ON DESIGN ERRORS AND SYSTEM DEGRADATION IN SEISMIC SAFETY

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SUMMARY

Recent probabilistic studies on seismic aspects of reactor safety have not specifically included the possible influence of design errors or of some forms of system degradation during plant life-time. In this paper a rough approach is outlined for the inclusion of possible design errors in such probabilistic estimates. Also, the possible influence of existing flaws on piping system failure under severe seismic shaking is examined.

In WASH-1400 it was concluded that the earthquake risk was negligibly small compared to other reactor accident risks. One important factor was the assumption that the simultaneous failure of two systems is required to produce a core melt accident. Based on Newmark's work, the probability of damage to a safety system with a built-in safety factor of 20 is of the order of 1.5×10^{-3}. The joint failure probability of two safety systems, using log normal mean, would be of the order of 10^{-5}. When this damage probability is multiplied by the estimated earthquake probability, the probability of core melt per reactor-year, due to a design basis earthquake, would be of the order of 10^{-7} to 10^{-8}, and thus the above conclusion was reached.

In estimating the conservatism in the nuclear reactor facility seismic design, Newmark suggested a safety factor of 20 for the safe shutdown earthquake (SSE) which resulted from the conservatism in the predicted free field ground motion and in the predicted structural responses.

In this paper an estimate of the possible number and influence of seismic-related design errors is obtained by examining the historic record of such errors for a specific reactor and assuming that, with inclusion of a factor to represent a learning curve, this record would be representative of other reactors. The historic errors were divided into three categories; major, considerable and moderate. About twenty of each of the first two categories were found. The range of undiscovered seismic design errors was estimated, and Newmark's safety factor was reduced proportionately depending on the seriousness of the design error.

It was assumed that at least ten failure paths exist, rather than the single path assumed in WASH-1400; also, it was estimated that for some failure paths (e.g., liquefaction) the original safety margin would be much less than 20. Also, a closely coupled failure in redundant systems from seismic events was assumed.

Using the same relationship between recurrence frequency and acceleration as used in WASH-1400, the probability of severe core damage from seismic events was then estimated to be between 8×10^{-5} and 8×10^{-4} per reactor year.

While the absolute values obtained are subject to large error and uncertainty, they are substantially larger than the values obtained in WASH-1400, and suggest further study is needed.

The number of cycles to failure of a pipe undergoing about 100 plastic cycles due to a severe earthquake was related to original flow size, using a crude method based on measurements by Cherepanov. As an example of the results, it was found that for flaws initially about 1 inch in size, fifty cycles having about 1% plastic strain might cause failure.

1. Introduction

Recent probabilistic studies on seismic aspects of reactor safety [1,2] have not specifically included the influence of design errors or of some forms of system degradation during plant lifetime. In this paper we examine the potential for existing cracks to cause pipe failure in a severe earthquake and attempt to estimate the potential influence of design errors on the probability of a seismically-induced severe nuclear reactor accident.

On examining the potential for existing cracks in piping systems to grow to critical size during a severe earthquake, it is concluded that for some materials the extent of crack growth with cyclic plastic strain could be substantial and finally lead to failure. A simple and crude method of relating the initial crack size, the range of strain produced in a component, and the number of cycles to failure, is proposed and discussed.

Subjective extrapolation of past records on design and construction errors, which may or may not be seismic-related, is used to infer the number of seismic-related design errors. Also, subjective values of safety-factor-reduction are assigned for these errors. Using the Newmark correlation between the probability of system failure and safety factor, as employed in WASH-1400 [1], the effect of design errors and system degradation is factored into an estimate of the probability of a core-melt accident due to seismic events. In this estimate, allowance is made for multiple paths to failure, the potential for initially reduced safety margins for some failure paths, and the possibility of closer coupling of failure for redundant but not diverse failure paths.

2. Crack Induced Failure under Seismic Loading

The presence of defects in piping has a non-negligible probability. In this paper a crude and simple analysis of the critical crack size, crack growth, and their relationship to the failure probability of major components under severe seismic loading will be performed. Rigorous treatment is beyond the scope of this study.

2.1 Critical Crack Size

For a situation in which linear elastic fracture methanics (LEFM) applies, the stress intensity factor K, is related to the crack size "ℓ" (crack half length) by

$$K = Q\sigma\sqrt{\pi\ell} \qquad (1)$$

where Q is a flaw shape parameter and is dimensionless and σ is the nominal stress. When the value of K approaches that of the fracture toughness K_c of the material, the risk of brittle fracture arises.

LEFM applies well to materials without any notable ductility. It may also be applicable to ductile materials when the stress state is "plane strain," in which the applied stress is elastic and the crack tip plastic zone size is small compared to the thickness of the structure or equipment. For ductile materials with a relatively thin wall, empirical techniques have been used successfully. Methodology and data can be found in References [3-6]. Depending on the material properties, the wall thickness, etc., the size of the critical flaw may range from as little as an inch to tens of inches.

2.2 Low Cycle Fatigue Crack Growth

Operational heat-up and cool-down during the life of a power plant ($\sim$40 yrs) may be in the order of a few hundred cycles. However, the induced stress is expected to be smaller than the seismic loading and reports have indicated that fatigue growth due to operational cycles is negligible ($\leq$ 10%) [7].

Fatigue crack growth data are generally correlated well with the range of the stress intensity factor by:

$$\frac{d\ell}{dN} = c_o (\Delta K)^m \quad , \qquad (2)$$

where $\frac{d\ell}{dN}$ is the crack growth per cycle; c_o, m are constants, and ΔK is the range of stress intensity corresponding to the cyclic load, and can be evaluated from eq. (1). Alternatively, the number of cycles after which an initial crack size of ℓ_i will grow to the critical crack size ℓ_{cr} can be expressed, after integrating eq. (2) as

$$N_f = \frac{2c_1}{m-2} \Delta\sigma^{-m} \left[\frac{1}{\ell_i^{\frac{m-2}{2}}} - \frac{1}{\ell_{cr}^{\frac{m-2}{2}}} \right] \qquad (3)$$

for m > 2; and,

$$N_f = c_2 \Delta\sigma^{-2} \, \ell n \, \frac{\ell_{cr}}{\ell_i} \qquad (4)$$

for m = 2.

Here $c_1 = \frac{1}{c_o Q^m \pi^{m/2}}$, and $c_2 = \frac{1}{c_o Q^2 \pi}$.

For $c_o = 3.69 \times 10^{-9}$, m = 2.4, [3] and $\Delta\sigma$ changes equal to the yield stress, say σ_y = 64 ksi, crack growth would be less than 10% for N_f = 200 cycles.*

Begley [8] proposed the following crack growth formula:

$$\ell n \, \frac{\ell_f}{\ell_i} = \Delta\sigma^2 \, \frac{10}{E^2} \, Y^2 N \qquad (5)$$

which relates the final crack size (ℓ_f) to the initial crack size (ℓ_i) by the variables: stress range ($\Delta\sigma$), Young's Modulus (E), crack shape factor (Y), and the number of stress cycles (N).

Assuming a strong ground motion effect with

N = 200 cycles

$\Delta\sigma = 6 \times 10^4$ psi

and Y = 2, it is found that

$$\frac{\Delta\ell}{\ell_i} = 3.3\%$$

Thus, the growth of the initial crack is probably insignificant compared to the uncertainty of the critical crack size prediction, if we do not exceed yield.

Parameters other than the stress intensity factor have been studied for crack growth prediction, including crack tip opening displacement [9], absorbed hysteresis energy per cycle [10], etc. Generally speaking, more research and experimental data are needed before practical engineering application of these parameters can be realized. However, particularly for specimens which contain sizeable plastic zones, $\Delta\varepsilon_p$, the applied plastic strain range has been widely used. A few examples are shown in Figure 1. Curve No. 1, the Coffin-Manson relation [11]

$$N_f = \frac{1}{4} \left(\frac{\varepsilon_f}{\Delta\varepsilon_p} \right)^2 \quad , \qquad (6)$$

*Actually, a much smaller number of large peak ground acceleration velocities and displacements is expected in a strong earthquake. However, N_f values of 100 or 200 cycles are used for purposes of illustration.

relates the cycles to failure N_f to the applied plastic strain range $\Delta\varepsilon_p$. In eq. (6), ε_f is the fracture strain. In Figure 1, ε_f=28.6% is approximately equivalent to an area reduction of 75%, a typical value for stainless steel. Curve No. 2 represents Coffin's law [12]:

$$N_f^n \, \Delta\varepsilon_p = c \quad , \tag{7}$$

where n is independent of the material, and to a first order approximation $n \cong 0.5$ and $c \cong 0.65$ for a number of ductile materials. Curve No. 3 was based on Iino's crack growth rate experimental data [13], assuming that the compressive strain is relatively small. Curve No. 4 is a transposition of one of Das' data for stainless steel [14].

In addition to the stress intensity factor and fracture toughness, Cherapanov [15] introduced a new material constant, β, which can be related to the specific dissipation energy of the material, for all materials manifesting plastic properties. The fatigue crack extension rate is obtained as:

$$\frac{d\ell}{dN} = \beta\left[\frac{K_{max}^2 - K_{min}^2}{K_c^2} + \ell n\,\frac{K_c^2 - K_{max}^2}{K_c^2 - K_{min}^2}\right] , \tag{8}$$

in which β and K_c must be determined experimentally. Table I shows some of the data presented in the reference.

Since the critical crack size could be as low as an inch, and the interesting number of cycles could be about 100 cycles, a crack growth rate greater than 10^{-2} in/cycle is significant. It can be seen from Table I that a material with $\beta \geq 10^{-2}$ inches could be damaged during strong earthquake motion.

A crack growth law which takes the stress ratio R (the min. stress/max. stress during a cycle) into account and having a form like [10]

$$\frac{d\ell}{dN} = c(\Delta K)^m / \left[(1-R)K_c - \Delta K\right]^n \tag{9}$$

would also predict a rapid crack growth when ΔK approaches K_c in the case of R = 0.

2.3 Cycles-to-Failure as a Function of Initial Crack Size Under Plastic Deformation

In a nuclear power plant, components of Seismic Category I are expected to have some degree of plastic deformation under SSE influence. In the following, a methodology for estimating the number of cycles-to-failure, as a function of initial crack size, is outlined for severely stressed components in which plastic deformation has occurred.

The critical crack size is determined first from the stress intensity factor and the fracture toughness, keeping the strain amplitude constant for each cycle. However, the stress intensity factor range must be modified, since the strain is beyond the domain of LEFM. Eq. (9) of Reference [16] can be used for this purpose.

$$\Delta K' = \Delta K\left(1 + \frac{\Delta\varepsilon_N^p}{\Delta\varepsilon_N^e}\right)^{\frac{1}{2}} , \tag{10}$$

where ΔK' is the modified stress intensity range, and superscripts p and e denote plastic and elastic, respectively. Assuming an initial crack size ℓ_i slightly smaller than the critical one ℓ_{cr}, eq. (8) is used to determine the number of cycles-to-failure. Since no closed-form solution similar to eq. (3) can be found, a cycle-by-cycle crack growth process must be followed. N_f is obtained whenever ℓ_i becomes greater than ℓ_{cr} at the $(N_f + 1)$th cycle. For smaller ℓ_i's, the entire process is repeated until N_f is greater than the value predicted by the curves in Figure 1.

As an example, an embedded crack growth history has been obtained for stainless steel 310 ($\beta = 3.94 \times 10^{-2}$ inch, $K_c = 130$ ksi$\sqrt{\text{in.}}$) and is shown in Figure 2. Approximations made in the process are:

a. The stress-strain relation remains unchanged from cycle to cycle.

b. $K_{min} = 0$, assuming that the crack has closed during compression with a negligible strain amplitude.

Figure 3 is based on information contained in Figure 2. It shows that components with crack lengths less than ~0.06 inch could be regarded as not having cracks at all, since their sizes have no effect on the cycles-to-failure under plastic strain. It also indicates that at about 0.1 inch size, the crack starts to grow much more rapidly, which agrees with the experimental observation of Gowda, [16].

2.4 Sample Calculation

One inference from the work of Wilson [4] is that the circumferential joint made in field construction is the weakest point with respect to crack detectability. The probability of an undetected crack, with its length ranging from 1 to 8 inches and with a depth up to about 1/10 of the wall thickness, is estimated in Reference [4] to be about 5×10^{-2} per joint for the worst case. Assuming that 10 such joints exist for a typical piping system, for the earthquake which induces stresses that substantially exceed the yield point, the conditional probability of seismic failure of the piping system may be as high as 0.5.

3. On Design Errors and Seismic Risk

3.1 Introduction

In WASH-1400 [1], it was concluded that the earthquake risk was negligibly small compared to other reactor accident risks. The main argument was that the simultaneous failure of two systems is required to produce a core melt accident. Based on Newmark's work [17], the probability of damage to a safety system with a built-in safety factor of 20 is of the order of 1.5×10^{-3}. The joint failure probability of two safety systems, using log normal mean, would be of the order of 10^{-5}. When this damage probability (P_D) was multiplied by the estimated earthquake probability [18], the probability of core melt per reactor year, due to a design basis earthquake, was of the order of 10^{-7} to 10^{-8}.

In estimating the conservatism in the nuclear reactor facility seismic design, Newmark suggested a safety factor of 20 for the SSE, which resulted from the conservatism in the predicted free field ground motion and in the predicted structural responses. Based on the assumption that all the related parameters follow a log-normal distribution, a correlation was established between the probability of failure due to seismic events and the safety factor. Newmark also indicated that a substantial margin-to-failure exists for earthquakes that are significantly larger than SSE, and that the reduced safety factor is inversely proportional to the magnitude of earthquake-induced motion.

However, there are many possible failure paths, each of which could lead to a core-melt accident independently. For the convenience of this parametric risk assessment study, we shall limit ourselves to the following postulated ten failure paths:

1) containment collapse
2) scram failure, with small primary-system leaks
3) reactor building foundation failure (such as soil liquefaction)

4) complete loss of AC power
5) complete loss of DC power
6) component cooling water system failure
7) ultimate heat sink system failure
8) multiple failure in the primary coolant system
9) residual heat removal (RHR) system failure
10) loss-of-coolant accident (LOCA), plus loss of minimum engineered safety features (ESF)

We shall group together paths No. 1 and 2 as group A, paths No. 3, 4, and 5 as group B, and paths No. 6 through 10 as group C; later, we shall assume that the Newmark factor of 20 is applicable for group A, a much lower safety factor, 6, for group B, and an intermediate factor, 10, for group C. For those failure paths which involve redundant subsystems, such as the RHR system, we shall assume that the safety margin is not improved by redundancy, since very strong earthquake vibration has a considerable chance of effecting all the components at the same time.

Reasons for choosing much lower safety factors for group B include the following:

1) A site in which soil liquefaction is unlikely under SSE could undergo liquefaction if the SSE is roughly doubled [19].
2) Concurrent occurrences of failure of the standby emergency power system and the loss of off-site power have been reported in the past [20], and loss of off-site power is a high probability event for a large earthquake.

The group C paths generally involve hundreds of components, and the majority of the components in a loop could be in series in the reliability sense. A typical RHR system loop is composed of piping, pipe fittings, valves, a pump, a heat exchanger, and a cooling water system for the heat exchanger which involves many components. Failure of any of these components would disable the decay-heat removal function.

3.2 Some Historical Perspectives on Design Errors

In this paper, an estimate of the possible number and influence of seismic-related design errors is obtained by examining the historic record of such errors for a specific reactor (called the reference plant) and assuming that, with inclusion of a factor to represent a learning curve, this record would be representative of other reactors. The historic errors were divided into three categories: I. Errors of major importance to reactor safety which might lead to a complete loss of plant function or a large reduction in original design margin. II. Errors of considerable significance which can cause an appreciable reduction in availability of an important system. III. Errors of moderate significance to safety.

For the reference plant, about twenty Category I and twenty Category II errors were noted. It was estimated that Category III errors would be twice as frequent as Category II. Examples of Category I errors are given in Table II.

In order to find a crude basis for estimating the number of errors which might occur in the seismic design, the following approach was used:

1) Assume that in later plants a lesser number of non-seismic design errors exist than in the reference plant. (Take the factor to range from 0.5 to 0.75).
2) Estimate the ratio of seismic engineering design effort to other safety-related engineering design. (Take the factor to range from 0.1 to 0.4).

3) Estimate the relative probability of a design error in the seismic design to that in other areas of design per engineering hour. (Take the factor to range from 0.1 to 0.4).

By combining either the smallest or the largest ends of the above ranges, a range of seismic-related design errors (designated minimal and maximal, respectively) was estimated as shown in Table III. Also shown in Table III is the estimate of reduced safety-margin for each category of error.

3.3 Parametric Assessment of Earthquake Risk

The design errors are assumed to be uniformly distributed among the ten failure paths. The reduced safety factor for each failure path, due to design errors in the three categories, is estimated from the following expression:

$$F_{rj} = \prod_{i=1}^{3} (R_i)^{n_i} \quad , \qquad (11)$$

where R_i's are the ratios of the reduced safety factors to the Newmark factor; i refers to the category, and n designates the number of design errors. F_{rj} gives the overall ratio of the reduced safety margin for the paths of the jth group to Newmark's factor. Newmark's correlation between the safety factor and the probability of failure, reproduced in Figure 4, is used to estimate the probability of damage $(P_j)^*$ to the reactor core, due to the path of the jth group. The conditional probability of damage leading to core melt per reactor (P_D), assuming SSE or a larger earthquake, is then calculated as follows:

$$P_D = 1 - \prod_{j-1}^{3} (1-P_j)^{m_j} \quad , \qquad (12)$$

where m_j is the number of paths in the jth group.

Tables IV and V present the results of the parametric evaluation.

By combining the conditional probabilities (P_D) with earthquake probabilities, an estimated core-melt per reactor year (P_{CM}), due to earthquakes, can be obtained. Table VI shows the estimated probability of earthquake damage resulting in core melt for an average site in the Eastern United States. As in Reference [1], the values for the frequency of earthquake occurrence for different peak accelerations were taken from Reference [18].

Thus, given all the assumptions herein, the total earthquake risk of a serious accident, which is obtained by summing all accelerations $\geq$ 0.2g, would range from 8×10^{-5} to 8×10^{-4} per reactor year.

3.4 Discussion

The analysis suggests that earthquakes could be significant contributors to serious reactor accidents. With a uniform safety factor of 20 for each possible failure path which could independently lead to core melt, together with minimal or no seismic design errors, the probability of core melt per reactor year due to an earthquake is about the same as that estimated in WASH-1400 when all accident initiators are included. If significant (maximal) design errors prevail and, if at the same time, many failure paths have a reduced safety margin (less than the Newmark factor), the probability of a core melt per reactor-year would be one order of magnitude higher, and the seismic event would be the predominant cause. For the convenience of comparison and of discussion, Table VII reproduces information contained in the two tables on page 67 of WASH-1400, together with results of our analysis.

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*P_j is the probability of failure corresponding to a safety factor equal to F_{rj} multiplied by the original safety factor for the failure path in the jth group.

From a review of WASH-1400, it appears that the reported total probability of core melt per reactor year did not include the contributions for 0.3g, 0.4g, 0.6g, 0.7g, 0.8g and 0.9g. If these probabilities were included, we estimate that the total probability of WASH-1400 would become 2×10^{-6}, instead of 5×10^{-7}.

The large difference in total probability between our results and those of WASH-1400 can be attributed to the following considerations:

1) We assume that there are more than ten independent failure paths that could lead to a core-melt accident.
2) We assume that an earthquake would affect all the components simultaneously, and hence would have a stronger potential for common-mode failure.
3) We allow for seismic design and construction errors.
4) For some systems or failure paths, we assume that the built-in safety factor could be lower than the Newmark factor.

It must be emphasized that the quantitative results reported herein depend heavily on the assumptions. For example, the use of earthquake probabilities from [18] may lead to higher than expected accelerations at many sites. Similarly, the assumptions concerning reduction in safety margin for each category design error are arbitrary, as are the assumptions concerning original safety margin for some failure paths. Hence, the failure probabilities estimated herein could readily be calculated to be at least an order of magnitude lower.

The purpose of the study is primarily to illustrate a possible method for the incorporation of design errors in reliability analysis and safety assessment, and to look in some detail at the results reported for seismic risk in WASH-1400. Further details are to be found in References [21] and [22].

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Table I. Crack Growth Rate (in/cycle) for Some Steels

Material	K_c ($ksi\sqrt{in}$)	β (in)	$\frac{K_{max}}{K_c}$			
			0.4	0.6	0.8	1.0
Martensitic aging Steel 250	200	7.87×10^{-3}	1.4×10^{-4}	7.9×10^{-4}	3.7×10^{-3}	1.5×10^{-2}
Stainless Steel 310	130	3.94×10^{-2}	7.9×10^{-4}	3.9×10^{-3}	2.0×10^{-2}	7.9×10^{-2}
Stainless Steel 301	197	1.57×10^{-1}	2.4×10^{-3}	1.4×10^{-2}	6.7×10^{-2}	3.5×10^{-1}
Steel B-95	56	2.31×10^{-2}	3.5×10^{-4}	2.0×10^{-3}	9.8×10^{-3}	3.9×10^{-2}

Table II. Some Examples of Design and Construction Errors for Reference Plant

1. The local creek and the local drainage patterns would allow plant flooding in the event of a probable maximum precipitation.
2. Use of mercury switches in the CO_2 systems could deprive the diesel generators of all air cooling when actuated by a seismic event which destroyed the off-site power.
3. Possibility of coupled failure of "redundant" piping, should failure of one pipe cause erosion of the local filled soil supporting contiguous essential water lines, and hence failure of redundant system.
4. Disregard of distant (but not local) dam failures as a source for flood damage.
5. Tornado vulnerability, single failure, and seismic vulnerability of the original ventilation system for the control room and for other critical rooms.
6. Failure to properly specify the dynamic loads in the torus in the event of a LOCA involving large ruptures.
7. The potential flooding of the safety systems due to the failures of non-safety grade components.
8. Failure of HPCI or RCIC and of the primary loop pipes in the secondary containment with certain coupled valve failures could result in an intolerable environment for the needed equipment.
9. Inadequate separation for fire protection.
10. Potential loss of the fuel pool water due to fuel cask dropping into the pool.

Table III. Estimated Seismic Related Design Errors in a Nuclear Power Plant

Category	Minimal Design Errors	Maximal Design Errors	Reduced Safety Margin
I	0.1	5	1/3
II	0.1	5	2/3
III	0.2	10	9/10

Table IV. The Conditional Probability of Damage to the Reactor Core (P_D), Assuming that Each of the Ten Failure Paths Has a Built-in Safety Factor of 20 for Reactors Designed for SSE with 0.2g

Ground Acc.	No Design Errors	Minimal Design Errors	Maximal Design Errors
0.2g	0.015	0.017	0.16
0.5g	0.18	0.20	0.72
1.0g	0.58	0.61	0.97

Table V. The Conditional Probability of Damage to the Reactor Core (P_D), Assuming Three Failure Path Groups With Built-in Safety Factors of 20, 6, and 10, Respectively.

Ground Acc.	No Design Errors	Minimal Design Errors	Maximal Design Errors
0.2g	0.16	0.17	0.60
0.5g	0.67	0.69	0.98
1.0g	0.96	0.98	1.0

Table VI. The Probability of Core Melt per reactor Year [P_{CM}] Due to Earthquake, for an Average Site in the Eastern United States Designed for SSE of 0.2g

Ground Acc.	Freq. of Ground Acc. Per Yr	Minimal Design Errors: Uniform Safety Factor of 20	Minimal Design Errors: Three Safety Factor Groups	Maximal Design Errors: Uniform Safety Factor of 20	Maximal Design Errors: Three Safety Factor Groups
0.2g	$7x10^{-4}$	$1x10^{-5}$	$1x10^{-4}$	$1x10^{-4}$	$4x10^{-4}$
0.5g	$5x10^{-5}$	$9x10^{-6}$	$3x10^{-5}$	$3x10^{-5}$	$5x10^{-5}$
1.0g	$1x10^{-5}$	$6x10^{-6}$	$1x10^{-5}$	$1x10^{-5}$	$1x10^{-5}$
	Total*	$8x10^{-5}$		$3x10^{-4}$	$8x10^{-4}$

*Including P_{CM}'s for 0.3, 0.4, 0.6, 0.7, 0.8, 0.9g.

Table VII. Comparison of Probabilities of Core Melt Due to Earthquake for an Average Site in the Eastern United States with 0.2g SSE

Ground Acc.	Newmark (safety) factor	Earthquake Prob. per year	Reference Analysis: Prob. of damage single system	Reference Analysis: Prob. of damage two systems	Reference Analysis: Core Melt per reactor year (P_{CM})	Our Analysis: Prob. of reactor damage due to one path	Our Analysis: Prob. of reactor damage due to ten failure paths	Our Analysis: Core melt per reactor-year, minimal or no design errors	Our Analysis: Core melt per reactor-year, maximal design errors
0.2g	20	$7x10^{-4}$	0.001	$3x10^{-5}$	$2x10^{-8}$	0.0015 to 0.017*	0.016 to 0.16*	$1x10^{-5}$	$1x10^{-4}$
0.5g	8	$5x10^{-5}$	0.02	$3x10^{-3}$	$2x10^{-7}$	0.02 to 0.12	0.19 to 0.72	$9x10^{-6}$	$3x10^{-5}$
1.0g	4	$1x10^{-5}$	0.1	$3x10^{-2}$	$3x10^{-7}$	0.083 to 0.30	0.58 to 0.97	$6x10^{-6}$	$1x10^{-5}$
				Total	$5x10^{-7}$		**Total	$8x10^{-5}$	$3x10^{-4}$

*The range of the numbers corresponds to "no design errors to maximal design errors."

**Including P_{CM}'s for 0.3, 0.4, 0.6, 0.7, 0.8, 0.9g, and uniform safety factor.

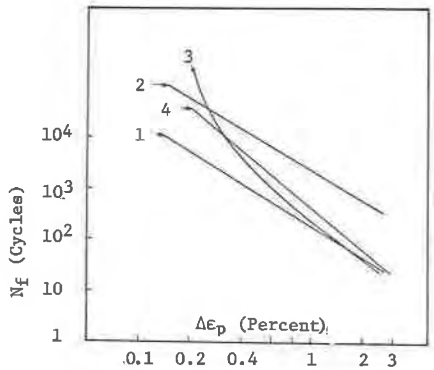

Figure 1. Number of Cycles to Failure (N_f) for Specimens With Sizeable Plastic Strain

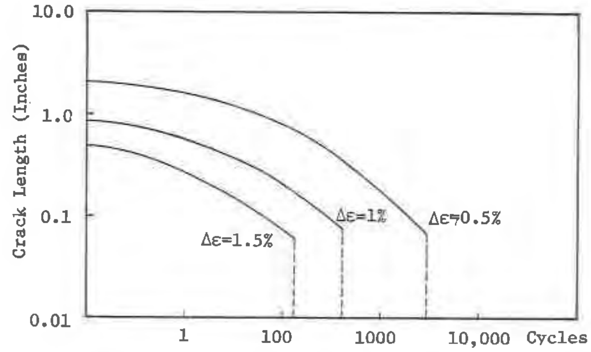

Figure 2. Number of Cycles to Failure as a Function of Initial Crack Length (for Stainless Steel 310)

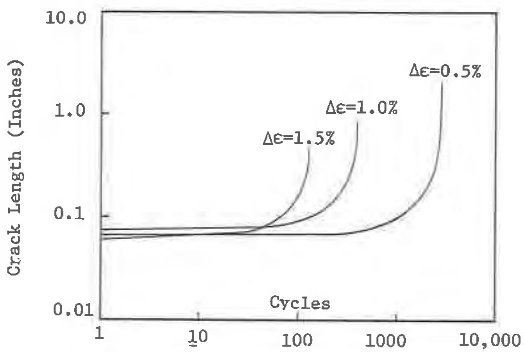

Figure 3. Crack Growth History (for Stainless Steel 310)

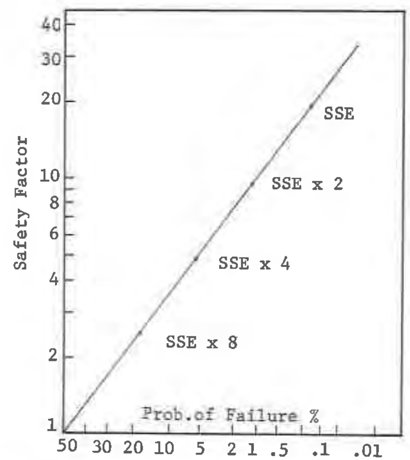

Figure 4. Newmark's Correlation Between Safety Factor and Probability of Failure for Nuclear Facilities