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U. S. Atomic Energy Commission

Docket No. 50-286
Exhibit B-5

Regulatory Suppl File Cy.

CONSOLIDATED EDISON COMPANY OF NEW YORK, INC.
INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

FIFTH SUPPLEMENT TO:
PRELIMINARY SAFETY ANALYSIS REPORT

4823

PREFACE

Supplement 5 to the Indian Point Nuclear Generating Unit No. 3 consists of additional information to supersede or augment information presented in Supplement 1 to the Unit No. 3 PSAR and of page revisions to Supplement 1.

Additional information is presented on the following general subjects which correspond to tab numbers in this supplement.

1) Modified ECC System
2) Primary Auxiliary Building Description
3) Conduct of Operations and Initial Testing Program
4) Quality Assurance/Quality Control Program
5) Hydrogen Generation and Recombiner
6) Containment Spray R & D
7) Provisions for Post Loss of Coolant Accident Protection
8) Off-Site Dose Evaluation
9) Thermal Shock Status
10) Control and Protection System Status
11) Flooding at the Site
12) Sodium Hydroxide Injection
13) Seismic Stress Limit Curves
14) Dose Calculations
15) Miscellaneous
    a) Power Supplies to the Site
    b) Tornado
    c) Leak Rate Detection Capability
    d) Steam Break Analysis*
    e) Loss-of-Coolant Design Criteria*
    f) Minimum DNBR Criteria*
    g) Criteria for Testing Air Cleanup System*
    h) Instrumentation Systems*
    i) Locked Open Valves in the Containment *

*Pages attached on these items for insertion into text to replace the corresponding pages of Supplement 1 to the PSAR.

MODIFIED ECC SYSTEM

A number of improvements have been incorporated into the Indian Point Unit No. 3 emergency core cooling system.

Completely separate lines have been provided from each residual heat exchanger to the reactor coolant system. Residual heat exchanger No. 1 supplies water to cold legs 1 and 2 while residual heat exchanger No. 2 supplies water to cold legs 3 and 4. In the event of a passive failure to one line, instrumentation and remotely operated valves are provided to enable the operator to detect the failure and to isolate the appropriate heat exchanger. Adequate cooling is provided by one heat exchanger train. See Figure 1.

The original system had the capability of using either residual heat exchanger for the recirculation mode head to high head pumps. This capability has been retained by providing individual feeder lines from each heat exchanger to the common recirculation high head line.

Provision for redundancy in the auxiliary cooling system (shown in Figure 2) is discussed in Supplement 1 to the PSAR in Item 17 (F-6.0) and the modified service water system is shown in Figure 8 of this supplement.

The boric acid injection tank has been placed in the discharge side of the high head safety injection pumps, enabling boric acid to be injected into the cold legs of the reactor coolant system by means of a simple direct tank pump through. This improvement is in line with current Westinghouse practice.

The containment sump suction line is contained within a concentric guard pipe which is connected to the containment liner and terminates within a leak tight compartment. This modification obviates the necessity for the use of the internal containment sump isolation valve.

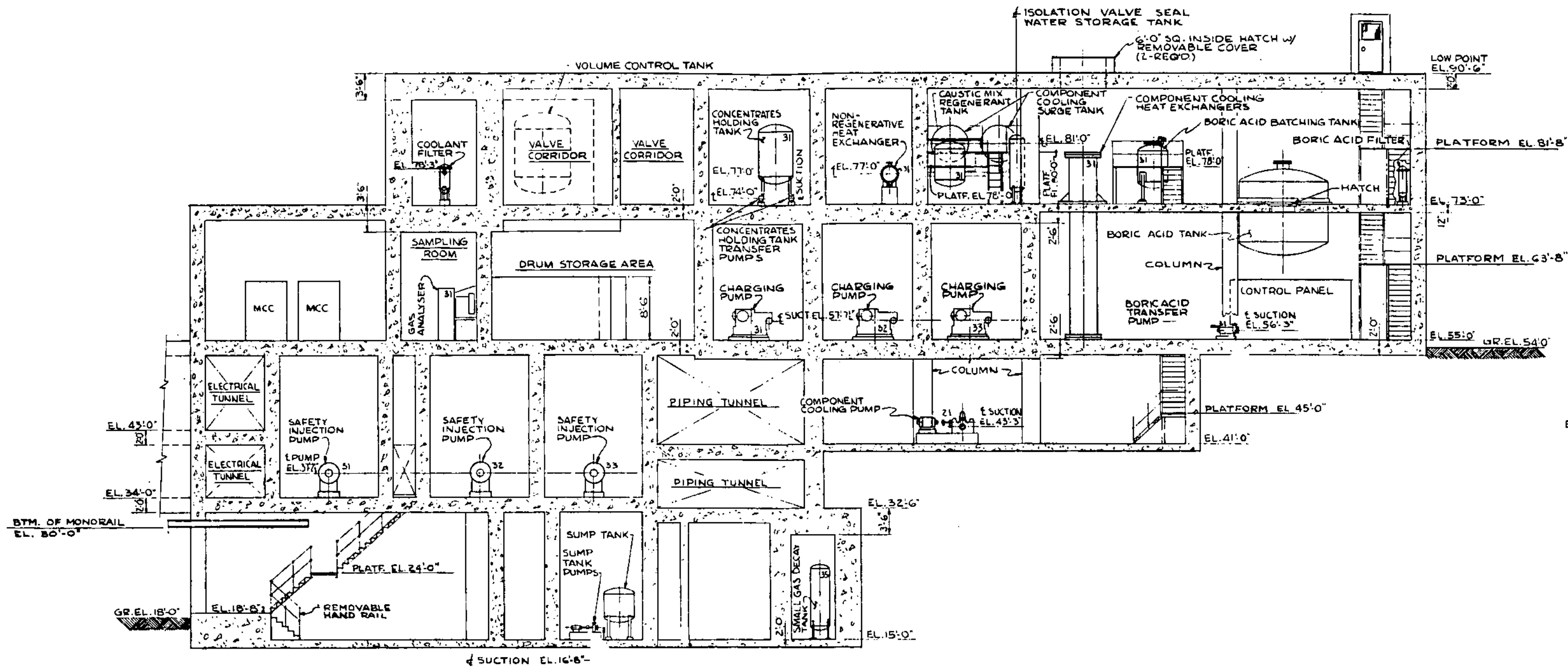
Provison has been made for a second containment sump line to be used in the event that the post loss of vessel coolant accident proves to be a credible accident.

PRIMARY AUXILIARY BUILDING

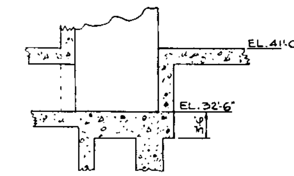
The auxiliary building layout with the required provisions for space for the addition equipment specified discussion of PLOCAP in this supplement is shown in Figures 3 through 6. Figure 7 shows the auxiliary building ventilation system along with the containment air filtration system.

Figure 1
Supplement 5

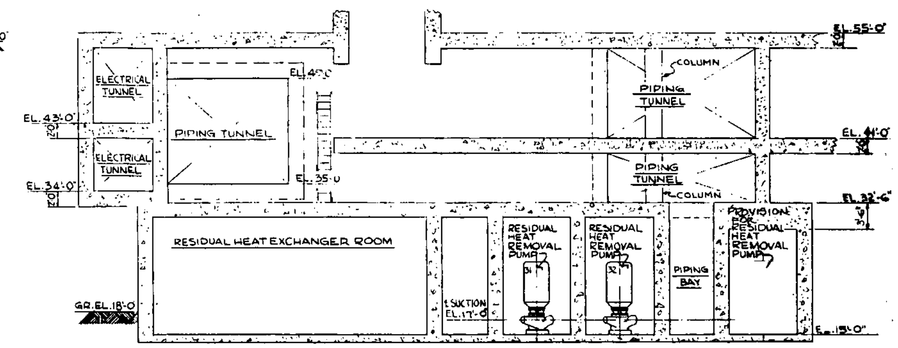
Figure 2
Supplement 5



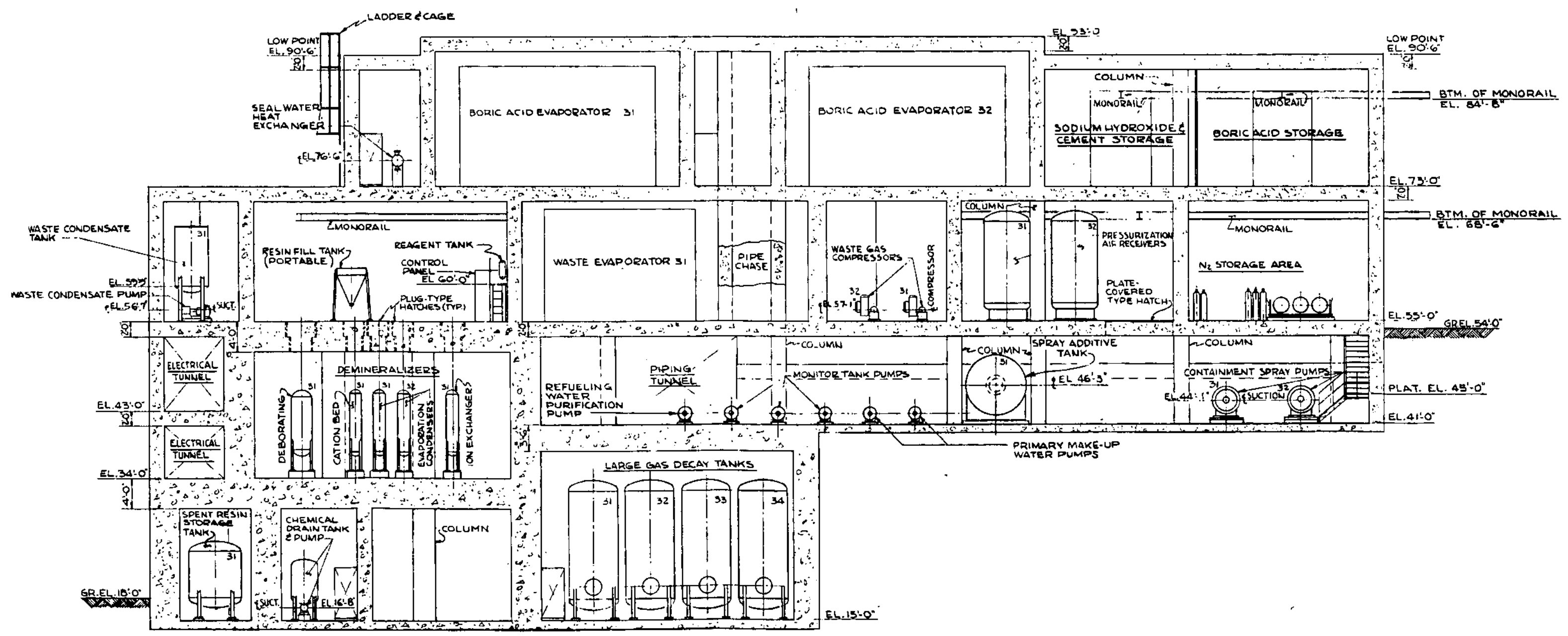
ELEVATION "B-B"
LOOKING NORTH
SCALE: 1/8" = 1'-0"



SECTION "H-H"
LOOKING NORTH
SCALE: 1/8" = 1'-0"

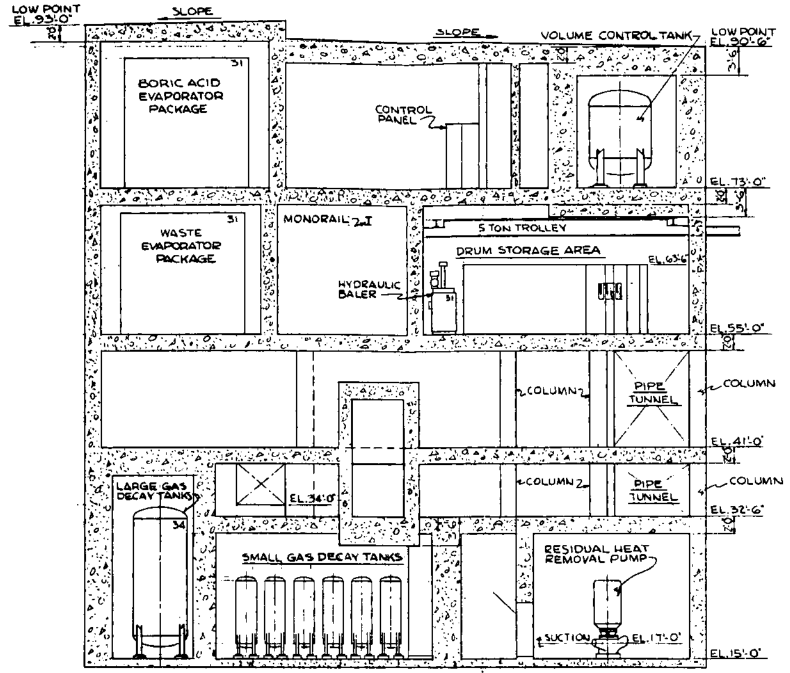


PARTIAL ELEVATION "G-G"
LOOKING NORTH
SCALE: 1/8" = 1'-0"

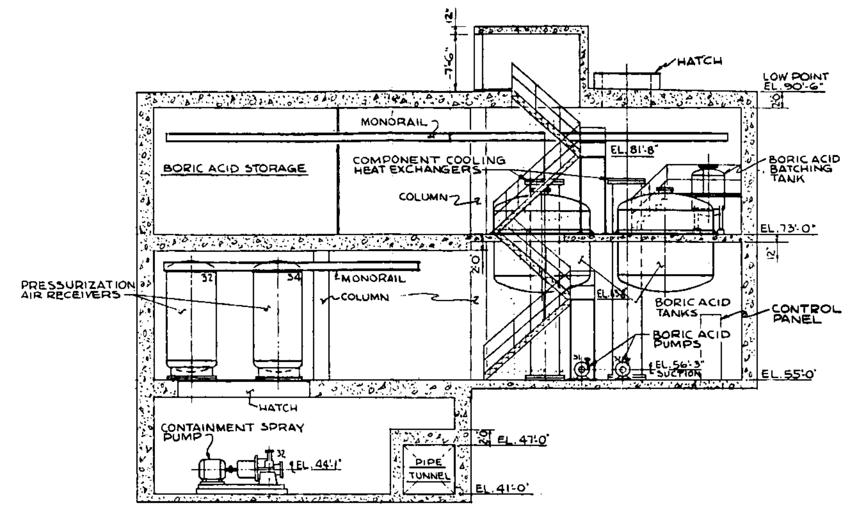


ELEVATION "A-A"
LOOKING NORTH
SCALE: 1/8" = 1'-0"

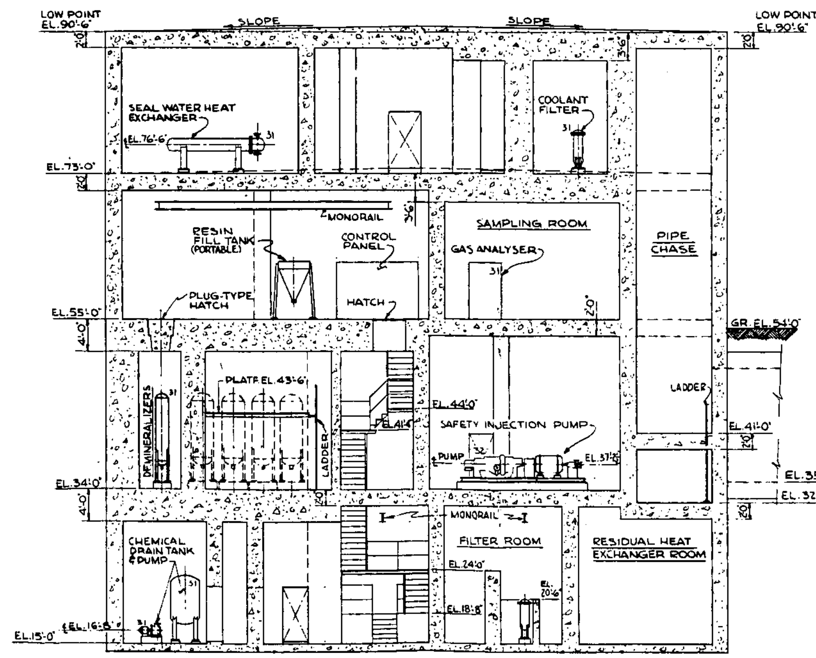
Figure 3
Supplement 5



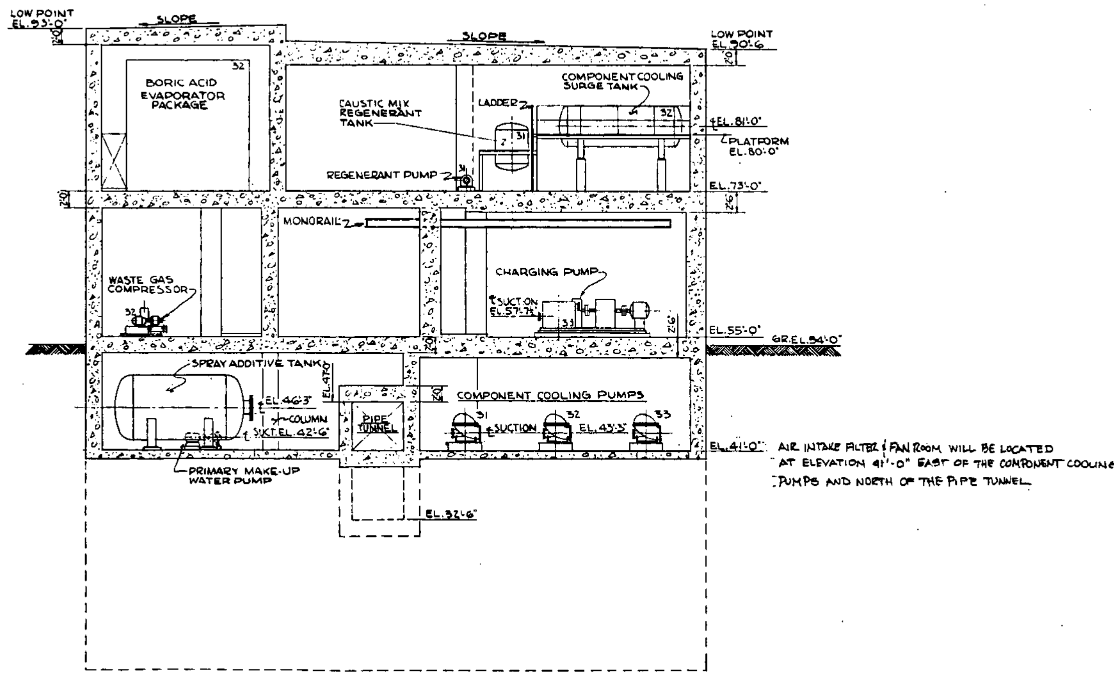
ELEVATION "D-D"
LOOKING WEST
SCALE: 1/8" = 1'-0"



ELEVATION "F-F"
LOOKING WEST
SCALE: 1/8" = 1'-0"

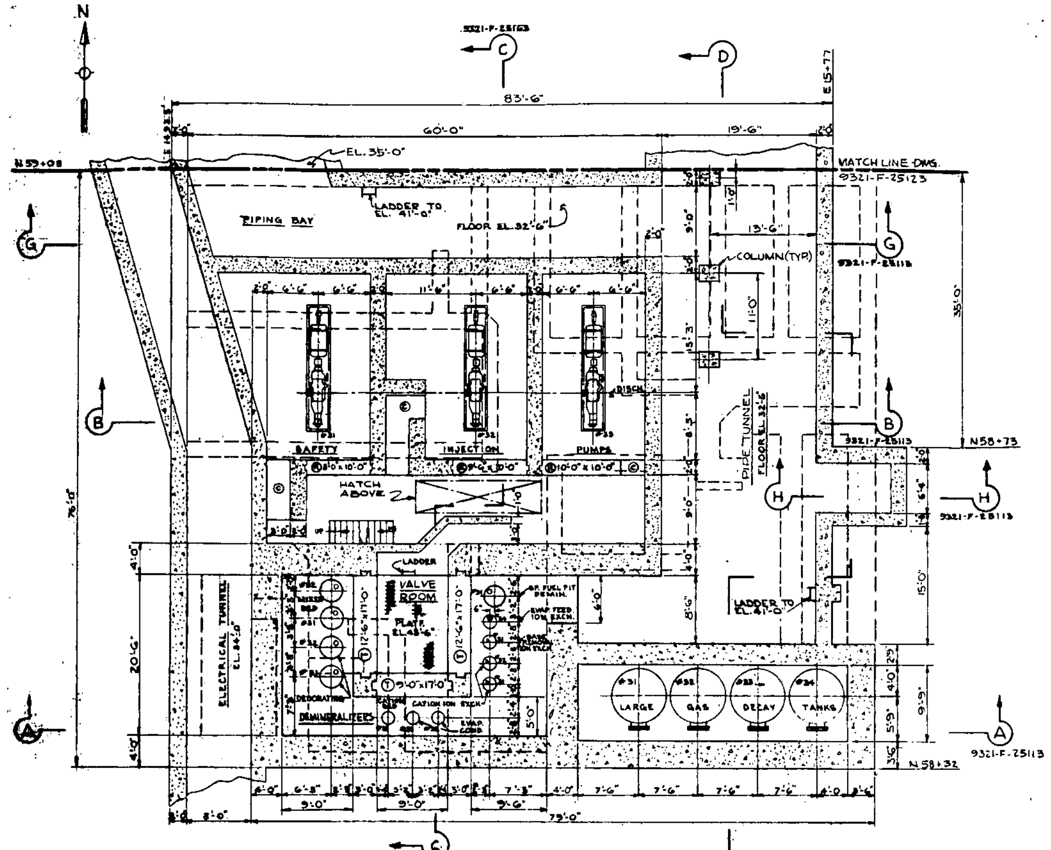


ELEVATION "C-C"
LOOKING WEST
SCALE: 1/8" = 1'-0"

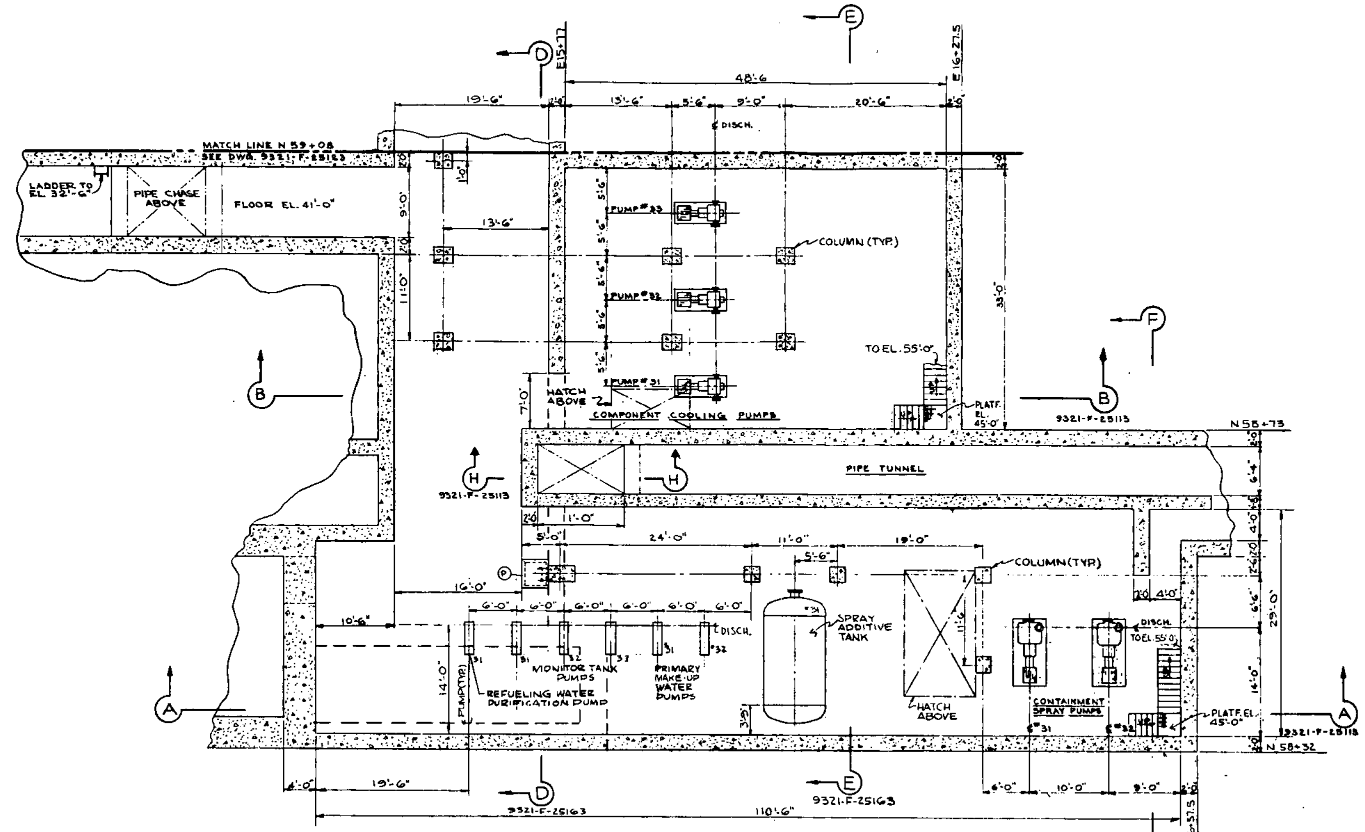


ELEVATION "E-E"
LOOKING WEST
SCALE: 1/8" = 1'-0"

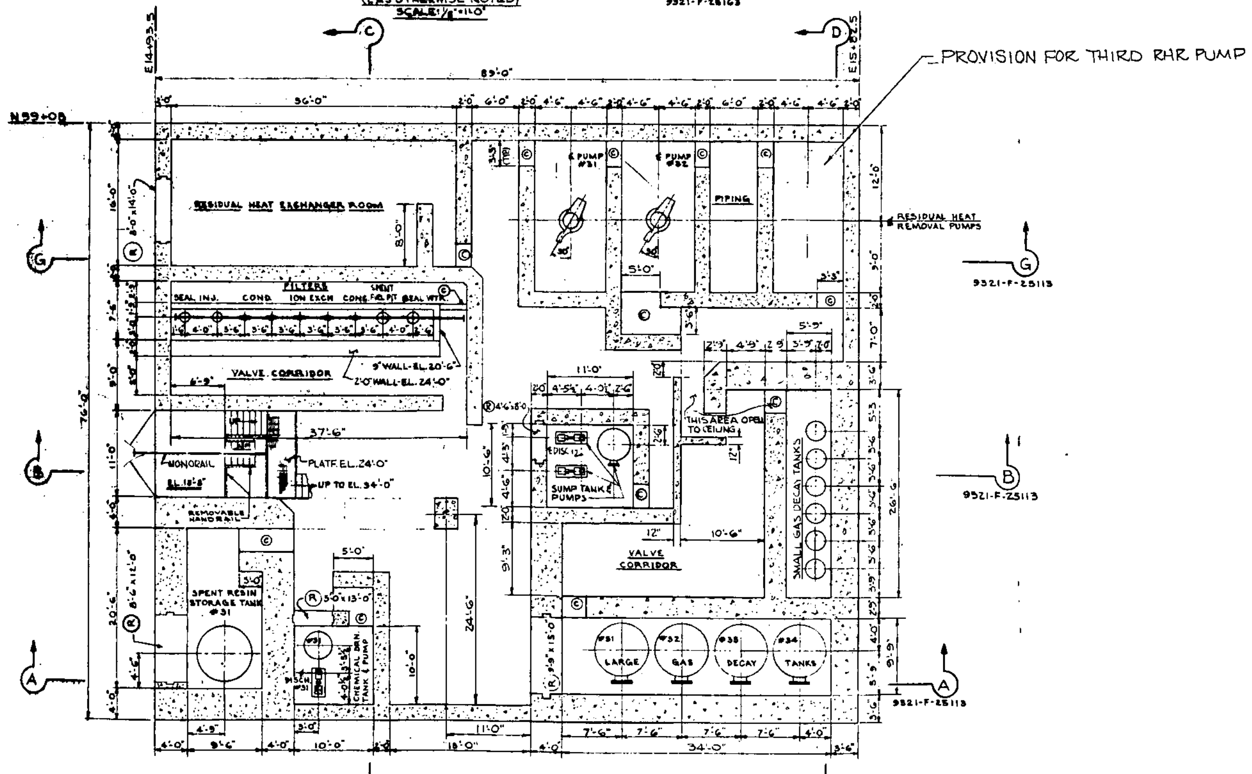
Figure 4
Supplement 5



PLAN @ EL. 34'-0"
(EAS OTHERWISE NOTED)
SCALE: 1/8"=1'-0"



PLAN @ EL. 41'-0"
SCALE: 1/8"=1'-0"



PLAN @ EL. 15'-0"
SCALE: 1/8"=1'-0"

LEGEND

Ⓒ - CURB (4" HIGH)
Ⓟ - PLUG TYPE HATCH
Ⓡ - REMOVABLE WALL (DIMENSIONED WIDTH x HEIGHT ON PLAN)
Ⓣ - TEMPORARY OPENING IN WALL
Ⓗ - HATCH W/REMOVABLE COVER

Figure 5
Supplement 5

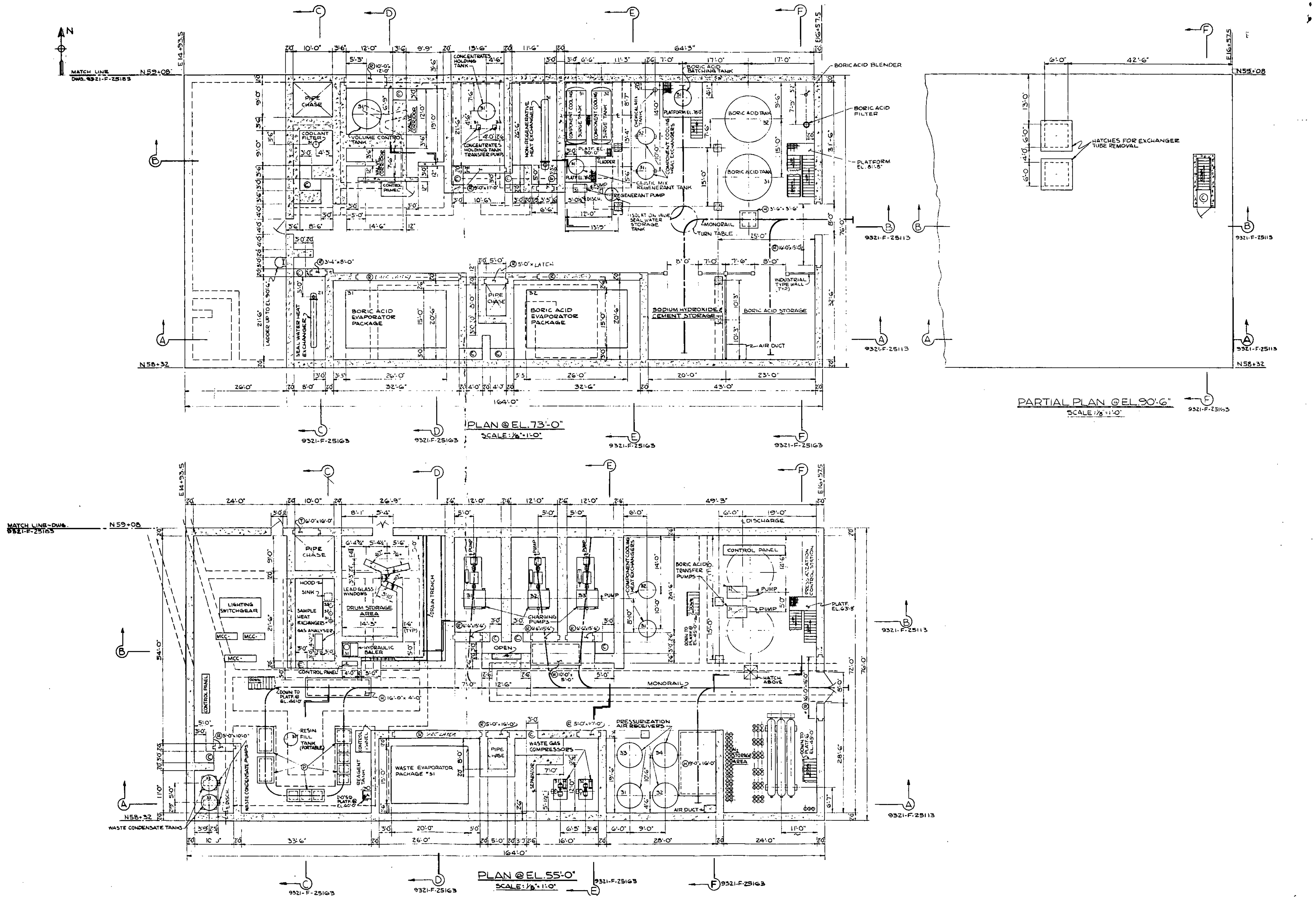


Figure 6
Supplement 5

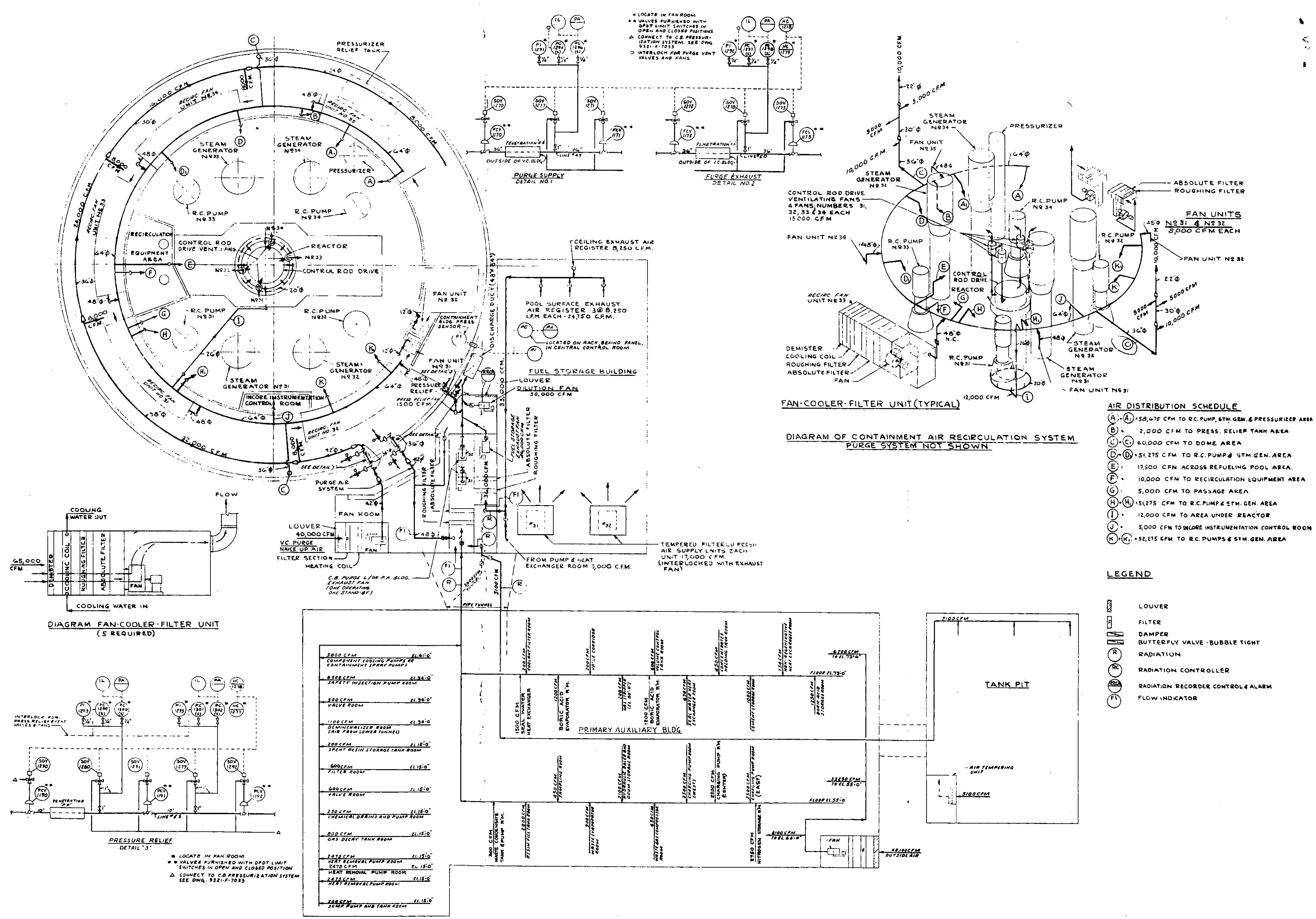


Figure 7
Supplement 5

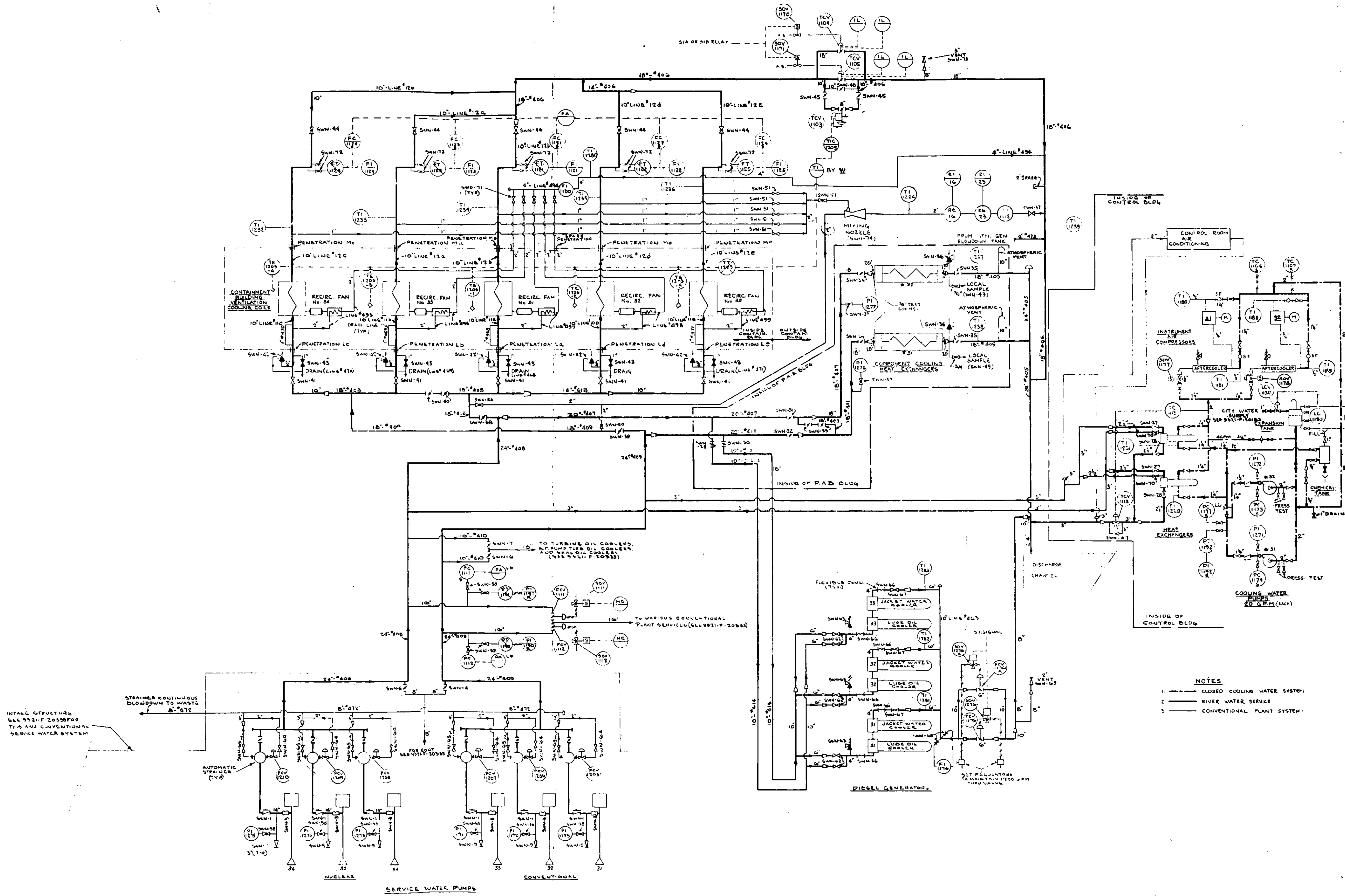


Figure 8
Supplement 5

CONDUCT OF OPERATIONS AND INITIAL TESTING PROGRAM

I Station Organization

The proposed organization chart for the administering and manning of Indian Point as a three unit station parallels closely the organization structure presented as Exhibit O-1A submitted in August, 1961 by Consolidated Edison in support of its application for a Provisional Operating License for its Unit No. 1 at Indian Point. Generally, the lines of authority remain unchanged. Certain changes from the organization that existed at that time have occurred, and are as follows:

1. The Reactor Engineer originally carried line responsibility for day to day nuclear plant operations. We presently feel that the technical and administrative demands of this function dictate the advisability of detaching this man from these duties and making him responsible for technical staff activities, in addition to his basic responsibility for providing core physics surveillance, burnup follow, etc. Accordingly, we have separated this title from the line organization.

2. All of the maintenance forces, including those responsible for unit overhauls, will report to the General Superintendent of the Station through the Maintenance Supervisor and the Superintendent.

II Staff Requirements

The existence of a trained, experienced operating organization for Indian Point Unit No. 1, and the current efforts at expanding that organization for Unit No. 2 responsibility, make very much simpler the task of preparing for the operation of Unit No. 3. At the upper level of station management, most, if not all of those presently responsible for the key positions associated with the operation of Unit No. 1 and the preparations for operation of Unit No. 2 will be responsible for Unit No. 3. We expect the addition of a single Assistant to the Superintendent to be sufficient for the upper station management duties resulting from the needs of Unit No. 3.

The minimum requirements for the General Superintendent will be that he hold a B.S. Degree in Engineering. He must also have a minimum of ten years of experience in engineering, construction, operation, or management of an electric generating facility, of which a minimum of three years will be nuclear reactor design or operating experience.

The minimum requirements for the Superintendent will be that he hold a B.S. Degree in Engineering, and a minimum of three years of nuclear reactor design or operating experience. If, including the required reactor operating experience, he has had a minimum of ten years of experience in the construction, operation, or management of an electric generating facility, the requirement of an engineering degree may be waived. He must hold a Senior Reactor Operator's License for the facility.

The minimum requirements for the Reactor Engineer will be as established for the General Superintendent except that the requirement of ten years of conventional plant experience will not be required. He must hold a Senior Reactor Operating License for the facility.

The minimum requirements for Performance Superintendent will be as established for the Reactor Engineer.

The minimum requirements for the Test Engineer (Health Physicist) will be that he hold a B.S. Degree in Engineering or one of the physical sciences. He must also have had a minimum of five years of experience in a radiation safety program involving the direct administering of that program, and must have demonstrated proficiency in all significant phases of the execution of the program for which he is to be responsible.

The minimum requirements for Maintenance Supervisor will be that he have had a minimum of five years of experience in installation, construction and maintenance of heavy machinery. He must have shown adequate proficiency in the owner's training course in radiation protection.

The instrumentation and control supervisor will have had a minimum of five years of experience in the installation, testing and maintenance of instruments and controls of which no less than three years shall have been in a nuclear facility.

The chemistry supervisor will have as minimum requirements a B.S. in Chemistry or Chemical Engineering and three years experience in chemical and radio-chemical organizations.

At the shift level, a complete separate force will naturally be required, including Watch Foremen and the necessary number of operators and operating mechanics to fill the posts described herein. Similar to the policy applied in preparing for the startup of Unit No. 2, a pre-requisite for candidacy for the startup Watch Foreman and Control Operator "A" (reactor control board operator) on Unit No. 3 will be that he hold a Senior Reactor Operator's License on Unit No. 1 or 2 and a Reactor Operator's license on Unit No. 1 or 2 respectively. In this manner, we have assured ourselves of the man's competence as to his ability to accept and qualify in keeping with the Atomic Energy Commission's licensing procedures. At the same time, we have provided the very best possible background for those men approaching the job of being educated in the specifics of the unit for which they are to be responsible.

The operating crew for Unit No. 3 will be as described in organization chart form included herein. The Watch Foreman will hold a Senior Operator's license for the Unit No. 3 facility, and there will be a minimum of four in number. The Control Operator "A" will hold a minimum of a Reactor Operator's License for the Unit No. 3 facility, and there will be a minimum of four in number. The Operating Mechanics will hold responsibility for operating and minor maintenance surveillance in two major operating areas of the facility. They will not be licensed under the Commission's regulations, and there will be a minimum of nine in number. It is likely, although uncertain at this point in time of planning, that most, and possible all of these personnel will cycle in their duties with their counterpart operators and supervisors assigned to Unit No. 2 and possibly even Unit No. 1

III Training

As stated earlier, the Unit No. 3 personnel, both staff and operating force, will be drawn largely from the ranks of the Units Nos. 1 and 2 organizations; consequently, the basic training for the assumption of Unit No. 3 duties will have been acquired in the form of direct operating responsibilities for the earlier units. Specific training for Unit No. 3 will be provided by making both the Unit No. 3 Watch Foremen and Control Operators "A" candidates available approximately one year before fuel loading in order for them to receive the training needed to learn the specifics of Unit No. 3.

This will including a training period administered by the Westinghouse Electric Corporation on the design philosophy of the facility, system descriptions, engineering flow diagrams, component descriptions, instrumentation and control information, and generally, all of the information necessary to render competent, from a design background standpoint, those individuals who are to have operating responsibility. This will be followed by assignments to those individuals of a system or systems, in which he is expected to make himself fully expert, and for which he is to write the operating procedures and test specifications. These procedures and specifications will then be subject to final review and approval by the General Superintendent of the station, the Production Department general office staff, the Westinghouse Electric Corporation, and the Company's Nuclear Facility Safety Committee. As his expertise in his assigned responsibilities reaches fruition, he is then required to impart his knowledge to each of the others through the medium of regular classroom sessions.

As construction approaches completion on the various systems and sub-systems, those supervisors and operators who have been assigned these responsibilities will supervise the pre-startup testing of their systems. As that testing is completed, the systems are then accepted, one at a time, by these people, under the direction of a cognizant individual at the Superintendent level or above, and this continues until the facility has been completed and fully turned over to the operating organization. This program for training and startup operations was utilized for Unit No. 1 and is presently being utilized for Unit No. 2 with considerable success.

During the period that these men are training for the assignments noted above, courses of instruction in various subjects pertinent to background knowledge and specific preparation for their license examinations will have been given. These are administered at the Indian Point facility by the Station Staff as a continuing program for the training of men for Unit #2 and for replacements for Unit #1. They will continue for Unit #3. The subjects covered in these training sessions are listed below with their approximate instruction hours, not including the field instruction which accompany the sessions:

	Lecture Hours
Reactor Theory	180
Core Design	40
Primary System Design	40
Auxiliary Systems	10
Operating Characteristics	50
Reactivity Control	50
Safety Systems	20
Emergency Systems	20
Containment and Shielding	10
Operating Procedures	30
Radiation Monitoring	10
Health Physics	15
Facility License including Technical Specifications	20
Core Loading, Procedures	20
Radioactive Waste Handling	10
Total	525

Consolidated Edison employees assigned to Indian Point Station are thoroughly trained in the areas of fire prevention and protection. General rules and regulations pertaining to the prevention of fires at any of its electric 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 any 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 and matters. Invaluable experience is gained at this school in as much as actual fires of the type encountered in generating stations are controlled and extinguished by the trainees.

The detail fire-emergency procedure presently in effect at Indian Point Station will be revised to broaden its scope to include Unit No. 3. It is not anticipated that the procedural actions will be sustantially changed, however, as the same general plan of fire control will be applicable to both units.

Adequate familiarization with the fire-emergency procedure 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 fire drills are generally initiated without prior announcement.

IV Written Procedures

Operating procedures will be developed in advance of startup of the facility. These procedures will be maintained in the possession of all licensed Reactor and Senior Reactor Operators, and at the various key operating locations within the facility.

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 Station General Superintendent and Consolidated Edison's Nuclear Facility Safety Committee.

Procedures to be followed in the event of an unscheduled radioactive release to the environment in excess of regulation limits will closely parallel those procedures presently in effect for Unit No. 1, and will involve off-site surveys by Consolidated Edison personnel, as well as a close cooperative effort with outside agencies such as the Atomic Energy Commission Division of Compliance, the A.E.C. New York Operations Office Radiological Emergency Assistance Team, the New York State Department of Health, the New York State Department of Police, and the United States Coast Guard. The extent of involvement with these agencies will, as with Unit No. 1, depend on the magnitude of the release, how that release relates to the requirements of the facility Technical Specifications, and the consequent degree of need for cooperative efforts of such agencies.

V Records

Records concerning facility operations will be maintained in the form of log books, charts, and such other internal reports as may be needed to document pertinent operating conditions. The principal logs to be maintained will be those in the Central Control Room, Watch Foreman's Office, the shift chemist, and the shift health physics technician. These logs will include descriptions of the operating conditions which exist at the time, descriptions of significant operational efforts accomplished during the shift, and such operating events or circumstances as are deemed pertinent to maintain proper continuity of knowledge and understanding of such matters as responsibility in those areas is passed on from shift to shift.

A record of radiation safety conditions, internal and environmental, is maintained in the form of appropriate log entries, and continuous recording chart information in those functional systems and areas provided with radiation survey instruments. In addition, Radiation Work Permit survey information provides the necessary record of radiation exposure conditions prior to job commencement, and actual personnel radiation exposure information is maintained in the form of film badge and dosimeter records. Records of controlled radiation releases to the environment will be maintained by the station chemical and health physics groups, and all necessary information describing specific radioactivity concentrations, total volumes to be released, along with any dilution requirements, will be entered on the Radioactive Waste Release Permit prepared for each release.

VI Review and Audit of Operations

In matters such as design changes to the facility involving unreviewed safety questions, changes to operating procedures, or changes to the technical specifications, a review of the question by the Nuclear Facility Safety Committee will be requested by the station General Superintendent. If the Committee concludes that such a change is acceptable from the standpoint of safety, the change will be approved by the Committee, or, if approval by the Atomic Energy Commission is required, a change request for permission to make the change will be initiated by the Committee.

A continuing review of operations is 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. In addition, there is periodic review of facility operations by the Nuclear Facility Safety Committee. Frequent communications, both written and oral, between the station General Superintendent and the Chairman of the Nuclear Facility Safety Committee assure the degree of awareness by that Committee of facility operations necessary for it to meet its audit responsibilities. There are frequent meetings of the Committee, at which time it reviews shutdowns of the facility and the reasons therefore, unusual operating conditions, releases to the environment, and proposed changes to the facility and its operating procedures. Periodically, the meetings of the Committee are held at Indian Point so as to allow the entire Committee an opportunity to tour the facility and scrutinize its operations. Approximately once a month, a different member visits the station for a personal audit of facility activities.

Nuclear Facility Safety Committee

The Nuclear Facility Safety Committee was established on April 23, 1962 for Indian Point Station, Unit No. 1. This Committee will also perform the same function for Units No. 2 and No. 3.

The purpose of the Nuclear Facility Safety Committee will be to review the operation of the facility, the operating organization, the procedures for operation, changes in the facility and the conduct of tests or experiments therein:

a. Membership

The Committee shall consist of:

1. The Manager of the System Operation Department, who shall be Chairman.

2. The Reactor Engineer at the Indian Point Station.

3. An engineer from the Mechanical Engineering Department having experience in nuclear engineering.

4. An engineer from the Mechanical Engineering Department or from The Chemical Bureau of the Production Department having experience in nuclear chemistry.

5. An engineer from the Electrical Engineering Department or from the Mechanical Engineering Department having experience in nuclear instrumentation and control.

6. The Radiation Safety Officer of the Company.

In the temporary absence of any member of the Committee, the member shall designate an Alternate with similar experience to attend meetings of the Committee with full authority to act in place of the absent member. In addition to designating an Alternate in his capacity as a member of the Committee, the Chairman shall also designate the member of the Committee who is to act as Chairman in his absence.

Designations of alternates will be made in writing. If any member of the Committee becomes incapacitated or resigns or is otherwise unable or unwilling to serve, his replacement shall-be designated by the Vice President of the Company to whom such member reports in his regular duties.

b. Functions

1. When requested by the General Superintendent of the facility, the Committee shall review (a) any proposed change in the facility, (b) any proposed change in the procedure and (c) any proposed test or experiment to be conducted therein, to reach a determination whether such proposed change, test or experiment involves an unreviewed safety question as defined in the facility Operating License or involves a change in or deviation from the Technical Specifications.

2. The Committee shall review facility operations, including equipment, from time to time to determine their adherence to the requirements of said License and shall make reports of the result of such review to the General Superintendent and to the Vice President of the Company in charge of Engineering, Production and Distribution Operations.

3. The Committee shall make such other studies and analyses of station organization, operations and procedures as may be requested by the General Superintendent or by corporate officers of Edison.

c. Procedure

1. If the 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 invlove 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 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.

2. If the 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 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 General Superintendent, together with a statement of the reasons for the Committee decision and the General Superintendent may then 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, Production and Distribution Operations for their review with a copy to the General Superintendent. One of said Vice Presidents shall thereupon forward the report and request to the Atomic Energy Commission for approval unless, after review, the three Vice Presidents either (a) disagree with the opinion of the Committee that approval of the Atomic Energy Commission is required, or (b) decide that the proposed change, test or experiment is not necessary from the standpoint of Company policy or operations.

3. If during a review by the Committee of facility operations and equipment the Committee determines that a variation from the Technical Specifications or an unreviewed safety question exists, the Committee shall immediately notify the General Superintendent who shall take any steps needed to assure safety.

   The Committee shall then prepare a report recommending, as appropriate, a change in the facility, in the operating procedures or in the Technical Specifications, together with any appropriate safety analysis. The Committee shall forward its report, together with a request for approval by the Atomic Energy Committion, to the above designated Vice presidents, with a copy to the General Superintendent.

VII Testing

Testing of systems prior to acceptance and startup will be performed in accordance with the basic philosophy for testing and acceptability as set forth in Consolidated Edison's Exhibit T, in the matter of its application for a Provisional Operating License for its Unit No. 1 at Indian Point, Docket 50-3.

Initial Tests and Operation

The initial testing and start-up operation of the unit systems prior to full power operation of the unit will include tests prior to reactor fueling, core loading, precritical tests, zero power tests and power level escalation.

The purpose of this program will be to test and operate the reactor and its various systems (1) to make certain that the equipment has been installed and will operate in accordance with the design requirements (2) to provide safe procedures for initial fuel loading and to determine zero power values of core parameters significant to the design and operation and (3) to bring the unit to its rated capacity in a safe and orderly fashion.

Procedures stating the test purpose, conditions, precautions, and limitations will be prepared for each test. The procedures will include a delineation of administrative procedures and test responsibility, equipment clearance procedures, and an overall sequence of startup operations.

The test program described in the following sections is based upon the reference plant design and experience gained during startup of other units. Detailed procedures will specify the sequence of tests to be conducted and the conditions under which each will be performed to ensure the relevance and consistency of the results obtained. This will include expected values and acceptance criteria which demonstrate the degree to which the facility does meet design criteria.

Tests Prior to Reactor Fueling

The following tabulation is the sequence of major start-up tests and operations performed to place all equipment in the specified system in service. Consolidated Edison Company of New York, Inc. in cooperation with Westinghouse Electric Corporation will prepare detailed test procedures prior to scheduled initial testing of systems and determination of reactor physics parameters.

The test objectives incorporate testing of redundant equipment where it is involved.

Abnormal plant conditions may be simulated during testing when such conditions do not endanger personnel, equipment or contaminate clean systems. Where predicted emergency or abnormal conditions are involved in the testing program the detailed operation will be provided in the test procedure.

Acceptance Criterion for all components and systems will be that the test results are acceptable when the test objectives are met within the design specification limits and within the applicable Technical Specification.

1. Switchgear System
2. Voice Communications System
3. Service Water System
4. Fire Protection System
5. Instrument and Service Air Systems
6. Nitrogen Storage System

7. Reactor Coolant System Cleaning
8. Reactor Containment Air Recirculation and Filtration System
9. Feedwater and Condensate Circulation Systems
10. Auxiliary Coolant System
11. Chemical Feed System
12. Chemical & Volume Control System
13. Safety Injection System
14. Fuel Handling System
15. Containment Isolation and Isolation Valve Seal Water Systems
16. Containment Penetration and Weld Channel Pressurization System
17. Reactor Containment High Pressure Test.
18. Cold Hydrostatic Tests
19. Radiation Monitoring System
20. Nuclear Instrumentation System
21. Radioactive Waste Disposal System
22. Sampling System
23. Instrumentation Calibration
24. Hot Functional Test
    - Reactor Coolant System
    - Chemical & Volume Control System
    - Sampling System
    - Waste Disposal System
25. Primary and Secondary Systems Safety Valves Tests
26. Turbine Steam Seal & Blowdown Systems
27. Emergency Diesel-Electric System
28. Containment System
    - Strength Test
    - Gross Leak Rate Test
    - Sensitive Leak Rate Test
29. Containment Spray System

<u>Objectives of Tests Prior to Reactor Fueling</u>

System or Test	Test Objective
1. Switchgear System (Electrical Tests)	To ensure continuity, circuit integrity, and the correct and reliable functioning of electrical apparatus. Electrical tests will be performed on transformers, switchgear, turbine-generator, motors, cables, control circuits, excitation switchgear, D-C System, annunciator system, lighting distribution switchboard, communication system and miscellaneous equipment.
2. Voice Communication Systems	To verify proper communication between all intra plant stations, for interconnection to commercial phone service and to balance and adjust amplifiers and speakers.
3. Service Water System	To verify, prior to critical operations, the design head-capacity characteristics of the service water pumps, that the system will supply design flow rate through all heat exchangers, and will meet the specified requirements when operated in the safeguards mode.

System or Test	Test Objective
4. Fire Protection System	To verify proper operation of the system by ensuring the design specifications are met for the fire service booster pump and fire service pumps, checking that automatic start functions operate as designed, and that level and pressure controls meet specifications.
5. Instrument & Service Air Systems	To verify the operation of all compressors to design specifications, the manual and automatic operation of controls at design setpoints, design air-dryer cycle time and moisture content of discharge air, and proper air pressure to each instrument served by the system.
6. Nitrogen Storage System	To verify system integrity, valve operability, regulating and reducing station performance and the ability to supply nitrogen to interconnecting systems as required.
7. Reactor Coolant System Cleaning	To flush and clean the reactor coolant and related primary systems to obtain the degree of cleanliness required for the intended service. Provisions to maintain cleanliness integrity and protection from contamination sources will be made after system cleaning and acceptance. The system, component, or section of a system shall be considered clean when the flush cloth shows no grindings, filings or insoluble particulate matter larger than 40 microns (lower limit of naked eye visibility).
8. Reactor Containment Air Recirculation and Filtration System	To verify, prior to critical operation, the fan capacities, and the remote and automatic operation of system louvers and valves in accordance with the design specifications.
9. Feedwater and Condensate Circulation Systems	To verify proper operation of feedwater and circulating water pumps according to specifications, valve and control operability and set points, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing will be performed when a steam supply is available.
10. Auxiliary Coolant System	To verify component cooling flow to all components, and to verify proper operation of instrumentation, controllers and alarms, Specifically each of the three loops, i.e. component cooling loop, residual heat removal loop and spent fuel pit cooling loop will be tested.

System or Test	Test Objective
11. **Chemical Feed System**	To verify valve and control operability and set-points, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing will be performed when a steam supply is available.
12. **Chemical and Volume Control System (CVCS)**	To verify, prior to critical operation, that the CVCS functions as specified in the system description and appropriate technical manuals.
13. Safety Injection System (SIS)	To verify prior to critical operation, response to control signals and sequencing of the pumps, valves, and controllers of this system as specified in the system description and the manufacturers technical manuals; and check the time required to actuate the system after a safety injection signal is received.
14. **Fuel Handling System**	To show that the system design is capable of providing a safe and effective means of transporting and handling fuel from the time it reaches the plant until it leaves the plant.
15. **Containment Isolation & Isolation Valve Seal Water Systems**	To verify the capability for reliable operation and demonstrate the manual and automatic operation of the system. Demonstrate the operation and proper sequence of isolation valve closure and seal water addition. Demonstrate function of Isolation Valve Seal Water System independent of other systems. Demonstrate the operation and system response time induced by an isolation signal. Manual valves will be manipulated to assure proper operation of the seal gas injection portion of the system.
16. **Containment Penetration and Weld Channel Pressurization System**	To verify air system and nitrogen backup system integrity, operate valves, check flowmeters and pressure gauges as required to ensure pressure differential meets design specifications.
17. **Reactor Containment High Pressure Test**	To verify prior to critical operation, the structural integrity and leak tightness of the containment.

System or Test	Test Objective
18. Cold Hydrostatic Tests	To verify the integrity and leak tightness of the Reactor Coolant System and related primary systems with the performance of a hydrostatic test at the specified test pressure with no visible leakage, nor distortion.
19. Radiation Monitoring System	To verify the calibration, operability, and alarm setpoints of all radiation level monitors, air particulate monitors, gas monitors and liquid monitors which are included in the Operational Radiation Monitoring System and the Area Radiation Monitoring System.
20. Nuclear Instrumentation	To ensure that the instrumentation system is capable of monitoring the reactor leakage neutron flux from source range through 120% of full power and that protective functions are operating properly.
21. Radioactive Waste Disposal System	To verify satisfactory flow characteristics through the equipment; to demonstrate satisfactory performance of pumps and instruments; to check for leak-tightness of piping and equipment, and to verify proper operation of alarms, instrumentation and controls.
22. Sampling System	To verify that a specified quantity of representative fluid can be obtained safely and at design conditions from each sampling point.
23. Instrument Calibration	Instrumentation and control devices will be checked to assure their accuracy. Primary sensing elements, transducers, transmitters, receivers, recorders and indicators will be thoroughly inspected and adjusted for accuracy of their set point characteristics. Interconnecting piping and wiring will be checked for continuity and functional requirements. Each device will then be tested in accordance with established test procedures. Limit switches used for initiating indicating lights, alarms and inter-lock functions will be checked under actual or simulated operating conditions. Control devices will be exercised to assure proper operation with the required accuracy and response characteristics. Set points for devices will be checked and adjusted to their specified values.

System or Test	Test Objective
23. Instrument Calibration (Cont'd)	Each individual circuit of the reactor and turbine protection systems will be tested to verify that appropriate signals initiate reactor and turbine trips. As a signal level corresponding to the particular condition is reached, trip or cutback functions will annunciate as provided for the particular channel under test.
24. Hot Functional Tests	The Reactor Coolant System will be tested to check heatup (using pump heat) and cooldown procedures; to demonstrate satisfactory performance of components prior to installation of the core; to verify proper operation of instrumentation, controllers and alarms; and to provide operating conditions for checkout of auxiliary systems. The Chemical and Volume Control System will be tested to determine that water can be charged at rated flow against normal Reactor Coolant System pressure; to check letdown flow against design rate for each pressure reduction station; to determine the response of the system to changes in pressurizer level; to check procedures and components used in boric acid batching and transfer operations; to check operation of the reactor makeup control; to check operation of the excess letdown and seal water flowpath; and to verify proper operation of instrumentation, controllers and alarms. The Sampling System will be tested to determine that a specified quantity of representative fluid can be obtained safely and at design conditions from each sampling point. The Auxiliary Coolant System will be tested to evaluate its ability to remove heat from reactor coolant; to verify component cooling flow to all components; and to verify proper operation of instrumentation, controllers and alarms. The Safety Injection System will be tested to check the time required to actuate the system after a safety injection signal is received; to check that pumps and motor operated valves are properly sequenced; and to verify proper operation of instrumentation, controllers and alarms.

System or Test	Test Objective
24. Hot Functional Tests	The Radioactive Waste Disposal System will be tested to verify satisfactory flow characteristics through the equipment; to demonstrate satisfactory performance of pumps and instruments; to check for leak-tightness of piping and equipment; and to verify proper operation of alarms. The Ventilation System will be tested to adjust proper flow characteristics of ducts and equipment; to demonstrate satisfactory performance of fans, filters, and coolers; and to verify proper operation of instruments and alarms.
25. Primary and Secondary Systems Safety Valves Tests	To test and set pressurizer and boiler safety and relief valves to ensure each valve lifts as specified, relieves excessive pressure down to the blowdown set point and reseats clean and tight.
26. Turbine Steam Seal & Blowdown Systems	To verify valve and control operability and setpoints, flushing and hydro as applicable, inspection for completeness and integrity. Functional testing will be performed when a steam supply is available.
27. Emergency Diesel Electric System	To demonstrate that the system is capable of providing power for operation of vital equipment under power failure conditions.

28. Containment System

a. Strength Test:

A pressure test will be made on the completed building using air. During this test, measurements and observations will be made to verify the adequacy of the structure design.

b. Gross Leak Rate Tests:

The basis for the integrated leak rate tests which will be performed on the completed building will be the reference volume method. This leakage test will be performed with the double penetration and weld channel zones open to the atmosphere. After it has been assured that there are no defects remaining from construction, a sensitive leak rate test will be conducted.

c. Sensitive Leak Rate Test:

The sensitive leak rate test will include only the volume of the weld channels and double penetrations. Because this volume is about 1000 times smaller than the containment free volume, the sensitivity and accuracy attainable in this leak test is increased correspondingly over that attainable by integrated leak rate testing. The sensitive leak rate test will be conducted with the penetrations and weld channels at pressure and with the containment building at atmospheric pressure.

29. Containment Spray System

a. Component Testing

All active components in the Containment Spray System will be tested both in pre-operational performance tests in the manufacturer's shop and in-place testing after installation. Initially the containment spray nozzle availability will be tested by blowing smoke through the nozzles and observing the flow through the various nozzles. The air test lines for checking the spray nozzles, connect downstream of the isolation valves. During the initial pre-operation tests of the spray system, the flow bypass through the spray additive tank will be checked. Subsequent system tests are made with the spray additive tank bypass valves closed.

b. System Testing

The functional test of the Safety Injection System demonstrate proper transfer to the emergency diesel generator power source in the event of a loss of power. A test signal simulating the containment spray signal will be used to demonstrate operation of the spray system up to the isolation valves on the pump discharge using the test pumps. The isolation valves will be blocked closed for the test. These isolation valves will be checked separately.

INITIAL TESTING IN THE OPERATING REACTOR

Initial Criticality

Initial criticality will be established by withdrawing the shutdown and control banks of RCC units from the core, leaving the last-withdrawn control bank inserted far enough to provide effective control when criticality is achieved, and then slowly and continuously diluting the heavily borated reactor coolant until the chain reaction is self-sustaining.

Successive stages of RCC bank withdrawal and of boron concentration reduction will be monitored by observing change in neutron count rate as indicated by the regular plant source range nuclear instrumentation as functions of RCC bank position and, subsequently, of primary water addition to the reactor coolant system during dilution.

Primary safety reliance will be based on inverse count rate ratio monitoring as an indication of the nearness and rate of approach of criticality of the core during RCC bank withdrawal and during reactor coolant boron dilution. The rate of approach toward cirticality will be reduced as the reactor approaches extrapolated criticality to ensure that effective control is maintained at all times.

Relevant procedures specify alignment of fluid systems to allow controlled start and stop and adjustment of the rate of which the approach to criticality may proceed, indicate values of core conditions under which criticality is expected and identify chains of responsibility and authority during reactor operations.

Zero Power Testing

Upon establishment of criticality a prescribed program of reactor physics measurements will be undertaken to verify that the basic statics and kinetic characteristics of the core are as expected and that the values of kinetic coefficients assumed in the safeguards analysis are indeed conservative.

Measurements made at zero power and primarily at or near operating temperature and pressure include verification of calculated values of RCC group and unit worths, of isothermal temperature coefficient under various core conditions, of differential boron concentration worth and of critical boron concentrations as function of RCC control group configuration. Preliminary checks on relative power distribution are made in normal and abnormal RCC unit configurations.

Concurrent tests will be conducted on the plant instrumentation including the source and intermediate range nuclear channels. RCC unit operation and the behavior of the associated control and indicating circuits will be demonstrated and the adequacy of the control and protection systems will be verified under zero power operating conditions.

Detailed procedures specify the sequence of tests and measurements to be conducted and the conditions under which each is to be performed to ensure the relevancy and consistency of the results obtained. These tests will cover a series of prescribed control rod configurations with intervening measurements of differential control rod worths and boron worth during boron dilution or boron injection. As the successive configurations are established, the measurement techniques to be used will be:

1. Dynamic Temperature Coefficient Measurement - Differential moderator coefficient measurement will be made by continuously increasing or decreasing the moderator average temperature and observing the resultant change in core reactivity.

2. Dynamic Pressure Coefficient Measurement - Differential moderator pressure coefficient measurements will be made by continuously increasing or decreasing the moderator pressure and observing the resultant change in core reactivity.

3. Control Rod Worth Measurements by Rod Drop at Zero Power - Integral control rod worth measurements will be made at zero power by dropping one or more control rods from a just critical configuration and determining the resultant change in core reactivity by observing the transient flux level response to the negative reactivity insertion.

4. Dynamic Control Rod Worth Measurements - Control rod differential worth measurements will be made by momotonically withdrawing or inserting selected control rods or groups of rods and part length rods and observing the resultant change in core reactivity.

5. Dynamic Boron Worth Measurements - Differential boron worth measurements will be made by monotonically increasing or decreasing main coolant boron concentration and observing the resultant change in core reactivity.

Power Level Escalation

In order to ensure that operation of the core is as expected in all respects, and that achievement of rated power is under carefully controlled conditions, a Power Escalation Test Program will be established to carry the plant from completion of zero power physics testing through full power operation. The Power Escalation Test Program provides for stepwise achievement of full power, with carefull review of significant core parameters at each step, to ensure that fuel and control rod mechanical performance, flux distribution, temperature distribution hot channel factors and reactivity control worths are acceptable, before additional escalation is undertaken.

The Power Escalation Test Program will provide for measurements to be made at convenient power levels in the vicinity of minimum self sustaining power, discrete levels approaching 100%, and at rated power. In each case, progress to higher levels is contingent upon acceptable core performance.

The following tests are to be conducted during the Power Escalation Test Program.

Electrical Trip Testing

Electrical tripping relays that are initiated by plant on-power malfunctions will be retested and the consequent trip sequence rechecked under operating conditions for correct operation and sequence.

Turbine Trip Testing

The turbine protection system will be checked to confirm that the appropriate initiation will either trip the turbine through the main trip solenoid or will mechanically trip the turbine. As the various set-points or status conditions are reached, the trip or runback functions will be verified.

Elevated Power Reactivity Coefficient Evaluation

During the approach to full power and during initial operation at power a sequence of reactor physics measurements will be carried out to experimentally determine power and temperature coefficients and power defects at various power levels, differential (full and part length) control rod worth and boron worths during boron dilutions, and xenon worth during initial operation. Measurements techniques are:

1. Dynamic Differential Power Coefficient - Differential power coefficient measurements are to be made at elevated power over a limited range in power level by initiating a small power level change. The change in core reactivity associated with the compensating control rod motion, is to be related to the net change in power level.

2. Dynamic Power Defect Measurements - The change in reactivity defect associated with a relatively large change in power level is to be measured by adjusting control rod positions during a ramp change in power level to maintain moderator average temperature at the prescribed value and by observing the compensating change in core reactivity due to control rod movement as indicated by the reactivity computer.

3. Dynamic Control Rod Worth Measurements - Control rod differential worth measurements are to be made at elevated power and by initiating a transient change in boron concentration in the coolant by adjusting control rod position during the transient to maintain moderator average temperature and power level essentially constant, and by observing the compensating change in core reactivity due to control rod movement as indicated by the reactivity computer.

4. Dynamic Boron Worth Measurements - Differential boron worth measurements are to be made at elevated power by monotonically increasing or decreasing main coolant boron concentration. Compensation for the reactivity effect or the boron concentration change will be made by withdrawing or inserting, respectively, control rods to maintain moderator average temperature and power level constant and observing the resultant accumulated change in core reactivity corresponding to successive rod motion steps.

5. Dynamic Xenon Transient Worth Measurements - Integral xenon worth transient measurements are to be made at elevated power, after a change in power level, by adjusting control rod position to maintain moderator average temperature and power level constant during the reactivity transient associated with the transient change in effective xenon concentration and observing the resultant accumulated change in core reactivity corresponding to successive compensating rod motion steps.

6. Elevated Power Transient Response Evaluation - As the power level is increased during the initial power escalation, a series of transient response measurements will be made to determine plant response to load changes. The test technique in each case will consist of establishing the transient change in plant conditions and closely monitoring system response during and after the transient period. The responses of system components are measured for 10% loss of load and recovery, loss of load with steam dump,turbine trip, loss of reactor coolant flow and trip of a single RCC units, reactor coolant coastdown is also measured.

7. Elevated Power Determination of Power Distribution - At successive power levels and in prescribed control rod configurations (full and part-length), measurements of flux and power distributions within the core will be made and nuclear hot channel factors will be evaluated. Use will be made of the miniature in-core flux detector system, and of the in-core temperature sensors, to determine the nuclear power and thermal and hydraulic conditions within the core. Ex-core nuclear instrumentation will be calibrated to indicate actual in-core axial power distribution.

8. Determination of Primary Coolant Flow Rate - Secondary heat balances will be made to insure that the several indications of plant power level are consistent and to provide a basis for calibration of power range nuclear channels and for determination of primary coolant flow.

9. Verification of Remote Control Stations - After the plant has been certified to operate at elevated power levels,the capability for manually taking the plant to hot shutdown from stations remote from the control room will be verified. This test will demonstrate that controls and information available in the local control stations are functioning properly and are sufficient to permit the operators to trip the plant, control heat removal, and borate in an orderly manner to reach and maintain th reactor in a hot shutdown status should the control room ever become uninhabitable.

b. Post Operational and In-Service Performance Testing

Post operational and in-service testing will be performed to the extent necessary to prove with reasonable frequency the functionability and reliability of all systems, the functionability and reliability of which might be required to limit the consequences of a potential accident as analyzed in the Safety Analysis Report, or which is needed to assure the availability, protection, or industrial safety of the facility. These tests will employ the most recent accepted practices for such testing at the time of their adoption.

VIII Emergency Planning

The planning for in-plant emergencies will closely parallel the planning employed for Unit No. 1 and described in the Emergency Procedure section of Exhibit O, submitted in support of Consolidated Edison's application for a Provisional Operating License for Unit No. 1. Generally, it will consider all credible accidents and serious occurrences for which rapid operator response is required, and will describe to the fullest extent practical what actions, notifications and cooperative efforts will be required. Accidents involving off-site considerations, as, for example, an uncontrolled release of radioactivity from the facility, will require, as with Unit No. 1 today, the close cooperation and assistance of a number of off-site authorities and agencies.

The Indian Point Station current emergency procedure to be followed in the event of an emergency situation is included below. The procedure is applicable to Unit #3.

Meetings have been held with all of the indicated public agencies, and the cooperative efforts on the part of those agencies which would be required have been discussed with them. The first such meeting was held on September 27, 1961, approximately one year before the startup of Indian Point Unit No. 1. At that time, the initial draft of the procedure was reviewed and subsequently submitted to all interested parties. As revisions to the procedure have been made, all involved organizations have received those revisions.

Internal practice exercises are performed routinely in Unit #1, including drills for the removal of injured personnel from the radiation area, evacuation of personnel from the reactor containment vessel in the event of a fuel handling accident,and the weekly communication by radio contact between the licensed reactor operator at the facility and the system operator such as might be required by the provisions of this procedure. However, practice drills involving outside agencies have not been performed. Periodic contact with these outside agencies for the purpose of maintaining current names and telephone numbers of persons to be reached, is of course, made.

Procedure in the Event of a Radiation Incident where Radioactive Material is Released Through the Nuclear Facilities Ventilation System Discharge Line to the Stack

In the event of a radiation incident at the Indian Point Station, it is believed that Iodine 131 will constitute the most critical radioactive material. The release of Iodine 131 is considered to be the most credible radioactive material having the lowest value of maximum permissible concentration to which personnel outside of the controlled area might be exposed in the event that primary water flashed into steam inside the containment. Any other releases of radioactive materials to the atmosphere which might be generated from other than the primary water system will be identified and released in accordance with Section 4.2.9, Waste Disposal, in Appendix A to Operating License DPR-5. Any release in excess of these stated values will be reported to Region 1, Division of Compliance of the Atomic Energy Commission in accordance with regulations "Standards for Protection Against Radiation" (10CFR Part 20) in the same manner as described for a release of Iodine 131.

Upon the receipt of an alarm indicating an accidental release of radioactive material through the nuclear facilities ventilation system discharge line to the stack, the Licensed Reactor Operator in the central control room will first act to verify that proper isolation is still in effect or has occurred, then

a. Contact the shift chemist and health physics tester on duty to obtain data as to the isotopic composition of the release with specific emphasis on I-131 and verification of release rate.

b. Notify the General Watch Foreman of the findings.

The General Watch Foreman will notify the General Superintendent, or his Deputy, of the incident at the magnitude of the release rate.

The Health Physics Tester will notify the Health Physicist and the Radiation Safety Officer of the incident, and the magnitude of the release rate.

The Health Physicist will conduct an area survey downwind of the stack and will report his findings to the General Superintendent. The survey will include vegetation and air analysis for I-131 and external radiation measurements. Observation will be made to a distance of 5 miles from the plant in the downwind quandrant.

1. <u>If the accidental discharge of I-131 to the atmosphere exceeds 2.4 micro-curies per second but is less than 24 micro-curies per second</u> the General Superintendent, or his Deputy will notify the Emergency Foreman No. 9 at 708 First Avenue, 212-576-3052, (a post which is manned continuously 24 hours per day every day of the year) to advise the N.Y. State Health Department and Region 1, Division of Compliance of the Atomic Energy Commission, that a release has occurred, that no emergency assistance is required, and that they will be kept advised of our findings.

2. <u>If the accidental discharge of I-131 to the atmosphere exceeds 24 micro-curies per second</u> the General Superintendent, or his Deputy will request the Emergency Foreman No. 9 to ask for assistance from the AEC N.Y. Operations Office Radiological Emergency Assistance Team. The Emergency Foreman No. 9 will also be instructed to notify the N.Y. State Health Department that a possible emergency exists and that a technical advisor is requested at the station.

   After a review of area survey data, elapsed time of release and known condition inside of reactor containment, the General Superintendent or his Deputy may request Emergency Foreman No. 9 to notify the N.Y. State Police and the U.S. Coast Guard that a release of radioactivity has occurred so that they may be prepared to administer whatever precautionary action may be required.

<u>Addresses and Telephone Numbers</u>

The Appendix to this Procedure includes the title or name of the contact representative of each public agency, their addresses and telephone numbers; also a list of the Con Edison representatives to be notified regarding <u>any</u> radiological incident at the Indian Point Station.

Notes: 1 - The Emergency Foreman will be the Company's contact agent between the Indian Point Staff and all outside agencies. He shall transmit to and receive from the public authority all reports and requests for information regarding an incident.

2 - Telephone notifications to Region 1, Division of Compliance of the Atomic Energy Commission will be confirmed by telegram.

<u>APPENDIX</u>

<u>Public Agencies to be Notified</u>

<u>State of New York</u> - Department of Health Bureau of Radiological Health Service, 84 Holland Avenue, Albany, New York. Phone numbers of individuals are kept current as a part of this procedure.

<u>Atomic Energy Commission</u> - New York Operations Office
376 Hudson Street, New York, N. Y. - Telephone 989-1000
To report an incident - Ask for Division of Compliance, Extension 381 or 382
(After office hours - The local compliance officer's home phone is kept on file)
To request radiological assistance - Ask for Radiation Duty Officer

New York State Police - Troop K - Hawthorne, N. Y.
Telephone: Area Code 914 769-2600

U.S. Coast Guard Captain of the Port of New York, Governors Island, New York
8:30 a.m. - 5:00 p.m. Call 264-8753 (Dangerous Cargo Officer)
5:00 p.m. - 8:30 a.m. Call 264-8770 (Operations Duty Officer)

COMPANY REPRESENTATIVES TO BE NOTIFIED - (Home phones are kept on file)

1 - Manager - System Operation, or
Assistant Vice President - Operations

2 - Manager - Production, or
Vice President - Production

3 - Chief Mechanical Engineer, or
Vice President - Engineering

4 - Medical - Executive Director

5 - General Attorney

6 - System Representative on duty

7 - Director of Public Information, or
General Director - Public Relations

IX Medical Procedures

One important area of consideration not discussed in the above procedure is the matter of medical procedures to be followed in the event of an in-plant medical emergency. An arrangement for close cooperative effort has been developed with members of the staffs of the New York University and Peekskill Hospitals to provide for the necessary medical assistance, either on-site or off-site, should such assistance be required. Frequent disucssions are held with appropriate representatives of those staffs, and visits have been made, both to those facilities by Consolidated Edison personnel, and by members of those organizations to the Indian Point facility. Initial contact for medical assistance is made through the Medical Director or the Company physician on 24-hour call duty. First aid is administered by station personnel trained in first aid procedures and precautions as related to radiation incidents.

Medical Facilities

Indian Point Unit No. 3 will have a regular first aid station in the nuclear service building to render routine emergency care at the site. In the nuclear service building adjacent to the containment sphere of Unit 1, but outside the external concrete biological shield, there are located special Medical Department facilities. Each of two rooms covers an area of approximately 330 square feet. The Decontamination Room is specially designed for the care of radioactively contaminated personnel. It is planned to permit the admission of several mixed ambulatory and non-ambulatory pantients. The floor is of stainless steel and the walls and ceiling are of special vinyl tile material. In the center of the Decontamination Room there is a fixed pedestal-type stainless steel table (autopsy table) which has been adapted for decontamination and treatment of non-ambulatory patients. This fixture offers the additional advantage of allowing copious irrigation in cleansing wounds without the spread of radioactive contaminants to other part of the room. Other stainless steel fixtures in this room for use by ambulatory patients include a double-bowl sink at waist level for treatment of hands and arms and another double-bowl sink at knee level for treatment of feet and legs. A shower area is situated at one end of the room to provide a continuity of flow from the decontamination area to the dispensary area. Radiation monitoring to determine possible residual activity will be performed in the vestibule area at the shower exit. A clean clothing cabinet is located immediately at the entrance of the dispensary area from the monitoring vestibule.

The Dispensary Area consists of a conventional examining room which can be used for first aid after decontamination of ambulatory and litter cases. It can also be utilized for general clinic services. Medical and first aid supplies are kept in the Dispensary and are passed through as they are needed into the Decontamination Room. All liquid waste from the fixtures in the Decontamination Room drain into a special retention tank where they are monitored before any further disposal.

Emergency facilities are available in the Decontamination Room in planning for the care of injured radiating personnel. The probability of such an occurrence is extremely remote but the possibility does exist of radioactive material embedded in a localized area of the body, or even generally distributed. Such cases might not be readily movable to an outside medical facility. It is planned that a surgical team could be brought in to perform emergency procedures and removal of impregnated radioactive materials.

Plans for any major radiation accident have been completed at University Hospital in New York City some 40 miles away. Despite the distance involved, the capability of the radiation team at this medical center yields very tangible benefits in treatment. Arrangements have also been completed with the local community hospital for a possible minor incident.

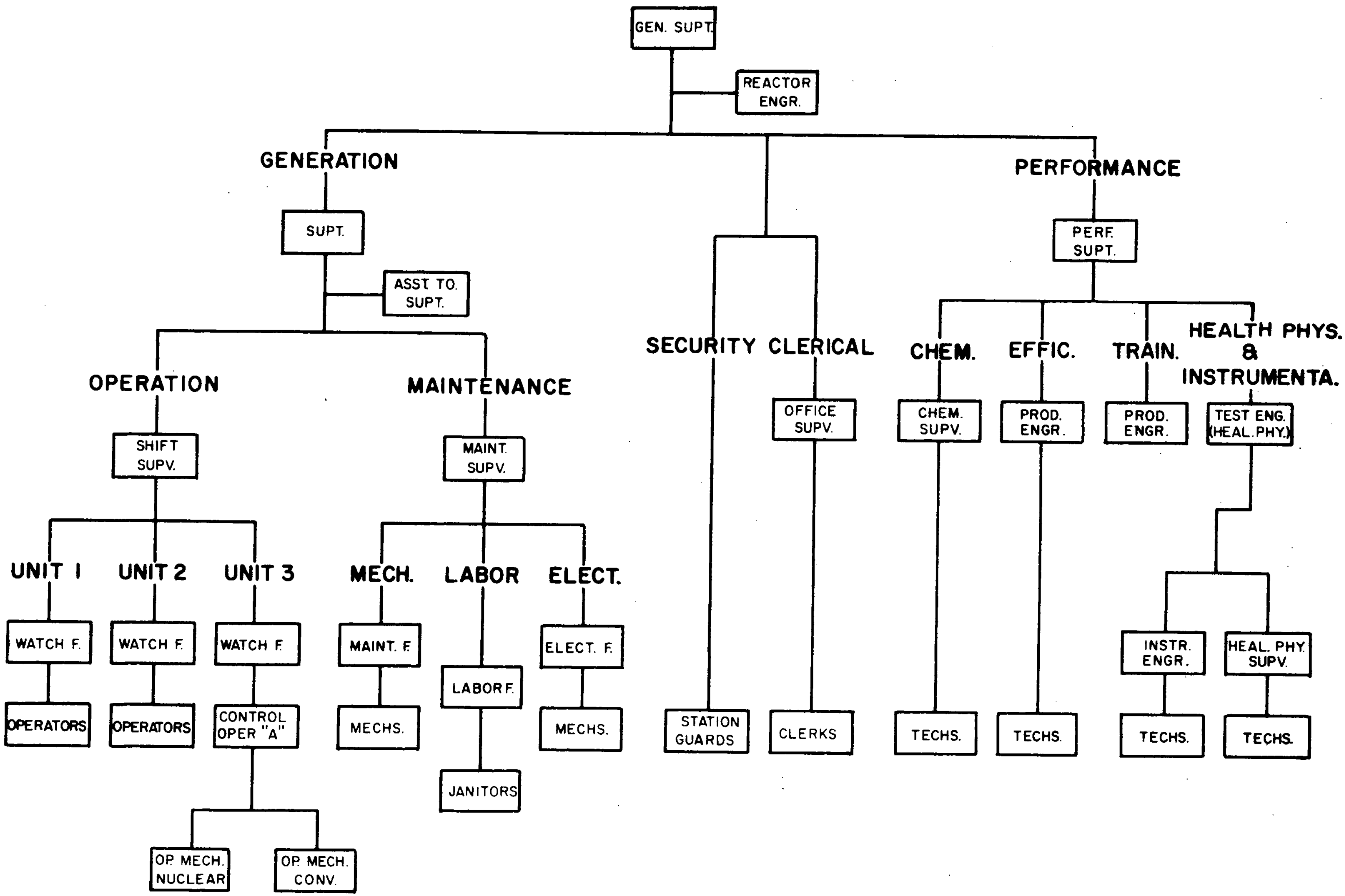
The hospital would be notified before a contaminated patient or patients are sent for admission so that bringing them into the hospital can be planned to avoid spreading high levels of radioactivity along the route. A hospital radiation team will be organized at that time with a radiological safety officer who has had experience either in the department of radiology or the radioisotope laboratory.

Incident Involving Unit Other Than No. 3

Generally, the design of each of the three units is intended to permit, from the standpoint of operating personnel radiation safety, the operation of any and all units subsequent to an accident on any adjacent unit. From an electric service reliability standpoint this design criterion is mandatory, since the three unit Indian Point complex will ultimately represent nearly two and one half million kilowatts of generating capability, and to lose and be forced to evacuate the entire site following an accident on one unit would lead to serious consequences, indeed, from the standpoint of electric service continuity. In the event of such an accident, however, extra measures would be taken in the form of additional area monitoring and personnel dosimetry to ensure that plant personnel are not being subject to excessive radiation exposures. All personnel not required for continued operation of the station including visitors at the exhibition building would leave the area. Actual experience with on-site construction workers shows that approximately 400 people leave the site within 15 minutes at the end of a working day. It is expected that no more than 400 people, assuming a peak visitors day, would have to leave the site during an emergency.

PROPOSED ORGANIZATION CHART FOR STAFFING OF THREE UNITS AT INDIAN POINT

(OPERATING ORGANIZATION FOR UNIT No. 3 ONLY SHOWN)



QUALITY ASSURANCE PROGRAM

The following information is provided to supplement that contained in Item 5 of Supplement I.

PART I

Figure A is a detailed Quality Control Relationship Chart for Indian Point Unit No. 3. All associated organizational levels of the principle parties involved, namely Consolidated Edison, Westinghouse and United Engineers and Constructors together with the U.S. Testing Company who are acting as a Consultant to Consolidated Edison are shown:

The chart depicts:

(a) Lines of Direct Responsibility.

(b) Lines of Internal Organization Co-ordination and Communication.

(c) Lines of External Organization Co-ordination and Communication.

(d) Lines of Audit Responsibility.

(e) Lines of Surveillance.

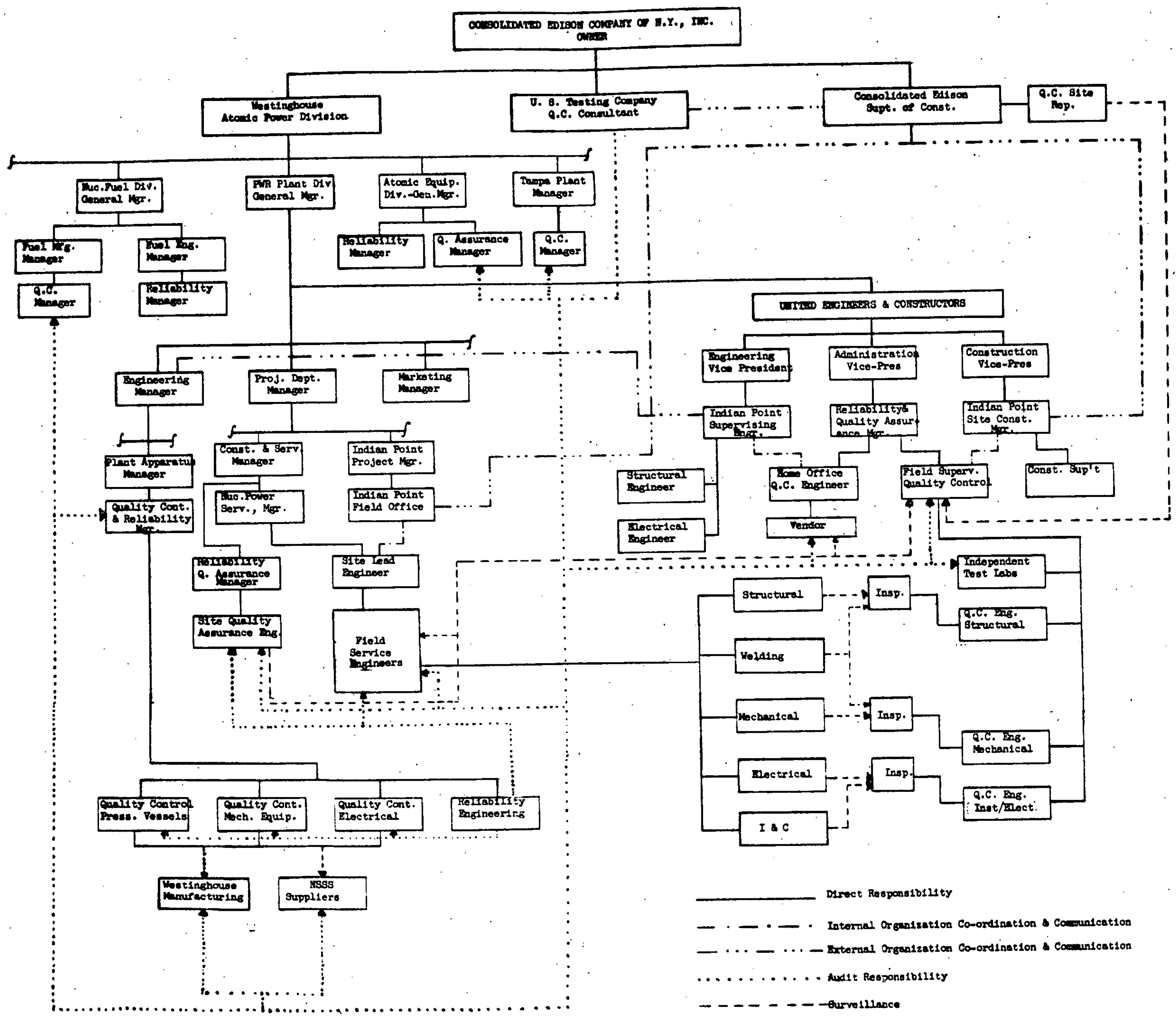


FIGURE A - INDIAN POINT NUCLEAR GENERATING UNIT NO. 3
Q. C. RELATIONSHIPS
CONSOLIDATED EDISON CO. OF NEW YORK, INC.

PART II

In the capacity of Prime Contractor, Westinghouse is responsible for the provisions of all material and equipment and for all construction. In discharging this responsibility, Westinghouse recognizes the importance of Quality Assurance throughout all stages of design, fabrication and construction, and accordingly maintains a comprehensive overall Quality Control Program. WAPD may be divided into a number of functional groups, each of which has both direct and indirect responsibility for certain aspects of the overall design, fabrication and construction phases of the project. Close association and interchange of information at all levels exists between the respective functional groups, including those associated with both the Applicant and all subcontractors. In addition, specific channels exist to ensure the correct interchange of information between all groups in relation to their respective scopes and associated responsibilities.

Table I illustrates the interrelationships between the official information exchanger channels (or flow paths) and the functional groups. For example, contractural requirements originate in the Project Group, and they are officially distributed to the Plant Safeguards and Licensing Group, the System Functional Requirement Groups, the System Design Groups and the Equipment Design and Procurement Groups. It can be seen that all aspects of the project are carefully considered at each stage in the overall program, with the respective "lead" functional group co-ordinating the overall efforts of the associated functional groups; the "lead" functional group being the group which has the direct responsibility for the associated aspect of the project which is under consideration.

To illustrate schematically the extent of the interrelationships between respective functional groups, Figure B has been produced.

Although the interrelationships described above are only implemented in the case of Westinghouse supplied equipment, similar interrelationships are established for the equipment, comprising a balance of the plant, which is supplied by the Architect Engineer. All such procedures reflect the fact that Westinghouse, as the Prime Contractor, has the overall responsibility for the whole plant.

Through the implementation of these procedures in the manner described throughout the whole of the design, fabrication and construction phases of the project, Westinghouse is able to meet the required levels of safety, operability, maintainability, and reliability in the finished plant.

TABLE I

WAPD FUNCTIONAL GROUPS QUALITY ASSURANCE FLOW CHART

Flow Path No.	Flow Path Definition	Origin	PROJECT A	PLANT SAFEGUARDS AND LICENSING B	SYSTEM FUNCTIONAL REQUIREMENT GROUPS C	SYSTEM DESIGN GROUPS D	EQUIPMENT DESIGN AND PROCUREMENT GROUPS E	QUALITY CONTROL AND RELIABILITY GROUPS F	MATERIALS ENGINEER GROUPS G	ARCHITECT ENGINEER H	EQUIPMENT SUPPLIERS I	CONSTRUCTORS J	ON-SITE SERVICE AND INSPECTION GROUPS K	APPLICANT L
0	Design and Construction Follow and Approval (to and from all groups)	A	A*	B*	C*	D*	E*	F*	G*	H*	I*	J*	K*	L*
1	Contractual Requirements	A		B	C	D	E							
2	Safety Requirements	B				D	E							
3	Functional Requirements	C				D								
4	System Design	D		B	C									
5	Concurrence on Design	B				D		F						
6(a)	Concurrence on System Design	C				D								
6(b)	Design Review	F			C	D	E		G				K	
7	Equipment Functional Requirements	D					E							
8	System Functional Requirements	D								H			K	
9	Quality Control Plan	F					E							
10	Equipment Specification (including Quality Control Requirements)	E		B		D		F	G					
11	Concurrence on Equipment Specifications	BDFG					E							
12	Final Equipment Specifications	E				D		F			I		K	
13	Design, Manufacturing and Inspection Procedures	(a) I					E							
		(b) E						F	G					
14	Concurrence on Design, Manufacturing and Inspection Procedures	(a) FG					E							
		(b) E									I			
15(a)	Surveillance, Process Audits and Quality Control Plan Actions	F									I			
15(b)	Corrective Action, Follow through on Deficiencies	F											K	

TABLE I (Continued)

Flow Path No.	Flow Path Definition	Origin	PROJECT A	PLANT SAFEGUARDS AND LICENSING B	SYSTEM FUNCTIONAL REQUIREMENT GROUPS C	SYSTEM DESIGN GROUPS D	EQUIPMENT DESIGN AND PROCUREMENT GROUPS E	QUALITY CONTROL AND RELIABILITY GROUPS F	MATERIALS ENGINEER GROUPS G	ARCHITECT ENGINEER H	EQUIPMENT SUPPLIERS I	CONSTRUCTORS J	ON-SITE SERVICE AND INSPECTION GROUPS K	APPLICANT L
16	Materials Evaluation	(a) I					E							
		(b) E							G					
17	Materials Concurrence	(a) G					E							
		(b) E									I			
18	System Layout	H		B		D	E							
19	Concurrence on System Layout	BDE								H				
20	Final System Layout and Field Follow	H										J		
21	Inspection and Erection, Requirements and Procedures	J					E	F	G				K	
22	Concurrence on Inspection and Erection, Requirements and Procedures	EFKG										J		
23	Equipment for Receipt Inspection	I											K	
24	Final Equipment Inspection and Erection, Requirements and Procedures	K										J		
25	Erection Follow	K										J		
26	System Installed	J											K	
27	Testing Requirements and Procedures	D		B	C		E							
28	Concurrence of Testing Requirements and Procedures	BCE				D								
29	Final Testing Requirements and Procedures	D											K	
30	System Performance Test Results	K		B	C	D								
31	Concurrence on System Performance	BCD											K	
32	Plant	K												L
33	Operating and Emergency Instructions	D		B	C		E							

TABLE I (Continued)

Flow Path No.	Flow Path Definition	Origin	PROJECT A	PLANT SAFEGUARDS AND LICENSING B	SYSTEM FUNCTIONAL REQUIREMENT GROUPS C	SYSTEM DESIGN GROUPS D	EQUIPMENT DESIGN AND PROCUREMENT GROUPS E	QUALITY CONTROL AND RELIABILITY GROUPS F	MATERIALS ENGINEER GROUPS G	ARCHITECT ENGINEER H	EQUIPMENT SUPPLIERS I	CONSTRUCTORS J	ON-SITE SERVICE AND INSPECTION GROUPS K	APPLICANT L
34	Concurrence on Operating and Emergency Instructions	BCE				D								
35	Final Operating and Emergency Instructions	D											K	
36	Technical Specifications	B			C	D	E		G				K	
37	Concurrence on Technical Specifications	CDEGK		B										
38	Final Technical Specifications	B								H				L
39	Plant Operation Information	L											K	
40	Plant Operation Information (to all Appropriate Westinghouse Groups)	K	A*	B*		D*	E*		G*				K*	
41	Audit of Site Quality Assurance	F											K	
42	Analysis of Plant Operating and Maintenance Information	F					E		G		I			

* Not shown on Figure B

WAPD FUNCTIONAL GROUPS
QUALITY ASSURANCE SCHEMATIC FLOW DIAGRAM

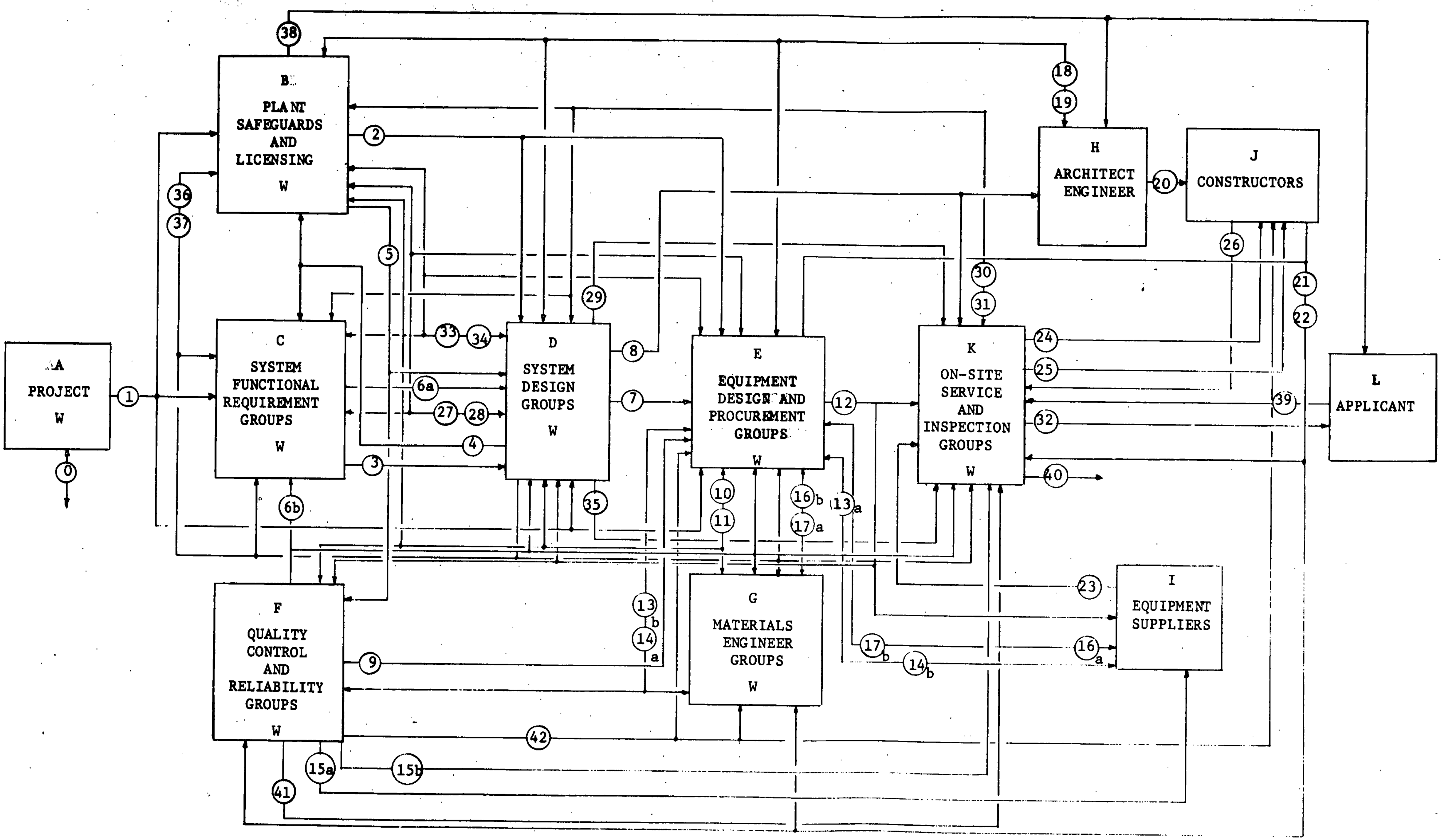


FIGURE B
Supplement 5

CODES AND STANDARDS FOR INSTRUMENTATION AND CONTROL EQUIPMENT

The Westinghouse Atomic Power Division's comprehensive equipment specifications and instrumentation and control standards individually establish required detail design and quality control requirements for the mechanical and electrical design material procurement, fabrication, testing, inspection, cleaning, packaging, and shipment. The standards and specifications cover equipment such as:

- Electrical instruments
- Control relays
- Control and selector switches
- Indicator lights
- Instrument and control power distribution panel board(s)
- Mounting, wiring, and piping of subcontractor equipment

These Westinghouse standards generally incorporate national standards such as:

- NEMA Standard, Pub. No. SG5-1959 - Power Switchgear Assemblies
- NEMA Standard, Pub. No. IC1-1965 - Industrial Control
- ASA Standard, Pub. No. C1-1963 - National Electric Code
- IPCEA Standard, Pub. No. S-61-402 - Thermoplastic-insulated Wire and Cable
- ISA Recommended Practices,

and supplement these standards where required.

PART III

WESTINGHOUSE
DISCREPANT AND DEFECTIVE MATERIAL: ACTION PROCEDURES

Westinghouse PWR-Plant Division has written procedures for identifying, reporting, and making disposition of material and equipment found discrepant or defective in the manufacturing, shipping, and erection stages.

A. Manufacturing

As explained in the details of the overall quality control system, each supplier, under the conditions of the purchase order requirements, has primary responsibility for the quality control and in-process control of the materials and components as they are processed through his shop. This responsibility includes identifying, segregating, reporting on, and awaiting disposition of materials and components judged to be discrepant.

Westinghouse PWR-PD Quality Control engineers and representatives survey the suppliers' quality control and process control functions to ensure the supplier's conformance to the quality control requirements specified in the purchase orders. Normally nothing but material and components which have been accepted by the supplier's quality control system, as conforming to the purchase order requirements, are submitted to Westinghouse Quality Control representatives for their final acceptance.

Most discrepant or non-conforming material found in the supplier's shop by the quality control system is rejected and segregated for scrap. However, in some cases, certain non-conforming material, either identified by the supplier or Westinghouse Quality Control representatives, is submitted to the PWR-PD for consideration. The method of submittal is by way of a formal written procedure.

Westinghouse Quality Control in the form of Deviation Notice Disposition Requests (DNDR) submits all discrepancies of all materials and parts uncovered in all phases of their surveillance to PWR-PD engineering for consideration and disposition. In this DNDR form, a Quality Control representative of Westinghouse identifies and describes the non-conforming characteristics on a DNDR form. The purchase order specification limitation on the particular characteristic is also indicated on the form. The discrepancy is considered by a group including a design engineer, the Quality Control engineer on the specific equipment, and the cognizant materials and process engineers if and when a discrepancy concerns materials or processes which are non-conforming.

Consideration is given to restoring the equipment or material under question to a conforming condition wherever possible, or scrapping the material and starting over. After due consideration one of three following dispositions is made:

1. Scrap - This disposition is made when the material cannot be repaired or used "as is".

2. Repair - This disposition is made when there is no compromise in the design and the method of repair is approved by PWR-PD.

3. Accept - This disposition is made when, after due consideration is made by the design engineers concerning the function of the equipment, it is determined that the use of the equipment "as is" will not adversely affect the performance of equipment.

Upon the completion of the review the disposition is recorded on the DNDR form and PWR-PD notifies the supplier as to the disposition given.

Particular attention is paid in the purchase order requirements that all non-conforming material shall be kept identified, segregated, and recorded in the supplier's shop to prevent its intermingling with conforming material. The supplier is required to take necessary actions to preclude its further use prior to disposition. Particular attention is given by the Westinghouse Quality Control engineers and representatives to the supplier's procedures formal handling of non-conforming material in his shop during the entire surveillance cycle.

B. Plant Site

The PWR-PD field quality assurance personnel have a field system for identifying, reporting and obtaining disposition of non-conforming, discrepant or defective material equipment or practices discovered at the plant site. The report is called a Field Deficiency Report (FDR). The Field Quality Assurance people fill out the FDR report with all the pertinent information to provide the cognizant group usually PWR-PD engineering with the vital information necessary to resolve the problem and make proper and timely disposition on the matter in question. After the cognizant personnel make disposition, the disposition is properly noted on the FDR form and communicated back to the field. FDR files are maintained to record all field deficiencies and to provide for long-term corrective action. The field must discontinue work on the non-conforming equipment until disposition is made.

UNACCEPTABLE CONDITIONS (WORK STOPPAGE)

A. Any item which in any way does not meet the Quality Control requirements of the specifications shall be noted as unacceptable and appropriately tagged by the UE&C Field Quality Control Group. The non-conforming conditions are identified in a Q.C. Report and requesting corrective action, and distributed to the UE&C Construction Superintendent, cognizant Departmental Supervisor, Purchasing Agent, Field Accounting, with a copy to the Project Engineer and the Manager of Reliability and Quality Assurance in the Home Office.

If the Construction Superintendent deems it advisable to use the affected item in the deficient condition, he shall have a request prepared in writing stating:

1. The description of the item and the deficiency.

2. End use of the item.

3. Reason for requesting the waiver of the specification requirement.

The request shall be distributed to the UE&C Construction Manager and/or the Project Engineer for review and disposition and the Westinghouse Field Office for concurrence with a copy to the UE&C Field Supervisor - Quality Control for follow-up action.

1. If the "Waver Request" is approved, the Field Supervisor - Quality Control shall remove the "Hold Tag" and the item shall be released for use.

2. If the "Waiver Request" is denied, the item shall be replaced or repaired as required and normal quality control procedures will be implemented regarding repair or replacement.

B. If a condition arises wherein the Field Supervisor - Quality Control determines that project work must stop in order to preserve the quality of the project, he shall so inform the Construction Manager. In the event the Construction Manager, from the total project standpoint, does not agree with the recommendation of the Field Supervisor, - Quality Control, and decides to continue the work, the Field Supervisor - Quality Control then will report the matter and his recommendations to the Manager of Reliability and Quality Assurance in the Home Office for resolution action with UE&C Management.

INDIAN POINT UNIT NO. 3

CON EDISON SURVEILLANCE PLAN

As part of the Con Edison Quality Assurance Program for Indian Point Unit No. 3, a surveillance plan has been prepared. This plan consists of a program to monitor the quality control and other quality related items on a spot check, and in some instances, full review basis. A table giving the general scope of this plan is attached.

The direct administration of this plan is done by two separate organizations of Con Edison. On site the direct administration of the plan is performed by the Construction Representative of Con Edison's Major Projects-Inside Plant Bureau of the Construction Department and his staff. Off site the direct administration of this plan is performed by the Nuclear Division Engineer of the Mechanical Plant Bureau of Con Edison's Mechanical Engineering Department and his staff. To assist these two organizations, the Company has engaged the United States Testing Company. Technical and specialized competence are available to these two organizations by the members of both their staffs and by the engineers of the Mechanical Plant Bureau and Structural Bureau of the Mechanical Engineering Department and the engineers of divisions of the Inside Plant Bureau of the Electrical Engineering Department. These two organizations can also call upon specialized assistance from other departments of Con Edison and Con Edison's consultants.

All reports by U. S. Testing are transmitted to both these organizations and are circulated to higher levels of management of their respective departments. They are also circulated for review to other divisions of Con Edison's engineering departments as discussed above when the content of the report applies to the particular division's areas of special competence. Reports of deficiencies in quality assurance or quality control by either of these two organizations or by other engineering divisions of Con Edison or by Con Edison's contractors are circulated in the same manner.

The Nuclear Division Engineer or delegated members of his staff may identify to Westinghouse materials or components being manufactured or fabricated off site that do not conform to specified requirements. He may also advise the Construction Representative and the Westinghouse Crop. that work performed on site should be rejected. The Construction Representative may stop or reject on site work. He may also advise the Nuclear Division Engineer and the Westinghouse Corp. that materials and components manufactured or fabricated off site should be rejected. Should there be any disagreement between the above parties regarding rejection of some portion of the job, it will be resolved on the basis of a technical evaluation which demonstrates that the reasons for the rejection are invalid or valid prepared by the designer and reviewed by competent persons in Con Edison. Con Edison may call upon it's consultants to aid in the review.

In the event that corrections are to be made in rejected work or deviations from specifications are to be made, the corrections or deviations will be allowed on the basis of a technical evaluation prepared by the designer and reviewed by competent persons in Con Edison or Westinghouse. Con Edison may call upon it's consultants for aid in the review. This procedure may not be followed when the corrections or deviations are allowed for in the specifications, codes or standards under which the work is being performed.

Materials and components which are damaged during shipment, handling, storage or installation may be rejected by either the Construction Representative and delegated members of his staff or the Nuclear Division Engineer and delegated members of his staff. Inspection for damage of components and materials which are included in the surveillance plan will be made. All materials and components included in the surveillance plan will be inspected for damage at least after installation. Damage inspections will usually be visual and will normally be made by the Construction Representative or member of his staff. The Construction

Representative may call upon U. S. Testing or members of the engineering departments to aid in damage inspections. Non-destructive examinations or tests may be performed on components or materials to detect damage if the Construction Representative or Nuclear Division Engineer believes such examinations or tests are indicated.

U. S. Testing Company personnel periodically witness the mixing of concrete at the batch plant located at Verplanck, New York. The weighing of the various aggregates is included in this portion of the surveillance. Also, the reports detailing the results of the batch mixture analysis are reviewed.

At the pouring site, U. S. Testing Company witnesses on a surveillance basis the general contractor pouring procedures including placement of forms. Air content of cement is monitored and slump tests which indicate concrete consistency are witnessed. The concrete compression test reports produced by the testing laboratory responsible to United Engineers are reviewed by U. S. Testing Company.

If U. S. Testing Company considers that either the batch plant or site operations are questionable, then concrete cylinder samples are taken by U. S. Testing Company and independently tested for compression and mixture composition.

Written reports of all surveillance visits and/or independent concrete tests by U. S. Testing Company are submitted to Con Edison.

Resolution of matters involving deficiencies in quality assurance are normally made at the level of the Construction Representative and the Nuclear Division Engineer or delegated members of their staffs. Reports of such resolutions are circulated to higher management levels of the Construction Department and the Mechanical Engineering Department which may reject the resolution reached or require further evaluation and justification. The assigned levels of decision making responsibility are made on the basis of a need to keep the decision making function at a low enough level to achieve involvement and knowledge of the subjects to be decided upon and high enough to achieve necessary responsibility.

The Indian Point Unit No. 3 project is a turnkey project. As such, matters involving scheduling and construction costs are under the direct management of Westinghouse. Because of this, it is possible to utilize almost all organizations and personnel of Con Edison to achieve quality assurance without any direct conflict of interest occurring.

Since the Indian Point Unit No. 3 project is a turnkey project, the engineering and design responsibility belongs to the Westinghouse Corp. Con Edison engineering departments do not prepare or approve the design drawings or specifications for the project. However, the design drawings and specifications are given to Con Edison and are reviewed by Con Edison's engineering departments. Changes may be made by Con Edison as a result of mutual agreement with Westinghouse or as a result of a change ordered by Con Edison. Westinghouse must institute changes required by Con Edison unless they can justify refusal on the basis that the change would not meet the safety requirements of the project.

The enclosed table of scope indicates areas that come under the Con Edison quality assurance surveillance plan. This scope may be expanded as experience dictates. Some portions of the surveillance plan may be changed or eliminated if it appears justified. An X is used to indicate those things that U. S. Testing is doing for Con Edison. For performance tests, Con Edison is utilizing it's own personnel.

Many of the final specifications have yet to be prepared and receive final approval. The following table is not intended to indicate that all of the listed tests will be performed. It is only intended to indicate the scope of U.S. Testing surveillance where the specifications may call for such testing.

The surveillance indicated on the enclosed table may be performed by witnessing the item or by witnessing performance of representative process at the suppliers plant, or by verification of adequate documentation.

ITEM	MILL				LIQUID PERET P. T.	ULTRASONIC U. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	PERFORMANCE	CLEANING BEF. SHIP.	RADIOGRAH R. T.
	PHYS.	CHEM.	U. T.	M. T.										
1. Internals									X				X	
a. Cage														
(1) Upper Barrel	X	X	X		X			X	X					X
(2) Upper Support Skirt (Hat Section - Forging)	X	X			X			X	X					X
(a) Upper Support Ring	X	X	X		X									
(3) Lower Barrel Top Shelf	X	X	X		X									
(4) Lower Barrel Bottom Shelf	X	X	X		X									
(5) Upper Core Plate	X	X	X					X	X					
(6) Lower Core Plate	X	X	X					X	X					
b. Baffle Plates	X	X							X					
c. Control Rod Guides														
(1) Tubes	X	X	X					X						
d. Thermal Shield Weldments and Attachments	X	X	X		X			X	X					X
e. Nozzles	X	X	X											
f. Deep Beam	X	X			X			X						
(1) Plate														
(2) Bar														
(3) Ring														
g. Upper Support Columns	X	X			X			X						
h. Diffuser Plant, Incermediate	X	X						X						
i. Lower Support Columns	X	X			X			X						
j. Core Support Casting	X	X			X			X						X

ITEM	MILL PHYS.	MILL CHEM.	MILL U. T.	MILL M. T.	LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.		
2. REACTOR PRESSURE VESSEL						X		X		X						
a. Control Rod Housing	X	X	X		X											
b. Studs (Forging)	X	X	X	X												
c. Nuts	X	X	X	X												
d. Sealing Rings and Sealing Surfaces at Flange	X	X						X								
e. Supporting Castings	X	X			X											
f. Nozzles	X	X	X	X			X									
g. Vessel Washers	X	X														
h. Bottom Instrument Tube	X	X	X													
i. Closure Head	X	X	X	X			X				UT7					
j. Shell Plates	X	X	X	X			X		X		PT					
k. Flanges (Forging)											UT7 PT					
(1) Closure Head	X	X	X	X												
(2) Vessel	X	X	X	X												
l. Head Adapters (Forging)	X	X	X		X	X										
m. Nozzle Safe Ends	X	X	X		X									X		

ITEM	MILL PHYS.	MILL CHEM.	MILL U. T.	MILL M. T.	LIQUID PENET. P. T.	ULTRASONIC H. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE			
3. CONTROL RODS MECHANISM								X	X							
a. Housing	X	X	X		X											
b. Fabrication of Parts (electrical & mechanical)																
c. Mechanism Assemblies and Housings	X	X						X	X							
d. Rod Drive Assembly																
e. Drive Shafts	X	X			X	X		X	X							

ITEM	MILL				LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.	EDDY CURRENT E. T.	VISUAL (BENDS)
	PHYS.	CHEM.	U. T.	M. T.												
4. <u>STEAM GENERATOR</u>										X		X				
a. Shell	X	X	X								7PT					
b. Channel Head (Casting)	X	X		X			X				7PT			X		
c. Steam Drum	X	X	X													
d. Tube Sheet Plate (Forging)	X	X	X	X							UT/PT					
e. Tube Supports	X	X														
f. Nozzles	X	X	X	X			X									
g. Tubes, Manufacturing	X	X	X					X							X	X
h. Tube to Tube-Sheet	X	X	X		X											

ITEM	MILL PHYS.	MILL CHEM.	MILL U. T.	MILL M. T.	LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.		
5. PRESSURIZER										X						
a. Heads																
(1) Casting	X	X					X							X		
(2) Clad					X											
b. Shell										X						
(1) Plates	X	X	X			X	X									
(2) Clad					X											
c. Heaters																
(1) Tubing	X	X			X	X										
(2) Centering of Element														X		
d. Nozzles	X	X	X	X	X	X										
e. Weldments																
(1) Shell, longitudinal							X							X		
(2) Shell, circumferential							X							X		
(3) Cladding					X											
(4) Nozzles					X		X									
(5) Nozzle Safe Ends					X											
(6) All welds after hydrotest																

ITEM	MILL PHYS.	MILL CHEM.	MILL U. T.	MILL M. T.	LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.		
6. <u>PIPING</u>																
a. Reactor Coolant	X	X	X		X	X		X								
b. Pressurizer Surge Line	X	X	X		X	X		X								
c. Residual Heat Removal Piping	X	X	X			X		X								
d. Safety Injection Piping	X	X	X			X		X								
Weldments:																
a. Longitudinal																
b. Circumferential																

ITEM	MILL				LIQUID PENET U. T.	ULTRASONIC U. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.		
	PHYS.	CHEM.	U. T.	M. T.												
7. FITTINGS All Pipe Fittings connecting the pipes of the Primary system.								X		X						
a. Reactor Coolant	X	X														
b/ Pressurizer Surge Line	X	X														
c. Residual Heat Removal Piping	X	X														
d. Safety Injection Piping	X	X														
I. Castings					X									X		
II. Forgings					X	X										

ITEM	MILL PHYS.	MILL CHEM.	MILL U. T.	MILL M. T.	LIQUID PENET. P. T.	ULTRASONIC U. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.	PRESSURE DROP	
8. <u>VALVES</u>																
a. Residual Heat Removal System	X	X			X					X		X		X		
(1) Aux. Coolant System																
(a) Air Operated	X	X			X					X		X		X		
(b) Relief	X	X			X					X				X		
(c) Check	X	X			X					X		X		X		
b. Primary System Relief	X	X			X					X		X		X		
c. Seal Water System, Isolation																
(1) Motor	X	X			X					X		X		X		
d. Chemical & Vol Control System																
(1) Diaphragm	X	X			X					X		X		X		
e. Reactor Coolant Pressurizer,	X	X			X					X		X		X		
f. Safety Injection System, Isolation																
(1) Air Operated	X	X			X					X		X		X		
(2) Relief	X	X			X					X		X	X			
(a) Pump Discharge Header																
(b) Accumulator Tank																
(3) Check	X	X			X					X		X		X		
(a) Accumulator Tank																
(4) Motor	X	X			X					X		X		X		
(5) Manual	X	X			X					X		X		X		

ITEM	MILL PHYS.	MILL CHEM.	MILL U. T.	MILL M. T.	LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.	SHAFT BALANCING	
9. <u>PUMPS</u>																
a. Residual Heat Removal										X						
(1) Casing Casting	X	X			X											
(2) Internals															X	
b. Safety Injection (High Head)	X	X			X					X					X	
c. Containment Spray	X	X			X					X					X	
d. Service Water	X	X	X							X					X	
e. Primary Coolant	X	X			X									X	X	
g. Component Cooling	X	X			X					X					X	
h. Auxiliary Feedwater	X	X								X					X	
i. Turbine Bearing Oil	X	X								X					X	

ITEM	MILL PHYS.	MILL CHEM.	MILL U.T.	MILL M.T.	LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PAR M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.
10. <u>TANKS</u>														
a. Accumulators	X	X			X					X	X			X
b. Refueling Water Storage	X	X			X					X X				X
c. Boron Injection	X	X			X									X

ITEM	MILL				LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PAR M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.	EDDY CURRENT E. T.	
	PHYS.	CHEM.	U. T.	M. T.												
11. HEAT EXCHANGERS																
a. Residual Heat										X		X				
(1) Shell	X	X					X							X		
(2) Domes	X	X												X		
(3) Tubes, Manufacturing	X	X	X													
(4) Nozzles	X	X					X									
b. Component Cooling										X		X				
(1) Shell	X	X					X							X		
(2) Domes	X	X		X												
(3) Tubes, Manufacturing	X	X	X					X							X	
(4) Nozzles	X	X												X		

ITEM	MILL				LIQUID PENE P. T.	ULTRASONIC U. T.	MAGNETIC PA M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE			
	PHYS.	CHEM.	U. T.	M. T.												
12. VENTILATION SYSTEM								X	X							
a. Recirculation Fan																
b. System Controls																
(1) Shield Building																
(2) Containment Vessel																

ITEM	MILL				LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PART M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE			
	PHYS.	CHEM.	U. T.	M. T.												
13. ELECTRIC MOTORS																
All motors inside containment which are required for engineered safeguards													X			

ITEM	MILL				LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PAR M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE			
	PHYS.	CHEM.	U. T.	M. T.												
14. <u>DIESELS AUXILIARY</u>																
a. Diesel Generator													G			
b. Switching System																

ITEM	MILL PHYS.	MILL CHEM.	MILL U. T.	MILL M. T.	LIQUID PENET P. T.	ULTRASONIC U. T.	MAGNETIC PAR M. T.	DIMENSIONAL	ALIGNMENT ASSEMBLY	HYDROSTATIC	CLADDING	LEAKAGE	PERFORMANCE	RADIOGRAPHY R. T.	PHYSICAL	VISUAL
15. CONSTRUCTION																
a. Concrete																
(1) Field																X
(2) Plant																X
(3) Sampling															X	X
b. Rebars															X	X
c. Cadwelds															X	X
d. Shields															X	X
e. Containment Liner	X	X	X									X				X
f. Hooks of Polar Crane															X	
g. Penetration Points for cables and tubes into the containment vessel.								X	X			X				X
h. Bolting & Welding - structural steel					X			X								X
i. Storage Facilities																X

HYDROGEN PRODUCTION & RECOMBINER INFORMATION

The production of hydrogen from corrosion of aluminum in the alkaline spray environment is presented at the end of this discussion. While it is not possible to account for all aluminum surfaces in the containment specifically at this stage in procurement, it is possible to make an estimate based on experience with Unit No. 2. Major sources whose aluminum contribution are known were itemized, and an allowance made for those presently unidentified. In the aggregate, a total of 262 square feet was estimated, and no limit on thickness was assumed.

The rate of alkaline corrosion assumed was 1000 $mg/dm^2/day$* on all aluminum surfaces. No credit was taken for anodizing or protective coatings. Complete and continuous immersion at the temperature of the containment was postulated although it would be expected that most of the objects considered would cease to be in contact with alkaline solution after the sprays are turned off (probably during the first few hours post accident).

Aluminum Corrosion Hydrogen Production	7 days	1000 standard ft^3
	10 days	1200 standard ft^3
	20 days	2000 standard ft^3
	40 days	4000 standard ft^3
	90 days	10,000 standard ft^3
	100 days	11,400 standard ft^3
Zr - Water Reaction (2%)	—	6600 standard ft^3

* Representative of corrosion rates measured by Westinghouse for several aluminum alloys at about 212°F in pH 9.5 solution.

DESCRIPTION OF RECOMBINER SYSTEM CONCEPT

Following a major loss of coolant accident in a PWR reactor, hydrogen may be generated inside the containment by the mechanisms of radiolysis, zirconium-water reaction, and the reaction of alkaline spray solution with aluminum. Because of the high level of radioactivity in the containment which may also result form the accident, the containment must be sealed for an extended period to prevent the spread of contamination to the environment. If hydrogen production is conservatively estimated, a hazardous concentration can be reached prior to the time the radioactive gas and hydrogen could be safely purged to the environment.

It is proposed, therfore, to provide equipment for the controlled recombination of hydrogen at a safe concentration. The system selected is a flame combustor using containment atmosphere (containing a low concentration of hydrogen) as primary oxidant and supplemental hydrogen as a fuel. The product of combustion, water vapor, is cooled and condensed from the atmosphere by the vital cooling systems of the containment. Operation of the system will control buildup of hydrogen to less than 2 v/o, or one-half of the lower flammable limit.

Catalytic recombination was considered but not adopted, because of the major problem of assuring long catalyst life in the atmosphere of the containment containing a diverse and unpredictable mixture of trace compounds, many of which could be potential catalyst poisons. A secondary problem to be solved in relation to catalytic combiners is the need for a reliable preheater to ensure low relative humidity entering the catalyst bed. The construction and licensing schedule for reactors of the current generation would not accommodate a comprehensive experimental program dealing with these probelms, and there was a considerable doubt that they could be resolved effectively in the long run. Development of a catalytic recombiner for this purpose, therefore, was set aside.

Other methods, including cryogenic devices and wet chemical (absorption-reaction) processes were considered and discarded because of complexity and prohibitive space requirements.

Basis of Design

A generation rate of hydrogen has been conservatively estimated based on the maximum credible loss of coolant accident. If allowed to proceed unchecked, this generation would result in the following approximate concentrations within the containment:

Days Post Accident	Volume percent (v/o) Hydrogen in Steam-Free Air
2.7	1.0
9.8	2.0
51	4.1 (lower flammable limit)
>100	10.0

The basis of this estimate is as follows:

1. Hydrogen (uncombined) from 2% assumed zirconium water reaction
2. Corrosion of exposed (estimated) aluminum surface at 1000 $mg/dm^2/day$
3. Radiolysis of water in core by absorbed gamma at 0.44 mol/100 ev.
4. Radiolysis of water in containment by beta and gamma of 50% of core halogens at 0.30 mol/100 ev.
5. Reactor thermal power = 3216 MWt
6. Containment free volume = 261 x 106 cu. ft.

Prediction of the hydorgen generation rate shows that the rate of hydrogen production diminshes as time after the accident increases. The object of the system design is to maintain the hydrogen level in the containment under the lower flammable limit of 4.1 v/o with adequate safety factor to allow for imperfect mixing and errors in sampling. For design purposes, a criterion was established that an average concentration of 2.0 v/o hydrogen should not be exceeded when one of two recombiners is in use. From the hydrogen generation data it was ascertained that the following processing rates would maintain the average hydrogen concentration constant at the value shown if the recombiner were started when the concentration first reached that value:

To Maintain H_2 Concentration At	Recombiner Must Start At	Processing Atmosphere At Rate Of
1 v/o	2.7 days	293 scfm*
2 v/o	9.8 days	96.6 scfm
4.1 v/o	51.0 days	27.9 scfm

*scfm defined at 14.7 psia and 32°F

The recombiner units provided will each process about 331 scfm, consuming essentially all of the contained hydrogen. If one of these units is started 9.8 days after the accident, when the conservatively estimated hydrogen concentration reached 2.0 v/o, it will consume 0.02 x 331 or 6.6 scfm of hydrogen. The production rate at 9.8 days has decayed to about 1.74 scfm. One of the units, therefore, meets the design criterion with considerable margin, and the second unit provides the redundancy of a spare system of equal capacity. Expressing the margin in a different way, the first unit could be operated intermittently or turned down to about 26% of rated capacity without violating the design criterion. As the production rate of hydrogen continuously decreases with time, the margins cited become larger.

To assure that stratification effects or sample errors would not permit all or parts of the containment to hold hydrogen in excess of the lower flammable limit (4.1 v/o), when the measured concentration is 2.0 v/o, the air recirculation system is operated to maintain a high degree of mixing throughout the containment, including the dome and the reactor loop compartments.

Sampling error may be minimized by proper administrative control of chemical standards and by repeat sampling. Analytical error is small. The probable error by the chromatographic method can be judged on the following basis:

sensitivity	0.02 volume percent
reproductibility	± 5% of measured value

Hence, the sum of all errors is well below the margin between two volume percent to be maintained and four volume percent lower flammable limit.

Description

Inside the containment are two complete combustor systems, one a spare. Each system consists of a blower to circulate containment air to the combustor, a combustion chamber complete with main burner, two igniters (one a spare), pilot burner, and a dilution chamber downstream from the flame zone where products of combustion are mixed with a large excess of containment air to reduce the temperature of gas leaving the system below 300°F.

Gaseous hydrogen and oxygen are stored in cylinders outside the containment. The hydrogen is piped directly to the combustor. Oxygen is bled into the containment vessel through a separate penetration to be mixed with containment gases by the main containment ventilating blowers. The external hydrogen is required for fuel because the hydrogen level in the containment ambient atmosphere is maintained below flammable limits. Because oxygen in the containment is depleted by combustion, and because the burner is expected to be unstable below some minimum oxygen level (to be determined by test), oxygen must be added to the containment from an external source. The oxygen and hydrogen facilities, including storage, metering, piping, and penetrations, are remote from each other.

Fuel addition is controlled to produce a temperature in the combustor of 1500 - 1600°F. This is substantially in excess of the temperature required for complete combustion of hydrogen in air. Oxygen makeup is proportioned to hydrogen flow to maintain the required stoichiometry.

The decision of when to start, stop, or throttle the combustion system is based on intelligence from containment air samples analyzed for hydrogen and oxygen in the control laboratory. It is intended that the combustor will be ignited when the hydrogen in the containment atmosphere reaches two volume percent. It may be run full throttle until the hydrogen is reduced to about 1.5 percent and then shut off for several days, or it may be cut back by reducing fuel flow to just burn enough containment hydrogen to maintain the desired concentration, as verified by successive samples. It should be noted that the flame process is self-regulating in the sense that the quantity of containment air which will be heated to reaction temperature as it passes through the combustion chamber is dependent on the rate at which hydrogen is burned. When operating at the system design point, essentially all of the containment gas passing through the combustion chamber is heated to combustion temperature. The system can be throttled* by reducing the hydrogen fuel flow rate so that the flow of containment air in the combustion chamber is in excess of that which could be heated uniformly to combustion temperature. The surplus air does not participate in the combustion process, but is merely entrained and heated to a lower temperature by mixing with the combustion products.

Combustor ignition is provided by a capacitance type system equipped with two surface gap plugs designed for operation in a wet environment. The capacitor is located outside the containment and the plugs located on the combustion chamber. Ignition leads into the combustor are completely housed into a pressure tight system with the wire connection at the ignitor field brazed at the pressure tight head on the ignitor. This provides absolute continuity from the capacitor outside the containment to either of the plugs inside the containment.

Design of the thermocouple system which indicates pilot flame and main burner ignition parallels that of the ignition system. Each combustor system contains two thermocouples (one a spare). Thermocouple leads are fabricated by a procedure similar to that used for igniter system leads.

The external systems incorporate the following features for operational safety and reliability:

1. Two complete control systems for fuel gas.
2. Isolation provisions for each fuel gas line to prevent outleakage when not in use, consisting of a check valve inside the containment, and at least two series normally closed valves outside the containment.
3. A block-and-bleed provision for each fuel gas and oxygen makeup line to prevent inleakage when not in use.
4. Provisions for purging fuel lines with nitrogen from the cylinder manifold before introducing combustible gas.
5. Alarm functions to alert the operator in case of loss of blower pressure, low combustor temperature (flameout), and low fuel gas or oxygen manifold

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* 6:1 operating range is expected, to be verified during the test program.

6. Provision to include an optical system to give verification of ignition independent of thermocouple readout if tests currently in progress prove that such a system is feasible.
7. Capability to test the complete control systems at any time by carrying out a complete dry-run startup using artificially generated thermocouple signals to simulate lightoff.
8. Capability to test the system at any time by a complete test and verification of ignition while the reactor is operating. This test can be conducted from operating stations outside the containment, although direct access to the units is permissable during power operation.
9. Piping design in conformance with ASA-B-31.1 Code for Pressure Piping.
10. Pneumatic control valves provided with an emergency air supply in the event of loss of instrument air.
11. System redundance such that no single active component failure can disable both combustor systems.

Instruments, controls, and suitable panels are provided to perform all these functions in a safe and reliable manner. An operator will be stationed at the control panel of a particular recombiner system whenever that system is in operation or under test. Radiation levels in the vicinity of the control panel one day after the maximum hypothetical accident (TID-14844 model source levels) will permit continuous acess. The following features are incorporated in the control system and panel design to ensure operational safety and reliability:

1. Wiring and electrical equipment in conformance with the National Electric Code, NEMA Standards, and the proposed IEEE Criteria for Nuclear Power Plant Protection Systems(where applicable).
2. A separate panel for each recombiner system.
3. Physical and functional separation of redundant features such that no single failure can invalidate both features.

Combustor Test Program

It is intended to test a recombiner system at a suitable site for two purposes

- to demonstrate that the design is sound (proof testing).
- to determine certain limits for the combustor's performance.

First of all, the combustor will be operated at design condition and the dampers in the air delivery lines to the combustion chamber and the dilution section will be set for proper air delivery. Once finally set, the dampers will remain at the same position when installed in the containment. All test runs will be made with no further adjustment or control of blower air delivery except for the test described under (4) below. The following performance information will then be determined by suitable test.

1. General operating performance of combustor and basic auxiliaries at lightoff and under operation with hydrogen fuel rate varied to provide a combustion zone outlet temperature of 300°F to 1800°F.
2. Starting with air to the combustor, oxygen content will be lowered and nitrogen content raised to determine the limit of flame stability with diminished oxygen.
3. The combustor will be operated at conditions simulating its two design points and with 1-2% hydrogen in the air to the burner (see Figure 6). Outlet hydrogen from the combustor will be measured to determine the efficiency of hydrogen removal.
4. The stability range of the burner will next be checked for the following conditions.
   a. Pilot ignition with variable air flow
   b. Main burner ignition with variable air flow
   c. Check of burner stability with constant air flow but rapidly varying fuel flow.
5. The effect of various quantities of steam and/or entrained water on light-off and normal burner operation will be determined.

All runs shall be made with 1-2 volume per cent hydrogen in inlet air to combustor.

CONTAINMENT SPRAY PROGRAM

Definition of Safety Related Problem

The addition of reactive chemicals to the containment sprays has been proposed as a means to reduce the iodine content of containment leakage to within site regulatory limits. Data which has already been obtained in engineering seal tests confirms the absorptive capacity of the chemically modified sprays. Further refinement of the analytical model is being pursued in order to be able to justify additional performance of the sprays and to evaluate non-ideal factors in extrapolating to large containment vessels. It must be established also that in no way does the use of the proposed additive chemicals jeopardize the performance or integrity of the containment or emergency core cooling systems. The basis for the design and evaluation are described in the PSAR. The discussion below is an amplification of the R&D program.

The following technical considerations and areas are being investigated in order to demonstrate the full capability of the spray system.

A. In extending the height of the chamber in which spray absorption takes place, the possibility of more interaction (i.e. coalescence) between droplets arises due to their longer residence time.

B. Simplifying assumptions which were made in preliminary analyses and verified in intermediate size tests, namely that absorption rate is gas-film controlled, must be re-examined to determine whether liquid phase mass transfer and/or chemical reaction may influence overall absorption rate in a large system.

C. The effect of non-uniformity of spray droplet size on the surface area for absorption has been incorporated in previous performance analyses. It is desired to consider non-uniformity effects on other aspects of the problem, inlcuding (in addition to collision frequency mentioned above) the increased residence time of small drops, gas phase mixing, and the depletion of the capacity of small drops to react with iodine due to their smaller volume-to-surface ratio.

D. It must be shown that the use of chemical additives does not promote corrosion or other degradation of the integrity of the containment or ECCS such that the safety function of these systems could be impaired.

E. The maximum rate of hydrogen generation from corrosion or radiolysis of water under post accident conditions must be assessed in order to establish the level of protective action to be taken against the accumulation of a flammable or explosive atmosphere. It is necessary that the basis for such an assessment include any effect on hydrogen production due to the presence of spray additive chemicals.

Specific Technical Information to be Obtained

A. Droplet Coalescence

For purposes of a conservative system evaluation, it will be assumed that collisions between droplets occurring with the spray trajectory patterns by virtue of unequal velocities and/or intersecting spray patterns will result in coalescence of the colliding drops. The spray development program will seek a quantitative evaluation of the effect of this coalescence on the calculated absorptive capacity, so that iodine removal rate can be assessed accordingly. The model will be applied to the NSPP and CSE experiments to show consistency with presently available data obtained with the shorter residence times of these experiments.

This analysis will be completed by Westinghouse, evaluated, and results made available in the third quarter of 1969.

B. Liquid Phase Mass Transfer Resistance

For purposes of conservative system evaluation, the existence of a liquid film mass transfer resistance will be reflected in the analytical model values of liquid film mass transfer coefficient and partition factor will be derived from literature data and applied to this model to show whether or not the liquid film resistence would limit performance in a large containment. As a test of the reasonableness of the values chosen, the same model will be applied to the NSPP and CSE experiments. A lack of consistency would indicate that the liquid film resistance of the model was too conservative and could be relaxed.

This analysis will be completed by Westinghouse, evaluated, and results made available in the third quarter of 1969.

C. Non Uniformity of Spray Drop Size and Coverage

The spray droplet population will be treated as having a non-uniform spectrum in all aspects. The data from previous Westinghouse measurements with the spray nozzles proposed for plant installation will define this spectrum at the point of introduction into the containment atmosphere. As mentioned earlier, changes in this spectrum due to coalesence during drop fall will be included. The processes of gas phase mixing will be treated also to determine the effect of limited regions of the containment which are not reached by spray.

This work will be completed by Westinghouse, evaluated, and results made available in the third quarter of 1969.

D. Materials Compatibility

Considerable data on the corrosion of major construction materials have been obtained under the Westinghouse spray development program to date. By further correlating and documenting this work it can be shown that general corrosion is not significant for the materials which have been used in the containment structure and in engineered safety features. Aluminum which is present in the containment is attacked, but is not used in vital components or structures. Documentation will cover, in particular, the temperature dependency of corrosion over the full range of accident conditions and will cover as well the degree of sensitivity of stressed and welded specimens. It is expected that data previously obtained in Westinghouse programs and supplemented by published literature will largely cover these concerns. The data which as developed in this program will be reviewed with the Staff.

As part of a general program to prove the durability of safeguards-related components under accident environments, tests are being conducted by Westinghouse and certain of its suppliers. Exposure to the spray solution under these conditions will provide information as to the compatibility of electrical components, lubricants, sealants, and typical insulating and protective coating systems. Sump pH v.s. time and spray pH at the nozzle are not fixed at this time. A criterion has been established which places the minimum containment sump pH at 8.5 when iodine is to be retained, and the maximum pH of sprayed solution at 10.0 for materials expsoure. The spray chemical addition program will be designed to meet these criteria. At pH 10 or less, the data obtained in a proprietary Westinghouse study indicate general corrosion to be quite acceptable, except for aluminum which must not be used for vital service in the spray environment. Further effort in the study just mentioned will be devoted to special considerations of corrosion in stressed and welded specimens, to provide assurance that localized attack is not a threat to the integrity of the containment and ECCS. While this is a continuing program, the status of information reflecting on the use of the proposed spray additives will be evaluated and reported in the first quarter of 1969.

E. Hydrogen Generation

Data on the corrosion of aluminum and other metals obtained under the materials compatibility phase of the program has provided a basis for predicting accumulation of hydrogen in the post accident containment atmosphere. Another Westinghouse program is in progress to evaluate the rate of radiolytic decomposition of spray and core cooling water. Its purpose is to ascertain the dependence of radiolysis on flow, temperature and chemical factors, including the effect of spray additives. Most of the data have been collected, and analysis is expected to extend through the last quarter of 1968, permitting an assessment of the magnitude of possible hydrogen accumulation at that time.

The capacity of the containment to accommodate hydrogen formation will not require further research or development, since the limits of flammability and the mechanisms for energy input from various types of recombination mechanisms are adequately understood.

<u>Basis for Assurance that R&D will Meet Objectives</u>

It is the overall objective of the spray development program to compile engineering data and analyses which justify reliance on the containment sprays to provide an average reduction factor of at least 5.3 in the iodine leakage from the Unit 3 reactor containment systems in the first two hours. Making the conservative assumption that 10% of the available iodine is not removable by sprays, it is apparent that the objective just stated requires a reduction factor of about 11 in the leakage of removable forms. The simple analytical model (which shows agreement with engineering scale test data, generally within a band of $\pm$ 20%) predicts that a reduction factor of approximately 60 could be expected if the effects of coalescence, liquid film resistance, drop size non-uniformity, etc., could be shown to be negligible, and if only one of the two spray systems were operated. Considering the agreement with test data and preliminary assessment of non-ideal scaling effects, there remains a very high confidence that the final evaluation will show an abundant margin between the predicted dose reduction factor for removable forms of iodine and the minimum acceptable factor 11, even when conservative allowance is made for coalescence, liquid film resistance and non-uniformity.

In the matter of materials compatiblity, there is also a high degree of assurance that results will confirm, rather than militate against the use of alkaline spray chemistry. This conclusion is based on the fact that this chemistry is one of the most favorable environments for stainless and carbon steels, and that there is flexibility in the selection or protection of other materials where safety is of concern should any adverse results be obtained.

In the area of hydrogen generation, there is presently sufficient data to make a conservative estimate of aluminum corrosion in the spray environment, and a maximum theoretical yield for the process of radiolysis. Thus at any time, an assessment of the need for protective action such as controlled hydrogen recombination or feed and bleed hydrogen purge could be conservatively made. Final data would only indicate the degree to which those actions might be deferred or avoided with better technical information at hand.

In summary, therefore, the probability is quite remote that there will be a need to adopt a back-up design in lieu of chemical sprays for iodine removal. Should this be necessary, provisions are made for charcoal filters in the air handling system to achieve the necessary overall reduction factor of 5.3 in overall organic and inorganic leakage.

It is expected that the spray development program will furnish all of the required analytical capability and reference data with which to demonstrate the full capability of the spray process by the third quarter of 1969.

PROVISONS FOR POST LOSS OF COOLANT ACCIDENT PROTECTION

The concept of PLOCAP (post loss of coolant accident protection) has been developed to meet the following bases:

a. PLOCAP will provide a means of covering and cooling the core thus preserving core heat transfer geometry in the event reactor vessel integrity is lost due to thermal shock caused by operation of the ECCS following a loss of coolant accident.

b. PLOCAP will be designed such that no new hazards to the health and safety of the public will be introduced by the equipment.

c. PLOCAP will be integrated into the existing ECCS in such a manner that in no way lessens the capability of the ECCS to meet its design objectives.

System Design

Additional equipment would be required to protect the Indian Point Unit No. 3 core from major damage in the event of a post accident vessel fracture. Consideration has been given to the additional plant requirements on the basis that the Indian Point Unit No. 3 systems will be adopted on the present design criteria and that vessel fracture will not be deemed credible. Provision will, however, be made in the design and layout of the plant to enable the installation of additional equipment if this proved to be necessary after the completion of construction and the plant operation.

Two services must be guaranteed by the systems in the event of an M.C.A. followed by vessel fracture:

(i) The core must be maintained in a water pool.
(ii) Energy must be removed continuously from the plant.

The additional duty imposed upon the systems is the maintenance of a water pool. This duty is to be met by a reactor vessel cavity flooding system and a recirculation system which delivers water to the top of the core.

Following an M.C.A. the reactor vessel cavity would be flooded to just below the reactor vessel nozzles such that in the event of a major crack developing in the lower section of the vessel, subsequent to the loss of coolant accident, the core would reamin in a water pool. Energy continues to be dissipated by the core and in the absence of the normal coolant flow boiling would take place in the core region.

If no further action were taken steam pressure build-up would commence in the upper part of the reactor vessel dome and eventually the water pool would be displaced from the core region by the growth of the steam bubble. Calculations suggest that the steam pressure relieving area provided by the original loss of coolant accident would not necessarily be sufficient for steam pressure relief.

In order to prevent the build-up of a steam bubble in the reactor vessel dome it is proposed to direct the low head injection flow and subsequent recirculation flow to the top of the core via the hot legs. Cold water flows to the plenum above the core and mixed with the hot water in and above the core region. The cold water flow must be sufficient to prevent steam production and subsequent release to the reactor vessel upper dome.

Cavity flood system - A standpipe incorporated over the instrument passageway leading to the cavity from the containment floor will permit retention of water in the reactor vessel cavity to a level above the core. The flooding of the cavity will be achieved by two sub systems, viz., the cavity flood tank system and the recirculation sump/cavity transfer pump system.

In the event of an M.C.A. the cavity flood valves would be opened by the S and accumulator low pressure signals. Water would flow to the cavity space from the flood tanks to fill the reactor cavity to just below the bottom of the reactor vessel. The capacity of the cavity tanks will be sized to limit the flood level to below the vessel bottom to prevent damage to the system in the event of a spurious opening of the cavity tank valves.

The combined S/A.L.P. signal would also open the valves in the discharge of the recirculation pumps. Water would be transferred from the recirculation sump to the reactor vessel cavity and thus fill the remaining part of the reactor vessel cavity.

Hot leg injection - The hot leg injection duty is required during both the injection phase and the recirculation phase. For each phase the design requirement is to prevent steam bubble accumulation in the upper dome of the reactor vessel.

During injection, water is drawn from the refueling water storage tank. During recirculation, water is drawn from the containment sump cooled in the residual heat removal heat exchangers and returned to the hot legs of the reactor coolant system.

Provisions

The main provisions to be incorporated into the plant may be listed as follows:

1. A standpipe over the instrument passageway leading to the cavity from the containment floor to permit the retention of water in the reactor vessel cavity to a level above the core.

2. Nozzles on each hot leg pipe to permit an upper core deluge system to be installed.

3. Provision for a second containment sump line will be made to enable high recirculation flow rates.

4. Space to be allocated in the primary auxiliary building to make provision for extra heat exchange capacity together with extra pumping capability.

5. Provisions made to enable the first stage cavity flood system to be installed, i.e., flood tanks and associated pipework.

6. Pipe layouts and plant arrangements will be made to take account of the extra pipework that would be required for the PLOCAP system together with the necessary containment penetrations.

The Conservatism of the Con Edison Off-Site Dose Evaluation Model

In the following discussion we shall attempt to demonstrate that the meteorological model used in the Indian Point Unit No. 3 loss-of-coolant dose evaluation is conservative for this site, even though it yields somewhat lower dosages than the AEC model.

In order to compare the two models, we have computed values of X/Q for each hour following an accident. The results are attached.

Inspection of the attached table indicates that the Con Edison model produces stronger doses than the AEC model does in all hourly classes except 2-8. In order to meet the AEC requirements in this class it will be sufficient to demonstrate that the actual wind speed under inversion conditions at the site will not fall below 1.86 m/s for a period of six consecutive hours following a two-hour period in which the wind speed was 1 m/s, with the wind direction steady in the same direction for the full 8 hours.

The experimental evidence that this triple combination of low wind speed, prolonged duration and absolute steadiness is highly improbable at the Indian Point site is contained in Figure 1.6-1 of Docket No. 50-286, Exhibit B, Volume 1, Section 1.6, reproduced herein as Figure 1.6-1, with 1 m/s and 2 m/s wind speed circles superimposed. The figure is based on measurements made with an Aerovane mounted on an anchored ship in the Hudson River near the site.

The figure shows the existence of low speed winds oriented predominantly along the 030°-210° direction, up-valley (toward the north) during daylight hours, and down-valley (toward the south) during nocturnal hours. These flows occur in the absence of geostrophic flow and are classically considered to be thermally generated by contact with valley walls which undergo cyclic heating and cooling by radiation. The strength of such flows is dependent upon valley geometry and orientation, number of tributary valleys and other factors enumerated and described more fully by B. Davidson (1961)*.

The Hudson River Valley in the vicinity of the Indian Point site has geometric proportions conducive to strong valley flows and, in addition, receives drainage from both the extension of the principal valley to the northwest and the tributary Conopus Creek Valley to the northeast. Thus, there is reason to believe that the measurements shown in Figure 1.6-1 are representative of not only the indicated two months, but of all occurrences of protracted inversions with large scale stagnation.

The initial stage of the AEC model, 1 m/s winds of constant direction for two hours, is seen to occur during the late morning (230° winds from 1000 to 1200 hours) and late evening (050° winds from 1900 to 2100 hours). During the following period, the two flows do not conform to the AEC model. Both flows experience

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* Valley Wind Phenomena and Air Pollution Problems, Journal APCA Vol. 11, No. 8, August

a rapid speed increase to about 2.5 m/sec. The down-valley flow is sustained in a rather narrow direction band centered on 210° for a period of about 12 hours, while the up-valley flow rotates counter-clockwise about 50° during a four hour period. Thereafter the winds subside and change direction very rapidly.

It is clear from the above that the down-valley wind is more critical for the diffusion model, but that it satisfies only two of the three AEC specified criteria: reasonable steadiness and duration. Wind speeds of 1 m/s and under may occur for durations up to 3 hours, but are accompanied by a large direction change. The wind speed which occurs simultaneously with the other two criteria is always greater than 2 m/s. This fact has been recognized in the Con Edison model by the assumption of a 2 m/s wind speed after the first two hours.

In summary, we believe that the Indian Point site has a sufficiently unique topography and microclimatology to warrant a relaxation of the AEC model in favor of a model based on site measurements.

	AEC Model					Con Edison Model					X/Q Ratio
Hour	Wind Speed m/s	Stability Class	Time Fraction	Steadiness	$10^4 X/Q$	Wind Speed m/s	Stability Class	Time Fraction	Steadiness	$10^4 X/Q$	Con Edison / AEC
0-2	1	F	1.00	Steady	3.54	1	I	1.00	Steady	3.80	1.07
2-8	1	F	1.00	Steady	3.54	2	I	1.00	Steady	1.90	.54
8-24	1	F	1.00	22.5°	0.73	2	I	1.00	Steady	1.90	2.60
24-96	2	F	.60	22.5°	0.44	2.03	I	.42			
	3	D	.40			2.80	N	.38	20°	0.18	1.29
96-720	2	F	0.33			5.24	L_2	.06			
	3	D	0.33	22.5°	0.10	1.74	L_1	.14			
	3	C	0.34								

Notes

1. In both models a virtual source displacement of 430 meters was used for the periods 0-2 hours and 8-24 hours.

2. Angles in Steadiness column are size of sector in which plume is assumed to meander uniformly.

3. Con Edison model assumed to run 31 days = 744 hours.

4. Weighted average X/Q for AEC model for period 24-720 hours is 0.14.

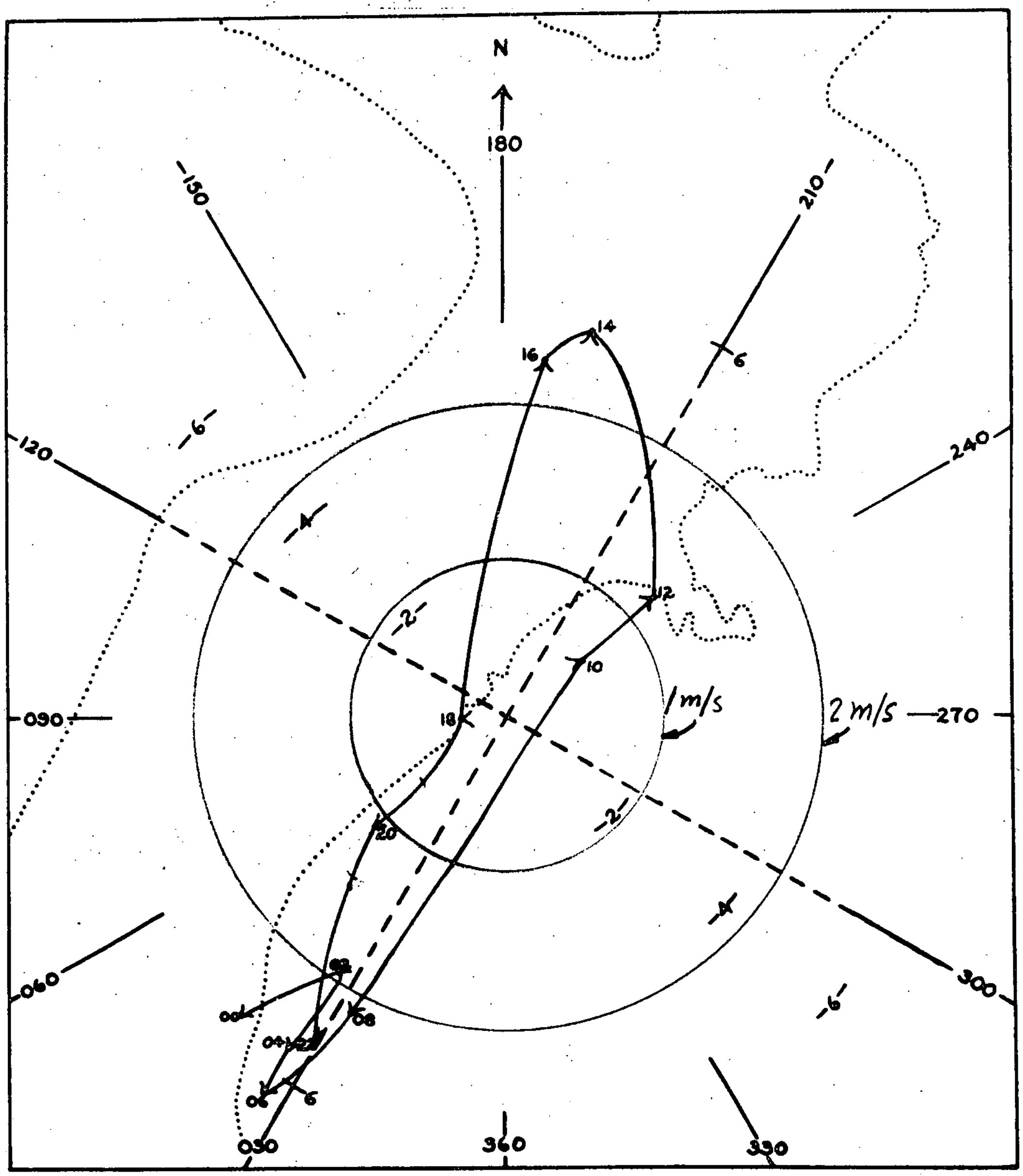

Figure 1.6-1

Diurnal variation of mean vector wind for virtually zero pressure gradient conditions, Sept.-Oct., 1955, 70 ft. above river.

Revised October 1968

THERMAL SHOCK STATUS

Potential failure modes for the reactor vessel in the event of a thermal shock initiated by emergency core cooling operation have been examined in detail. Three failure modes have been investigated; ductile yielding mode, fatigue mode and the brittle fracture mode.

In the ductile yielding mode a comparison of the material yield stress to the calculated stress indicates that under conservative assumptions the outer 82% of the actual base metal thickness remains below the minimum material yield strength at all times during a safety injection transient. Consequently local yielding may occur in the inner 18% of the base metal and in the cladding, and therefore this mode will not cause a breach of integrity of the reactor vessel.

In the fatigue mode the location in the vessel with the highest usage factor, the in-core instrumentation tube attachment welds to the vessel bottom head, was examined in detail.

A conservative analysis indicates that if fatigue were the governing failure mode, that 9 safety injection transients could be tolerated since the total accumulative usage factor would be .985 versus the code allowable of 1.0. This factor includes all other transients evaluated in the fatigue analysis and identified in Section 4 of the PSAR.

For the brittle fracture mode two analyses were performed: (1) the transition temperature approach, and (2) the fracture mechanics approach.

In the transition temperature approach*, a conservative nil ductility transition temperature (NDTT) was considered based on irradiation for a 40 year reactor vessel life and a large initial crack in the vessel to determine the crack arrest temperatures (CAT).

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* W.S. Pellini and P.P. Puzak, "Fracture Analysis Diagram Procedures for the Fracture-Safe Engineering Design of the Steel Structures," NRL Report 5920, March 15, 1963.

A comparison of the CAT curves with the corresponding stress-temperature conditions for the end of life transient show that 65% of the reactor vessel wall thickness remains below the crack-arrest curve at all times following safety injection. Therefore, if a crack were assumed present on the ID of the vessel and assumed to propagate it would not propagate further than 35% through the wall thickness in the extreme case.

In the fracture mechanics approach a detailed analysis using the methods of linearly elastic fracture mechanics was also performed. Analyses were performed for both a local flaw existing at arbitrary depths into the vessel and an axi-symmetrical flaw (circumferential crack) existing at arbitrary depths into the vessel wall. The resulting stress intensity factor versus crack detph for both types of flaws were calculated and compared to the fracture toughness of the material through the vessel wall to determine the extent of potential propagation. Fracture toughness bands as a function of irradiation and temperature were determined based on the latest available data. When considering the conservative lower bound estimates of fracture toughness the crack propagation for the extreme end of life condition would be less than 32% of the vessel wall. Considering best estimate fracture toughness properties, there would be no crack propagation throughout the life of the vessel.

Results of these analyses therefore show that under the postulated accident conditions the integrity of the reactor vessel will be maintained throughout the life of the plant.

CONTROL AND PROTECTION SYSTEM STATUS

The Indian Point Unit No. 3 control and protection system differs somewhat from that orignally proposed for Diablo Canyon Unit No. 1. The following is in response to concerns expressed by the AEC Staff and the ACRS in the Diablo Canyon review.

The level comparators on the steam generators have been deleted and a single level channel is used for both control and protection. The two additional narrow range level channels are used for protection purposes only. Continuous rod position indication will be provided for each control rod and period indication (decades/min.) will be provided for both the source and intermediate range channels. This system is the same as that proposed on all four-loop Westinghouse reactors since Diablo Canyon Unit No. 1 and is in accordance with the proposed IEEE criteria for Nuclear Power Plant Protection Systems (IEEE No. 279) and is essentially identical to that described in the FSAR for Indian Point Unit No. 2, Docket No. (50-247).

In particular, the following discussion explains the basis on which the system meets Section 4.7 of the IEEE Criteria relating to separation of control and protection.

Specific Control and Protection Interactions

Nuclear Flux

Four power-range nuclear flux channels are provided for overpower protection. Isolated outputs from all four channels are averaged for automatic control rod regulation of power. If any channel fails in such a way as to produce a low output, that channel is incapable of proper overpower protection. In principle, the same failure would cause rod withdrawal and overpower. Two-out-of-four overpower trip logic will ensure an overpower trip if needed even with an independent failure in another channel.

In addition, the control system will respond only to rapid changes in indicated nuclear flux; slow changes or drifts are overriden by the temperature control signals. Also, a rapid decrease of any nuclear flux signal will block automatic rod withdrawal as part of the rod drop protection circuitry. Finally, an overpower signal from any nuclear channel will block automatic rod withdrawal. The set point for this rod stop is below the reactor trip set point.

Coolant Temperature

Four T_{avg} channels are used for overtemperature-overpower protection. (See Figure 7.2-12 for single channel). Isolated output signals from all four channels are also averaged for automatic control rod regulation of power and temperature. In principle, a spuriously low temperature signal from one sensor would partially defeat this protection function and also cause rod withdrawal and overtemperature. Two out of four trip logic is used to insure that an overtemperature trip will occur if needed even with an independent failure in another channel.

In addition, channel deviation alarms in the control system will block automatic rod motion (insertion or withdrawal) if any temperature channel deviates significantly from the others. Automatic rod withdrawal blocks will also occur if any one of four nuclear channels indicates an overpower condition or if any one of four temperature channels indicates an over-temperature condition. Two-out-of-four trip logic is used to ensure that an overtemperature trip will occur if needed even with an independent failure in another channel. Finally, as shown in Section 14.1, the combination of trips on nuclear overpower, high pressurizer water level, and high pressurizer pressure also serve to limit an excursion for any rate of reactivity insertion.

Pressurizer Pressure

Four pressure channels are used for high and low pressure protection and for overpower-overtemperature protection. Isolated output signals from these channels also are used for pressure control and compensation signals for rod control. These are discussed separately below:

(1) Control of rod motion: One of the pressure channels is used for rod control with a low pressure signal acting to withdraw rods. The discussion for coolant temperature is applicable, i.e., two-out-of-four logic for overpower-overtemperature protection as the primary protection, with backup from multiple rod stops and "backup" trip circuits. In addition, the pressure compensation signal is limited in the control system such that failure of the pressure signal cannot cause more than about a 10°F change in T_{avg}. This change can be accommodated at full power without a DNBR less than 1.30. Finally, the pressurizer safety valves are adequately sized to prevent system overpressure.

(2) Pressure Control: Spray, power-operated relief valves, and heaters are controlled by isolated output signals from the pressure protection channels.

a) Low Pressure

A spurious high pressure signal from one channel can cause low pressure by spurious actuation of spray and/or a relief valve. Additional redundancy is provided in the protection system to ensure underpressure protection, i.e., two-out-of-four low pressure reactor trip logic and one-out-of-three logic for safety injection. (Safety injection is actuated on one-out-of-three coincident low pressure and low level.)

In addition, interlocks are provided in the pressure control system such that a relief valve will close if either of two independent pressure channels indicates low pressure. Spray reduces pressure at a lower rate, and some time is available for operation action (about three minutes at maximum spray rate before a low pressure trip is reached.)

b) High Pressure

The pressurizer heaters are incapable of overpressurizing the reactor coolant system. Maximum steam generation rate with heaters is about 15,000 lbs/hr., compared with a total capacity of 1,224,000 lbs/hr. for the two safety valves and a total capacity of 358,000 lbs/hr. for the two power-operated relief valves. Therefore, overpressure protection is not required for a pressure control failure. Two-out-of-three high pressure trip logic is therefore used.

In addition, either of the two relief valves can easily maintain pressure below the high pressure trip point. The two relief valves are controlled by independent pressure channels, one of which is independent of the pressure channel used for heater control. Finally, the rate of pressure rise achievable with heaters is slow, and ample time and pressure alarms are available for operator action.

Pressurizer Level

Three pressurizer level channels are used for high level reactor trip (2/3) and low level safety injection (1/3 logic level coincident with pressure). Isolated output signals from these channels are used for volume control, increasing or decreasing water level. A level control failure could fill or empty the pressurizer at a slow rate (on the order of half an hour or more).

(a) High Level

A reactor trip on pressurizer high level is provided to prevent rapid thermal expansions of reactor coolant fluid from filling the pressurizer: The rapid change from high rates of steam relief to water relief can be damaging to the safety valves and the relief piping and pressure relief tank. However, a level control failure cannot actuate the safety valves because the high pressure reactor trip is set below the safety valve set pressure. With the slow rate of charging available overshoot in pressure before the trip is effective is much less than the difference between reactor trip and safety valve set pressures. Therefore, a control failure does not require protection system action.

In addition, ample time and alarms are available for operator action.

(b) Low Level

For control failures which tend to empty the pressurizer, one-out-of-three logic for safety injection actuation on low level coincident with low pressure ensure that the protection system can withstand an independent failure in another channel.

In addition, a signal of low level from either of two independent level control channels will isolate letdown, thus preventing the loss of coolant. Also, ample time and alarms exist for operator action.

Steam Generator Water Level; Feedwater Flow

Before describing control and protection interaction for these channels, it is beneficial to review the protection system basis for this instrumentation.

The basic function of the reactor protection circuits associated with low steam generator water level and low feedwater flow is to preserve the steam generator heat sink for removal of long term residual heat. Should a complete loss of feedwater occur with no protective action, the steam generators would boil dry and cause an overtemperature-overpressure excursion in the reactor coolant. Reactor trips on temperature, pressure, and pressurizer water level will trip the plant before there is any damage to the core or reactor coolant system. However, residual heat after trip would cause thermal expansion and discharge of the reactor coolant to containment through the pressurizer relief valves. Redundant emergency feedwater pumps are provided to prevent this. Reactor trips act before the steam generators are dry to reduce the required capacity and starting time requirements of thse pumps and to minimize the thermal transient on the reactor coolant system and steam generators. Independent trip circuits are provided for each steam generator for the following reasons:

1. Should severe mechanical damage occur to the feedwater line to one steam generator, it is difficult to ensure the functional integrity of level and flow instrumentation for that unit. For instance, a major pipe break between the feedwater flow element and the steam generator would cause high flow through the flow element. The rapid depressurization of the steam generator would drastically affect the relation between downcomer water level and steam generator water inventory.

2. It is desirable to minimize thermal transient on a steam generator for credible loss of feedwater accidents.

   It should be noted that controller malfunctions caused by a protection system failure affect only one steam generator. Also, they do not impair the capability of the main feedwater system under either manual control or automatic T_{avg} control. Hence, these failures are far from being the worst case with respect to decay heat removal with the steam generators.

(1) Feedwater Flow

A spurious high signal from the feedwater flow channel being used for control would cause a reduction in feedwater flow and prevent that channel from tripping. A reactor trip on low-low water level, indepednent of indicated feedwater flow, will ensure a reactor trip if needed.

In addition, the three-element feedwater controller incorporates reset on level, such that with expected gains, a rapid increase in the flow signal would cause only a 12 inch decrease in level before the controller re-opened the feedwater valve. A slow increase in the feedwater signal would have no effect at all.

(2) Steam Flow

A spurious low steam flow signal would have the same effect as a high feedwater signal, discussed above.

(3) Level

A spurious high water level signal from the protection channel used for control will tend to close the feedwater valve. This level channel is independent of the level and flow channels used for reactor trip on low flow coincident with low level.

a) A rapid increase in the level signal will completely stop feedwater flow and actuate a reactor trip on low feedwater flow coincident with low level.

b) A slow drift in the level signal may not actuate a low feedwater signal. Since the level decrease is slow, the operator has time to respond to low level alarms. Since only one steam generator is affected, automatic protection is not mandatory and reactor trip on two-out-of-three low-low level is acceptable.

Steam Line Pressure

Three pressure channels per steam line are used for steam break protection (two-out-of-three low pressure signals for any steam line actuates safety injection) one of these channels is used to control the power-operated relief valve on that steam line. These valves are typically rated at 10% of the safety valve capacity. A spurious high pressure signal from the channel used for control will open the relief valve and cause low pressure. This is a slow rate of steam release, evaluated as a credible steam break in Section 14.2.5. In the analysis of steam breaks of this size, no credit is taken for the steam line pressure instrumentation. Safety injection is actuated by the pressurizer instrumentation. Therefore, control failure does not create a need for the protection, and two-out-of-three logic is acceptable.

The control and protection system is being reviewed to determine the potential effects of common failure modes as well as the single failure criteria in response to recent concerns expressed by the AEC staff and the ACRS for consideration during detailed design.

Flooding at the Site

Flooding due to one cause is not expected to result in water levels high enough to adversely affect engineered safeguards or any equipment required for safe shutdown. Two specific flood conditions will be examined to ascertain the water level.

The water surge due to a hurricane coincident with a runoff flow from heavy rainfall will be studied. Meteorological parameters determined by the flood flow will be utilized with a report for the Coastal Engineering Research Center* which predicts the surge caused by a hurricane.

The flood caused by a maximum rainfall coincident with a dam failure will also be determined.

In either case the design criteria for engineered safeguards or equipment required for safe shutdown is that they will be located or otherwise protected to assure operation with the highest predictable flood levels determined from these studies.

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* "Shore Protection, Planning and Design" Technical Report No. 4, Coastal Engineering Research Center.

SODIUM HYDROXIDE INJECTION

The addition of sodium hydroxide to the containment spray will be automatically actuated on the containment spray signal and will not require operator action for actuation. Provisions are being considered for the possible addition of a manual block, during the first few minutes following actuation of the containment spray system to obviate a spurious signal. Instrumentation and operator action will be examined for the design and the final actuation design will be presented to the AEC for their review.

STRESS LIMIT CURVES

The following stress limit curves, as referenced in Item 15, page 14, Note 1 are presented as supplementary information to be later published in Revision 2 to WCAP-5890.

302B, HOOP STRESS = 0.53 Sy

HOLLOW - CIRCULAR

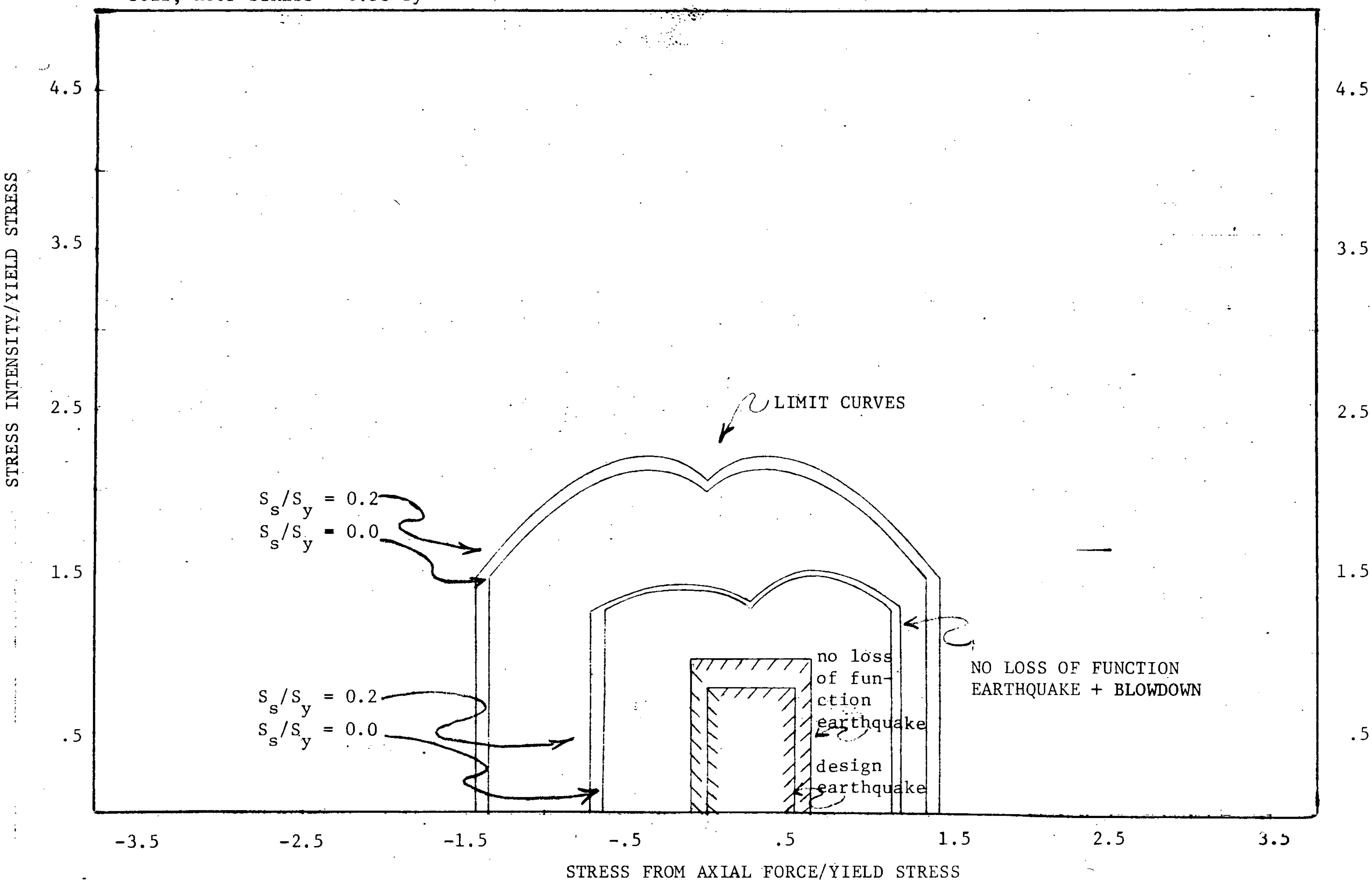


FIGURE A

304 HOOP STRESS = 0.90 Sy

HOLLOW - CIRCULAR

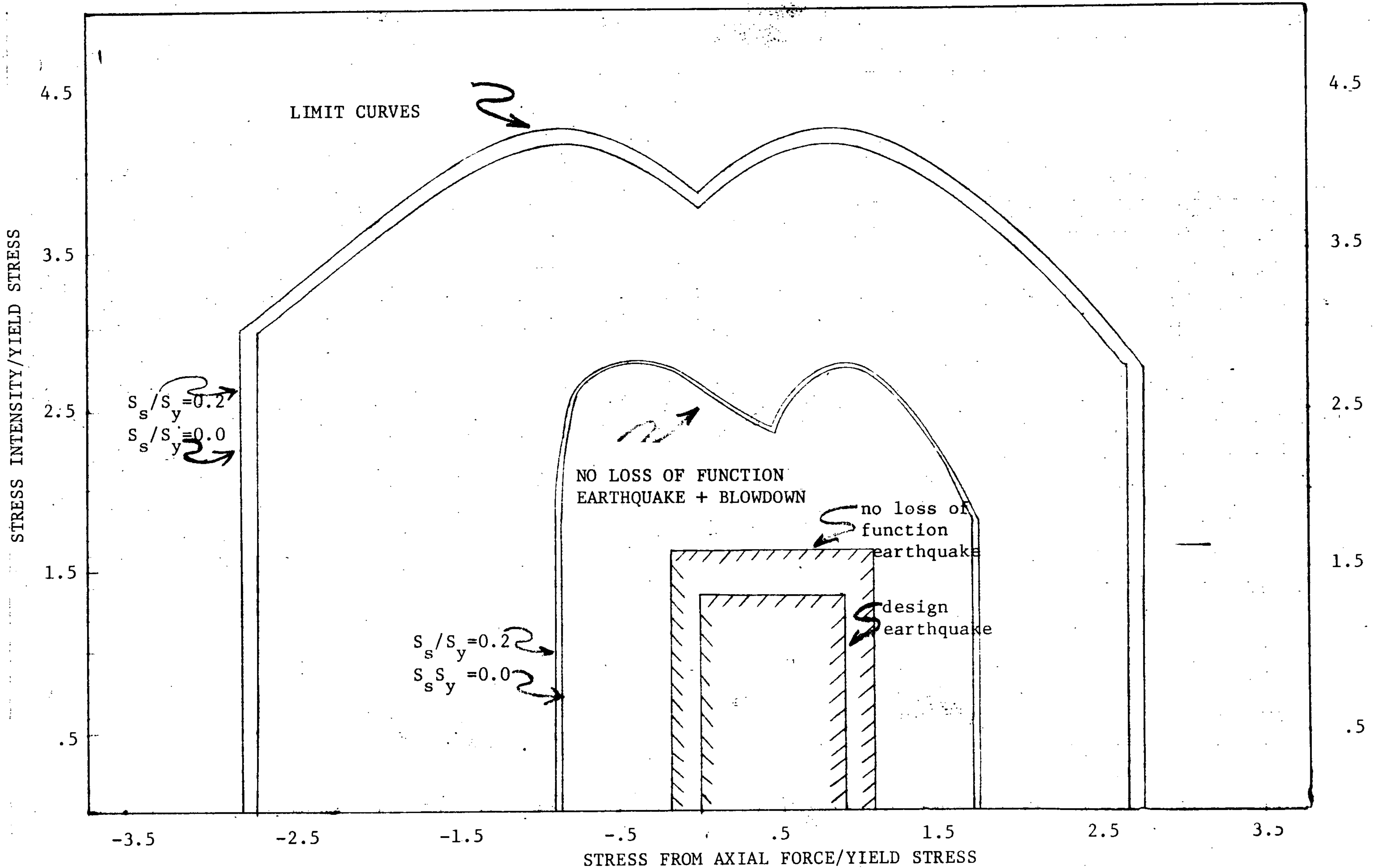


FIGURE B

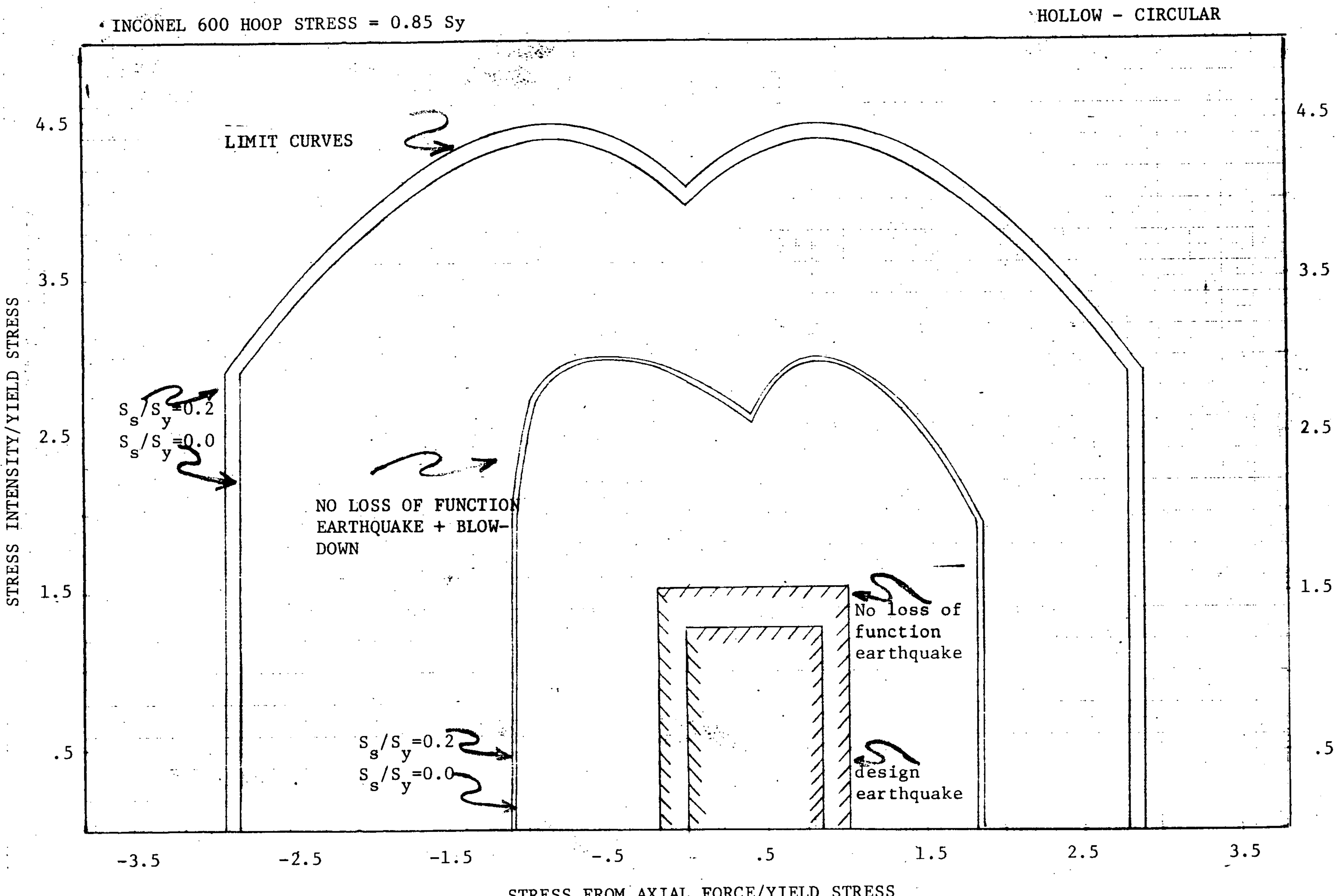

INCONEL 600 HOOP STRESS = 0.85 Sy
HOLLOW - CIRCULAR
LIMIT CURVES
S_s/S_y=0.2
S_s/S_y=0.0
NO LOSS OF FUNCTION EARTHQUAKE + BLOW-DOWN
No loss of function earthquake
S_s/S_y=0.2
S_s/S_y=0.0
design earthquake
STRESS INTENSITY/YIELD STRESS
4.5
3.5
2.5
1.5
.5
-3.5
-2.5
-1.5
-.5
.5
1.5
2.5
3.5
STRESS FROM AXIAL FORCE/YIELD STRESS

FIGURE C

304 HOOP STRESS = 0.90 S_y

RECTANGULAR

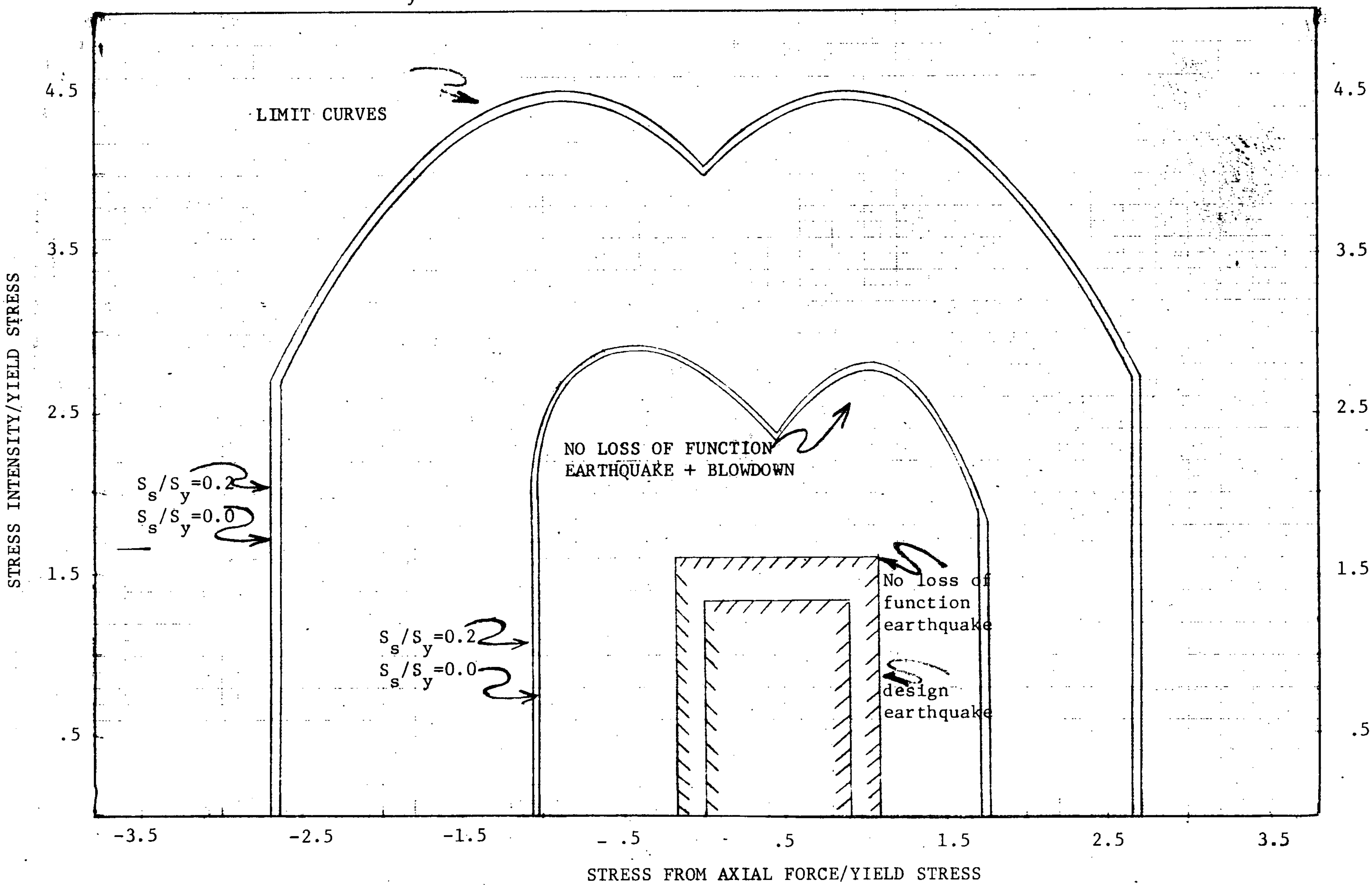
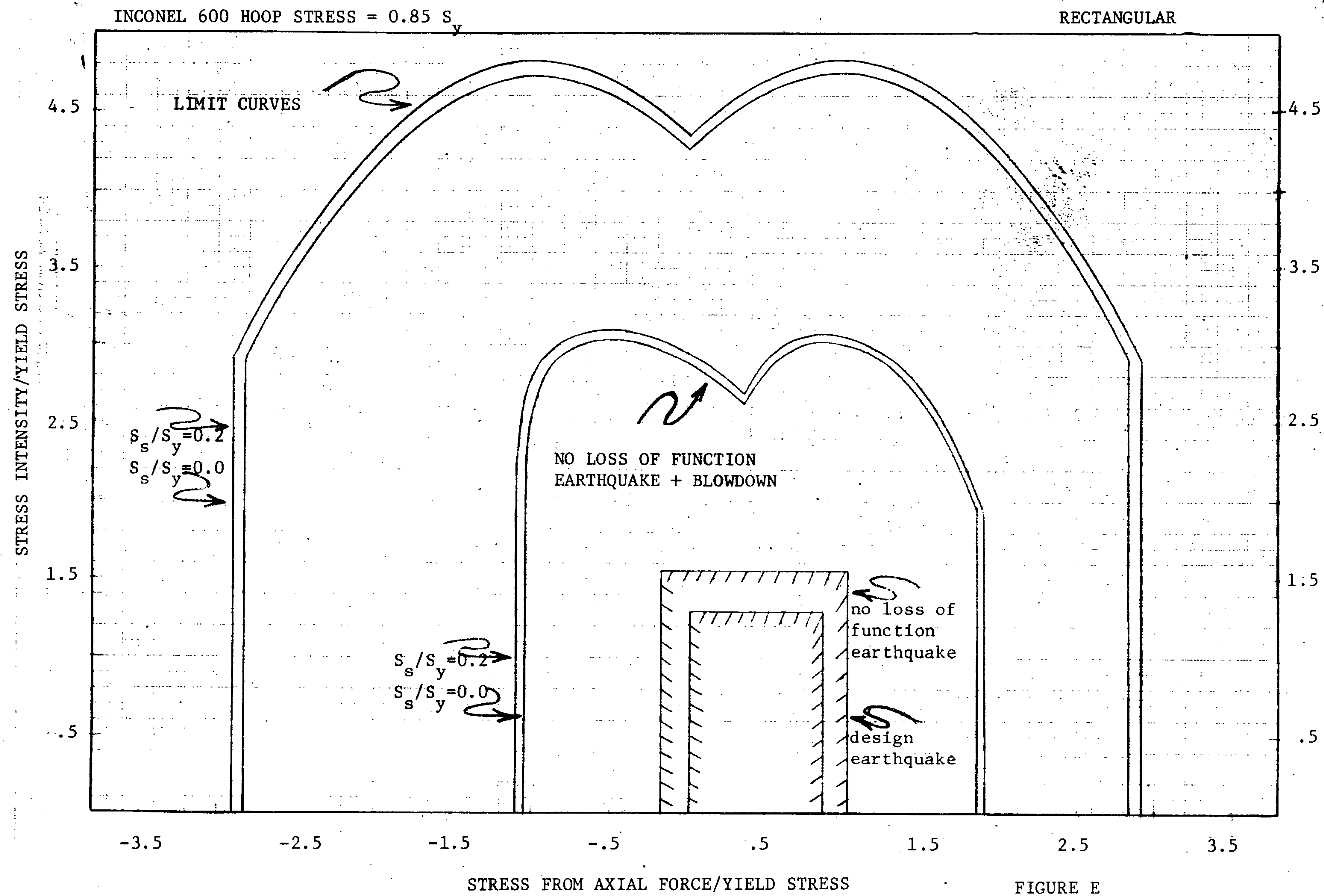


FIGURE D


INCONEL 600 HOOP STRESS = 0.85 S_y
RECTANGULAR
LIMIT CURVES
4.5
3.5
2.5
1.5
.5
S_s/S_y=0.2
S_s/S_y=0.0
NO LOSS OF FUNCTION
EARTHQUAKE + BLOWDOWN
S_s/S_y=0.2
S_s/S_y=0.0
no loss of
function
earthquake
design
earthquake
STRESS INTENSITY/YIELD STRESS
-3.5
-2.5
-1.5
-.5
.5
1.5
2.5
3.5
STRESS FROM AXIAL FORCE/YIELD STRESS
FIGURE E

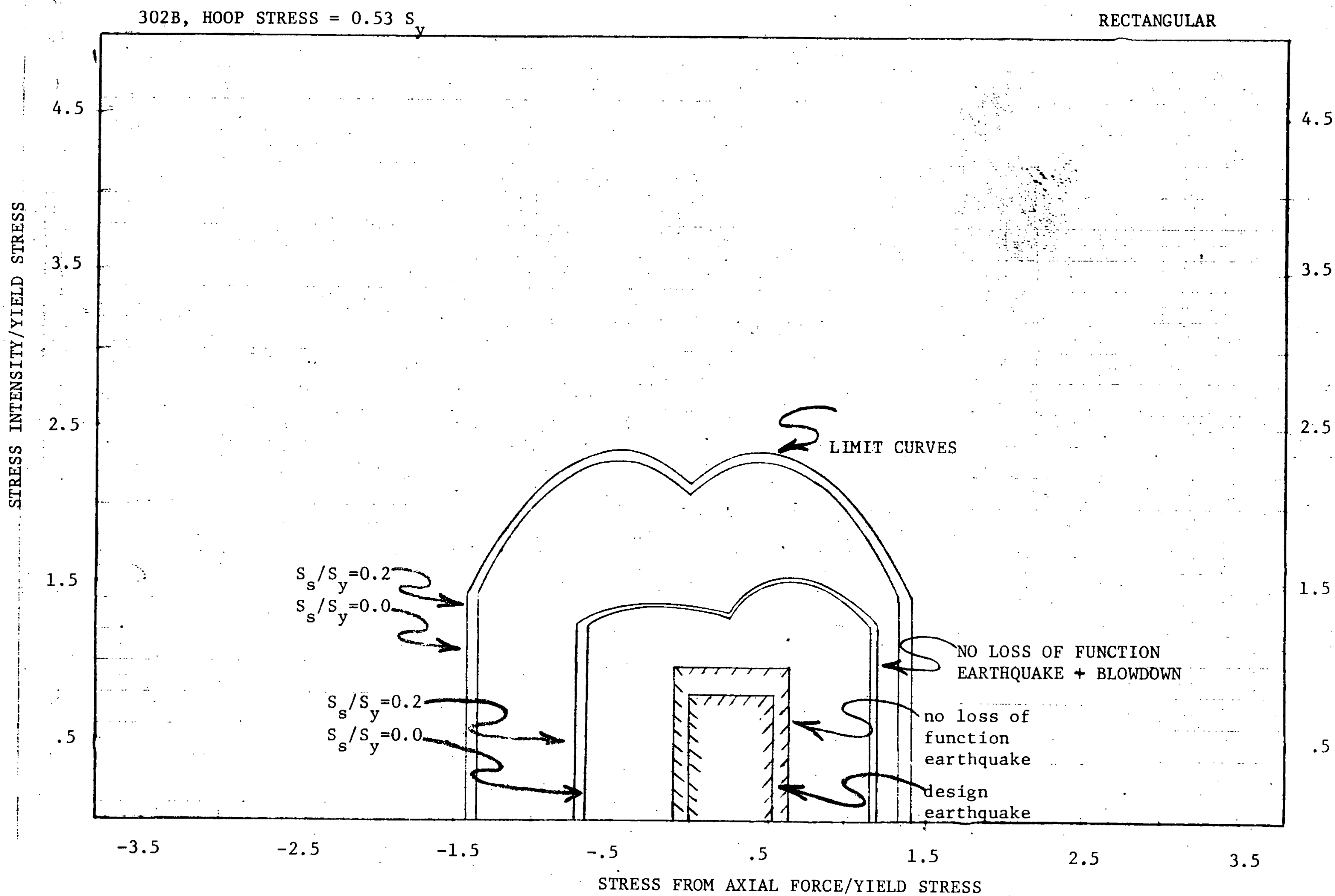

302B, HOOP STRESS = 0.53 S_y
RECTANGULAR
STRESS INTENSITY/YIELD STRESS
4.5
3.5
2.5
1.5
.5
LIMIT CURVES
S_s/S_y=0.2
S_s/S_y=0.0
S_s/S_y=0.2
S_s/S_y=0.0
NO LOSS OF FUNCTION
EARTHQUAKE + BLOWDOWN
no loss of
function
earthquake
design
earthquake
-3.5
-2.5
-1.5
-.5
.5
1.5
2.5
3.5
STRESS FROM AXIAL FORCE/YIELD STRESS

FIGURE F

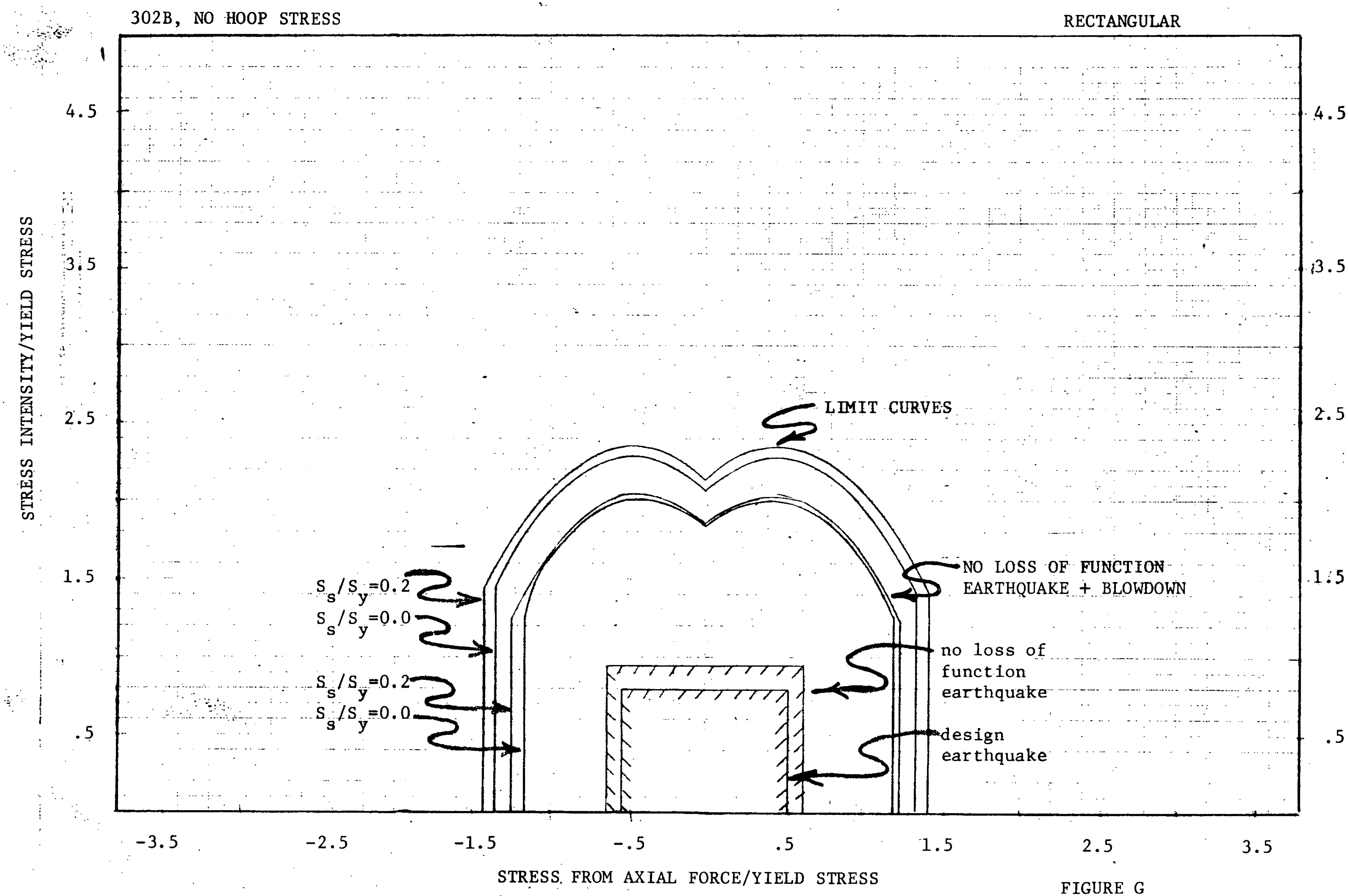

302B, NO HOOP STRESS
RECTANGULAR
STRESS INTENSITY/YIELD STRESS
4.5
3.5
2.5
1.5
.5
LIMIT CURVES
NO LOSS OF FUNCTION
EARTHQUAKE + BLOWDOWN
S_s/S_y=0.2
S_s/S_y=0.0
S_s/S_y=0.2
S_s/S_y=0.0
no loss of
function
earthquake
design
earthquake
-3.5
-2.5
-1.5
-.5
.5
1.5
2.5
3.5
STRESS FROM AXIAL FORCE/YIELD STRESS

FIGURE G

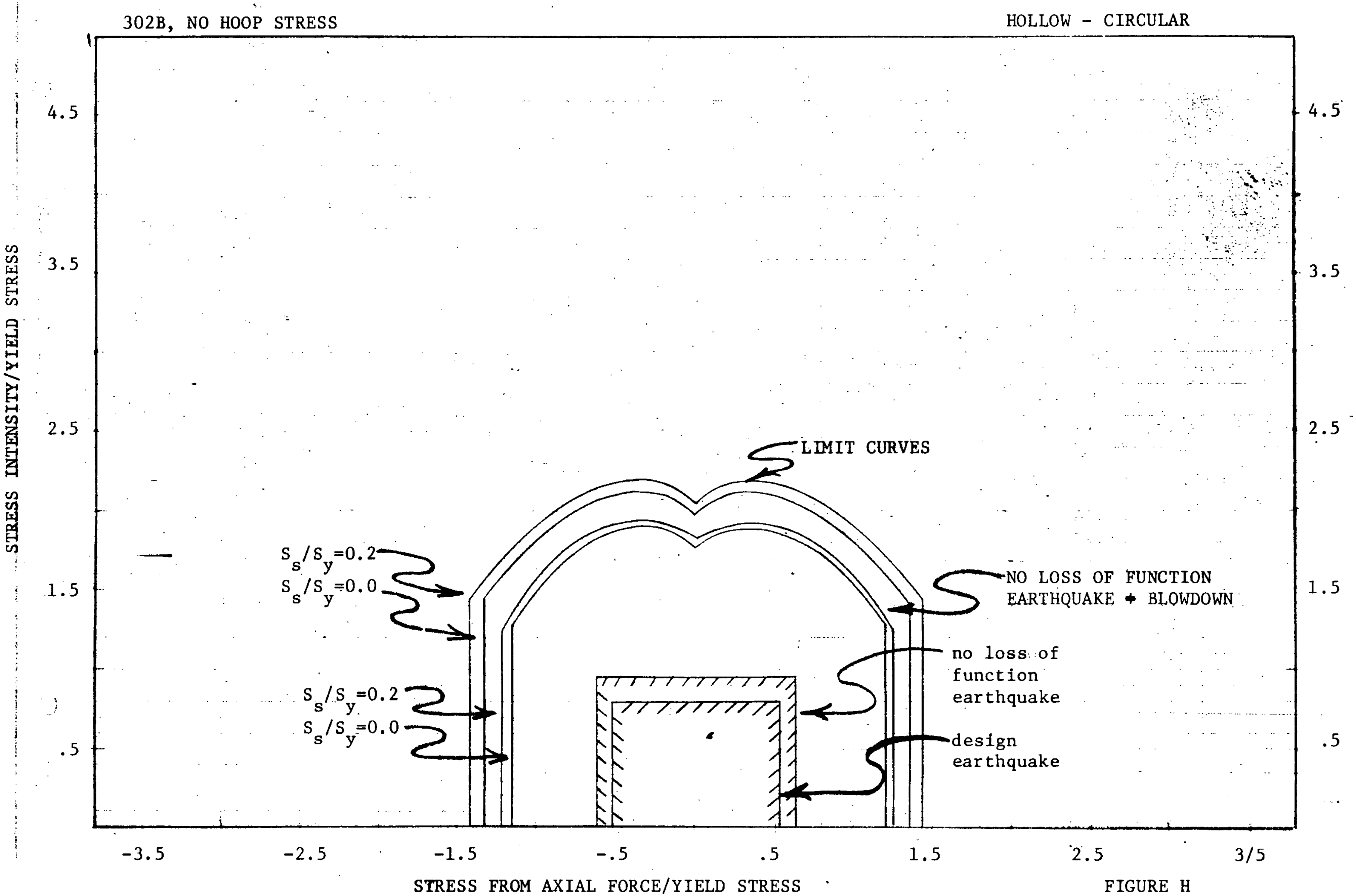

302B, NO HOOP STRESS
HOLLOW - CIRCULAR
STRESS INTENSITY/YIELD STRESS
4.5
3.5
2.5
1.5
.5
LIMIT CURVES
S_s/S_y=0.2
S_s/S_y=0.0
S_s/S_y=0.2
S_s/S_y=0.0
NO LOSS OF FUNCTION
EARTHQUAKE + BLOWDOWN
no loss of
function
earthquake
design
earthquake
-3.5
-2.5
-1.5
-.5
.5
1.5
2.5
3/5
STRESS FROM AXIAL FORCE/YIELD STRESS

FIGURE H

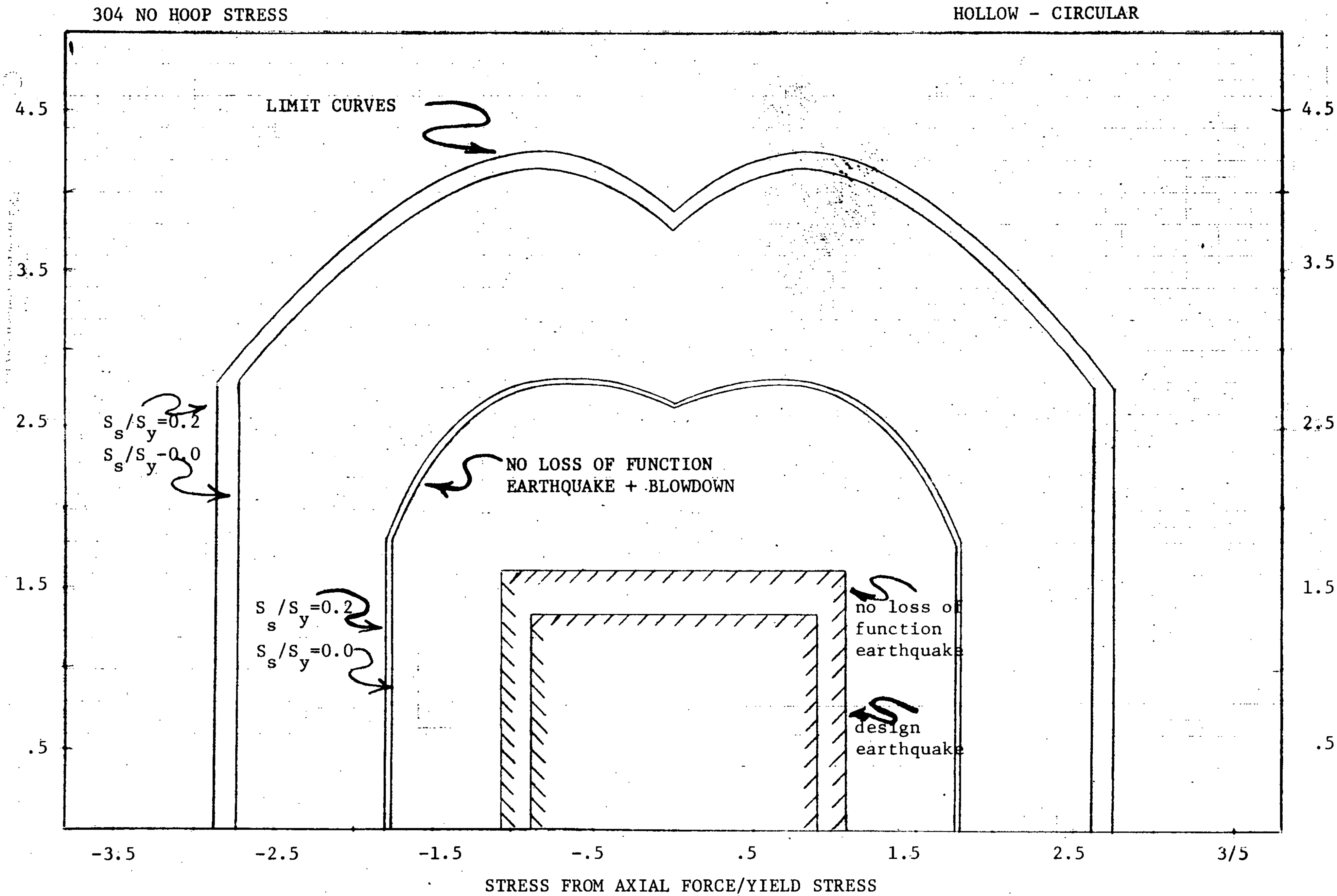

304 NO HOOP STRESS
HOLLOW - CIRCULAR
LIMIT CURVES
S_s/S_y=0.2
S_s/S_y-0.0
NO LOSS OF FUNCTION
EARTHQUAKE + BLOWDOWN
S_s/S_y=0.2
S_s/S_y=0.0
no loss of
function
earthquake
design
earthquake
4.5
3.5
2.5
1.5
.5
-3.5
-2.5
-1.5
-.5
.5
1.5
2.5
3/5
STRESS FROM AXIAL FORCE/YIELD STRESS

FIGURE I

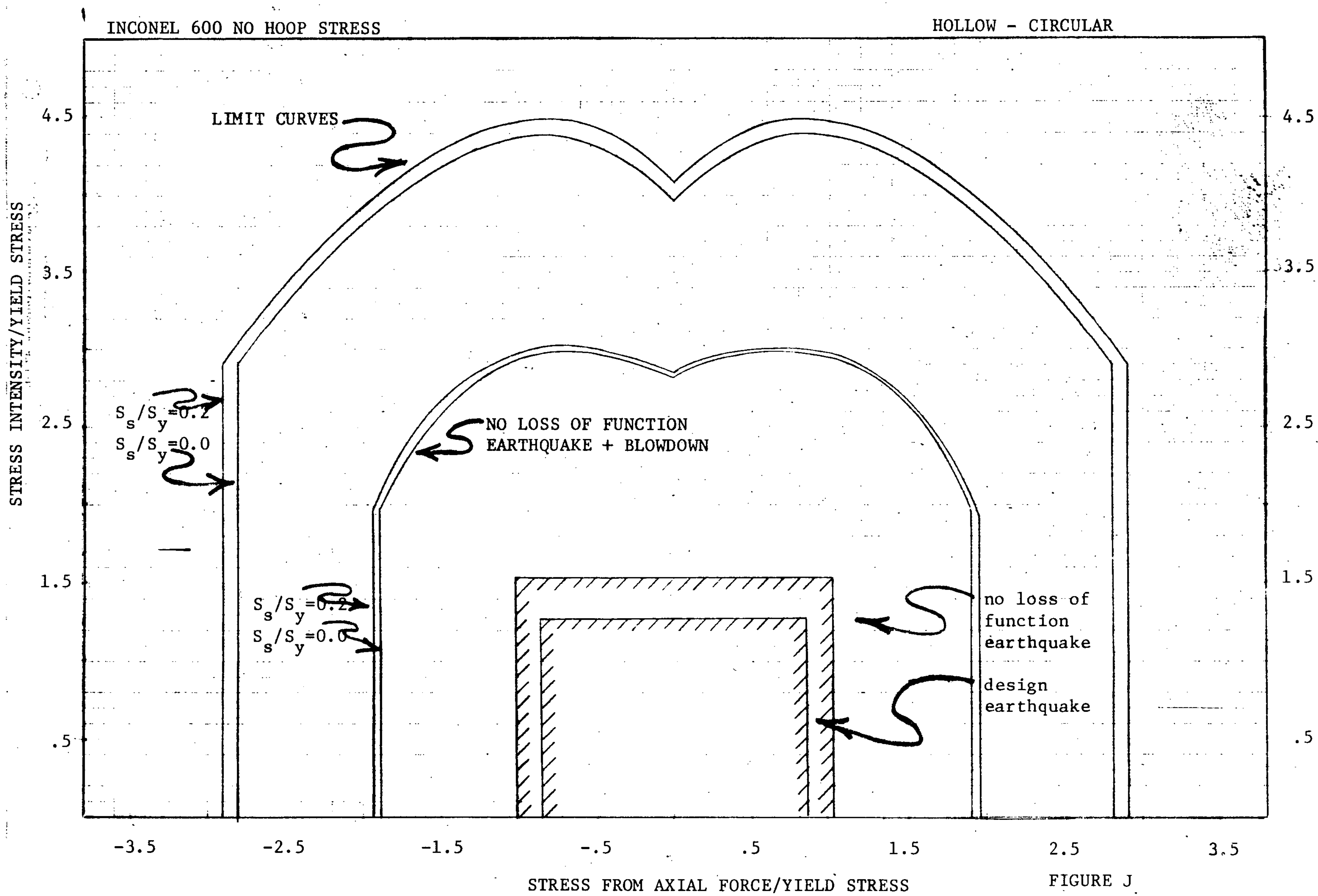

INCONEL 600 NO HOOP STRESS
HOLLOW - CIRCULAR
LIMIT CURVES
S_s/S_y=0.2
S_s/S_y=0.0
NO LOSS OF FUNCTION
EARTHQUAKE + BLOWDOWN
S_s/S_y=0.2
S_s/S_y=0.0
no loss of
function
earthquake
design
earthquake
STRESS INTENSITY/YIELD STRESS
4.5
3.5
2.5
1.5
.5
-3.5
-2.5
-1.5
-.5
.5
1.5
2.5
3.5
STRESS FROM AXIAL FORCE/YIELD STRESS

FIGURE J

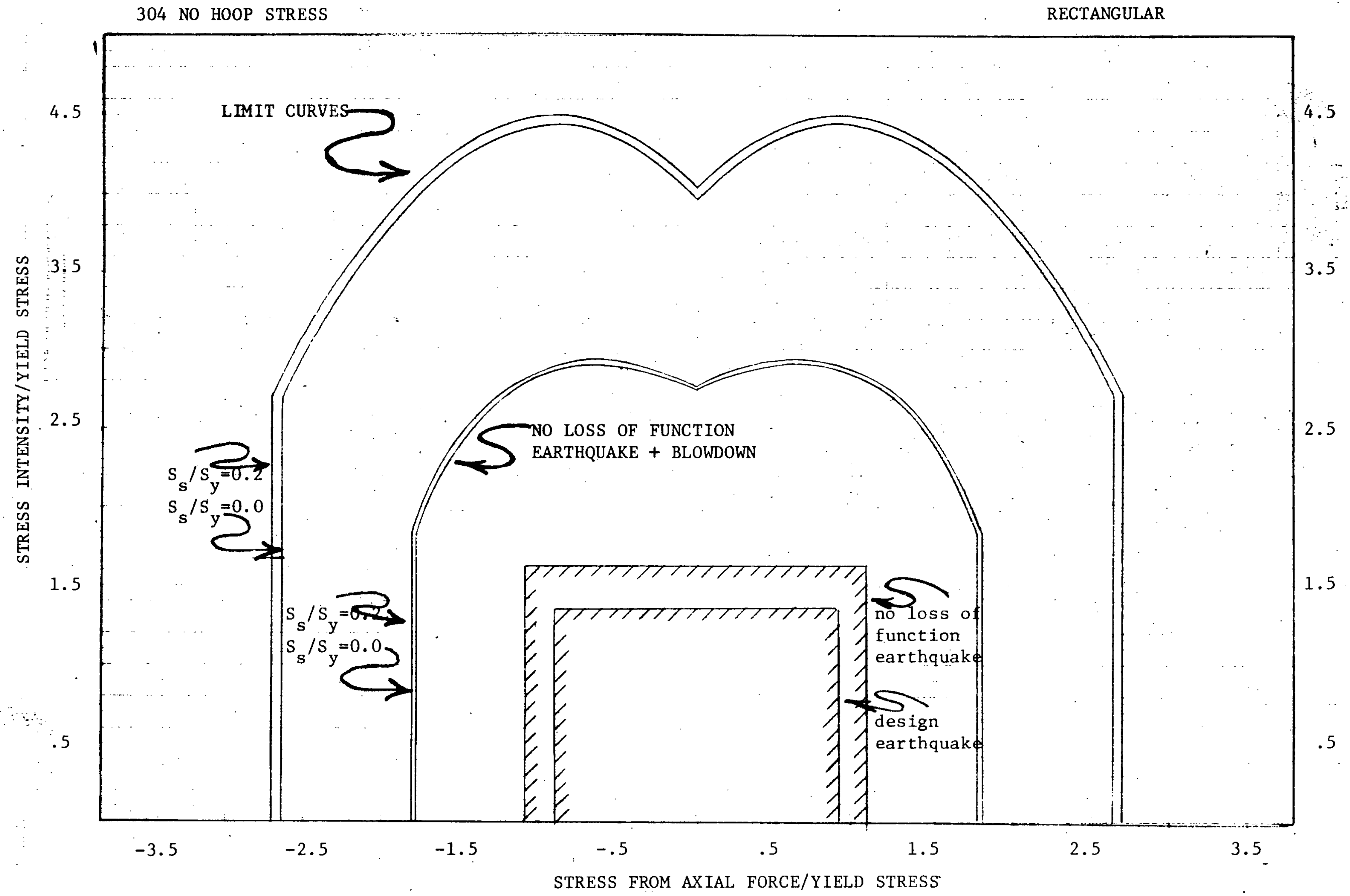
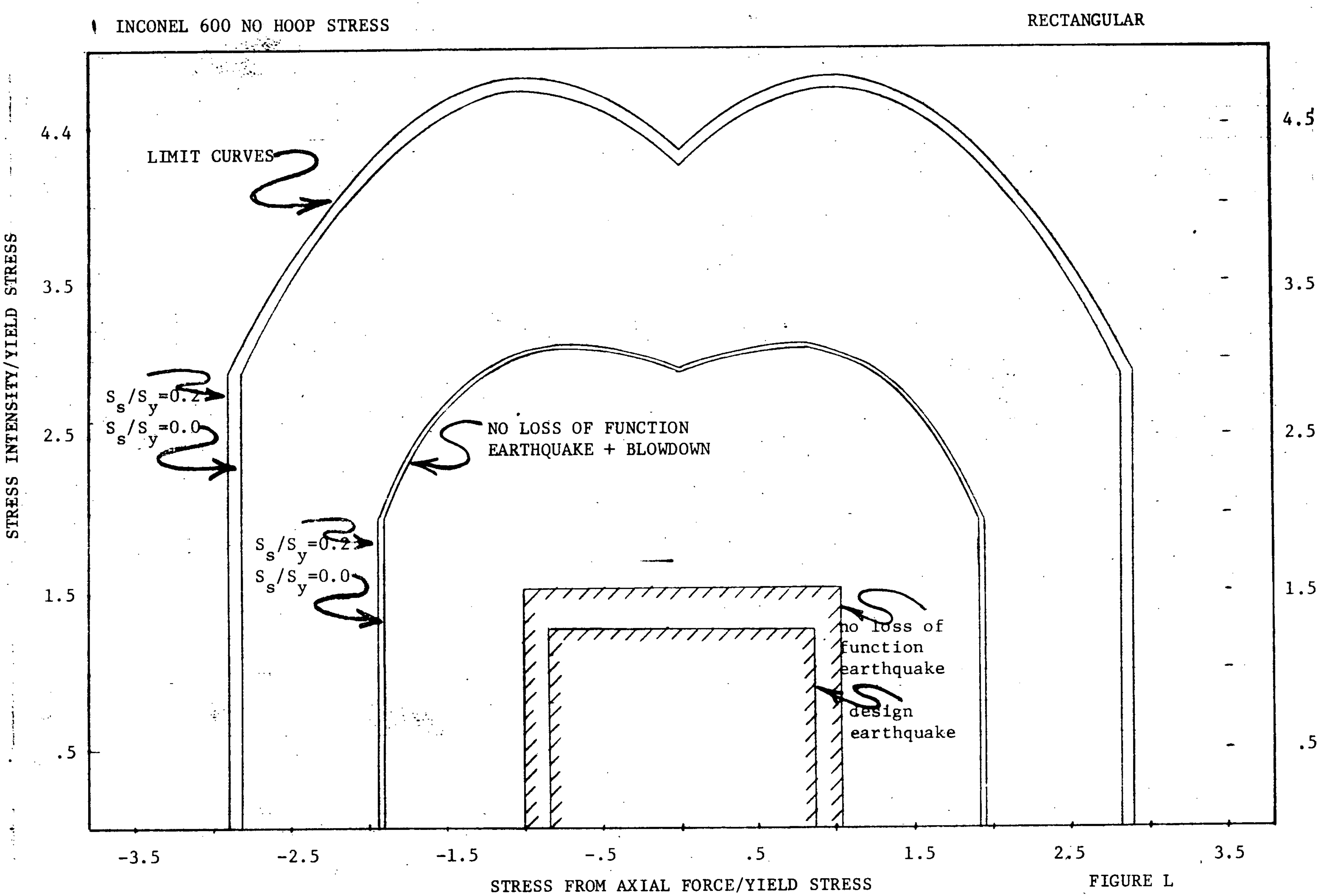

304 NO HOOP STRESS
RECTANGULAR
LIMIT CURVES
NO LOSS OF FUNCTION
EARTHQUAKE + BLOWDOWN
S_s/S_y=0.2
S_s/S_y=0.0
S_s/S_y=0.2
S_s/S_y=0.0
no loss of
function
earthquake
design
earthquake
STRESS INTENSITY/YIELD STRESS
4.5
3.5
2.5
1.5
.5
4.5
3.5
2.5
1.5
.5
-3.5
-2.5
-1.5
-.5
.5
1.5
2.5
3.5
STRESS FROM AXIAL FORCE/YIELD STRESS

FIGURE K



INCONEL 600 NO HOOP STRESS
RECTANGULAR
LIMIT CURVES
S_s/S_y=0.2
S_s/S_y=0.0
NO LOSS OF FUNCTION
EARTHQUAKE + BLOWDOWN
S_s/S_y=0.2
S_s/S_y=0.0
no loss of
function
earthquake
design
earthquake
STRESS INTENSITY/YIELD STRESS
4.4
3.5
2.5
1.5
.5
4.5
3.5
2.5
1.5
.5
-3.5
-2.5
-1.5
-.5
.5
1.5
2.5
3.5
STRESS FROM AXIAL FORCE/YIELD STRESS

FIGURE L

DOSE CALCULATIONS

Steam-Generator Tube Rupture

The analysis of the steam generator tube rupture accident is presented in the Supplement 1 to the PSAR, Item 2, Question 2 (2-5).

Further evaluation of the actual doses prior to diverting the air ejector to the containment has been performed. The transport time between the steam generator and air ejector plus the time to actuate the valve to divert the air ejector exhaust to the containment is approximately one minute. The corresponding whole body dose for one minute is 18 mrem. After approximately 30 minutes the pressure in the primary systems is brought down to the secondary safety valve setting such that the faulty steam generator can be isolated and further release of activity is terminated. If it is assumed that the the activity has not been diverted during this thirty minute period the offsite whole body dose is 0.5 rem.

STEAM LINE BREAK

The revised analysis of the steam line break is given in this supplement as revised pages to be inserted into Supplement 1, Item 16, 16 (E-3.5).

Dose calculations have been performed for the following steam line breaks.

A. Break occurs outside the containment

The following assumptions were used:

1) Leakage from primary to secondary is 10 GPM into the faulted steam generator.
2) Four hours for cooldown by steam generators prior to isolation
3) Activity equivalent to 1% fuel defects
4) No fuel clad damage occurs
5) The steam generator in the affected loop boils dry.

For this accident the inhalation dose at the site boundary is less than 10 rem during the first two hours and the corresponding whole body dose is less than 0.1 rem.

B. Break occurs inside the containment

The following assumptions were used:

1) Leakage from primary to secondary is 10 GPM into the remaining intact steam generators.
2) Four hours for cooldown by steam generators prior to isolation
3) Fuel clad damage up to 10% of the rods
4) Activity equivalent to 10% of the average gap activity in the core

For this accident the two-hour inhalation dose at the site boundary is less than 1 rem during the first two hours, and the corresponding whole body dose is less than 0.4 rem.

The above results are all well within the guidelines of 10 CFR 100.

ROD EJECTION ACCIDENT

For the rod ejection accident described in the PSAR the loss of clad integrity is estimated to affect about 10% of the fuel rods.

A 10 gpm leak from the primary to secondary is assumed. Isolation of all steam generators is assumed after 4 hours, after which the residual heat removal system is assumed to be operational. Assuming an activity equivalent to 10% of the average gap activity in the core, a whole-body dose of less than 0.4 rem is calculated for the two hour dose at the site boundary and the corresponding two-hour inhalation dose (thyroid) at the site boundary is less than 1 rem based on minimim dilution effects of the safety injection water. Whole-body and inhalation doses from containment leakage are negligible by comparison with the above values.

The above results are all well within the guidelines of 10 CFR 100.

POWER SUPPLIES TO INDIAN POINT NO. 3

There are three separate systems which are capable of supplying emergency electrical power to the Indian Point No. 3 nuclear unit. Each of these three systems is in itself a redundant system having at least two alternate sources of supply. The three separate systems of power supply are as follows:

1. The 138 Kv overhead supply from Buchanan Substation. This system consists of two separate 138 Kv overhead feeders, one of which goes from Buchanan Substation to Indian Point No. 2 and the second which is connected to Indian Point No. 3. The two 138 Kv feeders are on different tower systems and come from different positions on the Buchanan bus which are separated by at least two circuit breakers. The 138 Kv feeder that goes to Unit No. 2 is connected underground through two circuit breakers to the Indian Point No. 3 start-up transformer. The 138 Kv feeder to Indian Point No. 3 is connected to the same start-up transformer. There is no credible fault on one feeder that would cause the outage of the second feeder, including failure of any tower. The only common point for the two feeders is the connection to the 138 Kv transformer at Unit No. 3.

2. The second independent emergency power supply to Indian Point No. 3 is the 6.9 Kv connection to auxiliary bus sections No. 5 and 6. This supply is automatically connected on loss of the 138 Kv supply. This 6.9 Kv underground tie can be fed from any one of three separate sources. It can be connected to the gas turbine generator located in a rock cut adjacent to Unit No. 1. It can be supplied by a 13 Kv underground feeder from the Buchanan Substation or it can be supplied from the 6.9 Kv auxiliary bus of Unit No. 2 which in turn can be supplied either from Unit No. 2 generator or a 138 Kv feeder from Buchanan Substation. It should be noted that the 6.9 Kv supply is completely independent of the 138 Kv supply and is even equipped with its own prime mover - the 21 Mva gas turbine.

3. The third independent emergency power system is the diesel generator system which operates at 480 volts. The diesels can be connected directly to the station 480 volt busses and are, therefore, completely independent of either the 6.9 Kv supply or the 138 Kv supply to Indian Point No. 3. Only two of the three diesel generators are necessary to supply the emergency station load. Since these diesels supply separate 480 volt busses they are electrically independent and a fault on one will only affect that particular generator.

The above description shows why it is considered extremely unlikely and improbable that all emergency power could be lost. We know of no situation that could lead to the complete loss of all three of these systems. The 138 Kv feeders, the gas turbine generator and the diesel building are separated by substantial distances which preclude any accident affecting all three.

The question has arisen as to whether a tornado could conceivably disable all electrical power supplies to Unit No. 3. We do not believe that this is credible. To disable all electrical supplies would require that the tornado cut both 138 Kv feeders, then damage the gas turbine located in a deep rock cut between tall buildings and then proceed in such a direction as to then disable two of the three diesel generators located on the other side of Unit No. 3. This requires a tortuous path with several directional changes at precisely the right moment. The likelihood of such a path being followed by a tornado moving randomly is considered extremely improbable.

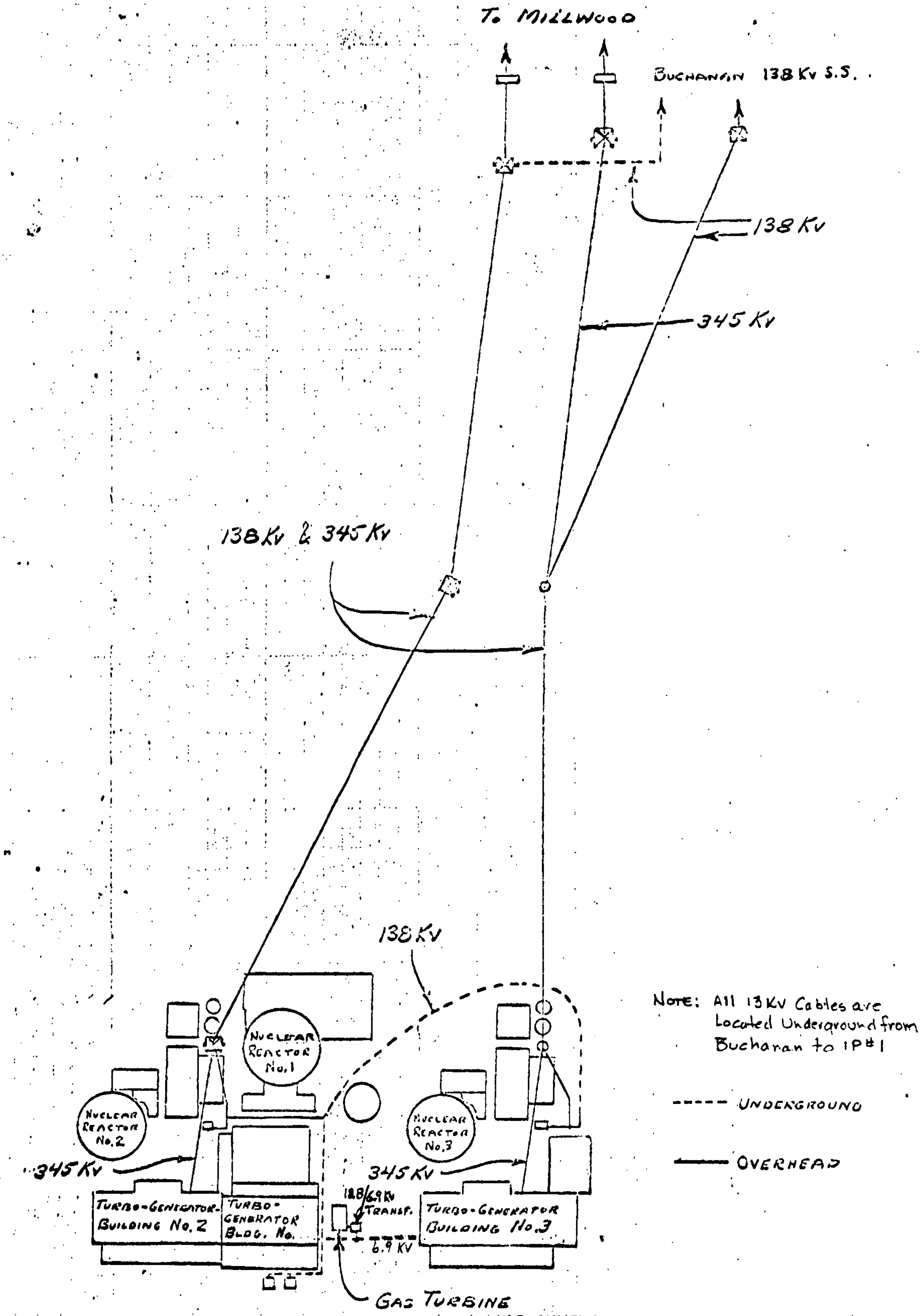


FIGURE A
SUPPLEMENT 5

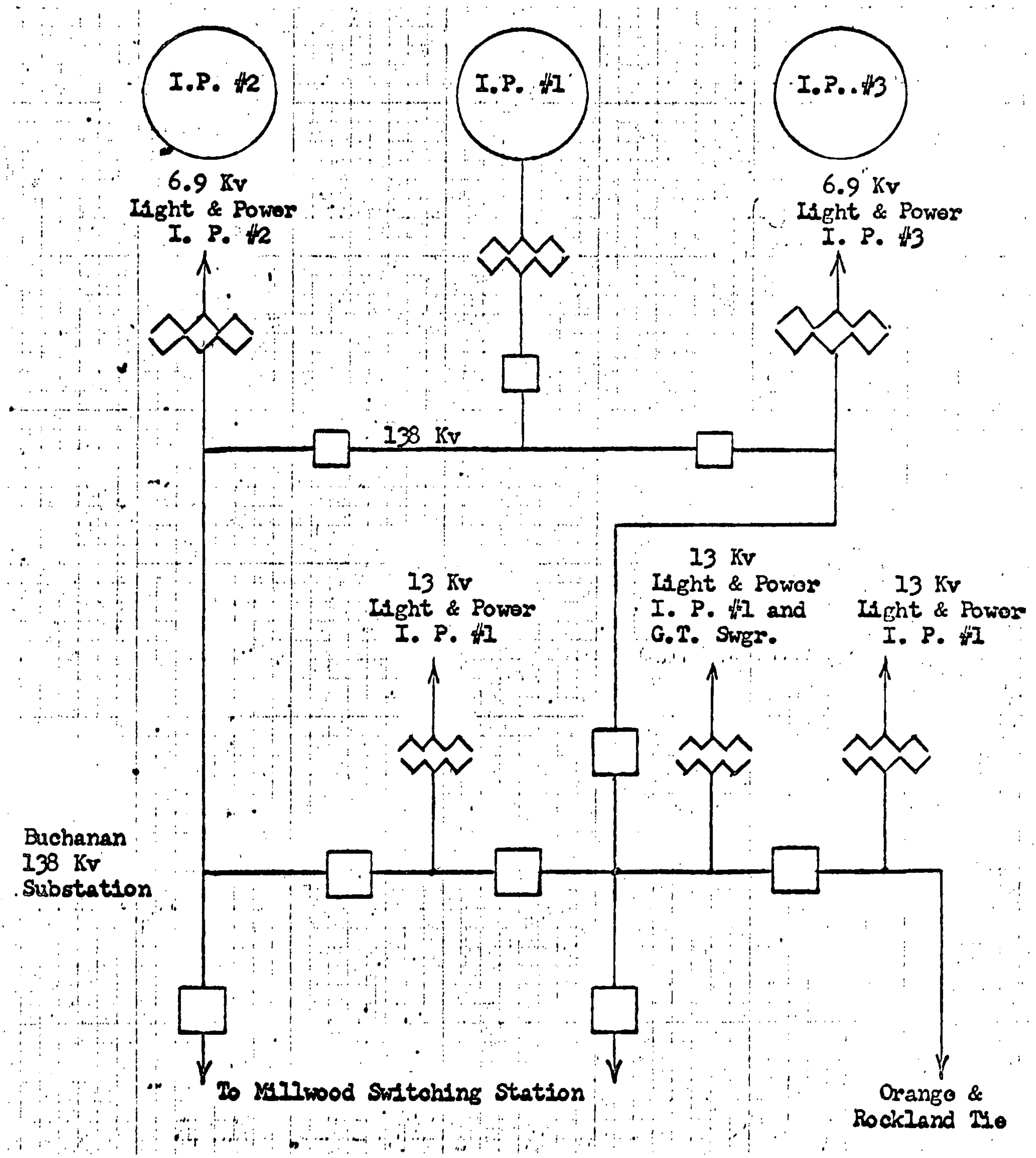


FIGURE B
SUPPLEMENT 5

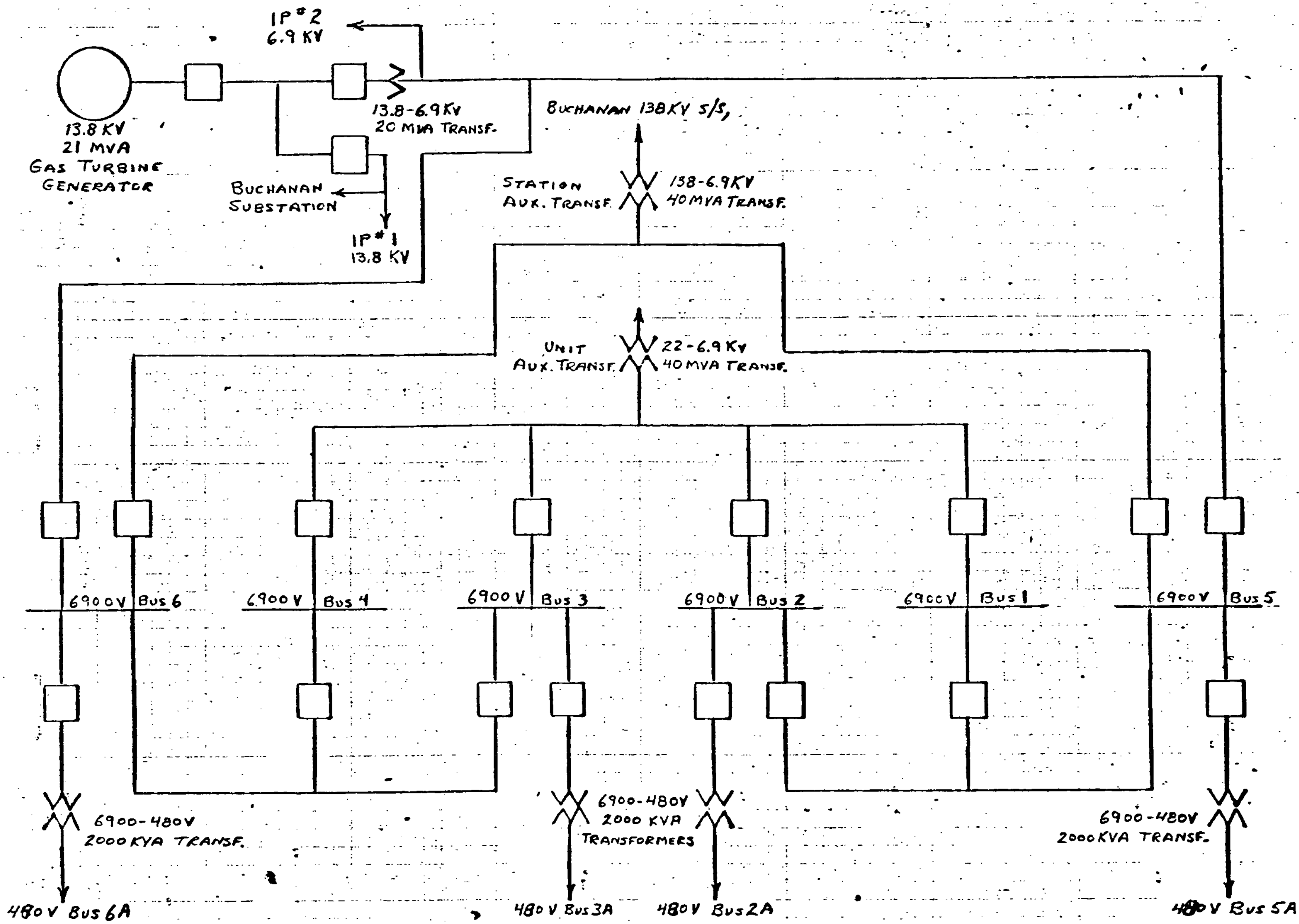


FIGURE C
SUPPLEMENT 5

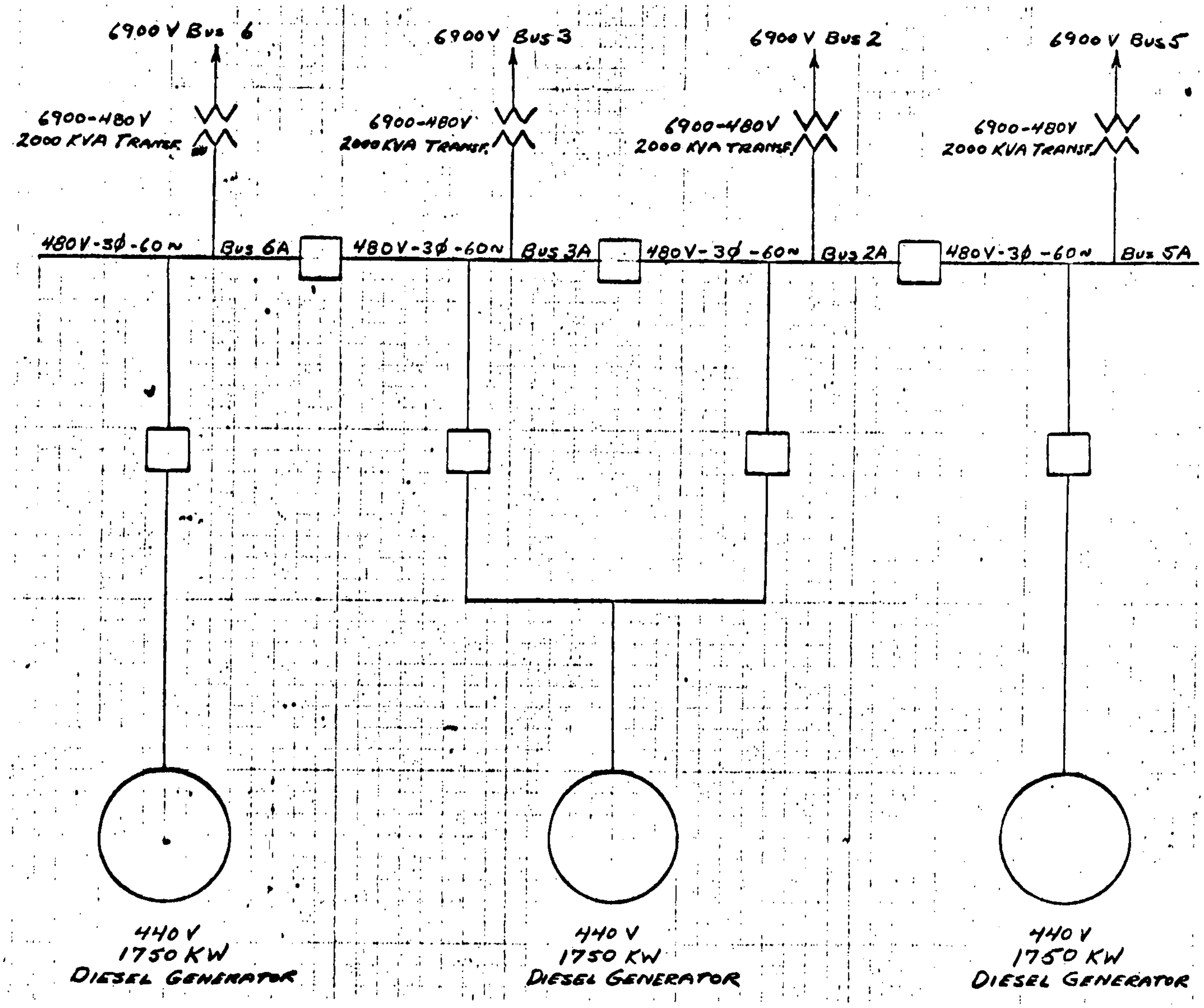


FIGURE D
SUPPLEMENT 5

SPENT FUEL PIT TORNADO EVALUATION

In addition to the tornado criteria specified in Supplement 1, Item 6, page 3 concerning the spent fuel pit, provisions will be made in the design of the pit such that a cover can be added at a later date should it be concluded that such protection is required. The cover would be designed so that it could not be a source of a hazardous missile, would withstand required pressure differentials and would absorb the necessary energy of missile such that the missile could not cause unacceptable fuel damage.

PRIMARY TO SECONDARY LEAK RATE DETERMINATION

Primary to secondary leak rate determination is accomplished by measuring the sodium 24 content, by gamma spectrometry, of the secondary blowdown water and comparing it with primary water sodium 24 content. Calculations used in the analysis take into account the blowdown rate and the delay time in the steam generator. The smallest primary to secondary leak rate measurable (based on Indian Point Unit No. 1 experience) is approximately 1 lb/hr equal to 1/500 gpm.

STEAM BREAK ANALYSIS

The steam break analysis presented in Supplement 1 to the PSAR has been revised and is included with this supplement as revised pages to the text of Supplement 1, Question 16 (E-3.5).

LOSS-OF-COOLANT DESIGN CRITERIA

The loss-of-coolant design criteria have been revised and the revisions have been included in this supplement as revised pages to Supplement 1 for Item 1, pages 57, 58, 59 and 60, and Item 16, A-16 (E-5.0), pages 1 and 2.

MINIMUM DNBR CRITERIA

Correction to the text concerning statements of the minimum DNBR are included as page revisions for Supplement 1 in this Supplement. The revisions are for Item 1, pages 37 and 38, and for Item 10, 10 (A-4.0).

CRITERIA FOR TESTING AIR CLEANUP SYSTEMS

The general design criteria in Item 1 of Supplement 1 to the PSAR have been revised in this supplement and are included as revised pages to Supplement 1. The revisions are on Pages 77 through 82 of Item 1.

INSTRUMENTATION SYSTEMS

Revisions have been made to Item 2, Question 2 (1-14) in Supplement 1 to the PSAR and are included in this supplement as revised pages for Supplement 1. The revised pages of Item 2 are 2 (1-14) through 2 (1-14), page 4.

LOCKED OPEN VALVES IN THE CONTAINMENT

Revisions to Item 2, Question 2 (2-8) have bee made and are included as page revisions to Supplement 1 to the PSAR, pages 2 (2-8) and 2 (2-9) page 2. These pages are included with this Supplement.