Offshore Power Systems

1st ACCEPTANCE REVIEW
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2 Plant Design Report

Offshore Power Systems

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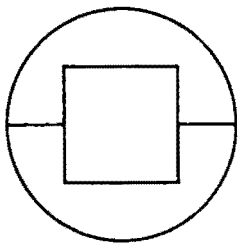

Plant Design Report

JANUARY 1973

PLANT DESIGN REPORT

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CHAPTER 3

DESIGN CRITERIA - STRUCTURES COMPONENTS AND SYSTEMS

3.1 RESPONSE TO AEC GENERAL DESIGN CRITERIA

CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection and testing of structures, systems and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

DISCUSSION

The structures, systems and components of the plant are classified according to their importance in the prevention and mitigation of accidents using the classification system developed by the American Nuclear Society. Each component is given a safety designation which identifies the codes, standards, quality control and quality assurance program applicable to each type of component and its class designation. Section 3.2 gives the referenced information for each component. Code applicability is based on 10 CFR 50.55a.

The Offshore Power Systems Quality Assurance Program described in Chapter 17 conforms with the requirements of 10 CFR 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants. Chapter 14 describes initial plant testing to assure performance of installed equipment commensurate with the importance of the safety function. Appropriate records of design, fabrication, construction and testing of safety components are defined in Chapter 17 and will be maintained in the possession of the applicant.

CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems and components shall reflect: (1) appropriate consideration of the most severe of natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the nautral phenomena and (3) the importance of the safety functions to be performed.

DISCUSSION

Structures, systems and components important to safety will be designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions.

Because the plant is a standardized design intended for use at many sites, it is designed to withstand the levels of natural phenomena which are discussed in Chapter 2.

Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena are considered in the design. The combinations used are described in the detailed design bases for the specific structures and systems.

CRITERION 3 - FIRE PROTECTION

Structures, systems and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Non-combustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems and components.

DISCUSSION

Non-combustible and fire resistant materials are used wherever practical throughout the plant, particularly in areas containing critical portions of the plant such as containment structure, control room and components of the protection and engineered safety features systems.

Redundant components of safety-related systems will be designed and located to render acceptable the effect of fires or explosions. Fire barriers and underwriter approved mechanisms of sufficient capacity are provided to reduce the effects of a fire. Equipment and facilities for fire protection, including detection, alarm and control systems are provided to protect both the plant and personnel from fire or explosion and the resultant release of toxic vapors.

The Fire Protection System shall be designed such that a failure of any component of the system:

1. Will not cause a nuclear accident or significant release of radioactivity to the environment.

2. Will not impair the ability to safely shutdown the reactor or limit the release of radioactivity to the environment in the event of a postulated accident.

The Fire Protection System is described in Section 9.5.1.

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing and postulated accidents, including loss-of-coolant accidents. These structures, systems and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit.

DISCUSSION

Structures, systems and components important to safety are specified and designed to accommodate the effects of and to be compatible with the pressure, temperature, humidity, and radiation conditions associated with normal operation, maintenance, testing, and postulated accident, including loss-of-coolant accidents in the area in which they are located.

Protective walls and slabs, local missile shielding, or restraining devices are provided to protect the containment and systems important to safety within the containment against damage from missiles generated by equipment failures. The containment structures serve as radiation shields and as missile barriers. A separate missile barrier is provided for the control rod drive mechanisms. Penetrations and piping extending to and including isolation valves are protected from damage due to pipe whip and external missiles, where such protection is necessary to meet the design basis.

Non-nuclear safety class piping will be arranged or restrained so that any failure of this piping will cause neither a nuclear accident nor loss of function of systems, components and structures important to safety.

Safety Class 1, 2, and 3 piping will be arranged or restrained such that in the event of a rupture, resulting pipe movement will not result in loss of containment integrity, unacceptable damage to the platform, or loss of function of the minimum systems, structures and components important to safety that are required to control the consequences of the rupture.

The containment structures are designed to sustain dynamic loads which could result from failure in major equipment and piping, such as jet thrust, jet impingement, and local pressure transients.

The containment and safety related structures are designed to withstand the effects of missiles as indicated in Chapter 3.

CRITERION 10 - REACTOR DESIGN

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

DISCUSSION

The reactor core and associated coolant, control, and protection systems are designed with adequate margins to:

1. Preclude excessive fuel damage during normal core operation and operational transients (Condition I) * or any transient conditions arising from occurrences of moderate frequency (Condition II).

2. Ensure return of the reactor to a safe state following a Condition III fault with only a small fraction of fuel rods damaged although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time.

3. Assure that the core is intact with acceptable heat transfer geometry following transients arising from occurrences of limiting faults (Condition IV).

* Definitions of Conditions I - IV are found in ANS-18.2.

Chapter 4.0 discusses the design bases and design evaluation of reactor components including the fuel, reactor vessel internals, and reactivity control systems. Details of the Control and Protection Systems Instrumentation design and logic are discussed in Chapter 7.0. This information supports the accident analyses of Chapter 15.0 which show that acceptable fuel design limits are not exceeded for Condition I and II occurrences.

CRITERION 11 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

DISCUSSION

Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the non-positive operational limit on moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the inherent design using low-enrichment fuel; the non-positive moderator temperature coefficient of reactivity is assured by administratively limiting the dissolved absorber concentration.

The core inherent reactivity feedback characteristics are described in Section 4.3, Nuclear Design. Reactivity control by chemical injection is discussed in Section 4.2.3, Reactivity Control Systems, and Section 9.3.4, Chemical and Volume Control System.

CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible and can be reliably and readily detected and suppressed.

DISCUSSION

Power oscillations of the fundamental mode are inherently eliminated by the negative Doppler and non-positive moderator temperature coefficient of reactivity.

Oscillations, due to xenon spatial effects, in the radial, diametral and azimuthal overtone modes are heavily damped due to the inherent design and due to the negative Doppler and non-positive moderator temperature coefficients of reactivity.

Oscillations, due to xenon spatial effects, in the axial first overtone mode may occur. Assurance that fuel design limits are not exceeded by xenon axial oscillations is provided as a result of reactor trip functions using the measured axial power imbalance as an input.

Oscillations, due to xenon spatial effects, in axial modes higher than the first overtone, are heavily damped due to the inherent design and due to the negative Doppler coefficient of reactivity.

The stability of the core against xenon-induced power oscillations and the function requirements of instrumentation for monitoring and measuring core power distribution are discussed in Section 4.3, Nuclear Design.

CRITERION 13 - INSTRUMENTATION AND CONTROL

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

DISCUSSION

Plant Instrumentation and Control Systems are provided to monitor variables in the reactor core, coolant system, and containment building over their predicted range for all conditions to the extent required. The installed instrumentation provides continuous monitoring, warning, and initiation of safety functions. The following processes are controlled to maintain key variables within their normal range:

1. Reactor power level (manual or automatic by control of thermal load).

2. Reactor coolant temperature (manual or automatic by control of rod position).

3. Reactor coolant pressure (manual or automatic by control of heaters and spray in the pressurizer).

4. Reactor coolant water inventory, as indicated by pressurizer water level (manual or automatic control of charging flow).

5. Reactor axial power balance (manual control of part-length rod position).

6. Reactor Coolant System boron concentration (manual or automatic control of makeup charging flow).

7. Steam generator water inventory on secondary side (manual or automatic control of feedwater flow).

The Reactor Control System is designed to automatically maintain a programmed average temperature in the reactor coolant during steady-state operation and to ensure that plant conditions do not reach reactor trip settings, as the result of a design load change.

The Reactor Protection System trip setpoints are selected so that anticipated transients do not cause a DNBR of less than 1.3

Proper positioning of the control rods is monitored in the control room by bank arrangements of individual meters for each rod cluster control assembly. A rod deviation alarm alerts the operator of a deviation of one rod cluster control assembly from its bank position. There are also insertion limit

monitors with visual and audible annunciation to avoid loss of shutdown margin. Each rod cluster control assembly is provided with a sensor to detect positioning at the bottom of its travel. This condition is also alarmed in the control room. Four ex-core long ion chambers also detect asymmetrical flux distribution indicative of rod misalignment.

Movable in-core flux detectors and fixed in-core thermocouples are provided as operational aids to the operator. Chapter 7 contains further details on instrumentation and controls. Information regarding the radiation monitoring system provided to measure environmental activity and alarm high levels is contained in Chapter 11.

Overall reactivity control is achieved by the combination of soluble boron and rod cluster control assemblies. Long-term regulation of core reactivity is accomplished by adjusting the concentration of boric acid in the reactor coolant. Short-term reactivity control for power changes is accomplished by the reactor control system which automatically moves rod cluster control assemblies. This system uses input signals including neutron flux, coolant temperature and turbine load.

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

DISCUSSION

The reactor coolant boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of unit operation including all anticipated transients and maintain the stresses within applicable stress limits. The design criteria, methods, and procedures applied to components of the reactor coolant pressure boundary are discussed in Section 5.2.1.

In addition to the loads imposed on the system under normal operating conditions, consideration is also given to abnormal loading conditions such as seismic and pipe rupture. Fracture prevention measures are included to prevent brittle fracture. Refer to the discussion under Criterion 31 for additional information.

The system is protected from overpressure by means of pressure-relieving devices as required by applicable codes.

The materials of construction of the pressure retaining boundary of the reactor coolant system are protected from corrosion by control of coolant chemistry which might otherwise reduce the system structural integrity during its lifetime. The pressure boundary has provisions for inspection, testing and surveillance of critical areas to assess the structural and leaktight integrity of the boundary.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

The Reactor Coolant System and associated auxiliary, control and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

DISCUSSION

The Reactor Coolant System and associated auxiliary, control and protection systems are designed to ensure the integrity of the reactor coolant pressure boundary with adequate margins during normal operation and during Condition I and Condition II transients. The system boundary accommodates loads due to the Operating Basis Earthquake during normal operation including normal operational transients (Condition II) within code stress limits for upset conditions. The system boundary accommodates loads due to the Design Basis Earthquake combined with loads due to piping failures such as circumferential pipe ruptures of reactor coolant pipes at junctures with equipment nozzles and connecting pipes at junctures to reactor coolant piping (Condition IV) without propagation of failure to remaining Reactor Coolant System loops, Steam Power Conversion System, or other piping or equipment needed for emergency cooling. The components of the reactor coolant system and associated fluid systems are designed in accordance with appropriate ASME and ANSI codes. These codes are identified in Chapter 3.2. The Protection System is designed in accordance with IEEE - Standard 279. The

protection system analyses are given in Subsection 7.2.2. Overpressure protection is provided by automatic controls, pressure relief valves, and spring loaded safety valves.

The selected design margins include operating transient changes due to thermal lag, coolant transport times, pressure drops, system relief valve characteristics and instrumentation and control response characteristics.

CRITERION 16 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

DISCUSSION

Containment structures, composed of a steel containment shell, a steel base plate, a reinforced concrete shield building, and interior steel and reinforced concrete structures, are provided which enclose the entire Reactor Coolant System. These structures are designed to sustain, without loss of required integrity, all effects of gross piping failures up to and including the double-ended rupture of the largest pipe in the Reactor Coolant System. Engineered safety features, comprising the Emergency Core Cooling System, the Containment Spray System, the Annulus Filtration System, the Ice Condenser, Air Return Fans, the Containment Isolation System, the Containment Hydrogen Recombiner System, and the Containment Hydrogen Venting System, then serve to cool the reactor core and return the containment to near atmospheric pressure. The containment systems and their associated engineered safety features are designed to assure the required functional capability of containing the release of radioactivity and the design conditions important to safety will not be exceeded for as long as postulated accident conditions require.

Periodic leak rate measurements ensure that the leaktight barrier is maintained. Containment structural design criteria can be found in Section 3.8.

CRITERION 17 - ELECTRICAL POWER SYSTEMS

An on-site electric power system and an off-site electric power system shall be provided to permit functioning of structures, systems and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The on-site electric power supplies, including the batteries, and the on-site electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the on-site electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all on-site alternating current power supplies and the other off-site electric power circuit, to assure that specified acceptable fuel design limit and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss-of-coolant accident to assure that the core cooling, containment integrity and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the on-site electric power supplies.

DISCUSSION

The off-site power sources will consist of two cable circuits to the mainland substation either of which will have the capacity to supply all the

protection and safety systems. Two full sized station auxiliary transformers which supply power for all operating conditions are provided, one connected to each cable circuit.

Both transformers are normally in service thus both circuits are available within a few seconds following a loss-of-coolant accident.

The on-site electric power sources including batteries, diesels, ac and dc distribution system will have sufficient independence, redundancy and testability to assure a power supply to safety and protective related functions even when such supply systems are subjected to the single failure criteria.

These systems are designed in accordance with IEEE No. 308 and are discussed in Chapter 8.

CRITERION 18 - INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically: (1) the operability and functional performance of the components of the systems, such as on-site power sources, relays, switches and buses; and (2) the operability of the systems as a whole, and under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the off-site power, and the on-site power system.

DISCUSSION

The essential service power supply buses and associated diesel generators are arranged for periodic testing of each system independently. The testing procedure will simulate a loss of bus voltage to start the diesel and connect it to the bus. Full load testing of the diesel can be performed by manually synchronizing to the normal power supply. These tests will prove the operability of the power supply system under conditions as close to design as practical to assess the continuity of the system and condition of the components.

The condition of the station batteries will be periodically monitored by checking and recording specific gravity and voltage of individual cells. The test records, over a period of time, should reveal a weak or weakening trend in any cell. Chapter 8.0 discusses the details of the Electric Power Systems.

CRITERION 19 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment in appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

DISCUSSION

Following proven power plant design philosophy, all control stations, switches, controllers and indicators necessary to operate and shut down the nuclear unit and maintain safe control of the plant will be located in the control room.

The design of the control room will permit safe occupancy during abnormal conditions. Shielding will be designed to maintain tolerable radiation exposure levels (see Section 12.1) in the control room for hypothetical accident conditions, including loss-of-coolant accidents.

The control room will be protected against unacceptable radiation levels by radiation equipment alarm and appropriate ventilation interlock circuitry.

Provisions will be made for control room air to be recirculated through high efficiency particulate and charcoal filters. Emergency lighting will be provided.

The control room will be continuously occupied by qualified operating personnel under all operating and accident conditions, except for the following. In the unlikely event that access to the control room is restricted, local control stations, or manual operation of critical components can be used to effect hot shutdown for an extended period of time from outside the control room. By use of appropriate procedures and equipment, the unit can also be brought to cold shutdown conditions. For this latter operation, it is assumed that off-site power will be available.

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed:

1. To initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

2. To sense accident conditions and to initiate the operation of systems and components important to safety.

DISCUSSION

The operational limits for the Reactor Protection System are defined by analyses of plant operating and fault conditions requiring rapid rod insertion to prevent or limit core damage. With respect to acceptable fuel design limits, the system design bases for anticipated operational occurrences are:

a. Minimum departure from nucleate boiling ratio (DNBR) shall not be less than 1.3.

b. Clad strain on the fuel element shall not exceed 1%.

c. No center melt shall occur in the fuel elements.

A region of permissible core operation is defined in terms of power, axial power distribution, and coolant pressure, flow and temperature. The protection system monitors these process variables (as well as other process

variables and plant conditions). If the limits of the region are approached during operation, the protection system will automatically actuate alarms, initiate load cutback, prevent control rod withdrawal, or trip the reactor depending on the severity of the condition.

Operation within the permissible region and complete core protection is assured by the overtemperature ΔT and overpower ΔT reactor trips in the system pressure range defined by the pressurizer high pressure and pressurizer low pressure reactor trips in the event of a transient which is slow with respect to piping delays from the core in the temperature sensors. In the event that a transient faster than the ΔT responses occurs, high nuclear flux and low coolant flow reactor trips provide core protection. Finally, thermal transients are anticipated and avoided by reactor trips initiated by turbine trip and primary coolant pump circuit breaker position.

The Protection System operates by interrupting power to the rod control power supply. All full length control and shutdown rods insert by gravity as a result. A Westinghouse Protection System design meets the requirements of IEEE 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations."

The Protection System measures a wide spectrum of process variables and plant conditions. All analog channels which actuate reactor trip, rod stop, and permissive functions are indicated or recorded. In addition,

visual and/or audible alarms are actuated for a reactor trip, partial reactor trip (any input channel), and any control variable exceeding its set point (any input channel). These measurements and indications provide the bases for corrective action to prevent the development of accident conditions. In the event of an accident condition, however, the reactor protection system will sense the condition, process the signals used for engineered safety features actuation, and generate the actuation demand. The conditions leading to engineered safety features actuation are:

a. Low pressurizer pressure coincident with low pressurizer water level

b. High containment pressure

c. High-high containment pressure

d. High steam line flow coincident with low steam line pressure or low-low average temperature

e. High steam line differential pressure

f. Manual.

The reactor trip system is discussed in Section 7.2 and the engineered safety features actuation system is duscussed in Section 7.3 of this report.

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

DISCUSSION

The protection system is designed to comply with IEEE-279, IEEE Criteria for Nuclear Power Generating Station Protection Systems. It provides high functional reliability and adequate independence, redundancy, and testability commensurate with the safety functions of the system. Actuation circuitry is provided with a capability of on-line testing without the loss of the minimum required redundancy. This extends to the final actuating device except where operational requirements prohibit actual operation of the device, e.g., turbine trip, steam line isolation, etc.

The Reactor Protection System is designed for high functional reliability by providing electrically isolated and physically separated, redundant analog channels and separate and independent trip logic trains. This assures that no single failure will result in the loss of any protection

function. Except for certain defined backup trip functions detailed in Chapter 7, the redundancy and independence provided in the Reactor Protection System allows individual channel test or calibration to be made during power operation without negating reactor protection or the single failure criterion. This testing will determine failures and losses of redundancy that may have occurred. This arrangement also permits removal from service of a channel while still maintaining the high reliability of the protection function. Details of the protection system design and testing provisions are contained in Chapter 7.

There are two series-connected circuit breakers through which power is supplied to the full-length rod drive mechanisms. A reactor trip signal is fed to the undervoltage coils of both breakers simultaneously and opening of either breaker will trip the reactor.

The Engineered Safety Features Actuation System is also designed to meet IEEE-279 requirements. It also utilizes redundant analog channels measuring the same parameter and redundant logic trains, which will actuate safety injection and/or containment spray.

The Engineered Safety Features Actuation System is testable at power with certain exceptions as detailed in Chapter 7. As with the components of the Reactor Protection System, both physical and electrical separation is provided for the Engineered Safety Features Actuation System to provide a high degree of availability for its safety function.

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

DISCUSSION

Protection system components are designed and arranged so that the environment accompanying any emergency situation in which the components are required to function does not result in loss of the safety function. Various means are used to accomplish this. Functional diversity, physical separation, electrical isolation and safe component and subsystem failure modes have been designed into the system. The extent of this functional diversity has been evaluated for a wide variety of postulated accidents. Generally, two or more diverse protection functions would automatically terminate an accident before unacceptable consequences could occur.

For example, there are automatic reactor trips based upon nuclear flux measurements, reactor coolant loop temperature measurements, pressurizer pressure and level measurements, and reactor coolant pump under frequency and under voltage measurements, as well as manually, and by initiation of a safety injection signal.

Regarding the Engineered Safety Features Actuation System for a loss-of-coolant accident, a safety injection signal can be obtained manually or by automatic initiation from two diverse sets of signals:

1. Coincident low pressurizer pressure and water level.
2. High containment pressure.

For a steam break accident, diversity of safety injection signal actuation is provided by:

1. High steam line flow coincident with low steam line pressure or low-low T_{avg}.

2. High steam line differential pressure.

3. For a steam break inside containment, high containment pressure provides an additional parameter for generation of the signal.

All of the above sets of signals are redundant and physically separated and meet the intent of the criteria.

High quality components, suitable derating and applicable quality control, inspection, calibration and tests are utilized to guard against common mode failure. Qualification testing is performed on the various safety systems

to demonstrate satisfactory operation at normal and post-accident conditions of temperature, humidity, pressure and radiation. Typical protection system equipment is subjected to type tests under simulated seismic conditions using conservatively large accelerations and applicable frequencies.

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

DISCUSSION

The protection system is designed with due consideration of the most probable failure modes of the components under various perturbations of energy sources and the environment.

Each reactor trip channel is designed on the de-energize to trip principle so that a loss of power or disconnection of the channel causes that channel to go into its tripped mode. In addition, a loss of power to the full-length rod cluster control assembly drive mechanisms causes them to insert by gravity into the core and trip the reactor.

In the event of a loss of the preferred off-site power source, on-site diesel generators will be available to power emergency loads and the station batteries to power the vital instrumentation loads. The diesels are capable of supplying power to the emergency core cooling equipment and associated valves. A loss of power to one train of emergency core cooling equipment will not affect the ability of the system to perform its function.

The protection system components have been tested and qualified for the extremes of the normal environment to which subjected. In addition, components are tested and qualified according to individual requirements for the adverse environment specific to the location which might result from post-accident conditions.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

The protection system shall be separated from control systems to the extent that failure of a single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

DISCUSSION

Protection and control channels in the facility protection systems will be designed in accordance with the IEEE-279, "IEEE Criteria for Nuclear Power Plant Protection Systems".

The Reactor Protection System itself is designed to maintain separation between redundant protection logic trains. Separation of redundant analog channels originates at the process sensors and continues along the wiring route and through containment penetrations to analog protection racks terminating at the Reactor Protection System logic racks. Isolation of wiring is achieved using separate wireways, cable trays, conduit runs and containment penetrations for each redundant channel. Analog equipment is separated by locating components associated with redundant functions in different protection racks.

Each redundant protection channel set is energized from a separate ac power feed.

The redundant (two) reactor trip logic trains are physically separated from one another. The Reactor Protection System is comprised of identifiable channels which are physically separated and electrically isolated.

Channel independence is carried throughout the system from the sensor to the logic interface. In some cases, however, it is advantageous to employ control signals derived from individual protection channels through isolation amplifiers contained in the protection channel. As such, a failure in the control circuitry does not adversely affect the protection channel.

The protection and control functions are thus separate and distinct. The design provides that failure of any single control system component or channel including any short or ground or applying available ac or dc voltages to the control side (output) of the isolation amplifier will not perceptibly disturb the protection side (input) of the amplifier.

The electrical supply and control conductors for redundant or backup circuits have such physical separation as is required to assure that no single credible event will prevent operation of the associated function by reason of electrical conductor damage.

Critical circuits and functions include power, control and analog instrumentation associated with the operation of reactor protection, engineered safety features, reactor and residual heat removal systems. The reactor protection system and the control systems are discussed in Sections 7.2 and 7.7, respectively.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

DISCUSSION

The protection system design assures that acceptable fuel design limits are not exceeded in the event of single reactivity control malfunction including accidental withdrawal of control rod groups. Reactor shutdown with control rods is completely independent of the control functions. The trip breakers will interrupt power to the full-length rod drive mechanisms to trip the reactor regardless of existing control signals.

The reactivity control systems, which are discussed further in Chapter 7, are also discussed in Criterion 10.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate fuel margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

DISCUSSION

Two independent reactivity control systems of different design principles are provided. One of the systems uses control rods; the second system employs dissolved boron (chemical shim).

Three functional categories of rods are employed. These categories are; (1) Full-length shutdown; (2) Full-length control; and (3) Part-length control. During operation the shutdown rod banks are fully withdrawn. The full-length control rod system automatically maintains a programmed average reactor temperature compensating for reactivity effects associated with scheduled and transient load changes. The part length control rods are used to shape the reactor axial power profile and maintain the axial peaking factor within design limits. The neutron absorber material used in the control rods is a silver-indium-cadmium alloy designed to shutdown

the reactor with adequate margin under conditions of normal operation and anticipated operational occurrences thereby ensuring that specified fuel design limits are not exceeded. The most restrictive period in core life is assumed in all analyses and the most reactive rod cluster is assumed to stick in the out of core position. The reactor protection system initiates reactor trip by interrupting power to the rod control power supply. This releases the magnetic latches and the full-length control and shutdown rods insert by gravity. The design is inherently fail-safe. The part length rods are not tripped during shutdown.

The boron system is capable of controlling the rate of reactivity change resulting from planned normal power changes including xenon burnout to assure fuel design limits are not exceeded. This system is capable of maintaining the reactor core subcritical under cold conditions with all rods withdrawn.

The control rod system and boron system are discussed in Sections 4.2.3 and 9.3.4, respectively.

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEM CAPABILITY

The reactivity control systems shall be designed to have a combined capability in conjunction with poison addition by the Emergency Core Cooling System, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

DISCUSSION

The Nuclear Steam Supply System is provided with the means of making and holding the core subcritical under any anticipated conditions and with appropriate margin for contingencies. These means are discussed in detail in Chapters 4 and 9.

Combined use of rod cluster control and chemical shim control permits the necessary shutdown margin to be maintained during long-term xenon decay and plant cooldown. The single highest worth control cluster is assumed to be stuck fully withdrawn upon reactor trip. Under accident conditions when the emergency core cooling system is actuated, concentrated boric acid is injected into the Reactor Coolant System. Reactivity effects of emergency core cooling are discussed in Section 6.3 and evaluated for accident conditions in Chapter 15.

CRITERION 28 - REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

DISCUSSION

Core reactivity is controlled by a chemical poison dissolved in the coolant, rod cluster control assemblies and burnable poison rods. The maximum reactivity insertion rates due to withdrawal of a bank of rod control cluster assemblies or by boron dilution are limited. These limits are set such that the peak heat generation rate and DNBR do not exceed the specified limits at overpower conditions. The maximum worth of control rods and the maximum rates of reactivity insertion employing control rods are limited to values which prevent rupture of the coolant pressure boundary or disruption of the core internals to a degree which would impair core cooling capability.

The reactor can be brought to the shutdown condition, and the core will maintain acceptable heat transfer geometry following a condition 4 event,

such as rod ejection or steam line break. Specifically, faulted condition stress limits are used to contain within tolerable limits the effects of extremely unlikely events.

The reactivity control systems are discussed in RESAR 3, Sections 4.2.3 and 4.3.

CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

DISCUSSION

The protection and reactivity control systems are designed to perform their required safety functions in the event of anticipated operational occurrences. Redundancy functional and spatial diversity, testability, use of safe failure modes, and analyses are design measures which are employed to assure performance of the required safety functions. Analyses of the systems verify this reliability. The protection system is further discussed under criteria 20 through 25 and in Section 7.2. The reactivity control systems are discussed in Sections 4.2.3 and 7.7.

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

DISCUSSION

Reactor coolant pressure boundary components are designed, fabricated, inspected and tested in conformance with ASME Nuclear Power Plant Components Code Section III.

Major components are classified as ANS N18.2 Safety Class 1 and are accorded the quality measures appropriate to this classification.

Leakage is detected by an increase in the amount of makeup water required to maintain a normal level in the pressurizer. The reactor vessel closure joint is provided with a temperature monitored leakoff between double gaskets.

Leakage inside the reactor containment is drained to the containment sump where it is monitored.

Leakage is also detected by measuring the airborne activity and activity of the condensate drained from the containment air recirculation units.

Monitoring the inventory of reactor coolant in the system at the pressurizer, volume control tank, and coolant drain collection tanks make available an accurate indication of integrated leakage. The reactor coolant pressure boundary leakage detection system is discussed in Subsection 5.2.4.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

DISCUSSION-

Close control is maintained over material selection and fabrication for the Reactor Coolant System to assure that the boundary behaves in a non-brittle manner. The Reactor Coolant System materials which are exposed to the coolant are corrosion resistant stainless steel or Inconel. The nil ductility transition temperature of reactor vessel material samples is established by Charpy V-notch and drop weight tests. These tests insure the selection of materials with proper toughness properties and margins as well as to verify the integrity of the reactor coolant pressure boundary.

As part of the reactor vessel specification, certain tests in addition to those which are specified by the applicable ASME codes are performed. These tests are listed below:

1. Ultrasonic Testing - Westinghouse requires the performance of 100 percent volumetric ultrasonic test of reactor vessel plate for shear wave and a post-hydro test ultrasonic map of all welds in the pressure vessel. Also, Westinghouse requires cladding bond ultrasonic inspection to more restrictive requirements than code in order to preclude interpretation problems during inservice testing.

2. Radiation Surveillance Program - In the surveillance programs, the evaluation of the radiation damage is based on pre-irradiation and post-irradiation testing of Charpy V-notch and tensile specimens.

   These programs are directed toward evaluation of the effects of radiation on the fracture toughness of reactor vessel steels based on the transition temperature approach and the fracture mechanics approach, and is in accordance with ASME-E-185-70, "Recommended Practice for Surveillance Tests for Nuclear Reactor Vessels."

The fabrication and quality control techniques used in the fabrication of the Reactor Coolant System are equivalent to those for the reactor vessel. The inspections of reactor vessel, pressurizer, reactor coolant pump casings, piping and steam generator are governed by ASME code requirements.

Administrative controls are placed on plant heatup and cooldown rates, using conservative values for the change in nil-ductility transition

temperature due to irradiation to control vessel stresses below acceptable levels over the life of the plant while considering both allowable and postulated flaws.

Details of the various aspects of the design and testing processes are included in Chapter 5.

CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

DISCUSSION

The design of the reactor vessel and its arrangement in the system provides the capability for accessibility during service life to the entire internal surfaces of the vessel and certain external zones of the vessel including the nozzle to reactor coolant piping welds and the top and bottom heads. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the area of pipe within the primary shielding concrete. The inspection capability complements the leakage detection systems in assessing the pressure boundary component's integrity.

Monitoring of the Nil Ductility Transition Temperature properties of the core region plates, forgings, weldments, and associated heat affected zones are performed in accordance with ASTM E-185 (Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors). Samples of reactor vessel plate materials are retained and cataloged in case future engineering development shows the need for further testing.

The material properties surveillance program includes not only the conventional tensile and impact tests, but also fracture mechanics specimens. The observed shifts in NDTT of the core region materials with irradiation will be used to confirm the calculated limits on heatup and cooldown transients.

To define permissible operating conditions below NDTT, a pressure range is established which is bounded by a lower limit for pump operation and an upper limit which satisfies reactor vessel stress criteria. To allow for thermal stresses during heatup or cooldown of the reactor vessel, an equivalent pressure limit is defined to compensate for thermal stress as a function of rate of change of coolant temperature. Since the normal operating temperature of the reactor vessel is well above the maximum expected NDTT, brittle fracture during normal operation is not considered to be a credible mode of failure. Additional details can be found in Section 5.2.

CRITERION 33 - REACTOR COOLANT MAKEUP

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available) the system safety function can be accomplished using the piping, pumps and valves used to maintain coolant inventory during normal reactor operation.

DISCUSSION

The Chemical and Volume Control System provides a means of reactor coolant makeup and adjustment of the boric acid concentration. Makeup is added automatically if the level in the volume control tank falls below a preset level. High pressure centrifugal charging pumps are provided which are capable of supplying the required makeup and reactor coolant seal injection flow with power available from either on-site or off-site electrical power systems. The safety injection system pumps will start automatically if the loss of coolant is greater than the capacity of normal makeup. These pumps are started from low pressurizer level and pressure. A high degree of functional reliability is assured by provision of standby components and assuring safe response to probable modes of failure. Details of system design are included in Chapters 6 and 9.

CRITERION 34 - RESIDUAL HEAT REMOVAL

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at the rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundance in components and features, and suitable interconnections, leak detection and isolation capabilities shall be provided to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available) the system safety function can be accomplished assuming a single failure.

DISCUSSION

The Residual Heat Removal System, in conjunction with the steam and power conversion system, is designed to transfer the fission product decay heat and other residual heat from the reactor core.

Suitable redundancy is accomplished with the two residual heat removal pumps, located in separate compartments with means available for draining and monitoring leakage, two heat exchangers and the associated piping and cabling. The Residual Heat Removal System is able to accommodate a single failure as well as operate on either on-site or off-site electrical power. Details of the system design can be found in Chapter 5.

CRITERION 35 - EMERGENCY CORE COOLING

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

An Emergency Core Cooling System will be provided which satisfies the intent of the "Interim Policy Statement Criteria for Emergency Core Cooling Systems for Light Water Power Reactors". The emergency core cooling system is described in Section 6.3 of this report.

CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

The Emergency Core Cooling System shall be designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping to assure integrity and capability of the system.

DISCUSSION

Design provisions are made to facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes and valves for visual inspection and for non-destructive inspection where such techniques are desirable and appropriate, or required by codes.

The components located outside containment are accessible for leaktightness inspection during operation of the reactor. The pressure containing systems are inspected for leaks from pump seals, valve packing, flanged joints, and safety valves during system testing.

Details of the inspection program for the reactor vessel internals are included in Chapter 4. Inspection of the Emergency Core Cooling System is discussed in Chapter 6.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The Emergency Core Cooling System shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

DISCUSSION

Appropriate components of the system located outside the containment are accessible for periodic tests. Each active component of the Emergency Core Cooling System may be individually actuated on the normal power source or transferred to emergency power source at any time during plant operation to demonstrate operability. The test of the safety injection pumps employs the minimum flow recirculation test line which connects back to the refueling water storage tank. Remote operated valves are exercised and actuation circuits tested. The automatic actuation circuitry, valves and pump breakers also may be checked during integrated system tests performed during a planned cooldown of the Reactor Coolant System.

Design provisions also include special instrumentation, testing and sampling lines to perform the tests during plant shutdown to demonstrate proper automatic operation of the Emergency Core Cooling System. A test signal is

applied to initiate automatic action and verification is made that the ECCS pumps attain required discharge heads. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.

These tests are described in Section 6.3.4.

CRITERION 38 - CONTAINMENT HEAT REMOVAL

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnection, leak detection, isolation and containment capabilities shall be provided to assure that for on-site electrical power system operation (assuming off-site power is not available) and for off-site electrical power system operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

The Ice Condenser System, which is a passive heat absorption system, and the Containment Spray System, which is an active heat absorption and heat removal system, function as emergency heat removal systems. These systems provide the full heat removal capability required in the event of an uncontrolled loss of fluid from the reactor coolant system or steam system and maintain the pressure below the containment design pressure.

If a loss of normal ac power to the Containment Spray System occurs in coincidence with the postulated accident, the system is powered by on-site emergency power supplies. The Containment Spray System performs its function assuming any single active failure in either the system itself or the power supply to the system. The systems are described in Section 6.2.

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

DISCUSSION

Inspection of the Ice Condenser is carried out before and after ice loading and at intervals throughout the plant operating lifetime. Selected flow passages are inspected visually to confirm that no significant changes are occurring. Selected ice baskets can be weighed to confirm that no significant weight loss has occurred. Provision is made for visual inspection of door panels which can be manually opened momentarily, to confirm that the doors are functioning properly. These inspections, except inspection of the lower door panels can be performed while the reactor is in operation.

The Containment Spray System pumps and heat exchangers are located outside the containment and can be inspected at any time. The remainder of the Containment Spray System, including ring headers and spray nozzles, and the containment sump can be inspected during cold shutdown.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

DISCUSSION

System valves, pumps and heat exchangers are designed to permit periodic testing for operability and performance. System components are arranged to allow testing for structural and leaktight integrity.

The operational sequence which actuates the Containment Spray System, including the transfer to alternate power sources, can be tested. The spray nozzles are tested by discharging air through the nozzles during plant shutdown.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

Systems to control fission products, hydrogen, oxygen and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associate systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation and containment capabilities to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available) its safety function can be accomplished assuming a single failure.

DISCUSSION

There are two systems which are designed to clean up the containment atmosphere after a postulated loss-of-coolant accident.

1. The containment spray system sprays a sodium tetraborate solution into the upper containment compartment from the containment sump during the recirculation phase to remove elemental iodine.

2. A hydrogen recombination system and an independent hydrogen purge system will be provided to prevent the buildup of excessive concentrations of hydrogen in the containment following a loss-of-coolant accident.

The containment spray and hydrogen control systems have sufficient redundancy, suitable interconnections, leak detection, isolation and containment capabilities with either on-site electric or off-site electrical power available to assure that their safety functions can be accomplished assuming a single active failure.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts and piping to assure the integrity and capability of the systems.

DISCUSSION

The containment atmosphere cleanup systems, with the exception of the spray headers and nozzles, are designed and located such that they can be inspected periodically as required.

The post-accident containment venting systems are located outside the containment and are accessible for inspection; the hydrogen recombining system can be inspected during reactor shutdown. Details are found in Section 6.2.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system such as fans, filters, dampers, pumps and valves, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

DISCUSSION

Testability of the containment spray system is discussed under Criterion 40.

The post-accident containment combustible gas control system is designed to allow appropriate periodic testing of the system through the full operational sequence and as close to design conditions as practical, including transfer to alternate power sources, to demonstrate system integrity as well as operability and performance of system active components. For further discussion of the testing capability of this system see Section 6.2.

CRITERION 44 - COOLING WATER

A system to transfer heat from structures, systems and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer to combined heat load of these structures, systems and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection and isolation capabilities shall be provided to assure that for on-site electrical power system operation (assuming off-site power is not available) and for off-site electrical power system operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

The cooling water systems provided to transfer heat during accident conditions from safety related components and systems to the ocean (ultimate heat sink) are the Essential Service Water System, the Component Cooling Water System (CCW), the Essential Raw Water System (ERW), and the Auxiliary Raw Water System (ARW).

The ESW and CCW Systems are closed systems which remove heat from essential components important to safety. The ESW System consists of four (4) independent subsystems or trains. Each ESW train consists of a pump, heat exchanger, piping, valves and instrumentation required for system operation. The CCW consists of three (3) pumps and three (3) heat exchangers, piping, valves and instrumentation required for system operation.

The ERW System is an open seawater system which removes heat from the ESW System and the Containment Spray System. The ERW System consists of four (4) separate subsystems or trains. Each ERW train consists of a pump, piping, valves and instrumentation required to perform its function. The ARW System is an open seawater system which removes heat from the CCW System. The ARW System consists of three (3) pumps, piping, valves and instrumentation required to perform its function.

The subsystem arrangement of the ESW, CCW, ERW, and ARW systems permits the system safety functions to be performed assuming a single failure.

Leak detection is accomplished by routine surveillance and/or monitoring instrumentation.

Electrical power for the operation of each system may be supplied from off-site or on-site emergency power sources, with distribution arranged such that a single failure will not prevent the system from performing its safety function.

The ESW and ERW systems do not operate during normal plant operating conditions. The systems only function is to provide essential component cooling during accident conditions. The CCW and ARW transfer heat from components and systems related to safety during normal operation.

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

DISCUSSION

The important components of the Essential Service Water (ESW), Component Cooling Water (CCW), Essential Raw Water (ERW), and Auxiliary Raw Water (ARW) Systems are located in accessible areas. The system design permits periodic inspection of important components such as heat exchangers, pumps, valves and piping. Additional details are given in Section 9.2.

CRITERION 46 - TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole, and under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

DISCUSSION

The Essential Service Water (ESW), Component Cooling Water (CCW), Essential Raw Water (ERW) and Auxiliary Raw Water (ARW) systems are designed to permit appropriate periodic pressure and functional testing of the system as a whole, including the functional sequence that initiates system operation and the transfer between normal and emergency diesel power sources. Periodic functional and pressure testing can be conducted to determine the structural and leaktight integrity of the systems, as well as the performance of the active components of the system. System details are found in Section 9.2.

CRITERION 50 - CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

DISCUSSION

The containment structure, including access openings and penetrations, and the containment heat absorption and heat removal systems are designed to accommodate the transient peak pressure and temperature associated with a postulated uncontrolled loss of fluid from the reactor coolant system or steam system.

The containment pressure and temperature transients are conservatively evaluated for various combinations of energy release. The analysis includes consideration of the heat release from stored heat sources in the containment and chemical reactions in the Reactor Coolant System. Where an active heat removal system is employed in containment cooling, the system performs its required function assuming the most restrictive single failure of an active component within the system. Containment design details are found in Sections 3.8.2 and 6.2.

CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady-state and transient stresses, and (3) size of flaws.

DISCUSSION

The exposed ferritic material of the containment pressure boundary will be shown to have non-brittle behavior by passing impact testing in accordance with subsection NE-2320 of Section III, ASME Boiler and Pressure Vessel Code.

Variations of material property and stress states due to temperature change will be considered in the design. Design margins are discussed in Section 3.8.

CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

DISCUSSION

The containment vessel is designed so that a pre-operational integrated leakage rate test can be performed at design pressure after completion and installation of penetrations and equipment. This test will form a baseline for subsequent testing. Periodic retesting of the containment leakage rate will be conducted to verify the continued performance of the structure. The retests may be performed at design pressure or at reduced pressures in accordance with ANS 7.60, "Standard for Leakage Rate Testing of Containment Structures for Nuclear Reactors."

Details of the containment leakage testing are found in Section 6.2. The testing is in conformance with 10CFR Part 50, Appendix J.

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

DISCUSSION

Provisions are made for conducting individual leakage rate tests on applicable penetrations. Penetrations and resilient seals and bellows will be visually inspected and pressure tested for leak tightness periodically. Containment testing and inspection details can be found in Section 6.2.

To ensure continued integrity of the containment isolation system, periodic closure and leakage tests shall be performed to ascertain that leakage will be within specified limits.

CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

DISCUSSION

Piping penetrating the containment is designed to withstand at least a pressure equal to the containment maximum internal pressure. The design basis requires a double barrier on all of the above systems between the atmosphere outside the primary containment and the containment atmosphere. Valves isolating penetrations serving Engineered Safety Features will not automatically close with a containment isolation actuation signal, but may be closed by remote operation from the control room to isolate any Engineered Safety Feature when required. Positions of these remotely operated valves on loss of actuating power are indicated in Subsection 6.2.4. Piping systems penetrating the containment shell will be designed to allow periodic testing to assure that valve leakage remains within allowable limits and that valves remain operable.

CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

2. One automatic isolation valve inside and one locked closed isolation valve outside containment, or

3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment, or

4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

DISCUSSION

The reactor coolant pressure boundary, as defined in ANS N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor

Plants", Section 5.4, is that boundary containing reactor coolant under operating temperature and pressure conditions. The entire reactor coolant system pressure boundary, as defined above, is located within the containment. Thus, there are no lines that form a part of the reactor coolant pressure boundary that penetrate the containment.

CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

2. One automatic isolation valve inside and one locked closed isolation valve outside containment, or

3. One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment, or

4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

DISCUSSION

Each line that connects directly to the containment atmosphere and penetrates containment is provided with containment isolation valves, except where it is demonstrated that the containment isolation provisions are acceptable.

With the exceptions noted below, all lines which connect directly to the containment atmosphere and penetrate the primary containment are provided

with two barriers in series where they penetrate the containment, so that failure of one barrier will not prevent isolation. The containment isolation system is described and evaluated in Section 6.2.4. Air operated containment isolation valves are designed to fail closed. Motor operated valves fail in the mode which they are when failure occurs. Different power sources for each valve in series ensure that isolation is not defeated by a single failure.

Exceptions to Criterion 56 and the alternate isolation means provided and design basis are as follows:

1. Containment Pressure Sensing Lines

   These instrument lines serve an engineered safeguards function. Due to the particular importance of the pressure measurements with respect to safety, alternate double barriers are provided which avoid installing isolation valves in the lines as required by Criterion 56. Sealed instrument sensing lines are utilized. Each pressure transmitter, located adjacent to the outside of the containment, is connected to sealed bellows located adjacent to the inside surface of the containment shell by means of a sealed fluid filled tube. This arrangement provides automatic double barrier isolation without operator action and without sacrificing reliability with regard to the safety function of the lines.

2. Containment Sump Recirculation Lines

The containment sump recirculation lines are utilized during the post-accident recirculation mode of emergency core cooling and Containment Spray System operation. The lines from the containment sump are each provided with a single remote manual gate isolation valve. This valve is surrounded by a chamber which is leaktight at containment design pressure. This chamber is located outside of the Shield Building. The portion of the line from the sump to the valve is enclosed in a concentric guard pipe which is also leaktight at containment design pressure. The isolation valve provides a barrier outside the containment to prevent loss of sump water should a leak develop in the recirculation line. Should a leak develop in the valve body or in the line between the sump and the valve, the fluid is contained by either the leaktight chamber or the guard pipe. Since the sump line connection to the sump is submerged under post-accident conditions. there is no requirement for a valve inside the containment. The arrangement meets all safety requirements and any additional isolation valves are unnecessary.

3. Post-Accident Containment Sampling Lines

Two manual isolation valves, locked closed except during post-LOCA containment atmosphere sampling, are provided in each line outside the containment. The double valve barriers provide the necessary isolation to prevent release of radioactivity to the environment.

CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

DISCUSSION

With the exceptions noted below, lines that penetrate the containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere are provided with at least one isolation valve located outside the containment, which is either automatic, locked closed or capable of remote manual operation. A simple check valve is not used as an automatic valve.

Exceptions to Criterion 57 and the alternate isolation means provided and the design basis are as follows:

1. Residual Heat Removal Supply

   The residual heat removal lines from two reactor coolant loops each contain two remote manual motor-operated valves inside the containment, which are closed during power operation. The valves are interlocked to prevent opening when the Reactor Coolant System

pressure is greater than the design pressure of the Residual Heat Removal System. The second valve, which defines the reactor coolant pressure boundary, is considered locked closed during power operation. Additional isolation valves outside the containment are considered unnecessary.

2. Auxiliary Feedwater Lines

The four auxiliary feedwater lines, which connect to the main feedwater lines prior to containment penetration by the feedwater lines, each contain a check valve. The auxiliary feedwater lines perform a safety function under accident conditions. The feedwater line comprises part of a closed loop inside the containment which serves as an isolation barrier. The isolation arrangement utilized provides necessary isolation without sacrificing reliability with regard to the safety function performed by the lines.

CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

DISCUSSION

Radioactive wastes generated during normal reactor operation including anticipated operational occurrences, are processed by the Waste Treatment Systems.

The liquid waste treatment system is provided with hold-up capacity and adequate processing equipment to provide decay such that the requirements of 10 CFR 20 can be satisfied. Most of the liquid waste is recycled to minimize quantities and the rest is diluted to reduce the concentration of the radioactivity released to the environment.

The radioactive gases are contained in the decay tanks for retention during the plant lifetime. The radioactive solid and concentrated liquid wastes are drummed for off-site disposal.

CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

DISCUSSION

The spent fuel pit and cooling system, fuel handling system, radioactive waste processing systems, and other systems that contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions.

1. Most of the components and systems in this category are in frequent use and no special testing is required. Those systems and components important to safety which are not normally operating will be periodically tested, i.e., the fuel handling equipment will be tested prior to each refueling.

2. The spent fuel storage racks are located to provide sufficient shielding water over stored fuel assemblies to limit radiation at the surface of the water to no more than 2.5 mr/hr during the storage

period. The fuel handling equipment is designed to move fuel under water and to utilize a combination of metal and water shielding to limit radiation at the surface of the water to no more than 2.5 mr/hr. The exposure time during refueling will be limited so that the integrated dose to operating personnel does not exceed the limits of 10CFR20. All other areas of the plant are designed with suitable shielding for radiation protection in accordance with 10CFR20 based on anticipated radiation dose rates and occupancy.

3. Components which contain significant radioactivity are located in areas kept at a slightly negative pressure and adequately ventilated with filtered discharges to monitored vents. In the unlikely event of an accidental release of radioactive materials, filtering systems are provided for air cleanup.

4. The spent fuel pit cooling and cleanup system provides cooling to remove residual heat from fuel stored in the spent fuel pit. The spent fuel pit cooling and cleanup system is designed for testability and is described in Section 9.5.

5. The spent fuel pit is designed to withstand the postulated tornado-driven missiles and seismic events without loss of significant pit water or significant damage to stored fuel. The spent fuel pit has no gravity drains, it cannot be drained by the rupture of any pipe.

The spent fuel pit is designed such that no postulated accident could cause excessive loss of coolant inventory. The piping connected to the fuel pit is designed so that a significant loss of fuel pit water will not occur due to a pipe rupture. Level instrumentation will indicate a reduction in fuel pit water level. Redundant sources of fuel pit makeup are provided which will ensure availability of makeup to the fuel pit even under loss of normal power conditions.

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

DISCUSSION

Criticality in the new and spent fuel storage racks is prevented by the following means: (1) borated water is used in the spent fuel storage pit and assures subcriticality at all times; (2) the new and spent fuel is stored in a vertical array in storage racks which are so constructed as to maintain and guarantee a sufficient center to center distance between assemblies to assure subcriticality ($K_{eff} \leq 0.9$). The distance between spent fuel assemblies is large enough to prevent criticality even if unborated water were introduced into the spent fuel pit. Criticality prevention is discussed in Section 9.1.

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

DISCUSSION

Monitoring systems are provided to alarm on excessive temperature, low or high water level in the spent fuel pit.

Radiation monitors and alarms are provided to warn personnel of increasing levels of radiation or airborne activity.

The radiation monitoring system is described in Chapters 11 and 12.

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

DISCUSSION

The containment atmosphere is continually monitored during normal and transient station operations, using the containment particulate, gas and iodine monitors. In the event of accident conditions, samples of the containment atmosphere will be obtained from the containment post-accident sampling system to provide data of existing airborne radioactive concentrations within the containment. The areas contiguous to the containment structure will be monitored by ventilation vent sample particulate and gas monitors. In addition, the Essential Raw Water (ERW) outlets from the containment spray heat exchangers will be monitored to ensure that any leakage of radioactive fluids into the ERW System will be detected. Radioactivity levels contained in the plant effluent discharge paths are continually monitored during normal and accident conditions by the station radiation monitoring system. The radiation monitoring system is described in Chapters 11 and 12.

The environmental monitoring program for the plant will be defined in the owner's application.

3.2 CLASSIFICATION OF STRUCTURES, SYSTEM AND COMPONENTS

3.2.1 SEISMIC CLASSIFICATION

The classification of structures, components and systems by safety class provides an adequate and proper assignment of the applicable seismic design requirements. Duplication by special seismic classification is unnecessary and in certain instances misleading when applied to systems and components.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

3.2.2.1 Safety Class Definitions-Mechanical Components

Mechanical components are classified in accordance with "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants" ANS 18.2 as Safety Class 1, Safety Class 2 or Safety Class 3 and non-nuclear safety (NNS) in accordance with their importance to nuclear safety. This importance, as established by class assignment, is considered in the design and manufacturing of the component. A single system may have components in more than one class.

The following definitions apply to fluid pressure boundary components and the reactor containment. Supports are in the same class as the components

for which they provide support if failure could cause a loss of safety function associated with the supported component.

Safety Class 1

Safety Class 1, SC-1, applies to components whose failure could cause a Condition III or Condition IV[1] loss of reactor coolant.

Safety Class 2

Safety Class 2, SC-2, applies to reactor containment and to those components:

1. Of the reactor coolant system not in Safety Class 1

2. That are necessary to:

    a. Remove directly residual heat from the reactor.

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(1) Condition III occurrences are faults which may occur very infrequently during the life of a particular plant. Condition IV occurrences are faults that are not expected to occur, but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Condition IV faults are the most drastic which must be designed against, and thus represent the limiting design case.

b. Circulate reactor coolant for any safety system [1] purpose.

c. Control within the reactor containment radioactivity released.

d. Control hydrogen in the reactor containment, or

3. Of safety systems located inside the reactor containment.

Safety Class 3

Safety Class 3, SC-3, applies to those components not in Safety Class 1 or Safety Class 2, the failure of which would result in release to the environment of radioactive gases normally required to be held for decay, or that are necessary to:

1. Provide or support any safety system function

2. Control outside the reactor containment airborne radioactivity released, or

3. Remove decay heat from spent fuel.

---

(1) A safety system (in this context) is any system that functions to shut down the reactor, cool the core or another safety system or (after an accident) the reactor containment, or that contains, controls, or reduces radioactivity released in an accident. Only those portions of the secondary system are included (a) that are designed primarily to accomplish one of the above functions or (b) whose failure could prevent accomplishing one of the above functions.

Table 3.2-1 tabulates components by safety class assignment, and applicable code class.

The applicant has selected the ANS N-18.2 safety classification system for classification of mechanical components. AEC Safety Guide No. 26 and Section 3.2.2 of the "Standard Format and Content for Safety Analysis Report for Nuclear Power Plants" offer another system for classification by quality group. Table 3.2-2 is provided to enable one to determine the design level by Safety Class, the applicable codes and standards applied, and the System Quality Group Class.

3.2.2.2 Safety Class Definitions-Electrical Components

Electrical systems and equipment are used to power and control their own and other systems, some of which may have reactor protection and safety functions. Electrical systems do not have single failure modes that would be directly involved in creating ANS Conditions III and IV faults but instead are used to power and control other systems which mitigate the results of such faults. Therefore, only one safety class is required. This classification has been defined at the November 2, 1972, meeting of the IEEE Joint Committee on Nuclear Power Standards as follows:

Class IE

"The safety classification of electric equipment and systems that are essential to emergency reactor shutdown, containment isolation, reactor core cooling, and reactor and containment heat removal or otherwise

are essential in preventing significant release of radioactive material to the environment."

Classification of electrical equipment by safety class IE will be used to establish the criteria by which such equipment is selected, designed, manufactured, assembled, installed and operated. Systems and equipment not included in the safety class will be constructed and installed employing established power plant practice and in accordance with applicable codes and standards. The Electrical System and Equipment Safety Classification List is shown in Table 3.2-3.

Instrument sensor wells will be classified according to the associated piping or apparatus ANS or ASME class or code.

3.2.2.3 Classification of Structures

Structures used to house or support components are divided into two safety categories. The first category, Category 1, are those structures whose failure could impair the safety function of Safety Class 1, 2 or 3 equipment. The second category, Category II, are those structures whose failure could not directly impair the safety function of Safety Class 1, 2 and 3 equipment, but the potential gross collapse of such structures might interact with safety class equipment and thereby indirectly impair the function of adjacent safety class equipment. Figures 3.2-1 through 3.2-4 show the boundaries of each structure.

Category I structures include the platform, containment, auxiliary shielded areas, fuel handling area, safeguards areas, control and penetration areas. Category II structures include the turbine area, hotel and administrative areas and the power transmission area.

Within the broad definition of Categories I and II structures all safety category buildings contain members and structural subsystems whose failure would not impair the safety function of Safety Class 1, 2 or 3 components. Examples of such systems would be personnel elevators, stairwells not required for access in the event of a postulated earthquake, all nonstructural partitions in non-safety related areas, etc. These substructures are treated as conventional structures.

GUIDE TO TABLE 3.2-1

Column 1 - System/Component

Column 2 - ANS Safety Classes as defined in Section 3.2.2

Column 3 - Applicable Code or Standard:

ABS	American Bureau of Shipping
AFI	Air Filter Institute
AMCA	Air Moving and Conditioning Association
ANSI	American National Standards Institute
ARI	Air Conditioning and Refrigerating Institute
ASA	American Standards Association
ASHRAE	American Society of Heating, Refrigerating and Air Conditioning Engineers
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing
CMAA	Crane Manufacturers' Association of America
EPA	Environmental Protection Agency
MIL	Mil-F-51068C, Military Specification for High Efficiency Particulate Air Filters
MMA	Monorail Manufacturers' Association
NBS	National Bureau of Standards
USCG	United States Coast Guard
UL	Underwriters Laboratory
ISA	Instrument Society of America

Column 4 - Quality Assurance Level (refer to Chapter 17, Figure 17.1-2)

Column 5 - Location of Equipment

AU	Auxiliary Area
C	Containment

CO	Control Area
PT	Power Transmission Area
SA	Safeguard Area
SE	Service Area
T	Turbine Area

Column 6 - Remarks

TABLE 3.2-1

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
REACTOR COOLANT SYSTEM					
Reactor Vessel	1	ASME III Class 1	(4)* N-1**	C	
Full Length Control Rod Drive Mechanism Housing	1	ASME III Class 1	(7) N-1	C	
Part Length Control Rod Drive Mechanism Housing	1	ASME III Class 1	(4) N-1	C	
Steam Generator (tube side)	1	ASME III Class 1	(7) N-1	C	
Steam Generator (shell side)	2	ASME III	(7) N-2	C	See note 2 1 & 10
Pressurizer	1	ASME III Class 1	(7) N-1	C	
Reactor Coolant Piping & Fittings	1	ASME III	(4) N-1	C	For safety class of other piping and assoc. valving in the Reactor Coolant System and for other Aux.Systems see Note 8.

* ( ) Parentheticals refer to remarks identifying Westinghouse Quality Administration Requirements.
Refer to page 3.2-31

** The definition of the applicant's Quality Control Levels can be found in Chapter 17.

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
REACTOR COOLANT SYSTEM (CONT)					
Surge Pipe, Fittings and Fabrication	1	ASME III Class 1	(4) N-1	C	
Loop Bypass Line (where applicable)	1	ASME III Class 1	(4) N-1	C	
Bypass Manifold	1	ASME III Class 1	(5) N-1	C	
Reactor Coolant Thermowells	1	ASME III Class 1	(5) N-1	C	
Safety Valves	1	ASME III Class 1	(4) N-1	C	
Relief Valves	1	ASME III Class 1	(4) N-1	C	
Valves to Reactor Coolant System Boundary	1	ASME III Class 1	(4) N-1	C	
Pressurizer Relief Tank	NNS	ASME VIII Class UW*	(5)	C	9
Control Rod Drive Mechanism Head Adaptor Plugs	1	ASME III Class 1	(5) N-1	C	
Reactor Coolant Pump Standpipe Orifice	NNS	No Code	(6)	C	

* UW, ASME B & PV Code Section VIII, Division 1. Welded only

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
REACTOR COOLANT SYSTEM (CONT)					
Reactor Coolant Pump Standpipe	NNS	ASME VIII	(6)	C	
Reactor Coolant Pump				C	
Casing	1	ASME III Class 1	(7) N-1	C	
Main Flange	1	ASME III Class 1	(7) N-1	C	
Thermal Barrier	1	ASME III Class 1	(7) N-1	C	
#1 Seal Housing	1	ASME III Class 1	(7) N-1	C	
#2 Seal Housing	2	ASME III Class 1	(7) N-2	C	1
CHEMICAL AND VOLUME CONTROL SYSTEM					
Regenerative Heat Exchanger	2	ASME III Class 2	(4) N-2	C	
Letdown Heat Exchanger (tube side)	2	ASME III Class 2	(4) N-2	AU	
(shell side)	3	ASME III Class 3	(4) N-3	AU	(1)

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
CHEMICAL AND VOLUME CONTROL SYSTEM (CONT)					
Mixed Bed Demineralizer	3	ASME III Class 3	(5) N-3	AU	
Cation Bed Demineralizer	3	ASME III Class 3	(5) N-3	AU	
Reactor Coolant Filter	2	ASME III Class 2	(4) N-2	AU	
Volume Control Tank	2	ASME III Class 2	(4) N-2	AU	
Centrifugal Charging Pumps	2	ASME III Class 2	(4) N-2	AU	
Positive Displacement Charging Pump	2	ASME III Class 2	(4) N-2	AU	
Seal Water Injection Filter	2	ASME III Class 2	(4) N-2	AU	
Seal Water Return Filter	2	ASME III Class 2	N-2	AU	
Seal Water Heat Exchanger (tube side)	2	ASME III Class 2	N-2	AU	
(shell side)	3	ASME III Class 3	N-3	AU	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL		LOCATION	REMARKS
CHEMICAL AND VOLUME CONTROL SYSTEM(CONT)						
Letdown Orifices	2	ASME III Class 2	(5)	N-2	AU	
Excess Letdown Heat Exchanger (tube side)	2	ASME III Class 2	(4)	N-2	C	
(shell side)	2	ASME III Class 2	(4)	N-2	C	
Boric Acid Tanks	3	ASME III Class 3	(5)	N-3	AU	
Boric Acid Transfer Pump	3	ASME III Class 3		N-3	AU	
Boric Acid Blender	3	ASME III Class 3	(5)	N-3	AU	
Boric Acid Filter	3	ASME III Class 3	(5)	N-3	AU	
Boric Acid Batching Tank	NNS	ASME VIII Class UW*	(5)		AU	
Chemical Mixing Tank	NNS	ASME VIII Class UW	(6)		AU	
Resin Fill Tank	NNS					

* UW, ASME, B & PV Code Section VIII, Division 1, Welded only.

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
BORON THERMAL REGENERATION SUB SYSTEM					
Moderating Heat Exchanger (tube side)	3	ASME III Class 3	(5) N-3	AU	
(shell side)	3	ASME III Class 3	(5) N-3	AU	
Letdown Chiller Heat Exchanger (tube side)	3	ASME III Class 3	(5) N-3	AU	
(shell side)	NNS	ASME VIII Class UW	(5)	AU	
Letdown Reheat Heat Exchanger (tube side)	2	ASME III Class 2	(5) N-2	AU	
(shell side)	3	ASME III Class 3	(5) N-3	AU	
Thermal Regeneration Demineralizer	3	ASME III Class 3	(5) N-3	AU	
Chiller	NNS		(5)	AU	3
Chiller Surge Tank	NNS	ASME VIII Class UW	(6)	AU	
Chiller Pumps	NNS	No Code	(5)	AU	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
LIQUID WASTE TREATMENT SYSTEM					
Waste Holdup Tank	3	ASME III Class 3	N-3	AU	
Waste Evaporator Feed Pump	3	ASME III Class 3	N-3	AU	
Waste Evaporator Feed Filter	3	ASME III Class 3	N-3	AU	
Waste Evaporator	3	ASME III Class 3	N-3	AU	
Waste Evaporator Distillate Demineralizer	NNS	ASME VIII		AU	
Waste Evaporator Distillate Filter	NNS	ASME VIII		AU	
Waste Evaporator Distillate Tank	NNS	No Code		AU	14
Waste Evaporator Distillate Pump	NNS	No Code		AU	13
Chemical Drain Tank	NNS	ASME III		AU	14
Chemical Drain Pump	NNS	No Code		AU	13
Spent Resin Storage Tank	3	ASME III Class 3	N-3	AU	
Spent Resin Sluice Pump	3	ASME III Class 3	N-3	AU	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
LIQUID WASTE TREATMENT SYSTEM (CONT)					
Spent Resin Sluice Filter	3	ASME III Class 3	N-3	AU	
Laundry and Hot Shower Tank	NNS	No Code		AU	14
Laundry and Hot Shower Pump	NNS	No Code		AU	13
Laundry and Hot Shower Strainer	NNS	ASME VIII		AU	
Laundry and Hot Shower Filter	NNS	ASME VIII		AU	
Floor Drain Tank	NNS	No Code		AU	14
Waste Monitor Tank	NNS	No Code		AU	14
Waste Monitor Pump	NNS	No Code		AU	13
Waste Monitor Tank Demineralizer	NNS	ASME VIII		AU	
Waste Monitor Tank Filter	NNS	ASME VIII		AU	
Floor Drain Tank Pump	NNS	No Code		AU	13
Drumming Header Strainer	NNS	ASME VIII		AU	
Floor Drain Tank Filter	NNS	ASME VIII		AU	
Waste Holdup Tank Demineralizer	NNS	ASME VIII			
Waste Holdup Tank Demineralizer Filter	NNS	ASME VIII			

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
LIQUID WASTE TREATMENT SYSTEM (CONT)					
Floor Drain Tank Strainer	NNS	ASME VIII		AU	
Reactor Coolant Drain Tank Heat Exchanger (tube side)	NNS	ASME VIII		AU	
(shell side)	NNS	ASME VIII		AU	
Reactor Coolant Drain Tank	NNS	ASME VIII		AU	
Reactor Coolant Drain Pump	NNS	No Code		AU	13
GASEOUS WASTE TREATMENT SYSTEM					
Waste Gas Compressor	3	ASME III Class 3	N-3	AU	
Gas Decay Tanks	3	ASME III Class 3	N-3	AU	12
Catalytic Hydrogen Recombiners	3	ASME III Class 3	N-3	AU	
SOLID WASTE TREATMENT SYSTEM					
Waste Process Feed Tank	3	ASME III Class 3	N-3	AU	
Waste Feeder	NNS	No Code		AU	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
SOLID WASTE TREATMENT SYSTEM (CONT)					
Mixer Feeder	NNS	No Code		AU	
Waste Dewatering Pump	NNS	No Code		AU	
Cement Train Equipment	NNS	No Code		AU	
Drum Capper	NNS	No Code		AU	
Bridge Crane	NNS	MMA		AU	
EMERGENCY CORE COOLING SYSTEM					
High Head Safety Injection Pumps	2	ASME III Class 2	N-2	SA	3
Safety Injection Pumps	2	ASME III Class 2	N-2	SA	3
Accumulators	2	ASME III Class 2	N-2	C	
Boron Injection Tank	2	ASME III Class 2	N-2	C	
Boron Injection Surge Tank	3	ASME III Class 3	N-3	C	
Boron Injection Recirculation Pumps	3	ASME III Class 3	N-3	C	

TABLE 1.3-2 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
BORON RECYCLE SYSTEM					
Recycle Holdup Tank	3	ASME III Class 3	N-3	AU	
Recycle Evaporator Feed Pump	3	ASME III Class 3	N-3	AU	
Recycle Evaporator Feed Demineralizer	3	ASME III Class 3	N-3	AU	
Recycle Evaporator Feed Filter	3	ASME III Class 3	N-3	AU	
Recycle Evaporator	3	ASME III Class 3	N-3	AU	
Recycle Evaporator Condensate Demineralizer	NNS	ASME VIII		AU	
Recycle Evaporator Condensate Filter	NNS	ASME VIII		AU	
RESIDUAL HEAT REMOVAL SYSTEM					
Residual Heat Removal Pumps	2	ASME III Class 2	N-2	SA	3
Residual Heat Exchanger (tube side)	2	ASME III Class 2	N-2	SA	
(shell side)	3	ASME III Class 3	N-3	SA	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
SPENT FUEL PIT COOLING AND CLEANING SYSTEM (CONT)					
Refueling Water Purification Pump Skimmer	NNS	No Code			
Refueling Water Purification Pump Skimmer Strainer	NNS	No Code			
CONTAINMENT					
Containment Shell	2	ASME III Class MC	N-2	C	
Containment Penetrations	2	ASME III Class MC	N-2	C	11
POST-LOCA HYDROGEN RECOMBINER SYSTEM	2	ASME III Class 2	N-2		
POST-ACCIDENT CONTAINMENT HYDROGEN VENTING SYSTEM					
Filters, Absolute	3	UL	N-3	C	
CONTAINMENT SPRAY SYSTEM					
Pumps	2	ASME III Class 2	N-2	SA	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
CONTAINMENT SPRAY SYSTEM (CONT)					
Heat Exchangers (tube side)	2	ASME III Class 2	N-2	SA	
(shell side)	3	ASME III Class 2	N-2	SA	
Nozzles, Spray	2	ASME III Class 2	N-2	C	
Refueling Water Storage Tanks	2	ASME III Class 2	N-2	SA	
COMPONENT COOLING SYSTEM					
Pumps	3	ASME III Class 3	N-3	SE, SA	
Heat Exchangers	3	ASME III Class 3	N-3	SE, SA	
Surge Tank	3	ASME III Class 3	N-3	SA	
ANNULUS FILTRATION SYSTEM					
Fans	3	AMCA	N-3	SA	
Filter Trains	3	AFI	N-3	SA	
Valves and Ducts	3	ASHRAE	N-3	SA	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
CONTAINMENT POST-ACCIDENT SAMPLING SYSTEM					
Vacuum Pump	3	ASME III Class 3	N-3	CO	
Sample Vessel	3	ASME III Class 3	N-3	CO	
EMERGENCY DIESEL FUEL OIL SYSTEM					
Fuel Oil Storage Tanks	3	ABS/USCG	N-3	SA	
Transfer Pumps	3	ASME III Class 3	N-3	SA	
Starting Air Tanks	3	ASME III Class 3	N-3	SA	
SPENT FUEL PIT COOLING AND CLEANING SYSTEM					
Spent Fuel Pit Pump	3	ASME III Class 3	N-3	AU	
Heat Exchanger (tube side)	3	ASME III Class 3	N-3	AU	
(shell side)	3	ASME III Class 3	N-3	AU	
Spent Fuel Pit Skimmer Pump	NNS	No Code		AU	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
ICE CONDENSER*					
Lower Support Structure	2	AISC	N-2	C	
Lattice Structure	2	AISC	N-2	C	
Ice Baskets	2	AISC	N-2	C	
Top Deck Beams	2	AISC	N-2	C	
Insulated Wall Panels on Crane Wall	2	AISC	N-2	C	
Lower Inlet Doors	2	AISC	N-2	C	
Intermediate Deck Doors	2	AISC	N-2	C	
Top Deck Doors	2	AISC	N-2	C	
Wall Panel Mounting Fasteners on Crane	2	AISC	N-2	C	
Lower Inlet Door Bolts	2	AISC	N-2	C	
Air Return Fans	2	AMCA	N-2	C	
Bridge Crane	NNS	MMA		C	
COMPRESSED AIR SYSTEMS					
Plant Air Compressor (2)	NNS	No Code -		T	
Plant Air Aftercooler/Separator (2)	NNS	ASME VIII-		T	

* All AISC references are 1969 edition

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
COMPRESSED AIR SYSTEMS (CONT)					
Plant Air Dryer (2)	NNS	ASME VIII		T	
Plant Air Receiver (2)	NNS	ASME VIII		T	
Emergency Instrument Air Compressor Package (2)	3	ASME III Class 3	N-3	SA	
*Aftercooler/Separator (2)	3	ASME III Class 3	N-3	SA	
*Air Receiver (2)	3	ASME III Class 3	N-3	SA	
Air Receiver	3	ASME III Class 3	N-3	SA	
Emergency Instrument Air Dryer	3	ASME III Class 3	N-3	SA	
FIRE PROTECTION SYSTEM					
Fire Pumps	NNS	UL		SA	13
PLATFORM DRAIN SYSTEM					
Platform Drain Pumps	NNS	USCG			

*Included in the compressor package.

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
NSSS SAMPLING SYSTEM					
Sample Heat Exchanger	NNS			AU	
Sample Vessel	NNS			AU	
Delay Coil	2	ASME III Class 2	N-2	C	
AUXILIARY RAW WATER SYSTEM					
Pumps	NNS			SE	
Heat Exchangers	3	ASME III Class 3	N-3	SA	
Surge Tank	3	ASME III Class 3	N-3	CO	
ESSENTIAL SERVICE WATER SYSTEM					
Essential Service Water Pumps	3	ASME III Class 3	N-3	SA	
ESSENTIAL RAW WATER SUPPLY					
Essential Raw Water Pumps	3	ASME III Class 3	N-3	SA	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
FUEL HANDLING SYSTEM					
Manipulator Crane	3		N-3	AU	
Miscellaneous Handling Tools and Fixtures	NNS			C	
Internals Handling Device	NNS			C	
Reactor Vessel Head Lifting Rig	2		N-2	C	Portions which transmit loading from CRDM seismic support only
Rod Cluster Control Change Fixture	3		N-3	C	
Internals Storage Stands	NNS			C	
New Fuel Elevator	NNS			AU	
New Fuel Storage Racks	NNS		N-3	AU	
Spent Fuel Storage Racks	3		N-3	AU	
Reactor Vessel Head Storage Pad	NNS	AISC		C	
Tool and Drive Shaft Storage Racks	NNS	AISC		C	
Control Rod Drive Mechanism, Seismic Support	2		N-2	C	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
AUXILIARY FEEDWATER SYSTEM					
Auxiliary Feedwater Pumps	3	ASME III Class 3	N-3	SA	
Auxiliary Feedwater Pump Turbine Drive	3	-	N-3	SA	
Auxiliary Feedwater Storage Tanks	3	ASME III Class 3	N-3	SA	
System Piping and Valves from Storage Tank to Last Check Valve Before Steam Generator	3	ASME III Class 3	N-3	SA	
From Check Valve to Steam Generator	2	ASME III Class 2	N-3	C	
GAS SUPPLY SYSTEM					
CO_2 Supply Package	NNS	ASME VIII		SE	
Piping for H_2 and O Supply Package		ANSI B31.8			
Piping for CO_2 and N_2 Supply Package		ANSI B31.8			
Valves for H_2 and O_2 Supply Package		ASA B16.5			
Valves for CO_2 and N_2 Supply Package		ASA B16.5			

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
FUEL AREA EXHAUST SYSTEM					
Fans	3	AMCA	N-3	AU	
Prefilters	3	NBS	N-3	AU	
Absolute Filters	3	UL	N-3	AU	
Charcoal Filters	3	MIL	N-3	AU	
Dampers	3	AMCA	N-3	AU	
Ductwork	3	ASHRAE	N-3	AU	
Instrumentation	3	ISA	N-3	AU	
SAFEGUARDS AREA EXHAUST SYSTEMS					
Fans	3	AMCA	N-3	SA	
Prefilters	3	NBS	N-3	SA	
Absolute Filters	3	UL	N-3	SA	
Charcoal Filters	3	MIL	N-3	SA	
Dampers	3	AMCA	N-3	SA	
Ductwork	3	ASHRAE	N-3	SA	
Instrumentation	3	ISA	N-3	SA	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
CONTROL ROOM AIR CONDITIONING SYSTEM					
Fans	3	AMCA	N-3	CO	
Filters	3	NBS	N-3	CO	
Absolute Filters	3	UL	N-3	CO	
Charcoal Filters	3	MIL	N-3	CO	
Ductwork	3	ASHRAE	N-3	CO	
Dampers	3	AMCA	N-3	CO	
Temperature Controls	3	ISA	N-3	CO	
Instrumentation	3	ISA	N-3	CO	
Refrigerator Condensers	3	ARI	N-3	CO	
Heating and Cooling Coils	3	ARI	N-3	CO	
ADMINISTRATION AND SERVICE AREAS AND EMERGENCY RELOCATION AREA SYSTEMS					
Fans	3	AMCA	N-3	SE	
Prefilters	3	NBS	N-3	SE	
Filters	3	ARI	N-3	SE	

TABLE 3.2-1 (CONT)

MECHANICAL COMPONENT CODE AND CLASSIFICATION SUMMARY

COMPONENT	ANS SAFETY CLASS	CODE	QUALITY CONTROL LEVEL	LOCATION	REMARKS
ADMINISTRATION AND SERVICE AREAS EMERGENCY RELOCATION AREA SYSTEMS (CONT)					
Absolute Filters	3	UL	N-3	SE	
Charcoal Filters	3	MIL	N-3	SE	
Ductwork	3	ASHRAE	N-3	SE	
Temperature Controls	3	ISA	N-3	SE	
Instrumentation	3	ISA	N-3	SE	
Refrigeration Condensers	3	ARI	N-3	SE	
Heating and Cooling Coils	3	ARI	N-3	SE	

NOTES TO TABLE 3.2-1

1 Code design requirements assigned are in excess of the requirement dictated by the applicable safety class.

2 Feedwater and steam relief required to perform safety function.

3 Portions of equipment containing component cooling water are SC-3, Code class 3.

(4)* Quality Administration: Westinghouse Administrative Specifications or ASME NA 4000 Section III with Society Survey and Westinghouse Administrative Specifications.

(5) Quality Administration: Westinghouse Administrative Specifications and ASME Code Inspection at Source, if applicable.

(6) Quality Administration: Westinghouse rights to inspect, no Quality Administration required.

(7) Westinghouse NES manufacturing divisions have quality programs which are in accordance with 10 CFR 50 Appendix B, AEC Quality Assurance Criteria.

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* ( ) Parentheticals refer to remarks identifying Westinghouse Quality Administration Requirements.

8 Classification of piping and valves shall be as defined by the systems engineering flow diagrams for the appropriate safety class. Exclusive of valves and piping in the reactor coolant pressure boundary which are Code Class 1, other valves and piping are assigned the code class and quality level commensurate with safety function of the system.

9 Non-nuclear safety components are defined as those whose classification do not fall within the jurisdiction of the safety classification definitions of ANS PWR Criteria, ANS 18.2.

10 The shell side of the steam generator conforms to the requirements for a Class 1 Vessel and is so stamped as permitted under rules of Section III; however, it is not a part of the reactor coolant boundary and need only comply with Safety Class 2.

11 All containment penetrations, including isolation valves, are Safety Class 2 regardless of the classification of a specific system.

12 National Board Fire Underwriters and Underwriters Laboratories certification required.

13 ASME Section III Class 3 design and fabrication only. No code stamp required.

14 ASME Section III Class 3 design and fabrication only. No code stamp required. (Atmospheric Pressure Tanks)

TABLE 3.2-2

COMPARISON TABLE FOR SAFETY CLASSIFICATION

COMPONENTS	SAFETY CLASS 1-QUALITY GROUP A	SAFETY CLASS -2 QUALITY GROUP B	SAFETY CLASS 3-QUALITY GROUP C	NON-NUCLEAR SAFETY-QUALITY GROUP D
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components-Class 1	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components-Class 2	ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components-Class 3	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1
Piping	As above	As above	As above	ANSI B 31.1.0 Power piping
Pumps	As above	As above	As above	*ASME Section VIII, Division 1
Valves	As above	As above	As above	ANSI B 31.1.0
Atmospheric Storage tanks	N.A.	As above	As above	ASME Section III, Class 3 Design and Fabrication only, no code stamp required (Atmospheric Pressure Tanks)
0-15 psig Storage Tanks	N.A.	As above	As above	

* Section VIII does not apply. Applicant will require design, manufacture and examination to ASME III Class 3 - no code stamp.

1971 Code and appropriate Addenda. See 50.55a for general guidance relating the Code and Addenda to be applied, the date the component was ordered, and the date of issuance of a construction permit.

GUIDE TO TABLE 3.2-3

Column 1 - System/Equipment

Column 2 - Safety Class IE is defined in Section 3.2.2.2.

Column 3 - Safety-related Quality Assurance (X = Yes; - = No) is subject to the requirements of the Quality Assurance Program defined in Chapter 17.

Column 4 - Location of Equipment within Plant:

- C - Containment
- CO - Control Room
- PRR - Process Rack Room
- RCR - Rod Control Room
- SA - Engineered Safety Features Area
- T - Turbine Area

TABLE 3.2-3

ELECTRICAL SYSTEMS AND EQUIPMENT SAFETY CLASSIFICATION LIST

SYSTEM/EQUIPMENT	SAFETY CLASS	QUALITY ASSURANCE REQUIRED	LOCATION
CABLE INTERCONNECTING SAFETY CLASS IE EQUIPMENT			
4 kV Power Cable	IE	X	SA
600 Volt Power Cable	IE	X	C, SA
125 Volt dc Cable	IE	X	SA, PRR, CO, RCR
120 Volt ac Control Cable	IE	X	SA, PRR, CO, T
Instrumentation Cable	IE	X	C, SA, PRR, CO
ELECTRICAL EQUIPMENT VENTILATION SYSTEMS			
Safeguards Area Fan Motors	IE	X	SA
Safeguards Area Fan Motor Control Devices	IE	X	SA, CO
Safeguards Area Damper Solenoids	IE	X	SA
Safeguards Area Damper Control Devices	IE	X	SA, CO
Control Room Fan Motors	IE	X	SA
Control Room Fan Motor Control Devices	IE	X	CO
Control Room Damper Solenoids	IE	X	SA
Control Room Damper Control Devices	IE	X	CO

TABLE 3.2-3 (CONT)

ELECTRICAL SYSTEMS AND EQUIPMENT SAFETY CLASSIFICATION LIST

SYSTEM/EQUIPMENT	SAFETY CLASS	QUALITY ASSURANCE REQUIRED	LOCATION
ELECTRICAL EQUIPMENT VENTILATION SYSTEMS (CONT)			
Process Rack Room Fan Motors	IE	X	PRR
Process Rack Room Fan Motor Control Devices	IE	X	PRR, CO
Process Rack Room Damper Solenoids	IE	X	PRR
Process Rack Room Damper Control Devices	IE	X	PRR, CO
Diesel Room Fan Motors	IE	X	SA
Diesel Room Fan Motor Control Devices	IE	X	SA, CO
Diesel Room Damper Solenoids	IE	X	SA
Diesel Room Damper Control Devices	IE	X	SA, CO
4 KV SAFETY RELATED POWER SYSTEM			
4 kV Safety Related Switchgear	IE	X	SA
4160/480V Safety Related Transformers	IE	X	SA
4 kV Safety Related Switchgear Control Devices (mains and ties)	IE	X	SA, CO

TABLE 3.2-3 (CONT)

ELECTRICAL SYSTEMS AND EQUIPMENT SAFETY CLASSIFICATION LIST

SYSTEM/EQUIPMENT	SAFETY CLASS	QUALITY ASSURANCE REQUIRED	LOCATION
480V SAFETY RELATED POWER SYSTEM			
480V Safety Related Switchgear	IE	X	SA
480V Safety Related Motor Control Centers	IE	X	SA
480V Safety Related Switchgear and MCC Control Devices- (mains and ties)	IE	X	SA, CO
125V DC STATION POWER SYSTEM			
125V Battery Chargers	IE	X	SA
125V Batteries	IE	X	SA
125V dc Distribution Panelboards	IE	X	SA
125V AC INSTRUMENT POWER SYSTEM			
125V dc/120V ac Instrument Bus Inverters	IE	X	PRR
120V ac Instrument Power Distribution Panelboards	IE	X	PRR
COMPONENT COOLING WATER SYSTEM			
Component Cooling Water Pump Motors	IE	X	C, SA
Component Cooling Water Valve Motors	IE	X	C, SA
Component Cooling Water Valve Pilot Solenoids	IE	X	C, SA

TABLE 3.2-3 (CONT)

ELECTRICAL SYSTEMS AND EQUIPMENT SAFETY CLASSIFICATION LIST

SYSTEM/EQUIPMENT	SAFETY CLASS	QUALITY ASSURANCE REQUIRED	LOCATION
COMPONENT COOLING WATER SYSTEM (CONT)			
Component Cooling Water Motor Control Devices	IE	X	C, SA, CO
Component Cooling Water Pilot Solenoid Control Devices	IE	X	C, SA, CO
ESSENTIAL SERVICE WATER SYSTEM			
Essential Service Water Pump Motors	IE	X	SA
Essential Service Water Valve Motors	IE	X	SA
Essential Service Water Motor Control Devices	IE	X	SA, CO
Essential Service Water Valve Pilot Solenoids	IE	X	SA
Essential Service Water Pilot Solenoid Control Devices	IE	X	SA, CO
ESSENTIAL RAW WATER SYSTEM			
Essential Raw Water Pump Motors	IE	X	SA
Essential Raw Water Valve Motors	IE	X	SA
Essential Raw Water Motor Control Devices	IE	X	SA, CO

TABLE 3.2-3 (CONT)

ELECTRICAL SYSTEMS AND EQUIPMENT SAFETY CLASSIFICATION LIST

SYSTEM/EQUIPMENT	SAFETY CLASS	QUALITY ASSURANCE REQUIRED	LOCATION
STEAM SYSTEM			
Main Steam Isolation Valve Pilot Solenoids	IE	X	SA
Main Steam Isolation Valve Pilot Solenoid Control Devices	IE	X	SA, CO
CONTAINMENT SPRAY SYSTEM			
Containment Spray Pump Motors	IE	X	SA
Containment Spray Valve Motors	IE	X	SA
Containment Spray Motor Control Devices	IE	X	SA, CO, PRR
CONTAINMENT ISOLATION			
Containment Isolation Valve Motors	IE	X	C, SA
Containment Isolation Valve Pilot Solenoids	IE	X	C, SA
Containment Isolation Valve Motor Control Devices	IE	X	C, SA, CO, PRR
Containment Isolation Valve Pilot Solenoid Control Devices	IE	X	C, SA, CO, PRR

TABLE 3.2-3 (CONT)

ELECTRICAL SYSTEMS AND EQUIPMENT SAFETY CLASSIFICATION LIST

SYSTEM/EQUIPMENT	SAFETY CLASS	QUALITY ASSURANCE REQUIRED	LOCATION
CONTAINMENT AIR SYSTEM			
Containment Air Recirculation Fan Motors	IE	X	C
Containment Hydrogen Recombiner	IE	X	C
Annulus Filtration Fan Motors Containment Air System Motor Control Devices	IE	X	SA
SAFETY INJECTION SYSTEM			
Safety Injection Pump Motors	IE	X	SA
Safety Injection System Valve Motors	IE	X	C, SA
Safety Injection System Motor Control Devices	IE	X	C, SA, CO
RESIDUAL HEAT REMOVAL SYSTEM			
RHR Pump Motors	IE	X	SA
RHR Valve Motors	IE	X	C, SA
RHR Valve Pilot Solenoids	IE	X	SA
RHR Motor Control Circuits and Devices	IE	X	C, CO, SA

TABLE 3.2-3 (CONT)

ELECTRICAL SYSTEMS AND EQUIPMENT SAFETY CLASSIFICATION LIST

SYSTEM/EQUIPMENT	SAFETY CLASS	QUALITY ASSURANCE REQUIRED	LOCATION
AUXILIARY FEEDWATER SYSTEM			
Auxiliary Feedwater Pump Motors	IE	X	SA
Auxiliary Feedwater Valve Pilot Solenoids	IE	X	SA
Auxiliary Feedwater Motor Control Devices	IE	X	SA, CO, PRR
Auxiliary Feedwater Air Operated Valve Control Devices	IE	X	SA, CO, PRR
NUCLEAR INSTRUMENTATION SYSTEM (PROTECTION)			
Nuclear Sensors	IE	X	C
Nuclear Instrumentation System Preamplifiers	IE	X	SA
Nuclear Instrumentation System Racks	IE	X	CO
PROCESS INSTRUMENTATION AND CONTROL SYSTEM (PROTECTION)			
Sensors	IE	X	C, SA, T
Process Instrumentation and Control Protection Racks	IE	X	PRR

TABLE 3.2-3 (CONT)

ELECTRICAL SYSTEMS AND EQUIPMENT SAFETY CLASSIFICATION LIST

SYSTEM/EQUIPMENT	SAFETY CLASS	QUALITY ASSURANCE REQUIRED	LOCATION
REACTOR PROTECTION AND ENGINEERED SAFETY FEATURES LOGIC AND ACTUATION			
Solid State Protection System Cabinets	IE	X	PRR
Control Devices	IE	X	CO, SA
Reactor Trip Switchgear	IE	X	RCR
Engineered Safety Features Test and Actuation Cabinets	IE	X	PRR
Engineered Safety Features Sequencer Cabinets	IE	X	SA
OTHER SYSTEMS			
Emergency Instrument Air Compressor Motors	IE	X	SA
Emergency Instrument Air Compressor Control Devices	IE	X	SA

3.3 WIND AND TORNADO DESIGN CRITERIA

3.3.1 WIND CRITERIA

Operating Basis Wind

The plant is designed to operate during a storm having a fastest mile wind speed of 180 mph at 30 feet above the water level. The design wind loads are based on ANSI A58.1 (1972) "American National Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures". This code gives loads for a maximum wind velocity of 130 mph. The design loads for a 180 mph wind velocity are obtained by increasing the loads given for a 130 mph wind velocity in proportion to the square of the velocities. Figure 3.3-1 shows the design wind loads on a typical rectangular building such as the turbine hall. Only external pressure loads are considered because the building is a completely enclosed structure.

The fastest mile wind speed of 180 mph is intended to correspond to a storm having a 50 year recurrence interval at a worst east or gulf coast location. Figure 3.3-2 shows information prepared by A. H. Glenn and Associates giving wind speed and duration characteristics for a typical offshore southeast Florida location. This location represents a maximum (or worst) condition with respect to prolonged severe storm winds for the east or gulf coasts of the United States.

Wind speed duration characteristics at the other locations are less severe; in particular at the proposed Atlantic Generating Station site the hundred year recurrence one minute duration wind is 156 mph instead of 197 mph for southeast Florida.

A.H. Glenn's curves in Figure 3.3-2 shows a fifty year recurrence storm having a wind velocity of 176 mph for one minute and a peak gust of 215 mph. ANSI A58.1 gives two different tables for the effective velocity pressures for use in the design of structures and parts of structures. These effective velocity pressures include a gust factor of 1.3 for structures and 1.4 for parts. Peak gusts are a local phenomenon and a part may be considered as the local area over which the gusts will occur. If we take the fastest mile wind velocity of 180 mph, structures will be designed for an effective velocity including the peak gust of $\sqrt{1.4} \times 180 = 213$ mph. The effective velocity of 213 mph to be used in the design of parts should therefore be compared to the peak gust of 215 mph given by A.H. Glenn.

3.3.2 TORNADO CRITERIA

The Floating Nuclear Plant will be located on the east or Gulf coast. The intensity of a tornado over water is expected to be less severe than a tornado over land based on observations of storms over the Great Lakes which dissipate as they move out over water but re-intensify when they pass over the opposite shore. Despite this reduced probability of a

large tornado, the Floating Nuclear Plant is designed for tornados as described in the following paragraphs:

1. Tangential wind velocity (rotational) = 440 fps (300 mph) at height greater than 64' above sea level

   = 290 fps (200 mph) at height less than 64' above sea level.

2. Translational wind velocity, maximum = 88 fps (60 mph)

   minimum = 29 fps (20 mph)

3. Pressure drop at center of vortex = 3.0 psi

3.3.2.2 Tornado Wind Pressure Field

The tangential wind and pressure profiles shown in Figure 3.3-3 are based on a two-region model where constant circulation is defined for the outer region and constant angular velocity for the inner region, with the boundary taken at point of maximum tangential wind velocity, V_c equal to 440 ft/sec. [2,10]

In the outer region wind velocity is determined

$$V = \frac{C}{r} \text{ for } \frac{r}{R_c} \geq 1$$

where

V = tangential wind velocity at radius r

r = radius r measured from the center of rotation

C = circulation constant = 10^5 square ft/sec.

R_c = radius to maximum wind velocity

The value of C has been determined to be reasonably constant based on the data from a number of tornadoes.[3,4,5,6,7,8,9] Obviously the equation breaks down as r approaches zero. In the central core region the angular velocity is assumed constant, (i.e., solid body rotation is assumed) and the central wind velocity is determined

$$V = \Omega r \text{ for } \frac{r}{R_c} < 1$$

where

Ω = constant angular velocity $\frac{V_c}{R_c}$

V_c = maximum tangential wind velocity

R_c = radius to maximum wind velocity = $\frac{C}{V_c}$

Since V_c is defined as 440 ft/sec and C as 10^5 square ft/sec, R_c is equal to 227 feet.

The resulting pressure distribution is determined by the cyclostropic wind equation.

$$\frac{1}{\rho}\frac{dp}{dr} = \frac{V(V)^2}{r}$$

where

ρ = ambient air density

p = air pressure

Substituting the expression for V(r) and integrating, the differential pressure at any point is defined by

$$P(r) = V_c^2 \left[1 - 0.5 \left(\frac{r}{R_c}\right)^2\right] \text{ for } \frac{r}{R_c} \leq 1, \text{ and}$$

$$P(r) = 0.5\, V_c^2 \left[\frac{R_c}{r}\right] \text{ for } \frac{r}{R_c} \geq 1$$

where

p = differential pressure (below atmospheric pressure at infinite radius)

The differential pressure as a function of cyclonic radius is shown in Curve A of Figure 3.3-3 with the maximum pressure drop occurring at radius equal to zero and one-half this value at R_c where the tangential wind velocity is a maximum. The maximum pressure drop is adjusted to the specified maximum of 3 psi. Tangential wind velocity as a function of cyclonic radius is shown in Curve B of Figure 3.3-3. In Figure 3.3-4 is shown the translational component of wind velocity combined with the tangential velocity. By the use of such curves, the free field pressure differential and wind or dynamic pressure load is defined at any point. The actual wind pressure load is determined by the equation :

$$q = 0.5\, p\, V^2$$

where

q = dynamic or wind pressure load

The total tornado pressure loading at any point i in a free field is expressed by:

$$W_i = p_i + q_i$$

3.3.2.3 Tornado Velocities at Low Elevations

The definition of the tornado in the previous paragraphs is applicable

to elevations where the effects of the ground are negligible. Closer to the ground the tangential velocities are reduced due to friction, while the pressure distribution is unaltered. This results in the observed air flow into the center of the tornado close to ground level. The magnitude of the tornado velocities close to the ground is dependent on the type of structures and topography at ground level. In the case of the floating plant the breakwater is larger than typical land structures and will provide some sheltering effect.

Figure 3.3-5 shows the maximum tangential velocity as a function of elevation based on the work of Hoecker [4] and Bates and Swanson[10]. The figure shows the following curves:

1. Dallas Tornado - maximum tangential velocity of 170 mph at a height of 100 feet based on Hoecker's analysis [4].

2. Dallas Tornado velocities increased by a factor of 300/170 with no change in the height ordinate. A more realistic magnification would also increase the height scale; the magnification of velocity with no change in height gives a conservative estimate of the velocity at a given height.

3. Tornado velocities based on equation proposed by Bates and Swanson[10]. These velocities assume a maximum tangential velocity of 300 mph at

a height of 200 feet above anemometer level (30 feet).

4. Design tangential velocities as follows:

Plant Elevation 94 feet and above	Maximum tangential wind velocity = 300 mph
Plant Elevation below 94 feet	Maximum tangential wind velocity = 200 mph

The plant elevation of 94 feet is approximately 64 feet above the water surface. It corresponds to the main operating level in the turbine and fuel handling areas.

The assumption of a 200 mph wind velocity over the full 64 feet above water level will give a conservative estimate of the total force on the plant for the "Standard" (300 mph) tornado. The radial component of the velocity is ignored because it does not occur at the same location as the maximum tangential component, acts perpendicular thereto and is one quarter to one half the maximum tangential velocity component.[4]

At plant elevations below 94 feet the tangential wind velocity distribution is taken as two thirds of the wind velocity above elevation 94 feet. The translational wind velocity of 60 mph maximum is added to the tangential component to give a maximum horizontal velocity of 360 mph above plant elevation 94 feet and 260 mph below this elevation.

3.3.2.4 Tornado Wind Design Load

Wind loads are calculated using the methods described in ASCE Paper No. 3269 "Wind Forces on Structures". This is generally the basis of the 1972 edition of ANSI A58.1 which is being used for the operating basis wind design. The ASCE paper provides more data on structures having dimensions similar to the floating nuclear plant. No gust factors are added to the specified tornado wind velocities.

Drag and lift coefficients and load distribution on building elements as a function of pressure differentials and wind velocity are determined in accordance with ASCE 3269.[1]

Wind loads used in the design of individual large structures such as the containment are based on a uniform wind velocity of 300 mph above plant elevations of 94' and 200 mph below this elevation. The wind pressure distribution on the containment is represented by Fourier Series and is shown in Figure 3.3-6. Wind loads on parts of structures are based on the maximum wind velocity of 360 mph.

3.3.2.5 Plant Tornado Loads

The tornado wind and pressure field defined in paragraph 3.3.2.2 represent the free field conditions when the tornado is passing over the sea. When the tornado passes over the breakwater and plant, there will be significant disturbance of these free field conditions.

Total tornado loads on the plant are analyzed based on the assumption that the plant and breakwater do not affect the free field wind and pressure distribution. The exposed area is subdivided into approximately 90 small areas as shown in Figure 3.3-7; the wind velocity and pressure force on each element is computed individually.

Wind Velocity Forces

Forces due to the wind velocity at a certain time are calculated for each exposed surface and are then combined to give the total forces on the platform at that instant of time.

In this manner time history forcing functions may be determined from time zero before the tornado affects the platform to some time after the tornado has passed.

The pressure on any exposed surface is calculated as:

$$q = 1/2P \; C_D \; V^2$$

where

q = pressure on the structure due to the wind velocity

P = density of air

C_D = drag coefficient

V = component of the free field wind velocity normal to the surface for a wall or parallel to the surface for a roof.

The drag coefficient C_D is based on the coefficients given in ASCE 3269 "Wind Forces on Structures" Table 4(a) for a closed hall and a height to width to length ratio of 1:4:4.

Windward face C_D = 0.9

Leeward face C_D = -0.3

Roof C_D = -0.5

Where one structure is sheltered behind another structure an equal area of both the sheltered and the sheltering structure are assumed to experience equal and opposite pressures which cancel each other out as far as the total

loads on the platform are concerned. No credit is taken for the sheltering effect of the breakwater; the wind loads are assumed to act from the water level up.

Where failure of siding will permit a flow of air through the building which does not act on significant interior structures, an effective area is taken for the siding equal to one third of the actual area. This reduction is applied to the turbine area and fuel handling area above the 94 foot level. (Below the 94 foot level the turbine support and spent fuel pit block any through flow).

Differential Pressure Forces

The forces due to the differential pressure are determined from the free field differential pressure on each exposed surface. An effective area of one third of the actual area is used for those elements whose siding will fail under tornado conditions.

3.3.2.6 Tornado Induced Plant Motions

The previous paragraph described the analysis of the total tornado loads on the plant. These time histories of tornado loads have been developed for a series of potential tornado paths and translational velocities. Typical time histories are shown in Figure 3.3-8 for tornados passing along the

longitudinal center line of the plant with translational velocities of 20 mph. and 60 mph.

The analysis of the response of the plant to these tornado loads is described in this paragraph. A simplified mooring system has been assumed for the response analysis in order to specify the motions which should be used for tornado design of the plant. The owner will provide a mooring system which results in tornado induced motions less severe than the motions specified in 3.7.1.

The first step in the analysis of the plant response is to calculate the hydrodynamic added masses associated with the six degrees of freedom of the plant. Once these masses have been calculated the response may be determined either by direct integration of the equations of motion or by modal analysis and integration of each mode individually.

Added Mass

The added masses have been analyzed using both a two dimensional and a three dimensional analysis. Generally two dimensional analyses are considered adequate for response motions; however, because of the nearly square platform configuration the effects of the three dimensional analysis were investigated. The hydrodynamic added mass is calculated from the pressure forces on the platform when the platform is vibrating in one

of its degrees of freedom. These pressures are calculated from an analysis of the flow in the vicinity of the platform. Assuming the water to be non-viscous incompressible fluid the analysis reduces to a Laplace equation for the two or three dimensional model. The method of dynamic relaxation [11] is used for solving the Laplace equation of velocity potential.

The model used for the three dimensional Laplace analysis is shown in Figure 3.3-9. The figure shows the outlines of the platform and basin used in the analysis together with a typical element and the locations of nodes for calculation of potential and velocities. A typical section of this model was used in the two dimensional analysis. On the fixed walls and bottom of the basin, the normal velocity is zero; along the wetted perimeter of the platform various normal velocities are assigned for each mode of motion of the platform. For instance, in sway (or surge) and heave the platform moves with accelerations either horizontally or vertically of arbitrary amount; in roll (or pitch), the platform rolls about its center of gravity. From the solution of the velocity potential, the hydrodynamic forces acting on the platform are calculated.

Figure 3.3-10 shows the added masses in heave, sway and roll of the platform as a function of the water depth or width around the plant. The added masses are practically independent of the distance between the breakwater and the platform for distances in excess of 40 feet. As would be expected both the heave and roll added mass are affected by the water depth below the platform; in addition the three dimensional effect is significant for both cases.

Figure 3.3-11 shows a comparison between the results for added mass obtained by the method described above with results obtained in the past by other investigators. [12] The analysis was carried out in two dimensions on a rectangular block. There is excellent agreement for B/D = 2; however for B/D = 3.3 there is a significant difference especially for the shallower water conditions. The answers given by Wilson[12] show the added mass for B/D = 3.3 to be less than the added mass for B/D = 2. The results obtained by the applicant indicate the reverse. A simplified one dimensional analysis in very shallow water indicated that the added mass should be proportional to B/D. Hence the applicant's results would seem to be more reasonable.

Response Analysis

The response of the plant is defined mathematically by six equations of motion. These equations of motion reduce to two independent equations for heave and yaw, two coupled equations for sway and roll, and another two coupled equations for surge and pitch. In the coupled equations, two natural frequencies and mode shapes are obtained. In sway-roll coupled equations, for example, the first mode is roll dominated where the second mode is sway dominated for an assumed mooring system having a stiffness of 10^7 pounds per foot in each direction. By modal analysis the coupled equations are reduced to two independent equations. The response of the platform is obtained by numerical integration for each component of the forcing function. These results are added together to give the total response in heave, sway, surge, yaw, roll and pitch.

Typical results

Preliminary analyses have been completed for some of the tornado loading conditions. Typical results are shown in this section for the case of the tornado passing along the center line of the platform. This tornado path is expected to produce the most severe platform response in roll and heave. The mooring is assumed to be linear elastic with spring constants as shown on the response plots.

As illustrated in Figure 3.3.12, the plant response varies with the translational velocity of the tornado with the maximum roll response occurring when the roll forcing function has the same frequency as the natural frequency in roll.

3.3.2.7 Tornado Missile Design Parameters

The full spectrum of potential tornado missiles falls into two categories, penetrating type missiles and impact missiles. The penetrating-type missiles are characterized by relatively high velocity and small area of impact and have their primary effect on the target structure by penetration. Impact-type missiles are characterized by relatively large mass and have their primary effect on the structure by impulse loading. The specific missile used to represent the penetrating-type missile is defined below.

There are no potential missiles of the impact type which have been identified as being likely to be in the vicinity of the plant.

1. Weight = 168 lb.

2. Horizontal velocity = 250 ft/sec.

3. Vertical velocity = 83 ft/sec.

4. Impact area = 0.35 sq. ft.

5. Length = 12.0 ft.

6. Diameter = 9.0 in.

7. Fragility = 1.0 percent

The velocity components for this missile are based on the methods of injection and the three dimensional wind field (tangential, radial, and vertical) developed by Bates and Swanson (10) particularized for the maximum wind velocity of 440 ft/sec. and 3.0 psi pressure drop used to define the design basis tornado. This analysis is described in WCAP 7897 (13). For design purposes only missile trajectories perpendicular to the surfaces of the structural element being designed are considered. Design of the

plant for the above tornado missile is described in Section 3.5

3.3.2.8 Structural Design

Category I Structures are designed such that the stresses remain within allowable values for the operating basis wind and such that they remain functional during and after the design basis wind. The design criteria are described further in Section 3.8.

Category II Structures are designed such that the stresses remain within allowable values for the operating basis wind. Main members are also designed to prevent collapse under the design basis wind. Under this condition the siding is designed to fail at predetermined locations in the walls. At these locations blow-out or blow-in panels will be provided. The design criteria are described further in Section 3.8

The design basis for tornado induced platform motions is defined in Subsection 3.7.1. These accelerations are based on a tornado passing over the plant at the translational velocity which creates the maximum platform response.

References

1. Wind Forces on Structures, Transaction, ASCE, V. 126, Part II (Paper No. 3269), 1124-1198, 1964.

2. Doan, P. L., "Tornado Considerations for Nuclear Power Plant Structures", Nuclear Safety, Vol. II, No. 4, July-August 1970.

3. Lewis, W. and Perkins, P. H., "Recorded Pressure Distribution in the Outer Portions of a Tornado Vortex", "Monthly Weather Review", December 1953.

4. Hoecker, W. H., Jr., "Wind Speed and Airflow Patterns in the Dallas Tornado of April 2, 1957", Monthly Weather Review, 88:167-180, May 1960.

5. Hoecker, W. H., Jr., Beebe, R. G., et. al., "The Tornadoes of Dallas Texas, April 2, 1957", U. S. Department of Commerce, Weather Bureau Research Paper No. 41, 1960.

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7. Hoecker, W. H., Jr., "History and Measurement of the Two Major Scottsbluff Tornadoes of 27 June 1055", Bulletin American Meterological Society, 40:117-133, March 1959.

8. Van Tassel, E. L., "The North Platte Valley Tornado Outbreak of June 27, 1955", Monthly Weather Review, 83:255-264, Nov., 1955.

9. Wagner, N.K., "A Brief Description and Analysis of the Austin, Texas Tornado of 10 May 1959", Bulletin American Meterological Society, 41:14-17, January, 1960.

10. Bates, F. C., and Swanson, A. E., "Tornado Design Consideration for Nuclear Power Plants", ANS Transactions, November 1967.

11. J. R. H. Otter, et al, Dynamic Relaxation, Proc. Inst. Civ. Eng. London December 1966.

12. B. W. Wilson, "The Energy Problem in the Mooring of Ships Exposed to Waves"

13. WCAP 7897 Topical Report-Characteristics of Tornado Generated Missiles.

3.4 WAVE DESIGN CRITERIA

3.4.1 WAVE CRITERIA DURING TOW

The plant structures and equipment are designed to resist the inertia loads and deformations imposed by the platform motion due to extreme weather conditions that might possibly be encountered while shipping the plant to its final location. In most cases, the equipment is not required to function during shipment, and the loads are considered in the same manner as temporary construction loads or wind loads. Temporary supports may be provided in some instances based on an economic evaluation of equipment changes versus temporary bracing.

The inertia loads and deformations are specified from the results of analyses of the platform response to extreme sea conditions. The sea conditions and method of analysis of the platform response are described in the following paragraphs.

The plant will be shipped when severe storms are unlikely. However, the plant is designed for the extreme sea conditions which would be associated with a severe storm. These sea conditions are defined by the International Ship Structures Congress (ISSC)[1] wave spectrum. This spectrum is defined in terms of the spectral density $2h^2(\omega)$. The area under the spectral density curve is equal to the square of the significant wave height, H_s^2.

$$2h^2(\omega) = \frac{2760\, H_s^2 T_s\, e^{-[690/(T_s\omega)^4]}}{(T_s\omega^5)}$$

Where: Significant wave height = H_s = 20 ft. (crest to trough).

Observed period = T_s = most severe period in range 5-20 seconds.

This wave spectrum is shown in Figure 3.4-1 for various significant periods. The ISSC committee report gives data for observed waves for ocean areas around the world. The area most representative of the east coast is Area 11 which covers the eastern seaboard below latitude 40°N (Philadelphia). Table 3.4-1 is extracted from the ISSC report and shows values for area 11 of period, height and direction of the waves. The table is based on 281,000 observations and shows that the significant wave height exceeds 6 meters 0.5 percent of the time. There is no specific period associated with these waves, but the table shows some probability for each range of periods. The response of the platform will be maximum for certain wave periods which are close to the fundamental frequencies of the platform. The design basis is therefore set such that the worst period should be assumed.

The design basis is based on observations for Area 11. Considering also Areas 1-a and 1-b which are applicable to the North Atlantic, significant waves exceed 6 meters for 1.5 percent of the time in area 1-a and 3.1 percent of the time in area 1-b. These are predominately waves from the west and would not be expected to be as severe close to the coastline. Therefore, the design basis is also applicable for these areas.

TABLE 3.4.1-1

OBSERVED SEA CONDITIONS OFF EAST COAST OF U.S.A.

period (Seconds)	Height in Meters ≤1/2	1 1/1-<3	3-<4 1/2	4 1/2-<6	6-<8	≥8	Total
<7	48.5	24.3	1.7	.1	.1	*	74.7
>7-9	2.3	8.2	1.7	.2	.1	*	12.5
>9-11	.8	2.8	1.3	.2	.1	*	5.2
>11-13	.3	.7	.5	.1	*	*	1.8
>13-15	.2	.2	.2	.1	*	*	.8
>15	2.1	.2	.1	.1	*	*	2.5
Total	54.2	36.4	5.6	0.8	0.4	0.1	97.5

Calm 2.5 Number OBS. 280,996

Direction	Height in Meters ≤1/2	1 1/2-<3	3-<4 1/2	4-1/2-<6	6-<8	≥8	Total
N	11.7	4.9	1.0	.2	.1	*	17.9
NE	5.5	4.8	.7	.1	*	*	11.1
E	11.2	4.9	.5	.1	*	*	16.7
SE	6.8	2.8	.3	*	*	*	9.9
S	9.1	4.6	.6	.1	*	*	14.3
SW	5.7	3.7	.5	.1	*	*	9.9
W	5.1	3.9	.8	.2	.1	*	10.1
NW	3.1	3.2	.8	.2	.1	*	7.4
Total	58. 2	32.8	5.0	.9	.3	.1	97.3

Calm 2.7 Number OBS. 313,969

Direction	Wave Period in Seconds <7	>7-9	>9-11	>11-13	>13-15	≥15	Total
N	8.5	1.9	.9	.3	.1	.3	12.1
NE	8.5	1.6	.7	.3	.1	.3	11.4
E	15.9	2.2	.8	.3	.1	.5	19.8
SE	8.2	1.0	.4	.1	.1	.3	10.1
S	12.1	1.6	.6	.2	.1	.4	15.0
SW	8.3	1.2	.4	.2	.1	.3	10.4
W	7.9	1.5	.7	.2	.1	.3	10.7
NW	5.3	1.3	.6	.2	.1	.2	7.8
Total	74.7	12.4	5.1	1.8	.8	2.5	97.5

Calm 2.7 Number OBS. 285,359

* Indicates <0.05 percent

NOTE: These figures are taken from ISSC committee No. 1 report for Area 11 which extends to longitude 60°E between latitudes 20°E and 40°N.

A number of different wave spectra are documented in the literature. These are summarized by W. H. Michel [2] Michel recommends the ISSC spectrum as the best philosophy for use at the present time. Since the observed period of the waves can vary over a wide range (see Table 3.4.1-1), the period used with the ISSC spectrum is assumed to vary between 5 and 20 seconds. An analysis using each of the five spectra described by Michel indicated that the ISSC spectrum with a varying significant period gives the most conservative results.

The platform was analyzed for its response in regular sea conditions using the charts presented in "Hydrodynamic Characteristics of Prismatic Barges" (3). These charts give the hydrodynamic added mass, damping and wave forces for prismatic barges in regular wave conditions.

Theoretically there are six coupled equations for six modes of motion, namely, heave, surge, sway, yaw, roll and pitch. The magnitudes of the motion are based on the maximum wave in either head or beam seas, occurring at the most severe wave length. The responses of the platform in oblique seas are less severe than these magnitudes due to the increased length along the diagonal. The symmetry of the platform reduces the equations of motion to one independent equation for heave motion and two sets of coupled equations for either pitch and surge or roll and sway. The equations of motion were solved to determine the displacements, velocities and accelerations under unit wave amplitude at a given wave period. The res-

ponse of the platform under unit wave amplitude is called the response amplitude operator. The platform responses in a regular sea condition are given by

$$X = \frac{H}{2} x$$

where X = platform response

H = regular wave height (crest to trough)

x = response amplitude operator

Figure 3.4-2 shows the response amplitude operators for the platform for heave, roll and pitch motions.

To determine the response of the platform to an irregular sea, a response spectrum is constructed for each motion (displacement, velocity or acceleration) by combining the wave spectrum and the response amplitude operator as follows:

$$2 h_r^2(\omega) = 2 h^2(\omega) \cdot x^2(\omega)$$

where $2 h_r^2(\omega)$ = response spectral density

$2 h^2(\omega)$ = wave spectral density (defined by the ISSC wave spectrum)

$x(\omega)$ = response amplitude operator.

A typical response spectrum is shown in Figure 3.4-3 for the heave response of the platform at a significant wave period of 14 seconds. This period is the period which gives the maximum value of the response amplitude operator. The area under the response spectrum curve is

calculated by numerical integration. The significant response in irregular seas is equal to the square root of this area. Since the period T_s appears as a parameter in the ISSC wave spectrum the responses are functions of T_s. By assigning various values of T_s in the formulation, the maximum response can be obtained. Figure 3.4-4 shows the significant responses in heave, roll and pitch as a function of the significant period, T_s.

Response spectra are also prepared for the equivalent static accelerations at typical locations of the plant. These accelerations are the significant magnitudes; they are increased by a factor of 1.67 to obtain statistically the magnitude corresponding to the one in a hundred wave which is generally considered to represent the observed maximum wave. The equivalent static accelerations obtained from this analysis are shown in Table 3.4.1-2.

The equivalent static accelerations used in design of the Floating Nuclear Plant are those corresponding to the one in a hundred wave and are summarized with the other plant motions in paragraph 3.7.1.

3.4.2 WAVE CRITERIA DURING OPERATION

The Floating Nuclear Plant is designed for wave conditions during operations as described in Paragraph 3.7.1.

TABLE 3.4.1-2

EQUIVALENT STATIC ACCELERATIONS DURING TOW

DIRECTION	LOCATION	SIGNIFICANT MAGNITUDE	AVERAGE OF ONE HUNDRETH GREATEST MAGNITUDES
Acceleration in plane parallel to platform (longitudinal or transverse)	El. 40'	0.10g	0.17g
	El. 140'	0.18g	0.30g
Accelerations normal to platform	Heave	0.04g	0.07g
	At distance of 100 ft. from longitudinal or transverse center line (due to roll or pitch)	0.09g	0.15g

REFERENCES

1. Committee No. 1, Report on Environmental Conditions, Proceedings of the Second International Ship Structures Congress, Delft, 1964.

2. W. H. Michel, "Sea Spectra Simplified" Marine Technology, Jan. 1968.

3. C. H. Kim, C. J. Henry, F. Chou "Hydrodynamic Characteristics of Prismatic Barges" Offshore Technology Conference, April 1971.

3.5 MISSILE PROTECTION CRITERIA

3.5.1 MISSILES POSTULATED WITHIN REACTOR CONTAINMENT

The characteristics and design evaluation of missiles postulated within the reactor containment are given in RESAR Revision 3, Sections 3.5.1 through 3.5.3.

3.5.2 MISSILES POSTULATED OUTSIDE THE REACTOR CONTAINMENT

Three sources of missiles have been postulated external to the reactor containment.

1. Tornado - borne missiles

2. Turbine - generator missiles

3. Missiles resulting from the crash of a helicopter on the plant

The missile protection criteria for these sources, missile design bases, and the analytical formulae used to determine missile penetration in concrete and steel are given in the following sections.

3.5.2.1. Missile Protection Criteria

The plant will be designed to meet the following criteria for any of the design basis missiles defined below. The missile(s) from one event must not cause:

1. Excessive flooding of the Floating Nuclear Plant.

2. Loss of reactor coolant accident (LOCA).

3. Sufficient damage to prevent safe shutdown assuming a single failure in the remaining equipment.

4. Offsite exposure in excess of the guidelines of 10 CFR 100.

No accidents other than loss of offsite power and those caused by the impact are assumed to occur simultaneous with the missile impact.

3.5.2.2 Tornado Missiles

The design basis tornado missile is defined in Section 3.3.2.7.

The impact of the tornado missile will be analyzed both for dynamic loading and for penetration of the structure or equipment hit. Initial impact of the tornado missile is assumed to occur anywhere on the exterior of the plant at or above the waterline. Preliminary analyses indicate that the defined tornado missile can be stopped before any equipment important to safety is disabled.

3.5.2.3 Missiles Generated by Turbine Generator Overspeed

The turbine-generator for the Floating Nuclear Plant is manufactured by Westinghouse Electric Corporation. The turbine is a single Westinghouse TC6F44, 1800 rpm machine with one high pressure (HP) and three double outward flow low pressure (LP) casings. Two moisture separator reheaters are mounted along side the low pressure casing - one on each side - above the turbine center line. The low pressure rotors have 44-inch last stage blades. The generator is directly coupled to the turbine. The turbine-generator is described in Section 10.2.

The design basis turbine-generator missile postulated for the Floating Nuclear Plant is four quarter disc segments of one of the discs shrunk-on the turbine shaft in one of the LP casings at design overspeed (120%

rated). The reasons for the selection of this missile, its characteristics and effects are discussed in the following sections.

The probability of unacceptable consequences arising from the turbine missile will be shown to be low. Many areas of the plant do not require protection from the turbine missile because impact is unlikely. Where the impact is considered possible either existing protection will be shown to be adequate or new protective measures will be provided, or the consequences of damage to equipment will be shown to be acceptable. Protective measures will be shown to be acceptable. Protective measures will be evaluated on their ability to withstand turbine missile penetration.

3.5.2.3.1 Source of Turbine Missile

Turbine-generators are large machines rotating at relatively high speed, and failure of the rotating components can result in the production of large steel missiles which may have sufficient energy to burst through the turbine casing and strike adjacent structures and equipment. The components most highly stressed by rotational forces in the turbine are the discs into which the moving blades are fitted. In the Westinghouse TC6F44 turbine the discs in the high pressure casing are integral with the turbine shaft while those in the low pressure casing are shrunk on and keyed to the shaft.

Missiles may also result from mechanical failures in the generator rotor.

Most of these will be contained within the generator by the stator frame and windings, and any that do emerge would be less energetic and damaging than those from the turbine. They are, therefore, not considered as design turbine-generator missiles.

Three potential rotational speeds may be identified for the turbine generator:

1. Rated speed.

   This is 1800 rpm for the Floating Nuclear Plant. This speed remains constant while the generator is synchronized to the 60 Hz frequency of the utilities electrical network.

2. Design overspeed.

   Design overspeed is 120% of rated for the Floating Nuclear Plant turbine-generator, and the rotating parts of the turbine are proof tested to this speed.

Normally the valves controlling steam to the turbine will be closed by the speed control system as the speed rises after the generator loses load, and the maximum overspeed will be approximately 110%. If the valves do not close on this signal, the turbine speed will rise a few percent more until either the electro-hydraulic or mechanical overspeed trip close the valves. The speed rise to 120% after trip is due to stored energy within the turbine.

It should be noted that the turbine is specifically designed to withstand stresses at design overspeed as an expected operational occurrence, and no missiles are expected at this speed.

3. Runaway

If the generator lost its load and steam were not shut off to the turbine, the unit would theoretically accelerate until aerodynamic losses rise to equal the driving force. For the Floating Nuclear Plant, runaway is approximately 180% of rated speed. At this speed several of the blade discs are calculated to be at or near their failure stress and many high energy missiles could be generated.

The turbine generator for the Floating Nuclear Plant will be

provided with control and protection equipment to reduce the probability of runaway to a negligibly low value (10^{-7} per year or less)

For turbine-disc failure to occur at or below design overspeed, any disc that fails must be flawed in some way, and fracture mechanics calculations by Westinghouse show that the critical flaw size is many times larger than any flaw that would be undetected during manufacture. Therefore, it is assumed that no more than one disc could fail at or below design overspeed.

3.5.2.3.2 Industry Experience of Turbine Missiles

Westinghouse turbine-generator units have never experienced a major structural failure of a rotating part that resulted in missile-like pieces leaving the turbine casing. A review of the records of all Westinghouse turbine-generator units in operation from 1938 through 1969 shows:

Total number of units in operation	792
No. of units with separate disc(s) on turbine shaft	453
Total number of discs in operation	1,429
Total number of disc years of service	15,430
Average number of discs per rotor	3.1
Average number of years per disc	10.8
Maximum years of service on a single disc	31
Disc failures	Zero

Three important factors have contributed to this record:

1. Factory test procedures

   Factory test procedures include destructive testing of material specimens taken from the disc forging and ultrasonic test of each disc following major heat treatment. These tests ensure sound discs with mechanical properties (tensile strength, yield strength, ductility and impact strength) equal to or exceeding the specified levels.

2. Redundancy in the control system

   As a minimum all Westinghouse steam turbines are provided with a main speed governing system to close the governing (and interceptor) valves plus a mechanical emergency overspeed trip system to close the separate stop and governing (plus reheat stop and interceptor) valves.

   On a unit trip two steam line valves (turbine stop and governing valves) are tripped closed to provide a redundant system.

It should be noted that each stop, governing, reheat stop and interceptor valve is spring closed. Thus it is only necessary to dump the high pressure fluid from under the servoactuators to close the valves.

3. Routine testing of the turbine steam valves and the mechanical emergency overspeed protective system while the unit is carrying load.

3.5.2.3.3 Probability of Turbine-Generator Missile at Design Overspeed

Westinghouse has calculated the probability of ejecting a turbine missile at design overspeed as 5×10^{-8} per year. This is based on the extremely conservative assumptions that a flaw much larger than detectable passes inspection and the disc material properties are unacceptably poor, and that the flaw then grows to critical size and fails the disc because of an excessive number of either cold starts (200 times the number anticipated) or design overspeed occurrences.

Using the Westinghouse data of Section 3.5.2.3.2 and General Electric Company data [1], an upper bound probability of missile ejection has been calculated by the applicant. Westinghouse experience with separate

discs is zero disc failures in 15,430 disc years of service and General Electric's is zero failures in 48,398 disc years for machines built between January 1938 and December 1969. The probability of catastrophic failure is then 4.7×10^{-5} per year at the 95% confidence level, or 1.1×10^{-5} per year at the 50% confidence level.

As design basis the applicant assumes that the probability of ejection of a turbine missile at design overspeed is no higher than 1×10^{-5} per year.

3.5.2.3.4 Characteristics of the Turbine Missile

Fracture of the discs into 90, 120, and 180 degree segments was considered. Calculations show that the 90 degree fragments pose the greatest threat as external missiles.

A 120 degree segment has an initial translational kinetic energy 12.5% greater than that of a 90 degree segment; however, it also has a 33% greater rim periphery resulting in greater energy loss while penetrating the turbine casing. This results in nearly equal kinetic energy of 90 and 120 degree segments leaving the turbine casing. However, since the 90 degree segments have the smaller impact areas, they represent a more severe missile.

The initial translational kinetic energy of a half disc is equal to that of a quarter disc. Because of kinematic considerations a half disc segment will always impact with the rotor after fracture. The 180 degree segment, due to its larger size, will subject the stationary parts to greater deformation. As a result the 180 degree segment will leave the turbine casing with lower energy than the 90 degree segment.

For the purpose of evaluating the missile containing ability of the turbine structure, the shrunk-on discs have been postulated to fail into four equal segments.

Turbine Missiles from the High Pressure Turbine

Calculations show that all fragments generated by a postulated failure of the HP turbine rotor would be contained by the HP turbine blade rings and casing.

Turbine Missiles from the Low Pressure Turbine

In each LP casing, there are ten discs shrunk on the shaft with five per flow side. These discs experience different degrees of stress when in operation. Disc # 2, starting from the transverse centerline,

experiences the highest stress while disc # 5 experiences the lowest. Calculated initial translational velocity and initial translational kinetic energies of disc quadrants leaving the turbine rotor assuming disc fractures at design overspeed (120% of rated speed) are:

DISC NO.	WEIGHT (LB)	VELOCITY of e.g. (FT/SEC)	TRANSLATIONAL K.E. (10^6 FT-LB)
1	3790	625	23.0
2	3750	594	20.6
3	2740	609	15.8
4	3190	532	14.0
5	3632	512	14.8

Summary of Individual Disc Results

DISCS NO. 1, 3 and 4

Discs No. 1, 3 and 4 are assumed to fracture at 120% of rated speed. The disc quadrants enter into collisions with the LP turbine stationary parts and are contained.

DISC NO. 2

Disc No. 2 is assumed to fracture at 120% of rated speed. At this speed the initial translational kinetic energy of a quadrant of Disc No. 2 is 20.6 x 10^6 ft-lbs. An initial collision occurs between the Disc No. 2 quadrants and the concentric blade ring. A section is sheared from the blade ring and secondary collisions occur between the disc quadrants, blade ring fragments and inner cylinders No. 1, No. 2 and outer cylinders. As a result quadrants of Disc No. 2 are ejected with a maximum translational kinetic energy of 7 x 10^6 ft-lbs.

DISC NO. 5

Disc No. 5 is assumed to fracture at 120% of rated speed. At this speed a quadrant of Disc No. 5 has a translational kinetic energy of 14.8 x 10^6 ft-lbs. The initial collision occurs between the disc fragments and the diffuser ring and the adjacent blade ring. As a result quadrants of Disc No. 5 are ejected through the end section of the outer cylinder with a maximum translational kinetic energy of 5 x 10^6 ft-lbs.

SUMMARY OF CALCULATED RESULTS

MISSILE	WEIGHT (LB)	EXIT VELOCITY (FT/SEC)	EXIT ENERGY (10^6 FT-LB)
Disc No. 1 Quadrant	3790	Missile contained	
Disc No. 2 Quadrant	3750	347	7.0
Disc No. 3 Quadrant	2740	Missile contained	
Disc No. 4 Quadrant	3190	Missile contained	
Disc No. 5 Quadrant	3632	298	5.0

Missile Impact Areas and Dimensions

Figure 3.5-1 shows the overall width and projected impact areas of a disc quadrant. The dimensions and areas for the disc quadrants of Discs No. 2 and 5 are as follows:

DISC NO.	2	5
A_1, sq ft	4.2	3.0
A_2, sq ft	2.04	2.45
A_3, sq ft	3.4	4.1
W, ft	6.0	5.2
L, ft	2.7	2.3

A_1 - Disc rim projected impact area.

A_2 - Disc end projected impact area.

A_3 - Disc hub projected impact area.

W - Maximum dimension of disc quadrant.

L - Radial dimension of disc quadrant.

Ejection Angles of Disc Quadrants

Based on test results, calculations show the ejection angles for disc quadrants leaving the turbine casing to be as follows:

1. Disc. No. 2: ±5 degrees measured from the vertical radial plane passing through the disc.

2. Disc No. 5: 5 degrees to 25 degrees measured from the vertical radial plane passing through the disc. The disc quadrant will eject only towards the adjacent coupling on the rotor shaft.

The quadrants are assumed to exit 90^{o} apart in elevation when looking along the turbine axial center line. Turbine rotation is clockwise looking at the high pressure end.

It should be remembered that only one disc in one LP cylinder is assumed to burst and produce missiles. This assumption is made because no disc failures are expected at design overspeed, since a disc must be unacceptably flawed to fail at this speed.

3.5.2.3.5 Probability of Impact of the Turbine Missile on Critical Plant Areas

The probability that a turbine missile will damage a critical component and violate the missile protection criteria (Section 3.5.2.1) can be assessed as the product of the three probabilities:

P_f the probability that turbine missiles will be generated and ejected from the turbine

P_i the probability that a turbine missile once ejected will strike a critical component

P_d the probability that a critical component once struck will be disabled.

From Section 3.5.2.3.3, P_f = 1 x 10^{-5} per year, and for preliminary analysis, P_d is assumed equal to 1.

For the Floating Nuclear Plant, P_f is the probability that one LP disc will burst into 90^0 segments at 120% rated speed.

On a single unit land plant, it is generally only necessary to follow the flight of one disc segment. Of the two segments ejected above the turbine shaft, only one will be thrown toward critical equipment while the other is thrown away from the plant. Both segments below the turbine will bury themselves in the turbine condensers or the ground near them. In the floating plant, those downward projected segments can potentially damage and breach the watertight shell below the waterline.

Missiles Ejected above the Turbine Shaft

Missiles ejected above the turbine shaft may be subdivided into two types.

1. Low trajectory missiles (LTM's). LTM's may strike structures or components which are within vertical line-of-sight of their point of ejection and are within the ejection angles (Section 3.5.2.3.4). Many plant areas are shielded from LTM impact by other structures.

2. High trajectory missiles (HTM's). HTM's may be thrown into the air and may drop back onto the plant. HTM's are not restricted to line-of-sight trajectories and potentially may land on more areas containing critical equipment with little loss of energy due to air resistance. However, the probability of impact on a specific area varies with both the distance and the angular orientation of the area with respect to the ejection point.

Preliminary calculations of P_i have been made using the method of Reference 2 for a number of critical plant areas for the single segment of concern considered both as an HTM and an LTM. In assessing P_i for HTM's, no credit has been taken for any protection afforded an area by structures above them. However, no allowance has been made for a falling missile being deflected into a critical area (an accessment of this effect will be made). For all areas the probability was calculated assuming bursting of the disc judged to give the highest probability of impact. Results obtained include the following:

Area	P_i	
	Due to LTM	Due to HTM
Reactor Containment		
a. On the side wall adjacent to the turbine hall	2×10^{-1}	3×10^{-4}
b. On the containment dome	0	3×10^{-4}
Control Room	0	4.3×10^{-4}
Spent Fuel Pit	0	6×10^{-4}
Safeguards Area #1	0	4×10^{-4}
Safeguards Area #2	0	4×10^{-4}
Safeguards Area #3	0	8.3×10^{-4}
Boric Acid Tanks and Pumps	0	12×10^{-4}
Waste Gas Decay Tanks Area	0	6×10^{-4}

The probability calculated for impact on the side wall of the containment assumes that impact occurs within 15° of the normal to the surface. Outside this angle, the missile strikes only a glancing blow, and does not penetrate. The highest probability of this impact is from bursting of the last disc in the generator end of the LP casing nearest the generator.

The probability that the HTM will breach watertight compartments has not been calculated using the method of Reference 2. However, since the platform can take at least the flooding of any two compartments with no unacceptable consequences, the problem becomes one of impact on the junction of compartments to cause three or four compartment flooding. The probability of a missile causing unacceptable multi-compartment flooding by the HTM has been assessed as no higher than 1×10^{-4}.

The combined probability ($P_f \times P_i$) of turbine missiles ejected above the turbine shaft for striking the portion of the side wall of the containment adjacent to the turbine hall is 2×10^{-6} per year, and no greater than 9×10^{-9} per year for any other critical area. The shield building wall is sufficiently thick (3 feet) that the design basis turbine missile will not penetrate it. In addition the containment shell, crane wall and operating deck provide additional missile barriers between the point of impact and the reactor coolant system. An evaluation of the effects of this impact will be made and the consequences will be shown to be acceptable.

Missiles Ejected below the Turbine Shaft

Detailed analyses will be made to determine the effects of, and any protection required against, missiles ejected below the turbine shaft, but some qualitative points may be made.

The trajectories of the missiles below center line are not independent of those above, for the disc quarter segments are assumed to be ejected 90° apart. Ejection of, for example, an HTM within approximately ± 10° of the vertical requires that of the other three segments one will go straight downwards and lose some energy passing through the condenser, while the other two will be ejected practically horizontal. The downward segment could penetrate the bottom shell of the platform and flood one compartment. The two horizontal missiles will not damage the platform unless they are deflected downwards into it. Similarly, for the LTM hit on the containment, the missiles ejected above the shaft will not damage the platform, though those ejected below could.

Not more than two out of the four segments are expected to have the potential to damage the platform and the missile protection criteria (Section 3.5.2.1) will be satisfied.

3.5.2.4 Missiles Resulting from the Crash of a Helicopter on the Plant

The Floating Nuclear Plant will be designed against the missiles which could result from the crash of a passenger-carrying helicopter during landing or take-off maneuvers. The design basis crash is that which would result from either the failure of one helicopter engine or from the helicopter flying into structures adjacent to the landing area. Since the most economic method for transporting supplies to the Floating Nuclear Plant will be by barge or boat, heavy cargo shipments would be lifted by helicopter only in highly unusual and infrequent circumstances. Therefore, it is assumed that only passenger-carrying helicopters will frequently fly near or land on the plants. Accidents caused by helicopters with suspended cargoes will not be the design basis. This basis does not preclude passenger helicopters carrying cargo internally to the plant as required.

Passenger carrying helicopters range from two-seat, single-engined machines to twin-engined helicopters carrying nearly 30 passengers.

A typical large helicopter (Sikorsky S61N, Mark II) used in offshore service can carry either 26 passengers or, alternatively, 8000 lbs of cargo. After failure of one engine, when airborne, this helicopter can sustain flight and safely land in air temperatures up to 90°F.

A review of helicopter accidents in U. S. general aviation for 1969 and 1970[3] shows that the largest single cause of substantial damage or destruction is powerplant failure or malfunction. Other important causes are hard landing, roll over, collision with ground or water, and collision with wires or poles. These accident statistics indicate that failure of one engine or collision with an adjacent structure are the only likely causes of a helicopter accident which could generate missiles while the helicopter is near the floating nuclear plant.

Each of these cases will be assessed, keeping in mind such factors as the ability of multi-engined machines to continue flying on one engine under certain conditions. Fires quite frequently occur after a helicopter crash and a maximum fuel fire will be defined and suitable protection provided. Since passenger-carrying helicopters are economical only over relatively short distances their maximum fuel capacities are of the order of a few hundred (less than 1,000) gallons, so the fire potential is small and no particular problems are foreseen. Specific attention is being paid to the outside air intake arrangements on HVAC Systems to ensure that critically situated intakes can be rapidly closed to prevent fire and smoke from being drawn into the plant.

3.5.2.5 Formulae used for Missile Penetration Calculations

The depth of penetration of missiles into steel or concrete slabs is estimated using the following formulae [4]

Concrete = (Modified Petry Formula)

$$D = K A_P \log_{10}\left(1 + \frac{V^2}{215{,}000}\right)$$

$$D' = D\left(1 + e^{-4\,[(T/D) - 2]}\right)$$

where

D' = actual depth of penetration (ft),

D = depth of penetration for an infinite slab (ft),

T = thickness of the slab (ft).

A_P = sectional pressure $\left(= \frac{\text{missile weight}}{\text{impact area of missile}} \right)$ (psf).

V = impact velocity (ft/sec),

K = experimentally obtained material coefficient for penetration

Steel (Stanford Formula)

$$\frac{E}{D} = \frac{S}{46,500} \left(16,000\ T^2 + 1,500\ \frac{W}{W_s}\ T\right),$$

where

E = critical kinetic energy required for perforation (ft-lb)

D = missile diameter (in.)

S = utilmate tensile strength of the target (steel plate) (psi)

T = Target plate thickness (in.)

W = length of a square side between rigid supports (in.)

W_s = length of a standard width (4 in.)

The ultimate tensile strength is directly reduced by the amount of bilateral tension stress already in the target. The equation is good within the following ranges:

$0.1 < T/D < 0.8$,

$0.002 < T/L < 0.05$,

$10 < L/D < 50$,

$5 < W/D < 8$,

$8 < W/T < 100$,

$70 < V_c < 400$

Where L is missile length (in.) and the missile is assumed to be cylindrical; V_c is velocity (ft/sec).

References

1. "Probability of Turbine-Generator Rotor Failure Leading to the Ejection of External Missiles", General Electric Company memo Report, March, 1971.

2. S. N. Semanderes, "Methods of Determining the Probability of a Turbine Missile Hitting a Particular Plant Region", WCAP-7681, Westinghouse Electric Corporation, Nuclear Energy Systems, February, 1972.

3. "Briefs of Aircraft Accidents Involving Rotorcraft - U. S. General Aviation, 1969, 1970", National Transportation Safety Board, NTSB -AMM-71-8 and NTSB-AMM-72-3.

4. Gwaltney, R. C., "Missile Generation and Protection in Light Water Cooled Power Reactor Plants", USAEC Report ORNL-NSIS-22, September 1968.

3.6 CRITERIA FOR PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH A LOSS OF COOLANT ACCIDENT

In the design of the Floating Nuclear Power Plant special provisions are made for protecting the public against the consequences of major mechanical accidents including a loss of coolant (LOCA) or steam/feedwater line break accident. This section describes the philosophy, criteria, and the measures that are used to assure such protection against the dynamic effects resulting from a LOCA can be shown to satisfy 10 CFR 100 guidelines. Consideration is given to the potential rupture of both reactor coolant pipe and reactor coolant branch lines.

The dynamic effects under consideration include the following:

1. Jet forces resulting from the release of high pressure steam or water from a ruptured line.

2. Pipe whip caused by the formation of a plastic hinge in a pipe due to a rupture somewhere else in the same pipe.

3. Missiles which can be generated in coincidence with an accident.

3.6.1 GENERAL CRITERIA AND PHILOSOPHY

To ensure that 10 CFR 100 limits can be satisfied, core and containment engineered safety features are provided for core cooling, boration, and

activity confinement in case of a loss-of-coolant accident.

To assure this capability, consideration is given to the consequential effects of a break in the reactor coolant pipe itself to the extent that:

1. The minimum performance capabilities of the safety features systems are not reduced below those required to protect against the postulated break.

2. The containment leaktightness is not decreased below the design value.

3. Propagation of damage is limited in type and/or degree to the extent that:

   a. A pipe break that is not a loss of reactor coolant or steam-feedwater line break would not cause a loss of reactor coolant or steam-feedwater line break.

   b. A pipe break that is a small loss of coolant (ID $\leq$ 4 inches) and which in itself would be unlikely to result in clad damage would not propagate to a larger pipe break where damage might be expected.

   c. A reactor coolant system pipe break would not cause a steam-feedwater system pipe break and vice versa.

To accomplish this in the design, a combination of component restraints, barriers, and layout ensures that for a loss of coolant or steam-feedwater line break, propagation of damage from the original event is limited, and that the containment, equipment supports, and engineered safety features are protected and available. For a pipe break that does not constitute a loss of reactor coolant, component supports, barriers, and layout provisions ensure that a loss of reactor coolant does not result, and that systems required for plant cold shutdown are protected.

3.6.2 REACTOR COOLANT PIPING RUPTURE

This subsection discusses analysis and design considered for a reactor coolant pipe rupture. Equipment, piping, containment, and supports are designed to resist the forces generated by this postulated accident. Subdivision 3.9.2.7 discusses pipe rupture criteria for piping in systems other than those which form the Reactor Coolant System boundary.

3.6.2.1 Location and Size of Rupture

Within the reactor coolant loop, circumferential and longitudinal breaks are postulated to determine the most critical loading conditions in order to design and analyze supports for the primary loop piping system and its equipment. Pipe rupture is assumed to occur anywhere within the reactor coolant system where large changes in flexibility occur; e.g., elbows, tees, anchors, and pipe vessel connections.

Eight break locations within the reactor coolant system piping are postulated. These locations and their anticipated effects are summarized in Table 3.6-1 and Figure 3.6-1.

The circumferential severance of a reactor coolant pipe is conservatively assumed to result in a double-ended pipe rupture; i.e., fluid issuing from both ends of the broken pipe. Circumferential breaks in connected branch lines; i.e., emergency core cooling system, residual heat removal system, etc., will in general result in a single-ended pipe rupture.

Fracture mechanics investigation of the growth of cracks in pipes has shown that longitudinal breaks need not be considered.[1] If a flaw does progress through the pipe wall, it will produce detectable leakage before becoming unstable and propagating to a larger rupture.[2] Similar studies have also shown that circumferential breaks need not be considered at points where no large change in flexibility occurs.[3]

Nevertheless, for conservatism, consideration is given to the occurrence of either a longitudinal or a circumferential break at each possible break location. Examination of the state of stress in the vicinity of the postulated break location may identify the most probably type of break. Design considerations will be on the basis of the postulated rupture occurrence of the most probably type of break. If no significant difference between hoop and axial stresses has been determined, then both types of breaks will be considered.

3.6.2.2 Thrust-Time Relationship

The dynamic forces caused by a rupture of a reactor coolant pipe comprise the specific jet thrust at the break location, and the internal hydraulic forces resulting from the acceleration of the fluid within the broken and unbroken loops. The pipe and equipment supports are analyzed for breaks in the hot, cold, and crossover legs. For the pipe breaks discussed above, a dynamic stress analysis of the reactor coolant loop support systems determines component and component support loadings.

The computer code used to generate the dynamic forcing function takes into account the dynamic fluid effects such as inertial forces, acoustic waves, and frictional losses.

The primary result of the hydraulic blowdown analysis is the dynamic pressure distribution. Pressure is the most important factor in determining the loads on the loop and components.

For the blowdown analysis of the reactor coolant loop support system, the points of force application occur where the flow changes direction as in the elbows around the loop, or where the boundary changes size, as through the pump, in the steam generator plenums, at the top of the steam generator tubes, etc. Thrust calculations are made at each of these nodes.

Forces are also developed within the reactor pressure vessel, and are considered along with the reactor coolant loop hydraulic forces. They affect

the design of the reactor vessel supports and also affect the primary loop stresses. Hydraulic forces are calculated for the broken as well as unbroken loops. Figure 3.6-2, Sections A-A, B-B, and C-C show the nodes where thrust values are calculated for a typical loop. The computer codes used for developing the dynamic forcing function and for thrust calculations are SATAN and STHRUST respectively, which are described in Appendix 3A.

3.6.2.3 Method of Dynamic Analysis for Blowdown Loading

The time-history method is used in the blowdown analysis of the reactor coolant loop (RCL) to determine the loads on the equipment and supports and stresses in the pipe due to postulated pipe ruptures.

The RCL including the reactor pressure vessel, steam generator, pump, and supports is represented by a mathematical model consisting of the loop geometry, lumped masses, single springs and the support stiffness matrices. The initial displacement configuration of the mass points is computed by applying the initial steady-state hydraulic forces to the unbroken RCL model using the computer program WESTDYN (see Appendix 3A). For this calculation, the support stiffness matrices for the static behavior are incorporated in the model.

For the dynamic solution, the RCL model is modified to simulate the physical rupture of the pipe for the postulated case under consideration. This model includes support stiffness matrices for dynamic behavior selected on the basis of anticipated displacement response at the support points. The

natural frequencies and normal mode for the modified RCL model are also determined by using the computer program WESTDYN. The hydraulic forces as described in subsection 3.6.2.2 are transformed to the RCL model coordinate system and applied to the appropriate mass points in the RCL model.

The support stiffness matrices representing the support structure in the RCL model are determined separately for each support considered. Different sets of matrices are evaluated for each support according to the anticipated movement of the support point. The computer program STRUDL (see Appendix 3A) used in these evaluations.

The initial displacements, the natural frequencies, normal modes, and the transformed time-history thrusts are used together as input to the computer program FIXFM (see Appendix 3A) to calculate the dynamic time-history displacement response for the dynamic degrees of freedom in the RCL model. The displacement response at support points is reviewed to validate the results. If the calculated response does not match the anticipated response, the dynamic solution is revised using a new set of support stiffness matrices for dynamic behavior. This procedure is repeated until the displacement response at support points agrees with that anticipated.

The time-history dynamic displacement solution is then used in combination with the support stiffness matrices to compute the support loads. It is also used to compute the dynamic loads on the equipment and dynamic stresses in the pipe, using the computer program WESTDYN-2 (see Appendix 3A).

3.6.2.4 Equipment Load Combinations and Allowable Stresses

Pipe ruptures associated with a loss-of-coolant accident fall into the category of "faulted conditions" under the operating conditions as defined in the ASME B & PV Code Section III. The load combination for the faulted condition is discussed in Section 3.9. The allowable stress shall conform with the draft Appendix F of the ASME B & PV Code Section III.

3.6.3 REACTOR COOLANT BRANCH LINE RUPTURE

This subsection describes the LOCA boundary limits and the protection criteria against dynamic effects resulting from rupture of branch lines. A branch line in this subsection is defined as large if its ID is greater than 4 inches, and as small if its ID is equal to or less than 4 inches.

3.6.3.1 Loss of Reactor Coolant Accident Boundary Limits

The branch line connections to the reactor coolant system are of five general cases, as illustrated in Figure 3.6-3. These cases are generally defined by the fact that they include lines either incoming to or outgoing from the reactor coolant system and the various valving arrangements. A rupture of this piping can cause an uncontrolled loss of reactor coolant, depending on the precise location of the break and the particular line configuration.

A loss of reactor coolant is assumed to occur from a pipe break upstream of the restraint of the second normally open automatic isolation valve (Case II) on outgoing lines and out to the second check valve (Case III) on incoming lines. A pipe break beyond the restraint or second check valve does not

result in an uncontrolled loss of reactor coolant provided that one of the two valves in the line closes. Therefore, both of the automatic isolation valves are suitably protected and restrained as close to the valves as possible so that a pipe break beyond the restraint will not jeopardize the integrity and operability of the valves. Further, periodic testing of the capability of the valves to perform their intended function is scheduled. This criterion takes credit for only one of the two valves performing the intended function.

For a normally closed isolation valve (Case I) or a normally closed incoming check valve (Case IV), a loss of reactor coolant will occur if the pipe breaks on the reactor coolant side of the valve, but will not occur if the break is on the other side of the closed valve.

3.6.3.2 Protection Criteria against Dynamic Effects Resulting from Rupture of Branch Lines

3.6.3.2.1 General Criteria

The general criteria are as follows:

1. The minimum performance capabilities of the engineered safety features are not reduced below those that are required to protect against a postulated break.

2. If a pipe rupture results in a LOCA, the containment integrity and leaktightness is maintained.

3. Where restraints on the lines are necessary to prevent impact on, and subsequent damage to, the adjacent equipment or piping, restraint type and spacing are provided so that a plastic hinge on the pipe at the two support points closest to the break is not formed.

4. If the layout is such that whipping of the two free sections cannot impact equipment or other pipes for which protection is required, or if barriers provide the necessary protection, plastic hinge formation may be allowed.

3.6.3.2.2 Large Line Rupture Resulting in a Loss of Coolant Accident

If one of the large lines connected to the reactor coolant system ruptures and results in a LOCA, the piping is arranged or restrained to meet the following criteria:

1. Loss of containment leaktightness must be prevented.

2. Rupture of the steam-feedwater lines must be prevented.

3. The break propagation criteria require that: 1) The necessary capacity of the accumulators and low-head safety injection system to the remaining loops must be maintained in order to meet the core cooling

criteria. More specifically, propagation of the break to the unaffected loop must be prevented. 2) The break size must not be permitted to enlarge such that the severity of the LOCA is substantially increased. More specifically, propagation of the break in the affected loop is permitted to occur but must not exceed 20 percent of the area of the line that initially ruptured.

3.6.3.2.3 Large Line Rupture Not Resulting in a Loss of Coolant Accident.

If a large line ruptures but does not result in a LOCA, the piping is arranged or restrained to meet the following criteria:

1. A loss of reactor coolant accident must not occur.

2. A steam-feedwater line break accident must not occur.

3. Boration capability must be maintained.

4. Cold shutdown capability must be maintained.

3.6.3.2.4 Small Line Rupture Resulting in a Loss of Coolant Accident

If one of the small pressurized lines connected to the reactor coolant system ruptures and results in a LOCA, the piping is restrained or arranged to meet the following criteria:

1. Loss of containment leaktightness must be prevented.

2. Rupture of the steam- feedwater lines must be prevented.

3. Break propagation must be limited to the affected leg.

4. Propagation of the break in the affected leg must be limited to 12.5 square inches (4 inches ID). The exception to this case is when the initiating small break is the high head safety injection line. Further propagation is not permitted for this case.

5. Damage to the high head safety injection lines connected to the other leg of the affected loop or to the other loops must be prevented.

6. Propagation of the break to the high head safety injection line connected to the affected leg must be prevented if the line break results in a loss of core cooling capability due to a spilling injection line.

As stated above, a small pipe break in one of the legs of a given loop must not cause a break in the opposite leg of the same loop or in other loops. A rupture of the resistance thermocouple device (RTD) bypass loop would constitute an exception to this criteria. If a break occurs at any location in the RTD bypass, however, the resulting blowdown, because of the relatively high flow resistance in the line, is less severe than from a rupture

directly adjacent to the reactor coolant pipe. Safety injection capacity for this case is provided via the high-head connections.

3.6.3.2.5 Small Line Rupture Not Resulting in a Loss of Coolant Accident

If one of the small pressurized lines ruptures but does not result in a LOCA, the piping is restrained or arranged to meet the same criteria as stated in paragraph 3.6.3.2.3.

3.6.4 PIPE SUPPORT ARRANGEMENT TO PREVENT PLASTIC HINGE FORMATION AFTER A PIPE BREAK

Where necessary to prevent pipe whip in the branch lines to meet the protection criteria, restraints are provided with the proper arrangement and spacing to prevent a plastic hinge mechanism from forming in the pipe as a result of the forces associated with a pipe rupture. The plastic hinge moment is evaluated using limit analysis and the load moment is calculated from the spacing of the restraints and the equivalent static force applied at the postulated break to prevent plastic hinge formation.

The load moment must be within the allowable limit as discussed in subsection 3.6.2.4.

Reactor coolant line pipe restraints are also required to serve the following purposes:

1. To limit displacement or rotation of the severed reactor coolant pipe and thereby reduce the nozzle loadings and also make the direction of the loading predictable.

2. To limit unrestricted displacement or rotation of reactor coolant pipe which might result from the formation of a plastic hinge mechanism.

The required pipe whip restraints are designed based on previous experience in nuclear plant design and are verified by dynamic analysis. Revisions are made as necessary in the final design.

3.6.5 EQUIPMENT SUPPORT PROTECTION CRITERIA

The equipment support structures (reactor pressure vessel, steam generator, and reactor coolant pump) must be protected from the impact of large whipping pipes and jet forces, or be designed to resist such impact. Adequate separation of equipment and piping,or the use of barriers or pipe restraints adequate to prevent the formation of a plastic hinge mechanism provide this protection (see Paragraph 3.6.4). If any branch pipes are supported from equipment support structures, the reaction force resulting from a rupture of these lines is considered in designing the equipment supports.

An adequate support design for the reactor pressure vessel, steam generator, and reactor coolant pump is achieved that will not overstress the primary

loop piping, the equipment, or the equipment support systems. The equipment support structures are also designed to prevent both uplift and overturning of the equipment under the action of normal and abnormal forces.

3.6.6 CONTAINMENT PROTECTION SYSTEMS

3.6.6.1 General

The containment shell, divider barrier, containment base plate and containment safety systems are designed for, or protected from, the forces and other effects associated with pipe breaks.

The containment shell, containment base plate, and containment safety systems are also designed to contain the pressure associated with reactor coolant system pipe ruptures up to and including the double ended severance of the reactor coolant piping. Containment shell overpressurization is prevented by protecting the containment safety systems and by properly supporting major components to prevent propagation of damage between the reactor coolant system and steam/feedwater systems.

3.6.6.2 Reactor Coolant System Ruptures

The secondary shield which includes the concrete and steel structures surrounding the reactor coolant loop is the principal protective barrier separating the reactor coolant system from the containment shell and containment safety systems. The containment shell and containment safety systems are outside the secondary shield.

The secondary shield is protected by 1) designing it for the jet loads and compartment pressures associated with the pipe break as discussed in subsection 3.6.7, and by 2) properly supporting major components to prevent them from hitting and damaging the barrier as discussed in paragraph 3.6.5.

The divider barrier is designed for the differential pressure between the lower and upper compartments inside the containment shell arising from pressurization due to a postulated coolant pipe rupture. It is also designed for, or protected from, the dynamic effects associated with a reactor coolant or steam line break inside the secondary shield. The essential function of the divider barrier (to direct flow to the ice condenser during a loss of coolant accident) is therefore always maintained.

3.6.7 CONSEQUENCE WITHIN CONTAINMENT OF PIPE FAILURE

The discharge of reactor coolant from a reactor coolant pipe rupture is accompanied by jet forces and pressurization associated with expansion of the steam-water mixture. The containment shell, containment safety systems, and engineered safety features limit the consequences of such a rupture and are not jeopardized by structural failures induced by these consequential jet and pressure loadings. This is assured by designing the containment interior structures to withstand the resultant forces, thereby preventing their collapse and damage to the above-mentioned essential systems. The jet and differential pressure forces are applied concurrently, but the peak

for each force will occur at different times. The containment interior structures are designed for the dynamic nature of these forces as discussed in subsection 3.8.2.9.

3.6.8 MAJOR EQUIPMENT FAILURE CONSIDERATIONS

Gross failure of vessels is not considered credible because of the material characteristics; inspections; quality control during fabrication, erection, and operation; conservative design; and prudent operation as applied to the particular component. Therefore, no special design provisions are made for such failures. This includes failure of:

Reactor vessel

Steam generators

Pressurizer

Reactor coolant pumps

Accumulators

Heat exchangers

TABLE 3.6-1

REACTOR COOLANT SYSTEM POSTULATED PIPE RUPTURE LOADINGS

Break Location[a]	Component(s) and/or Support(s) Expected to be Significantly Loaded	Main Type(s) of Loading Expected
1	steam generator reactor pressure vessel	vertical; horizontal moment
2	reactor pressure vessel steam generator	shear moment; shear
3	reactor pressure vessel reactor coolant pump	shear torsion
4	reactor pressure vessel reactor coolant pump	torsion shear
5	reactor coolant pump steam generator	vertical moment
6	reactor coolant pump steam generator	shear; moment shear; moment
7	reactor coolant pump steam generator	moment vertical
8	reactor coolant pump steam generator	moment vertical; horizontal

[a] As shown in Figure 3.6-1

REFERENCES

(1) Paris, P.C., The Fracture Mechanics Approach to Fatigue, Proc. Tenth Sagamore Army Materials Research Conference, August 1963, Syracuse University Press (1964).

(2) Eiber, R. J., et al, Investigation of the Initiation and Extent of Ductile Pipe Rupture, Batelle Memorial Institute - 1866 (July 1969).

(3) Szy Slow Ski, et al, Determination of Design Pipe Breaks for the Westinghouse Reactor Coolant System, WCAP-7503, Rev. 1, February 1972.

3.7 SEISMIC AND PLATFORM MOTION DESIGN

Motions to be used in the design of the Floating Nuclear Plant are defined in the following paragraphs. As described in Chapter 2 it is the responsibility of the Owner to design the site related structures such that the plant motions will not exceed those specified for the plant design.

3.7.1 INPUT CRITERIA

3.7.1.1 Seismic Input Criteria

The platform response spectra used for the design of safety related structures and components for the Floating Nuclear Plant are shown for the OBE and DBE in Figures 3.7-1, 3.7-2, 3.7-3 and 3.7-4 for the vertical and horizontal motions. Table 3.7.1-1 shows the percentage of critical damping values used for all safety related structures, systems and components.

3.7.1.2 Input Criteria for Wave Motions

The Floating Nuclear Plant is designed for wave conditions within the breakwater protected basin adjacent to the plant. The wave conditions are specified in two manners intended to be consistent, one used in the structural design of the platform, the other used to define plant motions.

TABLE 3.7.1-1

DAMPING VALUES

Component or Structure	Critical Damping (Percent)		
	Actual Stresses Not More than 0.5 Yield or 0.5 Concrete Ultimate Strength	Actual Stresses at or Just Below Yield or Concrete Ultimate Strength	Actual Stresses Above Yield
1. Reinforced Concrete	5.0	7.0	10.0
2. Prestressed Concrete	2.0	5.0	10.0
3. Steel Assemblies			
Bolted	5.0	10.0	20.0
Welded	2.0	5.0	7.0
4. Piping Systems	0.5	2.0	5.0
5. Fluid system and electrical components not otherwise defined above	2.0	5.0	10.0

Platform Structural Design

The platform is designed for a wave condition in the longitudinal or transverse directions with the length of the wave equal to the length or width of the platform respectively. The wave heights from crest to trough used in the platform structural design are as follows:

Operating Basis Wave,	V = 4 feet
Design Basis Wave,	V_S = 6 feet

Waves of greater amplitude are permitted provided their effect on the hull girder bending of the platform is less severe than the specified wave conditions. The higher amplitude wave pattern would be such that the wave length is significantly greater or less than 400 feet.

Safety-related superstructure and equipment are designed for platform deformations calculated for the above waves.

Platform Motions

Operating Basis (V)

Safety related structures and equipment are designed to operate for any combination of the following motions caused by wind and wave conditions:

1. Combined pitch and roll: Maximum accelerations not exceeding those due to motion having an amplitude of pitch plus amplitude of roll equal to 2 degrees at a period of 13 seconds.

2. Heave: Vertical acceleration = 0.02g.

3. Combined surge, sway & yaw: Horizontal acceleration at any point on platform in any direction equals 0.03g.

Note: The plant may be operated in an initial out-of-trim condition. However, under this condition the roll and pitch permitted will be decreased such that the sum of the out-of trim and motion heel angles about the longitudinal and transverse axes does not exceed 2°.

Design Basis (V_S)

Safety-related structures and equipment are designed to achieve and maintain safe shutdown for any combination of the following motions.

1. Combined pitch and roll: Maximum accelerations not exceeding those due to motion having an amplitude of pitch plus amplitude of roll equal to 3 degrees at a period of 13 seconds.

2. Heave: Vertical acceleration = 0.03g.

3. Combined surge, sway & yaw: Horizontal acceleration at any point on platform in any direction equals 0.03g.

3.7.1.3 Input Criteria for Wind Induced Motions

Operating Bases

The Operating Basis Wind is defined in subsection 3.3.1. The motions induced by the wind are small in comparison to the motion due to the wave. The specifications for motion included in paragraph 3.7.1.2 are considered to include Operating Basis Wind Motions.

Design Bases

The Design Basis Wind is the tornado defined in subsection 3.3.3.

The motions used in designing the plant to maintain safe shutdown are analyzed as described in Section 3.3 and are specified for design of safety-related equipment and structures as follows:

1. Combined pitch and roll: Maximum accelerations not exceeding those due to motion having an amplitude of pitch plus amplitude of roll equal to 2 degrees at a period of 13 seconds.

2. Heave: Vertical acceleration - 0.02g.

3. Combined surge, sway & yaw: Horizontal acceleration at any point on platform in any direction equals 0.02g.

3.7.1.4 Combination of Environmental Conditions

The Floating Nuclear Plant is designed for the following combinations of environmental conditions. The minimum depth of water at the site will ensure that the plant will not contact the ground under any condition. Instrumentation is provided to indicate the platform attitude, roll and pitch to the operator. This instrumentation is described in paragraph 3.7.4. The plant will be shut down if the Operating Basis conditions are exceeded.

Operating Basis Environmental Conditions	$H + W + V$
	$H + E$
Design Basis Environmental Conditions	$H_S + W + V_S$
	$H + W_S$
	$H + E_S$

Where H = Operating Basis Tide

H_S = Design Basis Tide

W = Operating Basis Wind

W_S = Design Basis Wind (=Tornado)

E = Operating Basis Earthquake

E_S = Design Basis Earthquake

V = Operating Basis Wave

V_S = Design Basis Wave

3.7.1.5 Equivalent Static Accelerations

All of the above motions except the vertical component of the seismic motion are considered as static loads in the design of the safety-related structures and equipment on the plant. These motions occur at a period in excess of 5 seconds and will not be amplified by the relatively rapid plant structures.

Table 3.7.1-2 summarizes the equivalent static accelerations for the motions defined in the previous paragraphs. Operating Basis Motions and Design Basis Motions are defined from these accelerations. Safety-related structures and equipment are designed for Operating Basis and Design Basis magnitudes of Motions, and Earthquakes and for towing as specified in the second part of Table 3.7.1-2.

The accelerations defined in Table 3.7.1-2 in a plane parallel to the platform are applicable to either the longitudinal or the transverse direction.

Each of these nearly horizontal accelerations will be taken individually and combined with the vertical accelerations for the design of safety-related structures and equipment.

TABLE 3.7.1-2

EQUIVALENT STATIC ACCELERATIONS DUE TO PLATFORM MOTION

Load	Acceleration in plane parallel to platform (longitudinal or transverse)		*Acceleration normal to platform		Notes
	Elev. 40'	Elev. 140'	Heave	At distance of 100' from longitudinal or transverse center line (due to roll or pitch)	
ENVIRONMENTALLY INDUCED MOTIONS					
Operating Basis Wave	0.046g	0.07g	0.02g	0.026g	
Design Basis Wave	0.069g	0.11g	0.03g	0.039g	
Operating Basis Wind	0.0	0.0	0.0	0.0	Included in Operating Basis Wave
Design Basis Wind	0.046g	0.07g	0.02g	0.026g	
DESIGN MOTIONS					
Operating Basis Motions	0.05g	0.08g	0.02g	0.026g	
Design Basis Motions	0.07g	0.11g	0.03g	0.004g	
Operating Basis Ethqk.	0.05g	0.05g			Vertical accelerations to be developed by dynamic analysis
Design Basis Earthquake	0.10g	0.10g			
Towing	0.17g	0.30g	0.07g	0.15g	

*The total equivalent static acceleration normal to the platform is given by the downward gravity force plus or minus the heave and pitch or roll accelerations listed.

The accelerations in Table 3.7.1-2 are shown for typical locations in the plant. Accelerations at other locations are determined by linear interpolation or extrapolation from the magnitudes specified.

3.7.2 SEISMIC SYSTEM ANALYSIS

3.7.2.1 Structures

Safety Category I structures are analyzed for the horizontal component of the earthquake using the equivalent static accelerations defined in subsection 3.7.1. Dynamic analyses for horizontal motion will be performed only on components such as tanks with a free liquid surface which may have natural frequencies below 0.5 cycles per sec.

Vertical dynamic analyses will be performed on Category I structures in order to develop equivalent static loads for the design of the structures and to develop floor response spectra for the design of the equipment. These analyses will be performed on lumped mass models. Modal analysis response spectra methods will be used for the design of the structures. Modal responses will be combined by the square root of the sum of the squares.

Modal analysis time history methods will be used to develop the floor response spectra. To account for possible variations in structural properties the calculated floor response spectrum will be shifted to ± ten percent of the period and the envelope will be specified as the floor re-

sponse spectrum.

The containment shell and shield building are modeled as shells of revolution and are analyzed for vertical motion using the axisymmetric shell program described in subsection 3.8.2.

All other structures are modeled with lumped masses and a set of spring stiffnesses representing the properties of the structure. In constructing these models the locations of lumped masses are chosen at floor levels and at other points carefully selected to adequately represent the dynamic and elastic properties of the structure.

Seismic design control measures are described in subsection 3.7.5. Damping factors used for the seismic design of Category I structures are identified in subsection 3.7.1. Allowable stress criteria are defined in subsection 3.8.2 for the containment structures, in section 3.12 for the platform and subsection 3.8.1 for other Category I structures.

3.7.2.2 Systems and Components

3.7.2.2.1 System Piping

The safety-related sections of piping systems identified on the system's flow diagrams as Safety Class 1, 2, or 3 will be analyzed as follows:

1. Seismic Criteria

Seismically designed piping will include earthquake loads represented by earthquake response spectra at the various locations in the Category I structures. For a piping system spanning between two or more locations, the spectrum curve associated with the location closest to, or more severe than, the center of mass of the piping system will be used.

Each piping system will be evaluated for seismic excitation in each of two orthogonal horizontal directions combined with excitation in the vertical direction. The maximum moments (stress) at any point in the piping system will be taken as the larger value obtained from either of these analyses.

For the evaluation of relative support motions in the seismic analysis of piping systems interconnecting two or more primary structures, the maximum relative movement between structures is assumed, and the piping system is subjected to these movements through the piping system supports and restraints. The stresses in the piping resulting from these imposed restraint movements are considered to act concurrently with other seismic and thermal stresses.

All piping will be classified into two categories, rigid or flexible. For the vertical mode, rigid piping will be that which has a natural frequency greater than 35 Hz. For the horizontal mode, rigid piping will be that which has a natural frequency greater than 0.5 Hz. All piping with natural frequencies less than 35 Hz and 0.5 Hz for the vertical and horizontal modes respectively will be classified as flexible.

2. Seismic Analysis for Rigid Piping

All rigid piping will be designed for a uniform static coefficient equal to the maximum floor acceleration corresponding to the appropriate building location for each piping system. Paragraph 3.7.3.8 describes the simplified analysis for rigid piping. Table 3.7.1-2 shows the equivalent static accelerations due to platform motion which will be used in the analysis.

3. Method of Analysis for Flexible Piping

Each pipe loop is idealized as a mathematical model consisting of lumped masses connected by elastic members. Lumped masses are located at carefully selected points in order to adequately represent the dynamic and elastic characteristics of the piping system.

The weight of valves and other in-line components are included in the model with eccentric masses considered. If the mass or flexibility of the equipment or supports can participate significantly in the response of the piping system, the equipment or supports are also included in the model; but if a piping system can be considered fixed at the equipment nozzle or at the support, only the effect of seismic anchor movement is considered. Appropriate damping factors, selected from Table 3.7.1-1 are also considered in the analysis of piping systems. The flexibility calculations include the effects of the torsional, bending, shear, and axial deformations. After the mathematical model of the system is constructed, a dynamic analysis is performed by the response spectrum technique utilizing the Westinghouse Electric Corporation's WESTDYN computer code or an equivalent code (A description of WESTDYN is given in Appendix 3A).

The analysis of safety-related piping and components supplied by Westinghouse is described in Subsection 3.7.2 of RESAR 3.

4. Simplified Dynamic Analysis for Flexible Piping

The conditions under which a simplified dynamic analysis of piping may be performed are given in Paragraph 3.7.3.8.

5. Static Analysis for Non-Seismic Motion Conditions

Piping systems will also be analyzed for effects due to platform motion conditions when the plant is either under tow or at the operating site. These motions occur at a period in excess of 5 seconds; therefore, a static analysis will be performed using the equivalent static accelerations shown in Table 3.7.1-2. The static analysis will consider relative movements between support points due to deformation of the platform as a result of temperature gradients and wind and wave action. Motion input criteria are described in Subsection 3.7.1.

3.7.2.2.2 System Components

In accordance with applicable response spectra curves as developed from the methods described in Paragraph 3.7.2.1, system components will be evaluated to verify their capability to withstand design loadings as described in Table 3.9.2-1. The equipment will be evaluated by theoretical analysis and/or performance testing. Where practical, equipment structural supports will be designed such that the fundamental frequency of the equipment and its support system will be above 35 Hz in the vertical direction in order that amplification of seismic disturbances within the support system can be avoided.

1. Cylindrical Shell Type Equipment and Components and Their Supports

The equipment specification for each tank, heat exchanger, and pressure vessel specifies the particular load for which the component must be designed. The loading combinations and stress criteria to which the component must be designed are shown in Table 3.9.2-1.

Seismic loading for each component is specified in the equipment specification in the form of response spectra. (See Figure 3.7-5). A dynamic analysis will be performed by either the manufacturer or the applicant to verify conformance with the specifications. The result of piping flexibility analysis shows the calculated loads on nozzles. These loads are compared with the design loads to assure a compatible design.

The acceleration response spectra for the applicable support area will be used to perform the analysis as follows:

All natural frequencies (below 35 Hz in the vertical direction) of the equipment will be determined. A modal analysis (if applicable) will be performed to determine the effects of each mode. The net effect of combining modes is equal to the square root of the sum of the squares of the effects of each mode. In determining

the adequacy of the equipment, two cases of simultaneous loading in the horizontal and vertical directions are considered. Horizontal equivalent static accelerations will be selected from Table 3.7.1-2.

The equipment specifications require that seismically designed mechanical equipment be capable of operating prior to, during and after a seismic occurrence.

2. Valves and Pumps

The analytical methods and criteria used to evaluate safety-related valves and pumps are described in Paragraph 3.9.2.9.

3. Reactor Core Support Structure Analysis

A description of the seismic analysis methods, stress criteria, and mathematical models used for the reactor core support structure is presented in Section 3.9 of RESAR-3.

4. Electrical Components

Seismic analysis and safety classifications for electrical components are defined in Section 3.10.

3.7.2.2.3 Criteria Used to Lump Masses

Refer to Paragraphs 3.7.2.2.1 and 3.7.3.9 for criteria to lump masses for systems and components.

Refer to Subsection 3.7.5 for the procedures used to assure that all the required inputs and/or responses required by the different design organizations for safety-class equipment are compatible.

3.7.2.2.4 Seismic Design Control

Seismic Design Control is described in Subsection 3.7.5.

3.7.2.2.5 Damping Factors

Damping factors used for the seismic analysis of safety-related equipment are defined in Table 3.7.1-1.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

In addition to the loads imposed on the system under normal operating conditions, the design of equipment and equipment supports requires that consideration be given to abnormal loading conditions such as would be experienced during an earthquake, or in a seaway when the plant is under tow, or moored at the operating site. Two types of seismic loadings are considered: Operational Basis Earthquake (OBE) and Design Basis Earthquake (DBE).

For the OBE loading condition, safety-related systems and equipment are designed to remain functional and to operate within the design stress limits established by Section III of the ASME B and PV Code for the Upset Condition.

Seismic design will assure capability to shutdown and maintain the plant in a safe condition following the DBE. In this case it is only necessary to ensure that required critical structures and equipment do not lose their capability to perform their safety function.

Two types of wave and wind motion (Operating Basis Motion and Design Basis Motion) and dynamic effects during towing are also considered. See Subsection 3.7.1 for a discussion of these motions. Table 3.8.2-1 shows how

these motion effects are applied in the loading and stress criteria.

For information on the seismic analysis of Westinghouse supplied components see RESAR-3, Subsection 3.7.2 and 3.7.3.

3.7.3.1 Earthquake Cycles for Subsystem Analysis

See Sections 3.7.1 and 3.7.2 for seismic criteria relating to the plant. The seismic fatigue criterion applicable to subsystems and components is as follows:

"20 occurrences of Operating Basis Earthquake and
20 cycles of maximum response for each occurrence."

3.7.3.2 Basis for Frequency Selection

The floating plant and its mooring provide a very flexible system in the horizontal plane. As discussed in Subsections 3.7.1 and 3.7.2, the horizontal natural frequency of the plant/mooring system is low enough to permit equivalent static acceleration factors to be used in lieu of dynamic analysis when analyzing equipment and supports with natural frequencies greater than 0.5 cycles per second. For the vertical direction, equipment and supports with natural frequencies less than 35 cycles per second will be subjected to dynamic analysis using the floor response spectra for

the particular location. Frequencies above 35 cycles per second are above the range of seismic excitation and will not cause vibratory amplification; therefore, equipment with vertical natural frequencies above 35 cycles per second will be analyzed using an equivalent static acceleration factor.

3.7.3.3 Modal Response Combinations by SRSS

The phrase "square root of the sum of the squares" and abbreviated by "SRSS" is used to describe the method of combining all modal responses when used herein.

3.7.3.4 Modal Response for Closely Spaced Frequencies

The criteria for combining modal responses (shears, moments, stresses, deflections, and/or accelerations) when modal frequencies are closely spaced and a response spectrum modal analysis is used are as follows:

1. The SRSS method of combining all modal responses is used.

2. All modes up to a frequency of 35 Hz are used in the analysis. A sufficient number of modes are considered to describe the motion.

3. Where closely spaced frequencies of two or more modes occur, only

these modes' responses are combined in an absolute manner; the resulting total is treated as that of a pseudo-mode and then combined with the rest of the modes in an SRSS manner.

For Westinghouse-supplied mechanical equipment when two significantly closely spaced modal frequencies occur and their eigenvectors are parallel, the combined total response is obtained by adding the SRSS of all other modes to the absolute value of one of the closely spaced modes as described in RESAR-3, Paragraph 3.7.3.1.

3.7.3.5 Equivalent Static Loads

The simplified seismic analysis method, where an equivalent static load is determined by using the peak of the floor response spectrum, is used only where the equipment is primarily a single-degree-of-freedom system and the frequency of the component or equipment is not analyzed. This method provides an approach to the analysis of complex equipment with some horizontal frequency modes below 0.5 cycles per second and some vertical modes below 35 cycles per second. The justification for this approach is provided where specific uses are given.

3.7.3.6 Design Criteria and Analytical Methods for Piping

The forces and moments induced during earthquake due to relative displace-

ments between piping support points, i.e., floors and components, at different elevations within the plant and between the various areas of the plant, are obtained by dynamic analysis in which the combined amplitude is obtained by calculating the square root of the sum of the squares of the amplitudes. In those cases where the fundamental frequencies of the floors, components or structural areas are closely spaced, the responses are combined as described in Paragraph 3.7.3.4. Seismic criteria and methods of analyses are described in Subsection 3.7.2.

3.7.3.7 Combined Response Loading for Seismic Design

See Subsection 3.7.2 for the seismic design of piping and components.

3.7.3.8 Simplified Dynamic Analysis for Piping

The simplified seismic analysis method and procedures used for seismic design require the piping system to be divided by means of restraints and anchors into a series of simple shapes for which the natural frequencies have been established by previous analyses or calculations. By causing the systems to be supported to fall within the rigid range or showing acceptable stresses at the peak value of the appropriate response curve, the stress limits specified by the Code are satisfied.

Numerical values indicating typical results from the simplified dynamic

analysis methods and the dynamic response spectra analytical methods will indicate the magnitude by which the resonant periods of the piping spans differ from the predominate supporting building and component periods. Generally, the frequency ratio will be greater than 2.

3.7.3.9 Torsion Effects in Seismic Analysis For Piping

The torsional effects of valves and other eccentric masses (i.e., valve operators) in the seismic piping analysis is considered by using a cantilevered member with a lumped mass at the free end to model the eccentric weight of the valve and motor operator.

3.7.3.10 Piping Between Supports and Structural Penetrations

The plant's "ship-like" hull provides a common bedplate support at elevation 40' for the containment, safeguards areas, auxiliary area and all other structures, thereby minimizing differential movements of pipe support and penetration points during seismic disturbances. To assure that piping and support stresses are within allowable limits, a dynamic analysis will be conducted in accordance with the methods described in Paragraphs 3.7.3.4 and 3.7.3.6.

3.7.3.11 Interaction Between Safety-Related and Non-Seismically Designed Piping

The protection of safety-related piping during an earthquake is accomplished by one or more of the following methods:

1. Safety-related lines are physically separated from non-seismically designed piping insofar as possible such that failure of a non-seismic line has no effect on safety-related piping.

2. All seismic boundary valves are designed to meet seismic criteria. A valve always serves as a pressure boundary and constitutes the seismic/non-seismic boundary. If failure in the non-seismic portion of the system could cause loss of function of the safety system, then an appropriate automatic or remote manual operator would be used if the valve is open during normal reactor operation.

3. The pressure boundary valve is protected by restraining and anchoring the non-seismic portion of the system.

3.7.3.12 Quality Assurance of Seismic Restraint Location

The location of seismic supports and restraints for safety-related piping systems and associated components have been determined by dynamic analysis

conducted to satisfy the following two conditions.

1. The locations selected must furnish the required response to control strain within allowable limits.

2. Adequate structural strength for attachment of the components must be available.

Drawings will indicate the location, type, loading and description of seismic supports and restraints. After installation, the location of these supports and restraints are checked against the drawings and instructions issued.

3.7.3.13 Seismic Design of Cranes

Seismic hold-down devices will be provided for the polar crane and fuel handling bridge crane to prevent dislodging the crane trolley or trucks from their rails in the event of a seismic excitation.

The manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper or derailing of the bridge or trolley in the event of a seismic excitation.

3.7.4 CRITERIA FOR SEISMIC AND WAVE MOTION INSTRUMENT PROGRAM

3.7.4.1 Seismic Instrumentation

Seismic instrumentation for the platform consists of both three direction acceleration time history and peak acceleration recording devices that meet the requirements of Safety Guide 12 as applied to a Floating Nuclear Station.

The accelerometers used to monitor and record strong seismic events are described in the following paragraphs.

3.7.4.1.1 Triaxial Accelerometer

An acceleration recording system is used which is self-actuating when a local earthquake exceeds a predetermined level of acceleration. The recording device is located in the Control Room, providing the operator with visual display of the recorder signal. The three element accelerometer transducer is rigidly mounted to the platform in the area of the containment. The platform behaves as a rigid body floating in a liquid pool. Acceleration at all points within the platform would be approximately the same. Lateral acceleration is minimal as compared to vertical acceleration. The seismic trigger is adjustable and is set at a level that avoids actuation due to non-earthquake generated motions. The triggering signal which starts the recorder is annunciated in the Control Room.

The seismic recording equipment continues to operate for approximately one minute after each trigger signal. At the completion of the run, the equipment remains in the standby condition. The level or set point of the seismic trigger will be determined during detail design.

The reactor operator will be able to determine peak acceleration as soon as the earthquake has passed. The equipment required to perform this function is located in the Control Room. The plant operating personnel are provided with a procedure and criteria to review the accelerations recorded in the Control Room. Seismic levels are established to determine whether or not the plant can continue operation or must be shut down. These criteria consider system design and dynamic analysis in establishing the acceptable levels for continued operation.

3.7.4.1.2 Peak Accelerometers

Peak acceleration recorders used to verify a dynamic analysis give the actual peak acceleration after a seismic event. The recorders are located in the general vicinity of the triaxial accelerometer and in specified selected locations on the platform.

3.7.4.2 Wave Motion Instrumentation

Wave motion instrumentation will be installed to monitor platform motion during transit (towing) conditions and wave motion within the breakwater.

This function will be performed by recording the six "degrees of platform motion." These motions are defined as follows:

Translational Motion

Heave	Translational motion along the vertical axis.
Sway	Translational motion along the transverse axis.
Surge	Translational motion along the longitudinal axis.

Torsional Motion

Yaw	Rotational motion about the vertical axis.
Roll	Rotational motion about the longitudinal axis.
Pitch	Rotational motion about the transverse axis.

3.7.4.3 Description

The instrumentation provided to monitor the six degrees of motion is described as follows:

Heave, surge, and sway will be monitored by the triaxial accelerometer already employed for seismic tremors. However, the wave action specified will be activated by a trigger different from the seismic trigger. Recorder speed will be adjusted automatically to provide an adequate record of the wave event.

Yaw will be monitored by one unidirectional accelerometer in addition to the above triaxial accelerometer. The additional unidirectional accelerometer will be mounted close to the edge of the platform.

Roll and pitch instrumentation for the platform consists of a gyroscopic or gravitational oriented sensing element which will monitor platform motion as to roll or pitch in any direction and record this parameter once a predetermined level has been reached. The wave action trigger will be actuated at a specified roll or pitch angle.

The actual element employed will be selected after careful analysis of commercially available equipment and is functionally described in the following paragraph.

A platform roll motion recording system is used which is designed to detect up to a maximum of approximately ± 15.0 degree roll in any direction. It will be capable of indicating a ± 1/10 degree roll as a minimum during normal period of operation. This system will be capable of discerning a 20 second angle of roll in any direction during periods of refueling (core loading). This is in addition to any anticipated static trim that might exist at the time.

No control functions are associated with this equipment. An alarm will be provided for roll or tilt in excess of 1°. The reactor operator will be able to monitor vessel trim as well as peak dynamic roll from a recorder and an indicator in the Control Room. Plant personnel are provided with a procedure for readjustment of platform trim and for refueling during periods of minimal roll.

3.7.5 SEISMIC DESIGN CONTROL MEASURES

Adequate seismic input data (including necessary feedback from structural and systems dynamic analysis) are specified to vendors of purchased safety related components and equipment. Seismic design criteria are prepared by the Applicant's Engineering Department and included in a procurement specification as a design requirement for purchased safety related structures, systems, and components. (See Figure 3.7-5).

Seismic requirements for mechanical components are specified by the Applicant's Engineering Department, and the supplier is required to design for these seismic requirements. The supplier shall perform the dynamic analysis or furnish detailed drawings with sufficient information to permit the Applicant's Engineering Department to perform a dynamic analysis on the equipment. The supplier may be required to revise his design to incorporate any changes necessary during the course of the dynamic analysis.

Seismic requirements for electrical equipment are specified by the Applicant's Engineering Department, and the supplier is required to design for these seismic requirements. The supplier or the Applicant will qualify the equipment in accordance with IEEE 344, IEEE Guide for Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations.

In the event that subsequent system analysis dictates a need for change in seismic criteria, the revised criteria and/or revised detailed design will be transmitted to the supplier as a contract change notice. The supplier will then provide revised detail drawings, or analysis and/or test data developed in accordance with the revised criteria, and, where required, the supplier will modify the appropriate components and equipment.

In all cases where the supplier is responsible for analysis and/or tests which demonstrate compliance to the specified seismic criteria, documented evidence developed by the supplier will be submitted to the Applicant for approval (see Figure 3.7-5) and retained for future commitment to the permanent plant files.

Offshore Power Systems' design control and design review are discussed in detail in chapter 17 of this report.

For information on the Westinghouse supplied components see RESAR 3, section 3.7.5.

A detailed procedure to implement the design control measures delineated in figure 3.7-5 will be in effect by November 1, 1973.

3.8 DESIGN OF CATEGORY I AND CATEGORY II STRUCTURES

3.8.1 STRUCTURES OTHER THAN CONTAINMENT AND PLATFORM

This subsection describes the design criteria for the superstructure which includes all structures above the 40-ft level with the exception of the containment structures described in Section 3.8.2. It also covers concrete construction below the 40-ft level in the safeguards and auxiliary areas. Both structural steel and reinforced concrete construction are used in the plant. Reinforced concrete is generally used only in areas where biological shielding is required. These structures surround the containment shield building and include the control area, safeguards area, and fuel handling and auxiliary area classified as Category I Structures, and the turbine area, power transmission area, and service areas classified as Category II Structures. These areas are shown in Figures 3.2-1, 3.2-2, 3.2-3, and 3.2-4. The superstructure is isolated from the containment shield building.

The interior of the plant is protected from wave action during tow by a watertight wave shield up to elevation 76 ft. which is 36 ft. above the top deck of the platform. The buildings are connected to the platform and their interaction with the platform is considered in the design. All buildings in the superstructure are designed in accordance with applicable codes, loadings and allowable stresses listed in the following paragraphs. The method of analysis and design of all these buildings is also described in the following paragraphs.

3.8.1.1 Category I Structures

3.8.1.1.1 General Description

Category I Structures include the control area, safeguards area, and fuel handling and auxiliary area. The layout of these structures is shown on the general arrangement drawings in Chapter 1.

The control area includes the control and instrumentation rooms. It is a combination of structural steel and reinforced concrete and is approximately 75 ft in length and 47 ft in width.

The safeguards area structures above 40 ft level are principally structural steel. The reinforced concrete located below the 40 ft level provides biological shielding to the safeguards equipment.

The fuel handling and auxiliary areas house the auxiliary systems which require biological shielding. These include the fuel pit, the spent fuel cask handling area, new fuel storage, filters, demineralizers, tanks and pumps. The structure in this area is reinforced concrete and is approximately 133 ft in length and 116 ft in width.

The reinforced concrete structures in the above areas will be anchored at the base and at the sides where necessary by means of dowels similar to Nelson Studs, which will transfer the direct and shear forces across the interface.

3.8.1.1.2 Design Basis

Category I Structures are designed to resist earthquake and tornado forces. All Structures will be vented to relieve the internal pressure during a tornado. Missile protection will be provided where necessary to protect safety related equipment. In the fuel handling and auxiliary area, the fuel pit will be protected from the tornado by a steel structure above the 94' level covering the entire length and width of the spent fuel pit. This steel structure is shown on the general arrangement plans in Section I. The cover and the rolling door at the ends of the structure are not designed for tornados. They act only as barriers to minimize air flow into the enclosure during fuel handling operations.

3.8.1.1.3 Design Loads and Allowable Stresses

The design loads and allowable stresses for the Category I Structures are as follows:

1. Design Loads

a. Dead load (D)

The dead load assumed in design consists of the weights of steelwork, concrete and all materials permanently fastened thereto or supported thereby.

b. Live load (L)

The live loads are taken as not less than those recommended in the Uniform Building Code, 1970. Appropriate amounts of these live loads are taken for each load combination.

c. Thermal load (T)

This is the overall and local temperature changes and gradients.

d. Operating basis environmental conditions (C)
Design basis environmental conditions (C_s)
Environmental conditions during tow (C_t)

These include winds, waves and seismic conditions as described in Sections 3.3, 3.4, and 3.7.

e. Jet forces (J)
Equivalent static loads are developed for these dynamic loads based on limited plasticity with a ductility ratio of 3.

2. Allowable Stresses

The allowable stresses, design load combinations, and load factors are listed in Table 3.8.1-1.

3.8.1.1.4 Structural Analysis and Design

Category I Structures are analyzed and designed for the loads described in paragraph 3.8.1.1.3.

The vertical seismic loads are determined by a dynamic analysis. The SAND Computer program is used for multidegree of freedom models. This program is described in appendix 3A.

3.8.1.2 Category II Structures

3.8.1.2.1 General Description

Category II Structures include the turbine area, power transmission area, and service areas.

The turbine area is structural steel. It houses the turbine-generator and its associated auxiliary equipment and is approximately 302 feet in length and 156 feet in width.

The power transmission area includes main and auxiliary transformers, and the switchyard, and is approximately 156 ft in length and 75 ft in width.

The service areas include administration areas, sleeping quarters, hotel facilities, general storage, laboratories, machine shops and the steel superstructure above the fuel and auxiliary building. The service area structures are structural steel.

3.8.1.2.2 Design Basis

Category II Structures are designed in accordance with conventional building design practice. No dynamic analysis is performed for seismic loads. Seismic design is based on the equivalent static horizontal acceleration listed in Table 3.7.1-2.

3.8.1.2.3 Design Loads and Allowable Stresses

The design loads and allowable stresses for Category II Structures

are considered as follows:

1. Design Loads

   Design loads used are the same as for Category I Structures described in paragraph 3.8.1.1.3 except for the jet forces which are not considered.

2. Allowable Stresses

   The allowable stresses, design load combinations, and load factors are listed in Table 3.8.1-2.

3.8.1.2.4 Structural Analysis and Design

Category II Structures are analyzed and designed for the loads described in paragraph 3.8.1.2.3.

These structures are designed to withstand the normal wind force defined in paragraph 3.8.1.1.3. The steel frame is designed to ensure that it can withstand a wind force greater than that at which the siding fails and can also withstand the tornado wind force after the siding has failed. The

magnitude of the tornado wind force used in design of the main steel members is assumed to be one-third of the full tornado wind force which would occur if the siding did not fail.

TABLE 3.8.1-1

LOAD COMBINATIONS AND LOAD FACTORS FOR CATEGORY I STRUCTURES

LOAD COMBINATIONS	CONCRETE-LOAD FACTORS							REMARKS
	D	L	T	C	C_s	C_t	J	
OB - a	1.4	1.7	1.0					
b	1.0	1.25	1.0	1.25				Allowable Stresses per ACI-318-71
DB - c	1.0	1.0	1.0	1.25			1.0	Code.
d	1.0	1.0	1.0		1.0			
Tow- e	1.0		1.0			1.25		
	STRUCTURAL STEEL-LOAD FACTORS							
OB - a	1.0	1.0	1.0					
b	1.0	1.0	1.0	1.0				Allowable stresses per AISC-69 Code, Part I, without usual increase permitted for seismic or wind.
DB - c	1.0	1.0	1.0	1.0			1.0	150% of allowable stresses per AISC-69 Code, Part I.
d	1.0	1.0	1.0		1.0			
Tow - e	1.0		1.0			1.0		Allowable stresses per AISC-69 Code, Part I, with usual increase permitted for seismic or wind

TABLE 3.8.1-2

LOAD COMBINATIONS AND LOAD FACTORS FOR CATEGORY II STRUCTURES

LOAD COMBINATIONS	CONCRETE LOAD FACTORS						REMARKS
	D	L	T	C	C_s	C_t	
OB - a	1.4	1.7	1.0				Allowable stresses per ACI-318-71 Code.
b	1.0	1.25	1.0	1.25			
DB - c	1.0	1.0	1.0		1.0		
TOW - d	1.0		1.0			1.25	
	STRUCTURAL STEEL LOAD FACTORS						
OB - a	1.0	1.0	1.0				Allowable stresses per AISC 69 Code, Part I, with usual increase permitted for seismic or wind.
b	1.0	1.0	1.0	1.0			
DB - c	1.0	1.0	1.0		1.0		150% of allowable stresses per AISC-69 Code, Part I.
TOW - d	1.0		1.0			1.0	Allowable stresses per AISC 69 Code, Part I, with usual increase permitted for seismic or wind.

3.8.2 CONTAINMENT STRUCTURES

3.8.2.1 General Description

The containment structures are composed of the following four major structures:

1. Shield building

2. Containment shell (or containment)

3. Containment base plate

4. Containment interior structure

The general arrangement of the structures is shown in Figure 1.2-1 through 1.2-13, and the physical description of each structure is given in paragraphs 3.8.2.5.1, 3.8.2.6.1, 3.8.2.8.1, and 3.8.2.9.1.

The containment shell, together with the penetration assemblies and associated piping systems out to and including the outermost containment isolation valves, and the containment base plate constitute the pressure-containing and leakage-limiting metal containment system.

3.8.2.2 Design Bases

The containment structures provide structural and biological barriers for containment of radioactive or hazardous effluents released from nuclear power system components located within the containment boundary. They also serve to protect the nuclear power system components from extreme environmental effects. The main function of the system is to protect the public from an uncontrolled release of radioactivity in the event of an accident.

The design bases for each structure under various loading conditions including testing and surveillance are given in later paragraphs furnishing detailed description of each structure.

3.8.2.3 Codes and Other Documents

Codes, standards, specifications, regulations, guides, and other documents used as bases for the design, construction, testing, and surveillance of the containment structures are listed below. Modifications to the rules of these documents are made where necessary to meet the specific requirements of the structures. These modifications are discussed in later paragraphs dealing with each structure.

1. American Concrete Institute (ACI) Manual of Concrete Practice, Part 1 to Part 3, Current Edition (December 1972).

2. American Institute of Steel Construction (AISC) Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, February, 1969.

3. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, 1971 Edition and Summer 1972 Addenda.

4. American Society for Testing and Materials (ASTM) Standards, Current Editions (December 1972).

5. Uniform Building Code, 1970 Edition, and other applicable regional building codes.

6. 10CFR50 "Licensing of Production and Utilization Facilities", Appendix B - Quality Assurance Criteria for Nuclear Power Plants.

3.8.2.4 Design Loads

The loadings considered in the design of the containment structures are defined as follows:

1. Dead Load (D)

Dead Load consists of the dead weight of the reinforced concrete, the weight of structural steel and miscellaneous embedments and attachments.

2. Live Load (L)

Live load is defined as the appropriate amount of the following loads:

a. Snow and ice load on the projected horizontal surface of the shield building dome roof.

b. Temporary loads during normal plant operating conditions.

c. Temporary loads during construction.

3. Transient Load during Tow (V_t)

For definition of the load, see paragraphs 3.12.2.2.1 and 3.7.1.

4. Operating Basis Pipe and Equipment Load (R)

This load covers the dead weight of various pieces of piping and equipment, including enclosed fluid, and the thermal expansion

load on piping and equipment supports during normal operating conditions. It also covers the loads imposed on the containment shell as a result of the interaction between the shell and the containment penetrations or attachments under normal operating conditions.

5. Operating Basis Temperature Load (T)

This is the overall and local temperature changes and gradients during normal operating conditions.

6. Operating Basis Environmental Load (C)

This load represents any one of the operating basis loads described in paragraph 3.7.1.4. For each structural component, only the applicable part of the load is considered.

7. Design Basis Accident Load (P, J, M_i, R_a, T_a)

This load consists of the following effects resulting from a loss-of-coolant or steam-feedwater line-break accident:

a. Pressure load (P)

There are two distinct types of pressures associated with a design basis accident. One is the short duration differential transient pressures existing in the various containment compartments right after the accident. The other is the steady state pressure after the accident. Both pressure types are described in detail in paragraphs 6.2.1.3.3 and 6.2.1.3.5.

b. Jet Impingement Load (J)

This is the dynamic impact load resulting from pressurized fluid jet being ejected from a postulated pipe rupture.

c. Internal Missile Load (M_i)

This is the postulated missile load associated with a loss-of-coolant accident. Both effects of the missile load, i.e., penetration and dynamic response of the target structure are considered. For internal missile criteria, see RESAR Revision 3 Section 3.5.

d. Pipe and equipment supports reaction load and, possibly, a pipe whip force (R_a)

e. Temperature Load (T_a)

f. Internal Hydrostatic Load (H)

Internal hydrostatic load covers the flooded conditions inside the containment shell following a loss-of-coolant accident.

8. Containment Shell External Pressure Load (P_e)

This is the containment shell internal vacuum pressure, based on the pressure differential across the vacuum relief systems, created by an accidental trip of a portion of the containment spray system during normal plant operations.

9. Design Basis Environmental Load (C_s)

This load represents any one of the Design Basis loads described in paragraph 3.7.1.4. For each structural component, only the applicable part of the load is considered.

10. Design Basis Earthquake Load (E_s)

This load is classified as a design basis environmental load but is singled out here for special consideration in the design of containment structures.

3.8.2.5 Shield Building

3.8.2.5.1 General Description

The shield building is a reinforced concrete structure composed of a right cylinder with a shallow dome roof. It is supported on the 40 foot level of the platform to which it is attached through structural steel connectors encased in the cylindrical concrete wall. These connectors will be designed to transmit axial, transverse, and radial shear loads. The cylinder has an inside diameter of 130 feet and a wall thickness of 3 feet. The dome roof is 2 feet thick and its crown rises 169 feet above the 40 foot level of the platform. The shield building houses the steel containment shell and provides biological shielding as well as protection against hazardous operating basis and design basis environmental loads.

A five-foot annular space is provided between the shield building and the containment shell for construction space and access to the steel shell and penetrations for testing and inspection. The annular space together with space beneath the containment base plate also provides volume for control of pressure under various plant operating conditions including a loss-of-coolant accident and as such serves as a redundant second containment barrier for control of leakage. The shield building seals are designed to accommodate all of these pressures except the tornado induced internal pressures. For details on the annulus ventilation system, see Section 6.4.

3.8.2.5.2 Design Basis

The shield building will be designed to maintain its function for those loads listed in Paragraph 3.8.2.5.3, both individually and in specific combinations, with their dynamic effects properly incorporated. The applicable codes and standards are discussed in Paragraph 3.8.2.5.5, and the allowable stresses and strains in Paragraph 3.8.2.5.7.

3.8.2.5.3 Design Loads

1. Dead Load (D)

2. Live Load (L)

3. Transient Load during Tow (V_t)

4. Operating Basis Pipe and Equipment Load (R)

5. Operating Basis Temperature Load (T)

6. Operating Basis Environmental Load (C)

7. Design Basis Accident Load (P, J, R_a, T_a)

a. Pressure load (P)

Pressure changes in the annular space due to temperature effects following a loss-of-coolant accident.

b. Jet impingement load (J)

Jet force resulting from a steam-feedwater pipe break outside the shield building.

c. Pipe and equipment supports reaction load and, possibly, a pipe whip force (R_a)

d. Temperature load (T_a)

8. Design Basis Environmental Load (C_s)

3.8.2.5.4 Structural Analysis

The static and dynamic analyses of the shield building under the design loads will make use of the following two computer programs:

Computer Program A - "Dynamic Stress Analysis of Axisymmetric Structure Under Arbitrary Loading", by S. Ghosh and E. Wilson,

University of California, Berkely, Earthquake Engineering Research Center Report No. 69-10.

Computer Program B - "Structural Analysis Program (STRAP)" developed at Newport News Shipbuilding and Dry Dock Co., Newport News, Virginia for use in the structural analysis of U. S. Naval vessels.

Outlines of the above computer programs are given in Appendix 3A, and the validity of applying these computer programs to the actual structural model will be verified by using well-established sample problems.

Computer Program A employs the finite element method to the solution of axisymmetric shell or toroidal structures under arbitrary static or dynamic loads. All applied loads under consideration are expressed in terms of Fourier components. In the dynamic analysis, the time history of any applied load can be input, and the response of the structure calculated by one of the following three methods: 1) step-by-step integration of the equation of motion, 2) modal analysis by mode superposition in time, and 3) modal analysis by response spectrum. Program A will be used as the main program for the structural analysis of the shield building.

Computer Program B is a large digital computer program for static and dynamic analysis of three dimensional structures by the finite element

approach. It is capable of solving elastic static structural problems, transient and steady state thermal problems, and linear dynamic structural problems. Program B will be used for the analysis of stress concentrations around large penetrations in the cylindrical wall. In this analysis of stress concentration, a substructure composed of penetrations and neighboring structural elements will be cut out from the cylindrical wall, and the stress resultants in the cylindrical wall obtained from Program A will be used as prescribed forces on the boundary of the substructure. The boundary of the substructure will be chosen remote enough from penetrations so that the stress redistribution in the substructure due to the presence of penetrations will have little effect on the prescribed boundary forces.

The structural analysis of impingement load associated with tornado-driven missiles or pipe whip force on the concrete shell involves the examination of the following two dynamic problems: 1) concrete penetration, and 2) response of the structure to the impulsive load. The first problem will be solved using the modified Petry formula as given in Reference 1. To solve the second problem, the following steps are required to convert the impact energy into equivalent static load:

1. Using the energy method as described in Reference 2 or the momentum transfer method for plastic impact in Reference 3 to obtain a forcing function.

2. Using elastic analysis or idealized elasto-plastic analysis with limited ductility factor to convert the forcing function to an equivalent static load.

3. Performing static stress analysis for the equivalent static load previously determined.

3.8.2.5.5 Design Methods and Applicable Codes

The rules governing the ultimate strength design method of reinforced concrete as given in ACI 318-71 Building Code Requirements for Reinforced Concrete will be the bases for the design of the shield building except as noted below:

1. In the area where a load combination gives rise to localized biaxial membrane tension zones in the shield building, the rules recommended by ACI Committee 334 in its report "Concrete Shell Structures Practice and Commentary" will be followed.

2. For temperature effects, methods based on cracked reinforced concrete sections as described in Reference 4 will be used.

3. Whenever a membrane compression combined with bending governs the design of a reinforced concrete section, an approximate method based

on the moment magnifier principle (similar to the one adopted in ACI Code) will be used to account for the slenderness effect of the shell structure. To be specific, if:

N=membrane compression per unit strip

M=bending moment per unit strip

r= radius of cylindrical or spherical shell

t=thickness of shell

E_c=modulus of elasticity of concrete

N_c=critical membrane compression

$$= .58 \frac{E_c t^2}{r}$$ for meridional compression

$$= .25 \frac{E_c t^3}{r^2}$$ for hoop compression

δ = ratio of the EI value per unit strip as defined by Equation (10-7) or (10-8) of the ACI Code to the theoretical EI value per unit strip of the concrete shell

ϕ = capacity reduction factor as per ACI Code

Then, the magnified design moment M_c will be:

$$M_c = \frac{1}{1 - \frac{N}{\delta \phi N_c}} M$$

In general, the slenderness effect of a concrete shell structure like the shield building is very small.

4. The design of structural steel and embedded plates will be based on AISC-69 Specifications for the Design, Fabrication and Erection of Structural Steel for Buildings with slight modifications to accommodate the load factors presented in Paragraph 3.8.2.5.6.

3.8.2.5.6 Load Combinations and Load Factors

Refer to Table 3.8.2-1. These load combinations and load factors are based on the ACI Code and on current and past nuclear power plant design practice.

3.8.2.5.7 Allowable Stresses and Strains

The allowable stresses and strains as specified in the codes and standards presented in Paragraph 3.8.2.5.5 are used for all load combinations except

the following:

1. When tension of reinforcing steel governs the design of a concrete section, the steel strain is allowed to go beyond yield point by the amount equal to the temperature effect provided the concrete compressive strain does not exceed 0.003.

2. Limited plasticity without loss of function will be allowed in the shield building when it is subjected to tornado-driven missiles or pipe whip or jet forces. The reinforcing steel and concrete strains will be limited by the lowest value of the following ductility factors:

   a. $\mu = \frac{0.05}{p - p'} \leq 10$ for flexural concrete sections, where p, p' are tensile and compressive steel ratios, respectively.

   b. $\mu = 1.3$ for slabs in region requiring no shear reinforcement to resist slab shear stresses.

   c. $\mu = 3.0$ for slabs in region requiring shear reinforcement to resist slab shear stresses.

   These ductility factors are established from References 5 and 6.

3.8.2.5.8 Materials

The following material will be used for the design and construction of the shield building:

1. Concrete-4000 psi compressive strength based on 28 day test.

2. Reinforcing bar-ASTM A 615, Grade 40.

3. Structural steel and embedded plates - ASTM A36.

For material specification and quality control, see Subsection 3.8.3.

3.8.2.6 Containment Shell

3.8.2.6.1 General Description

The containment shell is a freestanding welded steel structure with a 102 feet high vertical cylinder and a hemispherical dome roof. It is welded at the cylinder base to the 40 foot level of the platform. The inside diameter of the cylinder and the dome is 120 feet. The containment shell houses the three major compartments of the ice condenser containment system, namely, the lower compartment, the ice condenser compartment, and

the upper compartment. For more details on the compartments, see Paragraph 6.2.1.3.5. The containment penetrations provide access into the containment shell. These penetrations are described in Paragraph 3.8.2.7.

3.8.2.6.2 Design Basis

The containment shell will be designed to assure that an acceptable upper limit of leakage of radioactive material as specified in Sections 12.4 and 15.4 will not be exceeded under operating and design basis accident conditions. The requirements of Subsection NE, Section III, of the ASME Boiler and Pressure Vessel Code, Summer 1972 Addenda will be used, whenever applicable for the design, construction, examination and testing of the containment shell including all penetration assemblies and associated piping systems. For those configurations and loading cases not explicitly treated by the Code, design criteria as discussed in Paragraphs 3.8.2.6.5 and 3.8.2.6.6 will be used to supplement the code requirements. These design criteria are developed to assure that the containment shell maintains its functional integrity under all specified loading conditions.

Subsection NE, Section III of the ASME Code does not make provisions for stamping pressure vessels of this geometry. Therefore, the containment shell as well as the containment base plate will not receive an ASME Code Stamp. The personnel lock will receive a code stamp.

3.8.2.6.3 Design Loads

1. Dead Loads (D)

2. Live Loads (L)

3. Transient Load during Tow (V_t)

4. Operating Basis Pipe and Equipment Load (R)

5. Operating Basis Temperature Load (T)

6. Operating Basis Environmental Load (C)

7. Design Basis Accident Load (P, R_a, T_a, H)

   a. Pressure load (P)

   b. Pipe and equipment supports reaction load (R_a)

   c. Temperature load (T_a)

d. Internal hydrostatic load (H)

8. Containment Shell External Pressure Load (P_e)

9. Design Basis Environmental Load (C_s)

9a. Design Basis Earthquake Load (E_s)

3.8.2.6.4 Structural Analysis

The structural analysis of the containment shell will rely mainly on the two computer programs as described in Paragraph 3.8.2.5.4. The stress concentration problems around large penetrations in the cylindrical shell will be handled using the method of substructuring similar to the one described also in Paragraph 3.8.2.5.4.

3.8.2.6.5 Design Methods

As mentioned in Paragraph 3.8.2.6.2, the design of the containment shell will basically follow the requirements of Subsection NE, Section III of the ASME Code. For miscellaneous steel attachments not within the scope of the ASME Code, the design will be based on the AISC Specifications.

3.8.2.6.6 Load Combinations and Allowable Stresses

Load combinations will be as shown in Table 3.8.2-2. Following the requirements of Subsection NE, the allowable stress intensities for each of the construction, normal, and design conditions will be taken as the four stress-intensity limits of NE-3221.1, NE-3221.2, NE-3221.3, and NE-3221.4, and the special stress limits of NE-3227. For the emergency condition, the allowable stress intensities will be taken as the greater of 1) 1.20 times the stress limits for the afore-mentioned three loading conditions, or 2) the stress limits specified in ASME Code Figure NE-3221-1, with the design stress-intensity values, S_m, replaced by the yield strength values, S_y, of the ASME Code Table I-2.1. The allowable stress intensities for the condition during tow will be limited to 1.20 times the allowable stress intensities for normal conditions.

3.8.2.6.7 Materials

The containment shell is comprised of the following materials:

1. Cylindrical and hemispherical shell-SA 516 Grade 60

2. Stiffening rings and penetration assemblies-SA 516 Grade 60

3. Miscellaneous structural steel members-ASTM A36.

For material specification and quality control, see Subsection 3.8.3.

3.8.2.6.8 Coatings

The interior surface of the steel shell and metal surfaces of attachments will be given a coating system capable of withstanding the containment environment under normal operating conditions as well as design basis accident conditions. The exterior surface of the steel shell and metal surfaces of attachments will be coated with a system capable of providing protection against atmospheric exposure.

3.8.2.7 Containment Penetrations

3.8.2.7.1 Equipment Hatch

The equipment hatch will be composed of a cylindrical sleeve in the containment shell and a dished head 34.0 feet in diameter with mating bolted flanges. The flanged joint will have double O-ring seals with an annular space for pressurization for leak testing of the seals. The leak test pressure is vented to the outside of the containment and then plugged for

normal operation. The equipment hatch will be designed, fabricated and tested in accordance with Section III, Subsection NE, of the ASME Boiler and Pressure Vessel Code, Summer 1972 addenda. Details of the equipment hatch are shown in Figure 3.8-1 sheet 1. Figure 3.8-1 sheet 2 illustrates the hatch closure head handling and storage. This hatch will also function as the fuel transfer penetration with the crane passing through the hatch with the assembly fully shielded.

3.8.2.7.2 Personnel Access Hatches

Two personnel access hatches will be provided. Each hatch will have double doors with an interlocking system to prevent both doors from being opened simultaneously. Remote indication will be provided to indicate the position of the far door. Quick-acting type equalizing valves will be used to equalize the pressure inside the hatch when entering or leaving the containment. Double seals will be provided on the outer doors. The annular space between the seals for the outer door will be plugged for normal operation. The personnel access hatch will be a completely pre-fabricated and assembled welded steel subassembly designed, fabricated, tested and stamped in accordance with Section III, Subsection NE, of the ASME Boiler and Pressure Vessel Code. Details of the personnel access hatches are shown on Figure 3.8-2.

3.8.2.7.3 Mechanical Penetrations

Mechanical penetrations are classified as either a hot penetration type (temperature of process fluid is greater that 150°F) or a cold penetration type (temperature of process fluid is equal to or less that 150°F).

The hot type penetration will:

1. Accommodate thermal movement of hot process piping.

2. Insulate the process piping from the concrete to prevent heating the concrete above 150°F.

3. Endure design stresses without causing a breach of containment integrity.

4. Permit process fluid to flow only into the containment vessel and not into the containment vessel annulus by means of a guard pipe, if a process piping break occurs between the containment vessel and the shield building.

5. For main steam and feedwater only, the penetration guard pipe assembly will extend further into the containment and penetrate

the crane wall. This assembly will be supported to avoid transmitting loads to the containment vessel.

The cold type penetration will:

1. Endure design stresses without causing a breach of containment integrity.

2. Permit process fluid to flow only into the containment vessel and not into the containment vessel annulus by means of a guard pipe, if a process piping break occurs between the containment vessel and the shield building.

Both types of penetration will be designed to Section III, Subsection NE, of the ASME Boiler and Pressure Vessel Code. Details of these penetrations are shown in Figures 3.8-3, 3.8-4, and 3.8-5.

3.8.2.7.4 Electrical Penetrations

Medium voltage electrical penetrations for reactor coolant pump power as shown on Figure 3.8-6 will use sealed bushings for conductor seals.

The assemblies will incorporate two of these seals along the axis of each conductor to give a double barrier.

Low voltage power, control and instrumentation cables will enter the containment vessel through penetration assemblies which will be designed to provide two leak tight barriers along the axis of each conductor. Details of these assemblies are shown on Figure 3.8-6.

All electrical penetrations will be designed to maintain containment integrity for design basis accident conditions including the effects of high pressure, temperature and radiation. Double barriers will permit testing of each penetration assembly as required to verify that containment integrity is maintained. Design and test will be in accordance with IEEE Standard No. 317.

3.8.2.8 Containment Base Plate

3.8.2.8.1 General Description

The containment base plate consists of the upper deck of the platform at the 40-foot level inside the shield building, the bulkhead around the in-core instrumentation sump, and the innerbottom of the sump at the 7-foot, 6-inch level. It serves as the lower boundary of the metal con-

tainment system as defined in paragraph 3.8.2.1. The base plate, together with the girders and stiffeners welded to it, transmits all of its transverse loadings to the platform bulkheads supporting the base plate. The description of the fill concrete attached to the base plate is given in paragraph 3.8.2.9.1.

3.8.2.8.2 Design Basis

The base plate will be designed to assure that an acceptable upper limit of leakage of radioactive material as specified in Sections 12.4 and 15.4 will not be exceeded under operating and design basis accident conditions. Design criteria are developed herein which are consistent with the intent of the requirements of the ASME Code, Section III, Subsection NE.

These criteria are presented in paragraphs 3.8.2.8.4 through 3.8.2.8.7.

3.8.2.8.3 Design Loads

1. Dead Load (D)

2. Live Load (L)

3. Transient Load during Tow (V_t)

4. Operating Basis Pipe and Equipment Load (R)

5. Operating Basis Temperature Load (T)

6. Operating Basis Environmental Load (C)

7. Design Basis Accident Load (P, R_a, J, M_i, T_a, H)

   a. Pressure load (P)

   b. Pipe and equipment supports reaction load and, possibly, a pipe whip force (R_a)

   c. Jet impingement load (J)

   d. Internal missile load (M_i)

   e. Internal hydrostatic load (H)

8. Design Basis Environmental Load (C_s)

8a. Design Basis Earthquake Load (E_s)

3.8.2.8.4 Structural Analysis

Computer Program B as described in paragraph 3.8.2.5.4 will be used as the main program for the analysis of the stress distribution in the base plate as well as the interaction between the base plate and the superstructures attached to it. Method of substructuring will be employed whenever applicable. The presence of in-plane stresses in the upper deck due to the overall bending of the platform will have certain "magnifying effects" on the transverse deflections of steel panels and, consequently, the stress resultants from the analyses bases on Program B. Here, a panel is defined as an orthogonally reinforced flat or curved plate bounded by structural supports (e.g. platform bulkheads) which impose significant restraints in the transverse direction of the plate, and girders are considered as heavy stiffeners in a panel. Based on the platform framing as shown in Figures 3.12-1 through 3.12-6, the plate-girder combination predominately governs the stiffness and, consequently, the deflection of every panel in the base plate. This kind of framing practically breaks up the behavior of a panel into one-way actions with the plate-girder combination spanning across structural supports in one direction and the plate-angle combination spanning across girders in the other direction. Following this approach, the magnifying effects of the panels can be approximated by using the simplified expression for an amplification factor as given in Reference 7 (similar to the

expression used in the moment magnifier principle discussed in paragraph 3.8.2.5.5). Limited plasticity without loss of function will be allowed in the base plate when it is subjected to jet impingement loads, internal missile loads, or pipe whip forces. Methods of analysis for the impingement loads will be similar to that presented in paragraph 3.8.2.5.4, and the allowable stresses and strains are given in paragraph 3.8.2.8.7.

3.8.2.8.5 Design Method

Since all structural members of the base plate are composed of steel plate elements, the design of each member will be based on the maximum shear stress theory (Tresca Yield criterion) for plates. The stress intensity approach as given in Subarticle NE-3200, Section III of the ASME Code, will be used herein with the following modifications:

1. Definition of stress intensity will be the same as defined in NE-3213.1 except, for plate elements, the third principal stress is considered to be zero.

2. Classification of stress intensities is modified as follows to relate more directly to the stress distribution in the base plate:

a. General primary membrane stress (P_m): Axial stress in the plate-stiffener combination of a panel due to the overall bending of the platform and other in-plane loads distributed rather uniformly through the whole panel, where the definition of a panel is found in Paragraph 3.8.2.8.4.

b. Primary bending stress (P_b): Bending stress in the plate-stiffener combination in the central portion of a panel due to a fairly uniformly distributed transverse loading on the panel.

c. Local primary membrane and bending stress (P_L): Axial plus bending stress in the plate-stiffener combination in the central portion of a panel due to the specified pipe and equipment load.

d. Secondary stress (Q): 1) General thermal stress, and 2) bending stress in the central portion of a local plate element bounded by stiffeners due to a uniform load.

e. Peak stress (F): Same as defined in NE-3213.11.

3.8.2.8.6 Load Combinations (Table 3.8.2-3)

3.8.2.8.7 Allowable Stresses and Strains

Allowable stress intensities for the five loading conditions listed in Table 3.8.2-3 can be summarized as follows:

Construction, Normal, and Design Conditions

1. $P_m \leq S_m$ (1)

2. $P_m + P_b \leq 1.15$ (2) S_m

3. $P_m + P_L \leq 1.72\ S_m$

4. $P_m + P_L + P_b \leq 1.72\ S_m$

Emergency Condition

1. $P_m \leq 1.2\ S_m$ or S_y (3) whichever is greater

2. $P_m + P_b \leq 1.38\ S_m$ or $1.15\ S_y$ whichever is greater

3. $P_m + P_L \leq 2.07\ S_m$ or $1.72\ S_y$ whichever is greater

4. $P_m + P_L + P_b \leq 2.07\ S_m$ or $1.72\ S_y$ whichever is greater

Condition During Tow

1. $P_m \leq 1.2\ S_m$

2. $P_m + P_b \leq 1.2\ S_m$

3. $P_m + P_L \leq 1.8\ S_m$

4. $P_m + P_L + P_b \leq 1.8\ S_m$

Note (1): S_m denotes the design stress intensity values specified in the ASME Code Table I-10.1.

Note (2): Since bending stress in the plate-stiffener combination of a panel under design basis accident conditions predominates in the design of the base plate, a conservative number of 1.15 is used to account for the plastic shape factor of a wide flange section.

Note (3): S_y denotes the yield strength values specified in the ASME Code Table I-2.1.

When jet impingement loads, internal missile loads, or pipe whip forces are involved, plastic hinges in the base plate will be allowed with yield stresses limited to the values given in ASME Code Table I-2.1, and the

ductility factor limited to μ = 3 which is established from references 5 and 6.

The allowable stresses for bulkheads supporting the base plate are presented in paragraph 3.12.2.3.

3.8.2.8.8 Materials

All plating forming a part of the containment pressure boundary is SA 516 Grade 60 steel. Other steel members are ABS Grade A, B, or C steel.

3.8.2.8.9 Coatings

See Subsection 3.12.6.

3.8.2.9 Containment Interior Structure

3.8.2.9.1 General Description

The interior structure consists mainly of concrete structures which include a circular crane wall, an operating deck, a refueling canal with stainless steel liner, a primary biological shield enclosing the reactor vessel, and a fill concrete base. All concrete structures are attached

to the upper deck or, through the in-core instrumentation sump, to the support structures of the platform. Concrete structures requiring integral connections to the steel structures of the platform are provided with structural steel connectors embedded in the concrete which transmit the direct and shear forces across the interface. The primary functions of the interior structure are as follows:

1. To provide pressure boundaries for the lower, ice condenser, and upper compartments.

2. To provide support for major equipment, ice condenser, and refueling operation.

3. To act as a biological shield, and also as a missile shield to withstand internally generated missiles, jet forces, and pipe whip forces.

A divider barrier is defined which consists of those structures which separate the lower and upper containment compartments. It is an essential pressure boundary in the ice condenser containment system which requires that the steam and air flowing into the lower compartment in the event of a loss-of-coolant accident be routed to the upper compartment via the ice bed. The divider barrier includes the walls of the ice condenser compartment,

the operating deck, the steam generator enclosures, the removable missile shields of the reactor cavity compartment, and portions of the walls of the refueling canal.

3.8.2.9.2 Design Basis

The design of concrete structures will be based on the ultimate strength design method as given in ACI-318-71 code, and the steel structures based on the working stress design method found in AISC-69 specification, with some modifications. Additional conservatism will be provided for in the design of the divider barrier. Details of analysis and design are presented in the following paragraphs.

3.8.2.9.3 Design Loads

Same as listed in paragraph 3.8.2.8.3.

3.8.2.9.4 Structural Analysis

Because of the complexity in configurations as well as loadings, the analysis of the internal structure is divided into two computational procedures. First, a gross model representation of the entire structural system is analyzed for various loadings using either Computer Program A or B as

described in paragraphs 3.8.2.5.4 to obtain a general view of the stress distributions. Then, the gross model representation is divided into several more detailed substructures which include details such as large openings and interfaces between structural components. Program B is used to analyze the substructures with the stress resultants determined from the gross model representation prescribed as force boundary conditions. Conventional representation of structural elements as beams, columns, one and two way slabs or frames will also be used whenever possible to supplement the computer analyses.

For structures subject to jet impingement load, internal missile load, or pipe whip force, limited plasticity without loss of function will be allowed. Methods of analysis for the impingement load will be the same as described in paragraph 3.8.2.5.4 and the allowable stresses and strains are given in paragraph 3.8.2.9.7.

3.8.2.9.5 Design Methods

Criteria presented in paragraph 3.8.2.5.5 are directly applicable to the design of the interior structure.

3.8.2.9.6 Load Combinations and Load Factors (Table 3.8.2-4)

For the divider barrier, an additional overall load factor of 1.2 is

applied to all load combinations.

3.8.2.9.7 Allowable Stresses and Strains

Same as those set forth in paragraph 3.8.2.5.7 with the addition that for steel structures subject to impingement loads, the allowable stresses and strains specified in paragraph 3.8.2.8.7 apply.

3.8.2.9.8 Materials

Same as listed in paragraph 3.8.2.5.8 with the addition that SA516 Grade 60 will be used for vital steel structures subject to design basis accident pressure load and other impactive loads.

3.8.2.10 Interfaces between the Containment Superstructures and the Platform

Methods of substructuring similar in essence to those described in paragraphs 3.8.2.5.4, 3.8.2.8.4, and 3.8.2.9.4 will be used extensively to find the structural interactions between the containment superstructures and the platform. In particular, the stress distributions in the following areas will be obtained through these methods:

1. Base Plate

2. Base plate support structures of the platform.

3. Steel connectors connecting concrete superstructures to the base plate.

4. Lower parts of the superstructures which are either attached to the base plate or affected by the attachments.

Computer program B as described in paragraph 3.8.2.5.4 will be employed for the analyses of the substructures. Also, the displacements or the stress resultants or both, from the overall structural analyses of the superstructures and the platform will be used as the prescribed boundary conditions on the substructures.

3.8.2.11 Structural Preoperational Testing

The containment shell and base plate forming the containment pressure boundary, and appurtenances will be tested in accordance with Subarticle NE-6320 of Article NE-6000 of Subsection NE, Section III of the ASME Code.

TABLE 3.8.2-1

CONTAINMENT SHIELD BUILDING LOAD COMBINATIONS AND LOAD FACTORS

Load Combinations	D	L	R	R_a	T	T_a	C	C_s	V_t	P	J
Service Category											
a	1.4	1.7	1.4		1.0						
b	1.0	1.25	1.0		1.0		1.25				
Factored Category											
c	1.0	1.0							1.25		
d	1.0	1.0		1.0		1.0				1.25	1.0
e	1.0	1.0		1.0		1.0	1.25			1.25	1.0
f	1.0	1.0	1.0		1.0			1.0			

TABLE 3.8.2-2

CONTAINMENT SHELL LOAD COMBINATIONS

Load Combinations	D	L	R	R_a	T	T_a	V_t	E_s	C	C_s	P	H	P_e
Construction Condition at Ambient Temperature													
a	1.0	1.0											
Normal Condition													
b	1.0	1.0	1.0		1.0								
c	1.0	1.0	1.0		1.0				1.0				
Design Condition													
d	1.0	1.0		1.0		1.0					1.0	1.0	
e	1.0	1.0		1.0		1.0			1.0		1.0	1.0	
f	1.0	1.0	1.0		1.0								1.0
Emergency Condition													
g	1.0	1.0	1.0		1.0					1.0			
h	1.0	1.0		1.0		1.0		1.0			1.0	1.0	
Condition during Tow													
i	1.0	1.0					1.0						

TABLE 3.8.2-3

CONTAINMENT BASE PLATE LOAD COMBINATIONS

Load Combinations	D	L	R	R_a	T	T_a	V_t	E_s	C	C_s	P	J	M_j	H
Construction Condition at Ambient Temperature														
a	1.0	1.0												
Normal Condition														
b	1.0	1.0	1.0		1.0									
c	1.0	1.0	1.0		1.0				1.0					
Design Condition														
d	1.0	1.0		1.0		1.0					1.0	1.0	1.0	1.0
e	1.0	1.0		1.0		1.0			1.0		1.0	1.0	1.0	1.0
Emergency Condition														
f	1.0	1.0	1.0		1.0					1.0				
g	1.0	1.0		1.0		1.0		1.0			1.0	1.0	1.0	1.0
Condition during Tow														
h	1.0	1.0					1.0							

TABLE 3.8.2-4

CONTAINMENT INTERIOR STRUCTURE LOAD COMBINATIONS AND LOAD FACTORS

Load Combinations	D	L	R	R_a	T	T_a	V_t	E_s	C	C_s	P	J	M_j	H
Service Category														
a	1.4	1.7	1.4		1.0									
b	1.0	1.25	1.0		1.0				1.25					
Factored Category														
c	1.0	1.0					1.25							
d	1.0	1.0		1.0		1.0					1.25	1.0	1.0	1.25
e	1.0	1.0		1.0		1.0			1.25		1.25	1.0	1.0	1.25
f	1.0	1.0	1.0		1.0					1.0				
g	1.0	1.0		1.0		1.0		1.0			1.0	1.0	1.0	1.0

3.8.3 CONCRETE MATERIAL SPECIFICATIONS AND QUALITY CONTROL

Codes and Standards

The following codes, standards, and safety guides are used to establish the specifications and procedures governing the concrete construction. Unless otherwise indicated by the year following the code or standard such codes or standards are those in force effective at the date of this application.

ACI 214 - Recommended Practice for Evaluation of Compression Test Results of Field Concrete.

ACI 301 - Specification for Structural Concrete for Buildings.

ACI 306 - Recommended Practice for Cold Weather Concreting.

ACI 315 - Manual of Standard Practice for Detailing Reinforced Concrete Structures.

ACI 318 - Building Code Requirements for Reinforced Concrete.

ACI 347 - Recommended Practice for Concrete Formwork.

ACI 605 - Recommended Practice for Hot Weather Concreting.

ACI 613 - Recommended Practice for Selecting Proportions for Concrete.

ACI 614 - Recommended Practice for Measuring, Mixing, and Placing Concrete.

ASTM-C94 -Standard Specification for Ready-Mixed Concrete.

AEC Safety Guide 10 - Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments.

AEC Safety Guide 15 - Testing of Reinforcing Bars for Concrete Structures.

Materials for Concrete

Cement

Cement shall conform to the requirements of ASTM C-150 "Standard Specification for Portland Cement", Type I or II. Where high early strength is required Type III cement will be used.

Aggregates

Aggregate shall conform to the requirements of ASTM C-33 "Standard Specifications for Concrete Aggregates".

Aggregates shall consist of inert materials that are clean, hard and durable, free from organic matter and uncoated with dirt or clay. Fine aggregate

shall be natural or crushed sand and the coarse aggregate crushed stone or gravel. Aggregate shall not be potentially reactive to the type of cement used.

Mixing Water

Water for mixing concrete shall be clean and free from injurious amounts of oils, acids, alkalis, salts, and organic materials, or other substances which may be deleterious to concrete or steel.

Wherever possible, potable water defined by U. S. Public Health Standards shall be used.

Non-potable water may be used only if it produces mortar test cubes having 7 day and 28 day strengths equal to at least 90% of the strengths of similar specimens made with potable water. The strength comparison shall be made in mortars prepared and tested in accordance with ASTM C-109, "Method of Test for Compressive Strength of Hydraulic Cement Mortars (Using 2 in. Cube Specimens)".

Admixtures

When Admixtures are to be used they shall conform to the following specifications:

Air-Entraining Admixtures. Air-entraining admixtures shall conform to

the requirements of ASTM C-260, "Air-Entraining Admixtures for Concrete".

Fly Ash and Pozzolanic Materials. Fly ash and pozzolanic materials shall conform to the requirements of ASTM C-618, "Fly Ash and Raw or Calcined Natural Pozzolans for Use in Portland Cement Concrete".

Chemical Admixtures. Chemical admixtures shall conform to the requirements of ASTM C-494, "Chemical Admixtures for Concrete". Calcium chloride will not be used as a concrete admixture.

Reinforcing Steel

Mild steel reinforcing bar shall conform to ASTM A-615, "Standard Specifications for Deformed Billet-Steel Bars for Concrete Reinforcement". Either Grade 40 or Grade 60 will be used. A certification of physical properties and chemical content of each heat of reinforcing steel will be required from the steel supplier.

Quality Control

Cement: In addition to the tests required by the cement manufacturer, the following tests will be performed in order to ascertain conformance with ASTM Specification C-150:

ASTM C114 - Standard Methods for Chemical Analysis of Hydraulic Cement

ASTM C204 - Standard Method of Test for Fineness of Portland Cement by Air Permeability Apparatus

ASTM C151 - Standard Method of Test for Autoclave Expansion of Portland Cement

ASTM C191 - Standard Method of Test for Time of Setting of Hydraulic Cement by Vicat Needle

ASTM C109 - Standard Method of Test for Compressive Strength of Hydraulic Cement Mortars

Certified mill reports for each bin of cement as manufactured will be required. In addition, the tests described in ASTM C191 and ASTM C109 will be performed periodically during construction to check storage environment effects on cement characteristics. These tests will supplement visual inspection of material storage procedures.

Concrete

Concrete mixes will be designed in accordance with ACI-301-66 Section 308, Method 2. Proportions of ingredients will be determined and tests conducted in accordance with ACI 613-54. After trial mixes have been approved and construction has commenced, the methods used in sampling, making, curing, and testing concrete specimens will be in accordance with the following ASTM Standards:

ASTM C 39 Standard Method of Test for Compressive Strength of Molded Concrete Cylinders

ASTM C143 Standard Method of Test for Slump of Portland Cement Concrete

ASTM C172 Standard Method of Sampling Fresh Concrete

ASTM C192 Standard Method of Making and Curing Concrete Test Specimens in the Laboratory

ASTM C231 Standard Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method

One set of test cylinders will be made and cured in accordance with ASTM Designation C-31 for each 150 cubic yards of concrete or portion thereof placed per day per class. Each set of test cylinders will consist of six cylinders. Each cylinder will be capped and tested in accordance with procedures outlined in ASTM Designation C-39. Two cylinders will be tested at 7 days and two cylinders at 28 days. If results of 7 or 28 day tests indicate any cause for concern that concrete does not meet specification requirements, the remaining cylinders shall be saved and tested at a later date. Compression tests of concrete will be evaluated in accordance with ACI 301-66 Chapter 17. Whenever it appears that tests of the laboratory-cured cylinders fail to meet strength requirements set forth in this specification additional tests of the concrete in question shall be made. These tests

shall be made in accordance with ASTM C-42. If these tests fail to prove that the questionable concrete is of specified strength, the concrete will be replaced.

REFERENCES

1. "Design of Protective Structures", by A. Amirikian, NavDocks P-51.

2. "Effects of Impact and Explosion", Vol. 1, AD 221586.

3. "Structural Design for Dynamic Loads", by C. H. Norris, et. al., McGraw-Hill Book Co., 1959.

4. "Specifications for the Design and Construction of Reinforced Concrete Chimneys", ACI 307-69.

5. "Design of Structures to Resist Nuclear Weapon Effects", ASCE Manuals of Engineering Practice No. 42, 1964 Edition, P. 111 and P. 124.

6. "Introduction to Structural Dynamics", by J. M. Biggs, McGraw-Hill Book Co., 1964, p. 224.

7. "Theory of Elastic Stability", by S. P. Timoshenko and J. M. Gere, 2nd Edition, McGraw-Hill Book Co., 1961, P. 15.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

The scope of the different dynamic analysis techniques and methods used to evaluate mechanical systems and components of the Floating Nuclear Plant is very extensive; however, the more important and pertinent methods are presented as an overview of the type of methods utilized. Information presented in this section supplements the basic analytical methods described in Subsections 3.7.2 and 3.7.3 of this report

Seismic qualification of safety-related mechanical equipment will be accomplished by the dynamic analysis methods described in Subsections 3.7.2 and 3.7.3, and by the methods described in Subdivision 3.9.2.9.

Safety-related systems will be operated through their normal and emergency modes, where practical, to determine whether excessive movement or vibration occurs.

3.9.1 ASME CODE CLASS 1 COMPONENTS

A description of the methods of analyzing and testing ASME Code Class 1 Components is presented in Section 3.9 of RESAR 3.

3.9.2 ASME CODE CLASS 2 AND 3 COMPONENTS

3.9.2.1 System and Component Functional Design Basis

Design pressure, temperature, and other functional design conditions for mechanical systems and equipment are described on a system basis in Chapters 5, 6, and 9.

3.9.2.2 Design Loading

The design loading combinations in the component or system design include but are not limited to the following:

1. Internal and external pressure.

2. Weight of the component and normal contents under operating conditions, including additional pressure due to static and dynamic head of liquids.

3. Superimposed loads such as other components, operating equipment, insulation, corrosion-resistant or erosion-resistant linings, and piping.

4. Inertia and gyroscopic effects due to wind loads, wave action, vibrations, and seismic loads.

5. Reactions of supporting lugs, rings, saddles, or other types of supports.

6. Temperature effects.

3.9.2.3 Stress Criteria

The safety related mechanical components are classified as shown in Table 3.2-1, and will be designed in accordance with the codes listed. The loading conditions and stress criteria shown in Table 3.9.2-1 will be used in the design of systems and components.

3.9.2.4 Stress Limits

All safety related mechanical components will be designed in accordance with the stress criteria indicated in paragraph 3.9.2.3.

3.9.2.5 ASME and ANSI Code Case Interpretations

Applicant will comply with the ASME Boiler and Pressure Vessel Code and ANSI Standard B31.1 as appropriate. In the event that code case interpretations are used during the course of plant design, the interpretations will be identified and justification will be presented for their use.

3.9.2.6 Active Pumps and Valves

Active components are those whose operability is relied upon to perform a safety function such as safe shutdown of the reactor or mitigation of the consequences of a postulated pipe break in the reactor coolant pressure boundary.

Active pumps which are not a part of the reactor coolant pressure boundary are as follows: Essential Service Water Pumps, Essential Raw Water Pumps, High Head Safety Injection Pumps, Containment Spray Pumps, Auxiliary Feedwater Pumps, Diesel Fuel Oil Transfer Pumps, Safety Injection Pumps, and Residual Heat Removal Pumps.

Active containment isolation valves required to perform a safety function are identified by either a "T" or "P" symbol on the system flow diagrams. Active valves in other systems are identified on a system basis.

Equipment specifications define operating modes and require analytical calculations to be performed by the Supplier or applicant to assure that active components will function as designed. The methods of evaluation for system components are as described in Subsection 3.7.3 and Paragraph 3.9.2.9.

3.9.2.7 Criteria for Protection of Critical Systems and Containment Against the Effects of Pipe Whip

Protection criteria against the effects of pipe whip in the Reactor Coolant System boundary are presented in Section 3.6. The design criteria discussed in this section to assure the protection of safety-related systems pertain to postulated pipe breaks other than those discussed in Section 3.6. These criteria are as follows:

1. The system postulated to rupture is the Main Steam System.

2. Breaks are postulated to occur only as a result of fatigue; consequently, breaks are considered to occur only at point of high stress cycling. Typically, these points are located in the higher stressed areas of elbows, tees, nozzles, and at points of abrupt change in section properties. Circumferential breaks are considered at these locations only. Figure 3.9-1 shows the postulated breaks in the main steam piping between the steam generator and the first restraint beyond the second check valve outside the containment.

For the systems postulated to rupture, the static equivalent blowdown force is assumed to be:

$F = 1.25\ P_{max}\ A$ - for Main Steam System

where P_{max} = maximum internal pressure at time of postulated rupture

$A = \frac{\pi}{4} (I.D.)^2_{pipe}$

3. Pipe hangers and restraints are designed to prevent pipe whip. When calculating the forces on the restraints, credit may be taken for the ability of the ruptured pipe to resist pipe whip using the lower bound plastic bending moment for a fully plastic section.

4. Containment integrity is assured.

5. Safety-related systems minimum performance requirements are assured. This may be accomplished by physical separation of redundant safety systems, installation of barriers and pipe restraints, and isolation of vital equipment.

3.9.2.8 Safety and Relief Valves

Design analysis and installation criteria for safety and relief valves located outside the containment are as follows:

1. Piping and its support-restraint system are designed to accommodate and/or restrain the piping for both dynamic and static loadings as applicable such that stresses produced are within the limits established by the ASME III code for the following:

   a. Dead weight effects

   b. Thermal loads and movements

c. Seismic loads and deflections

d. Safety valve dynamic thrust and moment

e. Maximum absolute differential movement between structures.

f. Motion effects due to wave and wind conditions.

Applicable loads are combined in accordance with the appropriate sections of ASME III.

2. Nozzles are designed such that code allowable stresses are maintained for:

   a. Internal pressure

   b. Safety valve thrust

   c. Safety valve moment

Pressure relief for the main steam lines outside the Containment is accomplished through the use of sufficient safety valves to meet code requirements. The safety valves are set for progressive relief in intermediate steps of pressure within the allowed range of pressure

settings to prevent more than one valve actuating simultaneously. A stress analysis is performed to determine the effects on the main steam lines assuming that all steam generator safety and relief valves discharge concurrently. The piping system is designed to maintain stresses within ASME Section III Code allowable limits.

3.9.2.9 Safety-Related Pump and Valve Design Analysis

Analytical methods and criteria used to evaluate stresses and deformations in safety-related pumps and valves including safety and relief valves are established by the Applicant's Engineering Department, and, where applicable, are in accordance with Section III of the ASME B and PV Code. In cases not explicitly addressed by ASME III, the design conditions are evaluated and compared to similar design conditions addressed by ASME III. Acceptance criteria are as defined by ASME III for similar type loadings.

Specific criteria and methods applied to pumps and valves are as follows:

1. Pumps

   The pump specifications contain the criteria for determining nozzle loads and seismic and motion accelerations under which the pump must be designed to operate.

Piping flexibility analyses are performed to obtain the loads on the pump nozzles for each condition considered. The calculated loads are compared with the design loads to assure a compatible design.

Pumps have been found to be rigid bodies rigidly mounted when compared to seismic frequencies. Consequently, the only significant seismic design requirement is that the pump be required to operate under the influence of the lateral and vertical accelerations of the structure to which it is supported. The manufacturer is required to verify the functional capability of a pump and to provide for adequate strength and stiffness in its design. The structural integrity of the pump is verified by the Engineering Department of the Applicant. (See Figure 3.7-5).

2. Valves

The valve design specifications contain seismic design criteria that are considered to be more than adequate to assure capability for the calculated loading. The design specification for valves requires that the valves and operators be designed for specified umbrella acceleration values. Valves have been found to be rigid bodies, and their resonant frequencies are in excess of those significant exciting frequencies found in nuclear plants.

Where it is possible to do so, the specific acceleration level is determined by the intended physical location and the attendant building acceleration level in which the valve is to be used. Valves bought for general use are required to be designed to the umbrella acceleration level because their use location is undefined at the time of purchase. Piping systems are modeled for analysis so as to obtain from the results the modal accelerations of the valves and the operators. These accelerations are combined and compared with the design accelerations to determine the need for additional seismic restraints.

The manufacturer is required to verify the functional capability of a valve and to provide for adequate strength and stiffness in its design. The structural integrity of the valve is verified by the Engineering Department of the Applicant. (See Figure 3.7-5).

3.9.2.10 Component Design Conditions and Loadings

Component functional design conditions and mechanical loadings are described in the applicable system descriptions in Chapters 6 and 9. The design conditions for all components that are to be constructed in accordance with Section III of the ASME B and PV Code, Subsections NC, ND, and

NE are specified in the equipment specification of individual components as required by the Code.

3.9.2.11 ASME III Code Compliance Calculations and Tests

ASME III Components' analytical calculations and/or experimental testing done to demonstrate compliance with the code will be completed by the suppliers or the Applicant after contracts are awarded for the individual components. Reports of these analyses and/or tests will be prepared; however, the physical nature (various size drawings, calculations, and computer printouts) of these reports will not lend themselves to being part of the report.

3.9.2.12 Tanks

Tanks are listed in Table 3.2-1 and will be designed in accordance with the following:

1. Structural tanks (those tanks which are built into, and which form, part of the structure of the platform) are designed in accordance with ABS/USCG rules using the same criteria used for the platform as described in Section 3.12.

2. Safety-related, pressurized tanks in Nuclear Safety Class 1, 2 or 3 applications are designed in accordance with the requirements of the ASME Section III Code for Class 1, 2 or 3 equipment, respectively.

3. Non-nuclear safety tanks which are designed for low pressure, i.e., atmospheric drain tanks and tanks operating at 0-15 p.s.i.g., are designed and constructed in accordance with ASME Section III Class 3 requirements, except that no code stamp is required.

3.9.3 COMPONENTS NOT COVERED BY ASME CODE

Safety-related mechanical components furnished by Westinghouse, but which are not covered by the ASME B and PV Code, are described in Subsection 3.9.3 of RESAR.

TABLE 3.9.2-1

STRESS CRITERIA FOR ASME CODE CLASS 1, 2 and 3 COMPONENTS OF FLUID SYSTEMS

Conditions	LOADS			Criteria
	Piping	Mechanical Penetrations Shell Type Equipment & Components & their Supports	Piping Supports, Restraints & Anchors	
Normal	Thermal Expansion Pressure Weight Operating Basis Motion (OBM)	Thermal Expansion Pressure Weight Piping Loads Support Reactions OBM	Thermal Expansion Pressure Weight OBM	ASME B and PV Code Section III
Upset	Thermal Expansion Pressure Weight Operating Basis Earthquake (OBE) or (OBM)	Thermal Expansion Pressure Weight Piping Loads Support Reactions OBE or OBM	Thermal Expansion Pressure Weight OBE or OBM	ASME B and PV Code Section III
Emergency	Pressure Weight Design Basis Earthquake (DBE) or Design Basis Motion (DBM)	Thermal Expansion Pressure Weight Piping Loads Support Reactions DBE or DBM	Thermal Expansion Pressure Weight DBE or DBM	ASME B and PV Code Section III
Towing	Weight Towing Motion Effects	Weight Piping Loads Support Reactions Towing Motion Effects	Weight Towing Motion Effects	ASME B and PV Code Section III (Same as For Upset Conditions)

TABLE 3.9.2-1 (cont.)

STRESS CRITERIA FOR ASME CODE CLASS 1, 2 AND 3 COMPONENTS OF FLUID SYSTEMS

Conditions	LOADS			Criteria
	Piping	Mechanical Penetrations Shell Type Equipment & Components & their Supports	Piping Supports, Restraints & Anchors	
Faulted	Pressure Weight Pipe Rupture DBE or Design Basis Motion (DBM)	Thermal Expansion Pressure Weight Support Reactions DBE or DBM Pipe Rupture	Thermal Expansion Pressure Weight DBE or DBM Pipe Rupture	ASME B and PV Code Section III *

* Stress criteria for the faulted condition will be in accord with the draft Appendix F of the ASME B & PB Code Section III.

3.10 SEISMIC DESIGN OF SAFETY RELATED INSTRUMENTATION AND ELECTRICAL EQUIPMENT

RESAR-3, Section 3.10 is applicable to this section.

Seismic qualification of safety related electrical equipment and supports will be in accordance with IEEE Standard 344-1971, "IEEE Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations."

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

This section provides information on the environmental conditions and design bases for which the mechanical and electrical portions of the engineered safety features and reactor protection system are designed to assure acceptable performance in normal and worst postulated environments.

3.11.1 QUALIFICATION OF SAFETY RELATED MECHANICAL AND ELECTRICAL EQUIPMENT

Table 3.11-1 identifies all safety related mechanical and electrical equipment, indicating location, normal and postulated environmental conditions, and qualification of the equipment to perform in these environments. The condition causing the postulated environment is also indicated. General categories of equipment tabulated in 3.11-1 are detailed further in Tables 3.2-1 (Mechanical System Components Design Criteria Summary) and 3.2-3 (Electrical Systems and Equipment Safety Classification List) of the report.

Cable design and qualification is presented in Paragraph 8.3.1.2.3 of the report.

Containment electrical penetrations design and qualification is presented in Paragraph 3.8.2.7.4 of the report.

3.11.2 AIR CONDITIONING AND VENTILATION CRITERIA FOR CONTROL AREA

Temperature in the control area (Control Room, Process Rack and Rod Control Room) can be maintained in the range of 72 to 75 ($\pm 2^{\circ}F$); humidity a maximum of 50% RH. Two 100 percent Safety Class 3 air handling systems are provided for the Control Room. Two 100 percent refrigeration machines are provided for the cooling of the Control Room, Process Rack and Rod Control Rooms. The temperature will not be affected in the Control Room by a single failure of an air handling unit or refrigeration machine. In the event the Process Rack and Rod Control Room supply unit fails the return air fan will be engaged in providing the necessary air changes to maintain the room temperature well within the limits set by the design specifications for no loss of protective function.

A description of the air conditioning and ventilation systems is provided in Section 9.4 of the report.

Notes on Table 3.11-1:

1. These parameters are tabulated only where significant.

2. Dosage over life of plant is much less than post-LOCA dosage.

3. The neutron detectors will be designed for continuous operation at 135°F (57°C) (the normal operating environment is designed to be below this value) and will be capable of operation at 175°F (79°C) for eight hours.

   Air temperature detectors will be located in the suction ducts of the reactor cavity cooling fans with indication and alarm in the control room.

4. Local instrumentation is located to avoid extreme environmental conditions.

5. Diesel generators are designed for 122° F (50° C) ambient. Diesel room ventilation maintains the generator ambient at 110° F (43° C) with the diesel running.

Notes on Table 3.11-1 (Cont.)

6. The nature of mechanical and piping components and equipment is such that the post-accident environment is anticipated to have little if any affect on their design. However, the environmental parameters given in the Table will be incorporated into the design requirements for the respective items. The ability of the items to satisfy their design parameters will be verified by analysis and/or tests as required. Use will be made of type testing of components and materials, especially for passive items such as tanks, piping, and shielding.

7. If the ambient temperature of this item exceeds the postulated temperature for a significant period of time, this item will be considered out of service, and action will be taken per Chapter 16 of the report.

References on Table 3.11-1:

1. Locante, J. and Igne, E. G., "Environmental Testing of Engineered Safety Features Related Equipment (NSSS-Standard Scope)", WCAP-7744, Volume I, August 1971.

2. Wilson, J. F., "Electric Hydrogen Recombiners for PWR Containments". WCAP-7820, Supplement #1.

3. "IEEE Trial Use Guide for Type Tests of Continuous-Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations" (IEEE Standard 334-1971).

TABLE 3.11-1

ENVIRONMENTAL REQUIREMENTS FOR MECHANICAL AND ELECTRICAL EQUIPMENT

Item	Item Location	Normal Environment				Postulated Environment				Cause of Worst Postulated Environment	Qualification
		Temp.	Press. psia Note 1	Rel. Hum. at 100°F (Note 1)	Rad. Dosage (Note 1)	Temp.	Press. psia Note 1	Rel.Hum. at 100°F (Note 1)	Rad. Dosage (Note 1 & 2)		
Emergency Core Cooling System	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
	Safeguards Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Residual Heat Removal System	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
	Safeguards Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Condensate Feedwater System (selected sections of systems)	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
	Safeguards Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Main Steam System (Selected sections of system)	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
	Turbine Hall Safeguards Area	< 120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Annulus Filtration System											Note 6
	Aux. Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Ice Condenser	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
	Safeguards Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Hydrogen Recombiner System	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
Post-Accident Containment Hydrogen Venting System	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
Containment Spray System	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
	Safeguards Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Containment Isolation System	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
	Safeguards Area Aux. Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	

TABLE 3.11-1

ENVIRONMENTAL REQUIREMENTS FOR MECHANICAL AND ELECTRICAL EQUIPMENT (cont.)

Item	Item Location	Normal Environment				Postulated Environment				Cause of Worst Postulated Environment	Qualification
		Temp.	Press. psia (Note 1)	Rel.Hum at 100°F (Note 1)	Rad. Dosage (Note 1)	Temp.	Press. psia Note 1)	Rel.Hum. at 100°F (Note 1)	Rad. Dosage (Note 1 & 2)		
Component Cooling System	Containment	<120°F									
	Safeguards Area Auxiliary Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Auxiliary Raw Water System	Various	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	Note 6
Essential Service Water System	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
	Safeguards Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Essential Raw Water Supply	Safeguards Area Watertight Compt.	<120°F				40°F 120°F	15	100		Faulted heating Air Cond	Note 6
Auxiliary Feedwater System	Safeguards Area Watertight Compt.	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	Note 6
Compressed Air System (selected sections of system)	Turbine Area Safeguards Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	Note 6
Post Accident Sampling System	Containment	<120°F				250°F	30	100	2×10^8	LOCA	Note 6
	Control Area	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	
Emergency Diesel Fuel Oil System	Safeguards Area Watertight Compt.	<120°F				40°F 120°F	15	100		Faulted heating Air Cond.	Note 6

ENVIRONMENTAL REQUIREMENTS FOR MECHANICAL AND ELECTRICAL EQUIPMENT (cont.)

Item	Item Location	Normal Environment				Postulated Environment				Cause of Worst Postulated Environment	Qualification
		Temp.	Press. (psia (Note 1)	Rel.Hum. at 100°F (Note 1)	Rad. Dosage (Note 1)	Temp.	Press. (psia Note 1)	Rel.Hum. at 100°F (Note 1)	Rad. Dosage (Note 1 & 2)		
1. Engineered safety features (ESF) Motors: Containment air recirc fans	Containment	< 120°F	15		Note 2	250°F	30	100%	2×10^8 rads	LOCA	Type tests will be conducted in accordance with reference (3).
Containment lower compartment vent fans	Containment	< 120°F (49°C)	15								Type tests will be conducted in accordance with reference (3)
Other ESF motors	ESF areas outside containment	<104°F (40°C)				104°F (40°C) (Note 7)				Faulted air conditioning/ventilation to ESF area.	Industry experience and acceptance of motors designed for 40° C ambient.
2. ESF valve motors	Containment	< 120°F (49°C)	15		Note 2	250°F (121°C)	30	100%	2×10^8 rads	LOCA	Type tests have been conducted per reference (1).
	ESF areas outside containment	<104°F (40°C)				104°F (40°C) (Note 7)				Faulted air conditioning/ventilation to ESF area	Industry experience and acceptance of motors designed for 40°C ambient.
3. Hydrogen Recombiner	Containment	< 120°F (49°C)	15		Note 2	250°F (121°C)	30	100%	2×10^8 rads	LOCA	Type tests have been conducted per reference (2).
4. Instrumentation and control necessary for reactor trip and ESF actuation: a) Neutron detectors	Containment	135°F (57°C)	15			175° F (79°C) (Notes 3 and 7)	15			Faulted reactor cavity cooling fans	Power range detector has been tested in temperatures greater than 175°F (79°C) for 16 hrs. with negligible decrease in insulation resistance; insulation resistance is the governing factor for severe environments.

TABLE 3.11-1

ENVIRONMENTAL REQUIREMENTS FOR MECHANICAL AND ELECTRICAL EQUIPMENT

Item	Item Location	Normal Environment				Postulated Environment				Cause of Worst Postulated Environment	Qualification
		Temp.	Press. (psia Note 1)	Rel. Hum. at 100°F (Note 1)	Rad Dosage (Note 1)	Temp.	Press. (psia Note 1)	Rel. Hum. at 100°F (Note 1)	Rad Dosage (Note 1 & 2)		
4. b) Process Sensors	Containment	120°F (49°C)	15		Note 2	250° (121°C)	30	100%	2×10^8 Rads	LOCA	Type tests have been conducted per reference (1).
	ESF areas outside containment (Note 4)										
c) Nuclear instrumentation System Racks	Control Room	72-75°F (22-24°C)		50% max.		72-75°F (22-24°C)		50% max.		Single failure of redundant control room air conditioning/ventilation	Supplier has demonstrated operation of equipment up to 120°F (49°C) 95% RH with no loss of protective function. Detailed results retained by supplier.
d) Process Instrumentation and control racks	Process rack room	72-75°F (22-24°C)		50% max.		104°F (40°C) (Note 7)		50% max.		Faulted process rack room air conditioning/ ventilation	Supplier has demonstrated operation of equipment up to 120°F (49°C) 95% RH with no loss of protective function. Detailed results retained by supplier.
d) Solid state protection system cabinets	Process rack room	72-75°F (22-24°C)		50% max.		104°F (40°C) (Note 7)		50% max.		Faulted process rack room air conditioning/ ventilation	Supplier has demonstrated operation of equipment up to 120°F (49°C) 95% RH with no loss of protective function. Detailed results retained by supplier.
5. Reactor trip switchgear	Rod Control Room	72-75°F (22-24°C)		50% max.		104°F (40°C) (Note 7)		50% max.		Faulted rod control room air conditioning/ ventilation	Industry experience and acceptance of switch gear designed for 40°C ambient.
6. Vital Instrument bus inverters and AC distribution panels (4)	Process Rack room	72-75°F (22-24°C)				104°F (40°C) (Note 7)				Faulted process rack room air conditioning/ ventilation	Industry experience and acceptance of inverters and distribution panels designed for 40°C ambient.
7. Batteries and DC distribution panels (4)	ESF switchgear rooms (4)	<104°F (40°C)				104°F (40°C) (Note 7)				Faulted ESF switchgear room air conditioning/ ventilation	Industry experience and acceptance of batteries and distribution panels designed for 40°C ambient.

TABLE 3.11-1

ENVIRONMENTAL REQUIREMENTS FOR MECHANICAL AND ELECTRICAL EQUIPMENT

Item	Item Location	Normal Environment				Postulated Environment				Cause of Worst Postulated Environment	Qualification
		Temp.	Press. (psia Note 1)	Rel. Hum. at 100°F (Note 1)	Rad. Dosage (Note 1)	Temp.	Press. (psia Note 1)	Rel. Hum. at 100°F (Note 1)	Rad. Dosage (Note 1 & 2)		
8. Diesel generators (4) (see Note 5)	Diesel Rooms (4)										
9. ESF 4160V and 480V switchgear, ESF load sequencers (4)	ESF switchgear rooms (4)	<104°F (40°C)				104°F (40°C) (Note 7)				Faulted ESF switchgear room air conditioning/ventilation	Industry experience and acceptance of switchgear and relay cabinets for 40° Ambient.
10. ESF motor control centers (4)	Rooms adjacent to diesel rooms (4)	<104°F (40°C)				104°F (40°C) (Note 7)				Faulted area air conditioning/ventilation	Industry experience and acceptance of motor control centers design for 40° ambient.

3.12 PLATFORM STRUCTURE

The platform serves as the foundation for the Floating Nuclear Plant. Its primary function is to provide a stable watertight envelope to support the plant equipment and structures under operating and design bases conditions. Its secondary function is to house some of the systems necessary for plant operations and safe shutdown.

3.12.1 Description

The platform configuration is approximately 400 feet long, 378 feet wide and 40 feet deep with 40 foot (full depth) bulkheads in both the longitudinal and transverse directions extending the length and width of the platform to form a grid. Along the 378 foot width, bulkheads are located at regular intervals of about 37 feet, 9 inches. Along the 400 foot length bulkhead spacing varies from approximately 47 feet to 82 feet. The bulkhead spacing is arranged to provide support for major structures on the platform, provide shear area required for platform girder bending, and subdivide the platform into watertight compartments. The platform structure is designed using a plate and stiffener framing system of all welded mild steel construction. The material used for the platform construction is described in Subsection 3.12.4. Welding and nondestructive testing of the platform hull is discussed in Subsection 3.12.5.

The longitudinal and transverse strength of the platform is provided by the top deck at the 40 foot level, the bottom shell at elevation zero, sides, full depth bulkheads and all associated framing. Longitudinal framing, spaced 37-3/4 inches, is used for the top deck and bottom shell in conjunction with deep transverse girders spaced 10 to 12 feet between transverse bulkheads. All girders are built-up sections with typical depths of 4 feet, 6 inches for deck girders and 7 feet, 6 inches for bottom shell girders.

Bulkheads and sides are framed horizontally in conjunction with vertical webs (girders) in line with the 40 foot level and bottom shell girders to give planes of stiffness. The bulkhead and side plating stiffeners are spaced on 32 inch centers and vary in size according to the hydrostatic pressure head they are designed for.

The platform is basically a large rectangular box girder with all edges square except the bottom forward which has a 16 foot, 10 inch radius. This radius is provided to reduce resistance for towing the plant from the construction facility to the operating site.

The platform structural drawings will be reviewed by the American Bureau of Shipping. Fabrication and construction can be observed by an ABS surveyor for compliance with approved drawings and good construction details.

Figures 3.12-1 through 3.12-6 show the structural arrangement, basic member sizes and welding. Corrosion control measures are described in Subsection 3.12.6 and inspection and maintenance after construction is discussed in Subsection 3.12.7.

3.12.2 DESIGN BASES

3.12.2.1 General

The design of the platform as a whole and its individual components, meets or exceeds the requirements of the American Bureau of Shipping "Rules for Building and Classing Steel Vessels" where they are applicable, using loading conditions and stress levels similar to that required by ABS for ocean going vessels. Hull girder bending in still water has been determined using all dead loads on the platform, including platform weight, plus live loads for normal operation. The platform was also designed as a beam on a "standard" wave in a heavy sea. The longitudinal and transverse directions for both the hogging and sagging conditions were analyzed with all dead and live loads that will be on the platform during tow.

Compartmentation bulkheads within the platform and decks supporting vital equipment within the platform have been designed as watertight structures to prevent progressive flooding in the event of damage. The plating thickness and stiffener sizes meet or exceed the requirements of both the American Bureau of Shipping and the United States Navy for damage control structures.

3.12.2.2 Design Loads and Stress Categories

The platform is a stiffened-plate structure whereby the plating is supported

by a system of longitudinal and transverse stiffeners. The plating may be subjected to loads that fall into two general groups:

1. Loads which are applied in the plane of the plate and develop stresses which are tension, compression, shear or a combination of these stresses, called "in-plane" loads.

2. Loads which are applied perpendicular to the plane of the plate and develop tensile or compressive bending stresses in the plane of the plate and shear stresses, called "normal" loads.

In addition to adhering to the American Bureau of Shipping rules for building and classing ocean going vessels, the platform structure is being designed using an allowable stress approach. The allowable stress design is performed by specifying operating and design bases loading conditions and allowable stresses. The platform is being designed for loads and stress categories as follows:

3.12.2.2.1 Loads

D = Dead loads - weight of all structures and equipment.

L = Live loads - liquids and stores.

V = Operating basis wave within basin - see Section 3.7.

V_s = Design basis wave within basin - see Section 3.7.

V_t = Design basis environmental conditions during tow - standard trochoidal wave with length equal to the length or width of the platform and wave height from crest to trough equal to $1.1\sqrt{\text{wave length}}$ in conjunction with 40 knot wind.

E = Operating basis earthquake - see Section 3.7.

E_s = Design basis earthquake - see Section 3.7.

W = Operating basis wind - see Section 3.3.

W_s = Design basis wind - see Section 3.3

P = Design basis accident loads - pressure load and hydrostatic load from loss-of-coolant accident.

F = Load due to one compartment flooded.

F_s = Load due to two adjacent compartments flooding.

T = Load from thermal gradient in platform structure.

Hydraulic buoyancy forces and mooring reactions are included with all loading conditions to be compatible with applied loading.

3.12.2.2.2 Stress Categories

Three types of stresses are considered in the platform structure analysis.

Primary stresses - Plating develops in-plane stresses and stiffeners develop axial stresses from hull girder bending.

Secondary stresses - The plate-stiffener combinations bend locally under applied normal loads and the plate develops additional in-plane stresses since it acts as a flange with the stiffener, and stiffeners develop bending stresses.

Tertiary stresses - The plate panel develops bending stresses because, between stiffeners, it bends out of the mean surface of the stiffener system under applied normal loads. Tertiary stresses are not combined with primary and secondary stresses but are considered in the design of plating in paragraph 3.12.2.5.

The analysis of plating will be based on the maximum shearing distortion theory (Mises Theory) for a biaxial state of stress and the buckling criteria of plate panels.

3.12.2.3 Load Combinations and Allowable Stresses

The platform structure is being designed for the loading combinations and allowable stresses specified below.

Operating Basis Conditions

1. D + L

   Primary stresses do not exceed one-half the yield strength of the material.

2. D + L + P

3. D + L + E + P

4. D + L + V + W + P

5. D + L + F

6. D + L + T

   Combined primary and secondary stresses do not exceed two-thirds the yield.

Design Basis Conditions

1. $D + L + V_s$

2. $D + L + W_s$

3. $D + L + E_s + P$

4. $D + L + F_s$

   Combined primary and secondary stresses do not exceed 90% of yield.

Towing Conditions

1. $D + L$

   The section modulus of the platform hull girder is equal or greater than 25.3% of the bending moment from a one-dimensional analysis in still water, in accordance with American Bureau of Shipping Rules for ocean going barges. Section modulus is expressed in in^2-feet and moment is expressed in feet-long tons.

2. $D + L + V_t$

   Primary stresses do not exceed two-thirds the yield strength of the material.

Combined primary and secondary stresses do not exceed 90% of yield.

3.12.2.4 Buckling Criteria

The theory of stability of rectangular plates under compression or shear is well established. The stress at buckling depends on the ratio of plate size to thickness and on the degree of fixity at the panel edges. The degree of fixity is usually indeterminate except when plating is under essentially uniform normal loading as in the case of hydraulic pressures, the load is symmetrical across the panel supports, and it is usual to consider the panel edges as simply supported. In general buckling analyses will be based on simply supported edge conditions and for various loading conditions based on buckling theory. References 1 and 2 give the theory on buckling of rectangular plates. Reference 3 presents curves which eliminate all but the essential calculations.

Plating and stiffeners when loaded with various combinations of the primary and secondary stresses will be designed to meet the following criteria:

1. Combinations of primary and secondary stresses in panels of plating shall not exceed the buckling strength of the plating panel.

2. Combinations of primary and secondary stresses in panels of plating shall not be more than 80% of the ultimate strength of plating given by Figure 10 of Reference 3. This assumes the edge support is capable of sustaining the yield strength. Where this does not occur the stresses given by Figure 10 of Reference 3 shall be reduced by the ratio of column strength of the plate-stiffener combination to the yield strength.

3. For plate-stiffener combinations under combined axial and bending stresses the sum of the ratios of the primary and secondary stresses to the allowable compressive and bending respectively shall not exceed unity. The allowable compressive stress shall be 67% of the column strength for operating basis conditions and 80% of column strength for design basis conditions using Figure 1 of Reference 3, to determine the column strength. The allowable bending stresses are specified in paragraph 3.12.2.3.

Hence:

$$\frac{S_{axial}}{0.67\ S_{critical}} + \frac{S_{bending}}{0.67\ S_{yield}} \leq 1 \quad \text{for operating basis conditions}$$

$$\frac{S_{axial}}{0.80\ S_{critical}} + \frac{S_{bending}}{0.90\ S_{yield}} \leq 1 \quad \text{for design basis conditions}$$

3.12.2.5 Normal Loads

Methods of determining the stress and deflections of an isotropic plate under normal loads are dependent upon the relationship of the plate thickness to its deflection. As plating deflects under normal loads, the length of a strip between supports increases. This introduces a tension in the plate called "membrane effect" in addition to the bending stresses. For small deflections less than about 20% of the plate thickness, loads are resisted mostly by bending or "beam" action, but for deflections greater than about 50% of the plate thickness membrane effect is appreciable and stresses obtained by beam theory are not applicable.

The theory of bending has no practical significance for plates of a ductile material such as mild steel, which have considerable further strength due to membrane tension stresses. By accepting a small permanent set, a design load may be obtained that is larger than that allowable by elastic theory.(4)

Analysis of bulkhead plating and atmospheric tanks to withstand hydrostatic heads will be based on the requirements of American Bureau of Shipping "Rules for Building and Classing Steel Vessels", and United States Navy Specifications for damage control structures and tank boundaries. These requirements are based on theoretical analysis, experimental data and recorded data from actual ship installations.[5]

The effect of normal pressure on the buckling strength of rectangular plates is discussed in Chapter XII of Reference 1. The normal load does not prevent the usual buckling pattern, but delays it by raising the critical load. Reference 1 recommends computing the buckling strength of plating subjected to hydraulic pressures as if the normal pressure were not present. This is a more conservative analysis than attempting to combine the two conditions; hence, the buckling analysis does not include the effects of normal loads on plating panels.

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3.12.3 STRUCTURAL ANALYSIS

This subsection describes the techniques used for the structural analysis of the platform. The forty-foot level, bottom shell, side shell, and bulkheads with their associated framing are the items which provide the hull girder strength of the platform. Figure 3.12-7 shows this structure. The method used to analyze both the static and the dynamic conditions of the platform structure is based on static stress analysis.

3.12.3.1 Types of Analysis

The three classes of static analysis used are hull girder primary bending, framing member bending and axial compression or tension, and buckling of hull plating, plate stiffeners and of columns. Shear stresses based on the linear elastic theory are inherent in the analyses.

3.12.3.1.1 Hull Girder Primary Bending Stress Analysis

The primary stresses are calculated from the hull bending analysis. Preliminary strength calculations were performed for the still water condition using both a one-dimensional and two-dimensional analysis. The platform was also checked as a beam on a "standard wave" (the design basis wave during tow, paragraph 3.12.2.2.1). Additional two-dimensional analyses

will be accomplished in order to define the critical situation for each of the loading conditions defined in paragraph 3.12.2.2.1. Expansion and verification of the structural member strength calculations is being accomplished using a three-dimensional, finite element analysis for the still water condition and will be done to define the structural member stresses for the loading combinations mentioned in paragraph 3.12.2.3. One of the loading combinations will include a thermal stress analysis. The temperature gradients within the platform will be determined using a three dimensional, finite difference analysis by the computer program GNATS described in Appendix 3A.

3.12.3.1.2 Girder and Stiffener Bending and Axial Stress Analysis

The secondary stresses are combined with the primary stresses in order to evaluate the strength of the members of platform structure.

For the preliminary stress analysis the bending moments and shear forces which produce the secondary stresses are calculated using standard formulas from the various handbooks (1), (2) and also by the method of moment distribution or method for the analysis of statically indeterminate structures.

For these calculations the member strength characteristics, i.e. section properties, include a portion of deck plating as a flange of the girder

or stiffener.

The effective width of plating associated with any girder is determined essentially by expressions presented in reference (8). The section properties were taken from references (7) and (8) where applicable. Preliminary strength calculations were performed for the still water operating condition. Final analysis will include calculations for the loading combinations discussed in paragraph 3.12.2.3.

3.12.3.1.3 Plate Buckling and Plate Stiffener and Column Buckling

The critical buckling force or stress is calculated and compared with the actual force or stress in order to verify the design. Critical buckling stresses of the plate elements and critical buckling forces in the framing members are calculated using the procedure discussed in subsection 3.12.2.4. The calculations of the compressive strength of hull plating and hull framing members were performed for the still water operating condition and they will be completed for the other loading conditions described in paragraph 3.12.2.3.

3.12.3.2 Results of Analysis

The following is a summary with specific examples of the preliminary

stress analyses.

3.12.3.2.1 Hull Primary Bending

The one-dimensional analysis used the standard hull girder-beam analogy in both the longitudinal and transverse directions. The moment and shear curves for these one-dimensional analyses are shown in Figure 3.12-8 for still water operating basis condition and the design basis wave during towing conditions.

In accordance with American Bureau of Shipping Rules for ocean-going barges the section modulus of the platform hull girder was determined and compared with the bending moment (the required section modulus is to be greater than or equal to 25.3% of the bending moment).

	Bending Moment (feet-long tons)	.253 x Bending Moment	Section Modulus ($inches^2$-feet)
longitudinal	1,104,000	279,000	298,000
transverse	1,090,000	276,000	289,000

The average axial stresses in the 40 foot level deck and the bottom shell are also indicated on Figure 3.12-8. The results were obtained following the procedures outlined in reference (6).

The two-dimensional analysis used a bulkhead and shell plating beam analogy and a grid analysis technique. The grid used to model the platform consisted of beam elements connected at 49 nodes arranged to gain a basic understanding of the platform stresses.

The family of bending moment and shear curves for this two-dimensional analysis is shown in Figure 3.12-9. The maximum axial stress at the 40 foot level deck and the bottom shell are listed on this figure. A typical application of the distortion energy criterion to the results of the two-dimensional analysis follows:

Hull Bottom Plating adjacent to the intersection of bulkhead F and bulkhead 4.

Primary Stresses:

Transverse (S_y) = 16,000 psi tension

Longitudinal (S_x) = 13,200 psi tension

Shear (S_{xy}) = 1,900 psi

Allowable stress (S_a) = 1/2 yield stress (S_{ys})

Distortion energy stress criterion

$$S^2_{max} = S_y^2 - S_y S_x + S_x^2 + 3 S_{xy}^2$$

$$S^2_{max} = 16{,}000^2 - 16{,}000 \times 13{,}200 + 13{,}200^2 + 3 \times 1{,}900^2$$

S_{max} = 15,200 psi

so: $S_a \geq S_{max}$ therefore the design is satisfactory

The results were obtained using the computer code SAND, described in Appendix 3A and following procedures suggested by reference (5).

The detailed analysis of the hull stresses are to be calculated using a finite element solution technique for models consisting of discrete beam and plate elements. The arrangement of nodes in three orthogonal directions for one model of this problem is shown in Figure 3.12-10. This model consists of nodes at the hull bottom and the 40 foot level deck. The nodes are located at the intersection of each bulkhead and at the intersection of each transverse girder with each bulkhead and side shell. Using such a model the primary stresses can be found for each plate element and the secondary stresses can be found for each transverse deck girder, vertical bulkhead girder and bottom girder.

The results will be obtained using the computer code STRAP, described in the Appendix 3A and following procedures suggested by References (3) and (4).

The resultant stresses will be checked by the criteria defined in paragraph 3.12.2.2.2 in order to verify that the design is satisfactory.

3.12.3.2.2 Girder and Stiffener Bending and Axial Stresses

The appropriate section properties and loads were calculated for the transverse girders and longitudinal stiffeners. After the loading conditions and section properties of a girder are found the maximum fiber stress and average shear stress are determined using the appropriate for formula from Reference (1) or (2) or by the moment distribution method.

These secondary stresses are combined with the primary stresses and these combined stresses are checked by the criteria defined in paragraph 3.12.2.2.2 in order to verify that the design is satisfactory.

A typical example of a girder (see Figure 3.12-4) stress analysis follows:

Hull Bottom Transverse Girders between Bulkhead G and H

Local Direct Pressure

Assumptions: Platform at 30 ft. draft (d) operating condition

Uniform hydrostatic load along girder

Fixed end beam

Efficient bracket such that maximum stress occurs at center of span

Spacing (s) = 11.75 feet

Density (v) = 64 pounds per cubic foot

Length (L) = 453 inches

Section property of girder:

Girder (20" x 40.8# Plate flange on 90" x 20.4# Plate web on 67.5" x 1.125" Shell Plate)

Section Modulus at flange (Z_1) = 2839 in^3

Section Modulus at shell (Z_2) = 6465 in^3

Shear Area (A_1) = 45.5 in^2

Bending moment, center of span (BM) = d x s x v x $L^2/24$

Flange stress ($S_{1,2}$) = BM/Z_1 or BM/Z_2

$S_1 = 30 \times 64 \times 11.75 \times 453^2/(24 \times 2839 \times 12)$

$S_1 = 5{,}700$ psi tension at flange

$S_2 = 30 \times 64 \times 11.75 \times 453^2/(24 \times 6465 \times 12)$

$S_2 = 2{,}500$ psi compression at shell

Shear force near end of span (V) = d x v x s x L/2

Shear strees (S_v) = V/A

$S_v = 30 \times 64 \times 11.75 \times 345/(12 \times 45.5 \times 2)$

$S_v = 7100$ psi shear near ends of girder span

Hull girder - Primary Stress

Transverse Direction (S_y) = 16,000 psi tension at shell

(S_y) = 10,000 psi tension at girder flange

Longitudinal Direction (S_x) = 13,200 psi tension at shell

Shear (S_{xy}) = 1900 psi

(reference the two-dimensional analysis)

Primary plus secondary stress

$S_1 + S_y = 5{,}700 + 10{,}000$

= 15,700 psi tension at flange

$S_2 + S_y = -2500 + 16{,}000$

= 13,500 psi tension at shell

Distortion energy stress criterion

Allowable stress (S_a) = 2/3 yield stress (S_{ys}) Ref. 3.12.2

$$S^2_{max} = S_x^2 - S_xS_y + S_y^2 + 3 \times S_v^2$$

For shell plate $S_{max} = (13,200^2 - 13,200 \times 13,500 + 13,500^2 + 3 \times 1900^2)^{1/2}$

$= 13,800 \text{ psi} < 2/3\ S_{ys}$

For web of girder $S_{max} = (16,000^2 + 3 \times 7127^2)^{1/2}$

$= 20,200 \text{ psi} < 2/3\ S_{ys}$

For flange of girder $S_{max} = 15,700 < 2/3\ S_{ys}$

3.12.3.2.3 Plate, Plate Stiffener and Column Buckling

Section properties and loads were calculated for the vertical columns on the bulkheads, the transverse girders and the longitudinal stiffeners which are subject to a compressive load. Utilizing the section properties the critical buckling load or stress is calculated for each member. If the critical stress is found to be less than the yield stress then the allowable stress for the column or plate is considered to be reduced to that stress defined in paragraph 3.12.2.4. The stress analysis would then proceed the same as for the girder and stiffener bending and axial stress analysis in order to verify the design. A typical example of a bulkhead column (Figure 3.12-4) stress analysis follows.

Column (Vertical Girder) on Longitudinal Bulkhead

Local Direct Pressure on Bottom

Assumptions: Platform at 30 ft. draft (d)

operating conditions

Uniform hydrostatic load on bottom

Area supported (A_1) = 11.75 feet x 37.75 feet = 443.56 $feet^2$

Length (L) = 28 feet

Water density (v) = 64 pounds per cubic foot

Section property of girder

Girder (15" x 30.6# plate flange on 54" x 17.85# plate web on 30.6# plate bulkhead)

Compressive Load (P) = 30 x 443.56 x 64 = 852,000 pounds

Critical Compressive Load (Pcr) = Compressive cross sectional area of bulkhead, web and flange (A_{bhd}, A_{web}, A_{flg}) x Critical buckling stress (S_{cr})

$$P_{cr} = A_{bhd} \times S_{cr} + A_{web} \times S_{cr} + A_{flg} \times S_{cr}$$

$$= 31.50 \times 32,000 + 32.6 \times 14,000 + 11.25 \times 32,000$$

$$P_{cr} = 1,455,000 \text{ pounds}$$

Allowable Compressive load (P_a) = 2/3 P_{cr}

P_a = 969,965 pounds 851,634 pounds

therefore, design is satisfactory.

3.12.3.3 Detail Design

Emphasis is placed on minimizing stress concentrations in the platform structure. Some of the requirements to be implemented are: continuous welding for all members; radius corners for rectangular openings; suitable reinforcement for large openings; and chocks, brackets, stiffeners, etc. at the intersection or junction of members to give better stress distribution.

REFERENCES

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3.12.4 PLATFORM HULL MATERIAL

The evaluation of hull steel considers the design and its stresses caused by variable loadings both during transport at sea and during operation within the breakwater, the resistance to brittle fracture, the ease of fabrication, and protection from corrosion.

Subsection 3.12.2 outlines the load combinations and allowable stresses to be used for the operating and design bases conditions for analyzing the platform structure. The primary stresses in the platform hull from hull girder bending in still water conditions will not exceed 1/2 the yield strength of the material. The combined primary and secondary stresses will not exceed 2/3 the yield strength of the material for operating basis conditions or 90% of the yield strength for design basis conditions. The stresses approach the maximum allowables near the center of the platform and taper off toward the edges.

Analyses to date indicate the hull girder bending mode will be such that the bottom flange is in tension and the top flange is in compression for operating basis conditions. The stresses will vary from shifting of loads on the platform, wave conditions inside the basin, wind forces, etc.; however, no condition investigated would cause a complete reversal of stresses

in the hull girder. Under normal operating conditions the stresses do not vary more than approximately 4 ksi from still water conditions. Based on analyses, the probability of fatigue fracture is extremely small.

Figure 3.12-11 is an elevation showing the approximate waterline at which the plant will float. The submerged portion of the platform will be subjected to the temperature of the surrounding water. The lowest expected temperature of the sea water would be approximately $30^{o}F$. The 40 foot level is completely enclosed, except for a small docking area, by buildings located on the platform. The deck temperature does not vary significantly from the temperature of the enclosed buildings, i.e., $40^{o}F$ minimum. It approaches the temperature of the outside air only in the event of a prolonged loss of heating and even then the deck would be in compression.

A minimum service temperature of $+30^{o}F$ has been specified for determining a steel that will be ductile under the most severe conditions of temperature and stress. This temperature is conservative for the immersed portion of the platform and for the almost completely enclosed 40 foot level.

The exterior sides above the waterline would be subjected to the surrounding air temperature; however, there would be a temperature gradient between the immersed portion and the 40 foot level. A thermal analysis of the platform structure will be performed to determine the temperature gradient

and thermal stresses. Minimum temperatures from the thermal analysis will be related to maximum stresses in the side shell determined from detailed analyses of various loading combinations to ensure that the material will not fracture in a brittle manner.

American Bureau of Shipping Grade CS steel will be used for the outer skin and 40 foot deck of the platform hull. This material is a fully killed, normalized carbon steel made with fine grain practice, developed for use in special applications in shipbuilding. The internal platform structure including subdivision bulkheads, deep girders and plate panel stiffeners will be constructed of ABS grade steels in accordance with the Rules for ocean going vessels. Table 3.12.4-1 gives requirements for hull structural steels. Paragraph 3.8.2.8.8 specifies the material for the platform plating forming a path of the containment pressure boundary.

Mill inspection and construction are conducted in accordance with American Bureau of Shipping requirements in addition to the requirements specified in Chapter 17. The mill inspection consists of an ABS Surveyor witnessing mechanical tests, examining chemical analysis report, and applying the appropriate stamp or symbol signifying that the material has been tested in accordance with the rules. An ABS Surveyor at the construction site inspects steel plates during fabrication and erection to make sure that the materials used in the construction are those which were approved for use on the particular vessel and those which passed the mill inspection.

The material requirements for the exterior plating of the platform hull girder are based upon:

1. $30^{o}F$ as minimum service temperature.

2. Nil-Ductility-Transition Temperature (NDTT) as the parameter to express notch sensitivity.

3. Material NDTT must be $60^{o}F$ below minimum service temperature or $-30^{o}F$.

4. Drop weight test (ASTM E208 or equivalent) at $60^{o}F$ below minimum service temperature, or $-30^{o}F$. Test would be no-break performance at specified testing temperature, based on two tests for each plate.

Figure 3.12-12, the fracture analysis diagram (FAD) developed at the Naval Research Laboratory, describes the general conditions of stress, temperature and flaw size that are required for fracture initiation and propagation. The fracture transition elastic (FTE) temperature is the highest temperature of fracture propagation for purely elastic stresses. For all steels in the mild steel, or carbon steel classification FTE= NDT + $60^{o}F$. As shown by Figure 3.12-12, the selection of a material with an NDTT $60^{o}F$ below minimum service temperature provides a material in which fracture propagation would not occur in the elastic range for stresses up to the yield strength.

Figure 3.12-13 shows average charpy-V notch impact values versus temperature and the nil-ductility-transition temperature for various hull steels used in Navy and commercial shipbuilding. Based on the data, ABS Grade CS steel meets the requirements specified.

Welding and non-destructive testing of the platform material is discussed in subsection 3.12.5. Corrosion control of the platform hull is discussed in subsection 3.12.6 and in-service inspection and maintenance is discussed in subsection 3.12.7.

TABLE 3.12.4-1

REQUIREMENTS FOR ORDINARY STRENGTH HULL STRUCTURAL STEEL

Grades	A	B	C	CS	D	E	R
Process of Manufacture	For all grades: Open Hearth, Basic Oxygen or Electric Furnace.						
Deoxidation	Any method	Semi-killed or killed	Fully killed, fine grain practice[1]	Fully killed, fine grain practice[1]	Semi-killed or killed	Fully killed, fine grain practice[1]	Semi-killed or killed
Chemical Composition (Ladle analysis)							
- Carbon, %		0.21 max.[2,6]	0.23 max.[4]	0.18 max.	0.21 max[5]	0.18 max.[5]	[5]
- Manganese, %		0.80-1.10[2,6]	0.60-0.90[2]	1.00-1.35	0.60-1.40[5]	0.70-1.50[5]	2.5xCmin.[5]
- Phosphorus, %	0.05 max.	0.05 max.	0.05 max.	0.05 max.	0.05 max.	0.05 max.	0.05 max.
- Sulphur, %	0.05 max.	0.05 max.[3]	0.05 max.	0.05 max.	0.05 max.	0.05 max.	0.05 max.
- Silicon, %			0.10-0.35	0.10-0.35	0.35 max.	0.10-0.35	
Heat Treatment			Normalized over 35.0 mm(1.375 in.) thick[9]	Normalized		Normalized	
Tensile Test							
- Tensile Strength	For all grades: 41-50 kgs per sq. mm[7] or 58,000-71,000 lbs/sq. in. or 26-32 tons/ sq. in.						
- Elongation	For all grades: 21% in 200 mm (8 in.)[7] (see 43.3.4c and 43.3.4d or 24% in 50 mm (2 in.) or 22% in 5.65 $\sqrt{A}$. (A equals area of test specimen)						

TABLE 3.12.4-1(CONT)

Grades	A	B	C	CS	D	E	R
Bend Test	For all grades: 180° around a diameter equal to 3 times thickness of specimen.[8]						
Impact Test, Charpy Standard, V-Notch - Temperature - Energy, avg., min. - No. of Specimens					0°C (32°F) 4.8 kg. m. (35 ft-lbs) 3 from each 40 tons	-10°C(14°F) 6.2 kg. m. (45 ft-lbs) 3 from each plate	
Stamping	AB/A	AB/B	AB/C (as rolled) AB/CN (normalized)	AB/CS	AB/D	AB/E	AB/R

NOTES:

1 The requirement for fine grain practice may be met by either (a) A McQuaid-Ehn austenite grain size of 5 or finer in accordance with ASTM Designation E 112 for each ladle of each heat; (b) Minimum acid-soluble aluminum content of 0.020% for each ladle of each heat; or (c) Minimum total aluminum content of 0.025% for each ladle of each heat.

2 Upper limit of manganese may be exceeded, provided carbon content plus 1/6 manganese content does not exceed 0.40%.

3 When the silicon content is 0.10% or more (killed steel), the minimum manganese content may be 0.60%.

4 For normalized Grade C plates, the maximum carbon content may be 0.24%.

NOTES (CONT):

5 Carbon content plus 1/6 manganese content shall not exceed 0.40%.

6 Where the use of cold flanging quality has been specially approved (3.1), the manganese content may be reduced to 0.60-0.90%.

7 The tensile strength of cold flanging steel shall be 39-46 kg/mm^2 (55,000-65,000 psi) and the elongation 23% minimum in 200 mm (8 in.).

8 The bend test requirements for cold flanging steel are:

- 19.0 mm (0.75 in.) t and under ..180° flat on itself;
- Over 19.0 mm (0.75 in.) t to 32.0 mm (1.25 in.) t, 180° around diameter of 1 x specimen thickness;
- Over 32.0 mm (1.25 in.) t . .180° around diameter of 2 x specimen thickness.

9 Grade C plates over 35.0 mm (1.375 in.) in thickness are to be ordered and produced in the normalized condition when intended for important structural parts.

10 A tensile strength range of 41-56 kg/mm^2 (58,000-80,000 psi) may be applied to Grade A shapes provided the carbon content does not exceed 0.25% by ladle analysis.

3.12.5 WELDING AND NONDESTRUCTIVE TESTING

The Applicant's Quality Control and Welding Engineering Section will exert technical control over all welding operations required in manufacture of the Floating Nuclear Plant hulls and will conduct, interpret and document all nondestructive tests required to establish the compliance of such welding with the acceptance standards specified by drawings, specifications and quality assurance and reliability instructions. This Section is divided into four functional groups, each of which has a role in the manufacture of the platform hull. These groups are as follows:

1. Quality Control

2. Welding Engineering

3. Nondestructive Testing

4. Materials Evaluation

Each of these groups will provide a unique generic role in establishing the adequacy of the platform hull to meet designed objectives.

3.12.5.1 Quality Control

Quality Control's primary function is to insure that the Floating Nuclear Plant hulls are manufactured in accordance with drawings, specifications

and Quality Assurance and Reliability Criteria, and that the tests necessary to prove such compliance are properly performed and documented. With respect to the platform hull these tests will include, but not be limited to, the following:

1. Inspection and testing of purchased material for compliance with specifications.

2. Inspection of in-process workmanship and documentation of all inspections performed.

3. Conducting and documenting final inspections.

4. Input to and maintenance of the Quality Control data bank.

5. Utilizing the output from (4) to effect remedial action plans as indicated, directed towards quality improvement goals.

3.12.5.2 Welding Engineering

The welding engineering program as applied to the production of the Floating Nuclear Plant hulls will take into account the requirements of the ASME Code-Section IX for qualifications, Section II, Part C, for consumables and Section III for nuclear components and systems.

Welding engineering criteria for the platform will be more stringent than are applied in commercial shipbuilding. The program will be structured as follows:

1. Welding procedures will be in accordance with Section IX of the ASME Code.

2. Welding procedure qualification will be in accordance with Section IX of the ASME Code.

3. Welder and Welding Operator testing and qualification will be in accordance with Section IX of ASME.

4. Welding consumables will be purchased in accordance with the requirements of Section II, Part C, of ASME, and where required will also meet ABS approval criteria.

5. Welding consumables entering into construction of the platform will be inspected on receipt for conformance to purchase specification, and furthermore will be tested on a statistical sampling basis. Non-conforming materials will be segregated and disposed of in accordance with Quality Assurance and Reliability procedures.

6. A welding engineering facility will provide support for, and perform,

the above operations. Additionally, this activity will furnish design liaison and will provide technical control of all welding performed in manufacture of the Floating Nuclear Plant hulls.

3.12.5.3 Nondestructive Testing

Nondestructive testing (N.D.T.) as applied to the platform structure will result in an inspection level and frequency which affords a margin of safety above the American Bureau of Shipping requirements. In addition, the application of a quality assurance plan in accordance with the requirements of 10 CFR 50, Appendix B will result in a reliability level higher than is normally applied in commercial shipbuilding. The N.D.T. program will follow the following guidelines:

1. N.D.T. Procedures will be in accordance with Section V of the ASME Code.

2. N.D.T. procedures will be qualified in accordance with Section V of ASME. N.D.T. personnel will be qualified in accordance with the requirements of SNT-TC-1A of the American Society for Non-Destructive Testing and will be so certified.

3. All N.D.T. equipment and techniques employed will meet the requirements of ASME Section V as a minimum.

4. Nondestructive testing of welds in the platform (hull) component of the Floating Nuclear Plants will be conducted as follows:

a. Longitudinal seams. All longitudinal plating seam welds (shell bottom, sides, deck, and internal bulkheads) will be examined by the shear wave ultrasonic technique using approved procedures for approximately one-fifth of the total lineal length of each such weld. This inspection will be conducted primarily for process control purposes during flat panel manufacture in the panel shop. This inspection level (20%) significantly exceeds the requirements of the American Bureau of Shipping.

b. Hull erection seams and butts in the platform shell. Welds executed in erecting the platform shell (deck, sides and bottom) will be tested nondestructively (ultrasonic or radiographic) on a statistical reliability basis as follows:

Level 1: One inspection location (12" long) for each 5 feet of welding produced until a total of 500 lineal feet of welding has been sampled.

Level 2: One inspection location (12" long) for each 10 feet of welding produced until a total of 2,000 lineal feet have been sampled at this level.

Level 3: One inspection location (12" long) for each 20 feet of welding produced until a total of 4,000 lineal feet have been sampled at this level.

Level 4: One inspection location (12" long) for each 50 feet of welding produced thereafter.

In applying the above sampling plan, defective welding revealed at a specific sample location will be fully evaluated by non-destructively testing each side of the original inspection location until the full extent of the defective area is established. This system of inspection "extension" is in accordance with code requirements. Weld quality accept/reject standards will be in accordance with American Bureau of Shipping and ASME requirements. Defective welds identified by nondestructive testing will be removed, repaired and re-inspected.

In operating the statistical sampling plan, an acceptance quality level (AQL) will be established based on the ratio of defective to acceptable weld length. When this AQL is exceeded in production, an increase in inspection frequency to the next higher level will be imposed until process control is re-established and defect cause-occurrence is within acceptable limits.

A system will be utilized which will identify all welding to the welders or welding operators involved. Used in conjunction

with the above AQL method, this system will permit early recognition of substandard individual performance and serve as a basis for remedial training or other action.

c. Hull welds in the containment area. Certain welds in the hull structure are identified as forming part of the containment pressure boundary. Such welds will be nondestructively tested to the same criteria and extent as specified for the remainder of the containment structure.

d. Hull structure attachment welds. Fillet welds in the hull structure will be visually inspected for defects in accordance with Section V of the ASME Code, and when they are identified, such defective areas will be repaired and re-inspected.

As a basis for comparing the statistical inspection sampling plan to a system currently in use, attention is directed to the latest revision of NAVSHIPS-0900-000-1000. This document is titled "Fabrication, Welding and Inspection of Ship Hulls" and is currently used as a control document in Naval Shipbuilding programs. In Table 6.1 of this document the inspection plan for "Ultrasonic or Radiographic Inspection Requirements for Surface Ships" is delineated. It should be noted that the United States Navy has developed these inspection criteria from engineering data and operational experience, and states that this system provides a level of safety adequate to insure satisfactory performance of the hulls of Navy

combatant surface ships under all conditions of dynamic loading encountered in the performance of assigned combat missions, e.g. uninterrupted performance under all sea-state conditions, maximum resistance to combat-induced shock damage and adequate protection of embarked personnel. It should also be noted that the Naval Reactors Branch of the Atomic Energy Commission has sanctioned the installation and operation of nuclear propulsion plants in Naval surface ships, the hulls of which have been and are being inspected to the above criteria.

Tables 3.12.5-1 and 3.12.5-2 can be used in comparing the inspection plan for the floating nuclear plant hulls to the U.S.N. requirements mentioned above.

Table 3.12.5-3 shows such a comparison and indicates that the inspection plan for the floating nuclear plant hulls will afford an inspection level approximately 4 to 10 times as stringent as that applied to the hulls of nuclear powered surface ships being supplied to the U.S. Navy.

3.12.5.4 Materials Evaluation

When required by Quality Control Specifications or other controlling documents, materials used in fabrication of the platform hull will be subjected to specified tests by the applicant's Materials Evaluation group. Capability will be provided to conduct the following types of tests:

1. Static and dynamic tests for mechanical properties of materials.

2. Metallographic/metallurgical properties determinations.

3. Chemical and physical properties analyses of materials.

4. Paints and coatings tests and evaluations.

5. Corrosion tests.

6. Destructive testing.

TABLE 3.12.5-1

ULTRASONIC OR RADIOGRAPHIC INSPECTION PLAN FOR WELDS IN THE FLOATING NUCLEAR PLATFORM HULLS

Category		Inspection Extent
A-1	Seam (longitudinal) joints in the hull plating. (Deck, sides, bottom and internal bulkheads)	20 percent of the length of each joint.
B-1	Seams and butts in the platform shell welded during erection:	
	Level 1	20 percent of the length of each joint.
	Level 2	10 percent
	Level 3	5 percent
	Level 4	2 percent
C-1	Seams and butts in hull plating designated as part of the containment pressure boundary.	100 percent

TABLE 3.12.5 -2

ULTRASONIC OR RADIOGRAPHIC INSPECTION REQUIREMENTS FOR SURFACE SHIPS

Category of structure		Extent of Inspection
	A-SHELL PLATING	
A-1	Intersections of butt-welded joints in plating over 3/8 inch within the midship 3/5 length.	10 percent
A-2	Intersections of butt-welded joints in the sheer strake to other shell plating over 3/8 inch within the midship 3/5 length	All intersections.
A-3	Butt (transverse) joints in plating over 3/8 inch within the midship 3/5 length	1 random location in each 20-foot length.
A-4	Seam (longitudinal) joints in plating over 3/8 inch within the midship 3/5 length.	1 random location in each 40-foot length
A-5	Butt (transverse) joints in the flat and vertical keel within the midship 4/5 length.	10 percent of the length of each joint.
A-6	Butt (transverse) joints in the bilge and steer strakes within the midship 3/5 length.	10 percent of the length of each joint.
A-7	Butt (transverse) joints in shell plating over 1 inch outside the midship 3/5 length.	5 percent of the length of each joint.
	B-STRENGTH DECK(S)	
B-1	Intersections of butt-welded joints in the stringer strakes to other deck plating over 3/8 inch within the midship 3/5 length of the main strength deck(s) or inner-bottom tank top	All intersections.
B-2	Butt (transverse) joints in the stringer stakes of main strength deck(s) or inner-bottom tank top within the midship 4/5 length	10% of the length of each joint

TABLE 3.12.5-2 (CONT)

Category of structure		Extent of Inspection
	B-STRENGTH DECK(S) CONT.	
B-3	Butt (transverse) joints in plating over 1 inch in main strength deck(s) or inner bottom tank top within the midship 3/5 length.	5 percent of the length of each joint.
	C-UNDERWATER SIDE PROTECTIVE SYSTEM OR BALLISTIC PLATING OVER 3/8 INCH	
C-1	Butt-welded joints in plating.	1 random location in each 20-foot length
C-2	Intersections of butt-welded joints.	10 percent
C-3	Butt-welded joints welded from one side only	100 percent
	D-SUPERSTRUCTURE DESIGNED FOR BLAST LOADING	
D-1	Intersections of butt welded joints in outside plating over 1/4 inch.	10 percent
D-2	Butt-welded joints in outside plating over 1/4 inch.	1 random location in each 20-foot length.
D-3	Butt joints in supporting structure stiffening member over 8 inches in depth and over 3/8 inch thick.	10 percent of the length of each joint.

TABLE 3.12.5-3

WELD INSPECTION ON THE FLOATING NUCLEAR PLANT HULL

Category	No. of Inspections (12" long)
A-1	Approx. 15,600
B-1	
Level 1	Approx. 3,400
Level 2	Approx. 1,700
Level 3	Approx. 860
Level 4	Approx. 340
C-1	Approx. 5,000
Total Weld Footage in above categories	Approx. 95,200
Total Footage Inspected (Assume B-1 Level 3)	Approx. 21,460
Percentage Inspection	= 22.5%

Weld Inspection, Typical Nuclear Carrier	13,000 Locations
Weld Inspection, Typical Nuclear Frigate	700 Locations

Estimated percentage hull inspection, U.S.N. = 2 - 5 %

3.12.6 CORROSION CONTROL

The platform design incorporates various means of corrosion control which have proved successful in a saltwater environment. The plant is designed for a forty year life and the time span has been considered in the selection of corrosion control means, relative to durability, inspection and maintenance. There are four distinct zones of the platform to be considered in combating corrosion. These are immersed, splash and atmospheric zones of the exterior; and the internal surfaces of plating and stiffeners. All exterior (weather exposed) steel surfaces will be abrasive blasted prior to coating, to the quality known as "White Metal Blast Cleaned Surface Finish," in accordance with Steel Structures Painting Council SSPC-SP-5. This surface is free of all oil, grease, dirt, mill scale, rust corrosion products, oxides, paint and other foreign matter. Immediate application of a primer will retain this rust-free condition until coating can be accomplished, or cathodic protection established.

3.12.6.1 Immersed Zone

The exterior platform plating forming the bottom and sides up to the waterline (submerged portion) will be protected by an impressed current cathodic protection system described in Subsection 9.6.3. So long as the external current is maintained, cathodic protection will provide an effective means of reducing corrosion rate of the submerged portion of the outside hull

virtually to zero for an indefinite period. In addition, a redundancy is included by adding a corrosion allowance increment. The hull plating (bottom and sides up to the 40 foot elevation) has a 0.250 inch corrosion allowance increment added to the thicknesses required for strength and stiffness considerations.

The 0.250 inch increment was selected after a review of literature on corrosion including measured results of exposure of steel to corrosion by sea water.

Table 3.12.6-1 (from: H. H. Uhlig, "Corrosion Handbook", John Wiley and Sons, N. Y., 1948) gives results for various geographical locations and duration of exposure. The generally accepted number for relatively uniform corrosion rate for steel in sea water is 0.005 inches per year. This represents a time-averaged value, with the initial rate of attack usually being greater than the final rate. Therefore, the expected loss of steel on the underwater portion of the hull from uniform corrosion over a 40 year period would be approximately 0.200 inches if no means of corrosion control were provided. The above average rate does not take into account non-uniformity of corrosion in the form of pitting; however, with carbon steel, pitting penetration relative to total corrosion diminishes to a minor fraction after long times such as are considered here.

3.12.6.2 Splash Zone

The splash zone is the most critical zone to protect because of the continual change in submergence and aeration. The extent of this zone in the present case will be determined by model tests, together with known climate effects. The splash zone will be protected with a coating which will be compatible with the impressed current cathodic protection system for the immersed portion. Coatings are available such as molded rubber, a glass-flake and polymeric resin claddings that should last up to 10 years maintenance-free. The splash zone including two feet below the water line can be recoated as required, by trimming the plant within operating limits to expose one side at a time for maintenance.

3.12.6.3 Atmospheric Zone

The atmospheric zone is the area of least corrosion rate of the exterior exposed structures. Much experience and information is available from the offshore oil industry on the use of coatings on steel structures exposed to salt air atmosphere. Coatings will be used for this area which are expected to be maintenance free for up to 8 years after initial application. After 8 years service touch-ups and repairs will be required periodically.

3.12.6.4 Internal Surfaces

The interior of the platform except for specific areas such as ballast tanks, will be dry voids, but will be subject to condensation and dampness. The internal surfaces will be abrasive blasted to the quality consistent with the corrosion control coating specified. Details of design and fabrication are considered in that items to be coated are fully accessible and all joints, closures, intersections, etc., are accessible and continuously welded. It is anticipated that coatings in these areas would be maintenance free for up to 15 years service. In the case of ballast tanks, cathodic protection can be used in conjunction with the coating to extend still further the maintenance-free life of submerged surfaces. If this is done, the possibility of hydrogen generation will be precluded by proper alloy selection if sacrificial anodes are used, or by potential-limiting components if an impressed current system is employed.

3.12.6.5 Inservice Inspection

Inservice inspection for corrosion is discussed in Subsection 3.12.7.

TABLE 3.12.6-1

RESULTS OF EXPOSURE OF STEEL AND IRON TO CORROSION BY SEA WATER UNDER NATURAL CONDITIONS

Steel Specimens--Continuously Immersed

Line	Original Condition of Surface	Duration of Exposure, years	Corrosion Loss, in./yr.	Maximum Depth of Pitting, inches	Geographical Location
1	Pickled	2	.0058	----	Sandy Hook, N. J.
2*	Mill scale**	4	.0042	.009	Briston Channel, Weston-super-Mare
3*	Pickled	1	.0038	.018	Dunkerque, France
4*	Pickled	1	.0033	.017	Concarneau, France
5*	Pickled	1	.0042	.031	Le Havre, France
6*	Pickled	1	.0077	.031	Brest, France
7	Pickled	15	.0048	.075	Halifax, N. S.
8	Mill Scale	15	.0043	.68	Halifax, N. S.
9	Pickled	15	.0030	.043	Aukland, N. Z.
10	Mill Scale	15	.0033	.147	Aukland, N. Z.
11	Pickled	15	.0024	.065	Plymouth, Eng.
12	Mill Scale	15	.0024	.156	Plymouth, Eng.
13	Pickled	15	.0036	.064	Colombo, Ceylon
14	Mill Scale	15	.0039	.240	Colombo, Ceylon
15	Machined	3	.005	----	Eastport, Me.
16*	Sandblasted	1.2	.0049	----	Gosport, Eng.
17*	Sandblasted	1.2	.0051	----	Gosport, Eng.
18*	Mill Scale	0.5	.0060	.050	Gosport, Eng.
19	Pickled	1	.0041	----	Kill-van-Kull, N. J.
20	Pickled	0.5	.0066	----	San Francisco, Calif.
21	Pickled	1	.0038	.030	Cape May, N. J.
22	Pickled	1	.005	.028	Freeport, Texas
23	Mill Scale	1	.004	.080	Freeport, Texas
24*	Pickled	5	.0060	.115	Kure Beach, N. C.

TABLE 3.12.6-1 (Cont.)

Line	Original Condition of Surface	Duration of Exposure, years	Corrosion Loss, in./yr.	Maximum Depth of Pitting, inches	Geographical Location
25*	Mill Scale	7.5	.004	.25 (Perf.)	Kure Beach, N. C.
26*	Pickled	6.5	.005	.21	Kure Beach, N. C.
27	Pickled	6.5	.002	.035	Kure Beach, N. C.
28	Pickled	6.5	.003	.091	Kure Beach, N. C.
29*	Mill Scale	7.5	.005	.094	Kure Beach, N. C.
30*	Mill Scale	7.5	.004	.063	Kure Beach, N. C.
31*	Mill Scale	4	.0035	.029	Briston Channel Weston-super-Mare
32	Mill Scale	10	.0045	.084	Kure Beach, N. C.
33	Mill Scale	31	.001	----	Dry Tortegas Island
34	Mill Scale	7.5	.0037	.098	Boston, Mass.

*0.20-0.34 Copper in alloy.

** Note: The presence of mill scale is an abnormal condition which will not apply to the platform hull; the effect of such scale is to accelerate pitting, especially in the short term.

3.12.7 INSPECTION AND MAINTENANCE AFTER CONSTRUCTION

All compartments within the platform will be accessible for inspection and maintenance. Each watertight compartment will have watertight closures in accordance with American Bureau of Shipping requirements. Inspection of structural connections, coatings, watertight closures, etc., and ultrasonic inspection of plate thicknesses for corrosion can be made from the interior. Clearance between the bottom of the platform and the ocean bottom will provide for underwater inspection or repairs. Underwater brushing equipment is available for fast and thorough cleaning of any underwater structure subject to marine fouling. An experienced operator can clean approximately 3000 sq. ft. per hour and equipment is available that provides power and air for two divers to work simultaneously. (1)

Visual inspection by a diver is the most basic underwater inspection technique; however, numerous tools are available for underwater inspection of offshore structures. Underwater photography is usually accomplished with a 35-mm camera although a special camera has been developed to take excellent pictures in zero visibility water. Underwater television systems are relatively simple, reliable and present a satisfactory picture in almost any water condition. A videotape recording can be used to provide a permanent

record of a television image. Ultrasonic thickness measuring instruments can be used to gage material losses due to erosion or corrosion. Cathodic protection potential measuring devices are available whereby an inspector can obtain readings of cathodic protection potential at any point on the structure. (2)

Underwater repairs to the platform hull are feasible if damage or cracks ever occurred in the outer skin. Numerous reports are available on research and applications of underwater repairs. Reference 3, prepared for the U. S. Navy, discusses the state of the art. Since there are numerous processes available for both underwater cutting and welding of steels, consideration must be given to a specific application when selecting the proper process. Some factors to consider in the selection of underwater cutting processes are: nature of the work, depth of water in which work is to be performed, material condition, visibility, etc. The oxygen-arc cutting process has been used quite extensively by the Navy for salvage and other operations, and by commercial organizations in repair and maintenance of offshore structures. Underwater oxygen-arc cutting normally requires less skill and training than other cutting processes and is especially well suited to cutting mild steels up to approximately three inches in

thickness.

Underwater welding can be performed in either a "wet" or "dry" environment. The term "wet welding" refers to methods whereby the welder, structure, welding torch and electrodes are exposed to the water. The most widely used underwater wet welding method is the shielded metal-arc process. This process is commonly known as "covered-electrode" or "stick-electrode" welding for above-water applications. Wet welding has been used in patching holes, repairing cracks and making attachments to ship's hulls, and in the repair of damaged underwater pipelines. Mechanical properties of welds made in water are lower than similar welds made in air because of the effects of rapid quenching. Underwater welds in mild steel have about 80 percent of the strength and 50 percent of the ductility of comparable welds in air.

The term "dry" welding refers to processes in which the welding is performed inside watertight enclosures built around the area to be repaired. Applications of this technique have been used in pipeline repairs and repair of offshore structures where damage occurred in water depths of a few hundred feet.

It is anticipated that most underwater repairs to the outer skin of the platform hull will be made using the "dry" welding method. "Wet" welding will be used only where reduced mechanical properties obtained can be tol-

erated for the type repairs involved, or where temporary patches are required until "dry" weld repairs can be accomplished. The use of a watertight enclosure will also facilitate execution of required inspections and assure integrity of the repaired areas.

References:

1. Marine Engineering/Log, Monthly Publication, July, 1971, page 66.

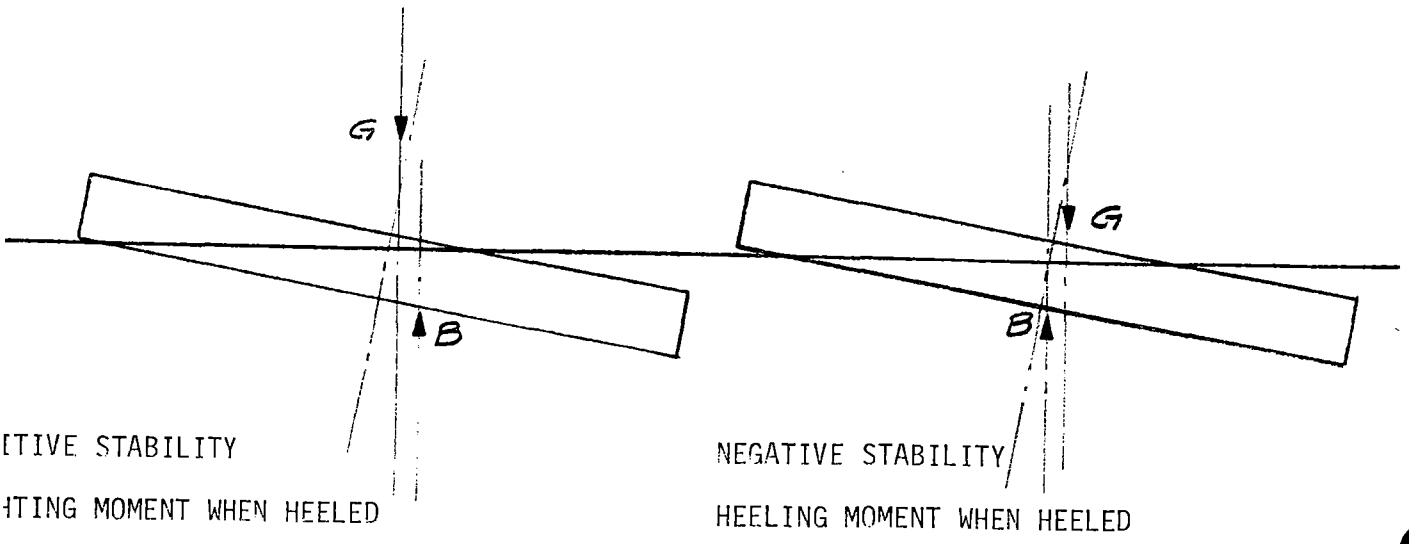
2. Ocean Industry Magazine, Monthly Publication, April, 1972, page 57.

3. Vogi, J. J., Mishler, H. W., and Randall, M.D., "Report on Underwater Cutting and Welding State of the Art", Battelle Memorial Institute, Columbus Laboratories, Columbus, Ohio.

3.12.8 STABILITY AND COMPARTMENTATION

3.12.8.1 Stability

Two resulting forces, an upward force of buoyancy and a downward force of weight, act upon a freely floating undisturbed body in a liquid. The resultant of the buoyant forces is a single force acting through the center of buoyancy, while the resultant of the weights is a single force acting through the center of gravity. A body is in equilibrium when the two resultants act through the same vertical line. When any body is in equilibrium in such a manner that it tends to right itself when its aspect to the horizontal plane is changed, stable equilibrium exists, and the condition is one of positive stability. Neutral equilibrium exists when the body remains in its displaced position. Negative stability is when the body moves farther from its original position by itself. Sketch (1) shows conditions of positive and negative stability.



SKETCH (1)

When comparing one vessel to another the index of stability normally used is the metacentric height (GM). GM is defined as the distance between the center of gravity and the metacenter (M). The metacenter is the line of intersection between a plane through the center of buoyancy and center of gravity in the upright position with a vertical plane through the center of buoyancy in the inclined position.

The metacenter is dependent on the waterplane inertia at which the vessel floats and the volume of displacement. When the plant is inclined as in sketch (2), W_1L_1 intersects WL at the center of flotation. The shaded wedges, one immersed and one emersed, are equal and the displacement has not changed. The center of buoyancy moves in a direction parallel to a line connecting the centers of gravity, g_1 and g_2, of the wedges and a distance BB_1.

If v = Volume of each wedge

V = Volume of displacement

$d\Theta$ = Angle of inclination

g_1g_2 = Distance between centroids of wedges

Then $BB_1 = \frac{v \cdot g_1g_2}{V}$

and $BM = \frac{BB_1}{\tan d\Theta} = \frac{v \cdot g_1g_2}{V \cdot \tan d\Theta}$ (Eq.1)

If y is the half-breadth of the waterline at any point x along the length of the vessel, then the area of a section through the wedge is 1/2 (y)·(y tan dθ) and the distance between centroids of the wedges is 2·2/3 (y).

Hence $v \cdot g_1 g_2 = \int 1/2\ (y) \cdot (y \tan d\theta)\ 2 \cdot 2/3\ (y)\ dx$

or $$\frac{v \cdot g_1 g_2}{\tan d\theta} = 2/3 \int_0^L y^3\ dx = I$$

where I = Moment of inertia of the waterplane,

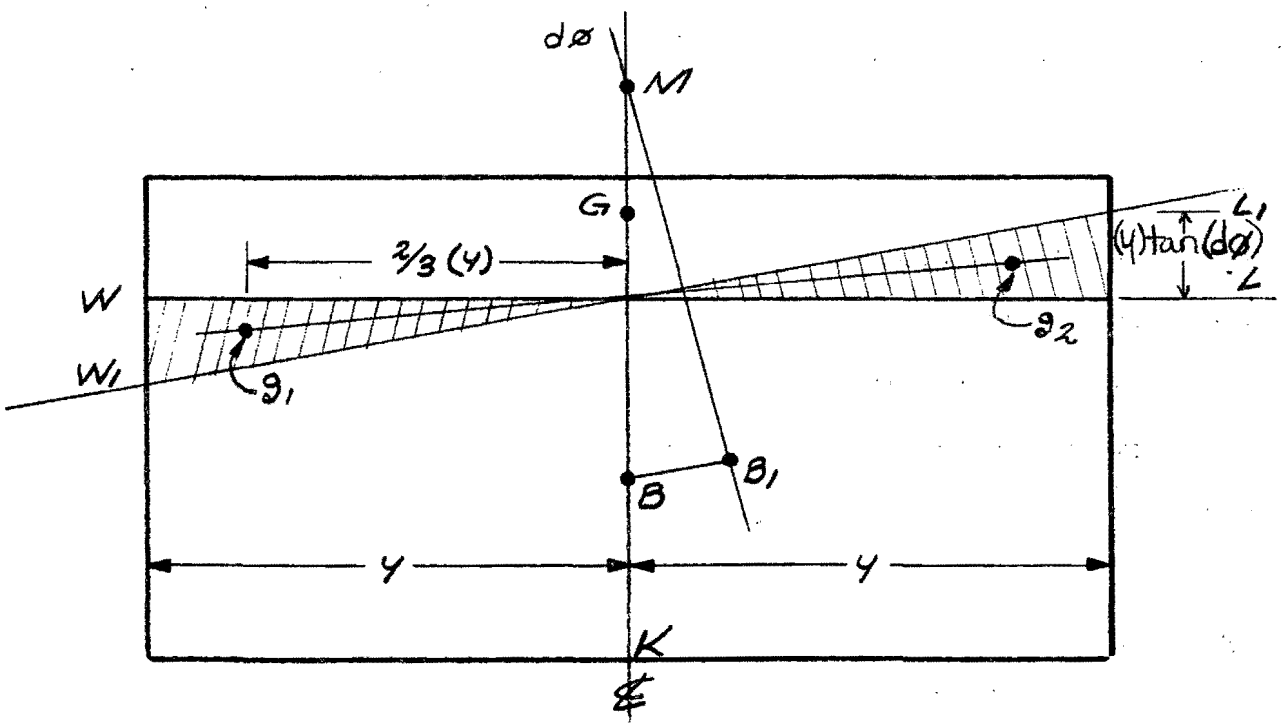
Therefore from Eq 1 above $BM = \frac{I}{V}$ (Eq. 2)

KB = Height of center of buoyancy above baseline, or bottom of vessel,

KM, height of metacenter above baseline = KB + BM

Therefore

GM = KM - KG (height of center of gravity above baseline)



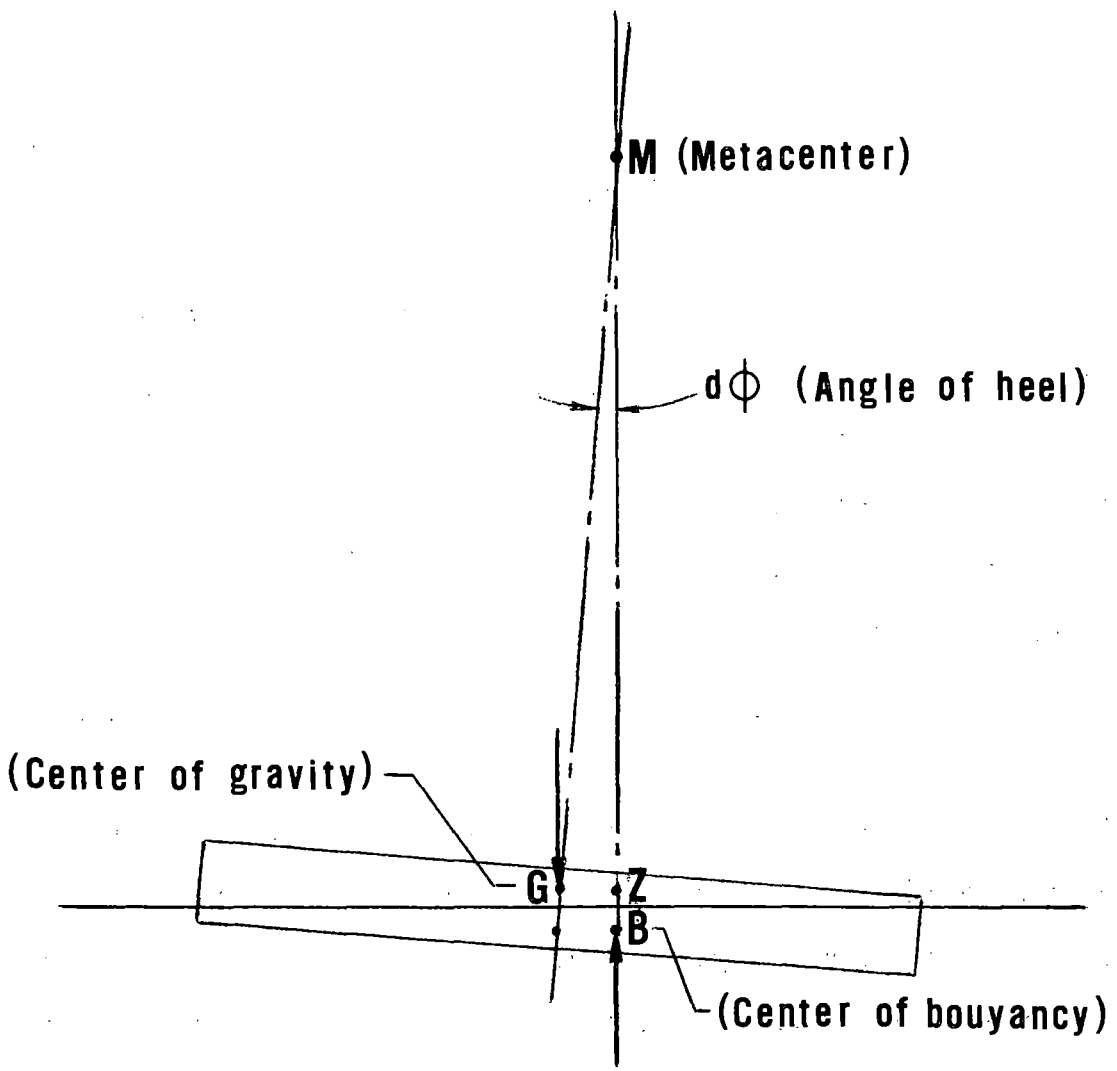
SKETCH (2)

A vessel with positive GM will tend to float upright and resist initial inclining forces. A vessel with negative GM will cease to float upright when disturbed by an outside force. The GM of the Floating Nuclear Plant is such that it will remain extremely stable under all operating and design basis conditions. Table 3.12.8-1 gives a comparison of the transverse metacentric height of the plant along with some typical transverse metacentric heights of various types of ships in the operating condition.

TABLE 3.12.8-1

Type of Vessel	Approximate Transverse Metacentric Height in Feet
Floating Nuclear Plant	345
75,000 ton displacement tanker	15
300,000 ton displacement tanker	25
Cargo & passenger ships	4 to 8

The GM is normally used in determining the moment required to incline a vessel from its upright position. Sketch (3) shows the metacenter M, center of gravity and buoyancy, G and B_1, and angle of heel for a heeled vessel. The center of buoyancy has moved off the vessel's centerline as a result of the inclination, and the lines along which the resultants of weight and buoyancy act are separated by a distance GZ, the righting arm.


M (Metacenter)
dϕ (Angle of heel)
(Center of gravity)
G
Z
B
(Center of bouyancy)

SKETCH (3)

The vertical line through the center of buoyancy intersects the centerline at the metacenter. Knowing the metacentric height, the righting arm for small angles of heel up to approximately 3 degrees can be readily calculated with sufficient accuracy by the formula:

$GZ = GM \sin\theta$ (Eq. 3)

The moment required to heel a vessel from equilibrium conditions is equal to the moment of weight and buoyancy, or W.GZ when the vessel is heeled to some angle.

Since $W \cdot GZ = W \cdot GM \sin\theta$ from Eq. 3

Moment to heel one degree = $W \cdot GM \sin 1^{o}$ (Eq. 4)

Within the range of inclinations in which the metacenter is stationary, the angle of heel produced by a given moment can be found by dividing the moment by the moment to heel one degree.

The change in draft can be determined by substituting one inch divided by the length or width or the platform in inches for $\sin 1^{o}$.

Moment to trim one inch = $\frac{W \cdot GM}{12\ L}$ (Eq. 5)

The Floating Nuclear Plant weighs approximately 140,000 tons and has a transverse metacentric height of approximately 345 feet.

Therefore:

Moment to heel one degree = $W \cdot GM \sin 1^{o}$

$= 140000 \cdot 345 \cdot 0.0175$

$= 845{,}000$ ft.-tons

Moment required to trim the Floating Nuclear Plant one inch in the longitudinal direction is:

GM_L = 395 feet

L = 400.25 feet

Moment to trim one inch = $\frac{140000 \cdot 395}{12 \cdot 400.25}$

= 11,500 ft.-tons

A plot of the righting arm versus angle of inclination is the most satisfactory means of presenting a complete picture of stability. Figure 3.12-14 shows the righting arm curve for the Floating Nuclear Plant. From the righting arm curve: the righting arm at any inclination, GM, angle of maximum righting moment, and range of stability can be determined.

Many variable characteristics of the plant are useful in predicting and computing flotation situations or stability information. The displacement, height of metacenter, position of the center of buoyancy, tons per inch immersion, tank capacities, and flooding effects are examples of such useful date. Since these variables are functions of the form of the plant, they may be tabulated or shown on curves and made available to the customer in a stability and loading data booklet. The information to be provided is as follows:

1. Draft and stability characteristics for the operating and design bases conditions.

2. Table of tank capacities.

3. Compartmentation diagram.

3.12.8.2 Compartmentation

The platform is divided into watertight compartments as shown on the Compartmentation Diagram Figure 3.12-1. The compartments are sized so that:

1. Flooding of any two adjacent compartments does not cause any part of the top deck (40-foot level) to sink below the water level.

2. Flooding of any two adjacent compartments does not cause any part of the platform bottom to come in contact with the ocean bottom inside the breakwater. This criterion is set to prevent "grounding" in the unlikely event of extensive flooding.

3. Flooding of any one compartment does not change the angle of trim at any point more than one degree. This criterion is set so the plant can continue to operate in the event that one compartment is flooded. Trim tanks approximately 80 feet by 38 feet by 40 feet deep are provided at each corner of the

platform to adjust trim or heel during operating and maintenance.

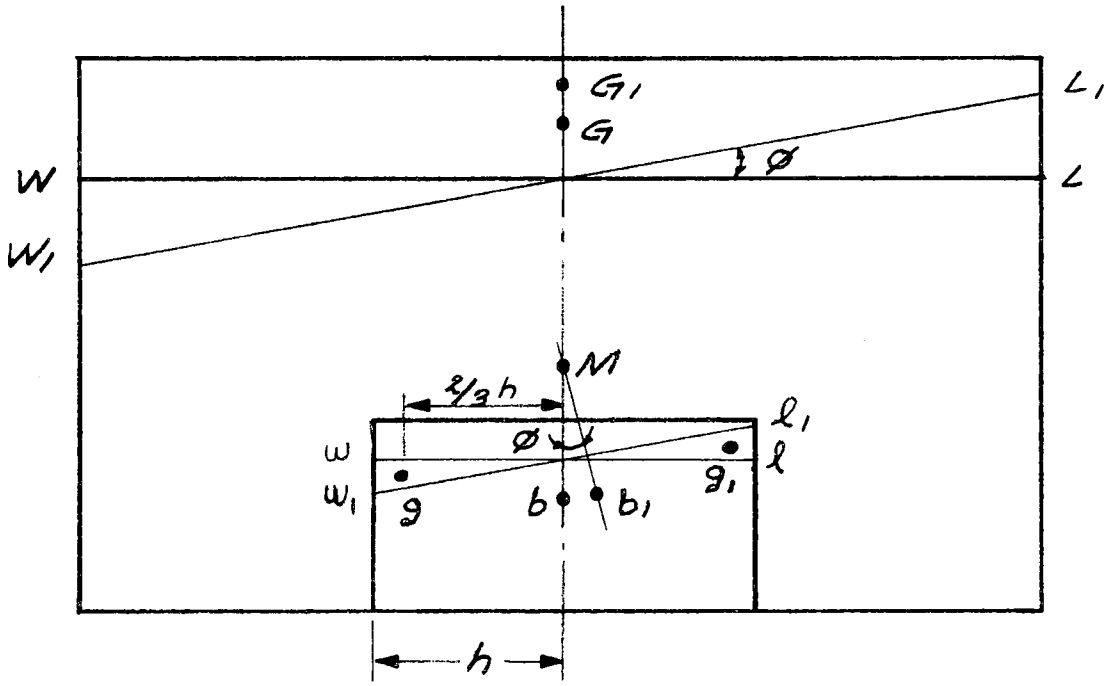
4. The containment boundary and the safeguard areas are bounded by two or more watertight bulkheads and a watertight inner bottom at the 7'-6" level to provide redundancy in the event of flooding from the exterior.

The Navships Hull Characteristics Computer Program, utilizing the lost buoyancy method, computes the results of flooding any one compartment or the combination of any two adjacent compartments. The computer prints out: the righting arm, transverse, vertical and longitudinal center of buoyancies, draft, and trim for any angle of heel. From the print out: the trim, heel angle, and draft at the corners of the platform are computed for the flooding condition in addition to plotting a righting arm curve. As a check the lost buoyancy method outlined in reference (1) was used. The U. S. Navy, U. S. Coast Guard, and most major shipyards utilize this computer code for damage stability calculations. Figure 3.12-15 shows the flooding condition for two adjacent compartments that results in the deepest draft at any corner. The trim tank was assumed completely empty before flooding.

3.12.8.3 Free Surface

When a liquid partially fills a tank, it has free surface. When the vessel

heels, the free surface tends to remain horizontal and the liquid moves to the low side of the tank adding to the inclining moment. To better understand free surface effect, consider the movement of the liquid as changing the apparent height of the center of gravity, see Sketch (4). From reference (2), pages 214-216, the result of the derivation based on small angles of inclination is a reduction in GM equal to i/V where i is the transverse



SKETCH (4)

moment of inertia of the water surface about its longitudinal center line and V is the volume of displacement. The Floating Nuclear Plant has a $GM_{transverse}$ and $GM_{longitudinal}$ approximately equal to 345 feet and 395

feet respectively. The reduction in $GM_{transverse}$ and $GM_{longitudinal}$ due to free surface of all tanks on board is less than 5 feet and 10 feet respectively; therefore the effect of free surface on the Floating Nuclear Plant is not significant. The free surface effect from post-accident flooding of the containment would reduce the GM approximately 2.5 feet or less than one percent.

References

1. Comstock, John P., Ed. Principles of Naval Architecture, published by The Society of Naval Architects and Marine Engineers, 1967.

2. Gillmer, Thomas C., Fundamentals of Construction and Stability of Naval Ships, published by U. S. Naval Institute, 1956.

APPENDIX 3A

DESCRIPTION OF CHAPTER 3 COMPUTER PROGRAMS

3A.1 INTRODUCTION

The following sections present outlines of computer programs used in the analyses of structures, systems, and components as described briefly in various sections of Chapter 3.

3A.2 DYNAMIC STRESS ANALYSIS OF AXISYMMETRIC STRUCTURES UNDER ARBITRARY LOADING

This program employs the finite element method for the dynamic analysis of complex axisymmetric structures subjected to any arbitrary static or dynamic loading or base acceleration. The three-dimensional axisymmetric continuum is represented either as an axisymmetric thin shell or as a solid of revolution or as a combination of both. The axisymmetric shell is discretized as a series of frustrums of cones and the solid of revolution as triangular or quadrilateral "toroids" connected at their nodal point circles.

Hamilton's variational principle is used to derive the equations of motion, and the diagonal mass matrix formulations are adopted to economize the

computer storage and time. These equations of motion are solved numerically through the time domain either by direct integration or by mode superposition. In both cases the numerical scheme adopts to the step-by-step integration procedure. For an earthquake analysis, the method of modal analysis by response spectrum can be called for.

The program has the following features:

1. It can solve five loading cases: dead load, arbitrary static load, arbitrary dynamic load acting on the structure, horizontal and vertical components of earthquake acceleration record applied at the base of the finite element model.

2. Any arbitrary loading is approximated by a Fourier cosine series with a finite number of terms. The structural response corresponding to each Fourier component is solved and then added up to obtain the total response.

3. Three dynamic response subroutines are available in the program, namely, direct integration, mode superposition, and response spectrum subroutines.

4. Either isotropic or anisotropic material properties can be specified for each shell or solid element, and variable thicknesses of shell

elements prescribed at different nodal point circles.

Modification by the Offshore Power Systems to the original program has added the following features:

1. Fourier sine series approximation to an arbitrary loading, and a combination of Fourier sine and cosine series.

2. Capabilities of prescribing abrupt changes in shell thicknesses and modifying the stiffness matrix to facilitate modeling of stiffening rings as either short thick shell elements or fictitious shell elements with smeared out stiffeners.

3A.3 WESTDYN

WESTDYN is a special purpose program designed for the static and dynamic solution of redundant piping systems with arbitrary loads and boundary conditions. It computes, at any point in the piping system, the stresses, forces, moments, translations, and rotations which result from the imposed anchor or junction loads in any combination of the three orthogonal axes. The section properties have been specialized to piping cross sections plus the addition of curved members or elbows. Valves may also be represented as stiffener members. The piping system may contain a number of sections, a section being defined as a sequence of straight and/or curved

members lying between two network points. A network point is 1) a junction of two or more pipes, 2) an anchor or any point at which motion is prescribed, or 3) any arbitrary point.

Any location in the system may sustain prescribed loads or may be subject to elastic constraint in any of its six degrees of freedom. For example, hangers may be arbitrarily spaced along a section and may be of the rigid, flexible, or constant force type.

The response to seismic excitation is determined by using normal mode techniques with a lumped mass system. The maximum spectral acceleration is applied for each mode at its corresponding frequency from response spectra. A basic assumption is that the maximum modal excitation of each mode occurs simultaneously. The modal participations are then summed.

3A-4 WESTDYN-2

This program is a slightly modified version of the WESTDYN program. The program accepts (does not calculate) time-history displacement vectors as boundary conditions and proceeds to an usual WESTDYN static solution. In addition to the usual stress solution, the program also calculates axial stress, shear stress and stress intensity.

3A.5 FIXFM

FIXFM is a digital computer program which determines the time-history response of a three-dimensional structure excited by an internal forcing function. FIXFM accepts (input) normalized mode shapes, natural frequencies, forcing functions and an initial deflection vector. The program sets up the modal differential equations of motion. The modal differential equations are then solved numerically by a predictor-corrector technique of numerical integration. The modal contributions are then summed at various nodal or mass points throughout the structure to get the actual time-history response.

3A.6 SATAN

The SATAN (System Accident and Transient Analysis of Nuclear Plants) computer code performs a comprehensive space-time dependent analysis of a LOCA and is designed to treat all phases of the accident. The three major phases are:

1. A subcooled phase where the rapidly changing pressure gradients in the subcooled fluid exert an influence upon the RCS internals and support structures.

2. A two-phase depressurization phase.

3. A refill-phase where the emergency core cooling system floods the core.

The code employs a one-dimensional analysis in which the entire reactor coolant loop system is divided into control volumes. The fluid properties are considered uniform and thermodynamic equilibrium is assumed in each element. Pump characteristics, pump coast-down and cavitation, are incorporated in the code.

The fundamental equations of conservation of mass, energy, and momentum are applied for single and two phase system in order to calculate the main dependent variables: density, internal energy and mass flow rate. A set of auxiliary dependent variables are defined by the system governing algebraic equations.

A critical flow calculation for subcooled, two phase, or superheated break flow is incorporated in the analyses. The subcooled break flow is calculated using a modified version of the Zaloudek correlation. The results are consistent with test results and do not produce a discontinuity between the subcooled and two-phase break flows. The two-phase break flow is calculated by using the Moody two-phase correlation.

The SATAN Code is equipped to analyze a maximum of two loops plus "special" elements such as the reactor core, pressurizer and accumulators.

In the analysis, the RCS is divided into a finite number of control volumes. Separate groups of control volumes are assigned to simulate the broken and unbroken loops and two control volumes are used to describe the coolant in the core region. The current multi-control volume analysis, particularly with two control volumes in the core, permits execution of a detailed parametric survey of the important phenomena affecting the blowdown process.

The adequacy of the SATAN code to predict the hydraulic behavior during blowdown of the reactor coolant system during a loss-of-coolant accident has been verified by comparing the SATAN results of loss-of-coolant accidents of various sizes and locations with experiments performed at the Idaho and Battelle Northwest facilities.

3A.7 STHRUST

STHRUST code computes blowdown hydraulic loads on the primary loop components from the blowdown information calculated by SATAN code. The entire primary system is represented by the same two-loop model employed in the SATAN blowdown calculation.

The force nodes are selected along the two-loop geometric model of a

reactor plant where the vector forces and their components in a global coordinate system are calculated. Each force node is associated with a control volume which may contain one or two blowdown (SATAN) control volumes depending on the location of the force node in the system. Each force control volume in turn, has one or two associated apertures (flow area). The force is calculated at each aperture.

The major input information required for the code are:

1. Blowdown hydraulic information which is read directly from the SATAN result tape.

2. The orientation of the force node in the system which is input as three projection coefficients along the three coordinate axes of the global coordinate system.

3A.8 STRUDL

STRUDL, part of the ICES civil engineering computer system, is widely used for the analysis and design of structures. It is applicable to linear elastic two-and three-dimensional frame or truss structures, employs the stiffness formulation, and is valid only for small displacements.

Structure geometry, topology, and element orientation and cross-section

properties are described in free format. Member and support joint releases such as pin and rollers, are specified. Otherwise, six restraint components are assumed at each end of each member and at each support joint.

The STRUDL system performs structural stability and equilibrium checks during the solution process and prints error messages if these conditions are violated. However, the system cannot detect geometry or topology errors. Type, location, and magnitude of applied loads or displacements are specified for any number of loading conditions. These can be combined as desired during the solution process.

One important feature of STRUDL is that any desired changes, deletions, or additions can be made to the structure model during the solution process. This produces results for a number of structure configurations, each with any number of loading conditions.

Printed output content, specified by input commands, includes member forces and distortions, joint displacements, support joint reactions, and member stresses. Westinghouse has expanded its output capabilities to include selective punched card data that is used as input in the THESSE program.

3A.9 THESSE

The THESSE computer program was developed by Westinghouse to accomplish

RCS equipment support structures analyses and evaluation. Two versions are used: 1) one for Normal Upset and Emergency Condition loading using AISC-69 allowable stress equations and 2) the other for Faulted Condition loading where LOCA loads are read in time-history form and ultimate stress equations are used.

Input is by punched cards (except that time history LOCA loads are read from tape) and includes:

1. Appropriate problem labels and dimension statements.

2. Six components of forces on the structure for each of Thermal, Weight, Pressure, and OBE or DBE and LOCA loads.

3. Member cross section dimensions, type code, and material.

4. 6 x 6 member influence coefficient array for each end of each member.

Loads on the structure are combined, transformed to the structure-coordinate system, and multiplied by member influence coefficients. The resulting member forces are then used with member properties in stress and interaction equations to determine the adequacy of each member in the structure.

All input data are included in the printed output along with maximum member force and stresses.

3A.10 STRUCTURAL ANALYSIS PROGRAM (STRAP)

The Newport News Shipbuilding and Dry Dock Co. Technical Services Structural Analysis Program (STRAP) provides the capability of static and dynamic analysis of linear, three-dimensional structures to which arbitrary mechanical and thermal loads and displacement boundary conditions may be applied. The dynamic portion of STRAP contains shock, harmonic, and transient analyses. The individual analyses are common in that each used the model analysis method. Structural elements incorporated into the program include beam members and triangular and quadrilateral plate members which have both membrane and plate bending stiffnesses. Basic assumptions which apply to each element are small deformations, linear elastic behavior, plane sections remain plane after deformation, and isotopic, homogeneous material properties. For the static portion of STRAP static loads may be applied in the form of concentrated joint forces and moments, thermal forces, normal pressure, and concentrated, uniform, or linear beam member loads. Program output includes joint displacements, beam forces and moments, beam stresses, triangle and quadrilateral plate stresses and for the dynamic analysis characteristic frequencies, phase angles, or mode shapes as appropriate. The computer program was developed at Newport News Shipbuilding and Dry Dock Co. (NNS & DD Co.) for use in the structural analysis of U. S. Naval vessels under contract to NNS & DD Co. and has been used successfully in the structural analysis of several naval vessels.

The static analysis performed by STRAP is based on the direct stiffness method of analysis. In this method of analysis, the structure is approximated by a model that is an assembly of discrete structural elements connected at a finite number of points. A matrix relating the nodal displacements and the associated nodal forces is derived in a coordinate system unique to each element. Once the element stiffness matrix is calculated, it is transformed into the global or structural axis system. The structural stiffness matrix is then formed by lumping or adding the element stiffnesses at the nodal points. A force matrix composed of externally applied nodal forces is also formed. After the structural stiffness matrix $[K]$ and the force matrix $\{F\}$ have been formed, the equation

$$[K]\{\Delta\} = \{F\}$$

can be solved for the nodal displacements $\{\Delta\}$, resulting in six displacement components at each node (referred to the global axis system). With the nodal displacements now defined, the internal stresses are calculated for each element in the structure.

The dynamic analyses performed by STRAP are based on the lumped mass method of model analysis. In this method of analysis the items whose motion is considered important are approximated by discrete masses which are interconnected by an assembly of discrete structural elements. The matrix relating mass displacements with the associated structure is derived from

istics of a wide variety of engineering systems. Complex heat transfer problems have been solved to determine temperature distributions in structures of nuclear submarines, aircraft carriers and cryogenic tankers.

3A.12 SUPPLEMENTAL COMPUTER PROGRAMS

Computer programs described in this section are available for use by the Offshore Power Systems design engineers. These programs are used as an alternate to one of those described in the previous sections of this appendix, or they are used as a checking procedure to make a comparison or to verify another computer run.

3A.12.1 STASYS-2

The STASYS-2 computer program is a proprietary Westinghouse program (WCAP 7651) developed by Dr. J. Swanson. It provides a procedure for the solution of a large class of one, two, and three-dimensional structural analysis problems. Included among the capabilities of the program are static elastic and plastic analysis, steady state and transient heat transfer, dynamic mode shape analysis, materially linear and non-linear dynamic analysis, and static and dynamic large deflection analysis.

The method of analysis used in the program is based on the finite element idealization of the structure. The matrix displacement method is used

for each finite element. The governing equations for each element are assembled into a system of simultaneous linear equations for the entire structure. A "wave front" direct solution technique is employed to give accurate results in a minimum of computer time. The main feature of this technique is that as the total system of equations is assembled from the element matrices arranged in a sequence, the equations for a nodal point which occurs for the last time are algebraically solved in terms of the remaining unknowns and eliminated from the assembled matrix. The equations for a nodal point which occurs for the first time are simply added to the assembled matrix. Thus, the assembled matrix grows and shrinks in size as nodal points made their first and last appearance in the element specifications. The varying size of the assembled matrix is the wave front size which can be kept to a minimum by properly arranging the input data.

The library of finite elements includes two or three-dimensional spars, beams, pipes, cables, plates, axisymmetric shells, solids, friction interface elements, springs, masses, dampers, thermal conductors, hydraulic conductors, convection elements and radiation elements.

Current capacity of the program allows approximately 15,000 degrees of freedom and a wave front size of 500 for static and thermal analyses. For dynamic mode shape problems, the limitation is 85 dynamic degrees of freedom with option for specifying higher degrees of freedom. For dynamic time history problems, the limitations are essentially the same

a reduction of the stiffness matrix used in the static analysis.

The limitations of the static problem which STRAP may analyze are a maximum of 2000 nodal points (each having 6 degrees of freedom), a maximum bandwidth of 500 coefficients in the resulting stiffness matrix, a maximum of 2000 different element shapes and orientations, a maximum of 18 elements connected to any given node, and a maximum of 30 nodes connected to any given node.

The number of elements is limited only by the amount of peripheral storage available on the computer for storage of element stiffness matrices.

The largest dynamic problem which STRAP can analyze is one of 100 degrees of freedom with non-critical damping.

3A.11 GENERALIZED NUMERICAL ANALYSIS OF THERMAL SYSTEMS (GNATS)

The Newport News Shipbuilding and Dry Dock Co. Technical Services Generalized Numerical Analysis of Thermal Systems (GNATS) computer program provides the capability to determine the temperature distribution in complex structures.

The Finite Difference Method is employed in this computer program by subdividing the structure into small discrete regions where each region is represented by a node. An equation is then written for the unknown temper-

ature at each node in terms of the unknown temperatures at the surrounding nodes or known temperatures at the boundary of the problem. The modes of heat transfer allowable are conduction, radiation, and convection, with the latter two modes being non-linearly related to temperature. Thus, the system of equations to be solved requires a double solution process. Initial estimates are made at the temperatures and a linearized approximation to the non-linear system follows. This is solved and the solution used to relinearize the system which is solved in turn. This process generally converges in around seven iterations for a steady-state problem. The code also solves transient problems where two different solution methods are available, explicit or implicit. The first method computes its own time increment depending on the changing problems, while the implicit method uses increments specified by the user but requires at each step the double solution process described for the steady-state problem. The program can accommodate a wide variety of boundary conditions, variations in physical properties and fluid flow conditions. A simplified data input procedure for any nodal network is incorporated in the program which allows a minimum of data handling. Also, data error check indicating input errors is included in the program to permit rapid elimination of errors. The program can accommodate systems with up to 3000 nodes, with a maximum of 80 terms per equation; with only slight modification, larger systems can be handled.

The GNATS Code has been utilized to evaluate the heat transfer character-

as for static problems.

3A.12.2 STATIC AND DYNAMIC ANALYSIS OF FRAME STRUCTURES: COMPUTER PROGRAMS SAND, MODAL AND SPECTA

This Westinghouse document (WCAP-7351) by R. S. Orr covers three computer programs which are described separately in the following:

Program SAND is developed from a program originally written by the Jet Propulsion Laboratory. It is a matrix structural analysis program for the static and dynamic analysis of two or three-dimensional frame structures. The program sets up the stiffness matrix for the structure. Then, this matrix is solved in combination with an applied force, temperature, or mass matrix to determine static deflections and stress resultants, or natural frequencies and mode shapes. The program has a maximum capacity of 170 degrees of freedom.

Program MODAL is a computer program for the time history analysis of structures whose fundamental modes have been determined by program SAND. Input to the program consists of structural data and mode shapes determined by SAND and a forcing function which may be either a ground acceleration or a force applied at any one of the joints. Output from the program gives the time history of deflections and accelerations at all joints; acceleration response spectra at selected joints, and member forces at certain intermediate times.

Program SPECTA is developed to carry out a response spectrum analysis based on the first six frequencies and mode shapes calculated by SAND. SPECTA computes the modal participation and effective mass. The user provides input giving the value of the response spectrum at each of the first six frequencies. Output consists of deflections, accelerations and member forces in each mode.

APPENDIX 3B

CODES AND STANDARDS

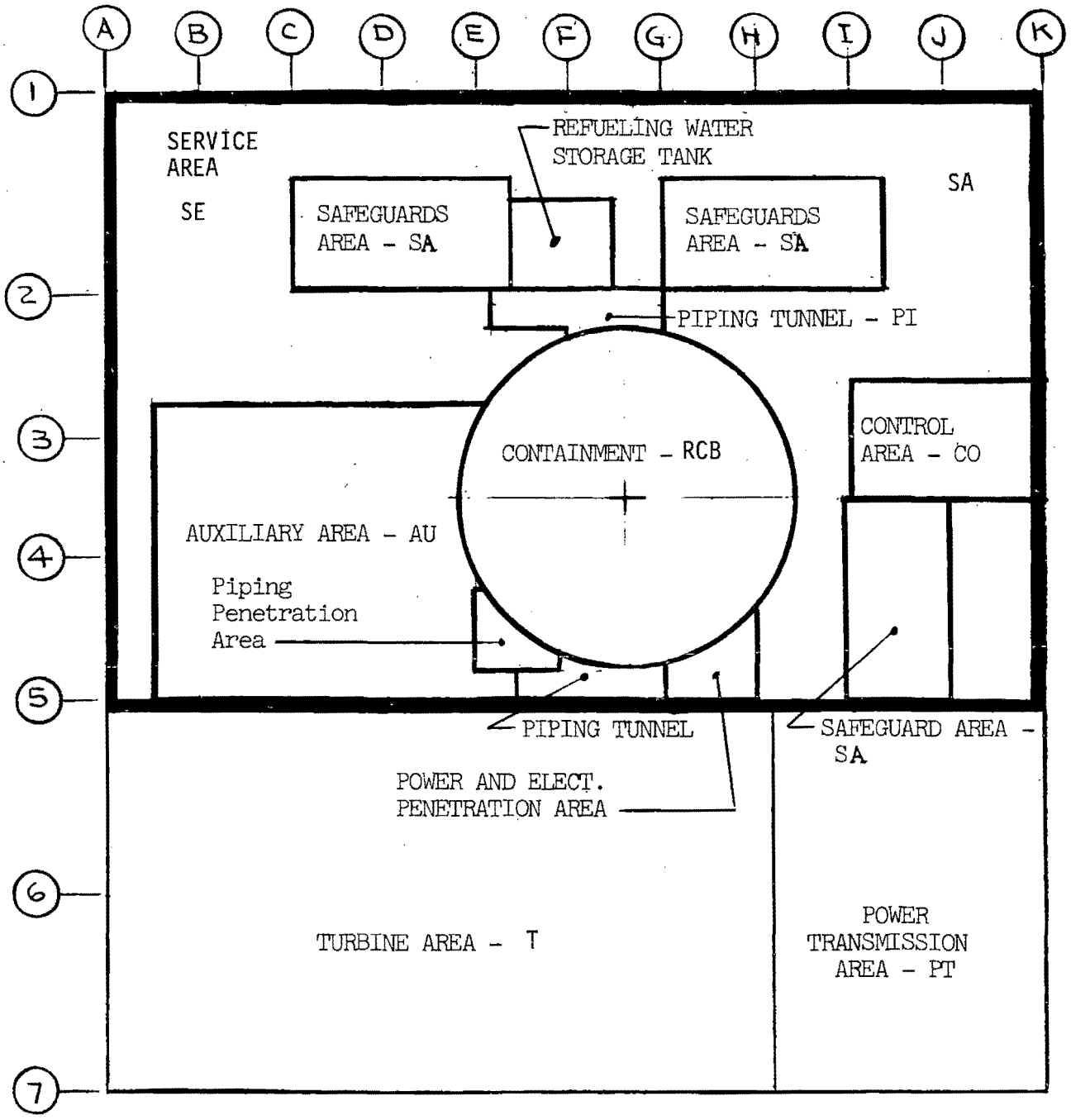
The Floating Nuclear Plants' structures, systems and components will be designed, fabricated, constructed, tested and inspected to standards which accord with the importance of the function to be performed. These standards are incorporated in 10 CFR 50.55a, Codes and Standards, and throughout the Plant Design Report.

As a minimum the Floating Nuclear Plants' structures, systems and components will meet the requirement of codes and standards in effect at a fixed time interval prior to issuance of a manufacturing license. For the pressure vessel the fixed time interval is 18 months, for pumps, valves and protection systems the time interval is 12 months and for piping and inservice inspection requirements the time interval is six months. If the pressure vessel, pumps, valves or piping are ordered after the issuance of a manufacturing license, the codes and standards complied with, will be those in existence on the date of order.

In general, new codes or standards, or revisions and addenda to existing codes or standards and regulatory requirements will be evaluated every six months, and when appropriate and feasible, will be incorporated into the standard design of the Floating Nuclear Plant. These new or revised codes and standards will be applied to all plants in design, fabrication or testing to the extent they can be applied without resulting in unusual

difficulties without a compensating increase in the level of quality and safety. However, those items imperative to safety will be incorporated immediately.

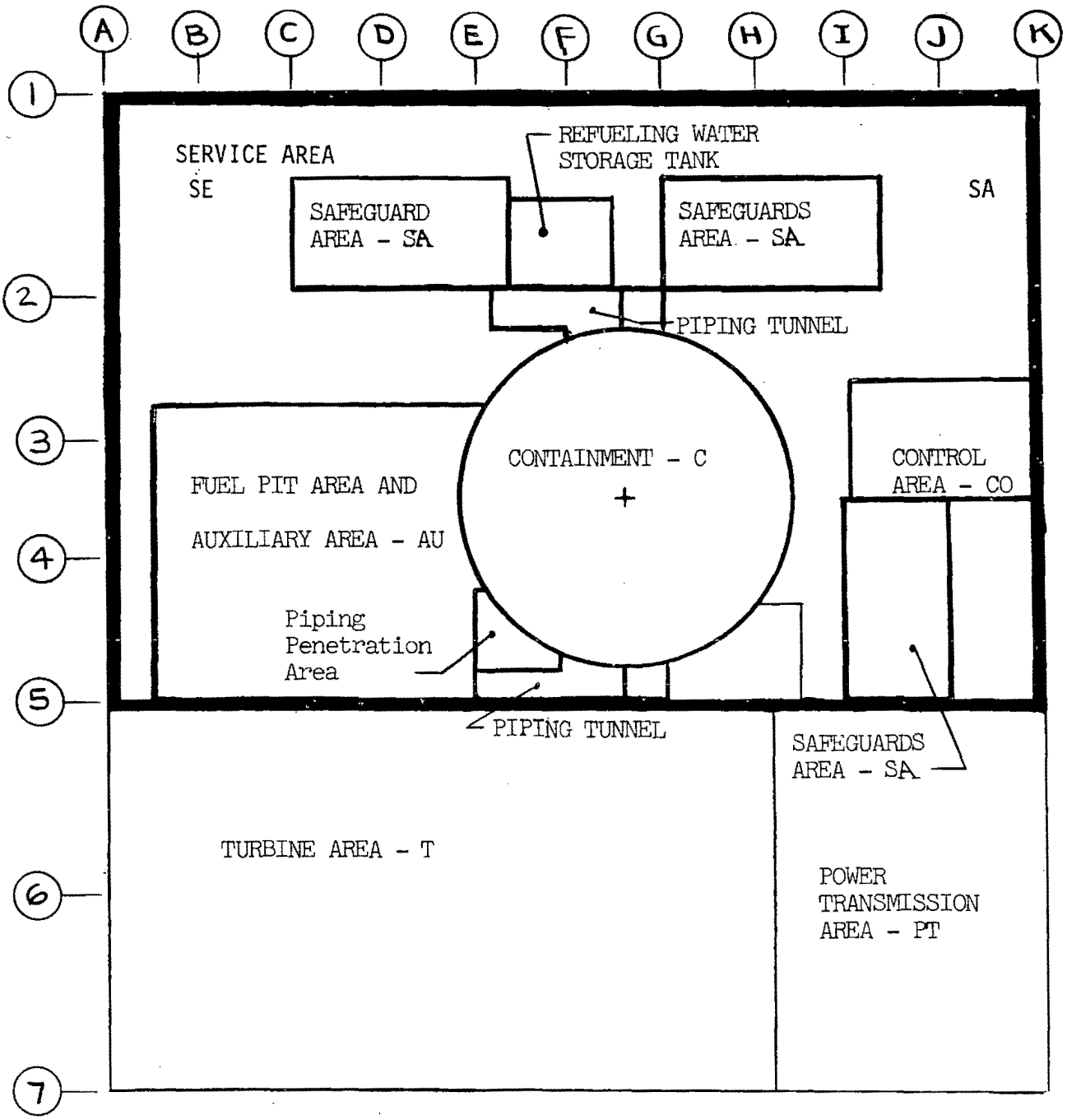
Codes and standards applicable to the design of the Floating Nuclear Plant will be promulgated in a way which gives guidance to each successive level of design, fabrication and testing and facilitates the implementation of new codes and standards.



Note: STRUCTURES WITHIN HEAVY LINES ARE CATEGORY I

CATEGORY I STRUCTURE AT 40'0 LEVEL

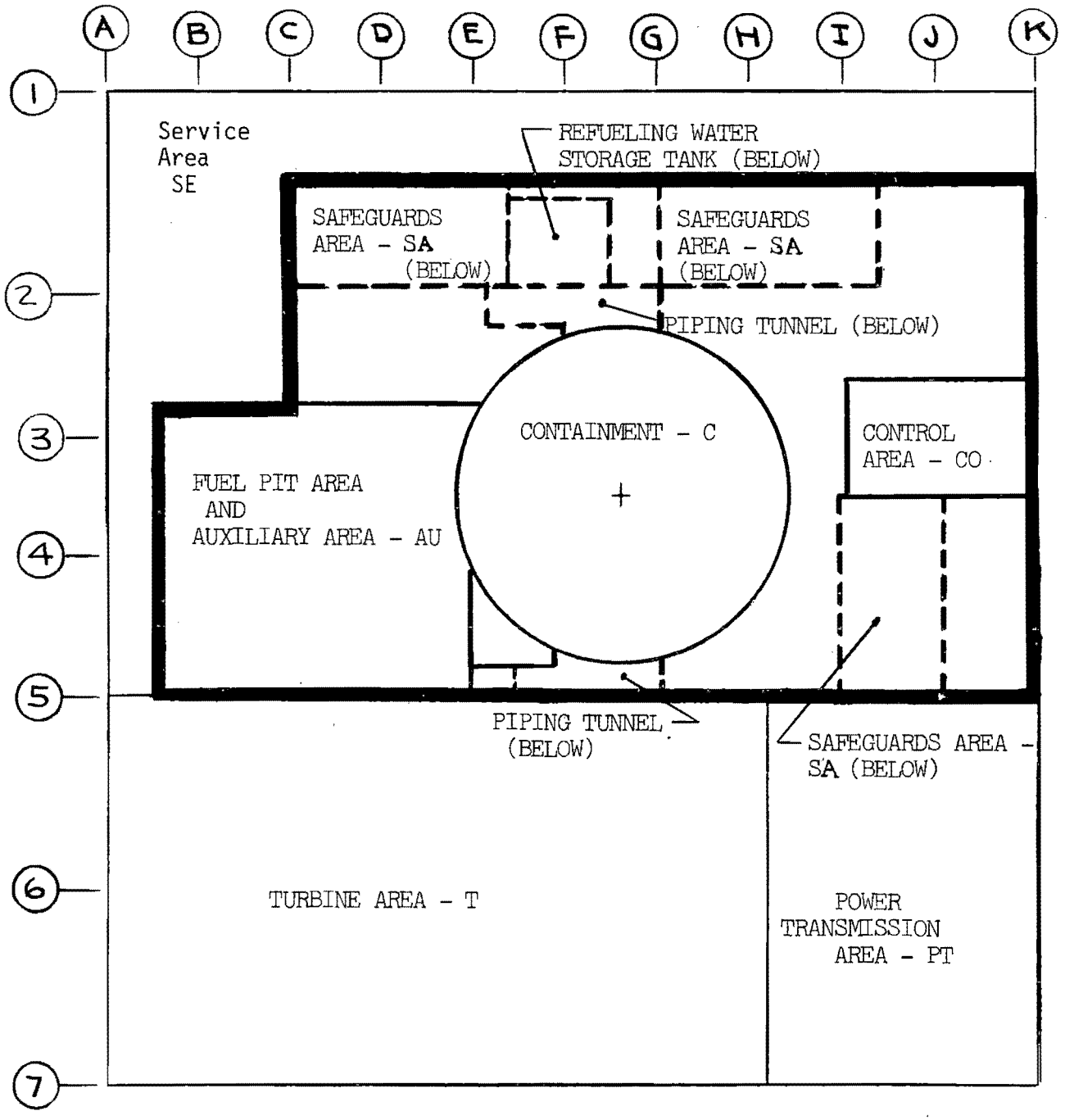
11-1-72 Figure 3.2-1



NOTE: STRUCTURES WITHIN HEAVY LINES ARE CATEGORY I.

CATEGORY I STRUCTURE AT
58'0 LEVEL

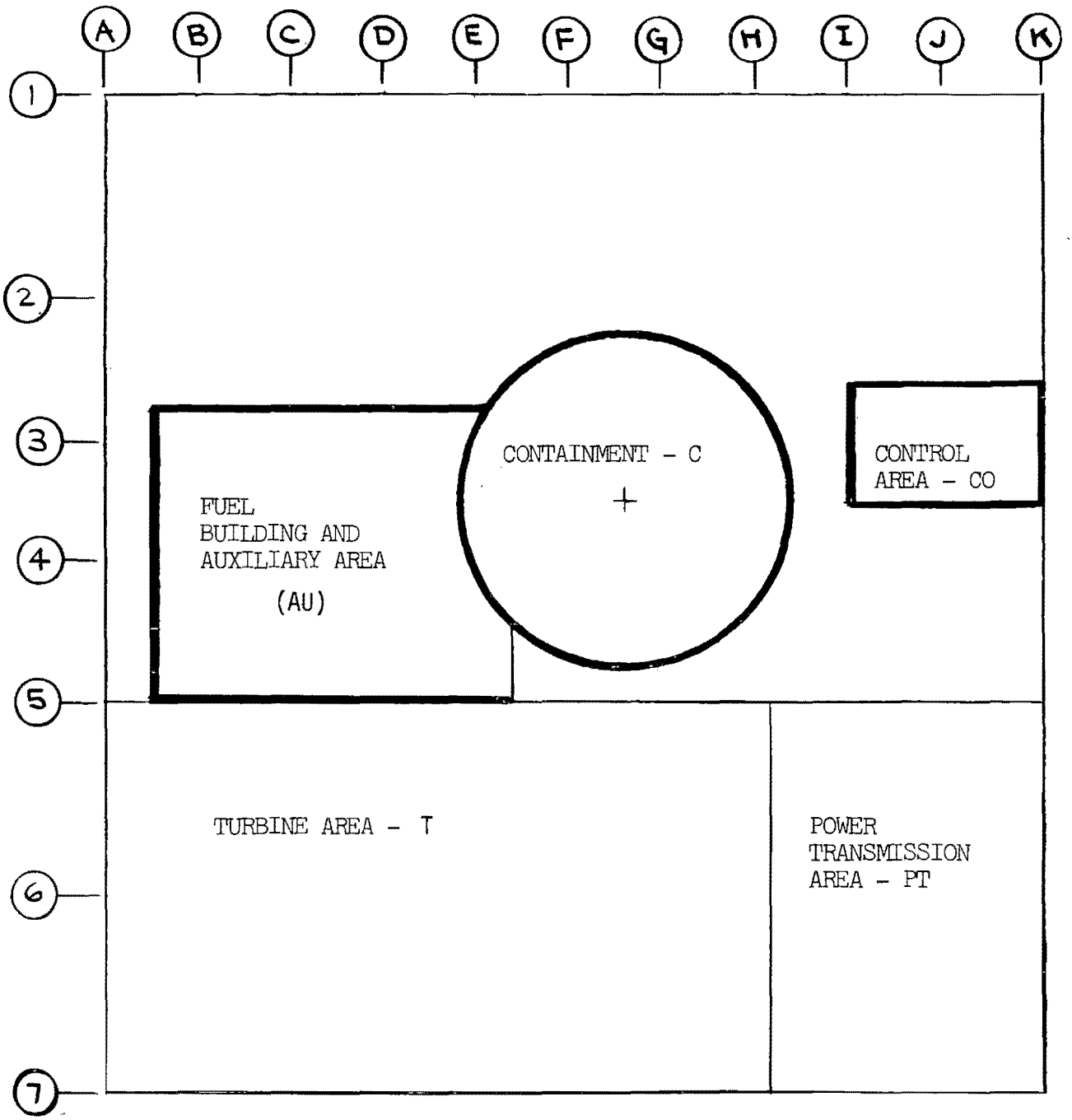
11-1-72 Figure 3.2-2



NOTE: STRUCTURES WITHIN HEAVY LINES ARE CATEGORY I.

CATEGORY I STRUCTURE AT
76'0 LEVEL

11-1-72 Figure 3.2-3

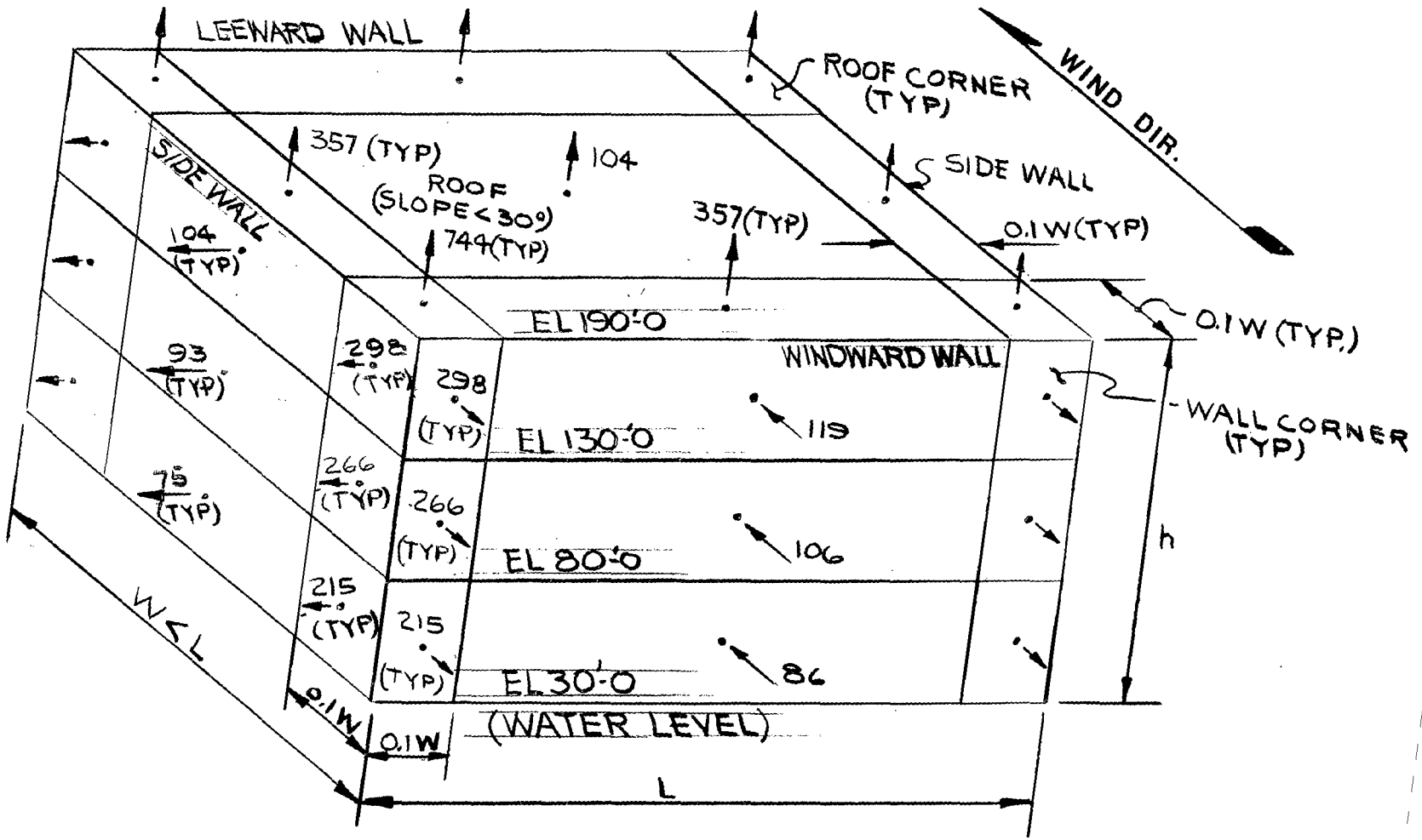


NOTE: STRUCTURES WITHIN HEAVY LINES ARE CATEGORY I

CATEGORY I STRUCTURE AT
94'0 LEVEL

11-1-72 Figure 3.2-4

NOTE: ALL LOADS ARE IN POUNDS PER SQUARE FOOT



OPERATING BASIS WIND LOADS

11-1-72

Figure 3.3-1

Figure No. 3.3-1 (Con't)

NOTES:

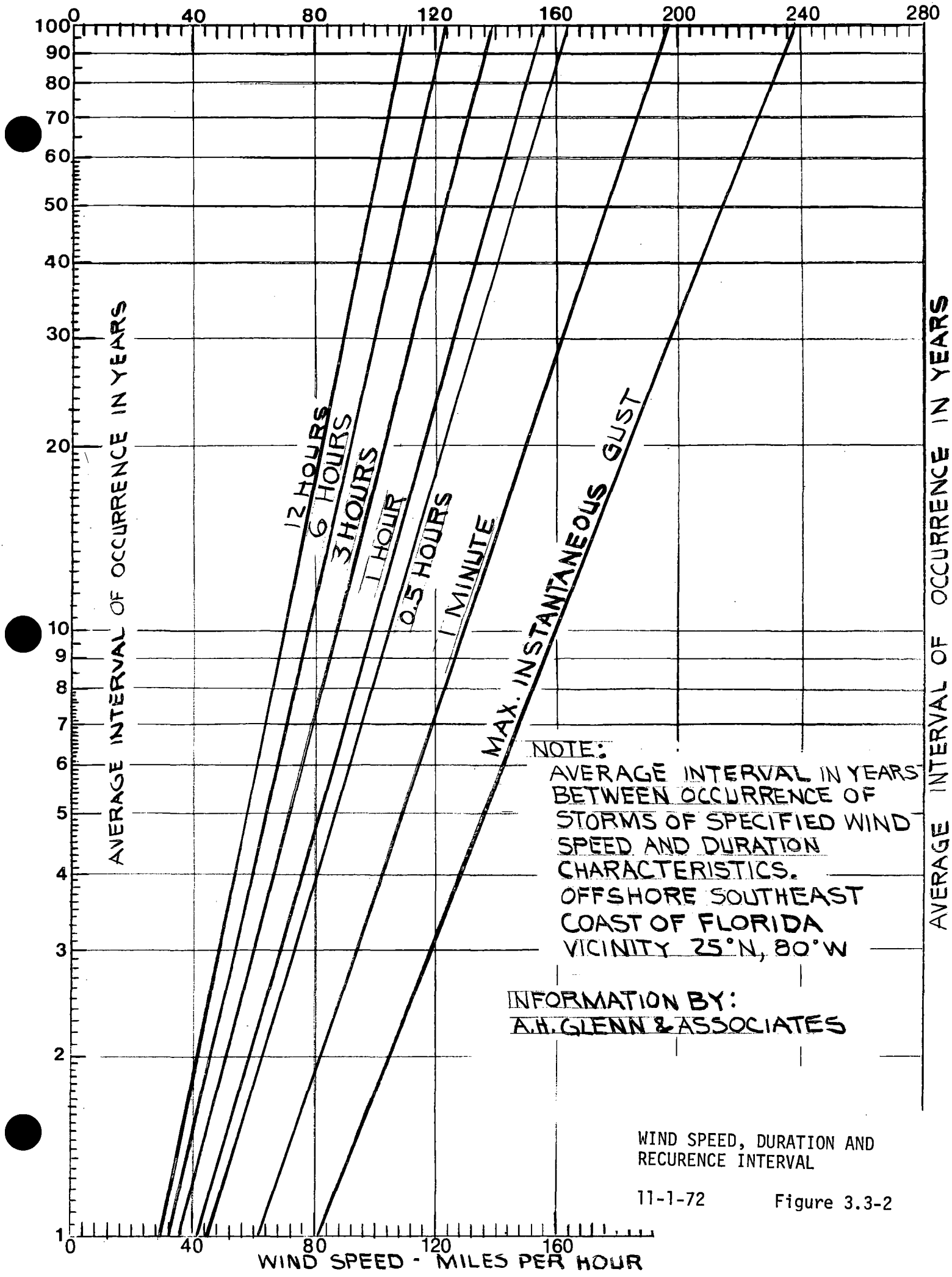
1. Solid arrows (⟶) indicate wind loads for external pressure.

2. Dotted arrows (- - -→) indicate wind loads for internal pressure.

3. Loads due to internal pressure are based on no openings in the walls. These loads are applied either inwards or outwards.

4. Leeward wall wind loads not shown will be obtained by multiplying the windward wall wind loads by the ratio of 0.5 to 0.8 for a height-width and height-length ratios less than 2.5. These loads will be applied outwards.

5. Wind loads shown on 0-1W strips are used in designing the local areas. They will not be included in designing the overall structure.

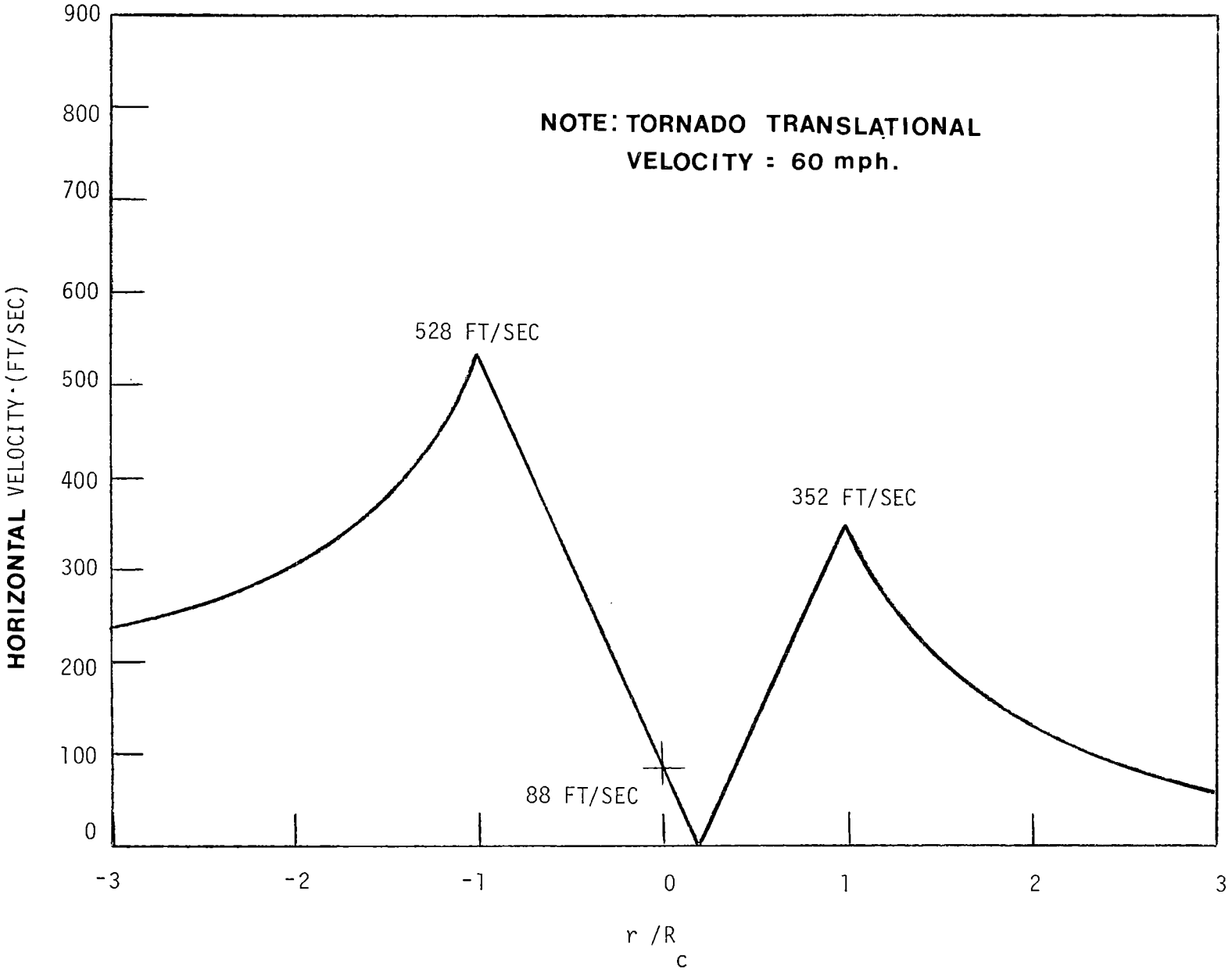


WIND SPEED, DURATION AND RECURENCE INTERVAL

11-1-72 Figure 3.3-2

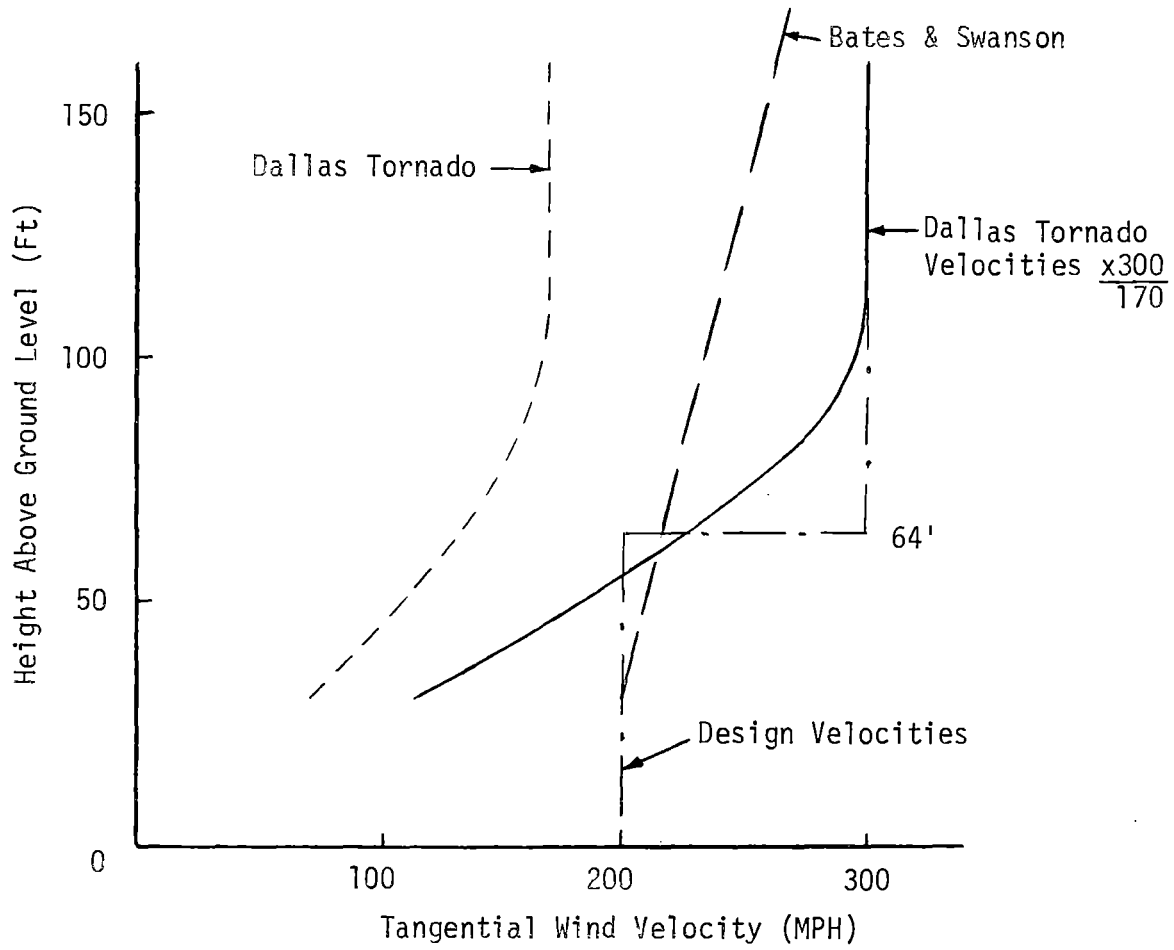
VARIATION OF DIFFERENTIAL PRESSURE AND TANGENTIAL VELOCITY AS A FUNCTION OF CYCLONIC RADIUS

11-1-72 Figure 3.3-3



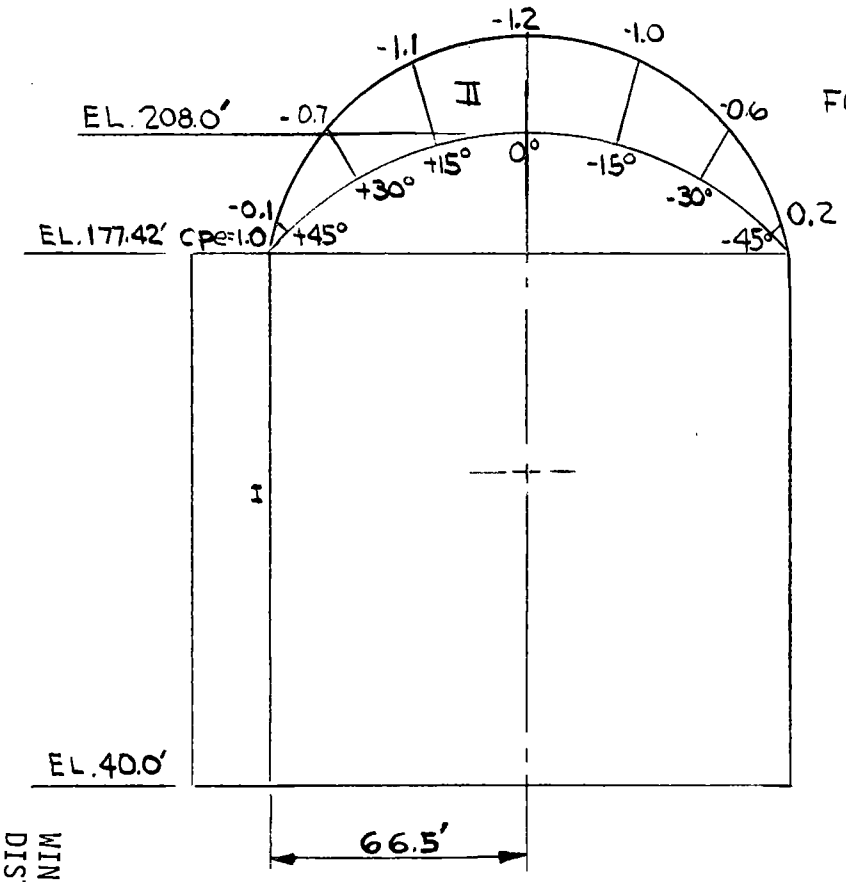
HORIZONTAL VELOCITY
DISTRIBUTION PARALLEL TO
DIRECTION OF TRANSLATION

11-1-72 Figure 3.3-4



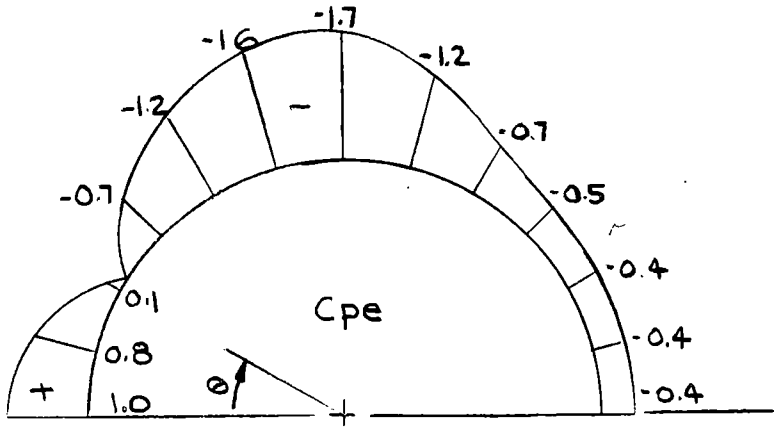
TORNADO TANGENTIAL VELOCITY
AS A FUNCTION OF HEIGHT

11-1-72 Figure 3.3-5



FOR PART II P = 1.6 Cpe

PRESSURE DISTRIBUTION ON PART II SYMMETRICAL ABOUT THE WIND DIRECTION.



PART I

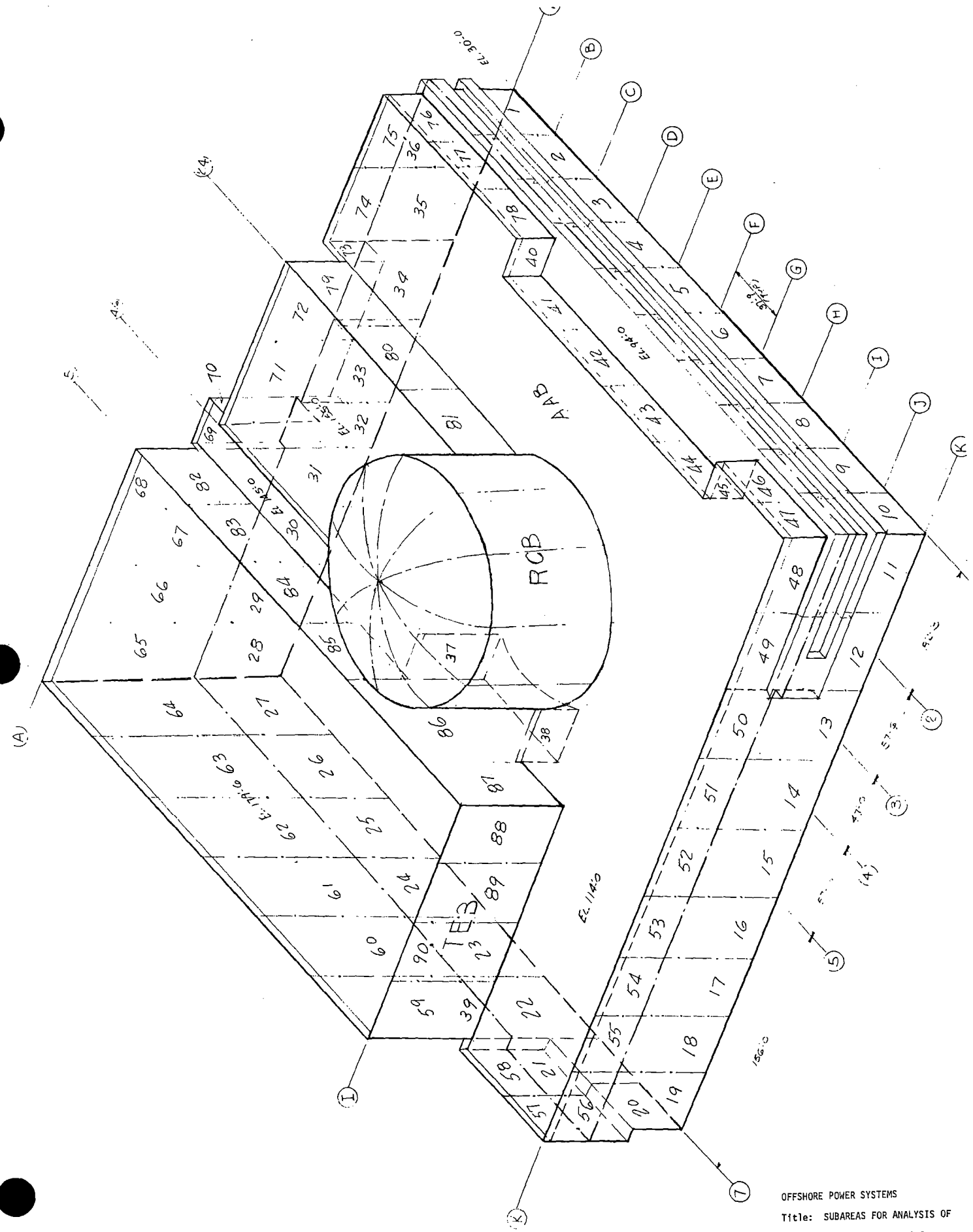
$$P = -0.96 + 0.49\cos\theta + 1.48\cos 2\theta + 0.62\cos 3\theta - 0.10\cos 4\theta + 0.03\cos 5\theta + 0.12\cos 6\theta + 0.01\cos 7\theta - 0.06\cos 8\theta - 0.01\cos 9\theta + 0.01\cos 10\theta - 0.01\cos 11\theta$$

Cpe = PRESSURE COEFFICIENT

P = PRESSURE FOR 300 M.P.H. WIND IN P.S.I.

WIND PRESSURE DISTRIBUTION ON THE REACTOR BUILDING

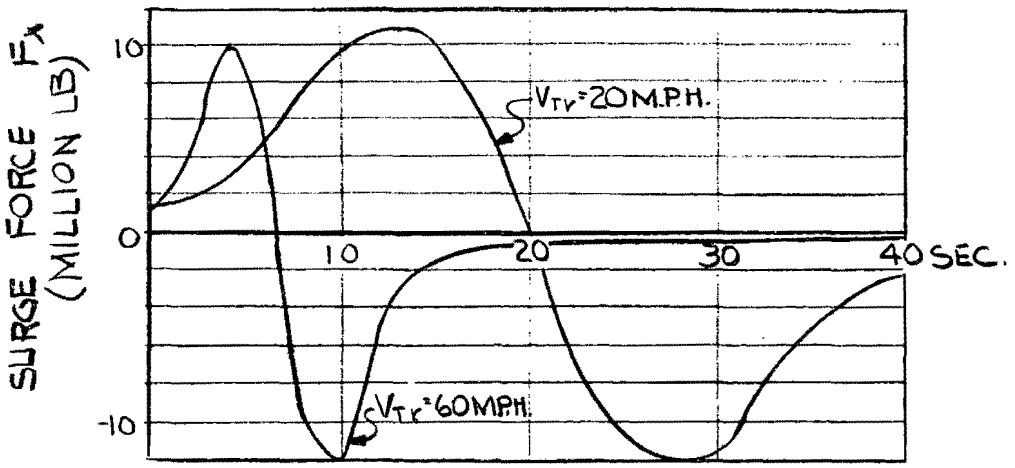
11-1-72 Figure 3.3-6

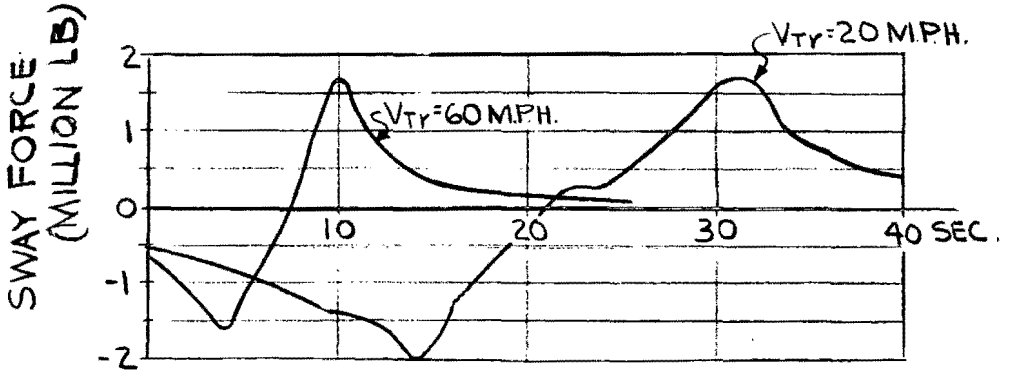


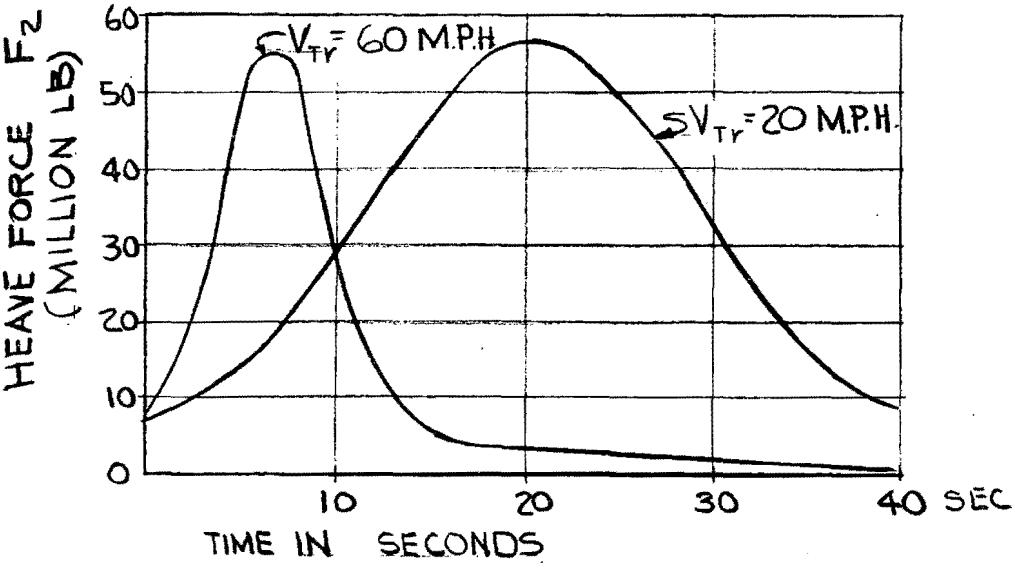
OFFSHORE POWER SYSTEMS

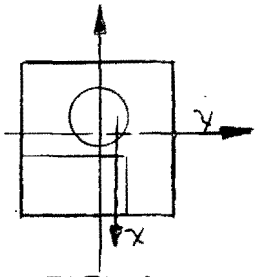
Title: SUBAREAS FOR ANALYSIS OF PLANT TORNADO LOADS

11/1/72 Figure: 3.3-7







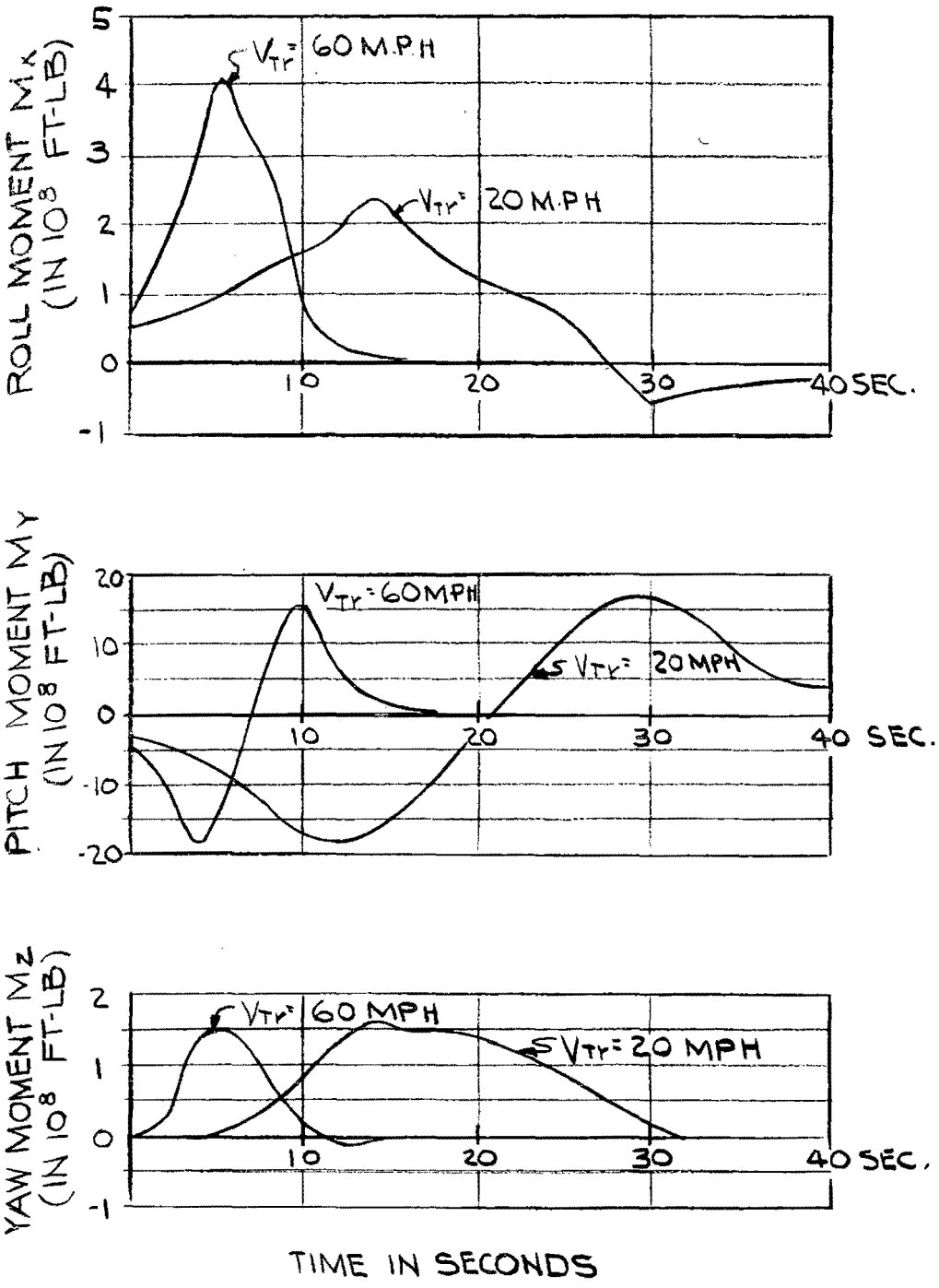


V_{Tr} = TORNADO TRANSLATIONAL VELOCITY

TORNADO FORCES

11-1-72 Figure 3.3-8

Sheet 1 of 2

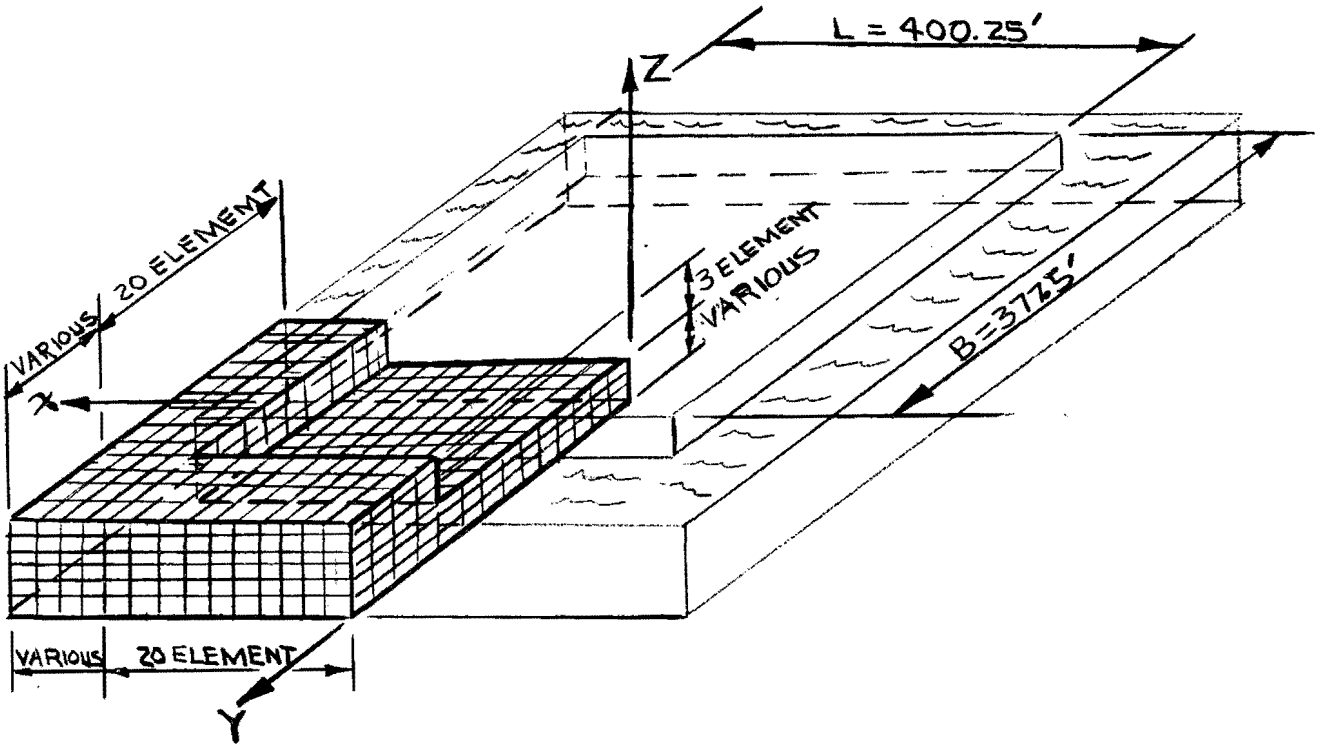


TORNADO FORCES

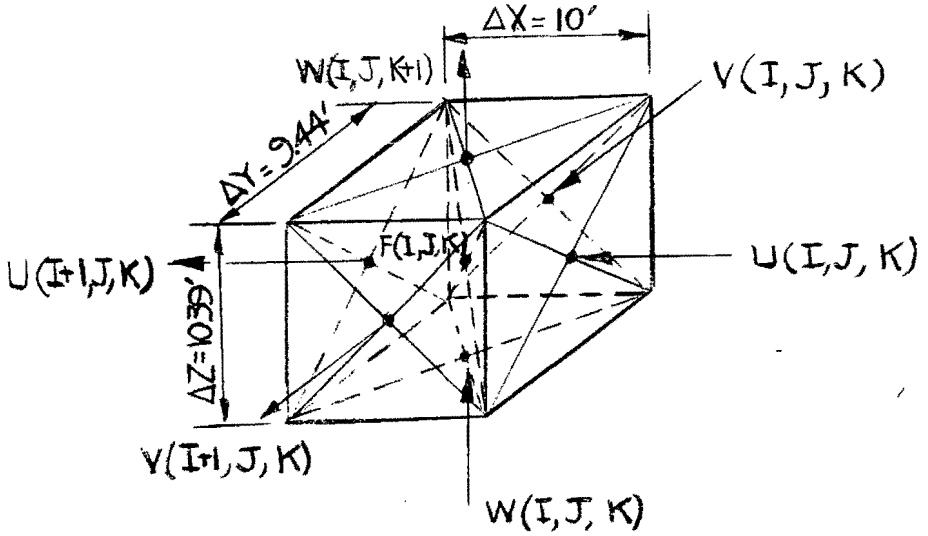
11-1-72

Figure 3.3-8

Sheet 2 of 2

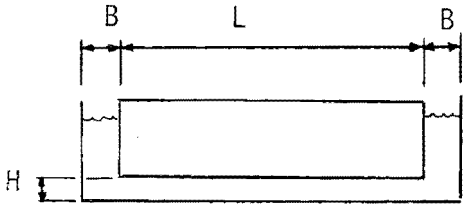


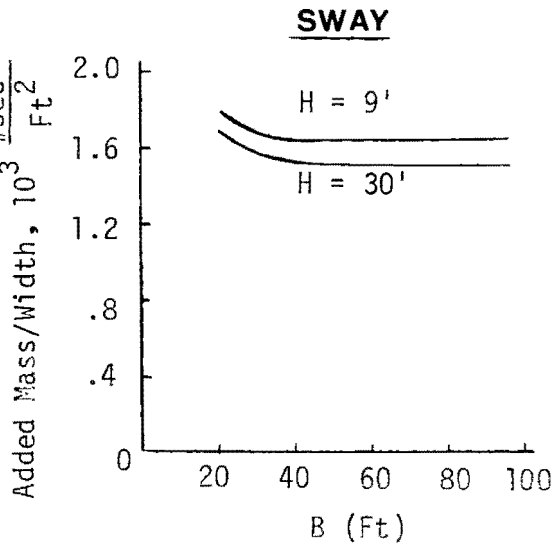
THREE DIMENSIONAL MESHES OF WATER SURROUNDING THE BARGE (ONLY ONE QUARTER IS SHOWN)



LOCATIONS OF NORMAL VELOCITIES AND POTENTIAL FOR A TYPICAL WATER ELEMENT

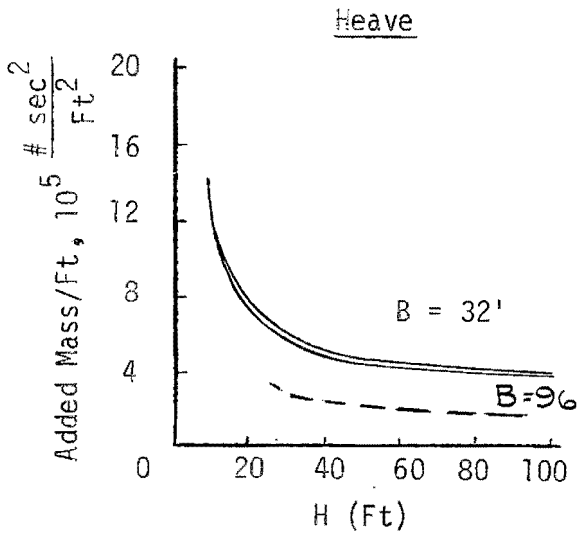
11-1-72 Figure 3.3-9

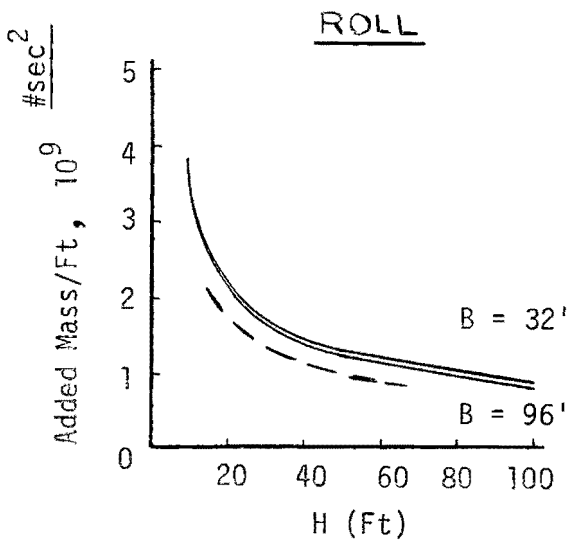




——— 2 DIMENSIONAL RESULTS

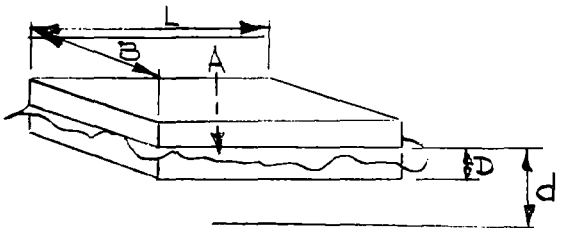
- - - - 3 DIMENSIONAL RESULTS

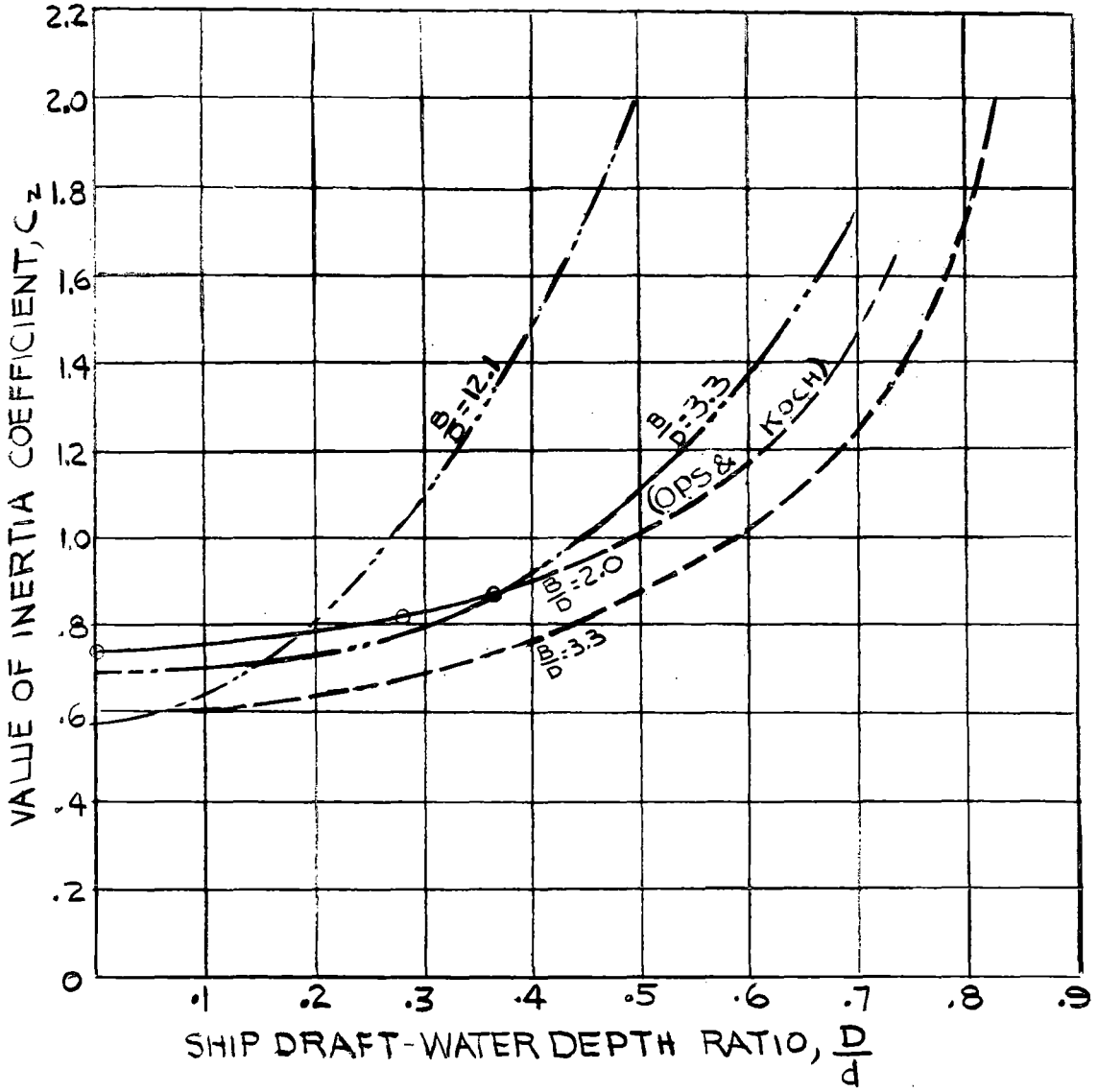




PLATFORM ADDED MASSES

11-1-72 Figure 3.3-10





LEGEND

—o— WENDEL (THEORY) FOR CASE B=2D

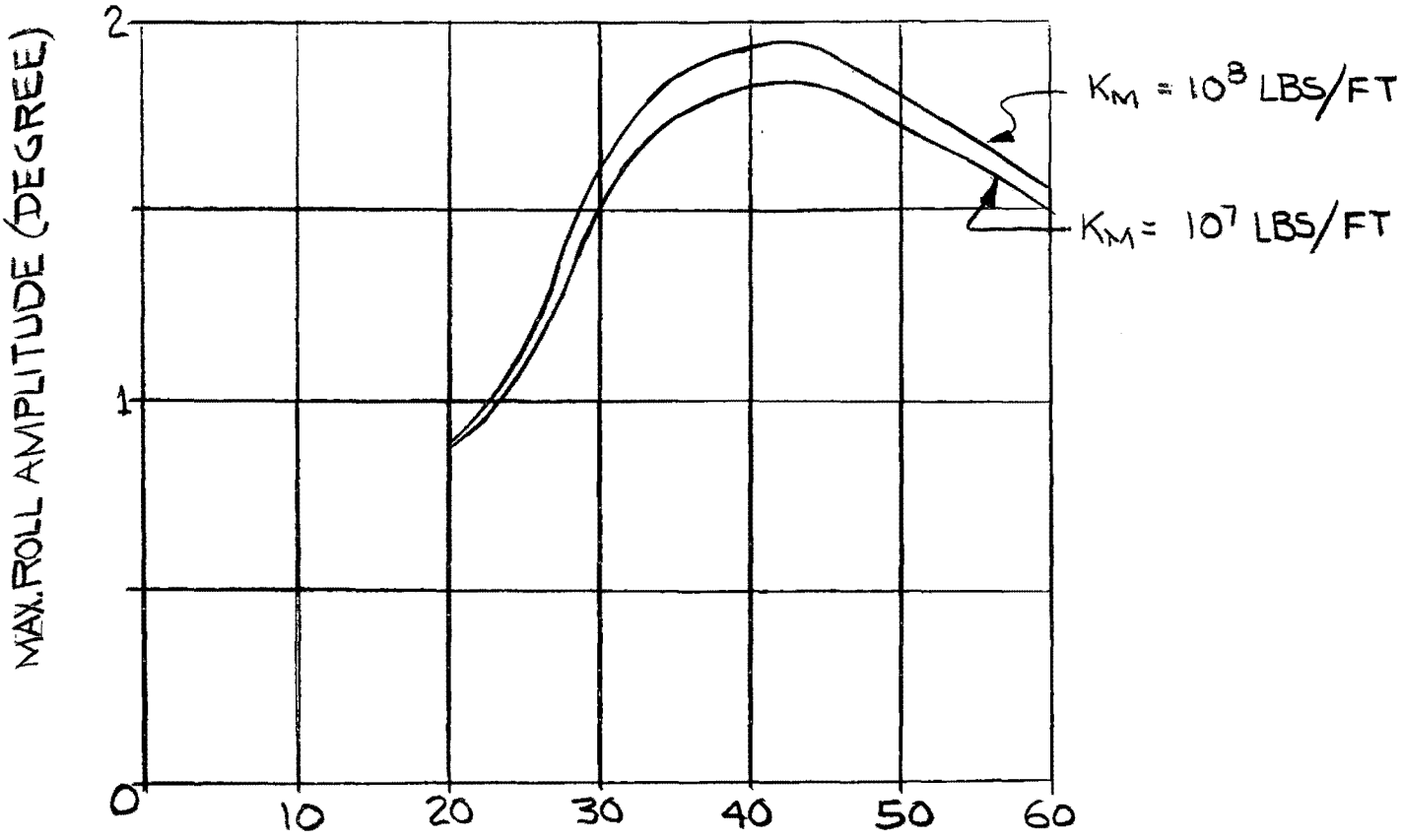
--- ADAPTED FROM KOCH (EXPERIMENT; ELECTRICAL ANALOG)

—·—· OFFSHORE POWER SYSTEMS

NOTE: $C_Z = \frac{4 \times (\text{ADDED MASS})}{\pi \rho B^2 L}$

DEPENDENCE OF ADDED MASS (FOR RECTANGULAR BLOCK) ON SHIP DIMENSIONS AND WATER DEPTH IN HEAVING MOTION

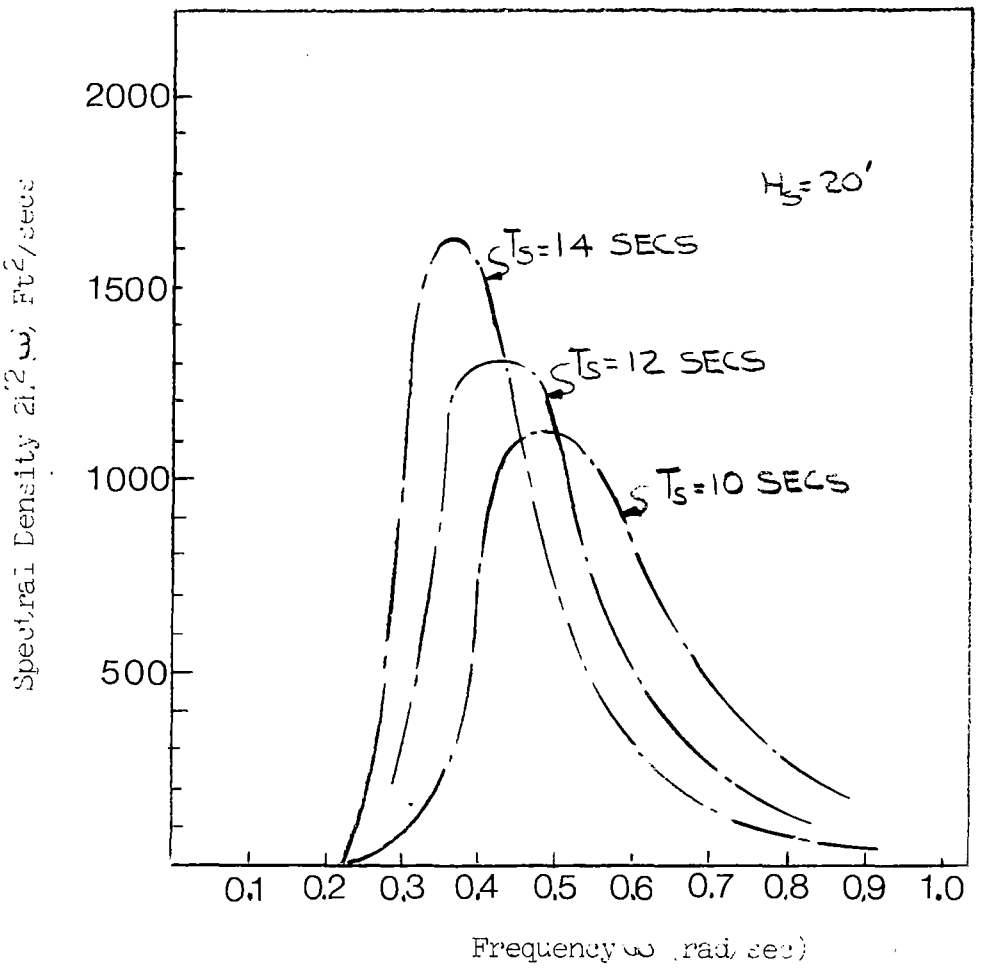
11-1-72 Figure 3.3-11



NOTE: K_M = HORIZONTAL MOORING STIFFNESS

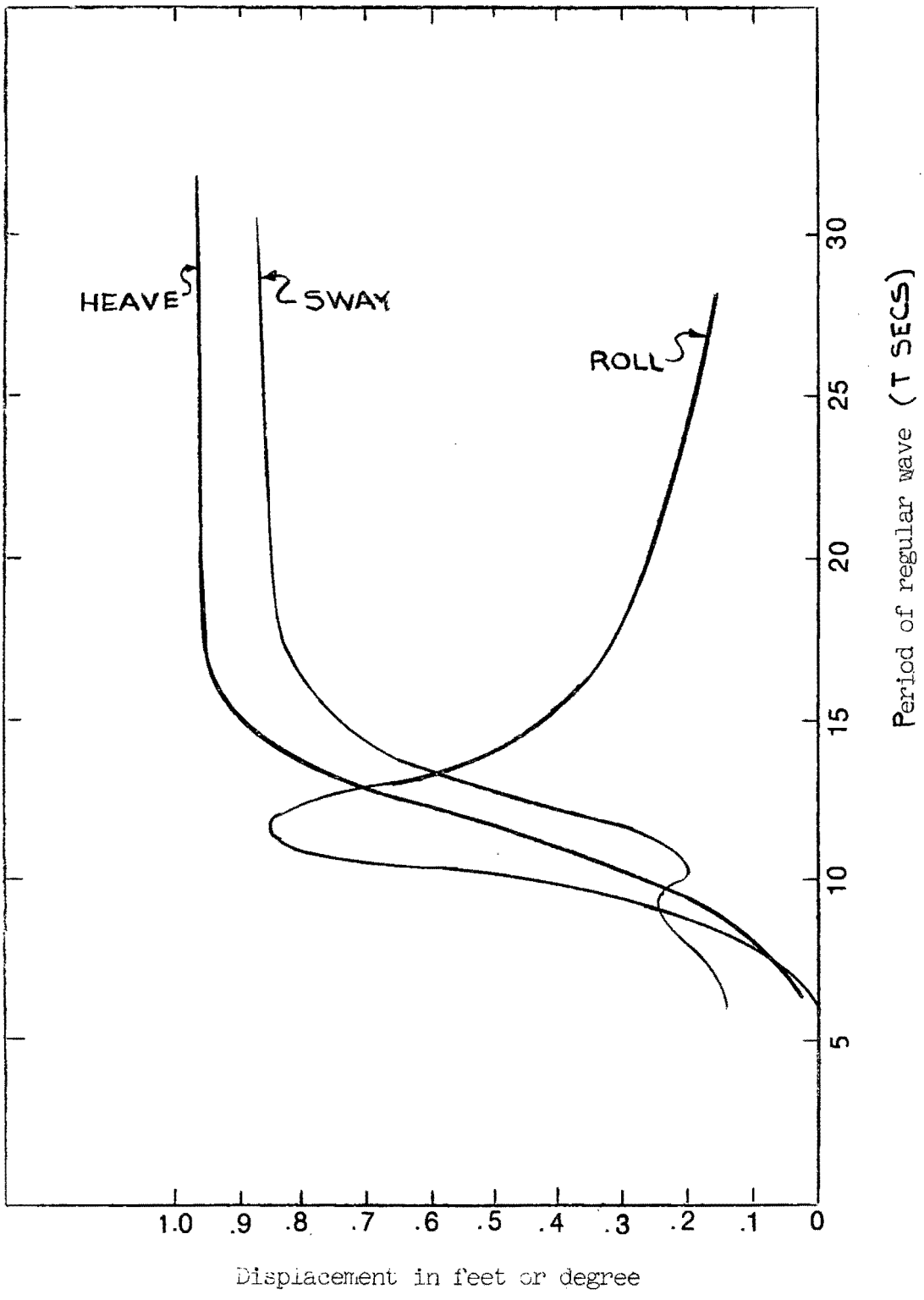
VARIATION OF MAXIMUM PLANT ROLL RESPONSE WITH TORNADO TRANSLATIONAL VELOCITY

11-1-72 Figure 3.3-12



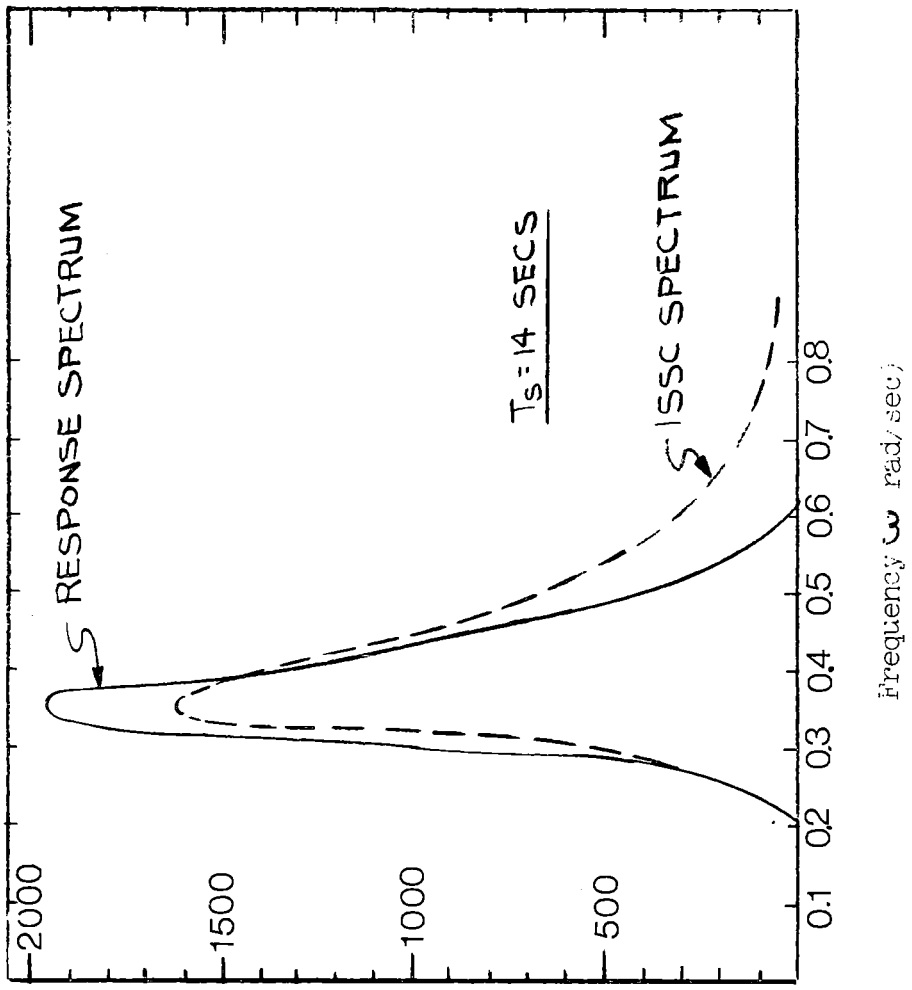
ISSC WAVE SPECTRUM

11-1-72 Figure 3.4-1



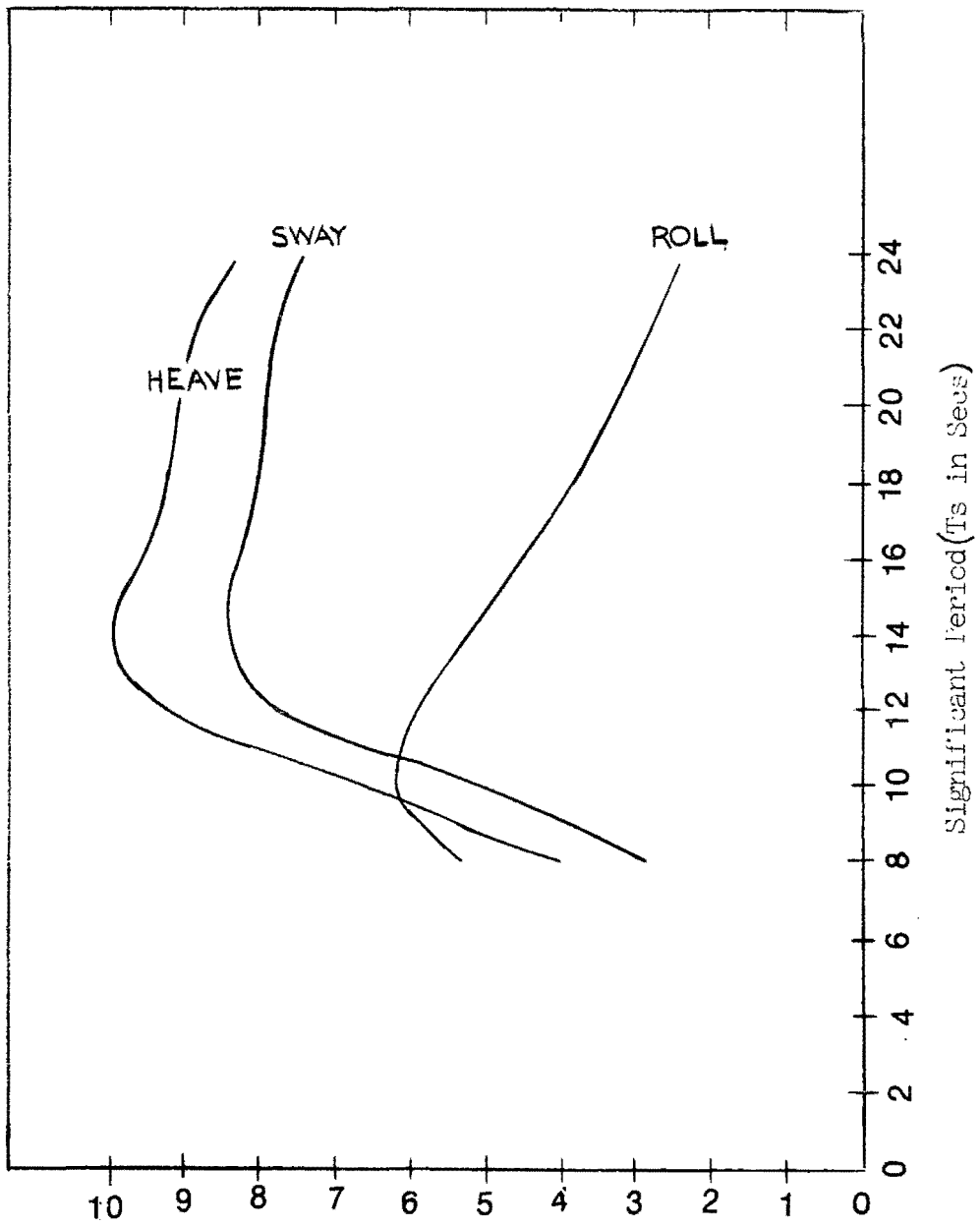
PLANT RESPONSE
AMPLITUDE OPERATORS

11-1-72 Figure 3.4-2



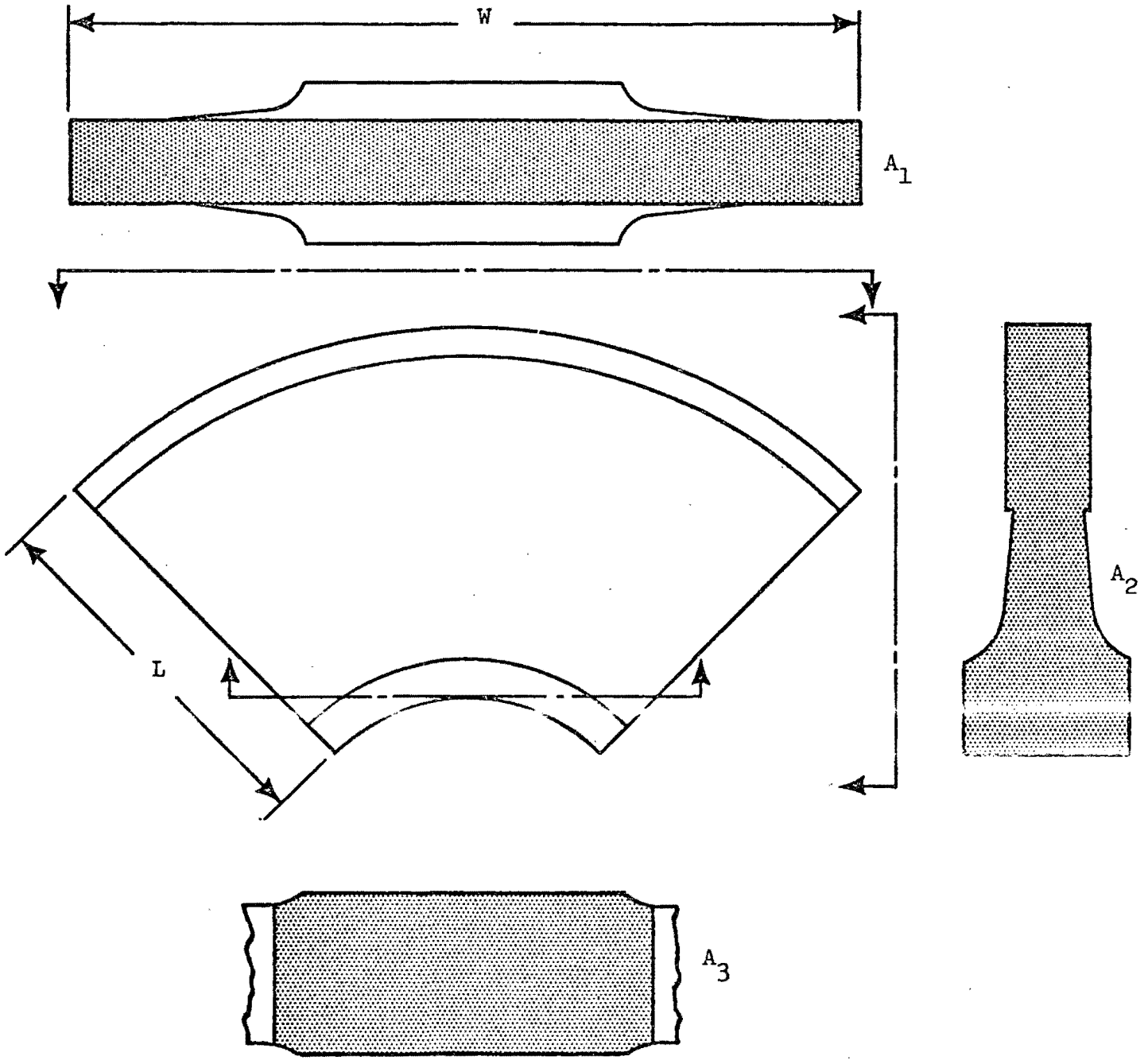
RESPONSE SPECTRUM IN HEAVE

11-1-72 Figure 3.4-3



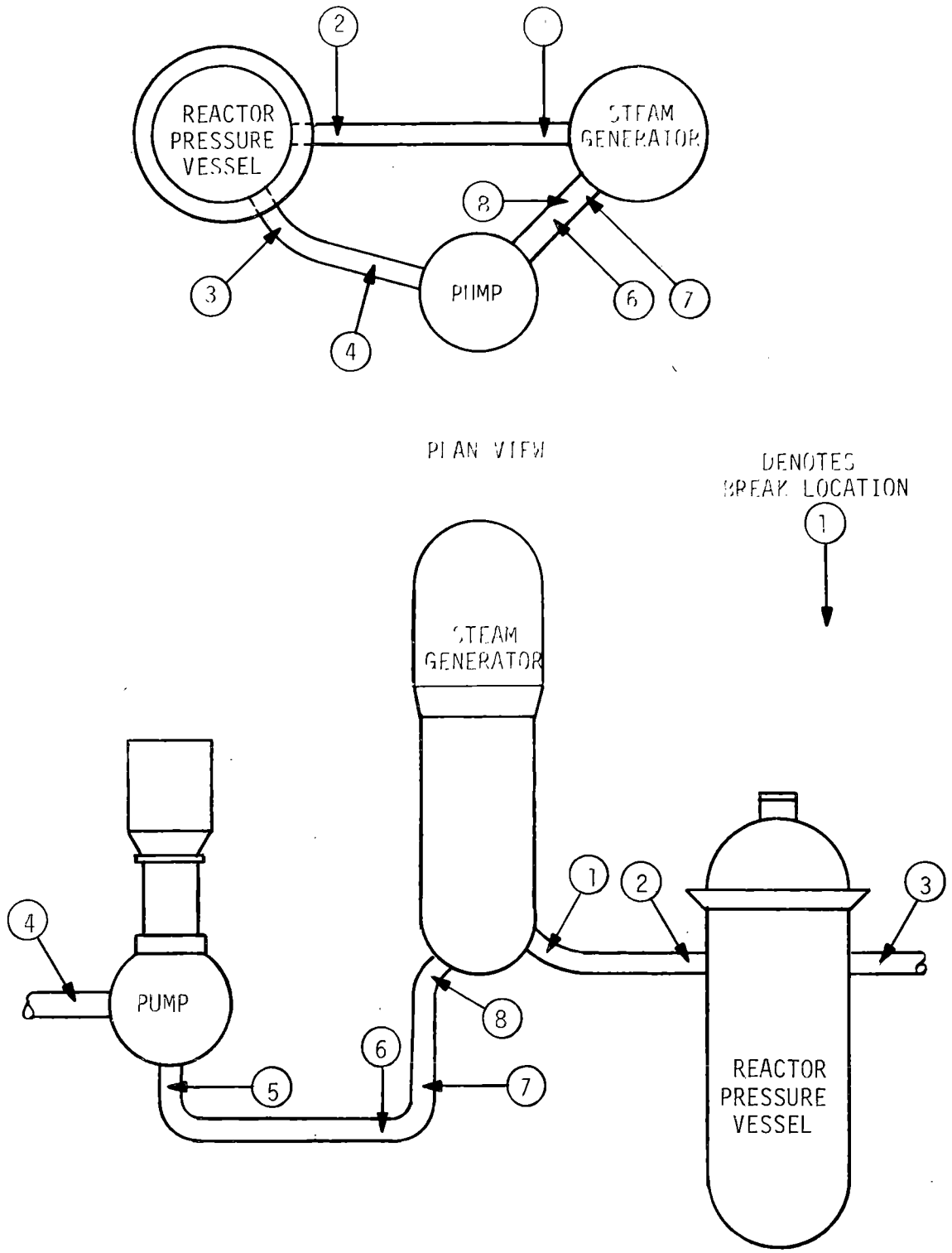
PLATFORM RESPONSE AS A FUNCTION
OF SIGNIFICANT PERIOD

11-1-72 Figure 3.4-4



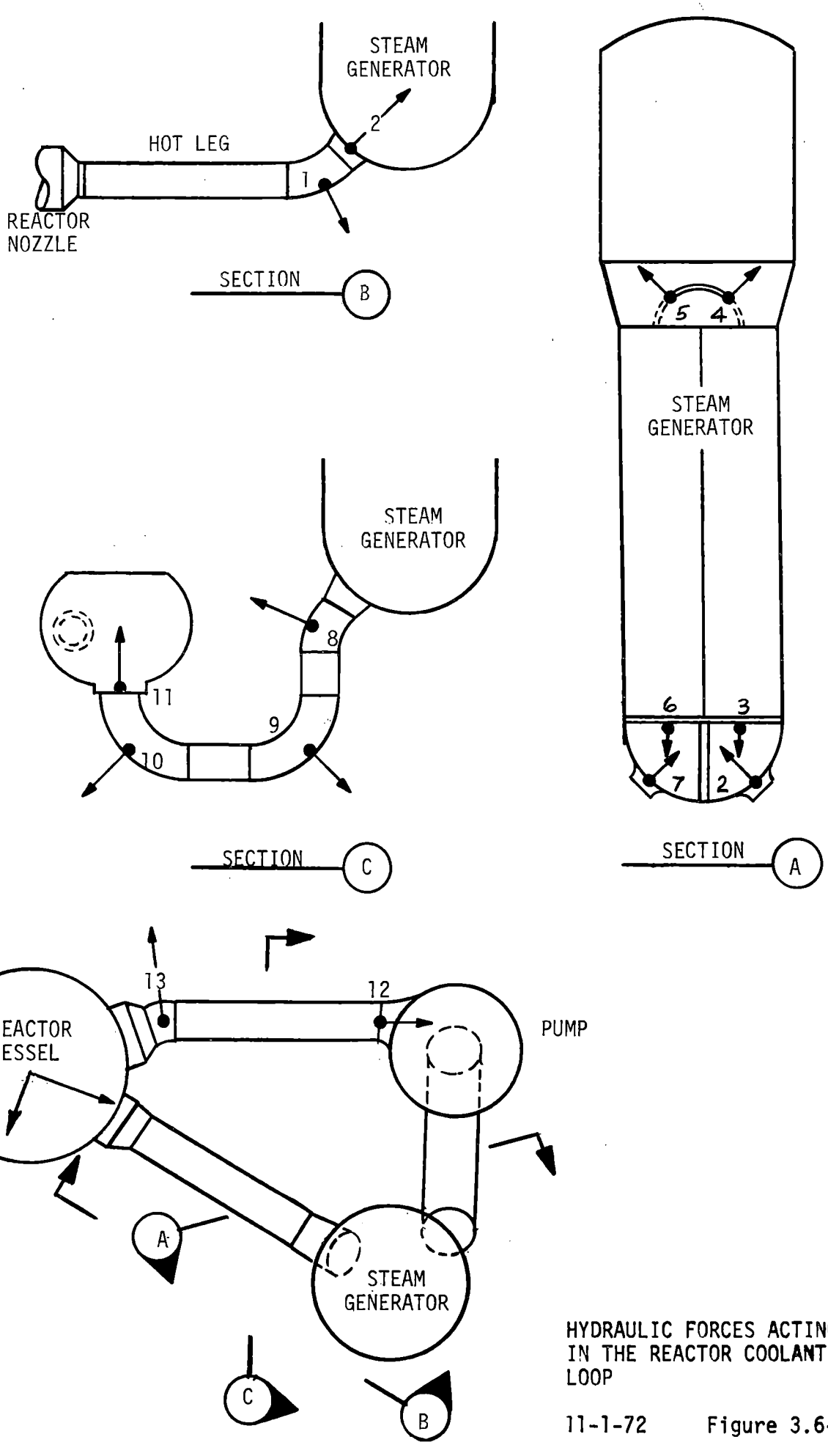
PHYSICAL DIMENSIONS OF MISSILE

11-1-72 Figure 3.5-1



REACTOR COOLANT LOOP LOCATION
OF POSTULATED BREAKS

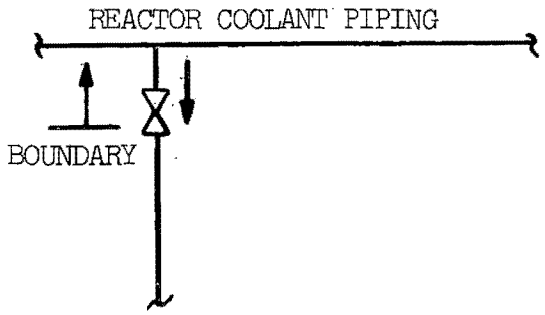
11-1-72 Figure 3.6-1



HYDRAULIC FORCES ACTING IN THE REACTOR COOLANT LOOP

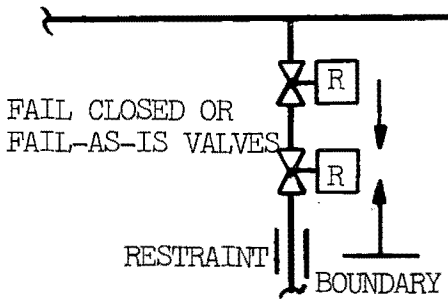
11-1-72 Figure 3.6-2

CASE I OUTGOING LINES WITH NORMALLY CLOSED VALVE.



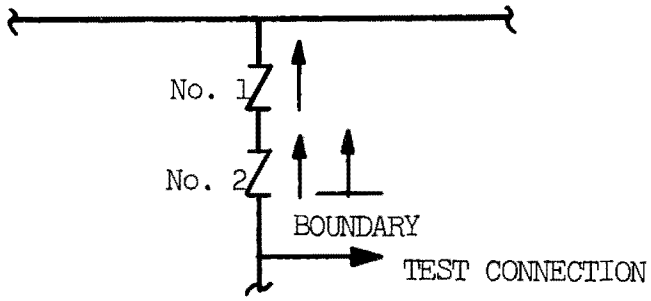
NOTE: Pressurizer Safety valves are included under this case.

CASE II OUTGOING LINES WITH NORMALLY OPEN VALVES.

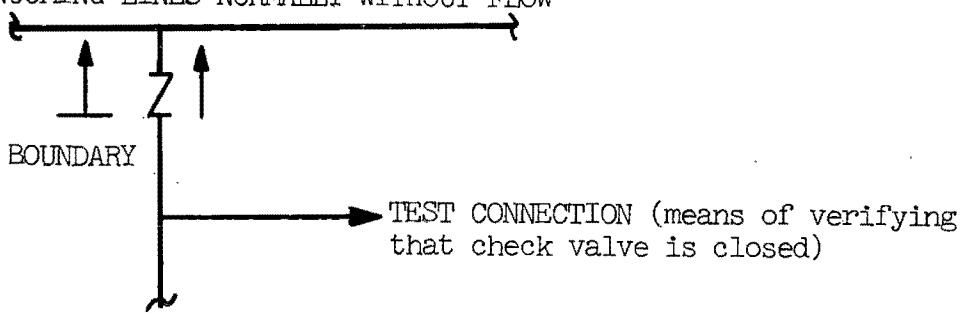


NOTE: The reactor coolant pump No. 1 seal is assumed to be equivalent to first valve.

CASE III INCOMING LINES NORMALLY WITH FLOW



CASE IV INCOMING LINES NORMALLY WITHOUT FLOW

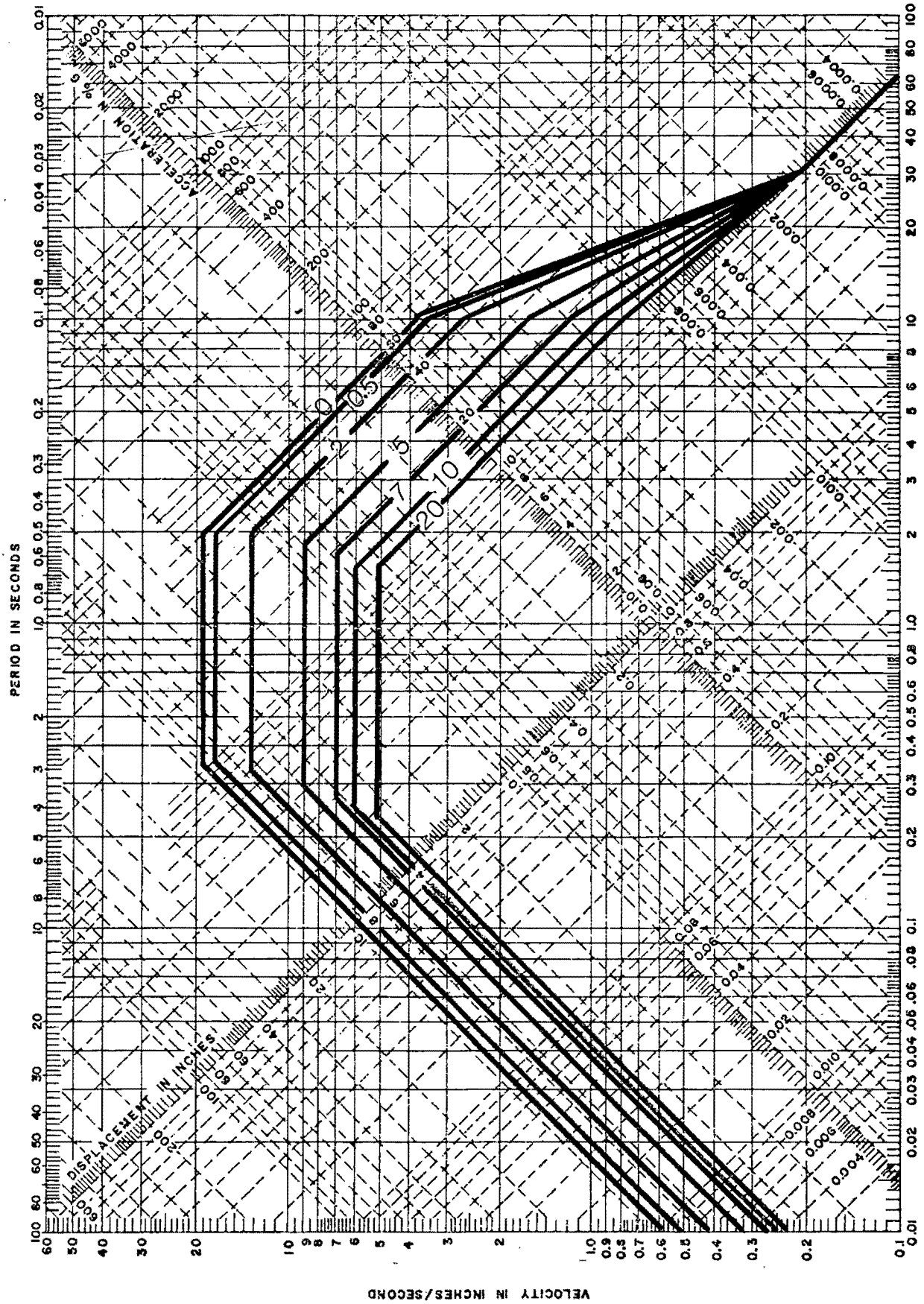


CASE V

All instrumentation tubing and instruments connected directly to the reactor coolant system is considered as a boundary. However, a break within this boundary results in a relatively small flow which can normally be made up with the charging system.

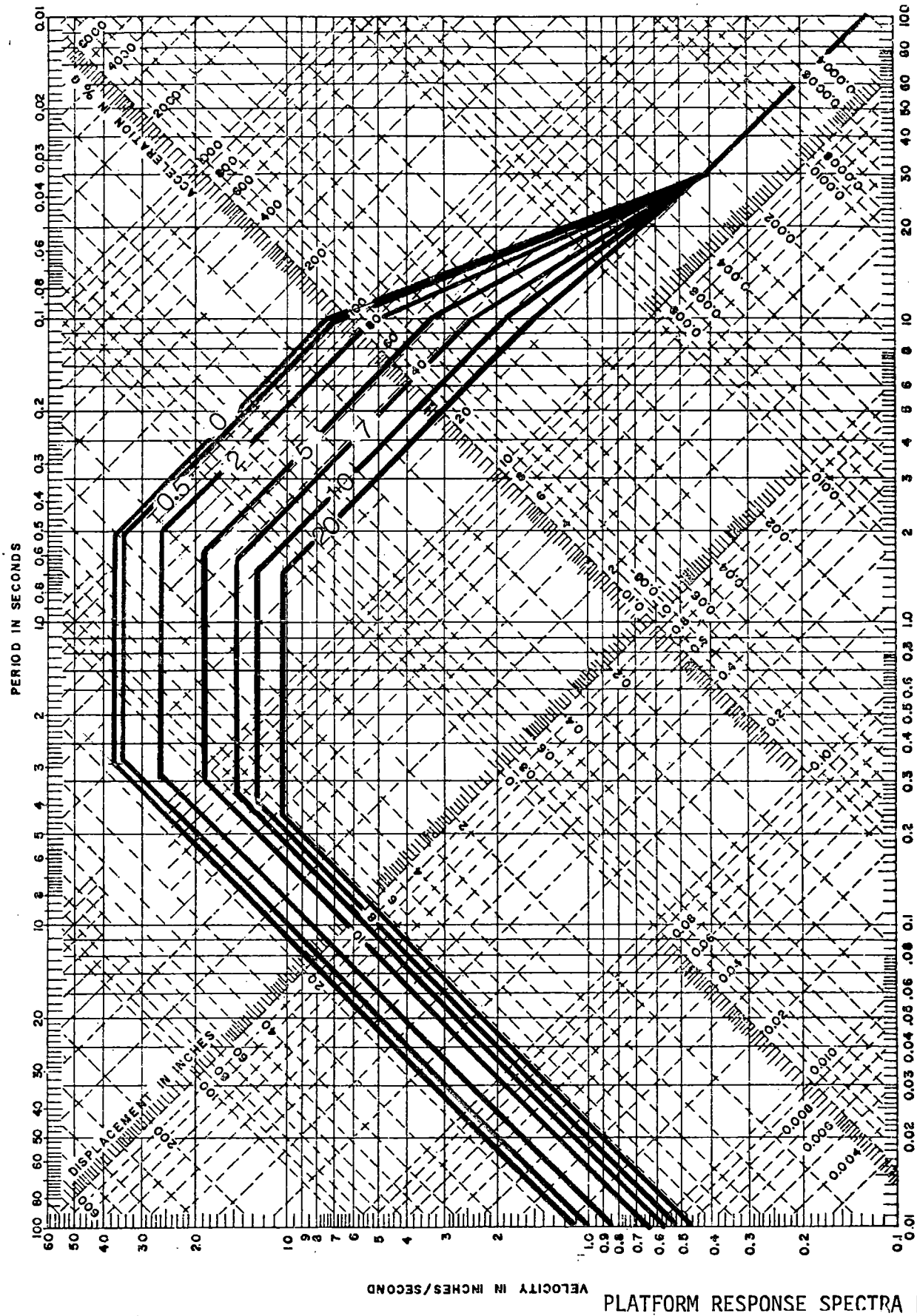
LOSS OF REACTOR COOLANT ACCIDENT BOUNDARY LIMITS

11-1-72 Figure 3.6-3



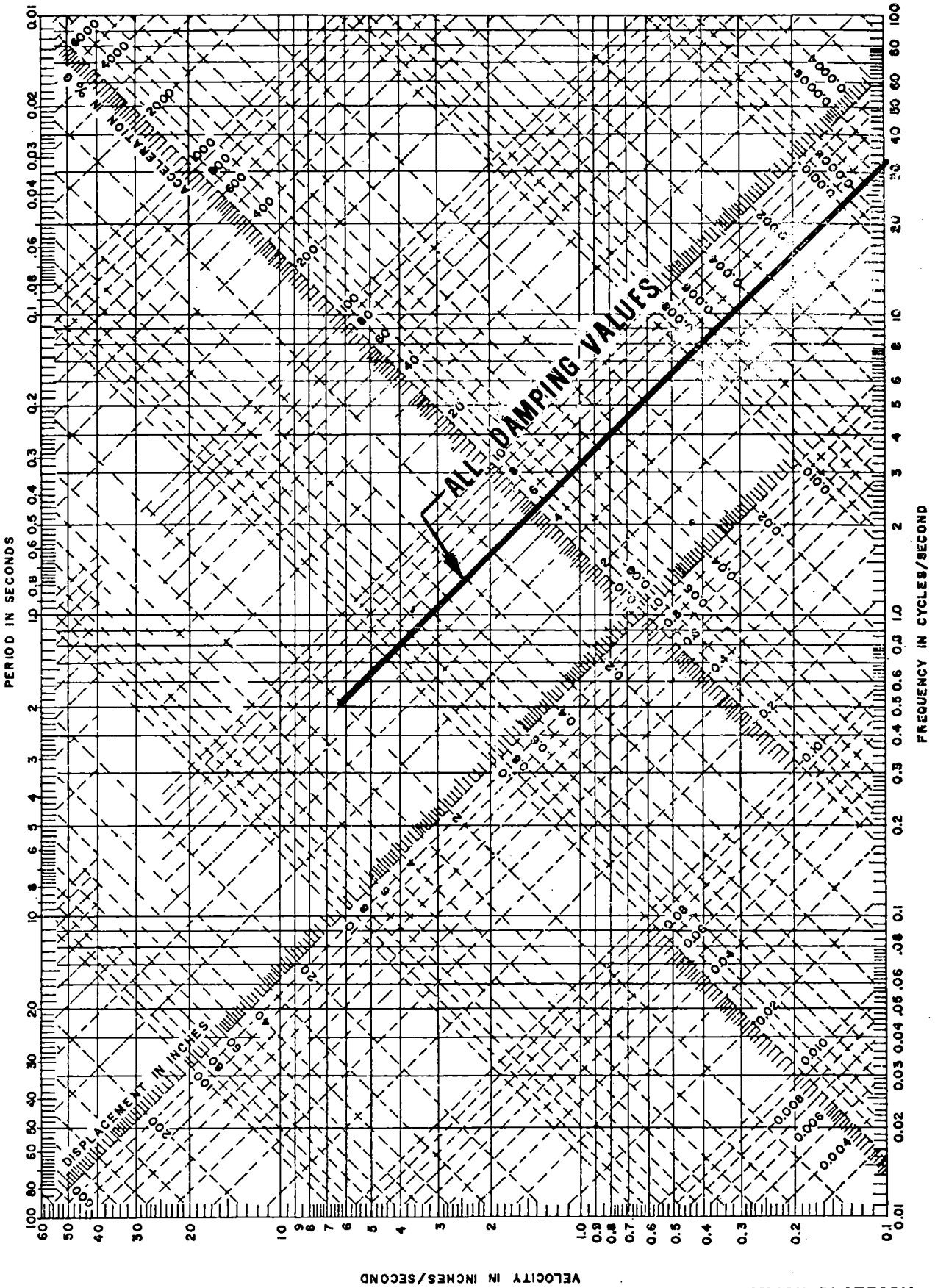
PLATFORM RESPONSE SPECTRA FOR VERTICAL MOTION (OPERATING BASIS EARTHQUAKE).

11-1-72 Figure 3.7-1



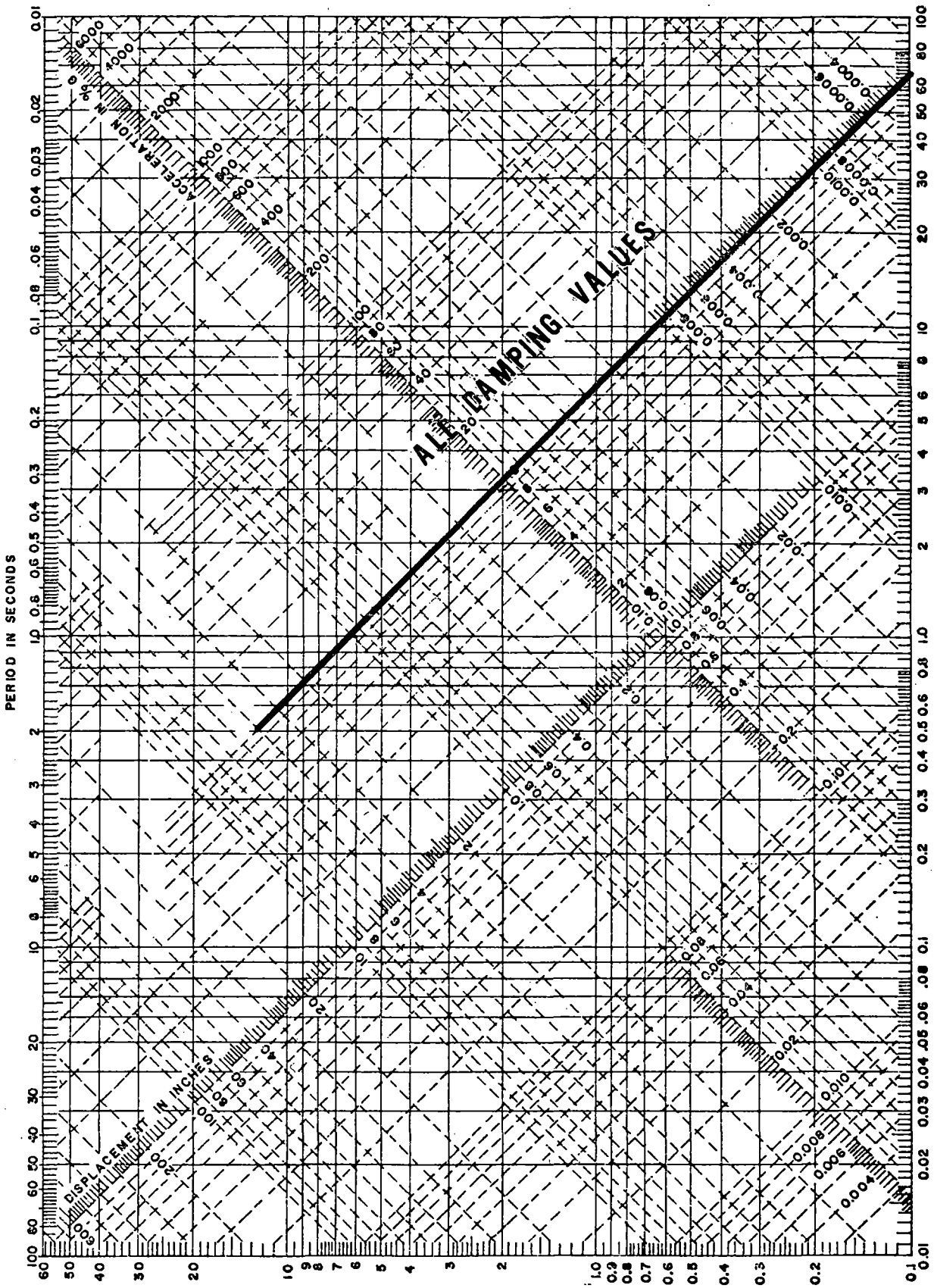
PLATFORM RESPONSE SPECTRA FOR VERTICAL MOTION (OPERATING BASIS EARTHQUAKE)

11-1-72 Figure 3.7-2



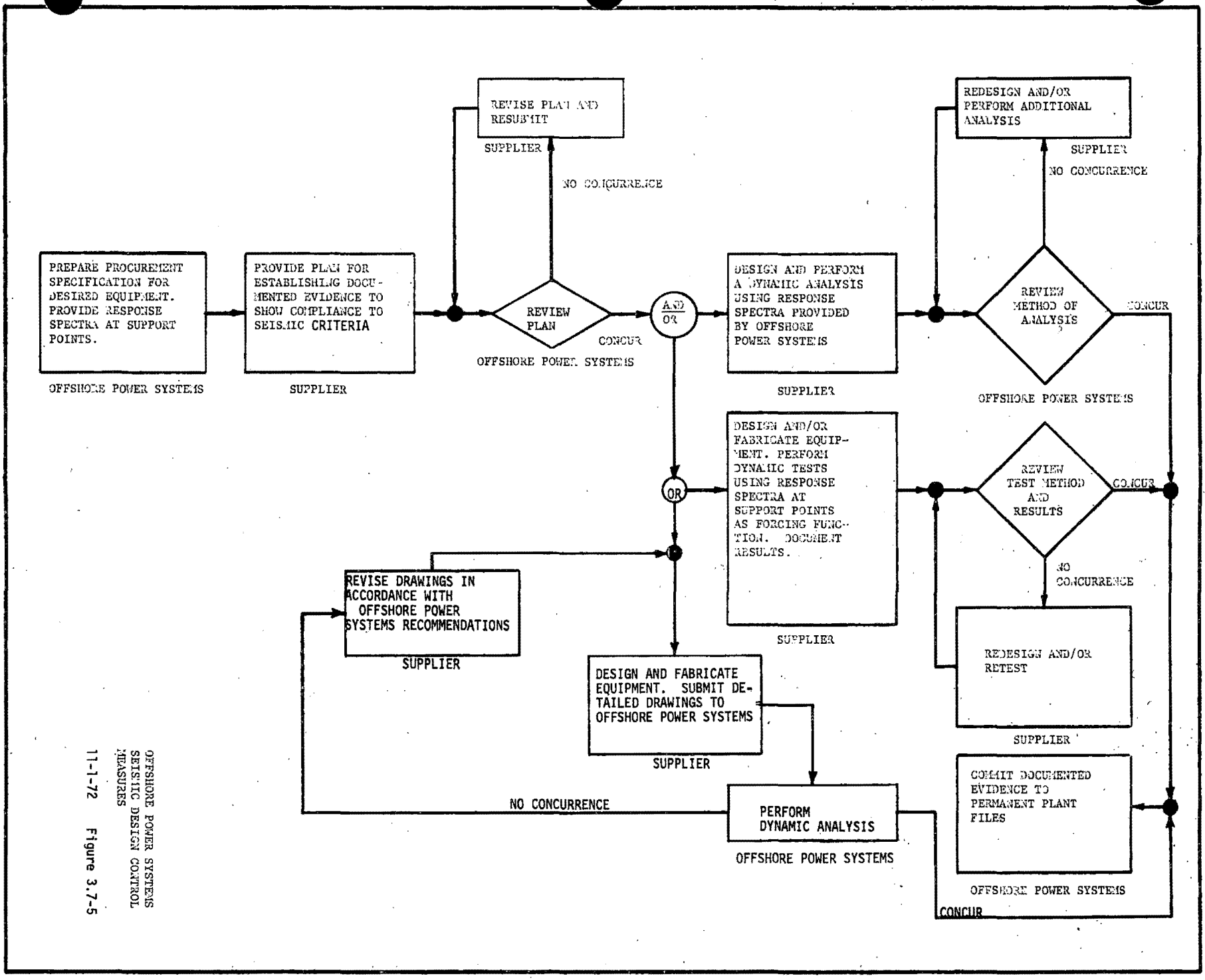
HORIZONTAL MOTION PLATFORM SPECTRA FOR (DESIGN BASIS EARTHQUAKE)

11-1-72 Figure 3.7-3



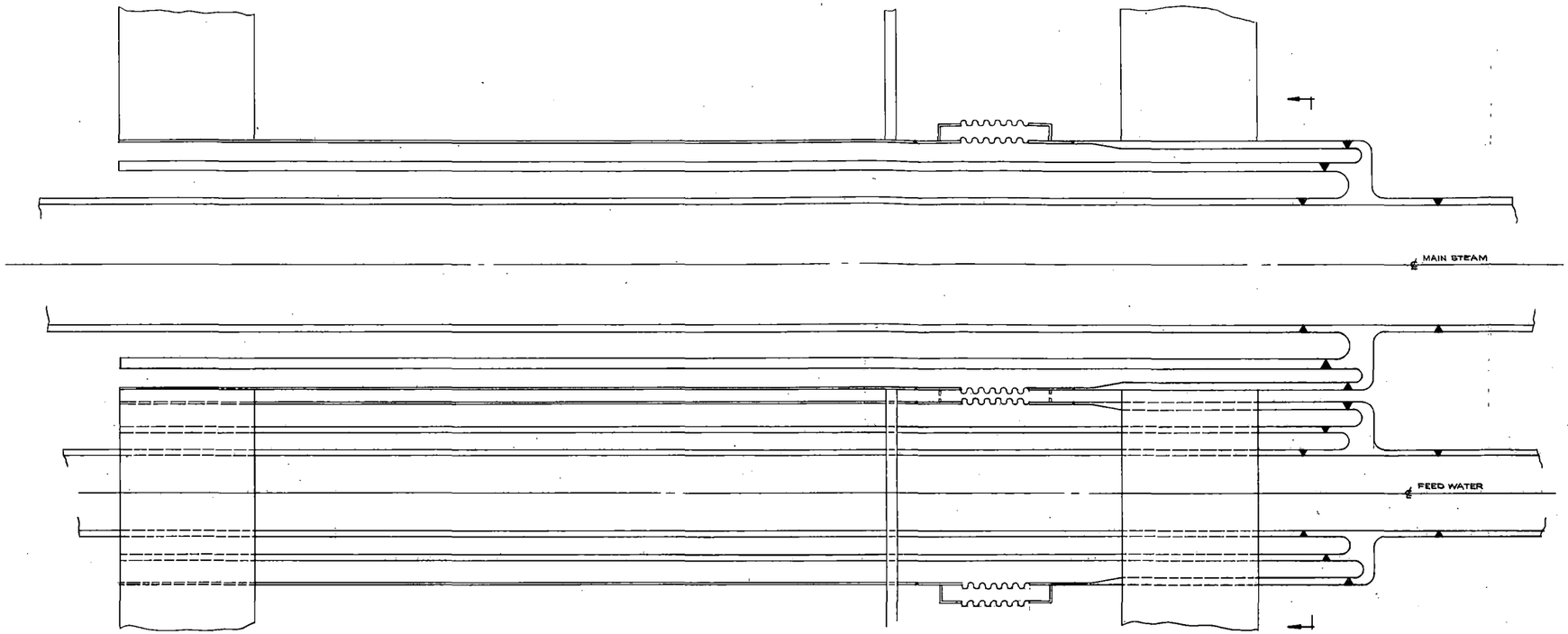
HORIZONTAL MOTION PLATFORM
RESPONSE SPECTRA FOR
(DESIGN BASIS EARTHQUAKE)

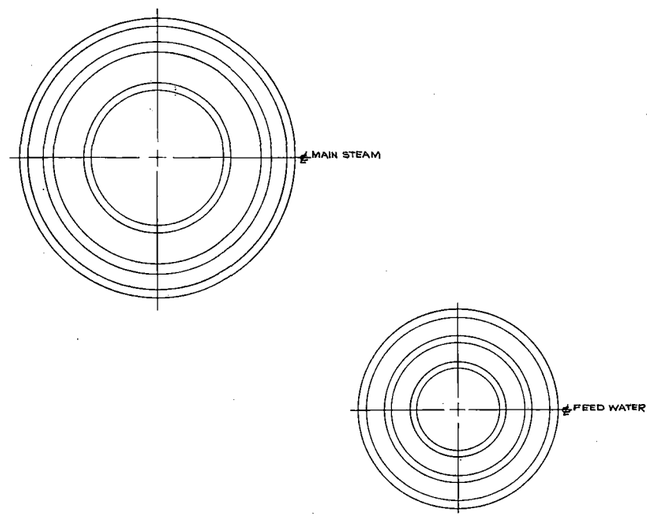
11-1-72 Figure 3.7-4



OFFSHORE POWER SYSTEMS SEISMIC DESIGN CONTROL MEASURES

11-1-72 Figure 3.7-5



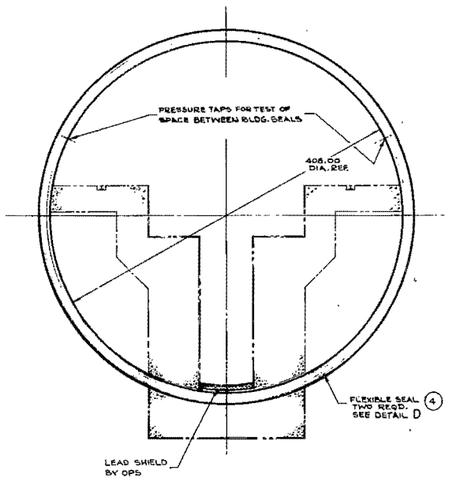


SECTION

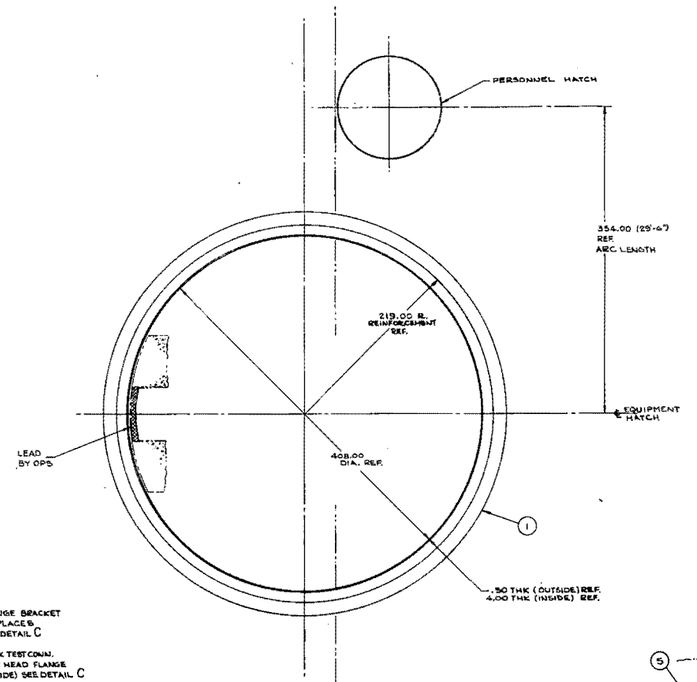
OFFSHORE POWER SYSTEMS

Title: MAIN STEAM AND FEEDWATER
CONTAINMENT PENETRATION DETAILS

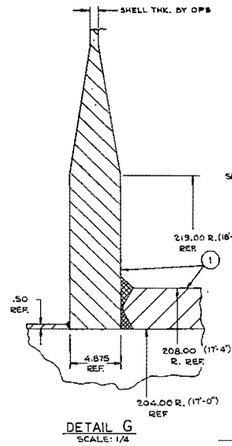
11/1/72 Figure: 3.8-5



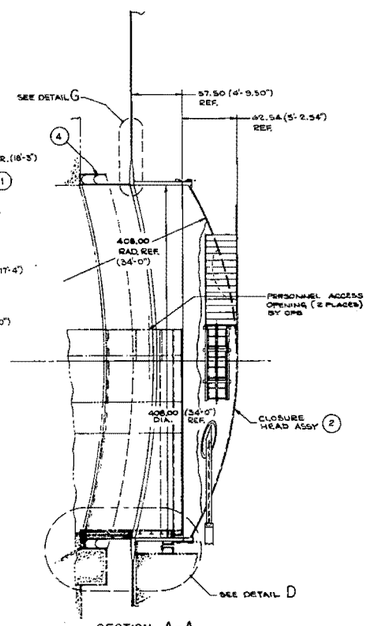
SECTION E-E
SCALE: 1/4"=1'-0" ROTATED 90° CCW



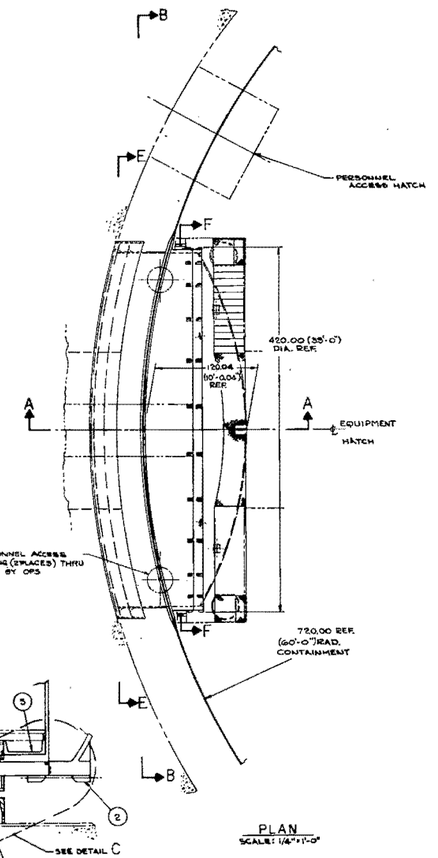
SECTION B-B
DEVELOPED PROJECTION
SCALE: 1/8"=1'-0"



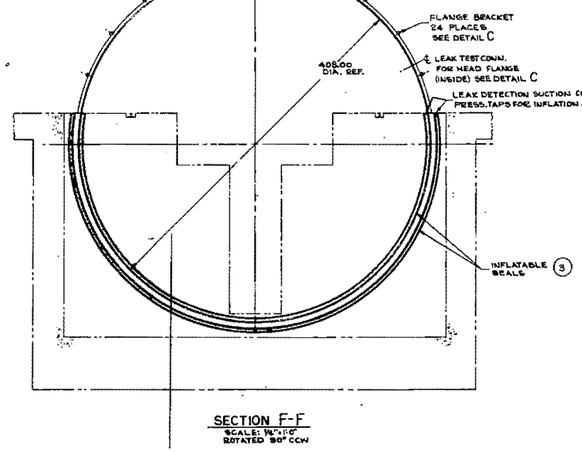
DETAIL G
SCALE: 1/4



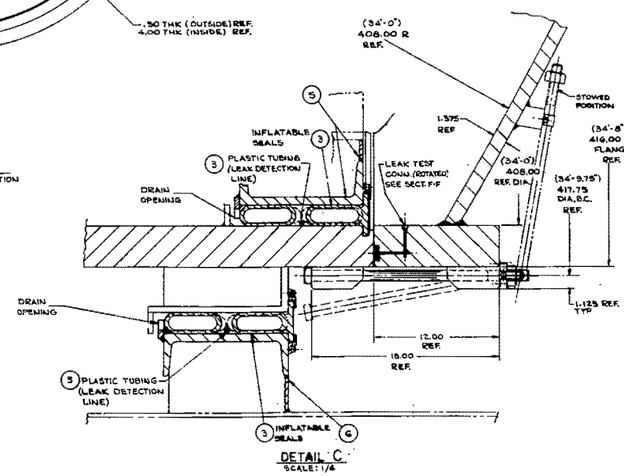
SECTION A-A
SCALE: 1/4"=1'-0"



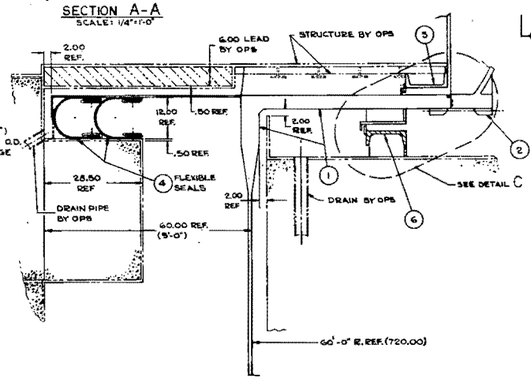
PLAN
SCALE: 1/16"=1'-0"



SECTION F-F
SCALE: 1/4"=1'-0" ROTATED 90° CCW



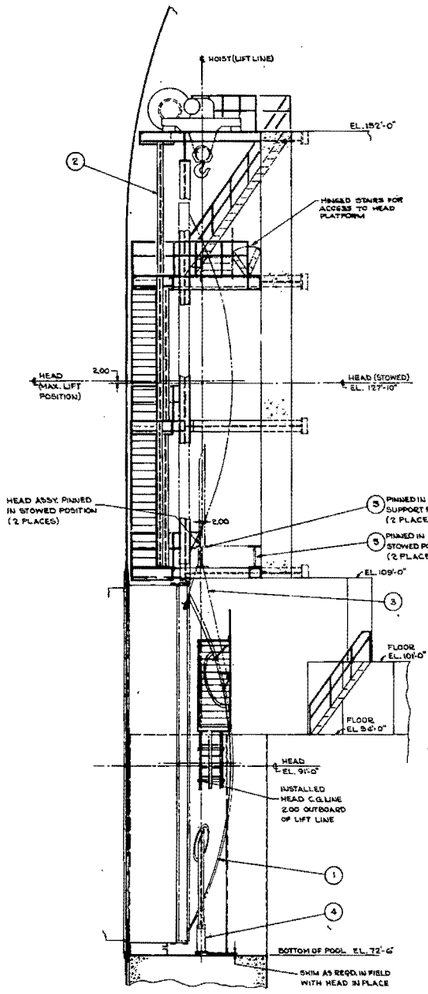
DETAIL C
SCALE: 1/4



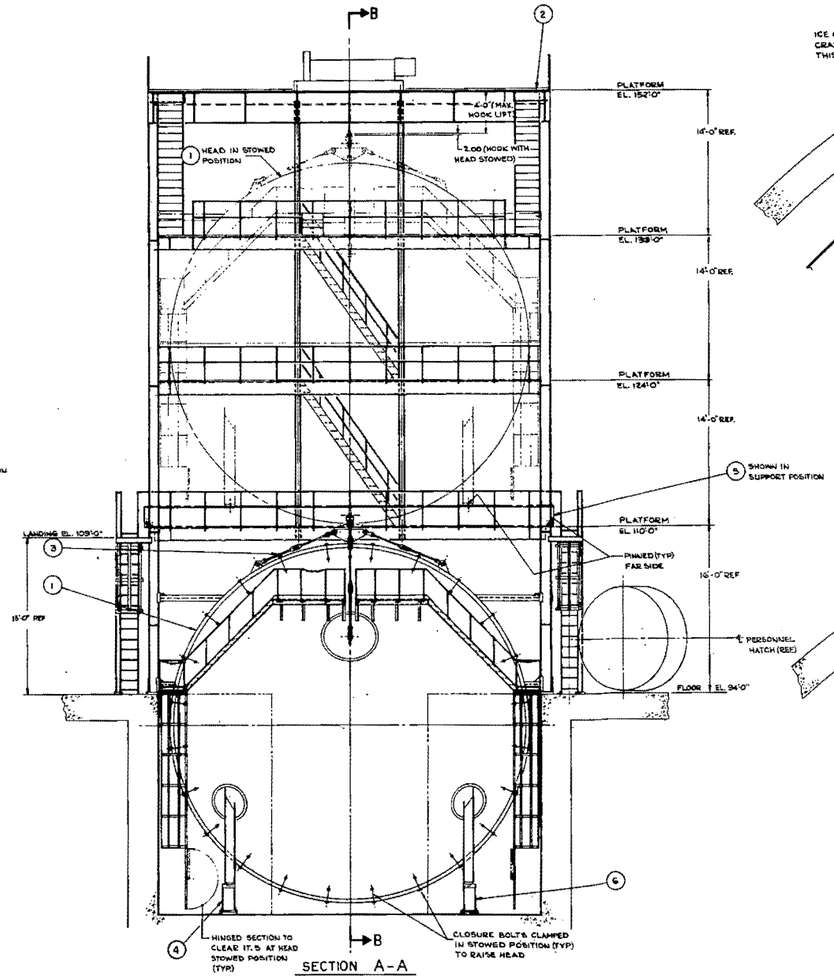
DETAIL D
SCALE: 1" = 1'-0"

REFUELING HATCH ASSEMBLY

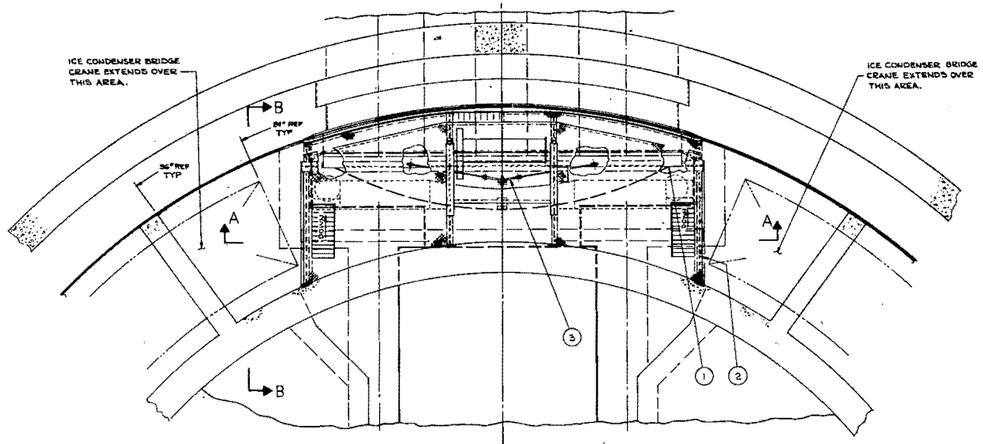
11-1-72 Figure 3.8-1
Sheet 1 of 2



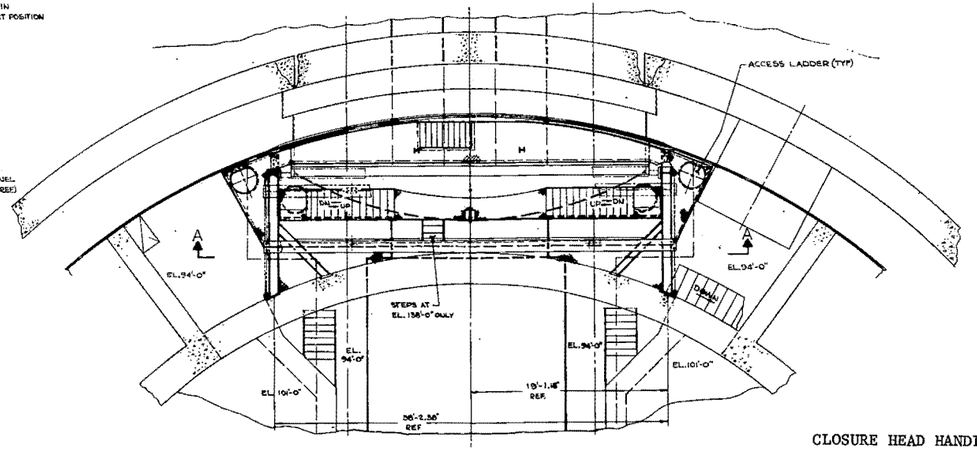
SECTION B-B



SECTION A-A



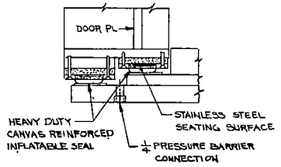
PLAN AT EL. 152'-0"



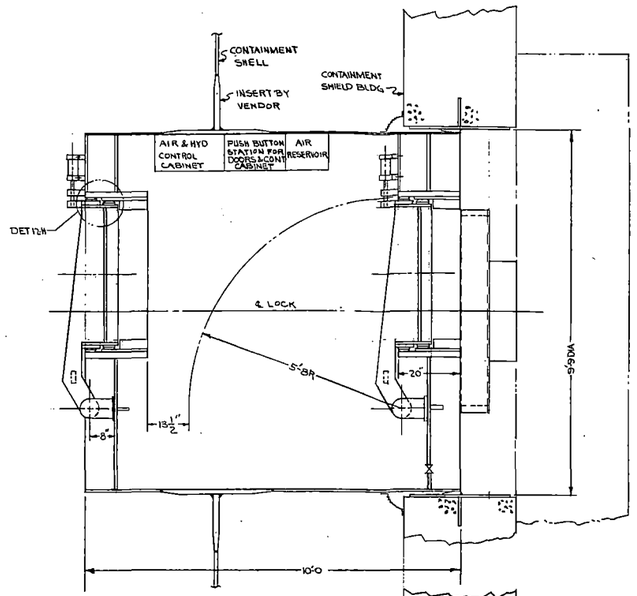
SECTION AT EL. 112'-0"

CLOSURE HEAD HANDLING AND STORAGE ASSEMBLY

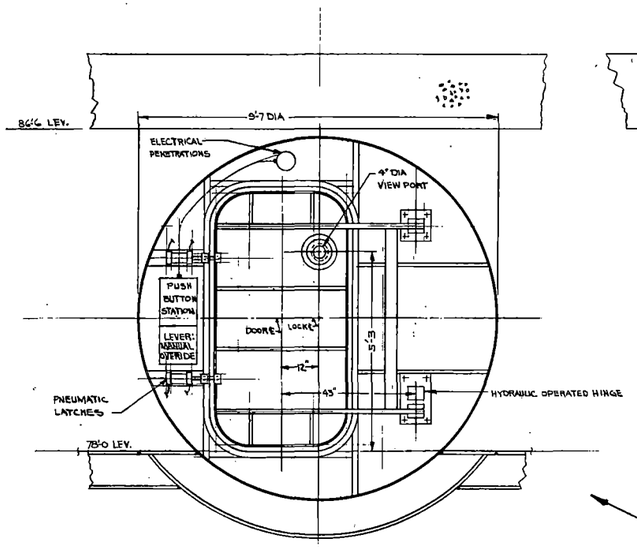
11-1-72 Figure 3.8-1
Sheet 2 of 2



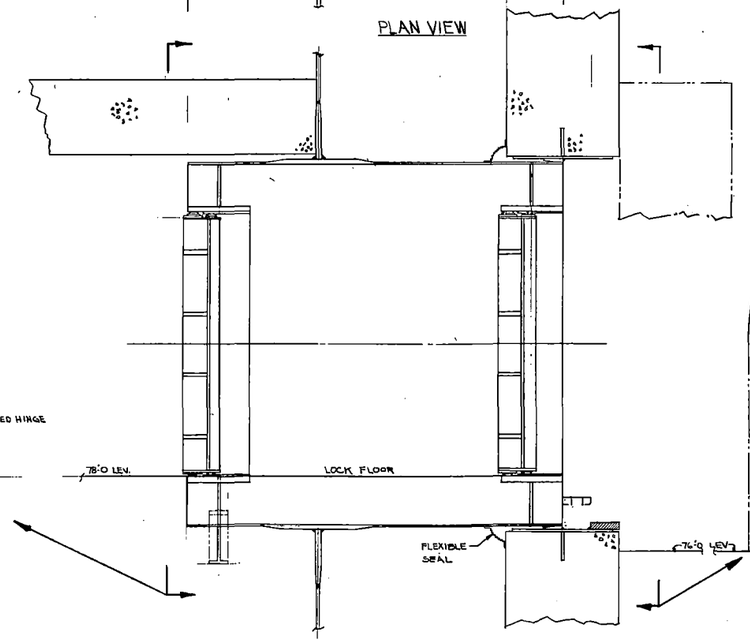
DETAIL 12-H
NO SCALE



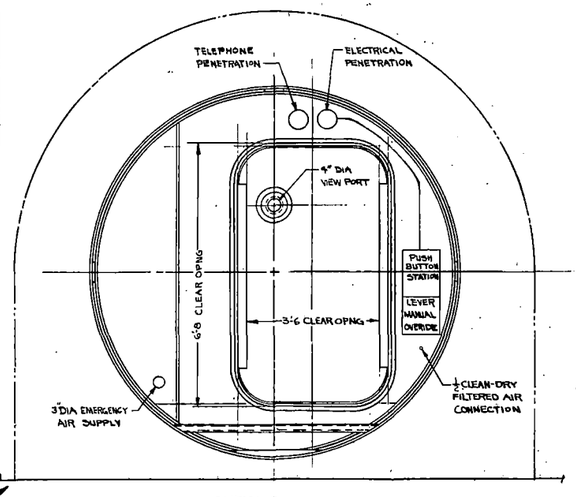
PLAN VIEW



REACTOR SIDE



ELEVATION

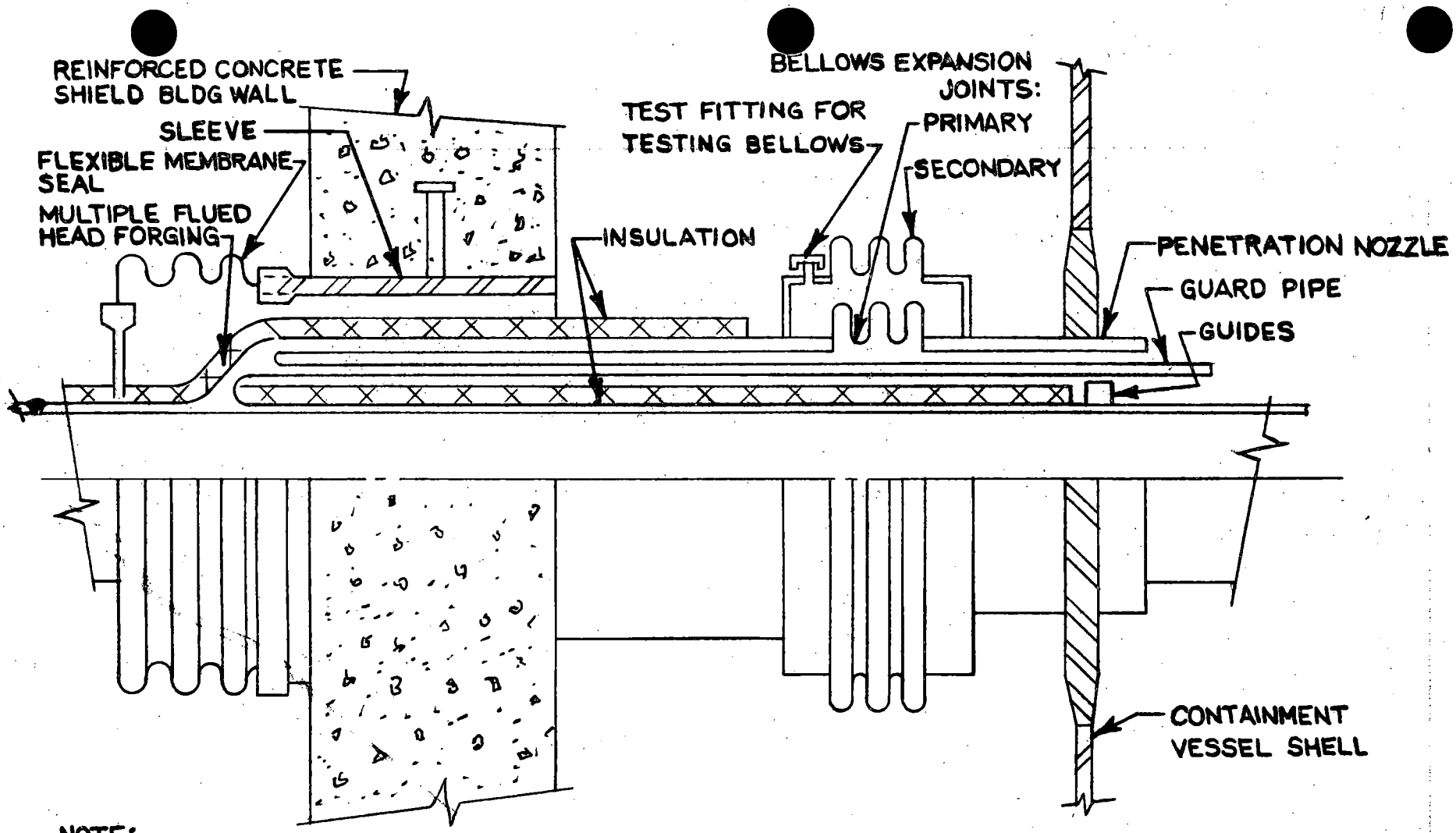


OUTSIDE

OFFSHORE POWER SYSTEMS

Title: PERSONNEL AIR LOCK
GENERAL ARRANGEMENT

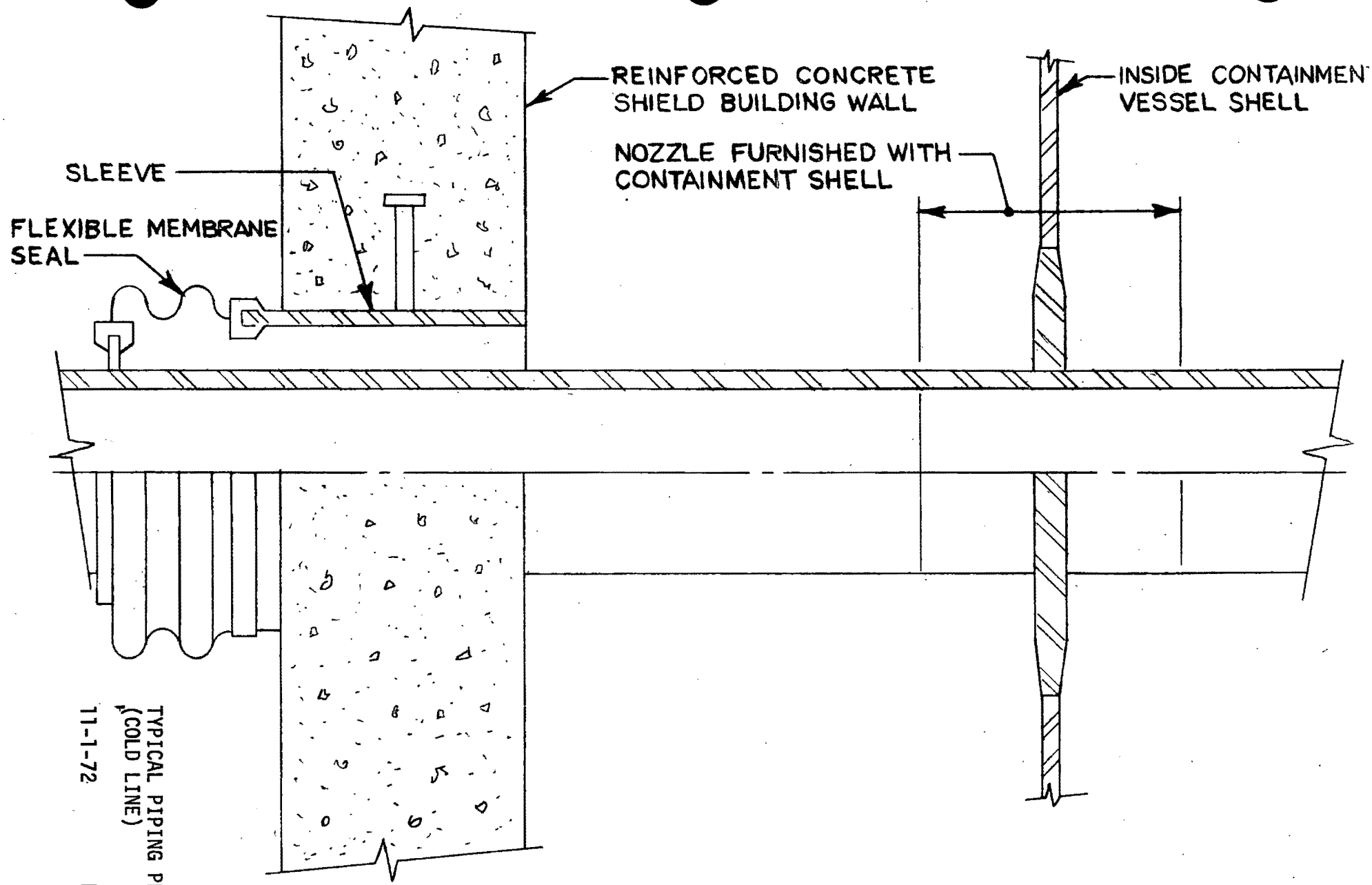
11/1/72 Figure: 3.8-2



NOTE:
PIPE SUPPORTS, STOPS & GUIDES ALLOW PIPE LINE GROWTH AND MOVEMENT IN PENETRATION

TYPICAL PIPING PENETRATION (HOT LINE)

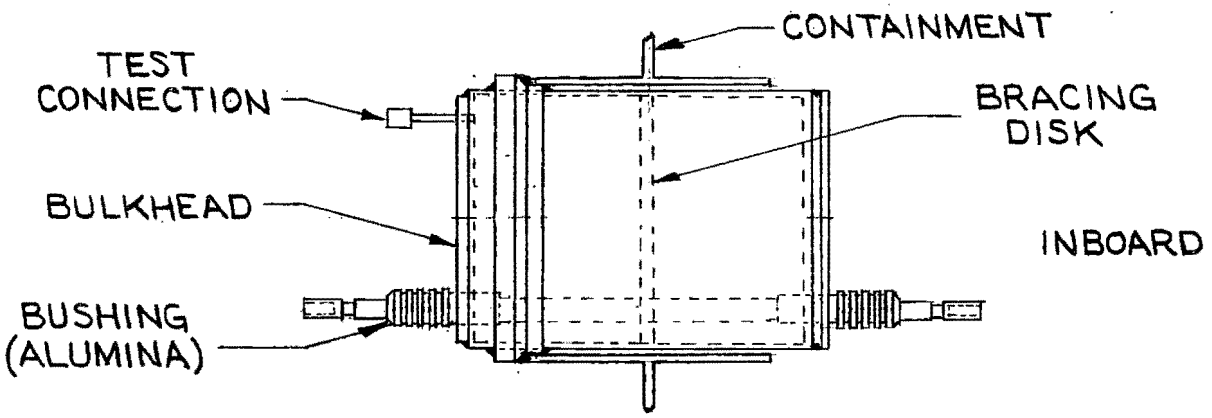
11-1-72 Figure 3.8-3



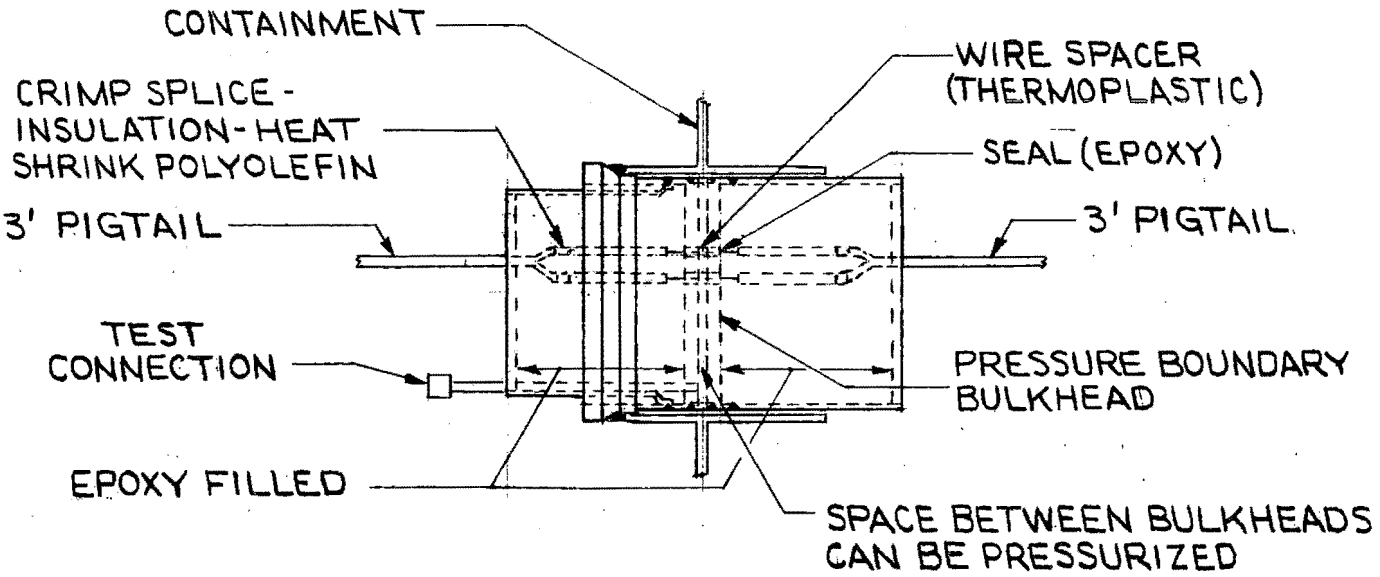
TYPICAL PIPING PENETRATION
(COLD LINE)

11-1-72

Figure 3.8-4



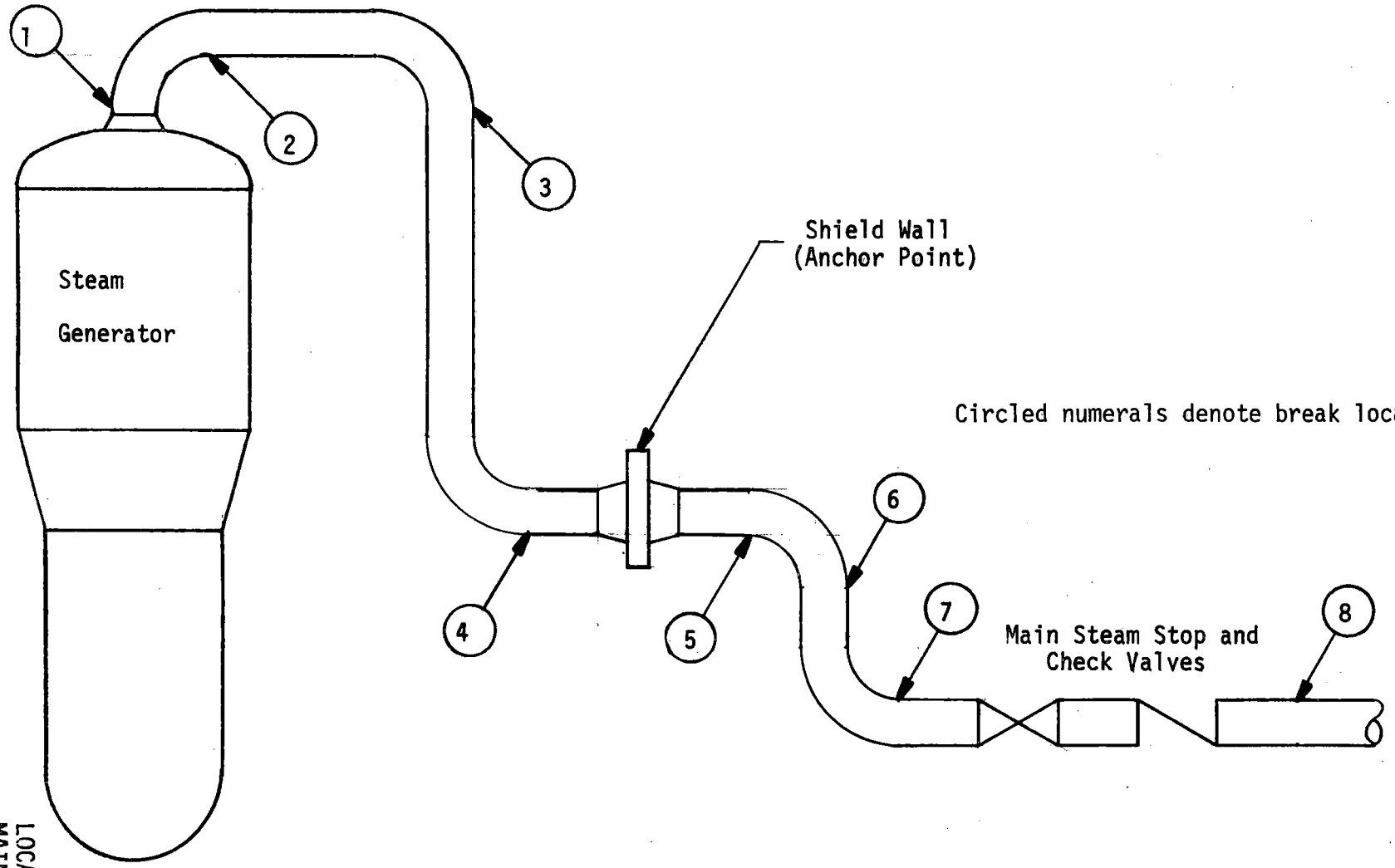
TYPICAL MEDIUM VOLTAGE POWER PENETRATION



TYPICAL LOW VOLTAGE POWER, CONTROL & INSTRUMENTATION PENETRATION

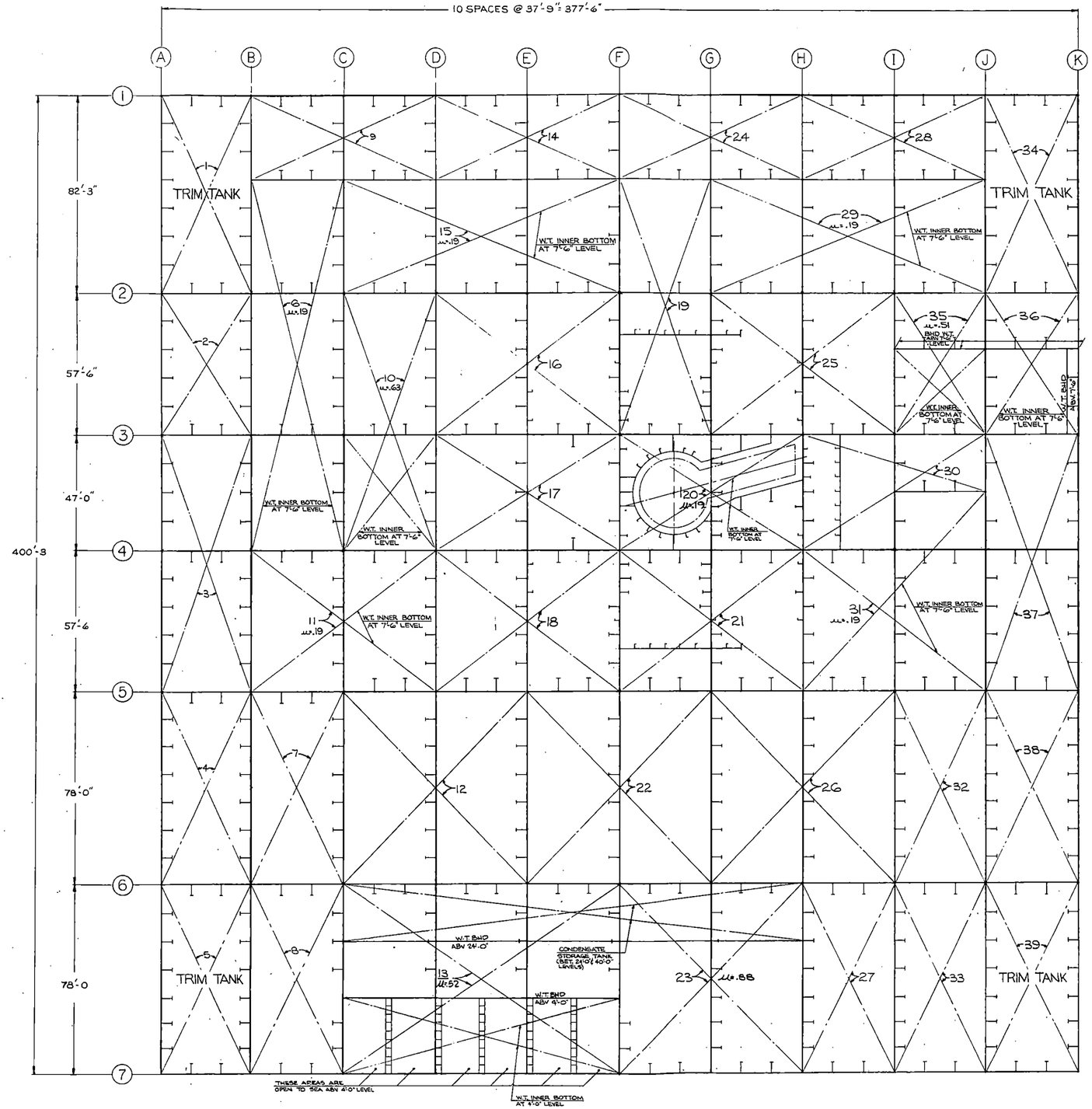
TYPICAL LOW VOLTAGE POWER, CONTROL & INSTRUMENTATION PENETRATION

11-1-72 Figure 3.8-6



LOCATION OF POSTULATED
MAIN STEAM LINE BREAKS

11-T-72 Figure 3.9-1

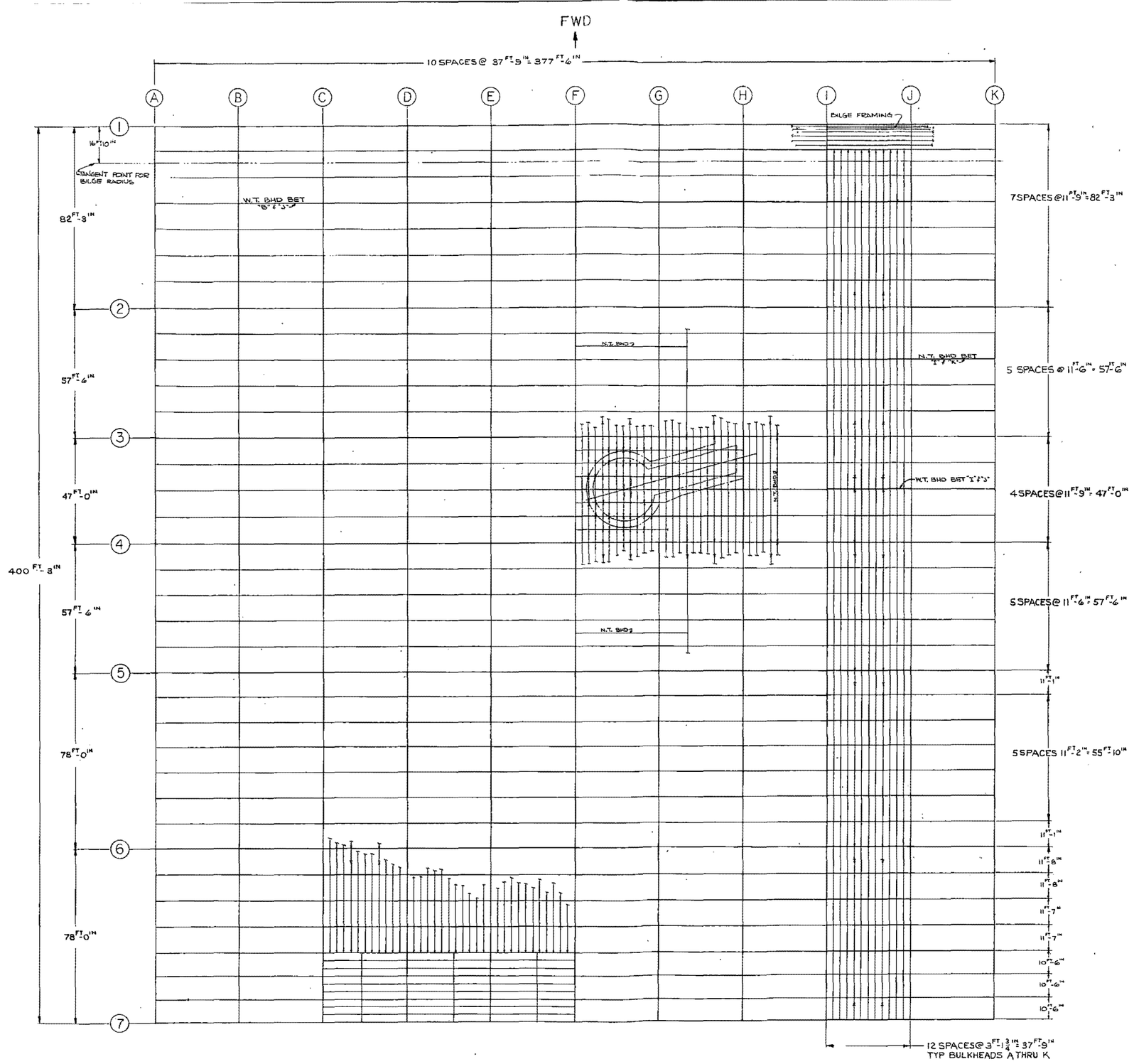


NOTES:
PERMEABILITY, μ = 0.95 EXCEPT AS NOTED.

OFFSHORE POWER SYSTEMS

Title: COMPARTMENTATION DIAGRAM

11/1/72 Figure: 3.12-1



OFFSHORE POWER SYSTEMS

Title: PLAN OF PLATFORM BOTTOM FRAMING

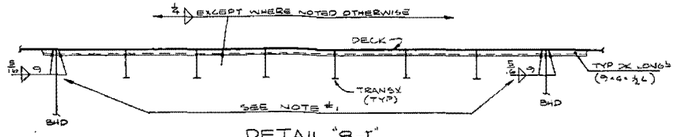
11/1/72 Figure: 3,12-2

NOTES

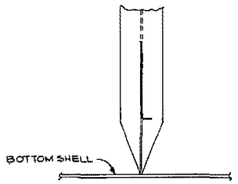
1. WHERE ¼" CONTINUOUS WELDING IS INCREASED TO ⅝" LOCALLY AT THE SUPPORTS OF MEMBERS, THE SMALLER WELD SIZE SHOWN MAY BE USED FOR THE ENTIRE SPAN WHEN DEEP PENETRATION WELDS ARE DEPOSITED WITH AN AUTOMATIC FILLET WELDING MACHINE.

PLATFORM PRINCIPAL DIMENSIONS

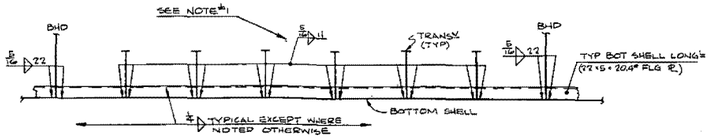
LENGTH	400'-3"
WIDTH	377'-6"
DEPTH	40'-0"
DRAFT	30'-0"
DISPLACEMENT	149,000 SHORT TONS



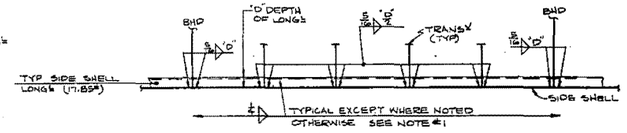
DETAIL "8-I"
TYPICAL FOR WELDING DECK LONGLS
⅜" = 1'-0"



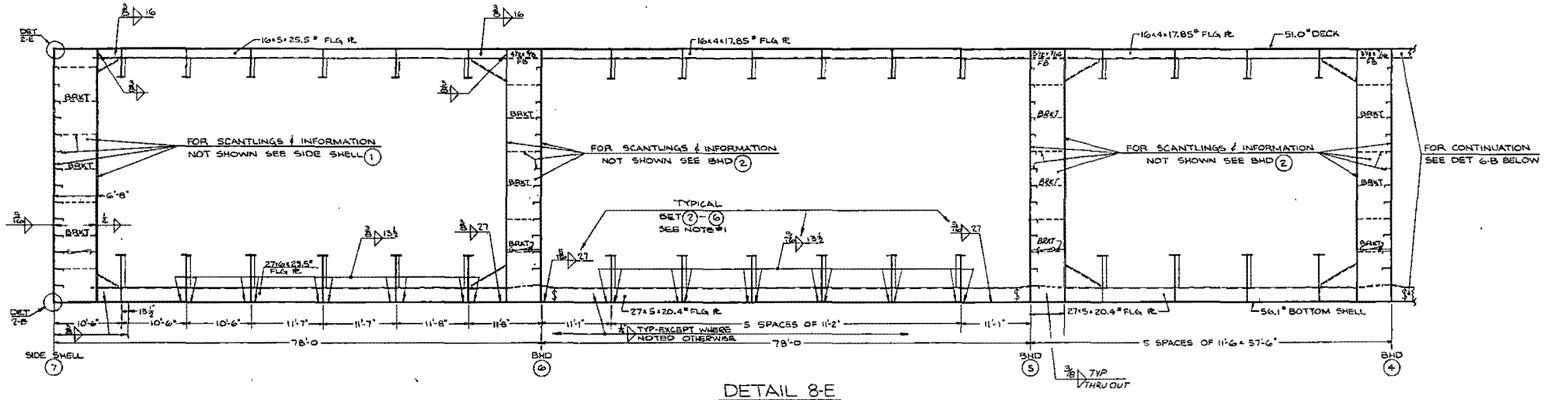
SECT "10-H"
½" = 1'-0"



DETAIL "8-G"
TYPICAL FOR BOTTOM SHELL LONGLS
⅜" = 1'-0"

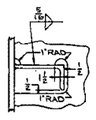


DETAIL "2-G"
TYPICAL FOR WELDING SIDE SHELL LONGLS
⅜" = 1'-0"

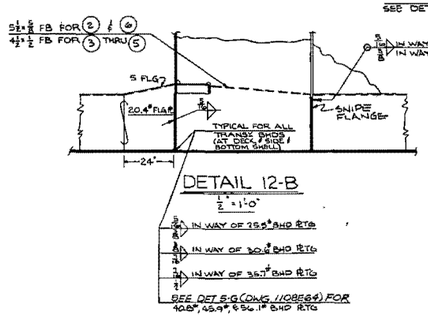


DETAIL 8-E
⅛" = 1'-0"

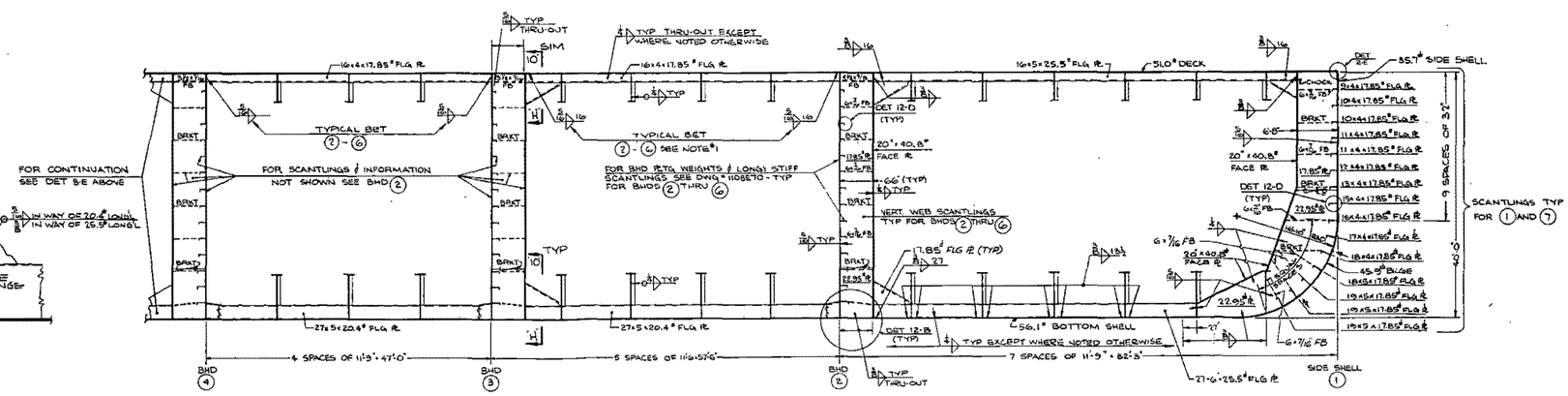
DETAIL 2-E
6" = 1'-0"



DETAIL 12-D
1½" = 1'-0"



DETAIL 12-B
½" = 1'-0"



DETAIL 6-B
⅛" = 1'-0"

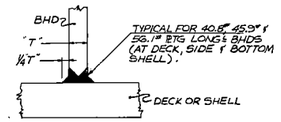
OFFSHORE POWER SYSTEMS

Title: SCANTLING PLAN-
TYPICAL LONGITUDINAL ELEVATION

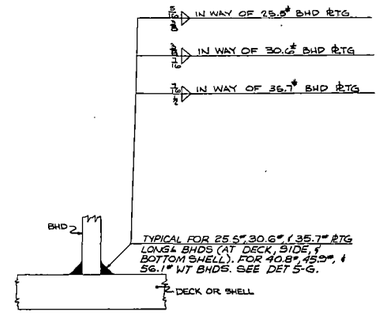
11/1/72 Figure: 3.12-3

PLATFORM PRINCIPAL DIMENSIONS

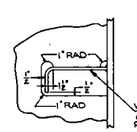
LENGTH	400'-3"
WIDTH	377'-6"
DEPTH	40'-0"
DRAFT	30'-0"
DISPLACEMENT	140,000 SHORT TONS



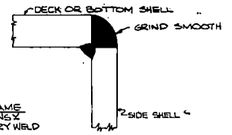
DETAIL 5-G
6" = 1'-0"



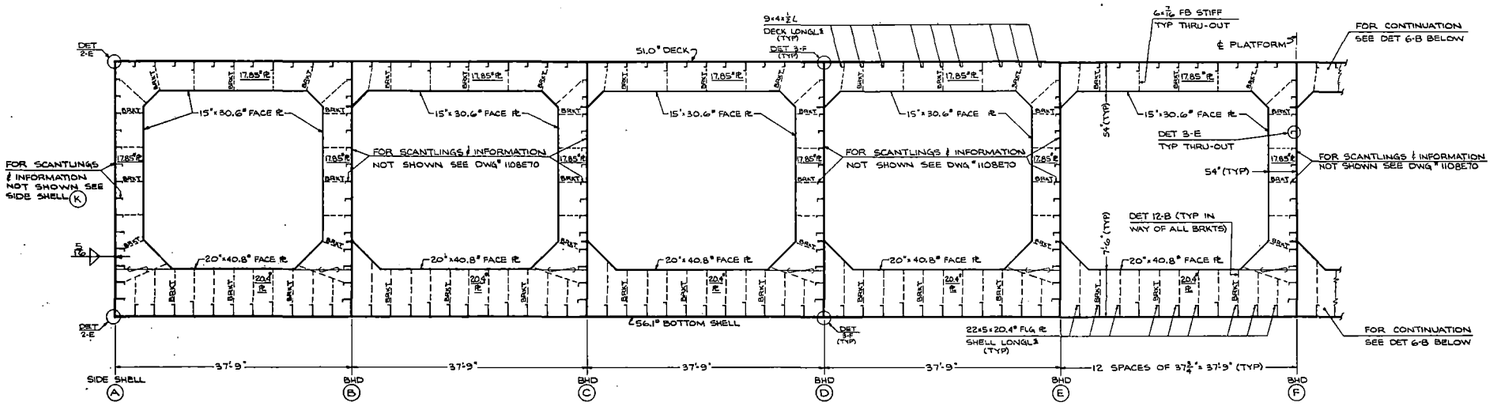
DETAIL 3-F
6" = 1'-0"



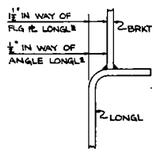
DETAIL 3-E
1½" = 1'-0"



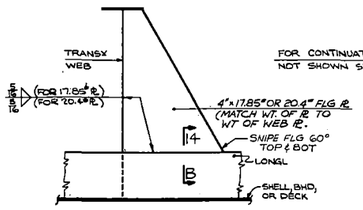
DETAIL 2-E
6" = 1'-0"



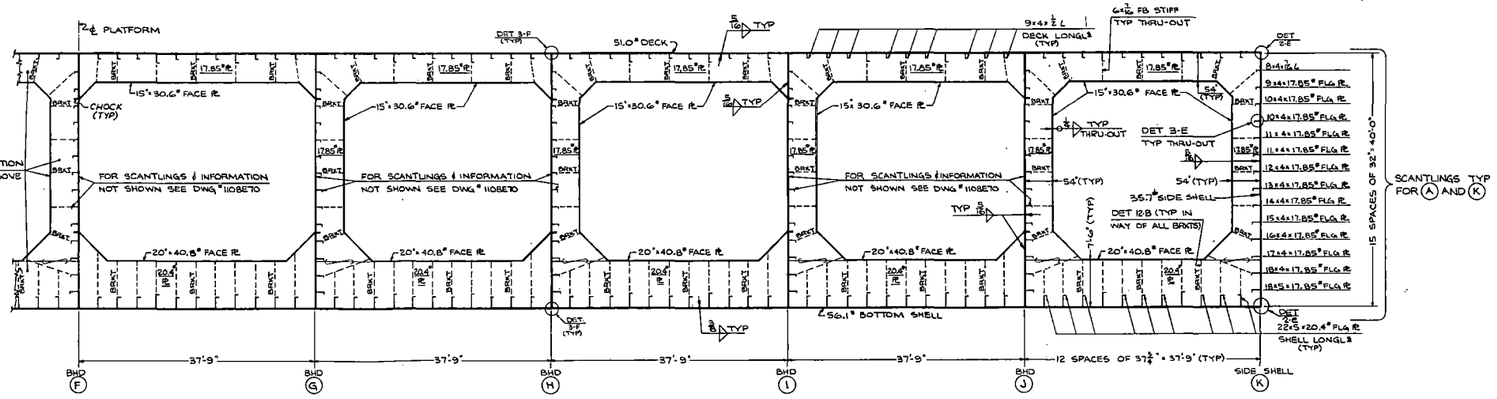
DETAIL 10-D
⅛" = 1'-0"



SECTION 14-B
3" = 1'-0"



DETAIL 12-B
TYPICAL BRACKET
½" = 1'-0"



DETAIL 6-B
⅛" = 1'-0"

OFFSHORE POWER SYSTEMS

Title: SCANTLING PLAN
TYPICAL TRANSVERSE ELEVATION

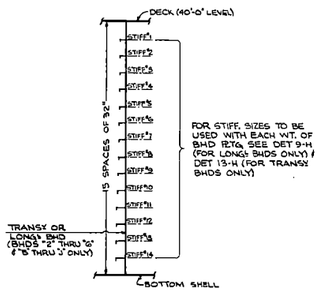
11/1/72 Figure: 3.12-4

SCANTLINGS FOR TRANSV BHD STIFFS			
STIFF.#	BHD PTG WT		
	25.5#	30.6#	35.7#, 45.9# OR 56.1#
1	9"x4"x17.85# FLG PL	9"x4"x17.85 # FLG PL	9"x4"x17.85 # FLG PL
2	10"x4"x17.85#	10"x4"x17.85#	10"x4"x17.85#
3	10"x4"x17.85#	10"x4"x17.85#	10"x4"x17.85#
4	11"x4"x17.85#	11"x4"x17.85#	11"x4"x17.85#
5	11"x4"x17.85#	11"x4"x17.85#	11"x4"x17.85#
6	12"x4"x17.85#	12"x4"x17.85#	12"x4"x17.85#
7	12"x4"x17.85#	12"x4"x17.85#	12"x4"x17.85#
8	13"x4"x17.85#	13"x4"x17.85#	13"x4"x17.85#
9	14"x4"x17.85#	14"x4"x17.85#	14"x4"x17.85#
10	15"x4"x17.85#	15"x4"x17.85#	15"x4"x17.85#
11	16"x4"x17.85#	16"x4"x17.85#	16"x4"x17.85#
12	17"x4"x17.85#	17"x4"x17.85#	17"x4"x17.85#
13	17"x4"x17.85#	17"x4"x17.85#	17"x4"x17.85#
14	18"x4"x17.85# FLG PL	18"x4"x17.85 # FLG PL	18"x4"x17.85 # FLG PL

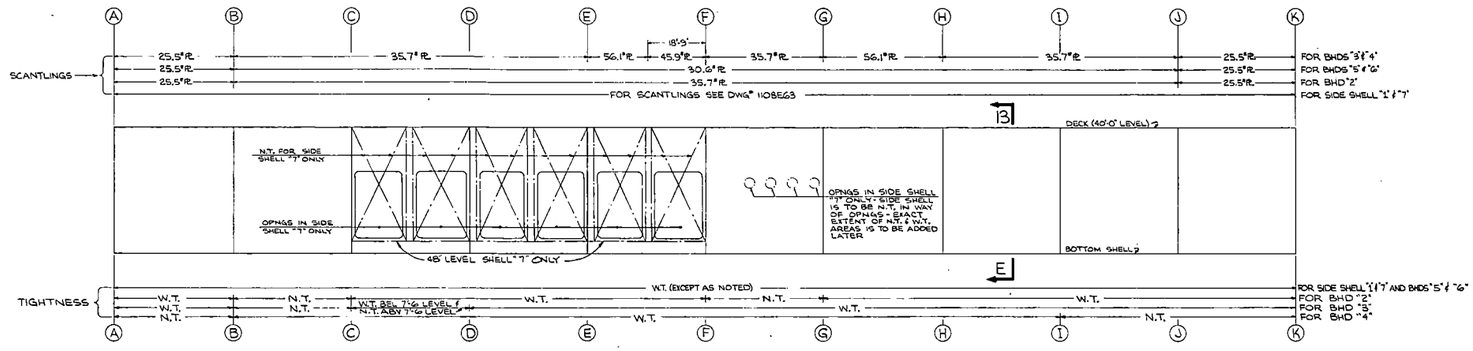
DETAIL 13-H
TABLE OF STIFF. SCANTLINGS
(FOR TRANSV. BHDS ONLY)

SCANTLINGS FOR LONGL BHD STIFFS			
STIFF.#	BHD PTG WT		
	25.5#	30.6#	35.7#, 40.8#, OR 45.9#
1	8"x3 1/2"x L	8"x3 1/2"x L	8"x4 1/2"x L
2	9"x4"x17.85# FLG PL	9"x4"x17.85# FLG PL	9"x4"x17.85# FLG PL
3	10"x4"x17.85#	10"x4"x17.85#	9"x4"x17.85#
4	10"x4"x17.85#	10"x4"x17.85#	10"x4"x17.85#
5	10"x4"x17.85#	10"x4"x17.85#	10"x4"x17.85#
6	11"x4"x17.85#	11"x4"x17.85#	11"x4"x17.85#
7	11"x4"x17.85#	11"x4"x17.85#	11"x4"x17.85#
8	12"x4"x17.85#	12"x4"x17.85#	12"x4"x17.85#
9	13"x4"x17.85#	13"x4"x17.85#	13"x4"x17.85#
10	14"x4"x17.85#	14"x4"x17.85#	14"x4"x17.85#
11	14"x4"x17.85#	14"x4"x17.85#	14"x4"x17.85#
12	15"x4"x17.85#	15"x4"x17.85#	15"x4"x17.85#
13	16"x4"x17.85#	16"x4"x17.85#	16"x4"x17.85#
14	17"x4"x17.85# FLG PL	17"x4"x17.85# FLG PL	17"x4"x17.85# FLG PL

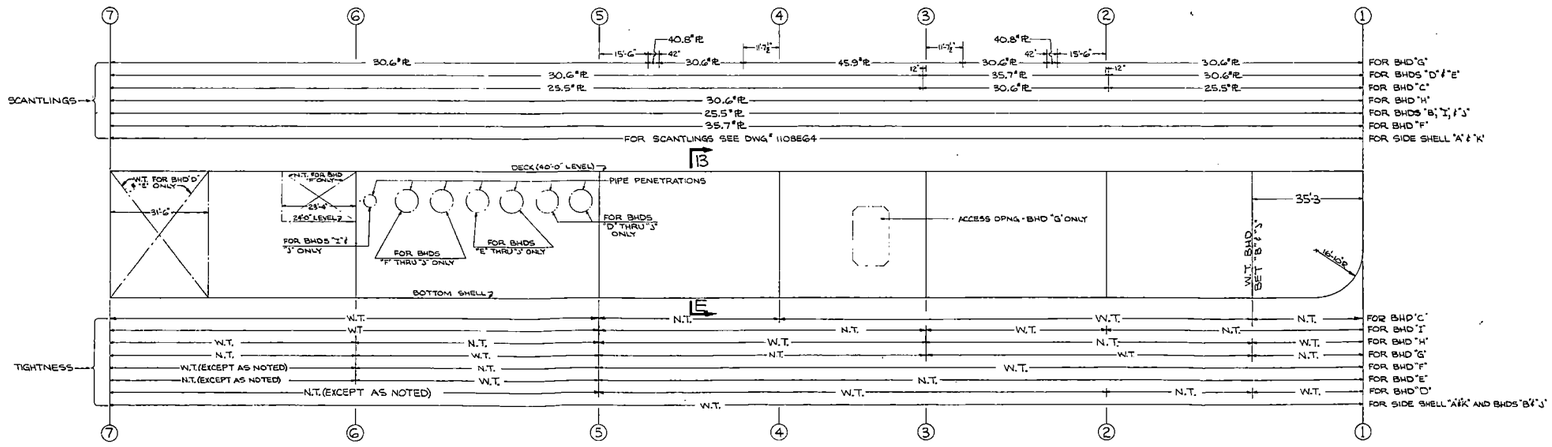
DETAIL 9-H
TABLE OF STIFF. SCANTLINGS
(FOR LONGV BHDS ONLY)



SECTION 13-E
3/8"=1'-0"



DETAIL 6-E
TRANSV SIDE SHELL & BHD SCANTLINGS
& TIGHTNESS DIAGRAM
1/2"=1'-0"

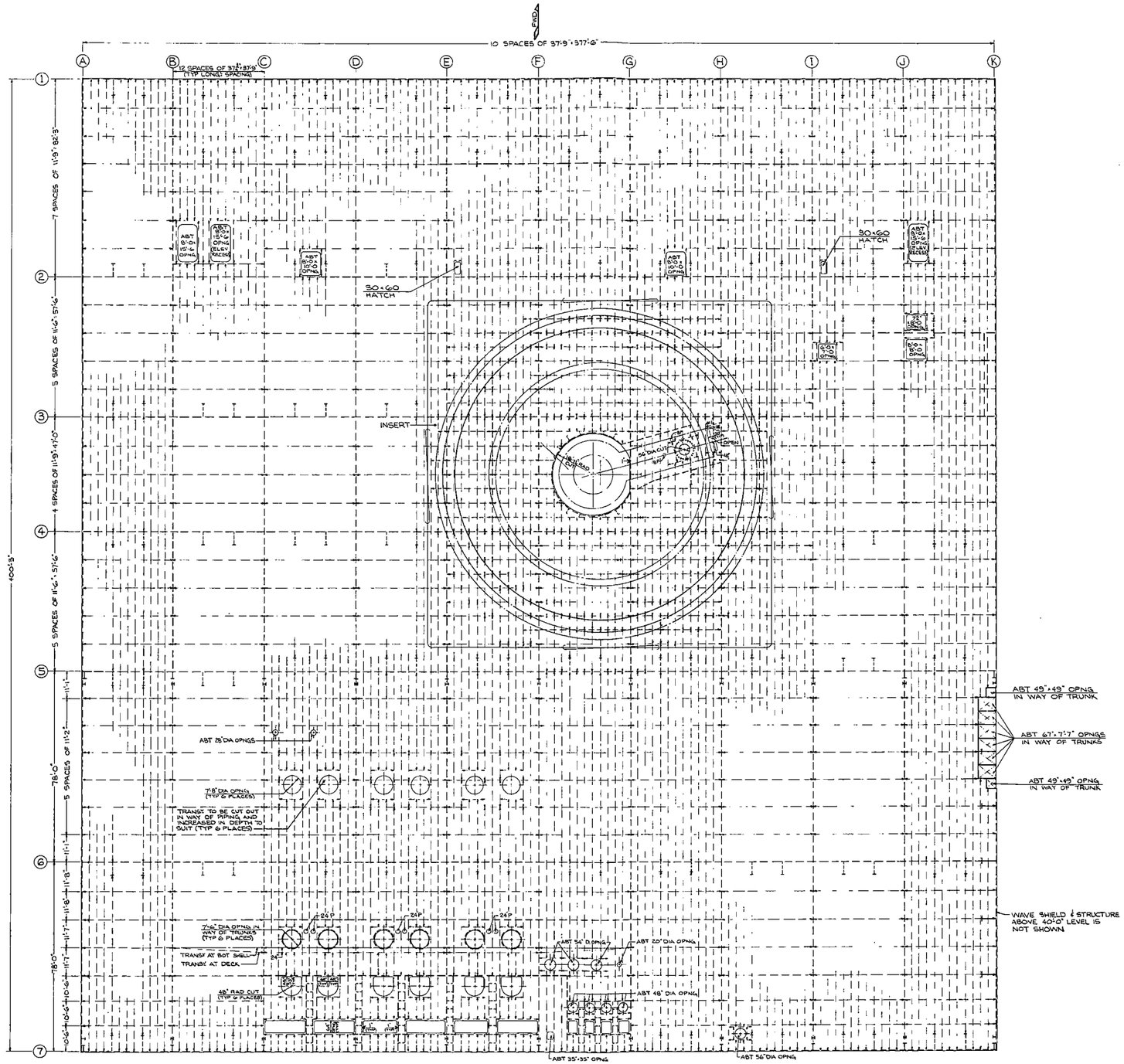


DETAIL 6-B
LONGL SIDE SHELL & BHD SCANTLINGS
& TIGHTNESS DIAGRAM
1/2"=1'-0"

OFFSHORE POWER SYSTEMS

Title: SCANTLING PLAN AND TIGHTNESS DIAGRAM FOR BULKHEADS AND SIDE SHELL

11/1/72 Figure: 3.12-5

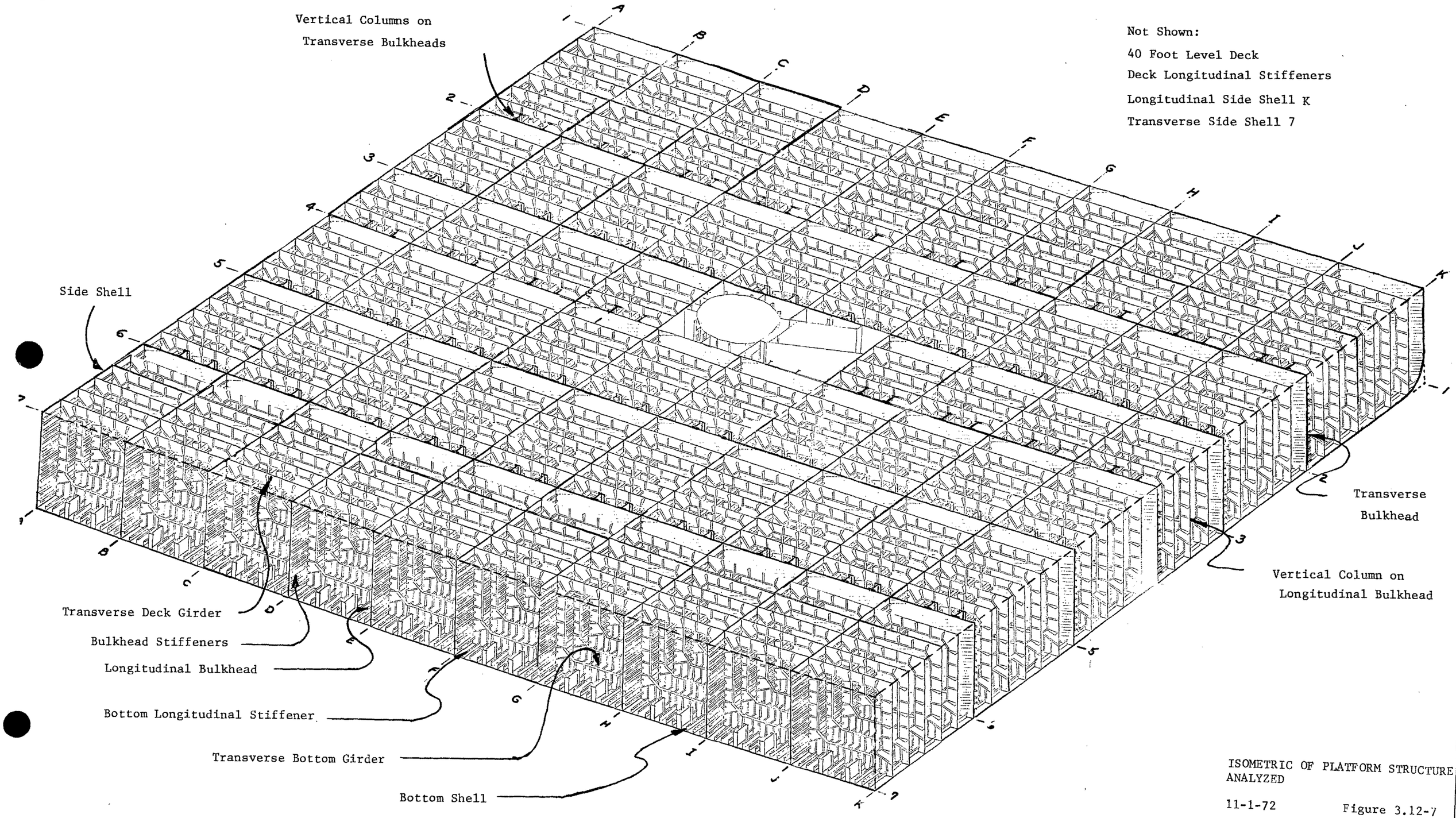


DETAIL 7-B
PLAN VIEW AT 40'-0" LEVEL

OFFSHORE POWER SYSTEMS

Title: PLAN OF 40'-0" LEVEL FRAMING

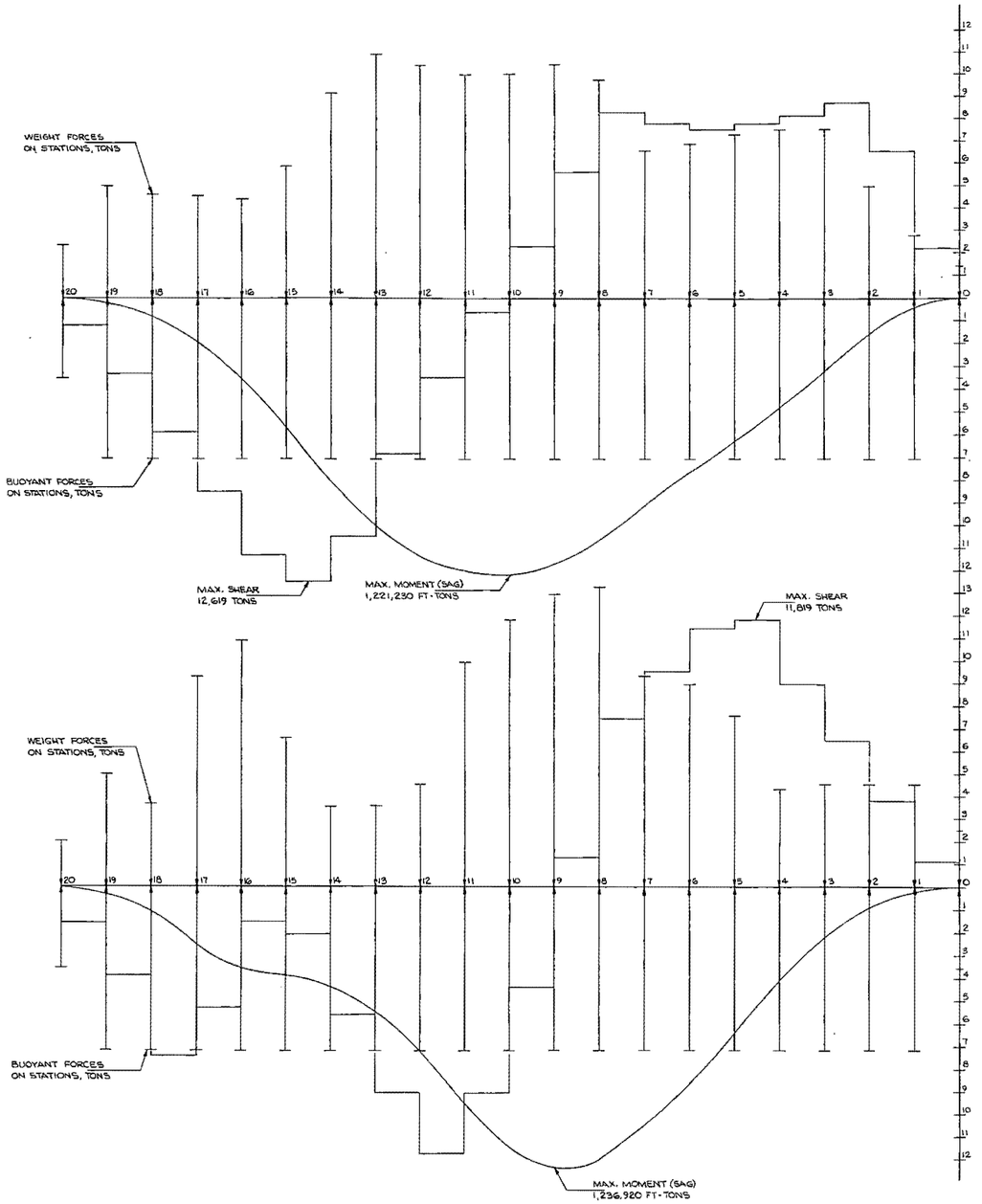
11/1/72 Figure: 3.12-6



ISOMETRIC OF PLATFORM STRUCTURE ANALYZED

11-1-72

Figure 3.12-7



TRANSVERSE STRENGTH

BENDING STRESSES

MAXIMUM MOMENT	1,221,230 FT-TONS
AVERAGE STRESS IN DECK	8.51 KIPS/IN^2
AVERAGE STRESS IN KEEL	8.46 KIPS/IN^2

MEAN DRAFT, FWD 29'-9$\frac{3}{64}$"; AFT 29'-9$\frac{63}{64}$"

LONGITUDINAL STRENGTH

BENDING STRESSES

MAXIMUM MOMENT	1,236,920 FT-TONS
AVERAGE STRESS IN DECK	8.64 KIPS/IN^2
AVERAGE STRESS IN KEEL	8.31 KIPS/IN^2

MEAN DRAFT, FWD 29'-9$\frac{1}{64}$"; AFT 29'-9$\frac{53}{64}$"

PRINCIPLE DIMENSIONS

LENGTH OVERALL	400'-8"
BEAM	377'-6"
DEPTH	40'-0"
DRAFT	30'-0"

SECTION MODULUS USED WITH LONGL BENDING

DECK	3,435,000 IN^3
BOTTOM	3,512,000 IN^3

SECTION MODULUS USED WITH TRANSV BENDING

DECK	3,448,000 IN^3
BOTTOM	3,464,000 IN^3

SCALES

HORIZONTAL

LONGITUDINAL	1 IN. = 20.0125 FT.
TRANSVERSE	1 IN. = 18.875 FT.

VERTICAL

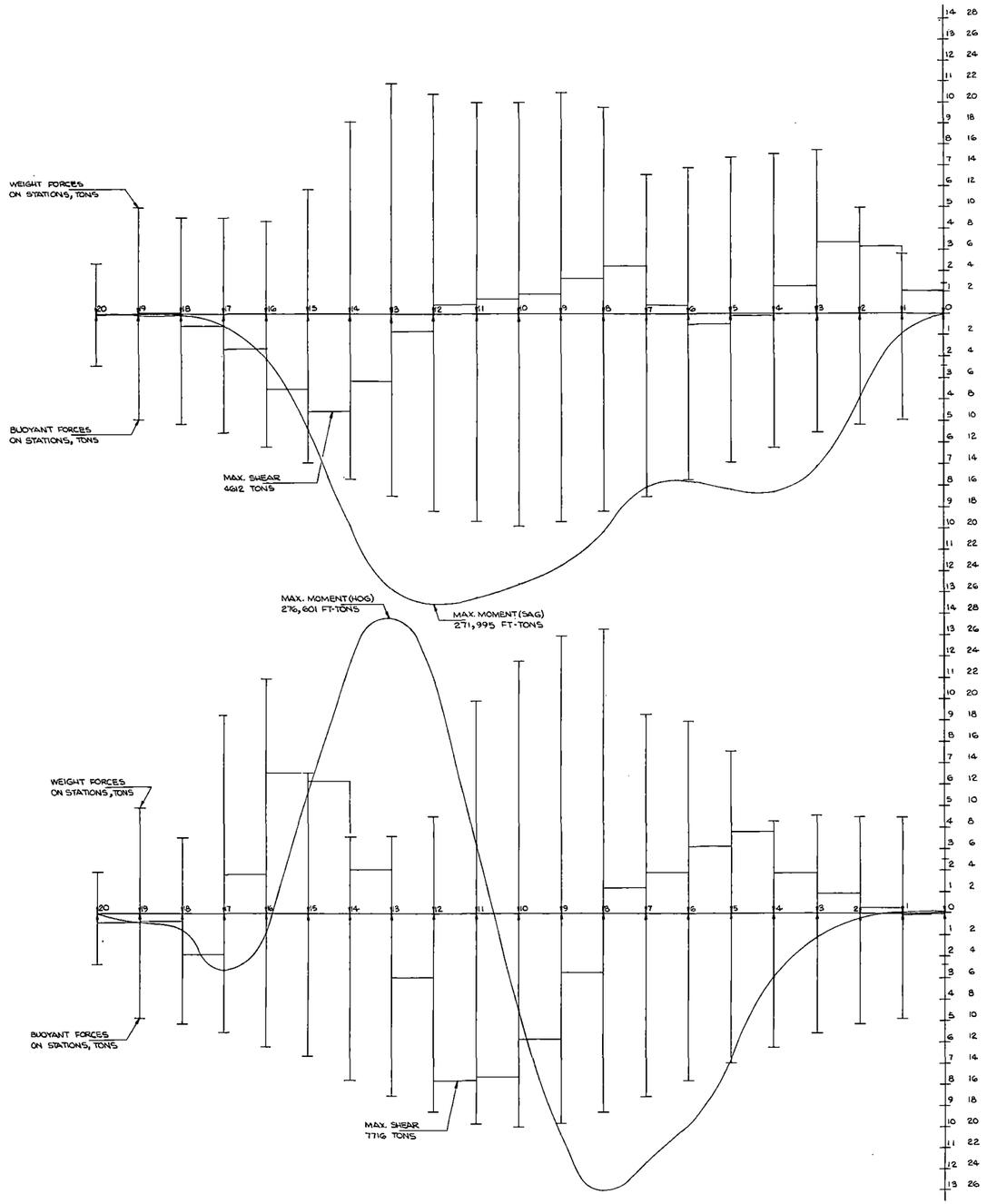
WEIGHT	1 IN. = 2000 TONS
BUOYANCY	1 IN. = 2000 TONS
SHEAR	1 IN. = 2000 TONS
MOMENT	1 IN. = 200,000 FT-TONS

STILL WATER SHEAR AND MOMENT CURVES

Title: MOMENT AND SHEAR CURVES ,
DIMENSIONAL ANALYSIS

11/1/72

Figure 3.12-8
Sheet 1 of 3



TRANSVERSE STRENGTH

BENDING STRESSES

MAXIMUM MOMENT	271,995 FT-TONS
AVERAGE STRESS IN DECK	1.90 KIPS/IN^2
AVERAGE STRESS IN KEEL	1.88 KIPS/IN^2
MEAN DRAFT, FWD	20'-0 13/64"; AFT 20'-0 13/32"

SCALES

HORIZONTAL

LONGITUDINAL	1 IN. = 20.0125 FT
TRANSVERSE	1 IN. = 18.875 FT

VERTICAL

WEIGHT	1 IN. = 2000 TONS
BUOYANCY	1 IN. = 2000 TONS
SHEAR	1 IN. = 2000 TONS
MOMENT	1 IN. = 40,000 FT-TONS

HOGGING TROCHOIDAL WAVE SHEAR AND MOMENT CURVES

LONGITUDINAL STRENGTH

BENDING STRESSES

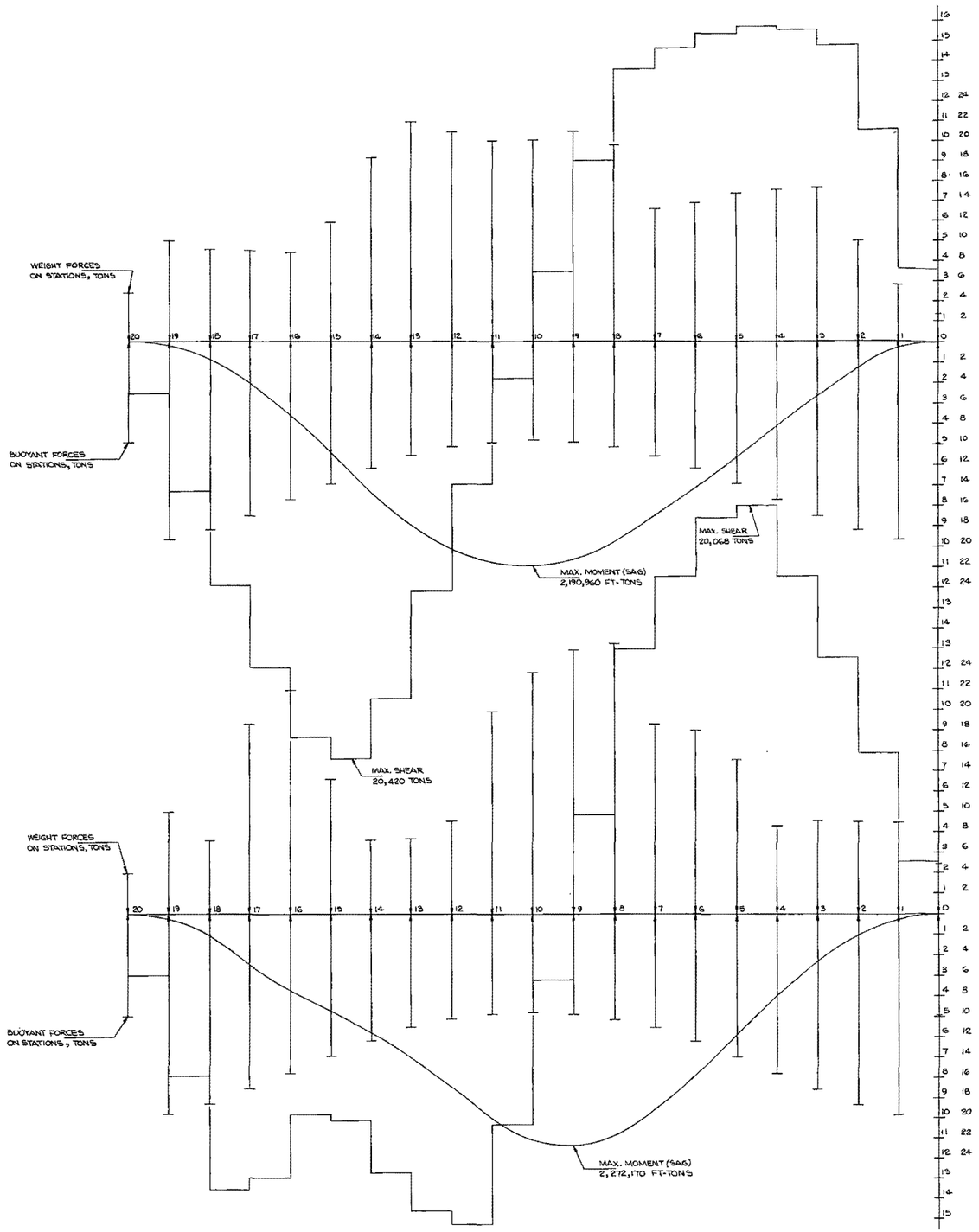
MAXIMUM MOMENT	276,601 FT-TONS
AVERAGE STRESS IN DECK	1.93 KIPS/IN^2
AVERAGE STRESS IN KEEL	1.86 KIPS/IN^2
MEAN DRAFT, FWD	19'-8 53/64"; AFT 19'-8 49/64"

OFFSHORE POWER SYSTEMS

Title: MOMENT AND SHEAR CURVES FOR ONE-DIMENSIONAL ANALYSIS

11/1/72

Figure 3.12-8
Sheet 2 of 3



TRANSVERSE STRENGTH

BENDING STRESSES

MAXIMUM MOMENT	2,190,960 FT-TONS
AVERAGE STRESS IN DECK	15.26 KIPS/IN^2
AVERAGE STRESS IN KEEL	15.18 KIPS/IN^2

MEAN DRAFT, FWD 41'-4 $\frac{23}{24}$"; AFT 41'-5"

LONGITUDINAL STRENGTH

BENDING STRESSES

MAXIMUM MOMENT	2,272,170 FT-TONS
AVERAGE STRESS IN DECK	15.88 KIPS/IN^2
AVERAGE STRESS IN KEEL	15.26 KIPS/IN^2

MEAN DRAFT, FWD 41'-8 $\frac{19}{32}$"; AFT 41'-8 $\frac{27}{32}$"

SCALES

HORIZONTAL

LONGITUDINAL	1 IN. = 20.0125 FT
TRANSVERSE	1 IN. = 18.875 FT

VERTICAL

WEIGHT	1 IN. = 2000 TONS
BUOYANCY	1 IN. = 2000 TONS
SHEAR	1 IN. = 2000 TONS
MOMENT	1 IN. = 400,000 FT-TONS

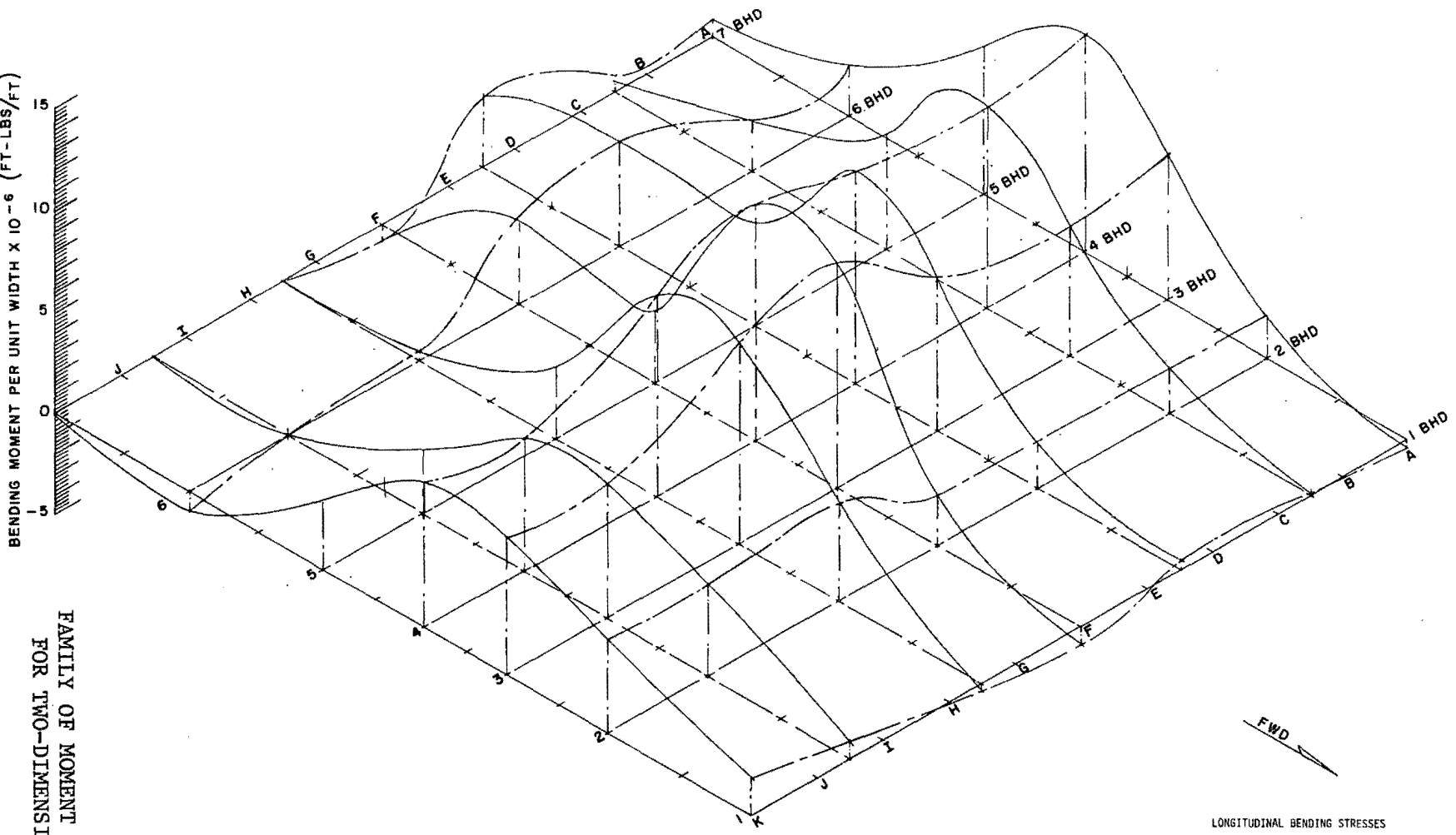
SAGGING TROCHODAL WAVE SHEAR AND MOMENT CURVES

OFFSHORE POWER SYSTEMS

Title: MOMENT AND SHEAR CURVES FOR ONE-DIMENSIONAL ANALYSIS

11/1/72

Figure 3.12-8
Sheet 3 of 3



DISTRIBUTION OF LONGITUDINAL BENDING MOMENTS

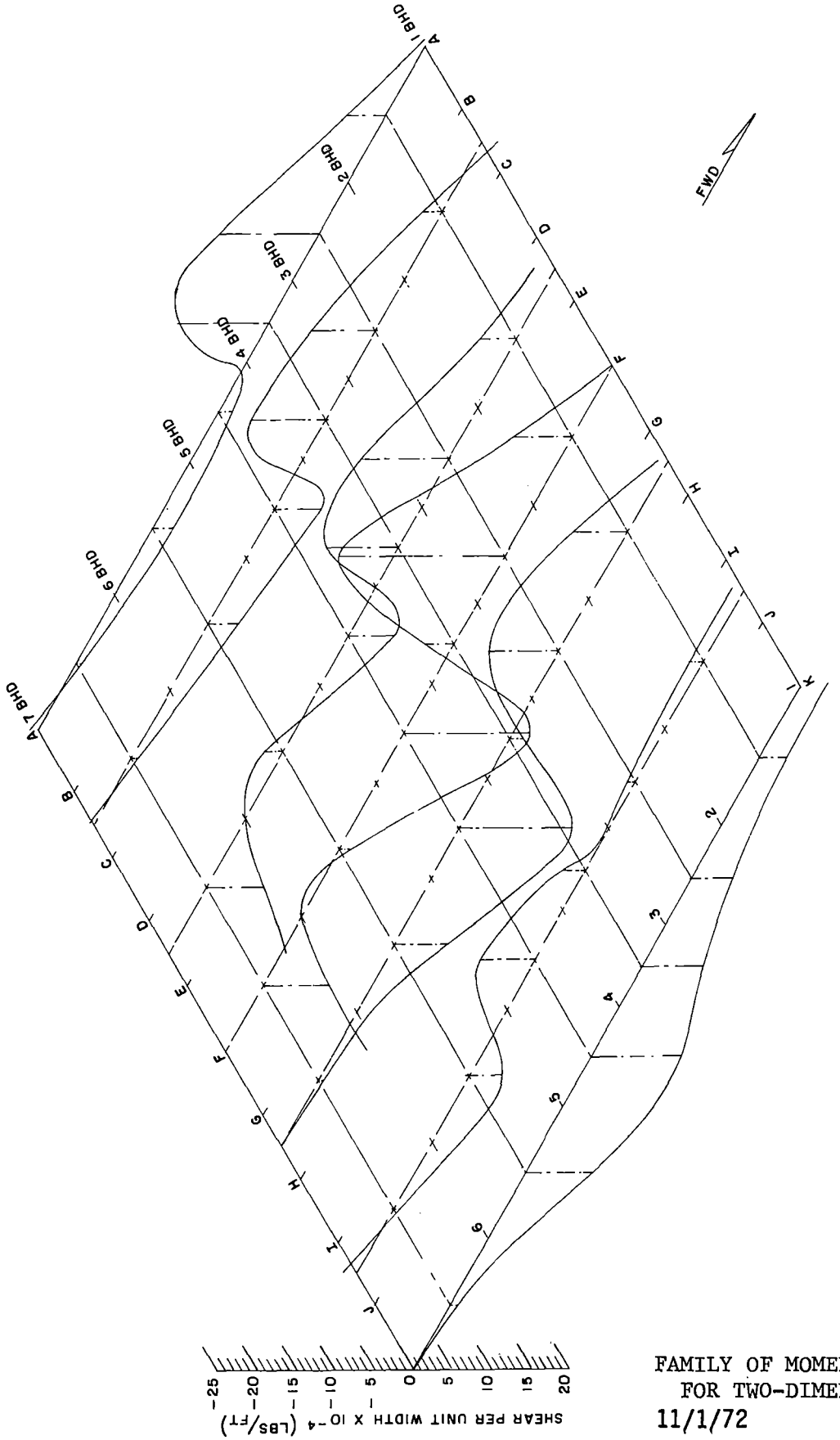
LONGITUDINAL BENDING STRESSES

Element Bulkhead	Maximum Bending Moment (in - lb)	Section Modulus Deck (in^3)	Section Modulus Bottom (in^3)	Maximum Stress Deck (psi)	Maximum Stress Bottom (psi)
A	2.458×10^9	$.1970 \times 10^6$	$.2251 \times 10^6$	12,475	10,915
B - C	9.160×10^9	$.6836 \times 10^6$	$.8011 \times 10^6$	13,398	11,432
D - E	9.471×10^9	$.6927 \times 10^6$	$.8118 \times 10^6$	13,672	11,666
F	5.387×10^9	$.3521 \times 10^6$	$.4092 \times 10^6$	15,298	13,163
G - H	9.167×10^9	$.6927 \times 10^6$	$.8118 \times 10^6$	13,233	11,291
I - J	6.052×10^9	$.6836 \times 10^6$	$.8011 \times 10^6$	8,852	7,753
K	1.629×10^9	$.1970 \times 10^6$	$.2251 \times 10^6$	8,265	7,232

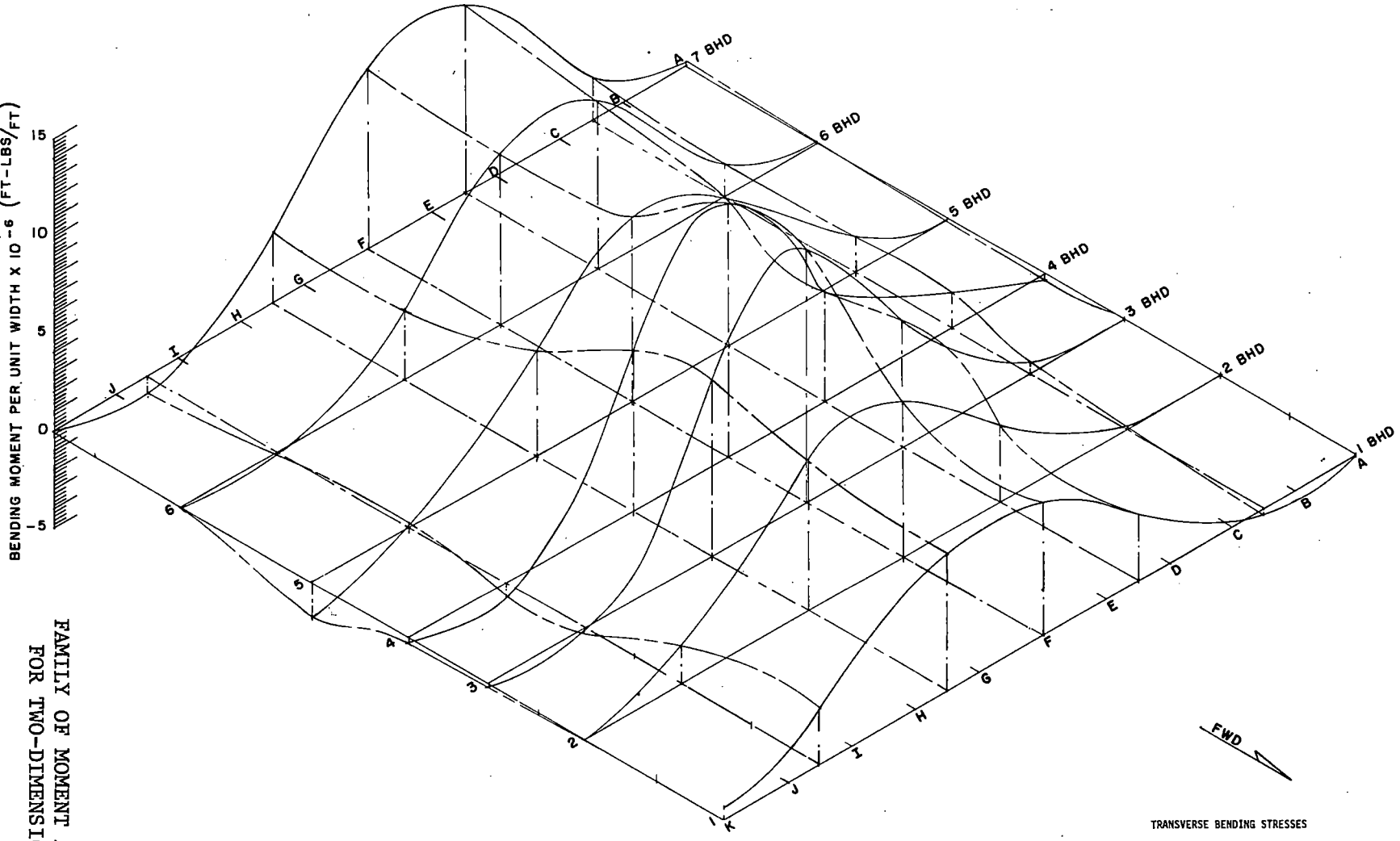
FAMILY OF MOMENT AND SHEAR CURVES
FOR TWO-DIMENSIONAL ANALYSIS

11/1/72

Figure 3.12-9
Sheet 1



FAMILY OF MOMENT AND SHEAR CURVES
FOR TWO-DIMENSIONAL ANALYSIS
11/1/72 Figure 3.12-9
Sheet 2



DISTRIBUTION OF TRANSVERSE BENDING MOMENTS

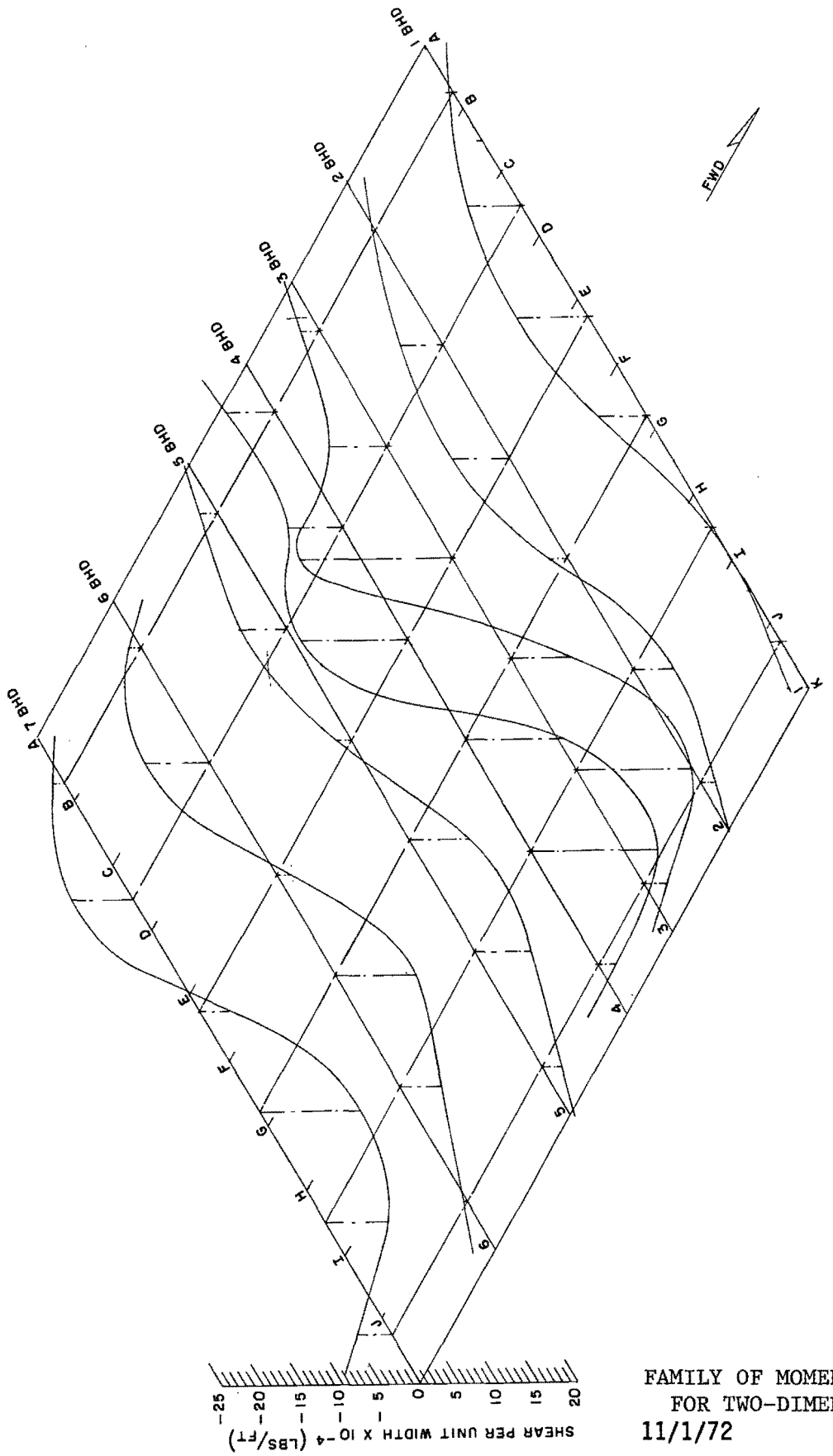
FAMILY OF MOMENT AND SHEAR CURVES
FOR TWO-DIMENSIONAL ANALYSIS

11/1/72

Figure 3.12-9
Sheet 3

TRANSVERSE BENDING STRESSES

Element Bulkhead	Maximum Bending Moment (in - lb)	Section Modulus Deck (in^3)	Section Modulus Bottom (in^3)	Maximum Stress Deck (psi)	Maximum Stress Bottom (psi)
1	3.383×10^9	$.3734 \times 10^6$	$.4267 \times 10^6$	9,060	7,927
2	6.641×10^9	$.6047 \times 10^9$	$.6911 \times 10^6$	10,982	9,609
3	8.139×10^9	$.5032 \times 10^6$	$.5032 \times 10^6$	15,997	15,997
4	8.148×10^9	$.5032 \times 10^6$	$.5032 \times 10^6$	16,016	16,016
5	7.730×10^9	$.5851 \times 10^6$	$.6687 \times 10^6$	13,210	11,559
6	8.328×10^9	$.6718 \times 10^6$	$.7678 \times 10^6$	12,394	10,845
7	4.334×10^9	$.3579 \times 10^6$	$.4690 \times 10^6$	12,111	10,597



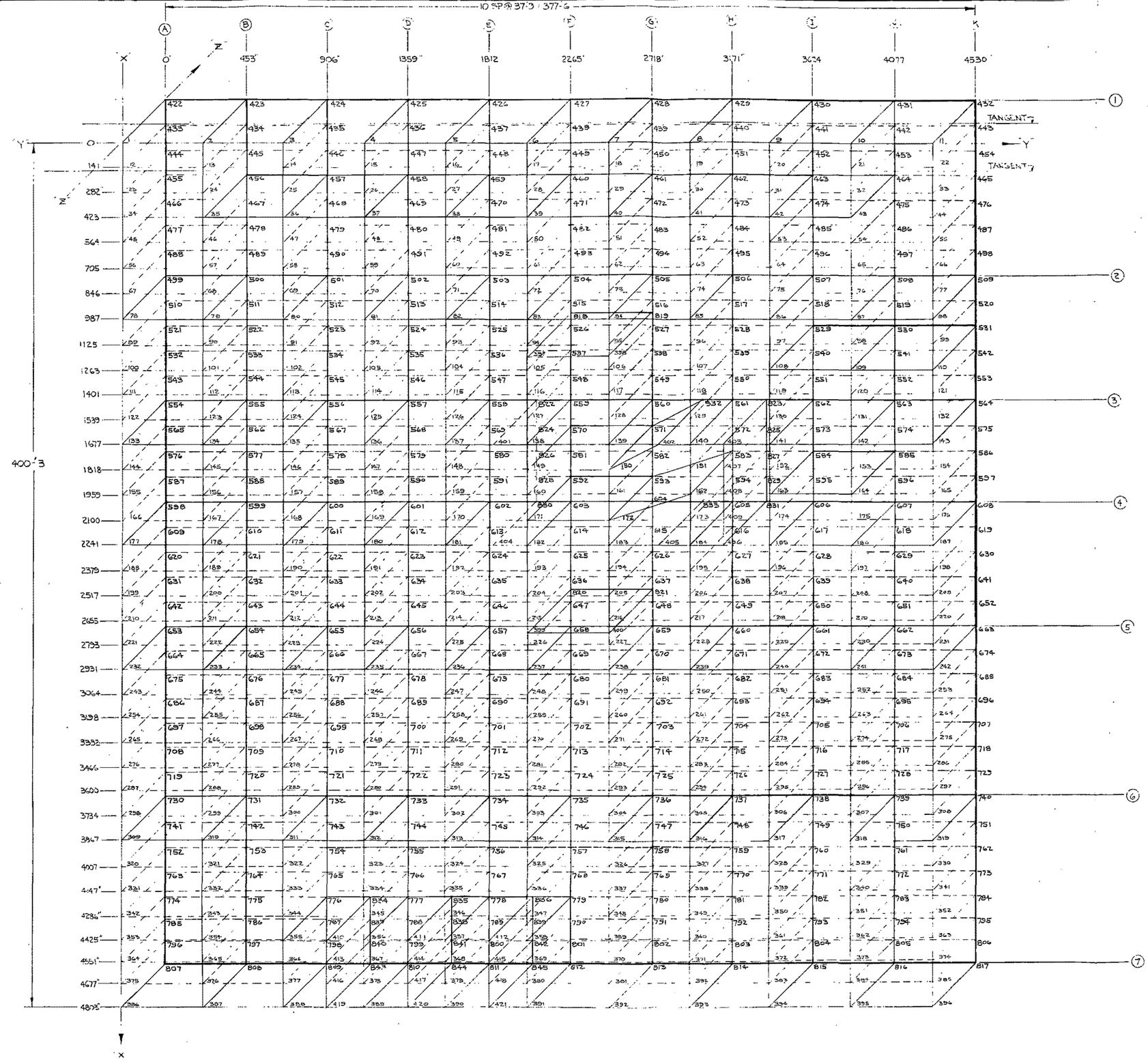
DISTRIBUTION OF TRANSVERSE SHEAR

FAMILY OF MOMENT AND SHEAR CURVES
FOR TWO-DIMENSIONAL ANALYSIS

11/1/72

Figure 3.12-9

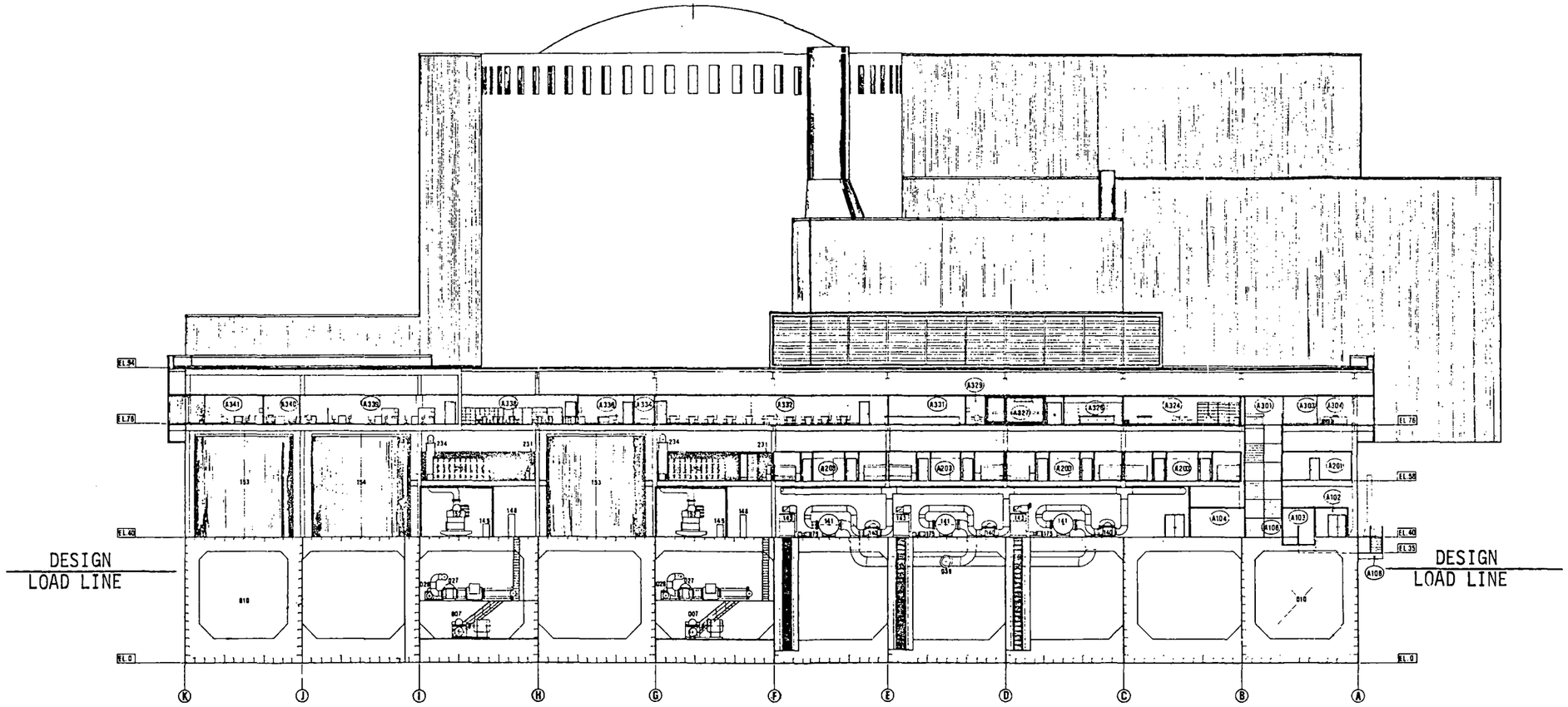
Sheet 4



OFFSHORE POWER SYSTEM

Title: 3 DIMENSIONAL STRESS ANALYSIS-ARRANGEMENT OF NODES

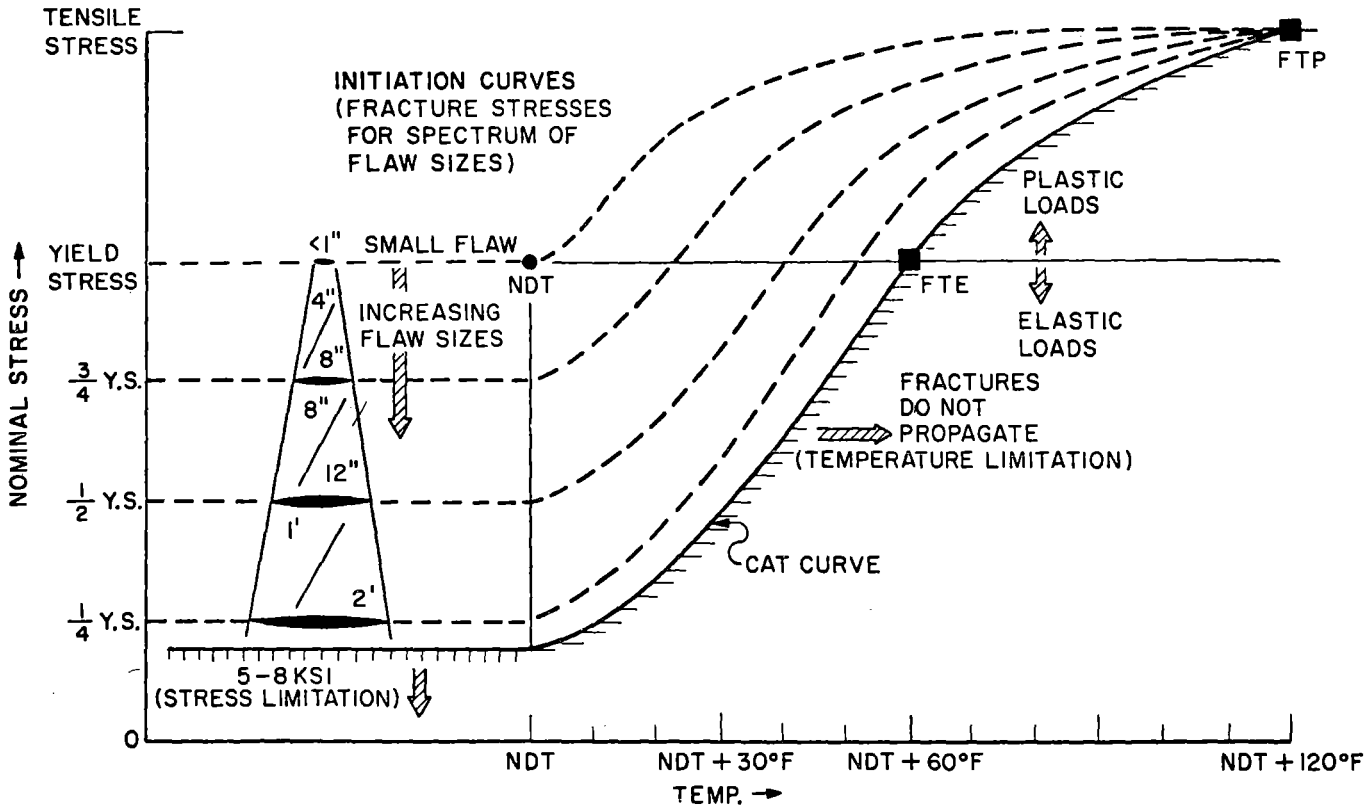
11/1/72 Figure: 3.12-10



OFFSHORE POWER SYSTEMS

Title: ELEVATION SHOWING DESIGN LOAD LINE

11/1/72 Figure 3.12-11



GENERALIZED FRACTURE ANALYSIS DIAGRAM, AS REFERENCED BY THE NDT TEMPERATURE

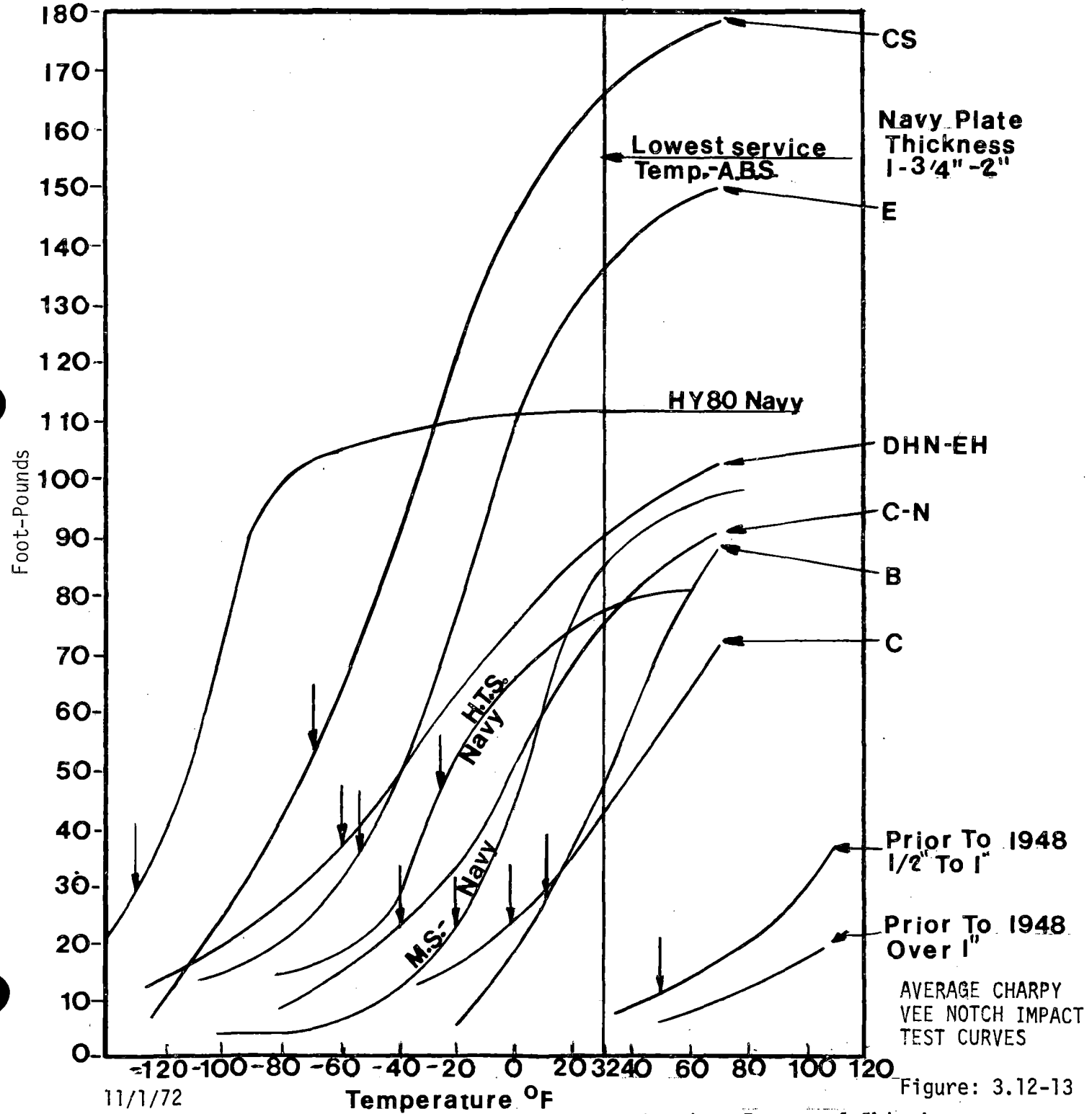
11/1/71

Figure: 3.12-12

HULL STRUCTURAL STEEL

	Ordinary Strength							Higher Strength
	Prior to 1948		Present ABS Grades					Present ABS Grades
	1/2" to 1"	Over 1"	B	C	C-N	CS	E	DHN-EH
No. of Plates -	93	8	61	23	24	25	56	32
Avg. Gage -	.72"	1.23"	.84"	1.19"	1.46"	1.45"	1.42"	1.25"
Avg. C -	.24	.24	.10	.17	.17	.13	.14	.17
Avg. MN -	.43	.43	.96	.73	.71	1.09	.93	1.27
Avg. SI -	.03	.04	.04	.21	.21	.21	.22	.34

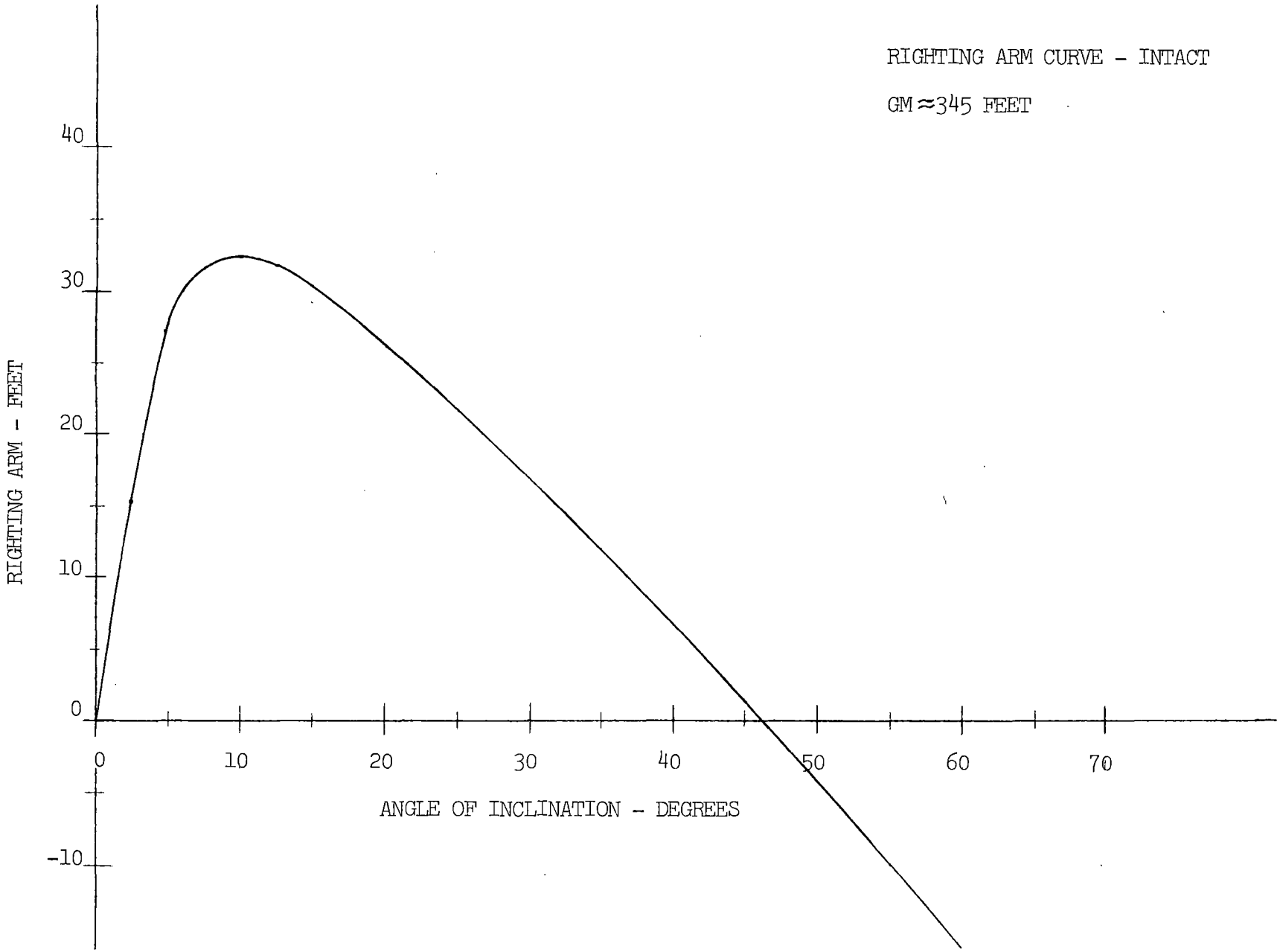
NDT INDICATED BY ↓ (LIMITED DATA) SHATTER POINT



11/1/72

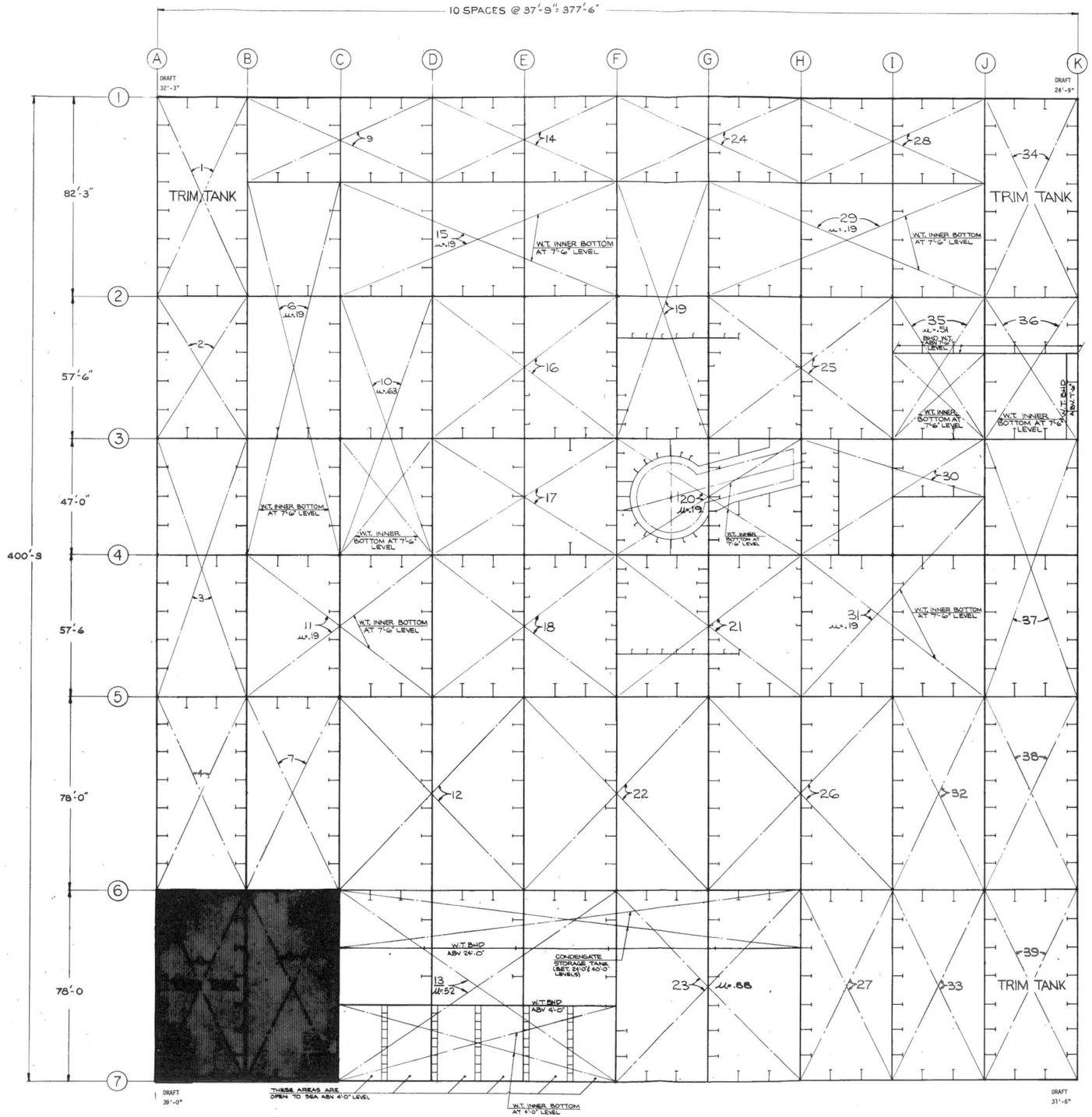
Figure: 3.12-13

American Bureau of Shipping
January 1970



RIGHTING ARM CURVE - INTACT

11/1/72 Figure 3.12-14



NOTES:
PERMEABILITY, μ = 0.95 EXCEPT AS NOTED.

OFFSHORE POWER SYSTEMS

Title: MOST SEVERE CONDITION FOR TWO COMPARTMENT FLOODING

11/1/72 Figure: 3.12-15