS AND THE EXPECTED RESULTS

EXPECTED POPOUT SHEAR STRESS VALUES

AREA OF HATCH $= \pi (8)^2 = 201\ ft^2$

Total Load $= 47\ \#/in^2 \left(210 \times 144\ in^2\right) = 1422^k$

$\frac{1422}{2\pi 8} = 28.2\ ^k/FT$

$47\ \#/in^2 \times 7.5\ in \times 12\ in = 4.25\ ^k/FT$ (LOAD CALCULATED @ CENTROID OF 15" WIDE

TOTAL $= 28.2 + 4.25 = \underline{32.45}\ ^k/FT$

$32.45\ ^k/FT \times 2\pi(8) = \underline{\underline{1634}}\ ^k$

DESIGN OF RADIAL BARS TO RESIST POPOUT SHEAR

$f_s = 51\ ^{ksi}$ $\quad s = t/3 = 90/3 = 30"$

$A_v = \frac{Vs}{f_s d} = \frac{(1.5)(32.45^k)(30)}{(51)(87)} = .330\ \square"/FT$

USE #8 @ 12" $= .79\ \square" >> .330\ \square"/FT$

USING COMPUTER RESULTS

MAX SHEAR $= 13.23^k/in \times 12\ in/FT = 159\ ^k/FT$

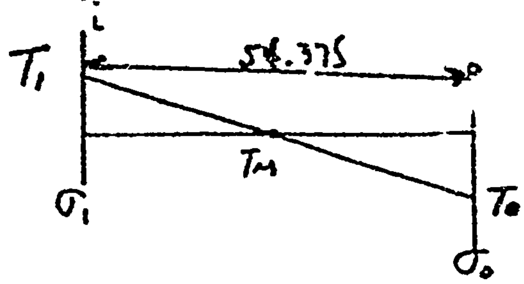
$A_v = \frac{(159)(30)}{(51)(87)} = 1.07\ \square"/FT$

USE #9 @ 12"

COMPUTER RESULTS ARE CONSERVATIVE USING THE SQUARE ROOT OF THE SUM OF THE SQUARES. THIS SHEAR OCCURS @ ONLY ONE POINT IN A QUADRANT OF THE BOSS AND IS SUBSTANTIALLY HIGHER THAN THE VALUES SHOWN IN THE HAND CALCULATI

Thermal Correction Factors

1 Cylinder - Load Combination 3



$T_i = 248°F$
$T_o = -50°F$
$T_M = 50.6337$
$\lambda = -5.7124$

$$T_1 = 50.6337 + \frac{1}{2}(54.375)(5.7124)$$
$$= 50.6337 + 155 = 205.6337\ °F$$

$$\sigma_i = \frac{(29\times10^3)(6.5\times10^{-6})(-248+205.6339)}{1-.3} = -11.4\ ksi$$

2 Cylinder - Load Combination 2

$T_1 = 227°F$
$T_o = -50°F$
$T_M = 47.43$
$\lambda = -5.24$

$$T_1 = 47.43 + \frac{1}{2}(54.375)(5.24) = 189.93°F$$

$$\sigma_1 = \frac{(29\times10^3)(6.5\times10^{-6})(-227+189.93)}{.7} = -10\ ksi$$

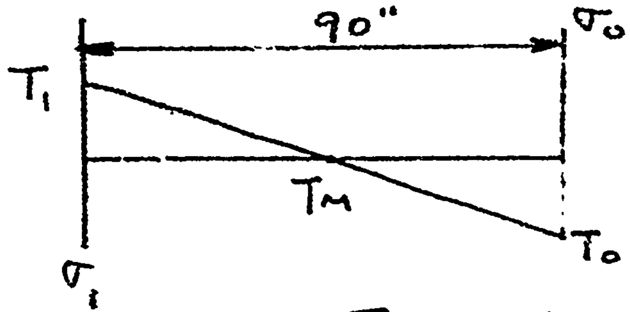
3 Cylinder - Load Combination 3

$T_1 = 188°F$
$T_o = -50°F$
$T_M = 41.18$
$\lambda = -4.38$

$$T_1 = 41.18 + \frac{1}{2}(54.375)(4.38) = 160.18$$

$$\sigma = \frac{(29\times10^3)(6.5\times10^{-6})(-188+160.18)}{.7} = -7.5\ ksi$$

4. Boss - Load Combination 3



$T_1 = 248°F$
$T_o = -50°F$
$T_M = 77.47$
$\Upsilon = -3.143$

$T_1 = 77.47 + \frac{1}{2}(90)(3.143) = 218.97° F$

$\sigma = \frac{(29 \times 10^3)(6.5 \times 10^{-6})(-248 + 218.97 °F)}{.7} = -8.15 \text{ KSI}$

5. Boss - Load Combination 2

$T_1 = 227°F$
$T_o = -50°F$
$T_M = 77.70$
$\Upsilon = -2.93$

$T_1 = 77.70 + \frac{1}{2}(90)(2.93) = 204.70 °F$

$\sigma = \frac{(29 \times 10^2)(6.5 \times 10^{-6})(-227 + 204.70)}{.7}$

$= -6.49 \text{ KSI}$

6. Boss - Load Combination 1

$T_1 = 188°F$
$T_o = -50°F$
$T_M = 63.84$
$\Upsilon = -2.54$

$T_1 = 63.84 + \frac{1}{2}(90)(-2.54)$
$= 178.34 °F$

$\sigma = \frac{(29 \times 10^3)(6.5 \times 10^{-6})(-188 + 178.34)}{.7} = -2.60 \text{ KSI}$

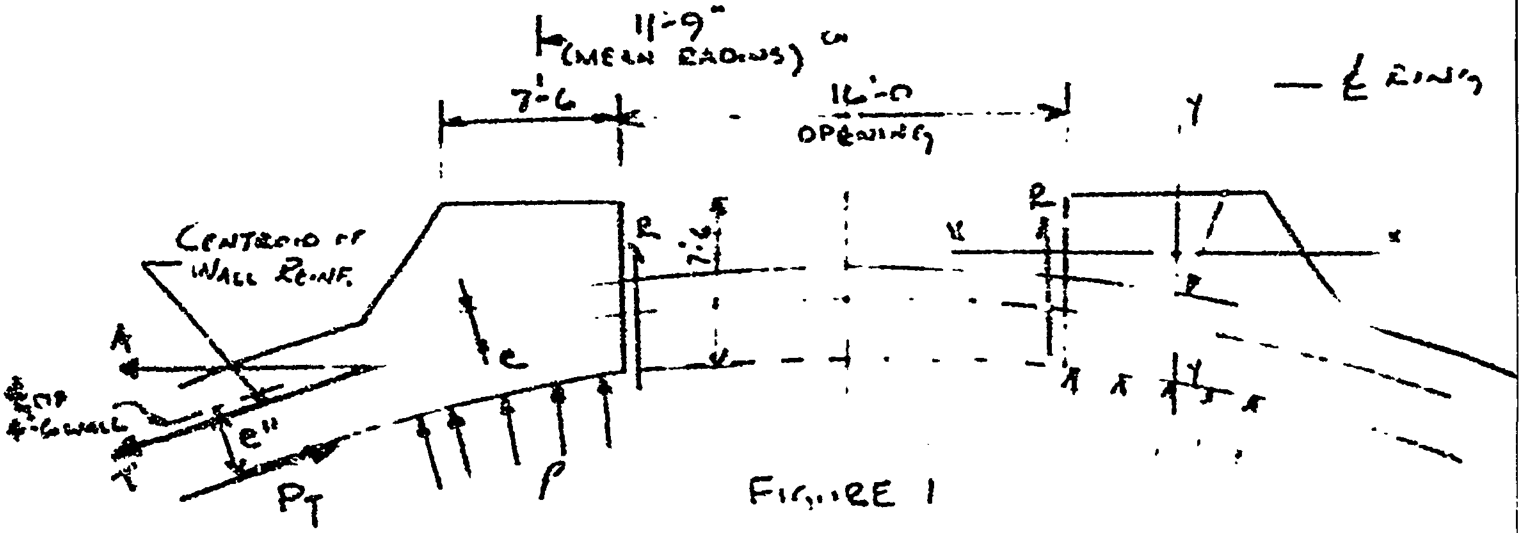


FIGURE 1

<u>1.5P CASE</u> (Bending about x-x Axis)

<u>Secondary Moments Caused by Torsion</u> (tendency of R & p to turn the ring inside out) SEE FIG 2

$M_z = M_x R_c$ From Pg 13l Eq (a) Strength of Materials Part II by S. Timoshenko

$M_x = R\,(3.75') + p\,\dfrac{(3.75')^2}{2}$

R is from the pressure of 70.5 psi acting over the hatch

$= \dfrac{70.5\,(\pi)(8)^2(144)}{\pi(16)} = 40.5\ ^{K}/_{FT}$

$M_x = 40.5\ ^{K}/_{FT} \times 3.75' + 70.5\ ^{\#}/_{in^2} \times 144\ ^{in^2}/_{ft^2} \times \dfrac{(3.75\ ft)^2}{2} = 223.5\ ^{FT}$

$M_z = 223.5\ ^{FT\text{-}K}/_{FT} \times 11.75\ FT = \underline{\underline{2620}}\ ^{FT\text{-}K}$ (tension on outside face)

<u>Secondary Moment Caused by Curvature of the Structure</u>
The moment comes from the eccentricity of the load applied by the circumferential loading and the vertical loading. e is the distance from the centroid of the ring to the centroid of the reinforcing including the liner

e for circumferential bars is $45" - \dfrac{8"}{12.5}$ (steel on outside face / " " inside ") x

$= 27.7$ in

e for vertical bars is $45 - \dfrac{4}{8.5} \times 27, 32.3$ in

R-112

FORM 381

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

J. O. NO.
NAME OF COMPANY
SHEET NO. 78 OF
DATE
SUBJECT CONTAINMENT - EQUIPMENT HATCH AREA
COMP. BY B.E.S. C'K'D BY

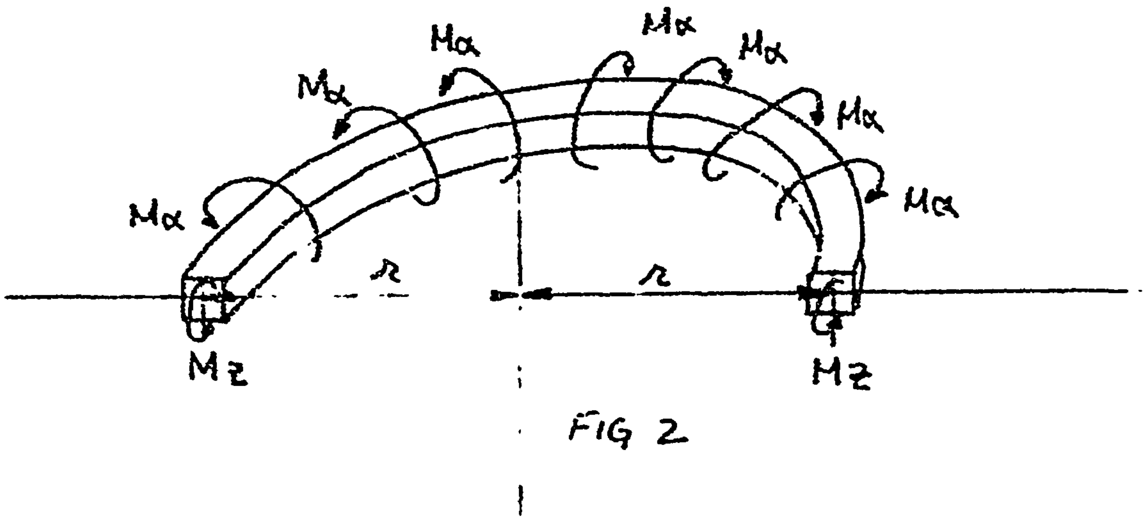
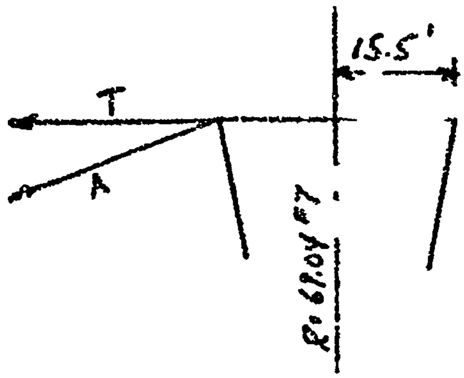


FIG 2

R-113



$\tan\theta = \frac{15.5}{69.04} = .2245$

$\theta = 12.66°$

$A \quad T \cos 12.66° = .976T$

USE $T = A$

Total Vertical Rebar force = 24.1 k/in² × 15.73 in²/FT 379 k/FT
" Hoop " = 35.4 k/in² × 21.43 in²/FT = 761 k/FT
The above values are from the containment design report for Point 5 which is close to the ℄ of hatch. The values for forces include the effects of D.L, L.L and Temperature effects

Circumferential Loading

Moment Per @ Pt. 10 (about x-x axis)

$761\ ^{k}/_{FT} \times \left(\frac{27.7}{12}\right)^{FT} \times 11.75^{FT} = \underline{\underline{20,650}}^{1k}$ (Tension on inside face)

The moment @ Pt 9 is equal to 0 (about x-x axis)

Vertical Loading
Moment @ Point 9

$379 \times \left(\frac{132.3}{12}\right) \times 11.75 = \underline{\underline{12,000}}^{1k}$ (Tension on inside face

Secondary Moment caused by Heating @ Liner

Use appropriate strains in the liner calculations for 440 sec. The total liner strain for vertical and horizontal bars are combined, considering Poisson's Ratio Effect to get the final liner stress. The appropriate strains are from the containment design report for Point 5 which is close to the Equipment Hatch Opening

$$\text{Vert Stress} = \left(\frac{-.520 - \frac{.180}{4}}{.9375}\right)E = 17.5 \text{ ksi}$$

From Elements of Strength of Materials by S. Timoshenko, & D.H. Young, P. 57 Eqts (3.6)

$$\text{Hor. Stress} = \left(\frac{-.180 + \frac{.520}{4}}{.9375}\right)E = 9.58 \text{ ksi}$$

Vertical Force = $17.5\ ^{K}/_{IN^2} \times .375^{IN} \times 12\ ^{IN}/_{FT} = 78.7\ ^{K}/_{FT}$

Horizontal Force = $9.58\ ^{K}/_{IN^2} \times .375 \times 12\ ^{IN}/_{FT} = 43.1\ ^{K}/_{FT}$

Circumferential Loading (P_T)

@ P_T 10 $\quad 43.1\ ^{K}/_{FT} \times 2.25\ FT \times 11.75\ FT = \underline{\underline{1139}}$ FT-K (tension on out.

Vertical Loading

@ P_T 9

$78.7\ ^{K}/_{FT} \times 2.25\ FT \times 11.75\ FT = \underline{\underline{208}}$ FT-K (tension on ou

Point 10 (Tension on outside +)

$M = 2620 - 20,650 + 1139 = 16,891$ FT-K (TOTAL ON SECTION

$$\frac{16,891^{FT-K} \times 12\ ^{IN}/_{FT}}{90^{IN}} = \underline{\underline{2250}}\ \frac{IN\text{-}K}{IN}$$ (Tension on in

Point 9 (Tension on outside +)

$M = 2620 - 12,000 + 2080 = 7,300$ FT-K (TOTAL ON SECTION)

$$\frac{7300 \times 12}{90} = \underline{\underline{973}}\ \frac{IN\text{-}K}{IN}$$ (Tension on inside)

Check stress in rebars. See Dwg 9321-F-1146

Point 9

Resisted by inside vertical bars $\#18 @ 6 = .667\ IN^2/IN$

$12 - \#18 = \left(\frac{12 \times 4}{90}\right) = .530\ IN^2/IN$

$1.197\ IN^2/IN$

$d = 85"$

$M = A_s f_s (d - \frac{a}{2})$

$$\frac{M}{A_s} = f_s \left(d - \frac{A_s f_s}{2(.85) f_c b}\right)$$

$$\frac{973}{1.197} = f_s \left(85 - \frac{1.197\, f_s}{2(.85)(3)}\right)$$

$813 = 85 f_s - .235\, f_s^2$

$fs^2 - 362 fs + 3460$

$$\frac{362 \pm \sqrt{362^2 - 13,840}}{2} = \frac{362 - 342}{2} = 10 \text{ ksi}$$

Add tensile effects from Pg 3 for all loads, with a stress concentration factor of 3 and assuming this load acts over the entire 90° section

$$\frac{379^k/FT}{12} = 31.6^k/in$$

A_s = 13 - #18 S hoops = 2.50 in²/in
26 - #18 S verticals = 1.15 in²/in
2 - #18 S E6 @ 75° seism.
$\frac{(2)(4)}{6} \sin^2 75° = 1.24$ in²/in
= 5.19 in²/in

$$\frac{31.6^k/in \times 3}{5.19 \text{ in}^2/in} = 18.3 \text{ KSI}$$

Add this to stress from bending to get

$18.3 + 10 = \boxed{28.3 \text{ ksi}}$

Point 10

Bending resisted by inside hoop bars 3-#18 E6" = i
12-#18
1.8

$$\frac{2250}{1.8} = fs\left(85 - \frac{1.86\, fs}{2(.85)(3)}\right)$$

$1210 = 85 fs - .365 fs^2$
$fs^2 - 232.5 fs + 3310$

$$\frac{232.5 \pm \sqrt{232.5^2 - 13,240}}{2} \qquad \frac{232.5 - 200}{2} = 16$$

Add tensile effects from Pg 3 for all loads with a stress concentration factor of 3 assuming this load acts over the entire 20" section

$$\frac{761^k/FT}{12} = 63.5^k/in$$

FORM 306

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

J. O. NO. 9-71-01

NAME OF COMPANY ____ SHEET NO. 82 OF ___

DATE ____

SUBJECT ____ COMP. BY B.B.S. C'K'D BY

As = 63 - #18 S hoops - 2.80 in²/in
52 - #18 S hoops - 2.30 in²/in
2 - #18 S B6" @ 75° Serm. - 1.24 in²/in

6.34 in²/in

$\frac{63.5 \times 3}{6.34 \text{ in}^2/\text{in}}$ k/in = 30 k/in²

Add this to the stress from bending to get

30 + 16.25 = $\boxed{46.25 \text{ k/in}^2}$

The maximum rebar stress occurs @ point 10 and is equal to 46.25 k/in². This considers bending moment due to all loads and tensile strength due to all loads. The values for loads away from the hatch are obtained from the Containment Design Report Table 3.3

FORM 102

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC. Revised 4/2/70

J. O. NO. 9321-01
SHEET NO. 23 OF
DATE
COMP BY BBS CH'K'D BY F

NAME OF COMPANY

SUBJECT Containment – Equipment Hatch Area

Bending About The Z axis in Figure I

In actuality the pressure and temperature will cause tension on the outside of the wall at the equipment hatch because the hatch will not allow the wall to rotate at this point

In order to investigate this moment as a bending on the section rather than a torsion, as requested in IPP-25 1/6, the pressure loads will be assumed to cause tension on the inside with the section rotating about the point where the thickened wall meets the 4'-6" wall. It is felt this is very conservative, The moment from the curvature and temperature will not rotate the section as they are taken in the hatch itself

Moment = 223.5 FT-K/FT = 223.5 IN-K/IN

This is resisted by the circumferential legs of the stirrups and the radial fill in bars

1 - #11 @ 5" radial fill in bars = .3175 in²/in
1 - #8 @ 6" main stirrup = .1315 in²/in
.4490 in²/in

$$\frac{223.5}{.4490} = f_s\left(85 - \frac{.4490 f_s}{5.1}\right)$$

$$497 = 85 f_s - .0871 f_s^2$$

$$f_s^2 - 967 f_s + 5650$$

$$\frac{967 \pm \sqrt{967^2 - 22{,}600}}{2} \qquad \frac{967 - 956}{2} = \underline{\underline{5.5 \text{ KSI}}}$$

The calculations indicate that this moment has a small effect on the stress in the stirrups

R-118

FORM 801

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

J. O. NO. 9321-01

NAME OF COMPANY ...... SHEET NO. R4 OF

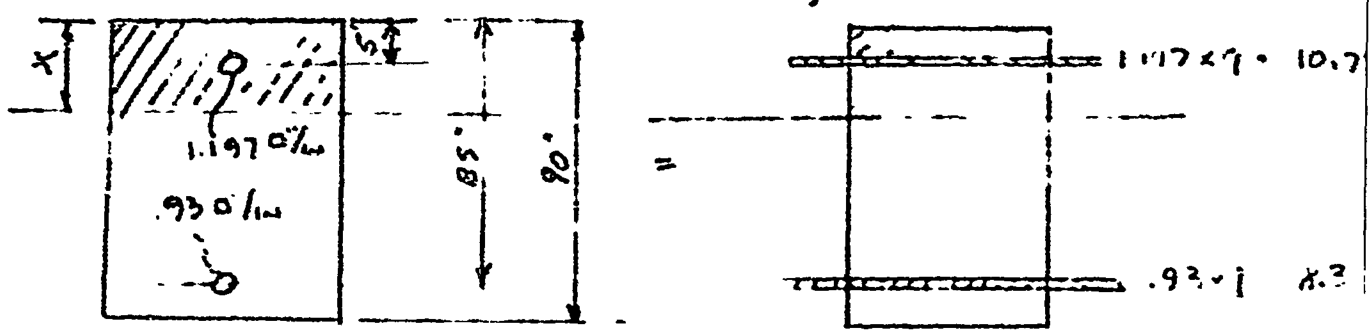
DATE

SUBJECT Containment - Equipment Hatch Area COMP BY RAS C'K'D BY

Check of the stress put back in main wall reinforcing from the rotation caused by the 223.5 FT-K/FT moment

$\theta = \frac{M_0 T^2}{E I_x}$ from Pg. 139 Eqt. 126 Strength of Materials Part II by S Timoshenko

$= M r / E I_x$ Consider the following section



The steel consists of the bars on the outside and inside face of the containment Plus hoops around the equipment hatch 12 additional on the inside 12 additional on the outside

To find N A (Moments above - Moments below)

$\frac{x^2}{2} + 10.75 (x - 5) = 8.37 (85 - x)$

$x^2/2 + 10.75x - 53.75 = 711 - 8.37x$

$x^2/2 + 19.12x - 764.75 = 0$

$x^2 + 38.24x - 1529.50 = 0$

$\frac{-38.24 \pm \sqrt{38.24^2 + 6118.0}}{2} = \frac{-38.24 + 86.9}{2} = \underline{24.35''}$

$I = \frac{1}{3}(24.35)^3 + 10.75(19.35)^2 + 8.37(60.65)^2$
$= 4810 + 4030 + 30800 = 39,640 \text{ in}^4/\text{in}$

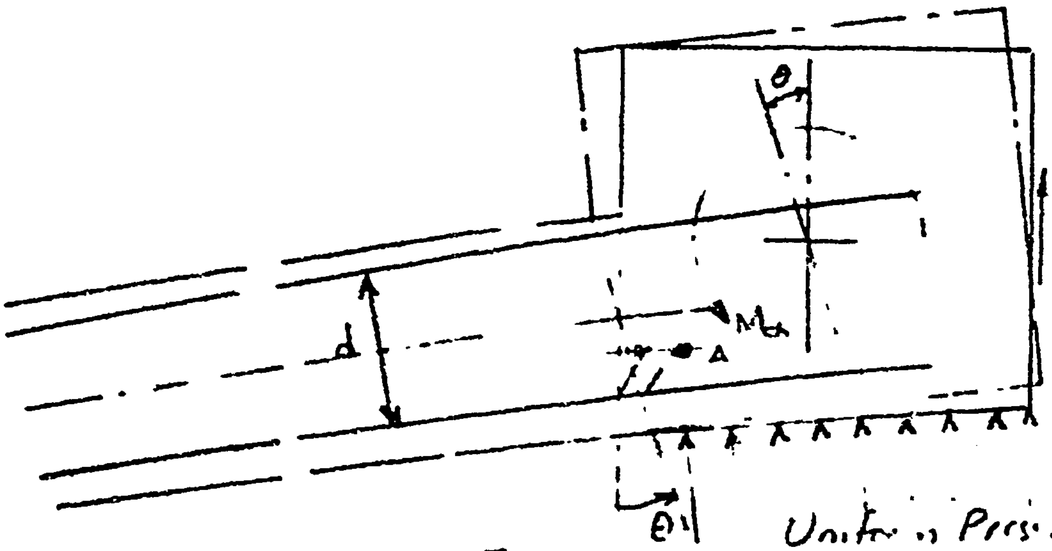
in 90" $= 39,640 \times 90 = 3,570,000 \text{ in}^4$

$\theta = \frac{(223.5^{FT-K/FT} \times 11.75^{FT})(12)(11.75 \times 12)}{(3.2 \times 10^3)(3,570,000)}$

$= \frac{2.235 \times 10^2 (1.175 \times 10^1)(1.2 \times 10^1)(1.175 \times 10^1)(1.2 \times 10^1)}{(3.2 \times 10^3)(3.57 \times 10^6)} = .3885 \times 1$

R-119

$\theta = .00039$



Uniform Pressure p

FIGURE 3

$$\tan\theta = \frac{\Delta}{d/2} = \frac{2\Delta}{40} = \frac{\Delta}{20} = .00039$$

$$\Delta = .0078''$$

Assume the elongation takes place over $\frac{d}{2} = 20''$ (very conservative)

$$\epsilon = \frac{.0078}{20} = .00039$$

Stress $= .00039 \times 29 \times 10^3 = 11.3$ k/in²

From Pg. 2 the hoop rebar stress @ this point = 35.4 ksi

$35.4 + 11.3 = 46.7^{ksi} < 54^{ksi}$ o.k.

Therefore the hoop bars have ample capacity to adjust to ring rotation and no bars will be added

Again it should be noted that the restraint from the equipment hatch has been ignored, therefore the stresses from the rotation above will be much lower.

FORM 001

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

J. O. NO. 9321-01
SHEET NO. 86 OF
DATE
COMP BY B.J. C'K'D BY E.N.

OF COMPANY Con Edison - Indian Point No 2

SUBJECT Containment - Personnel Lock Area

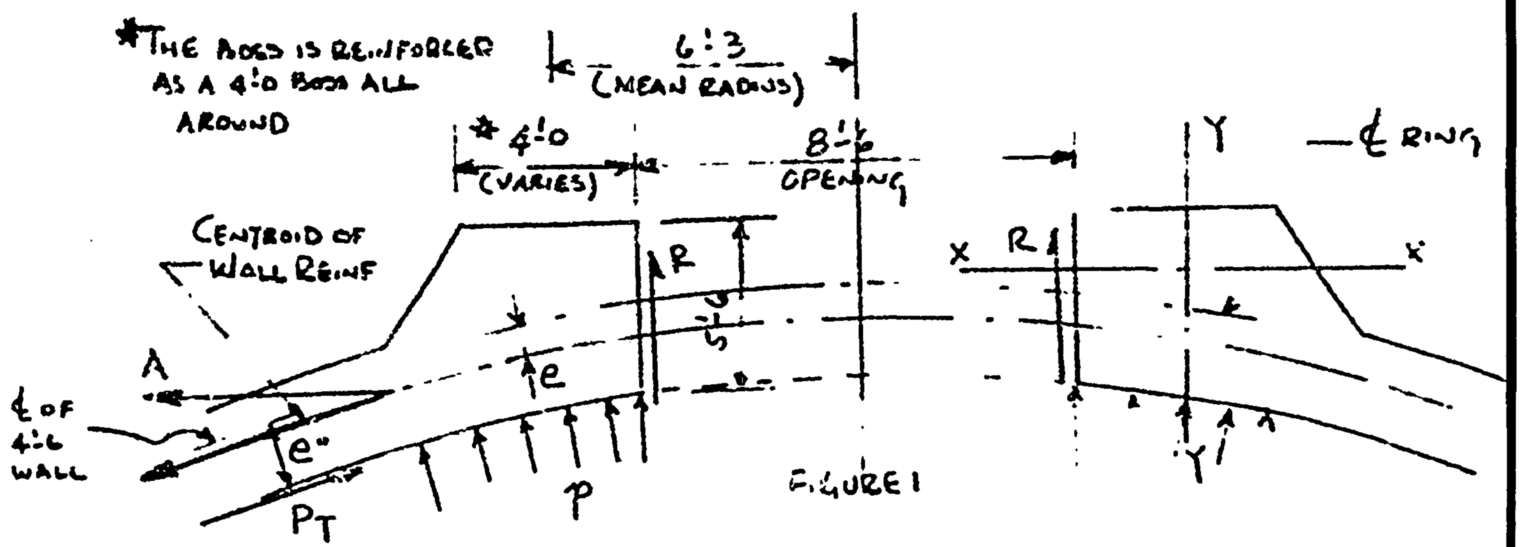


FIGURE 1

ISP CASE (Bending about x-x Axis)

Secondary Moments Caused by Torsion (tendency of R & p to turn the ring inside out) (SEE FIG 2 EQUIPMENT HATCH AREA CALCS)

$M_z = M_x\, r$ From P. 138 EQT (a) STRENGTH OF MATERIALS PART II by S. TIMOSHENKO

$M_x = R\,(2') + p\dfrac{(2)^2}{2}$

R is from the pressure of 70.5 psi acting over the hatch

$$\frac{70.5\,(\pi)\,(4.25)^2\,(144)}{\pi\,(8.5)} = 21.5\ ^{K}/_{FT}$$

$$M_x = 21.5 \times 2 + 70.5 \times 144 \times \frac{2^2}{2} = 63.3\ ^{FT\text{-}K}/_{FT}$$

$M_z = 63.3\ ^{FT\text{-}K}/_{FT} \times 6.25\ ^{FT} = \underline{395.5}\ ^{FT\text{-}K}$ (tension on outside

Secondary Moment Caused by Curvature of the Structure

The moment comes from the eccentricity of the load applied by the circumferential loading and the vertical loading. e is the distance from the centroid of the ring to the centroid of the reinforcing including the liner

e for circumferential bars is $33 - \dfrac{8}{12.5} \times 27 = 15.7$ IN

e for vertical bars is $33 - \dfrac{5.\,?}{10.\,?} \times 27 = 18.09$ IN

R-121

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

FORM 906

J. O. NO. ......
SHEET NO. 82 OF ......
DATE ......

NAME OF COMPANY ......

SUBJECT Containment - Personnel Lock Area   COMP. BY R.O.S. CH'KD BY EX

Total Vertical Rebar Stress = $17.8\,^{k}/_{in^2} \times 18.85\,^{in^2}/_{FT} = 335.5\,^{k}/_{FT}$
" Hoop = $35.4\,^{k}/_{in^2} \times 21.43\,^{in^2}/_{FT} = 761\,^{k}/_{FT}$

The above values are from the containment design report for Point 4 (Vertical) and Point 5 (horizontal - point 4 is still in the region affected by discontinuity and does not represent the lock area). The values for stress include effects of DL, LL & Temp

Circumferential Loading

Moment = Per @ Pt 10 (about x-x axis)

$761\,^{K}/_{FT} \times \left(\frac{15.7}{12}\right) \times 6.25\,^{FT} = \underline{\underline{6210}}\,^{FT-K}$ (Tension on inside face

The moment @ Pt 9 is equal to 0 (about x-x axis)

Vertical Loading

Moment @ Pt 9

$335 \times \frac{18.09}{12} \times 6.25 = \underline{\underline{3140}}\,^{FT-K}$ (Tension on inside face)

Secondary Moment caused by Heating @ Liner

Use appropriate strains in the liner calculations for 440 sec. The total liner strain for vertical and horizontal bars are combined considering Poisson's Ratio Effect to get the final liner stress. The appropriate strains are from the Containment Design Report for Point 4 which is closest to the Personnel Lock opening

$$\text{Vert Stress} = \left(\frac{-.767 - \frac{.400}{4}}{.9375}\right) E = 26.75 \text{ ksi}$$

$$\text{Hor Stress} \left(\frac{-.400 - \frac{.767}{4}}{.9375}\right) E = 18.20 \text{ ksi}$$

from Elements of Strength of Materials by S. Timoshenko & D H Young Pg. 57 Eqn (36)

R-122

FORM 163

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

REVISED 4/2/80

J. O. NO. ....
SHEET NO. 88 OF ....
NAME OF COMPANY ....
DATE ....
SUBJECT: CONTAINMENT – PERSONNEL LOCK AREA
COMP BY E.B.S. C'K'D BY [illegible]

Vertical Force = $26.75 \,^{K}/_{IN^2} \times .375 \,^{IN} \times 12 \,^{IN}/_{FT} = 120.2 \,^{K}/_{FT}$

Horizontal Force = $18.30 \,^{K}/_{IN^2} \times .375 \,IN \times 12 \,^{IN}/_{FT} = 82.5 \,^{K}/_{FT}$

Circumferential Loading (P_T)

@ PT 10 $82.5 \,^{K}/_{FT} \times 2.25 \,FT \times 6.25 \,FT = \underline{1160}$ FT-K (tension on o

Vertical Loading
@ PT 9 $120.2 \,^{K}/_{FT} \times 2.25 \,FT \times 6.25 \,FT = \underline{1690}$ FT-K (tension on out

Point 10 (Tension on outside +)

$M = 395.5 - 6210 + 1160 = 4654.5$ FT-K (TOTAL ON SECTION)

$\dfrac{4654.5 \text{ FT-K} \times 12 \,^{IN}/_{FT}}{48} = \underline{1163} \; \dfrac{IN\text{-}K}{IN}$ (Tension on inside)

Point 9 (Tension on outside +)

$M = 395.5 - 3140 + 1690 = 1054.5$ FT-K (TOTAL ON SECTION)

$\dfrac{1054.5 \times 12}{48} = \underline{263.6} \; \dfrac{IN\text{-}K}{IN}$ (Tension on inside

Check stress in rebars See Dwg 9321-F-1102

Point 9

Resisted by inside vertical bars #18 @ 6 = .667 in²/in
6 - #18 = $\left(\dfrac{6 \times 4}{48}\right) =$ = .500 in²/in
1.167 in²/in
d = 60 INCHES

$M = A_s f_s \left(d - \frac{a}{2}\right)$

$\dfrac{M}{A_s} = f_s \left(d - \dfrac{A_s f_s}{2(.85) f'_c b}\right)$

$\dfrac{263.6}{1.167} = f_s \left(60 - \dfrac{1.167 \, f_s}{2(.85)(3)}\right)$

$226 = 60 f_s - .2285 \, f_s^2$

$f_s^2 - 262 f_s + 988 =$

NOTE The resisting m[oment] is essentially t[he] same as for t[he] Equipment Hat[ch] Area.

R–123

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC. REVISED 6/2/70

J. O. NO.
NAME OF COMPANY ... SHEET NO. ... OF ...
DATE
SUBJECT Containment - Personnel Lock Area COMP BY J.D.S C'K'D BY

$$\frac{262 \pm \sqrt{262^2 - 3952}}{2} = \frac{262 - 254.5}{2} = 3.75\ ksi$$

Point 10

Resisted by inside hoop bars 2-#18SGG = 1.330 in²/in

8-#18 = $\frac{8 \times 4}{48}$ = .667

1.997 in²/in

$$\frac{1162}{1.997} = fs\left(60 - \frac{1.997\ fs}{5.1}\right)$$

$$584 = 60fs - .391 fs^2$$

$$fs^2 - 153.5fs + 1470 = 0$$

$$\frac{153.5 \pm \sqrt{153.5^2 - 5380}}{2} \qquad \frac{153.5 - 134.6}{2} = 9.45\ ksi$$

Point 9 Add tensile effects on pg 2 for all loads with a stress concentration factor of 3 assuming this load acts over the entire 48" section

$$\frac{335.5\ ^{k/FT}}{12} = 27.9\ ^{k/in}$$

As = 46-#18S hoops = 3.84 in²/in
12-#18S vert. = 1.00 in²/in
4.84 in²/in

$$\frac{27.9\ ^{k}/in \times 3}{4.84\ in^2/in} \qquad 17.3\ ksi$$

Add this to the stress from bending 17.3 + 3.75 =

$$\boxed{21.05\ ksi}$$

Point 10

Add tensile effects from pg 2 for all loads with a stress concentration factor of 3 assuming this load acts over the entire 48" section

$$\frac{761\ ^{k/FT}}{12} = 63.5\ ^{k/in}$$

As = 46-#18S hoops = 3.84 in²/in
24-#18S hoops = 2.00 in²/in
5.84 in²/in

FORM 201

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY ______ J. O. NO. ______ SHEET NO. 90 OF ___

SUBJECT: CONTAINMENT – PERSONNEL LOCK AREA DATE ______ COMP. BY ______ C'K'D BY ______

$$\frac{63.5 \times 3}{5.84} = 32.7 \ K/in^2$$

Add this to the stress from bending to get $32.7 + 9.45 = \boxed{42.15 \ K/in^2}$

There is a slight possibility that during an accident the inside door of the personnel lock could be open and the temperature and pressure effects from within cause additional load on the hoops closest to the lock

Assuming a ΔT of $306°F$

Stress $= \alpha \Delta T E = (6.5 \times 10^{-6})(306)(29 \times 10^3) = 57600 \times 10^{-3}$
$= 57.6 \ K/in^2$ compressive stress in the Lock

$57.6 \ K/in^2 \times (3/8 \times 5.5 \times 12) \ in^2 = 1428^K$

This must be resisted by the hoops in tension

$28 \text{-} \#18S = 28 \times 4 \quad 112 \ in^2$

$$\frac{1428^K}{112 \ in^2} = 12.7 \ KSI$$

Pressure Effects

$(12)(70.5)(59) = 50 \ K/FT$

$$A_s = \frac{(28 \times 4)}{5'\text{-}6''} = 20.35 \ in^2/FT$$

$$\frac{50 \ K/FT}{20.35 \ in^2/FT} = 2.46 \ K/in^2$$

$$\begin{array}{r} 12.70 \\ + \ 2.46 \\ \hline 15.16 \end{array} \ K/in^2$$

$\therefore$ the maximum stress @ Point 9 $21.05 + 15.16 = \boxed{36.21 \ KSI}$

the maximum stress @ Point 10 $15.16 + 42.15 = \boxed{57.31 \ KSI}$

R-125

FORM 101

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC. Revised 4/2/70

J. O. NO. ______

NAME OF COMPANY ______ SHEET NO. 11 OF ______

DATE ______

SUBJECT Containment - Personnel Lock Area COMP BY B.B.S. C'K'D BY F?

Bending About the z axis in Figure I

Moment = 63.3 FT-K/FT = 63.3 IN-K/IN

This is resisted by the circumferential legs of the stirrups and the radial fill in bars

1-#11 @ 5" radial fill in bars = .3175 in^2/in
1-#8 @ 6" main stirrup = .1315 in^2/in
.4490 in^2/in

$$\frac{63.3}{.4490} = f_s\left(60 - \frac{.4490 f_s}{5.1}\right)$$

Note: This resisting rebar is same as that used for the Equipment Hatch Area

$$141 = 60 f_s - .0879 f_s^2$$

$$f_s^2 - 683 f_s + 1603$$

$$\frac{683 \pm \sqrt{683^2 - 6412}}{2} \qquad \frac{683 - 676}{2} = \boxed{3.50 \text{ KSI}}$$

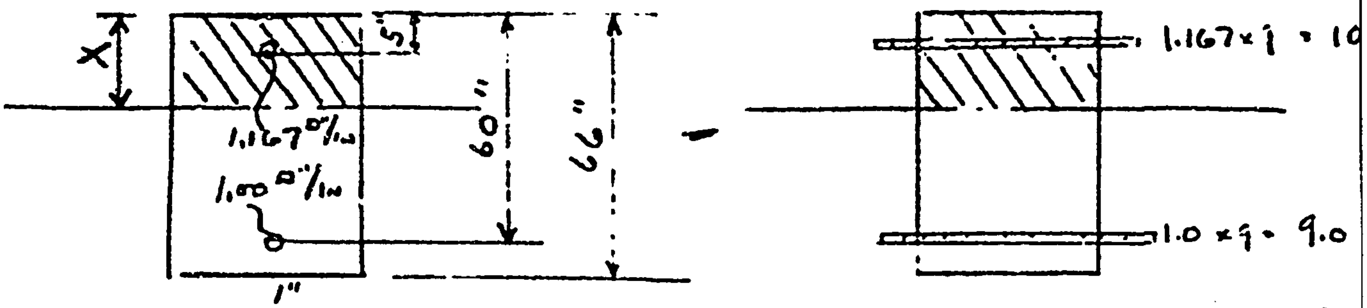
Check the stress out back in the main wall reinforcing from the rotation caused by the 63.3 in-k/in moment

$$\theta = \frac{M_x r^2}{E I_x} = \frac{M r}{E I_x}$$

From Pg 137 Eqt 126
Strength of Materials Part II
by S Timoshenko

R-126

Consider the following section



The steel consists of the bars on the outside and inside face of the containment plus hoops around the equipment hatch 6 additio on the inside & 6 additional on the outside. This is the same steel used in the previous design

To find N A. (Moments above = Moments below)

$$\frac{x^2}{2} + 10.50\,(x - 5) = 9.0\,(60 - x)$$

$$x^2/2 + 10.5x - 52.5 = 540 - 9x$$

$$x^2/2 + 19.5x - 592.5 = 0$$

$$x^2 + 39x - 1185 = 0$$

$$\frac{-39 \pm \sqrt{39^2 + 4740}}{2} \qquad \frac{-39 + 79}{2} = \underline{\underline{20}}\,''$$

$$I = 1/3\,(20)^3 + 10.50\,(15)^2 + 9\,(40)^2$$
$$= 2666.7 + 2360 + 14{,}400 = 19{,}426 \text{ in}^4/\text{in}$$

$$\text{in } 48'' = 19{,}426 \times 48 = 933{,}000 \text{ in}^4$$

$$\theta = \frac{(63.3^{\text{FT-K}}/\text{FT} \times 6.25)(12)(6.25 \times 12)}{(3.2 \times 10^3)(936{,}000)}$$

$$= \frac{(6.33 \times 10^1)(6.25)(12 \times 10^1)(6.25)(12 \times 10^1)}{(3.2 \times 10^3)(9.36 \times 10^5)} = 1.9 \times 10^{-4}$$

$$\theta = .00019$$

R-127

SEE FIG. 1 EQUIPMENT HATCH AREA

$$\tan\theta = \frac{\Delta}{d/2} = \frac{2\Delta}{40} = \frac{\Delta}{20} = .000119$$

$$\Delta = .00238$$

Assume the elongation takes place over $d/2 = 20''$ (very conservative)

$$\epsilon = \frac{.00238}{20} = .000119$$

$$\text{Stress} = .000119 \times 29 \times 10^3 = 3.45 \text{ K/in}^2$$

From Pg. 2 the hoop rebar stress @ this point = 28.1 K/in^2

$3.45 + 28.1 = 31.55 \text{ K/in}^2 < 54 \text{ KSI}$ ∴ O.K.

Therefore the hoop bars have ample capacity to adjust to ring rotation and no bars will be added.

It should be noted that the restraint from the Personnel lock has been ignored therefore the stresses from the rotation above will be much lower.

FORM 801

GENERAL COMPUTATION SHEET
UNITED ENGINEERS & CONSTRUCTORS INC.

NAME OF COMPANY: CON ED – INDIAN POINT #2 — J. O. NO. 9321-01 — SHEET NO. 94 OF ___

SUBJECT: CONTAINMENT STRUCTURE — DATE 4/2/70 — COMP. BY BDS — C'K'D BY ___

POPOUT SHEAR FOR PERSONNEL LOCK

AREA OF LOCK HEAD WHEN INSIDE DOOR IS CLOSED

$\pi(4.25)^2 = 56.7\ FT^2$

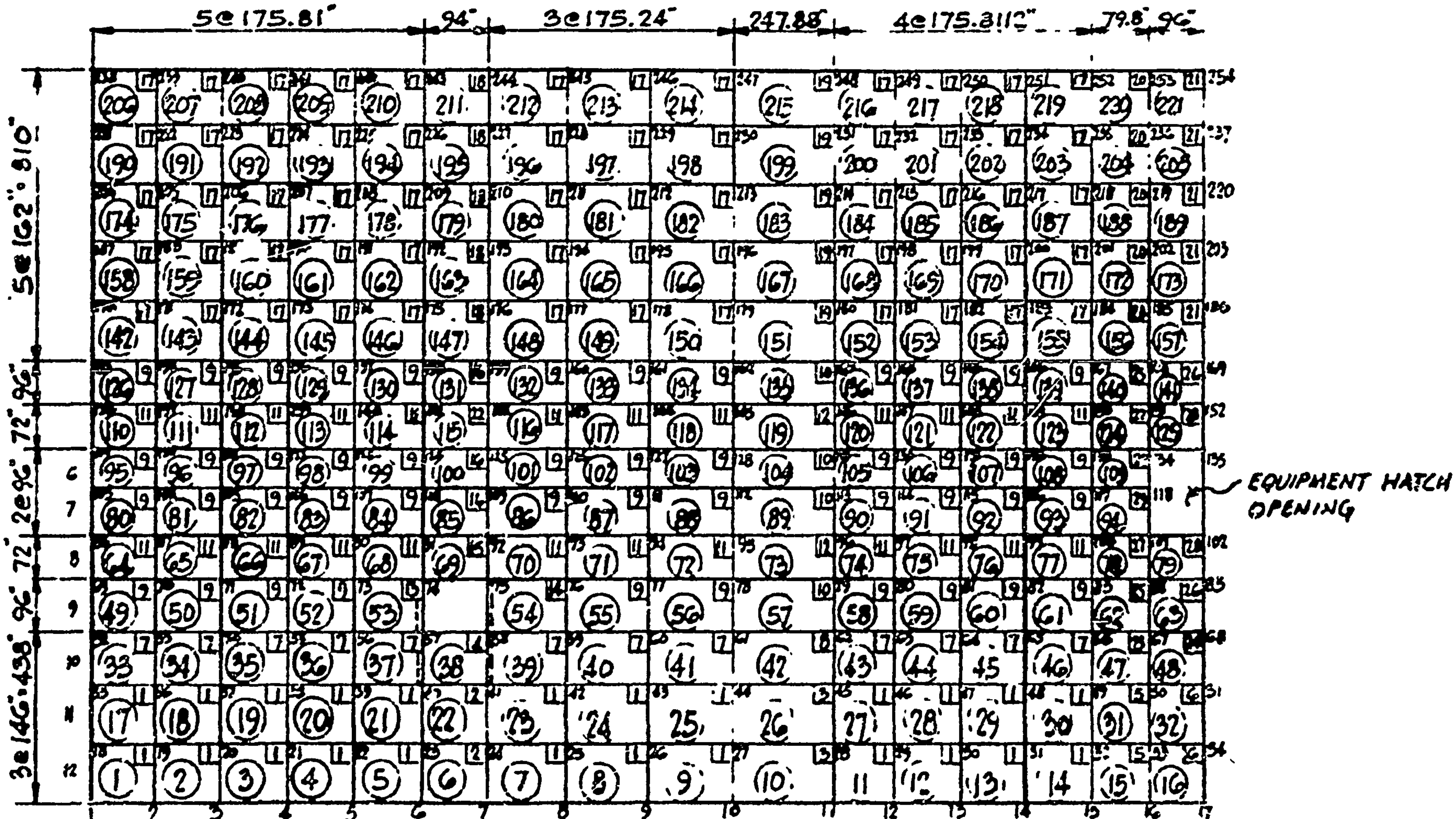
TOTAL LOAD $= 47^{\#}/IN^2\ (56.6^{7} \times 144) = 384^K$

$\frac{384^K}{2\pi(4.25)} = 14.35^K/FT$

THE AREA OF RADIAL BARS TO RESIST POPOUT SHEAR IN THE EQUIPMENT HATCH WAS DESIGNED BASED ON COMPUTER RESULTS. NO COMPUTER ANALYSIS WAS PERFORMED ON THE PERSONNEL LOCK THEREFORE A RATI[O] POPOUT SHEARS FOR PERSONNEL LOCK TO EQUIPMENT HATCH WILL BE USED AS A MULTIPLIER OF REQUIRE[D] EQUIPMENT HATCH STEEL TO DETERMINE THE AREA OF STEEL REQUIRED FOR THE PERSONNEL LOCK.

$\frac{14.35^K/FT}{28.20^K/FT\ (SHT^{\#}74)} \times 1.07\ \square''/FT\ (SHT^{\#}74) = .545\ \square''/FT$

USE #7 @ 12"



WESTINGHOUSE ELECTRIC CORPORATION

FOR

CONSOLIDATED EDISON COMPANY

INDIAN POINT GENERATING STATION - UNIT NO. 2

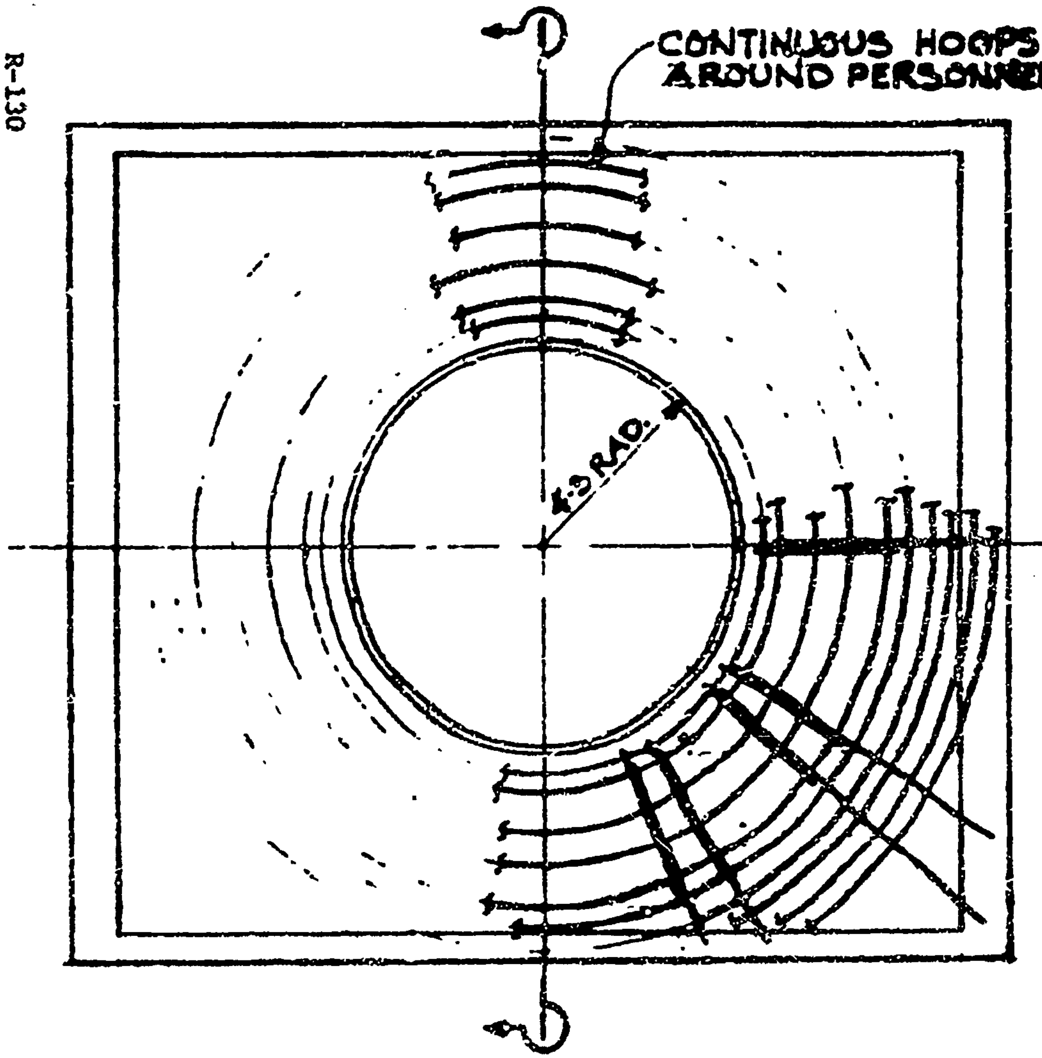
CONTAINMENT

FINITE ELEMENT MODEL

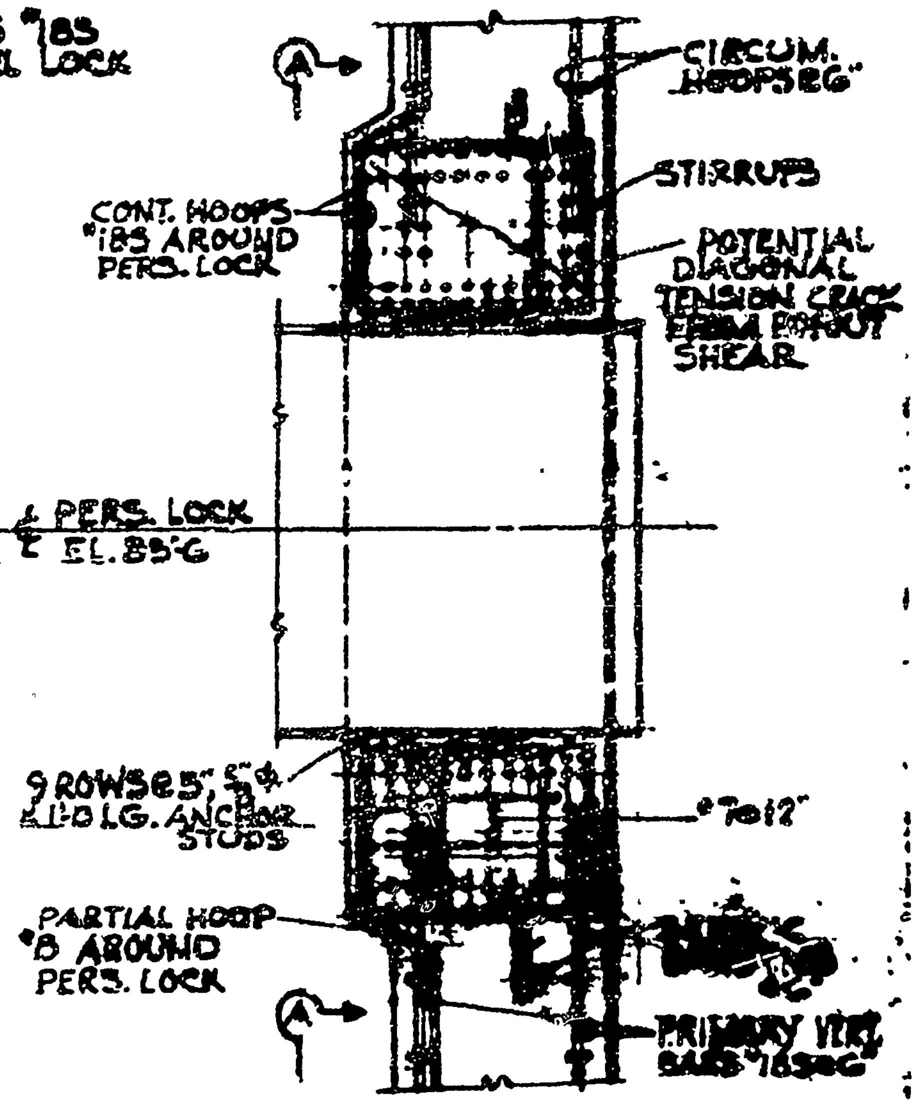
COARSE GRID ANALYSIS

UNITED ENGINEERS & CONSTRUCTORS INC.

R-130



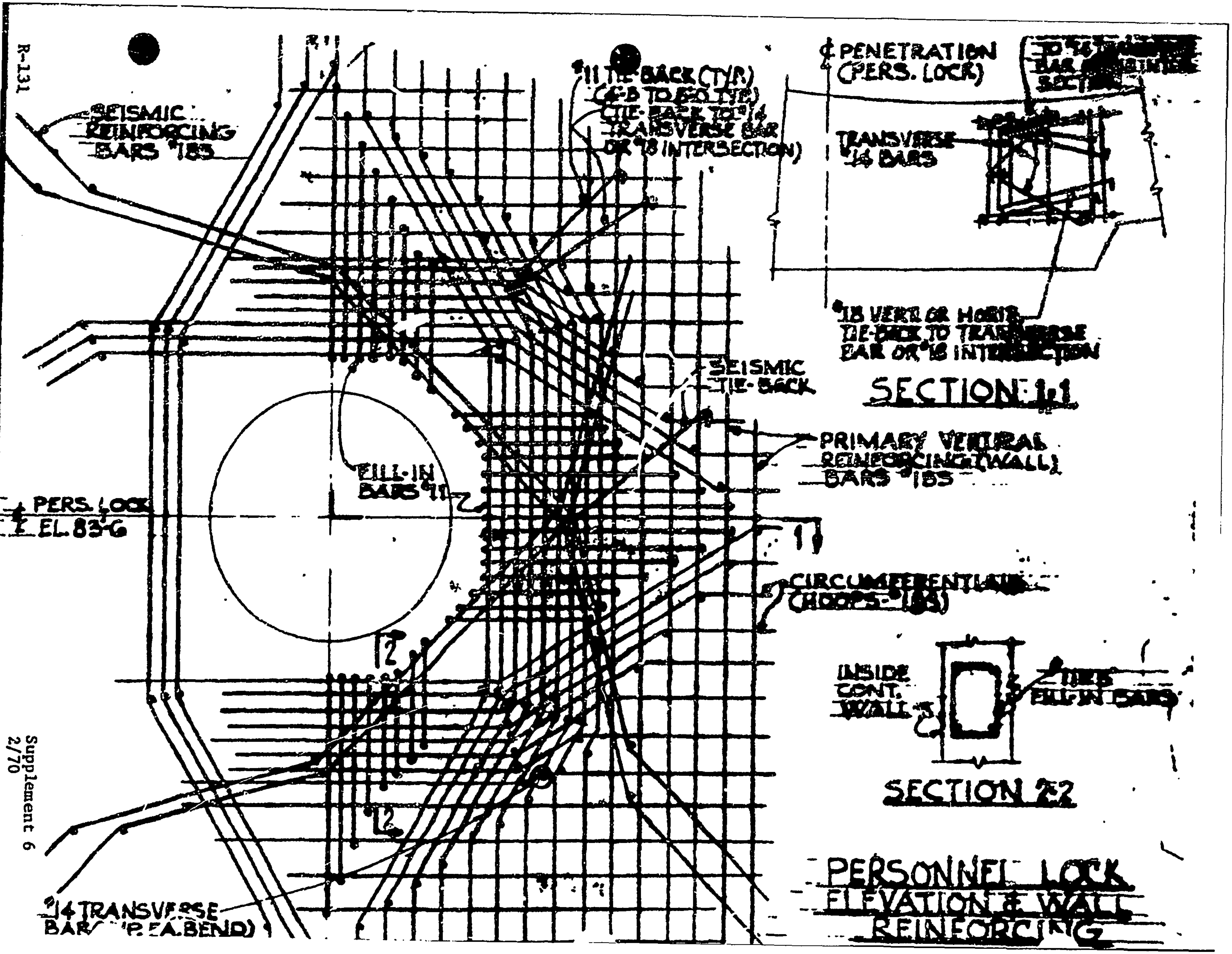
ELEVATION A-A



SECTION 1-1

PERSONNEL LOCK
PLAN & SECTION
BOSS REINFORCING DETAILS

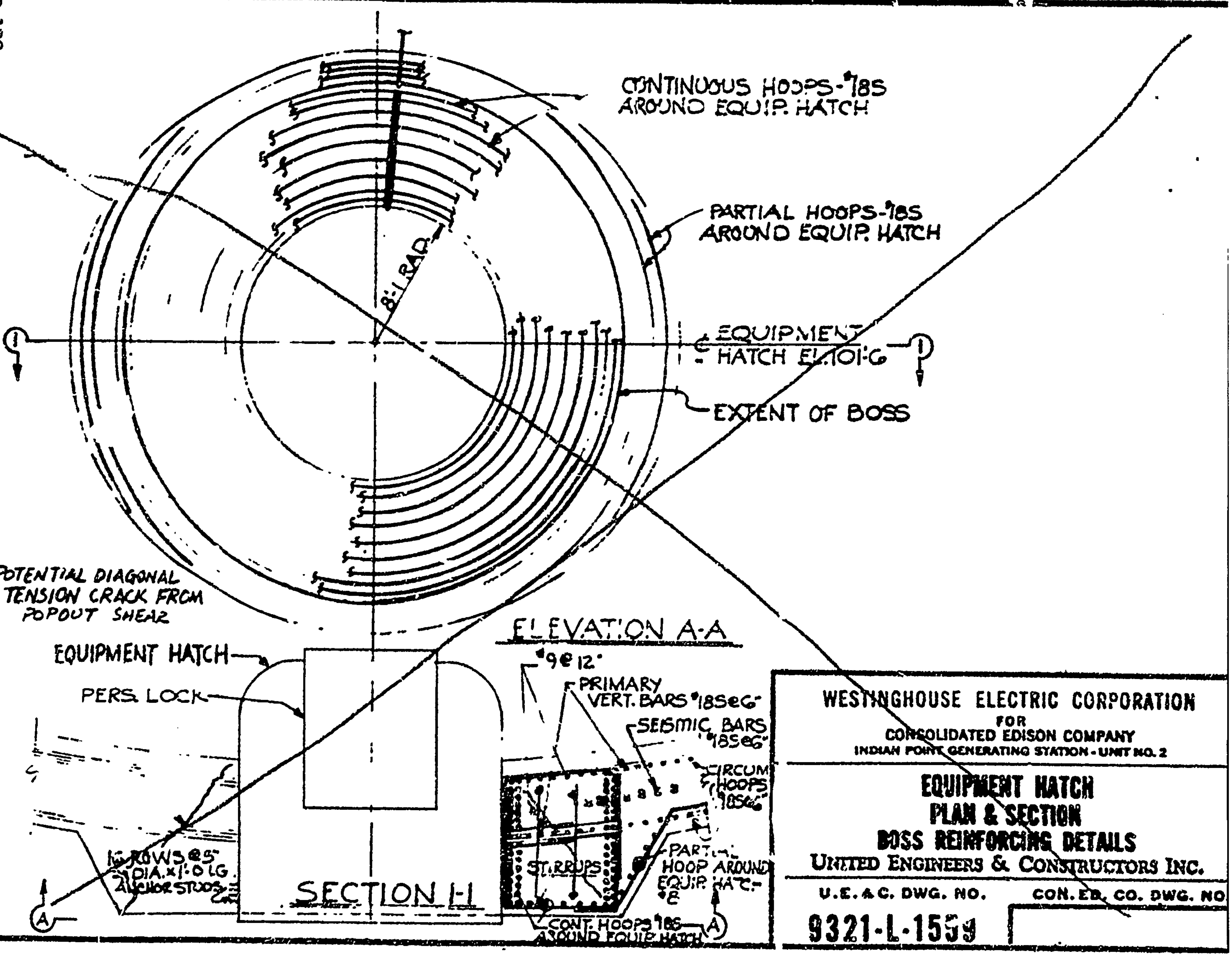
Supplement 6
2/70



PERSONNEL LOCK ELEVATION & WALL REINFORCING

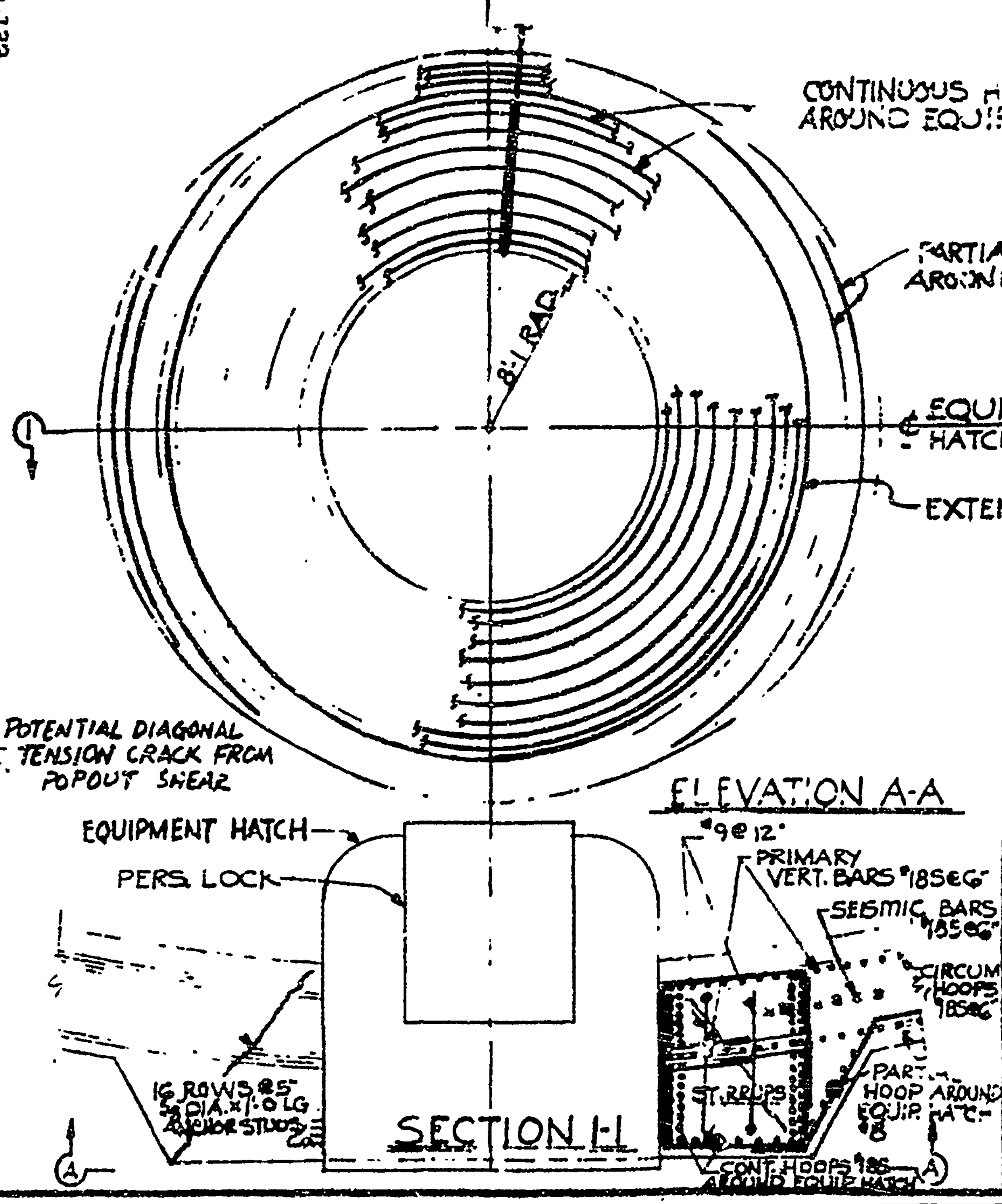
R-132

Superseded by Amend. #18, Suppl. #9



Supplement 6
2/70

R-132



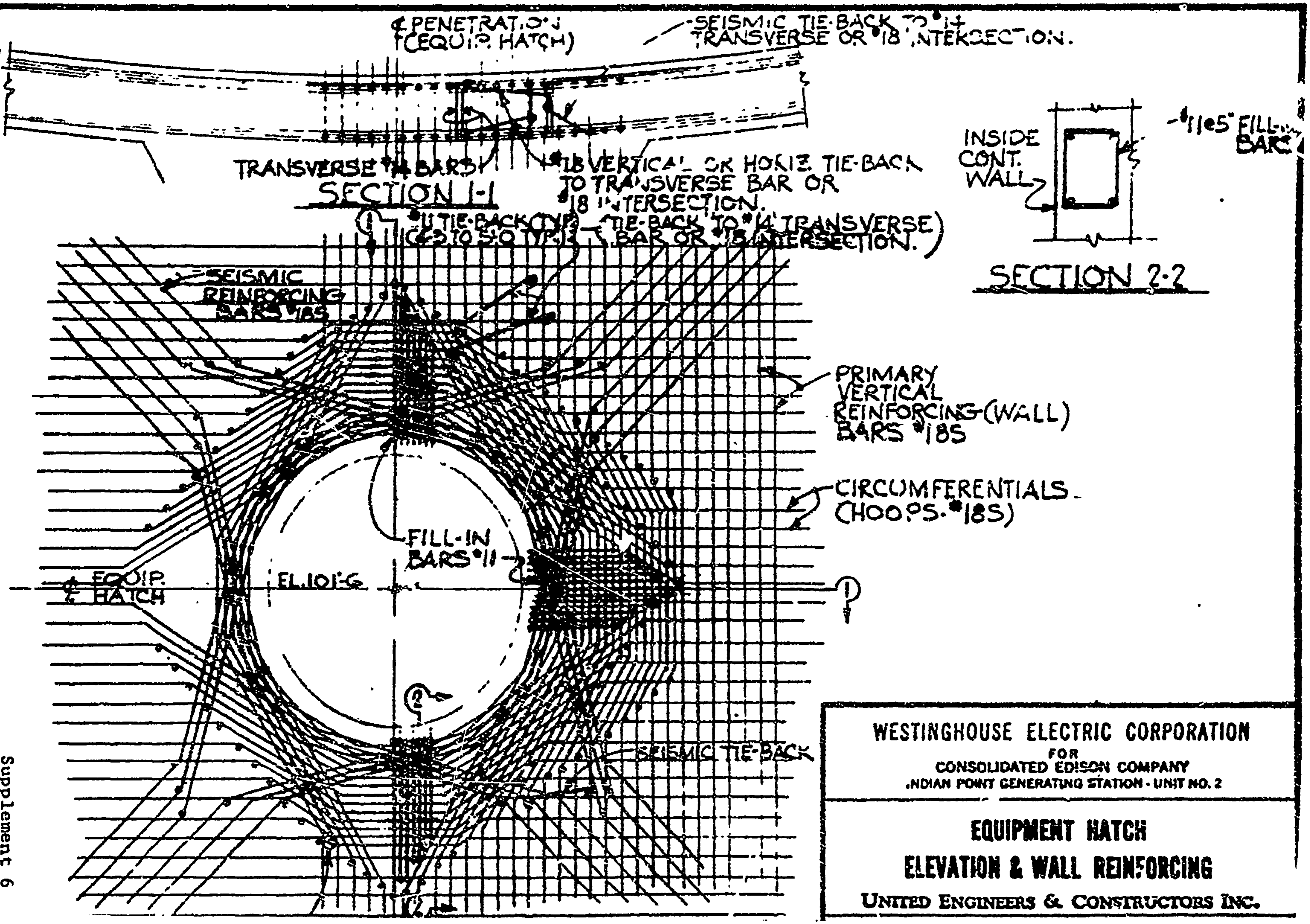
WESTINGHOUSE ELECTRIC CORPORATION
FOR
CONSOLIDATED EDISON COMPANY
INDIAN POINT GENERATING STATION - UNIT NO. 2

EQUIPMENT HATCH
PLAN & SECTION
BOSS REINFORCING DETAILS

UNITED ENGINEERS & CONSTRUCTORS INC.

U.E.& C. DWG. NO.	CON. ED. CO. DWG. NO.
9321-L-1559	

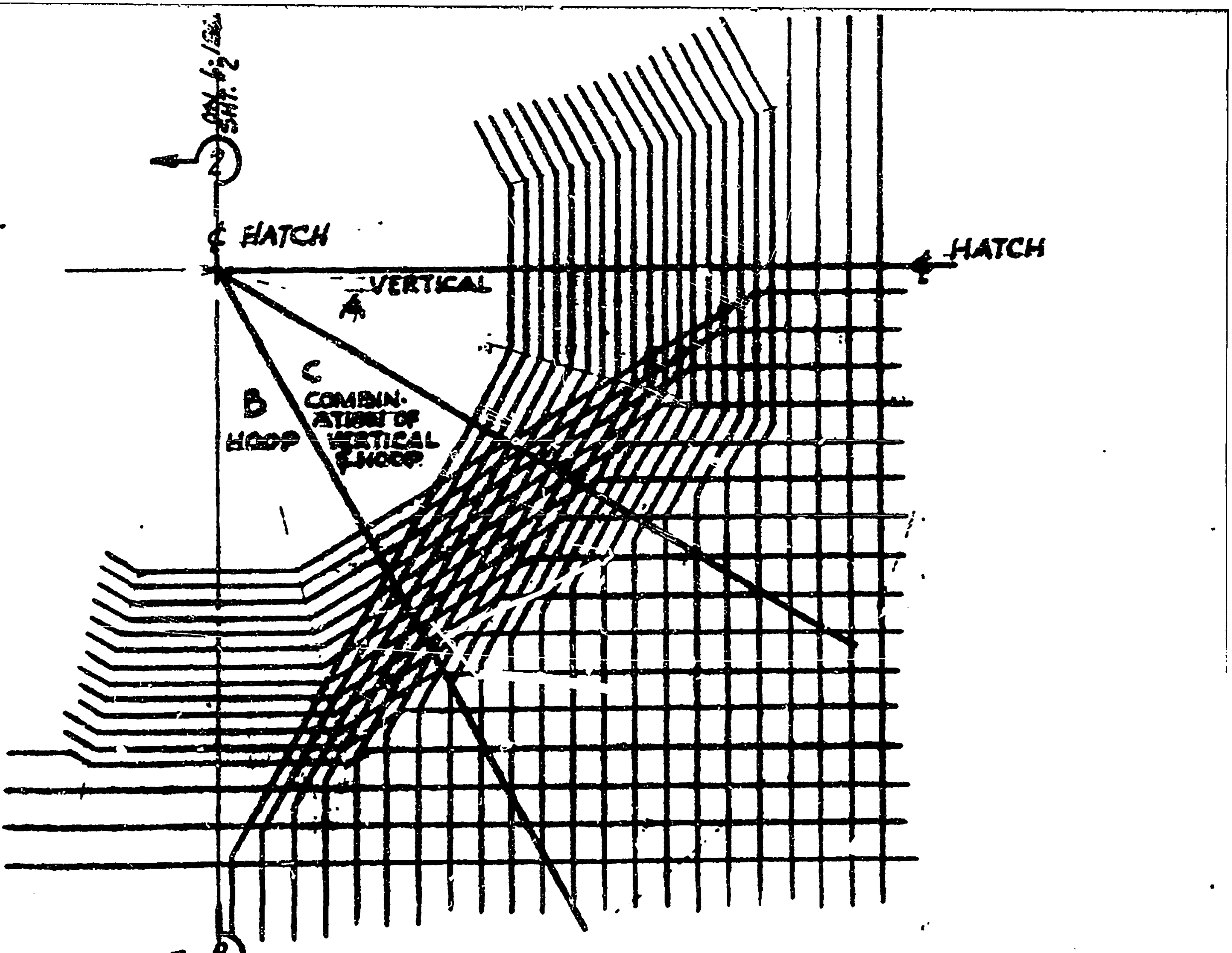
Supplement 6
2/70



WESTINGHOUSE ELECTRIC CORPORATION
FOR
CONSOLIDATED EDISON COMPANY
INDIAN POINT GENERATING STATION - UNIT NO. 2

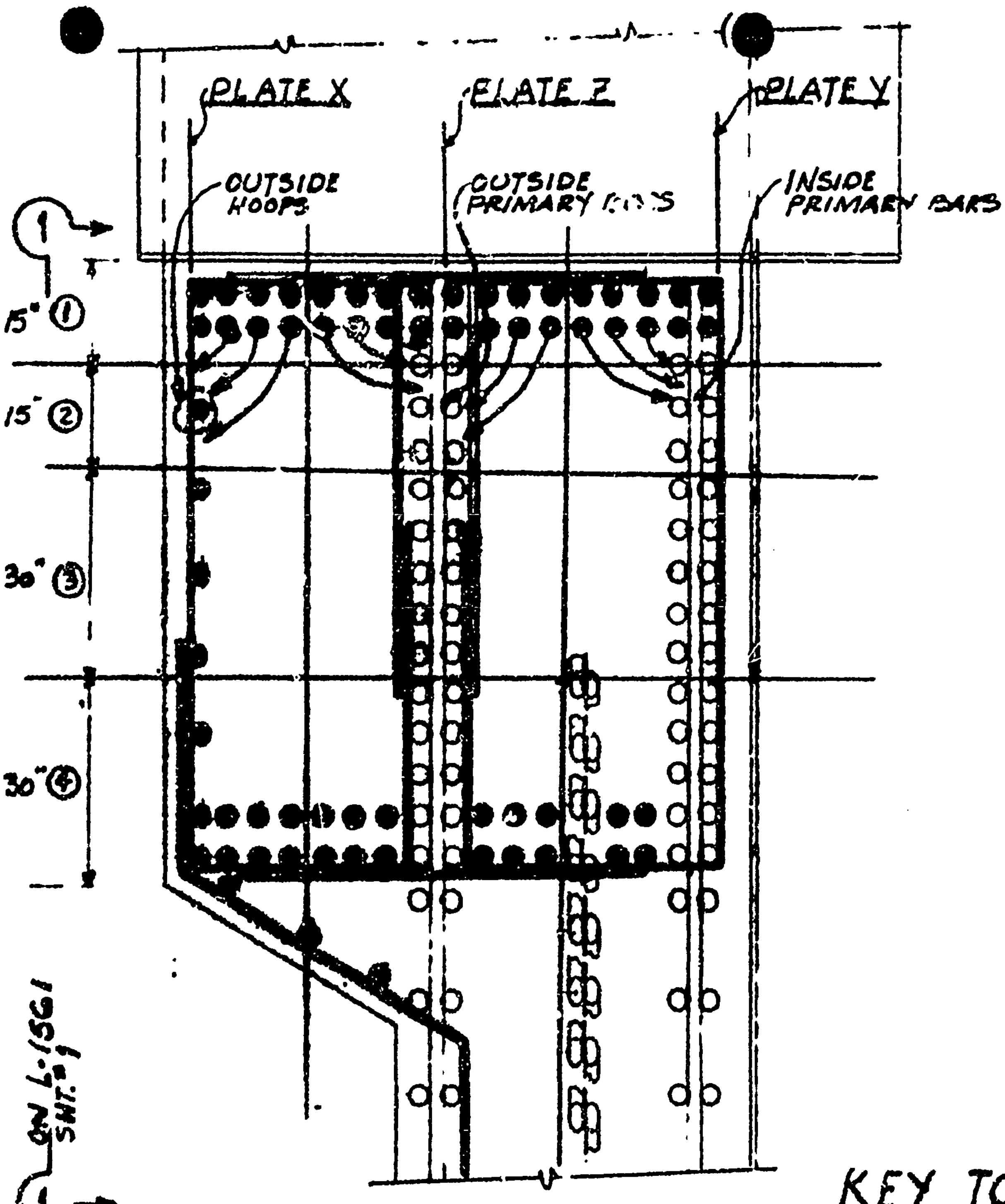
EQUIPMENT HATCH
ELEVATION & WALL REINFORCING

UNITED ENGINEERS & CONSTRUCTORS INC.



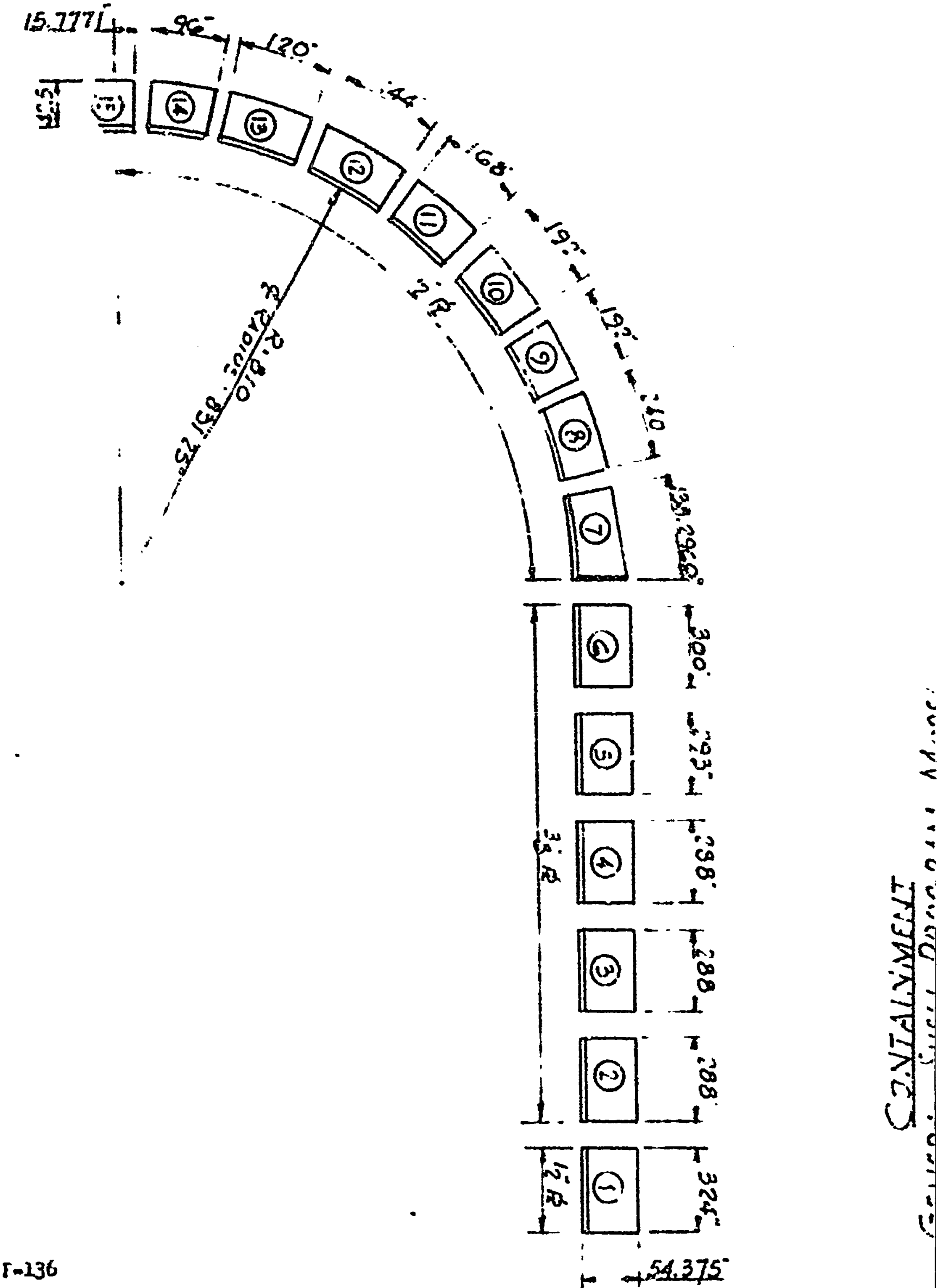
OUTSIDE ELEVATION 1-1

KEY TO EQUIPMENT HATCH REBAR IDEALIZATION

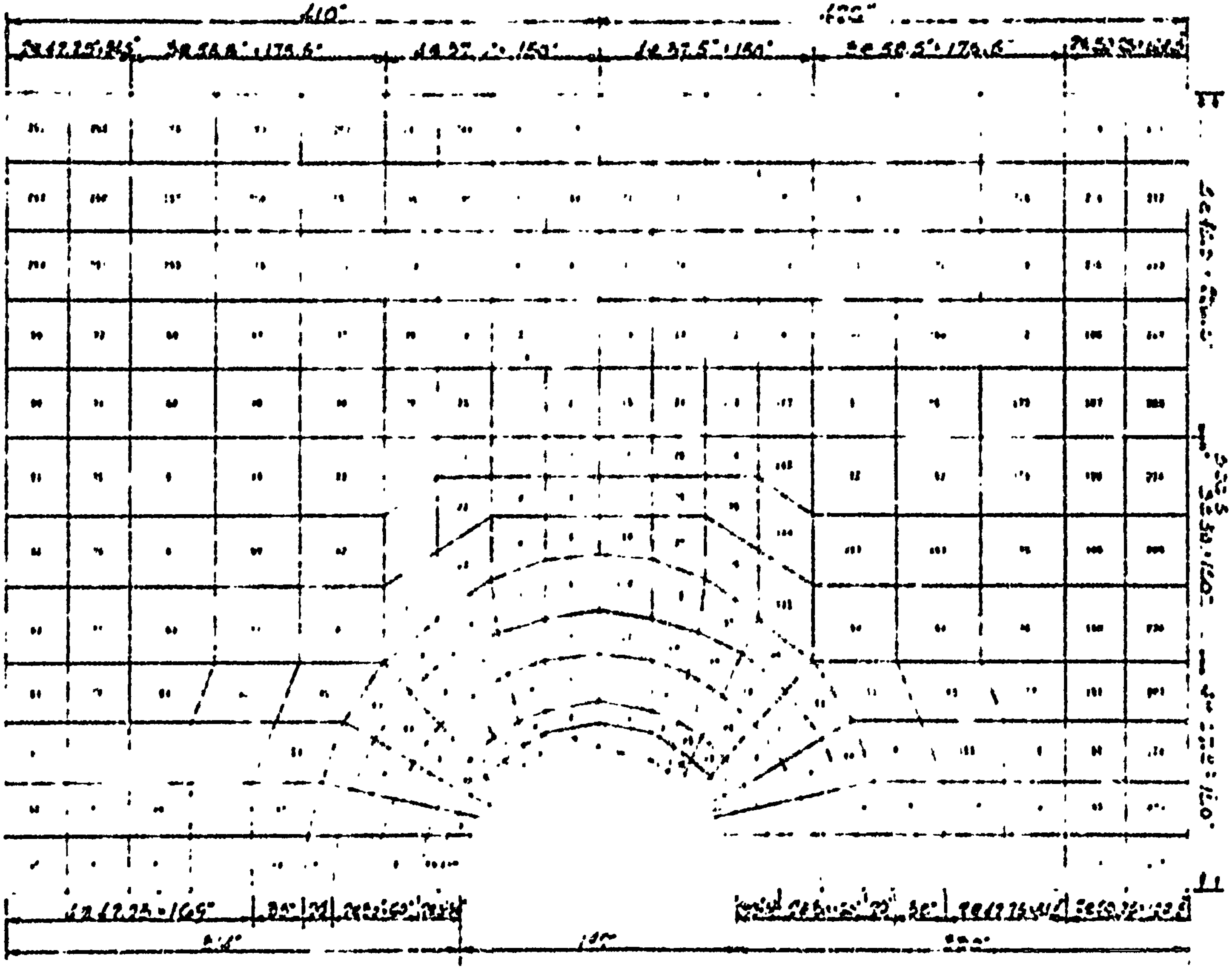


SECTION 2-2

KEY TO EQUIPMENT HATCH REBAR IDEALIZATION

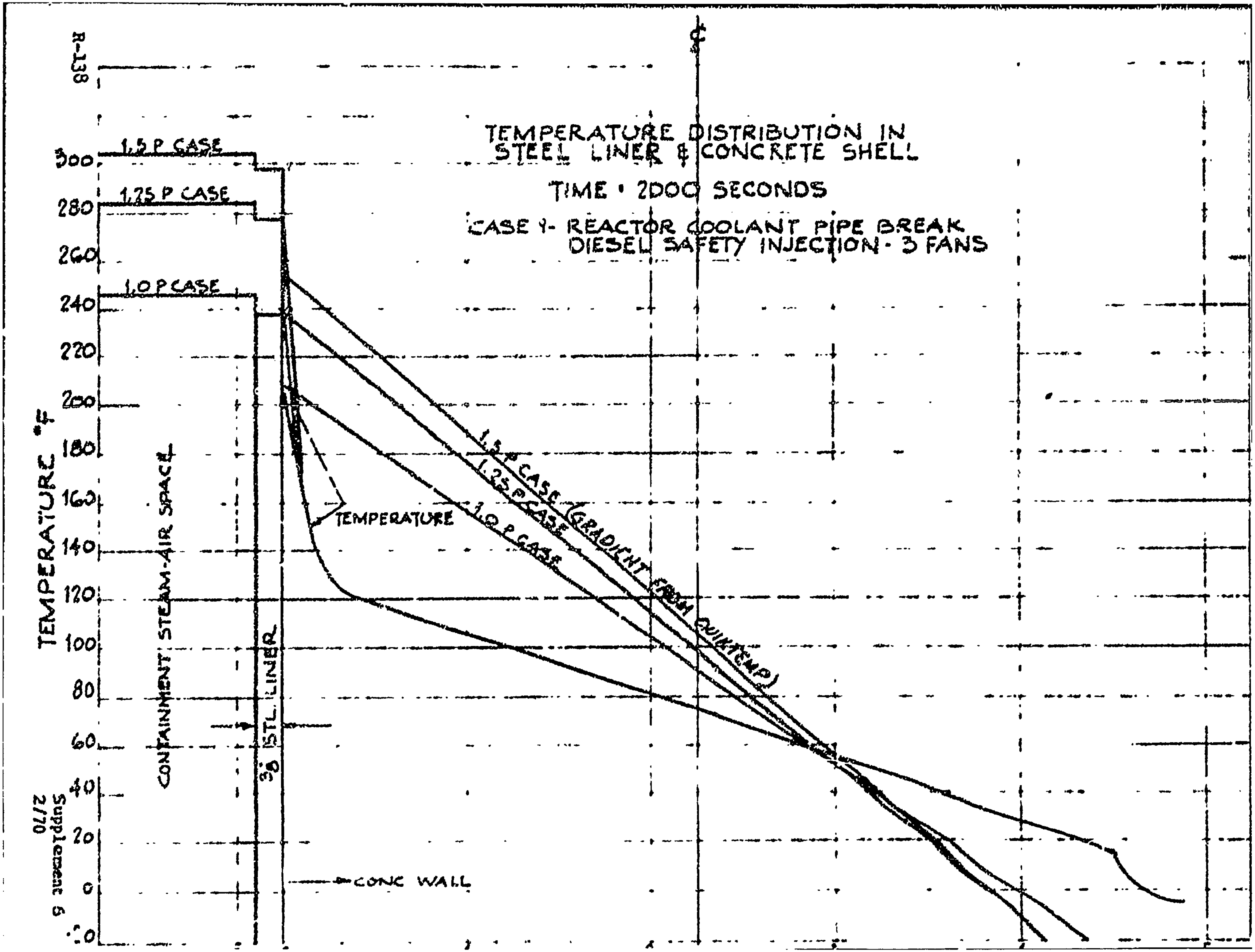


15.777
96
120
44
168
192
192
240
139.2968
300
293
288
288
288
324
54.375
R. 810
℄ RADIUS 831.75
2 ft.
3½ ft.
1½ ft.
CONTAINMENT



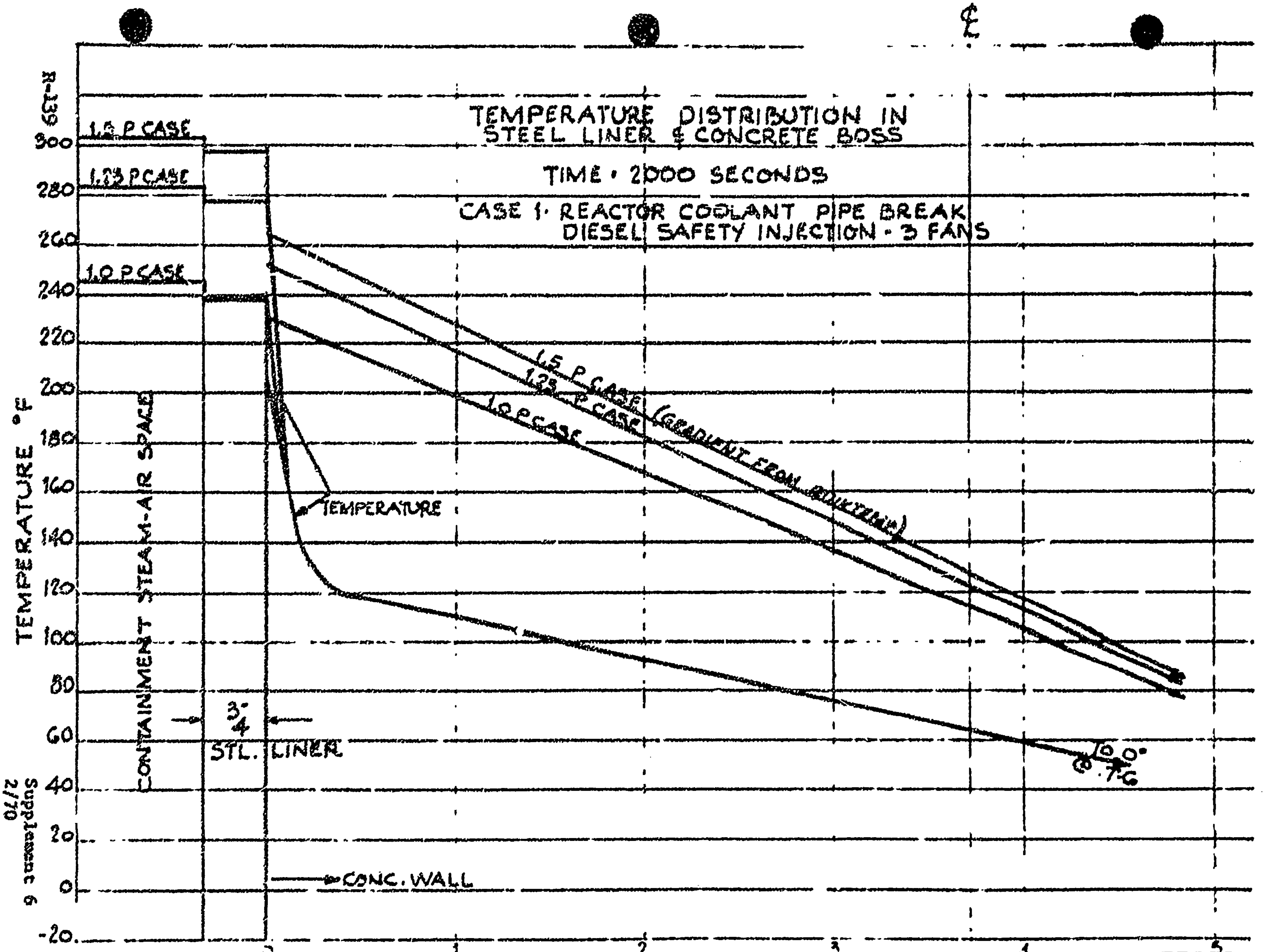
CONTAINMENT
FINITE ELEMENT MODEL
FINE GRID ANALYSIS

9321-LL-1564

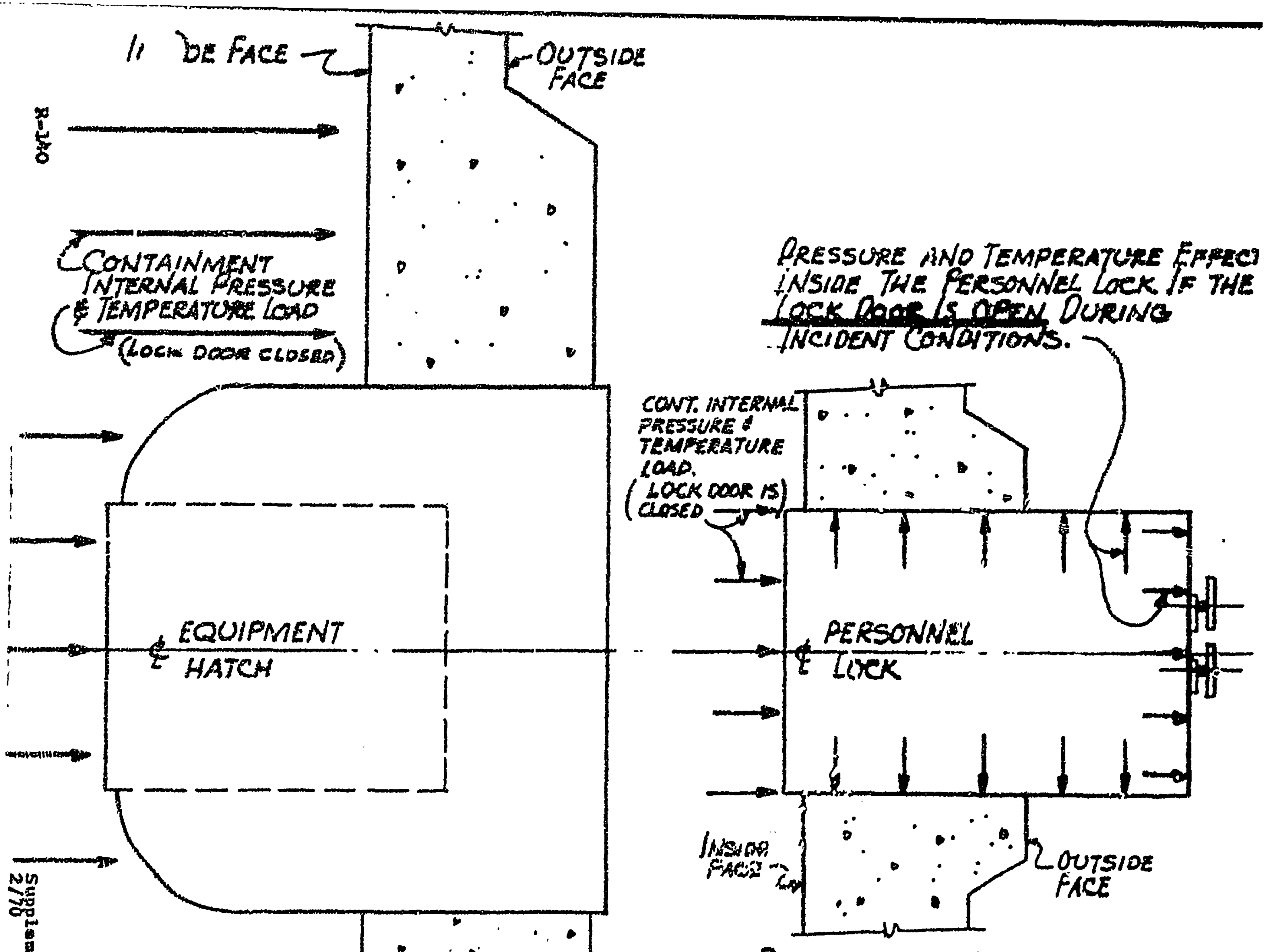


R-138
TEMPERATURE DISTRIBUTION IN STEEL LINER & CONCRETE SHELL
TIME = 2000 SECONDS
CASE Y- REACTOR COOLANT PIPE BREAK DIESEL SAFETY INJECTION - 3 FANS
1.5 P CASE
1.25 P CASE
1.0 P CASE
TEMPERATURE °F
CONTAINMENT STEAM-AIR SPACE
3/8 STL LINER
TEMPERATURE
1.5 P CASE
1.25 P CASE
1.0 P CASE
(GRADIENT FROM QUIKTEMP)
CONC WALL
Supplement 6 2/70

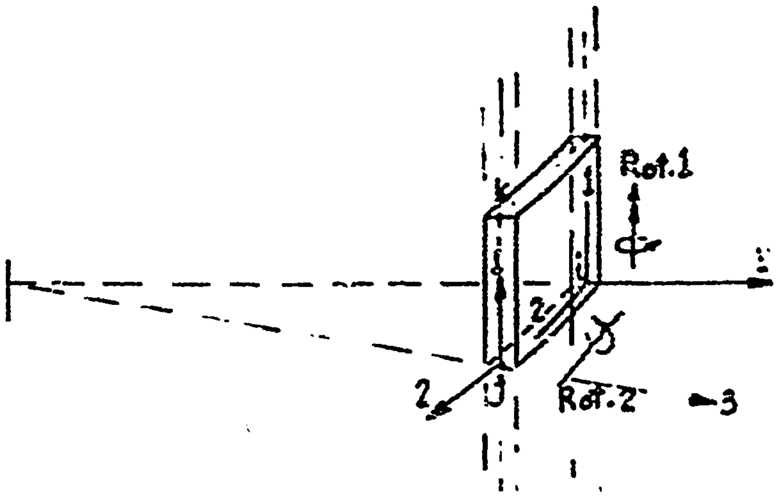
R-139



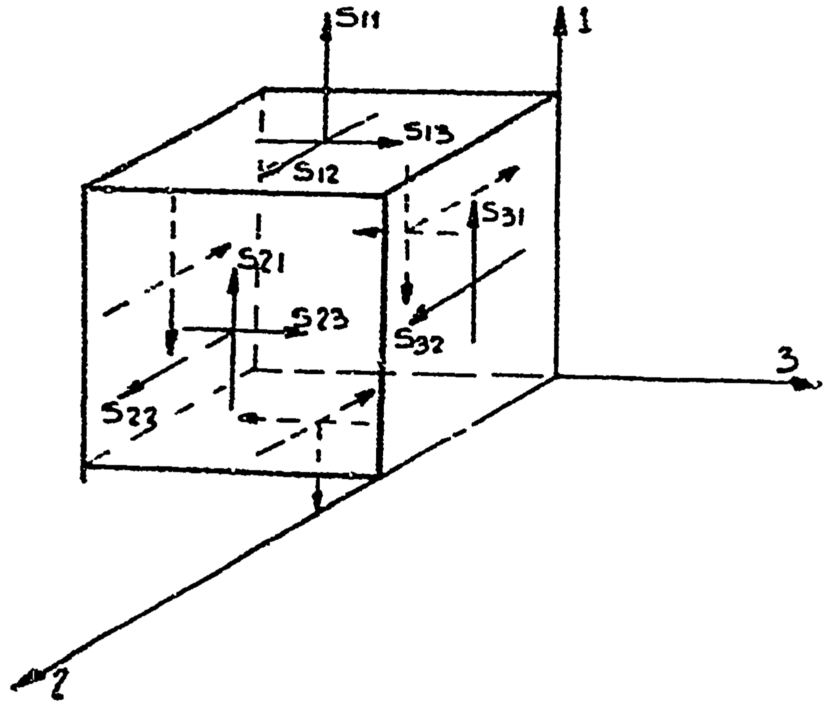
Supplement 6
2/70



PERSONNEL LOCK LOADING

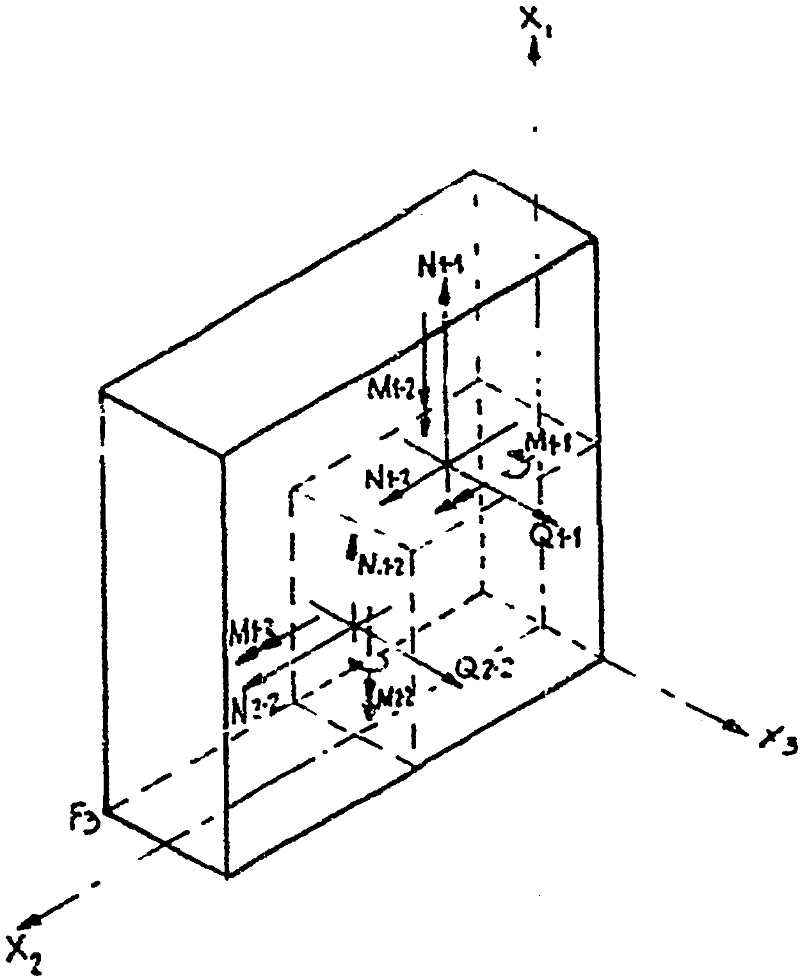


NODAL DISPLACEMENTS AND ROTATION
IN SHELL COORDINATE SYSTEM



STRESSES AT PANEL MIDPOINT

FELAP PROGR
SIGN CONVENTION
SHT 1 9321-L-156B 8-



CORNER AND EDGE FORCE RESULTANTS

FELAP PROGRAM
SIGN CONVENTION

SHT. 2 9321-L-1569 9-2-69

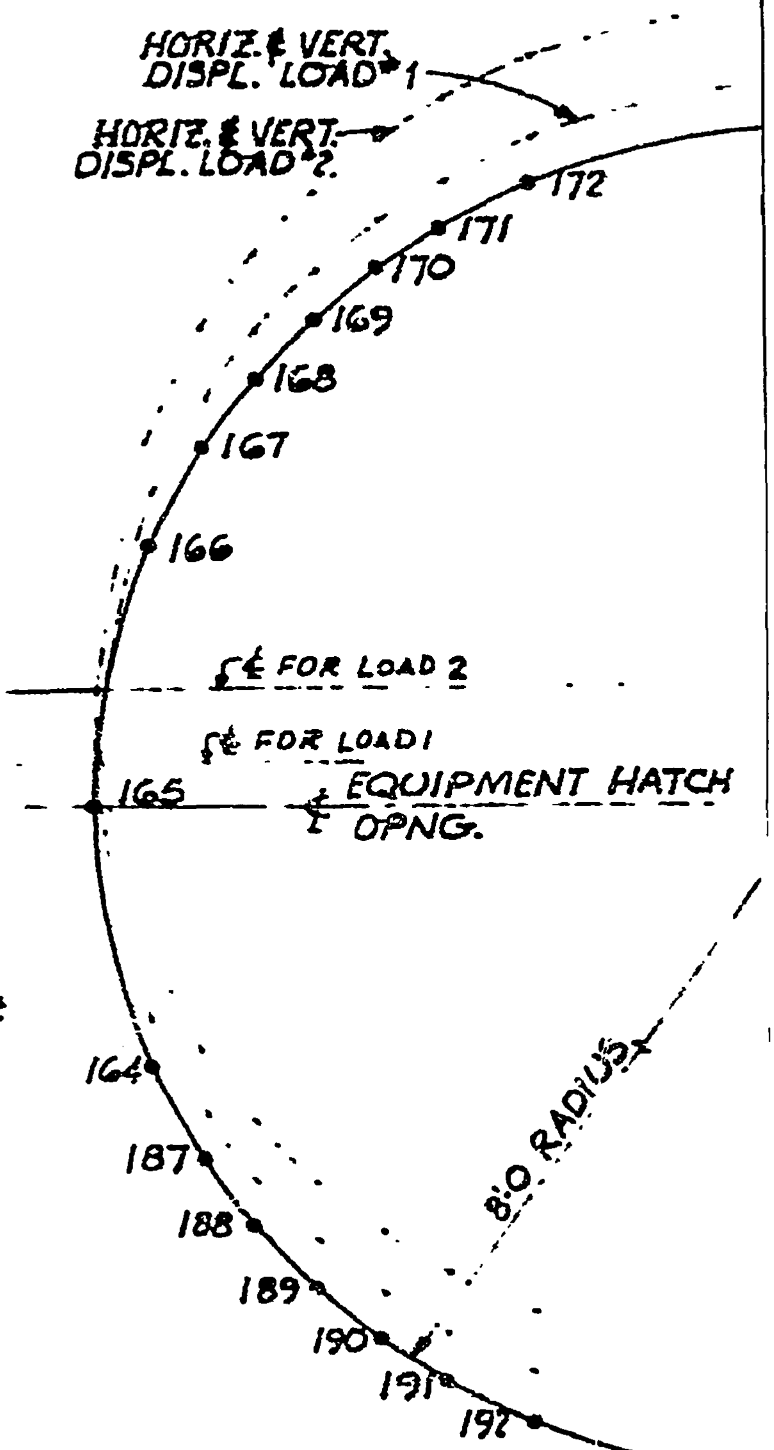
Supplement 6
2/70

NODES AT HATCH OPNG.
1.0P CASE (47 PSI)
LOAD #1

NODE #	VERTICAL DISPL.	HORIZONTAL DISPL.
173	.2413	.0
172	.2496	.00399
171	.2559	.00772
170	.2618	.01005
169	.2666	.01220
168	.2690	.01324
167	.2687	.01233
166	.2669	.01027
165	.2572	.00827
164	.2476	.01008
187	.2459	.01210
188	.2457	.01295
189	.2482	.01187
190	.2532	.00967
191	.2594	.00751
192	.2660	.00358
193	.2748	.0

NODES AT HATCH OPNG.
1.0 VERT. E.Q. + .95 D.L. + 1.25P + 1.25E + 1.0 THERMAL LOAD #2
LOAD #2

NODE #	VERTICAL DISPL.	HORIZONTAL DISPL.
173	.6535	.0
172	.6619	.0103
171	.6679	.0179
170	.6724	.0243
169	.6741	.0309
168	.6690	.0327
167	.6614	.0317
166	.6571	.0276
165	.6233	.0223
164	.5950	.0253
187	.5860	.0290
188	.5785	.0299
189	.5736	.0280
190	.5754	.0217
191	.5801	.0154
192	.5862	.0082
193	.5948	.0



DEFLECTIONS PLOTTED TO SCALE 1" = 2"

ELEVATI

OVALING EFFECT AT EQUIPMENT HATCH JF
9321-L-1570

5-2-69

R-143

Supplement 6
2/70

REFERENCES

(1) Timoshenko, S. P., and Woinowsky-Krieger, S., Theory of Plates and Shells, 2nd Ed., McGraw Hill Book Co., New York, 1959.

(2) Nuclear Reactors and Earthquakes, TID 7024, Division of Technical Information, USAEC August, 1963.

(3) Blume, J., Newmark, N., et al, Design of Multistory Reinforced Concrete Buildings for Earthquake Motion. Portland Cement Association, 1961.

(4) American Concrete Institute, "Specification for the Design and Construction of Reinforced Concrete Chimneys (ACI 505-54)," ACI Manual of Concrete Practice, Part 2, 1967.

4.0 CONTAINMENT COMPONENT DESIGN

4.1.0 CONTAINMENT SUMPS

There are three containment sumps which cause projections of the bottom of the containment base mat. The largest is the containment reactor sump which is a key shaped reinforced concrete pit located in the center of the base slab (Figure 4.1). This sump which is 52.5 feet long and 25'-6" deep encloses the bottom section of the reactor vessel and the in-core instrumentation leads. The side walls and floor of the sump are 4.5 feet thick supporting the 1/4" steel liner. An additional 2 feet of concrete is poured over the liner.

Since the reactor sump walls and floor are poured directly against the rock foundation, rigid support conditions have been considered in the design at the sump structural elements to withstand load. Also since this sump is located in the central portion of the base slab which is poured directly on the rigid rock foundation, negligible bending shears and moments exist in the base slab at the sump location under all load conditions. The reinforcing steel in the sump includes an extension of the reinforcement with the standard detailing procedures specified in ACI-315 being followed. Temperature steel is included in the sump to meet the requirements of ACI-318.

The next largest sump encloses the intakes for the recirculating pumps and consists of a rectangularly shaped reinforced concrete pit 18 feet by 12 feet in plan and 12 feet deep. The side walls and floor of this sump are 9 feet thick supporting the sump liner with an additional 3 feet covering the pit liner floor and 1 foot covering the liner enclosing the sides of the sump (Figs. 4.2, 4.3, 4.4, and 4.5). As in the case of the reactor sump, the walls and floor of this sump are supported by the rigid rock foundation and the sump is located in a region of negligible bending stresses in the base mat. The walls and floor of the sump are considered structurally as part of the base mat.

The smallest sump encloses the containment sump intake and measures 7.5 feet by 7.5 feet by 5.75 feet deep. It has side walls and floor 7.25 feet thick with a 1 foot covering on the liner. As in the case of the recirculating water sump, the walls of the sump are considered as part of the base mat and are located in a region of negligible bending moment and shear.

The three sumps and in particular the concrete cover over the sump liners, also serve as excellent shear keys in transferring seismic or thermal shear loading from the containment internal structure to the base mat. While it is anticipated most of the shear load would be transmitted by friction between the containment base liner and the containment mat the concrete cover area of the sumps acting alone is capable of transmitting full seismic shear load for a 0.15 earthquake at an average shearing stress of 120 psi.

4.2.0 Containment Base Mat

The containment base mat is a reinforced concrete slab 146 ft. in diameter and 9 ft thick, (Figure 4.6). The base slab is designed as a flat circular plate supported on a rigid non-yielding foundation. For loads applied uniformly around the slab, the analysis considers a one foot wide beam fixed at a point where the vertical shear is equal to zero. This is the point where the downward pressure on the mat and the dead weight overcome the uplift at the containment wall base mat juncture from pressure and earthquake loadings.

4.2.1 Shear Reinforcement Design of Slab:

The limiting loading condition for shear is defined by the 1.25P factored load equation which results in the base mat loads as shown in Figure 4.7. The external shear load per foot of 1 foot wide section of the mat is determine from Eqs. 3.1.18 and 3.1.20.

The maximum shear stress permitted on an unreinforced web subjected to combined shear and bending is given by (ACI-318; Eq. 17-2)

$$v_c = \emptyset \left(1.9 \sqrt{f_c'} + 2500 \frac{p_w Vd}{M}\right) \qquad (4.2.1)$$

where:

v_c = shear stress carried by concrete

$\emptyset$ = capacity reduction factor for shear (0.85)

p_w = reinforcement ratio (As/bd)

V = total shear at section

M = bending moment at section

d = depth of section from compression fact to centroid of tensile steel (100 in)

Solving Eqs. 4.2.1 and 4.2.3 for the loads defined in Figure 4.7 the distance x determined as the cut off point for shear reinforcement is 16.0 ft. or just inside of the crane wall.

The shear load V used for the design of shear reinforcement is determined at a distance d from the edge of the slab. This value for the loading given in Figure 4.7 is 209 k/ft psi. The shear load which is assumed carried by other than shear reinforcement is determined as shown in Eq. 3.2.10 equal to 108 psi.

$$V_c = vbd = 108 \text{ k/ft}$$

where:

$v = 2\emptyset\sqrt{fc'}$ (ACI-318, section 1701); $\emptyset = 0.85$
d = effective depth of base mat slab (100 in.)
b = width of wedge shaped section at x = d = 100 in.

The required area of shear reinforcement per foot is determined by Eqs. 3.2.9 and 3.2.7 as shown in Figure 4.8.

4.2.2 Moment Reinforcement Design of Slab:

As in the case for shear, moment was calculated by writing equations for moment in terms of x using the center of the containment wall-base slab juncture as the origin with x increasing toward the center of the containment building. For the 1.5P limiting case the discontinuity moment is 1210 k.ft/ft, the discontinuity shear is 157 k/ft as shown in Figure 4.9. The expressions for the moment as a function of x are shown in Equations 3.1.21 and 3.2.23.

The loading diagram in the mat is shown in Figure 4.9. The equation for moment as a function of x is set equal to zero and the distance x at which the condition of tension in the top of the mat would discontinue is found to be 6.5 feet. The expression for shear is also set equal to zero and the distance x at which the maximum positive moment (1208 k.ft) occurs is found to be 20.6 feet.

The moment steel provided for the maximum negative moment of 1210 k ft/ft which occurs along the perimeter of the slab is also assumed to carry one half of the discontinuity shear of 157 k/ft as an axial load which results in a direct stress of 18.4 psi. The section is designed according to Part IV-B Structural Analysis and Proportioning of Members - Ultimate Strength Design of the ACI-318-63 Code as shown in Section 3.2.3 of this report.

The value of fy used is reduced to 41.6 KSI since 18.4 KSI is taken by the discontinuity shear and the ultimate moment is found to be 1,250 K;ft/ft which is greater than the maximum applied negative moment value of 1210 K ft/ft

For all combinations of pressure, dead loa' and earthquake loadings which tend to cause uplift in the base slab, the dead weight of the crane wall greatly reduces uplift. This forms a rigid central region in the base slab which is supported on an essentially rigid non-yielding foundation. The model used to analyze this condition is a circular and solid flat plate with a central rigid portion subjected to an external moment (Figure 4.10). The maximum radial stress at the inner edge is given by[1].

$$\sigma_R = \beta \frac{M_{EXT}}{a\ t^2} \qquad (4.2.2)$$

where:

M_{EXT} = external overturning moment

β = parameter which depends on ratio of a to the radius of the central rigid portion of the slab.

a = radius of the circular slab (875 in)

t = thickness of the slab

The radial stress due to an internal moment is

$$\sigma R = \frac{M_{INT}C}{I} = \frac{6M_{INT}}{t^2} \qquad (4.2.3)$$

where:

M_{INT} = internal moment in base slab

C = distance from neutral axis to outer fiber of section

I = moment of inertia of section

By equating Eqs. 2.11 and 2.12 the expression for internal moment as a function of the external overturning moment is

$$M_{INT} = \frac{\beta M_{EXT}}{6a} \qquad (4.2.4)$$

The external overturning moment M_{EXT} is that due to the seismic shear forces. The maximum positive moment acting on the slab base occurs for the 1.25 P factored load case at the crane wall. The uplift pressure is added to the internal moment due to the seismic overturning moment.

Temperature steel was also added in the base mat to meet the requirements of article 807 of the ACI 318 Code. In the circumferential direction reinforcement is placed in the top and bottom of the base slab. In the central region of the base slab for a radius of 28 feet the temperature steel is placed in an orthogonal grid pattern.

4.3.0 Containment Cylinder Walls

The analysis of the cylinder was accomplished by the superposition of membrane forces resulting from gravity, internal design basic accident, temperature and pressure and overturning due to earthquake using the factored load equation presented in Section 2.1.12. The cylindrical walls are reinforced circumferentially with steel hoops and vertically with straight bars.

For the vertical axial load in the cylinder the 1.25P loading condition governs the design. The axial force in the cylinder due to the pressure loading on the dome is given by Eq. 3.1.5.

The uplift force in the cylinder due to the horizontal earthquake is given by Eqs. 3.1.29 and 3.1.32. The dead weight force in the cylinder is obtained by taking the total weight of the dome as the force acting at the top of the cylinder and the total weight of the dome and cylinder as the force acting at the base. The uplift force in the cylinder due to the pressure loading on the dome and the uplift due to the horizontal earthquake are combined with the dead weight load in the cylinder. The resultant load diagram varies from an uplift force of 330 k/ft at the base of the cylinder to 276 k/ft at the springline.

For the hoop direction the 1.5P case controls since the dead weight and earthquake effects are zero. The force in the hoop direction is given by Eq. 3.1.6.

The seismic loads were determined as described in Section 3.1.5. To provide for the seismic steel, diagonal bars are placed in the center of the cylinder walls in both directions at an angle of 45°.* Seismic steel reinforcement is as shown in Figures 4.11 and 4.12.

---

* The design of the diagonal steel is such that its horizontal component is equal to the maximum value of the shear flow which is equal to twice the average shear on the cross-section. Since the diagonals are assumed to act in diagonal tension only, half of the total area of the 45° diagonal seismic bars is assumed active to resist seismic shear effects at any given instant.

4.4.0 Containment Dome

The thickness of the dome is small in comperison with the radius of curvature (1/15) and there are no discontinuities such as sharp bends in the meridional curves, therefore the stresses due to dead weight, pressure, or earthquake, were calculated by considering a uniform distribution across the wall thickness. All membrane tensile stresses are assumed taken by the steel reinforcement and none by the concrete unless they are compressive stresses since the concrete is assumed to have no tensile strength.

The membrane analysis of the hemispherical dome has been performed by the superposition of forces resulting from gravity and accident pressure. The dead weight forces in the dome are computed by using the procedure outlined in the Portland Cement Association Bulletin ST55, "Design of Circular Domes." The total vertical dead load acting downward for a given central angle from the apex is given by Eq. 3.1.4.

The meridional thrust (T) is given by Eq. 3.1.3 and the circumferential thrust (H) is given by Eq. 3.1.2.

The membrane force due to the internal design pressure is equal throughout the dome and is given by Eq. 3.1.5.

Analysis has shown that the earthquake effects are small in the dome, therefore the critical design condition is the 1.5P factored load case. The membrane forces due to the 1.5 factored internal design pressure of 70.5 psi are added vectorily to the membrane forces due to 95 percent of dead weight and the total force per foot is divided by the allowable yield stress of the rebar (57 KSI) to determine the area of steel required. All of the combined direct stresses are developed in the reinforcing steel encased in the concrete.

The vertical steel in the cylindrical concrete wall is extended into the dome such that a continuity between the dome and the cylinder is achieved. At an angle of 60° from the springline the 18S bars come together to a 6 inch spacing. The bars are connected to splice plates by means of Cadwell mechanical splices such that for every two bars coming together there is one 18S bar extending beyond this point. At an angle of 75° from the springline the bars again come together to a 6 inch spacing and are cadwelded to a splice plate to increase the spacing to 12 inches. Similarly the 18S bars are connected to splice plates at an angle of 83° and 86.5° from the springline. At the apex of the dome 18S bars at a constant spacing of 12 in connect the splice plates which are 3.5° from the center of the dome as shown on Figure 4.13.

To provide the required earthquake resistance the seismic steel in the cylinder is extended into the dome to a point which is 30° above the springline as shown in Figure 4.14.

Above 30° from the springline the membrane steel in the dome is sufficient to carry the seismic shear. The maximum stress in the rebar due to an earthquake is determined by resolution of the principal tensile stress into components parallel the rebar. In addition the dome liner has sufficient capacity to carry seismic loads under operation or accident conditions.

4.5.0 Containment Liner

Details of the containment liner design can be found in Appendix C of the FSAR.

4.6.0 CONTAINMENT CYLINDER, BASE, AND DOME AT POINTS OF DISCONTINUITY

Discontinuity stresses occur at changes in section or direction of the containment shell. The juncture 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 diamter somewhat differently than the dome. To compute the unrestrained dimensional changes the dome and cylinder have been considered as steel membranes equivalent to the area of reinforcing steel in the hoop direction. As shown in Section 3.1.3, the unrestrained radial deformation of the dome and cylinder are nearly equal therefore the discontinuity moments and shears are insignificant and there is no steel required at the dome to cylinder juncture due to the discontinuity effects.

The juncture of the cylindrical wall and the base mat is also a point of discontinuity. In determining the discontinuity moments and shears, the base mat was considered as offering complete fixity, therefore the only discontinuity is that due to the unrestrained radial expansion of the cylinder. As for the dome to cylinder juncture the unrestrained radial expansion of the cylinder has been computed by considering the cylinder to be a steel membrane equivalent to the area of reinforcing steel in the hoop direction. The method of analysis for the discontinuity moment and shear and its distribution into the cylindrical walls is given in Section 3.1.3.

The maximum discontinuity moment at the base occurring under the 1.5P factored load condition is 1210 K.FT/FT and the maximum discontinuity shear is 157/K/FT. The limiting discontinuity moments and shears are distributed as shown in Figs. 5.1-11, 5.1-12 and 5.1-13 of Section 5 of the FSAR (attached hereto). The placement of steel to carry discontinuity shears and moments is shown in Figs. 4.15 and 4.16

The required area of shear reinforcing as determined from Eq. 17-6 of the ACI Code is given in Eq. 3.2.7. The allowable value of f_y used as the basis for f_s in Eq. 3.2.7 is reduced from 60 KSI to 47 KSI since part of the stress is assumed taken by the axial force due to uplift. The point where the minimum web reinforcement required is less than the .15 percent of the area b_s the provisions of ACI-318 Code Article 1706b apply.

The allowable shear which may be taken by the concrete alone is found from Article 1701 e) of the ACI Code and is given by

$$v_c = 3.5\phi \sqrt{f'_c (1 + 0.002N/A_g)}$$

where

v_c = allowable shear stress carried by the concrete.

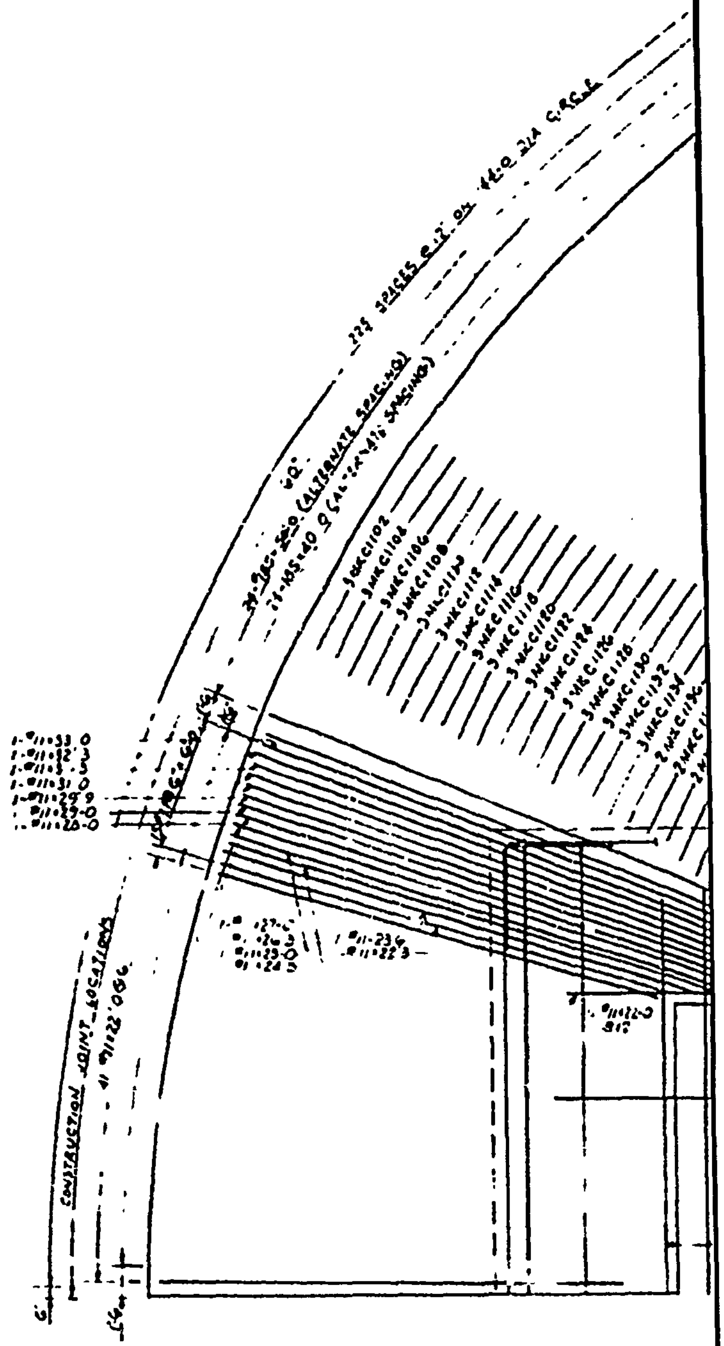
f'_c = concrete design compressive strength

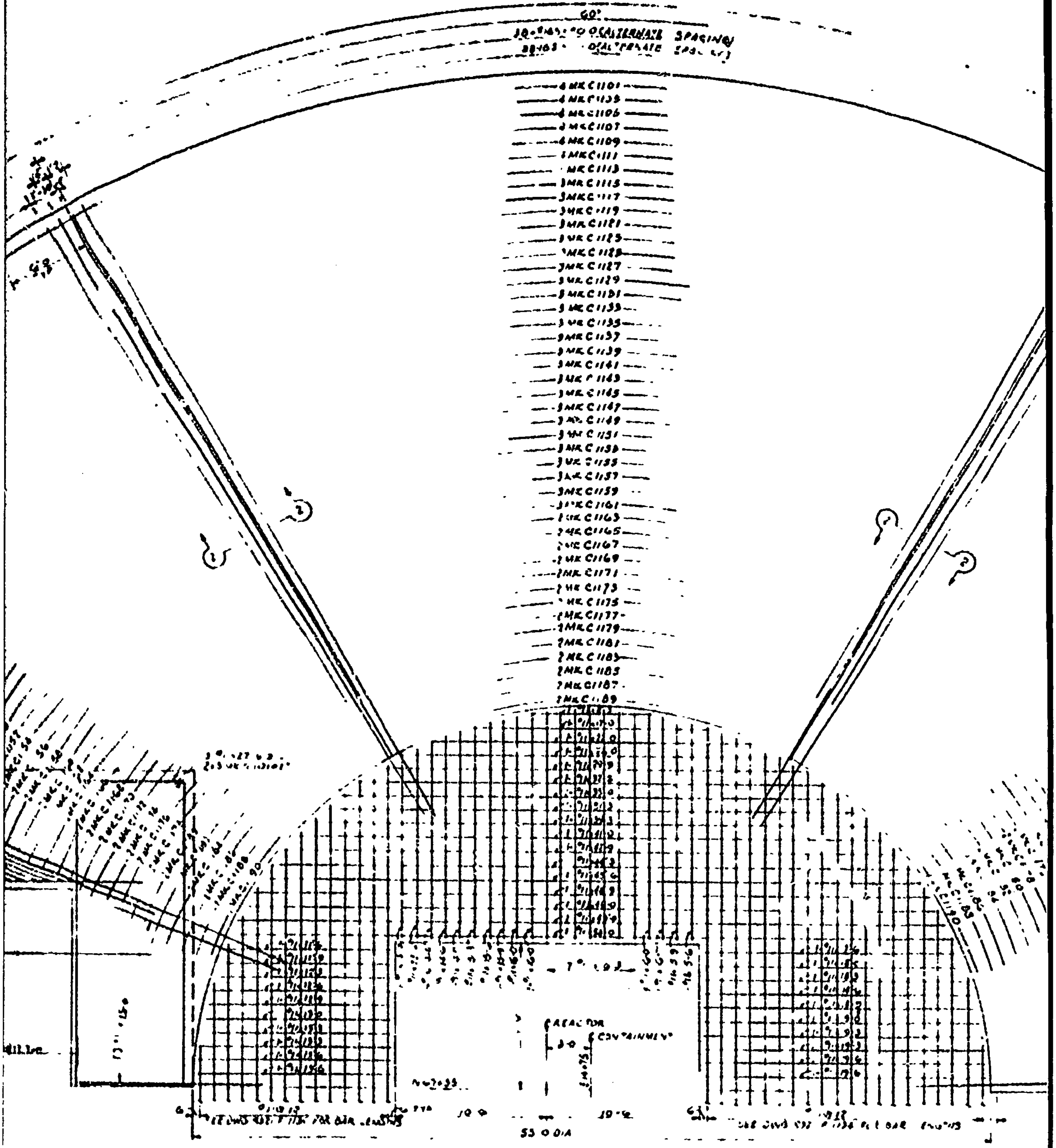
N = load normal to the cross section where N is negative for tensile loads

A_g = gross area of cross section

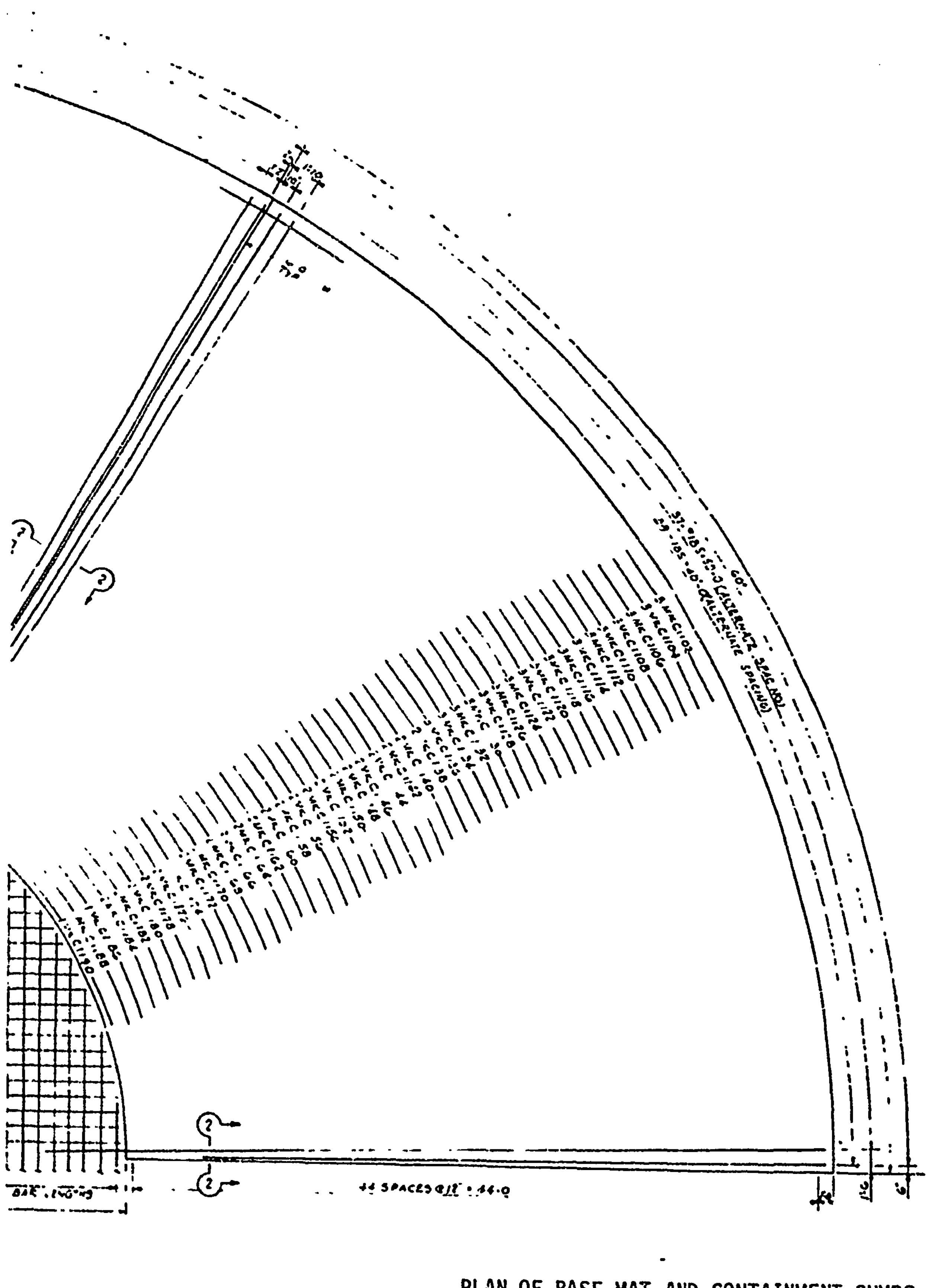
Reference

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


CONSTRUCTION JOINT LOCATIONS

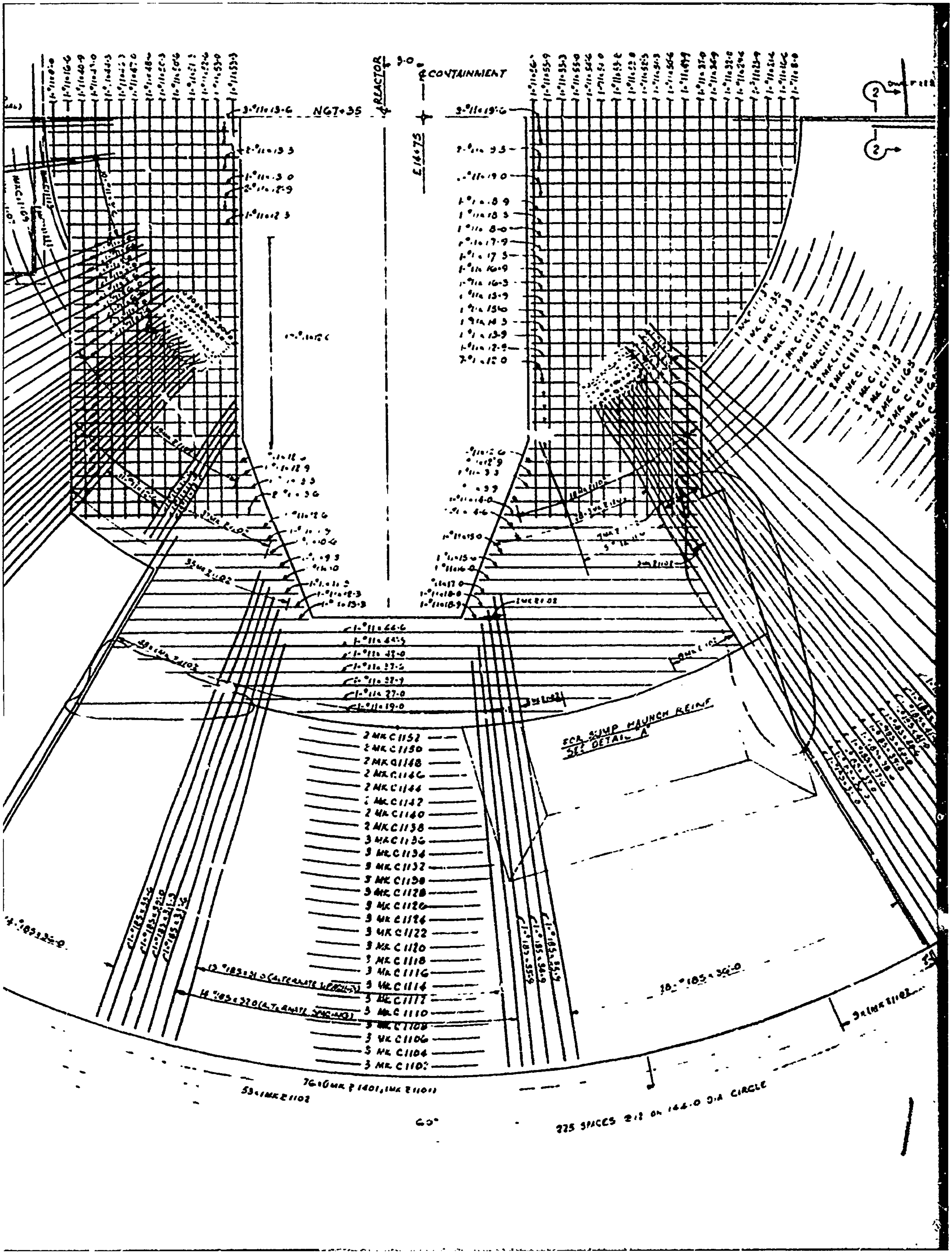


60°
REACTOR
CONTAINMENT
55.0 DIA

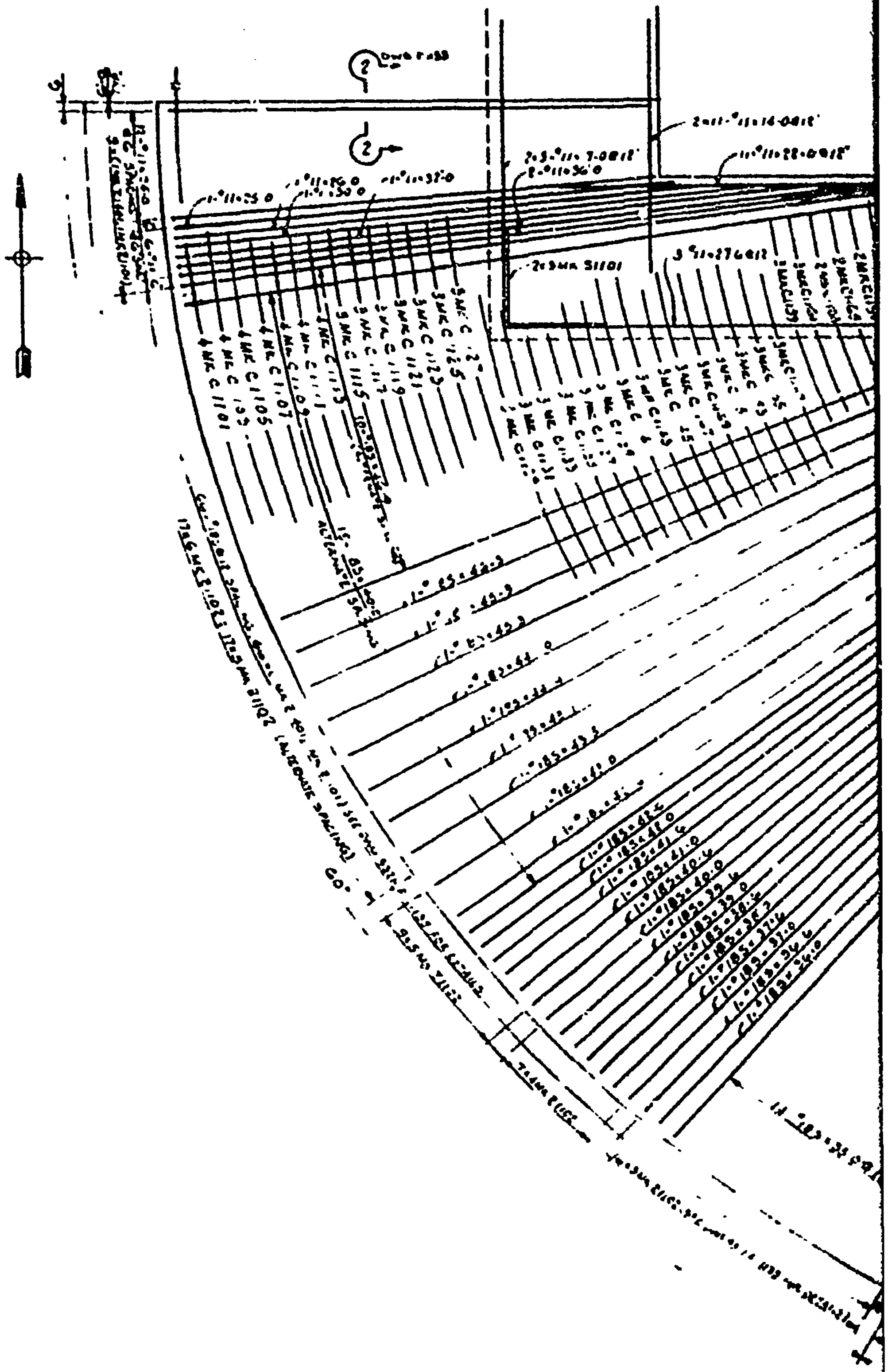


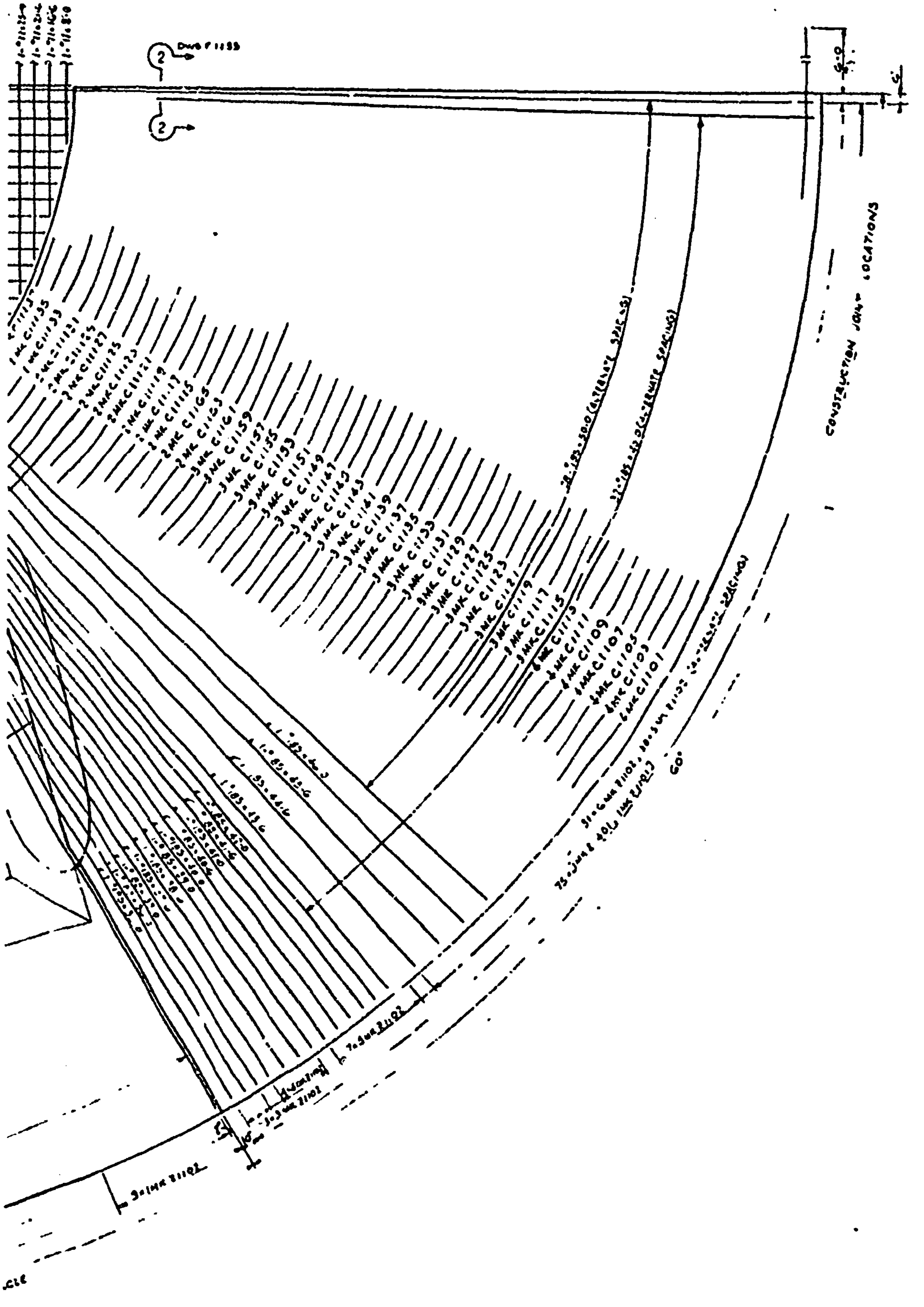
PLAN OF BASE MAT AND CONTAINMENT SUMPS

FIGURE 4.1



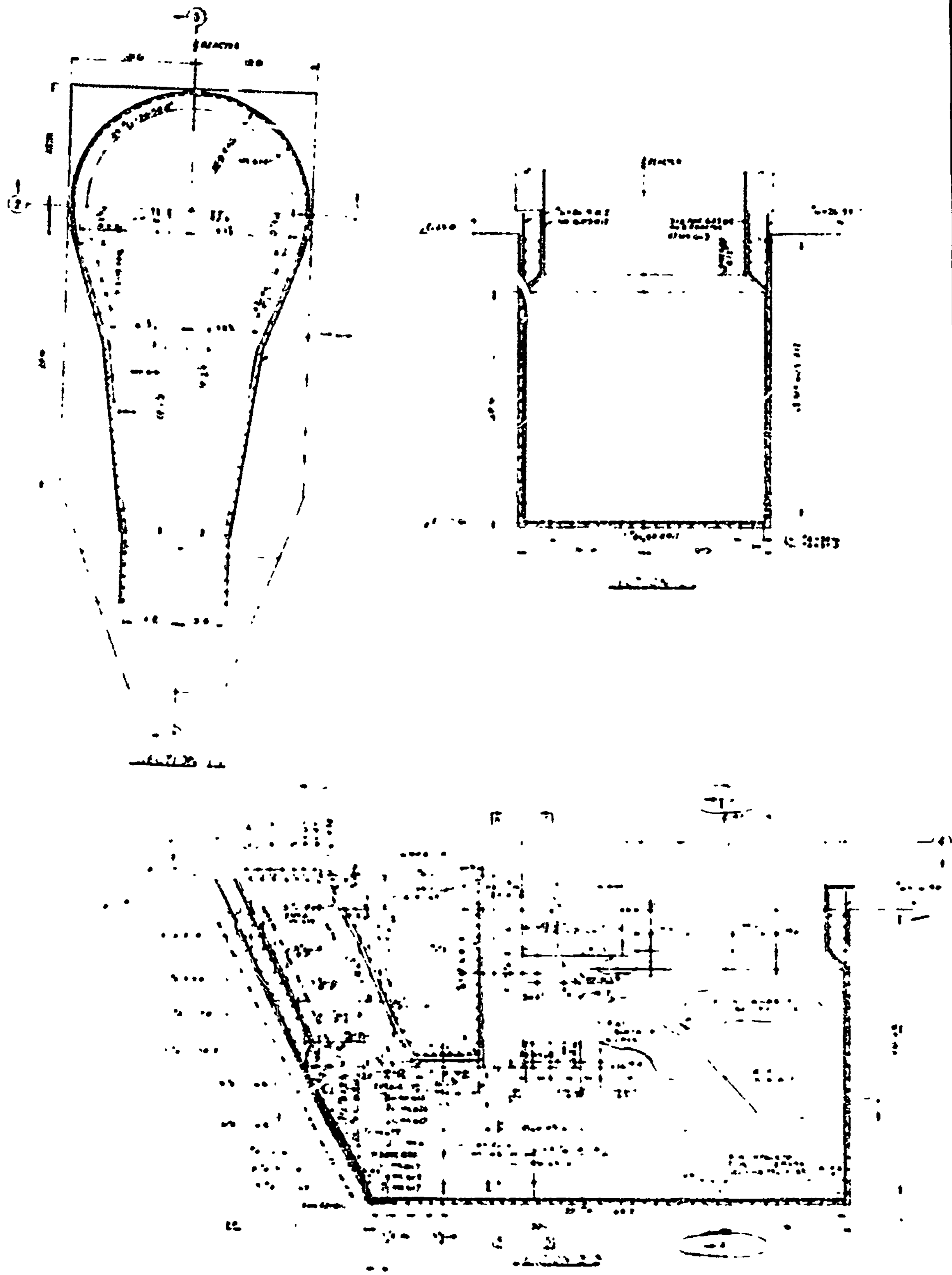
℄ REACTOR
℄ CONTAINMENT
NG7+35
E14+73
FOR SUMP HAUNCH REINF SEE DETAIL A
2 MK C1152
2 MK C1150
2 MK C1148
2 MK C1146
2 MK C1144
2 MK C1140
2 MK C1138
3 MK C1136
3 MK C1134
3 MK C1132
3 MK C1130
3 MK C1128
3 MK C1126
3 MK C1124
3 MK C1122
3 MK C1120
3 MK C1118
3 MK C1116
3 MK C1114
3 MK C1112
3 MK C1110
3 MK C1108
3 MK C1106
3 MK C1104
225 SPACES @ 2 ON 144.0 DIA CIRCLE





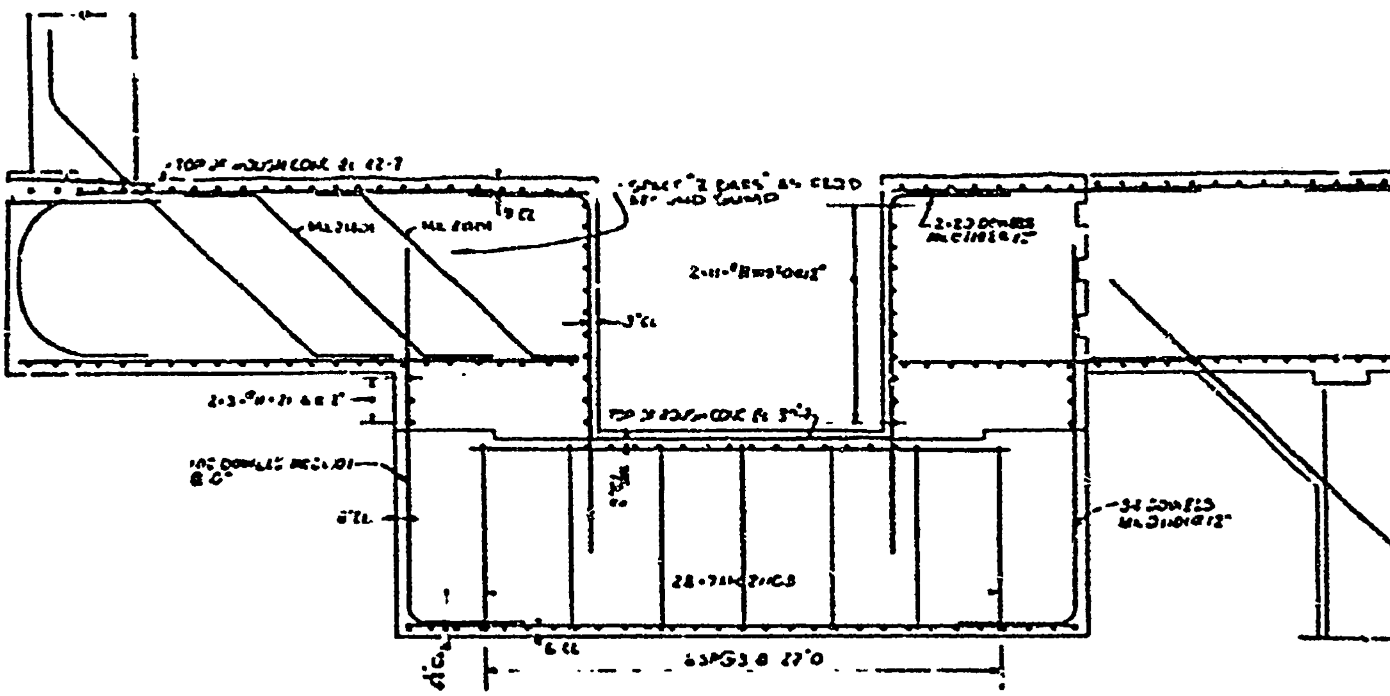
PLAN OF BASE MAT AND CONTAINMENT SUMPS

FIGURE 4.2



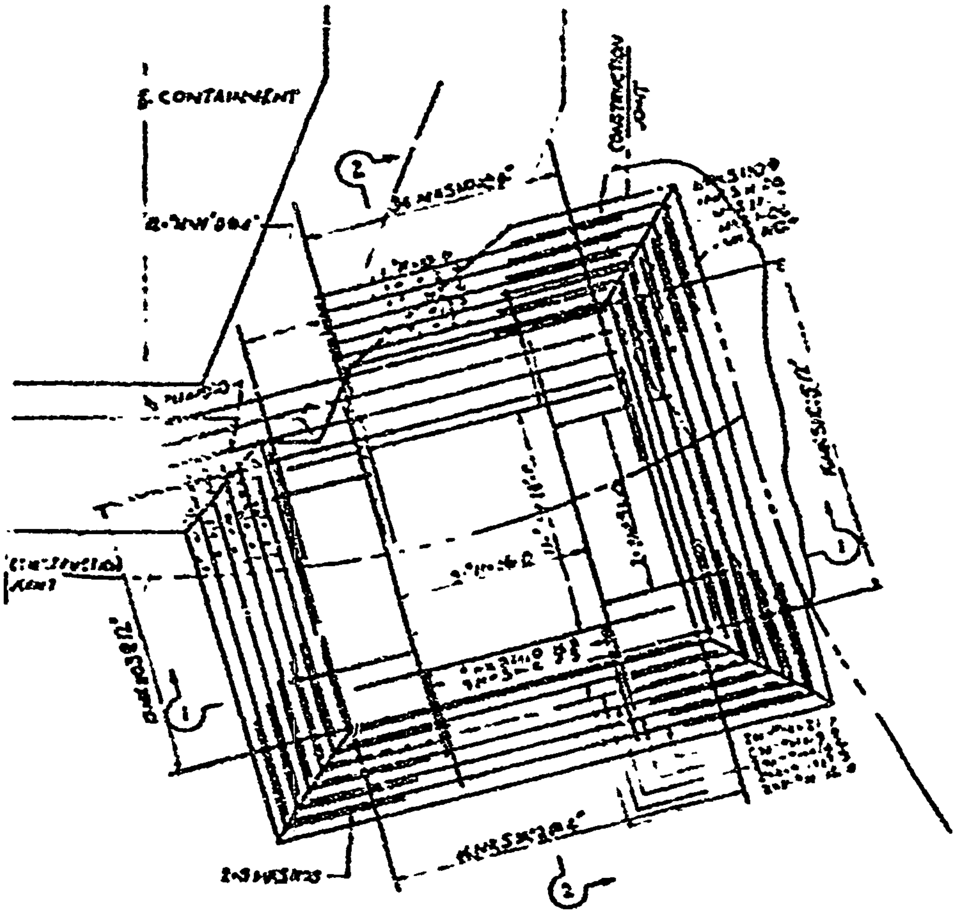
CONTAINMENT REACTOR SUMP

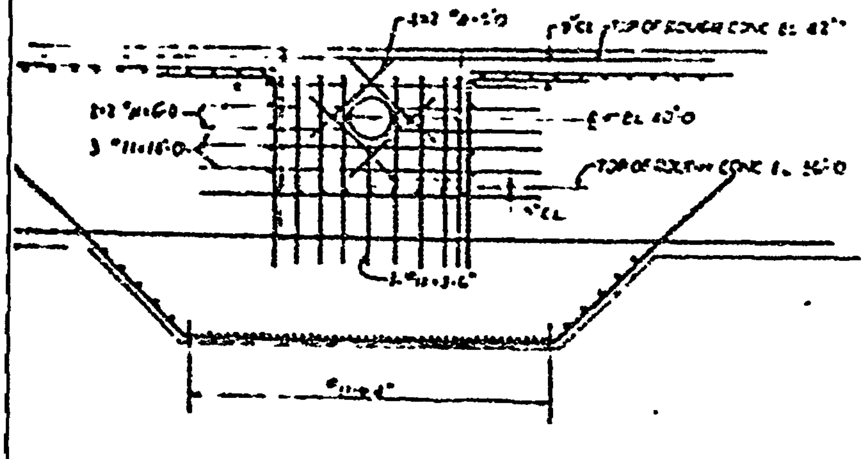
FIGURE 4.3



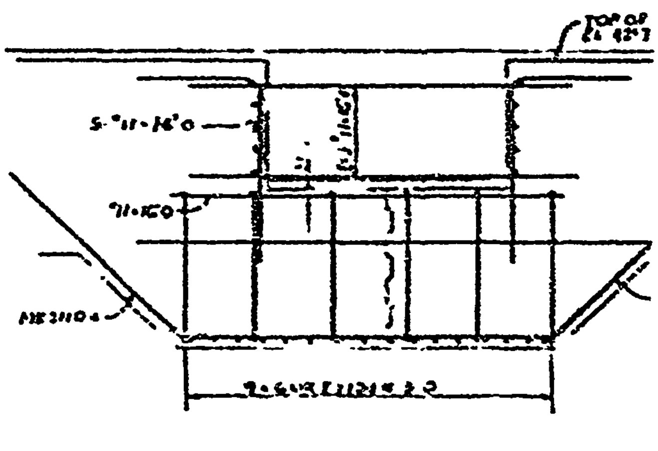
SUMP FOR RECIRCULATING PUMPS

FIGURE 4.4





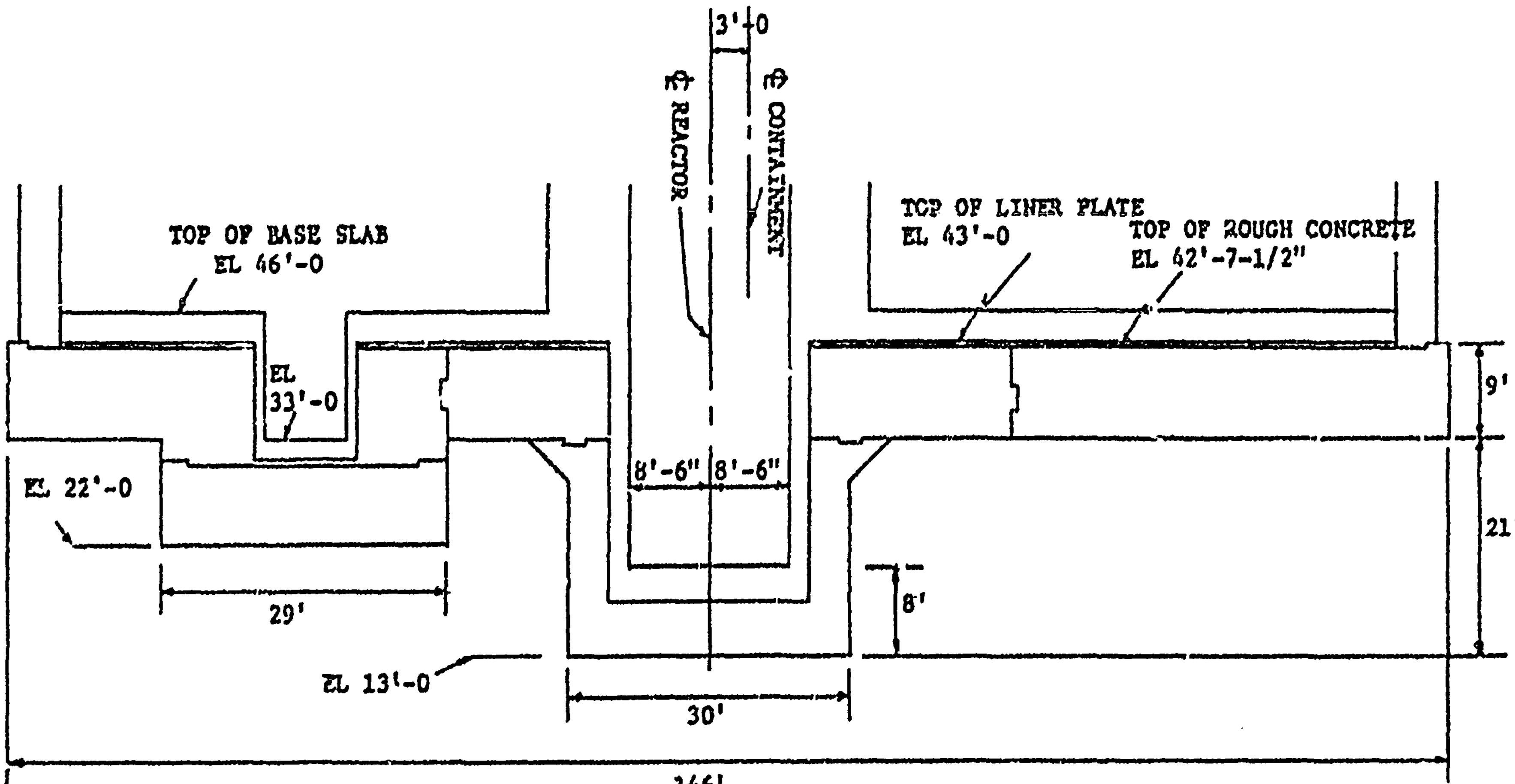
SECTION 1-1



SECTION 2-2

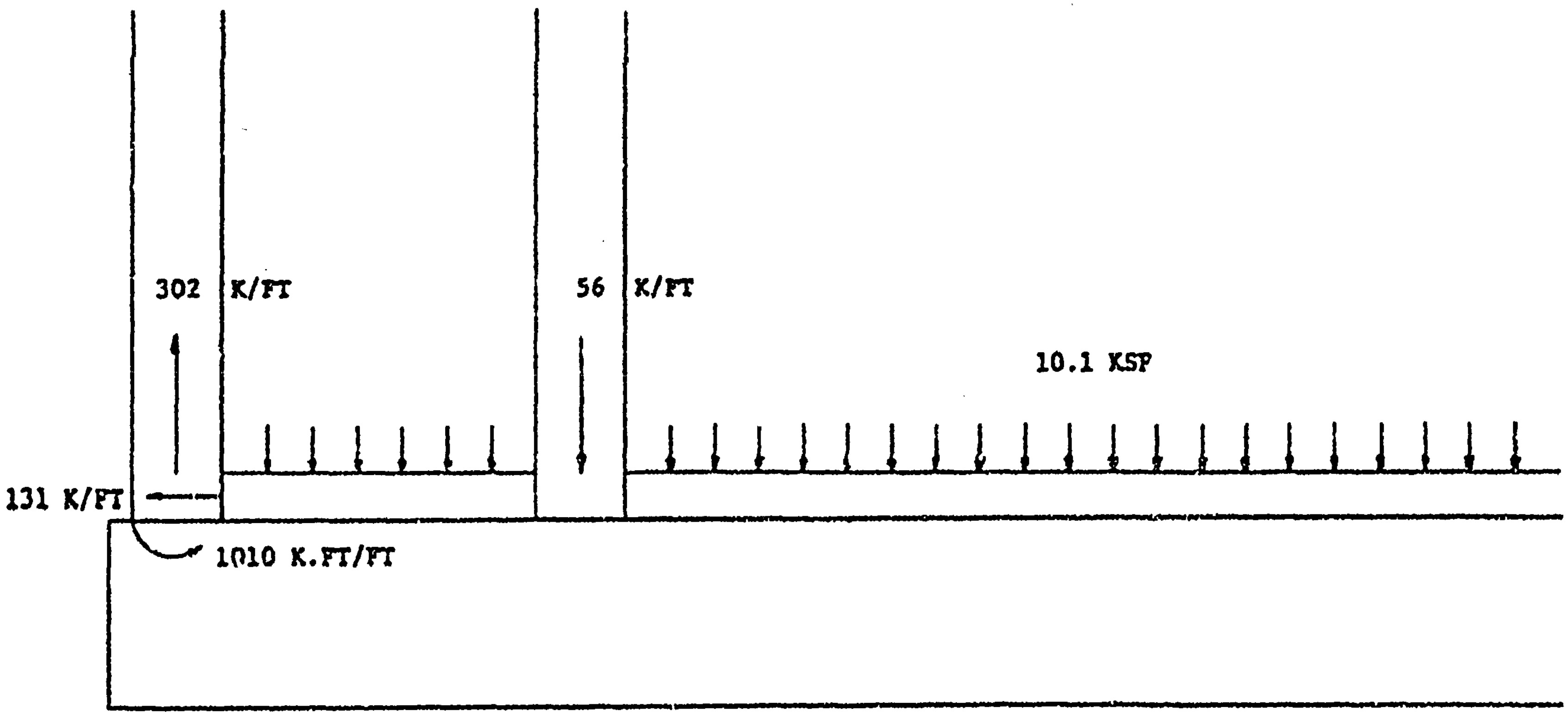
CONTAINMENT INTAKE SUMP

FIGURE 4.5



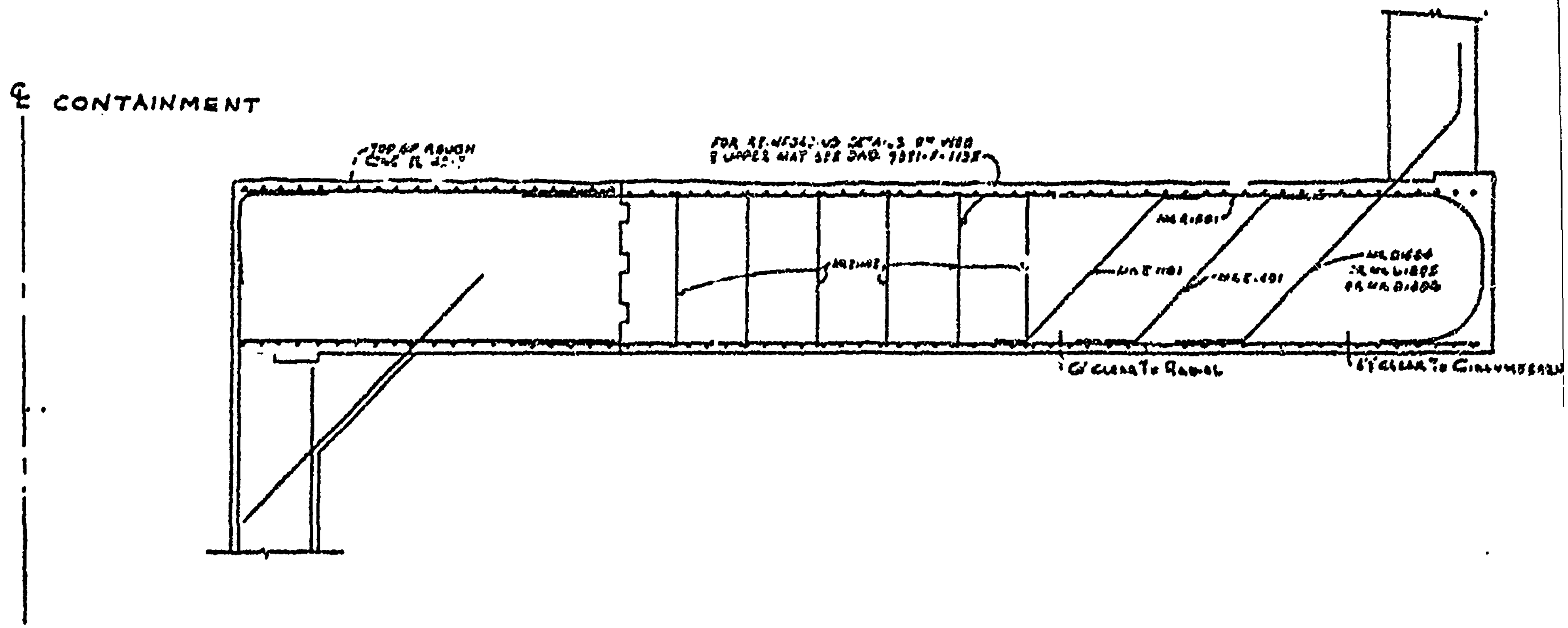
CONTAINMENT BASE MAT

FIGURE 4.6



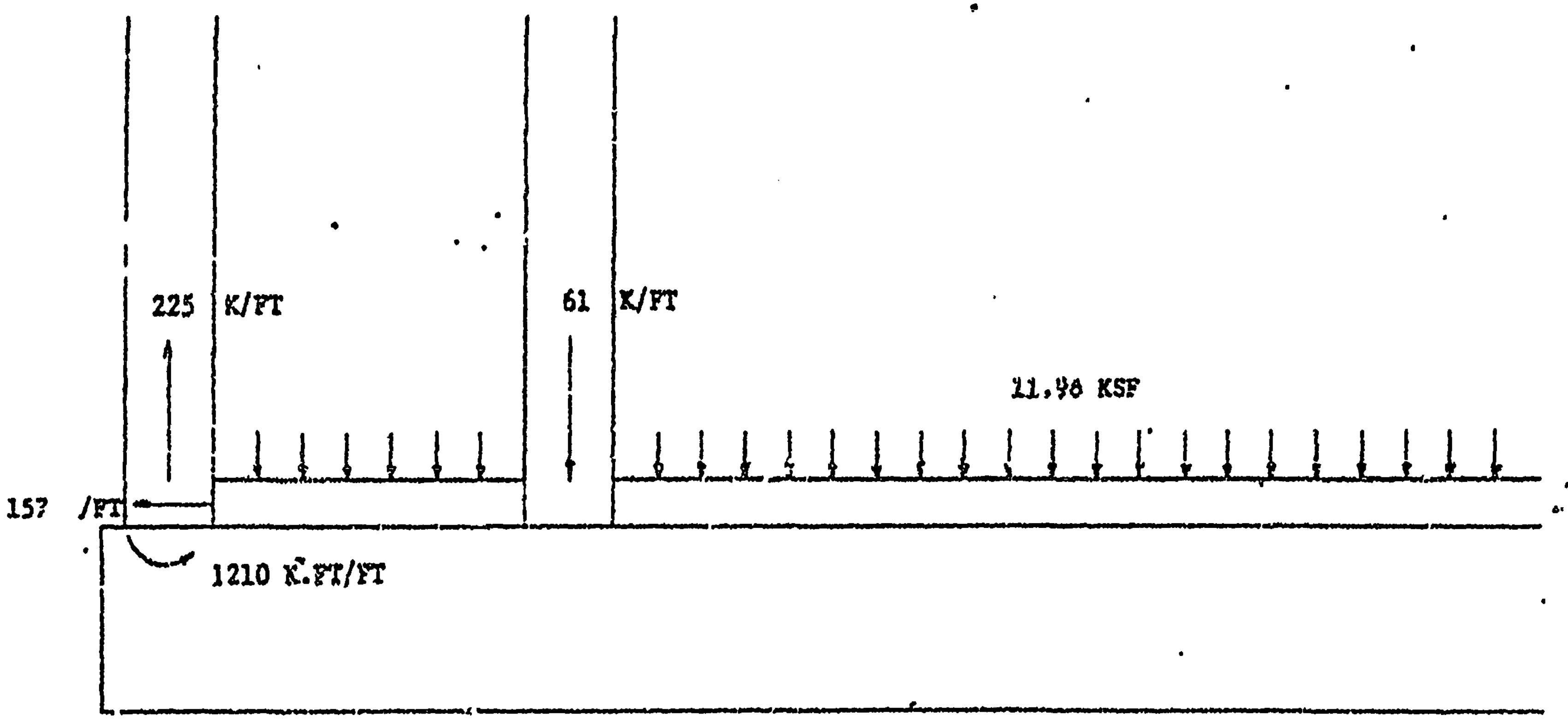
LOADING DIAGRAM IN MAT FOR 1.25 P CONDITION

FIGURE 4.7



SHEAR REINFORCEMENT IN BASE MAT

FIGURE 4.8



LOADING DIAGRAM IN MAT FOR 1.5P CONDITION

FIGURE 4.9

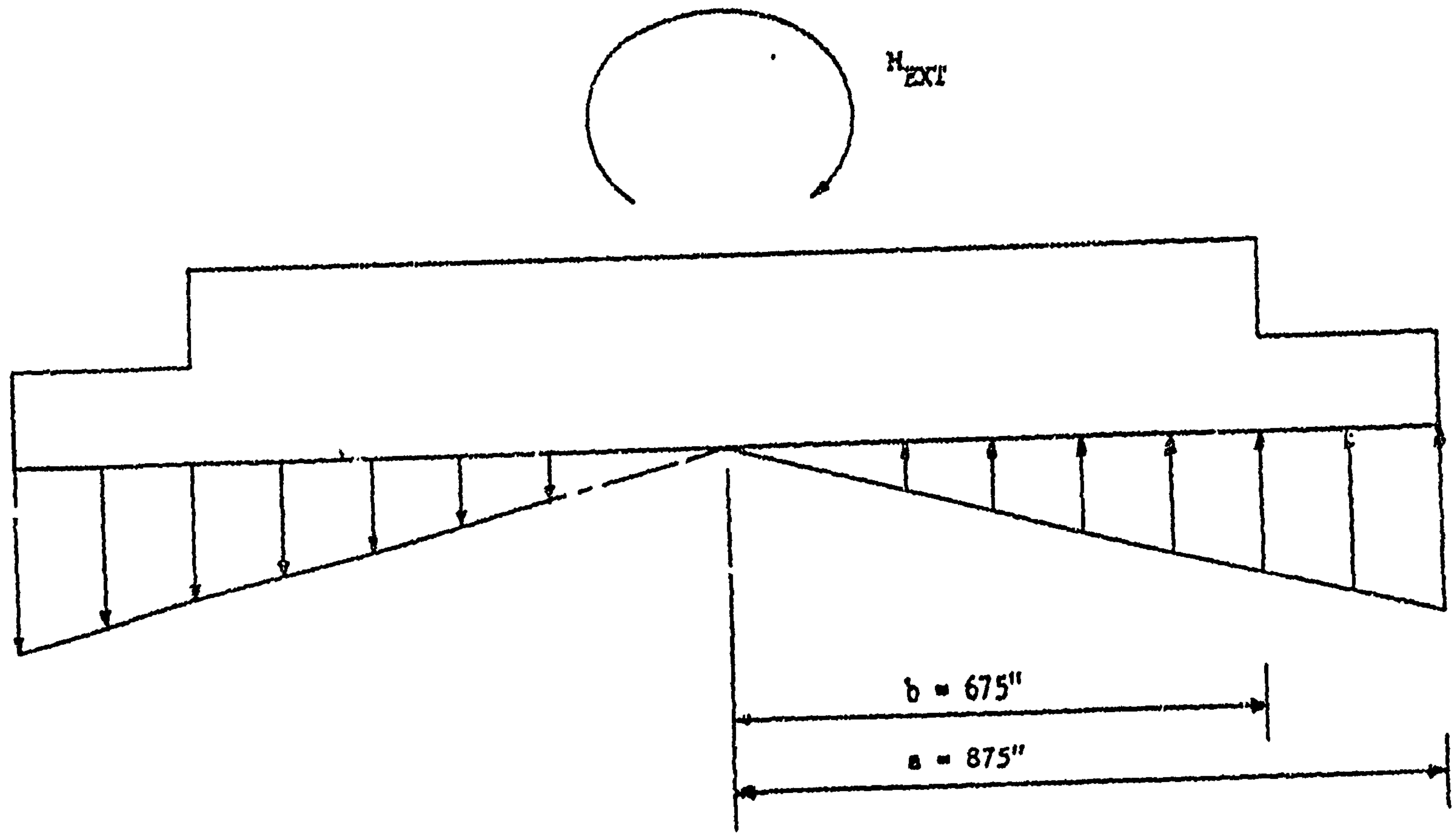
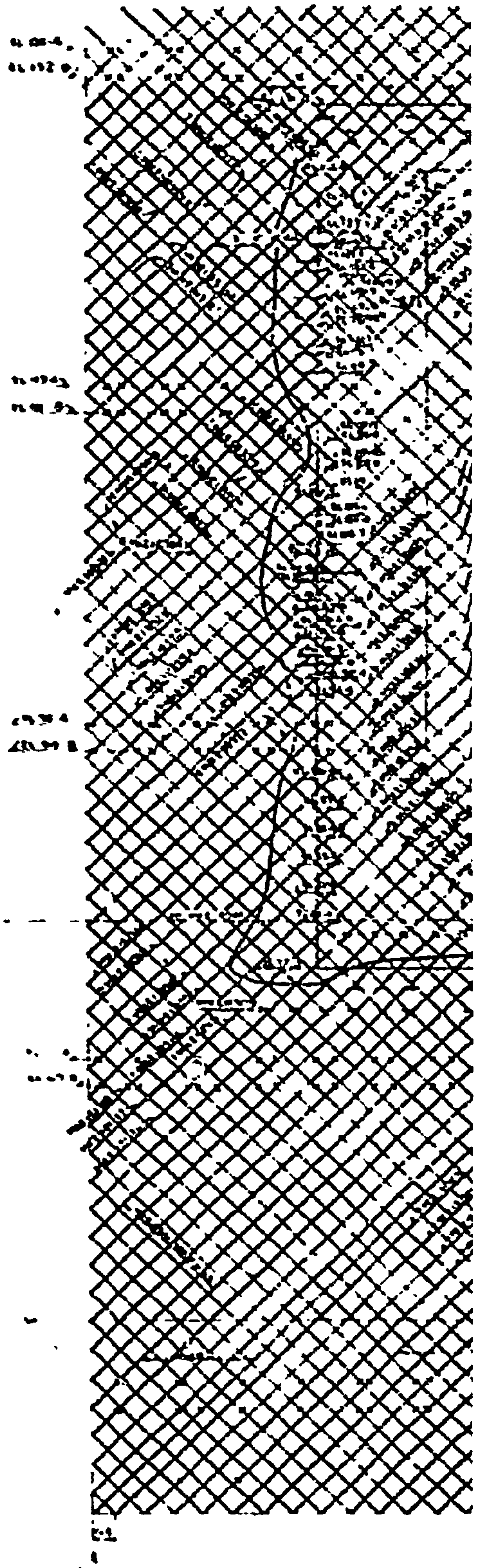
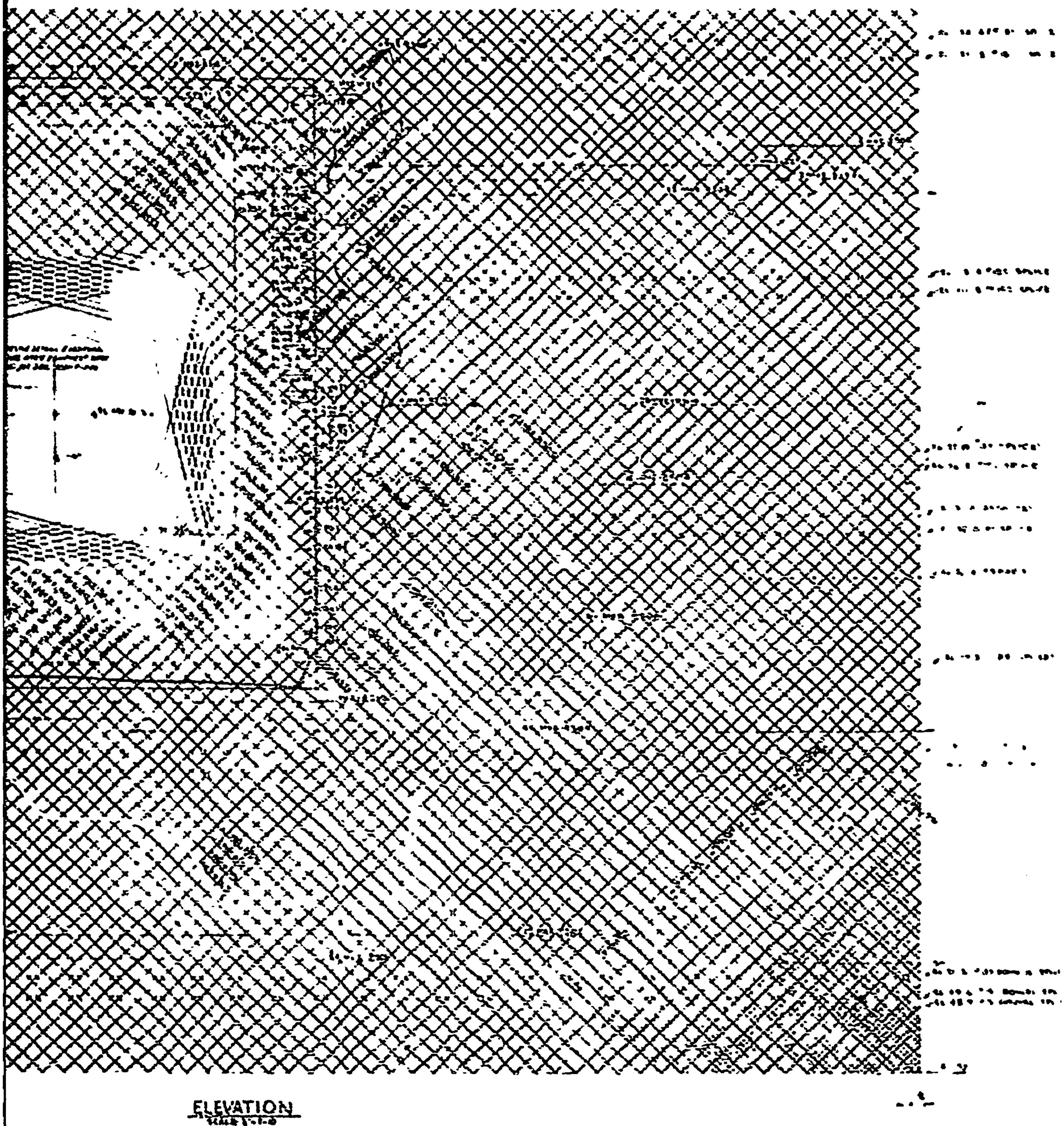


FIGURE 4.10

CIRCULAR AND SOLID FLAT PLATE WITH A CENTRAL RIGID PORTION

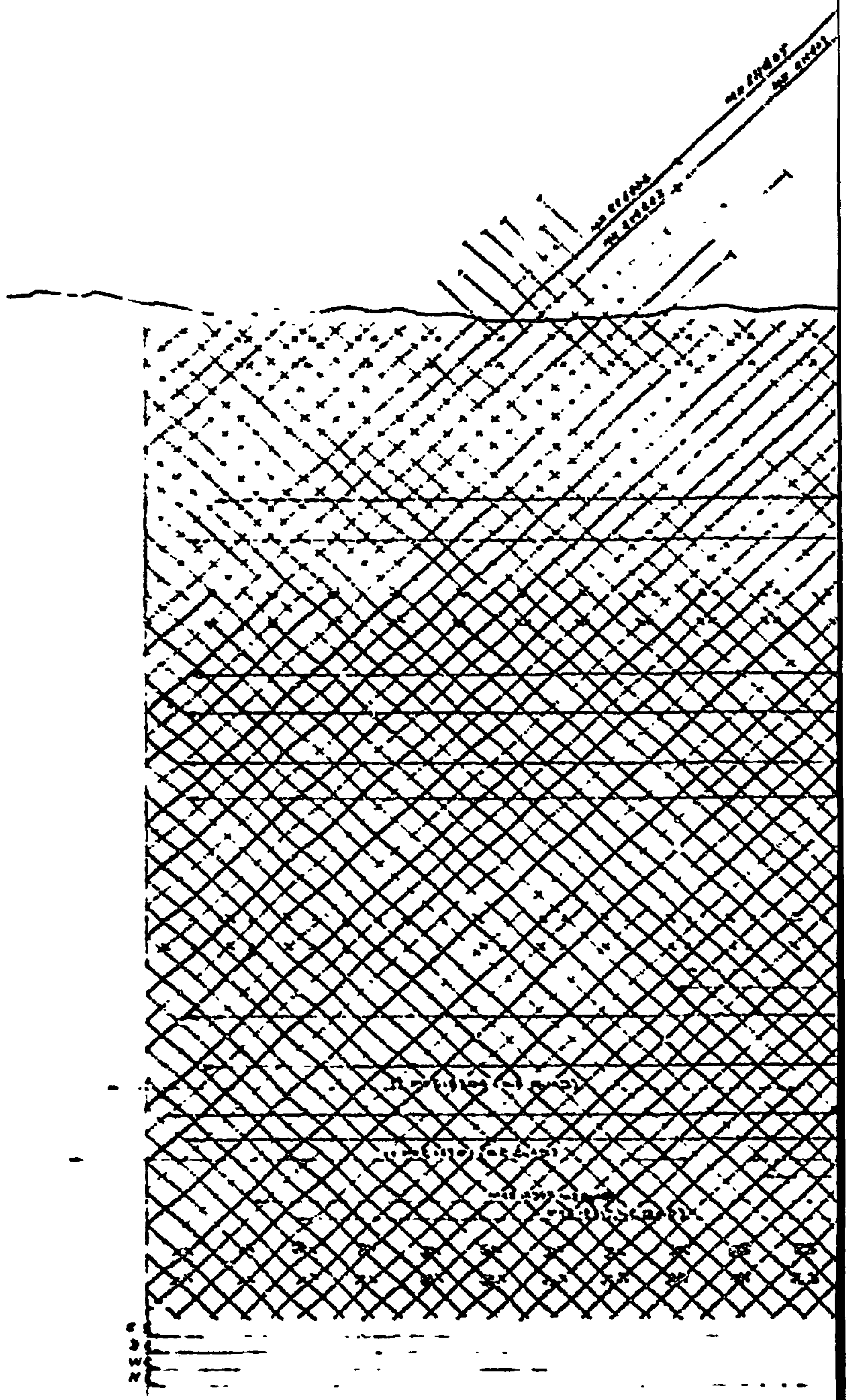




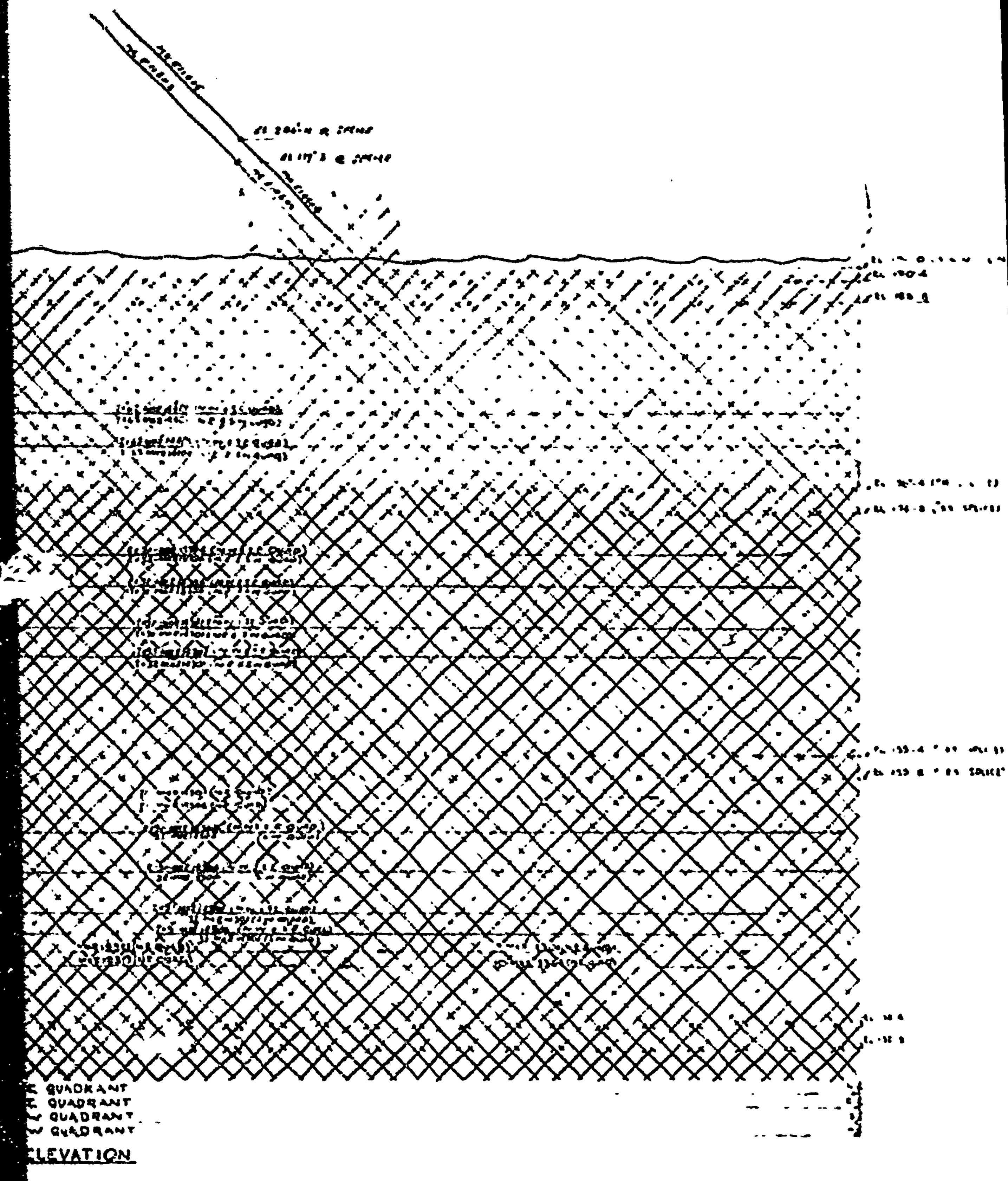
SEISMIC STEEL REINFORCEMENT IN CYLIND

(EL. 43'-0 to EL. 132'-8")

FIGURE 4.11

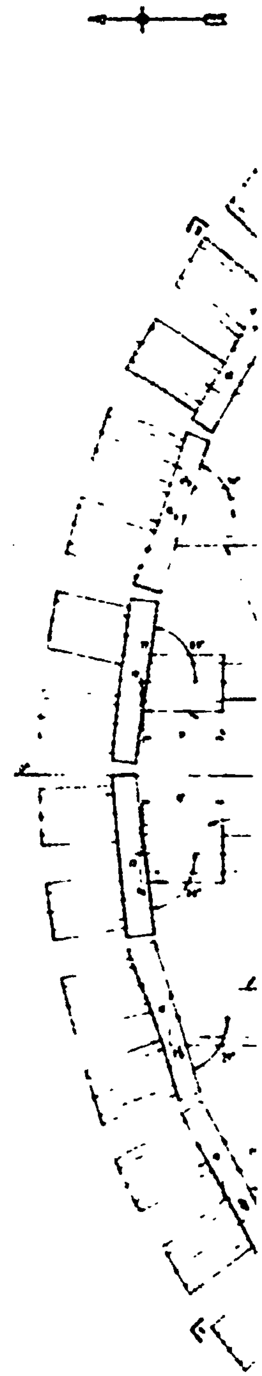


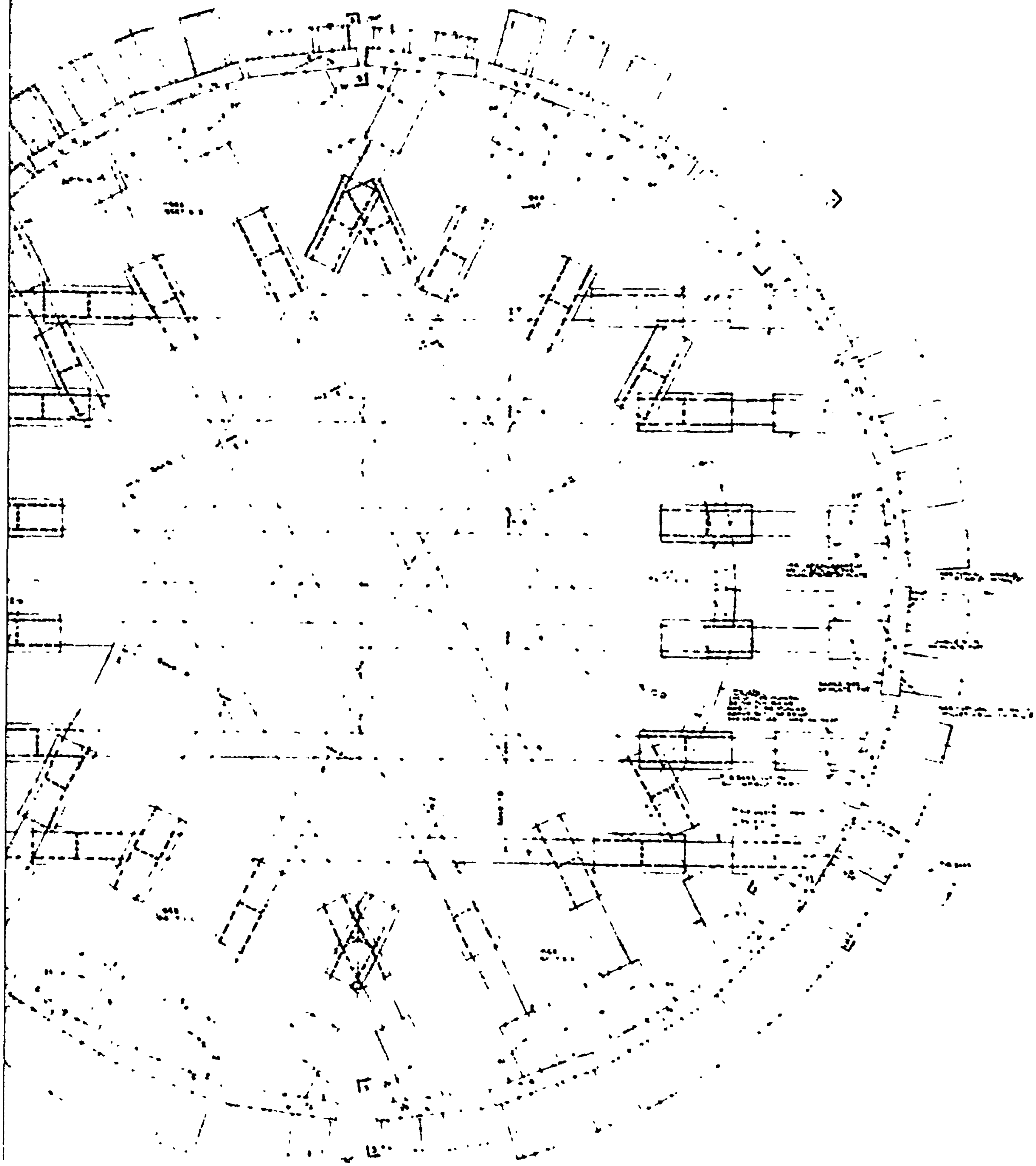
XX INDICATES SPLICES WHICH DO NOT
APPEAR IN N.E. QUAD



SEISMIC STEEL REINFORCEMENT IN CYLINDER
(EL. 132'-8" to EL. 191'-0)

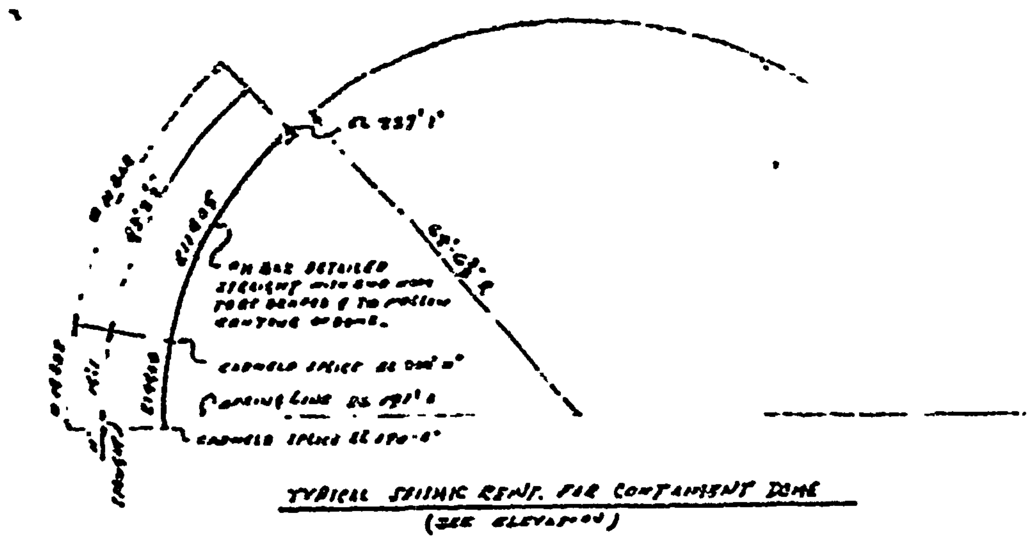
FIGURE 4.12

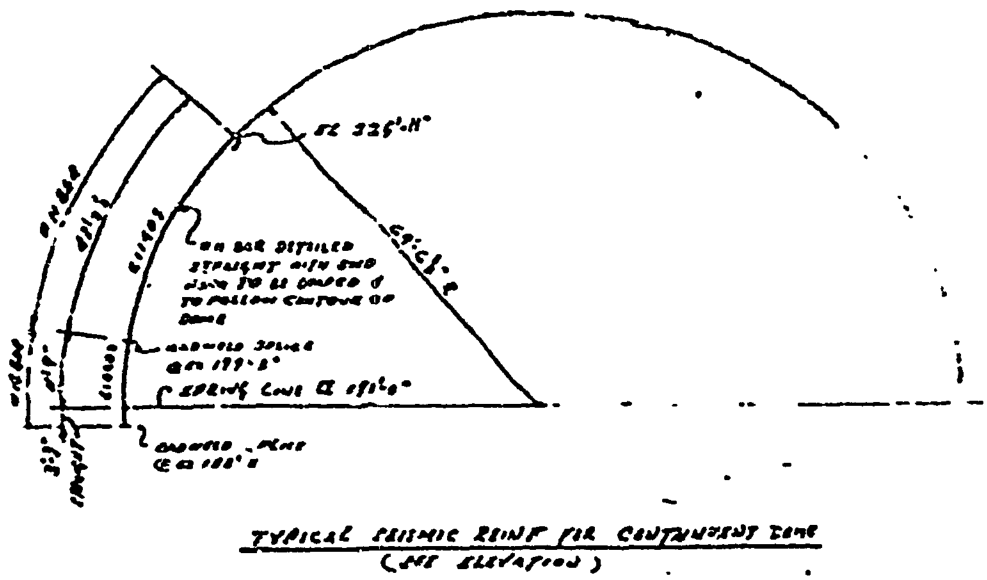


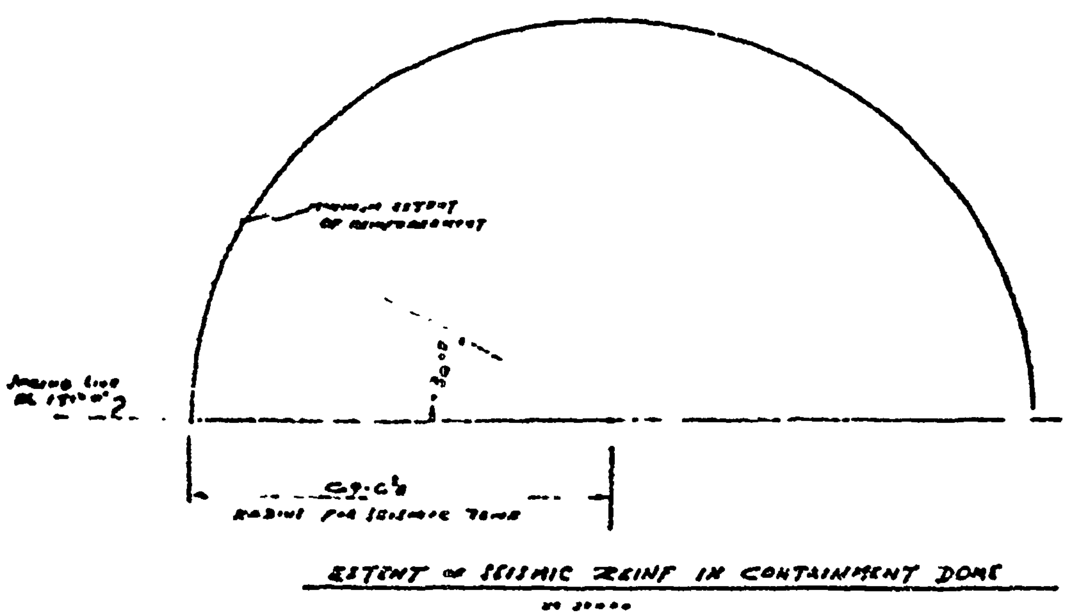


REINFORCEMENT PATTERN AT APEX OF DOME

FIGURE 4.13

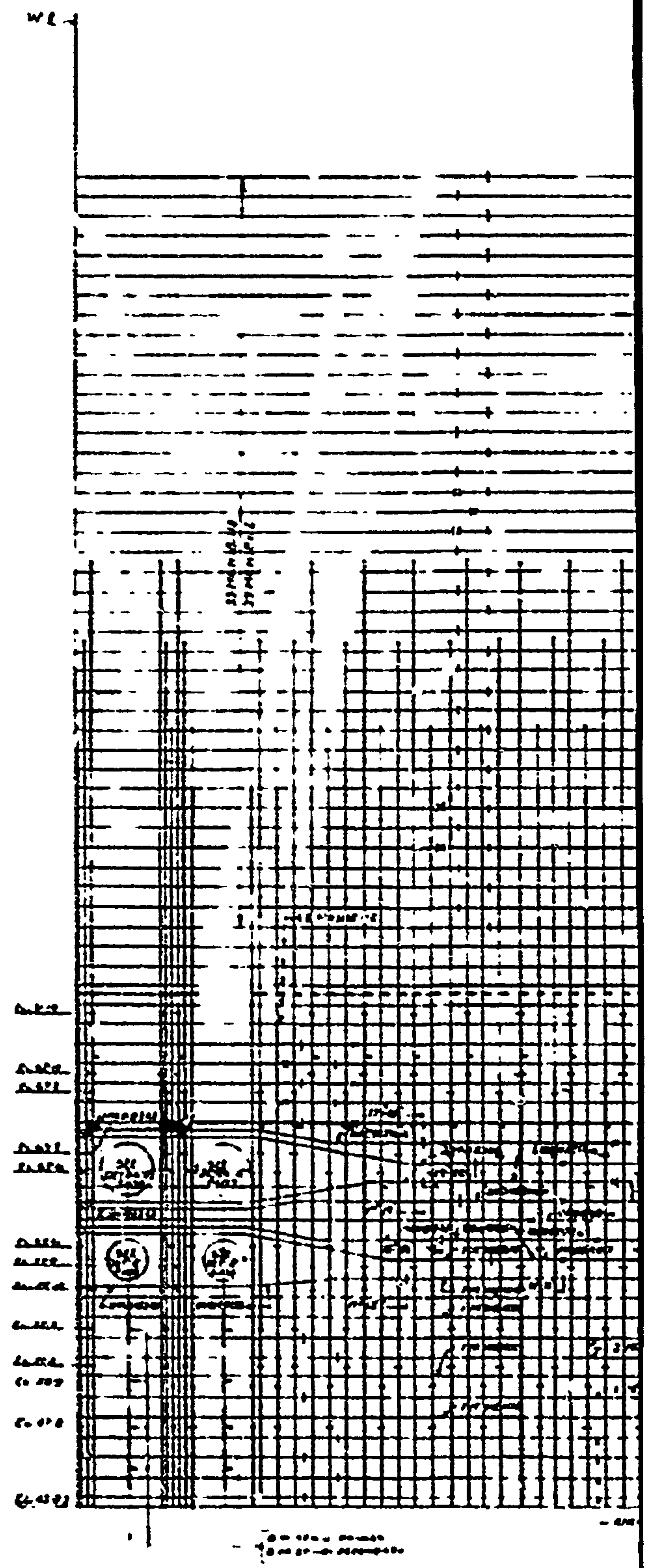


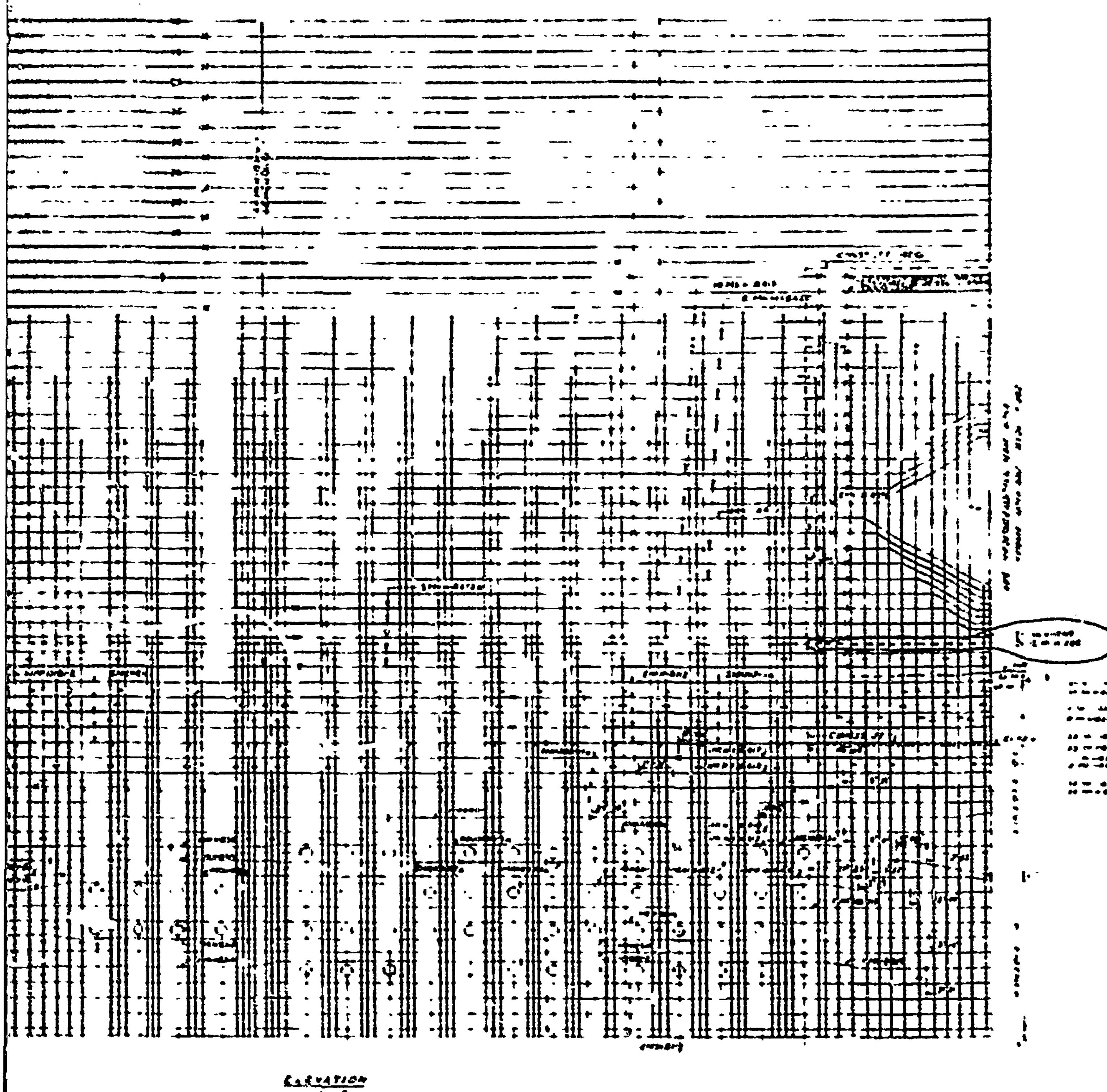




SEISMIC STEEL IN CONTAINMENT DOME

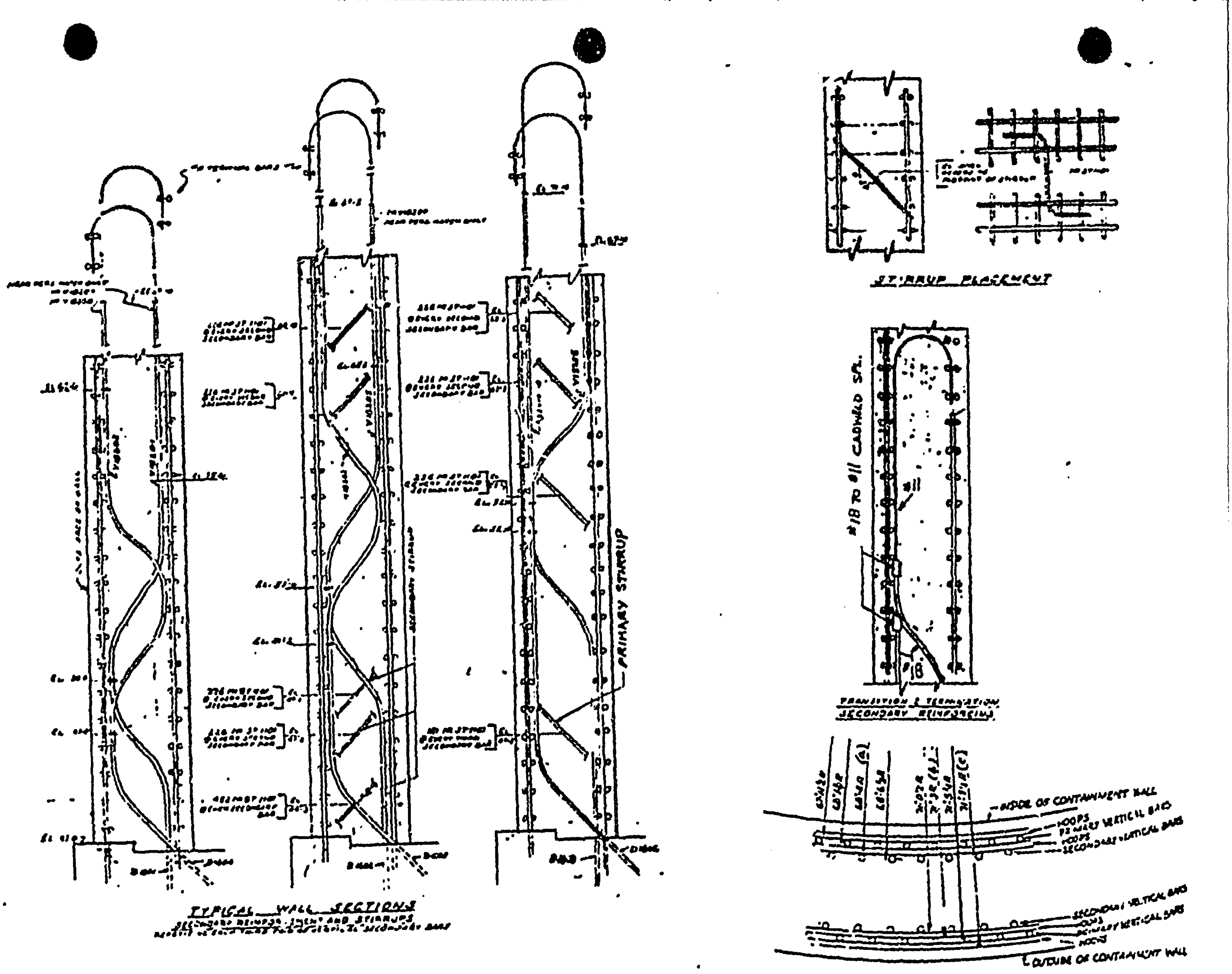
FIGURE 4.1





DISCONTINUITY MOMENT STEEL IN CYLINDER WAL

FIGURE 4.15


TYPICAL WALL SECTIONS
PRIMARY STIRRUP
STIRRUP PLACEMENT
#18 TO #11 CADWELD SPL.
#11
#18
TRANSITION & TERMINATION SECONDARY REINFORCING
INSIDE OF CONTAINMENT WALL
HOOPS
PRIMARY VERTICAL BARS
HOOPS
SECONDARY VERTICAL BARS
SECONDARY VERTICAL BARS
HOOPS
PRIMARY VERTICAL BARS
HOOPS
OUTSIDE OF CONTAINMENT WALL

FIGURE

5.0 CONTAINMENT MATERIAL PROPERTIES, FABRICATION AND ERECTION PROCEDURES

5.1.0 CONCRETE

Concrete used in the containment structure was designed to have a minimum compressive strength in 28 days of 3000 psi. The concrete mix was designed to produce a strength of fifteen percent above the minimum design strength as determined by the average strength of three laboratory tests of the specified design mix including satisfactory plasticity qualities.

The minimum cement factor specified was 5 sks/cu.yd. The maximum slump permitted was limited to 5 inches, except in localized regions of extreme congestion where 7 inch slump was permitted. Concrete was prepared in ready mix equipment conforming to ASTM specifications C94.

5.1.1 CEMENT

The cement used was Portland Cement Type II conforming to ASTM designation C-150. Cement used in the ready mix batch process was stored in weatherproof bins so as to prevent deterioration or contamination.

5.1.2 WATER

Concrete mix water was supplied from the drinking water supply of the city of Verplanck, New York and as such is clean, clear and free of significant impurity. Cloride content of mix water has been maintained below 200 ppm as determined by water sample analysis.

5.1.3 AGGREGATES

Fine aggregate consisted of sand conforming to the requirements of ASTM Specification C-33. The natural sand was supplied by the Southern Dutchess Sand and Gravel Company of Fishkill, N.Y.

Typical properties of the sand are as follows:

SIEVE ANALYSIS		ASTM C-33
Sieve Sizes	% Passing by wt.	Specifications
38"	100	100
#4	97.8	95-100
#8	84.9	80-100
#16	61.8	50-85
#30	42.7	25-60
#50	18.1	10-30
#100	3.3	2-10
Fineness Modulus	2.91	
Specific Gravity (SSD)	2.67	
Absorption %	0.7	
Clay Lumps %	Negative	1.0 Max.
Coal & Lignite %	Negative	0.5 Max.
Material Finer than No. 200 Sieve %	0.6	3.0 Max.
Organic Impurities	Standard	Standard
Soundness 5 Cycles, % Loss	10.9	
Unit wt. (dry-rodded) lbs/ft^3	104.3	

Coarse aggregate consisted of crushed gravel conforming to the requirements of ASTM Specification C-33 and was supplied by the Southern Dutchess Sand and Gravel Company of Fishkill, N. Y. Typical properties of the crushed grave are as follows:

COARSE AGGREGATE

30% - 40% Crushed Gravel - Southern Dutchess S&G Fishkill, N.Y.

SIEVE ANALYSIS Sieve Sizes	% Passing by wt.	ASTM C-33 Specification
1 1/2"	100.0	100
1"	97.2	95-100
3/4"	71.5	-
1/2"	30.9	25-60
3/8"	12.4	-
#4	0	0-10
Fineness Modulus	7.16	
Specific Gravity	2.67	
Absorption %	0.7	
Clay Lumps %	Negative	0.25 Max
Soft Particles %	Negative	5.0 Max
Unit Wt. (dry-rodded) lbs/ft^3	102.2	
Magnesium Sulfate Soundness		
5 Cycles, % loss	14.8	18 Max
Los Angeles Abrasion, % loss	41.7	50 Max

5.1.4 ADMIXTURES

The only admixture used in the concrete mix design was a plasticizer "Placewell" manufactured by the Union Carbide Corporation. The plasticizer is provided in the proportion of 16.5 oz/cu.yd. of concrete to increase ease of concrete placement in highly congested areas. No other admixture in the form of air entraining agents, set retarders or set accellerators have been us

5.1.5 PLACEMENT AND CURING

Placing and Curing of concrete conform to the provisions of Chapter 6 of the ACI 318-63.

5.2.0 REINFORCING STEEL

Reinforcing steel used for the dome, cylindrical walls and base mat is high-strength deformed billet steel bars conforming to ASTM Designation A-432 "Specification for Deformed Billet Steel Bars for Concrete Reinforcement with 60,000 psi Minimum Yield Strength." This steel has a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi, and a minimum elongation of 7 per cent in an 8-in. specimen. The design limit for a tension member (i.e., the capacity required for the design load) was based upon the yield stress of the reinforcing steel. No steel reinforcement experiences average strains beyond the yield point at the factored load. The load capacity so determined has been reduced by a capacity reduction factor "Ø" which provides for the possibility that small adverse variations in material strengths, workmanship, dimensions, and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in under capacity. For tension numbers, the factor "Ø" was 0.90 for flexure and 0.85 for diagonal tension, bond and anchorage.

5.2.1 CADWELD SPLICES

All reinforcing bar design to carry membrane tension or in the size range 14S and 18S where jointed by means of mechanical butt splices known as a Cadweld splice which is a standard commercial product manufactured by Erico Products Inc., Cleveland, Ohio. All splices used were either type T-1891, T-18101, or T-1491, T-14101, which are designed to develop the specified minimum ultimate strength of the ASTM A-432 reinforcing bar or greater even though the specified requirement on splice strength was set at 125 per cent of specified minimum yield (83.3 per cent of minimum ultimate).

All splices were made in accordance with the recommendations of the manufacturer and included the following specific requirements:

1. Bar ends to be spliced were squared by torch cutting, shearing or sawing.
2. Bar ends were cleaned by means of a wire brush of dirt, oil, moisture, concrete and heavy rust.
3. Preheating of the bar to a temperature 100°F if air temperature was below 40°F.

After approximately 30 percent of the total number of splices were placed in the containment a more detailed procedure was developed which included the following additional provisions:

1. Bar ends were cleaned to a degree of cleanliness as represented by heating the bar uniformly to surface temperature of 200°F to 300°F, power wire brushing to bare metal, and wire brushing to remove any resulting dust or loose material.
2. No manufacturer's mill marks were included on that portion of the re-bar inserted within the splicing sleeve.
3. A punch mark was made in reinforcing bar 12 inches from bar ends to insure proper entering of bars in the splice.

5.3.0 FORMWORK

Concrete form work was erected to conform to the shape, lines and dimension of the concrete elements as called for on the drawings and sufficiently tight to prevent leakage of mortar.

For all permanently exposed surfaces of concrete the form facing was constructed of new unscarred plywood, re-used plywood in good condition or metal pans. Forms were removed in such a manner and at such a time as to insure the complete safety of the structure as determined by concrete cylinder tests of pours for the concrete sections being stripped.

5.4.0 WATERPROFING CONTAINMENT WALLS BELOW GRADE

All areas of the containment shell in contact with backfill were water proofed. Water proofing consisted of a 5/8 inch thick hardboard insulation (Homasote) placed between two coats of bitumastic. Both coats of bitumastic No. 50 were applied at a rate of 55-70 sq. ft. per gallon. All joints of the insulation were butted with a 1/4" maximum gap. Preparation and application of waterproofing material was in accordance with manufactured printed instructions. The bitumastic coating extended 6 inches beyond finished grade while the insulation was cut off 1 foot below grade.

5.5.0 CONTAINMENT LINER

5.5.1 MATERIAL

The steel liner plate is carbon steel conforming to ASTM Designation A-442 "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 percent 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 3/4-in. thick at penetrations and 3/8-in. thick for remaining portion of the cylindrical walls and 1/2-in; thick in the dome. The liner material was impact tested at a temperature 30°F. lower than the minimum operating temperature of the liner material. For the liner steel the factor "Ø" was 0.95 for tension.

5.5.2 FABRICATION

The steel liner plate was fabricated from hot rolled plate in the Greeneville, Pennsylvania and New Castle, Delaware shop of the Chicago Bridge and Iron Co. The plate was shop fabricated into approximately 9' by 30' section and rolled to desired curvature. The Nelson stud anchors were welded to the containment liner shell after the plate was erected.

5.5.3 ERECTION

The difference between the minimum and maximum inside diameters at any cross section does not exceed 0.25 percent of the nominal diameter at the cross section under consideration. Maximum diameter 135'-2, minimum diameter and 134'-10" below elevation +95. Above +95 tolerance does not exceed .50 percent of the nominal diameter of cross section under consideration. The liner was erected true and plumb not to exceed $\frac{1}{500}$ of height at cross section under consideration with allowance for 2" buckling in the plates.

Particular care was taken in matching edges of cylindrical and hemispherical sections to insure that all joints were properly aligned. Maximum permissible offset of completed joints was 25 percent of nominal plate thickness.

5.6 LINER INSULATION

The liner insulation consists of 1-1/4 in polyvinylchloride insulation as manufactured by Johns-Mansville Corporation. The insulation was fabricated into 44" x 84" flat panels. The insulation panels were attached to the steel containment liner by means of 3/16 in. stainless steel studs welded to the liner on the basis of 6 per panel. Insulation adhesive was FNIZ Adhesive as manufactured by [illegible] Brand Division of Johns-Manville Corporation. The insulation panels were protected by .019 in. stainless steel jacketing on the exposed face.

The insulation was designed to meet the following operational requirements:

1. Normal operating temperature - 120°F.
2. Under accident conditions rise in liner temperature not to exceed 80°F.
3. Insulation panels rated non-burning in accordance with ASTM procedure D-1692.

5.7.0 PENETRATIONS

In general containment penetrations for pipe, electrical conduit, duct or access hatches consist of sleeves imbedded in the concrete section and welded to the containment liner. The weld to the liner is shrouded by a continuously pressurized channel which is used to assure the leak tightness of the penetration to liner weld joint. Differential expansion between sleeve and pipes passing through is accommodated by bellows type expansion joints between the outer end of the sleeve and the outer plate.

5.7.1 Materials

The materials for penetrations including the personnel and equipment access hatches, together with the mechanical and electrical penetrations will be carbon steel, conform with the requirements of the ASME Nuclear Vessels Code and exhibit ductility and welding characteristics compatible with the main liner material. As required by the Nuclear Vessels Code, the penetration materials meet the necessary Charpy V-notch impact values at a temperature 30°F below lowest service metal temperature which is 50°F within containment and -5°F outside the containment.

The stainless steel bellows of the hot penetration expansion joints will be protected from damage in transit and during construction by sheet metal covers fastened in place at the fabricator's shop. These can be left in place permanently if there is no interference with nearby piping or equipment.

The specific materials used in penetrations may be found in Section 2.2.6.

5.7.2 Design

Those portions of penetrations not backed up by concrete are designed to meet the requirements of ASME Code Section VIII. Those portions of penetrations backed up by concrete are designed considering strains and stresses compatible with the deformation of the concrete wall sections and as such have the same governing design criteria as does the containment liner. As such no primary load strains greater than the guaranteed yield point under factored loads are permitted. However, strains due to stress concentrations and other localized secondary load effects are limit to 0.5 percent strain.

5.7.3 Fabrication

Th qualification of welding procedures and welders has been in accordance with Section IX, "Welding Qualifications" of the ASME Boiler and Pressure Vessel Code. The repair of defective welds has been in accordance with Para. UW-38 Section VIII "Unfired Pressure Vessels."

6.0 QUALITY CONTROL METHOD AND PREOPERATIONAL TEST PROCEDURE

6.1.0 QUALITY CONTROL ORGANIZATION AND CHAIN OF COMMAND

The responsibility for implementation of the on-site quality control program for UE&C rests with the Field Supervisor - Quality Control who reports directly to the Manager of Reliability and Quality Assurance Department in the home office who in turn reports directly to the Vice President - Administration.

Reporting directly to the UE&C Field Supervisor - Quality Control at the project site are Quality Control Engineers assigned primarily to a specific discipline (e.g., structural, mechanical, electrical, and piping/welding), Quality Control Inspectors, clerks, and subcontracted testing service personnel.

No one in this quality control chain of command is directly responsible for production or schedules.

The UE&C Quality Control group and/or the subcontracted testing service personnel conduct the first level inspection and tests of all construction of structural elements of the vapor containment building except for the field fabrication and erection of the containment liner, penetrations where the construction subcontractor has first level responsibility subject to audit and surveillance by the UE&C Field Quality Control group.

In all cases, all quality control activity is audited by the Prime Contractor, Westinghouse Electric Corp., the Owner, Consolidated Edison Co., and the Owner's Surveillance Group, United States Testing Laboratories.

All necessary records and documentation are compiled and maintained by the UE&C Site Quality Control Group.

6.2.0 SUMMARY OF MATERIAL TEST RESULTS

6.2.1 CONCRETE

A minimum design strength of 3000 psi is specified. The average of all 28 day strength tests to date is 3705 psi. To date no test cylinders strengths under 3000 psi 28 day strength have been determined. One set of two test cylinders to include a 7 day and 28 day test cylinder has been tested per each 100 cubic yards placed. Approximately 18,000 cubic yards of concrete have been placed in the containment structure to date with a total pour of 20,000 cubic yards expected.

6.2.2 REINFORCING STEEL

Material mill test reports are required for each heat of steel received. Results of all tests show conformance with ASTM specification requirements. In addition to the mill test reports a total of 18 user tests have been performed on the bar received. All tests have met minimum specified strength requirements. The average yield strength of user's tests is 64.8 ksi and average ultimate strength is 101 ksi for the ASTM A 432 steel (f_y = 60,000 psi; f_u = 90,000 psi).

6.2.3 STRUCTURAL STEEL

Various types of structural steel were furnished and erected. Structural steel was furnished to ASTM Specification in job lots substantiated by mill certification covering each job lot.

6.2.4 INSULATION

No insulation on site as of this date. Letters of certification covering material requirements substantiated by test results are being furnished by the manufacturer.

Installation of insulation will be checked by UE&C Quality Control to assume tight joints as well as material dimensions and insulation studs.

6.2.5 CONTAINMENT LINER

All heats of steel used in the fabrication of the liner plate are covered by mill test certificates showing chemical analysis, mechanical test results, and Charpy impact test results.

Each liner plate is marked or coded to a specific heat of steel. These heat numbers are recorded on the as-built drawings. Material control (heat number continuity) is maintained by subcontractor and checked by UE&C Quality Control.

The same method of heat identification, certification, and recordation is maintained for the penetration material as for the liner plate.

Weld rod control (only E 7018 rod used on liner plate) is maintained by subcontractor and audited by UE&C Quality Control.

Dimensions of erected material were checked by the UE&C field survey group, recorded on marked-up drawings in the quality control file. Any dimension found ou of tolerance is reported to the subcontractor, corrected and rechecked by the survey group.

6.3.0 QUALITY CONTROL TESTS ON FABRICATED ELEMENTS

6.3.1 LINER, PENETRATIONS, LOCKS, AND EQUIPMENT HATCH

Nondestructive testing of these items consists of the following:

Coupon Testing - In locations on the liner where radiography is not possible, such as floor plates, and lower courses of the shell where back-up plates are used, the subcontractor welds a 2" long overrun coupon which is broken off, marked for location and given to UE&C for destructive examination or radiography.

a. Vacuum Box Test

Bottom liner plate welds and all liner plate seam welds in the cylindrical walls and dome are vacuum box tested with at least a 5 psi pressure differential by the subcontractor. No leaks are permitted.

b. Strength Tests

After successful vacuum box testing all liner plate welds (bottom, cylinder, and dome) channels are welded on the seam wel' and the channel welds by pressurizing the channel with air at 54 psig for 15 minutes. No leaks are permitted. Strength testing shall be by predetermined zones, and includes channels and gaskets of the personnel locks.

c. Leak Test

After strength tests of liner seam welds and channels, these welds and penetration sleeve weld channels, and personnel lock weld channels are leak tested by pressurization to 47 psig with a 20% by weight Freon-air mixture. The entire run of plate weld and the channel to plate welds are then traversed with a halogen leak detector.

The sensitivity of the leak detector is 1×10^{-9} standard cc per second. Any halogen indication indicates a leak requiring repair and retest. In addition, the zone of channels tested is held at test pressure for at least 2 hours, with no indication of drop in pressure.

The strength and leaks tests are also performed on the gaskets and seals on the lock penetrations by pressurizing the space between the gaskets and seals as above.

6.3.2 CADWELDS

All Cadwelds are visually inspected by the UE&C Quality Control group on site. Details of Cadwelding operations, operator qualification criteria, testing frequencies and criteria, inspection procedures and acceptance standards are included in Appendix C. "Summary of United Engineers & Constructors Inc. Experience in Utilizing Cadweld Reinforcing Bar Splices."

This Appendix shows current probability of all splices in the structure exceeding 60,000 psi at approximately 0.998 and probability of all splices exceeding 75,000 psi (125% min. yield) at approximately 0.990.

6.3.3 STUD ANCHORS ON THE LINER

A procedure is set up whereby after qualification, the first stud welded each day by each welder is tested by cold bending the stud to an angle of 45°. This test is repeated after the lunch break.

6.4.0 PREOPERATIONAL PERFORMANCE TESTING

After completion of the vapor containment structure the building will be pressurizer to 54 psig with increments at 18 and 36 psig as described in the Section 5 of the FSAR.

Strain gauges will have been attached as called for on the approved drawings. Gauges are sufficiently redundant to overcome possible construction damage. As a minimum each gauge shall have one gauge as back-up.

Strains and deformations will be recorded at each increment of pressure and the corresponding stresses calculated for the reinforcing and liner.

After the structure has been proof-tested at 54 psig for a minimum of one hour, the pressure shall be reduced to 47 psig and held for 24 hours.

Instrumentation will have a capability of measuring strains from 0 to 0.003 inches per inch with an accuracy of $\pm$ 0.1%.

During the 24 hour hold at 47 psig the structure will be investigated for cracks and manual measurements recorded.

In the quadrant of "boss" around the equipment hatch and personnel lock and in the 10' width between elevation 43'-0" and 73'-0", concrete surface will be whitewashed and detailed measurements of crack width and spacing shall be recorded. A detailed test procedure is currently being prepared.

6.5.0 FIELD PROBLEMS ENCOUNTERED IN CONSTRUCTION

During the construction of the vapor containment structure, two conditions were found to exist which were determined to be problem areas. These were "Cadwelds" and a "Liner Bulge."

In the case of the Cadweld problem, a declining in the strength of routine samples tested was noted. When the downward trend continued and approached the limit of design requirements all Cadwelding was stopped. An engineering investigation and testing program followed which determined the cause of the problem, verified the existing splices, produced new procedures for Cadweld operations, and outlined statistical controls to verify confidence levels in future work. A detailed summary of the situation may be found in the attached Appendix C "Summary of United Engineers & Constructors Inc. Experience in Utilizing Cadweld Reinforcing Bar Splices."

In the case of the liner bulge, during a routine inspection of the liner by quality control personnel, a buckle was noted in the liner plate near the fuel transfer tube between elevations 56'-76" and 59'-7".

An evaluation of the condition was made and corrective action taken to eliminate the problem.

A detailed report of the situation is contained in Appendix D Report on the Containment Building Liner Plate Buckle in the Vicinity of the Fuel Transfer Canal.

7.0 SUMMARY AND CONCLUSIONS

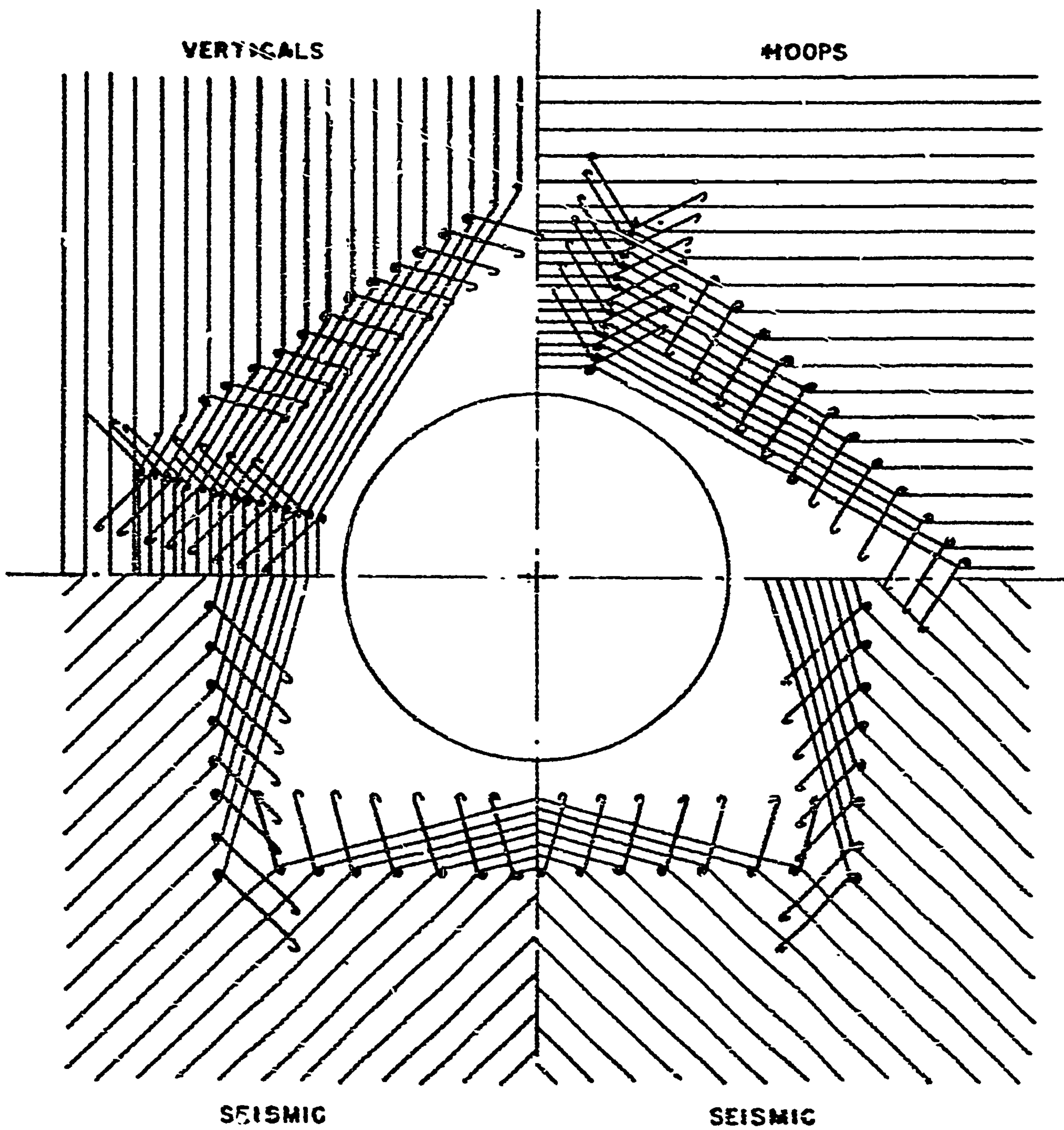
In this report are found the essential elements on which the structural design adequacy of the Indian Point Generating Station Unit No. 2 is based. These essential elements of design can be categorized as follows:

1. Design load assumptions and criteria.
2. Design stress strain or behavior limits criteria.
3. Development of design equations based on recognized conservative behavior models for the various structure elements which make up containment.
4. Limiting assumptions and parameters used in conjunction with the design equations.
5. Comparison of computed stresses or strains to limiting criteria to demonstrate adequacy.

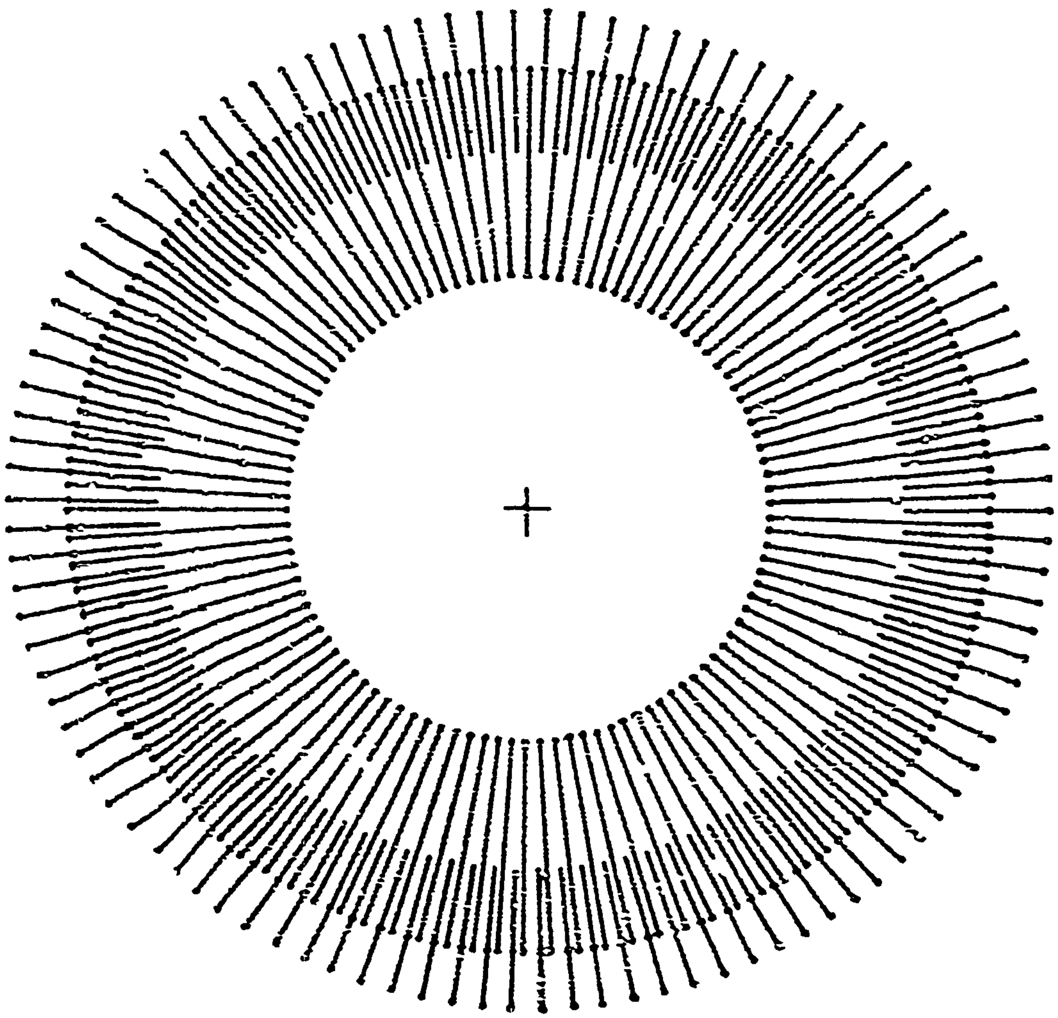
In addition to the verification of design adequacy this report presents or references those features of containment detailing, fabrication and erection to include quality control and assurance which demonstrate that the containment structure was constructed to the highest standards of nuclear industry practice.

It is the opinion of the authors that this report presents in concise form sufficient design and construction information to clearly illustrate the as built integrity of the Indian Point Generating Station Unit No. 2 Containment. Complete records concerning containment design, detailing fabrication and erection can be found in the files of United Engineers and Constructors, Inc., Philadelphia, Pa.

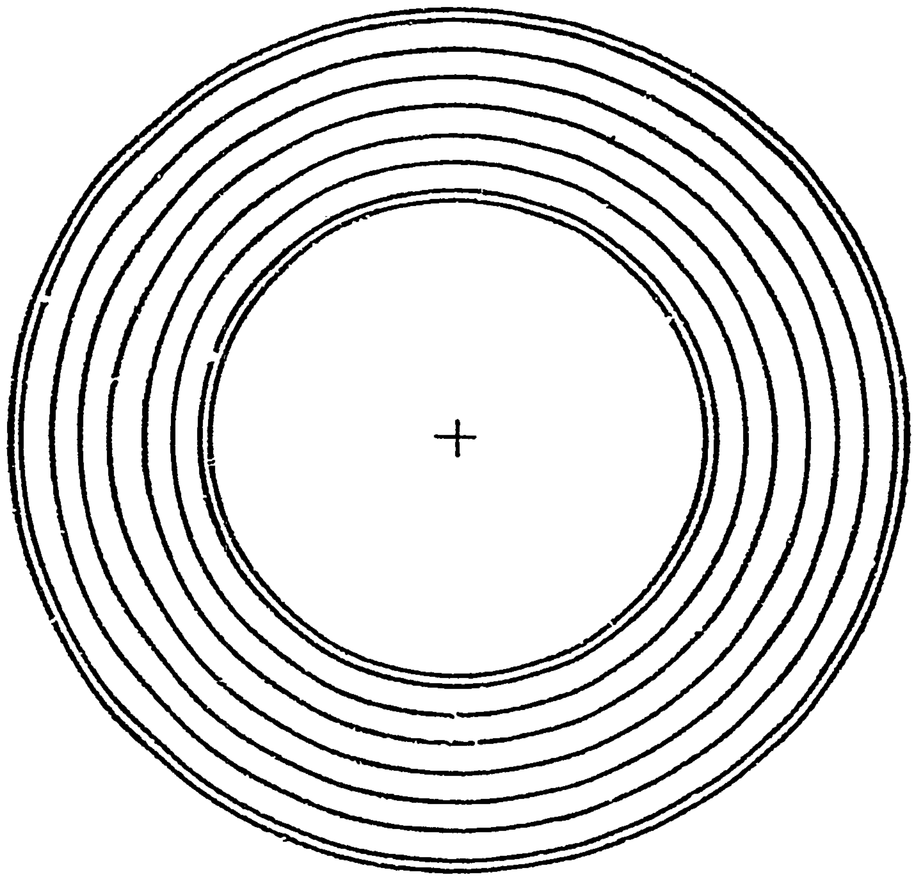
APPENDIX A



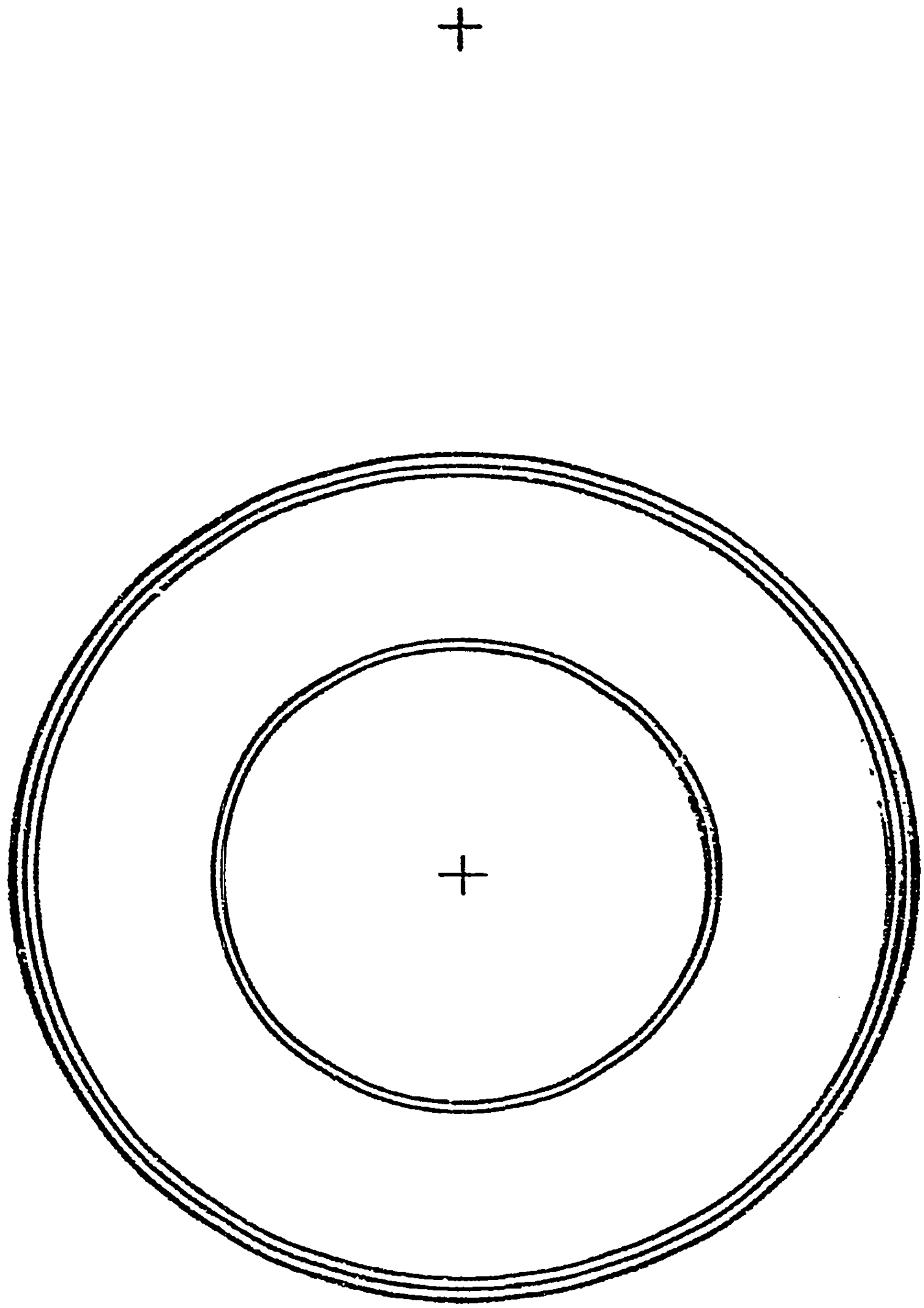
TYPICAL TIE-BACK BARS



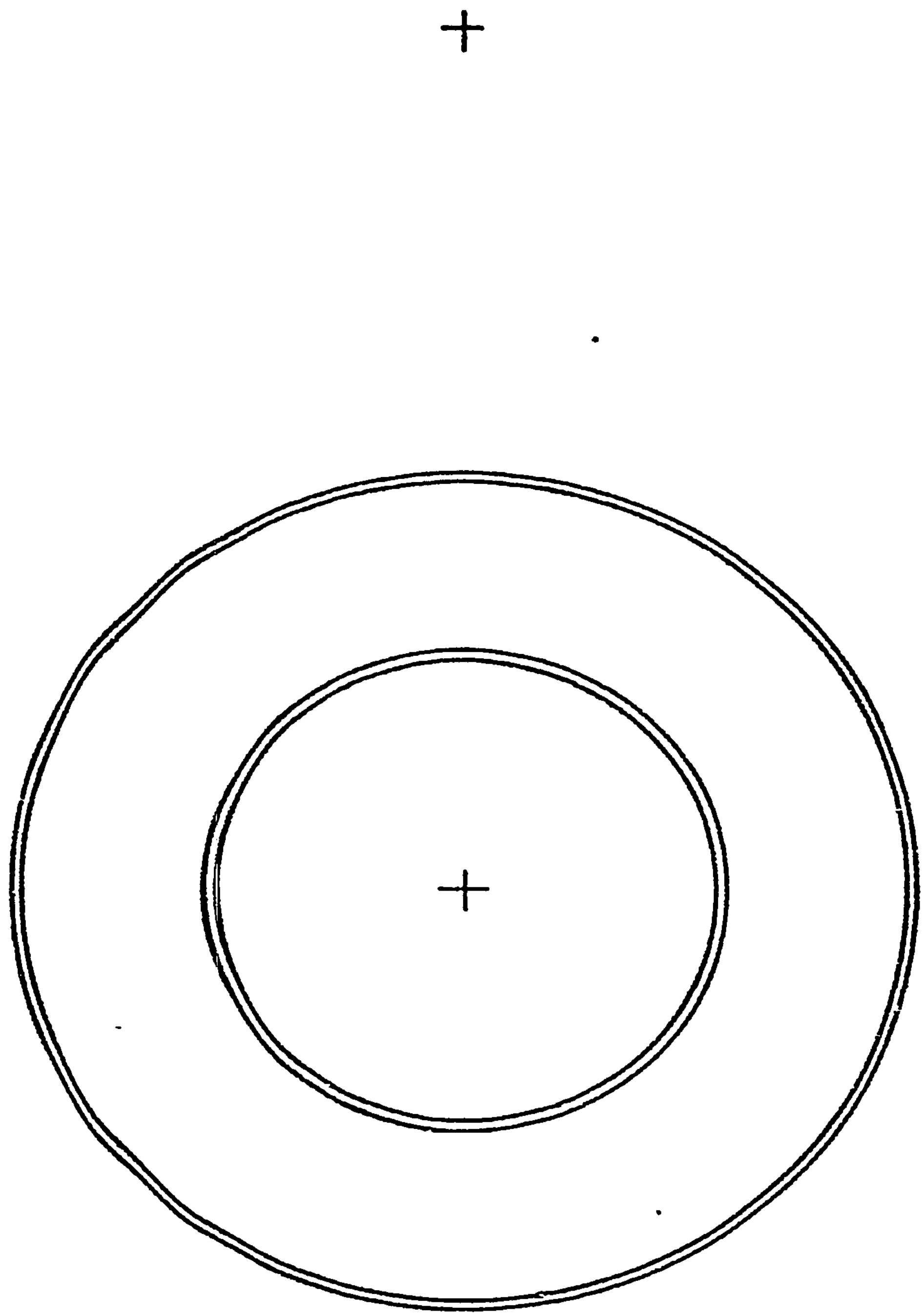
SECTION 24



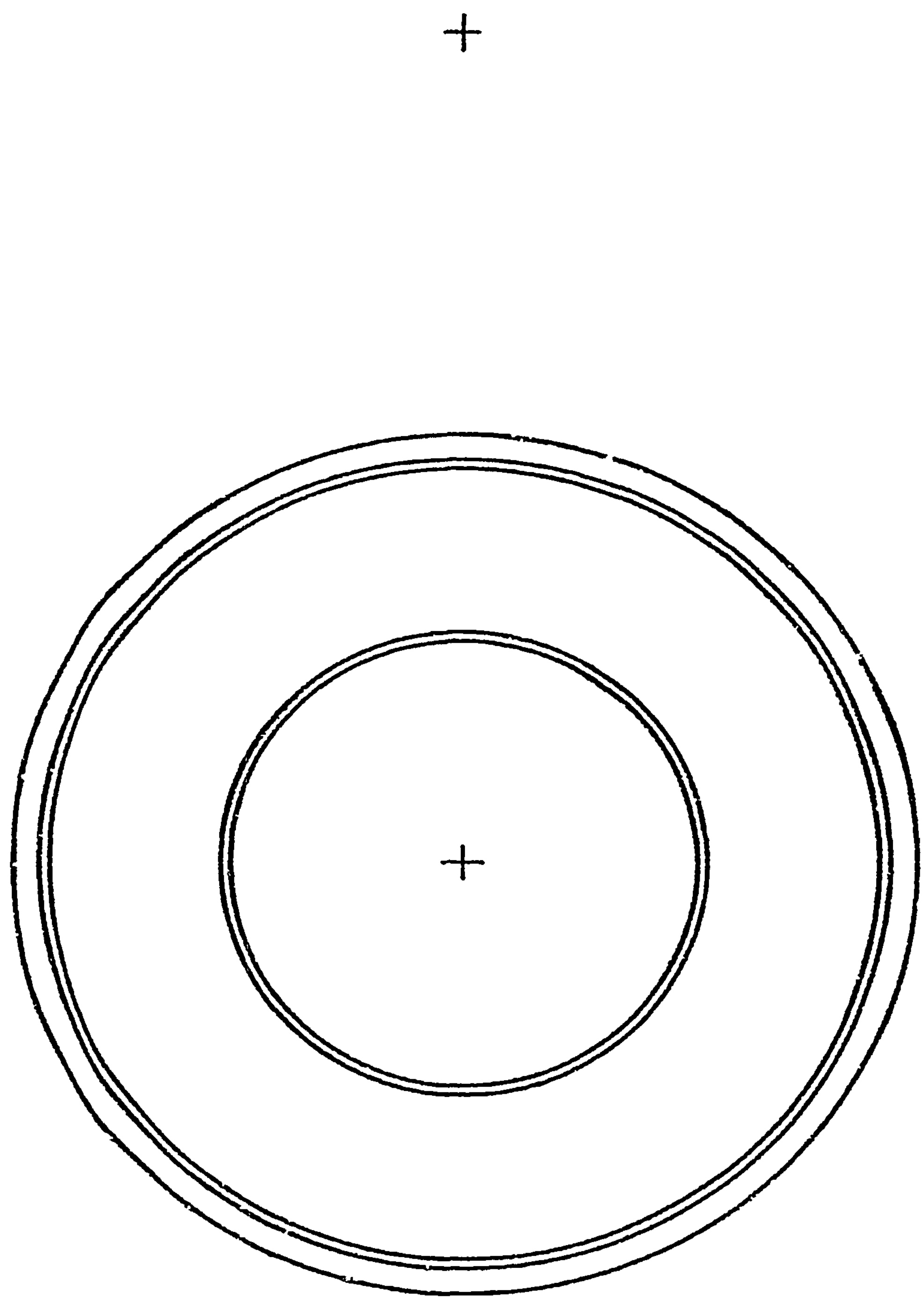
SECTION 23



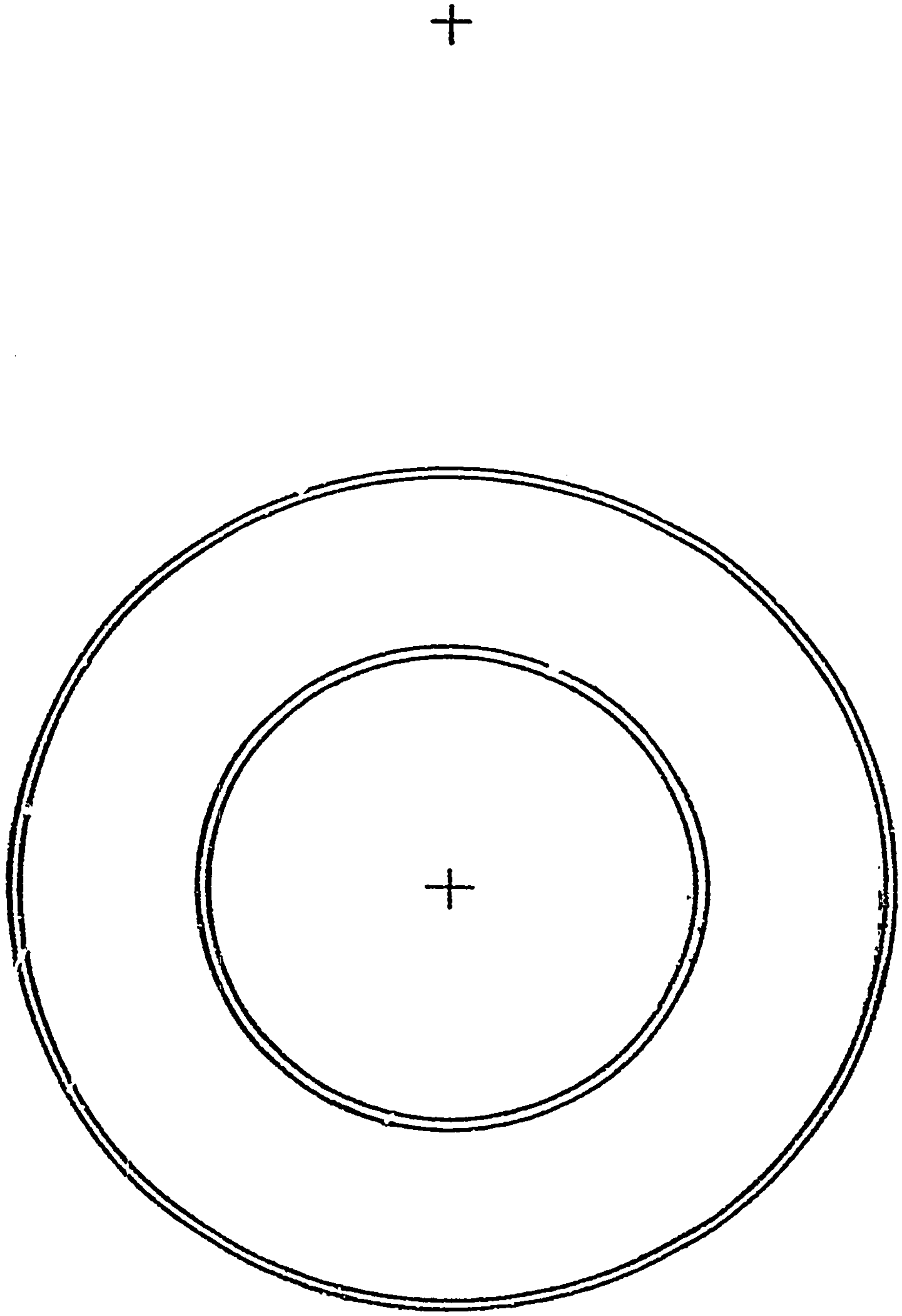
SECTION

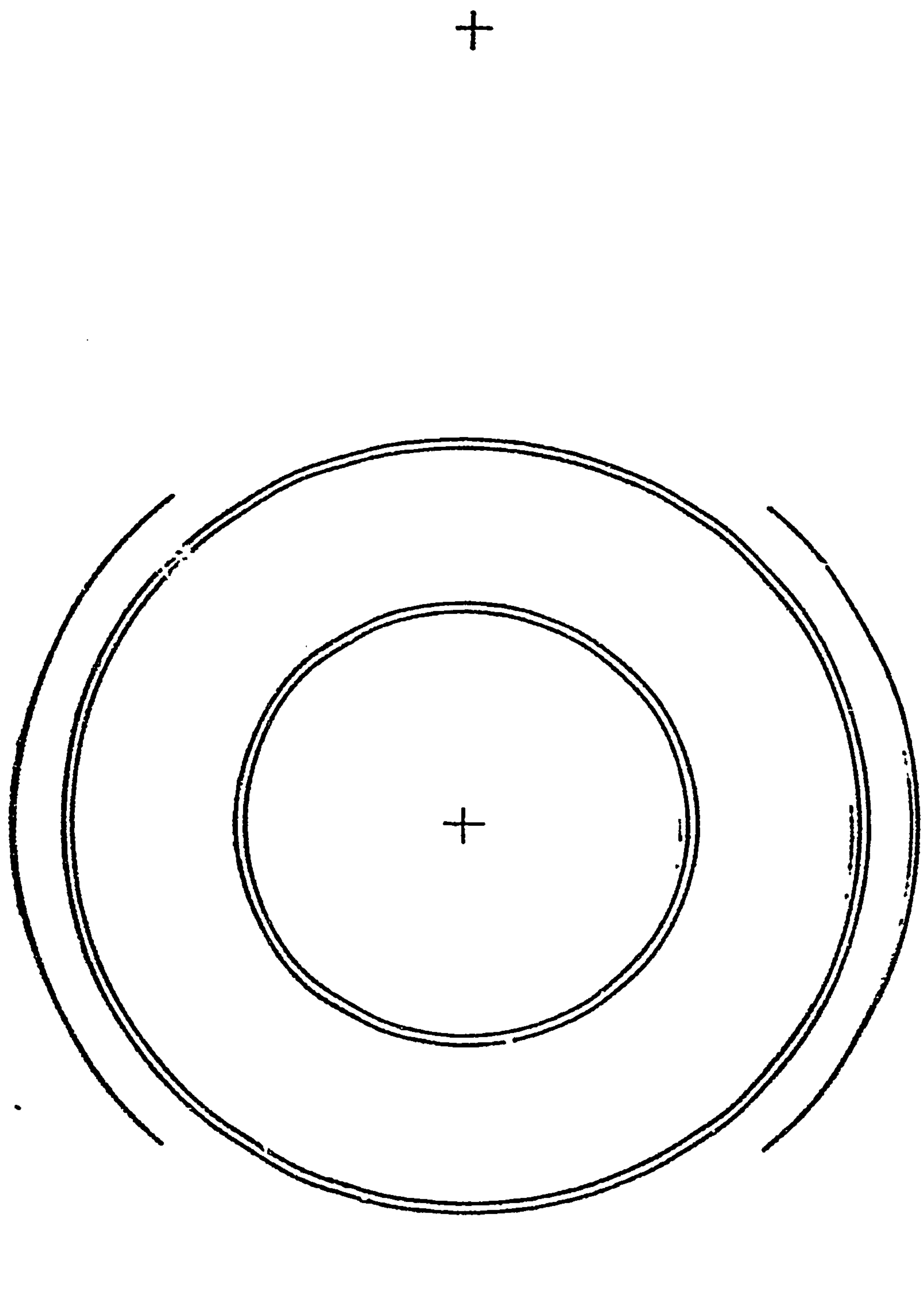


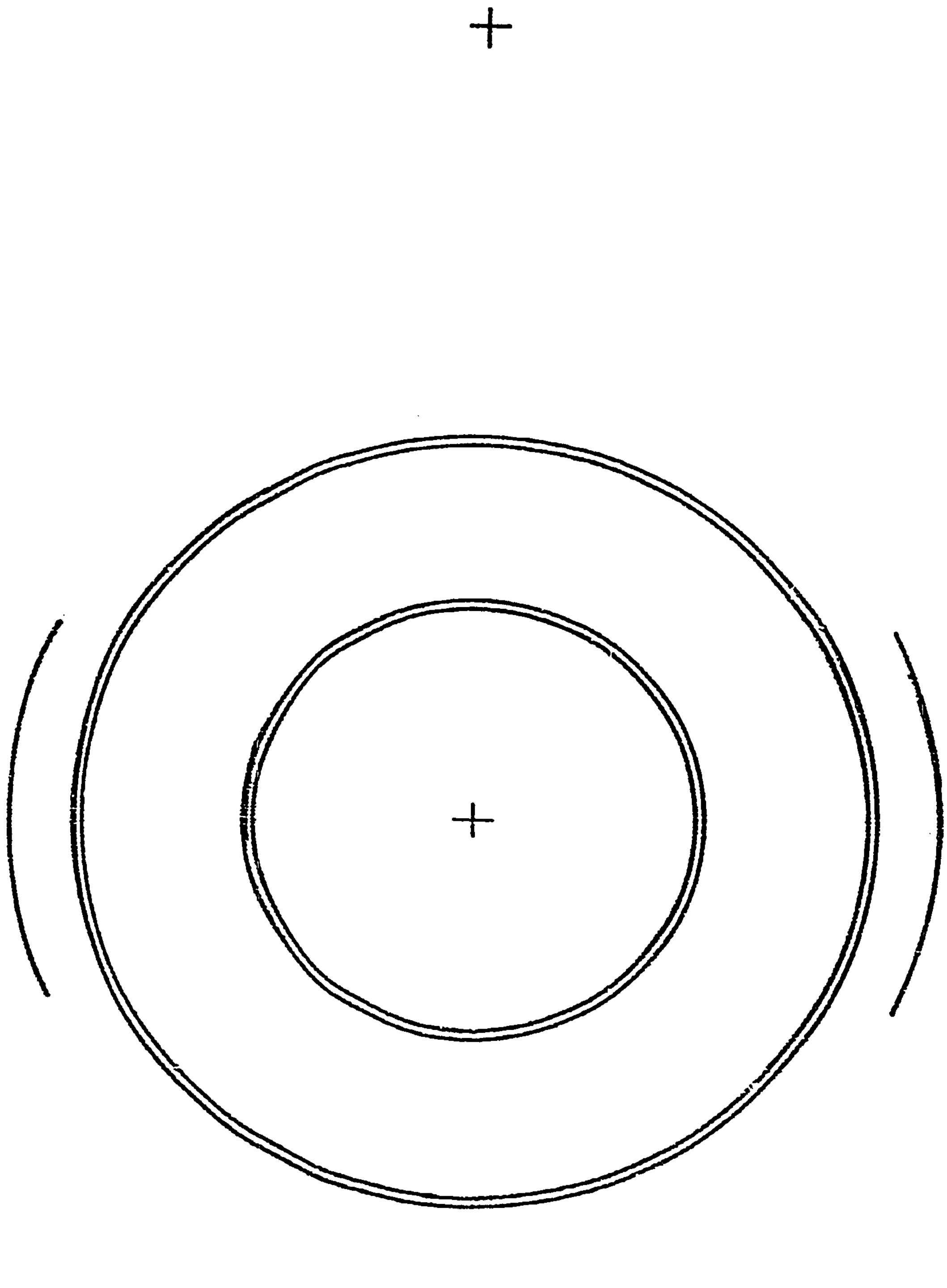
SECTION



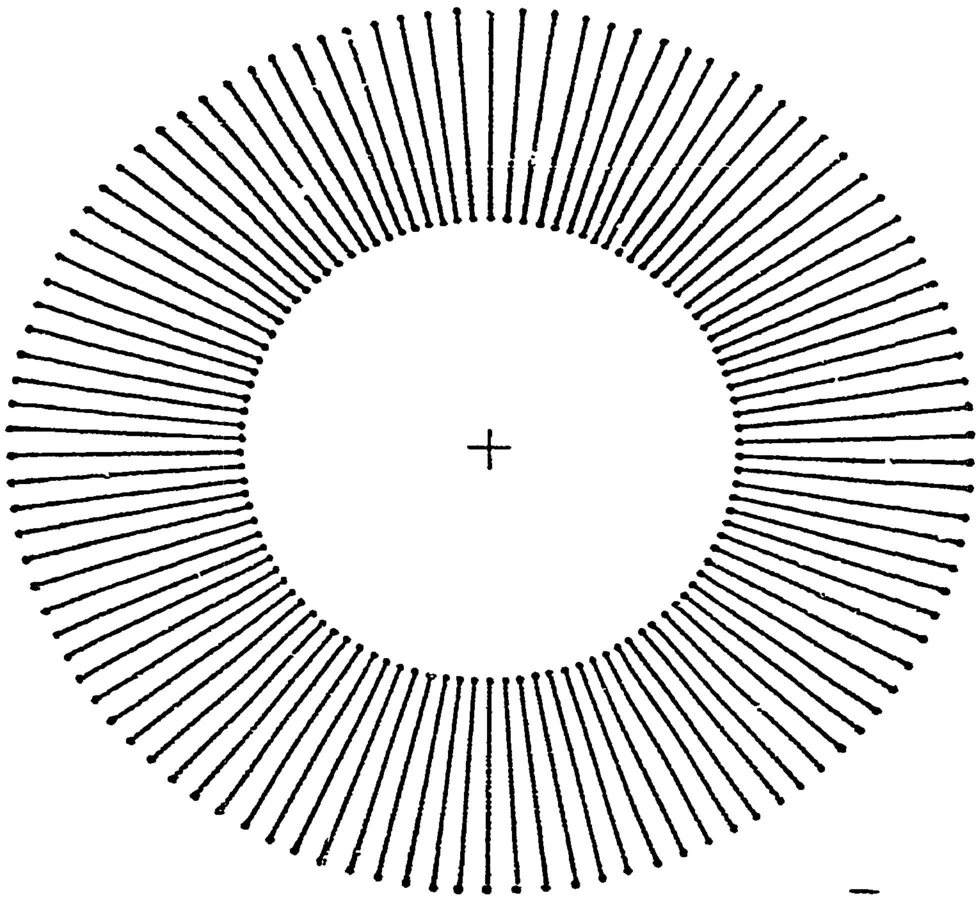
SECTION 2



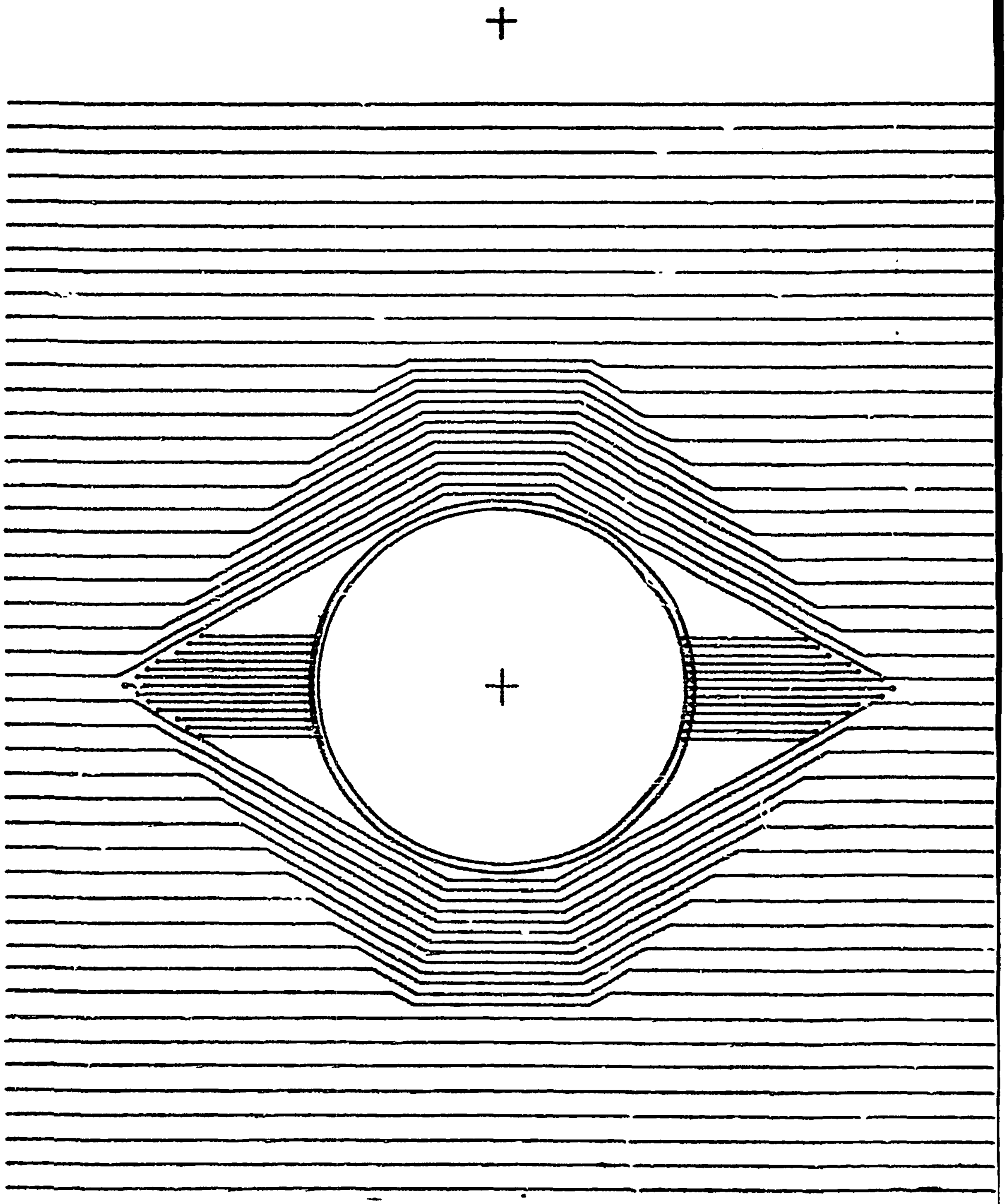




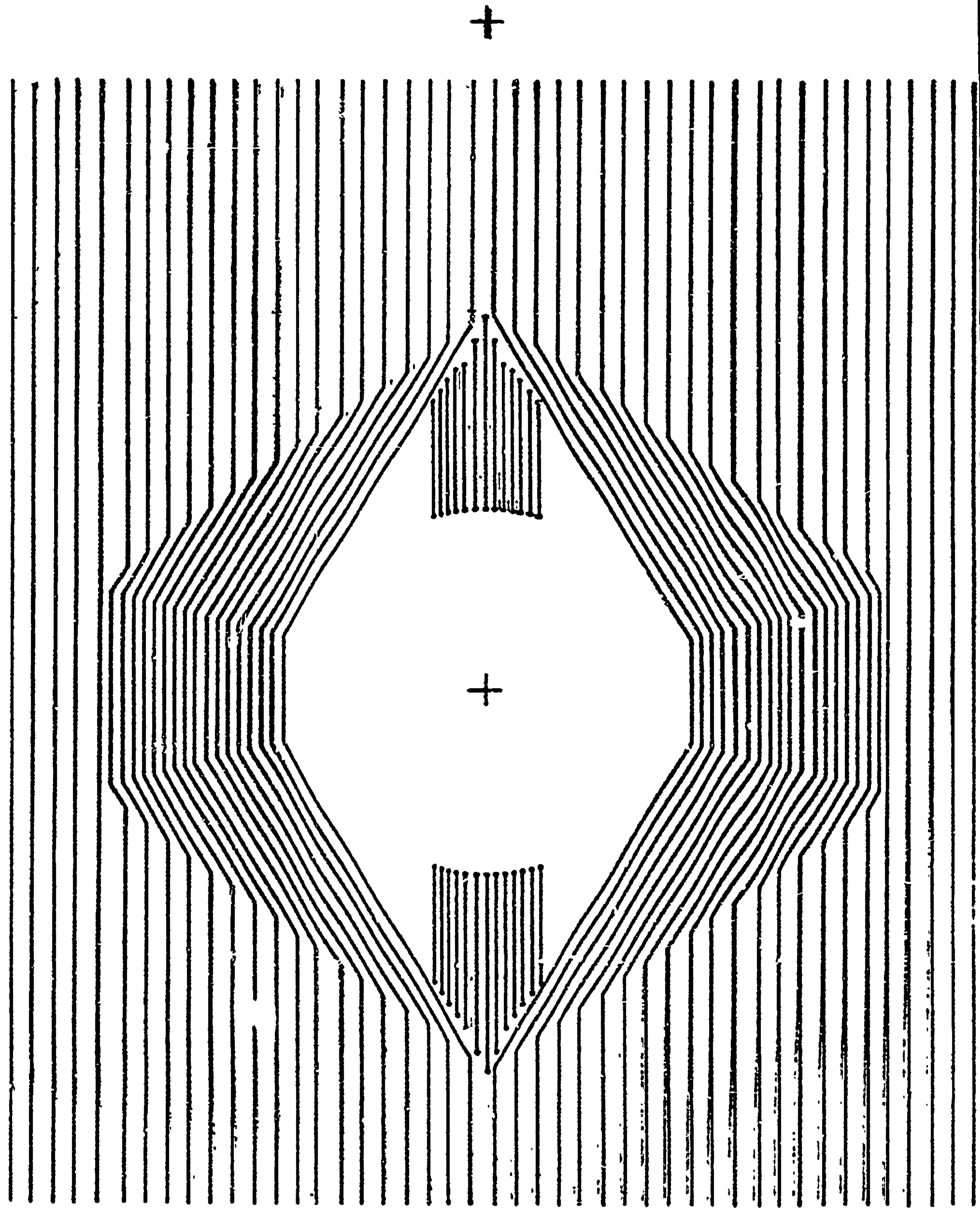
SECTION I



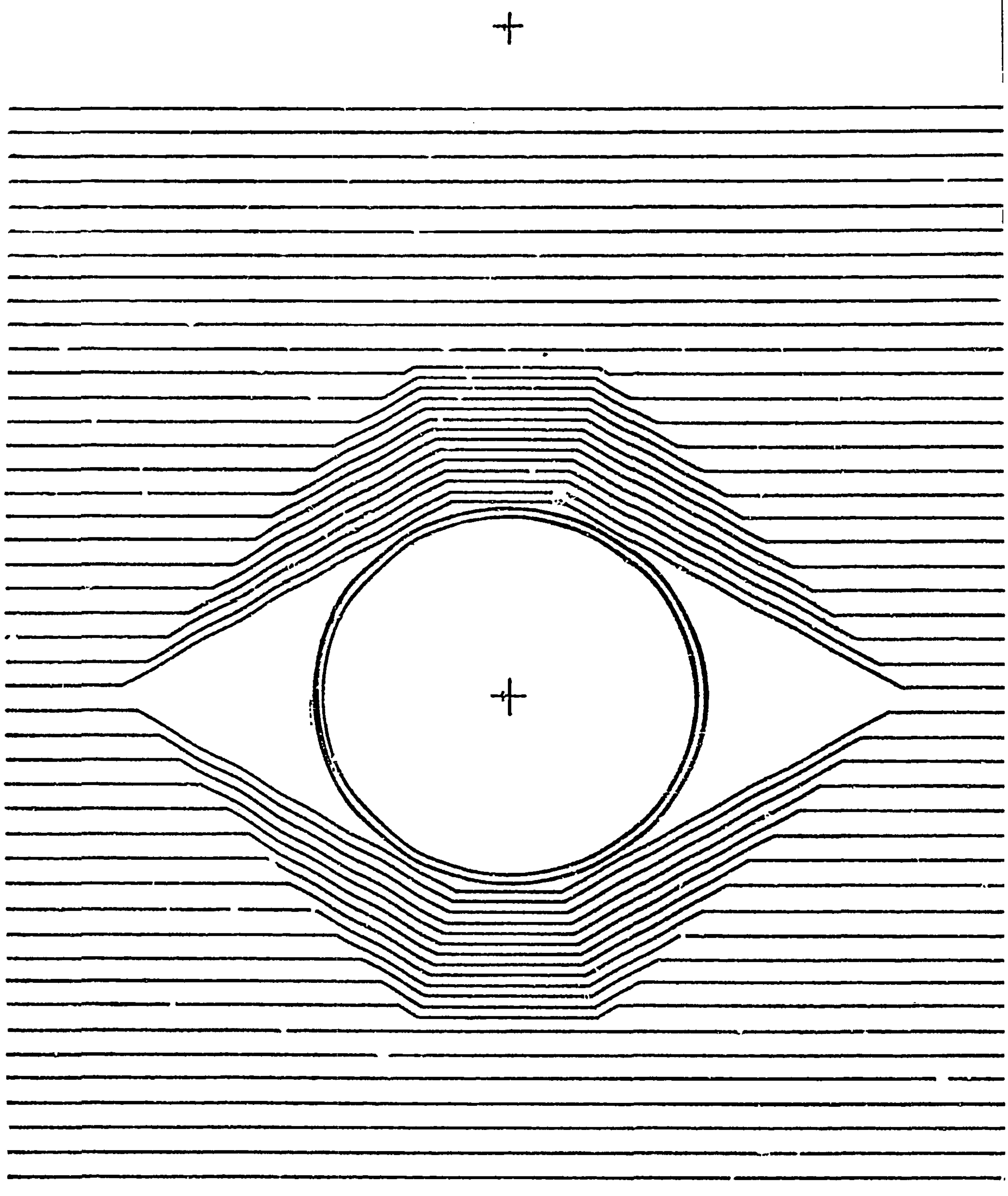
SECTIC



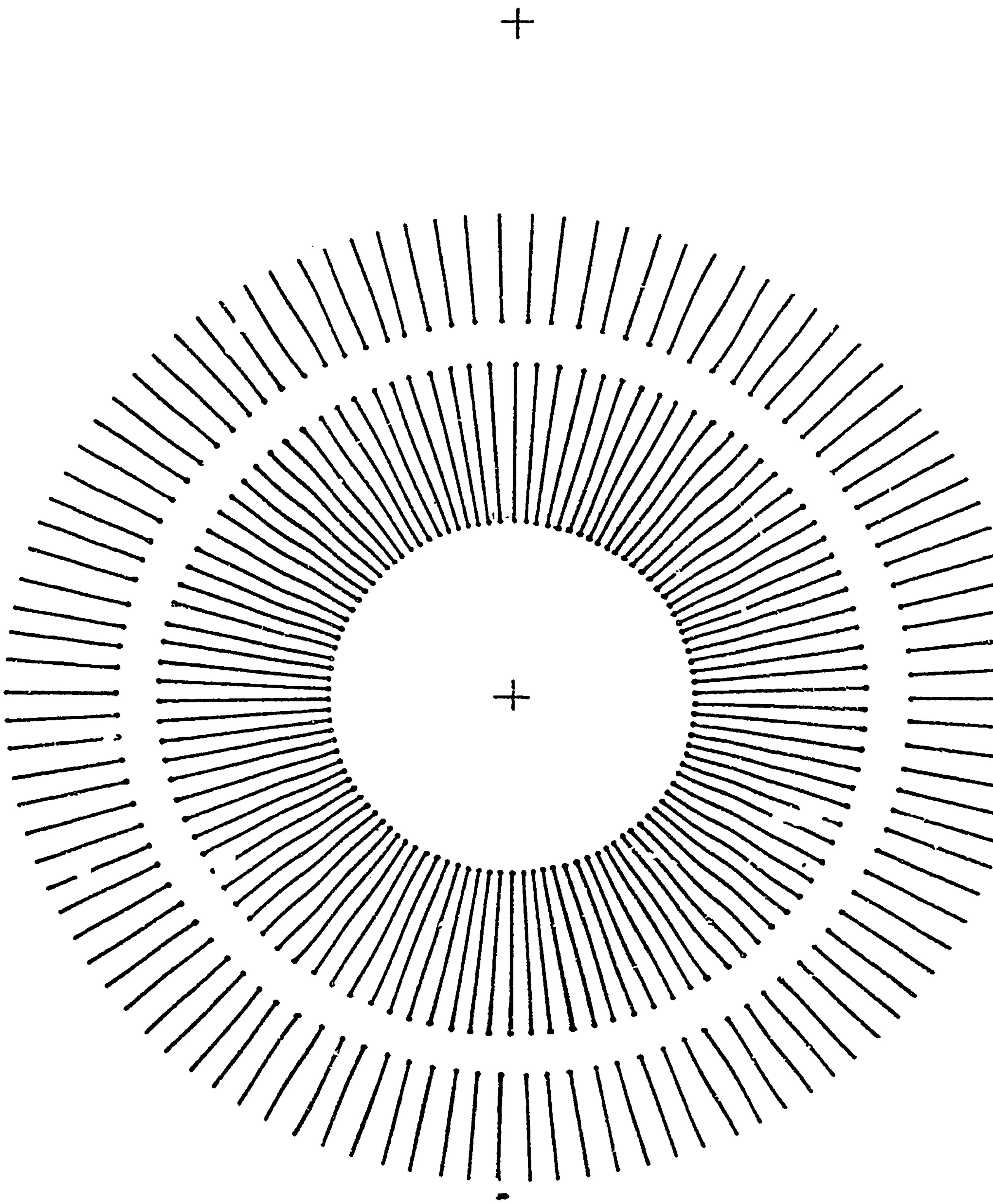
SECTION 15



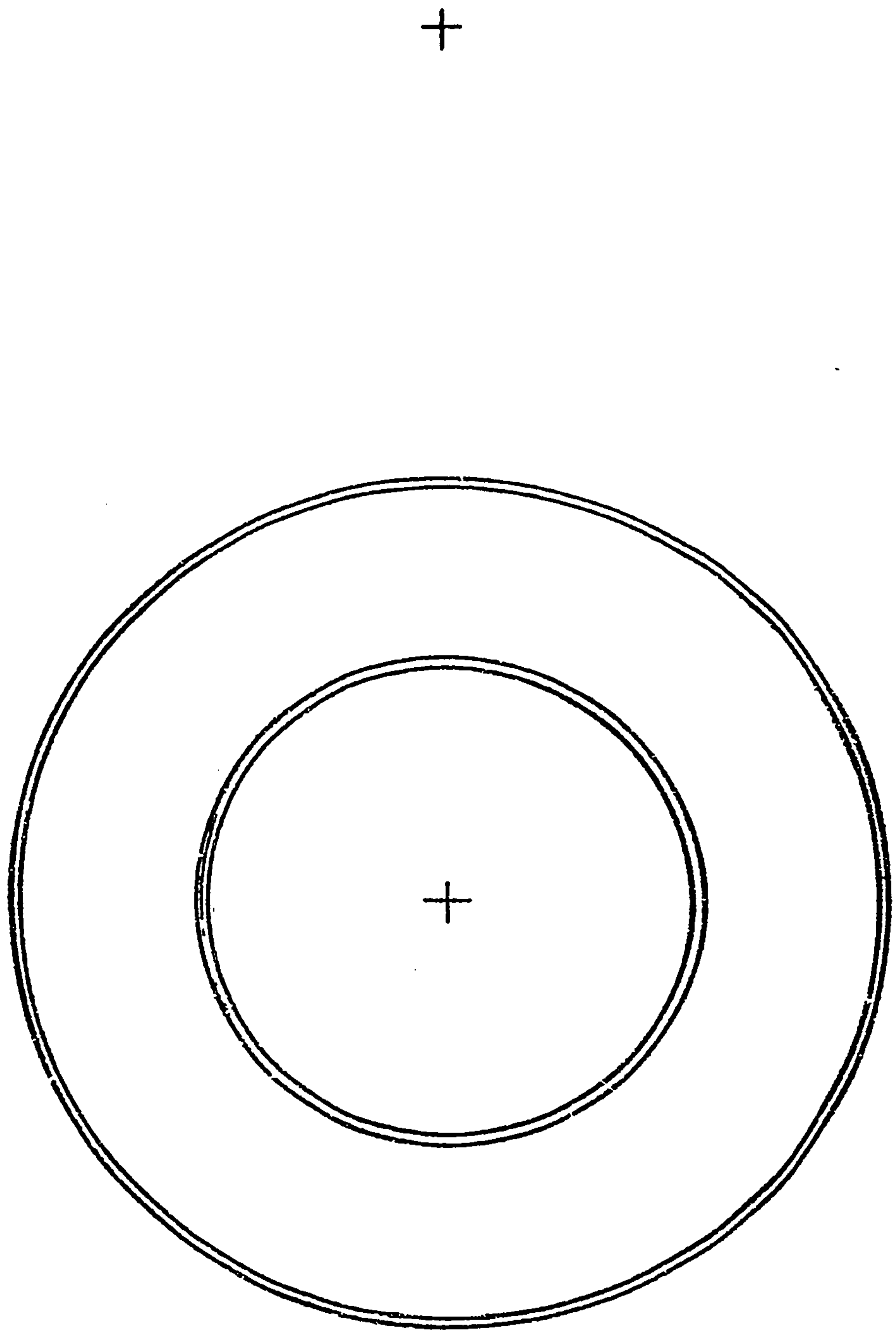
SECTION 14



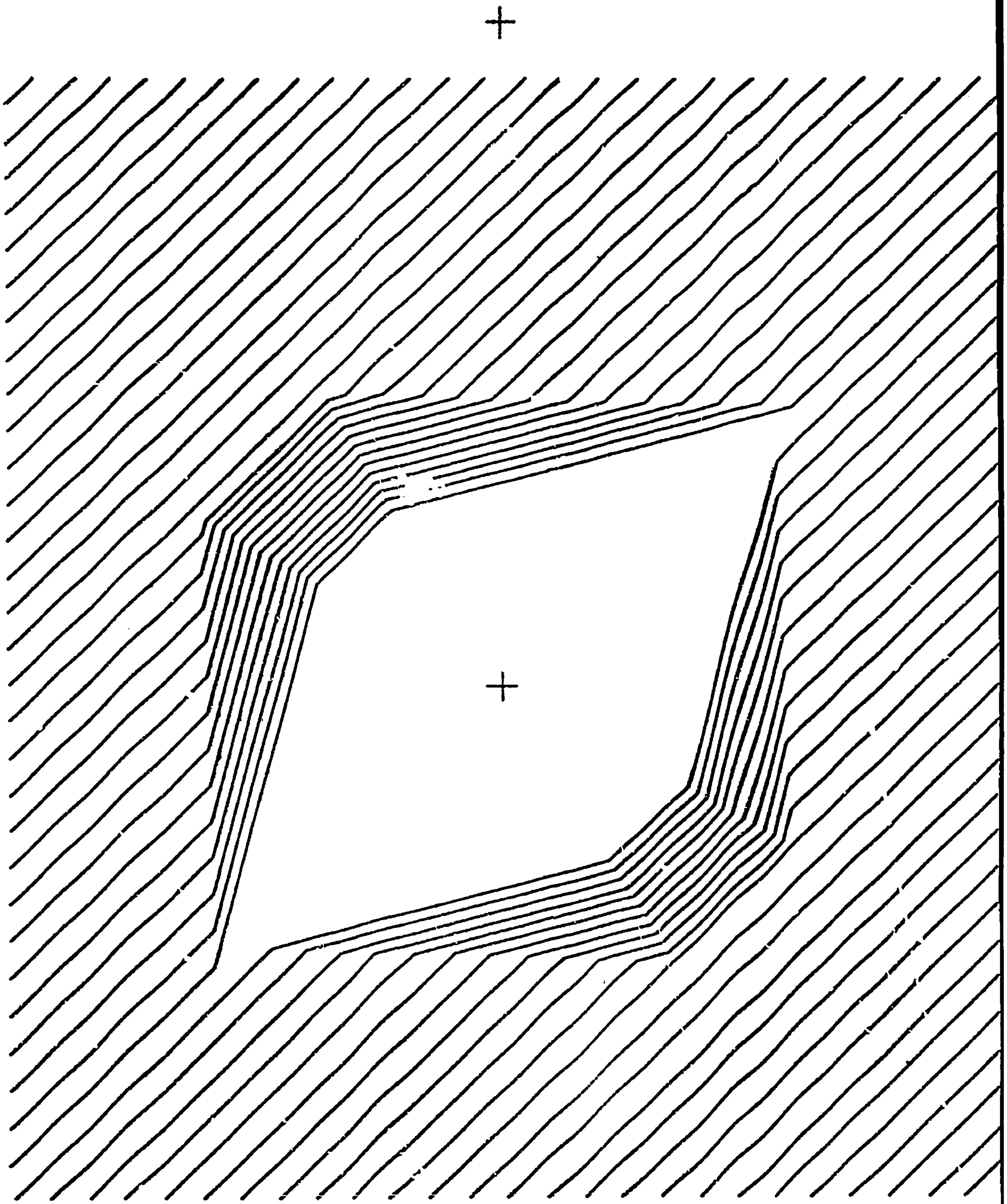
SECTION 1:



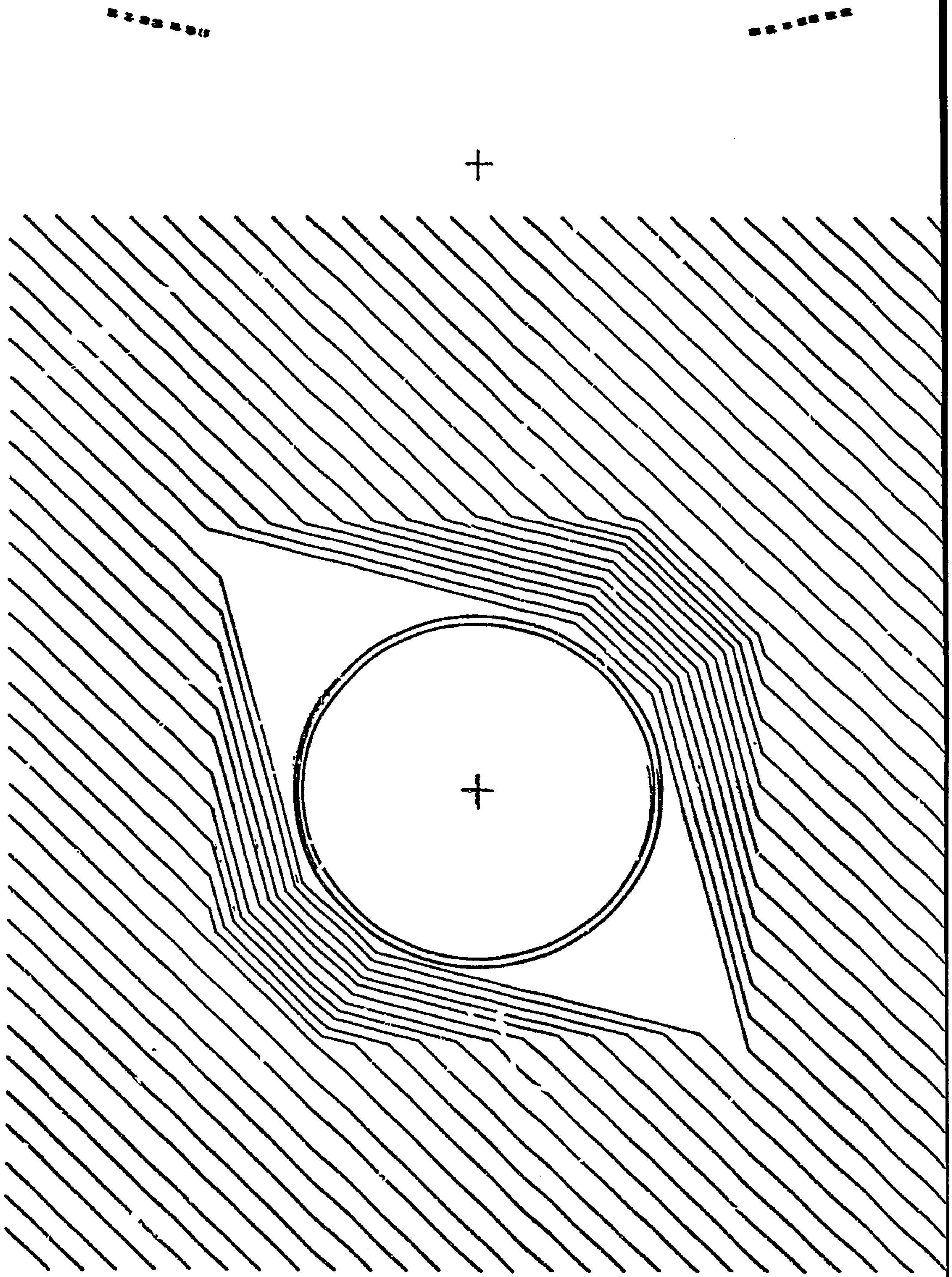
SECTION I



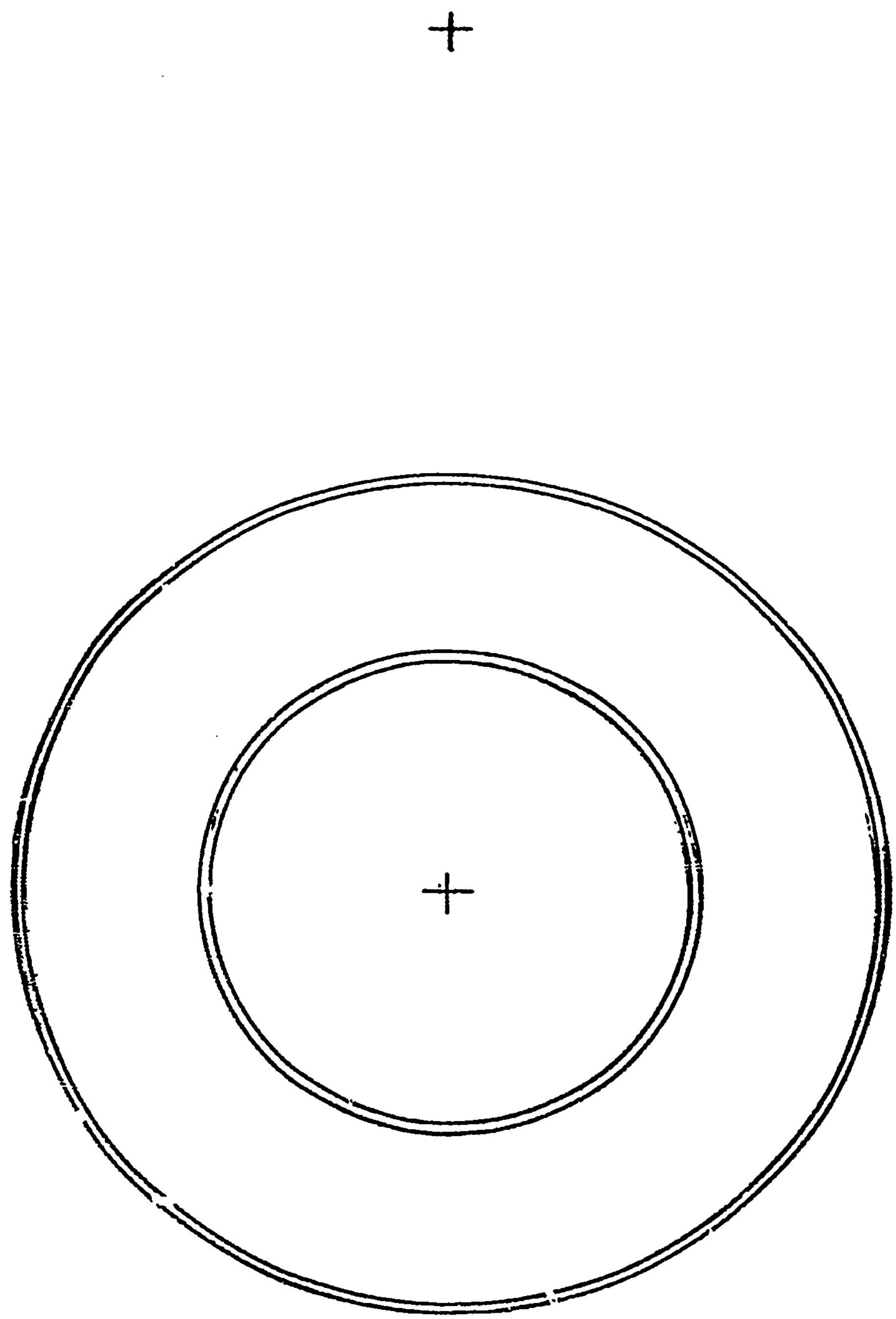
SECTION 9,10,11



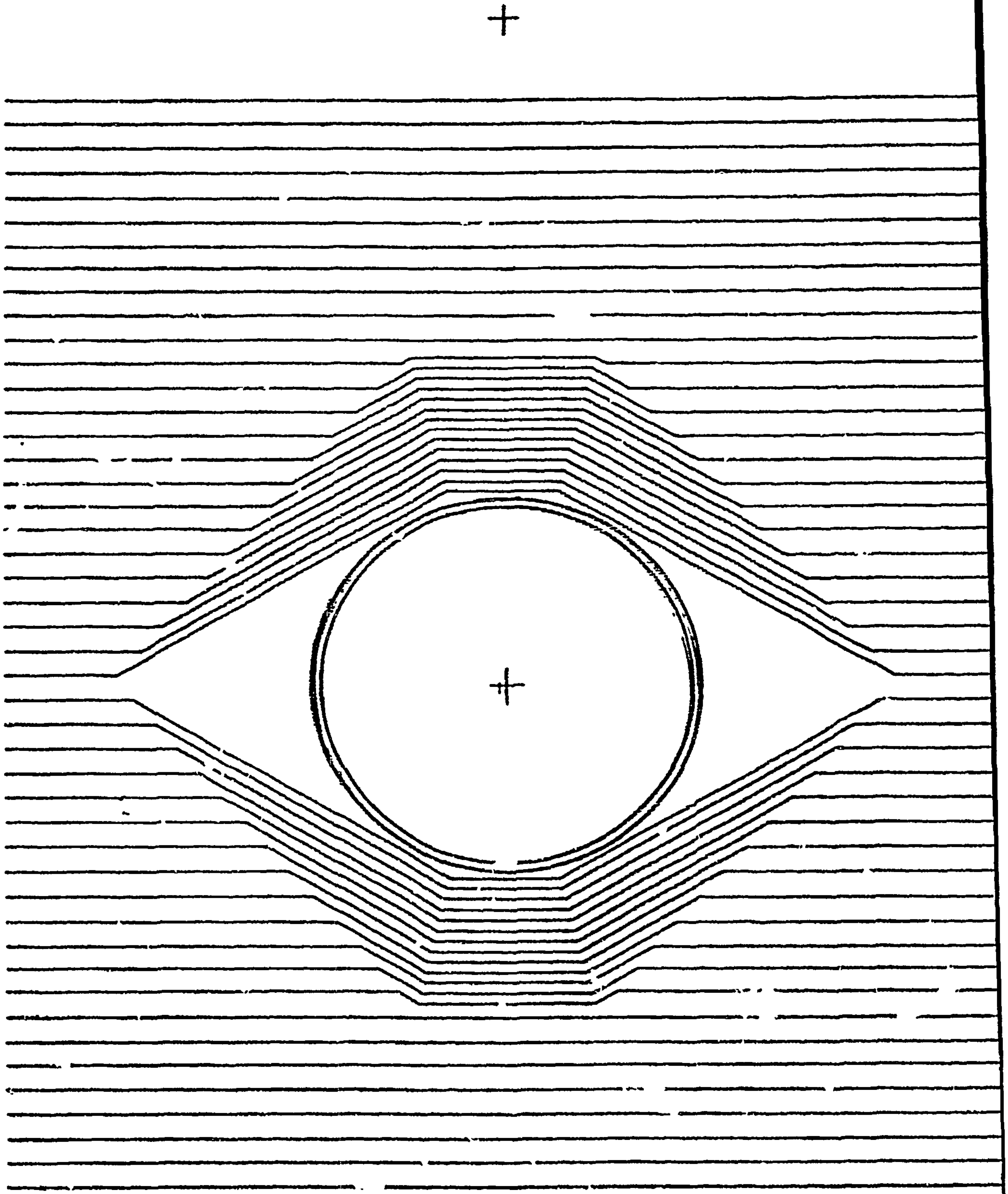
SECTION 8



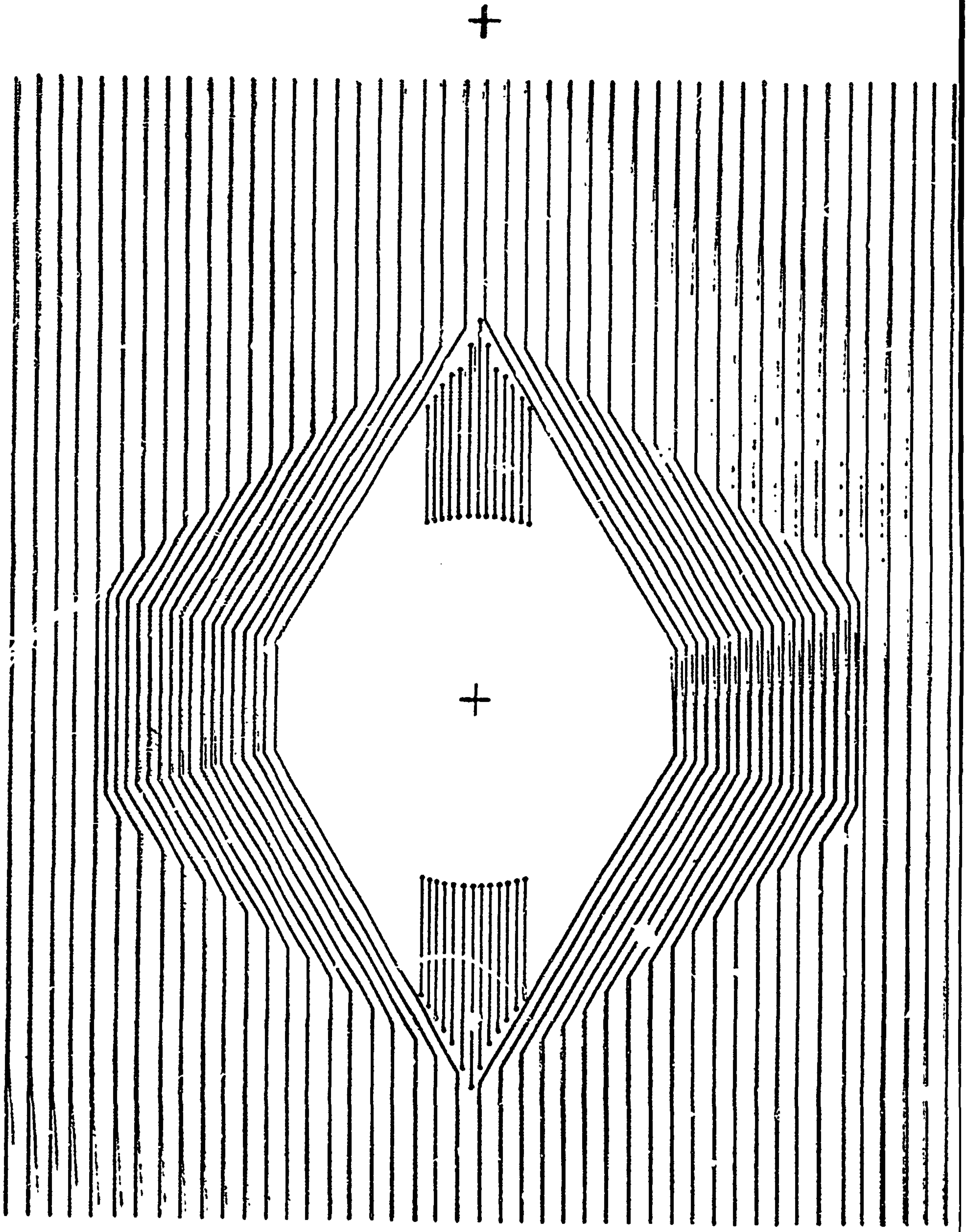
SECTION 7



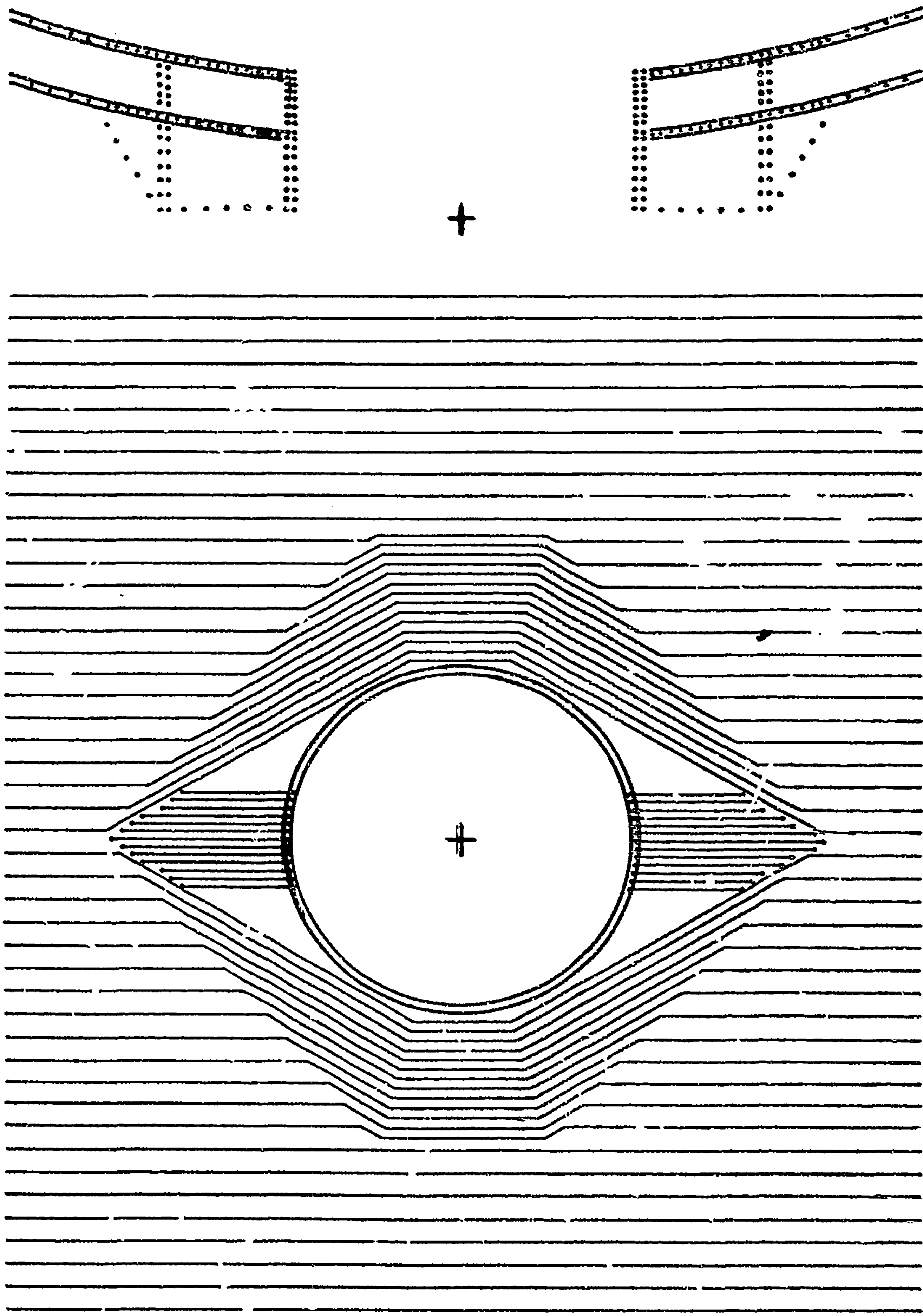
SECTION 5,6



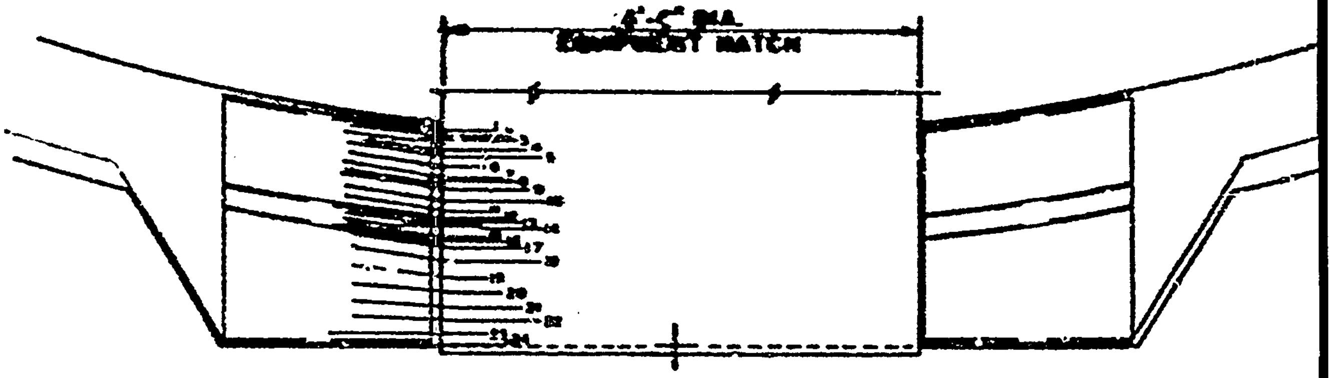
SECTION

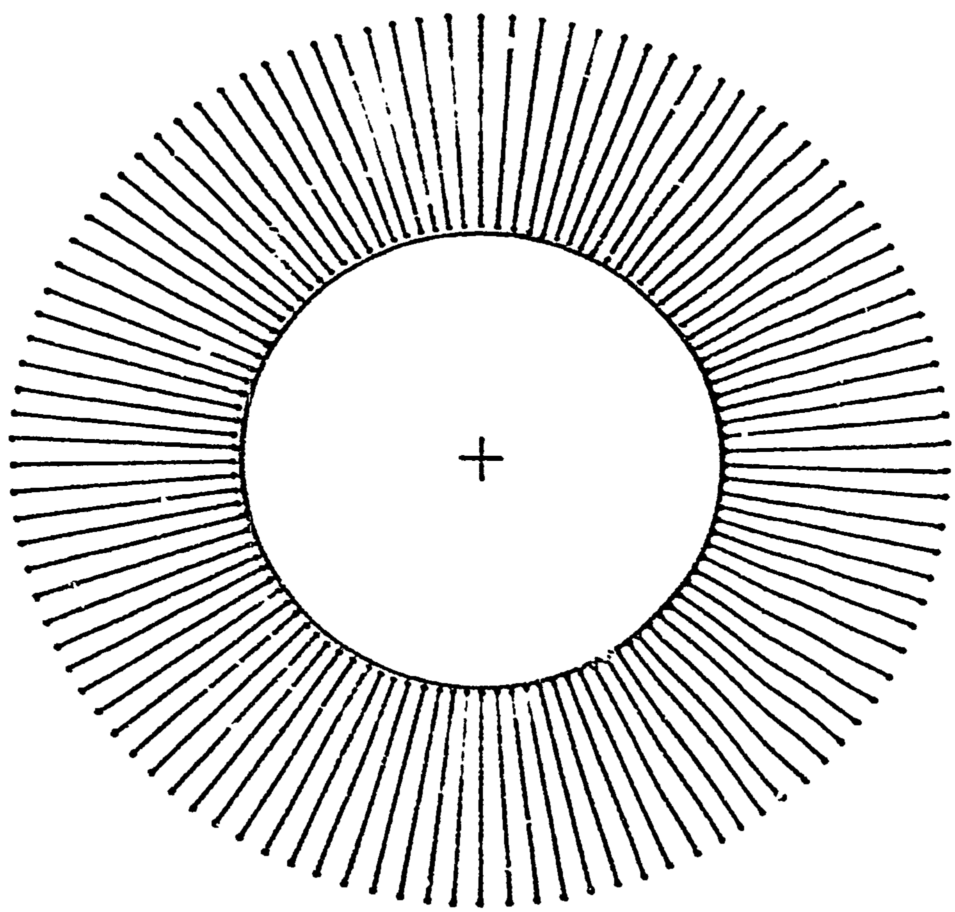


SECTION 3



SECTION 2





SECTION

APPENDIX B

SUMMARY
OF
TORNADO EFFECT ANALYSIS
ON
INDIAN POINT UNIT NO. 2
CONTAINMENT STRUCTURE

The design of Indian Point Unit No. 2 did not contain any tornado design requirement. However, the containment has been analyzed to determine tornado effect on the design structure.

The analysis of effects of tornados on Indian Point Unit No. 2 containment structure is based on the following characteristics:

Missiles:

a. Corrugated sheet of siding 4' x 8', 100 lbs., traveling at 225 mph.
b. Bolted wood decking 12' x 4' x 4", 430 lbs., traveling at 200 mph.
c. Passenger car, 4000 lbs., traveling on the ground at 50 mph.
d. Cedar fence post, 33 lbs., 6" by 6", traveling at 150 mph.

Wind Load:

a. Wind load associated with 300 mph horizontal wind distributed on structures in accordance with ASCE paper 3269.
b. Pulsed negative pressures associated with tornado winds having horizontal component of 300 mph and vertical component of the same magnitude.

Results of this anlaysis show that the containment structure can withstand wind loadings, pressure differentials, and missiles resulting from a tornado as defined without overturning or loss of integrity.

The Indian Point II Containment building is a monolithic steel lined reiefo[rced]
concrete upright cylinder with an integrated hemispherical reinforced roof. [The]
foundation is a 9' thick reinforced concrete mat resting on solid rock. The
uphill side of the foundation is approximately 30' below grade, while the
downhill side rests on concrete fill, thus exposing itself completely to the
winds. The cylindrical wall thickness is about 4.5 feet, while the mean
thickness on top is about 3.5 feet.

Principal dead weight of the containment building parts which can be consid[ered]
as minimum and structurally integral with the foundation are listed in Table
I.

TABLE I

Principal Dead Weights of the Containment

Containment Shell*	60.0×10^6 #
9' Matting Foundation	19.0×10^6 #
3' Mat above Liner	6.3×10^6 #
Primary and Biological Shields	7.3×10^6 #
Missile Shields	1.0×10^6 #
Crane, Reactor, Boilers and Miscelleneous**	3.0×10^6 #
Total	96.6×10^6 #

According to ASCE paper 3269, the dynamic pressure of winds is:

$$q = 0.002558 \ V^2$$

Where q is the pressure in psf

V is the wind speed in MPH

Then, a 300 MPH wind represents a dynamic pressure of:

$$q = 0.002558 \times (300 \quad)^2 = 230 \text{ psf}$$

The local pressure at any point on the surface of a building is equal to
the product of the dynamic pressure and a pressure coefficient is strongly
dependent on the relative position of the point in question with respect
to the direction of the wind, and of the shape and nature of the building
itself. Normally, this coefficient is positive at points facing the wind

*Reinforced concrete specific gravity is taken as 2.35

**No credit is given to weight of earth above foundation and negative at points "sideway" or "behind" the wind. The positive coefficients give rise to a pushing force, while the negative coefficients give rise to a suction force, both acting on the structure to effect an overturning moment in the direction of the wind. Fig. 1 shows the pressure coefficients for an upright cylinder and an exposed sphere.

The pressure differential at any point of the airtight structure is:

$$\Delta p = (P_a + C_p q) - P_i$$

in which P_a is the normal atmosphere pressure, taken as 14.7 psi

C_p is the pressure coefficient

q is the wind dynamic pressure

P_i is the working pressure inside the structure

In order to calculate the net force and overturning moment of the wind, angular integration of the coefficient C_p (α) has to be effected. Paper ASCE 3269 has adopted data of tests performed by Ackeret of the Institute of Aerodynamics of Zurich, which give simple calculational approximations for the total force acting on an upright cylinder. This force is:

$$F = C_D \, q A_{cyl}$$

where q is the wind dynamic pressure

C_D is the overall drag coefficient; $C_D = 0.45$ for upright cylinder of rise ratio 1, with smooth outer surface

A_{cyl} is the vertical cross section of the structures, equal to the product of the diameter and the height.

As for elevated rounded roofs, paper ASCE 3269 gives pressure coefficients for four segments of circular arches: the center half, the leeward and windward quarters. For an elevated dome of rise ratio 0.5 as in the case of Indian Point II Containment, these coefficients are:

$0° < \alpha < 45°$	$C_{pe} = 0.7$
$45° < \alpha < 135°$	$C_{pe} = -1.2$
$135° < \alpha < 180°$	$C_{pe} = -0.5$

The net horizontal force on which the dome can thus be considered as:

$$F = (0.7 + 0.5) \, q \, A_D$$

where q is the wind dynamic pressure

A_D is the corss-sectional area of the dome below 45°.

perpendicular to the wind direction.

If the whole containment building is considered as a smooth upright cylinder with the same height as that of the uppermost point of the structure i.e. 223 ft., then the horizontal wind force according to the formula given above is 3.24×10^6 lbs. The overturning moment due to this force would be 3.62×10^8 ft lbs.

If separate consideration is considered for the spherical dome and the cylindrical body, then we have:

Force over cylindrical body = 2.25×10^6 lbs.

Force over spherical dome = 1.10×10^6 lbs.

which totals 3.35×10^6 lbs., or about the same as the force acting on the upright cylinder of the same height. Also,
Overturning moment due to force over cyl. body = 1.78×10^8 ft. lbs.
Overturning moment due to force over spherical dome = 1.98×10^8 ft. lbs.
which totals 3.74×10^8 ft. lbs., or again very similar to the overturning moment due to the force acting on the upright cylinder of the same height.

Table II summarizes data and results of these calculations.

TABLE II

300 MPH HORIZONTAL WIND FORCES AND OVERTURNING MOMENTS ACTING ON INDIAN POINT II CONTAINMNET

	C_D	Moment Arm (ft)	Total Force (#)	Overturning Moment (ft #)
Hypothetical smooth upright cylinder of max. containment height	0.45	112	3.24×10^6	3.62×10^8
Actual Containment				
Cyl. body	0.45	78	2.25×10^6	1.76×10^8
Sph. dome	(*)	180	1.10×10^6	1.98×10^8
Total			3.35×10^6	3.74×10^8

* A step function of angle as previously shown

It should be noted that the horizontal shear force associated with the earthquake design requirement is approximately 16×10^6 lbs. and the overturning moment is approximately 23×10^8 lb-ft.

If the pivotal point is taken at the uphill side, corner of the foundation, the minimum stability moment due to the containment dead weight is:

Stability moment = 96.6×10^6 # x 70 ft. = 67.6×10^8 ft. # of containment

And the ratio of the maximum 300 MPH wind overturning moment to the minimum stability moment is:

$$\frac{\text{Maximum Overturning Moment}}{\text{Minimum Stability Moment}} = 5.5\%$$

According to ASA Standard A58.1-1955, Sect'on 5.8.1, the overturning moment due to wind load shall not exceed 66.7% of he stability of the structure due to dead loads. It is thus concluded th t the Indian Point II Containment is inherently safe from overturning due to tornadoes.

Bursting Effect

As presented in paper ASCE 3269 and shown in figure 1, the maximum negative coefficient of the external pressure is - 1.7 for a cylinder with a rise ra of about 1 to - 1.2 for a sphere, both happening at points of the wall perpendicular to the wind. At 300 MPH wind, this represents a maximum drop normal atmospheric pressure of 2.7 psi for a cylinder and 1.9 psi for the spherical dome.

If the vertical component of the tornado is taken into account, and is assumed to be also 300 MPH, then the maximum dynamic wind pressure will increase by a factor of $\sqrt{2}$ to 325 psf. The maximum pressure drop would be 3.84 psi at points of the cylindrical body sideway with respect to the wind, and 2.7 psi for some particular points over the spherical dome.

Since the Indian Point II Containment building is structurally designed and built to withstand pressure up to 70.5 psi to accommodate safely the maximum hypothetical accident, it is seen that maximum pressure differential created by tornadoes do not cause any accountable bursting effects.

Lifting Effect

ASCE paper 3269 gives the lifting force acting on a dome roofed, air-tight reservior as:

$$F_D = (Pi - Pa)\,A$$

where P_i is the inside working pressure, $A = \pi d\ 2/4$ and P_a is the external pressure (C_{pe} being taken as -1.0)

Using this formula, the upward lift acting on the containment is:

$$F_D = 230 \times \pi \times (140)2 = 3.54 \times 10^6\ \#$$

Comparing with the dead weights of the containment vessel and internals, this represents:

$$\frac{\text{Lifting force}}{\text{Minimum Dead Weight}} = 3.7\%$$

For the case of a maximum hypothetical accident which the containment building is designed to withstand, the internal building pressure may reach 70.5 psi. The lifting force due to 300 MPH tornado represents only 2.9% of the lifting force introduced by this MCA internal pressure. It is thus seen that the lifting effect introduced by tornadoes is insignificant compared either to the minimum building dead weight or to the maximum lifting force that the building is structurally designed to withstand.

The equation used for missile penetration in concrete is the Modified Petry formula.

$$D = \frac{KW}{A} \log_{10} \left(1 + \frac{V^2}{215,000}\right)$$

in which D is the depth of penetration in feet

K is an experimentally determined coefficient taken as 4.76×10^{-3} $ft^3/\#$ for reinforced concrete of 1.4% reinforcement at 3200 psi compressive strength, as 8×10^{-3} $ft^3/\#$ for concrete in mass without reinforcement.

W is missile weight in lb.

V is striking velocity in ft/sec

A is missile frontal area in ft^2

This formula predicts the depth of penetration into an infinite slab.

For a finite reinforced concrete slab of thickness T a correction factor must be applied to the preceding formula to determine the predicted penetration D' which is defined as follows:

$$D' = D\,(1 + e^{-4\,(T/D\,-2)})$$

Results of penetration in concrete of the specified missiles are shown in Table III.

TABLE III

Penetration of Missile in Concrete

	Corrugated Sheet of Siding	Bolted Wood Decking	Passenger Car	Cedar Fence Post
Impact Velocity				
MPH	225	200	50	150
FPS	330	294	73.5	220
Weight (lb)	100	450	4000	33
Penetration (ft)				
Reinforced Concrete	0.129	0.234	0.074	0.053
Mass Concrete	0.216	0.392	0.124	0.093

Considering the containment building wall as reinforced concrete slab of thickness 4.5 feet, the correction factor for finite thickness is very close to 1. Therefore, the expected penetrations in these walls would be very similar to those shown in the Table III. The most serious penetration is about a quarter of a foot, which is at least an order of magnitude below the thickness of the walls. The containment building and the concrete areas of the primary auxiliary building are thus considered invulnerable to perforation caused by the missiles considered.

APPENDIC C

SUMMARY

OF

UNITED ENGINEERS & CONSTRUCTORS INC.

EXPERIENCE

IN

UTILIZING

CADWELD REINFORCEMENT

BAR SPLICES

1.0 SUMMARY AND CONCLUSIONS

The design of the containment structure for Unit No. 2 at the Indian Point Generating Station is predicted on the utilization of a mechanical splicing technique for joining reinforcement bars consisting of a "Cadweld" splice as supplied by Erico Co. of Cleveland, Ohio.

The initial intent of the design and the Quality Control program was to provide reasonable assurance that all splices tested have an ultimate strength of 125% (75,000 psi) of a minimum of yield (60,000 psi). To provide a margin, the design stress is 95% of the minimum yield or 57,000 psi.

The sampling process through April 10, 1967, which consisted of a destructive test of one out of 100 splices and a plot of the average strength of failure, gave every indication that the average stress in the splices was well in excess of 75,000 psi. (Average values were approximately 90,000 psi). This sampling program continued through June 29, 1967.

The average strength of the splices tested beginning April 10, 1967 indicated a deterioration of the strength of the splices. The average strength for tests made during the month of April dropped to approximately 80,000 psi. This reduction in indicated strength of the splices was due primarily to the failure of 8 splices made over the period of April 10, 1967, to April 30, 1967. While the average value was still in excess of 125% yield strength, the trend of the test results was unsatisfactory. No reason for the decreasing trend was readily apparent. Hence, a test program was undertaken to determine the cause, evaluate the existing situation, and propose remedies.

All cadwelding was stopped in the field on June 29, 1967 and the Quality Control program re-evaluated. At the time work was stopped the following splices were in place:

(a)	Vertical	1250
(b)	Horizontal	1250
(c)	Seismic	700

1.0 SUMMARY AND CONCLUSIONS (Cont'd)

At that time 57 splices had been removed from the structure. 37 splices had been tested with results recorded. 20 additional splices were under te and these results were added to those previously received and mode of failure and the stress at which the failure occurred for all tests were as follows:

Mode of Failure	No.	Stress (Avg.)
Bar break	12	80,100 psi
Pull out	45	64,600 psi
	57	

The results of these tests were reviewed and subsequent evaluation led to development of the next step in determining the cause of the decreasing trend of splice test results.

A meeting was held in Philadelphia on July 15, 1967 to review the splicing problems and to determine a program to correct these problems. In summary, the program consisted of:

1. A review of the mode of failure and the variables involved.
2. A test program to determine the effects of the variables.
3. A review of the field procedures for making the splices.

The initial phase of the test program noted above consisted of randomly selecting 68 additional splices from those in place in the structure at the time. Of these, 18 were seismic splices and 50 were horizontal or vertical. It was subsequently found that 3 of the 50 splices had inadverten been cut from another series of 50 splices being made under strict surveillance and ideal conditions (called 25/25 program).

1.0 SUMMARY AND CONCLUSIONS (Cont'd)

The results of these 3 tests were not used in the following evaluation:

Mode of Failure	No.	Stress (Avg.)
Bar break	14	70,700 psi
Pull out	51	86,800 psi
	65	

(NOTE: The 25/25 program mentioned above consisted of making 50 splices under the strict surveillance of UE&C supervisory personnel and Erico representatives and under ideal conditions. 25 of these splices were cut out for testing and 25 remain in the structure. Of the 25 to be tested, 20 were tension tested and 5 were sectioned for macrographic examination. These results were used in determining new production procedures but due to the "laboratory" nature of their production, results were not used in the statistical survey. An analysis of the test results on production splices indicated that splices that failed at low strength could be traced to three principle causes:

1. Rebars that were nicked or scarfed at or adjacent to the sleeve usually resulted in a "bar break."

2. Rebars that were not centered in the sleeve usually resulted in pull outs.

3. Rebars that were mill marked (mill stampings on portion of bar in sleeve) usually resulted in pull out..

Of equal importance, bars not scarfed, nicked, or mill marked and properly centered in the sleeve, usually resulted in a high strength splice.

A revised procedure was issued to the field on September 19, 1967 for performing the splices. This procedure called for full inspection to eliminate mill marked or nicked bars, and prohibit scarfing, and provide a guide to assure proper centering of the bars in the sleeve.

All splices in the existing structure were visually inspected and 1182 splices were radiographed (609 vertical and 573 seismic.) Any splices failing visual inspection and/or the radiograph test (to indicate if bar was adequately centered) were removed and replaced. This program provided assurance that in existing splices all bars were properly centered in the sleeve, and all scarfed or mill marked bars were removed. Based on this assurance, all test data based on these two modes of failure were eliminated from the statistical analysis of the test program. This phase of work was completed on October 15, 1967.

The overall program resulted in a sound quality control program to assure high reliability for all future splices and a comprehensive and complete check of splices that were made prior to the revised quality control program.

Independent consultants were engaged to assist in determining the program and to evaluate the results. These consultants were Dr. T. C. Kavanagh and The Franklin Institute. The opinion of the consultants support the position and procedures used by UE&C.

Cadwelding work was resumed in the field on August 25, 1967 for replacement work; production splicing resumed September 19, 1967. This work was performed in accordance with the new procedures and at a higher sampling rate. The higher sampling rate consisted of a program of initially testing one out of five splices made in the structure and incrementally increasing to testing one out of 100.

From November 1966 to December 1, 1967, the following tests were made:

1. Tests on Splices Made From November 1966 to June 29, 1967

Description	No. of Splices	Mean Value
Horizontal	50	88,800 psi
Vertical	49	83,100 psi
Seismic	23	72,800 psi
	122	

2. Tests on Splices Made From June 29, 1967 to December 1, 1967 <u>Under New Production and Testing Procedures</u>

Description	No. of Splices	Mean Value
Horizontal	65	92,000 psi
Vertical	77	98,500 psi
Seismic	34	91,700 psi
	176	

3. <u>Tests on Total Splices as of December 1, 1967</u>

Description	No. of Splices	Mean Value
Horizontal	115	90,600 psi
Vertical	126	92,500 psi
Seismic	57	84,000 psi
	298	

Based on the completed work to date, including the replacements made subsequent to June 29, 1967, and the new work performed under the procedures established September 19, 1967, there is every reason to believe that the completed structure will more than meet the minimum strength requirements that have been established.

APPENDIX D

REPORT ON THE
CONTAINMENT BUILDING LINER
PLATE BUCKLE
IN THE VICINTIY 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 b lge 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 proble 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 concr 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 ps and a minimum tensile strength of 60,000 psi with an elongation of 22 per ce in an 8-in. gauge length at failure. The liner is 1/4-in. thick at the bott 1/2-in. thick in the first three courses except 3/4-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 t 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 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 of this report are available and on file in the UE&C field offices for review as required.

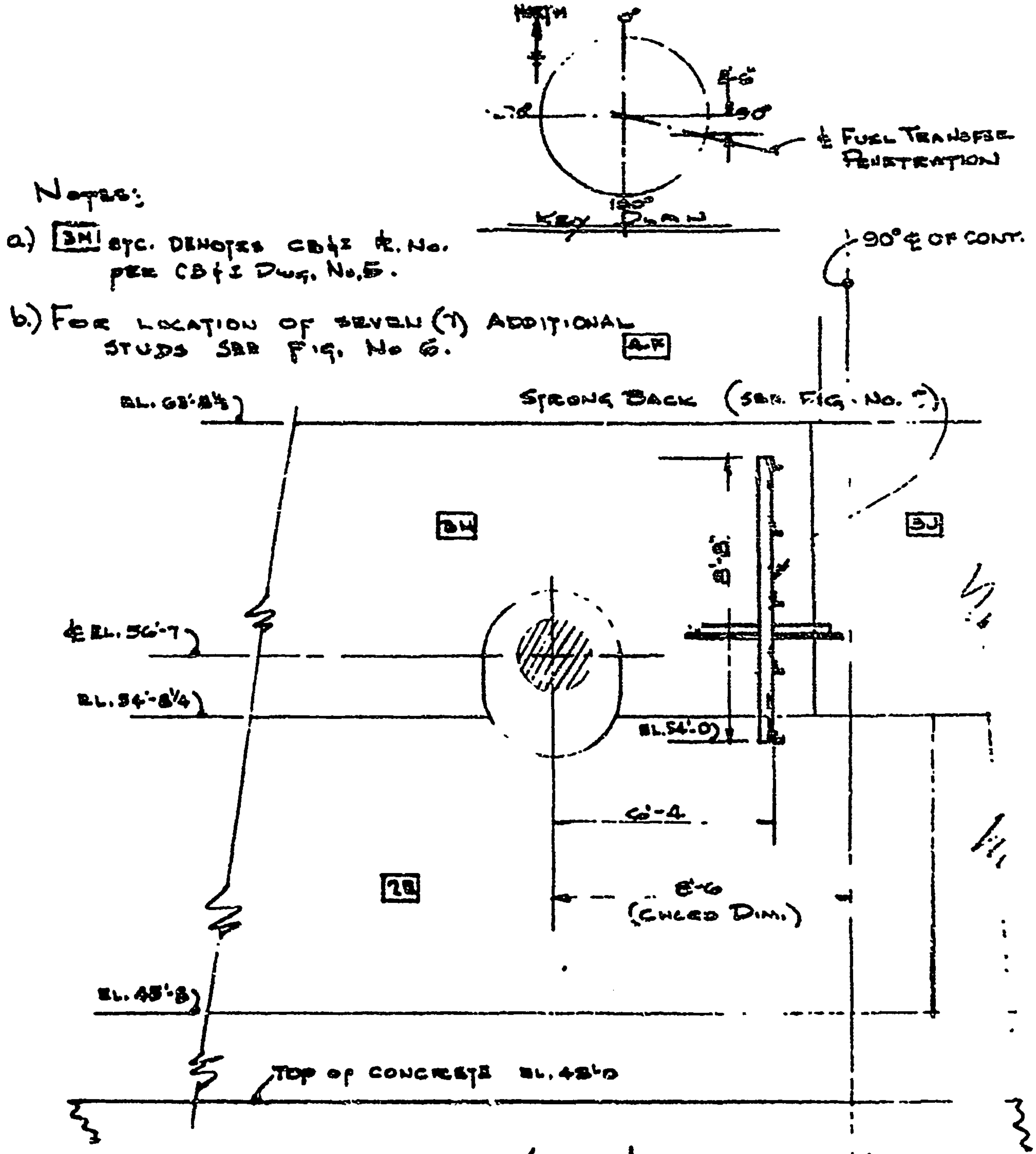
MEASUREMENTS AND EVALUATION

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 attach Figures 2 and 3. Figure 4 is a plot of this data in the form of a contour ma of the buckled zone.

Specifications 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.

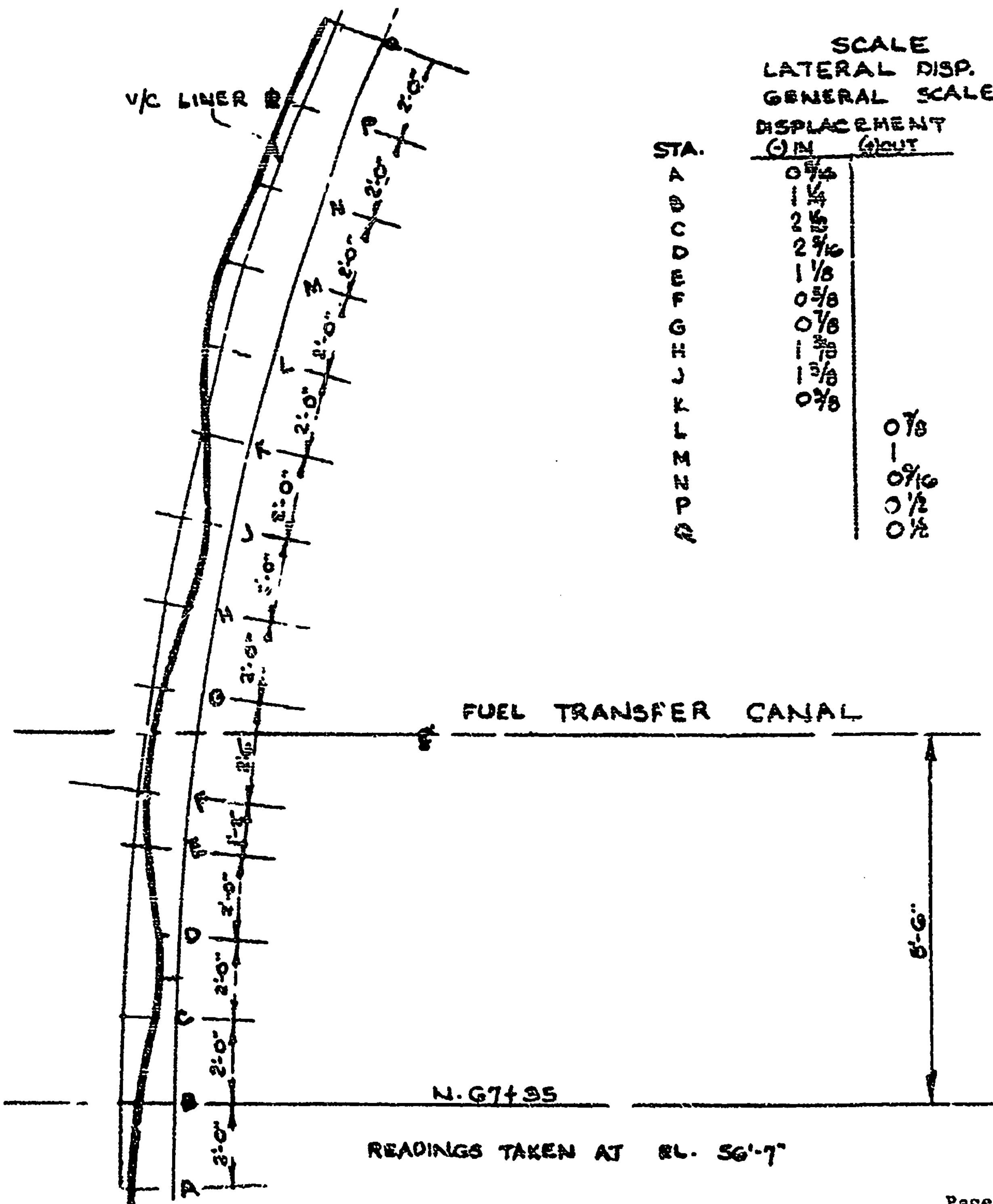
FIGURE 1



View Looking West

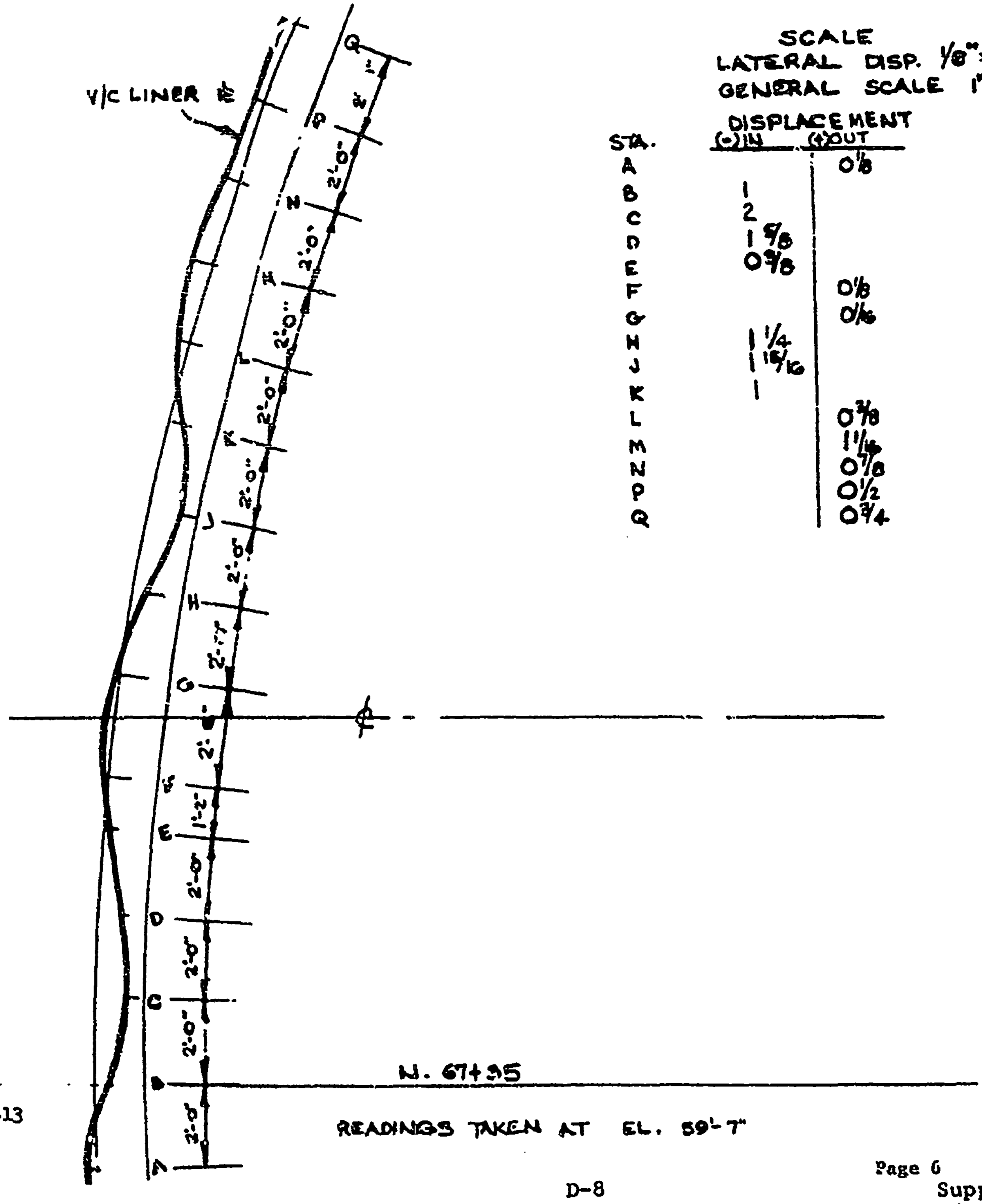
Scale ¼" = 1'-0"

FIGURE 2



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

FIGURE 3

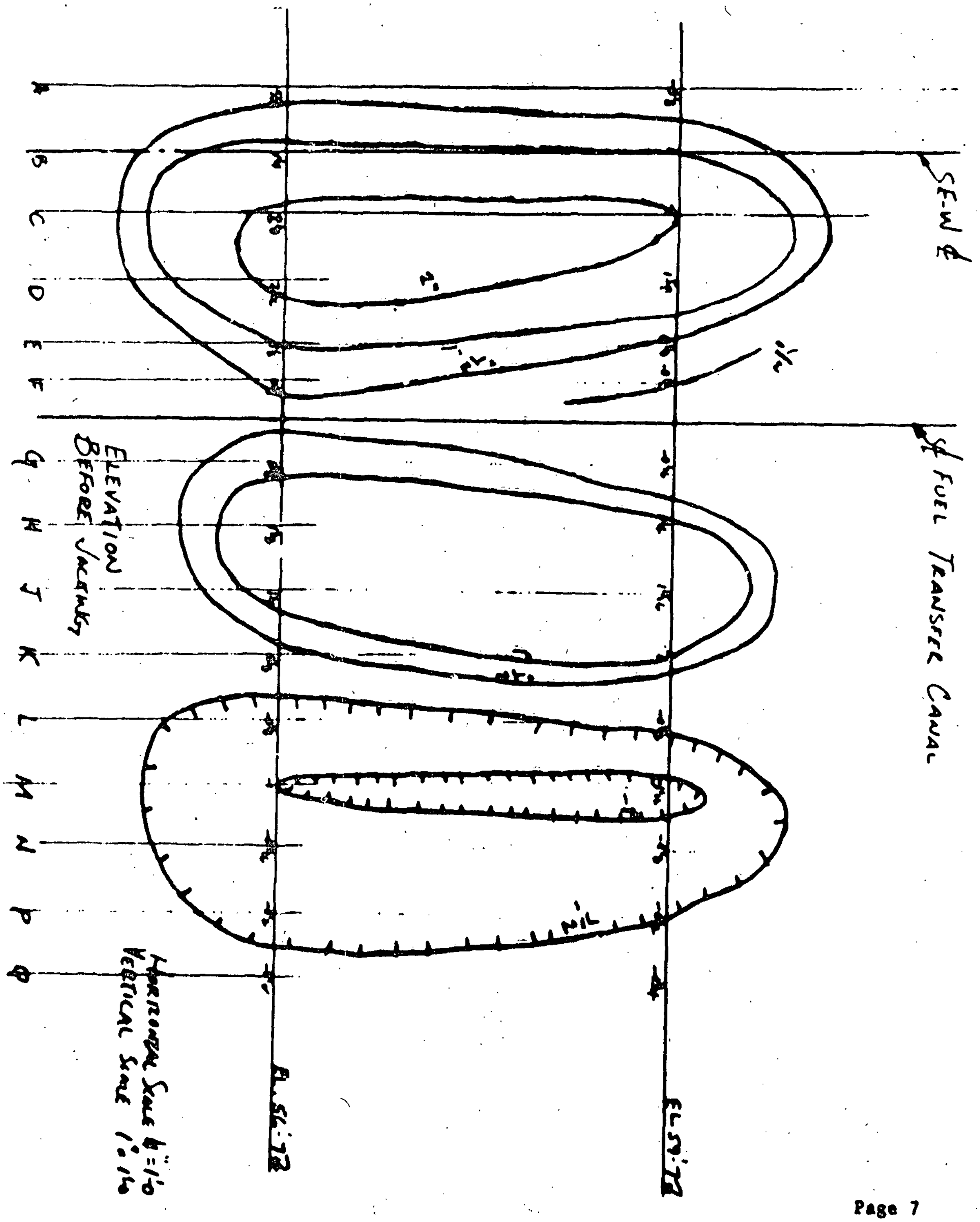


SCALE
LATERAL DISP. 1/8" = 1"
GENERAL SCALE 1" = 3'

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

M-13

FIGURE 4



D-9

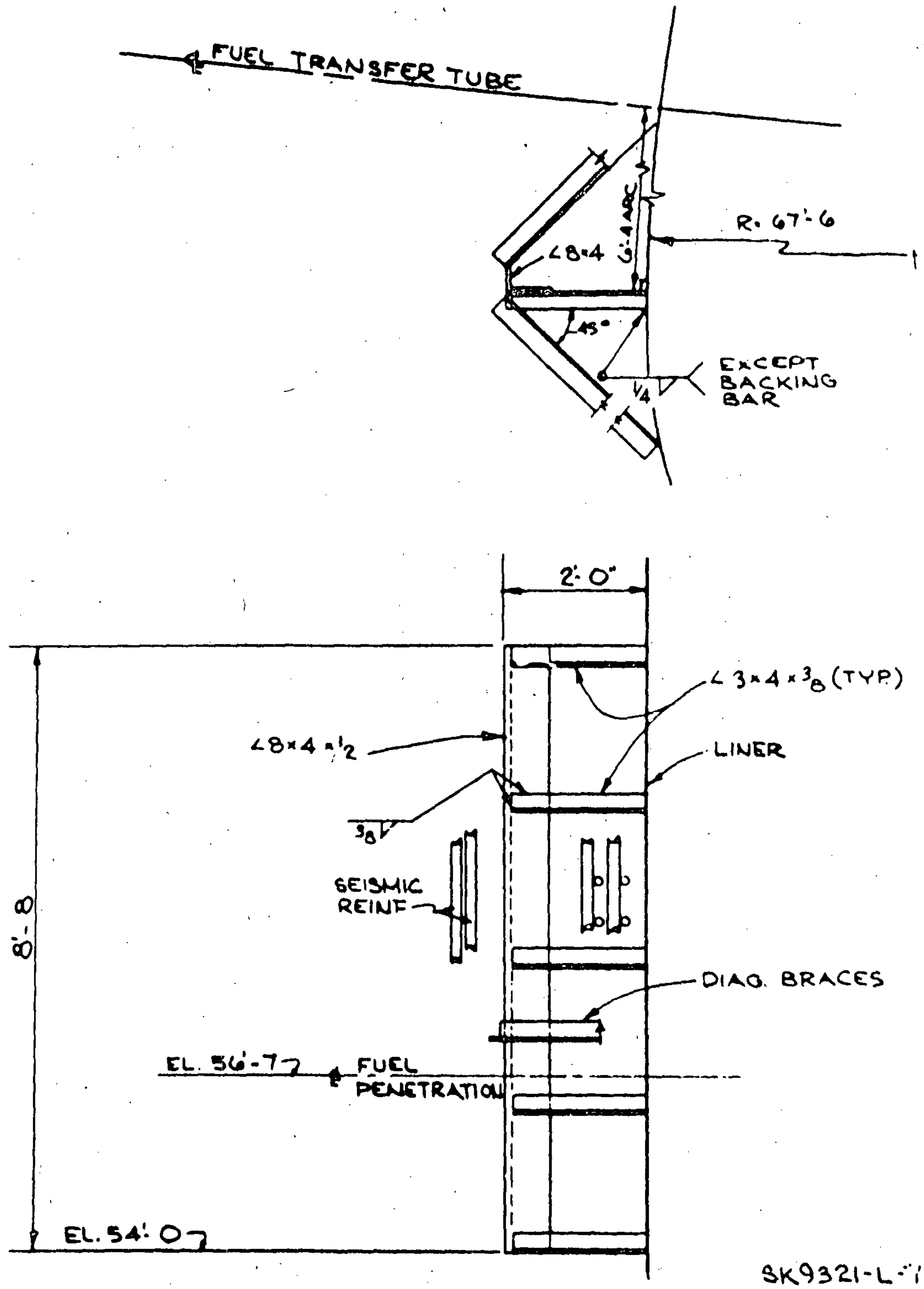
CORRECTIVE ACTION

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.

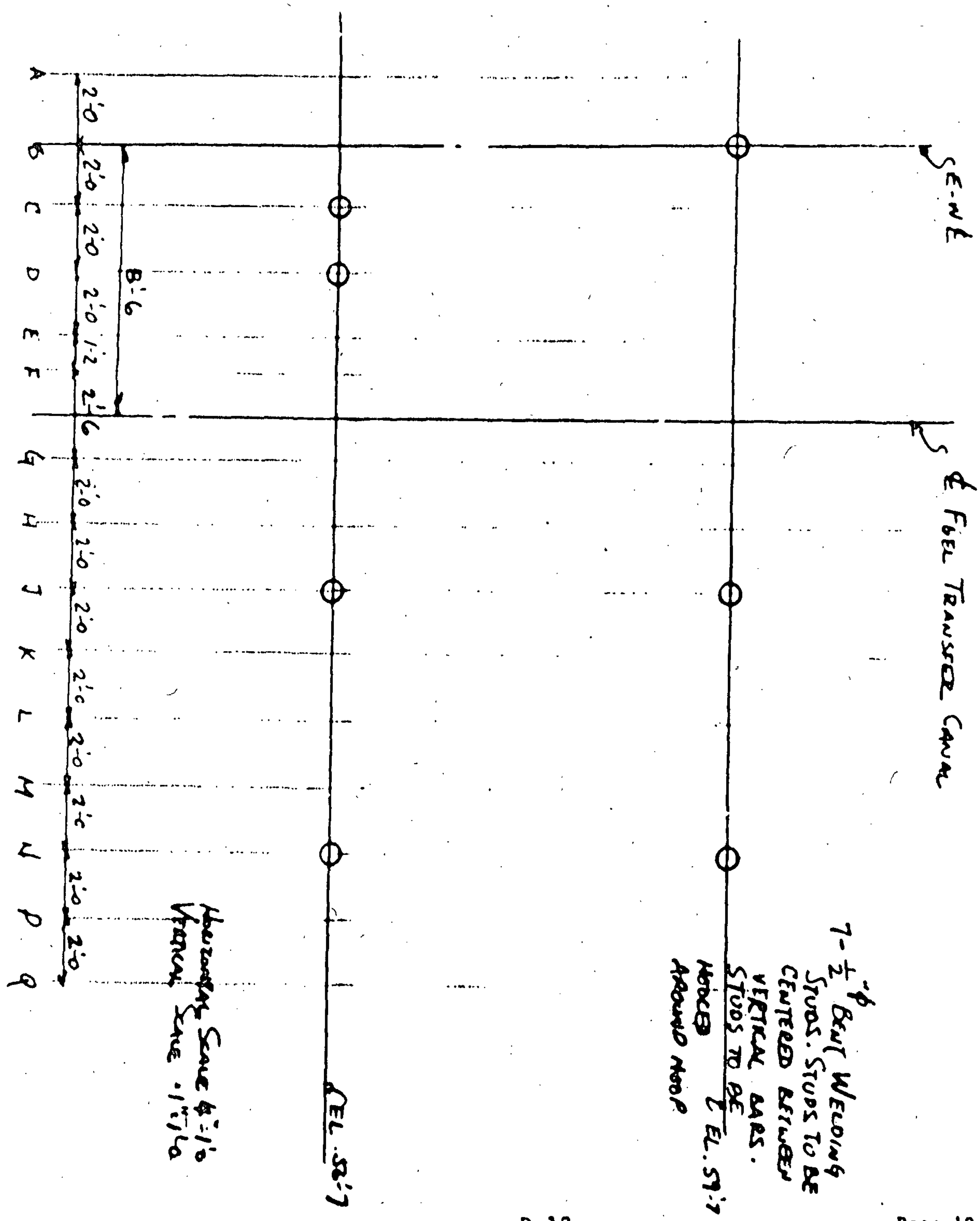
Figures 7, 8 and 9 show the liner configuration after corrective action was taken. Figure 10 is a contour map of the buckled zone after corrections and in addition, all of the studs including the original installation and the additional studs are shown. At all points the liner is within the allowable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

FIGURE 5



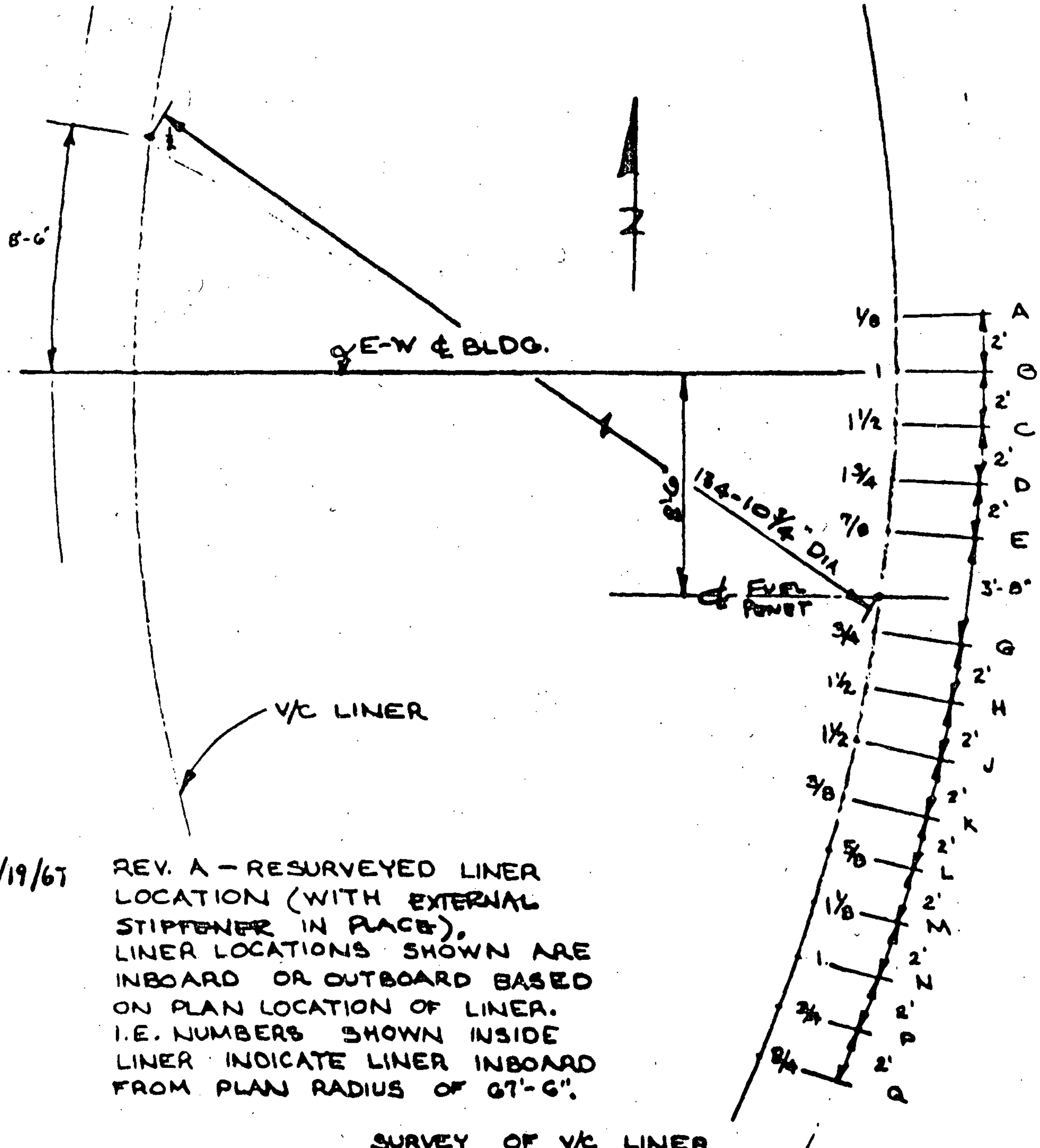
SK9321-L-1677

FIGURE 6



A
B
C
D
E
F
G
H
J
K
L
M
N
P
Q
8'-6
SE-NE
℄ FUEL TRANSFER CANAL
℄ EL. 53'-7
7-½"Ø BENT WELDING STUDS. STUDS TO BE CENTERED BETWEEN VERTICAL BARS. STUDS TO BE HOOKED AROUND HOOP
℄ EL. 57'-7

FIGURE 7



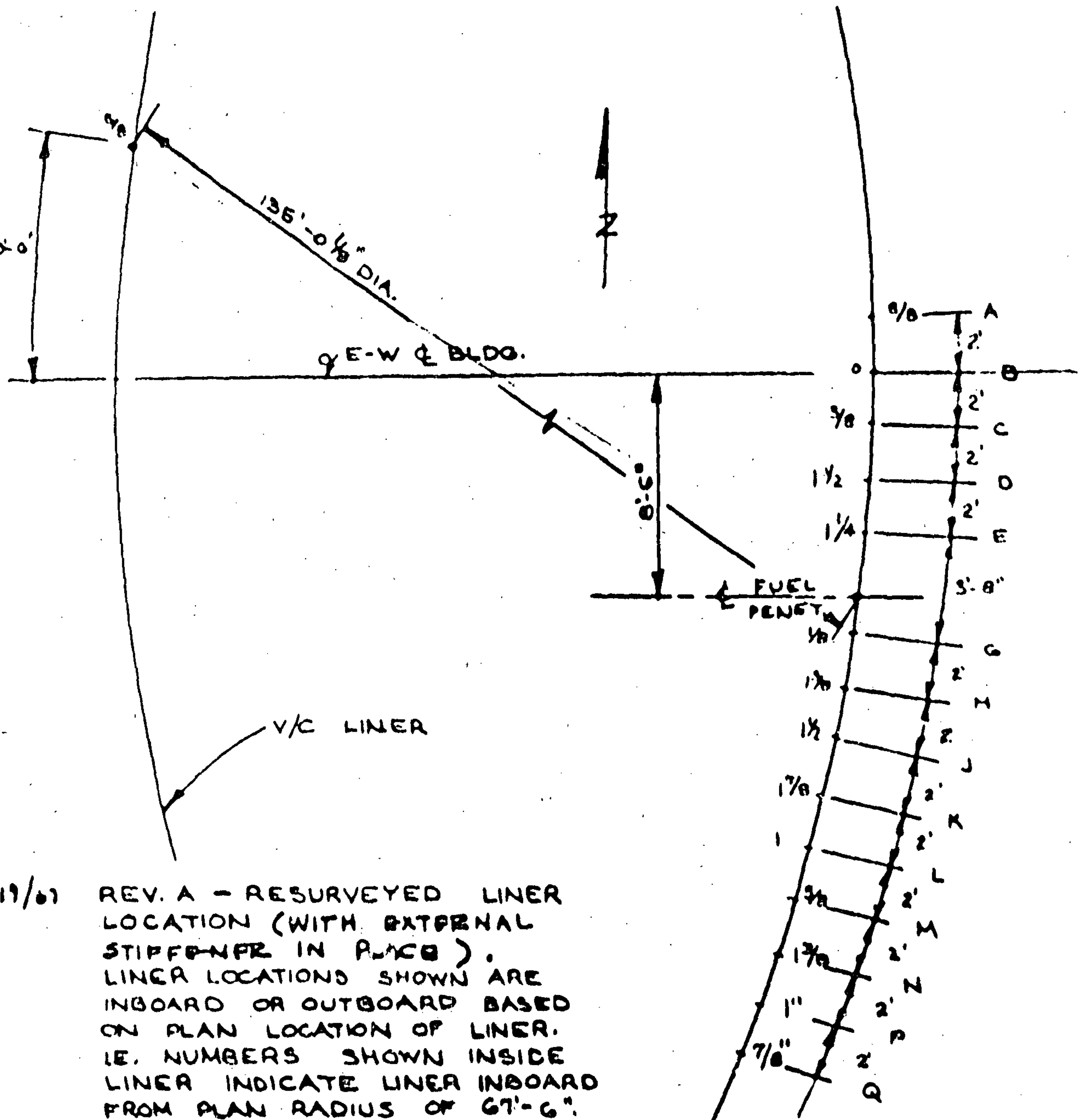
12/19/67 REV. A – RESURVEYED LINER LOCATION (WITH EXTERNAL STIFFENER IN PLACE). LINER LOCATIONS SHOWN ARE INBOARD OR OUTBOARD BASED ON PLAN LOCATION OF LINER. I.E. NUMBERS SHOWN INSIDE LINER INDICATE LINER INBOARD FROM PLAN RADIUS OF 67'-6".

SURVEY OF V/C LINER

EL. 56'-7'

M-18

FIGURE 8

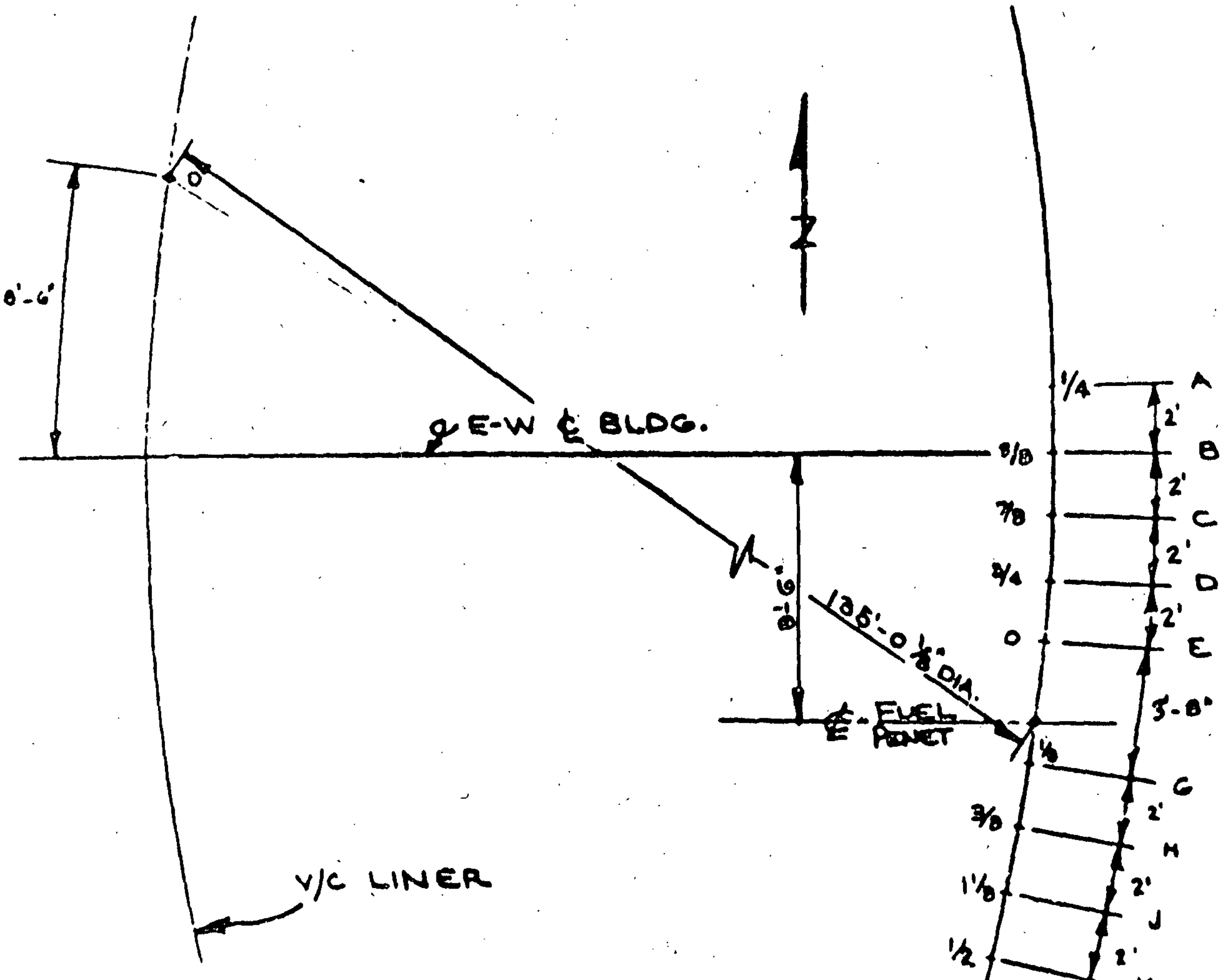


12/19/67 REV. A – RESURVEYED LINER LOCATION (WITH EXTERNAL STIFFENER IN PLACE). LINER LOCATIONS SHOWN ARE INBOARD OR OUTBOARD BASED ON PLAN LOCATION OF LINER. I.E. NUMBERS SHOWN INSIDE LINER INDICATE LINER INBOARD FROM PLAN RADIUS OF 67'-6".

SURVEY OF V/C LINER

EL. 59'-1"

FIGURE 9



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".

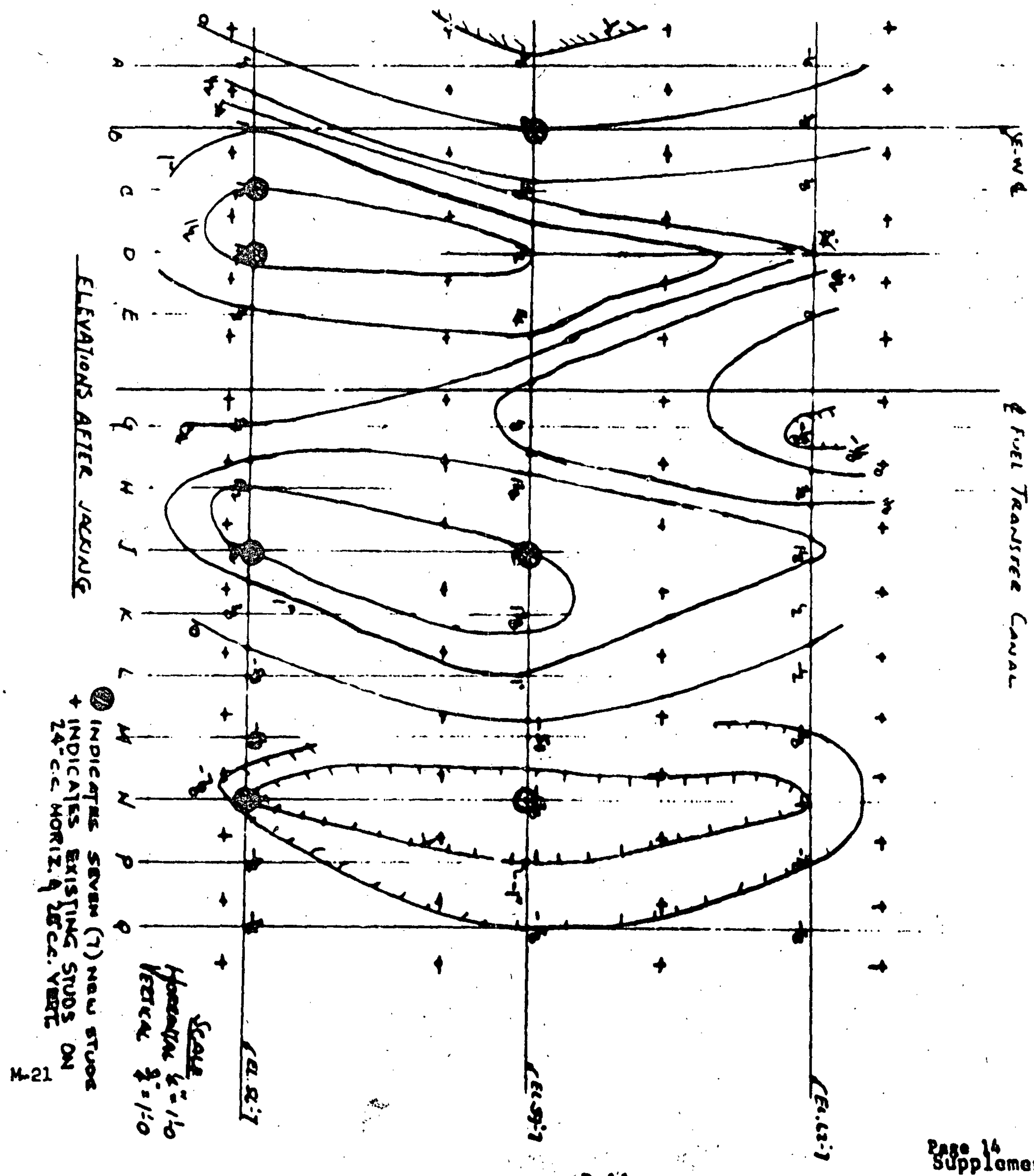
M-20

SURVEY OF V/C LINER

EL. 82'-7"

D-15

FIGURE 10



ELEVATIONS AFTER JACKING
€ FUEL TRANSFER CANAL
INDICATES SEVEN (7) NEW STUDS
INDICATES EXISTING STUDS ON 24" C.C. HORIZ. & 16" C.C. VERT.
SCALE
M-21