Enclosure 6

PROPOSED REVISION 2 OF REGULATORY GUIDE 1.99

VALUE-IMPACT ANALYSIS

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CONTENTS

1.0 INTRODUCTION 1.1

2.0 PROPOSED ACTION AND POTENTIAL ALTERNATIVES 2.1

3.0 AFFECTED DECISION FACTORS 3.1

4.0 VALUE-IMPACT ASSESSMENT SUMMARY 4.1

5.0 UNQUANTIFIED RESIDUAL ASSESSMENT 5.1

6.0 DEVELOPMENT OF QUANTIFICATIONS 6.1

PUBLIC HEALTH 6.1

OCCUPATIONAL EXPOSURE (ACCIDENTAL) 6.7

PUBLIC PROPERTY 6.9

ONSITE PROPERTY 6.9

INDUSTRY IMPLEMENTATION 6.10

INDUSTRY OPERATION COST 6.11

NRC IMPLEMENTATION COSTS 6.12

7.0 CONCLUSIONS 7.1

REFERENCES R.1

FIGURES

1. Heat-up P-T Limits Based on Rev. 1 Guidelines 6.2

2. Heat-up P-T Limits Based on Rev. 2 Guidelines 6.2

3. Cool-down P-T Limits Based on Rev. 1 Guidelines 6.3

4. Cool-down P-T Limits Based on Rev. 2 Guidelines 6.3

TABLES

1. Summary of Changes in P-T Limits Resulting from a Change from Rev. 1 to Rev. 2 of Regulatory Guide 1.99 . . . . 1.1

2. V/I Analysis Decision Factors. 3.1

3. Effects on Change in Failure Probability per Transient from Revision 1 to Revision 2. 6.4

4. Changes to Release Category Frequency (per event) 6.5

5. Dose Conversion Factors (man-rem/release). 6.6

6. Avoided Calculated Public Dose (man-rem/startup-shutdown). 6.6

7. Summary of Avoided Public Health Risk 6.7

8. Summary of Avoided Occupational Exposure 6.8

9. Summary of Avoided Public (Offsite) Property Damage. 6.9

10. Summary of Avoided Onsite Property Damage. 6.10

PROPOSED REVISION 2 OF REGULATORY GUIDE 1.99 VALUE-IMPACT ANALYSIS

1.0 INTRODUCTION

This report presents a value-impact assessment of implementing Revision 2 of Regulatory Guide 1.99, "Radiation Damage to Reactor Vessel Materials." In practice, neutron radiation damage to the reactor vessel is compensated for by shifting the Pressure-Temperature (P-T) limits up the temperature scale every few years by an amount corresponding to the shift in the Charpy test transition temperature produced by the accumulated neutron fluence. The NRC regulates this process on the basis of Appendices G and H, 10 CFR Part 50. Paragraph V.A of Appendix G requires: "The effects of neutron radiation ... are to be predicted from the results of pertinent radiation effects studies ...". Regulatory Guide 1.99 provides a means for determining results in the form of calculational procedures that are acceptable to NRC, and it describes acceptable procedures for using plant-specific surveillance data when they become available.

The objectives of Regulatory Guide 1.99 are best described in terms of the transition temperature approach to fracture prevention. In this approach, the margin of safety against fracture is given as the difference between the operating temperature of the vessel and the temperature at which brittle fracture could occur, and the measure of radiation damage is a transition temperature shift. Thus, the ability to calculate the shift has a direct impact on the margin of safety against fracture. The objective of Revision 2 of Regulatory Guide 1.99 is to make the calculational procedures consistent with the present knowledge of radiation damage. Table 1 presents a summary of the changes that would result from the changes in Pressure-Temperature (P-T) limits that are proposed in Revision 2 of Regulatory Guide 1.99. The values in this table were obtained from the NRC staff (P. N. Randall, 1984).

TABLE 1. Summary of Changes in P-T Limits Resulting from a Change from Rev. 1 to Rev. 2 of Regulatory Guide 1.99

Effect of Change From Rev. 1 to Rev. 2	Number of Operating Reactors			Number of Plants Undergoing Licensing		
	PWR	BWR	Total	PWR	BWR	Total
Ratchet 50°F to 100°F	4	4	8			
Ratchet 20°F to 50°F	16	17	33	34	11	45
No Change (+20°F)	23	7	30	3	10	13
Benefit 20°F to 50°F	7	1	8			
Benefit 50°F to 100°F	1	0	1			
Benefit 100°F to 150°F	1	0	1			
TOTALS	52	29	81	37	21	58

(a) As used herein, ratcheting means an increase in P-T limits up the temperature scale.

(b) As used herein, benefit means a decrease in P-T limits down the temperature scale.

The following sections present the results of a value-impact analysis done on the effects of implementing Revision 2 of Regulatory Guide 1.99. The scope of this analysis is limited to analyzing the changes in failure probability of the reactor vessel during normal startup-shutdown procedures from Revision 1 to Revision 2. The effects on the operational translents and the pressurized thermal shock issue (PTS) have not been addressed in this study.

The value-impact assessment uses the methods developed in The Handbook for Value-Impact Assessment (Heaberlin et. al. 1983), the data developed for Safety Issue Prioritization (Andrews et. al. 1983) and the results of calculations done using a NRC Vessel Integrity Simulation Analysis (VISA) code (D.L. Stevens et. al. 1983).

2.0 PROPOSED ACTION AND POTENTIAL ALTERNATIVES

It has been proposed that Revision 2 of Regulatory Guide 1.99 be implemented by operating plants as well as those undergoing licensing review. This proposal is based on considerations of public risk, occupational dose and cost impacts.

Improvements in knowledge about material properties warrant the use of the guidelines given in Revision 2 of Regulatory Guide 1.99 in determining the Nil Ductility Transition Reference Temperture (RT_{NDT}). Use of Revision 2 guidelines will effect the RT_{NDT} of the majority of the operating nuclear reactors and those undergoing licensing. The change in RT_{NDT} will subsequently change the P-T limits set in the Technical Specifications of the nuclear power plants. This change and the number of plants affected were presented in Table 1.

The alternatives to implementation of Revision 2 are to leave Revision 1 in place or to eliminate the Guide all together. The solution is not feasible since the NRC staff reviews more than 10 Pressure-Temperature limits per year and needs a published basis for its review. Currently, there is no ASME code equivalent to Regulatory Guide 1.99. ASTM Standard Guide E-900-83 contains an equation relating the Charpy shift to copper content and fluence, but it is out of date. Also, the ASTM Standard does not contain guidance on the use of plant-specific surveillance data. Revison 1 of Regulatory Guide 1.99 was used extensively from the time it was published in 1977 until late 1982 when the review of radiation damage resulting from the pressurized thermal shock (PTS) issue revealed the need for change. Also, the surveillance data base has increased to the point where Revision 2 is based almost entirely on surveillance data, whereas Revision 1 was based primarily on test reactor data.

3.0 AFFECTED DECISION FACTORS

Parameters considered in the value/impact analysis and those affected in this study are shown in Table 2.

TABLE 2. V/I Analysis Decision Factors

Decision Factors	Causes Quantified Change	Causes Unquantified[a] Change	No Change
Public Health	X		
Occupational Exposure (Accidental)	X		
Occupational Exposure (Routine)			X
Public Property	X		
Onsite Property	X		
Regulatory Efficiency			X
Improvements in Knowledge			X
Industry Implementation C	X		
Industry Operation Cost	X		
NRC Development Cost			X
NRC Implementation Cost	X		
NRC Operation Cost			X

(a) Unquantified means not readily estimated in dollars.

4.0 VALUE-IMPACT ASSESSMENT SUMMARY

Decision Factors	Best Estimate	Lower Estimate	Upper Estimate
VALUES[a] (man-rem)			
Public Health	1.1E+04	0	2.2E+04
Occupational Exposure (Accidental)	6.7E+01	0	2.2E+02
Occupational Exposure (Routine)	N/A		
Regulatory Efficiency	N/A		
Improvements in Knowledge	N/A		
Total Quantified Value		0	
IMPACTS[b,c] ($)	1.1E+04		2.2E+04
Industry Implementation Costs	-2.0+5	-1.2E+5	-4.0E+5
Industry Operating Cost	-1.01E+08	-5.01E+07	-1.7E+08
NRC Development Cost	N/A		
NRC Implementation Cost	-1.3E+05	-8.6E+04	-3.4E+05
NRC Operation Cost	0	0	0
Public Property	3.1E+06	0	3.2E+07
Onsite Property	3.5E+07	0	6.7E+07
Total Quantified Impact	-6.3E+07	-5.0E+07	-7.2E+07

(a) A decision term is a value if it supports NRC goals. Principal among these goals is the regulation of safety.

(b) Impacts are defined as the costs incurred as a result of the proposed action. ~~Negative~~ Positive impacts indicate cost savings (avoided cost).

(c) Assuming 5% discount rate.

N/A = Not Affected

5.0 UNQUANTIFIED RESIDUAL ASSESSMENT

There are no unquantified decision factors in the assessment of this action.

6.0 DEVELOPMENT OF QUANTIFICATIONS

Development of quantifications included the following factors: public health, occupational exposure (accidental), public property, onsite property, industry implementation, industry operation cost and NRC implementation cost.

PUBLIC HEALTH

A risk analysis was performed to assess the effects of implementing Revision 2 of Regulatory Guide 1.99. It can be seen from Table 1 that implementation of this revision will change the P-T limits of a majority of the operating nuclear reactors and those plants undergoing licensing. The effects of this revision on those plants that are ratcheted by 0°F to 100°F were analyzed.

The VISA code was used to develop estimates of the failure probabilities of reactor vessels. The P-T limits of a representative case were developed based on Revision 1 and Revision 2 values of RT_{NDT}, following the procedures prescribed in Appendix G to Section III of ASME code and Appendix G of 10 CFR 50. Figures 1 through 4 show the heatup-cooldown P-T limit curves developed for our assumed case and based on Revision 1 and Revision 2 guidelines.

A conservative estimate of the failure probability was determined by choosing a plant that would be ratcheted by 100° F if Revision 2 was to be implemented. Following are the characteristics of the assumed representative case:

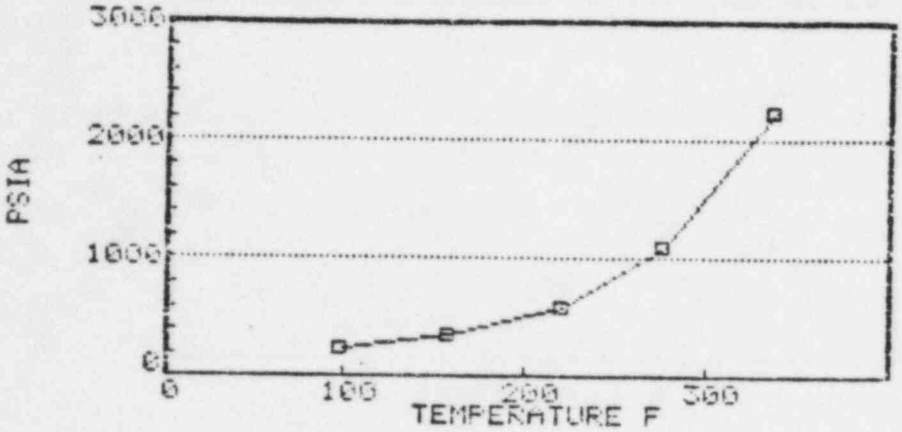
CU% = 0.20 NI% = 1.00 F = 1E+19 (F = fluence)

An estimate of 4.9E-07 per transient for the change in failure probability of the reactor vessel from Revision 1 to Revision 2 was obtained. This was assumed to be an upper estimate for this analysis.

Most of the ratcheted plants are those whose P-T limits are affected by 20°F to 50°F. An intermediate estimate was determined by considering a plant that would be ratcheted by about 50°F. The change in failure probability of the reactor vessel from Revision 1 to Revision 2 is determined to be zero. However, for our analysis, a best estimate of half the upper estimate (2.5E-07 per transient) was assumed.

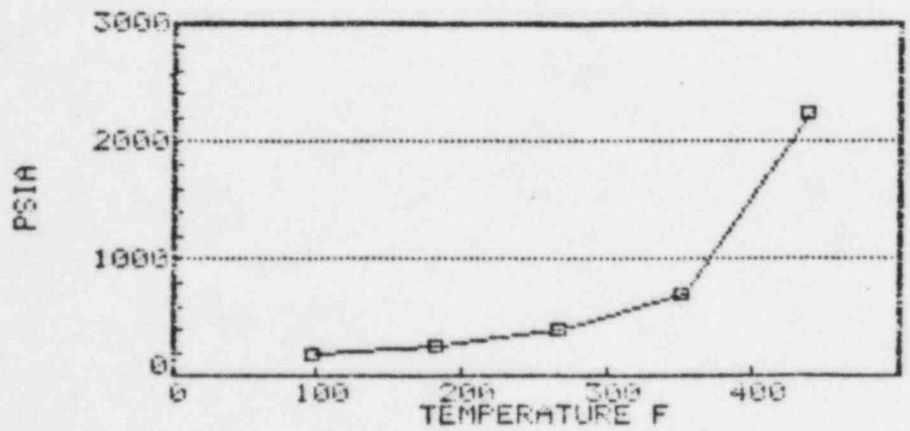
The lower estimate of the change in failure probability of the vessel was assumed to be zero, since there are about 30 plants that will see no change or improvement in their P-T limits due to implementation of Revision 2.

Modifications were also made to VISA in order to obtain estimates of the reactor vessel failure probability. The main modification was to revise the flaw size distribution given in VISA. A listing of the flaw size distribution follows:



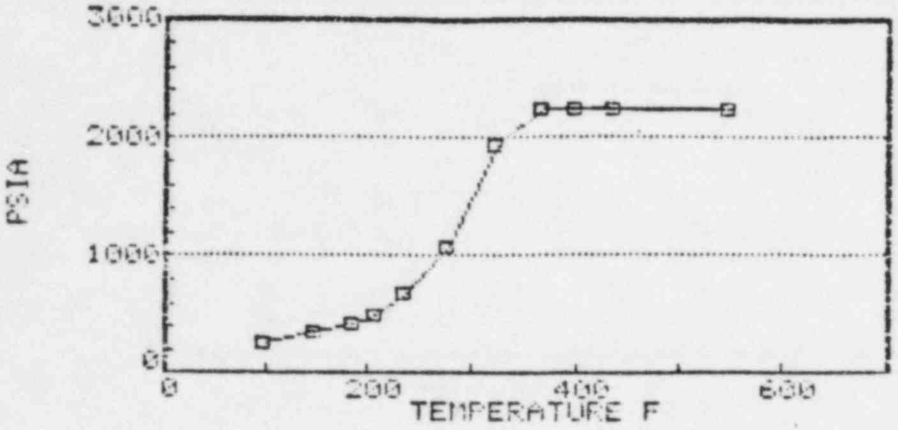
HEAT-UP LIMITS, REV. 1

FIGURE 1. Heat-up P-T Limits Based on Rev. 1 Guidelines



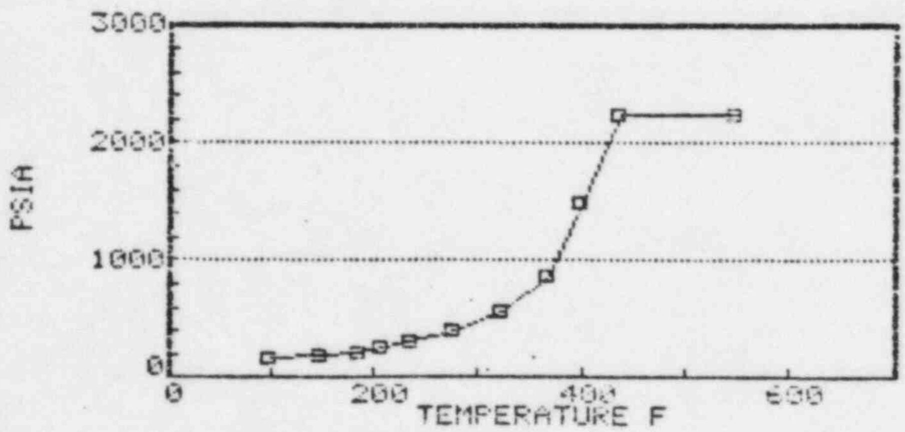
HEAT-UP LIMITS, REV. 2

FIGURE 2. Heat-up P-T Limits Based on Rev. 2 Guidelines



COOL-DOWN LIMITS, REV. 1

FIGURE 3. Cool-down P-T Limits Based on Rev. 1 Guidelines



COOL-DOWN LIMITS, REV. 2

FIGURE 4. Cool-down P-T Limits Based on Rev. 2 Guidelines

Flaw Size (Inches)	Probability
0.000	0.91767661
0.125	0.05507015
0.250	0.02256957
0.500	0.00422063
1.000	0.00036718
1.500	0.00007085
2.000	0.00001667
2.500	0.00000500
3.000	0.00000250
3.500	0.00000083

This distribution was used during the preliminary stages of our analysis. However, due to the small probabilities given in the distribution no reactor vessel failure was simulated by VISA. Therefore, to get any kind of an estimate, we assumed that the probability of a 1/4 T (Thickness) flaw is 1. For our purposes the thickness of the reactor vessel was assumed to be 8 inches. The failure probability results were then reduced by a factor of 3500 based on the frequency of this flaw size. This value was obtained by the ratio of the results given by two VISA runs: the first with the adjusted flaw size distribution and the second with the original flaw size distribution. These values were then multiplied by 6 to account for 6 welds in the reactor vessel belt line.

Table 3 shows the effects of change in failure probability of vessels from Revision 1 to Revision 2. The values in Table 3 should be interpreted as the "reduced" failure probability or the failure probability that will be "avoided" by implementing Revision 2 of Regulatory Guide 1.99. These estimates are very conservative due to the conservative nature of the analysis and the assumptions.

TABLE 3. Effects of Change in Failure Probability per Transient from Revision 1 to Revision 2

Best Estimate	Upper Bound	Lower Bound
2.5E-07	4.9E-07	0

A conservative estimate of risk was made by assuming that a propagating crack would result in breach of the reactor pressure vessel and subsequent core melt. This conservative assumption was used to perform the Pressurized Thermal Shock Value-Impact analysis (Andrews et. al. 1983) and it is believed that core-melt following a PTS event is more probable than during normal heatup-cooldown procedures, therefore the subsequent risk analysis done for this report is believed conservative. The containment failure modes, likelihoods, and release categories are assumed to be the same as for sequences S_1D (for PWRs) and SI (for BWRs) Appendix A of the Guidelines (Andrews 1983).

Table 4 shows the changes to release category frequency based on the previously specified release categories.

TABLE 4. Changes to Release Category Frequency (per event)[a]

Reactor Type	Estimate	Release Category 1	2	3	4	5	6	7	Total Core Melt[(b)]
PWR	Best Estimate	2.5E-09		5.0E-08		1.8E-09		2.0E-07	2.5E-07
	Upper Bound	4.9E-09	0	9.8E-08	0	3.6E-09	0	3.9E-07	5.0E-07
	Lower Bound	0	0	0	0	0	0	0	0
BWR	Best Estimate	2.5E-09	2.5E-07	0	0	—	—	—	2.5E-07
	Upper Bound	4.9E-09	4.9E-07	0	0	N/A[(c)]	—	—	4.98E-07
	Lower Bound	0	0	0	0	—	—	—	0

(a) Event refers to startup-shutdown.

(b) The values in table are per (startup-shutdown). For our purposes 6 (startup-shutdown)/reactor is assumed.

(c) N/A = not affected

Application of the dose conversion factors in Table 5 to the changes in release category frequencies given in Table 4 results in the avoided public dose shown in Table 6. The uncertainty is conservatively propagated by employing the extremes (e.g., high estimate dose conversion times upper bound release category frequency change).

TABLE 5. Dose Conversion Factors (man-rem/release)[a]

Release Category	Whole Body Dose Consequence (man-rem)
PWR 1	5.4E+06
PWR 2	4.8E+06
PWR 3	5.4E+06
PWR 4	2.7E+06
PWR 5	1.0E+06
PWR 6	1.5E+05
PWR 7	2.3E+05
BWR 1	5.4E+06
BWR 2	7.1E+06
BWR 3	5.1E+06
BWR 4	6.1E+05

(a) From CRAC, with guidelines and quantities of radioactive isotopes used in WASH-1400. Estimates are based on the meteorology of a typical Midwest site (Byron-Braidwood) with a uniform population density of 340 people/square mile, no evacuation and 50-mile radius model.

TABLE 6. Avoided Calculated Public Dose (man-rem/startup-shutdown)

Reactor	Best Estimate	Upper Bound	Lower Bound
PWR	3.3E-01	6.5E-01	0
BWR	1.8E+00	3.5E+00	0

To estimate the total public risk averted, the per-startup-shutdown estimates must be multiplied by the number of affected facilities, average number of startup-shutdown (events) per year and by their average remaining lifetimes. The number of affected facilities was assumed to be 86, given in Table 1. The average number of startup-shutdown for a plant is assumed to be 6 per year. The average remaining lifetime for the plants is assumed to be 25 years. Use of these values yields the total avoided public dose estimates shown in Table 7.

TABLE 7. Summary of Avoided Public Health Risk[a,b]

Total Avoided Dose (Person-rem)		
Best Estimate	Upper Estimate	Lower Estimate
1.1E+04	2.2E+04	0

(a) For all the affected PWRs and BWRs.
(b) These estimates include both the operating plants and those undergoing licensing. Refer to Table 1 for exact number of affected plants.

OCCUPATIONAL EXPOSURE (ACCIDENTAL)

The avoided occupational exposure from accidents can be estimated as the product of the change in total core-melt frequency and the occupational exposure likely to occur in the event of a major accident. The estimated change in core-melt probabilities are presented in Table 4. The occupational exposure in the event of a major accident has two components. The first is the "immediate" exposure to the personnel on site during the span of the event and its short-term control. The second is the longer-term exposure associated with the cleanup and recovery from the accident.

The final data required are the number of affected facilities and their remaining lifetimes. The number of plants affected is given in Table 1 and their average remaining lifetimes is assumed to be 25 years. The total avoided occupational exposure is then calculated as follows:

$$D_{TOA} = NTD_{OA} = \Delta F \; (D_{IO} + D_{LTO})$$

where

D_{TOA} = total avoided occupational dose

N = number of affected facilities

T = average remaining lifetime

D_{OA} = avoided occupational dose per reactor-year

ΔF = change in core-melt probability

D_{IO} = "immediate" occupational dose

D_{LTO} = long-term occupational dose

Table 8 shows the values taken as best estimates and bounds for these parameters. Uncertainties are conservatively propagated by use of extremes (e.g., high estimate D_{IO} + high estimate D_{LTO}).

TABLE 8. Summary of Avoided Occupational Exposure[a]

	Change In Core-Melt Probability (events/reactor-yr) PWR	Change In Core-Melt Probability (events/reactor-yr) BWR	Immediate[b] Occupational Dose (man-rem/event)	Long-Term[c] Occupational Dose (man-rem/event)	Total Avoided Occupational Exposure (man-rem)
Best Estimate	2.5E-07	2.5E-07	1.0E+03	2.0E+04	6.7E+01
Upper Estimate	5.0E-07	4.50E-07	4.2E+03	3.0E+04	2.2E+02
Lower Estimate	0	0	0	1.0E+04	0

(a) For both the operating plants and those undergoing licensing. For the exact number of plants affected refer to Table 1.

(b) Based on initial (4 month) occupational exposure following the accident at TMI.

(c) Based on cleanup and decommissioning estimates. NUREG/CR-2601 (Murphy, 1982).

PUBLIC PROPERTY

The effect of the proposed action upon reducing risk to public (i.e., offsite) property is calculated by multiplying the change in accident probability by a generic offsite property damage estimate. This estimate was derived from the mean value of results of CRAC2 calculations, assuming an SST1 release (i.e., major accident) for 154 reactors (Strip 1982). The damage estimate is converted to present value by discounting at 10%.

The following discounting formula was used:

$$D = V \quad \frac{e^{-0.10t_l} - e^{-.10t_f}}{0.10}$$

where

D = discount value
V = damage estimate
t_l = years before reactor begins operating 0 for operating reactor
t_f = years remaining until end of life.

The average number of years of remaining life is 25. Therefore, the discount D/V = 9. This must be multiplied by the number of affected facilities to yield the total effect of the action. Table 9 summarizes these results. The high and the low estimates are values for Indian Point No. 2 and Palo Verde No. 3 calculated from Strip (1982).

The cost estimates have also been calculated using a 5% discount rate. This was done as a sensitivity analysis to determine the impact of discount rate on the overall value-impact ratio.

TABLE 9. Summary of Avoided Public (Offsite) Property Damage

	Offsite Property Damage ($/event)	Value of Avoided Offsite Property Damage ($)	
		10%	5%
Best Estimate	1.7E+09	2.0E+06	3.1E+06
Upper Bound	9.2E+09	2.1E+07	3.2E+07
Lower Bound	8.3E+08	0	0

ONSITE PROPERTY

The effect of the proposed action on reducing the risk to onsite property is estimated by multiplying the change in accident probability by a discounted onsite property cost. This discounted property cost was developed from the generic onsite property cost taken from Andrews et. al. (1983). It includes an estimate for replacement power.

This value is discounted at 10% using the following formula:

$$D = V \left[\left(\frac{e^{-0.10 t_i}}{0.01M} \right) \left(1-e^{-0.10(t_f - t_i)} \right) \left(1-e^{-0.10M} \right) \right]$$

where

D = discounted value

V = damage estimate

t_i = years before reactor begins operation, 0 for operating reactors

t_f = years remaining until end of life

M = period of time over which damage cost is paid out (recovery period in years)

Assuming that the remaining reactor life is 25 years and that the recovery period is 10 years, the discount D/V = 5.8.

To obtain the total effect of the actions, the per-reactor results are multiplied by the number of affected facilities (86). The results are summarized in Table 10. The uncertainty bounds given in the table reflect a +50% spread in the generic property cost coupled with the bounds on core-melt probability. This was estimated to be indicative of the uncertainty level.

TABLE 10. Summary of Avoided Onsite Property Damage

	Onsite Property Damage ($/event)	Value of Avoided Onsite Property Damage ($)	
		10%	5%
Best Estimate	1.65E+09	1.2E+06	3.5E+07
Upper Bound	2.5E+09	3.6E+06	6.7E+07
Lower Bound	8.2E+08	0	0

INDUSTRY IMPLEMENTATION

The primary cost element associated with implementing Revision 2 of Regulatory Guide 1.99 consists of changing the P-T limits given in a plant's technical specification. The time required to make this change is estimated to be one man-week per plant. This cost is taken to be $2270/week (Andrews et. al. 1983). The number of affected plants is assumed to be 86 (from Table 1). This includes both the operating plants and those undergoing licensing. Therefore, the total cost of industry implementation is estimated to be:

(86 plants) ($2270/plant) = $2.0E+05

This value is taken to be our best estimate. For our purposes an upper estimate of 2 weeks and a lower estimate of 3 days have been assumed for the time required to make the changes in the Technical Specifications.

INDUSTRY OPERATION COST

The main effect of Revision 2 of Regulatory Guide 1.99 is to shift the P-T limits up the temperature scale. This will require the plants to warm up the vessel to a higher temperature before reaching to their normal operating conditions. It is estimated that the maximum "lost time" for the ratcheted plants would be 2 hours assuming 50°F/hr heat up rate. This estimate is derived by studying the revised P-T limits and discussions with selected industry representatives. This estimate is converted to a monetary value as follows:

(2 hours lost time)(1 day/24 hours)($300,000/day) = $2.5E+04

The $300,000/day estimate represents the cost of one day delay in startup (full power operation) to the industry. Multiplying $2.5E+04/plant by the number of affected plants, number of assumed startup-shutdowns/yr, and the remaining lifetime of the plants (assumed to be 25 years) yields the total industry operation cost. Therefore, the total industry operation cost is:

(8 plants)($2.5E+4/plant)(6 startup-shutdown/yr) (25 yr) = $3.0E+07 (1)

This value represents the industry operation costs for those plants that are ratcheted by 100° F. There are about 78 plants that are operating and undergoing licensing and that are ratcheted by up to 50° F. Their "lost time" will therefore be half of the ones that are ratcheted by 100° F. The industry operating cost for these plants is:

(78 plants)($1.25E+4/plant)(6 startup-shutdown/yr)(25 yr) = $1.5E+08 (2)

Adding these values (i.e., $3.0E+07 and $1.5E+08) gives a total industry cost of $1.8E+08.

Assuming 5% discount over the next 25 years, the present value of total industry operation cost becomes $1.01E+08. At 10% discount this estimate is $6.54E+07.

Assuming a 100°F/hr heatup rate, can reduce this estimate by 50 percent. An estimate of 30°F/hr heatup rate is chosen for the upper bound cost calculations.

Some of the older plants (i.e., those that are ratcheted significantly), there is concern that their P-T limits are already close to the saturation curve and implementation of Revision 2 will cause their P-T limits to get closer

to the saturation curve and thereby further limit their startup procedures. Operations to cover heatup-cooldown rates may be needed to avoid this problem. Additional costs would be incurred if this approach is taken.

NRC IMPLEMENTATION COST

The impact of proposed changes with respect to staff review time will be minimal. It will be limited to reviewing the revised P-T limits given in the Technical Specifications. It is estimated that less than 1 person-week of staff review time would be required (estimated cost = \$1500/plant). The cost per-plant might range from \$1000/plant to \$4000/plant. For 86 plants, this yields a total NRC impact of \$1.3E+05 with bounds ranging from \$8.6E+04 to \$3.4E+05.

7.0 CONCLUSIONS

The summary results for the value-impact assessment are shown below.

Summary of Value-Impact Assessment

Value (man-rem)			Impact(a) ($)					
Best Estimate	Upper Estimate	Lower Estimate	Best Estimate		Upper Estimate		Lower Estimate	
			5%	10%	5%	10	5%	10%
1.1E+04	2.2E+04	0	-6.3E+07	-6.2E+07	-7.2E+07	-7.7E+07	-5.0E+07	-3.2E+07

(a) Assuming a 5% and 10% discount rate.

The best estimates for cost and dose reductions indicate that the cost of one man-rem avoided is in the $3500 - $5600 range. This high cost estimate, therefore, prevents us from recommending the implementation of this revision. This conclusion is not sensitive to the assumed discount rate.

There is no doubt that the Revision 2 guidelines are more up-to-date and more accurate than Revision 1. However, there are several drawbacks associated with the proposed revision. The main one is the fact that the implementation of this revision will slow down the startup time of the majority of the operating plants and those undergoing licensing. The other drawback is that the P-T limits of some of the older plants is already close to the saturation curve. Implementation of this revision will further limit their operational procedures.

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