CHAPTER 5 – CONTAINMENT SYSTEM AND OTHER SPECIAL STRUCTURES

TABLE OF CONTENTS

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

TABLE OF CONTENTS (cont'd)

CHAPTER 5 – CONTAINMENT SYSTEM AND OTHER SPECIAL STRUCTURES

LIST OF TABLES

CHAPTER 5 – CONTAINMENT SYSTEM AND OTHER SPECIAL STRUCTURES

LIST OF FIGURES

LIST OF FIGURES (cont'd)

LIST OF FIGURES (cont'd)

CHAPTER 5 – CONTAINMENT SYSTEM AND OTHER SPECIAL STRUCTURES

APPENDICES

5.0 CONTAINMENT SYSTEM AND OTHER SPECIAL STRUCTURES

Containment for this unit is provided by the Reactor Building including its steel liner and the Reactor Building isolation systems. The concrete structure also provides shielding from the fission products which could be airborne in the building under accident conditions. The Reactor Building spray system and the Reactor Building emergency cooling system each provide cooling capability to assure that internal temperature and pressure remain within design conditions following an accident.

The containment for this unit has been designed and constructed to limit the leakage rate to 0.1 percent by weight of contained atmosphere in 24 hours at the design pressure of 55 psig. An additional leakage blocking system, consisting of the penetration pressurization system, has been provided. In the highly unlikely event of an accident it is anticipated that the action of this special system, in conjunction with the design leak rate integrity of the containment, will reduce the postaccident leak rate from containment to essentially zero.

Other special structures are those structures which house equipment which is vital to maintaining containment integrity, vital to safe shutdown of the reactor, or contain quantities of radioactive materials. These structures are designed to withstand a hypothetical aircraft incident.

5.1 STRUCTURAL DESIGN CLASSIFICATION

The plant structures, components, and systems have been classified according to their function and the degree of integrity required to protect the public.

5.1.1 CLASSES OF STRUCTURES AND SYSTEMS FOR SEISMIC DESIGN

5.1.1.1 Class I

Those structures, components, and systems, including instruments and controls, whose failure might cause or increase the severity of a loss of coolant accident or result in an uncontrolled release of radioactivity, and those structures and components which are vital to safe shutdown and isolation of the reactor are designated Class I. When a system as a whole is referred to as Class I, certain portions not associated with loss of function of the systems may have been designated under Class II or III as appropriate. A partial listing of Class I structures, components, and systems follows:

Class I - Structures, Components, and Systems

a. Buildings and Structures

Reactor Building including all penetrations, equipment hatch and air locks, concrete shell, liner, and interior structures.

Auxiliary Building

Fuel Handling Building and fuel storage pools

Control Building

Diesel Generator Building

Intermediate Building (portions designated on Figure 5.1-1)

Intake screen and pump house

Heat exchanger vault and access tunnel-vault to Auxiliary Building

Air intake structure (portion below ground)

Chem Cleaning Building Basin

b. NSSS Components

Reactor internals (including fuel elements and control rods)

Control rod drive mechanisms (and support)

Pressure retaining parts of Incore Monitoring System

Steam generators (and supports)

Pressurizer (and supports)

Reactor coolant piping including pressurizer surge and spray piping and valves

Reactor coolant pumps and motors and supports

c. Engineered Safeguards Systems

Makeup and Purification System (high pressure injection system) including: makeup pumps, makeup tank, letdown coolers, letdown filters, seal return cooler, process and instrument piping and valves

Core flooding tanks including process and instrument piping and valves

Decay Heat System (low pressure injection system) including: decay heat pumps, decay heat coolers, process and instrument piping and valves

Borated water storage tank

Reactor Building Spray System including: spray pumps, spray headers and nozzles, process and instrument piping and valves

Reactor Building emergency air cooling units including: fans and motors, demisters, cooling coils and the duct system supplying the air handlers

Combustible Gas Control System and Hydrogen Monitoring System

Reactor Protection Systems

Engineered Safeguards Actuation System

All Reactor Building piping penetrations and associated Reactor Building isolation valves

d. Vital Cooling Water Systems

Decay Heat Services Cooling Water Systems A and B including: surge tank, pumps, heat exchangers, process and instrument piping and valves

Decay Heat River Cooling Water Systems including: pumps, heat exchangers, process and instrument piping and valves

Nuclear Services Closed Cooling Water System including: pumps, heat exchangers, process and instrument piping and valves

Nuclear Services River Cooling Water System including: pumps, heat exchangers, process and instrument piping and valves

Reactor Building Emergency Cooling Water System including: pumps, process and instrument piping and valves and cooling coils.

Control Building Ventilation Chilled Water System

e. Emergency Power Supply System

Diesel Generators and fuel oil day tanks

DC Power Supply System and inverters

Power distribution lines to equipment required for emergency

Switchgear and power centers supplying the engineered safety features

Main Control consoles and panels

Motor control centers

f. Spent Fuel Cooling System including: spent fuel pumps, heat exchangers, all process and instrument piping and valves, etc.

g. Vital Ventilation Systems:

Ventilation System for Spent Fuel Cooling pump room

Ventilation System for Intake Screen and Pump House

Ventilation System for Diesel Generator rooms

Ventilation system for Nuclear Service Closed Cooling Water System Pumps and Decay Closed Cooling Water System Pump rooms

Reactor Building Purge Containment Isolation Valves

Ventilation system for Emergency Feedwater Pump rooms

Emergency Ventilation System for Control Building, not including the Automatic Temperature Control (ATC) Compressed Air System.

h. Miscellaneous Vital Systems and Components

Feedwater (from FW-V-12 A/B to OTSG's)

Those portions of the Emergency Feedwater System required for decay heat removal including: pumps, condensate storage tanks (excluding hotwell), steam generator pressure and level indications, auxiliary (emergency) feedwater control valves, and atmospheric dump valves.

Underground diesel fuel tank

Instrument and Control Air System

Hydrogen and Nitrogen Supply System Including:

Nitrogen manifold (portions supplying core flood tanks, vital to penetration pressurization system

New and spent fuel storage racks

Reactor building polar crane

Fuel Handling Building crane

River pump service crane

Diesel generator air start tanks and piping

Sodium hydroxide storage tank*

i. Waste Disposal System

Reactor coolant bleed tanks

Miscellaneous waste storage tank

Reactor coolant drain tank

Spent resin storage tank

Used filter precoat tank

Reclaimed boric acid tanks

Neutralizer tank

Neutralizer feed tank

Evaporator condensate storage tanks

Neutralized waste storage tank

---

* - The NaOH Tank is isolated and no longer used for storage of NaOH. It is required to be maintained Seismic Class I to ensure the integrity of connected piping.

Laundry waste tank

Reactor coolant drain tank cooler

Reactor coolant drain tank pump

Liquid outlet piping to second isolation valve downstream from each of the above tanks and the process piping associated with the reactor coolant drain tank

j. Sampling System

All piping and valves inside containment through the isolation valve outside containment (as described in Section 9.2.2)

Class I equipment and components have not been located within Class II structures.

Seismic Class for FLEX (SIf)

SSC which are relied upon for beyond design basis seismic event mitigation strategies (FLEX) are designed or evaluated to ensure the SSC can perform the required FLEX function after a safe shutdown earthquake (SSE). Those SSC are classified class SI or class SIf. [Reference 79]

5.1.1.2 Class II

Those structures, components, and systems which are important to reactor operation but not essential to safe shutdown and isolation of the reactor and whose failure could not result in the release of substantial amounts of radioactivity are designated Class II. A partial listing of Class II structures, components, and systems follows:

Class II - Structures, Components, and Systems

a. Buildings and Structures
None

b. Reactor Coolant, NSSS, and Engineered Safeguards Systems

All piping and valves in vent and drain lines downstream of the second isolation valve off the primary system.

All piping and valves in vent and drain lines downstream of the isolation valve off the Safeguards Systems process piping.

c. Ventilation Systems

Reactor Compartment Ventilation System

Reactor Building Steam Generator Compartment Ventilation System

Fuel Handling Building Ventilation Supply System

Fuel Handling Building Ventilation Exhaust System

Auxiliary Building Ventilation Supply System

Auxiliary Building Ventilation Exhaust System

Controlled Access Area and Hot Tool Room Ventilation System (ventilation system for Control Building first floor)

Control Building Normal Ventilation System, including the Automatic Temperature Control (ATC) Compressed Air System.

d. Refueling Equipment

Fuel handling bridges

Fuel transfer carriages

e. Waste Disposal System

All components, piping, and valves not included in Class I or Class SIf

f. Sampling System

All piping and valves not included in Class I (up to sample sink isolation valves as shown on 302-671)

g. Chemical Addition System

Boric acid mix tank

Boric acid pumps and motors

Filter precoat skid including tank, pumps, piping, and valves

Piping and valves

5.1.1.3 Class III

Those structures, components, and systems which are not related to reactor operation or containment are designated Class III.

Class III - Structure, Components, and Systems

a. Buildings and Structures

Turbine Building (including Auxilliary Boiler House)

Intermediate Building (portions not designated Class I on Figure 5.1-1)

Service Building

Warehouse

Water Pre-treatment Building

Circulating water pump house

Natural draft cooling towers

Mechanical draft cooling tower basin

Access bridge

Circ. Water Chemical Addition Building

Circ. Water Biocide Building

b. Chemical Addition System

Reclaimed water tank

Reclaimed water pressure tank

Reclaimed water pump

Resin add tank

Caustic mix tank

Lithium hydroxide mix tank

Caustic pump

Lithium hydroxide pump

Hydrazine pump

Zinc Injection Skid

c. Reactor Building Leakage Rate Test System

Air cooler

Air dryer

All piping and valves not included in Class I or Class II

d. Miscellaneous Components and Systems

Computer

Intermediate cooling system

Secondary services cooling water system

Turbine hall crane

Condensate-feedwater system (except Class I portion of Feedwater)

Turbine generator and auxiliaries

Condenser-hotwell and auxiliaries

Auxiliary boilers and auxiliaries

Traveling screens

Trash rakes

House Service Air System

Instrument Air System (portions not included in Class I)

New fuel elevator

Ventilation system for cooling Reactor Building penetrations

Reactor Building purge and Kidney Filter System (excluding Purge Containment Isolation Valves and debris screen)

Reactor Building Penetration Air Cooling System fan-coolers

Turbine Building & Heater Bay H&V System

Feedwater Pumps Pedistal Ventilation System

Intermediate Building HVAC System

Fuel Handling Area ESF Ventilation System

Small, local instruments, such as pressure gauges, thermometers, etc. and appurtenances, such as drains and some sampling valves, attached to seismically designed systems have not been procured with seismic analysis, as the failure of these devices due to a seismic disturbance does not impair the proper operation of the related system. They are for normal operational use, and are in no way related to plant safety.

5.1.2 SEISMIC DESIGN BASES

5.1.2.1 Class I Design Bases

All structures, components, and systems designated as Class I have been designed in accordance with the following criteria:

5.1.2.1.1 Class I Structure Design

Seismic loading for each Class I structure has been developed as described in Sections 5.2.4 and 5.4.4.

a. Primary steady state stresses, which include the seismic stress resulting from the design earthquake* ground acceleration of 0.06g acting horizontally and 0.04g acting vertically and occurring simultaneously, have been maintained within the allowable working stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards, e.g., ASME Boiler & Pressure Vessel Code, Building Code Requirements for Reinforced Concrete, ACI 318 and AISC Specifications for the Design and Erection of Structural Steel for Buildings.

b. Primary steady state stress, which include the seismic stress resulting from the maximum hypothetical earthquake** ground acceleration of 0.12g acting horizontally and 0.08g acting vertically and occurring simultaneously, has been limited so that the function of the structure is not impaired as to prevent a safe and orderly shutdown of the plant.

   For loading combinations, see Sections 5.2.3 and 5.4.3.

* The term defined here is equivalent to current terminology used as Operating Basis Earthquake.

** The term defined here is equivalent to current terminology used as Safe Shutdown Earthquake.

5.1.2.1.2 Class I Systems and Equipment Design

Components and systems classified as Class I have been designed in accordance with the following criteria:

a. Primary steady state stresses, which included the seismic stress resulting from the design earthquake ground acceleration of 0.06g acting horizontally and 0.04g acting vertically and occurring simultaneously, have been maintained within the allowable working stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards, e.g., ASME Boiler & Pressure Vessel Code and USAS B31.1.0, Code for Pressure Piping.

b. Primary steady state stress and corresponding strains, which include the seismic stress resulting from the maximum hypothetical earthquake ground acceleration of 0.12g acting horizontally and 0.08g acting vertically and occurring simultaneously, have been limited so that the function of the component, system, or structure is not impaired as to prevent a safe and orderly shutdown of the plant.

   Stresses resulting from the simultaneous occurrence of the maximum earthquake and the loss of coolant accident shall be limited to permit a safe shutdown of the plant. Refer to Chapter 4. For piping stress criteria, refer also to Section 5.4.4.

c. As an alternative to the methodology described in a) and b) above, seismic experience data utilized in accordance with the Seismic Qualification Utilities Group (SQUG) methodology may be used to verify the seismic adequacy of existing, new, modified and replacement items on a case-by-case basis. Such evaluations are performed in a controlled and systematic manner to ensure that the item of equipment is properly represented in the earthquake experience or generic testing classes and that applicable caveats are met. In particular, each new or replacement item must be evaluated for any design changes that could reduce the seismic capacity of the equipment from that reflected in the experience data base, and all such evaluations must be documented in accordance with established procedures. SQUG methodology is applied in accordance with the SQUG Generic Implementation Procedure (Reference 66) and implementation of the SQUG methodology is controlled and documented in accordance with the Exelon Nuclear procedure (Reference 67). All evaluations performed using the SQUG methodology use as input the amplified response spectra contained in EQE Report 50097-R-001 (Reference 68), GPUN Report 990-2362 (Reference 69) and EQE Calculation 42105-C-004 (Reference 70). The use of the SQUG methodology is limited to the scope of equipment covered by the equipment classes described in the SQUG Generic Implementation Procedure (GIP). The methodology is not used to verify the seismic adequacy of equipment not included within the scope of the equipment classes described in the GIP.

   In addition to the restrictions, inclusion rules and caveats described in the preceding paragraph and those specified by the GIP, the following restrictions as described in the Exelon Nuclear procedure (Reference 67) are applied to use of the SQUG methodology for verification of seismic adequacy of equipment at TMI:

   The SQUG methodology is not utilized to verify seismic adequacy of equipment that is part of the systems described in Sections 7.1.1.8, 7.1.3 and 7.3.2 of the TMI FSAR.

1. Comparisons of seismic capacity to demand are performed using Method A from Table 4-1 of the SQUG GIP for equipment located in the Intake Screen and Pump House (ISPH). For equipment located in other areas of the plant, Method B is used unless the use of the Method A is justified as part of the evaluation.

2. The anchor bolt allowables contained in GIP Appendix C, Section C.2, are used only to verify the adequacy of existing equipment, within the scope of USI A-46, which is known to not contain essential relays and any other existing equipment which does not contain any relays. Anchorage for existing equipment that contains essential relays is designed using the allowables specified in the GIP, reduced by one fourth as required by the GIP. Anchorage for all new equipment with or without essential relays is designed using allowable capacities specified in TMI-1 procedures and specifications.

3. Relay evaluations for new or modified safety related relays are performed based on a comparison of demand to capacity. Reliance on chatter being acceptable or on the ability of operators to take manual action is not used as a basis for seismic qualification of new safety related equipment or modifications to existing safety related equipment. Safety related replacement items are evaluated on the basis of a comparison of capacity to demand or on the basis of equivalency to the existing equipment being replaced.

5.1.2.2 Class II Design Bases

Structure, components, and systems classified as Class II have been designed for a ground acceleration of 0.06g in accordance with the requirements of the Uniform Building Code.

In addition to existing analytical/test methodology, seismic experience data applied in accordance with SQUG methodology may be used to verify seismic adequacy on a case by case basis, as described in Section 5.1.2.1.2.c.

5.1.2.3 Class III Design Bases

Structures, components, and systems classified as Class III have been designed in accordance with applicable building code requirements.

5.1.2.4 Site Seismic Surveillance

A strong motion accelerograph has been installed to measure ground motion and structural vibrating response caused by earthquakes occurring in the vicinity of the site. The installation complies with AEC Safety Guide Number 12.

The instrument used for this purpose is a strong motion recording system consisting of:

Two triaxial sensor units installed on the south side of the Reactor Building. One unit is attached directly to the Reactor Building base mat (elevation 281 ft), outboard of the containment wall. The second unit is attached to the Reactor Building ring girder, (elevation 455 ft). Peak reading accelerographs have been installed on the following representative Class I items to verify the seismic response determined analytically:

1. Nitrogen Manifold - Auxiliary Building elevation 331 ft.
2. 15 KVA Inverter IC - Control Building elevation 322 ft.
3. D.H. Surge Tank DE-T1A-Fuel Handling Building elevation 329 ft.

The time history of ground motion and resultant vibrating response will be recorded and stored digitally. The threshold seismic condition alarm will be energized at a preset seismic acceleration for either triaxial accelerometer. The operating basis earthquake alarm will be energized if the event exceeds a preprogrammed seismic spectra curve, based on data obtained from either triaxial accelerometer. These alarms are annunciated through audio and visual indications.

5.1.3 AIRCRAFT PROTECTED STRUCTURES

Those structures that are vital to protection of the reactor coolant pressure boundary, safe shutdown of the plant, and/or contain radioactive materials are designed for the aircraft and associated loadings as specified in Appendix 5A.

The structures designed to the aircraft impact criteria are:

a. Reactor Building

b. Fuel Handling Building

c. Designated portions of the Auxiliary Building (see Figure 5.1-1)

d. Designated portions of the Intermediate Building (see Figure 5.1-1)

e. Control Building

f. Intake screen house and pump house

g. Heat exchanger vault

h. Air intake structure (below ground)

i. Access tunnel-vault to Auxiliary Building

5.2 REACTOR BUILDING

The Reactor Building is a reinforced concrete structure with cylindrical wall, a flat foundation mat, and a shallow dome roof. The foundation slab is reinforced with conventional mild steel reinforcing. The cylindrical wall is prestressed with a post-tensioning system in the vertical and horizontal directions. The dome roof is prestressed using a three way post-tensioning system. The inside surface of the Reactor Building is lined with a carbon steel liner to ensure a high degree of leak tightness during operating and accident conditions. Nominal line plate thickness is 3/8 inch for the cylinder and dome and 1/4 inch for the base.

The foundation mat is bearing on sound rock and is 9 ft thick with a 2 ft thick concrete slab above the bottom liner plate. The cylinder portion has an inside diameter of 130 ft, wall thickness of 3 ft 6 inches, and a height of 157 ft from top of foundation slab to the spring line. The shallow dome roof has a large radius of 110 ft, a transition radius of 20 ft 6 inches, and a thickness of 3 ft. The Reactor Building is shown on Figure 5.2-1, Penetration Details on Drawings 521034, 521044, E-222-54, and Personnel and Equipment Access Opening Details on Figure 5.2-3.

The Reactor Building has been designed to contain radioactive material which might be released from the core following a loss of coolant accident (LOCA) at a maximum leak rate of 0.1 percent by weight of contained atmosphere in 24 hours at the design pressure. The prestressed concrete shell ensures that the structure has an elastic response to all loads and that the structure strains are limited such that the integrity of the liner is not compromised. The liner has been anchored so as to ensure composite action with the concrete shell.

The general construction sequence for the Reactor Building has been as follows:

a. Immediately after excavation of the rock for the foundation and a dewatering system was established, a lean concrete fill was placed to seal the rock to prevent weathering.

b. After the foundation was poured, the knuckle and bottom plate of the liner was installed and tested. Concrete was then placed on top of the base of the liner.

c. The liner was erected and individual welds tested prior to the placing of reinforcement, tendon conduit, and concrete. Concrete work on the cylinder proceeded prior to completion of the cylindrical portion of the liner.

d. The dome liner was erected and individual welds tested prior to the placing of dome reinforcement, tendon conduit, and concrete.

e. The prestressing tendons were installed, stressed, and sealed off with end caps.

To support the 2009 Steam Generator Replacement (SGR) Project, a Containment Opening was made in the Reactor Building concrete wall and liner plate at the 290^{o} azimuth, and directly above the existing Equipment Hatch to gain building access for rigging and handling of the steam generators. The design for this activity ensured that the opening area was restored to a condition meeting Reactor Building design requirements.

5.2.1 STRUCTURAL DESIGN PARAMETERS

The Reactor Building is a Class I structure as defined in Section 5.1.1. The internal free volume is at least 2.00 x 10^6 ft^3. It is designed for a positive internal pressure of 55 psig with a coincident temperature of 281°F at accident conditions, an additional positive internal pressure of 3 psig during a tornado, and a negative internal pressure of 2.5 psig during normal operation of the plant (see Table 5.2-2). Due consideration was given to the dead load, live load, temperature gradients, and penetrations at accident and working conditions. The functional requirements for the liner and penetrations are covered in detail in Section 5.2.2.4.

The design criteria for the Reactor Building are covered in Subsection 5.2.3 and include design moments, shears, deflections, membrane forces, and stresses.

5.2.1.1 Design Leakage Rate

The Reactor Building has been designed to limit the leakage rate to 0.1 percent by weight of contained atmosphere in 24 hours at the design pressure.

5.2.1.2 Design Loads

The design loads for the Reactor Building have been determined based on operating and accident requirements, as specified below, in addition to regular loads as required by applicable codes.

Scaled plots of stress resultants, stress couples, shear, and deflections for the individual loads are shown on Figures 5.2-4 through 5.2-24.

5.2.1.2.1 Loss of Coolant Accident

a. Postulated Accident Conditions

The Reactor Building encloses the reactor and the Reactor Coolant System and is designed to ensure that an acceptable upper limit of radioactive material leakage will not be exceeded under the maximum LOCA as described in Chapter 14, Safety Analysis. The accident is based on a double-ended pipe break in the main coolant system and produces pressures and temperatures that are influenced by the safeguard system, heat sinks, and energy sources. This is described in Chapter 6, Engineered Safeguards, and Chapter 14, Safety Analysis.

b. Energy and Mass Releases

Additional energy and mass will be available for release into the Reactor Building from the following sources:

1) Stored heat in the reactor

2) Reactor decay heat

3) Stored heat in the Reactor Coolant System

4) Metal-water reaction

The energy released from these sources is discussed in Chapter 14.
The energy contribution from the secondary steam system is not included in the calculations for the Reactor Building design pressure and temperatures. The supports for the Reactor Coolant System are designed to withstand the forces associated with a break in the reactor coolant pipe so that a rupture in the secondary system will not be considered to act simultaneously with a reactor coolant pipe break.

c. Contribution of Engineered Safeguards Systems

The contribution of the Engineered Safeguards Systems is discussed in Chapter 6. These systems will be actuated to minimize the accident conditions by removing heat from the reactor core, inserting negative reactivity into the reactor, and isolating the Reactor Building.

These safeguards systems are:

1) A high pressure injection system (Makeup and Purification)

2) A low pressure injection system (Decay Heat Removal)

3 A core flooding system

4) A Reactor Building emergency cooling system

5) A Reactor Building spray system

6) A Reactor Building isolation system

5.2.1.2.2 Dead Load

The dead load consists of the weight of the complete structure as shown on Figure 5.2-1. Scaled plots of stress resultants, stress couples, shear, and deflections for dead load are shown on Figure 5.2-4.

5.2.1.2.3 Prestress Load

The concrete shell has been prestressed sufficiently to eliminate tensile stresses due to membrane forces from design loads. Membrane tension due to factored loads is as described in Item a. of Subsection 5.2.3.2.5. The prestress load applied to the Reactor Building is divided into three groups:

a. Local effect due to temporary jacking force equal to 80 percent of maximum guaranteed ultimate capacity of the wires. The tendon hardware has been designed to develop the ultimate capacity of the tendon for this maximum load as described in Section 5.2.2.3. In addition, stresses in the concrete, such as bearing stresses, bursting stresses, and spalling stresses, were checked.

b. Initial prestress force equal to 70 percent of ultimate capacity when locked off (shimmed in place). The supporting concrete and anchorage reinforcement has been analyzed and designed for this load.

c. Final prestress force at the end of the plant life, that is, 40 years. This load takes into account time dependent losses such as shrinkage, creep, and steel stress relaxation. To support the 2009 SGR Project, a Containment Opening was made into the Reactor Building concrete wall and liner plate at the 290^{o} azimuth, and directly above the existing Equipment Hatch to gain building access for rigging and handling of the steam generators. Design evaluations of the restoration of the Containment Opening ensured that the concrete stress in the area of and surrounding the restored opening at the end of plant life was within the applicable code allowables under design basis loading conditions. Note that these design evaluations utilized a 60 year plant lifetime, which is sum of the total of the initial 40 year Operating License and a potential 20 year License Renewal. These design evaluations relied upon an existing concrete compressive strength of 7579 psi (increase from minimum compressive strength value of 5000 psi at 28 days for original concrete) and a concrete compressive strength of 6800 psi at 7 days for the replaced concrete at the time of the steam generator replacement.

Scaled plots of stress resultants, stress couples, shear, and deflections for prestress loads are shown on Figures 5.2-5, 5.2-6, and 5.2-7.

5.2.1.2.4 Live Load

Applicable loads on the Reactor Building shell during normal operation and factored loads are:

	During Operation	Factored Loads
Pipe penetration	Normal Reactions	Pipebreak & Earthquake
Pipe supports	Normal Reactions	Pipebreak & Earthquake
Polar crane	Normal Reactions	Earthquake

The structures inside the Reactor Building have been designed for normal operating loads and loads caused by LOCA as described in Subsection 5.2.5.

5.2.1.2.5 Wind, Snow, or Ice Load

The Reactor Building has been designed to withstand a snow or ice load of 35 lb/ft^2. The wind velocity as a function of height and drag coefficients has been established on the basis of Reference 1.

The basic wind velocity (wind 30 ft above ground) has been based on the fastest mile of wind with a 100 year period of recurrence. The Three Mile Island site has been classified as a coastal area and the wind velocity as a function of height is expressed as:

$$V_z = V_{30}\left[\frac{z}{30}\right]^x$$

When:

V_Z = velocity at height z above grade

V_{30} = fastest mile of wind at 30 ft above ground

z = height above grade

x = exponent for Ekmon spiral ranges from 0.3 to 0.143

For the Three Mile Island site, the following values have been used:

V_{30} = 80 mph

z = 144 ft (The pressure at this elevation was considered for the entire height of the cylinder)

x = 0.263

V_Z = 118 mph

The wind force has been applied in the vertical plane as shown on Figure 5.2-8.

The wind pressure can then be defined as:

$$g = 1/2\ PV_Z^{\ 2}$$

Where P is a function of air density

For standard air at 0.07651 lb/ft^3 and V^2 in mph:

$$g = 0.002556\ V_z^2\ Psf = 0.0000177\ V_z^2\ psi$$

or:

$$g = 0.246\ psi$$

The wind pressure distribution in the horizontal plane is shown on Figure 5.2-8 and is defined by the Fourier series.

$$\begin{aligned} f(x) = {} & -0.1682 + 0.0738 \cos x + 0.228 \cos 2x + 0.101 \cos 3x \\ & -0.0123 \cos 4x + 0.0033 \cos 5x + \\ & \ 0.0164 \cos 6x - 0.002 \cos 7x - 0.0124 \cos 8x \\ & -0.003 \cos 9x \end{aligned}$$

Scaled plots of stress resultants, stress couples, shear, and deflections for wind load are shown on Figures 5.2-9 and 5.2-10.

5.2.1.2.6 Tornado Load

The Reactor Building has been designed to withstand short term tornado loadings including tornado generated missiles. The tornado design requirements are:

a. Tangential wind velocity of 300 mph. (A gust factor of 1.3 has been used for those structures designed for tornado loadings.)

b. An external vacuum of 3 psig.

c. Missile equivalent to a utility pole 35 ft long, 14 inches in diameter, weighing 50 pcf, and traveling at 150 mph.

d. Missile equivalent to a 1 ton automobile traveling at grade at 150 mph.

A 300 mph wind has been applied in accordance with standard wind design practice and using applicable pressures, shape factors, and drag coefficients from Reference 1. The vacuum of 3 psig is conservative considering that most measured pressure drops have been in the order of magnitude of 1.5 psig.

Scaled plots of stress resultants, stress couples, shear, and deflections for tornado load are shown on Figures 5.2-11 and 5.2-12.

5.2.1.2.7 External Pressure

The Reactor Building has been designed for an external atmospheric pressure of 2.5 psi greater than that of the internal pressure that could be caused by an accidental discharge from the spray system.

5.2.1.2.8 Internal Pressure

The design pressure for the Reactor Building is 55 psig caused by a hypothetical LOCA. The vessel is also designed with the assumption that a tornado reduces the external pressure to 3 psig below atmospheric applied as an equivalent internal pressure. (See Table 5.2-2 for load combinations). The Reactor Building was pressure tested at 63.3 psig. Scaled plots of stress resultants, stress couples, shear, and deflections for internal pressures of 55, 63.3, 68.75, and 82.5 psig are shown on Figures 5.2-13, 5.2-14, 5.2-15, and 5.2-16.

5.2.1.2.9 Operating Temperature

Figure 5.2-17 shows the normal operating temperature profile through the containment shell. The temperature profile shown for normal operation was obtained by using boundary conditions of 110°F inside containment and 21.4°F, which is the lowest average monthly temperature ever recorded by the United States Weather Bureau at Harrisburg, Pennsylvania, outside containment. Lower temperatures will be experienced for periods shorter than 1 month. However, these short term low temperatures only affect a small portion of the wall thickness as proven by preliminary analysis and, therefore, were not considered. Scaled plots of stress resultants, stress couples, shear, and deflections for operating temperature are shown on Figure 5.2-18.

The thermal gradients for the typical wall sections of the Reactor Building during startup, shutdown, and normal operation are shown on Figure 5D-18 of Appendix 5D of the FSAR. The temperature profiles were obtained as described in Section 4 of Appendix 5D.

The thermal gradient used for the design of the Reactor Building wall is the steady-state linear temperature gradient as shown on Figure 5D-18, Winter Normal Operation.

The transient temperature gradients have been considered, and the following conclusions were obtained:

a. The tensile stress on the outside face of the wall is maximum for the steady state temperature gradient used in the design.

b. The tensile force is maximum for the steady state temperature gradient used in the design.

c. The compressive force is maximum for the steady state temperature gradient used in the design.

d. The compressive stress on the inside face of the wall will be greater during startup and shutdown conditions than for the steady state temperature gradient used in the design.

In order to determine the magnitude of the increase in compressive stress during startup and shutdown conditions, the typical wall was analyzed for the temperature gradient existing at one half day after startup. It was found that the maximum fiber stress increased by about 150 psi based on an uncracked section. The 150 psi is equivalent to a percentage increase of approximately 7 percent above the allowable stress of 0.45 f'_c during final prestress conditions. Based on the following reasons, it was concluded that the additional compressive stresses resulting from the startup and shutdown conditions will not be detrimental to the performance of the structure.

a. The allowable stresses of 0.45 f'_c during final prestress conditions and 0.60 f'_c during initial prestress conditions are low when considering the biaxial state of stress in the Reactor Building.

b. The thermal stresses due to startup and shutdown conditions are nonlinear across the wall thickness, thereby limiting the overstress to a small portion of the thickness of the wall.

c. Startup and shutdown conditions are temporary.

d. The calculated stresses which include the linear temperature gradient do not reach the allowable stresses except at localized points at discontinuities.

Based on actual operating experience, an analysis has been performed using a containment temperature of 130°F for elevations above 320' and 120°F for elevations below 320'. This analysis was performed in accordance with Working Stress Design provisions of ACI318-63. Acceptance criteria was based on the allowable stresses specified in Section III, Division 2, issued January 1, 1975 of the ASME Boiler and Pressure Vessel Code. The stresses resulting from the increased operating temperature were found not to be detrimental to the performance of the structure. In addition, the most critical section under the loading conditions specified above was checked using the Ultimate Strength Design provisions of ACI318-63. A safety factor of 1.63 was found using this design method and confirmed the adequacy of the structure.

5.2.1.2.10 Accident Temperature

The design basis event for the design of the RB structure is a LBLOCA. The peak containment temperature after a MSLB may exceed the RB design temperature (281°F). The effect of RB pressure and temperature conditions after a MSLB (Figure 6B-15 & 6B-18) on the RB structure were evaluated (Reference 78) and confirmed that the LOCA is the limiting event.

Figure 5.2-19 shows the accident LOCA temperature gradient throughout the concrete wall of the Reactor Building at time T = 10,000 seconds after initiation of an accident. Figure 5.2-19 also shows the temperature gradient across the concrete wall and the liner plate at various time intervals after initiation of an accident. The temperature profiles were obtained as described in Section 4.4 of Appendix 5D.

The temperature gradient used for the design of the Reactor Building is a combination of the liner plate temperature of 281°F occurring at T = 700 seconds and the linear concrete temperature gradient occurring at T = 0 seconds. This means the nonlinear portion of the temperature gradient through the concrete at time T = 700 seconds is ignored. The linear concrete temperature gradient used for design results in a maximum tensile stress at the typical wall of approximately 99 percent of the value that would be obtained if the nonlinear concrete temperature gradient at T = 700 seconds were used.

The thermal stresses resulting from the temperature gradients occurring at the time intervals shown on Figure 5.2-19 were considered, and it was found that the maximum tensile stress and maximum tensile force occur at time T = 700 seconds after initiation of an accident. This is due to the fact that the tensile stress in the concrete at any given time is a function of both the temperature gradient through the concrete and the existing temperature of the liner.

The increase in compressive stress on the inside (hot) face of the concrete due to the nonlinear portion of the accident temperature gradients is not critical because of the following:

a. The accident temperature occurs in combination with the accident pressure which relieves the high compressive stresses due to the prestress forces.

b. The allowable compressive stress increases from 0.45 f'_c or 0.60 f'_c during normal operation and to 0.85 f'_c during accident conditions.

5.2.1.2.11 Earthquake Load

The vertical component is taken as two thirds of the horizontal component of the response spectrum.

The seismic design of the Reactor Building is based on the response to a ground acceleration as described in Section 5.1.2.1.

The Reactor Building shell was analyzed for horizontal and vertical ground motion separately. The respective vertical and horizontal seismic components at any point on the shell have been added by summing the absolute values of the response (i.e., stress, shear, moment, or deflection) of each contributing mode due to vertical motion to the corresponding absolute values of the response of each contributing mode due to horizontal motion.

Damping Factors

The following damping factors have been applied to the maximum earthquake in the seismic design of components and structures:

	Component or Structure	Percent of Critical Damping	
1.	Reactor Building	2.0	
2.	Concrete support structures inside the Reactor Building	2.0	
3.	Assemblies and structures		
	a) Bolted or riveted	2.5	
	b) Welded	1.0	
4.	Vital piping systems	5.0	@ $\leq$ 10 Hz
		Linearly decreasing from 5.0 to 2.0	Between 10 Hz to 20 Hz
		2.0	@ $\geq$ 20 Hz
5.	Other concrete structures above ground	5.0	

Scaled plots of stress resultants, stress couples, shear, and deflections for seismic load are shown on Figures 5.2-20, 5.2-21, 5.2-22, 5.2-23, and 5.2-24.

The modal damping values expressed as a percentage of critical damping values shown in Table 1 of U.S. Atomic Energy Commission Regulatory Guide 1.61 should be used for viscous modal damping for all modes considered in an elastic dynamic seismic analysis of NRS/ITS electrical raceway systems (Reference 63).

5.2.1.2.12 Polar Crane Load

The polar crane is a Class I component and has been designed along with its supporting structure to satisfy the criteria for Class I structures and components.

In order to ensure stability during an earthquake, the crane trolley and bridge are restrained against overturning at all times during unit operation.

The standard WF-section used as a crane bracket has been welded directly to a continuous bent plate ring that is thicker than the liner. See Figure 5.2-25. The wheel loads from the polar crane are carried by the runway girder and transferred to the brackets. The bracket moment and shear, as a result of the wheel loads is restricted by two built-up tees embedded in the wall.

The shear is transferred from the bracket web plate, into the stiffener plates, and resisted by bearing on the tee web plates.

The horizontal load components are transferred directly, at the elevation of the top flange of the runway girder, to structural tees embedded in the wall.

5.2.1.2.13 Groundwater and Floods

The top of the Reactor Building foundation mat is at elevation 279 ft. The groundwater elevation is approximately 280 ft. The foundation slab design takes into consideration groundwater pressure. Fluctuations in the groundwater due to flood and normal variation have been given due consideration in designing the foundation mat. Refer to Sections 2.6, Hydrology, and 2.7, Engineering Geology and Foundation Considerations.

5.2.1.2.14 Aircraft Impact Loadings

The loadings due to aircraft impact described in Appendix 5A have been evaluated on the apex and other locations of the dome and at significant locations on the cylinder. The results of these evaluations, as presented below, provide assurance that the hypothetical objects are prevented from penetrating the containment structure or causing it to collapse.

The interior structures and major components are supported from the base mat independent of the containment shell. Consequently, their response due to the hypothetical aircraft incident is minimal (i.e., less than that associated with the seismic disturbance). The only other significant items attached to the containment shell include the polar crane and the spray header piping. The trolley and trucks of the crane are restrained during plant operation to ensure that no overturning will occur due to the most severe disturbance resulting from either the maximum seismic load or the hypothetical aircraft incident. The spray header piping has been checked to ensure collapse will not occur. It can therefore be concluded that a LOCA will not occur due to the disturbance associated with the hypothetical aircraft incident and there will be no release of radioactive material to the environment.

5.2.1.2.15 Other Missiles

Tornado Missiles

The Reactor Building has been checked to verify its integrity due to tornado missiles as listed in Subsection 5.2.1.2.6.

5.2.2 MATERIALS AND SPECIFICATIONS

This design is based upon specifications giving acceptable limitations of physical and chemical properties for the structural materials used.

5.2.2.1 Concrete

Structural concrete work has been performed in accordance with ACI 301-66 (Reference 2), modified as necessary for the more exacting requirements of the Reactor Building. The prestressed concrete has a minimum compressive strength of 5000 psi in 28 days. The base mat also consists of 5000 psi strength concrete.

Portland Cement conforms to ASTM C-150, Type II (Reference 3), modified for low heat of hydration.

Concrete aggregates conform to the Pennsylvania Department of Highways Specifications (Reference 4). The type and size of aggregate, slump, and additives have been established to minimize shrinkage and creep. Neither calcium chloride nor any admixture containing calcium chloride or other chlorides, sulphides, or nitrates were used. Mixing water was controlled so as not to contain more than 100 ppm of each of the above chemical constituents.

The restoration of the temporary opening in the Reactor Building that was created for access to enable the change-out of the OTSGs was performed utilizing a certified concrete mix design similar to that used in original construction, except that it was modified to reduce creep and shrinkage and qualified for a minimum compressive strength of 6800 psi in 7 days. Sufficient testing was performed prior to placement to ensure that the design strength was attainable.

5.2.2.2 Mild Steel Reinforcement

The mild steel reinforcing for the Reactor Building provides capacity in bending and shear only and therefore was designed in accordance with ACI 318-63 (Reference 5). In addition, a minimum amount of mild steel reinforcement (0.15 percent of the wall section) was placed near the exposed surface of the concrete shell for crack control.

The mild steel reinforcing was deformed bars conforming to ASTM A 615-68, Grade 40 (Reference 6). It is to be noted that Grade 40 reinforcing steel, which is the type material used for this structure, is the lowest strength material commonly used for construction. Furthermore, no reliance is placed on special high strength properties and therefore any interchange of higher strength material would not jeopardize the strength of the structure.

The allowable bond and anchorage stresses used for the mild steel reinforcing were specified by the ACI 318-63 Code (Reference 5). Bond for deformed bars relies chiefly on the bearing of the deformations on the concrete and not on the adhesion or friction between the concrete and steel. Because crack-controlling reinforcement is provided, the width of the cracks occurring parallel to the reinforcing will be relatively small and should not affect the bearing capacity of the deformations or the bond characteristics of the deformed bars. It is also to be noted that the anchorage and bond stresses allowed by the ACI 318-63 Code (Reference (5) for slabs in which cracks do occur parallel with the reinforcing are equal to the values allowed by the ACI 318-63 Code (Reference 5) for beams in which cracks normally do not occur in a direction parallel to the reinforcing.

Splices at points of maximum tensile stress were avoided insofar as possible. Alternate splices for concrete reinforcement were staggered a minimum of 6 ft when the center to center spacing of bars was less than 12 inches.

A retaining wall provides access for field installation and tensioning of tendons. This retaining structure will also keep ground water away from the containment shell, thereby eliminating the potential for corrosion of rebar due to stray currents. Therefore, no tack welding of reinforcement was performed. Arc welding was not used to splice concrete reinforcement. Tension splices for bar sizes larger than No. 11 were made with Cadweld splices.

Arc welding was not permitted as a means of splicing mild steel reinforcement during original construction of the Reactor Building. An exception to this statement has been made for the following case during the 2009 SGR Project. To support the 2009 SGR Project, a Containment Opening was made in the Reactor Building concrete wall and liner plate at the 290^{o} azimuth, and directly above the existing Equipment Hatch to gain building access for rigging and handling of the steam generators. In order to restore the concrete opening, new rebar was spliced to existing No. 11 and No. 8 rebar using BarSplice BPI XI swaged couplers, which consisted of a cold-swaged sleeve installed using a portable hydraulic press and a pair of swaging dies. As an acceptable design equivalent to cadweld splicing of rebar, these mechanical splice devices complied with ASME Section III, Div. 2, Subsection CC to be capable of developing not less than 125% of the specified yield strength of the bars in question. For areas where mechanical splices could not be used, direct-butt fusion (arc) welded splices were employed. Welding of reinforcement was localized to select areas at the corners of the construction opening to minimize the amount of reinforcement with residual stresses. Welded splices were welded and inspected in accordance with AWS D1.4-1998, Structural Welding Code - Reinforcing Steel. Welding of reinforcement is permitted in both the original construction code (ACI-318-1963, Chapter 805) as well as the SGRP Design Code ACI-318-2002, Section R12.14.3) as long as the welded splice develops at least 125 percent of the specified yield strength of the reinforcing bar. The rebar butt-welds will be at least 125% of the rebar yield strength in accordance with AWS D1.4-98.

To support the 2009 SGR Project, a Containment Opening was made in the Reactor Building concrete wall and liner plate at the 290^{o} azimuth, and directly above the existing Equipment Hatch to gain building access for rigging and handling of the steam generators. In order to restore the concrete opening, the new rebar was spliced to original rebar using BarSplice BPI XI swaged couplers, which consisted of a cold-swaged sleeve installed using a portable hydraulic press and a pair of swaging dies. As an acceptable design equivalent to cadweld splicing of rebar, these mechanical splice devices complied with ASME Section III, Division 2, Subsection CC in being capable of developing not less than 125 percent of the specified yield strength of the bars in question. For areas where mechanical splices could not be used, direct-butt fusion welded splices were employed. Welded splices were welded and inspected in accordance with AWS D1.4-1998, Structural Welding Code - Reinforcing Steel. With the development of at least 125 percent of the specified yield strength in tension or compression for the rebar, as required, Section R12.15.4 of ACI-318-02 did not require staggering alternate swaged-coupler splices in restoring the Containment Opening. Additionally, type B lap splices were used to splice new rebar to new rebar, staggered a minimum of 6 ft when center to center spacing of bars was less than 12 inches. New rebar employed was Type A615 Grade 60, as opposed to the Type A615 Grade 40 rebar that was used in the original construction of the Reactor Building wall. Use of Type A615 Grade 60 rebar for this application is acceptable per ASME Section III, Division 2 CC4333.2.4, which permits splicing of different grades of rebar, provided that the original splice qualification is made to the higher of the two grades being joined in the production splice. A new vertical rebar configuration and additional horizontal rebar was added to the outside face of the Containment Opening in order to meet design acceptance criteria due to post-construction loading. Additionally, a new interior rebar mat was installed adjacent to the liner plate within the Containment Opening to carry the bending loads associated with post-tensioning of tendons on the inside face.

5.2.2.3 Post-Tensioning System

The prestressing system used for the Reactor Building is the BBRV system using a maximum of 169 1/4 inch diameter wires (See Figure 5.2-26). The wires consist of high tensile steel, bright, cold drawn, and stress relieved conforming to ASTM A 421-65T, Type BA (Reference 7). This type of wire is not susceptible to stress corrosion such as hydrogen cracking (Reference 11). The BBRV system uses parallel wires with cold-formed buttonheads at the ends which bear upon a perforated steel anchor head, thus providing a positive mechanical means for transferring the prestress force. The anchorage hardware is designed and fabricated for the use of 170 wires. However, one hole located on the outer perimeter of the holes in the anchor head was used to accommodate a removable unstressed surveillance wire.

Details including dimensions of anchorage components are as shown in Appendix 5B which also describes tests conducted to confirm the adequacy of the hardware.

To support the 2009 SGR Project, a Containment Opening, was made in the Reactor Building concrete wall and liner plate at the 290^{o} azimuth, and directly above the existing Equipment Hatch, to gain building access for rigging and handling of the steam generators. The Containment Opening was restored to its original design requirements using the following:

1. New vertical and horizontal tendon sheaths within the SGR construction opening area were installed at the original locations to replace those removed for creation of the opening. The tendon sheaths used carbon steel pipe rather than the spiral wound thin gauge corrugated steel sheaths originally used. The sheaths were secured and otherwise supported to allow installation of the tendons prior to placing concrete. The sheath-to-sheath connections were sealed to prevent concrete intrusion or grease leakage.

2. The following tendons were removed for creation of the opening and were replaced with new tendons which were tensioned as a part of the closure process:

    a. Verticals - V131 thru V140;
    b. Horizontals - H-46-030 thru H-46-039; and,
    c. Horizontals - H-51-028 thru H-51-039.

3. The following tendons were de-tensioned during creation of the opening, and re-tensioned as a part of the closure process:

    a. Verticals - V113 thru V130;
    b. Verticals – V141 thru V157;
    c. Horizontals - H-46-028 thru H-46-029;
    d. Horizontals - H-46-040 thru H-46-042; and,
    e. Horizontals - H-51-040 thru H-51-042.

4. Re-tensioning of the horizontal and vertical tendons commenced after a 72 hour cure time and the concrete reached a compressive strength of 5800 psi. Tendon re-tensioning as controlled and sequenced to preclude overstressing of the concrete. The

tensioning force met the requirements of the UFSAR, and the minimum value required at the end of the plant life.

5. The tendons were greased after tendon tensioning using Visconorust 2090P-4 grease.

5.2.2.3.1 General Criteria

The tendons, including anchorage components, and the anchorage zone were designed in accordance with the criteria for the Reactor Building as defined in Subsection 5.2.3 and are classified as Class I elements per the definition contained in Section 5.1.1.1.

The strength of the Reactor Building is not dependent upon developing the ultimate strength of the tendons and anchorage components, but is controlled by stresses and strains in the concrete and liner. However, the tendons and hardware have been designed and tested to severe criteria to ensure development of the ultimate capacity of the tendons. For example, 10 percent of the anchor nuts are proof-tested for the ultimate load of the tendons as an acceptance criteria. In addition, performed tests indicate that the nut has an ultimate capacity approximately 50 percent greater than the minimum specified ultimate strength of the tendon.

The anchorage zone has been designed in accordance with the ACI codes and the PCI recommendations. The reinforcement in the anchorage zone has been designed to the provisions of the ACI 318-63 (Reference 5). For example, the reinforcement in the anchorage zone has been designed for 70 percent of the ultimate capacity of the tendon with an allowable stress of 20 ksi (for 40 ksi yield material).

When the reinforcement required for "Working Stress Design" under construction and operating conditions was greater at a section than that required for "Ultimate Strength Design" under factored loads, the reinforcement required for "Working Stress Design" governed.

In addition to the load combination described in Subsection 5.2.3.2 where design is based upon an "Ultimate Strength Design" approach, the Reactor Building is also designed to accommodate construction and the controlling operating load combination in accordance with ACI 318-63. The analysis of the Reactor Building for operating condition includes an evaluation of thermal transients for initial startup, startup during cold weather, protracted shutdown, and seasonal variations during operating condition.

5.2.2.3.2 Anchorage Zone Analysis

Examples of the analysis and design of the anchorage zones for the prestressed tendons are presented in Appendix 5D. The factors considered in the design of the anchorage zones are:

a. Bearing stresses

b. Spalling stresses

c. Transverse tensile splitting stresses in vertical and horizontal direction

d. Transfer of unbalanced tendon forces

The reinforcement in the anchorage zone was designed for the most unfavorable condition to control possible cracks in the concrete.

A check of the design was performed using the method described in Appendix 5D.

5.2.2.3.3 <u>Prestressing Tendons</u>

a. Friction and Efficiency Tests

A series of tests have been conducted on curved tendons using the BBRV wire system to evaluate the efficiency of the tendon and the friction factors. These tests are as follows:

<u>Location</u>	
Frick, Switzerland	121-7 mm wires, 180° horizontal curvature
South Haven, Michigan	90-1/4 inch wires, approx. 107° horizontal curvature
Middletown, Pennsylvania	90-1/4 inch wires, approx. 30° horizontal and 50° vertical curvature

The combined results of the above give a coefficient of friction of 0.1217 and a wobble coefficient of 0.000343. The details of these tests were reported in Reference 8. The coefficients used for design are 0.16 and 0.0003, respectively.

Three tests from the Frick series were also used to determine the ultimate strength efficiency of the curved tendon. These tests indicate that the tendon had not less than 95 percent efficiency. This efficiency is based upon 180 degree curvature and stressing from one end of the tendon. However, the design of the Reactor Building does not require that the ultimate strength of the tendon be reached. The load in the tendon will not be greater than 70 percent of the minimum guaranteed ultimate strength under any combination of loadings.

b. Tendon Redundancy

Section 5.2.3.2.5, Loading Conditions-At Factored Loads, states that the load capacity determined for tensile membrane stresses will be reduced by a capacity reduction factor "ϕ" of 0.95 which will provide for the possibility that small variations in material strengths, workmanship, dimensions, and control may combine to result in under capacity. Considering the above "ϕ" factor, it is possible to have a symmetrical failure of up to 5 percent of the tendons and meet the design criteria for the factored loads.

A study was performed to determine the effect of the total loss of three adjacent 169 wire tendons either vertically or circumferentially in the cylinder or in the dome. This study indicated that the loss of three adjacent tendons will not jeopardize the capability of the Reactor Building to withstand the design accident loading condition.

c. Low Temperature Characteristics

The Reactor Building has been designed so that it is not susceptible to a low temperature brittle fracture. This does not mean, however, that each element in the structural system which is a ferritic material complies with an NDTT criterion of 30°F below minimum service metal temperature.

The transition temperature, defined by an impact test, was not considered relevant for the design of the concrete containment shell for the following reasons:

1) History

a) To the best of our knowledge, no prestressed or reinforced concrete members have failed due to a low temperature brittle fracture.

b) No suspension spans, which use high strength wire, have failed due to a low temperature brittle fracture.

c) To the best of our knowledge, no prestressed concrete primary vessels, during proof test or when tested to destruction, have failed due to a low temperature brittle fracture.

d) Modern design concepts, such as ultimate strength design and energy absorption methods for a seismic design, acknowledge that the criterion of NDTT, as defined by Charpy impact tests, is not applicable to uniaxially stressed rebars and tendons.

2) Uniaxial Stress

Essentially only uniaxial stresses are applied to the mild steel reinforcing and tendon material. The triaxial stresses which may induce brittle behavior at higher temperatures by restricting plastic flow are thus avoided. Field inspection ensures the absence of mechanical and metallurgical notches from the material.

3) Strain Rate

The tendons are stressed more highly during the jacking operation than they are during any design condition. Prestress losses exceed the minimal increase of tendon stress during the load application. Because the strain during pressure loading of the prestressed elements is primarily a function of the concrete strains and because the application of the accident pressure load is considerably slower than the application of an impact load, the steel elements would not experience the high rate of strain associated with impact loading.

4) Residual Stresses

No splicing of mild steel reinforcement by arc welding is permitted, thereby avoiding residual stresses produced by welding. The tendon material is stress relieved, thereby beneficially tempering heat affected zones and favorably altering the material's microstructure.

Arc welding was not permitted as a means of splicing mild steel reinforcement during original construction of the Reactor Building. An exception to this statement has been made for the following case during the 2009 SGR Project. To support the 2009 SGR Project, a Containment Opening was made in the Reactor Building concrete wall and liner plate at the 290° azimuth, and directly above the existing Equipment Hatch to gain building access for rigging and handling of the steam generators. In order to restore the concrete opening, new rebar was spliced to existing No. 11 and No. 8 rebar using BarSplice BPI XI swaged couplers, which consisted of a cold-swaged sleeve installed using a portable hydraulic press and a pair of swaging dies. As an acceptable design equivalent to cadweld splicing of rebar, these mechanical splice devices complied with ASME Section III, Div. 2, Subsection CC to be capable of developing not less than 125% of the specified yield strength of the bars in question. For areas where mechanical splices could not be used, direct-butt fusion (arc) welded splices were employed. Welding of reinforcement was localized to select areas at the corners of the construction opening to minimize the amount of reinforcement with residual stresses. Welded splices were welded and inspected in accordance with AWS D1.4-1998, Structural Welding Code - Reinforcing Steel. Welding of reinforcement is permitted in both the original construction code (ACI-318-1963, Chapter 805) as well as the SGRP Design Code ACI-318-2002, Section R12.14.3) as long as the welded splice develops at least 125 percent of the specified yield strength of the reinforcing bar. The rebar butt-welds will be at least 125% of the rebar yield strength in accordance with AWS D1.4-98.

5) Fatigue Tests

Lehigh University has advised that fatigue tests performed on 270K wire at room temperature and at 0°F show no change in properties at the lower temperature.

These facts, coupled with experience to date in concrete construction, indicate that transition temperatures, as defined by an impact test, are not indicative of the behavior of concrete structures at low temperatures whether they be mild steel reinforced or prestressed.

Properties at temperatures substantially lower than the lowest service temperature have been established for cryogenic LNG storage tanks. Further testing of one or more full size prototype stressed tendon anchorage systems were performed by increasing tendon stress at a temperature at least 30°F lower than the lowest service metal temperature.

d. Fabrication Procedures

1) Tubing (sheathing): Tubing is galvanized steel. Material purchased in multiples of required length.

a) Operations

1. Tubing is saw-cut to length.

2. Tubing is inserted in central hole of bearing plate and secured using a seal weld using a low hydrogen electrode.

b) Effects

"Carbon steel material with less than 0.30 percent carbon and wall thickness less than 3/4 inch generally does not crack during or after welding nor do any significant changes take place in their physical properties or metallurgical structure" (Reference 9).

2) Bearing Plate: Material purchased as hot-rolled plate.

a) Operations

1. Plate is flame-cut to size.

2. Central hole is flame-cut.

3. Four mounting holes are jig-drilled and tapped.

4. Bearing plate is then painted with No. 722 epoxy zinc rich coating.

b) Effects

1. "There is no detrimental effect when low and mild carbon nonalloy steels containing less than 0.35 carbon are flame-cut" (Reference 10). The steel surface after flame-cutting has reached a temperature of approximately 1600°F and the removed metal oxidized to form Fe_3O_4. Residual quantities of this oxide will remain on this surface. The effect of heating and subsequent air cooling is similar to normalizing.

2. In machining the drilled and tapped holes, small residual stresses are produced on the machined surface. Also, this operation may produce stress risers. The location of these holes (outside the stressing chair area) make this of little concern.

3) Transition Funnel: Material purchased as hot-rolled or cold-rolled sheet.

a) Operations

Funnel is spun-formed to shape and galvanized.

b) Effects

Spinning is a cold working operation. The funnel, however, is not part of the structural system.

4) Split Shims: Material purchased as hot-rolled plate.

a) Operations

Plate is flame-cut to size and shape.

b) Effects

"There is no detrimental effect when low and mild carbon nonalloy steels containing less than 0.35 carbon are flame-cut" (Reference 10). The steel surface after flame-cutting has reached a temperature of approximately 1600°F and the removed metal oxidized to form Fe_3O_4. Residual quantities of this oxide will remain on this surface. The effect of heating and subsequent air cooling is similar to normalizing.

5) Anchor Head (Washer or Composite Washer): Material purchased as bar, saw-cut to length sufficient for facing both sides.

a) Operations

1. Piece is machined to finished size.

2. Wire holes are drilled.

3. Wire holes are deburred and checked for tolerances.

4. Part is heat-treated, per Specification MIL-H 6875b, to a hardness of 40 to 44 measured on the Rockwell C scale. In order to address industry embrittlement concerns affecting anchor heads, ductility was enhanced for the new anchor heads employed in restoring the Containment Opening that was created to support the 2009 SGR Project. The new anchor heads were heat-treated to a lower hardness (335 to 370 measured on the Brinnell Scale, which is approximately 36 to 40 measured on the Rockwell C scale).

5. Threads are then machine-cut and checked for tolerances.

b) Effects

1. Any residual stress resulting from machining prior to heat-treating is essentially cancelled during heat-treating.

2. During heat-treating, parts are placed in a controlled atmosphere oven and brought up to approximately 1550°F, which is above the upper critical range (austenitic range) then quenched in oil to approximately 800°F and air-cooled to ambient. This will produce the mechanical properties described previously.

3. Cutting the threads will leave small residual stresses of little significance. The metal surface exposed after machining is more subject to corrosion and will be protected as described hereafter in Item e, Corrosion Protection.

6) Washer-Nut: Material purchased as bar with center hole trepanned to approximate (smaller) inside diameter, saw-cut to length sufficient for facing both faces.

a) Operations

1. Piece is finished to size.

2. Part is heat-treated per Specification MIL-H 6875 b to a hardness of 40 to 44 measured on the Rockwell C scale.

3. Threads are then machine-cut and checked for tolerances.

b) Effects

1. Any residual stress resulting from machining prior to heat-treating is essentially cancelled during heat-treating.

2. During heat treating, parts are placed in a controlled atmosphere oven and brought up to approximately 1550°F, which is above the upper critical range (austenitic range) then quenched in oil to approximately 800°F and air cooled to ambient. This will produce the mechanical properties described previously.

3. Cutting the threads will leave small residual stresses of little significance. The metal surface exposed after machining is more subject to corrosion and will be protected as described hereafter in Item e, Corrosion Protection.

e. Corrosion Protection

The corrosion protection system is designed to protect the tendon (and its components) during the various stages of tendon fabrication, installation, and throughout the life of the plant. Refer to Section 5.2.2.3.7 for historical description of tendon grease.

5.2.2.3.4 <u>Prestressing Sequence</u>

The dome and wall tendons were installed and tensioned in a prescribed sequence so as to minimize stress concentration in the shell. The stressing operation for the vertical tendons started at four positions approximately equally spaced along the circumference of the cylinder and proceeded in a prescribed sequence.

The hoop tendons were stressed in sets of three tendons comprising a complete 360 degree band at six positions and proceeded in the prescribed sequence.

The stress-strain curves for the production lots used were reviewed by the Engineer along with the final gauge reading and elongation for each stressed tendon. If the loss of prestress force due to failure of wires or buttonheads exceeded 1/2 of 1 percent, the engineer was advised.

Force and strain measurements were made by measurement of elongation of the prestressing steel after taking up initial slack and comparing it with the force indicated by the jack-dynamometer or pressure gauge. The gauge indicates the pressure in the jack within plus or minus 2 percent. Force-jack pressure gauge or dynamometer combinations were calibrated against known precise standards just before application of prestressing forces and calibrations were so certified prior to use.

Pressure gauges and jacks so calibrated were always used together. During stressing, records were made of elongations as well as pressures obtained. Jack-dynamometers or gauge combinations were checked against elongation of tendons, and the cause of any discrepancy exceeding plus or minus 5 percent of that predicted by calculations (using average load elongation curves) were corrected and, if caused by differences in load-elongation from averages, were so documented. Calibration of the jack-dynamometer or pressure gauge combinations were maintained accurate within the above limits.

5.2.2.3.5 Prestressing Arrangement

The configuration of the tendons in the dome (Figure 5.2-1) is based on a three way tendon system consisting of three groups of tendons oriented at 120 degrees with respect to each other. A large concrete ring girder is provided at the intersection of the dome and wall. The cylindrical wall is prestressed with a system of vertical and horizontal tendons. The horizontal system consists of a series of rings. Each ring is made up of three tendons, each subtending an angle of 120 degrees. Six buttresses are used as anchorages with the tendons staggered so that adjacent rings do not have tendons anchored at the same buttress. Each hoop and dome tendon is stressed from both ends so as to reduce the friction losses. The vertical system consists of vertical tendons anchored in the foundation mat and ring girder. For typical tendon arrangement, see Figures 5.2-1 and 5.2-3.

5.2.2.3.6 Prestress Losses

In accordance with the ACI 318-63 (Reference 5), the design makes allowance for the following prestress losses:

a. Seating and anchorage

b. Elastic shortening of concrete

c. Creep of concrete

d. Shrinkage of concrete

e. Relaxation of steel stress

f. Frictional loss due to intended or unintended curvature in the tendons

The above losses have been predicted within safe limits. The environment of the prestress system and concrete is not appreciably different in this case from that found in numerous bridge and building applications.

5.2.2.3.7 Historical - Tendon Grease

The machined tendon anchorage components are thoroughly cleaned and completely immersed in Visconorust 1601 Amber. Immersion results in complete coverage of the components, including the interior surfaces of all holes. Visconorust 1601 Amber is a solvent cut back of high melting point soaps and oxidized petrolatums with the addition of synthetic wetting agents and rust preventive additives. The 1601 Amber provides a dry, tack free coating satisfactory for temporary protection. Each tendon wire is also completely coated with 1601 Amber during tendon fabrication so that the finished tendon will be protected from corrosion prior to bulk filling the conduit.

Following the insertion of the tendons in the conduit, the conduit is filled with Visconorust 2090 P Casing Filler providing a bulk filling of the void in the conduit. The Casing Filler is pumped hot into the conduit. The pumping temperature of 120°F to 150°F is greater than the maximum the conduits will experience at any time during the life of the plant, including accident conditions. Bulk filling at these elevated temperatures effectively provides for any expansion of the Casing Filler. The Casing Filler is pumped into the vertical conduits from the bottom. A flow through system of filling is used for the horizontal and dome tendons where the filler is either recirculated into another conduit or bled off until the conduit voids have been completely filled. The bulk casing filler displaces both air and moisture as it is pumped through the conduit and effectively seals off the tendon bundle and anchorage from the elements necessary for corrosion.

Visconorust 2090 P Casing Filler has been developed for use primarily in the nuclear power plant construction industry. The melting point range is 125°F to 135°F. It is composed of stable petroleum base micro-crystalline waxes (petrolatums) and long chain saturated paraffinic mineral oils that are resistant to oxidation and physical and chemical degradations at expected operating temperatures. Compounded with these barrier agents are polar active additives, one of which is lanolin, that enhance the wetting of the metal surfaces with petrolatum. Petroleum sulfonates are used as surface active agents and water displacing agents.

Visconorust 2090 P Casing Filler is a thixotropic product containing no solvents. It has undergone gamma radiation to 5×10^8 rads, and has shown no change in the physical or chemical specifications.

Visconorust 2090P-2 or Visconorust 2090-4 has been used as a substitute casing filler grease at TMI-1 since the production of the original casing filler grease 2090P was discontinued in 1974. Visconorust 2090P-4 is currently used as bulk filler grease at TMI-1.

The current requirement for corrosion protection medium or casing filler grease at TMI-1 is that the samples from each end of each tendon examined shall meet the requirements of Subsection IWL of Section XI of the ASME Boiler and Pressure Vessel Code, as incorporated by reference into 10 CFR 50.55a.

TMI Technical Specification 4.4.2.1 requires that the surveillance program for structural integrity and corrosion protection conforms to the requirements of Subsection IWL of Section XI of the ASME Boiler and Pressure Vessel Code, as incorporated by reference into 10 CFR 50.55a.

TABLE IWL-2525-1 of IWL-2525, ASME B & P V Code Section XI and Regulatory Guide 1.35 require same acceptance limits for water soluble chlorides, nitrates, sulfides and reserve alkalinity in filler grease samples analyzed according to same national standards for analyzing water soluble chlorides, nitrates, sulfides and reserve alkalinity.

Corrosion medium or casing filler samples are obtained during inspection intervals and are analyzed for reserve alkalinity, water content, and concentration of water soluble chlorides, nitrates and sulfides. Analyses are performed in accordance with the procedures specified in Table IWL-2525-1 of ASME B & P V Code Section XI.

The majority of grease currently in the tendons at TMI-1 is type 2090P. The chemical analyses of grease tested over the past tendon periodical surveillances have all resulted in the grease meeting acceptance criteria of Regulatory Guide 1.35 or ASME Section XI, Subsection IWL, regarding performance requirements for the grease. All chemical analyses of grease tested over the past surveillances indicate that Visconorust 2090P-2 or Visconorust 2090P-4 is adequate as substitutes for Visconorust 2090P.

5.2.2.4 Liner Plate and Penetrations

The Reactor Building is a steel-lined concrete shell in the form of a vertical right cylinder with an ellipsoidal dome and flat base. The concrete thickness is 3 ft 6 inches for the cylindrical wall and 3 ft for the dome. The liner has been designed as a free standing vessel during erection. Temporary erection loads, including loads resulting from using the liner as a form for concrete work, have been considered in the design. The liner is an element of the composite steel and concrete shell. The degree of leaktightness ensures a containment leak rate of no greater than 0.1 percent by weight of contained atmosphere in 24 hours at 50.6 psig.

The general construction sequence of the liner is described in Section 5.2.

The steel plate for the liner main shell, including the dome, cylindrical wall, and base, conforms to:

a. ASTM A 283-67, Grade C, (Reference 11). Plate thickness 1/4 inch and 3/8 inch.

b. ASTM A 516-67, Grade 55, (Reference 12). Plate thickness 3/4 inch.

c. ASTM A 36-67, (Reference 13). Liner attachments, anchors for polar crane support, and rolled sections including test channels and stiffeners.

The materials for penetration sleeves, including the personnel and equipment access hatches as well as the mechanical and electrical penetrations, conform to the requirements of the ASME Nuclear Vessels Code for Class "B" Vessels. Materials for penetrations exhibit impact properties as required for Class "B" Vessels. The equipment hatches and personnel access lock material is SA-516 Grade 60 modified to SA-300. Other penetrations materials are SA-516

Grade 60 modified to SA-300 for insert plates and A333 Grade 1 and SA-516 Grade 60 modified to SA-300 for sleeves, or equal.

The header plate for electrical penetration 201E is AS240 Grade 304.

To support the 2009 SGR Project, a Containment Opening was made in the Reactor Building concrete wall and liner plate at the 290° azimuth, and directly above the existing Equipment Hatch to gain building access for rigging and handling of the steam generators. A section of the liner plate approximately 23'-6" wide and 22'-0" tall, centered about the 290° azimuth and elevation 378'. The removed section of 3/8" liner plate was restored via full penetration welds with backing. The vacuum box test and 50.6 psig integrated leak rate test confirmed leak tightness.

5.2.2.4.1 Codes

The Reactor Building liner and penetrations conform in all respects to the applicable sections of ASA N 6.2-1965 (Reference 14). The personnel locks and that portion of the equipment access door extending beyond the reinforced concrete shell conform to the requirements of the Nuclear Vessels Code for Class "B" Vessels. The selection of materials for penetration sleeves and insert plates, equipment door, and personnel lock considers a minimum service metal temperature of -5°F.

Principal load-carrying components (except for the prestressing system) of ferritic materials for the containment vessel exposed to the external environment were selected and tested to confirm that their nil ductility transition temperature is at least 30°F below the minimum service metal temperature. The ferritic materials exposed to the external environment that must meet this requirement consist of the penetrations and large openings (equipment access hatch and personnel locks) for which materials have been selected to conform with the ASME Boiler and Pressure Vessel Code, Section III, for Class "B" Vessels.

To support the 2009 SGR Project, a Containment Opening was made in the Reactor Building concrete wall and liner plate at the 290° azimuth, and directly above the existing Equipment Hatch to gain building access for rigging and handling of the steam generators. Codes utilized for removal and restoration of the liner plate in creating and restoring this temporary opening are described in UFSAR Section 5.2.3.1.

5.2.2.4.2 Design Requirements

The liner has been designed to support dead load and wind and snow loads during the erection period. Normal operating and accident load requirements are described in Subsections 5.2.1 and 5.2.3.

5.2.2.4.3 Technical Parameters

The technical conditions applicable to the complete containment vessel are as follows:

Vessel inside diameter, ft	130
Cylinder height, ft	157
Ellipsoidal Dome,	
°Short radius, ft	20.5

°Long radius, ft	110
Design pressure, psig	55
Design temperature,°F	281
Operating pressure, range, psig	+1 to -1
Operating temperature, Range,	
° Inside vessel,°F	+90 to 110
° Outside vessel,°F	-5 to 100
Design vacuum (Internal), psig	2.5
Liner leak rate at 50.6 psig/hours	0.1%/24

One equipment access hatch is provided with a minimum inside diameter of 22 ft 4 inches.

Two personnel air locks are provided with an inside diameter of 10 ft. One of these personnel hatches is mounted in the equipment access hatch. The air lock in the equipment access hatch measures 15 ft from door to door. The personnel access air lock measures 9 ft from door to door. Doors are pressure-seated type for pressure within containment, 3 ft 6 inches by 6 ft 8 inches, and provisions are made to test between doors at 63.3 psig.

Mechanical, electrical, and fuel transfer penetrations are provided.

Erection tolerances:

1) Overall out of roundness ±3 inches.

2) Deviation from round in 10 ft is 1 1/2 inches except at seams.

3) Overall deviation from a plumb line ±3 inches.

4) Deviation from line between tangent points - at cylinder to dome transition and base to cylinder transition - ±3 inches.

5) Shell plate edges shall butt for a minimum of 75 percent of wall thickness.

Tolerances for penetrations are as specified in Subsection 5.2.2.4.8. These tolerances are maintained in the final erected position.

5.2.2.4.4 Plate Thickness

The steel plate for the main shell, including the cylindrical walls and the dome but excluding specially reinforced areas, is 3/8 inch thick.

The steel plate for the containment base liner, including the pits and sumps, has a minimum thickness of 1/4 inch.

5.2.2.4.5 Opening Reinforcement

The liner is reinforced about openings in accordance with the ASME Unfired Pressure Vessels Code (i.e., by replacing cut out area of 3/8 inch liner plate).

5.2.2.4.6 Test Channels

Steel channels have been provided along weld seams that were inaccessible. The channels have been segmented so as to ensure no length of weld covered by any one channel segment exceeds the greatest dimension of one plate. One plug fitting has been provided in each channel segment and extends through any covering material, including concrete. The fittings on the base have been protected by sleeves at the base weld to ensure no weld failure during placing of concrete. Test channels have been located on the face of the liner inside containment. Flanged heads have also been installed to cover penetration sleeve to liner plate welds.

5.2.2.4.7 Material Damages

Materials have been carefully handled so that members or parts which were damaged since fabrication have been repaired without heating by methods which did not produce fracture or other injury. Any members which were so damaged that it was inadvisable to correct them in the field were replaced with new members. Hammering which would injure or distort the members was not permitted.

5.2.2.4.8 Penetration And Openings

a. Personnel Access Locks

Two personnel access locks are provided, one of which penetrates the dished head of the equipment hatch. Each personnel hatch is a welded steel assembly with double doors equipped with double gaskets to provide an air space that can be pressurized to the Reactor Building design pressure for leak testing or fluid blocking.

b. Equipment Access Hatch

An equipment hatch with an inside diameter of 22 ft 4 inches has been provided to enable passage of large equipment and components into the Reactor Building during a plant shutdown. The items brought into the vessel include, in part, the reactor coolant pumps and motors, and reactor vessel O-rings.

A steel test channel was provided over the field weld of the penetration ring to the sleeve for replacement penetration 201E, to allow for pressure testing of the weld.

The following applies to both the equipment and personnel openings in the containment liner:

1) Flanged joints have been designed for the use of a double gasketed seal. This seal between the gaskets has been designed to a pressure equal to the design pressure.

2) The material used in the construction of the openings is compatible with the liner material metallurgical characteristics.

3) The personnel opening doors are interlocked to prevent both doors being opened simultaneously. Interlocks are so connected that one door must be completely closed before the opposite door can be opened.

4) For the personnel openings, remote indication is provided in the Control Room to indicate whenever the interlock mechanism is defeated and the doors are not locked closed.

5) Personnel locks have an interior lighting system which is capable of operating from the emergency power supply.

6) Personnel locks have an emergency communication system.

7) Provisions on personnel locks have been provided to permit bypassing the door interlocking system to allow doors to be left open when plant is shut down.

8) The floor system of the personnel locks has been designed so that it can be easily removed.

9) The personnel locks have been designed, fabricated, tested, and stamped in accordance with the ASME Nuclear Vessels Code as Class "B" Vessels.

10) Personnel locks have been designed to be capable of withstanding a test of 63.3 psig in the interspace between doors.

11) Personnel lock hinges are capable of a three dimensional adjustment to assist proper seating. Hinges are capable of independent adjustment.

12) Seals, gaskets, O-rings, or other seating materials are suitable to withstand design temperature conditions.

13) Personnel lock equalizing valves are of the quick-acting type.

c. Mechanical Penetrations

The following applies to the fabrication and testing of penetrations:

1) Details for mechanical penetrations are as shown on Figure 5.2-2.

2) For hot pipelines (operating temperature equal to or greater than 150°F) an expansion joint was provided between the pipe and sleeve at the second barrier to accommodate the calculated axial and lateral pipe motions.

3) The penetration material is compatible with liner materials.

4) "Unibestos" material was used where thermal insulation was required.

5) Pipe insertions are of greater wall thickness than wall thickness of the pipe elsewhere in the system.

6) Bellows, expansion joints, gaskets, canopies, protectors, or other flexible members were designed for a minimum of 250 cycles for the movement associated with each penetration.

7) The locations of penetrations with regard to azimuth location are within 1/2 inch measured on the circular section. The horizontal and vertical dimensions associated with the radial dimensions are 1/2 inch for the pipelines.

8) Penetrations were installed in the respective plate sections prior to testing.

9) Mechanical penetrations have double barriers designed for pressurization with air to 63.3 psig for leak testing and 60 psig during accident conditions.

10) Mechanical spares have both ends of the sleeves capped. Each sleeve is equipped with a 3/4 inch test connection outside containment. These sleeves are 12 inch, Schedule 80 pipe.

11) The containment penetration boundary welds are Category C, corner groove welds per Paragraph N-461, ASME Section III, 1968 Edition. Corner groove welds were not considered radiographable. Weld acceptance was by surface examination in accordance with Subparagraph N-1350(b), ASME Section III.

d. Electrical Penetrations

Electrical penetrations have been designed as follows:

1) Penetration cartridges have been installed in the penetration sleeves.

2) The penetration sleeves are 12 inch, Schedule 80 carbon steel pipe. Penetration sleeves were shop-welded to the reinforcement plate.

3) The weight of the liner cartridges is not more than 500 lb.

e. Leak Testing

The penetrations have been tested as follows:

1) A proof test has been applied to each penetration by pressurizing the penetration annulus to 63.3 psig. The pressure was reduced to 55 psig and held at this pressure to soapbubble test welds and mated surfaces. Leaks were repaired and retested. This procedure was followed until no leaks existed.

2) Tests were conducted in accordance with ANS-7.60, Appendix A (Reference 15).

3) After installation and prior to the initial integrated leakage rate test of the Reactor Building, the spaces between the dual resilient seals on each of the doors of the personnel access air locks were pressurized with air at 50.6 psig and both sides of the seal checked with a sonic leak detector and/or soapbubble. Subsequently, with the inner and outer doors both closed, the space between both doors was pressurized to 50.6 psig with air and sonic leak detection and/or soapbubble applied to the seals in the nonpressurized areas (i.e., inside and outside the Reactor Building).

4) Penetration 201E was replaced by a single header plate design, utilizing mechanically swaged ferrule seals with a qualified polysulfone resilient seal. The 24 feedthrough ports and the field weld between the sleeve extension ring and the penetration sleeve, which has an applied leak test channel, was tested in accordance with paragraph 1) above.

The subsequent surveillance procedure that will be used for door seals will pressurize the spaces between the dual seals to 10 psig and measure the flow rate to maintain this pressure. The double gasketed seal on the equipment hatch will be periodically tested at 50.6 psig by use of a flow meter, in accordance with Reference 16.

f. Penetration Appurtenances

The following applies to the penetrations:

1) Reinforcing was designed to support the penetration in the liner for shop testing, shipping, and field erection.

2) Bellows were fabricated from stainless steel having metallurgical characteristics compatible with mechanical design requirements. Bellows are suitably protected against field damage and are a part of the permanent installation.

g. Special Penetrations

Two penetrations which required special attention are the containment supply and exhaust purge ducts. The following additional requirements were imposed on these penetrations:

1) Formed heads were supplied with the penetration for use during liner tests.

2) Each penetration is provided with two test connections, one 3/4 inch pipe size to test the annulus space and one 2 inch pipe size to test the purge duct.

5.2.2.5 Corrosion Protection

The containment structure is protected against external corrosive influences by the following means:

a. Deleted.

b. The Reactor Building is surrounded by a circular retaining wall extending from top of foundation to elevation 304 ft. The retaining wall provides access space for stressing and inspecting tendons. At the base of the retaining wall between the wall and the Reactor Building, a drainage system is provided to prevent significant accumulation of water in the access space.

Therefore, waterproofing of the lower portion of the cylinder was not required. There was no waterproofing or membrane used on the base slab. The retaining wall precludes contact of groundwater with the shell.

c. A cover of concrete was provided in excess of that for normal construction as exemplified by code requirements. The concrete cover provisions for reinforcing steel and prestressing is as shown in Table 5.2-1. The minimum cover specified by ACI-318 (Reference 5) is also tabulated for comparison.

d. A galvanized steel conduit for tendons was used with the added precaution of a thicker-walled, rigid steel conduit in the base mat and extending immediately above into the cylinder.

e. An inboard-oriented haunch which results in only nominal tensile stresses of the outer fibers; these stresses are within the capacity of concrete in tension.

The exposed surface of the liner has been given a protective coating. The tendons and included anchorages are coated with a temporary corrosion inhibitor for protection prior to bulk-filling the conduct. The casing filler provides an environment for the tendon where corrosion is highly unlikely. The retaining wall, and drainage system, as described previously, eliminates groundwater corrosion problems on the liner.

The tendons were inserted in galvanized steel conduit embedded in the concrete which provides an additional water barrier, as well as an electrical shield against stray currents. The inner surface of the steel conduit, as well as the tendons including anchorages, is protected with a heavy wax base corrosion inhibitor. The end anchorages are covered with a metal container. The retaining wall and drainage system around the Reactor Building provides excellent protection for the liner and tendons against corrosion, and therefore, no cathodic protection system was provided. Metallic components including the liner plate and tendon conduit are electrically connected to prevent stray current corrosion. The tendons are enclosed in a metallic tube so as to isolate them from outside electrical influences. The 2009 SGR Project replaced the existing tendon conduits within the opening with identical 4-3/4" OD galvanized tubing with a 0.065" wall thickness. Similar to the existing tendon conduit configuration, a stainless steel medium-pressure 4" diameter pipe repair clamp 7-1/2 " long with 3 bolts was used as a tendon coupler to attach the existing and new galvanized tendon conduits.

Permanent reference electrode stations have been installed to facilitate measurements of structure potentials. These stations consist of a plastic pipe extending from the ground surface to the point at which the structure potential measurement was required. The plastic pipe functions as a salt bridge. Standard reference half cells placed into the pipe, down to the groundwater level, can be used to make structure potential measurements.

5.2.3 STRUCTURAL DESIGN CRITERIA

The prestressed concrete Reactor Building has been designed to have a low strain elastic response to all design loads, thereby ensuring that the integrity of the vapor barrier is never breached. The design has been based upon various combinations of factored loads by which loads are increased to approach the limit of an elastic response. These factors were developed in a similar manner to the Ultimate Strength Design provisions of ACI 318-63 (Reference 5) where factors are applied for those factors outlined in the ACI 318-63 (Reference 5). In the case of this design wherein a more exact analysis is performed than contemplated by ACI 318 (Reference 5) the load factors primarily provide for a safety margin on the applied loads. The

Reactor Building has been analyzed to ensure proper performance of all components including the liner, concrete shell, and reinforcement under the following loading conditions:

a. During construction but prior to prestressing

b. During prestressing

c. At normal operating conditions

d. At test conditions

e. At factored loads

In addition, the Reactor Building has been evaluated for aircraft impact and other missiles (See Appendix 5A). Although the probability of an aircraft incident is extremely remote, the purpose of this study is to assure that hypothetical objects are prevented from penetrating the external reactor containment structure or causing its integrity to be violated.

5.2.3.1 Codes

The Reactor Building has been designed under the following codes:

a. Regulations for Protection From Fire and Panic - Commonwealth of Pennsylvania (Reference 17)

b. Building Code Requirements for Reinforced Concrete, ACI 318-63 (Reference 5)

c. Specifications for Structural Concrete for Buildings, ACI 301-66 (Reference 2) except as modified in the design and quality control of this building

d. AISC Manual of Steel Construction (Reference 18)

e. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessel; Section VIII, Unfired Pressure Vessels; Section IX, Welding Qualifications (applicable portions)

AEC Publication TID-7024 (Reference 19) as amplified herein has been used as the basic design for seismic analysis.

Structural design for normal operating conditions has been governed by the applicable design codes. The design for the LOCA and maximum seismic condition ensures no loss of functions related to public safety.

To support the 2009 SGR Project, a Containment Opening was made in the Reactor Building concrete wall and liner plate at the 290^{o} azimuth, and directly above the existing Equipment Hatch to gain building access for rigging and handling of the steam generators. Creation and restoration of the Containment Opening employed the following codes and standards:

a. AISC Manual of Steel Construction-1989, 9th Edition (ASD).

b. ANSI/AWS D1.1-2000, Structural Welding Code - Steel.

c. ANSI/AWS D1.4-1998, Structural Welding Code - Reinforcing Steel.

d. ASME Boiler and Pressure Vessel Code, Section III, Division 2, 2001 Edition through 2003 Addenda (NOTE 1).

e. ASME Boiler and Pressure Vessel Code, Section VIII, 2001 Edition through 2002 Addenda.

f. ASME Boiler and Pressure Vessel Code, Section XI, 1995 Edition through 1996 Addenda (NOTE 1).

g. ASME Boiler and Pressure Vessel Code, Section XI, 1992 Edition/Addenda for (NOTE 1).

h. Uniform Building Code (UBC), 1997.

i. ACI 318-02, Building Code Requirements for Structural Concrete and Commentary.

j. AISC Code of Standard Practice, 1986 Edition.

k. ANSI/N45.2.2, Packaging, Shipping, etc. of items for Nuclear Power Plants, 1978 Edition.

l. ANSI N45.2.11, Quality Assurance Requirements, 1974 Edition.

m. ASME NQA-1, Quality Assurance Program Requirements for Nuclear Facilities, 1994 Edition.

n. ASME Code Case N-649, Alternative Requirements for IWE-5240 Visual Examination, Section XI, Division 1.

o. ACI 349-97, Code Requirements for Nuclear Safety Related Concrete Structures.

p. ACI 301-99, Specifications for Structural Concrete.

NOTE 1: Code reconciliation to the original Construction Code is performed by the design for creating and restoring the Containment Opening, as applicable.

5.2.3.2 <u>Loading Conditions</u>

The design has been based upon limiting load factors which were used as the ratio by which accident, earthquake, and wind loads have been multiplied for design purposes to ensure that the load deformation behavior of the structure is one of elastic, low strain response. The loads utilized to determine the required limiting capacity of any structural element on the Reactor Building were computed based on the load combinations shown below and those in Table 5.2-2.

a. $C = (1.0 \pm 0.05) D + 1.5 P + 1.0 T$

b. $C = (1.0 \pm 0.05) D + 1.25 P + 1.0 T' + 1.25 E$

c. $C = (1.0 \pm 0.05) D + 1.0 P + 1.0 \underline{T} + 1.0 E'$

d. $C = (1.0 \pm 0.05) D + 1.0 W_t + 1.0 P_t$

Symbols used in the above equations are defined as follows:

C: Required load capacity of section

D: Dead load of structure

P: Accident pressure load

$\underline{T}$: Thermal loads based upon temperature transient associated with 1.5 times accident pressure

T': Thermal loads based upon temperature transient associated with 1.25 accident pressure

T: Thermal loads based upon temperature transient associated with accident pressure

E: Design earthquake (OBE) (0.06 g ground motion)

E': Maximum Hypothetical Earthquake (SSE) (0.12 g ground motion)

W_t: Wind loads based on a 390 mph (300 mph times a gust factor of 1.3) tornado. See Subsection 5.2.1.2.6.

P_t: Pressure load based on an external pressure drop of 3 psig between inside and outside of the Reactor Building

Should the required resisting capacity on any structural component as a result of wind load on any portion of the structure have exceeded that resulting from the design earthquake, the wind load "W" would have been used in lieu of "E" in the second equation. In fact the wind load controlled no aspect of the design. All structural components have been designed to have a capacity, as defined hereafter, required by the most severe loading combination.

The load combinations considered in the analysis are given in Table 5.2-2. The stress resultants, stress couples, and normal shears due to the 34 load combinations described in Table 5.2-2 were calculated using a computer program for 16 typical sections on the Reactor Building.

Openings required for equipment and/or personnel access have been reinforced to withstand computed stress concentrations as described in Appendix 5C. The design of the opening reinforcement ensures approximate strain compatibility within the shell. See loading combination at normal operating conditions, and at factored loads, Subsections 5.2.3.2.3 and 5.2.3.2.5.

Penetrations for process piping, ventilation ducts, fuel transfer, instrumentation lines, and electrical cables have been designed to withstand the following loads:

a. Incident pressure and temperature including response of the Reactor Building shell.

b. Pipe reactions based on thermal flexibility and seismic loads.

c. Expansion of containment shell under test conditions.

d. Pipe thrust loads to ensure the vapor barrier is not breeched due to the rupture of any piping system.

The openings in the concrete shell have been reinforced with mild steel reinforcement. Locations of these penetrations are such as to minimize tendon deflection and related stress concentrations.

Reinforcement has been provided as required for radial, hoop, and meridional moments and shears. Minimum reinforcement at discontinuities complies with the requirements of Chapter 26 of ACI 318-63 (Reference 5) for shear reinforcement of webs.

Pipe penetrations are normally anchored at the Reactor Building shell. Penetrations were made of material which exhibits by test a transition temperature at least 30°F below the minimum service metal temperature.

Typical piping, duct, and electrical penetrations are shown on Drawings 521034, 521044 and E-222-54. Penetrations are of the double barrier type. Where temperature changes require, the second barrier of piping penetrations includes expansion bellows. Penetrations have been designed to provide a captive air space that can be pressurized to the Reactor Building design pressure for leak testing and accident conditions. Piping systems that may be open to Reactor Building atmosphere have been designed for adding a fluid block. (The Fluid Block System has been deactivated and removed. Refer to 5.3.5.2).

The stress levels in the dome, shell, cylinder-base junction, and in the ring girder region in the containment structure which result from the chosen loading combinations are tabulated in Table 5.2-3.

The stress levels at the base of the wall are based upon a straight transition of the liner. A detailed analysis of the liner at the base to cylinder transition is more fully described in loading combination at factored loads, subsection 5.2.3.2.5.

The stresses associated with operating and accident temperatures are based on the temperature gradients as shown on Figures 5.2-17 and 5.2-19.

The stresses at initial operation (assumed at maximum winter temperature gradient; see Figure 5.2-18) are the maximum compressive stresses induced in the concrete. These stresses are temporary in that creep due to operating temperature has not occurred.

The asterisk indicates stresses that occur at discontinuities and exceed 0.45 (f'_c) but less than the 0.60 (f'_c) specified by ACI 318, Chapter 26 (Reference 5) for temporary loads. Concrete sections that have temporary stresses that exceed 0.45 (f'_c) in compression have been designed with reinforcement to limit compressive strains.

The use of the working stresses and ultimate stresses as presented in the ACI 318-63 (Reference 5) in structures where two and three dimensional stress fields are predominant is discussed in Reference 20. The relevant information can be found in Section 5.3.3, "Allowable Stresses and Yield Strengths."

5.2.3.2.1 <u>During Construction But Prior to Prestressing</u>

The Reactor Building has been designed as a conventional reinforced concrete structure subjected to dead, live, wind, and construction loads with allowable stresses in accordance with the limits established by ACI 318-63 (Reference 5).

5.2.3.2.2 <u>During Prestressing</u>

The Reactor Building has been designed for prestress loads and has been checked to ensure that the concrete stress does not exceed 0.6 f'_c at initial transfer. Stresses due to shrinkage, creep, and elastic shortening of concrete have been taken into account considering the time dependent phenomena where appropriate. For analysis and design of anchor zones for the prestressed tendons see Section 5.2.2.3.2 and Appendix 5D.

5.2.3.2.3 <u>At Normal Operating Conditions</u>

The loads due to normal operating conditions are:

a) External pressure of 2.5 psig

b) Operating temperature transients

c) Dead load

d) Live load

e) Prestress load

f) Seismic load

g) Snow and wind load

The stresses in the concrete and reinforcing steel resulting from these loads are in accordance with ACI 318-63, Chapter 26 (Reference 5). The stresses and strains are such that the integrity of the liner has been maintained.

Containment vacuum can only occur during operating condition and therefore could only influence liner buckling during that time. The deflection associated with a vacuum load of 2.5 psi and an axial stress of about 24 ksi is approximately 0.010 inches. The height of the middle ordinate due to the curvature of the plate is 0.054 inches. Considering this deflection and tolerances in construction the vacuum will not influence the buckling of the liner. The liner anchors and welds are designed to withstand a vacuum load of 2.5 psi and are also designed such that the weld or anchor will fail before the liner is breached.

Scaled plots of stress resultants, stress couples, shear, and deflection for Normal Operating Condition (Load Combination 17) are shown in Figure 5.2-27.

The above analysis was based on design operating temperatures of 110°F inside the Reactor Building. Based on actual operating experience, the above analysis has been reperformed using a containment temperature of 130°F for elevations above 320 ft and 120°F for elevations below 320 ft. This analysis was performed in accordance with Working Stress Design provisions of ACI 318-63 (Reference 5). Acceptance criteria were based on the allowable stresses specified in Section III, Division 2, issued January 1, 1975 of the ASME Boiler and Pressure Vessel Code. The stresses resulting from the increased operating temperature were found not to be detrimental to the performance of the structure. In addition, the most critical section under the loading conditions specified above was checked using the Ultimate Strength Design provisions of ACI 318-63 (Reference 5). A safety factor of 1.63 was found using this design method and confirmed the adequacy of the structure.

5.2.3.2.4 Test Loads

The Reactor Building has been designed to function under the following loads at test conditions:

a) Internal pressure of 1.15 times design pressure = 63.3 psig.

b) Dead load

c) Live load

d) Prestress load

e) Temperature transients at test conditions

The design is in accordance with ACI 318-63 Part IV-B and Chapter 26 (Reference 5).

The vessel has been adequately instrumented to verify during the pressure test that the structural response of the principal strength elements is consistent with the design as described in Section 5.7.1.

Scaled plots of stress resultants, stress couples, shear, and deflection at Test Load (Load Combination 25) are shown in Figure 5.2-28.

5.2.3.2.5 At Factored Loads

The building has been checked for the factored loads and load combination given in Section 5.2.3.2, and compared with the yield strength of the structure. The load capacity of the structure is defined for our design, as the upper limit of elastic behavior of the effective load carrying structural materials.

For steels (both tendon wires and mild steel reinforcement), this limit is considered to be guaranteed minimum yield strength. For concrete, the yield strength is limited by the ultimate values of shear (as a measure of diagonal tension) and bond per ACI 318-63 (Reference 5) and the 28 day ultimate compressive strength for flexure (f'_c). A further definition of "load capacity" is that deformation of the structure which will not cause strain in the steel liner plate to exceed 0.005 inches/inch, nor average tensile strains to exceed that corresponding to the minimum yield stress.

The load capacity of load-carrying structural elements has been reduced by a capacity reduction factor (ϕ) as stated in the basic structural design criteria. This factor provides for "the possibility that small adverse variations in material strength, workmanship, dimensions control, and degree of supervision while individually within required tolerances and the limits of good practice, occasionally may combine to result in under-capacity" (refer to ACI 318-63, p. 66 - footnote) (Reference 5).

The load factor of 1.0 applied to equations c and d of Subsection 5.2.3.2 were used because the loadings caused by a safe shutdown earthquake with a 0.12g ground motion or a 390 mph tornado are considered extreme environmental loadings which have a remote possibility of occurrence.

Scaled plots of stress resultants, stress couples, shear, and deflection for three Factored Loads (Load Combination 26, 27, 31 which correspond to equations a, b, and c of Section 5.2.3.2) are shown on Figures 5.2-29, 5.2-30, and 5.2-31.

a. Principal Tensile Stresses - Flexural Stresses

The allowable tensile capacity of concrete for membrane stresses (i.e., excluding all flexural and thermal stresses) due to the factored loads is $3\sqrt{f'_c}$. The allowable tensile capacity of concrete for maximum fiber stresses due to the factored loads including the thermal load plus other secondary effects is $6\sqrt{f'_c}$. Where tensile fiber stresses exceed the allowable, mild steel reinforcement was added on the basis of cracked section design. The addition of mild steel reinforcement and the increase in steel stress due to temperature effects was determined in a manner similar to that contained in ACI 505-54 (Reference 21). The procedure given by ACI 505-54 (Reference 21) was modified as described below, in order to include the effects of both the applied axial load and moment.

1) Analyze the section for P plus M based on a cracked section and a linear strain diagram.

2) Determine the location of the neutral axis, the slope (S_l) of the strain diagram, and the resulting stresses.

3) Obtain the slope (S_t) of the strain diagram due to the applied temperature gradient based on an uncracked section.

4) Equate the slope (S_c) of the final strain diagram for P plus M plus temperature gradient to the sum of S_l plus S_t.

5) Knowing the slope of the final strain diagram and equating the sum of the forces of the final stress diagram to the applied axial force, the location of the neutral axis for the combined loading of P plus M plus temperature gradient can be determined.

6) Calculate the resulting stresses.

The adequacy of the minimum steel on the exposed face of 0.15 percent of the cross-section area of the concrete was also verified using the above procedure.

Mild steel reinforcing was not provided on the inside (liner) face of the concrete in the typical wall and typical dome for the following reasons:

a) During the normal operating conditions and startup and shutdown conditions, the stress on the inside face of the concrete remains in compression.

b) During accident conditions, the tensile stresses at the inside face of the concrete are resisted by a reduction in the compressive stresses in the liner plate.

The cracking limit of the concrete in principal tension is governed by the allowable values of the shear as a measure of diagonal (principal) tension. The allowable shear values are as follows:

a) When membrane tension or when membrane compression less than 100 psi exists, the section was designed to the ultimate shear provisions of Chapter 17 of ACI 318-63 (Reference 5).

b) When membrane compression of greater than 100 psi exists, the principal membrane tension was limited by the ultimate shear provision of Chapter 26 of ACI 318-63 (Reference 5).

The concrete shell has been prestressed sufficiently to eliminate tensile stresses due to membrane forces from design loads. Membrane tension due to factored loads was permitted to the limits described above. On those elements carrying primarily tensile membrane forces, any secondary tensile stresses due to bending could cause partial cracking. Mild steel reinforcing was provided to control this cracking by limiting crack width, spacing, and depth. The load capacity that determined tensile membrane stresses was reduced by a capacity reduction factor of 0.95. The coefficient "ϕ" for flexure, shear, and compression will be in accordance with Section 1504 or ACI 318-63 (Reference 5).

Tensile stresses in the concrete resulting from diagonal tension have been permitted. The nominal shear stresses as a measure of this diagonal tension will be less than the maximum value stipulated in ACI 318 (Reference 5).

The concrete compressive stresses in the shell at design load, p = 55 psig, including operating and accident temperature, are below 0.85 (f'_c).

The preliminary design of the Reactor Building shell was done according to the ultimate strength design method. In the final design, concrete and steel stresses were verified using a triangular stress distribution.

b. Shear

The formation of diagonal cracks in one way slabs takes place in approximately the same manner as in beams. However, in two way slabs, the stresses in the third dimension influence the ability of the material to resist the stresses in the other two dimensions. Thus, the behavior of a slab can not be directly compared to the behavior of a beam.

It has been verified through experimental investigations of reinforced concrete slabs that under certain conditions the slab will behave similarly to a beam in shear (Figure 8-1, Equation 8-14 of Reference 22). As the ratio of column size to slab thickness, r/d, from the above report, approaches infinity the slab will have comparatively little slab action and will tend to behave like a wide shallow beam. Therefore, as r/d approaches infinity, the value of μ would approach the corresponding shear strength of a beam.

The circumferential stresses in the shell are normally uniform; that is, meridional shear is zero. Consequently, the radial displacement of the shell is uniform, with little slab action. Therefore, the ultimate shear strength for a beam was used as a measure of diagonal tension for the shell structure.

The ultimate shear values used in the design have been in accordance with Chapter 26, (Reference 5), except as noted below in Ultimate Strength-Load Factors.

The influence of in-plane shear stresses, resulting from wind and earthquake loads, on the allowable stresses and ultimate strength in compression, and in tension for cases where the direction of the mild steel reinforcing does not coincide with the direction of

the principal stress, is taken into account by increasing or decreasing the calculated compressive or tensile stress in the direction of the mild steel reinforcing by the calculated in-plane shear stress. The above procedure results in designing the section under consideration for the forces acting normal to the section (in the direction of the mild steel reinforcing), including the normal component of the principal force due to the in-plane shear stress.

c. Ultimate Strength - Load Factors

The load factors utilized in the criteria are based upon the load factor concept employed in Part IV-B, of Reference 5. The load factor of 1.0 ±0.05 applied to dead load represents the accuracy of dead load calculations considering the greater severity of reduced dead loads for tension members. The load factor applied to the pressure loads due to the Maximum Hypothetical Accident of 1.5 is consistent with that suggested by Reference 23 as the limit of low strain behavior on prestressed concrete pressure vessels for nuclear reactors. This factor is also consistent with the proposed set of "French Regulations Concerning Concrete Reactor Pressure Vessels" wherein it is stated that "The design pressure shall not exceed 2/5 of the pressure calculated to bring about destruction of the structure by rupture of the cables." The load factor considering a stress of 0.6 f_u at factored load would thereby equal 0.6 ÷ 2/5, or 1.5. The load factor is applied to the pressure load when the design earthquake load is consistent with that utilized in ACI 318 (Reference 5). The reduction in the load factor applied to the pressure load, when the design earthquake is considered, is also consistent with the reduction in ACI 318 (Reference 5).

The shear stress limits and shear reinforcing for radial shear used in the design has been in accordance with Chapter 26, (Reference 5), except as follows:

1) In equation 26-12 (Reference 5) the shear increment between flexural and diagonal tension cracking $(0.6b'd\sqrt{f_c'})$ was modified based upon the results of testing in accordance with Reference 25. The resulting equation is:

$$V_{ci} = K_{delta-v} b'd\sqrt{f_c'} + \frac{Mcr}{\frac{M}{v} - \frac{d}{2}} + V_d$$

Where:

$$K_{delta-v} = 1.75 - \frac{0.036}{np} + 4.0\ np$$

In accordance with ACI 318 (Reference 5) the factor $K_{delta-v}$ is not considered to be greater than 0.6.

2) Requirements for minimum shear reinforcement as called for in Equation (26-11) of ACI 318 (Reference 5) were provided only at discontinuities.

d. The Liner

The Reactor Building liner has been designed for the loads that are specified in Section 5.2.1.2, Design Loads, and combined as specified in Section 5.2.3.2, Loading Conditions.

The stress levels in the liner are as tabulated in Table 5.2-3. The liner has been designed so that the critical buckling stress is greater than the proportional limit of the steel. Present analysis indicates that the basic accident conditions produce a strain of approximately 0.0022 inches/inch in the liner.

e. Liner Anchor

The Reactor Building liner anchors are vertical angles, as shown on Figure 5.2-1, and are spaced horizontally at 18 inches center to center.

The liner has been analyzed as a flat plate. This assumption is conservative in that the liner will have to buckle against its own curvature. For analysis, it is assumed that the liner is fixed at the angles and that there will not be any differential radial movement of the boundaries.

The following analysis is based on interaction curves given in Reference 26.

The critical stress resultants N_1 and N_2 are the stresses induced in the plate (Figure 5.2-32) and are defined as:

$N_1 = K_5N_e$ where $K_5 = 6.97$

$N_2 = K_3N_e$ where $K_3 = 4.00$

and:

$$N_e = \frac{\pi^2 E^2 t}{12(1-\mu^2)} \times \frac{1}{b^2}$$

It is seen from the interaction curve (Reference 26) that for, "a" equal to infinity, the influence from N_1 can be neglected.
Therefore:

$N_{critical} = 47.0$ ksi

The liner anchors were designed and spaced so that the critical buckling stress is greater than the proportional limit of the liner.

The liner anchors are designed to resist the loads induced when a section of the liner between anchors exhibits greater stresses than the adjacent panel. These loads are as shown on Figure 5.2-33.

The liner anchors have been designed so that the welds connecting the anchors to the liner will fail before the liner is breeched. Where the anchor angles did not conform to the curvature of the plate, such that the specified weld could not be made, the angle was reshaped to conform to the configuration of the plate. The design of the welds between the liner and anchors was based on the minimum acceptable thickness and the minimum guaranteed ultimate strength of the liner.

f. Load Transfer Through the Liner

For transfer of load through the liner, the steam generator has been bolted down through the base liner, as shown on Drawing 521017. The controlling factors in doing so are based on the fact that the total net uplift and overturning forces were too large to utilize the slab above the liner to transfer the forces to the nearby walls.

g. Cylinder to Base Junction

The analysis of the knuckle plate at the cylinder to base transition (see Figure 5.2-35) has been carried out by the methods described in Subsection 5.2.4. Based upon the present analysis, the following loads, movements, and strains were applied to the knuckle:

1) A vertical strain of +0.0030 inch and a lateral movement of +0.00074 inch applied at the top of the knuckle due to the design accident conditions.

2) A vertical strain of -0.0036 inch and a lateral movement of -0.00603 inch applied at the top of the knuckle due to dead load and prestress load.

3) Internal pressures corresponding to the design accident condition.

The compressible material is such that the knuckle plate can deform and absorb the strains produced by the operating and design basis accident conditions.

Based upon a similar solution described in Reference 27, for more severe motion and pressure loading, it has been concluded that maximum tensile stresses will be less than yield stress.

h. General

Figures 6.6-2 through 6.6-7 and Figures 6.6-9 through 6.6-12 show the Reactor Building pressure as a function of time for a range of Large Break LOCA break sizes. This analysis has been performed using the digital computer code "CONTEMPT" (Reference 28). Results of this code were used in determining what maximum pressures result from the Large Break LOCA's and serve to describe what variations of pressure loads the liner is subjected to with time. The peak pressures are summarized in Table 6.6-1 where it is seen that the 7.0 ft^2 pump suction break results in the highest peak pressure. The pressure profile for the 7.0 ft^2 pump suction break is shown in Figure 6.6-3.

The procedure for the analysis of the liner and the concrete shell is provided in Section 5.2.4. The stress concentration in the liner around the remaining opening has been determined by the methods suggested in Reference 29. Reinforcing of the liner at openings has been in accordance with the ASME Boiler and Pressure Vessel Code, Section VIII unless analysis indicated greater reinforcement was required. Stress limits are in accordance with the ASME Nuclear Vessels Code. A more detailed description of the analytical method for penetrations is contained in Reference 27.

Piping penetrations have been designed to ensure that the liner was not breeched due to the rupture of any process pipe. The load imposed on the containment shell was based upon the full plastic moment capability of the pipe with the moment calculated on the basis of the ultimate strength of the pipe. In addition to the foregoing, the penetrations have been designed for those loadings detailed in Items a through c of Section 5.2.3.2. Therefore the piping penetration and its anchorage into the liner is designed as a stronger element than the piping system. Refer to Subsection 5.2.4.1 for the method of analysis used.

5.2.3.2.6 Missiles

Missile protection for the Reactor Building liner:

a) The building and liner are protected from loss of function due to damage from such missiles as might be generated in a loss of coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe.

b) The engineered safeguard components required to maintain containment integrity and to meet the site criteria of 10CFR100 are protected against loss of function due to damage by the missiles defined below.

During the detailed plant design, the missile protection necessary to meet the above criteria has been developed and implemented using the following considerations:

a) The reactor and reactor coolant system are each surrounded by reinforced concrete and steel structures designed to withstand forces associated with double-ended rupture of a main coolant pipe and any missiles that may be generated.

b) The structural design of the missile shielding has taken into account both static and impact loads and is based upon a barrier cross section with energy absorption capacity at least 25 percent greater than that required when considering a potential missile.

c) Components of the reactor coolant system have been examined to identify and classify potential missiles according to size, shape, kinetic energy, and driving force for purposes of analyzing their effects.

d) The types of missiles for which missile protection is provided are:

Valve stems up to and including the largest size used

Valve bonnets

Instrument thimbles

Various sizes of nuts and bolts

Reactor vessel head bolts

Control rod drive mechanisms

5.2.4 METHOD OF ANALYSIS

The shell of the Reactor Building has been analyzed to determine all stresses, moments, shears and deflections due to the static and dynamic loads.

5.2.4.1 Reactor Building Shell

5.2.4.1.1 Static Solution

The static load stresses and deflections that are in a thin, elastic shell of revolution are calculated by an exact numerical solution of the general bending theory of shells. This analysis employs the differential equations derived by Reference 30. These equations are generally accepted as the standard ones for the analysis of thin shells of revolution. The equations given in Reference 30 are based on the linear theory of elasticity, and they take into account the bending as well as the membrane action of the shell.

The method of solution is the multisegment method of direct integration, which is capable of calculating the exact solution of an arbitrary thin, elastic shell of revolution when subjected to any given edge, surface, and temperature loads. This method of analysis was published in Reference 31 and has found wide application by many engineers concerned with the analysis of thin shells of revolution.

The actual calculation of the stresses produced in the shell and foundation was carried out by means of a computer program written by Reference 32. This computer program makes use of the exact equations given in Reference 30, and solves them by means of the multisegment method mentioned above. The program can solve up to four layers in a shell and these layers can have different elastic and thermal properties and can vary in thickness in the meridional direction. Applied loads can vary in meridional and circumferential directions.

The method of analysis for thermal loading of the shell at the penetrations and all pipe reactions and moments is performed by methods as suggested in References 33 and 34. Stress concentration factors used to analyze membrane stresses around the penetration are based upon these references.

With the exception of the openings for the equipment hatch and personnel lock, the next largest opening is the purge line sleeve which has a diameter of 48 inch. The diameter to wall thickness ratio is about 1:1.4. This opening and other smaller openings are designed as described in Sections 5.2.2.4 and 5.2.3.2.

Large openings in the Reactor Building:

1 - Equipment Hatch, 22 ft 4 inches inside diameter

1 - Personnel Lock, 10 ft inside diameter

The equipment access and personnel access openings are designed for the loads and load combination as specified in Sections 5.2.3.2.3 and 5.2.3.2.5. The analysis of discontinuity stresses resulting from these large openings utilize the finite element method. A complete description of the analysis and design, as applied to the equipment opening, is described in Appendix 5C.

5.2.4.1.2 Dynamic Solution

The stresses and displacements of the response of a shell of revolution to the excitation of an earthquake can be calculated by superimposing the normal modes of free-vibration of the shell. The modes of vibration are calculated by means of the general bending theory of shells derived by Reference 30. The translatory inertia terms in the normal, meridional, and circumferential direction of the shell are taken into account. The mass distribution is the actual mass distribution of the shell and no approximations are made. E. Reissner's shell theory is such that it predicts exactly the complete spectrum of natural frequencies of the shell without any approximations.

The differential equations given by Reference 30 are solved by means of the multisegment direct integration method of solving eigenvalue problems, which was published in Reference 35. According to this method, the eigenvalue problem of a shell of revolution is reduced to the solution of a frequency equation which vanishes at a natural frequency. The frequency equation consists of exact solutions of Reference 30 equations, and no approximations are made.

The calculation of the natural frequencies and the corresponding mode shapes of each mode of free-vibration is performed by means of a computer program (Reference 35). The computer program has been used for the calculation of the dynamic characteristics of many types of shells of revolution and its results have been verified with experiments on many occasions. The program calculates the natural frequencies of any rotationally symmetric thin shell within a given frequency interval and for the determined natural frequency calculates all the stresses, stress resultants, rotations, and displacements at any prescribed point on the meridian of the shell.

The normal modes of free-vibration need only be added in order to construct the response of the shell to an earthquake. The vertical and horizontal seismic components at any point on the shell are added by taking the sum of the absolute value of the contributing vertical frequencies and adding it to the sum of the absolute value of the contributing horizontal frequencies. The relationship between free-vibration and a given excitation is given by the following equation:

$$\mathrm{Y}\,(\mathrm{x},\mathrm{t}) = \sum_{\mathrm{i}=1}^{\mathrm{N}} \mathrm{Y}_{\mathrm{i}}\,(x)\,\frac{C_i S_{vi}}{W_i N_i}$$

where:

Y (x,t)	=	fundamental variables of the response
$Y_i(x)$	=	fundamental variables of the ith mode
C_i	=	constant for the ith mode
W_i	=	natural frequency of the ith mode
N_i	=	constant for the ith mode
S_{vi}	=	Maximum velocity from the response spectrum for a single degree of freedom system for a given value of Wi for the ith mode

For analysis purposes the Reactor Building shell is divided into structural parts, and each part is divided into a specified number of segments as shown in Figure 5.2-1.

The Static Analysis and Dynamic Analysis have been used by the following companies for the analysis of thin shells:

Martin Company - Orlando, Florida

Pratt and Whitney Aircraft - East Hartford, Connecticut

Central Electricity Generating Board - London, England

The Static Analysis has been evaluated in Reference 36.

The Dynamic Analysis is described and its results compared to experiment in Reference 37.

5.2.4.2 Interior Missiles

The missiles within the Reactor Building are relatively small as compared with the mass of the surrounding reinforced concrete structures. The high velocity and small impact area will cause the missiles to penetrate into the protective structures until its energies are dissipated. The analytical procedure in design of missile protection shielding is divided into the following three groups:

a. The Impact Area

The depth of penetration is computed based on an empirical correlation. The protective reinforced concrete shield must be of a thickness not less than three times the depth of penetration.

b. The Local Effect

The collapse mechanism investigated has the form of a circular fan (Reference 40). Energy absorbed by the protective structure, according to the Yield Line Theory, is based on an ideal condition of plastic impact (Reference 39). Mass ratio between the missile and part of the structure considered effective in absorbing the energy during the impact is assumed equal to 0.1. Based on the above condition, the kinetic energy to be absorbed by response of the collapse mechanism can be computed as $W_e = (W_1+W_2)(V_2')^2/2g$ (Reference 39, p. 128). Then work dissipated in the fan is $W_i = M\phi$ (Reference 40, Eq. 14, p. 28). Then, by equating $W_e = W_i$, the required minimum ultimate moment capacity of the structure, according to ACI 318-63 Chapter 16 (Reference 5) can be computed as $M_e = W_e/\phi$. The required structural dimensions and reinforcement are specified by ACI 318-63 Chapter 16 (Reference 5).

c. The Overall Structural Effect

The protective structure was investigated for a concentrated load applied at the point of missile impact. The overall structural stability was then verified.

5.2.5 INTERIOR STRUCTURE

The Reactor Building interior structure comprises the following elements:

a. Basement floor

b. Intermediate floor

c. Operating floor

d. Reactor cavity, the surrounding concrete wall is referred to as the primary shield wall.

e. Two steam generator compartments, the surrounding concrete walls are referred to as the secondary shield walls.

f. Refueling pool (located between the steam generator compartments and above the reactor cavity).

g. Equipment supports

h. Piping supports and pipe-whipping restraints

The interior structure of the Reactor Building is designed to withstand the following loading conditions:

Intermediate floor live load P = 200 lb/ft^2

Operating floor live load P = 750 lb/ft^2

Pipe support loads determined in the piping analysis

The refueling pool walls are designed for the hydrostatic pressure.

The reactor vessel is located in the reactor cavity which serves as a biological shield wall. The steam generator compartment walls serve as secondary shield walls and support the pressurizer, steam generators, and reactor coolant pumps. The supports of this equipment are designed for normal operating loads combined with a design basis high energy line break plus maximum hypothetical earthquake. Pipe whipping restraints are provided for the main steam and feedwater pipes.

The design pressure differential across walls of enclosed compartments in the internal structure is as follows:

Primary shield wall P = 165.3 psi

Secondary shield wall P = 15 psi

The reinforcing steel provided in the reactor pressure vessel cavity (primary shield) in the hoop direction is one layer of No.14S bars spaced at 8 inches vertically on both faces of the wall.

5.2.5.1 <u>Reactor Building Compartments Integrity Analysis</u>

A detailed pressure-temperature transient analysis was performed for two Reactor Building compartments, namely the reactor cavity and the steam generator compartment. The maximum differential pressures were obtained and the jet impingement forces due to various pipe ruptures have been determined. In all numerically performed analyses sufficiently small time steps (in the range of 0.0001 to 1 second) were used to assure proper mathematical convergence and limit truncation errors. All models analyzed include assumptions to obtain conservative results. Due to the relatively large relief areas of subcompartments investigated, moisture carryover effect was not considered.

Reactor cavity pressurization could potentially cause the rupture and possible ejection of some of the reactor cavity shield plugs. Disintegration of the shield plugs would create a source of loose sand in the area. The 2 inch reactor cavity drain valve WDL-V-520 is maintained in the closed position by administrative controls. This prevents direct transfer of potential sand particles in the reactor cavity area to the Reactor Building sump and to the Decay Heat pump suction. The shield plugs are capable of withstanding the pressurization forces and will not rupture. No sand will be transported to the reactor building sump.

5.2.5.2 Subcompartment Calculation Data*

Due to the inherent geometry of the TMI Nuclear Power Plant, the evaluation of subcompartment integrity could be treated analogous to the containment integrity analysis. Each compartment was treated independently and a postulated major occurrence was considered in any affected compartment. A summary of the compartments considered with the respective postulated occurrence is summarized below:

Subcompartment	Occurrence
Primary Shield Cavity	8.55 ft^2 reactor coolant pipe break
	14.1 ft^2 (double-ended guillotine) reactor coolant pipe break
Steam Generator	Double-ended 36 inch ϕ reactor coolant pipe break (14.1 ft^2)

* B&W Report on the effects of asymmetric LOCA loadings in the region of the reactor pressure vessel (B&W-1621 dated July, 1980) has been submitted to the NAC. This report was the result of a directive on January 25, 1978 by the NRC's Division of Operating Reactors requesting all pressure vessel Licensees to proceed with an evaluation of asymmetric LOCA loadings. When accepted by the NAC the following will be revised accordingly.

The energy and mass releases associated with the above occurrences are summarized in Table 5.2-4. Calculation of the subcompartment pressure-temperature transients was accomplished by use of the digital computer code, CONTEMPT-M (Reference 41). The analytical mode was restricted to the subcompartment volume and structure with relief into ambient conditions so as to obtain conservative pressure differentials. The justification of the applicability of the code itself to the solution of these analyses can be verified by CVTR (Reference 42) experiments and current state of the art.

a. Primary Shield Cavity

The unit has incorporated in its design an annular ring and shield plugs covering the annular space between the primary shield and reactor vessel flange for the purpose of shielding and prevention of neutron activation. In the event of a reactor coolant pipe rupture the ring and plugs are designed to unseat and then provide adequate relief area for the dissipation of pressure buildup in the cavity. Thus, at the instant of the assumed rupture in the cavity the available vent area is only through the tolerances around the main coolant piping which afford an area of 25.06 ft^2.

Since these pipes are insulated, it was assumed that the insulation will be expelled without resistance at the instant of rupture and thus, the total relief area immediately available was taken to be 43.30 ft^2 and the 98.3 ft^2 of relief area available due to the movement of the shield plugs around the reactor vessel flange annulus was added at 0.069 seconds after the rupture for a final total vent area of 141.6 ft^2. The CONTEMPT-M (Reference 41) computer program was then applied to determine the pressure transient in the primary cavity for the above conditions. The maximum pressure transient peaks obtained are given in Table 5.2-5. In addition to the above parameters, several assumptions were considered in this analysis, i.e.,

1) A constant back pressure (pressure in the containment) of 14.7 psia was assumed.

2) Heat losses to the structures were neglected.

b. Steam Generator Compartment

The pressure and temperature transient for the steam generator compartments resulting from a loss of coolant accident were calculated for the design limiting case of the double ended severance of a 36 inch diameter reactor coolant pipe. The analytical model was restricted to the steam generator compartment volume and the analysis performed using the CONTEMPT-M (Reference 41) computer code. The discharge mass and energy flow rate through the break were determined from the Reactor Coolant System blowdown and core thermal transient analysis performed by B&W and are given in Table 5.2-4.

A conservative analysis was performed by selecting the steam generator compartment which afforded the least flow relief area as the model. As in the analysis of the primary shield cavity, the balance of the containment was assumed to remain at ambient pressure and temperature of 14.7 psia and 120F during the transient. No heat transfer to structures was allowed.

The results of the analysis are summarized in Table 5.2-5.

5.2.5.3 Jet Impingement Forces

Blowdown forces resulting from jet impingement are determined by applying the momentum theorem of fluid dynamics. At the instant at which the rupture occurs and with all momentum transferred normal to the surface of impingement, the jet force is:

$F_j = pAV^2$

where p = fluid density

A = cross sectional area

V = fluid velocity

Assuming no losses, the force at the plane of rupture will be equal to the force of impingement. The instantaneous jet force at the break throat exist is $F_j = p_B A_B V_B^2$ where subscript B denotes conditions at the break. For a subcooled fluid, a metastable flow condition may exist initially at the throat and the jet force will be given by:

$F_j = 2PA_B$

where P is the system pressure (gauge). A saturated fluid or gas will be subject to the limitations of choked flow. Then the jet force may be given by:

$F_j = KPA_B$

where K is a thrust coefficient. For saturated water and steam, K is set equal to 1.26 based on Reference 43.

The pressure exerted by the jet on the structure may be given by:

$P_j = F_j / A_j$

$= KPA_B / A_j$

The area of jet impingement, A_j, is determined by the half-angle of jet expansion 22.5 degrees and distance from the break.

The maximum forces due to jet impingement were calculated utilizing a break area equal to the cross sectional area of the ruptured pipe. The system operating pressure was used and an instantaneous leak opening time was assumed. The results obtained are:

		Jet Force, kips	Size Break, ft^2
Reactor Coolant System			
Hot Leg	=	4,550	7.07

Cold Leg	=	2,750	4.28
Pump Suction	=	3,375	5.23
Main Steam Line	=	570	3.14
Feedwater Line	=	490	2.00

5.2.5.4 Blowdown Pressure Pulse

a. Primary Shield Wall

The primary shield wall design was based on the rupture of a pipe having a double-ended break area of 8.5 ft^2. This rupture resulted in a design pressure differential across the wall of 165.3 psi which produces a stress in the reinforcing steel equal to 40.4 ksi.

The rupture of a pipe having a double-ended break area of 14.1 ft^2 inside the primary shield wall results in a differential pressure across the wall of 186 psi. This pressure differential produces a stress in the reinforcing steel equal to 45.5 ksi. The stress of 45.5 ksi exceeds the specified minimum yield strength of 40.0 ksi but is about equal to the average yield strength of 45.2 ksi obtained from the user tests performed on the 14 S reinforcing bars used at the site.

For the latter loading, a dynamic analysis of the primary shield wall resulted in a ductility factor of $\mu \cong 1.0$.

A commonly used criterion for moderate damage is μ = 3 which implies considerable yielding of steel or cracking of concrete but no significant impairment of the resistance to future loading. A ductility factor μ = 20 may be permissible to prevent collapse under one load application (Reference 44).

Following a 14.1 ft^2 rupture inside the primary shield wall, the yielding of reinforcing steel should be essentially nil.

b. Secondary Shield Wall

The secondary shield wall design was based on the rupture of a pipe having a double-ended break area of 14.1 ft^2. This rupture resulted in a design pressure differential across the wall of 15 psi. The stress resulting in the reinforcing steel of the secondary shield wall from this 15 psi differential was less than the design maximum allowable stress of 36 ksi. Also, the design concrete compressive stress of the secondary shield wall used in the analysis was 2550 psi whereas the 28 day minimum compressive stress of the concrete actually used in the wall was 5000 psi.

5.2.5.5 Jet Impingement

a. Primary Shield Wall

Because of the geometric configuration, the jet impingement resulting on the primary shield wall from a pipe rupture within it is either tangential to its inner cylindrical surface or parallel to its vertical axis if the rupture is of the "fishmouth" type or, the intersection of a disk with its axis perpendicular to that of the cylinder it lies within if the rupture is of the circumferential type. In either case, the resultant force applied to the wall from this type of impingement is insignificant relative to the force applied because of the differential pressure across it. It may cause some slight erosion and spalling of the inner surface of the primary shield wall but otherwise does not have any significant effect on the integrity of the wall.

b. Secondary Shield Wall

The potential worst case of jet impingement on the secondary shield wall is a 7.07 ft^2 "fishmouth" rupture in the 36 inch reactor coolant pipe normal to and 18 inch from the secondary shield wall at the highest possible point above elevation 346 ft.

The initial 4550 kip load imposed on the affected wall area by this rupture results in an initial shear stress in the concrete of 480 psi. Although this is higher than design ultimate shear strength of 310 psi for the concrete, the condition exists for less than one second before the load reduces to result in a shear stress of less than 310 psi as the blowdown mass rate decreases. Therefore, it is concluded that there will be no puncture failure of the secondary shield wall.

However, there will be considerable spalling and erosion of the interior shield wall surface. Because of the jet impingement on it, the debris from this spalling and erosion will be pulverized into pieces too small to do any damage to other equipment or systems that could increase the severity of the incident.

In addition, the secondary shield wall was analyzed for the bending moment imposed on it due to the 4550 kip load imposed on it by the jet impingement. The analysis considered the reinforcing steel embedded in the concrete as the only portion of the structure resisting the load. Based on a nominal design minimum yield strength of 40 ksi of the reinforcing steel, the maximum bending moment load the wall can withstand is 5000 kips. Therefore, it is concluded that the wall can withstand the bending moment imposed on it without over stress.

The material used for the interior structural elements are:

1)	Reinforcing steel	ASTM A615, Grade 40
2)	Concrete	f'_C = 5000 psi
3)	Structural steel	A36 and A572

TABLE 5.2-1
(Sheet 1 of 1)

MINIMUM CONCRETE COVER

Location	Type of Steel	Minimum Cover To Be Used	Minimum Cover By ACI-318
Dome	Reinforcing	18 and 14S: 2-1/4 in. Others: 2 in.	2-1/4 and 1-3/4 in. 1-1/2 in.
	Prestressing	6 in.	1-1/2 in.
Cylinder	Reinforcing	18 and 14S: 2-1/4 in. Others: 2 in.	2-1/4 and 1-3/4 in. 1-1/2 in.
	Prestressing	6 in.	1-1/2 in.
Base Mat	Bottom reinforcing	18 and 14S: 3 in. Others: 3 in.	3 in. 3 in.
	Top reinforcing	18 and 14S: 2-1/4 in. Others: 2-1/4 in.	2-1/4 and 1-3/4 in. 1-1/2 in.

LOADING COMBINATIONS
Table 5.2-2
(Sheet 1 of 2)

	Loading Combination Number	Dead Load	Initial Pre-Stress	Prestress At Operating Condition*	Prestress Accident Condition*
Basic Loads		1.0			
			1.0		
				1.0	
					1.0
Construction Loading Combinations	1	1.0±.05			
	2	1.0±.05			
	3	1.0±.05			
	4	1.0±.05			
	5	1.0±.05	1.0		
Normal Operating Loading Combinations	6	1.0±.05		1.0	
	7	1.0±.05		1.0	
	8	1.0±.05		1.0	
	9	1.0±.05		1.0	
	10	1.0±.05		1.0	
	11	1.0±.05		1.0	
	12	1.0±.05		1.0	
	13	1.0±.05		1.0	
	14	1.0±.05		1.0	
	15	1.0±.05		1.0	
	16	1.0±.05		1.0	
	17	1.0±.05		1.0	
	18	1.0±.05		1.0	
	19	1.0±.05		1.0	
	20	1.0±.05		1.0	
	21	1.0±.05		1.0	
	22	1.0±.05		1.0	
	23	1.0±.05		1.0	
	24	1.0±.05		1.0	
Test Loading Combination	25	1.0±.05			1.0
Accident Loading Combinations	26	1.0±.05			1.0
	27	1.0±.05			1.0
	28	1.0±.05			1.0
	29	1.0±.05			1.0
	30	1.0±.05			1.0
	31	1.0±.05			1.0
	32	1.0±.05			1.0
	33	1.0±.05			1.0
	34	1.0±.05			1.0

*at end of plant life

LOADING COMBINATIONS
Table 5.2-2
(Sheet 2 of 2)

	Wind Load	Tornado Load	.06 g Earth quake	55-psig Internal Pressure	Operating Temperature	Accident Temperature
Basic Loads	1.0					
		1.0				
			1.0			
				1.0		
				1.15		
				1.25		
				1.5		
					1.0	
						1.0
Construction Loading Combinations	1.0					
	-1.0					
			1.0			
			-1.0			
Normal Operating Loading Combinations					1.0	
				-0.045	1.0	
	1.0					
	-1.0					
	1.0				1.0	
	1.0			-0.045	1.0	
	-1.0				1.0	
	-1.0			-0.045	1.0	
			1.0			
			-1.0			
			1.0		1.0	
			1.0	-0.045	1.0	
			-1.0		1.0	
			-1.0	-0.045	1.0	
		1.0		0.055	1.0	
		-1.0		0.055	1.0	
		1.0		0.010	1.0	
		-1.0		0.010	1.0	
Test Loading Combination				1.15		
Accident Loading Combinations				1.5		1.0
			1.25	1.25		1.0
			-1.25	1.25		1.0
	1.25			1.25		1.0
	-1.25			1.25		1.0
			2.0	1.0		1.0
			-2.0	1.0		1.0
	2.0			1.0		1.0
	-2.0			1.0		1.0

*At end of plant life

TABLE 5.2-3
(Sheet 1 of 1)

LOAD COMBINATION STRESSES

		Meridional Concrete			Hoop Concrete		
		Outside	Inside	Liner	Outside	Inside	Liner
Typical wall	No. 17	-39	-1,694	-21,520	-474	-2,093	-27,724
	No. 25	-421	-418	-3,597	-430	-429	-3,675
	No. 26	+781	-680	-36,000	-1,003	-603	-36,000
	No. 27	+551	-912	-36,000	+769	-843	-36,000
	No. 31	-486	-320	-36,000	-296	-297	-36,000
	No. 32	-93	-65	-36,000	-295	-250	-36,000
Typical dome	No. 17	-1,004	-1,839	-24,164	-902	-1,980	-25,352
	No. 25	-659	-603	-5,042	-593	-448	-3,968
	No. 26	+577	-490	-36,000	+840	-437	-36,000
	No. 27	+312	-671	-36,000	+528	-689	-36,000
	No. 31	-654	-239	-36,000	-513	-290	-36,000
	No. 32	-598	-243	-36,000	-409	-258	-36,000
Base of wall	No. 17	-677	-668	-1,218	-261	-633	-8,707
	No. 25	-1,450	+723	+5,615	-481	-47	-245
	No. 26	-2,040	+2,001	-36,000	-601	-194	-36,000
	No. 27	-1,987	-1,596	-36,000	-629	-305	-36,000
	No. 31	-342	-419	-36,000	-409	-71	-36,000
	No. 32	-1,326	-719	-36,000	+23	+81	-36,000
Top of wall	No. 17	-1,246	-2,235	-26,720	-540	-1,355	-17,951
	No. 25	-129	+504	+4,252	-344	-419	-3,662
	No. 26	-1,163	-870	-36,000	+733	-992	-36,000
	No. 27	-1,152	-1,137	-36,000	-770	-1,007	-36,000
	No. 31	-118	-175	-36,000	-273	-319	-36,000
	No. 32	-142	-19	-36,000	+413	-415	-36,000
Base of dome	No. 17	+520	-3,029*	-37,060	+438	-1,364	-19,237
	No. 25	-804	-798	-6,851	-695	-800	-6,869
	No. 26	+430	-882	-36,000	+323	-1,398	-36,000
	No. 27	+461	-1,284	-36,000	+419	-1,322	-36,000
	No. 31	-431	-830	-36,000	-247	-434	-36,000
	No. 32	-455	-714	-36,000	-387	-586	-36,000

* Compressive mild steel reinforcement provided.

TABLE 5.2-4
(Sheet 1 of 1)

BLOWDOWN DATA

HOT LEG

14.1 ft^2 Rupture Time

Time (Sec)	Mass Rate (lb/sec)	Energy (Btu/lb)
0	128,780	622.42
0.5	116,240	621.59
1.0	81,260	625.19
2.5	69,310	607.92
4.5	45,360	650.57
5.5	27,940	668.93
6.5	17,570	684.12
7.5	10,530	705.60
8.5	5,180	841.69
9.5	1,129	1151.36

8.55 ft^2 Rupture

0	1.83×10^5	550
0.05	1.05×10^5	550
0.10	9.90×10^4	560
0.5	8.95×10^4	570
1.0	7.88×10^4	580
2.0	7.33×10^4	590
4.4	6.20×10^4	600
6.4	4.33×10^4	630
8.4	1.99×10^4	710
10.0	8.62×10^3	810
13.0	4.6×10^3	930
15.0	1.04×10^3	1160

TABLE 5.2-5
(Sheet 1 of 2)

SUMMARY OF RESULTS

Subcompartment	Compartment Volume (ft^3)	Breaks Investigated (ft^2)	Total Areas Relief Considered (ft^2)
Primary Shield Cavity	10,000	8.55*	141.6
		14.1*	141.6
Steam Generator	85,000	14.1	910.0

* Values of mass and energy release are given in Table 5.2-4.

TABLE 5.2-5
(Sheet 2 of 2)

SUMMARY OF RESULTS

Subcompartments	Peak Sub-Compartment Pressure	Pressure Differential Across Wall	Time at Which Peak Occurs
Primary Shield Cavity	180.0	165.3	0.50
	200.7	186	0.60
Steam Generator	29.7	15.0	0.2

5.3 ISOLATION SYSTEM

5.3.1 DESIGN BASES

The general design basis governing isolation valve requirements is:

Leakage through fluid penetrations not serving accident-consequence-limiting systems is minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss of isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the Reactor Building, and various types of isolation valves.

Reactor Building partial isolation occurs on a signal of approximately 4 psig in the Reactor Building. A 30 psig signal provides isolation for certain lines not isolated by the 4 psig signal.

There are additional containment isolation signals such as the reactor trip, high radiation, 1600 psig RCS pressure (ESAS), and pipeline break signals. These signals assure that radioactive material is not inadvertently transferred out of the Reactor Building even if a 4 psig isolation signal is not reached.

The isolation system closes fluid penetrations not required for operation of the engineered safeguards in order to prevent leakage of radioactive materials to the environment. Fluid penetrations serving engineered safeguards also meet the design basis.

Remotely operated Reactor Building isolation valves are provided with position limit indicators in the Control Room.

5.3.2 SYSTEM DESIGN

The fluid penetrations that require isolation after an accident are classified as follows:

Type I — Each line connecting directly to the Reactor Coolant System has two Reactor Building isolation valves. One valve is external and the other is internal to the Reactor Building. These valves may be either a check valve and a remotely operated valve or two remotely operated valves, depending on the direction of normal flow.

Type II — Each line connecting directly to the Reactor Building atmosphere has two isolation valves. At least one valve is external and the other may be internal or external to the Reactor Building. These valves may be either a check valve and a remotely operated valve or two remotely operated valves, depending on the direction of normal flow.

Type III — Each line not directly connected to the Reactor Coolant System or not open to the Reactor Building atmosphere has at least one valve, either a check valve or a remotely operated valve. This valve is located external to the Reactor Building. A closed loop, which has a low probability of rupture during an accident, may be used as the second isolation barrier.

Type IV Lines that penetrate the Reactor Building and are connected to either the building or the Reactor Coolant System, but which are never opened during reactor operation, have two normally closed barriers. (e.g. blind flange, closed valve).

Table 5.3-2 identifies all of the reactor building fluid penetrations (i.e. all penetrations except electrical penetrations and sealed spares) except the personnel and equipment hatches. Table 5.3-2 identifies which “Type” design criteria is applicable and thereby demonstrates compliance with the criteria above (i.e., Type I, II, III or IV).

There are two exceptions on Table 5.3-2 to the penetration isolation design criteria: 1) RB Sump to DH suction header penetrations #352 & 354; and 2) RB Pressure Instrument penetrations #340, 350, 351. These penetrations serve “accident consequence limiting systems” (reference Section 1.4.53) and the accident mitigating function is performed more reliably without redundant containment barriers. Penetrations 352 & 354 to valves DH-V-6A&B is enclosed by a guard pipe as a supplemental design feature. The containment barrier for penetrations 340, 350 & 351 includes a closed loop of instruments and associated equipment outside of the Reactor Building.

The seismic Class III Industrial Cooling System piping, which provides a closed system inside containment, satisfies the criteria of having a low probability of rupture during an accident. The Reactor Building penetrations for the Industrial Cooling System and the Reactor Building Cooling Units through which the cooling water piping runs are Seismic Class I. The Industrial Cooling System is not a high energy piping system, and additional review has confirmed that this system is not subject to any design basis accident (MSLB/LOCA) missile or jet impingement effects inside containment. The closed system piping inside containment is included in the Local Leak Rate Test Program to verify closed system integrity.

The individual system flow diagrams show the manner in which each Reactor Building isolation valve arrangement fits into its respective system. Table 5.3-2 lists the mode of actuation, the type of valve, its normal position, and its isolation signal. The specific system penetrations to which each of these arrangements is applied are also presented. Each valve will be tested periodically during normal operation or during shutdown conditions to ensure its operability when needed. Plant operation is not adversely affected by the use of the on-line testing capability.

There is sufficient redundancy in the instrumentation circuits of the safeguards actuation system to minimize the possibility of inadvertent tripping of the isolation system. This redundancy and the instrumentation signals that trip the isolation system are discussed further in Chapter 7.1.3.

The diverse containment isolation system meets the requirements of IEEE Standard 279 (see 7.5 Reference 2). Electrical power for nuclear safety related actuation logic portions of the system is provided from Class IE uninterruptible sources.

All lines open to the Containment atmosphere or connected directly to the RCS (either normally or intermittently which can result in transfer of radioactivity out of Containment) are isolated on reactor trip except:

1. Required for Emergency Core Cooling Systems

2. Required for RCP operation
3. Required for Reactor Building cooling
4. RCS letdown to the makeup system
5. RB atmosphere radiation monitor
6. RB hydrogen recombiners
7. RB hydrogen monitors

In order to maintain support services for RCP operation, the following non-ECCS service lines do not close on a 4 psig Reactor Building isolation signal:

A. Reactor coolant pump seal return valves MU-V25 and 26 are isolated on 30 psig Reactor Building pressure signal or by the operator through remote manual operation on high radiation alarm.

B. Nuclear services closed cooling water and intermediate closed cooling water entering and exiting the Containment are isolated on a 30 psig Reactor Building pressure signal and by a pipeline break isolation signal coincident with a safety injection (HPI) signal.

Emergency Reactor Building cooling water system is automatically initiated on a 4 psig Reactor Building pressure signal or by a 1600 psig reactor coolant pressure signal. This system will take the place of the Normal Reactor Building Cooling Unit Coils System, which is isolated on a 4 psig Reactor Building Pressure signal or a 1600 psig Reactor Cooling Pressure signal.

RCS letdown is isolated on a 1600 psig Reactor Coolant Pressure signal. This line is not isolated on reactor trip to avoid thermal cycles on the letdown coolers.

RB atmosphere radiation monitor is isolated on a 1600 psig Reactor Coolant Pressure signal. This line is not isolated on a reactor trip to maintain the capability to detect RCS leakage.

Hydrogen monitor and hydrogen recombiner containment isolation valves are operable from the control room. These valves are normally closed and are opened for required surveillance activities and accident mitigation. These systems are periodically tested for leakage at a pressure at or above RB design pressure to ensure containment integrity is maintained when these valves are open.

There is no automatic isolation of the RB penetrations serving RCP seal injection or the Main Steam lines. Although the reactor coolant pumps are not required to mitigate the consequences of an accident, forced circulation of reactor coolant is the preferred mode for normal and most emergency conditions. Maintaining seal injection increases the reliability of the reactor coolant pump for forced circulation cooling. With seal injection in service, system pressure is well above containment design pressure and therefore leakage through this penetration is not possible. This penetration can be isolated from the control room after reactor coolant pump seal injection is not required.

The main steam line reactor building penetrations provide a flow path for RCS heat removal. For most design basis events, main steam line pressure is greater than containment pressure and therefore containment leakage through these penetrations is not possible. These penetrations will be isolated from the control room following a LBLOCA or if a loss of OTSG integrity results in containment pressure above OTSG pressure.

Leakage through the RB penetrations for the RB Fan Motor Coolers is prevented by a closed loop inside containment. To ensure the reliability of the ES RB Fans the outside RB isolation valves (NS-V-52 & 53 A/B/C) fail open on loss of air. In addition, maintaining NS system pressure above RB design pressure as described in Section 9.6.2.3 prevents leakage into the NS system inside containment.

The radiation monitors used for containment isolation actuation are classified as non-safety related.

The line break isolation for the intermediate cooling water and nuclear services cooling systems is designed to meet the following requirements:

a. Provide leak detection and isolation protection for Reactor Building pressure conditions which are below 30 psig such that if there were a leak in the piping system inside the Reactor Building during an accident, then that leak would be into rather than out of the Reactor Building.

b. Detect and alarm (in the Control Room) system leakage rates which, if undetected, would reduce the surge tank volume from high to low level within one hour or less.

c. Close the containment isolation valves when the level in the surge tank reaches the low-low level only if a safety injection (HPI) signal had been initiated.

Containment isolation signal overrides are provided on an individual signal source basis such that overriding the isolation signal due to one source will still allow the valves to be isolated by a second isolation source if it is activated. The override will be to the isolation signal which will not automatically reopen the isolation valves. Individual containment isolation valve logic does not allow override of the automatic isolation signal. Automatic isolation override capability for each penetration is described in Table 5.3-2. The high Reactor Building pressure signal cannot be bypassed. The override capability for 4 psig RB isolation is only functional after the high pressure signal is present. Operator action to reopen selected containment isolation valves will be required after the signal override has been accomplished. Prior to opening any containment isolation valve which has automatically closed, containment and RCS radiation levels, and the integrity of the system outside of containment will be evaluated.

High radiation signals that are bypassed are annunciated in the Control Room since the bypassing action of a high radiation signal could cause blocking of an isolation signal.

In response to Generic Letter 96-06, containment penetration overpressure protection devices were installed on pipe segments that could thermally overpressurize during an accident. Technical Data Report TDR-1212 (Reference 65) documents the susceptibility evaluation. Table 5.3-2 identifies which penetrations are protected by relief valves to prevent this condition.

5.3.2.1 Design Evaluation And System Operation

The reactor trip signal is used as a diverse containment isolation signal. This is conservative, since a reactor trip signal occurs on low reactor coolant pressure (1900 psig); in addition, it is anticipatory of an Engineered Safeguards Actuation Signal (ESAS) and occurs prior to ESAS initiation.

Lines which are isolated on reactor trip are:

a. Reactor Building sump

b. RCDT gas vents and liquid discharge

c. RCS sample lines

d. Containment purge lines

e. Demineralized water

f. OTSG sample lines (due to primary to secondary leaks)

g. Core flood makeup and sample lines

Closure of these paths by a signal that is not dependent on building pressure ensures that there will be no uncontrolled radioactivity release from containment for design basis events.

With the exception of the RCDT vents, the above lines are normally isolated. If these lines receive an isolation signal after a reactor trip, the plant condition is not degraded.

The use of the Reactor Protection System (RPS) would provide isolation for the following events:

a. Rod withdrawal accidents

b. Loss of coolant flow

c. Feedwater line break or loss of feedwater

d. Small steam line break accident outside containment (isolation of containment lines is still desirable)

e. Ejected rod accident

f. Boron dilution accident

g. Cold water addition

h. Iodine spikes or crud burst after trip

i. Loss of offsite power or station blackout

The 1600 psig reactor coolant pressure ESAS signal does not isolate containment for Items a, b, c, f, g, h, and i. Isolation on 1600 psig reactor coolant pressure ESAS for Items d and e does not cover a full spectrum of events.

Individual high radiation signals are used to prevent releases of radioactivity outside containment from the:

a. Reactor Coolant System letdown line

b. Reactor coolant drain tank vent and pump discharge

c. Reactor building purge

d. Reactor coolant sample lines

e. OTSG sample lines

f. RC pump seal return (alarm only)

g. Intermediate closed cooling water (alarm only)

Intermediate closed cooling water is alarmed on high radiation in order to prevent inadvertent releases due to letdown cooler leakage into the ICCW system. Isolation of the ICCW system does not jeopardize operation of the reactor coolant pumps since normally functioning seal water injection provides adequate cooling for the seals.

Individual radiation isolation has been chosen in lieu of a general radiation isolation signal because reactor trip isolation is anticipatory of a high radiation condition and individual isolation is more sensitive to isolating the source of activity.

5.3.2.2 Safety Evaluation

Diverse containment isolation signals on high radiation, reactor trip, and 30 psig Reactor Building pressure do not compromise plant safety for the following reasons:

a. The system is designed as nuclear safety related and single failure proof (except for high radiation isolation). Thus, the system will perform its safety function when required. The probability of containment isolation occurring on demand is increased.

b. The change to use high radiation, reactor trip and 30 psig RB pressure for valve closure signals on valves which were previously closed by 4 psig RB pressure is acceptable. Spurious closure of these valves was previously evaluated.

5.3.3 REACTOR BUILDING PURGE SYSTEM ISOLATION

With the exception of the equipment access hatch and the personnel airlock, the two Reactor Building purge lines are the largest penetrations of containment. Further, both of these penetrations periodically form a direct path between the Reactor Building atmosphere and the environment when the Reactor Building atmosphere is being purged.

In this respect, the valves used to isolate these penetrations are important in establishing and maintaining containment integrity.

5.3.3.1 Design Bases

The following design requirements were applied to the Procurement of the four 48 inch purge isolation valves:

a. Exterior Valves

1) Must be capable of closing against a differential pressure of 60 psi.

2) Must close fully in 2 seconds from the fully open position.

3) When closed, the valve must seal bubble-tight, that is, no air bubbles appear in a pool of water, with 125 psig air pressure applied across the closed face.

b. Interior Valves

1) Must be capable of closing against a differential pressure of 60 psi.

2) Must close fully in 5 seconds from the fully open position.

3) When closed, the valve must seal bubble-tight under the same conditions as Item a.3) above.

The Reactor Building purge isolation valves outside of containment (AH-V-1A/D) are limited to less than 31 degrees open and the purge isolation valves inside of containment (AH-V-1B/G) are limited to less than 33 degrees open by positive means while purging is conducted. The purge isolation valves can be opened fully only when the reactor is in cold shutdown or refueling shutdown. All of the purge isolation valves are required to close in less than or equal to 5.0 seconds from the nominal 30 degree open position to minimize post-accident release through the containment purge path. The valves are procedurally stroke tested from the nominal 30 degree open position to verify closure times. An analysis has been performed to demonstrate that the ductwork downstream of the purge exhaust containment isolation valves will withstand the peak post-LOCA containment pressure. This analysis demonstrates that the ductwork maintains its integrity and there will be no hostile environment for safety-related equipment or structures downstream of the purge valves.

5.3.3.2 Component Description

a. Valves

The Reactor Building purge system isolation valves are butterfly type 48 inch diameter, fabricated steel body with 125 lb flanges, fabricated steel disc with stainless steel seating edges, stainless steel shaft and Ethylene Propylene seat. The pressure retaining parts and bolts meet the material and non-destructive tests requirements of USAS B31.7 (February 1968 Draft, including June 1968 Errata) Class I, and meet Seismic Class I Requirements.

Both sets of valves, inside and outside the Reactor Building, are capable of satisfactory performance in the post-accident ambient conditions. All four valves are in areas protected from damage by missiles, pipe whip or equipment rupture. The valves inside the Reactor

Building are subject to normal operating temperature and pressure, but under design basis accident conditions are subject to a maximum of 281ºF and 55 psig. The valves outside the Reactor Building are subject to atmospheric pressure and temperatures ranging from 40ºF to 140ºF, with relative humidity ranging from 40% to 100%.

Under normal power operating conditions, including startup and hot shutdown, all four Purge Isolation valves are normally closed. Containment purging during these conditions is allowed in accordance with TMI-1 Technical Specification 3.6.

b. Valve Air Operators (Outside Reactor Building)

The valve operators on the valves located outside of the Reactor Building are spring return air cylinder type with the spring driving the valve to the closed position. Each valve is controlled with two 3-way solenoid valves, either one capable of permitting the valve to close from the 30 degree (nominal) open position in less than or equal to five seconds.

The air-operated valves have an adjustable piston travel stop installed in the air cylinder. The length of the mechanical stop corresponds to a maximum valve opening of 30 degrees (nominal), which is the open travel limit during normal plant operations. During cold shutdown or refueling operations, retraction of the piston travel stop allows the valves to be open fully (90 degrees).

c. Valve Motor Operators (Inside Reactor Building)

The valve operators on the valves located inside the Reactor Building are electric motor driven with a totally enclosed, non-ventilated type motor and gear drive.

Motor limit switches and circuitry for the motor-operated valves are set to limit the maximum valve open position to 30 degrees (nominal) open during normal plant operation. From this position, the motor-operated valves will close in less than or equal to five seconds. During cold shutdown or refueling operations, readjustment of the motor limit switches allows the valves to be open fully (90 degrees).

5.3.3.3 Component Evaluation

The purge valves will effectively maintain the containment isolation since:

a. Their design capabilities exceed the maximum postulated postaccident conditions of temperature and pressure.

b. The maximum closing time is sufficiently small so that postaccident release through the open containment will be minimized.

c. All four valves will close automatically upon receipt of the engineered safeguards actuation signal, a high radiation signal, or a manual override signal. In addition, diverse signals are provided to isolate the purge system, which will cause the valves to close automatically on a safety grade reactor trip or high containment pressure signals.

d. The overriding of any one type of isolation signal will not block any of the other signals from performing their isolation function.

e. The overriding or resetting of any isolation actuation signal will not cause automatic reopening of any purge isolation valve.

f. Key-operated permissive switches are installed in each of the purge valve open circuits. It is necessary for an operator to obtain a key and operate the permissive switch before a purge valve can be opened. Operation of a permissive switch will cause an alarm to be initiated in the Control Room.

g. The valves on the exterior portion of the containment building will automatically close on loss of control air pressure or loss of electrical power.

h. All valves are in locations protected from damage by the various postulated missiles.

i. The valves located inside of the Reactor Building have a debris screen installed on their inlet side to ensure the valves perform their containment isolation function, in the event that a LOCA causes debris (i.e., insulation, etc.) to be drawn toward the valves by escaping air and steam. The debris screens have been designed for all postulated loads, including seismic and the maximum containment LOCA pressure applied against a fully blocked debris screen.

5.3.4 LEAK RATE TEST SYSTEM

The leak rate test system is designed to enable initial integrated leakage rate testing of the Reactor Building prior to operation and to perform subsequent integrated leakage rate tests periodically during the life of the plant. A flow diagram of the Leak Rate Test (LR) system is shown on Drawing 302725.

Note: An alternate pressurization / depressurization flow path, through the Chemical Cleaning System piping, is shown in 302196. For the alternative path the Compressor Vendor is responsible to provide the cooling, drying and pressure control provisions that are described below as being permanently installed in the LR System.

The containment pressurization portion of the LR system is designed to produce dry, pressurized air for use as the testing medium. The pressurized air is to be supplied by a bank of portable air compressors which will be located in the yard outside the Reactor Building. The permanently installed part of the system is designed to accommodate compressors capable of delivering 6000 scfm at 100 psig. Permanently installed relief valves, check valves, and stop valves are located downstream of each compressor to facilitate removing or installing a compressor without interrupting pressurization for a test. High pressure air discharged from the compressors enters a permanently installed aftercooler and cyclone separator which reduces the air temperature from 220°F to 105°F and removes condensed moisture.

The cooled air then passes through an air dryer which further lowers the moisture content of the air. The air leaving the dryer is at a temperature of 105°F (75°F dewpoint) and 90 psig. The air then goes through a pressure reducing station where its pressure is reduced to the desired value before it enters the Reactor Building.

All piping within the pressurization portion of the system is designed in accordance with USAS B31.1.0. All equipment and supporting systems are seismic Class III, except as shown on Drawing 302725.

The containment pressurization portion of the system is provided with check valves, globe valves, pressure reducing valves, pressure and temperature instruments, flow meters, readout equipment, and other instrumentation required for safe and proper operation.

A leak rate test panel, located outside containment, contains the instruments for recording pressure, temperature, and dew point of the air inside containment. Flow meters are also provided in the panel for use in obtaining a known leak rate.

Provisions for controlled depressurization of containment following completion of a test are contained within the containment pressurization portion of the system.

5.3.5 LEAKAGE BLOCKING SYSTEMS

5.3.5.1 Penetration Pressurization System

The original Penetration Pressurization (PP) System had many automatic functions which have been eliminated by authorized modifications. The existing PP has the following non-nuclear safety related functions:

1. Convenient manually initiated source of air pressure for the periodic integrated leak rate test of the normally depressurized reactor building personnel and emergency access hatches, and for the periodic leak rate testing of the normally depressurized reactor building purge valves.

2. Provides convenient test connections for the periodic leak testing of the normally pressurized resilient seals on the equipment access hatch flange.

3. Test connections for the leak testing of the normally depressurized welded reactor building penetrations. (If required)

4. An automatic source of backup air supply (from the purge interspace pressurization tanks PP-T1A/B) to the Instrument Air system through an interconnecting orificed check valve.

5. Constant pressurization and leakage indication during reactor operation for the following containment leakage barriers, through air flow rotameters:

    a) Equipment Hatch Flange "O" rings to 60 psig with air.

    b) Electrical Penetrations (with epoxy seals) to 30 psig with nitrogen.

(See Table 5.3-4 for gas supplies to the PP System)

The Penetration Pressurization (PP) System was originally an Emergency Safeguards(ES) actuated system intended, post-accident, to block leakage from reactor building penetrations. Though many penetrations were pressurized full-time during plant operation, a few others, like the interspace between purge valves, and the reactor building door seals were intended to pressurize automatically post-accident in response to high pressure in the reactor building. These ES actuated supplies were provided with automatic backup air / nitrogen supplies in case the primary supply failed.

In addition to penetrations with bellows, resilient, and epoxy seals the original Penetration Pressurization System pressurized mechanical penetrations which had welded boundaries. Since the Penetration Pressurization system was not a qualified safety system and could not be used for benefit in the Safety Analysis it was converted into a strictly manual system with a much more limited function as indicated above.

The bases for constant pressurization of certain containment leakage barriers is that they have elastomer "O" rings or epoxy seals which are more prone to degradation than welds, bellows, valve seats, or spiral-wound type gaskets. The epoxy seals on electrical penetrations are not periodically replaced so that degradation is cumulative over plant life.

Periodic PP system flow meter readings for the constantly pressurized components are documented and evaluated to assist in timely determination and resolution of any leakage problem. However, the official leak rate testing of the penetrations, to meet 10 CFR 50 Appendix J criteria, is by separate Tech Spec Surveillance (Local Leak Rate Testing for the "O" rings and RB Integrated Leak Rate Testing for the epoxy seals). If the PP system were intact post-accident the pressurization would be expected to limit the potential leakage of these components. The PP system is not, however, assumed by the SAR to exist post-accident. Only those check valves or closed isolation valves, where the PP system interfaces with Nuclear Safety Related systems or components, are designed/maintained as safety grade. The PP system supply manifolds are purposely vented for the periodic Tech Spec-required leak rate tests to mock up a failed PP system.

The Purge Interspace Pressurization Tanks PP-T1A/1B are not assumed to be nuclear safety related though they act as a back-up air supply for the Instrument Air (IA) System. The interconnection between the IA system and PP-T1A/1B is through a check valve with orifice in the disc that severely limits potential leakage from the IA system if the tank fails. However, it allows full pipe flow from PP-T1A/1B to the Instrument Air system if needed.

5.3.5.2.1 Fluid Block System

The Fluid Block (FB) System was deactivated and ripped out due to a demonstrated lack of benefit to safety. That system originally provided automatic pressurization, post-LOCA, for the bonnets of numerous containment isolation gate valves. The system design and construction was not safety grade so that TMI could take no credit for it's existence post-accident. Also, the required valve leak-tightness could be easily maintained and the valves could be periodically leak-tested without the FB system. Consequently it created an unreasonable maintenance and operation burden.

TABLE 5.3-1

DELETED

(Sheet 1 of 5)
TABLE 5.3-2
REACTOR BUILDING ISOLATION VALVE INFORMATION

				INSIDE REACTOR BUILDING						OUTSIDE REACTOR BUILDING					
Pent No	**Description**	**Type**	**Flow**	**In Isolation**	**Valve Type**	**Size (in.)**	**Valve Operator**	**Auto Valve Closure**	**Normal Position**	**Out Isolation**	**Valve Type**	**Size (in.)**	**Valve Operator**	**Auto Valve Closure**	**Normal Position**
336	RB PURGE EXHAUST	II	OUT	AH-V-1B	BUTTERFLY	48	EMO	7	CLOSED	AH-V-1A	BUTTERFLY	48	SPRING	1,4,5	CLOSED
423	RB PURGE SUPPLY	II	IN	AH-V-1C	BUTTERFLY	48	EMO	7	CLOSED	AH-V-1D	BUTTERFLY	48	SPRING	1,4,5	CLOSED
301	BUILDING SPRAY A	II	IN	BS-V-30A	CHECK	8			CLOSED	BS-V-1A	GLOBE	8	EMO		CLOSED
308	BUILDING SPRAY B	II	IN	BS-V-30B	CHECK	8			CLOSED	BS-V-1B	GLOBE	8	EMO		CLOSED
340	RB PRESSURE INSTRUMENTS	**							OPEN	CLOSED LOOP					
350	RB PRESSURE INSTRUMENTS	**							OPEN	CLOSED LOOP					
351	RB PRESSURE INSTRUMENTS	**							OPEN	CLOSED LOOP					
351	RB PRESSURE INSTRUMENTS	**							OPEN	CLOSED LOOP					
307	RECLAIMED WATER	II	IN	CA-V-192	CHECK	2			CLOSED	CA-V-189 CA-V-443	GATE RELIEF	2 ¼	SPRING SPRING	1,5	CLOSED CLOSED
328	RCS & PRESSURIZER SAMPLE LINE	I	OUT	CA-V-13	GLOBE	1	EMO	1,4,5	OPEN	CA-V-2 CA-V-446	GLOBE RELIEF	1 ½	SPRING SPRING	1,4,5	OPEN CLOSED
213	Capped Spare														
214	Capped Spare														
106	CHEM CLEANING	IV	BOTH	FLANGE		4			CLOSED	FLANGE		4			CLOSED
105	CHEM CLEANING	IV	BOTH	SPEC FLG		8			CLOSED	SPEC FLG		8			CLOSED
348	CF TANK A MAKEUP/N2	I	IN	CF-V-12A	CHECK	1			CLOSED	CF-V-19A CF-V-46A	GATE RELIEF	1 ¼	SPRING SPRING	1,5	CLOSED CLOSED
349	CF TANK B MAKEUP/N2	I	IN	CF-V-12B	CHECK	1			CLOSED	CF-V-19B CF-V-46B	GATE RELIEF	1 ¼	SPRING SPRING	1,5	CLOSED CLOSED
348	CF TANK A SAMPLE	I	OUT	CF-V-2A	GLOBE	1	EMO	1,5	CLOSED	CF-V-20A	GATE	1	SPRING	1,5	CLOSED
349	CF TANK B SAMPLE	I	OUT	CF-V-2B	GLOBE	1	EMO	1,5	CLOSED	CF-V-20B	GATE	1	SPRING	1,5	CLOSED
108	RB ATMOS SAMPLE RETURN	II	IN							CM-V-1 & 2	BALL	1	SPRING	1,2	OPEN
108	RB ATMOSPHERE SAMPLE	II	OUT							CM-V-3 & 4	BALL	1	SPRING	1,2	OPEN
306	RCS DROPLINE TO DH	I	OUT	DH-V-2 DH-V-37	GATE RELIEF	12 ¾	EMO SPRING	3	CLOSED CLOSED	DH-V-3	GATE	12	EMO		CLOSED
303	DH A INJECTION LINE	I	IN	DH-V-22A	CHECK	10			CLOSED	DH-V-4A	GATE	10	EMO		CLOSED
310	DH B INJECTION LINE	I	IN	DH-V-22B	CHECK	10			CLOSED	DH-V-4B	GATE	10	EMO		CLOSED
320	AUX SPRAY LINE	I	IN	DH-V-69	CHECK	1.5			CLOSED	DH-V-64	GLOBE	2	HW		CLOSED
352	RB SUMP TO DH A	**							OPEN	DH-V-6A	GATE	14	EMO		CLOSED
354	RB SUMP TO DH B	**							OPEN	DH-V-6B	GATE	14	EMO		CLOSED
110	EMERGENCY FEEDWATER A	III	IN	CLOSED LOOP						EF-V-12A	CHECK	6			CLOSED
111	EMERGENCY FEEDWATER B	III	IN	CLOSED LOOP						EF-V-12B	CHECK	6			CLOSED
212	FUEL TRANSFER TUBE EAST	IV	BOTH	FLANGE		30			CLOSED	FH-V-1A	GATE	30	HW		CLOSED
332	FUEL TRANSFER TUBE WEST	IV	BOTH	FLANGE		30			CLOSED	FH-V-1B	GATE	30	HW		CLOSED
216	FIRE SERVICE	IV	IN	FS-V-401	GLOBE	4	HW		CLOSED	FLANGE		4			CLOSED
227	MAIN FEEDWATER A	III	IN	CLOSED LOOP						FW-V-12A	CHECK	20			OPEN
103	MAIN FEEDWATER B	III	IN	CLOSED LOOP						FW-V-12B	CHECK	20			OPEN

(Sheet 2 of 5)
TABLE 5.3-2
REACTOR BUILDING ISOLATION VALVE INFORMATION

				INSIDE REACTOR BUILDING						OUTSIDE REACTOR BUILDING					
Pent No	Description	Type	Flow	In Isolation	Valve Type	Size (in.)	Valve Operator	Auto Valve Closure	Normal Position	Out Isolation	Valve Type	Size (in.)	Valve Operator	Auto Valve Closure	Normal Position
210	OTSG B BLOWDOWN & LIQUID SAMPLE SYSTEM	III	OUT	CA-V-4B	GLOBE	2	EMO	1,4,5	CLOSED	CA-V-5B CA-V-449	GLOBE RELIEF	2 1/2	AIR/SPG SPRING	1,4,5	CLOSED CLOSED
211	OTSG A BLOWDOWN & LIQUID SAMPLE SYSTEM	III	OUT	CA-V-4A	GLOBE	2	EMO	1,4,5	CLOSED	CA-V-5A CA-V-449	GLOBE RELIEF	2 1/2	AIR/SPG SPRING	1,4,5	CLOSED CLOSED
420	RB H2 MONITOR A RETURN	II	IN	HM-V-3A	GLOBE	0.5	SPRING		CLOSED	HM-V-1A	GLOBE	0.5	SPRING		CLOSED
101	RB H2 MONITOR B RETURN	II	IN	HM-V-3B	GLOBE	0.5	SPRING		CLOSED	HM-V-1B	GLOBE	0.5	SPRING		CLOSED
420	RB H2 MONITOR A INLET	II	OUT	HM-V-4A	GLOBE	0.5	SPRING		CLOSED	HM-V-2A	GLOBE	0.5	SPRING		CLOSED
101	RB H2 MONITOR B INLET	II	OUT	HM-V-4B	GLOBE	0.5	SPRING		CLOSED	HM-V-2B	GLOBE	0.5	SPRING		CLOSED
240	HYDROGEN PURGE	IV	OUT						OPEN	HP-V-1 & 6	GATE	6	HW		CLOSED
415	H2 RECOMBINER SUPPLY	II/IV	OUT	HR-V-22A/B FLANGE	GLOBE	2 6	SPRING		CLOSED CLOSED	HR-V-2A/B FLANGE	GLOBE	2 6	HW		CLOSED CLOSED
416	H2 RECOMBINER RETURN	II/IV	IN	HR-V-23A/B FLANGE	GLOBE	2 6	SPRING		CLOSED CLOSED	HR-V-4A/B FLANGE	GLOBE	2 6	HW		CLOSED CLOSED
109	INSTRUMENT AIR	IV	IN	IA-V-20	GLOBE	2	HW		CLOSED	IA-V-6	GLOBE	2	HW		CLOSED
302	ICCW RETURN	III	OUT	IC-V-2	GATE	6	EMO	8,9	OPEN	IC-V-3 IC-V-103	PLUG RELIEF	6 ¼	AIR SPRING	8,9	OPEN CLOSED
333	ICCW SUPPLY (RCP & LD COOLERS)	III	IN	IC-V-18	CHECK	6			OPEN	IC-V-4	PLUG	6	AIR	8,9	OPEN
334	ICCW SUPPLY TO CRDM	III	IN	IC-V-16	CHECK	3			OPEN	IC-V-6	GLOBE	3	SPRING	8,9	OPEN
417	RB LEAK RATE TEST SUPPLY	IV	IN	FLANGE		6			CLOSED	SPEC FLG		6			CLOSED
113	MAIN STEAM LINE B (OTSG A)	III	OUT	CLOSED LOOP						MS-V-1A/B,2A MS-V-17/20B	STOP CHECK,GATE RELIEF	24,12 4x6	EMO SPRING		OPEN CLOSED
413	OTSG A ANNULUS DRAIN	III	OUT	CLOSED LOOP						MS-V-1A/B,2A	STOP CHECK,GATE	24,12	EMO		OPEN
112	MAIN STEAM LINE A (OTSG A)	III	OUT	CLOSED LOOP						MS-V-1A/B,2A MS-V-17/21A	STOP CHECK,GATE RELIEF	24,12 5x6	EMO SPRING		OPEN CLOSED
114	MAIN STEAM LINE C (OTSG B)	III	OUT	CLOSED LOOP						MS-V-1C/D,2B MS-V-17/21C	STOP CHECK,GATE RELIEF	24,12 5x6	EMO SPRING		OPEN CLOSED
419	MAIN STEAM LINE D (OTSG B)	III	OUT	CLOSED LOOP						MS-V-1C/D,2B MS-V-17/20D	STOP CHECK,GATE RELIEF	24,12 4x6	EMO SPRING		OPEN CLOSED
102	OTSG B ANNULUS DRAIN	III	OUT	CLOSED LOOP						MS-V-1C/D,2B	STOP CHECK,GATE	24,12	EMO		OPEN
321	HPI LINE A	I	IN	MU-V-107A	CHECK	2.5			CLOSED	MU-V-16A	GLOBE	2.5	EMO		CLOSED
322	HPI LINE B	I	IN	MU-V-107B	CHECK	2.5			CLOSED	MU-V-16B	GLOBE	2.5	EMO		CLOSED
338	HPI LINE C	I	IN	MU-V-107C	CHECK	2.5			CLOSED	MU-V-16C	GLOBE	2.5	EMO		CLOSED
339	HPI LINE D	I	IN	MU-V-107D	CHECK	2.5			CLOSED	MU-V-16D	GLOBE	2.5	EMO		CLOSED
323	NORMAL RCS MAKEUP	I	IN	MU-V-219	CHECK	2.5			OPEN	MU-V-18	GLOBE	2.5	SPRING	1,2	OPEN
337	RCP SEAL INJECTION	I	IN	MU-V-116	CHECK	1.5			OPEN	MU-V-20	GLOBE	4	SPRING		OPEN
329	RCP SEAL RETURN	I	OUT	MU-V-25	GLOBE	4	EMO	8	OPEN	MU-V-26	GLOBE	4	SPRING	8	OPEN
309	RCS LETDOWN	I	OUT	MU-V-2A/B	GLOBE	2.5	EMO	1,2,4	OPEN	MU-V-3 MU-V-238	GLOBE RELIEF	2.5 ¼	AIR/ SPRING SPRING	1,2	OPEN CLOSED
307	NITROGEN	IV	IN							NI-V-26 & 27	GLOBE	1	HW		CLOSED

(Sheet 3 of 5)
TABLE 5.3-2
REACTOR BUILDING ISOLATION VALVE INFORMATION

				INSIDE REACTOR BUILDING						OUTSIDE REACTOR BUILDING					
Pent No	**Description**	**Type**	**Flow**	**In Isolation**	**Valve Type**	**Size (in.)**	**Valve Operator**	**Auto Valve Closure**	**Normal Position**	**Out Isolation**	**Valve Type**	**Size (in.)**	**Valve Operator**	**Auto Valve Closure**	**Normal Position**
346	NSCCW SUPPLY TO RB	III	IN	NS-V-11	CHECK	8			OPEN	NS-V-15	GATE	8	EMO	8,9	OPEN
347	NSCCW RETURN FROM RB	III	OUT	NS-V-35	GATE	8	EMO	8,9	OPEN	NS-V-4	GATE	8	EMO	8,9	OPEN
401	NS SUPPLY TO AH-E-1A MOTOR	III	IN	CLOSED LOOP						NS-V-52A	GATE	1	AIR		OPEN
402	NS SUPPLY TO AH-E-1B MOTOR	III	IN	CLOSED LOOP						NS-V-52B	GATE	1	AIR		OPEN
403	NS SUPPLY TO AH-E-1C MOTOR	III	IN	CLOSED LOOP						NS-V-52C	GATE	1	AIR		OPEN
404	NS RETURN FROM AH-E-1A MOTOR	III	OUT	CLOSED LOOP						NS-V-53A NS-V-36A	GATE RELIEF	1 ½	AIR SPRING		OPEN CLOSED
405	NS RETURN FROM AH-E-1B MOTOR	III	OUT	CLOSED LOOP						NS-V-53B NS-V-36B	GATE RELIEF	1 ½	AIR SPRING		OPEN CLOSED
406	NS RETURN FROM AH-E-1C MOTOR	III	OUT	CLOSED LOOP						NS-V-53C NS-V-36C	GATE RELIEF	1 ½	AIR SPRING		OPEN CLOSED
421	RB INDUSTRIAL COOLING SUPPLY	III	IN	CLOSED LOOP						RB-V-2A	GATE	8	EMO	1,2	OPEN
422	RB INDUSTRIAL COOLING RETURN	III	OUT	CLOSED LOOP RB-V-1A,B,C, D,E	 RELIEF	 3x1/2	 SPRING		 CLOSED	RB-V-7	GATE	8	EMO	1,2	OPEN
410	RB EMERG COOLING SUPPLY A	III	IN	CLOSED LOOP						RR-V-3A	GATE	12	EMO		OPEN
411	RB EMERG COOLING SUPPLY B	III	IN	CLOSED LOOP						RR-V-3B	GATE	12	EMO		OPEN
412	RB EMERG COOLING SUPPLY C	III	IN	CLOSED LOOP						RR-V-3C	GATE	12	EMO		OPEN
407	RB EMERG COOLING RETURN A	III	OUT	CLOSED LOOP						RR-V-4A RR-V-11A	GATE RELIEF	12 ½	EMO SPRING		CLOSED CLOSED
408	RB EMERG COOLING RETURN B	III	OUT	CLOSED LOOP						RR-V-4B RR-V-11B	GATE RELIEF	12 ½	EMO SPRING		CLOSED CLOSED
409	RB EMERG COOLING RETURN C	III	OUT	CLOSED LOOP						RR-V-4C/D RR-V-11C	GATE RELIEF	12 ½	EMO SPRING		CLOSED CLOSED
104	SERVICE AIR	IV	IN	SA-V-3	GLOBE	2	HW		CLOSED	SA-V-2	GLOBE	2	HW		CLOSED
104	OTSG DRAIN	IV	OUT	SPEC FLG		2			CLOSED	SPEC FLG		2			CLOSED
304	FUEL XFER CANAL FILL/DRAIN	IV	BOTH							SF-V-22 & 23	GATE	8	HW		CLOSED
330	RC DRAIN TANK VENT	III	BOTH	WDG-V-3	GLOBE	2.0	EMO	1,4,5	OPEN	WDG-V-4	GLOBE	2	SPRING	1,4,5	OPEN
331	REACTOR COOLANT DRAINS TO WDL	III	OUT	WDL-V-303	GATE	3.0	EMO	1,4,5	CLOSED	WDL-V-304 WDL-V-727	GLOBE RELIEF	3 ¼	SPRING SPRING	1,4,5	CLOSED CLOSED
353	RB SUMP DRAIN TO WDL	II	OUT							WDL-V-534&535	GATE	6	SPRING	1, 5	CLOSED
414	OUTAGE ROUTING	IV	BOTH	FLANGE		12			CLOSED	FLANGE		12			CLOSED
241	INCORE REPLACEMENT	IV	BOTH	FLANGE		18			CLOSED	FLANGE		18			CLOSED
221E	OUTAGE ROUTING	IV	BOTH	FLANGE		12			CLOSED	FLANGE		12			CLOSED
222E	OUTAGE ROUTING	IV	BOTH	FLANGE		12			CLOSED	FLANGE		12			CLOSED

TABLE 5.3-2
(Sheet 4 of 5)

REACTOR BUILDING ISOLATION VALVE INFORMATION

GENERAL NOTES:

- All valves with electric motor operators are also equipped with hand wheels (except MS-V-1A/B/C/D).
- ‘Type’ refers to penetration barrier design criteria type (** are exceptions), see text in 5.3.2.
- All spring operated valves are air or solenoid to open, and spring to close.
  All solenoid operated valves are energize to open, spring to close.
- All electric motor operators, air and solenoid operated valves may be remote operated manually.
- All valves which require air to close have reserve air supplies which are not dependent upon the IA system, except for NS-V-52 & 53, see Section 5.3.2.

THE SYSTEM ABBREVIATIONS ARE DEFINED AS FOLLOWS:

MU Makeup and Purification System

DH Decay Heat Removal System

RB Reactor Building Cooling System

SF Spent Fuel Cooling System

WD Waste Disposal System

CA Chemical Addition and Sampling System

BS Reactor Building Spray System

IC Intermediate Cooling System

NS Nuclear Services Cooling Water System

RM Radiation Monitoring

LR Leak Rate Test

IA Instrument Air

SA Service Air

TABLE 5.3-2
(Sheet 5 of 5)

REACTOR BUILDING ISOLATION VALVE INFORMATION

ABBREVIATIONS (continued):

CF	Core Flooding
NI	Nitrogen
HR	Hydrogen Recombiner
FS	Fire System

VALVE CLOSURE SIGNAL SOURCES:

1) 4 Psig Reactor Building Pressure Isolation
(includes common defeat capability)

2) RCS 1600 Psig (Reactor Building) Isolation
(includes common bypass and defeat capability)

3) RCS 400 psig DH system overpressure protection

4) High Radiation (Non-Safety) Isolation
(includes individual signal bypass capability)

5) Reactor Trip Isolation
(includes common defeat capability)

6) Deleted

7) AH-V-1B, 1C Auto Close on Reactor Trip and 4 psig RB and on Hi Radiation. Bypass and override capabilities are the same as 5, 1 and 4 except that in addition an individual valve key switch must be operated to open the valve.

8) 30 Psig Reactor Building Pressure Isolation
(no defeat or bypass capability)

9) Line Break Isolation Signal
(includes common override capability through bypass or defeat of 1600# signal)

TABLE 5.3-3

DELETED

TABLE 5.3-4
(Sheet 1 of 2)

SOURCES OF GAS SUPPLY TO THE PENETRATION PRESSURIZATION

SUPPLY		MODE OF OPERATION
FROM	TO	
Instrument Air Receivers	Penetration Pressurization System (Mechanical Penetrations Portions	Primary Source During Normal and Accident Conditions
Nitrogen Manifold	Penetration Pressurization System (Electrical Portions Only)	Primary Source During Normal Operation
Service Air Receivers	Penetration Pressurization System Via the Instrument Air Supply System	Available by manual valve alignment if the Penetration Pressurization System Drops Below 57 psig

TABLE 5.3-4
(Sheet 2 of 2)

SOURCES OF GAS SUPPLY TO THE PENETRATION PRESSURIZATION

SUPPLY		MODE OF OPERATION
FROM	TO	
Instrument Air & Service Air Receivers	a. Electrical Penetrations b. Inner & Outer Door of Normal & Emergency Air Locks	Penetration Pressuriza-Tion Actuation Occuring ona Containment Isolation Signal
Purge Interspace Pressurization Tank	Containment Build-ing Purge Inter-space between the two sets of Purge Isolation Valves	During Accident Conditions or for Periodic Purge Valve Leak Tests. Manually Controlled
Purge Interspace Pressurization Tank	Penetration Pressurization System via Instrument Air Supply System	Normal alignment, can be manually isolated, if desired, to supply containment Building Purge Interspace independent of the Instrument Air Supply System.
Penetration Pressurization System	Purge Interspaces	Manually Actuated, if desired, to maintain purge interspaces pressurized

*During normal operation of nitrogen manifold supplies the electrical portion of the Penetration Pressurization System.

TABLE 5.3-5
(Sheet 1 of 1)

PENETRATION PRESSURIZATION SYSTEM

Design Codes and Criteria

Air Tanks -	(PP-TIA and PP-TIB) - 125 psig and 100°F ASME Section VIII Seismic I
Valves -	Design pressure 63.3 psig Design temperature 120°F Material - carbon steel and brass Seismic Class I/III (As shown on Drawings 302706 and 302707)
Flowmeters-	63.3 psig 120°F Material carbon steel Seismic Class III
Piping -	carbon steel 125 psig 120°F Seismic Class I/III (As shown on Drawings 302706 and 302707) USAS B31.1.0 Power Piping Code

5.4 OTHER SPECIAL STRUCTURES

The special structures for this station are:

a. Control Building

b. Fuel Handling Building

c. Designated portions of the Auxiliary Building (see Figure 5.1-1)

d. Designated portions of the Intermediate Building (see Figure 5.1-1)

e. Diesel Generator Building

f. Intake screen house and pump house

g. Heat exchanger vault

h. Air intake structure

i. Equipment support

j. Access Tunnel - vault to Auxiliary Building

These structures are Class I structures, that is, they protect vital components, systems, including instruments and controls, whose failure might jeopardize a safe shutdown of the reactor. See Figures 5.1-1 and 5.4-1.

5.4.1 STRUCTURAL DESIGN PARAMETERS

The loads used in design of the special structures have been determined based on operating and accidental requirements, as specified below, in addition to regular loads as required by applicable codes.

5.4.1.1 <u>During Operation</u>

The loads due to normal operating conditions are:

a. Dead load

b. Live load

c. Snow and wind load

d. Equipment support

e. Design earthquake (OBE), see Section 5.1.2.1.1

5.4.1.2 Protection of Safeguards

The protective structures as listed above in Section 5.4 have been designed for such incidents as:

a. Tornado loads, see Section 5.2.1.2.6

b. Hypothetical aircraft incident, See Appendix 5A (except for Diesel Generator Building and Chemical Cleaning Building Basin).

c. Main steam turbine missiles

d. Tornado missiles, see Section 5.2.1.2.6

e. Maximum hypothetical earthquake (SSE), see Item b) of Section 5.1.2.1.1.

5.4.2 MATERIALS AND SPECIFICATIONS

See Section 5.2.2.

5.4.2.1 Concrete

Concrete has a minimum compressive strength of 5000 psi in 28 days. See Section 5.2.2.1.

5.4.2.2 Mild Reinforcing Steel

The mild steel reinforcing conforms to ASTM A-615-68 grade 60 for 14s and 18s rebars. All the smaller size rebars conform to ASTM A-615-68 grade 40. See Section 5.2.2.2.

5.4.3 STRUCTURAL DESIGN CRITERIA

This design has been based on ACI 318 (Reference 5), "Working Stress Design" for normal operating conditions, and "Ultimate Strength Design" for aircraft and missile impact. Although the probability of an aircraft incident is extremely remote, the purpose of this study is to assure that hypothetical objects are prevented from penetrating the structure or causing it to collapse.

5.4.3.1 Codes

Same as Items a, b, c, and d of Section 5.2.3.1.

Inspection of structural weldments that are under the purview of American Welding Society Standard D.1.1 or other non-ASME class structures shall be conducted in accordance with the provisions of Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants (VWAC), Revision 2. VWAC is not utilized for inservice inspections required by ASME Section XI.

5.4.3.2 Loading Combinations

The design has been based upon normal operating loads, earthquake load, and accident loads as described in Sections 5.4.1.1 and 5.4.1.2.

5.4.3.2.1 At Normal Operating Conditions

The stresses in the concrete and reinforcing steel resulting from combinations of those loads listed in Subsection 5.4.1.1 have been in accordance with ACI 318-63 (Reference 5), "Working Stress Design."

5.4.3.2.2 Tornados

The special structures have been designed to withstand short term tornado loadings including tornado-generated missiles where such structures house systems and components whose failure could result in an inability to safely shut down and isolate the reactor. Structures that are so designed include the following:

a. Control, relay, and battery rooms in the Control Building

b. Designated areas of Auxiliary and Intermediate Buildings, see Figure 5.1-1

c. Fuel Handling and Diesel Generator Buildings

d. Intake screen and pump house

The tornado design requirements are as described in Section 5.2.1.2.6. The structural design has been done in accordance with ACI 318-63 (Reference 5), "Ultimate Strength Design."

The following Class I components are not designed for protection from tornado-generated missiles:

a. Borated water storage tank

b. Condensate storage tanks

c. Plant vent

d. Air intake structure portion above ground

5.4.3.2.3 Aircraft Impact

Those structures vital for plant shutdown and protection of safeguards as listed in Items a, b, c, d, and f of Section 5.4 are designed for the aircraft loadings as explained in Appendix 5A.

5.4.3.2.4 Main Steam Turbine Missiles

The Nuclear Regulatory Commission (NRC) requires that utilities in the United States consider the effects of turbine missiles as well as many other hypothetical events that might effect the operation of nuclear power plants. Utilities commonly demonstrate protection against the effects of turbine missiles by the use of probability-based analyses. The NRC has developed guidelines that limit the maximum annual probability for various hypothetical events. In the case of TMI Unit 1, the limit for the annual probability for generation of a turbine missile is 1×10^{-5}.

The proprietary GE Co. report to the NRC dated January 1984 and entitles, “Probability of Missile Generation in General Electric Nuclear Turbines,” described GE’s methodology for evaluating the probability of wheel missile generation for nuclear turbines manufactured by GE. The methodology includes consideration of the probability of unit overspeed, wheel materials, in-service inspection capabilities, and the potential for wheel containment by stationary turbine structures.

The GE analysis methodology considers two fundamental failure modes that can lead to missile generation: (1) failure of rotating components operation at or near normal operating speed (brittle failure) and (2) failure of components that control admission of stream to the turbine, resulting in destructive shaft rotating speed (ductile failure). The brittle and ductile failure modes are statistically independent.

The turbine originally supplied to TMI Unit 1 utilized built-up LP rotors with shrunk-on wheels and axial keyways. As a result of the Turbine Steam Path Retrofit (1R14), the built-up rotors were replaced with ones manufactured from monoblock forgings. Therefore, the missile analysis that applied to the original built-up rotors no longer applies. GE provided an assessment on the probability of generating a turbine missile for the new monoblock design.

a. Turbine Rotor Failure at or Near Operating Speed

Turbine and generator rotor failures at or near rated speed have resulted from the combination of severe strain concentrations in relatively brittle materials. The brittle fracture failure mechanism is due to the growth of keyway stress corrosion cracks to critical size. The probability of this failure mode is also dependent on speed, temperature, and material properties. However, due to the nature of stress corrosion cracking, the probability of the failure mechanism is also a function of inspection methods and intervals. For a shrunk-on wheel operated in the speed ranges considered by GE, the probability of bursting, and thus of missile generation is dominated by this fracture mechanism.

Stress corrosion cracks began to be detected in the early 1980s at the wheel bore region of conventional rotors with shrunk-on wheels and axial keyways. High stresses in the wheel keyway and the potential for stress corrosion cracking due to a condensation mechanism which resulted in an enrichment of oxygen levels in the location of the highly stressed keyway, allowed for the possibility of a brittle wheel burst. Although no GE nuclear unit had experienced a wheel burst or missile incident, GE developed an in-service testing capability and a probabilistic missile analysis methodology which was submitted to the NRC. Since approval of the GE methodology by the NRC in 1986 (reference NUREG-1048, appendix-U), wheel inspection intervals for turbines with shrunk-on wheels have routinely been established short enough to keep the probability of wheel burst at or near normal operating speeds within acceptable limits.

The conditions necessary to support brittle failure mechanism, high stress levels, and keyway stress corrosion cracking mechanism, are eliminated by installation of the monoblock rotors. Therefore, the probability of missile generation at normal operating speeds is negligible and no longer a concern.

b. Turbine Generator Overspeed Failure

Improvements in rotor quality reduce the chance of failures at operating speed but tend to increase the hazard level associated with unlimited overspeed because of the greater missile energy associated with higher bursting speed. Thus, the turbine overspeed protection systems are as follows:

1) Main and secondary steam inlets have the following valves in series:

   a) Control valves - controlled by the speed governor and tripped closed by emergency governor and backup overspeed trip.

   b) Stop valves or trip throttle valve - actuated by the emergency governor and backup overspeed trip. Emergency main stop valves of the stem-sealed design have been used on General Electric steam turbines 10,000 kW and larger since 1948. There have been over 650 turbines shipped and put in service during this period and there has been no report of the main steam stop valve failing to close when required to protect the turbine. Impending sticking has been disclosed by means of a full-closed test feature so that a planned shutdown could be made to make the necessary correction. This almost always involves the removal of the oxide layer which builds up on the stem and bushing and which would not occur on a low temperature nuclear application.

   c) Combined stop and intercept valves in cross-around systems where required to control overspeed to the values mentioned above. These are actuated by the speed governor and emergency and backup overspeed trips. These valves also include the testing features described above.

2) Uncontrolled Extraction Lines to Feedwater Heaters

   If the energy stored in an uncontrolled extraction line is sufficient to cause dangerous overspeed, then positive closing non-return valves are provided, to be actuated by the emergency governor and backup overspeed trip. These are designed for remote manual periodic tests to ensure proper operation. The unit piping, heater, and check valve system were reviewed during the design stages to make sure the entrained steam cannot overspeed the unit beyond safe limits.

   Special field tests have been made of new components both to obtain design information and to confirm proper operation. Such tests include the capability of controls to prevent excessive overspeed on loss of load.

   Careful analysis of past failures has led to design, inspection, and testing procedures to substantially eliminate destruction overspeed as a possible cause of failure in modern design units.

To keep the probability of a significant overspeed event very low, periodic maintenance and inspection of valves and other overspeed protection components are performed.

The probability of ductile failure is considered a function of speed, temperature, and material tensile strength. With stress below unlimited strength, the probability of a ductile failure is negligible. The GE probabilistic analysis of turbine overspeed was also documented in the 1984 NRC report, and is applicable to units with LP monoblock rotors. The overspeed analysis considers the characteristics of the turbine control system, the unit configuration, and test requirements for the steam valves and other overspeed protection devices. This overspeed analysis showed that the probability of attaining a given overspeed decreases rapidly as the overspeed increases. As long as the control system is maintained in accordance with GE's recommendations, the annual probability of attaining an overspeed of 120% or greater is in the range of 10^{-7} to 10^{-8}. Thus, in the case of a turbine with monoblock rotors, such as TMI Unit 1 following completion of the Turbine Steam Path Retrofit project, protection against missile generation can be shown by avoiding the potential for ductile failure at any operating speed below 120%.

GE, in the course of designing the new turbine for TMI Unit 1, evaluated tensile stresses in every rotating component. All of the rotating components have sufficient margin to tensile strength (at design component temperatures) to support operating speeds well in excess of 120% of normal. For example, the overspeed capability of the un-bucketed HP and LP rotors is over 200%. The limiting components, per design, for the bucketed rotors are the L-O buckets which have overspeed capability of 170%.

Therefore, the GE design ensures that the generation of turbine missiles as the result of ductile failure remains well below the maximum annual probability threshold established by the NRC for TMI Unit 1.

5.4.3.2.5 <u>Tornado Missiles</u>

Equipment within the Auxiliary Building vital to the safe shutdown of the station is included in the tornado protection criteria.

There is no equipment located in the Turbine Building that is required for safe shutdown of the plant. The emergency feedwater pumps are located in the Intermediate Building and have been protected by a structure designed for the tornado loads as stated in Section 5.4.3.2.2.

The Condensate Line from Condensate Storage Tank CO-T-1A to the Intermediate Building runs underneath the turbine building floor and is encased in a trench. Vent valve CO-V-135 for this line is located in the trench under a protective plate. Per Section 10.6.1c, the Condensate Storage Tanks and Piping outside of the Intermediate Building are redundant and do not need to be tornado missile protected.

5.4.4 PIPING DESIGN CRITERIA

5.4.4.1 <u>Seismic Class I, II and III Piping System Design</u>

The piping design criteria for the primary loop piping and related pipe lines is discussed in Chapter 4. The following design criteria cover the other pipe lines which are not described in Chapter 4.

Nuclear class piping systems have been fabricated, erected, and inspected with the intent of satisfying the applicable sections of the Code for Nuclear Power Piping, USAS B31.7 February 1968 Draft with Errata through June 1968 (issued for trial use and comment).

The basic guideline for the design of piping has been the Code for Pressure Piping, USAS B31.1.O-1967, and those portions of Code Cases N7, N9, and N10 that pertain to design criteria. Pertinent sections from this Code apply to Class I and II piping described hereafter:

a. Code Paragraph 101.5.3, Earthquake

"The effect of earthquakes, where applicable, shall be considered in the design of piping, piping supports, and restraints, using data for the site as a guide in assessing the forces involved. However, earthquake need not be considered as acting concurrently with wind."

b. Code Paragraph 101.5.4, Vibration

"Piping shall be arranged and supported with consideration of vibration."

c. Code Paragraph 121.2.5, Sway Braces

"Sway braces or vibration dampeners shall be used to control the movement of piping due to vibration."

d. Code Paragraph 102.3.2 (d), Additive Stresses

"The sum of the longitudinal stresses due to pressure, weight, and other sustained loads shall not exceed the allowable stress in the hot condition S_h. Where the sum of these stresses is less than S_h, the difference between S_h and this sum may be added to the term 0.25 S_h in Equation (1) for determining the allowable stresses range S_A.

Eq. (1)

$$S_A = f\,(1.25\,S_c + 0.25\,S_h)$$

Where:

S_c = basic material allowable stress at minimum (cold) temperature from the Allowable Stress Table.

S_h = basic material allowable stress at maximum (hot) temperature from the Allowable Stress Tables.

f = stress range reduction factor for cyclic conditions for total number N of full temperature cycles over total number of years during which system is expected to be in operation.

The longitudinal pressure stress S_{lp} shall be determined by dividing the end force due to internal pressure,

$$F = \frac{p\,\pi d^2}{4}$$

by the cross-sectional area of the pipe wall,

$$A = \frac{\pi}{4}(D_0{}^2 - d^2)$$

or:

$$S_{lp} = \frac{F}{A} = pd^2/(D_0{}^2 - d^2)$$

Where:

S_{lp} = longitudinal pressure stress, psi

p = internal design pressure, psi

d = nominal inside diameter of the pipe, inches.

D_0 = nominal outside diameter of the pipe, inches.

e. Code Paragraph 102.3.3, Limits of Calculated Stresses Due to Occasional Loads

During Operation

The sum of the longitudinal stresses produced by internal pressure, live and dead loads and those produced by occasional loads such as the temporary supporting of extra weight may exceed the allowable stress values given in the Allowable Stress Tables by the amounts and durations of time given in Paragraph 102.2.4 (as follows):

Ratings: Allowance for Variations From Normal Operation

It is recognized that variations in pressure and temperature inevitably occur, and therefore the piping system shall be considered safe for occasional operation for short periods at higher than the design pressure or temperature.

Either pressure or temperature, or both, may exceed the design values if the stress in the pipe wall calculated by the formulas using the maximum expected pressure during the variation does not exceed the S-value allowable for the maximum expected temperature during the variation by more than the following allowances for the periods of duration indicated:

1) Up to 15 percent increase above the S-value during 10 percent of the operating period.

2) Up to 20 percent increase above the S-value during 1 percent of the operating period.

Major portions of Classes I and II piping systems as herein described have been checked by thermal, dynamic, (seismic), dead load and pressure stress analysis. For those pipe lines operating at elevated temperatures thermal analyses have been made. For those pipe lines which fall within the seismic criteria, dynamic analyses have been made as herein described giving due attention to designed supports and restraints. In addition, due recognition has been taken of stresses caused by internal pressure and the dead weight load of piping.

For the critical piping systems, the effect of differential movement between floors at different elevations, the in phase and out of phase displacements at different elevations between equipment, piping, and buildings has been evaluated, and the increase in stresses was found to be insignificant.

The results of these various analyses have been compared with allowable stress values for the various limiting conditions as follows:

a. Piping stress for plant operation during the Operating Basis Earthquake (OBE) shall not exceed 1.2 x S_h (USAS B31.1.0 Code Tables A-1 and A-2 and Code Cases N7, N9 and N10).

b. The allowable piping stress for plant operation during the maximum hypothetical earthquake shall not exceed 1.8 x S_h (USAS B31.1.0 Code Tables A-1 and A-2 and Code Cases N7, N9 and N10); but not more than 1.5 S_y where S_y is the material yield strength at operating temperature.

c. Thermal stresses shall not exceed S_A as defined by Code Paragraph 102.3.2.

d. Pressure stresses shall be calculated in accordance with Code Paragraph 102.3.2.

e. Maximum estimated dead weight load stresses shall be calculated in accordance with conventional beam formulas.

The stresses for each section of piping have been evaluated and combined to ascertain the total maximum stress value. Giving due consideration to the pipe line conditions, this totalized stress does not exceed the Code allowable stress. This evaluation checks the piping design for the condition of operating load plus earthquake load.

The most severe load condition imposed on the piping is the effect caused by possible pipe rupture. To assure that this type of accident is contained, main steam, main feedwater, and sections of emergency feedwater piping within the Reactor Building are constrained from excessive deflection caused by this effect by a series of structural rings. Criteria used for the design of these assume instantaneous reaction from emission of pipe contents to the containment atmosphere.

In analyzing the forces acting on the pipe due to a steam or feedwater pipe break, the following conservative assumptions are made:

a. The maximum break size is equal to the cross-sectional area of the pipe.

b. The break is assumed to be an ideal nozzle.

c. Critical mass flow rate occurs.

The net energy developed from pipe travel and collision loss has been absorbed by the pipe whip restraints and designed to yield and form plastic hinges.

Piping not falling within Classes I and II has been considered conventional (Class III) and has been designed and checked by adherence to the Code for Pressure Piping, USAS B31.1.0 1967, without consideration of seismic effects.

The piping, support, and restraint systems have been designed in accordance with good design practice which has been previously proven on similar type systems. Piping systems will be observed during the initial operation of the unit to determine if any unusual operating characteristics are present such as water hammer or excessive vibration. Particular emphasis will be placed on high energy dissipation processes such as pump recirculation, heater drains, main steam dump to the condenser, high pressure sampling systems, etc. Conditions considered to be detrimental to reliable unit operation will be corrected as required.

Planned observations of piping supports during preoperational and startup tests of newly installed systems:

1) Prior to functional testing of systems, the equipment and piping hangers and supports are inspected. These inspections will include the following:

   a. Assure that hangers and supports are located in the proper locations as indicated on the appropriate drawings.

   b. Assure that hangers are carrying load and permit expected pipe motion during heatup and operation.

   c. Assure that nuts on hanger rods are fitted with jam nuts which are tightened.

   d. Assure that spring-type hangers are correctly adjusted for cold conditions.

   e. Assure that seismic restraints are located as indicated on the appropriate drawings and should not interfere with expected pipe motion during heatup and operation.

2) During the cold functional testing of systems, station personnel and/or assigned test personnel observe system piping and equipment for vibration and pipe motion upon initiation of system operation and record vibration data for rotating equipment and report any observed vibration or motion of piping and equipment.

3) During system heatup, hot functional testing, initial system, hot operation system, plant personnel, and/or assigned test personnel inspect and observe systems for the following:

   a. Assure that hangers and supports do not interfere with system equipment and piping expansion.

   b. Assure that seismic restraints do not interfere with equipment and piping expansion.

   c. Observe spring-type hangers for proper hot loading.

   d. Observe system equipment and piping for vibration and report observations to the appropriate responsible person.

During startup testing, as power level and thus, steam, feedwater, and heater drain flows increase, system is inspected and observed for equipment and piping vibration.

During surveillance testing of systems or equipment, station personnel observe system equipment and piping for motion upon initiation of the system and for vibration after system initiation, report their findings to the Control Room Supervisor.

During plant operation the nuclear plant auxiliary operators observe equipment and piping of operating systems during their plant shift tours, reporting any vibrations noted to the Control Room Supervisor on duty.

Inspections of equipment and piping supports, hangers, and seismic restraints, will be performed in accordance with the ASME Section XI, to assure that they are functioning properly and that spring-type hangers are assuming the proper loading. Maintenance procedures require that hangers or supports which require blocking or removal during the specific maintenance operation are returned to the proper condition upon completion of the job.

5.4.4.2 Auxiliary Steam System Piping Design

The auxiliary steam system does not perform any safety related function and is not required for normal plant operation or for safe plant shutdown. However, the portions of the piping located in the Auxiliary, Fuel Handling and Control Buildings must maintain pressure boundary to prevent potential damage to safety related components and systems located within these structures. Therefore, these portions of the auxiliary steam system piping were analyzed for combined operating and seismic conditions to ensure that a rupture of the pipe will not occur. The pipe is designed for normal operating and occasional loads in accordance with USAS B31.1.0 code and, in addition, the portions described above are analyzed for seismic loads in accordance with the conservative deterministic failure margin (CDFM) methodology described in EPRI NP-6041-SL (Reference 73). Details of the methodology and the results of the CDFM analysis are contained in report 240046-R-001 (Reference 74). Important aspects of the CDFM methodology as applied to the analysis of the auxiliary steam system piping are summarized below:

a) The CDFM methodology consists of an explicit analysis of portions of the subject piping system combined with a detailed walkdown of the entire system to identify potential vulnerabilities.

b) The seismic inputs to the analysis using the CDFM methodology are obtained from 5% damped amplified response spectra in place of the damping specified for Seismic Class I piping analysis. The analysis includes loads resulting from operating conditions and the SSE.

c) The amplified in-structure response spectra used for the evaluation of the auxiliary steam piping is the spectra developed for the resolution of USI A-46 as described in EQE report 50097-R-001 (Reference 68). These spectra are more conservative than the median centered amplified response spectra permitted by the CDFM approach.

d) The CDFM approach uses ASME service level D allowables (e.g. 3.0 Sh) for evaluating pipe and pipe components for load combinations that include the SSE. Ultimate strength values are used for the evaluation of support members and steel and concrete structures for SSE conditions. The 20% reduction in calculated stresses permitted by the CDFM methodology to account for inelastic energy absorption is not utilized.

e) Expansion anchor bolt allowables are based on a factor of safety equal to three on ultimate capacity.

5.4.5 METHOD OF ANALYSIS

The vertical and horizontal seismic components are assumed to occur simultaneously, and for systems designed to the original basis the absolute values of the seismic components were added. Seismic evaluations of existing, new and replacement structures, systems and components may be performed by combining the responses due to each of the three directional components of the earthquake using the square root of the sum of the squares (SRSS) approach as described in USNRC Regulatory Guide 1.92 (Reference 71). For piping systems which were reanalyzed for USNRC I.E. Bulletin 79-14 the seismic components were combined per USNRC Reg. Guide 1.92 (Reference 64). Table 5.4-1 presents a comprehensive list of Class I structures, systems, and components, their applicable design criteria, and how they were analyzed for seismic loading. Seismic experience data methods (SQUG) may be used on a case-by-case basis as described in Section 5.1.2.1.2.c.

For Class I structures, components, and systems, the method of analysis used was either a modal analysis wherein modal shapes, frequencies, stresses, and participation factors are determined, or were developed as follows:

a. The natural period of vibration of the structure, component, or system was determined.

b. The response acceleration of the component to the seismic motion was taken from the response spectrum curve at the appropriate natural period.

c. Stresses and deflections resulting from the combined influence of the normal loads and the additional load from the design earthquake (OBE) were calculated and checked against the limits imposed by the design standard or code.

d. Stresses and deflections resulting from the combined influence of the normal loads and the additional loads from the maximum hypothetical earthquake (SSE) were calculated and checked to verify that deflections do not prevent functioning and that stresses do not produce rupture or excessive distortion.

e. The dynamic analysis of critical piping systems (i.e., Class I systems) is a modal analysis based upon either a distributed or lumped mass solution depending upon the complexity of the system. Piping configurations of the larger sized pipe lines classified as SI and SII were analyzed using the lumped mass model method briefly described in Section 5.4.5.1. A general spacing criterion for use by the field to locate seismic restraints for small field run piping was developed using a distributed mass model. In general, this small piping is cold (less than 150°F) and the factor of thermal expansion requires a minimum of consideration.

f. For the critical piping systems, the effects of differential movement between floors at elevations have been evaluated both for horizontal and vertical movements. The increase in stresses due to differential movement was found to be insignificant.

The vertical seismic input at the ground is assumed to be two thirds of horizontal input. For equipment and piping systems anchored to the walls and on floors close to the walls, no amplification due to the floors is assumed. For those anchored at the middle of the floors, the amplification due to the vibrations of the floors would be investigated. Floors with fundamental frequencies 30 cps or up would be considered to be rigid, and no amplification is assumed to apply to equipment on rigid floors.

In case the piping is anchored to a piece of flexible equipment, the equipment should be modeled as a part of the piping. In case the mass of the equipment is very large in comparison with that of the piping one may exclude the equipment from the model, but the response curve at the point of attachment to the equipment should be used instead of the floor response curve.

5.4.5.1 Piping Analysis

Lumped-Mass Analysis

The system is represented by a series of concentrated masses. First, the space coordinates are established for the system, and coordinates of mass points are determined. Using a static analysis, flexibility matrices corresponding to these mass points are computed. Next, the equations of motion are written in matrix form. Responses of modes with frequencies 10 percent within each other are added absolutely. Then the square root of the sum of the squares approach is taken for the rest of the modes. Since the absolute addition of the close modes method was incorporated into the computer program after certain portions of the pipe system were run, a general check of these systems was made. The effect of absolute addition increases the response less than 12 percent. Hence, for conservative reason, the response of those systems without absolute addition are multiplied by a factor of 1.12 to account for the effect. A more complete description of the analytical methods employed is contained in Reference 45. The application of the amplification curves cited in Reference 45 is explained in Reference 46. In Reference 46, the single degree-of-freedom model was not used to compare with the multi mass system directly. Instead, it was used to compare with one of the normal modes of the multi-mass system. This is the standard procedure in the method of normal mode. Even with the time history approach the same comparison is used, if the normal mode method is employed. An approximate comparison of our floor response curves with these derived from time history method (References 47 and 48) indicated that the method of Reference 46 is no less conservative than the time history method. However, this fact was recently verified by analyzing the lumped mass model of the Reactor Building interior concrete as shown on Figure 5.4-3 for:

a. El Centro normalized time history method

b. Golden Gate normalized time history method

The resulting unsmoothed ground response spectra are shown on Figure 5.4-4 and compared with the Three Mile Island ground response spectrum. It is interesting to note that the Golden Gate spectra has a low response at the period of the fundamental mode of the building (0.09 sec). This in turn results in the relative large difference in the floor response spectra as shown on Figures 5.4-5, 5.4-6, 5.4-7, 5.4-8, and 5.4-9. The dotted curve shows the floor response using Reference 46 method with the unsmoothed Golden Gate response spectrum as shown on Figure 5.4-4. System and component masses less than 10 percent of the structure mass were not included in the dynamic analysis of the structure.

In addition to the earthquake response for the pipe system, the models described above were used to determine forces and moments with resulting stresses for any transient or permanent displacements which were induced at the support point.

Stresses arising from the combination of each horizontal excitation with the vertical excitation are compared, in order to arrive at the maximum values.

The interaction of the piping with the structure is taken into account by the floor response curve as described in Reference 49.

The model for the buried piping is described as a semi-infinite beam on elastic foundation. The transient and, if applicable, permanent displacements of the structure can be determined. The displacements of the structures are determined independently of the piping system. The displacements are applied to the buried pipe at its point of attachment to the structures, in order to calculate maximum bending moments and shear forces.

This technique for analyzing seismic stresses in buried pipes is valid only if the soil surrounding the pipe experiences uniform motion due to the earthquake excitation. This requires the use of a homogeneous backfill material for the complete length of the pipe. This was provided.

The concrete pipes have been connected to the structures with a bell and spigot connection. Steel pipes have been rigidly anchored to the structures resulting in stresses up to 8000 psi during maximum hypothetical earthquake.

Technical examples of seismic Class I piping system stress analysis results are presented on Figure 5.4-10.

5.4.5.2 Other Missiles

See Section 5.2.4.2.

5.4.5.3 Empirical Test

Static or dynamic analyses of certain equipment, or system components, would be extraordinarily difficult and of questionable value in determining whether the equipment could meet Class I requirements. Therefore, in some cases, testing under simulated seismic conditions was substituted in lieu of an analysis.

For example, a battery charger was mounted on a test machine table. The table was oriented at an angle to the horizontal such that the charger could be subjected to a given horizontal acceleration with two thirds of that acceleration acting vertically. The input acceleration corresponded to the predicted structure acceleration at the equipment location. Structure accelerations were determined by dynamic analysis based on the response spectrum curve with 5 percent damping for concrete structures above ground. The charger was subjected to shocks in the front-to-back and side-to-side directions. equipment functioned satisfactorily during and after the test with no visible damage.

5.4.5.4 Pipe Whip Restraint Criteria

For a Main Steam or Feedwater Rupture inside containment:

a. The containment integrity and leaktightness shall be preserved.

b. Broken steam or feedwater pipe shall not damage the other steam generator and its piping.

c. Broken steam or feedwater pipe shall not jeopardize the Reactor Coolant System nor impair the function of the emergency cooling system.

5.4.5.5 Pipe Whip Forces

The force generated by a fluid escaping from a ruptured pipe consists of two components:

a. The "rocket effect" or thrust due to the change in momentum of the fluid, equal to the product of the exit velocity and the mass flow rate.

b. The exit plane pressure force, equal to the product of the exit plane pressure and the exit plane cross-sectional area.

In calculating the force exerted by a ruptured main steam line, the following conservative assumptions are made:

a. Ruptured pipe forms a perfect nozzle, i.e., entropy change is equal to zero.

b. Area of the break is equal to the cross-sectional area of the pipe.

c. Critical flow occurs at the break.

d. Velocity of the fluid approaching the break is zero, i.e., maximum momentum change occurs.

The approach is basically that advocated by Reference 50. A sample calculation for a main steam line break follows:

Initial Conditions

P_1 = 1104 psia (highest set safety valve blowing)

T_1 = 575F

H_1 = 1212 Btu/lb

S_1 = 1.40

V_1 = 0

Exit Plane Conditions

P_2 = 643 psia

H_2 = 1165 Btu/lb

Critical pressure ratio - 0.584, based on a K = 1.13

From Reference 24, Figure 14, W = 49.4 lb/hr - in^2 - psi (flow rate)

From Figure 12, Reference 24, velocity equals 1468 ft/sec.

Force = (mass flow rate) (velocity) + (pressure) (area):

F =

$$\frac{(49.3\ \text{lb/hr - in}^2\text{- psi})\ (382.3\ \text{in}^2)\ (1104\ \text{psi})\ (1468\ \text{ft/sec})}{(3600\ \text{sec/hr})\ (32\ \text{lbmft/lbfsec}^2)}$$

$$+\ (643\ \text{psia} - 14.7)\ \text{psig}\ (382.3\ \text{in}^2) = 506{,}000\ \text{lb}$$

The above figure was used for main steam restraint design and, from the basic assumptions made, is quite conservative. In Reference 50, the author takes credit for the appreciable pressure drop occurring in the pipe approaching the break. As the break may occur very near or far from a steam generator, no attempt was made to consider pressure losses approaching the break. Forces resulting from a ruptured feedwater line are determined in a similar manner.

5.4.5.6 Pipe Whip Restraint Structural System

The main steam and feedwater lines are postulated to rupture either by a side split or a guillotine break. The rupture force accelerates the pipe through a small distance until it strikes a restraint collar which is anchored to the floor beam and girder system absorbing the energy imparted to it which then transfers the load through the structural system to the foundation mat.

The design involves the following aspects:

a. Spacing of the Pipe Whip Restraint Collars

Circular steel collars are located and anchored to the floor beams or directly to the secondary shield wall. An allowance is incorporated into the inside diameter of the collar so that the pipe is permitted unrestricted freedom during plant operation. These collar restraints do not provide any seismic, thermal, or dead load support for the pipe. Their sole function is that of a pipe rupture restraint. The maximum spacing of the restraints is determined by the theory of plastic mechanism.

b. Collar Design

The design of the collar is similarly treated as a plastic mechanism.

c. The Floor Framing Design

The steel framing systems are designed to resist the pipe thrust load by developing plastic mechanisms. By use of diaphragm beams, the load is distributed into the structure. Shear studs are designed to withstand horizontal forces and to activate sufficient mass to reduce the effect of collision. In addition, the columns are designed to prevent uplift resulting from the rupture force. Horizontal loads are taken down into the mat by bracing and by the secondary shield wall.

5.4.5.7 Method of Analysis of Rupture Restraint Support System

The efforts of engineering research are not presently capable of applying rigorous methods of analysis to the design of actual structures to withstand the forces generated by pipe rupture and collision. However, certain basic assumptions have been made using the conservation of energy and momentum theories, thereby generating the design parameters required for the design of the structural system.

Associated with pipe whip restraints are several phenomena:

a. The whipping pipe can be described as a complex elasto-plastic multi-degree of freedom motion. However, Reference 51 proposes that a single degree of freedom system, considering the response as elasto-plastic or rigid plastic will yield reasonably good results.

b. The collision of the effective mass of the pipe, which has been given an acceleration due to the steam or feedwater rupture force, with the effective mass of the supporting system, produces an impulse loading. This impulse load is determined by the kinetic energy imparted to the struck body as suggested by Reference 52 by postulating a plastic collision and making use of the conservation of momentum and energy theorems (Reference 53). The effective mass of beam elements are determined by Rayleigh's Principle (Reference 52).

c. In addition to the above impingement, the steam or feedwater thrust is conservatively assumed to act on the supporting structural element as a suddenly applied constant load of infinite duration.

Essentially, the method used can be described as a single degree of freedom system excited by a suddenly applied constant load of infinite duration together with an initial condition of velocity applied at the time of collision with a responding rigid plastic material.

The cases of maximum impinging velocity and constant force for side split and guillotine ruptures are investigated to determine the capability of the supporting structure to absorb the energy imparted to it. The allowable limit strain in the beams is incipient strain hardening of the compression-tension stress-strain curve, this being the general limit of unsupported beam flanges as shown on Figure 14 of Reference 54. This results in an allowable ductility factor of 12 for ASTM A36 steel. Therefore, a design utilizing plastic mechanisms to resist the rupture thrust force will give satisfactory results for the structural system.

When a shear mode of response governs the situation, Hall and Newmark (Reference 55) have shown that the ductility factor for compact rugged structural shapes by far exceeds 12.

Table 5.4-1
(Sheet 1 of 6)

LIST OF CLASS I
STRUCTURES, SYSTEMS, AND COMPONENTS

Structures	Design Code	Seismic Design
Reactor Building	FSAR 5.2.2, 5.2.3.1	Dynamic
Auxiliary Building	FSAR 5.4.3.1	Dynamic FSAR 5.4.5
Fuel Handling Building	FSAR 5.4.3.1	Dynamic FSAR 5.4.5
Control Building	FSAR 5.4.3.1	Dynamic FSAR 5.4.5
Diesel Generator Building	FSAR 5.4.3.1	Static
Intermediate Building	FSAR 5.4.3.1	Dynamic FSAR 5.4.5
Intake screen and pump house	FSAR 5.4.3.1	Dynamic FSAR 5.4.5
Access tunnel vault to Auxiliary Building	FSAR 5.4.3.1	Aircraft Impact Controls
Air intake structure (below ground)	FSAR 5.4.3.1	Aircraft Impact Controls
Chemical Cleaning Building Basin	FSAR 5.4.3.1	Static
Engineered Safeguards Systems		
Process and instrument piping	FSAR 6.1.2.4	Dynamic, Static
Valves	FSAR 6.1.2.4	Static
Reactor Building Spray System		
Piping	FSAR 6.2.2.4	Dynamic
Valves	FSAR 6.2.2.4	Static
Reactor emergency air cooling units	SAR 6.3.2 Table 6.3-2	Static
Ducts and supports	AISC	Static

Table 5.4-1
(Sheet 2 of 6)
LIST OF CLASS I
STRUCTURES, SYSTEMS, AND COMPONENTS

Structures	Design Code	Seismic Design
Hydrogen monitoring system	FSAR 6.5.2	Dynamic
Reactor Protection System	-	Shake Test
Engineered Safeguards Actuation System	-	Shake Test
All piping penetrations and associated isolation valves	FSAR 5.2.2.4	Dynamic - Piping Static - Valves
Vital Cooling Water System		
Decay Heat Removal	FSAR 5.4.4	Dynamic
Decay heat closed cycle	FSAR 5.4.4	Dynamic
Decay heat river water - steel	FSAR 5.4.4	Dynamic
Decay heat river water - concrete	A.W.W.A	Dynamic
Nuclear services closed cycle	FSAR 5.4.4	Dynamic
Nuclear services river water - steel	FSAR 5.4.4.1	Dynamic
Nuclear services river water – concrete	A.W.W.A	Dynamic
Reactor Building emergency cooling - steel	FSAR 5.4.4	Dynamic
Reactor Building emergency cooling - concrete	A.W.W.A	Dynamic

Table 5.4-1
(Sheet 3 of 6)

LIST OF CLASS I
STRUCTURES, SYSTEMS, AND COMPONENTS

Structures	Design Code	Seismic Design
River water tail pipe – concrete	A.W.W.A	Dynamic
Emergency Power Supply System		
Diesel generators	DEMA, IEEE, ISA, ASME	Static
Fuel oil storage tanks	-	Static
dc power supply system	USASI, NEMA, ASME	Static, Shake Test
Inverters	USASI, NEMA, ASME	Shake Test
Power distribution lines to equipment required for emergency	AISC, IPCEA	Static
Switchgear and power centers supplying the engineered safety features	NEMA, USASI, IEEE	Shake Test
Control console	-	Static

Table 5.4-1
(Sheet 4 of 6)

LIST OF CLASS I
STRUCTURES, SYSTEMS, AND COMPONENTS

Structures	Design Code	Seismic Design
Motor control center	NEMA, USASI, IEEE	Static
Spent Fuel Cooling System		
Piping	FSAR Table 9.4-1	Dynamic
Valves	USAS B31.1.0	Static
Penetration Pressurization System		
Process piping	-	Static
Valves	USAS B31.1.0 USAS B16.5	Static
Flow meters	Mil S -901C	Shake Test
Purge interspace tanks	ASME III-C and ASME VIII	Static
Fluid Block System		
Fluid block tanks	ASME III-C	Static
Process piping	FSAR Table 9.4-1	Dynamic
Valves	USAS B31.1.0	Static
Vital Ventilating Systems	-	Static
Emergency Feedwater System		
Pumps	-	Static
Condensate storage tanks	-	Static

Table 5.4-1
(Sheet 5 of 6)

LIST OF CLASS I
STRUCTURES, SYSTEMS, AND COMPONENTS

Structures	Design Code	Seismic Design
Piping	FSAR 5.4.4	Dynamic
Valves	FSAR 5.4.4	Static
Underground diesel fuel tank	-	Static
Instrument air system	-	Static
Nitrogen manifold	USAS B31.1.0 CGA Standards	Static
Reactor Building polar crane	AISC EOCI-61	Static
Fuel Handling Building crane	AISC EOCI-61	Static
River pump service crane	AISC EOCI-61	Static
Waste Disposal System		
Reactor coolant bleed tanks	ASME III-C	Static
Miscellaneous water storage tank	ASME III-C	Static
Spent resin storage tank	ASME III-C	Static
Used filter precoat tank	ASME III-C	Static
Concen. radioactive waste storage tank	ASME III-C	Static

Table 5.4-1
(Sheet 6 of 6)

LIST OF CLASS I
STRUCTURES, SYSTEMS, AND COMPONENTS

Structures	Design Code	Seismic Design
Reclaimed boric acid tanks	ASME III-C	Static
Neutralizer tank	ASME III-C	Static
Neutralized waste storage tank	ASME III-C	Static
Laundry waste storage tank	ASME III-C	Static
Reactor coolant drain tank cooler	ASME III-C, ASME VIII	Static
Auxiliary Building sump pumps	None	Static
Sampling System		
Piping and valves	FSAR 5.4.4	Dynamic

5.5 QUALITY CONTROL

5.5.1 GENERAL

The organization, responsibilities, and general provision for quality assurance are described in Chapter 1. The specific quality control provisions that have been imposed by Specification requirements for the Containment System are described herein. Applicable quality control and quality assurance procedures are outlined in Chapter 1.

5.5.2 CONCRETE

The following quality control measures are followed for the structural concrete.

5.5.2.1 Preliminary Tests

a. The services of a testing laboratory were obtained prior to commencing concrete work. The testing laboratory made determinations of controlled mixes, using the materials proposed and consistencies suitable for the work, in order to determine the mix proportions necessary to produce concrete conforming to the type and strength requirements specified. Aggregates are tested in accordance with ASTM Specifications C 29-60, C 40-66, C 127-59, C 128-59, and C 139-63. Compression tests conform to ASTM Specifications C 39-64 and C 192-66.

b. The proportions for the concrete mixes are determined by Method 2 of Section 308 of ACI 301-66 and as herein specified.

5.5.2.2 Field Tests

During concrete operations, the testing laboratory had an inspector at the batch plant who certified the mixed proportions of each batch delivered to the site and sample and test periodically concrete ingredients. A concrete batch plant was utilized which complied in all respects, including provisions for storage and precision of measurements, with Reference 56. The testing agency maintained an inspector at the batch plant to ensure that the mix proportions complied with those for the design mixes with water content modified as required by measurements that were made of content of surface moisture on the aggregates. This inspector tested periodically the mix ingredients and ensured that a ticket was provided for each batch documenting the time loaded, actual proportions of the mix, amount of concrete, concrete design strength, destination as to portion of structure, identification of transit mixer, and reading of revolution counter at first addition of water.

The ready mixed concrete was mixed and transported in accordance with Reference 56. The minimum amount of mixing in truck mixers loaded to maximum capacity was 70 revolutions of the drum or blades after all of the ingredients, including water, were in the mixer. The maximum number of revolutions at mixing speed was 100. Records were maintained as to the time and reading of the revolution counter when concrete was discharged.

Other inspectors, at the construction site, inspected reinforcing and form placements, made slump tests, made test cylinders, checked air content, and recorded weather conditions. Except as noted hereinafter, test cylinders were molded, cured, capped, and tested in accordance with ACI 301-66 (Reference 2). For the Reactor Building shell, a set of six cylinders was made for each 50 yd^3 or fraction thereof placed in any 1 day. Two cylinders were tested at 7 days, two cylinders at 28 days, and the remaining cylinders at 90 days. Slumps tests were made at random with a minimum of one test for each 10 yd^3 of concrete placed. Slump tests were also made on the concrete batch used for test cylinders.

Requirements for placing and consolidating concrete have been as detailed in ACI 301-66 (Reference 2). Placing temperatures have been limited per the requirements for mass concrete.

5.5.2.3 Test Evaluation

The evaluation of test results was in accordance with Chapter 17 of ACI 301-66 (Reference 2). Sufficient tests were conducted to provide an evaluation of concrete strength.

5.5.2.4 Deficient Concrete

When tests of the laboratory cured cylinders failed to meet the requirements, the Contractor was required to:

a. Order changes to the proportions of the mix to increase the strength.

b. Order changes to improve procedures for protecting and curing the concrete.

c. Require additional tests in accordance with Reference 5.

5.5.3 REINFORCING STEEL

The Technical Specification for structural concrete includes the following quality control measures for reinforcing steel:

a. Testing and inspection of reinforcing steel was performed at the mill to ASTM requirements. The certified mill test reports were reviewed for each heat of steel covering chemical composition and specification requirements in mechanical properties. Each bar was branded in the deforming process to carry identification as to manufacturer, size, type, and yield strength.

b. Tests were performed on reinforcing steel by a testing laboratory to confirm compliance with physical requirements and verification of mill test results. The frequency of testing was two specimens taken from each heat of material in excess of 10 tons and within one heat of material, a series of tests for each 25 tons of steel. The tests determined yield, ultimate strength, and elongation. If test results did not meet specification requirements, further testing of that heat of material and an engineering investigation would have been performed.

c. Reinforcing bar material was kept separated by size and heat on the fabricator's yard. In addition, when loaded for mill shipment, the bars were properly separated by size and heat and tagged with the manufacturer's identification number.

5.5.4 CADWELD SPLICES

The Technical Specifications for structural concrete include the following quality control measures for cadweld splices:

a. Tension splice for bar sizes larger than No. 11 were made with a cadweld splice which developed the ultimate strength of the bar. To ensure the integrity of the cadweld splice, the quality control procedure provided for a random sampling of splices in the field. The selected splices were removed and tested to destruction.

b. A sampling of twenty splices was initially tested to destruction to develop an average (X) and standard deviation. The resulting average was 76,905 psi and standard deviation was 2842 psi. Sufficient samples were therefore tested to provide 99 percent confidence level that 95 percent of the splices develop 125 percent of the minimum guaranteed yield strength of the bar. The actual specification for the cadweld splices was developed on the basis of using the splice designed to develop the ultimate strength of the bar, or with the use of deformed bars conforming to ASTM 615-68, Grade 40 (Reference 58), a minimum tensile stress of 70,000 psi. The specification for random sampling was amplified to require that the average (X) equals or exceeds the minimum ultimate strength.

   The distribution established on this basis permitted the development of the lower limit. If the result of any test fell below this limit, the subsequent or previous splice was sampled. If this result was above the lower limit, the process was considered to be in control. If this result was again below the lower limit, the process average must have changed and an engineering investigation was made to determine the cause of the excess variation and reestablish control. The work of each splicing crew as individually sampled. The control limit based on the initial sample was 69,678 psi. Actual tensile strength of tested splices is plotted on Figure 5.5-1.

Prior to the production splicing of reinforcing bars, each operator or crew prepared and tested a joint for each bar size and position (i.e., vertical, horizontal, side entry) used in the production work. To qualify, the completed splices had to meet the following acceptance standards for workmanship:

a. Sound, nonporous filler material shall be visible at both ends of the splice sleeve and at the top hole in the center of the sleeve. Filler material is usually recessed 1/4 inch from the end of the sleeve due to the packing material, and is not considered a poor fill.

b. Splices which contain slag or porous metal in the riser, top hole, or at the ends of the sleeve shall be rejected. A single shrinkage bubble present below the riser is not detrimental and should be distinguished from the general porosity as described above.

c. There will be evidence of filler material between the sleeve and the bar for the full 360 degrees; however, the splice sleeves need not be exactly concentric or axially aligned with the bars.

d. The strength of the cadweld splice shall be equal to or greater than the specified minimum ultimate tensile strength of the bar.

A manufacturer's representative, experienced in cadweld splicing of reinforcing bars, was present at the jobsite at the outset of the work to demonstrate the equipment and techniques for making quality splices. He was also present for at least the first 25 production splices to observe and verify that the equipment was used correctly and that quality splices were obtained. The following quality control procedures were followed to ensure acceptable splices:

a. The splice sleeve, powder, and molds were stored in a clean dry area with adequate protection from the elements to prevent absorption of moisture.

b. Each splice sleeve was visually examined immediately prior to use to ensure the absence of rust and other foreign material on the inside surface.

c. The molds were preheated to drive off moisture at the beginning of each shift when the molds are cold or when a new mold is used.

d. Bar ends to be spliced were brushed to remove loose mill scale, rust, concrete, and other foreign material. Prior to brushing, water, grease, and paint were removed by heating the bar ends with a torch.

e. A permanent line was marked from the end of each bar for a reference point to confirm that the bar ends are properly centered in the splice sleeve.

f. Before the splice sleeve was placed into final position, the bar ends were examined to ensure that the surface was free from moisture. If moisture was present, the bar ends were heated until dry.

g. Special attention was given to maintaining the alignment of sleeve and guide tube to ensure a proper fill.

h. When the temperature was below freezing, the splice sleeve was preheated after the materials and equipment were in position.

i. Completed splices were visually inspected at both ends of the splice sleeve and at the top hole in the center of the splice.

5.5.4.1 <u>Rebar Splicing Performed as part of TMI-1 Steam Generator Replacement</u>

To support the 2009 SGR Project, a Containment Opening was made in the Reactor Building concrete wall and liner plate at the 290° azimuth, and directly above the existing Equipment Hatch to gain building access for rigging and handling of the steam generators. In order to restore the concrete opening, new rebar was spliced to existing No. 11 and No. 8 rebar using BarSplice BPI XI swaged couplers, which consisted of a cold-swaged sleeve installed using a portable hydraulic press and a pair of swaging dies. As an acceptable design equivalent to cadweld splicing of rebar, these mechanical splice devices complied with ASME Section III, Div. 2, Subsection CC to be capable of developing not less than 125% of the specified yield strength of the bars in question. For areas where mechanical splices could not be used, direct-butt fusion welded splices were employed. Welded splices were welded and inspected in accordance with AWS D1.4-1998, Structural Welding Code - Reinforcing Steel.

5.5.5 LINER PLATE

5.5.5.1 Materials

The liner plate was tested at the fabrication shop to meet those requirements enumerated in Section 5.2.2.4. ASTM standard test procedures were employed to ascertain compliance with ASTM specifications. Certified copies of mill test reports describing the chemical and physical properties of the steel were submitted to the user for approval. Tests for qualifying welding procedures and welders have been performed by the fabricator and monitored by the user. These tests have provided confirmation on weldability and weld ductility. Tests on nil ductility have been performed for materials used in the penetrations to the extent described in ASTM A 300. The user, or his authorized representative, has monitored shop test procedures at the fabrication shop and audited all records. The user has not duplicated tests performed by the steel supplier and the fabricator.

The non-destructive testing procedure which was employed on the thickened plates welded into the containment liner at the steam generator supports was as specified in Note 2 of Drawing 521017. Note 2 of Drawing 521017 also applies to the thickened plate welded into the containment liner at the polar crane brackets. Namely, the plates "shall be ultrasonically inspected to insure the steel is free from gross internal discontinuities such as pipes, ruptures, and laminations". The procedure and acceptance standards shall be in accordance with ASTM, A 435-65 (Reference 59).

5.5.5.2 Liner Welding And Non-Destructive Testing

The Technical Specification for the Reactor Building liner includes the following quality control measures for welding:

The qualification of welding procedures and welders has been in accordance with Section IX, "Welding Qualifications," of the ASME Boiler & Pressure Vessel Code. The repair of defective welds was made in accordance with Paragraph N-528 of the ASME Nuclear Vessels Code.

Meridional and circumferential welded joints within the main shell, the welded joint connecting the dome to the cylindrical side walls, and welded joints within the dome have been inspected by the liquid penetrent method and spot radiography. Penetrations, including the equipment access door and the personnel locks, have been examined in accordance with the requirements of the ASME Nuclear Vessels Code for Class "B" Vessels. Other shop-fabricated components, including the reinforcement about openings, were fully radiographed. Other joint details were examined by the liquid penetrant method.

Components built to Nuclear Class "B" requirements have been radiographed in accordance with the procedure and governed by the acceptability standards of Paragraph N-624 of the ASME Nuclear Vessels Code. Two percent of the welds have been spot radiographed, in accordance with the procedures and governed by the acceptability standards of Paragraph UW-52 of the ASME Unfired Pressure Vessels Code.

Twenty percent of the welds have been tested by liquid penetrant methods, in accordance with Appendix VIII of the ASME Unfired Pressure Vessel Code.

The crane bracket, bracing attachment, and anchor welds for the polar crane support have been examined by the liquid penetrant method with 100 percent coverage.

To control porosity, the following steps have been followed:

a. Thoroughly cleaning weld preparations

b. Cleaning slag from tack welds and previous passes

c. Controlling welding current and power input

d. Making certain welding materials and base material were dry before welding

e. Shielding welding arcs from winds and drafts

f. Controlling weave and whipping

Porosity where used in evaluating spot radiography and meets the required standards of Appendix IV of Section VIII of the ASME Nuclear Pressure Vessel Code.

The liner angle welds have been tested by the liquid penetrant method at the same frequency and to the same standards for the liner plate. Welding procedures have been in accordance with the ASME Boiler & Pressure Vessel Code, Section IX.

Quality control procedures and standards for field welding have been in accordance with the requirements of Section VIII of the ASME Boiler & Pressure Vessel Code.

Inspection of the liner seam welds was as follows:

a. One hundred percent visual inspection

b. Twenty percent liquid penetrant test

c. Two percent radiographic inspection for radiographable welds

d. One hundred percent vacuum box test or halogen leak test or test channels and seams covered by test channels.

Construction records were maintained at the jobsite. After completion of construction, the construction records were put into the possession of Metropolitan Edison Company for the life of the plant, except for those records required by code to be maintained by the Fabricator.

5.5.5.3 Preliminary Tests

The Technical Specification includes preliminary tests using test channels and a vacuum box as follows:

a. Test Channels

Weld seams on the base and pits and between penetration sleeves and liner plate and any additional weld seams that are covered by steel test channels were examined by detecting leaks at 63.3 psig. Embedded test connections bear metal stamped identification tags firmly affixed and referenced on drawings to test channel section to which they are attached. Other test connections were dimensioned on as-built drawings. Any field changes were subject to Engineer's approval and revised drawings issued.

b. Vacuum Box

Weld seams not covered with test channels were tested using soap film and vacuum box. The pressure differential was more than 4 psig. The rate of inspection did not exceed 2 feet of weld per minute. The box overlaps a minimum of 6 inch over the previously tested section. Detectable leaks were corrected.

5.5.6 PRESTRESSING TENDONS

The detailed procedure for quality control during fabrication, shipping, and erection was submitted by the tendon supplier (Ryerson) to the Construction Manager (UE&C) for approval. This procedure includes a detailed outline of the following:

a. Purchase of materials and mill test including chemical and physical properties and records

b. Handling, storage, and fabrication

c. Acceptance and rejection criteria and the fabrication tolerances

d. Corrosion protection during storage, fabrication, shipping, and erection

e. Shipping and erection

f. Installation sequences and stressing operations

g. Field tolerances

The approved quality control procedures detailed provisions to assure that each manufactured tendon shipped to the construction site was assigned an individual number and identified in such a manner that each such tendon could be identified at the construction site.

The Construction Manager audited the manufacturer's shop quality control program, assuring compliance with the quality control plan and receiving and material control, testing, dimensional checks, documentation, corrosion protection, coiling, and packaging for shipment, as described in the detailed procedures.

The coiled tendons were shipped in a handling frame. Each coiled tendon was banded in the shop during coiling and was enclosed in an approved protective cover. The handling frame was securely attached to the frame of the carrier.

Before unloading, each tendon carrier assembly was visually inspected. On evidence of damage to the assembly, the protective cover was removed and the tendon visually examined by the UE&C Field Quality Control Receiving Inspector (QCRI). Disposition of damaged tendons was documented.

During handling and erection, particular attention was given to protection from mechanical damage and corrosive elements.

The identification of tendons, anchor heads, and shims with mill test certificates and test reports were documented and forwarded to the Field Supervisor - Quality Control.

Fabrication and erection tolerances complied with the detailed procedures for fabrication, erection, and quality control as submitted by Ryerson. The allowable tolerance between the actual elongation and that indicated by the Jack-Dynamometer was as described in Section 5.2.2.3.4.

During fabrication, each tendon was protected with a coating of temporary corrosion inhibitor (Item e of Section 5.2.2.3.3) and enclosed in an approved container. The tendons were protected from damage during handling and shipping.

The conduit was checked for cleanliness and dryness prior to tendon installation. Conduit was capped at all times after the tendon was installed and capped with the permanent enclosure after the tendons were installed and tensioned.

The prestressing tendons for the containment were inspected by both the tendon supplier and quality control representatives.

The supplier's inspection consisted of the following:

a. Obtained certified mill test reports of chemical and physical properties of each reel of wire and submitted them. Material met requirements of ASTM A 421-65 Reference 60, Type BA.

b. Cut coupons from beginning and middle of each reel of wire, form buttonheads on the specimens, and tested them in tension to destruction. These tests ensured that the wire ruptured before failure of the buttonhead and that the wire met the physical requirements of ASTM A 421-65. Coupons and the reels they represented, not meeting the aforementioned requirements, were rejected. Records were maintained for each coupon test and for the tendons in which each coil of wire was used.

c. Cut wires within $\pm$ 3/32 inch of the specified length of less than 50 ft and within $\pm$ 1/8 inch if greater than 50 ft.

d. Checked each buttonhead by visual examination for general appearance and splits.

e. Inspected for discontinuities and imperfections produced on the bearing surface and fillet by chipped or damaged gripping dies.

f. Inspected for gross defects such as incompletely formed and double-upset buttonheads and buttonheads crushed by gripping dies.

g. Inspected for sharp indentations in the wires caused by the wire gripping dies.

h. Ten percent of the buttonheads, selected by random, were checked dimensionally. Diameter was not less than 0.372 inch nor more than 0.395 inch Head length was not less than 0.260 inch nor more than 0.300 inch. The buttonhead had a bearing surface on all sides.

The quality control representatives performed the following:

a. Submitted certified mill test reports to the Engineer, Gilbert Associates, Inc., for their review and comment.

b. Monitored the shop procedures and inspection by the supplier.

c. Inspected each tendon at the supplier's shop before shipment to ensure conformance to specifications and proper preparation for shipment.

d. Inspected for Discontinuities and imperfections produced on the bearing surface and fillet by chipped or damaged gripping dies.

e. Inspected for Gross defects such as incompletely formed and double-upset buttonheads and buttonheads crushed by gripping dies.

f. Inspected for Sharp indentations in the wires caused by the wire gripping dies.

g. Ten percent of the buttonheads, selected by random, were checked dimensionally. Diameter was not less than 0.372 inch nor more than 0.395 inch. Head length was not less than 0.260 inch nor more than 0.300 inch. The buttonhead had a bearing surface on all sides.

The quality control representatives performed the following:

a. Submitted certified mill test reports to the Engineer, Gilbert Associates, Inc., for their review and comment.

b. Monitored the shop procedures and inspection by the supplier.

c. Inspected each tendon at the supplier's shop before shipment to ensure conformance to specifications and proper preparation for shipment.

5.6 REACTOR BUILDING AIR HANDLING SYSTEMS

5.6.1 DESIGN BASES

These systems maintain suitable ambient conditions for personnel and equipment during normal operating conditions, and assist in reducing the effects of the postaccident conditions.

5.6.2 SYSTEM FUNCTION

In maintaining the general design basis, the systems have the following functions:

a. During normal operating conditions, recirculate air through cooling coils to maintain bulk air temperatures at or below 130°F above elevation 320 ft, and at or below 120°F below elevation 320 ft. During shutdown periods, maintain a minimum temperature of 60°F. System capacities for normal and postaccident conditions are indicated in Section 6.3.

b. During normal operating conditions, maintain airflow around the reactor vessel, seal ring, and nozzles so that the sustained concrete temperature adjacent to these items does not exceed 150°F, except for local areas, such as around a penetration, which shall not exceed 200°F. Higher temperatures than those given in the previous sentence may be allowed for concrete if tests are provided to evaluate the reduction in strength and this reduction is applied to design alllowables. Also, evidence shall be provided that verifies that the increased temperatures do not cause deterioration of the concrete either with or without load. The design basis is based on code from ACI 349-97, which was accepted by the NRC in Regulatory Guide 1.142. Thermocouples installed in the six reactor coolant system primary shield wall penetrations measure air temperatures at local points. Air temperature readings up to 300°F do not result in concrete strength loss based on ACI code 216.1-07 Fig. 2.12(a), "Compressive strength of siliceous aggregate concrete at high temperatures and after cooling," consequently, sustained air temperatures measured by these thermocouples shall not exceed 300°F.

c. During the integrated leak rate test, the air handling system maintains a constant temperature with minimum air temperature gradient throughout all areas and elevations.

d. The postaccident function of the air handling system is discussed in Section 6.3.

e. During Reactor Building purging, supply filtered (85 percent - NBS) and tempered air at a rate of 0-50,000 cfm depending on purge isolation valve position. When the reactor is in cold shutdown or refueling shutdown, continuous purging is permitted with the Reactor Building purge isolation valves opened fully. However, during all other plant conditions, the purge isolation valves shall be limited to a maximum of 30° open (90 degrees being fully open). This restricts air flow to less than 14,000 cfm. The purge air exhausted from the Reactor Building is filtered through roughing, HEPA, and charcoal filters and then discharged through the unit vent.

f. Filter recirculated air through prefilter, HEPA, charcoal absorber, and HEPA filter.

g. Reduce the airborne radioactivity level in the Reactor Building during normal operation.

h. Deleted

5.6.3 DESCRIPTION

The Reactor Building air handling systems are comprised of the following subsystems: Reactor Building Recirculation Cooling system, Reactor Compartment Cooling system, Operating Floor Ventilation system, Steam Generator Compartment Ventilation system, Reactor Building Purge And Vent system, Reactor Building Hydrogen Purge system, and Reactor Building Atmospheric Cleanup system.

5.6.3.1 Reactor Building Systems

The Reactor Building systems consist of:

a. Main containment cooling and filtering system, three units furnished, any two normally required.

b. Fans for steam generator compartments, two 100 percent capacity units furnished.

c. Fans for operating floor, two 50 percent capacity units furnished.

d. Reactor compartment system, two 100 percent capacity systems furnished, each including filter, cooling coil, and fan.

e. Unit heaters.

5.6.3.2 Reactor Building Purge Systems

The Reactor Building purge systems consist of:

a. Purge supply system, two 50 percent capacity units furnished, each including filter, heating coil, and fan.

b. Purge exhaust system including two 50 percent capacity fans and one 100 percent capacity filter plenum with roughing, HEPA, and charcoal filters.

c. The postaccident hydrogen purge system is designed to maintain the hydrogen concentration of the postaccident containment atmosphere below the lower flammability limit. This is accomplished by introducing fresh outside air into the Reactor Building and allowing the displaced containment atmosphere to be discharged in a controlled manner to the plant vent through the purge exhaust filters. Use of the hydrogen purge system is subject to evaluation of off-site dose.

5.6.3.3 Reactor Building Atmospheric Cleanup System

The purpose of the Reactor Building atmospheric cleanup system (kidney system) is to reduce the airborne radioactivity level in the Reactor Building during normal operation. The system performs no safety related function. The kidney system was completely installed in the Reactor Building during the first refueling shutdown. This kidney system is not safety related and therefore need meet only good standard industrial practices.

The kidney system consists primarily of a filter plenum, a centrifugal fan, and an internal water spray system for fire protection. The banks of filters in the plenum are arranged as follows: roughing, HEPA, and charcoal, HEPA. Each of these banks has an individual externally mounted differential pressure gauge that indicates the pressure drop across that bank.

5.6.4 METHODS OF OPERATION

a. During normal operation all three of the Reactor Building recirculation fan cooler units operate continuously, although only two of the three units are required to operate. The two-speed fan motors operate at high speed during normal operations. Water to the normal cooling coils is supplied from a closed circuit industrial cooler water system. The units are operated from the Control Room. Temperature is controlled by a thermostat which varies the cooling water temperature by varying the capacity of the industrial cooler or by manual adjustments of the capacity of the industrial cooler.

b. Fans supplying air to the operating floors, steam generator compartments, and reactor compartment operate continuously. Units are operated from the Control Room.

c. Fan motors for Operating Floor Ventilation and Steam Generator Compartment Ventilation systems are monitored for excessive vibration, with alarms annunciated in the main control room. Low air flow rates are annunciated in the main control room for the Reactor Compartment Cooling, Operating Floor Ventilation and Steam Generator Compartment Ventilation systems.

d. The Reactor Building Purge system has been provided with controls to stop fans and alarm in the Control Room on high temperature indication, alarm in the Control Room on loss of air flow, and stop fans and alarm in the Control Room on detection of combustible vapors in the purge supply system. The system is operated from the Control Room.

e. Samples of the containment atmosphere will be taken periodically and analyzed, starting approximately 100 hours after a loss of coolant accident and Reactor Building isolation. Should a periodic sample analysis indicate that the volumetric hydrogen concentration has reached 2.5 percent and is projected to exceed 3 percent, preparations for purging the Reactor Building will begin. A portable compressor will be obtained and connected to the leak rate test connections outside the Intermediate Building. The containment monitoring subsystem will be used to monitor the activity and hydrogen concentration of the containment. At a sufficient time prior to starting purging, the compressor will be used to raise the Reactor Building pressure to approximately 1 psig (a 500 scfm compressor will raise the pressure 1 psig in approximately 4-1/2 hours). The operator in the Control Room will monitor wind speed and direction and direct control of the purge rate in accordance with a specific purge plan. The makeup subsystem will be operated as necessary to maintain adequate Reactor Building pressure.

f. The Reactor Building Atmospheric Cleanup system air flow pattern is arranged so that air is drawn from the Reactor Building 308'-0" elevation. Separation of the Kidney System inlet from the purge supply duct allows concurrent operation of the two systems without any impedance, and serves to enhance Reactor Building atmospheric cleaning capability via improved iodine removal. The air then flows through the kidney filter plenum before being freely discharged to the intermediate floor level space through the fan discharge connection. The air in this space is then drawn into the reactor compartment cooling systems and the Reactor Building recirculation cooling system.

5.6.5 SYSTEM EVALUATION

The Reactor Building Recirculation system is a seismic Class I system which operates following a LOCA, and is therefore discussed in Section 6.3.

The Reactor Building purge system operation is limited during Startup, Hot Standby and Power Operation with the purge isolation valves limited to a maximum 30º open (90 degrees being fully open). This helps assure postaccident valve closure. Refer to Section 5.3.3 for details on Reactor Building Purge System isolation.

The atmospheric cleanup system serves no safety related function, however, it is used to reduce the airborne radioactivity to improve working conditions during a Reactor Building entry while the plant is in operation.

5.7 TESTING

5.7.1 STRUCTURAL PROOF TEST AND INSTRUMENTATION

Because of the importance of the containment structure to public safety, its integrity was verified by a pressure test. This pressure test permitted verification that the structural response of the principal strength elements was consistent with the design. The Reactor Building was subjected to internal pressure from 0 psig to 63.3 psig and back to 0 psig. The maximum pressure 63.3 psig is 15 percent in excess of the design pressure. This test pressure level was selected so as to impose, insofar as practical with a static pressure test, maximum stresses on principal strength elements which are reasonably consistent with those stresses imposed by a loss of coolant accident and the design earthquake. The displacement data are obtained from three meridians with emphasis on the vessel behavior at discontinuities. These displacement readings were monitored during the test and describe the building response properly, including secondary effects such as out of roundness. A complete description of the test and acceptance criteria is included in Appendix 5E. For additional test description and results refer to Reference 61.

5.7.2 PREOPERATIONAL LEAK TESTING (HISTORICAL INFORMATION that does not change with time.)

Preoperational leak testing was employed to detect any leaks which may affect the integrity of the containment vessel and the results of the initial integrated leak rate test. The tests were performed at a pressure not less than 55 psig. (For a complete description of the preoperational integrated leak rate test see Reference 62.)

The following areas were locally tested for leakage:

a. Welds in Reactor Building liner plate

b. Containment penetrations (mechanical and electrical)

c. Equipment and personnel access hatches

d. Fuel transfer tubes

e. Isolation valves, blind flanges, spectacle flanges and other isolation devices in lines penetrating the containment boundary

Any unacceptable valve leakage was repaired. A comprehensive program of testing equipment in the manufacturer's plant plus quality assurance during installation kept the leaks detected during preoperational leak monitoring at a minimum.

Examples of the pretesting are: the steel liner welds have been radiographed, liquid penetrant inspected, and leak tested during the normal installation; all penetrations, including the equipment and personnel access locks, are examined in accordance with the requirements of the ASME Nuclear Vessel Code for Class "B" Vessels; the fabricator's plant and all valves have been pressure tested for leaktightness. After installation and prior to the initial integrated leak rate test, local tests were conducted to ensure that any leakage was found and corrected.

Table 5.7-1 lists valves individually leakage rate tested prior to the initial integrated leakage rate test.

5.7.3 INITIAL INTEGRATED LEAK RATE TEST (HISTORICAL INFORMATION that does not change with time.)

An initial integrated leak rate test was performed on the Reactor Building following the completion of building construction and the installation of all systems penetrating the containment boundary. The Reactor containment was subjected to a preoperational integrated leak rate test during the period of March 19, 1974 to March 22, 1974. For a description and results of this test refer to Reference 62. The purpose of this test was to demonstrate the acceptability of building leakage rates at internal pressures of 55 psig (pd) and 30 psig (pt). In addition a reference test was established for subsequent periodic integrated leak rate tests at 30 psig. (However, subsequent periodic testing has been performed at calculated accident pressure, not 30 psig.)

The initial integrated leakage rate test was performed without the penetration pressurization and the fluid block systems (system subsequently ripped-out) being activated. The results of the initial tests provided information for establishing the leakage rate versus pressure characteristics of the building. The absolute pressure-temperature method of measuring leakage from the Reactor Building was employed during the leakage rate test. The objectives of the tests were:

a. To verify that the integrated leakage rate does not exceed the design basis accident leakage rate, La, which is 0.1 percent by weight of the contained atmosphere in 24 hours.

b. To establish maximum allowable leakage rate at reduced pressure (Lt) for use during subsequent reduced pressure integrated leakage rate tests as the criterion for test frequency. (Though reduced pressure testing has not subsequently been authorized or used.)

c. To establish a representative operational leakage rate relation between Ltm and Lt for use during subsequent reduced pressure integrated leakage rate tests as the criterion for criticality. (Though reduced pressure testing has not subsequently been authorized or used.)

d. To obtain a leakage rate history and leakage characteristic of the containment system.

The leakage rate was determined at 30 psig and then at 55 psig by measuring the leakage from the Reactor Building over a period of at least 24 hours after the building temperature and pressure were established. Each 24 hour period was followed by a 12 hour supplemental test for a verification of the test Instrument. During each 36 hour period of testing reactor containment building internal temperature was maintained at $71.5 \pm 0.3°F$. (Except for hour 8, at reduced pressure where temperature reading was out of the confidence level). To ensure equal temperature distribution, fans were provided to circulate the air in the Reactor Building during the test. Verification of the leakage rate was obtained by measuring leakage from the Reactor Building while a known leakage rate was superimposed on the normal building leakage rate. The difference between the total leakage and the superimposed known leakage resulted in the actual leakage rate. This leakage rate (difference) was compared with the original leakage rate as a check against its accuracy.

High accuracy instrumentation was provided to ensure that the temperature, pressure, and dew points of the building atmosphere were accurately measured. For instrument calibration and summary of description refer to Reference 62. For the preoperational test leakage rates based on the total time method of analysis were found to be 0.043 percent per day at 50.6 psig and 0.031 percent per day at 30 psig. These leakage rates are well below the allowable leakage rates of 0.075 percent per day at 50.6 psig and 0.058 percent per day at 30 psig.

Testing was performed in conformance with the requirements of 10CFR50, Appendix J and ANSI N45.4-1972.

5.7.4 POST OPERATIONAL LEAKAGE RATE TESTS

Post operational integrated leakage rate tests have been conducted periodically at 50.6 psig to ensure that the integrity of containment is maintained and to determine if any leakage problems have developed since the previous integrated leakage rate test.

The post operational leakage rate testing also consists of individual local leakage rate tests of certain components associated with penetrations of the containment structure. Within test scope are those penetration boundaries that constitute potential primary containment atmospheric pathways during and following a Design Basis Accident (DBA). But excluded from this scope are those penetration boundaries that are:

1. sealed with a qualified seal system, or
2. test or drain connections which are $\leq$1 inch diameter and administratively secured closed and consist of a double barrier.

Since the approval of the ‘performance-based’ testing option of 10CFR50 Appendix J for TMI-1 in 1997, the test details have been moved from Technical Specification 4.4.1 to the plant’s “Reactor Building Leakage Rate Testing Program” (Reference 72).

5.7.4.1 Integrated Leakage Rate Tests

These tests are conducted periodically in the same manner as the initial leakage rate test. Precautions are employed to protect reactor equipment and instrumentation from being damaged.

5.7.4.2 Local Leak Detection Tests - Penetrations

Components which penetrate and seal the containment boundary with seals, gaskets, or sealant compounds which are resilient, or piping penetrations fitted with an expansion bellows as the only barrier to leakage from containment are leak tested at periodic intervals to ensure their continuing integrity. The local leak rate shall be measured for components listed in Table 5.7-2 using a Type "B" test as defined in 10CFR50, Appendix J.

Except for penetration 201E, all electrical penetrations have at least one epoxy insulator, structurally bonded between wire and steel, and full penetration, steel to steel welds as the barrier between the containment atmosphere and the environment.

Penetration 201E was replaced by a single header plate design, utilizing mechanically swaged ferrule seals with a qualified polysulfone resilient seal. The 24 feed-through ports are able to be pressurized for leak testing by utilizing the pressurization space provided in the header plate, normally connected to the Penetration Pressurization System.

Of the penetrations in the category requiring periodic leak detection testing, only the equipment hatch "O" rings are continuously pressurized with air, at a pressure in excess of 55 psig, by the penetration pressurization system. During reactor operation, continuous indication of leakage through the resilient seals of these items is provided by the leak detection rotameters of the penetration pressurization system.

5.7.4.3 Local Leak Detection Tests And Operability Tests - Valves

Isolation Valves

Periodic tests are conducted to determine the operability and leaktightness of valves serving an isolation function. The local leak rate shall be measured for isolation valves listed in Table 5.7-3 using a Type "C" test as defined in 10CFR50, Appendix J. The safeguards actuation system test circuitry provides the means for testing isolation valve operability.

5.7.5 TENDON STRESS SURVEILLANCE

Provisions have been made for an inservice surveillance program, throughout the life of the plant, intended to provide sufficient inservice historical evidence to maintain confidence that the integrity of the Reactor Building is being preserved. This surveillance program is implemented in accordance with Technical Specification Sections 3.19.1 and 4.4.2.1.

The surveillance program for structural integrity and corrosion protection conforms to the requirements of Subsection IWL of Section XI of the ASME Boiler and Pressure Vessel Code, as incorporated by reference into 10 CFR 50.55a.

5.7.5.1 Deleted

5.7.5.2 Current Inservice Tendon Surveillance Program Requirements And Criteria

The requirements and acceptance criteria are in accordance with the ASME Section XI Boiler and Pressure Vessel Code which are incorporated by reference into 10 CFR 50.55a and are consistent with the surveillance period requirements.

5.7.5.2.1 Sample Selection

For selecting tendons for testing, sampling will be made from the following types of tendons: Vertical, Hoop, Dome, Verticals affected by the SGRP, and Hoops affected by the SGRP. Tendon Sampling from each group is selected as specified by ASME XI as incorporated by 10 CFR 50.55a.

5.7.5.2.2 Visual Inspection

a. Containment Surfaces

The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, is evaluated during the shutdown for each Type A containment leakage rate test (Technical Specification 4.4.1.1) by a visual inspection of these surfaces. This inspection is performed prior to the Type A containment leakage rate test in accordance with 10 CFR 50, Appendix J. Any abnormal degradation must be documented.

b. End Anchorages and Adjacent Concrete Surfaces

The structural integrity of the end anchorages of all tendons scheduled for inspection and the adjacent concrete surfaces is determined by a visual inspection. The condition of the end anchorage and adjacent concrete is recorded. Concrete cracks with widths greater than 0.010 inch are recorded and evaluated. Any crack with a width greater than 0.040 inch is investigated to determine the effect upon the Reactor Building's structural integrity. Any changes in the condition of the end anchorage or the concrete from that previously recorded shall be noted. Any condition or change in condition which indicates abnormal material or structural behavior shall be evaluated as prescribed in IWL-3300 (10CFR50.55a); any violation of the acceptance criteria is subject to the review and reportability requirements of 10CFR50.72 and 50.73.

c. Tendon Surveillance Previous Inspections

The tendon surveillance includes the reexamination of all abnormalities (i.e., concrete scaling, cracking, grease leakage, etc.) discovered in the previous inspections to determine whether conditions have stabilized. The inspection program shall be modified accordingly if obvious deteriorating conditions are observed.

d. Inspection for Crack Growth at Dome Tendons in the Ring Girder Anchorage Areas

The concrete around the dome tendon anchorage areas was inspected for crack growth during the 10th, 15th and 20th year inspections. A minimum of nine dome tendon anchoring areas were selected,with preference given to those anchorage areas previously identified as having concrete crack widths greater than 0.005 inch. The width, depth (if depths could be measured with simple existing plant instruments, i.e., feeler gauges, wire) and length of selected cracks were measured and mapped by charting. No significant changes occurred between the 10th and the 20th year inspection.

5.7.5.2.3 Prestress Monitoring Tests

The tendons selected for each surveillance period are tensioned to determine the force required to loosen the tendon shims. This "lift-off" force is a measure of the prestress remaining in the tendon. The prestressing force measured for each tendon is compared to the Base Values predicted for the specific tendon at the specific time of the test as described in Regulatory Guide 1.35, Revision 3.

Any violation of the acceptance criteria which follows is subject to the review and reportability requirements of 10CFR50.72 and 50.73.

As prescribed in IWL-3221.1, tendon forces are acceptable if:

a. The average of all normalized * tendon lift-off forces, including those measured in b.2 below, for each type of tendon (vertical, dome, or hoop) is equal to or greater than the required minimum average tendon force at the anchorage for that type of tendon.

NOTE

Required minimum average tendon force is:

1033 Kips for Vertical Tendons
1064 Kips for Dome Tendons
1108 Kips for Hoop Tendons

b. The measured force in each individual tendon is not less than 95% of the Predicted Base Value (Predicted Force) obtained from C-1101-153-E410-046, unless the following conditions are satisfied:

b.1 the measured force in not more than one tendon is between 90% and 95% of the predicted force;

b.2 the measured forces in two tendons located adjacent to the tendon in b.1 are not less than 95% of the predicted forces (Predicted Base Values): and

b.3 the measured forces in all the remaining sample tendons are not less than 95% of the predicted force.

* In order for the tendon lift off forces to be indicative of the average level of prestress, each lift off force is adjusted for differences which exist among the tendons due to initial lock off force and elastic shortening loss.

c. If from consecutive surveillances, the measured prestressing forces for the same tendon or tendons in a group indicate a trend of prestress loss larger than expected and the resulting prestressing forces are projected to be less than the minimum required for the group (as specified in paragraph a above) before the next scheduled surveillance, additional lift-off testing shall be done to determine the cause and extent of such occurrence. The condition shall be evaluated as prescribed in IWL-3300.

d. During detensioning and retensioning of tendons, if the elongation corresponding to a specific load differs by more than 10% from that recorded during the last measurement, an investigation shall be made to ensure that the difference is not related to wire failures or slip of wires in anchorages. A difference of more than 10% shall be identified as prescribed in IWA-6000.

If the total population of each group of sampled tendons meets all the above criteria, the structural integrity of the containment shall be considered acceptable.

If a condition is observed which potentially affects containment structural integrity, an investigation of the impact on containment integrity shall be conducted in accordance with T.S. 3.19.1.

5.7.5.2.4 Tendon Material Tests and Inspection

a. A previously stressed tendon wire (strand) from one tendon in each group is removed for testing and examination over its entire length to determine if evidence of corrosion or other deleterious effects is present. At each successive inspection, the samples are selected from different tendons. In addition, all wires or strands found to be broken and not previously documented shall be removed for tensile testing and visual examination.

b. Tensile tests are performed on at least three samples cut from each removed wire or strand, one at each end and one at mid-length. The samples are the maximum length practical for testing and the gauge length for the measurement of elongation is in accordance with the relevant ASTM specification. The following information is obtained from each test:

   1. Yield strength
   2. Ultimate tensile strength
   3. Elongation

c. If any wire sample fails to achieve a minimum ultimate tensile strength of 240,000 psi or if there is rejectable corrosion or pitting as defined in Procedure 1301-9.1, this condition shall be evaluated as prescribed in IWL-3300, and is subject to the review and reportability requirements of 10 CFR 50.72 and 50.73.

5.7.5.2.5 Inspection of Filler Grease

Tendon grease samples are taken and analyzed in accordance with ASME XI as incorporated by 10 CFR 50.55a.

5.7.5.2.6 Reports

A written report of the results of each tendon surveillance as well as evaluations performed pursuant to 10 CFR 50.55a (IWL-3300 and IWA-6000) shall be documented as required by T.S. 4.4.2.1.6.a. Any reports submitted to the NRC pursuant to the requirements of 10 CFR 50.73 shall include the information required by T.S. 4.4.2.1.6.b.

5.7.6 Inservice Inspections

Effective September 9, 1996, the Nuclear Regulatory Commission mandated a comprehensive change to the Inservice Inspection Requirements for the examination of metal and concrete containments. The rulemaking was made by an amendment to the Code of Federal Regulation, Title 10, Part 50.55a, to incorporate the requirements of the ASME Code, Section XI, Subsections IWE and IWL using the edition consistent with the surveillance period. In addition, specific modifications to the Section XI requirements were included in the final rule making and are specified in 10 CFR 50.55a(b)(2)(ix) and 10 CFR 50.55a(b)(2)(x). An expedited examination schedule was implemented as required by the rule. By inclusion of these components within the Section XI framework, programs relevant to applicable containment components will implement the inspection, repair and replacement, evaluation and acceptance, oversight and reporting requirements contained in Section XI.

Subsection IWE pertains to class MC pressure retaining components and their integral attachments, and of metallic shell and penetration liners of class CC pressure retaining components and their integral attachments. Subsection IWE imposes visual examinations on a number of accessible containment metal components. These include the wall liner, dome liner, mechanical penetrations, electrical penetrations, fuel transfer tubes, and the equipment and personnel hatches. A number of these examinations were completed on an expedited schedule as directed by 10 CFR 50. These examinations have been incorporated into the TMI-1 ISI Program. IWE also contains reference to the requirements for completion of Integrated Leak Rate Testing (ILRT) by direct reference to Appendix J. This examination will be conducted through TMI-1 surveillance procedures.

Subsection IWL pertains to the reinforced concrete and post-tensioning system of class components (concrete containment), excluding steel portions not backed by concrete, shell metallic liners and penetration liners extending through the surrounding concrete shell. Subsection IWL contains rules for examination and testing of containment post tensioning systems (tendons) on a five year interval. Previous to IWL, the only requirements for conduct of these examinations and tests were contained in Regulatory Guide 1.35. Many of the regulatory guide requirements were incorporated into IWL.

UFSAR section 5.7.5 contains additional description and history of the TMI-1 tendon surveillance program. IWL also imposed additional visual examinations encompassing 100% of the accessible containment concrete surface on a 5-year interval. IWL examinations are completed through TMI-1 surveillance procedures.

TABLE 5.7-1
(HISTORICAL INFORMATION that does not change with time.)
(Sheet 1 of 4)

Information on this page deleted.

TABLE 5.7-1
(HISTORICAL INFORMATION that does not change with time.)
(Sheet 2 of 4)

VALVES LEAK-CHECKED
PRIOR TO INITIAL INTEGRATED LEAK TEST OF REACTOR BUILDING
DURING 1974

	Isolation Valves	Size	Remarks
Containment Building pressure Instrumentation	*BS-V37A *BS-V37B *BS-V37C *BS-V37D	3/4 3/4 3/4 3/4	Connect directly to containment.
Hydrogen purge discharge and containment radiation monitoring	HP-V1 HP-V6 CM-V1 CM-V2 CM-V3 CM-V4	6 6 1 1 1 1	Connect directly to containment.
Service and instrumentation air supply to Reactor Building	IA-V6 IA-V20 SA-V2 SA-V3	2 2 2 2	Connect directly to containment.
Fuel transfer canal fill and drain	SF-V23 SF-V22	8 8	Connect directly to containment.
Fuel transfer tubes	Double-gasketed blind flange Double-gasketed blind flange	30 30	Connect directly to containment.
Demin. water to Reactor Building	CA-V192 CA-V189	2 2	Connects directly to containment.
Nitrogen to Reactor Building	NI-V27 NI-V26	1 1	Connects directly to containment.
Reactor coolant drain tank vent	WDG-V3 WDG-V4	2 2	Could be connected to primary coolant system if reliefs are open.
Core flooding tank supply CF-V12B	CF-V19B 1 CF-V19A CF-V12A	1 1 1	Connects to RCS.

TABLE 5.7-1
(HISTORICAL INFORMATION that does not change with time.)
(Sheet 3 of 4)

VALVES LEAK-CHECKED
PRIOR TO INITIAL INTEGRATED LEAK TEST OF REACTOR BUILDING
DURING 1974

	Isolation Valves	Size	Remarks
Core flooding	CF-V2A	1	Connects to RCS.
tank sample	CF-V20A	1	
	CF-V20B	1	
	CF-V2B	1	
Decay heat	DH-V22B	10	Connects to RCS.
	DH-V4B	10	
	DH-V22A	10	
	DH-V4A	10	
	DH-V64	2	
	DH-V63	2	
	DH-V2	12	
	DH-V3	12	
	DH-V6B	14	Connects to containment.
	DH-V6A	14	
Reactor coolant	WDL-V303	3	Connects to RCS.
drain tank	WDL-V304	3	
Reactor Building	AH-V1A	48	Connects directly to
purge lines	AH-V1B	48	containment.
	AH-V1C	48	
	AH-V1D	48	
Reactor coolant	NS-V11	8	Inside secondary shield.
pump motors and	NS-V15	8	
coolersNS-V35	8		
	NS-V4	8	
Reactor Building	WDL-V534	6	Connects to containment.
sump drain	WDL-V535		

TABLE 5.7-1
(HISTORICAL INFORMATION that does not change with time.)
(Sheet 4 of 4)

VALVES LEAK-CHECKED
PRIOR TO INITIAL INTEGRATED LEAK TEST OF REACTOR BUILDING
DURING 1974

	Isolation Valves	Size	Remarks
Reactor Building leak rate test	LR-V1	6	Presently LR-V10, flange and spectacle blind flange
	LR-V49	6	
	LR-V2 and blind flange	6	
	LR-V3 and blind flange	6	
	LR-V4 and blind flange	2	No longer exists.
	LR-V5 and blind flange	2	No longer exists.
	LR-V6 and blind flange	2	No longer exists.
Steam generator Cleaning Penetrations 210 and 215	Blind flange inside and outside of Reactor Building at Penetration 210	8	Connects directly to containment.
	Blind Flange inside and outside of Reactor Building at Penetration 215	3	Connects directly to containment.
Steam generator drain pump discharge	Blind flange inside and outside of Reactor Building at Penetration 104	3	Connects directly to containment.
Incore insertion P-241	Blind flange inside and outside of Reactor Building	18	Connects directly to containment.

Table 5.7-2
(Sheet 1 of 1)

COMPONENTS REQUIRED TO BE TESTED USING A TYPE "B" TEST

1. Personnel air lock door gaskets and other seals.

2. Emergency air lock door gaskets and other seals.

3. Resilient seals on the equipment hatch and fuel transfer tube blind flanges.

4. Blind flanges on both ends of pipe through the following penetrations:

PENETRATION NUMBER	DESCRIPTION
104	OTSG drains
105	OTSG cleaning
106	OTSG cleaning
221	Outage Equipment Access Penetration
222	Outage Equipment Access Penetration
241	Incore Instrument Transfer Tube Access
414	Outage Equipment Access Penetration
415	Leak Rate Test Bleed Line
416	Leak Rate Test Bleed Line
417	Leak Rate Test Supply Line

5. Electrical penetrations utilizing resilient seals on the conductors:

PENETRATION NUMBER	DESCRIPTION
201E	Neutron Monitoring Electrical Penetration

TABLE 5.7-3
(Sheet 1 of 1)

CONTAINMENT ISOLATION VALVES REQUIRED TO BE TESTED
USING A TYPE "C" TEST

	VALVES	DESCRIPTION
1.	AH-V1A/1B/1C/1D	Reactor Building Purge Supply/Exhaust
2.	CA-V2,13,446	Primary Sampling
	CA-V189,192,443	Reclaimed Water
	CA-V4A,4B,5A,5B,449	Steam Generator Blowdown and Sampling System
3.	CF-V2A,2B,12A,12B, 19A CF-V19B,20A,20B,46A,46B	Core Flood
4.	CM-V1,2,3,4	Containment Monitoring
5.	FS-V401*	Fire Service
6.	DH-V64,69	Decay Heat
7.	HM-V1A,1B,2A,2B, 3A,3B,4A,4B	Postaccident Hydrogen Monitor
8.	HP-V1,6	Hydrogen Purge
9.	HR-V2A,2B,4A,4B HR-V22A,22B,23A,23B	Hydrogen Recombiner
10.	IA-V6,20	Instrument Air
11.	IC-V2,3,4,6,16,18,102	Intermediate Cooling
12.	MU-V2A,2B,3,18,20 MU-V25,26,116,219,238	Makeup and Purification
13.	NI-V26*,27	Nitrogen
14.	NS-V4,11,15,35,211	Nuclear Services Closed Cooling
15.	PP-V210,211,212,213	Penetration Pressurization
16.	RB-V2A,7	Reactor Building Industrial Cooling
17.	SA-V2,3	Service Air
18.	SF-V22*,23	Spent Fuel Cooling
19.	WDG-V3,4	Waste Gas Header
20.	WDL-V303,304,727	Waste Disposal Liquid
21.	WDL-V534,535	Reactor Building Sump Gravity Drains

* FS-V401, NI-V26, and SF-V22 do not meet the criteria for mandatory Type “C” testing, but they are normally tested for information and trending.

5.8 REFERENCES

1. ASCE Paper No. 3269, Wind Forces On Structures.

2. ACI-301-66 "Specifications for Structural Concrete for Buildings," American Concrete Institute.

3. ASTM C-150, "Specifications for Portland Cement."

4. "Specifications for Concrete Aggregates," Pennsylvania Department of Highways Specifications.

5. ACI 318-63, Building Code Requirements for Reinforced Concrete.

6. ASTM A 615-68, "Deformed Billet-Steel Bars for Concrete Reinforcement."

7. ASTM A 421-65T, "Specifications for Uncoated Stress Relieved Wire for Prestressed Concrete."

8. "Friction Testing on Large Multi-Wire Post-Tensioning Tendons,"by H. Wahl and T. Brown, presented at the 1968 ASCE National Meeting on Transportation in San Diego.

9. "Applications of Welding," Welding Handbook, 5th Edition, American Welding Society, Section 5, p. 85, 1967.

10. Procedure Handbook of Arc Welding, 6th Edition.

11. ASTM A 283-67,"Specification for Low and Intermediate Tensile Strength Carbon Steel Plates of Structural Quality."

12. ASTM A 516-67, "Specification for Carbon Steel Plates for Pressure Vessels for Moderate and Lower Temperature Service."

13. ASTM A 36-67, "Specification for Structural Steel."

14. ASA N 6.2-1965, "Safety Standard for Design, Fabrication, and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors."

15. "Proposed Standard for Leak Rate Testing of Containment Structures for Nuclear Reactors," ANS-7.60, Appendix A.

16. TMI-1 Surveillance Procedures 1303-11.25, "Reactor Building Local Leakage – Access Hatch Door Seals" and 1303-11.18, "Reactor Building Local Leak Rate Testing."

17. Regulations for Protection From Fire and Panic-Commonwealth of Pennsylvania.

18. AISC Manual of Steel Construction.

19. AEC Publication TID-7024, "Nuclear Reactors and Earthquake".

20. "A Study of the Design and Construction Practices of Prestressed Concrete and Reinforced Concrete Containment Vessels," Final Report F-C2121, August 1969, by Chen Pang Tan of the Franklin Institute Research Laboratories. Page 102 through Page 107.

21. ACI 505-54, "Specification for the Design Construction of Reinforced Concrete Chimneys".

22. "Shear and Diagonal Tension," Report of ACI-ASCE Committee 326.

23. T.C. Waters, N.T. Barrett, "Prestressed Concrete Pressure Vessels for Nuclear Reactors," Journal of British Nuclear Energy Society 2 (31) pp. 315-325, July 1963.

24. ASME Steam Tables.

25. "Diagonal Tension Cracking in Concrete Beams with Axial Forces," Alan H. Mattock, Journal of the Structural Division Proceedings of the American Society of Civil Engineers, September 1969, page 1887.

26. A. Pfluger: "Stabilitasprobleme der Elastostik" Springer-Verlag, Berlin, pp. 410 thru 413, 1975.

27. Preliminary Facility Description and Safety Analysis Report for Brookwood Nuclear Station, Unit No. 1 (Docket 50-244).

28. Phillips Petroleum Company.

29. "Reinforcement of a Small Circular Hole in a Plane Sheet Under Tension," Levy, McPherson, and Smith, Journal of Applied Mechanics, June 1948, Page 160.

30. Reissner E., "American Journal of Mathematics," Vol. 63, 1941, pp. 177-184.

31. "Journal of Applied Mechanics," Vol. 31, 1964, pp. 467-476.

32. Professor A. Kalnins of Lehigh University, Bethlehem, Pennsylvania.

33. Maghdi, A.K., and Eringen, A.C., "Stress Distribution in a Cylindrical Shell with a Circular Cut Out," Ingenievs - Archiv, 1964.

34. Savin, G.N., Stress Concentration Around Holes, Pergamon Press, London, 1961.

35. Kalnins, A. "Journal of the Acoustical Society of America," Vol. 36, 1964, pp. 1355-1356.

36. Kraus, H., Welding Research Council Bulletin, No. 108, September 1965.

37. Williams, J.J., "Natural Drought Cooling Towers-Ferry Bridge and After," Institution of Civil Engineers publication, June 1967.

38. "Design Criteria Structural Theory for Protective Construction," Department of the Navy Bureau of Yards & Docks, April 1951.

39. Norris, C.H., et al, Structural Design for Dynamic Loads, McGraw-Hill Book Co. 1959, p. 128.

40. Wood, R.H., "Plastic and Elastic Design of Slabs and Plates," Ronald, Press Co., 1961, p. 29.

41. Modified version of CONTEMPT: L.C. Richardson, L.T. Finnegan, R.J. Wagner, and J.M. Waage, "CONTEMPT - A Computer Program for Predicting the Containment Pressure - Temperature Response to a Loss-of-Coolant Accident", IDO-17220, June 1967.

42. R.C. Schmitt, G.E. Bingham and J.A. Norberg - "Simulated Design Basis Accident Tests of the Carolina Virginia Tube Reactor Containment Final Report", UC-80, December 1970.

43. F.J. Moody, "A Pressure Pulse Model for Two-Phase Critical Flow," Journal of Heat Transfer, 1968.

44. Biggs, John M., "Introduction to Structural Dynamics," McGraw-Hill Book Company, New York.

45. Associates, Inc., Topical Report No. M29, "Dynamic Analysis of Vital Piping Systems Subjected to Seismic Motion."

46. Professors J.M. Biggs, and J.M. Roesset's "Seismic Analysis of Equipment Mounted on a Massive Structure."

47. "Summary of Current Seismic Design Practice for Nuclear Reactor Facilities", USAEC TID-25021, pp. 87-98.

48. FSAR of Oconee Nuclear Power Station, Duke Power Co.

49. GAI Report No. 1729.

50. F.J. Moody - "Prediction of Blowdown Thrust and Jet Forces," ASME Technical Paper 69-HT-31.

51. R. Gwaltney - "Missile Generation and Protection in Light-Water-Cooled Power Plants," ORNL-NSIC-22, UD-80.

52. S. Timoshenko - Vibration Problems in Engineering, 3rd Ed., 1955, D. Van Nostrand Co.

53. S. Timoshenko and D. Young - Engineering Mechanics, 3rd Ed., 1951, McGraw-Hill.

54. G. Haaijer - "Plate Buckling in the Strain-Hardening Range," Transactions, ASCE. Vol. 124, 1959.

55. W. Hall and N. Newmark - "Shear Deflection of Wide-Flange Steel Beams in the Plastic Range," Proceedings ASCE, October 1955.

56. ASTM C 94-68, "Standard Specifications for Ready-Mixed Concrete."

57. ASTM C 42-64, "Methods of Obtaining and Testing Drilled Cores and Sawed Beams of Concrete."

58. Same as Reference 6.

59. ASTM A 435-65, "Standard Method and Specification for Ultrasonic Testing and Inspection of Steel Plates of Fire Box and Higher Quality."

60. ASTM A 421-65, "Specifications for Uncoated Stress Relieved Wire for Prestressed Concrete."

61. GAI Report No. 1838, Reactor Containment Building Structural Integrity Test, June 15, 1974.

62. GAI Report No. 1837, Reactor Containment Building Preoperational Integrated Leak Rate Test, June 1974.

63. URS/John A. Blume & Associates, Report on "Analytical Techniques, Models and Seismic Evaluation of Electrical Raceway Systems".

64. GAI Report 79-14, Design Input Record: (DIR) TMI Unit 1 Piping Analysis, Rev. 3, dated 2/12/82 (Task NM-35, W.O. 04-4692-665).

65. GPUN TDR-1212, "TMI Response to Generic Letter 96-06".

66. "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Seismic Qualification Utility Group (SQUG), Revision 3A, December 2001.

67. CC-AA-320-002, "Use of SQUG Methodology for Seismic Qualification of New and Replacement items."

68. EQE Report No. 50097-R-001, "Conservative Design In-Structure Response Spectra for Resolution of Unresolved Safety Issue A-46 for the Three Mile Island Nuclear Generating Station," Revision 1, July 1993.

69. GPUN Report No. 990-2362, "TMI Intake Screenhouse and Pumphouse SSE Spectra," Revision 0, May 1994.

70. EQE Calculation 42105-C-004, "TMI Diesel Generator Building SSE Spectra," Revision 0, May 1995.

71. Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis," U.S. Nuclear Regulatory Commission, Revision 1, February 1976.

72. ER-TM-380-1001, "TMI-1 Reactor Building Leakage Rate Testing Program."

73. EPRI NP-6041SL, “A Methodology for Assessment of Nuclear Power Plant Seismic Margin,” Electric Power Research Institute, Revision 1, July 1991.

74. Report 240046-R-001, “Seismic Verification Review of Auxiliary Steam System Piping,” EQE International, Revision 1, June 5, 1998.

75. ACI 349-97, "Code Requirements for Nuclear Safety Related Concrete Structures," Reported by ACI Committee 349, April 11, 1997.

76. ACI 216.1-07, "Code Requirements for Determining Fire Resistance of Concrete and Masonry Construction Assemblies," Reported by ACI Committee 216, June 2007.

77. C-1101-153-E410-046, “TMI-1 Reactor Building Post-Tensioning System In-Service Inspection Program Tendon Force Prediction,” Revision 0.

78. ECR 10-00286 Attachment 3, “Review of MSLB temperature effect on RB structure.”

79. ECR 13-00208, "FLEX Design Specification."