Report of the U.S. Nuclear Regulatory Commission Piping Review Committee

Summary — Piping Review Committee Conclusions
and Recommendations

U.S. Nuclear Regulatory
Commission

Prepared by the Piping Review Committee

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FOREWORD

The Executive Director for Operations (EDO) of the U.S. Nuclear Regulatory Commission (NRC) requested that a comprehensive review be made of NRC requirements in the area of nuclear power plant piping. In response to this request, the NRC Piping Review Committee was formed. The activities of this committee were divided into four tasks handled by appropriate task groups, namely:

- o Pipe Crack Task Group
- o Seismic Design Task Group
- o Pipe Break Task Group
- o Other Dynamic Loads and Load Combination Task Group

Each task group has prepared a report appropriate to its scope. In addition, the Committee has prepared this overview document. The five volumes make up the "Report of the U.S. Nuclear Regulatory Commission Piping Review Committee." The subtitles of the five volumes are

- Volume 1 - Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants
- Volume 2 - Evaluation of Seismic Designs - A Review of Seismic Design Requirements for Nuclear Power Plant Piping
- Volume 3 - Evaluation of Potential for Pipe Breaks
- Volume 4 - Evaluation of Other Dynamic Loads and Load Combinations
- Volume 5 - Summary - Piping Review Committee Conclusions and Recommendations

Volume 5 summarizes and sets priorities for major issues, reviews interfaces, and presents the Committee's recommendations and conclusions with respect to present NRC requirements and NRC staff positions on nuclear power plant piping and research deemed necessary to resolve outstanding issues.

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EXECUTIVE SUMMARY

The overall scope of the Piping Review Committee (hereinafter called the Committee) was to review nuclear piping in the context of current regulations, regulatory guides, standard review plans, and other pertinent documents. Four task groups were established under the Committee to prepare recommendations in the following areas: (1) Stress-Corrosion Cracking in Piping of Boiling Water Reactor Plants, (2) Evaluation of Seismic Designs, (3) Evaluation of the Potential for Pipe Breaks, and (4) Evaluation of Other Dynamic Loads and Load Combinations.

The task groups prepared Volumes 1 through 4 of NUREG-1061 dealing with the issues cited. These reports have been reviewed by the Committee (1) to identify and coordinate interactions among the four reports, (2) to examine the conclusions and recommendations, (3) to develop priorities for changes in regulatory requirements,* and (4) to establish priorities for research.

The suggested changes are quite substantial. The collective judgment of the Committee is that the implementation of the changes will have positive effects on both the licensing process and the safety and reliability of nuclear reactor power plants. Essentially all the principal suggested changes would lead ultimately to a simplification of licensing, and, if implemented, plant piping would be more accessible and inspectable, thereby reducing occupational radiation exposures, costs, and likelihood of undetected defects.

The Committee recognizes that implementation of the recommendations will require changes in regulations, regulatory guides, and standard review plans, as well as changes in the ASME Code. The necessary actions should begin as soon as possible because in the absence of definitive value-impact studies, it is the judgment of the Committee that there will be major payoffs in the cases of all actions suggested in the high-priority category and substantial, albeit lesser, payoffs in the second- and third-priority categories.

The Committee has completed its review of the four reports and has established three priority categories (A, B, and C) as well as subcategories within each. There are greater differences between A and B than between A-1 and A-2; for example, the Committee feels that the Category A items are highly significant in the context of both regulatory requirements and research needs. Only Category A items are covered in this summary.

The first priority item would relax the loss-of-coolant-accident (LOCA) criteria and apply leak-before-break (LBB) criteria instead. The implementation of such an action would permit removal of pipe restraints and jet impingement barriers, thereby enhancing the accessibility and inspectability of piping.

*"Requirements," as broadly referred to in this report, include NRC regulations, regulatory guides, standard review plan acceptance criteria, and staff positions delineated in various NUREG reports and are, therefore, not limited to the strict legal definition.

Changes in seismic damping values, the second priority item, have been accepted on a case-by-case basis. Broader implementation of these changes could substantially reduce the excessive number of piping supports, particularly snubbers. Some NRC Regions have already questioned the need for snubbers where a rigid support would suffice.

The third priority item, a change in operating basis earthquake (OBE) accelerations, would have a major impact on the design of new plants and extend well beyond piping considerations. In this instance, a decision is necessary concerning what is an appropriate ratio of OBE to SSE (safe shutdown earthquake). The ultimate aim is that the OBE not control plant design, which can be achieved by decoupling the OBE from SSE.

In the case of BWR intergranular stress corrosion cracking (IGSCC), the Committee suggests as the fourth priority item the further implementation of ongoing practices to replace existing recirculation piping with materials known to be resistant to IGSCC. The procedures exist and have been used to implement such actions.

The fifth priority item represents an interface between the first three items, particularly Recommendation A-1. We believe it is appropriate to decouple the LOCA from seismic events; the evidence confirms such a position.

The sixth priority item relates to improvement and modification of leak-detection systems. This issue impacts on the overall LBB issue as enunciated in Volumes 1 and 3.

It is the intent of this report that these recommendations be applied to operating reactors, plants under construction, and future plant designs.

The following is a summary of Category A recommendations for regulatory change in order of priority. Categories B and C are described in Section 10.

Category & Rank Order	Recommendation	Documents Requiring Change
A-1	Use LBB criteria rather than double-ended guillotine break criteria in design of piping so that terminal and intermediate breaks can be eliminated when certain acceptance criteria are met. This would lead to exclusion of dynamic effects such as pipe whip, jet impingement, and subcompartment pressurization. The major impact would be on General Design Criterion 4, <u>Environmental and Missile Design Bases</u> . The requirement to postulate arbitrary intermediate breaks should be eliminated.	10 CFR Part 50 (Appendix A, GDC-4, -30, -31, -32) SRP 3.6.2 R.G. 1.46

Category & Rank Order	Recommendation	Documents Requiring Change
A-2	Modify seismic damping values currently used in seismic design. The suggested values have been incorporated into ASME-III and accepted by NRR on a case-by-case basis. This modification could lead to changes in support design and spacing and consideration of nozzle loads as well as reducing the number of snubbers.	R.G. 1.61 SRP 3.9.2
A-3	Decouple OBE from SSE.	10 CFR Part 100 (Appendix A)
A-4	Replace 316SS or 304SS in BWR recirculation piping with alloys resistant to IGSCC to eliminate this mode of pipe cracking. Possible types are 316NG, 304NG, 347NG.	10 CFR Part 50 (Appendix A, GDC-30 (possibly)) NUREG-0313 R.G. 1.44
A-5	Decouple seismic and LOCA events in systems where LBB is applicable.	SRP 3.9.3
A-6	Modify leak-detection requirements. This issue impacts BWR-IGSCC as well as Recommendation A-1.	NUREG-0313 Tech. Specs. R.G. 1.45

The following is a listing of Category A items for research in order of priority. Categories B and C are described in Section 9.

Category & Rank Order	Recommendation
A-1	The full-scale pipe fracture experiments of the NRC Degraded Piping Program should be completed. Of primary interest is the development and/or validation of fracture mechanics analysis techniques for ductile piping. Experimental variables should include flaw geometries, material toughness, axial-to-bending load ratios, and static/dynamic loads.

Category & Rank Order	Recommendation
A-2	Advanced techniques and procedures for crack detection and depth sizing should continue to be developed and incorporated into Code requirements. Included should be analysis of the human factor, equipment qualification and certification, and inspection techniques for detection and dimensioning of flaws in pipes repaired by the weld overlay process.
A-3	Test programs (e.g., EPRI's piping capacity tests) for verifying seismic design margins and identifying failure modes for typical piping systems should be supported. Test results should be evaluated and recommendations provided for criteria changes (e.g., reclassification of seismic inertial stresses as "secondary"), as appropriate. Both cracked and uncracked piping systems should be tested.
A-4	Work under way at the Lawrence Livermore National Laboratory on Babcock and Wilcox and General Electric reactor coolant loop piping designs should be completed to learn whether earthquake in combination with reactor coolant loop double-ended guillotine break may be excluded for these designs.
A-5	Work should be performed to determine the reliability of methods to predict leak rate and validate the reliability of leak-detection systems.

1. INTRODUCTION

1.1 Background

This report documents a comprehensive review of NRC requirements* regarding nuclear power plant piping to determine whether current requirements need modification. A preliminary review of the subject had identified several regulatory positions in need of updating. For example, requirements relating to piping design were developed in the early stages of the commercial development of nuclear energy with conservatisms added to compensate for lack of conclusive data and service experience. As relevant data became available and service experience was gained, the need for appropriate changes to these positions became apparent. Conservative positions have caused the addition of numerous, massive pipe whip restraints and supports, including snubbers. These devices may lead to stiff piping systems that result in relatively high thermal stresses and nozzle loads.

In addition, if these devices are improperly installed or failure occurs during operation, they can result in excessive pipe loads. Their presence may adversely affect the construction and operation, including maintenance and inspection, of piping systems.

Other design requirements such as seismic criteria, pipe break criteria, and dynamic loads and load combination criteria were also in need of reassessment. It was anticipated that this reassessment could lead to revisions to the requirements that would reflect a more balanced approach to designing piping systems for normal operating and accident conditions.

While experimental and analytical evidence confirmed the excessive conservatism of some requirements, service experience raised concerns regarding our knowledge of the actual response of large BWR piping to degradation due to intergranular stress corrosion cracking (IGSCC). This concern is further exacerbated by uncertainties associated with crack detection and sizing of IGSCC in austenitic stainless steel piping welds.

The greatly expanded data base, including service experience, pertaining to nuclear piping and new developments in analysis techniques such as fracture mechanics analysis and probabilistic risk analysis permit an objective reassessment of existing regulatory requirements and help to identify significant improvements.

1.2 Regulatory Issues

Four groups of regulatory issues were identified as needing reassessment at the time the Committee was established:

1. Pipe cracking due to intergranular stress corrosion that has occurred in larger-diameter BWR piping more extensively than previously forecast. The issues principally relate to inservice

*"Requirements," as broadly referred to in this report, include NRC regulations, regulatory guides, standard review plan acceptance criteria, and staff positions delineated in various NUREG reports and are therefore not limited to the strict legal definition.

inspections, evaluation of repair techniques (including those for replacement materials), and bases for allowing continued operation of plants that have suffered IGSCC in piping.

2. Seismic design issues relating principally to the fact that the operating basis earthquake (OBE), although not directly safety related, usually controls design because of lower allowable stress levels and lower damping values. Issues also identified include criteria for enveloping spectral input, damping values, peak broadening requirements for floor response spectra, and industry design practices.
3. Pipe break issues relating to the present requirements for protection against dynamic effects due to double-ended guillotine breaks (DEGB) (specifically, designing for full flow area pipe breaks and selecting break locations).
4. Certain load combinations, particularly the loss-of-coolant-accident (LOCA) plus safe shutdown earthquake (SSE) load combination, represent severe design requirements leading to massive supports on piping and their components. No analytical or physical evidence supports a causal relationship between pipe break and earthquake, and, for the particular case of the primary loop of a pressurized water reactor (PWR), it has been demonstrated that earthquakes are extremely unlikely to induce a full flow area pipe break.

Other dynamic loads treated under this issue include hydrodynamic loads such as water hammer and loads resulting from safety relief valve (SRV) discharge, and vibrational loads.

1.3 Approach

The Committee was established to gather and review available information related to nuclear piping, including pertinent data from foreign sources, and evaluate this information for use in reaching conclusions and making recommendations relating to NRC requirements on the construction and operation of piping in nuclear power plants.

The Committee was predominantly composed of NRC personnel from the Offices of NRR, RES, and IE, and from the Regions, with assistance from OELD and an observer from the ACRS. Substantial use was made of expert consultants who helped to prepare position papers, assess data, and prepare summaries for further review. Significant information and comments were also received from industry representatives and foreign sources. In the area of IGSCC of BWR piping, an international review team was assembled to comment on the task group report.

Four technical task groups were formed, each responsible for a number of technical issues and the preparation of a report on the subject. The membership of these task groups and a list of supporting consultants are given in Appendix A. Each of these task groups has prepared one volume of this report. The Pipe Crack Task Group prepared Volume 1 entitled "Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants"; the

Task Group on Seismic Design prepared Volume 2 entitled "Evaluation of Seismic Designs - A Review of Seismic Design Requirements for Nuclear Power Plant Piping"; the Pipe Break Task Group prepared Volume 3 entitled "Evaluation of Potential for Pipe Breaks"; and the Task Group on Other Dynamic Loads and Load Combinations prepared Volume 4 entitled "Evaluation of Other Dynamic Loads and Load Combinations."

The Committee has reviewed these task group reports as well as various comments on the subject and has prepared this Volume 5 of the series. Volume 5 summarizes the major issues, reviews the interfaces, and presents the Committee's conclusions and recommendations for revision of NRC requirements, if appropriate, on these issues. Volume 5 also suggests research or other work that may be required to respond to issues not amenable to resolution at this time.

2. OBJECTIVE

The objective of the Committee was to review and evaluate current regulatory requirements for the construction and operation of piping systems in light-water-reactor nuclear power facilities using available domestic and foreign information, to provide recommendations on where and how the NRC should modify current requirements, and to identify areas requiring further action. The review was not intended to impact on existing ongoing regulatory actions prior to the issuance and implementation of the final report, nor was it intended to impede the resolution of any specific piping problem.

3. SCOPE

The scope of this review covers those pipes that are in the safety-related systems and those high-energy lines important to safety in new and operating nuclear power plants. With respect to postulated pipe break, the scope covers all high-energy lines. The review was performed on a system-integrated basis considering all ongoing programs. Safety-related piping systems are defined as those piping systems needed to ensure the integrity of the reactor coolant pressure boundary, to shut down the reactor and maintain it in the safe shutdown condition, and to mitigate the consequences of accidents. It is the intent of this review that the recommendations be applied to operating reactors, plants under construction, and future plant designs.

4. IGSCC IN BWR PIPING

4.1 Background

The EDO concurred in the recommendations of the Committee to Review Generic Requirements (CRGR) that the "NRC should develop an integrated program plan to deal with the entirety of the stress corrosion cracking problem in BWR piping in order to avoid a piecemeal approach to the problem. Such a program should enlist the aid of recognized experts in the area and result in a coordinated resolution that takes into consideration the entire spectrum of BWR plants."

This Committee attempted to comply with this charter. A review of cracking incidents in both BWRs and PWRs since the release of NUREG-0531 (primarily BWRs) in 1979 and NUREG-0691 (PWRs) in 1980 reveals that incidents due to all types of cracking for both BWRs and PWRs illustrate no unusual trends other than intergranular stress corrosion cracking (IGSCC) in the recirculating system of BWRs. Therefore, a decision was made to limit the Committee's scope on pipe cracking to IGSCC in BWRs.

In essence, the overall scope is quite comparable to that of NUREG-0531 with one notable exception. The Committee has made a conscious effort to relate recommendations to regulatory documents in which changes would be required if such recommendations were implemented.

Stress corrosion cracking, particularly IGSCC, is not a new phenomenon in nuclear reactor piping. For example, IGSCC has occurred in a range of piping sizes in BWRs over the past 25 years. To the Committee's knowledge, the most severe leaking crack was leaking at or below the technical specification limits; therefore, high leak rates have not been a problem, and IGSCC has induced no structural failures. The fact that the current problem affects the larger recirculation piping has resulted in increased concern.

The Committee considers that IGSCC is a serious problem requiring some changes to current regulatory practice. However, it is not believed that IGSCC represents such an urgent problem that it necessitates immediate additional regulatory action. The ongoing BWR surveillance and monitoring programs, e.g., those described in SECY-83-267C or in NRC Generic Letter 84-11, are adequate short-term responses to the problem.

Both value-impact and risk studies indicate that pipe failures, even assuming the higher rates due to IGSCC used in this report, would not be a major contributor to core melt. However, the following recommendations represent good engineering practice and enhance safety and should markedly reduce levels of inservice inspection and reactor outage time in terms of inspection sampling and frequency.

4.2 Technical Issues, Conclusions, and Recommendations

4.2.1 Long-Term Fixes

IGSCC occurs because of a synergism between the three causative factors: sensitization, an environment conducive to initiation and propagation of IGSCC, and high-tensile stresses (primarily residual but other loads are also contributors). It is recommended that at least two of the three causative factors

should be modified to minimize IGSCC although best practice calls for modification of all three. The preferred action to combat IGSCC is to replace commonly used materials such as Types 304 and 316 stainless steel, which are prone to sensitization, with a sensitization-resistant material such as Type 316NG stainless steel.

Experience with materials to mitigate IGSCC such as 347NG used in Germany and 304NG and 316NG used in Japan has been excellent. Other materials used in the United States include 304L and 316L. All these alloys are more resistant to IGSCC than conventional Types 304 or 316 stainless steel. Based on United States data and prior use, Type 316NG stainless steel offers an additional margin of resistance to IGSCC, and utilities that choose to replace pipe should be strongly encouraged to use it. One specific caveat applies. Although low-carbon stainless steels with nitrogen additions such as 316NG have been successfully fabricated and welded in Japan and Europe, United States experience with these materials is limited. It appears that greater care must be exercised in the control of composition and fabrication variables to limit cracking during hot-forming or welding.

For piping systems in future plants or for replacement of existing piping systems, the material, design of pipe joints, and accessibility from both sides of the weld should be optimized for inspections using ultrasonic testing (UT). A substantial percentage of welds are limited to single-sided access. This will remain the case even after recirculation systems are replaced with a material such as 316NG where the requirement should be mandatory to provide UT access for all components with the exception of existing items such as pumps, valves, and vessels. It is recognized that retention of the original pumps, valves, vessel nozzles, etc., results in regions adjacent to the weld that are too short for UT.

Attention to stress reduction with induction heating stress improvement (IHSI) should be considered for these components. (See Section 4.2.2, "Short-Term Fixes," for more information on IHSI.)

BWR water chemistry controls should be modified to minimize IGSCC. These modifications should include both a substantial reduction in the levels of ionic species entering the primary coolant and a control of the oxygen level. The current work on reducing oxygen through hydrogen additions should be closely followed, considering the possibility that it may be employed to further reduce the electrochemical potential of the stainless steel to a level at which SCC, either IGSCC or transgranular stress corrosion cracking, will not occur. It appears that hydrogen water chemistry is an effective IGSCC countermeasure. However, ongoing work regarding potential adverse effects on other reactor components should be closely observed in order to confirm the acceptability of this countermeasure.

Since operating experience and fracture mechanics evaluations indicate that leak before break is the most likely mode of piping failure, it is recommended that reasonably achievable leak-detection procedures be in effect in operating plants. Current sump pump monitoring systems are sensitive enough to provide additional margin against leak before break if more stringent requirements on surveillance intervals and unidentified leakage are imposed. Therefore, it is recommended that the limits on unidentified leakage in BWRs be decreased to 3 gpm and that the surveillance interval be decreased to 4 hours or less.

Additional fracture mechanics analyses, material properties characterization, and large-scale pipe tests should be performed to understand further the implications of stainless steel flux weld and cast material fracture toughness properties in flawed pipe evaluations. Furthermore, in this regard, the Committee recommends active NRC support of the ASME Section XI Task Group currently evaluating the concerns that have been raised regarding IWB-3640.

4.2.2 Short-Term Fixes

There are several measures capable of reducing the tensile residual stress level in the heat-affected zone of weldments in pipe. With appropriate control of parameters, a number of these processes should be acceptable.

IHSI is considered to be a more effective mitigating action for IGSCC than heat sink welding and last pass heat sink welding, in part because more data are available to demonstrate that the process does produce a more favorable residual stress state. All the residual stress improvement remedies are considered to be much more effective when applied to weldments with no reported cracking.

The use of IHSI on weldments with detectable cracking should be considered on a case-by-case basis. However, for relatively short cracks (approximately 20 percent of the circumference in length), even large errors in crack sizing or the prediction of flaw growth will lead only to small leakage, and the decision on whether an additional repair is required should be determined by analysis. For longer cracks, repair should be required.

For relatively short axial cracks, analysis can be used to justify long-term operation with or without weld overlays because errors in crack-depth measurement or flaw-growth predictions for these cracks will lead at worst to relatively small leaks that will be easily detectable long before the crack can grow long enough to cause failure. For circumferential cracks, weld overlay is considered an acceptable repair procedure for a maximum of two refueling outages unless reliable techniques for sizing cracks through the overlay or for monitoring crack growth are developed.

Flaw evaluation criteria should limit the length of the cracks accepted for continued operation without repair. The limitation on acceptable crack length is primarily a result of the lack of confidence in flaw-depth sizing capability and is intended to ensure leak-before-break conditions. The maximum allowable throughwall crack length can be determined based on weld joint specific loads and material properties.

The maximum crack length allowable without repair for a specific weld joint should be the minimum of either (1) the throughwall crack length demonstrated by elastic-plastic fracture mechanics analyses to be stable under normal operating plus SSE loading conditions* or (2) the maximum crack length that would result in a leak rate greater than the plant's normal makeup capacity. Shorter cracks can be evaluated using the IWB-3640 criteria as modified by the NRC staff in SECY-83-267C. Calculations indicate that in the majority of cases

*Limit-load analysis can be used if it is demonstrated that the material being evaluated has adequate fracture toughness to justify the assumptions associated with limit-load analysis or, if appropriate, factors of safety on fracture toughness are included in the analysis.

the maximum crack length acceptable under the above criteria will be approximately 25 to 30 percent of the pipe circumference.

On the basis of fracture mechanics evaluation for bounding and typical stress conditions and weld toughness properties, it is concluded that IWB-3640 provides an adequate basis for evaluating the majority of the weld connections in BWR recirculation piping. This is especially true because many of the cracks will be in higher toughness zones adjacent to the lower toughness welds.

4.2.3 IGSCC Detection and Sizing

A major problem cited in NUREG-0531 was the inability of ultrasonics to reliably detect and size IGSCC in austenitic alloys. Substantial progress has been made in the reliability of detection, assuming proper operator training and procedures. However, more work is required to develop a better understanding of the variability from operator to operator. The existing techniques are adequate to size the crack length to acceptable limits of accuracy. With regard to crack-depth sizing, the techniques that have been practiced are inadequate and require more work although significant improvements have been recently achieved in the sizing of IGSCC by state-of-the-art ultrasonic techniques and personnel training and qualification programs.

Although IGSCC detection has improved to the point that it is considered acceptable under optimum conditions and procedures, the detection reliability, as impacted by variability in operator procedure and equipment performance along with field conditions, needs further study and improvement. While length sizing of cracks is acceptable, depth sizing is currently inadequate. It is recommended that advanced techniques and procedures for crack detection and depth sizing continue to be developed and incorporated into Code requirements to provide data to reduce the need for extremely conservative fracture mechanics evaluations.

The current activities in personnel and procedure qualification and performance demonstration represent steps in the right direction, and the resultant process being implemented is acceptable in the interim. However, further improvement is needed. Therefore, it is recommended that ongoing industry and NRC activities to develop adequate criteria for qualifying the entire inspection process in order to achieve more reliable field inspection be completed and the criteria be implemented on a high-priority basis.

Code-minimum UT procedures result in totally inadequate IGSCC detection. Easily implementable modifications to these procedures have resulted in some improvement. These modifications have been incorporated into Code Case N-335. Therefore, it is recommended that Code Case N-335 should immediately be made mandatory for all augmented inspections until better procedures are developed.

Inspection techniques should be developed for detecting and dimensioning flaws in pipes repaired by the weld overlay process.

All Types 304 and 316 austenitic stainless steel piping systems operating at a temperature over 200°F (90°C) should receive augmented inservice inspection unless they have been treated with effective countermeasures.

The inspection schedule for welds should be based on the resistance to IGSCC of the materials and the effectiveness of the mitigating processes applied to the welds. The materials considered resistant are:

- 304L, 316L, 316K, 304NG, 316NG, 347NG, 308L,
- low-strength carbon steels,
- approved nickel-based materials,
- cast low-carbon/high-ferrite austenitic stainless steels,
- welds solution heat-treated after fabrication and welding, and
- others, as approved by NRC.

5. SEISMIC DESIGN

5.1 Background

Seismic design criteria of nuclear power plant piping evolved over a period of years through a series of often discrete regulatory actions without an overall assessment of their collective effect on the actual systems constructed. Some criteria were established without an adequate data base. The existing requirements, along with prevailing industry design practice, generally result in inherently stiff piping systems because of the increased use of supports, including snubbers. Because stiff systems increase thermal stresses and nozzle loads, they may be more adversely influenced by construction and operation errors, including maintenance and inspection errors. In addition, snubbers may suffer degradation or aging during operation, which may increase piping stresses because of snubber freeze-up.

5.2 Technical Issues

The Committee, after reviewing current regulatory positions, identified the following issues.

5.2.1 OBE and SSE

Appendix A to 10 CFR Part 100, "Reactor Site Criteria," requires that nuclear power plants be designed to both the operating basis earthquake (OBE) and the safe shutdown earthquake (SSE). Each plant is designed to remain structurally and functionally intact under the OBE (the earthquake determined to have a reasonable probability of occurrence during the plant design life) and then checked to maintain safety function under the SSE (the largest earthquake that potentially could occur at the plant site). Paragraph V(a)(2) in Appendix A to 10 CFR Part 100 also specifies that the OBE shall be at least one-half the magnitude of the SSE. Thus, the level of OBE is directly coupled with that of SSE. As a result, the original intent of OBE, defined in paragraph III(d) of Appendix A to 10 CFR Part 100 as being an earthquake that could reasonably be expected to affect the plant site during the operating life of the plant, may have been lost. Such inconsistency in defining OBE and SSE was further exacerbated by present regulatory positions that require different damping values for plant structures and piping under OBE and SSE conditions. The impact is twofold: (1) The OBE rather than the SSE often controls piping seismic design and (2) both OBE and SSE must be evaluated. Since designing piping systems to SSE is sufficient to ensure safety, the level of OBE should be defined as having a reasonable probability of occurrence but should be decoupled from the SSE.

5.2.2 Damping Values and Spectral Modifications

Damping values are specified in Regulatory Guide 1.61 and spectra peak broadening criteria are specified in Regulatory Guide 1.122. As relevant data on damping and assessment of peak broadening became available, the high degree of conservatism inherent in these positions became apparent. Because of a lack of understanding of the parameters affecting damping, lower-bound values were mandated for use in the seismic design. Regulatory Guide 1.122 specifies that the peaks of floor response spectra used for piping design be broadened by ± 15 percent to account for uncertainty in the calculation of soil-structure-piping system frequencies. In reality, a piping system will excite only one

frequency at each spectral peak. Thus, the peak broadening requirement unrealistically increases the total energy input to the piping system.

The Pressure Vessel Research Committee (PVRC) recently recommended an interim position on damping values that are dependent on piping modal frequency. In addition, the PVRC has also recommended an alternative to peak broadening whereby response spectra peaks would be shifted throughout the ± 15 percent range. The American Society of Mechanical Engineers (ASME) has incorporated the PVRC damping position (Code Case N-411) and the spectral modification position (Code Case N-397 and revised Appendix N) in Section III of the ASME Boiler and Pressure Vessel Code. The Committee believes that the ASME/PVRC recommendations are more realistic and should be implemented immediately.

5.2.3 Supports and Snubbers

Current design criteria address piping, piping supports, and component supports independently. A more integrated approach is needed for designing piping systems with proper procedures to optimize support placement and minimize the number of supports. The Committee believes that the ASME/PVRC should be requested to develop guidance on support design and that research should be encouraged by industry to support criteria improvement. In addition, snubbers are widely used in piping systems to restrain seismic motions. Operating experience has indicated that mechanical and hydraulic snubbers do not always perform reliably in service. For improving overall reliability of piping systems, the use of snubbers to meet piping system design requirements should be limited. The Committee believes that a nonmandatory "snubber reassessment" program for all plants should be initiated and that the nuclear industry should be encouraged to develop methods and procedures to limit the use of snubbers.

5.2.4 Component Nozzle Flexibility and Nozzle Loads

At present, piping system flexibility calculations (used for evaluating both static and dynamic loadings) do not always consider the flexibility of component nozzles. This omission, combined with the low nozzle allowable loads generally used today, also contributes to stiffening piping systems. The NRC and industry should develop design guidance on component nozzle flexibilities and stress limits.

5.2.5 Inelastic Analysis

At present, acceptance criteria for evaluating inelastic response analysis based on an amount of strain or deflection have not yet been developed. Since SSE is a low-probability event, it is appropriate to accept some inelastic behavior in the design of piping systems in order to fully use their capability to absorb and dissipate energy. Pseudolinear-elastic estimation methods should be developed and procedures designed to account for inelastic response.

5.2.6 Seismic Spectral Input

Current seismic design criteria are based on the premise that added conservatism in seismic design results in increased overall safety. One example is the use of enveloped spectra as a uniform seismic input to all piping supports

instead of the realistic consideration of individual seismic input for different supports. The result is piping that can be stiff and possibly less reliable. As a result of recently completed evaluations of multiple-response spectra methods (NUREG/CR-3811) by Brookhaven National Laboratory, the Committee feels that regulatory positions should be revised to permit the use of such methods in seismic analysis.

5.2.7 Summary of Issues

In summary, based on the foregoing discussion of problems in piping seismic design and the proposed changes in regulatory positions, the revised design criteria would lead to more flexible piping systems. Such systems would naturally result in greater displacements during seismic events. Regulatory positions should address such displacement effects for possible interaction with surrounding structures, systems, and equipment. The Committee recognizes that its investigations were solely based on available information and the state of technology at this time.

5.3 Conclusions and Recommendations

The Committee has completed evaluations of current regulatory positions with regard to the seismic design of piping systems in nuclear power plants. First, there is inconsistency in defining the OBE. Second, seismic design criteria are generally overly conservative because of inadequate supporting data and because of the perception that added conservatism in seismic design improves safety. Such seismic criteria have led to overly stiff piping and excessive use of snubbers and supports, which, in fact, could result in less reliable piping systems.

The Committee recommends that

- o Rulemaking amending Appendix A to 10 CFR Part 100 be undertaken to permit decoupling of the OBE and SSE.
- o ASME Code Cases N-411, pertaining to damping values, and N-397, pertaining to spectra modification, as developed by the PVRC piping committee and cited in WRC Bulletin 300, be expeditiously implemented. Regulatory Guides 1.61 and 1.122 should be revised or deleted to be consistent with these Code cases.
- o Section 3.9.2 of the NRC standard review plan be revised: (1) to address component nozzle flexibility, (2) to incorporate the performance goals for conducting inelastic seismic analysis, and (3) to permit the use of the multiple-spectra method.
- o A nonmandatory snubber reassessment program be initiated for all facilities to limit the use of snubbers.
- o The ASME Code Committee be requested to develop guidance on component nozzle flexibility.

6. POTENTIAL FOR PIPE BREAK

6.1 Background

6.1.1 Evolution of Current Requirements

The "design basis accident," "maximum credible accident," or "maximum hypothetical accident" have been terms used to describe what was generally known as the double-ended guillotine break (DEGB). The concept was originated by the U.S. Atomic Energy Commission for the multiple purpose of sizing containments and establishing "accident" doses and later for sizing emergency core cooling systems. The original concept was quite straightforward; namely, an instantaneous DEGB of a major pipe in the primary system of a light-water reactor would maximize the fluid release and establish an upper bound for the design pressure established for a containment. This determined the containment volume in relationship to a reasonable design accident pressure.

Later changes in regulatory philosophy, primarily with regard to seismic design, tended to shift the DEGB from a hypothetical accident to one with increasing credibility. It was a relatively short step from the hypothetical to a belief in major pipe breaks. A natural consequence of an accepted pipe break was the assumption of a terminal end (reactor pressure vessel nozzle) break and the asymmetric loading of the reactor pressure vessel (Unresolved Safety Issue [USI] A-2). If one accepts a DEGB, then massive pipe restraints to minimize pipe deflection become a natural consequence, and backfitting requirements follow automatically.

6.1.2 Current Requirements

1. Commission's Regulations

Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," requires postulation of pipe breaks and provision of appropriate protection against associated dynamic effects. The regulations also impose other design requirements stemming from postulated pipe breaks, e.g., emergency core cooling system (ECCS) (§50.46 of 10 CFR Part 50), containment (GDC-16, -50), other engineered safety features (GDC-34, -38, -41), and the environmental qualification of equipment (§50.49 of 10 CFR Part 50). However, the scope of this report is limited to addressing only the dynamic effects resulting from postulated pipe breaks. In that regard, the effective regulation is GDC-4 in the context of the definition of loss-of-coolant accidents (LOCAs), both of which are reproduced below.

"Criterion 4 - Environmental and missile design bases.

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit."

"Loss-of-coolant accidents - Loss-of-coolant accidents means those postulated accidents that result from the loss of reactor coolant at a rate in excess of the capability of the reactor coolant make-up system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.¹"

¹Further details relating to the type, size, and orientation of postulated breaks in specific components of the reactor coolant pressure boundary are under development.

The footnote to the definition of LOCAs warrants further discussion. Criteria relating to the type, size, and orientation of postulated breaks were developed by the staff, although not published in the regulations. These criteria were published first in a regulatory guide and later in standard review plan (SRP) sections both of which are described in Volume 3 of the Committee report.

2. Implementation of Commission's Regulations

The Commission's regulations (e.g., GDC-4 and the definition of the LOCA) as currently implemented by the applicable standard review plans and regulatory guides, impose the postulation of piping ruptures in high-energy fluid systems, both inside and outside containment, as part of the design bases for structures, systems, and components important to safety. These postulated ruptures include circumferential and longitudinal breaks, up to and including DEGBs in piping, which also encompasses the largest pipe in the reactor coolant system. The direct result of such postulated piping ruptures led to the establishment of USI A-2, "Asymmetric Blowdown Loads on PWR Primary Systems," and criteria to protect structures, systems, and components important to safety against the consequences of pipe breaks in all other high-energy fluid systems. Protective measures include physical isolation from postulated pipe rupture locations, if feasible, or the installation of pipe whip restraints, jet impingement shields, or compartments.

6.2 Technical Issues

6.2.1 Application of Leak-Before-Break Concept

The application of fracture mechanics technology in the licensing process to demonstrate the integrity of nuclear facility high-energy piping systems in lieu of requiring the postulation of nonmechanistic DEGBs has become known as leak before break (LBB). The LBB approach originated some years ago to resolve the asymmetric load issue (USI A-2) for certain PWR facilities* although fracture mechanics technology is much older. With the exception of the Systematic Evaluation Program, the NRC staff has limited the application of LBB technology

*Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops (NRC Generic Letter 84-04, February 1, 1984).

to addressing the dynamic effects of postulated DEGBs of PWR primary coolant system piping because these lines are of high quality, do not have a history of stress corrosion or fatigue cracking, and are not subject to water hammer loads. In resolving the blowdown asymmetric load issue, these lines have been subjected to in-depth technical reviews, including deterministic and probabilistic fracture mechanics studies, both of which led to the conclusion that the likelihood of complete failures of (or even leaks in) these lines is extremely small. LBB technology is also applicable to other systems and other types of nuclear facilities such as BWRs. To facilitate the broader application of the LBB approach, the NRC has initiated rulemaking to modify GDC-4. Without this modification, GDC-4 together with the definition of a LOCA requires the postulation of a DEGB or its equivalent as a longitudinal split in the largest pipe in a system. Thus, the application of LBB technology to date has required exemptions to the regulations.

The rationale used by the NRC staff and the Pipe Break Task Group (in Volume 3 of this report) for limiting the applicability of LBB technology to addressing the "dynamic effects" of pipe break are as follows:

- o In-depth technical and regulatory analyses have been performed that lead to the conclusion that facility safety is not jeopardized, and indeed is enhanced, by the elimination of certain protective devices such as pipe whip restraints and jet impingement shields. Thus, there is an immediate payoff in terms of large safety benefits (particularly a significant reduction in occupational radiation exposure during inspections and maintenance) and reduced impacts (the costs of designing and installing the protective devices that now are not needed).
- o Other sources of blowdown from either the primary or secondary systems within containment such as from malfunctioning safety relief valves, pump seals, etc. cannot be eliminated by analyses and must be considered. Thus, containments and ECCSs are still necessary even though pipe ruptures may be eliminated. Further in-depth studies of these matters are required before potential modifications to containment and ECCS design requirements can be considered.
- o Rulemaking to address the dynamic effects of postulated pipe ruptures only involves a modification of GDC-4 and hence can be accomplished in a relatively short time. Modification of containment and ECCS design requirements would involve many portions of the regulations, and rulemaking for this purpose would probably take a number of years.
- o The NRC staff/Committee approach of limiting the LBB application to dynamic effects has been discussed with, and is generally supported by, representatives of the nuclear industry.

In addition to pipe whip and jet impingement, there are other dynamic effects to be considered, i.e., the loads on piping and component supports and sub-compartment differential pressurization.

As mentioned earlier, the original application of LBB technology was to address and resolve the asymmetric LOCA load issue. This involved potentially large

asymmetric loads within the reactor cavity of PWRs and within the reactor vessel. These loads are reduced to insignificant values via the LBB approach. If pipes are demonstrated not to rupture, other subcompartment differential pressure loads are also minimized. However, consideration must still be given to pressure loads from other sources of blowdown.

If pipe rupture loads are eliminated, the design loading requirements for piping and component supports are also reduced. The NRC staff, however, has taken the position that design margins for component supports should be retained until this matter is subjected to further review by the staff. The Committee supports this decision.

A detailed discussion of limitations and general technical guidance for the application of the LBB approach is contained in Volume 3 of this report. It is not recommended that the LBB concept be applied to piping systems that are susceptible to IGSCC or to water hammer and to piping subject to erosion, such as portions of steam extraction systems. Volume 3 also discusses the currently available ductile fracture mechanics analysis algorithms and the need for appropriate associated materials properties; the use of a throughwall crack that can be detected during normal operation by in-plant leakage monitoring systems; the use of postulated accident loads; and recommended margins on postulated load or flaw size to ensure adequate resistance to fracture.

The discussion also includes various computational techniques, benchmark comparisons with currently available pipe experimental results, and consideration of appropriate materials data for use with analysis methods.

Ductile fracture mechanics methods employ analytical techniques ranging from elaborate finite-element models (FEMs) to various fracture mechanics estimation procedures to simple limit-load analyses. FEM analyses are expensive and time consuming to perform, and the purpose of the simple models is to facilitate the performance of fracture mechanics analyses in a timely and relatively inexpensive manner.

Although all fracture mechanics methods are based (at least to some extent) on theory, it is necessary to include in them certain idealizing assumptions related to crack shapes, consistent geometry, and crack behavior if the crack initiates and grows as a result of increased loads. Also, under most circumstances, it is necessary to obtain materials property data from other than the component being evaluated. However, actual flaws can have complex shapes, the component being evaluated may deform under high loads particularly in the vicinity of the flaw (e.g., a pipe may ovalize and its wall may become thinner near the flaw), and a growing crack may develop shear lips. These reasons plus the inherent variability of material properties from specimen to specimen lead to the conclusion that perfect correspondence between analytical and experimental results should not be expected. On the other hand, to be useful at all, analytical methods should be able to predict results within an acceptable uncertainty band that can then be accounted for by appropriate margins.

6.2.2 Arbitrary Intermediate Breaks

The position on pipe rupture postulation is given in detail in the Branch Technical Position MEB 3-1 as presented in Section 3.6.2 of the SRP. This position is intended to comply with the requirements of GDC-4 of Appendix A

to 10 CFR Part 50 for the design of safety-related nuclear power plant structures and components. The rules stated in this position intend that available piping design information be used for postulating pipe rupture at locations having relatively higher potential for failure so that an adequate and practical level of protection may be achieved.

In addition to the terminal ends, component connections, and other high-potential break locations, MEB 3-1 required protection at any location in Class 1 piping where calculated stress reaches $2.4 S_y$ (or 80 percent of yield) or where the usage factor reaches 0.1. This requirement was initiated to account for the effect of combined stress and fatigue. For additional protection, MEB 3-1 further required that two locations be selected along the intermediate portion of the pipe even if the calculated stress and usage factor do not exceed the specified limits. These two locations are selected at the two highest stress locations, even if their stresses are below $2.4 S_y$. Similar criteria are postulated for Classes 2 and 3 piping with the exception that fatigue is not a design consideration. The intent of MEB 3-1 is to obtain additional protection. As a result of these so called "arbitrary intermediate-break criteria," many pipe whip restraints have been installed. These restraints have resulted in many problems, which are described as follows:

- o Complications in Pipe System Design. Designing for the two arbitrary intermediate breaks is a difficult process because the location of the two highest stress points tends to change several times as a result of the iterative process involved in the seismic design of piping systems. The SRP (NUREG-0800, dated July 1981) provides criteria intended to reduce the need to relocate intermediate-break locations when the high stress points shift as a result of piping reanalysis; in practice, these criteria provide little relief. The two locations selected by the stress calculation may not be the actual locations of highest stress because the mathematical model may differ from the actual piping system. If the locations are not actually representative, proper protection may not be provided in accordance with the system's design.
- o Cost Factors. As a result of the arbitrary intermediate-break requirements, an excessive number of pipe rupture protection devices have to be designed and constructed. The cost for the design, construction, and operational service and maintenance is estimated to be from \$4 million for nine major systems to \$30 million for all systems.
- o Restricted Access for Inservice Inspection. The LBB concept can be implemented only when inservice inspection and/or leak-detection systems provide early detection of possible cracks and potential leaks in the system. However, the pipe rupture protection devices block access to welds and thus hinder inservice inspection. The removal and reinstallation of the pipe rupture protection devices will add to the time required to perform necessary inservice inspections. Restricted access will also increase occupational radiation exposure during repair, maintenance, and decontamination operations.
- o Increased Heat Loss to Surrounding Environment. Because pipe whip restraints fit closely around the high-energy piping, the piping

insulation must often be cut back in these areas to avoid interferences, thus creating convection gaps adjacent to the restraints. This creates an overall increase in heat loss to the surrounding environment and is a major contributor to the tendency for many containments to operate at temperatures near technical specification limits.

- o Unanticipated Thermal Expansion Stress. Pipe rupture protection devices are designed not to restrict pipe-free thermal expansion. Should these devices inadvertently come into contact with the pipe itself, unanticipated stresses due to restraint of thermal expansion can be introduced. The precise consequences of this incident are difficult to assess. Probabilistic analyses performed by the Lawrence Livermore National Laboratory indicate in general that the resultant reduction in flexibility reduces the overall safety margin of the pipe system.

6.3 Conclusions and Recommendations

6.3.1 Exemption Requests

As previously described in Volume 3 and other sections of this Volume 5, exemptions to GDC-4 of Appendix A to 10 CFR Part 50 with respect to the resolution of USI A-2 were justified, both on a technical and on a regulatory analysis basis. The Committee to Review Generic Requirements (CRGR) observed, after reviewing the staff's evaluation of the fracture mechanics analysis performed for the Westinghouse A-2 Owners' Group plants, that this technology is equally applicable to other piping systems. The Committee agrees with the CRGR.

As enumerated elsewhere in this report, there are large safety, occupational-radiation-exposure, and economic benefits that can accrue by the use of fracture mechanics to address the issue of piping integrity instead of postulating nonmechanistic accidents such as double-ended breaks of ductile piping. Therefore, the Committee recommends that, in parallel with expedited rulemaking, the NRC continue to grant plant-specific exemptions to GDC-4 to PWR applicants and licensees who provide justification for such requests, both on a technical and safety benefit basis, for their primary coolant piping. Such exemptions should relate to the requirement to postulate pipe breaks up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. Furthermore, the scope of the exemptions should only be applicable to the measures required for protection against the dynamic effects (e.g., pipe whip, jet impingement) of postulated pipe ruptures. It should pertain at this time neither to the definition of a LOCA nor to its relationship to the regulations addressing design requirements for ECCS (§50.46 of 10 CFR Part 50), containment (GDC-16, -50), and other engineered safety features.

The Committee further believes that LBB technology has advanced sufficiently so that the use of advanced fracture mechanics technology may be applied as an alternative to the postulation of pipe breaks in other facilities and in other high-energy fluid systems as defined in Section 3.6.2 of the SRP. High-energy fluid systems both inside and outside the containment may be included in the application of this technology if the recommended acceptance criteria stipulated in Volume 3 are met. At this time, the results of a successful demonstration of LBB should be restricted to the elimination of the dynamic

effects associated with postulated full flow area circumferential or longitudinal breaks in the piping. The specific dynamic effects that may be excluded are

1. Pipe whip and other pipe break reaction forces,
2. Jet impingement forces,
3. Vessel cavity or subcompartment pressurization,* including asymmetric transient effects, and
4. Pipe-break-associated transient loadings in functional systems or portions thereof whose pressure-retaining integrity remains intact.

The Committee recommends that the NRC consider granting exemptions to GDC-4 for this expanded application of LBB technology during the rulemaking process. To optimize the potential benefits available from carrying out this recommendation, the Committee believes that any implementation that occurs during the rulemaking process may require schedular treatment. The benefits to be gained are sufficiently great to warrant schedular treatment in order to achieve the in-plant objective of implementing the exclusion of consideration of the above dynamic effects at the earliest possible time.

6.3.2 Rulemaking

In parallel with the Committee activities, the NRC has initiated rulemaking to modify GDC-4 permitting the use of advanced fracture mechanics (applying the LBB concept) as an alternative approach to requiring the postulation of pipe ruptures. The Committee supports this initiative and strongly recommends that rulemaking be expeditiously pursued. The resources necessary for expeditious rulemaking (e.g., forming a task force and retaining consultants) should be made available on the highest-priority basis. The reason for this recommendation, as supported in other sections of this report, is that a large gain in safety, as well as economic gain, accrues from the elimination of massive protection devices in nuclear power facilities, particularly those intended to prevent whipping of ruptured pipes. The LBB approach will benefit licensees and applicants for operating licenses as well as applicants for future construction permits.

6.3.3 Arbitrary Intermediate Breaks

Pipe rupture protection devices can introduce many negative effects on plant operations and do not contribute to the plant safety as originally intended. Therefore, deletion of the requirement to postulate arbitrary intermediate breaks and the need to protect against the dynamic effects of jet impingement and pipe whip is warranted. Environmental qualification of equipment in the vicinity of these lines should be reviewed on a case-by-case basis until definitive criteria are developed.

*Pressurization and environmental effects due to leakage must be evaluated.

The Committee recommends that Section 3.6.2 (MEB 3-1) of the SRP be revised to incorporate proposed changes eliminating the requirements for mechanical pipe rupture protection against arbitrary intermediate breaks in all systems, including those subject to stress corrosion cracking, fatigue, and dynamic loads. The SRP should include definitive criteria related to environmental qualification of equipment.

7. OTHER DYNAMIC LOADS AND LOAD COMBINATIONS

7.1 Background

This section treats a number of unrelated piping issues not considered explicitly elsewhere. Event combinations deals primarily with the potential for simultaneous occurrence of earthquakes with pipe ruptures and water hammers; response combinations includes methods for evaluating the performance of multiply supported piping with independent inputs; stress limits and dynamic allowables cover inelastic allowables and strain rate effects in piping; water hammer loading treats code and design specifications for these loadings; and piping vibration loads deals with evaluation procedures for estimating other than seismic vibratory loads. Relief valve opening and closing loads are incorporated with this issue.

7.2 Technical Issues

7.2.1 Event Combinations

Event combinations refers to the assumed or postulated occurrence of distinct loads that are treated for design purposes as existing simultaneously. The focus is on infrequent and intermittent events, usually dynamic in character and of short duration, that may be independent or dependent on a common source or on each other. Normal operating loads such as operating temperature and pressure and deadweight loads will always be assumed to act concurrently with the infrequent and intermittent events.

There has never been a well-developed rational basis for considering concurrent earthquake and large loss-of-coolant-accident (LOCA) loads in the design basis. In the early 1960s, the double-ended guillotine rupture of reactor coolant loop piping was postulated for containment sizing and emergency core cooling system (ECCS) performance. Later this pipe rupture was combined with earthquake and applied to containment structural design and subsequently to the design of other plant features, including nuclear reactor piping and its support systems. The evolution of seismic design requirements over the last two decades has led to increases in seismic stresses. Likewise, large increases in the calculation of pipe rupture loads have taken place since the 1960s. Thus, designing to meet the requirements of this event combination has become progressively more difficult. Field evaluations of piping at conventional power plants and petrochemical facilities have indicated that ruptures in the type of piping found in nuclear power plants in general do not occur during severe earthquakes. Moreover, recent probabilistic assessments demonstrate that for the particular case of PWR primary system piping, pipe rupture is extremely unlikely under any transient condition, including earthquakes, although special attention must be directed toward maintaining the reliability of heavy component supports.

While undue conservatism may have been exercised in combining certain pipe rupture events with postulated earthquakes, the same conclusion cannot be reached for other combinations of dynamic loads such as water hammer, safety relief valve (SRV) discharge, turbine trips, and vibratory loads. Since water hammer occurrences have resulted in damage to piping and piping supports in nuclear plants, water hammer was designated an Unresolved Safety Issue (USI A-1), and this issue was technically resolved in March 1984 (see NUREG-0927). Nonetheless,

water hammer will continue to recur (despite design and operating precautions) because of the nonanticipatory nature of the phenomenon.

7.2.2 Response Combinations

Response combinations treats questions regarding the use of independent support motion methods in place of the presently approved uniform response spectrum techniques specified in the standard review plan (SRP). Additionally, issues relating to the sequence of combinations between directional and modal components and to the treatment of high-frequency modes are included.

The NRC position on multiply supported piping with independent seismic inputs was developed at a time (during the early 1970s) when the urgency to establish criteria did not allow for a complete assessment of the problem. As a consequence, criteria were selected that would provide conservative results without, however, indicating the effect that these criteria might have on overall reliability. These criteria were based on conservative assumptions.

Recent studies have indicated that, in most cases, analyses based on these assumptions can considerably overestimate the seismic response when compared to time-history solutions that do not embody these conservatisms.

These investigations demonstrate that significant reductions in calculated responses used for design can be achieved through the implementation of improved methods, without, however, leading to unconservatisms.

7.2.3 Stress Limits and Dynamic Allowables

This section deals with two issues relating to allowable limits for piping analyses. The first issue involves the appropriate allowables (stress or strain limits) that should be used for piping if inelastic piping analyses are performed. The second issue involves the appropriate treatment of strain rate effects in piping analyses. Strain rate effects involve the increase in measured material yield strength when the specimen is rapidly loaded. Both issues are relevant to criteria for infrequent dynamic design events postulated for piping systems. As a result of a review of all available information, the Committee recommends no substantive changes to present practices regarding these matters.

7.2.4 Water Hammer Loads

Water hammers have occurred in nuclear power plants since the late 1960s; since that time, approximately 150 water hammer occurrences have been reported. The staff's concerns were founded on the increasing frequency of occurrence in the early 1970s and, in particular, the feedwater line rupture at the Indian Point 2 plant in December 1972 due to a steam generator water hammer. Since that time, only one additional incident (i.e., at Maine Yankee in January 1983) has resulted in a pressure boundary failure due to water hammer. The other water hammer occurrences have resulted primarily in damage to piping supports and/or equipment supports.

Because of the multidisciplinary nature of the problem, there does not exist a systematic and uniform treatment of water hammer in developing design specifications, except for major events such as turbine stop valve closure, feedwater

line break, and SRV discharge in nuclear power plants. It is not always clear whose responsibility it is to determine the susceptibility of a system to water/steam hammer (i.e., system designer versus piping designer). If these events are not mentioned in the design specification, it is possible that the system will not be evaluated for them.

7.2.5 Piping Vibration Loads (Including Relief Valve Loads)

The need for consideration of hydrodynamic loads from the operation of SRVs came from observations of the relatively high magnitude of loading due to the pressure-suppression phenomenon in BWR plants. Requirements for the consideration of these loads were incorporated in the July 1981 revision to the SRP. Also, the anticipated water/steam hammer loading was emphasized as a source of vibratory loading.

Unanticipated vibratory loads, however, have always been considered important for integrity and functionality of piping systems and are dealt with under the dynamic testing requirement in the SRP.

Piping system design for "high frequency" (33 to 100 hertz) vibratory loading is generally performed by using in-structure acceleration response spectra. There is reason to believe that the use of these acceleration response spectra for piping design may lead to overestimating the actual loading.

Results of SRV tests and studies analyzing the test data are now available. It is clear from these studies that the high-amplitude responses are consistently overpredicted by analytical means. The prevailing view is that anticipated vibratory loads should be accounted for by a combination of analysis and pre-operational testing, and reliance must be placed on testing for consideration of unanticipated vibratory loads.

7.3 Conclusions and Recommendations

Concerning regulatory changes, as opposed to needed research, the following findings are endorsed:

- o When adequate technical evidence is presented, the event combination of earthquake and double-ended guillotine pipe rupture may be excluded from the design basis for the mechanical design of components and their supports. Such evidence already exists for the reactor coolant loop piping of Westinghouse and Combustion Engineering designs, and this event combination should be eliminated for these vendors. This evidence is based on probabilistic studies performed at the Lawrence Livermore National Laboratory. The staff emphasizes that it believes evidence on only primary circuit piping exists at this time. Information for Babcock and Wilcox and General Electric reactors does not exist but is now being developed. Requirements for ECCS performance and containments are not affected by this revision. The Committee recommends that the decoupling of SSE and LOCA loads be permitted for the design of the reactor coolant loop of those designs for which the technical evidence as discussed above has been verified.

- o Water hammer events should be considered in the pipe stress analysis and pipe support design process for which the ASME Code-required design specification includes such requirements. The design specification shall define the load and specify the applicable Code Service Stress Limit. The NRC design requirements are based on endorsement of ASME Code requirements, and the development of adequate design specifications is incumbent on the applicant or his designer.
- o The independent support motion response spectrum method should be allowed as an option in calculating the response of multiply supported piping with independent inputs. This method could replace the uniform support motion procedure in the SRP. Special rules have been developed by the Task Group on Other Dynamic Loads and Load Combinations for calculating the inertial, pseudostatic, and combined responses. Additionally, new provisions for treating high-frequency modes are associated with this recommendation.
- o The conduct of vibration testing programs in accordance with the latest ANSI/ASME OM3 standard, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems," should be allowed and accepted in the SRP.
- o For vibratory loads other than seismic and with significant loading in the frequency range of 33 to 100 hertz, it is acceptable to perform nonlinear analysis to account for gaps between pipes and pipe supports provided that verification of the predicted nonlinear response is made.
- o The functionality criterion for piping will be maintained. Current ASME Code Class 1 or Class 2 stress evaluation procedures, not to exceed Level C limits, will be used. These limits are similar to those being used on a case-by-case basis to satisfy the functionality criterion. It is recommended that the upcoming EPRI/NRC pipe tests be evaluated to confirm this position and to determine whether it is appropriate to use the current higher Level D stress limits.

Excluding the combination of SSE and the reactor coolant loop double-ended guillotine break from the design basis will result in a large increase in calculated safety margins of reactor internals, heavy component supports and systems, and components and structures inside the containment of existing plants. In the event that the seismic hazard is increased or design deficiencies are discovered in operating plants, adequate margins may still be shown to exist without undertaking any plant modifications. For any future plants, relaxed and more realistic design standards will prevail.

The revisions discussed above regarding multiply supported piping with independent inputs will lead to more accurate and more realistic estimations of piping behavior. Significant predicted reductions in response can be expected in general for all response quantities. Adoption of these procedures could lead to the removal of pipe supports from operating plants without violating Code allowables.

With respect to piping vibration loads, it is generally recognized that high-frequency (33 to 100 hertz) loading as currently evaluated by analytical techniques tends to overpredict piping response. By allowing nonlinear analysis with appropriate verification through preoperational testing, it would be possible to evaluate more realistic response of piping. This should be particularly useful to utilities making modifications to safety-related piping.

As in the case for multiply supported piping with independent inputs, this could lead to a reduction in the number of piping supports and perhaps an improved reliability of piping systems to accommodate vibratory loads. It is expected that some additional testing may result from the proposed changes. However, the benefits from reduced piping supports and a more reliable piping system could outweigh the cost.

Concerning future investigations, the following suggestions are offered:

- o Work under way at the Lawrence Livermore National Laboratory should be completed on Babcock and Wilcox and General Electric (Mark I) reactor coolant loop systems to learn if the probability of a double-ended guillotine break combined with earthquake is sufficiently low so that this event combination can be excluded from the design basis for these two particular vendors. Later, other General Electric configurations (Mark II and Mark III) may be considered.
- o Currently planned research efforts related to evaluating flawed (degraded) ductile piping response to dynamic loads such as simulated seismic and water hammer loads would be useful for developing predictive techniques for estimating design margins.
- o Characterization of hydrodynamic loads and the prediction of the response of a piping system subjected to such loads are subject to several sources of uncertainty. Significant improvement in the licensing review process can be achieved by benchmarking both the thermal-hydraulic transient load and the piping response calculations through the development of standard problems and acceptable solution bounds.
- o Additional effort on phase correlation between support motions and the impact on the recommendations for multiply supported piping with independent inputs would assist in removing uncertainties.

8. INTERFACES AMONG ISSUES

8.1 Introduction

The purpose of this section is to reconcile and to clarify where necessary the conclusions and recommended positions of the respective task groups responsible for developing the following volumes of NUREG-1061.

- Volume 1 - Investigation and Evaluation of Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants
- Volume 2 - Evaluation of Seismic Designs - A Review of Seismic Design Requirements for Nuclear Power Plant Piping
- Volume 3 - Evaluation of Potential for Pipe Breaks
- Volume 4 - Evaluation of Other Dynamic Loads and Load Combinations

8.2 Applicability of Leak-Before-Break Approach

The term leak before break (LBB) has recently been used in many different forums and in various contexts. Some confusion as to the definition of LBB and its application has been the result. In general, the LBB failure mode in piping refers to the concept that a crack (either postulated or real) will propagate through the wall of a pipe by some defined mechanism (e.g., IGSCC, fatigue) and will result in a stable throughwall crack that can be reliably detected by leakage. The development of advanced fracture mechanics technologies applicable to pressure-retaining components, including piping systems, are based on theory and validation by experiments. The conclusion reached from the fracture mechanics studies and from many reactor years of operating experience is that flawed piping is much more likely to leak before it breaks. Both Volumes 1 and 3 of NUREG-1061 use the LBB technology; however, they apply it in a different manner and for a different purpose.

Volume 1, which addresses IGSCC in piping of operating BWRs, uses fracture mechanics (the LBB approach) in flaw evaluation for determining the disposition of pipes containing cracks detected during service. When a crack is detected in service, an evaluation must be performed to determine if repair or replacement of the pipe is necessary or if the crack is sufficiently small to return the cracked pipe to operation for some specified operating time. To establish allowable crack sizes for relatively short cracks, Volume 1 recommends using the allowable crack sizes provided in subsubarticle IWB-3640 of Section XI of the ASME Code with a factor to account for uncertainties in crack depth sizing.

The acceptable flaw sizes presented in IWB-3640 are intended to provide a factor of safety against pipe failure of at least 2.77 for nominal operating conditions and at least 1.38 for emergency and faulted conditions. The critical crack dimensions in IWB-3640 were determined using net-section collapse, also referred to as limit-load or plastic-collapse analysis. Limit-load analysis, when applied to ductile materials with adequate toughness, is a valid LBB analysis technique.

In contrast, Volume 3 uses various fracture mechanics analysis techniques to demonstrate that LBB is valid for certain high-energy piping. General Design Criterion 4 of Appendix A to 10 CFR Part 50 requires postulation of pipe breaks and provision of appropriate protection against associated dynamic effects. Note that the regulations also impose other design requirements stemming from postulated pipe breaks, e.g., ECCS (§50.46 of 10 CFR Part 50), containment (GDC-16, -50), other engineered safety features (GDC-34, -38, -41), and the environmental qualification of equipment (§50.49 of 10 CFR Part 50). However, the scope of Volume 3 is limited to addressing only the dynamic effects resulting from postulated pipe breaks. The objective is to demonstrate by deterministic analyses that the detection of small flaws either by inservice inspection or by leakage monitoring systems is ensured long before the flaws can grow to critical or unstable sizes that could lead to large-break areas such as the double-ended LOCA or its equivalent.

Considering the objective of Volume 3, one of the limitations that applies to the mechanistic evaluation of pipe breaks in high-energy fluid system piping is:

"The LBB approach should not be considered applicable to high-energy fluid system piping, or portions thereof, that operating experience has indicated particular susceptibility to failure from the effects of corrosion (e.g., IGSCC), water hammer, or low and high cycle (i.e., thermal, mechanical) fatigue."

The reason for this limitation should be self-evident. In addition, Volume 1 and its implementation documents address all aspects of IGSCC in BWR piping.

8.3 Limit-Load Analyses (Net-Section Collapse)

Limit-load analyses of circumferentially and axially cracked pipes have been successfully applied in many cases. Both Volumes 1 and 3 use this approach in their evaluations. Net-section collapse, plastic instability, and flow stress-dependent analyses are terms frequently used interchangeably in reference to limit-load analyses. The inherent assumption in applying such an analysis is that the material toughness is adequate to ensure that failure loads are controlled by the material strength. Net-section collapse analysis predicts the maximum load based on the initial crack size. Hence, another assumption in applying the net-section collapse approach is that there is negligible crack growth between the load where extension of the initial crack begins and the maximum load capacity of the pipe.

The net-section collapse analysis for circumferentially cracked pipe was originally developed for applications to stainless steel pipes. The concepts were based on center cracked flat plate experiments and, in subsequent research, on pipes in pure bending. For throughwall, circumferentially cracked pipe under pure axial tension stress, comparisons with existing experimental data show that the net-section collapse analysis gives a good estimate of the maximum load. For combined bending and pressure loading on throughwall, circumferentially cracked pipes, very little data are available to assess the validity of the net-section collapse approach.

The above comparisons are encouraging in regard to using the simple limit-load analyses; however, certain limitations need to be addressed. The major concern is that for low toughness materials, the limit-load approach may be nonconservative. The degree of nonconservatism is currently unknown since tests conducted to date have been on relatively small-diameter piping and fairly high toughness materials. In addition, many previous tests were performed without measurements of crack extension prior to maximum load. Furthermore, many previous tests were conducted using system compliances that do not appropriately model real piping compliance characteristics that may result in lower load-carrying capacity than indicated by the limit load.

Volume 3 stated that, in the absence of experimental pipe fracture data on low-toughness materials and/or representative piping system compliances, the net-section collapse analysis cannot be shown to give accurate estimates of the maximum load-carrying capacity for all the potential pipe cases of interest to the nuclear industry. While the toughness of most materials used in nuclear facility piping systems is considered to be adequate, there are currently a number of exceptions such as the toughness of stainless steel submerged arc weldments. Volume 3 recommended that a toughness comparable to or better than that of A106 Grade B carbon steel be demonstrated to justify using the limit-load approach. When adequate material toughness is established, it is suggested that the limit-load approach can be used when the calculated limit load is greater than the service load of interest by a factor of at least three. The use of limit-load analysis with a factor of three is intended to demonstrate that double-ended guillotine breaks will not occur.

Because of the nature of the current IGSCC problems in BWRs, Volume 1 was published before the other three volumes of NUREG-1061. However, the issue of potential low-toughness weldments and materials was recognized in Volume 1 in relation to the use of IWB-3640 which, as previously stated, is a limit-load approach.

On this issue, Volume 1 stated:

"Regarding the issue of lower resistance to crack extension in weld deposits relative to wrought materials and the potential for ligament instability, the concern is that the assumptions associated with net-section collapse may be invalid. Specifically, the assumption that no crack extension will occur prior to reaching maximum load may be incorrect. Destructive examination of a limited number of cracked weldments removed from service has revealed that cracks which initiate in the HAZ by SCC may grow into the weld metal. This type of cracking behavior also has been observed in laboratory tests. Although crack growth into welds is rare and, in service has been associated with low-ferrite-content repair welds, the possibility that IGSCC cracks may grow into the weld material, regardless of the ferrite level, cannot be ignored. In relation to IWB-3640, this issue must be addressed for both weld metal and cast stainless steel materials which also have reduced fracture toughness."

Volume 1 further states:

"However, the Task Group has recently learned that a significant percentage (approximately 30%) of the circumferential welds in BWR

pipings are shop fabricated submerged arc welds. It has also been learned that some fraction of these welds, the exact number of which has not been tabulated, were not solution annealed. These findings are important because limited data show that ductile fracture toughness properties for submerged arc welds are significantly lower than those used in this evaluation. The Task Group is therefore initiating a high priority effort to evaluate the impact of the submerged arc weld fracture toughness data on the calculations presented here. The Task Group also recommends that NRC and industry initiate high priority efforts to generate more fracture toughness data for the range of weld types existing in BWR plants."

In order to reconcile the criteria of Volumes 1 and 3 in this area, NRR is working with the Section XI group responsible for developing and revising IWB-3640, the ASME Task Group on Pipe Flaw Evaluation. This Task Group is embarking on the following:

"To ensure that the advertised IWB-3640 safety margins are maintained when the flaw evaluation standards are applied for flux welds, an adjustment to the loads ($P_1 + P_2$) in Tables IWB-3641 can be made. This adjustment increases the applied loads by an amount that represents the reduced load-carrying capacity of the flux welds relative to limit load. Application of this procedure effectively reduces the allowable flaw sizes acceptable for continued service while retaining the advertised safety margins and the convenience of Tables IWB-3641. The method used to evaluate the apparent reduction in load-carrying capacity for flux welds is based on the J-Integral and the associated tearing modulus crack stability criterion. Because part-through J solutions are not available in the literature, the reduction in load-carrying capacity is based on throughwall cracks. Computations will be performed for both submerged arc welds (SAW) and shielded metal arc welds (SMAW)."

The preceding work is continuing and should be completed in early 1985. The NRC should continue to follow and participate in the work of the ASME Task Group on Pipe Flaw Evaluation and determine if the resulting Code procedures are acceptable in lieu of the flaw evaluation procedures proposed in Volume 1 of this report.

8.4 Replacement Criteria

A preferred wording of Recommendation 3 on page 2 of Volume 4 should be:

- o 3. A replacement pipe rupture for combination with the safe shutdown earthquake should be investigated. (The existing text in Volume 4 uses "developed" instead of "investigated.")

8.5 Load Uncertainties

In the course of this review, the Committee discussed water hammer loads. Because of the uncertainty of the magnitude of these loads, resolution of the issue is recommended by minimizing the potential for water hammer events.

Another phenomenon that can lead to significant uncertainties in stress magnitudes is thermal fatigue such as has occurred in PWR feedwater lines. The Committee did not address this issue because it was adequately covered in NUREG-0691, prepared by a previous Pipe Crack Study Group.

9. RECOMMENDED RESEARCH

This section includes the more significant recommendations for research from Volumes 1 through 4 although the wording used in the other sections of this report has not necessarily been preserved. An effort has been made to combine recommendations that tend to overlap so that they appear as one item. The terminology appearing under the heading Category and Rank Order, i.e., Vol. 1, etc., identifies the volume in which the item appears.

The Committee has assigned three priority categories, i.e., high (A), medium (B), and low (C), to distribute the various recommendations. Further, the recommendations within a given priority category are rank ordered. In the Committee's estimation, the differences in priorities within a given category such as A are substantially less than between categories such as the last item in A versus the first item in B.

The priorities appearing in Table 9.1 recognize the level of work under way by industry as well as by the NRC. If a given research area is being addressed primarily by industry, the Committee uses lower priorities than the issue would warrant on purely technical bases.

Table 9.1

SUMMARY OF RECOMMENDATIONS FOR RESEARCH

Category & Rank Order	Recommendation
A-1 (Vols.1,3)	The full-scale pipe fracture experiments of the NRC Degraded Piping Program should be completed. Of primary interest is the development and/or validation of fracture mechanics analysis techniques for ductile piping. Experimental variables should include flaw geometries, material toughness, axial-to-bending load ratios, and static/dynamic loads.
A-2 (Vol.1)	Development of advanced techniques and procedures for crack detection and depth sizing should continue for incorporation into Code requirements. Included should be analysis of the human factor, equipment qualification and certification, and inspection techniques for detecting and dimensioning flaws in pipes repaired by the weld overlay process.
A-3 (Vols.2,3)	Test programs (e.g., EPRI's piping capacity tests) for verifying seismic design margins and identifying failure modes for typical piping systems should be supported. Test results should be evaluated and recommendations provided for criteria changes (e.g., reclassification of seismic inertial stresses as "secondary"), as appropriate. Both cracked and uncracked piping systems should be tested.
A-4 (Vol.4)	Work under way at the Lawrence Livermore National Laboratory on Babcock and Wilcox and General Electric reactor coolant loop piping should be completed to learn whether earthquake in combination with reactor coolant loop double-ended guillotine break may be excluded for these designs.
A-5 (Vol.3)	Work should be performed to determine the reliability of methods to predict leak rate and validate the reliability of leak-detection systems.
B-1 (Vols.1,3)	A comprehensive data base of tensile and ductile fracture toughness properties should be developed for piping materials.
B-2 (Vol.2)	The ongoing work at the Oak Ridge National Laboratory to develop improved design guidance on nozzle stress limits and flexibilities should be completed.
B-3 (Vols.2,4)	The damping tests at the Idaho National Engineering Laboratory should be completed, and additional studies should be made to determine damping values for nonseismic loadings.

Table 9.1 (Continued)

Category & Rank Order	Recommendation
B-4 (Vol.1)	Studies of modifications to BWR water chemistry to minimize IGSCC in austenitic piping should be continued. The studies should include the potential for adverse effects such as the impact of nitrogen-16 on personnel and hydrogen absorption by fuel cladding.
C-1 (Vol.2)	The uncertainty range of seismic spectral frequencies, including uncertainties in piping system modeling, should be assessed. The appropriateness of new, simple spectral-broadening procedures based on equivalent energy input should be evaluated.
C-2 (Vols.2,4)	<p>Additional analytical studies should be performed to</p> <ol style="list-style-type: none"> 1. Clarify the impact of phase correlations between support groups on the recommendations for the independent support motion method, 2. Evaluate methods for calculating the effect of closely spaced modes, and 3. Establish the transition frequency between high and low frequency when implementing the algebraic summation rule for high-frequency modal combinations.
C-3 (Vols.3,4)	Tests of high-energy piping should be conducted to determine what, if any, replacement pipe rupture and pipe crack criteria should be used.
C-4 (Vol.1)	The long-term effects of decontamination procedures on IGSCC need to be investigated.
C-5 (Vol.1)	It is recommended that a formal study and updating of the pipe failure data base be undertaken.
C-6 (Vol.1)	It is recommended that the PRAISE computer code be used to (1) investigate the impact of IGSCC in primary piping according to pipe size, (2) study the failure frequencies of components due to IGSCC, fatigue, etc., and (3) calculate the conditional probability of multiple failures associated with IGSCC to check the point value and to test the assumption of a lognormal distribution.
C-7 (Vol.2)	Pseudolinear-elastic estimation methods should be developed and procedures designed to account for inelastic response.

Table 9.1 (Continued)

Category & Rank Order	Recommendation
C-8 (Vol.4)	The response of piping to thermal-hydraulic transients should be further benchmarked to help reduce uncertainties.
C-9 (Vol.2)	The Pacific Northwest Laboratory study of Licensee Event Reports related to snubber performance should be completed to identify failure causes and effectiveness of various snubber types and to suggest methods of improving performance.
C-10 (Vol.2)	The gathering and assessing of earthquake experience data for piping systems should be continued. One objective would be to establish and justify an earthquake level that piping systems can sustain with sufficient confidence that no seismic analysis is needed.
C-11 (Vol.2)	Studies should be conducted to 1. Investigate the effect of support gap size on piping system behavior for both seismic and thermal loadings, and 2. Assess the performance of various piping supports.

10. RECOMMENDATIONS - PRIORITIES AND IMPLEMENTATION

This section includes the more significant recommendations from Volumes 1 through 4 although the wording used in the other sections of this report has not necessarily been preserved. An effort has been made to combine recommendations that tend to overlap so that they appear as one item. The terminology appearing under the heading Category and Rank Order, i.e., Vol. 1, etc., identifies the volume in which the item appears.

The Committee has assigned three priority categories, i.e., high (A), medium (B), and low (C), to distribute the various recommendations. Further, the recommendations within a given priority category are rank ordered. In the Committee's estimation, the differences in priorities within a given category such as A are substantially less than between categories such as the last item in A versus the first item in B.

Table 10.1 includes the various recommendations rank-ordered by category as well as the various documents that would require changing if a given recommendation were implemented. While a substantial effort was expended in compiling a listing of affected documents, there is no assurance that the list is complete. However, the Committee believes that all the significant documents are included.

The priorities appearing in Table 10.1 recognize the level of work under way by industry as well as by the NRC. If a given problem is nearing resolution through such work, the Committee uses lower priorities than the issue would warrant on purely technical bases.

Table 10.1

SUMMARY OF RECOMMENDATIONS

Category & Rank Order	Recommendation	Documents Requiring Change
A-1 (Vol.3)	Use leak before break (LBB) rather than the DEGB so that terminal and intermediate breaks would be eliminated when certain acceptance criteria are met. This would lead to exclusion of dynamic effects such as pipe whip, jet impingement, and subcompartment pressurization. The major impact would be on General Design Criterion 4 - <u>Environmental and Missile Design Bases</u> . The requirement to postulate arbitrary intermediate breaks should be eliminated.	10 CFR Part 50 (Appendix A, GDC-4, -30, -31, -32) SRP 3.6.2 R.G. 1.46
A-2 (Vol.2)	Modify seismic damping values currently used in seismic design. The suggested values have been incorporated into ASME-III and accepted by NRR on a case-by-case basis. This modification could lead to changes in support design and spacing and consideration of nozzle loads as well as reducing the number of snubbers.	R.G. 1.61 SRP 3.9.2
A-3 (Vol.2)	Decouple OBE from SSE.	10 CFR Part 100 (Appendix A)
A-4 (Vol.1)	Replace 316SS or 304SS in BWR recirculation piping with alloys resistant to IGSCC to eliminate this mode of pipe cracking. Possible types are 316NG, 304NG, 347NG.	10 CFR Part 50 (Appendix A, GDC-30 (possibly)) NUREG-0313 R.G. 1.44
A-5 (Vol.4)	Decouple seismic and LOCA events in systems where LBB is applicable.	SRP 3.9.3
A-6 (Vols.1,3)	Modify leak-detection requirements. This issue impacts BWR-IGSCC as well as Recommendation A-1.	NUREG-0313 Tech. Specs. R.G. 1.45
B-1 (Vol.4)	Emphasize operator awareness to minimize the incidence of dynamic events such as water and steam hammers.	NUREG-0927 Tech. Specs.
B-2 (Vol.1)	Modify BWR water chemistry to minimize potential for IGSCC.	NUREG-0313 Tech. Specs.

Table 10.1 (Continued)

Category & Rank Order	Recommendation	Documents Requiring Change
B-3 (Vol. 1)	Develop nondestructive examination (NDE) processes for weld overlays on IGSCC in austenitic piping. Unless these processes are developed, limit such overlays to two fuel cycles.	NUREG-0313
B-4 (Vol.1)	Require the use of residual stress reversal processes such as induction heating stress improvement to reduce probability of IGSCC in existing BWR piping systems.	NUREG-0313
B-5 (Vols.2,4)	Consider independent support motion as a design basis for seismic loads.	SRP 3.9.2
B-6 (Vol.2)	Develop criteria pertaining to nozzle flexibility.	SRP 3.9.2
B-7 (Vol.1)	Develop austenitic piping joint designs to maximize the probability of flaw detection using ultrasonics. In the short term, this should be done for replacement piping. The long-term goal should be to make all weldments accessible for volumetric NDE in the event new plants are designed.	NUREG-0313 10 CFR Part 50 (Appendix A, GDC-32)
B-8 (Vol.1)	Modify existing crack acceptance criteria to develop a position relevant to IGSCC with a crack depth and circumferential length that is acceptable to ASME-XI (IWB-3640) and to NRR.	NUREG-0313
B-9 (Vol.1)	Improve ultrasonic techniques to cover better crack sizing, personnel and procedures qualification, use of ASME-XI Code Case N-335, and ability to examine overlay cladding.	10 CFR Part 50 (Appendix A, GDC-32)
C-1 (Vol.2)	Spectral shifting for seismic design.	R.G. 1.122
C-2 (Vol.4)	Accept ASME OM3 as a basis for analyses of vibrations in piping.	Develop new Reg. Guide
C-3 (Vol.4)	Accept vibrational loads by either analyses or testing or a combination of both.	SRP 3.9.2 SRP 3.9.3 R.G. 1.92
C-4 (Vol.4)	Permit the use of nonlinear analyses for dynamic loads (other than seismic) in handling support piping gaps.	SRP 3.9.1 SRP 3.9.2

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APPENDIX A

NRC PIPING REVIEW COMMITTEE MEMBERS, CONSULTANTS, AND INTERNATIONAL REVIEW TEAM

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APPENDIX B

GENERIC ISSUES OF NUREG-0933 IMPACTED BY THIS REPORT

<u>Issue No.</u>	<u>Description</u>
A-1	Water Hammer
A-2	Asymmetric Blowdown Loads on Reactor Primary Coolant Systems
A-7	Mark I Long-Term Program
A-8	Mark II Containment Pool Dynamic Loads - Long- Term Program
A-13	Snubber Operability Assurance
A-14	Flaw Detection
A-18	Pipe Rupture Design Criteria
A-21	Main Steamline Break Inside Containment - Evaluation of Environmental Conditions for Equipment Qualification
A-22	PWR Main Steamline Break - Core, Reactor Vessel, and Containment Building Response
A-40	Seismic Design Criteria - Short-Term Program
A-41	Long-Term Seismic Program
A-42	Pipe Cracks in Boiling Water Reactors
B-6	Loads, Load Combinations, Stress Limits
B-11	Subcompartment Standard Problems
B-16	Protection Against Postulated Piping Failures in Fluid
B-51	Assessment of Inelastic Analysis Techniques for Equipment and Components
14	PWR Pipe Cracks
86	Long-Range Plan for Dealing with Stress Corrosion Cracking in BWR Piping
89	Stiff Pipe Clamps

<p><small>NRC FORM 335 (2-84) NRCM 1102, 3201, 3202</small></p> <p style="text-align: center;">BIBLIOGRAPHIC DATA SHEET</p> <p><small>SEE INSTRUCTIONS ON THE REVERSE</small></p>		<p><small>U.S. NUCLEAR REGULATORY COMMISSION</small></p> <p><small>1 REPORT NUMBER (Assigned by TIDC add Vol No., if any)</small></p> <p style="text-align: center;">NUREG-1061 Volume 5</p>									
<p><small>2 TITLE AND SUBTITLE</small></p> <p>Report of the U.S. Nuclear Regulatory Commission Piping Review Committee Volume 5: Summary - Piping Review Committee Conclusions and Recommendations</p>		<p><small>3 LEAVE BLANK</small></p>									
<p><small>5 AUTHOR(S)</small></p> <p>Piping Review Committee</p>		<p><small>4 DATE REPORT COMPLETED</small></p> <table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 50%; text-align: center;"><small>MONTH</small></td> <td style="width: 50%; text-align: center;"><small>YEAR</small></td> </tr> <tr> <td style="text-align: center;">March</td> <td style="text-align: center;">1985</td> </tr> </table> <p><small>6 DATE REPORT ISSUED</small></p> <table border="1" style="width: 100%; border-collapse: collapse;"> <tr> <td style="width: 50%; text-align: center;"><small>MONTH</small></td> <td style="width: 50%; text-align: center;"><small>YEAR</small></td> </tr> <tr> <td style="text-align: center;">April</td> <td style="text-align: center;">1985</td> </tr> </table>		<small>MONTH</small>	<small>YEAR</small>	March	1985	<small>MONTH</small>	<small>YEAR</small>	April	1985
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<p><small>13 ABSTRACT (200 words or less)</small></p> <p>This document summarizes a comprehensive review of NRC requirements for Nuclear Piping by the U.S. NRC Piping Review Committee. Four topical areas, addressed in greater detail in Volumes 1 through 4 of this report, are included: (1) Stress Corrosion Cracking in Piping of Boiling Water Reactor Plants, (2) Evaluation of Seismic Design, (3) Evaluation of Potential for Pipe Breaks, and (4) Evaluation of Other Dynamic Loads and Load Combinations. This volume summarizes the major issues, reviews the interfaces, and presents the Committee's conclusions and recommendations for updating NRC requirements on these issues. This report also suggests research or other work that may be required to respond to issues not amenable to resolution at this time.</p>											
<p><small>14 DOCUMENT ANALYSIS - a KEYWORDS/DESCRIPTORS</small></p> <p>piping pipe break pipe crack seismic design IGSCC</p> <p><small>b IDENTIFIERS/OPEN ENDED TERMS</small></p>		<p><small>15 AVAILABILITY STATEMENT</small></p> <p style="text-align: center;">Unlimited</p> <p><small>16 SECURITY CLASSIFICATION</small></p> <p style="text-align: center;">(This page) Unclassified (This report) Unclassified</p> <p><small>17 NUMBER OF PAGES</small></p> <p><small>18 PRICE</small></p>									

LIST OF ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ASME	American Society of Mechanical Engineers
BWR	Boiling Water Reactor
CFR	Code of Federal Regulations
CRGR	Committee to Review Generic Requirements
DEGB	Double-Ended Guillotine Break
ECCS	Emergency Core Cooling System
EDO	Executive Director for Operations
FEM	Finite Element Models
GDC	General Design Criterion
HAZ	Heat Affected Zone
IE	Office of Inspection and Enforcement
IGSCC	Intergranular Stress Corrosion Cracking
ISHI	Induction Heating Stress Improvement
ISI	Inservice Inspection
LBB	Leak Before Break
LOCA	Loss-of-Coolant Accident
NDE	Nondestructive Examination
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
OBE	Operating Basis Earthquake
OELD	Office of the Executive Legal Director
PVRC	Pressure Vessel Research Committee
PWR	Pressurized Water Reactor; also Pipe Whip Restraint
RES	Office of Nuclear Regulatory Research
SAW	Submerged Arc Weld
SCC	Stress Corrosion Cracking
SMAW	Shielded Metal Arc Weld
SRP	Standard Review Plan
SRV	Safety Relief Valve
SSE	Safe Shutdown Earthquake
USI	Unresolved Safety Issue
UT	Ultrasonic Test
WRC	Welding Research Council