CHAPTER 3

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CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

The design of the Oyster Creek Nuclear Generating Station (OCNGS) began approximately in January 1964. First concrete was poured in February 1965 and the Reactor Building was completed in November 1967. Fuel loading was started on April 10, 1969 and Commercial Operation achieved on December 23, 1969.

As part of the application for a Full Term Operating License, the design of the station, as of March 6, 1972, was evaluated against the requirements of 10CFR50.34, Appendix A, General Design Criteria for Nuclear Power Plants, in effect on July 7, 1971. The discussions presented in this section reflect this evaluation, which was submitted as Amendment 68 to the original Facility Description and Safety Analysis Report (FDSAR).

Conformance with NRC General Design Criteria (GDC) for the OCNGS has also been established as part of the Systematic Evaluation Program (SEP). The SEP topics (as detailed in NUREG-0822) have been summarized in Table 3.1-1. The table lists the specific topic or topics which address each GDC, and identifies the corresponding FSAR section or sections which discuss subjects pertinent to each GDC. A detailed summary of the SEP is presented in Section 1.10.

3.1.1 Criterion 1 - Quality Standards and Records

Criterion

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 intent of Criterion 1 was satisfied. A quality assurance program was implemented during the design, fabrication, installation and erection of the station. In planning and implementing the quality assurance program, particular attention was given to its application to those structures, systems and components important to safety. These were designed, fabricated, erected, and tested commensurate with the importance of the safety function to be performed. to quality standards and applicable codes and regulations in effect at the time. When recognized industry codes and standards were used, they were evaluated to determine their

applicability, adequacy, and sufficiency, and then modified or supplemented as necessary to provide assurance of a quality product that satisfied safety related functions.

Appropriate records of the design, fabrication, erection, and testing of structures, systems and components important to safety are maintained by or for the licensee and are available for review or recall.

The classification of structures, systems, and components in relation to the importance of the safety functions to be performed is addressed in Section 3.2. The design bases for structures, systems, and components, as covered in Section 3.8, 3.9, 3.10 and 3.11 have been formulated to reflect the importance of safety functions to be performed.

Table 3.1-1 summarizes the SEP topics which address GDC 1. Section 1.10 provides a summary of the program. The operational assurance program is discussed in Section 17.2.

3.1.2 Criterion 2 - Design Bases for Protection Against Natural Phenomena

Criterion

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 the 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 natural phenomena and, (3) the importance of the safety functions to be performed.

Discussion

The plant equipment that is important to safety was designed to permit safe plant operation and to accommodate all Design Basis Accidents without loss of capability for the appropriate natural phenomena at the site. The capability of these designs was based on relevant site historical data, with suitable margin allowances for uncertainties.

Natural phenomena at the site such as tornadoes, floods and earthquakes are discussed in Sections 2.3, 2.4 and 2.5, respectively. Design magnitudes were based upon the most severe of the natural phenomena recorded for the site or the site vicinity with appropriate margins to account for uncertainties in the historical data. The criteria for determining the effects of these natural phenomena on structures, systems, and components are discussed in Sections 3.3, 3.4, 3.5, and 3.7, as applicable.

The design of safety related structures, systems, and components are discussed in Sections 3.8, 3.9, 3.10 and 3.11. The combinations of the effects of normal and accident conditions with the effects of the natural phenomena are addressed in those sections to the extent that these combinations were considered in the original design.

The classification of structures, systems, and components in relation to the importance of the safety functions to be performed is addressed in Section 3.2. The design bases for structures, systems, and components, as covered in Sections 3.8, 3.9, and 3.10, have been formulated to reflect the importance of safety functions to be performed.

Table 3.1-3 summarizes the SEP topics which address GDC 2. Section 1.10 provides a summary of the program.

3.1.3 Criterion 3 - Fire Protection

Criterion

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. Noncombustible and heat resistant materials shall be used wherever practical throughout the units, 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

Design provisions were included to minimize the occurrence of fire, explosions, and their effects, through the use of noncombustible and fire resistant materials throughout the plant. Combustible materials were limited to those applications where noncombustible materials would be impractical. Separation of redundant wireways was implemented to protect against fire damage causing loss of functions important to safety.

Redundant wiring in the Control Room panels designated for core cooling and isolation functions were not physically separated but their circuits were protected by fuses and circuit breakers such that a damaging electrical fire was not considered credible. The DC power distribution systems are redundant and separated.

In response to NRC rules, regulations and guidelines, the OCNGS Fire Protection System has been upgraded to comply with the requirements of GDC 3. Guidance on the implementation of this criterion is provided in Appendix A of Branch Technical Position 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants." As discussed in Section 9.5, a Fire Hazards Analysis program was implemented for the facility. This analysis was used as the basis for the comparison to Standard Review Plan Section 9.5-1, Appendix A.

3.1.4 Criterion 4 - Environmental and Missile Design Bases

Criterion

Structures, systems, and components important to safety shall be designed to accommodate the effects of, and to be compatible 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 were designed for compatibility with the environmental conditions associated with normal operation, maintenance and testing. Further, equipment used to mitigate the consequences of accidents either were designed to be compatible with postulated accidents or protected against the dynamic effects of postulated accidents.

The extent to which the design bases for structures were formulated to accommodate the above effects is given in Sections 3.8 and 3.9. The assessment of capability of mechanical and electrical components to withstand environmental conditions are discussed in Section 3.11.

These structures, systems, and components are appropriately protected (where necessary) 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 facility. Section 3.5 discusses missile protection. The protection against dynamic effects is discussed in Section 3.6.

Table 3.1-1 summarizes the SEP topics which address GDC 4. Section 1.10 provides a summary of the program.

3.1.5 Criterion 5 - Sharing of Structures, Systems and Components

Criterion

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Discussion

The Oyster Creek Station is a single unit facility.

3.1.6 Criterion 10 - Reactor Design

Criterion

The reactor core and associated coolant, control, and protection system 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 design, in combination with the plant equipment characteristics and nuclear safety systems, was based on providing margins to ensure that fuel is not damaged during normal operation or as a result of anticipated abnormal operational transients.

As discussed in Section 4.4, the design basis for the thermal and hydraulic characteristics incorporated in the core design, in conjunction with the plant equipment characteristics, nuclear instrumentation (Section 7.5), and the Reactor Protection System (Section 7.3), is to ensure that no fuel damage will occur in normal operation or operational transients caused by reasonably expected single operator error or equipment malfunction.

Table 3.1-1 summarizes the SEP topics which address GDC 10. Section 1.10 provides a summary of the program.

3.1.7 Criterion 11 - Reactor Inherent Protection

Criterion

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 characteristic tends to compensate for a rapid increase in reactivity.

Discussion

The reactor core was designed to have: (a) a reactivity response that regulates or damps changes in power level and spatial distributions of power production to a level consistent with safe and efficient operation; (b) a negative reactivity feedback consistent with the requirements of overall nuclear hydrodynamic stability; and (c) a strong negative reactivity feedback under severe power transient conditions.

Nuclear design is discussed in Section 4.3.

3.1.8 Criterion 12 - Suppression of Reactor Power Oscillations

Criterion

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 or can be reliably and readily detected and suppressed.

Discussion

Refer to the discussion for GDC 11. In the Oyster Creek core design, four types of stability have been considered:

a. Process control system stability

b. Nuclear Hydraulic stability

c. Inter channel hydraulic stability

d. Xenon spatial stability

A detailed discussion of reactor stability is provided in Section 4.3.

3.1.9 Criterion 13 - Instrumentation and Control

Criterion

Instrumentation and control shall be provided to monitor variables and systems over their anticipated range for normal operation and accident conditions, and to maintain them within prescribed operating ranges, including those variables and systems which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems.

Discussion

The fission process is monitored and controlled for all conditions from source range through power operating range. The Neutron Monitoring System detects core conditions that threaten the overall integrity of the fuel barrier due to excess power generation and provides a signal to the Reactor Protection System. Fission counters, located in the core, are used for the source range through power operating range. The detectors are located to provide maximum sensitivity to control rod movement during startup, and to provide optimum monitoring in the intermediate and power range.

Instrumentation and controls are discussed in Chapter 7. Table 3.1-1 summarizes the SEP topics which address GDC 13. Section 1.10 provides a summary of the program.

3.1.10 Criterion 14 - Reactor Coolant Pressure Boundary

Criterion

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 Pressure Boundary (RCPB) was designed and fabricated in accordance with the ASME and ASA codes which were in effect during the mid 60's. Leakage from the RCPB is monitored during reactor operation to assess RCPB integrity.

The RCPB is discussed in detail in Section 5.2, which also discusses the Primary Containment Leakage Detection System. Table 3.1-1 summarizes the SEP topics which address GDC 14. Section 1.10 provides a summary of the program.

3.1.11 Criterion 15 - Reactor Coolant System Design

Criterion

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 2pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

Discussion

The Reactor Coolant System (RCS) consists of the reactor vessel and appurtenances, the Reactor Recirculation System, the nuclear pressure relief system, the main steam lines, the Isolation Condensers, a portion of the Feedwater System, Core Spray System, and the Shutdown Cooling System. These systems and components were designed to appropriate codes and standards to provide a high integrity reactor boundary throughout the plant lifetime. These systems provide an integrated energy transport capability which is highly reliable and provides sufficient margin to assure that the design conditions of the Reactor Coolant Pressure Boundary are not exceeded during any condition of normal operations, including anticipated operational occurrences.

A summary description of the Reactor Coolant System and connected systems is presented in Section 5.1. Table 3.1-1 summarizes the SEP topics which address GDC 15. Section 1.10 provides a summary of the program.

3.1.12 Criterion 16 - Containment Design

Criterion

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

The Primary Containment System was designed, fabricated, and erected, to accommodate without failure the pressures and temperatures resulting from, or subsequent to, the double ended rupture or equivalent failure of any coolant pipe within the Primary Containment. The Reactor Building, enveloping the Primary Containment (drywell) System, provides secondary containment when the drywell is closed and in service.

The Reactor Building further provides the primary containment function when the drywell is open. The Primary Containment and its associated safety systems were designed, and are maintained, so offsite doses that could result from postulated design basis accidents remain below the guideline values in 10CFR100. The structural design of the Primary Containment and the Reactor Building is discussed in Section 3.8. The Containment System functional design is presented in Section 6.2.

3.1.13 Criterion 17 - Electrical Power Systems

Criterion

An onsite electric power system and an offsite 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 onsite electric power supplies, including the batteries, and the onsite 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 onsite electric distribution 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 onsite alternate current power sources and the other offsite electric power circuit, to assure that specified acceptable fuel design limits 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 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 onsite electric power supplies.

Discussion

The capacity of either the onsite or the offsite electric power systems is adequate to accomplish all required safety functions under postulated design basis accident conditions. The station batteries are redundant as are the distribution systems for both ac and dc power.

Separation of equipment and wireways has been maintained insofar as practicable to make the redundant distribution systems immune to localized damage.

Physical independence of motive and control power for services required for safe shutdown under accident conditions was attained as follows:

a. All 4160 volt power cables were run in independent steel enclosures and follow separate routes to redundant equipment.

b. All dc motor operated valves' power cables were run in raceways, with proper separation or fire barriers.

c. Controls for redundant power equipment were run in separate raceways.

a. Power cables from the two 480 volt emergency ac switchgear lineups were run in separate raceways, with proper separation or fire barriers.

A summary description of the electric power system is provided in Section 8.1. Full descriptions of the offsite and onsite power systems are included in Sections 8.2 and 8.3, respectively. Table 3.1-1 summarizes the SEP topics which address GDC 17. Section 1.10 provides a summary of the program.

3.1.14 Criterion 18 - Inspection and Testing of Electrical Power Systems

Criterion

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 system shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite 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 the applicable portions of the protection system and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

Discussion

Availability of electrical power is assured through periodic inspection and testing during operation. Continuous monitoring is provided so there is no requirement for testing for availability. However, periodic verification of operability of power available monitors is possible and the status of each power supply, i.e., voltage frequency and presence of grounds (in ungrounded systems), is continually indicated. Individual circuits of the energize-to- operate safeguards systems were provided with loss of control power annunciation.

Inspection and testing requirements of critical components have been established in the Technical Specifications. Inspection and testing is discussed in Section 8.3. Table 3.1-1 summarizes the SEP topics which address GDC 18.

3.1.15 Criterion 19 - Control Room

Criterion

A Control Room shall be provided from which actions can be taken to operate the nuclear power unit safety 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 at 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

The plant is equipped with a centralized Control Room having adequate instrumentation and controls for safe operation of the plant under normal conditions and to maintain the plant in a safe condition in accident situations.

The Control Room has adequate shielding, fire protection, air conditioning, and access facilities to permit continuous occupancy while maintaining radiation exposures under 10CFR20 dose limits during all design basis accident conditions.

Although it should never be necessary to evacuate the Control Room, the station design does not preclude the capability of bringing the reactor unit to a safe, cold shutdown from outside the Control Room. However, some areas of the Reactor Building may not be readily accessible following an accident.

Hot and cold shutdown can be accomplished from outside the Control Room. The reactor will be scrammed prior to control room evacuation, or will be scrammed from outside the control room by deenergizing the Reactor Protection System locally. This is the “tripped” condition for the RPS System and will result in all rods inserting. Alternate shutdown capability is available to meet 10CFR50, Appendix R requirements, and is used for achieving and maintaining hot and cold shutdown from outside the control room. Alternate shutdown capability allows for decay heat removal using the Isolation Condenser System and reactor coolant makeup using the Control Rod Drive System. Cooling water makeup to the Isolation Condenser System can be accomplished by use of Demineralized Water, Condensate Transfer or the fire pumps and local valve operation. Reactor isolation can be accomplished prior to control room evacuation or from outside the control room. Alternate means of boron injection for reactivity control are available in accordance with Emergency Operating Procedures. Other support systems or components can be operated outside the control room using the Remote Shutdown Panel, local shutdown panels or local manual operation. Alternate shutdown capability allows plant cooldown to cold shutdown condition using the Shutdown Cooling System. Process monitoring instrumentation is available to aid the operators in placing and maintaining the plant in a safe hot shudown condition and to subsequently reach cold shutdown. Refer also to Section 7.5.2.4.2.

The Control Room is located in the seismic Class I portion of the Turbine Building. The ventilation system for the Control Room is discussed in Section 9.4. Control Room habitability is discussed in Section 6.4. Table 3.1-1 summarizes the SEP topics which address GDC 19. Section 1.10 provides a summary of the program.

3.1.16 Criterion 20 - Protection System Functions

Criterion

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 and (2) to sense

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

Discussion

The Reactor Protection System was designed to automatically override normal operational controls and initiate reactor protection (scram) action, and the operation of Containment Isolation, Emergency Core Cooling and Standby Gas Treatment Systems whenever the monitored conditions exceed pre-established limits. Each of the protective function actions is initiated by a variety of sensed conditions.

The reactor trip parameters and the respective trip setpoints were selected on the basis of safety analysis. The trip setpoints are selected so that no core design limits will be exceeded as a result of any anticipated operational occurrence. The Reactor Protection System is discussed in Section 7.2. The initiating instrumentation for engineered safety features is discussed in Section 7.3.

3.1.17 Criterion 21 - Protection System Reliability and Testability

Criterion

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, (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 Reactor Protection System was designed for high functional reliability and inservice testability which was considered to be commensurate with the safety function to be performed. Redundancy and independence designed into the system is sufficient to assure that no design basis single failure results in loss of the protective function. Removal from service for brief intervals during anticipated testing, calibration or maintenance procedures was accounted for as not to disable the protective function and high reliability of the function.

Active components in the system and redundant subsystems are capable of being tested or removed from service during reactor operation without compromising the protective function, even in the event of a subsequent single failure.

The Reactor Protection System was designed to permit periodic testing of each active element when the reactor is in operation, with the exception of the main steam tunnel high temperature switches.

The control circuit for the Automatic Depressurization System relief valves can be tested during plant operation.

The Reactor Protection System is discussed in Section 7.2. Table 3.1-1 summarizes the SEP topics which address GDC 21. Section 1.10 provides a summary of the program.

3.1.18 Criterion 22 - Protection System Independence

Criterion

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

The Reactor Protection System was 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 protective function.

The design employed spatial and/or functional diversity sufficient to meet a single failure criterion which allows unlimited damage to any single active device or module and unlimited damage to any single panel outside the Control Room.

The qualification testing program is discussed in Section 3.10. The conformance to Section 4 of IEEE 279 is discussed in Section 7.2 for the Reactor Protection System and in Section 7.3 for the engineered safety features.

3.1.19 Criterion 23 - Protection System Failure Mode

Criterion

The protection system shall be designed to fall 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 system was designed so that failure of any one Reactor Protection System input or subsystem will not prevent a subsequent trip signal, or a tripped condition on both channels, from initiating the protective function.

The reactor scrams and isolation control trips to the safe condition on loss of power. Motive power for redundant isolation valves which do not fail closed on loss of power (or air) is from two independent supplies. Control and motive power for redundant core cooling equipment is from two independent supplies. All control equipment and motive power sources were

designed or upgraded after initial operation to operate under conditions of design basis environmental extremes (refer to Section 3.11 for discussion).

Alternate power sources are discussed in Chapter 8. Control and protection systems are discussed in detail in Chapter 7.

3.1.20 Criterion 24 - Separation of Protection and Control Systems

Criterion

The protection system shall be separated from control systems to the extent that failure of any 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 protective system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Discussion

Sensors and electrical circuits (except in certain portions of Control Room panels) necessary to the functioning of the Reactor Protection Systems were physically (by means of distance or barriers) and electrically separated to prevent any single event, including single failures in the control systems, from compromising the protection function.

There are no active elements of the reactor or nuclear plant control systems whose failure can impair the operation of the Reactor Protection System, Containment Isolation System or Emergency Core Cooling System. All active sensory equipment for the reactor protection, isolation and safeguards functions were independent of those used for automatic level are also used for water level control but there are redundant sets of sensor trains arranged such that failure of one set will not disable the protective function.

Conformance with IEEE-279 is discussed in Sections 7.2 through 7.6. Control and protection systems are covered in detail in those sections.

3.1.21 Criterion 25 - Protection System Requirements for Reactivity Control Malfunctions

Criterion

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 safety analyses discussed in Chapter 15 demonstrate that acceptable fuel damage limits will not be exceeded for any reactivity transient resulting from a single equipment malfunction or operator error.

The reactor design, which is discussed in Chapter 4, has considered reactivity control malfunctions. The operation of the Reactor Protection System is discussed in Section 7.2.

3.1.22 Criterion 26 - Reactivity Control System Redundancy and Capability

Criterion

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 operations, including anticipated operational occurrences, and with appropriate 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 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

The reactor unit contains two independent, different principle reactivity control systems. Control of reactivity is operationally provided by a combination of movable control rods, burnable poison, and Reactor Recirculation System flow. These systems accommodate fuel burnup, load changes, and long term reactivity changes.

Reactor shutdown by the Control Rod Drive System is sufficiently rapid to prevent fuel damage limits being exceeded during either normal operation or any operational transients. A Standby Liquid Control System (Liquid Poison System) was provided as an independent backup shutdown system to cover emergencies of the operational reactivity control system. This system was designed to maintain the reactor in a shutdown condition as it cools.

The reactivity control system was designed to provide sufficient reactivity compensation under conditions of normal operation to make the reactor always subcritical from its most reactive condition. Means were provided for continuously regulating the reactor core excess reactivity and reactivity distribution.

Reactor physics is discussed in Chapter 4. The Liquid Poison System is described in Section 9.3.

3.1.23 Criterion 27 - Combined Reactivity Control Systems Capability

Criterion

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

In addition to the Control Rod Drive System (Sections 4.6 and 3.9); the Liquid Poison System (Section 9.3), the Recirculation Flow Control System (Section 4.4), and the Control Rod Velocity Limiter (Section 4.6), combine to provide reactivity control for the reactor. The evaluation of combined performance is presented in Subsection 4.6.5. Also refer to Subsection 3.1.22.

3.1.24 Criterion 28 - Reactivity Limits

Criterion

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 orother 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

The Reactivity Control System was designed to compensate for positive and negative reactivity changes resulting from nuclear coefficients, fuel depletion, and fission product transients and buildup. The system design limits control rod worths and the rate at which reactivity can be

added. These design limits assure that a design basis reactivity accident is not capable of damaging the Reactor Coolant System and disrupting the reactor core, core support structures, or other vessel internals sufficiently to impair the Emergency Core Cooling System effectiveness.

The control requirements for excess reactivity and the methods of control are discussed in Section 4.3. Accident analyses for various reactivity problems are discussed in Chapter 15.

3.1.25 Criterion 29 - Protection Against Anticipated Operational Occurrences

Criterion

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

A high functional reliability of the Reactor Protection and Reactivity Control Systems was achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail safe design, and inservice testability. The instrumentation and control for the Reactor Protection System, Containment Isolation System

Emergency Core Cooling System were designed to assure an extremely high probability of accomplishing their safety functions in the event of any anticipated operational occurrence including single operator error or single equipment malfunction.

The high probability of correct protection system and reactivity control system response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance.

Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions, even in the event of a subsequent single failure.

Consideration was given to a steam release in the drywell which can raise the temperature to a value which could cause failure of neutron monitoring cables (IRM and APRM). During the design of the OCNGS, operational data from plants which had experienced such events had demonstrated the failure mode to be in the safe (upscale) direction of the APRMs (LPRM cables) in most cases. In addition, it has been demonstrated that scram from high drywell pressure (at about 2 psig) would occur before any neutron monitoring cable damage could occur so cable failure considerations were not of primary importance to safe shutdown of the reactor.

The Reactor Protection System is discussed in Section 7.2, the Reactivity Control Systems are discussed in Sections 3.9, 4.4, 4.6 and 9.3. The Containment Isolation System and the Emergency Core Cooling System are described in detail in Sections 6.2 and 6.3, respectively. Environmental design considerations are discussed in Section 3.11. Anticipated operational occurrences are addressed in Section 5.2, and accident analyses are presented in Chapter 15. Pipe break analyses are covered in Section 3.6.

3.1.26 Criterion 30 - Quality of Reactor Coolant Pressure Boundary

Criterion

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

The ASME and ASA Codes in effect at the time of design (C.1964) were used as the established and acceptable criteria for design, fabrication, and operation of components of the Reactor Coolant Pressure Boundary (RCPB). The RCPB was designed and fabricated (Criteria 30, 31) to meet the following codes:

a. Reactor vessel: ASME Boiler and Pressure Vessel Code, Section I and Case Interpretations 1270 N and 1273 N.

b. Pumps and valves: ASA B31.1 and ASME B&PV Code Sections I, III and VIII.

c. Piping: All piping connected to the Reactor Pressure Vessel up to the first isolation valve, except for CRD and Feedwater (includes the remainder of the reactor recirculation loops: ASME Section I; for CRD piping from the Reactor Vessel up to the inlet side of their respective vent valves and to the inlet side of the Hydraulic Control Unit isolation valves: ASME Section I; for Feedwater piping connected to the Reactor Vessel up to the and including the first isolation valve outside the Drywell: ASME Section I; the remaining RCPB piping not delineated above; ASA B31.1.

The Reactor Coolant System was given a final hydrostatic test at 1560 psig in accordance with Code requirements prior to initial reactor startup. A hydrostatic test, not to exceed system operating pressure, (about 1020 psig), is made on the Reactor Coolant System following each removal and replacement of the reactor vessel head. In addition, a hydrostatic test of 1135 psig is performed for the ASME ten year required hydro and if weld repairs are made. The system is checked for leaks, and any abnormal conditions are corrected before reactor startup. The minimum vessel temperature during hydrostatic testing was established at 60°F above the calculated NDT (nil ductility transition) temperature prior to pressurizing the vessel.

The inherent safety features of reactor core design, in combination with certain engineered safety features and the Reactivity Control System, will limit the consequences of all postulated accident conditions. These consequences are limited to prevent either motion or rupture caused damage to the Reactor Coolant Pressure Boundary.

Leakage detection from the RCPB is discussed in Section 5.2. Radioactivity monitoring to detect leakage is discussed in Section 11.5. Table 3.1-1 summarizes the SEP topics which address GDC 30. Section 1.10 provides a summary of the program.

3.1.27 Criterion 31 - Fracture Prevention of Reactor Coolant Pressure

Criterion

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 nd 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

The brittle fracture failure mode of the Reactor Coolant Pressure Boundary components has been precluded by control of the notch toughness properties of the ferritic components. This control was exercised in the selection of materials, fabrication of equipment and components, and by limiting radiation exposure to levels below which the nil ductility transition (NDT) temperature remains unaffected. The design considered the different notch toughness requirements of the various ferritic steel forms, including weld and heat affected zones. In this way, brittle fracture would be prevented under all potential service loading temperatures.

A temperature based rule, with modifications drawn from fracture mechanics technology, was used to establish the requirements for preventing brittle fracture.

Refer to Sections 5.2 and 5.3 for additional discussion.

3.1.28 Criterion 32 - Inspection of Reactor Coolant Pressure Boundary

Criterion

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 leak tight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Discussion

All piping, pumps and valves as defined by ASME Boiler and Pressure Vessel Code Section XI (January 1, 1970 issue)* to be part of the primary coolant pressure boundary and all components of the reactor vessel above the biological shield, (such as: the closure head, head to flange, vessel to flange, and nozzles), were designed to be accessible for inspection during efueling. Also, the reactor pressure vessel within the biological shield is accessible for inspection at the nozzles through removal of shield plugs and thermal insulation around the nozzles. Surveillance specimens have been provided for periodic testing.

Inservice inspection of the RCPB is discussed in Section 5.2. These activities are presently conducted in conformance with the NRC approved Inservice Inspection Program for the OCNGS. The vessel surveillance program is discussed in Section 5.3. Periodic inspections for leak tightness are addressed in the Technical Specifications.

3.1.29 Criterion 33 - Reactor Coolant Makeup

Criterion

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 onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite 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.

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* ASME B&PV Code Section XI which was the original Code of Record, was revised in 1974 (Summer 1975) and did not define what is part of the RCPB. As a result, the 1970 Edition was used for boundary definition.

Discussion

During normal operation, water level in the reactor vessel is maintained by the Feedwater System. For small breaks in the Reactor Coolant Pressure Boundary, the makeup capability is normally provided by the Feedwater System, and the Control Rod Drive System. The Emergency Core Cooling System provides core cooling for breaks beyond the capability of the Feedwater System or the Control Rod Drive System. Both the Emergency Core Cooling System and the Control Rod Drive System can be powered from the Emergency Diesel Generator System in the event that offsite power is not available.

The analysis of small breaks in the RCPB is discussed in Chapter 15. A description of the Feedwater System and its performance for reactor water level control is presented in Section 10.4. The Control Rod Drive System is discussed in Section 3.9, and the Head Cooling System (which is supplied water from the control rod drive pumps) is described in Section 5.4. Details of the Emergency Core Cooling system, and the degree of protection provided for small breaks is discussed in Section 6.3. Protection functions are provided even in the event of loss of offsite power and are discussed in Section 8.3.

3.1.30 Criterion 34 - Residual Heat Removal System

Criterion

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

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

Discussion

The station is provided with a Shutdown Cooling System (SCS) for normal shutdown, and Core Spray and Containment Systems for accident conditions, to ensure the integrity of the reactor core.

The Isolation Condenser System (ICS) acts as a backup, standby means of decay heat removal in the event that the Main Condenser is not available as a heat sink following a reactor scram.

The Emergency Core Cooling System (ECCS) was designed to prevent fuel cladding melt over the entire spectrum of postulated design basis reactor primary system breaks. Such capability is available concurrent with loss of all offsite power. The ECCS was designed to high levels of component redundancy so that a single active component failure, in addition to the accident, will not affect the system's ability to provide sufficient core cooling.

The initial cooling and removal of decay heat, immediately following a shutdown of the turbine and reactor, is accomplished by means of the Turbine Bypass System (Section 10.4). At

approximately 350°F, the SCS is placed in operation (refer to Section 5.4). The SCS is not safety related.

The Isolation Condenser System is described in Section 6.3. Table 3.1-1 summarizes the SEP topics which address GDC 34. Section 1.10 provides a summary of the program. Protection functions are provided even in the event of loss of offsite power.

3.1.31 Criterion 35 - Emergency Core Cooling

Criterion

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 onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Discussion

Following a Loss-of-Coolant Accident (LOCA), the Core Spray System will provide cooling water to the reactor. The Core Spray System is a low pressure system and is fully operational only after reactor pressure drops to about 285 psig. In the event of large breaks, the system is designed to be fully operational immediately following vessel blowdown. For small breaks beyond the makeup capability of the Feedwater System, or when the Feedwater System is not available, initial reactor depressurization is accomplished by the Automatic Depressurization System.

The Core Spray System is highly redundant and is capable of performing its protective function even in the event of loss of offsite power, assuming a single failure. Both the Core Spray system and the Automatic Depressurization System are described in detail in Section 6.3.

3.1.32 Criterion 36 - Inspection of Emergency Core Cooling System

Criterion

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 the integrity and capability of the system.

Discussion

Design provisions were made to enable physical and visual inspection of the ECCS components. All piping and valves within the Containment are inspectable during reactor shutdown, and piping, pumps and valves external to the Containment are inspectable at any time. The components of the ECCS within the reactor vessel are inspectable during refueling.

Inspection of the ECCS is discussed in Section 6.3. Frequency of inspection is addressed in the Technical Specifications, and in the NRC approved Inservice Inspection Program for Oyster Creek.

3.1.33 Criterion 37 - Testing of Emergency Core Cooling System

Criterion

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

To assure that the ECCS will function properly, specific provisions were made for testing and sequential operability and functional performance of each individual system.

Minimum testing requirements for the ECCS are detailed in the Technical Specifications. Testing is also addressed in Section 6.3.

3.1.34 Criterion 38 - Containment Heat Removal

Criterion

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 acceptable low levels.

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

Discussion

Provisions were made for the removal of heat from within the Primary Containment for as long as necessary to maintain the integrity of the Containment following the various postulated design basis accidents.

The Containment Spray System is designed to remove heat energy from the Primary Containment. It is used with the Core Spray System as the means of removing the decay heat of the reactor core from the Containment in the event of a LOCA. The Emergency Service

Water System provides cooling for the Containment Spray Heat Exchangers thereby providing the heat sink for the energy released during a LOCA.

The Containment Spray and Emergency Service Water Systems are highly redundant, and will perform their function in the event of a loss of offsite power, assuming a single failure. These systems are described in detail in Section 6.2.

3.1.35 Criterion 39 - Inspection of Containment Heat Removal System

Criterion

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

Discussion

The capability to test the functional performance and to inspect the containment heat removal system has been provided. This capability is discussed in Section 6.2. The inspection schedule is addressed in the Technical Specifications.

3.1.36 Criterion 40 - Testing of Containment Heat Removal System

Criterion

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 the 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

Specific provisions have been made in the design of the Containment Spray System to allow for testing the operability of the system. Minimum testing requirements are addressed in the Technical Specifications. Testing is also discussed in Section 6.2.

3.1.37 Criterion 41 - Containment Atmosphere Cleanup

Criterion

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 associated systems, the concentration and quality 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 onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

Discussion

The Containment Atmosphere Control System was designed to maintain an inert atmosphere within the Primary Containment to preclude energy releases from a possible hydrogen oxygen reaction following a postulated Loss-of-Coolant Accident which could jeopardize the integrity of the Containment. Following postulated accidents, the control of fission products, hydrogen and oxygen is accomplished by routing the containment atmosphere to the Standby Gas Treatment System, which consists of two parallel, 100% capacity systems. The SGTS will fulfill its safety function even in the event of loss of offsite power.

The Containment Atmosphere Control System is described in Section 6.2. The Standby Gas Treatment System is discussed in Section 6.5.

3.1.38 Criterion 42 - Inspection of Containment Atmosphere Cleanup

Criterion

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

Design of the Containment Atmosphere Control System permits periodic inspection and functional testing as well as testing of the active components of the system. Refer to Section 6.2 for additional information.

3.1.39 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 systems 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

Design of the Containment Atmosphere Control System permits periodic functional testing, as well as testing of the active components of the system. Refer to Section 6.2 for additional information.

3.1.40 Criterion 44 - Cooling Water

Criterion

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 the 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 onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Discussion

The Reactor Building Closed Cooling Water System was designed to remove heat rejected by the Shutdown Cooling System during plant cooldown and plant shutdown, and from other Reactor Building heat exchangers during operation. The Emergency Service Water System was provided to remove heat rejected by the Containment Spray System. Cooling water requirements were based on accident cooling water demands of the equipment served.

The Containment Spray System, via the Emergency Service Water System, transfers heat from the containment to the Ultimate Heat Sink following a LOCA. Refer to Subsection 3.1.34.

The Shutdown Cooling System is described in Section 5.4, and the description of the Reactor Building Closed Cooling Water System is presented in Section 9.2. The Containment Spray System and the Emergency Service Water System are described in Section 6.2. Details of the Ultimate Heat Sink can be found in Sections 9.2 and 10.4.

Table 3.1-1 summarizes the SEP topics which address GDC 44. Section 1.10 provides a summary of the program.

3.1.41 Criterion 45 - Inspection of Cooling Water System

Criterion

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

Discussion

The capability to test the functional performance and to inspect the major components in the Shutdown Cooling and Reactor Building Closed Cooling Water Systems was provided.

The inspection provisions for the Containment Spray and Emergency Service Water Systems are discussed under GDC 39 (Subsection 3.1.35).

The Reactor Building Closed Cooling Water System is described in Section 9.2, inspection provisions are also discussed in that section. The Shutdown Cooling System is described in Section 5.4. Table 3.1-1 summarizes the SEP topics which address GDC 45. Section 1.10 provides a summary of the program.

3.1.42 Criterion 46 - Testing of Cooling Water System

Criterion

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing. This testing assures:

a. the structural and leaktight integrity of its components,

b. the operability and the performance of the active components of the system, and,

c. 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 both reactor shutdown and Loss-of-Coolant Accidents. This includes operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Discussion

The Reactor Building Closed Cooling Water System is normally in operation. Testing of the system includes the closure timing of containment isolation valves, inservice testing of the pumps and valves, and calibration of the local pump discharge pressure gauges. The Shutdown Cooling System has been designed to permit periodic testing. The testing provisions of the Containment Spray and Emergency Service Water System are covered under GDC 40.

The Reactor Building Closed Cooling Water System and the Shutdown Cooling System are described in Sections 9.2 and 5.4, respectively.

3.1.43 Criterion 50 - Containment Design Basis

Criterion

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 in steam generators and 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 Primary Containment structure, including access openings and penetration, was designed to withstand the peak accident pressure and temperatures that could occur during the postulated design basis Loss-of-Coolant Accident. The Containment design includes considerable allowance for energy addition from metal-water or other chemical reactions beyond those that could occur during the accident. The integrity of the complete Containment was designed, and will be maintained, to limit offsite doses from postulated design basis accidents to a value below the guideline values stated in 10CFR100.

The Primary Containment structural design is discussed in detail in Section 3.8. The basis for defining the functional capability of the Containment Systems are discussed in Section 6.2. Table 3.1-1 summarizes the SEP topics which address GDC 50. Section 1.10 provides a summary of the program.

3.1.44 Criterion - Fracture Prevention of Containment Pressure Boundary

Criterion

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 non-brittle manner, (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 stage and transient stresses, and (3) size of flaws.

Discussion

The Primary Containment was designed to the version of the ASME Boiler and Pressure Vessel Code Section VIII, plus nuclear code cases, in effect at the time of design (C.1965). The containment vessel is designed to be at a minimum temperature of NDT + 30°F during any plant condition at which the Primary Containment could be pressurized.

For further discussion, refer to Sections 3.8 and 6.2.

3.1.45 Criterion 52 - Capability for Containment Leakage Rate Testing

Criterion

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 station design allows pressure and leak rate testing of the Primary Containment. Leakage rate testing is discussed in the Technical Specifications.

The structural design of the Containment is presented in Section 3.8, whereas the functional design of the Primary Containment and Containment Systems is covered in Section 6.2.

3.1.46 Criterion 53 - Provisions for Containment Testing and Inspection

Criterion

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

Primary Containment design provides for individual leak testing of penetrations, except those piping penetrations which are welded directly to the Containment shell. Refer to Sections 3.8 and 6.2 for further detail.

3.1.47 Criterion 54 - Piping Systems Penetrating Containment

Criterion

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

Provisions were made to demonstrate functional performance of the Containment System isolation valves and permit leak testing of selected penetrations. The Containment Isolation System is described in Section 6.2.

Table 3.1-1 summarizes the SEP topics which address GDC 54. Section 1.10 provides a summary of the program.

3.1.48 Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

Criterion

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 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 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 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.

<u>Discussion</u>

The Reactor Coolant Pressure Boundary consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, valves and pipes which extend from the reactor pressure vessel up to and including the outermost isolation valve. The lines of the Reactor Coolant Pressure Boundary which penetrate the Primary Containment are capable of isolating the Containment, thereby precluding any significant release of radioactivity. Similarly, for lines which do not penetrate the Primary Containment but which form a portion of the Reactor Coolant Pressure Boundary (such as connecting lines up to and including the second isolation valve), the design ensures that isolation from the Reactor Coolant Pressure Boundary can be achieved.

Influent lines to the Reactor Coolant Pressure Boundary (RCPB) include the following:

a. Feedwater lines

b. Isolation Condenser condensate return lines

c. Reactor Cleanup System return line to recirculation loop

d. Shutdown Cooling System return to recirculation loop

e. Core Spray System injection lines

f. Head Spray Cooling System Injection line

g. Control Rod Drive System return line

h. Standby Liquid Control System (Liquid Poison System) injection lines.

Effluent lines from the RCPB include the following:

a. Main Steam lines

b. Reactor Cleanup System suction line

c. Shutdown Cooling System suction line

d. Main steam drain lines

e. Isolation Condenser steam supply lines.

f. RCS sample lines.

Isolation provisions for these influent and effluent lines are discussed in Section 6.2. The section also provides a listing of valve positions during normal operation and accident conditions.

In order to assure protection against the consequences of accidents involving the release of radioactive material, lines which form the Reactor Coolant Pressure Boundary were shown to provide adequate isolation capabilities on a case by case basis. Adequate isolation capabilities were also demonstrated for lines that connect to the Reactor Coolant Pressure Boundary outside the Primary Containment. In all cases, a minimum of two barriers were used to protect against the release of radioactive materials.

In addition, in meeting the isolation requirements stated in Criterion 55, the pressure retaining components which comprise the Reactor Coolant Pressure Boundary were designed to meet other appropriate requirements which minimize the probability or consequences of an accidental rupture. The quality requirements for these components ensure that they were designed, fabricated, and tested to the highest quality standards of all reactor plant components.

Table 3.1-1 summarizes the SEP topics which address GDC 55. Section 1.10 provides a summary of the program.

3.1.49 Criterion 56 - Primary Containment Isolation

Criterion

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

Those lines which penetrate the Primary Containment and communicate with the Containment atmosphere may be grouped into two categories: (1) lines which penetrate the Primary Containment and connect directly to the suppression chamber, and (2) lines which penetrate the Containment and connect directly to the Containment atmosphere.

Criterion 56 required that these lines have two isolation valves; one inside the containment, the other outside. This criterion presented a significant problem in the BWR suppression pool type of Containment. An isolation valve located inside the lines near the bottom of the suppression chamber would result in placement of a valve underwater. In effect, this would result in introducing a potentially unreliable valve in a highly reliable system thereby compromising design. For this reason, these lines were designed to incorporate two valves outside the Containment, the first of which was located as close to the Containment as possible. Provisions for containment isolation and valve positions for various plant conditions are discussed in Section 6.2.

In order to assure protection against the consequences of accidents involving release of significant amounts of radioactive materials, pipes that penetrate the Primary Containment were demonstrated to provide isolation capabilities in accordance with Criterion 56. In all cases, these pipes were provided a minimum of two protective barriers against Containment leakage.

In addition to meeting the isolation requirements stated in Criterion 56, the pressure retaining components of these systems were designed to the same quality standards as the Containment.

Table 3.1-1 summarizes the SEP topics which address GDC 56. Section 1.10 provides a summary of the program.

3.1.50 Criterion 57 - Closed System Isolation Valves

Criterion

Each line that penetrates primary reactor containment and is neither part of the reactor coolant 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

Lines that penetrate Primary Containment and were not part of the Reactor Coolant Pressure Boundary and were not open to the Containment atmosphere were equipped with at least one remote operated isolation valve outside Containment or a check valve on the influent line inside Containment. Refer to Section 6.2 for further detail.

Table 3.1-1 summarizes the SEP topics which address GDC 57. Section 1.10 provides a summary of the program.

3.1.51 Criterion 60 - Control of Releases of Radioactive Materials to the Environment

Criterion

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 reaction 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

The station radioactive waste control systems, which include the liquid, gaseous and solid radwaste subsystems, were originally designed to limit the offsite radiation exposure to levels below doses set forth in 10CFR20. The station engineered safeguards, including the Containment barriers, were designed to limit the offsite dose under various postulated design basis accidents to levels significantly below 10CFR100. The air ejector offgas system was designed with sufficient holdup capacity to prevent the controlled release of radioactive materials from exceeding the established release limits at the elevated station stack during normal station operation.

Subsequently, (C.1977) new radioactive liquid, gaseous and solid waste systems were designed and constructed to process radioactive materials to the degree required for effluents to meet the requirements of Appendix I to 10CFR50. Chapter 11 presents the discussion of those features of the facility designed to control releases of radioactive materials to the environment.

Table 3.1-1 summarizes the SEP topics which address GDC 60. Section 1.10 provides a summary of the program.

3.1.52 Criterion 61 - Fuel Storage and Handling and Radioactivity Control

Criterion

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 appropriate periodic 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 station has appropriate plant fuel handling and storage facilities to preclude accidental criticality and to provide sufficient cooling for spent fuel. Fuel handling and storage facilities are housed in the Reactor Building. Irradiated fuel transfer operations are conducted underwater for shielding purposes.

The Spent Fuel Pool Cooling and Cleanup System was designed to maintain pool water temperature below acceptable limits, control water clarity to permit visual inspection and control of operation, and reduce water radioactivity to protect personnel.

Water depth in the pool provides sufficient shielding for normal Reactor Building occupancy (10CFR20) by operating personnel.

Fuel handling and storage are discussed in Section 9.1. Table 3.1-1 summarizes the SEP topics which address GDC 61. Section 1.10 provides a summary of the program.

3.1.53 Criterion 62 - Prevention of Criticality in Fuel Storage and Handling

Criterion

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

Discussion

The new fuel storage vault racks, located inside the Reactor Building, were designed for top entry and to prevent an accidental critical array even in the event the vault becomes flooded. Vault drainage was provided to prevent possible water collection.

The handling and storage of spent fuel, which are entirely within the Reactor Building (the Secondary Containment) is done in the Spent Fuel Storage Pool. The pool has provisions to maintain water clarity, temperature control, and instrumentation to monitor water level.

The spent fuel racks, in which irradiated fuel assemblies are placed, were designed and arranged to ensure subcriticality in the storage pool. For further discussion, refer to Section 9.1.

3.1.54 Criterion 63 - Monitoring Fuel and Waste Storage

Criterion

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

Station radiation and process monitoring systems were provided to monitor the significant process parameters and the station environmental effluents. These systems provide alarms and signals to permit appropriate corrective action.

Spent Fuel Storage Pool level and temperatures are displayed in the Control Room. Area radiation monitors are provided throughout the facility. Monitoring of fuel storage is presented in Section 9.1, area monitoring is described in Section 12.3.

3.1.55 Criterion 64 - Monitoring Radioactivity Releases

Criterion

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

This criterion was met in the original design with the exception that monitoring of the Primary Containment during and following a Loss-of-Coolant Accident had not been provided. The radiation levels experienced during a Loss-of-Coolant Accident or sizable leak in the Primary Containment would be beyond the capacity of the normal radiation monitoring instrumentation. However, alterations were implemented to permit monitoring radiation levels throughout the ranges anticipated during accidents.

Details of monitoring for radioactivity releases are presented in Sections 11.5 and 12.3. The functional objective of the radioactive waste processing systems is to minimize the release of radioactivity to the environment. Liquid, gaseous and solid wastes are collected, treated and made suitable for reuse or disposal in accordance with the requirements of Appendix I to 10CFR 50, and the limits of 10CFR20. Accidental releases of radioactivity are minimized to the extent that the likelihood of occurrence is minimal and, if such releases do occur the radiological consequences will be within 10CFR100 limits.

Table 3.1-1
(Sheet 1 of 9)

SEP TOPIC SUMMARY

Criterion		SEP Topic		Corresponding Section
GDC-1	Quality Standards and Records	III-1	Classification of Structures and Systems	3.2, 17.2
		III-7.B	Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria	
		VI-1	Organic Materials and Post Accident Chemistry	
GDC-2	Design Basis for Protection Against Natural Phenomena	II-3.B	Flooding Potential and Protection Requirements	2.3, 2.4, 2.5 3.2, 3.3, 3.4,
		II-3.B.1	Capability of Operating Plant to cope with Design-Basis Flooding conditions	3.5, 3.7, 3.8 3.9, 3.10, 3.11
		II-3.C	Safety Related Water Supply (Ultimate Heat Sink)	
		II-2	Wind and Tornado Loadings	
		III-3.A	Effects of High Water Level on Structures	
		III-3.C	Inservice Inspection of Water Control Structures	
		III-4.A	Tornado Missiles	
		III-6	Seismic Design Considerations	
		III-7.B	Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria	

Table 3.1-1
(Sheet 2 of 9)

SEP TOPIC SUMMARY

Criterion		SEP Topic		Corresponding Section
GDC-2	Design Basis for Protection Against Natural Phenomena	IV-2	Reactivity Control Systems Including Functional Design and Protection Against Single Failures	
		VI-7.C.1	Appendix K - Electrical, Instrumentation and Control Re-Reviews	
		VII-3	Systems Required for Safe Shutdown	
		VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	
		VIII-4	Electrical Penetrations of Reactor Containment	
GDC-3	Fire Protection	IX-6	Fire Protection	3.8, 9.5
GDC-4	Environmental and Missile Design Bases	III-4.B	Turbine Missiles	3.5, 3.6, 3.8,
		III-4.D	Site Proximity Missiles (including Aircraft)	3.11, 5.4, 3.9
		IV-5.A	Effects of Pipe Break on Structures, Systems and Components Inside Containment	
		III-5.B	Pipe Break Outside Containment	
		III-7.B	Design Codes, Design Criteria, Load Combinations, and Reactor Cavity Design Criteria	
		VI-7.C.1	Appendix K - Electrical, Instrumentation and Control Re-Reviews	

Table 3.1-1
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SEP TOPIC SUMMARY

Criterion		SEP Topic		Corresponding Section
GDC-4	Environmental and Missile Design Bases	VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	
		VIII-4	Electrical Penetrations of Reactor Containment	
		IX-5	Ventilation Systems	
GDC-5	Sharing of Structures, Systems and Components	VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	1.2
		VIII-4	Electrical Penetrations of Reactor Containment	
GDC-10	Reactor Design	XV-1	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	4.4
GDC-11	Reactor Inherent Protection	-		4.1 to 4.6
GDC-12	Suppression of Reactor Power Oscillations	-		4.1 to 4.6
GDC-13	Instrumentation and Control	III-8.A	Loose Parts Monitoring and Core Barrel Vibration Monitoring	3.11, 11.5, Ch. 7
		VII-3	Systems Required for Safe Shutdown	

Table 3.1-1
(Sheet 4 of 9)

SEP TOPIC SUMMARY

Criterion		SEP Topic		Corresponding Section
GDC-14	Reactor Coolant Pressure Boundary	V-12.A	Wa8ter Purity of BWR Primary Coolant	5.2
		VI-1	O8rganic Materials and Post-Accident Chemistry	
GDC-15	Reactor Coolant System	XV-1	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	5.1 to 5.4
GDC-16	Containment Design	-		3.8, Ch. 6
GDC-17	Electrical Power Systems	VI-7.C.1	Appendix K - Electrical, Instrumentation and Control Re-Reviews	8.1, 8.2, 8.3
		VIII-2	Onsite Emergency Power System (Diesel Generator)	
		VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	
		VIII-4	Electrical Penetrations of Reactor Containment	
GDC-18	Inspection and Testing of Electrical Power Systems	VI-7.C.1	Appendix K - Electrical, Instrumentation and Control Re-Reviews	8.3
		VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	
		VIII-4	Electrical Penetrations of Reactor Containment	

Table 3.1-1
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SEP TOPIC SUMMARY

Criterion		SEP Topic		Corresponding Section
GDC-19	Control Room	V-10.B VIII-3.B	Residual Heat Removal System Reliability DC Power System Bus Voltage Monitoring and Annunciation	6.4, 9.4, Ch. 7
GDC-20	Protection System Functions	-		7.3
GDC-21	Protection System Reliability and Testability	VI-10.A	Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing	7.2, 7.3
GDC-22	Protection System Independence	-		7.2, 7.3
GDC-23	Protection System Failure Modes	-		Ch.7
GDC-24	Separation of Protection and Control Systems	-		Ch. 7
GDC-25	Protection System Requirements for Reactivity Control Malfunctions	-		15.4, 7.2, Ch. 4
GDC-26	Reactivity Control System Redundancy and Capability	-		Ch. 4, Ch. 9

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SEP TOPIC SUMMARY

Criterion		SEP Topic		Corresponding Section
GDC-27	Combined Reactivity Control Systems Capability	-		Ch. 4, Ch. 9
GDC-28	Reactivity Limits	-		4.3, 15.4
GDC-29	Protection Against Anticipated Operational Occurrences	-		Ch. 6, Ch. 5, 4.5, 4.6, 9.3
GDC-30	Quality of Reactor Coolant Pressure Boundary	V-5	Reactor Coolant Pressure Boundary Leakage Detection	Ch. 5, Ch. 11
GDC-31	Fracture Prevention of Reactor Coolant Pressure Boundary	-		5.2
GDC-32	Inspection of Reactor Coolant Pressure Boundary Coolant Pressure Boundary	-		5.3, 5.2
GDC-33	Reactor Coolant Makeup	-		9.3
GDC-34	Residual Heat Removal	V-10.B VII-3	Residual Heat Removal System Reliability Systems Required for Safe Shutdown	5.4

Table 3.1-1
(Sheet 7 of 9)

SEP TOPIC SUMMARY

Criterion		SEP Topic	Corresponding Section
GDC-35	Emergency Core Cooling	-	6.3
GDC-36	Inspection of Emergency Core Cooling System	-	6.3
GDC-37	Testing of Emergency Core Cooling System	-	6.3
GDC-38	Containment Heat Removal	-	6.2
GDC-39	Inspection of Containment Heat Removal System	-	6.2
GDC-40	Testing of Containment Heat Removal System	-	6.2
GDC-41	Containment Atmosphere Cleanup	-	6.2
GDC-42	Inspection of Containment Atmosphere Cleanup Systems	-	6.2
GDC-43	Testing of Containment Atmosphere Cleanup Systems	-	6.2

Table 3.1-1
(Sheet 8 of 9)

SEP TOPIC SUMMARY

Criterion			SEP Topic	Corresponding Section
GDC-44	Cooling Water	III-3.C	Inservice Inspection of Water Control Structures Control Structures	9.2
GDC-45	Inspection of Cooling Water System	III-3.C	Inservice Inspection of Water Control Structures	9.2
GDC-46	Testing of Cooling Water System	-		9.2
GDC-50	Containment Design Basis	VIII-4	Electrical Penetrations of Reactor Containment	3.8, 6.2
GDC-51	Fracture Prevention of Containment Pressure Boundary	-		3.8
GDC-52	Capability for Containment Leakage Rate Testing	-		3.8, 6.2
GDC-53	Provisions for Containment Testing and Inspection	-		3.8, 6.2
GDC-54	Piping Systems Penetrating Containment	VI-4	Containment Isolation System	6.2

Table 3.1-1
(Sheet 9 of 9)

SEP TOPIC SUMMARY

Criterion		SEP Topic		Corresponding Section
GDC-55	Reactor Coolant Pressure Boundary Penetrating Containment	VI-4	Containment Isolation System	6.2
GDC-56	Primary Containment Isolation	VI-4	Containment Isolation System	6.2
GDC-57	Closed System Isolation Valves	VI-4	Containment Isolation System	6.2
GDC-60	Control of Releases of Radioactive Materials to the Environment	IX-5	Ventilation Systems	11.2, 11.3, 11.4
GDC-61	Fuel Storage and Handling and Radioactive Control	IX-5	Ventilation Systems	3.8, 9.1, 9.3
GDC-62	Prevention of Criticality in Fuel Storage and Handling	-		9.1
GDC-63	Monitoring Fuel and Waste Storage	-		9.1, 12.3
GDC-64	Monitoring Radioactivity Releases	-		11.5, 12.3

3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS AND SYSTEMS

3.2.1 Seismic Classification

Guidance for determining the seismic classification of structures, systems and components is provided in Regulatory Guide 1.29, "Seismic Design Classification." The Oyster Creek Nuclear Generating Station (OCNGS) was designed prior to the promulgation of this regulatory guide.

The original seismic design for the Oyster Creek Nuclear Generating Station critical structures and equipment was based on dynamic analyses using acceleration response spectrum curves which were based on a ground motion of 0.11g. The design is such that a safe shutdown could be achieved during a ground motion of 0.22g.

The two classes of structures to which earthquake design requirements were applied are:

a. Class I - Structures and equipment whose failure could cause significant release of radioactivity, or which are vital to a proper shutdown of the plant and the removal of decay heat.

b. Class II - Structures and equipment not designated as Class I.

Structures and equipment originally categorized as Class I are listed in Table 3.2-1. Class II items were designed following the normal practice for the design of power plants in the State of New Jersey in effect at the time of design; as a minimum this was not less than given in the "Uniform Building Code" for Zone 1.

3.2.2 System Quality Group Classifications

General Design Criterion 1 of Appendix A to 10CFR50, as implemented by Regulatory Guide 1.26, requires that structures, components and systems important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of safety functions to be performed. Although the OCNGS was designed prior to issuance of Regulatory Guide 1.26, the codes used for the design, fabrication, erection and testing of the Oyster Creek plant were compared with current codes as part of the Systematic Evaluation Program. A general summary of this program is presented in Section 1.10.

Conformance with General Design Criterion 1 is given in Section 3.1. The responses to staff positions detailed in NUREG-0822 for Topic III-1 are included in Section 1.10. Drawings JCP-19431, 3E-211-A1-001, 3E-213-A1-001, 3E-241-A1-001, 3E-223-A1-001, 3E-212-A1-001, 3E-411-A1-002 (Sheets 1 & 2, 3E-424-A1-001, 3E-243-A1-001, 3E-532-A1-001, 3E-541-A1-001 (Sheets 1, 2, &3), 3E-225-A1-001 and 002, 3E-251-A1-001 are the inservice diagrams for the

OCNGS; these figures are coded to reflect quality group classifications of piping as defined in the NRC approved Inservice Inspection Program for the OCNGS

3.2.3 Systematic Evaluation Program (Sep) Topic III-I, Classification Of Structures, Components And Systems

10 CFR 50 (GDC 1) requires that structures, systems and components important to safety be designed, fabricated, erected and tested to quality standards commensurate with the importance of safety functions to be performed.

During the integrated assessment of the SEP the NRC staff requested information in certain areas as documented in Section 4.2 of the Integrated Plant Safety Assessment (NUREG 0822). GPU Nuclear provided the information requested by NUREG 0822 by letter dated September 29, 1989. The NRC responded in their letter dated April 3, 1990, concluding that the information provided by GPU Nuclear is "adequate and acceptable."

Brief description of the information submitted to the NRC is provided below.

A. Fracture Toughness

ASME Code, Section III imposes minimum fracture toughness requirements on certain carbon steel components. Design and construction information was gathered on the 40 items and compared to the established criteria exempting components from fracture toughness requirements.

B. Radiographic Requirements

Tables were provided to document the construction radiography for the Oyster Creek Class 2 vessels, Class 1 and 2 piping and valves, and Class 1 and 2 pumps.

At the time of the NUREG 0822 letter (09/29/89), the inspection requirements for Class 2 vessels, Class 1 and 2 piping and valves, and Class 1 and 2 pumps were contained in the Oyster Creek Inservice Inspection Program. The program was per Section XI of the ASME Boiler and Pressure Vessel Code, 1974 edition with addenda through Summer 1975. The program defines the inspection requirements for the vessels, piping, valves and pumps. Volumetric inspections were performed per the requirements of Section XI, where volumetric inspection was not meaningful or possible appropriate exemptions and reasons were listed in the program.

C. Valves

Current ASME Code, Section III design requirements regarding body shapes and Service Level C stress limits for Class 1 valves and pressure-temperature ratings for Class 2 and 3 valves are different from those used when the plant was designed.

An investigation was made, on a sampling basis, whether Class 1 valve stress limits meet current criteria for body shape and Service Level C conditions and if the pressure-temperature ratings of Class 2 and 3 valves are comparable to current standards.

D. Pumps

A table was provided to list pumps by system, materials and design code used.

E. Storage Tanks

Compressive stress requirements for atmospheric storage tanks and tensile stress requirements for 0- to 15- psig storage tanks designed according to ASME Code, Section III, Class C (1965) or ASME Code, Section VIII (1965) differ from those in the current ASME Code, Section III, Class 2.

The following evaluations were performed to:

a. Confirm the atmospheric storage tanks meet current compressive stress requirements.

b. Confirm that the 0- to 15- psig storage tanks meet current tensile allowables for biaxial stress field conditions.

c. Re-evaluate the design and construction of the liquid poison system and liquid waste system tanks against current criteria because these tanks were designed to American Petroleum Institute (API) Standard 650 and the requirements of API 650 are not comparable to those of present design codes, and identify whether adequate safety margins exist.

d. Re-evaluate the design and construction of the condensate tank against current criteria.

F. Piping

Five representative piping systems were selected to be evaluated under the current ASME Class 1 Code criteria. These systems are:

a. Core Spray System

b. Main Steam System

c. Emergency Isolation Condenser System

d. Control Rod Drive Housing

e. Automatic Depressurization System

Significant transients for each piping system and the corresponding occurrence cycle anticipated for the operating life of the plant were considered in the analysis.

The conclusion reached by the evaluation of these sample piping systems for thermal transient induced fatigue under the current Class 1 piping code criteria for ASME Section III, Subsection NB shows these piping systems meet the usage factor criteria (i.e., total usage 1.0). The fatigue analyses are based on calculations and descriptions of all the significant transients encountered during the operating life of the plant.

For current transient and fatigue information, see Sections 5.2.1 and 5.3.1.

TABLE 3.2-1
(Sheet 1 of 2)

CLASS I CRITICAL STRUCTURES AND EQUIPMENT

FOR THE ORIGINAL FACILITY DESIGN*

Class I Critical Structures

Drywell, Vents, Torus and Penetrations
Reactor Building
Control Room (and supporting part of Turbine Building)
Spent Fuel Pool
Ventilation Stack

Class I Critical Equipment

Nuclear Steam Supply System Components:
- Reactor Vessel
- Reactor Vessel Supports
- Control Rods and Drive System (including equipment necessary for scram operation)
- Control Rod Drive Thimble Supports
- Fuel Elements
- Core Shroud
- Core Supports
- Steam Separator
- Steam Dryer
- Recirculation Piping System (including valves and pumps)
- All piping connections from the Reactor Vessel (up to and including the first isolation valve external to the drywell)
- Isolation Valves

Reactor Emergency Systems:
- Isolation Condenser System
- Liquid Poison System
- Core Spray System
- Containment Spray System

Emergency Service Water System

Standby Gas Treatment System

Fuel Storage Facilities (including spent fuel and new fuel storage equipment)

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* This table lists those structures and equipment that were designed as Class I for the original plant. Refer to system descriptions in the appropriate FSAR sections for more detailed and updated information.

TABLE 3.2-1
(Sheet 2 of 2)

CLASS I CRITICAL STRUCTURES AND EQUIPMENT*

FOR THE ORIGINAL FACILITY DESIGN*

Standby Electrical Power Systems:

Station Battery
Diesel Generators
Emergency Buses (and other electrical gear and power to critical equipment)

Instrumentation and Controls:

Reactor Pressure and Level Instrumentation
Standby Liquid Control System Instrumentation
Manual Reactor Control System
Control Rod Position Indicating System
Reactor Protection System
Neutron Monitoring System
In Core Neutron Monitors
Area Monitors

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* This table lists those structures and equipment that were designed as Class I for the original plant. Refer to system descriptions in the appropriate FSAR sections for more detailed and updated information.

3.3 WIND AND TORNADO LOADINGS

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

All of the Oyster Creek Nuclear Generating Station (OCNGS) structures were originally designed to withstand ASA 58.1-1955 (reconstituted as USAS A58-1-1955 design wind loadings for buildings and other structures for Ocean County, New Jersey. The Reactor and Turbine Buildings were originally designed to withstand at least 25 psf between elevations 0 to 30 feet, 30 psf between 30 feet to 50 feet, 40 psf between elevations 50 feet to 100 feet and 45 psf between elevations 100 to 500 feet.

The Reactor Building structure and the Ventilation Stack were later studied (References 1, 2) to determine their ability to withstand the loading applied by a wind storm with a 100 year period of recurrence. The significance of the 100 year storm was based on a probability of 33 percent of encountering this storm in an assumed 40 year design life. For the OCNGS, the 100 year wind storm has a reference wind velocity of 100 mph. The vertical distribution of wind velocity for a base wind of 100 mph (fastest mile of wind at 30 feet above ground level) is as follows:

Height Above Ground in feet	Wind Velocity in mph
0 to 50	100
50 to 150	125
150 to 400	155

The calculations of Reactor Building pressure were based on a 1.1 gust factor. The calculations for stack pressures were based on a 1.3 gust factor. For additional details on site meteorology refer to Section 2.3.

Wind loadings have also been evaluated under Topic III-2 of the Systematic Evaluation Program for the OCNGS. Section 1.10 provides a summary of this program. The results of the evaluation are provided in Section 3.8.4.5.4.

3.3.1.2 Determination of Applied Forces

In the original studies conducted for the Reactor Building structure and the Ventilation Stack, the wind velocity was converted into an equivalent pressure on the structures by using the methods described in ASCE Paper No. 3269 (1961).

For the later study of the Reactor Building (Reference 1), wind pressures corresponding to the velocities shown in Subsection 3.3.1.1 are 40.3 psf from 0 to 50 feet and 62.8 psf from 50 to 150 feet. These building pressures are based on a 1.1 gust factor and a 1.3 shape factor for a rectangular building.

In the later study of the ventilation stack (Reference 2), wind pressures were calculated to be 34.6 psf, 53.9 psf and 83.0 psf for the ranges of 0 to 50 feet, 50 to 150 feet and 150 to 400 feet above ground, respectively. These stack pressures were based on a 1.3 gust factor and a 0.8 shape factor for a cylinder oriented perpendicular to the direction of flow and appropriate for a Reynolds number in the range between 9×10^6 and 3×10^7.

In these later analyses, normal allowable working stress levels were increased by one third for loading combinations of dead load, live load and wind load.

The capacity of the various plant structures subjected to wind and tornado loads has been evaluated as part of the Systematic Evaluation Program and the reactor building superstructure has been modified. The resulting wind speed limits are described in Section 3.8.4.5.4. The probabilities of exceeding these wind speeds at the OCNGS site are shown in Figure 3.3-1. This figure is obtained from Appendix 2.3A and incorporates NRC comments. Generally, Class I equipment is enclosed in the Class I structures and is therefore protected within the limits shown. The Outdoor Service Water Pumps and Startup Transformer are capable of withstanding 200 mph winds and a depressurization of 2 psi.

3.3.2 Tornado Loadings

For the original design of the OCNGS, the significance of the 100 year storm (Subsection 3.3.1) was based on a probability of 33 percent of encountering this storm in an assumed 40 year design life. This 100 year storm is not the highest wind to which the building could be subjected. A tornado was considered as a possible source of higher loadings; however, the tornado frequency period for the Oyster Creek Site was 2190 years corresponding to a probability of 1.81 percent for a 40-year life.

Thus, the OCNGS does not meet the tornado wind and pressure drop loading to the absolute limits specified in Regulatory Guide 1.76. Nevertheless, the safety margin is adequate when considering the historical records for tornado occurrences at the site. In addition, the reactor building superstructure has been modified to increase its capacity to withstand wind and tornado loads. Details of this effort are provided in Section 3.8.4.5.4.

The Control Room, Battery Room, Emergency Diesel Generator Building, emergency switchgear and related wiring have been designed for tornado protection.

A reevaluation of the tornado wind speeds and associated pressure drops has been performed as part of the Systematic Evaluation Program. A summary of the program is presented in Section 1.10, and results are described in Section 3.8.4.5.4.

3.3.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

The Turbine Building capacity to resist tornado loads, and the effects of failure of the building on other structures has been evaluated as part of the Systematic Evaluation Program, refer to Section 1.10 for details.

3.3.4 References

(1) Engineering Evaluation for Facility Design and Safeguards Report, "Extension of Wind Load in Structure - Reactor Building," The Ralph M. Parsons Company. May 15, 1967.

(2) Engineering Evaluation for Facility Design and Safeguards Report, "Extension of Wind Load in Structure - Stack," The Ralph M. Parsons Company, 1967.

3.4 WATER LEVEL (FLOOD) DESIGN

Flood potential for the Oyster Creek Site and flood protection for structures and components are discussed in Section 2.4. Plant structures were originally designed for hydrostatic loads up to 15 ft above Mean Sea Level (MSL) in combination with horizontal seismic earth loading.

The effects of high water level on structures was reviewed under Topic III-3.A of the Systematic Evaluation Program (SEP). The NRC has concluded that based on factors of safety obtained against floatation, the adequacy of subgrade walls, and the adequacy of bearing capacity, the facility can adequately withstand a ground water level at El. 23' MSL. Section 1.10 provides a summary of SEP topics, and discusses the conclusions of the program and the plant modifications implemented as a result of the SEP.

3.5 MISSILE PROTECTION

3.5.1 Missile Selection and Description

This subsection identifies the source/type of postulated missiles internal and external to the plant. The following subsections discuss the criteria used for missile selection.

3.5.1.1 Internally Generated Missiles (Inside Drywell)

The Oyster Creek Nuclear Generating Station (OCNGS) has been designed to preclude the possibility that missiles penetrate the Primary Containment. This has been accomplished in practice through the specific design of the Primary Containment and contained systems, which takes into account the potential for generation of missiles and minimizes the possibility of containment violation.

The design of the Primary Containment and piping systems has considered the possibility of missiles being generated in the following forms:

a. Valve bonnets (large and small)

b. Valve stems

c. Thermowells

d. Vessel head bolts

e. Instrument thimbles

f. Nuts and bolts

g. Pieces of pipe

The driving force for potential missiles was assumed to come from the energy within the working fluid. In the case of a break in a pipe carrying liquid the maximum liquid velocity attainable at the break is 200 fps because of choking. Similarly, the velocity of fluid from a steam line break is limited to the critical velocity of 1500 fps at the break. The drag force of the fluid which propels any potential missile is proportional to the product of the density and the velocity square. Even though the velocity of the steam exceeds that of the water, the even larger ratio of water density to steam density at Primary Containment ambient conditions means that projectiles originating from a water line will have a greater drag force applied, and will therefore achieve a larger kinetic energy.

Thus, missiles originating from steam lines were neglected as being insignificant relative to missiles originating from liquid lines where the action was from the same distance and water lines were present. All small missiles propelled by liquid were assumed to achieve and maintain the maximum liquid velocity of 200 fps until impact. This is conservative because a missile, after being dislodged, requires a finite time for acceleration before it can approach a velocity of 200 fps. In addition, for missiles directed in a horizontal direction, there is a tendency for a missile, which is traveling slower than the driving jet, to fall out of the jet as it is acted upon by gravity. Therefore, the driving force acts for a shorter time and the missile achieves a lower maximum velocity.

Extensive tests were conducted by the Stanford Research Institute (SRI) during which rod shaped missiles (traveling at velocities that could possibly be produced within the drywell) were impacted against square steel plates having clamped edges. The results of the tests can be described by the following expression for minimum energy per unit diameter of missile required for perforation of a steel plate:

$$E/D = U\,(0.344T^2 + 0.032T)$$

where:

E = critical kinetic energy required for penetration, ft-lbs.
D = diameter of missile, inches
U = ultimate tensile strength, psi
T = plate thickness, inches

This equation was plotted for the various thicknesses of the drywell shell and is shown in Figure 3.5-1.

Using the above conservative design criteria and the results of the SRI tests, it was found that no small missiles (e.g., thermowells, small valve components, etc.) originating from the liquid lines would achieve sufficient energy to penetrate the drywell, nor was there sufficient strain energy in the pressure vessel head bolts to cause penetration.

The most serious potential missile appeared to be a dislodged valve bonnet originating from a recirculation loop valve. It was assumed that the face of the bonnet (35 inch diameter) was acted upon by the water jet, and that the massive (approximately 3000 lbs) bonnet stem assembly impacted the Primary Containment with the stem (4 inch diameter) making initial contact. This is a conservatively chosen event because it requires that all bolts holding the bonnet sever completely, that the bonnet and the stem move as a massive unit and that the stem end (smallest impact area) strike the Primary Containment first.

Since the valve bonnet is so heavy, it would achieve a kinetic energy of 1,860,000 ft-lbs if it were traveling at 200 fps. Therefore, a more refined calculation was necessary to show that a velocity of 200 fps is not actually attained. This calculation was made and it was found that the bonnet would have to accelerate a distance of 15 feet, and reach a velocity slightly in excess of 30 fps in order to acquire the energy (44,000 ft-lbs) necessary for the 4 inch diameter stem on the bonnet to penetrate the Primary Containment. It was determined from the arrangement of components within the drywell that, even though the recirculation valves are oriented such that a dislodged valve bonnet would strike the Primary Containment directly, there is not sufficient distance available between the stem and drywell to achieve the energy necessary to penetrate.

It has been shown in experiments conducted by the Chicago Bridge and Iron Co. (Reference 1) that large, slowly applied loads acting over an area of 1.08 ft^2 on a plate 3/4 inch thick would not cause cracking to develop in the plates until a deflection greater than 3 inches had occurred. Since the drywell shell is reinforced by concrete, located nominally 2 3/4 inches away, it appears improbable that objects having a large impact area will be able to penetrate the steel without also penetrating the concrete. Small missiles do not achieve a high enough velocity, based on the assumed design criteria, to attain an energy level sufficient to penetrate sound Containment shell material. Therefore, it is concluded that missile penetration of the Containment is a highly improbable event.

Safety systems inside Containment are redundant and physically separated sufficiently to ensure that missiles could not affect both redundant portions of the system.

A reactor vessel instrument tube anti-ejection device restricts the movement of the instrument tube flange to approximately 1/2 inch, and thus maintains the tube within the confines of the reactor vessel penetration annulus.

3.5.1.2 Internally Generated Missiles (Outside Drywell)

The drywell vessel is completely enclosed in a reinforced concrete structure having a thickness of 5 to 7 1/2 feet. This structure, in addition to serving as the basic biological shielding for the reactor's primary system, also provides a major mechanical barrier for the protection of the Primary Containment drywell and the reactor's primary system against potential missiles generated external to the Primary Containment.

The reactor's primary system is further protected from missiles generated outside the Secondary Containment by means of the Reactor Building's concrete structure which consists of poured in place, reinforced concrete exterior walls and internal floors up to the refueling floor level. Above this level, the building structure is of steel frame construction with insulated metal siding. The concrete wall adds 1 1/2 feet of concrete for a total thickness of 6 1/2 to 9 feet of reinforced concrete surrounding the Primary Containment.

In addition, the equipment necessary for emergency core and containment cooling, including heat exchangers and pumps, is located in the Reactor Building in compartments located below the refueling floor level and generally separated by reinforced concrete walls, which prevents missile damage.

Missile or shielding protection of other equipment and systems critical to station safety is afforded by the reinforced concrete structures and separations which surround this equipment. In addition, the heavy casings around the rotating parts in pumps, motors, generators and turbines provide a highly effective barrier to contain loose pieces which might come from potential failure of rotating parts.

Missile generation outside the drywell was also investigated as part of the Systematic Evaluation Program (SEP), discussed in detail in Section 1.10.

3.5.1.3 Turbine Missiles

GPUN has replaced the Oyster Creek LPC turbine rotor with a rotor which has last stage buckets which weigh more than the original last stage buckets. Therefore, GPUN has re-evaluated the potential for a turbine generated missile to affect safe operation of the OCNGS. Specifically, GPUN has evaluated the consequences of worst case turbine generated missiles striking safety related structures in the worst possible orientation. As described in Reference 11, sufficient protection is in place to prevent penetration of any of the structures by any credible turbine generated missile.

The configuration of the OCNGS LP turbine after replacement of the LPC rotor is:

TURBINE SECTION	ROTOR TYPE	LAST STAGE BUCKET WEIGHT (Lbs.)
LPA	Built-Up	14
LPB	Built-Up	14
LPC	Monoblock	28

The original, built-up, low pressure turbine rotors at OC are susceptible to wheel bore cracking. If a wheel bore crack reaches a critical crack size and the turbine is over-sped, a wheel failure could occur, potentially resulting in a turbine missile (Reference 10). The turbine OEM (General Electric) has identified four aspects of turbine maintenance, operations, and testing that factor into the probability of generating a turbine missile;

1. Periodic Ultrasonic inspections of the wheel bores on an OEM recommended frequency assures that new wheel bore cracks are identified, and existing wheel bore cracks are re-examined, before they reach a critical crack size. (Reference 15)
2. Pre-warming the Low Pressure turbine rotors from cold starts lowers the stresses on the wheels and reduces the growth rate of existing wheel bore defects. (Reference 18)
3. The frequency of testing of turbine valves affects the probability of experiencing a turbine overspeed event. (References 14 and 16)
4. The frequency of testing of overspeed protective trip devices affects the probability of experiencing a turbine overspeed event. (References 16 and 19)

Changing the frequency of occurrence of any of these four aspects directly impacts the probability of generating a turbine missile.

The results of Low Pressure turbine rotor wheel bore inspections are evaluated by GE using a probabilistic methodology developed by General Electric, the results of which determine the wheel burst probability as a function of time, for specified overspeed trip test and turbine valve test frequencies, with and without pre-warming (Reference 16). The evaluation methodology has been approved by the NRC for use by Licensees in establishing turbine maintenance, operations, and testing requirements that maintain the probability of generating a turbine missile below NRC limits (Reference 20).

The turbine system reliability criteria established by the NRC, for an unfavorably oriented turbine, is as follows:

Criterion	Annual Probability	Required Licensee Action
(A)	$P_1 < 10^{-5}$	This is the general, minimum reliability requirement for the loading of the turbine and bringing the system on line.
(B)	$10^{-5} < P_1 < 10^{-4}$	If this condition is reached during operation, the turbine may be kept in service until the next scheduled outage, at which time the licensee is to take action to reduce P_1 to meet the appropriate (A) criterion (above) before returning the turbine to service.
(C)	$10^{-4} < P_1 < 10^{-3}$	If this condition is reached during operation, the turbine is to be isolated from the steam supply within 60 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate (A) criterion (above) before returning the turbine to service.
(D)	$10^{-3} < P_1$	If this condition is reached at any time during operation, the turbine is to be isolated from the steam supply within 6 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate (A) criterion (above) before returning the turbine to service.

P_1 – probability of turbine missile generation external to the turbine casing.

OCNGS committed to the NRC in response to SEP Topic III.4.B, as identified in NUREG 1382 Section 3.5.1.4, to adhere to GE recommendations for maintaining the probability of missile generation below NRC criteria (Reference 17). As such, Oyster Creek will adjust turbine rotor inspection intervals, valve testing frequencies, overspeed protective trip frequencies and the need to pre-warm the Low Pressure turbine rotors, based on GE recommendations from rotor inspection findings, that satisfy the requirements of the NRC with respect to the turbine system reliability criteria listed above.

To ensure that nuclear safety related structures, systems or components would not be impacted in the unlikely event that a turbine missile is generated, OCNGS has evaluated the consequences of both bucket and wheel fragment missiles in the following sections:

3.5.1.3.1 Low Trajectory Missiles

Low angle trajectory missiles exit the turbine casing on a path which directly intersects an intervening wall, slab or other structure. A review of potential low angle trajectory missiles has been performed by Quad Cities as shown in References 4 and 5. Based on this report and input from G.E., the most damaging potential low trajectory missile is a bucket vane which fails at the root. As described in paragraph 3.5.1.3, the new LPC buckets weigh 28# each while the LPA and LPB buckets weigh 14# each. Failure of the root is expected at 175% of rated speed. An analysis has been performed to determine penetration depths of a bucket missile into various structures at Oyster Creek. This analysis utilizes 180% of rated speed as the condition at which the bucket becomes a missile. As shown in Table 3.5-1 and in Reference 11, none of the structures will be perforated by either the 14# bucket or the 28# bucket. Penetration depths are calculated for all missiles using both the NDRC equation and the more recent CEA-EDF equation. The NDRC equation was used in the original missile penetration analysis. However, recent work (Reference 12) indicates that the CEA-EDF formula more accurately predicts penetration depths for missiles typical of low trajectory turbine generated missiles. Penetration depths in all cases are less using the CEA-EDF formula. However, as shown in Table 3.5-1, sufficient protection is provided in all cases even if penetration depths are calculated using the NDRC equation.

3.5.1.3.2 High Trajectory Missiles

High angle trajectory missiles essentially fall onto the target after being projected into the air. The mass of the missile is therefore the controlling parameter. An analysis of high angle trajectory missiles is described in Reference 6. Based on this analysis, a 120° wheel fragment is the controlling missile. Damage due to all other potential missiles except the turbine generator coupling are bounded by the wheel fragment damage assessment. Due to its location and configuration, the coupling cannot impact any safety related structures even if it becomes a missile.

The wheel fragment missile is not applicable to the monoblock rotor as described in References 10 and 13. The single piece monoblock rotor can withstand speeds up to 225% of rated speed while physical configuration limits turbine speed to 220% of rated speed even if no component failures occur at lower speeds. Therefore, a wheel fragment missile from the monoblock turbine rotor is not considered.

As described in Reference 11, the penetration depth of a high trajectory missile has been calculated. The results demonstrate that the 120° wheel fragment penetrates 11.1" into a reinforced concrete structure of finite thickness. All other potential missiles penetrate less than 11" into reinforced concrete. Therefore, a slab thickness of 22.2" is sufficient to stop any postulated missile. The 84" thick reactor cavity shield plugs and 62" thick spent fuel pool slab provide significantly greater depth than the required 22.2". The control room ceiling is 20" thick which is slightly less than the required thickness of 22.2". However, the missile must pass through the turbine building roof (twice) consisting of metal decking with built-up roofing, through the 8" cable spreading room concrete roof and through components in the cable spreading room. Since none of these intervening structures is considered in the analysis, the 20" concrete slab provides adequate protection for the equipment in the control room.

3.5.1.3.3 Deleted

3.5.1.4 Missiles Generated by Natural Phenomena

3.5.1.4.1 Tornado Missiles

Tornado wind velocity probabilities have been evaluated at Oyster Creek, as presented in Section 2.3. Further evaluation of tornadoes and tornado missiles at the Oyster Creek site has been conducted as part of the Systematic Evaluation Program; this program is summarized in Section 1.10. Wind velocity probabilities determined during this effort are presented in Appendix 2.3A and in Section 3.3.

In the design of the OCNGS, the method of analysis to determine the protective capability of Class I buildings and equipment against various sized missiles and missile penetration at tornado velocities was based on the Modified Petry Formula (Navy Bureau of Yards and Docks NP3726).

The missiles assumed were a wood utility pole, 35 feet long by 14 inches in diameter having a velocity of 200 mph, and a one ton missile, such as a compact type automobile traveling at 100 mph with a contact area of 25 ft^2.

The results of the analysis indicated that no perforation of the 18 inch thick Reactor Building walls or the 12 inch thick Control Room walls would occur, although spalling of the inside concrete face would be expected.

As part of the systematic evaluation program, the potential for damage to safety related (Q) equipment or systems due to impact from tornado generated missiles was reviewed. The results indicate that the precast panels on the north wall of the Control Room and the Emergency Diesel Generator (EDG) Bldg. do not provide protection from tornado generated missiles traveling at the speeds described above. In addition, the equipment access door provides essentially no protection for a small number of safety related (Q) components located in the vicinity of this door.

OCNGS, therefore, re-evaluated these three areas to assess capability to withstand tornado generated missiles. The analysis demonstrates that the EDG Bldg. will withstand missiles associated with a wind speed of 168 mph and the north wall of the Control Room will withstand missiles associated with a wind speed of 160 mph. Based on the low probability of exceeding a wind speed of 160 mph, the extremely low probability of a tornado generated missile striking these small targets and the redundancy of equipment, sufficient protection for the safety related (Q) equipment and systems in the vicinity of these structural components is in place.

In addition, GPUN has determined that safe plant shutdown can be achieved even if the components in proximity to the equipment access door are damaged by a tornado missile. Thus adequate protection is also provided from tornado missiles striking the equipment access door.

There was no consideration of missile protection in the design of the Class I pumps at the intake structure. However, further analysis indicated that it is unlikely that more than one of the redundant pumps would become inoperable as a result of tornado missiles.

Further analysis concluded that the Spent Fuel Storage Pool 62 inch thick slab could not be perforated by the postulated missiles. The design of the Reactor Building and the arrangement of equipment is such that there is little chance that either the Spent Fuel Storage Pool or the fuel stored in the pool could be seriously damaged as a result of tornado effects on the building or its contents. (See Section 9.1.)

3.5.1.4.2 Floods

No missiles associated with floods have been identified for the OCNGS.

3.5.1.5 Missiles Generated by Events Near the Site

3.5.1.5.1 External Explosion Missiles

Potential hazards due to external explosions have been considered. Possible sources of explosion are essentially limited to transportation accidents (Reference 3). The Systematic Evaluation Program Integrated Plant Safety Assessment (NUREG-0822) concluded that, on the basis of existing State of New Jersey laws and the low probability of explosive shipments in the vicinity of the site, the intent of Regulatory Guide 1.91 is satisfied and that missiles generated by explosions near the site need not be further evaluated. However, a NRC letter dated May 23, 1990 stated that the NRC staff has concluded, after their discussion with the State of New Jersey, that there is sufficient increase in truck traffic in the vicinity of the Oyster Creek Nuclear Generating Station, specifically on U.S. Route 9, to warrant a reassessment of the frequency of hazardous material shipments and, potentially, the level of risk associated with the shipment. GPUN agreed to perform a detailed characterization of traffic and hazardous material shipments, based on more recent data in the vicinity of OCNGS, that would permit verification that the risk due to nearby transportation is acceptably low (Reference 8). Missile sources resulting from accidental transportation explosions in the vicinity of OCNGS have not been

characterized or evaluated because of the low probability of such events. Additional procedural measures, described in Section 2.2.3.2, address Control Room actions in the event of potential hazardous material incidents in the vicinity of OCNGS. See Section 2.2.3.2 regarding the OCNGS Hydrogen Water Chemistry System hydrogen storage facility.

Two natural gas pipelines are installed in the vicinity of Oyster Creek. One is for normal distribution and the other is adjacent to the combustion turbines. In addition to the low probability of an accident occurring, the detonation of an unconfined natural gas dispersal in air is not a credible event

3.5.1.5.2 External Turbine Missiles

Combustion turbines are located outside the plant protected area approximately 1/4 mile from the nearest safety related structure at Oyster Creek.

A potential turbine missile hazard affects only the intake canal due to the position of the Combustion Turbines. No Safety Related (Q) or Augmented Quality (A) Structure is located within the Turbine Missile Strike Zone as defined in NRC Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles."

A Turbine Missile Hazard will not affect the safety function of the intake canal due to the following:

a) As delineated in Section 3.5.1.3 of the FSAR, the failure of turbines to the extent of producing missiles is highly unlikely.

b) The CT's have metal enclosures which will reduce the velocity of the missiles, if produced.

c) If a partial blockage should occur due to the effects of a turbine missile, a lesser percentage of the total canal area can supply the cooling water requirements. A total blockage due to a turbine missile is not considered possible. Also, due to the existence of redundant Emergency Service Water pumps on the intake structure, it is not considered credible for more than one set of pumps to be affected by a missile.

3.5.1.6 Aircraft Hazards

An evaluation of aircraft accidents for the OCNGS is presented in Reference . Further evaluation of aircraft hazards is presented as part of the Systematic Evaluation Program in Section 1.10.

3.5.2 Structures, Systems and Components to be Protected from Externally Generated Missiles

The Class I (Seismic) Structures for the OCNGS include the following:

a. Reactor Building exterior concrete walls

b. Reactor Building insulated metal siding

c. Reactor Building roof decking

d. Reactor Building steel for craneway enclosure

e. Control Room walls

f. Ventilation Stack

g. Battery Room (interior room)

h. Diesel Generator and Fuel Oil Tank Vaults

Generally, safety related equipment is enclosed in the listed Class I structures. An analysis of missile protection for safety related equipment was performed as part of the SEP. This program is discussed in detail in Section 1.10.

3.5.3 References

(1) Thullen - "Loads on Spherical Shells," Chicago Bridge & Iron Co.., Oak Brook, Illinois, August 1964.

(2) Not Used.

(3) Not Used.

(4) Application for License, Quad Cities Units 1 and 2, Docket No. 50-254/265, Amendment 3.

(5) Ibid, Amendment 4.

(6) Application for License, Browns Ferry, Amendment 6, Question C-8.

(7) "An Analysis of Turbine Missiles Resulting from Last Stage Wheel Failure," GE Topical Report TR67SL211.

(8) NUREG-0822, Integrated Plant Safety Assessment, Systematic Evaluation Program. Oyster Creek Nuclear Generating Station, Docket No. 50-219, Final Report. January 1983.

(9) OCNGS Procedure 2000-ABN-3200.33, "Toxic Material/Flammable Gas Release - No Radiation Involved".

(10) Letter from G. E. (Martin O'Connor) to GPUN (Frank Collado), dated June 7, 1996.

(11) LES Calculation No. 72-01-01, Turbine Missile Analysis for New Monoblock Rotor and Blades," October, 1996. Revision 3.

(12) Journal of the Structural Division, "Assessment of Empirical Concrete Impact Formulas," By George E. Sliter, Dated May 1980, page 1035-1036.

(13) Letter from GE Power Generation (George Reluzco) to General Electric Company (J. Hess), Dated March 28, 1996.

(14) GE Technical Information Letter (TIL) 969-3 Revision 1, December 27, 1993.

(15) GE Technical Information Letter (TIL) 1008-3 Revision 1, December 27, 1993.

(16) GE, "LPA Wheel Dovetail Inspection Report", December 2000.

(17) NUREG-1382, "Safety Evaluation Report Related to the Full Term Operating License for Oyster Creek Nuclear Generating Station", Docket No. 50-219, January 1991.

(18) General Electric, GEK-46518 "Importance of Rotor Prewarming During Cold Starting", March 1975.

(19) GE Technical Information Letter (TIL) 1165-3, May 1, 1995.

(20) NUREG-1048 Appendix U, "Probability of Missile Generation in General Electric Nuclear Turbines", July 1986.

TABLE 3.5-1

COMPONENT	MISSILE	PENETRATION BASED ON NDRC FORMULA (1)	PENETRATION BASED ON CEA-EDF FORMULA (2)	MINIMUM THICKNESS OF STRUCTURAL BARRIER
CONTROL ROOM	14#	29.1”	19.8”	69.1” (4)
	28#	(3)	(3)	
CONTAINMENT	14#	22.8”	18.8”	78” (4)
	28#	39.3”	26.2”	78” (4)
SPENT FUEL POOL	14#	22.8”	18.8”	76.6” (4)
	28#	(3)	(3)	
4160V SWITCHGEAR	14#	(3)	(3)	
	28#	45”	26.7”	114” (4)

NOTES:

1) Modified National Defense Research Committee formula used in the original Oyster Creek Analysis.

2) Recent empirical formula for concrete penetration (Reference 12) indicates that this formula is more applicable to turbine missiles than the NDRC equation.

3) Due to their relative locations, it is not possible for a low trajectory missile from this turbine stage to strike the subject target.

4) To be an effective barrier using either penetration formula, the thickness must be equal to or greater than twice the predicted penetration depth.

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

3.6.1 Design Considerations

The design of the Oyster Creek Nuclear Generating Station (OCNGS) regarding rupture of high energy piping inside or outside Primary Containment considered the following:

a. An instantaneous pipe break and the resulting pipe whip is considered to be 9improbable. The piping is considered ductile, and if a break occurs it will not be instantaneous nor have sufficient energy to cause a secondary failure.

b. The safety systems within the Primary Containment are arranged to take advantage of the redundancy of equipment.

c. The specific design of the Primary Containment and contained systems takes into account the potential for generation of missiles and minimizes the possibility of Primary Containment violation (see Section 3.5).

d. Conservative piping design utilizing proven engineering design practice, the proper choice of piping materials and the use of conservative quality control standards and procedures for piping fabrication and installation ensure that pipes will not break in such a manner as to bring about movement of the broken pipe sufficient to damage the components of the Emergency Core Cooling Systems or the Primary Containment.

e. Inservice inspection of piping is preferable to massive pipe restraints which can only have postaccident effectiveness.

3.6.2 Design Evaluations

3.6.2.1 Emergency Core Cooling System Evaluation

The following evaluation was performed for the Emergency Core Cooling System (Reference 1): there are two independent core spray loops physically located on opposite sides of the Primary Containment entering at El. 61'-9", azimuth 245° and El. 79'-4", azimuth 100°. In addition, the level and pressure sensors (that actuate emergency cooling) and lines are on opposite sides at El. 66'-6" and azimuths 60° and 212°. Therefore, given the unlikely event of a pipe break and whiplash, the worst that could occur would be the elimination of one of the core spray loops, leaving the other loop operational. It is also very possible that neither core spray loop would be affected depending upon the location of the break.

The high drywell pressure sensors do not extend beyond the inner wall of the drywell so they could not be damaged by a pipe whiplash. In addition, they are located at El. 47'-0" and at azimuths of 284°, 286°, 300°, 325°, and 335°.

Similarly, the hydraulic lines to the control rod drives are located at opposite sides of the pedestal. Due to the safety design of the control rod drives, a severance of the hydraulic lines will automatically scram the drives with reactor pressure. The remainder of the drives will scram with a safety system scram signal. Drives will scram in less than 10 seconds if vessel pressure is 550 psi, based on test results. At 8 seconds after a recirculation line break, reactor

pressure is at 600 psi. The feedwater lines enter the Primary Containment approximately 13 feet apart and then go in opposite directions within the Primary Containment.

The worst pipe break with respect to damage within the Primary Containment would be a circumferential break of a recirculation line at an elbow. This could generate the large side thrust necessary to whip the free end of the pipe. However, structural members reduce the probability of the whip damaging equipment within the Primary Containment. For example, there are two tiers of structural members in the drywell at El. 24' and El. 46', from which equipment is supported. The radial members of these tiers are 21 inch wide flange beams which would reduce the energy of whipping and lower the probability of damage.

Radial motion of the pipe at the lower tier will be resisted by an 8 inch channel bow assembly around each recirculation pipe. The upper tier includes an 8 inch channel assembly in the circumferential direction attached to each of the radial wide flange beams at approximately a 17 foot radius from vessel center. Below this upper tier will also be the 8 ton monorail system which will include about 270° of arc around the vessel as added resistance to any radial pipe motion.

From the above, it was concluded that pipe whip will neither affect the ability to safely shut down nor the adequacy of core cooling.

3.6.2.2 Evaluation of Other Pipes Inside Containment

For all pipes inside the Primary Containment that can possibly strike the drywell shell or other safeguard systems when whipping (due to single failure), the licensee was requested to identify those pipes that can exceed the threshold energy and specify the energy they can have, and to describe any modifications that would be required to prevent the whipping (Reference 2). The following response was provided:

Calculations were performed to show that pipe breaks which result in the maximum energy available for pipe whip are circumferential in nature and occur at the end of a long run of pipe in the vicinity of an elbow. The worst break is one in which the jet force from the elbow acts in a lateral direction such that the long run of pipe acts as a cantilever. As previously indicated, long pipe runs such as this encounter obstructions enroute to the Primary Containment wall which minimize the probability of penetration. In addition, the smooth shape of a curved elbow and the relatively large area of impact imply that large deformation of the Primary Containment shell would be required before complete penetration would occur. Experiments conducted by Chicago Bridge and Iron, Inc. (see Section 3.5) showed that for an impact area of 1.08 sq ft, the Primary Containment shell would deform to the point of contacting the concrete (radial deformation of approximately 2 3/4 inches) without causing the shell material to crack.

Some additional calculations were made to determine if a whipping pipe could possibly achieve the amount of energy needed to penetrate the Primary Containment, regardless of whether it was restrained by other structural members (i.e, girders, beams, concrete, etc.). The calculations were extremely conservative so that if the results indicated that penetration was not possible the results would be conclusive.

Some of the conservatism included in the calculations is as follows:

a. The Primary Containment shell was not permitted to deform to the point where credit could be taken for the support provided by the concrete.

b. The area of impact of a pipe with the Primary Containment was chosen as the undeformed area existing at the time of impact. No credit was taken for the large impact area that results from flattening of the pipe.

c. All of the energy from the break was assumed to propel the pipe. No losses due to pipe distortion were considered.

Four large pipes were examined which were assumed to be representative of the worst case. The summary of results is shown in Table 3.6-1.

The pertinent results found in the table are found under the heading "energy." The first column indicates the energy per foot of movement that a whipping pipe would attain for a circumferential (elbow) or longitudinal pipe break. The second column indicates the energy required to perforate the Primary Containment (based on the Stanford Research Institute formula). It can be seen from these energy comparisons that, based upon extremely conservative calculations, the energy released from an elbow break of a 26 inch recirculation line is excessive for even 1 foot of travel. However, the conservatism in the calculations did not permit the conclusion that the Primary Containment would be penetrated.

Subsequently (Reference 3), an evaluation was made of the feasibility of various "reasonable and practical measures" that could be relied upon to reduce the probability and consequences of pipe whip in the Oyster Creek drywell as was requested by the AEC. It was concluded that, while it may have been possible at the stage of near completion of the Oyster Creek plant to install the numerous, massive pipe restraints that would be required to restrain all of the various pipes in the drywell that have the potential for damaging the metal wall of the drywell, or interrupting the operation of engineered safeguard equipment, in the unlikely event that such a pipe would fail precipitously, that, in fact, the installation of a system of massive pipe restraints at that time would tend to create a less safe situation than could be established by other means. The nature of the pipe restraints that it might have been possible to provide at that time would have seriously reduced the capability to inspect for, and detect, small flaws and incipient cracks.

This finding resulted from the fact that the restraints would have to have been of such a size that it would have been impractical to remove the restraints at the regular intervals that inservice inspection of the piping in the drywell would be normally performed. Since the restraints would have been installed near sensitive sections of the pipes which would be restrained, if the restraints were to be effective in their function, it was also clear that the restraints, because of the size of the equipment that would be used, would have had to be removed to permit a competent inspection to be performed.

Preventive maintenance and regular inspection of sensitive pipe runs was concluded to be a safer method to be followed in assuring that large pipes in the reactor drywell will not fail rather than assuming that very unlikely failures can occur and providing massive restraining equipment that in themselves compromised the opportunity to perform maintenance and inspection activities.

The investigation of the consequences of an instantaneous pipe failure was made without regard to the credibility of such an event occurring in a high quality system. It was recognized that if an undetected fault could lead to an abrupt failure of any pipe in the reactor drywell area, it would be most likely that the failure would take place in a small diameter line. The

investigation of the potential of various pipe runs to cause damage in the event of a pipe failure concluded that there are no failures of small pipe systems which would initiate a sequence of events that would lead to penetration of the Primary Containment. However, the Recirculation System piping can generate sufficient energy to penetrate the Primary Containment if the pipe breaks in certain specific locations. The same is true of the main steam and feedwater piping and one of the isolation condenser return pipe runs.

The results of an AEC sponsored Reactor Primary Coolant System Rupture Study were used to evaluate the probability of complete severance of a primary system pipe. The failure rates and critical crack sizes for the piping of interest were evaluated using the results of the AEC study. The conditions for critical (or unstable) crack growth were based on the assumptions that the cracks grow to critical size by mechanically or thermally induced cyclic loading or stress corrosion cracking or some other mechanism characterized by gradual crack growth.

Earthquake and normal vibration stresses, since they were included in the piping design criteria, were considered in the determination of critical crack size. The case against pipe severance relies on the high probability that a crack will be detected by one or more of a number of surveillance techniques before it reaches the critical size due to the gradual growth characteristics of such cracks.

By a very conservative analysis, the probability of a critical severance break of a critical pipe was shown to be less than 1×10^{-6} in the plant life of 40 years. This conclusion was based on the following factors:

a. The rate of occurrence of leaking failures in reactor piping was estimated from failure history for BWR's to be 5.3×10^{-4} leaks per piping component per year. This rate was assumed to apply to all lines, with no credit taken for higher fabrication and inspection standards on large, critical lines.

b. It was assumed that a crack will grow until it is either detected or causes severance of the line. The leak detection capability in the Primary Containment was then estimated to be of the order of 5 gpm.

c. The critical crack size for complete severance was estimated from experimental data from the results of the AEC Pipe Rupture Study. The leak rate expected from this crack was calculated by a conservative analysis and found to be at least an order of magnitude greater than the detection capability of the Oyster Creek plant.

d. The probability that a critical leak will not be detected can be calculated by a comparison of the expected leak rate from a given crack to the detectable leak rate and taking into consideration the uncertainties associated with each event. The pipe failure probability prediction was obtained by combining the probability of the leak occurring with the probability that the leak will not be detected.

A detailed evaluation of all piping within the drywell was performed to establish the need for and the extent of pipe restraints. The philosophical basis that was taken for the evaluation was "for any one pipe, for any mode of failure," restrain it in a manner such that in spite of its failure:

a. Containment integrity will be maintained,

b. At least one core spray loop, including its instrumentation, will remain operable, and

c. At least one set of reactor pressure vessel level instrumentation will remain operable.

The design criteria for the application of restraints were as follows:

a. For any single mode of pipe failure, where the energy accumulated by the pipe in motion exceeds the energy absorption capability of the containment shell (E_{pipe} greater than or equal to $E_{containment}$), a restraint should be applied.

b. Restrain the pipe in a manner such that the full plastic moment of pipe is not exceeded.

c. Very conservative loading conditions shall be pressure times flow area:

$$F = P \times A$$

where:

$$P = 1250 \text{ psig - design condition}$$

The pipe energy is approximated by force times distance traveled:

$$E = F \times L$$

d. For longitudinal splits the pipe shall be assumed to be capable of developing its full plastic bending moment.

e. A value of 80 percent of ultimate material stresses shall be used in the design of the restraints and also for the calculation of the plastic moment of the pipe.

f. Restraint devices shall be so designed and located that accessibility and inspectability will not be compromised.

g. Restraint devices shall be designed and locat88ed such that they in no way affect the normal pipe movement.

h. The restraint devices shall be designed to accommodate the impact loading which results from the operating clearances required to satisfy g. above.

The method that was utilized to determine the energy of the pipe was:

a. Assume a pipe rupture.

b. Conservatively assume no restraint resulting from the normal pipe suspension system.

c. "Track" the resultant flight path of the pipe until its energy was dissipated in other piping or structure, or until it strikes the Primary Containment.

d. Conservatively assume no time delay effects for the pipe in flight.

e. Determine energy from the relationship E = P x A x L; where E_p is greater than or equal to E_{cont}, a restraint device should be applied. It must be noted that the 1250 psig pressure is applied only when the pipe is directly fed from the reactor pressure vessel. Reduced pressures are justified for several rupture cases when the pipe is fed from other than the reactor pressure vessel; for example, steam fed from the turbine stop valve steam chest through the steam line flow restrictor.

Based on the criteria and methods listed above, the pipes in the drywell were evaluated to establish their acceptability when compared to the criteria. Wherever E_{pipe} was greater than or equal to E_{cont} the need for a restraint device was indicated. Wherever any of the other criteria were judged to be potentially violated, a restraint device was to be added.

However, it was finally concluded that while restraints could be applied to the steam and feedwater lines to satisfy the criteria, it was not possible to apply restraints to the recirculation loops to meet all the criteria. The physical arrangement of the loops and the adjacent structure, and the load bearing capability of this structure, made the addition of restraints extremely difficult, if not impossible. Moreover, the addition of any substantial restraints to the already compact arrangement would significantly reduce the accessibility and inspectability of these areas. That is why inservice inspection of piping was found preferable to massive pipe restraints which can only have a post accident effectiveness.

Finally, it was reported (Reference 4) that detailed evaluations of the arrangement of instrument and process lines and of sensors within the drywell led to the conclusion that there are no process pipes that can fail and whip in such a manner as to prevent automatic actuation of the engineered safety features.

Considerations in this evaluation included: location of breaks with relationship to source of driving force (i.e., from reactor vessel side or outside the Primary Containment); stiffness of pipe; direction and distance of movement; momentum exchange between intervening structural and process components; location of instrument piping; and redundancy of instrumentation including operating modes permitted by the then proposed Technical Specifications.

Movement of smaller pipes did not lead to failure of larger pipes in any case examined.

The report concluded that failure and movement of pipes within the drywell would not damage the reactor vessel support structure. Due to possible break configurations combined with direction of travel and distance, no pipes would have the energy potential to damage the support structure.

3.6.2.3 Isolation Condenser Piping Evaluation

Analyses were performed to determine the dynamic response of the inside Primary Containment portion of the Isolation Condenser piping system following pipe rupture. The report on these analyses is presented in Appendix 3.6A.

The Isolation Condenser piping system was selected for study, as it was considered to be "typical" of the inside Primary Containment piping in the following respects:

a. The piping is of moderate size, and operating conditions are representative of the high energy systems in the plant.

b. The magnitude of blowdown forces associated with rupture is average for the systems.

c. The piping is located relatively close to potential load carrying structures for attachment of rupture restraints, and hence, the design difficulties for the restraint system are average.

d. Congestion and interference with adjacent systems are typical of the inside Primary Containment portion of the plant.

A mathematical model of the Isolation Condenser System from the reactor vessel connection to the postulated break location was constructed for use in the dynamic response analyses. The model consisted of lumped masses connected by massless pipe elements having nonlinear inelastic structural properties and a nonlinear inelastic spring representing the rupture restraint provided for the pipe. Dynamic time history analyses were performed for shock loading resulting from an instantaneous circumferential break at the postulated break location. The analyses included the effects of pipe impact on the rupture restraints, generalized yield behavior of the piping and kinematic strain hardening of the piping and restraint.

The blowdown force time histories were calculated at selected points in the piping system, and applied as time varying concentrated loads in the nonlinear structural response analyses. Total fluid force time histories were calculated from the time varying pressure and momentum conditions following rupture, and included the effects of pipe friction and two phase compressible fluid flow.

Both the response of the piping system and the rupture restraint provided were found within acceptable limits and the maximum energy absorbing capabilities of the restraint were utilized. The maximum displacements at all points in the piping system were sufficiently small to ensure that no interference with adjacent systems will occur from pipe whip.

It should be noted that the rupture restraint system and design details developed from the results of these analyses are applicable to the particular piping system for a specific break case, and would not adequately restrain the piping for other potential break locations, or longitudinal breaks at the same location.

3.6.2.4 Analysis of Pipe Breaks Inside Primary Containment

An evaluation was performed in order to determine compliance with Regulatory Guide 1.46, Protection Against Pipe Whip Inside Containment. The report on this evaluation is presented in Appendix 3.6B. The study concluded that of the 150 possible rupture points only 13 have stress levels greater than 50 percent of the allowable stress in Regulatory Guide 1.46. The OCNGS surveillance program reflects the actions taken as a result of these conclusions. The report referenced in Appendix 3.6B as the response to Question 2.a is Appendix 3.6A.

3.6.2.5 Biological Shield Wall Evaluation

Analyses of the biological shield wall have been performed to determine its capability to withstand the effects of postulated pipe failures.

For gross damage evaluation, a dynamic analysis was performed to determine the response of the shield wall upon being struck by a 24 inch line in the worst location. A review of resultant

moment and shear loads in combination with other loads, as required, indicated that the load carrying capability of the wall was substantially greater than the loads encountered. Therefore, it was concluded that the wall would withstand the full spectra breaks without suffering gross damage.

For local damage evaluation, penetration calculations were performed utilizing conservative impact assumptions. Even with these conservative assumptions, the predicted penetration was limited to 2/3 of the wall thickness under the most severe loads. An evaluation of spalling and gross damage due to whipping pipes indicated that damage would be restricted to the local area of impact and spalling would not take place due to the plates provided on the interior wall.

Except for jet impingement analyses, damage analyses were not performed for the Primary Containment vessel as it was conservatively assumed that all pipe ruptures resulting in impact on the Primary Containment would cause failure of the vessel. Any failure of the vessel was considered unacceptable and postulated breaks that could cause this type of damage are provided with increased surveillance.

For jet impingement loading, the Containment was analyzed and proven capable of withstanding such loads.

Appendix 3.6C entitled "Evaluation of Structural Integrity of the Biological Shield Wall Under Pipe Whip Loadings" describes the analyses performed in detail.

3.6.2.6 Effects of Postulated Pipe Failures Outside Containment

On December 18, 1972 the Regulatory Staff (AEC) requested a detailed design analysis to determine the consequences of a postulated rupture in any high energy fluid piping system outside the Primary Containment. The report in response to this request was submitted on July 1, 1974 as Amendment No. 75 to the Facility Description and Safety Analysis Report.

Based on this submittal and on additional information provided on December 24, 1974, March 24, 1974, April 25, 1975, June 1, 1976 and September 21, 1976, the NRC concluded in their Safety Evaluation transmitted by their letter, dated December 27, 1976 that the OCNGS reactor can be safely shutdown and maintained in a safe condition for any postulated pipe break in any high energy system outside the Primary Containment. The acceptance criteria used to review the effects of a pipe break outside containment were provided in the enclosure to the December 18, 1972 letter mentioned above. The December 18, 1972 acceptance criteria are also referenced in IE Bulletin 79-01B, "Environmental Qualification of Class IE Equipment" as a basis for evaluating qualification of Class IE equipment under harsh environment due to a high energy line break (HELB) outside containment.

In 1989 the NRC released a generic letter (GL 89-10) which required GPUN to review the motor operated valves (MOVs) at Oyster Creek. Included among these MOVs are the isolation valves associated with the isolation condenser and cleanup systems. As a result, the full guillotine high energy line break analyses for these systems were redone to establish design requirements for the MOVs. In addition, there has been a clarification to the list of equipment which will be environmentally qualified for the mitigation of the OC HELB events. The analyses performed and the equipment used to mitigate the events are summarized in Paragraphs 3.6.2.6.1 and 3.6.2.6.2.

3.6.2.6.1 Isolation Condenser System HELB

The Isolation Condenser system will isolate following a persistent high flow signal indicative of a pipe failure. The Isolation Condenser system is evaluated to assess the impact of a line failure in the system outside the primary containment. The piping failure assessments are conducted with and without the system in operation. The details of the assumptions and scenarios used in this evaluation are documented in Paragraph 3.6.3, Reference 6. The assessment of these line breaks is performed to ensure that the isolation valves isolate the system in an acceptable amount of time against the expected system pressures. The criteria used to establish acceptable isolation valve operation is:

a. The valve isolation prevents the core from uncovering ensuring adequate core cooling (Reference 6).

b. The valve isolation time protects the equipment required to mitigate the accident from excessive accident environmental conditions (Reference 7).

c. The valve isolation time ensures that the offsite release remains well within the 10 CFR part 100 accident environmental conditions (Reference 7).

In addition to the discussion in Amendment 75 (Reference 5), the following clarification regarding primary systems assumed to be available to mitigate a HELB in an isolation condenser system is provided.

a. The core spray system is assumed to be available for reactor vessel inventory makeup. Previously Amendment 75 excluded this system from the required systems, however, the system is now considered to be required. Each core spray system is single failure proof and can perform the inventory makeup function despite any interaction identified within Amendment 75 which may disable the redundant system. Only Core Spray System 2 may be disabled by a break in the Reactor Building.

b. The ADS system and the redundant isolation condenser system is assumed to be available for reactor vessel pressure control. Amendment 75 has identified a potential interaction which may disable the electrical redundancy of the ADS system. Either ADS Div. I or II may be disabled by a break inside the Drywell. However, the redundant ADS is still relied upon for pressure control along with the intact isolation condenser providing redundancy.

3.6.2.6.2 REACTOR WATER CLEANUP SYSTEM HELB WITH FEEDWATER AVAILABLE

It has been demonstrated analytically that a cleanup line break outside primary containment will not produce a lo-lo RPV water level signal if feedwater is available to supply makeup, Paragraph 3.6.3, Reference 8. To address this concern a temperature monitoring system was installed to isolate the system when the area temperature exceeds 180°F.

In addition to the discussion in Amendment 75 (Reference 5), the following clarification regarding primary systems assumed to be available to mitigate a HELB in the Cleanup System is provided.

1. Amendment 75 states that the CRD pumps and IC's are required to cope with a cleanup line break. The CRD pumps are not part of the E equipment list for qualification in HELB environments. The core spray system, however, is included in this E list. Core spray was not relied upon in Amendment 75 since one system could be failed due to pipe whip. This would leave the remaining system susceptible to a single active failure. However, the core spray systems are designed to be single failure proof (Section 6.3.1). Therefore, if pipe whip were to fail one core spray system the other would remain available even with a single active failure. Manual control of core spray is all that is required for mitigation of this event. Based upon the above argument, the pipe whip interaction with the one core spray system is acceptable and the core spray system can be used to mitigate the HELB. The CRD pumps are not required to mitigate the event and need not be included on the EQ list.

2. Amendment 75 states that the Auto Depressurization System I may be damaged by the pipe break effects while System II is not. In the interaction evaluation, Amendment 75 states: "The auto depressurization system is not required to cope with break B-15 event". Loss of system redundancy is permissible. A residual heat removal system is necessary to remove decay heat and cool the core. Therefore, without the auto depressurization system, the isolation condenser provides this function.

In summary, at least one core spray system and both isolation condensers need to be available to cope with the failure of a cleanup line system pipe in the Reactor Building. In addition, no further interactions that would disable auto depressurization system II and containment spray system I is allowed.

3.6.2.7 Assessment for the Systematic Evaluation Program

The issue of the effect of pipe breaks both inside and outside Primary Containment was reassessed under the Systematic Evaluation Program (SEP) developed for Oyster Creek. SEP Topics III-5.A and III-5.B, relating to pipe breaks, are discussed in Section 1.10.

3.6.3 References

1. Oyster Creek Nuclear Power Plant Unit No. 1, Facility Description and Safety Analysis Report, Docket No. 50-219, Amendment No. 5, June 5, 1967.

2. Oyster Creek Nuclear Power Plant Unit No. 1, Facility Description and Safety Analysis Report, Docket No. 50-219, Amendment No. 11, August 30, 1967.

3. Oyster Creek Nuclear Power Plant Unit No. 1, Facility Description and Safety Analysis Report, Docket No. 50-219, Amendment No. 34, March 22, 1968.

4. Oyster Creek Nuclear Power Plant Unit No. 1, Facility Description and Safety Analysis 2Report, Docket No. 50-219, Amendment No. 38, July 8, 1968.

5. Oyster Creek Nuclear Power Plant, Unit No. 1, Facility Description and Safety Analysis Report, Docket No. 50-219, Amendment No. 75, Rev. 0, July 1, 1974, through Rev. 5, September 21, 1976.

6. GPUN Calculation C-1302-153-5450-083, Rev. 1, "Evaluation of Isolation Condenser MOV Gear Mod Impact on RPV Response to a HELB."

7. GPUN Calculation C-1302-822-5450-059, Rev. 2, "Impact of MOV Gear Modification on RB Environment and Offsite Release."

8. GPUN Calculation C-1302-153-5450-070, Rev. 0, "OC RELAP5 Analysis of Cleanup Line Break."

9. GPUN Calculation C-1302-215-E610-061, Rev 1 “RBEQ Profiles Cleanup Line Modified Break Detection”.

10. GPUN Memo from A. F. Hertz to P. N. Hansen, "Response to PSC-93-001/Memo 5450-93-0053", August 24, 1993.

11. GPUN PSC 93-001 Safety Determination, dated September 9, 1993.

12. SE315403-038, Modification to provide additional cleanup System Break Detection.

TABLE 3.6-1
SUMMARY OF PIPE BREAK RESULTS

	Length of Pipe Considered	Maximum Force Per Unit Break Area	Theroretical Displacement at Collapse, Inches		Energy/Foot ** Of Displacement of Break 10^3 ft-lb/ft		Energy (10^3 ft-lb) Required to Perforate Containment With Thickness		
Pipe	Feet	10^3 psf	Elbow Break	Longitudinal Split	Elbow Break	Longitudinal Split	0.64 Inch	0.722 Inch	1.1875 Inch
10-inch Feedwater	40	103.00	2.70	3.50	47.60	0	70.0	95.0	300.0
18-inch Feedwater	40	81.70	19.9	2.50	96.00	0	95.0	125.0	415.0
26-inch Steam	40	175.5 or 58.1*	4.6	0.57	131.40	134.0	115.0	155.0	500.0
26-inch Recirc.	40	175.70	7.13	0.89	500.00	128.20	115.0	155.0	500.0

* Break upstream or downstream of flow limiter

** Calculated from thrust force (with break area equal to pipe flow area) times displacement minus plastic moment energy

3.7 SEISMIC DESIGN

The original seismic design for the Oyster Creek Nuclear Generating Station critical structures and equipment is based on dynamic analyses using acceleration response spectrum curves which were based on a peak ground acceleration of 0.11g for the Operating Basis Earthquake (OBE) and 0.22g for the Safe Shutdown Earthquake (SSE). Beginning in September, 1995, seismic design of equipment and structures is based on a peak ground acceleration of 0.092g for the OBE and 0.184g. for the SSE.

The two classes of structures which earthquake design requirements were applied to include:

a. Class I - Structures and equipment whose failure could cause significant release of radioactivity, or which are vital to a proper shutdown of the plant and the removal removal of decay heat.

b. Class II - Structures and equipment which may be either essential or nonessential to the operation of the station but which are not essential to a proper shutdown.

Structures and equipment originally categorized as Class I are listed in Table 3.2-1. Class II items were designed following the normal practice for the design of power plants in the State of New Jersey in effect at the time of design; as a minimum this was not less than given in the "Uniform Building Code" for Zone 1.

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The design of Class I structures and major pieces of equipment (Table 3.7-1) was based on a dynamic analysis using the acceleration response spectrum curves shown in Figure 3.2-1 which are based upon the recommendations of Dr. George W. Housner (Reference 1). These curves are based on the seismology, geology, and other pertinent data at the site. The acceleration response spectra curves for the recirculation system are contained in Reference 4 and are based on the uniform hazard ground motion spectra developed and recommended by the USNRC for Oyster Creek in Reference 5. This spectra provides an SSE zero period acceleration of 0.165g.

Beginning in September, 1995, equipment, components, supports and structural subsystems are designed on the basis of the SSE design response spectra shown in Figures 3.7-18 and 3.7-19 and contained in Reference 6. The response spectra were developed by Weston Geophysical Corporation using the 84 percent non-exceedance probability from 67 horizontal records and the corresponding 34 vertical records. The SSE Site Specific Response Spectra (SSRS) have a peak ground acceleration of 0.184g. horizontal and 0.0952g. vertical. These spectra were approved by US NRC in March, 1992. (Reference 7).

3.7.1.2 Design Time History

An acceleration time history for the Oyster Creek Site was developed by URS/Blume, and reported in the "Seismic Acceleration Floor Response Spectra for the Reactor Building at Oyster Creek Nuclear Power Plant", in December 1981. This report is included as Appendix 3.7A.

For the SSI analysis using the SSRS developed by Weston, three artificial time histories (two horizontal and one vertical) are generated by EQE International and presented in Reference 8. These time histories are shown in Figures 3.7-20 to 3.7-22. The time histories envelope the SSE target spectra in that no more than 5 points of the time history response spectra fall below the target spectra and no more than 10% below at any point. The comparisons of the time history response spectra and the SSRS are shown in Figures 3.7-23 to 3.7-25.

3.7.1.3 Critical Damping Values

The percentages of critical damping used for the seismic analyses of Class I structures, systems and components are listed in Table 3.7-2. The percentages of critical damping values used for the recirculation system piping and the Seismic Category I structures and components in the New Radwaste Building are those specified in NRC Regulatory Guide 1.61.

The percentages of critical damping used for seismic analysis after September, 1995, are those specified in NRC Regulatory Guide 1.61 or ASME Code Case N-411 for piping.

3.7.1.4 Supporting Media for Seismic Category I Structures

The base of the Reactor Building is approximately 140 ft x 140 ft and is embedded 50 feet below grade. The outside walls of the Reactor Building are constructed of reinforced concrete up to El. 119'-3". Above this elevation is a steel braced frame superstructure which carries the building's overhead crane.

The stack foundation was placed on the very dense sands, below the finite clay layer separating
the geologic formations at the site. The New Radwaste Building rests on about four feet of compacted backfill. The structural foundation is approximately 86 feet x 114 feet in plan. The height of the building is 44 feet. A constant soil density of 78 lb/ft^3 (buoyant weight of soil) was conservatively assumed for seismic analysis.

Details on the evaluation of the effects of soil properties on those parameters critical to seismic design are contained in Appendix 3.7A.

For use in generating new floor response spectra, a soil profile based on site specific borings and other data was developed by Geomatrix Consultants and provided in Reference 9. A table of the best estimate low strain soil properties is shown in Table 3.7-4. Reductions in shear modulus and damping ratios at higher strains are shown in Table 3.7-5.

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods

The seismic analysis for the design of the Reactor Building was performed using the following method:

a. The building was treated as a flexible cantilever system with masses lumped at each floor and roof level, and a point between the foundation mat and the first

floor. The resulting eight masses were considered to be supported by weightless elastic columns.

b. Natural frequencies and mode shapes of the equivalent eight mass system were computed with the aid of a digital computer.

c. After determining the natural frequencies and mode shapes for the building a maximum absolute acceleration for each mode of vibration was calculated.

d. The modal maximum acceleration of the first three modes were combined by taking the root mean square for each modal maximum acceleration.

e. Rocking of the building due to earthquake was then investigated. The acceleration effects of rocking and the first three modes were combined on a root mean square basis to produce acceleration, shear, moment and displacement curves which show a distribution over the height of the Reactor Building.

3.7.2.2 Natural Frequencies and Response Loads

The seismic analysis of Category I (Class I) structures are based upon the response spectrum normal mode method using lumped mass models of individual structures. This method utilizes the natural frequencies, mode shapes and appropriate damping coefficients of the system.

The Reactor Building model and analysis is described in Appendix 3.7A. Figures 3 and 4 of Appendix 3.7A show the horizontal and vertical models. Table 1 in Appendix 3.7A lists the maximum relative displacements, maximum absolute acceleration, maximum story shears and maximum overturning moments at each node point.

3.7.2.3 Procedure Used for Modeling

The Modal analysis response spectra method was used to determine seismic response of the structure using ground response spectra as shown in Figures 3.7-5 through 3.7-8.

The structure was modeled as a cantilever with masses at the floors, connected by weightless elastic springs. The structure has been analyzed in the two orthogonal horizontal directions and the vertical direction. Each mass was considered to have three degrees of freedom in each of the horizontal analyses, and one degree of freedom in the vertical analysis.

The horizontal mathematical model had a total of 12 degrees of freedom. The first six modes having frequencies less than 33 cps. The vertical model had a total of four degrees of freedom, with the first mode being the only one with a frequency less than 33 cps. To ensure participation of all significant modes of vibration, all modes causing an increase in response of 10 percent or more were included. Natural frequencies and response loads were determined for the horizontal and vertical directions, for the SSE with 7 percent damping. The representative maximum value of a particular response was obtained using the procedures outlined in Regulatory Guide 1.92 (12/74).

For use in generating new floor response spectra from the Weston Design response spectra, three-dimensional lumped-mass dynamic models of the reactor building, turbine building and intake structure have been developed. The reactor building model includes the reactor building,

drywell vessel, drywell shield wall, biological shield wall, and reactor pressure vessel. Damping is based on stiffness proportional composite damping calculated for the fixed base structure. Details of the model are provided in Reference 8 and the 3-D coupled model is shown in Figure 3.7-26. Details of the model for the remaining buildings are shown in Reference 10.

3.7.2.4 Soil Structure Interaction

A simplified lumped mass and soil spring approach was used to characterize the soil structure interaction of the New Radwaste Building. Translational, rocking, vertical and horizontal springs were calculated by considering elastic solid behavior and values of dynamic modulus of elasticity and Poisson's ratio. A dynamic modulus of elasticity of 3450 tons/ft^2 and a Poisson's ratio of 0.44 were used. To account for possible variations in the soil modulus, cases 30 percent above and below the stated value have been considered in the analysis.

Figure 3.7-2, 3.7-3 and 3.7-4 show the mathematical models for seismic analyses of the New Radwaste Building.

Soil structure interaction was also considered in developing the floor response spectra for the Reactor Building. The analysis report is presented in Appendix 3.7A. Soil springs were used to represent the stiffness of the soil. The Reactor Building responses are calculated for a soil shear modulus, G, of 6,000 KSF. Two values of G, 6000 KSF and 4000 KSF were used to generate floor response spectra and the results were enveloped.

In generating new floor response spectra based on the Weston design response spectra, Soil-Structure-Interaction (SSI) models have been generated for the reactor building, turbine building and intake structure. The key features of the reactor building SSI models are :

- Use of fully three-dimensional models of the soil and structure.
- Use of a rigid foundation for the portion of the building in contact with soil below grade.
- Sufficient discretization of the soil model to accurately model frequencies of 12 Hz and below based on the high strain best estimate solid profile.
- Full bonding between the structure and the soil.
- Use of three soil profiles: best estimate, lower bound and upper bound.

These features are based on US NRC Regulatory Guide 1.60 and the results of sensitivity studies described in Reference 8. Further details and descriptions of the reactor building model are contained in Reference 8 and details of the models for the remaining structures are contained in Reference 10.

3.7.2.5 Development of Floor Response Spectra

Horizontal and vertical broadened and smoothed acceleration response spectra are generated for the Reactor Building. A synthetic time history is generated whose spectrum closely matches the NRC site specific spectrum for the Oyster Creek plant. The derivation of the site specific spectrum is described in Appendix 3.7B. Soil-structure interaction is accounted for by using springs to represent the stiffness of the soil. The response of the structure is computed in the frequency domain to properly consider the frequency dependence of the soil impedences. Two bounding values of soil shear moduli of 6000 KSF and 4000 KSF are used for generation of floor response spectra which are then enveloped to produce the final spectra.

New floor response spectra for the reactor building, turbine building, emergency diesel generator building, and intake structure were generated in September, 1995, by EQE International. The spectra for the emergency diesel generator building are contained in Reference 13 and the spectra for the other structures are contained in Reference 10. Details of the methodology used for response spectra development are contained in Reference 8. The methodology described in Reference 8 is used to develop the response spectra for all four structures. Use of the spectra is accepted by the NRC in the SER attached to their letter dated February 23, 1995 (Reference 11).

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Two horizontal and one vertical spectra are generated at each specified location in the reactor building, intake structure, emergency diesel generator building, and turbine building. Three soil cases, a best estimate, an upper bound spectra are broadened ±10% and the best estimate spectra are broadened ±15%. The three spectra at each location are then enveloped to obtain the final spectra contained in Reference 10.

3.7.2.6 Three Components of Earthquake Action

For the design of Class I structures and equipment, the maximum horizontal acceleration and the maximum vertical acceleration were considered to occur simultaneously. The vertical acceleration assumed was equal to 2/3 the horizontal ground acceleration. Stresses arising from an earthquake in both the horizontal and vertical direction, and which occur simultaneously at a particular location are added directly and linearly. For all piping systems analyzed using response spectra methods, one vertical and two horizontal acceleration curves (Figure 3.7-1) will be applied simultaneously to a dynamic computer model of the piping system configuration with the modal results through a minimum of 33 Hz being combined by the square root sum of the squares SRSS).

For equipment and seismic subsystems analyzed or evaluated using the new EQE response spectra contained in Reference 10, the three directional components of the earthquake are combined using SRSS.

3.7.2.7 Combination of Modal Responses

In the analysis of the Reactor Building the modal maximum acceleration for the first three modes were combined by taking the root mean square of each modal maximum acceleration. The acceleration effects of rocking and the first three modes were then also combined using the root mean square method to produce the final distributions of shear, moment and acceleration.

3.7.2.8 Interaction Between Class I and Class II Structures and Components

The relative motion of the Turbine Building and Reactor Building has been analyzed. The displacement of the Reactor Building and Turbine Building was determined from the method outlined in Reference 14.

The maximum relative displacements between the Reactor Building and Turbine Building has been recalculated based on EQE analysis (Reference 10 and 14) and are shown on Figure 3.7-17. The summary of relative displacements is presented in Table 3.7-3. Cable trays and conduit that are interconnected between the Reactor and Turbine Buildings are provided with expansion fittings and cable slack to allow for differential building displacements.

The maximum expected relative displacement between the buildings is 0.338 inches at the top elevation (El. 74') of the Turbine Building. Maximum movements were considered in setting the minimum separation of the buildings at 3 inches (2 inches are constructed) to assure no contact between the structures during earthquakes. The existing minimum 2 inch separation occurs only at the concrete floor and roof slabs and at the end walls, and is closed with sliding plates and flexible flashing. Walls, structurally part of the Turbine Building serve as common walls above grade; below grade the walls of the separate building are 3 feet apart, allowing space for flexible sleeves for interconnecting piping and for flexible conduit connections for interconnecting cables. The stress analyses of piping systems crossing the building interface consider the range of predicted building deflections. Above grade, cables are in trays at predicted building deflections. Above grade, cables are in trays at various elevations up to El. 57', where maximum combined deflection is expected to be 0.326 inch; trays terminate at each side of the building interface and cable slack sufficient for a 2 inch movement is provided. Differential settlement of the buildings has been considered and should not significantly affect interconnecting pipe or cable.

The New Radwaste Building has been designed so that failure of non-Category I structures and components will not affect Seismic Category I structures.

In the original design of the facility, the Service Water System was designated as a Class I system whereas the intake and discharge structures were designed as Class II structures. It became necessary to perform a structural analysis to determine the extent of damage that might result to these structures due to a maximum earthquake and to relate this damage to the degree of operability of the Service Water System.

The structures were evaluated and found adequate to withstand seismic loads of 0.11g and 0.22g and remain within the allowable stresses for Class I structures.

3.7.2.9 Effect of Parameter Variation on Floor Response Spectra

The most significant parameters in the lumped parameter approach to soil structure interaction is the shear modulus, G, of the soil. In determining the floor response spectra for the Reactor Building two values of G were used, 6000 KSF and 4000 KSF, and the results enveloped. For further details see Appendix 3.7A.

Three soil profiles, a best estimate, a lower bound and an upper bound are used in the EQE analysis. These profiles are described in Reference 8. In addition, the effects of other

parameter variations are incorporated into the analysis based on the results of various sensitivity studies as described in Reference 8.

3.7.2.10 Methods Used to Account for Torsional Effects

Torsional effects have been taken into account by considering: 1) a torsional degree of freedom at each modal point; 2) the polar mass moment of inertia of each mode; 3) the eccentricity between the center of mass and center of rigidity at each floor level; and 4) a torsional soil spring attached to the foundation.

Torsional and rocking effects are included directly in the EQE analysis by massless rigid links connecting the center of mass to the extreme locations at each floor. The spectra at the extreme locations are enveloped together with the spectra at the center of rigidity to obtain the spectra contained in Reference 10.

3.7.2.11 Comparison of Responses

Only the response spectrum method was used to determine the earthquake forces in the structure. Therefore comparison with the time history method was not made.

3.7.2.12 Methods for Seismic Analysis of Category I (Class I) Dams

No dams are used on this project. Therefore, this section does not apply.

3.7.2.13 Determination of Seismic Category I (Class I) Structure Overturning Moments

Overturning effects on Seismic Category I structures in the New Radwaste Building were developed considering horizontal seismic loading in combination with the effects of vertical seismic loads and rotational moments at each mass point. The results of the dynamic analysis were converted to equivalent static loads and moments at the mass points.

3.7.2.14 Analysis Procedures for Damping

An equivalent modal damping procedure was used to represent the combined damping of the structure and the radiation damping of the soil. This method was used by URS/Blume in its parametric study. Subsequent analysis performed by URS/Blume to generate the floor response spectra used the frequency domain approach. This is described in Appendix 3.7A.

In the EQE analysis, described in Reference 8, stiffness proportional composite damping values are obtained from a fixed base model and incorporated into the SSI analysis.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods

Analyses were performed on various components to determine their capability to meet Class I requirements. The evaluation of the equipment was performed using the following procedures:

a. Used the peak values given in Reference 1, normalized to 0.11g ground acceleration (see Figure 3.7-1).

b. Using this acceleration, checked the adequacy of the anchor bolts and support feet or legs as appropriate. If stresses were within allowable, no further analysis was required. For anchor bolts embedded in concrete, the embedment lengths used were those required by the Uniform Building Code, 1967 Edition, for the allowable forces used in the analyses. Actual embedment lengths are all greater than those assumed.

c. If stresses exceeded allowable, a conservative estimate was made of the period of the equipment. Using this period and the acceleration response spectrum in Reference 1, rechecked the anchor bolts and support feet. If stresses were within allowable, no further analysis was required.

d. If stresses exceeded allowable, recommendations were made to insure that equipment meet Class I requirements.

This procedure was considered adequate since most of the equipment analyzed is located on the Turbine Building basement floor, and the response spectra from Reference 1 could be used for this floor as the floor response spectra.

The components that were determined to meet Class I requirements are:

a. Condensate Pumps

b. Steam Jet Air Ejector Intercondenser

c. Steam Jet Air Ejector Aftercondenser

i. Low Pressure Feedwater Heaters

j. Feedwater Heater Drain Coolers

k. Feedwater Pumps

l. Main Condenser

Those components which did not meet Class I requirements were:

a. Condensate Demineralizer Mixed Bed Tanks

b. Intermediate Pressure Feedwater Heaters

c. High Pressure Feedwater Heaters

d. Condensate Storage Tank

e. Condensate Pre-Filter Vessels

3.7.3.2 <u>Dynamic Design of Pipe Systems</u>

3.7.3.2.1 Dynamic Analysis of Core Spray and Containment Spray Pipe Systems

The dynamic analysis of the Core Spray and Containment Spray Pipe Systems, from the Suction Header connections to the Torus, through to the Main Core Spray Pumps and the Containment Spray Pumps Suction Nozzles is governed by the Mark I Containment Long- Term Program rules described in Section 3.8.2 of the UFSAR.

The dynamic analysis of the Core Spray and Containment Spray Systems from the discharge nozzles of the Core Spray Main Pumps and the Containment Spray Pumps to the RPV and Containment is governed by the rules described in Section 3.9.3.1 of the UFSAR, the details of which are described in Specification SP-1302-12-294.

3.7.3.2.2 Dynamic Analyses of the Remaining Eleven Seismic Class I Pipe Systems

The dynamic analyses of the remaining eleven Seismic Class I systems are governed by the rules described in Section 3.9.3.1 of the UFSAR, the details of which are described in Specification SP-1302-12-294.

3.7.3.2.3 Dynamic Design of Pipe Systems Other Than Seismic Class I Large Bore Pipe

Small Bore Pipe and Large Bore Pipe systems such as the Condensate and Feedwater piping in the Turbine Building evaluated for meeting Class I standards used the procedure presented below:

a. Knowing the period of the supporting building or structure, established the period of the pipe under conditions wherein the pipe is rigid, flexible or resonant.

b. Established the maximum span for pipe of various diameters to sustain a load of 0.5g and not stressed more than 1500 psi. (Following the requirements of Table 121.4 in Power Piping Code USAS B31.1-1967)

c. Established the resonant limits for pipe of various diameters by using the curves in Figure 3.7-14.

d. After selection of the length of pipe, used Figures 3.7-15 and 3.7-16 to determine the deflection and reaction at the supports.

The displacement and support reactions were increased, where required, by a factor of three due to magnification of response of the equipment over ground accelerations. Span lengths were reduced to account for valves or branch lines. For 90-degree bends, either leg is no more than L/2 were L is the sum of the span length for both legs.

Supports are located in a manor to avoid the resonant range. Pipe spans in the rigid range of Figure 3.7-14 have acceptable deflections, reactions and stresses. Pipe spans in the flexible range were analyzed for seismic loads and stresses induced by these loads were held to acceptable values. Supports and snubbers for pipe in this category were designed to limit stress to acceptable values.

3.7.3.3 Seismic Verification of Control Room Cabinets/Consoles

A seismic evaluation of the Control Room cabinets/consoles containing Safety Related (Q) and Augmented Quality (A) instrumentation and controls was performed in accordance with Seismic Qualification Utility Group (SQUG) and Senior Seismic Review and Advisory Panel (SSRAP) guidelines. The evaluation was based on the performance of similar cabinets/consoles, devices and sub-components in strong-motion earthquakes.

To verify the applicability of the experience data base for the seismic qualification of these components, certain caveats and requirements had to be met. The site SSE response spectrum is enveloped by the Generic Bounding Spectrum. The Control Room is located within 40 feet of grade, the devices adequately anchored to preclude unacceptable movement and interactions and the equipment is covered by the experience data base. Good workmanship during contruction is evident. Therefore, all existing Control Room cabinets/consoles, excluding anchorage, are verified as adequate for the design SSE seismic loading of OCNGS. Complete cabinet/console anchorage will be verified during the USI A46 walkdown.

All control room cabinets/consoles were evaluated during the walkdowns performed by GPUN to resolve USI A46. The results of this evaluation are described in Reference 12. During this walkdown, all anchorage was evaluated and either shown to be adequate or modified to ensure that all equipment fulfills its intended function during a SSE.

3.7.3.4 Seismic Subsystem Analysis Subsequent to September, 1995

Beginning in September, 1995, equipment and seismic subsystems are analyzed, evaluated, and designed using the new EQE response spectra described in Reference 8 and contained in Reference 10. These spectra are used in conjunction with damping values specified in US NRC Regulatory Guide 1.61 or in ASME Code case N-411 for piping. The three components of the earthquake are applied simultaneously using SRSS.

In addition, the new EQE spectra is used together with earthquake experience data and methodologies developed by the seismic qualification utilities group to evaluate the seismic adequacy of existing, new and replacement equipment.

> GPUN will utilize seismic experience data as an acceptable alternative to existing analytical or test methodologies to demonstrate seismic adequacy of existing, new and replacement equipment within the scope of the Generic Implementation Procedures (GIP) (Reference 3.7.4.15).

Use of experience data for existing, new and replacement equipment will be implemented with appropriate procedures, controls and limitations. Each application will be documented and signed by individuals who are qualified in accordance with the GIP and appropriately trained. Documentation and certification of seismic adequacy will be maintained for future reference.

3.7.4 References

1. Housner, "Report on the Earthquake Design Criteria for Jersey Central Nuclear Power Plant", March 1964, and Revisions dated 14 May 1964.

2. Woodward-Moorhouse and Associates, "Geotechnical Study, Proposed Radwaste and Off Gas Buildings, Oyster Creek Nuclear Power Station," February 1975.

3. Weston Geophysical, "Eastern United States Tectonic Structures and Provinces Significant to the Selection of a Safe Shutdown Earthquake," August 1979.

4. "Seismic Response Spectra for Reanalysis of Recirculation System Piping - Oyster Creek Nuclear Generating Station," URS/J.A. Blume and Associates, Engineers, San Francisco, California, October, 1986".

5. USNRC Letter LS05-81-06-068 dated June 17, 1981 forwarding site specific ground motion spectra for Oyster Creek.

6. Letter, G. C. Klimkiewicz, Weston Geophysical Corp. to A. P. Asfura, EQE, "Site Specific Response Spectra, Oyster Creek Nuclear Generating Station", October 14, 1992.

7. Letter, A. Dromerick, NRC to J. J. Barton, GPUN, Review and Evaluation of the Site Specific Response Spectra - Oyster Creek Nuclear Generating Station (M68217), Docket No. 50-219, March 18, 1992.

8. EQE International, "Design Criteria for Soil-Structure Interaction Analysis of the Reactor/Containment Building at GPUN Oyster Creek Nuclear Generating Station", Rev. 0, June, 1993, File No. 990-2191.

9. Geomatrix Consultants, "Soil Profile and Dynamic Soil Properties for Soil Structure Interaction Analysis of Reactor Building, Oyster Creek Nuclear Generating Station, New Jersey," Report No. 1957-1, Rev. 0, October, 1991.

10. EQE International, "Design Basis Seismic Response Analyses for the Oyster Creek Nuclear Generating Station Reactor Intake, and Turbine Buildings," Report No. 50069-R-001, Rev. 0, September, 1995.

11. Letter, A. Dromerick, NRC to J. J. Barton, GPUN, "Review and Evaluation of the Soil Structure Interaction Analysis and Approach Proposed for Generation of Instructure Spectra (M69467)", Docket No. 50-219, February 23, 1995.

12. EQE International, "Oyster Creek Nuclear Generating Station, USI A46 Seismic Evaluation Report," Report No. 42112-R-001, Rev. 0, March 1, 1996.

13. EQE International, "In-Structure Response Spectra for the Oyster Creek Nuclear Generating Station, Compilation of Response Spectra for Use in USI A-46 Program," Report No. 50124-R-001, Rev. 0, May, 1994.

14. EQE International, “OCNGS Reactor Building & Turbine Building Displacements” Calculation No. 200494-C-001, Rev 0, November, 1997.

15. Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 2, as corrected by the NRC SER Supplement No. 1 to generic letter 87-02, dated May 22, 1992.

TABLE 3.7-1

CLASS I CRITICAL STRUCTURES AND EQUIPMENT*

Deleted - Refer to Table 3.2-1

TABLE 3.7-2
(Sheet 1 of 1)

DAMPING FACTORS

Item	Percent of Critical Damping
Reinforced Concrete Structures	10.0
Steel Frame Structures	2.0
Welded Assemblies	1.0
Bolted and Riveted Assemblies	2.0
Vital Piping System	0.5
Reinforced Concrete Stack	5.0

TABLE 3.7-3
(Sheet 1 of 1)

SUMMARY OF RELATIVE DISPLACEMENTS
REACTOR BUILDING TO TURBINE BUILDING

(East-West Direction SSE)

Elevation (feet)	Foundation Motion (inches)		Relative Displacement (inches)		Total Displacement (inches)
	Translation	Rocking	Turbine Building	Reactor Building	
0.0	0.23	Note 1	Note 1	Note 1	0.255 Note 1
9.0	0.23	0.029	0.002	0.003	0.264
23.5	0.23	0.038	0.005	0.006	0.279
46.5	0.23	0.053	0.007	0.013	0.304
63.5	0.23	0.065	0.013	0.018	0.326
74.0	0.23	0.072	0.015	0.021	0.338

Note 1: Total displacement was not calculated for this elevation but was obtained from Figure 3.7-17 by projecting the linear curve between elevation 9'-0" and 23'-6" to elevation 0'0".

Table 3.7-4

Best Estimate Low Strain Soil Properties

Depth (ft)	Elevation (ft.)		Shear Wave Velocity (fps)	Compressional Wave Velocity (fps)	Poisson's Ratio	Total Unit Weight (pct)	Soil Type
0.0 - 3.0	+23	+20	310	730	0.39	120	Sand
3.0 - 10.5	+20	+12.5	600	1,413	0.39	120	Sand
10.5 - 18.0	+12.5	+5	660	1,554	0.39	120	Sand
18.0 - 26.0	+5	-3	735	5,200	0.49	115	Clay
26.0 - 34.0	-3	-11	810	5,200	0.49	115	Clay
34.0 - 43.0	-11	-20	930	5,600	0.49	125	Sand
43.0 - 52.0	-20	-29	985	5,600	0.48	125	Sand
52.0 - 55.0	-29	-32	1,170	5,600	0.48	125	Sand
55.0 - 61.5	-32	-38.5	1,270	5,600	0.47	125	Sand
61.5 - 68.0	-38.5	-45	1,145	5,600	0.48	125	Sand
68.0 - 78.0	-45	-55	1,130	5,600	0.48	125	Sand
78.0 - 80.0	-55	-57	1,260	5,600	0.47	125	Sand
80.0 - 85.0	-57	-62	1,415	5,600	0.47	125	Sand
85.0 - 92.5	-65	-69.5	1,455	5,600	0.46	125	Sand
92.5 - 100.0	-69.5	-77	1,400	5,600	0.47	125	Sand
100.0 - 108.0	-77	-85	1,185	5,600	0.48	125	Clay
Below 108.0	Below	-85	1,550	5,900	0.46	125	Sand

Note: Poisson's Ratio calculated as: $= 1/2 \dfrac{(V_p/V_s)^2-2}{(V_p/V_s)^2-1}$

Table 3.7-5

Shear Modulus Reductions and Equivalent Damping Ratios at Selected Strain Levels

(A) Shear Modulus Reductions

	Shear Modulus Reduction	
Strain (%)	Clay	Sand
0.000100	1.000	1.000
0.000316	0.999	0.985
0.001000	0.997	0.961
0.003162	0.972	0.897
0.010000	0.903	0.791
0.031623	0.772	0.600
0.100000	0.535	0.372
0.316228	0.293	0.206
1.000000	0.134	0.087

(B) Equivalent Damping Ratios

	Equivalent Damping Ratios (%)	
Strain (%)	Clay	Sand
0.000100	2.0	0.9
0.000316	2.3	1.0
0.001000	2.7	1.4
0.003162	3.3	2.2
0.010000	4.3	4.1
0.031623	6.0	7.5
0.100000	8.6	11.9
0.316228	13.7	18.3
1.000000	20.2	22.8

3.8 DESIGN OF CATEGORY I STRUCTURES

The General Electric Company was the prime contractor for Jersey Central Power and Light Co. in the design and construction of the Oyster Creek Nuclear Generating Station (OCNGS). Thus, General Electric had the overall responsibility for the Containment System as a part of the total plant.

General Electric Company engaged the services of Burns & Roe, Inc. for engineering assistance and construction management. General Electric furnished the conceptual information drawings, design criteria, and design specifications. Burns & Roe was responsible for the detailed design, construction drawings, specifications, and management of the actual construction and installation. All Burns & Roe drawing information was supplied to General Electric who had the privilege of review and approval.

Burns & Roe, Inc., subcontracted the design, construction, and testing of the drywell and torus vessel, and vent system work to Chicago Bridge & Iron Company.

Subsequent to the initial design and the start of commercial operation, certain modifications were made to the torus under the Mark I Containment System Evaluation Program. This program is further discussed in Subsection 3.8.2.

Evaluations of the structural soundness of the Drywell were performed during 1986 and 1987. The results of these evaluations showed evidence of Drywell wall thinning at various locations. These evaluations, the results thereof, and mitigative measures, as applicable, are discussed in Section 3.8.2.8.

In addition, under the Systematic Evaluation Program (SEP), and independent review was conducted of the seismic design aspects of the OCNGS as they relate to overall design margins. The report "Seismic Review of the Oyster Creek Nuclear Power Plant as Part of the Systematic Evaluation Program", NUREG/CR-1981, UCRL-53018, RD, RM, was issued to summarize the evaluation program. The SEP is summarized in Section 1.10.

3.8.1 Concrete Containment

Not applicable

3.8.2 Steel Containment

The Function of the Primary Containment System is to accommodate, with minimum leakage, the pressures and temperatures resulting from the break of any enclosed process pipe, and thereby limit the release of radioactive fission products to values which will insure offsite dose rates well below 10CFR100 guideline limits. The design integrated leak rate for the system is no greater than 0.5 percent of its total volume per day at 35 psig.

The development, design, fabrication and construction of the OCNGS Primary Containment are discussed in detail in Reference 1. For the design and construction of the Primary Containment, Burns & Roe prepared a detailed design specification and bid package from design criteria information supplied by General Electric. Chicago Bridge & Iron assumed responsibility for providing the primary components of the Containment System. All design and construction drawings were submitted to Burns & Roe for approval and to General Electric for review prior to construction. Included in this package were openings and sleeves (nozzles) through the drywell

wall to accommodate the penetration of process piping, instrumentation, and electrical lines. The actual penetration line fixtures and seals design, fabrication, and testing was subcontracted by Burns & Roe to piping or electrical fabricators as appropriate.

Subsequent to the design completion and start of commercial operation, additional loading conditions which arise in the functioning of the pressure suppression concept utilized in the Mark I Containment System design were identified. These additional loading conditions resulted in an industry wide reanalysis and modification program which is briefly described in the following paragraphs.

<u>Mark I Containment System Evaluation Program</u>

<u>Background</u>

The original design of the Mark I Containment System considered postulated accident loads previously associated with containment design. These included pressure and temperature loads associated with a Loss-of-Coolant Accident (LOCA), seismic loads, dead loads, jet impingement loads, and hydrostatic loads due to water in the suppression chamber. However, after establishment of the original design criteria, additional loading conditions which arise in the functioning of the pressure suppression concept utilized in the Mark I Containment System design were identified. These additional loads resulted from dynamic effects of drywell air and steam being rapidly forced into the suppression pool (torus) during a postulated LOCA and from suppression pool response to safety relief valve (SRV) operation generally associated with plant transient conditions.

Because these hydrodynamic loads had not been considered in the original design of the Mark I containment, the Nuclear Regulatory Commission (NRC) required that a detailed reevaluation of the Mark I containment system be made. In February and April 1975, the NRC transmitted letters to all utilities owning BWR facilities with the Mark I containment system design, requesting that the owners quantify the hydrodynamic loads and assess the effect of these loads on the containment structure. The February 1975 letters reflected NRC concerns about the dynamic loads from SRV discharges, while the April 1975 letters indicated the need to evaluate the containment response to the newly identified dynamic loads associated with a postulated design basis LOCA.

As a result of these letters from the NRC, and recognizing that the additional evaluation effort would be very similar for all Mark I BWR plants, the affected utilities formed an "ad hoc" Mark I Owners Group, and GE was designated as the Group's lead technical organization. The objectives of the Group were to determine the magnitude and significance of these dynamic loads as quickly as possible and to identify courses of action needed to resolve any outstanding safety concerns. The Mark I Owners Group divided this task into two programs: a Short Term Program (STP) and a Long Term Program (LTP).

<u>Short Term Program</u>

The objectives of the Short Term Program (STP) were to verify that each Mark I Containment System would maintain its integrity and functional capability when subjected to the most probable loads induced by a postulated design basis LOCA, and to verify that the licensed Mark I BWR facilities could continue to operate safely without endangering the health and safety of the public while a methodical, comprehensive Long Term Program (LTP) was being conducted.

The STP structural acceptance criteria used to evaluate the design of the torus and related structures were based on providing adequate margins of safety; i.e., a safety to failure factor of 2, to justify continued operation of the plant before the more detailed results of the LTP were available.

The results of the Short Term Program evaluation of the Oyster Creek torus were submitted to the NRC by Jersey Central Power and Light in 1976. As a part of that program, a drywell to wetwell differential pressure was imposed to reduce LOCA loads and a quencher was installed on the SRV discharge line to reduce SRV discharge transient induced loads. The conclusion of the Short Term Program evaluation was that the Oyster Creek torus met the criteria established for the Short Term Program.

The NRC concluded that a sufficient margin of safety had been demonstrated to assure the functional performance of the containment system and, therefore, any undue risk to the health and safety of the public was precluded. These conclusions were documented in the "Mark I Containment Short Term Program Safety Evaluation Report,"

NUREG-0408, dated December 1977. The NRC granted the operating Mark I facilities an exemption relating to the structural factor of safety requirements of 10CFR50.55(a) for an interim period while the more comprehensive LTP was being conducted.

<u>Long Term Program</u>

The objectives of the Long Term Program (LTP) were to establish conservative design basis loads that are appropriate for the anticipated life of each Mark I BWR facility (40 years), and to restore the originally intended design safety margins for each Mark I Containment System. The plans for the LTP and the progress and results of the program were reviewed with the NRC throughout the performance of the program.

The LTP consisted of:

a. The definition of loads for suppression pool hydrodynamic events

b. The definition of structural assessment techniques

c. The performance of a plant unique analysis (PUA) for each Mark I facility

The generic aspects of the Mark I Owners Group LTP were completed with the submittal of the "Mark I Containment Program Structural Acceptance Criteria, Plant Unique Analysis Application Guide" (PUAAG), NEDO-24583-1. The NRC concluded that load definitions and structural acceptance criteria documented in these two reports were acceptable for use in the plant-unique analysis of each plant. The NRC conclusions and comments were presented in the "Mark I Containment Long Term Program Safety Evaluation Report", NUREG-0661, dated July 1980.

<u>Summary of Results</u>

The analysis of the Oyster Creek torus and vent system has been performed in conformance with the requirements of the Mark I Containment Long Term Program. As a result, a number of structural modifications were designed for installation in the OCNGS Primary Containment as part of the Long Term Program.

The results of the analysis, which assumed that the modifications were completed, show that all components of the torus and vent system meet the criteria of the Mark I Long Term Program. Thus, the functional performance of the OCNGS Containment System will be assured for both Loss-of-Coolant Accidents (LOCA) and Safety Relief Valve (SRV) discharge suppression pool hydrodynamic loading conditions. Specific results of the analysis are given in the report "Plant-Unique Analysis Report, Suppression Chamber and Vent System", MPR-733 dated August 1982.

No evaluation of the Oyster Creek drywell was required in the Mark I Containment Long Term Program, since the maximum drywell pressure specified for Oyster Creek in the Long Term Program (NEDO-24572 Rev 2) is well within the design value specified in the original containment design.

The analysis of the piping systems attached to the Oyster Creek torus and vent system has been completed in conformance with the requirements of the Mark I Containment Long Term Program.

A number of piping and pipe support structural modifications were designed for installation as part of the Long Term Program. The analyses are based on the piping arrangement with all modifications installed. The loads used in the analyses of the piping are based upon the response of the Oyster Creek Containment modified as described in the report "Plant-Unique Analysis Report, Suppression Chamber and Vent System", MPR-733, dated August 1982.

The results of the analyses of piping systems attached to the Oyster Creek torus and vent system show that all piping, pipe hangers and supports, nozzles and related components meet the criteria of the Mark I Containment Long Term Program with the modifications completed. Specific results of the analyses are given in the report "Plant Unique Analysis Report, Torus Attached Piping", MPR-734, dated August 1982. These results were updated in MPR-999, Revision 3, "Addendum to MPR-734." (Reference 41)

An evaluation of the nozzles in the vent system for the Electromatic Relief Valves piping penetrations has been performed. The results, as presented in the report MPR-772, "Plant Unique Analysis Supplemental Report," indicate that all stresses are below ASME Code allowables and therefore, the penetrations meet the requirements of the Mark I Containment Long Term Program.

The Mark I Containment Long Term Program Confirmation Order dated January 19, 1982 required plant modifications needed to comply with the Acceptance Criteria in Appendix A of NUREG-0061, Mark I Containment Long Term Program, dated July 1980. This program is now complete for OCNGS.

Subsequent to the completion of this Mark I Containment Long Term Program, the high pressure actuation setpoints, specified by the Technical Specifications, were increased by 15 psig (Reference 45). To support this increase, an evaluation of the impact of the increased setpoints on Mark I results was completed (Reference 46). This evaluation utilized an estimation of, not a determination of, the resulting increases in stress levels. The results of this estimation were accepted as sufficient bases for assessing the impact of the setpoint increase on previously determined Mark I long term results.

3.8.2.1 Description of the Containment

The Primary Containment consists of a pressure suppression system with two large chambers as shown in Figure 3.8-1. The drywell houses the reactor vessel, the reactor coolant recirculating loops, and other components associated with the reactor system. It is a 70 ft diameter spherical steel shell with a 33 ft diameter by 23 ft high cylindrical steel shell extending from the top.

The pressure absorption chamber* is a steel shell in the shape of a torus located below and around the base of the drywell. It has a major diameter of 101 ft, a chamber diameter of 30 ft, and is filled to approximately 12 ft depth with demineralized water. The structure is made up of 20 mitered wedge shaped sections or bays with internal stiffening rings or ring girders at each miter.

The two chambers are interconnected through 10 vent pipes 6 ft 6 in in diameter equally spaced around the circumference of the pressure absorption chamber which feed into a common header inside the pressure absorption chamber. This header also takes the shape of a torus of 101 ft major diameter by 4 ft 7 in minor diameter. There are 120 downcomer pipes, 2 ft in diameter, uniformly spaced which have their open ends extending 3 ft below the minimum water level in the pressure absorption chamber. Gas phase return lines with vacuum breaker valves feed back gas to the drywell in case its pressure is less than the absorption chamber.

The base of the drywell is supported on a concrete pedestal conforming to the curvature of the vessel. For erection purposes a structural steel skirt was first provided supporting the vessel. A portion of the steel skirt was left in place to serve as one of the shear rings intended to prevent rotation of the drywell during an earthquake.

After erection, concrete was poured up to the level of the vessel floor providing uniformity in the support by following the contour of the drywell vessel.

A three inch clearance has been provided between the steel vessel of the drywell and the concrete drywell shield wall to provide for a regulated expansion of the drywell steel shell. This clearance was achieved by applying a compressible material to the outside of the drywell vessel prior to placement of the shield wall concrete. For further detail refer to Subsection 3.8.2.4.

The vent header is supported by pinned columns inside the absorption chamber. The downcomers are connected in pairs by pinned braces.

Projecting downward from the vent pipe header are downcomer pipes, terminating below the water surface of the pool. During a Loss-of-Coolant Accident (LOCA), the upward reaction from the downcomers is resisted by columns to the bottom of the absorption chamber. Due to the vent clearing jet forces the columns are pinned top and bottom to accommodate the differential horizontal movement between the header and the pressure absorption chamber. The horizontal reaction from the downcomers is resisted by the pinned braces.

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* The pressure absorption chamber is identified often in various reference documents, drawings, and figures as suppression chamber, wetwell, or torus.

Jet deflectors are provided in the drywell at the entrance of each vent pipe to prevent possible damage to them from jet forces which might accompany a pipe break in the drywell.

Access to the pressure absorption chamber from the Reactor Building is provided through two manholes with double gasketed bolted covers which can be tested for leakage.

Access to the drywell is provided through the equipment hatch and personnel air lock and through the double gasketed drywell head cover, all of which have provisions for being individually leak tested.

The pressure absorption chamber is supported on columns located on the outer and inner radii of the torus at the miters. At the center of each bay, a sliding saddle is provided to support the torus, resist upward forces caused by a LOCA, and allow for thermal expansion of the chamber.

The outer columns were pinned at the bottom and the inner columns are pinned at the top and bottom to allow radial growth of the absorption chamber due to temperature and pressure changes. Support for horizontal forces and lateral stability is provided by cross bracing between the outer support columns.

Additional details on the Containment System penetrations and on the equipment hatch and personnel air lock are presented in Subsection 3.8.2.4. The Appendices to Reference 1 provide details and dimensions of these penetrations and the personnel air lock. General arrangement drawings showing the relationship of the Containment System to the surrounding structures are presented as Drawings 3E-153-02-001 through 009. Overall dimensions and volumes of the Containment System are given in Table 3.8-1.

3.8.2.2 Applicable Codes, Standards and Specifications

The design, materials, fabrication, construction and inspection of the Containment System conform to, but are not necessarily limited to, the applicable sections of the following codes and specifications which are used to establish or implement design bases and methods, analytical techniques, material properties, construction techniques and quality control provisions.

Other tests and standards identified by the lead documents listed and in effect or promulgated at the time the design or construction was performed, shall also be considered as viable controlling documents.

The design and construction of the Containment System involved two basic stages:

- Original Construction (Basic Design)
- Subsequent Design Modification

Codes, standards and specifications are presented in the following paragraphs relative to these two stages.

Original Construction (Basic Design)

a. American Society of Mechanical Engineers
Boiler and Pressure Vessel Code, Sections VIII and IX, latest edition at the time of design, with all applicable addenda; nuclear case interpretation 1270 N-5, 1271 N, 1272 N-5 and other applicable case interpretations.

Boiler and Pressure Vessel Code, Section II, latest edition at the time of design with all applicable addenda, for the following material specifications:

SA-201 Carbon-Silicon Steel Plates of Intermediate Tensile Ranges for Fusion-Welded Boilers and Other Pressure Vessels

SA-212 High Tensile Strength Carbon-Silicon Steel Plates for Boilers and Other Pressure Vessels

SA-300 Steel Plates for Pressure Vessels for Service at Low Temperature

SA-333 Seamless and Welded Steel Pipe for Low Temperature Service

SA-350 Forged or Rolled Carbon and Alloy Steel Flanges, Forged Fittings, and Valves and Parts for Low Temperature Service

b. American Society for Testing and Materials Standards

A36 Structural Steel

A193 Specification for Alloy Steel and Stainless Steel Bolting Material for High Temperature Service

A307 Specification for Low Carbon Steel Externally and Internally Threaded Standard Fasteners

c. American Institute of Steel Construction

Specification for the design, fabrication and erection of structural steel for buildings.

d. Federal Specifications

TT-P-86c Paint; Red-Lead Base, Ready Mixed

e. Steel Structures Painting Council Specifications

SSPC-SP-3 Power Tool Cleaning

SSPC-SP-6 Commercial Blast Cleaning

f. State of New Jersey Laws, Rules and Regulations

g. Burns & Roe Specifications

S-2299-4 Design, Furnishing, Erection and Testing of the Reactor Drywell and Suppression Chamber Containment Vessels

Design Modification

Modifications subsequent to the basic Containment System design and construction have transpired over a number of years after being initiated in 1975. As such, numerous codes and code revisions have been utilized in carrying out the design and construction efforts.

The following codes, standards and specifications have been supplied to indicate the basic nature of the documents being employed. Specific information relative to actual governing documents used, must be obtained from the individual modification's "System Design Description" for the Oyster Creek plant.

a. American Society of Mechanical Engineers

ASME Boiler and Pressure Vessel Code, Section III, Subsection NE," Class MC Components," (1977 Edition through Summer 1977 Addenda).

ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 1, (1977 Edition through Summer 1978 Addenda).

ASME Boiler and Pressure Vessel Code, Section II, "Material Specifications," (1977 Edition through Summer 1978 Addenda).

ASME Boiler and Pressure Vessel Code, Section III, Subsection NF, "Component Supports," (1977 Edition through Summer 1977 Addenda).

b. American Concrete Institute

ACI 349-76, "Code Requirements for Nuclear Safety-Related Concrete Structures," (through 1979 Supplement).

3.8.2.3 Load and Loading Combinations

The Primary Containment is designed to withstand all credible conditions of loading, including preoperational test loads, normal loads, severe environmental loads, extreme environmental loads, and abnormal loads. These loads are considered in the applicable load combinations to assure that the response of the structure will remain within the design limits prescribed in Subsection 3.8.2.5.

The loads and load combinations provided below are extracted from Reference 1. Loads and load combinations relative to the modifications implemented after start of commercial operation are contained in References 2 through 11.

a. Design Loadings

The loadings considered in the design of the drywell, absorption chamber and interconnecting elements include:

- loads caused by temperature and internal or external pressure conditions.
- Gravity loads from the vessels, appurtenances and equipment supports.
- Horizontal and vertical seismic loads acting on the structures.
- Live loads.
- Vent thrusts.
- Jet forces on downcomer pipes.
- Water loadings under normal and flooding conditions.
- The weight of contained gas in the vessels.
- The effect of unrelieved deflection under temporary concrete loads during construction.
- Restraint due to compressible material.
- Wind loads on the structures during erection.

b. Description of Loads

1. Pressures and Temperatures Under Normal Operating Conditions

During reactor operation the vessels will be subjected to temperatures up to 150°F at close to atmospheric pressure. The absorption chamber will also be subject to the loads associated with the storage of up to 91,000 cubic feet of water distributed uniformly within the vessel.

2. Pressures and Temperatures Under Accident Conditions

The drywell and the vent system are designed for an internal pressure of 44 psig coincident with a temperature of 292°F and for an internal pressure of 35 psig coincident with a temperature of 281°F. The 35 psig and 281°F have been considered to prevail for a period of 4 to 5 days as a design condition. The absorption chamber is designed for an internal pressure of 35 psig coincident with the loads associated with the storage of absorption pool water increased in volume up to 91,000 cubic feet and a temperature of 150°F.

3. Jet Forces

The drywell shell and closure head are designed to withstand jet forces of the following magnitudes in the locations indicated from any direction within the drywell:

Location	Jet Force (Max.)	Interior Area Subjected to Jet Force
Spherical Partof Drywell	566,000 pounds	3.14 square feet
Cylinder and Sphere to Cylinder Transition	466,000 pounds	2.54 square feet
Closure Head	16,000 pounds	0.09 square feet

These jet forces consist of steam and/or water at 300°F maximum in the impingement area. The jet forces do not occur simultaneously. However, a jet force is considered to occur coincident with internal design pressure and a temperature of 150°F.

The spherical and cylindrical parts of the drywell are backed up by reinforced concrete with a layer of compressible material and an air gap between the outside of the drywell and the concrete to allow for thermal expansion. It is assumed that local yielding will take place, but it has been established that a rupture will not occur. This assumption is discussed more fully in Section III-2.4 of Reference 1.

Where the shell is not backed up by concrete (closure head), the primary stresses resulting from the combination of loads previously defined does not exceed 0.9 times the yield point of the material at temperature.

However, the primary plus the secondary stresses are limited to three times the allowable stress values given in Table UCS-23, Section VIII, ASME Boiler and Pressure Vessel Code. Supporting data is available in the report, "Loads on Spherical Shells", prepared by CB&I following a series of load tests on spherical plates. This report is included as an Appendix in Reference 1.

The absorption chamber and vent system are designed to withstand jet force reactions associated with the design basis LOCA. The design reaction on each 24 inch diameter downcomer pipe is 21,000 pounds. Stresses resulting from these reactions are limited to ASME Code allowables.

4. Gravity Loads to be Applied to the Drywell Vessel

- The weight of the steel shell, jet deflectors, vents and other appurtenances.
- Loads from structural members used to support equipment.
- An allowance for the weight of the compressible material applied to the exterior of the vessel and as described in the B&R, Inc. report "Expansion of the Drywell Containment Vessel", which is included as an Appendix in Reference 1.
- The live load on the access opening: 11 tons or 150 pounds per square foot, whichever is more severe.

- The live load for the depth of water on the water seal at the top flange of the drywell with the drywell hemispherical head removed.

- The weight of contained gas during the tests.

- Dead and live loads on the welding pads provided on the inside of the containment sphere shoulders, spaced at 8 foot centers in each direction. Permanent loads are 200 pounds on each pad, with 800 pounds of live load on any two adjacent pads.

- A temporary load due to the pressure of fluid concrete which was placed directly against the compressible material attached to the exterior of the drywell and vents. The fluid concrete pressure was controlled by limiting the rate of placement per hour in order to have a pressure limit of 3 psi on the compressible material.

5. Gravity Loads to be Applied to the Absorption Chamber

- The weight of the steel shell including catwalk, vent header, downcomer pipes and other shell appurtenances.

- The absorption pool water stored in the vessel as specified above.

- The weight of contained air during the tests.

6. Lateral Load

The drywell vessel which was exposed above grade, prior to construction of the Reactor Building, was designed to withstand wind loads on the projected area of the circular shape in accordance with the height zones listed below. These loads were analyzed in combination with other loads applicable during this stage, with stresses limited to 133 percent of the ASME allowable stresses.

Height Above Grade in Feet	Wind Load in Pounds per Sq. Foot
0 - 30	15
30 - 50	18
over 50	24

The effects of the lateral loads at the blanked off vessel penetrations were investigated.

A lateral static coefficient equal to 22 percent, and a vertical static coefficient equal to 10 percent, of the permanent gravity load was assumed as acting simultaneously with each other.

This load was taken concurrently with permanent gravity loads, accident pressure conditions and other lateral loads as shown in Figures 3.8-4 and 3.8-5. These values were based on studies and criteria described in Section 2.5.

The static coefficients listed were used by CB&I to develop the design of the drywell and absorption chamber. After completion of this design and fabrication of the vessels, John A. Blume & Associates were engaged by G.E.to perform a dynamic analysis of the structure under seismic conditions. The complete analysis performed by Blume has been included in Appendix III-2.4 (Item 3) in Reference 1. The results of these calculations list coefficients equal to those utilized by CB&I in their calculations, corroborating the adequacy of the seismic design performed by CB&I.

c. Loading Combinations Used in the Basic Design of the Drywell and Vent System

1. Case I - Initial Test Condition at Ambient Temperature at Time of Test

- Gravity load of vessel and appurtenances
- Design pressure
- The weight of contained air
- Lateral load due to wind or seismic forces whichever is more severe
- Vent thrusts
- Vertical seismic load

2. Case II - Final Test Condition at Ambient Temperature at Time of Test

- Gravity load of vessel and appurtenances
- Gravity load from equipment supports
- Gravity load of compressible material
- Gravity load of welding pads
- Design pressure
- Seismic loads
- Effect of unrelieved deflection under temporary concrete load
- Restraint due to compressible material
- Vent Thrusts

3. Case III - Normal Operating Condition at Operating Temperature Range of 50°F to 150°F

- Gravity load of vessel and appurtenances
- Gravity load from equipment supports
- Gravity load of compressible material
- Seismic loads
- Vent thrusts
- Restraint due to compressible material
- Gravity load on welding pads
- Effect of unrelieved deflection under temporary concrete load
- External pressure of 2 psig
- Live load on personnel air lock

4. Case IV - Refueling Condition with Drywell Hemispherical Head Removed, at Operating Temperature Range of 50°F to 150°F

   - Gravity load of vessel and appurtenances
   - Gravity load from equipment supports
   - Gravity load of compressible material
   - Gravity and live load on welding pads
   - Water load on water seal at top flange of drywell
   - Seismic loads
   - Effect of unrelieved deflection under temporary concrete load
   - Restraint due to compressible material
   - Vent thrusts
   - External pressure of 2 psig
   - Live load on access opening

5. Case V-Accident Condition at Temperature Listed Below

   - Gravity load of vessel and appurtenances
   - Gravity load from equipment supports
   - Gravity load of compressible material
   - Gravity load on welding pads
   - Seismic loads
   - Design Pressure: Maximum positive pressure of 44 psig at 292°F decaying to 35 psig at maximum temperature at 281°F, to maximum negative pressure of 2 psig at 205°F.
   - Effect of unrelieved defl8ection under temporary concrete load
   - Restraint due to compressible material
   - Vent thrusts
   - Jet forces

d. Load Combinations Used in the Basic Design of the Absorption Chamber

1. Case I - Initial and Final Test Condition at Ambient Temperature at Time of Test

    - Gravity load of vessel and appurtenances
    - Absorption pool at the operating maximum of 91,000 cubic feet of water
    - Seismic loads
    - Design pressure
    - Vent thrusts

2. Case II - Temporary Condition at Ambient Temperature During Construction

    - Gravity load of vessel and appurtenances
    - Seismic loads

3. Case III - Normal Operating Condition at Operating Temperature Range of 50°F to 150°F

    - Gravity load of vessel and appurtenances
    - Absorption pool at the operating minimum of 82,000 cubic feet of water
    - Seismic loads
    - Vent thrusts

4. Case IV - Accident Condition at 150°F Maximum

    - Gravity load of vessel and appurtenances
    - Absorption pool at the operating maximum of 91,000 cubic feet of water
    - Seismic loads
    - Design pressure of 35 psig
    - Vent thrusts

- Jet forces on downcomer pipes

3.8.2.4 Design and Analysis Procedures

The design and analysis procedures described herein are those presented in Reference 1. Subsequent to the initial design, certain modifications were made to Primary Containment penetrations. The original design, modifications and analyses related to them are discussed in Subsection 3.8.2.4.3.

3.8.2.4.1 Drywell

Primary Membrane Stresses

The membrane stresses are based on the assumption that the thin shell resists the imposed loads by direct stress only. In addition, for earthquake design, it has been assumed that the shell as a free standing circular cantilever beam of variable cross section. Stresses have been computed at various points along the vertical axis of the drywell as shown on Figures 3.8-6 and 3.8-7. The notations adopted in these calculations are defined as follows:

T_1 = Latitudinal force in pounds per inch of meridional arc length

T_2 = Meridional force in pounds per inch of arc length

S_1, S_2 = Unit stresses corresponding to T_1 and T_2 and are equal to T_1 or T_2 divided by t

W = Total gravity load above the plane, in pounds

P = Internal or external pressure in lbs/in^2.

R = Radius of the cylinder or sphere as applicable, in inches

t = Plate thickness in inches

q = Vertical angle between vertical axis and point in the shell being computed

The internal force per unit width is computed from the following relationships:

Cylindrical Portion of Drywell:

$T_1 = PR$ and $T_2 = PR/2$ for internal or external pressure

$T_2 = W/2\ \ p\ R$ for gravity loads

$T_2 = -T_1 = Meq/S$ for earthquake loads

$T_2 = Mw/S$ for wind loads

where Meq and Mw are the moments due to earthquake and wind, respectively, and S is the Section Modulus of the Section.

Spherical Portion of the Drywell:

$T_1 = T_2 = PR/2$ for internal or external pressure

$T_2 = -W/2\ \ p\ R\ \sin^2 q$; $T_1 = -PR\cos q\ \ \ -T_2$ for gravity loads

$T_2 = -T_1 = Meq/\ p\ R^2(\sin^3 q)$ for earthquake load

$T_2 = T_1 = Mw/\ p\ R^2(\sin^3 q)$ for wind load

Load Deflection Tests

Design pressure for the drywell requires a relatively thin walled steel vessel. However, the vessel has relatively little capability to resist concentrated jet forces. Such loads are, however, readily accepted by the massive concrete shield which surrounds the vessel. Accordingly, the space between the steel drywell vessel and the concrete shield outside has to be sufficiently small so that, although local yielding of the steel vessel can occur under concentrated forces, yielding to the extent causing rupture will be prevented. Space has been provided to allow the drywell to expand when in its stressed condition in order for it to function as a pressure vessel. In addition, the vessel is subject to thermal expansion due to exposure to operating and possible accident temperatures which are significantly higher than ambient.

In order to investigate whether or not a steel shell could deflect up to three inches locally without failure as a result of a concentrated load, CB & I conducted a series of tests on a steel plate formed to simulate a portion of the drywell vessel. The tests also provided data on loading required to produce a given deflection, and the strain at various points of the shell. In performing these tests, it was assumed that permanent deformation is not considered as failure.

The basic test section was designed and fabricated to simulate a 70 foot diameter sphere. The material and plate thickness used were typical of the type used in pressure suppression containment system applications. By modifying the basic section through the addition of an 18 inch diameter fitting with insert type reinforcing, a typical penetration was simulated. Again by the removal of the insert type fitting and the insertion of an 18 inch diameter fitting with pad type reinforcing, another typical penetration was simulated.

Step by step procedures, description of the tests, as well as load deflection and load strain curves are included in the CB & I report "Loads on Spherical Shells" in Appendix III-2.4 (Item 2) of Reference 1. The results of these tests indicate that spherical steel shells of this diameter and thickness, as well as fittings with insert type reinforcing located in a spherical steel shell are capable, under concentrated loading, of withstanding a substantial localized deflection without failure. Graphs of the theoretical radial strain in the shell, calculated assuming the shell to be a membrane, are included in this report. They indicate that the experimental data conforms rather well to the theoretical values. This confirms that the shell was acting in close conformity to the approximate theoretical mode.

Expansion of the Drywell Containment Vessel

The load deflection tests performed by CB&I on steel plates provided the basis for selecting three inches as the maximum acceptable space between the cold drywell shell and the biological concrete shield which surrounds it.

The three inch space precludes the use of a conventional forming system for the inner face of the concrete shield.

The approach taken was to fill the space permanently with a material having sufficient compressibility to permit the expected vessel movement and yet be rigid enough so as not to deform under the fluid pressure of concrete. This pressure can be controlled by limiting the rate of placement of the concrete.

To eliminate the need for a continuous internal pressure in order to prevent compressive forces on the vessel, an inelastic compressible material was selected; such a material can be permanently compressed once by simulating the conditions causing the greatest vessel expansion. The residual air gap created by the inelastic compression of the material will then offer no resistance to subsequent repetitions of vessel expansion.

After careful consideration, testing, and investigations as to the type of material to be utilized, an asbestos fiber magnesite cement product was selected. To determine the required minimum thickness of the material, it was necessary to establish the extent to which it was compressed. This was determined by the expansion of the vessel associated with its highest postulated temperature for any future operating or accident condition, and by the procedure planned for expanding the vessel to create an air gap larger than required to accommodate any future conditions.

Information and discussions pertaining to the performance, design and analysis aspects of the inelastic compressible material is given in Subsection 3.8.2.4.3.

An internal pressure of 35 psig (saturated steam pressure at a temperature of 281°F) resulted in an expansion which exceeded postulated accident or operating expansion, and hence, was a criterion for determining spacing dimensions.

At the most critical location, the point on the sphere most distant from the bottom embedment, thermal expansion was expected to be about 1.06 inches. Tests on the spacing material to measure the pressure required to reduce its thickness by this amount, and also taking into account the compression resulting from the fluid concrete pressure before setting, indicated an initial thickness requirement of about 2 1/2 inches. The design pressure transmitted to the concrete shield wall by the spacing material during initial expansion of the vessel would be 20 psi, which is tolerable from the standpoint of the concrete strength. Some tolerance on thickness of the compressible material had to be allowed. A workable limit of ±1/4 inch was chosen. Since the design pressure on the wall assumed 2 1/2 inches minimum, thickness of 2 3/4 inches ±1/4 inch was specified.

In considering the acceptability of the three inch gap as a maximum between the steel vessel and the concrete shield, it should be noted that this distance would be reduced by: the compression of the material under the fluid concrete pressure; the thermal expansion of the vessel in going from ambient temperature during construction to an operating temperature at which the design

accident might occur; and the fully compressed thickness of the material. These conditions were expected to reduce the three inches to well below the 3.125 inch minimum failure deflection of the CB&I jet load simulation tests, particularly in view of the conservative approach used in those tests. It was thus concluded that a gap of three inches between the drywell vessel and the biological concrete shield would be satisfactory.

The construction schedule required that the compressible material be applied to the exterior of the vessel prior to the construction of the concrete shield wall.

The mixing and foam injection, as well as the application procedure for the compressible material to the vessel was performed in accordance with that developed by the manufacturer, All Purpose Fireproofing Corp. The material was built up in three coats to make a total thickness of 2 3/4 inches ±1/4 inch for the upper hemisphere. Since the lower hemisphere of the cylindrical section will have less total expansion, 2 1/2 inches ±1/4 inch of the compressible material was applied over their surfaces. A polyethylene sheet reinforced with glass fibers was used to prevent bonding of the spacing material and the concrete. The actual application was completed in about two weeks.

After completion of the material application, any damages noted were repaired. Testing and inspection services were provided to assure that the quality and workmanship were as required.

After the biological concrete shield wall was poured against the compressible material and cured, the vessel was prepared for the expansion operation. Expansion of the vessel was accomplished by pressurizing with heated air by means of portable compressors, electric duct heaters and fans placed at various locations within the vessel.

A temperature recorder was used to monitor temperature. Several of the existing vessel penetrations, consisting of pipes welded into the vessel and extending out through the concrete shield wall through sleeves, were used to monitor vessel expansion.

The expansion operation was conducted as planned, and pressure, temperature and expansion recorded throughout the procedure. The concrete shield wall exterior was examined periodically and particularly at maximum temperature and pressure; no evidence of distress was observed. An inspection of the interior of the drywell immediately after the expansion operation and again some 12 hours later gave no evidence of distress. The maximum displacement recorded during expansion was 0.61 inches which was less than the time temperature performance value calculated by computer program method. This measurement together with the favorable results of the examination of the shield wall and drywell vessel interior corroborated the assumptions made in the drywell design. Complete step by step procedures, initial criteria and conclusions drawn from this expansion procedure are included in the B&R, Inc. report "Expansion of the Drywell Containment Vessel" in Appendix III-2.4 (Item 1) of Reference 1. See also Subsection 3.8.2.4.3.

Maximum Primary Membrane Stresses in the Shell

The maximum primary membrane stresses in the shell result from the following combination of loads.

Internal pressure of 44 psig, dead load of the shell and appurtenances lateral and vertical seismic loads, gravity load on welding pads and gravity load of the compressible material. The internal pressure load causes by far the greatest stress.

The maximum stress is 19,200 psi which is less than that allowed by the code. It occurs in the cylindrical portion of the drywell. Other stresses computed at other points along the drywell are lower in magnitude.

In addition to maximum stresses computed for the cylindrical and spherical portions of the drywell, stresses have been computed on the elliptical head of the vessel taking into account the effect of jet forces since this portion of the vessel is not backed up by concrete. The maximum stress on the head and been found to be 29,340 psi and it results from jet forces combined with an internal pressure of 44 psig. The design specification allowance for this loading combination is 31,500 psi.

Since the personnel and equipment hatch had no concrete backing to take the effect of jet forces, this portion of the drywell as well as its components was investigated and designed for jet forces in conjunction with the other load combinations as set forth in Figures 3.8-4 and 3.8-5. The effect of eccentricities on possible jet forces was also analyzed and the design provided reinforcements and stiffeners as required to maintain stresses within specified limits.

In conclusion, the design of the personnel and equipment hatch is adequate, and provides a safe and well engineered structure.

<u>Flooded Condition</u>

The drywell vessel has been analyzed for its ability to withstand loading resulting from partial flooding and for maximum flooding to El. 74'-6" (see Figure 3.8-8).

In each case, the maximum stress computed for various locations on the shell are below the ASME Code allowables. In addition, critical buckling of the vessel under flooded conditions has been analyzed. The results of this analysis show that there is ample margin of safety under either flooding condition.

<u>Buckling Considerations</u>

The drywell shell must be capable of resisting the compressive stresses resulting from the external pressure, the dead load of the shell and appurtenances, the dead load of the compressible material, the live load on the access and beam loads, the gravity loads on the weld pads, plus the wind or seismic loads. These loads produce uniaxial compressive stresses of varying magnitude at different points along the drywell shell.

Section VIII of the ASME B&PV Code (1950), permits an allowable compressive stress of 1,800,000 (t^2/R) for uniaxial compression. Later editions of the code do not include this equation as such, but include tables for allowable external pressures which are based on this allowable.

The state of stress at any point in the spherical shell may be expressed as a biaxial compressive stress plus a uniaxial compression. By combining the T_1 and T_2 stresses acting at the point algebraically, the allowable compressive stress is then given by the relationship:

$$(T_2 - T_1)/1.8 \times 10^6(t^2/R) + T_1/9 \times 10^5(t^2/R)\text{£}1,$$

where T_1 and T_2 are compressive stresses. This relationship applies to buckling of the spherical shell under biaxial compression. Also:

$$T_2/1.8 \times 10^6(t^2/R)\text{£}1,$$

which is the axial buckling of the cylindrical shell.

The stress values at the different points along the shell are summarized in Table 3.8-2 and are below the ASME allowables.

Summary

Since all possible loads, as well as their combinations, have been taken into consideration, and the maximum stresses computed are all within the design specifications and ASME Boiler and Pressure Vessel Code allowables, the drywell design is adequate.

3.8.2.4.2 Torus

Following the original design of the facility, additional design, analysis and modification work was performed for the torus under the Mark I Containment Systems Evaluation Program. These efforts are described in detail in References 2 through 11. The general analytical procedures and computer techniques utilized in the design modifications of the suppression chamber are provided in Reference 10. The discussion that follows was extracted from Reference 1, the Primary Containment Design Report.

Primary Membrane Stresses

The absorption chamber is supported on twenty pairs of columns located on the inner and outer peripheries and equally spaced. An internal ring girder of variable cross section has been provided at each of the supporting points to reduce local stresses and to add stiffness to the section. Although the principal stresses computed on the absorption chamber were circumferential, detailed analyses have been performed to determine the magnitude of localized stresses at the points of column and downcomer supports, vents, etc., to determine the need for and provide additional stiffeners and reinforcing as required.

The notation adopted in the calculations for the absorption chamber is similar to that used for the drywell. The internal force per unit width was computed from the following relationship:

$T = PR$ for Internal or External Pressure (see Subsection 3.8.2.4.1 for nomenclature).

Due to the complexity of the analysis involved in the determination of maximum stresses under various loads and load combinations, CB&I set up a computer program for each of the major loading combinations. These combinations were the initial and final condition at ambient temperature at time of the acceptance test, and the accident condition at 150°F maximum. In addition, a flooded condition was analyzed.

The CB & I calculations for the absorption chamber, including the printout sheets for the computer program, are on file at General Electric Co., San Jose, California.

Accident Condition at 150°F Maximum

The maximum primary membrane stresses in the shell and ring girder result from a combination of loads as follows: downcomer thrusts of 21,000 lbs each, internal pressure of 35 psi at 150°F or external pressure of 1 psi, dead load of shell and appurtenances, absorption pool of 91,000 cubic feet of water, lateral and vertical seismic loads and vent thrusts of 44 psig at 292°F or 35 psi at 281°F. The equations used in the computer program and the nomenclature are given in Figure 3.8-9, influence coefficients are given in Figure 3.8-10, and resulting moments as well as horizontal and vertical loads are given in Figure 3.8-11.

Stresses have been computed at critical points along the girder. The maximum stresses are 11,430 psi on the outside and -5670 psi on the inside which are well below the ASME Code allowables.

Flooded Condition

An analysis was performed on the absorption chamber to determine the magnitude of the stresses that could develop under a flooded condition. The water level assumed at flooding was found adequate to cover reactor fuel as shown in Figure 3.8-8.

The loads on the absorption chamber are shown on Figure 3.8-12, the equations for the computer solution are shown in Figure 3.8-13 and the moments and horizontal and vertical loads in Figure 3.8-14. The maximum stresses computed were 19,250 psi outside and 18,000 psi inside, which are within the Code allowables.

Buckling Considerations

The absorption chamber shell as well as its elements have been thoroughly analyzed to determine their resistance to buckling. Stiffener stresses at the point of column support are in the order of 7,400 psi versus 8,900 psi allowable (in accordance with the NASA technical note D-163 of September 1959). The absorption chamber shell was also analyzed for shear buckling under a flooded condition. The analysis of a typical curved panel shows a maximum shear stress of 1,800 psi against an allowable of 6,560 psi using Roark's Case L, the factor of safety therefore being 3.65.

Supports

The absorption chamber support columns have been analyzed for each loading condition listed under Subsection 3.8.2.3 and under flooding as described in Subsection 3.8.2.4. Horizontal seismic forces and the methods of transmittal of forces have been investigated. The structural steel sections used provide a safe design that conforms to the requirements of the ASME Code and the AISC.

Header, Downcomer and Vent Pipes

These components of the absorption chamber have also been analyzed and adequately sized for plate thickness and reinforcements, as required, and in conformance to the ASME Code.

Summary

The analyses and investigations performed in connection with the design of the absorption chamber and all of its components clearly indicate that the design is adequate. Working stresses have been

kept below the ASME Code and design specification allowables and the design assumptions are in conformance with sound engineering practices.

3.8.2.4.3 Primary Containment Penetration Design

The Primary Containment is penetrated at several locations by piping, instrument lines, ventilation ducts, and electrical leads. These lines must be attached to the Primary Containment during both normal operating and abnormal conditions. In order to minimize post accident containment leakage, the containment penetrations are designed to withstand the normal environmental conditions during plant operation, and to retain their integrity during and following postulated accidents. Penetrations are also provided for personnel, equipment access, and for refueling. The criteria for the system piping and drywell penetrations includes expansion joints, guides, piping anchors and leak testing.

Burns & Roe reviewed and approved all nozzle and penetration designs prior to release for fabrication. This included materials, specifications, fabrication procedures, and quality control requirements as well as design calculations. The designs fulfill the requirements of the ASME Boiler and Pressure Vessel Code, Section III for Class B vessels for all modes of plant operation, including any design accident condition.

Pipe penetrations which contain bellows type expansion joints and all gasketed type penetrations are designed to make it possible to perform leak tests without pressurizing the entire containment system. Leak testing to the design pressure of the drywell may be accomplished by pressurizing the penetration between the double seals utilizing the pressure tap.

Type 1 Pipe Penetrations

Type 1 pipe penetrations are those which must accommodate thermal movement and experience relatively high thermal stress (see Figure 3.8-15).

These are high temperature lines such as the steam and feedwater system lines. These pipes are capable of exerting a reaction force due to line thermal expansion or containment movement which cannot be restrained by the containment shell and are therefore provided with a bellows expansion seal. Where necessary, these lines are anchored outside the containment to limit the movement of the line relative to the containment. This design assures integrity of the penetration during plant operation.

The penetration nozzle passes through the concrete and is welded to the primary containment vessel. The process line which passes through the nozzle is free to move axially, with the bellows expansion joint accommodating the movement. A guard pipe immediately surrounds the process line and is designed to protect the bellows and maintain the penetration seal, should the process line fail within the penetration. A seal arrangement is also provided which permits periodic leakage testing of these penetrations.

The design of the penetration takes into account the simultaneous stresses associated with normal thermal expansion, internal pressure, dead loads, seismic loads, and loads associated with a Loss-of-Coolant Accident within the drywell. For these conditions the resultant stresses in the pipe and penetration components do not exceed the Code allowable design stress. In addition, for random failures of the process lines, the design takes into account the jet force loadings resulting from the failure. The design criteria and evaluation methodology for pipe penetrations are as stipulated in FDSAR Amendments 11, 50 and 51.

Type 2 Pipe Penetrations

Type 2 pipe penetrations are those in which the piping or ventilation duct penetrations are welded directly to the nozzles (see Figure 3.8-16).

Bellows and guard pipes are not necessary in this design, since the thermal stresses are small and are accounted for in the design limits. Typical of this type of penetration are the shutdown cooling piping and standby liquid control piping.

Type 2A Pipe Penetrations

In 13R, the isolation condenser penetrations were replaced along with their process pipelines. Type 2A penetrations (see Figure 3.8-16A) are designed with no reactor coolant pressure retaining welds within the guard pipe. A flued collar is welded to the OD of a process pipe fabricated from IGSCC resistant stainless steel (316 NG).

Type 2B Pipe Penetrations

In 13R, the reactor water cleanup penetrations were replaced along with their process pipelines. By doing this, welds inside penetrations were eliminated. Also, the modification eliminates the integral fluid collar/ process pipeline welds. The new design attaches the flued collar to the outside diameter of the process pipeline (see Figure for 3.8-16B).

Type 3 Pipe Penetrations

Type 3 pipe penetrations are those pipe lines which penetrate the containment, where the reactive forces can be restrained directly by the containment shell and are provided with full strength attachment welds between the pipe and the containment shell (see Figure 3.8-17). These penetrations are designed for long term integrity without the use of a bellows seal. Typical of this type of penetration are the containment spray supply piping and the core spray supply piping.

Electrical Penetrations

The electrical penetrations include electrical power, signal, and instrument leads (see Figures 3.8-18 and 3.8-19). Although two types of electrical penetrations are shown, they are basically of the same design. Electrical penetrations require a special design to achieve minimum leakage because of the problem imposed by creepage of electrical insulation. The penetration nozzle is welded to the Primary Containment vessel, and the ends are field welded in place during installation. A bonding resin is utilized in the seals where the cable emerges. This arrangement provides a leak tight configuration which is leak tested after installation, and provides a means for periodic leak testing thereafter.

The penetration seals are designed and constructed to minimize leakage to acceptable limits under conditions of pressure and temperature within the drywell as specified below:

	Normal Operating	Maximum
Pressure	0 - 2 psig	62.5 psig
Temperature	150°F	310°F
Relative Humidity	20 to 100%	100%

Personnel Access Lock and Equipment Hatch

The personnel lock and equipment hatch details are combined to form one integral unit. The personnel lock is approximately 8 ft in diameter and serves as part of the cover for the 10 ft diameter equipment hatch. Details and general arrangement of the personnel lock and equipment hatch are shown on Figure 3.8-20.

The equipment hatch closure is effected by bolting the cover to the equipment hatch barrel. The bolting detail allows the bolts to be rotated free of the cover upon loosening the nuts. The bolts also supply the seating force necessary to seal the double tongue and groove gasket. The annular space between the two gaskets provides a means of testing the closure without pressurizing the entire vessel. This is accomplished by pressurizing the annular space and observing for leaks. A pressurizing port is furnished for this purpose.

A floor that can be removed in small lightweight sections has been provided in the equipment barrel to allow each section to be handled manually.

The personnel lock and hatch cover are supported by an integral wheel carriage. The wheel carriage is mounted on two parallel I-beam rails that allow the lock assembly to be rolled out of the equipment hatch barrel. Once out of the equipment barrel, the entire lock and carriage can be removed from the area by lifting the weight of the assembly off its wheels and pulling the inner pins of the wheel carriage. This allows the wheels to swing away and frees the unit from the rails.

Operation of the personnel lock is completely mechanical. Each of the two doors can be operated from inside the containment vessel, inside the lock, or outside the containment vessel. Approximately five and one half revolutions of a hand wheel will lock one door, open an equalizing valve at an opposite door, unlock that door and swing it completely open. About 20 pounds of handle effort is required to actuate the hand wheel. Each door is normally pressure seated and thus opens inward towards the center of the containment vessel.

The 2 ft 6 in x 6 ft rectangular doors are mechanically interlocked so that one door can be opened only when the opposite door is completely locked and the exhaust valve closed. This safety feature can be voided by removing a padlock and retaining pin. This in turn allows for straight passage through the air lock.

An indicator is located at each hand wheel of each door both inside and outside of the lock. This indicator reveals the position of the door. Thus, an operator knows, at a glance, the position of both door interlocks, the exhaust valves and the door latching mechanisms. Limit switches provide the capability for remote indication of both door positions.

The handwheels for the two doors of the lock are mounted on three separate gear boxes, one inside the lock and one on the outside of each door bulkhead. All actuating mechanisms are completely contained in the gear boxes.

The door hinge detail provides for adjustment in all directions. The detail also allows the door to bear uniformly against the gasket which is a high temperature elastometer material. A machined tongue on the door is forced into the gasket by the latching mechanism on the end of the hinge arm. The latching mechanism is adjustable to provide for increased permanent set in the gasket.

Figures 3.8-21, 3.8-22 and 3.8-23 are representative of the shop details and fabrication procedures used in this type of penetration. Additional fabrication details can be found in Appendix VI-3.0 of Reference 1.

General Design and Analysis Information

All Primary Containment penetrations were designed to the following criteria:

a. Accident design condition - For this condition the particular penetration being analyzed shall take into account all normal thermal expansion, live and dead, and seismic loads, in addition to accommodating the incident thermal growth and pressure conditions resulting from a loss of coolant incident within the drywell. For these conditions, the resultant stresses in the drywell penetration shall not exceed the Code allowable design stress of 1.5 Sm equal to 28,875 psi.

b. Pipe rupture design condition - For this condition the particular penetration being analyzed, the pipe connected thereto assumes to rupture, at random, on either side of the drywell. The analysis shall take into account the loadings given in a. above, in addition to the jet force loadings resulting from the rupture.

c. Maximum earthquake design condition - This design condition is the same as in b. above, with the following exceptions:

    1. Seismic forces for this analysis shall be two times the forces used to analyze conditions a. and b.

    2. For this condition, stresses may exceed the yield point of the material; however, it shall be established that a failure of the Containment does not occur.

d. The design and location of piping anchors, position stops and guides shall be such as to satisfy all conditions listed above, and thereby preclude any cause for failure of the drywell penetration and one of its associated isolation valve(s).

e. If any of the drywell piping penetrations are located or arranged such as to require a bellows expansion joint between the drywell and the process line to accommodate pipe and equipment motions, then a "guard pipe" shall be located concentrically between the process pipe and the drywell penetration with its bellows expansion joint. the "guard pipe" shall be designed for the same process conditions as the process line it encloses.

Calculations of the stresses in the containment vessel at the penetrations are in accordance with the methods outlined in Bulletin 107, Welding Research Council, August 1965.

The containment vessels, including penetrations, have been designed in accordance with the ASME Boiler and Pressure Vessel Code and Code Case 1272N, except for allowable stresses, as noted above. Even though Code jurisdiction terminates at the first circumferential joint, the design, fabrication and inspections of the penetration assemblies welded to the containment vessel nozzles have been made to conform to the requirements of the Code. Therefore, no other inspection techniques, other than leak check and those specified in the Code were required.

Following the initial design, a reanalysis of Primary Containment penetrations was performed directed towards the possibility of wrinkling or buckling of the metal liner of the drywell because of excessive interface or external pressure as the liner expands following a Loss-of-Coolant Accident.

A requirement was established for an air gap and gap material (Firebar D) placed between the drywell shell and the concrete which is to provide a relatively uniform annular gap which is available to accommodate the differential thermal expansion between the shell and the concrete during any mode of plant operation. The gap is sufficiently large that the interface pressure created by expansion of the shell during any postulated operating condition (normal or accident) does not exceed the allowable external (interface) pressure on the shell.

To determine the maximum acceptable distance between the steel vessel and the concrete wall, load deflection tests were made by the Chicago Bridge and Iron Co. These tests, described in the CB&I report "Loads on Spherical Shells", August 1964, were to determine the maximum deflection consistent with no rupture but accepting yielding of prototype sections of the vessel under a load simulating the postulated jet force accompanying the Loss-of-Coolant Accident.

Two of the three test runs were terminated without rupture (due to limitations of the testing apparatus) at deflections of 3.3 and 3.25 inches of deflection. The third was terminated with the development of a crack at a deflection of 3.125 inches. Conservatisms in the test included: distribution of the load over only 1.08 square feet of plate whereas the pipe, the failure of which is the postulated source of the jet, has an area of 2.54 square feet in the cylinder zone of the drywell and 3.14 square feet in the sphere; use of flat insert plates in simulating a penetration for the one test where a crack developed rather than plates dished in the direction of load to conform to the curvature of the main shell plate; and application of load as a static load at normal ambient temperature rather than as an impact load with steel at the higher operating temperature of the vessel.

In order to determine required minimum thickness of the gap material it was necessary to establish the extent to which it would be compressed. This amount would be determined by the expansion of the vessel associated with its highest postulated temperature for any future operating or accident condition not concurrent with high internal pressure or by the procedure planned for expanding the vessel to create the air gap if this procedure would expand the vessel to a greater extent than the future conditions.

At the most critical location, the point of the sphere most distant from the embedment, thermal expansion at 281°F was expected to be 1.057 inches, as discussed later in this subsection. Tests on the compressible material reducing its thickness this much, plus that due to compression from the fluid concrete pressure, from an initial thickness of about 2 1/2 inches set the design pressure which would be transmitted to the concrete wall during vessel initial expansion at 20 psi. Some tolerance on thickness of compressible material had to be allowed; a feasible limit was ±1/4 inch. Since the design pressure on the wall assumed 2 1/2 inches minimum, a thickness of 2 3/4 inches ±1/4 was indicated.

In considering the acceptability of the three inch upper limit of tolerance in relation to the maximum acceptable distance between steel and concrete, it was noted that the initial three inches would be reduced by: the compression of the material under the fluid concrete pressure, the thermal expansion of the vessel in going from ambient temperature during construction to an operating temperature at which the Loss-of-Coolant Accident could occur, and the fully compressed thickness of the material.

The conditions were expected to reduce the 3.0 inches to an amount adequately below the 3.125 minimum failure deflection of the CB&I jet load simulation tests, particularly in view of the many conservatisms in those tests.

It was subsequently determined by tests and measurements during the expansion operation that these conditions would reduce a 3.0 inch gap as follows:

Compression under fluid concrete pressure:	0.10 inch
Vessel thermal expansion @ 100°F:	0.15 inch
Residual thickness of lining:	0.25 inch
	TOTAL 0.50 inch
Maximum Reduced Thickness:	2.50 inch

The United States Testing Company, having previous experience with the material and having an operation at the jobsite (for concrete testing) was engaged to provide inspection services.

The development tests had demonstrated that the desired compression characteristics were achieved by the formulation producing a material having a dry density of eight pounds per cubic foot and that this dry density corresponded to a wet density at the spray nozzle of 29 pounds per cubic foot. Hence the principal inspection effort was directed toward monitoring wet density and of course, the gaging of thickness. Density was checked approximately hourly; at the infrequent deviations from the acceptable range of 27 to 31 1/2 pounds per cubic foot, work was stopped and the variables affecting density, the foam and/or water content, were adjusted until satisfactory density was demonstrated by another check.

Thickness was gauged during the spraying be observing the projection of the uniform length insulation pins stuck on the vessel at four foot intervals and was checked by probe by United States Testing. Deficient or excess thicknesses were corrected as the work progressed. Thickness was subsequently checked by Burns and Roe before release to the general contractor for each concrete placement against the material; no out of tolerance thickness was found in these subsequent inspections. Initially no record of actual thickness was made other than that it conformed to the specified thickness within the tolerance. It was later decided, after about a third of the concrete wall height had been placed, that a knowledge of actual material thickness might serve some future purpose; hence a record of measured thickness at about 3 foot intervals was kept but only at and above elevation 47 feet.

There was little question that the material, properly formulated and applied, would have sufficient strength to resist the load to be applied during the placing of wet concrete against it without excessive deformation since the development testing had shown that resistance to compression associated with vessel expansion in the order of twice the ideal would have to be accepted. However, with the concrete wall designed for a 20 psi load expected during the operation of compressing the material, some check was indicated to insure against greater resistance to compression.

Twenty one samples of the material were obtained as the work progressed by spraying into 18 inch square plywood boxes located on the work platform at the point of application, to the depth in the box equal to the thickness being applied. The samples were taken at intervals such that each would represent the material applied to a specific area of the vessel, and were labeled with elevation and azimuth to identify that area.

Since the concern was to determine stress strain characteristics at maximum strength, the compression testing of these production samples was deferred until they were thoroughly cured and dried. Tests were performed in August 1966 by United States Testing using the same procedures and equipment previously employed in testing of the development samples.

The variation in strength between different samples was wider than expected considering that the previous development samples had been made with the same equipment; resistance of 9 of the 21 samples at 1 inch of compression exceeded the 20 psi maximum observed in the development samples and used as a design load for the concrete wall.

A question of whether the samples were representative of the work in place was largely dispelled by subsequent tests of two samples cut off the wall of the vessel. These two tests also demonstrated the variation shown by the production samples and the more resistant of these two also exceeded the maximum strength of the development samples.

The production samples, their resistance to compression having been established by the tests, provided a means of calibrating a hand penetrometer which was used to spot check the initially applied material and to check the strength of replacement material applied to areas where the original lining was damaged and removed.

During the first stage of a Loss-of-Coolant Accident, the ambient temperature and pressure in the drywell both rise very rapidly. Later during the accident as the containment spray is initiated, the pressure begins to decrease rather sharply while the temperature decreases in a somewhat slower fashion (see Figure 3.8-24A). As the drywell temperature rises, heat is transferred to the metal containment liner which causes it to expand radially relative to the cold concrete and crushable material which surround the liner. As the differential thermal expansion continues, a thermally induced interface pressure is produced between the liner and the compressible material which, it should be noted, is thermal in nature and therefore self relieving.

The compressible material, a porous, asbestos compound known commercially as Firebar D, was used to cover the steel containment liner prior to pouring the concrete outer shell. After the concrete was poured, the compressible material provided an annular space between the liner and the concrete to allow for differential thermal expansion. The physical characteristics of the Firebar D are such that upon compression of the material, 80 percent of the deformation is permanent while 20 percent is elastic (i.e., 20 percent rebound). A series of tests conducted by the United States Testing Company verified that these characteristics were prevalent at both ambient temperature (approximately 70°F) and at a temperature of 300°F. In order to determine the pressure versus displacement compression characteristics of the material, 21 samples were obtained as work progressed on the containment by spraying the Firebar D into 18 inch plywood boxes located on the work platform at the point of application to the vessel to a depth in the box equal to the thickness being applied (approximately 2 3/4 in. ± 1/4 in.). In addition, two samples of Firebar D were taken from the wall of the containment in the as applied condition and were subjected to similar testing. The curves shown in Figure 3.8-24B give the pressure versus displacement characteristics of the two off wall samples. There was considerable scatter in the data points obtained for the other samples; however, the average of all the data fell below the upper curve on Figure 3.8-24B. Therefore, for conservatism, all calculations were based on the upper curve which gives a higher than average value of interface pressure for a given deflection. Furthermore, it was assumed that as the load is removed from the Firebar D, the pressure deflection curve is linear from the point representing the maximum pressure (and deflection) achieved to zero pressure at a displacement equal to 80 percent of maximum. This accounts for

the 20 percent rebound effect. Upon reloading and further compression, the Firebar D is assumed to exhibit pressure deflection characteristics such that the elastic unloading curve describes the pressure on reloading until the former maximum value of displacement is once again achieved. Then the nonlinear curve is followed until a new value of maximum pressure is reached. Similarly, this pressure, on unloading, varies linearly to zero at a value of displacement equal to 80 percent of the new maximum displacement achieved. This assumption is illustrated on Figure 3.8-24B where the deflection goes from A to B on initial loading, B to C on unloading, C to B on reloading, B to D on further loading and D to E on unloading.

An air gap was provided between the metal liner and the concrete to permit relatively unrestrained differential thermal expansion between the metal and concrete. This gap was formed by intentionally compressing the Firebar D between the metal and concrete by heating (to 180.5°F) and pressurizing (to 42 psig) the liner itself.

Since the containment shell is embedded in concrete at its base, the thermal expansion of the liner was calculated using the base as a reference point rather than the center of the sphere. Therefore, during the expansion of the liner relative to the concrete, contact between the liner and the compressible material first occurs in the spherical shell in the vicinity of the knuckle and the differential pressure is calculated to reach a maximum at this point. This calculated pressure varies from a maximum at the knuckle to zero at the point of embedment. However, it was observed from deflection measurements made during compression of the Firebar D that the interface pressure acting on the metal liner did not reach a maximum value at the knuckle and vary to zero close to the point of embedment as predicted by calculation, but rather assumed a uniform value of pressure as indicated by the relatively uniform radial displacement of the shell. The displacement measurements were made at the various penetration locations which were accessible for measurement.

Maximum measured vessel expansions compared with the calculated expansions at teach gauging point are tabulated below:

DRYWELL MOVEMENT AT TIME OF MAXIMUM AVERAGE TEMPERATURE AND PRESSURE
180.5°F; 42 psig; 9:45 pm, March 11, 1967

				Movement (inches)					
				Calculated			Measured		
Designation	Penet. No.	Elev.	Azimuth	Hor.	Vert.	Total*	Hor.**	Vert.	Total*
2	X-66	27'	38°	0.42	0.22	0.48	0.46	0.01	0.46 0.43 (avg)
4	X-2A	27'	171°-51'	0.42	0.22	0.48	0.40	0.05	0.40
1	X-32	36'	125°	0.42	0.34	0.54	0.43	0.02	0.43
3	X-36	44'	290°	0.43	0.44	0.61	0.38	0.20	0.43
5	X-63	62'	340°	0.31	0.66	0.73	0.32	0.50	0.59 0.49 (avg)
6	X-12A	62'	240°	0.31	0.66	0.73	0.36	0.10	0.37
7	X-19	86'	190°	0.21	0.93	0.95	0.31	0.52	0.61 (cyl)

Figure 3.8-4C shows the locations by azimuth and elevation of the points at which measurements were taken. The figure shows that the azimuthal spacing of measurement points is relatively uniform while the elevation locations adequately represent the outer surface of the sphere. Therefore, it was concluded that the assumed uniform radial expansion of the liner was not a local phenomenon but was representative of the entire sphere.

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* $(Hor.^2 + Vert.^2)^{1/2}$

** Dial readings corrected for 140°F avg. penetration pipe temperature.

In order to further substantiate the assumption that the expansion was uniformly radial and on the average to 0.43 inches for the pressure and temperature conditions of testing, a thermal expansion calculation was performed using the center rather than base of the sphere as the origin of the expansion. The radial expansion, DR, was assumed to be

$$R = R\, a\Delta Teq,$$

where:

R = radius of sphere, in. = 420 in.

α = mean coefficient of thermal expansion, °F
= $6.6 \times 10^{-6}\ F^{-1}$ (steel)
= $5.5 \times 10^{-6}\ F^{-1}$ (concrete)

ΔTeq = equivalent temperature change of liner, °F
= ΔT (metal liner) + ΔT (differential pressure)

Δ T (differential pressure = temperature equivalent of radial expansion caused by differential pressure (i.e., 1 psi ΔP -1F ΔT).

At the beginning of the test, the concrete temperature is 43°F (and remains constant throughout test) and the metal temperature is 58°F (for zero measured deflection). At test conditions of T = 180.5°F and P = 42 psig, assume that the interface pressure is approximately 14 psi (obtained from Figure 3.8-24B for a deflection of 0.43 in). Then DTeq = DT (metal liner) + DT (differential pressure) = (180.5 - 58) + (42 -14) = 150.5°F and DR = R aDTeq = (420) (6.6) (10^{-6}) (150.5) = 0.42 in. At DR = 0.42, the external pressure is still approximately 14 psi so another iteration is not required. The calculated value of DR = 0.42 is within 3 percent of the measured value (DR = 0.43) assuming the expansion occurs from the center of the sphere. Hence, all subsequent calculations utilized this same assumption.

When the vessel is cooled and depressurized it contracts to a displacement of 80 percent x 0.43 inch = 0.34 inch at which point the interface pressure becomes zero. Further contraction of the vessel causes the gap to enlarge even more. An even larger expansion gap will be obtained when the reactor begins initial operation because the concrete containment shell will be at a much higher average temperature during normal reactor operation. The nominal drywell ambient temperature is 135°F during reactor operation while the ambient temperature in the Reactor Building is never permitted to drop below 60°F. The average temperature of the concrete for these conditions of temperature is calculated to be approximately 100°F (3°F allowance for gamma heating) even though the average temperature is expected to be somewhat higher during hot summertime periods.

Therefore, the minimum additional radial gap provided by the concrete in heating from the test temperature of 43°F to the normal operating temperature of 100°F is $\Delta R = R\, a_{c0}DT$ (concrete) or ΔR = (420) (5.5) (10^{-6}) (100 - 43) = 0.13 in. If the liner were cooled to its original temperature of 58°F, the gap between the liner and insulation would be 0.34 + 0.13 = 0.47 in (concrete at an average temperature of 100°F). It is assumed that the maximum drywell ambient temperature just before a Maximum Credible Accident (MCA) is 140°F (conservative) since an annunciator will alert the reactor operator if the normal drywell ambient temperature (135°F) is exceeded by more than 5°F. Therefore, at the start of an MCA, the gap has been reduced by the amount ΔR= (420) (6.6) (10^{-6}) (140 - 58) = 0.23 to a value of 0.47 - 0.23 = 0.24 inches. The liner first makes contact

with the Firebar D at a liner temperature to T = [0.47 ¸ (420)(6.6)(10^{-6})] + 58 = 228°F after the MCA. Subsequent temperature increase causes an increase in external pressure on the liner as the Firebar D is compressed.

The analysis of drywell liner temperature response (Subsection 6.2.2.3.3) indicates that there is some uncertainty in the selection of condensing and convective transfer coefficients which greatly affects the liner temperature response after a Loss-of-Coolant Accident. However, it is anticipated that the heat transfer coefficients will fall somewhere in the bounded region. Corresponding to the curves which bound the liner temperature response are two curves which bound the liner external interface pressure response, the upper pressure curve corresponding to the upper temperature curve and lower pressure curve to the lower temperature curve. The external pressure was calculated by selecting a liner temperature and a corresponding drywell internal pressure at some time after the beginning of the Loss-of-Coolant Accident. Using the procedure outlined earlier, a ΔT_{eq} was calculated considering both the temperature change and an assumed pressure differential.

A radial liner displacement was determined from $\Delta\text{-}_{eq}$; then using the graph of physical characteristics of Firebar D as shown in Figure 3.8-24B, an external pressure was determined. As an example of the calculation, select the point 50 seconds after the MCA where the maximum liner temperature is 262°F and the internal pressure is 20 psi. For this condition it is arbitrarily assumed that the interface pressure will be 16 psi (this must be iterated if not closed correctly) as that ΔT (metal liner) = 262 - 58 = 204°F and ΔT (differential pressure) = 20 - 16 = 4°F or ΔT_{eq} = 208°F. The radial growth of the liner for this case is ΔR = (420) (6.6) (10^{-6}) (208) = 0.58 inches of which 0.47 inches is required to close the annular gap and 0.11 inches is required for compressing the Firebar D. Therefore, to determine the external pressure on the liner, the compression of the Firebar D is measured from the point of initial contact (0.34 inches deflection on Figure 3.8-24B) to 0.11 inches beyond that point (0.45 inches) where the pressure is actually found to be approximately 16 psi. Later during the transient, at t = 200 - 400 seconds, the internal pressure decreases to a point where a net excess external over internal differential pressure exists (approximately 5 psi maximum).

The liner external pressure curves are plotted versus time on the same figure with liner temperature, ambient pressure, ambient temperature and allowable external pressure for the spherical portion of the drywell. Since the external pressure of the cylinder, calculated using the same procedure as for the sphere, in no case exceeds the internal pressure, its plot was not shown on the graph.

The concern about possible buckling is tempered by the self relieving nature of the interface pressure which diminishes to zero with a very small uniform inward displacement of the shell. The stability equation for buckling of a spherical shell, applicable where external pressure is constant and continuously applied, is:

$$P_{cr} = 2Et^2/R^2(3(1\text{-}v\text{-}2))^{1/2} \qquad \text{(Reference 12)}$$

where:

P_{cr}	= critical buckling pressure, psi	
E	= modulus of elasticity for steel, psi	= 30 x 10^6psi
t	= minimum thickness of spherical shell, in	= 0.722 in
R	= radius of spherical shell, in	= 420 in
v	= Poisson's ratio for steel	= 0.3

The critical buckling pressure obtained from this equation becomes:

P_{cr} = 107 psi

Von Karman indicates (Reference 12) that experimental values of buckling pressure are often times as little as 1/3 or 1/4 the theoretical value because of eccentricity, nonuniform wall thickness, etc. in the actual vessel. This implies an actual buckling load of between 27 and 36 psi for continuous overpressure on the spherical portion of the drywell. Even though the calculated critical buckling pressure is not precisely applicable for this type of self relieving external load, it does serve as a basis for comparison to reveal any safety margin which may be inherent in the design.

For conservatism the lower value of buckling load, i.e., 27 psi, will be used as a design guide in determining the allowable excess interface pressure (excess external over internal pressure). It is shown on Figure 3.8-24A as a curve 27 psi above the internal pressure curve and is the limit for allowable external pressure. If this value of overpressure is achieved or exceeded, wrinkling or local buckling could occur; however, it should be remembered that since the external pressure is being supplied by the thin layer of Firebar D and is self-relieving as the shell deflects under the load, gross collapse of the shell by buckling as if under the influence of a continuously applied hydrostatic or pneumatic external pressure can never occur.

An inspection of the consolidated graph (Figure 3.8-24D) indicates that the external pressure on the metal liner does not exceed the allowable external pressure (Figure 3.8-24A) during any time after the Loss-of-Coolant Accident. Therefore, neither local nor general buckling should occur in the liner as a result of the differential thermal expansion between the metal liner and the concrete which accompanies the Loss-of-Coolant Accident.

A test program was conducted specifically for the application of the Firebar Type D which was directed essentially toward developing information regarding its pressure versus deflection characteristics under compressive loading. The tests were arranged to simulate the conditions under which the material would be used in order to demonstrate stability and adhesion to the steel surface.

The material differs from asbestos fiber - magnesite cement products used for insulation and fire proofing since about 1885 only in that its density is controlled by a foaming agent which plays no part in the final structure of the material. Prior to consideration of its use for this special application, Firebar had been subjected to a standard fire test in accordance with ASTM E119 during which a 1/8 inch thickness of the material applied to a vertical metal duct withstood a 2 1/2 hour exposure to fire resulting in a 900°F metal temperature without separating from the metal.

Tests specifically for this application of the material conducted by the United States Testing Company, were to determine increments of pressure required to cause increments of deflection up to 50 percent of sample thickness. The samples were made using the production equipment and procedure to spray onto metal surfaces; the tests were made with samples in vertical and horizontal positions, at ambient temperature and at 300°F. Material loss after compaction was measured on test panels compressed in the vertical position; loss was about 1 percent of compressed sample weight; it was observed that loss was occurring at the break in the samples at the perimeter of the compression shoe, a discontinuity which would not occur in service. The reduction in thickness of the samples results principally from the collapse of the cellular structure impacted by the foam and maintained by the magnesite cement, however, some elastic compression of the asbestos fibers would be expected. The test samples were retained by the

testing agency for periodic observation of rebound; rebound stabilized at 20 percent of total deflection.

The tests and evaluations indicated that the foamed asbestos fiber magnesite cement product has the required compression characteristics and stability, and would be unaffected by long term exposure to radiation and heat.

Further evaluation of the design of Primary Containment penetration is presented in References 13 and 14.

3.8.2.5 Structural Acceptance Criteria

The Structural Acceptance Criteria relating the design and analysis results for the loads and load combinations given in Subsection 3.8.2.3 to the allowables, is presented in Subsection 3.8.2.4 and other referenced documents. The Basic Design phase of the Containment System is given in Subsection 3.8.2.4 and the references listed in Subsection 3.8.6. These reference documents must be addressed to obtain complete information.

A summary of allowable stresses considered in the original design of the facility used in conjunction with certain seismic loading combinations is given in Table 3.8-3.

3.8.2.6 Material, Quality Control and Special Construction Techniques

Materials used for the steel containment vessel and its penetrations include steel plates, bolts, pins, seamless pipe and welding filler materials, coatings and other miscellaneous items. These materials are listed in Subsection 3.8.2.2. Silicone rubber seals are used in the personnel access lock and equipment hatch.

Quality control procedures employed for the fabrication and erection of the containment vessels were in keeping with the normal requirements of the codes, standards and specifications identified in Subsection 3.8.2.2. Suppliers of materials were required to furnish Certified Materials Test Reports (CMTR), prepared in accordance with requirements commensurate with applicable material. CMTRs include results of all required chemical analysis, physical tests, mechanical tests, examinations including radiographic film, repairs and heat treatments performed on the material.

There are no special construction techniques.

3.8.2.6.1 Basic Materials

Plates

A-212-61T, Grade "B", made to ASTM A-300 requirements. The minimum Charpy vee notch impact test values were 20 ft-lbs at 0°F instead of 13 ft-lbs on full size specimens as permitted by Code Case 1317. Test specimens were taken both parallel to and transverse to the direction of final rolling of the plate.

Forgings

A-350, Grade LF1. Minimum Charpy vee notch impact test values were 13 ft-lbs at 0°F in addition to required Charpy keyhole impact tests.

Pipe

A-333, Grade "0", seamless. Minimum Charpy vee notch impact test values were 13 ft-lbs 0°F on full size specimens in addition to required charpy keyhole impact tests.

Miscellaneous Plate and Structural Steel (not within Scope of ASME-A36)

All permanent structural attachments and lugs, welded to the shells, were made of impact tested material for a distance of not less than 16 times the plate thickness. The erection skirt supporting the drywell was also made of impact tested material.

3.8.2.6.2. Fabrication Approach and Controls

The steel required for the fabrication of this facility was rolled at Eastern Mills. In order to insure that only materials of the highest quality were used, incentives were offered to shop employees finding defects which had been passed over at the mill, such as plate laminations. Furthermore, the certified mill test reports were reviewed to assure their compliance with the material specifications.

Workmen experienced in this type of fabrication prepared templates, dies, and jigs for the manufacturing process. The spherical plates of the drywell were formed on a 1500 ton hydraulic press designed and manufactured by the fabricator in order to meet production quality requirements. Plate edges were trimmed and beveled with semiautomatic burning torches after the forming process was completed. Heavy cylindrical sections, such as flange rings, were formed on a press brake and lighter weight cylindrical sections were formed utilizing standard plate rolls. Fabrication tolerances were closely controlled in order to minimize excessive gaps in weld joints and the accompanying excessive distortion from welding.

Shop Welding

Components were shop welded, where possible, into large size shipping pieces, utilizing either submerged or metallic coated arc techniques. In either case, low hydrogen electrodes were used, thus assuring the notch toughness requirements to meet the ASME Code Impact Tests.

a. Seam Welds

All seam welds in the shell of the Containment were of the double bevel butt type. All butt joints in any accessories subject to the ASME Code were also of the double welded type or equivalent, and all the tee joints were full penetration welds. Welding details for nozzles are of approved types. All welds subject to the Code were radiographed or otherwise examined in accordance with ASME Code Case 1272 N-5. All mandatory provisions of this code were followed and all recommended provisions were also followed where practical.

All welders and welding procedures were qualified in strict accordance with the requirements of Section IX of the ASME Code and of the Burns & Roe, Inc. (B&R) specifications.

In addition, the following requirements were incorporated into the design and fabrication:

- The design, method and sequence of welding was subject to the review and approval of B&R and General Electric Co. (GE) prior to performance of welding.
- Prior to welding, all protective coating was chemically or mechanically removed from all areas within two inches of the seam to be welded.
- In manual arc welding the electrodes were of the low hydrogen type, and were such that the physical and chemical properties of the resulting welds met the full requirements of the physical and chemical properties of the base metal.
- All automatic welding was done by the submerged arc process, and physical and chemical properties of the resulting welds met the full requirements of the physical and chemical properties of the base metal.
- Preheating at 200° minimum was applied to all seams whose thicknesses exceeded 1 1/4 inches regardless of the surrounding air temperature. Preheating at 100°F as also applied to all seams whose thicknesses were less than 1 1/4 inches when the surrounding air temperature was below 40°F.

b. Test Plates

Test plates were made and tested in accordance with Paragraph UG-84 of the ASME Code as modified by ASME Code Case 1317, employing a test temperature of 0°F and using the same material and thickness range as in the shell for each welding position used in construction for:

- Each brand of low hydrogen electrode used in construction.
- Each combination of wire and flux for automatic welding used in construction.

Only those low hydrogen electrodes and combinations of wire and flux producing welds which at least met the impact values of the parent material, as specified, were utilized in the construction.

c. Heavy Welds

Heavy weldment and penetration weldments were furnace stress relieved as follows:

- Any plate segment wholly containing a penetration, nozzle, or column connection was furnace stress relieved at the shop after insertion of the penetration.

- All large penetrations intersecting more than one shell plate were stress relieved as follows: Any portion of a penetration containing seams joining metal over 1 1/2 inch thick at the joint was furnace stress relieved as a unit before welding into a penetrations assembly or into the shell.

d. Weld Inspection

All shop welds were radiographed in the shop. All welds in those parts of the work subject to the ASME Code were radiographed by methods complying with Paragraph UW-51 of the Code. Unsatisfactory negatives were rejected and new radiographs were made of the portions of the work covered by the unsatisfactory negatives until satisfactory results were obtained. All negatives were examined and all welds which failed to meet the standards of radiographic quality set forth in the ASME Code were cut out, rewelded, and reradiographed until all such welds satisfied the standards of the ASME Code.

The radiographic film was fine grain, high contrast type, equivalent to Eastman Kodak AA. Lead intensifying screens were used. The film density was within the range of 1.5 to 2.5 as determined by either film density specimen or densitometer.

Welds on which radiographing was impractical were examined for cracks by the magnetic particle method of inspection in accordance with Appendix VI of the ASME Code.

All negatives and certified interpretations thereof were submitted to B&R for further examination and are on file at the OCNGS fire proof vault.

Hatches and Locks

Lock doors and other appurtenances requiring shop finishing were machined on either a large vertical boring mill or on smaller equipment, there being adequate equipment to end-face the door flanges on the lock and equipment hatch. The personnel lock was shop fabricated and tested, both mechanically and for leaktightness, before shipment.

Cleaning and Coating

Prior to shipment all materials were cleaned and painted in accordance with the B&R, Inc. specifications as follows.

All oil, grease, dirt, loose rust, loose mill scale, and other foreign substances were removed. The removal of oil and grease was accomplished before mechanical cleaning was started, using mineral spirits or other paraffin free solvents having a flash point higher than 100°F. Clean cloths and clean fluids were used to avoid leaving a film or greasy residue. The use of chipping tools which could produce cuts, burs, and other forms of excessive roughness was not permitted.

The areas described below received one shop coat of Carboline Carbo-Zinc 11 paint. Surface preparation and application was in accordance with the paint manufacturer's recommendations. The steel was pickled in accordance with the 3-Bath Horton Pickling Process. After assembly all areas were sandblast cleaned in accordance with Steel Structures Painting Council Specification SSPC-SP6-63 for commercial blast cleaning. Surfaces painted in the above manner were the interior surfaces of the drywell above the concrete floor, including jet deflectors, the exterior of the drywell above the water seal support bracket, the interior and exterior surfaces of the absorption

chamber, the interior and exterior surfaces of the downcomers and header, and the exterior surface of the vents within the absorption chamber. The interior surface of the drywell below El. 12'-3" and the exterior surface of the drywell in direct contact with final support concrete was not painted.

All other surfaces of the drywell, the absorption chamber and the vent pipes were pickled and, after cleaning, given one coat of Carboline primer conforming to Federal Specification TT-P-86c Type 1.

After erection and testing, all field welds and abraded places on the shop paint were cleaned by sandblasting and painted as specified above.

The vessels have been inspected for compliance with the ASME Boiler and Pressure Code. Manufactures' data reports as well as certificates of shop and field inspection are included in Appendix IV - 1.0 (Item 2) of Reference 1.

In 1983, the Torus interior surfaces, the interior of the Vent System up to the drywell and all external surfaces of the Vent System were grit blasted to SSPC-10 or SSPC-5 at 1 1/2 - 3 mils profile.

Pitted surfaces of immersed Torus shell were repaired by welding. Rough areas of Torus shell were blended by grinding. Mobil 46-X-16 Epoxy Filler was applied to selected pitted areas of the Torus immersed shell portion prior to coating. Surfaces in the Vent System thinned by corrosion were repaired by welding.

The immersed bottom half of the Torus shell, the interior of the downcomer and the entire interior surfaces of the Vent System were given 3 coats of Mobil 78 Hi-Build Epoxy (DFT-16 mils). The vapor phase upper half of the Torus shell, exterior of the Vent Header and vent lines portions inside the Torus were given two coats of Mobil 78-Hi Build epoxy (DFT-10 mils).

Following coating application, the entire Torus interior was heat cured at 108°F for 48 hours. Demineralized water was put back in the Torus. Subsequent to this coating application, minor coating repairs have been performed using BRUTEM-15 (UT-15).

3.8.2.6.3 Erection Approach and Controls

A 100 ton capacity guyed derrick was erected on a 60 foot tower to be used in conjunction with the construction of the drywell and pressure absorption chamber. It also served the adjoining subassembly area for fitting, welding and radiographing of the various weldments.

The foundation for the derrick was furnished and installed in accordance with the design information supplied by CB&I. The derrick foundation had to be located immediately adjacent to the pressure absorption chamber to allow maximum derrick coverage of both the drywell and subassembly area. One piece of auxiliary lifting equipment was utilized during a portion of the erection time for miscellaneous lifting requirements, such as unloading materials and setting the pressure chamber sections outside of the reach of the derrick itself.

The 70 foot diameter spherical drywell and neck were field assembled and welded. The transition knuckle and top head flanges were field stress relieved in accordance with the ASME Code, utilizing the latest techniques developed as the result of having performed over 150 field stress reliefs using luminous flame techniques.

The heavy plate flanges for the 33 foot diameter cover and neck flanges of the drywell were subassembled in segments, welded, x-rayed and stress relieved as complete units.

The machining was accomplished by special equipment designed and constructed by CB&I. This machining unit has been successfully used on several projects where close tolerances on double "O" ring seals have been specified. The top cover was subassembled to one flange and a cylinder was subassembled to the lower flange, thus both sections were ready for final installation on the vertical cylinder of the drywell.

The rigid integral flanges supplied by this technique, which will hold pressure without seal welding, and the design comply with the ASME Code.

The pressure absorption chamber was subassembled into convenient erection segments, including the circumferential stiffeners and the column supports. The connecting vent pipes and expansion joints were subassembled into appropriate lengths for maximum ease in erection.

Multiplate sections were made for the spherical drywell in the subassembly area using an automatic welding machine setup in a subassembly yard.

The pressure absorption chamber was erected on foundations already in place, concurrent with the placement of the internal vent headers inside for later assembly and positioning. A minimum clearance of 30 inches was provided for access between any concrete foundations and walls where welding and radiography was to take place. The vent headers and downcomers were fabricated as subassemblies and placed inside the suppression chamber as erection progressed around the torus, and eventually were welded into position and the necessary ties made.

Special subassemblies had to be made prior to assembling and welding the drywell. The large transition knuckle between the spherical and cylindrical sections was subassembled, and the seams radiographed and fully stress relieved. A temporary furnace was constructed around the knuckle for this operation.

A clear area, approximately 200 ft by 300 ft in size was made available near the derrick in order to have a continuous flow of materials from the unloading yard through subassembly activities and into the hole.

The drywell was erected on a skirt, thereby eliminating the use of support columns at the sphere equator and their removal after embedment of the lower sphere segment. The balance of the spherical drywell, the transition knuckle, and the cylindrical neck were then erected. The vent pipes and attendant expansion joints, connecting the drywell to the absorption chamber were then fitted and installed in their final positions. Field x-ray laboratories were used for radiographing the various weldments.

All field installed penetrations and locks were placed in their final positions in the sequence most suitable for actual field conditions and good erection practice.

All seams, where thicknesses exceeded 1 1/4 inches, were preheated to 200°F minimum before welding. Seams, whose thicknesses were equal to or less than 1 1/4 inches, were preheated to 100°F when the ambient temperature was found to be lower than 40°F.

All completed shell plate assemblies, with penetrations installed, were stress relieved after fabrication. All butt welds were 100 percent x-rayed. Other welds which could not be 100 percent x-rayed were magnafluxed before and after stress relieving.

3.8.2.6.4 Material Parameters Used in Design Modifications

The material parameters used in the Design Modification of the Suppression Chamber are based on the requirements of the 1977 Edition with Addenda through Summer 1978 of the ASME B&PV Code, Section III, Division I, Subsections NE and NF (Subsection 3.8.2.2). These parameters included Young's modulus, coefficient of thermal expansion, Poisson's ratio, yield strength, and allowable stress. The appropriate values have been used corresponding to the material temperature existing at the time of each loading condition.

The material of the principal structures of the original torus and vent system was steel plate specified as ASME SA-212, Grade B. This material specification has been superseded in the ASME Code applicable for this evaluation (Subsection 3.8.2.2) by ASME SA-516, Grade 70. Therefore, this current specification is used for purposes of defining material parameters.

All modifications use ASME Code materials and material properties as specified in the referenced ASME Codes (Subsection 3.8.2.2). Weld materials comply with the referenced ASME Code requirements, thus weld material properties are based on the ASME Code.

3.8.2.6.5 Erection Tolerances

The requirements in Paragraph UG-80 and UG-81 of the ASME Code shall be followed. The average elevation of top head flange shall be kept within ±3 inches of the theoretical elevation and top flange shall not deviate more than ±1/2 inch from true horizontal plane. At no place shall the location of the vertical centerline deviate more than 3 inches from the theoretical location.

3.8.2.7 Testing and Inservice Requirements

The pressure testing and inservice inspection program consists of the integrated leak rate test and inservice leak rate testing of the vessels as discussed in Section 6.2. The preoperational structural integrity test and visual examination are described as follows.

Since no new or previously untried design approaches are used for the Containment Vessels, there is no special inservice pressure testing requirements other than the inservice leak rate testing discussed in Section 6.2.

Testing and inservice inspection work is covered by three basic areas:

a. Structural Pressure and Leak Rate Testing

b. Response of Suppression Chamber to Relief Valve Actuation

c. Suppression Chamber Corrosion Pitting Investigation

3.8.2.7.1 Structural Pressure and Leak Rate Testing

To demonstrate that the pressure containment vessels (drywell and suppression chamber) will respond satisfactorily to the postulated internal pressure loads, a preoperational structural integrity test (SIT) was performed at 1.15 times the original maximum design pressures of 62 psig for the drywell and 35 psig for the suppression chamber. These tests were conducted as follows.

General Procedure

The procedures for the overload tests fulfilled the requirements of Section VIII of the ASME Code and Code Case 1262 N-5. The drywell was pneumatically air tested to 115 percent overload. The absorption chamber was also pressurized with air to 115 percent overload in two separate tests, first, with the vessel empty and second, with the vessel filled to design level with water.

The method used for the leakage rate tests consisted, basically, of initially comparing the pressure in the containment vessel with an airtight inner chamber which is an integral part of a reference system.

The location of the inner chamber, inside of the containment vessel and approximately at the center of the air mass, enabled the average temperature of the air in both inner and outer vessels to be reasonably close during the daylight hours and practically equal during the late night hours. Data obtained from previous tests have shown, during the midnight to dawn periods of normal atmospheric conditions, that the air temperature becomes relatively uniform throughout the containment vessel and that the temperature at the geometric center represents the average air temperature throughout the vessel.

With negligible difference in average air temperature between the inner chamber and the containment vessel, the possibility of a pressure differential being caused by temperature changes can be eliminated. Consequently, any relative decrease in containment vessel pressure must be assumed to occur as a result of external leakage. By measuring the difference in pressure between the two air volumes with a water manometer, a high degree of sensitivity to this pressure differential can be accomplished.

The report of the tests conducted ("CB&I Report of Initial Overload Tests and Leakage Rate Determination of the Pressure Suppression Vessels") in Appendix IV-3.0 of Reference 1, describes the relationship of differential pressure measurements to the percent leakage. The leakage test of the drywell was conducted with the vessel in the "dry" condition, with no free water present. The leakage tests of the absorption chamber were first conducted with the vessel in the "dry" condition and later with the vessel in the "wet" condition (with the vessel filled to design level). In the "wet" test of the absorption chamber, measurements of vapor pressures and temperatures were taken and were included in the calculation for leakage.

Pressure Testing at the Site

The detailed steps of the preliminary tests, the overload tests, and the leakage rate tests are given in the "CB&I Report of Initial Overload Tests and Leakage Rate Determination of the Pressure Suppression Vessels" and are summarized below.

Before the overload and leakage rate tests were conducted at Oyster Creek, preliminary testing on all subassemblies and test equipment was performed in the shop and in the field. All shop welded manholes and nozzles were magnafluxed after shop stress relief without indication of cracks or defects. The personnel air lock was shop assembled and pressure tested for structural adequacy. A tightness check of the locks was performed in the shop and included gasket seals, equalizing valves, shaft penetrations and nozzles. The chamber and appurtenances for the reference systems were shop tested with freon at approximately 71.3 psig.

Reference System Testing

At the Oyster Creek site, the reference systems for the drywell and pressure absorption system were initially checked for tightness. After installation, the vessels were tested by pressurizing with freon to about 71.3 psig and using a halide test detector. This was followed by a holding period with nitrogen at about 71.3 psig when pressures and temperatures were recorded.

The holding period for the drywell system was initiated on September 29, 1965. Because minute leakage was indicated during the holding period, a second reference system was installed, immediately adjacent to the first system, and the holding period continued with freon. A study of the pressure temperature data during the holding period indicated that the pressure in the systems followed the vapor pressure of the freon for the measured temperature. Thus it was concluded that an unstable condition existed because of the probable vaporization and condensation of the freon vapor.

The systems were then purged of freon and pressurized again with nitrogen for a holding period. The subsequent data showed that the desired pressure temperature relationship followed the gas law, which was the basis for comparison during the holding period. The data from the No. 1 reference system was not entirely satisfactory and as a consequence, this system was not used for any further leakage rate testing. The No. 2 reference system was held from October 16 through November 10, 1965, when air was pumped into the drywell for the overload test.

The holding period for the reference system for the absorption chamber "dry" test was initiated on January 4, 1966 and extended into January 7, when the overload test was started. The holding period for the reference system for the absorption chamber "wet" test was from January 5 into February 25, 1966.

The two gaskets of the personnel lock were checked by pressurizing the space between the gaskets and inspecting both gaskets for possible leakage. The tightness of the lock was checked in the field by a soapsuds test.

Overload Tests for Drywell and Personnel Access Lock

After the successful checking of the No. 2 reference system in the drywell, the vessel was closed for the overload test. No water was introduced into the differential manometer until the start of the leakage rate test. Figure 3.8-25 illustrates the piping layout for the overload test. The drywell was pumped to 5 psig on November 10, 1965 and held overnight. On November 11, the vessel was inspected with soapsuds. Minor gasket leakage was found on the bottom manhole, which was eliminated by replacing the gasket. Leakage found on one of the nozzles was eliminated by tightening the bolts on the flange. Leaks were also found in the temporary welds on the downcomer caps. The pressure was released and the leakage eliminated by rewelding on the temporary welds of the downcomer caps. On November 12, the pressure was raised again to 5 psig. Soapsuds inspection revealed a few more leaks at the downcomers, which were repaired after the pressure was released.

Starting at 5 P.M. on November 12, 1965, the pressure was applied in increments until the test pressure of 71.3 psig was reached at 5:50 A.M. on November 13.

The vessel proved to be adequate for the test pressure and after 30 minutes, the drywell pressure was reduced to 66 psig. The personnel access lock was then pressurized to 71.3 psig and held at that pressure for 10 minutes, proving its adequacy for the test pressure. The lock pressure was then reduced to 66 psig and both the lock and the drywell were then brought to the design pressure of 62 psig. The soapsuds inspection was made at 8 A.M. on November 14 at a pressure of 62 psig.

The soapsuds inspection at the 62 psig pressure for both the drywell and personnel lock found the following: three minor leaks in temporary welds on downcomer caps; one leak on a nozzle cap connection; slight leakage at the tapped hole provided for testing the gaskets of the top head; one of the nozzle connections showed leakage at its point of attachment to the shell plate. Three small leaks were noted. These leaks were repaired after completion of the leakage rate test and were checked by pressurizing a box, welded to the shellplate, over the connection.

Overload Tests for the Pressure Absorption Chamber

After the successful checking of the reference system in the "dry" absorption chamber, both the drywell and the absorption chamber were closed for the overload test of the absorption chamber. No water was present in either vessel. Air was simultaneously introduced into both vessels at 10:15 A.M. January 8, 1966. The drywell was pressurized at the same time to avoid an external pressure on the vent pipes and header inside of the absorption chamber. Figure 3.8-26 illustrates the piping layout for the overload test.

To perform the soapsuds test of the expansion joint welds on the vent lines, the protective steel coverings were removed. During this operation the expansion joint that was installed first was found to be distorted. The joint was judged adequate for the test, with a new joint to be installed later. Periodic checks of the joint were made during the pressurization and after the overload test.

The soapsuds test at 5 psig was conducted on January 8, 1966 and then the pressure was increased in intervals to the test pressure of 40.25 psig, which was reached at 11:30 P.M. After one half hour holding time and the vessel having been proven adequate for the test pressure, the pressure was reduced to the design pressure of 35 psig for the final soapsuds inspection and "dry" leakage rate test. The "dry" leakage rate test was satisfactorily completed by 10 A.M. of

January 10, 1966. The final soapsuds inspection after the "dry" test found no leakage. The pressure was then released from both vessels.

Subsequent to the testing described above, the distorted expansion joint was replaced. After the new expansion joint was installed, the absorption chamber was filled to design level with water for a test of both drywell and absorption chamber. The lower part of the absorption chamber, below the water level, was inspected for tightness.

The drywell was pumped up to 5 psig by 3 P.M. on February 25, 1966 and the rewelded parts of the new expansion joint and the vent pipe were checked with a soapsuds solution with satisfactory results. The pressure was increased to the test pressure of 71.3 psig at 4:55 A.M. on February 26 and held for 10 minutes to the satisfaction of the Hartford Inspector, B&R, GE, and CB&I.

The pressure was reduced to 62 psig and the rewelded parts of the new expansion joint and the vent pipe were again checked with a soap solution. The drywell pressure was then lowered to 45 psig.

The manhold of the absorption chamber was closed and the pressure increased to 5 psig, starting at 7 A.M. February 26, 1966. The rewelded parts of the expansion joint and all connections that had been opened since the "dry" test (January 1966) of the absorption chamber were satisfactorily checked with a soap solution. The absorption chamber was then pumped up to the test pressure of 40.25 psig at 11:15 A.M. on February 26 and held for one half hour to the satisfaction of the Hartford Inspector, B&R, GE, JCP&L, and CB&I.

The pressure in the absorption chamber was lowered to 35 psig on February 26 for the holding period of the leakage test. The leakage test described below was completed by 8 A.M. on February 28 and the air pressure released from both vessels.

The above effort completed the overload testing for the drywell and pressure absorption chamber.

Leakage Rate Tests

Prior to the start of the leakage rate test of the drywell, the blowoff valve at the bottom of the vessel was opened for the purpose of draining off any condensate that might have accumulated during the pressurization. No condensate was blown off.

At 12:45 P.M. on November 13, 1965, oil with a specific gravity of 1.0 was introduced into the d2ifferential manometer of the reference system of the drywell. Additional air was withdrawn from the vessel in order to establish a differential on the manometer. Figure 3.8-27 illustrates the equipment layout for the drywell test.

The pressure and temperature readings were recorded hourly from 1 P.M. on November 13 until 11 A.M. on November 15, 1965. After a review of the leakage rate data and acceptance of the test as successful, the pressure was released from the drywell on November 15. The reference system was then removed from the drywell. Using the average midnight to 6 A.M. data from the two successive periods, the calculated leakage (as a negative value) per 24 hour period was -0.078 percent, which was less than the allowable -0.2 percent stipulated by the specifications. The results were considered acceptable by B&R, GE, and CB&I.

The holding period for the "dry" leakage rate test of the absorption chamber started at 3 A.M. on January 9, 1966, after oil had been previously introduced into the manometer and a differential established. The readings continued until 10 A.M. on January 10, 1966, when the air pressure was released. A slightly higher pressure was held in the drywell during the test in order to avoid an external pressure on the vent lines and header. The reference system was removed from the absorption chamber after acceptance of the test. Figure 3.8-28 illustrates the equipment layout for the "dry" absorption chamber test.

Using the average of the three hours of uniform ambient temperature during each period, the calculated leakage (as a negative value) per 24 hour period was -0.041 percent, which was less than the allowable -0.2 percent stipulated by the specifications. The results were considered acceptable by B&R, GE and CB&I.

Readings for the "wet" leakage rate test of the absorption chamber were initiated at 3 P.M. on February 26, 1966, after oil had been previously introduced into the manometer and a differential established. The pressure in the drywell was maintained at about 35 psig during the holding period. Figure 3.8-29 illustrates the equipment layout for the "wet" absorption chamber test.

Internal fans were used in the absorption chamber for circulation of the air and water vapor in order to obtain uniformity in the air vapor space. To obtain a dew point temperature (and water vapor pressure), three dew cells were located about 90 degrees apart (plan view) in the vapor space of the absorption chamber. Six resistant bulbs were used for temperatures. One bulb was located in the water and one just above the water. Three were adjacent to the dew cells and one was located near the top of the vessel.

The "wet" leakage rate test of the absorption chamber was concluded at 8 A.M. of February 28 after the results were accepted. Using the average data (without vapor pressure correction) from the two successive midnight to dawn periods, the preliminary leakage (as a negative number) per 24 hour period was -0.062 percent. Considering only the change in water vapor pressure, the apparent loss (as a negative number) was +0.005 percent. Combining the above calculated values, the corrected loss (as a negative number) was -0.067 percent, which was less than the allowable leakage of 0.2 percent stipulated by the specifications. The results were considered acceptable by B&R, GE, and CB&I.

The above concluded satisfactorily the testing of the drywell and pressure absorption chamber.

3.8.2.7.2 Mark I Program Testing

Subsequent to the completion of the original design and start of commercial operation, behavioral loading phenomena were identified which required further reanalysis. Test work performed to verify the analysis of the response of the torus to relief valve actuation is documented in References 15 through 18.

3.8.2.7.3 Torus Inspections

Inspection of the surface areas of the torus above the water line is possible during refueling outages, or other periods when the reactor is shut down without draining the water from the absorption chamber. The water in the chamber can be lowered below normal water level for inspection.

Torus inspections to assess corrosion thinning are documented in References 16 and 19.

3.8.2.8 Drywell Corrosion

The potential for corrosion of the drywell vessel was first recognized when water was noticed coming from the sand bed drains in 1980. Corrosion was later confirmed by ultrasonic thickness (UT) measurements taken in 1986 during 11R. During 12R (1988) the first extensive corrective action, installation of a cathodic protection system, was taken. This proved to be ineffective. The system was removed during 14R (1992).

The upper regions of the vessel, above the sand bed, were handled separately from the sand bed region because of the significant difference in corrosion rate and physical difference in design. Corrective action for the upper vessel involved providing a corrosion allowance by demonstrating, through analysis, that the original drywell design pressure was conservative. Amendment 165 to the Oyster Creek Technical Specification (Ref. 48) reduced the drywell design pressure from 62 psig to 44 psig. The new design pressure coupled with measures to prevent water intrusion into the gap between the vessel and the concrete will allow the upper portion of the vessel to meet ASME code for the remainder life of the plant.

In the sand bed region laboratory testing determined the corrosion mechanism to be galvanic. The high rate of corrosion in this region required prompt corrective action of a physical nature. Corrective action was defined as; (1) removal of sand to break up the galvanic cell, (2) removal of the corrosion product from the vessel and (3) application of a protective coating. Keeping the vessel dry was also identified as a requirement even though it would be less of a concern in this region once the coating was applied. The work was initiated during 12R by removing sheet metal from around the vent headers to provide access to the sand bed from the Torus room. During operating cycle 13 some sand was removed and access holes were cut into the sand bed region through the shield wall. The work was finished during 14R.

After sand removal, the concrete floor was found to be unfinished with improper provisions for water drainage. Corrective actions taken in this region during 14R included; (1) cleaning of loose rust from the drywell shell, followed by application of epoxy coating and (2) removing the loose debris from the concrete floor followed by rebuilding and reshaping the floor with epoxy to allow drainage of any water that may leak into the region.

During 14R, UT measurements were taken from the outside surface of the drywell vessel in the sand bed region. Measurements were taken in each of the ten sand bed bays. The results of this inspection and the structural evaluation of the "as found" condition of the vessel is contained in Reference 44. As documented in the TDR, the vessel was evaluated to conform to ASME code requirements given the deteriorated thickness condition. In general these measurements verified projections that had been made based on measurements taken from inside the drywell. Several areas were thinner than projected. In all cases these areas were found to meet ASME code requirements after structural analysis.

The cleaning, floor refurbishing and coating effort completed in 14R will mitigate corrosion in the sand bed area. Since this was accomplished while the vessel thickness was sufficient to satisfy ASME code requirements, drywell vessel corrosion in the sand bed region is no longer a limiting factor in plant operation. Inspections will be conducted in future refueling outages to ensure that the coating remains effective. In addition, UT measurements will also be taken from inside the drywell. The frequency and extent of the coating inspections and UT thickness measurements will be per Reference 47, as follows:

1. For the upper elevations, UT measurements will be made during the 16th refueling outage (September, 1996) and during every second refueling outage, thereafter. After each inspection, a determination will be made if additional inspection is to be performed.

2. For the sandbed region, visual inspection of the coating as well as UT measurements of the shell will be made during the 22nd refueling outage (2008) and during every second refueling outage, thereafter. After each inspection, a determination will be made if additional inspections are to be performed.

3. For water leakage not associated with refueling activities, an investigation will be made as to the source of the leakage. Oyster Creek will take corrective actions, evaluate the impact of the leakage and, if necessary, perform an additional drywell inspection about three months after the discovery of the water leakage.

Reference 51 and 53 provides the evaluation of the latest drywell UT inspections through the next scheduled inspection.

Oyster Creek will notify NRC prior to implementing any changes to the drywell thickness measurement inspection program (Reference 43).

3.8.3 Concrete and Steel Internal Structures of Steel or Concrete Containments

The following section contains the physical description, design, construction and fabrication codes, and loading information as they relate to the structures housed within the drywell and torus. This information is extracted from References 20 through 27.

The drywell vessel, major penetrations and hatches are covered in Section 3.8.2. The refueling pools and the Reactor Building operating floor are covered in Section 3.8.4.

3.8.3.1 Description of the Internal Structures

A general description of the structural configuration and framing arrangement and reactor support is shown in the North-South section through the Reactor Building shown in Figure 3.8-1.

The following major internal structures are located within the Drywell. Pipe supports and structural framing systems are discussed in Subsection 3.8.2.4 and in other references identified therein.

3.8.3.1.1 Fill Slab

The fill slab is reinforced concrete placed in the bottom of the drywell vessel up to El. 10'-3" to provide a working base for supporting the reactor pedestal and other loads from the internal structures and equipment. The fill slab transfers all imposed loads to the drywell vessel foundation through direct bearing only. No dowels or other anchorages are provided.

3.8.3.1.2 Reactor Pedestal

The reactor pedestal is a reinforced concrete cylinder with an outside diameter of 26 feet 0 inches. The cylinder is doweled into the fill slab at El. 10'-3" and extends up to El. 36'-10 & 3/4" to support the reactor vessel and the Biological Shield Wall. Loads from the floor framing system are transferred to the pedestal by either direct connection, posts or through the Biological Shield Wall.

3.8.3.1.3 Biological Shield Wall (Sacrificial Shield)

The Biological Shield Wall is a composite steel, concrete cylinder with an inside diameter of 20 feet-11 3/4 inches. The wall is framed with a vertical grillage of 27 inches deep wide flange members and covered with 5/16 inch plate on the outside and 1/4 inch plate on the inside. The area between the plates is filled with concrete. Between El. 56'-5 3/4" and El. 69'-5 3/4", a special dense concrete mix is utilized to meet additional shielding requirements. The shield wall extends from the top of the reactor pedestal (bottom of base plate at El. 36'-7 1/4") up to El. 82'-2".

The bottom is fixed to the reactor pedestal by two concentric rings of anchor bolts. The top is stabilized by lateral supports attached to the drywell shell, thus creating a propen cantilever structure. Horizontal forces from the floor framing at El. 46' - 1 & 5/8" is transmitted to the Biological Shield Wall by brackets.

3.8.3.1.4 Floor Framing

There are two basic floors within the drywell vessel, these are located at El. 23'-6" and El. 46'-1 5/8". The main framing for the upper floor consists of radial beams which are supported off the drywell shell by hangers, posted to the Reactor Pedestal to transfer vertical forces and attached to the Biological Shield Wall to transfer horizontal forces. The main framing for the lower floor consists of radial beams which are seated on a bracket on the drywell shell and attached to the reactor pedestal to transfer both horizontal and vertical forces. Forces transmitted to the drywell shell are limited to the vertical direction by nature of the upper floor hanger and lower floor sliding type connections. The radial floor beams for both floors are braced by chord members at various interior locations and continuously on the outer circumference. Numerous openings are framed out for piping and equipment penetrations. The structural framing is topped with 1 1/2 inch deep open grating to provide a walking surface and the free passage of air and water.

3.8.3.1.5 Deleted

3.8.3.2 Applicable Codes, Standards and Specifications

The codes, standards and specifications relating to the design and construction of the internal structures is as given in Subsection 3.8.4.2.

The two classes of structures for which earthquake design requirements apply are as follows:

Class I - Structures and equipment whose failure could cause significant release of radioactivity or which are vital to a proper shutdown of the plant and the removal of decay heat.

Class II - Structures and equipment which are both essential and nonessential to the operation of the station, but which are not essential to a proper shutdown.

For the reactor vessel supports, the allowable stresses were as follows:

a. Seismic - Allowable stress = normal AISC allowable stresses.

b. Seismic + Jet - Allowable stress = 150 percent of normal AISC allowable stresses.

c. 2 Seismic - Allowable stress = 150 percent of normal AISC allowable stresses.

3.8.3.3 Design and Analysis Procedure

Biological Shield Wall (Sacrificial Shield)

The effects of the LOCA which were considered in the design of the sacrificial shield were limited to those associated with jet force reactions on the reactor vessel resulting from a complete circumferential failure of either a recirculation inlet or outlet line or one of the main steam lines. The failure was assumed to occur at the pipe-to-vessel nozzle weld resulting in lateral forces being applied to the vessel. The Sacrificial Shield Wall provided support for the vessel by means of stabilizers which are installed between the vessel and the top of the wall. The shield wall has been designed to withstand the shears and moments resulting from the jet force reactions on the reactor vessel. The recirculation inlet and outlet nozzles project partially through the shield wall. The steam outlet nozzles are above the top of the wall; therefore, the sacrificial shield has not been specifically designed for differential pressure or direct jet impingement.

The sacrificial shield consists of a series of vertical columns spaced around the reactor vessel with 1/4 inch thick steel plates welded to the column flanges to form two concentric cylinders which are tied together by the column webs. Concrete for shielding was placed in the space between the cylinders. The total wall thickness is essentially equal to the depth of the column sections.

Only the steel portions of the shield have been used for structural and strength purposes. The concrete is considered only for shielding purposes and has been conservatively neglected insofar as the structural design of the wall is concerned. The wall has also been analyzed and designed to withstand the loadings associated with the 0.11 g and 0.22 g ground acceleration earthquakes in addition to and in combination with the jet forces of the reactor vessel.

3.8.3.4 Structural Acceptance Criteria

a. Reactor Pedestal

Information relevant to acceptance criteria for the reinforced concrete pedestal is given in Table 3.8-4.

b. Floor Framing

Information relevant to acceptance criteria for the structural steel framing system is given in Table 3.8-5.

3.8.4 Other Seismic Category I Structures

3.8.4.1 Description of the Structures

A key of the main power plant complex is shown on Figure 3.8-30. General arrangement plans and sections of the various structures are shown in Section 1.2.

The two classes of structures for which earthquake design requirements apply are as follows:

Class I - Structures and equipment whose failure could cause significant release of radioactivity or which are vital to a proper shutdown of the plant and the removal of decay heat.

Class II - Structures and equipment which are both essential and nonessential to the operation of the station, but which are not essential to a proper shutdown.

Class I - Critical Structures (Category I)

a. Reactor Building
b. Control Room (and supporting part of Turbine Building)
c. Spend Fuel Pool
d. Ventilation Stack
e. Emergency Diesel Generator Building

Class II Structures

f. Turbine Building
g. Radioactive Waste Building (Original)
h. Service Building
i. Office Building
j. Screenhouse Superstructures
k. Intake and Discharge Structures
l. Radwaste Building (New)
m. Chlorination Building
n. Pretreatment Building
o. Miscellaneous Yard Equipment
p. Heating Boiler House
q. Machine Shop and Storage Building
r. Site Emergency Building
s. Exhaust Tunnel & Fan Foundation

The structures are described in the following paragraphs. These structures incorporate no unique or new design or construction features.

Masonry walls that are in the proximity to or have attachments from safety related piping or equipment such that wall failure could affect a safety related system have been identified in accordance with IE Bulletin No. 80-11. The masonry wall evaluation description and program schedule is given in Appendix 3.8A.

3.8.4.1.1 Reactor Building

The Reactor Building is constructed entirely of reinforced concrete to the refueling floor level at El. 119'-3". Above the refueling floor, the structure is steel framework with insulated, corrosion resistant metal siding.

The foundation mat is 146 feet square by about 10 feet thick, with the finished top surface at El. (-)19'-6" (42 feet 6 inches below grade). The drywell support pedestal in the center of the mat is a concrete cylinder about 67 feet in diameter and 19 feet 5 inches high. The torus is supported from the mat by structural framework.

The Reactor Building is a square structure up to El. 23'-6" and rectangular for the remainder of the height having a grid of walls and columns. The drywell containment vessel, which is shaped as an inverted light bulb, is surrounded by a heavy concrete shield wall which essentially follows the contour of the vessel from the foundation of the drywell up until it intersects the fuel pools, where it then continues to the operating floor (El. 119'-3") as a vertical cylinder. The drywell shield wall is an integral part of the building structure.

As part of the drywell corrosion mitigation project (Section 3.8.2.8), the sand bed area of the Oyster Creek drywell vessel was modified during the 14R outage. The sand was completely removed, the vessel was cleaned of corrosion product and coated with two coats of epoxy paint. In order to accomplish this work, ten twenty-inch round access holes were cut in the concrete shield wall. At the end of the outage the access holes were filled with sand bags and boron bags to provide a barrier to the high radiation area that exists in the sand bed during operation. Only one stack of boron bags was used and is located near the end of the hole closest to the torus room.

The upper part of the building does not extend to the west wall of the basement; thus the drywell is closer to the west wall of the upper structure than it is to the other walls. The refueling floor at El. 119'-3" is served by a traveling crane for handling heavy equipment and spent fuel casks. Fuel rods are handled by special equipment. For refueling, the concrete plugs over the top of the drywell and the steel drywell cover and reactor vessel head are stored on the refueling floor. This floor, and the entire structure, are designed to bear these exceptionally heavy loads. Spent fuel can be loaded into casks at the refueling floor, then lowered through hatches to a railroad car at grade level.

The building as a whole is airtight and is kept at a slight negative pressure so that any leakage is into the building. All air is exhausted to the stack. The building can be entered through six personnel air locks; two at grade, two to the Turbine Building basement, and two to the Office Building at upper floors. The railroad siding also enters through an air lock.

General arrangement drawings of the Reactor Building are shown in Drawings 3E-153-02-001 through 3E-153-02-003.

All piping from the Reactor vessel to the Turbine between the Reactor Building and the Turbine Building runs through a tunnel above grade.

The drywell has a combined personnel and equipment air lock on the ground floor at El. 23'-6". Rolling concrete shield doors in front of the equipment air lock prevent the escape of radiation. The Reactor Building has two major stairways and an elevator. The southeast stairs extend from the basement to the elevator machinery room above the refueling floor, and are enclosed in a stairwell with fire doors at each level. The other stairways are open.

Access to the entire Reactor Building is controlled, but most areas can be entered during normal operation. Exceptions are the drywell, and the following shielded areas with labyrinth entrances:

a. In-core monitor equipment at El. 33'-3".

b. Shutdown heat exchangers at El. 51'-3" and El. 38'-0".

c. Cleanup system components at El. 51'-3" and El. 75'-3".

A system has been devised to isolate the corner rooms located at the lowest level of the Reactor Building. The system isolates each corner room from the remainder of the Reactor Building at that level.

In general, the isolation entails:

a. Providing water tight doors for the corner rooms.

b. Sealing pump suction piping penetrations through the corner room walls.

c. Installing a system of floor drains with check valves that would prevent backflow of water from one closed room through the floor drains into another room.

Two of the corner rooms have pump suction pipes running through the lower area of the entrance doorways. These doorways are sealed by encircling the suction pipe in a steel plate that extends across the doorway and by water tight, bulkhead doors above these steelplates. The doors are designed for a head of 20 feet. For simplicity in design and procurement, the remaining two corner room doorways are similarly sealed with the steel plate, water tight door system outlined above.

The sealed piping penetrations are the pump suction lines: two run through the aforementioned corner room doorways and the other two penetrate through the corner room walls. The penetration seal in all cases will be of 1/8 inch steel fabrication with a flexible "U" joint in the horizontal direction to allow for expansion. The "U" joint is welded on one end to a ring attached to the suction piping and on the other end to a sleeve through the wall or door plate. The penetrations leaving the corner rooms through their outer walls are already sealed as part of secondary containment requirements.

Check valves are provided in the sumps and floor drains of both the corner rooms and the inner torus room the prevent backflow into any of these rooms.

3.8.4.1.2 Control Room (and Supporting Part of Turbine Building)

The Control Room is located on the northeast corner of the Turbine Building Operating Floor at El. 46'-6". The Cable Rooms serving the controls are located immediately above and below the Control Room at El. 36'-0" and El. 63'-9". The mechanical equipment (HVAC) for the Control Room is housed immediately above the Control Room at elevation 63'-9", in the Upper Cable Room.

The rooms are supported by heavy reinforced concrete members which make up part of the Turbine Building structure. Walls for the rooms are reinforced concrete not less than 18 inches thick. The roof of the Upper Cable Room is reinforced concrete. An overhead crane supported on structural steel framework traverses the roof above this room.

General arrangement drawings of the Control Room are shown on the Turbine Building drawings 3E-151-02-006 through 3E-151-02-008.

3.8.4.1.3 Spent Fuel Pool

The Spent Fuel Pool is a rectangular tank like structure, lined with a stainless steel sheet and constructed of massive reinforced concrete walls and bottom slab. The pool liner is ¼" thick on the bottom of the pool, the north wall, and at the opening to the reactor cavity. The liner is a nominal 1/8" thick on the remaining walls. The pool is located immediately north of the drywell with its top being at the refueling floor at El. 119'-3". The massive walls are build integral with the drywell concrete walls. The plan dimensions of the pool are normally 39 feet in the east-west direction and 27 feet in the north-south direction. The bottom of the pool is at El. 80'-6" making the pool 38 feet 9 inches deep. In addition to being attached to the drywell the pool structure is built monolythic with the remainder of the Reactor Building concrete structure and is supported above the foundation mat, at El. (-) 19'-6", by concrete columns.

General arrangement drawings of the Spent Fuel Pool are shown on Drawings 3E-153-02-006 through 3E-153-02-008.

3.8.4.1.4 Ventilation Stack

The 394 foot reinforced concrete stack (368 feet above grade) is linked by tunnels to the Reactor Building, Turbine Building, and both Radwaste Buildings. These tunnels contain piping and air ducts between the buildings and the stack. The off-gas piping is buried; it is not run through the tunnels.

The top of the stack foundation mat is at El.(-)3'-0". Floors at El. 23'-6" and El. 35'-0" are reached through the heating boiler house next to the stack. Off-gas piping enters the stack at El. 0'-0", and penetrates both floors. Absolute filters in the larger of the two pipes are housed below grade in a controlled area accessible only through the tunnel. The differential pressure indicator and isolating valve are located above the floor at El. 23'-6". Exhaust fans for the Reactor Building ventilating ducts are located outdoors at grade level and discharge to the stack above the second floor level. The tunnels are accessible through a stairway to a concrete block enclosed entrance at grade level directly west of the stack.

3.8.4.1.5 Emergency Diesel Generator Building

The Emergency Diesel Generator and Diesel Oil Storage Tank Building is located southwest of the Turbine and Pretreatment Buildings, at the bend in the plant road and east of the discharge canal.

The building is constructed of reinforced concrete with a 16 foot 0 inch by 16 foot 9 inch appendage for the diesel oil storage tank and two 17 foot 6 inch by 72 foot 6 inch compartments for housing the diesel generators. The walls are 18 inches thick with additional 4 inch by 4 inch welded wire fabric reinforcement of the inside face of all exterior walls.

The foundation slab for the diesel generator compartments is 24 inches thick with the top coinciding with the finished grade at El. 23'-0". The tank compartment foundation slab is 24 inches thick with the top located at El. 18' 4", 4 feet 8 inches below finished grade. Sumps and drain lines are included in the foundation slab for drainage purposes. Conduit exits through the north end of the diesel generator foundation slab into buried conduit banks.

Personnel entrance ways are provided in the south end of the building for all compartments and additionally through the east walls for the diesel generator compartments. A 1 foot thick labyrinth wall is provided at all exterior door openings. The entrance to the tank compartment is raised to El. 25'-6", providing a 7 foot 2 inch deep reservoir. An 18 inch concrete wall is provided between the oil tank area and the adjacent diesel generator compartment. An 8 inch by 8 inch square opening has been supplied in the above wall at El. 33'-0" for electrical conduits.

The two entrances to the emergency diesel generator building are at elevation 23 ft. MSL, which is 6 inches below the flooding level which would be caused by local Probable Maximum Precipitation (PMP) at elevation 23.5 ft. MSL (Reference 54). A 6 inch high asphalt dike is provided at these entrances mitigate the entry of surface water into the emergency diesel generator building during a PMP. Flowing water will still have the ability to impinge the dikes and some overflow of water into the EDG is expected. In this case, the building is designed with an internal drain system to port water into the discharge canal.

The walls of the building extend 14 feet above grade to El. 37'-0". The roof of the building compartments are provided with 12 inch thick removable precast concrete sections which are surfaced with paving material for weather protection. A 6 foot 6 inch diameter opening is provided in the roof of each diesel generator compartment for an exhaust stack. A 10 foot wide by 16 foot long opening, provided with heavy duty, 6 inch deep grating, is located 6 feet 6 inches in from the north and south end of the engine compartment for intake and exhaust purposes.

All exposed structural steel is galvanized.

Switchgear is housed in an outdoor, walk in metal enclosure located just north of the generators.

General arrangement of the Emergency Diesel Generator Building is shown on Drawing 3E-157-02-001.

3.8.4.1.6 Turbine Building

The Turbine Building is a reinforced concrete structure directly to the west of the Reactor Building. The building is about 265 feet long and 171 feet wide. The foundation mat is 6 feet to 8 feet thick and the finished top is at El. 0'-0". The Turbine Building foundation mat overlaps the Reactor Building mat where the two buildings abut. Concrete walls extend from the basement level to the operating floor at El. 46'-6" (about 23 feet above grade). Steel framework and insulated metal siding are used over the turbine generator area. Much of the operating floor is not enclosed. The

area to the west contains ventilating fans. The area to the east is partly taken up by the Control Room and Cable Rooms. The area over the three feedwater heaters on the mezzanine floor is served by an outdoor traveling crane with access through hatches to the heaters. The area over the Turbine Generator is served by a traveling crane just beneath the roof. Equipment may be raised and lowered to the floor at grade level through an equipment hatch to the west of the generator.

At the basement level, heavy concrete walls with labyrinth entrances shield the Main Condensers, reheaters, and moisture separators. Six 72 inch circulating water pipes to the three condensers penetrate the west wall. Circulating water enters and discharges via reinforced concrete tunnels below the foundation mat. Shield walls and labyrinth entrances are also provided for the Steam Jet Air Ejectors and Steam Packing Exhausters, the condenser vacuum pump, and the regeneration system waste tanks. Condensate pumps, feedwater pumps, and local instrument racks are located in a separate room to the east of the main condensers. This area is shielded and it can usually be entered when the plant is in operation. Access to the remainder of the floor is unrestricted.

The mezzanine floor surrounds the Main Condenser area at several levels. The Intermediate Pressure and High Pressure feedwater heaters at El. 23'-6" are located on an open gallery above the pump room, accessible by open stairs from the basement floor. This area is not entered when the plant is operating. The hallway on the north side has unrestricted access. The south side of the mezzanine floor at El. 23'-6" houses the Condensate Pre-Filter System, 4160 volt switchgear and a railroad siding. The condensate polishers on this floor are surrounded by heavy concrete shield walls with a labyrinth entrance. An unrestricted valve space contains the polishers control panel and all manual valves for polishers control.

Access to the enclosed portion of the operating floor is controlled, although the area can be entered during normal operation. Since radiation levels decrease with distance from the high pressure turbine, the main stairs and the elevator are located at the south end of the building. The stairs and elevator extend from the basement to the operating floor, and the elevator has a door to the machine shop at the mezzanine level. Interior stairs, at Columns 3-4 (east of Column Line F) and at the north side of the building, lead to the Control Room and Office Building. There are exterior doors at the north and south sides of the building, and a double door to the machine shop on the south side.

General Arrangement drawings of the Turbine Building are shown in Drawings 3E-151-02-001 through 3E-151-02-009.

3.8.4.1.7 <u>Radioactive Waste Building</u> (Original)

This structure and facility have been replaced by a new facility, which is described in Subsection 3.8.4.1.8. However, some of the equipment is used to support radwaste processing operations at the new facility. The following description is given to provide general interface information in conjunction with any reviews made relative to the original plant.

The Old Radwaste Building is directly east of the Reactor Building, and is linked to the Reactor, Turbine and New Radwaste Buildings by underground pipe tunnels. It is a single story reinforced concrete building surmounted by a two story penthouse. A smaller basement area at El. 6'-6" contains equipment and small pumps.

Most of the building is inaccessible during operation, and much of the equipment is contained in concrete cells accessible only through concrete roof hatches. The main floor, at or near grade, contains a control room, a filter aid and precoat room, a change room operating gallery, drum transfer aisle, drum storage room, compactor area, and north personnel access building. A change room on this floor is the only entrance to the accessible controlled areas (two pump rooms, a part of the basement area, the operating gallery for the drum capping machine, and the penthouse floor at El. 45'-6"). The rest of the first floor area is taken up by inaccessible tank and equipment cells. The latter areas are accessible for limited periods following a check of radiation levels. The penthouse contains space for two waste centrifuges and a cement chute for filling the concrete mixer on the level below.

The interior is maintained at a negative pressure, and all ventilating air is exhausted to the stack through a separate underground tunnel. Exhausted air is filtered before discharge. The filters are located in a room at the basement level (El. 6'-6") which is accessible only from the tunnel.

The roof at El. 39'-3" is used as a laydown area for hatch covers. Permissible live load is 400 psf, which means that covers must be stacked in a single layer. The higher roof (El. 49'-10" to El. 50'-6") has a permissible live load of 30 psf. Hatches or equipment must not be placed on this roof.

During 1985, GPUN accomplished the removal of drums of filter media type radwaste and abandoned drum capping and handling equipment from the old Radwaste Building. Decontamination and recovery of the drum storage area of this building was accomplished by January, 1986. This area is now used for the storage of non-waste radioactive material.

3.8.4.1.8 Radwaste Building (New)

Figure 3.8-37 shows the structural configuration of the New Radwaste Building with the seismic Category I elements identified.

The New Radwaste Building houses the facilities for solid and liquid radwaste processing. The basic functions of the building are to provide radiation protection during operating conditions and to ensure no leakage of radioactive materials to the surroundings during extreme environmental conditions.

Following is a physical description of the building:

The building is rectangular in plan. It has three main floors: grade, intermediate and operating. A large door opening for truck access is provided at grade level in the east wall. The door is designed to provide for conventional weather protection. A concrete curb is provided to retain any spillage inside the building. Reinforced concrete walls are provided to the operating level and above this level where liquid retention is required. The remaining wall area is insulated metal siding or of solid concrete block construction. The roof area is covered by insulated metal deck and roofing and by concrete slabs where radiation shielding is required. Interior shield walls of concrete block are provided for protection of operating and maintenance personnel.

3.8.4.1.9 Masonary Walls

On May 8, 1980 the NRC issued Inspection and Enforcement (IE) Bulletin 80-11, Masonary Wall Design, which required reevaluation of the design adequacy of safety related masonary walls under postulated loads, including seismic load.

There are safety related walls at the Oyster Creek Nuclear Generating Station (OCNGS). These walls are typically single width, except for one 24 inch thick shield wall in the Reactor Building. Unreinforced and reinforced walls are found at the plant. Both stacked and running bond construction are encountered.

The masonary wall functions and materials for OCNGS are as follows:

Wall Functions

Shielding, fire barrier, security, personnel partition

Construction Materials

Hollow Concrete Unit	
Non load bearing	C-129
Load bearing	C-90
Mortar	C-270, Type M
Bar reinforcement	A-15, intermedite grade, deformed bars per ASTM 305 (Grade 40).
Horizontal joint reinforcement	Extra heavy Dur-O-Wal.

Following reanalyses and modification, NRC concluded that there is reasonable assurance that the safety related walls at OCNGS will withstand the specified design load conditions without impairment of: a) wall integrity, or b) the performance of the associated safety systems.

3.8.4.2 Applicable Codes, Standards and Specifications

The design, materials, fabrication, construction and inspection of other seismic Category I structures conform to, but are not necessarily limited to, the applicable sections of the listed codes and specifications which are used to establish or implement design bases and methods, analytical techniques, material properties, construction techniques and quality control provisions.

The design of the Oyster Creek Nuclear Generating Station evolved over a number of years starting shortly after the letter of intent was issued in December 1963 to the General Electric Company as the prime contractor to provide Jersey Central Power and Light Company with a complete operating plant. The dates and revisions for the codes listed would have been those in effect during the different stages of design and construction. Other tests and standards identified by the lead documents listed and in effect or promulgated at the time the design or construction was performed, shall also be considered as viable controlling documents.

The following chronology is provided to establish the applicable time frame in which the design and construction was performed.

Milestone Event	Date
Design and Construct Letter of Intent	December, 1963
Preliminary Safety Analysis Report Filed	March, 1964
Site Preparations Started	September, 1964
Construction Permit Effective	January 15, 1965
First Concrete Placed	February, 1965
Reactor Building Completed	November, 1967
Fuel Loading Started	April, 1969
Placed in Commercial Operation	December, 1969

The New Radwaste Building (Subsection 3.8.4.1.8) was completed to support the scheduled initial operation of the Augmented Offgas System in March 1977 and the scheduled operation of the modified Liquid and Solid Radioactive Waste Treatment Systems in February 1978.

All construction and fabrication contracts were awarded on the basis that all work be conducted in accordance with the latest issue of the applicable publication as of the contract date. The following contracts, contract date and scheduled completion is provided to define a more specific time frame for code issue date reference purposes.

Burns & Roe Specification No.	Contract Title	Contract Date	Contract Completion
S-2299-15	Production and Delivery of Concrete	9/64	**
S-2299-17	Reactor Building Foundations	10/64	3/65
S-2299-33	Turbine Building Foundations and Circulating Water System Structures	4/65	3/66
S-2299-34	Structural Steel	6/65	10/66
S-2299-45	Superstructure Construction	3/66	12/66

** Concrete supplied throughout the construction schedule.

Unless otherwise identified for a specific structure(s), the following codes, standards and specifications are applicable to all the structures identified in Subsection 3.8.4.1.

American Concrete Institute Publications

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 613	Recommended Practice for Selecting Proportions for Concrete
ACI SP-4	Formwork for Concrete

American Society for Testing and Materials Standards

A15	Specification for Billet-Steel Bars for Concrete Reinforcement
A36	Structural Steel
A53	Welded and Seamless Steel Pipe
A123	Zinc (Hot-Galvanized) Coatings on Products Fabricated from Rolled, Pressed and Forged Steel Shapes, Plates, Bars and Strip
A141	Structural Rivet Steel
A185	Specification for Welded Steel Wire Fabric for Concrete Reinforcement
A305	Specification for Minimum Requirements for the Deformation of Deformed Steel Bars for Concrete Reinforcement
A307	Low Carbon Steel Externally and Internally Threaded Standard Fasteners
A325	High Strength Steel Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers
A408	Specification for Special Large Size Deformed Billet-Steel Bars for Concrete Reinforcement
B43	Seamless Red Brass Pipe, Standard Sizes
B75	Seamless Copper Tube
C31	Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field
C39	Method of Test for Compressive Strength of Molded Concrete Cylinders
C40	Method of Test for Organic Impurities in Sands for Concrete

C42	Methods of Securing, Preparing and Testing Specimens from Hardened Concrete for Compressive and Flexural Strengths
C87	Method of Test for Measuring Mortar-Making Properties of Fine Aggregate
C88	Method of Test for Soundness of Aggregate by Use of Sodium Sulfate or Magnesium Sulfate
C94	Ready Mixed Concrete
C117	Method of Test for Materials Finer than No. 200 Sieve in Mineral Aggregates by Washing
C123	Method of Test for Lightweight Pieces in Aggregate
C127	Method of Test for Specific Gravity and Absorption of Coarse Aggregate
C128	Method of Test for Specific Gravity and Absorption of Fine Aggregate
C131	Method of Test for Abrasion of Coarse Aggregate by Use of the Los Angeles Machine
C125	Definition of Terms Relating to Concrete and Concrete Aggregates
C136	Method of Test for Sieve or Screen Analysis of Fine and Coarse Aggregates
C138	Weight per Cubic Foot, Yield and Air Content (Gravimetric) of Concrete
C142	Method of Test for Clay Lumps in Natural Aggregates
C143	Method of Test for Slump of Portland Cement Concrete
C156	Water Retention Efficiency of Liquid Membrane Forming Compounds and Impermeable Sheet Materials for Curing Concrete
C150	Specification for Portland Cement
C172	Method of Sampling Fresh Concrete
C173	Air Content of Freshly Mixed Concrete by the Volumetric Method
C192	Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Laboratory
C227	Method of Test for Potential Alkali Reactivity of Cement Aggregate Combinations (Mortar Bar Method)
C231	Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method
C235	Scratch Hardness of Coarse Aggregate Particles

C260	Specification for Air Entrainment Admixtures in Concrete
C295	Recommended Practice for Petrographic Examination of Aggregates for Concrete
C309	Liquid Membrane Forming Compounds for Curing Concrete
C360	Ball Penetration in Fresh Portland Cement Concrete
C440	Cotton Mats for Curing Concrete
C494	Chemical Admixtures for Concrete
D43	Creosote for Priming Coat with Coal-Tar Pitch in Dampproofing and Waterproofing
D88	Test for Viscosity, by Means of the Saybolt Viscosimeter
C92	Test for Flash and Fire Points by means of Cleveland Open Cup
C155	Test for Color by Lubricating Oil and Petrolatum by Means of ASTM Union Colorimeter
D173	Woven Cotton Fabrics Saturated with Bituminous Substances for Use in Waterproofing
D390	Creosote
D412	Method of Tension Testing of Vulcanized Rubber
D450	Coal Tar Pitch for Roofing, Dampproofing and Waterproofing
D735	Elastomer Compounds for Automotive Applications
D746	Method of Test for Brittleness Temperature of Plastics and Elastomers by Impact
D882	Tensile Properties of Thin Plastic Sheets and Films
D1760	Pressure Treatment of Timber Products
E8	Methods of Tension Testing of Metallic Materials
E96	Measuring Water Vapor Transmission of Materials in Sheet Form

American Institute of Steel Construction Publications:

Code of Standard Practice for Steel Buildings and Bridges.

Specifications for Structural Joints Using ASTM A325 Bolts.

Specification for the Design, Fabrication and Erection of Structural Steel for Buildings

American Association of State Highway Officials Specifications

T-26	Water for Use in Concrete

American Welding Society

D1.0	Standard Code for Arc and Gas Welding in Building Construction
D12.1	Recommended Practice for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction

Concrete Reinforcing Steel Institute

CRSI	A Manual of Standard Practice for Reinforced Concrete Construction

Federal Specifications

AAA-S-121b	Scales, Weighing; General Specifications & Am-1
CCC-C-467	Cloth, Jute (or Kenof), Burlap & Am-1
HH-C-581a	Cotton Fabric; Woven, Asphalt-Saturated
HH-F-141a	Felt; Asphalt-Saturated (for) Flashings, Roofing, and Waterproofing
HH-F-191	Felt, Asphalt Saturated (for) Flashings, Roofing and Waterproofing
HH-F-201	Felt, Coal-Tar Saturated (for) Roofing and Waterproofing
HH-F-341-a	Filler, Expansion Joint, Preformed, Nonextruding and Resilient Types (for Concrete)
LLL-F-321B	Fiberboard; Insulating & Am-1
LLL-H-35	Hardboard, Fibrous-Felted (Fiberboard)
LLL-I-535(1)	Insulated Board, Thermal and Insulation Block, Thermal
L-S-137	Screening, Plastic Coated Fibrous Glass, Insect & Am-1
PPP-T-60	Tape, Pressure Sensitive Adhesive, Waterproof for Packaging & Am-2 and Sealing

QQ-C-576b	Copper Flat Products with Slit, Slit and Edge Rolled, Sheared, Sawed or Machined Edges, (Plate, Bar, Sheet and Strip)
RR-G-661a	Gratings, Steel, Floor
SS-A-666	Asphalt (for) Builtup Roofing, Waterproofing, and Dampproofing
SS-A-701	Asphalt Primer; (for) Roofing and and Waterproofing
SS-S-164	Sealer; Hot Poured Type, for Joints in Concrete & Am-1
TT-C-598	Compound, Caulking; Plastic (for Masonry and Other Structures) & AM-2
TT-E-489c & AM-2	Enamel, Alkyd, Gloss (for Exterior and Interior Surfaces)
TT-E-506c	Enamel, Tints and White, Gloss, Interior
TT-E-508	Enamel, Interior, Semigloss, Tints and White & AM-4
TM-E-543	Enamel Undercoat, Interior, Tints and White
TT-P-29b	Paint, Latex Base, Interior, Flat White and Tints
TT-P-30b	Paint, Alkyd, Odorless, Interior, Flat, White and Tints
TT-P-56b	Primer Coating (Primer Sealer), Pigmented Oil, Plaster and Wallboard
TT-P-86c	Paint, Red Lead Base, Ready Mixed
TT-P-641b	Primer, Paint; Zinc Dust Zinc Oxide (for Galvanized Surfaces)
TT-S-00227	Sealing Compound; Rubber Base, Two Component
UU-P-264a	Paper, Concrete Curing, Waterproofed (Kraft)
W-P-404c	Pipe, Steel (Seamless and Welded, Black and Zinc Coated) (Galvanized)
WW-P-406b	Pipe, Steel (Seamless and Welded) (for Ordinary Use)

Federal Standard

No. 158.a	Cements, Hydraulic; Sampling, Inspection and Testing

Military Specifications

MIL-S-12935B	Sealer, Surface, Knot
MIL-C-15328B	Coating, Pretreatment (Formula No. 117 for Metals)

Other Codes and Specifications

UBC-1964	Uniform Building Code

Codes and Specifications Used for the Ventilation Stack

ACI-505-54	American Concrete Institute Specification for the Design of Reinforced Concrete Chimneys
ACI-318-63	Building Code Requirement for Reinforced Concrete
ASCE Paper No. 3269	Wind Forces on Structures, Final Report of the Task Committee on Wind Forces

Codes and Specifications Used for the New Radwaste Building

ACI 318-71	Building Code Requirement for Reinforced Concrete
ACI 301-72	Specifications for Structural Concrete for Buildings
ACI 306	Recommended Practice for Cold Weather Concreting
ACI 311	Manual of Concrete Inspection
ACI 315	Manual of Standard Practice for Detailing Reinforced Concrete Structures
ACI 347	Recommended Practice for Concrete Formwork
ACI 605	Recommended Practice for Hot Weather Concreting
ACI 614	Recommended Practice for Measuring, Mixing and Placing Concrete
ACI 211	Recommended Practice for Selecting Proportions for Concrete
ACI 214	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
AISC	Specification for the Design, Fabrication and Erection of Structural Steel Buildings
AISC	Code of Standard Practice
AISC	Specification for Structural Joints Using ASTM A325 or A490 Bolts
AWS	Code for Welding in Buildings D1.1
ASME	Boiler and Pressure Vessel Code, Section VIII, Pressure Vessel, Division I
ASCE Paper No. 3269	Task Committee Report "Wind Forces on Structures"

UBC-1973	Uniform Building Code of the International Conference of Building Officials
BOCA	The BOCA Basic Building Code

3.8.4.3 Loads and Load Combinations

The seismic Category I structures discussed herein were designed, as applicable, for all credible conditions of loadings, including normal loads, loads due to severe and extreme environmental conditions, abnormal loads and missiles.

The design, subsequent design reviews and the addition of new structures has occurred over the design and operating life of the plant. This has resulted in some changes and/or additions to load definitions and load combinations as a consequence of reassessing the original designs or designing new structures to later design criteria.

The following Design Load Definitions have been listed to cover all loads used in the applicable design or review.

Load combinations are first listed as generally applicable to all structures in accordance with the original design of the plant. Where subsequent reviews or new structures have been performed to different loads or load combinations, a separate list of load combinations is provided.

3.8.4.3.1 Design Loads

The definition of the loads used in the design of the other seismic Category I structures include the following:

a. Normal Startup, Operational and Shutdown Loads

Loads encountered during the normal plant operation are:

1. Dead Loads

Dead loads includes the weight of the structural components and architectural appurtenances. Includes all permanent loads.

2. Live Loads

Live loads include loads which can be expected throughout the reactor structure, i.e., roof and crane loads, movable equipment, the shipping cask, stored supplies, piping, cable trays, etc.

3. Operating Loads

   Operating loads consists of gravity loads from all equipment and piping, and for the weight of water over the reactor during refueling and in the storage pools. Includes the restraint to thermal movement of the structure.

4. Thermal Loads

   Effects and loads during normal operating or shutdown conditions.

5. Earth Pressures

   Lateral earth pressure loads acting along the embedded depth of the structure including lateral loads induced by above grade surcharge.

6. Hydrostatic Pressures

   Pressure loads from liquids (from ground water)

b. Severe Environmental Loads

Loads encountered infrequently during the plant life are:

1. Design Earthquake (OBE)

   These are the loads generated by the Operating Basis Earthquake, which is the earthquake of an intensity enveloping the maximum anticipated for the site.

2. Wind Load (W)

   Wind loads generated by the design wind defined for the plant. See Section 3.3.

c. Extreme Environmental Loads

Extreme environmental loads are those loads which result from postulated events which are credible, but highly improbable. These are:

1. Safe Shutdown Earthquake (SSE)

   To assure that the plant can be shut down with containment and heat removal facilities intact, such principal structures are designed to accommodate a ground acceleration equal to twice the design earthquake.

2. Tornado Loads

   Wind loads generated by the design tornado specified for the plant site. See Section 3.3.

3. Tornado Depressurization Loads

   Differential pressure loads due to a tornado oriented rapid atmospheric pressure drop. Section 3.3.

4. Tornado Generated Missiles

   The local and overall effects of tornado generated missiles are considered.

d. Specific Design Floor Loadings

   Permissible floor loadings used in the design of the Turbine and Reactor Buildings are given in Table 3.8-6.

3.8.4.3.2 Load Combinations

Various load combinations were considered in the design to determine the strength requirements of the structure.

a. Normal Load Conditions are those encountered during testing and normal operation and are referred to in the Standard Review Plan as service load conditions. They include dead load, live load and loads occurring during startup and shutdown. Normal loading also includes the effect of an Operational Basis Earthquake and normal wind load.

b. Unusual Load Conditions are those resulting from combinations of accident, wind, tornado, safe shutdown earthquake and live and dead loads, and are referred to in the Standard Review Plan as factored load conditions.

Original Plant Design Load Combinations

Load combinations for concrete and steel structures are given in Table 3.8-7 and 3.8-8, respectively.

Tornado Wind Load Combinations

Load combinations used in responses to questions raised during the integrated plant safety assessment under the Systematic Evaluation Program (SEP) are given in Table 3.8-9. The program is summarized in Section 1.10.

New Radwaste Building Load Combinations

Load combinations used in the design of this structure are given in Table 3.8-10 and nomenclature definitions are given in Table 3.8-11.

Ventilation Stack

Load combinations used in the design of the ventilation stack are given in Table 3.8-12.

3.8.4.4 Design and Analysis Procedure

3.8.4.4.1 Design and Analysis

Category I structures other than the containment are generally constructed of reinforced concrete with some structural steel framing used in conjunction with enclosing the upper portions of certain structures. More description of these structures is given in Subsection 3.8.4.1. The design and analysis of these structures was performed utilizing classical approaches consistent with the applicable codes and standards. The solution to mathematical equations of compatibility for indeterminate structural systems were essentially derived by the Moment Distribution Method with consideration for lateral translations. Unless specifically noted, no computer programs were utilized in the original analysis or design of the basic Category I structures. Statically determinate systems were analyzed by standard equations relating the physical behavior of the element under design.

Concrete structures have been designed as integral structures utilizing floor slabs as diaphragms to transfer lateral forces to the walls. The walls2 are utilized to transfer the lateral forces to the foundation mats. The structures have been divided into appropriate structural subsystems for ease of analysis. The behavior of the basic structure has been maintained in the subsystem analysis by the use of relative boundary conditions which define appropriate rotation, translation and deformation characteristics. The typical approach was to analyze and design systems of bents comprised of beams and columns acting along a given plane through the structure. Slabs are generally designed as two way systems, utilizing standard ACI design provisions for determining the distribution of forces. The stiffness contribution of the Drywell Shield Wall was taken into account in distributing and resisting forces and transferring lateral shears and vertical forces to the foundation mat.

The walls below grade are designed for inplane seismic shears and lateral earth and ground water pressures. The continuity of the below grade wall system and the inter-related behavior of the coupled internal framing system was incorporated in the analysis and design. The influence of various stages of construction was accounted for in the below grade walls as it related to changing lateral wall support and backfill conditions.

The structural steel framing above the operating floors of the Reactor and Turbine Buildings are designed to standard AISC requirements. These framing systems are designed and constructed essentially as rigid bents with respect to lateral forces and as braced framing to provide structural capabilities in the longitudinal direction. The end walls of the Reactor Building are also designed as braced systems to resist lateral forces (References 28, 29, 30, 31). The end walls of the Turbine Building are of rigid frame construction.

The design, subsequent design reviews and the addition of new structures has resulted in some procedures being defined specifically with respect to given structures. This information is provided for structures so identified.

a. Ventilation Stack

See Subsection 3.8.4.5 for discussions and results of design and analysis work for the ventilation stack.

b. New Radwaste Building

The building is constructed on a foundation mat at grade resting on compacted backfill. Steel framing and metal decking are provided for support of the reinforced concrete floor slabs. Only those floor slabs within the cubicle housing the concentrated liquid waste tanks are considered to be seismic Category I. All other framing and floor slabs are conventionally designed. Exterior walls and other interior walls required for retention of spilled liquids are constructed of reinforced concrete and are treated as seismic Category I elements.

Since the building is composed of a combination of seismic Category I and non seismic elements, both the failure and non failure of non seismic elements has been considered in design to determine the controlling case. Lateral loads due to wind, tornado, and earthquake are transferred to the foundation mat through the stiff reinforced concrete walls. Distribution of these lateral loads takes into account the flexural and torsional rigidities of the walls. Although not designed for these lateral loads, the floor slabs are considered adequate to effect the load transfer. The foundation mat is designed as a seismic Category I structure and its analysis takes into account the relative flexibility of the mat and the supporting soil.

Design of structural elements, except as modified herein, is based on the requirements of ACI 318 and AISC reference codes listed in Subsection 3.8.4.2.

3.8.4.5 Structural Acceptance Criteria

The basic codes utilized for structural acceptance design criteria are ACI 318 and AISC as applicable to the specific structure.

The earthquake design is based on ordinary allowable stresses as set forth in the applicable codes. A one third increase in allowable working stresses because of earthquake loading is not used. The design is such that a proper shutdown can be made during ground motion having twice the intensity of the spectra shown for the design earthquake (OBE) (Section 3.7).

The foregoing design criteria are for Class I items. Class II items are designed following the normal practice for the design of power plants in the State of New Jersey, but as a minimum, this will not be less than given in the "Uniform Building Code" for Zone 1.

The stresses resulting from the earthquake accelerations shown on the family of curves given in Section 3.7, when combined with functional loading stresses, are within the established code allowable working stresses for the particular materials involved.

The combined stresses resulting from functional loadings and from an earthquake having a ground acceleration of 0.22g are such that a safe shutdown can be achieved. The combined earthquake and functional load stresses probably would not exceed yield stress. However, where calculations indicated that a structure or piece of equipment would be stressed beyond the yield

point, an analysis was made to determine its energy absorption capacity. The capacity is such that it exceeds the energy input from the earthquake. In addition, the design has been reviewed to assure that any resulting deflections or distortions would not prevent the proper functioning of the structure or piece of equipment and would not endanger adjacent structures or components.

Information relevant to the response and implementation program associated with IE Bulletin No. 79-02 and IE Bulletin No. 80-11 are documented in References 32 through 37.

Information relevant to specific structures' acceptance criteria can be obtained from the Tables identified below:

a. Concrete Structures (concrete and reinforcement)

Structure	Reference Table
1. Reactor Building	3.8-7
2. Control Room Slabs and Walls	3.8-7
3. Battery Room Slabs and Walls	3.8-7
4. Emergency Diesel Generator and Tank Vault	3.8-7
5. Intake Structure (Service Water and Circulating Water Pump Area)	3.8-7
6. Startup Transfer Foundation	3.8-7
7. New Radwaste Building	3.8-10
8. Ventilation Stack	3.8-12
9. Drywell Concrete Shield Wall	3.8-13

b. Structural Steel Framing

Structure	Reference Table
1. Reactor Building (above El. 119'-3")	3.8-8
2. Reactor Building Platforms	3.8-8

3.8.4.5.1 Effects of Earthquake on the Reactor Building

During an earthquake having a ground acceleration of 0.11g (i.e., the Oyster Creek design earthquake) cracking of the concrete portion of the Reactor Building is not anticipated. The maximum shear stresses based on the dynamic analysis having a ground acceleration of 0.11g with 10 percent damping and shear distributed by relative stiffness was found to be at the exterior

north and south walls at El. 23'-6". Calculations indicate the maximum concrete shearing stress is 134 psi which is less than the allowable 150 psi and considerably less than the ultimate shearing stress. The maximum shear stresses based on 5 percent damping is 171 psi.

The structural steel frame of the Reactor Building will deflect when subjected to the ground motions due to the design earthquake; however, the structure is designed to accommodate the forces caused by these ground motions. The siding is designed with caulked interlocking vertical joints and overlapping horizontal joints. The horizontal joints are overlapped sufficiently to insure leaktight integrity. To further insure the leak tightness of the siding, all joints are provided with mechanical fasteners to provide positive connections to the structural steel. In the event of a design earthquake causing movement of structural steel frame, some of the siding connections may be distressed; however, it is expected that the mastic joint sealant will provide sufficient resilience to prevent the exfiltration of air.

3.8.4.5.2 New Radwaste Building

Referring to the loading combinations listed in Table 3.8-10, the following defines the allowable limits which constitute the structural acceptance criteria.

	Loading Combination	Limit
a.	Concrete Structures All combinations	U
b.	Steel Structures	Not applicable

Where:

U = The section strength required to resist design loads based on the ultimate strength design methods described in ACI-318-71

3.8.4.5.3 Failure Mode Analysis of the Ventilation Stack

The 368 foot ventilation stack was designed at normal allowable stresses for the forces resulting from an earthquake with a ground acceleration of 0.11g. An investigation of the stack was also made to assure that the stack will not be adversely affected by the application of forces created by an earthquake with a ground acceleration of 0.22g. Therefore, there is more than reasonable assurance that the stack will not topple. Some cracking may be possible under the extreme loading condition. The following discussion has been prepared to summarize failure modes and resultant damage to the plant.

The normal mode of failure for a reinforced concrete stack is cracking with pieces of concrete falling to the ground. The following tabulation indicates the location of the failure regions measured down from the top of the stack, for which some degree of damage to the principal plant structure might occur, assuming the failed section topples as a tree:

Failure Zone	Building(s) Affected
0-65'	None
65'-85'	Reactor Building
85'-210'	Reactor and Old Radwaste Buildings
210' to base	Reactor, Old Radwaste and Turbine Buildings

The standby diesel generator vault is located such that the stack would fall short even if it toppled about its base.

The location of the other major plant structures is such that a failure of the stack could occur, including overturning about the base of the stack, and not strike any of the buildings. For example, the included angle between lines drawn from the center of the stack to the southwest and northeast corners of the Reactor Building is approximately 65 degrees. The included angle between lines drawn from the center of the stack to the southeast corner of the Turbine and Radwaste Buildings is approximately 180 degrees.

In instances where stacks have failed and toppled, it is usually due to repetitive stresses with the upper one third or less of the stack falling. In the case of the Oyster Creek stack, 1/3 of the stack height is approximately 130 feet. Since the horizontal distance from the centerline of the stack to the centerline of the reactor is approximately 160 feet, a failure of the one third point would result in the failed section falling short of the reactor and drywell vessels and the 7 foot thick concrete shield plug located over the drywell vessel.

An analysis has been made to determine the capability of the Reactor Building refueling floor and reactor shield plugs to resist penetration of various sized stack sections. The basis for the analysis was "The Design of Barricades for Hazardous Pressure Systems," KAPL-M-6446 (CVM-24) by C.V. Moore. The results of the analysis are tabulated below indicating a size of stack section which the various structural elements can withstand without allowing penetration:

Section Analyzed	Length and Weight of Stack Section
16" Floor Slab	approximately 20 feet, 34 kips
12" Floor Slab	approximately 15 feet, 20 kips
7'-0" thick Shield Plugs	approximately 40 feet, 86 kips

In each case the stack section was assumed to strike on end as a cylinder and was assumed to fall from the top of the stack.

The equipment necessary for emergency core and containment cooling and other equipment necessary to shut down the reactor, including heat exchangers and pumps, are located in the Reactor Building in compartments located below the refueling floor level. Thus, even such an improbable event as failure of the stack would probably not impair the ability to safely shut down.

Analysis by the above referenced method is conservative in this case for the following reasons:

a. Energy absorbed by crushing of the stack concrete upon impact has not been considered.

b. Energy absorbed by the Reactor Building superstructure roof and wall elements has not been considered.

c. It is highly unlikely that the stack will fall in such a manner that sections of the stack of the size indicated would strike the structure on end as a cylinder. If such an improbable event as stack failure occurred, the sections of the stack which could fall on the building would most likely fall with the axis of the cylinder more nearly parallel to the slabs and plugs. This would, of course, mean that the effective contact area would be much larger than assumed in the analysis.

From the investigations and analyses made, the following conclusions have been reached:

a. Stack failure as a result of the maximum earthquake is highly unlikely since the stack has been designed to withstand the loads from such an earthquake.

b. The physical location of the stack with respect to other plant structures provides some degree of assurance (and in some cases, complete assurance) that even if the stack did fail or topple, it would not strike any of these structures.

c. In the unlikely event that some part of the stack should fall on the Reactor Building, there is reasonable assurance that the ability to safely shut down the reactor would not be impaired.

Confidence in the integrity of the stack is based on the conservatisms in the criteria design and construction.

The design was performed in accordance with the procedures prescribed in the American Concrete Institute Specification for the Design of Reinforced Concrete Chimneys, ACI-505. Design and construction were by a specialty contractor of established reputation in the field. Design wind gust velocity was set at a minimum of 110 mph at the base and increasing with height in accordance with the relation used in ACI-505 whereas the U. S. Weather Bureau records show maximum wind speed of any duration (including gusts) and at any height ever measured on the New Jersey Coast to be 91 mph.

A dynamic analysis of the stack was performed by John A. Blume & Associates, Engineers. The stack was treated as a flexible cantilever fixed at the base. Thirty nine (39) lumped mass points were considered in the analysis. A total of ten (10) modes were considered in the analysis using a damping value of five (5) percent. The analysis produced envelopes of maximum displacement, shears and overturning moments which formed the basis for stack seismic design.

Allowable stresses established by the ACI Specification followed are conservative, 1250 psi for the 5000 psi strength concrete specified and which has been verified by daily tests as having been produced. The ACI Specification contemplates use of 40,000 psi yield strength reinforcing steel for which allowables of 12,500 and 15,000 psi for wind and earthquake, respectively, are specified by ACI; 60,000 psi yield strength steel was used for which allowable stresses were set at 15,000 and 18,000 thus providing greater conservatism than the ACI specification.

The design which satisfied wind and design earthquake requirements (separately) was analyzed for the maximum (double) earthquake and the reinforcing was increased to maintain steel stress below yield under this loading. Concrete stress under double earthquake is relatively low, 2400 psi maximum for 5000 psi concrete. This design was analyzed to determine maximum acceptable

wind velocity as a measure of the safety factor on specified wind inherent in the final design; with stresses at the levels used for double earthquake the chimney can accept a maximum wind gust velocity of 192 mph in the upper 268 feet.

The foundation was designed very conservatively since schedule required its construction before final resolution of the chimney design; stresses are approximately half normal allowable at design loads.

The placing of the foundation on the very dense sands below the finite clay layer separating the geologic formations at the site not only provides a foundation support stratum having a bearing capacity of at least twice the maximum (edge) bearing pressure imposed under double earthquake but also provides a 33 ft depth of embedment of the structure below grade. With the resistance of passive earth pressure provided by this embedment, the factor of safety against overturning is at least 1.5 under double earthquake.

Although the possibility of liquefaction of the very dense sands characteristic of the supporting formation would be expected to be extremely remote, this possibility has been investigated following the procedures recently suggested by Leed. A factor of safety in the order of 4 against liquefaction is indicated.

The maximum overturning moments on the stack due to the maximum earthquake (0.22g) and the maximum wind are 94,000 kip-feet and 68,270 kip-feet, respectively. The stack foundation design is such that a factor of safety of at least 1.5 would exist should the structure be subjected to these overturning moments. Therefore, the conclusion can be drawn that the stack is designed to prevent failure under the most severe loading conditions.

As stated previously, some portions of either the Reactor, Old Radwaste, or Turbine Buildings could be damaged if the stack failed at its base and toppled as a tree. In case of such an unlikely event, the stack could strike the roof of the Reactor Building. As the stack continued its fall, energy would be absorbed while the superstructure roof truss, wall framing and metal panel siding was being crushed. Assuming that the stack is still intact, it would then strike the edge of the Reactor Building formed by the intersection of the wall and refueling floor slab. Energy would be expended by the crushing of the concrete of the stack and the building. Assuming that the stack now fails at the point of contact with the floor to wall edge, the upper 250 feet (approximately) would continue to topple and fall onto the reactor building roof. Before the stack could hit the floor of the building, it would have to crush the remaining roof trusses, the wall framing and wall panels on the west side of the building and the reactor building crane girders and rails. A great deal of energy will obviously be absorbed during this part of the event. Additionally, parts of the roof and wall elements will fall to the floor providing further cushioning at the time the stack impacts on the refueling floor, resulting in a spreading of the load. Thus the contact area will be increased and the chances of the stack penetrating the shielding slabs over the reactor and drywell vessels of the floor slabs decreased. Furthermore, because of the relative locations of the Reactor Building and stack, it is possible that the stack would strike the building a "glancing blow" and slide along either the south or east wall of the Reactor Building, instead of falling across the structure. Therefore, the analysis of the case of a part of the stack striking the building on end as a cylinder represents the more severe condition. Thus there is reasonable assurance that the ability to shutdown the reactor and to maintain it in the shutdown mode would not be impaired even though such an unlikely event as stack failure occurred.

3.8.4.5.4 Acceptance of Class I Structures Under Tornado Loadings

The following discussion corresponds to the original design of the Reactor Building, it should be noted that tornado and wind loading analyses have been performed as part of the SEP, this program is summarized in Section 1.10.

The tornado frequency for Oyster Creek is 2190 years. This means a probability of 1.81 percent for a 40 year life. (Refer to Section 2.3 and 3.3.)

The 100 year wind storm would be a storm with the greatest intensity that can statistically be expected to occur. The wind velocities and wind pressure would be:

Height Zone (ft)	0 to 50	50 to 150
Velocity (mph)	100	125
Building pressure (psf)	40.3	62.8

These building pressures are based on 1.1 gust factor and 1.3 shape factor for a rectangular building. In the analysis, normal allowable working stress levels were increased by one third for loading combinations of dead load, live load and wind load.

For the concrete system consider the wind acting normal to one wall; the forces distributed between the reactor biological shield wall (37 percent) and the two exterior walls parallel to the wind (63 percent).

For the steel, horizontal wind loads are transmitted to the column and distributed to the concrete and roof system. The load on the roof is transmitted to vertical cross bracing at the faces of the building and analyzed in tension only.

Table 3.8-14 includes the design criteria and stresses the Reactor Building is subjected to during a 100 year wind storm. The results of this investigation indicate the Reactor Building will be able to safely withstand the 100 year wind storm.

The following discussion provides the results of the tornado and wind load analysis performed as part of the Systematic Evaluation Program (SEP).

As part of the SEP, GPUN evaluated the various plant structures to determine the maximum permissible wind velocity associated with each structure. The results of the evaluation are presented in Table 3.8-15. These values were approved by the U.S. NRC in their safety evaluation dated December 7, 1992 (Reference 49).

Damage from high winds or tornadoes is caused by either of two phenomena, the wind load itself or impact load from tornado generated missiles. For the concrete structures listed in Table 3.8-15, normal allowable working stresses increased by one third are used to determine maximum permissible wind speeds. The assessment of the effects of tornado generated missiles is performed in accordance with BC-TOP-9-A, Revision 2 (Reference 50). The maximum permissible wind speeds provided in Table 3.8-15 are the lowest of the acceptable wind speeds from the two analyses.

The Reactor Building superstructure has been modified to increase its capability to withstand wind loads. The values presented in Table 3.8-15 reflect the capacity of the structure after completion

of the modification. The analysis does not rely on the roof membrane to transfer load to the building columns. In addition, loads are calculated assuming the roof deck and metal siding remain intact. A failure of the siding or roof deck will increase the capability of the structure to withstand wind loads. Tornado generated missiles are not a concern for this structure due to its elevation above grade. Normal AISC allowable stresses increased by 1.6, but less than yield, are used to determine maximum permissible wind speeds for this structure.

3.8.4.5.5 Masonry Walls

In general, the materials, testing, analysis, design, construction and inspection of safety related concrete masonry walls should conform to the criteria developed by the Structural Geotechnical Engineering Branch (SGEB) of the NRC, the Uniform Building Code, and ACI 551-79.
The masonary walls at OCNGS were evaluated using the following criteria:

1. Allowable stresses were based on ACI 531-79.

2. Loads and load combinations are those presented in this FSAR.

3. The working stress design method was used to qualify the walls.

4. For unreinforced walls, 2% damping for OBE and 4% for SSE were used, for reinforced walls, the values used were 4% damping for OBE and 7% for SSE.

These criterions were found to be acceptable under SGEB guidelines.

3.8.4.6 Materials, Quality Control and Special Construction Techniques

The primary materials of construction are concrete, reinforcing steel and structural steel (rolled shapes and plates). Information is supplied as follows relevant to structures noted.

3.8.4.6.1 Basic Facility Construction

Basis Facility Construction refers to the construction of those structures built as part of the original plant development and does not include those structures built at a later date such as the New Radwaste Building (3.8.4.1) or the Site Emergency Building.

a. Concrete

1. Cement

Cement used was an approved brand of Portland Cement conforming to ASTM Specification C-150; Type II, low alkali cement with the alkali content limited to 0.60 percent total alkali. The low alkali requirement for cement was waived provided that tests in accordance with ASTM C295 and C227 demonstrated no potential alkali reactivity for all aggregates proposed for use in the work. Total alkali is defined as the sum of sodium oxide (Na_20) and potassium oxide (K_20) calculated as sodium oxide.

Cement was tested for compliance with the specifications and in accordance with Federal Test Method Standard No. 158a including the heat of hydration and the false set limitations specified. Mill tests were made by

the supplier and subjected to verification. A notarized test certificate was furnished for each shipment of cement delivered to the batch plant.

Cement was specified to be free from lumps and otherwise undamaged when used in concrete. Sacked cement, if used, was plainly marked to show the type, brand and maker's name. Sacked cement was protected from the weather during shipment and storage, and used in the chronological order in which it was delivered. Bulk cement, when used, was delivered in weather tight carriers and unloaded by means of weather tight conveyors or other suitable means to protect the cement completely from exposure to moisture. Bulk cement was specified to be stored in weather tight bins. Bins were to be emptied and cleaned when so directed with the intervals between required cleanings not to be less than four months.

2. Aggregate

Fine aggregate consisted of natural sand, manufactured sand, or a combination of natural and manufactured sands. Coarse aggregate consisted of gravel, crushed gravel, crushed stone or a combination thereof.

Tests were performed as necessary to determine that the proposed aggregate would produce concrete of acceptable quality and durability, meeting the requirements of the specifications and would be available in sufficient quantity to meet the requirements of the project.

Aggregate as delivered to the batch bins was from approved sources and consisted of clean, hard, durable particles conforming to the requirements of the specifications. Discretionary tests and analysis of aggregates during the various stages in the processing and handling operations were made to determine compliance with the specification requirements. During construction, aggregates were sampled as delivered to the batch plant to determine compliance with the specification requirements. These samples were obtained using the specified ASTM test methods.

For fine aggregate as delivered to the batch plant the maximum percentages of deleterious substances including material passing the No. 200 sieve was limited to the values set forth in the following table:

The sum of the percentages of deleterious substances including material passing the No. 200 sieve was not to exceed three percent by weight.

	Percent by Weight	Test
Material Passing No. 200 sieve	3	ASTM C117
Coal and Lignite	1	ASTM C123
Clay Lumps	1	ASTM C142

All fine aggregate was free from injurious amounts of organic impurities. Aggregate subjected to the colorimetric test for organic impurities in accordance with ASTM C40 and producing a color darker than the standard was rejected unless mortar strength requirements were met.

The use of aggregate showing a darker color than that of the samples originally approved for the work, was withheld until satisfactory mortar strength tests were made to determine whether the change in color was indicative of an injurious amount of deleterious substances.

Fine aggregate was well graded from coarse to fine and when tested by means of laboratory sieves in accordance with ASTM C136 conformed to the following requirements:

Sieve Designation Square Openings	Percent by Weight Passing Sieves
3/8 inch	100
No. 4	95 - 100
No. 8	80 - 100
No. 16	50 - 85
No. 30	25 - 60
No. 50	10 - 30
No. 100	2 - 10
No. 200	0 - 3

The above gradation represented the extreme limits to determine suitability for use from all sources of supply. The gradation was reasonably uniform and not subject to the extreme percentages of gradation specified above. Fineness modulus was maintained within the range of 2.30 to 3.10. The grading of fine aggregate was controlled so that the fineness moduli of test samples of the fine aggregate, as delivered to the batch plant, did not vary more than 0.20 from the average fineness modulus of all samples previously taken unless suitable adjustments were made in concrete proportions to compensate for the difference in grading. The fineness modulus was determined by dividing by 100 the sum of the cumulative percentages retained on sieves Nos. 4, 8, 16, 30, 50 and 100.

Fine aggregate, when subjected to the mortar strength test in accordance with ASTM C87 had a compressive strength, at 7 and 28 days using Type II Portland cement, of not less than 90 percent of that developed by the same cement and graded Ottawa sand having a fineness modulus of 2.4 plus or minus 0.10 and the same water cement ratio and consistency.

When the fine aggregate was subjected to five alternations of the sodium sulfate soundness test, in accordance with ASTM C88, the weighted loss did not exceed eight percent.

When tested in accordance with ASTM C128, the fine aggregate had a specific gravity of not less than 2.60 and the absorption did not exceed 1.5 percent.

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The amount of deleterious substances in coarse aggregate as delivered to the batch plant did not exceed the following limits:

	Percent By Weight	Test
Soft Fragments	5.00	ASTM C235
Coal and Lignite	0.25	ASTM C123
Clay Lumps	0.25	ASTM C142
Material passing the No. 200 sieve	1.00	ASTM C117

Percentage of loss for gravel or crushed stone did not exceed 40 percent when subjected to the Los Angeles Wear Test, in accordance with ASTM C131.

Coarse aggregate was separated into two sizes, meeting the following grading requirements by weight when tested by means of laboratory sieves in accordance with ASTM 0136:

Sieve Designation	Percent By Weight Passing Sieves		
Square Opening	Large Size 1-1/2" to 3/4" to #4	Small Size 3/4" to #4"	Combined 1-1/2"
2"	100		100
1-1/2"	90 - 100	100	95
1"	20 - 55	100	
3/4"	0 - 15	90 - 100	35 – 70
3/8"	0 - 5	20 - 55	10 – 30
#4		0 - 10	0 – 5
#8		0 - 5	

The above gradation represented the extreme limits to determine suitability for use from all sources of supply. The gradation used was reasonably uniform and approached the mean of the percentage of gradation specified above.

Coarse aggregate was limited to less than 5 percent of flat or elongated particles as defined by ASTM C125, having a maximum dimension greater than 5 times the minimum dimension.

When the coarse aggregate was subjected to five alternations of the sodium sulfate soundness test, in accordance with ASTM Method C88, the weighted loss did not exceed ten percent.

When tested in accordance with ASTM C127, the coarse aggregate had a specific gravity of not less than 2.60 and the absorption did not exceed 1.5 percent.

All aggregates were stored in such a manner as to prevent segregation and the inclusion of foreign materials in the concrete. Aggregates which became mixed with foreign matter of any kind were not used. All fine aggregate remained in free drainage storage at the batch plant for at least 72 hours immediately prior to its use. Any washing, screening, classifying, blending or other operations on the aggregates, required to meet these specifications, was performed prior to delivery to the batch plant stockpile.

3. Water

Water used in mixing concrete was free from injurious amounts of impurities. Potable water was considered acceptable. If doubt existed as to the suitability of the water, it was tested by one or both of the following tests:

When used in a mix of an approved cement and standard Ottawa sand the mortar strength ratio shall not be less than 90 percent of that obtained from the use of distilled water with equivalent cement and sand.

American Association of State Highway Officials Method T-26 was adhered to for impurities.

4. Admixtures

An approved air entraining agent conforming to ASTM C260 was included where required in accordance with the manufacturer's recommendations in proportions such that air entrainment of 4 to 6 percent by volume of the concrete will be produced as determined by ASTM C138, C173 or C231. In order that the proportions may be adjusted to produce the specified percentage of air under varying conditions, the agent was not combined with the cement or other admixture prior to batching.

A water reducing, set retarding agent conforming to ASTM C494 was included where required in accordance with the manufacturer's recommendations. Certified laboratory test reports were submitted demonstrating the performance of the proposed agent when used in the manufacturer's recommended proportions in a mix with the cement and aggregates used in the work.

5. Proportioning

a. Classes of Concrete

Concrete was specified to ensure plastic, workable concrete of homogeneous structure which, when hardened, would have durability, impermeability and the strengths for each class of concrete tabulated below.

Class of Concrete	28 Day Strength psi	Max. Size Aggregate inches	Admixtures (WRR = water reducing retarding) (AE = Air entraining)
4L	4000	1-1/2	WRR
4LA	4000	1-1/2	WRR, AE
4S	4000	3/4	WRR
4SA	4000	3/4	WRR, AE
3L	3000	1-1/2	WRR
3LA	3000	1-1/2	WRR, AE
3S	3000	3/4	WRR
3SA	3000	3/4	WRR, AE
2L	2000	1-1/2	

b. Slump

Consistency of concrete was determined in accordance with ASTM C143. The slump was not less than two inches nor more than four inches. Tests were made during the progress of the work.

c. Control

The strength quality of the concrete proposed for use was established by tests made in advance of the beginning of operations, using the maximum slump specified herein. Specimens were made and cured in accordance with ASTM Standard C192 and tested in accordance with ASTM Standard C39. Curves representing relation between the water content and the average 28 day compressive strength, were established for a range of values including the compressive strengths specified. Curves were established by at least three points, each point represented average values from at least four test specimens. The maximum allowable water content for each class of concrete was determined from these curves and corresponded to a compressive strength 15 percent greater than that specified for that class.

d. Strength Tests During the Work

Three cylinders were taken from each 150 cubic yards or fraction thereof, or each day's delivery, whichever was less, for each class of concrete delivered. Test specimens were made and cured in accordance with ASTM Standard C31 under laboratory conditions. When the possibility existed for

the air temperature to fall below 40°F, a discretionary equal number of additional cylinders were made for curing under field conditions to indicate the adequacy of protection and curing of the concrete by others. All cylinders were tested in accordance with ASTM Standard C39.

b. Reinforcement

Reinforcing steel was shop fabricated to the required shapes and dimensions and was placed where indicated on the drawings or where required to carry out the intent of the drawings and specifications. Before being placed, reinforcing steel was thoroughly cleaned of loose or flaky rust, mill scale, or coating, including ice, and of any other substance that would reduce or destroy the bond. Reinforcing steel reduced in section was not used. Previously placed reinforcing steel left for future bonding was inspected and cleaned. Reinforcing steel was not permitted to be bent or straightened in a manner injurious to the steel. Bars with kinks or bends not shown on drawings were not placed. The use of heat to bend or straighten reinforcing steel was not permitted without special permission of the entire operation. Reinforcing steel was not spliced at points of maximum stress. Laps or splices were of adequate length to transmit stresses and, unless otherwise indicated, conformed to the table in ACI315. Splices not occurring in zones of tension were lapped a minimum of 24 bar diameters. Splices in adjacent bars were staggered. Splices in columns, piers, and struts were lapped sufficiently to transfer the full stress by bond.

All splices of No. 14S and 18S reinforcing bars, and No. 11 bars where indicated on the drawings, were made by welding or by approved mechanical connections. Welded splices were thermit welded or arc welded.

Bars to be thermit welded were cut off square by means of an oxyacetylene torch or by sawing. Ends of bars to be arc welded were beveled. Ends of bars within the welded or mechanical splice area were free of dirt, oxide, scale, cement, paint, oil, grease, moisture or other foreign matter prior to splicing.

Procedure for thermit welding or for installing mechanical connections were approved and in accordance with the manufacturer's recommendations. Procedures for arc welding were submitted for approval. Arc welding conformed to the Recommended Practice for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction (D12.1) of the American Welding Society.

Prior to the making of production splices, a procedure qualification test was performed which consisted of making test splices at the site using the same procedures and mechanics to be employed in making the splices in the structure. At least six (6) test splices of each bar size to be used in the work were made with at least three bars in the horizontal position and at least three with bars in the vertical position, for each splicing procedure proposed. Each of the test specimens so formed were tension tested by an approved testing laboratory.

Every mechanic employed in the making of thermit welds or mechanical splices made at least one acceptable test splice at the site using the same procedure to be used in making the splices in the structures. Welders were qualified by tests as prescribed in the Standard Code for Arc and Gas Welding in Building Construction (D1.0) of the

American Welding Society. Test specimens were tension tested by an approved testing laboratory and all test reports were submitted for the records. Any mechanic having made a test splice failing to meet acceptance requirements was not employed in making splices in the structures without a written waiver of this requirement.

Splices were required to have a minimum ultimate tensile strength of 100 percent of the specified minimum ultimate tensile strength of the bar. Pouring gates remaining after making of connections were removed. All connections were examined visually; evidence of porosity or lack of fusion in welds and deviations of mechanical connections from approved test connections were cause for rejection. Rejected splices were cut out and replaced.

c. Structural Steel

1. Mill Inspection

Arrangements were made with the mills which rolled the material so that the Engineer could witness all operations and tests in the mill and have satisfactory opportunity for thorough surface inspection and gauging of each piece of material after it came from the mills and prior to loading.

2. Test Reports

Certified reports of mill tests of the specified physical and chemical requirements for each heat of steel were furnished to the Engineer before any fabrication was started on the material covered by the reports.

3. Shop Inspection

All material was subject to inspection by the Engineer. The Engineer had at all times access to all parts of the shop while material was being fabricated and was provided with all reasonable inspection facilities. No material was painted or shipped without release by the Engineer unless inspection was waived in writing by the Engineer.

3.8.4.6.2 New Radwaste Building

The materials used for construction of the New Radwaste Building, together with the quality control standards and inspection requirements during construction are described in the following subsections.

a. Concrete

All structural concrete for the New Radwaste Building has a minimum compressive strength of 5000 psi at 28 days. The concrete materials and specifications confirming their suitability are discussed in the following paragraphs:

1. Cement - The cement used in developing the design mix as well as in the production mix conforms to ASTM C150, Type II cement. Testing at the mill is in accordance with ASTM C183 and C150 at the frequency listed below:

- Chemical requirements listed in Table I of ASTM C150 - every 500 barrels
- Fineness requirement noted in Table II of ASTM C150 - every 500 barrels
- Physical requirements listed in Table II of ASTM C150 - every 2000 barrels.

In addition to these qualifying tests, the production cement is tested monthly in accordance with ASTM C109 and ASTM C191. Each shipment received by the production plant is tested for physical and chemical properties in accordance with ASTM C150.

2. Aggregate - Aggregate for concrete work are qualified in accordance with ASTM C33. In addition to the qualifying tests in that specification, a periodic inspection control program is conducted by an independent testing laboratory during concrete production to ensure continued conformance with specification requirements. The tests and frequency of testing are as follows:

a. Fine Aggregates

Test	Applicable ASTM Standard	Frequency of Tests
Material Finer than No 200 Sieve	C117	Daily
Coal and Lignite	C123	Daily
Clay Lumps	C142	Daily
Organic Impurities	C40	Daily
Gradation Including Fineness Modulus	C136	(1)
Mortar Strength	C87	Bimonthly or (2)
Soundness	C88	Bimonthly or (2)
Moisture Content	C566	(3)
Water Soluble Chlorides	C1411	Weekly or (2)

b. Coarse Aggregate

Soft Particles	C125	Daily
Coal and Lignite	C123	Daily
Clay Lumps	C142	Daily
Material Finer than No. 200 Sieve	C117	Daily
Abrasion	C131	Every 3 mos. or (2)
Gradation	C136	(1)
Flat or Elongated Particles	CRD-C119-53(4)	Weekly or (2)
Soundness	C88	Bimonthly or (2)
Moisture Content	C566	(3)

Water Soluble Chlorides	D1411	Weekly or (2)

Notes:
(1) Before first batch and each 70 tons.
(2) Whenever it appears the type of grading of aggregate has changed as determined by Construction Manager.
(3) Before first batch and every hour thereafter.
(4) Corps of Engineers Standard.
Organic matter
(Volatile Solids) less than 20 ppm AASHO Designation, T26

3. Water - It was ascertained that water proposed to be used for concrete mixes and for making ice for hot weather concrete production met the following specific requirements:

   a. Setting time for portland cement not altered more than 25 percent as compared to that obtained by use of distilled water when tested in accordance with ASTM C191.
   b. Compressive strength of mortar, when tested in accordance with ASTM C109, is not reduced by more than 10 percent when compared with results obtained using distilled water.

   c. In addition, the mixing water was checked for compliance with Section 3.1.3 of AASHO Designation M157 and the following:

	Maximum Limit	Test
Chlorides	30 ppm	ASTM D512
Sulfates	150 ppm	ASTM D516
pH Value	4.5 to 8.5	AASHO Designation, T26
Total Solids in Water (Total Residue)	less than 500 ppm	AASHO Designation, T26

   During construction, the testing described in items b and c above was conducted on a monthly basis in order to ensure continued conformance with the specification requirements.

4. Admixtures - An air entraining agent conforming to ASTM C260 is used. The amount of the agent is adjusted to produce air entrainment of 4 to 6 percent by volume as determined by ASTM C231.

   A water reducing agent conforming to ASTM C494, Type D is specified for use in concrete for hot weather production. For cold weather production a water reducing agent conforming to ASTM C494, Type A is specified.

5. Concrete - Tests and Inspection - Trial design mixes are proportioned in accordance with Method 1 outlined in Paragraph 3.8.2.1 of ACI301. The design mix proportions are required to produce a concrete of the following specified strength and slump.

Concrete Class	Strength f'_c at 28 Days	Slump(Inches) Min.	Max.
5SA	5000 psi	1	4
2SS	2000 psi	1	4

Concrete was furnished by the Civil Construction Contractor in accordance with the requirements of the B&R specification for this work. A qualified testing laboratory was retained by the Contractor to test and inspect all concrete ingredients. A Quality Control Program was established to confirm that the required tests and inspections were made and recorded to show that the concrete did meet the specification requirements.

The Contractor's batch plant inspector tested all mix ingredients periodically and certifies the mix proportions for each batch. The inspector ensured that a ticket was provided for each batch documenting the time loaded, actual mix proportions (including water), amount of concrete, concrete design strength, amount of water which may have been added at the site, location of concrete placement, and identification of transit mixer.

Other inspectors at the construction site were assigned to inspect reinforcing and form placements, to make slump tests, to check air content, and to record weather conditions. Field control was in accordance with the "Manual of Concrete Inspection" as reported by ACI Committee 311 and testing was performed as described hereunder:

1. A set of nine (9) cylinders was made for each 100 cubic yards, or fraction thereof, of each class of concrete delivered each day. They were tested as follows:

Concrete	Age in Days 7	29	90
5000 psi	3	3	3
2000 psi	3	3	3

Cylinders were made in accordance with ASTM C31 from samples taken at point of concrete placement. After an initial period of field storage and curing for about 24 hours, the cylinders were cured under laboratory conditions until time of test. Compressive tests were made in accordance with ASTM C39. The 28 day tests were evaluated in accordance with ACI 214 as follows:

- The coefficient of variation is 15 percent or less.

- Strength level of concrete is considered satisfactory if the average of any three consecutive tests of the laboratory cured specimens representing each specified strength of concrete is equal to or greater than the specified strength,

and no individual strength test result falls below the specified value by more than 500 psi.

2. Slump, air content, and temperature are taken when cylinders are cast and as follows:

- For concrete mixed in stationary mixers at concrete plant the slump, air content, and temperature are tested for at least the first two (2) truck loads of each class each day.

- At the site, slump and temperature are tested for each load of concrete after any final addition of water. Air content is checked for the first two (2) loads in each placement, every 40 cubic yards in each placement, and whenever the total mixing water between consecutive loads of the same slump varies by more than one gallon per cubic yard. Slump tests are performed in accordance with ASTM C143. Air content is determined in accordance with ASTM C231.

b. Reinforcement Steel

All reinforcing steel conformed to ASTM A615 Grade 60. Maximum bar size is No. 11 bar. All splices are tension lap splices. Welding of reinforcing steel was prohibited. Testing and inspection at the mill was performed in accordance with ASTM A615 except that tensile tests are based on the full section of the bar as rolled. Certified mill test reports of the chemical and physical properties, including measurement of bar deformations, were submitted to the Architect/ Engineer for review. Before placement, additional tensile testing was conducted in the field to check compliance with the specification. For each lot of a particular bar size furnished from heat, one test specimen for each 50 tons, or fraction thereof, was tested. If the test results did not meet the requirements of ASTM A615 or deviated more than 10 percent from the mill test results, two additional samples from the same 50 ton segment of the lot were tested.

If these additional tests showed nonconformance with ASTM A615 or a deviation from the mill tests by more than 10 percent, the entire lot of bars of that size produced from the heat tested were rejected.

c. Structural Steel

Since none of the structural steel is seismic Category I this section is not applicable.

d. Construction

Codes used to establish the specifications and procedures governing the construction of the New Radwaste Building are given in Subsection 3.8.4.2.

3.8.4.7 Testing and Inservice Inspection Requirements

Testing and inservice inspection requirements for the OCNGS are established in the Technical Specifications and in the current version of the approved Inservice Inspection Program for Oyster Creek.

3.8.5 Foundations

3.8.5.1 Description of the Foundations

The general locations, relationships and basic dimensions for the individual structures' foundations are given in Subsection 3.8.4.1. Additional general relationships for the major structures are shown in Figures 3.8-38, 3.8-39 and 3.8-40.

Foundations for seismic Category I structures (Class I) are constructed of conventionally reinforced concrete mats of varying thicknesses founded on firm soil.

The walls of the three largest structures extend below grade to the continuous foundation mats (see Figure 3.8-39). The exterior surfaces of the walls and slabs have waterproofing materials applied to seal against the intrusion of ground water. Material specifications are given in Subsection 3.8.4.2.

Excavations have been backfilled and compacted against the foundation walls with sound soil material suitable for the purpose. Compacted, engineered backfill is capable of resisting lateral foundation forces by passive resistance although this property has not been necessary to assure the performance of the foundations.

There are no unique features utilized for foundations.

3.8.5.2 Applicable Codes, Standards, and Specifications

Codes, standards and specifications given in Subsection 3.8.2.2 (Containment) and Subsection 3.8.4.2 (Other Category I Structures) are applicable to the respective structure foundations.

3.8.5.3 Loads and Load Combinations

The foundation for the drywell vessel is incorporated as part of the Reactor Building. Loads, load combinations and the design approach used in the combined foundation mat is consistent with the drywell and superstructure analysis and design and is given in Subsections 3.8.2.3 and 3.8.4.3, respectively.

The other Category I structure foundations' loads, loading combinations and design approaches are given in Subsection 3.8.4.3.

Foundations and structures have been checked for sliding and overturning due to earthquakes, winds and tornados and for flotation due to high ground water levels.

Lateral earth pressures are considered as applicable (see Reference 38).

3.8.5.4 Design and Analysis Procedures

The foundations of Category I structures are analyzed and designed to determine maximum stresses in reinforcing, concrete and bearing soils using the appropriate load combinations identified in Subsection 3.8.5.3.

3.8.5.4.1 Boundary Conditions and Expected Behavior

The foundations are founded directly on firm soil and are designed to sustain all credible loads being transferred from the structures they support. Design procedures for all structures insured that the foundation bearing pressures did not exceed 13,000 psf. This allowable bearing pressure was established by rational engineering procedures.

Each building has an individual foundation system; no common foundations are employed except for the support of the Control Room by part of the Turbine Building structure. The effect of lateral pressures created by adjacent structures has been taken into consideration. (See Reference 39 and 40).

All loads are transferred to the foundation through the walls, or in the case of the drywell shield wall and reactor vessel, through supporting pedestals. Reinforcement continues from the walls into the foundation mat and is embedded in the mat and wall by hooks, bends or straight extensions of sufficient length to develop the design strength of the bars, forces generated within the drywell vessel are transmitted to the external walls and pedestal by direct bearing; no anchorage is provided through the drywell shell. In all cases the forces are adequately transferred from the structures to the foundations.

The foundation mats are designed as continuous slabs and beams on rigid nonyielding media.

3.8.5.4.2 Vertical Loads, Lateral Loads and Overturning Moments

Vertical loads are carried by direct bearing on the firm soil media. Overturning moments and bearing pressures on the soil (including maximum toe pressure) were investigated in accordance with established safety factors.

Base shears were calculated to be transferred by friction between the bottom of the foundation and the interfacing face of the soil without incorporating the resistance of lateral passive earth pressures which is actually available through embedment of the foundation mat and wall system.

Conventional hand calculations were used for the design of the foundations. Uplift and overturning forces were determined by static methods using forces which were provided by the seismic analysis.

Design factors of safety against sliding, overturning and buoyancy are given in Subsection 3.8.5.5.

3.8.5.4.3 Computer Programs

There were no computer programs used in the design of the foundations.

3.8.5.5 Structural Acceptance Criteria

The acceptance criteria relating to the behavior of structural materials, such as concrete and reinforcement, is given in Subsection 3.8.4.5.

The design of the foundations is such that the safety factors for buoyancy, sliding and overturning will not be less than the following, for loads as defined in Subsection 3.8.4.3.

Load	Factor of Safety		
	Overturning	Sliding	Flotation
Normal	1.5	1.5	-
Severe	1.5	1.5	-
Extreme	1.1	1.1	-
Dead Load/Flooding	-	-	1.1

The use of waterproofing materials against the exterior concrete surfaces of walls and slabs does not prevent the structures from achieving the resistance to sliding as considered in the structure design. The lateral resistance of passive earth pressure has been kept in reserve in checking the foundation factors of safety.

3.8.5.6 Materials, Quality Control and Special Construction Techniques

The primary materials of construction are concrete and reinforcing steel. Their descriptions and basic quality control procedures are discussed in Subsection 3.8.4.6.

There were no special construction techniques.

3.8.5.7 Testing and Inservice Surveillance Requirements

The ability of the drywell and torus to transmit pressure associated loads to the soil media via the foundations has been demonstrated by the structural integrity test described in Subsection 3.8.2.7.

No preoperational or inservice surveillance tests are required for the other Category I structure foundations.

3.8.6 References

(1) Oyster Creek Nuclear Power Plant Unit No. 1, Facility Description and Safety Analysis Report, Docket No. 50-219, Amendment No. 15, Primary Containment Design Report, September 11, 1967.

(2) NUREG-0661. Safety Evaluation Report, Mark I Containment Long Term Program Resolution of Generic Technical Activity A-7. July 1980.

(3) NUREG/CR-1083, LBL-6754. Aslam, M; Godden, W.G.; and Scalise, T. Sloshing of Water in Annular Pressure Suppression Pool of Boiling Water Reactors under Earthquake Ground Motions. Lawrence Berkeley Laboratory for U.S.N.R.C. October 1979.

(4) NUREG/1082, LBL-7984. Aslam, M; Godden, W.G.; and Scalise, T. Sloshing of Water in Torus Pressure-Suppression Pool of Boiling Water Reactors under Earthquake Ground Motions. Lawrence Berkeley Laboratory for U.S.N.R.C. October 1979.

(5) NUREG-0408. Mark I Containment Short Term Program: Safety Evaluation Report. December 1977.

(6) NEDO-21888 (Revision 2). Mark I Containment Program: Load Definition Report. November 1981.

(7) NEDO-24572 (Revision 2). Mark I Containment Program: Plant Unique Load Definition Oyster Creek Nuclear Generating Station. July 1982.

(8) NEDO-24583-1. Mark I Containment Program Structural Acceptance Criteria: Plant-Unique Analysis Application Guide. October 1979.

(9) NEDC-23702-P. Arain, S.M. Mark I Containment Program: Seismic Slosh Evaluation. March 1978.

(10) MPR-733. Oyster Creek Nuclear Generating Station Mark I Containment Long-Term Program: Plant-Unique Analysis Report Suppression Chamber and Vent System. August 1982.

(11) MPR-734. Oyster Creek Nuclear Generating Station, Mark I Containment Long Term Program: Plant-Unique Analysis Report Torus Attached Piping. August 1982.

(11a) MPR-722. Oyster Creek Nuclear Generating Station, Mark I Containment Long Term Program: Plant-Unique Analysis Supplemental Report. July 1983.

(12) Von Karman, "The Buckling of Spherical Shells by External Pressure," Pressure Vessel and Piping Design, Collected Papers, ASME, 1960, pp. 633 to 640.

(13) Oyster Creek Nuclear Power Plant Unit No. 1, Facility Description and Safety Analysis Report, Docket No. 50-219, Amendment No. 50, Primary Containment Penetration Design, March 1969.2

(14) Oyster Creek Nuclear Power Plant Unit No. 1, Facility Description and Safety Analysis Report, Docket No. 50-219, Amendment No. 51, Supplemental Information Regarding Primary Containment Penetrations, March 21, 1969.

(15) Letter from I. R. Finfrock, Jr. (JCP&L) to George Lear (NRC), dated November 1, 1977, on Torus Pool Swell-Relief Valve Actuation.

(16) Letter from George Lear (NRC) to I.R. Finfrock, Jr. (JCP&L), dated March 24, 1977, Summary of March 4, 1977 Meeting Results, Related to Torus Inspection for Corrosion and Staggered Relief Valve Set Points.

(17) Letter EATJM-190, March 22, 1977, Report on Steam Vent Cleaning Phenomenon.

(18) Letter EATJM-29, January 10, 1977, Report on Steam Vent Clearing Phenomenon.

(19) Letter EA-76-686, July 16, 1976, Oyster Creek Torus Shell Thickness Evaluation.

(20) Drwg 4104-1 Biological Shield Wall

(21) Drwg 4205-1 Biological Shield Wall, Sections & Details

(22) Drwg 4069-4 Radial Beam Framing (Inside Drywell)

(23) Drwg 2063-4 General Arrangement-Reactor Building Sections (from Print Book; shows drywell internals).

(24) Drwg 4049-7 Reactor Building Floor Plan & Sections (Outside Drywell Shell).

(25) Drwg CBI 34-3 Floor Framing Bracket

(26) Drwg CBI 35-3 Floor Framing Hanger

(27) Calculations 19-62 Drywell Steel Framing at El. 46'-08" to 19-102

(28) Calculations 29-19

(29) Calculations 9-126 to 9-304

(30) Calculations 21-32 to 21-56

(31) Calculations 6-1 to 90

(32) Letter from D.A. Ross (JCP&L) to B.H. Grier (NRC:I&E), dated December 7, 1979, Re: IE Bulletin 79-02

(33) Letter from D.A. Ross (JCP&L) to B.H. Grier (NRC:I&E), dated August 3, 1979, Re: IE Bulletin 79-02.

(34) Letter from D.A. Ross (JCP&L) to B.H. Grier (NRC:I&E), dated July 6, 1979, Re: IE Bulletin 79-02.

(35) Letter from P.B. Fiedler (GPUN) to D.M. Crutchfield (NRC: DRL), dated November 2, 1983, Re: IE Bulletin 80-11.

(36) Letter from P.B. Fiedler (GPUN) to D.M. Crutchfield (NRC: DRL), dated August 11, 1983, Re: IE Bulletin 80-11.

(37) Letter from I.R. Finfrock, Jr. (JCP&L) to B.H. Grier (NRC: I&E), dated September 19, 1980, Re: IE Bulletin 80-11.

(38) Calculations, Sheets 9-1 to 9-25, Frame 37.

(39) Drawings 4075-7, 4049-7, 4103-4.

(40) Calculation Sheets: 1-1 to 1-128, Frame 2;
to 28-130, Frame 36;
to 27-94, Frame 31;
to 3-59, Frame 18;
to 9-125, Frame 37.

(41) MPR-999, Rev. 3, Oyster Creek Nuclear Generating Station, Mark I Containment Long Term Program: Addendum to MPR-734 Plant-Unique Analysis Report Torus Attached Piping, 12/88

(42) GPUN Safety Evaluation SE-000243-002, (Current Revision), "Drywell Steel Shell Plate Thickness Reduction".

(43) Letter, GPUN to NRC, dated March 25, 1993, NRC SER for License Amendment No. 163, dated April 6, 1993.

(44) GPUN Technical Data Report TDR-1108, "Summary Report of Corrective Action Taken from Operating Cycle 12 through 14R", April 28, 1993.

(45) Letter, NRC to GPUN, dated February 21, 1995, NRC SER for License Amendment No. 177, dated February 21, 1995.

(46) MPR-1434, Evaluation of Proposed Increase in Technical Specification Limits for EMRV Setpoint Pressure on Mark I Containment Long-Term Program Analyses. March 1994.

(47) Letter , NRC to GPUN, dated November 1, 1995, Subject: "Changes in the Oyster Creek Drywell Monitoring Program," as clarified by Letter, NRC to GPUN, dated February 15, 1996, same subject.

(48) NRC SER for License Amendment No. 165, dated September 13, 1993.

(49) Letter, A. Dromerick, NRC, to J. J. Barton, GPUN, “Evaluation of Upper Reactor Building and Non-Safety Architectural Components Subjected to Tornado-Wind Loading – Items 1 and 11 of SEP Topic III-2 (M79165),” Docket No. 50-219, December 7, 1992.

(50) BC-TOP-9-A, Revision 2, “Design of Structures for Missile Impact,” Bechtel Corporation.

(51) Calculation “C-1302-187-E310-037 Statistical Analysis of Drywell Vessel Thickness Data Through September 2000,” Revision No. 0.

(52) ECR 02-01441, Revision 0, “Oyster Creek Drywell Vessel Corrosion Assessment.”

(53) Calculation C-1302-187-E310-041, “Statistical Analysis of Drywell Vessel Sandbed Thickness Data 1992, 1994, 1996, and 2006.”

(54) OCNGS Response Letter SE Topic II-3-C Flooding Potential and Protection Requirements, 06/06/1983

TABLE 3.8-1
(Sheet 1 of 1)

PRESSURE SUPPRESSION SYSTEM DIMENSIONS

DRYWELL	
Cylindrical Section – Diameter	33 ft
Cylindrical Section – Length	23 ft 3 in
Spherical Section - Diameter	70 ft
Free Air Volume	180,000 ft^3
Wall Plate Thickness	
Spherical Shell	0.676, 0.722, 0.770, 1.154 in
Spherical Shell to Cylindrical Neck	2 5/8 in
Cylindrical Neck	0.640 in
Top Head	1 3/16 in
VENT SYSTEM	
Number of Vent Pipes	10
Internal Diameter	6 ft 6 in
Break Area/Vent Pipe Area	0.0194
Downcomer Pipes	
Number of Downcomer Pipes	120
Internal Diameter	1 ft - 11 1/2 in
Submergence Below Absorption Pool Water Level	3 ft (*)
PRESSURE ABSORPTION CHAMBER	
Water Volume (max. operating)	91,000 ft^3
Free Air Volume	122,400 ft^3
Chamber Inner Diameter	30 ft
Torus Major Diameter	101 ft

(*) This dimension has changed due to revision as a result of the design changes identified in Subsection 3.8.2.

TABLE 3.8-2
(Sheet 1 of 1)

DRYWELL BUCKLING STRESS

POINT	EXTERNAL PRESSURE		DRYWELL SHELL & APPURTENCES		DRYWELL COMP. MATERIAL		22% LATERAL E.Q.		10% VERTICAL E.Q.		H_20** SEAL		ACCESS LL & BEAM LOADS		GRAVITY LOADS WELD PADS		TOTAL STRESSES (LBS/IN)		PLATE THICK-NESS	BUCKLING STRESS
	T_2	T_1	T_2	T_1	T_2	T_1	T_2	T_1	T_2	T_1	T_2	T_1	T_2	T_1	T_2	T_1	T_2	T_1		
B*	-198	-785	-185	-249	-13	-18	-108	146	-19	27	-555	-768	-	-	-	-	-1078	-1993	0.64	0.290
C[1]	-272	-1570	-265	-1183	-19	-69	-142	-526	-28	-125	-659	-2452	-	-	-	-	-1385	-5930	2.562	0.362
C	-420	-1344	-382	-1181	-27	-78	-186	-459	-42	-129	-878	-2102	-	-	-11	-27	-1935	-5293	2.562	0.495
D	-420	-420	-352	+280	-24	+22	±153	±153	±38	±31	-730	+730	-	-	-9	+9	-1726	+805	0.722	0.772
E	-420	-420	-198	+163	-21	+21	±82	±82	±24	±21	-313	+313	-	-	-16	+16	-1074	+196	0.722	0.480
F	-420	-420	-357	+395	-41	+42	±197	±197	±209	±213	-296	+296	-530	+530	-16	+16	-2066	+1269	0.770	0.813
G	-420	-420	-1240	+1356	-128	+128	±1136	±1136	±549	±559	-630	+630	-1361	+1361	-39	+39	-5503	+4789	1.154	0.964

* Axial buckling of cylindrical shell = $ST_2/1.8 \times 10^6(t^2/R)$£1
** Stresses = water seal - head
*** Buckling stress ratio - $(ST_2 - T_1)/1.8 \times 10^6$ (t^2/R) + $ST_1/9 \times 10^5$ (T^2/R)£1 (where ST_1 and ST_2 are compressive stresses).

TABLE 3.8-3
(Sheet 1 of 1)

ALLOWABLE STRESSES - PRIMARY CONTAINMENT (Basic Design)

Material SA212 Grade B to SA300

Loading Condition	General Membrane		General Bending Plus Local Membrane Plus General Membrane		General Membrane Plus Secondary Stresses	
	Allowable Stress psi	Percent of Y.S.	Allowable Stress psi	Percent of Y.S.	Allowable Stress psi	Percent of Y.S.
1. Dead Load Plus Operating Loads Plus Loss of Coolant Loads Plus Seismic Loads (0.11g) (OBE)	19,250	50.7	28,875	75.97	52,500	138
2. Dead Load Plus Operating Loads Plus Loss of Coolant Loads Plus Seismic Loads (0.22 g) (SSE)	Safe Shutdown of Plant can be achieved (See Note 1 below)					

Y.S. = Minimum yield stress of the material.

NOTE 1:

The combined stresses resulting from functional loadings and from an earthquake having a ground acceleration of 0.22g are such that a safe shutdown can be achieved. The combined earthquake and functional load stresses probably would not exceed yield stress. However, where calculations indicate that a structure or piece of equipment will be stressed beyond the yield point, an analysis will be made to determine its energy absorption capacity. The capacity is such that it exceeds the energy input from the earthquake. In addition, the design has been reviewed to assure that any resulting deflections or distortions would not prevent the proper functioning of the structure or piece of equipment and would not endanger adjacent structures or components.

TABLE 3.8-4
(Sheet 1 of 1)

ALLOWABLE STRESSES FOR REACTOR VESSEL CONCRETE PEDESTAL

Loading Condition	Reinforced Steel Maximum Allowable Tension Stress	Reinforced Steel Maximum Allowable Compression Stress	Concrete Maximum Allowable Compression Stress	Concrete Maximum Allowable Shear Stress
1. Dead Load Plus Equipment Load Plus Jet Load Plus Temperature Plus Design Earthquake (OBE)	0.25 Fy	0.10 Fy	0.133 f'_c (bending) 0.116 f'_c (direct)	0.55 $(f'_c)^{1/2}$
2. Dead Load Plus Equipment Load Plus Jet Load Plus Temperature Plus Double Design Earthquake (SSE)	0.25 Fy	0.10 Fy	0.267 f'_c (bending) 0.232 f'_c (direct)	1.1 $(f'_c)^{1/2}$

NOTES: a. Jet plus seismic loads per J.A. Blume's "Earthquake Analysis: Reactor Pressure Vessel." (See Section 3.7)

b. Temperature = 40°F maximum gradient

Table 3.8-5
(Sheet 1 of 1)

ALLOWABLE STRESSES FOR STRUCTURAL STEEL FRAMING SYSTEM

Allowable Stresses for Structural Steel for:
a) Reactor Building Roof Steel, Craneway Columns, Crane Girders, Vertical Bracing
b) Drywell Radial Steel Framing and Recirculating
c) Reactor Building Platforms

	Loading Condition	Tension on Net Section	Shear On Gross Section	Compression	Bending Tension & Compression	A325 U.S. Bolts With Thread Excluded From Shear Plane: Tension (psi)	A325 U.S. Bolts With Thread Excluded From Shear Plane: Shear (psi)	A141 ivets: Tension (psi)	A141 ivets: Shear (psi)
1.	Dead Load Plus Live Load Plus Operating Load Plus Design Earthquake (OBE)	0.60 Fy	0.40 Fy	Varies with Slenderness Ratio	0.60 Fy* to 0.66 Fy	40,000 (0.50 Fy)	22,000 (0.27 Fy)	20,000 (0.56 Fy)	15,000 (0.42 Fy)
2.	Dead Load Plus Live Load Plus Operating Load Plus Wind (for steel listed as item "a" above)	0.80 Fy	0.533 Fy	Varies with Slenderness Ratio	0.80 Fy* to 0.88 Fy	53,300 (0.667 Fy)	29,300 (0.37 Fy)	26,700 (0.74 Fy)	20,000 (0.56 Fy)

- 0.60 Fy or 0.80 Fy is reduced for members with excessive unbraced compression flange length in accordance with AISC Specifications.

TABLE 3.8-6
(Sheet 1 of 1)

TURBINE BUILDING AND REACTOR
PERMISSIBLE LIVE FLOOR LOADS

TURBINE BUILDING

Area	Elevation	Live Load (psf)
Basement	0'-0", 3'-6"	350
Heater Bay	23'-6"	300
Railroad Track	23'-6"	Flatcar with generator and cribbing
Other Areas of Mezzanine Floor (except between Column Lines 2 & 3 and D & F)	23'-6"	250
Mezzanine Floor Between Column Lines 2 & 3 and D & F	23’-6”	125
Condensate Demineralizers	28'-0"	350
Cable Vault	35'-0"	200
Control Room	46'-6"	175
Operating Floor (Enclosed)	46'-6"	600
Open Deck (East), Columns 1-2 & 3-4	46'-6"	600
Open Deck (East) Columns 2-3	46’-6”	400
Open Deck (East), Remainder	46'-6"	200
Open Deck (West), Columns 7-10	46'-6"	200
Open Deck (West), Remainder	46'-6"	30
3Upper Cable Room	58'-0"	150
Roofs	- - -	30
Roof	110'-7"	25
Stairs and Landings	- - -	100

REACTOR BUILDING

Columns A - C, 4 - 7; D - F, 1 - 2	119'-3"	1000
Remainder	119'-3"	800
New Fuel Storage	95'-3"	800
Floor of Storage Pool		2000 (including water)
Floor of Spent-Fuel Pool		4500 (including water)
Remainder	95'-3"	400 plus 20K conc.
	75'-3"	400
	51'-3"	400
	38'-0"	250
Columns A - C, 2 and 3	23'-6"	Flatcar with 300 K capacity
Remainder		250
Stairs and Landings		2 100

TABLE 3.8-7
(Sheet 1 of 1)
LOAD COMBINATIONS FOR CONCRETE STRUCTURES

Allowable Stresses for:

a) Reactor Building Floor Slabs, Beams, Columns, Walls, Storage, Pools and Foundations
b) Control Room Slabs and Walls
c) Battery Room Slab and Walls
d) Emergency Diesel Generator and Tank Vault
e) Intake Structure (Service Water and Circulating Water Pump Areas)
f) Startup Transformer Foundation
g) Exhaust Tunnel & Fan Foundation+

	Loading Condition	Reinforced Steel Maximum Allowable Tension Stress	Reinforced Steel Maximum Allowable Compression Stress	Concrete Maximum Allowable Compression Stress	Concrete Maximum Allowable Shear Stress	Reinforced Concrete Shear Walls Allowable Unit Stress	Concrete Maximum Allowable Peripheral Shear	Concrete Maximum Allowable Bearing
1.**	Dead Load Plus Live Load Plus Operating Load Plus Design Earthquake (OBE)	0.5 Fy	0.34 Fy	0.45 f'_c	1.1 $(f'_c)^{1/2}$	0.05 f'_c	2 $(f'_c)^{1/2}$	0.25 f'_c (full A) 0.375 f'_c(<1/3 A)
2.**	Dead Load Plus Live Load Plus Operating Load Plus Wind	0.667 Fy	0.454 Fy	0.60 f'_c	1.467 $(f'_c)^{1/2}$		2.667 $(f'_c)^{1/2}$	0.333 f'_c (full A) 0.50 f'_c (<1/3 A)
3.***	Dead Load Plus Live Load Plus Operating Load Plus Double Design Earthquake (SSE)	0.9 Fy	0.6 Fy	0.9 f'_c	1.7 $(f'_c)^{1/2}$		3.4 $(f'_c)^{1/2}$	0.333 f'_c (full A) -.50 f'_c (<1/3A)

* Although the intake structure is defined as a Class II structure, the supporting slabs and walls for the service water and circulating water pumps have been investigated and have been found adequate to withstand seismic loads of 0.11g and 0.22g using the tabulated allowable stresses.

** Loading Conditions 1 and 2 are for Normal Load Conditions (Service Load Conditions)

*** Loading Condition 3 covers Unusual Load Conditions (Factored Load Condition)

\+ Although the Exhaust Tunnel & Fan Foundation is defined as a Class II structure, the structure has been investigated and has been found adequate to withstand seismic loads of 0.092g for OBE and 0.184g for SSE.

TABLE 3.8-8
(Sheet 1 of 1)

LOAD COMBINATIONS FOR STEEL STRUCTURES

Allowable Stresses for Structural Steel for:
a) Reactor Building Roof Steel, Craneway Columns, Crane Girders, Vertical Bracing
b) Drywell Radial Steel Framing and Recirculating
c) Reactor Building Platforms

	Loading Condition	Tension on Net Section	Shear on Gross Section	Compression	Bending Tension and Compression	A325 U.S. Bolts With Thread Excluded From Shear Plane Tension (psi)	A325 U.S. Bolts With Thread Excluded From Shear Plane Shear (psi)	A141 Rivets Tension (psi)	A141 Rivets Shear (psi)
1.**	Dead Load Plus Live Load Plus Operating Load Plus Design Earthquake (OBE)	0.60 Fy	0.40 Fy	Varies with Slenderness Ratio	0.60 Fy* to 0.66 Fy	40,000 (0.50 Fy)	22,000 (0.27 Fy)	20,000 (0.56 Fy)	15,000 (0.42 Fy)
2.	Dead Load Plus Live Load Plus Operating Load Plus Wind (for steel listed as Item "a" above	0.80 Fy	0.533 Fy	Varies with Slenderness Ratio	0.80 Fy* to 0.88 Fy	53,300 (0.667 Fy)	29,300 (0.37 Fy)	26,700 (0.74 Fy)	20,000 (0.56 Fy)

* 0.60 Fy or 0.80 Fy is reduced for members with excessive unbraced compression flange length in accordance with AISC Specifications.

** Loading Conditions 1 and 2 represent Normal Load Conditions (Service Load Conditions)

TABLE 3.8-9
(Sheet 1 of 1)

TORNADO WIND LOAD COMBINATIONS

Load combinations used to determine the effect of tornado wind induced loads. These loading conditions are considered Unusual Load Conditions (Factored Load Conditions)

1. DL + LL + TL + Ww

2. DL + LL + TL + Wp

3. DL + LL + TL + Ww + 0.5 Wp

DL = Dead Loads
LL = Live Loads
TL = Thermal Loads
Ww = Tornado Wind Pressure
Wp = Tornado Generated Differential Pressure

Live loads and thermal loads are included in the load combinations only if they made the total loads more critical.

The local and overall effects of tornado generated missiles are considered.

TABLE 3.8-10
(Sheet 1 of 1)

NEW RADWASTE BUILDING LOAD COMBINATIONS

The stress resultants (axial loads, moments, and shears) obtained from each of the loads considered in design are combined to simulate the worst credible combinations of loadings. These loading combinations, including associated load factors, are as follows:

a. Concrete Structures:
Normal Load Conditions (Service Loads)

1. 1.4D + 1.4F + 1.7L + 1.7H
2. 1.4D + 1.4F + 1.7L + 1.7H + 1.9E
3. 1.4D + 1.4F + 1.7L + 1.7H + 1.7W
4. (1.4D + 1.4F + 1.7L + 1.7H + 1.7T + 1.7R)
5. (1.4D + 1.4F + 1.7L + 1.7H + 1.9E + 1.7T + 1.7R)
6. (1.4D + 1.4F + 1.7L + 1.7H + 1.7W + 1.7T + 1.7R)
7. 1.2D + 1.9E
8. 1.2D + 1.7W
9. 0.9D + 1.4F
10. 0.9D + 1.7H

Unusual Load Conditions (Factored Loads

11. $D + L + F + H + T + R + E^1$
12. $D + L + F + H + T + R + W^1$
13. $9D + E^1$
14. $9D + W^1$

Both cases of "L" having its full value or being completely absent are considered.

b. Steel Structures:

Steel Structures are not considered as being seismic Category 1

Nomenclature definitions for loads are given in Table 3.8.11.

TABLE 3.8-11
(Sheet 1 of 1)

NOMENCLATURE DEFINITIONS FOR TABLE 3.8-10 LOADS

(New Radwaste Building)

D - Dead loads of the structure and all other permanent loads including buoyant pressure from design flood where applicable.

L - Live loads on floors and roof including moveable equipment loads, piping, cable trays and any other loads which vary in intensity and occurrence.

T - Thermal effects and loads during normal operating or shutdown conditions.

R - Pipe reactions during normal operating or shutdown conditions based on the most critical transient or steady-state condition.

H - Lateral earth pressure and surcharge.

F - Lateral pressure from liquids including design flood.

W - Loads generated by the design wind.

E - Loads generated by the operating basis earthquake.

E^1 - Loads generated by the safe shutdown earthquake.

W^1 - Loads generated by the design tornado. Tornado loads include loads due to the tornado wind pressure and tornado created differential pressure.

TABLE 3.8-12
(Sheet 1 of 1)

ALLOWABLE STRESSES FOR CONCRETE VENTILATION STACK

	Loading Condition	Reinforced Steel Maximum Allowable Tension Stress	Reinforced Steel Maximum Allowable Compression Stress	Concrete Maximum Allowable Compression Stress	Concrete Maximum Allowable Shear Stress
1.	Dead Load Plus Wind Load	0.30 Fy	-	0.375 f'_c	1.1 $(f'_c)^{1/2}$
2.	Dead Load Plus Wind Plus Temperature	0.54 Fy	-	0.67 f'_c	1.1 $(f'_c)^{1/2}$
3.	Dead Load Plus Design Earthquake Load (OBE)	0.30 Fy	-	0.375 f'_c	1.1 $(f'_c)^{1/2}$
4.	Dead Load Plus Design Earthquake (OBE) Plus Temperature	0.54 Fy	-	0.67 f'_c	1.1 $(f'_c)^{1/2}$
5.	Dead Load Plus Double Design Earthquake (SSE)	0.96 Fy	-	0.67 f'_c	1.1 $(f'_c)^{1/2}$

NOTES:
- a. Maximum wind velocity = 110 mph
- b. Seismic loads per J. A. Blume's "Earthquake Analysis: Ventilation Stack."
- c. Temperature = 100^{o}F maximum gradient
- d. Loading Conditions 1 through 4 are for Normal Load Conditions (Service Load Conditions)
- e. Loading Condition 5 covers Unusual Load Conditions (Factored Load Condition)

TABLE 3.8-13
(Sheet 1 of 1)

ALLOWABLE STRESSES FOR DRYWELL CONCRETE SHIELD

	Loading Condition	Reinforced Steel Maximum Allowable Tension Stress	Reinforced Steel Maximum Allowable Compression Stress	Concrete Maximum Allowable Compression Stress	Concrete Maximum Allowable Shear Stress	Concrete Maximum Allowable Peripheral Shear
1.	Dead Load Plus Live Load Plus Overpressure Plus Maximum Temperature Plus Design Earthquake	0.5 Fy	0.34 Fy	0.45 f'_c	1.1 $\sqrt{f'_c}$	2 $\sqrt{f'_c}$
2.	Dead Load Plus Live Load Plus Maximum Temperature Plus Overpressure Plus Double Design Earthquake	0.5 Fy	0.34 Fy	0.45 f'_c	1.467 $(f'_c)^{1/2}$	2.677 $(f'_c)^{1/2}$
3.	Dead Load Plus Live Load Plus Maximum Temperature Plus Design Earthquake Plus Jet Force	0.667 Fy	0.454 Fy	0.60 f'_c	1.467 $(f'_c)^{1/2}$	2.677 $(f'_c)^{1/2}$

NOTES:
a. Dead loads and live loads include contributing maximum loadings from building floors, columns and walls, pool walls, slabs and plug.
b. Overpressure = Maximum 20 psi reaction from compressible joint.
c. Temperature = 55°F maximum gradient.
d. Design earthquake = Loads due to 0.11g basic ground acceleration and includes proportions of the drywell vessel, the surrounding building, the reactor vessel and reactor building equipment.
e. Jet Force = 566 kips over a 3.14 ft^2 area at any point of the spherical portion and
466 kips over a 2.54 ft^2 area at any point of the cylindrical portion (below El. 94' - 8").

TABLE 3.8-14
(Sheet 1 of 1)

REACTOR BUILDING DESIGN CRITERIA AND STRESSES

	Stress (psi)			Deflection (in.)			Foundation	
	Concrete		Steel					
	Shear	Flexural	Tension	Conc.	Steel	Total	(KSF) Load	F.S.
Evaluation Criteria	50	450	7350	--	--	1.8	35.2	3.0
Biological Shield Wall	6.7	18		--	--	--	--	--
Exterior Walls	13.0	48		--	--	--	--	--
Diagonal Bracing	--	--	3000	--	--	--	--	--
Concrete at El. 119'-3"	--	--	--	0.017	--	--	--	--
Steel at El. 168'-0"	--	--	--	--	0.252	--	--	--
Total Defl. At El. 168'-0"	--	--	--	--	--	0.27	--	--
Foundation Loading	--	--	--	--	--	--	--	--
D.L. + 0.8 LL Wind Load (max)							8.75 ±0.44	
Total Foundation	--	--	--	--	--	--	9.19	3.84

Table 3.8-15
(Sheet 1 of 1)

Maximum Permissible Wind Speeds

STRUCTURE		WIND SPEED (MPH)
Concrete Portion of Reactor Building		300(+)[3]
Reactor Building Steel Superstructure		190
Reactor Building Metal Siding		190
Control Room - North Wall		160[1]
Control Room - Remainder		300(+)[3]
Intake Structure		300(+)[3]
Ventilation Stack		180
Diesel Generator & Oil Tank Vaults		168[1]

NOTES:

(1) Permissible Wind Speed Governed by Tornado Missile Loads
(2) All Structures are concrete unless otherwise noted in the Table
(3) The probability of a 300 mph wind speed is 1×10^{-7} per year. Therefore, speeds in excess of 300 mph are not considered in the analysis.

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 System Design Basis

The original design of the Oyster Creek Nuclear Generating Station (OCNGS) incorporated design methods, codes and standards which were current as of that time and which were acceptable to the governing regulatory authorities.

3.9.2 Seismic and Operating Cycles Design

During plant life, systems and components will be subjected to various operating cycles. These conditions were evaluated to determine their effect on the equipment. The operating cycles and their effects are discussed in Section 5.2.

The seismic design classification for major systems and components is discussed in Section 3.2, and the seismic input loads and accelerations are discussed in Section 3.7. The design of Category I (seismic) structures is presented in Section 3.8.

3.9.3 ANSI/ASME Code Class I Components

3.9.3.1 Analytical Procedures for Piping

The eleven Class I (seismic) pipe systems, with the exception of buried pipe, as specified in Section 3.9.3.5 must satisfy the design stress requirements specified in Section 3.9.3.1.1 through Section 3.9.3.1.4.

3.9.3.1.1 Design Code

The Design Code for pipe and for Component Standard Supports will continue to be B31.1-1955 with supplements, addenda and applicable code cases as of 1966. However, the code for evaluation will be B31.1-1983 through Winter 1984 Addenda, and supplemental criteria. The supplemental criteria are identified in the design code reconciliation report, MPR-1938. These supplemental criteria are exceptions to the requirements of B31.1-1983 through Winter 1984 Addenda that are necessary to maintain compliance with B31.1-1955 with supplements, addenda and code cases through 1966. In addition, Component Standard Supports will also be designed in accordance with MSS-SP-58, 1975.

The Design Code for Field Fabricated Supports and Supplementary Steel will continue to be the AISC Specification 1978. The design code reconciliation report, MPR-1938 discusses that this Code and the original design code are essentially the same.

3.9.3.1.2 Seismic Loads

The seismic loads are specified to be the Operating Basis Earthquake (OBE) and the Safe Shutdown Earthquake (SSE) in accordance with the seismic response spectra, as specified in the UFSAR Section 3.7.3.4. Damping values are those specified in Code Case N-411 or Regulatory Guide 1.61, as specified in the UFSAR Section 3.7.1.3. The vertical spectra are not assumed to be equal to 2/3 of the horizontal. Rather, they are explicitly calculated from the data, as were the horizontal spectra.

3.9.3.1.3 Allowable Stress Values

The FDSAR uses wording very carefully when it discusses structural design issues. Subsection 3.1.2 of Amendment 3 states that the seismic design is based on an earthquake of 0.11g. The facility was designed for this earthquake using the ordinary allowable stress values from the various codes. The FDSAR goes on to describe that the "…design is such that a safe shutdown…" can be achieved during a ground motion of 0.22g. This describes a condition that the structure will survive such an earthquake. The FDSAR discusses the techniques that could be used to evaluate the facility for a 0.22g earthquake in Amendment 3.

Hence, the FDSAR describes two processes; the first is a design as evidenced by the existence of design calculations that qualify the structure for a 0.11g earthquake. The second is a condition that describes that the structure would not fail and/or deflect to the point where safe shutdown could not be achieved for a 0.22g earthquake.

The retrievable FDSAR information contains inconsistencies concerning the design of the four heat removal systems: Core Spray, Containment Spray, Shutdown Cooling and Isolation Condenser. In Subsection 3.1.2 of Amendment 3, the FDSAR describes that the four heat removal systems and the containment are designed to accommodate an earthquake of 0.22g and, indeed, the design calculations for the containment qualify the containment for a 0.22g earthquake. Therefore, one would expect that the design calculations for the four heat removal systems would be to qualify them for a 0.22g earthquake also. However, the design calculations that qualify these systems for the load combinations to B31.1 are not retrievable. The FDSAR is inconsistent because, Amendment 11 presents a response that the design is for 0.11g with stresses held to Sh, and for 0.22g, safe shutdown can be achieved. This response is consistent with the overall design philosophy for the facility; however, it makes no mention of the exception for the four heat removal systems presented in the text cited above. Thus, the inconsistency.

Faced with this inconsistency, the only conservative way to proceed is to provide stress limits consistent with the more conservative of the two design philosophies. Hence, it is assumed, since the calculations are not retrievable, that the designers intended to design the four heat removal systems to a specific B31 allowable stress for a 0.22g earthquake.

In an effort to understand the design intent for "such principal structures" the design for the containment is examined, since that is the other principal structure that the FDSAR states was designed for the 0.22g earthquake. Detailed design bases information for the containment is available.

The Design Specification for the containment states: "The design, fabrication, erection and testing of the vessels shall conform to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section VIII, latest edition and all applicable addenda; Nuclear Case Interpretations 1270N5, 1271N, 1272N5 and all other applicable case interpretations…"

The reference to Code Case 1272N5 is particularly important. This Code Case allows a 10% increase in the Section VIII primary membrane allowable stress for the load combinations that include "…wind, snow, or other specified live loads…" (Reference Paragraph (5) (g) (1)) as long as the design complies with the specified Section VIII allowables for the temperatures and pressures imposed during normal operation (Reference Paragraph (5) (c)). This is an allowance over the nominal code allowable values for the occasionally applied load.

Consequently, it is reasonable to expect that the designers allowed the 1.2Sh for the occasional load combinations, i.e., seismic at 0.22g, specified in B31.1-1955.

Therefore, converting the original design criteria allowable stress discussed above to modern design criteria allowable stresses we have the following.

Allowable stress for pipe for the OBE is 1.0Sh. The allowable forces for OBE for Component Standard Supports will be the vendor supplied catalog values, which will remain compliant with B31.1-1983. The allowable for OBE for Field Fabricated Supports and Supplementary Steel will be nominal allowable from the AISC Code without the Code permitted 33% increase. This complies with the FDSAR Amendment 3 statement that the “ordinary allowable” is used.

Allowable stress for pipe for the SSE is 1.2Sh for the four heat removal systems, Shutdown Cooling, Core Spray, Containment Spray and Isolation Condenser Systems, using the B31.1 Code. The allowable force values for SSE for Component Standard Supports, Field Fabricated Supports and Supplementary Steel for the four heat removal systems will be the allowable values derived for the OBE factored by 1.2 except for Bergen-Paterson hydraulic snubbers, which will remain unfactored. This complies with the FDSAR specific design requirement cited above.

The allowable stress for pipe for the SSE is 2.4Sh for the remaining systems, using guidance from ASME Section NC-3611.2, level D. The allowable force values for SSE for Component Standard Supports, Field Fabricated Supports and Supplementary Steel for the remaining systems will be the allowable values derived for the OBE factored by values found in AmerGen Specification SP-1302-12-295. These were derived using guidance in NRC NUREG-0800 for load combinations involving the SSE. These criteria were selected to provide specific evaluation equations and allowable values for the original design intent of “Safe shutdown can be achieved.” Hence, this complies with the original FDSAR design criteria.

The loading conditions and the associated allowable stresses for Class I (seismic) pipe developed above are shown in detail in Table 3.9-2.

3.9.3.1.4 Dynamic Analysis

Section 3.7.3.2 describes the Dynamic Analysis performed for various types of pipe systems at Oyster Creek.

The Dynamic Analysis of torus attached pipe is governed by the Mark I Containment Long-Term Program rules described in Section 3.8.2 of the UFSAR.

3.9.3.2 Allowable Stresses for Reactor Vessel Supports

The allowable stresses for the reactor vessel supports are shown on Table 3.9-2.

3.9.3.3 Investigation of Stainless Steel Piping

An investigation was conducted of stainless steel critical piping to develop positive identification of all material and verify that these materials meet all characteristics required by the design specifications. This investigation is documented in Reference 1, and summarized briefly in the following paragraphs.

3.9.3.3.1 Background

The verification program was designed to develop positive identification of all material in stainless steel piping and determine whether all the characteristics required by the specifications were met. For purposes of the program and to provide conservatism to the approach, it was assumed that there was zero confidence in the validity of the manufacturer's record data and of the certifications. Accordingly, tests were designed to obtain data that could be compared with data submitted by the manufacturer. In this manner, record validity and the adequacy of the piping could be confirmed or denied.

3.9.3.3.2 Summary of Results

The documentation accounting program identified the potential questionable areas to be 8, 10, 12, and 16 inch piping installed in the Isolation Condenser (piping outside drywell replaced during 13R) System, Core Spray System, and Shutdown Cooling System, because the final radiographs of the seam welds could not be located.

Results of all metallurgical testing for the above piping and two lengths of excess piping confirmed the vendor recertifications that the material conforms to all chemical, physical and heat treatment requirements of ASTM-A-358 specifications.

It should be noted that there was an error in marking of some of the pipe, as it was marked A-312 rather than A-358.

Actual measurements of wall thickness show values 58, 47, 43, 52, and 46 percent in excess of that required by ASME Boiler and Pressure Vessel Code, Section I, for the 8, 10, 12, 14, and 16 inch pipe, respectively.

There is overwhelming evidence to support the vendor's contention that radiography was performed on the seam welds. The finding of the vendors in-process radiographs and the many repair areas found during the reradiography program established beyond question that radiography had been performed at the vendor's plant.

The reradiography program revealed a question whether approximately 9 feet of 12 inch piping contained either a valley between bands, a lack of penetration, or valley between bead and base metal. Triangulation radiography established this condition to be a surface condition on the inside surface of the pipe.

Positive verification of the valley condition was obtained by cutting a 5 inch diameter segment from the installed piping. This demonstrated that it was a surface condition between the weld bead and base metal. The valley was measured as 0.027 inches deep with a bottom radius of approximately 1/64 inch. The removed segment satisfactorily passed a dye penetrant test and a severe bend test across the valley. Polished cross sections of the removed segment showed the entire weld to be sound and free of defects.

Triangulation radiography and ultrasonic soundness examinations were made for all the areas where radiography revealed the evidence of the valley condition. The UT inspection was performed using a 45 degree notch, 0.025 inch deep, in a sample section as the ultrasonic standard. The use of this standard was considered conservative since the UT indication obtained from this standard was approximately 60 percent of the UT indication obtained from the acceptable valley conditions of the removed disc as described above.

Ultrasonic examination revealed three areas where the ultrasonic indication exceeded the standard used and the indication appeared to be linear. These areas were all in spool piece NE-1-6 and were 2 1/2 inches, 6 inches, and 5 1/2 inches in length. All other areas were less than the ultrasonic standard (45 degree V notch, 0.025 inch deep), and consequently less than the indication for an acceptable valley condition.

Maximum possible depth of ultrasonic indications was determined to be 0.159 inch.

Stress analysis revealed the notch effect of the standard valley condition to be an insignificant factor in the fatigue life of the piping.

Stress analysis of the most severe condition located by ultrasonics also showed the condition to be adequate for the service.

However, in order to provide the same conservatism that exists in all the remaining piping, all of the areas where the ultrasonic testing showed the indications to be greater than the surface valley condition were removed and replaced with new pipe lengths.

The total investigation shows that the original piping and fitting installation was satisfactory for its intended service.

3.9.3.4 Experience with Cracking of Stainless Steel Piping

3.9.3.4.1 Isolation Condenser System Piping

The Isolation Condenser System (ICS) is described in Section 6.3. On March 22, 1984 a leak was detected in the ICS Condensate piping during hydrostatic testing of the system. The reactor was in the shutdown mode at the time of the incident.

It was determined that the cracking associated with the ICS stainless steel piping was Intergranular Stress Corrosion Cracking (IGSCC). Appropriate inspection and repairs were performed in a manner acceptable to NRC. The staff concluded in a safety evaluation that the plant could be safely returned to power.

During 13R, all Isolation Condenser piping within the four drywell penetrations and outside the drywell was replaced with IGSCC resistant material. This modification also reduced the number of welds outside the drywell, upgraded the welds between the pipe spools to Category A (Ref. 3) and eliminated the uninspectable reactor coolant pressure boundary welds within the penetrations.

3.9.3.4.2 Feedwater System Nozzles at the Reactor Pressure Vessel

In 1977, inspection of the feedwater nozzle region of the reactor pressure vessel (RPV) identified blend radius and bore region cracks. In the original sparger/thermal sleeve arrangement, leakage occurred. The cooler feedwater leakage mixed with much hotter downcomer flow creating turbulent eddies. The result of this unanticipated mixing was that the nozzle blend radius region, in particular, was alternately wetted by hot coolant then by cooler feedwater at high frequency. High cycle thermal fatigue initiated cracks through the cladding. Normal thermal duty propagated the high cycle initiated cracks into the base metal. Periodic inspection ensures that significant bypass leakage does not develop in the replacement feedwater sparger/thermal sleeve assemblies.

To prevent reoccurrence, a piston ring seal thermal sleeve was installed in each of four feedwater nozzle penetrations. The piston ring seal is intended to reduce leakage between the thermal sleeve and the feedwater nozzle inner diameter. The nozzle region was examined during the 1988 (12R) refueling outage using remote automated ultrasonic equipment. The welds, inner radii and inner bores were examined from external vessel and nozzle surfaces. No reportable indications were detected.

3.9.3.4.3 Augmented Inspections of Stainless Steel Piping

IGSCC inspection and overlay repairs are presently performed at each refueling outage. Commencing with the 13R outage, augmented inspections are performed in accordance with TR-050 (Ref. 3). The inspection program was submitted to the NRC for approval prior to implementation, and the results of the inspection and repairs program are evaluated by the NRC prior to restart.

3.9.3.5 As-Built Seismic Analyses for Safety Related Piping Systems

IE Bulletin 79-14 required verification that the seismic analysis for the safety related systems reflected as-built configuration of the piping. Prior to IE Bulletin 79-14, other generic seismic issues were identified which could cause seismic analyses of safety related piping systems to yield non conservative results.

One issue involved algebraic summation of loads in some seismic analyses, the other involved the accuracy of the information input to seismic analyses, particularly relative to pipe supports and valve weights. These issues had been addressed in IE Bulletins Nos. 79-02, 79-04 and 79-07. The discussions covered in IE Bulletin No. 79-04 ("Incorrect Weights for Swing Check Valves Manufactured by Velan Engineering Corporation", March 30, 1979) and IE Bulletin No. 79-07 ("Seismic Stress Analysis of Safety Related Piping", April 14, 1979) were found to be not applicable to OCNGS.

However, IE Bulletin 79-02, "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts", dated March 8, 1979 and revised and supplemented on June 21, 1979, on August 20, 1979, and on November 8, 1979, required mathematical verification of loads in piping analyses and/or a testing program for anchor bolts. IE Bulletin 79-14 was subsequently supplemented on August 15, 1979 and September 7, 1979.

In response to IE Bulletins 79-02/14, GPUNC initiated a reanalysis and field verification program. The eleven piping systems within the program were:

1. Liquid Poison
2. Shutdown Cooling
3. Core Spray
4. Emergency Service Water
5. Control Rod Drive/Scram Discharge Volume
6. Containment Spray
7. Isolation Condenser
8. Feedwater
9. Cleanup Demineralizer
10. Main Steam
11. Reactor Recirculation

3.9.3.6 Thinning of Pipe Walls

Since 1978, GPUN has investigated small cracks and wall reduction of various pipes at the OCNGS.

Component selection for thickness measurements is based, in part, on the following criteria:

1. Process Fluid: single phase, steam and two phase with temperature above 200°F, pH and oxygen content (all systems included--no exclusions made), and moisture content (2% and above)

2. Pipe Material (carbon steel, lack of alloying metals such as Chromium, Molybdenum, and Copper)

3. Geometry (each system is evaluated for arrangement of components-configurations conducive to flow disturbances)

4. Fluid Velocity (all systems included - prioritization of inspection selections emphasizes components with higher velocities)

5. Additional inspection prioritization criteria include:

    a. System Category (e.g., required for shutdown; prevention of challenges to safety systems)

    b. Operating Frequency

    c. Design Margin (based on the ratio between nominal and minimum design pipe wall thickness)

    d. Maintenance Experience

Details of inspections and examinations under the OCNGS Pipe Wall Thinning Program have been periodically reviewed by the NRC.

3.9.4 Control Rod Drive System

This subsection describes the Control Rod Drive System which includes the control rod drive mechanisms, the hydraulic supply subsystem, the scram subsystem and the cooling subsystem.

For a description of the Scram Subsystem and its backup, the Alternate Rod Injection System, refer to Subsections 3.9.4.2.3 and 3.9.4.2.4, respectively.

3.9.4.1 Control Rod Drive Mechanisms

3.9.4.1.1 General Description

The basic drive mechanism is a double acting, mechanically latched, hydraulic cylinder using reactor grade water as the operating fluid. It is identified as Model 7R-DB-144 and is shown on Figure .9-1. The original Model 7R-DB-144-A1 has been replaced in their entirety by Models 7R-DB-144-B and E and these models are currently being replaced by Model 7R-DB-144-F.

The individual drives are mounted below the reactor vessel, where they position the control rods in the reactor core. Bottom entry control rods are used because they provide axial flux shaping for greater fuel economy. Being bottom mounted, the drives do not interfere with refueling, and are operative even when the head is removed from the reactor. The use of reactor water as the operating fluid eliminates the need for special hydraulic systems. Drives are able to utilize simple piston seals with leakage into the reactor coolant, thus minimizing contamination and helping to cool the mechanism.

The drive is capable of inserting or withdrawing the rod at a slow, controlled rate as well as providing rapid insertion in an emergency. A locking mechanism (the collet) allows a drive to be positioned at short increments of stroke (6 inches) and will hold the rod in a fixed position.

Each drive is an integral unit wholly contained in a housing (actually a reactor vessel extension) protruding below the reactor vessel. The lower end of each drive housing terminates in a flange which mates with the drive flange. In order to allow removal of the drive without disturbing external hydraulic connections, the lines are welded to the top and continue through to the lower face of the housing flange, where they are sealed with static face seals. These seals, together with the reactor static seal, are compressed by the eight mounting bolts used to attach the drive to the housing flange.

The control rod drive mechanism is similar in operating principle and general construction to those used in Commonwealth Edison's Dresden Unit 1, Consumers Power's Big Rock Point Plant, and Pacific Gas and Electric's Humboldt Bay No. 3 in the United States, and in the SENN plant in Italy. Similar drives, except for stroke length, are also used in the KRB plant in Germany. Improvements in the design of the Control Rod Drive System for this plant over previous designs are as follows:

a. Nitrided Type 304 stainless steel is used for the index tube, replacing 17-4 PH1100. Nitrided Type 304 has superior resistance to galling.

b. The guide sleeve, a part which has been the source of friction and abrasion in some drive mechanisms, is eliminated.

c. Additional screens are included in the design to prevent the entrance of foreign material into the drive.

d. The operating forces and collet return forces are higher.

e. The unit are longer, to accommodate the 144 inch control rod travel.

f. All tubular members which are subject to column loading are larger in diameter for greater column strength.

g. Other material changes and refinements in material processing and structural improvements have been made.

3.9.4.1.2 Principle of Drive Operation

The drives operate using condensate at 1250 psig as the actuating hydraulic medium. Water is taken from the water quality line at the 8 inch condensate pump discharge or from the Condensate Storage Tank and is raised in pressure to approximately 1500 psig by the control

rod drive pump which has a capacity of about 100 gpm. This flow, for use in the drive system, is only a small percentage of the total feedwater flow. After filtering, the pressure is regulated by standard water pressure regulators to approximately 1400 psig for scram accumulator charging, to about 250 psig above reactor pressure for normal drive actuating and to approximately 20 psig above reactor pressure for drive cooling. Normally each drive requires between 0.2 and 0.3 gpm for cooling and about 4 gpm when moving at shim speed. The additional pump capacity is an allowance for increased cooling requirements for regulation, and to provide for accumulator charging. Water in excess of the drive requirements and water exhausted during drive operation to the exhaust header is discharged to the reactor.

The basic principles of drive operation and construction are shown in Figure 3.9-2. This figure, used to illustrate the following explanation, is not the actual mechanical arrangement but illustrates the functional elements of the drive and hydraulic system.

The four valves labeled "insert valves" and " withdraw valves" make up a four way, closed center, reversing valve which can accomplish the following modes of drive operation:

a. Apply driving pressure below the piston and connect the above piston area to the exhaust header.

b. Apply driving pressure over the piston and connect the below piston area to the exhaust header.

c. Shut off all driving pressure when no motion is required.

Table 3.9-3 describes the flow modes for each CRD system operation.

Rod Insertion

Drive insertion is accomplished by opening the two "insert" valves. This applies driving pressure to the bottom of the piston, and opens the chamber above the drive piston to the exhaust header and thence to the reactor. Driving pressure is reactor coolant pressure plus 250 psi. The 250 psi differential acting on the drive piston area of four square inches exerts an upward force of up to 1000 pounds. This force is greater than the drive friction force (normally less than 100 lbs) plus the weight of the control rod and index tube, so the drive inserts the rod.

The construction of the latch is such that it is cammed open and acts as a ratchet during rod insertion. The speed at which the drive moves is determined by the pressure drop through the insert speed control valve. During normal motion, this pressure drop accounts for all but 80 to 90 psi of the operating differential pressure. However, if the drive slows down for any reason, the full differential pressure is available for continued insertion.

Rod Withdrawal

Drive withdrawal is, by design, more involved. First, in the actual drive design, the latch must be raised to reach the unlocked position (rather than withdrawn horizontally as Figure 3.9-2 indicates). The latch piston (or collet piston) area is small enough that this is impossible when opposed by the latch return spring, drive line weight, and the force of the driving pressure applied to the area above the drive piston. Second, the notches in the index tube and the latch mechanism are shaped so that downward forces on the index tube hold the latch in place. The index tube, therefore, must be lifted before the latch can be released. This is done by opening the drive "insert" valves (in the manner described in the preceding paragraph) for approximately

one second; an automatic sequence timer is provided. The "withdraw" valves are then opened (by the sequencing timer mechanism) applying driving pressure above the main piston and opening the area below the piston to the exhaust header. Pressure is simultaneously applied to the latch piston to release the latch.

This pressure must be set and maintained high enough to overcome the force of the latch return spring plus the force of reactor pressure opposing movement of the latch piston. When this occurs, the drive is unlatched and free to move in the "withdraw" direction. Water displaced by the drive piston flows out through the withdraw speed control valve, which regulates the speed at which the drive moves. The speed control valves are used to provide separate insert and withdraw speed adjustments. The entire valving sequence is automatically controlled and is initiated by a single operation of the operating switch.

Scram Actuation

During a scram, a separate set of valves comes into play. These valves, the inlet and exhaust scram valves, open, admitting the pressure in the accumulator (1050 to 1150 psi) under the main drive piston and venting the area over this piston to one of two discharge volumes which are maintained at atmospheric pressure during normal plant operation. The large differential pressure (initially about 1100 psi and always several hundred psi-depending on reactor pressure) produces a large upward force on the index tube and control rod, giving the rod a high initial acceleration and providing a large margin of force to overcome any possible friction or binding. This initial scram force is a maximum of 5600 lbs under cold reactor conditions and 2800 lbs when the reactor is at operating pressure. The characteristics of the hydraulic system are such that after the initial acceleration (less than 30 milliseconds after start of motion) the desired scram velocity of about 5 ft/sec is achieved and the drive travels at a fairly constant velocity. This characteristic provides a high initial rod insertion rate and a high operating force margin that cannot be achieved by a drive designed to utilize gravity forces. As the drive piston nears the top of its stroke, the piston seals close off the large passage in the exhaust line, and the drive is slowed down. The illustration (Figure 3.9-2) shows only two passages; in the drive mechanism there are eight which are progressively closed, providing a more gradual deceleration.

Each drive requires about 2.5 gallons of water during the scram stroke. There is adequate water capacity in each drive's accumulator to complete a scram in the required time at low reactor pressures. At higher reactor pressures, the accumulator is assisted by reactor pressure reaching the drive through the ball check valve. As water is drawn from the accumulator, the accumulator discharge pressure falls below reactor pressure. This causes the check valve to shift its position to admit reactor pressure under the drive piston. Thus, reactor pressure furnishes the force needed to complete the scram stroke at higher reactor pressures, while the accumulator alone will accommodate the low pressure scrams. When the reactor is up to full operating pressure, the accumulator is actually not needed to meet scram time requirements. With the reactor at 1000 psig, the scram force is still over 1000 pounds without an accumulator.

3.9.4.1.3 Mechanical Arrangement

The actual mechanical arrangement of the drive is discussed in the next section and illustrated in Figure 3.9-1. In comparing this arrangement with the schematic in Figure 3.9-2, one should note the following:

a. A conventional hydraulic cylinder, such as is shown in the schematic, is normally operated with a fixed cylinder and a moving piston and piston rod. To be double acting, the piston rod must pass through a seal or rod packing. The locking grooves shown on the schematic obviously cannot be passed through a seal, so the normal arrangement is reversed in the actual drive. The fixed stop piston and piston tube correspond to piston and piston rod. The index tube and drive piston correspond to cylinder and rod seal, with these being the moving members. Because of column loading and water flow considerations, external seal rings and an outer cylinder have been added to the "conventional" arrangement. These allow driving pressure to be routed through the flange to the area below the drive piston for rod insertion. The area inside the index tube and above the stop piston is open to reactor pressure in the actual drive rather than utilized for driving, as is illustrated in Figure 3.9-2.

b. The stationary piston rod (piston tube) is hollow. A hydraulic passage is provided inside this tube to carry water to and from the area above the drive piston. The progressive orifice drilling for scram deceleration is in this member.

c. The latch mechanism is made up of six fingers attached to an annular piston operating in an area at the upper end of the drive. Spring action in the fingers themselves hold them against the index tube, except when pressure is applied to the latch (or collect) piston to hold them in the unlocked position.

d. The annular space between the drive and the housing becomes the hydraulic passage which connects reactor pressure to the ball check valve. This assures the reactor pressure is always available at the lower end of the piston (for a scram), and that the drive can be actuated in the withdraw direction only by pressures higher than reactor pressure. The integrity of this passage is vital to the fail-safe properties of the drive. A second annular passage is provided in the outer or drive cylinder structure, which is a double walled tube. This passage carries the operating pressure on the upper side (rather than a separate line to the reactor as shown on the schematic).

e. The schematic in Figure 3.9-2 does not show the position indicator probe, nor the filters which are located in all passages through which reactor water flows to enter the drive mechanism. These and other details are shown in Figure 3.9-1 and explained below.

A more representative simplified sketch of the control rod drive mechanism is shown in Figure 3.9-1.

3.9.4.1.4 <u>Description of Drive Mechanism</u>

<u>Drive Piston and Index Tube</u>

The main drive piston is mounted at the lower end of the index tube, which functions as a piston rod. These parts (drive piston and index tube) make up the main moving assembly in the drive. The drive piston operates between positive end stops, with a hydraulic cushion provided at the upper end only. The piston has both inside and outside seal rings, operating in an annular space between an inner and outer cylinder. The inner piston rings are conventional radial tangential packing sets with coil springs. The upper seal set is used for cushioning the drive at

the upper end of the stroke. As this type of seal is effective in only one direction, the lower sets of seal rings are mounted with one set sealing in each direction. A pair of bushings is provided to prevent metal to metal contact between the piston assembly and the inner cylinder surface. The outer piston rings are segmented step cut seals with expander springs holding the segments against the cylinder wall. A pair of split bushings on the outside of the piston prevents contact with the cylinder wall. The effective area under the piston, used during "up" travel or "insert", is 4.1 square inches. The effective area for "down" travel or "withdraw" is about 1.2 square inches. This difference in driving area tends to balance out the rod weight and makes it possible always to have higher "insert" forces than "withdraw" forces.

The upper end of the index tube is threaded to receive a coupling spud. The coupling, shown in Figure 3.9-3, is designed to accommodate a small amount of angular misalignment between the drive and control rod. Six spring fingers allow the coupling spud to enter the mating socket on the control rod. A lock plug then enters the spud and prevents uncoupling. Two means of uncoupling are provided. The lock plug may be raised, against the spring force of approximately 50 lbs, by a rod extending up the center of the control rod to an unlocking handle. The control rod, with the unlocking plug raised, can then be lifted from the drive. The locking plug may also be lifted from below, if it is desired to uncouple a drive without removing the vessel head for access. In this case, the central portion of the drive is raised to lift the uncoupling rod assembly. This rod lifts the lock plug and allows the coupling spud to disengage the socket as the drive piston and index tube are driven down. Note that a control rod weight of 100 lbs or higher is sufficient to force the spud fingers to enter the socket and push the lock up, allowing the spud to enter the socket completely and the plug to snap back into place. Therefore, the drive can be coupled to the control rod using only the weight of the control rod. However, with the lock plug in place, a force in excess of 50,000 lbs is required to pull the coupling apart.

The index tube is a long hollow shaft. This tube has locking grooves spaced every six inches along the outer surface. These grooves transmit the weight of the control rod to the locking device (collet fingers). The inside of the index tube above the stop piston is open to reactor pressure through the coupling spud, thus allowing the effect of reactor pressure to be confined to a small area--roughly the cross sectional area of the index tube. A closely woven stainless steel screen prevents the entry of foreign particles with water flowing into this space.

<u>Locking Mechanism</u>

A ratchet type locking device is located in the upper end of the drive mechanism. This device requires a hydraulic pressure higher than reactor pressure to unlock for "down" movement. Due to the cam action of the index tube locking grooves, no unlocking signal is needed for "up" movement.

Locking is accomplished by six fingers mounted on the collet piston at the top of the drive cylinder. In the locked or latched position the fingers engage a locking groove in the index tube and thus carry the weight from the index tube to the outer drive cylinder. The collet fingers are hard surfaced for wear resistance. The collet piston is normally held in this position by a force of approximately 150 lbs supplied by a spring. Metal piston rings are used to seal the collet piston from reactor pressure. A pressure approximately 100 psi above reactor pressure is required to overcome spring force, slide the collet up against the conical surface in the guide cap, and spread the fingers out so that they do not engage the locking grooves. The collet piston is nitrided to allow rubbing against the surrounding cylinder surfaces.

Fixed in the upper end of the drive assembly is a guide cap. This member provides the unlocking cam surface for the collet. It also serves as the upper bushing for the index tube and is nitrided to provide a compatible bearing surface for the index tube. Mounted on the guided cap is a filter through which water passes when it is drawn down into the drive during a scram at elevated reactor pressures.

Piston Tube and Stop Piston

Extending up inside the drive piston and index tube is an inner cylinder or column called the piston tube. This cylinder is fixed to the bottom flange of the drive and remains stationary during rod movement. Water is brought to the upper side of the drive piston through this tube. A series of orifices at the top of the tube provides a progressive water shutoff, thus cushioning the drive piston at the upper end of its scram stroke.

A stationary piston, called the stop piston, is mounted on the upper end of the piston tube. This piston provides the seal between reactor pressure and the area above the drive piston. It also functions as a positive end stop at the upper limit of rod travel. A stack of spring washers just below the stop piston helps absorb the final mechanical shock at end of travel. The piston rings are similar to the outer drive piston seals, being segmented, step cut seal rings. The spring used to expand the seal is, in effect, a seal ring also, sealing the gaps between ring segments. The piston rings are arranged in two pairs, with a bleed off passage to the center of the piston tube. This arrangement allows seal leakage from the reactor (during a scram) to be bled directly to the discharge line, rather than to the critical area above the drive piston. The lower pair of seals is used only during the cushioning of the drive piston at the upper end of the stroke.

Position Indicator

The center tube of the drive mechanism is a well containing the position indicator probe. The position indicator probe is an aluminum extrusion attached to a cast aluminum housing. Mounted on the extrusion are hermetically sealed magnetically operated position indicator switches held by spring clips. Each switch is sheathed in a braided glass sleeve, and the entire probe assembly is protected by a thin walled stainless tube. The switches are actuated by a ring magnet carried at the bottom of the drive piston. The drive piston, piston tube and indicator tube or well are all of non magnetic stainless steel, allowing the individual switches to be operated by the magnet as the piston passes. One switch is located at each position corresponding to an index tube groove, thus allowing indication at each latching point. One switch is located at each midpoint between latching points, allowing indication of this intermediate point during drive motion. Duplicate switches are provided for full-in and full-out. One additional switch (an overtravel switch) is located at a level below the normal full-out position. As the limit of "down" travel is normally provided by the control rod as it reaches the backseat position, a drive can pass this position only if uncoupled. The overtravel switch is thus a convenient means to verify that the drive and rod are coupled after installation of a drive or at any time during plant operation.

Flange and Cylinder Assembly

Welded to the drive cylinder is a heavy flange. A sealing surface on the upper face of this flange is used in making a static seal to the drive housing flange. Stainless steel rings are used for these seals. In addition to the reactor vessel seal, two hydraulic control lines to the drive are sealed at this face. A drive can thus be replaced without removing the control lines, which are permanently welded into the housing flange. The drive flange contains the integral ball or two-way check valve. This valve is so situated as to direct reactor pressure or driving pressure,

whichever is higher, to the underside of the drive piston. Reactor pressure is admitted to this valve from the annular space between the drive and the housing through passages (not shown) in the flange. An additional screen is provided to intercept foreign material at this point.

The outer cylinder is double walled to provide an annular passage for water used to operate the collet piston. The inner tube is honed to provide the surface required for the drive piston seals.

Both the cylinder tube and outer shell are welded to the drive flange, but the top of the tubes have a sliding fit to allow differential expansion to take place. The latch housing, welded to the outer shell, is provided with ports to allow free passage of water from the clearance space between index tube and cylinder wall.

3.9.4.2 Control Rod Drive Hydraulic System

3.9.4.2.1 System Description

The Control Rod Drive Hydraulic System which supplies and controls the pressure and flow combinations to the drives consists of three subsystems:

a. Supply Subsystem

b. Scram Subsystem

c. Cooling Subsystem

There is one supply subsystem for all the control rod drives of a system but each drive has a separate and independent scram subsystem and cooling subsystem. A simplified sketch of the control rod drive hydraulic control unit, scram discharge volumes, and scram air valving is shown in Figure 3.9-6.

A control rod drive hydraulic control unit piping and instrumentation drawing are shown in Drawing GE237E487. An illustration of the hydraulic control unit is provided in Figure 3.9-5, and a piping and instrumentation drawing of the Hydraulic Control Unit is provided in Drawing GE197E871. The main components of the system are identified by equipment piece numbers, the numbers below the line in the component identification circles. Note that piping, pipe fittings and standard valves are not identified by this means.

3.9.4.2.2 Supply Subsystem

This subsystem is made up of supply pumps, filter, control valves, and associated instrumentation and controllers. In general, the supply subsystem takes water from the CRD water quality line, from either the condensate pump discharge or the Condensate Storage Tank, pressurizes it, filters it, and with three pressure stages regulates its three output pressures. The first pressure stage supplies the highest pressure, nominally 1400 psig, to the scram subsystem for charging the accumulators independent of reactor pressure. The second and third stage pressures vary directly with reactor vessel pressure changes. This is accomplished by using control valves acting as orifices which develop constant pressure drops due to the constant flow from the first stage. The second and third pressure stages are connected in series and the outlet of the third stage exhausts to reactor pressure. Therefore, both these pressure stages will vary with reactor pressure changes. The second pressure stage is adjusted to hold a differential pressure nominally at 250 psi above reactor pressure and supplies water for normal

drive operation. The third stage is adjusted to hold a differential pressure nominally at 20 psi above reactor pressure. These three supply pressures plus reactor vessel pressure in the exhaust water header are the four operating pressures of the control rod drive hydraulic system.

Pumps

A supply pump pressurizes the system, with a spare pump that is 100 percent capacity. Changeover from one unit to the other is manual. System's controls and indication are provided in the Control Room. Each pump is installed with a suction strainer, two isolation valves for pump maintenance, a throttling valve for discharge pressure control, and a discharge check valve to prevent bypassing flow backwards through the non operating pump.

A minimum flow bypass connection between the pump discharge and the Condensate Storage Tank prevents overheating the pump in the event that the pump discharge valve is inadvertently closed. The pump discharge pressure is indicated at the pump by a pressure indicator.

Filters

The two parallel filters remove foreign material larger than 50 microns from the hydraulic system water. When isolated, a filter can be drained, cleaned, and vented for reuse while the other is in service. A differential pressure indicator and alarm monitor the filter element as it collects foreign material. Strainers in the filter discharge lines guard the hydraulic system in the event of a filter element failure.

First Pressure Stage

The pressure of the first stage is maintained automatically by a flow sensing control system and an air operated flow control valve. By throttling the flow control valve so as to maintain constant flow through the flow sensing control system, the pump is caused to operate at the point on its characteristic curve which corresponds to the required pressure. A parallel spare valve is provided with isolation valves to permit maintenance of the non controlling valve. This first pressure stage supplies water to the scram subsystem by the accumulator charging header. The pressure in this header is monitored in the Control Room with a pressure indicator and low pressure alarm.

Second Pressure Stage

The second stage pressure is automatically maintained at approximately 250 psi above the reactor vessel pressure by the combined operation of a pressure control valve and two bypass valves. The pressure control valve is a motor operated valve which is manually adjusted from the Control Room for a pressure drop of 230 psid. When this valve is adjusted, both bypass valves are open. This bypass flow is adjusted using a flowmeter to correspond to the flow required by a drive when moving--the flow through one corresponding to the flow while inserting, and the flow through the other corresponding to the flow while withdrawing. Electrically, one valve is connected so that it closes when the "insert" valves for any drive are actuated, the other is closed when the "withdraw" valves for any drive are actuated. In this manner the flow through these valves always balances the flow to the drives through the 1 inch drive water header; the flow and differential pressure through the pressure control valve is substantially constant, and the required pressure is maintained in the one inch drive header. The variation in flow requirements between drives is small enough so that the corresponding pressure variation is within acceptable limits.

A manual valve is provided to allow for temporary local pressure control during maintenance on the flow control valve.

Filters are installed upstream of the bypass valves for protection. Isolation valves are provided for the maintenance of the bypass valves. A flow element and indicator are installed for measuring the flow through the bypass valves so that they can be adjusted to the drive flows.

The flow element and indicator located in the drive water header are used to measure flow to the drives for adjustments and testing. A differential pressure indicator in the reactor Control Room shows the differential pressure between the reactor vessel and the drive water header. This is used when adjusting the second stage pressure with the motor operated pressure control valve.

Third Pressure Stage

The third pressure stage is automatically maintained at approximately 20 psi above reactor vessel pressure. This pressure is used to supply cooling water to the drives. The pressure drop which maintains the pressure for this stage is developed by a motor operated valve. This valve is manually adjusted from the Control Room to produce a drop of approximately 10 psig. This valve is also provided with isolation valves and a manual bypass valve for maintenance. The operation of the first and second pressure control stages is such that the flow through the motor operated valve is substantially constant; therefore the valve is able to maintain a constant differential pressure. Changes in the setting of these valves are required only to adjust for changes in the cooling requirements of the drive mechanism, as the seal characteristics change with time and variations in pump flow characteristics.

The cooling water is monitored by a flow indicator. A differential pressure indicator in the Control Room indicates the difference between reactor pressure and the cooling water pressure.

Exhaust Header

The exhaust header takes water discharged by the drives during operation, water discharged from the bypass valves, and water discharged by the third stage pressure controller, conducting this water to the reactor. Furthermore, water is continually discharged into the exhaust header via a minimum flow line which bypasses the flow and pressure control valves of the hydraulic system. The purpose of this line (with restricting orifice) is to allow a minimum flow during a scram to bypass the fully closed flow control valves, keeping the CRD Hydraulic System Reactor Vessel Nozzle cool. This makes the scram a minor transient with regards to thermal cycling at the nozzle. Periodic inspection of the CRD return line nozzle region ensures that thermal fatigue cracking would be detected.

The piping is sized to maintain a low differential (approximately 5 psi) above reactor pressure in this header. A check valve permits isolating this line from the reactor vessel and automatically prevents reactor water from flowing into this line should the supply subsystem pressure fall. A flow element and indicator permit measuring the exhaust line flow during plant operation. A bypass line from the pump output to a point upstream of this flow meter allows checking pump flows.

3.9.4.2.3 Scram Subsystem

The scram subsystem is made up of the following components:

a. Accumulators, which are charged from the charging header through a valve

b. Scram valves and their pilot valves

c. Scram discharge volumes

d. Two drain valves and two vent valves for each scram discharge volume

e. Associated instrumentation and controllers

The nitrogen precharge system, scram pilot valve, air supply system, and reactor protection electrical system operate in conjunction with the scram subsystem, but are discussed elsewhere. The operation of the scram subsystem is as described in Subsection 3.9.4.1.2.

For a description of the backup Alternate Rod Injection System which comes into play in the unlikely event that the RPS does not cause a reactor scram in response to an operational transient, refer to Subsection 3.9.4.4, Alternate Rod Injection System.

Accumulator

The accumulator on each drive is an independent source of stored energy to scram that drive. The top of the accumulator contains water while the bottom is precharged with nitrogen.

To ensure that the accumulator is always capable of producing a scram, it is continuously monitored for water and nitrogen leakage. A float type level switch will actuate and alarm if water leaks past the barrier and collects in the bottom of the accumulator. A pressure indicator and pressure switch are connected to monitor nitrogen pressure. During normal plant operation, a loss of nitrogen causes an accumulator barrier to move onto a stop and any further loss of gas will cause a decrease in the nitrogen pressure. The accumulator barrier will not move down beyond the stop, and compress the reduced amount of gas back up to pressure. A decrease in nitrogen pressure will actuate the pressure switch, illuminate a local indicating light and sound a common alarm in the Control Room. An isolation valve allows all of these instruments to be isolated and serviced.

A stop check valve in the charging line allows isolation of the accumulator for maintenance and prevents backflow from the accumulator to the header. It assures that the accumulator will retain its charge even if the supply subsystem fails or the connecting pipe ruptures.

Scram Pilot Valves

During normal plant operation, each of the two reactor protective system buses energizes one of the two three way solenoid scram pilot valves. When energized, these pilot valves supply air to the operators of both the inlet scram valve and the outlet scram valve causing both scram valves to close. During a scram, both of the reactor protective systems are de-energized and both pilot valves open, venting the scram valves and allowing them to open. To protect against spurious scrams, the pilot valves are interconnected so that both pilot valves must be de-energized for the scram valves to vent. On the other hand, failure of either electric power to both solenoids or instrument air will produce a scram. The design of the pilot valves is selected

based on simplicity, a minimum of moving parts, fast opening time (approximately 0.050 seconds) and statistical operating history on similar units.

For added protection, the instrument air header to all the pilot valves has two sets of backup scram pilot valves. Upon a scram signal, these solenoid three way valves close off the air supply and vent this section of the instrument air header. This will scram any drive should its scram pilot valves malfunction.

Scram Valves

The inlet scram valve is a globe valve which is opened by the force of an internal spring and closes when air pressure is applied on top of the diaphragm operator. The opening force of the spring is approximately 700 lbs. Each valve has a position indicator switch which energizes a light in the Control Room as soon as both valves on an HCU are open. The scram valve is selected based on high operating force, fast opening time (approximately 0.1 second) and operating history on similar units.

The outlet scram valve is identical to the inlet scram valve except that it is smaller.

Scram Discharge Volumes

The scram discharge volumes are used to limit the loss of and contain the reactor vessel water from all the drives during a scram. Two volumes are provided in the scram discharge headers, one in the North bank (69 CRDs) and one in the South bank (68 CRDs). During normal plant operation, the discharge volumes are empty with both the drain and vent valves open. These valves operate very much like the scram valves. With a scram signal, the reactor protective system is de-energized and the scram discharge volume pilot valves actuate the scram discharge volume vent and drain valves causing them to close. Position indicator switches on the main valves indicate, with lights in the Control Room, the position of the vent and drain valves.

During a scram, the discharge volumes partly fill with the water from above the drive pistons. While scammed, the control rod drive seal leakage continues to flow to the discharge volumes until the volume pressure equals reactor vessel pressure. When the scram signal is removed from the reactor protection system, the scram valves may be closed and the discharge volumes may be vented and drained. A control system interlock will not allow the drives to be withdrawn until the discharge volumes are emptied to a safe level.

A series of level switches connected to the two scram discharge instrument volumes indicate when they have emptied after a scram. There are three level switches in each scram discharge instrument volume. The level switches also guard against the discharge volume being inadvertently full when a scram is required. Should the discharge volume start to fill with water, an alarm will sound and rod block will be established; if filling continues, the reactor will automatically scram.

Test pilot valves allow the discharge volume valves to be tested without disturbing the reactor protection system. Closing the discharge volume valves allows the outlet scram valve seats to be leaked tested.

3.9.4.2.4 Cooling Subsystem

The cooling subsystem is made up of the third stage pressure control, the cooling water header, and a check valve which admits water to the underside of the drive piston. Although the drive can function without cooling water, the life of the graphitar seals and elastomer O-rings is shortened by exposure to reactor temperatures, thus cooling water is provided to protect these members. When a drive is in motion, the pressure under the piston is higher than the cooling water pressure, and the check valve is closed. The check valve opens to admit cooling water when the drive is stationary.

3.9.4.2.5 Directional Control

Four solenoid directional control valves are used for switching the drive and exhaust headers to the two drive ports as described in Subsection 3.9.4.1.2. By energizing and opening two valves at a time, the drive water header can be connected under or over the control rod drive piston while the exhaust header is connected to the opposite side. Two directional control valves, which include speed control valves, are connected so that they always pass the flow to or from the area under the piston. This is approximately 4 gpm when the drive is moving at the normal speed of 3 inches/second. The balance of forces in the drive mechanism is such that the pressure under the piston is approximately 90 psi when the drive is inserting or withdrawing. Proper speed is obtained when the speed control element in the insert valve is set, so that 4 gpm produces a pressure drop of 160 psi (260-90) when inserting; similarly, the withdraw valve is adjusted so that 4 gpm produces a pressure drop of 85 psi (90-5) when withdrawing. The directional control valves are protected by filters.

The cooling, control and scram subsystems for each drive use common piping to the drive; therefore, the directional control valves are periodically subjected to scram pressure. The two directional control valves connected to the drive water header can be opened by this higher pressure on their outlet ports. The check valve prevents significant loss of water to the drive water header during scram. Similarly a check valve provides the identical function for the cooling water header.

Directional Control Valve Sequencing

As describe above, insert motion is obtained by opening the proper pair of valves. In order to unload the collet so it can be unlocked, this pair of valves is also opened for approximately one second during a withdraw operation, after which the withdraw pair of valves are opened. This is accomplished by an electrical sequence timer, and occurs automatically when the rod withdraw operating switch is closed.

Notching is accomplished in a similar manner; when this mode of operation is selected by the operator, the proper pair of valves is energized electrically long enough to allow the drive to move to the next notch position, at which time the valves are automatically de-energized, even if the operator holds the switch closed. This feature relieves the operator of having to estimate the time required to accomplish a single notch movement.

If all four directional control valves are closed while the drive is in a position between notches, water displaced by the drive piston must leak past the drive seals in order for the drive to "settle" into a latched position. With normal seals (including those well worn) this settling speed is a fraction of normal withdraw speed. To speed up settling and latching, the "settle" circuit delays the closing of the withdraw valve for approximately 5 seconds. This allows the drive to withdraw at about 1/2 normal speed to the next latch position. The notch withdraw time interval is shortened so that the settle cycle begins before the drive has withdrawn a full notch.

3.9.4.3 Control Rod Drive Housing Design

The General Electric Company intensively investigated the potential modes of control rod drive housing failure. Particular attention was given in design and fabrication to minimize the probability for failure of the housings, including the following:

a. The housings are designed to Section I of the ASME Boiler and Pressure Vessel Code.

b. Two bolted flanged joints are made on each control rod drive. The major bolted joint is the flange to housing connection. Eight one inch bolts are used for this joint. The other bolted joint on each control rod drive is the position indication equipment flange. Six 1/2 inch bolts are used for this joint. On each of these joints, each bolt is stressed to less than 1/6 of its ultimate strength at the reactor system relief valve set pressure, and two bolts could supply the total holddown force without exceeding their ultimate strength.

c. The stress levels and the potential for cyclic stress fatigue in the reactor vessel and control rod drive housing has been evaluated. It has been determined that the most critical area is the welded joint between the housing and the nozzle.

d. The piping used for each housing was hydrostatically tested to 1800 psig and either dye penetrant, ultrasonic or X-ray and dye penetrant tested. The housing to vessel welds were dye penetrant tested. Additionally, the vessel with all control rod drives in place was hydrostatically tested to 1.50 times the reactor operating pressure as required by ASME Code.

e. The control rod drive housings are in a very low neutron flux region, resulting in a negligible increase in the nil ductility transition (NDT) temperature.

The intensive design evaluation and testing, the methods for preventing reactor overpressure, the low neutron flux and the potential for detection of initiation of a rupture prevent a control rod housing or hydraulic system failure that could cause a control rod ejection. Nevertheless, additional plant protection is provided by the control rod drive housing support structure. This equipment is positioned below the control rod drives and designed for the maximum force which could be imposed by a ruptured control rod drive housing, so that axial motion would be prohibited or limited.

The ability of the collet fingers to stop rod ejection has been investigated using dynamic drop tests at APED. Free fall drop tests of weights equal to the rod weight were conducted to simulate index tube impact on the collet. Height of the free fall was varied to cover a range of impact velocities from zero to 15 feet per second (maximum possible rod ejection velocity in the control rod drive is calculated to be 10 feet per second). Instrumentation recorded impact velocity and instant of collet finger engagement in index tube while high speed motion pictures recorded the deformation of one of the collet fingers. In each test the ability of the collet to stop the ejection and hold the index tube was demonstrated. Thus even in the event of a housing failure, the control rod would not be ejected from the core.

3.9.4.4 Alternate Rod Injection System

The Alternate Rod Injection (ARI) System (Figures 3.9-6 and Drawing Br 2013, Sheet 6) provides a method diverse from the Reactor Protection System (RPS) for depressurizing the scram air header in the unlikely event the RPS does not cause a reactor scram in response to an operational transient. The effects of an operational transient will be mitigated quickly by the ARI System, because control rod insertion makes the core subcritical much more quickly than boron injection does. Control rod insertion takes less than 10 seconds after the SCR scram valves open, whereas it takes minutes to inject sufficient boron into the reactor vessel to achieve hot shutdown and to distribute boron throughout the core. Plant safety is enhanced because the reactor is shutdown more quickly by an ARI induced control rod insertion than by boron injection.

To minimize inadvertent ARI System actuation, ARI System setpoints have been selected so as to initiate ARI after RPS setpoints have been exceeded.

The ARI System employs five normally deenergized solenoid valves on the scram air header. When the ARI valves are energized, incoming air to the scram air header from the control air system is blocked and the scram air header is vented to depressurize the scram air header. Depressurization of the scram air header causes the CRD scram valves to open resulting in control rod insertion and the SDIV vent and drain valves to close resulting in SDIV isolation. During normal (pressurized) operation, the ARI valves do not restrict the normal leakage makeup air flow from the filter/regulator to the scram air header. During a scram, the ARI valves are not in the scram air header backup scram valve vent path.

The ARI System utilizes the Fuel Zone Pressure loops PT-622-1018 (Channel C) and PT-622-1019 (Channel D) which are part of the Reactor Plant Instrumentation System. The ARI System utilizes these loops for the RV high pressure inputs to the ARI logic. ARI automatic level and pressure initiation sensor loops are arranged in a single channel, two-out-of-two logic. Since both sensor loops must trip in order to automatically initiate the ARI System, the failure of one sensor loop will not cause inadvertent ARI actuation.

3.9.5 Reactor Pressure Vessel Internals

3.9.5.1 Design Arrangements

For a general view of the vessel internals see Figure 3.9-8.

3.9.5.1.1 Core Shroud

The core shroud is a cylinder which surrounds the core and provides a barrier to separate the upward flow of coolant through the core from the downcomer recirculation flow. The lower shroud and cone is the portion of the shroud which is below the seal surface between the core support plate and shroud. The upper portion of the shroud is the portion of the shroud which is above the seal surface. (See Figure 3.9-9.)

Mounted at the top of the shroud is the shroud headsteam separator assembly. A discharge plenum at the top of the core provides a mixing chamber before the steam water mixture enters the steam separators. The recirculation inlet and outlet plenums are separated by the shroud and cone support. The cone support is designed to sustain the differential thermal expansion of the ferritic reactor vessel and the austenitic stainless steel shroud without high stresses. The cone support essentially sustains all of the vertical weight of the core structure (except those weights transmitted to the guide tube) and the steam separator assembly; the differential

upward pressure loading on the shroud under operating conditions; and the vertical and sidewise thrusts developed on the core and core structure during an earthquake. The entire weight of the fuel is carried by the guide tubes except for 12 peripheral fuel assemblies.

The cylindrical shroud is joined to the core support cone with a full penetration weld and thirty-six (36) equally spaced redundant (lug/clevis assemblies). The principal stresses produced in the shroud are due to differential pressure loading, differential thermal expansion, dead weight loadings, earthquake loadings, and 12 peripheral fuel assemblies.

In response to Generic Letter 94-03, Oyster Creek performed inspections of the redundant brackets which span welds H7 and H8 and all other horizontal shroud welds during the 15R outage. Upon discovery of cracking in one of the circumferential welds, Oyster Creek elected to repair the shroud in lieu of continued shroud inspections (References 4, 5).

The shroud repair is an alternative design to the requirements of the ASME Boiler and Pressure Code pursuant to Title 10, CFR Part 50.55a(a)(3)(i). The design provides structural integrity for all circumferential shroud welds without taking credit for the weld integrity.

The design (Figure 3.9-14) of the Core Shroud repair consisted of ten (10) steel tie-rod/rad.
An inspection of the reactor vessel internals during original construction (1967) revealed the presence of weld defects on the shroud support ring. In order to protect against failure of the shroud support ring, a lug/clevis assembly was designed to bridge the area. The assembly consists of a lug welded to the core support cone, a clevis welded to the shroud, and a pin joining the lug and clevis. Thirty-six (36) of these assemblies were installed, each equally spaced around the circumference of the shroud support ring bridging the circumferential welds (Reference 7). The assemblies were designed to support the shroud, whether or not the shroud support ring remained intact.

An inspection of the reactor vessel internals during original construction (1967) revealed the presence of weld defects on the shroud support ring. In order to protect against failure of the shroud support ring, a lug/clevis assembly was designed to bridge the area. The assembly consists of a lug welded to the core support cone, a clevis welded to the shroud, and a pin joining the lug and clevis. Thirty-six (36) of these assemblies were installed, each equally spaced around the circumference of the shroud support ring bridging the cicumferential welds (Reference 7). The assemblies were designed to support the shroud, whether or not the shroud support ring remained intact.

As part of the shroud repair evaluations (1994), a reanalysis demonstrated that only 26 of the 36 lug/clevis assemblies (without the tie-rod assemblies) are required to maintain ASME Code allowable stresses (Reference 6). This analysis superseded the original design (Reference 7). However, Oyster Creek elected to install the tie-rod radial restraint assemblies during the 15R outage. With the installation of the tie-rod assemblies to the 10 lug/clevis assemblies, this design is now sufficient to maintain structural integrity and supersedes the previous evaluations.

3.9.5.1.2 Core Support Plate

Four fuel assemblies rest on the fuel support mounted on top of each control rod guide tube (see Figure 3.9-9). Each guide tube, with its fuel support, bears the weight of four fuel assemblies and rests on the control rod drive housing, which is welded to the stub tube mounted on the vessel bottom head.

The core plate provides lateral guidance for the bottom of the fuel assemblies. Each fuel support contains four flow orifices, one for each of the four fuel assemblies. Each flow passage has an orifice to regulate flow through the fuel assemblies.

During refueling outage 17R, in September 1998, eight (8) core support plate lateral restraints (wedges) were installed in the annulus space between the outside diameter of the core support plate and the inside diameter of the core shroud. The wedges are installed around the core support plate at the following azimuthic locations: 60°F, 96°F, 132°F, 204°F, 240°F, 276°F and 312°F. The wedges provide redundant lateral support for the core support plate assembly and insure lateral alignment for control rod drive insertion in the event that the holddown bolts degrade to a point where they no longer maintain the required core support plate and control rod drive alignment.

The modification has been reviewed and approved by the NRC (Reference 8).

The installation of the wedges also eliminates the need to inspect the core support plat holddown bolts.

3.9.5.1.3 Top Grid

The upper core grid provides lateral support and alignment at the top of the fuel assemblies. It consists of a grid located at the top of the core with four fuel assemblies contained in each opening. The grid assembly is supported from the core shroud. The top grid and core plate limit lateral movement of the fuel assemblies.

3.9.5.1.4 Control Rod Guide Tubes

The control rod guide tubes extend from the control rod drive housings through holes in the core plate. Each tube is designed as a lateral guide for a control rod and as vertical support for the four fuel assemblies surrounding the control rod. In addition, the guide tubes protect withdrawn control rods from crossflow in the inlet plenum region.

The downward vertical loads from the fuel assemblies are directly transferred to the guide tubes and to the bottom vessel head. The guide tubes are locked into a sleeve mounted on the control rod drive housing. This sleeve is inserted in the control rod drive housing and extends from the housing flange almost the entire housing length. This locking device prevents upward movement of the guide tube.

3.9.5.1.5 Feedwater Spargers

The feedwater spargers are mounted to the reactor vessel wall above the downcomer annulus formed by the shroud and vessel. The spargers discharge water radially inward. This arrangement permits the cooler feedwater to mix with the downcomer recirculation flow before coming in contact with the reactor vessel. This mixing also minimizes carryunder and increases the recirculation pump suction subcooling.

3.9.5.1.6 Core Spray Spargers

The core spray spargers with spray nozzles are mounted along the inside of the core shroud in the discharge plenum at the top of the core. A more detailed description is presented in Section 6.2.

3.9.5.1.7 Standby Liquid Control Sparger

The ring sparger for the injection of liquid neutron absorber is mounted on the inside shroud surface below the core.

3.9.5.1.8 Steam Separator and Dryer

The steam separator assembly consists of a base into which are welded an array of standpipes, with a steam separator located at the top of each standpipe. The steam separator base assembly forms the shroud head which isthe top of the core discharge plenum. The fixed centrifugal type steam separators have no moving parts.

In each separator, the steam water mixture rising through the standpipe impinges on vanes which impart a spin to establish a vortex which separates the steam from the water. Steam leaves the separator at the top and passes into the wet steam plenum below the dryer. The separated water enters the pool that surrounds the standpipes to enter the downcomer annulus.

The steam dryer assembly is mounted to the reactor vessel wall above the separator assembly. A shroud extends from the bottom of the dryer assembly and extends below the water level to form a seal separating the steam entering and leaving the dryer. Steam from the separators flows upward and outward through the drying structures. Moisture removed by the dryers flows through a system of troughs and drain tubes, the drain tubes extending below the water level. Dry steam enters the vessel head cavity and is directed into the steam outlet nozzles. Vertical guide rods on the inside of the vessel provide a guide train for the dryer assembly and shroud head when being installed. The steam separator and the dryer assembly are bolted to the core shroud flange by long holddown bolts that extend above the separator and dryer for easy access. The separator base is guided to its final position with locating pins. The vertical track guides are used to rough position the shroud head.

3.9.5.2 Loading Conditions

An analysis was performed to determine the integrity of the core structure for a recirculation line break, a steam line break, and an earthquake. The lower shroud and core support, the upper shroud, the core support plate, the guide tubes and fuel channels were examined from the view point of fulfilling the requirements of the accident design criteria for both the maximum credible recirculation line break and steam line break. If buckling was not a possible failure mode and the stresses were within ASME Code Section III allowables for Class A vessels, then it was assumed that the design criteria were automatically satisfied. For those components falling outside the above limitations, the elastic stability of the structure (if buckling is the failure mode) or deformations were examined to determine if the design requirements were fulfilled.

Additional analyses were performed prior to Cycle 19 to evaluate operation with GE11 fuel. These analyses demonstrated the acceptability of operation with GE11 fuel by evaluating the impact of the limiting accident condition (steamline break inside the velocity limiter) combined with seismic loads. The results of these analyses are documented in References 9, 10, 11 and 12.

3.9.5.2.1 Structural Design Criteria

3.9.5.2.1.1 Normal Operating Transients and Steady State Conditions

The core structure was designed and constructed to meet all the primary and secondary stress requirements of Section III ASME Code for Class "A" vessels. Earthquake loading was defined as a normal operating transient.

3.9.5.2.1.2 Accident Loadings

Deformations and deflections are limited under an accident loading such that all structural components maintain their "cage like" configuration and the performance of the control rod drives and the core spray is not affected.

3.9.5.2.2 Recirculation Line Break

Table 3.9-4 outlines the collapse loads and stress resultants of the structural components previously discussed. The stresses on the upper shroud and guide tubes are so small that their integrity under maximum design accident conditions is obvious.

3.9.5.2.2.1 Lower Shroud and Cone Support

The maximum differential pressure is externally applied so that the elastic stability of the structure must be examined (the resultant stresses are quite low). The structure was examined as a short cylindrical shell extending from the vessel attachment to the shroud flange where the core support plate sits. The extra stiffness afforded by the ring where the shroud sits on the support core was conservatively neglected in the analysis. Both the stainless steel and inconel portions of the structure were examined separately. The diameter of the Inconel core support was taken as the mean diameter between the vessel I.D. and shroud O.D. The stainless steel becomes limiting because of its relatively lower strength. Formulation for determining the buckling strength of short cylindrical shells were varied both in the method of deviation and the results. Listed in Table 3.9-5 are the results from application of several formulas.

The prediction of buckling loads for a short cylinder by the use of various formulae concluded to a wide variation of results, as shown in Table 3.9-5. For this reason, the Winderberg Test results were utilized (Reference 1). The reported pressure was based on a model that most closely represented the lower shroud and support cone section. Since the test model did not have exactly the same geometrical characteristics as the structure, the buckling strengths of both structures were computed. The lower shroud and support core was found to have a 60 percent higher buckling strength than the Winderberg Test Model. However, the Winderberg Tests were conducted on mild steel at room temperature.It is believed that the 60 percent higher buckling capability of the lower shroud and core support compensates for the above factors.

3.9.5.2.2.2 Core Support Plate

The collapse load on the core support plate was based on a limit load analysis utilizing an extremely conservative analytical model.

Collapse of the core support plate is not to be confused with buckling but it is due to local yield hinges being formed on the structure such that it behaves similar to a linkage mechanism with the yield hinges acting as mechanism hinges.

The structure was examined as a clamped-clamped beam, the length of which corresponds to the diameter of the plate. The cross section examined was as schematically shown in Figure 3.9-10 ("B"). The uniform pressure applied to the beam is equivalent to that pressure acting on a 24 inch wide span, which is the spacing between the beams.

The beam model is a conservative approximation for the limit analysis of the structure. A plate is much more resistant to collapse than a beam. As an example, if one were to determine an equivalent solid simply supported plate which would deflect equally under a given distributed load as the actual structure, the computed collapse pressure would be 700 psi. It must be admitted that the beam approximation is more realistic than an equivalent plate, but it is emphasized that the beam approach represents a lower bound solution.

The stress strain curve as shown in Figure 3.9-10 ("C") was utilized in the analysis. The 24,000 psi stress utilized would be the maximum fiber stress allowed by the ASME Code if the member were subjected to a primary bending load. The use of this stress-strain curve is conservative in that no significant strain hardening is taken into account. A more realistic stress-strain curve is outlined by the dotted line shown in Figure 3.9-10 ("C").

Although the reported collapse pressure is close to the pressure applied during a maximum credible accident, it must be remembered that the assumptions utilized in the model represent a lower bound solution; probably the actual collapse pressure would be closer to 250 - 300 psi.

The consequence of the pressure applied to the plate would be to plastically deform the plate such that there would be about a 1 1/2 inch permanent deformation at the center. This deformation in no way affects the insertion of the control rod drives.

Because of the plastic deformation sustained by the core support plate, it is important to determine if radiation has affected the material properties, especially the ductility. The total integrated flux received by the core plate top surface is conservatively predicted as 1.5×10^{20} fast NVT (neutrons with an energy equal to or greater than one mev). The irradiation analysis was conservatively analyzed assuming that the control rod drives are in their fully withdrawn position for the 40 year design life of the reactor. No significant changes in the mechanical properties of annealed 304 stainless steel are expected to occur until an exposure of 5×10^{20} fast NVT is reached; since the total flux calculated on a conservative basis was a factor of three below the above figure, no deleterious irradiation effects on the core support plate is expected to occur during the reactor design life.

3.9.5.2.2.3 Fuel Channel

An external differential pressure will be applied to a fuel channel which will be about same magnitude as the predicted collapse pressure. Note that this pressure is the maximum value applied to the channel bottom. No differential pressure is applied at the top of the channel. The analytical model utilized was conservative, but not to the extent that a definite statement can be made that no inward collapse will occur. However, this collapse will not be detrimental in preventing control rod blade insertion. As shown by Figure 3.9-11 the inward motion is limited by the fuel clad which is in turn substantially supported along its length by the inlet casting and the fuel spacers. As evidenced by Figure 3.9-11, the clearance between the control rod blade and fuel channel will be increased.

3.9.5.2.3 Steam Line Break

Table 3.9-6 outlines the reactor internal pressure differentials and selected collapse loads on various critical components of the core structure. Pressure differentials are tabulated for both a line break inside and also outside the velocity limiter. The original plant structural analysis demonstrated that the structural integrity of the reactor internals would be maintained.

Additional analyses were performed prior to Cycle 19 to evaluate operation with GE11 fuel. These analyses evaluated the impact of the limiting accident condition (steamline break inside the velocity limiter) combined with seismic loads. The analyses demonstrated that the structural integrity of the reactor internals would be maintained. The results of these analyses are documented in References 9, 10, 11 and 12.

3.9.5.2.3.1 Core Support Plate

Since the core support plate sustains only one third of the applied recirculation design break pressure, it will be more than adequate to withstand the steam line differential pressure. However, since the differential pressure acts upward, the hold down bolts must be examined. The hold down bolts are of equal size. Because of the non axial symmetric nature of the plate, the loading on the bolts is not uniform. Since the applied stresses are about one half of yield strength, the core plate will be more than adequate to withstand the applied differential pressure.

The core support plate was re-analyzed prior to Cycle 19 to evaluate operation with GE11 fuel. The analysis demonstrated that the structural integrity of the core support plate would be maintained (Reference 12).

3.9.5.2.3.2 Guide Tube

Because of the externally applied pressure, the guide tube must be examined as to its collapse susceptibility. As in the case of the shroud and core support, a number of formulae were utilized to predict the collapse pressure. Unfortunately, the use of Winderberg tests is not applicable because the geometry of the guide tube falls outside the test range. Use of the ASME curves Section VIII indicated the extreme sensitivity to wall thickness. For the minimum wall as called out for a 10 inch Schedule 10 pipe, the ASME curves predict a buckling load of 45 psi. Using the average wall thickness, the collapse pressure is increased to over 70 psi. Using empirical relations for tubes over the critical length, the calculated collapse pressure is over 100 psi.

The ASME curves will predict that the collapse pressure is reached at 54 psi at a wall thickness of 0.150 inches, which is 6 mils over the minimum for a 10 inch Schedule 10 pipe.

Since the ASME curves are obtained on a conservative basis, it is reasonable to assume no collapse. Also analysis has indicated that the control rod blade will be 70-90 percent inserted by the time the maximum external pressure is applied to the guide tube.

The guide tube was re-analyzed prior to Cycle 19 to evaluate operation with GE11 fuel. The analysis demonstrated that the structural integrity of the guide tube would be maintained (Reference 12).

3.9.5.2.3.3 Fuel Channel

The fuel channel collapse load due to an internally applied pressure was examined utilizing a fixed-fixed beam analytical model under a uniform load. Tests were conducted at APED to verify the applicability of the analytical model. The results indicated that the analytical model was conservative. Although collapse will not occur, the fuel channels may bow sufficiently outward to cause some interference with movement of the control rod blade. This was examined in detail. There are about 15 factors such as fuel channel bow, core plate hole tolerance, top guide beam location, etc., that go to determine the clearance between the control rod blade and fuel channel. If each of these tolerance factors are assumed to be at the worst extreme of the tolerance range, then a slight interference under an 18 psi pressure would develop. At the top of the blade there is a roller to guide the blade as it is inserted. The clearance between channels would be 70 mils less than the diameter of the roller causing it to slide or skid instead of roll. As the blade is inserted about halfway there is a tendency for the blade sheath to push inward on the channel. This is a blade surface to channel surface contact. A worst cast study indicates a possibility of a 50 mil interference.

The possibility of a worst case developing is extremely remote. If a statistical analysis is undertaken utilizing a normal distribution for each of the 15 variables, then no interference will occur within three sigma limits. However, even if interference occurs, the result would be negligible. It takes about one pound of lateral force to deflect the channel inboard one mil. The friction force developed would be an extremely small percentage of the total force available to the control rod drives or compared to the weight of an individual fuel assembly.

All the above discussion presupposes the control rod blade has not moved when the channel experiences the largest magnitude of pressure drop. Analysis indicates that the blade will be about 70-90 percent inserted. If the blade is beyond 70 percent inserted, then no interference is likely to develop because all the channel deformation is in the lower portion of the fuel assembly, whereas the roller is at the top portion of the blade. Also, the blade is beyond the position where surface to surface contact can occur.

The fuel channel was re-analyzed prior to Cycle 19 to evaluate operation with GE11 fuel. This analysis calculated a maximum pressure differential of $<$13 psi. The analysis demonstrated that the structural integrity of the fuel channel would be maintained (Reference 12). GNF2 channel maximum pressure differential conditions are bounded by GE11 fuel (Reference 13). Structural integrity of the fuel channel is maintained.

3.9.5.2.4 Earthquake Analysis

All earthquake loadings must be within the design criteria for normal operating conditions. The vertical component does not contribute to any significant loadings on the core structure. The vertical weight of the fuel is not applied to the core support plate. The vertical weight is applied to the guide tube which in turn transfers the load to the bottom head of the reactor vessel.

The horizontal component was examined in detail. Not only must the stresses in the various components be examined but also the deflection of the shroud in the region of the core spray connection line. In essence, the shroud is a vertical cantilevered structure with the lateral force of the fuel component transferred to the shroud by the core plate and top guide. The analytical model utilized is shown in Figure 3.9-12. It turns out that the structure is more than adequate to sustain an earthquake load. To demonstrate this capability, a one "g" lateral force was applied to the structure. This is not to imply that the design earthquake loading is of this magnitude, but rather to demonstrate the capability of the core structure to sustain an earthquake loading. Under a one "g" lateral force, the maximum bending stress in the shroud support cone would be

4400 psi, and the shroud would deflect less than one eighth of an inch in the region of the core spray pipe connection. The deflection was conservatively obtained assuming all the loads were applied in the region of the core spray pipe connections. Both bending and shear deflections were taken into account.

The above discussion is based on the original plant analysis. Additional analyses were performed prior to Cycle 19 to evaluate operation with GE11 fuel. These analyses evaluated the impact of the limiting accident conditions <u>in combination with</u> seismic loads. All reactor internal components have been evaluated considering the limiting seismic loads (References 9, 10, 11 and 12). GNF2 fuel assemblies were evaluated prior to Cycle 23 (Reference 13) and were shown to be bounded by the previous fuel types.

3.9.5.2.5 <u>Conclusions</u>

The structural analysis has always been based on the pressure differentials obtained under maximum credible accident conditions (an instantaneous double ended break of the largest pipe line). Once these assumptions are relaxed, that is, if the line break is not the maximum size or the break is assumed to occur in some finite time, the resulting pressures are reduced and so are the stresses sustained by the core structure components. The above assumptions do not have to be relaxed to any degree for the resulting stresses to be within ASME Code limits.An example is illustrated in Figure 3.9-13. The core support plate was examined under a recirculation line break. Plotted is the time of break and percent design break. The shaded area represents those break times and percent design breaks where the maximum stress would be within ASME Code limits. As can be seen the unshaded area represents a small portion of the total area. In the real world, the probability of a small break is much greater than the maximum design break. Also the probability of the break occurring in a finite time is a much more realistic approximation of the real event.

In other words, the core structure analysis was very conservative and, even so, the structure withstands the accident conditions; therefore, it may be rationally concluded that the core structure is more than adequate to sustain the accident loading; if such an unlikely event occurs, it still retains its "cage-like" configuration.

3.9.5.3 <u>Design Bases</u>

The reactor internals were designed to withstand a design basis earthquake.The internal components of the reactor vessel (in conjunction with the reactor vessel) were designed to allow adequate core cooling to be maintained during normal operation and accident conditions and not fail during normal operation and accident conditions.

To meet the above, the internal components were designed to:

a. Provide support for the fuel, steam separators, dryers, etc., during normal operation and accident conditions.

b. Maintain required fuel configurations and clearances during normal operation and accident conditions.

c. Circulate reactor coolant to cool the fuel.

d. Provide adequate separation of steam from water.

The core structural components were designed to accommodate the loadings applied during normal operation and maneuvering transients considering both stress and deflection. Deflections are limited by design so that the normal functioning of the components under these conditions will not be impaired. Where deflection was not the limiting factor, the ASME Boiler and Pressure Vessel Code, Section III was used as a guide to determine limiting stress intensities and cyclic loadings for the core internal structure where code was not mandatory.The internal structural components were designed to also accommodate the earthquake conditions specified in Section 3.7. Structural

earthquake loading on the internals was based on the calculated reactor vessel acceleration response of 0.20g to 0.30g, depending upon the vertical elevation on the vessel, due to the specified ground motion of 0.11g.

Deflections of components supporting the fuel and control systems are limited by design to allow safe operation during this transient. Stress intensity in these components is also maintained within the ASME Pressure Vessel Code, Section III.

The reactor internals were designed to preclude failure which would result in any part being discharged through the steam line in the event of a steam line break outside of the steam line isolation valve.

The structural components which guide the control rods were examined to determine the loadings which would occur in a loss of coolant accident (including a steam line break). The core structural components were designed so that deformations produced by accident loadings will not prevent insertion of the control rods.

3.9.5.4 Surveillance and Testing

Rigid quality control requirements insured that the design specifications ofthe vessel internal components were met. These quality control methods were utilized during the fabrication of the individual components as well as during the assembly process.

Preoperational performance tests of the core spray spargers demonstrated the satisfactory operation of the system.

3.9.6 Inservice Testing of Pumps and Valves

Inservice testing of pumps and valves is performed in accordance with the latest version of the Oyster Creek Inservice Inspection Program and the plant Technical Specifications.

3.9.7 References

(1) Oyster Creek Nuclear Power Station Unit No. 1, Facility Description and Safety Analysis Report, Docket No. 50-219, Amendment No. 53, Investigation of Stainless Steel Piping, June 1969.

(2) Deleted.

(3) GPUN Response to Generic Letter 88-01 and NUREG 0313, Rev 2 TR-050. 8/93.

(4) GPUN Safety Evaluation 403037-001, "Reactor Vessel Shroud Repair".

(5) NRC Safety Evaluation Regarding the Oyster Creek Core Shroud Repair, dated November 25, 1994.

(6) Oyster Creek Vessel Shroud Lug/Clevis Assembly Stress Analysis GE Report # GENE 523-A151-1094, dated October 1994.

(7) FDSAR Amendment 40.

(8) NRC Safety Evaluation regarding Core Support Plate Wedge Modification for Oyster Creek Nuclear Generating Station 17R Outage, dated November 6, 1998.

(9) MPR-2407, “Oyster Creek Reactor Internals Structural Analyses of Core Shroud and Core Shroud Repair with GE9 and GE11 Fuel Loads”, Rev. 1, July 2002.

(10) SIA Letter Report RAM-02-079, “Evaluation of the Oyster Creek Shroud Support System for Revised RIPDs and Seismic Loads”, July 3, 2002.

(11) SIA Letter Report MLH-02-030, “Oyster Creek Top Guide Evaluation: Impact of Revised RIPD and seismic Loads”, July 16, 2002.

(12) General Electric Report GE-NE-0000-0001-0959-01P, “GE11 Fuel Design Cycle-Independent Analyses For Oyster Creek Generating Station”, Rev. 0, August 2002.

(13) GNF2 Fuel Design Cycle-Independent Analyses for Exelon Oyster Creek Generating Station," GEH-0000-0118-3544-R0, Revision 0, October 2010.

TABLE 3.9-1
(Sheet 1 of 2)

ALLOWABLE STRESSES FOR CLASS I PIPE
FOUR HEAT REMOVAL SYSTEMS

	Loading Condition	Allowable Stress
1.	Thermal Expansion	$S_A + f\ (S_h - S_{Sustained})$
2.	Pressure + Deadweight	$1.0\ S_h$
3.	Pressure + Deadweight + OBE	$1.0\ S_h$
4.	Pressure + Deadweight + SSE	$1.2\ S_h$

$S_{\mathbf{Sustained}}$ = Stress Due to Pressure + Deadweight

S_A = $f\ (1.25\ S_c + 0.25\ S_h)$

where:

f = stress range reduction factor for cyclic conditions

S_c = allowable stress in cold condition per ANSI B31.1 – 1983 through Winter 1984 Addenda or Specification SP-1302-12-294, Appendix B

S_h = allowable stress in the hot condition per ANSI B31.1 – 1983 through Winter 1984 Addenda or Specification SP-1302-12-294, Appendix B

The two classes of components subject to the earthquake design requirements are as follows:

Class I - Structures and equipment whose failure could cause significant release of radioactivity or which are vital to a proper shutdown of the plant and the removal of decay heat.

- Class II Structures and equipment which are both essential and nonessential to the operation of the station, but which are not essential to a proper shutdown.

TABLE 3.9-1
(Sheet 2 of 2)

ALLOWABLE STRESSES FOR CLASS I PIPE
REMAINING SYSTEMS

	Loading Condition	Allowable Stress
1.	Thermal Expansion	S_A+f (S_h-$S_{Sustained}$)
2.	Pressure + Deadweight	1.0 S_h
3.	Pressure + Deadweight + OBE	1.0 S_h
4.	Max Pressure + Deadweight + SSE	2.4 S_h

$S_{\textbf{Sustained}}$ = Stress Due to Pressure + Deadweight

S_A = f (1.25 S_c + 0.25 S_h)

where:

f = stress range reduction factor for cyclic conditions

S_c = allowable stress in cold condition per ANSI B31.1 – 1983 through Winter 1984 Addenda or Specification SP-1302-12-294, Appendix B

S_h = allowable stress in the hot condition per ANSI B31.1 – 1983 through Winter 1984 Addenda or Specification SP-1302-12-294, Appendix B

The two classes of components subject to the earthquake design requirements are as follows:

Class I - Structures and equipment whose failure could cause significant release of radioactivity or which are vital to a proper shutdown of the plant and the removal of decay heat.

Class II - Structures and equipment which are both essential and nonessential to the operation of the station, but which are not essential to a proper shutdown.

TABLE 3.9-2
(Sheet 1 of 1)

ALLOWABLE STRESSES FOR REACTOR VESSEL SUPPORTS

1. Seismic - Allowable stress = normal AISC allowable stresses.

2. Seismic + Jet - Allowable stress = 150 percent of normal AISC allowable stresses.

3. Seismic - Allowable stress = 150 percent of normal AISC allowable stresses.

TABLE 3.9-3
(Sheet 1 of 2)

FLOW FUNCTIONS THROUGH PIPING ASSEMBLY RISERS

Control Rod Drive Functions and Flow Functions

Riser	Drive-Insert	Drive-Withdraw	Scram	Drive Stationary
Accumulator Charging	No Flow	No Flow	Water from accumulator charging header to scram accumulator water cylinder	No Flow
Drive-Withdraw	Discharge waterCRD over-piston port to manifold	Drive water from manifold to CRD over-piston port.	Scram discharge water from CRD over-piston port to manifold	No Flow
Drive Water	Water from CRD hydraulic system drive water header to manifold	Water from CRD hydraulic system drive water header to manifold	No Flow	No Flow
Exhaust Water	Water from manifold to CRD hydrraulic system exhaust water header	Water from manifold to CRD hyddraulic system exhaust water header	No Flow	No Flow

TABLE 3.9-3
(Sheet 2 of 2)

FLOW FUNCTIONS THROUGH PIPING ASSEMBLY RISERS

Control Rod Drive Functions and Flow Functions

Riser	Drive-Insert	Drive-Withdraw	Scram	Drive Stationary
Drive-Insert	Drive water from manifold to CRD underpiston port.	Discharge water from CRD underpiston port to manifold	Accumulator water from manifold to CRD underpisoton port	Cooling water from manifold to CRD underpiston port.
Cooling Water	No Flow	No Flow	No Flow	Water from CRD hydraulic system cooling water header to manifold.
Scram Discharge	No Flow	No Flow	Discharge water from manifold to CRD hydraulic system scram discharge header	No Flow

TABLE 3.9-4
(Sheet 1 of 1)

CORE STRUCTURE ANALYSIS FOR MAXIMUM CREDIBLE RECIRCULATION LINE BREAK

Structural Component	Applied Differential Pressure	Collapse Load	Resultant Stress
Lower Shroud and cone support	125 psi (inward)	320 psi	7500 psi
Upper Shroud	7 psi (outward)	----	420 psi
Core Support Plate	132 psi (downward)	145 psi	----
Guide Tube	132 psi (outward)	----	4300 psi
Fuel Channel	22 psi (inward)	22 psi	----

TABLE 3.9-5
(Sheet 1 of 1)

LOWER SHROUD AND CONE SUPPORT ANALYSIS RESULTS
(RECIRCULATION LINE BREAK)

Method of Analysis	Type of Analysis	Computed Collapse Pressure	Safety Factor Applied Press.
ASME (1933) PressureVessel ResearchCommittee Curve		450	3.6
Flugges Formula	Analytical	620	5.0
Windberg Test	Tests	320	2.5
Southwell	Analytical	1800	14.4

TABLE 3.9-6
(Sheet 1 of 1)

CORE STRUCTURE ANALYSIS FOR MAXIMUM CREDIBLE STEAM LINE BREAK

Structural Component	Applied Differential Pressure: Outside Velocity Limiter*	Applied Differential Pressure: Inside Velocity Limiter [1,2,3]	Collapse Load
Lower Shroud and Cone Support	45 psi (out)	59 psi	----
Upper Shroud	11 psi (out)	40 psi	----
Core Support Plate	34 psi (up)	33 psi	
Guide Tube	34 psi (in)	33 psi	107 psi
Fuel Channel	11 psi (out)	<13 psi	25 psi

* - Historical results, not re-evaluated for GE11 Fuel Introduction

1) Limiting Faulted Condition (22.6% Power / 110.7% Core Flow / Feedwater Temperature Reduction)

2) A 1-psi conservatism is added to the Shroud Support, Core Support Plate, Guide Tube and Fuel Channel

3) Values bound operation with GE11 Fuel and all 8x8 Fuel Types

3.10 SEISMIC QUALIFICATION OF SEISMIC CATEGORY I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Safety related instrumentation is discussed in Sections 7.2 through 7.6. Safety related electrical equipment is discussed in Section 8.3. Refer to those sections for additional detail.

3.10.1 Seismic Qualification Criteria

The design of the Oyster Creek Nuclear Generating Station (OCNGS) began approximately in January 1964. At that time, the seismic design for equipment structures was based on dynamic analyses using acceleration response spectrum curves which were based on a ground motion of 0.11 g. The intent for the design was to ensure a safe shutdown for ground motions of 0.22 g. For further information on seismic input refer to Section 3.7.

The NRC initiated a generic program to develop criteria for the seismic qualification of electrical and mechanical equipment in operating plants as an unresolved safety issue (USI A-46). Under this program, an explicit set of guidelines that should be used to judge the adequacy of the seismic qualifications of safety related equipment at all operating plants was developed (Reference 3.10.4.1).

The resolution of USI A-46 for Oyster Creek was by implementation of the generic criteria and methodology of the Generic Implementation Procedure for seismic verification of Nuclear Plant Equipment developed by the Seismic Qualification Utility Group. (Reference 3.10.4.5)

3.10.2 Methods and Procedures for Qualifying Electrical Equipment and Instrumentation

For methods and procedures for qualifying electrical equipment and instrumentation, see Section 3.7.3.4.

3.10.3 Methods and Procedures of Analysis or Testing of Supports of Electrical Equipment and Instrumentation

Studies have been conducted in accordance with the SEP to determine the effect of a seismic event on components in switchgear, motor control centers and panels. In these studies tensile pullout loads and shear loads on anchors were calculated by taking into account seismic and deadweight loads (Reference 3.10.4.2). The calculation of loads due to seismic events were performed in accordance with the methodology described in NRC Regulatory Guide 1.92. Damping values were in accordance with Regulatory Guide 1.61. Seismic loads were defined by horizontal and vertical response spectra, as provided in Reference 3.10.4.3. Actual tests and test data are discussed in Reference 4.

3.10.4 References

(1) NUREG-0822, Integrated Plant Safety Assessment/Systematic Evaluation Program, Oyster Creek Nuclear Generating Station. Final Report, January 1983.

(2) Oyster Creek Nuclear Generating Station, "Seismic Analysis of 4160 Volt Switchgear and 460 Volt Unit Substation Cabinets" (MPR-794), MPR Associates Inc., November 1983.

(3) NUREG/CR-1981 UCRL-53018, "Seismic Review of the Oyster Creek Nuclear Power Plant as part of the Systematic Evaluation Program," April 1981. (Revised on June 23, 1981)

(4) Teledyne Summary Report, TR-3501-2, "Generic Response to USNRC I&E Bulletin 79-02: Base Plate/Concrete Expansion Anchor Bolts," August 30, 1979.

(5) Letter, GPUN to NRC, "Response to Generic SER on SQUG Resolution of Unresolved Safety Issue A-46", October 13, 1988.

3.11 ENVIRONMENTAL DESIGN OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT

The Environmental Qualification (EQ) Program provides assurance that specific electrical equipment will perform its intended safety function. Specifically, the objectives of the EQ Program are:

a. Document the qualification of electrical equipment important to safety as required by 10CFR50.49.

b. Establish the maintenance/surveillance required to maintain the qualification of this electrical equipment over the life of the plant.

3.11.1 Equipment Identification and Environmental Conditions

3.11.1.1 Identification of Electrical Equipment

3.11.1.1.1 Criteria for Selection of Equipment

The EQ program addresses all electrical equipment important to safety as defined in 10CFR50.49(b)(1),(2), and (3). The EQ Master List, GPUN document No. 990-1464, identifies electrical equipment or components which must be environmentally qualified for use in a harsh environment. Equipment important to safety which is exposed only to a mild environment during postulated accident conditions is not included in the EQ Program.

3.11.1.1.2 Class 1E Equipment and Interfaces

A detailed review of each system (Figure 3.11-1) was performed to identify all of the systems' electrical components and instrumentation in accordance with 10CFR50.49. This was performed by listing each major electrical component (motor operator, pump motor, instrument, etc.) and then identifying the auxiliary electrical equipment within the circuit to the component (cable, terminal blocks, splices, etc.). The identification of the major electrical equipment was carried out through a review of the electrical one lines, elementary wiring diagrams, Technical Specifications, Emergency Operating Procedures, (as needed to support 10CFR50.49 Program requirements) and Process and Instrumentation Diagrams.

Certain Class 1E equipment was classified as commodity items. A variety of plant walkdowns were performed to provide reasonable assurance that qualification attributes were accurate.

3.11.1.1.3 NUREG-0737 and Regulatory Guide 1.97 Equipment

Supplement 1 to NUREG 0737 requires that certain post accident monitoring instrumentation be provided to enable operators to assess plant and environmental conditions during and following an accident. The post accident monitoring instrumentation is selected using the guidance provided by ANSI/ANS 4.5-1980 as endorsed by Regulatory Guide 1.97.

3.11.1.1.4 EQ Position on DOR Guidelines, NUREG-0588 and R. G. 1.89

All equipment within the scope of this program has been evaluated for compliance with either the DOR Guidelines NUREG-0588, Category I, or 10CFR50.49 with guidance from Regulatory Guide 1.89.

Oyster Creek was an operating plant when the DOR Guidelines were issued in November 1979. Therefore, installed equipment was required to meet the requirement of the DOR Guidelines.

Replacement parts must be qualified to 10CFR50.49 except where there are determined acceptable "Sound Reasons to the Contrary" as defined in Regulatory Guide 1.89, R1, Section C, paragraph 6, subparagraphs "a" through "g".

3.11.1.2 Environmental Conditions

The environmental parameters for each plant area were determined for both normal and accident service conditions. These parameters are documented in Reference 3.11.6.3.

The plant environmental conditions applicable to a specific piece of equipment can be obtained from the EQ files identified on the EQ Master List.

3.11.1.2.1 Normal Service Conditions

The plant normal service conditions include all aspects of normal operation, including all levels of power operation, shutdown condition, cold shutdown, or refuel mode as defined by the Technical Specifications and any other normally anticipated operational occurrence (which includes a loss of HVAC).

The normal service conditions for a specific component are given in Reference 3.11.6.3 and encompass the applicable temperature, pressure humidity, and radiation conditions postulated to occur at the specific equipment location during normal operation of the plant. The methodology used to define the normal service conditions are described below.

a. Temperature/Pressure

The temperatures inside the drywell were obtained from measurements taken during normal operation. The temperatures in the various rooms in the reactor and turbine buildings were based on calculated average annual temperatures for those areas of the plant which are ventilated by outside air under normal operating conditions. Atmospheric pressure (14.7 psia) is assumed for all areas outside the drywell and drywell pressure is assumed to be 16 psia.

b. Humidity

The humidity inside the drywell was obtained from measurements taken during normal operation (use of 50% is deemed conservative). For areas outside the drywell, the humidity can vary reaching a maximum of 100%. Thus, a value of 100% RH has been used unless a lower value has been justified and documented. Humidity variations during normal operation will not be evaluated in accordance with 10CFR50.49(e)(2).

c. <u>Radiation</u>

Normal power operation general area radiation dose rates for the drywell, reactor and turbine buildings are taken from health physics dose rate surveys as well as from predicted normal operating dose rates.

The radiation dose is integrated over a 40 year plant operating time.

3.11.1.2.2 <u>Accident Service Conditions</u>

The development of accident service conditions considered the environmental conditions resulting from a postulated Loss-of-Coolant-Accident (LOCA), Main Steam Line Break (MSLB) inside the drywell, and High Energy Line Break (HELBS) outside of the drywell in the reactor and turbine buildings.

The analyses of these postulated accidents address the following environmental parameters:

a. Temperature/Pressure

b. Humidity

c. Chemical Spray

d. Submergence

e. Radiation

The specific analyses performed and their results are discussed in greater detail below.

a. <u>Temperature/Pressure</u>

A description of the analyses performed to determine the containment temperature and pressure response to a LOCA is found in Reference 3.11.6.3. The main steam line break provides the most severe containment temperature and the DBA LOCA provides the most severe containment pressure response. The resulting time dependent temperature and pressure profiles which are used in the Oyster Creek environmental qualification program are shown in Figures 3.11-2 and 3.11-3 (Reference 3.11.6.3).

A High Energy Line Break (HELB) can produce a harsh environment in the reactor and turbine buildings. The peak temperatures and pressures are based on the worst case HELB affecting a specific EQ zone.

Temperature profiles were generated for five double ended guillotine pipe breaks in the areas where the environmental response for electrical equipment would be most severe for equipment qualification (Reference 3.11.6.3). The pipe breaks are as follows:

	System	Break Size and Location
1.	Main Steam	24" main steam pipe within the steam tunnel at elevation 23'-6" in Reactor Building
2.	Cleanup and Demineralizer	6" pipe to Cleanup Auxiliary Pump at Elevation 51'-3" in Reactor Building.
3.	Emergency Condenser	16" emergency condenser pipe break at elevation 75'-3" in Reactor Building.
4.	Reactor Feedwater	14" reactor feed line break in the feedwater pump room elevation 14'-11" in Turbine Building basement floor.
5.	Main Steam	24" main steam pipe at elevation 23'-6" in Turbine Building mezzanine floor.

b. Humidity

Environmental qualification of equipment during the above mentioned postulated accidents is based on 100 percent RH for the drywell, reactor and turbine buildings, and the standby gas treatment system tunnel. A lesser value may be used, when justification is developed for a given component and included in the EQ file for that component.

c. Chemical Spray

The demineralized water spray inside the drywell is considered to have an insignificant effect on metallic or non-metallic components. However, the moisture intrusion effects of a water spray have been considered in the qualification process.

d. Submergence

The plant areas which could be submerged during postulated accident conditions are in the reactor building, drywell and the steam tunnel. For all other plant areas, the design of floor drains will prevent submergence of electrical equipment. There is no electrical equipment in the EQ Program which is submerged.

e. Radiation

For equipment qualification purposes, the accident radiation conditions postulated to occur result from a non isolable pipe break in the reactor coolant system inside of containment and were developed based upon the bounding requirements of the DOR Guidelines of NUREG-0588, as applicable to the component being qualified.

The accident radiation doses, gamma and beta (if applicable) were integrated over the duration of the accident (which is usually taken as 1 year). This is a conservative approach if the required component operating time is appreciably less than the radiation integration time (e.g., hours versus months, respectively).

3.11.1.2.3 Equipment Operability Time

The operability duration is the length of time during and following an accident that equipment must maintain its ability to perform its intended safety function. The safety function includes:

1. The ability to initiate short term protective action.

2. The ability to place the plant in a controlled condition.

3. The ability to keep the plant in a stable condition after the accident until personnel are able to enter the plant to inspect, repair, or replace equipment.

Operating time may be defined on a case by case basis. However, most components requiring environmental qualification were those devices required to perform one of the functions delineated in Table 3.11-1 suitable for the equipment to accomplish its intended safety function (Reference 3.11.6.2 as modified by the Technical Specifications Section No. 5.2.)

3.11.2 Qualification Tests and Analyses

3.11.2.1 Acceptance Criteria

Electrical equipment was evaluated to ensure that it will function as required after exposure to its normal and postulated accident environments. All qualification conforms to the requirements of either the DOR Guidelines, or NUREG 0588-Category 1 (IEEE 323-1974).

3.11.2.1.1 Accident Environments

Each piece of equipment entered into the Oyster Creek EQ Program was evaluated to determine if it would function as required during exposure to postulated accident conditions. The components need only be qualified to the accident parameters of the accidents the component is required to mitigate, for the time period it is required to function. The accident parameters are specified in individual EQ files which are identified on the EQ Master List.

3.11.2.1.2 Margins

Equipment within the scope of the Oyster Creek EQ Program was qualified to accident environmental profiles which enveloped the plant parameters. The conservatisms included in these profiles are judged to be sufficient to account for uncertainties associated with the analytical techniques, definitions of performance requirements, and variations in commercial production.

3.11.2.1.3 Connection Interfaces

Equipment exposed to steam conditions coincident with pressure is provided with a seal where required.

3.11.2.1.4 Performance Specifications

The EQ Program includes an evaluation of equipment to ensure that performance specifications are achieved under conditions existing during and following postulated accidents. This evaluation reviews functional requirements, and/or loop accuracy based upon function, location, environment and performance requirement.

3.11.2.1.5 Voltage and Frequency

Safety related electrical equipment is subject to variations in power supply characteristics such as voltage and frequency. For the AC distribution system, these are comprised of the expected off-site power supply variations, including degraded grid conditions, and the expected variations of the diesel generator if off-site power has been lost. These conditions are addressed in the EQ files for the specific equipment.

3.11.2.1.6 Synergistic Effects/Phase Changes

The equipment qualification effort did consider synergisms/phase changes to the extent identified as follows:

a. If the vendor identified a synergistic effect/phase change, it was evaluated.

b. If the reviewer was aware of a synergistic effect/phase change, it was evaluated. As additional synergistic effect/phase change data became available it was evaluated and factored into the program.

c. If neither a. nor b. existed, then no further actions were taken to determine if any synergistic effect/phase change were known (e.g. literature research).

3.11.2.1.7 Field Verification

a. Walkdowns

Adequate field inspection of as-installed Class 1E equipment was performed by walkdowns. The field inspection, provided a reasonable assurance that there is: (a) a traceable link between the equipment installed at the plant and the equipment that was qualified, (b) a direct verification that any special installation requirements identified in the qualification program were applied, and (c) a verification that external gaskets, seals, protective covers, etc. have been installed.

3.11.3 Qualification Test Results

3.11.3.1 Documentation

The qualification documentation for electrical equipment are assembled into EQ files. An EQ file is prepared for a group of components having the same manufacturer but different plant tag numbers. The EQ file contains all necessary reports, analyses, and correspondence submitted by the vendor to satisfy Purchase Order (P.O.) requirements, thereby establishing a direct link between plant installed equipment and providing the basis for environmental qualification in accordance with 10CFR50.49.

3.11.3.2 Independent Verification

Each EQ file is prepared and verified to ensure the completeness and accuracy of the data presented. The results of this review are documented in Environmental Qualification files, as necessary, based upon the requirements of the DOR Guidelines or NUREG 0588-Category I.

3.11.4 Loss of Ventilation

A loss of ventilation is considered a possible normal operational occurrence and, therefore, does not establish a harsh environmental condition. Loss of HVAC is not considered a design basis accident for environmental qualification purposes, unless the loss of HVAC is a direct consequence of a design basis accident. Therefore, the interpretation of 10CFR50.49, as it applies to loss of HVAC is that, "an area whose environment results from normal plant operation or an anticipated operational occurrence is still considered, and should be classified, as a mild environment". (Reference 3.11.6.3)

3.11.5 Estimated Chemical and Radiation Environment

3.11.5.1 Chemical Spray

The Containment Spray System is described in OCFSAR Section 6.2.2.

3.11.5.2 Radiation

The accident radiation conditions postulated are from a non-isolable pipe break in the reactor coolant system inside of containment, and were developed based upon the bounding requirements of the DOR Guidelines of NUREG-0588.

The accident radiation doses, gamma and beta (if applicable) were integrated over the duration of the accident (which is usually taken as 1 year). This is a conservative approach if the required component operating time is appreciably less than the radiation integration time (e.g., hours versus months, respectively).

3.11.5.2.1 Inside Drywell - General Areas

The percent of core inventory assumed to be released from the fuel for a LOCA meets the NUREG-0588 requirements of:

100 percent of the Noble Gas Core Inventory; 50 percent of the Iodine Core Inventory; and, 1 percent of the other nuclides in the Core Inventory.

a. Gamma Dose

The source term, basic assumptions and model used to develop the total gamma dose radiation service condition for Class 1E equipment located in general areas inside the drywell are described in Reference 3.11.6.3. An estimated value of 5.0 x 10E7 RADs for the total airborne gamma dose in the drywell (1 year integrated dose from airborne iodine and airborne noble gases).

b. Beta Dose

The source term, basic assumptions and model used to develop the total beta radiation dose for Class 1E equipment located in general areas inside of the drywell are described in Reference 3.11.6.3.

An estimated value of 9.6 x 10E8 RADs for the total airborne beta dose in the drywell (1 year integrated dose from airborne iodine and airborne noble gases has been utilized). Per the DOR Guidelines, electrical cable is considered to be the most vulnerable to damage from beta radiation from the general classes of electrical equipment. The beta surface dose was reduced by a factor of 10 within 30 mils of the surface of electrical cable insulation, therefore, for cables with 30 mils or greater insulation, beta radiation is 9.6 x 10E7 RADs.

The above radiation service conditions inside the drywell are applicable to equipment in the drywell vapor space. A calculation was not performed for equipment submerged in the torus fluids as none exist.

3.11.5.2.2 Outside Primary Containment

Radiation values outside containment conservatively assumed a 100% fuel failure even though for HELB's outside containment, the radiation exposure will be significantly less due to isolation of the line break and expected minimum fuel damage.

Reference 3.11.6.3 defines the calculated general area radiation maps and radiation exposures at specified equipment locations. The analysis included the effects of radiation shine from the drywell, airborne radioactivity due to leakage from the primary containment and gamma doses due to recirculation piping and components. The analysis calculated beta radiation and gamma radiation.

3.11.6 References

NRC Letter LS05-85-05-031 to P.B. Fiedler, dated May 28, 1985, "Safety Evaluation for Final Resolution of Environmental Qualification of Electric Equipment Important to Safety".

(1) NRC Letter LS05-85-05-031 to P.B. Fedler, dated May 28, 1985, “Safety Evaluation for Final Resolution of Environmental Qualification of Electric Equipment Important to Safety.”

(2) Impell Report 02-0370-1293 Rev. 0, dated December 14, 1984, "Identification of Safety Related Equipment".

(3) ES-027, GPUN Technical Functions Standard "Environmental Parameters - Oyster Creek NGS".

(4) USNRC Letter to GPUN, dated August 8, 1986, "Inspection Report No. 05000219/86-08.

(5) TR-028 Rev. 3, dated October 12, 1992, "OC Response to USNRC R.G. 1.97".

Table 3.11-1
(Sheet 1 of 1)

SUCCESS CRITERIA/DURATIONS

FUNCTION	PARAMETER LIMITS	DURATION
Emergency Reactor Shutdown	All control rods inserted	<12 hrs*
Reactor Core Cooling	Fuel Clad Temperature <2200°F	>48 days
Reactor Heat Removal	RPV Pressure <1375 psig Cooldown <100°F/hr if possible	>48 days
Containment Isolation	All required isolation valves closed	<I hr
Containment Heat Removal	Drywell Temperature <292°F Torus Temperature <150°F Drywell Pressure <44 psig Torus Pressure < 35 psig	>48 days
Prevention of Release to Environment	Oxygen Concentration <5% by Volume Hydrogen Concentration <6% by Volume	>1 day >1 day
	Radioactivity Release <small percentage of 10 CFR 100 limits	<1 day (Vent)
	All required isolation valves closed	<1 hr (Isolation)

* Small steam line breaks result in a high drywell pressure scram and are used to mitigation of RWCU Line Break (Manual Scram)
Other DBEs will result in a reactor scram in less than one minute.