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الجامعة الأردنيـــــة
كلية الدراسات العليا

التاريـــخ: 2011/ 01 / 06

نموذج رقم (18)
اقرار والتزام بالمعايير الأخلاقية والأمانة العلمية
وقوانين الجامعة الأردنية وأنظمتها وتعليماتها
لطلبة الماجستير

أنا الطالب: مهند صالح مصطفى أبوحمدة الرقم الجامعي: (8062083)

تخصص: هندسة مدنية/ إنشاءات الكليـــــــة: الهندسة و التكنولوجيا

عنوان الرسالة: Nuclear Power Plants Containment Structures

اعلن بأنني قد التزمت بقوانين الجامعة الأردنية وأنظمتها وتعليماتها وقراراتها السارية المفعول المتعلقة باعداد رسائل الماجستير عندما قمت شخصيا" باعداد رسالتي وذلك بما ينسجم مع الأمانة العلمية وكافة المعايير الأخلاقية المتعارف عليها في كتابة الرسائل العلمية. كما أنني أعلن بأن رسالتي هذه غير منقولة أو مستلة من رسائل أو كتب أو أبحاث أو أي منشورات علمية تم نشرها أو تخزينها في أي وسيلة اعلامية، وتأسيسا" على ما تقدم فانني أتحمل المسؤولية بأنواعها كافة فيما لو تبين غير ذلك بما فيه حق مجلس العمداء في الجامعة الأردنية بالغاء قرار منحي الدرجة العلمية التي حصلت عليها وسحب شهادة التخرج مني بعد صدورها دون أن يكون لي أي حق في التظلم أو الاعتراض أو الطعن بأي صورة كانت في القرار الصادر عن مجلس العمداء بهذا الصدد.

توقيع الطالب: .... التاريخ: 2011/ 1 /6

تعتمد كلية الدراسات العليا
هذه النسخة من الرسالة
التوقيع ..... التاريخ ٩/ ١/ ٢٠١١

NUCLEAR POWER PLANTS CONTAINMENT STRUCTURES

By

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This Thesis was Submitted in Partial Fulfillment of the Requirements for the Master's Degree of Science in Civil Engineering

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December, 2010

Committee Decision

This Thesis/Dissertation (Nuclear Power Plants Containment Structures) was Successfully Defended and Approved on 8/12/2010

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LIST OF ABBREVIATIONS

ABWR	:Advanced Boiling Water Reactor
ACI	:American concrete Institute
AGR	:Advanced Gas-Cooled Reactor
AIJ	:Architectural Institute of Japan
ANS	:American Nuclear Society
ANSI	:American National Standards Institute
APWR	:Advanced Pressurized Water Reactor
ASCE	:American Society of Civil Engineers
ASME	:American Society of Mechanical Engineers
AWS	:American Welding Society
BPVC	:Boilers and Pressure Vessel Code
BWR	:Boiling Water Reactor
CANDU	:CANada Deuterium Uranium
CCD	:Concrete Capacity Design
CFR	:Code of Federal Regulations
DBA	:Design Basis Accident
FBR	:Fast Breeder Reactors
FEM	:Finite Element Method
FSAR	:Final Safety Analysis Report
HTR	:High Temperature Reactors
IAEA	:International Atomic Energy Agency
INSAG	:International Nuclear Safety Advisory Group
JEAG	:Japan Electrical Association
LOCA	:Loss of Coolant Accident
MITI	:Ministry of International Trade and Industries
NISA	:Nuclear & Industrial Safety Agency
NPP	:Nuclear Power Plant
NRC	:National Regulatory Commission
NSC	:National Safety Council
NUREG	:Nuclear Regulation
OBE	: Operational Basis Earthquake
PA	:Preplaced Aggregate
PCPV	:Prestressed Concrete Pressure Vessel
PGA	:Peak Ground Acceleration
PHWR	:Pressurized Heavy Water Reactors
PWR	:Pressurized Water Reactor
RCCV	:Reinforced Concrete Containment Vessel
R/B	:Reactor Building
SIT	:Structural Integrity Test
SSE	:Safe Shutdown Earthquake
SSC's	:Structures Systems and Components
SSI	:Soil-Structure Interaction

LIST OF SYMBOLS

E_s :Steel Modulus of Elasticity
E_c :Concrete Modulus of Elasticity
f_y :Steel Yield Stress
f_s :Steel Stress (f_{st} tension, f_{sc} compression)
f_c' :Concrete Compression Strength
P_u :Ultimate Capacity
P_a :Actual Capacity
ν :Poisson's Ratio
D :Dead loads
L :Live loads
F :Loads resulting from the application of prestress
P_1 :Normal Operating Pressure
R_1 :Normal Operating Piping Loads
T_1 :Normal Operating Thermal Loads
P_2 :LOCA accident pressure
R_2 :LOCA accident piping loads
T_2 :LOCA accident thermal loads
G :Loads resulting from relief valve or other high-energy device actuation
T_o :Thermal effects and loads during normal operating or shutdown conditions
R_o :Pipe reactions during normal operating or shutdown conditions
P_v :External pressure loads
P_t :Pressure during the structural integrity and leak rate tests
T_t :Thermal effects and loads during the test
W :Loads generated by the design wind specified for the plant site
E_o :Loads generated by the operating basis earthquake
E_{ss} :Loads generated by the safe shutdown earthquake
W_t :Tornado loading including the effects of missile impact
W_{tq} :The loads due to tornado wind pressure
W_{tp} :The differential pressure loads due to rapid atmospheric pressure change
W_{tm} :The tornado generated missile impact effects
H_a :Load on the containment resulting from internal flooding
P_a :Design Pressure load within the containment generated by the DBA
T_a :Thermal effects and loads generated by the DBA including T_o
R_a :Pipe reaction from thermal conditions generated by the DBA including R_o
R :The local effects on the containment due to the DBA
R_{rr} :Load on the containment generated by the reaction of a ruptured high energy pipe during the postulated event of the DBA
R_{rj} :Load on the containment generated by jet impingement from a ruptured high-energy pipe during the postulated event of the DBA
R_{rm} :The load on the containment resulting from the impact of a ruptured high-energy pipe during the DBA
γ :Unit Weight
Q : Shear stress
S :Axial Stress
M :Bending Moment

NUCLEAR POWER PLANTS CONTAINMENT STRUCTURES

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Abstract

Nuclear containment structures are one of the most important facilities of the nuclear power plants due to their function to provide safety and security.

What's make the nuclear structures special is not only their design to withstand normal loads but also to provide a physical radiation protection to the surrounding environment and to protect systems and components from exterior natural or human action effects during a different operation conditions varying from normal operation to accidental situations .

This study is an attempt to focus on this new kind of structures to be build in Jordan covering definition, importance, different types, design and analysis from structural point of view and a summary of radiation protection design. Reviewing some of the codes covering these structures with some details for the USA codes and standards, from Europe, and Japan. A comparison between some of USA main codes and Japanese codes was presented. Simplified StaadPro 2007 model analysis for a containment performed to show structure behavior under seismic excitation using an USA regulatory guide response spectrum and response spectrums for Aqaba city. The analysis showed a noticeable difference between the USA Reg.1.60 response spectrum and Aqaba city response spectrums.

Chapter 1

INTRODUCTION

The International Atomic Energy Agency (IAEA) defines the Confinement as a barrier which surrounds the main parts of a facility containing radioactive materials and which is designed to prevent or mitigate the uncontrolled release of radioactive material to the environment in operational states and Design Basis Accident (DBA). Confinement is similar in meaning to containment, but is typically used to refer to the barriers immediately surrounding the radioactive material, whereas containment refers to the additional layers of defense intended to prevent the radioactive materials reaching the environment if the confinement is breached. Hence, for example, in a nuclear power plant confinement may be provided by the reactor pressure vessel, whereas the building housing the reactor may provide containment.

Containment refers to methods or physical structures designed to prevent as low as reasonably achievable the dispersion of radioactive substances. Although approximately synonymous with confinement, containment is normally used to refer to methods or structures that prevent radioactive substances being dispersed in the environment if confinement fails.

The term Nuclear Power Plant (NPP) is used to refer to nuclear power plants for production of electrical energy and not to other types of plants that may be used for distillation, propulsion, supply of hot water, research reactors, etc.

The overall organization of the NPP always involves the following main buildings:

- Reactor building,
- the turbine hall,
- the intake and outlet of cooling water with or without cooling towers,
- the switchyard,

- annex auxiliary building and fuel (new and spent) building.

The NPP is designed, constructed, operated, and controlled in such a way as to reduce consequences of an accident to an acceptable level. In spite of this, series of incidents or accidents are postulated by the safety authorities including leakage and even rupture in the primary coolant system and its consequences. The containment is designed to resist and contain the effects of such accidents. The containment is the most characteristic structure of NPP both for its architectural representativety and its basic purpose is safety.

Practically all plants built during the last few decades include a containment, which in case of internal accident (such as Loss of Coolant Accident (LOCA) with pressures and temperature increase in the containment) or an external event such as aircraft crash, explosions, missile and earthquakes, constitutes the ultimate barrier against the dissemination of fissile products towards the general public. Depending on the type of plant and external hazards considered (such as seismicity), the forces that may be exerted on the containment in case of an accident will differ and so will affect the design of the containment.

Some containments are metallic with a cylindrical or spherical shape, others like (RBMK) which is an abbreviation for the Russian *Reaktor Bolshoy Moshchnosti Kanalniy*: Graphite-Moderated Nuclear Power Reactor is designed to resist lower accident forces and are equipped with box type containments.

Most of the recent containments (approximately 95%) are shell type concrete structures, reinforced concrete or more frequently prestressed concrete, usually cylindrical in shape with varying dimensions depending on the type of the NPP and the specific features of the containment (either single wall or double wall structures with or without liner).

The containment is a complex structure considering the numerous large sizes of penetrations (openings), the magnitude and number of applied loads and load combinations at different situations, the specific regulations and associated inspections performed by safety authorities. Design requires adequate structural knowledge and feedback from previous experience. Construction is closely inspected for the quality of material and of execution. Monitoring and inspections are carried out during the entire lifetime of the plant to ensure that safety requirements remain satisfied.

Chapter 2

CONTAINMENT ROLE IN SAFETY

The main subject of this thesis is the containment, it is necessary to make a brief but more general presentation of NPP safety where the containment is a pivotal component.

2.1 Definition

International Atomic Energy Agency (IAEA) defines Safety as: the achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards.

2.2 Safety Issues for NPPs

Three successive barriers provide the most important issue, prevention of releases of radioactivity:

- The fuel element cladding,
- the primary circuit (core vessel, piping loops connecting pumps, steam generators and pressurizer),
- the containment.

The containment is the third and last barrier. Its integrity under any normal or accidental conditions must be ensured and for this reason, operators and safety organizations closely control it.

2.3 Safety Reports and Regulations

The design engineers and operator presents the safety reports (e.g. Safety Analysis Report (SAR)) to public authorities for approval prior to any authorization of construction, commissioning (start-up), operation, closing, decommissioning and dismantling of the plant.

The safety authorities (regulatory body and related ministries) must always approve regulations, standards and guides.

Depending on the country's practice and specific requirements, its number of operators and nuclear suppliers, the extent of guidance by the safety authorities to the nuclear industry and operators may differ. For example, although the set of regulations is complete and self supporting both in France and in the USA the safety authorities are more guiding in the USA where operators are more diverse. However, the general organization is similar.

In France for instance, documents issued by safety authorities are:

- General technical regulations with many organizational aspects
- Fundamental safety rules presenting clearly the goals to be achieved.

Other types of documents are issued by nuclear industry or the operators are necessarily analyzed and approved by the safety authorities, such as:

- The Rules for Conception and Construction (RCC) are related to:
  - Civil works (RCC-G)
  - Mechanical (RCC-M)
  - Electrical (RCC-E)
  - Fire protection (RCC-I)
  - Fuel (RCC-C)

RCC-G, which is the regulatory document for conception, design and construction of nuclear civil works (and especially the containment), meets the authority's r safety regulations and is approved by safety authorities. RCCG refers to different civil work codes applied in France (each as BAEL and BPEL) or internationally (model code CEB 78), adapts, and completes them where necessary.

Specific documents for a particular plant (such as site conditions, seismic levels, external aggression risk)

In United States of America, the Nuclear Regulatory Commission (US-NRC) is the nuclear licensing authority. The NPP's general design criteria documents are reviewed based on the Code of Federal Regulations (10CFR50) and Nuclear Regulations (NUReg.). For the industrial codes and standards, the American Society of Mechanical Engineers (ASME) - Boiler and Pressure Vessel Code (BPVC) is considered the fundamental code for analysis and design alongside with other industrial codes published by American Concrete Institute (ACI), American Welding Society (AWS) which complies with NRC guidelines.

In Sweden and Finland, Nuclear Structures Systems and Components (SSCs) design and analysis USA guides, industrial codes and standards have been followed in principles.

In Japan, the licensing procedures are not different from other countries such as France and the USA. Although the safety of nuclear power plants is double-checked by the Japan Atomic Energy Safety Commission (JAESC), the Ministry of International Trade and Industries (MITI) plays a major role in licensing review.

The regulatory documents relating to concrete containment vessels are as follows:

- MITI Notice 452 (Technical Standard for concrete containment vessels for nuclear power plants(1990))

- MITI Notice 501(Technical Standard for structural design of mechanical components of nuclear power facilities (1980))
- JAESC (Regulatory Guide for Aseismic Design of Nuclear Power Reactor Facilities (1981))
- Japan Electrical Association (Technical Guidelines for Aseismic Design of Nuclear Power Plants (1984)), JEAG 4601-1987 translated as NUREG/CR-6241, BNL-NUREG-52422.

2.4 The Concept of " Defense In-Depth "

2.4.1 Levels of Accidental Situations

From the International Nuclear Safety Advisory Group IAEA INSAG-10 document. Although the safety requirements tend help to avoid accidental situation, it is assumed that an accident may occur. The defence in depth approach consists of classifying the situations into five different levels and imposing the actions aimed at limiting the consequences to one level and the avoiding them reaching the next and worse level. The successive levels are as follows:

- 1st level: Preventing of failure of any component under normal operation conditions, including the most sever conditions (operational basis earthquake for instance) through prudent design and quality of construction.
- 2nd level: preventing of the development of accidental situation through reliable regulation systems (temperature and pressure increase for instance) enabling the plant to stay within operational conditions even in cases of a deviation. A program for checking abnormal conditions is required (for containments: in service inspection and pressure tests).

- 3rd level: in spite of the actions taken in view of avoiding the first two levels, a series of incidents and postulated (deterministic approach) including instantaneous and complete rupture of a primary loop LOCA. Specific measures are taken to limit the effect of such accidents and avoid radioactive release. They include systems which are only related to safety and not to the operating capacity of the plant:

  - Water injection systems in the primary loop and in steam generators and release of containment,

  - provision of a containment structure capable of withstanding the pressure and temperature effects while remaining sufficiently leak tight.

- 4th level : the risk of multiple failure leading to accidents which are not included in level 3 are considered, which may lead to more severe conditions such as core fusion and as consequence a higher risk of radioactive confinement as possible. The aim of level 4 is to reduce the probability of occurrence of such failure and to maintain as high level of radioactive confinement as possible.

- 5th Level: as contingency, postulated failure of the first 4 levels (including radioactive risks) is assumed, and plans for protection, information and evacuation of the public are set up.

2.4.2 Loss of Coolant Accident (LOCA)

This is considered as the basic accidental load for the containment whatever the initiating event to this accident. It has been seen in the previous section that LOCA is a 3rd level accident, which requires the containment to be capable of withstanding the resulting effects. The Structural Integrity Test (SIT) checks this capability before start-up.

Simplified Description

A simplified presentation of the postulated accident is:

- A complete and instantaneous piping rupture occurs in the primary loop connecting the vessel with the steam generator and the pump at the worst position (between the pump and the vessel known as the cold branch). Immediate loads (in the range of 15 MN) are applied to the reactor building internal structures.

- Pressure lowers rapidly in the loop while pressure and temperature increase in the containment.

- Lack of liquid water (replaced by steam) around fuel elements reduces the nuclear chain reaction (negative reactivity effect in light water reactors) even before automatic lowering of control rods, but heat (over 800^{o}C) and pressure increases in the core while the fuel elements with a risk of rupture of Zircalloy sheath (cladding).

- Water from the accumulator is automatically emptied by gravity into the primary loops. The safety water injection system then comes into operation automatically and the water level increases in the core while the fuel elements stay surrounded with steam due to their temperature and are cooled progressively. The aspiration system of the containment comes into operation simultaneously. Pressure and temperature reduce progressively in the containment. The cooling by recirculation cold water may last for months.

Effect on the Containment

The escape of steam creates a fast sudden (but not dynamic) increase in pressure (in the range of 0.5 MPa absolute in Pressurized Water Reactor) and simultaneously an increase in temperature (in the region of 150^{o}C).

After the initialization of the safety injection systems, the pressure and temperature lower gradually as shown Figure (1).

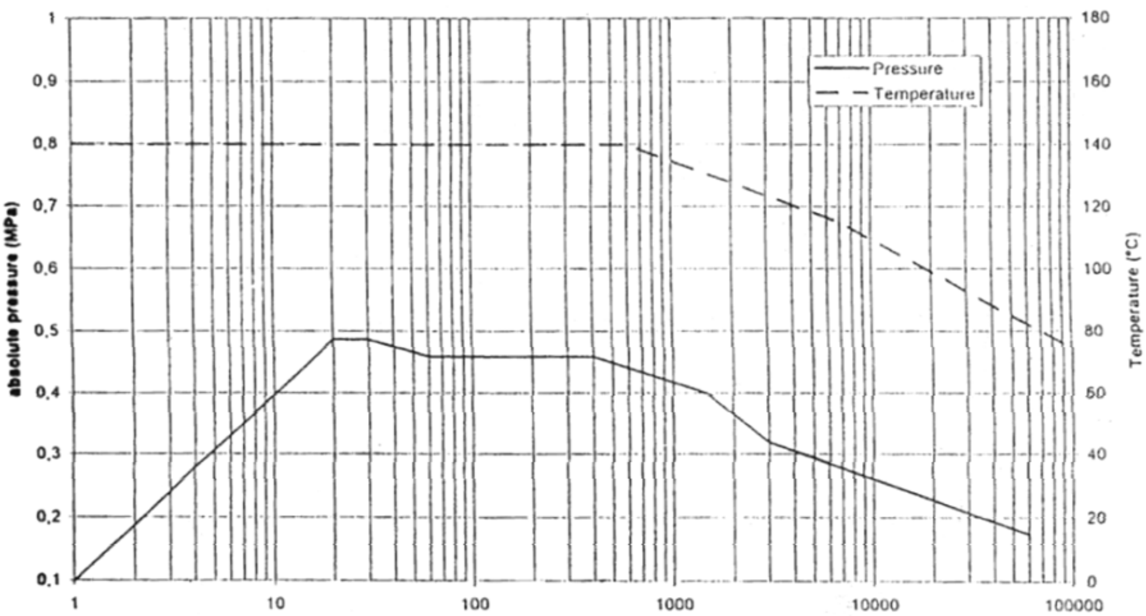


Figure 1.Typical pressure and temperature in case of LOCA versus time (seconds) [18]

Calculation of Loads on Containment In Case of LOCA

The effects of LOCA are calculated by modeling the thermo hydraulic behavior of system throughout the process of the accident. The calculations are carried out with enveloping assumptions to reach conservative results. The calculations normally are carried out by the supplier and strictly controlled by the safety authorities.

Chapter 3

EXAMPLES OF CONTAINMENT DESIGNS

3.1 Common Reactor Types

This chapter presents short descriptions of several concepts for containment systems for the most common reactors now in use or in an advanced stage of design/construction. Out of 436 on operation reactors, there are 54% Pressurized Water Reactors (PWR), 21% Boiling Water Reactor (BWR) and 10% Pressurized Heavy Water Reactor (PHWR) most commonly known as Canada Deuterium Uranium (CANDU) according to IAEA reports as illustrated in Figure (2). The descriptions are not comprehensive but are intended to provide a general overview of how certain containment subsystems have been combined to carry out the containment functions.

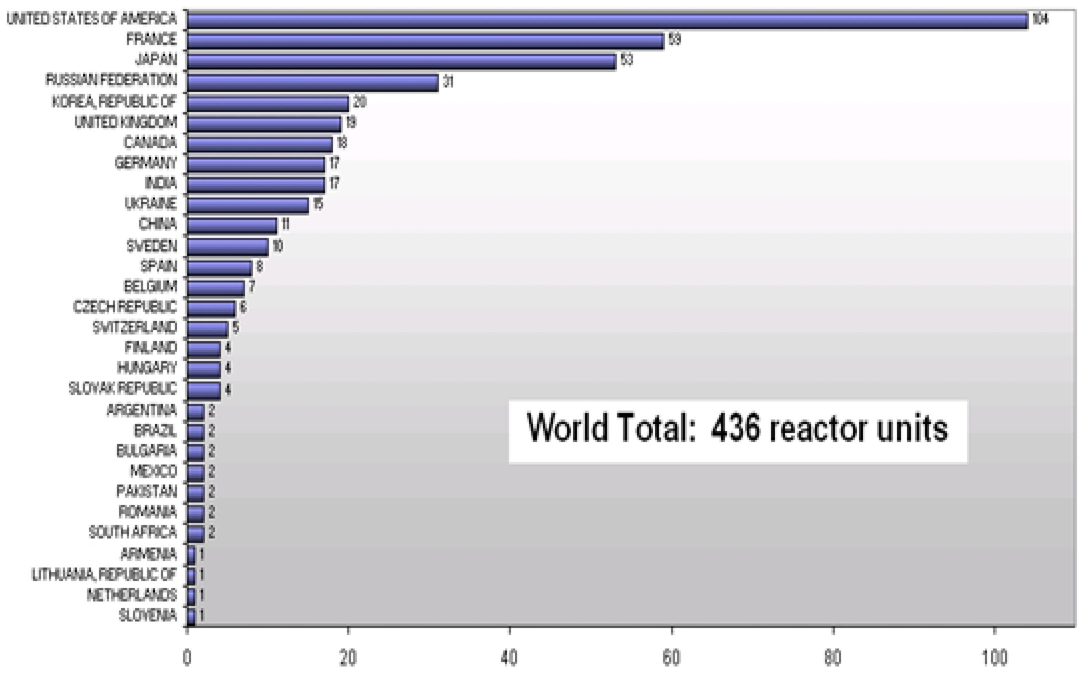


Figure 2. Number of Nuclear Reactors in the World [IAEA]

3.1.1 General Description of Pressurized-Water Reactors (PWR)

Geometry

The shape of containment usually consists of a concrete cylinder topped with a partly spherical dome resting on concrete basement, the specifications are:

- Inner-diameter: from 37m (as a minimum for 900 Mega Watt Electrical (MWe)) to 45m (1300 MWe).
- Wall and dome thickness: from 0.8 m to 1.3 m.
- Base mat thickness: from 1 m (solid rock or resting on basement building for VVER-PWR Russian version) to 5 m (softer foundation material, high seismicity up to 7 m, prestressing gallery within base mat).

Penetrations (Openings)

The containment, which is necessary for safety considerations, is part of complete NPP and must therefore allow for numerous penetrations of various diameters. The largest ones being: the equipment hatch (for instance 8m diam.), the personal air-locks (for instance 3m diam.) the steam penetrations (for instance 1.3m diam.) and numerous electrical or mechanical penetrations.

Main Loads Influencing Design

- LOCA pressure: usually in the range of 0.5 MPa (approximately 5 atm.) absolute pressure, temperature usually in the region of 150 ^{o}C for peak temperature. The pressure test is a cold test with usually 1.15 LOCA relative pressure if there is a steel liner, so as to represent the effect of temperature on the liner creating an outer thrust on concrete shell, or a pressure test equal to LOCA pressure if there is no steel liner.

The pressure effect creates membrane tensile forces in the concrete shell, which are generally balanced by resisting membrane forces due to prestressing tendons or due to passive reinforcement in some containment.

- Earthquake (Operational Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE)).

  These can vary considerably from one site to another. A minimum SSE with high frequency acceleration of (0.15g) is usually taken into account, even in non-seismic areas. Seismic forces induce vertical tensile and shear forces in the shell, bending in base mat with possible uplift from the foundation, and dynamic effects at junctions with mechanical parts (response spectra).

- Extreme environmental conditions such as aircraft or missile impact or external fire and blast effects. Forces are exerted either directly on the containment in case of single wall containment or on outer shell for a double wall type containment.

- Average stress under normal operation conditions: the average concrete stresses in cylindrical part of a typical prestressed containment shell under normal operation conditions are in the region of 10 MPa in the tangential direction and 7MPa in the vertical direction, which evidently requires concrete with sufficient strength (nominal strength in region of 40 MPa).

Main Structural Components

- **Liner**

Most PWR containments have a metallic liner of about 6 mm thick on the inner face of the containment. The liner provides leak tightness where the concrete (reinforced or usually prestressed) ensures stability and resistance to loads.

The concept is clear and satisfactory although the difficulties are numerous and require careful design and construction due to the following:

- The amount of welding and associated weld inspections,
- Stresses at junction with penetrations and at all discontinuities,
- Thermal effects creating additional outward forces, which are exerted on the concrete and so requires a high density of connectors, which might create tensile forces in the liner after accident.

Containments with a steel liner are usually single wall structures, as imposed criteria for leakage in case of an accident are satisfied.

- **Single or Double Wall Concept**

  The basic idea is the separation of two types of actions:
  - Internal Actions (such as pressure, temperature, local forces) acting on the inner (usually prestressed) shell,
  - External Actions or events (such as missiles), acting on the outer shell.

The double wall concept improves the control of any possible leakage through the inner containment, which would then be collected in the annulus between inner and outer shell, which is maintained under slightly negative pressure. In case of an accident, any radioactive leakage would then be collected, filtered and rejected.

A steel containment liner is no longer necessary, as the limited leakage through the inner containment concrete is sufficiently low to be collected without difficulty. Therefore, the acceptable rate of leakage through the inner containment is higher than the acceptable rate of leakage through a single wall containment, which is not collected, and goes directly into the environment.

The double wall concept also ensures better protection of the inner equipment in the case of sever external conditions such as missiles or aircraft crash. It has; however, the inconvenience of lengthening all pipes coming out of the containment (such as the secondary steam piping system) also creates numerous additional penetrations through the concrete of the outer shell.

In the US, Russia, Japan and Ukraine the favored concept is that of single wall containment.

In France, the double wall concept has been applied to all reactors of the 1300 MWe and 1400 MWe series accompanied by the omission of steel liner of the inner containment. The leakage of the inner containment is measured during the preoperational pressure test and periodically tested to ensure that it can be collected safely in the annulus.

In Belgium, the latest containments are of the double wall design with a steel liner on the inner shell.

3.1.2 PWR Example: Full Pressure Dry Containment The concept is illustrated in Figure (3), the primary containment envelope is a steel shell or a concrete building (cylindrical or spherical) with a steel liner that surrounds the nuclear steam supply system. The containment encompasses all components of the reactor coolant system under primary pressure. It is designed as full pressure containment; i.e. it is able to withstand the increases in pressure and temperature that occur in the event of any DBA, especially a LOCA. The atmospheric pressure in the containment envelope is usually maintained at a substantial negative gauge pressure during normal operations by means of a filtered air discharge system (i.e. a fan and High Efficiency Particulate Air (HEPA) filter).

Energy management in the building can be accomplished by an air cooler system or by a water spray system. In addition, the free volume of the containment and the structural heat sinks (the containment envelope and the structures within it) are used to limit peak pressures and temperatures in postulated conditions for pipe rupture accidents. The initial supply of water for the spray system and for the emergency core cooling system is held in a large tank. When this water has been used, suction for both the spray system and the emergency core cooling system is switched to the containment building sump. Water that is recirculated to the reactor vessel is sometimes cooled by means of heat exchangers.

In most designs, the recirculation water for the spray headers – which is also used to limit contamination of the containment atmosphere – is cooled by means of heat exchangers. When pipes rupture in systems other than the reactor coolant system, only the spray system is operated in the recirculation mode.

IAEA Safety Guide No. NS-G-1.10.

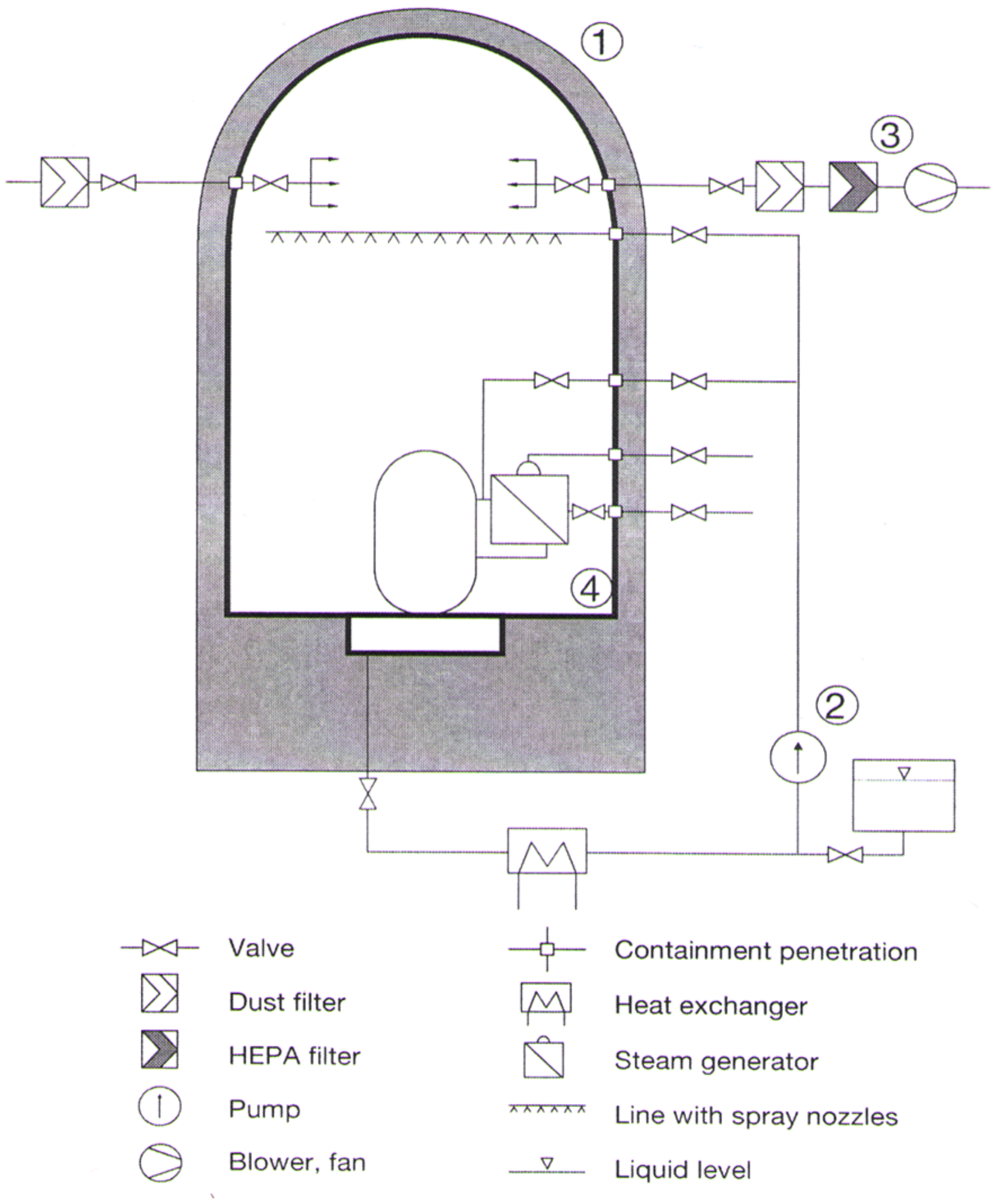


Figure 3. Schematic Diagram of a Full Pressure Dry Containment System for a Pressurized Water Reactor [7]
(1: containment; 2: containment spray system; 3: filtered air discharge system; 4: liner)

3.1.3 General Description of Boiling Water Reactors (BWR)

Geometry

The general shape is again a cylinder, resting on a thick slab and topped with a prestressed slab with a metallic removable lid to enable direct access to the reactor vessel. The containment volume (in the range of 12000 m^3) is much less than for the PWR system. As the only equipment within it are the reactor pressure vessel and the dry and wet well. The overall dimensions for a 1200 MWe BWR unit are in the region of 26 m internal diameter and 35 m in height and for an Advanced BWR (ABWR) 1350 MWe unit in region of 29 m internal diameter and 29.5m in height. The containment is a single wall type but is integrated in the reactor building which provides protection form environmental loads. There is a steel liner of 6 to 10 mm thick.

Penetrations (Openings)

The total number of penetrations is less than the number of penetrations in PWR. As there is more limited equipment within the containment, there is no equipment hatch. The personal air locks are in region of 2.5 m diameter.

The Main Loads Influencing Design

The same types of loads as for a PWR are taken into account. A LOCA is in the region of 0.60 MPa absolute with a temperature with a temperature of 170 ^{o}C. The pressure test is run at 1.15 relative LOCA pressure. The aircraft impact is resisted by the reactor building.

3.1.4 BWR Example: Pressure Suppression Containment

The pressure suppression containment system in boiling water reactors is shown in Figure (4).It is divided into two main compartments: a dry well housing the reactor coolant system and a wet well partly filled with water, whose function is to condense steam in the event of a LOCA. Pipes that are submerged in the water of the wet well connect the two compartments. Spray systems are usually installed in both the dry well and the wet well. The reactor building surrounding the containment forms a secondary confinement, which captures leaks from the containment. The containment envelope usually consists of either a concrete structure with a steel liner for leaktightness or a steel shell.

The purpose of the pressure suppression system is to reduce the pressure if a pipe in the reactor coolant system ruptures. The steam from a leak in these pipes enters the dry well and is passed through pipes into the water of the suppression pool (wet well), where it condenses, and the pressure in the dry well is reduced. The pressure suppression system helps in reducing the concentrations of airborne radioiodines by scrubbing radionuclides from the steam.

The wet well is also used as a heat sink for the automatic pressure relief system. This serves to limit the pressure rise in the reactor coolant system when the reactor cannot discharge steam to the turbine condenser system.

The steam still produced by residual heat after shutdown of the reactor is passed into the water in the wet well via safety relief valves connected to the steam pipes within the dry well.

The concrete or steel structure of the reactor building surrounding the containment serves as protection against external events.

The reactor building is held at a slightly negative gauge pressure in both operational states and accident conditions. In the event of an accident, leaks from the dry well into the reactor building are extracted and filtered by an air removal system to permit the use of controlled emission from the plant stack.

IAEA Safety Guide No. NS-G-1.10.

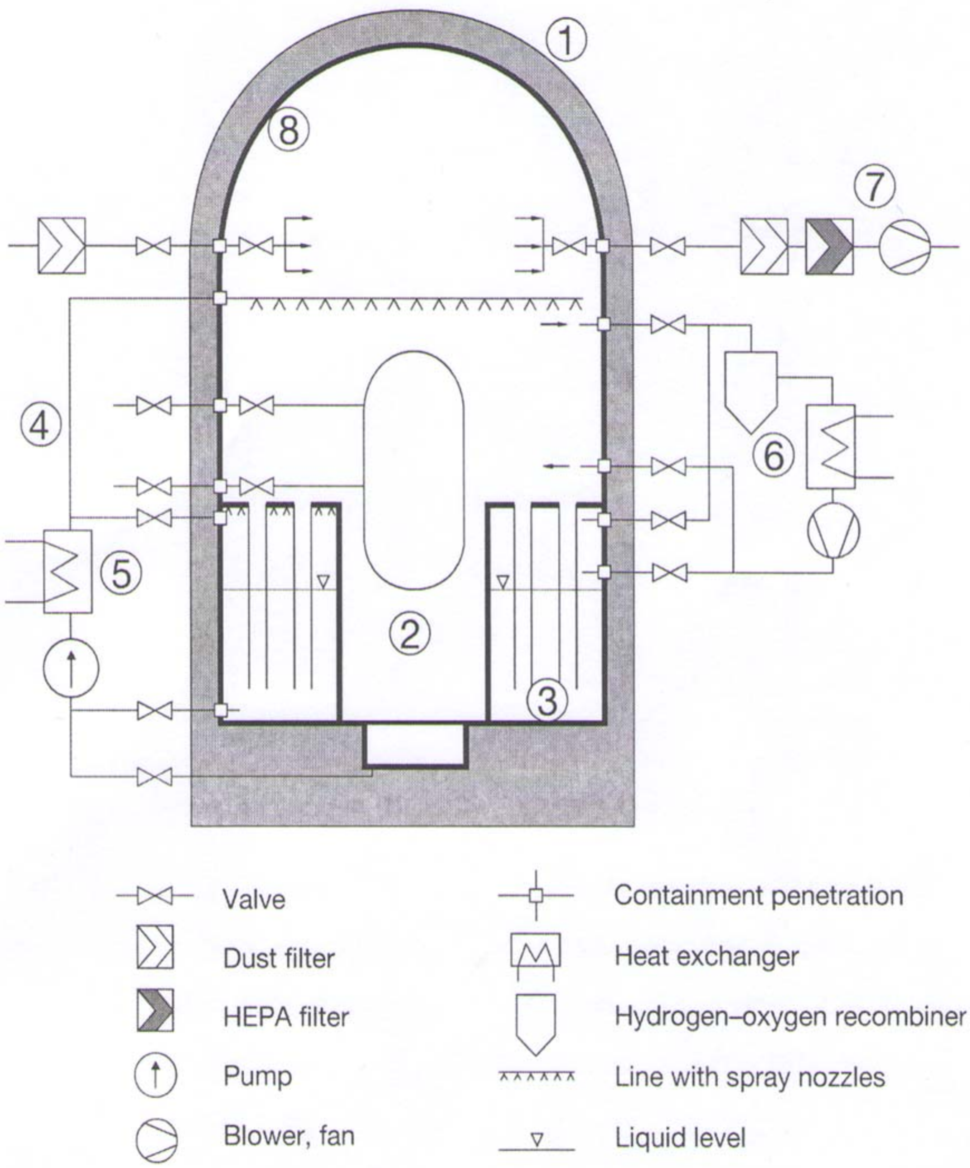


Figure 4. Schematic Diagram of a Pressure Suppression Containment System [7]
(The reactor building with its confinement function is not shown) for a boiling water reactor (1: containment; 2: dry well; 3: suppression pool (wet well); 4: containment spray system; 5: suppression pool cooling system; 6: hydrogen control system; 7: filtered air discharge system; 8: liner)

3.1.5 Pressurized Heavy Water Reactors (PHWR)

Geometry

The general shape of the containment is a cylindrical topped by a partly hemispherical dome. The pressurized heavy water transfers the heat to a steam generator within the containment which leads to organization and dimensions of the containment similar to that of PWR but the LOCA design pressure is considerably lower (less than 0.3MPa absolute) and may become even lower if a vacuum building is provided so as to increase the volume for steam expansion in case of an accident. To allow for this: the containment and the vacuum building may be of prestressed or reinforced concrete; also the liner may be metallic or organic or have no coating at all for double wall containment. The reactor building is based, with some exceptions, on the single shell concept.

3.1.6 PHWR Example: Pressurized Containment

The pressurized containment system used in pressurized heavy water reactors for single unit plant designs Figure (Error! Reference source not found.5) typically consist of the following subsystems:

a) A containment envelope comprising a prestressed, post-tensioned concrete reactor building and its extensions;

b) An energy suppression system that consists of a dousing tank and a spray system that suppress the initial peak pressure;

c) Reactor building cooling system to depressurize the containment in the longer term;

d) Filtered air discharge system to help to maintain sub-atmospheric pressure within the containment envelope in the long term after an accident, and an atmospheric control system to aid in cleanup operations for the containment.

Upon the detection of radioactivity or high pressure in the reactor building, the isolation system closes the appropriate penetrations of the containment envelope.

When high pressure is detected in the reactor building, the dousing system is also activated. The initial peak pressure following a LOCA is suppressed by the condensation of steam through the dousing spray system. Long term energy management is provided by the atmosphere control system (building air coolers) and by the heat exchangers in the recirculation system of the emergency core cooling system. Radionuclide management is accomplished by plate-out on the internal surfaces of the containment envelope, by washout afforded by the dousing spray system, by the leak-tightness of the containment envelope and in some plants by pH control buffers in the sump.

IAEA Safety Guide No. NS-G-1.10.

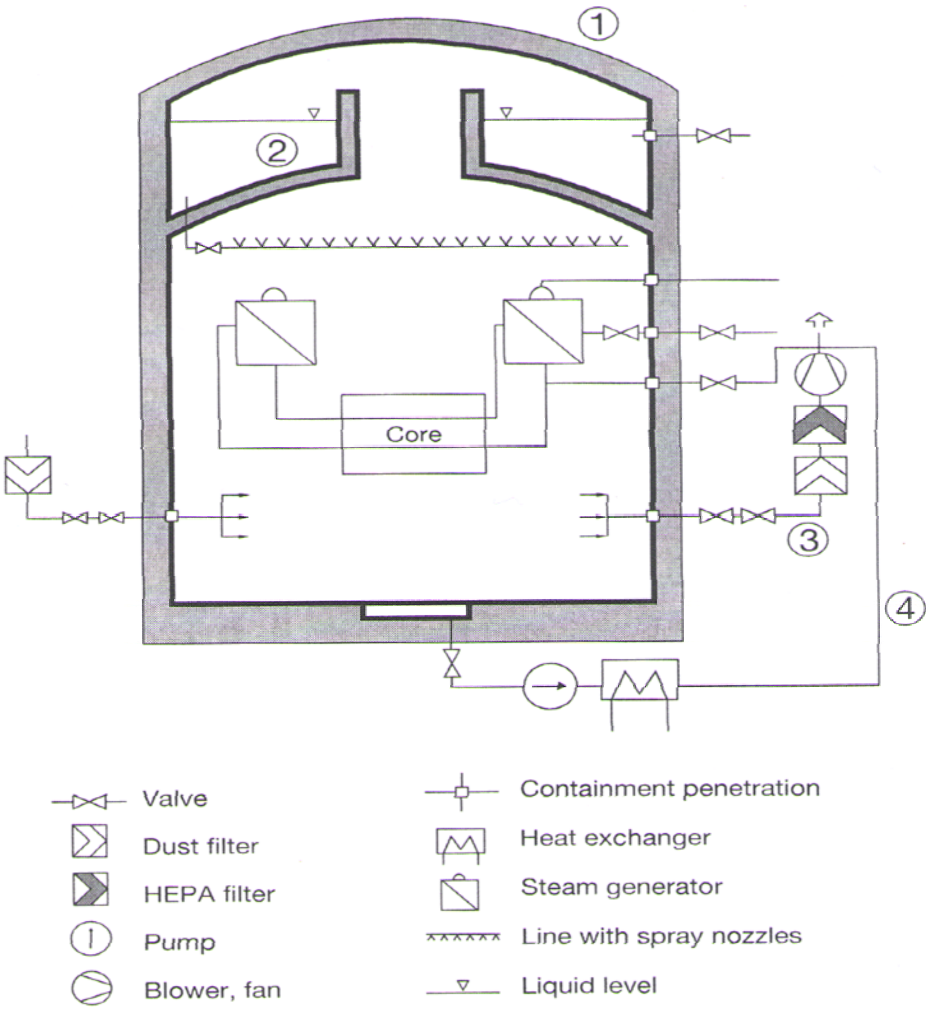


Figure 5. Schematic Diagram of a Pressurized Containment System for a Pressurized Heavy Water Reactor [7]
(1: containment; 2: dousing tank and spray system; 3: filtered air discharge system; 4: emergency core cooling system)

3.2 Less Common Types of Reactors

-Advanced Gas-Cooled Reactor (AGR)

Here, the reactor fuel is uranium oxide, with graphite acting as moderator and CO_2 coolant gas to transfer heat to the boilers. As there is a prestressed concrete pressure vessel, there is no containment. The prestressed concrete pressure vessel encloses the reactor and the pressurized primary coolant during operation of the plant. There are 7 AGR Nuclear power stations in the UK.

-High Temperature Reactors (HTR)

Such reactors constructed mainly in Germany and the USA in the sixties have been decommissioned in the eighties and for this reason are not documented in new design reports. The 300MWe HTR at Schemehausen of Germany was a single barrier cylindrical building in prestressed concrete, designed for LOCA with an overpressure of 0.47MPa and resistance to external chemical explosions, aircraft impact, earthquake, etc.

The general designs for a number of plants in USA of 770MWe and 1160MWe types had reinforced concrete containments in the shape of the cylinder with a domed roof. The containment surrounds the Prestressed Concrete Reactor Vessel (PCRV), which houses the reactor core, steam generators, helium circulators, etc. the design overpressure is 0.35MPa.

-Fast Breeder Reactors (FBR)

In the later design of FBRs, containments have been incorporated. The design pressures are quit low, 0.05-0.15MPa. The containments may be concrete or steel. This type of reactor has not yet had considerable industrial development.

-Graphite-Moderated Nuclear Power Reactor (RBMK)

This type of NPP has been developed and constructed only in the former Soviet Union and satellite countries: it uses uranium oxide within pressure tubes, graphite being the moderator. In case of LOCA, the steam is forced through a basin-bubbler to keep the pressure in the containing compartments rather low 0.3MPa absolute. No shell type containment is provided for this type of reactor. The containing compartments are in reinforced concrete and are non-hermetic, the activity of the steam being comparatively low. More stringent safety standards (N.R.B.96) have been used in Russia requiring the steam compartment to be transformed into containment.

Chapter 4

CONTAINMENTS ANALYSIS AND STRUCTURAL BEHAVIOR

4.1 Introduction

American Society of Mechanical Engineers code design procedures for concrete containments (as well as metal containments) are based on elastic structural analysis, using specified allowable stresses and loads and the design procedures incorporate numerous conservatisms. In addition to the allowable stress factors, the nominal strength and loads are specified subjectively and usually very conservatively. In addition, the inelastic load carrying capacity and ductility of steel are ignored. Numerous analyses and tests of scaled models over the past decade have confirmed that this reserve capacity is well in excess of the design basis internal pressure.

For example, summery of the calculated ultimate capacities, P_u , of six reinforced concrete containments designed in the 1960's to 1980's where failure was defined as yielding of all circumferential reinforcement, lists factors of safety (P_a:actual capacity) P_u/P_a in the range of 2.5 to 6.3. Similarly, studies of steel containments indicated a range of P_u/P_a from 2.2 to 5.6 based on limiting hoop strain equal to twice the yield strain; membrane action of the containment shell was the limiting factor in all cases. Later studies led to similar results and conclusions: median vales of P_u/P_a reported were 3.0 for reinforced concrete containments and 3.4 for steel containments. Assessments of containments safety margins through fragility modeling or other probability-based analysis requires, foremost, an estimate of the median capacity of the containment system at load levels in excess of the design basis. At such levels, the containment response as a whole is well into the inelastic range, and local strains may approach the ductility limit of the material. Steel containments can be modeled as thin shell structures, with stiffeners in both meridional and circumferential directions, and numerous transitions in shell thickness in regions where shell penetrations are required for piping and requirement access. Elastic methods of analysis or simple methods of limit analysis are inadequate for predicting the complex behaviors that occur at such load levels, as the studies above show. Any finite-element analysis used to perform the containment analyses at loads in excess of the design basis must have the capability of handling nonlinear material constitutive behavior, temperature dependence of strength and stiffness, the geometric nonlinearities due to large deformations. Such finite-element methods are essential not only for fragility modeling purposes, particularly for estimating the median capacity, but also for assessing the variability in capacity due to factors known to affect containment behavior that are uncertain in nature.

A fragility assessment clearly must be tied directly to the performance requirement of the containment system, and such requirements must be couched in the context of a nonlinear structural analysis. The primary function of the containment is to confine hazardous materials in the event of an accident. Thus, its most important performance limit is loss of integrity in the pressure boundary. However, this loss of integrity can take a number of forms, with vastly different consequences, ranging from leakage involving depressurization over a period of hours to days and with the possibility of accident mitigation measures, to catastrophic rupture leading to depressurization in seconds and virtually immediate release of radionuclides.

Such performance limits must be related to structural limit states involving response parameters that can be obtained from finite-element analysis and local or general structure or material failure criteria.

This mapping that must occur from the performance requirement space to the structural response analysis space is exceedingly difficult and a source of significant uncertainty in the fragility assessment process.

Some of the difficulties in containment structural analysis can be gleaned from the results of scaled model tests of steel containments similar to one conducted at Sandi National Laboratory during the past decade, which have provided insights into the complexity of metallic pressure boundaries. Such test suggest that structural failure of the containment occurs when the maximum local strains exceed the fracture ductility of the material, typically on the order of 0.25 for carbon steel. While these local strains generally occur adjacent to penetrations or transitions in shell thickness and have little impact on the global structural response of the containment, they are the points where tears initiate that lead to sudden depressurization of the containment. Lesser but still significant local strains in the vicinity of shell penetrations can cause ovalization (structural distortion such that circular parts become ovals) of the penetrations and lead to failure of seals and leakage. In concrete containments with steel liners, loss of integrity is associated with liner tearing that initiates at the point where the liner studs interface with the concrete shell. Such failure can initiate when the far-field hoop strains are in the order of 0.02.

Other performance requirements in addition to integrity of the pressure boundary also play a role in the fragility assessment. Nuclear power plant structural systems are closely integrated with other safety-related mechanical and electrical systems.

Excessive general shell deformations may cause malfunction of appurtenant equipment. For example, large containment shell deformations may cause interference with the polar crane bridge, piping and adjacent structures.

In a BWR Mark I containment, radial expansion of the containment shell may cause sufficient axial deformation in the bellows to crush the bellows and cause leakage. Such performance limits are difficult to relate to the structural responses computed from a nonlinear Finite Element Analysis (FEA).

Thus, the structural analysis of NPP structures is exceedingly complex from the standpoint of first having to perform a nonlinear, large-deformation FEA and next having to identify specific structural response quantities, that can be related in a physically meaningful way to the significant performance requirements of concern (the issue of developing appropriate load model from postulated accident scenarios using principles of thermodynamics and fluid mechanics introduces an additional level of complexity that we have not attempted to address here).

The post-processing and interpreting of the results is particularly difficult, and it is only recently that the computational resources have become sufficient for these tasks to be performed with some confidence.

(Robert E. Melchers and Richard Hough, 2007).

Regarding the steel liner analysis and design, the ASME code shows: "CC-3120 METALLIC LINER

CC-3121 General

The liner shall not be used as a strength element. Interaction of the liner with the containment shall be considered in determining maximum strains.

CC-3122 Liner

The general requirements to be used in the design of metallic liners:

(a) The liner shall be designed to withstand the effects of imposed loads and to accommodate deformation of the concrete containment without jeopardizing leaktight integrity."(ASME code, sec3d2cc3: Page 57)

From Table CC-3270-1 (ASME code, sec3d2cc3: Page 79)

"Stress-Strain Allowable

Construction for Membrane : $f_{s\ tension} = f_{s\ compression} = 2/3\ f_y$"

4.2 Analysis Procedures

Methods of analysis which are based on accepted principles of engineering mechanics and which are appropriate to the geometry of the containment shall be used. In the design of local sections, consideration shall be given to the redistribution of moments and forces in a statically indeterminate structure because of cracking of the concrete, and to the stiffening effect of buttresses or other integral portions of the containment.

Short-term as well as long-term foundation soil properties shall be considered. In order to ensure consideration of the critical condition, a range of values of soil constants shall be considered.

For prestressed containments, the analytical methods selected for construction and normal category load combinations shall account for the creep characteristics and the thick section geometry that is characteristic at ring girders and buttresses.

The ASME code acceptable methods of analysis to determine the stresses and stress intensities required to ensure the adequacy of a design as defined in Section3NB-3200.

The methods presented in SEC3D1 sec3apapa are not intended to exclude others such as computer programs working directly with shell equations or finite element breakdowns of the component under investigation.

Shells

Containments are normally thin shell structures. Elastic behavior shall be the accepted basis for predicting internal forces, displacements, and stability of thin shells.

Effects of reduction in shear stiffness and tensile membrane stiffness due to cracking of the concrete shall be considered in methods for predicting maximum strains and deformations of the containment. Equilibrium check of internal forces and external loads shall be made to ensure consistency of results. Although shell analysis may be based on membrane theory, additional considerations is required for bending and shear forces at penetrations, intersection with base mat, discontinuities, and the stresses and strains caused by temperature variations. The stability of the containment shall be verified, considering the possible reduction in the buckling capacity caused by large deflections, creep effects, and specified construction tolerances.

Model tests may be used instead of the design analysis if they are conservative and represent the prototype containment. In addition, model tests may be used to check the validity of assumptions involved in mathematical analysis.

Base mat, frames, box type structures, and assemblies of slabs

Analyses based on elastic behavior, or other methods generally accepted in conventional practice, shall be used. Effects of discontinuities and loading from the foundation soils shall be considered.

Penetrations and openings

Careful attention shall be given to the analysis of the containment near openings. The effect of an opening on the overall containment shall be considered and the containment shall be thickened around the opening, if necessary, to satisfy allowable stresses and facilitate concrete placement.

The thermal stresses caused by process piping passing through the wall shall be considered.

Chapter 5

RADIATION SHIELDING

5.1 Introduction

Design of NPP structures has a unique feature that in addition to withstand physical load combinations it also provide a mean of radiation shield.

One of the standards used for that regard is the American Nuclear Society (ANS) Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants, which contains methods and data needed to calculate the concrete thickness required for radiation in NPP. Where possible, specific recommendations are made regarding radiation attenuation calculations, shielding design and standards. The standards provide guidance to architect-engineers, utilities and reactor vendors who are responsible for the shielding design of stationary nuclear plants. This standard does not consider sources of radiation other than those associated with NPP. It also excludes considerations of economics aspects of shielding design.

The ANS standard includes a discussion of the nature of concrete, which is a mixture of materials with different proportions that differ from application to application, and by emphasizing those variable aspects of the material, which are important to the shield designer.

5.2 Calculation Methods

5.2.1 Introduction

Various methods are available to the shield designer for calculating the radiation field outside concrete shields. Some of the techniques, mainly kernel methods, are simple enough to allow computation by hand; however, computerizes calculations are preferable and are readily performed. The discrete ordinates and Monte Carlo approaches must be carried out by digital computer, since they require thousands of repetitive calculations.

The decision on which technique to use is not always an obvious one. If some of the simpler methods can be justifiably applied, they should be used first. In the case of NPPs, kernel methods are generally applicable, although there are several notable exceptions where more methods that are sophisticated are needed to better represent the radiation transport involved. These are the primary shield and its reactor sources; and areas involving scattering problems, e.g., labyrinths, ducts, piping penetrations, and (skyshine or airshine) situations (in which the radiation scattered by air).

5.2.2 Calculation Methods

Some of methods used shield analysis: Point Kernel Methods, Discrete Ordinates Method, Monte Carlo Method, Matrix Method, Direct Integration Methods and the Moments Method.

While a shielding engineer may use any calculation method as long as his design accommodates the computational uncertainties, the following observations may be used as a guide in the selection of one of these methods for treatment of a particular problem:

- The Point Kernel Method provides the simplest, most straightforward approach when it can be applied properly.
- The Discrete Ordinates Method provides a detailed map of the radiation field and is best applied where most of the media are dense materials, and deep penetration is involved.
- The Monte Carlo Method is particularly useful in the treatment of radiation transport in complex, asymmetric, three-dimensional configurations where

scattering is important; in some cases, however, considerable user experience might be required where variance reduction techniques are required in lieu of running an inordinate number of particle histories.

The following are typical applications of these three principal methods of shield design:

- The Point Kernel Method may be used in the design of shielding for equipment which contains gamma-emitting fluid, such as demineralizers, heat exchangers, filters, pipes, tanks, steam lines, etc.
- The Discrete Ordinates Method is used to design the primary reactor shield because it readily treats coupled neutron and gamma attenuation.
- The Monte Carlo Method is used for complex radiation transport problems that involve scattering, such as neutron streaming or skyshine.

Chapter 6

MAIN DESIGN FACTORS

6.1 The Design Parameters and Mechanical Properties of Materials

The design parameters necessary for design are independent of the design code but specifications that are used to define the design values may vary depending on the design codes that are used. It is important to differentiate between the following values:

- Values imposed or proposed by regulations:
  - General structures codes and standards in addition to particular codes/standards devoted to analysis, design and construction of NPP when not included in general codes,
  - Specific design criteria and specification for one particular NPP.
- Values resulting from testing
  - Qualification tests and certifications, mainly for steel components,
  - Laboratory and on site testing, mainly for concrete and geotechnics and for prestressing (friction factor).
- Values used for design, resulting as mentioned previously either from codes or testing but which are checked throughout construction.

In Table (1) and Table (2) of the origin of main design values, whether derived from regulations or testes, should in no way be considered as exhaustive but as typical. For more precise and detailed information one must refer to the different existing codes, standards, rules and specifications. The design parameters that are presented in Table (1) and Table (2) do not include those such as aggregate, admixtures, cement, or the chemistry of water, which are necessary for specifying the concrete but are not directly used to design the containment.

Table 1.The Origin of Main Design Values 1/2

Material	**Design parameter under consideration**	**Values from regulation**		**Values from testing**		**Values testing during construct-ion**
		General and particular codes	**Specific design criteria**	**Qualification test certification**	**Laboratory and site testing**	
Concrete	Compressive strength	1	2		x	x
	Young's modulus and Poisson ratio (a)	1	2		x	x
	Shrinkage	1	2		x	
	creep	1	2		x	
	Thermal coefficient (b)	2				
	Density	2				
	Damping factor	2				
	Stress/strain curve	2				
Reinforc-ment Bars	Yield stress	2		x		
	Young's modulus	2				
	Stress/strain curve	2				
Steel Liner	Grade	2		x		
	Yield stress average	2		x		
	Yield stress minimum	2		x		
	Young's modulus and Poison's ratio(a)	2		x		
	Thermal coefficient (b)	2				
	Stress/strain curve	2				

Table 2.The Origin of Main Design Values 2/2

Material	The Considered Design Parameter	Values From Regulation		Values From Testing		Values Testing During Constructi on
		General and particular codes	Specific design criteria	Qualification test certification	Laboratory and site testing	
Pre-stressing	Type of tendon	1	2	x		
	Ultimate tensile stress	1	2	x		
	Tensile yield stress	1	2	x		
	Young's modulus	2		x		
	Relaxation losses	1	2	x		
	Max. stress at tensioning	2				x
	Anchorage slip	1	2	x	x	x
	Friction factor	1	2		x	x
	Dimensions of ducts	1	2	x		
	Allowable curvatures	1	2	x		
	Stress/strain curves	1		x		
	Corrosion protection	1	2	x		
Geo-thecnics	Soil strata characteristic		2			
	Modulus and Poisson's ratios		2			
	Water table levels		2			
	Damping ratio	1	2			

(a) Young's modulus and Poisson's ratio distinguish dynamic values, long-term values, and temperature effects.
(b) Thermal effects include thermal expansion coefficient, transmission coefficient between air and concrete, heat capacity. (1) Denotes “Proposed” and (2) Denotes “Imposed”.

6.2 Loads Exerted on the Containment

Load categories can be classified into:

- Those relative to the external hazards such as wind, earthquake, explosion, missiles and aircraft crash,
- Those relative to internal events such as operation or accidental reactor conditions: radiation, pressure during Structural Integrity Test (SIT) or pressure and thermal effects in the case of the DBA.

Depending on the overall concept of the containment (single or double shell), the loads may be exerted entirely on one shell (single wall containment) or separated between inner and outer shells (double wall containment).

The elementary effects to be taken into account in design are not basically dependent on the type of regulation that is applied, but the values and combination of the loads or actions and also the safety factors on loads and the stresses in materials are dependent on the applied regulations.

According to different standards listed below, loads may be classified as outlined Table (3) Loads Classification from Different Codes (ASME, RCCG and MITI Load Comparison):

- ASME for containments,
- RCCG (French Design and Construction rules for PWR [EDF-RCCG (1998)],
- MITI Notice 4.5.2 (Japanese Notice for Concrete Containment Vessel).

Table 3. Loads Classification from Different Codes

ASME(USA)	RCCG(France)	MITI(Japan)
• **Service Loads** 1-Normal Loads D. Dead Loads including hydrostatic and permanent equipment L. Live Loads including movable equipment F. Prestress Loads G. Loads from relief valves or other devices T_o Thermal loads during Normal Operating (N.O.) R_o Pipe reaction during N.O. P_v Pressure variation inside or outside containment 2- Construction Loads D, L, F, T_o applicable but with construction conditions 3- Test Loads D, L, F, based on test conditions P_t : pressure during SIT (Standard Integrity Test) T_t : thermal effects during SIT • **Factored Loads** 1-Severe environmental loads w : Design wind E_o: OBE (Operational Basis Earthquake) 2-Extreme environmental loads (1) ESS: SSE (Safe Shutdown Earthquake) Wt: Tornado 3-Abnormal Loads Pa: Design pressure load (Design Basis Accident) T_E: Thermal Effects DBA Ra: Pipe reaction and local effects due to DBA	• **Permanent Actions** -Dead weight -Weight of fixed equipment -Lateral thrust of earth and water table -Thermal actions, shrinkage, creep -Ground reaction -Prestressing • **Variable Actions** -Loads during construction -Operating Loads -Climatic Actions -Thermal Actions -Variation of water table -Pressure during SIT -Pressure variations during Normal Operation • **Accidental Actions** -SSE -Explosion -Aircraft crash -Piping rupture -Site Flooding -Fire -DBA -Internal missiles	• **Load Category I** D. Dead Weight including fixed equipment L. Live Loads including movable equipment F. Prestress loads P1. Normal Operating Pressure R1. Normal Operating Piping Loads T1. Normal Operating Thermal Loads • **Load Category II** 1-Safety Relief Valve Operating. D, L, F, P1, R1, T1 2-Testing D, L, F 3- Snow Load with Normal Operating Loads • **Load Category III** 1-Storm Wind Loads with Normal Operating Loads 2- S1 Seismic Load with Normal Operating Loads 3- L(1)- accident (including peak loads immediately after LOCA) D, L, F P2- LOCA accident pressure R2- LOCA accident piping loads T2- LOCA accident thermal loads 4- L(2)-accident +S1 (long-sustaining loading condition 10e-1 year after LOCA when combined with S1) • **Load Category IV** 1- S2 seismic D, L, P1, R1, S2 seismic load 2- L(3) accident D, L, F, R2, P3, 1.5times design pressure 3- J-accident D, L, F and Jet Force 4- L(4)- accident + S1 D, L, F, P2, R2, and S1 seismic load where the maximum pressure and piping loads are taken into account 5- L(5)- accident + snow D, L, F, P2, R2 6- L(6)-accident + Storm D, L , F, P2, R2, and Wind load

Chapter 7

ASME BOILER AND PRESSURE VESSEL CODE

Disclaimer: All loads categories, loads definitions, loads combinations and sections titles are from the ASME BPVC Ref.[11].

In this chapter, a review for the ASME/ACI Committee 359 code: Boiler and Pressure Vessel Code (BPVC) will be presented.

The containment shall be designed to resist the loads and load combinations given in Table 5. The design shall not be limited to the loads specified herein if any other loads are applicable to the particular site conditions.

7.1 Load Categories and Definitions

7.1.1 Service Loads

1. Normal Loads

Normal loads are loads, which are encountered during normal plant operation and shutdown. The nomenclature is as follows:

D = Dead loads, including hydrostatic and permanent equipment loads.

L = Live loads, including any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressures.

F = Loads resulting from the application of prestress.

G = Loads resulting from relief valve or other high-energy device actuation.

T_o = Thermal effects and loads during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

R_o = Pipe reactions during normal operating or shutdown conditions, based on the most critical transient or steady state condition.

P_v = External pressure loads resulting from pressure variation either inside or outside the containment.

2. Construction Loads

Construction loads are loads, which are applied to the containment from start to completion of construction.

The definitions for D, L, F, and T_o -previously mentioned- are applicable but shall be based on construction conditions.

3. Test Loads

Test loads are applied during structural integrity or leak rate testing.

The definitions for D, L, and F -previously mentioned- are applicable but shall be based on test conditions.

In addition, the following shall also be considered:

P_t = Pressure during the structural integrity and leak rate tests.

T_t = Thermal effects and loads during the test.

7.1.2 Factored Loads

1. Severe Environmental Loads

Severe environmental loads are loads that could infrequently be encountered during the plant life.

W = Loads generated by the design wind specified for the plant site.

E_o = Loads generated by the operating basis earthquake. Only the actual dead load and existing live load weights need be considered in evaluating seismic response forces.

2. Extreme Environmental Loads

Extreme environmental loads are loads, which are credible but are highly less probable.

E_{ss} = Loads generated by the safe shutdown earthquake. Weights considered shall be the same as for E_o.

W_t = Tornado loading including the effects of missile impact. Included in W_t are the following:

W_{tq} = the loads due to tornado wind pressure.

W_{tp} = the differential pressure loads due to rapid atmospheric pressure change.

W_{tm} = the tornado generated missile impact effects.

The type of impact, such as plastic or elastic, together with the ability of the structure to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the impact.

3. Abnormal Loads

Abnormal loads are loads generated by the DBA.

H_a = Load on the containment resulting from internal flooding, if such an occurrence is defined in the Design Specification as a design basis event.

P_a = Design Pressure load within the containment generated by the DBA, based upon the calculated peak pressure with an appropriate margin.

T_a = Thermal effects and loads generated by the DBA including T_o.

R_a = Pipe reaction from thermal conditions generated by the DBA including R_o.

R_r = The local effects on the containment due to the DBA. The local effects shall include the following:

R_{rr} = Load on the containment generated by the reaction of a ruptured high energy pipe during the postulated event of the DBA. The time-dependent nature of the load and the ability of the containment to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the effects of R_{rr}.

R_{rj} = Load on the containment generated by jet impingement from a ruptured high-energy pipe during the postulated event of the DBA. The time-dependent nature of the load and the ability of the containment to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the effects of R_{rj}.

R_{rm} = the load on the containment resulting from the impact of a ruptured high-energy pipe during the DBA. The type of impact, for example, plastic or elastic, together with the ability of the containment to deform beyond yield shall be considered in establishing the structural capacity necessary to resist the impact.

7.1.3 Other Loads

1. Static and Seismic Loads

Static loads are defined as those loads which are considered to remain constant with respect to time or which have a long period of application or rise time relative to the response period of the containment. This category also includes seismic loads for which the dynamic effects have been included in their determination.

The following are examples of loads in this category:

(a) Dead load *D,* live load *L*, and prestress *F*;

(b) Accident pressure P_a;

(c) Pipe reactions during normal and postulated accident conditions R_o and R_a;

(d) Design wind *W*, tornado wind pressure W_{tq}, and differential pressure W_{tp};

(e) Operating and safe shutdown earthquake, E_o and E_{ss}, except when combined with impulse loading and impact effects.

2. Impulse Loads

Impulse loads are time dependent and include the following:

(a) The dynamic effects of accident pressure P_a where rate of loading affects the response of the structure;

(b) The effects of pipe rupture reactions R_{rr} and jet impingement loading R_{rj};

(c) The dynamic effects of valve actuation G such as steam relief valve or other high energy device actuation effects where rate of loading affects the response of the structure.

3. Impact Effects

Impact effects are those that can be specified in terms of kinetic energy at impact. These include the impact energies resulting from tornado missiles $W_{tm,}$ pipe rupture generated missiles $R_{rm,}$ and any other specific site-dependent missiles, including the case where a gap exists between the pipe and its structural restraint.

7.2 Load Combinations

Table (4) lists the loads, loads combinations and applicable load factors for which the containment shall be designed. The live load shall be considered to vary from zero to full value for all load combinations.

The maximum effects of P_a , T_a , R_a , R_r and G shall be combined unless a time history analysis is performed to justify lower combined value. For each row you have a laod combination (Category, Service, Test = D (1) + l (1) +F (1) + P_t (1) + T_t (1), were 1 is a factor).

Table 4. Loads and Loads Combinations

Category	D	$L_{(1)}$	F	P_t	G	P_a	T_t	T_o	T_a	E_o	E_{ss}	W	W_t	R_o	R_a	R_r	P_v	H_a
Service																		
Test	1.0	1.0	1.0	1.0	----	----	1.0	----	----	----	----	----	----	----	----	----	----	----
Construction	1.0	1.0	1.0	----	----	----	----	1.0	----	----	----	1.0	----	----	----	----	----	----
Normal	1.0	1.0	1.0	----	1.0	----	----	1.0	----	----	----	----	----	1.0	----	----	1.0	----
Factored																		
Severe Environmental	1.0	1.3	1.0	----	1.0	----	----	1.0	----	1.5	----	----	----	1.0	----	----	1.0	----
	1.0	1.3	1.0	----	1.0	----	----	1.0	----	----	----	1.5	----	1.0	----	----	1.0	----
Extreme Environmental	1.0	1.0	1.0	-----	1.0	----	----	1.0	----	----	1.0	----	----	1.0	----	----	1.0	----
	1.0	1.0	1.0	-----	1.0	----	----	1.0	----	----	----	----	1.0	1.0	----	----	1.0	----
Abnormal	1.0	1.0	1.0	----	1.0	1.5	----	----	1.0	----	----	----	----	----	1.0	----	----	----
	1.0	1.0	1.0	----	1.0	1.0	----	----	1.0	----	----	----	----	----	1.25	----	----	----
	1.0	1.0	1.0	----	1.25	1.25	----	----	1.0	----	----	----	----	----	1.0	----	----	----
Abnormal/severe Environmental	1.0	1.0	1.0	----	1.0	1.25	----	----	1.0	1.25	----	----	----	----	1.0	----	----	----
	1.0	1.0	1.0	----	1.0	1.25	----	----	1.0	----	----	1.25	----	----	1.0	----	----	----
	1.0	1.0	1.0	----	1.0	-----	----	1.0	----	1.0	----	----	----	----	----	----	----	1.0
	1.0	1.0	1.0	----	1.0	-----	----	1.0	----	----	----	1.0	----	----	----	----	----	1.0
Abnormal/Extreme Environmental	1.0	1.0	1.0	----	1.0	1.0	----	----	1.0	----	1.0	---	---	---	1.0	1.0	---	---

(1)Includes all temporary construction loading during and after construction of containment

D: Dead Load, L: Live Load, F: Prestress Load, P_t: Pressure Leak Rate Test Load, G: Relief Valve Load, P_a: DBA Pressure, T_t: Test Thermal Load, T_o: Normal Thermal Loads, T_a: DBA Thermal Loads including T_o, E_o: OBE Loads, E_{ss}: SSE Loads, W: Wind Load, W_t: Tornado Load, R_o: Normal Pipe Reaction Loads, R_a: DBA Pipe Reaction Loads including R_o, R_r: DBA Local Effects on The Containment, P_v: External Pressure Loads, H_a: Internal Flooding Load

7.3 Design Criteria for Impulse Loadings and Missile Impact

Containment and liner shall be designed to resist the effects of impulse loadings from pipe rupture and the impact of missiles resulting from pipe rupture, tornadoes, or any other missile specified in the Design Specification in accordance with load classifications outlined in sub-sections 7.1.2 and 7.1.3.

7.3.1 Design Allowables

– Normal and Severe Environmental Load Categories.

Structural members designed to resist loads in the normal and severe environmental load categories are not allowed to exceed yield.

– Abnormal, Extreme Environmental, Abnormal and Extreme Environmental Load Categories.

Structural members designed to resist impulse loads and dynamic effects in the abnormal, extreme environmental, and abnormal and extreme environmental categories are allowed to exceed yield strain and displacement values. Design adequacy is controlled by limiting the ductility (μ : is defined as the ratio of maximum deformation or strain of the member at the point of collapse to the maximum elastic deformation or strain) assumed in evaluating the energy absorption capability or resistance function of the structure.

7.3.2 Stress Allowables

The allowables applicable to the determination of section strength are given in 7.4 Containment Analysis and Design flow-charts (pages 49-57) in determining (f_y) values, the dynamic effect of the loading may be considered.

7.3.3 Ductility Limits

For Impulse Loads ductility limits shall not exceed one-third the ductility determined at failure. For impact, ductility limits shall not exceed two-third the ductility determined at failure.

7.3.4 Design Assumptions

1. Penetration Formulas and Impulse or Impact Effects

Empirical penetration formulas are assumed to govern design local to the missile impact area. Missile penetration shall be limited to 75% of total section thickness.

Local areas for missile impact are defined as having a maximum diameter equal to (10) times the mean diameter of the impacting missiles, or ($5\sqrt{t}$) plus the mean diameter of the impacting missile where (t) is defined as the total section thickness in feet, whichever is smaller. The effect of damage in the local missile impact area shall be considered in the overall structural integrity of the section.

2. Effective Mass during Impact

For a concrete section, the effective diameter of the section to be used in determining the kinetic energy transferred on missile or dynamic characteristics of the structural response shall be equal to the mean diameter of the missile plus one section thickness (t). Larger values of effective mass may be used if test or analytical verification is available to substantiate the use of larger values.

7.4 Containment Analysis and Design

Analysis and design procedures according to (ASME-ACI Committee 359) BPVC code (Concrete Containment with or without liner) will be summarized in the next flow-charts. The ASME standard covers the proper analysis, design and construction of concrete structures that form parts of a nuclear power plant which have nuclear safety-related functions, but does not cover concrete reactor vessels (as defined by Joint ASME-ACI Committee 359).

ANALYSIS	**ASME CODE SEC3D1** Concrete Containment	**Sec3apapa**
DESIGN	**ASME CODE SEC3D2** Concrete Containment	**Sec3d2cc3**

Figure 6. Flow Chart ASME Analysis and Design of Concrete Containment

STRESS ANALYSIS

A-2000 Analysis of Cylindrical Shells	A-3000 Analysis of Spherical Shells	A-4000 Design Criteria and Equations for Torispherical and Ellipsoidal Heads
A-5000 Analysis of Flat Circular Heads	A-6000 Discontinuity Stresses	A-7000 Thermal Stresses
A-8000 Stresses in Perforated Flat Plates	A-9000 Interaction Method	

Figure 7. Flow Chart ASME Stress Analysis

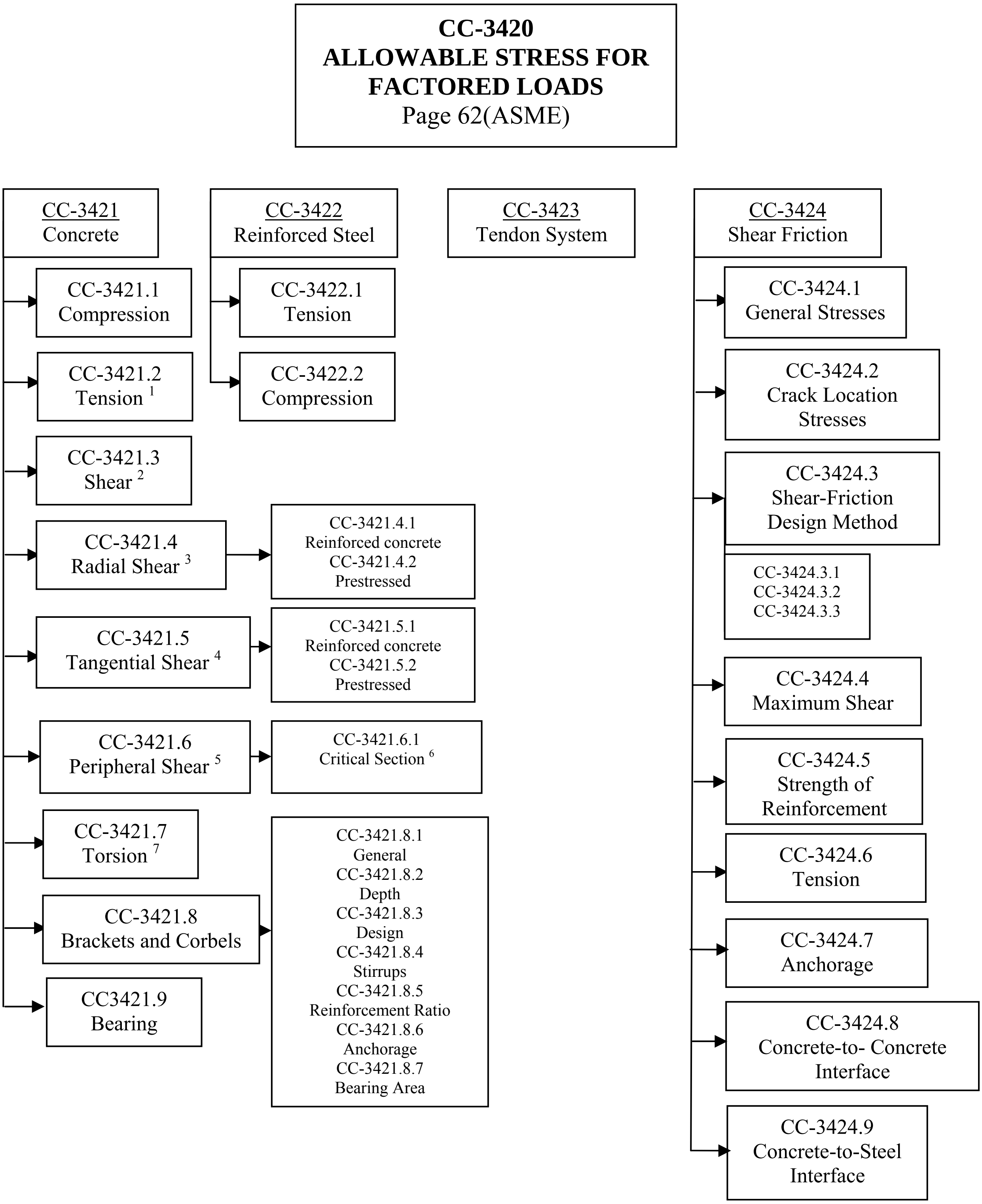


Figure 8. Flow Chart Allowable Stress for Factored Loads

Remarks

1. Concrete Tensile Strength shall not be relied upon to resist flexural and membrane tension.

2. Shear if the calculated shear is greater than the allowables given in CC-3421.4, CC-3421.5, CC-3421.6 and CC-3421.7, then reinforcement or prestressing forces shall be provided in accordance with CC-3520.

3. Radial Shear Radial shear is a transverse shear and is similar to shear in beam analysis. It occurs near discontinuities in shell flexural or membrane behavior. An example of radial shear is the shear caused by self-constraint of a cylinder and base mat during pressurization of the containment. Another example is the shear in the base mat caused by primary vertical forces in structures supported by the mat. A third example is the shear resulting from discontinuity effects, which can occur at the perimeter of penetrations or near other concentrated loads. In this example, peripheral shear must also be considered.

4. Tangential Shear Tangential shear is a membrane shear in the plane of the containment shell resulting from lateral load such as earthquake, wind, or tornado loading.

5. Peripheral Shear is a transverse shear and is similar to punching shear in slab analysis. It is the shear resulting from a concentrated force or reaction acting transverse to the plane of the wall. An example of peripheral shear is the transverse shear associated with a local concentrated load. Another example of peripheral shear is the transverse shear, which can occur at the perimeter of penetrations. In this example, radial shear must also be considered.

6. Critical Section The failure surface for peripheral shear shall be perpendicular to the surface of the containment and located so that its periphery is at a distance *d*/2 from the periphery of the concentrated load or reaction area, except for impact loads where the critical section is defined in CC-3931.

7. Torsion Torsional shear stress is a local, in-plane shear stress produced in the containment wall by a direct external torsional loading applied about an axis normal to the containment wall. In the case of piping penetrations normal to and anchored in the containment wall, the applicable loading is the torsional moment in the penetration. When the penetration is on a skew from the containment wall, the applicable loading is the sum of the components, along an axis normal to the containment, of the internal moments (torsion and bending) in the penetration. Such a loading and anchorage will produce in-plane shear stress in the concrete normal to a radius from the centerline of the penetration.

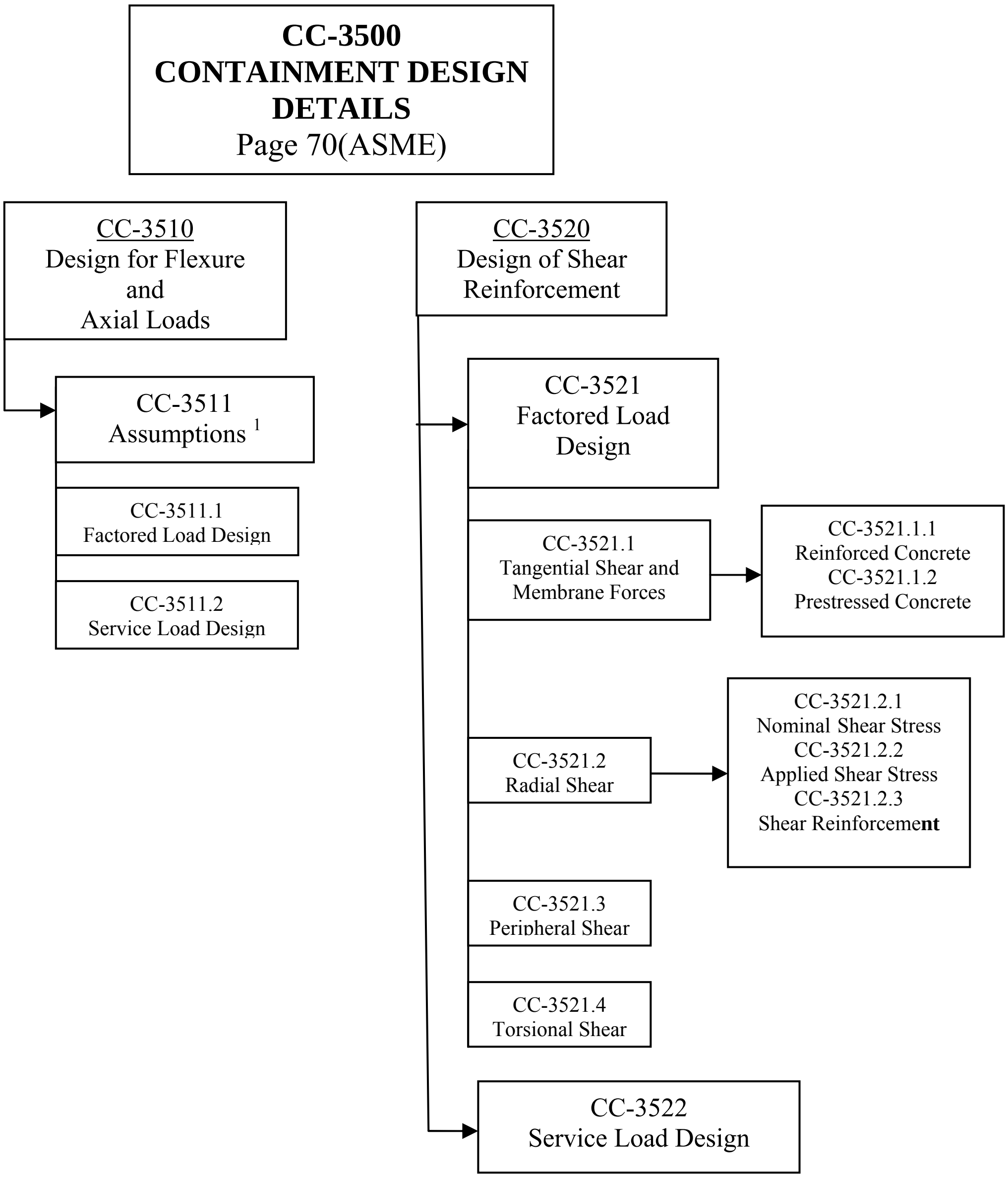


Figure 9. Flow Chat Containment Design Details 1/3

1. Assumptions

Factored Load Design

- The design of sections for flexure and membrane loads shall be based on the assumptions given in this paragraph and on satisfaction of the applicable conditions of equilibrium and compatibility of strains.

- Strain in the reinforcing steel and concrete shall be assumed directly proportional to the distance from the neutral axis.

- Stress in reinforcement below 0.9 of the specified yield strength for the grade of steel used shall be taken as E_s times the steel strain. For strains greater than that corresponding to ($0.9f_y$), the stress in the reinforcement shall be considered independent of strain and equal to ($0.9\,f_y$).

- Tensile strength of the concrete shall be neglected in flexural calculations of reinforced concrete.

- The relationship between the concrete compressive stress distribution and the concrete strain used in the analysis of sections may be assumed to be a triangle, parabola, or any other shape which results in prediction of stress and strains in substantial agreement with the results of comprehensive tests. The stresses determined shall be compared to the stress limits of (**CC3420)** to ensure design adequacy.

Service Load Design

The straight-line theory of stress and strain shall be used and the following assumptions shall be made.

- A section plane before bending remains plane after bending; strains vary as the distance from the neutral axis.

- The stress–strain relation for concrete is a straight line under service loads within the allowable stresses; stresses vary as the distance from the neutral axis.

- Tensile stress of the concrete shall be neglected in flexural calculations of reinforced concrete.

- The modular ratio, $n= E_s /E_c$, may be taken as the nearest whole number but not less than 6. In doubly reinforced members, an effective modular ratio of ($2E_s /E_c$) may be used to transform the compression reinforcement for stress computations.

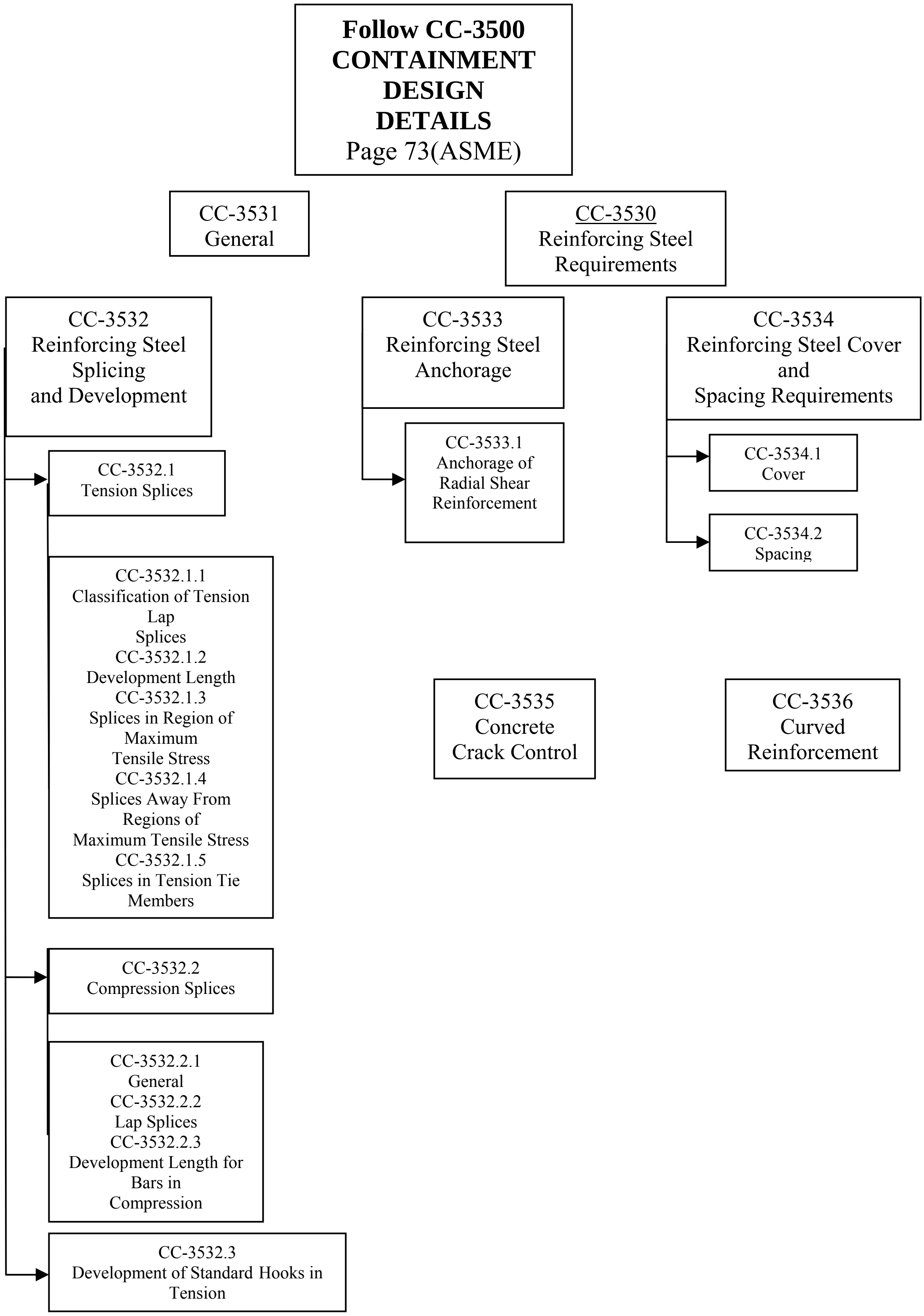


Figure 10. Flow Chart Containment Design Details 2/3

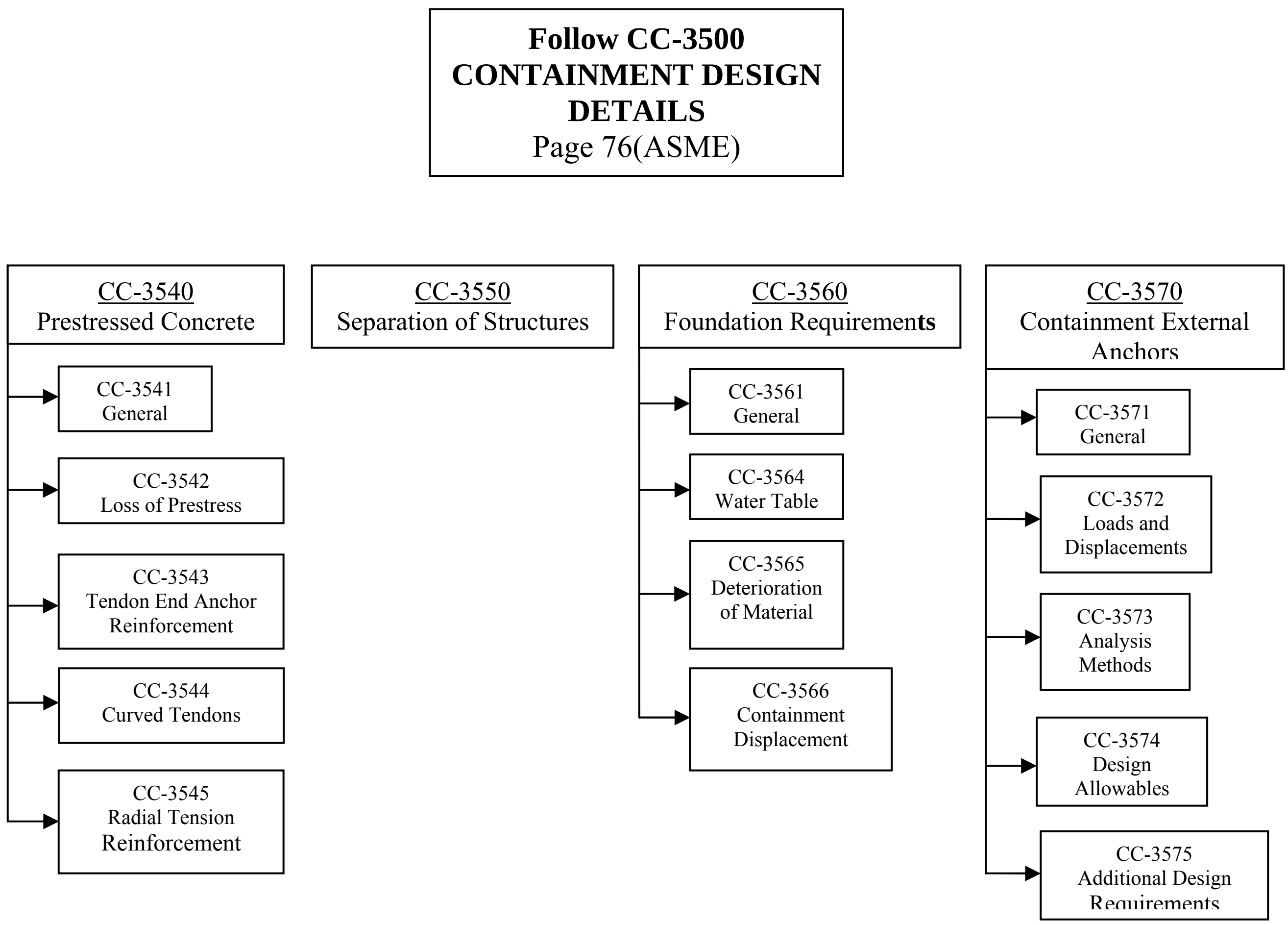


Figure 11. Flow Chart Containment Design Details 3/3

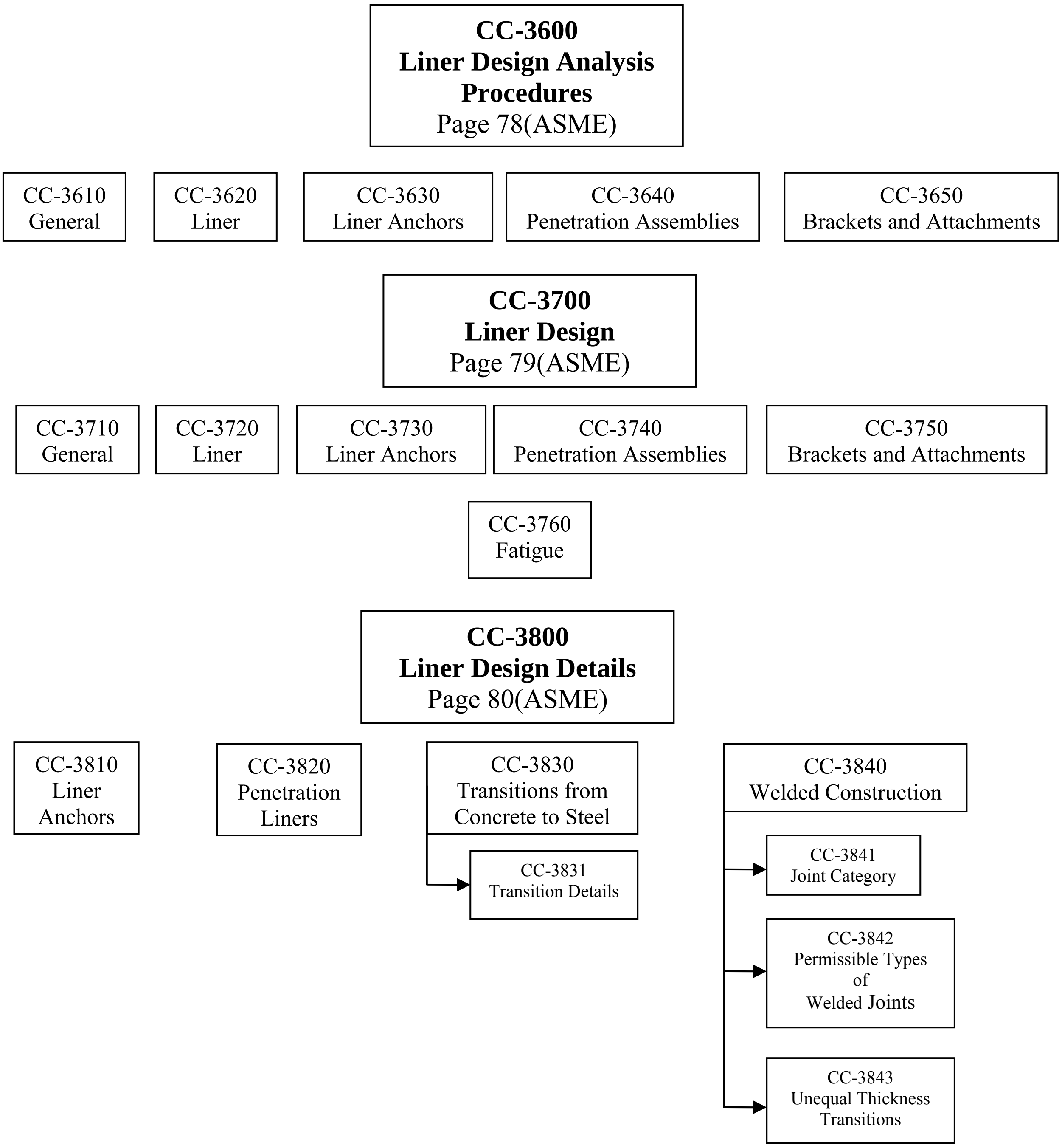


Figure 12. Flow Chart Liner Design and Analysis

7.5 Components Classifications

The ASME code classifies the components into different categories according to its function, working environment and their important to the safety issues.

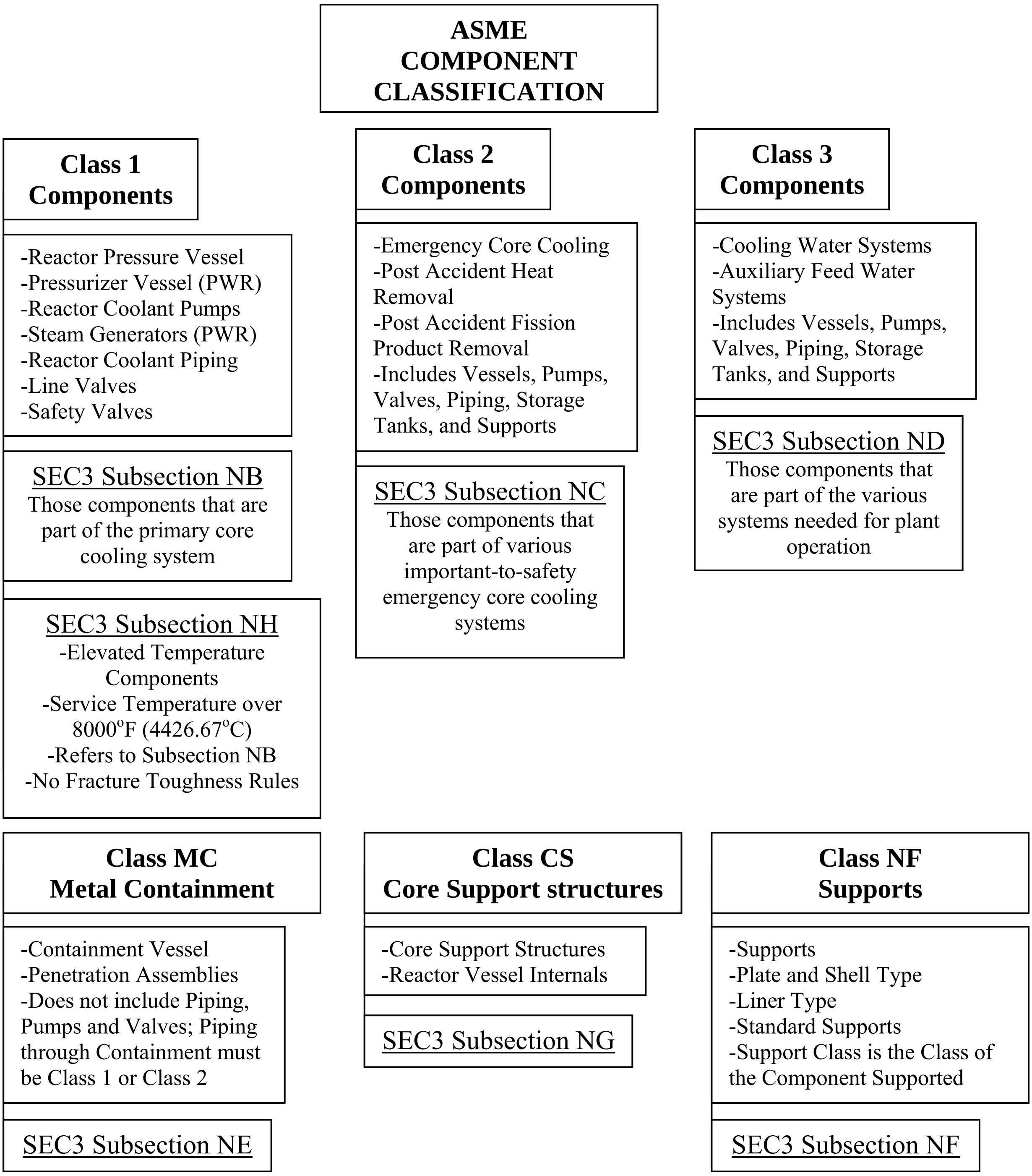


Figure 13. Flow Chart ASME Component Classification

Chapter 8

A COMPARISON BETWEEN THE USA'S NUCLEAR STRUCTURE CODES AND JAPAN'S CODES

8.1 Japan's Nuclear Structure Codes Review

8.1.1 Review

Japan's Examination Guide for a Seismic Design of Nuclear Power Reactor Facilities was under the process of upgrading during 2007-2008 by the Nuclear Safety Commission (NSC) of Japan. The major points of the upgrading are related to the new developments of seismology, the safety concept for public understanding, and the reflection of the government policy to handle seismic margins of nuclear facilities. (Shibata H. 1994) The works and discussions are currently still ongoing.

The recommendations describe in (The Examination Guideline) by the Nuclear Safety Commission are as follows:

1. The safety function of the important facilities including safety protection facilities should never be spoilt even if the plant is attacked by the earthquake ground motion presumed to occur in quite small probability from the viewpoint of the geology, the geological structure around the site, and the seismology within a certain period of service life of the NPP facilities.
2. The above facilities should be designed to have suitable safety margins based on the existence of the certainties in determining the above earthquake ground motion and the uncertainties (dispersion) in the seismic capacities of the NPP facilities.

8.1.2 Classification of Safety Importance in Seismic Design

Through the works for upgrading of the present Examination Guideline, it is required to evaluate the residual risk taking into account the existence of the uncertainty in the seismic capacities of the facilities and the uncertainties in determining a design earthquake ground motion as small as possible.

For such a risk, it has been said that the risk is kept small enough by taking enough margins in the detailed design of the facilities against the seismic load by the design earthquake ground motion of S_2.

Also it is considered desirable to decrease the residual risk from the viewpoint of improving the safety much more. Based on this concept from 2007, it is proposed to revise the classification methodology in the Examination Guideline that the class (A) component, which has a function to redundant accident conditions when accident occurs, is changed over to the present class (As) so that the whole structures, systems, and components in the class (A) will be categorized into the present class (As)

Table (5). shows the SSCs classification pre-2007 and post-2007.

For the name of every class, it is proposed to take (Seismic Class I, II, and III) in expression to avoid confusion. The functional importance classification of the NPP facilities in seismic design based on the way of thinking described above is shown in Table (6).

Table 5. Japan's SSCs Classification

Pre 2007	As	A	B	C
Post 2007	As and A (Class S)	B	C	
Sesismic Design	I	II	III	

Table 6.The Functional Importance Classification of Facilities in Sesmic Design

Seismic Class	Function
Seismic Class I **Class As ,Class A SSC**	A SSC which has radioactive materials inside or a SSC directly related to other SSC having radioactive materials inside, then the function loss of the SSC might be a cause of radioactive material release in the atmosphere. Also a SSC needed to avoid radioactive material release and a SSC needed to reduce an influence by the radioactive material release in the atmosphere in addition those influence and the effect are large.
Seismic Class II **Class B SSC**	SSCs whose influence and effect is small as compared to the above mentioned phenomena in the seismic class I
Seismic Class III **Class C SSC**	SSCs other than the seismic classes of I and II

8.1.3 Overview of Design Criteria for NPP

The seismic requirements of nuclear power facilities were determined according to the importance classification, As, A, B and C, as listed in Table (7) Examples of Aseismic Importance Classification for BWR and PWR, Table (8) shows Seismic Requirements for Classification Pre 2007.

Table 7. Examples of Aseismic Importance Classification

Aseismic Importance	Major Equipment	
	BWR	**PWR**
Class As	(i) Nuclear reactor pressure vessels; vessels, pipes, pumps, and valves within the nuclear reactor coolant pressure boundary. (ii) Spent fuel pool (iii) Control rods, control rod driving mechanism, control rod driving hydraulic system (scram function) (iv) Residual heat removal system (cooling mode in shutdown state) (v) Nuclear reactor containment vessel; piping and valves within the boundary of the nuclear reactor containment vessel.	(i) Nuclear reactor pressure vessels; vessels, pipes, pumps, and valves within the nuclear reactor coolant pressure boundary. (ii) Spent fuel pool (iii) Control rods, control rod driving mechanism, control rod driving hydraulic system (scram function) (iv) Residual heat removal system (v) Nuclear reactor containment vessel; piping and valves within the boundary containment vessel.
Class A	(i) Emergency nuclear core cooling system (ii) Standby gas treatment system (iii) Reactor internal structures	(i) Safety injection system (ii) Annular air cleaning equipment (iii) Reactor internal structures
Class B	(i) Waste disposal system (ii) Steam turbine, condenser, feed water heater (iii) Fuel pool cooling system	(i) Waste disposal system (ii) Spent fuel pit water cleaning system
Class C	(i) Sample collecting system, floor drainage system, etc. (ii) Main generator/transformer	(i) Sample collecting system, floor drainage system, etc. (ii) Turbine equipment, main generator / transformer

Table 8. Seismic Requirements Pre 2007

	Aseismic Importance Classification	Required Analysis	Design Earthquake	
			Horizontal	**Vertical**
Building & Structures	As	Dynamic	S_2	$1/2\ S_2$
	As, A	Dynamic Static	S_1 $3.0\ C_1$	$1/2\ S_1$ C_v
	B	Static	$1.5\ C_1$	______
	C	Static	C_1	______
Equipment & Piping	As	Dynamic	S_2	$1/2\ S_2$
	As, A	Dynamic Static	S_1 $3.0\ C_1$	$1/2\ S_1$ $1/2\ C_v$
	B	Static	$1.8\ C_1$	______
	C	Static	$1.2\ C_1$	______

Note: S_1= extreme design earthquake
S_2 = maximum design earthquake
C_1= static seismic coefficient (= 0.2)

In Table (8), the S_1 earthquake is probably somewhat higher than the OBE of the U.S. practice, and the S_2 earthquake is roughly equivalent to the SSE. Most parts of reactor buildings and containment structures are classified as class (As) facilities. In general, static and liner dynamic analyses are performed for the S_1 earthquake, and nonlinear dynamic analyses for the S_2 earthquake.

8.1.4 Earthquake Ground Motion for Use in the Evaluation of Seismic Safety

It is proposed to treat the earthquake ground motion for use in the evaluation of seismic safety. The earthquake ground motion of S_s, which is defined as an earthquake ground motion, presumed to occur, or possibly occur, though its possibility is quite small, around the site from the viewpoint of seismology and earthquake engineering within a certain period of the plant life.

The earthquake ground motion of S_s is proposed to be designed based on the following:

1. It should be taken into account of past earthquake ground motion and ground motion caused by active faults "Seismo-techtonic" knowledge is also considered for reference.
2. It should be taken into account as earthquake ground motion to be considered at least as earthquake ground motion presumed without specifying the seismic sources. It is presented that the common way of thinking to determine a response spectrum based on a probabilistic study and/or past earthquake records obtained in the neighborhood of epicenters without seismic fault in the inland cluster earthquakes.
3. The probability of the ground motion level of S_s is checked after design.
4. Earthquake ground motion in the vertical direction at free field should be also determined.

Figure (14) shows the current proposed methodology for determining the design ground motion of S_s.

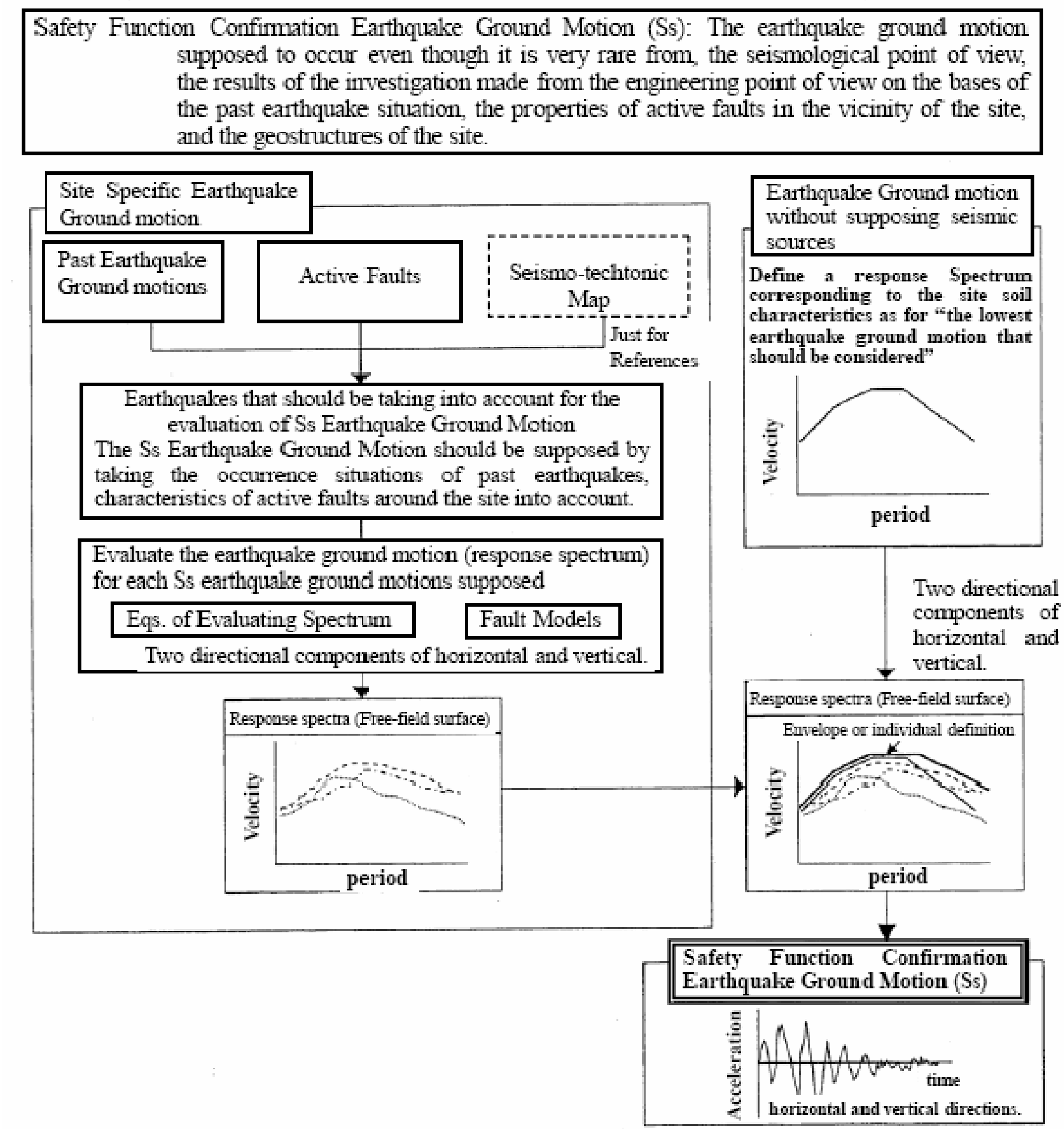


Figure 14. Flow chart for generating design earthquake ground motion [23]

The Architectural Institute of Japan (AIJ) recommendations for design of reactor building the concept (allowable state) has been introduced which is similar to the classification of levels, A, B, C and D limits (i.e., normal, upset, emergency and faulted) in the ASME code. The classification of the allowable states I, II and III, are summarized briefly in Table (9).

Table 9. Allowable States

Allowable States	Plants Condition	Allowable Concrete Compression Stress	Other Allowable Stresses	Thermal Stiffness Reduction Factor
I	Normal Operation	1/3 f_c'	Long-Term allowable	1/2
II	S_1 EQ, Storm, Snow	2/3 f_c'	1.5 times above	1/3
III	S_2 EQ, Accident	0.85 f_c' (ε = 0.3%)	Materail Strength	Neglect

8.1.5 Design Requirements for Containment Structures

The design of concrete containment structures is performed based on MITI Notification No. 452. This document, and in particular the background information upon which this standard is based, may be useful as the test results of large-scale containment structures are extensively utilized. Some unique features are highlighted herein :

Loading State: According to Notification No. 452, the structural design of containment is performed based on the Loading States Table (10). Compared to the AIJ recommendations for reactor building Table 10, the basic design requirements may be considered to be similar.

Thermal State: The evluation of the thermal stresses is pereformed according to the following procedure:

- Reduce the elastic stiffness (i.e. Young's Modulus) by a factor of 1/2 for loadingStates-I and II, and 1/3 for Loading State - III, and calculate the thermal stresses.
- Calculate stresses for other loads using the original elastic stiffness.
- Combine the above stresses.
- For the Loading State - IV (S_2 earthquake and accident), the thermal stress is neglected.

Table 10. Loading State for Containment Structures

Loading State	Plant Condition	Allowable Concrete Compressive Stress	Other Allowable Stresses	Thermal Stiffness Reduction Factor
I	Normal	1/3 f_c'	Long - term allowable (RC standard)	1/2
II	Relief Valve / Test			
III	S_1 EQ	2/3 f_c'	Short - term allowable (RC standard)	1/3
IV	S_2 EQ / Accident	Strain limit 0.3% for Concrete 0.5% for Steel		Neglect

8.1.6 Japanese Seismic Design Review

Through 2006-2008, the Japanese seismic design guide of nuclear power reactor facilities was revised and some of the new aspects can be summarized in Table (11).

Design Guide Revision:

Table 11. Design Guide Revision

Item	Before	Revised
Design Base Earthquake Definition (DBE)	- S_1: Return period more than 10000y Stay in Elastic Region* - S_2: Return period more than 50000y Keep Function* * for Class As, A component	- One (DBE) S_s: Consider active fault hereafter late Pleistocene (80000-130000y before) Keep Function* - S_d for design (not earthquake) To Stay in Elastic Region* $S_d = \alpha\ S_s$, $\alpha \geq 0.5$ * Class S component
Geological Survey		Use most updated knowledge and technique
Consideration of Vertical Seismic Forces	$F_V = 1/2\ F_H$ (Static)	Define Fv Dynamically
Over DBE Earthquake		Possibility of over DBE earthquake cannot be defined. Risk by over DBE is to be assessed for reference
Seismic Classification	As, A, B, C	S (old As and A), B, C → Old A class ranked up to As
Phenomena accompanying earthquake		Consider the effect of: - Tsunami, - Collapse of around inclined plane

Each Item in Table (11) described next

1. DBE Definition

1.1 DBE Definition - Earthquake Research Flow

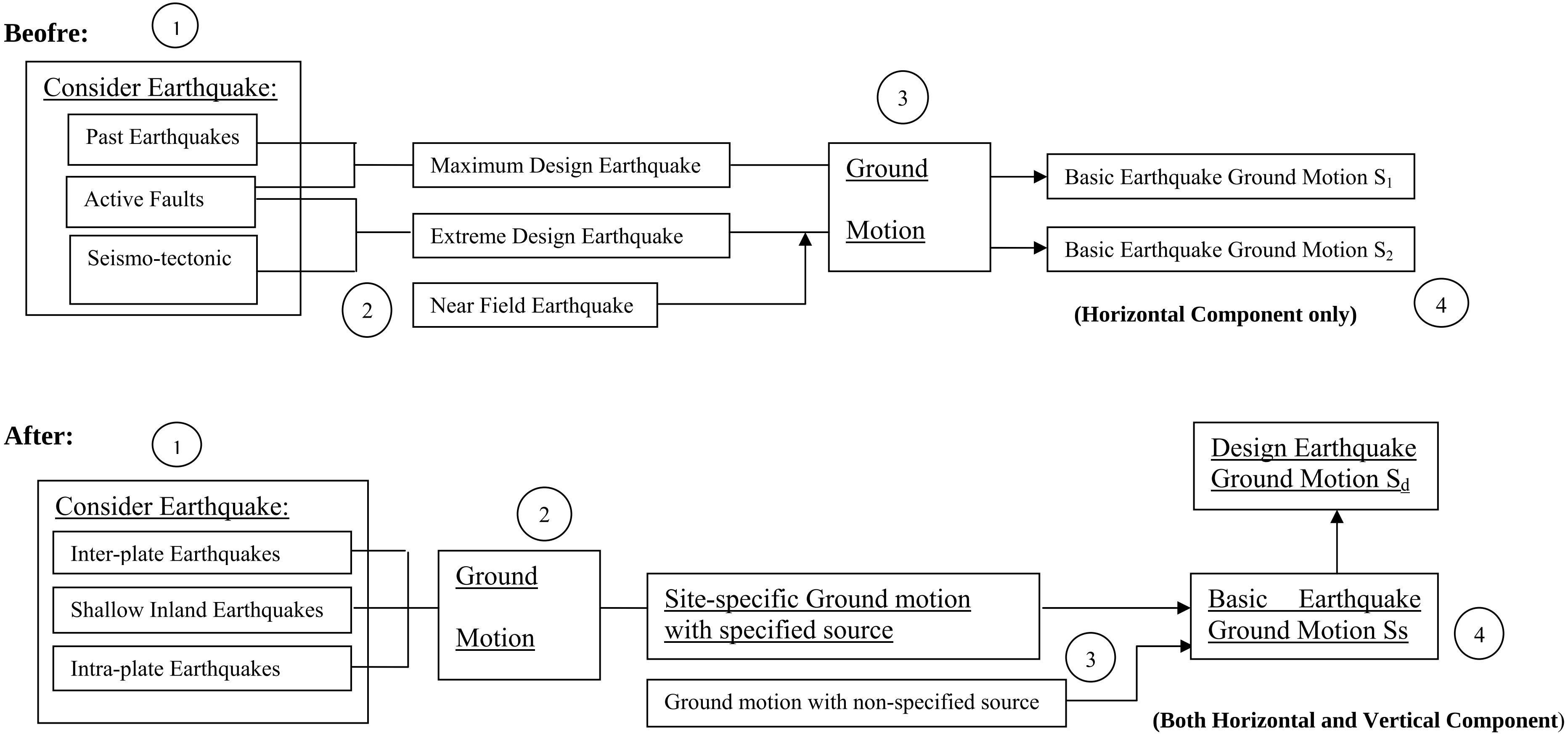


Figure 15. Flow Chart Design Bases Earthquake Research

1.2 DBE Definition – Earthquake Consideration

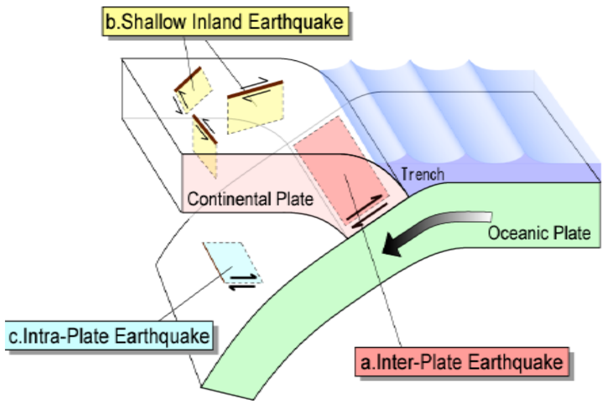
Before:
Consider With each research methods

- Earthquake documents
- Active faults research
- Seismicity near site

→

Past Earthquakes
Active Faults
Seismo-tectonic

Revised:
Consider with each source type



→

Inter-plate Earthquakes
Shallow Inland Earthquakes
Intra-plate Earthquakes

Figure 16. Source Types

1.3 DBE Definition – Ground Motion Evaluation

Before:
Empirical Methods (Response Spectrum Evaluation)

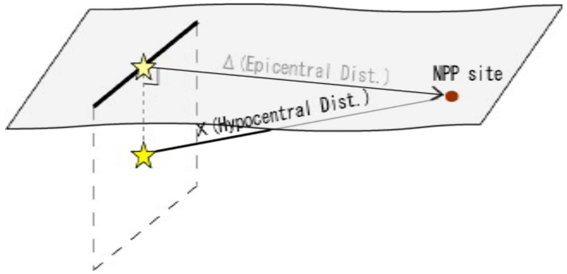


Figure 17. Response Spectrum Evaluation

Revised:
Empirical methods + Strong motion evaluation using Earthquake source model

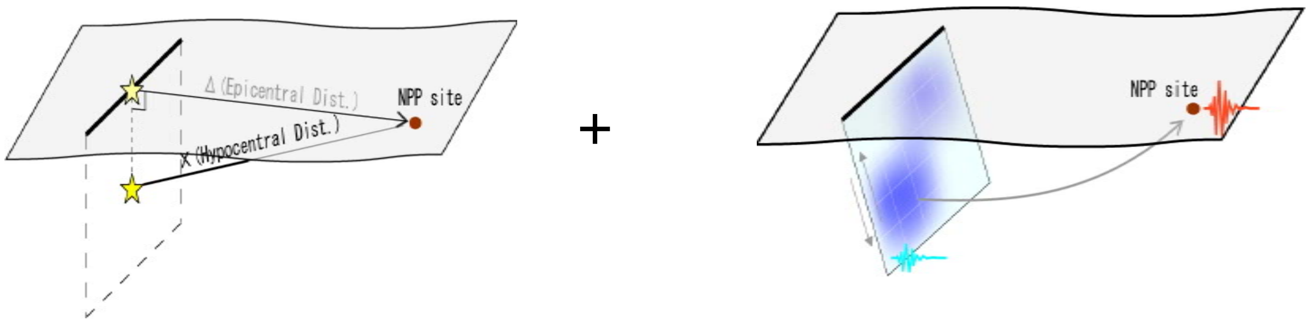


Figure 18. Response Spectrum Evaluation + Effect of Fault Plane

1.4 DBE Definition – Near-Field Earthquake

Before: Consider Near-field Earthquake by way of precaution.

Revised: Estimate the upper level of the ground motion due to the earthquakes source of which are difficult to specify in spite of detailed survey near the site, directly based on near-source strong motion records.

2. Active Faults Consideration

Before:

Consider the active faults that have activity in 50,000 years:

Active Fault of Low activity (Return period more than 50,000 years) → Consider as the source of S_2.

Active Fault of high activity (Return period more than 10,000 years) → Consider as the source of S_1.

Revised:

For S_s, consider the active faults that has activity in the late Pleistocene (referring to last Interglacial strata about 80,000 – 130,000 years ago)

Consider as the source of Inland Earthquakes for S_s.

3. Geological Survey

Revised:

- In land: Seismic profiling controlled seismic source.

-Off–Shore: Super Sonic Waves. (Over 10 km beneath the sea bottom can be searchable now)

4. Consideration of Vertical Seismic Force

Before: Consider Vertical Seismic Force as 1/2 as Horizontal, statically.

Revised: Consider Both Horizontal and Vertical Seismic Force dynamically.

5. Seismic Classification

Before:

There were four classes (As, A, B and C) (defined in 8.1.3 Table (7))

For which Class (As) designed with S_2 (Main Safety Function), also designed with S_1 (Remains within Elastic Limit).

Class A designed with S_1 (Remains within Elastic Limit). Table (12) Pre 2007 Seismic Classification and Table (13) Pre 2007 Load Combination and Allowable Limit.

Table 12. Pre 2007 Seismic Classification

Present		Example of Major Facilities	
Seismic Force	Aseismic Importance	BWR	PWR
Basic Earthquake Ground Motion S_2	As	-Containment Vessel -Control Rod -Residual Heat Removal System -Emergency Diesel Generator -Reactor Pressure Vessel,...etc	-Containment Vessel -Control Rod -Residual Heat Removal System -Emergency Diesel Generator -Reactor Vessel….etc
Basic earthquake ground motion S_1 or 3.0Ci [1] either large	A	-Emergency Corel Cooling System,..etc	-Safety injecting System,…etc
Seismic force of 1.5Ci	B	-Waste Disposal System -Turbine equipment[2],...etc	-Waste Disposal System,…etc
Seismic force of 1.0Ci	C	-Main Generator,...etc	-Main Generator -Turbine equipment[2] ,..etc

Notes 1: Ci Story shear coefficient to Static force required by civil code for non-nuclear structure
2: although turbine equipment is classified into C class according to a functional classification, turbine equipment of BWR is B class

Table 13. Pre 2007 Load Combination and Allowable Limit

Present			Facilities
Allowable Limit	Load Combination	Aseismic Importance	
Capability fully deformation (margin ductility) as a structure and appropriate safety margin to ultimate strength. Allowable Stress based on a suitable standard.	Basic earthquake ground motion S_2 and normal load …etc. Either basic earthquake ground motion S1 or static load and normal load, etc.	As	Building / Structures
Allowable stress based on suitable standard.	Basic earthquake ground motion S_1or static load and normal load,…etc	A	
Same as above	Static load and normal load ,… etc	B	
Same as above	Same as above	C	
Even when the structure of a portion carries out plastic deformation fairly, excessive modification a crack, breakage, etc arise and the function of facility is not affected. Yield stress or allowable limit or equivalent safety.	Basic Earthquake ground motion S_2 and operating load,…etc Basic earthquake ground motion S1 or static load and operating load,…etc	As	Equipment / Piping
Yield stress or the allowable limit of equivalent safety	Basic earthquake ground motion S_1 or static load and operating load , etc	A	
Allowable stress based on a suitable standard and standard	Static load and operating load,…etc	B	
Same as above	Same as above	C	

Revised:

Classes As and A are integrated into Class S. For which Class S designed with S_s (Main Safety Function), also designed with S_d (Remains within Elastic Limit). Where: $S_d = \alpha\ S_s$ ($\alpha \geq 0.5$), Table (14) and Table (15) Revised Load combination and allowable limit.

Has been changed into a higher rank from the present

Table 14. Revised Load combination and allowable limit 1/2

Example of Major Facilities		Revised	
BWR	PWR		
-Containment Vessel -Control Rod -Residual Heat Removal System -Emergency Diesel Generator -Reactor Pressure Vessel,...etc	-Containment Vessel -Control Rod -Residual Heat Removal System -Emergency Diesel Generator -Reactor Vessel,...etc	Aseismic Importance	Aseismic Importance
-Emergency Corel Cooling System, etc.	**-Safety injecting System, etc.**	S	S
-Waste Disposal System -Turbine equipment[2] , etc	-Waste Disposal System,…etc	B	B
-Main Generator, etc.	-Main Generator -Turbine equipment[2], etc	C	C

Table 15. Revised Load combination and allowable limit 2/2

Facilities	Revised		
	Aseismic Importance	Load combination	Allowable limit
Building / Structures	S	1. Basic earthquake ground motion Ss and normal load, etc. 2. Elastic Design ground motion Sd or static load and normal load, etc.	Same as present
	B	Same as present	Same as present
	C		
Equipment / Piping	S	1. Basic earthquake ground motion Ss and operating load, etc. 2.Elastic Design ground motion S_d or static load and operating load, etc.	1. Stress Analysis is same as the present. 2. The check of active component to basic earthquake ground motion S_s is based on comparison with acceleration using the actual probed examination.
	B	Same as present	Same as present
	C		

6. Consideration to the Phenomena Accompanying Earthquake

Before:

The concrete demand is not described.

The demand to the natural disaster of a landslide, tsunami or high tide and others is specified independently.

Revised:

The following factors should be taken into account in the seismic design:

1. Influence on the safety function on the facilities by collapse of circumference slope.
2. Influence on the safety function on the facilities by tsunami.

Table 16. Tsunami Situation/Procedure

Situation	Procedure
The Maximum Height of Tsunami associated with The Water Level at the Time of High Water	Height Installation of Plant Water Proof Design of Facilities or equipment, etc.
The minimum Water Level of the tsunami	Management by the design of the facilities or equipment, etc.

8.2 USA's Nuclear Structure Codes Review

8.2.1 Applicable standards

The design of NPPs follows, as expected, a much more rigorous design and evaluation process than conventional or even other critical storage facilities since the consequence of failure is a much more serious issue. The U.S. Nuclear Regulatory Commission developed the primary guidance documents that contain the Standard Review Plan NUREG-0800 associated with Regulatory Guides (such as RG 1.60). These guidance documents provide significant detail on design and analysis procedures required for safe system designs. Other developments undertaken by NRC over the years have addressed a number of issues with respect to plant design. Fortunately, this effort continues to improve assessment of uncertainties in these methods. More recent evaluations of site seismic hazard have been developed by the U. S. NRC and have made use of probabilistic hazard estimation to address issues of consistency in probability of non-exceedance in development of design response spectra (Regulatory Guide 1.165).

The primary differences in process design between NPPs and conventional facilities:

- The much longer return period used to develop the parameters of the design response spectrum and
- Prevention of any inelastic behavior in structural responses.

  The result of these two differences generally leads to extremely robust designs for Category I facilities (Seismic category I Structures, systems, and components that are designed and built to withstand the maximum potential earthquake stresses for the particular region where a nuclear plant shall be located).

8.2.2 Ground Response Spectra Definitions

The basic ground motion design spectrum has typically been defined at a probability of non-exceedance set at a level of median 1×10^{-5}, which is approximately equivalent to mean $1x10^{-4}$ (mean 10 000-yr event). This compares with the 500-yr event used for ordinary structures or 2500-yr events used for the some critical facilities. In the graded approach described in ASCE 43-05, the 10,000-yr event is considered only for the highest design category. Coupled with the requirement of elastic response, the NRC design process then corresponds to the most stringent design conditions considered in the graded approach used for design of critical facilities by other U.S. agencies.

In the original designs approved for older nuclear power plants, the basic input design response spectrum and corresponding enveloping ground motions were based on Reg. Guide 1.60 spectral shapes scaled to Peak Ground Acceleration (PGA) selected to match the site seismic hazard.

This spectral shape was selected independently of site condition (rock or deep soil, for example) and specified PGA. Figure (19) presents a plot of this spectral shape (scaled to a PGA of 0.2 g) together with the NUREG-0098 shape (N. M. Newmark and W. J. Hall 1978). This recommended spectrum was developed later in time and was used in re-evaluations of some older plants. Both are considered appropriate shapes to represent large magnitude events. As may be noted, the Reg. Guide 1.60 spectrum is significantly more conservative than the 0098 spectrum, particularly at frequencies less than 10 Hz, the frequency range of interest from a damage potential point of view. However, in addition to not being able to characterize anticipated site-specific behavior, these spectral shapes are hazard inconsistent; that is, unfortunately, from a hazard consistent point of view, these spectral shapes are not consistent; that is, the return period associated with the spectra is frequency dependent.

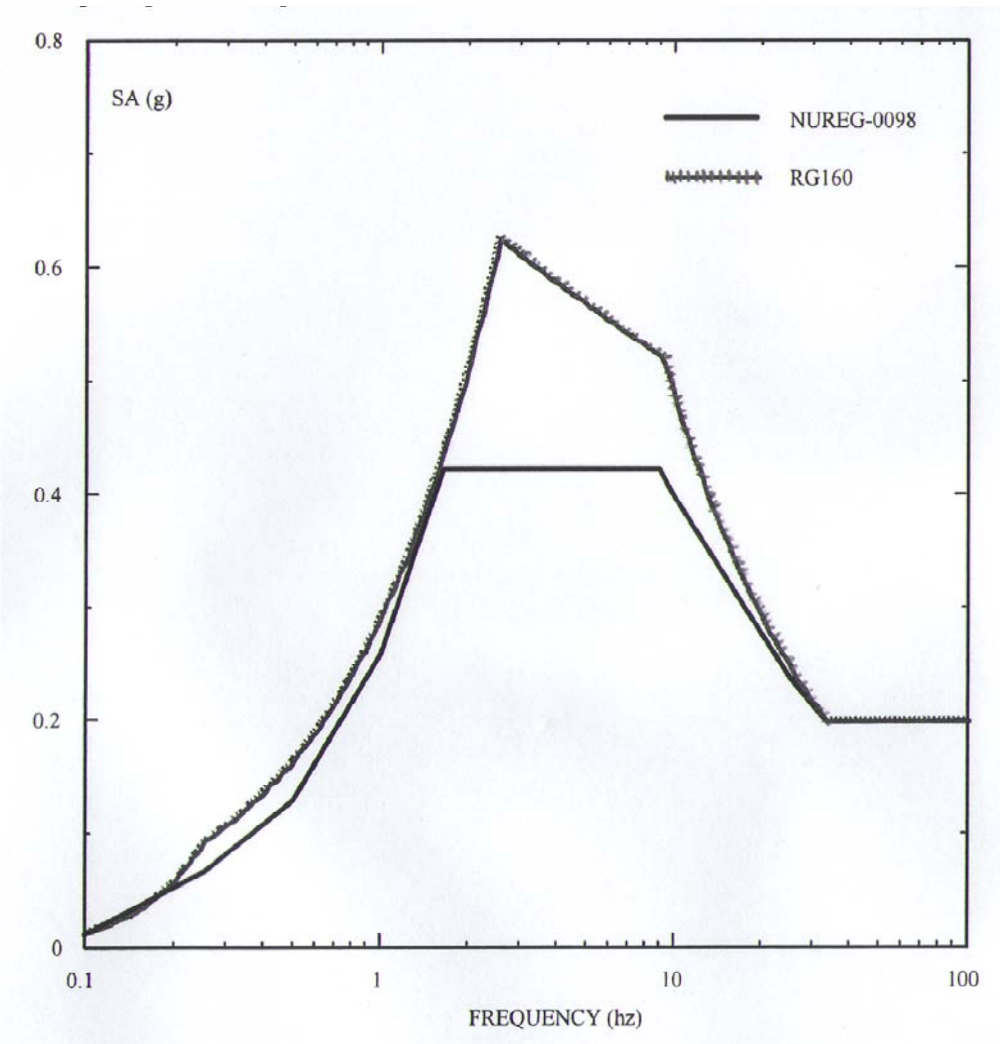


Figure 19. Comparison of Deterministic Spectra for Large Magnitude Events [34]
(5% damped spectral shapes used for evaluation of nuclear structures)

8.3 Comparison between Japan's and USA's Codes

8.3.1 Codes and Guides

A comparison between Japanese seismic design review guide of nuclear power reactor facilities according to Japan Electric Association Guide (JEAG) 4601 and the USA’s guide (ASME1998 Section III, Regulatory Guides) is shown in Table (17).

Table 17. Comparison of JEAG and USA Guides/Codes

Field	Japan			USA		
Design Floor Response Spectrum	10% Peak Broadening to absorb model or analysis uncertainty			Regulatory Guide 1.122 - Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components		
Loading Conditions		Reactor Vessel	Primary Piping		Reactor Vessel	Class 1 Piping
				Level B Limit, Upset (OBE)	1.5 S_m	1.8 S_m 1.5 S_y
	S_s Earthquake	1.5 S_y S_u	2.25 S_m	Level C Limit, Emergency	1.8 S_m 1.5 S_y	2.25 S_m 1.8 S_y
	S_d Earthquake	S_u	3S_m	Level D Limit, Faulted (SSE)	3.6 S_m S_u	3 S_m 2 S_y
Comparison Damping (%) Value		S_s, S_d		OBE (or 1/2 SSE)	SSE	
	<u>Concrete Structures</u> -Reinforced Concrete -Prestressed (PCCV)	5.0 3.0		4.0 2.0	7.0 5.0	
	Welded Steel Structures	1.0		2.0	4.0	
	Bolted Structures	2.0		4.0	7.0	
	<u>Piping</u> Function of type and number of supports, with and without thermal insulation.	0.5 to 2.5		Piping	OBE	SSE
				<u>RG1.61</u> Large Dia. 12 < D_o	2.0	3.0
				Small Dia. 12 > D_o	1.0	2.0
				<u>ASME Code Case N-411</u> Function of Frequency for Response Spectrum Analysis	5.0 (≤10 Hz) 2.0 (≥20Hz)	

It is important to note the following in relation to Table (17):

Basic Earthquake Ground Motion S_1: Past earthquakes and earthquakes caused by the active fault, which has been active for the past 10,000 years, are evaluated. It is the basic earthquake ground motion determined by the Maximum Design Earthquake which covers these earthquakes.

Basic Earthquake Ground Motion S_2: Earthquakes caused by the active fault that has been active for the past 50,000 years, seismotectonic structure and (M6.5) earthquake direct beneath the site are evaluated. It is the basic earthquake ground motion determined by the Extreme Design Earthquake which covers these earthquakes.

S_m: Design Stress Strength, S_y: Design Yield Point, S_u: Design Tensile Strength.

8.3.2 Load Combination Comparison between USA Guides and Japan Guides

Table (18) shows no difference between the Japan guides and USA guides in regard of loads and loads combinations.

Table 18. Japan’s load combination compared to USA ASME load combination

Load Category	Load Load Combination	Dead Load	Live Load	Prestress Loads	Normal Operating Pressure	Normal Operating Piping Loads	Normal Operating Thermal Loads	L-Accident Pressure	L-Accident Piping Load	L-Accident Thermal Load	Jet Force	S_1 Seismic Load	S_2 Seismic Load	Snow Load	Wind Load	Test Pressure	Equivalent to ASME BPV Sec.3Div2
I	Normal Operation	1.0	1.0	1.0	1.0	1.0	1.0										Normal
II	Safety Relief valve Operation	1.0	1.0	1.0	1.0	1.0	1.0										
	Testing	1.0	1.0	1.0												1.0	Test
	Snow	1.0	1.0	1.0	1.0	1.0	1.0							1.0			Normal
III	Storm	1.0	1.0	1.0	1.0	1.0	1.0								1.0		Severe
	S_1 Seismic	1.0	1.0	1.0	1.0	1.0	1.0					1.0					
	L(1) Accident	1.0	1.0	1.0				1.0	1.0	1.0							Abnormal
	L(2) Accident + S_1	1.0	1.0	1.0				1.0	1.0	1.0		1.0					Abnormal/Severe
IV	S2 Seismic	1.0	1.0	1.0	1.0	1.0							1.0				Extreme
	L(3) Accident	1.0	1.0	1.0				1.5	1.0								Abnormal
	L-Accident	1.0	1.0	1.0							1.0						Abnormal/Extreme
	L(4) Accident+ S_1	1.0	1.0	1.0				1.0	1.0			1.0					Abnormal/Severe
	L(5) Accident + Snow	1.0	1.0	1.0				1.25	1.0					1.25			
	L(5) Accident + Storm	1.0	1.0	1.0				1.25	1.0						1.25		

Notes:
-Safety relief valve operating condition in load category II is applicable to BWR only.
- L (1) accident condition in load category III includes peak loads immediately after LOCA.
- L (2) accident condition in load category III is long sustaining loading condition 10^{-1} year after LOCA which is combined with S_1.
- L (3) accident condition in load category IV is LOCA loading condition where 1.5 times the design pressure is taken into account.
- L (4) accident condition in load category IV is LOCA loading condition combined with S_1 where the maximum pressure and piping loads are taken into account.
- L(5) accident condition in load category IV is LOCA loading condition combined with Snow and Storm where 1.25 times the maximum pressure and piping loads are taken into account.

Chapter 9

CONTAINMENT ANALYSIS - CASE STUDY

A model of United States - Advanced Pressurized Water Reactor (US-APWR) is analyzed using STAAD.Pro2007 software under load combination from Table (4) page 46, Factored -Extreme Environmental:

Dead load (1) + Live Load (1) + Prestress Load (1) + Relife Valve Load (1) + Normal Normal Thermal Loads (1) + Safe Shut Down Earthquake Loads (1) + Normal Pipe Reaction Loads (1) + External Pressure Loads (1).

For the Safe Shut Down Earthquake Loads three different response spectrums were used Figure (20) Response Spectrums Charts; the first is the American response spectrum from Regulation 1.60 Ref.[28] (will be refered to it by Reg1.60), the second is AQABA1995 response spectrum based on records from Sinai/Egypt for 1995 gulf of Aqaba earthquake on a rock bed site (Hisham H. Mohammed and Magdy M. Wahba, 2005), and the third is mean response spectrum for Aqaba city (Hana Kabalawi, 1997) , covered the major earthquakes in north red sea zone including the Nov.22 1995 earthquake, with a moment magnitude M_W = 7.1 and a local magnitude M_L= 6.2, the mena Aqaba city response spectrum will be called (Study1997)

The Nov.22 1995 response spectra value was built after records obtained from two stations the first one located near the shore (in Study1997 Hotel Station) with soil characteristics medium alluvium formation and the second is the Civil Defense Station on land with shallow dense and stiff alluvium formation.

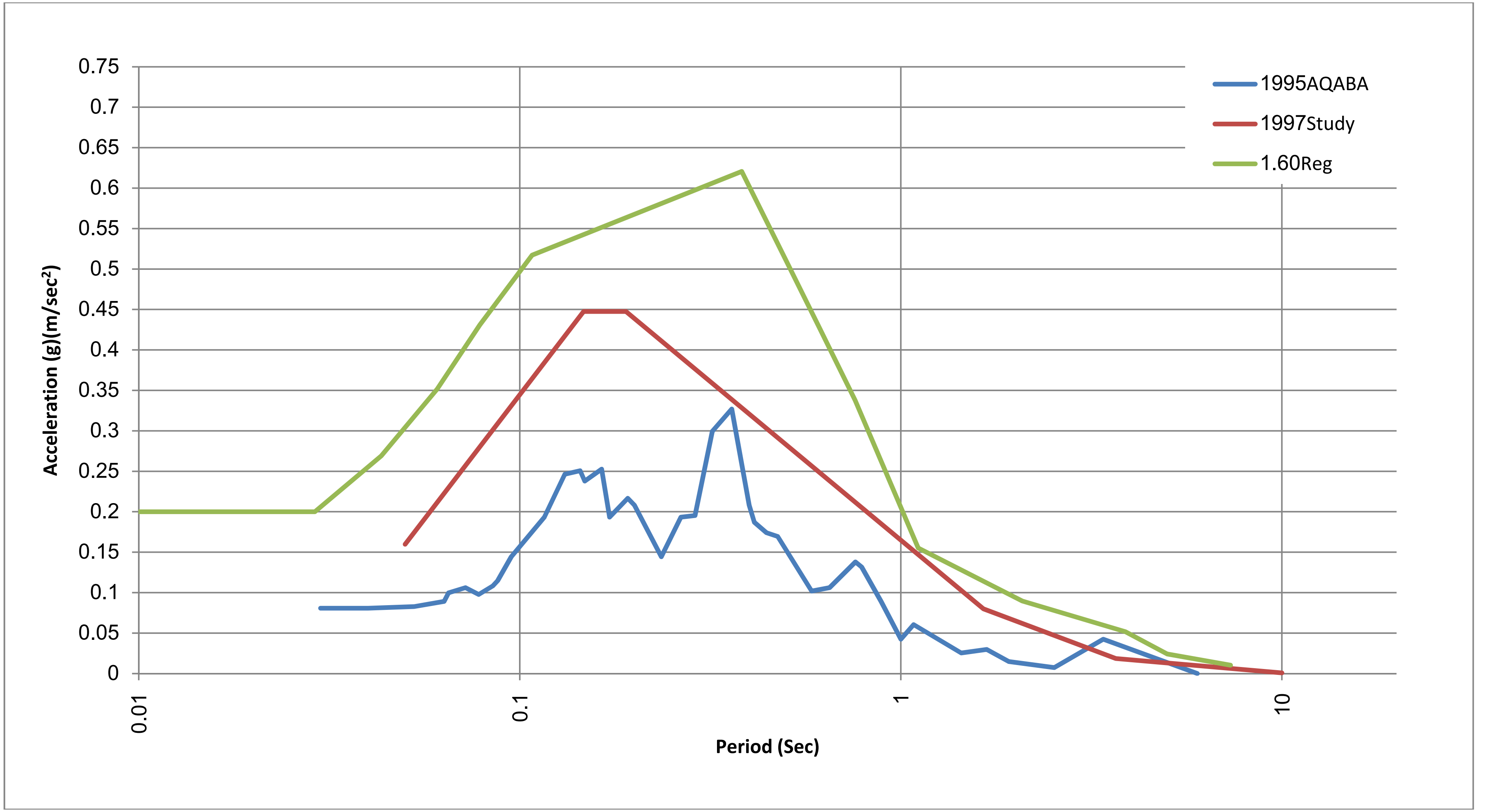


Figure 20. Response Spectrums Charts

The design specifications documents of US-APWR originally submitted to US-NRC for licensing and they are available at the NRC site: www.nrc.gov .

Some of the figures are presented to illustrate: Figure (21) Dimensions and Figure (22) Containment Penetrations.

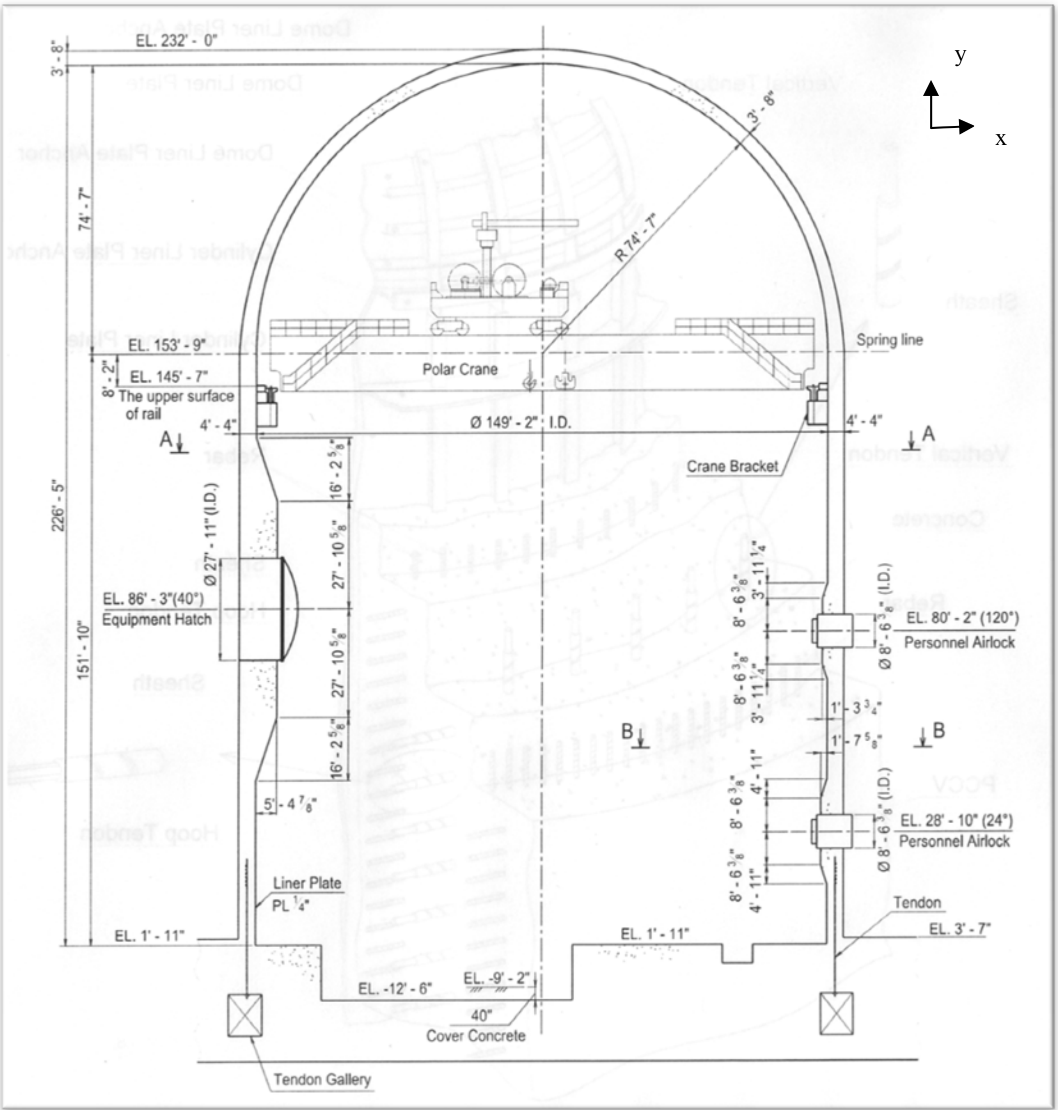


Figure 21. Dimensions of US-APWR containment

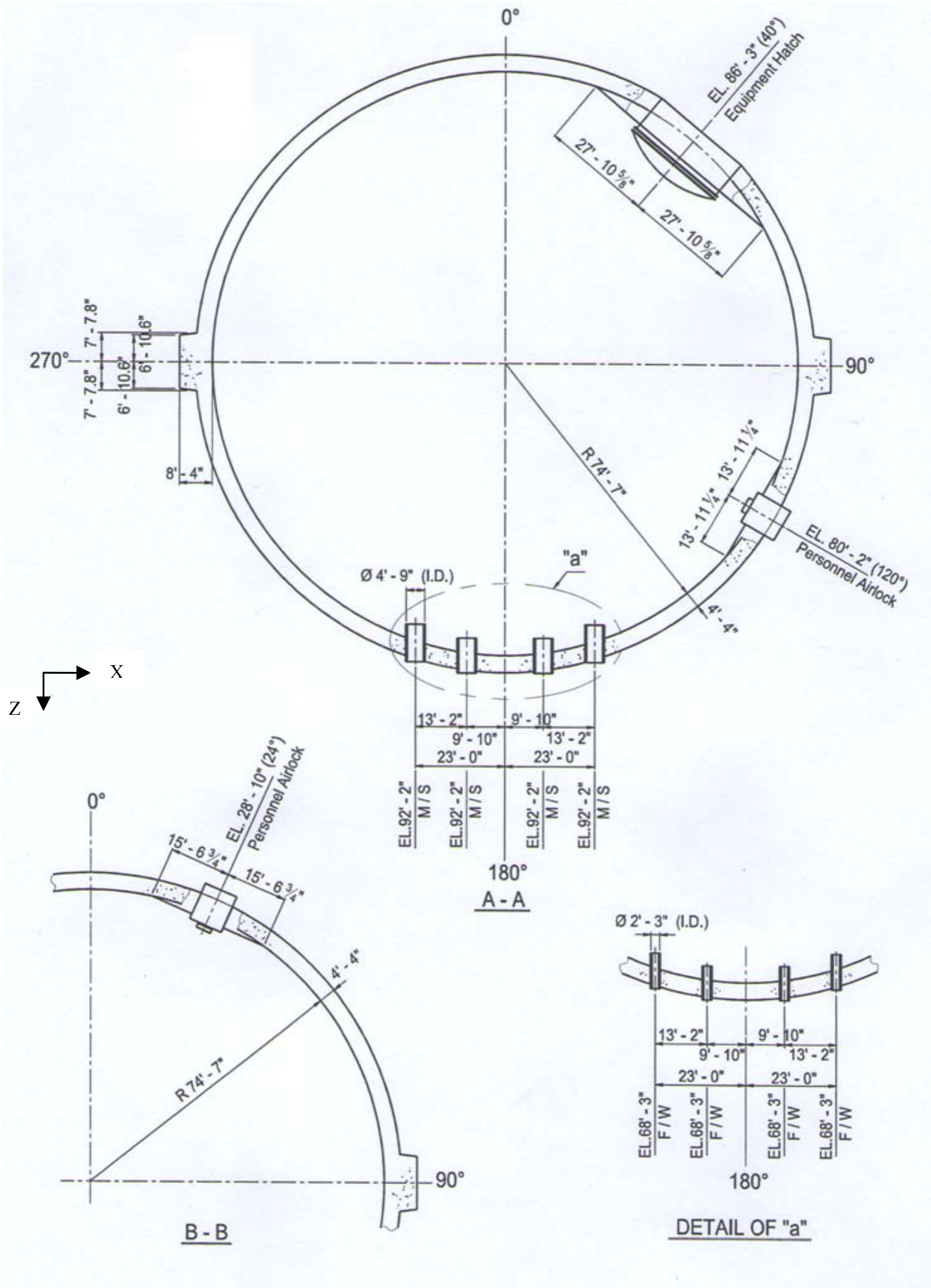


Figure 22.Containment Penetrations

The STAAD.Pro2007 model represents a simpler containment design showed in Figure (23) with global coordinates showed at the lowe left side, Figure (24) Model Penetrations. Model Dimensions Table (19), and Table (20) Concrete Properties.

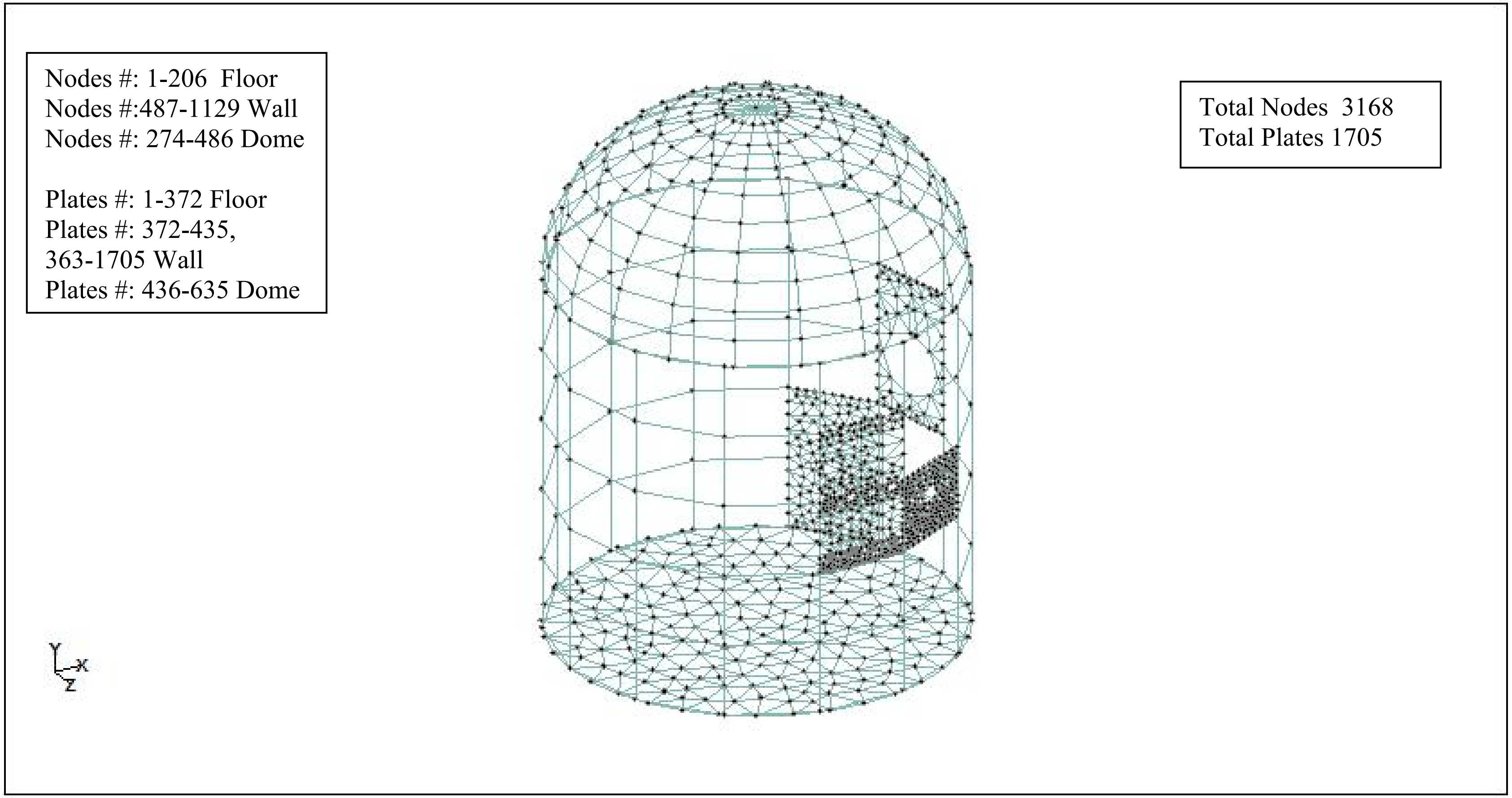


Figure 23. STAAD.Pro2007 Model

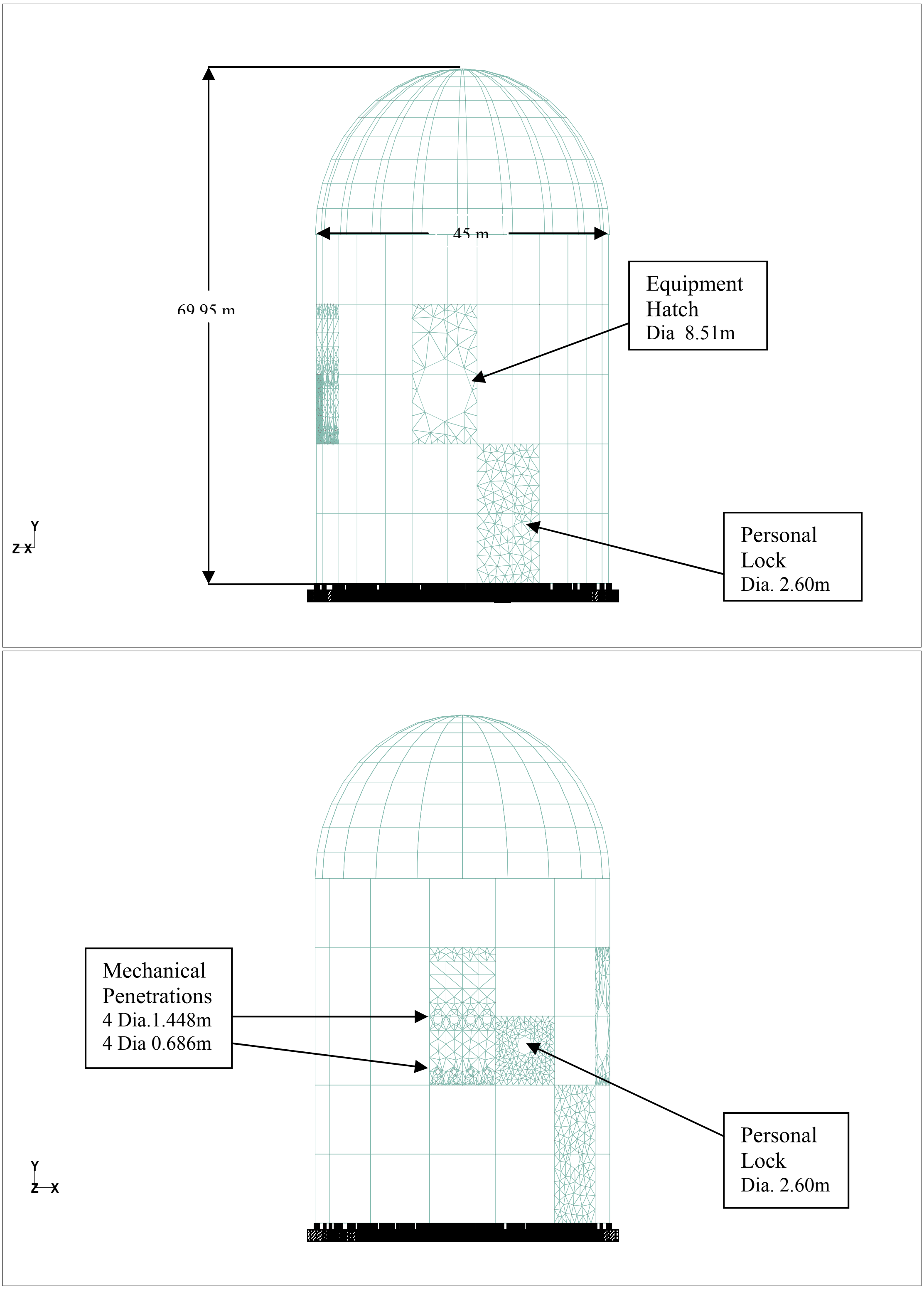


Figure 24. Model Penetrations

Table 19. Model Dimensions

Description	Dimension
Height	69.95 m
Diameter	45 m
Spherical Cap Radius	22.5 m
Concrete Shell Thickness	Dome 1.15 m Side Wall assumed an average of 2 m Transition Section between Dome and Side Wall 2 m to 1.12 m
Floor Thickness	4 m

Table 20. Concrete properties

Concrete Properties	Value
Concrete Strength f_c'	7000 psi (48.3 = 49 MPa)
Modulus of Elasticity E	4769 ksi (32664 MPa)
Poisson's Ratio ν	0.17
Thermal Expansion Coefficient	0.99×10^{-5}/°C
Unit Weight γ	150lb/ft^3 (2400 kg/m^3)

- Linear analysis was performed only to stay in the elastic region.

-The nuclear containment rests on the nuclear island which is considered as the foundation, to model the containment only (not the whole nuclear island) the Fixed Supports was used at the floor nodes.

- The analysis was performed without the liner, adding the liner (0.006 mm Steel) to the Staad model will lead to analyze the liner only without the concrete containment.

The following Loads were considered:

- Dead Load in (-Y) direction.

- Live Load = 200 lb/ft^2 (during normal operation) = 10 kPa on the floor Figure (25) Live Load.

- Thermal Load =10C^o difference between the out and inside the containment Figure (26) Thermal Load.

- Prestressing Load were assumed to equale the minimum requirment = D + 45psi (310kPa) Ref.[31] although steel tendons, paths / pattern and the details of the nuclear island were not avaliable. Figure (27) Prestress Load

- External Pressure = 100kPa on the outer walls Figure (28) External Pressure.

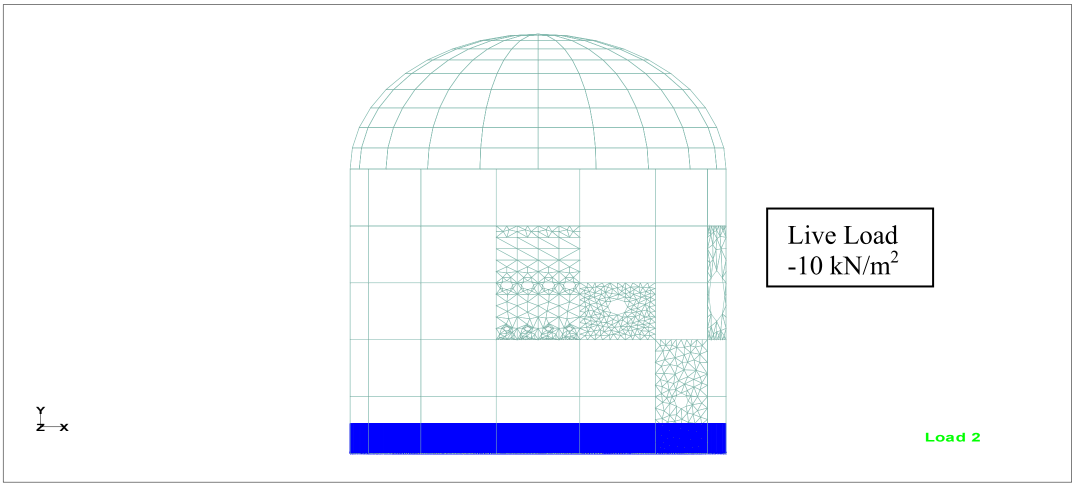


Figure 25. Live Load

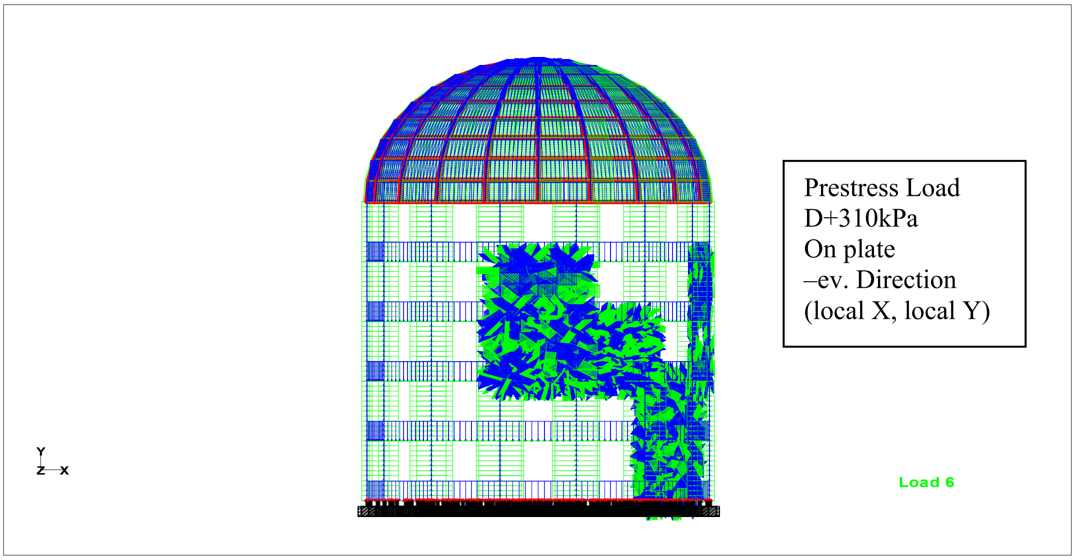


Figure 26. Prestress Load

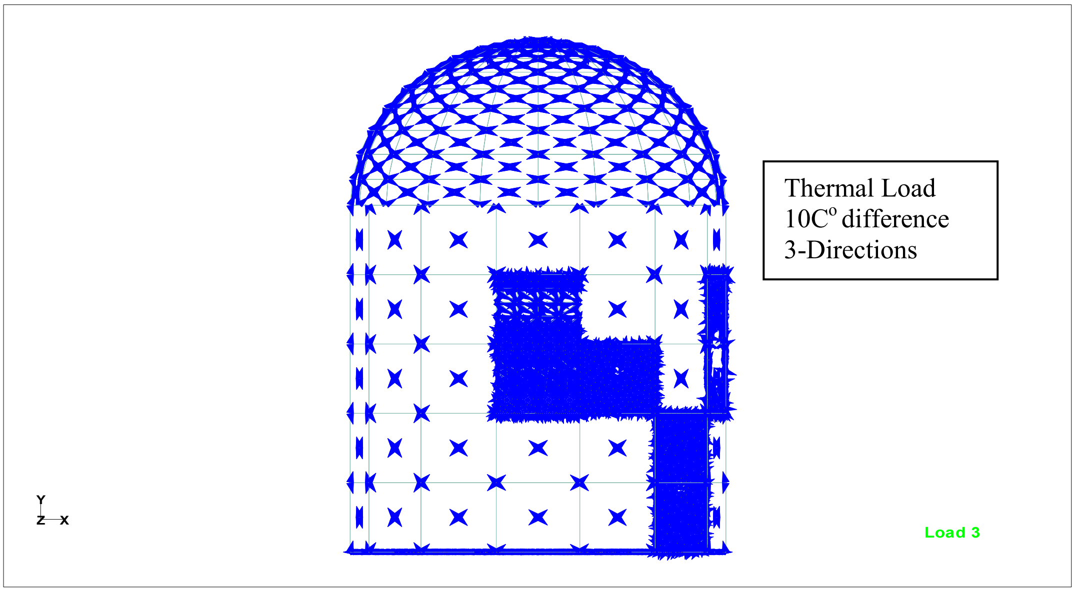


Figure 27. Thermal load

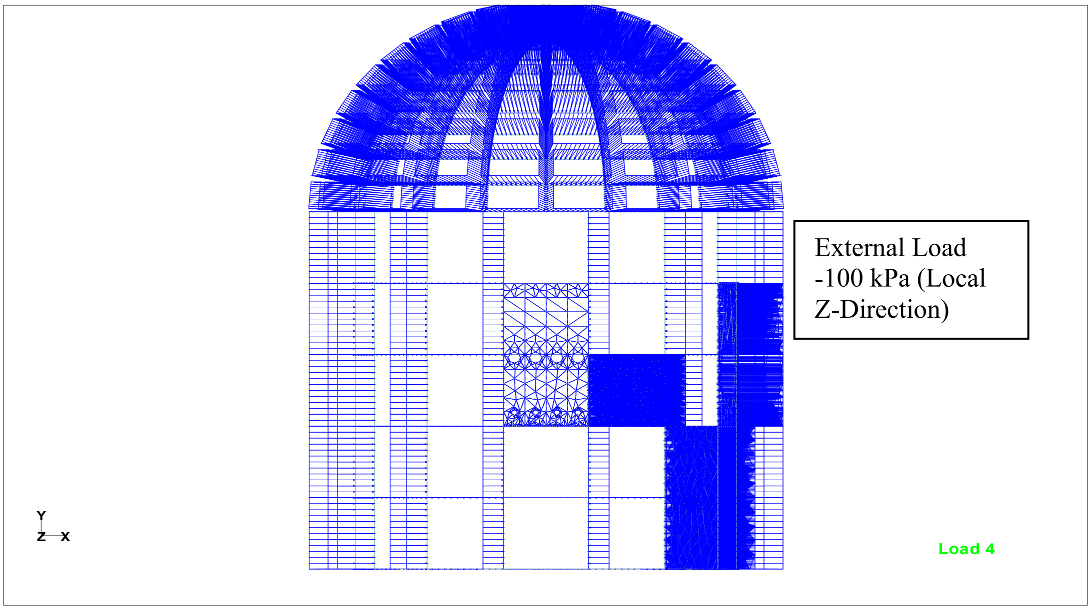


Figure 28. External Load

- Safe Shut Down Earthquake = using response spectrum analysis Ref. [26], for the two horizontal and one vertical components were combined based on 100 - 40 - 40 Ref.[30].

Damping Ratio = 5%

Directions:H1= Direction 1 = Plant east-west = Global X-axis
H2= Direction 2 = Plant north-south = Global Z-axis
V= Direction 3 = Vertical Up-Down = Global Y-axis

Two cases for each response spectrum X-Y-Z 100- 40- 40, Z-Y-X 100- 40- 40.

- Due to insufficient data related to Relife Valve Load and Normal Pipe Reaction Load their values were set eqaule Zero.

- The analysis was first performed with Prestress Load (G) then it was performed without that laod.

- Dead Load: DL, Live Load: LL, Thernmal Load: TL, External Load: EL

Earthquake Load: EQL, Prestress Load: PRESTRESSL,

Load Combination: COMBINATION

RESULTS

Table 21. Structure Frequencies X-Direction Dominates

Mode	Frequency (Hz)	Period (seconds)	Participation X (%)	Participation Y (%)	Participation Z (%)
1	1.342	0.745	25.288	0.000	0.000
2	1.961	0.510	0.000	0.000	0.000
3	5.413	0.185	0.074	0.000	0.000
4	6.123	0.163	0.092	0.000	0.000
5	7.405	0.135	11.525	0.000	0.000
6	7.729	0.129	0.348	0.000	0.000

Table 22. Structure Frequencies Z-Direction Dominates

Mode	Frequency (Hz)	Period (seconds)	Participation X (%)	Participation Y (%)	Participation Z (%)
1	0.778	1.286	0.000	0.000	18.003
2	5.191	0.193	0.000	0.000	0.045
3	5.348	0.187	0.000	0.000	2.521
4	5.888	0.170	0.000	0.000	15.923
5	6.163	0.162	0.000	0.000	0.083
6	6.954	0.144	0.000	0.000	0.062

STAAD.Pro 2007 Result for the load combination with the Prestressed Load:

Q: Shear Stress, S: Membrane (Axial) Stress, M: Bending Moment, (t): top, (b): bottom.
E 3 = x1000

Table 23. Nodal Displacement Prestressed Load X-Direction Reg. 1.60 1st run

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	303	7:COMBINATI(	**1.93E 3**	191.589	-123.716	1.95E 3	0.027	0.023	0.097
Min X	322	6:PRESTRESS	**-1.35E 3**	311.165	70.826	1.39E 3	-0.034	0.013	-0.087
Max Y	320	6:PRESTRESS	-1.28E 3	**417.454**	22.643	1.34E 3	-0.046	-0.047	-0.088
Min Y	311	7:COMBINATI(	1.66E 3	**-1.03E 3**	544.826	2.03E 3	-0.070	-0.050	0.068
Max Z	309	7:COMBINATI(	1.75E 3	-971.240	**683.285**	2.11E 3	-0.068	0.001	0.074
Min Z	289	7:COMBINATI(	-1.12E 3	-870.322	**-788.474**	1.62E 3	0.067	0.001	-0.066
Max rX	292	7:COMBINATI(	-1.02E 3	-932.352	-636.240	1.52E 3	**0.070**	-0.050	-0.059
Min rX	311	7:COMBINATI(	1.66E 3	-1.03E 3	544.826	2.03E 3	**-0.070**	-0.050	0.068
Max rY	305	7:COMBINATI(	1.79E 3	-218.406	153.168	1.81E 3	-0.008	**0.049**	0.084
Min rY	299	6:PRESTRESS	1.19E 3	271.330	63.834	1.23E 3	0.029	**-0.106**	0.052
Max rZ	301	7:COMBINATI(	1.9E 3	378.057	-138.404	1.94E 3	0.042	-0.035	**0.100**
Min rZ	320	7:COMBINATI(	-1.28E 3	380.756	20.788	1.34E 3	-0.041	-0.036	**-0.092**
Max Rst	309	7:COMBINATI(	1.75E 3	-971.240	683.285	**2.11E 3**	-0.068	0.001	0.074

Table 24. Nodal Displacement Prestressed Load X-Direction Reg. 1.60 2nd run

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	299	7:COMBINATI(	**276.248**	-46.640	2.859	280.172	0.002	0.004	0.019
Min X	315	7:COMBINATI(	**-176.664**	-42.523	8.233	181.896	0.002	0.003	-0.015
Max Y	311	6:EQL	70.439	**16.143**	21.671	75.444	0.002	0.000	0.002
Min Y	307	7:COMBINATI(	73.144	**-325.431**	299.167	448.059	-0.023	0.002	0.002
Max Z	307	7:COMBINATI(	73.144	-325.431	**299.167**	448.059	-0.023	0.002	0.002
Min Z	288	7:COMBINATI(	76.069	-311.269	**-294.958**	435.517	0.023	0.002	0.001
Max rX	288	7:COMBINATI(	76.069	-311.269	-294.958	435.517	**0.023**	0.002	0.001
Min rX	307	7:COMBINATI(	73.144	-325.431	299.167	448.059	**-0.023**	0.002	0.002
Max rY	322	7:COMBINATI(	10.160	-184.032	-117.539	218.601	0.014	**0.020**	-0.006
Min rY	292	7:COMBINATI(	149.737	-189.575	-122.043	270.656	0.014	**-0.019**	0.009
Max rZ	299	7:COMBINATI(	276.248	-46.640	2.859	280.172	0.002	0.004	**0.019**
Min rZ	318	7:COMBINATI(	-176.252	-37.925	-1.059	180.289	-0.001	0.004	**-0.016**
Max Rst	307	7:COMBINATI(	73.144	-325.431	299.167	**448.059**	-0.023	0.002	0.002

Table 25. Nodal Displacement Prestressed Load Z-Direction Reg. 1.60 1st run

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	303	7:COMBINATI(	**1.84E 3**	266.637	-111.800	1.86E 3	0.029	0.025	0.095
Min X	322	7:COMBINATI(	**-1.37E 3**	299.713	10.006	1.4E 3	-0.023	0.024	-0.090
Max Y	320	7:COMBINATI(	-1.34E 3	**429.539**	17.763	1.41E 3	-0.039	-0.037	-0.093
Min Y	311	7:COMBINATI(	1.56E 3	**-958.370**	555.063	1.92E 3	-0.067	-0.048	0.065
Max Z	309	7:COMBINATI(	1.65E 3	-866.858	**715.336**	1.99E 3	-0.064	0.001	0.071
Min Z	289	7:COMBINATI(	-1.2E 3	-772.227	**-787.360**	1.63E 3	0.071	0.000	-0.068
Max rX	292	7:COMBINATI(	-1.1E 3	-859.345	-648.947	1.54E 3	**0.073**	-0.048	-0.061
Min rX	311	7:COMBINATI(	1.56E 3	-958.370	555.063	1.92E 3	**-0.067**	-0.048	0.065
Max rY	305	7:COMBINATI(	1.7E 3	-111.899	186.905	1.71E 3	-0.003	**0.048**	0.081
Min rY	299	6:PRESTRESS	1.19E 3	271.330	63.834	1.23E 3	0.029	**-0.106**	0.052
Max rZ	301	7:COMBINATI(	1.83E 3	424.914	-129.452	1.89E 3	0.044	-0.036	**0.099**
Min rZ	320	7:COMBINATI(	-1.34E 3	429.539	17.763	1.41E 3	-0.039	-0.037	**-0.093**
Max Rst	309	7:COMBINATI(	1.65E 3	-866.858	715.336	**1.99E 3**	-0.064	0.001	0.071

Table 26. Nodal Displacement Prestressed Load Z-Direction Reg. 1.60 2nd run

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	299	4:EL	**62.723**	-13.632	0.264	64.188	0.000	0.000	0.005
Min X	318	4:EL	**-58.068**	-11.855	-1.468	59.284	-0.001	0.000	-0.005
Max Y	307	6:EQL	4.665	**90.801**	13.405	91.904	0.004	0.000	0.000
Min Y	307	4:EL	1.989	**-89.400**	81.524	121.006	-0.006	0.000	0.000
Max Z	486	6:EQL	0.214	0.316	**89.039**	89.039	0.004	0.000	0.000
Min Z	288	4:EL	2.533	-86.544	**-82.369**	119.503	0.006	0.000	-0.000
Max rX	288	4:EL	2.533	-86.544	-82.369	119.503	**0.006**	0.000	-0.000
Min rX	307	4:EL	1.989	-89.400	81.524	121.006	**-0.006**	0.000	0.000
Max rY	303	4:EL	20.534	-57.968	37.623	72.093	-0.003	**0.005**	0.002
Min rY	311	4:EL	-16.233	-57.755	38.654	71.367	-0.003	**-0.005**	-0.002
Max rZ	299	4:EL	62.723	-13.632	0.264	64.188	0.000	0.000	**0.005**
Min rZ	318	4:EL	-58.068	-11.855	-1.468	59.284	-0.001	0.000	**-0.005**
Max Rst	307	4:EL	1.989	-89.400	81.524	**121.006**	-0.006	0.000	0.000

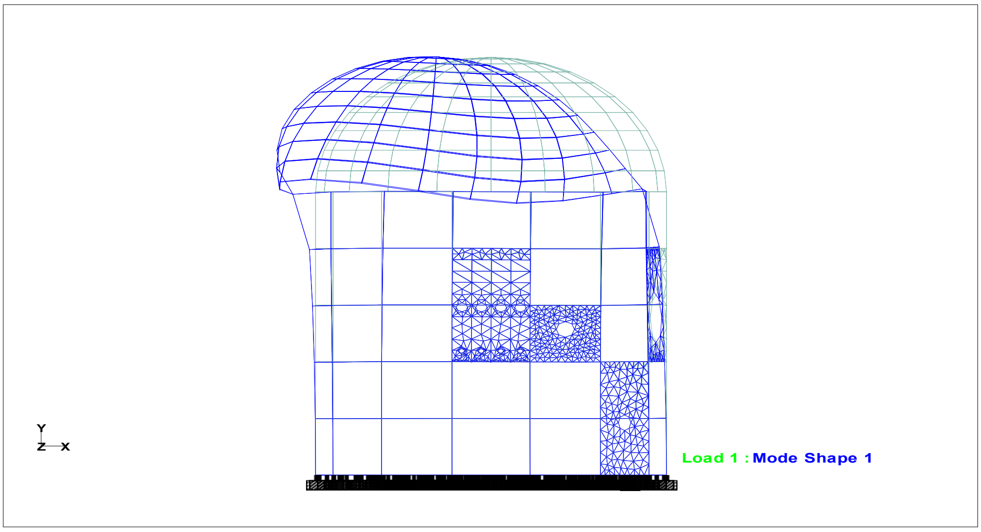


Figure 29. STAAD 1st Mode Shape X-Direction

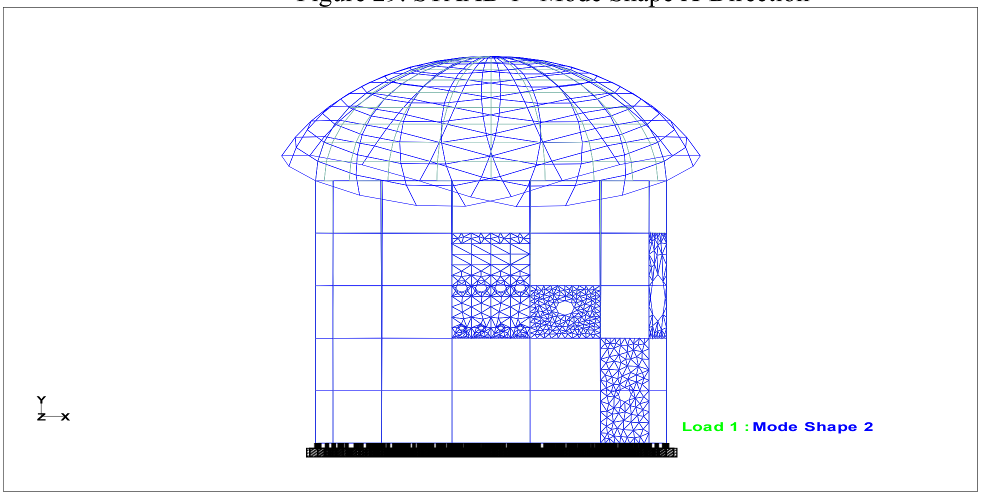


Figure 30. STAAD 2nd Mode Shape X-Direction

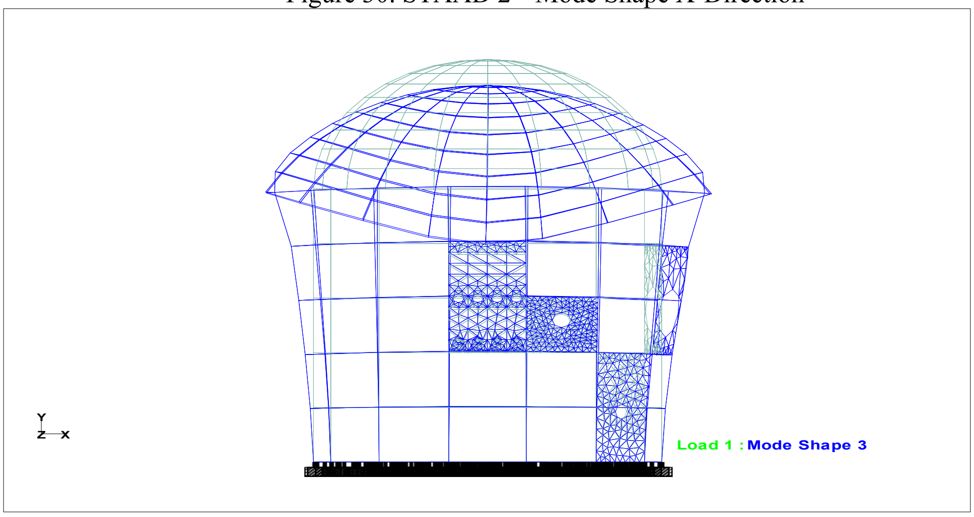


Figure 31. STAAD 3rd Mode Shape X-Direction

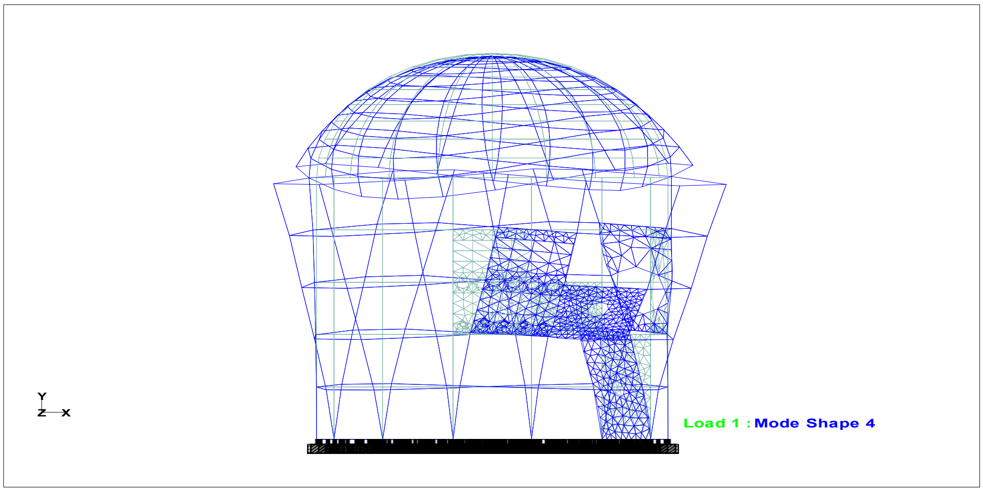


Figure 32. STAAD 4th Mode Shape X-Direction

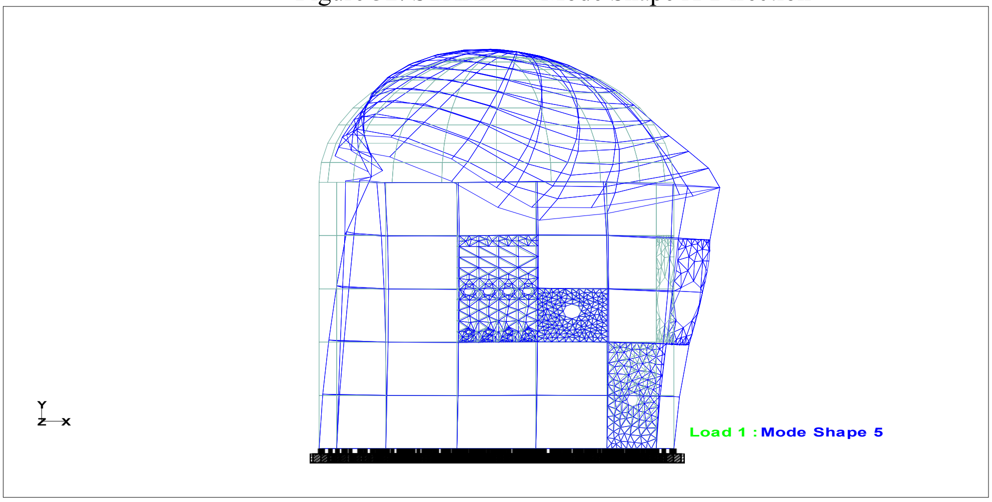


Figure 33. STAAD 5th Mode Shape X-Direction

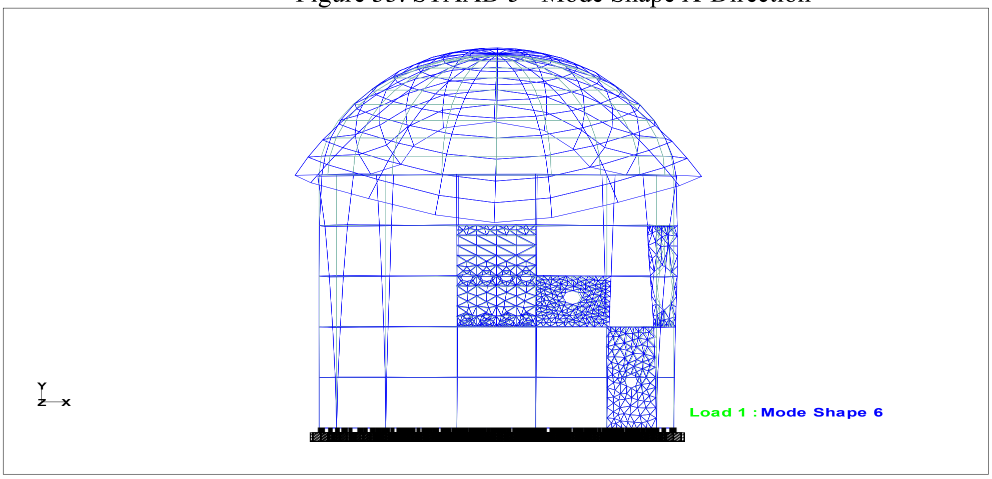


Figure 34. STAAD 6th Mode Shape X-Direction

STAAD.Pro2007 Results for the load combination without the Prestressed Load:

Table 27. Nodal Displacement X-Direction Reg. 1.60

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	294	6:COMBINATI(	**118.124**	-19.470	1.374	119.726	0.003	-0.003	0.005
Min X	315	4:ExternalLoad	**-47.589**	-9.527	1.052	48.545	0.001	0.000	-0.004
Max Y	311	5:EQLoad	62.503	**15.503**	20.557	67.599	0.002	0.000	0.002
Min Y	307	6:COMBINATI(	59.976	**-107.297**	106.149	162.411	-0.009	0.002	0.002
Max Z	307	6:COMBINATI(	59.976	-107.297	**106.149**	162.411	-0.009	0.002	0.002
Min Z	288	6:COMBINATI(	60.945	-103.848	**-103.871**	159.021	0.009	0.002	0.001
Max rX	324	6:COMBINATI(	58.781	-82.758	-71.669	124.260	**0.009**	0.006	0.001
Min rX	307	6:COMBINATI(	59.976	-107.297	106.149	162.411	**-0.009**	0.002	0.002
Max rY	320	6:COMBINATI(	-2.841	-18.412	2.377	18.781	0.003	**0.008**	-0.004
Min rY	292	6:COMBINATI(	91.103	-49.469	-27.003	107.127	0.006	**-0.007**	0.004
Max rZ	275	6:COMBINATI(	-32.340	-2.424	0.213	32.432	0.000	-0.000	**0.010**
Min rZ	281	4:ExternalLoad	31.556	-6.428	-0.627	32.210	-0.000	-0.000	**-0.004**
Max Rst	307	6:COMBINATI(	59.976	-107.297	106.149	**162.411**	-0.009	0.002	0.002

Table 28. Max. Stress Summary X-Direction Reg. 1.60

			Shear		Membrane			Bending		
	Plate	L/C	Qx (N/mm^2)	Qy (N/mm^2)	Sx (N/mm^2)	Sy (N/mm^2)	Sxy (N/mm^2)	Mx (kNm/m)	My (kNm/m)	Mxy (kNm/m)
Max Qx	451	6:COMBINATI(	**3.819**	1.329	-15.591	-7.172	-7.620	2.22E 3	4.15E 3	2.99E 3
Min Qx	648	6:COMBINATI(	**-1.680**	1.564	2.467	0.956	-0.674	-1.74E 3	-8.45E 3	3.34E 3
Max Qy	1652	6:COMBINATI(	-1.510	**4.868**	-0.426	-13.729	-2.160	7.67E 3	23.5E 3	-13.3E 3
Min Qy	1691	6:COMBINATI(	0.796	**-1.953**	-0.522	2.300	-0.566	-3.57E 3	-1.02E 3	2.7E 3
Max Sx	1254	6:COMBINATI(	0.218	-0.439	**6.190**	1.352	-0.031	-20.4E 3	-2.71E 3	6E 3
Min Sx	441	6:COMBINATI(	3.800	1.275	**-15.681**	-7.211	-7.661	2.22E 3	4.21E 3	3.06E 3
Max Sy	1059	6:COMBINATI(	-0.132	2.133	0.937	**7.500**	0.117	-2.39E 3	-22.6E 3	-7.27E 3
Min Sy	1652	6:COMBINATI(	-1.510	4.868	-0.426	**-13.729**	-2.160	7.67E 3	23.5E 3	-13.3E 3
Max Sxy	1608	6:COMBINATI(	2.018	1.705	-6.925	-5.674	**9.949**	18.7E 3	-6.61E 3	28.1E 3
Min Sxy	441	6:COMBINATI(	3.800	1.275	-15.681	-7.211	**-7.661**	2.22E 3	4.21E 3	3.06E 3
Max Mx	1608	6:COMBINATI(	2.018	1.705	-6.925	-5.674	9.949	**18.7E 3**	-6.61E 3	28.1E 3
Min Mx	724	6:COMBINATI(	0.537	0.103	0.291	-0.614	-0.446	**-24.5E 3**	-6.04E 3	-8.3E 3
Max My	1660	6:COMBINATI(	-0.526	3.787	0.080	-0.026	-0.507	2E 3	**25.2E 3**	-4E 3
Min My	686	6:COMBINATI(	-0.573	1.768	-0.727	-1.929	0.753	-2.65E 3	**-37.4E 3**	14.7E 3
Max Mxy	1608	6:COMBINATI(	2.018	1.705	-6.925	-5.674	9.949	18.7E 3	-6.61E 3	**28.1E 3**
Min Mxy	1652	4:ExternalLoad	-1.461	2.716	-0.643	-12.661	-1.223	5.2E 3	21.1E 3	**-16.4E 3**

Table 29. Max. Principal Stress Summary X-Direction Reg. 1.60

			Principal		Von Mis		Tresca	
	Plate	L/C	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)
Max (t)	1608	6:COMBINATI(	**24.945**	-24.945	36.446	12.688	41.826	13.970
Max (b)	1608	6:COMBINATI(	24.945	**-24.945**	36.446	12.688	41.826	13.970
Max VM (t)	1608	6:COMBINATI(	24.945	-24.945	**36.446**	12.688	41.826	13.970
Max VM (b)	441	6:COMBINATI(	15.317	-7.453	20.108	**47.044**	22.771	49.137
Tresca (t)	1608	6:COMBINATI(	24.945	-24.945	36.446	12.688	**41.826**	13.970
Tresca (b)	441	6:COMBINATI(	15.317	-7.453	20.108	47.044	22.771	**49.137**

Table 30. Reaction Summary X-Direction Reg. 1.60

			Horizontal	Vertical	Horizontal	Moment		
	Node	L/C	FX (kN)	FY (kN)	FZ (kN)	MX (kNm)	MY (kNm)	MZ (kNm)
Max FX	203	6:COMBINATI(	**51.5E 3**	164E 3	230.172	-680.098	778.862	-213E 3
Min FX	222	3:ThermalLoad	**-46.3E 3**	17.9E 3	-6.52E 3	60.6E 3	46.2E 3	104E 3
Max FY	6	6:COMBINATI(	-43.6E 3	**165E 3**	14.2E 3	-14.3E 3	-8.49E 3	223E 3
Min FY	226	4:ExternalLoad	-2.69E 3	**-40.1E 3**	-10.7E 3	-34.3E 3	2.18E 3	-8.38E 3
Max FZ	220	6:COMBINATI(	-2.22E 3	94.1E 3	**50.5E 3**	17.8E 3	-44E 3	126E 3
Min FZ	213	3:ThermalLoad	-11.3E 3	-2.91E 3	**-46.4E 3**	-224E 3	73.723	53.9E 3
Max MX	226	3:ThermalLoad	7.71E 3	68.988	46.4E 3	**223E 3**	915.437	-54.2E 3
Min MX	213	3:ThermalLoad	-11.3E 3	-2.91E 3	-46.4E 3	**-224E 3**	73.723	53.9E 3
Max MY	222	3:ThermalLoad	-46.3E 3	17.9E 3	-6.52E 3	60.6E 3	**46.2E 3**	104E 3
Min MY	220	6:COMBINATI(	-2.22E 3	94.1E 3	50.5E 3	17.8E 3	**-44E 3**	126E 3
Max MZ	217	6:COMBINATI(	-41.3E 3	136E 3	-16.5E 3	-32E 3	2.18E 3	**231E 3**
Min MZ	203	3:ThermalLoad	47.9E 3	706.455	211.359	-1.72E 3	349.999	**-230E 3**

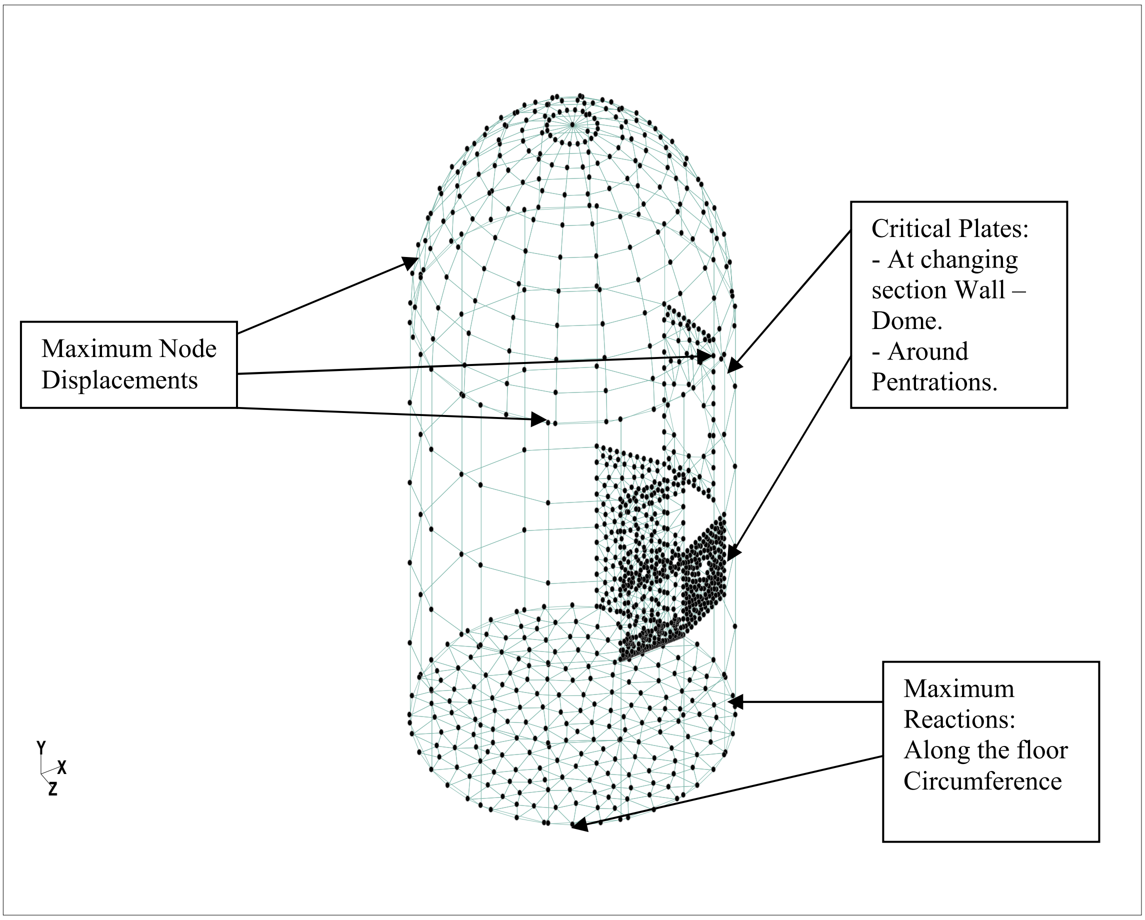


Figure 35. X-D Reg1.60 Stress/Displacement Location

Table 31. Nodal Displacement Z-Direction Reg. 1.60

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	299	6:COMBINATI(	**102.346**	9.883	1.391	102.831	0.003	0.001	0.007
Min X	318	6:COMBINATI(	**-83.936**	13.035	-0.019	84.942	0.001	0.001	-0.006
Max Y	307	5:EQL	4.665	**90.801**	13.405	91.904	0.004	0.000	0.000
Min Y	307	4:EL	1.989	**-89.400**	81.524	121.006	-0.006	0.000	0.000
Max Z	307	6:COMBINATI(	7.479	-42.415	**135.948**	142.607	-0.006	0.000	0.000
Min Z	288	6:COMBINATI(	8.451	-37.914	**-110.541**	117.167	0.014	0.000	0.000
Max rX	288	6:COMBINATI(	8.451	-37.914	-110.541	117.167	**0.014**	0.000	0.000
Min rX	307	4:EL	1.989	-89.400	81.524	121.006	**-0.006**	0.000	0.000
Max rY	303	6:COMBINATI(	35.746	-12.679	60.650	71.532	-0.002	**0.009**	0.003
Min rY	311	6:COMBINATI(	-18.255	-12.355	60.135	64.048	-0.002	**-0.007**	-0.002
Max rZ	299	6:COMBINATI(	102.346	9.883	1.391	102.831	0.003	0.001	**0.007**
Min rZ	281	6:COMBINATI(	72.404	-9.857	0.092	73.072	0.000	0.000	**-0.007**
Max Rst	307	6:COMBINATI(	7.479	-42.415	135.948	**142.607**	-0.006	0.000	0.000

Table 32. Max. Stress Summary Z-Direction Reg. 1.60

			Shear		Membrane			Bending		
	Plate	L/C	Qx (N/mm^2)	Qy (N/mm^2)	Sx (N/mm^2)	Sy (N/mm^2)	Sxy (N/mm^2)	Mx (kNm/m)	My (kNm/m)	Mxy (kNm/m)
Max Qx	451	6:COMBINATI(	**2.687**	0.701	-16.517	-5.067	-7.812	1.14E 3	4.52E 3	2.52E 3
Min Qx	1652	6:COMBINATI(	**-2.502**	4.850	0.142	-22.662	-6.737	2.34E 3	7.4E 3	-6.59E 3
Max Qy	1652	6:COMBINATI(	-2.502	**4.850**	0.142	-22.662	-6.737	2.34E 3	7.4E 3	-6.59E 3
Min Qy	1691	6:COMBINATI(	1.016	**-2.028**	-1.051	4.039	-1.123	-624.112	-2.28E 3	1.15E 3
Max Sx	1254	6:COMBINATI(	-0.071	-1.285	**13.565**	1.703	-1.285	-7.62E 3	362.236	483.518
Min Sx	440	6:COMBINATI(	-2.092	0.719	**-16.823**	-5.313	9.173	940.481	4.6E 3	-2.3E 3
Max Sy	1045	6:COMBINATI(	0.493	-1.311	1.880	**17.740**	-0.372	-722.510	-10.3E 3	532.279
Min Sy	1652	6:COMBINATI(	-2.502	4.850	0.142	**-22.662**	-6.737	2.34E 3	7.4E 3	-6.59E 3
Max Sxy	1608	6:COMBINATI(	1.157	1.660	-12.700	-12.733	**16.291**	6.4E 3	-1.2E 3	6.33E 3
Min Sxy	441	6:COMBINATI(	2.560	0.573	-16.803	-5.118	**-7.890**	1.18E 3	4.61E 3	2.64E 3
Max Mx	1596	6:COMBINATI(	1.911	-1.409	-6.047	0.741	0.766	**11.6E 3**	9.54E 3	-8.15E 3
Min Mx	1254	6:COMBINATI(	-0.071	-1.285	13.565	1.703	-1.285	**-7.62E 3**	362.236	483.518
Max My	1654	6:COMBINATI(	2.139	0.252	-2.554	-2.962	3.527	-1.12E 3	**11.9E 3**	2.01E 3
Min My	1045	6:COMBINATI(	0.493	-1.311	1.880	17.740	-0.372	-722.510	**-10.3E 3**	532.279
Max Mxy	1608	6:COMBINATI(	1.157	1.660	-12.700	-12.733	16.291	6.4E 3	-1.2E 3	**6.33E 3**
Min Mxy	1596	6:COMBINATI(	1.911	-1.409	-6.047	0.741	0.766	11.6E 3	9.54E 3	**-8.15E 3**

Table 33. Max. Principal Stress Summary Z-Direction Reg. 1.60

			Principal		Von Mis		Tresca	
	Plate	L/C	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)
Max (t)	1608	6:COMBINATI(	**35.227**	-35.227	46.582	22.618	52.816	25.475
Max (b)	1608	6:COMBINATI(	35.227	**-35.227**	46.582	22.618	52.816	25.475
Max VM (t)	1608	6:COMBINATI(	35.227	-35.227	**46.582**	22.618	52.816	25.475
Max VM (b)	441	6:COMBINATI(	17.717	-11.923	25.832	**43.533**	29.640	45.467
Tresca (t)	1608	6:COMBINATI(	35.227	-35.227	46.582	22.618	**52.816**	25.475
Tresca (b)	441	6:COMBINATI(	17.717	-11.923	25.832	43.533	29.640	**45.467**

Table 34. Reaction Summary Z-Direction Reg. 1.60

			Horizontal	Vertical	Horizontal	Moment		
	Node	L/C	FX (kN)	FY (kN)	FZ (kN)	MX (kNm)	MY (kNm)	MZ (kNm)
Max FX	203	6:COMBINATI(	**20.9E 3**	151E 3	5.32E 3	7.81E 3	337.985	-34E 3
Min FX	222	6:COMBINATI(	**-25.1E 3**	15.9E 3	-11E 3	-8.51E 3	9.24E 3	58.4E 3
Max FY	203	6:COMBINATI(	20.9E 3	**151E 3**	5.32E 3	7.81E 3	337.985	-34E 3
Min FY	226	4:EL	-472.168	**-54.8E 3**	-9.6E 3	-8.14E 3	1.05E 3	-16.5E 3
Max FZ	220	6:COMBINATI(	1.22E 3	77.7E 3	**26.4E 3**	-51.9E 3	-13.1E 3	43E 3
Min FZ	213	3:TL	-4.34E 3	-1.78E 3	**-17.5E 3**	-61.8E 3	4.769	15.1E 3
Max MX	223	3:TL	-6.98E 3	325.088	15.9E 3	**62.2E 3**	1.93E 3	7.28E 3
Min MX	213	3:TL	-4.34E 3	-1.78E 3	-17.5E 3	**-61.8E 3**	4.769	15.1E 3
Max MY	222	3:TL	-19.9E 3	8.4E 3	-4.06E 3	8.37E 3	**12.2E 3**	36.2E 3
Min MY	220	6:COMBINATI(	1.22E 3	77.7E 3	26.4E 3	-51.9E 3	**-13.1E 3**	43E 3
Max MZ	215	6:COMBINATI(	-13.4E 3	24.3E 3	-3.96E 3	7.57E 3	3.03E 3	**92.5E 3**
Min MZ	209	6:COMBINATI(	15E 3	23.2E 3	-4.13E 3	6.6E 3	-2.85E 3	**-88.6E 3**

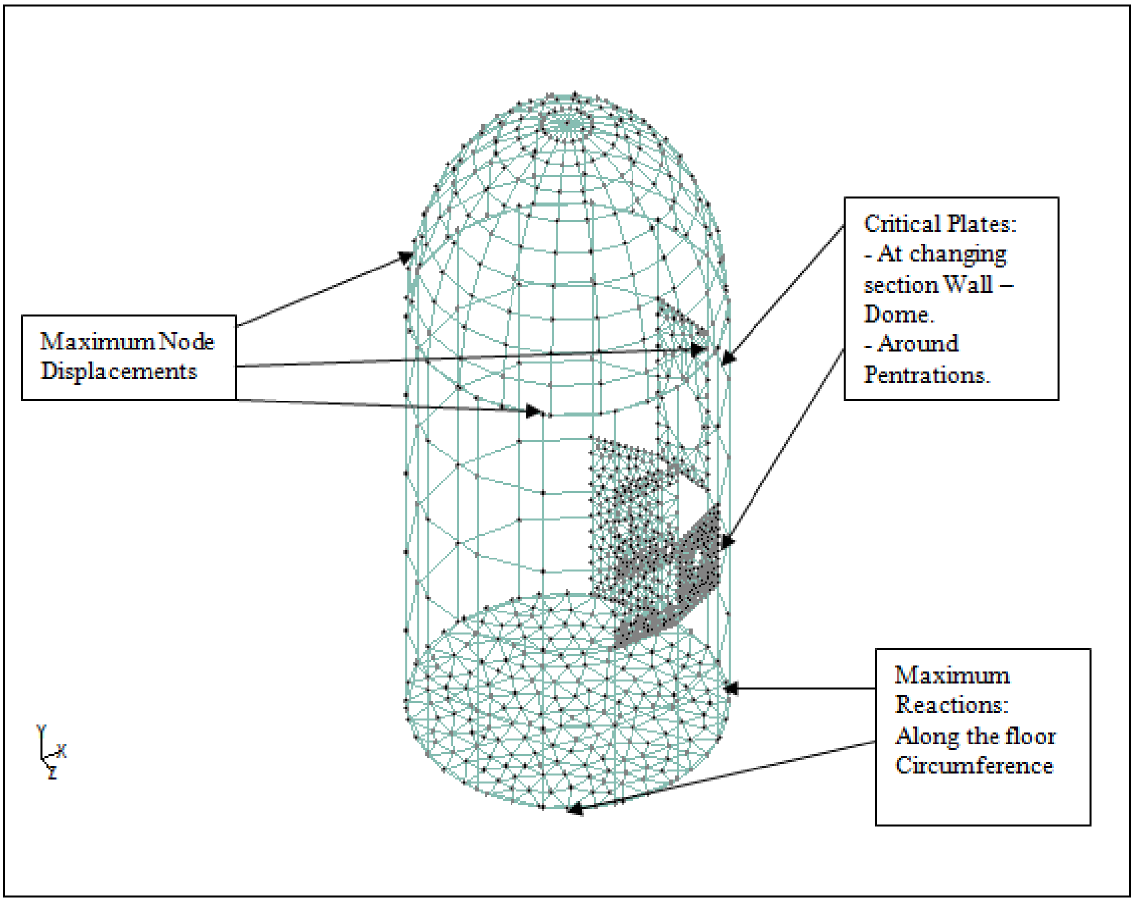


Figure 36. Z-D Reg1.60 Stress/Displacement Location

Table 35. Nodal Displacement X-Direction AQABA1995

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	299	6:COMBINATI(	**79.868**	-11.287	0.122	80.662	0.001	-0.001	0.005
Min X	315	6:COMBINATI(	**-73.803**	-10.389	1.132	74.539	0.001	0.001	-0.005
Max Y	476	3:ThermalLoad	-0.113	**7.141**	-0.002	7.142	0.000	-0.000	-0.000
Min Y	307	6:COMBINATI(	3.554	**-108.598**	104.595	150.819	-0.009	0.000	0.000
Max Z	307	6:COMBINATI(	3.554	-108.598	**104.595**	150.819	-0.009	0.000	0.000
Min Z	288	6:COMBINATI(	4.115	-104.688	**-105.044**	148.360	0.009	0.000	-0.000
Max rX	288	6:COMBINATI(	4.115	-104.688	-105.044	148.360	**0.009**	0.000	-0.000
Min rX	307	6:COMBINATI(	3.554	-108.598	104.595	150.819	**-0.009**	0.000	0.000
Max rY	303	6:COMBINATI(	29.756	-66.762	46.446	86.602	-0.004	**0.007**	0.003
Min rY	311	6:COMBINATI(	-22.128	-66.306	47.217	84.354	-0.004	**-0.007**	-0.003
Max rZ	275	6:COMBINATI(	-46.962	-4.637	0.092	47.190	-0.000	-0.000	**0.007**
Min rZ	281	6:COMBINATI(	51.195	-5.887	-0.842	51.539	-0.000	-0.000	**-0.007**
Max Rst	307	6:COMBINATI(	3.554	-108.598	104.595	**150.819**	-0.009	0.000	0.000

Table 36. Max. Stress Summary X-Direction AQABA1995

			Shear		Membrane			Bending		
	Plate	L/C	Qx (N/mm^2)	Qy (N/mm^2)	Sx (N/mm^2)	Sy (N/mm^2)	Sxy (N/mm^2)	Mx (kNm/m)	My (kNm/m)	Mxy (kNm/m)
Max Qx	441	6:COMBINATI(	**2.551**	0.598	-17.170	-7.843	-8.228	223.669	3.9E 3	2.42E 3
Min Qx	440	6:COMBINATI(	**-2.509**	0.675	-17.180	-7.972	8.322	108.327	3.89E 3	-2.31E 3
Max Qy	1652	6:COMBINATI(	-2.113	**3.849**	-0.602	-17.169	-2.810	5.53E 3	15.1E 3	-20.3E 3
Min Qy	1691	6:COMBINATI(	0.762	**-2.084**	-0.542	1.930	-0.593	-4.2E 3	-2.93E 3	2.03E 3
Max Sx	1254	4:ExternalPres	0.048	-0.127	**4.245**	0.329	-0.328	-3.92E 3	-880.439	1.49E 3
Min Sx	440	6:COMBINATI(	-2.509	0.675	**-17.180**	-7.972	8.322	108.327	3.89E 3	-2.31E 3
Max Sy	1045	4:ExternalPres	0.070	0.016	0.630	**6.324**	-0.395	-459.007	-3.72E 3	-926.975
Min Sy	1652	6:COMBINATI(	-2.113	3.849	-0.602	**-17.169**	-2.810	5.53E 3	15.1E 3	-20.3E 3
Max Sxy	440	6:COMBINATI(	-2.509	0.675	-17.180	-7.972	**8.322**	108.327	3.89E 3	-2.31E 3
Min Sxy	441	6:COMBINATI(	2.551	0.598	-17.170	-7.843	**-8.228**	223.669	3.9E 3	2.42E 3
Max Mx	1608	4:ExternalPres	1.126	1.000	-7.305	-4.826	5.061	**18.4E 3**	-4.81E 3	13.5E 3
Min Mx	693	6:COMBINATI(	-0.348	0.392	-2.098	0.297	1.239	**-26.5E 3**	-4.62E 3	16.8E 3
Max My	1652	4:ExternalPres	-1.461	2.716	-0.643	-12.661	-1.223	5.2E 3	**21.1E 3**	-16.4E 3
Min My	686	6:COMBINATI(	-0.641	1.562	-1.126	-2.037	0.627	-4E 3	**-41.4E 3**	12.5E 3
Max Mxy	1608	6:COMBINATI(	1.641	1.212	-8.724	-7.072	8.314	11.4E 3	-8.76E 3	**22.2E 3**
Min Mxy	1652	6:COMBINATI(	-2.113	3.849	-0.602	-17.169	-2.810	5.53E 3	15.1E 3	**-20.3E 3**

Table 37. Max. Principal Stress X-Direction AQABA1995

			Principal		Von Mis		Tresca	
	Plate	L/C	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)
Max (t)	1608	6:COMBINATI(	**24.316**	-24.316	30.223	11.592	33.837	13.011
Max (b)	1608	6:COMBINATI(	24.316	**-24.316**	30.223	11.592	33.837	13.011
Max VM (t)	1608	6:COMBINATI(	24.316	-24.316	**30.223**	11.592	33.837	13.011
Max VM (b)	441	6:COMBINATI(	16.511	-16.511	24.174	**41.583**	27.748	42.611
Tresca (t)	1608	6:COMBINATI(	24.316	-24.316	30.223	11.592	**33.837**	13.011
Tresca (b)	441	6:COMBINATI(	16.511	-16.511	24.174	41.583	27.748	**42.611**

Table 38. Reaction Summary X-Direction AQABA1995

	Node	L/C	Horizontal FX (kN)	Vertical FY (kN)	Horizontal FZ (kN)	Moment MX (kNm)	MY (kNm)	MZ (kNm)
Max FX	203	6:COMBINATI(	**49.9E 3**	144E 3	128.010	-2.75E 3	470.223	-217E 3
Min FX	222	6:COMBINATI(	**-49.5E 3**	31.6E 3	-12.9E 3	38.9E 3	40.1E 3	101E 3
Max FY	203	6:COMBINATI(	49.9E 3	**144E 3**	128.010	-2.75E 3	470.223	-217E 3
Min FY	226	4:ExternalPres:	-2.69E 3	**-40.1E 3**	-10.7E 3	-34.3E 3	2.18E 3	-8.38E 3
Max FZ	220	6:COMBINATI(	-3.23E 3	84.6E 3	**49.6E 3**	5.81E 3	-46E 3	116E 3
Min FZ	213	3:ThermalLoad	-11.3E 3	-2.91E 3	**-46.4E 3**	-224E 3	73.723	53.9E 3
Max MX	226	3:ThermalLoad	7.71E 3	68.988	46.4E 3	**223E 3**	915.437	-54.2E 3
Min MX	213	3:ThermalLoad	-11.3E 3	-2.91E 3	-46.4E 3	**-224E 3**	73.723	53.9E 3
Max MY	222	3:ThermalLoad	-46.3E 3	17.9E 3	-6.52E 3	60.6E 3	**46.2E 3**	104E 3
Min MY	220	6:COMBINATI(	-3.23E 3	84.6E 3	49.6E 3	5.81E 3	**-46E 3**	116E 3
Max MZ	6	3:ThermalLoad	-44.1E 3	-3.52E 3	9.06E 3	-21.1E 3	-7.58E 3	**224E 3**
Min MZ	203	3:ThermalLoad	47.9E 3	706.455	211.359	-1.72E 3	349.999	**-230E 3**

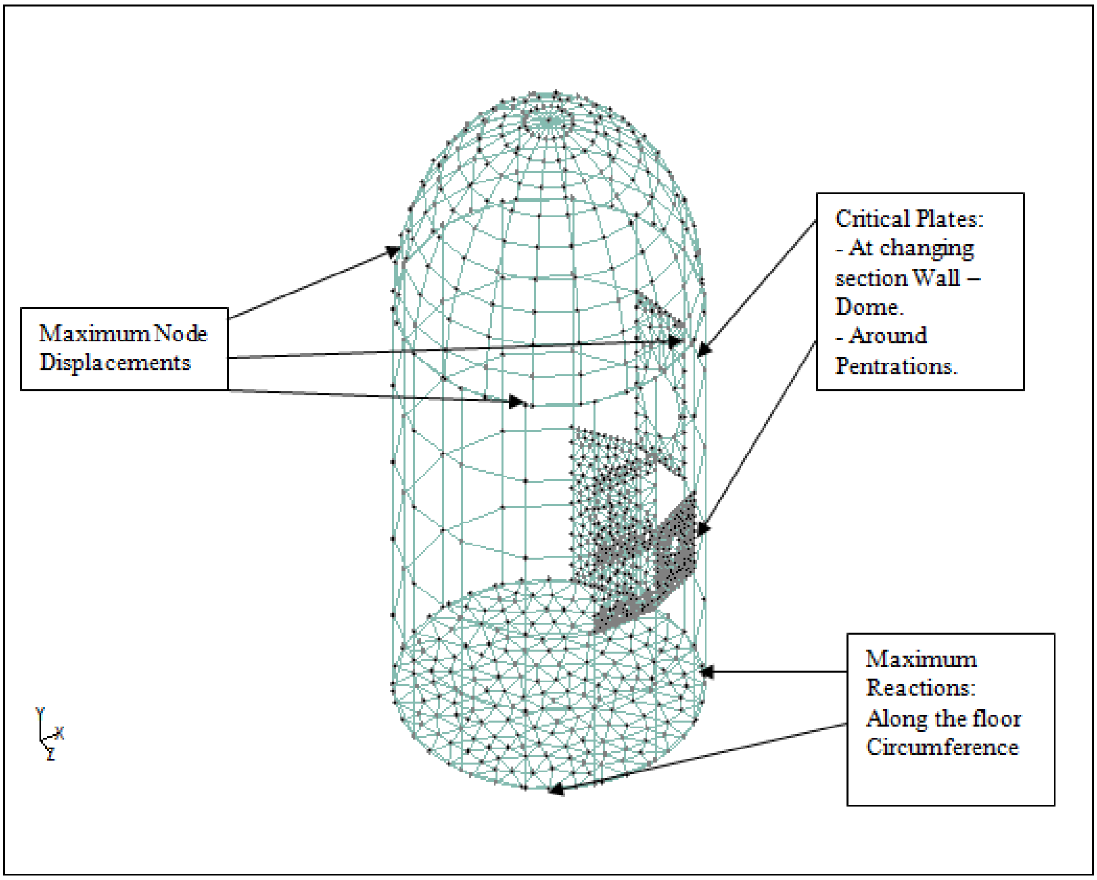


Figure 37. X-D AQABA1995 Stress/Displacement Location

Table 39. Nodal Displacement Z-Direction AQABA1995

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	299	6:COMBINATI(	**97.925**	-16.606	-0.206	99.323	0.001	0.000	0.007
Min X	318	6:COMBINATI(	**-90.721**	-13.379	-1.464	91.714	-0.001	0.000	-0.007
Max Y	278	6:COMBINATI(	-0.730	**8.805**	-26.042	27.500	-0.001	0.000	0.000
Min Y	307	6:COMBINATI(	3.000	**-130.589**	123.068	179.466	-0.010	0.000	0.000
Max Z	307	6:COMBINATI(	3.000	-130.589	**123.068**	179.466	-0.010	0.000	0.000
Min Z	288	6:COMBINATI(	3.968	-125.812	**-123.280**	176.188	0.010	0.000	-0.000
Max rX	288	6:COMBINATI(	3.968	-125.812	-123.280	176.188	**0.010**	0.000	-0.000
Min rX	307	6:COMBINATI(	3.000	-130.589	123.068	179.466	**-0.010**	0.000	0.000
Max rY	303	6:COMBINATI(	32.542	-83.149	56.019	105.408	-0.005	**0.008**	0.003
Min rY	311	6:COMBINATI(	-25.944	-82.607	57.588	103.987	-0.005	**-0.008**	-0.003
Max rZ	299	6:COMBINATI(	97.925	-16.606	-0.206	99.323	0.001	0.000	**0.007**
Min rZ	318	6:COMBINATI(	-90.721	-13.379	-1.464	91.714	-0.001	0.000	**-0.007**
Max Rst	307	6:COMBINATI(	3.000	-130.589	123.068	**179.466**	-0.010	0.000	0.000

Table 40. Max. Stress Summary Z-Direction AQABA1995

			Shear		Membrane			Bending		
	Plate	L/C	Qx (N/mm^2)	Qy (N/mm^2)	Sx (N/mm^2)	Sy (N/mm^2)	Sxy (N/mm^2)	Mx (kNm/m)	My (kNm/m)	Mxy (kNm/m)
Max Qx	681	6:COMBINATI(	**2.447**	-1.382	-3.865	-0.045	-3.278	-4.91E 3	-1.2E 3	-2.28E 3
Min Qx	1652	6:COMBINATI(	**-2.546**	4.704	-0.084	-25.161	-6.868	2.23E 3	6.55E 3	-6.69E 3
Max Qy	1652	6:COMBINATI(	-2.546	**4.704**	-0.084	-25.161	-6.868	2.23E 3	6.55E 3	-6.69E 3
Min Qy	1691	6:COMBINATI(	0.932	**-2.148**	-1.119	3.795	-1.181	-750.585	-2.45E 3	1.07E 3
Max Sx	1254	6:COMBINATI(	-0.133	-1.304	**12.648**	1.482	-1.553	-7.74E 3	285.788	371.118
Min Sx	440	6:COMBINATI(	-2.352	0.439	**-17.654**	-7.990	8.595	666.738	4.51E 3	-2.39E 3
Max Sy	1045	6:COMBINATI(	0.473	-1.418	1.750	**16.639**	-0.616	-757.512	-10.4E 3	413.649
Min Sy	1652	6:COMBINATI(	-2.546	4.704	-0.084	**-25.161**	-6.868	2.23E 3	6.55E 3	-6.69E 3
Max Sxy	1608	6:COMBINATI(	0.912	1.576	-13.342	-13.094	**15.640**	5.52E 3	-1.48E 3	6.02E 3
Min Sxy	441	6:COMBINATI(	2.294	0.294	-17.651	-7.797	**-8.476**	902.227	4.54E 3	2.55E 3
Max Mx	1596	6:COMBINATI(	1.683	-1.683	-6.930	0.206	0.232	**11E 3**	8.67E 3	-8.83E 3
Min Mx	1254	6:COMBINATI(	-0.133	-1.304	12.648	1.482	-1.553	**-7.74E 3**	285.788	371.118
Max My	1654	6:COMBINATI(	1.995	0.196	-2.618	-3.504	3.437	-1.21E 3	**11.3E 3**	1.66E 3
Min My	1045	6:COMBINATI(	0.473	-1.418	1.750	16.639	-0.616	-757.512	**-10.4E 3**	413.649
Max Mxy	1608	6:COMBINATI(	0.912	1.576	-13.342	-13.094	15.640	5.52E 3	-1.48E 3	**6.02E 3**
Min Mxy	1596	6:COMBINATI(	1.683	-1.683	-6.930	0.206	0.232	11E 3	8.67E 3	**-8.83E 3**

Table 41. Max. Principal Stress Z-Direction AQABA1995

			Principal		Von Mis		Tresca	
	Plate	L/C	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)
Max (t)	1608	6:COMBINATI(	**35.385**	-35.385	44.815	21.952	50.393	24.770
Max (b)	1608	6:COMBINATI(	35.385	**-35.385**	44.815	21.952	50.393	24.770
Max VM (t)	1608	6:COMBINATI(	35.385	-35.385	**44.815**	21.952	50.393	24.770
Max VM (b)	441	6:COMBINATI(	14.391	-13.832	24.444	**44.552**	28.223	46.728
Tresca (t)	1608	6:COMBINATI(	35.385	-35.385	44.815	21.952	**50.393**	24.770
Tresca (b)	441	6:COMBINATI(	14.391	-13.832	24.444	44.552	28.223	**46.728**

Table 42. Reaction Summary Z-Direction AQABA1995

			Horizontal	Vertical	Horizontal	Moment		
	Node	L/C	FX (kN)	FY (kN)	FZ (kN)	MX (kNm)	MY (kNm)	MZ (kNm)
Max FX	203	6:COMBINATI(	**20.8E 3**	150E 3	61.515	-2.19E 3	59.460	-34E 3
Min FX	222	6:COMBINATI(	**-26E 3**	11.7E 3	-11.6E 3	-12.1E 3	8.77E 3	54E 3
Max FY	203	6:COMBINATI(	20.8E 3	**150E 3**	61.515	-2.19E 3	59.460	-34E 3
Min FY	211	6:COMBINATI(	5.51E 3	**-61E 3**	-6.17E 3	-48.7E 3	-2.95E 3	-37.3E 3
Max FZ	104	3:ThermalLoad	4.61E 3	-0.000	**26.6E 3**	-8.87E 3	-11E 3	1.54E 3
Min FZ	184	3:ThermalLoad	17.4E 3	0.000	**-28.2E 3**	9.41E 3	4.4E 3	5.78E 3
Max MX	223	3:ThermalLoad	-6.98E 3	325.088	15.9E 3	**62.2E 3**	1.93E 3	7.28E 3
Min MX	213	3:ThermalLoad	-4.34E 3	-1.78E 3	-17.5E 3	**-61.8E 3**	4.769	15.1E 3
Max MY	222	3:ThermalLoad	-19.9E 3	8.4E 3	-4.06E 3	8.37E 3	**12.2E 3**	36.2E 3
Min MY	220	6:COMBINATI(	566.075	73.5E 3	23.7E 3	-53.9E 3	**-13.2E 3**	42.2E 3
Max MZ	215	6:COMBINATI(	-14.8E 3	15.6E 3	-5.76E 3	6.39E 3	2.85E 3	**90E 3**
Min MZ	209	6:COMBINATI(	13.3E 3	11.2E 3	-6.75E 3	5.66E 3	-2.92E 3	**-90.5E 3**

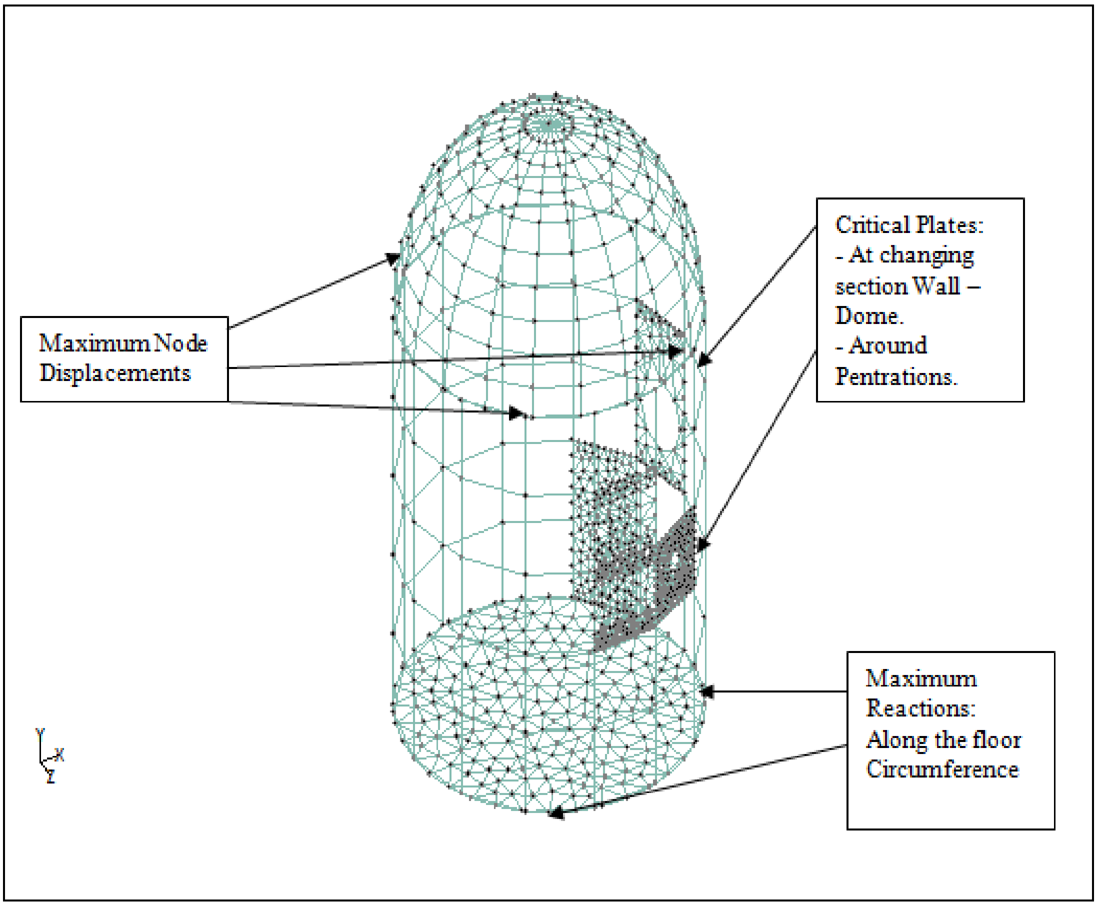


Figure 38. Z-D AQABA1995 Stress/Displacement Location

Table 43. Nodal Displacement X-Direction Study1997

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	299	6:COMBINATI(	**131.211**	-15.476	2.446	132.143	0.001	0.003	0.008
Min X	315	6:COMBINATI(	**-58.453**	-13.503	4.438	60.156	0.001	0.003	-0.006
Max Y	311	5:EQLaod	56.304	**12.442**	16.636	60.014	0.002	0.000	0.001
Min Y	307	6:COMBINATI(	55.714	**-131.521**	123.872	189.066	-0.010	0.002	0.001
Max Z	307	6:COMBINATI(	55.714	-131.521	**123.872**	189.066	-0.010	0.002	0.001
Min Z	288	6:COMBINATI(	57.127	-126.600	**-121.944**	184.827	0.010	0.002	0.001
Max rX	288	6:COMBINATI(	57.127	-126.600	-121.944	184.827	**0.010**	0.002	0.001
Min rX	307	6:COMBINATI(	55.714	-131.521	123.872	189.066	**-0.010**	0.002	0.001
Max rY	320	6:COMBINATI(	-15.869	-31.574	-4.886	35.674	0.003	**0.009**	-0.005
Min rY	292	6:COMBINATI(	89.728	-70.635	-42.910	121.991	0.007	**-0.008**	0.004
Max rZ	275	6:COMBINATI(	-49.457	-5.425	0.379	49.755	0.000	0.000	**0.009**
Min rZ	318	6:COMBINATI(	-58.322	-11.896	0.894	59.530	-0.000	0.003	**-0.006**
Max Rst	307	6:COMBINATI(	55.714	-131.521	123.872	**189.066**	-0.010	0.002	0.001

Table 44. Max. Stress Summary X-Direction Study1997

			Shear		Membrane			Bending		
	Plate	L/C	Qx (N/mm^2)	Qy (N/mm^2)	Sx (N/mm^2)	Sy (N/mm^2)	Sxy (N/mm^2)	Mx (kNm/m)	My (kNm/m)	Mxy (kNm/m)
Max Qx	451	6:COMBINATI(	**3.378**	0.933	-16.460	-7.339	-8.076	2.54E 3	4.6E 3	3E 3
Min Qx	1026	6:COMBINATI(	**-2.058**	2.293	6.576	2.755	-1.179	-4.85E 3	-4.36E 3	3.13E 3
Max Qy	1652	6:COMBINATI(	-1.910	**5.837**	-0.001	-21.199	-5.807	2.99E 3	9.6E 3	-4.53E 3
Min Qy	1692	6:COMBINATI(	1.637	**-2.089**	0.370	2.456	-2.300	-5.04E 3	1.1E 3	1.2E 3
Max Sx	1254	6:COMBINATI(	0.043	-1.231	**15.316**	2.037	-0.925	-7.68E 3	399.118	448.030
Min Sx	440	6:COMBINATI(	-1.279	1.017	**-16.534**	-7.540	8.974	2.37E 3	4.64E 3	-1.85E 3
Max Sy	1045	6:COMBINATI(	0.540	-1.207	2.069	**19.085**	-0.400	-667.810	-10.3E 3	493.450
Min Sy	1652	6:COMBINATI(	-1.910	5.837	-0.001	**-21.199**	-5.807	2.99E 3	9.6E 3	-4.53E 3
Max Sxy	1608	6:COMBINATI(	1.084	2.235	-11.314	-11.100	**18.034**	8.27E 3	-1.23E 3	7.5E 3
Min Sxy	441	6:COMBINATI(	3.350	0.847	-16.532	-7.373	**-8.112**	2.56E 3	4.67E 3	3.12E 3
Max Mx	1596	6:COMBINATI(	1.812	-1.644	-6.425	0.417	0.430	**13.4E 3**	10.2E 3	-7.05E 3
Min Mx	1254	6:COMBINATI(	0.043	-1.231	15.316	2.037	-0.925	**-7.68E 3**	399.118	448.030
Max My	1654	6:COMBINATI(	2.584	0.561	-2.468	-3.333	3.691	-935.707	**13.7E 3**	1.82E 3
Min My	1045	6:COMBINATI(	0.540	-1.207	2.069	19.085	-0.400	-667.810	**-10.3E 3**	493.450
Max Mxy	1608	6:COMBINATI(	1.084	2.235	-11.314	-11.100	18.034	8.27E 3	-1.23E 3	**7.5E 3**
Min Mxy	1596	4:ExternalLoad	1.402	-1.125	-5.487	-1.064	1.292	10.6E 3	6.67E 3	**-7.29E 3**

Table 45. Max. Principal Stress X-Direction Study1997

			Principal		Von Mis		Tresca	
	Plate	L/C	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)
Max (t)	1608	6:COMBINATI(	**36.046**	-36.046	52.495	23.805	60.228	26.399
Max (b)	1608	6:COMBINATI(	36.046	**-36.046**	52.495	23.805	60.228	26.399
Max VM (t)	1608	6:COMBINATI(	36.046	-36.046	**52.495**	23.805	60.228	26.399
Max VM (b)	441	6:COMBINATI(	17.109	-6.441	21.081	**49.465**	23.550	52.274
Tresca (t)	1608	6:COMBINATI(	36.046	-36.046	52.495	23.805	**60.228**	26.399
Tresca (b)	441	6:COMBINATI(	17.109	-6.441	21.081	49.465	23.550	**52.274**

Table 46. Reaction Summary X-Direction Study1997

			Horizontal	Vertical	Horizontal	Moment		
	Node	L/C	FX (kN)	FY (kN)	FZ (kN)	MX (kNm)	MY (kNm)	MZ (kNm)
Max FX	203	6:COMBINATI(	**21.7E 3**	168E 3	-65.454	-1.36E 3	118.287	-31.5E 3
Min FX	222	6:COMBINATI(	**-24E 3**	12.5E 3	-10.5E 3	-10.7E 3	8.83E 3	56.1E 3
Max FY	203	6:COMBINATI(	21.7E 3	**168E 3**	-65.454	-1.36E 3	118.287	-31.5E 3
Min FY	211	6:COMBINATI(	9.72E 3	**-58.5E 3**	-5.19E 3	-48.1E 3	-2.7E 3	-35.3E 3
Max FZ	104	3:ThermalLoad	4.61E 3	-0.000	**26.6E 3**	-8.87E 3	-11E 3	1.54E 3
Min FZ	184	3:ThermalLoad	17.4E 3	0.000	**-28.2E 3**	9.41E 3	4.4E 3	5.78E 3
Max MX	223	3:ThermalLoad	-6.98E 3	325.088	15.9E 3	**62.2E 3**	1.93E 3	7.28E 3
Min MX	213	3:ThermalLoad	-4.34E 3	-1.78E 3	-17.5E 3	**-61.8E 3**	4.769	15.1E 3
Max MY	222	3:ThermalLoad	-19.9E 3	8.4E 3	-4.06E 3	8.37E 3	**12.2E 3**	36.2E 3
Min MY	220	6:COMBINATI(	855.293	80.9E 3	24.2E 3	-42.8E 3	**-12.7E 3**	46.4E 3
Max MZ	215	6:COMBINATI(	-12.6E 3	16.6E 3	-4.27E 3	11.6E 3	3.17E 3	**96.2E 3**
Min MZ	228	6:COMBINATI(	14.8E 3	10.7E 3	8.19E 3	-2.58E 3	2.59E 3	**-85E 3**

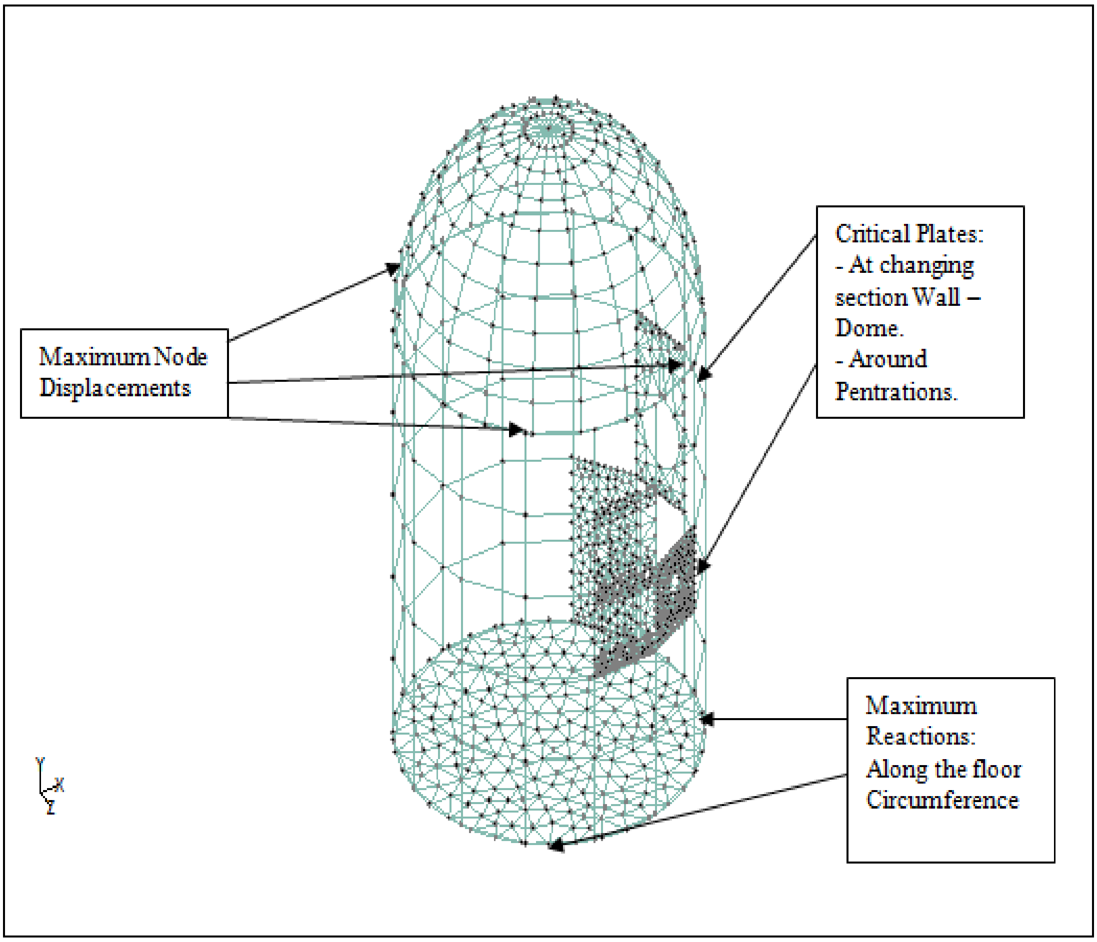


Figure 39. X-D Study1997 Stress/Displacement Location

Table 47. Nodal Displacement Z-Direction Study1997

	Node	L/C	X (mm)	Y (mm)	Z (mm)	Resultant (mm)	rX (rad)	rY (rad)	rZ (rad)
Max X	299	6:COMBINATI(	**102.038**	15.152	1.340	103.165	0.003	0.001	0.007
Min X	318	6:COMBINATI(	**-84.860**	18.269	-0.087	86.804	0.002	0.001	-0.006
Max Y	307	5:EQLaod	3.849	**108.050**	11.281	108.706	0.005	0.000	0.000
Min Y	307	4:ExternalLoad	1.989	**-89.400**	81.524	121.006	-0.006	0.000	0.000
Max Z	307	6:COMBINATI(	6.663	-25.166	**133.823**	136.332	-0.005	0.000	0.000
Min Z	288	6:COMBINATI(	7.636	-20.715	**-112.660**	114.803	0.015	0.000	0.000
Max rX	288	6:COMBINATI(	7.636	-20.715	-112.660	114.803	**0.015**	0.000	0.000
Min rX	307	4:ExternalLoad	1.989	-89.400	81.524	121.006	**-0.006**	0.000	0.000
Max rY	303	6:COMBINATI(	35.228	1.220	59.865	69.472	-0.001	**0.009**	0.003
Min rY	311	6:COMBINATI(	-19.612	1.598	59.783	62.938	-0.001	**-0.007**	-0.002
Max rZ	299	6:COMBINATI(	102.038	15.152	1.340	103.165	0.003	0.001	**0.007**
Min rZ	281	6:COMBINATI(	72.368	-9.844	0.044	73.034	0.001	0.000	**-0.007**
Max Rst	307	6:COMBINATI(	6.663	-25.166	133.823	**136.332**	-0.005	0.000	0.000

Table 48. Max. Stress Summary Z-Direction Study1997

			Shear		Membrane			Bending		
	Plate	L/C	Qx (N/mm^2)	Qy (N/mm^2)	Sx (N/mm^2)	Sy (N/mm^2)	Sxy (N/mm^2)	Mx (kNm/m)	My (kNm/m)	Mxy (kNm/m)
Max Qx	451	6:COMBINATI(	**2.620**	0.714	-16.672	-4.530	-7.726	1.16E 3	4.52E 3	2.52E 3
Min Qx	1652	6:COMBINATI(	**-2.496**	4.872	0.172	-22.273	-6.735	2.35E 3	7.51E 3	-6.59E 3
Max Qy	1652	6:COMBINATI(	-2.496	**4.872**	0.172	-22.273	-6.735	2.35E 3	7.51E 3	-6.59E 3
Min Qy	1691	6:COMBINATI(	1.015	**-2.029**	-1.051	4.031	-1.130	-625.178	-2.28E 3	1.15E 3
Max Sx	1254	6:COMBINATI(	-0.079	-1.282	**13.457**	1.685	-1.308	-7.63E 3	363.525	486.737
Min Sx	440	6:COMBINATI(	-2.139	0.750	**-16.933**	-4.784	9.270	984.884	4.61E 3	-2.3E 3
Max Sy	1045	6:COMBINATI(	0.490	-1.330	1.865	**17.572**	-0.409	-728.983	-10.3E 3	534.495
Min Sy	1652	6:COMBINATI(	-2.496	4.872	0.172	**-22.273**	-6.735	2.35E 3	7.51E 3	-6.59E 3
Max Sxy	1608	6:COMBINATI(	1.204	1.674	-12.730	-12.696	**16.411**	6.52E 3	-1.15E 3	6.39E 3
Min Sxy	441	6:COMBINATI(	2.512	0.602	-16.914	-4.589	**-7.792**	1.23E 3	4.62E 3	2.65E 3
Max Mx	1596	6:COMBINATI(	1.956	-1.355	-6.035	0.701	0.729	**11.7E 3**	9.7E 3	-8.02E 3
Min Mx	1254	6:COMBINATI(	-0.079	-1.282	13.457	1.685	-1.308	**-7.63E 3**	363.525	486.737
Max My	1654	6:COMBINATI(	2.160	0.254	-2.555	-3.063	3.513	-1.1E 3	**12E 3**	2.07E 3
Min My	1045	6:COMBINATI(	0.490	-1.330	1.865	17.572	-0.409	-728.983	**-10.3E 3**	534.495
Max Mxy	1608	6:COMBINATI(	1.204	1.674	-12.730	-12.696	16.411	6.52E 3	-1.15E 3	**6.39E 3**
Min Mxy	1596	6:COMBINATI(	1.956	-1.355	-6.035	0.701	0.729	11.7E 3	9.7E 3	**-8.02E 3**

Table 49. Max. Principal Stress Z-Direction Study1997

			Principal		Von Mis		Tresca	
	Plate	L/C	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)	Top (N/mm^2)	Bottom (N/mm^2)
Max (t)	1608	6:COMBINATI(	**35.296**	-35.296	46.907	22.810	53.227	25.687
Max (b)	1608	6:COMBINATI(	35.296	**-35.296**	46.907	22.810	53.227	25.687
Max VM (t)	1608	6:COMBINATI(	35.296	-35.296	**46.907**	22.810	53.227	25.687
Max VM (b)	441	6:COMBINATI(	18.305	-11.850	26.314	**43.327**	30.155	45.270
Tresca (t)	1608	6:COMBINATI(	35.296	-35.296	46.907	22.810	**53.227**	25.687
Tresca (b)	441	6:COMBINATI(	18.305	-11.850	26.314	43.327	30.155	**45.270**

Table 50. Reaction Summary Z-Direction Study1997

			Horizontal	Vertical	Horizontal	Moment		
	Node	**L/C**	**FX** (kN)	**FY** (kN)	**FZ** (kN)	**MX** (kNm)	**MY** (kNm)	**MZ** (kNm)
Max FX	203	6:COMBINATI(	**20.9E 3**	151E 3	4.43E 3	7.71E 3	287.103	-34E 3
Min FX	222	6:COMBINATI(	**-25.2E 3**	15.3E 3	-11.1E 3	-8.99E 3	9.16E 3	57.9E 3
Max FY	203	6:COMBINATI(	20.9E 3	**151E 3**	4.43E 3	7.71E 3	287.103	-34E 3
Min FY	226	4:ExternalLoad	-472.168	**-54.8E 3**	-9.6E 3	-8.14E 3	1.05E 3	-16.5E 3
Max FZ	104	3:ThermalLoad	4.61E 3	-0.000	**26.6E 3**	-8.87E 3	-11E 3	1.54E 3
Min FZ	184	3:ThermalLoad	17.4E 3	0.000	**-28.2E 3**	9.41E 3	4.4E 3	5.78E 3
Max MX	223	3:ThermalLoad	-6.98E 3	325.088	15.9E 3	**62.2E 3**	1.93E 3	7.28E 3
Min MX	213	3:ThermalLoad	-4.34E 3	-1.78E 3	-17.5E 3	**-61.8E 3**	4.769	15.1E 3
Max MY	222	3:ThermalLoad	-19.9E 3	8.4E 3	-4.06E 3	8.37E 3	**12.2E 3**	36.2E 3
Min MY	220	6:COMBINATI(	1.11E 3	77.1E 3	25.9E 3	-52.1E 3	**-13.1E 3**	42.9E 3
Max MZ	215	6:COMBINATI(	-13.5E 3	23.2E 3	-4.19E 3	7.6E 3	3.03E 3	**92.8E 3**
Min MZ	209	6:COMBINATI(	14.8E 3	21.4E 3	-4.52E 3	6.61E 3	-2.84E 3	**-88.3E 3**

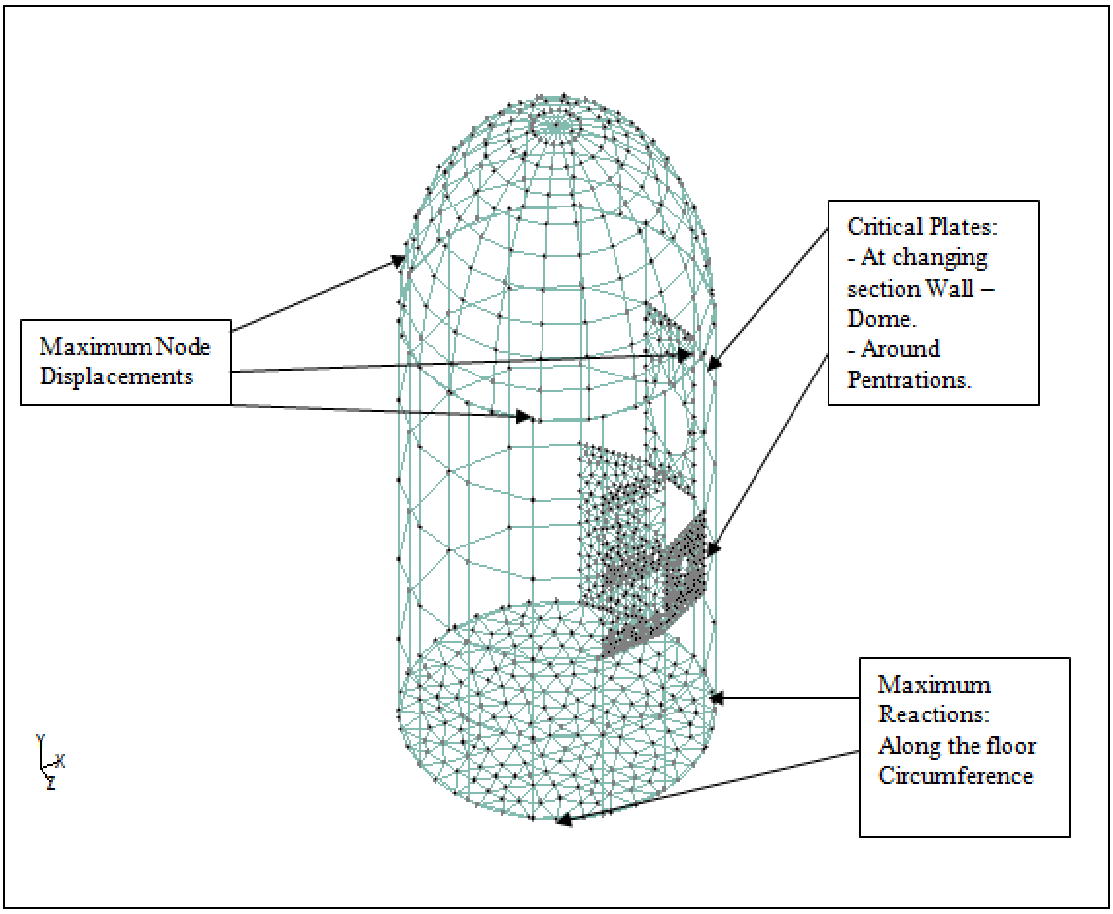


Figure 40. Z-D Study1997 Stress/Displacement Location

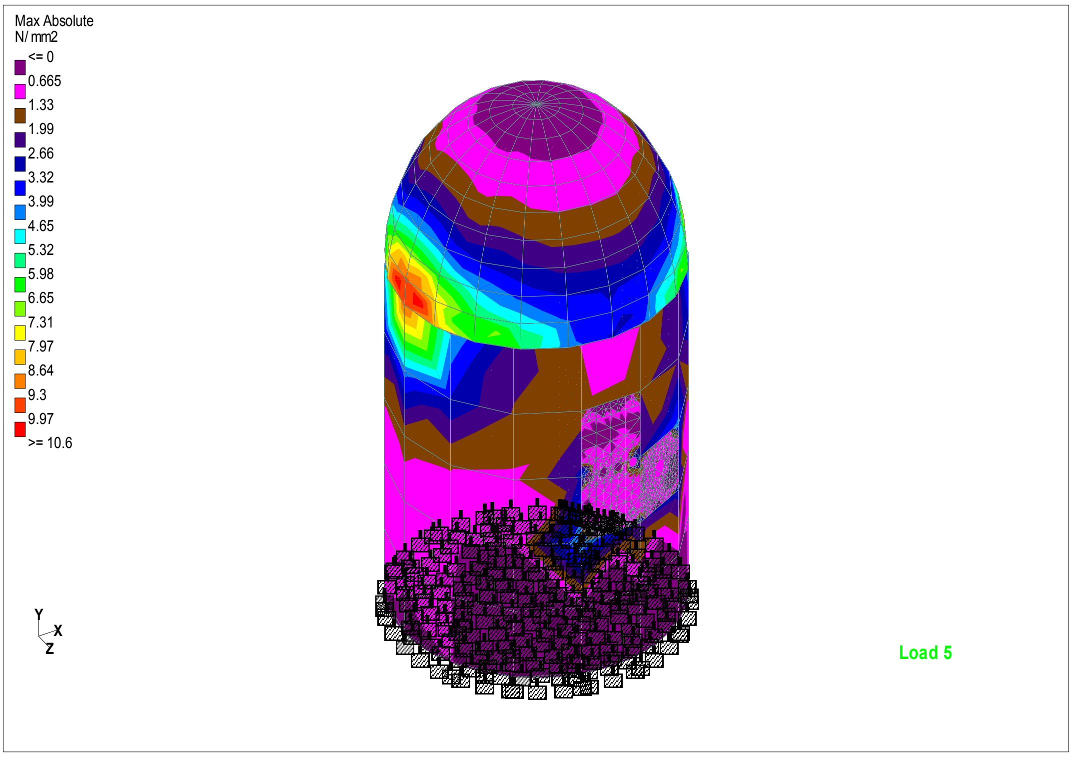


Figure 41. X-Direction Max. Absolute Stress

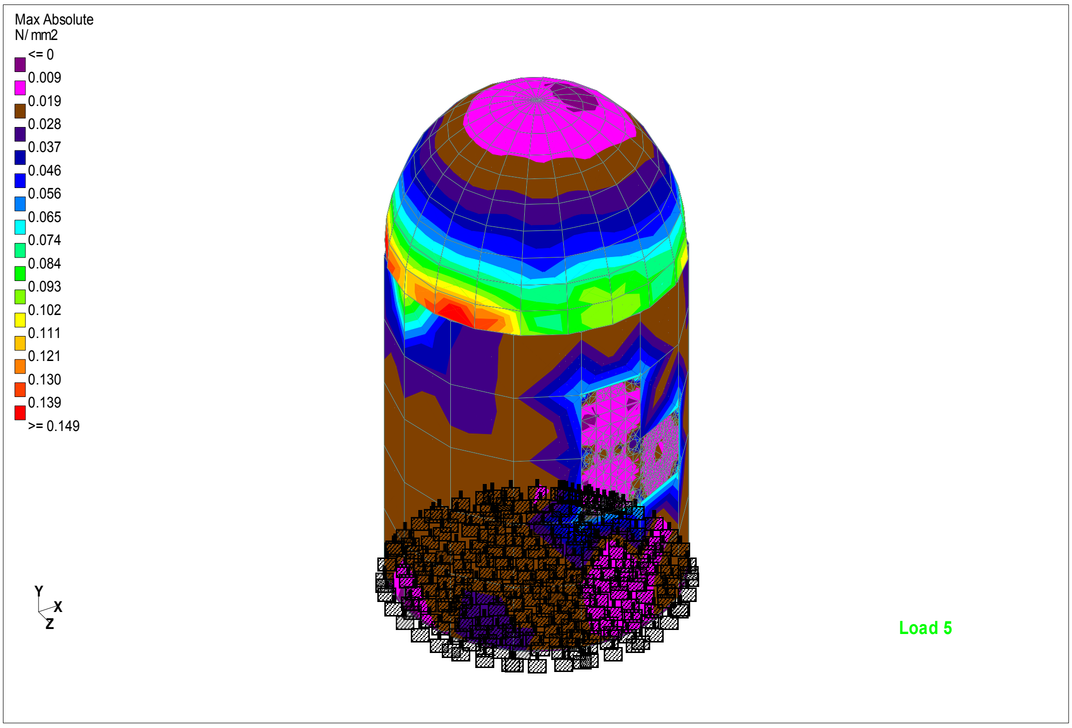


Figure 42. Z-Direction Max. Absolute Stress

DISCUSSION

The aim of this study is to represent NPPs containment structures covering definition, importance, some types, shield design, and analysis / design according to ASME standards, a comparesion between USA, Japan and some European countries regarding analysis/design references.

Model Analysis

For the prestress load from Table (23) to Table (26) the results were unstable for the same model/input data so the Prestress Load was not included, the analyses only included:

Dead Load + Live Load + Thermal Load + External Load + Safe Shut Down Earthquake Load, all of them were multiplied by a factor of (1)

The most critical zones were at the transition sections:

- Upper part where changing shape from the cylindrical wall to the semispherical dome, the change of thickness and shape give that high stress values,
- around the penetrations, where the thickness also changes and the stress redistribution occurs near the opening,
- at the outer circumference of the floor where the punishing shear takes place.

Some of the plates in the critical zones reached Von Mis. or Tresca failure stress specially near the penetrations where they are in common practice are metallic plates fitted then casted with concrete

The maximum and minimum displacement (shift/rotation) values where obtained at nodes located at the separating line between the wall and the dome.

The response spectrum combination method used is the 100 – 40 – 40 Ref. [30], the two horizontal components were X-Direction and Z-Direction and the vertical Y-Direction, for each component there was a case at which the horizontal component took 100 dominating factor (i.e. X-Direction 100 – Y-Direction 40 – Z-Direction 40).

Since its unlikely that the seismic excitation happens along a certain direction only, its preferable using a way to simulate that exaltation in all directions at the same time, to cover a higher probability event.

There were no vertical component case, the vertical component took only 0.4 out the Reg.1.60, AQABA1995 and Study 1997 response spectrums.

The AQABA1995 results showed lower values than the Study1997 due to soil nature at the response spectrum site (rock bed vs. sediment soils).

The analysis results showed that the Reg.1.60 response spectrum gave a very conservative results higher values (compeered to the other response spectrums) since it was used in USA during the sixties as a tool to represent a comprehensive response spectrum in NPPs analysis and design during at which time computers hard ware and software were less advanced and it was benifitial to use such a fixed response spectrum.

In the new analysis philosophy of the nuclear structures, the importance of a site response spectrum is essential regarding merging different structure behavior: the elastic behavior to prevent deformations (cracks) under normal and severe operation states (including seismic excitation), hence preventing nuclear leakage.

The importance of a representative site response spectrum comes into scene, by providing a background for adequate visible design to meet the elastic design seismic loads, in addition to that, the nuclear power reactor containment in new modern designs must also withstand crash of a commercial air plane by using plastic analysis (in the seventies the military air planes were considered like F-4 Phantom) which under that circumstances the structure may deformed to absorbed the impact to some limit with the possibility of leakage (a risk that could be taken to protect the whole structure from total destruction).

Codes and Standards

In USA the last NPP entered operation was in 1996 (the licensing process started in the mid eighties), the turn for clean energy sources in the new millennium was the motive to reconsider nuclear power energy again (after the mid eighties pausing due to Three Mile Island and Chernobyl accidents), some vendors applied their proposals to NRC in order to get new license (of which Mitsubishi Nuclear Industries with the US-APWR version). The USA codes and standards regarding the response spectrums for seismic design of nuclear structures were considered to some point out dated by only applying a fixed form response spectrums referred to them as guides Reg1.60, NUREG0098, (during the sixties and seventies) . Now the new approach is to use the site response spectrum in two horizontal components and one vertical.

Nuclear industry started in the USA and the American industrial codes, standards are considered main references worldwide in nuclear industry, and they are a complete set for analysis, design, testing and inspection.

The concrete structures US codes:

- Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary ACI 349M-06,
- Reinforced Concrete Design for Thermal Effects on Nuclear Power Plant Structures ACI 349.1R-07,
- Guide to the Concrete Capacity Design (CCD) Method— Embedment Design Examples ACI 349.2R-07,

- Evaluation of Existing Nuclear Safety-Related Concrete Structures ACI 349.3R-02

Do not cover the Concrete Reactor Vessels either Reinforced Concrete or Prestressed Concrete.

Committee 359 from ACI now is working as a part of ASME to publish the Boiler and Pressure Vessel Code covering the containment structures.

All the codes and standards used in U.S NRC to approve the nuclear power plants design with the Title 10 (Energy) Code of Federal Regulations (CFR10) can be seen in the NRC web site: wwwnrc.gov.

Noting that the steel liner in some of the nuclear containment designs analyzed and designed under the ASME BPVC and the construction is inspected under the American Welding Society (AWS) codes.

The steel structures in the containment designed under the ASME BVPC, but the AISC Specification for Safety-Related Steel Structures for Nuclear Facilities covers the auxiliary structures.

The American codes and standards do cover most of the nuclear power designs worldwide and can be compared and adopted by most of other codes and standards especially for approving nuclear power plants in USA. With exception to the CANDU design which feature (and the only design) a horizontal reactor core called (Calandria) in that design (not vertical as common).

Some nuclear contractors as AREVA, Mitsubishi, Toshiba Westinghouse uses for US NRC COLA application documents advanced software different from what we are familiar with, capable of analyzing a large number of elements/nodes (such as ANSYS+CivilFEM software which is capable of analyzing 32000 nodes), for foundation/nuclear island analysis and design ACS SASSI PREP software.

CONCLUSIONS

1- Using a site representative response spectrum in analysis will give a more efficient nuclear containment structure design regarding safety, function and cost.

2- If its considered to use seismic isolators, they will be located at the peripheral side of the floor.

3- The analysis of complex structures needs a powerful software capable of handling large number of elements under different load conditions using different materials (composite sections) in order to get validate, reasonable and stable results.

RECOMMENDATIONS

The nuclear structures are a new field of study and practice in Jordan. Hence, it is recommended to perform further studies in the following facets:

- Building a representative site response spectrum for any proposed nuclear power plant site.
- Using a powerful software which can deals with more elements/nods (ANSYS software) for analysis.
- Analysis of nuclear structures (containment or auxiliary structures, waste treatment facility, mining) dynamically using finite element methods, stick model.
- Studying the shielding properties of local concrete mixtures for different components properties, w/c ratios, thicknesses, temperatures, curing methods and periods.
- Modeling of structures and components (such as lumped masses, stiffness elements, dampers) under different vertical and horizontal excitations, different load combinations, isolating methods with assistant from other engineering field like mechanical, material engineers.
- Geothectical studies and foundations (nuclear island) design.
- Concrete aging effect under local circumstances.
- Inspection methods, especially, non-destructive methods, quality control and quality assurance.

Studying new materials like carbon fiber polymers, carbon nano-tube, polymers and their applications in shielding and lightweight concrete.

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منشآت الإحتواء في مفاعلات الطاقة النووية

إعداد

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المشرف

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ملخص

منشآت الإحتواء النووية هي واحدة من أهم المرافق في محطات الطاقة النووية نظرا لوظيفتها في ضمان الأمان والأمن وتوفير العملاتية لتحقيق الغاية منها. و ما أعطى تلك المنشآت هذه الخصوصية او الفرادة ليس فقط تصميمها لإحتمال القوى او الأحمال التي قد تتعرض لها مثل المنشآت الأخرى بل ايضا لتوفير الحماية المادية من الإشعاع للبيئة المحيطة و حماية النظم والمكونات الداخلية من الأحمال الخارجية سواءالطبيعية أو الناتجة عن الانشطة البشرية خلال ظروف تشغيلية مختلفة تتفاوت من ظروف تشغيل طبيعية إلى حالات الطوارئ العرضية في حال حدوثها مع ضمان استمرار المنشأة في القيام بالوظيفية الاساسية لها وهي توفير السلامة. هذه الدراسة هي محاولة لإستعراض هذا النوع من المنشآت خاصة و أنه لم يسبق تشييدها في الأردن، و هي تشتمل على : التعريف ، الأهمية ، الأنواع المختلفة ، التصميم والتحليل من جهة نظر إنشائية مع تقديم ملخص لتصميم الوقاية الإشعاعية ، استعراض بعض كودات البناء المتعلقة بهذه المنشآت في الولايات المتحدة الأمريكية بنوع من التفصيل و التعرض لبعض الكودات اليابانية و الاوروبية. عقد مقارنة مختصرة بين كودات البناء المستخدمة في اليابان و ما يقابلها في الولايات المتحدة. تم تضمين الدراسة تحليلا لنموذج مبسط لمنشأة إحتواء باستخدام برنامج STAADPro.2007 لإيضاح بعض خصائص هذه المنشأة تحت تأثير الزلازل وفق الإستجابة الطيفية المعتمدة سابقا في الولايات المتحدة و أخرى تمثل مدينة العقبة. وأظهرت الدراسة بأن هناك إختلافا ملموسا في نتائج التحليل ما بين الإستجابة الطيفية المعتمدة في الولايات المتحدة وتلك الممثلة لمدينة العقبة.