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Value/Impact Assessment of Jet Impingement Loads and Pipe-to-Pipe Impact Damage

Revised Methods and Criteria

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ABSTRACT

To account for effects that might result from a loss-of-coolant accidents (LOCA), nuclear power plant designers have been required to analyze the effects of double-ended guillotine breaks (DEGB) in high-energy piping. The U.S. Nuclear Regulatory Commission (NRC), through its Standard Review Plan (SRP), requires that plant designers follow certain prescribed methods and criteria in the estimation of dynamic effects associated with the postulated rupture of piping.

The work reported in this NUREG is intended to provide the basis for NRC decisions on adopting revisions to parts of the SRP 3.6.2 entitled "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping." The revisions considered in this work evaluated updated prescriptions for calculating jet impingement forces on critical systems and the requirement to consider pipe-whip damage to a new population of pipes.

In accordance with the procedures documented in NUREG/CR-3586 entitled "A Handbook for Value-Impact Assessment", this report found indication that substantial costs and occupational radiation exposure would result from the proposed action without substantially reducing the risks to public health and safety.

EXECUTIVE SUMMARY

The Code of Federal Regulations (Title 10, Part 50) requires that structures, systems, and components important to the safety of nuclear power plants in the United States be designed to withstand the effects of natural phenomena and the effects of normal and accident conditions. General Design Criterion 4 (GDC-4), "Environmental and Dynamic Effects Bases," requires that structures, systems, and components important to safety be designed to accommodate the effects of postulated accidents, particularly the loss-of-coolant accident (LOCA). To account for effects that might result from a LOCA, especially the protection of critical reactor monitoring and control-related systems, nuclear power plant designers have been required to analyze the consequences of double-ended guillotine breaks (DEGB) in high-energy piping. Consistent with this need, the U.S. Nuclear Regulatory Commission (NRC), through its Standard Review Plan (SRP), suggests that plant designers follow certain prescribed methods and criteria in the estimation of dynamic effects associated with the postulated rupture of piping.

On occasion, however, the NRC deems it appropriate to modify the content of the SRP when scientific evidence--either from reactor operating experience or independent studies--supports such a position. The work reported in this NUREG is intended to provide the basis for NRC decisions on adopting revisions to parts of the SRP 3.6.2 entitled "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping." The revisions considered in this work address prescriptions for calculating jet impingement forces on critical systems and the requirement to consider pipe-whip damage to a new population of pipes (pipes of diameter and wall thickness equal to the whipping pipe). This study provides an analysis of the value-impact of these SRP modifications to the general public (health and safety risks only), the nuclear power industry, and to the U.S. Nuclear Regulatory Commission. Controlling value and impact are associated with the proposed changes to the SRP regarding pipe-to-pipe impact damage considerations. This is because the proposed changes to the SRP regarding jet impingement forces do not lead to any elimination or installation of jet impingement barriers or shields. However, proposed changes to the SRP regarding pipe-to-pipe impact would lead to considerable impact to the nuclear power industry in terms of both occupational radiation exposure and dollar costs with only a negligibly small increase in value to the public health and safety.

As a result of estimating value and impact in this study, the following conclusions can be made.

- Incorporation of the proposed change to SRP 3.6.2 regarding the estimation of jet impingement forces would most likely require the nuclear power industry to perform some re-analysis of their postulated pipe breaks. Although the proposed changes generally lead the user to predict reduced jet impingement forces on a given target (less conservative), a prediction of wider jet plumes (wider than predicted by the old 10-degree recipe) at some distances from the postulated breaks may result in the industry's need to slightly modify (increase the effective protected area) some jet shields adjacent to stable breaks (postulated breaks in pipes that are fitted with restraints). Because the proposed change regarding jet impingement forces would not result in the addition or removal of jet impingement barriers or shields in either operating plants or plants under construction, decision factors analyzed in this regulatory analysis were not sensitive--not impacted--by this proposed change.

- Incorporation of the fact in SRP 3.6.2 that severe damage can result from pipe-to-pipe impact between pipes of equal diameter and wall thickness would require the nuclear power industry to re-analyze all unstable breaks (postulated breaks in pipes that are not restrained) and install new pipe-whip restraints where appropriate. Values and impacts associated with changes to pipe-to-pipe impact analysis methods are estimated to be as follows (for 118 plants):

	Best Estimate	High Estimate	Low Estimate
Public health value (man-rem)	9E-4	2	4E-6
Occupational exposure impact (man-rem)	6E+3	2E+4	1E+3
Impact ($)	80M	200M	20M

- The proposed action requiring re-analysis of pipe-to-pipe impact damage suggests a check on the importance of the pipe impact damage probability to value-impact results. Studies aimed at determining this sensitivity showed a direct affect to factors dependent on core-melt frequency [i.e., public health risk, accidental occupational radiation exposure (ORE), and onsite and offsite economic risks] and no affect to the other factors (routine ORE and industry costs). A doubling of the pipe impact damage probability, for example, caused a likewise doubling of the negligibly small public health risk, and a doubling of the accidental ORE and industry economic risks. However, because of the insensitive nature of the controlling routine ORE, total ORE is insensitive to changes or inaccuracies in pipe impact damage probability. Likewise, because of the insensitive nature of the controlling industry implementation cost, total costs are insensitive to changes or inaccuracies in pipe impact damage probability.

- Sensitivity studies aimed at determining the effects of introducing the leak-before-break (LBB) philosophy to PWR piping showed that use of LBB in larger-diameter piping ($D \geq 2$ in.) had very little impact on public health risk. On the other hand, the introduction of LBB did play a significant role in lowering routine ORE and industry implementation and operating costs.

ACKNOWLEDGEMENTS

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The value-impact assessment described in this report was based largely on information supplied by various sources within the nuclear power industry, a majority of which is documented in past value-impact studies performed by the Lawrence Livermore National Laboratory (LLNL) under the leadership of Dr. G.S. Holman. In addition to the use of specific data regarding the number and costs of pipe-whip restraints, the authors recognize the helpful suggestions provided by Dr. Holman directly.

The authors would also like to express their appreciation for many helpful suggestions from Dr. D.N. Norris of the Electric Power Research Institute, and Drs. Shou-nien Hou and P.T. Kuo of the Mechanical Engineering Branch, and Dr. J.A. O'Brien of the Structural and Seismic Engineering Branch, U.S. Nuclear Regulatory Commission. Also key in the conduct of the value-impact assessment was the guidance provided by Drs./Messrs. F.A. Simonen, S.H. Bush, P.J. Pelto, T.V. Vo, and M.F. Mullen, all of the Pacific Northwest Laboratory.

<u>CONTENTS</u>

FIGURES

TABLES

INTRODUCTION

The Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), requires that structures, systems, and components important to the safety of nuclear power plants in the United States be designed to withstand the effects of natural phenomena and the effects of normal and accident conditions [1]. The U.S. Nuclear Regulatory Commission (NRC), through its regulations, Regulatory Guides, Branch Technical Positions, and the Standard Review Plan (SRP) has required that various accident loads and loads caused by natural phenomena be considered, both individually and in appropriate combinations, in the analysis of safety-related structures, systems, and components.

Designing safety-related structures, systems, and components to withstand the effects of loss-of-coolant accidents (LOCAs) is one important load requirement. To account for these effects, the design basis of nuclear power plants has historically included postulation of double-ended guillotine breaks (DEGBs) in certain high-energy systems, such as reactor coolant piping. These requirements have necessitated analyses to evaluate hydrodynamic loads and the resultant response of structures and mechanical components, and has led to the placement of massive pipe-whip restraints and jet impingement barriers (or "jet shields") near piping as protection against the dynamic effects of postulated pipe breaks.

The NRC position on postulation of pipe ruptures is presented in the Standard Review Plan, Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping" [2]. The SRP is prepared for the guidance of staff reviewers in the Office of Nuclear Reactor Regulation in performing safety reviews of applications to construct or operate nuclear power plants. The principal purpose of the SRP is to assure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. It is also a purpose of the SRP to make information about regulatory matters widely available and to improve communication and understanding of the staff review process by interested members of the public and the nuclear power industry.

The safety review is primarily based on the information provided by an applicant in a Safety Analysis Report (SAR). Section 50.34 of 10 CFR 50 of the Commission's regulations requires that each application for a construction permit for a nuclear facility shall include a Preliminary Safety Analysis Report (PSAR) and that each application for a license to operate such a facility shall include a Final Safety Analysis Report (FSAR). The SAR must be sufficiently detailed to permit the staff to determine whether the plant can be built and operated without undue risk to public health and safety. Prior to submission of an SAR, an applicant should have designed and analyzed the plant in sufficient detail to conclude that it can be built and operated safely. The SAR is the principal document in which the applicant provides the information needed to understand the basis upon which this conclusion has been reached.

The NRC is now considering an update to SRP 3.6.2 to include the latest methods and criteria for determining jet impingement loads and pipe-to-pipe impact phenomena. As a result of this consideration, the Pacific Northwest Laboratory[a] (PNL) was placed under contract to develop a basis (including a value/impact evaluation) as to whether the above revised methods and criteria should be endorsed by the NRC. This study is therefore based on evaluating the values and impacts to the general public, the nuclear power industry, and the U.S. NRC as a result of the NRC formally adopting the latest methods and criteria for determining jet impingement loads and pipe-to-pipe impact in their SRP. The change or modification in NRC policy that this study is based on is defined by the differences in guidance provided in the current SRP 3.6.2 (defines the old or current methods and criteria) and the final draft of the American National Standard (ANS) 58.2 [3] (defines the proposed methods and criteria). These differences in guidance provided by the current SRP 3.6.2 and the draft ANS 58.2 are defined in this study as the proposed action.

Incorporation of the proposed action regarding jet impingement forces would not result in the elimination of current, or installation of new, protective barriers or shields in operating pressurized water reactors (PWRs) or boiling water reactors (BWRs). This is because the newer guidance is generally less conservative (predicts lower jet impingement forces) than the current guidance provided in SRP 3.6.2. However, jet force prediction using the proposed methods will still lead design engineers to incorporate barriers and jet shields as before. Incorporation of the proposed action regarding pipe-to-pipe impact will require the industry to re-analyze, for potential damage, all current unstable breaks (postulated pipe breaks in high-energy piping not restrained for one reason or another) that could involve interaction with adjacent piping of equal diameter and wall thickness. All potential pipe-to-pipe interactions involving pipes (whipping and target) of equal diameter and wall thickness will require the installation of new pipe-whip restraints. It is the requirement of re-analysis and installation of new restraints that results in the controlling values and impacts determined in this study.

Values and impacts associated with the proposed action were estimated and displayed using the major decision factors and format suggested in NRC's Handbook for Value-Impact Assessment [4].

(a) Operated for the U.S. Department of Energy by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830.

PROPOSED ACTION

The proposed action is defined in this study as the differences in engineering-design guidance provided in the current SRP 3.6.2 [2] and the proposed ANS 58.2 [3] regarding the prediction of jet impingement forces and pipe-to-pipe impact damage. The pertinent parts of these documents and their differences are discussed in this section.

Under the proposed action, the following modifications to the current SRP 3.6.2 would be adopted by the NRC from ANS 58.2.

1. The methods and criteria for estimating jet impingement loads described in Appendices C and D in the latest draft of ANS 58.2.

2. The potential for damage from pipes of equal size impacting together as expressed in the May 1988 draft of ANS 58.2 and supported by NUREG/CR-3231 [5].

In all cases, the basis for defining the proposed action should be consistent with current NRC positions on pipe rupture and leak-before-break (LBB).

JET IMPINGEMENT FORCES

SRP 3.6.2

Under the proposed change to SRP 3.6.2, jet impingement forces are to be determined per instructions in the latest draft of ANS 58.2. Under SRP 3.6.2-7 and -8, the following is stated as a guideline for review of analysis of jet impingement forces.

> "These analyses should show that jet impingement loadings on nearby safety-related structures, systems, and components will not be such as to impair or preclude essential functions. Assumptions that are acceptable in modeling jet impingement forces are:
>
> a. The jet area expands uniformly at a half angle not exceeding 10 degrees.
>
> b. The impinging jet proceeds along a straight path.
>
> c. The total impingement force acting on any cross-sectional area of the jet is time and distance invariant, with a total magnitude equivalent to the jet thrust force as defined in Subsection III.2.c(4), above.
>
> d. The impingement force is uniformly distributed across the cross-sectional area of the jet, and only the portion intercepted by the target is considered.

e. The break opening may be assumed to be a circular orifice of cross-sectional flow area equal to the effective flow area of the break.

f. Jet expansion within a zone of five pipe diameters from the break location is acceptable if substantiated by a valid analysis or testing, i.e., Moody's expansion model [6]. However, jet expansion is applicable to steam or water-steam mixtures only, and should not be applied to cases of saturated water or subcooled water blowdown.

Analyses of pipe break dynamic effects on mechanical components and supports should include the effects of both internal reactor pressure vessel asymmetric pressurization loads and expand asymmetric compartment pressurization loads, as appropriate, as discussed for PWR primary systems in Reference 7." (NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," Resolution of Generic Task Action Plan A-2.)

ANS 58.2

Under ANS 58.2, jet impingement loads are defined as the force exerted by the jet impinging on a target and being turned or diverted to a different direction. The loads are assumed to be a function of jet properties of steam quality, velocity, pressure, temperature, and cross-sectional area at the point of interaction with the target and the shape of the target itself. The jet impingement load may be calculated by establishing the pressure distribution on the component and integrating the pressure over the target surface or by calculating the momentum change of the jet caused by the target. Appendix D of ANS 58.2 reviews acceptable methods for determining jet impingement forces and jet pressure. ANS 58.2 goes on to say that the jet impingement loading rate is important and shall be given proper consideration in the evaluation of jet impingement loads on target equipment and structures. The response of the target is a function of the stiffness characteristics of the target and the jet impingement loading rate. This response shall be determined from a dynamic analysis utilizing the actual impingement force loading rate, or from an equivalent static analysis with the use of a dynamic load factor.

ANS 58.2 also states that the movement of the jet centerline caused by pipe whip shall be taken into account in the characterization of jet impingement loads on a target. For example, the jet impingement load on a target located between the initial and final resting position of a whipping pipe can be characterized as an impulse load with a time width equal to the time of jet/target inter- action and an amplitude equal to the load average of the time interval.

Sandia Model

In an attempt to reduce over-design from the use of Moody's jet load model [6], NRC provided funding to Sandia National Laboratories (SNL) to study the use of modern multidimensional computational methods to evaluate the two-phase jet load on target geometries. The governing equations of mass, momentum, and energy were solved with a high resolution Eulerian method for all calculations. The calculations form a computational data base for evaluating jet and target pressures for axisymmetric target geometries. A two-phase jet load model, which provides both pressure and load distributions, was then developed using the computational data base [7].

Comparisons of SRP 3.6.2, ANS 58.2, and Sandia Model

To compare methods and criteria identified in Appendices C and D of ANS 58.2 and the more general guidelines of SRP 3.6.2, calculations were performed using the appropriate formulas for a series of sample cases. Depending on the degree of engineering judgement utilized in following the SRP guidelines, differences range from: 1) the SRP 3.6.2 guidelines always lead to greater jet forces (more conservative) than the proposed ANS 58.2 recipes, to 2) under certain conditions, ANS 58.2 guidance predicts essentially the same jet force values.

Examples of this range in values are provided by an analysis of two scenarios. Both scenarios assume an initial pressure of 2000 psi and 25°F subcooling. The first scenario uses the most conservative SRP jet force assumption possible, i.e., a jet expansion of 10 degrees (labeled "SRP 10 Degrees") and compares it with predictions of jet forces using ANS Appendix C and D (labeled "ANS", Figure 1). The second scenario uses Moody's assumption of jet expansion for a distance from the break opening to five pipe diameters for the SRP estimation (labeled "SRP2", Figure 1). In both scenarios 1 and 2, the jet target configuration was assumed to be another pipe of equal diameter to the one that suffered the break. Results show that ANS jet force predictions are always smaller (less conservative) than SRP predictions.

Because of its occasional use in the nuclear power plant industry, and because of its recognition in ANS 58.2, a jet impingement model developed at Sandia National Laboratory [7] was analyzed. An examination of the SNL model indicates that it was developed for estimating loading from two-phase jets on axisymmetric flat targets infinite in radial extent. As described in the model analysis, use of the SNL jet load model is not limited to targets with large radii. Except for a small subsonic region of flow near the center of the target, flow on the target is supersonic; thus, edge effects are negligible and the solution is essentially independent of target radius.

Representations of the other jet impingement models considered in our review (described in SRP 3.6.2 and ANS 58.2 Appendices C and D) have not specifically addressed the target type but have considered the size of the target to be small when compared to the jet cross section. These models essentially consider the uninterrupted expansion of the jet whereas the SNL analysis considers the target to completely stop the forward motion of the jet at the target plane. Thus, the modeling of these different types of jet impingement situations leads to different jet impingement forces at the target. However, for comparison purposes, predicted jet centerline pressures were evaluated as a function of distance from a postulated break for the three models (SRP 3.6.2, ANS 58.2 Appendices C and D, and SNL) (Figure 2). These graphical comparisons estimate jet centerline pressures for an initial pipe pressure of 2000 psi and 25°F subcooling. For the SNL result, the axial distance is the distance from the pipe break to the target and the reported pressure is the pressure on the target face. For the other models, the pressures are those in the jet.

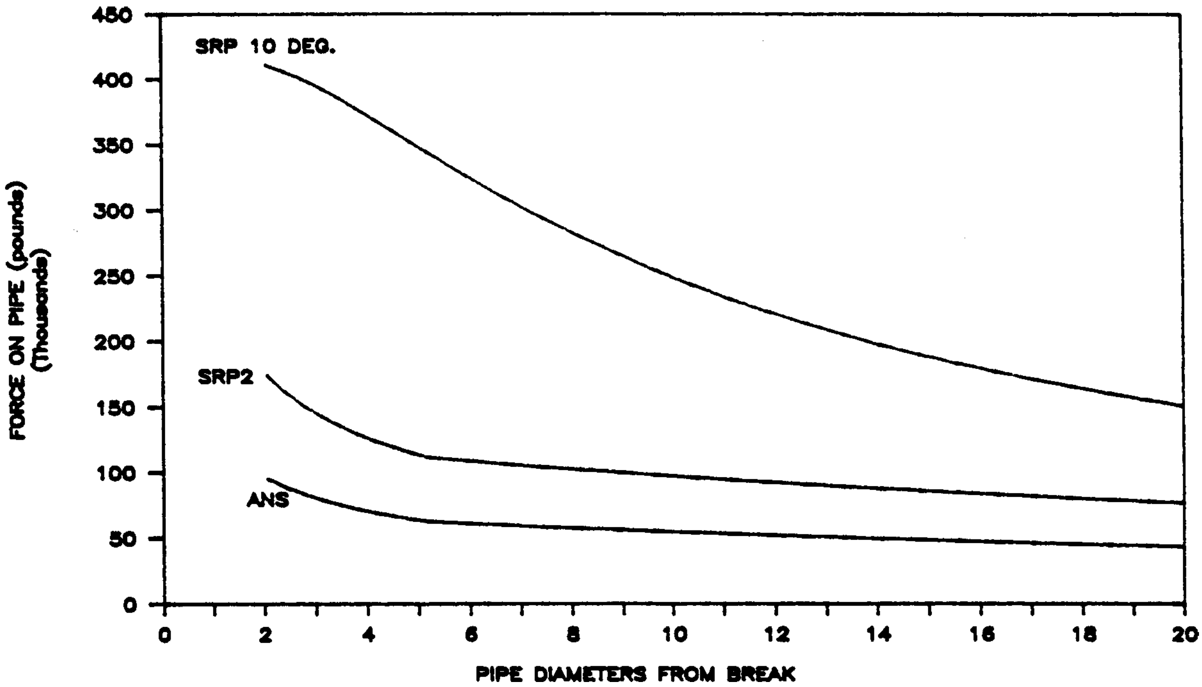

FIGURE 1. Jet Impingement Force on Pipe Through Center of Jet

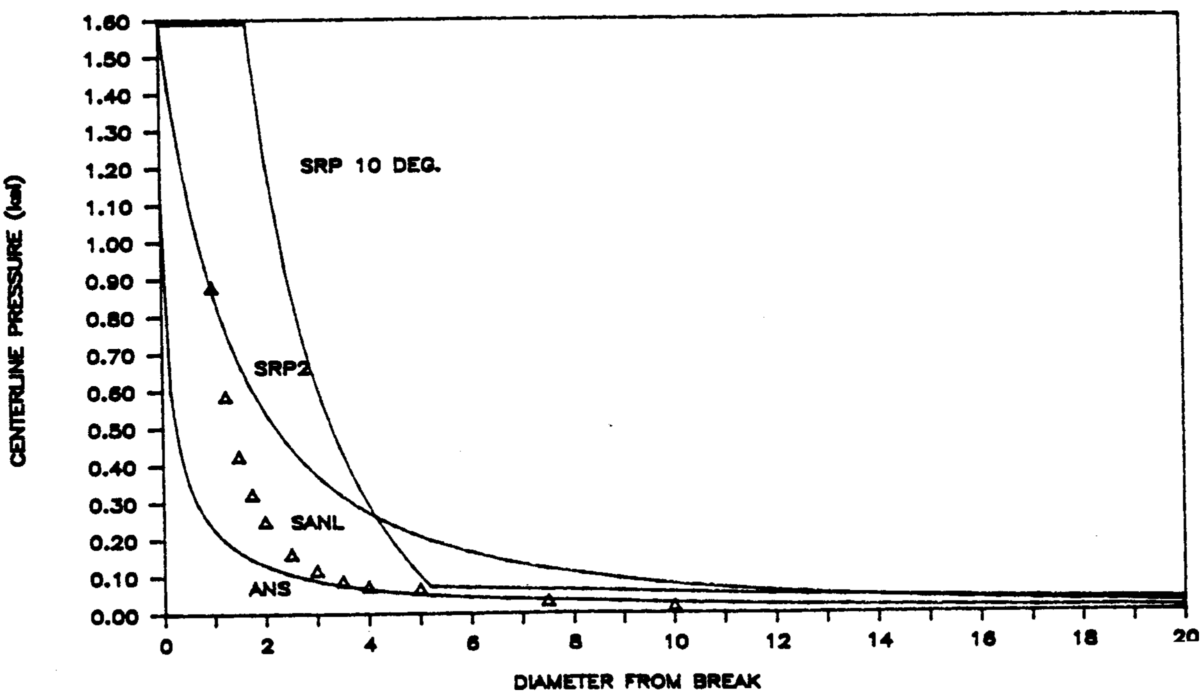

FIGURE 2. Jet Centerline Pressure as Predicted by Assumptions in SRP 3.6.2, ANS 58.2, and SNL Models

By comparing jet centerline pressures as predicted by the three models (Figure 2) and recalling that solutions are essentially independent of target size in the Los Alamos National Laboratory (LANL) model, it can be concluded that current methods (SRP 3.6.2) will predict target forces--on small targets--that are greater than, or equal to, forces predicted by the proposed action (ANS 58.2 methods). Therefore, incorporation of the proposed action in SRP 3.6.2 would not lead to any additions or eliminations of jet impingement barriers or shields in operating plants.

The following conclusions are made as a result of these comparisons.

1. SRP 3.6.2 methods predict jet impingement forces equal to or greater than those predicted by ANS 58.2 methods (i.e., SRP is more conservative than ANS).

2. Not shown in the figures, ANS 58.2 methods can predict slightly wider jet-plume cross sectional areas compared to the 10-degree SRP 3.6.2 model at some distances from the pipe break. Although the 10-degree model leads to greater overall jet force estimations on any target, slightly larger jet-target interaction zones can result when using the ANS 58.2 methods.

PIPE-TO-PIPE IMPACT

The current version of the pipe whip damage criteria, as expressed in the SRP 3.6.2 (Rev. 1, July 1981), is:

> "An unrestrained whipping pipe should be considered capable of causing circumferential and longitudinal breaks, individually, in impacted pipes of smaller nominal pipe size, and developing through-wall cracks in equal or larger nominal pipe sizes with thinner wall thickness, except where analytical or experimental, or both, data for the expected range of impact energies demonstrates the capability to withstand the impact without rupture."

Because of the Pipe-to-Pipe Program performed at PNL [5], ANS 58.2 was modified to read:

> "Pipe whip shall be considered capable of causing circumferential and longitudinal breaks, individually, in impacted pipes of smaller nominal pipe size, irrespective of pipe wall thickness, and developing through-wall cracks in equal or larger nominal pipe sizes with equal or thinner wall thickness. Analytical or experimental data, or both, for the expected range of impact energies may be used to demonstrate the capability to withstand the impact without rupture; however, loss of function caused by reduced flow in the impacted pipe should be considered."

The proposed modification causes pipes of equal diameters and thicknesses to be regarded as an additional category that is susceptible to rupture in an impact event.

AFFECTED DECISION FACTORS

Major decision factors affected by the proposed action and addressed in this value-impact assessment are summarized in Table 1.

TABLE 1. Major Decision Factors Affected by the Proposed Regulatory Action

Decision Factors	Causes Quantified Change	Causes Unquantified Change
Public health risk	X	
ORE[a] (accidental)	X	
ORE (routine)	X	
Offsite economic risk	X	
Onsite economic risk	X	
Regulatory efficiency[b]		X
Improvements in knowledge[b]		X
Industry implementation cost	X	
Industry operation cost	X	
NRC development cost	X	
NRC implementation cost	X	
NRC operation cost	X	
Power replacement cost[b]		X

(a) Occupational Radiation Exposure
(b) See Page 10 for definition.

VALUE-IMPACT ASSESSMENT SUMMARY

Table 2 summarizes the results of the regulatory analysis. Values or impacts are presented as appropriate for each of the major decision factors affected by the proposed action. In this table, positive "value" represents reductions in human exposure, public or occupational; positive "impacts" are added costs associated with the proposed action. (Refer to appropriate section of report to justify the numbers.)

TABLE 2. Results of Value-Impact Assessment (Total for 118 Plants)

Factors	Best Estimate	High Estimate	Low Estimate
Values (man-rem)			
Public health risk	9E-4	2	4E-6
ORE (accidental)	1E-3	7	3E-6
ORE (routine)	-6E+3	-2E+4	-1E+3
Values subtotal	-6E+3	-2E+4	-1E+3
Impacts ($)			
Industry implementation cost	8E+7	2E+8	2E+7
Industry operating cost	4E+6	1E+7	8E+5
NRC development & implementation cost	0	0	0
NRC operating cost	0	0	0
Power replacement cost	0	0	0
Offsite economic risk	-1	-3E+4	0
Onsite economic risk	-2	-9E+3	-4E-3
Impact subtotal	8E+7	2E+8	2E+7

Notes:

1. Operating costs reflect a 10% discount rate.
2. Related to implementation of the proposed action, power costs due to accidents are included under onsite economic risk.
3. Economic risk estimates reflect a 10% discount rate.

UNQUANTIFIED RESIDUAL ASSESSMENT

The following decision factors were not quantified in our assessment of the proposed action.

- Regulatory efficiency - It is anticipated that incorporation of the proposed changes to the SRP will increase the allocation of NRC resources for technical review and follow-up with the industry. That is, the industry will be required to perform a re-analysis of certain postulated pipe breaks and determine the appropriate action(s); the NRC will have to review these re-analyses and make appropriate recommendations to the industry.

- Improvements in knowledge - The proposed SRP revisions, if formalized, will require the operating industry to re-analyze both stable terminal end breaks and unstable pipe breaks. These additional investigations will add insight into the extent to which more general exclusions of pipe breaks--such as would be permissible under the broad scope General Design Criterion 4 (GDC-4) rule change--could potentially be applied.

- Power replacement costs - This analysis assumes that any industry actions taken to implement the proposed changes in the SRP revision would be performed either prior to initial startup (construction plants) or during scheduled or forced outages (operating plants). Consequently, no power replacement costs uniquely associated with implementing the proposed action would be incurred. Power replacement costs resulting from accidents are included as part of onsite economic risk.

DEVELOPMENT OF QUALIFICATION

This section presents the detailed development of value impact for the proposed action. The assessment is based on the following general assumptions.

1. A total of 77 PWR plants are considered, of which 67 are operating and 10 are under construction effective March 31, 1988. Appendix A gives a vendor-by-vendor breakdown of operating plants or plants under construction. The PWR plants have a total remaining life time of 2370 plant-years (py)[a], assuming a design lifetime of 40 yr, and an overall average lifetime of 30.8 yr (29.5 yr for operating plants only). It was assumed that plants under construction have an average forward-fit (time to operation) of zero years.

2. A total of 41 BWR plants are considered, of which 38 are operating and 3 are under construction effective March 31, 1988. The Appendix gives a vendor-by-vendor breakdown of operating plants and plants under construction. The 41 plants have a total remaining lifetime of 1250 py, and an overall average remaining lifetime of 30.6 yr (29.8 yr for operating plants). It was assumed that plants under construction have an average forward-fit of zero yr.

3. The combined group of 118 plants has a total remaining lifetime of 3620 py, and an overall average remaining lifetime of 30.7 yr (29.6 yr for operating plants only). It was assumed that plants under construction have an overall average forward-fit of zero yr. The zero forward-fits for both PWR and BWR plants indicates that all non-operational units are in advanced stages of construction.

4. The only values and impacts considered in this analysis are those uniquely associated with the introduction of the proposed changes to the SRP. Values and impacts resulting from the general elimination of pipe breaks as a plant design basis [such as avoided costs and occupational routine exposure associated with whip restraints on pipes associated with arbitrary intermediate breaks (AIB) locations or approved for the LBB philosophy] are addressed in the GDC-4 and AIB regulatory analyses [8,9].

PUBLIC HEALTH RISK

The change in "value" corresponding to the proposed NRC action is caused by the change in risk to public health and safety as a result of the nuclear power industry following the new directives of the proposed action. Risk is determined by the product of frequency (probability) of a component failure/accident and the resulting damage or consequence of the failure. Changes in risk will, therefore, result from changes in either probabilities and/or consequences associated with a reference accident. The accident of reference

(a) Note that py denotes plant years and /py denotes per plant year throughout the document.

in this study is the nuclear reactor core-melt accident and the frequency for this accident is represented by the sum of all events that lead to accident sequences that in turn lead to component failures in systems important to preventing a core-melt accident. Therefore, changes in risk will result from adding new pipe-whip restraints to existing piping systems (operating plants) or planned systems (construction plants) of nuclear power plants. That is, an increase in the number of pipe restraints will result in a small decrease (risk reduction) in the estimated frequency of failures to a nearby safety system (e.g., a pipe of equal diameter and wall thickness) from a pipe break event.

Therefore, changes to public health or risk associated with the proposed action were determined by developing values for the following probabilities and dose estimates:

- probability that a single unstable (unrestrained) high-energy pipe will suffer a guillotine rupture (break)
- probability that the ruptured pipe will undergo an unstable whipping action
- probability that the whipping pipe will cause serious damage to an adjacent pipe of equal size and wall thickness on impact
- changes in core-melt frequency caused by the failure of the struck pipe
- probabilities of postulated release sequences following core melt
- radiation dose estimates for each of the release sequences and core-melt frequencies.

The following discussion explains how changes to public health were determined and notes the data bases that were used in the calculations. It is assumed that the proposed change to the SRP regarding jet impingement load determination does not result in a change to risk and core-melt frequency. This assumption is judged to be reasonable as the proposed action will not result in the addition or subtraction of jet barriers or shields; it only results in changes to estimated jet loads on a jet impingement target. However, incorporation of the proposed change to the SRP regarding pipe-to-pipe impact will likely cause the industry to add pipe-whip restraints that will theoretically decrease the core-melt frequency.

Changes in risk and core-melt frequency stemming from the proposed changes to pipe-to-pipe impact damage prediction were developed by following the above bulleted sequence. First, the probability that a high-energy pipe will undergo a guillotine break is determined. By considering only "unstable" initial breaks (pipes having no restraints), the probability that the "unstable break" will whip is considered unity. Then the probability that an adjacent pipe of equal diameter and wall thickness will undergo a failure when struck by the whipping pipe is determined. The probability that the impacted pipe will fail is then applied to appropriate accident sequences to estimate the change in probability of a core melt and the resultant change in public risk from radioactive releases. PNL based its assessment of the change to public risk based on the following general assumptions.

- The risk associated with any given unstable guillotine break depends solely on the diameter of the affected piping. This approach is analogous to that taken towards LOCAs in WASH-1400 [10], which divided pipe breaks into three size ranges: large LOCA (D > 6 in.) henceforth designated by LLOCA, small LOCA (2 in. ≤ D ≤ 6 in.) hence-forth designated as S1LOCA, and small-small LOCA (1/2 in. ≤ D < 2 in.) henceforth designated as S2LOCA. These size ranges were used in this analysis and allowed the generic accident sequences and release categories from WASH-1400 to be used.

- For a given plant, the probability of having at least one essential pipe failure resulting from the whipping action of an adjacent ruptured pipe of equal diameter is represented by the number of equally sized pipe pairs (whipping pipe and adjacent target pipe) that exist, multiplied by a generic break probability dependent only on pipe diameter (consistent with WASH-1400 [10] assumptions), multiplied in turn by the probability that the target pipe once struck will be severely damaged or fail.

- Only pipe damage/failure caused by pipe whips inside containment contribute to changes in plant risk.

This study used the same data base accumulated by Lawrence Livermore National Laboratory (LLNL) in an earlier value-impact study [8] to determine the number of postulated breaks generically applicable to all plants (PWR and BWR). The LLNL data were based on plant-specific information supplied to them by several plant owners. Note that it would, in principle, be more accurate to evaluate risk on a system-by-system basis. This however, would ideally require for each system, that:

- location and walk-down confirmation of all candidate breaks be performed to determine the number of unstable breaks that could involve interaction with adjacent piping of equal diameter and wall thickness for each reactor

- a detailed fracture mechanics evaluation be performed to estimate rupture probabilities based on normal operating and postulated accident conditions for each unstable break involving interaction with adjacent piping of equal diameter and wall thickness

- a detailed analysis of unstable pipe-whip dynamics be performed to determine the energy imparted to adjacent piping of equal diameter and wall thickness

- a detailed fracture mechanics evaluation be performed to estimate the fraction of adjacent piping that suffer severe impact damage

- the effect of damaged adjacent piping on overall plant safety be assessed.

However, for this analysis, a simplified approach was followed. The approach utilized the above mentioned generic plant data to lead to estimates of the total number of postulated pipe breaks typically found in a PWR and a BWR reactor, Table 3. These data were then used to estimate the best, high, and low estimates for the number of unstable breaks inside containment that required restraints (i.e., were located adjacent to high-energy piping of equal diameter and wall thickness), Table 4. Note that this approach adds a new set of postulated breaks that require restraints; thus, the resultant analyses represent a true incremental value-impact analysis associated with the proposed action.

Probability of Pipe Guillotine Break

The best-estimate probabilities of unstable pipe ruptures were based on the results of prior LLNL probabilistic studies of reactor coolant piping reliability in PWR and BWR plants [11-14]. These evaluations focused on estimating the probability of a double-ended guillotine break (DEGB) in PWR coolant loops and in BWR main steam, feedwater, and recirculation loop piping. To estimate the probability of a DEGB, PNL considered two causes of pipe break: 1) pipe fracture caused by the growth of cracks at welded joints (direct DEGB), and 2) pipe rupture indirectly caused by the seismically-induced failure of critical supports or equipment (indirect DEGB). In the LLNL studies, the probability of direct DEGB was estimated using a probabilistic fracture mechanics model that calculated the growth of as-fabricated surface flaws at welded joints, taking into account loads on the piping caused by normal operating conditions as well as seismic events. Other factors, such as the capability to detect cracks by nondestructive examination and the capability to detect pipe leaks, were also considered by LLNL. A detailed evaluation of Westinghouse plants [11] yielded a best-estimate system probability of direct DEGB of 1.0E-12/plant-year (/py) for plants east of the Rocky Mountains, with a 90th-percentile value (i.e., 90% confidence limit) of 1.0E-10/py; this latter value also bounded the direct DEGB probabilities for west coast plants and for Combustion Engineering plants [12]. Although the probability of direct DEGB was not explicitly estimated for Babcock & Wilcox plants, LLNL [8] concluded, by a review of reactor coolant loop stress information, that the probability of crack-induced pipe break should be similarly low. Therefore, this estimate was assumed applicable to all PWR plants.

Therefore, based on the studies performed by LLNL, PNL established best estimates for the annual probability of pipe rupture per each unstable break considered:

- $D > 6$ in.; $P = 1.2E-12$/postulated unstable break/yr
- 2 in. $< D \leq 6$ in.; $P = 1.4E-10$/postulated unstable break/yr
- 1/2 in. $\leq D \leq 2$ in.; $P = 1.4E^{-9}$/postulated unstable break/yr.

PNL assumed that the low estimate of break probability for each size range was simply an order-of-magnitude less than the corresponding best-estimate probability.

TABLE 3. Estimated Pipe Break Locations in PWR Data Base [8]

Plant Name	Units	Break Locations (IC) LLOCA	S1LOCA	S2LOCA	TOTAL	Break Locations (OC)	Total Break Locations
Beaver Valley 2	1	40	76	16	132	246	378
Byron 1,2	2	80	36	140	256	60	316
Braidwood 1,2	2	80	36	140	256	60	316
Catawba 1,2	2	50	42	64	156	10	166
McGuire 1,2	2	50	42	64	156	10	166
Seabrook 1	1	44	46	96	186	124	310
South Texas 1,2	2	124	76	8	208	264	472
Vogtle 1,2	2	34	136	14	184	180	364
WNP-1	1	52	102	42	196	108	304
Best estimate (per unit)		64	64	68	196	110	306
High estimate (per unit)		94	100	120	314	204	518
Low estimate (per unit)		34	28	16	78	0	94

TABLE 4. Estimated Pipe Break Locations in BWR Data Base [8]

Plant Name	Units	Break Locations (IC) LLOCA	S1LOCA	S2LOCA	TOTAL	Break Locations (OC)	Total Break Locations
Clinton	1	66	24	0	90	0	90
Hope Creek	1	54	8	0	62	100	162
LaSalle	2	12	8	0	20	0	20
Best estimate (per unit)		36	12	0	48	50	48
High estimate (per unit)		64	20	0	84	100	84
Low estimate (per unit)		8	4	0	12	0	12

NOTES: 1. Location: Inside containment (IC), outside containment (OC)
Size: LLOCA (D > 6 in.), S1LOCA (2 in. < D $\leq$ 6 in.), S2LOCA (1/2 in. $\leq$ D $\leq$ 2 in.)

2. High and low estimates reflect $\pm 1\sigma$ averaged over all units.

The high estimate of break probability is based on the NRC position stated in the GDC-4 rule change, which allows exclusion of dynamic effects associated with DEGBs when the probability of a break can be demonstrated to be extremely low. The GDC-4 revision defines "extremely low" as a system rupture probability on the order of 1E-6/yr when all rupture locations are considered in the fluid system piping or portions thereof. PNL divided this probability by four break locations (two intermediate plus two terminal ends) to establish a high estimate of break frequency (2.6E-7/yr) independent of pipe diameter.

Probability of Pipe-to-Pipe Impact Failure

To determine the probability that a whipping pipe (a pipe that suffers a direct or indirect DEGB) will cause a failure in an adjacent pipe, the results of an earlier PNL testing program were used [5].

In the PNL study, a series of impact tests were conducted to establish the pipe-to-pipe failure conditions for pipes of varying sizes. The results of the test program displayed some differences from results that are assumed by observation of the current SRP 3.6.2 [2]. An important difference was stated in the following program conclusions.

- There is the possibility that pipes of equal diameter and equal thickness could fail under impact conditions. This combination of piping size is not included within the criteria as a category that could experience failure.

- The probability of failure of the impacted pipe was estimated by using judgement to attach numerical values to the important constituent probabilities. The total probability, P_0, given an impact event was estimated to be at least one impact in 50 (2E-2) will result in a failure [5].

The total probability, P_0, was determined by considering the following constituents (i.e., $P_0 = P_1 \times P_2 \times P_3 \times P_4 \times P_5$):

- moving pipe has a thicker wall ($P_1 = 0.25$)
- motion exceeds three to four diameters ($P_2 = 0.075$)
- jet force is normal to moving pipe ($P_3 = 0.5$)
- impacting pipes are normal ($P_4 = 0.25$)
- support/hinge is several diameters from the break ($P_5 = 0.75$).

Because the proposed action is specific in terms of pipe diameters and wall thicknesses being equal, P_1 was set equal to 0.1, which yields an overall best-estimate probability, P_0, of 7E-3.

Therefore, the resultant "initiating event" frequencies, summarized in Table 5, take into account the single break probability for each piping size range, the number of unstable-break/adjacent-pipe pairs, and the probability that the impacted pipe will fail once struck by the moving pipe.

It is noted here that although these initiating event frequencies are considered "generic", the frequencies for different pipe-diameter ranges cannot be combined into a meaningful single frequency for an individual PWR or BWR

TABLE 5. Initiating Frequencies for Pipe Failure Caused by Pipe-to-Pipe Impact

	Size Range LLOCA	S1LOCA	S2LOCA	Total Locations (IC)
Unstable breaks per PWR plant[a]				
Best estimate	3	3	3	9
High estimate	5	5	6	16
Low estimate	2	1	1	4
Unstable breaks per BWR plant[b]				
Best estimate	4	1	0	5
High estimate	6	2	0	8
Low estimate	1	0	0	1
Single DEGB frequency (/yr)				
Best estimate	1.2E-12	1.4E-10	1.4E-9	
High estimate	2.5E-7	2.5E-7	2.5E-7	
Low estimate	1.2E-13	1.4E-11	1.4E-10	
Estimated failure frequency for impacted pipe				
Best estimate	7.0E-3	7.0E-3	7.0E-3	
High estimate	2.0E-2	2.0E-2	2.0E-2	
Low estimate	1.0E-3	1.0E-3	1.0E-3	
Pipe failure initiating frequency (/py), PWR plants				
Best estimate	2.5E-14	2.9E-12	2.9E-11	
High estimate	2.5E-8	2.5E-8	3.0E-8	
Low estimate	2.4E-16	1.4E-14	1.4E-13	
Pipe failure initiating frequency (/py), BWR plants				
Best estimate	3.4E-14	9.8E-13	0	
High estimate	3.0E-8	1.0E-8	0	
Low estimate	1.2E-16	0	0	

(a) The number of unstable breaks requiring restraints is estimated to be 5% of the PWR pipe break locations shown in Table 3.

(b) The number of unstable breaks requiring restraints is estimated to be 10% of the BWR pipe break locations shown in Table 4.

plant; nor can PWR and BWR results be combined into a meaningful single frequency for all plants. The frequencies must be analyzed separately because the accident sequences and release categories used to estimate core-melt frequency and public risk are different for each plant type and each pipe-diameter range.

Core-Melt Frequency

These initiating event frequencies were combined with WASH-1400 accident sequences [10] for large, small, and small-small LOCAs to estimate the change in core-melt frequency, ΔF, resulting from implementation of the proposed action.

For illustration purposes, the relevant accident sequences leading to core melt for a large LOCA inside PWR containment with their /py frequencies are as follows (see Table 6):

AB with frequency = 1.2E-9/py
ACD with frequency = 6.0E-11/py
AG with frequency = 9.1E-9/py
AHF with frequency = 1.2E-10/py
AD with frequency = 2.0E-6/py
AH with frequency = 1.0E-6/py
AF with frequency = 1.0E-8/py
ADF with frequency = 2.0E-10/py

The overall probability of core melt caused by a large LOCA is, therefore, 3.1E-6/py. WASH-1400 [10] assumes a large LOCA frequency of 1.0E-4/py; therefore, the conditional probability of core melt is 0.031 per large LOCA event.

Similarly, the overall WASH-1400 probabilities of core melt caused by a small LOCA and a small-small LOCA are, respectively, 6.1E-6/py and 1.7E-5/py (Table 6). Here WASH-1400 assumes a small LOCA frequency of 3.4E-4/py and a small-small LOCA frequency of 1.0E-3/py (WASH-1400, Appendix III). Thus, the conditional probabilities of core melt caused by small or small-small LOCA are about 0.018 and 0.017, respectively.

Multiplying these conditional probabilities of core melt by the large, small, and small-small pipe failure initiating frequencies from Table 5 yields best-estimate core-melt frequencies of:

- $D > 6$ in. 7.8E-16
- 2 in. $< D \leq 6$ in. 5.3E-14
- 1/2 in. $\leq D \leq 2$ in. 5.5E-13

for a total best-estimate core-melt frequency of 5.5E-13/py for PWR plants.

The estimated frequencies of core melt resulting from large, small, and small-small pipe failures in BWR plants (caused by whipping pipes) are

TABLE 6. Accident Sequences and Release Categories Used in Pipe-to-Pipe Impact Risk Analysis (PWR Plants)

LARGE LOCA ACCIDENT SEQUENCES AND RELEASE CATEGORY FREQUENCIES (D > 6 in.)

WASH-1400 Accident Sequence	Release Category PWR-1	PWR-2	PWR-3	PWR-4	PWR-5	PWR-6	PWR-7	Accident Sequence Frequency
AB	1.0E-11	1.4E-10				1.0E-9		1.2E-9
ACD	5.0E-11			1.0E-11				6.0E-11
AG	9.0E-11		9.0E-9					9.0E-9
AHF		2.0E-11				1.0E-10		1.2E-10
AD			2.0E-8		4.0E-9		2.0E-6	2.0E-6
AH			1.0E-8		3.0E-9		1.0E-6	1.0E-6
AF	1.0E-10		1.0E-8					1.0E-8
ADF						2.0E-10		1.0E-10
TOTALS	2.5E-10	1.6E-10	4.9E-8	1.0E-11	7.0E-9	1.3E-9	3.0E-6	3.0E-6

WASH-1400 S1LOCA frequency, /py: 1.0E-4
LLOCA core-melt frequency, /py: 3.1E-6
Conditional core-melt probability: 0.031

SMALL LOCA (S1) ACCIDENT SEQUENCES AND RELEASE CATEGORY FREQUENCIES (2 in. < D ≤ 6 in.)

WASH-1400 Accident Sequence	Release Category PWR-1	PWR-2	PWR-3	PWR-4	PWR-5	PWR-6	PWR-7	Accident Sequence Frequency
S1B	3.0E-11	5.0E-10				2.0E-9		2.5E-9
S1CD	7.0E-11			1.0E-11				8.0E-11
S1F	3.0E-10		3.0E-8					3.0E-8
S1G	3.0E-10		3.0E-8					3.0E-8
S1HF		6.0E-11			4.0E-10			4.6E-10
S1D			3.0E-8		6.0E-9		3.0E-6	3.0E-6
S1H			3.0E-8		5.0E-9		3.0E-6	3.0E-6
S1DF						3.0E-10		3.0E-10
TOTALS	7.0E-10	5.6E-10	1.2E-7	1.0E-11	1.1E-8	2.3E-9	6.0E-6	6.1E-6

WASH-1400 S1LOCA frequency, /py: 3.4E-4
S1LOCA core-melt frequency, /py: 6.1E-6
Conditional core-melt probability: 0.018

TABLE 6. (Contd)

SMALL LOCA (S1) ACCIDENT SEQUENCES AND RELEASE CATEGORY FREQUENCIES (1/2 in. ≤ D ≤ 2 in.)

WASH-1400 Accident Sequence	Release Category PWR-1	PWR-2	PWR-3	PWR-4	PWR-5	PWR-6	PWR-7	Accident Sequence Frequency
S2B	1.0E-10	1.4E-9				8.0E-9		9.5E-9
S2F	1.0E-9		1.0E-7					1.0E-7
S2CD	1.0E-10					2.0E-8		2.0E-8
S2G	9.0E-10		9.0E-8					9.1E-8
S2C	2.0E-9		2.0E-6					2.0E-6
S2HF		2.0E-10				1.0E-9		1.2E-9
S2D			9.0E-8		2.0E-8		9.0E-6	9.1E-6
S2H			6.0E-8		1.0E-8		6.0E-6	6.1E-6
S2DG				1.0E-12				1.0E-12
TOTALS	2.2E-8	1.6E-9	2.3E-6	1.0E-12	3.0E-8	2.9E-8	1.5E-5	1.7E-5

WASH-1400 S2LOCA frequency, /py: 1.0E-3
S2LOCA core-melt frequency, /py: 1.7E-5
Conditional core-melt probability: 0.017

similarly derived from the WASH-1400 accident sequences in Table 7. Combining these with the appropriate estimated BWR pipe failure probabilities yields best-estimate values of:

- D > 6 in. 6.1E-17
- 2 in. < D ≤ 6 in. 8.4E-16
- 1/2 in. ≤ D ≤ 2 in. 0

for a total best-estimate core-melt frequency of 9.0E-16/py.

Summing the results for PWR and BWR plants yields the following core-melt frequencies associated with the proposed action:

best estimate = 5.5E-13
high estimate = 1.8E-9
low estimate = 2.6E-15

where the high and low estimates reflect the high and low estimates of large, small, and small-small pipe failure frequencies previously derived.

Public Health Risk

Public risks were estimated for the proposed action by assuming that the WASH-1400 release sequences [10] for large, small, and small-small pipe failures inside the PWR and BWR containments apply. To illustrate, the dominant large LOCA release sequences for PWR plants (WASH-1400, Table V.3-14, Appendix V) are:

AB-α	(PWR-1)	with frequency	= 1E-11/py
ACD-α	(PWR-1)	with frequency	= 5E-11/py
AG-α	(PWR-1)	with frequency	= 9E-11/py
AF-α	(PWR-1)	with frequency	= 1E-10/py
AB-γ	(PWR-2)	with frequency	= 1E-10/py
AB-δ	(PWR-2)	with frequency	= 4E-11/py
AHF-γ	(PWR-2)	with frequency	= 2E-11/py
AD-α	(PWR-3)	with frequency	= 2E-8/py
AH-α	(PWR-3)	with frequency	= 1E-8/py
AF-δ	(PWR-3)	with frequency	= 1E-8/py
AG-δ	(PWR-3)	with frequency	= 9E-9/py
ACD-β	(PWR-4)	with frequency	= 1E-11/py
AD-β	(PWR-5)	with frequency	= 4E-9/py
AH-β	(PWR-5)	with frequency	= 3E-9/py
AB-ϵ	(PWR-6)	with frequency	= 1E-9/py
AHF-ϵ	(PWR-6)	with frequency	= 2E-10/py
ADF-ϵ	(PWR-6)	with frequency	= 2E-10/py
AD-ϵ	(PWR-7)	with frequency	= 2E-6/py
AH-ϵ	(PWR-7)	with frequency	= 1E-6/py

WASH-1400 assumes a medium-large LOCA frequency of 1.0E-4/py. Replacing this value with the previously estimated probability of a large pipe break (2.5E-14/py, Table 5) and using the WASH-1400 dose estimates for each release category results in the following best estimate of risk from the occurrence of at least one large pipe failure caused by a whipping pipe:

$$\begin{aligned}\text{Risk} &= (6.3\text{E-}20/\text{py})\ (5.4\text{E+}6\ \text{man-rem}) + (4.0\text{E-}20/\text{py})\ (4.8\text{E+}6\ \text{man-rem})\\ &+ (1.2\text{E-}17/\text{py})\ (5.4\text{E+}6\ \text{man-rem}) + (2.5\text{E-}21/\text{py})\ (2.7\text{E+}6\ \text{man-rem})\\ &+ (1.8\text{E-}18/\text{py})\ (1.0\text{E+}6\ \text{man-rem}) + (3.3\text{E-}19/\text{py})\ (1.5\text{E+}5\ \text{man-rem})\\ &+ (7.6\text{E-}16/\text{py})\ (2.3\text{E} + 6\ \text{man-rem})\\ &= 7.1\text{E-}11\ \text{man-rem/py}.\end{aligned}$$

The best estimates of public health or risk associated with small and small-small pipe failures in PWR piping are similarly developed using the small and small-small LOCA release frequencies, respectively, from WASH-1400 (Table 6). As for estimating core-melt frequency, the WASH-1400 small and small-small LOCA frequencies (3.4E-4/py and 1.0E-3/py, respectively) were replaced in the present analysis by the appropriate pipe failure frequencies, Table 5.

TABLE 7. Accident Sequences and Release Categories Used in Pipe-to-Pipe Impact Risk Analysis (BWR Plants)

LARGE LOCA ACCIDENT SEQUENCES AND RELEASE CATEGORY FREQUENCIES (D > 6 in.)

WASH-1400 Accident Sequence	Release Category BWR-1	BWR-2	BWR-3	BWR-4	Accident Sequence Frequency
AE	2.0E-9	4.0E-8	1.0E-7		1.4E-7
AJ	1.0E-10	1.0E-9	1.0E-8		1.2E-8
AHI	1.0E-10	2.0E-9	1.0E-8		1.2E-8
AI	1.0E-10	2.0E-9	1.0E-8		1.2E-8
AGJ				6.0E-11	6.0E-11
AEG				7.0E-10	7.0E-10
AGHI				6.0E-11	6.0E-11
TOTALS	2.3E-9	4.6E-8	1.3E-7	8.2E-10	1.8E-7

WASH-1400 LLOCA frequency, /py: 1.0E-4
LLOCA core-melt frequency, /py: 1.8E-7
Conditional core-melt probability: 0.0018

SMALL LOCA (S1) ACCIDENT SEQUENCES AND RELEASE CATEGORY FREQUENCIES (2 in. < D ≤ 6 in.)

WASH-1400 Accident Sequence	Release Category BWR-1	BWR-2	BWR-3	BWR-4	Accident Sequence Frequency
S1E	2.0E-9	8.0E-8	1.0E-7		1.8E-7
S1J	3.0E-10	7.0E-9	3.0E-8		3.7E-8
S1I	4.0E-10	7.0E-9	4.0E-8		4.7E-8
S1HI	4.0E-10	6.0E-9	2.0E-8		2.6E-8
S1C				2.0E-10	2.0E-10
S1GJ				2.0E-10	2.0E-10
S1GI				1.0E-10	1.0E-10
S1EG				2.0E-10	2.0E-10
S1GHI				2.0E-10	2.0E-10
TOTALS	3.1E-9	1.0E-7	1.9E-7	9.0E-10	2.9E-7

WASH-1400 S1LOCA frequency, /py: 3.4E-4
S1LOCA core-melt frequency, /py: 2.9E-7
Conditional core-melt probability: 0.00086

TABLE 7. (Contd)

SMALL-SMALL LOCA (S2) ACCIDENT SEQUENCES AND RELEASE CATEGORY FREQUENCIES (1/2 in. $\leq$ D $\leq$ 2 in.)

WASH-1400 Accident Sequence	Release Category BWR-1	BWR-2	BWR-3	BWR-4	Accident Sequence Frequency
S2J	1.0E-9	2.0E-8	8.0E-8		1.0E-7
S2I	1.0E-9	2.0E-8	9.0E-8		1.1E-7
S2HI	1.0E-9	2.0E-8	9.0E-8		1.1E-7
S2E	5.0E-10	1.4E-8	4.0E-8		5.5E-8
S2C			8.0E-9		8.0E-9
S2CG				6.0E-11	6.0E-11
S2GHI				6.0E-10	6.0E-10
S2EG				3.0E-10	3.0E-10
S2GJ				6.0E-10	6.0E-10
S2GI				2.0E-10	2.0E-10
TOTALS	3.5E-9	7.4E-8	3.1E-7	1.8E-9	3.9E-7

WASH-1400 S2LOCA frequency, /py: 1.0E-3
S2LOCA core-melt frequency, /py: 3.9E-7
Conditional core-melt probability: 0.00039

The best estimate of per-plant risk due to any pipe failure caused by a whipping pipe of equal size and wall thickness becomes:

Risk = 7.1E-11 man-rem (D > 6 in.)

+ 5.9E-9 man-rem (2 in. < D < 6 in.)

+ 3.7E-7 man-rem (2 in. < D $\leq$ 1/2 in.)

= 3.8E-7 man-rem/py.

Multiplying this result by the total remaining lifetime of PWR plants (+ 2370 py) results in a best-estimate total public risk decrease from the proposed action of 9.0E-4 man-rem.

Equivalent results were obtained for BWR plants by combining the estimated BWR DEGB frequencies and pipe-impact failure frequencies with the appropriate BWR release sequences from WASH-1400, Table 7. The resultant best estimate for risk is 5.2E-9 man-rem/py; when multiplied by the total remaining lifetime of BWR plants (1250 py) yields a total public risk of 6.6E-6 man-rem from damaged pipes resulting from whipping pipes of equal diameter and wall thickness.

Summing the results for PWR and BWR plants, therefore, yields the following incremental decreases in total public risk resulting from the proposed action:

best estimate = 9.0E-4 man-rem

high estimate = 1.6E-0 man-rem

low estimate = 4.3E-6 man-rem.

This analysis indicates a decrease in the estimate of risk to public health and safety. Note that the high estimate of risk reflects both NRC minimum reliability guidelines (i.e., system failure probability on the order of 1E-6/yr) and the added protection provided by an estimated 1200 restraints in PWR plants and an additional estimated 330 restraints in BWR plants.

OCCUPATIONAL RADIATION EXPOSURE - ACCIDENTAL

The decreased ORE from accidents can be estimated as the product of the change in total core-melt frequency and the ORE likely to occur in the event of a major accident. The nominal reduction in core-melt frequency was estimated to be 5.5E-13/py. The ORE in the event of a major accident has two components [8]. The first is the immediate exposure to personnel onsite during the time of the accident and its short-term control. The second is the long-term exposure associated with cleanup and recovery from the accident. The incremental ORE to an accident is calculated as follows:

$$D(TOA) = NTD(OA)$$

$$D(OA) = (\Delta F)[D(IO) + D(TO)]$$

where: D(TOA) = total accidental occupational dose
N = number of affected facilities
T = average remaining lifetime, py
D(OA) = accidental occupational dose, py
ΔF = change in core-melt frequency
D(IO) = immediate occupational dose
D(TO) = long-term occupational dose.

Table 8 presents the resultant ORE caused by accidents, based on cleanup and decommissioning estimates given in NUREG/CR-2800 [15].

OCCUPATIONAL RADIATION EXPOSURE - ROUTINE

Additional routine occupational radiation exposure (ORE) is expected to be experienced by the industry as a result of incorporating the proposed action. This additional exposure will result from a need to perform confirming walk-down inspections as part of a re-analysis of pipe breaks in containment, from the installation of any required restraints, and from the inspection and maintenance of added restraints. In terms of estimating jet impingement forces, the proposed action will not result in the addition of jet impingement barriers or jet shields. This is caused by the less conservative nature of

TABLE 8. Occupational Radiation Exposure Caused by Accidents

	Best Estimate	High Estimate	Low Estimate
Increase in core-melt frequency	5.5E-13	1.8E-9	2.6E-15
Immediate dose (man-rem/event)	1.0E+3	4.0E+3	0
Long-term dose (man-rem/event)	2.0E+4	3.0E+4	1.0E+4
Total exposure (man-rem)	1.3E-3	6.8	2.9E-6

the revised methods and criteria for determining jet impingement forces proposed for introduction into the SRP. That is, the revised methods will generally lead to smaller estimated jet loads on essential systems. Although the conservatism of the proposed jet impingement methods and criteria will lead operating plant managers to the conclusion that existing jet shields are generally over-designed, they are not expected to replace old shields with lighter ones.

One exception to the conservative nature of the newer methods for estimating jet impingement forces stems from their prediction, in general, of slightly wider jet plumes. A re-analysis of stable breaks may, therefore, lead plant managers to conclude that shields designed by the old 10-degree recipe are undersized in shield area by a small margin. Additional ORE would, therefore, be expected from confirmatory inspections (walk-downs); and if re-analysis dictates, the installation of larger or modified shields would be necessary. Because data on the number of jet impingement shields that would be affected were not readily available from the industry, values and impacts were not estimated for this effect. However, because shields are not being removed, only slightly modified in a few cases, values and impacts are not expected to be comparable to those estimated for the addition of pipe-whip restraints as discussed below.

Incorporating the more conservative approach of analyzing unstable breaks involving interaction with adjacent piping of equal diameter and wall thickness will result in additional routine ORE for the industry. The additional ORE will result from the following new activities:

- confirmatory inspections in containment (walk-downs)
- installation of needed restraints inside containment
- in-service inspection (ISI) of the added restraints
- routine restraint maintenance
- restraint gap verification.

Quantitative estimates of routine ORE are presented only for the addition of pipe-whip restraints. Because only restraints added inside containment would significantly contribute to personnel radiation exposure, restraints that may be added outside containment were not considered.

Table 9 summarizes the sources of routine ORE considered in this evaluation; it is not, however, necessarily intended as an exhaustive list of all sources of added ORE resulting from the proposed action. The specific activities described below and their associated dose estimates represent a composite of information documented in past LLNL value-impact assessments [8,9]. Note that dose values are best estimates; high and low estimates in Table 9 reflect a ±50% uncertainty range, except where noted otherwise.

In-Service Inspection of Piping Welds

Pipe-whip restraints restrict access to piping welds for routine ISI. As a result, personnel exposure is increased because the restraints must often be removed and then reinstalled to perform ISI. Even when restraints are specifically designed so that ISI can be performed without their removal (as is the case for several later-generation plants), their presence still reduces efficiency and, therefore, causes workers to remain longer in high radiation areas.

Based on dose values used in the LLNL GDC-4 evaluation [9], it was assumed here that each added restraint would increase exposure (due to either cause) by 1.0 man-rem per ISI. Assuming a 10-yr inspection interval implies that about 4.0 man-rem would be experienced per excluded restraint over a 40-yr plant lifetime.

Routine Restraint Maintenance

It is anticipated that restraints will be visually inspected once every 5 yr, resulting in 0.5 man-rem per restraint total exposure over the 40-yr plant lifetime based on the experience of one plant owner providing input to our study.

Restraint Gap Verification

It is anticipated that restraint gaps will be verified every 10 yr. Assuming that exposure averages 0.1 man-rem per verification implies an added dose of 0.4 man-rem per restraint over a 40-yr plant lifetime [13].

In addition to these specific activities, the addition of pipe-whip restraints would add congestion inside of containment and generally reduce the efficiency of maintenance activities. The associated ORE increases are difficult to quantify, being highly sensitive to plant-specific variations in the type and frequency of maintenance activities performed. Consequently, added ORE associated with increased plant congestion was left unquantified in this study.

TABLE 9. Summary of Plant Routine Activities Affected by the Addition of Pipe-Whip Restraints with Related Radiation Dose Estimates (man-rem/device)

	Operating Plants			Construction Plants		
Routine Activity	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
1. Plant walk-downs for re-analysis	0.05	0.1	0.01	0.05	0.1	0.01
2. Restraint installation	4	6	2	NA	NA	NA
3. In-service inspection	4	6	2	4	6	2
4. Restraint maintenance	0.2	0.3	0.1	0.2	0.3	0.1
5. Gap verification	0.4	0.6	0.2	0.4	0.6	0.2
Total added ORE (man-rem/device)	8.7	13.1	4.3	4.7	7.1	2.3

NOTES:

1. All values cumulative over a 40-yr plant life.
2. Values given are for pipe-whip restraints only.

Increased plant congestion would also increase personnel exposure in certain "non-routine" situations, such as recovery from unusual plant conditions. In the event of a radioactive release or spill, for example, decontamination operations would be less effective--and personnel exposure greater--if protective device support structures, with their complex shapes, were added. Access for fire control in certain plant areas would also be hindered by increasing the number of pipe restraints.

Development of Overall Added ORE

Separate per-plant estimates of added ORE for PWR and BWR plants were determined by combining the per-restraint dose estimates described above with the respective number of pipe-whip restraints added inside containment only (Table 10). These results were then multiplied by the number of affected PWR and BWR plants to obtain the following overall estimates of avoided ORE resulting from the proposed action:

best estimate = -6E+3 man-rem

high estimate = -2E+4 man-rem

low estimate = -1E+3 man-rem

where the high and low estimates reflect uncertainty in both the per-restraint dose values (±50%) and in the number of added restraints. Note also that in

TABLE 10. Overall Added Routine Occupational Radiation Exposure

	PWR Plants			BWR Plants		
	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
Added Routine ORE						
Added restraints[a] (in containment)	9	16	4	5	8	1
Operating plants (man-rem/plant)	58	150	13	32	78	3.2
Construction plants (man-rem/plant)	78	210	17	43	100	4.3

Affected Plants	PWR Plants	BWR Plants	All Plants
Operating plants	67	38	105
Construction plants	10	3	13
Average remaining life (operating plants only)	29.5	29.8	29.6
Total added routine ORE (man-rem):			
Best estimate	-4.6E+3	-1.4E+3	-6.0E+3
High estimate	-1.2E+4	-3.3E+3	-1.6E+4
Low estimate	-1.0E+3	-1.4E+2	-1.2E+3

(a) Assumes one pipe restraint for each unstable break that requires restraining.

developing the overall avoided ORE that the per-plant values for operating plants were prorated by the ratio of the number of remaining plant-years to total plant life (≈32/40).

OFFSITE ECONOMIC RISK

The effect of the proposed action on offsite economic risk is calculated by multiplying the change in accident frequency by a generic estimate of offsite accident costs. For severe accidents that result in a substantial release of radioactive material to the environment, the offsite impacts would include health effects as well as the radiologically induced economic costs of taking population protection measures such as evacuation and relocation, agricultural product disposal, decontamination of land and tangible property, and land interdiction. The cost estimates used in this regulatory analysis are those presented in the NRC Safety Goal Evaluation Report [16]. As described in NUREG/CR-3673 [17], these costs were developed by Sandia Laboratories from the results of calculations made with new economic models developed for the MELCOR series of risk assessment codes. The best-estimate costs follow those established by LLNL in an earlier value-impact assessment [8] and presume an major accident release for an 1100-MWe PWR plant (Palisades) with a typical surrounding population density; upper- and lower-bound cost estimates were developed for similar plants located in areas of high- and low-population density (Indian Point and Palo Verde, respectively). The original cost estimates, expressed in 1982 dollars, are presented in Table 11; note that these do not include litigation costs, impacts to areas receiving evacuees or institutional costs. In the current analysis, the costs presented in the Safety Goal Evaluation Report have been adjusted to 1988 dollars by assuming a 4% real discount rate in accordance with the NRC Safety Goal Policy [18].

Following standard NRC regulatory analysis guidelines, estimated costs (in 1988 dollars) are converted to present value by discounting at 10%; a 5% discount rate is also considered as a sensitivity case. The following discounting formula is employed:

$$\frac{D}{V} = \frac{e^{-It_i} - e^{-It_f}}{I}$$

where: D = discounted value
V = estimated accident costs (in 1988 dollars)
t_i = years to reactor operation; 0 for operating and construction plants
t_f = years remaining until end of plant life
I = discount rate.

For purposes of this assessment, no distinction is made between operating and planned reactors; the average remaining life of the total population of 118 reactors is 30.7 yr. The 10% discount factor is therefore 9.54, the 5% discount factor 15.7. The offsite economic risk, V_{FP}, is estimated as:

$$V_{FP} = N(\Delta F)D$$

where N and ΔF are the number of reactors and the change in core-melt frequency, respectively. The results are summarized in Table 11; upper and lower bounds are cost estimates developed in Ref. 32 for Indian Point and Palo Verde coupled with the bounds on the core-melt frequency.

ONSITE PROPERTY RISK

The effect of the proposed action on the risk to onsite property is estimated by multiplying the change in accident frequency by a generic estimate of onsite accident costs. The cost estimates used in this regulatory analysis are those presented in the NRC Safety Goal Evaluation Report that were derived from NUREG/CR-3673 [17] for an SST1 release (major accident). These costs include onsite cleanup and decontamination, replacement power, facility repair and restoration, and the capital cost of damaged plant equipment. In the event of a major accident causing plant contamination, immediate decommissioning would offer an alternative to plant repair and restoration; in this case, incurring related costs sooner than anticipated would result in real costs because of present value considerations.

TABLE 11. Reduced Offsite Economic Risk

5% discount factor:	15.7	(applied to present dollars)
10% discount factor:	9.54	
Cost adjustment factor, 4%:	1.27	(present value of 1982 dollars)

	Best Estimate	High Estimate	Low Estimate
Change in core-melt frequency (/py)	5.5E-13	1.8E-9	2.6E-15
Offsite costs ($/event)[a]			
• Economic costs	-1.1E+9	-1.0E+10	-5.0E+8
• Health costs	-2.0E+8	-2.0E+9	-1.0E+8
Total offsite costs (1982 dollars)	-1.3E+9	-1.2E+10	-6.0E+8
Total offsite costs (1988 dollars)	-1.6E+9	-1.5E+10	-7.6E+8
Discounted offsite costs ($/event)			
• 5% discount rate	-2.6E+10	-2.4E+11	-1.2E+10
• 10% discount rate	-1.6E+10	-1.5E+11	-7.2E+9
Total offsite economic risk ($)			
• 5% discount rate	-1.7E+0	-5.1E+4	-3.7E-3
• 10% discount rate	-1.0E+0	-3.1E+4	-2.3E-3

(a) From U.S. NRC Safety Goal Evaluation Report (April 1985).

The original cost estimates presented in the Safety Goal Evaluation Report, expressed in 1982 dollars, are given in Table 12. Costs not included are utility "business" costs, nuclear power industry costs, and costs for onsite litigation. Utility business costs are those that might result from altered risk perceptions in the financial markets combined with the need for the plant licensee to replace the income once generated by the operating plant. Nuclear power industry costs are those associated with elimination of or a slow down in nuclear industry growth. Litigation costs would include damage awards and associated legal fees.

As with offsite costs, the current analysis adjusts the original onsite costs from 1982 to 1988 dollars by assuming a 4% real discount rate. The adjusted costs are then discounted using the following formula:

$$\frac{D}{V} = \frac{1}{mI^2} \{e^{-It_i}\} \left\{(1 - e^{-Im}) \left[1 - e^{-I(t_f - t_i)}\right]\right\}$$

TABLE 12. Reduced Onsite Economic Risk

Recovery time (yr):	10
5% discount factor:	12.4 (applied to present dollar)
10% discount factor:	6.0
Cost adjustment factor, 4%:	1.27 (present value of 1982 dollars)

	Best Estimate	High Estimate	Low Estimate
Core-melt frequency	5.5E-13	1.8E-9	2.6E-15
Onsite costs ($/event)[a]			
• Onsite cleanup costs	-1.7E+9	-2.5E+9	-8.0E+8
• Replacement power costs	-1.7E+9	-2.5E+9	-8.0E+8
• Pro-rata facility costs	-3.0E+8	-5.0E+8	-2.0E+8
Total onsite costs (1982 dollars)	-3.7E+9	-5.5E+9	-1.8E+9
Total onsite costs (1988 dollars)	-4.7E+9	-7.0E+9	-2.3E+9
Discounted onsite costs ($/event)			
• 5% discount rate	-5.8E+10	-8.6E+10	-2.8E+10
• 10% discount rate	-2.8E+10	-4.2E+10	-1.4E+10
Total onsite economic risk ($)			
• 5% discount rate	-3.8E+0	-1.8E+4	-8.8E-3
• 10% discount rate	-1.9E+0	-9.0E+3	-4.3E-3

(a) Best-estimate values from U.S. NRC Safety Goal Evaluation Report (April 1985). High and low estimates assume a ±50% uncertainty.

where: D = discounted value
V = estimated accident costs (in 1985 dollars)
m = years required for plant recovery (= 10 yr)
t_i = years to reactor operation; zero for operating plants
t_f = years remaining until end of plant life
I = discount rate.

For purposes of this assessment, no distinction is made between operating and planned reactors. The 10% discount factor is therefore 6.0; the 5% discount factor 12.4. The risk to onsite property, V_{OP}, is estimated as:

$$V_{OP} = N(\Delta F)U$$

where: U = per-reactor onsite costs (1985 dollars)
ΔF = change in accident frequency
N = number of affected facilities.

The results are summarized in Table 12; the uncertainty bounds reflect a ±50% spread in the generic property onsite cost estimate coupled with the bounds on core-melt frequency.

INDUSTRY IMPLEMENTATION COSTS

Significant costs to the industry are anticipated as a result of the proposed action, primarily for the added costs associated with the re-analysis of pipe breaks and the subsequent addition of pipe-whip restraints. Table 13 presents a breakdown of the estimated costs[a] (in 1988 dollars) assumed applicable to pipe-whip restraints. Cost items applicable to operating plants and plants under construction include the following items:

- Design engineering--All engineering costs for device design, including civil engineering, drafting, and field follow costs during construction.

- Hazard engineering--Analysis costs for determining postulated unstable break locations, evaluating pipe-whip loads, and target response. These costs include iterative analysis costs to redefine break points in the event; for example, that field interferences cause piping to be rerouted. They do not include non-mechanistic analysis of environmental effects (pressure, temperature, humidity) caused by pipe break.

- Other manpower costs--Quality assurance follows during construction, miscellaneous manpower, and paper costs.

- Hardware and fabrication--Device fabrication, including material and other hardware costs.

- Installation--Device installation.

The per-device implementation costs for operating and plants being constructed were assumed to be equal for design engineering, QA, and materials and fabrication. However, for operating plants, costs for hazards engineering and installation were considered to be twice those for plants under construction.

Development of Overall Added Implementation Costs

Per-plant implementation costs were developed by multiplying the per-restraint estimates by the number of added pipe-whip restraints (Table 14). These results were then multiplied by the number of affected PWR and BWR plants to obtain the following estimates of total costs:

best estimate = \$80 million

high estimate = \$200 million

low estimate = \$20 million

(a) Costs in 1982 dollars were obtained from a previous LLNL value-impact study [8].

TABLE 13. Summary of Implementation Costs for Pipe-Whip Restraints ($K/device)

Cost Item	Operating Plants			Construction Plants		
	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
1. Design engineering	7.9	11.0	5.6	7.9	11.0	5.6
2. Hazard engineering	45.0	59.0	38.0	23.0	29.0	19.0
3. QA, design follow, miscellaneous	2.3	4.5	1.1	2.3	4.5	1.1
4. Materials and fabrication	10.0	14.0	9.0	10.0	14.0	9.0
5. Installation	23.0	34.0	18.0	11.0	17.0	9.0
Total costs ($K/device)	88.0	120.0	72.0	54.0	75.0	44.0

NOTES:

1. All costs in 1988 dollars.

where the high and low estimates reflect not only the high and low cost estimates from Table 13, but the high and low estimates in number of added restraints as well.

As noted earlier, industry implementation costs have been estimated on the basis of generic per-device costs assumed applicable to all pipe-whip restraints. In reality, the cost of pipe-whip restraints can vary widely depending on size, complexity, and the operating characteristics of the piping system with which it is associated. Table 15 presents a LLNL summary [8] of actual pipe-whip restraint costs for a sample PWR plant, all for high-energy piping systems. Note that even restraints for small-diameter piping can cost much more than the approximately $50K per device assumed in developing the implementation costs in Table 13.

Use of Undiscounted Costs

Generally, if costs are anticipated at some time in the future and are planned for by, for example, establishment of suitable contingency funds, then these costs should be present-valued at an appropriate discount rate. If these anticipated costs are later avoided through regulation changes, clearly their present value--and not their undiscounted value--reflects the true financial benefit to a plant owner. However, undiscounted implementation costs were used in this analysis for the following reasons.

- Construction plants would be expected to immediately implement the proposed action through such measures as installing pipe-whip restraints at unstable break locations. Therefore, because little time would elapse between regulation action and implementation, it seems most appropriate to use undiscounted costs.

TABLE 14. Overall Added Implementation Costs

	PWR Plants			BWR Plants		
	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
Added restraints	9	16	4	5	8	1
Implementation costs ($/plant)						
• Operating plants	7.9E+5	1.9E+6	2.9E+5	4.4E+5	9.7E+5	7.2E+4
• Construction plants	4.9E+5	1.2E+6	1.8E+5	2.7E+5	6.0E+5	4.4E+4

Affected Plants	PWR Plants	BWR Plants	All Plants
• Operating plants	67	38	105
• Construction plants	10	3	13
Overall Implementation Costs			
• Best estimate	5.8E+7	1.8E+7	7.5E+7
• High estimate	1.4E+8	3.9E+7	1.8E+8
• Low estimate	2.1E+7	2.9E+6	2.4E+7

NOTES:
1. All costs given in 1988 dollars.

TABLE 15. Summary of Actual Pipe-Whip Restraint Costs for a Sample PWR Plant ($K/restraint)

Pipe Diameter	No. of Restraints	Average Cost	High Cost	Low Cost
2 in.	3	57.6	79.3	39.3
3 in.	4	80.3	126.0	31.4
4 in.	25	77.7	102.0	36.1
6 in.	17	44.8	80.5	17.2
12 in.	2	96.2	100.0	92.0
16 in.	5	27.1	37.6	18.1
32 in.	3	124.0	168.0	95.8

- Likewise, operating plants would be expected to implement the proposed action after several months of re-analysis.

INDUSTRY OPERATING COSTS

The proposed action would result in the industry adding a number of pipe-whip restraints now not required. This would result in additional operating costs due to additional ISI, routine restraint maintenance, and restraint gap verification. Costs for ISI outweigh the other two costs because of the requirement to remove and replace restraints blocking access to welds.

Table 16 itemizes specific maintenance activities considered in this evaluation that would be affected by the installation of additional pipe-whip restraints; it is not, however, necessarily intended as an exhaustive list of all costs associated with the proposed action. The specific activities described below and their associated cost estimates represent a composite of information provided by LLNL in earlier value-impact studies [8]. Note that the following cost values are best estimates in 1988 dollars; high and low estimates in Table 16 reflect a +100%, -50% uncertainty range, respectively.

In-Service Inspection of Welds

Based on industry estimates presented in earlier LLNL value-impact studies [8,9], it was assumed in this study that each added restraint would increase costs by $1K per ISI. Assuming a 10-yr inspection interval implies that each restraint would cost about $4.5K over a 40-yr plant lifetime.

This estimate represents only those direct costs for weld inspection, and does not include ancillary costs associated with restraint removal. Depending on utility practice, removal may also involve tagging each component of the restraint, storage in a controlled warehouse, and then retrieval for

TABLE 16. Summary of Plant Maintenance Activities Affected by the Addition of Pipe-Whip Restraints ($K/device)

	Operating Plants			Construction Plants		
Maintenance Activity	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
1. In-service inspection	4.5	9.0	2.3	Same as operating plants		
2. Routine restraint maintenance	0.56	1.1	0.34	"	"	"
3. Gap verification	0.90	1.8	0.45	"	"	"
Total added costs ($K/device)	6.0	12.0	3.0	"	"	"

NOTES:
1. All values are in constant 1988 dollars and are cumulative over a 40-yr plant life.

installation. These activities may end up being more costly than examination of the weld.

Routine Restraint Maintenance

This study has assumed that a restraint will be visually inspected once every 5 yr, resulting in a total cost of $0.56K per restraint over the 40-yr plant lifetime based on the experience of one plant owner providing input to an earlier LLNL value-impact study [8].

Restraint Gap Verification

It is anticipated that restraint gaps will be verified every 10 yr. Assuming an average cost of $0.2K per verification implies a total cost of $0.9K per restraint over a 40-yr plant lifetime [9].

In addition to these specific activities, the addition of pipe-whip restraints would increase congestion inside of containment and generally reduce the efficiency of maintenance activities. The associated costs are difficult to quantify, being highly sensitive to plant-specific variations in the type and frequency of maintenance activities performed. Consequently, costs resulting from increased plant congestion were left unquantified in this study.

The cost figures in Table 16 apply both to operating plants and plants under construction when figured over a 40-yr plant lifetime.

Development of Overall Added Operating Costs

Separate per-plant estimates of avoided operating costs for PWR and BWR plants were developed by combining the per-restraint cost estimates described above with the respective number of pipe-whip restraints added inside containment only (Table 17). These results were then multiplied by the number of affected PWR and BWR plants to obtain the following overall estimates of added operating costs resulting from the proposed action:

best estimate = $ 4 million

high estimate = $14 million

low estimate = $0.8 million

where the high and low estimates reflect the uncertainty in both the per-restraint costs (±50%) and in the number of added restraints. Note also that in developing the overall operating costs, the per-plant costs for operating plants were prorated by the ratio of the number of remaining plant-years to total plant life (≈30/40).

Note that the overall industry operating costs is about a factor of 10 less than the industry implementation cost. This is because of the high implementation costs associated with re-analysis and restraint installation

TABLE 17. Overall Added Operating Costs

	PWR Plants Best Estimate	PWR Plants High Estimate	PWR Plants Low Estimate	BWR Plants Best Estimate	BWR Plants High Estimate	BWR Plants Low Estimate
Restraint Costs						
• Restraints added	9.0	16.0	4.0	5.0	8.0	1.0
• Operating plants ($K/plant)	54.0	190.0	12.0	30.0	95.0	3.0
• Construction plants ($K/plant)	54.0	190.0	12.0	30.0	95.0	3.0

Affected Plants	PWR Plants	BWR Plants	Total Plants
• Operating plants	67.0	38.0	105.0
• Construction plants	10.0	3.0	13.0
• Average remaining life (operating plants only)	29.5	29.8	29.6
Overall Added Operating Costs ($)			
• Best estimate	3.2E+6	9.3E+5	4.1E+6
• High estimate	1.1E+7	3.0E+6	1.4E+7
• Low estimate	7.2E+5	9.5E+4	8.2E+5

NOTES:

1. All values are in constant 1988 dollars and are cumulative over a 40-yr plant life.

for both operating plants and plants under construction. However, routine ISI, restraint maintenance, and gap verification are activities that must be accomplished regardless of the proposed action.

Effect of Cost Discounting

The preceding estimation of increased industry operating costs ignores present worth considerations. Because these costs continually recur over all remaining years of plant life, NRC value-impact assessment guidelines recommend that they be present-valued to account for the real opportunity cost of money. The present value of future operating costs is estimated according to the following formula:

$$PV = C_A \left[\frac{(1 + r)^t - 1}{r(1 + r)^t}\right] \left[\frac{1}{(1 + r)^n}\right]$$

where:
- PV = present value of recurring operating costs
- C_A = annual operating costs in constant dollars
- r = real discount rate
- t = annuity period over which costs recur
- n = years before regulatory action is implemented.

If per-plant operating costs average \$85K over a 40-yr lifetime, annual recurring costs average \$2.1K in 1988 constant dollars. It seems reasonable to expect that plant maintenance costs will more or less follow the inflation rate from year to year, which in turn implies that the annual costs would be the same magnitude in constant dollars; on this basis, use of the above aggregation formula is valid. The average remaining lifetime of the 118 plants is 29.6 yr; the 10% present worth factor is therefore 9.4, the 5% factor is 15.3. The total discounted operating costs for these plants are about 31.8% (= 9.4/29.6) and 51.7% (= 15.3/29.6) of the undiscounted costs, or about \$1.3 million and \$2.1 million, respectively, down from \$4 million.

The above analysis assumes that operating costs would begin to be avoided immediately upon adoption of the proposed action. If implementation were delayed, however, the present worth of the avoided costs could be further reduced because of additional discounting. Assuming that 5 yr elapse between the proposed SRP revision and its implementation by plant licensees, the total discounted operating costs would be about \$910,000 based on a 10% present worth factor and an average remaining operating life of 24.6 yr for the 118 plants affected. Note that here the remaining plant life reflects an additional 5 yr of operation for plants now operating and also 5 yr of operation (i.e., zero years average forward-fit) for plants currently under consideration.

The actual effect of discounting will, of course, depend on the financial policies of individual licensees. It could be argued, for example, that if a utility pays operating costs directly out of its annual rate base, then revenues and expenditures would be sufficiently near-term that discounting would have a negligible effect; a constant-dollar analysis might therefore most accurately reflect the true costs (or cost savings) associated with the proposed SRP revision. In any case, the overall effect of discounting is not sufficiently large to alter the basic conclusions of the regulatory analysis.

NRC DEVELOPMENT AND IMPLEMENTATION COST

Development of the proposed SRP modification is complete. However, costs associated with the following activities will have to be covered by the NRC:

- a review by the CRGR committee
- generation and sending of a generic letter or bulletin to the nuclear power industry
- review of the responses by each plant manager.

NRC OPERATING COST

No additional NRC operating costs related to the proposed action are anticipated.

SENSITIVITY STUDIES

To obtain a feeling for the sensitivity of the major decision factors to parameters such as the probability of pipe failure upon impact, the number of added restraints, and the effects of LBB, several additional cases were evaluated. The results of these additional cases are described below.

PIPE IMPACT FAILURE FREQUENCY

Because the proposed changes to the SRP were driven in part by PNL studies of pipe-to-pipe impact damage [5], the sensitivity of the major decision factors to changes in pipe impact failure frequency (Table 5) were evaluated. Because changes to this frequency directly affect core-melt frequency, all decision factors depending on this parameter were affected. That is:

- risk to the public health and safety
- accidental ORE
- offsite economic risk
- onsite economic risk

are all directly affected by changes in pipe impact failure frequency.

On the other hand:

- routine ORE
- industry implementation cost
- industry operating cost

are all independent of the value chosen for pipe impact failure frequency.

NUMBER OF ADDED RESTRAINTS

General

Sensitivity of the decision factors to the number of added restraints was evaluated. The number of added restraints affects ORE and the industrial costs, and is proportional to the number of breaks stabilized. Because the number of breaks stabilized directly affects core-melt frequency, which affects all the remaining decision factors, virtually all decision factors are affected by changes in number of added restraints.

Restraints for Specific Pipe Sizes

Although the above result seems obvious for changes to number of restraints in general, the sensitivity of the decision factors to number of restraints for a single pipe-diameter range is not. This different result is because of the dependence of the decision-factor values on different functional relationships for the LLOCA, S1LOCA, and S2LOCA pipe-diameter ranges. For example, if breaks in the LLOCA pipes of a PWR are eliminated in this study (a decrease in the "best-estimate" breaks of 33%, Table 5), the following results are obtained:

- best values for public health risk, accidental ORE, offsite economic risk, and onsite economic risk all remain essentially unchanged

- best values for routine ORE and industry implementation and operating costs decrease by approximately 25%.

The above results represent an important deviation to the general results of this study; they represent a likely result of a LBB philosophy by a plant manager in dealing with the larger-diameter coolant loop piping in a PWR [9]. That is, should the NRC formalize the proposed changes to the SRP, a plant manager will be required to perform a re-analysis of all his or her unstable breaks and add restraints where dictated by the analysis. If the reactor is a PWR type, the manager has the option to seek relief from adding new restraints by introducing the LBB philosophy to as many affected pipes as possible. At the present time, the NRC has promoted the use of the LBB philosophy for all PWR main coolant lines [19].

As pointed out in the summary report of the NRC's Piping Review Committee [19], probabilities of crack initiation and propagation in the large primary pipes of PWRs is quite low; however, it is somewhat higher for BWRs because of the increased probability for intergranular stress corrosion cracking (IGSCC). In LLNL studies funded by the NRC and performed by Science Applications Inc. (SAI) [20], probabilities of about 10E-6 for leakage and about E-12 for failure per plant lifetime using Zion (Westinghouse PWR) as the model were used. Median values for direct DEGB in Babcock & Wilcox (B&W) and Combustion Engineering (CE) plants varied somewhat, but generally were below E-10.

LLOCA and S1LOCA Breaks Eliminated for PWR

In a similar study, all PWR LLOCA (D > 6 in.) and PWR S1LOCA (2 in. < D $\leq$ 6 in.) breaks were eliminated from the base study analysis (67% decrease in number of postulated breaks or added restraints). The results indicated the following:

- best value for public health risk essentially remained the same

- best values for accidental ORE, offsite and onsite economic risks decreased by approximately 10%

- routine ORE, industry implementation, and operating costs decreased by approximately 52%.

Again, the best value of the estimated public health risk remained essentially the same and the decision factors affected most directly by the number of added restraints (routine ORE and the industry costs) were most sensitive. However, there was a slight but quantifiable decrease or sensitivity in accidental ORE and industry economic risks. These results amplify the nonlinear effects present in the value-impact model used in this study when varying non-uniformly the number of added restraints within the pipe-diameter ranges.

Leak-Before-Break Philosophy

At the present time, lethargy by plant managers to applying LBB philosophy to operating-plant piping is real. This lethargy is most likely caused by the unfamiliarity of plant managers with promoting the LBB approach combined with their tendency not to change anything in their machine that is working successfully. However, the sensitivity studies discussed above indicate that plant managers will have additional incentive to apply LBB philosophy to unstable breaks that involve interaction with adjacent piping of equal diameter and wall thickness (subject of the proposed action). Similarly, there will be incentive by plant owners to apply LBB in plants under construction. One example of using the LBB approach in a plant under construction is provided by the Duquesne Light Company and their Beaver Valley Unit 2. Recognizing the value of pipe-whip restraint removal, Duquesne proposed LBB criteria to the NRC to show that certain unit 2 restraints were not needed. NRC agreed with this conclusion in NUREG-1057 [21] and Duquesne did not install the restraints.

SUMMARY

DISCUSSION OF RESULTS

The results of the regulatory assessment are summarized in Table 18. In this table, value represents a reduction in public health risk (man-rem) and impact represents an increase in ORE (man-rem) and costs (1988 dollars) to the power reactor industry. As can be seen from the results, there is a potentially significant impact to the industry with only a negligibly small increase in value to the public.

The following general observations can be made by reviewing the assumptions made in the assessment and analyzing the results depicted in Tables 2 and 18 and the sensitivity study.

- Incorporation of the proposed change to SRP 3.6.2 regarding the estimation of jet impingement forces would most likely require the nuclear power industry to perform some re-analysis of their postulated pipe breaks. Although the proposed changes generally lead the user to predict reduced jet impingement forces on a given target (less conservative), a prediction of wider jet plumes (wider than predicted by the old 10-degree recipe) at some distances from the postulated breaks may result in the industry's need to slightly modify (increase the effective protected area) some jet shields adjacent to stable breaks (postulated breaks in pipes that are fitted with restraints). Because the proposed change regarding jet impingement forces would not result in the addition or removal of jet impingement barriers or shields in either operating plants or plants under construction, decision factors analyzed in this regulatory analysis were not sensitive--not impacted--by this proposed change.

- Incorporation of the result that severe damage can result from pipe-to-pipe impact between pipes of equal diameter and wall thickness may require the nuclear power industry to re-analyze all unstable breaks (postulated breaks in pipes that are not restrained) per 10 CFR 50 (GDC-4) and modified SRP 3.6.2. Because this proposed change in SRP 3.6.2 would result in the addition of pipe-whip restraints to some high-energy piping in both PWRs and BWRs, the decision factors analyzed in this regulatory analysis were controlled by this proposed change to SRP 3.6.2. Values and impacts estimated for the incorporation of the proposed action regarding pipe-to-pipe impact are given in Table 18.

TABLE 18. Summary of Value Impact (Total for 118 Plants)

	Best Estimate	High Estimate	Low Estimate
Public health value (man-rem)	9E-4	2	4E-6
Occupational exposure impact (man-rem)	6E+3	2E+4	1E+3
Impact ($)	80M	200M	20M

- The proposed action requiring re-analysis of pipe-to-pipe impact damage suggests a check on the importance of the pipe impact damage probability to value-impact results. Studies aimed at determining this sensitivity showed a direct affect to factors dependent on core-melt frequency [i.e., public health risk, accidental occupational radiation exposure (ORE), and onsite and offsite economic risks] and no affect to the other factors (routine ORE and industry costs). A doubling of the pipe impact damage probability, for example, caused a likewise doubling of the negligibly small public health risk, and a doubling of the accidental ORE and industry economic risks. However, because of the insensitive nature of the controlling routine ORE, total ORE is insensitive to changes or inaccuracies in pipe impact damage probability. Likewise, because of the insensitive nature of the controlling industry implementation cost, total costs are insensitive to changes or inaccuracies in pipe impact damage probability.

- Sensitivity studies aimed at determining the effects of introducing the LBB philosophy to PWR piping showed that use of LBB in larger-diameter piping ($D \geq 2$ in.) had very little impact on the value of public health risk. On the other hand, the introduction of LBB did play a significant role in lowering routine ORE and industry implementation and operating costs.

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APPENDIX

PLANTS INCLUDED IN THE STUDY

APPENDIX

PLANTS INCLUDED IN THE STUDY

This appendix lists all plants considered to be affected by the regulatory analysis. Plant status as of March 31, 1988 is based on information published by the Office of Scientific and Technical Information, U.S. Department of Energy. The publication listing is: Nuclear Safety, Vol. 29, No. 3, July-September, 1988.

Years of remaining plant life for operating and construction reactors are estimated on the following bases.

1. All plants are assumed to have a design lifetime of 40 yr.

2. Plant operating life is calculated from the date of commercial operation. While it is recognized that the effective number of operating years for an individual plant may be measured from a different starting point (such as issuance of the construction permit, or fuel loading, or low-power testing), the selected basis was applied for the sake of consistency and because it offers a reasonable and convenient measure of the actual time of operation. Also, it should be noted that no credit (i.e., an increase in effective operating life) has been taken for nonoperational periods during either planned or forced outages.

3. All plants under construction are considered to have a zero forward fit; i.e., zero years until operation commences. This assumption is believed to be valid for the purposes of this study because the majority of construction plants are in the final construction or operating-license pending phase.

4. All plants that are "inactive" or "mothballed" (e.g., WNP-1) have been excluded from the construction plant bases.

TABLE A.1. Westinghouse (W) Plants

Reactor, (PWR)	Net Power, MWe	Start Oper, mo/yr	Start Oper, plant yr	Remaining Life, plant yr
Yankee Rowe	175	11/60	60.92	12.17
Haddam Neck	582	8/67	67.67	18.92
San Onofre 1	436	1/68	68.08	19.33
Robert E. Ginna	490	7/70	70.58	21.83
Pt. Beach 1	497	12/70	71	22.25
H. B. Robinson 2	700	3/71	71.25	22.50
Pt. Beach 2	497	10/72	72.83	24.08
Surry 1	788	12/72	73	24.25
Turkey Pt 3	693	12/72	73	24.25
Surry 2	788	5/73	73.42	24.67
Turkey Pt 4	693	9/73	73.75	25.00
Prairie Is 1	530	12/73	74	25.25
Zion 1	1040	12/73	74	25.25
Kewaunee	535	6/74	74.5	25.75
Indian Point 2	873	8/74	74.67	25.92
Zion 2	1040	9/74	74.75	26.00
Prairie Is 2	530	12/74	75	26.25
Donald C. Cook 1	1030	8/75	75.67	26.92
Indian Point 3	965	4/76	76.33	27.58
Trojan	1130	5/76	76.42	27.67
Beaver Valley 1	852	10/76	76.83	28.08
Salem 1	1090	6/77	77.5	28.75
Joseph M. Farley 1	829	12/77	78	29.25
North Anna 1	907	6/78	78.5	29.75
Donald C. Cook 2	1100	7/78	78.58	29.83
North Anna 2	907	12/80	81	32.25
Joseph M. Farley 2	829	7/81	81.58	32.83
Sequoyah 1	1148	7/81	81.58	32.83
Salem 2	1115	10/81	81.83	33.08
McGuire 1	1180	12/81	82	33.25
Sequoyah 2	1148	6/82	82.5	33.75
Virgin C. Summer 1	900	1/84	84.08	35.33
McGuire 2	1180	3/84	84.25	35.50
Callaway 1	1171	12/84	85	36.25
Diablo Canyon 1	1086	5/85	85.42	36.67
Catawba 1	1145	6/85	85.5	36.75
Catawba 2	1153	8/85	85.67	36.92
Wolf Creek 1	1170	9/85	85.75	37.00
Byron 1	1120	9/85	85.75	37.00
Diablo Canyon 2	1119	3/86	86.25	37.50
Millstone 3	1150	4/86	86.33	37.58
Vogtle 1	1113	6/87	87.5	38.75

TABLE A.1. (Contd)

Reactor, (PWR)	Net Power, MWe	Start Oper, mo/yr	Start Oper, plant yr	Remaining Life, plant yr
Byron 2	1120	8/87	87.67	38.92
Beaver Valley 2	836	11/87	87.92	39.17
Comanche Pk 1	1150			40
Comanche Pk 2	1150			40
Vogtle 2	1113			40
Watts Bar 1	1177			40
Watts Bar 2	1177			40
So Texas Proj 1	1250			40
So Texas Proj 2	1250			40
Seabrook 1	1200			40
Braidwood 1	1120			40
Braidwood 2	1120			40

....

Total operating W plants:	44
Total W plants under construction:	10
Total W plants considered:	54
Average remaining life, operating reactors, py:	29.6
Average remaining life, plants under construction, py:	40
Average remaining life, all W plants, py:	31.5
Remaining plant years, all W plants:	1700

TABLE A.2. Combustion Engineering (CE) Plants

Reactor, (PWR)	Net Power, MWe	Start Oper, mo/yr	Start Oper, plant yr	Remaining Life, plant yr
Palisades	805	12/71	72	23.25
Main Yankee	790	12/72	73	24.25
Fort Calhoun 1	478	6/74	74.5	25.75
Calvert Cliffs 1	845	5/75	75.42	26.67
Millstone 2	870	12/75	76	27.25
St. Lucie 1	830	12/76	77	28.25
Calvert Cliffs 2	845	4/77	77.33	28.58
Arkansas 2	912	3/80	80.25	31.50
St. Lucie 2	830	6/83	83.5	34.75
San Onofre 2	1070	8/83	83.67	34.92
San Onofre 3	1080	1/84	84.08	35.33
Waterford	1104	9/85	85.75	37.00
Palo Verde 1	1270	2/86	86.17	37.42
Palo Verde 2	1270	9/86	86.75	38.00
Palo Verde 3	1270	1/88	88.08	39.33

Total operating CE plants:	15
Total CE plants under construction:	0
Total CE plants considered:	15
Average remaining life, operating CE plants, py:	31.5
Average remaining life, plants under construction, py:	0
Average remaining life, all plants, py:	31.5
Remaining plant years:	472

TABLE A.3. Babcock & Wilcox (B&W) Plants

Reactor, (PWR)	Net Power, MWe	Start Oper, mo/yr	Start Oper, plant yr	Remaining Life, plant yr
Oconee 1	887	7/73	73.58	24.83
Oconee 2	887	9/74	74.75	26.00
Oconee 3	887	12/74	75	26.25
Arkansas 1	850	12/74	75	26.25
Rancho Seco	918	4/75	75.33	26.58
Crystal River 3	825	3/77	77.25	28.50
Davis Besse 1	906	7/78	78.58	29.83
TMI-1	906	12/78	79	30.25
Bellefonte 1	1213	inactive		0
Bellefonte 2	1213	inactive		0

Total operating B&W plants:	8
Total B&W plants under construction:	0
Total B&W plants considered:	8
Average remaining life, operating plants, py:	27.3
Average remaining life, plants under construction, py:	0
Average remaining life, all B&W plants, py:	27.3
Remaining plant years:	218

TABLE A.4. Summary of All PWR Plants

Total operating PWR plants:	67
Total remaining life, py:	1980
Average remaining life, py:	29.5
Total plants under construction:	10
Total remaining life, py:	400
Average remaining life, py:	40
Total plants:	77
Total remaining life, py:	2370
Average remaining life, py:	30.8

TABLE A.5. General Electric (GE) BWR Plants

Reactor, (BWR)	Net Power, MWe	Start Oper, mo/yr	Start Oper, plant yr	Remaining Life, plant yr
Big Rock Pt	72	3/63	63.25	14.50
Nine Mile Pt 1	620	12/69	70	21.25
Oyster Creek	650	12/69	70	21.25
Dresden 2	794	6/70	70.5	21.75
Millstone 1	660	3/71	71.25	22.50
Monticello	545	6/71	71.5	22.75
Dresden 3	794	11/71	71.92	23.17
Vermont Yankee	514	11/72	72.92	24.17
Pilgrim 1	655	12/72	73	24.25
Quad Cities 1	789	2/73	73.17	24.42
Quad Cities 2	789	3/73	73.25	24.50
Cooper	778	7/74	74.58	25.83
Peach Bottom 2	1065	7/74	74.58	25.83
Browns Ferry 1	1065	8/74	74.67	25.92
Peach Bottom 3	1065	12/74	75	26.25
Duane Arnold	538	2/75	75.17	26.42
Browns Ferry 2	1065	3/75	75.25	26.50
FitzPatrick	821	7/75	75.58	26.83
Brunswick 2	821	11/75	75.92	27.17
Edwin Hatch 1	777	12/75	76	27.25
Browns Ferry 3	1065	3/77	77.25	28.50
Brunswick 1	821	3/77	77.25	28.50
Edwin Hatch 2	795	9/79	79.75	31.00
Susquehanna 1	1065	6/83	83.5	34.75
LaSalle 1	1078	1/84	84.08	35.33
LaSalle 2	1078	10/84	84.83	36.08
WNP-2	1100	12/84	85	36.25
Susquehanna 2	1065	2/85	85.17	36.42
Grand Gulf 1	1250	7/85	85.58	36.83
Limerick 1	1055	2/86	86.17	37.42
Limerick 1	1055	2/86	86.17	37.42
River Bend 1	934	6/86	86.5	37.75
Hope Creek 1	1067	2/87	87.17	38.42
Shearon Harris 1	900	5/87	87.42	38.67
Perry 1	1205	11/87	87.92	39.17
Clinton 1	933	11/87	87.92	39.17
Fermi 2	1093	1/88	88.08	39.33
Nine Mile Pt 2	1080	3/88	88.25	39.50
Shoreham	819			40
Limerick 2	1065			40
Clinton 2	933			40

TABLE A.5. (Contd)

Total operating GE plants:	38
Total GE plants under construction:	3
Total GE plants considered:	41
Average remaining life, operating plants, py:	29.8
Average remaining life, plants under construction, py:	40
Average remaining life, all GE plants, py:	30.6
Remaining BWR plant years:	1250

TABLE A.6. Summary of All (PWR & BWR) Reactor Plants

Total operating plants:	105
Total remaining life, py:	3110
Average remaining life, py:	29.6
Total plants under construction:	13
Total remaining life, py:	520
Average remaining life, py:	40
Total plants:	118
Total remaining life, py:	3620
Average remaining life, py:	30.7