LIMERICK GENERATING STATION

UNITS 1 & 2

PSAR PAGE REVISIONS

The following PSAR pages, tables and figures are to be inserted in your copy of the PSAR. These pages are either replacements or new pages as indicated below:

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Page 3.6-4	Page 3.6-4	dated July 30, 1971
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Page 5.2-6	Page 5.2-6	dated July 30, 1971
Fig. 5.2.5	Fig. 5.2.5	dated July 30, 1971
Page 6-i	Page 6-i	dated July 30, 1971
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	(New) Page 6.7-1	dated July 30, 1971
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Fig. 7.7.1c	Fig. 7.7.1c	dated July 30, 1971
Fig. 7.7.1e	Fig. 7.7.1e	dated July 30, 1971

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Fig. 12.2.2	Fig. 12.2.2	dated July 30, 1971
Fig. 12.2.3	Fig. 12.2.3	dated July 30, 1971
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	Volume 5	
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the fuel depletion process with spatial neutron flux and energy distributions typical of reactor operating conditions. At selected burnup intervals, the nuclide concentrations are used to recalculate revised flux and material weighted cross sections with the lattice model and these are again recycled through two dimensional diffusion theory.

A large capacity, nodal, three dimensional boiling water reactor simulation code which provides for representation and calculation of spatially varying voids, control rods, burnable poisons and other variables is used to compute three-dimensional power distributions, exposure and reactor thermal-hydraulic characteristics at the beginning of core life and as burnup progresses. In addition, it can serve to determine control rod strategy through life, power response to changes in core flow and to calculate assembly as well as core exposure.

3.6.5.2 Reactivity Control

The excess reactivity designed into the initial core is controlled with a control rod system supplemented by gadolinia-urania burnable poison rods. The core is designed to permit the energy extraction of 19,000 MWD/T of uranium averaged over the initial core load. This exposure can be achieved with the reactor operating at full power at the end of each cycle. The average fuel enrichment for the initial core load is chosen to provide excess reactivity in the fuel assemblies sufficient to overcome the neutron losses due to core neutron leakage, moderator heating and boiling, fuel temperature rise, and equilibrium xenon and samarium poisoning; also included is an allowance for fuel depletion. Following the initial cycle, more new fuel may be added to achieve annual refueling during the desired refueling month. During fuel burnup, control rods are used, in part to counteract the power distribution effect of steam voids indicated by the in-core flux monitors. In combination, the control rod and void distributions can be used to flatten gross power beyond that which is possible in a non-boiling core. The design provides considerable flexibility in the control of gross power distribution. This permits regulation of fuel burnup and isotopic composition throughout the core to the extent necessary to counteract the effect of voids on axial power distribution at the end of a fuel cycle, when few control rods remain in the core.

The control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during the equilibrium fuel cycle operation. The initial core loading, however, has an excess reactivity somewhat higher than that of the equilibrium core. Thus, the design basis for the initial burnable poison loading is that it shall compensate the reactivity difference between the control rod system

3.7 THERMAL AND HYDRAULIC DESIGN

3.7.1 Power Generation Objective

The objective of the thermal and hydraulic design of the core is to achieve power operation of the fuel over the life of the core without sustaining fuel damage.

3.7.2 Power eration Design Bases

1. The thermal hydraulic characteristics of the core shall provide the ability to achieve rated core power output throughout the design lifetime of the fuel without sustaining fuel damage.

2. The thermal hydraulic characteristics of the core shall provide thermal margin which, in conjunction with the plant equipment characteristics, nuclear instrumentation, and the reactor protection system assures that no fuel damage will occur during normal operation or abnormal operational transients caused by reasonably expected single operator error or equipment malfunction.

3.7.3 Safety Design Bases

1. The thermal hydraulic design of the core shall establish limits for use in setting devices of the Nuclear Safety Systems so that no fuel damage occurs as a result of abnormal operational transients. (See Section 14,"Plant Safety Analysis".)

2. The thermal hydraulic design of the core shall establish a thermal hydraulic safety limit for use in evaluating the safety margin relating to the public safety consequences of fuel barrier failure.

3.7.4 Thermal and Hydraulic Limits

3.7.4.1 Steady State Limits

For purposes of maintaining adequate margin during normal steady state operation, the minimum critical heat flux ratio (MCHFR) is maintained in excess of 1.9 relative to the design correlation limit lines (ref.1); the maximum linear heat generation rate is maintained below 18.5 kilowatts per foot. Operating power and peaking factors are not specified; these parameters are determined subject to a number of constraints, including the thermal limits noted previously. To accommodate uncertainties and to assure that no fuel damage results even during the worst anticipated transient condition at any time in life, the core and fuel design basis for steady state operation (i.e., MCHFR 1.9 and LHGR 18.5 kw/ft) has been

f. TIP System guide tubes are provided with an isolation valve which closes automatically upon receipt of proper signals and after the TIP cable and fission chamber have been retracted. In series with this isolation valve, an additional or backup isolation shear valve is included to assure integrity of the containment in the unlikely event that either the isolation valve should fail to close or the chamber drive cable should fail to retract if it should be extended in the guide tube during the time that containment isolation is required.

g. Isolation valves are either actuated by various signals or are remote manually operated, as appropriate. Refer to Table 7.3.1, Lines Penetrating the Primary Containment.

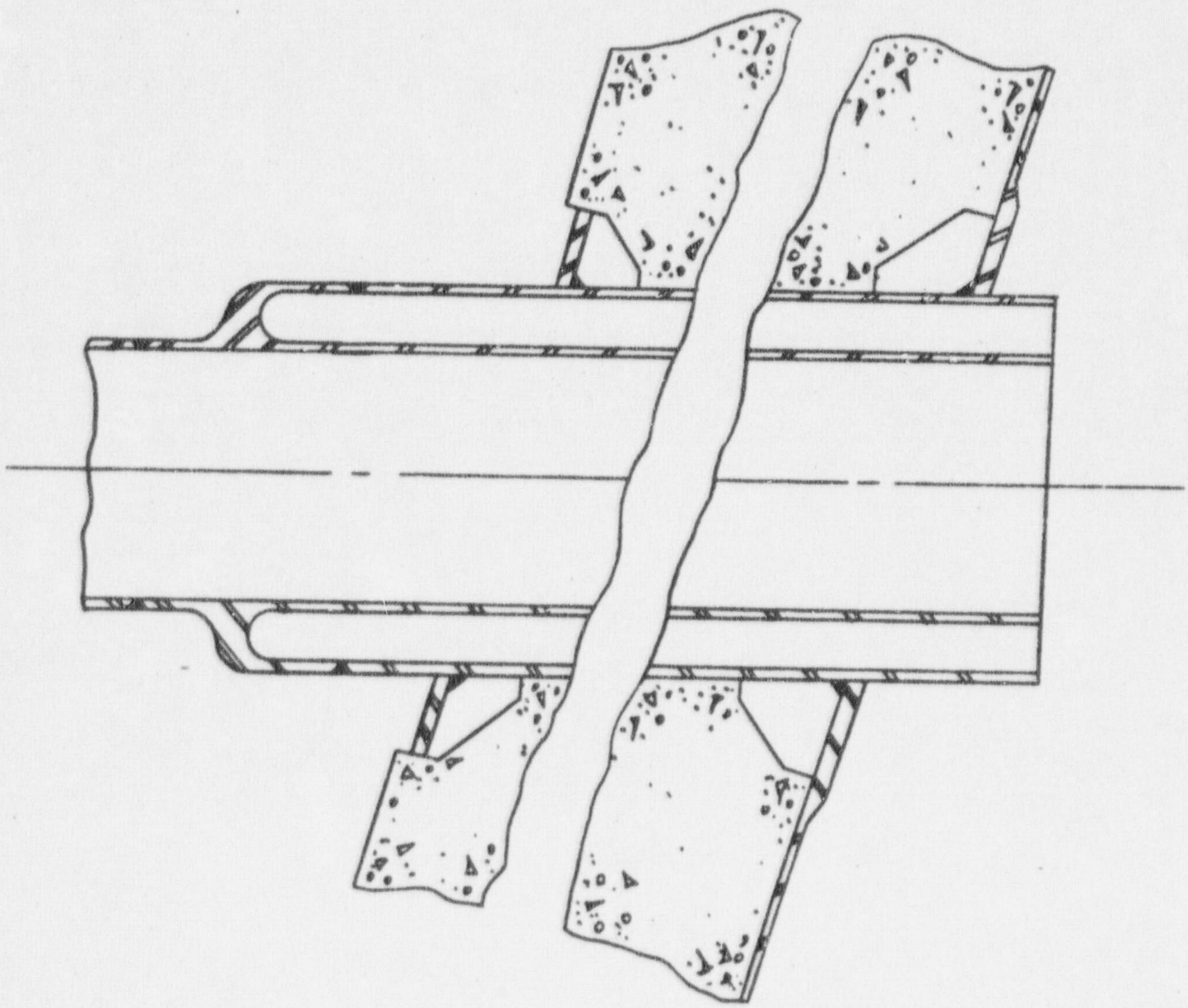
5.2.3.5 Primary Containment Venting and Vacuum Relief System

The primary containment is vented as required to eliminate pressure fluctuations caused by temperature changes during various operating modes. This is accomplished through ventilation purge connections which are normally closed while the reactor is at a temperature greater than 212°F. The suppression chamber is vented separately. (Refer to Figure 5.3.1, Reactor Building Ventilation and Standby Gas Treatment.)

5.2.3.6 Primary Containment Cooling and Ventilation System

The Primary Containment (Drywell) Cooling System utilizes fancoil units distributed inside the drywell. (Refer to Figure 5.2.8, Drywell Cooling.) Each fancoil unit consists of two cooling coils and two direct connected, two speed motor-driven fans. Each cooling coil is connected to a separate water sup ply and return piping system inside the drywell to permit the use of either the Chilled Water System for normal service or the Reactor Building Cooling Water System for standby service. Should the Chilled Water System malfunction, the standby system continues to cool the drywell. Each fancoil unit is manually controlled from outside the primary containment. Each of the two fans in a fancoil unit may run individually or simultaneously. Drywell space temperatures , and inlet and outlet temperatures of the fancoil units, are indicated outside the primary containment. High fan discharge terperature is annunciated in the main control room.

Each fancoil units are designed to run at 340°F and 55 psig high radiation environment and will operate under ony one of the following modes:



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PRIMARY CONTAINMENT PROCESS LINE PENETRATION

FIGURE 5.2.5

July 30, 1971

SECTION 6

CORE STANDBY COOLING SYSTEMS

CONTENTS

taken for the pressure reduction effect of the cold LPCI water in the reactor vessel.

The effective range of LPCI alone (3 or 4 pumps) for the spectrum of steam or liquid line breaks is shown in Figure 6.2.1. The half-width portion of the bar shows the overlap with other Core Standby Cooling Systems.

To assure continuity of core cooling, signals to isolate the primary or secondary containments do not operate any LPCI valves. This arrangement satisfies safety design basis 6.

The two testable check valves (one in each loop) are the only LPCI equipment in the primary containment required to actuate during a loss-of-coolant accident which require consideration for the high temperature and humidity environment in the containment from the accident. The type of valve chosen actuates on flow through the pipeline, independent of any external signal. The actuator is provided only for test. Thus, neither the normal nor accident environment in containment affects the operability of the Low Pressure Coolant Injection equipment for the accident. It is concluded that safety design basis 9 is satisfied.

Using the suppression pool as the source of water for LPCI establishes a closed loop for recirculation of LPCI water escaping from the break. It is concluded that safety design basis 11 is satisfied.

The LPCI and appropriate portions of the recirculation loops are designed as Class I Nuclear so that they meet design basis 8.

6.5.3 Integrated Operation of the Core Standby Cooling Systems

The previous discussion has described the performance and operation of each of the CSCS individually. This discussion is directed toward the integrated performance of the CSCS, i.e., how the CSCS operate together to provide core cooling for the entire spectrum of loss-of-coolant accidents, viz., a break of a liquid line and a break of a steam line. The primary emphasis of the discussion

6.5.3.1 Liquid Line Breaks

1. Large Breaks

The double-ended recirculation line break is one of the bases for the design of the Core Standby Cooling Systems and the containment response calculations. The containment response is discussed in Section 14, "Plant Safety Analysis".

This accident is analyzed with a nine node reactor volume simulation model. The nine nodes are the lower plenum, the core, the upper plenum, the leakage region, the separation zone, the steam dome, the downcomer and the two recirculation loops. The jet pump modeling assumes that conservation of momentum and drive pump trip can be included. The core and leakage regions each have 10 common pressure subnodes. Included in the model is a method of calculating the movements of the liquid level in the separator region. A vapor to liquid relative velocity of 1 ft/sec is assumed in these calculations (refs. 3, 7).

It is assumed that the reactor is operating at design power when a complete circumferential rupture instantly occurs in one of the two recirculation system suction lines. An interlock assures that the valve in the equalizer line between the jet pump headers will be closed when both recirculation pumps are operating; thus, the area available for coolant discharge from the reactor vessel would be the sum of 10 jet pump nozzle areas and the cross sectional area of the main recirculation line and the reactor water cleanup suction line (4.82 ft^2). This is the worst case for the CSCS analysis.

Immediately after the break, critical flow would be established at the break. The large increase in core void fraction that would be caused by the decreasing vessel pressure would be sufficient to render the core subcritical. High drywell pressure would initiate mechanical scram of the control rod system in less than one second. In about 9 seconds the liquid inventory in the downcomer and the separator region of the vessel would be depleted and the break flow would switch from liquid to vapor; this would result in a large increase in the vessel depressurization rate.

6.7 CORE STANDBY COOLING SYSTEM IMPROVEMENTS

The core standby cooling systems originally proposed for Limerick (described in Sections 6.1 through 6.5) are adequate in that the ECCS provided in depth protection with redundant systems meeting all criteria with significant margin. However, Philadelphia Electric will provide additional cooling margin and system reliability for ECCS. The following improvement to further enhance the margins already present will be made to the core standby cooling systems for the Limerick Units:

1. The HPCI system will discharge into a core spray header rather than into a feedwater line.

2. The HPCI system will be analyzed in detail to determine whether the present equipment can attain effectiveness as a core spray.

3. The LPCI system will discharge into the vessel inside the core shroud through four separate penetrations rather than into the recirculation piping.

6.7.1 HPCI System Improvements

Injection of the HPCI fluid through the core spray sparger results in more efficient utilization of the flow than if the water is injected outside the core shroud. The reason for this is that by injecting the water over the core, it must pass through the core before it is lost from the primary system. Thus, water levels inside the shroud would be maintained higher for longer times which result in better core cooling and therefore further lowering of peak clad temperatures.

In addition to the above improvement, the HPCI flow rate may be increased by optimal pump impeller design. The exact magnitude of the gain will be established during detail system design. The purpose of this increase is to maximize the probability that the HPCI can function as a core spray over that section of the break spectrum for which pressure is available. If indeed the HPCI can be claimed as a core spray, this would result in peak clad temperatures remaining below 700°F for breaks up to approximately 0.6 ft^2. That is, core heat up for this range of break sizes would be precluded by the early actuation of core spray.

6.7.2 LPCI System Improvements

Revising the LPCI System such that it discharges directly through four seperate nozzles into the reactor vessel eliminates the possibility of losing the LPCI mode of the RHR system due to a single component failure (such as an injection valve). This approach is a further improvement in LPCI system reliability. In addition, the use of smaller, faster opening valves result in significantly faster flooding time to give even greater cooling margins.

The LPCI water injection into the top of the core instead of into the lower plenum provides some core cooling before the lower plenum is filled with water. However, credit has not been taken for this cooling in any of the preliminary analysis. It is expected that these improvements to the low pressure coolant injection system will result in a peak fuel temperature reduction during the worst case accident (DBA plus single failure) of between 200 to 280°F.

SECTION 7

CONTROL AND INSTRUMENTATION

CONTENTS

7.2.6 ADDITIONAL INFORMATION

Studies have been conducted to evaluate the effects of a postulated failure to scram under anticipated transients. The General Electric Company report NEDO-10349, March 1971, has been submitted to AEC for review.

The postulated failure of the total scram protection function is not considered a credible event and has not been our design basis. However, provisions will be made in the Limerick design such that the function of tripping the recirculation pumps, as described in the above report, will be added.

SUBSECTION 7.2 REACTOR PROTECTION SYSTEM

REFERENCE

1. Hentschel, M. K. et al., "Compliance of Protection Systems to Industry Criteria: General Electric BWR Nuclear Steam Supply System", June 1970 (NEDO - 10139)

7.3.4.4 Logic

The basic logic arrangement is one in which the operation of an automatic isolation valve is controlled by two trip systems. Where many isolation valves close on the same signal, two trip systems control the entire group. Where just one or two valves must close in response to a special signal, two trip systems may be formed from the instruments provided to sense the special condition. Valves that respond to the signals from common trip systems are identified in the detailed descriptions of isolation functions.

Each trip system has a pair of logics, each logic of which receives input signals from at least one channel for each monitored variable. Thus, two channels are required for each essential monitored variable to provide independent inputs to the logic of one trip system. A total of four channels for each essential monitored variable is generally provided for the logics of both trip systems except for HPCI excess flow, which is 1:2 logic. Figures 7.3.2 and 7.3.3 illustrate typical isolation control arrangements for motor-operated valves and for the main steamline isolation valves.

The actuators associated with one logic pair provide inputs into each of the actuator logics for that trip system. Thus, either of the two logics associated with one trip system can produce a trip system trip. The logic is a one-out-of-n arrangement, where n may be two or more.

To initiate valve closure the actuator logics of both trip systems must be tripped. The overall logic of the system could be termed one-out-of-two taken twice.

The basic logic arrangement just described does not apply to class C isolation valves and testable check valves. Exceptions to the basic logic arrangement are made for the HPCI and RCIC isolation valves as described below.

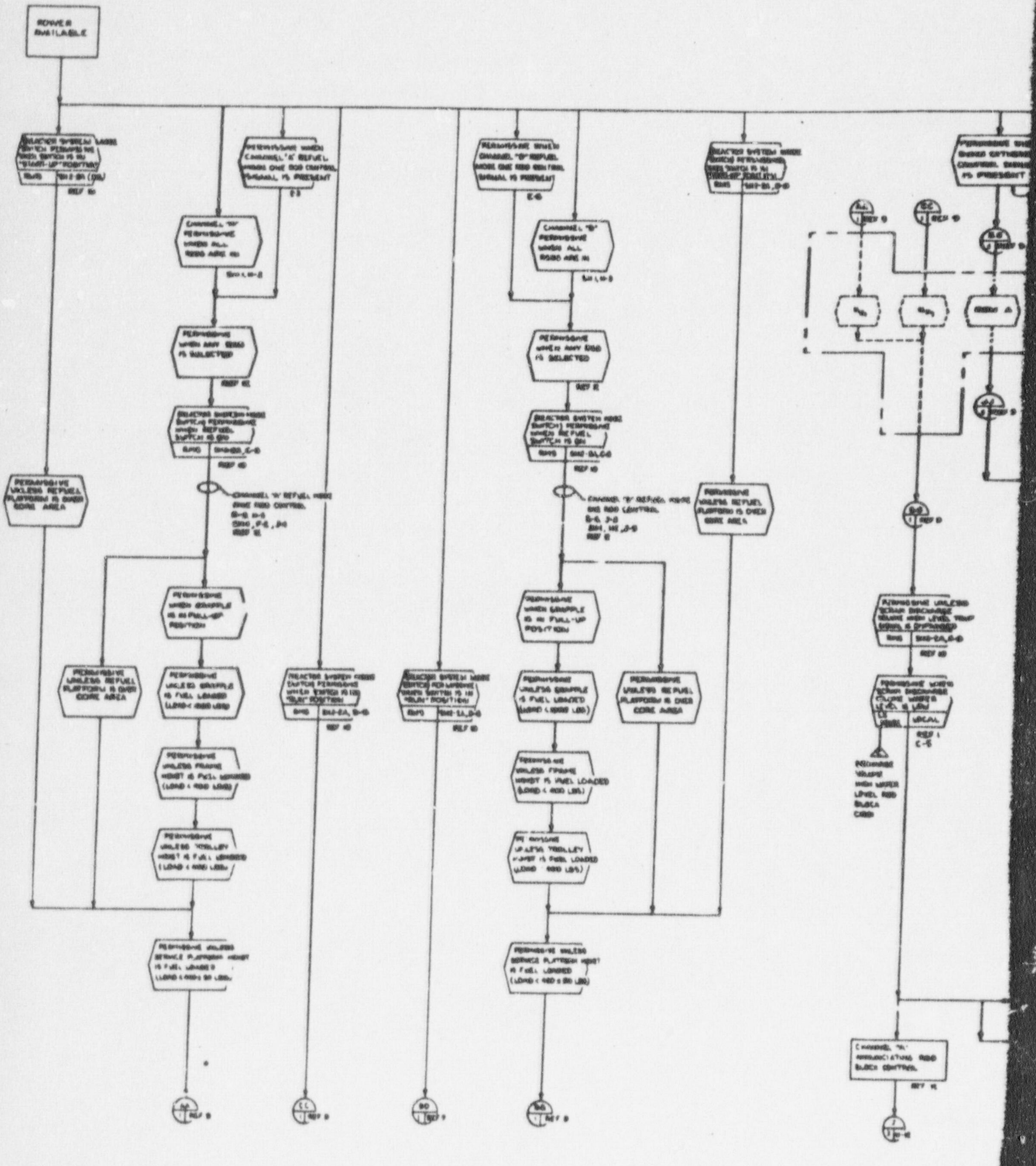
7.3.4.5 Operation

During normal operation of the plant, when isolation is not required, sensor and trip contacts essential to safety are closed; channels and trip logics are normally energized. Whenever a channel sensor contact opens, its auxiliary relay deenergizes causing contacts in the trip logic to open. The opening of a sufficient number of contacts in the

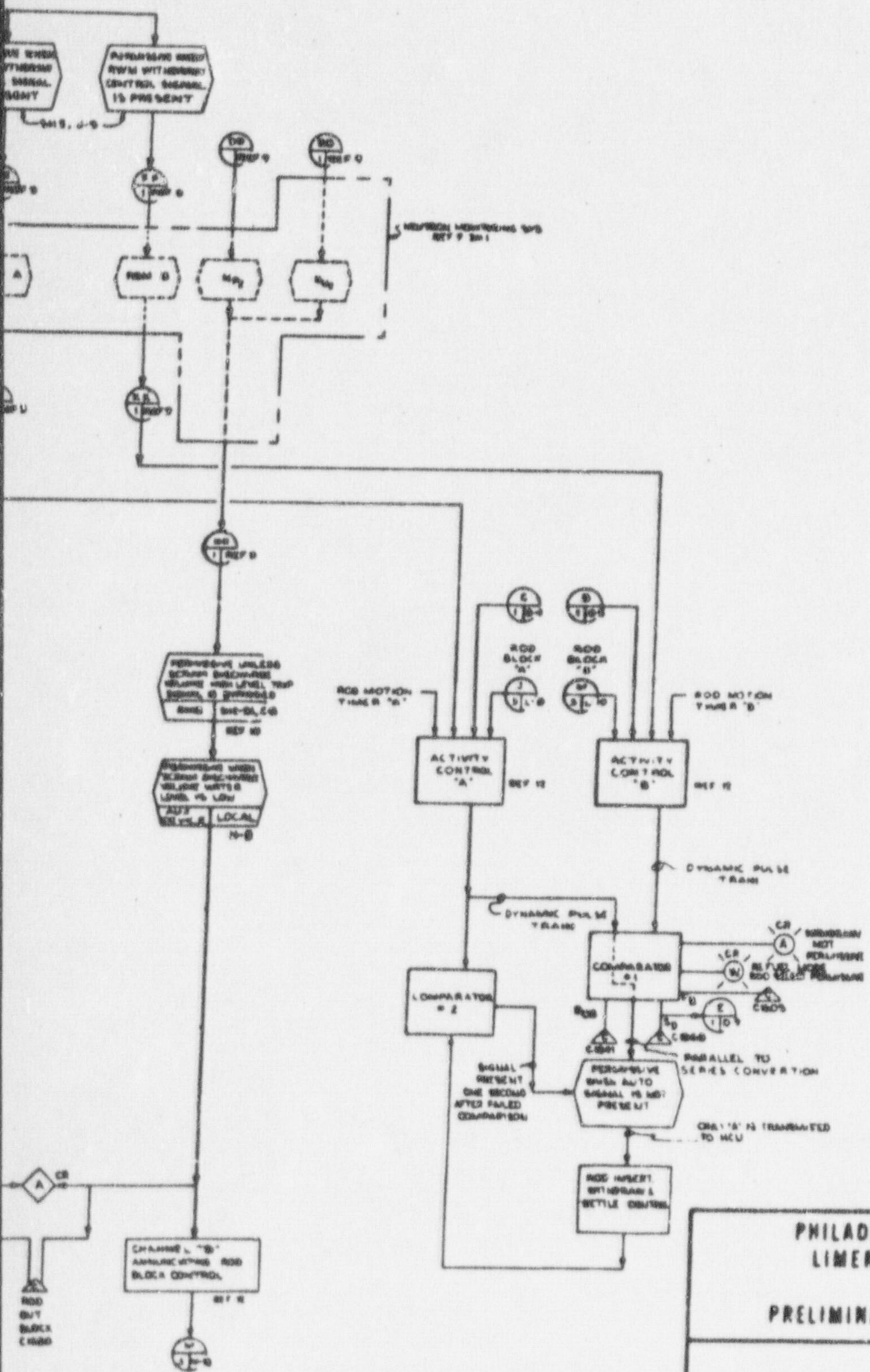
The main steam line high flow trip setting was selected high enough to permit the isolation of one main steam line for test at rated power without causing an automatic isolation of the rest of the steam lines yet low enough to permit early detection of a gross steam line break.

5. <u>Low Steam Pressure at Turbine Inlet</u> (Table 7.3.1, Signal P)

Low steam pressure upstream of the turbine stop valves while the reactor is operating could indicate a malfunction of the pressure regulator in which the turbine control valves or turbine bypass valves open fully. This action could cause rapid depressurization of the nuclear system. From part-load operating conditions, the rate of decrease of nuclear system saturation temperature could exceed the design rate of change of vessel temperature.


POWER AVAILABLE
PERMISSIVE UNLESS REFUEL PLATFORM IS OVER CORE AREA
CHANNEL "A" PERMISSIVE WHEN ALL RODS ARE IN
CHANNEL "B" PERMISSIVE WHEN ALL RODS ARE IN
PERMISSIVE WHEN ANY ROD IS SELECTED
PERMISSIVE WHEN GRAPPLE IS IN FULL-UP POSITION
PERMISSIVE UNLESS GRAPPLE IS FUEL LOADED
CHANNEL "A" ANNUNCIATING ROD BLOCK CONTROL

761E913AD SHT. 3 REV. 0



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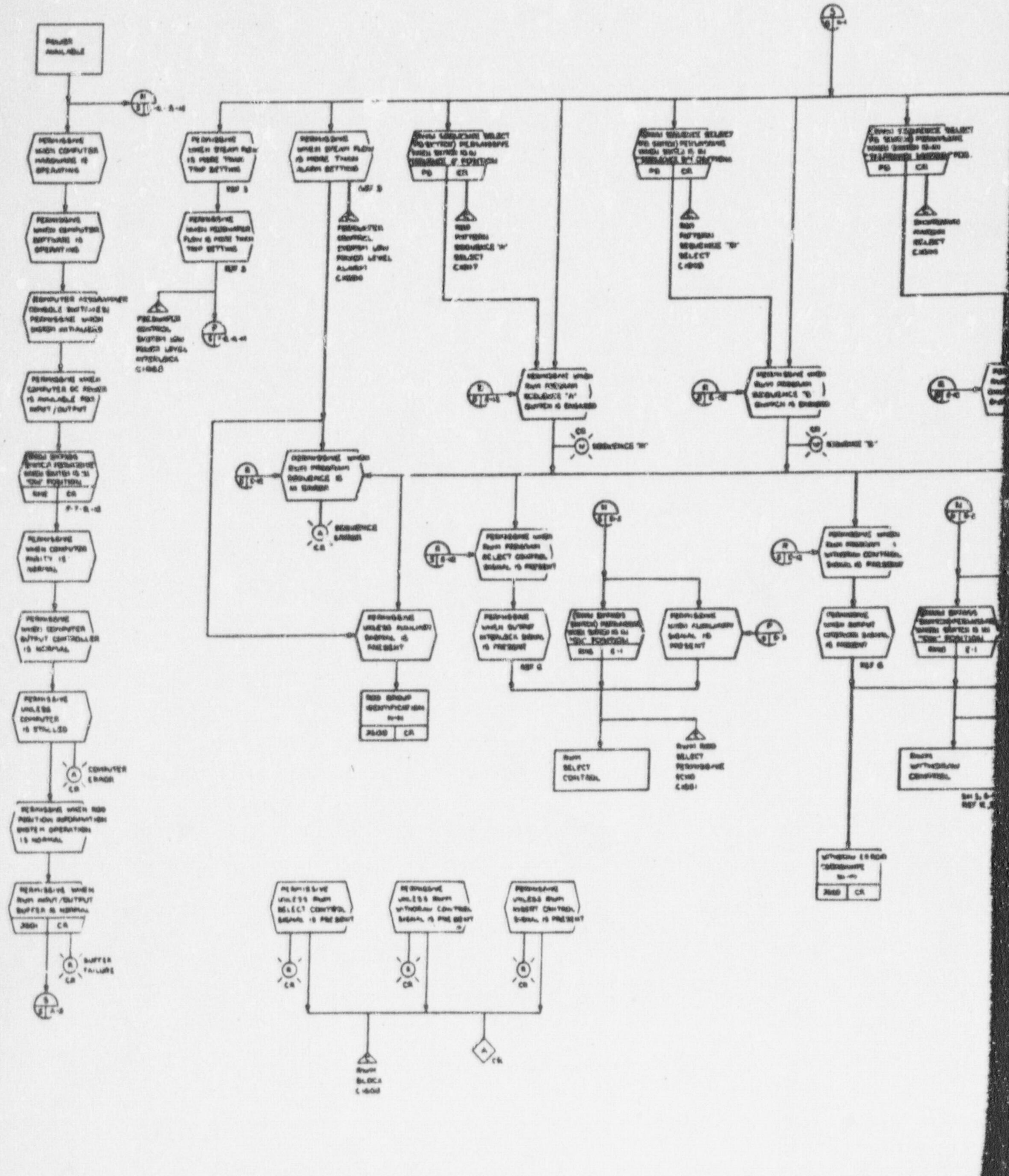
CONTROL ROD DRIVE HYDRAULIC SYSTEM, FUNCTIONAL CONTROL DIAGRAM, SHEET 3

2


POWER AVAILABLE
PERMISSIVE WHEN COMPUTER HARDWARE IS OPERATING
PERMISSIVE WHEN COMPUTER SOFTWARE IS OPERATING
COMPUTER ERROR
SEQUENCE ERROR
PERMISSIVE WHEN RWM INPUT/OUTPUT BUFFER IS NORMAL
BUFFER FAILURE
RWM SELECT CONTROL
RWM WITHDRAW CONTROL
PERMISSIVE UNLESS RWM SELECT CONTROL SIGNAL IS PRESENT
PERMISSIVE UNLESS RWM WITHDRAW CONTROL SIGNAL IS PRESENT
PERMISSIVE UNLESS RWM INSERT CONTROL SIGNAL IS PRESENT
RWM BLOCK

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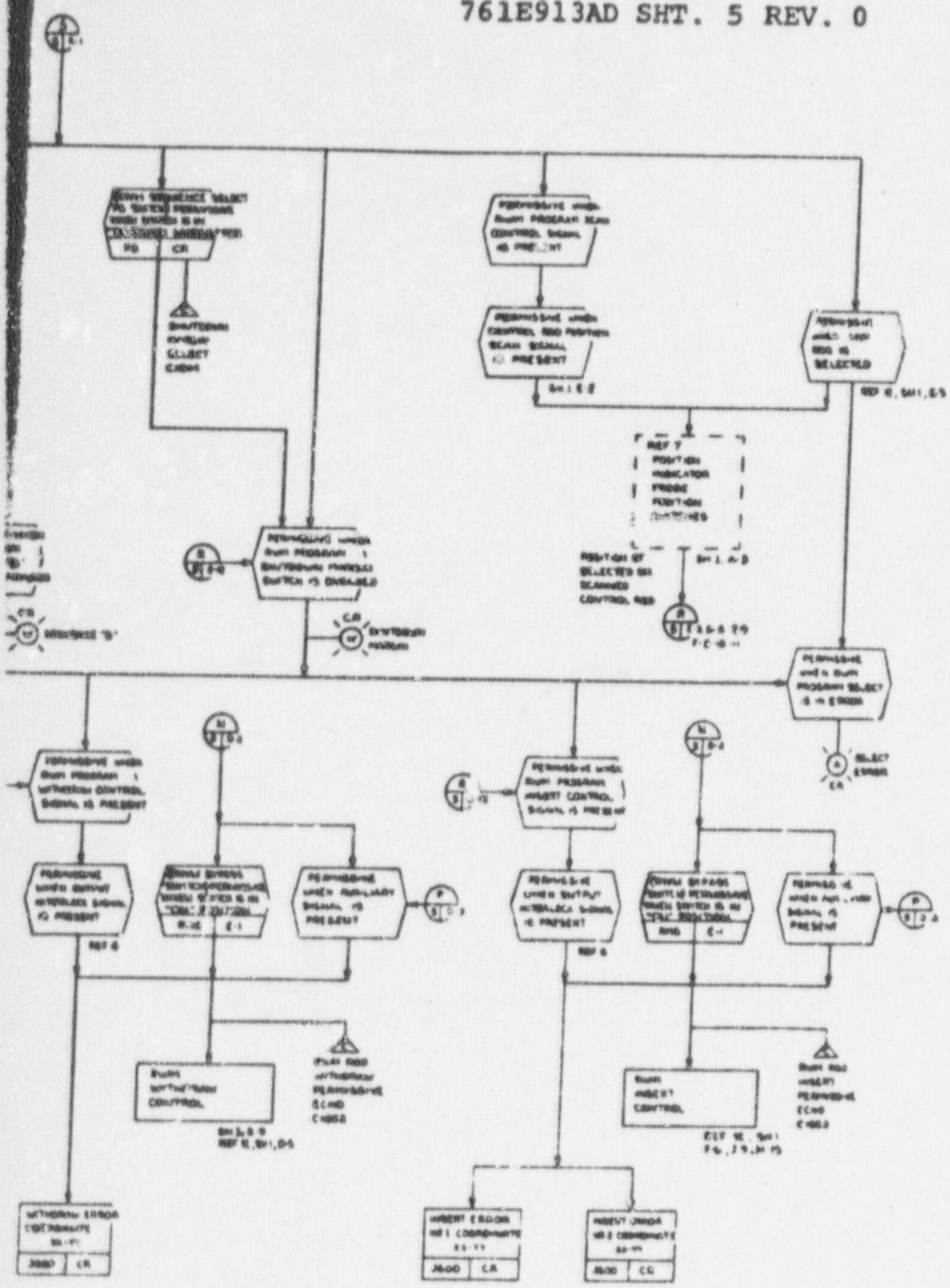

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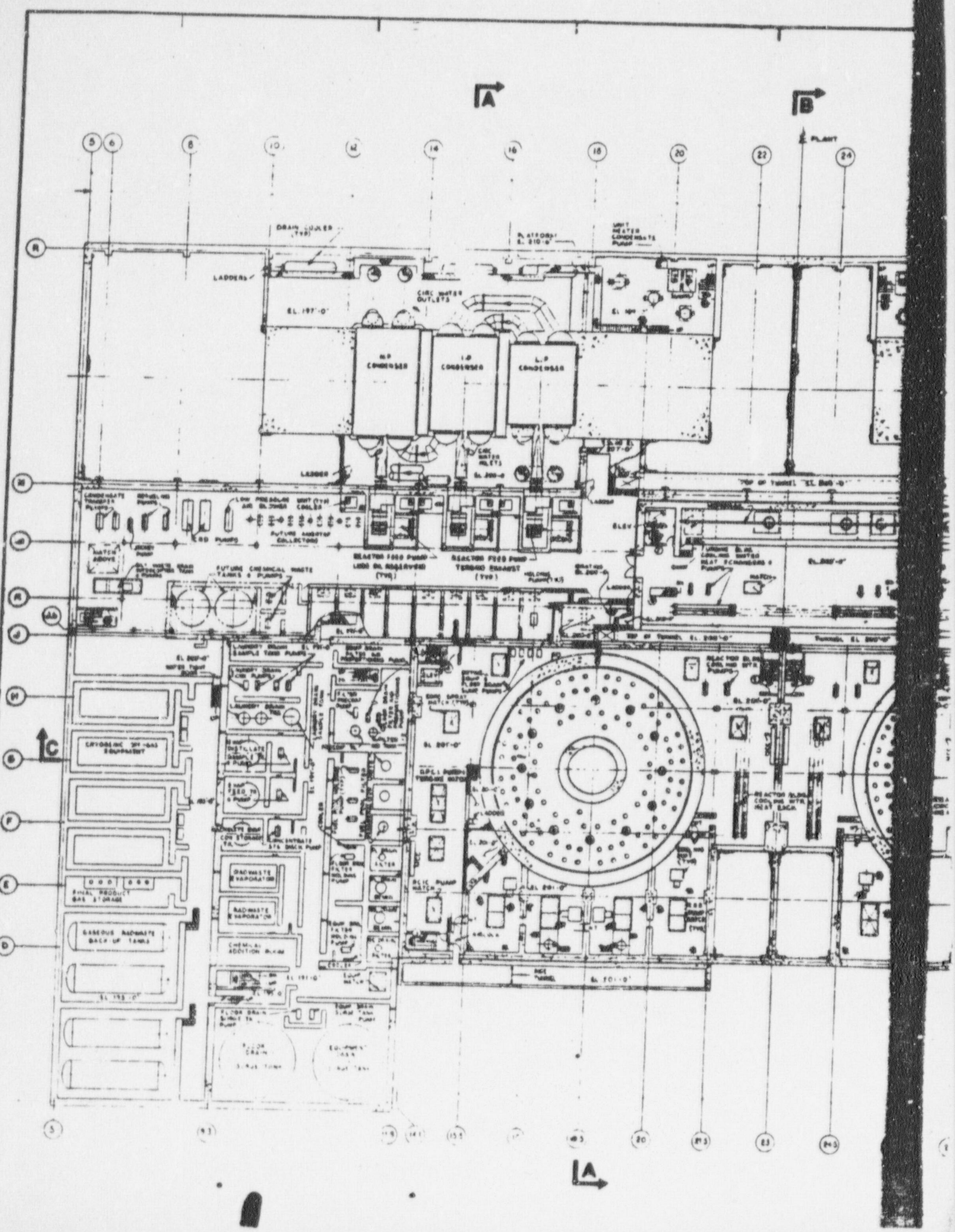
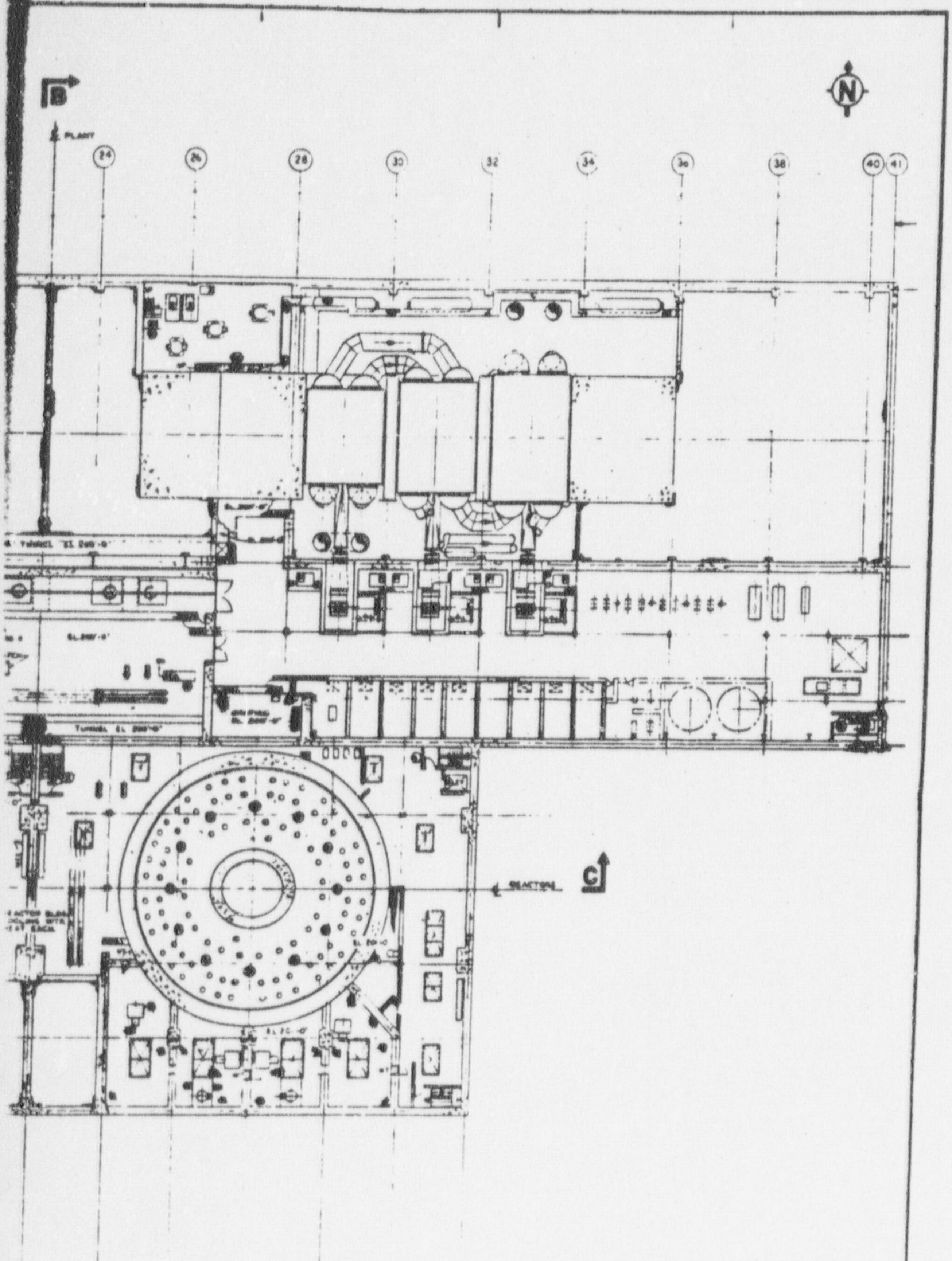
CONTROL ROD DRIVE
HYDRAULIC SYSTEM, FUNCTIONAL
CONTROL DIAGRAM, SHEET 5

2



A
B
C
H.P CONDENSER
I.P CONDENSER
L.P CONDENSER
DRAIN COOLER (TYP)
LADDERS
CIRC WATER OUTLETS
EL. 197'-0"
LADDER
REACTOR FEED PUMP
FUTURE CHEMICAL WASTE TANKS & PUMPS
CRYOGENIC OFF-GAS EQUIPMENT
FINAL PRODUCT GAS STORAGE
GASEOUS RADWASTE BACK UP TANKS
RADWASTE EVAPORATOR
CHEMICAL ADDITION BLDG
FLOOR DRAIN SURGE TANK
ELEV



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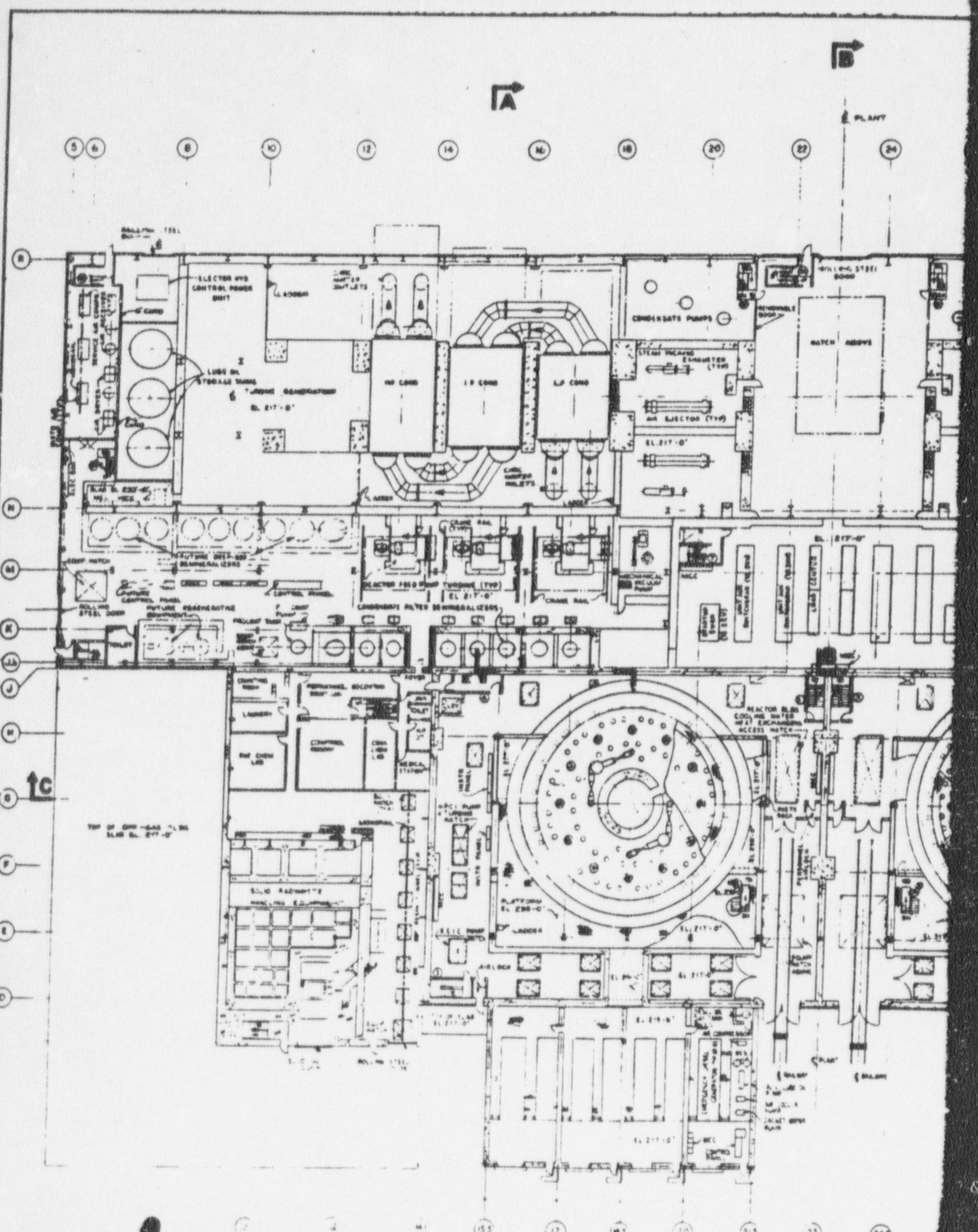
REACTOR BUILDING, RADWASTE BUILDING, OFF GAS BUILDING, AND TURBINE BUILDING, PLAN AT ELEVATION 201'-0"



A
B
C
PLANT
5
6
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14
16
18
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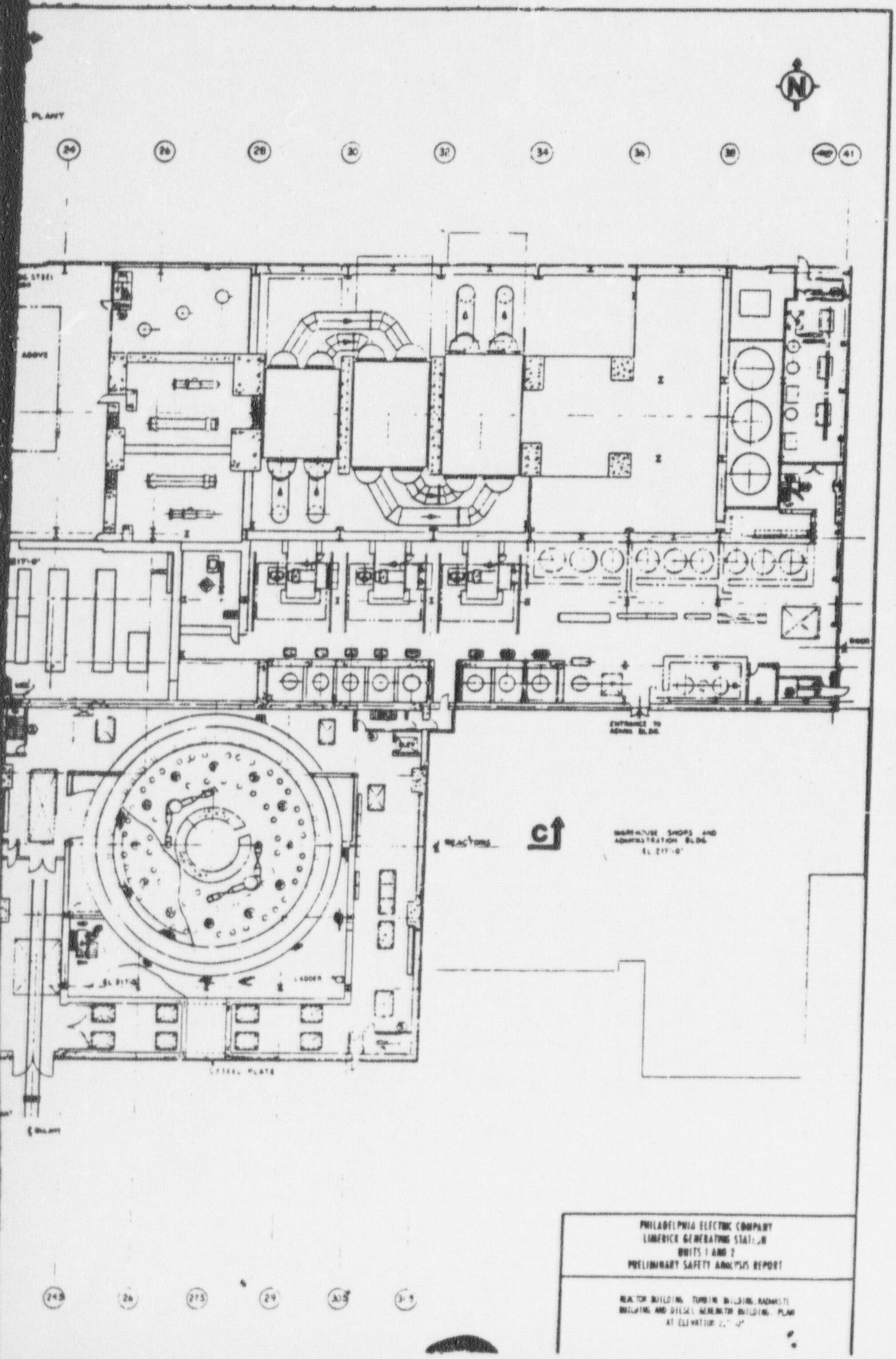
22
24
R
N
M
K
J
H
G
F
E
D
CONDENSATE PUMPS
HATCH ABOVE
TURBINE GENERATOR
EL. 217'-0"
AIR EJECTOR (TYP)
ROLLING STEEL DOOR
LAUNDRY
CONTROL ROOM
SOLID RADWASTE HANDLING EQUIPMENT
PLATFORM EL. 290'-0"
AIRLOCK
LADDER



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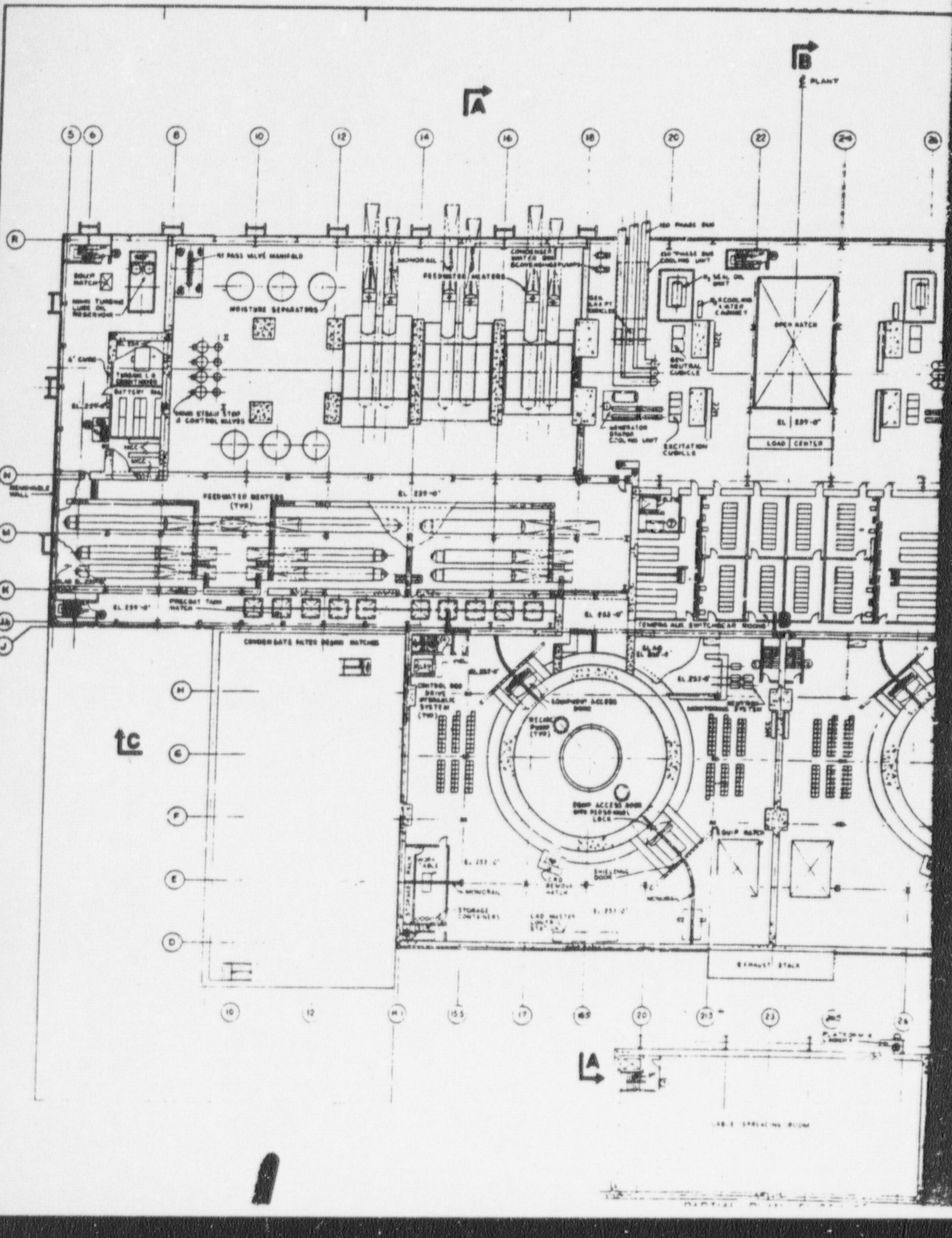
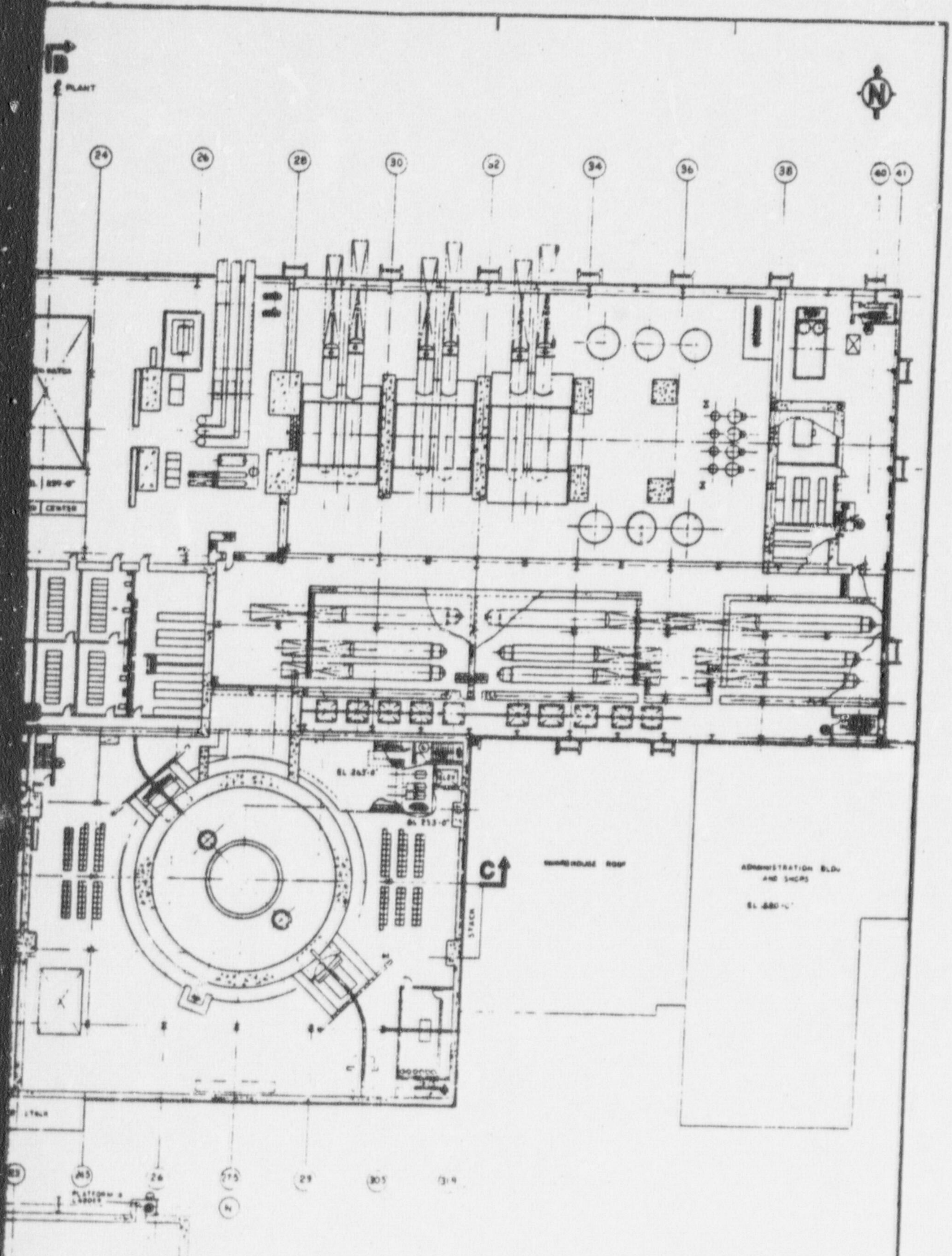
REACTOR BUILDING, TURBINE BUILDING, RADWASTE BUILDING AND DIESEL GENERATOR BUILDING, PLAN AT ELEVATION



MOISTURE SEPARATORS
FEEDWATER HEATERS
MONORAIL
OPEN HATCH
LOAD CENTER
EXCITATION CUBICLE
GENERATOR STATOR COOLING UNIT
GEN NEUTRAL CUBICLE
BATTERY
FEEDWATER HEATERS (TYP)
EL 239'-0"
REMOVABLE WALL
CONDENSATE FILTER DEMIN HATCHES
CONTROL ROD DRIVE HYDRAULIC SYSTEM (TYP)
EQUIPMENT ACCESS DOOR
EQUIP HATCH
SHIELDING DOOR
STORAGE CONTAINERS
EXHAUST STACK
CABLE SPREADING ROOM



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REACTOR BUILDING, TURBINE BUILDING, AND RADWASTE BUILDING PLAN AT ELEVATIONS 259'-0" AND 253'-0"

2

design work. This equipment is designed to meet a detailed functional requirement specification. The design is supported by field experience and test experience.

2. Special Supplemental Methods

Some complex equipment (e.g., reactor internals) is normally sized by rational stress analysis techniques and requires supplemental criteria in areas where industrial codes do not apply.

A.3.1.1 Piping

Class I Seismic piping is classified as either rigid or flexible. Rigid piping is that which has a fundamental frequency in the rigid range of the spectrum curves for the building locations. This generally corresponds to frequencies greater than about 30 cps. These piping systems are analyzed with static loads corresponding to the acceleration in the rigid range of the spectrum curves.

The dynamic analysis of flexible Class I Seismic Piping Systems for seismic loads is performed using the spectrum response method. The percentage of critical damping for all modes is 0.5 for the Operating Basis Earthquake (OBE) and 0.5 for the Design Basis Earthquake (DBE).

The vertical and horizontal floor response spectra applied to the Piping Systems are developed as part of the seismic analysis for the building in which the piping is located.

When the seismic load is due to the Design Basis Earthquake (0.12 g horizontal plus 0.08 g vertical), the vectorial combination of all longitudinal primary stresses does not exceed material yield stress at temperature unless higher allowable limits are calculated and substantiated by the methods outlined in PSAR Volume 5, Appendix C.

The main steam line (MSL) from the MSL isolation valve up to and including the turbine stop valve and the turbine bypass line from the main steam lines to the bypass valve header, including the header, and their associated restraints will be designed by the use of a dynamic seismic analysis to withstand the OBE and DBE loads within the limits of the ANSI B31.7 Class II piping code and Appendix A. The dynamic input for design of the MSL will be derived from a time history modal analysis (or an equivalent method) of the pertinent supporting structures.

The Class II Turbine Building, housing the MSL's may undergo some plastic deformation under the DBE; however, the plastic deformation will be limited to a ductility factor of 2 and an elastic multi-degree of freedom system analysis will be performed. The MSL supporting structures (those portions of the Turbine Building) will be such that the MSL and its supports can perform their safety function under the Class I Seismic loading conditions. The stress allowable and associated deformation for piping will be limited to 1.2 times the stress allowable (S_h) for OBE and yield stress for DBE.

A.3.1.2 Equipment

Equipment is supported or restrained to accommodate seismic loading determined in accordance with the criteria defined in Appendix C, "Structural Loading Criteria."

A.3.2 Materials

A.3.2.1 Brittle Fracture Control for Ferritic Steels

The fracture or notch toughness properties and the operating temperature of ferritic materials in systems which form the reactor coolant and primary containment pressure boundaries are controlled to ensure adequate toughness when the system is pressurized to more than 20 percent of the

APPENDIX C

STRUCTURAL DESIGN CRITERIA

CONTENTS

APPENDIX C

STRUCTURAL DESIGN CRITERIA

TABLES

C.4 NSSS EQUIPMENT LOADING DESIGN CRITERIA

C.4.1 Loading Criteria

Class I equipment is examined to assure its ability to withstand seismic requirements. Experienced designers determine which specific portions of systems and components require further examination.

The techniques used in this determination fit into two general categories. They are:

a. Normal analytical techniques using empirical design methods, as defined by appropriate design codes.

b. Special techniques (employed to supplement code calculations, or to cover conditions not considered by existing codes).

C.4.1.1 Normal Design Techniques

All class I equipment is designed in accordance with applicable industrial codes. The limits contained in the applicable design codes will not be exceeded. Some codes utilize empirical design methods for equipment which cannot be sized by conventional rational stress analysis methods, and which do not require a detailed stress analysis for primary design work. This equipment is designed to meet a detailed functional requirement specification. The design is supported by empirical field experience and test experience. Examples are valve bodies and pump cases.

C.4.1.2 Core Support Structures

The stress, deformation and fatigue criteria presented in Tables C.4.5, 6 and 7 are used. These criteria are supplemented, where applicable, by the criteria of Table C.4.4 but in no case are the criteria presented in Tables C.4.5, 6 and 7 exceeded for core support structures.

C.4.1.2.1 Bolting

The design stress intensity limits used in the design of bolting for reactor core support structures are as follows:

1. The maximum value of the primary plus secondary membrane stress intensity, including stress from preload, averaged across the area of either the shank or threads, shall be no greater than the lesser of 90% of the yield strength or 2/3 of the ultimate strength, both at temperature.

2. The primary plus secondary membrane stress plus bending stress at the periphery of the bolt shall be no greater than the lesser of 1.2 times yield strength or 8/9 of the ultimate strength both at temperature.

3. The average value of shear stress in the threads is no greater than 0.6 of yield strength at temperature.

4. The average value of bearing stress under the head of bolt is no greater than 2.7 times yield strength at temperature.

The above stated criteria are used for normal, upset, emergency and faulted conditions.

C.4.1.3 Reactor Internal Structures Other than Core Support

The stress, deformation and fatigue criteria listed in Table C.4.4 or empirical methods such as described in paragraph C.4.1.6 are used in the design of the reactor internal structures.

As noted in Table C.4.1, the loading conditions are classified into four categories, with the plant requirements specified for each. In turn, a minimum safety factor is imposed based on the plant requirements, for example, a higher safety factor or margin between normal operation and failure is required for normal conditions where equipment must continue in operation, whereas a lower safety factor (but still greater than 1) or margin is allowed for faulted conditions where the system is not required to remain operational, but need only shutdown safely. The safety factors used in the loading criteria analyses are shown in Table C.4.2.

It is not planned to use stress limits associated with faulted conditions as shown in Table C.4.4 for the equipment and components which (1) are not part of the reactor coolant pressure boundary and (2) are covered by applicable design codes.

Table C.4.3 lists supplementary criteria which are used in the design of the station. The deformation, buckling stability and fatigue limits included in Table C.4.4 are included for completeness but are not necessarily applied to all components. Where it is clear that the fatigue, excess deformation or buckling limit is not applicable to a particular structure or component, a formal analysis with respect to that limit will not be performed.

Two limiting criteria are considered in Table C.4.4 which negate the need for specific strain limits. These are the deflection limits and plastic instability limits described below.

The deflection limit requires that maximum permissible deformation under combination loading be limited to 80% of the loss of function (LOF) deformation (calculated on a conservative basis). As a practical matter, the stresses in most of the critical components are so low that these deformation limits are not invoked.

When combination loading stresses do exceed the yield stress, the plastic instability design criterion would permit a maximum load equal to 80% of the plastic instability load. This criterion is more conservative than the recent edition of Section III of the ASME Boiler and Pressure Vessel Code which permits 90% of the plastic instability load. Using this criterion, the strain corresponding to this load varies from about 10% (non-strain hardening materials) to about 35% (strain hardening materials) of the ultimate strain at temperature as determined by standard ASTM tensile tests. It has not been necessary to use this criterion in the past, however, the method does represent the upper bound of strain permitted within the criterion. Primary stresses due to fault conditions are limited for design purposes to 2 S_m under combination loading. Since S_m implies a minimum factor of safety of 3 (e.g., S_m 1/3S ultimate) the minimum factor of safety on load obtained from this criterion would be 1.5.

The fact that the maximum load permitted is only 90% of the maximum load permitted by ASME Section III, a code generally recognized as being quite conservative, should demonstrate the adequate margin of safety present in the criteria.

C.4.1.3.1 Bolting

The design stress intensity limits used in the design of bolting for reactor internal structures, other than core support, are the limits specified in Table C.4.4 for ductile metal components.

C.4.1.3.2 Fatigue Limits

The fatigue limit criteria for analysis shown in Table C.4.4 are essentially identical to the fatigue limits of Section III of the ASME Boiler and Pressure Vessel Code except that a factor of safety of 20 on cycles below the mean fatigue data curves for the material is already contained in the Section III fatigue design curves and a cumulative usage factor of 1.0 is permitted whereas the criteria shown in Table C.4.4 permits the use of the mean fatigue data directly with the factor of safety of 20 being applied to the cumulative usage (i.e., usage is limited to 0.05 in Table C.4.4 rather than 1.0). This is an equivalent procedure which permits a fatigue analysis to be performed directly for materials which may not be covered by applicable industry codes and for which there are no code fatigue curves available.

The alternate criterion for an actual fatigue test represents a margin of safety of 3 below failure or loss of function. The experimental programs would be designed so as to insure that conservatism is present in all aspects of the test including geometry, tolerances, loading conditions, etc., so that a factor of safety of 3 should be more than ample.

C.4.1.4 Pressure Vessels, Piping, Pumps and Valves

The criteria for emergency and fault conditions of equipment constructed in accordance with the requirements of Quality Group (See paragraph A.2) B or C codes and standards are as follows:

$$\frac{E_{B-C}}{U_{B-C}} = \frac{E_A}{U_A}$$

$$\frac{F_{B-C}}{U_{B-C}} = \frac{F_A}{U_A}$$

Where U_A, E_A and F_A are the criteria listed in the applicable code for upset, emergency and fault conditions, respectively, for Group A equipment, U_{B-C} are the criteria listed in the applicable code for upset conditions for Group B or C equipment, and E_{B-C} and F_{B-C} are the criteria to be used for emergency and fault conditions respectively, for Group B or C equipment.

Table Q 15.22 contains a list of the Quality Group B and C equipment furnished by General Electric along with the loading conditions and stress limits for design.

Where analysis is required for the faulted condition on Quality Group A pumps ASME Section III stress limits will be used. Where analysis is required for the faulted condition on Quality Group A valves B31.7 Code case 70 stress limits will be used.

C.4.1.5 Structural Steel

Stress and deformation criteria of structural steel equipment shall be 0.9 of the yield stress for emergency design conditions and either the yield stress or plastic deformations that do not prevent accomplishment of the equipment safety functions for fault conditions.

C.4.1.6 Other Equipment

For other equipment the criteria shall be based on the criteria established in applicable codes and standards for similar equipment, by manufacturers standards (e.g., turbines), or by empirical methods based on field experience and testing.

TABLE C.4.1

GENERAL DEFINITIONS FOR LOADING CRITERIA

Class I Structures and Equipment - Structures and equipment that are essential to the safe shutdown and isolation of the reactor or whose failure or damage could result in significant release of radioactive material.

Class II Structures and Equipment - Structures and equipment that are important to reactor operation but are not essential to the safe shutdown and isolation of the reactor and whose failure cannot result in a significant release or radioactive material.

Class III Structures and Equipment - Structures and equipment that are not essential to the operation, safe shutdown or isolation of the reactor and whose failure cannot result in the release of radioactive material.

Note: Class II and III items shall not degrade higher class items.

Normal Conditions (Expected during 40-year operation)

Any condition anticipated to occur in the course of operation of the plant under planned, expected conditions (in the absence of upset, emergency, or faulted conditions).

Upset Conditions (Likely or possible during 40-year operation)

Any deviations from normal conditions anticipated to occur often enough that station design should include a capability to withstand the conditions with the station remaining operational or being capable of regaining its operational status.

Upset conditions include: abnormal operational transients caused by a fault in a system component requiring its isolation from the system; transients due to loss of load or power, any system upset not resulting in a forced outage, and operating basis earthquake.

Emergency Conditions (Low probability during 40-year operation)

Any deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system.

Emergency conditions have a low probability of occurence but are evaluated to provide assurance that no gross loss of structural integrity will result as a concomitant effect of any damage developed in the system.

TABLE C.4.1 (Continued)

Faulted Conditions (Extremely low probability)

Extremely low probability postulated events or combinations of conditions whose consequences are such that the integrity and operability of the nuclear system may be impaired to the extent where considerations of public health and safety are involved.

Operating Basis Earthquake (OBE) - An improbable event, which however may be considered possible during the 40-year station design life, and therefore an upset condition for which the station must be capable of remaining operational, or regaining its operational status.

Design Basis Earthquake (DBE) - A low probability event, and therefore an emergency condition, which however is evaluated to assure station capability for safe shutdown.

Pipe Rupture - The low probability rupture of a small Class I system pipe, which must therefore be considered an emergency condition; or the extremely low probability of rupture of a major Class I system pipe - such as the recirculation line break or main steam line break - which are used as design basis accidents for safety evaluations of station capability for protecting the public health and safety; or the extremely low probability of pipe rupture in conjunction with a design basis earthquake.

Minimum Safety Factor, SF_{MIN} - Minimum safety factors appearing in loading criteria used for design of high reliability Class I equipment. They are based on the operational or safe shutdown requirements placed upon the station, and the nature and severity of the loading condition.

TABLE C.4.2

MINIMUM SAFETY FACTORS

Condition (40 yr. Probability)	Governing Loading Condition	SF min
Upset (Likely or Possible)	N and A_D --or-- N and U	2.25
Emergency (low probability)	N and R	1.5
	--or-- N and A_M	1.5
	--or-- Other	2.25 to 1.5
Fault (extremely low probability)	N and A_M and R	1.125
	--or-- Other	1.5 to 1.125

Where:

N = normal loads

U = upset loads excluding earthquake

A_D= operating basis earthquake including any associated transients

A_M= design basis earthquake including any associated transients

R = any pipe rupture loading including any associated transients

TABLE C.4.3

Table C.4.3 ULTIMATE DESIGN STRESS VALUES FOR PIPING AND PRESSURE VESSEL MATERIALS

ITEM	REVISED GE CRITERIA		
	Limit	% Above Normal	% of Ultimate
Emer - Ferr - PV	$2.25S_m$	50	~50
Emer - Aust - PV	$2.25S_m$	50	~50
Emer - Ferr - Pipe	$2.25S_m$	50	~50
Emer - Aust - Pipe	$2.25S_m$	50	~50
Fault - Ferr	1.33LLB ($1.5S_m$)	100 (-)	67 (-)
Fault - Aust	1.33LLB ($1.5S_m$)	100	67
*Fault - PV	0.8 LPI	140	80
*Fault - Pipe	0.8 LPI	140	80
*Experiment	0.89 (ULT)	167	89

*These conditions will not be used prior to further discussion with the AEC Staff.

NOTE: S_m, LLB, and LPI are defined in Table C.4.4.

TABLE C.4.4

SUPPLEMENTARY LIMIT CRITERIA

FOR REACTOR INTERNAL STRUCTURES

PRIMARY STRESS LIMIT

No.	Any One of (Only One Required)	General Limit
P1	$\frac{\text{Elastic Evaluated Primary Stresses}}{\text{Permissible Primary Stresses}}$	$\frac{SE}{SP} \leq \frac{2.25}{SF_{min}}$
P2	$\frac{\text{Elastic Evaluated Primary Stress}}{\text{Conventional Ultimate Strength at Temperature}}$	$\frac{SE}{SU} \leq \frac{0.75}{SF_{min}}$
P3	$\frac{\text{Elastic-Plastic Evaluated Nominal Primary Stress}}{\text{Conventional Ultimate Strength at Temperature}}$	$\frac{SEP}{SU} \leq \frac{0.9}{SF_{min}}$
P4	$\frac{\text{Permissible Load}}{\text{Largest Lower Bound Limit Load}}$	$\frac{LP}{LLB} \leq \frac{1.5}{SF_{min}}$
P5*	$\frac{\text{Permissible Load}}{\text{Plastic Instability Load}}$	$\frac{LP}{LPI} \leq \frac{0.9}{SF_{min}}$
P6*	$\frac{\text{Permissible Load}}{\text{Ultimate Load from Fracture Analysis}}$	$\frac{LP}{LUF_r} \leq \frac{0.9}{SF_{min}}$
P7*	$\frac{\text{Permissible Load}}{\text{Ultimate Load or Loss of Function}}$	$\frac{LP}{LU_x}$ or $\frac{LP}{LLF_x} \leq \frac{1.0}{SF_{min}}$

S_m = The tabulated value of ASME III, or its equivalent, allowable stress at temperature.

SE = Primary stresses evaluated on an elastic basis. The effective membrane stresses are to be averaged through the load carrying section of interest. The simplest average bending, shear, or torsion stress distribution which will support the external loading will be added to the membrane stresses at the section of interest.

SP = Permissible primary stress levels under normal or upset conditions under applicable industry code.

SU = Conventional ultimate strength at temperature or loading which would cause a system malfunction as delineated in the design specification, whichever is more limiting.

TABLE C.4.4 (Continued)

PRIMARY STRESS LIMIT (Continued)

SEP = Elastic plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress strain curve at the temperature of loading or any approximation to the actual stress strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule may be used.

LP = Permissible load under stated emergency or fault conditions.

LLB = Lower bound limit load with yield point equal to 1.5 S_m. The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (non-strain hardening) material where formations increase with no further increase in applied load. The lower bound load is one in which the materi-l everywhere satisfied equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case. S_m is the tabulated value of ASME III, or its equivalent, allowable stress at temperature.

LPI = Plastic instability load. The "plastic instability load" is defined here as the load at which any load bearing sections begins to diminish its cross-sectional area at a faster rate than the strain hardening can accomodate the loss in area. This type analysis requires a true stress-true strain curve or a close approximation based on monotonic loading at the temperature of loading.

LUF_r = Ultimate load from fracture analyzer. For components which involve sharp discontinuities (local theoretical stress concentration 3) the use of a "Fracture Mechanics" analysis where applicable utilizing measurements of plain strain fracture toughness may be applied to compute fracture loads. Correction for finite plastic zones and thickness effects as well as gross yielding may be necessary. The methods of linear elastic stress analysis may be used in the fracture analysis where its use is clearly conservative or supported by experimental evidence. Examples where "Fracture Mechanics" may be applied are for fillet welds or end of fatigue life crack population.

LU_x or LLF_x = Ultimate load or loss of function load as determined from experiment. In using this method, account shall be taken of the dimensional tolerances which may exist between the

TABLE C.4.4 (Continued)

PRIMARY STRESS LIMIT (Continued)

actual part and the tested part or parts as well as differences which may exist in the ultimate tensile strength of the actual part and the tested parts. The guide to be used in each of these areas is that the experimentally determined load shall use adjusted values to account for material properties and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

DEFORMATION LIMIT CRITERIA**

No.	Any One of (Only One Required)	General Limit
D1	$\frac{\text{Permissible Deformation}}{\text{Analyzed Deformation Causing Loss of Function}}$	$\frac{DP}{DLF_A} \leq \frac{0.9}{SF_{min}}$
D2*	$\frac{\text{Permissible Deformation}}{\text{Experimental Deformation Causing Loss of Function}}$	$\frac{DP}{DLF_x} \leq \frac{1.0}{SF_{min}}$

DP = Permissible Deformation under stated normal, upset, emergency, or fault conditions.

DLF_A = Analyzed Deformation which would cause a system loss of function as delineated in the design specification.

DLF_X = Experimentally Determined Information which would cause a system loss of function as delineated in the design specification.

"Loss of Function" can only be defined quite generally until attention is focused on the component of interest. In cases of interest where deformation limits can affect the function of Class I equipment and components, they will be specifically delineated.

Examples where such limits apply are: control rod drive alignment and clearances for proper insertion, core support deformation causing fuel disarrangement, excess leakage of any component.

BUCKLING STABILITY LIMIT***

No.	Any One of (Only One Required)	General Limit
B1	$\frac{\text{Permissible Load}}{\text{Code Normal Event Permissible Load}}$	$\frac{LP}{LN} \leq \frac{2.25}{SF_{min}}$

TABLE C.4.4 (Continued)

BUCKLING STABILITY LIMIT (Continued)

No.	Any One of (Only One Required)	General Limit
B2*	$\frac{\text{Permissible Load}}{\text{Ultimate Buckling Collapse Load from Test}}$	$\frac{LP}{LUB_X} \leq \frac{1.0}{SF_{min}}$
B3	$\frac{\text{Permissible Load}}{\text{Stability Analysis Load}}$	$\frac{LP}{LS_A} \leq \frac{0.9}{SF_{min}}$

LP = Permissible Load under stated normal, upset, emergency, or fault conditions.

LN = Applicable code normal event permissible load.

LUB = Ultimate buckling collapse load as determined from experiment. In using this method, account shall be taken of the diagonal tolerances which may exist between the actual part and the tested part. The guide to be used in each of these areas is that the experimentally determined load shall be adjusted to account for material property and dimension variations, each of which has no greater probability than 0.1 of being exceeded in the actual part.

LS_A = Stability analysis load. The ideal buckling analysis is often sensitive to otherwise minor deviations from ideal geometry and boundary conditions. These effects shall be accounted for in the analysis of the buckling stability loads. Examples of this are ovality in externally pressurized shells or eccentricity of column members.

FATIGUE LIMIT CRITERIA***

No.	Any One of (Only One Required)	
	Summation of fatigue damage usage with design and operation loads following Miner hypotheses**-shall not exceed F1 F2, or F3 as appropriate:	
F1*	= Mean Fatigue cycle usage from analysis	$F_A \leq 0.05$
F2*	= Mean Fatigue cycle usage from test	$F_X \leq 0.33$
F3	= Design fatigue cycle usage from analysis#	$F_D \leq 1.0$

*Fatigue failure is defined here as the more limiting of:

TABLE C.4.4 (Continued)

FATIGUE LIMIT CRITERIA (Continued)

1. a 25 per cent area reduction for a load carrying member which is required to function, or

2. excessive leakage

In the fatigue ev.luation the methods of linear electric stress analysis may be used when the $3S_m$ range limit of ASME III has been met.

If $3S_m$ is not met, account will be taken of:

(a) increases in local, strain concentration, (b) strain ratcheting, (c) redistribution of strain due to elastic-plastic effects. The February, 1968, draft of the USAS B31.7 Piping Code may be used where applicable or detailed elastic-plastic methods may be used. With elastic-plastic methods, strain hardening may be used not to exceed in stress for the same strain, the steady state cyclic strain hardening measured in a smooth low cycle fatigue specimen at the average temperature of interest.

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*Equations P5, P6, P7, D2, F1, F2, and B2 will not be applied unless supporting data are submitted for evaluation by the AEC staff.

**Minor, M.A., "Cumulative Damage in Fatigue," Journal of Applied Mechanics, Vol. 12, Trans. ASME, Vol. 67, pp. A159-A164, Sept. 1945. It is acceptable to use the ASME Section III Design Fatigue Curves in conjunction with a cumulative usage factor of 1.0 (using Miner's hypothesis) in lieu of using the mean fatigue data curves with a limit on fatigue usage of 0.05 since the two methods are approximately equivalent.

***Formal analysis required only where qppropriate.

#Using method from Table C.4.5

TABLE C.4.5

CORE SUPPORT STRUCTURES,

STRESS CATEGORIES AND LIMITS OF STRESS I

FOR NORMAL AND UPSET CONDITION

STRESS CATEGORY	PRIMARY STRESSES Membrane, P_m (Notes 4,7,8)	 Bending P(Notes 4,7,8)	SEC Memb Second
Normal and Upset	$\boxed{P_m}$ S_m Or Elastic Analysis (Note 6) $.67L_L$ Or Limit Analysis (Note 10) $.44L_u$ Test (Note 11)	$\boxed{P_m+P_b}$ $1.5S_m$ Or Elastic Analysis (Note 6) $.67L_L$ Or Limit Analysis (Note 10) $.44L_u$ Test (Note 11)	P_m+ F T P (

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E C.4.5

T STRUCTURES,

MITS OF STRESS INTENSITY

UPSET CONDITIONS

	SECONDARY STRESSES	PEAK STRESSES
,7,8)	Membrane & Bending Secondary Q(Notes 2,4,6)	Peak F (Notes 2,4,6)

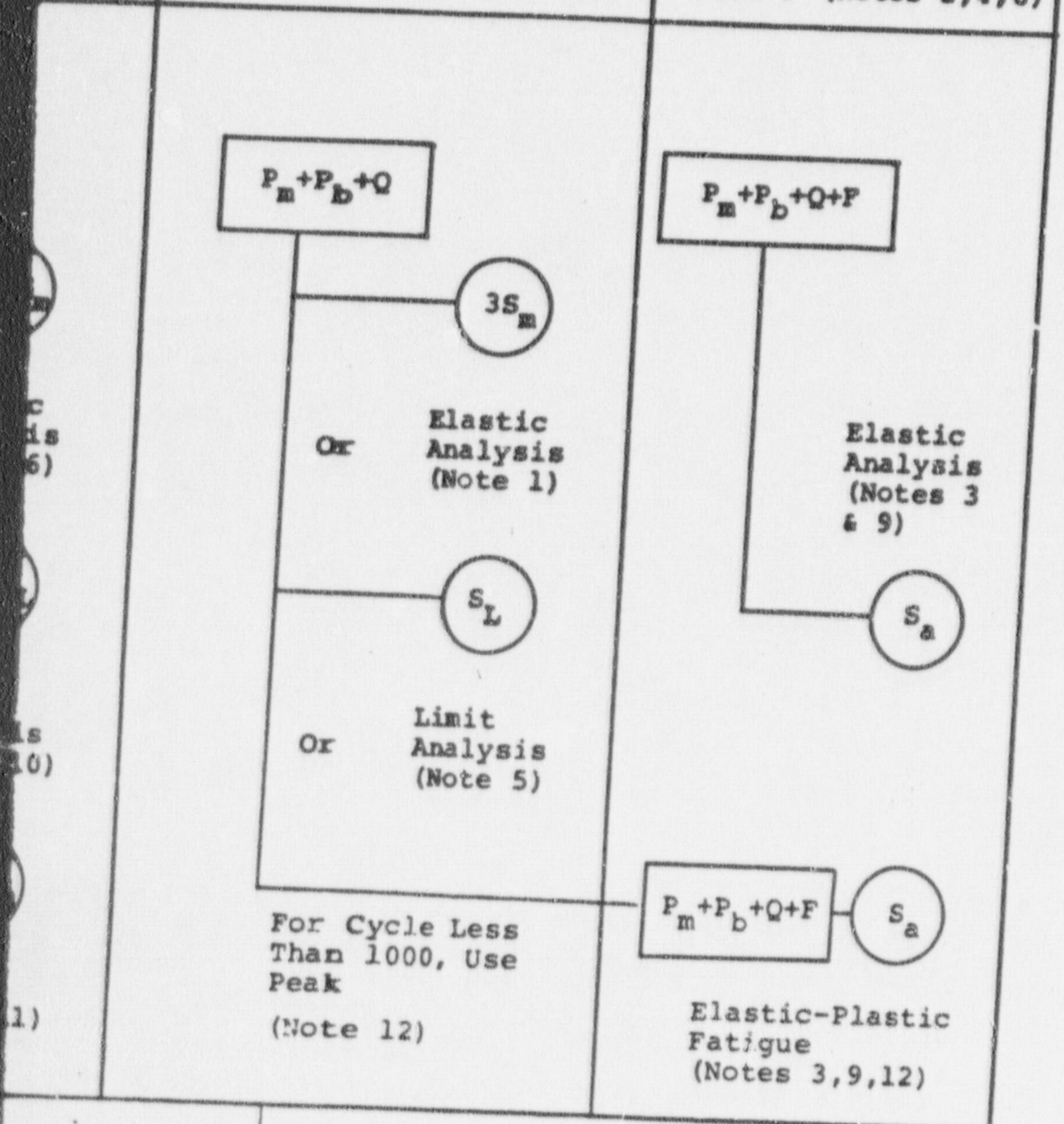

TABLE C.4.5 NOTES

NORMAL AND UPSET CONDITIONS

NOTE 1 - This limitation applies to the range of stress intensity. When the secondary stress is due to a temperature excursion at the point at which the stresses are being analyzed, the value of S_m shall be taken as the average of the S_m values tabulated in Tables N-421, N-422, and N-423 of ASME Boiler and Pressure Vessel Code, Section III, (ASME III) for the highest and the lowest temperature of the metal during the transient. When part of the secondary stress is due to mechanical load, the value of S_m shall be taken as the S_m value for the highest temperature of the metal during the transient.

NOTE 2 - The stresses in Category Q are those parts of the total stress which are produced by thermal gradients, structural discontinuities, etc., and do not include primary stresses which may also exist at the same point. It should be noted, however, that a detailed stress analysis frequently gives the combination of primary and secondary stresses directly and, when appropriate, this calculated value represents the total of $P_m + P_b + Q$ and not Q alone. Similarly, if the stress in Category F is produced by a stress concentration, the quantity F is the additional stress produced by the notch, over and above the nominal stress. For example, if a plate has a nominal stress intensity, $P_m = S$, $P_b = 0$, $Q = 0$ and a notch with a stress concentration K is introduced, then $F = P_m (K - 1)$ and the peak stress intensity equals $P_m + P_m (K - 1) = KP_m$.

NOTE 3 - S_a is obtained from the fatigue curves, Figures N-415 of ASME III. The allowable stress intensity for the full range of fluctuation is 2 S_a.

NOTE 4 - The symbols P_m, P_b, Q, and F do not represent single quantities, but rather sets of six quantities representing the six stress components σ_t, σ_l, σ_r, τ_{tl}, τ_{lr}, and τ_{rt}.

NOTE 5 - S_L denotes the structural action of shakedown load as defined in par. N-412 (9) of ASME III calculated on a plastic basis as applied to a specific location on the structure.

NOTE 6 - The triaxial stresses represent the algebraic sum of the three primary principal stresses ($\sigma_1 + \sigma_2 + \sigma_3$) for the combination of stress components. Where uniform tension loading is present triaxial stresses are limited to 4 S_m.

TABLE C.4.5 NOTES (Continued)

NOTE 7 - For configurations where compressive stresses occur, the stress limits shall be revised to take into account critical buckling stresses (see par. N-410 of ASME III). For external pressure, the permissible "equivalent static" external pressure shall be as specified by the rules of par. N-417.8 of ASME III. Where dynamic pressures are involved, the permissible external pressure shall be limited to 25% of the dynamic instability pressure.

NOTE 8 - When loads are transiently applied, consideration should be given to the use of dynamic load amplification, and possible change in modulus of elasticity.

NOTE 9 - In the fatigue data curves, where the number of operating cycles are less than ten, use the S_a value for ten cycles; where the number of operating cycles are greater than 10^6, use the S_a value for 10^6 cycles.

NOTE 10 - L_L is the lower bound limit load with yield point equal to 1.5 S_m (where S_m is the tabulated value of allowable stress at temperature as contained in ASME III). The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (non-strain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

NOTE 11 - For normal and upset conditions, the limits on primary membrane plus primary bending need not be satisfied in a component if it can be shown from the test of a prototype or model that the specified loads (dynamic or static equivalent) do not exceed 44% of L_u, where L_u is the ultimate load or the maximum load or load combination used in the test. In using this method, account shall be taken of the size effect and dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested part to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under the postulated loading for Normal and Upset Conditions.

TABLE C.4.5 NOTES (Continued)

NOTE 12 - The allowable value for the maximum range of this stress intensity is $3S_m$ except for cyclic events which occur less than 1000 times during the design life of the plant. For this exception, in lieu of meeting the $3S_m$ limit, an elastic-plastic fatigue analysis in accordance with ASME III or ASA B 31.7 may be performed to demonstrate that the cumulative fatigue usage attributable to the combination of these low cycle events plus all over cyclic events does not exceed a fatigue usage value of 1.0.

TABLE C.4.6

CORE SUPPORT STRUCTURES

STRESS CATEGORIES AND LIMITS OF STRESS INTE

FOR EMERGENCY CONDITIONS

STRESS CATEGORY	PRIMARY STRESSES		SECONDARY
	Membrane, P_m (Notes 1,2,&10)	Bending, P_B (Notes 1,2&10)	Membrane Second
Emergency (Note 9)	P_m $1.5S_m$ Elastic Analysis (Note 3) Or L_L Limit Analysis (Note 4) Or $1.5S_m$ Plastic Analysis (Note 6) Or $.6L_e$ Test (Note 7) Or S_E Stress-Ratio Analysis (Note 8)	P_m+P_B $2.25S_m$ Elastic Analysis (Note 3) Or L_L Limit Analysis (Note 4) Or $2.25S_m$ Plastic Analysis (Notes 5 & 6) $.5S_u$ (Note 5) Or $6L_e$ Tests (Note 7) Or KS_E Stress-Ratio Analysis (Note 8)	Evaluat Not Requ

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TABLE C.4.6

SUPPORT STRUCTURES

AND LIMITS OF STRESS INTENSITY

ERGENCY CONDITIONS

	SECONDARY STRESSES	PEAK STRESSES
ing, P_B 1,2&10)	Membrane & Bending Secondary-Q	Peak F
S_m Elastic Analysis (Note 3) Limit Analysis (Note 4) S_m Plastic Analysis (Notes 5 & 6) /u (Note 5) Tests (Note 7) Stress-Ratio Analysis (Note 8)	Evaluation Not Required	Evaluation Not Required 2

TABLE C.4.6 NOTES

EMERGENCY CONDITIONS

NOTE 1 - The symbols P_m, P_b, Q, and F do not represent single quantities, but rather sets of six quantities representing the six stress components σ_t, σ_l, σ_r, τ_{tl}, τ_{lr}, and τ_{rt}.

NOTE 2 - For configurations where compressive stresses occur, the stress limits shall be revised to take into account critical buckling stress. For external pressure, the permissible "equivalent static" external pressure shall be taken as 150 percent of that permitted by the rules of par. N-417.8 of ASME Boiler and Pressure Vessel Code, Section III. Where dynamic pressures are involved, the permissible external pressure shall satisfy the preceding requirements or be limited to 50% of the dynamic instability pressure.

NOTE 3 - The triaxial stresses represent the algebraic sum of the three primary principal stresses ($\sigma_1 + \sigma_2 + \sigma_3$) for the combination of stress components. Where uniform tension loading is present, triaxial stresses should be limited to $6S_m$.

NOTE 4 - L_L is the lower bound limit load with yield point equal to 1.5 S_m (where S_m is the tabulated value of allowable stress intensity at temperature as contained in ASME III). The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (non-strain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

NOTE 5 - S_u is the ultimate strength at temperature. Multiaxial effects on ultimate strength shall be considered.

NOTE 6 - This plastic analysis uses an elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress-strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the shear or strain energy of distortion flow rule shall be used to account for multiaxial effects.

NOTE 7 - For emergency conditions, the stress limits need not be satisfied if it can be shown from the test of a prototype or model that the specified loads (dynamic or

TABLE C.4.6 NOTES (Continued)

static equivalent) do not exceed 60% of L_e, where L_e is the ultimate load or the maximum load or load combination used in the test. In using this mehtod, account shall be taken of the size effect and dimensional tolerances which may exist between the actual part and the tested part or parts as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested parts to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading for emergency conditions.

NOTE 8 - Stress ratio is a method of plastic analysis which uses the stress ratio combinations (combination of stress that consider the ratio of the actual stress to the allowable plastic or elastic stress) to compute the maximum load a strain hardening material can carry. K is defined as the Section Factor; $S_E \leq 2S_M$ for primary membrane loading.

NOTE 9 - Where deformation is of concern in a component, the deformation shall be limited to two-thirds the value given for Emergency conditions in the Design Specification.

NOTE 10 - When loads are transiently applied, consideration should be given to the use of dynamic load amplification and possible change in modulus of elasticity.

TABLE C.4.7

CORE SUPPORT STRUCTU

STRESS CATEGORIES AND LIMITS OF ST

FOR FAULTED CONDITION

STRESS CATEGORIES	PRIMARY STRESSES	
	Membrane P_m (Notes 1,2&3)	Bending P_b (Notes
Faulted (Note 9)	$\boxed{P_m}$ $2.4\ S_m$ Elastic Analysis Or $.75S_u$ (Note 5) Or $1.33L_L$ Limit Analysis (Note 4) Or $.67S_u$ Plastic Analysis (Notes 5&6 Or $.8L_F$ Tests (Note 7) Or S_F Stress-Ratio Analysis (Note 8)	$\boxed{P_m + P_B}$ $3.$[illegible]S_m Ela Ana Or $1.33L_L$ Or $.75S_u$ Pla Anal (Not Or $.8L_F$ Test (Not Or KS_F Stre Rati Anal (Note

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RES PPORT STRUCTURES

TRESS D LIMITS OF STRESS INTENSITY

NS LTED CONDITIONS

	SECONDARY STRESSES	PEAK STRESSES
1,2, ing P_b (Notes 1,2,&3)	Membrane & Bending Secondary-Q	Peak F
P_B asti alys $3.0S_m$ Elastic Analysis r $1.33L_I$ r asti lys tes $.75S_u$ Plastic Analysis (Notes 5&6) r ts te $.8L_F$ Tests (Note 7) r ess io ys te KS_F Stress-Ratio Analysis (Note 8)	Evaluation Not Required	Evaluation Not Required

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TABLE C.4.7 NOTES

FAULTED CONDITIONS

NOTE 1 - The symbols P_m, P_b, Q, and F do not represent quantities but rather sets of six quantities representing the six stress components, σ_t, σ_l, σ_r, τ_{tl}, τ_{lr}, and τ_{rt}.

NOTE 2 - When loads are transiently applied, consideration should be given to the use of dynamic load amplification and possible changes in modulus of elasticity.

NOTE 3 - For configurations where compressive stresses occur, the stress limits shall be revised to take into account critical buckling stresses. For external pressure, the permissible "equivalent static" external pressure shall be taken as 2.5 times that given by the rules of par. N-417.8 of ASME Boiler and Pressure Vessel Code Section III. Where dynamic pressures are involved, the permissible external pressure shall satisfy the preceding requirements or shall be limited to 75% of the dynamic instability pressure.

NOTE 4 - L_L is the lower bound limit load with yield point equal to 1.5 S_m (where S_m is the tabulated value of allowable stress intensity at temperature as contained in ASME III). The "lower bound limit load" is here defined as that produced from the analysis of an ideally plastic (non-strain hardening) material where deformations increase with no further increase in applied load. The lower bound load is one in which the material everywhere satisfies equilibrium and nowhere exceeds the defined material yield strength using either a shear theory or a strain energy of distortion theory to relate multiaxial yielding to the uniaxial case.

NOTE 5 - S_u is the ultimate strength at temperature. Multiaxial effects on ultimate strength shall be considered.

NOTE 6 - This plastic analysis uses an elastic-plastic evaluated nominal primary stress. Strain hardening of the material may be used for the actual monotonic stress-strain curve at the temperature of loading or any approximation to the actual stress-strain curve which everywhere has a lower stress for the same strain as the actual monotonic curve may be used. Either the maximum shear stress or strain energy of distortion flow rule shall be used to account for multiaxial effects.

TABLE C.4.7 NOTES (Continued)

NOTE 7 - For Faulted Conditions, the stress limits need not be satisfied if it can be shown from the test of a prototype or model that the specified loads (dynamic or static equivalent) do not exceed 80% of L_F, where L_F is the ultimate load or load combination used in the test. In using this method, account shall be taken of the size effect and dimensional tolerances as well as differences which may exist in the ultimate strength or other governing material properties of the actual part and the tested parts to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading for Faulted Condition.

NOTE 8 - Stress ratio is method of plastic analysis which uses the stress ratio combinations (combination of stresses that consider the ratio of the actual stress to the allowable plastic or elastic stress) to compute the maximum load a strain hardening material can carry. K is defined as the Section Factor; S_F is the lesser of 2.4 S_m or 0.75 S_u for primary membrane loading.

NOTE 9 - Where deformation is of concern in a component, the deformation shall be limited to 80% of the value given for Faulted Conditions in the Design Specifications.

QUESTION 5.20

State the design differential pressure across the floor at elevation 236'00" (Figure 5.2.1) under loss-of-coolant accident conditions. Also, indicate what initial and subsequent testing of the floor with regard to strength and leakage will be performed, and how it will be carried out. Specify the maximum allowable leakage which will not result in overpressure of either upper or lower compartments and state the allowable design leakage through the floor. Include the method and assumptions by which these leakage values are determined.

ANSWER:

The design pressure conditions of 55 psig in the drywell and 25 psig in the suppression chamber have been selected based upon design basis loss-of-coolant accident. These conditions define a 30 psi design differential pressure across the drywell floor slab which envelopes the most severe loading of this component.

Structural and pressure integrity tests are to be performed prior to plant operation and additional pressure integrity tests may be made subsequently during plant shutdown. Initial tests are to be conducted at 115% of the following design conditions:

a. A design pressure condition of 55 psig in both the drywell and suppression chamber.

b. A design pressure condition of 55 psig in the drywell and 25 psig in the suppression chamber.

The differential pressure test of the drywell floor slab described in item (b) above is to be accomplished by capping the downcomers above the drywell floor slab upper surface.

A liner plate, as described in the answer to Question 5.21, has been added to the upper surface of the drywell floor slab. This liner plate is of the same material and meets the same quality assurance requirements as the liner plate at the primary containment boundary. Details of the drywell floor slab liner plate are shown in PSAR Supplement 3, Figure Q5.20.1, dated June 1971. Based upon the information supplied in response to Question 14.12 submitted in PSAR Supplement No. 5, it can be concluded that only large drywell floor slab bypass (break) areas (exceeding about 2 square feet in area) will result in overpressurization of the containment. Bypass areas approaching this magnitude will be detected during the structural and pressure integrity test described above. Therefore, no quantitative leakage tests across the drywell floor slab will be performed.

However periodic low differential pressure (less than 4 psid) tests will be conducted subsequent to initial startup to insure no gross leakage path exists between the drywell and suppression chamber gas space.