Estimation of the Safety Margins between Design and Failure Conditions of PWR Containments

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Abstract

The design of a nuclear power plant imposes to build an external barrier, nominally the containment, made of concrete reinforced with steel bars which very often are pre-stressed in order to augment the overall capability to resist to tensile stresses. The state of the art of these huge structures did not advance lastly but the new features of future plants (bigger than the current standard in size and power or based on innovative technologies such as the Fast Gas Reactors with higher operating pressures) are going to push the industry rethink the current standards in order to meet new requirements at least in cost terms. The key variable for the containment building is the design of an adequate margin which, from a structural point of view, corresponds to determine the strengthen configuration of concrete and steel as a response to internal and external loads. In this paper we analyze the existing and more common configurations, as they were adopted by the US nuclear market, from to Sixties to these days, and then, provide some calculations of the safety margins for a classical large PWR dry containment. The example provided in the calculations is executed mainly by means of the data provided in the Safety Analysis Report, SAR, of an existing plant, the Indian Point Unit 3, IP3.

The work is organized as follow: Sections 2 and 3 of this paper give an overview of the different types of containment built in the US. In Section 4, the detailed model structure is provided and the scenarios analyzed together with the results of the study are described. Finally, a discussion of the obtained results and some future utilizations of the model are given.

In addition to the calculations performed as an illustrative exercise for the 314 class at MIT, this paper also analyzes the current regulatory framework and standards of the ASME and ACI codes and emphasizes, in the conclusions, the excessive use of conservativism and the contradictions in the adoption of these codes.

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Source of the figure: layout of the EPR containment from AREVA.

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1. Introduction: some Historical Notions about Containments

The containment building, which constitutes the ultimate barrier for each unit, has the fundamental task to protect the people and the environment around it in case of an accident. Because of this, the so called, nuclear island (basically everything that is inside the containment) is subjected to massive amount of regulations which are often very complex to decipher and apply. A containment building, in its most common usage, is a steel or concrete structure enclosing a nuclear reactor.

Containment buildings are an intricate and expensive part of a nuclear plant and the attention to them is also lately increasing together with security concerns.

Designers of US containment are mainly: GE who produced the BWRs containments, while PWRs containments have been provided by the Combustion Engineering Co., the Backcock Wilcox Co., and the Westinghouse Electric. Co. The Stone Webster Co. also designed several containments with reduced internal air pressure versus the more common dry-containment type. In the United States, the design and thickness of the containment are governed by 10 CFR 50.55a.

Figure 1: A LWR containment building. Source: anEPRI study on Aircraft Crash Impact to Demonstrate Nuclear Power Plant's Structural Strength" December 2002.

Some historical notes about containments: the first one used to house a large power reactor (> 1000 MWe) was the Connecticut Yankee, North-east Utilities deformed-bar, reinforced-concrete containment for a Whestinghouse PWR reactor with a design started in 1962 and completed in 1967 by the Stone and Webster Engineering Co. The design essentially used the working strength design provisions of the then-current ACI Standard 318-63 Building Code augmented by agreements made between the Atomic Energy Commission (AEC) and the Utility Owner of the plant as documented in the Safety Analysis Report for the plant. [14]

Note finally that in the US last containments were built in 1976 and that modern containments types can be found mainly in Japan, while in the Soviet Union it was normal practice not to build containment buildings. This, along with the unstable nature of the RBMK reactors, led to the catastrophe of the Chernobyl accident. In the case of these types of reactors it would be more proper to refer to the building housing the reactor as a reactor building rather than as a containment building.

2. LWR Containment Types and Characteristics

Pressurized water reactor (PWR) containments typically consist of heavily steel-reinforced concrete cylinders ranging in thickness from 1 meter to 1.3 meters, capped by a hemispherical dome of steel-reinforced concrete. The cylinder is typically 40 meter high, with a 40-meter diameter. Reinforcement bars that form a cage within the concrete are typically Grade 60 #18 steel bars on 30 to 40 centimeters centers. A #18 rebar is 5.6 centimeters in diameter – about the size of a man's forearm. Pressurized water reactors constitute about two-thirds of the 104 reactors operating in the United States.

Boiling water reactor (BWR) containments typically consist of a steel containment vessel surrounded by a reinforced concrete shield that typically has a thickness of four feet or greater and is housed within the reactor building. The primary containment of a BWR is typically one-third the diameter of PWR containment.

Figure 2: Different types/configurations of containment building as they are in the US.

PWR common designs are categorized as either "large-dry," "sub-atmospheric," or "ice-condenser."

For BWRs, the containment and missile shield fit close to the reactor vessel. The reactor building wall forms a secondary containment during refueling operations. The containment designs are referred to by the names Mark I (oldest; drywell/torus), Mark II, and Mark III (newest). All three types house a large body of water used to quench steam released from the reactor system during transients.

Different properties and features of these containment types are reported in detail in Table 1.

Table 1: Main containment features and characteristics of the US fleet divided by design type (Pressure Water reactors and Boiling Water Reactors).

Characteristics of the US Containments (109 units)	BWR Containment Design/Types			PWR Containment Design/Types (5)		
	Mark I	Mark II	Mark III	Sub-atmospheric	Ice Condenser	Large-Dry
Number of Units	24	8	4	7	9	**57**
Pressure Suppression	Yes	Yes	Yes	No	Yes	**No**
Number of Barriers	2	2	3	3	3	**3**
Volume, (10^3*m^3)	12	15	48	52-70	36- 40	**46-100**
Heat Capacity, billion of BTU	1.7	1.3	1.3	-	-	-
Design Pressure, MPa	0.528	0.48	0.20	0.41	0.30	**0.42-0.52**
LOCA Pressure, MPa	0.4	0.4	0.16	-	-	**0.34**
Reactor thermal power (MW_{th})	1593 - 3293	3293 - 3323	2894 - 3833	2441 -3411	3411	**1500 - 3800**
Containment free volume (ft^3) drywell wetwell	200.000 - 320.000 110.000- 180.000 90.000 - 140.000	200.000-310.000 140.000 - 190.000 340.000 - 500.000	1.440.000- 1.800.000 250.000 - 280.000 1.165.000- 1.550.000	1.800.000	1.200.000	**2.600.000**
Containment free volume (10^3*m^3)	15.86	25.63	49.43	54.93	36.62	**79.34**
Cont. volume to thermal power ratio (m^3/MW_{th})	7.53-3.64	4.51-4.56	12.52-16.59	20.52-21.30	11.14	**26.32-30.67**
Containment strength Containment design pressure (MPa) Median containment failure press (MPa) in IPE	0.49-0.53 0.78 - 1.41	0.41-0.48 1.07- 1.42	0.2 0.49-0.75	0.41-0.52 0.93-1.00	0.18-0.31 0.35-0.76	**0.38-0.52** **0.72-1.41**
Containment construction	22 steel 2 concrete	1 steel 7 concrete	2 steel 2 concrete	7 concrete -	7 steel 2 concrete	**7 steel** **50 concrete**
Vapor pressure suppression system	Vent header with vertical bents DW/WW vacuum breakers	Vertical vents DW/WW vacuum breakers	Horizontal vents and SPMU(1) DW/WW vacuum breakers(2)	No	Ice condenser and recirculation fans	**No**
Containment heat removal system (3)	RHR(1) system in SPC(1) or DWS(1) mode	RHR system in SPC or DWS mode	RHR system in SPC or DWS mode(4)	Containment spray(6) and fan coolers	Containment spray(6) and fan coolers	**Containment spray(6) fan coolers**
Combustion gas control	Inerted by N_2	Inerted by N_2	Igniter System	Hydrogen recombiner (for design-basis)	Hydrogen igniters	**Hydrogen recombiner (for design-basis)**
Containment venting for pressure control	Hardened vent pipe requested by CPI(1)	Hardened vent pipe not requested byCPI	Hardened vent pipe not requested by CPI	-	-	-
Allowable Leak Rate (volume %/day)	0.5	0.5	0.4	0.1	0.25	**0.1**
Capability Pressure (MPa)	0.91 (7)	1.07	0.52	1.03	0.45	**0.92**

(1) RHR - Residual Heat Removal; SPC - Suppression Pool Cooling; DWS - Drywell (or Containment Spray System; SPMU - Suppression Pool; (2) River Bend does not have an SPMU systems or DW/WW vacuum breakers; (3) There is also a fax cooler system for CHR during normal plant operation. It is not a safety system and is usually not credited in the PRA; (4) River Bend does not have a containment spray system but has two safety-related containment unit coolers; (5) From NUREG-1560, November 1996; (6) Recirculation spray, taking suction from the containment sump; (7) CPWG utilized the capability pressure predicted for Browns Ferry

3. Specific and Useful Notions about Containments

3.1 Containment Functions

The primary function of the containment building is to contain radioactivity in case of an accident from public and to maintain the internal pressure without leaking in case a major accident occurs:

• Public Protection
- Retention of radioactivity
- Retention of missiles (internal missiles)

But beside those, there are further implications/functions of primary importance which have to deal with the protection of the plant itself and we reported as:

• Protection of Plant Systems
- Natural elements (flood, storms, hurricanes, tornados and wind)
- Human actions (crashes and explosions)
- Fires
- Missiles and planes (external missiles)

Finally, consider that at the containment are anchored many of the systems it contains so it has to support their weight and other permanent load such as dead loads:

• Structural Support of Systems
- Routine
- Seismic or other dynamic effects
- Internal loads during accidents

Note that from an economic point of view, assess all these functions all together is impossible and that is most of the efforts about containment is to reduce costs by means of opportune optimization techniques find the right tradeoffs among all of the functionalities we reported above. Also consider that as being the last barrier of the plant design conditions should be very complex to figure and take into account even more of the functionalities introduced (or we think about new reactor designs using sodium as coolant or with higher pressure operating and accident conditions as the S-CO2 FGR at MIT).

3.2 Current Regulations

In this Section we want to give a flavor of the exceptional and sometimes atypical amount of obligations designers have to follow in the case of concrete containment buildings. Among the many regulations, guidelines and codes, the guiding one is the NUREG-0800: "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants"[3]. This constituted the reference regulatory guides for decades but today as a new regulatory framework is expected to review and add new features and constrictions to the containment design because of the list of factors reported below:

1. US nuclear power plants have an average age of 40 years and some initially undersigned age mechanisms are showing up all around the US nuclear fleet; typically, for our purposes here aging of concrete.
2. As a consequence of that many plants faced and are facing re-licensing which exposes them to more restrictive inspections.
3. New designs, as Fast Gas Reactors, are expected to render the environment within the containment more aggressive and hostile (higher values of the main thermodynamic variables such as the pressure at which they operate or higher gradients of temperature are classical examples) or others as IRIS pose new challenging specifications and functions to the traditional containment design.

Thus in the last years the Nuclear regulation Commission's (NRC) staff was currently evaluating certain regulatory guides for adequacy for use in new reactor licensing and a preview of the new out coming rules regarding containment buildings (specifically, SRP sections 3.8.1, 3.8.2 and 3.8.3) will prescribe new mandatory prescriptions on concrete and steel containments. In addition to this the NRC Regulatory Guides - Power Reactors (Division 1) also explicitly addresses In-service Inspection of Ungrouted Tendons in Prestressed Concrete Containments (Regulatory Guide, RG, 1.35 and 1.35.1), Containment Isolation Provisions for Fluid Systems (RG 1.141) and Performance-Based Containment Leak-Test Program (RG 1.163) to which descends the RG 1.136 regulates Construction and Testing of Concrete Containments. Containment System Leakage Testing Requirements addresses a regulatory position which addresses new containment system leakage testing necessities. Ultimately, the Reactor License Renewal process gives further regulations for the future installations of plants in the united states: Reactor License Renewal Guidance Documents: NUREG-1611"Aging Management of Nuclear Power Plant Containments for License Renewal" gives evidence of needing in terms of in-service inspection requirements (as promulgated in 10 CFR 5.55a for license renewal).

Containment construction criteria, which include design loads, load combinations and acceptable behavior applicable to concrete nuclear containments, developed originally as a unique combination of mechanical and civil structural engineering procedures. They are composed of procedures considered in design and analysis of boiler and pressure vessel components developed by the ASME Boiler and Pressure Vessel Code Committee, and those procedures used in design of conventional concrete building structures as developed by the American Concrete Institute (ACI) 318 Code Committee. This situation naturally follows from an understanding that such containments perform a dual function: (1) to be a building structure used to house, and to protect from design-basis hazards, nuclear safety-related structures, mechanical and electrical components, and distribution systems associated with a reactor coolant system; and (2) to serve a primary function as an engineered safeguard to contain the postulated radiological consequences of a loss-of-coolant accident in the nuclear steam supply system.

In addition to the pure regulatory framework imposed by the NRC, we also found a historical contraposition of the rules that of the ASME imposed to all boiling and vessel systems and the specifications given by the ACI in case the containment is going to be built with concrete.

3.3 Containment Load: Factored Loads

The load, to which the containment should be exposed to, constitutes a major source of uncertainty and it reflects the various functionalities presented in the previous section. Problems here are determining the different loads and types of loads and their combination concurring to the "worst possible scenario". Moreover also note that all the uncertainties coming from the PRA I and PRA II analyses, involving the deepest barriers of the plant sort of cumulate in containment analysis, and also sum up with the uncertainties naturally arising from the definition of a proper environmental model of dispersion after this ultimate barrier fails. Furthermore also the evolution of these loads within the accident time constitutes a major concern especially regarding new under design technologies such as the S-CO2 FGR here at MIT where the design of the containment will be crucial because of the different amount of heat (due to the different fuel and coolant) and of the different heat transport mechanisms.

Another interesting aspect we have to consider is that the distributions of loads could be both of dynamic and static nature but in the particular example we are going to consider, the Indian Point 3 plant, are not computed to keep calculation easier.

Figure 3: Load combinations for IP3 [10].

In order to incorporate the earthquake ground acceleration load, which has a dynamic nature we should use the model proposed by Wen [13] to combine them as shown in figure 3 (Table 3 offer an idea of the final output that is usually provided by these models). Because of all the possible sources of uncertainties already mentioned, ASME and ACI codes reduce them by factorizing loads into different well specified load factors and also as in Section III of the B&PV ASME Code factorize loads and take into account the standard deviation and mean values of the concrete properties. It is thus obvious that a probabilistic framework would fit fine in this context and later section will go further explaining it.

3.4 Containment Capacity and Accident Evolution

The capacity of a given structure is subjected both to vary over time and to uncertainties around its value at a defined instant time. To address uncertainties o structural strength of the containment we here provide a list of related meaningful considerations:

1) The retaining of the CG (presence of leaks, ruptures) and the size of the leak or penetration (which determines a depressurization rate and also a possible release of contaminated material);
2) The chemical reactions between steel and oxygen, concrete and CO_2 or other minor reactions capable to affect, in the few hours of the accident, the strength of the concrete walls or liner;
3) Temperature on the wall of the CG (ASME code provides different margins depending on T);
4) The stratification of the gas mixture or other mechanisms that can reduce locally the strength of the concrete or of the liner surface.

In the model we are using in these paper, as shown in next section, a normal distribution will take into account of possible uncertainties in its mean value[1]. The capacity provided in figure 4 as the upper curve is always thought as a constant line but actually over the reactor's lifetime it would not be appropriate to consider it as a constant. In reality by recalling the factors listed above a concurrent series of mechanisms are going to modify/affect its initial design value, and consequently its value over time:

1. Corrosions or any other Aging Mechanisms
2. Chemical reactions with CO_2 (carbonation of the concrete)
3. Fragility curve of the steel (ASME code for the temperature effects)
4. Penetration due to impact with internal or external missiles
5. Water release within the containment

The discussion provide her is actually given for completeness but actually none of these mechanisms will be analyzed in the present analysis. But the hope is that, as emphasized in the previous section, the reader will be aware that new regulations (by new we mean the last 15 years or so..) are trying to consistently address material degradation mechanisms and that a complete analysis, as recent papers in the field are showing, will include the effects of uncertainties as time change.

Now we need to connect the concept of load to the concept of capacity. The load is the evolution of pressure (which is varying in a more stringent time that is the time of the evolution of a possible accident) Figure 4 provides a possible example of pressure load history during different scenarios of evolution of the accident and leading to different reliability end states. The capacity (expected failure pressure) is the demarcation zone for scenarios ending in a failure state. Note that from this figure could be better explained trough the set of observations that follow.

[1] The Capacity is given primarily by the external pre-stressed concrete barrier's strength so in this case it's deducible from the mechanical properties of a single component and this because the internal steel liner plays a marginal role in supporting the structure.

The black line shown in figure 4 represents the pressure evolution during an accident. The accident can follow different paths or scenarios. Until the peak, the accident can be thought as the DBA or classical LOCA where the energy and amount of inventory reversed from primary trough containment are essentially determining it. The peak pressure is used to determine the design basis accident pressure by increasing it of a factor that usually ranges from 10 to 25%[2]. The bifurcation point is, in this illustrator example, the last point in which to achieve or not the desired mitigation trough i.e. the ECCS system. If we don't have the expected mitigation the pressure evolves following the path of a severe scenario with meltdown where the pressure is going to pass the boundaries imposed by the design pressure and then ultimately ends in a failure state of the containment: this is the region we are going to refer. The region between the red and the brown line constitute the safety margins in terms of pressure for the reactor building. Passing the red line means going below the required margins for safety, and, passing the brown one means going through a failure mode of the containment. This second upper boundary is also known as capability of the system and we are going to refer to it later on in this work when the phase of conceptualization of a calculation will require that. Finally note that the graph presented below refers to pressure and not to stresses. A proper safety margin representation will be in a stress-time space and would take into account not just pressure but, as we anticipated in the previous Section, also other factors piling up with it such as temperature, dynamic loads.

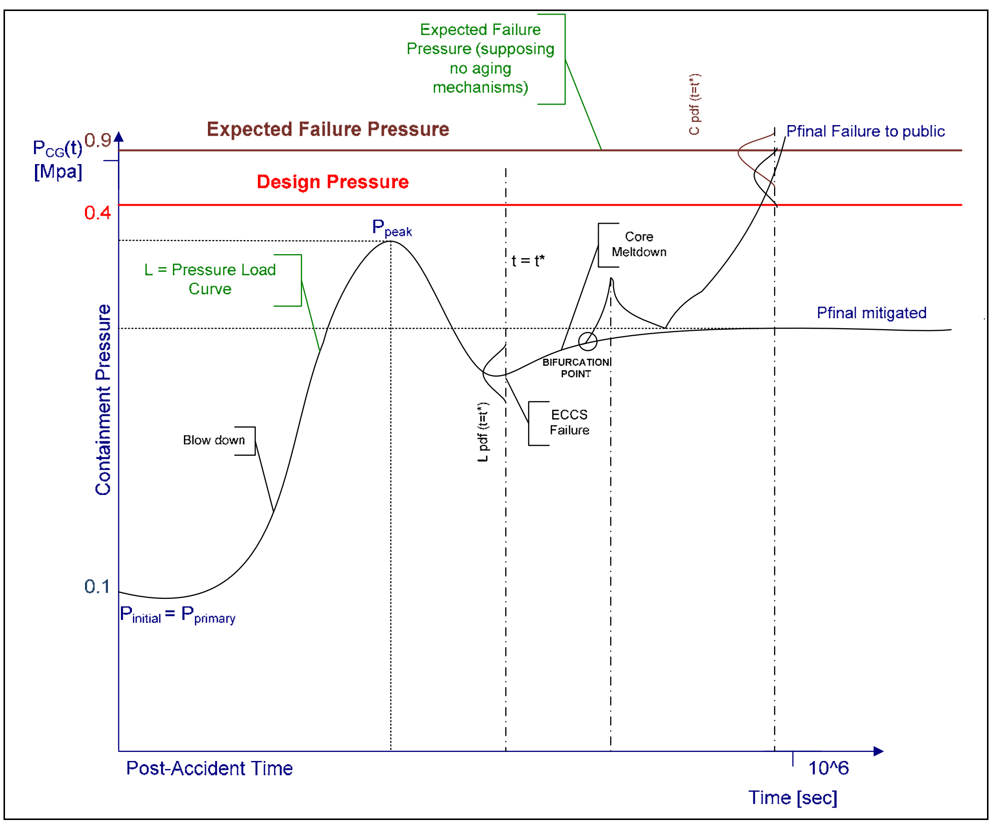

Figure 4: Pressure histories in the containment after a large LOCA leading to reactor meltdown.

[2] Containment design pressures typically have been defined as 10%-25% above maximum pressure. Initial design of containment typically used a 40% margin for containment barriers and 20% for interior structures that did not serve a barrier function. These margins were reduced to half these values when all the geometry of the interior compartments are known as part of the final design. An even lower margin can be used if rigorous modeling and dynamic load effects are considered.

3.5 Safety Margins in a Probabilistic Context

The safety margin nomenclature does not find an official definition in the literature. Despite its wide

In general, the Safety Margin, SM, has been devised to deal with uncertainty. But different types of

The more general definition of safety margin was cast for structural-mechanics analyses, recognizing

for load and capacity, which shape the bases for the more general definition of safety margin.

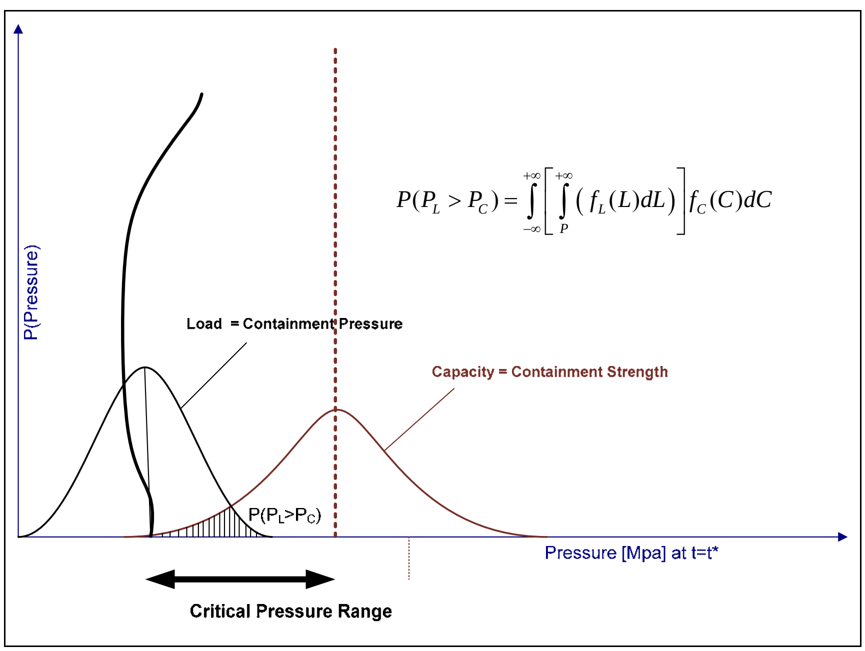

Figure 5: Capacity and load distributions within the containment and SM def.

The quantity, safety margin, SM, describes the reliability of a barrier or system in light of load-strength or load-capacity considerations. It is computed from the following equation:

$$SM = \frac{\overline{C} - \overline{L}}{\sqrt{\sigma_C^2 + \sigma_L^2}} \tag{1.1}$$

where C is the mean capacity or containment strength, L is the mean load, σ_c is the capacity standard deviation, and σ_L is the load standard deviation[3]. Thus, the safety margin results to be an indirect measure of the overlap in the probability density functions and can be used to estimate the probability that the load does exceed the capacity (i.e., the probability of failure):

$$P(L > C) = \int_{-\infty}^{+\infty} \left[\int_{C}^{+\infty} \left(f_L(L)dL \right) \right] f_C(C)dC \tag{1.2}$$

Under the strength-stress model (see Fig. 5), for failure mode i[4], the reliability is defined as the probability that the strength, C, is greater than the stress, L. The strength and stress are in general sense. Any resistance, such as a yield strength, allowable force, or allowable deflection can be considered as strength, and any loading-type quantity, such as a bending stress, external force[5], or deflection, can be considered as a stress.

For normally distributed C, L, the probability of failure (i.e., 1-reliability) can be expressed as a

sole proxy for reliability in many applications. The other reason is that the design goal (in the nuclear industry, especially in the related field of the pressure vessel construction as presented here) is to build components and systems that have negligible failure probabilities. This can be attained by having sufficient safety margin (i.e., a large separation between mean capacity and load relative to their combined standard deviations). This solidified the generalization that having adequate safety margin is a sufficient condition for high reliability. Thus, a highly reliable system (i.e., one in which the probability of failure is negligible) shows no overlap between the probability densities of capacity and load. On the other hand, refer to figure 5, a non reliable system shows a configuration of the system that is defective because it shows a common area for the capacity of the system and the load at the accident.

With the definitions provided in the last two Sections, we can now define the Capacity and Load condition in the case of the specific example of the Indian Point 3 unit for which data (in terms of distributions) are available. Before doing that please note once again that the capacity as reported in

steel are subjected to several degradation mechanism (as reported in the list given in the previous Section) not reported here and for which we assume to be all negligible over the lifetime of the reactor.

[3] The ratio in Eq. 1.1 means that for a given distance between C an L can be reduced if L or C present big uncertainties. This is a measure of the probability for the two curves to overlap in presence of flat distribution curves and its aim is different from the (C-L)/C ratio that we are going to use later which is typically showing the variation of the margin with regard of the original C (mean) value. Note the two formulations can be both used and they are consistent with each other.

[4] Note that failure mode for containment are not addressed in this paper so we refer to the general definition of a potential crack which is going to reach a defined size which in turn allow the gas to be released in the environment. This definition in its generality includes the loss of almost all the functionalities of the containment as presented in Section 3.1.

[5] Or combinations of loads as it will be the case of the factored load that we have to deal with in the example provided in this paper.

3.6 Capital Expenditures for the Containment Building

We conclude this preliminary section about the containment by providing a range of possible estimation of a large dry atmospheric containment. We will recall later on these numbers in Section 4 when it will be required to evaluate a less "safe margined" solution in terms of economical benefits.

The containment building is usually costing between 8 to 12 % of the total initial expenditure in construction and it is particularly sensible to delay in construction.

Cost of containments varies widely by design (PWR vs. BWR) and location of the plant. A BWR typically has more concrete and rebar due to wetwell /drywell configuration. The location is important because of the seismic zone. As an example, the ABWR that the Shaw Group is building in Taiwan the ground force acceleration is very large and as a result the amount of concrete/rebar is very high. There is also a variation in techniques or philosophies adopted worldwide; some France plants like the "Brennilis" are, regardless of costs, completely pre-stressed including the base-mat while an US plant like "Ginna" is partially pre-stressed axially and just in the cylinder.

Also the labor rates will impact in addition to material costs. For instance, in the Northeast US labor rates are much more than in the Gulf Coast. This will vary worldwide also and likely depend greatly on labor productivity. QA Cat 1 concrete runs about $120/yd^3 these days[1]. I can take that a complete new nuclear plant holds 100,000 m^3 of concrete and roughly 50% goes to the containment. The labor is hard to predict so it will be excluded from present economic evaluations.

Anyway if we select today's cost and suppose a containment to be costing around 20 Million man-hours for units (which is reasonable and deducible from past average construction times) done several years ago. By discounting the obtainable value by about 25% and then applying perhaps 25% to the containment we could also be able to do a rough estimation of the labor force required too. However, the concrete, rebar and labor don't make the whole containment. Most containment in fact have liners and most have a lot of embedded/attached equipment that should bring total costs closer to 15 % of the total initial capital expenditure (so something around 30 million dollar for the containment of a 2 billion dollar LWR's plant is going to be our maximum value in the calculations).

About the liner, consider that all US containments have one. In the US is part of normal practice to built it because of it provides a leak-tight membrane whose integrity is obviously important to preserve. Beside this main scope, it is anchored to the concrete shell with systems of channels and angels or grid of studs which serve to limit large scale deflections and, in turn, limit strains.

We conclude giving the assumptions we are going to use. Imagining not to consider labor costs, specific costs for concrete are 270 $/m^3 [1]and for the steel used for the bars a price estimate for steel equal to 2.5 $/kg (consider that steel has a density of 7850 kg/m^3).

4 Calculations and Results

4.1 Preliminary Calculations from the IP3 PWR

This Section constitutes the core, in computational terms, of the present paper and its aim is to evaluate the safety margins of a concrete-steel reinforced containment. The objective is to answer, or at least provide a solid basis, to answer the question if it is possible to double the pressure of a LWR's containment and maintaining appropriate margins as well. The evaluation of the safety margins of any building implies the knowledge of the capacity (or maximum strength or capacity = C) of the structure and the load (or maximum stress or load = L) to which the structure is exposed to. The safety margins can be roughly determined by the difference of the two mean values of the strength and load; our focus here is first to provide an estimate of the safety margins under initial proper load conditions which will be then doubled in a second moment. The different load conditions are deduced by elaborating some results available from studies conducted on a selected reference plant (Indian Point unit 3) which had been chosen for both its representativeness and the availability of studies related to it [9][11].

4.1.1 Plant's Description and Basic Assumptions

The initial geometry of the problem is defined by Figure 6 which sketches the containment of the Indian Point while figure 7 and 8 are showing the configuration of the bars immersed in the concrete we are going to refer to. Table 4 summarizes the characteristic of the Indian Point unit 3 plant.

We here report the material properties and geometry of the IP3 plant as extracted from the various pertinent sources available.

Material Properties

The material properties intrinsically involve epistemic and aleatory uncertainties. Hence, an appropriate probabilistic model should for instance consider the material strengths as random variables while Young's modulus and Poisson ratio as deterministic ones. The properties of the concrete and reinforcements are summarized as follows as referred to the Indian Point unit 3's containment:

A. Concrete: the <u>concrete</u> has a minimum compressive strength f'c (or design minimum ultimate strength) equal to 3000 psi (20.68 MPa) and which we should assume is normally distributed and statistical data of available tests provides a μ,σ= 4896, 627 in psi (34.37, 4.32 MPa) [10], [11] . Young's modulus and Poisson Ratio are equal to E_c=3.1 MPsi (21.373 GPa) and 0.2 which is equal to the bar value while bars have a E_s=29 MPsi (20 GPa) . The statistic used for the steel <u>bar</u> yielding strength are obtained by considering the predominant reinforcement as constituted by 18 bars as clearly shown by Table 2. Thus from same data, f_y result to be normally distributed with μ,σ= 71.8, 5.18 in Ksi (495, 35.7 MPa). Note that the E_c value of the concrete has been varied between the 30% and 100% in order to evaluate different behaviors of the concrete with cracks.

B. Steel Liner: the plate steel liner is carbon steel conforming with ASTM designation A442-65, grade 60. This steel has a minimum yield strength of 32,000 psi ($\sigma_{y,min}$= 220MPa) and minimum tensile strength of 60,000 psi ($\sigma_{t,min}$= 413 MPa) with an elongation of 22% in an 8-inch gauge length of failure.

Geometry

Figure 7 shows the IP3 containment analyzed.

Figure 6: Containment with spherical shell on the top of the PWR type as for the Indian Point 3 unit.

Main dimensions are:

R+L=H m R= 20.574 m H= 66.7512 m L= 46.1772 m Ss= 1.0668 m Sc=1.3716 m

Containment Capabilit

Containment capability is given from the probabilistic safety study of Indian Point Plant in the appendix 4.4.1 [11]. It is an essential element of risk evaluation for degraded core events because it defines the realistic lower bound ultimate containment building capability for withstanding internal pressure loads. This had been determined to be 0.972 MPa (or 141 psia) for IP3 based on a failure criterion defined as the state of having reached general yield in the reinforced walls. At that pressure the strains start to significantly increase and radial deflections of the containment wall may become unacceptably large. This value is consistent with the average value for capability we provided in Table 1, so it confirms that the IP3 is a good representative example of large dry containment building. The capability was defined as the maximum combination of temperature and pressure to produce a general yield state.

Load factors

Load factors used in IP3 safety report were calculated by following the ACI 318-63: load factors in the design primarily provide for a safety margin on the load assumptions. Specific combinations used in the design are presented below. The design included the consideration of both primary and secondary stresses. The load capacity (the capacity in the old way to define it) was based on the ultimate strength values presented in part IV B of ACI-318 as reduced by a capacity factor $\phi = 0.9$ for flexure and 0.85 for diagonal tension, bond and anchorage. For the liner steel it is equal to 0.95 for tension while for compression is maintained below 0.95 yield.

The loads used to determine the required limiting capacity of any structural element on the containment structures are:

(1) $L=D+/- 0.05\ D +1.5\ P + 1.0\ (T +T_L)$

where D=dead load of the structure, P= pressure associated to the DBA accident as shown in transient calculations and T = load due to the maximum temperature gradient trough the concrete shell upon temperatures associated with 1.5 P while T_L= load exerted by the liner based on 1.5P.

Three other loads were defined by inserting also other possible contributions such as earthquakes, winds and tornado. We here report just the first condition which mean that the containment has the capacity to withstand loadings at least 50% greater than those calculated for the LOCA alone[6].

The loads we are going to consider are the DEAD LOAD, the ACCIDENTAL PRESSURE and TEMPERATURE and the PRESTRESSING LOAD.

[6] All the structural components are designed to have a capacity required by the most several loads combination. The loads resulting from the use of these equations will hereafter be termed "Load Factors".

Results of the SM analysis conducted on the IP units 2 and 3.

Containment can withstand a pressure of 0.972 MPa which is 2.7 times the DBA pressure (0.32 MPa) and this without impairing the functional capability of the containment. This analysis at that time was conducted manually at different levels of the structure and in correspondence of critical regions like membrane region of the dome and cylinder, base-mat, liner or close to large penetrations. The critical region was to be found in the cylinder just above the base-mat (first elevation in Table 2) in a zone where the seismic reinforcing steel is reduced.

The 2.7 factor found corresponds to the conservatism applied in the original design as summarized below:

1. Application of load factors
2. Application of capacity reduction factors
3. Strength of liner not accounted for
4. Minimum strength of the material accounted for
5. Seismic rebar resisting LOCA loads
6. Designer conservatism

So all of these considerations concur to prove that the actual capability is higher than 0.972 MPa. Note finally that the role of the concrete cracking on shell stiffness is also incorporated into the calculations as required.

4.1.2 Calculations and Analyses

The present analysis is conducted only on the main cylindrical wall and steel liner and upper dome are not considered so any analysis of the dome over the cylinder is provided here.

A welded steel liner with a minimum thickness of one-quarter inch it is attached to the inside face of the concrete shell. The load carrying capacity of the liner is usually disregarded from calculations because this is not functional to the scope of the liner which is to ensure a high degree of leak tightness.

Note that the surface of the containment is imagined without penetrations, personnel locks and equipment hatches.

The containment wall is reinforced with hoop meridian and diagonal rebar. The typical rebar arrangement we refer to for the cylindrical wall is shown in Figure 8. The hoop and meridional rebars are divided into two groups and each group is placed close to the wall surface. We consider that the tendons move in the longitudinal direction the same as the concrete. So the rigidity is calculated on the basis that "plane sections remain plane" and that transverse Poisson's ratio of the bars can be treated the same as the concrete. The details of the rebar arrangements for the cylindrical portion of the IP3 containment are shown by Table 2 [11]. The sense of this table is to observe that rebar is changing at different elevations (note that a proper calculation would inspect the various elevations by means of a traditional finite element mash but also the safety report did not conducted it at that time). An average value among all the elevations has been used to perform our calculations. Among the different elevations and point in space we selected the region near the base mat. We considered the cylinder to be built-in at the position it joins the base mat (zero radial deflection, zero slope of radial deflection).

About pre-stressing conditions we assume there is no bending of the shell in the unpressurized condition and that pre-stressing is going to be achieved by "post-tensioning".

Finally we assumed elastic behavior and we define flexural rigidity on the basis of the section of material shown in Fig. 8.

Figure 7: Rebar arrangement for Cylindrical Wall.

Figure 8: Section of the reinforcing steel array for IP3.

Evaluate the impact of changing the capacity of the bars in the longitudinal tendon on the required pre-stress level to prevent tensile in the concrete upon pressurization and verify that the obtained values stay beyond the prescribed margins requires calculating the required pre-stress level of the generic containment shell.

First we compute some constants that will be used and then perform an analysis of the single piece of concrete based on the thin shell theory as cited in L54 notes of the 314J class. Beside the different values assigned to the material and some differences in the design, the calculation proposed here, which is qualitative but provides the order of magnitude we deal with in safety analysis is probabilistic and not deterministic. Assumptions of the analysis are reported at the end of the present Section.

In order to compute calculation and uncertainties propagations we made use of the Crystall Ball [17] software which allows using as an input the different load distributions composing the factored load formula, and easily provides as an output the probability of failure of the structure or, in other words, the margins we are interested to. The software in use was also useful to optimize the results as a function of costs and to perform Monte Carlo sampling of crucial parameters such as Ec.

Remember the intent is to verify the margins under accident conditions, so we will refer mainly to regulations directly imposing specifics and affecting the margins in question; mainly NUREG-0800 and Section CC-3540 of ACI-359 applies specifically to the pre-stressed containment designs, however, all manner of ASME III and ACI codes apply. Also Code of Federal Regulations 10CFR 50.55a applies and in accordance with ASME Section XI Subsections IWE and IWL. The minimum value usually is reported to be equal to 2 (see NUREG-800), in our calculations, conservatively we are going to assume it in the range that goes to 2 to 3 (IP3 has a 2.7 [11] value of applicable margins, see eq. 1.8 in the next Section).

<u>Calculation 1:</u> Safety Margins of the IP3 plant with our Model

The safety margins of the structure as reported in figure 6 from the shell theory and thus referred to figure 7 and 8 is computed as follow:

$$P_a = |\sigma_\theta - \sigma_r| \tag{1.3}$$
$$L = P_a\gamma + D + F \tag{1.4}$$
$$\gamma > 1 \rightarrow (1 + 25\%) \tag{1.5}$$
$$C = \varphi f'_y$$
$$\varphi < 1 \rightarrow = 90\% \tag{1.6}$$
$$F < 85\% f_c \tag{1.7}$$
$$C - L > 2 \tag{1.8}$$

So, according to our assumptions, the load factor "L" of Eq. 1.4, or the design pressure, is mainly defined from the accident sequence pressure Pa and increased of the margin we previously described. Here the γ value is conservatively (because here we don't know all the details of the load) settled equal to 1.25. The design pressure in Eq. 1.3 is obtained analyzing the concrete with the thin shell theory. F and D are then obtained by assuming that the dead load is mainly the weight of the structure and its uncertainties are overshadowed by the large variability introduced by the other two, so it has a minor effect on the limit state probabilities, thus its value is going to be deterministic and equal to 150 lbs/ft^3. The accidental pressure is indeed a random variable and has been treated as a Gaussian with mean value of 47 psi = 0.32 MPa and std dev of 5.02 Psi (0.0346 MPa) , according to the data of ref. [11]

The capacity "C" is in contrast decreased to take into account possible imperfections of the material and uncertainties regarding the bending moment acting on the structure and here corresponds to a residual factor=90%. Note that the value we choose for f'y is set to be equal to 50 % of the yield strength of the material as imposed to be for average tensile stresses for bounded reinforcing steel.

In addition to this, we have to avoid that the concrete faces creep deformations and thus we impose a limit on the pre-stress F, Eq. 1.7, as given from the ACI-359.

Thus constriction imposed on the material translates into the tensions within the shell of reference:

$$\text{MAX}(\sigma_{Hs,preS} = (1-Xhs)/Xhs * \sigma_{Hc,max} \ , \mu\sigma_{Ls,preS} = (1-Xls)/Xls * \sigma_{Lc,max}) < 0.85fc \tag{1.9}$$

Equation 1.8 provides ultimately the safety margins in a deterministic formulation while the probabilistic one, assuming that both, Load and Capacity, are normally distributed as was given in Eq. 1.1, is at the moment not considered. Because the system of the equation identified revealed to be linear, the formulation of Eq 1.2 can be used and thus the margins we found are going to be expressed in terms of their relative initial capacity (see also footnote 3 at page 13) :

$$SM\ \% = (C-L)/C \tag{1.10}$$

Note that, as stated before, we expect the absolute value (C-L) to be equal to 2 at least (see NUREG-800) but our calculation initially find evidence of bigger margins because we don't consider abnormal load conditions (see Table 5 where SM=8.05). By varying the coefficient γ in equation 1.5 (Table 5 makes use of a typical 1.25 factor which is used for DBA Accidents)in typical ranges are used to express abnormal load conditions (to value of 2 and 3) we can take into account of other possible load combing into the final load factor such as wind or earthquakes.

So the calculation we settled takes into account all this limits and optimize the parameters we identified as crucial which are the Area of the steel bar=As, the thickness of the shell=Sc=t, the Young Modulus =Ec for the concrete which should be subject to cracking [16], the Pressure (by means of γ) within the containment and the Volume V_c which is actually based on the pressure itself (See figure 9). The software is calculating the safety margins following the criteria already defined[7] and if the margins are not respected the area of the steel in the structure is increased or Ec increased in such a way to fit the constraints given by the problem and optimize costs.

Figure 9: Free volume and equilibrium pressure LOCA.

[7] Note there are a set of different criteria we can establish and use to set up our limits or boundaries to the system which are related to the different performances desired from the structure (here we referred to strength and thus to measures of performance based on the yielding stress because a mere structural criteria was adopted, but, there are other performances we should be interested to, such as serviceability, stability, plasticity and vibrations).

A final clarification about the way we determined costs; assuming there is no change of technology or construction methods corresponding to thickness variations of the concrete, and, of the area of the steel bars used to pretension, costs are simply calculable by multiplying the corresponding volumes of material in use for their costs for kg:

$$C_{steel} = V_{steel} * UC_{steel} \tag{1.11}$$

$$C_{concrete} = V_{concrete} * UC_{concrete} \tag{1.12}$$

Results provided by Table 5 gives an overall cost that correspond to 14.7 % of the initial investment cost we hypothesized to be for a PWR of this type (1000 MWe) around 2 billion $.

Calculation 2: Sensitivities to Material and Geometry

After the calculation of the SM, we run a couple of sensitivity analysis to verify some properties of the material such as the Young Modulus for the concrete, Ec, and to verify cost variations as a function of the parameter A_S.

A_S is responsible of both the strength of the shell element and also of the weight of steel costs over the concrete ones. The results are reported in the tables below as a function of costs and cost variation in percentage but this time with regard of the initial value given by "Calculation 1". By varying A_s above its mean value of 0.00255 m^2 we obtained a maximum 1-L/C % equal to 11.39 % for an As equal to 0.02707 m^2 corresponding to a overall cost increase equal to 16.56 % above the base case.

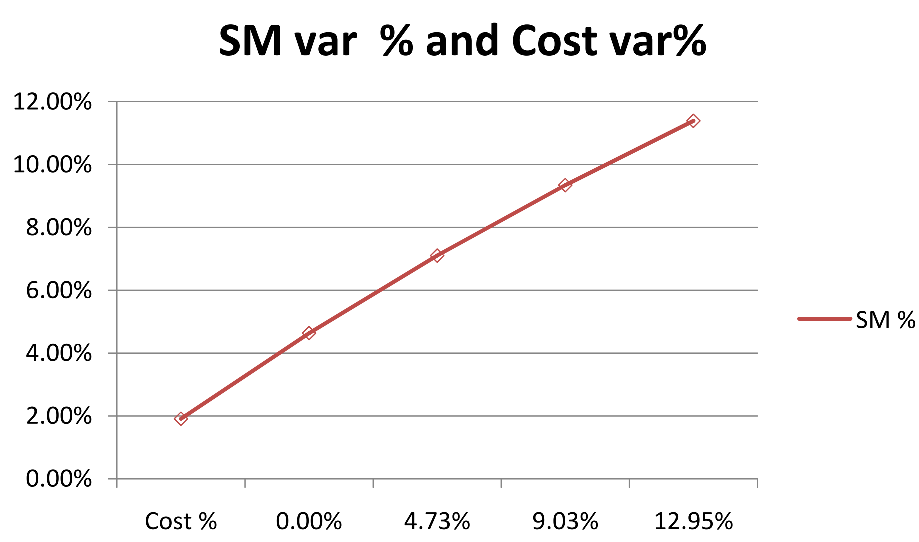

Fig 10: SM % as it varies with cost (capital cost as var %).

Third calculation we computed is a sensitivity analysis on the parameter E_c or Elastic modulus of the concrete to see how the results are varying depending on the possible initial degradation of the material such as small cracks or imperfections [16]. Results are reported below.

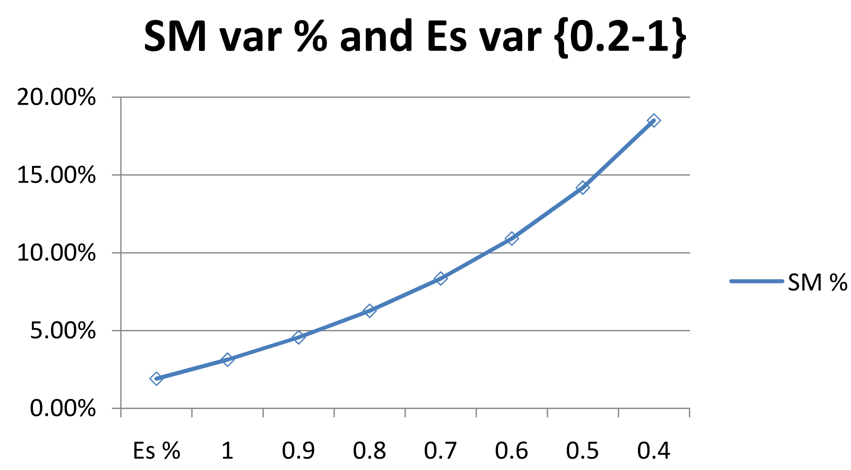

Fig 11: SM % as it varies with the young Modulus of concrete (crack effect).

<u>Final note on other calculations of interest</u>

Also a preliminary calculation has not been included in the tables but we cite it here as an interesting one; which is reformulating the same exercise in the case we use a stainless steel containment instead of a concrete pre-stressed one. Calculations are pretty straightforward here and from the basic thin shell theory predict a 40% increase of the thickness. Formulas, which are reported in the tables, have been used as a reference to validate and compare the results of the calculations performed for the concrete case. Evidently the steel containment is more expensive and not able to account all the functionalities of the concrete one. But if we were going to consider it we should clearly take into account the strength variation of the material at different temperatures (things about the restrictions given from ASME codes to the 16Cr-12Ni-2Mo Plate SA-240 type 316 of which IP3's liner is made of) thus forcing to take into incorporate temperature loads into the margins on the capacity side (which could be a good example of one of the mechanism we referred into Section 3.4 where the upper boundary of the system is varying with time, in other words C=C(t)).

4.2 Comments and Final Remarks

In this paper an overall methodology to calculate the SM for a pre-stressed concrete containment was given. By referring to a specific plant with has been particularly studied by the literature the basis to compute probabilistic calculations of the safety margins were provided. The analyses tries to put emphasis a wide discussion that has been both in class and during the writing of the paper with different Professor of the departments of Civil and Nuclear Engineering here at MIT , regarding the current methodology to address safety margins. The basic conclusions I would like to concentrate on here are even if not always clearly and explicitly addressed or still to be accomplished in possible future works, are summarized the below.

The current methodology used to address safety margins can be interpreted in respect of the following sentence that appears at the end of the safety report of the Surry's containment: "For Surry, there were no intersections of the load distributions with the containment strength distribution, and thus the DHC issue for Surry can be resolved on containments loads alone."

So current containments are adequately safe and, as partially but quantitatively presented in this paper, is opinion of the author that, thanks to developed state of the art of probabilistic criteria, the current regulation, that looks obsolete and redundant in many parts, could be replaced with appropriate ad hoc SM evaluations.

Future part of the work should consider all the basic factors composing load factors and runs the same optimization used here along all the possible beyond DBA scenarios (we here in fact just selected one and we limited the factors in the load factor formula to 3) comprehending all the loads involved in the factorized formula of the L.

A clearer framework would help to feel the gap that the hyper conservativism intrinsic to the regulation and caused by the overlapping of different standards as ASME and ACI, especially in view of the imminent development of future nuclear technologies with innovative (and thus costly) containments (See IRIS, ESBWR and also ITER). The NISTI report entitled "Structures Division Prediction of Cracking in Reinforced Concrete Structures" [16] is giving good clarifications in this sense and provides a set of different model which takes into account the different misinterpretations we cited.

Table 2: Spacing at different elevations and divided by layers for the IP3 PWR Cont [11]

elevation		*hoop*	*meridional*		*diagonal*
(m)		*(dmnl,dmnl,cm)*	*primary*	*secondary*	
		#layer #bar pitch	#layer #bar pitch	#layer #bar pitch	#layer #bar pitch
7.6		2 18 35.5	1 18 30.5	1 18 30.5	1 18 76.2
13.8		2 18 35.5	1 11 30.5	1 11 30.5	1 18 76.2
15.3		2 18 35.5	1 18 30.5	1 11 91.5	1 18 76.2
16.4		2 18 35.5	1 11 30.5	1 11 91.5	1 18 76.2
33.7		2 18 35.5	1 18 30.5	1 18 30.5	1 18 76.2
45.1		2 18 35.5	1 11 30.5	1 11 30.5	1 18 152.4

Table 3: Mean duration of the combined load acting on the containment (IP3) [11]

Load combo	***Occurrence rate [combined occurrences/year]***	***Cumulative occur. [# occurrences]***	***mean duration [sec]***
D+P	2.16E-03	8.64E-02	1200
D+E	1.64E-02	6.56E-01	15
D+P+E	1.36E-09	5.46E-08	14.81

Table 4: The Indian Point 3 Unit. Features of the Containment and of the Plant [11]

LOADS

D=dead load=150 lbs/ft3= 2226.586 kg/m3

P=accidental pressure=(m,s)=N(0.288, 0.034)
E=earthquake loads
T=accidental temperature corresponding to P

GEOMETRY

R+L=H m

R= 20.574 m

H= 66.7512 m

L= 46.1772 m

Ss= 1.0668 m

Sc=1.3716 m

MATERIAL

Concrete:

E_c=21.373 GPa

v_c=0.2

$f'_c=f'_c(m,s)$=N(33.75, 4.32) [MPa]

f'_c,min=20.68 MPa

Reinforced bars:

E_s= 200 GPa

v_s=0.2
$f_y=f_y(m,s)$=lognormal(495, 35.71)

REACTOR FEATURES

Reactor type: PWR

24 MI N of New York City, NY

Docket Number: 05000286
Operating License: Issued - 04/05/1976, Expires - 12/15/2015

Operator: Entergy Nuclear Operations, Inc.

Electrical Output: 979 MWe

Reactor Vendor/Type: Westinghouse Four-Loop
Containment Type: Dry, Ambient Pressure

Table 5: Final Results of the IP3 Calculation for Safety Margins of a Concrete Containment

GEOMETRY INPUT			*STRUCTURAL CALCULATIONS*		
t=Sc	1.371	m	εLc,max	6.69286E-05	
R	20.6	m	Wp=W,max	0.004671207	
d	0.057	m	W		
de	0.165	m	εhc,max	0.000226758	
PL	0.165	m	εbL(z)		
PH	0.2	m	εbL,max(z=0)	0	
X	-0.6855	m	σLc,max	2431003.13	Pa
Xs	0.5205	m	σHc,max	5277691.233	Pa
L	46.1772	m	σLs,preS	105323971.4	Pa
R+L=H	66.7512	m	σHs,preS	136501287.9	Pa
Ss	1.0668	m	***LOADS AN CAP***	**mean**	**[MPa]**
MATERIAL INPUT	**nominal/ distribution**		γ	1.25	dmnl
Es	2E+11	Pa	Pa*γ	142.1888416	MPa
Ec	2.137E+10	Pa	D	21.8205428	MPa
Pi	450000	Pa	F	136.5012879	MPa
vc=vs	0.2	dmnl	T	127.9699574	MPa
f'c=f'c(m,s)	33750000	Pa	L=tot	428.4806298	MPa
fy=fy(m,s)	495000000	Pa	ϕ	0.9	dmnl
fuc		Pa	C=ϕ*fy	445.5	MPa
OTHER QUANTITIES	**nominal/ distribution**		SM	8.509685116	MPa
σr	-275000	Pa	***VOLUMES***	**mean**	**[m^3]**
As	0.0025518	m2	Volume (free)	61406.58409	m^3
Xls	0.0225605	dmnl	Vcil	8456.642372	m^3
D	6.644E+09	Pa*m3	Vs/bar	0.117833069	m^3
Xhs	0.0372248	dmnl	# bars	484	dmnl
NL	3605000	Pa*m	Vs=Vs/bar*# bars	114.06	m^3
EL	2.176E+10	Pa	Vc=Vcil-Vs	8342.579961	m^3
Eh	2.143E+10	Pa	***COSTS***	**mean**	**value**
E'	2.627E+10	Pa	Unit Cost concrete	268.5646061	$/m^3
α	1.875E+09	Pa	Unit Cost steel	2.5	$/kg
γ	512503.67		ρs (steel density)	7850	kg/m^3
β	0.241911		Unit Costs	0.000318471	$/m^3
OPTIMIZATION ON COSTS			Tot Costc=Vc*Ucc	2.240521701	Million $
max F, max P, max L-S	given f'c, fy		Tot Costs=Vs*Ucs	285.1560264	Million $
	Total invest costs	2 Billion $	Tot Cost= Cs+Cc)tot	287.3965481	Million $
	Cost fraction	14.37%			

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