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Volume 4

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

Evaluation of Other Dynamic Loads and Load Combinations

U.S. Nuclear Regulatory Commission

Prepared by the Other Dynamic Loads and Load Combinations Task Group

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Manuscript Completed: September 1984
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The Other Dynamic Loads and Load Combinations Task Group

**U.S. Nuclear Regulatory Commission
Washington, D.C. 20555**

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FOREWORD

Six topical areas were covered by the Task Group on Other Dynamic Loads and Load Combinations as described below:

1. Event Combinations, dealing with the potential simultaneous occurrence of earthquakes, pipe ruptures, and water hammer events in the piping design basis.

2. Response Combinations, dealing with multiply supported piping with independent inputs, the sequence of combinations between spacial and modal components of response, and the treatment of high frequency modes in combination with low frequency modal responses.

3. Stress Limits/Dynamic Allowables, dealing with inelastic allowables for piping and strain rate effects.

4. Water Hammer Loadings, dealing with code and design specifications for these loadings and procedures for identifying potential water hammer that could affect safety.

5. Relief Valve Opening and Closing Loads, dealing with the adequacy of analytical tools for predicting the effects of these events and, in addition, with estimating effective cycles for fatigue evaluations.

6. Piping Vibration Loads, dealing with evaluation procedures for estimating other than seismic vibratory loads, the need to consider reciprocating and rotary equipment vibratory loads, and high frequency vibratory loads.

NRC staff recommendations for regulatory changes and additional study appear in Sections 1 through 5 of this report. Section 5 combines the topical areas "Relief Valve Opening and Closing Loads" and "Piping Vibration Loads" in a single section.

ACKNOWLEDGMENTS

The NRC staff that defined the scope of issues, monitored our consultants, and prepared the staff recommendations were Goutam Bagchi, John Fair, Mark Hartzman, John O'Brien, and Al Serkiz. Our consultants, who were responsible for preparing position papers to be used, in part, to draft recommendations, were Bob Guenzler of INEL, Bob Kennedy of SMA, John Stevenson of Stevenson and Associates, and Everett Rodabaugh of E. C. Rodabaugh Associates, Inc. Al Serkiz of the NRC, besides being a member of the Task Group, also served as a consultant and prepared a position paper. Our industry coordinators were Sid Bernsen of the Bechtel Energy Corporation and Don Landers of Teledyne Engineering Services. Additionally, comments were received from Spence Bush of Review & Synthesis Associates, Paul Bezler of Brookhaven National Laboratory, Tom Esselman of Westinghouse, Louis Nieh of Stone and Webster Engineering Corporation, and Norm Edwards of NUTECH Engineers. Pat Higgins of the Atomic Industrial Forum assisted the Task Group as well. Foreign information was obtained from sources in Belgium, Canada, France, Italy, Japan, Sweden, and the Federal Republic of Germany. This information is presented in Appendix A to this report.

Finally, the Task Group wishes to thank Shirley Poms for rendering word processing services and Louise Gallagher for the technical editing.

EXECUTIVE SUMMARY

This report partially fulfills and complies with the requirements of the July 13, 1983 memorandum from Harold Denton and Robert Minogue to William Dircks entitled "Proposal for Reviewing NRC Requirements for Nuclear Power Plant Piping." In accordance with that memorandum, the Task Group on Other Dynamic Loads and Load Combinations has developed recommendations for revising present requirements for nuclear reactor piping and has made suggestions for additional effort to respond to issues not currently amenable to resolution. This summary provides recommendations for modifying present regulatory standards in general terms and, in addition, offers guidance on potentially useful future research.

More detailed information and qualitative value impacts of the recommendations are found in Sections 1 through 5, as well as in Appendix B. Particular sections of the Standard Review Plan (SRP), regulatory guides, and sections of 10 CFR are cited in the latter parts of these sections.

Recommended Revisions to NRC Criteria

The principal recommendations of the Task Group are as follows:

1. The event combination of earthquake and double-ended guillotine rupture of primary system piping in Westinghouse and Combustion Engineering reactor systems should be eliminated from the design basis.

2. 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 potential for water hammer and water/steam hammer should be given proper consideration in the development of these design specifications.

3. The independent support motion method should be allowed as an option to the uniform support motion method for multiply supported piping with independent inputs. Also, algebraic combinations should be used for high frequency modes in place of the present square root of the sum of the squares (SRSS) technique, and any combinational sequence between modal and spacial components should be allowed.

4. A major shift to inelastic analysis of piping systems using strain limits for piping analysis is not justified at this time. No change is recommended in the current SRP procedure, which allows the inelastic piping analysis on a case-by-case basis.

5. The SRP should allow increases in minimum design yield strength greater than 10 percent due to strain rate effects for pipe whip restraint design when an adequate basis is provided.

6. The responsibility for including water hammer in the design specification should rest with the plant owner or applicant and the NRC

should not be called upon to define an all-inclusive checklist. Efforts to reduce and minimize the incidence of unanticipated water hammer should continue with emphasis on operator training and awareness of potential water hammer occurrence.

7. 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.

8. The SRP should allow and accept the conduct of vibration test programs in accordance with ANSI/ASME OM3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems."

9. Explicit reference to vibrational loads from reciprocating and rotating equipment should be made in the SRP.

10. The SRP should indicate that it is acceptable to perform the evaluation of vibratory loads transmitted by supporting structure to piping by analysis, testing, or a combination of analysis and testing.

Recommendations For Additional Studies

The following represents potentially useful areas of future research:

1. Work should be completed on Babcock and Wilcox and General Electric reactor coolant loop piping to learn if earthquake and reactor coolant loop double-ended guillotine break may be excluded for these vendors.

2. 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.

3. A replacement pipe rupture for combination with the safe shutdown earthquake should be developed.

4. Investigations should be undertaken to establish the transition frequency between high and low frequency when implementing the algebraic summation rule for high frequency modal combinations.

5. The impact of phase correlations between support groups on the recommendations for the independent support motion method should be better clarified.

6. Additional effort is warranted on appropriate methods for calculating the effect of closely spaced modes.

7. Additional benchmarking of piping response to thermal-hydraulic transients will help to reduce uncertainties.

8. It should be determined whether the recently approved PVRC (Pressure Vessel Research Committee) pipe damping values for seismic design can be extended to higher frequency (33 to 100 hertz) vibratory loadings.

1. STAFF RECOMMENDATIONS ON EVENT COMBINATIONS

1.1 Introduction

This section deals with proposed revisions to NRC criteria and suggested research on Event Combinations for nuclear reactor piping. Event Combinations refers to the assumed or postulated concurrence 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 dead weight loads will always be assumed to act concurrently with the infrequent and intermittent events and are not further discussed herein. The events of principal concern are earthquake (OBE and SSE), pipe rupture (including pipe whip and jet impingement), and water hammer. Piping vibration loads and safety relief valve loads are treated in Section 5 of this report.

1.2 Historical Development of Technical Issues

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 1960's, 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 their support systems. The evolution of seismic design requirements over the last two decades has led to increases in seismic stresses by a factor of two to three. Likewise, large increases in the calculation of pipe rupture loads have taken place since the 1960's (estimated at a factor of between 1.5 and 2.5). Thus, design 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 the primary system piping of PWRs, 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. Progress in advancing the leak-before-break hypothesis and increasing confidence in its applicability are leading to a situation wherein serious consideration is being given to excluding certain pipe ruptures entirely from the design basis. Should this occur, event combinations involving these events automatically vanish.

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 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. Therefore, recognition of water hammer potential should be maintained in the preparation of system design specifications and plant operating procedures and in operator training. Section 4 of this report discusses this phenomenon, underlying causes, and systems affected.

1.3 Summary and Assessment of Available Information

Both deterministic and probabilistic advanced fracture mechanics evaluations for PWR primary system piping indicate that fatigue crack growth from all transient sources, including earthquakes, will not lead to a double-ended guillotine rupture. Studies of indirect sources of double-ended guillotine rupture in which a seismically induced failure elsewhere in the plant causes a pipe rupture in primary piping confirm the improbability of these events. The limited historical record supports these analytical results. Work to date has been limited to Westinghouse and Combustion Engineering reactor systems but is being extended at this time to Babcock and Wilcox PWR configurations and General Electric BWR reactor coolant loop piping. The methodology would be applicable to other nuclear power plant piping and has received the endorsement of the Advisory Committee on Reactor Safeguards. Additionally, in the Federal Republic of Germany, the double-ended rupture is no longer postulated for new PWR primary systems.

With respect to water hammer events, approximately 150 have occurred during the last 20 or so years, the majority being relatively minor or within the design basis. The likelihood that some of these water hammer events would occur during a major earthquake or a plant dynamic event is not small. The staff view is that anticipated water hammer events should be combined with earthquakes and plant dynamic events (an SRSS procedure is an acceptable method). Benefits and uniformity would result from the preparation of checklists to identify all water hammer events that may affect plant safety in the development of system design specifications.

With respect to unanticipated or accidental water hammers, these events are driven by the same underlying phenomena or operator actions attributable to the anticipated class. Operator awareness and training have been stressed and are recommended for avoiding such water hammers. Water hammer in the PWR secondary system(s) is the most significant such unanticipated water hammer, and the associated loads can be large. Use of bounding-type analyses for such load estimates leads to massive pipe supports. On the other hand, such water hammer occurrences have not resulted in major pipe ruptures (with the exception of Indian Point, Unit 2, in 1972) despite repeated recurrences. Damage to pipe hangers and pipe supports has been the principal effect. Rather than requiring additional load combinations, the staff recommends that continued emphasis on proper plant operating procedures and operator training should be maintained. At present, the staff opinion is that loads from unanticipated water hammer should not be included in the design basis but that continued emphasis should be devoted to reducing the incidence and effects of unanticipated water hammer. Water hammer considerations have already been incorporated in designing reactor system features (e.g., J-tubes, vacuum breakers, keep-full systems) for avoiding and minimizing water hammer occurrences.

1.4 Recommendations for Revisions to Present NRC Criteria

The following general revisions are recommended:

- 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. The staff emphasizes that it believes only evidence on primary circuit piping exists at this time. This recommendation influences plants already licensed in that they may now take credit for improved safety margins resulting from the relaxed criteria. Definite information for Babcock and Wilcox and General Electric reactors does not exist but is now being developed. Requirements for equipment qualification, ECCS performance, and containments are not affected by this revision. Replacement criteria for the event combination of pipe rupture and safe shutdown earthquake are addressed in NUREG/CR-1061, volume 3, section 10.6.

- o With respect to water hammer, these 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. For clarification, it should be noted that the potential for water hammer and water/steam hammer occurrence should also be given proper consideration in the development of design specifications. (See Section 4 of this report, "Staff Recommendations on Water Hammer Loading," for additional information on water hammer.)

- o Regulatory Guide 1.48 should be withdrawn since updated guidance is now provided in SRP Section 3.9.3, Appendix A, for the material covered by the regulatory guide.

1.5 Recommendations for Additional Study

The Task Group recommends the following as high-priority fields of investigation:

- o Work should be completed on Babcock and Wilcox and General Electric (Mark I) reactor coolant loop systems to learn if the leak-before-break hypothesis can be extended to these vendors and 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 III) may be considered.

- 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.

1.6 Qualitative Value Impacts of Recommended Revisions

Excluding the combination of SSE and the reactor coolant loop double-ended guillotine break from the design basis will have a large impact on the perceived reliability and safety margins of reactor internals, heavy component supports and systems, and components and structures inside the containment. In the event that the seismic hazard is increased or design deficiencies are discovered in operating plants, margins may still be shown to exist without undertaking any plant modifications. For any future plants, relaxed and more realistic design standards will prevail leading to simpler and less costly designs. On the other hand, the Task Group recommendations on water hammer do not impose any new requirements although encouragement of checklists may enhance safety if these checklists lead to the identification of water hammer events that warrant consideration in design.

2. STAFF RECOMMENDATIONS ON RESPONSE COMBINATIONS

2.1 Introduction

This section of the Task Group report treats questions regarding the use of independent support motion (ISM) methods in place of the presently approved uniform response spectrum (URS) techniques specified in SRP Section 3.9.2. Additionally, issues relating to the sequence of combinations between directional and modal components and to the treatment of high frequency modes are included.

2.2 Historical Development of Technical Issues

The NRC position on multiply supported piping with independent seismic inputs was developed at a time (during the early 1970's) 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 the following conservative assumptions:

1. A single uniform response spectrum that enveloped all the independent response spectra applied to the different support groups was used.

2. With peak group displacements occurring at the same moment, these peak displacements were combined in the most unfavorable way to calculate the seismic anchor motion (pseudostatic) component of seismic response.

3. The inertial and pseudostatic response was absolutely combined to obtain the total response.

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.

An item that was not addressed during the early 1970's is the combinational sequence between modal and directional components of piping response. This combinational sequence is a consideration only when closely spaced modes comes into play, under which conditions combining directional components first will give a more conservative result. This issue is not addressed in the SRP or in regulatory guides but is treated in branch technical positions. Recent studies have shown that in some situations the choice of one sequence over another leads to maximum differences in response estimates of about 20 percent. However, in the majority of practical cases where this item was addressed, the results show only minor differences in final responses. Therefore, present thinking is that this issue is more an academic one than an issue seriously impacting safety.

Difficulties with combining high frequency modes by the square root of the sum of the squares (SRSS) approach were pointed out in 1979 in the course of responding to Task Action Plan A-40. Here high frequency modes means modes beyond the maximum input excitation frequency where dynamic amplification is essentially zero. For this situation, the high frequency modes are all nearly in-phase with the input motion, and, as a result, in-phase with each other. This implies that the algebraic combination of high frequency modal responses is appropriate.

2.3 Summary and Assessment of Available Information

Brookhaven National Laboratory (BNL) in a report prepared for the Nuclear Regulatory Commission entitled "Alternate Procedures for the Seismic Analysis of Multiply Supported Piping Systems," NUREG/CR-3811, May 1984, recommended that "The independent support motion response spectrum method should be certified as acceptable for the evaluation of the dynamic component of response." This recommendation was endorsed by this Task Group's consultant and the NRC staff -- however, with a significant exception. BNL (with support from NUTECH) advocated that combinations between support groups be by the use of the SRSS rule. The NRC staff and our consultant recommended the absolute sum rule instead. Westinghouse offered the view that absolute summation should be implemented "unless the groups are from different structures (or if from the same structure, they can be shown to be phase uncorrelated), then SRSS should be used." For the dynamic and pseudostatic component of response, our consultant and BNL both endorse a newly developed procedure called grouping by attachment points (BNL offers an additional option, grouping by elevations, for preliminary design). In this grouping procedure, structural support points that are attached to a rigid floor or structure (so that the same translationary motion, without rotation, is experienced) are considered as one group of supports. Supports should not be considered rigid for any frequency. After the individual group responses are determined, they are combined by the absolute sum method. The aforementioned BNL NUREG report demonstrates that significant reductions in predicted responses can be achieved without leading to unconservatisms. It is the consensus of all parties that the total response should be obtained by combining the inertial and pseudostatic responses by the SRSS rule, which would be a relaxation over the present absolute sum rule.

Evaluations of the issue on the sequential combination of directional and modal components indicate that it is relatively insignificant and our recommendations reflect this observation.

Available evidence also strongly supports the algebraic summation of high frequency modes or a procedure equivalent to algebraic summation. After the high frequency modes are combined by algebraic summation, this quantity is combined with the response to lower frequency modes by the SRSS rule to obtain the total response.

2.4 Recommendations for Revisions to Present NRC Criteria

There are three principal recommendations for the material of this section as follows:

1. Independent Support Motion Method

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 should be implemented under the following rules for response combination.

a. For Inertial or Dynamic Components

(1) Group responses for each direction should be combined by the absolute sum method.

(2) Modal and directional responses should be combined by the SRSS method without considering closely spaced frequencies.

b. For the Pseudostatic Components

(1) For each group, the maximum absolute response should be calculated for each input direction.

(2) These should then be combined by the absolute sum rule.

(3) Combination of the directional responses should be by the SRSS rule.

c. For the Total Response

Dynamic and pseudostatic responses should be combined by the SRSS rule.

2. Sequence of Combinations

Any sequence may be selected between spacial and modal components, that is, modes may be obtained first or spacial components may be combined first. The reason is that consideration of closely spaced frequencies need not be taken into account.

3. High Frequency Modes

Algebraic combinations should be used for high frequency modes as described in the position paper on Response Combinations in Section B.2 of Appendix B to this report. The high frequency modes should be combined with low frequency modes by the SRSS rule.

The procedure for independent support motions should be added to SRP Section 3.9.2. Regulatory Guide 1.92 should be modified to reflect the inclusion of the high frequency modal effects.

2.5 Recommendations for Additional Study

The studies delineated below reflect the Task Group's view as to fruitful fields of future effort.

- Investigations should be undertaken to establish the transition frequency between high and low frequency when implementing the algebraic summation rule for high frequency modes.

- Additional effort on phase correlation between groups and the impact on the BNL recommendations is needed. BNL, using the Lawrence Livermore National Laboratory (LLNL) data from Zion, were unable to quantify the influence of phase correlations. Thus, uncertainties exist as to potential limitations on the recommendations.

- Additional effort is warranted on appropriate methods for calculating the effect of closely spaced modes.

2.6 Qualitative Value Impacts of Recommended Revisions

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 (by a factor of two or more) 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. On the other hand, for very stiff piping systems, the high frequency mode combination recommendation could result in higher response predictions under certain conditions. The degree to which these response predictions increase depends on the importance of the high frequency modes in deciding the total response.

3. STAFF RECOMMENDATIONS ON STRESS LIMITS/DYNAMIC ALLOWABLES

3.1 Introduction

This section of the report 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. These issues are currently addressed in Appendix F to the ASME Boiler and Pressure Vessel Code.

3.2 Historical Development of Technical Issues

Criteria for inelastic system analysis stress or strain limits for ASME Class 1 components have been included in the ASME Boiler and Pressure Vessel Code since the incorporation of Appendix F. Although these criteria could be used for piping analyses, the standard industry practice has been to use the special stress limits for piping in conjunction with Code Equation 9 and an elastic system analysis. Similar stress limits were also developed for ASME Class 2 and 3 analyses.

The stress limits for piping in Appendix F, as well as the stress limits for Class 2 and 3 piping, allow components to be loaded substantially above the material yield strength for many piping components. As stated in the accompanying position paper (Section B.3 of Appendix B), these limits could result in certain components being loaded above their theoretical limit moments. However, the limits were selected based on judgments that conservatisms existed in the application of the Code criteria that would preclude reaching the point of structural instability. The NRC staff, recognizing that the Code stress limits were high, developed a set of functionality criteria incorporated in Section 3.9.3 of the SRP to ensure that piping systems maintained dimensional stability when the higher Code limits were used.

In addition to the elastic piping analysis limits, the Appendix F criteria for inelastic analysis have been addressed in Section 3.9.1 of the SRP. The SRP requires a case-by-case review of stress-strain relationships and analytical procedures employed in the analyses.

Criteria for considering strain rate effects have been recently added to Appendix F to the Code. The criteria allows for the adjustment of the shape of the curve but does not increase the Code-allowable stresses. Use of the criteria as written would not result in any apparent benefit in terms of the load-carrying ability of a given component but would improve the accuracy of the system analysis.

3.3 Summary and Assessment of Available Information

Both criteria for inelastic allowables and criteria for consideration of strain rate effects are contained in the current Appendix F to the ASME Boiler and Pressure Vessel Code. In addition, SRP Section 3.9.1 requires case-by-case review for the application of inelastic component analysis.

In order to apply general strain criteria for inelastic analysis, strain limits that would result in a uniform margin of safety would first have to be developed, considering different component geometries and material properties, including weld properties. In addition to strain limits, inelastic computer codes for piping analysis would have to be developed and properly benchmarked. Based on these considerations, a major shift to inelastic analysis of piping systems using strain limits for piping analysis is not justified at this time. The current SRP procedure, which allows inelastic analysis on a case-by-case basis with appropriate justification, is adequate for current piping analyses.

The use of strain rate effects in piping system analyses would require more complex computer codes than are currently used in the industry. As discussed in the position paper (Section B.3 of Appendix B), most of the test data available today on strain rate effects is based on uniform tensile test specimens. Piping system analyses result in complex stress patterns in some components that would require consideration of three-dimensional effects. Therefore, the analysis on an entire system would be extremely complex, and the available test data might not be directly applicable. The most benefit obtained from the application of strain rate effects occurs during impactive-type loadings such as those involved with whip restraint design. Since the whip restraint is generally less complex than an entire piping system, consideration of strain rate effects would be practical for this application. Currently, SRP Section 3.6.2 allows a 10 percent increase in minimum specified design yield strength to account for strain rate effects. This should be changed to allow justification of higher values on a case-by-case basis.

3.4 Recommendations for Revisions to Present NRC Criteria

- No change in current NRC criteria for inelastic analysis stress or strain limits is recommended.
- Section 3.6.2 III. 2.a of the SRP should include a statement that allows increases in minimum design yield strength greater than 10 percent because of strain rate effects for pipe whip restraint design provided a report that includes a detailed description of the basis for the values and the analysis methods used for strain rate effects is submitted for review.

3.5 Recommendations for Additional Study

- None

3.6 Qualitative Value Impacts of Recommended Revisions

The recommended change in the SRP will have minimal impact since the position is already being implemented in the licensing review process.

4. STAFF RECOMMENDATIONS ON WATER HAMMER LOADING

4.1 Introduction

This section deals with staff recommendations regarding water hammer loading on piping components and fittings.

4.2 Historical Development of Technical Issues

Water hammers have occurred in nuclear power plants since the late 1960's; 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 1970's 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.

The staff (and its subcontractors) have carefully reviewed these occurrences and concluded that:

1. Total elimination of water hammer occurrence is not possible because inherent in the design of nuclear power plants is the possible existence of steam, water, and voids in the various plant systems. Experience shows that design inadequacies and operator- or maintenance-related actions have contributed about equally to initiating water hammer occurrences.

2. Proven design changes (e.g., use of J-tubes to minimize PWR steam generator water hammer and "keep-full" systems and vacuum breakers in BWRs) should be maintained.

3. Operator awareness to water hammer potential and training for avoidance should be stressed.

The staff's technical findings are reported in NUREG-0927 entitled "Evaluation of Water Hammer in Nuclear Power Plants-Technical Findings Relevant to Unresolved Safety Issue A-1." SRP Sections 3.9.3, Rev. 1, "ASME Code Class 1, 2, and 3 Components Supports and Core Support Structures"; 3.9.4, Rev. 2, "Control Rod Drive Systems"; 6.4.6, Rev. 3, "Reactor Core Isolation Cooling System (BWR)"; 5.4.7, Rev. 3, "Residual Heat Removal (RHR) System"; 6.3, Rev. 2, "Emergency Core Cooling System"; 9.2.1, Rev. 3, "Station Service Water System"; 9.2.2, Rev. 2, "Reactor Auxiliary Cooling Water Systems"; 10.3, Rev. 3, "Main Steam Supply System"; and 10.4.7, Rev. 3, "Condensate and Feedwater System," were revised to reflect staff findings and to maintain proven practices.

4.3 Summary and Assessment of Available Information

As noted above, NUREG-0927 reports the staff's technical findings regarding water hammer. Appendix B to this paper contains consultant position papers dealing with water hammer and the other dynamic loads.

4.4 Recommendations for Revisions to Present NRC Criteria

Designing for water hammer piping loads are dealt with in SRP Section 3.9.3, Appendix A, Rev. 1. Since water hammer occurrence cannot be predicted, the potential for such loads should be considered in preparing design specifications for normal operation, upset, and faulted conditions as defined in specified service-loading combinations identified for ASME Class 1 components and Class CS Support Structures per the ASME Boiler and Pressure Vessel Code, Section III, Div. 1. Table I of Appendix A to SRP Section 3.9.3 was modified as follows:

> "These events must be considered in the pipe-stress analysis and pipe-support design process when specified in the ASME Code-required design specification. The design specification should define the load and specify the applicable Code Service Stress Limit. For clarification, it should be noted that the potential for water hammer and water (steam) hammer occurrence should also be given proper consideration in the development of design specifications."

Thus, 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. The adequacy of these design specifications is, therefore, the key question when addressing dynamic loads (such as water hammer) and combined dynamic loads.

Because of the multidisciplinary nature of the problem, there does not exist a systematic and uniform treatment of water hammer, or other dynamic loads in developing design specifications, except for major events such as turbine stop valve closure, feedwater line break, and safety relief valve (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 these events.

NUREG-0927 contains summary tables identifying systems that have experienced water hammer, the underlying causes, and remedial actions that could be taken. Tables 4-1 and 4-2 (extracted from this report) are included for ease of reference.

Therefore, a checklist of water hammer design considerations could be developed. Underlying causes such as potential line voiding, steam pocket formation, flashing and unstable condensation due to entrapped condensate can be derived from Tables 4-1 and 4-2. Certain system design features have proved effective; certain systems have been more susceptible to water hammer. However, the wide variety in plant designs and operations works against development of such a generic checklist. Therefore, the responsibility of including water hammer considerations into design specifications must rest with the plant owner or applicant, and the NRC should not be called upon to define an all-inclusive checklist and institute adoption thereof. The revised SRP sections identified in Section 4.2 identify systems warranting review for water hammer design adequacy.

In summary, efforts to reduce or minimize the incidence of unanticipated water hammer should continue with an emphasis on operator training and awareness to potential water hammer occurrence. Since loads from likely unanticipated water hammer are similar to those that can be designed against, the design specifications dealing with upset, emergency, and faulted conditions should be used to deal with such occurrences. The proper development of design specifications rests with the plant designers.

4.5 Recommendations for Additional Study

Additional studies or research based principally on water hammer occurrence postulates are not warranted. Any proposed experimental programs should be preceded by properly structured analytical studies that would define the extent and magnitude of postulated problems. Studies in the following areas would be helpful:

- The sensitivity of piping supports to dynamic loads (e.g., vibratory, SRV, water hammer) and determination of excess design margins, etc., for various piping systems (treat PWRs and BWRs as two different classes).

- Evaluation of combined load effects on degraded (or flawed) piping coupled with dynamic loads (such as water hammer). Such studies would shed light on where emphasis should be placed in developing design specifications, as well as providing an analytical basis for determining which code design requirements warrant reconsideration. For example, the recently reported LLNL studies on "stiff" versus "flexible" piping (see NUREG/CR-3718) might warrant an extension to evaluate all postulated dynamic load effects singularly and then in combination, thereby providing a basis for recommending load combinations.

4.6 Qualitative Value Impacts of Recommended Revisions

Since no additional restrictions are being proposed for water hammer loads in combination with other loads, the result is a zero impact to the industry.

TABLE 4-1

BWR SYSTEM WATER HAMMER CAUSES AND PREVENTIVE MEASURES

SYSTEM	PRIMARY CAUSES OF WATER HAMMER	PREVENTIVE MEASURES* DESIGN	PLANT OPERATION
RHR	Voiding, steam bubble collapse	Void Detection (3.1), Keep-Full System (3.2), Venting (3.3)	Void Detection and Correction (3.1), Venting (3.2), Operating Procedures (3.12), Operator Training (3.11)
HPCI	Steam water entrainment, turbine inlet valve operation	No Opening Seal-In in Manual Mode (3.5a), Gradual Opening (3.5b)	Valve Opening Sequence (3.5c), Operator Training (3.11), Operating Procedures (3.12)
	Steam water entrainment drain pot malfunction	Proper Drain System Including Drain Pot Sizing and Level Verification (3.8)	Verification of Drain Pot Level (3.8), Operating Procedures (3.12)
	Turbine exhaust line bubble collapse	Exhaust Line Vacuum Breakers (3.7)	
	Pump discharge line voiding	Void Detection (3.1), Keep-Full System (3.2), Venting (3.3)	Void Detection and Correction (3.1), Venting (3.2), Operating Procedures (3.12), Operator Training (3.11)
Core Spray	Voiding steam bubble collapse	Void Detection (3.1), Keep-Full System (3.2), Venting (3.3)	Void Detection and Correction (3.1), Venting (3.2), Operating Procedures (3.12), Operator Training (3.11)
Essential Service Water	Voiding column separation	Void Detection (3.1), Keep-Full System (3.2), Venting (3.3), Open Loop Line Analysis (3.4)	Void Detection and Correction (3.1), Venting (3.2), Operating Procedures (3.12), Operator Training (3.11)

*Refers to section in NUREG-0927 providing details of preventive measures.

TABLE 4-1 (Continued)

SYSTEM	PRIMARY CAUSES OF WATER HAMMER	PREVENTIVE MEASURES* DESIGN	PLANT OPERATION
Main Steam	Steam hammer relief valve discharge	Valve Closure (3.9) and Relief Valve Discharge Loads (3.10)	
	Steam water entrainment		Operating Procedures (3.12), Operator Training (3.11)
Feedwater	Feedwater control valve instability	Feedwater Controller Design Verification (3.6a, b, and c)	
RCIC	Exhaust line steam bubble collapse	Exhaust Line Vacuum Breakers (3.7)	
Isolation Condenser	High reactor water level		Operating Procedures (3.12), Operator Training (3.11)
**Control Rod Drive	Rapid valve motion	Actuation Loads (3.14)	

**Control Rod Drive events have not been reported but have been analytically postulated.

TABLE 4-2

PWR SYSTEM WATER HAMMER CAUSES AND PREVENTIVE MEASURES

SYSTEM	PRIMARY CAUSES OF WATER HAMMER	PREVENTIVE MEASURES*	
		DESIGN	PLANT OPERATION
Feed-water	Feedwater control valve (FCV) over-sizing & instability	FCV-Design Verification (3.6)	
	Unknown and operator-error-induced steam bubble collapse		Operating Procedures (3.12), Operator Training (3.11)
Main Steam	Steam hammer (valve closure)	Include Valve Closure Loads in Pipe Support and Component Design Basis (3.9)	
	Relief valve discharge	Include Relief Valve Discharge Loads in Pipe Support and Component Design Basis (3.10)	
	Steam water entrainment, unknown		Operating Procedures (3.12), Operator Training (3.11)
Reactor Coolant (Pressurizer)	Relief valve discharge	Include Relief Valve Discharge Loads in Pipe Support and Component Design Basis (3.10)	
RHR	Voiding	Venting (3.3)	Operating Procedures (3.12), Operator Training (3.11)
ECCS	Voiding	Venting (3.3), Void Detection (3.1)	Operating Procedures (3.12), Operator Training (3.11)
CVCS	Steam bubble collapse or vibration		Operating Procedures (3.12), Operator Training (3.11)

TABLE 4-2 (Continued)

SYSTEM	PRIMARY CAUSES OF WATER HAMMER	PREVENTIVE MEASURES*	
		DESIGN	PLANT OPERATION
Essential Cooling Water	Voiding	Venting (3.1), Filling Essential Cooling Water (3.4), Analysis (3.4)	Filling Essential Cooling Water (3.4), Operating Procedures (3.12), Operator Training (3.11)
Steam Generator	Line voiding followed by steam bubble collapse	BTP ASB 10-2 Provisions (3.13): Top Discharge, Short Line Lengths, External Header (B&W Only)	BTP ASB 10-2 Provisions (3.13): Testing, Keeping Line Full. Automatic AFW Initiation

*Refers to section in NUREG-0927 providing details of preventive measures.

5. STAFF RECOMMENDATIONS ON PIPING VIBRATION LOADS

5.1 Introduction

Staff recommendations on consideration of vibratory loads to ensure structural and functional integrity of piping systems are based on a review of the current requirements and two consultant reports: (1) Position Paper on Vibration Load Considered As a Design Basis For Nuclear Power Plant Piping by J. D. Stevenson (see Section B.6 of Appendix B) and (2) Position Paper on Piping System Dynamic and Thermal Stress Response Induced By Thermal-Hydraulic Transients by R. C. Gunzler (see Section B.5 of Appendix B). The three types of vibratory loads considered here are (1) high stress and low cycles such as those caused by transient operation and seismic loading, (2) very high stress and a few cycles such as those caused by blast or shock type of loading, and (3) low stress and high cycles such as those caused by steady operation of rotating machinery. For type 1 vibratory loading, the earthquake component is well understood and is amply covered under the current criteria. However, hydrodynamic loads caused by plant transients such as discharge through pressure relief devices, anticipated water hammer loads, and loads caused by flow control devices are the subject of major emphasis for coverage in the proposed NRC requirements. The type 2 vibratory loading is appropriately covered by NRC requirements in SRP Section 3.9.3 under design basis pipe break loading. Type 3 loading does not pose a serious concern since piping systems are subjected to preoperational testing and the ASME Code uses a conservative stress limit for sustained loads.

5.2 Historical Development of Technical Issues

The need for consideration of hydrodynamic loads came from observations of relatively high magnitude of loading due to pressure suppression phenomenon in BWR plants. Requirements for the consideration of these loads were incorporated in the July 1981 revision to SRP Section 3.9.3. Also, the anticipated water/steam hammer loading was emphasized as a source of vibratory loading in the recently revised SRP Section 3.9.3.

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 SRP Section 3.9.2.

Piping system design for "high frequency" (33 to 100 hertz) vibratory loading is generally performed by using in-structure acceleration response spectra in much the same way as the design for earthquake loading, for which cut-off frequency for significant energy input is considered to be 33 Hz. Sufficient experience has been gathered in the United States regarding piping design for "high frequency" vibratory loading and in Europe with respect to aircraft impact loading for over a decade. There is reason to believe that the use of acceleration response spectra for piping design may lead to overestimating the actual loading.

5.3 Summary and Assessment of Available Information

Results of Kuosheng 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.

Also, the two consultant studies in this area reviewed both characterization of the loading as well as the response of piping system to the loading. The prevailing view is that anticipated vibratory loads should be accounted for by a combination of analysis and preoperational testing, and reliance must be placed on testing for consideration of unanticipated vibratory loads.

5.4 Recommendations for Revisions to Present NRC Criteria

For the type 1 vibratory loading, seismic design requirements are covered by the activities of the Task Group on Seismic Design. However, other vibratory loads such as hydrodynamic loading and water/steam hammer loading are addressed in the recommendations indicated below. Type 2 vibratory loading is adequately covered in the current NRC requirements, and no change is considered necessary. The use of inelastic response analysis methods will continue to be acceptable for dealing with type 2 vibratory loading. Type 3 vibratory loading has not been a source of concern for piping systems since conservative allowable stresses are used for sustained loading and consideration of vibration on aging is given in qualification of equipment, including piping nozzles.

Following are the specific changes recommended to the SRP:

(1) SRP Section 3.9.2, Page 3.9.2-5, article 1.

Add after line 8: The conduct of vibration testing program in accordance with the latest ANSI/ASME OM3 standard, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems," or an equivalent is considered for acceptability of the proposed vibration testing program.

(2) SRP Section 3.9.2, Page 3.9.2-5, article 1B.

Add: (5) Opening and closing of flow control valves

(3) SRP Section 3.9.2, Page 3.9.2-9f.

Add a new paragraph: For vibratory loads other than seismic and with significant loading in the frequency range of 33 to 100 Hz, it is acceptable to perform nonlinear analysis in order to account for gaps between pipes and pipe supports and the ability of the pipe supports to transmit vibration displacements of limited amplitude provided that verification of the predicted nonlinear response is made by conducting preoperational vibration testing.

(4) SRP Section 3.9.2, Page 3.9.2-15

Add a new item 7 as follows:

7. It is acceptable to perform the evaluation of vibratory loads transmitted by supporting structures to the piping by analysis, testing, or a combination of analysis and testing. Acceptability of analytical procedures and testing methods is discussed in subsections II.2.a and II.1, respectively.

(5) SRP Section 3.9.3, Page 3.9.3-2.

Delete the word "downstream" from the seventh line under item 2.

In addition to the above changes, a number of changes proposed in the consultant paper on consideration of vibration loads are also endorsed. These proposed changes are listed below for convenience.

(1) Reference I. Areas of Review 1.

In the 11th line, the following words should be added:

. . . withstand flow-induced <u>and reciprocating and rotating equipment</u> dynamic loadings . . .

(2) Reference I. Areas of Review

Add a new item 7 on page 3.9.2-4, the text of which is as follows:

7. A discussion should be provided that describes methods to be used to evaluate equipment and piping system to confirm their structural design adequacy when subjected to transient, accident, and extreme environment (other than seismic) vibratory loads. Such vibratory loads typically result from response of equipment and piping system supporting structures when such support structures are subjected to vibratory loads of significant amplitudes.

(3) Reference II. Acceptance Criteria 1.

Rewrite Section 1 as follows:

1. Relevant requirements of GDC <u>1, 2, 4,</u> 14, and 15 are met if vibration, thermal expansion, and dynamic effects testing are conducted during start-up functional testing for specified high- and moderate-energy piping, and their supports and restraints. The purpose of these tests is to confirm that the piping components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions that will be encountered during service as required by the Code and to confirm that no unacceptable restraint of normal thermal motion occurs. <u>Results of vibrational tests may also be used directly or by interpolation to confirm design adequacy of high- and moderate-energy piping, components, restraints, and supports to accident and extreme environmental loads.</u>

An acceptable test program to confirm the adequacy of the designs should consist of the following:

a. A list of systems that will be monitored. <u>This list may be limited to those systems based on experience that undergo significant thermal expansion, vibration, and dynamic effects.</u>

b. A listing of the different flow modes of operation and transients such as pump trips, valves closures, etc. to which the components will be subjected during the test. (For additional guidance see Reference 8.) For example, the transients associated with the reactor coolant system heat up tests should include, but not necessarily be limited to:

   (1) Reactor coolant pump start.

   (2) Reactor coolant pump trip.

   (3) Operation of pressure-relieving valves.

   (4) Closure of a turbine stop valve.

c. A list of selected locations in the piping system at which visual inspections and measurements (as needed) will be performed during the tests. For each of these selected locations, the deflection (peak-to-peak), <u>maximum velocity</u>, or other appropriate criteria, to be used to show that the stress and fatigue limits are within the design levels, should be provided.

d. A list of snubbers on system which experience sufficient thermal movement to measure snubber travel from cold to hot position.

e. A description of the thermal motion monitoring program, that is, verification of snubber movement, adequate clearances and gaps including acceptance criteria and how motion will be measured.

f. If vibration is noted beyond the acceptance levels set by the criteria of c. above, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If, during the test, piping system restraints are determined to be inadequate or are damaged, corrective restraints should be installed and another test should be performed to determine that the vibrations have been reduced to an acceptable level. If no snubber piston travel is measured at those stations indicated in d. above, a description should be provided of the corrective action to be taken to assure that the snubber is operable.

(4) Reference II. Acceptance Criteria 2.

Add the following new paragraph as the last paragraph of II.5, page 3.9.2-15.

High frequency (greater than 30 Hz) vibratory loads other than seismic, analyses methods for all Category I systems, and components equipment and their supports (including supports for conduit and cable trays and ventilation ducts) are reviewed. In addition, other significant effects that are accounted for in the high frequency vibratory load analysis such as nonlinear response and plastic stress levels in the materials are reviewed.

(5) Reference III. Review Procedures 1.

Rewrite Section 1 as indicated.

1. During the CP stage, the PSAR is reviewed to assure that the applicant has provided a commitment to conduct a piping steady-state vibration, thermal expansion and operational transient test program. The applicant may also commit a simulated accident or natural phenomena vibration test program in lieu of analysis.

(6) Reference IV. Evaluation Findings 2.

In the fifth line, add the words "or test" after analysis.

(7) Reference IV. Evaluation Findings 4.

In the sixth line, add the words "or test" after analysis.

5.5 Recommendations for Additional Study

The following are recommended for further studies and action:

- Characterization of hydrodynamic loads and the prediction of response of 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 by developing standard problems and acceptable solution bounds.

- A regulatory guide should be prepared to endorse the industry standard ANSI/ASME OM3, "Requirements for Preoperational and Initial Start-Up Vibration Testing of Nuclear Power Plant Piping Systems." Consideration should be given in developing the regulatory guide to supplementing the provisions of OM-3 to provide more restrictive acceptance standards for vibratory stresses in those limited areas where crack initiation from other service conditions can be anticipated. Also, the need should be considered for more rigorous evaluation of vibratory stresses to be performed in those areas, to the extent necessary to evaluate stress levels consistent with those limits. Supplementary acceptance standards for those areas should be based on the capability of such stresses to contribute to crack propagation rather than be based on the crack initiation potential for such stresses as in a normal fatigue design evaluation.

- o If the Task Group on Seismic Design proposes changes to piping damping values to be used in analytical modeling, a closure study should be made regarding their applicability to analytical evaluation of piping systems subjected to high frequency (33 to 100 Hz) vibratory loading.

5.6 Qualitative Value Impacts of Recommended Revisions

It is generally recognized that high frequency" (33 to 100 Hz) loading as currently evaluated by analytical techniques tends to overpredict piping response. By allowing nonlinear analysis with appropriate verification through properational 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.

This could lead to a reduction in the number of piping supports and perhaps an improved reliability of piping systems to accommodate such vibratory loads.

Proposed changes are likely to increase attention to preoperational testing for vibratory loads. As opposed to reliance on purely analytical methods of calculating usage factor for fatigue effects due to vibratory loads, the staff has relied on preoperational vibration testing in addition. It should be noted that the staff had always used criteria that are similar to the ANSI/ASME OM3 criteria for allowable vibration limits, and the latest version of the OM3 standard provides a convenient document for the industry to follow. 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.

APPENDIX A

FOREIGN INFORMATION

(This appendix was prepared under contract by John D. Stevenson for the Task Group on Other Dynamic Loads and Load Combinations.)

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POSITION PAPER
CURRENT FOREIGN REGULATORY PRACTICES
ASSOCIATED WITH DYNAMIC LOADS
(other than seismic and pipe break)
AND LOAD COMBINATIONS FOR NUCLEAR POWER PLANT PIPING

John D. Stevenson

AUGUST 1984

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1.0 INTRODUCTION

This paper presents and summarizes the response of a number of non-U.S. organizations regarding the current practices used in the design of nuclear power plant piping subjected to dynamic loadings and load combinations other than seismic and pipe break. The material presented herein was developed as a result of responses to a questionnaire sent to several foreign agencies both government and industry in December 1983, as shown in Appendix A attached hereto. The questionnaire was meant to be comprehensive with regard to piping design. Hence it included requests for information which are beyond the scope of the limited set of dynamic loads design considered in this position paper.

In addition, informal discussions were held in the offices of several of the organizations contacted with Dr. Stevenson in January 1984. The following organizations have responded in whole or in part to the questionnaire, or individuals from these organizations have discussed the questionnaire personally with Dr. Stevenson:

(1) Belgium-
  Electrobel
  Tractionel

(2) Canada-
  Ontario Hydro
  Atomic Energy of Canada Ltd., AECL

(3) France-
  Framatome
  French Electricity Authority -- EDF
  French Atomic Energy Commission, CEA

(4) Italy-
  Ansaldo Impianti

(5) Japan-
  "Procedures, Analysis and Research on Earthquake Resistant Design for Nuclear Power Plant" Presented at the Tadotsu User's Seminar, Tadotsu, Japan, May 1983

(6) Sweden-
  Swedish Nuclear Power Inspectorate

(7) Federal Republic of Germany-
  Kraftwerk Union, KWU
  TUV Rheinland
  Company for Reactor Safety, GRS

The position taken by the various organizations are summarized by topic areas defined herein and by country. The country positions are a composite of the information received and were transmitted to the various contributing organizations to solicit their review comments and correction as necessary. Information contained in this paper should in general be considered unofficial and does not necessarily reflect formal regulatory policy, or as is often the case in foreign countries, formal regulatory policy has not been formulated in the technical area discussed. The text of the country positions presented in this report follow as closely as possible the wordings or translation of the wording in the response to the questionaire.

Considerable detailed background information on dynamic load design of nuclear power plant facilities in several foreign countries is also contained in NUREG/CR-3020[1], and it is recommended that this reference be used in conjunction with this position paper.

2.0 DYNAMIC LOADS OTHER THAN SEISMIC AND PIPE BREAK

High frequency vibratory loads (greater than $20H_z$) are developed in piping from flow-induced and rotating equipment vibration, as well as vibratory response of structural supports to airplane crash, BWR suppression pool safety relief valve discharge, and postulated pipe break blow down loads. Water and Steam Hammer loads are another category of dynamic loads which are considered in piping design.

2.1 Belgium

2.1.1 High Frequency Vibration Loads

Aircraft crash (AOE) loads including vibratory response are considered as a design basis accident. Flow-induced vibration level is checked during preoperational test, but specific calculations are performed only for critical applications. Analytical methods for AOE treatment are similar to earthquake simulation; three directional AOE response spectra are considered; modal combination is performed using R.G. 1.92 rules modified with absolute sum of low frequency modes and directional combination is SRSS.

Testing methods for flow-induced vibration are as follows: Peak velocities and acceleration are measured with a full-range general - purpose accelerometer near any significant flow restriction. Results are compared with a general curve based on velocity requirements for pipe and stress and acceleration requirements for supports. When the limits are not met, induced stresses are estimated from the maximum velocity recorded and compared with the margins in the pipe stress report. No explicit fatigue evaluation is performed for piping vibration.

2.1.2 Water (Steam) Hammer Loads

In water systems (main feedwater e.g.), the water hammer problem is explicitly addressed and taken into account. In gas systems (incl. steam) the only influence being considered is a small overpressure. Water slugs in pressurizer relief and safety valve discharge systems are explicitly considered. The latest state of the art, including RELAP 5 -MOD 1 thermohydraulic calculations, are used to determine the corresponding piping and support loads. Hardware modifications (slug heating) are introduced to moderate these loads.

Analyses are performed on water systems to evaluate the possible water hammer effects (e.g. cavitation at pump section when pumps in series); layout criteria (e.g. no swing check valve in vertical run) and design provisions (use of equilibrium chambers, etc.) are ruled out. Systematic control of rapidly closing valves is performed with use of damping devices or pressure-operated safety valves on water circuits (letdown, shutdown cooling). In the conventional portions of the plant, administrative provisions guarantee the absence of gas bags in water circuits before start-up; moreover, all isolation valves are opened before pump start-up.

Water hammer events are classified as Service Level B and combined with other loads as required per ASME code.

2.2 Canada

2.2.1 High Frequency Vibration Loads

At the design stage no specific evaluation for high frequency pipe vibrations is generally carried out. Good engineering practice and experience is used as a guide in designing the pipe lines to minimize the adverse effect from high frequency vibrations. Emphasis is given to the inspection and observation by the field staff to identify and decide if trouble is anticipated during the life of the system from its everyday vibration, whatever the cause. A corrective action is then taken if necessary, and incorporated in other current and future designs.

For the latest CANDU design (i.e. Darlington GS) a more detailed approach to high frequency pipe vibrations is proposed. The following answers pertain mostly to this latest approach proposed by Ontario Hydro for Darlington GS.

For frequencies up to 30 Hz, flow-induced vibration analysis is performed. As well, pipe whip computations (pipe hitting containment or other structures) are carried out.

2.2.1.1 Setting Allowable Limits For Flow-Induced Vibrations.

The piping modes below 30 Hz, computed routinely for seismic analysis purposes, are used to provide an ensemble of possible distortions which provide known velocity maximum values for unknown distributions of moments within the structure. From the modal moment, one computes corresponding ASME code stress intensities (the maximum in the structure noted) corresponding to the maximium velocity. Hence, for the mode, one obtains a stress to velocity ratio.

Assuming that one wishes to limit the alternating stress intensity to a value likely to ensure design adequacy for, say, 10^{11} cycles, one uses the maximum stress to maximum velocity ratio in conjunction with this admissible stress and computes an admissible modal velocity.

In practice, the modal admissible velocities vary from mode to mode depending on whether the maximum moments for the mode happen to occur at a high stress indice point (such as a tee) or at a low stress indice point. The lowest modal admissible velocity encountered is the one which is deemed significant. To account for possible synergy between two modes of maximum at different points, but whose stresses may prove additive at the same point (a pessimistic assumption), this minimum admissible modal velocity is halved to produce the final system specific allowable velocity.

The numbers that emerge in this manner are purely for internal testing purposes. It is relatively easy for field staff to observe where a piping structure is exhibiting maximum vibration velocity and so decide whether trouble is anticipated during the life of the system from its normal vibrations, whatever their cause.

2.2.1.2 Analysis to Reduce Flow-Induced Vibrations

The method consists of frequency response analysis of the piping system for all modes up to 30 Hz. A flat spectrum is then applied at all elbows and tees and the responses are combined in a conservative manner. The method identifies the few modes which are most susceptible to excitation. During plant construction supports are installed, and left untied to the piping. During the plant operation, these supports are tied one-by-one (at hot conditions) as the need arises until measured vibration velocities are below the allowables established. It should be noted that the analysis described herein is a fatigue oriented calculation. Since it has greatest use on ASME III Class 2, 3, or B31.1 systems, combination with other fatiguing loads is often not possible. At present, there is no special cognizance of safety class for these purely formal computations.

2.2.2 Water (Steam) Hammer Loads

Originally, water hammer computations were limited to guarding against catastrophic pressure loading. For this purpose, a network analysis program for computing the pressure transient has been used. An Ontario Hydro internally developed program has been the workhorse of such a computation. The results of the analysis identify design changes to avoid pressure loading above allowables. Use has also been made of the RELAP5 and SURNAL programs in recent years.

Recently, anticipated rapid change in the pressure caused by a rapid change in fluid velocity is required to be considered in design. If the system operational characteristics cannot be changed, protective features are often installed to reduce water hammer loading due to rapid valve closure or pump trip to that below other operational transients.

Steam hammer loading has been computed successfully (as compared with experimental (commissioning) data. As a time history analytical procedure, it remains in the developmental stage. Maximum pressure surge on valves have been hand-computed for some time now to guard against catastrophic overpressurizations.

Protection against water hammer includes consideration of the following features:

(a) (1) Design against overpressurization as above: including sequencing of values;

(2) Avoidance of dead legs containing columns of liquid near boiling and other "common sense" design methods;

(3)
- -- vacuum breakers to let air into break voids
- -- acoustic filters
- -- system logic for valve opening and closure(sequencing and timing)
- -- butterfly valves gear-operated
- -- accumulators on small piping with solenoid valves
- -- spring loaded check valves
- -- cyclone to take air out of inlet
- -- "eaton" type wave arrestor
- -- one-way surge tanks to fill voids following pump trip prior to startup
- -- small vessel containing pressurized nitrogen with solenoid valve to inject after pump trip at sufficient pressure
- -- controlled air outlet from piping (small orifice on air release valves)
- -- control valve or pump by-pass to reduce flow into system during startup
- -- increased rotor inertia to avoid rapid pump rundown.

(b) & (c) It is deemed inappropriate to leave to human beings too much operational/administrative decision making which, if done erroneously, could lead to serious water hammer. Human error is designed out of the system as far as possible through the use of control system logic.

Computation of water hammer piping loads is through time history analysis and the extraction of maximum moment loads. These are combined with other applicable loadings (yielding their own sets of moments).

2.3 France

2.3.1 High Frequency Vibratory Loads

Aircraft considered in France are small and thus generate limited excitation for piping which are not analyzed for that effect.

For vibration induced by flow and valve or pump operation, the current position is to try, in the design, to avoid some effects such as cavitation, to evaluate some other effects as it is mentioned in Section 2.3.2, and for the rest to rely upon the hot functional tests to reveal latent problems, especially in small lines (vent, drain or instrumentation lines), recognizing the fact that high frequency vibration leads to quick failure (within hours or days) and is difficult to diagnose by a vibration test due to the short time and the number of system (valves, pumps) configurations to be considered.

2.3.2 Water (Steam) Hammer Loads

Taking into account water or steam hammer is not explicitely required by French regulations. The status is the following:

(1) - Steam hammer in main steam line due to rapid closure of the turbine stop valve or the MSIV has been studied on one plant and is not considered severe.

(2) - Steam hammer in feedwater line due to partial voiding of the line close to steam generator has been solved by installing J tubes on the feedwater ring inside the steam generator.

(3) - Pressurizer discharge line has been extensively tested in-situ and in laboratory with steam, with and without water seals upstream of the valve, with cold water, with hot water. Computer programs have been validated and these loads are taken into account in the design, when necessary.

Other relief and safety valves discharge lines see similar loadings.

(4) - Water hammer generated by the rapid closure of a valve (a check valve for example) in water filled piping are studied presently on a R&D basis.

(5) - Pressure waves generated by coupling of a valve elastic drive and a fluid column (veina, or slug), called elastic instability by some people, are evaluated when they cause damage (see Section 2.3.1), examples are: 1) the operation of spring loaded safety valves with water upstream, especially water slug which generate

self-maintained small amplitude valve stem displacement coupled with plane pressure waves in the fluid upstream, and 2) the possible elastic vibration of an air operated butterfly valve with water flow.

Problems have occurred in the past on steam and/or water spring loaded safety valves, during cold and hot functional tests, and on loops tests, this led some people to think of replacing these valves by pilot valves which had been used for a time by the French Navy. Extensive tests in laboratory and in-situ have been carried out and the solution is being implemented progressively on three and four loop plants in France, on the pressurizer discharge lines and on the RHR lines first.

In addition to the design considerations mentioned above, operational procedures are taken to minimize the potential for water hammer caused by rapid closure of a valve.

2.4 Italy

2.4.1 High Frequency Vibratory Loads

High frequency loading induced from airplane crash and BWR suppression pool response are considered in designs of piping. In general analytical rather than testing methods are used to determine and evaluate these loads. Fatigue analysis is performed for ASME-III Class 1, 2, and 3 piping.

2.4.2 Water (Steam) Hammer Loads

Water hammer events are limited primarily by administrative and operational procedures. Anticipated water hammer events are considered analytically in design.

2.5 Japan

2.5.1 High Frequency Vibratory Loads

High frequency vibration of piping induced by postulated external events such as aircraft or other missile impact is not evaluated in Japan, but loads induced by flow and valve or pump operation are considered in design of piping systems. Flow and valve or pump operation are considered in design of piping systems by past operating experience and testing.

2.5.2 Water (Steam) Hammer Loads

Design considers both aniticipated and unanticipated water and steam hammer phenomena where it is reasonable to consider these phenomena in the design. However, different resolutions such as preventions by operational procedures, administrative control, piping layout, etc., are also used as appropriate.

2.6 Sweden

2.6.1 High Frequency Vibration Load

High frequency excitations are considered in some cases (e.g. when evaluating effects of BWR suppression pool condensation oscillations). Analytical, finite element and experimental methods when required are used to design against vibration load effects. For Class 1 components fatigue analysis is considered according to ASME III. For other classes it is considered when required because of actual problems.

2.6.2 Water (Steam) Hammer Loads

In design, rapid valve closure is postulated to occur both under normal operation and after a pipe break outside the containment in steam and feed-water lines. Administrative and operating procedures as well as design features are used to control potential water hammer effects.

2.7 Federal Republic of Germany

2.7.1 High Frequency Vibration Load

Pipings (as well as other components) are designed against high-frequency cyclic loads (20 Hz). In these loads are comprised aircraft impact, gas cloud explosions, fluid reaction and impingement induced loads, suppression pool dynamic loads resulting from safety relief valve and postulated DBA's discharges in BWR systems, opening and closing of valves, and pump operations.

The design of piping systems in response to aircraft impact is based on the load assumptions of the RSK Guidelines for pressurized water reactors (Section 19.1):

- collision load-time-diagram (see Figure 2.1)
- area of impact: 7 m^2 circular
- angle of impact: normal to the tangent plane at the point of impact
- crash weight: 200 kN
- speed of impact: 215 m/s

As in the loading case for "earthquake", in the design of the piping systems against the vibrations caused by aircraft impact there are determined acceleration transient responses or floor response spectra for the corresponding site. In Figure 2.2 are shown the determinative points of impact (1 to 6) for a reactor building of a pressurized water reactor. The enveloping acceleration processes are determined for these points of impact and used for the design of the piping (except as permitted by the simplified procedure given below). Figure 2.3 shows a typical comparison of the floor response acceleration spectra for safe shutdown earthquakes, aircraft impact and gas cloud explosions.

The RSK Guidelines for pressurized water reactors also admit a simplified procedure for the loading case aircraft impact as proof of the stability of the components and systems in the reactor building. This proof is given by the assumption of a static substitute load upon the piping, resulting from a defined acceleration, in a horizontal and vertical direction up to 16 Hz. The degree of acceleration depends on the construction of the building. For KWU pressurized water reactors there can be taken an acceleration of 0.5 g for the reactor building. In the frequency range above 16 Hz, it must be made certain that the relative displacements between component and support can be elastoplastically absorbed up to 1mm.

2.7.2 Water (Steam) Hammer Load

Loads that occur due to the opening or the closing of a valve will be determined with the aid of dynamic analyses and taken into consideration in the design of the piping systems.

Therein, special attention is paid to free-swinging non-return valves which are installed in the emergency cooling and residual heat-removal systems. The calculations are based on load-time diagrams (e.g., square-wave impulse). The modal analysis is conducted with the aid of the direct integration.

Generally, structural measures are taken as protection against loads caused by water hammer. They consist of an adequate arrangement of support structures, of the installation of attenuation elements, of the timely limiting of the opening and closing of valves and flaps, and in the limitation of aperture angles in the case of non-return valves. Attention has to be paid during operation so that the pipings are vented. No administrative control measures are applied.

Water hammer loads are superposed in pipings with the operating pressure and the inherent weight of the piping. For supports, the dead weight of the pipings is superposed with the water hammer loads.

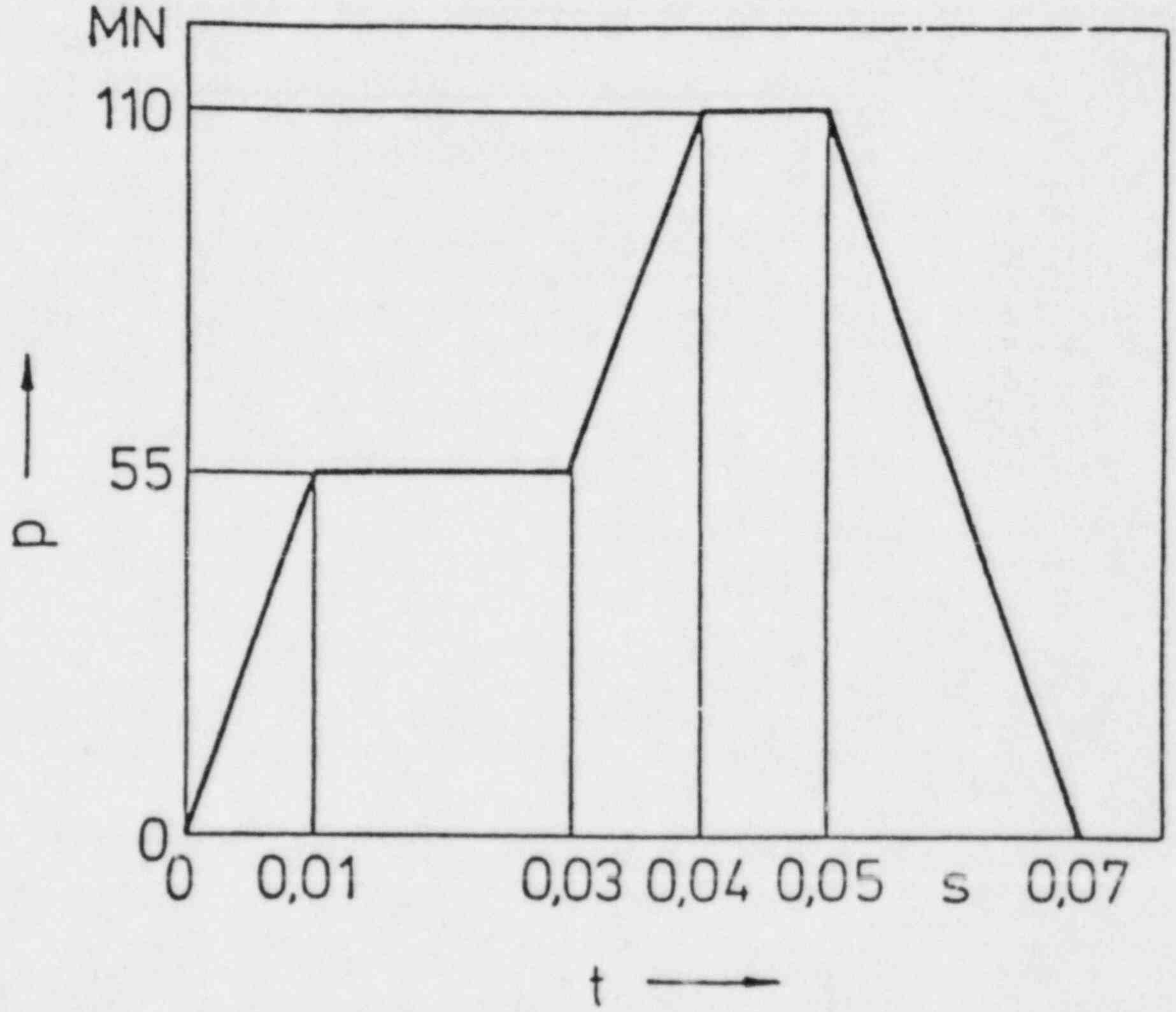

Figure2.1:Load-time diagram for the loading case aircraft impact

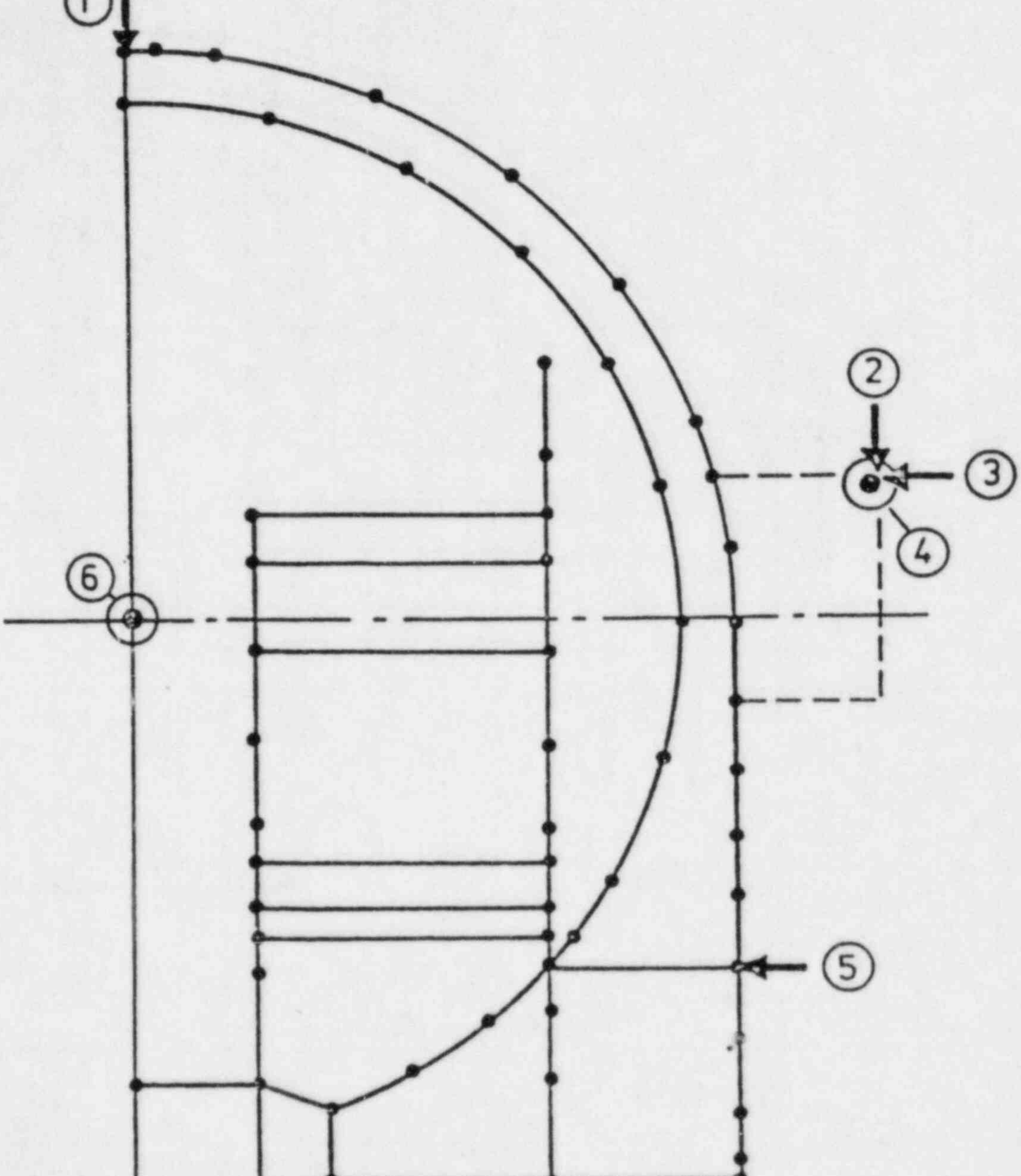

Figure 2.2: Load impact points of the reactor building of a pressurized water reactor in the load case "aircraft impact"

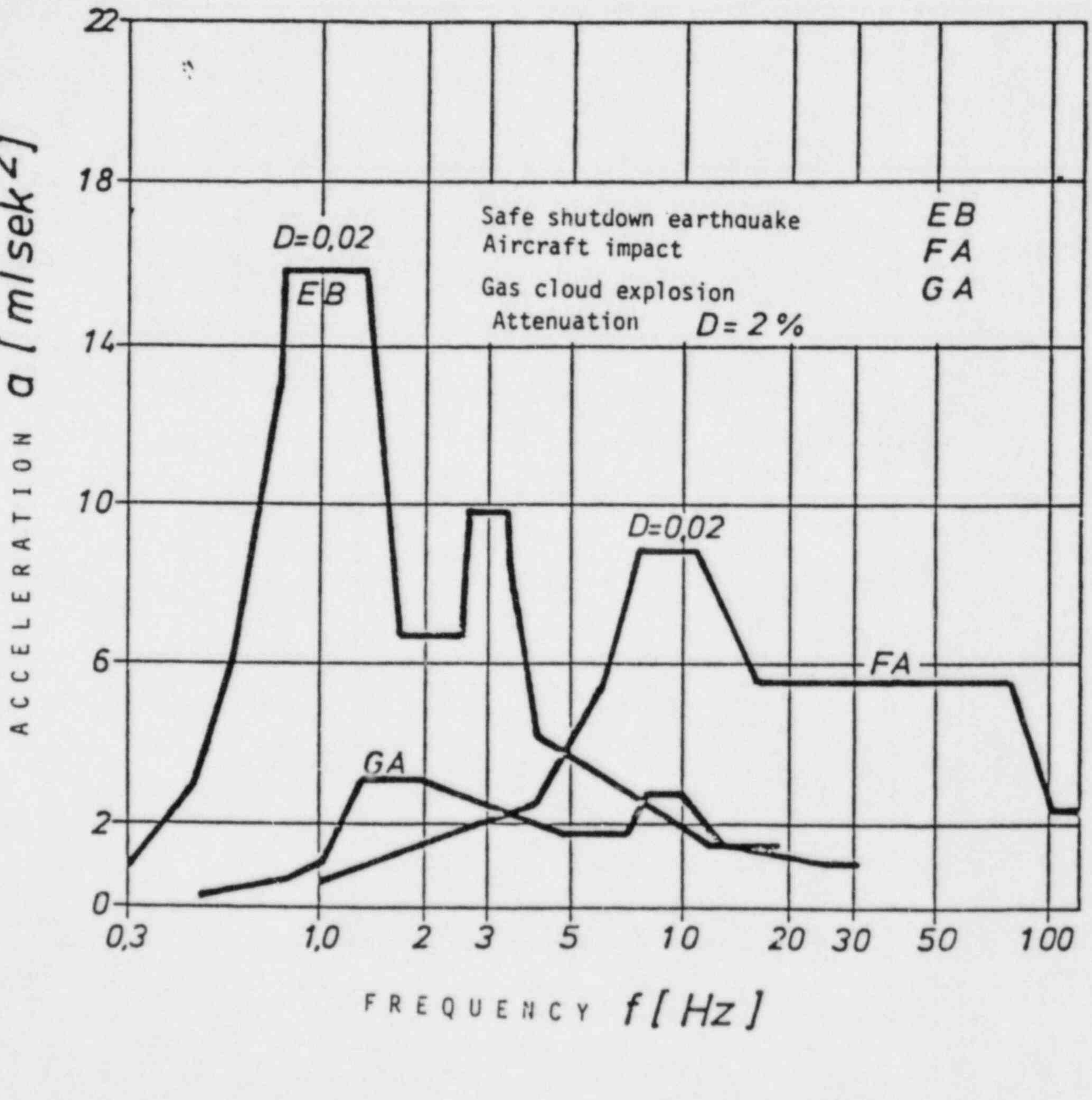

Figure2.3:floor response spectra

3.0 LOAD COMBINATIONS

3.1 Belgium

The criteria of the ASME code are used as the main guide, using the flowing specific guidelines:

(a)- the SSE and AOE are considered faulted conditions (Service Level D)

(b)- all pipe breaks are considered faulted conditions (Service Level D)

(c)- post-accidental operation of safety systems (e.g.ECCS) is considered a normal operation condition for the system, even if corresponding to a faulted plant condition (Service Levels A and B are used in design)

(d)- secondary stresses in piping systems are not limited for C and D Service Levels, but integrity of the supports is required for primary equilibrium purposes.

LOCA and SSE loads are combined on a SSRT basis.

3.2 Canada

General Load Combinations for applicable to Canadian nuclear power plants are summarized in Table K-1 of NUREG-1061 Vol. II..

Short-term loads, such as loads due to pipe break, water/steam hammer, seismic, etc., are normally categorized in other than ASME Service Level A conditions. These loads, therefore, get combined accordingly in the ASME Code equations. For earthquake load combinations please see CAN3-N289.3-M81 (Section 6.3.2).

Both plus and minus signs are attached to the dynamic loads and then combined with others to get the worst combination. However, if time history method of analysis is carried out, then the magnitudes with the associated signs are considered in the combination.

3.3 France

Water hammer loads as identified in Section 2.3.2 (1), (3) and (5) are combined with other loads. The water hammer load described in Section 2.3.2 (4) is currently under investigation and will be combined with other loads.

3.4 Italy

Generally applicable load combination used in design of Italian nuclear power plants are contained in Tables K-4 and K-5 of NUREG-1061 Vol. II. Specific loads applicable to BWR pressure suppression pool safety relief valve and LOCA discharge response are found in the proprietory General Electric Co. Document NEDO 1070. Time phasing of pressure suppression pool loading is indicated in Figure 3.1 of this paper.

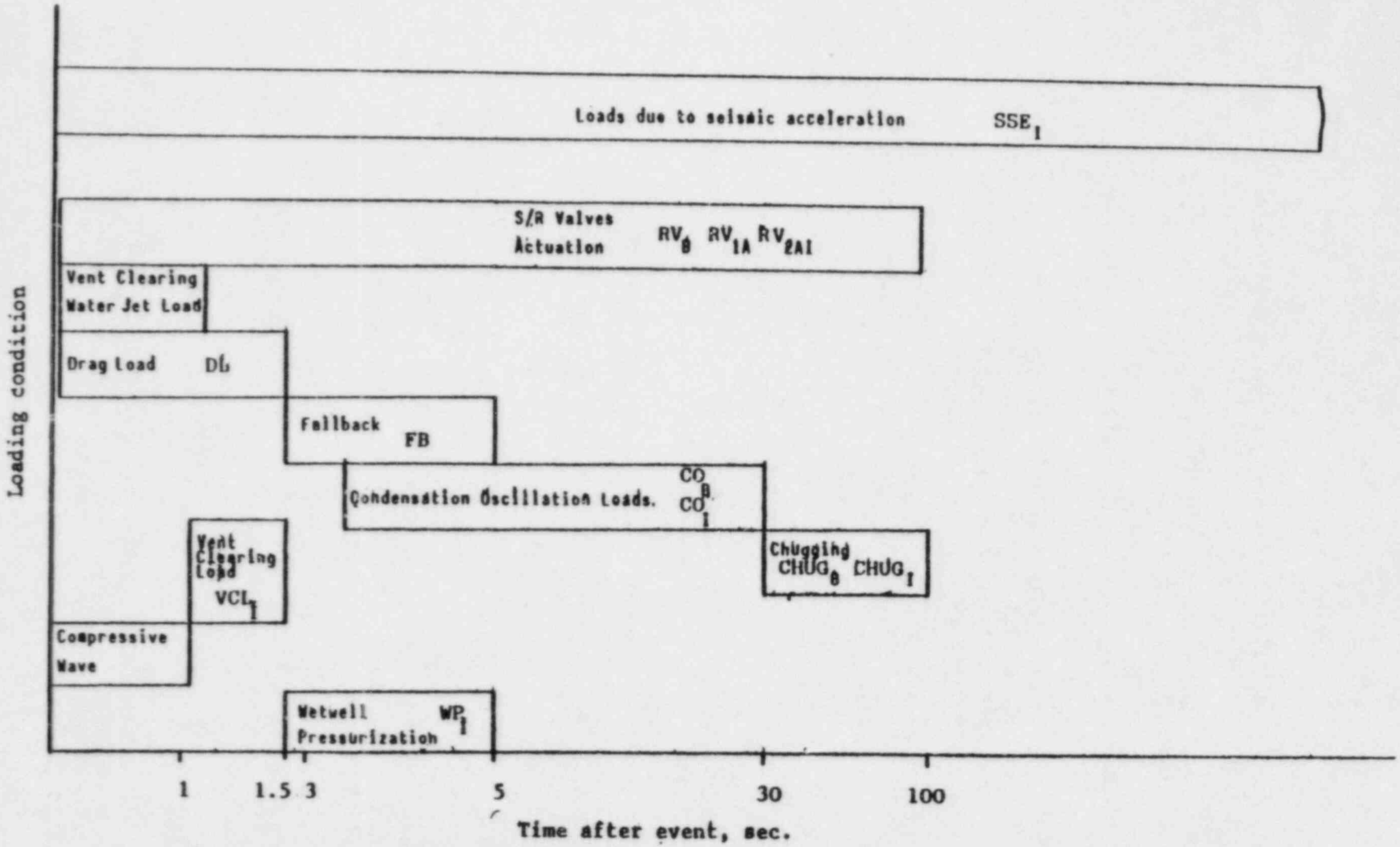

Figure 3.1. Piping Located within Suppression Pool-loading Chart for LOCA o

3.5 Japan

Load combinations other than earthquake are the same as ASME Code Section III (Author's Note: ASME Code for piping does not specifically specify load combinations).

A distinction is made between long and short term loadings but procedures used have not been specified. Different dynamic loads are combined on an absolute sum basis.

3.6 Sweden

A distinction is made between long and short term loading but procedures used have not been specified. Independent short term dynamic loads are combined on a SRSS basis.

3.7 Federal Republic of Germany

Load combinations and the thereto pertaining limits are compiled, by way of example, in Tables 3.2 and 3.3. Other applicable load combinations can be found in Appendix K-1 of NUREG-1061, Vol. II. The compilation can be considered as being representative. No distinction is made between long-term and short-term acting loads as their effects individually or in combinations. Loads resulting from dynamic analyses are superposed absolutely with the other loads.

4.0 BEHAVIOR AND ACCEPTANCE CRITERIA APPLICABLE TO HIGH FREQUENCY, VIBRATING AND WATER HAMMER LOADS

4.1 Maximum Stress and Behavior Criteria

4.1.1 Belgium

Stresses resulting from high frequency vibrating loads associated with aircraft crash (AOE) appear to use Service Level D acceptance criteria. Flow-induced or operational high frequency vibration stress are determined experimentally. It is not clear what acceptance criteria is used for velocity levels in the pipe and stress or deflection levels in the supports.

4.1.2 Canada

In Canada, static loads - dead weight, thermal and slow dynamic loads - seismic, steam/water hammer, valve thrust are considered with allowable stress limits defined by the code equations. For fast dynamic loads - pipe-break the plastic strain is limited to half the ultimate strain in current practice. For high frequency vibrating loads, acceptance appears to be based on velocity criteria. For a discussion of velocity used as an acceptance criteria as considered by Ontario Hydro, - see Appendix B.

4.1.3 France

4.1.3.1 Piping

Allowable stresses and loading combination are defined in RCC-M Code, B-3600, C-3600, D-3600 chapters of Section 1. Functional capability is assured by leveling up stress criteria.

Detailed fatigue analyses are required for Class 1 piping and frequently call for more than the simplified methods described in B-3600. Finite element analyses are needed to minimize the effects of thermal transients, the results being combined to other terms of B-3600 equations (pressure, earthquake, other mechanical loads, thermal expansion, and support settlement).

The French Code, RCC-M, requires that an evaluation of protection against fast fracture risk in piping be done for class 1 piping.

This is required in Section B 3611.5, which refers to Section B 3260, which refers to appendix ZG for the analysis methods which can be used for feritice steels.

Rules in appendix ZG for austenitic steel are in preparation.

Piping degradation in service is not explicitly considered in design. Defects detected during in-service inspection and which are difficult to be repaired immediately, are subjected to a crack propagation and stability analysis which, if successful, enables the plant to wait until the next outage to repair the defect.

With regard to overall design margins, Class I piping and components have to conform to Arrete des Mines of 26 February 1974 in which specific coefficients by which loads should be multiplied without damage, are included in paragraph 10.

The coefficients are the following:

DAMAGE	LOADING CONDITIONS		
	First Category	Third Category	Fourth Category
Excessive deformation	1.5	1.2	
Plastic instability	2.5	2	1.1
Elastic or elastoplastic instability	2.5	2	1.1

– Similarly, the Manufacturer shall demonstrate that, under Second Category conditions, there is no risk of progressive deformation or fatigue crack growth during the designed operating life of the CPP.

(First Category loading conditions refers to design conditions, Second Category to Normal and Upset, Third to Emergency and Fourth to Faulted, in the United State terminology. CPP includes all the reactor coolant loops and all auxiliary lines up to the second isolation valve.)

It is the objective of RCC-M Chapter B to meet this regulation

Loading Cases	Int. Press.	Dead Weight: Pipe	Dead Weight: Operat. Fluid	Dead Weight: Insulat.	Thermal Expansion	Support Displacem.	Displacem. of Nozzle	Prestressing	Water Ham.	Dynamic: Design B. Earth.	Dynamic: Save Sh. Earthq.	Dynamic: Aircraft Impact	Dynamic: Gas Cloud Explos.	Transient L.		Pipes: Equation	Pipes: Allowable Stress
Normal Operation																	
	X	X		X					X							9	$1{,}5\ S_m$
	X	X		X						X						9	$1{,}5\ S_m$
	X	X	X	X												9	$1{,}5\ S_m$
Normal a. Abnormal Op.																	
	X				X				X					X		S_n	$3{,}0\ S_m$
	X				X					X				X		S_n	$3{,}0\ S_m$
	X		X		X									X		S_n	$3{,}0\ S_m$
Test Conditions																	
(Pressure Test)	X	X	X	X												9	$1{,}35\ R_{pQ}$
Emergency / Faulted Cond.																	
Water Hammer	X	X		X					X							9	$3{,}0\ S_m$
Save Shutdown Earthquake	X	X		X					X		X					9	$3{,}0\ S_m$
Aircraft Impact	X	X		X								X				9	$3{,}0\ S_m$
Gas Cloud Explosion	X	X		X									X			9	$3{,}0\ S_m$
LOCA						X	X									S_e	$3{,}0\ S_m$

Table 2: Example Steam Line (PWR)

Stresses acc. to General Specification „Basic Safety" Class A1

Loading Cases	Superposition of Loads															Pipes	
	Int. Press.	Dead Weight			Thermal Expansion	Support Displacem.	Displacem. of Nozzle	Prestressing	Water Ham.	Dynamic				Transient L.		Equation	Allowable Stress
		Pipe	Operat. Fluid	Insulat.						Design B. Earth.	Save Sh. Earthq.	Aircraft Impact	Gas Cloud Explos.				
Normal Operation																	
Design Cond.	x	x	x	x												S_{Sh}	1,0 S_h
Normal a. Abnormal Op.																	
	x	x	x	x					x								1,2 S_h
	x	x	x	x						x							1,2 S_h
					x												1,0 S_A
Test Conditions																	
Pressure Test	x	x	x	x													1,35 $R_{p0,2}$
Emergency a. Faulted C.																	
	x	x	x	x							x						2,4 S_h
	x	x	x	x								x					2,4 S_h
	x	x	x	x									x				2,4 S_h
LOCA						x	x										3,0 S_c

Table 3.3: Example Emergency Core Cooling System (PWR)

Stresses acc. to General Specification „Basic Safety" Classes A2/A3

4.1.3.2 Supports

RCC-M code Section 1 Volume H covers supports (linear type and plate elements, except embedded plates themselves); it is not usually used for primary steel frames.

For <u>expansion bolts</u> for which criteria are not included in RCC-M, FRAMATOME has developed a procedure that guarantees a good fixture for bolts and plates. The procedure is based on recognizing that usual drilling (with tungsten carbide drills) is very often difficult and leads to holes which are out of tolerance (diameter, surface, uprightness, angle with concrete surface), not to speak of the impossibility to cut a rebar.

The method is based on diamond drilling associated, for the buildings where it is not permitted to cut rebars (the reactor buildings), with a detection of the location of rebars through:

-Examining the rebars drawing
-Checking the rebar locations with a magnetic detector

<u>For embedded plates</u>, tests have been performed to qualify the design in terms of:

-resistance of welded attachment of the embedded (curved) bar to the plate,
-resistance of the embedded bar,
-necessity of stiffening the plate,
-M-O relationship,
-concrete behaviour,
-validation of design loads (computer program, nomographs),
-validation of a detailed analysis model used for interpretation (non-linear finite element model)

Design margins in steel support design are greater than those used in piping design: steel supports have an elastic general behaviour, even in faulted conditions, per RCC-M.

In faulted conditions, piping thermal expansion effects are computed in order to determine loads on supports which are then designed to withstand these loads, wereas thermal expansion generates in the piping stresses which are secondary and unbounded per the RCC-M code.

4.1.4 Italy

Acceptance criteria applicable to loads for Italian nuclear power plant piping and support are summarized in Tables K-4 AND K-5 of NUREG-1061 Vol. II.

4.1.5 Japan

Acceptance criteria applicable to loads for Japanese nuclear power plant piping and supports are summarized in Section 6.3.1.5 of NUREG-1061 Vol. II and in Section 4.4 of NUREG/CR-3020.[7.1].

4.1.6 Sweden

Behavior criteria for water hammer type levels are established consistent with ASME III service levels as determined in the design specification. For high frequency vibrating loads, ASEA-ATOM has developed acceptance criteria for the Swedish State Power Board based primarily on measured reponse of piping similiar to procedures developed in the ASME-OM-3 standard [7.2].

4.1.7 Federal Republic of Germany

Acceptance criteria applicable to all loads for FRG nuclear power plant piping and supports are summarized in Appendix K-1 of NUREG-1061 Vol. II.

4.2 Fatigue Analysis Requirements

4.2.1 Belgium

Fatigue analysis in general are performed in accordance with the ASME III Code.

4.2.2 Canada

Fatigue analysis requirements for loads identified for analytical purposes are similar to those defined in ASME III. Fatigue evaluation for high frequency vibration are discussed in Section 2.2.1.1 of this paper.

4.2.3 France

Detailed fatigue analyses are required for Class 1 piping and frequently call for more than the simplified methods described in B-3600. Finite element analyses are needed to minimize the effects of thermal transients, the results being combined to other terms of B-3600 equations (pressure, earthquake, other mechanical loads, thermal expansion, support settlement).

4.2.4 Italy

Fatigue analysis is performed for Class 1, 2 and 3 piping according to ASME Section III requirements.

4.2.5 Japan

Explicit fatigue analysis requirements are described in the MITI Code.

4.2.6 Sweden

For Class 1 components fatigue analysis is performed according to ASME III. For other classes of pipes fatigue analysis is performed considering actual problem experience.

4.2.7 Federal Republic of Germany

Fatigue analyses are required for the piping systems. Fatigue analyses are conducted both for the primary loop and, according to the Basic Safety Criteria contained in the General Specifications of Table 2.1.3, included herein for such systems as listed in the 1st Attachment to the RSK guidelines for pressurized water reactors . For all other systems, a fatigue analysis is not generally conducted. However, in accordance with the regulations for pressure vessels, a definition is made for static and dynamic loading cases (see AD Memorandum (pamphlet) S1).
For the prevention of failures due to fatigue under changing stresses, a fatigue analysis is conducted for the components of the primary loop and for those of the External Systems. For pipings of the primary loop a difference is made between

- simplified proof of safety against fatigue
- elastic fatigue analysis, and
- simplified elastic-plastic fatigue analysis.

Details of the procedure are specified in the safety regulations KTA 3201.2, Section 7.8.

The criteria for the performing of fatigue tests and the applicable calculation procedures for the External Systems are represented in the General Specification "Basic Safety" (Attachment 2 to the RSK directives for pressurized water reactors, Chapter 4.2). The criteria for the conducting of fatigue tests and the permissible calculation procedures can be obtained from the ASME-Code, Section III, Subsection NB and NC.

5.0 MODELING ASSUMPTIONS ASSOCIATED WITH HIGH FREQUENCY VIBRATORY AND WATER HAMMER LOAD ANALYSIS

5.1 Vibrating Loads

In all the foreign countries surveyed, high frequency vibrating loads induced during normal operation (e.g. flow-induced) are not normally considered analytically. It is usual to consider such phenomenon experimentally during plant start up testing as needed on a case-by-case basis. Therefore, analytical models, except as they may relate measured velocity and displacement to stress as a function of a series of simple pipe geometries, are not considered explicitly.

High frequency loads induced by aircraft impact or BWR suppression pool dynamic response are typically applied to the seismic analytical model of the piping system and its supports. Input to this model in Belgium, Italy and the U.S. appear to be an acceleration response spectra while in the FRG a constant g value is used. For the FRG in the frequency range above 16 Hz, it must be assured that the relative diplacements between components and supports can be absorbed elasto-plastically.

Testing Class	Component	Fatigue Analysis	
		Criteria	Methods
A 1	Pressure Vessel, Pumps, Valves	Fatigue analysis if the number of cycles >1 000	Fatigue analysis e.g. in accordance with ASME NC 3219.2 or NB 3222.4 Determination of the number of cycles in accordance with ASME NC 3219 (All stress amplitudes $>0.2\ S_m$. Thermically induced load changes are determined on the basis of ΔT as it occurs in the wall of the component)
	Pipes	Fatigue analysis always required	Fatigue analysis e.g. in accordance with ASME NB 3653.4 or NB 3222.4
A 2	Pressure Vessel, Pumps, Valves	Fatigue analysis at special points if the number of cycles >1 000	Fatigue analysis e.g. in accordance with ASME NC 3219.2 or a suitable stress index method
	Pipes	Fatigue analysis always required	Stress index method in accordance with ASME NC 3611 (Limitation of the S_A value as a function of the number of cycles)
A 3	Pressure Vessel, Pumps, Valves	No fatigue analysis (design temperature $<100\ ^{o}C$)	
	Pipes	Fatigue analysis always required	Stress index method in accordance with ASME NC 3611

TABLE 2.1.3 GENERAL SPECIFICATION "BASIC SAFETY"

Criteria for the Implementation of Fatique Anlayses and Permissible Calculation Methods

Table 1

However, in Italy an inelastic spectral input with ductilities taken equal to 2.0 to 3.0 is permitted in response to aircraft impact effects. This loading condition does not effect design in France and Canada because of the type of reactor systems used and the aircraft crash criteria considered. It is not clear what inputs are considered in Japan and Sweden for this loading condition.

5.2 Water Hammer Loads

Except for the FRG anticipated water hammer loads such as rapid large valve opening and closure, local pressure transient, determined by simplified hand calculations, are generally performed. Sophisticated thermal - hydraulic - structural computer analysis and associated dynamic models of the system have seen very limited application except in the FRG where such computations are routinely performed for systems with safety significance. For other countries such calculations tend to be used only in those cases where simplified methods are thought to give overly conservative results, or where water hammers have occurred in a particular system. Structural models of the piping systems analyzed for water hammer are usually similiar to those used for seismic analyses.

6.0 CONCLUSIONS AND RECOMMENDATIONS

6.1 Conclusions

Review of the design practices associated with high frequency vibrating and water hammer loads considered in the foreign countries surveyed indicate that these countries generally take a less mechanistic approach than the U.S. to the problems that may arise from such loads. For impact high frequency resultant loads, there is a general recognition that such loads do not cause the damage that their magnitude as determined by inertia response acceleration would indicate. As a result such loads have been limited in some countries by use of a frequency cut off, as in the FRG, and a non-linear spectrum input, as in Italy. It is also recognized that displacements associated with these loads are quite small (typically less than the tolerance gaps which exist between the pipe and its pipe support), hence the motion of the support is not sufficient to excite the pipe. The resultant stresses in the pipe are much less than would be indicated by the calculated response of a linear elastic model of the piping system.

The flow-induced or operational high frequency vibration of piping systems are generally not considered analytically. This is true because such effects can be observed and measured experimentally relatively cheaply and accurately during plant start up. In addition, analytical definition of vibrating-forcing functions, due to flow or other operational perturbation of the system, are generally not possible with any accuracy. It is also recognized, because of the small deflections and gaps in the pipe support systems, that it would be exceedingly difficult to predict stress resultants in the piping system analytically.

Except for the FRG which tends to use more rigorous analytical techniques, anticipated water hammer effects in the foreign countries surveyed are evaluated by simplified hand calculations which, historically for conventional high energy piping systems (fossil fuel and petrochemical plants), have given satisfactory results. In foreign countries in general, there seems to be much less reliance on rigorous calculation or computational results and a much greater willingness to substitute experience and technical judgment in developing an adequate design for piping systems for all applied loads.

6.2 Recommendations

It is recommended that foreign operating experiences, particularly those associated with water hammer and fatigue failures be reviewed in detail to determine if their experiences are significantly different than those in the U.S. for nuclear power plant piping. Based on such a review, it may be possible to determine, at least on a statistical basis, if the higher level of analytical effort expended in the U.S. provides a significant difference in the level of plant reliability as defined by unanticipated occurrences, excess vibration and failures in piping and their supports.

7.0 REFERENCES

7.1 Stevenson, J.D., and Thomas, F.A., "Selected Review of Regulatory Standards and Licensing Issues for Nuclear Power Plants, "NUREG/CR--3020 United States Nuclear Regulatory Commission, November 1982.

7.2 ANSI/ASME-OM3-1982, "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Piping Systems", ASME, Sept. 1982

7.3 AFCEN, "Design and Construction Rules for Mechanical Components of PWR Nuclear Islands -- RCC-M, Section 1, January 1981

APPENDIX A

Questionnaire on

Criteria, Assumptions and Analytical Methods Used in Design of Nuclear Power Plant Safety Related Piping - All Classes

I. LOADING

A. SEISMIC

1. What forms of seismic load definition are permissible for piping design, as for example, a) ground spectra with amplification factors, b) floor or amplified response spectra - using dynamic multidegree of freedom analysis, c) one times peak of floor response spectra applied to mass distribution of pipe, d) a multiple of one times the peak of the floor spectra applied to the mass distribution of the pipe. If a multiple of one times the peak of the floor spectra is used, how is this value defined? Are other forms of seismic load definition permissible? If so, what are they?

2. How many simultaneous directions of seismic loading are considered, a) one horizontal only, b) one horizontal plus one vertical, c) two horizontal plus one vertical?

3. If more than one direction of seismic loading is considered simultaneously, are they considered of equal magnitude (e.g., 100% 100% 100%)or some other combination (e.g., 100% 40% 40%)?

4. Are inelastic floor response spectra input permitted? If yes, under what circumstances and how and what values of ductility are defined?

5. How many levels of earthquake (e.g., OBE, SSE or S1, S2) are actually considered in the design of pipe? If two levels of earthquake are considered, which level usually controls design? How are different input response spectra for piping located at different support locations considered in design, a) by using single envelope spectra, b) by use of input from several spectra located at different support points. If single envelope spectra are used, how is envelope spectra developed? If several spectra from multiple support points are used please describe means or give appropriate references as to how these spectra are considered in developing seismic response of the piping.

6. What values of percent critical damping are used in design and analysis of piping? If more than one value is used please describe functional relationships and basis of selection, a) damping as a function of frequency, b) damping as a function of mode, c) damping as a function of support type and support gap size?

7. Please describe how different spacial and modal components are combined to determine resultant forces and moments about the three principal local axes of the pipe. How are closely spaced modes considered? What sequence of load combination is used, a) by mode first and then by direction or b) by direction and then mode?

B. HIGH FREQUENCY VIBRATION

1. Are pipes evaluated for high frequency (> 20 Hz) cyclic loads (aircraft or other missile impact, flow induced, valve or pump operation, etc.)?

2. If yes, please describe methods (analytical or testing) of evaluation.

3. Are explicit fatigue analyses required for piping? If yes, please describe the procedure used as a function of the safety classification of piping.

C. WATER (STEAM) HAMMER

1. To what extent is the rapid change in the pressure of a fluid in a pipe caused by a rapid change in the fluid velocity (water, steam hammer) required to be considered in design, a) anticipated - rapid valve closure, b) unanticipated - water slugs, steam condensation?

2. What measures are used to protect against water hammer, a) design, b) administrative and c) operational procedures.

3. Are water hammer loads combined with other loads?

4. Is water hammer considered with degraded pipe?

D. DEAD WEIGHT, PRESSURE, THERMAL AND LIVE LOADS

1. Are differential support settlement explicitly considered in design?

E. LOADS AND LOAD COMBINATIONS

1. Please give load combinations explicitly considered in design of piping and indicate applicable behavior criteria limits.

2. Do you distinguish between long term or short term loads as to their effect or combination?

3. Are dynamic loads combined on other than an absolute sum basis? If yes, what is the basis of combination?

F. PIPING SYSTEM DESIGN RESPONSIBILITIES

1. Please describe the organization used to develop the overall piping design.

II. BEHAVIOR CRITERIA

A. PIPE

1. Please define the allowable stresses and/or deformations permitted in piping as a function of safety class or give applicable construction standard reference in design for

   a. Dead + Live Load + Pressure (Primary Stresses)

   b. Dead + Live + Pressure + (OBE or SSE) Earthquake (Primary Stresses)

   c. Dead + Live + Pressure + Thermal + Support Settlement (Primary + Secondary Stresses)

   d. Dead + Live + Pressure + (OBE or SSE) Earthquake + Thermal + Settlement (Primary + Secondary Stresses)

2. Are specific fatigue analyses required? If yes, under what circumstances and what acceptance criteria is used?

3. Are specific brittle fracture analyses required? If yes, under what circumstances and what acceptance criteria is used?

4. Is the potential for ratcheting explicitly considered in design?

5. Do you distinguish between allowable stress limits associated with dynamic loads (slow - seismic; and fast - pipe break) and static loads? If yes, how are these distinctions made?

6. Is piping degradation in service explicitly considered in design? If yes, how?

7. How are nozzle load limits on equipment determined?

8. Are overall design margins identified? If yes, how are they determined?

B. SUPPORTS

1. Please provide the same information as (IIA.) above applicable to supports instead of piping including criteria governing design of anchor bolts

2. Are seismic supports required to be rigid (e.g., the support plus contributing mass from pipe has fundamental frequency greater than 33 Hz)? Or is there some minimum ratio required between the stiffness of the support and the pipe it supports?

3. Are there any restrictions or special requirements on the use of snubbers, a) hydraulic, b) mechanical?

4. Are vertical rod type pipe hangers used in seismically designed lines? If yes, what analytical assumptions are made if there is a net compression or upward load in the hanger under seismic loading?

5. Are the design margins used in support design greater or less than those used for the piping?

III. MODELING & LAYOUT ASSUMPTIONS

A. PIPE

1. For thermal analysis of piping, a) is a computer analysis considering all supports required, b) are all fixed supports (hangers, U bolts, etc.) considered rigid?

2. Are spring constants for spring hangers and snubbers considered in the, a) dead weight analysis, b) thermal analysis, c) seismic analysis? If yes, how are they determined?

3. How are constant spring hangers considered in the, a) dead weight analysis, b) thermal analysis, c) seismic analysis?

4. Are nonlinear analyses of the piping system permitted? If yes, what are the circumstances?

5. For seismic analysis of piping are all fixed supports (hangers, U bolts, etc.) considered rigid? If yes, what is the basis for this consideration?

6. For seismic analysis of piping are variable and constant spring hangers considered as restraints in the analysis?

7. Are maximum permissible gaps between pipe and supports specified? In the U.S. such gaps are typically specified at ± 0.06 inches, in Canada such gaps are taken as ± 0.25 inches. Larger gaps consistent with adequate restraint of the piping based on experimental tests appear to result in higher damping, hence, lower seismic stresses. Have you formulated a policy in this area?

8. Are support gaps ever used to reduce thermal loads and thereby reduce the need for snubbers? If yes, on what basis?

B. SUPPORTS

1. Is the use of snubber type supports actively encouraged or discouraged for, a) hydraulic, b) mechanical? If yes, what procedures are used?

2. How are support stiffnesses considered in design for, a) thermal, b) seismic, c) pipe whip?

APPENDIX B

VIBRATION VELOCITY AS A GENERAL SEVERITY CRITERION

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SUMMARY

Vibration velocity is better than either displacement or acceleration as a direct indicator of vibration severity and associated equipment distress. Furthermore, the allowable magnitudes of velocity lie in a relatively narrow range, even for a wide variety of systems, equipment and structures. Thus, adopting velocity as the standard quantity for general use will reduce the need for system-specific investigation and analysis to determine acceptable limits. This approach has been identified and proven by reference to existing standards of acceptable vibration, to a recent CEA Research project which confirmed a strong theoretical correlation between vibration-velocity and dynamic stress, and to data from a wide range of actual field problems.

1.0 INTRODUCTION

Vibration data can be collected and reported in many different forms. Firstly, there is a choice of the QUANTITY to be recorded, i.e. displacement, velocity or acceleration. And secondly, there is a choice of the particular format, i.e. peak, average or root-mean-square; all-pass or filtered; time waveform or spectral components, etc. Various formats are also used in STANDARDS of acceptable vibration levels. Most standards are expressed in either displacement (peak-to-peak), velocity (peak) or velocity (r.m.s.). In practice the choice of quantity and format is normally determined by some combination of specified requirements, past practice, transducers and equipment readily available, and personal preferences.

Most people seem more familiar with either displacement or acceleration, as they have a ready physical reference; i.e. one can readily grasp and appreciate the displacement (peak-to-peak) as the total excursion of the vibratory motion; and one can imagine a dynamic inertial loading equal to the vibrating mass times the maximum acceleration. It should also be noted that most theoretical analysts and most large general-purpose computer programs work in terms of displacement or acceleration. However, velocity is widely used, due to its appearance in various standards for machinery, and to the availability of velocity transducers which require neither a stationary reference point, nor complex electronics.

The state-of-affairs outlined above can result in uncertainty and wasted effort in several situations. For example, when standards and field data are both expressed in a variety of ways, the overall reference data base is fragmented; we lose the benefit of being able to COMPARE and EXTRAPOLATE across a range of applications. When faced with a commissioning or operating problem one always wants to know what level of vibration is acceptable. Suppose the application is 'new' (i.e. where neither a direct experience-base nor an applicable standard is available). There are then, loosely speaking, two alternatives: decide arbitrarily, or do an analysis. The first alternative is certainly undesirable, and where safety is involved, not acceptable. To do an analysis involves an expense which might be avoidable, if a broader data base could be utilized; in addition, any acceptance criterion developed analytically may be presented in terms which the field crew is not equipped to measure readily.

Out of this existing situation there arise two obvious questions: Does any one of the various vibration quantities relate most directly to distress and potential damage? If so, wouldn't the use of that particular quantity lead to simplifications and cost reduction in most vibration work? The answer to both questions is affirmative: VIBRATION VELOCITY is the preferred quantity; and it's adoption as a standard WILL simplify and reduce the work involved. These conclusions are based upon three elements of support: existing standards; a small applied research project; and experience-data from a variety of field problems. The main body of this paper will present the important details of the supporting arguments, expand upon the results, conclusions and limitations, and present recommendations on how to apply the results.

2.0 THEORETICAL BASIS, CEA RESEARCH PROJECT

As noted above, most standards of acceptable vibration levels for rotating machinery use vibration velocity as the reference indicator of severity. There are some common misconceptions about the underlying basis for choosing velocity. Many people think it is based mainly upon ease of measurement using velocity transducers. Others consider velocity appropriate because it lies midway between displacement (which falls off at high frequencies) and acceleration (which falls off at low frequencies), and therefore should be applicable over a broader central frequency range. And most people believe that the actual magnitudes of velocity allowed have been developed empirically.

All of the above may be true to some extent. However, there is a much more fundamental reason for using velocity, not only for machinery, but for any vibration problem on any component or structure. By piecing together bits of information, it began to appear that there is a strong and persistent direct correlation between vibration velocity and dynamic stresses. To explore the analytical basis of this correlation, and the range of its validity for practical application, a small CEA research contract was initiated /1/. The contract was awarded to the Centre de Recherche Industrielle du Quebec (CRIQ), and was performed by the second and third authors of this paper.

The work on the project consisted mainly of literature review, and parametric calculations using formulae available for various configurations. At the time of writing the technical work was complete, and the final report was being prepared. The main findings and conclusions are as follows:

2.1 First it was proven that for any linear structure there is a simple relation between space-average mean square vibration velocity and space average mean-square stress. This relationship is a consequence of the equality of maximum kinetic and potential energies in the vibrating structure. It has the form

$$\sigma_{rms} = E \frac{V_{rms}}{c} \qquad (1)$$

and is independant of the form of the structure, and of the particular mode of vibration. This provides a very sound, fundamental starting point. However it must be noted that failures are more closely related to maximum stresses rather than the root-mean-square stresses of Equation 1.

2.2 Next, it was found that a basis for relating the maximum vibratory stress to the maximum vibratory velocity had existed in the mechanical engineering literature for twenty years. In a footnote to a 1962 paper /2/, Ungar showed that for beams and plates vibrating at resonance, the maximum dynamic strain is related to the maximum vibration velocity according to:

$$\epsilon_{max} = \sqrt{3} \frac{V_{max}}{c} \qquad (2a)$$

Interestingly, Ungar interpreted this relation in terms of the strain being proportional to the 'Mach Number' of the oscillation, i.e. the velocity of the oscillation divided by the velocity of sound in the material).

By simply introducing Hookes Law $\sigma = E \epsilon$, Equation 2a becomes:

$$\sigma_{max} = \sqrt{3}\, E \frac{V_{max}}{c} \qquad (2)$$

2.3 Based upon Equations 2 and 1, there naturally develops a two-point hypothesis as follows:

- Firstly, it is expected that for any structure vibrating at resonance there will be a simple relationship of the form:

$$\sigma_{max} = (\text{constant})\ (E)\ (\frac{V_{max}}{c}) \qquad (3)$$

- Secondly, the 'constant' is not expected to vary greatly, even over a wide range of system size, geometry, vibration mode and frequency.

The main theme of the CEA project was to evaluate these two hypotheses, i.e. their basic validity in broad terms, the variability of the proportionality constant, and the potential for practical application.

2.4 To test the hypotheses, a very straightforward approach was taken. First, a list of elements of practical interest was developed, i.e. rods, shafts, beams, plates and shells. Next the technical literature was searched for available analytical solutions for vibration frequencies and modal deflection shapes. Given such solutions one can determine the maximum vibration velocity and the associated maximum stress (i.e. vibration velocity is proportional to maximum deflection times natural frequency, and maximum stresses are determined by section stiffness and curvature of the deflected shape). The proportionality factors are then calculated by simple division.

Emphasis was placed on covering as wide a range of element types and geometry, and as many modes of vibration as could be accommodated within a limited project budget. The main results of these parametric calculations are summarized in Appendix I.

2.5 From a glance at Appendix I it can be seen that the proportionality factors for the great majority of cases fall in a reasonably narrow range. This confirms the basic validity of both hypotheses, and it remains only to define the limits of valid application, and the range of variation of the proportionality factor. This cannot be done definitively and rigorously from the limited study completed, but several important features have been demonstrated as follows:

- The correlation works well for flexural vibration of beams and plates with any practical section shapes and boundary conditions. The proportionality factors for most practical cases lie between 1.5 and 2.5.

The range of extreme proportionality factors is from about 1 to 4. Included in this range is the effect of having clamped boundary conditions; this increases the factors by approximately one third as compared to the simply-supported case as a reference. Also included is the effect of section cross-sectional shape; as might be expected the extreme low and high factors are associated with efficient and inefficient shapes respectively, e.g. from approximately 1 for WF beams, up to about 3 for Tees and triangles.

- Remarkably perhaps, the correlation also works well for the non-flexural cases tried. For axial vibration of bars, the proportionality factor is 1. For torsional vibration of shafts it ranges from 1.2 to 1.6 depending upon section shape.

- Some exceptions have been identified, where the correlation breaks down. These include:

- beams of non-uniform cross section (such as the tapered cantilever, where there is a factor of five variation even for the lower modes)

- Clamped circular plates and supported/free rectangular plates, where the proportionality factors are less than one for some modes

- clamped annular plates where the proportionality factor exceeds four for some modes

- add masses which can shift the proportionality factor from the 'uniform' reference case (in general the factors can shift in either direction; of greatest concern in practice is the case where sizeable added masses can lead to substantially higher factors, making the approximate approach non-conservative.

- for uniform cylindrical shells, membrane stresses can be significant for certain modes, yielding extremely large proportionality factors.

- From the analytical relationships developed (e.g. Equations 1 and 2) and from the limited parametric studies peformed, it can be concluded that:

- For elementary structures vibrating at a natural frequency, there is a remarkably simple direct relationship between vibration velocity and nominal dynamic stress. It is of the form:

$$\sigma_{max} = (\text{contant})\ (E)\ \left(\frac{V_{max}}{c}\right) \qquad (3)$$

  which may also be expressed as:

$$\frac{\sigma_{max}}{V_{max}} = \text{constant} \times \sqrt{\rho E} \qquad (4)$$

- For a wide range of practical structural and machine elements, the proportionality factor will fall in the range,

  proportionality factor = 1.2 to 2.8

  whereby Equation 4 may be rewritten as a rough but very useful practical approximation:

$$\frac{\sigma_{max}}{V_{max}} = 2\sqrt{\rho E} \pm 40\% \qquad (5)$$

Substituting the values of density and modulus of elasticity for various materials there results the following table of 'stress-per- velocity' constants:

Material	$\frac{\sigma_{max}}{V_{max}}$, kPa per mm/sec
Steel	77.2
Copper	65.9
Brass	59.4
Cast Iron	56.8
Aluminum	27.6

3.0 EXISTING AND DEVELOPING STANDARDS

As noted in the Introduction, various existing standards for allowable vibration utilize different quantities and formats (i.e. displacement or velocity, peak or r.m.s., etc.). There are some recent and ongoing developments in standards writing, from which some relevant trends can be determined. These are described briefly, for three different applications, as follows.

3.1 Rotating Machinery

There are many international, national, and manufacturer's standards for classification or limits of vibration severity. Some of them utilize lines of constant velocity to define various categories of vibration severity. Others retain the 'constant velocity' criterion over the mid-range of frequency, but switch over to limits on displacement at low frequency, and acceleration at high frequency. There are some situations, particularly for continuous machinery-protection monitoring, where the choice of parameter is quite obvious. Shaft-to-bearing radial clearance and rotor-axial-positon for example, clearly call for relative displacement as the most direct and relevant quantity.

Although there are exceptions, velocity appears to be the most frequent choice as a general descriptor of machinery levels, particularly when absolute bearing cap or casing measurements are used. A recent paper by Plummer /3/ recommends using velocity as the criterion for 'periodic-inspection' monitoring of pumpsets; Plummer's argument is based upon the expected direct relationship between vibration velocity and dynamic stress. Also, an ANSI committee on machinery is considering adopting ISO standards which use velocity as the reference quantity.

3.2 Power Plant Piping

The American Society of Mechanical Engineers is developing a standard for piping vibration. Entitled 'Requirements for Preoperational and Initial Startup, Vibration Testing of Nuclear Power Plant Piping Systems', it has reached the final draft stage /4/. Eventually it will become an ANSI standard.

This standard allows for varying levels of complexity and effort to demonstrate acceptability as regards vibration. These options range from the application of very conservative system - independant screening criteria for maximum vibration, through detailed dynamic analysis to determine system-specific test points and permissible vibrations, and right up to direct measurement of dynamic strain by strain gages.

For purposes of this paper we are most interested in specification of acceptance criteria directly in term of a vibration quantity. Here the ASME standard allows the use of either displacement or velocity as the significant quantity. If displacement is chosen, the allowable limits will be system specific; that is, they will depend upon the size, layout and mode of vibration. If velocity is chosen, the limits will be nearly independant of these factors, since the approach is based upon the direct relationship of vibration velocity to dynamic stress. Thus using a velocity criterion provides a simpler more straightforward process of measurement, comparison with allowables, and reporting. For example the ASME standard, based upon conservative assumptions regarding layout, additional masses and stress concentration factors, includes a screening level below which any system is acceptable.

3.3 Structures

There is at least one code for steady-state vibration of structures. This is the German code DIN 4150 /5/ which applies for uniform concrete and wooden beams and plates. This code is another case where vibration velocity is used as the reference quantity, based upon the direct relationship with dynamic stress.

3.4 The Choice of Format; Peak Versus RMS

Even assuming acceptance of velocity as the best general descriptor of vibration severity, there remains the question of which format to use. The RMS format has an averaging effect which will smooth out the variability in long-term trend plots. The PEAK format on the other hand is more sensitive to intermittent vibration which in many cases is an indication of trouble. Some observations can be made based upon further reference to the existing and developing standards as follows.

For rotating machinery the North American practice has been to express velocity measurements in the PEAK format, whereas in Europe there is a preference for RMS. Apparently most European representatives on the ISO committee are satisfied with RMS, with the case for PEAK being made by some North American representatives. The discussion is ongoing.

For piping the ASME standard is very clear and specific. The reference quantity is to be the maximum PEAK vibration, not RMS. It is acceptable to make the measurements in the RMS format, but they must then be converted to PEAK by a demonstrably conservative multiplying factor.

The DIN 4150 code for structures is also clear. It uses PEAK vibration velocity as the standardized reference quantity.

3.5 Absolute Levels of Peak Vibration Velocity

Based upon these points of reference it is concluded that maximum PEAK vibration velocity should provide a well recognized, if not the best, general indicator of vibration severity. It is of interest to see how the absolute acceptance levels vary from one type of equipment to another. Figure 1 shows a comparison of levels taken from three standards: the ISO charts for rotating machinery /6/, the ASME standard for piping, and the DIN code for structures. It is somewhat remarkable to see how little variation there is. In fact Figure 1 suggests that, within a generous but practically-useful tolerance, vibration severity levels might be considered to be independant of the particular type of equipment.

4.0 FIELD EXPERIENCE

The best evaluation of vibration velocity as a general severity criterion is by correlation with actual field experience. To test the validity of extrapolation into areas where there is little background and experience, the emphasis should be on non-machinery applications. The cases should cover a variety of power plant and process plant systems and equipment, and various types of vibration problems. Such a test has been compiled, as per Appendix II and Figure 2.

Appendix II briefly describes the field cases considered. Although dealing mainly with piping, they can be seen to contain a good measure of variety. Some are from internal Ontario Hydro experience, while others are external having been drawn from the technical literature. Some cases involved failures, while others were rectified before any failures. The failures included cases of wear, fatigue and fracture. The sources of vibration excitation were varied, and the resulting vibration frequencies ranged from less than 1 Hz to nearly 3 kHz. These cases are not the result of any complete comprehensive search; rather, we have simply used data which was on hand or known to be readily available. On the other hand the cases have not been selected or screened in any way; we have included all cases for which quantitative data was available. Thus the cases of Appendix II are considered a fair preliminary test of the proposed correlation.

The accumulated field experience of Appendix II is presented graphically on Figure 2. Each case is shown by a single data point of peak velocity versus frequency of the vibration. Some lines of constant velocity are also shown for reference purposes (some of these lines represent standards; other have been added arbitrarily by the authors).

From Figure 2 some observations can be made in support of the theme of this paper. In particular:

- The use of peak velocity as the relevant parameter leads to a reasonably compact, and practically useful, correlation. All of the problems fall in the range of 6.4 to 254 mm/sec, a factor of 40. For piping and valves the range is from 12.7 to 254 mm/sec, a factor of 20; excluding the somewhat unusual case P1, this range is reduced to 38 to 254 mm/sec, a factor of about 7.

Had we attempted the correlation in terms of displacement or acceleration, the range of variation would have been several orders of magnitude, due to the wide frequency range.

- It is feasible to at least estimate the range of vibration velocity at which problems are likely to occur. For the general category of process and power plant piping circuits, there should be few if any problems below 40 mm/sec, while levels beyond 100 mm/sec are likely to require correction.

The ranges for other types of equipment could be estimated from similar plots.

It is also of interest to relate these actual field cases to the various standards. The main points of comparison are as follows:

- Process piping and associated equipment can apparently operate at vibration levels well above those recommended for machinery. There appears to be a pivotal band at about 25 to 40 $mm.sec^{-1}$ below which piping is acceptable, but above which even the largest machinery would be considered very rough.

- For the data shown, the ASME's screening level for piping appears to be overly conservative (i.e. 12.7 $mm.sec^{-1}$ allowed by the standard versus 40 $mm.sec^{-1}$ deduced from the data). On the positive side application of the ASME criterion during commissioning would have identified all of these cases for either rectification or more detailed analysis to justify higher allowables. But on the other hand, it would probably have identified many other cases which would also have required follow-up effort.

- The one data point on structural vibration is compatible with the DIN standard.

5.0 COMPARISON OF ANALYTICAL MODEL TO STANDARDS AND FIELD EXPERIENCE

It has been shown that there is a fundamental direct relationship between vibration velocity and the associated dynamic stress. Further, it has been demonstrated that vibration velocity is a reliable general indicator of vibration severity and potential equipment distress. Naturally then, one might expect that the realistic, practical, allowable levels could be derived, or at least rationalized, in terms of allowable stresses and failure theories. This can be achieved to some extent, but there are very definite limitations, as explained in the following.

The most direct and complete comparison can be made by considering a piping system. Let us further assume that the material is steel, for which the allowable alternating stress is 68 MPa. From Figure 2 we concluded that the acceptance level should lie between 40 and 100 mm/sec. Using the proportionality factor of 0.0772 MPa/mm sec^{-1} for steel, the corresponding nominal dynamic stresses are only about 3.1 to 7.7 MPa; these are obviously very low as compared to the allowable 68 MPa; using the ASME screening level of 12.7 mm/sec would limit the nominal dynamic stresses to the still lower value of 0.97 MPa. Clearly then, the practical vibration limits correspond to surprisingly low dynamic stress levels as estimated by the approximate formula. This apparent discrepancy can be accounted for by two main factors as follows:

- The DETAILS OF THE LAYOUT can lead to localized stresses much higher than those estimated from the stress-versus-velocity approximation. The higher stresses arise because of complex three-dimensional layouts, additonal distributed masses from pipe contents and insulation, additional concentrated point masses such as large unsupported valves, and finally localized stress concentration factors at fittings, welds etc. The ASME standard, via these respective factors, allows for a total range of a factor of (1.9) (1.5) (8.33) (4) = 95 from the most to the least favourable extremes. The ASME screening level of 12.7 mm sec allows for the worst case, and is thus seen to be quite conservative.

- WEAR AND GENERAL DETERIORATION are as important a class of failure as actual fatigue failure of piping. Examples would include loosening wear, and impaired function of valve operators and other control components, and loosening and general deterioration of equipment supports and structural connections. The relationship of such failure processes to nominal or even to local dynamic stresses is not well developed. Thus there is little point in trying to derive or explain practical absolute allowables in terms of the approximate stress-versus-velocity relationship.

Thus, although the universal stress-versus-velocity relationship is a valid and useful basis for correlation and comparison, it cannot provide the absolute allowables, except for simple configurations. In general, either more-detailed analysis or relevant prior experience is required to determine the absolutes.

6.0 CONCLUSIONS AND RECOMMENDATIONS

- For resonant vibrations there is a fundamental, direct relationship between maximum vibration velocity and maximum dynamic stress. It has the form

$$\sigma_{max} \simeq C\sqrt{\rho E}\, V_{max}$$

and is thus independant of the details of the configuration.

It is recommended that this fact be exploited to reduce the need for detailed analysis to determine allowable levels.

- For a wide range of simple practical structural elements, the proportionality factor C, falls within the range of 1 to 3, independant of the particular mode of vibration.

For actual process systems and structures, the additional complexity of three-dimensional layouts, appended masses, and local stress concentration must be accounted for, as they yield significantly greater stress for a given vibration velocity.

- Maximum vibration velocity is the best general indicator of vibration severity and potential distress. Its use is recommended for any and all applications except where another parameter (e.g. clearance) is clearly superior.

The use of velocity should add to the value of all data bases by reducing scatter and permitting more confident extrapolation to new situations.

- The basic validity of this approach has been demonstrated by application to a variety of actual field problems.

For typical power and process plant piping systems, including appended equipment and supports, the allowable level is approximately 40 mm sec^{-1}.

- The allowables for process system piping and associated equipment are somewhat greater than for rotating machinery, i.e. 40 mm sec^{-1} for piping versus a maximum of 25 mm sec^{-1} for large machinery.

- The relationship between dynamic stress and vibration velocity, and the use of vibration velocity as a universal descriptor of severity, are both worth some further development.

ACKNOWLEDGEMENTS

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6. ISO Standard 2372-1974 (E), Mechanical Vibration of Machines with Operating Speeds from 10 to 200 Rev/sec - Basis for Specifying Evaluation Standards.

NOMENCLATURE

c, speed of sound in material
C, dimensionless proportionality constant
E, modulus of elasticity
v, vibration velocity
ϵ, strain
σ, dynamic stress
ρ, density

APPENDIX I

STRESS-VERSUS-VIBRATION PROPORTIONALITY FACTORS
SUMMARY OF PROMISING RESULTS FROM CEA PROJECT G 197

CONFIGURATION AND BOUNDARY CONDITIONS	GEOMETRY AND MODES CONSIDERED	S, FOR SECTION SHAPE	$C = \left\{\frac{\sigma_{max}}{E}\right\}\left\{\frac{V_{max}}{c}\right\}^{-1}$
UNIFORM BEAMS	tubular	$\sqrt{2}\,(1 + t/d)$	
	rectangular	$\sqrt{3}$	
	solid circular	2	
	various structural steel	1.08 to 3	
Simply-supported	all modes		C = S
Cantilevered	all modes		C = S
Clamped-clamped	all modes		C = 1.32 S
UNIFORM PLATES			
Rectangular Simply-supported	aspect ratio 1 to 9 first 16 modes		1.18 to 1.82
Simple-clamped	aspect ratio 0.5, 1, 2 first 36 modes		1.19 to 2.58
Clamped-clamped	aspect ratio 1 to 2 first 36 modes		1.65 to 2.43
Simple-free	aspect ratio 0.5, 1, 2 a few modes		0.62 to 1.53
Circular, clamped	first 18 modes		0.7 to 1.2
Annular, simple-simple	any radius ratio, up to 4 half-waves radially		1.6 to 2.04
clamped	any radius ratio, up to 4 half-waves radially		2.76 to 4.3
UNIFORM RODS			
Axial vibration	any mode		1
UNIFORM SHAFTS	any mode		C = 2 S(1)
Torsional vibration	circular or tubular	0.62	1.24
	square	0.65	1.30
	rectangular, 2:1 section	0.81	1.62

(1) Note: For torsional case stress σ is taken as twice the shear stress.

APPENDIX II

SUMMARY OF FIELD-EXPERIENCE CASES

MACHINERY

M1	Five problems with feed and circulating pumps in power plant, e.g., loose thrust bearing nuts, loss of oil from thrust bearing, motor core rub, worn journal bearing, support structure resonance at running speed.
M2	Cooling water pumps operated at low flow.
M3	Feed pump at low flow.
M4	Rotor dynamics torsional on pump start.

PIPING

P1	Process-plant gas circuit. Fatigue failure of bellows liner located downstream of butterfly flow-control valve. (Vibration measurements made externally on pipe wall.)
P2	Condensate piping, between level-control valves and deaerator. Broad band vibration due to cavitation. Failure of pipe at welded-on pipe support.
P3	Cold reheat piping in 600 Mw cycling unit. High-cycle fatigue failures of thermowells, drain pots and instrumentation attachments. Excitation was traced to blade passing frequency of last three stages of HP turbine.
P4	Main steam piping in power plant. Low-frequency flow-induced vibration. Unit load restricted until additional restraints added to reduce vibration.
P5	Refinery piping, low-frequency flow-induced vibration; vibration reduced by improved flow distribution.
P6	Refinery piping; low-frequency flow-induced vibration; failure of supports and propagation of crack into vessel.
P7	Thermal plant feedwater piping. Low-frequency vibration of some concern to operators.
P8	Steam reject piping in power plant. Low-frequency flow-induced vibration during high-flow steam-dump-to-condenser. Concern to operators.

P9 Process loop bypass line. Severe vibration and high noise caused by flow through an orifice located just upstream of an elbow. No failure, but rectification considered necessary.

P10 Intense vibration of pumps, piping and valves, in compact, three-dimensional, high-flow circuit. Excitation due to flow noise and pump vane-passing.

P11 Main steam piping in power plant. Low-frequency flow-induced vibration. Unit load restricted until additional restraints added to reduce vibration.

P12 Process plant. Large, complex effluent line. Intense flow-induced vibration caused fatigue failures of instrumentation items.

P13 Turbine inlet piping in power plant. Low-frequency flow-induced vibration. Dampers added to control vibration.

P14(T) Large circulating-pump test facility. Severe vibration of test-loop piping, during test of pump under partial voiding condition.

P12(T) Same system as P12. Occasional severe transients.

VALVES

V1 Power plant governer valves of multi-plug bar lift type. Acoustically induced valve-pipe instability in a limited load range caused failure of servo spindles and couplings.

V2 Rough operation of deaerator level-control valves. Low-frequency vibration caused fracture of valve yoke.

V3 Reheater safety valves on gas-fired steam generator. Severe vibration due to flow turbulence interacting with acoustics of stub column. Pivot pins vibrated through drop levers in a few months. Annual inspection revealed severe wear on valve internals.

V4 Severe high-frequency vibration of turbine inlet piping, downstream of governor valves, at low load. Caused fatigue failures of several large flange bolts, and load restriction until valve modified.

V5(T) Steam reject piping, same system as case P8. Appreciable transient displacements upon opening reject-to-condenser valves.

STRUCTURES

S1 Turbine hall concrete floor. Low-frequency vibration transmitted from steam piping through supports and restraints.

S1(T) Same system as case S1. Transient increase in floor vibration levels during valve testing.

APPENDIX B

CONSULTANT POSITION PAPERS

(Included in this appendix are the six position papers developed by consultants to the Task Group on Other Dynamic Loads and Load Combinations.)

83C1269
0353C

Position Paper

Event Combination Associated with Dynamic Load and Load Combinations Applicable to Nuclear Power Plant Piping

by: J.D. Stevenson

April 1984

Position Paper

Event Combinations for Dynamic Load and Load Combinations in Nuclear Power Plant Piping

by: J.D. Stevenson[1]

1.0 STATEMENT OF THE ISSUES

Combined dynamic events considered as a design bases for nuclear power plant safety related (Seismic Category I) piping should have combined event probabilities consistent with single design basis event probabilities. Single event probabilities which are considered as a design basis for nuclear power plant safety related piping range from 10^{-2} to 10^{-3}/year for the OBE, 10^{-3} to 10^{-4}/year for the SSE(1), and an estimated 10^{-5}/year[2] for a DEGB loss of coolant accident(8) and about 10^{-7}/year for the externally generated plant missiles.(2) Event probabilities should not be confused with radiological release consequence probabilities in excess of prescribed limits. Radiological release consequences probabilities of a postulated event in excess of the exposure guidelines of 10CFR100 are required to be less than about 10^{-7}/year.(3) In calculating radiological release probabilities in excess of 10CRF100 the mitigating effect of plant design features should be considered.

The dynamic events which currently normally must be considered in the design of safety related piping systems are identified as follows:

(1) Earthquake (OBE)
(2) Earthquake (SSE)
(3) Pipe Break (DEGB)

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[2] Pipe break includes both a postulated slot or longitudinal type rupture as well as the double ended guillotine break, DEGB. The DEGB is normally limiting in the piping system from a thermal-hydraulic energy release standpoint, but the slot break often governs the maximum reaction load on the piping and supports, and also governs jet impingement effects on adjacent components. For reactor coolant loop piping, the slot type break has generally been eliminated as a design basis. Rigorous stress analyses show the potential for a slot type pipe rupture is much less than for a DEGB. The DEGB is distinguished from other accident induced internal dynamic events because it has traditionally been combined with other dynamic events such as earthquake while other internal accident load have not.

(4) Water (Steam) Hammer
(5) External Missiles, Blast - Accident & Environmental
(6) Internal Missiles, Blast and Jet Impingement - Accident[3]

Vibratory loads typically associated with fluid flow or rotating equipment operation will be considered in a separate position paper.

It has been common practice since the mid 1960's to consider the first three dynamic events listed, OBE, SSE (or their equivalents),[4] and the DEGB as separate design basis events. From the beginning in the design of containment systems the OBE and SSE loads have been combined with the DEGB loads. Starting in 1967 the combination of OBE and SSE with the DEGB loads has been considered for design of the reactor coolant system. The manner in which these events were converted to loads used in design of piping systems and their supports with particular application to reactor coolant systems has historically undergone several major changes as discussed in Reference 4. An edited version of Reference 4 which provides historical background and perspective has been included with this position paper as Appendix A.

The purpose of this position paper is to discuss and recommend how the six dynamic events identified herein which individually and in combination are currently considered as design basis for nuclear power plant piping design should rationally be combined with other dynamic events to form design bases. Consideration of changes in individual dynamic event characterizations as a design basis while obviously of importance is beyond the scope of this position paper.

2.0 DISCUSSION OF ISSUES

2.1 Combined SSE and DEGB

Both individual earthquake and DEGB events as well as dynamic event combinations are undergoing intensive NRC study and reappraisal at this time. For example, the DEGB may be eliminated as a design basis LOCA given leak before break considerations and implementation of associated augmented in-service inspection and monitoring programs, in particular, for the reactor coolant system piping.(5)

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[3] Except for the earthquake, the containment and other Seismic Category I (Safety Related) structures are generally designed to preclude other environmental or external accident events from affecting safety related piping located within such structures.

[4] Many operating nuclear power stations used earthquake design nomenclature different from the OBE AND SSE designation. In general, all stations have used a two earthquake design criteria. The smaller earthquake should be considered equivalent to the OBE and the larger equivalent to the SSE.

The joint or combined consideration of earthquake (SSE) and DEGB as a design basis event for nuclear power plant reactor coolant and main steam and feedwater piping began about 1967 <u>and continues to this day as a formal regulatory requirement</u>. This event combination was developed as a regulatory requirement since at the time of its inception there was little technical data available to establish the degree of dependency between an SSE event and a resultant DEGB. Lacking any quantification of the dependency relationship the NRC postulated this event combination for design purposes.

The combined SSE and a single DEGB has long been assailed by the nuclear industry in the U.S. as being an irrational regulatory requirement.(6) The argument presented is as follows:

> It can easily be shown that if the SSE (upper bound probability of event $< 10^{-3}$/yr(1)) and the DEGB (probability of event $< 10^{-5}$/yr(8)) are independent then given the relative short duration of the two events[5] their joint probability of occurrence is less than 10^{-8}/yr regardless of their durations which, in general, would place this event combination probability below the 10^{-7}/yr threshold for consideration as a design basis.[6]
>
> The probability of the simultaneous occurrence of two independent events of finite duration t_1 and t_2 in a year can be formulated as follows(7):
>
> $$\text{Pr (Two event Combination)} = \frac{Pr(1)\ Pr(2)\ (t_1 + t_2)}{T}$$
>
> where:
>
> Pr(1) = Probability of Event(1) per year: In the case of the SSE = 10^{-3}
>
> Pr(2) = Probability of Event (2) per year: In the case of the DEGB = 10^{-5}
>
> T = One year (minutes)
>
> t_1 = Duration of Event (1) For SSE Assume = 1.0 minute
>
> t_2 = Duration of Event (2) For DEGB Assume = 1.0 minute

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[5] For a detailed discussion of the probability of the Simultaneous Occurrence of Rare Independent Events see Reference 7.

[6] A similar argument can be made for the containment design basis but this is outside the scope of this paper.

Therefore, the probability of the simultaneous occurrence of an SSE and DEGB per year assuming independence and a finite duration for each event on one minute each is:

$$\text{Pr (SSE and DEGB)} = \frac{10^{-3} \cdot 10^{-5} (1+1)}{5.26 \times 10^{5}} \simeq 4.0 \times 10^{-14}/\text{yr.}$$

The probability of the simultaneous occurrence of two dependent events of finite duration is a function of the degree of dependence between the events. The probability for the simultaneous occurrence of events 1 and 2 considering dependence is defined:

Pr (Two Event Combination) = Pr (2/1) Pr (1)

where:

Pr (2/1) = is the conditional probability for the occurrence of event 2, during the occurrence of event 1, given the occurrence of event 1.

In Reference 9 a conditional probability of DEGB from a seismic event causing a support system failure was estimated at 10^{-7}/yr. If the conditional probability of the DEGB from all seismic causes is conservatively estimated at 10^{-6}/yr., then the

$$\text{Pr (SSE and DEGB)} = 10^{-6} \cdot 10^{-3} = 10^{-9}/\text{yr.}$$

which again is well below the thershold probability level of 10^{-7}/yr. for a design basis consideration.

It should be noted that once dependence has been established between two events the duration of the two events in developing joint probabilities becomes less important in defining joint probability level. In the limit for completely dependent events

Pr (2/1) = 1

and the joint probability of Pr (1 and 2) reduces to Pr (1) regardless of the duration of either event 1 or 2.

Therefore, if independence between the SSE and DEGB can be established then their combination should not be a design basis. Alternatively, if independence cannot be estalished between the SSE and DEGB then the probability of simultaneous occurrance of event SSE and DEGB as a function of the degree of dependence varies from about 10^{-13}/yr. to 10^{-3}/yr. In general it would be irrational to assume as a design basis that the actual degree of correlation would be such that only a single pipe break (DEGB) would occur as a result of an SSE. Either there is strong correlation where several DEGB's resulting from earthquake should be postulated or there is weak or no correlation in which case no combination is required.

This is not to say that some lesser level of LOCA should not rationally be considered in conjunction with the SSE but, in general, resultant loads on the reactor coolant system piping and supports for this combined loading would be less with a DEGB acting alone.

NRC consideration is being given to decoupling the SSE from the DEGB for PWR reactor coolant systems design.[5,9] Based on a rigorous technical evaluation of a particular PWR reactor coolant system,[8] which effectively has established independence between the SSE and DEGB. The text of Reference 5 and 9 are included with this position paper as Attachments 1 and 2.

The results of the Reference 8 study may be quite applicable to other PWR and BWR reactor coolant systems and probably to other high energy safety related systems in BWR and PWR plants as well. However, detailed consideration should be given to the significant differences between the various reactor coolant and auxiliary systems before a generic recommendation is made.

These significant differences fall in the following categories:

(1) Materials
(2) Stress Levels
(3) Stress Corrosion Potential
(4) Support Capability

Materials in reactor coolant system piping include both austenitic and ferritic steels. The sensitivity of results of the Reference 8 study which considered an austenitic steel to different materials including welds and heat effected zones should be performed to assure general applicability to the different types of materials in use. This issue has been explored in considerable depth in NUREG/CR-2301[10].

Thermal stress levels in reactor coolant system piping are dependent on the amount of restraint in the system. Westinghouse and Combustion Engineering PWR and General Electric BWR reactor coolant systems employ moving major reactor coolant system components which tends to minimize piping restraint, thereby reducing thermal stresses in the piping. Major reactor coolant components in some B&W PWR reactor coolant system are fixed, hence, thermal stresses in the piping must be accomodated by reactor coolant piping flexibility. In addition, as a result of the major component support restraint of the piping some additional restraint of free end displacement stress would be developed in the component supports.

As in the case of thermal stresses, seismic stresses in the various reactor coolant systems as a function of system geometry and elevation above the containment base also tend to differ. GE and B&W reactor coolant systems extend more than 100 feet above the containment base. Westinghouse and Combustion Engineering reactor coolant piping typically are within 20 feet of the containment base mat. This difference in elevation may result in significantly different stress levels in the piping for the same seismic design ground response spectra, with higher seismic stresses expected for GE and B&W systems.

Conclusions reached as to the potential for pipe rupture based on stress levels in one reactor coolant system piping may not be directly applicable to other systems.

Corrosive stress induced cracking in reactor coolant systems is highly dependent on system chemistry. The higher level of dissolved oxygen in BWR systems as compared to PWR's tends to increase relative stress corrosion cracking potential in BWR's and has resulted in cracking in BWR recirculation loop piping.

Finally, reactor coolant system support capacity has been identified(9) as the dominate consideration in the potential of a DEGB in RCL piping. Support failure analysis of 46 Westinghouse major reactor system components, pumps and steam generators(11) described in Reference 9, developed a median estimate of 10^{-7}/yr for a seismic induced probability of failure of the support which would result in a DEGB. Direct fracture mechanics crack growth induced DEGB from all postulated transients effects including seismic have been estimated in the 10^{-10} to 10^{-17} per year range. Similiar analyses have been performed for the Combustion Engineering reactor coolant system and supports with preliminary results indicating similar or lower probabilities of failure.

In this regard it should be understood that the support designs evaluated in Reference 11 were controlled by DEGB LOCA loads. These loads are typically four to ten times larger then SSE loads.(4) Therefore, use of DEGB based loads in design leads to the very low probabilities of failure which permits elimination of the combined SSE and DEGB LOCA as a design basis event. It should also be understood that elemination of DEGB events for PWR major reactor coolant system component does not necessarily extend to BWR components. While the BWR reactor vessel and supports are designed for the DEGB event, the recirculation pump has not always been so designed(4). For this reason, under SSE loadings, the probability of a recirculation pump support failure in a particular application may be significantly higher than the values presented in Reference 11 for Westinghouse PWR reactor coolant systems. Another reason for the low failure probabilities associated with earthquake induced loads is the relatively low stress levels permitted in piping and supports for seismic loads as compared to stress levels necessary to cause failure. Seismic stresses for all levels of design basis earthquakes are considered as primary stresses which restricts response to the essentially elastic range. Maximum calculated stresses as a result of the limiting SSE load in both the piping and supports are usually well below the maximum permitted by the ASME Code. This is because the OBE load tends to control seismic design of the system (1) and DEGB loads tend to control the overall design of the system.

In summary, there are many differences between reactor coolant and auxiliary systems among PWR's as well as between PWR's and BWR's. In spite of these differences and the difficulties in assessing the impact of the combined SSE and DEGB, it is my opinion that the conclusions reached regarding the technical acceptability of eliminating the combined SSE and DEGB for the Westinghouse PWR reactor coolant system main piping and the similar analysis performed on the Combustion Engineering RCS system, should also be applicable to other reactor coolant systems and high energy piping. However, specific evaluation of these other systems should still be conducted to affirm such a judgement.

2.2 Combined Water Hammer and OBE or SSE Events

In discussing water hammer events it is necessary to distinguish between anticipated and unanticipated water hammer events. Anticipated water hammer events should be considered as design basis events to the same extent than any other anticipated operating transient is considered. Unanticipated water hammer dynamic events by definition are not considered as a design basis since they are not identified a priori, and safety related piping response to them must be accommodated by design margins built into normal operation or by the design basis accident, DBA. Water hammer has been identified by the NRC as Safety Issue 1-A. Reference 12 presents a recent summary and evaluation of water hammer events in nuclear power plants as well as an identification of a variety of conditions which can lead to the phenomenon.

Unanticipated water hammer events can in general be catagorized as accidents even though in most cases they do not lead to rupture or leakage of the effected system. They do, however, typically result in piping and piping supports responding well into the inelastic region which may damage and tend to reduce the usage factor or future load carrying capacity of the piping and its supports.

Anticipated water hammer as a design basis for safety related piping has not seen wide application in plant design in the past. Water hammer loads as individual design bases dynamic events are currently being highlighted by the NRC and addressed in proposed revisions to several Standard Review Plans.[13,14,15,16,17,18,19]

Water hammer combined with other dynamic events should be based on a causative relationship between such events and water hammer. For example, a major earthquake in the absence of a low frequency filter, would be expected to cause a turbine trip. This trip would result in steam line relief valve operation. In BWR's, this results in a safety relief valve discharge into the containment suppression system. In general, any relatively rapid actuation of a valve (typically less than a few seconds) either opening or closing can cause water (steam) hammer. When such valve operation or other transient operation which can cause water hammer results from an earthquake, the two events should be combined as a design basis. Such an event combination should be based on anticipated system behavior and be included in the ASME Boiler and Pressure Vessel Code mandated Design Specification [20] used to define design loading requirements to the designer.

2.3 Combined Water Hammer and Engineered Safety System Operation (Accident)

Other potential combined dynamic events involving water hammer are found in the engineered safety systems. As a result of a LOCA or other DBA, the engineered safety systems may be required to perform their design function, which usually requires rapid actuation and transient operation of the system. Given the dynamic consequences of a LOCA or DBA which triggered the actuation of the safety system, it may be necessary to consider a combination of the two dynamic events, LOCA or DBA plus water hammer in the safety system as a design basis. However, such interactions are highly system design dependent. Hence, it is difficult to generalize whether such combined dynamic events should be considered as a design basis for a particular system.

2.4 Other DBA Combined With Earthquake

The dynamic events identified in Section 1.0 which are applicable to other safety related piping systems are based on a postulated DBA event (other then DEGB, LOCA) and potentially includes internal missiles, blast (rapid differential pressurize rise) and jet impingement. While in theory the DBA is considered in combination with the SSE as a NRC regulatory requirement, designers have generally layed out their safety related systems such that the effects in the broken system do not interact with other safety related systems or otherwise reduce redundancy below acceptable limits as permitted in the Standard Review Plan 3.6.2. This is done by installing pipe whip restraints, barriers and restricting jet impingements. Design of such restraints and barriers, consistent with the current requiremtns for LOCA, should consider the SSE event in combination with the pipe break event. Based on the decoupling of earthquake and pipe break research associated with leak before break considerations, it may be possible to eliminate this combination in the future.[7]

The evaluation of pipe break in a given system includes consideration of OBE seismic stresses. Based on a causative relationship between earthquake and pipe break research discussed in Section 2.1 and its continuance, it may be possible in the future to eliminate earthquake as having a causitive effect in pipe break for all high energy systems.

3.0 PROPOSED RECOMMENDATIONS

In this section general recommendations are made. These recommendations are based on the review performed of existing information and current research. They are also based on a judgement as to what continuing and needed future research on the relationship between earthquake and pipe rupture for all types of reactor coolant systems and other safety related piping systems will conclude. Specific recommendations to changes to NRC regulatory requirements are also included in this section.

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[7] It is also anticipated that the pipe break dynamic event as a design basis for all safety related piping will be greatly reduced or eliminated by "leak before break" considerations in the future.

3.1 General Recommendations

General recommendations relative to combined dynamic events are made as follows:

(1) Combined earthquakes and DBA in piping need not be postulated in design when evidence is presented to exclude such a combination from occurring using either deterministic or probabilistic arguments. Such evidence now exists on Westinghouse and Combustion Engineering reactor coolant systems and is being developed for B & W and General Electric reactor coolant systems. Further evaluation is needed to extend this decoupling criteria to auxiliary piping systems.

(2) The SSE earthquake combined with a limiting LOCA which rationally has a joint probability of occurrence in the 10^{-5}/yr range should be considered as a design basis event.[8] This combined event should be specifically identified (e.g., SSE combined with pressurized surge line break). This combination will serve as a replacement criteria for item 1 above for a LOCA DEGB combined with the SSE.

(3) Emphasis should be focused on having the licensee identify plant conditions where an earthquake (OBE or SSE) would result in an anticipated water hammer event (safety relief valve opeation) and the resultant earthquake and water hammer event combination be considered as a design basis event.

(4) Emphasis should be focused on having the licensee identify where actuation of an engineered safety system resulting from a plant dynamic event would result in an anticipated water hammer. In such cases, the plant dynamic event and the resultant anticipated water hammer event combination should be considered as a design basis event.

3.2 Specific Recommendation

Current NRC regulatory documents applicable to dynamic event combinations have been reviewed with recommended changes indicated herein. Recommended specific changes to General Criteria 2 and 4 are as shown in Figure 1.

3.2.1 USNRC Regulatory Guides 1.48 and 1.67

These documents should be cancelled since they have been superceded by Appendix A of SRP 3.9.3.

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[8] It is also anticipated that the pipe break dynamic event as a design basis for all safety related piping will be significantly reduced or eliminated by "leak before break" considerations in the future.

3.2.2 USNRC Standard Review Plan 3.9.2

Ref. Section II. Acceptance Criteria 5.

The reference to the "most severe" LOCA should be deleted. It is suggested that the term "design basis" LOCA be substituted. Elsewhere in the SRP the term design basis LOCA should be defined as follows:

> The Design Basis LOCA is defined as that LOCA event either alone or in combination with other events where the LOCA event or event combination has a probability of occurrence greater than about 10^{-5}/year.

Ref. Section IV. Evaluation of Findings 4.

The reference to "postulated" loss of coolant accident should be changed to "design basis". The "postulated" main steam line rupture (for a BWR) should also be changed to "design basis". The term Design Basis Main Steam Line Break should also be defined in a manner similar to that given above for the Design Basis LOCA.

3.2.3 USNRC Standard Review Plan 3.9.3

Ref. Appendix A, Section A. Introduction

In paragraph 2, 3 and 4 specific reference should be made to the approved ANSI standards ANSI/ANS-51.1-1983 and ANSI/ANS 52.1-1983 which contain the ANS compiled safety criteria for light water reactors. These are intended to provide the guidance applicants require with regard to the selection of acceptable design and service stress limits. Obviously, there may not be comlplete NRC agreement with the 51.1 and 52.1 positions and exceptions would be taken as appropiate.

Ref. Appendix A.

The use of the term LOCA in Table 1 needs clarification because LOCA should not necessarily mean or include a double ended pipe break.

The specific recommendations needed to modify current regulatory requirements to decoupled DEGB LOCA and SSE as recommended herein are few. However, this change would have a significant effect on piping support design, particularly if the LOCA DEGB event alone is significantly modified as the result of leak before break considerations. Obviously, an extension of the leak before break concept to all high energy piping would have a major impact on nuclear plant design and costs.

4.0 REGULATORY VALUE/IMPACT

4.1 Combined LOCA DEGB and SSE

Value/Impact with regard to decoupling the SSE and LOCA DEGB while significant are not as great as might be assumed without detailed study. This is because the LOCA DEGB loads dominate the SSE load by a factor which typically ranges from 4.0 to 10.0. Therefore the addition of the SSE loading to the LOCA DEGB is a relatively small incremental change . In reference 4 the hardware cost of the LOCA DEGB plus SSE combination were estimated in January 1980. Assuming a 25 percent increase between 1980 and 1984 dollars the load combination is estimated as follows:

A. PWR - 1300MWe

1. Reactor Building Internal Structure		\$112,500
2. RCS Supports		\$750,000
	Total	\$862,500

B. BWR - 1300MWe

1. Reactor Building Internal Structure		\$ 32,000
2. RCS Piping Supports		\$375.000
	Total	\$407,500

Not included in the above totals for PWR's and BWR's is any consideration of containment structure costs, the effect on reactor vessel internals or engineering costs associated with consideration of the combined load case. Evaluation of containment structure and reactor internal costs tend to be plant specific. As to engineering assume 10,000 additional engineering manhours allocated to this load case for a PWR 15,000 engineering manhours for a BWR reactor coolant system design. Resultant engineering cost differentials at \$50.00/hour would be \$500,000 and \$750,000 respectively in 1984 dollars for PWR's and BWR's. Please note these are direct cost estimates (1984 dollars) in the total amount of \$1,362,500 for a 1300 MWe PWR and \$1,157,500 for a 1300 MWe BWR for combined SSE and LOCA DEGB loads. Total costs which include all indirect plus direct costs would be approximately three times the values shown. These estimates do not include cost effects on the containment structure or the reactor internals due to the load combination. Such information would require additional study beyond the scope of this paper.

4.2 Other Combined Load Events

It is not possible, based on the data currently available, to estimate with any accuracy the cost of the other combined load events discussed in Sections 2.2, 2.3 and 2.4 of this paper. However, it is my opinion that hardware costs associated with such combinations are relatively small (less than \$2,000,000 in direct cost per plant) but the engineering effort necessary to establish such requirements is significant. I would estimate at least 200,000 manhours of additional engineering time is spent on this combined load engineering effort at a direct cost in excess of \$10,000,000.

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(14) Standard Review Plant 5.4.7, "Residual Heat Removal (RHR) System," Proposed Rev. 3 NUREG-0800.

(15) Standard Review Plan 6.3, "Emergency Core Cooling System," Proposed Rev. 2 NUREG-0800.

(16) Standard Review Plan 9.2.1, "Station Service Water System," Proposed Rev. 3 NUREG-0800.

(17) Standard Review Plan 9.2.2, "Reactor Auxiliary Cooling Water Systems," Proposed Rev. 2 NUREG-0800.

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(19) Standard Review Plan 10.4.7, "Condensate and Feedwater System," Proposed Rev. 3 NUREG-0800.

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APPENDIX A

COST AND SAFETY MARGIN ASSESSMENT OF THE EFFECTS OF DESIGN FOR COMBINATION OF LARGE LOCA AND SSE LOADS

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SECTION 1

INTRODUCTION

This report assesses the effect on safety and cost of the requirement to combine loss-of-coolant-accident (LOCA) and safety-shutdown earthquake (SSE) loads in the design of nuclear power plants. Analysis is limited mainly to plants recently completed or near completion, where current definitions of LOCA and SSE loading phenomena require or may require substantial modification to as-built or in-place structures and equipment. This effort is being performed to provide information regarding LOCA-SSE decoupling efforts for the Load Combination Program conducted at Lawrence Livermore National Laboratory.

Since 1967, (1) light water reactors in the U.S. have been designed to withstand combined SSE and maximum LOCA loadings. However, the actual SSE and LOCA loads considered in design have undergone significant change since that date. This report deals mainly with the evolution of SSE and LOCA loadings and their effect on the safety and cost of plants now in the active construction phase.

SECTION 2

HISTORICAL DEVELOPMENT OF THE SSE AND LOCA LOADS

2.1 SAFE SHUTDOWN EARTHQUAKE LOAD (SSE)

Design of nuclear power plants with respect to seismic requirements generally paralleled that of conventional structures until 1964. Before that time, in regions of low seismic activity in the East, South, and Midwest, no seismic design requirement usually existed. If one was imposed, it was typically in the range of 0.02 to 0.05 g taken as a static g-force on both equipment and building. Such small seismic forces were of little or no consequence in design. In regions of high seismic activity, the Uniform Building Code or equivalent local building codes were applied and equipment was typically designed for a static accleleration of 0.20 g.

Development of today's seismic design criteria parallels the growth of the commercial nuclear power industry. The current criteria came into being in 1963 with the publication of TID 7024. (2) Before that date, seismic design simply considered a static horizontal load at the center of gravity of the equipment or the plant building and equipment. Dynamic analysis, including response spectrum, damping, and resonance effects, had been available since the late 1950's in connection with development programs for strategic missiles and Navy weapons; however, these methods normally were not used except by a few defense contractors and specialized consultants. Starting about 1964, the Atomic Energy Commission required utilities and the architect/engineers who design nuclear power facilities to adopt the methods of dynamic analysis to seismic design of equipment and structures.

Dynamic response spectrum analysis was limited at first to building structure design. Calculation of seismic loads on equipment assumed that either the building or the equipment was rigid. In the first assumption, the ground motion passed directly through the building to the equipment. In the second one, the equipment simply received the inertia loading felt by the building at the point of attachment. In some instances, particularly for boiling water reactor (BWR) plants, equipment was evaluated by use of a floor response spectrum. This was derived from the ground response spectrum by increasing the zero-period acceleration to equal the floor acceleration determined from the building dynamic analysis. This philosophy characterized the period from 1964 to 1967.

Beginning in 1967, the potential for resonance between the building and the equipment was considered in equipment design. This approach generated "amplified" floor response spectra to be used in design of equipment located at a specific point in the building. Subsequent work has centered mainly on the development of more conservative response spectra. Steady movement occurred away from the Housner type (3) response spectra (1964-1971), which were based on a weighted averaging of individual response spectra, toward those of the more conservative modified-Newmark type (4) (1971-1973), based on an approximate enveloping of individual response spectra, and finally to the Regulatory Guide 1.60 spectra, (5) or Newmark, Blume, Kapur (NBK) spectra (6). Based on one standard deviation from the mean value, these have formed the NRC basis for licensing since 1973.

Data from the 1971 San Fernando earthquake also significantly affected seismic design. They resulted in a general requirement, starting in 1973 East of the Rocky Mountains, to consider vertical zero-period ground acceleration equal to the horizontal acceleration in the frequency range between 3.5 and 33 Hz. A vertical acceleration equal to two-thirds of the horizontal spectra was used previously. Also in 1973, a spectrum specifically applicable to vertical response appeared for the first time in R.G. 1.60. In addition, it became a general requirement, in 1973, to combine two orthogonal, independent horizontal components with one component of vertical earthquake motion on an SRSS basis where only one horizontal resultant combined with vertical was considered previously. The values of structural damping used in nuclear plant design underwent a similar evolution, as did response spectra. The damping criteria ranged from Housner (3) to Newmark (4) to Regulatory Guide 1.61 (7). Little change has occurred since 1973 in the seismic design procedures typically used in the design of nuclear power plants.

Definitions of earthquake input and acceptable behavior criteria have changed also. Starting in 1964, the dominant or independent earthquake considered in design was usually termed the Design Basis Earthquake (DBE). It was considered to be the largest earthquake ever recorded at the site. For the DBE, structures and equipment were required to meet existing design code requirements.

The significance for both concrete and steel structures was that a one-third increase in normal code allowable stresses was permitted. For piping, the then-applicable USAS B31.1 Code permitted a 20 percent increase over normal allowables by classifying earthquakes as occasional loads occurring less than 10 percent of the time. In general, no increase was allowed for other mechanical and electrical components. No generally acceptable structural design codes existed at that time for mechanical components other than vessels and piping; hence, in most instances it was left to the manufacturer to define in the preliminary safety analysis report (PSAR) the stress limits that would be permitted in meeting DBE induced loads. In 1967, the Nuclear Regulatory Commission (NRC) disallowed the one-third increase in stresses permitted for structures under the DBE loading. The PSAR made statements about the continued operability of active components (pumps, valves, electrical instrumentation, controls, and power supplies) in the event of a DBE. Similar statements were incorporated into procurement specifications, but means for demonstrating such operability were not usually defined.

In addition to the DBE, a maximum hypothetical earthquake (MHE) was defined as having twice the zero-period ground acceleration of the DBE. A generally acceptable earthquake nomenclature had not been developed at the time. Thus, the term described above as the DBE was often defined as the operational basis earthquake (OBE) and the term DBE was often applied to what is defined above as the MHE. Appendix A to 10 CFR 100, (8) published in 1971, finally established the current definitions of the smaller OBE and larger safe shutdown earthquake (SSE). It established the manner in which the SSE would be determined, and made the smaller OBE dependent on the size of the SSE.

The behavior criterion originally established for use with the MHE or SSE was "no loss of function," an expression with no well-defined meaning. An alternative criterion was "within yield stress after load redistribution";

that is, plastic hinge formation was permitted, but a failure mechanism was not. Currently, SSE loads combined with other applicable loads are carried by structures with a 1.6 increase in normal allowable stresses (9). For ASME equipment, Service Level D stress limits specified by the code (10) are used for passive components and Service Level B* stress limits for active components not otherwise qualified by test. Combined stresses for passive components of non-ASME equipment have typically been limited to yield stress.

Table 1 compares the amplifications associated with the various design ground response spectra that have been used in the past.

The evolution of the seismic design requirements from the late-1960's to the mid-1970's has introduced the changes in response spectra and damping values, the development of floor response spectra for three independent components of earthquakes, and the manner in which modes of response are combined. The impact on plant design has been to increase the seismic stress resultants in plant equipment by a factor of 2 to 3 for the same zero-period ground acceleration. Note, however, that equipment seismic stress resultants for some plants designed in the 1970's have decreased, in comparison to the earlier plants, because of better plant layout, more sophisticated analysis, and improved modeling techniques.

In addition to the increase in seismic stress resultants, a large increase in LOCA loads has taken place since the 1960's (see Section 2.2 of this report). These developments have greatly affected later plants, which have to be designed for the combined new SSE and LOCA loads.

2.2 LOSS OF COOLANT OR DESIGN BASIS ACCIDENT LOAD

As with earthquake load development for nuclear power plants, design for effects of loss of coolant accident (LOCA) has also changed. Before 1965, designing for resistance to LOCA effects were generally limited to containment and core support. A convenient design basis for containment pressurization, selected for both boiling water reactor (BWR) and pressurized water reactor (PWR) nuclear plants, consisted of the release of the reactor coolant inventory to the containment volume. For BWRs this was accomplished through a pressure suppression system, later designated the Mark I containment. The mechanism by which this release would take place was selected as the double-ended rupture of the largest reactor coolant line. This permitted a thermal-hydraulic analysis of reactor system blowdown through the break opening for calculating containment pressure and temperature transients and loading on core support structures. From these, containment design pressure and temperature as well as resultant loads on steam and feedwater containment penetrations could be selected. The postulated pipe break also causes other effects generally not considered in design, except for some earlier plants such as Dresden-2 that did consider pipe rupture restraint. These effects are

* Recent changes in the ASME Section III Code have increased Service Level B allowable stresses for Class 1 components. Compared to the 1980 Edition of the ASME Code, Service Level A would be more applicable.

as follows:

a. Break reaction loads on structure, components, or supports restraining the broken pipe
b. Formation of fluid jets at the point of break and the effects of their potential impingement on other components or structures
c. Transient pressurization of local compartments within the containment
d. Differential transient pressurization of local compartments within the containment
e. Transient LOCA loads on other componments in unbroken systems

Starting in 1965 for PWRs, the designers have considered the break reaction loads of supports for reactor coolant components (pumps, steam generators, reactor vessels). Their purpose is to ensure that the steam generator can sustain LOCA reaction load of the attached pipe without gross deformation that could rupture the attached feedwater or steam lines. Such a secondary rupture would, in turn, release inventory from the secondary side of the nuclear steam supply system to the containment. The PWR containment structure is not designed to accomodate blowdown of both a primary and a secondary nuclear steam supply system. LOCA reaction load effects on the broken system are not considered in BWRs except in regard to core support structures and containment penetrations, since primary and secondary systems are not separated, hence, a LOCA in a BWR will result in blowdown of both the reactor coolant (recirculation system) and the steam system.

From 1965 until 1968, LOCA reaction loads were usually treated as staticallyapplied loads

$$F = p_0 A \quad (1)$$

where:

F = the static applied load at the postulated point of break perpendicular to the break area

p_0 = system operating or design pressure, typically 2500 psi in a PWR and 1100 psi in a BWR

A = area of the postulated break; varies from 4.5 to 9.5 ft^2.

Since about 1968, the dynamic characteristics of these loads have been considered in the form

$$F = K_1 K_2 p_0 A \quad (2)$$

where

K_1 = dynamic load factor due to sudden application of the load; typically taken as a value between 1.0 and 2.0, depending on the amount of ductility assumed in the system

K_2 = thrust coefficient; $K_2 = 2.0$ for subcooled water and 1.26 for steam for two phase, mixed or transient flow cases; thermo-hydraulic computer codes typically are used.

By 1972, time-history forcing functions for multidegree-of-freedom dynamic models of the reactor coolant system began to be available (11). Design of local compartments within containment for transient pressurization also began about 1965. However, differential transient pressurization of local compartments was not considered in both PWRs and BWRs until 1975. Fluid jet impingement loads have had little practical effect to-date on design of the reactor coolant systems, because pipe whip restraints severely restrict the amount of displacement for postulated guillotine ruptures, thus impeding the formation of jets. Nonetheless, jet formation from postulated slot or longitudinal rupture due to crack formation and stability is still the subject of considerable research. (12,13) It has long been the contention of the nuclear power industry, both in the U.S. and elsewhere, that substantial crack lengths must develop, particularly in the longitudinal direction, before instability occurs and the crack can open enough to form a significant jet. Such cracks, it is though, would cause leak before break as they grow by fatigue, and the leaks would give enough warning to permit depressurization and repair. This is commonly referred to as the leak before break criterion. (14) In addition, it can generally be shown that the stress field in the longitudinal direction of reactor coolant piping is significantly greater than that in the hoop direction. Consequently, guillotine ruptures are far more likely and reactor manufacturers have not seriously considered the possiblity of slot-type ruptures. (15) Jet impingement effects are not considered in this assessmnt of LOCA plus SSE.

The effect of transient LOCA loads on other components in unbroken systems is generally not considered, except where such loads produce response input applicable to the component. Such response input to other systems as a consequence of LOCA is essentially limited to BWR containments, which employ a pressure-suppression water pool system. The discharge of pressure relief valves (PRVs) in BWRs can occur independently of a LOCA but not an SSE and produce responses similar to those of a LOCA in systems located outside the primary shield wall. Although it is reasonable to postulate a LOCA independent of an SSE, a PRV discharge occurs as a consequence of a strong motion earthquake. This effect began to be considered in design about 1975. Because of changes in LOCA load determination from 1965 to the present, the calculated break reaction loads have increased by a factor of 1.5 to 2.5.

The historical development of behavior criteria used to evaluate LOCA effects are similar to those established for SSE loads. In general, no behavior criterion is specified for the pipe segment containing the postulated LOCA break. In PWRs, the no behavior criterion segment is defined as the broken leg of the broken loop. In the unbroken leg of the broken loop and in other affected components (reactor vessel, steam generator and attached steam and feedwater piping, reactor coolant pumps and their supports), the same behavior criteria as used for the SSE are required (ASME Section III Service Level D for passive components and Service Level B for active components). The behavior criterion used in the reactor core for PWRs is normally associated with deformation and is not limited by stress such that control rod engagement and cooling paths remain open, as determined by test. For BWRs the requirements of ASME III Service Level D also apply to the core.

For BWRs no behavior criterion applies to the broken system or loop. The behavior criteria for unbroken loops and the reactor core internals as a result of LOCA in a BWR are the same as in PWR. Note that the postulated broken segments or systems for both PWRs and BWRs are pipe whip restrained which cannot interact and cause the loss of function of other safety related systems.

SECTION 3

DEVELOPMENT OF STRUCTURAL BEHAVIOR CRITERIA TO ACCOMMODATE SSE AND LOCA LOADING

The structural behavior criteria used for combined SSE and LOCA effects are essentially the same as for either SSE or LOCA acting alone.

SECTION 4

IMPACT OF CURRENT COMBINED SSE AND LOCA REQUIREMENTS ON EXISTING PRESSURIZED WATER REACTOR NUCLEAR PLANTS

The effect of combined SSE and LOCA for PWR stations is essentially limited to the following structures, systems, and components:

a. Reactor coolant compartment surrounding the broken loop
b. Primary shield wall surrounding the reactor vessel
c. Broken reactor coolant-loop components (reactor coolant pumps, steam generator) and supports and piping in the unbroken leg of the broken loop
d. The emergency core cooling (ECC), residual heat removal (RHR), and chemical volume control (CVC) systems attached to the unbroken leg of the broken loop
e. Reactor vessel and its support; reactor core and its supports
f. Containment structure

The 1973 introduction of the Reference R.G. 1.60 ground response spectra, the damping values of R.G. 1.61, and the redefinition of the reference earthquake motion as one of two simultaneously acting horizontal components rather than a single horizontal resultant has had the most effect in increasing the seismic loads.

Introduction in 1975 of the effect of asymmetrical-transient compartment pressurization due to LOCA has had the most effect in increasing LOCA load. By far the most significant result of asymmetric transient pressurization is associated with postulated breaks within the primary shield wall surrounding the reactor vessel. This factor is most predominant in Westinghouse plants, where in-service inspection of the reactor vessel from the inside permits a gap of 6 inches between the vessel and the shield wall. However, in plants where external reactor vessel inspection is intended it is common to have a 2-foot gap that greatly increases the vent area and reduces the asymmetric transient loading effect. The effect of asymmetry is most pronounced on the reactor vessel, its supports, and the reactor internals. Table 2 summarizes the typical LOCA and SSE design loads and their relative effects on design.

4.1 EFFECT OF COMBINED SSE AND LOCA ON REACTOR COOLANT COMPARTMENT

To the extent that the operating decks above the reactor coolant compartment obstruct flow from the compartment if a LOCA occurs within that space, a net uplift on the compartment walls will result. The overturning effect of the SSE adds directly to the uplift. Thus, the combined LOCA and SSE add to the requirements of vertical reinforcement and anchorage for the walls of the reactor coolant compartment. Without the combined LOCA and SSE, it may be possible to anchor the reactor coolant compartment walls in the fill slab above the containment liner. With the combined LOCA and SSE, vertical anchorage, which connects through the leak-tight containment liner into the containment base mat, usually must be provided. The SSE and LOCA combination adds some 90 mechanical anchors. It increases, by an average of 15 percent, about 30 tons of the vertical reinforcement steel. This effect normally is considered in both current and anticipated future designs.

4.2 EFFECT OF COMBINED SSE AND LOCA ON PRIMARY SHIELD WALL

Since the primary shield wall surrounding, and in most cases* supporting, the reactor vessels is typically 5 to 6 feet thick, the effect of asymmetric LOCA and loading of the reactor pressure vessel support on the shield wall is minimal, because they tend to produce high bending stresses that the thick concrete wall can readily accommodate. The combination of SSE and LOCA has little or no effect on the design of the primary shield wall beyond what would be considered for either acting alone.

4.3 EFFECTS OF COMBINED SSE AND LOCA ON BROKEN LOOP COMPONENTS

Table 2 shows typical calculated reaction loads from LOCAs and SSE on the steam generator and reactor coolant pump and on their support structures as a function of the period during which the design was performed. The combination of LOCA and SSE has had little or no effect on the design of the components themselves or the attached piping, but the combination has increased component support costs 10 to 25 percent. The total direct in-place support cost per 300 MWe in PWR plants is $650,000 for a steam generator and $200,000 for a reactor coolant pump. Hence, approximately $150,000 per 300 MWe is currently chargeable to the SSE plus LOCA combination.

4.4 EFFECTS OF COMBINED SSE AND LOCA ON ATTACHED STEAM AND FEEDWATER SYSTEMS

Because the steam generator is effectively restrained by snubbers designed to accommodate the combined LOCA and SSE, the LOCA and SSE loads induced in the primary system are essentially isolated from the steam and feedwater line and thus do not affect their design. The cost of the isolating snubber is included in the support costs for the steam generator.

4.5 EFFECTS OF COMBINED SSE AND LOCA ON THE ATTACHED SYSTEMS OF THE UNBROKEN LEG OF THE BROKEN LOOPS

The emergency core cooling (ECC), residual heat removal (RHR), and chemical volume control (CVC) systems attached to the unbroken leg of the broken loop are assumed to function in the event of a LOCA. Therefore, they must accommodate the thermal hydraulic transients and displacements associated with LOCA in the broken loop, plus the SSE loading, without loss of function. It is difficult to assess the influence of the combination without evaluating, in detail, layout geometries that tend to be plant specific. Seismic requirements have generally dictated the support design of these systems, independently of any pipe whip restraints. Combining LOCA and SSE does not appear to be important in design of these systems, since the broken leg of a broken loop is isolated from the unbroken leg by a major component whose deformation in response to LOCA reaction loads is limited. For the purposes of this study, these attached lines do not have any significant effect on overall design or cost relative to combined SSE and LOCA effects.

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* Most of the PWRs designed by Stone and Webster Engineering Corporation had the reactor vessel supported by a steel neutron shield tank, not by the primary shield wall.

4.6 EFFECTS OF COMBINED SSE AND LOCA ON REACTOR VESSEL, SUPPORTS, AND REACTOR CORE AND SUPPORTS

a) SSE load effects have had a greater impact than LOCA on the reactor core and core supports, as shown in Table 2. The explanation is that some of the blowdown load bypasses the reactor core and the core tends to have an amplified response to seismic excitation of the vessel.
b) An asymmetric LOCA load within the primary shield wall has significant lateral impact on the vessel, its supports, and its internals, and it tends to add directly to earthquake effects.
c) Nuclear steam supply system (NSSS) suppliers prefer to substitute zircoloy, with its superior nuclear properties, for inconel core structures, thus reducing the core structural load carrying capacity up to 20 percent.

Studies to evaluate the effect of current LOCA (post-1975) combined with old (pre-1973) SSE loads on PWR reactor internals are still underway. Results are expected in June 1980. (16) Since blowdown areas affect the lateral loads on the reactor vessel, caused by asymmetric loading of postulated LOCA inside the primary shield, pipe displacement restraints of the type shown in Fig. 1 may need to be installed on existing plants to minimize asymmetric loading. In fact, the existing analysis for postulated breaks outside the primary shield wall suggests that such restraints should be installed. It is estimated that their in-place cost would be $50,000 per 300 MWe and they would be required in Westinghouse plants only.

Primary shield wall restraints may not be strong enough to accommodate current LOCA plus SSE loads on the reactor core and core supports. A modified core with sufficient strength would cost about $8 million and the core support structures another $4 million. This modified design would also tend to reduce plant performance. Consequential costs of such reductions have not been considered in this study. It is highly unlikely that a substantial modification of the reactor vessel support would be required, but the backfit cost for an existing plant would be $30 million to $50 million, assuming the modification was feasible.

4.7 EFFECTS OF COMBINED SSE AND LOCA ON THE CONTAINMENT STRUCTURE

Containment structures have always been designed for combined LOCA and SSE, and neither the localized dynamic amplification factor nor the asymmetric loads have had an appreciable effect on containment design pressures. Modification resulting from SSE seismic load phenomena has been minimized by the use of more rigorous analytical techniques. Historical changes in LOCA and SSE effects have not influenced design, but their combination as a design requirement adds to the vertical and diagonal shear* reinforcement in concrete containment structures. The current effect of SSE and LOCA combination on concrete containment design is to increase vertical reinforcement 15 percent to 300 tons and require diagonal reinforcement in a deformed bar concrete containment to 600 tons.

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* Diagonal shear reinforcement is not required in prestressed concrete containments.

SECTION 5

IMPACT OF CURRENT COMBINED SSE AND LOCA REQUIREMENTS ON EXISTING BOILING WATER REACTOR NUCLEAR PLANTS

The current effect of combined SSE and LOCA is more complicated for BWR than for PWR plants. The complication arises from the effect of the LOCA blowdown on the suppression pool and from the development of secondary loads, such as chugging and condensation-oscillation, which tend to be dynamic and periodic. Since the loads act directly on the containment and containment internal structure, they tend to excite the reactor building with a resultant response spectrum. This spectrum, when combined with the seismic spectra, may control design of supports for mechanical and electrical equipment and distribution system throughout the containment structure and reactor building*.

Determining the effect of the combined LOCA and SSE is further complicated by the fact that the main steam safety relief valves (SRVs) discharge into the containment and suppression pool. The discharge generates loads similar to LOCA resulting directly from a turbine trip that occurs during any significant seismic disturbance. Because an SSE (or a lesser earthquake) causes an SRV discharge, their resultant effects cannot clearly be separated. As far as possible, this study disassociates the effects of (SSE + LOCA) + SRV from those of EQ + SRV. Figures 2 through 6 compare the response spectral curves developed for LOCA with the SRV curves, showing that LOCA may control design inside the shield wall. This factor cannot be determined by simple comparison of spectral curves, since the behavior limits associated with SRV discharge are more restrictive than those of LOCA.

The effects of combined SSE and LOCA for BWR stations are, for practical purposes, limited to the following structures, systems, and components:

a. Shield wall surrounding the reactor vessel
b. Reactor vessel support pedestal and skirt
c. Reactor core and core supports
d. Steam, feedwater, and recirculation lines in the unbroken loop
e. Containment internal structure
f. Mechanical and electrical equipment and distribution systems in the containment
g. Containment structure

The effects of a LOCA on equipment in a PWR tend to concentrate on the components and on the unbroken leg of the broken loop. In a BWR, they tend to concentrate on the other unbroken loop and on other seismic Category 1 systems located within the shield wall surrounding the vessel. Outside the shield wall, OBE combined with SRV discharge tends to govern design. These distinctions between BWRs and PWRs arise because the effects of the postulated LOCA need not be isolated from the steam and feedwater system in a BWR, since

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* Mark I and Mark II containments are located within the reactor building and supported by a common foundation. For a typical Mark III containment, a reactor auxiliary building houses those systems not located within the containment. The auxiliary building and containment may have separate foundations.

the energy from all three systems in a BWR is assumed released to containment in the event of a LOCA. Also, as mentioned earlier, some of the current BWR LOCA loads are periodic or oscillatory. They excite the entire containment-reactor building structure, imparting shock spectra loading to components not otherwise involved in the LOCA. Similar excitations of building equipment do not occur in PWR dry containments.

5.1 EFFECTS OF COMBINED SSE AND LOCA ON SHIELD WALL SURROUNDING THE REACTOR VESSEL

Because the shield wall surrounding the reactor vessel is 3 to 5 feet thick, the design of the vessel in a BWR is not particularly affected by the LOCA-SSE combination. Reinforcement of the wall is controlled predominantly by LOCA-induced pressurization. The current consideration of asymmetrical LOCA pressurization produces on the wall a lateral local force that augments the SSE overturning effect, thereby adding to the needed amount of steel acting in the vertical direction. The combination adds about 5 percent to the vertical steel, an increase of 5 tons of reinforcement.

5.2 EFFECTS OF COMBINED SSE AND LOCA ON THE VESSEL SUPPORT PEDESTAL AND SKIRT

As in the shield wall, the dominant lateral load is developed by the asymmetric LOCA pressurization of the annulus between the shield wall and the reactor vessel. This effect is more pronounced in Mark III containments because there is no upper lateral restraint stabilizing the reactor vessel. The effect of LOCA and SSE combination on the concrete support pedestal and anchorage system of the reactor vessel would be to increase vertical steel requirements slightly in the pedestal and require anchor bolts of higher strength. One can assume a 5-ton increase in pedestal vertical reinforcement or cylinder wall thickness and a $5,000 additional cost due to the change in bolt material. The pedestal design may be either of reinforced concrete or of concentric steel cylinders with the annulus filled with concrete.

5.3 EFFECT OF SSE AND LOCA COMBINATION ON REACTOR CORE AND CORE SUPPORTS

LOCA-induced asymmetric annulus pressurization imparts a large, impulsive excitation to the reactor vessel, inducing a response in the core and core supports that adds directly to the SSE-induced response. For a BWR plant designed to sustain an SSE having 0.2-g zero-period ground acceleration (ZPGA), the LOCA induced loads, including asymmetric annulus pressurization, produces horizontal load effects on the core and core supports roughly equal to two-thirds of the SSE effect. For vertical loading, the LOCA effects are several times those of the SSE; hence the LOCA-SSE vertical load combination has little effect on the design load of the core and core support compared to LOCA and SSE treated separately.

Based on the analyses performed to date, the General Electric Company (11) does not expect a need to modify core or core supports for the combination of currently defined SSE and OBE loading. If core and core support modifications were required, we estimate that their costs would be somewhat more than those estimated for PWRs. They would be roughly $12 million for the core and $6 million for the core supports on existing plants because BWR internals weigh more than those of PWRs.

5.4 EFFECT OF SSE AND LOCA COMBINATION ON UNBROKEN STEAM FEEDWATER AND RECIRCULATION LINES

The effect of SSE plus LOCA on the unbroken steam, feedwater, and recirculation lines is quite pronounced within the primary shield wall. Figures 4, 7, and 8 show typical input load effects resulting from SSE and LOCA on the recirculation line. Figure 2 is a LOCA-induced response spectrum for the recirculation line as a result of a postulated feedwater line rupture. Figure 7 is an SSE response spectrum for 2 percent equipment damping for a BWR-6, Mark III system. Figure 8 is an OBE response spectrum, also applicable to the recirculation line. Figure 3 is a response spectrum applicable to the main steam line as a result of postulated feedwater line rupture. Figures 4 and 5 show the input spectra for the shield wall resulting from a postulated LOCA in the feedwater line and SRV discharge.

A comparison of SSE, LOCA, and SRV discharge spectra in the graphs shows that the spectrum for LOCA-induced feedwater line break is one to three times the SRV spectrum, depending on location within the primary shield wall. The average LOCA effect in unbroken lines within the shield wall is roughly 65 percent of the effect of the SSE load, which is 30 percent greater than the effect of SRV discharge. Consequently, the extreme load combination for LOCA plus SSE tends to govern design within the shield wall. This effect should be more pronounced for postulated steam and recirculation line LOCA, since these lines are larger than the feedwater line and because blowdown would be more rapid. Outside the shield wall, however, the effect of SRV discharge tends to be similar to LOCA loading; hence, given the more conservative SRV behavior criteria typically associated with ASME Section III Service Level B stress limits, the LOCA-SSE load combination is less likely to control design.

5.5 EFFECT OF COMBINED SSE AND LOCA ON THE CONTAINMENT INTERNAL STRUCTURE

As can be seen by comparing Fig. 6 with Fig. 7, design of the reactor shield wall and pedestal support is controlled primarily by a postulated LOCA-induced asymmetric pipe break within the shield wall. The containment internal structures, other than the reactor shield wall and reactor pedestal, tend to be controlled by earthquake plus the pool dynamics response associated with SRV discharge. The pool dynamic spectra resulting from SRV discharge are similar to those developed from LOCA outside the shield wall. Since SRV discharge loading has a dependency relationship with earthquakes, the decoupling of SSE and LOCA would not affect the relationship. Design and cost differentials for containment internal structures other than the shield wall and pedestal are not significantly affected by the SSE and LOCA combination.

5.6 EFFECT OF COMBINED SSE AND LOCA ON MECHANICAL AND ELECTRICAL EQUIPMENT AND DISTRIBUTION SYSTEMS IN THE CONTAINMENT AND REACTOR BUILDING

The combination of SRV-induced pool dynamics and OBE earthquake spectra normally governs design of mechanical and electrical equipment and distribution system within the containment and reactor building. Since the pool dynamic spectra resulting from SRV and LOCA are similar, the decoupling of SSE and LOCA loads would not appreciably affect design or the cost of equipment and distribution systems. This is not to say that the design and cost of such equipment and systems has not been significantly affected, since pool dynamic loads have been explicitly considered in design.

5.7 EFFECT OF COMBINED SSE AND LOCA ON CONTAINMENT STRUCTURE

Most Mark III containments in the U.S. are structural steel, with torispherical dome and right circular cylinders anchored to a flat slab of reinforced concrete. To counter the effect of localized pool dynamic loads on the steel containment, vertical and horizontal stiffners are added to the containment shell. These stiffners also increase the buckling resistance of the containment shell to accommodate earthquake-induced overturning compressive stresses. The net effect of SSE plus LOCA on containment shell design is negligible, since SRV discharge in the worst case has effects similar to those of LOCA. The same conclusion can be reached relative to MARK II and Mark I containment systems.

SECTION 6

ASSESSMENT OF THE EFFECT OF SSE AND LOCA COMBINATION ON EXTREME LOAD AND NORMAL OPERATING SAFETY MARGIN

The possibility has long been recognized that, in designing nuclear plant facilities, normal plant operation may be affected by extreme loads such as LOCA and/or SSE. LOCA plus SSE effects on PWR plants require heavier supports or restraints on components of reactor coolant loops. These supports should have little or no effect on normal operation of the components. For both PWRs and BWRs, reactor system components, including reactor core and core supports, until now have been designed to optimize operating performance. None of the LOCA, SSE, or combined LOCA and SSE loadings have been limiting load case. Re-analysis associated with the currently redefined larger SSE and LOCA load effects may require modification of reactor component internals, particularly for the reactor core and core supports. This concern appears more applicable to PWRs than to BWRs. If hardware modifications are required, they will certainly impact cost and may also impact operating performance.

In Section 5.4 of this report, it was concluded that the LOCA and SSE load combination may control the design of BWR piping systems within the shield wall. Additional supports usually are required to provide more restraints for the piping system in accommodation of the combined LOCA and SSE loads. These restraints normally are snubbers, which are assumed, in design, not to affect normal thermal loads. However, the existence of such restraints inevitably reduces the overall system reliability in normal operation because an ideal snubber has never been designed.

To quantify the effect that SSE + LOCA has on design of piping systems, a typical BWR recirculation line originally designed without consideration of the currently defined SRV or LOCA was reevaluated, using the spectra presented in the graphs, for the following load cases:

	ASME Section III Service
1. DL + press + thermal	(Condition A)
2. DL + press + OBE + SRV	(Condition B)
3. DL + press + thermal + OBE + SRV	(Condition B)
4. DL + press + LOCA	(Condition D)
5. DL + press + SSE + SRV	(Condition D)
6. DL + press + (SSE + LOCA)	(Condition D)

where:

DL = dead load
Press = design pressure
Thermal = design temperature
OBE = operating basis earthquake
LOCA = loss of coolant accident (feedwater line)
SRV = safety relief valve discharge
SSE = safe shutdown earthquake

The results are shown in Table 3. This analysis indicates that the SSE + LOCA is a limiting load case resulting in a 153 percent overstress compared with 135 percent for the OBE + SRV. Two additional snubbers are required in the recirculation line to bring resultant stresses within code limits. The LOCA + SSE effect would tend to be even more pronounced for postulated main steam or recirculation line rupture on other lines within the shield wall, since the LOCA blowdown rate would be somewhat greater for these systems than for the feedwater line. It is estimated that the combination of SSE + LOCA requires 20 to 40 additional snubbers for the piping located within the shield wall. Outside the shield wall, the OBE + SRV load case would govern design; hence, the combination of SSE + LOCA would not affect normal operation of the plant.

It can be concluded, therefore, that the only significant impact of the SSE + LOCA load combination, outside the reactor vessel, on normal operation is the addition of pipe supports to piping located within the shield wall on BWRs. The supports reduce the reliability of such systems. The effect of the combination on PWR reactor internals is still being evaluated. The General Electric Company concluded that the combination should not affect core internals in BWRs. (17)

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(11) Stevenson, J. D., "Criteria and Design of Pressurized Water Reactor Coolant System Support Structures - State of the Art", Paper F 5/4, Proceedings 1st International Conference on Structural Mechanics in Reactor Technology, Berlin, 1971.

(12) Agres, D. J., "Research for Rationalization of Nuclear Power Plant Pipe Break Criteria", Paper F 5/1, Proceedings 5th International Conference on Structural Mechanics in Reactor Technology, Berlin, 1977.

(13) WCAP 9558, Mechanistic Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through Wall Crack, Westinghouse Electric Co., August 1979.

(14) Duff, C. G., "National Policies - Panel Presentation", Proceedings of International Seminar on Probabilistic and Extreme Load Design of Nuclear Plant Facilities", San Francisco, California, August, 1977.

(15) WCAP 9628, Westinghouse Owners Group, "Asymmetric LOCA Loads Evaluation - Phase B", Westinghouse Electric Co., November 1979.

(16) Stevenson, J. D., "Seismic Margins as They Affect the Verification of Seismic Design Adequacy of Mechanical and Electrical Components," presented at Atomic Industrial Form Workshop on Reactor Licensing and Safety", December 1974

(17) Correspondence, Subramanian, C. V., General Electric Company, to Stevenson, J. D., Woodward-Clyde Consultants, May 28, 1980.

Table 1. Seismic response, with one degree of freedom maximum amplification factor, compared to peak ground motion.

Percent Critical Damping	Newmark[a] Acc.	Vel.	Disp.	Housner Acc.	Vel.[b]	Disp.[c]
0	6.4	4.0	2.5	6.2	2.7	1.4
0.5	5.8	3.6	2.2	4.6	---	---
1.5	5.2	3.2	2.9	3.1	---	---
2.0	4.3	2.8	1.8	2.3	1.6	1.2
5.0	2.6	1.9	1.4	1.5	1.3	1.0
7.0	1.9	1.5	1.2	---	---	---
10	1.5	1.3	1.1	---	---	---
20	1.2	1.1	1.0	---	---	---
	Modified Newmark				NBK[d]	
0	6.4	---	---	---	---	---
0.5	5.8	---	---	5.95	3.7	3.2
1.0	---	---	---	---	---	---
2.0	3.5	---	---	4.25	3.2	2.5
5.0	2.6	---	---	3.13	2.4	2.05
7.0	1.9	---	---	2.72	2.1	1.88
10	1.5	---	---	2.28	1.8	1.70
20	---	---	---	---	---	---

[a] Based on a standard earthquake; maximum values of ground motion:

Acceleration = 0.10 g
Velocity = 4.8 inches/second
Displacement = 3.6 inches

[b] Relative to base at a period of 2.0 seconds (Fig. 1.19 in TID 7024)

[c] Relative to base at a period of 3.0 seconds (Fig. 1.23 in TID 7024)

[d] Acceleration amplification maximum at 2.5 Hz, decreasing approx. 20 percent at 9 Hz and back to no amplification for all values of damping at 33 Hz and above. Displacement and velocity amplification based on a maximum of 0.25 Hz

Table 2. Historical summary of typical LOCA and SSE (0.2-g ZPGA) equivalent static loads on reactor coolant system components

Item	Nominal Component Weight	Load/period (Kips)					
		LOCA 1965-1968	SSE 1965-1968	LOCA 1968-1975	SSE 1968-1973	LOCA 1975-pres.	SSE 1973-pres.
I PWR							
Reactor vessel	1500-2500	1600-3200K	750	3200-6400	1000	5000-8000	2000
Reactor internals	800-1200	800-1600	500	1000-3200	800	3500-6000	1600
Steam generator	1000-1400	1600-3200	500 to 700	3200-6400	1000-1400	3500-7000	2000-3000
Reactor coolant pump	200	1600	100	3200	200	3200	300
II BWR							
Reactor vessel	4000	800	1600	1600	3200	7000	6000
Reactor internals	2700	400	1600	800	3200	3500	4000
Recirculation pump[a]	180	---	100	---	300	540	540

[a]LOCA pipe reaction loads on pump in broken loop not considered

ATTACHMENT 1 Reference 5

OCT 1 7 1983

Mr. Warren H. Owen
Executive Vice President
Engineering and Construction Department
Duke Power Company
422 South Church Street
Charlotte, NC 28242

Dear Mr. Owen:

Your letter of September 19, 1983 concerning pipe break design considerations has been referred to me for reply. In that letter you cite the work done by the industry in developing the leak-before-break concept for PWR main coolant piping. You also expressed the interest of Duke Power Company in reflecting the results of this work in your stations.

It appears that sufficient technical justification exists to consider decoupling of safe shutdown earthquake and LOCA loads. For PWR main coolant loops probabilistic analyses have indicated that the probability of a safe shutdown earthquake (SSE) causing a double-ended pipe break is extremely low. (Reference attached NUREG/CR-2189: Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant.) On a generic basis we are considering changes to current regulatory requirements in this area. Specifically, Standard Review Plan Sections 3.6.2 and 3.9.3 would have to be revised to accommodate such changes and possibly a revision to General Design Criteria (GDC) 2 and 4 in Title 10, Code of Federal Regulations, Part 50, Appendix A would be needed. Processing and approval of such changes may take one or two years to complete.

In a closely related area, we also believe that the technology now supports consideration of the leak-before-break performance of PWR main coolant loops. This performance is based on fracture mechanics analysis to demonstrate crack stability under the applied loadings and sufficient leakage detection. We will be considering additional regulatory changes to permit application of this concept, where appropriately justified, for both new and existing designs. These changes would effectively decouple LOCA and SSE since the LOCA loads would be negligible. The timing anticipated for processing and approval of these changes in regulatory requirements is expected to be about the same as those mentioned above.

OCT 1 7 1983

As you know, we have met with representatives of Duke Power and other owners in a generic meeting on this subject. In this meeting it was agreed to separate the industry proposals into three phases of resolution. These phases will cover reactor coolant loop piping, reactor coolant loop branch piping and piping in other plant systems, and the treatment of arbitrary intermediate breaks in all classes of plant systems. We are in the process of developing a detailed regulatory approach to be implemented for each of these three phases. With respect to the first phase, we can now approve application of the concept to eliminate the whip restraints associated with the asymmetric LOCA loads. The three phased approach should permit some additional selected application of the leak-before-break concept prior to completing all of the changes in regulatory requirements discussed above.

In following the approach we are developing, it is our intention to work closely with you to bring about expeditious resolution of these issues.

Sincerely,

Original Signed by
H. R. Denton

Harold R. Denton, Director
Office of Nuclear Reactor Regulation

Enclosure: NUREG/CR-2189

ATTACHMENT 2 Reference 9

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOV 18 1983

Dr. Otfried Voigt
Kraftwerk Union
Aktiengesellschaft
Berliner Strasse 295-303
P.O. Box 962
D-6050 Offenback Am Main
Federal Republic of Germany

Dear Dr. Voigt:

Regarding your desire to learn more about U.S. NRC thinking on (1) double-ended pipe breaks and (2) the relative advantages of stiff versus flexible piping, the following information is provided.

Research Information Letter No. 117 dated April 10, 1981, (Enclosure 1) concluded that, based on probabilistic fracture mechanics assessments of PWR primary piping, "Through-wall cracks are about a million times more likely to occur than double-ended guillotine breaks. This appears to offer substantial quantitative support in a probabilistic format for the leak-before-break hypothesis." In another place, it is stated that "Fatigue crack growth due to all transients, including earthquakes, is an extremely unlikely mechanism for inducing large LOCA [double-ended pipe rupture]."

Subsequently, in a June 14, 1983 letter from the ACRS to the NRC Executive Director for Operations (Enclosure 2) discussing the work reported in Enclosure 1, it was stated that "The principal risk comes not from the direct growth of cracks to a size that would be ruptured by an earthquake, but from failure due to indirect causes such as the earthquake-induced failure of the supports of heavy components, for example, the steam generators and pumps. We find this procedure to be an acceptable and proper approach to the problem, and the decoupling of the loss of coolant accident and seismic loads to be appropriate."

In response to this letter, on July 29, 1983, the Executive Director for Operations in a letter to the Chairman of the ACRS (Enclosure 3) stated that "Contractors have investigated the seismic reliability of 46 heavy component support systems on Westinghouse PWRs. .... It was determined that the probability of a double-ended guillotine break resulting from the seismic failure of heavy component support systems ranged from 10^{-5} to 10^{-10} per reactor year with a median estimate of 10^{-7} per year."

Finally, in an October 17, 1983 letter from Harold Denton, Director, Office of Nuclear Reactor Regulation, to the Executive Vice President of the Duke Power Company (Enclosure 4) it is stated that "It appears that sufficient technical justification exists to consider decoupling of safe shutdown earthquake and LOCA loads. For PWR main coolant loops probabilistic analyses have indicated that the probability of a safe shutdown earthquake (SSE) causing a double-ended pipe break is extremely low." Elsewhere in the same correspondence, Mr. Denton concludes that "We also believe that the technology now supports consideration of the leak-before-break performance of PWR main coolant loops."

Future activities relate to extending these investigations to BWR piping and to piping other than primary circuit piping at PWRs. Several reports are presently in preparation concerning this research. In the meantime, under separate cover, I am sending you the nine volumes of NUREG/CR-2189 entitled, "Probability of Pipe Fracture in the Primary Loop of a PWR Plant."

Turning to flexible piping design versus stiff piping design, certain advantages are immediately apparent for flexible piping systems (those with fewer pipe supports such as rigid restraints or snubbers) as indicated below:

- Flexible piping provides easier access for plant maintenance.
- Flexible piping reduces radiation exposure during maintenance.
- Flexible piping reduces thermal stresses during plant operation.
- Flexible piping costs less.

The central issue in our investigations over the last year or so has been how piping reliability is affected by stiffness and flexibility. We have concentrated on snubber-supported piping, and we have assumed in our investigations that snubbers have a non-zero failure rate and may fail in the "free" or "locked" mode. Our studies to date have included high, moderate, and low energy piping. The only failure modes we have investigated so far are pipe rupture and leaking, although we plan to extend these efforts to include the effect of flexibility and stiffness on the reliability of components on piping such as pumps and valves. The principal conclusions to date are:

- For high energy piping, assuming realistic snubber failure rates, too many snubbers placed to reduce seismic loads actually reduce overall reliability.

  Failure in the locked mode (typical of mechanical snubbers) contributes significantly to this reduction in reliability while failure in the free mode (typical of hydraulic snubbers) essentially leads to a less flexible piping whose reliability would be only slightly different than if the hydraulic snubber had functioned properly.

- For low energy piping, assuming realistic snubber failure rates, snubbers placed to reduce seismic loads increase reliability slightly. Nonetheless, snubbers are infrequently placed on low energy piping.

These investigations, which are motivated by our desire to learn under what conditions we may safely remove snubbers from nuclear reactor piping, will continue for the next year or so.

Enclosures 5 and 6 describe our fiscal year 1984 work activities at Lawrence Livermore National Laboratories related to these matters. A special Piping Review Committee has been established to help integrate these research results into the licensing process. I hope you find this information useful to your needs. I look forward to receiving similar information from you on this subject.

Sincerely,

G. A. Arlotto

Guy A. Arlotto, Director
Division of Engineering Technology
Office of Nuclear Regulatory Research

Enclosures: As stated

POSITION PAPER ON RESPONSE COMBINATIONS

by

R. P. Kennedy

March, 1984

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1.0 STATEMENT OF THE ISSUES

Dynamic analyses of piping systems are generally performed by either the time history analysis method or the response spectrum analysis method with the response spectrum method being most commonly used. Several response combination issues arise when the response spectrum method is used. The important issues are:

1. How should independent support motion response spectra be used for multiple supported subsystems such as piping?

2. How should modal responses be combined for well-spaced modes, closely-spaced modes, and high-frequency modes to determine the total inertial responses. High-frequency modes are those modes with frequencies above the frequency at which spectral accelerations begin to reduce to about the zero period acceleration (ZPA).

3. How should responses due to different spatial components of the input motion be combined?

4. Should spatial component responses be combined before or after modal responses?

5. How should multiple support displacement responses be combined to determine the total support displacement (secondary) responses?

6. How should the total support displacement (secondary) responses be combined with the total inertial (primary) responses?

These six (6) issues are currently addressed for piping systems by Standard Review Plan (SRP) Sections 3.7.2, 3.7.3 and 3.9.2 and Regulatory Guide 1.92 (Reference 1). Current practice with respect to the six issues above as addressed by the SRP and R.G. 1.92 is:

1. Use a single uniform support motion response spectrum which envelopes all of the independent support motion response spectra appropriate for the multiple piping supports.

2. Combine well-spaced and closely-spaced modes in accordance with any one of the acceptable methods of R.G. 1.92. No guidance is given for the combination of high-frequency modes and practices differ.

3. Responses due to different spatial components are combined by the square-root-sum-of-the-squares (SRSS) method.

4. No guidance is given as to whether spatial component responses should be combined before or after modal component responses. When closely-spaced modes or high-frequency modes exist, the order of the response combination (spatial versus modal) influences the end results. In my experience, it has been general practice to combine modal responses prior to combining spatial component responses.

5. Multiple support displacement responses are combined in the most unfavorable way, i.e., absolutely, to determine the total support displacement (secondary) responses.

6. Total support displacement (secondary) responses are combined absolutely with the total inertial (primary) responses.

Some of these response combination practices as defined by the SRP and Reg. Guide 1.92 are controversial, potentially excessively conservative, and not well-founded theoretically. Therefore, several efforts have been initiated by the NRC to develop recommended changes to the SRP and Reg. Guide 1.92. One such effort was performed as part of the Task Action Plan A-40 effort to identify and quantify the conservatism inherent in the seismic design sequence of current NRC criteria. Reference 2, which was developed as part of this program, recommended in 1979 changes to SRP 3.7.2 and Reg. Guide 1.92 to incorporate more realistic, technically sound, and less conservative modal combination rules for closely-spaced modes, and to provide explicit guidance for modal combination of high-frequency modes. Although these recommendations were specifically made for civil structures, they are equally applicable for piping systems.

A second effort specifically directed toward response combination rules for multiply-supported piping systems is currently ongoing at Brookhaven National Laboratory (BNL). Table 1 presents interim NRC staff-recommended rules for combining responses using the Independent Support Motion Response Spectrum Analysis Method (ISMA) based upon this ongoing BNL research (Reference 28).

In general, I consider the response combination rules laid out in Table 1 to be well thought out and reasonable. Certainly those rules represent an improvement (less conservative and more realistic) over the earlier SRP requirements. My consulting comments will be based upon the assumption that Table 1 represents the current NRC staff position and will be directed toward some further improvements in the response combination rules summarized in Table 1.

2.0 DISCUSSION OF THE ISSUES

2.1 DISCUSSION OF TABLE 1 RESPONSE COMBINATION ALGORITHM

Dynamic responses of multiply-supported piping systems can be subdivided into inertial (primary) responses and relative support displacement-induced (secondary) responses. Table 1 treats these two response components separately calling the inertial responses dynamic components and the relative support displacement-induced responses pseudo-static components. For simplicity, I will call these two components primary and secondary.

2.1.1 Multiple Support Response Spectra

To obtain the primary response component by the response spectrum method for multiply-supported piping systems with differing input at each support, one must first decide whether to use the Uniform Support Motion Response Spectrum Analysis (USMA) technique or the Independent Support Motion Response Spectrum Analysis (ISMA) technique. With the USMA technique, a single response spectrum which envelopes each of the multiple support input response spectra is developed and input at all of the supports for a particular input directional component. In many cases, this approach leads to excessive conservatism. As a result, the ISMA technique has recently come into vogue. By this technique, a single response spectrum is applied to a group of supports, but different input response spectra are applied to different groups of supports. In the extreme, each support might have a different input response spectrum. With the ISMA technique, one group of supports is moved at a time using the input response spectrum specified for these supports, with all other groups being stationary. The primary concern is how to combine the responses from each of the individual grouped analyses.

Brookhaven National Laboratory (BNL) is currently conducting extensive studies (Reference 29) on the ISMA technique. One of the primary questions being studied is how to combine grouped responses*. BNL has studied the absolute sum (ABS), algebraic sum (Algebraic) and the square-root-of-the-sum-of-the-squares (SRSS) methods of combining grouped responses.

The BNL preliminary results clearly indicate that the ISMA technique with ABS combination of grouped responses is consistently conservative when compared to time-history results. Sometimes the method is excessively conservative. The ISMA technique with ABS combination of grouped responses is sometimes more conservative than the USMA technique. Therefore, if ABS combination of grouped responses is required as indicated by Table 1, the ISMA technique will often not offer any significant advantage over the current USMA technique.

The Algebraic combination of grouped responses assumes that responses of all supports are essentially in-phase. For the case of different floor responses in the same structures, this assumption is often reasonable. However, in some cases, this assumption may be substantially incorrect and unconservative. The BNL studies indicate that the ISMA technique with Algebraic combination is generally conservative but can sometimes be unconservative.

The SRSS combination of grouped responses assumes that each of the independent response spectra are uncorrelated (random relative phasing of each frequency). Within my experience, the primary system (civil

---

* Grouped responses refers to responses computed from a common input applied to a specific group of supports in the ISMA technique. To obtain the responses due to input applied at all supports, the individual grouped responses must be combined.

structure) leads to considerable phase correlation between these independent response spectra. Therefore, the SRSS combination of grouped responses cannot be supported on any theoretical grounds for the case of different floor responses in the same structure. However, for reponses between different structures, this assumption is probably reasonable. The BNL studies indicate that the ISMA technique with SRSS combination is generally conservative but can sometimes be unconservative. The tentative BNL recommendation (Reference 29) is to use the SRSS combination of grouped responses. Because the SRSS combination method has no theoretical basis for the combination of responses from individual input groups within the same structure, and because of the limited data available, I cannot support the recommendation. At this time, I would recommend that if one has not retained relative phasing information, then it would be prudent to combine group responses by ABS as suggested in the NRC staff recommendations in Table 1 even though such a combination may often be excessively conservative. Only if one can demonstrate that the responses are reasonably phase uncorrelated should group responses be combined SRSS. Reasonable phase uncorrelation is likely between different structures.

The most appropriate way to combine independent group responses is to retain the relative phasing provided by the primary system. Techniques have been proposed for retaining and using this information. However, such techniques are still in their infancy and need further work before being accepted in the regulatory process. The NRC should encourage the development and use of such techniques in order to alleviate the excess conservatism of the ABS combination.

2.1.2 Modal Response Combination

Current accepted practice for modal response combination is based upon Reg. Guide 1.92 which suffers from two deficiencies:

1. Excessive conservatism in some cases for the combination of closely-spaced modes.

2. No guidance is given for the combination of high-frequency modes.

These topics will be discussed in detail in Sections 2.2 and 2.3, respectively.

2.1.3 Spatial Component Response Combination

Support input motions are defined in terms of x, y, and z orthogonal component directions. The total resultant peak x-directional support motion (XR) is made up of a combination of x, y, and z earthquake input components to the primary structure. If one assumes that these earthquake input components are uncorrelated (random relative phasing at each frequency) then the resultant peak support motions can be realistically obtained by the SRSS combination of the peak support motion due to each of the earthquake input components. Thus, if XX represents the peak x-direction support motion due to the x-direction earthquake input component, XY represents the peak x-direction support motion due to the y-direction earthquake input component, and YX represents the peak y-direction support motion due to the x-direction earthquake input component, etc., then the resultant peak support motions are given by:

$$XR = \sqrt{XX^2 + XY^2 + XZ^2}$$

$$YR = \sqrt{YX^2 + YY^2 + YZ^2}$$

$$ZR = \sqrt{ZX^2 + ZY^2 + ZZ^2}$$

So far as piping response is concerned, the question is how should piping responses due to XR, YR, and ZR be combined. There is no assurance that XR, YR, and ZR are uncorrelated even though the x-direction, y-direction, and z-direction earthquake input motions are uncorrelated. The peak support motion components XX, YY, and ZZ will be uncorrelated. However, the peak support motion components XX, YX, and ZX are likely to have substantial phase correlation due to the primary system (civil structure).

In most practical structures, XX predominates over XY and XZ while YY predominates over YX and YZ and ZZ predominates over ZX and ZY. In these cases, XR, YR, and ZR will be uncorrelated and the SRSS combination of directional piping responses, as defined in Table 1, is appropriate. However, it is possible for XX, YY, and ZZ to not predominate and in these rarer cases, the SRSS combination of directional piping responses might not be appropriate. This issue deserves further study.

2.1.4 Order of Modal and Spatial Combination

The NRC staff recommended combination algorithm (Table 1) suggests that the directional responses be combined by SRSS prior to combining modes by Reg. Guide 1.92. The more common practice has been to combine modes prior to combining directional responses. When closely-spaced or high-frequency modes which are not combined by SRSS are important, then the order of the combination can make a difference on the end results. In my experience, this difference is seldom greater than 20% for significant response quantities. The BNL studies (Reference 29) also indicate that the sequence of combination is not significant. Philosophically, it appears to me to be more appropriate to combine modes first and to combine directional components last rather than as shown in Table 1. However, either order of combination should be allowed.

So long as closely-spaced modes must be combined absolutely as currently required by Reg. Guide 1.92, then combining directions first and modes second will lead to higher combined responses than when modes are combined first and directions second. Thus, one might argue that the combination order in Table 1 is conservative. However, if closely-spaced modes are combined algebraically as is correct (see Section 2.2), then one cannot say which order of combination is conservative relative to the opposite order.

2.1.5 Combination of Groupings of Support Displacement Responses

The issue of combining secondary responses due to independent groupings of multiple support displacements is the same as that discussed for primary stresses by the ISMA technique in Subsection 2.1.1. An SRSS combination of grouped secondary responses would be questionable although often conservative. The most appropriate way to combine independent group responses is to retain the relative phasing of support motions provided by the primary system. If this relative phasing information has not been retained, then it is prudent to combine grouped secondary responses by ABS as recommended in Table 1 from the NRC staff. Such a combination is generally very conservative.

2.1.6 Combination of Directions of Support Displacement Responses

The spatial combination for secondary responses should be performed in the same manner as for primary responses. If the SRSS spatial combination method is judged acceptable for primary responses (see Subsection 2.1.3), it should also be adequate for secondary responses. Thus, I do not agree with the NRC staff interim recommendations (Table 1) that the spatial combination be by ABS for secondary responses while by SRSS for primary responses.

2.1.7 Combination of Secondary and Primary Responses

For piping systems, it is generally unnecessary to combine secondary (support displacement-induced) and primary (inertial-induced) responses. The ASME code contains separate stress allowables for primary and secondary stresses. However, in some cases such as fatigue evaluation, one might need a total combined response. Then the question arises as to how to do the combination.

Displacement-induced (secondary) responses and inertial-induced (primary) responses are not phase uncorrelated. In fact, they often have a negative phase correlation. Therefore, the SRSS combination of primary and secondary responses cannot be justified on theoretical grounds. However, peak primary responses and peak secondary responses would be highly unlikely to occur concurrently. Therefore, an ABS combination would generally be excessively conservative. An SRSS combination is preferable even though unjustified on theoretical grounds. Ibrahim (Reference 21) has demonstrated that the SRSS combined primary and secondary responses have a 96.4% non-exceedance probability. The BNL study (Reference 29) also recommends an SRSS combination.

2.2 CLOSELY-SPACED MODES

Many methods have been proposed and used for the combination of peak modal responses (References 1, 3 through 10). The common methods are:

1. ABS (absolute sum)
2. Algebraic Sum
3. SRSS (square-root-of-the-sum-of-the-squares) [Equation (3), Reference (1)]
4. Grouping Method [Equation (4), Reference (1)]
5. Ten percent method [Equation (5), Reference (1)]
6. DSC (Rosenblueth Double Sum Method) [Reference (3)]
7. NRC-DSC (NRC Double Sum Method) [Equation 8, Reference (1)]
8. CQC (Complete Quadratic Combination) [Equation (12) of Reference (9)]
9. ARC (Advanced Response Combination) [Reference (10)]

All of these methods can be expressed in either one or the other of the following two general equations which include certain modal coupling factors C_{jk}, (Reference 10):

$$R = \sqrt{\sum_j \sum_k C_{jk} R_j R_k} \tag{1}$$

$$R = \sqrt{\sum_j \sum_k C_{jk} |R_j R_k|} \tag{2}$$

where j and k are mode numbers and R_j and R_k are peak responses in modes j and k, respectively. In every case, when j equals k, $C_{jk} = 1.0$. Otherwise, the coupling factors and appropriate equation number (Equation 1 or 2) given in Table 2 apply.

Obviously, the ABS and Algebraic Sum Methods can be cast in a more simple format then Equations (1) or (2). However, they have been cast in this format for comparison purposes.

The ABS method is always conservative because it assumes worst-case phasing of all modes. It is generally excessively conservative and unrealistic.

The Algebraic Sum method is the appropriate modal combination method for high-frequency modes as will be discussed in Section 2.3. This combination method applies whenever modes are reasonably in-phase (phase differences less than about 35 degrees) at the time of peak response. Such conditions exist for high-frequency modes. However, this method has sometimes been misapplied to lower frequency modes where the assumption of random phasing is more realistic. The only difference between the Algebraic Sum Method and the ABS Method is the retention of the relative response signs (Equation 1 versus Equation 2).

The SRSS method is based upon the assumption of random phasing of peak modal responses at the time of peak combined response. This assumption works well for widely-spaced modes except at high frequencies where modes are reasonably in-phase. The SRSS method is deficient for closely-spaced modes and high-frequency modes which are essentially in-phase. All of the remaining methods in Table 2 are attempts to correct these deficiencies in the SRSS method. Methods based upon Equation (1) approach Algebraic Summation when $C_{jk} = 1.0$ and SRSS when $C_{jk} = 0.0$ and are in-between for values of C_{jk} between 0.0 and 1.0. Similarly, methods based upon Equation (2) approach ABS when $C_{jk} = 1.0$ and SRSS when $C_{jk} = 0.0$.

Both the DSC (Rosenblueth Double-Sum-Combination) and the CQC (Complete Quadratic Combination) methods are theoretically based in random vibration theories. Both methods use Equation (1) so both are consistent with Algebraic Summation when $C_{jk} = 1.0$. The C_{jk} coefficients are given by:

<u>DSC Method (Reference 3)</u>

$$C_{jk} = \left[1+\left(\frac{(\omega_j'-\omega_k')}{(\beta_j' \ \omega_j+\beta_k' \ \omega_k)}\right)^2\right]^{-1} \qquad (3)$$

in which

$$\omega_j' = \omega_j\sqrt{(1-(\beta_j)^2)} \qquad (4)$$

$$\beta_j' = \beta_j+\frac{2}{S\omega_j} \qquad (5)$$

ω_j = natural frequency of the jth mode.

β_j = critical damping ratio for the jth mode.

S = time duration of "white noise" segment of earthquake excitation. For actual earthquake records, this may be represented by the strong motion segment characterized by extremely irregular accelerations of roughly equal intensity.

<u>CQC Method (Reference 8 or 9)</u>

$$C_{jk} = \frac{8\sqrt{(\beta_j\beta_k\omega_j\omega_k)}(\beta_j\omega_j+\beta_k\omega_k)\omega_j\omega_k}{(\omega_j^2-\omega_k^2)^2+4\beta_j\beta_k\omega_j\omega_k(\omega_j^2+\omega_k^2)+4(\beta_j^2+\beta_k^2)\omega_j^2\omega_k^2} \qquad (6)$$

Equation (6) is only strictly appropriate when the duration of strong input motion is long compared with the modal natural periods and when the input response spectrum is smooth over a wide range of frequencies. More complex expressions for C_{jk} accounting for duration and frequency content details are given in Reference 8.

The ARC Method is also similar to the DSC and CQC Methods except the C_{jk} coefficients are empirically based (Reference 10) rather than based upon random vibration theory.

The NRC-DSC Method (Equation 8 of Reference 1) represents a modification of the original DSC Method (Rosenblueth Double Sum Method). The NRC-DSC Method differs from the DSC Method in that Equation 2 (Absolute Signs) is used in lieu of Equation 1 (Algbebraic Signs). I can find no theoretical or empirical justification for the NRC-DSC Method. The only basis appears to be that it always is more conservative than the DSC Method.

The Grouping Method and Ten-Percent Method as described in Table 2 are both approximations to the NRC-DSC Method. For 5% damped structures, when $\omega_k/\omega_j = 1.1$ (10% frequency difference), the value of C_{jk} from either Equation 3 or Equation 6 will be about 0.50. Furthermore, at 5% damping, with frequency differences less than 10%, C_{jk} will be closer to 1.0 than to 0.0. With frequency differences greater than 10%, C_{jk} will be closer to 0.0 than 1.0. These approximate methods, using the above characteristics, save a considerable amount of computational time for structures with more than about 10 modes with only a minor change of results from those obtained by the NRC-DSC Method. However, both of these methods suffer from the same lack of either a theoretical or empirical basis and from the possibility of excessive conservatism as does the NRC-DSC Method.

Studies (References 4, 5, 9 and 11) have illustrated that for dynamic models with significant closely-spaced modes (frequency differences less than about 10%), both the DSC and the CQC Methods more closely approximate time history computed responses than does the SRSS Method. Both methods give very similar results with good accuracy for all problems studied (Reference 11). The NRC-DSC Method often introduces excessive conservatism when compared with the DSC or CQC Method or time history computed results (Reference 9).

The only apparent problem with either the DSC or CQC Methods is the increased computational time associated with including all the cross product terms for dynamic models with more than about 10 modes. This problem is easily eliminated by a minor approximation. Only the cross product terms where $C_{jk} \geq 0.5$ need to be included in Equation 1. When $C_{jk} < 0.5$, it is reasonable to assume $C_{jk} = 0.0$ which means SRSS modal combination. In the case of low damping values ($\beta \leq 5\%$), C_{jk} will exceed 0.5 only when modal frequencies are within 10% of each other. Thus, a practical rule becomes:

Frequencies Within 10% of Each Other

Compute C_{jk} by DSC (Equation 3) or CQC (Equation 6) Methods

Frequencies More than 10% Apart

$C_{jk} = 0.0$

Within my experience, this simplification never introduces more than a ±15% error from results obtained including all cross coupling terms.

2.3 HIGH-FREQUENCY MODES

2.3.1 Background

In a 1979 submittal for the Lawrence Livermore Laboratory A-40 Program effort (Reference 2), I demonstrated the inaccuracies associated with the use of the SRSS combination method* for high-frequency modes (modes in excess of the frequency at which the spectral acceleration returns to approximately the zero period acceleration, ZPA, which is about 33 Hz in the case of the R.G. 1.60 spectrum). This problem had also been illustrated by Biswas and Duff (Reference 12) and Gwinn and Waal (Reference 13). The basic problem is that the SRSS method assumes random

* The SRSS combination method as referred to herein means the conventional square-root-sum-of-squares method as modified for closely-spaced modes per the comments in Section 2.2.

phasing of modal responses at the time of peak response. However, higher frequency modes are all nearly in-phase with the input motion and thus are all nearly in-phase with each other. As noted in Section 2.2, when modal responses are all nearly in-phase, the modes should be combined by Algebraic Summation rather than by SRSS.

It is now apparent that there are three modal combination zones of interest.

First, there is a lower frequency zone corresponding approximately to the frequency range where the response spectrum is in the amplified spectral velocity domain. This zone corresponds to frequencies less than f^1 which will be defined later. However, for the Reg. Guide 1.60 response spectrum, f^1 may be as low as 1.5 Hz to 3.0 Hz. Below f^1, the total modal response can be combined by the SRSS method modified for closely-spaced modes.*

A second zone corresponds to the frequency range above the frequency f^r where f^r is defined as the rigid frequency at which the spectral acceleration, S_a, roughly returns to the peak zero period acceleration, ZPA. At these high frequencies, the seismic input motion does not contain significant energy content and the structure simply responds to the inertial forces from the peak ZPA in a pseudo-static fashion. The phasing of the maximum response from modes at these high frequencies (roughly 33 Hz and greater for the Reg. Guide 1.60 response spectrum) will be essentially deterministic and in accordance with this pseudo-static response to the peak ZPA. The combined response from modes with frequencies above f^r can either be determined by a pseudo-static response analyses as defined in Appendix A (taken from Reference 2) or by Algebraic Sum of all of these higher frequency modal responses. Both approaches lead to identical results and are theoretically correct.

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* It should also be noted that the SRSS method is also incaccurate at very low frequencies but this problem is of little importance to stiff nuclear power facilities and is not addressed herein.

However, the pseudo-static technique of Appendix A is generally more simple to use and is less susceptible to numerical errors which sometimes occurs with the algebraic summation of high frequency modes.

The third zone between the frequencies f^1 and f^r represents a transition region within which a portion of the modal responses should be combined by SRSS as modified for closely-spaced modes, and a portion of the response should be combined by Algebraic Sum. Close to f^1 essentially all of the modal response should be combined SRSS while close to f^r essentially all of the modal response should be combined by Algebraic Sum. The exact distribution between the portion to be combined by SRSS and the portion to be combined by Algebraic Sum is uncertain and is the subject of considerable recent study (References 14 through 20). Unfortunately, this transition region is the region within which most of the important piping system response modes lie. Therefore, modal combination in this transition region needs to be further discussed.

2.3.2 Recent Research

The publication and dissemination of NUREG/CR-1161 (Reference 2) has resulted in new research on the combination of higher frequency modes, including Lindley and Yow (Reference 14), Hadjian (Reference 15), Gupta (References 16 through 19) and Singh (Reference 20). This new research has indicated that my 1979 recommendation did not go far enough. Basically, the problem with the SRSS response combination method and the transition to algebraic summation occurs at frequencies well below that at which the spectral acceleration, S_a, returns roughly to the ZPA. Whereas, I illustrated that the SRSS method should not be used at frequencies above 33 Hz for the USNRC R.G. 1.60 spectra, this newer research illustrates that the same problems extend down to lower frequencies as well.

All of these approaches incorporate the idea that the total peak response is made up of two parts consisting of a damped periodic relative peak response, R^p, and a rigid peak response, R^r. The total damped periodic relative response, R^p, is obtained by the SRSS method of combining modal "relative" responses based upon the assumption that the phasing of these "relative" responses are uncorrelated with each other. The total rigid response, R^r, is obtained by algebraic summation of modal "rigid" responses because this rigid portion of total response is all in-phase with the ground motion. In understanding these methods, three frequencies need to be defined:

f^1 = lower frequency below which rigid and damped periodic relative responses are not additive. Below this frequency, the separation into rigid modal responses and damped periodic modal responses is unnecessary and the total modal responses can be combined by the SRSS method.

f^2 = upper frequency above which the separation into damped periodic relative modal response and rigid modal response is unnecessary and the total response should be treated as being in-phase (rigid) and should be combined algebraically.

f^r = frequency at which spectral acceleration, S_a, roughly returns to the ZPA.

Gupta (References 16 through 18 as modified by Reference 19) defines f^1 and f^2 by:

$$f^1 = \frac{S_{amax}}{2\pi S_{vmax}} \tag{7}$$

$$f^2 = (f^1 + 2f^r)/3 \tag{8}$$

where S_{amax} and S_{vmax} are the maximum spectral acceleration and velocity, respectively. The frequency f^1 may be thought of as a corner frequency between the velocity and acceleration response domains. For a given response spectrum, f^1 is uniquely defined. Based on the R.G. 1.60 response spectrum, f^1 is 2.0 Hz at 0.5% damping, 1.7 Hz at 5% damping, and 1.5 Hz at 10% damping. The frequency f^2 is between 22 Hz and 23 Hz for the R.G. 1.60 spectrum.

Hadjian (Reference 15) indicates that f^1 lies between 2 and 3 Hz for the 1% damped R.G. 1.60 spectrum and arbitrarily assigns an f^1 value of 2.5 Hz. Hadjian does not need to explicitly define an f^2. However, this approach implicitly defines f^2 by:

$$f^2 \approx f^r \tag{9}$$

Thus, for R. G. 1.60, f^2 equals 33 Hz.

Even more important, Hadjian demonstrates that the separation into a relative response component (combined SRSS) and a rigid response component (combined algebraically) is only important for structures which contain multiple (more than one) significant modes with frequencies greater than 10 Hz for the R.G. 1.60 spectrum. In other words, with the R.G. 1.60 spectrum, for frequencies below 10 Hz the SRSS modal response combination method is perfectly adequate and modifications for higher frequency modes are unnecessary. Above 33 Hz, SRSS is not acceptable and algebraic summation should be used. Between 10 Hz and 33 Hz, a transition zone exists in which a portion of the modal responses should be combined SRSS and a portion should be combined algebraically for the R.G. 1.60 spectrum. For other spectra, these transition frequencies would differ somewhat.

Lindley and Yow (Reference 14) do not explicitly define f^1 or f^2. However, their approach is nearly identical to the Hadjian approach so that the transition zone defined for the Hadjian approach would also be applicable to their approach.

Singh (Reference 20) also does not explicitly define f^1 or f^2. However, a review of his approach would indicate that f^1 lies at about 6 Hz and f^2 at about 28 Hz. Significant rigid response effects do not occur at frequencies below about 10 Hz.

The Gupta, Lindley and Yow, and Hadjian approaches can all be cast into a common format for ease of comparison. Therefore, each of these approaches for this transition zone will be discussed further.

2.3.2.1 GUPTA APPROACH

1. Separate the total individual modal peak responses, R_i, into a rigid peak response, R_i^r, and a damped periodic relative R_i^p, by:

$$R_i^r = \alpha_i R_i \tag{10}$$

$$R_i^p = \sqrt{1-\alpha_i^2} \quad R_i \tag{11}$$

$$\text{where } \alpha_i = \frac{\log f_i/f^1}{\log f^2/f^1} \quad , \quad \text{except } 0 \le \alpha_i \le 1 \tag{12}$$

   Thus, at $f_i \le f^1$, $\alpha_i = 0$, and at $f_i \ge f^2$, $\alpha_i = 1.0$.

2. The damped periodic relative modal responses, R_i^p, are computed for modes with frequencies below f^2, and are combined SRSS to obtain the damped periodic relative response, R^p. The rigid modal responses, R_i^r, are computed for modes with frequencies above f^1, and are combined algebraically to obtain the rigid response, R^r. Note that modes with frequencies above f^r do not have to be computed. Rather, my 1979 recommendations (repeated in Appendix A) can be used to accurately incorporate the effects of all such modes.

3. The total response, R is obtained by the SRSS combination of R^p and R^r.

2.3.2.2 HADJIAN APPROACH

1. For modes with frequencies below f^1, the total modal responses are computed using the conventional pseudo spectral acceleration, S_{a_i}. These modal responses are combined by the SRSS method to obtain the total response, R_L, for all modes with frequencies less than f^1.

2. For frequencies above f^1, an "effective relative" spectral acceleration, S'_{ar_i}, is obtained by:

$$S'_{ar_i} = S_{a_i} - (ZPA) \tag{13}$$

which assumes that the relative response is in-phase (additive) with the rigid response. Next, an "effective relative" response is computed for each mode using S'_{ar_i} in lieu of S_{a_i}.

Note that S'_{ar_i} becomes zero at frequency f^r. Thus, only modes up to frequency f^r need be considered. All modal responses computed in this step are combined by the SRSS method to obtain the damped periodic relative response R^p which is based on the assumption that phasing of these relative response modes is uncorrelated.

3. The rigid response, R^r, is computed by my 1979 recommendations (repeated in Appendix A) except only modes with frequencies below f^1 are used to compute F_i (see Equation A1 of Appendix A). The combined rigid response, R^r, for all modes with frequencies above f^1 is obtained from a static analysis using the pseudo-static inertial forces given by Equation A3 of Appendix A

4. The total response, R_H, for all modes with frequencies higher than f^1 is obtained by the absolute sum combination of R^p and R^r. One must use an absolute sum combination of R^p and R^r to be consistent with the in-phase (additive) assumption upon which Equation (13) is based.

5. The higher frequency total response, R_H, and the lower frequency total response, R_L, are combined SRSS under the assumption that responses in these two frequency ranges are uncorrelated.

2.3.2.3 Lindley and Yow Approach

The Lindley and Yow Approach is identical to the Hadjian approach with the following exceptions:

1. The "effective relative" spectral acceleration, S'_{ar_i}, is obtained by:

$$S'_{ar_i} = (S^2_{a_i} - ZPA^2)^{1/2} \tag{14}$$

which assumes that the relative response is randomly phased with the rigid response.

2. The total response is obtained by the SRSS combination of R^p and R^r. This combination is consistent with the use of Equation 14 in lieu of Equation 13 to find the relative response, R^p.

2.3.2.4 Comparison of Lindley and Yow, Hadjian, and Gupta Approaches

The Lindley and Yow, Hadjian, and Gupta approaches can be directly compared by casting the Lindley and Yow, and the Hadjian approaches into the same format as the Gupta approach. There are basically two differences. First, the Lindley and Yow and the Hadjian approach are consistent with α_i being defined as:

$$\alpha_i = 0 \text{ for } f_i < f^1$$

$$(15)$$

$$\alpha_i = \frac{(ZPA)}{S_{a_i}} \quad \text{for } f_i > f^1$$

whereas Equation (12) is used to define α_i for the Gupta approach. This is the only difference from the Gupta approach for the Lindley and Yow approach. However, the Hadjian approach assumes in-phase (additive) phasing between the rigid response and the "effective relative" response whereas Gupta assumes uncorrelated phasing. Therefore, in the Hadjian approach:

$$R_i^p = (1-\alpha_i)R_i \qquad (16)$$

whereas Equation (11) based on SRSS combination is used by Lindley and Yow, and Gupta to obtain R_i^p. Because of the use of Equation (12) to obtain R_i^p in the Hadjian approach, one must combine the total relative response, R^p, and total rigid response, R^r, by absolute summation. In the Lindley and Yow, and the Gupta approaches, these two response components are combined SRSS to be consistent with Equation (11). These are the only differences.

The Hadjian approach contains a fundamental inconsistency in its logic. First, it assumes that all "effective relative" modal responses, R_i^p, are in-phase (additive) with the corresponding rigid modal responses, R_i^r. This assumption is the basis for Equation (16). Next, it assumes that all rigid modal responses, R_i^r, are in-phase with each other which is the basis for algebraic summation of the rigid modal responses, R_i^r, to obtain the total rigid response R^r. However, it also assumes all "effective relative" modal responses, R_i^p, are uncorrelated with each other so that they may be combined SRSS to obtain the total "effective relative" response, R^p. It is inconsistent to assume the relative modal responses are uncorrelated with each other (SRSS combination) and yet are in-phase with the rigid modal responses (Equation 16) which are all in-phase with each other (algebraic summation). This fundamental inconsistency does not exist with either the Lindley and Yow approach or the Gupta approach. For this reason, I prefer either the Lindley and Yow or the Gupta approach to the Hadjian approach.

2.3.2.5 Concluding Remarks on Recent Research

Recent research has indicated that the SRSS method of modal response combination when modified for closely-spaced modes is adequate so long as the dynamic model does not contain more than one significant mode at a frequency higher than that associated with the highly amplified spectral acceleration response domain (approximately 10 Hz for the R.G. 1.60 spectrum). In these cases, no special provisions are necessary for the modal response combination of higher frequency modes. However, if the dynamic model does contain more than one significant mode at a frequency higher than that associated with the highly amplified spectral acceleration response domain then provisions for Algebraic Summation of at least a portion of the higher frequency responses are necessary.

For the R.G. 1.60 spectra, it appears that any approach which uses Equations (10) and (11) and defines α_i so as to be less than about 0.6 at frequencies below about 10 Hz, and greater than about 0.8 at frequencies above about 25 Hz should lead to reasonable results. In other words, below 10 Hz responses should be predominantly SRSS combined and above 25 Hz reponses should be predominantly algebraic sum combined. Between 10 and 25 Hz, a transition zone should exist. These frequency ranges are for the R.G. 1.60 spectrum. For other spectra, these frequency ranges would shift somewhat.

2.3.3 Impact of Improperly Combining Higher Frequency Modes by SRSS

The SRSS response combination method even when modified for closely-spaced modes can lead to significantly unconservative computed responses near the base of stiff cantilever structures and near supports for stiff components such as a stiff piping system. This unconservatism only occurs near supports. Away from supports, the SRSS response combination method can lead to significant conservatism. For the R.G. 1.60 spectrum, the SRSS response combination method will tend to underestimate responses near supports for structures which contain more than one significant mode at frequencies exceeding 10 Hz. If only one significant mode exceeds 10 Hz, no problem exists. The problem of underestimation becomes most severe when the dynamic model contains more than one significant mode at frequencies exceeding 25 Hz for the R.G. 1.60 spectrum. The degree of unconservatism depends upon the importance of these high frequency modes on total response. Generally, the level of unconservatism is negligible and of academic interest only. However, for very stiff structures such as are sometimes encountered in nuclear plant designs, the level of unconservatism can be severe.

Based upon my own experience and a review of References 14 through 20, I would judge that under fairly extreme but realistic situations the ratio of SRSS computed to actual responses might be as low as:

Response Quantity	Ratio: SRSS Computed to Actual Response
Acceleration	0.60
Inertial Forces	0.60
Shears	0.75
Moments	0.90

These levels of unconservatism would only occur near the supports of structure models which contain more than one significant mode at frequencies above 25 Hz. Note that the unconservatism is most severe for accelerations and inertial forces. The underprediction of shears and moments is much less, because in these cases the SRSS method leads to overprediction of responses away from the supports and this reduces the unconservatism of shears and moments at supports.

Actually, an experienced or cautious analyst would catch these levels of unconservatism in their results. The only places I have seen this level of unconservatism in results occurs when the SRSS computed accelerations near supports are less than the ZPA of the support. Any analyst who makes this check would realize an analytical problem existed and would correct for it by adding in static inertial accelerations or would perform a time-history analysis. Thus, I would doubt if such large unconservatisms would exist in any analysis or design performed by an experienced or cautious analyst using the SRSS method. However, such unconservatism might exist in "cookbook" analyses performed by an analyst who was overly trusting in the accuracy of their computer program.

The impact of incorporating any of the proposed methods would be to eliminate this possible but generally unlikely source of severe unconservatism in design. The change would make clear the cause of this unconservatism and would eliminate the need for the use of approximate methods which have been used to correct this deficiency in the SRSS combined response. Once computer programs were modified, the added analytical costs and engineering efforts to incorporate any of these methods would be negligible.

2.4 COMBINATION OF SPATIAL COMPONENTS AND MULTIPLE RESPONSES

Regulatory Guide 1.92 states that when the response spectra method is used, spatial components should be combined SRSS. This requirement is based upon the reasonable assumption that the responses (frequency-by-frequency) of the three components of the ground motion are uncorrelated. For piping, it is further assumed that the three components of support motion are also uncorrelated (see Section 2.1.3 for discussion on this point). So long as one assumes a lack of phase correlation between the three spatial components of support motion, the SRSS combination of spatial components is fully justified.

The SRSS combination of spatial components works well when applied to a single final response quantity of interest such as a stress, displacement, or force. However, often one is interested in some combination of multiple response quantities. For instance, for pipe the Tresca or maximum shear stress given by:

$$\tau_{max} = (M_x^2 + M_y^2 + M_z^2)^{\frac{1}{2}} /2Z \qquad (17)$$

is generally the stress quantity of interest. In Equation 17, M_x, M_y and M_z are the moments in the local x, y, and z piping cross-section axes while Z is the section modulus. In applying Equation 17 for seismic response, one should use values of M_x, M_y, and M_z which occur concurrently. However, the SRSS combination of responses due to the three independent spatial components of support motions leads to maximum probable resultant responses, M_{XR}, M_{YR}, and M_{ZR}, in each of the three response directions (see Section 2.1.3). These maximum probable resultant responses are not likely to occur concurrently. Yet, the standard procedure is to substitute all three of these maximum probable resultant responses, M_{XR}, M_{YR}, and M_{ZR}, for the concurrent responses, M_x, M_y, and M_z in Equation 17. This substitution conservatively assumes that M_{XR}, M_{YR}, and M_{ZR} all occur at the same time. Within my experience, such an assumption leads to a 0% to 40% margin of conservatism in the combined response τ_{max} over that appropriate for the assumption of uncorrelated support motions. Unfortunately, this substitution of M_{XR}, M_{YR}, and M_{ZR} for M_x, M_y, and M_z in Equation 17 is the only practical approach with the SRSS method for the combination of spatial components so that this conservatism for multiple responses is unavoidable with this method.

However, a more sophisticated response combination method which avoids most of this unnecessary conservatism does exist (References 22 through 24). I will call this method the Gupta method. The Gupta method described in Reference 22 provides a rigorous solution for the maximum probable combination of multiple responses under the assumption of uncorrelated three-component input motions. As such, this method represents the "exact" method whereas the above-described SRSS method is a conservative approximation. Application of this "exact" Gupta method for piping systems is illustrated in Reference 23. Unfortunately, the "exact" Gupta method is very difficult to apply and so has not come into wide use. However, it does represent the "standard" against which other approximate methods should be measured. As such, Reg. Guide 1.92 should allow this method.

Reference 24 recommends an "approximate" Gupta method which is only slightly conservative (0% to 13% conservative in the case of piping stress analyses governed by Equation 17) as compared to the "exact" Gupta method and much easier to apply. Even this method is more difficult to apply than the SRSS method. A further simplification will be described herein which only slightly increases the uncertainty (-1% to +17% conservative). Each of these "approximate" Gupta methods are more accurate than the SRSS method and should be allowed by Reg. Guide 1.92.

Rosenblueth (Reference 25) has also proposed a method similar to the "approximate" Gupta method. However, I have not studied the Rosenblueth method in detail.

Also, Newmark (References 26 and 27) has proposed an approximate method for combining multiple responses from these spatial components of input motion. This method is called the 100-40-40 method. Within my experience, the Newmark 100-40-40 method introduces about the same level of conservatism as the SRSS method and is less accurate than either of the approximate Gupta methods. The Newmark 100-40-40 method should also be allowed by Reg. Guide 1.92.

All of these methods are founded on the same assumption of uncorrelated spatial components of input motion. Each of these methods is at least as valid as the SRSS method. Because the starting assumptions are the same, all of these methods could be called SRSS-equivalent methods. All of these methods should be allowed for spatial component combination. These methods are described and compared in Appendix B.

3.0 RECOMMENDATIONS

3.1 REVISIONS TO REG. GUIDE 1.92 AND STANDARD REVIEW PLAN

1. The algorithm given in Table 3 for combining responses using the independent support motion response spectrum analysis method (ISMA) should be added to Standard Review Plan 3.9.2. This algorithm represents a modification of the NRC staff-recommended algorithm contained in Table 1. The bases for this revised algorithm are given in Section 2.1.

2. The absolute signs should be removed from the Double Sum Combination (DSC) Method in Reg. Guide 1.92. Also, the Complete Quadratic Combination (CQC) Method should be added to Reg. Guide 1.92 without inserting an arbitrary set of absolute signs. A detailed discussion of the issues concerning closely-spaced modes is presented in Section 2.2.

3. Regulatory Guide 1.92 and/or the appropriate Standard Review Plan Sections should require the algebraic summation of all modes with frequencies exceeding f^r where f^r is defined as the frequency at which the spectral acceleration, S_a, roughly returns to the peak zero period acceleration, ZPA. The two methods of algebraic summation given in Appendix A should be allowed. Secondly, the SRP should allow the SRSS method of modal response combination as corrected for closely-spaced modes to be used if the dynamic model does not contain more than one significant mode at a frequency higher than that associated with the highly amplified spectral response domain (approximately 10 Hz for the Reg. Guide 1.60 spectrum). In other words, no special consideration of how to combine high-frequency modes is necessary in this case. Third, for dynamic models which contain more than one significant mode at frequencies above about 10 Hz, the SRP should require a gradual transition from the SRSS response combination which is appropriate for lower frequency modes and the algebraic summation appropriate at frequencies above f^r. Both the Gupta method and the Lindley and Yow method should be explicitly permitted. Any other rational method of treating this transition should also be allowed. Fine tuning of this transition is unwarranted. However, some consideration is necessary. A further discussion of higher frequency modal combination is contained in Section 2.3.

4. Regulatory Guide 1.92 should permit the use of any of the SRSS equivalent methods for the combination of effects from the three spatial components of input. The "exact" Gupta method, "approximate" Gupta method, and the Newmark 100-40-40 method are at least as valid as the SRSS method and are founded on the same theory. These methods are discussed in Section 2.4.

3.2 IMPACT OF RECOMMENDATIONS

All of these recommendations will lead to more accurately and rationally computed piping responses by the response spectra method. For most piping systems, these recommendations will result in a reduction in computed response. In some cases, this reduction will be substantial.

However, for very stiff piping systems, the high-frequency mode combination recommendation will result in an increase in support forces and responses near supports. Thus, these recommendations will properly penalize very stiff piping system designs and will benefit more flexible designs.

3.3 RECOMMENDATIONS FOR FURTHER RESEARCH

In my opinion, only a limited amount of further research in response combination methods is necessary in order to safely and rationally design piping systems and structures. If further research is performed it should concentrate on the following topics:

1. Research to develop practical ways to retain the relative phasing relationships caused by the primary system (civil structure) in the ISMA method for multiply-supported subsystems. This research would enable the actual relative phasing to be used in lieu of the conservative absolute summation of support group responses recommended in Table 3. This research should be directed toward both primary and secondary reponses with the primary benefit probably being with the secondary responses (see Sections 2.1.1 and 2.1.5).

2. Research on the correlation or lack of correlation of the three-directional components of input support motions for piping systems (see Section 2.1.3 and 2.1.6).

3. Research on the higher frequency transition zone from SRSS modal combination to algebraic sum modal combination (see Section 2.3).

I would rank these research topics in the order listed with 1 being highest and 3 being lowest.

3.4 ALTERNATE SIMPLIFIED RECOMMENDATIONS FOR MODAL COMBINATION

Accounting for closely-spaced modes and high-frequency modes as per the recommendations of Section 3.1 improves the accuracy of computed piping responses. However, the penalty for this improved accuracy is more complex modal combination techniques. A school of thought exists that says we don't need this improved accuracy to safety design piping systems, but we do need more simplified analysis techniques. I am in sympathy with this school of thought. In my judgment, adequate accuracy for safe design can be achieved by the following simpler modal combination rules:

1. Combine all modes with frequencies below f^r by SRSS where f^r is defined as the frequency at which the spectral acceleration, S_a, roughly returns to the zero period acceleration. No consideration of closely-spaced modes or a gradual transition to algebraic summation at higher frequencies need be included.

2. Combine all modes with frequencies greater than f^r by algebraic summation using either method given in Appendix A.

3. Combine the low (Rule 1) and high (Rule 2) frequency modal responses by SRSS.

In my judgment, there is sufficient conservatism in other aspects of dynamic analysis and design of piping systems to adequately cover any unconservatism introduced by the use of these simplified modal combination rules. I leave it to the NRC staff to decide whether improved accuracy or greater simplicity is the preferred goal.

TABLE 1

NRC STAFF-RECOMMENDED ALGORITHM FOR COMBINING RESPONSES USING THE INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM ANALYSIS METHOD (INTERIM)

(Reference 28)

A. Dynamic Components (primary)

1. For each mode and for each direction:
   Combine group responses by absolute sum (ABS).

2. For each mode:
   Combine direction responses by SRSS.

3. For each nodal point and degree of freedom:
   Combine modal responses by R.G. 1.92

This can be summarized as:

Displacements: GROUP (ABS) - DIRECTION (SRSS) - MODES (R.G. 1.92)

B. Pseudo-Static Components (secondary)

1. For each group, calculate maximum absolute response for each direction.

2. Combine for all groups and directions by absolute sum.

C. Total Dynamic Responses

Add dynamic and pseudo-static components by SRSS.

Note: For the design of piping, only the dynamic components are considered as primary. For piping or equipment support, both dynamic and pseudo-static components should be considered as primary.

TABLE 2

COUPLING FACTORS FOR MODAL COUPLING METHODS

Method	Equation	C_{jk}
1. ABS	(2)	All 1.0
2. Algebraic Sum	(1)	All 1.0
3. SRSS	(1) or (2)	All 0.0
4. Grouping Method	(2)	Modes arranged in ascending frequency order. Groups formed beginning with the lowest frequency such that all higher modes with frequencies within 10% of lowest mode in group are lumped into same group. No mode in more than one group. Within Same Group: $C_{jk} = 1.0$ Outside Same Group: $C_{jk} = 0.0$
5. Ten Percent Method	(2)	Modes arranged in ascending frequency order. If modal frequencies within 10% of each other, then $C_{jk} = 1.0$. Otherwise, $C_{jk} = 0.0$
6. DSC	(1)	C_{jk} from Equation (3)
7. NRC-DSC	(2)	C_{jk} from Equation (3)
8. CQC	(1)	C_{jk} from Equation (6)
9. ARC	(1)	C_{jk} from Reference (10)

TABLE 3

SUGGESTED REVISION TO RECOMMENDED ALGORITHM FOR COMBINING RESPONSES USING THE INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM ANALYSIS METHOD

A. Inertial or Dynamic Components (primary)

1. For each mode and for each input motion direction: Combine group responses by absolute sum (ABS) or preferably, by actual relative phasing if structural phasing information is retained. If it can be shown that group responses are reasonably phase uncorrelated (such as responses between different structures), then an SRSS combination may be used.

2. For each response quantity and each input motion direction: Combine modal responses by the Double Sum (DSC) or CQC method with provisions for high-frequency modes.

3. For each response quantity: Combine input motion direction responses by SRSS or equivalent method.

This can be summarized as:

GROUP (ABS or Actual) - MODES (DSC or CQC) - DIRECTION (SRSS equivalent)

B. Support Displacement or Pseudo-Static Components (secondary):

1. Group by common attachment point. For each group, calculate maximum absolute response for each input direction.

2. Combine for all groups by absolute sum or preferably, by actual relative phasing if structural phasing information is retained. If reasonable phase uncorrelation can be demonstrated, SRSS combination may be used.

3. Combine for input directions by SRSS or equivalent method.

C. Total Dynamic Responses

Add dynamic and pseudo-static components by SRSS

Note: For the design of piping, only the dynamic components are considered as primary. For equipment support, both dynamic and pseudo-static components are considered as primary.

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APPENDIX A

INCLUSION OF PSEUDO-STATIC RESPONSE FOR ALL MODES ABOVE THE RIGID FREQUENCY, f^r

1. Determine the modal responses only for those modes with natural frequencies less than that at which the spectral acceleration approximately returns to the ZPA (33 Hz in the case of the Regulatory Guide 1.60 response spectra). Combine such modes in accordance with rules for the SRSS combination of modes as modified for closely-spaced and higher frequency modes.

2. For each degree-of-freedom included in the dynamic analysis, determine the fraction of degree-of-freedom (DOF) mass included in the summation of all of the modes included in Step 1. This fraction F_i for each degree-of-freedom i is given by:

$$F_i = \sum_{m=1}^{M} PF_m * \phi_{m,\, i} \qquad \text{(A1)}$$

where

m is each mode number

M is the number of modes included in Step 1.

PF_m is the participation factor for mode m

$\phi_{m,i}$ is the eigenvector value for mode m and DOF i

Next, determine the fraction of DOF mass not included in the summation of these modes:

$$K_i = F_i - \bar{\delta} \qquad \text{(A2)}$$

where

$\bar{\delta}$ is the Kronecker delta which is one if DOF i is in the direction of the earthquake input motion and zero if DOF i is a rotation or not in the direction of the earthquake input motion.

If, for any DOF i this fraction $|K_i|$ exceeds 0.1, one should include the response from higher modes than those included in Step 1.

3. Higher modes can be assumed to respond in phase with the peak ZPA and thus with each other so that these modes are combined algebraically which is equivalent to pseudo-static response to the inertial forces from these higher modes excited at the ZPA. The pseudo-static inertial forces associated with the summation of all higher modes for each DOF i are given by:

$$P_i = ZPA * M_i * K_i \qquad \text{(A3)}$$

where

P_i is the force or moment to be applied at degree-of-freedom (DOF), i

M_i is the mass or mass moment of inertia associated with DOF i

The structure is then statically analyzed for this set of pseudo-static inertial forces applied at all of the degrees-of-freedom to determine the maximum responses associated with the high-frequency modes not included in Step 1.

4. The total combined response to high-frequency modes (Step 3) are combined SRSS with the total combined response from lower frequency modes (Step 1) to determine the overall structural peak response.

This procedure is easy because it requires the computation of individual modal responses only for the lower frequency modes (below 33 Hz for the Regulatory Guide 1.60 response spectrum). Thus, the more difficult higher frequency modes do not have to be determined. The procedure is accurate because it assures inclusion of all modes of the structural model and proper representation of DOF masses. It is not susceptible to inaccuracies due to an improperly low cutoff in the number of modes included.

Alternately, one can compute modal responses for a sufficient number of modes to ensure that an inclusion of additional modes does not result in more than a 10% increase in responses. Modes with natural frequencies less than at which the spectral acceleration approximately returns to the ZPA (33 Hz in the case of the Regulatory Guide 1.60 response spectrum) are combined in accordance with rules for the SRSS combination of modes as modified for closely-spaced and higher frequency modes. Higher mode responses are combined algebraically (i.e., retain sign) with each other. The total response from the combined higher modes are then combined with the total response from the combined lower modes.

APPENDIX B

DESCRIPTION AND COMPARISON OF SRSS - EQUIVALENT METHODS FOR SPATIAL AND MULTIPLE RESPONSE COMBINATIONS

In this appendix, I will describe the Gupta and Newmark methods and will compare τ_{max} results for an example typical piping response problem. The example piping response problem has the following individual component responses:

$M_{xx} = 10.0$ $\quad M_{xy} = 3.0$ $\quad M_{xz} = 2.0$

$M_{yx} = 4.0$ $\quad M_{yy} = 15.0$ $\quad M_{yz} = 3.0$

$M_{zx} = 2.0$ $\quad M_{zy} = 3.0$ $\quad M_{zz} = 6.0$

where, for example, M_{xy} represents the x-component maximum probable component response due to the y-direction input motion. For simplicity, it will be assumed that no closely-spaced modes exist. The presence of closely-spaced modes slightly modifies the combined response, ($2Z\ \tau_{max}$), obtained by the Gupta methods and has no influence on the other methods. The comparisons presented are equally valid with or without closely-spaced modes.

For this example problem, when spatial component responses are combined SRSS, the maximum probable resultant combined component responses are:

$M_{XR} = 10.6$

$M_{YR} = 15.8$

$M_{ZR} = 7.0$

If these responses are assumed to occur concurrently, then $(2Z\ \tau_{max})$ from Equation 17* is:

SRSS Approach

$$(2Z\ \tau_{max}) = 20.3$$

It will be shown that the "exact" maximum probable combined response consistent with the assumption of uncorrelated input motions as obtained by the "exact" Gupta method is:

"Exact"

$$(2Z\ \tau_{max}) = 17.8$$

Thus, for this example problem the SRSS method introduces 14% conservatism. For some other problems, the conservatism can be much greater. However, this example is representative of the majority of cases in which the conservatism is not excessive.

B.1 Gupta "Exact" Method (References 22 and 23)**

The Gupta "Exact" Method requires the development of combined modal responses:

$$M_{xe} = \left(\sum_{i=1}^{3} \sum_{m} \sum_{n} C_{mn} M_{xim} M_{xin} \right)^{\frac{1}{2}}$$

$$M_{ye} = \left(\sum_{i=1}^{3} \sum_{m} \sum_{n} C_{mn} M_{yim} M_{yin} \right)^{\frac{1}{2}} \qquad \text{(B-1)}$$

$$M_{ze} = \left(\sum_{i=1}^{3} \sum_{m} \sum_{n} C_{mn} M_{zim} M_{zin} \right)^{\frac{1}{2}}$$

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* Equation numbers in this Appendix which do not have a B- prefix refer to equations from the main body of this report.

** References for this Appendix are listed in Section 4 of the main body of this report.

and cross coupling terms:

$$r = \sum_{i=1}^{3} \sum_{m} \sum_{n} C_{mn} M_{xim} M_{yin}$$

$$s = \sum_{i=1}^{3} \sum_{m} \sum_{n} C_{mn} M_{xim} M_{zin} \qquad \text{(B-2)}$$

$$t = \sum_{i=1}^{3} \sum_{m} \sum_{n} C_{mn} M_{yim} M_{zin}$$

where C_{mn} is the mode coupling term ($C_{mn} = 1$ when $m = n$; otherwise C_{mn} is from Equation 3 for DSC Method or Equation 6 for CQC Method), and M_{xim} is the x-direction moment in the m-mode due to the i-direction input component.

In the absence of closely-spaced modes, Equation B-1 becomes the SRSS combination of spatial component responses. Thus:

No Close-Spaced Modes

$$M_{xe} = M_{XR}$$
$$M_{ye} = M_{YR} \qquad \text{(B-3)}$$
$$M_{ze} = M_{ZR}$$

and Equation B-2 becomes:

$$r = \sum_{i=1}^{3} M_{xi} M_{yi}$$

$$s = \sum_{i=1}^{3} M_{xi} M_{zi} \qquad \text{(B-4)}$$

$$t = \sum_{i=1}^{3} M_{yi} M_{zi}$$

Thus, for our example problem which does not have closely-spaced modes:

$M_{xe} = 10.6$ $\qquad$ $r = 91.0$

$M_{ye} = 15.8$ $\qquad$ $s = 41.0$

$M_{ze} = 7.0$ $\qquad$ $t = 71.0$

The Gupta "exact" method then requires the development of a set of equivalent modal responses, $\bar{M}_{x\alpha}$, $\bar{M}_{y\alpha}$, and $\bar{M}_{z\alpha}$, which also satisfy Equations B-1 and B-2. The number α must equal the number of response quantities being combined ($\alpha = 3$ in the case of Equation 17). For the case of Equation 17, these equivalent modal responses can be obtained from the following table:

Equivalent Modal Moments

Equivalent Mode, α	$\bar{M}_{x\alpha}$	$\bar{M}_{y\alpha}$	$\bar{M}_{z\alpha}$
1	M_{xe}	r/M_{xe}	s/M_{xe}
2	0	$(M_{ye}^2 - \bar{M}_{y1}^2)^{1/2}$	$(t-\bar{M}_{y1}\bar{M}_{z1})/\bar{M}_{y2}$
3	0	0	$(M_{ze}^2-\bar{M}_{z1}^2-\bar{M}_{z2}^2)^{1/2}$

Maximum probable concurrent responses are then given by:

$$M_x = \sum_\alpha k_\alpha \bar{M}_{x\alpha}$$

$$M_y = \sum_\alpha k_\alpha \bar{M}_{y\alpha} \qquad (B\text{-}5)$$

$$M_z = \sum_\alpha k_\alpha \bar{M}_{z\alpha}$$

where

$$\sum_\alpha k_\alpha^2 = 1 \qquad (B\text{-}6)$$

All possible combinations of K_α which satisfy Equation B-6 must be considered. These maximum probable concurrent responses are then used in Equation 17 to evaluate ($2Z\ \tau_{max}$).

The obvious problem with the Gupta "exact" method is that an infinite number of K_α values satisfy Equation B-6. One must find the set which leads to the maximum value of ($2Z\ \tau_{max}$) in order to find the "exact" maximum probable ($2Z\ \tau_{max}$). If one stops his search too early and does not find the "worst" combination leading to the maximum value, then one will unconservatively underestimate the maximum probable value of ($2Z\ \tau_{max}$).

For our example problem, a set of equivalent modal moments are:

Example Problem Equivalent Modal Moments

Equivalent Mode, α	$\bar{M}_{x\alpha}$	$\bar{M}_{y\alpha}$	$\bar{M}_{z\alpha}$
1	10.6	8.6	3.9
2	0	13.2	2.8
3	0	0	5.1

and some of the possible solutions of Equations B-5 and B-6 are:

Trial Solutions for $(2Z\ \tau_{max})$

Trial No.	K_1	K_2	K_3	M_x	M_y	M_z	$2Z\ \tau_{max}$
1	.55	.8	.24	5.8	15.3	5.6	17.3
2	.45	.86	.25	4.8	15.2	5.4	16.8
3	.50	.86	.10	5.3	15.6	4.9	17.2
4	.60	.70	.39	6.4	14.4	6.3	17.0
5	.67	.70	.25	7.1	15.0	5.8	17.6
6	.76	.60	.25	8.1	14.5	5.9	17.6
7	.72	.65	.24	7.6	14.8	5.8	1[illegible].6
8	.75	.65	.12	8.0	15.0	5.4	17.8
9	.70	.70	.14	7.4	15.3	5.4	17.8
10	.79	.60	.13	8.4	14.7	5.4	17.8
11	.74	.65	.17	7.8	14.9	5.6	17.8
12	.76	.65	0	8.1	15.1	4.8	17.8

After a wide search of possible <u>concurrent</u> solutions, one finds that the maximum probable value is:

<u>"Exact" Maximum Probable</u>

$$(2Z\ \tau_{max}) = 17.8$$

which is somewhat less than the simple SRSS combination of spatial components but more than the largest single component of response.

B.2 Gupta "Approximate" Method (Reference 24)

Because of the effort involved in evaluating all possible combinations which satisfy Equations B-5 and B-6, Gupta developed a conservative approximate solution. In this solution for the combination of three response components, 100% of one equivalent modal response is taken concurrent with 41.4% of a second equivalent modal response and with 31.8% of the third equivalent modal response. Gupta shows that this combination is always conservative compared with the exact solution and also provides the minimum conservativeness consistent with always being conservative. The level of conservatism ranges from 0 to 13%.

By this approach, there are six (6) possible combinations* for ($2Z\ \tau_{max}$). These are:

Approximate Solutions For ($2Z\ \tau_{max}$)

Combination No.	K_1	K_2	K_3	M_x	M_y	M_z	($2Z\ \tau_{max}$)
1	1.0	0.414	0.318	10.6	14.1	6.7	18.9
2	1.0	0.318	0.414	10.6	12.8	6.9	18.0
3	0.414	1.0	0.318	4.4	16.8	6.0	18.4
4	0.414	0.318	1.0	4.4	7.8	7.6	11.8
5	0.318	1.0	0.414	3.4	15.9	6.2	17.4
6	0.318	0.414	1.0	3.4	8.2	7.5	11.6

* If one must be concerned with + and - signs, then there are 8 times 6 or 48 combinations. However, for Equation 17 the response signs are unimportant.

Thus, the approximate maximum probable response is:

"Approximate" Maximum Probable

$$(2Z\,\tau_{max}) = 18.9$$

which is only 6% more conservative than the "exac." solution for this example problem.

B.3 Alternate Gupta "Approximate" Method

The approximate Gupta method can be further simplified by taking 100% of one equivalent modal response concurrent with 40% of all other equivalent modal responses. For the combination of three response components, this simplification reduces the problem to only 3 possible combinations with the possible level of conservatism ranging from -1% to +17%.

By this approach:

"Approximate" Maximum Probable

$$(2Z\,\tau_{max}) = 18.8$$

B.4 Newmark 100-40-40 Method (References 26 and 27)

The Newmark 100-40-40 Method requires that 100% of the responses due to one spatial component be assumed to act concurrently with 40% of the responses from each of the other two input spatial components.

When determining $(2Z\,\tau_{max})$, there are three possible combinations of the 100-40-40 rule. These are:

Combination 1 (100% x-input direction)

$$M_x = M_{xx} + 0.4(M_{xy} + M_{xz}) = 12.0$$

$$M_y = M_{yx} + 0.4(M_{yy} + M_{yz}) = 11.2$$

$$M_z = M_{zx} + 0.4(M_{zy} + M_{zz}) = 5.6$$

$$(2Z\ \tau_{max}) = 17.3$$

Combination 2 (100% y-input direction)

$$M_x = M_{xy} + 0.4(M_{xx} + M_{xz}) = 7.8$$

$$M_y = M_{yy} + 0.4(M_{yx} + M_{yz}) = 17.8$$

$$M_z = M_{zy} + 0.4(M_{zx} + M_{zz}) = 6.2$$

$$(2Z\ \tau_{max}) = 20.4$$

Combination 3 (100% z-input direction)

$$M_x = M_{xz} + 0.4(M_{xx} + M_{xy}) = 7.2$$

$$M_y = M_{yz} + 0.4(M_{xx} + M_{yy}) = 10.6$$

$$M_z = M_{zz} + 0.4(M_{zx} + M_{zy}) = 8.0$$

$$(2Z\ \tau_{max}) = 15.1$$

Combination 2 controls and thus:

Newmark 100-40-40

$$(2Z\ \tau_{max}) = 20.4$$

which in this case is identical to the SRSS spatial combination.

POSITION PAPER

ON

STRESS LIMITS/DYNAMIC STRESS ALLOWABLES FOR PIPING

BY

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APRIL, 1984

Position Paper

on

Stress Limits/Dynamic Stress Allowables for Piping

1.0 STATEMENT OF ISSUES

The NRC, through SRP 3.9.3, accepts the stress limits for piping pressure boundaries given in the ASME Boiler and Pressure Vessel Code, Section III, Div. 1[1]; hereinafter called the Code. The Code establishes stress limits for two types of loads:

Type-1: Loads which could cause gross plastic deformation. These loads include internal pressure, weight and inertia effects of earthquakes and other dynamic loadings. They are controlled by Code Eq. (9) which is based on limit load tests and theory.

Type 2: Loads which are deformation limited. These loads include thermal expansion, thermal gradients and relative anchor movement from any cause, including earthquakes or other dynamic loadings. They, in combination with Type-1 loads that may be repeated in service, are controlled by a fatigue evaluation method detailed in the Code.

Code stress limits for piping are different from those used by structural designers in that the Code stress limits, for Levels A, B, C and D, permit loads that cause plasticity in the piping. This piping concept dates back to the early 1950's and is embodied in ASA B31.1-1955[2] in the form of the stress range concept. However, Code stress limits do not explicitly consider the finite-time-duration (or energy content) of dynamic loads or strain rate effects.

The issues of this paper are:

(1) Are Code stress limits appropriate for control of dynamic loads when inelastic analysis methods are used?

(2) Are strain rate effects sufficient to warrant inclusion in a dynamic analysis?

2.0 DISCUSSION OF ISSUES

2.1 Code Stress Limits and Inelastic Analysis

The question of adequacy of stress limits cannot be separated from the question of how accurately the loads are calculated. This aspect can be discussed in terms of Code Eq. (9):

$$B_1 P D_o/2t + B_2 M_1/Z \leq S_L \tag{C9}$$

where* M_1 is a Type-1 moment resultant and S_L is the Code stress limit:

Table 1: Code Eq. (9) Stress Limits

Condition	Stress Limit, S_L	
	Class 1 Piping	Class 2/3 Piping**
Design	$1.5S_m$	$1.5S_h$
Level A	———	Lesser, $1.8S_h$, $1.5S_y$
Level B	Lesser, $1.8S_m$ or $1.5S_y$	Lesser, $1.8S_h$. $1.5S_y$
Level C	Lesser, $2.25S_m$ or $1.8S_y$	Lesser, $2.25S_h$, $1.8S_y$
Level D	Lesser, $3.0S_m$ or $2.0S_y$	Lesser, $3.0S_h$, $2.0S_y$

S_m = allowable stress intensity, S_h = allowable stress, S_y = material yield strength. The Code gives tables of S_m, S_h and S_y; they are functions of the material and temperature. Accordingly, the right-hand-side of Eq. (C9) is quantitatively defined. The left-hand-side of Eq. (C9) is defined to the extent that the Code gives the stress indices, B_1 and B_2, for commonly used

* The reader should see the Code for definition of other terms.

** At present (Jan. 1984), the Code does not contain the approved changes to make Class 3 Eq. (9) the same as Class 2.

piping components and defines D_o, t and Z. However, the Code does not tell how to calculate P and M_1 for dynamic loadings; e.g., relief valve operation.

The Code does contain a portion on "Expansion and Flexibility", NB/NC/ND-3672. A sub-portion is headed "Method of Analysis" reads:

> "All systems shall be analyzed for adequate flexibility by a rigorous structural analysis unless they can be judged technically adequate by an engineering comparison with previously analyzed systems."

It may be noted that the Code does not prohibit an inelastic analysis, even for the static loadings involved in restraint of thermal expansion.

While the Code does not address calculation of dynamic loads (P and M_1) for use in Eq. (9), and even for static loads does not prohibit inelastic analysis, Code users have almost always calculated these loads using an elastic analysis. The important point we wish to make is that no changes are needed in the Code to permit calculation of loads (P, M_1) by an inelastic analysis method which could include consideration of the energy content of the event; e.g., an earthquake.

SRP 3.7.2 and 3.7.3 infer, in many places, that linear elastic earthquake analysis methods are expected to be used; however, there is no ban on the use of inelastic methods. SRP 3.9.3, under "Design and Installation of Pressure Relief Valves" states:

> "The structural response of the piping and support system is reviewed with particular attention to the dynamic or time-history analysis employed in evaluating the appropriate support and restraint stiffness effects under dynamic loadings when valves are discharging."

Again, the implication is that linear elastic analysis methods are expected to be used but there is no ban on inelastic methods. However, because the SRP's address methods of calculation of dynamic loads (P and M_1 for use in Code Eq. (9), the SRP's should be revised to say that inelastic analysis methods are acceptable.

Having accepted inelastic analysis methods, the question arises: Are the Code limits shown in Table 1 acceptable in conjunction with inelastic analysis methods that consider the limited energy content of the event, the plastic energy absorption by the piping and strain rate effects? Note, in particular, that for Level D the stress limit is 2.0 S_y; where S_y is the material yield strength. A discussion of the basis of Code Eq. (9) is relevant to this question.

The background of the Code Eq. (9) is discussed in Reference (3). Briefly, the equation and the B-indices used therein are based on limit-moment tests and limit-moment theory. Static tests were used, with no limit on the energy input during the tests. The test limit moment was defined as that moment at which the displacement was two times the extrapolated elastic displacement. This is the same criterion used in the Code, II-1430, "Criterion of Collapse Load". The motivation for this criterion was to assure that displacements are kept close enough to elastically calculated displacements so that the results of the elastic piping system analysis would remain reasonably valid for support and equipment loads.

The bending limit moment of thin-wall pipe is $(4/\pi)ZS_y$, where Z is the section modulus of the pipe, S_y is the yield strength of the pipe material. For this simple case, all of the stress limits in Table 1 permit moments to be greater than the limit moment. The judgmental aspects that led to those seemingly high stress limits are discussed in Reference (3). They are (see p. 55 of Ref. (3)):

"(1) The presence of limit moment conditions at some location in a piping system does not mean that gross plastic deformation will necessarily occur. A collapse mechanism must be formed.

(2) With the exception of Ref. (24) tests, all test data and theory ignore time dependent effects; e.g., increases in yield strength for very short-time loading.

(3) With the exception of Ref. (23) and (24) tests, all test data and theory ignore cyclic strain hardening.

(4) The selection of an experimental limit load criteria, such as $\delta = 2\delta_e$, is essentially arbitrary. In many tests, maximum loads were substantially higher than limit loads and, in many piping systems, limit moments may be unduly conservative."

On page 59 of Ref. (3), additional aspects are cited:

"(c) increase in yield strength and/or decrease in structural response under short-time loadings

(d) the probability that actual yield strengths will be higher than Code-tabulated values."

The second part of (c) alludes to the limited energy content of some dynamic loads.

These considerations led to establishing stress limits such as $2.0S_y$ for Level D. Now, if an analysis is to be permitted that takes into account many or most of the cited aspects, is $2.0S_y$ still an appropriate and defensible Level D stress limit? The only answer we can give is: Not necessarily. We recommend that it be made clear that Code NB/NC/ND stress limits are <u>not</u> necessarily appropriate for use in conjunction with a rigorous inelastic analysis.

This rather non-committal answer can perhaps best be explained in an attempt to answer the question: What stress (or strain) limits should be placed on a rigorous plastic analysis of a piping system?

Appendix F of the Code, in particular F-1341.2, gives stress limits which, for A106 Grade B material, translate approximately into about 1% membrane strain and 4% membrane-plus-bending strain. For SA312 TP304 material, the stress limits translate into about 20% membrane strain and about 35% membrane-plus-bending strain. Code Case N-196, concerning use of a plastic analysis in liew of a shakedown analysis, states that the maximum accumulated local strain, as a result of cyclic operations to which plastic analysis is applied, must not exceed 5%. Code Case N47, T-1300, "Deformation and Strain Limits for Structural Integrity", T-1310, "Strain Limits for Inelastic Analysis" prescribes strain limits of:

1% averaged through the thickness
2% at surface due to a linearized distribution of strain through the thickness
5% local at any point

Each of the cited sources refers to deformation limits and two of the cited sources call attention to the problems of compressive stresses/buckling. None of the sources appear to distinguish between strain limits for base materials and those for weldments.

Appropriate strain limits are deemed to be a function of the particular base material; e.g., appropriate strain limits for an annealed austenitic steel might be higher than for a bolting material like SA193 Grade B7. However, perhaps more important, weldments may be less able to withstand plastic strains than the base materials. In piping, there are a large number of girth butt welds and, in addition, welds between run pipe and branch connections. The branch welds may be subjected to bi-axial or tri-axial strains; under which conditions the appropriate strain limit may be quite low. In piping, welds may be made to cast steel components, (e.g., valve bodies) and welds may be made between ferritic steel and austenitic steel; these kinds of weldments must be considered in establishing appropriate strain limits. Strain limits for compressive loads (e.g., compressive side of a pipe subjected to moment loading) must include potential buckling considerations.

In view of the complexities of appropriate strain limits discussed above, we recommend that NRC initiate a research program with the objective of developing acceptable strain limits for use with inelastic analyses.

2.2 Strain Rate Effects

2.2.1 Tensile Test Data

It has been known for many years that the strain response of ductile materials depends upon the loading rate. In 1938, Davis[4] reviewed the literature on the effect of speed of testing on the yield point of mild steel. In 1944, Manjoine [5] presented extensive data on the influence of rate of strain on the tensile properties of mild steel; Fig. 1 herein is from Manjoine's paper.

To bring the strain rates into perspective with standard methods of tensile testing of steel products, ASTM A 370, "Specification for Mechanical Testing of Steel Products", states that:

> "Any convenient speed of testing may be used up to one-half the specified yield point or yield strength. When this point is reached, the rate of separation of the crossheads under load shall be adjusted so as not to exceed 1/16 in. per minute per inch of gage length... . This speed shall be maintained through the yield point or yield strength. In determining the tensile strength, the rate of separation of the heads under load shall not exceed 1/2 in. per minute per inch of gage length. In any event the minimum speed of testing shall not be less than 1/10 of the specified maximum rates for determining the yield point or yield strength and tensile strength."

The maximum and minimum strain rates with units of in/in/second are:

Property	Maximum	Minimum
Yield	1.04×10^{-3}	1.04×10^{-4}
Tensile (Ultimate)	8.33×10^{-3}	8.33×10^{-4}

Figures 2, 3 and 4 show correlations of yield strength with strain rate. Figure 2 is from a paper by Bodner [6] in which he used the data from Manjoine[5] and, in the range of $\dot{e}$ from 10^{-3} to 10^{2}, represented the data for analytical purposes by the equation:

$$\dot{e} = 40.4\ (\sigma_y/\sigma_o - 1)^5 \qquad (1)$$

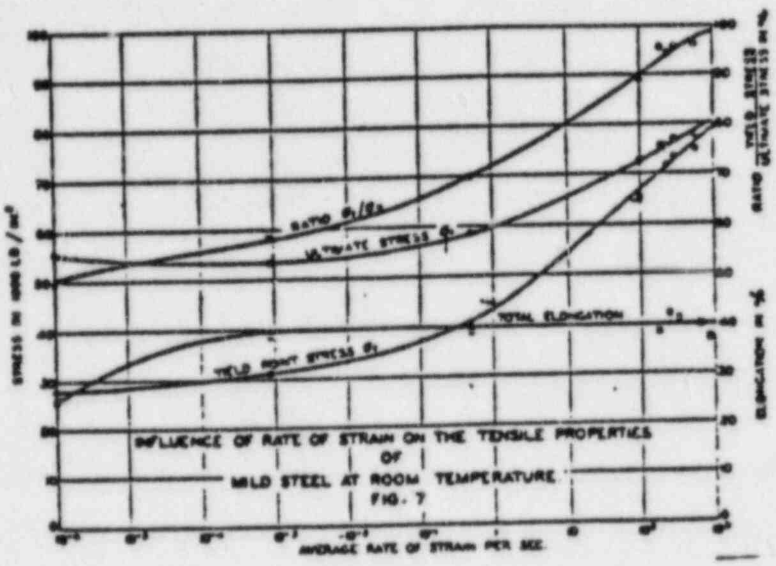

Fig. 1: Effect of Strain Rate on Tensile Properties of Mild Steel at Room Temperature, From Reference (5)

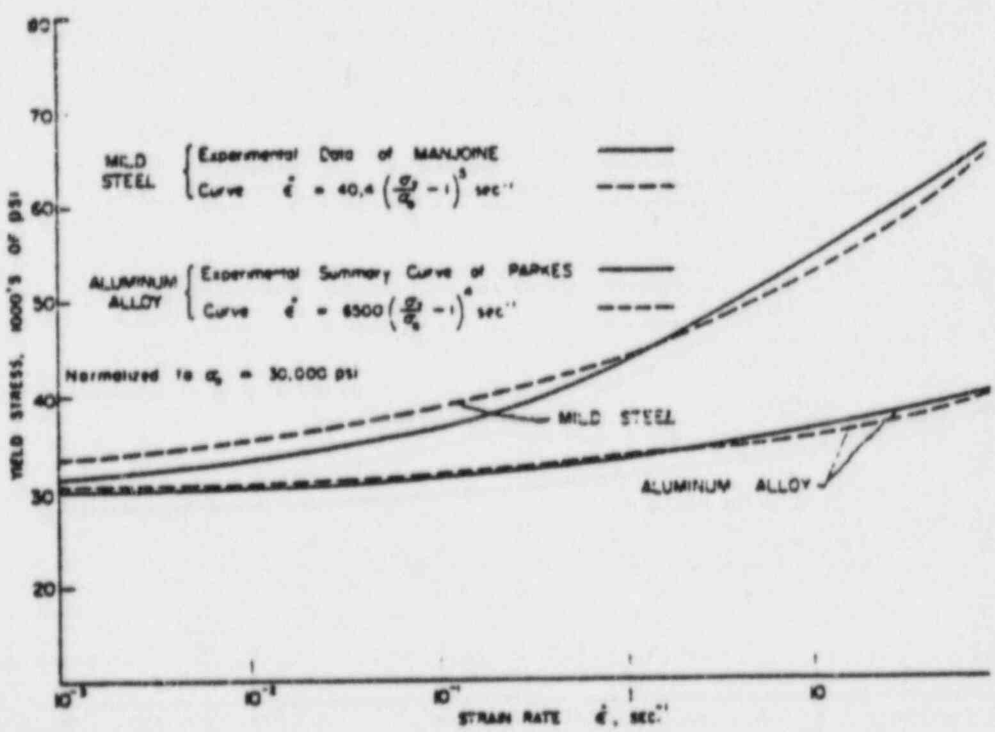

Fig. 2: Correlation Equations for Effect of Strain Rate on Yield Strength at Room Temperature, From Reference (6)

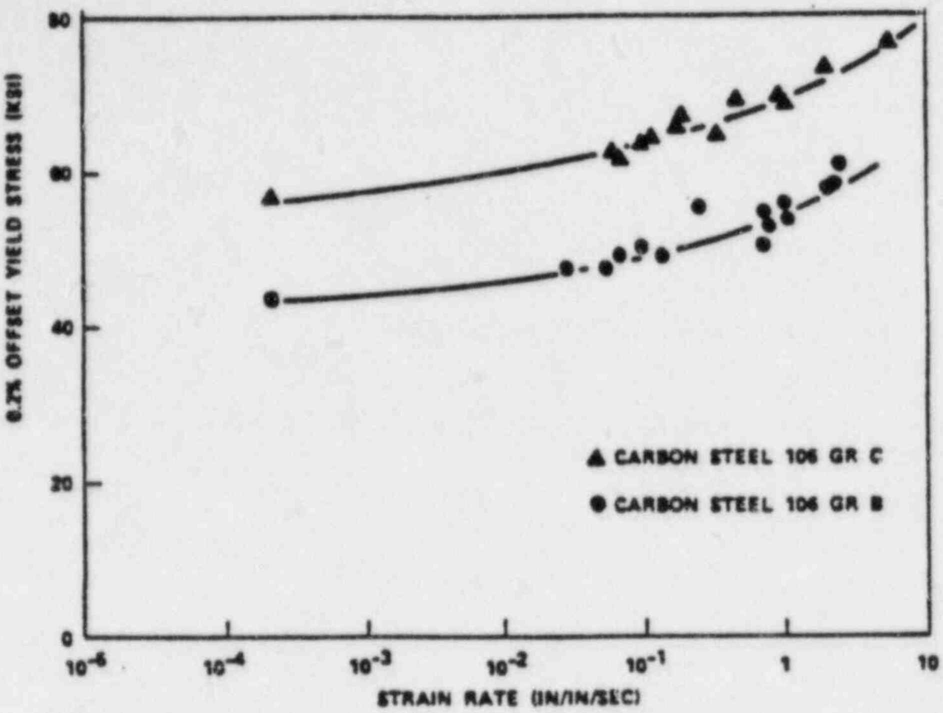

Fig. 3: Effect of Strain Rate on Yield Strength of SA106 Carbon Steel at Room Temperature, From Reference (7)

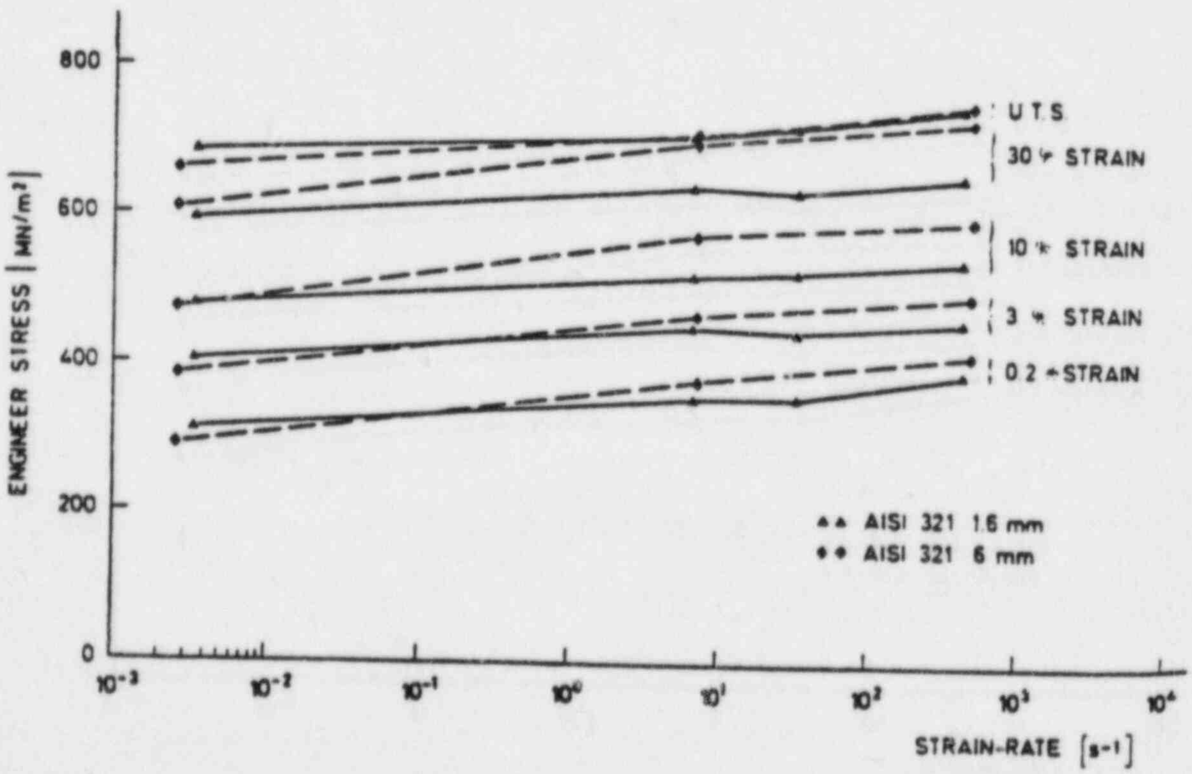

Fig. 4: Effect of Strain Rates on Flow Stresses of TP321 Stainless Steel at Room Temperature, From Reference (8)

where $\dot{e}$ = strain rate, in/in/sec

σ_y = yield strength, function of strain rate

σ_o = static yield strength, taken as 30 ksi

For the present purpose of seeing how the dynamic yield strength, σ_y, varies with strain rate, Eq. (1) can be written as:

$$\sigma_y = \sigma_o\left[(\dot{e}/40.4)^{0.2} + 1\right] \tag{2}$$

It is apparent in this form that $\sigma_y = \sigma_o$ only for $\dot{e} = 0$. At a standard testing strain rate of 1×10^{-3}/second, Eq. (2) gives $\sigma_y/\sigma_o = 1.12$. Accordingly, care must be taken in using Eqs. (1) and (2) if the static yield strength is determined by a "standard" tensile test. For example, Beazley[9] used the equation:

$$\sigma_y = \sigma_o\left[(\dot{e}/100)^{0.1} + 1\right] \tag{3}$$

to represent 0.2% yield strength of TP304 stainless steel at room temperature data given by Steichen[10]. From Eq. (3), Beazley states that a strain rate of 20 in/in/sec "increases the yield strength by as much as 85%". The coefficient of σ_o in Eq. (3), for $\dot{e} = 20$, is indeed 1.85. However, inspection of Steichen's data indicates a yield strength of about 32 ksi at the "standard" strain rate of 0.001 in/in/sec and a yield strength of about 47 ksi at a strain rate of 20 in/in/sec; giving a ratio of 1.47 rather than 1.85.

A strain rate of 100 in/in/sec corresponds to loading an elastic ($E=3\times10^7$psi) structure from zero to 60,000 psi in 0.00002 seconds. During this time, a stress wave in steel will travel only about 0.35 ft. For many dynamic events such as an impact on a pipe, a strain rate of 100 in/in/sec may be about an upper bound rate. From this viewpoint, the ratios of yield strengths at $\dot{e} = 100$ to those with $\dot{e} = 0.001$ (standard rate) are of interest. Some test-derived ratios are shown below.

Material	Ref.	Fig.	$\sigma_y(\dot{e} = 100)/\sigma_y(\dot{e} = 0.001)$
Mild Steel	(5)	1	2.2
A106 Grade B	(7)	3	1.9
A106 Grade C	(7)	3	1.7
TP321	(8)	4	1.4
TP304	(10)	-	1.5

As indicated in Figures 1 and 4, strain rates also influence the flow stress (stress to produce a given amount of strain), the ultimate tensile strength and the instability or maximum-load strain. Figure 5 shows complete stress strain curves for TP 321 material, tested at 20C. Out to about 30% strain, the flow stresses and ultimate tensile strengths increase with increasing strain rate but the instability strain decreases.

2.2.2 Use of Strain Rate Effects in Analyses

In an elastic analysis, strain rate effects could be used to defend somewhat higher allowable stresses than those established for static loading. For example, the minimum yield strength of SA312 TP304 material at 100F is 30 ksi. Assuming a dynamic event that involved strain rates in the elastic region of 20 in/in/sec, the Code 2 S_y limit might be taken as 2x30x1.3 ksi rather than 2x30 ksi.

Inelastic analysis methods have been used by several authors in making comparisons with test data. Bodner[6] states:

"Very good agreement is obtained by the inclusion of a strain rate-dependent yield stress into the governing equations."

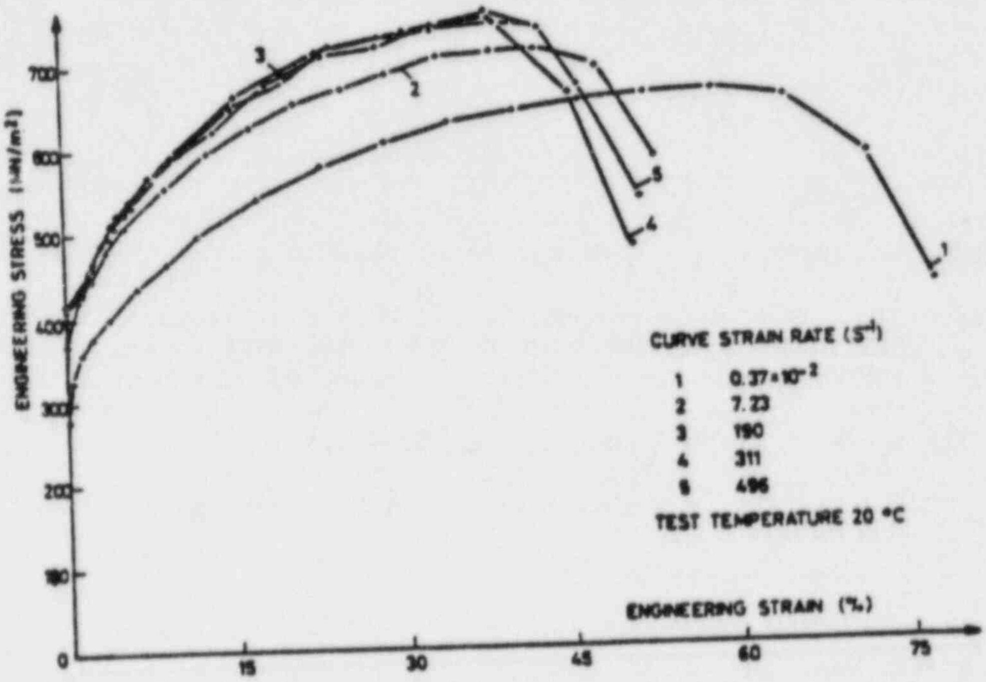

Fig. 5: Stress-Strain Curves at Various Strain Rates, TP321 Stainless Steel at Room Temperature, From Reference (8)

Anderson[11] states:

"The effect of strain rate on the initial yield level of a material cannot be neglected in systems subjected to impulsive or impactive loading."

Beazley[9] states:

"Material strain rate effects were found to be very important in the dynamic response and cannot be neglected without causing unnecessarily high degrees of conservatism."

The Code, in Appendix F, F-1322.3, "Material Behavior", states that:

"When performing a plastic analysis .... It is permissible to adjust the stress-strain curve to include strain rate effects resulting from dynamic behavior."

These statements indicate that, in the opinion of several workers in the field of inelastic dynamic analysis, strain rate effects are significant and should be utilized in analyses of impulsive or impactive loads. Our recommendation is:

In performing an inelastic analysis, it is permissible to include strain rate effects, provided a comprehensive report is prepared for review and acceptance by NRC. That report must include a detailed description of the basis for the strain rate effects and how strain rate effects are incorporated in the analysis.

The following comments are pertinent to the portion of our recommendation following the word "provided".

Bodner[6] investigates the relatively simple problem of a solid, rectangular-cross-section cantilever beam. He uses limit load (rigid-perfectly plastic) theory to estimate the beam resistance. In changing from strain-rate-independent to strain-rate-dependent analysis, his limit moment is increased as indicated by Eq. (2). Use of strain-rate-dependence, according to Bodner, completely changes the kinematics of the system for impulse loading. Bodner, in his Tables 3 and 4, shows test data and both strain-rate-independent and strain-rate-dependent analysis results. Cross comparisons show that, indeed, the strain-rate-dependent analysis checks better with the test data.

Anderson[11] investigates the relatively simple problem of a solid, rectangular-cross-section beam which may b- cantilevered or fixed at both ends. While Anderson uses Eq. (2) to describe strain effects, it is not clear how these are incorporated in his analysis. He does not show any comparisons between rate-independent and rate-dependent results; hence, the basis for his rather strong conclusion that "The effect of strain... cannot be neglected..." is not apparent from the paper. Anderson shows a large amount of calculated responses and a few measured responses but comparisons between them by the reader is difficult. One exception is in his Fig. 9(g) where he shows measured residual plastic deformations that are less than given by his analysis by a factor of about 5.

Beazley[9] investigates the considerably more complex problem of an impact (dropped weight) on a straight pipe. This is more complex because strains will vary in a complex manner in the pipe, both around the circumference and along the pipe axis. Beazley shows comparison of analyses results with test data; apparently the analyses include rate-dependent effects. There are then no analytical results for rate-independent so the basis for his conclusion that "Material strain rate effects were found to be very important..." is not apparent in the report.

Beazley[9] states:

> "For high rates of strain this relationship (Eq. 3 herein) predicts an increased yield stress, with the slope and shape of the hardening curve remaining the same."

This statement is analogous to the statement in Code Appendix F:

> "It is permissible to adjust the stress-strain curve to include strain rate effects resulting from dynamic behavior."

Now, questions arise as to just what is meant by these statements. We note that strain rates will vary widely depending on the circumferential/axial location on the pipe and location with respect to through-the-wall thickness. Further, Beazley's results show strain rates varying significantly during the time of the dynamic loading (high during the first millesecond, then much lower). Accordingly, it appears that "adjusting the stress strain curve" is (or should be) a rather complex process in which $\dot{e}$ is a function of location and time during the dynamic event. Beazley(9) states that the computer program ABACUS was used in his analysis and gives a brief but impressive description of its capabilities. However, while he devotes about two pages to what he calls "Analysis Parameters", there is no hint as to how strain rate effects were embodied in the analysis.

The preceding raises some questions concerning the adequacy of available material test data to confidently estimate strain rate effects in structures. In structures, bending often dominates; hence, the strain rates will be high and tensile on one surface, close to zero at the midsurface, high and compressive on the opposite surface. The available data is almost entirely restricted to tests in which a uniform tensile strain is applied.

(a) What are the strain rate effects in compression?

(b) What are the strain rate effects in bending?

(c) The available data is for a constant strain rate. In dynamic events, the strain rate may change. What would happen if, for example, a tensile test was run at a strain rate of 100 in/in/sec up to a strain of 0.01; then the strain rate was reduced to 1 in/in/sec?

Other questions could be added to this list. However, hopefully, the preceding explains the last portion of our recommendation. We think it inappropriate to prohibit use of strain rate effects but we would regard any such analysis with reservations unless convinced otherwise by a comprehensive description and defense of the analysis method.

3.0 RECOMMENDATIONS

3.1 Code Stress Limits and Inelastic Analysis

(a) Change NB-3672.6 and NC/ND-3673.1 to:

"Method of Analysis. All piping systems shall be analyzed by a structural analysis unless they can be judged adequate by an engineering comparison with previously analyzed piping systems. The stress limits provided in NB(NC, ND)-3650 were developed for use in conjunction with elastic analysis methods. Those stress limits are not necessarily appropriate when inelastic analysis methods are used. Inelastic analysis methods may be used provided the method and stress or strain limits used therewith are justified in the Design Report.

(b) Research Program on Strain Limits for Inelastic Analysis

A program should be initiated with the objective of developing acceptable strain limits for use with an inelastic piping system analysis. Strain limits should be established for all commonly used ferritic and austenitic piping materials and, in particular, weldments therein (e.g., girth butt welds and branch connection welds). Uniaxial, biaxial and triaxial strain fields should be addressed.

3.2 Strain Rate Effects

NRC should permit use of strain rate effects in an inelastic analysis; provided a comprehensive report is prepared for review and acceptance by NRC. That report should include a detailed description of the basis for the strain rate effects and how strain rate effects are incorporated in the analysis.

4.0 REFERENCES

(1) ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components", 1983 Edition with Summer 1983 Addenda, Published by ASME, 345 E. 47th St., New York, NY 10017.

(2) ASA B31.1-1955, "Code for Pressure Piping", Published by ASME, 345 E. 47th St., New York, NY 10017.

(3) Rodabaugh, E. C. and Moore, S. E., "Evaluation of the Plastic Characteristics of Piping Products in Relation to ASME Code Criteria", NUREG/CR-0261, July 1978. See also, Moore and Rodabaugh, "Background for Changes in the 1981 Edition of the ASME Nuclear Power Plant Components Code for Controlling Primary Loads in Piping Systems", ASME J. of Pressure Vessel Technology, Vol. 104, pp. 351-361, November 1982.

(4) Davis, E. A., "The Effect of the Speed of Stretching and the Rate of Loading on the Yielding of Mild Steel", ASME Trans, Vol. 60, p. A-137 (1938).

(5) Manjoine, M. J., "Influence of Rate of Strain and Temperature on Yield Stresses of Mild Steel", J. of Applied Mechanics, Vol. 11, ASME Trans., Vol. 66, pp. A-211-A218 (1944).

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POSITION PAPER ON WATER HAMMER

AND OTHER DYNAMIC LOADS

BY

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Generic Issues Branch

DST/NRR

REVISION 1
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POSITION PAPER

WATER HAMMER LOADS

1.0 Statement of Issues

Water hammer can occur as a result of pump start-up in voided lines, steam-driven slugs of water due to steam-pocket collapse, operating system(s) misalignments and design deficiencies. Since 1968, about 150 water hammers have been reported in U.S. nuclear power plants; damage has been confined principally to pipe hangers and snubbers. In two instances, the Indian Point-2 Plant in 1972 and the Maine Yankee Plant in 1983 experienced water hammers in the feedwater systems which resulted in breach of the secondary side pressure boundary. None of the water hammer occurrences have resulted in any release of radioactivity.

The USNRC staff has studied the water hammer issue generically and has concluded that the frequency and severity of water hammer occurrences has been significantly reduced through a) incorporation of preventive design features such as keep full systems, vacuum breakers, J-tubes, etc., and b) increased operator awareness and training. The staff's technical findings are reported in NUREG-0927[1]; these findings were utilized to revise portions of the SRP to ensure maintaining proven design concepts for minimizing or avoiding water hammer.

Water hammer piping loads are dealt with in SRP Section 3.9.3, Appendix A, Rev. 1. Since water hammer occurrence cannot be prevented the potential for such loads should be considered for normal operation, upset, and faulted conditions as defined in specified service-loading combinations identified for ASME Class 1 components and Class CS Support Structures per the ASME Boiler and Pressure Vessel code, Section III, Div. 1. Table I, Appendix A, of SRP Section 3.9.3 was modified as follows:

"These events must be considered in the pipe-stress analysis and pipe-support design process when specified in the ASME code-required Designed Specification. The Design Specification shall define the load and specify the applicable Code Service Stress Limit. For clarification, it should be noted that the potential for water hammer and water (steam) hammer occurrence should also be given proper consideration in the development of Design Specifications."

Thus, the NRC design requirements are based on endorsement of ASME code requirements and the development of adequate design specifications is incumbent on the applicant and his designer. The adequacy of these design specifications is therefore the key issue when addressing dynamic loads (such as water hammer) and combined dynamic loads. This subject is further discussed below.

2.0 Discussion of Issues

Total elimination of water hammer occurrence is not feasible, because inherent in the design of nuclear power plants is the possible coexistence of steam, water and voids in the various plant systems. Experience shows that design inadequacies and operator-or maintenance-related actions have contributed about equally to initiating water hammer occurrences. Therefore, the systems' design specifications become a focal point for preventive design measures.

2.1 Current Design Practice

Current design practices are based on ASME code requirements and, therefore, system design specifications are developed. These specifications are normally developed by a systems designer (ie, NSSS and AE designer) and cover such items as plant operating conditions (e.g., pressure, temperature, flows, etc.) and transients, expected loads, load combinations to be considered, etc. The system design specifications are then given to the piping and structural analysts for developing detailed analysis specifications.

Generally speaking, this current practice appears to be working since preventative design features have been incorporated into operational plants (based on operational experience) and are being proposed for plants in the OL cycle. A more specific identification of where water hammer can occur, and underlying reasons which could be assistance to the system designer can be extracted from NUREG-0927, NUREG/CR-2781[2], and NUREG/CR-2059[3].

2.2 Anticipated Water (Steam) Hammer Loads

An anticipated water or steam hammer is one which could result in a component performing in the manner for which it has been designed, and thus loading the system in its expected manner. Typical examples of anticipated water (steam) hammers are those caused by valve closures, pump trips, and pump start-up into voided lines. Anticipated water hammers that are generally included in piping-support system design considerations are: (a) steam hammers induced by turbine stop valve (TSV) closure, (b) possible control rod drive (CRD) insertion water hammers, and (c) water hammers caused by the trip and restart of open loop, safety-related service water pumps. These types of water (steam) hammers should be considered in developing design specifications, because they can occur when components such as TSVs and CRDs perform their intended function. TSV and CRD actuation occurs frequently enough to warrant their inclusion. Pump trips and start-ups are also frequent occurrences which should receive similar consideration.

In general, the closure, or opening of valves in most systems does not result in significant water hammers because typical valve closure times (5 to 120 seconds) are several orders of magnitude longer than the pressure wave sonic transit times ($\tilde{-}$.1 seconds) within the system lines. An exception to this is turbine stop valves that close in 0.1 seconds. However, because of the lower density and sonic velocity of steam, TSV loads are smaller than those occurring in water-filled lines. Reviews of typical analyses indicate that loads caused by TSV closure are large when compared to seismic and other piping loads and are generally included in design specifications. Except for TSV closure and CRD insertion loads, measurable loads from normal valve opening or closing have never been significant enough to be considered in nuclear power plant design. On the other hand, check-valve closures can result in high loads, particularly if inadvertent system misalignment occurs. Another load source is pump start-up into voided lines.

Although pump trip is a common occurrence in power plants, pump trip-induced water hammers have not generally been reported in nuclear power plants. This is the case because pump coastdown times (2 to 5 seconds) are long relative to pressure wave transit times. A potential exception is open-loop service water systems since water lines which run from the ultimate heat sink to the plant may be several thousand feet long. Additionally, the service water lines discharge at a low elevation and at ambient pressure. The high points on loop service water systems can have column separation and drainage leading to line voiding. Although such water hammers have not occurred during plant operation, analysis and preoperational testing has shown that water hammer caused by pump trip in an open-loop service water system is possible. Therefore, such water hammer loads warrant consideration in developing the design basis for service water systems if damage from these occurrences are to be minimized or avoided.

The start-up of pumps into voided lines has been a significant cause of previous water hammers (particularly in BWRs). Incorporation of keep full (or jockey) pumps appears to have minimized such water hammers. However, pump start-up into voided lines should be considered in developing system design specifications since it could lead to incorporation of design features for avoidance (i.e., use of void detection systems).

Anticipated water (steam) hammer loads should be combined with seismic loads because the events causing these loads can be initiated by a seismic event. Seismic and water hammer loads should be combined using SRSS methods rather than absolute summing for the reasons discussed below. Seismic loads have a short (milliseconds) distinct peak load that is significantly higher than other portions of the load. Individual piping segments exhibit peak response to water hammer loads for intermittent short (millisecond) periods. Therefore, although the probability of seismic and water hammer peak loads occurring simultaneously is low, it would be appropriate to sum these loads using SRSS methodology.

2.3 Unanticipated Water Hammers

An unanticipated water or steam hammer is one that would not be expected from a component or system operating in the manner for which it was designed. Examples of unanticipated water hammer include those caused by steam bubble collapse (i.e., SGWH), void filling (i.e., pump starting) and water entrainment in steam lines. The most recent occurrence of water hammers in the feedwater systems at the Salem plant on April 6, 1984, and at Calvert Cliffs 2 on April 22, 1984, are examples of failure to observe precautions in system operating or maintenance instructions. Thus, unanticipated water hammers cannot be specifically included in the design basis of piping.

Unanticipated water hammers are difficult to include in the design basis of piping for several reasons including:

- frequency of occurrence is low and unpredictable;

- such water hammers are often caused by plant operational upsets and maintenance causes and are generally introduced by operator or maintenance actions;

- postulating water hammer scenarios, yet more severe than experienced, is an open-ended endeavor which can lead to misleading conclusions.

As noted previously, unanticipated water hammers have not resulted in catastrophic failures. Generally speaking, such occurrences have been the result of plant operational transients (i.e., loss of feedwater, SG water level loss trip) and/or maintenance related. In other instances, audible water hammer has been noted; however, followup inspections have not revealed any damage. In some cases, piping supports have been severely damaged indicating that the water hammer loads far exceeded piping support systems' design margin. The Indian Point-2 plant (in 1972) experienced a water hammer in the feedwater (FW) system which ruptured a pipe. Maine Yankee experienced a water hammer in 1983 which cracked a FW pipe in the SG nozzle region (the nozzle already having incurred IGSC cracking). Water hammer in the feedwater system(s) of PWR steam generators employing a top feed-ring design is an example of large, unanticipated SGWH loads which should be anticipated. Therefore, the system designer in his preparation of those design and operational specifications should consider such unanticipated water hammers as probable and design for avoidance thereof.

Water hammer forces in liquid-filled lines can be propagated through piping with little attenuation except at branches. Therefore, a support system that could accommodate large water hammer loads would require installing very large supports at almost every piping segment. Such supports would make the piping system unnecessarily stiff and would create considerable access and inspection problems. The installation of such devices to partially mitigate events of low frequency of occurrence that have not had a significant effect on plant safety would reduce rather than increase plant safety. Therefore, it is recommended that, while efforts to reduce the incidence of unanticipated water hammers should continue, loads from hypothetical unanticipated water hammer should not be included in the design basis of piping systems.

NUREG-0927 can be used to derive expected "unanticipated" water hammers.

2.4 Classification of Water Hammer Behavior and Analysis

From the perspective of the piping design analyst, or systems analyst, there are two fundamental classes of water hammer which should be considered. These are:

1. simple pressure waves; and
2. two-phase water hammer.

For the first type of water hammer, there are well developed methods of analysis. The term "pressure waves" refers to classical water hammer encountered in hydraulic analysis and deals with the transmission, reflection and attenuation of abrupt changes in pressure throughout piping networks. Analysis of these one-dimensional pressure waves has resulted in well developed analysis techniques and many computerized methods are available for engineering design analyses.

"Two-phase water hammer" relates to situations involving both gas and liquid. These situations may range from the traditional column separation to condensation-induced slug acceleration and impact, or flow oscillations. They include such phenomena as pump discharge into voided lines, and transmission of pressure waves in liquid systems which can ingest air or other non-condensables. Computerized analyses of these two-phase water hammer situations is often limited due to physical computer modeling limitations of the physical phenomena and requires considerable judgment in the development and application of analytical methods.

2.4.1 Water Hammer Wave Analysis

There are five major elements in the analysis of water hammer events involving pressure waves:

1. identification and definition of load sources;

2. wave-guide analysis;

3. development of forcing functions;

4. structural analysis; and

5. comparison with acceptance criteria.

All of each of the above elements have been computerized to varying degrees.

Typical water hammer load sources include flow ramps due to control valves, abrupt flow stoppage (e.g., due to check valve slam), pump on/off transients, flow instability (e.g., due to limiting by automatic control systems). Quantitative definition of these sources depends on component specifications, ad-hoc analysis, and engineering judgment. Some computer codes for wave guide analysis incorporate selected versions of idealized load sources.

"Wave-guide analysis" is the subject of most so-called water hammer codes. Most of the computer codes used employ the method of characteristics (MOC) to track "shock waves" as they travel through and reflect throughout piping networks. These waves emanate from the source point and result in a distribution of pressure and velocity fields. Various analytical methods resolve these fields in either the space-time or frequency domains.

The water hammer analyst (or computer code specialist) supplies the piping analyst with a so-called "forcing function." The calculated pressure and velocity fields are converted to forces imposed on the piping system. Sometimes the pressure field itself is important to evaluate deformation of the piping due to hoop stresses. Although these forcing function calculations are sometimes computerized, they are more often done manually.

Thus, the piping analyses are dependent on the forcing function provided, and the structural codes then calculate the stress and defection of the piping, accounting also for piping restraints and external supports (e.g., hangers and snubbers).

Ultimately, the structural analyst compares the calculated piping and support stresses with allowable-limit criteria based on requirements for the class of piping or system being analyzed and the type of load (see also Table 1, SRP 3.9.3, Rev. 1). By these conformance methods, the analyst ensures that the piping is adequately supported and appropriately configured.

Well over a hundred computer codes amenable to analyzing classical water hammer loads are available in the United States. Some are available through public domain sources (i.e., National Laboratories), others can be obtained through commercial leasing, purchase or arrangements to use through various computer service companies. Some of the more commonly used codes are: WAVENET, PTA, RELAP, WHAM, WHAM 6, etc. Also, the major A-E's have developed highly specialized piping and support analysis codes which are proprietary to their respective companies.

Two-phase Water Hammer

Two-phase water hammer loads can occur in single-phase systems as well as two-phase systems (i.e., such as certain BWR systems or PWR steam generators that are designed to operate under two-phase fluid conditions). Also, in some liquid systems the second phase can be the result of either a gas source, or gas produced as the result of transmission of a flow change or pressure wave. Examples of situations with a gas source include: (a) SRV discharge of alternating gas and liquid (slug flow) into SRV piping, (b) top feedring water hammer initiated by ingestion of steam into the feedring from the steam generator vessel, or (c) vapor presence in BWR core-spray piping. Examples of two-phase situations caused by a fluid transient include the typical water column separation conditions, pump surge, or situations in high energy systems where pressure transients lead to flashing and void generation during a depressurization followed by cavity collapse and water hammer due to the subsequent compression wave.

The usual approach to two-phase water hammer relies on a sequence of identification, evaluation, understanding, quantification, and resolution. The resolution may involve design (or modification of existing hardware), but more often it also involves operating procedures and limits. Avoidance and preventing of the load is more often of value than strengthening the piping and supports.

Analysis is of use principally as an aid to understanding rather than as a rigorous predictive method. Theoretical two-phase loads are often grossly over conservative.

Appendix A discusses further PWR steam generator water hammer loads since such water hammers have resulted in PWR feedwater piping failures.

2.5 Severity of Water Hammer Occurrences

The USNRC and its subcontractors have periodically performed comprehensive reviews of water hammer events in the U.S. nuclear industry. These events are based on approximately 150 water hammer incidents since 1967. Only one incident led to a pipe rupture of the secondary system pressure boundary, this being at Indian Point No. 2 on November 13, 1973, and resulted in a rupture in an 18 inch feedwater pipe following impact of water slug resulting from a water hammer in the steam generator. This event and its details are described in NUREG-0291.[4] More recently, on January 25, 1983, a water hammer occurred at the Maine Yankee Plant which fractured an existing crack in the feedwater piping at the steam generator FW nozzle the initial crack being the result of prior IGSC. Other reported SGWH events resulted in either no damage (noises were heard) or damage to pipe hangers and snubbers, or damage was confined internally to the feedring and support structure.

Since opinions have been set forth regarding the possible occurrence of "catastrophic" water hammer occurrences in non-nuclear applications, a quick-look survey was undertaken in early 1984, and the findings are presented below.

1. Wilkinson and Dartnell[5] surveyed a 20 year period including 150 thermal power stations in the range 30 to 660 MW capacity. They state that "35 cases of failure were found," mainly

breakage of cast iron gate valves. They review one such incident in detail--that at Fiddler's Ferry station which involved a fatality. About half of the incidents involve flashing followed by condensation and water slug impact.

2. Signor[6,7], Smith, and Dubry[8] describe events in the steam distribution system maintained by the Detroit Edison Company. The system was comprised of over 50 miles of steam distribution piping, some of which had been in service since 1904. They mention several ruptures of this piping over a period of two decades as well as failure of a test pipe (which was constructed to evaluate the problem).

   On March 21, 1973, the Consolidated Edison steam distribution system experienced a severe expansion joint rupture in a section of 24 inch main.[9] "The explosive force of the rupture tore a 30 ft. by 18 ft. crater in the street and showered the area with mud and debris...several hundred windows in nearby buildings were broken." On October 11, 1977, a steamline rupture in a steam distribution system in Birmingham, Alabama,[10] resulted in the death of two workers. A steam main line ruptured and a control valve was fractured.

Common to the above events were:

a. low pressure steam distribution systems which were not required to be designed to ASME code requirements;

b. questionable operating procedures prevailed and underlying reasons were generally undocumented;

c. unexpected water being in the line with the result being rapid condensation and water slug impact; and

d. the possibility of faulty condensate traps.

It should be noted that expansion joints (which failed) are weak links in piping systems, cast iron valves are prone to brittle fracture, and poor (or lack thereof) conformances to proper design and construction procedures were involved in the accidents noted above.

Fossil fueled power plants also experience steam-water hammer events, particularly in lines connected to direct contact heaters (generally deaerators) which are common in such plants. The nonequilibrium conditions existing in a direct contact heater, along with the large number of lines carrying fluids at different thermodynamic states and flow rates, make direct contact heater systems more susceptible to water hammers than other systems. These events generally occur during rapid transients and off-design (generally low power) operating conditions, or when control components malfunction. Plants that serve swing and peaking functions have many transients at low power and are more prone to water hammer than base-load plants. Modifications to eliminate water hammers are made if it is felt that the events present a safety hazard or if it is cost effective to do so from an equipment protection standpoint. Typical examples are as follows:

a. In one two-unit, coal-fired plant, condensate lines under-went large water hammers following plant trips. Several water hammers had occurred with large (one foot) line movements that resulted in extremely loud sounds and support damage. However, no pipe cracking or leaks occurred.

b. Water hammers in another coal plant, originating in the direct contact heater system, resulted in considerable pipe hanger damage. The forces were large enough that movement of the heater occurred. No pipe cracking or leakage occurred. All structural damage was noted in areas of long, flexible piping runs. No damage or evidence of significant pipe motion was noted in areas containing short pipe segment lengths, or near piping anchors.

c. A five-unit, oil-fired plant was averaging two-to-three water hammer events per week for several years. The events were occurring in lines attached to deaerators and generally took place during low power transients or trips. These units undergo over 800 start-ups and shutdowns per year as well as many more rapid transients. Considerable pipe support and building structural damage, including crushed floor grating, has been observed. Evidence that a 90 foot long section of pipe had moved three feet is a more specific example, and some lines may have undergone plastic deformations. The evidence of these motions and damages has occurred in long, flexible lengths of piping. Pressure rises also caused relief valves to lift. However, no evidence of pipe rupture was noted even with the repetitiveness and magnitude of the events. Valve leakage and pump-seal leakage had been observed and this leakage, although repaired, was not significant enough to prevent plant operation. In one unit, the piping was supported more rigidly and evidence of water hammer induced damage was greatly reduced.

These fossil plant water hammer experiences illustrate non-nuclear plant water hammer occurrences and reveal that the affected piping can be subjected to large repetitive water hammer loads without loss of function. In addition, these examples may be pointing out the benefits to be gained from non-rigid piping support systems plus use of ductile piping. The limited effect of water hammer loads on piping integrity is likely due to the ductility and stength of power plant piping materials employed. Cast iron piping or valves were not employed in systems noted above. Also, castastrophic water hammer effects were not in evidence.

3.0 Load Combinations

Development of dynamic load combinations should be based on the following considerations:

1. the susceptibility of safety systems to dynamic loads, one of which is water hammer;
2. the frequency of occurrence;
3. the potential for simultaneous occurrence;
4. the safety implication(s) of piping failure; and
5. load magnitudes and load frequency distribution.

3.1 Safety System(s) Susceptibility

Safety system susceptibility is defined herein as that potential for dynamic loads to occur because of design features, or such systems for PWRs and BWRs. Water hammer has occurred in many of the identified systems, although incorporation of certain design features (i.e., keep full systems in BWRs and J-tubes in PWR steam generators employing a top feedring) and operator awareness have contributed to significantly reducing water hammer occurrence. NUREG/CR-2781 and NUREG-0927 detail and discuss water hammer occurrences, systems affected, and underlying causes.

3.2 Frequency of Occurrence

Frequency of water hammer occurrence and failure on demand models derived from reported events is reported in SAI's report[11] entitled, "Probabilistic Assessment of Unresolved Safety Issue A-1: Water Hammer, 'January 1983.'" For the PWR systems listed in Table 1, frequency of occurrence is in the range of 2.5×10^{-4}/yr to 1.7×10^{-2}/yr; for the BWR systems listed in Table 2, the range is 1.0×10^{-3}/yr to 3.5×10^{-2}/yr. On the other hand, SRV discharge occurrence in PWR main coolant and main steam systems is on the order of 10 occurrences/yr, BWR main steam SRV is also on the order of 10 occurrences/yr. Thus, if frequency were the only consideration in load combinations, water hammer loads should be adjusted downward accordingly relative to other dynamic loads. In contrast, vibrational loads are a continuous load throughout plant life and have resulted in piping failures.

3.3 Potential for Simultaneous Occurrence

Normally occurring vibrational loads have the highest likelihood of occurrence in conjunction with a seismic event. These vibrational loads are generally introduced by pump operation during normal and start-up plant operations. Although major vibrational loads are normally discovered during plant hot functional testing and eliminated by design changes, fluid flow induced vibrational loads exist throughout the operating life of the plant.

On the other hand, seismic events could result in loss-of-offsite power, turbine trip, etc. A scenario could be postulated (i.e., following a turbine trip) in which main isolation valves and turbine trip valves close, resulting in a steam hammer which would be followed by SRV discharge for BWRs, or main steam relief valve actuation for PWRs. Although these occurrences do not occur simultaneously, the steam hammer and SRV loads might occur while the seismic event is in progress. Due to the short duration of the seismic event, ECCS initiation would likely occur afterwards. Table 3 provides an overview of the above discussion and includes flow-induced vibration loads due to ECCS start-up.

Other dynamic load combinations (exclusive of seismic event occurrence) are summarized in Table 4.

Table 1: **PWR Safety Systems Susceptible to Other Dynamic Loads**

PWR Plånt Safety System	Potential for Vibrational Load	SRV Discharge	Potential for Water-Hammer Load	Actual Water Hammer Occurrence	Potential Steam-Hammer
Feedwater system			✓	✓	
Reactor coolant system	✓	✓	✓		
Main steam system		✓	✓	✓	✓
Auxiliary feed water system	✓		✓	✓	
Residual heat removal system	✓		✓	✓	
Chemical and volume control system	✓		✓	✓	
ECCS safety injection system	✓		✓	✓	
Containment spray system	✓		✓		
Auxiliary cooling water system	✓		✓	✓	
Spent fuel pool cooling system	✓		✓		

Table 2: **BWR Safety Systems Susceptible to Other Dynamic Loads**

BWR Plant Safety System	Vibration Potential	SRV Discharge	Water-Hammer Potential	Water-Hammer Occurrence	Steam-Hammer Potential
Feedwater	✓		✓	✓	
Residual Heat Removal System	✓		✓	✓	
High Pressure Coolant Inj. System	✓		✓	✓	
Reactor Core Isolation Cooling System	✓		✓	✓	
Safety Related Portions of the Main Steam Sys.		✓	✓	✓	✓
Auxiliary Cooling Water Systems	✓		✓	✓	
Reactor Recirculation System	✓				
Standby Liquid Control System	✓				
Spent Fuel Pool Cooling System	✓		✓		
Safety Related Portion of the Reactor Water Cleanup System	✓		✓	✓	
Control Rod Drive			✓		
Isolation Condenser	✓		✓	✓	

Table 3: Potential Seismic Induced Multiple Load Combinations

Seismic Induced Initiating Event	BWR Other Dynamic Loads Concurrent w/Seismic Event	BWR Systems Involved	PWR Other Dynamic Loads Concurrent w/Seismic Event	PWR Systems Involved
Loss of Offsite Power	Steam Hammer	Main Steam	Steam Hammer	Main Steam
	SRV Discharge	Main Steam Reactor Recirculation	SRV Dicharge	Main Steam Reactor Coolant
	Vibrational[3] (Flow Induced)	ECCS[1] Reactor Recirculation[2]	Vibrational[3] (Flow Induced)	Reactor Coolant[2] Emergency Feedwater System[1]
Turbine Trip	Steam Hammer	Main Steam	Steam Hammer	Main Steam
	SVR Discharge	Main Steam Reactor Recirculation	SVR Discharge	Main Steam Reactor Coolant
	Vibrational[3] (Flow Induced)	ECCS[1] Reactor Recirculation[2]	Vibrational[3] (Flow Induced)	Reactor Coolant[2] Auxiliary Feedwater System[1]

Footnotes:
1. Vibrational loads concurrent with seismic loads only if ECCS initiation prior to completion of the seismic event.
2. Pump induced vibrational loading until coast down of tripped pump.
3. Significant vibrational loads are identified and eliminated during preoperational testing.

Table 4: Potential Multiple Other Dynamic Load Combinations

Other Dynamic Load Combinations	BWR Systems Impacted	Cause for BWR Load Combinations	PWR Systems Impacted	Cause for PWR Load Combinations
Steam Hammer and Relief Valve Discharge	Main Steam	Turbine Stop Valve and/or main steam isolation valve closure	Main Steam	Turbine stop valve and/or main steam isolation valve closure
Pump Induced Vibration and Water Hammer	All standby and intermittent operating systems susceptible to flow into voided line water hammer	Flow into voided lines after pump start	All standby and intermittent operating systems susceptible to flow into voided line water hammer	Flow into voided lines after pump start

3.4 Safety Implications of Piping Failure

PWR Safety Systems

PWR safety systems which operate continuously or intermittently during full power operation include the feedwater, reactor coolant, main steam, chemical and volume control, auxiliary cooling water, and spent fuel pool cooling systems. Postulated worst case piping failures for these systems are summarized in Table 5--alternate and/or shutdown. Redundant shutdown paths provide a means for safe plant shutdown.

PWR systems normally in standby include the auxiliary feedwater, residual heat removal, ECCS, and containment spray systems. Other than those portions of these systems which may be used for normal plant start-up, shutdown, or abnormal conditions, these systems are in a standby mode. Most of the water hammer events in standby systems have occurred during tech spec testing (see NUREG/CR-2781) and have not affected the normally operating plant systems.

BWR Safety Systems

BWR safety systems which operate continuously or intermittently during full power operation include the main steam, auxiliary cooling water, reactor recirculation, spent fuel pool cooling, and reactor water clean-up systems. No postulated single worst case piping failure in these systems will prevent safe plant shutdown as shown in Table 6.

Systems normally in standby include the core spray, high pressure coolant injection, reactor core isolation cooling, and standby liquid control system. Other than plant start-up, shutdown, abnormal, or test conditions, these systems are in a standby mode. If system failure occurs during tech spec testing, there is no effect on normally operating plant systems, and redundant safety systems are still available.

TABLE 5: SAFETY SIGNIFICANCE OF PWR PIPING FAILURES

System	Postulated Worst Case[(1)] Failure	Alternate or Redundant Shutdown Paths
Normally Operating:		
Feedwater	Loss of normal feedwater	Auxiliary feedwater and all plant safety systems remain available for safe plant shutdown.
Reactor Coolant	Loss of coolant accident (LOCA)	ECCS and all other plant safety systems remain available.
Main steam	Main steam line break	ECCS and all other plant safety systems remain available.
Chemical & Volume Control System	LOCA	ECCS and all other plant safety systems remain available.
	Failure of boron concentration control capability	Control rods and reactor protection systems remain available.
Auxiliary Cooling Water	Loss of one cooling water loop.	Redundant loop remains available.
Spent Fuel Pool Cooling	Loss of one cooling loop	Redundant loop remains available. Total spent fuel pool cooling loss has no immediate adverse effect on plant safety.
Standby (2):		
Auxiliary Feedwater	Loss of auxiliary feedwater to one steam generator	Normal feedwater, residual heat removal, auxiliary feedwater, to other steam generators, and other safety systems remain available.
Residual Heat Removal	Total loss of residual heat removal	Auxiliary feedwater remains available.
ECCS	Loss of one safety injection loop.	Other safety injection loop or loops and accumulators remain available.
Containment Spray	Loss of one containment spray loop.	Redundant containment spray loop remains available.

Footnotes:

1. The postulated failures have never occurred. However, postulations were made to determine worst consequences.

2. No direct safety impact on plant if failure occurs during testing.

TABLE 6: SAFETY SIGNIFICANCE OF BWR PIPING FAILURES

System	Postulated Worst Case(1) Failure	Alternate or Redundant Shutdown Paths
Normally Operating:		
Feedwater	Loss of feedwater LOCA	ECCS systems available.
Residual Heat Removal	Loss of one loop.	Redundant cooling loop and other ECCS remain available.
Main Steam	Main steam line break.	ECCS and all other plant safety systems remain available.
Auxiliary Cooling	Loss of one cooling water loop.	Redundant loop remains available.
Reactor Recirculation	Loss of coolant accident (LOCA)	ECCS and all other plant safety systems remain available.
Spent Fuel Pool Cooling	Loss of one cooling water loop	Redundant loop remains available. Total spent fuel pool cooling loss has no immediate adverse effect on plant safety.
Reactor Water Cleanup	LOCA	ECCS and all other plant safety systems remain available.
Standby (2):		
Core Spray	Loss of one core spray loop.	Redundant core spray loop remains available.
High Pressure Coolant Injection (HPCI)	Loss of HPCI	Automatic depressurization system and other ECCS remain available.
Reactor Core Isolation Cooling (RCIC)	Loss of RCIC	Other ECCS and plant shutdown systems remain available.
Standby Liquid Control (SLC)	Loss of SLC	Control rods and reactor protection system remain available.
Control Rod Drive	Loss of insert line	Standby liquid control system available
Isolation	LOCA, loss of isolation condenser cooling capability	Feedwater and plant safety systems remain available.

Footnotes:

1. The postulated failures have never occurred. However, postulations were made to determine worst consequences.

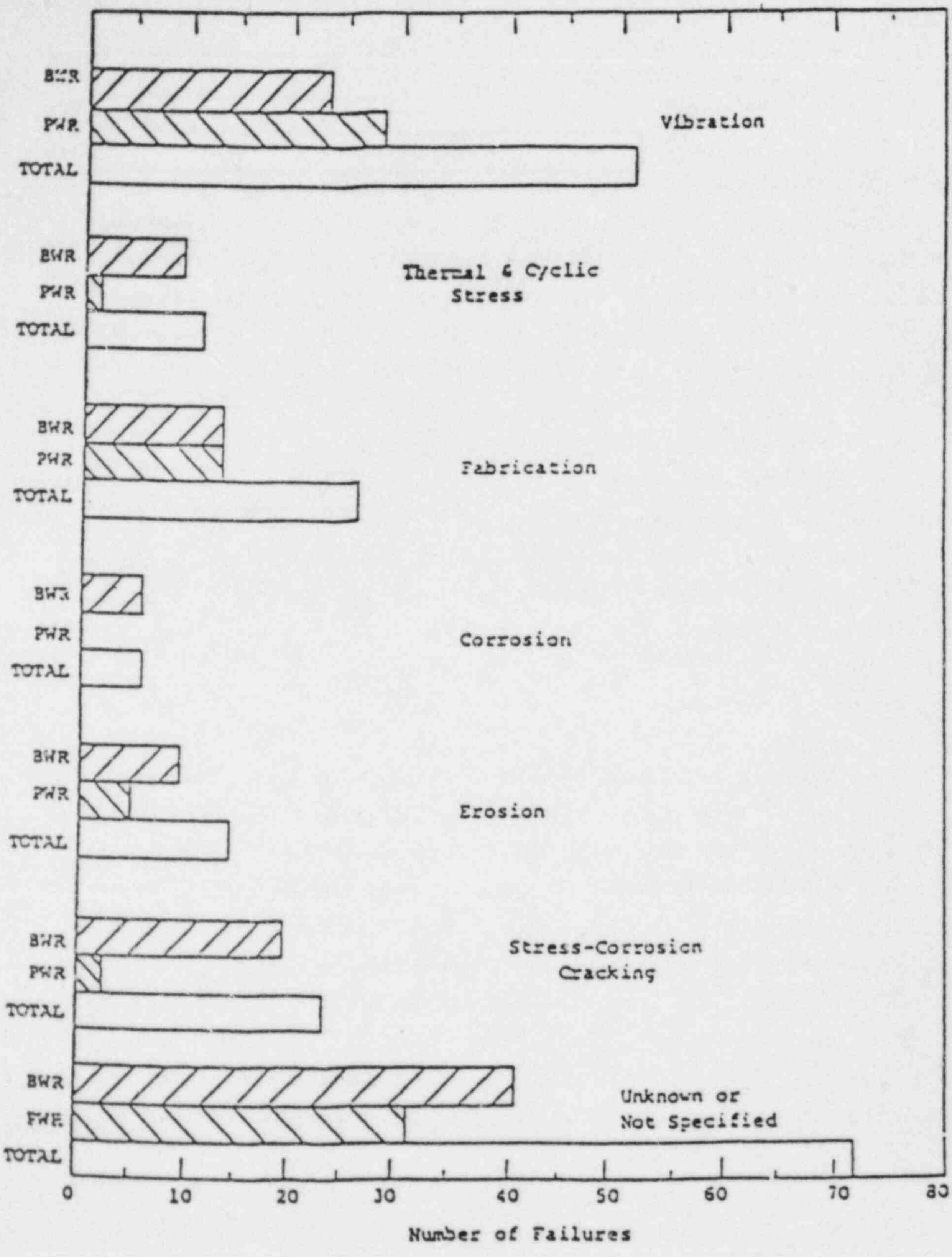
2. No direct safety impact on plant if failure during test.

Although water hammer loads which could lead to piping rupture, or failure, are the principal topic of this position paper, it should be recognized that piping failures have occurred from a wide variety of causes, including vibration loads and metallurgically-induced failures. This is illustrated in Figure 1, which was abstracted from EPRI's Report NP-438, "Characteristics of Pipe System Failures in LWRs," August 1977.

Although a more current survey (none was found by the author) would likely alter the data shown, Figure 1 is introduced as a caution against over rating the significance of only water hammer loads - either as the most significant dynamic load (which is not the case) or at the expense of ignoring other potential dynamic loads during development of the design specifications.

3.5 Load Magnitude Estimates

A conservative estimate of water hammer loads can be made by assuming the pipe to be rigid and the flow to stop instantly. The maximum pressure rise is the product of fluid density, wave velocity and the change in fluid velocity. In those cases where the water hammer is caused by rapid valve closure, valve closure time has a significant effect on the water hammer load. The valve closure time is generally compared with the wave travel time (2L/a) where "L" is the distance the wave has to travel before it is reflected, and "a" is the wave velocity. For example, if the valve closure time is 3 times the wave travel time, then the actual pressure rise will be 30 - 40 percent of the theoretical value.



NOTE: Abstracted from EPRI Report NP-438.

Figure 1 COMPARISON OF FAILURE MODES WITHIN BWR'S VERSUS PWR'S

Using the same method for steam-hammer loads caused by turbine stop valve closure, an estimate of that load can be obtained. For a main steamline with a flow area of 3 sq.ft. and a flow-rate of 1000 lb/sec of saturated steam at 1000. psi, the theoretical pressure rise is about 140 psi, which produces an axial load of about 60 kips. A computer analysis would produce a force time history for each pipe segment, and calculate a maximum peak load of approximately 40 kips which is somewhat less than simplified, one-dimensional analyses would predict.

Water hammer loads due to check valve closure in the feedwater line are on the order of 50 kips. The magnitude of this load is very sensitive to how rapidly the check valve closes. Ideally, the check valve should close as soon as the flow stops. Any delay from that point on will cause substantial increase in the loads.

Control Rod Drive (CRD) hydraulic valves open in 20 - 60 ms and can create water hammers. Analysis discussed in Reference 12 reports piping segment forces may reach 700 pounds and transient pressure peaks may reach 2800 psi. Both of those values are within the design capability of the piping system.

Estimating the SRV loads for BWR plants is more involved due to the complexity of the phenomena associated with a closed discharge system. The submerged portion of the discharge line contains a slug of water that has to be expelled before the air and then steam can be discharged. The water slug is rapidly accelerated and usually expelled in less than 0.5 seconds. As it makes a 90 degree turn in the discharge device (usually a sparger), it exerts a large axial force on the order of 50 - 100 kips on the discharge line. This force is in the form of a sharp spike with a mean width of 20 - 30 msec. The rest of the discharge line, i.e., the portion which is not submerged, experiences loads of much lower magnitude. These loads are due to pressure waves introduced by the inflow of steam and reflected back and forth between the water slug interface and the SRV.

To summarize, water hammer loads due to check valve closure in the feedwater line are less than 50 kips, and SRV loads range from several kips to about 100 kips for the submerged portion of closed discharge systems. Water hammer pressure loads in CRD lines are about 700 pounds with peak pressures as high as 2800 psi. Steam hammer loads due to TSV closure are less than 50 kips.

An unanticipated water or steam hammer is one that would not be expected from a component or system operating in the manner for which it was designed and for which proper operating procedures have been employed. Examples of unanticipated water hammer include those caused by steam bubble collapse, void filling and water entrainment in steam lines. Unanticipated water hammers generally involve bubble collapse, water entrainment or void filling. In all of these cases, a slug of water is accelerated through a void and is instantly stopped upon impact with a closed valve or a water-filled section of piping. PWR top feed-ring SGs and FW systems have shown susceptibility to unanticipated water hammers.

Because of the number of variables involved, unanticipated water hammer loads can only be estimated through bounding analyses. The range of observed forces due to unanticipated water hammers is very large. Some events caused no visible damage while others caused considerable damage to the piping support systems, indicating that the forces exceeded the design basis of the system. For instance, steam generator water hammer (SGWH) can produce local pressures as high as 6000 psi. Such pressure spikes, however, are not propagated down the piping because pressure is reduced by plastic deformation of the piping (bulging). A pressure rise of 2500 psi can be propagated through the piping producing a 500 kips force in an 18 inch feedwater line.

In summary, the frequency content (or load time) of water hammer forcing function depends on: a) wave speed in the pipe, b) pipe lengths in the system, c) segment lengths (between elbows), and d) location of the segments. Water hammer loads based on wave reflection theory predict step function loads that lead to high impulse loadings. On the other hand: a) the magnitudes of the forces are lower mainly due to the fact that in real life, flow stoppage does not happen instantly but takes a finite time, b) the forcing function is smooth and does not contain step changes. This is also a result of the finite time it takes to stop the flow, c) events slightly delayed - this is due to the fact that the actual wave speed is lower than the theoretical one, due to pipe expansion and other factors such as presence of gas bubbles, and d) the magnitudes of the forces decay rapidly due to various loss mechanisms such as mechanical, viscous, etc.

4.0 Proposed Recommendations

Because of the multi-disciplinary nature of the problem, there does not exist a systematic and uniform treatment of water hammer, or other dynamic loads, in developing design specifications except for major events such as turbine stop valve closure, feedwater line break and SRV discharge in nuclear power plants. The following comments, therefore, have to do with the implementation of the existing requirements and are not proposed changes to existing ASME code or NRC requirements:

1. As discussed in Section 1, the current ASME design codes and SRP Section 3.9.3 provide acceptable guidelines for incorporation of dynamic loads (including water hammer) into the development of design specifications. However, it is not always clear whose responsibility it is to determine the susceptibility of various plant systems to water hammer, or steam-water hammer (i.e., the systems designer versus piping designer paradox). If water hammer occurrence possibility is

not mentioned in the Design Specification(s), it is possible that water hammer loadings will not be evaluated.

2. Water hammer occurrences, underlying causes and corrective measures taken have been studied and are reported in NUREG-0927. However, because of the multi-disciplinary nature of the problem, there does not exist a systematic and uniform treatment of water hammer, or other dynamic loads 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 hammer or 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 these events. NUREG-0927 contains summary tables which identify systems that have experienced water hammer, the underlying causes, and remedial actions that could be taken.

   Underlying causes such as potential line voiding, steam pocket formation, flashing and unstable condensation due to entrapped condensate, etc., can be derived from NUREG-0927. Certain system design features have proven effective; certain systems have been more susceptible to water hammer. Thus, a common checklist could be developed. However, the wide variety in plant designs and operations works against development of a singular generic checklist. Therefore, the responsibility of including water hammer considerations into design specifications must rest with the plant owner or applicant, and the NRC should not be called upon to define an all-inclusive checklist and institute adoption thereof.

3. Efforts to reduce or minimize the incidence of water hammer should continue, with an emphasis in operator training and awareness to potential water hammer occurrence (see also NUREG-0927). Since loads from unanticipated water hammer are similar to those which can be designed against, the design specification(s) which deals with upset, emergency, and faulted conditions should be used to deal with such occurrences.

4. Design considerations related to water hammer loads in combination with degraded piping are beyond the scope of this position paper (and the scope of the task committee on other Dynamic Loads and Load Combinations). However, as illustrated by the Maine Yankee water hammer in January 1983, degraded piping and a water hammer can lead to a pipe crack. Thus, it is recommended that degraded piping in conjunction with anticipated dynamic loads (i.e., vibratory, SRV and water hammer) be given a broader review and consideration by the Piping Review Committee prior to arriving at conclusions dealing with the relaxation or change in piping support requirements.

4. More extensive discussions of dynamic loads, water hammers, analysis methods, etc., are contained in References 12 and 13, which were utilized by the author in preparing this position paper.

5. <u>Regulatory Value-Impact Assessment</u>:

   NUREG-0993, Revision 1, is the staff's regulatory analysis dealing with the resolution of the Unresolved Safety Issue A-1, Water Hammer. This report contains the value-impact analysis for this issue, public comments received, and staff response or action taken in response to those comments. The staff's technical findings regarding water hammer in nuclear power plants are contained in NUREG-0927.

Based on the USI A-1 technical findings, the following actions were implemented:

1. Issuance of the revised SRP Sections for forward-fit implementation, these being SRP Sections 3.9.3, 3.9.4, 5.4.6, 5.47, 6.3, 9.2.1, 9.2.2, 10.3 and 10.4.7.

2. Issuance of NUREG-0927 as a technical findings document. This staff report summarizes the staff's assessment of water hammer in nuclear power plants.

3. Ensure operator awareness and training with respect to avoiding water hammer through the use of the TMI Task Action Plan, Part I.C.5 and Part I.A.2.3, operator training evaluation criteria under current development by the Licensee Qualifications Branch.

4. Conclusion of current Operating License reviews through staff evaluations in progress.

The "forward fit" nature of these actions has minimal industry impact, and the suggestions made above (which are in keeping with current ASME code requirements) should likewise have minimal impact.

6.0 References

[1]NUREG-0927, Rev. 1, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," A. W. Serkiz, USNRC, March 1984.

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[4]NUREG-0291, "An Evaluation of PWR Steam Generator Water Hammer," J. A. Block, et al., Creare Inc., June 1977.

[5]Wilkinson, G. J. and L. M. Dartnell, "Water Hammer Phenomena in Thermal Power Station Feedwater Systems," Proceedings IME, Volume 194, March 1980, pages 17-25.

[6]Signor, C. E., "What We Learned About Water Hammer," ASHRAE Journal, 2(4), April 1960, pages 69-71.

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[8]Smith, E. T., and E. E. Dubry, "Water Hammer Experiments," Proceedings International District Heating Association, 1951, pages 130-133.

[9]Bennett, A. J., "Avenue "C" Expansion Joint Venture," Proceedings International District Heating Association, (no date), pages 42-49.

[10]Deposition of Mr. J. S. Laster under Civil Action No. CV-79-04267 in the Circuit Court of the Tenth Judicial Circuit, Jefferson County, Alabama, July 2, 1980.

[11]Amico, P. and Ferrell, W., "Probabilistic Assessment of Unresolved Safety Issue A-1,: Water Hammer," January 1913.

[12]NUREG/CR-3939, "Water Hammer, Flow Induced Vibrations and Safety Relief Valve Loads," August 1984.

[13]Creare TM-963a, "Water Hammer Review," April 1984.

APPENDIX A

PWR STEAM GENERATOR WATER HAMMER LOAD PREDICTION

v.

MEASURED LOADS

PWR Steam Generator Water Hammer (SGWH)

Approximately thirty SGWHs have occurred in nuclear plants since 1969 in those steam generators designed with a top feedwater ring. Water hammer due to slug impact in the feedwater piping to PWR steam generators has been evaluated extensively by NRC and was the subject of a major review and study by Creare in 1976.[A1&A2] This two-phase water hammer situation is useful to review, because it provides a mix of extensive experience from operating plants together with theory and data from laboratory tests and plant tests.

A theory was presented in Reference A1 for the one-dimensional collapse of a steam cavity in a liquid-filled pipe. This theory was derived from "water cannon" experiments which were conducted with a driving pressure of one atmosphere and records taken for many events with overpressures in the range 800 to 1200 psi. High speed motion pictures were taken of the motion of the liquid slug in transparent piping, and it was determined that the slug traveled at velocities of about 20 ft./sec. Thus, the measured overpressures and the observed velocities were consistent with the Joukowski relation ($\Delta P = -\ PC\Delta V$). Two pipe materials were used, with a factor of 3 difference in calculated celerity, and the measured overpressure also differed by a factor of 3 as expected.

In the Creare experiments, the measured overpressures were consistent with a condensation effectiveness theory also developed by Creare with a single parameter C^* to represent values of C^* in the range $0.3 < C^* < 0.4$ (over a factor of 5 in driving pressure range). For values of C^* greater than unity, the cavity collapse is inertially limited. The condensation is so rapid that the cavity depressurizes essentially to zero in a very short time, and the terminal velocity of the water slug is limited by the distance available to accelerate it. Finally, the overpressures measured in these experiments were approximately one-half of the theoretical maximum.

Creare also tested a second model which simulated the top feedring geometry of PWR steam generators. Measured overpressures were in the range 300 to 700 psi. This further mitigation was traced to three physical factors: reduction of driving pressure in the feedring, necessity to accelerate stagnant liquid along bottom of pipe, and irregular interface at impact.

Tables A1 and A2 present comparisons of bounding theory calculations with data for the water cannon and steam generator models, respectively. The first two columns are the impulse (pressure time duration) felt before and after slug impact. The third column is the peak overpressure, P_h. Thus in well controlled laboratory conditions, actual impulses and overpressures were far below the theoretical maximum even for a highly one-dimensional steam cavity.

In addition, a water hammer occurred in the Tihange plant in Belgium. The transient pressure data from the Tihange plant which was operating at full pressure revealed three key facts:

1. a rapid and nearly complete depressurization from 70 bar to almost zero in about 20 ms was recorded in two locations. This corresponds a value of $C^* \sim 2$ and represents the highest known value ever recorded;

2. a feedwater system overpressure in excess of 6000 psi was recorded before the pressure transducers failed--this is 75% of the theoretical maximum;

3. despite this extreme load, the Tihange piping was not damaged.

Table 3 compares key Tihange data with bounding theory calculations. Thus, the Tihange plant data provides evidence of very rapid condensation

TABLE A1

COMPARISON OF IDEAL CALCULATION WITH MEASURED WATERCANNON DATA

Test #	I_s (psi-msec)	I_h (psi-msec)	P_h (psig)	V_s (ft/sec)	V_h/V_s	I_h/I_s	L_s (ft)
Ideal	1000	2000	2000	34	1.0	2.0	2.3
Test 1	598	1000	1200	20.0	1	1.67	2.3
2	345	450	500	11.5	0.71	1.30	2.3
3	522	900	1000	17.4	0.93	1.72	2.3
4	428	950	1100	14.3	1.25	2.22	2.3
5	546	1200	1300	18.2	1.15	2.20	2.3
6	586	800	1000	19.6	0.83	1.37	2.3
7	400	800	1000	13.3	1.22	2.00	2.3
8	482	800	1000	16.0	1.02	1.66	2.3
9	506	750	1000	16.9	0.96	1.48	2.3
10	473	900	1100	15.8	1.14	1.90	2.3
11	549	1000	1200	18.3	1.08	1.82	2.3

TABLE A2

COMPARISON OF BOUNDING THEORY WITH PRESSURE TRACES FOR STEAM GENERATOR MODEL

Test #	I_s (psi-msec)	I_h (psi-msec)	P_h (psig)	V_s (ft/sec)	V_h/V_s	$\frac{I_h}{I_s}$	L_s (ft)	Q (gpm)
Theory	1700	1700	1000	34	1.0	1.0	4.0	1
Test 1	525	175	700	34.1	0.67	0.33	1.18	1
2	479	94	375	31.2	0.39	0.20	1.18	1
3	411	125	500	26.7	0.61	0.30	1.18	1
4	494	88	275	22.9	0.39	0.18	1.65	1
5	478	178	475	20.7	0.75	0.37	1.77	2
6	632	188	500	27.4	0.60	0.30	1.77	2
7	517	193	550	24.1	0.75	0.37	1.65	2
8	540	227	500	22.9	0.71	0.42	2.12	4
9	447	252	475	16.2	0.95	0.56	2.12	4
10	428	210	600	19.5	0.98	0.49	1.65	5.3

TABLE A3

COMPARISON OF BOUNDING THEORY WITH TIHANGE DATA

	Theory	Data
Depressurization (psi)	1,000	900
Impulse Is (psi-msec)	55,000	40,000
Overpressure P_h (psi)	8,300	6,000*

* Maximum value recorded at which time transducer failed.

phenomenon and slug impact, as well as the design margin available in piping and support designs. It should also be noted that the Tihange plant had not taken preventive design measures, such as installation of J-tubes, which are installed in many U.S. plants.

At Indian Point No. 2, the steam generator water hammer caused a 180° circumferential rupture of the feedpipe near where the pipe penetrated through containment, and also produced a bulge in the feedpipe near the steam generator nozzle. Although no pressure data are available from this incident, calculations in Reference A1 show that the pipe bulge is consistent with the collapse of a steam void 2.8 feet long acting on a water slug about 2 feet long at impact thereby supporting the validity of such calculations.

The recommendations for the design and operation of top feedring plants resulted in the following combination of four items (see also SRP 10.4.7, BTP ASB 10.2).

1. modify top discharge feedrings by installing J-tubes to avoid drainage and steam ingestion;

2. incorporate prompt restart of feedwater into operating procedures to reduce the degree of drainage through the thermal sleeve;

3. utilize a short external feedwater pipe (preferably with a downward elbow) to minimize the horizontal length available to trap steam;

4. place limits on feedwater flow to slowly fill the feedring in order to minimize flow turbulence and suppress the onset of rapid condensation; and

5. conduct preoperational tests to demonstrate the avoidance of water hammer occurrence.

Extensive tests were performed at the Trojan plant in 1975 to demonstrate that geometry alone (see Items 1 and 3 above) suffice. No water hammer was recorded in these tests even though the piping was intentionally drained and feedwater flow was supplied well above the limit. Tests of the Trojan geometry in Creare's laboratory model did result in slug impact in a few cases, but overpressure magnitudes were reduced by a factor of 5 to 10 relative to other possible configurations.

In summary, the resolution of water hammer in PWR steam generators with top feedwater ring relies on design modifications or operating procedures. Plant experience shows that loads near the theoretical maximums can be achieved. This appendix illustrates the kinds of calculations that can be performed and order of magnitude loads in an extreme situation.

Appendix A References

A1 Block, J. A., et al.; An Evaluation of PWR Steam Generator Water Hammer, NUREG-0291, U.S. NRC, June 1977.

A2 Rothe, P. H., Wallis, G. B. and and Crowley, C. J.; Water Hammer in the Feedwater Systems of PWR Steam Generators; Fluid Transients and Acoustics in the Power Industry, ASME, December 1978, pages 75-87.

POSITION PAPER ON
PIPING SYSTEM DYNAMIC AND THERMAL STRESS RESPONSE
INDUCED BY
THERMAL-HYDRAULIC TRANSIENTS

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ABSTRACT

Issues on thermal-hydraulic and structural dynamic response analysis of piping systems affected by safety or relief valve opening transients are discussed in this report. The presentation also contains a review of recent experimental vs. analytical studies, summaries of the individual analysis steps, and guidelines for performance of these analyses. In addition, recommendations resulting from this review are given.

CONTENTS

POSITION PAPER ON
PIPING SYSTEM DYNAMIC AND THERMAL STRESS RESPONSE
INDUCED BY THERMAL-HYDRAULIC TRANSIENTS

1. STATEMENT OF ISSUES

All nuclear power plant piping systems are subject to dynamic design events. Prediction and evaluation of piping system dynamic and thermal events are included in design bases for these power plant piping systems. Contained in these even.. are thermal-hydraulic transients which induce dynamic and thermal stresses.

The responsibility for review of the applicant's safety analysis report (SAR) of such transients is granted to the U.S. Nuclear Regulatory Commission (NRC) by the Code of Federal Regulations.[1,2] These regulations further reference the primary code utilized by the nuclear industry: the ASME Boiler and Pressure Vessel Code, Section III.[3] Additional interpretive guidelines are supplied by the USNRC Standard Review Plan, 3.9.3[4] and the Welding Research Council Bulletin 269.[5]

Analysis of these S/RV discharge transients are complex analyses involving multidisciplinary processes which include four links of the analysis and evaluation chain. These four links consist of: thermal-hydraulic analysis, mechanical load calculation, structural dynamics analysis, and transient thermal stress analysis. Each link of the chain contains uncertainties and potential errors due to inaccuracies of: (a) description of the physical system and initial conditions, (b) limitations of the representative governing equations, (c) generation of a consistent mathematical model, (d) algorithms and solution processes used, and (e) correct utilization or interpretation of results. In addition, due to the multidisciplinary nature of the task, potential communication problems may occur.

Several important issues arise when performing meaningful prediction and evaluation of piping system dynamic and thermal stress response induced by thermal-hydraulic transients. In particular, the issues relating to safety and relief (S/RV) discharge induced piping dynamic response are discussed in this paper and some recommended guidelines proposed. Basically, the question that must be answered is, "Do the postulated fluid and thermal loads and the resulting structural response evaluations accurately or conservatively describe the consequences created by SRV discharge?"

Specific issues are broken down into the various analysis processes:

- Thermal-Hydraulic Analysis
- Mechanical Loads Determination
- Structural Response Analysis
- Thermal Transient Stresses
- Results Evaluation.

All standards, guides, and codes specify what must be considered, under what circumstances, and how design analysis results are to be interpreted. The basic standard (a law of the land) is the Code of Federal Regulations,[1,2] wherein reference is made to the primary code utilized by the nuclear industry: the ASME Boiler and Pressure Vessel Code, Section III.[3] Recent interpretive guides pertaining to the subject are the USNRC Standard Review Plan, 3.9.3[4] and the Welding Research Council Bulletin 269.[5] In summary, Reference 4 states that S/RV discharge transients, when classified as design or service loading and when the system is Class 1, 2, or 3, shall be treated according to Appendix O of the ASME Code and the supplementary criteria given under II.2 of Reference 4. Appendix N of the ASME Code is supposed to provide guidance for fluid transient induced loads but is, at this time, "in course of preparation".

Although, for some simple mechanical systems subject to thermal-hydraulic induced transient loads, simple conservative analyses may be performed; discussions presented herein are concerned with more complex systems which require detailed analyses and often include computer code implemented analyses. It has been stated that simplified techniques and engineering judgment are sufficient requirements for good discharge piping design. However, if simplified techniques have not or cannot be validated or calibrated by either sophisticated techniques and/or experiment, the validity is in doubt. Also, it is emphasized that this discussion is not a thorough critique of the "state-of-the-art" but, rather, a brief discussion of those factors which need to be considered as prerequisites for accurate or conservative analyses. A more extensive discussion of the analysis and evaluation chain including background information is included in Appendix A.

1.1 Issues

The TMI-2 incident and others provided reasons for an increased emphasis within the nuclear power industry for more detailed standards and experimental programs relating to safety and relief valve (S/RV) discharge thermal-hydraulic transients and resulting attached piping system response. Within this section, some of the more pertinent issues related to piping analysis and evaluation for S/RV discharge events are explored.

1.2 Thermal-Hydraulic Issues

Thermal-Hydraulic analyses are always required to evaluate time dependent fluid temperatures and pressures acting on the pressure boundary of S/RV valves and associated piping. Issues that have been raised on evaluation of this loading environment usually relate to how well do these computations represent reality or a conservative set of design loads. These important issues usually include: (Appendix A contains further discussion and references.)

- What time step should be used?

- How much piping should be included?

- What fluid conditions need to be considered?

- How do multiple valve openings affect loadings?

- How does valve functioning affect loadings and are coupled mechanical valve behavior--hydraulic behavior analyses needed?

1.3 Mechanical Loads Issues

Utilization of thermal-hydraulics output for the determination of time-dependent mechanical loads is the link of the overall S/RV system analysis which is, at this time, the least systematized of the individual analysis processes. One reason for this is that, at this stage of the process, the thermal-hydraulics and structural response disciplines meet. The important issue here is: Does the thermal-hydraulics discipline communicate with the structural response discipline such that the analyses are compatible? Since this issue is different for each organization, further discussion will be limited to those guidelines contained in Appendix A. It is emphasized that a clear understanding of how the loads are generated and used is important to an adequate end result.

1.4 Structural Response Issues

Structural dynamic response to dynamic loads are always a consideration in evaluation of S/RV transients. This response has customarily been done using computer programs. However, dynamic time history, response spectra, and static (dynamic load factor) methods have been used. Issues of current importance include:

- What cut off frequency should be sufficient?

- What time step limit is adequate?

- What piping supports should be included?

- What damping is permitted?

- What dynamic load factors are adequate?

- Should axial effects be considered?

1.5 Thermal Transient Stress

It has been suggested that heat transfer between enclosed fluid and piping be neglected so that the thermal-hydraulics analysis is simplified. This does not imply that thermal stresses in the piping should be neglected. Rather, the heat transfer (fluid to pipe) may be decoupled from hydrodynamic calculations. A transient heat transfer and thermal stress analysis should be performed where required by ASME Code utilizing fluid temperatures obtained from the thermal-hydraulic analysis. There do not appear to be any strong issues in this area at this time.

1.6 Stress Results Issues

Two assumptions are made for the purpose of this discussion: the S/RV analysis is a portion of a design analysis (rather than an experimental study) and that the S/RV transient is specified as a service condition in the Design Specification. Thus, all resulting mechanical bending moments and thermal stresses must satisfy the requirements of the ASME Code.[3] The issues under this topic appear to be:

- How are loads to be combined?

- Should a fatigue evaluation be made?

- Should axial effects be considered?

2. DISCUSSION OF ISSUES

It is the purpose of this section to present a position on the issues listed above and recommend means for improving procedures where possible. It is emphasized that simplified or judgmental procedures are often proposed for S/RV transient analyses and evaluations. It remains the author's belief that unless these simplified techniques have been or can be validated by other sophisticated analyses or experiments, the validity is in doubt. The opinions presented here are further discussed in Appendix A and the 50 or so references included in that appendix. Appendix A provides numerous guiding comments for S/RV discharge transient analysis as well as background references for those interested in engaging in further research on the topic.

2.1 Thermal-Hydraulics

2.1.1 What time step should be used for fluid computations?

The answer to this question depends on a number of things. Typically, the maximum time step is limited by:

- The time step should be equal to or less than the wave travel time across the smallest fluid volume length.

- According to EPRI tests, the piping upstream of the SRV valve experiences pressure oscillations in the 170-260 Hz range when loop seal water passes through the valve. The time step should be small enough to represent these oscillations if the system may respond to these frequencies.

- Recent S/RV tests[6] have shown vibratory fluctuations caused by discharge in the 30-100 Hz range.

Therefore, the time step must be appropriate for the fluid conditions and geometry. Additionally, it should be adequate for structural response up to about 100 Hz.

2.1.2 How much piping should be included?

The results of numerous tests where fluid transients excite piping or components show that sufficient piping should be included in the model to define pressure and momentum forces accurately. This means the analysis should include effects of upstream boundary conditions, entrapped fluid in loop seals, planes of choking (orifices) and effects of submergence if they significantly affect S/RV discharge flow. Additionally, the fluid model needs to be defined in such a manner that time dependent forces are determined at points related to the structural geometry (i.e., elbows, orifices, T's, etc.).

2.1.3 What fluid conditions should be considered?

Again, consider all conceivable conditions which could occur. In other words, in addition to planned operating conditions, consider possible fluid leaks through the valves or liquid that can remain in the pipe. These conditions can create unexpected liquid slugs and associated pressure oscillations as the liquid is accelerated out of the piping system.

The fluid conditions typically producing maximum loads are liquid flow, high pressures, and low temperatures. Water slugs such as those occurring in loop seals create especially large forces when discharged through the system.

2.1.4 How do multiple valve openings affect loadings?

Present knowledge suggests multiple valve actuations can have a significant effect on pipe loadings and should be included in the design analysis. Where more than one valve actuates, it is difficult to establish a sequence of valve openings that produces maximum loading on the system. Adjusting the opening sequence to produce the most severe loading situation is a complex problem that could require many costly iterations. The solution to this problem for each plant is likely to be unique because of differing piping and support configurations between plants. Intuitively, adjusting the opening times such that the initial pressure waves from each valve arrive at a common junction downstream would produce seve.· loading in the vicinity of the common junction. Most plant installations, however, contain a significant amount of dynamic supports in the region of the common point, alleviating some of the potential high stresses in this region and isolating this region from the valves so as not to jeopardize operability of the valves or integrity of the valve inlet piping and pressurizer nozzles. Many licensees assume the valves to actuate simultaneously under multiple valve actuation conditions. This puts a large pressure wave in each valve discharge line at the same time, and assures that the waves from each valve will arrive at the common junction downstream within a short time of each other unless the individual discharge lines are of radically different lengths. The probability that other opening sequences would produce significantly greater loading should be small. Any peculiarities in specific plant installations should, however, be considered. Reference 6 contains data from a series of tests where effects of multiple valve openings were studied.

2.1.4 How does valve functioning affect loading and are coupled valve behavior--hydraulic behavior/analyses needed?

It has been observed that the effects of back pressure and other fluid-mechanical forces acting on a spring-loaded valve disc influence the position of the disc which in turn influences the valve flow characteristics. Appendix A contains further references and information. However at this time, solving the phenomonon of coupled behavior for mechanical-hydrodynamic forces in S/RV valves is not generally considered practical. Careful consideration of uncoupled response appears sufficient in many cases.

2.2 Mechanical Loads Issues

Due to dependence on system geometry, initial conditions, and codes used for S/RV system transient analyses, few general guidelines can be given. However, at locations where area or flow direction changes occur, obviously forces may be developed and the thermal-hydraulic model must be defined such that these forces may be accurately calculated. Particular attention must be paid at locations where the flow may become complicated such as at valves and tees.

2.3 Structural Response Issues

2.3.1 What cut-off frequency should be sufficient?

Little guidance is available concerning cut-off frequency (that maximum frequency of response considered in the particular analysis) for dynamic response analysis. It has been observed that fluid frequency ranges of 170-260 Hz exist in experiments and significant response has been measured in the 10-100 Hz range. This growing awareness indicates high frequency effects need to be considered to about 100 Hz and possibly higher if a particular design can experience the high frequency fluid transients and will respond to such loads. Additionally, the contribution to loads from frequencies beyond the cut-off frequency should be considered. For more discussion see Reference 48 of Appendix A.

2.3.2 What time step is adequate?

This question is tied directly to the previous question. Once a cut-off frequency has been established, the time step is then selected based on the convergence criteria of the solution algorithm used for dynamic response computations. Common practice is to set the time step equal to 0.1 times the period of the highest frequency of interest. Some methods can be shown to converge for larger time steps. However, the time step used must also be sufficiently small to closely approximate the applied hydrodynamic forces.

2.3.3 What piping supports need to be included?

This becomes a matter of engineering judgment. Those supports with support stiffness will affect system natural frequencies of vibration more than 10% should be included in the stiffness of the structural dynamic model. All support effects should be included so that support loads can be computed for design evaluation of the support. Those with less than the 10% effect can be treated in a simpler manner (i.e., fixed, etc.). Further guidelines are provided in Appendix A and particular piping analysis software guides.

2.3.4 What damping is permitted?

Recent damping research studies are providing information on piping damping as a function of: support type, stress level, frequencies, etc. New damping has also been proposed by the PVRC for comment. This will be an area of active change for several years. It is recommended that PVRC values be used for most cases. However, experimental values for similar piping and excitation levels should be permitted when properly justified.

2.3.5 What dynamic load factors are adequate?

Dynamic load factors (DLF) of 1.5 to 2.0 have been listed in some reports. Unless these DLF factors are developed and justified for each piping system with its particular configuration and set of fluid conditions, it is doubtful that they have any validity. The application of DLF factors developed for a single degree of freedom system loaded by a single impulse load simply do not apply to a series of fluctuations.

2.3.6 Should axial effects be considered?

It is the opinion of some authors that axial elongation of the piping should be modeled for the purpose of correctly approximating dynamic response of the piping system to the hydrodynamic loads. However, this does not mean that the ASME Code evaluations need to be modified to account for axial forces in the resulting stress computations. It merely means that structural dynamic response computations permit the axial deformation effects to be included when computing bending moments throughout the piping system.

2.4 Stress Results Issues

2.4.1 How are loads to be combined?

Clear-cut guidance does not appear available on this topic. The SRSS method of load combination provided in NUREG-0484 is generally accepted. Combination on an absolute value basis is also an acceptable but conservative approach.

2.4.2 Should a fatigue evaluation be made?

It is not clear how one should perform an ASME Code fatigue evaluation including S/RV transient induced stresses. It is believed that a fatigue evaluation should be conducted. The number of expected S/RV transient occurrences should be specified in the Design Specifications. However, little guidance is available, at this time, for determination of the number of effective stress cycles that should be specified for a given S/RV transient. Reference 6 has indications that there are about 7 to 100 cycles of significant motion in each discharge cycle. Further study in this area is certainly in order.

2.4.3 Should axial effects be considered?

As a final comment pertaining to stress results evaluation, the potential influence of piping elongation has been noted. Even though the ASME Code requires that only design mechanical bending moments rather than axial design mechanical forces be used for primary and secondary piping stress intensity evaluation, the axial extension of piping segments due to hydrodynamic loads should be considered in the structural dynamic response. The reason for this is that, especially for long straight bounded pipe segments, hydrodynamic load induced elongation of these segments induces bending moments which may not be negligible. However, it should not be interpreted as requiring the use of axial pipe forces in the ASME Code stress evaluations.

3. PROPOSED RECOMMENDATIONS

A brief summary has been made of issues raised by recent experimental and analytical studies relating to S/RV system thermal and structural response to discharge transients. Recommendations for resolution of these issues were generally made as the issue was discussed. Appendix A expands on these discussions and presents a wealth of additional resource information in the form of referenced reports and data. A relatively large number of experimental results are available with relatively few corresponding analytical comparisons. The comparisons that have been cited are, generally, not in "good" agreement with tests. It is recommended that where possible a comprehensive evaluation of these comparisons be undertaken so that a unified and more quantitative understanding of the ability to adequately perform S/RV system analyses is obtained. It is generally found that the thermal-hydraulic experimental vs. analytical comparisons are better than the structural response experimental vs. analytical comparisons. This is thought to be caused by: a) incomplete description of applied loads to the structure and b) error has been propagated in thermal-hydraulics computation and is further compounded in the structural response evaluation.

Appendix A and studies referenced in the appendix have shown that a complete S/RV system analysis is a complex multidisciplinary process involving several distinct analysis and evaluation steps. Perhaps the weakest link in the analysis chain is the utilization of hydrodynamic results for the prediction of mechanical load histories for subsequent input to structural dynamic response analysis. An additional recommendation is that a detailed evaluation of the load determination process be undertaken in conjunction with the additional experiemntal vs. analytical comparison study.

Finally, ASME Code evaluation of S/RV system transient results is required for safety evaluation. Here, the requirement for additional study is primarily in fatigue evaluation. Further evaluation of S/RV system transient test data and analysis results is required for the determination of a realistic number of stress cycles per transient that should be included in ASME Code fatigue evaluations. It should be noted here that S/RV transients affect piping both upstream and downstream of the valve.

In summary, analytical tools are available for accurately predicting and evaluating results of S/RV system transients. However, more work is required to learn how to effectively utilize those tools for realistic and effective analysis and evaluation of these systems. Additional work is also necessary to provide validation, improvement or elimination of simplified techniques for S/RV discharge response.

4. VALUE IMPACT

The value impact of the discussion of the above issues and subsequent recommendations is of mixed impact. The recommended actions are, in general, clarifications which lead to an improved design analysis. Therefore, the design loads should be more accurately defined and understood, resulting in a more reliable design. Correct application of dynamic principles such as selection of adequate time steps, inclusion of sufficient modes, inclusion of load effects beyond cut-off frequency, and determination of the number of significant cycles of stress increase cost very little in the design analysis process. It can provide great savings if a failure and/or a retrofit is prevented.

Cost of additional research into better determination of the number of fatigue cycles per S/RV discharge transient is probably minimal compared to the improved understanding of this problem. It can be piggybacked on other experiments and may even be extracted from data of experiments already performed. Additional research into improved loads evaluation and improved load application for dynamic response evaluation is expected to be an evolving process. Simplified methods, when qualified, should reduce analysis costs which to some extent will be counteracted by costs of methods qualification.

5. REFERENCES

1. 10 CFR Part 50, Paragraph 50.55a, "Codes and Standards," U.S. Code of Federal Regulations.

2. 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," U.S. Code of Federal Regulations.

3. ASME Boiler and Pressure Vessel Code, Section III, Division 1, "Nuclear Power Plant Components," American Society of Mechanical Engineers.

4. NUREG-0800 (Formerly NUREG-75/087), "USNRC Standard Review Plan," U.S. Nuclear Regulatory Commission, July 1981.

5. Bulletin No. 269, "Interpretive Report on Dynamic Analysis of Pressure Components - Second Edition," Welding Research Council, August 1981.

6. EGG-EA-6025, "Acceleration Data in the Suppression Pool Area from Kuosheng SRV Tests," G. K. Miller and C-Y. Yuan, EG&G Idaho, Inc., October 1982.

APPENDIX A

PIPING SYSTEM DYNAMIC AND THERMAL STRESS RESPONSE INDUCED BY THERMAL-HYDRAULIC TRANSIENTS

TECHNICAL LETTER REPORT

PIPING SYSTEM DYNAMIC AND THERMAL STRESS RESPONSE INDUCED BY THERMAL-HYDRAULIC TRANSIENTS

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EGG-EA-6506

PIPING SYSTEM DYNAMIC AND THERMAL STRESS RESPONSE INDUCED BY THERMAL-HYDRAULIC TRANSIENTS

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ABSTRACT

Complete thermal-hydraulic and structural dynamic response analysis of piping systems subjected to a safety or relief valve opening transient is a complex multi-step process. This presentation contains a review of recent experimental vs. analytical studies, summaries of the individual analysis steps, and guidelines for performance of these analyses. In addition, recommendations for further experimental and analytical study are given.

CONTENTS

PIPING SYSTEM DYNAMIC AND THERMAL STRESS RESPONSE
INDUCED BY
THERMAL-HYDRAULIC TRANSIENTS

1. OVERVIEW

Prediction and evaluation of piping system dynamic and thermal stress response induced by a thermal-hydraulic transient are complex multidisciplinary processes. A complete and accurate evaluation contains four links of the analysis and evaluation chain: thermal-hydraulic analysis, mechanical load calculation, structural dynamics analysis, and transient thermal stress analysis. Each link of the chain contains uncertainties and potential errors due to inaccuracies of: (a) description of the physical system and initial conditions, (b) limitations of the representative governing equations, (c) generation of a consistent mathematical model, (d) algorithms and solution processes used, and (e) correct utilization or interpretation of results. In addition, due to the multidisciplinary nature of the task, potential communication problems may occur. Thus, complete and accurate analyses of the subject mechanical systems must be carefully planned, the important parameters thoroughly understood, and the solution process accurately performed.

The primary purpose of this presentation is to outline and briefly discuss those factors which are necessary for meaningful prediction and evaluation of piping system dynamic and thermal stress response induced by thermal-hydraulic transients, exclusive of water hammer (subcooled hydraulic transients). In particular, safety and relief valve (S/RV) discharge induced piping response is addressed. Additionally, the current status of standards, codes, and experimental studies (S/RV systems) are briefly discussed. The following general outline is used:

1. Overview
2. Current Status (guides, standards, and experimental studies)

3. Thermal-Hydraulic Analysis
4. Mechanical Loads Determination
5. Structural Response Analysis
6. Thermal Transient Stresses
7. Results Evaluation
8. Summary and Recommendations
9. References.

Although, for some simple mechanical systems subject to thermal-hydraulic induced transient loads, simple conservative analyses may be performed; discussions presented herein are concerned with detailed computer code implemented analyses. Also, it is emphasized that this discussion is not a thorough critique of the "state-of-the-art" but, rather, a brief discussion of those factors which must be considered as prerequisites to useful subject analyses.

2. CURRENT STATUS

The TMI-2 incident provided reasons for an increased emphasis within the nuclear power industry for more detailed standards and experimental programs relating to safety and relief valve (S/RV) discharge thermal-hydraulic transients and resulting attached piping system response. Within this section, some of the more pertinent standards and experimental programs are reviewed.

2.1 Standards and Codes

All standards, guides, and codes specify what must be considered, under what circumstances, and how design analysis results are to be interpreted. The basic standard (a law of the land) is the Code of Federal Regulations,[1,2] wherein reference is made to the primary code utilized by the nuclear industry: the ASME Boiler and Pressure Vessel Code, Section III.[3] Recent interpretive guides pertaining to the subject are the USNRC Standard Review Plan, 3.9.3[4] and the Welding Research Council Bulletin 269.[5] In summary, Reference 4 states that S/RV discharge transients, when classified as design or service loading and when the system is Class 1, 2, or 3, shall be treated according to Appendix O of the ASME Code and the supplementary criteria given under II.2 of Reference 4. Appendix N of the ASME Code is supposed to provide guidance for fluid transient induced loads but is, at this time, "in course of preparation."

2.2 Experimental Studies

The TMI-2 incident prompted issuance of a series of USNRC NUREG's[6,7,8] which required the nuclear power industry to experimentally demonstrate operability of power operated relief and safety valves. The Electric Power Research Institute (EPRI) instituted a research program[9] which resulted in a large number of valve tests and, in addition, approximately 116 tests of PWR S/RV systems (with and without loop seals, steam, steam-water, and water). These system tests were conducted at the

Combustion Engineering test facility.[10] Results of some of the tests have been analyzed and/or compared with corresponding analysis predictions.[11-20]

A series of 29 relief valve discharge tests were conducted at the Kuosheng BWR-6/Mark III Nuclear Station[21,22] where the reactor was operating at 60% power. The test series consisted of single, consecutive, multiple, and extended valve actuations. Partial results and selected analysis comparisons concerning these tests may be obtained in References 23, 24 and 25.

The Federal Republic of Germany conducted a series of tests at the decommissioned Heissdampfreaktor (HDR) facility termed "German Standard Problem No. 4."[26] This system consisted of pressure vessel, primary piping, check valve, and rupture device. Although the tested system is not a conventional S/RV system, the thermal-hydraulic transient is analogous to a typical S/RV system transient. Results of analytical vs. experimental thermal-hydraulic and structural response comparisons are given in References 27 and 28, respectively.

Most of the test results vs. analytical comparisons contained some common important elements which are summarized as:

1. For both thermal-hydraulic and structural response models, construction of the mathematical model must be very detailed and accurate.

2. Small variations of assumed initial conditions, i.e., valve opening time--flow rate, significantly influence thermal-hydraulic predictions.

3. Coupled mechanical valve behavior--hydraulic behavior appears to be an important consideration that has not been adequately addressed.

4. Thermal-hydraulic models of multiple S/RV openings yield results not adequately comparable with experimental results.

5. Thermal-hydraulic predictions compare, in general, more favorably with test results than do structural response predictions. This may be partically attributed to cumulative error. However, inaccuracies of load calculation and structural modeling may also contribute to lack of test result--response prediction comparison.

In summary, the difficulty of adequately predicting thermal-hydraulic and structural response for S/RV systems subjected to valve discharge transients is demonstrated in these test vs. analytical comparisons. The following portions of this presentation deal with those factors which are necessary (however, not sufficient in the mathematical sense) to obtain adequate S/RV system transient response predictions.

3. THERMAL-HYDRAULIC ANALYSIS

The purpose of thermal-hydraulic analysis, for S/RV system design analysis, is the accurate prediction of those quantities necessary for realistic evaluation of the safety of the system. In particular, time dependent fluid temperatures and forces acting on the pressure boundary are required as input to additional analysis for final design safety evaluation. The thermal-hydraulic evaluation of a S/RV system is a complex process which requires extensive theoretical background and practical experience relating to two-phase thermo-hydrodynamic processes. The following paragraphs outline some of the more important topics that should be considered in thermal-hydraulic analyses of S/RV systems.

3.1 Computer Codes

A number of codes have been developed for general and special purpose thermal-hydraulic analysis. Three of the more general and widely used code families are RELAP,[29] TRAC,[30] and DAPSY[31]. Of these, the RELAP series is the most widely used for general two-phase thermo-hydrodynamic applications and will be used as a basis for further discussion. In particular, RELAP5/MOD1 appears, from the experimental vs. analytical comparisons cited previously, to be most applicable to S/RV system analysis.

RELAP5/MOD1 uses a two-fluid, five-equation (2 mass conservation, 2 momentum conservation, and an energy balance relation) model for twophase flow. An additional constraining relation is that one of the fluids is at the saturated state. The numerical mathematical model consists of control volumes, wherein scalar quantities are averaged, interconnected by nodes where vector quantities are defined. Since scalar quantities (pressure, temperature, density, etc.) are used as input for the additional structural dynamic response and thermal stress analyses, concern for the use of the thermal-hydraulic output should be considered during mathematical modeling.

3.2 Application Guidelines

Since a comprehensive study of all factors involved in thermal-hydraulic analyses is outside the scope of this discussion, only those topics which are particularly important are presented herein. References 32, 33, and 34 present thermal-hydraulic concerns as applied to S/RV systems. In addition, Reference 35 outlines those factors, relating to the use of RELAP5/MOD1 for S/RV system modeling, which were found to be important from many of the experimental vs. analytical studies cited previously. This outline is summarized as follows:

1. System modeling: The piping system should be represented by straight segments between the midplanes of consecutive elbows. The path length should be maintained. Segments about 2 ft in length should have 6 to 8 vol nodes. Segments from 2 to 5 ft in length need about 10 nodes. Segments from 5 to 10 ft need about 12 nodes. Longer pipes are unlikely but need no more than 12 nodes. Node segments should not be smaller than 0.25 ft. The choking option should be applied upstream of the valve at the orifice area representing the valve and at the exit junction. The option should not be applied in the downstream piping unless an area reduction is present. The valve flow orifice area should be sized to pass the measured or specified vapor flow rate at the specified pressure. The valve opening time should be set to the smallest measured or specified "pop time" (elapsed time for the valve to open completely from an assumed closed position after simmering) for vapor and liquid conditions upstream of the valve with the recommended ring settings. Since piping wall heat transfer is complex to model and adds to calculational difficulty, it is recommended that the effect be excluded. It is not necessary to model the relief tank since forces from the wave occur before significant flow exists at the exit. For multiple valve systems, piping loads in connecting runs are likely to be largest if waves from individually operating valves arrive simultaneously in the connecting piping. Thus, valve operation should be slightly staggered in time to insure wave addition to produce maximum piping loads.

2. Initial Conditions: Since downstream hydraulic forces are proportional to the initial downstream fluid density when piping heat transfer is not considered and, since leakage through the valve resulting in the downstream fluid being saturated steam is possible for non loop seal geometries, a reasonable initial quality should be assumed. Collection points for pools or slugs are possible and should be modeled as liquid full unless drains are present. For loop seal geometries, since valve simmering prior to pop is likely if the loop seal liquid is subcooled, liquid should be transported downstream assuming a constant enthalpy process. Liquid should be distributed in the first few downstream cells with vapor assumed in remaining piping if sufficient vapor is generated based on the assumed process and upstream mass.

3. Time Step: The maximum time step should be equal to or less than (smallest volume length)/(n·c) where c is the expected sonic velocity and n is equal to or greater than one. For two-phase or vapor flow conditions, the value of n should be set at 2 so that shock waves propagated downstream from the valve will not pass through a vol element in one time step. For subcooled liquids, it is recommended that n be set to 5 for optimum acoustic wave shape (assuming that c approximately equals 5000 ft/s).

As a final topic in thermal-hydraulic modeling, the phenomenon of mechanical valve behavior--hydrodynamic behavior coupling is briefly discussed. It has been observed[36] that the effects of back pressure and other fluid-mechanical forces acting on a spring-loaded valve disc influence the position of the disc which in turn influences the valve flow characteristics. References 36 and 37 present coupling models which account for the phenomenon. In addition, the model presented in Reference 37 has been used in conjunction with RELAP5/MOD1 calculations.

4. MECHANICAL LOADS DETERMINATION

Utilization of thermal-hydraulics output for the determination of timedependent mechanical loads is the link of the overall S/RV system analysis which is, at this time, the least systematized of the individual analysis processes. One reason for this is that, at this stage of the process, the thermal-hydraulics and structural response disciplines meet. Unfortunately, except in the rare case of the dual analyst, insufficient communication usually occurs between thermal-hydraulicist and structural analyst. A common result of this lack of communication is that the thermal-hydraulic available information is ill-suited for accurate mechanical load calculation. To circumvent this problem, the mechanical load determination process must be well planned, considering both thermal-hydraulics and structural response requirements, in advance of any calculational effort.

Two general formulations are used for mechanical loads determination: force balance and momentum balance. The first of these, the force balance method, equates resultant force transmitted from fluid to structural element as the sum of all pressure and frictional tractions acting on the wetted surface of the element. The momentum balance method equates resultant force on the element to the time rate of change of fluid momentum within a control volume. The following paragraphs briefly discuss these methods and present potential advantages and disadvantages of each method.

4.1 Force Balance Method

Many computer codes have been developed for force balance conversion of hydrodynamic output to time-dependent mechanical loads; two of which, designed for RELAP5/MOD1 output, are described in References 38 and 39. An advantage of the force balance method is that it is inherently stable due to the absence of time derivatives. In addition, it is relatively easily implemented due to its heavy dependence on pressure which is a principal variable of most hydrodynamic codes. The force balance method is particularly well suited to S/RV transients which involve liquid slug propagation due to its independence on time rate of momentum change.

The major difficulty encountered using the force balance method is the inherent difficulty of calculating fluid friction tractions acting on the wetted surface of an element. This is particularly troublesome where an S/RV transient involves only steam where pressure tractions are of the same order as friction tractions. Another potential disadvantage is that, for codes such as RELAP5 and where significant pressure differentials occur over a short length of pipe, the hydrodynamic model must be very finely divided due to pressure averaging in control volumes.

4.2 Momentum Balance Method

References 38, 40, and 41 describe computer codes that have implemented the momentum balance procedure for determining mechanical loads acting on piping systems. Briefly, the momentum balance equation results in a three term expression representing force on a given control volume. One term is the time rate of change of mass acceleration within the control volume which has been termed the "wave" or "acceleration" force. The other two terms involve pressure and momentum flux integrated over the inlet and outlet surfaces of the control volume. The pressure contribution to these terms has been termed "blowdown force." An advantage of the momentum balance method is that all quantities required for computing force are usually contained in the hydrodynamic code output. In addition, for RELAP5 type codes, the vector quantities utilized in the momentum balance are nonaveraged and located at nodes rather than being averaged over control volumes. Thus, distribution of resultant forces to structural nodes, where the hydrodynamic and structural nodes coincide, is relatively straightforward.

The principal disadvantage of the momentum balance method is the potential for numeric instabilities as mentioned previously. Thus, for applications where large time rates of change of vector quantities exist, caution must be exercised in the use of momentum balance methods. Perhaps the greatest usefulness of the momentum balance formulation is for S/RV transients which primarily involve fluids in the vapor state.

4.3 Modeling Guidelines

Due to dependence on system geometry, initial conditions, and codes used for S/RV system transient analyses, few general guidelines can be given. However, at locations where area or flow direction changes occur, obviously forces may be developed and the thermal-hydraulic model must be defined such that these forces may be accurately calculated. Particular attention must be paid at locations where the flow may become complicated such as at valves and tees. Another consideration, which may depend on the method of load calculation used, is convergence. In some cases, the load calculated may vary greatly depending on the time step chosen for the thermal-hydraulic analysis, even though the hydrodynamic results are stable. Forces calculated at locations where flow choking occurs may be particularly troublesome due to rapid variations of hydrodynamic variables. Tangent piping runs between adjacent elbows require special attention, particularly if they are relatively long. If the momentum balance formulation is used, "wave" or "acceleration" forces must be computed and appropriately applied. If force balance methods are used, forces developed by fluid friction tractions need to be computed and correctly applied. In summary, due to the complexity of all of the factors influencing load calculation, very great care must be exercised in the method by which the loads are calculated and in verification that the estimated loads are reasonable and as accurately computed as is possible.

5. STRUCTURAL RESPONSE ANALYSIS

Many computer codes may be used for the structural response prediction of piping systems subject to hydrodynamic transient induced force histories. Among these are SAP IV, NUPIPE II, ADINA, and ANSYS,[42-45] respectively. Utilization of most structural dynamics codes requires consideration of the following: (a) proper conditioning of input force data, (b) determination of the computational method to be used (direct integration of coupled equations of motion or modal superposition), (c) in the case of modal superposition, determination of the highest frequency to be considered, and (d) mathematical modeling considerations to ensure that forces are correctly applied and the model correctly includes boundary conditions and sufficient detail so that the highest frequencies of interest are accounted for. The following paragraphs briefly discuss some of the specific factors that must be considered in each of these areas.

5.1 Input Force Data Conditioning

Most thermal-hydraulics output histories and, hence, load histories are represented with unequal time steps. In fact, the time steps may vary from microseconds to milliseconds. Since many structural dynamics codes accept input loads defined at equal time intervals or at only a limited number of unequally defined time steps, it is often necessary to further process force histories to render them compatible with the structural response code utilized. Examples of codes which perform this function are the BLAZER codes described References 46 and 47. It is very important that the magnitude and distribution of the frequency content of the initial histories be preserved in the conditioning process. It is not sufficient to merely interpolate the initial data. A final consideration is a consequence of Shannon's sampling theorem: the maximum frequency content of the input loads and, hence, the output from the structural response (linear solution) is limited by one-half of the inverse of the input time step. Thus, if the input time steps are not sufficiently small, there is a possibility of neglecting important high frequency response.

5.2 Solution Algorithm Determination

Two methods of solution of the governing structural equations of motion are available: direct integration of the coupled equations and modal superposition method. In the case of direct integration, little more need be said at this time. Use of the modal method offers significant computational time savings for some analyses. However, as outlined in Reference 48, utilization of the modal superposition solution method for S/RV systems subject to hydrodynamic loads requires caution. One must assure that the frequencies of the modes included in the analysis envelope the frequency content of the input loads. The upper limit of significant input frequency content is normally about 100 Hz. It has also been demonstrated in Reference 48 that a pseudostatic high frequency response component must also be included in the modal superposition solution to avoid significant errors in the total response. The integration time step should be chosen, for either direct coupled equation integration or modal superposition, so that integration errors for the highest frequency of interest are acceptably small. The theoretical largest integration time step is 0.5/(highest frequency of interest). However, it is common practice to limit the time step to be equal to or less than 0.1/(highest frequency of interest). This results in acceptable integration error for the higher frequencies and correspondingly lower error for the lower frequencies.

5.3 Modeling Considerations

One of the most important considerations in modeling the structure is that boundary conditions (piping supports and associated structures in the case of piping systems) be correctly represented. Supports present the most significant nonlinearities in a piping system. If significant nonlinearities are present (gaps, hysteresis, etc.), then a more accurate nonlinear solution must be utilized or the conservatism of a linear analysis must be demonstrated. Model nodes capable of transmission of the external forces to the structure must be present at area changes, elbows, and tees. Some codes, such as NUPIPE II, allow for the definition of both structural connectivity nodes and mass point nodes. In this case, it is important that it is realized that some types of node points (structural connectivity only) do not allow for the input of external forces. Finally, the length of piping elements, specified between adjacent nodes, must be small enough to ensure that the structural response frequencies are greater than the input force significant frequency content. A common recommendation is to place the mass points no more than 1/4 wave length apart at the highest frequency of interest. This length may be computed by assuming the pipe to behave as a simple beam with a standing wave of the limiting frequency.

Little guidance or experience is available, at this time, for estimation of structural damping values to be used for dynamic response calculations. A common approach is to assume relatively low values, approximately 1% of critical, so that results are conservative. However, this practice may lead to overdesign with inherent economic penalties.

6. THERMAL TRANSIENT STRESSES

In Section 3, dealing with thermal-hydraulic analysis, it was suggested that heat transfer between enclosed fluid and piping be neglected so that the thermal-hydraulics analysis is simplified. This does not imply that thermal stresses in the piping should be neglected. Rather, the heat transfer (fluid to pipe) is decoupled from hydrodynamic calculations. A transient heat transfer and thermal stress analysis should be performed utilizing fluid temperatures obtained from the thermal-hydraulic analysis. The process of performing an analysis of this type is well known and will not be discussed herein. However two aspects of this type of analysis may significantly effect resultant thermal stresses and are briefly discussed.

The heat transfer coefficient between fluid and piping (film coefficient) is difficult to estimate for flow conditions as complex as exist during an S/RV transient. However, the value of this coefficient may significantly influence temperature and resulting stress gradients through the pipe wall. Thus, care must be exercised in utilization of fluid parameters for an accurate estimation of this coefficient.

In situations where flow stratification is possible resulting in variation of temperature with respect to the circumference of the pipe, two-dimensional heat transfer and thermal stress analyses may be required. In addition, potential pipe bending thermal distortions should be accounted for.

7. RESULTS EVALUATION

Two assumptions are made for the purpose of this section discussion: the S/RV analysis is a portion of a design analysis (rather than an experimental study) and that the S/RV transient is specified as a service condition. Thus, all resulting mechanical bending moments and thermal stresses must satisfy the requirements of the ASME Code.[3]

NUREG-0484, Rev. 1[3] provides the basis for the method by which S/RV transient induced mechanical loads are to be combined with all other design mechanical loads. The conclusion of Reference 49 states:

> "The staff considers the use of SRSS (square root sum of squares) appropriate for: (i) Combination of SSE and LOCA ----- (ii) Combining responses of dynamic loads other than LOCA and SSE provided a non-exceedence probability (NEP) of 84% or higher is achieved for the combined SRSS response. An acceptable method for achieving that goal is outlined in Section 4, Condition A and Condition B, paragraphs (i), (ii), and (iii)."

Thus, it is clear that, if the requirements of Reference 49 are met, mechanical loads resulting from S/RV transients may be combined with other design mechanical loads on an SRSS basis for ASME Code evaluation purposes. Failing this, they must be combined on an absolute basis.

Performance of an ASME Code fatigue evaluation including S/RV transient induced stresses is not as clear. It is believed that a fatigue evaluation should be conducted. The number of expected S/RV transient occurrences should be specified in the Design Specifications. However, little guidance is available, at this time, for determination of the number of effective stress cycles that should be specified for a given S/RV transient.

As a final comment pertaining to results evaluation, the potential influence of piping elongation is noted.[50] Even though the ASME Code requires that only design mechanical bending moments rather than axial design mechanical forces be used for primary and secondary piping stress intensity evaluation, the axial extension of piping segments due to hydrodynamic loads must be considered. The reason for this is that, especially for long straight bounded pipe segments, hydrodynamic load induced elongation of these segments induces bending moments which may not be negligible.

8. SUMMARY AND RECOMMENDATIONS

A brief summary of recent experimental and analytical studies relating to S/RV system thermal-hydraulic and structural response has been given. A relatively large number of experimental results are available with relatively few corresponding analytical comparisons. The comparisons that have been cited are, generally, not in "good" agreement with tests. It is recommended that a comprehensive evaluation of these comparisons be undertaken so that a unified and more quantitative understanding of the ability to adequately perform S/RV system analyses is obtained.

Summaries of those factors which are believed to be important for accurate S/RV system analyses have been presented. It has been shown that a complete S/RV system analysis is a complex multidisciplinary process involving several distinct analysis and evaluation steps. Perhaps the weakest link in the analysis chain is the utilization of hydrodynamic results for the prediction of mechanical load histories for subsequent input to structural dynamic response analysis. An additional recommendation is that a detailed evaluation of the load determination process be undertaken in conjunction with the additional experimental vs. analytical comparison study.

Finally, ASME Code evaluation of S/RV system transient results is required for safety evaluation. Here, the requirement for additional study is primarily in fatigue evaluation. Evaluation of S/RV system transient test and analysis results is required for the determination of a realistic number of stress cycles per transient that should be included in ASME Code fatigue evaluations.

In summary, analytical tools are available for accurately predicting and evaluating results of S/RV system transients. However, more work is required to learn how to effectively utilize those tools for realistic and effective analysis and evaluation of these systems.

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Complete thermal-hydraulic and structural dynamic response analysis of piping systems subjected to a safety or relief valve opening transient is a complex multi-step process. This presentation contains a review of recent experimental vs. analytical studies, summaries of the individual analysis steps, and guidelines for performance of these analyses. In addition, recommendations for further experimental and analytical study are given.

NOTE: This report was prepared for the Task Group on Other Dynamic Loads and Load Combinations, J. A. O'Brien, Chairman of the USNRC Piping Review Committee, R. H. Vollmer and L. C. Shao, Co-chairmen.

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Position Paper

Vibration Load
Considered as a Design Basis
for Nuclear Power Plant Piping

by: J.D. Stevenson

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1.0 STATEMENT OF THE ISSUES

All nuclear power plant piping are subject to vibratory motion where such motion is defined as(1) "a periodic motion of the particles of an elastic body or media in alternately opposite directions from the position of equilibrium when that equilibrium has been disturbed". All elastic bodies experience some level of vibration as an ambient condition. In passive components such as nuclear power plant piping, vibration is a safety concern and there is a potential for failure only when the endurance limit[1] of the material is exceeded. For stress levels above this value the number of cycles becomes a needed parameter to determine construction adequacy.

There are three categories of vibration in piping systems, 1) high stress and low cycles from transient operation and seismic considerations, 2) high stress and low cycles resulting from accident or environment induced blast or jet impulse and missile impact and 3) low stress and high cycles from steady state operation (piping attached to reciprocating or rotary equipment or flow induced).

At the present time in the first category two types of vibratory loads are usually considered as a design basis, earthquake and thermal operating cycles. In some cases vibratory loads resulting from valve operation have also been considered. For example, in ASME BPVC Section III Class 1 piping(2) a fatigue (vibration) analysis is required as defined in NB3653 whenever changes in mechanical or thermal loads occur. Vibratory loads caused by BWR suppression pool hydrodynamic loadings resulting from safety relief discharge are also in this category. These loads or load effects generally generate relatively high stresses and a relatively few (less than 5,000) number of cycles in the piping during the operating life of the facility.

The second category of vibratory loads is in response to accident or extreme environmental conditions. Major structures will generally vibrate at relative high frequency typically above 30 Hz in response to large impulse or impact loads. Such loads are typically found as the

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[1] The maximum stress that can be reversed an indefinitely large number of times without producing fracture of a material. For engineering purposes in steels this value typically corresponds to the stress level at 10^6 to 10^9 structural cycles without fracture with a suitable safety margin, say two on stress and 20 on cycles as applied to small, polished specimens.

result of a LOCA or other major high energy system rupture or as the result of a missile impact, (for example, aircraft, turbine or tornado) or external and internal blast or fluid oscillations. These building structure vibrations are passed on to supported equipment and piping as vibratory support effects in much the same way seismic effects have been passed on to the piping. However, because of the high frequency actual support displacements are much smaller than would be found for earthquakes for the same acceleration level of cyclic excitation. In the past this second category of vibratory loads as to its effect on the design adequacy of piping in the U.S. generally has not been considered explicitly except in response to suppression pool hydrodynamic loading in BWR plants resulting from a postulated DBA.

The third category of vibratory loads or stress in piping consists of reciprocating or rotary equipment and flow induced vibration and are not typically considered analytically as a design basis. This is not to say they are not considered in the qualification of nuclear power plant piping. Such qualification is currently performed as part of the preoperation and hot functional testing prior to plant start-up.

It is the purpose of this position paper to discuss and recommend changes in qualification for nuclear power plant piping for vibratory loads identified in Categories 1, 2 and 3 consistent with current and anticipated knowledge regarding such loads.

2.0 DISCUSSION OF ISSUES

2.1 Consideration of Operating Transients, Accident and Extreme Environment Induced Vibratory Loads on Piping

This grouping includes the first and second categories of vibratory loads for which some analytical evaluation is normally required.

For several years analyses of piping in BWR plants have included analytical evaluation of accelerations induced by suppression pool hydrodynamic loadings. These analyses have resulted in the development of limiting loads based on inertia accelerations developed from response spectrum analysis which tend to control pipe support design in the dry and wet wells of a BWR containment.

Relatively recent tests of the actuation of four pressure relief valves (60 percent power) for a BWR Mark III containment system resulted in the measurement of peak acceleration of 0.245 g in the structure for vibratory motion in the 30 to 100 Hz range.(3) Analytical prediction of these accelerations based on comparisons with measured results given in Table 2 of Reference 3 as shown in Appendix A to this paper appear in general to significantly over estimate actual measured accelerations. No recommendation was given in Reference 3 for extrapolation to a recirculation or main steam line break. However, it is suggested by ratios given in the report for one valve to two and three valve discharge that acceleration values given for four valves discharging at 60 percent power compared to 100 percent power and 7 valves operating can conservatively be increased by a factor of 1.5. For a DEGB recirculation or main steam line break accelerations might be increased by a factor of 3.0 or more. A simple comparison of peak floor acceleration values of 3 x 0.245 = 0.735 g which is well above a typical 0.4 g value for a second or third floor elevation zero period floor acceleration that might be expected for a 0.2 g ZPGA earthquake input. This result indicates that suppression pool hydrodynamic loads may control design of piping in the BWR containment wet and dry wells as shown in Figure 1 which is reproduced from Reference 16.

A similar type of situation where high frequency loading based on acceleration response spectra appears to control design is found in response of a German PWR plant facility to an aircraft impact as shown in Figure 2. Therefore, the German consideration of this loading case should be of interest.

Inertia acceleration as defined by an amplified response spectrum is a very poor measure of resultant stresses in components vibrating at relatively high frequency (greater than 30 Hz). In recognition of this the Federal Republic of Germany regulatory authorities have used 20 Hz as a guideline cut off frequency for response spectral acceleration required in design as the result of impact loading.(10) There is also increasing evidence that inertia acceleration is a poor measure of damage for low frequency (seismic) excitation in piping.

Measured velocity and displacements have been used to qualify piping subject to category 3 vibratory loads. Consideration should be given to applying these methods to analytical verification of category 1 and 2 high frequency loadings.

2.2 Consideration of Reciprocating and Rotary Equipment Operation and Flow Induced Vibration

Vibratory loads and stress resultants from steady state operation have not normally been analyzed as part of the stress analysis design verification (Design Report) prepared to qualify the design adequacy of the piping. Such phenomena are normally evaluated during plant start-up on the bases of observed or measured vibratory displacements or velocities and qualified on the basis of those test results.(4) Because of the complexities involved and the high uncertainty and potential variability of loading, it is not considered likely that analysis techniques will be used to qualify category 3 vibratory loads.

Since 1981 the U.S. NRC in the form of SRP 3.9.2(11) has had detailed requirements for vibration testing of nuclear power plant safety related piping. Since 1982 the ASME has had available a detailed standard intended to be used in preoperational and initial start-up vibration testing qualification of nuclear power plant piping systems.(4)

Experience has shown that acceleration independent of frequency is not a good measure of damage and has not even been considered in the preparation of the ASME Standard on Vibration Testing of Nuclear Power Plant Systems.[4] Recent research performed in Canada has made a strong case for the use of vibration velocity as a general and more universally applicable measure of damage or failure potential in vibratory systems.[5] However, velocity used as a preferred criteria for judging the damage potential from vibration has not received general acceptance in the industry[6,7] and there are at least two references which have shown possible instances of unconservative acceptance criteria being calculated using the velocity method.[8,9] As a result of this concern regarding conservatism it has been proposed that the next revision of Reference 4 have a frequency dependent correction factor to the velocity method.

Current industry practice would permit the use of either measured velocity or displacement as a means for qualifying piping subject to high frequency vibration. Vibratory displacement tends to be the easiest to measure in the field while velocity tends to give a more accurate measure of stress resultant over a wide range of system geometries.

2.3 Consideration of Changes in Design and Analysis Procedures in Seismic Design

There is a growing awareness that the current procedures used in seismic design and analysis which are based primarily on inertia accelerations being used to define resultant stress are not consistent with observed behavior.[12,13,14,15] However, eventhough seismic is a vibratory load its detailed consideration is outside the scope of this paper.

3.0 PROPOSED RECOMMENDATIONS

3.1 General

(1) The first category of vibratory load as identified in Section 1.0 of this paper, operating transient induced vibratory loads other than thermal, when identified in the design specification are typically high frequency (greater than 30 Hz) in nature. These loads are typically identified for design purposes by acceleration response spectra. As discussed in Section 2.0, such loading definitions appear to over estimate zero period acceleration when compared to experimental results even when such response is in the linear elastic range. In addition the high frequency nature of this load when characterized by the acceleration parameter tends to greatly over estimate the damage potential of this loading. For these reasons it is recommended that consideration be made to a frequency cut off used to define the acceleration based inertia design load similar to that used in the Federal Republic of Germany for aircraft impact effects when acceleration response spectra are being used as loading input.

(2) It is recommended that the second category of vibratory loads as identified in Section 1.0 of this paper resulting from response to accident or extreme environmental loads consider limitation on acceleration based inertia loads in the same manner as described in (1) above. In addition response in this case both for the structure transmitting the load and the piping will typically be into the inelastic range. Evaluation of this nonlinearity should be permitted in the analysis for this category of loading. It is recommended that non-linear methods be permitted in the analysis for this category of loads. An evaluation of various methods proposed to consider nonlinear response to dynamic loads should be performed to evaluate adequacy.[11,17,18]

(3) The present method used to qualify the third category of vibratory loads (machinery and flow induced) namely preoperational and start-up testing should continue to be the primary method of qualifying piping systems for such loadings. However, explicit applications to all high energy and category I seiemic should be limited to systems which historically have exhibited significant vibratory motion. An evaluation should be performed to identify such systems and operating conditions.

(4) For the first and second category of vibratory loads qualified by analysis, it is recommended that displacement and velocity based acceptance criteria used in testing for category 3 vibration loads[4,5,8,9] be evaluated for applicability.

3.2 Specific Recommendations

It is recommended that changes as indicated herein be made to SRP 3.9.2. These changes would require the explicit consideration of dynamic operational, environmental and accident loads on building or support structures that result in significant response vibration loads in supported piping systems. However, they would also permit explicit consideration of high frequency low damage characteristics of these loads and when appropriate nonlinear response characteristics to such loadings.

(1) Reference I. Areas of Review 1.

In the 11th line of the following words should be added:

> ...withstand flow-induced <u>and reciprocating and rotating equipment</u> dynamic loadings...

(2) Reference I. Areas of Review

Add a new item 7 on pg. 3.9.2-4 whose text is as follows:

7. A discussion should be provided which describes methods to be used to evaluate equipment and piping system to confirm their structural design adequacy when subjected to transient, accident and extreme environment (other than seismic) vibratory loads. Such vibratory loads typically result from response of equipment and piping system supporting structures when such support structures are subjected to significant impact or impulse loads.

(3) Reference II. Acceptance Criteria 1.

Rewrite Section 1 as follows:

1. Relevant requirements of GDC 1, 2, 4, 14, and 15 are met if vibration, thermal expansion, and dynamic effects testing are conducted during start-up functional testing for specified high-and moderate-energy piping, and their supports and restraints. The purpose of these tests is to confirm that the piping components, restraints, and supports have been designed to withstand the dynamic loadings and operational transient conditions that will be encountered during service as required by the Code and to confirm that no unacceptable restraint of normal thermal motion occurs. Results of vibrational tests may also be used directly or by interpolation to confirm design adequacy of high-and moderate-energy piping, components, restraints and supports to accident and extreme environmental loads.

   An acceptable test program to confirm the adequacy of the designs should consist of the following:

   a. A list of systems that will be monitored. This list may be limited to those systems based on experience which undergo significant thermal expansion, vibration and dynamic effects.

   b. A listing of the different flow modes of operation and transients such as pump trips, valve closures, etc. to which the components will be subjected during the test. (For additional guidance see Reference 8). For example, the transients associated with the reactor coolant system heat up tests should include, but not necessarily be limited to:

      (1) Reactor coolant pump start.

      (2) Reactor coolant pump trip.

      (3) Operation of pressure-relieving valves.

      (4) Closure of a turbine stop valve.

   c. A list of selected locations in the piping system at which visual inspections and measurements (as needed) will be performed during the tests. For each of these selected locations, the deflection (peak-to-peak) <u>maximum velocity</u> or other appropriate criteria, to be used to show that the stress and fatigue limits are within the design levels, should be provided.

d. A list of snubbers on systems which experience sufficient thermal movement to measure snubber travel from cold to hot position.

e. A description of the thermal motion monitoring program, that is, verification of snubber movement, adequate clearances and gaps including acceptance criteria and how motion will be measured.

f. If vibration is noted beyond the acceptance levels set by the criteria of c., above, corrective restraints should be designed, incorporated in the piping system analysis, and installed. If, during the test, piping system restraints are determined to be inadequate or are damaged, corrective restraints should be installed and another test should be performed to determine that the vibrations have been reduced to an acceptable level. If no snubber piston travel is measured at those stations indicated in d., above, a description should be provided of the corrective action to be taken to assure that the snubber is operable.

(4) Reference II. Acceptance Criteria 2.

Add the following new paragraph as the last paragraph of II.5. pg. 3.9, 2-15.

<u>High frequency (greater than 30 Hz) vibratory loads, other than seismic, analyses methods for all Category I systems, components equipment and their supports (including supports for conduit and cable trays, and ventilation ducts are reviewed. In addition, other significant effects that are accounted for in the high frequency vibratory load analysis such as nonlinear response and plastic stress levels in the materials are reviewed</u>.

(5) Reference III. Review Procedures 1.

Rewrite Section 1 as indicated.

1. During the CP stage, the PSAR is reviewed to assure that the applicant has provided a commitment to conduct a piping steady-state vibration, thermal expansion and operational transient test program. <u>The applicant may also commit a simulated accident or natural phenomena vibration test program in lieu of analysis</u>.

(6) Reference IV. Evaluation Findings 2.

In the fifth line add the words "or test" after analysis.

(7) Reference IV. Evaluation Findings 4.

In the sixth line add the words "or test" after analysis.

4.0 REGULATORY VALUE/IMPACT

4.1 Consideration of Operating Transient, Accident and Extreme Environment Induced Vibratory Loads on Piping

Design procedures which consider the different effect of high frequency vibratory excitation induced from operating loads as compared to low frequency seismic loads should be permitted. In addition nonlinear response of the building structure and piping for high frequency vibratory excitation induced from acccident and extreme environment should also be permitted. It is my opinion such consideration will result in these loads no longer controlling the design of piping supports in BWR Containment dry ard wet wells. The net effect is estimated to be elimination of approximately 100 snubbers on BWR piping per plant. Assuming an average installed hardware cost of $4,000.00 per snubber this would result in a direct cost saving of $400,000.00 per BWR at initial construction plus an addition of $80,000.00 per year in maintenance and inservice inspection costs. Assuming another 20,000 engineering manhours is used to evaluate this governing load case and support design for BWR dry and wet well piping per plant there would be a further reduction of $1,000,000.00 in direct engineering costs.

4.2 Consideration of Reciprocating and Rotary Equipment Operation and Flow Induced Vibration

It is estimated that pre-operational and hot functional vibration monitoring of piping systems is taking approximately 25,000 manhours per plant during start up. If, based on a review of past experience, the number of lines required to be monitored were reduced by 50 percent, a net direct cost savings of 12,500 X $40.00/hr. or $500,000.00 per plant would be possible.

5.0 REFERENCES

(1) Websters Collegiate Dictionary, CC. Merriam Co., 1980.

(2) ASME Boiler and Pressure Vessel Code Section III Subsection NB Class 1, Nuclear Components, American Society of Mechanical Engineers, 1983.

(3) Miller, G.K. and Yuan, C.Y., "Acceleration Data in the Suppression Pool Area from Kuosheng SRV Tests," EGG-EA-6025, dtd. October 1982.

(4) ANSI/ASME OM3-1982, "Requirements for Preoperational and Initial Start-up Vibration Testing of Nuclear Power Plant Piping Systems, American Society of Mechanical Engineers, 1982.

(5) Hartlen, R.T., Elmaraghy, R., and Slingerland, F., "Vibration Velocity As A General Severity Criterion," Presented at the CEA Thermal and Nuclear Power Section Spring Meeting, Montreal, 16 March 1982.

(6) Esselman, T.C. "Westinghouse Review and Comments on Position Papers - Other Dynamic Loads and Load Combinations", Westinghouse Electric Co., 21 February 1984.

(7) Landers, D.F. "Review of Position Paper - Vibrations Load Considered as a Design Basis for Nuclear Power Plant Piping" Letter to J. O'Brien, USNRC, Teledyne Engineering Services, 21 March 1984.

(8) Olson, D.E. and J.L. Smetters, "Conservatism Inherent in Simplified Qualification Techniques Used for Piping Steady State Vibration" - 7th International Conference on Structural Mechanics in Reactor Technology, 1983, Chicago, Ill.

(9) Stone, King, J.E. and R.C. Kryter, "Screening Procedures for Vibrational Qualification of Nuclear Plant Piping", ASME Paper 80-C2/PVP-4.

(10) Stevenson, J.D., "Review of High Frequency Excitation of Equipment, Design Requirements", Trip Report J.D. Stevenson Visit to TUV Rheinland and GRS, 19 May 1983.

(11) SRP 3.9.2, "Dynamic Testing and Analysis of Systems, Components and Equipment, NUREG-0800, July 1981.

(12) Broman, R., et.al., "Conceptual Task to Develop Revised Dynamic Code Criteria for Piping," EDS Nuclear, Prepared for the Electric Power Research Institute, May 1983.

(13) Letter Communication between J.D. Stevenson and M. Beeman, dtd. 1 December 1983.

(14) Rodabaugh, E. C., "Position Paper on Stress/Dynamic Stress Allowable for Piping", Prepared for U.S. Nuclear Regulatory Commission, February 1984.

(15) Stevenson, J.D., "Survey of Above Ground Piping Failures in Historical Earthquakes", sponsored by Lawrence Livermore Laboratory, (In Preparation - report expected 15 May 1984).

(16) Stevenson, J.D., "Cost and Safety Margin Assessment of the Effects of Design for Combination of Large LOCA and SSE Loads", UCRL-15340, Lawrence Livermore Laboratory, October 1980.

(17) Campbell, R.D., Kennedy, R.P., and Thrasher, R.D., "Development of Dynamic Stress Criteria for Design of Nuclear Piping Systems," SMA 17401.01, Structural Mechanics Associates, November 1982.

(18) Stevenson, J.D., "Proposal on Assessment of Design Requirements Additional to Dynamic Stress Criteria for Piping, Stevenson & Associates, April 1983.

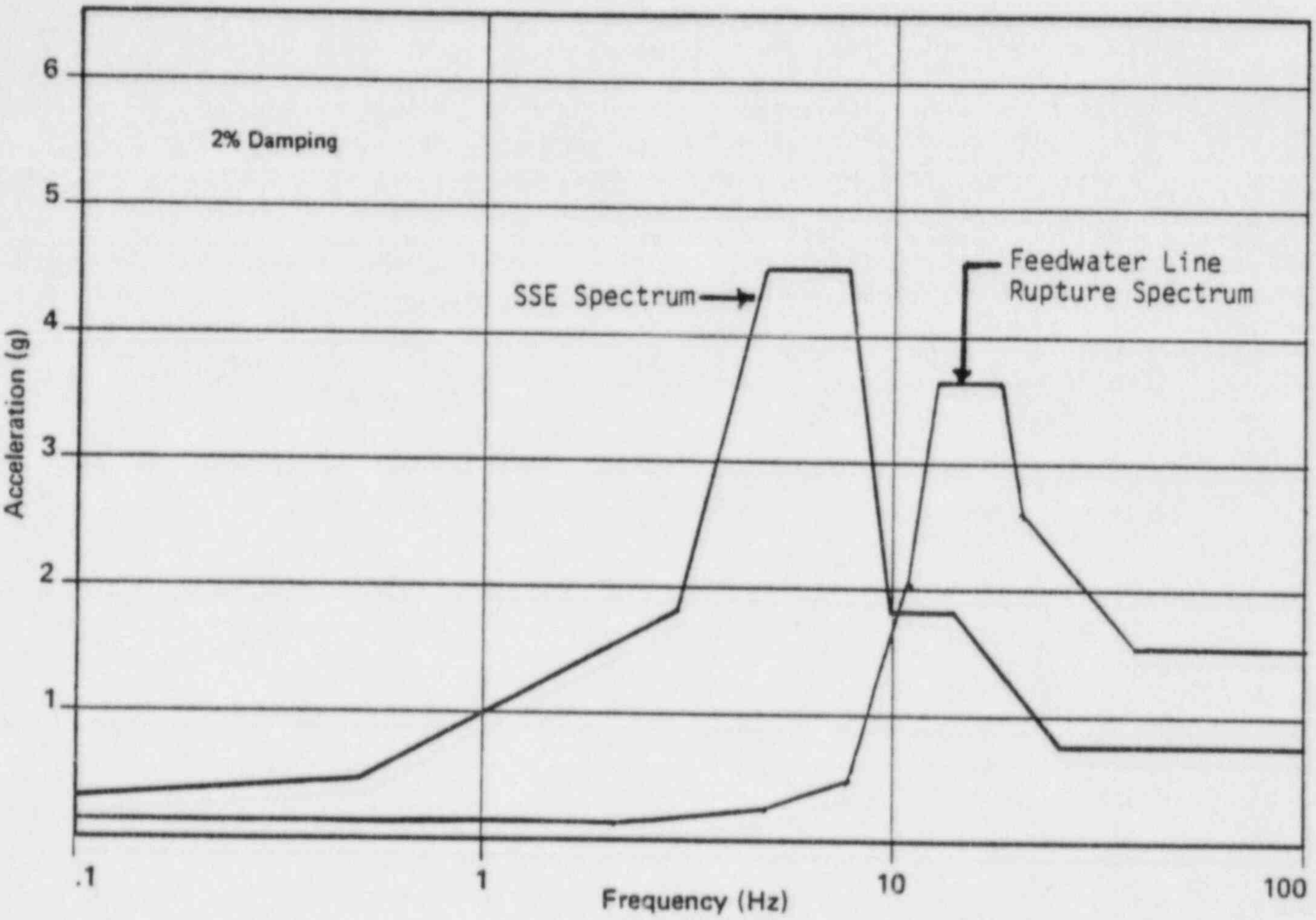

Figure 1. Typical SSE acceleration floor and feedwater line rupture acceleration response spectra for design of recirculation line.

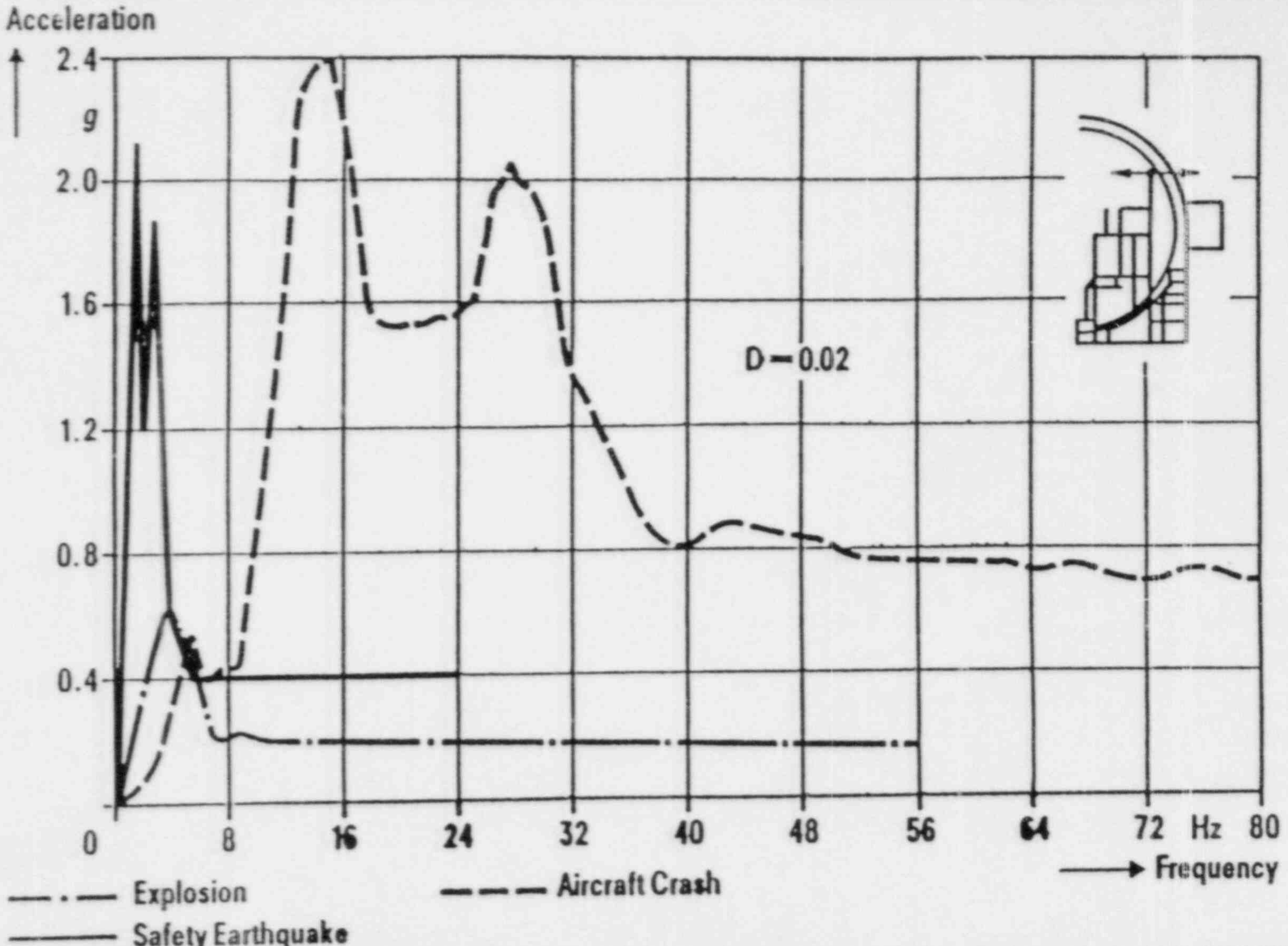

Figure 2 Comparison of Response Spectra Due to External Dynamic Loads
PWR Reactor Building/Upper Interior Cylinder, Radial

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TECHNICAL LETTER REPORT

ACCELERATION DATA IN THE SUPPRESSION POOL AREA FROM KUOSHENG SRV TESTS

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ABSTRACT

Acceleration data from the suppression pool area of a General Electric Boiling Water Reactor with Mark III containment system (BWR6/Mark III) are presented and studied. The acceleration measurements were obtained from the safety relief valve (SRV) discharge tests conducted at the Kuosheng Nuclear Power Station in Taiwan. The data included plots of acceleration time histories, the power spectral densities and the peak values of the accelerations. Comments on the data and recommendations for their use are offered. These data were requested by the USNRC for the purpose of characterizing the dynamic responses of the containment structure and the equipment inside caused by the hydrodynamic excitations associated with the SRV actuations.

SUMMARY

Acceleration data recorded during a safety relief valve (SRV) discharge test was studied to characterize the structural response in the suppression pool area. The tests were performed in August 1981 at the Kuosheng Nuclear Power Station in Taiwan by the Taiwan Power Company. EG&G Idaho was requested by the USNRC to provide this information for evaluation of equipment qualification in similar nuclear plants. The acceleration histories from accelerometers located in the suppression pool area were plotted. From these histories, peak acceleration amplitudes were obtained. In addition, power spectral densities were generated for all histories to determine their frequency content. Finally, the number of significant cycles in the acceleration histories were estimated.

The data examined in this study contained only two seconds of recorded history beyond valve opening. This is enough time to include response due to the SRV discharge loading but probably not all of the subsequent hydrodynamic loading. Thus, evaluations made in this study pertain primarily to the discharge loading, which is apparently more significant than any subsequent loading that may occur.

The data revealed that the magnitude of peak acceleration values on the walls above the suppression pool were relatively low, but the frequency range of the vibration motion extends well beyond that used for seismic analysis. Also a low number of significant cycles of motion per actuation were counted from all the acceleration time histories. The hydrodynamic loads due to SRV discharge must be combined with other dynamic loads in qualifying the equipment. Suggestions are also made regarding the method of load combination.

ACCELERATION DATA IN THE SUPPRESSION POOL AREA FROM KUOSHENG SRV TESTS

1. INTRODUCTION

This report presents results of a study performed by EG&G Idaho, Inc. on acceleration data obtained from the suppression pool and the Hydraulic Control Unit (HCU) areas during the safety relief valve (SRV) discharge tests at the Kuosheng Nuclear Power Station in Taiwan during August 1981. The HCU area is located just above the suppression pool. The purpose of the study was to characterize the structural response in the suppression pool area caused by SRV discharge loading in support of the Nuclear Regulatory Commission's equipment qualification program for BWR-6 Mark III plants. To qualify equipment for loading associated with the SRV discharge, the equipment support motion must be defined. The response of the containment structure in the suppression pool area is the support motion for equipment contained therein. Thus, recorded acceleration response from the suppression pool area is useful for qualification of equipment in this area for loading due to SRV actuation.

In qualifying equipment for seismic loading, the frequency content and amplitude of the seismic input motion are of particular interest. Similarly, frequency content and amplitude of the input accelerations are of interest in qualifying to SRV discharge loads. Thus, this report presents plots of recorded accelerations, maximum acceleration amplitudes, and frequency content of the acceleration histories. Since hydrodynamic loading typically imposes many more stress cycles on equipment than seismic loading, fatigue effects on equipment need consideration. Information on the number of cycles of motion encountered in the suppression pool area due to discharge loading, therefore, is also presented.

The data studied consists of acceleration readings from 17 accelerometers located on the containment and drywell walls and 7 accelerometers situated on pieces of equipment. The instrumented pieces of equipment were the jet pump control panel and the 3-inch power operated valve located on the HCU floor and the suppression pool drywell wall, respectively. Readings from these pieces of equipment give information on actual equipment response in the suppression pool area. According to Nutech International, the test contractor, the acceleration measurements inside the suppression pool are unreliable due to the high frequency and high amplitude acceleration of the pool liner.[1] Additionally, an accelerometer on the containment wall in the HCU area was faulty. Thus, readings from these accelerometers (A1 through A10, and A19) were not considered in this evaluation.

The Kuosheng SRV tests form the first such test program conducted on a BWR-6/Mark III reactor. The test results will be useful for considering similar type of plants in the future. The tests were performed while the reactor was operating at 60% power. Therefore, the acceleration data could need some adjustment in order to correspond with a 100% power condition.

Operation at full power is expected to increase peak pressures in the SRV piping system during discharge by 17% to 33%, depending on the nature of the valve actuation.[3] Higher pressures would probably increase the magnitude of response in the suppression pool area, but determination of the amount of increase would require further study.

2. DESCRIPTION OF INVESTIGATION

Accelerometers on the containment and drywell walls considered in this study were concentrated in the suppression pool areas near the SRV4 and SRV8 discharge lines. Accelerometers on the equipment were situated in the vicinities of SRV2 and SRV6. Table 1 and Figures 1 and 2 describe the locations and orientations of all accelerometers considered.

Accelerometer data evaluated were those recorded during the test MT-81, which was a simultaneous actuation of valves V4, V8, V11, and V16. Discharge loading for this test was more severe than the loadings for other discharge tests, yielding accelerations of the greatest amplitude. Frequency content of accelerometer readings among tests should be reasonably consistent.

Acceleration histories from all the accelerometers are plotted in Figures 3 through 26. Peak acceleration values from these histories are presented in Table 2. Also shown in the table are the predicted peak values as determined by analyses performed by the Bechtel Power Corporation. As shown in the table, there is only one exceedance among the measured responses.

The number of significant cycles of acceleration that occur during the discharge period at each accelerometer location are presented in Table 3. Significant cycles for any acceleration history were assumed to be those having an amplitude of at least 25% of the peak magnitude. Though the data recorded on magnetic tapes by Nutech extended for only two seconds beyond valve opening, the cycle estimates given in Table 3 correspond to a discharge time of approximately five seconds. The estimates were made by assuming that the number of cycles occurring between one and two seconds after discharge would be repeated for the next three seconds.

TABLE 1. LIST OF ACCELEROMETERS

1. At azimuth 307° to 344°

Accelerometer No.	Orientation[a]	Elevation	Location
A19[b]	V	17'-0"	Containment Wall
A20	R	17'-0"	Containment Wall
A21	T	17'-0"	Containment Wall
A43	V	-14'-0"	Drywell Wall
A44	R	-14'-0"	Drywell Wall
A53	V	16'-3"	Drywell Wall
A54	R	16'-3"	Drywell Wall
A55	T	16'-3"	Drywell Wall
A110	R	10'-0"	HRU Floor
A111	T	10'-0"	HCU Floor
A112	R	3'-0"	HCU Floor
A113	T	3'-0"	HCU Floor

2. At azimuth 198° to 254°

Accelerometer No.	Orientation[a]	Elevation	Location
A11	V	-14'-10"	Containment Wall
A12	R	-14'-10"	Containment Wall
A22	V	17'-4"	Containment Wall
A23	R	17'-4"	Containment Wall
A24	T	17'-4"	Containment Wall
A45	V	-13'-1½"	Drywell Wall
A46	R	-13'-1½"	Drywell Wall
A56	V	19'-10½"	Drywell Wall
A57	R	19'-10½"	Drywell Wall
A58	T	19'-10½"	Drywell Wall
A122	V	- 2'-6"	Close to Drywell
A123	R	- 2'-6"	Close to Drywell
A124	T	- 2'-6"	Close to Drywell

[a] V = Vertical
R = Radial
T = Tangential

[b] Acceleration readings from A19 are unreliable.

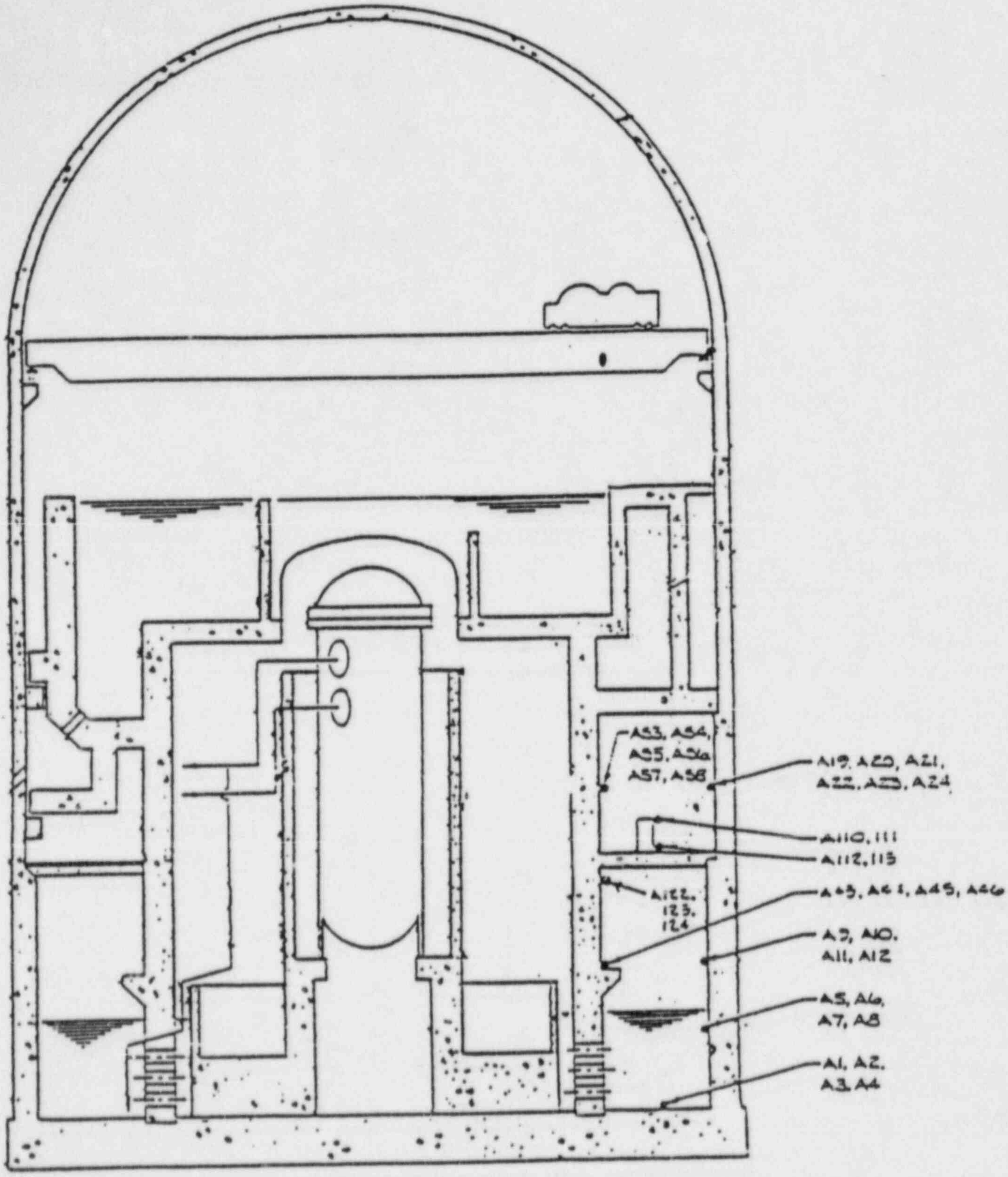

Figure 1. Location of accelerometers--elevation.

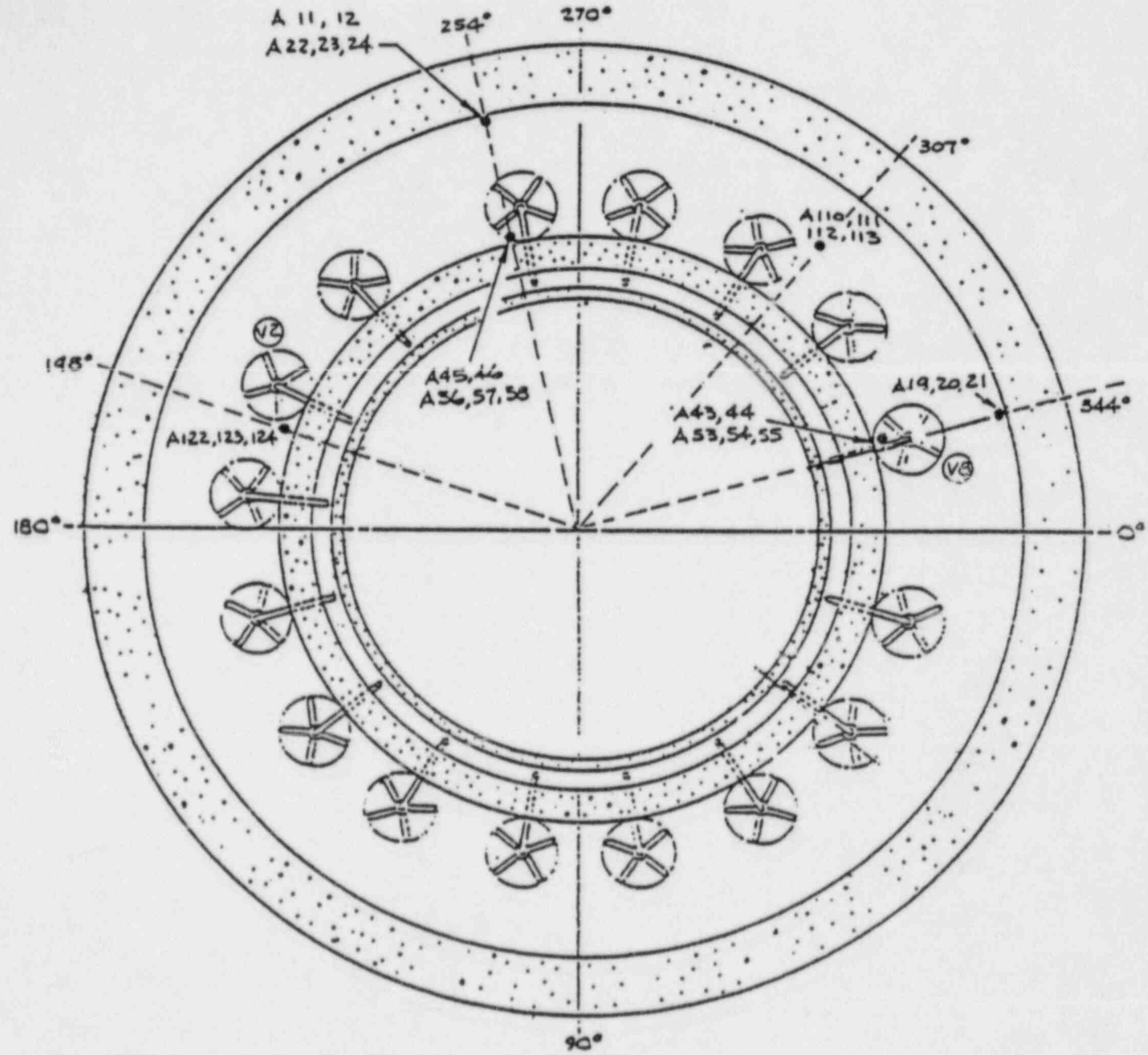

Figure 2. Location of accelerometers--plan view.

TABLE 2. PEAK ACCELERATION VALUES (TEST NO. MT-81)

Accelerometer No.	Expected Response[a] (g)	Test Value (g)	Exceedance[b] (g)
A11	0.103	.0361	
A12	0.154	.1147	
A20	0.087	.0730	
A21	0.087	.0273	
A22	0.114	.0494	
A23	0.087	.1078	.0208
A24	0.087	.0351	
A43	0.124	.0691	
A44	0.482	.2446	
A45	0.124	.0698	
A46	0.482	.1467	
A53	0.137	.0394	
A54	0.209	.0650	
A55	0.209	.0217	
A56	0.137	.0301	
A57	0.209	.0918	
A58	0.209	.0158	
A110	0.178	.1196	
A111	0.178	.1476	
A112	0.178	.0588	
A113	0.178	.0763	
A122	6.0	.1945	
A123	6.0	.1359	
A124	6.0	.2017	

a. Prediction based on Automatic Depressurization System (ADS) actuation involving seven ADS valves. No calculated data for a condition matching the above test is available.

b. Exceedance = Test Value - Expected Response.

TABLE 3. ESTIMATED NUMBER OF SIGNIFICANT CYCLES OF ACCELERATION (TEST NO. MT-81)

Accelerometer No.	Estimated Cycles of Motion
A11	7
A12	7
A20	24
A21	37
A22	8
A23	10
A24	11
A43	7
A44	27
A45	9
A46	8
A53	21
A54	48
A55	48
A56	31
A57	8
A58	75
A110	27
A111	97
A112	39
A113	67
A122	6
A123	49
A124	40

To obtain the frequency content of the accelerometer data, power spectral densities (PSD's) were generated for all acceleration histories. The PSD's indicate predominant frequencies contained in the data and are presented in figures 27 through 50. The range of significant frequencies contained in the histories are listed in Table 4.

TABLE 4. RANGE OF SIGNIFICANT FREQUENCIES (TEST NO. MT-81)

Accelerometer No.	Frequencies (Hz) High	Frequencies (Hz) Low
A11	70	30
A12	50	30
A20	85	30
A21	85	30
A22	70	30
A23	70	30
A24	70	30
A43	110	25
A44	100	30
A45	100	15
A46	90	15
A53	95	25
A54	95	30
A55	95	30
A56	130	30
A57	90	30
A58	100	30
A110	105	30
A111	100	20
A112	100	30
A113	95	40
A122	75	30
A123	30	20
A124	50	30

3. COMMENTARY

Because Nutech was primarily interested in initial dynamic transient response in evaluating the accelerometer data, the data contained on their magnetic tapes covered only the first two seconds of recorded response beyond the time of valve opening. Thus, observations on peak acceleration values and frequency content contained in this report pertain to hydrodynamic loading that occurs during this two-second time period. The number of significant stress cycles, though, was extrapolated over a discharge period of five seconds from the number of cycles occurring during the recorded time. The number of cycles estimated was extrapolated beyond two seconds because it appears that the recorded histories for several accelerometers would contain more significant cycles were the recorded times extended.

An examination of the acceleration histories reveals that the most significant response occurs within the first second of recorded data. This response can be attributed to the safety relief valve (SRV) discharge loading. Subsequent hydrodynamic loads contribute toward some lower magnitude acceleration response during the one to two second period after valve opening, but their effects beyond that time cannot readily be assessed from the available data. These loads, however, would not likely be as severe as the primary SRV loading at any of the accelerometer locations.

The highest acceleration value (0.245 g) occurred in the radial direction on the drywell wall of the suppression pool at a height of 26 feet from the basemat. The accelerations were generally of low magnitude and in only one case (containment wall, radial direction) exceeded Bechtel's predicted response. The above data was obtained from test MT-81, a four valve actuation condition involving V4, V8, V11 and V16. Based on the limited number of tests examined, the maximum acceleration readings in the vertical and radial directions for the four valve discharge test were about three times as large as for the one valve discharge cases. The two (adjacent) valve discharge responses were about 40% higher than the one valve discharge cases.

A review of the frequency content of the acceleration histories reveals that motion due to the hydrodynamic loading contains frequencies of up to 100 Hz and beyond. The highest frequencies seem to occur at the drywell wall and the Hydraulic Control Unit floor. The accelerations generally do not have much frequency content below 20 Hz. The predominant frequencies are in the range of 30 to 100 Hertz.

The number of cycles reported was based on the assumption that the incidence of significant cycles between one and two seconds would continue at the same rate for the remainder of the valve opening period. How many additional cycles may occur beyond that time could not be estimated from available data. Also in determining the cycling from the time history record, the significant cycles were assumed to be those having an amplitude of at least 25% of the peak value. The cycle count will be different if this criterion is changed or a longer record is available. In the area immediately above the suppression pool, the significant cycles are generally below ten. In the HCU area and on the equipment, the average number of cycles are between thirty and fifty.

4. RECOMMENDATIONS

Equipment in the suppression pool area of BWR-6 plants must be qualified for hydrodynamic loads due to safety relief valve actuations along with other loads such as seismic, LOCA and operating. Qualification to the hydrodynamic loads requires that a time history or response spectrum representing motion caused by the loading be generated for the equipment support location. The history or spectrum can be prepared either by analysis performed on the containment facility or from discharge tests such as those conducted at the Kuosheng plant. The information contained in this report concerning acceleration amplitudes and frequency content can be used to evaluate validity of these histories or spectra. This acceleration input will, of course, be affected by dimensions such as wall thicknesses of the containment structure and the plant operating power levels.

When analysis is performed to develop the input motion to equipment, it is necessary to calculate the containment response to the hydrodynamic loads. The structural model of the containment should be verified to ensure adequacy of the structural representation. The model should properly respond to the hydrodynamic loading in all directions so that calculated response of the containment will accurately define the hydrodynamic loading that must be sustained by attached equipment.

If the equipment is qualified by analysis alone, adequacy of any structural model used to perform the analysis must also be well verified. In addition, it must be demonstrated that structural integrity of the equipment is enough to guarantee its operability during and after the hydrodynamic loading. Otherwise, some testing would be required to demonstrate operability of the equipment during and after the loads. If it is impracticable to fulfill all the qualification requirements by testing or analysis alone, a combination of the two qualification methods is recommended.

In qualifications of the equipment, the hydrodynamic loads must be combined with other dynamic loads, such as seismic. Unfortunately, a defined time-phase relationship among loads frequently does not exist so that a straightforward addition of the equipment's responses to individual input time histories for each load would not be possible. Thus, other load combination techniques must be sought. A logical approach to combining dynamic loads is to combine the response spectra representing the individual dynamic excitations so as to form a combined spectrum to which the equipment can be tested or analyzed. Another approach is to calculate the response of the equipment to the individual response spectra and then combine the individual resonses. Both approaches, however, require that a suitable method for combining spectra or responses be used. No known combination method had been proven to be effective in all cases. Combining the spectra or responses by absolute sum (ABS) is often too conservative since no location in the structure is likely to incur maximum response to all of the loads simultaneously. Designing for an ABS combination can thus result in a system or structure that is too rigid to accommodate thermal expansion.[4] Use of the square-root-of-the-sum-of-the-squares (SRSS) method is more realistic but, according to studies performed by Brookhaven National Laboratory,[2] this method can often give nonconservative results when only two dynamic loads are being combined. Until future studies indicate otherwise, the combination method used in any particular situation should be justified. In the case of the SRV loading discussed in this report, most overlap in frequency content between seismic and hydrodynamic loads occurs in the 20 to 30 Hz range. Outside this range a load combination would essentially amount to only one or the other of the loads.

Fatigue effects due to significant stress fluctuations from SRV discharge loading can be accounted for in equipment qualification by assuring that the equipment sustains in a test program the number of acceleration cycles given in Table 3 multiplied by the number of valve actuations expected during plant life. Alternatively, the equipment can be analyzed for fatigue requirements of the ASME Code, Section III, for Class 1 components. In evaluating fatigue effects due to SRV discharges, operability of the equipment must be demonstrated both during and after application of all stress cycles. This may be difficult to accomplish if the equipment is qualified by analysis. The qualification of equipment to the hydrodynamic and seismic loads, including application of a sufficient number of significant stress fluctuations, should follow other forms of aging of the equipment. This verifies that the equipment will remain functional if significant dynamic events occur late in plant life.

5. REFERENCES

1. L. A. Conrad, Final Test Report, Safety Relief Valve Discharge Test, Kuosheng Nuclear Power Station, Unit No. 1, Revision A, (draft copy), Nutech International.

2. Brookhaven National Laboratory, Evaluation of the Simultaneous Action of Earthquake, LOCA and SRV on Mark-III Containment and Drywell Structures, 1981.

3. C. D. Shadinger, General Electric Company letter to M. Taylor, Jr., " Kuosheng Inplant SRV Test Amplification Factors", letter number KS-14,095, Rev. 1, May 20, 1982.

4. R. K. Mattu, Methodology for Combining Dynamic Responses, NUREG-0484, Rev. 1, May 1980.

APPENDIX C

INDUSTRY COMMENTS

(This appendix contains two sets of comments provided by industry in response to the Task Group's solicitation for review of (1) the consultants' position papers and (2) the draft report prepared by the staff that included the staff's tentative recommendations. Part I of this appendix contains comments on early drafts of the position papers. Part II contains comments on the draft report prepared by the staff at a later date. Subsequent to the receipt of these comments, both the consultant position papers and the staff recommendations were revised. Thus industry comments may not correlate well with what now appears in this final draft of the Task Group report.)

PART I

Industry Comments on Consultant Position Papers

PSE-SSD-1073

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
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February 21, 1984

Mr. P. Higgins
Reactor Licensing and Safety Projects Manager
Atomic Industrial Forum, Inc.
7101 Wisconsin Avenue
Bethesda, Maryland 20814

Dear Mr. Higgins:

SUBJECT: Westinghouse Review and Comments on Position Papers-
Other Dynamic Loads and Load Combinations

We welcome the current NRC Piping Review Committee activity. This Committee, as we know, is performing a comprehensive review of current regulatory requirements in the area of Nuclear Power Plant piping. We believe that many requirements need to be updated simply because of the rapid technology advancement and the availability of new and relevant data in the last few years.

In addition to the role of equipment manufacturer, Westinghouse, in the last decade, has gained substantial experience in the piping and support area (both in the Class 1 and BOP areas). Such experience tells us that new information and technology related to piping design allows for a fruitful evaluation of criteria and methodology. The methods used to obtain an "acceptable" design must be reviewed to assure that we have not become overconservative to the eventual detriment to the plant. We welcome the effort.

After a careful but quick review, this letter with attachments, provides Westinghouse comments and suggestions on five draft Position Papers that we received from the Atomic Industrial Forum, Inc. through Mr. S. A. Bernsen. We appreciate this opportunity to provide input to the Task Group of the Piping Review Committee, and hope to continue to support the effort to develop more appropriate regulatory guidelines and positions.

If we can provide further clarification, please contact us.

Very truly yours,

T. C. Esselman

T. C. Esselman, Manager
Engineering Mechanics

jm
cc: J. J. McInerney
J. A. O'Brien

ATTACHMENT

Westinghouse comments and suggestions are provided on the following five Position Papers:

1. "Stress Limits/Dynamic Stress Allowables for PIping", by E. C. Rodabaugh.

2. "Piping System Dynamic and Thermal Stress Response Induced by Thermal-Hydraulic Transient", by J. G. Arendts.

3. "Event Combination Associated with Dynamic Load Combinations Applicable to Nuclear Power Plant Piping", by J. D. Stevenson

4. "Vibration Loads Considered as a Design Basis for Nuclear Power Plant Piping", by J. D. Stevenson

5. "Position Paper on Response Combinations", by R. P. Kennedy.

ITEM 1 - Position Paper on "Stress Limits/Dynamic Stress Allowables for Piping", by E. C. Rodabaugh

Comments/Concerns

1. Other issues that should be addressed in developing a new stress limit/dynamic allowables are:
   - strain hardening
   - cyclic load failure

2. Realistic damping, such as recommended by the PVRC Sub-Committee should be allowed to be used in piping analysis. (This will probably be in another Position Paper).

3. Equation 9 stress indices and limits are intended for use with elastic system analysis to obtain the dynamic loads. If inelastic analysis methods are used, is Equation 9 still applicable or should the detailed methods of NB-3200 be used?

Suggestions

1. In order to avoid the complexities and expense of NB-3200 analysis methods, a new set of equations that correspond to new simplified inelastic analysis methods should be developed. These methods should include the actual failure mechanics for piping components subjected to cyclic loadings with inelastic strain.

ITEM 2 - Position Paper on "Piping Systems Dynamic and Thermal Stress Response Induced by Thermal-Hydraulic Transients" by J. G. Arendts.

Comments/Concerns

1. With respect to Item 5 statements on Page 5 and conclusion on Page 18, Westinghouse comparisons have showed good agreement between tests and calculations for both the thermal hydraulic and structural responses. This conclusion is documented in two references listed below.

2. In the system modeling paragraph on Page 7, the staggering of the time of valve operation to maximum the structural response is not necessary because of the low probability in occurrence and the overall conservatism in analytical methods used in the design.

3. New relevant test data and studies have demonstrated that 1% of critical damping required in the piping analysis is extremely conservative. A new position of using recent PVRC values should be recommended by this Position Paper. In general, however, in a short time transient analysis such as this, damping will not have a significant effect.

4. On Page 17, the effect of the axial extension of piping segments, due to hydrodynamic loads, has been considered in the calculation of bending moments in the pipe. The structural model allows for deformation in the axial direction, so that the induced bending moment is accurate. It is correctly pointed out that the ASME equations for stress calculation do not require inclusion of the axial forces.

Suggestions

We recommend inclusion of the following two papers in your references:

1. L. C. Smith and T. M. Adams, "Comparison of Analytically Determined Structural Solutions with EPRI Safety Valve Test Results", 4th National Congress on Pressure Vessel and Piping Technology, Oregon, 1983, PVP-Vol. 74.

2. L. C. Smith and K. S. Howe, "Comparison of EPRI Safety Valve Test Data with Analytically Determined Hydraulic Results", 7th International Conference on Structural Mechanics in Reactor Technology, Chicago, Ill., 1983, Vol. F, 2/6.

ITEM 3 - Position Paper on "Event Combination Associated with Dynamic Load Combinations Applicable to Nuclear Power Plant Piping", by J. D. Stevenson.

Comments/Concerns

We have found Stevenson's approach to be acceptable. The recommendations appear to be specific enough with respect to the definition of design basis. They also appear to be reasonable in eliminating the combination of earthquake with DEGB or maximum LOCA as a design basis event.

ITEM 4 - Position Paper "Vibration Loads Considered as a Design Basis for Nuclear Power Plant Piping", by J. D. Stevenson

Footnote [2]
of front page: The term endurance limit for the fatigue limit at 10^6 cycles is perhaps not appropriate for the discussion of the third category of vibration. Note that the stainless steel curves have been extended to 10" cycles, and OM.3 applies a reduction factor to the allowable stress at 10^6 cycles for carbon steel, even though the ASME curves already contain a safety margin of the larger of two on stress or twenty on cycles. Note that the term "endurance limit" will be deleted from the next revision of OM.3.

General

Much is made of the use of velocity as a criteria for judging the damage potential of vibration, based on the Hartlen, Elmaragby, and Stingerland paper. However, at least two papers have shown possible instances of unconservative acceptance criteria being calculated using the velocity method, and the OM.3 subcommittee will introduce, in the next revision, a frequency dependent correction factor to the velocity method. Unfortunately, addition of this frequency dependence removes part of the desirability from the velocity method.

(1) "Conservatism Inherent in Simplified Qualification Techniques Used for Piping Steady State Vibration"- 7th International Conference on Structural Mechanics in Reactor Technology, 1983, Chicago, Ill. by D. E. Olson and J. L. Smetters, Sargent & Lundy Engineers.

(2) Screening Procedures for Vibrational Qualification of Nuclear Plant Piping, ASME Paper 80-C2/PVP-4, J. E. Stoneking and R. C. Kryter, Dept. of Engineering Science & Mechanics, Oak Ridge National Laboratory, Univ. of Tennessee.

ITEM 5 - Position Paper on Response Combinations", by R. P. Kennedy

Comments/Concerns

1. Equation (4) on Page 10 should be β_j instead of β'_j.

Suggestions

1. The paper provides a fairly good discussion on the NRC Staff interim position (Table 1). Although much of the suggested changes (Table 2) are reasonable, some improvements appear to be desirable. These are provided in Table A, and are described in the following paragraphs:

   (a) For inertial or dynamic components (primary):
   Although the phase relationship cannot be easily defined within a primary structure, the phase should be uncorrelated for two different structures, (such as containment and interior concrete) which may both provide supports to a similar piping system. In such a case, group responses should be combined by the SRSS method.

   If, on the other hand, it can be shown that the supports are those uncorrelated even within the same structure (such as one support close to the base and the other at a high elevation), then again SRSS should be used.

   Consequently, the suggested revision is "For each mode and for each input motion direction: combine group responses by absolute sum (ABS), unless the groups are from different structures (or if from the same structure they can be shown to be phase uncorrelated), then SRSS should be used".

   (b) For support displacement or pseudo-static components, the same philosophy as described in (a) above should be used.

2. Although there is only a small difference in whether to combine modes first or directions first, there could be a substantial difference in computational efficiency. If directions are combined first, then for each solution, there will only be one modal response printout. Conversely, if the modes are combined first, then there will be three modal response printouts; one for each translational direction input. In terms of data management, combining directions first would then be more logical.

3. The following two references are recommended:

(a) Vashi, K. M. "Seismic Spectral Analysis for Structures Subject to Non-Uniform Excitation", ASME Paper 83-PVP-69.

(b) Lin, C.-W., Loceff, F. "A New Approach to Compute Spectrum Response with Multiple Support Response Spectra Input", Nuclear Engineering and Design, 60 (1980) pp. 347-352.

TABLE A

Westinghouse Suggested Revision to Recommended Algorithm for Combining Responses Using the Independent Support Motion Response Spectrum Analysis Method

A. Inertial or Dynamic Components (primary)

1. For each mode and for each input motion direction:
   Combine group responses by absolute sum (ABS), unless the groups are from different structures (or if from the same structure, they can be shown to be phase uncorrelated) then SRSS should be used.

2. For each response quantity:
   Combine input motion direction responses by SRSS or equivalent method.

3. For each response quantity and each input motion direction:
   Combine modal responses by the Double Sum (DSC) or CQC method with provisions for high-frequency modes.

This can be summarized as:
Group (ABS)/(SRSS with justification) - direction (SRSS equivalent) - Modes (DSC or CQC).

B. Support Displacement or Pseudo-Static Components (secondary):

1. For each group, calculate maximum absolute response for each input direction.
2. Combine for all groups by absolute sum, unless the groups are from different structures, or if from the same structure, they can be shown to be phase uncorrelated then SRSS should be used.
3. Combine for input directions by SRSS or equivalent method.

C. Total Dynamic Responses

Add dynamic and pseudo-static components by SRSS.

NOTE: For the design of piping, only the dynamic components are considered as primary. For piping supports or equipment supports, dynamic components clearly should be considered as primary. Pseudo-static loads applied to supports should be categorized as either primary or secondary. They are currently called primary, but we believe that they cannot cause failure like a dynamic load. This should be pursued further.

PSE-SSD-1084

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

February 28, 1984

Mr. J. A. O'Brien
U. S. Nuclear Regulatory Commission
Mail Stop NL-5650
Washington, D. C. 20555

Dear Mr.O'Brien:

SUBJECT: Comments on Position Papers - Other Dynamic Loads and Load Combinations

The attachment contains Westinghouse comments on the paper "Stress Limits/ Dynamic Stress Allowables for Piping" by E. C. Rodabaugh, which we received through AIF from Mr. S. A. Bernsen. We are pleased to have this opportunity to express our views to the U.S. NRC Piping Review Committee and to assist you in your effort to define new criteria for designing nuclear power plant piping.

E B Johnson, for

T. C. Esselman, Manager
Engineering Mechanics

cc: P. Higgins

jm

WESTINGHOUSE COMMENTS AND SUGGESTIONS ON POSITION PAPER
"STRESS LIMITS/DYNAMIC STRESS ALLOWABLES FOR PIPING"
BY E. C. RODABAUGH, 2/10/84

COMMENTS/CONCERNS

1. Adding 10 cycles of SSE loading to the current ASME Class 1 fatigue evaluation could have a significant impact on those systems with high thermal gradient stresses. In some instances, the currently evaluated cyclic loadings result in usage factors higher than 0.9. This clearly would not result in failure, but could require more sophisticated analysis techniques to be utilized.

2. Table 3 identifies the potential impact of the new criteria on thin-wall stainless steel pipe. Calculations should be made to cover Schedule 160 piping, which is common to all PWR's, and a comparison similar to Table 3 should be made.

3. It is not clear what is meant by the following recommendation, which is found on Page 39:

   "(1) For the purpose of evaluating support and equipment loads, the present Code limits should be met."

   It appears to require that the ASME primary stress limits be met for earthquake loadings only in the supports and the equipment nozzles (including valves, tanks and pumps), but not in the piping components. If this is the correct interpretation, it will likely lead to an artificially unbalanced system design, rather than a more desirable balanced design. For example, permitting inelastic behavior of a pipe support or tank nozzle may result in a more efficient and reliable overall system design.

4. The ratio of OBE to SSE loads of 1 to 2 is no longer commonly found in piping system analysis. This is due in a large part to conservatisms in the damping values for buildings and piping systems.

SUGGESTIONS

1. In order to avoid the complexities and expense of NB-3200 analysis methods, a new set of equations that correspond to new simplified inelastic analysis methods should be developed. These methods should include the actual failure mechanics for piping components subjected to cyclic loadings with inelastic strain.

   A possible economic approach is to represent the piping system with inelastic pipe elements for straight pipes and elbows and elastic elements for branches and tees. The elastic elements are then evaluated using the simplified method of Equation 9 of NB-3650 while the inelastic elements are evaluated using the more detailed methods of NB-3200.

2. The SSE is a one-time event with much fewer than 10 cycles of maximum response expected. Protection against fatigue failure due to earthquake events is already provided for Class 1 by evaluating the OBE loadings. The currently designated magnitude and cycles of the OBE is very conservative and, therefore, SSE need not be evaluated for fatigue. If SSE is evaluated for fatigue, it should not be combined with any other expected cyclic loadings (e.g., thermal gradient stresses).

3. The appropriate requirements to ensure operability of piping components for the OBE and SSE should be addressed in this Position Paper to provide a complete picture of the potential impact of the new proposed criteria.

4. In Class 2 and 3 piping systems that do not experience significant thermal transients, we suggest that the "f" factor be increased to correspond to the small number of cycles of the earthquake loadings (much less than 7000). Margin can be included by applying an appropriate factor to Markl's equation.

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Mr. John O'Brien, Chairman
Task Group on Other Dynamic
Loads and Load Combinations
USNRC Piping Review Committee
Nuclear Regulatory Commission
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Mail Stop NL 5650

February 14, 1984

COMMENTS ON POSITION PAPERS PREPARED
FOR NRC PIPING REVIEW COMMITTEE

The following comments are offered for the position papers prepared for the USNRC Piping Review Committee:

1. Position Paper on Response Combination - R. P. Kennedy

   This paper is an excellent summary of the theoretical studies on response combination. However, it is too academic and is of little use for the piping designer unless a simplified design formula is also provided. It ignores the vast amount of historical data which demonstrate that the existing design rule is adequate for piping and there is no need to engage in such sophisticated theoretical analysis when there is a large safety margin already built in the current design methodology. We suggest that this paper be used as the basis to justify the simplest combination method, such as the SRSS, for piping design without any further concern on closely spaced modes or high frequency response.

2. Piping System Dynamic and Thermal Stress Response Induced by Thermal -Hydraulic Transients - J. G. Areadts

   Section 5.3 Modeling Considerations - It has not been the industry's practice and it has been judged unnecessary to model the pipe support accurately to include nonlinearities in piping analysis. The degree of sophistication suggested is not consistent with the level of accuracy for the input and present design methods.

3. Position Paper on Stress Limits/Dynamics Stress, etc. - E. Rodabaugh

We believe that the pipe dynamic motions resulted from seismic and other dynamic loads typical in a power plant do not justify the use of strain rate effects in the analysis.

for Louis Nieh
Consulting Engineer
Stone & Webster Engineering Corporation

CC: Pat Higgins
Atomic Industrial Forum, Inc.
7101 Wisconsin Ave.
Bethesda, Maryland 20814

SWEC: A. W. Chan
A. L. VanSickel
D. A. VanDuyne

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 2 8 1984

MEMORANDUM FOR: John O'Brien, Chairman
Task Group on Other Dynamic Loads and Combinations
USNRC Piping Review Committee

FROM: John R. Fair
Engineering and Generic Communications Branch
Division of Emergency Preparedness and Engineering Response
Office of Inspection and Enforcement

SUBJECT: REVIEW OF CONSULTANTS POSITION PAPERS

I have reviewed the consultant's position papers and have the following comments:

Stress Limits/Dynamic Stress Allowables for Piping

1. The basic thrust of the paper deals with seismic conservatisms and does not address the issue of increased allowable for dynamic loads due to strain rate effects. The recommendations in this paper would have a major impact on seismic design and would be more appropriate in the Task Group on Seismic Design.

2. The paper has two inconsistent recommendations. On page 4 the statement is made that, "the SRP's should be revised to say that inelastic analysis methods are acceptable." Then on page 7 the statement is made, "Accordingly, in our opinion, rigorous inelastic analysis of piping systems is in an early research stage. An attempt to prescribe generally applicable stress or strain limits for such analyses is premature and not needed at this time." Based on previous experience with piping codes we should not endorse analysis methods until we have properly verified codes to use for benchmarking purposes.

3. The major recommendations in the paper appears to be based on an unpublished paper by Broman which concludes, "There is insufficient energy in typical seismic motions to cause the formation of primary collapse mechanisms in beam spans..." I would like to see this study before accepting this conclusion. While it seems feasible that this could be demonstrated for simple and continuous beam spans where load redistribution and progressive yielding results in large deflections at failure strains, it would be difficult to extrapolate this study to complex piping geometrics where strains could be more localized (elbows, fittings, valves, etc.). It should be noted that if the piping systems contained only simple straight

beams there would be no problem with stiff systems since reasonably long spans between supports are possible for these cases even using current criteria.

4. The paper recommends that current code limits be met in evaluating support and equipment loads. However, if the stress ranges are permitted to allow gross plastic deformation in the pipe, how can an accurate evaluation of support and equipment loads be made? Although it is generally assumed that loads are reduced when inelastic response occurs, this is not necessarily true for complex structural systems where significant load redistribution may occur.

5. It is not clear from the proposed changes how the Class I fatigue exemption rules of NB 3200 and NB 3600 will be considered. For example, NB 3200, which can be used for piping, allows exemption from fatigue analysis if six conditions are met. These conditions treat thermal, pressure, and mechanical loads separately. It should be noted that NB 3653(b) allows the use of NB 3200 when Equation 10 limits are exceeded.

6. The paper recommends a change to NB 3672.6 to allow use of inelastic methods provided the designer justifies appropriate stress or strain limits. This is an open ended criteria and does not appear appropriate for the cookbook section of the code. Also, the question arises as to the appropriateness of the code stress indices and design fatigue curves if the recommendations contained in the paper are implemented. These recommendations will allow gross inelastic deflections in the piping system whereas the code stress and fatigue evaluations are based on gross elastic behavior.

7. The proposed changes could result in Class 2 stress limits being less conservative than Class 1 limits. The 51 ksi limit for SSE is equivalent to an allowable stress range of 6 Sm for A-106 Grade B pipe for earthquake alone. With the Class 1 fatigue evaluation earthquake will be combined with thermal including thermal transient effects (the transients are not evaluated in Class 2) to calculate the total stress range.

8. The effect of these proposed changes along with the items considered by Task Group on Seismic Design (such as increased damping) may effectively eliminate all required earthquake restraints. When constraints such as low nozzle allowables are excluded, the basic problem in seismic piping analysis is meeting allowables when responses are near the peaks of the floor response spectra. Since it is difficult to straddle the peak with a piping system, most designers design for first mode frequencies on the high frequency side of the peak (typically the first and major building peak is at 5-6 Hz and the piping system first mode frequencies are greater than 8 Hz). On the flexible side of the peak accelerations decrease rapidly with decreasing frequency and the result yielding an almost constant first mode maximum moment as span length is increased (this occurs if the acceleration decreases linearly with frequency). For purpose of illustration using the 51 ksi criteria for SSE and assuming a simply supported piping span, a spectrum peak of 10-20 g's could be tolerated

at 5 Hz without causing an overstress. I suggest that sample analyses of actual piping systems be performed to assess the impact of these criteria changes. The national labs should have sample problems already coded and could easily remove or relocate restraints to evaluate stress allowable or spectrum modification changes.

<u>Piping System Dynamic and Thermal Stress Response Induced by Thermal-Hydraulic Transients</u>

1. The statement on page 17 concerning axial extension of piping segments inducing bending moments needs clarification. I would not expect this axial extension due to most hydrodynamic loads to be any greater than the extension due to design internal pressure which is not included in ASME code evaluations either.

2. The recommendations imply that current evaluation techniques are inadequate. If the techniques give unconservative results, we need recommendations for improvement and assessment of the significance.

3. The concern on the number of stress cycles due to S/RV transients needs clarification. Typical S/RV discharge lines are not ASME Class 1 and do not require fatigue considerations for mechanical loads. Is the recommendation that a fatigue evaluation be performed on Class 2 and 3 S/RV discharge lines?

<u>Vibration Loads Considered as a Design Basis for Nuclear Power Plant Piping</u>

1. I do not completely agree with the first general recommendation. Currently BWRs are evaluating effects of the containment responses due to LOCA on equipment and piping qualification.

2. The third general recommendation needs to be clarified in terms of how it would be accomplished and the impact due to the change.

3. The fourth general recommendation does not seem practical. Equating or extrapolating piping responses from system transients to earthquake response could not be performed directly since the load directions, frequency content, and load magnitudes are different. The recommendation should be more specific in terms of how this would be accomplished.

4. Specific recommendation 2 is not consistent with general recommendation 1.

<u>Event Combination Associated with Dynamic Load and Load Combinations Applicable to Nuclear Power Plant Piping</u>

1. General recommendation 1 is not supported by the last paragraph of Section 2.1. This recommendation is premature until the results from the Task Group on Pipe Break are obtained. General recommendation 2 cannot be impelemented until the first recommendation is formally accepted.

2. General recommendations 3 and 4 are generally contained in current SRP revisions and are implied by GDC 4. I don't think formal revision of GDC 4 is necessary.

Response Combinations

1. The paper cites several methods that have been proposed for the combination of model responses and the related accuracy or lack of accuracy of these methods. It is not clear, without reading the referenced papers, how the exact solutions are determined. The discussion references RG 1.60 spectra; however, the input to piping is a floor response spectra developed from the building response. Since the building motion frequency content can be significantly different from the ground response, are these studies applicable to piping response from actual building motions?

2. In my experience the unconservatism in inertial forces due to high frequency response of piping systems is more a consequence of model cut-off and infinite support stiffness assumptions used in the analysis then modal summation methods. This results in neglecting the rigid body response of stiff portions of the piping system. I agree with the recommendation that analysis techniques should be adjusted to account for ZPA forces in stiff portions of piping systems.

John R. Fair

John R. Fair
Engineering and Generic
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cc: R. L. Baer, IE
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28 February 1984

Dr. John O'Brien
Mechanical Engineering Research Branch
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U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear John:

Per your request enclosed herewith please find my comments on Rodabaugh's paper. In general I consider it an excellent review of the problem areas and concur in the recommendations concerning ASME Code changes. I have only three major areas where I differ with his recommendations.

(1) Different safety factors should be used on Service Level B as compared to Service Level D fatigue analysis limits. For example the Code specifies a "normal" safety factor limit of 20 on cycles and 2 on stress. I suggest for Service Level D this might be reduced to 10 on cycles and 1.5 on stress. Otherwise we are not consistent with procedures used with other Code allowables.

(2) Axial stresses in piping systems subject to differential support motions in real earthquake appear to be at least as important a contributor to failure as bending stresses. For this reason stresses induced in the piping by SAM (seismic support motions) should also include axial effects.

(3) The more conservative approach taken for supports as opposed to pipe design in my opinion currently results in over design of supports with the result that the pipe would be more likely to fail than the support given a limiting differential movement of the support. This is contrary to a balanced design concept where our primary goal is to maintain the structural and leak tight integrity of the pipe.

I suggest we might considered action taken by AISC in their approach to the problem (see attached).

Please advise if you require any clarification of my comments.

Sincerely,

John D. Stevenson
President

JDS:lap

Enclosure

nutech ENGINEERS

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DR. N. W. EDWARDS, P.E.
President

March 2, 1984
NWE-84-037

Mr. Donald Landers
Senior Vice President
Engineering Operations
Teledyne Engineering Services
130 Second Avenue
Waltham, Massachusetts 02254

Dear Don:

We have reviewed draft copies of position papers that are being generated by consultants to the NRC Piping Review Committee Task Groups on Seismic Design and Load Combinations/Other Dynamic Loads, and wish to offer comments. You are being contacted because of our understanding that you are industry's representative on these Task Groups. We appreciate the effort you and the others, such as PVRC, are putting forth to make piping analysis methods more realistic. We agree with the majority of the points made in these papers, but would like to offer the following comments:

1. Position Paper on Response Combinations

   By R. P. Kennedy

   In this paper, Kennedy endorses the NRC staff position of absolute sum combination of support group dynamic responses when using the independent support motion seismic analysis technique. We believe that this absolute sum rule, when used in conjunction with the already conservative procedures/methods used for today's seismic analysis, will result in unnecessary overall conservatism in seismic design. In other words, this would be counter to the intent of the Task Group's effort to identify more reasonable seismic design requirements. The absolute sum rule may be appropriate if the other seismic analysis conservatisms are adjusted. Thus, imposition of absolute sum methods should not take place unless the other changes are made concurrently. Meanwhile, NUTECH recommends that the SRSS rule be used in conjunction with today's analysis procedures. We believe that this recommendation is consistent with the preliminary recommendation made by Brookhaven at the January PVRC Steering Committee Meeting in Fort Lauderdale.

Limiting the Use of Snubbers in Nuclear Power Plants' Safety Related (Seismic Category I) Piping Systems

By J. D. Stevenson

Stevenson states that a snubber reduction in the range of 25 to 40 percent would be needed to offset the cost of the analysis effort, and that such a reduction might be possible if more realistic analysis procedures were adopted (such as higher damping, dynamic stress allowables, and so forth). This may be valid for those plants initially designed with just enough snubbers to enable the piping to meet code requirements. However, in our experience several plants have a considerable number of snubbers that could have been eliminated if a more complete analysis had been performed in the initial design phase. Examples are: use of snubbers at locations where piping thermal displacements are small, and at locations immediately adjacent to rigid supports and equipment. Some of these snubbers can be eliminated and others replaced by rigid struts for a small analysis cost.

We agree that there is a cost benefit consideration to be made by a utility in addressing the snubber question. We think that substantial snubber reductions can be achieved for less cost than the 30 to 40 percent mentioned above. Some snubbers can be removed for very little cost, and this should be done right away. Further reductions can be realized in conjunction with a comprehensive seismic reanalysis. There is a point of diminishing return in cost benefit considerations, but we are more optimistic about the potential reduction in snubbers than has been reflected in Stevenson's paper.

In the same paper, Stevenson proposes that a minimum pipe support gap (i.e., .125 inches) would be beneficial for seismic response. This may mislead some into thinking that large gaps would not be a concern. Until more test and/or analysis data on the effects of gap sizes for all loadings (including water hammer) become available, it may be prudent to also recommend a reasonable maximum gap size.

In this and several of the other position papers, the issue of excess snubbers for seismic design is emphasized; yet in his load combination paper Stevenson identifies other dynamic loads, such as water hammer, as being appropriate for consideration. We agree that consideration of other

nutech ENGINEERS

loads such as water hammer should be addressed in design of piping and supports (including snubbers). However, loads for water hammer are not well defined, nor are the analysis methods correlated with the phenomenon.

In the past, conservative loads and approaches to combining loads were used to avoid rigorous evaluation of every event scenario imaginable. This was cost prohibitive, and tools did not exist to perform the analyses. With today's analytical capabilities, more rigorous event combinations can be performed; but in so doing, it is appropriate to more accurately define the load as well as the time relationship for the events being combined.

3. Consulting Paper on Seismic Design of Piping

By R. P. Kennedy

We agree with Kennedy's position that one earthquake analysis is sufficient. We favor the concept of using SSE for the analysis, adding a provision for inclusion of seismic anchor motion secondary stresses for ASME Code Service Levels C and D.

I hope that NUTECH's comments will enable you to add to the other industry input being provided. It would be a very positive action if peer review were possible on more of these Task Force efforts. If you or any of the committee members have any questions on these comments, please call me, Jon Arterburn (404-955-1275), or Vic Weber (408-281-6229).

Sincerely,

Norm Edwards

Norman W. Edwards

NWE/bjm

cc: Mr. S. Hou (USNRC)
Dr. J. O'Brian (USNRC)

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MEMORANDUM FOR: John O'Brien, Chairman
Task Group on Other Dynamic Loads
and Load Combinations
U.S. NRC Piping Review Committee

FROM: Mark Hartzman
Mechanical Engineering Branch
Division of Engineering, NRR

SUBJECT: COMMENTS ON THE POSITION PAPER ON RESPONSE COMBINATIONS BY R. P. KENNEDY

This paper is a very good summary of current research in the technique of response spectrum analysis, and as such, it deserves further detailed study. However, based on the work done by BNL it seems to me that the important question is not the method for combining modal responses, but the method for combining group responses. I would therefore like to recommend the following modifications to Table 3.

1. Abandonment of all modal combination techniques except the SRSS algorithm. This will also take care of the question of the order of combinations for direction and mode. Since both are combined by SRSS, the order is irrelevant.

2. Include the high-frequency rigid body effects as outlined in the Appendix.

3. All supports are to be taken as elastic, that is, to have finite stiffness. Backup steel should be included in calculating the stiffness, if appropriate.

M. Hartzman

Mark Hartzman
Mechanical Engineering Branch
Division of Engineering, NRR

TABLE 3

SUGGESTED REVISION TO RECOMMEND ALGORITHM FOR COMBINING RESPONSES USING THE INDEPENDENT SUPPORT MOTION RESPONSE SPECTRUM ANALYSIS METHOD

A. Inertial or Dynamic Components (primary)

1. For each mode and for each input motion direction: Combine group responses by absolute sum (ABS).

2. For each response quantity and each input motion direction: Combine modal responses by SRSS.

3. For each response quantity: Combine input motion direction responses by SRSS.

This can be summarized as:

GROUP (ABS) - MODES (SRSS) - DIRECTIONS (SRSS)

B. Support Displacement or Pseudo-Static Components (secondary):

1. For each group, calculate maximum absolute response for each input direction.

2. Combine for all groups by absolute sum.

3. Combine for input directions by SRSS.

C. Total Dynamic Responses

Add dynamic and pseudo-static components by SRSS.

Note:

1. For the design of piping, only the dynamic components are considered as primary. For piping or equipment support, both dynamic and pseudo-static components should be considered as primary.

2. Supports should not be considered rigid for any frequency. (Model actual stiffness of support.)

3. High frequency modal effects should be included as outlined in the attachment.

Attachment

Recommended Procedure for Inclusion of High Frequency Modal Effects

1. Determine the modal responses only for those modes with natural frequencies less than that at which the spectral acceleration approximately returns to the ZPA.

2. For each degree-of-freedom included in the dynamic analysis, determine the fraction of degree-of-freedom (DOF) mass included in the summation of all of the modes included in Step 1. This fraction F_i for each degree-of-freedom i is given by:

$$F_i = \sum_{m=1}^{M} \left(\sum_{n=1}^{N} PF_{m,n} \right) * \phi_{m,i}$$

where

m	is each mode number
M	is the number of modes included in Step 1.
$PF_{m,n}$	is the participation factor for mode m and group n.
$\phi_{m,i}$	is the eigenvector value at DOF i for mode m and group n.

3. Determine the fraction of DOF mass K_i not included in the summation of these modes:

$K_i = F_i - \bar{\delta}$

where

$\bar{\delta}$ equals one if DOF i is in the direction of the earthquake input motion and zero if DOF i is a rotation or not in the direction of the earthquake input motion.

If, for any DOF i $|K_i|$ exceeds 0.1 the response from higher modes should be combined with those in Step 1.

4. Calculate the pseudo-static inertial forces associated with the summation of all higher modes for each DOF i, given by:

$P_i = ZPA * M_i * K_i$

where

P_i	is the force or moment to be applied at degree-of-freedom (DOF), i
M_i	is the mass or mass moment of inertia associated with DOF i

5. Analyze the structure statically for this set of pseudo-static inertial forces applied at all of the degrees-of-freedom to determine the maximum responses associated with the high frequency modes not included in Step 1.

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March 21, 1984

Mr. John O'Brien
Mechanical/Structural Engineering Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Dear John,

I have reviewed the Position Paper, "Vibration Loads Considered as a Design Basis for Nuclear Power Plant Piping," by John Stevenson dated January 1984 and have the following comments.

<u>Page 1, Footnote (2)</u>

The new changes to the Code fatigue curves, which now extend beyond 10^6 cycles, have changed the endurance limit for nuclear components from 10^6 to that point at which increased number of cycles does not require reduced alternating stress to preclude a fatigue failure.

<u>Page 1, 1st Paragraph</u>

As we are all aware, building structure vibrations associated with suppression pool hydrodynamic loading has been considered for some time in BWR Mark II and III plants. Loads associated with aircraft impact, etc., have not been analytically used for designing piping systems in the U.S.

<u>Page 2, Section 2.1</u>

The discussion on hydrodynamic loads is extraneous since it is now considered.

<u>Page 3, Section 2.1, 2nd Paragraph</u>

I have not studied the references in detail but, I think what is being said is that at high frequencies insufficient energy exists in the loading to produce failure of the piping. Certainly, the best measure of response and loading in a piping system is displacement. The subsequent loads produced by that displacement result in a stress level that can be compared with an allowable value. At high numbers of cycles the displacement (and subsequent stress) need only result in stresses beyond the endurance limit to be of concern. For socket welded systems the displacement of concern is significantly less than that for a butt weld system because of the high stress concentraitons that occur at socket welds. We must be cautious in addressing high cycle vibration problems in a general fashion. Just as pointed out in my comments on Everett's paper, I think the vibration problem is best dealt with by providing design tools up front, continuing to require preoperational testing, and enforcing plant operating personnel to report on vibrating systems during plant operation. Preoperational testing should, as a minimum, include those systems which experience tells us are problems. For example, feedwater systems have been a problem prior to nuclear power. There is a 1950 or 1955 paper by GE on feedwater vibration problems in fossil units. It's not new - yet we still have problems.

Page 4, Section 3.1 (1)

As discussed in my second comment, hydrodynamic suppression pool loads for Mark II and III BWR's are considered in piping design.

Page 4, Section 3.1 (3)

This may not be the total solution. See my discussion on Section 2.1, 2nd Paragraph.

Page 4, Section 3.1 (4)

Testing is great if you know what the load input really is.

Pages 5 and 6, Item 1, Last Sentence

In order to do this (exclusive of earthquake) the test procedure would have to be rather extensive and more elaborate than is currently used. This may not be the way to go.

In general there is not much in this paper that is of significance. The discussion on deflection or velocity versus acceleration is meaningful but more study needs to be done since no real recommendations are made. Vibration due to system operation has been a continuous problem and current approaches have not solved it. More work needs to be done in this area as I have already pointed out in my letter of March 14th on Everett's paper.

If you have any questions, please do not hesitate to call.

Very truly yours,

TELEDYNE ENGINEERING SERVICES

Don

Donald F. Landers
Senior Vice President

DFL/lh

ENGINEERING SERVICES

March 14, 1984

Mr. John O'Brien
Mechanical/Structural Engineering Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Dear John,

I appreciate all of the reports that you have been forwarding to me and I plan on reviewing and commenting on as many as possible in time for the comments to be of any value in preparing your draft report. The following are comments on the February 10, 1984 report by E. C. Rodabaugh on "Stress Limits/Dynamic Stress Allowables."

Page 1, Section 1.0, 4th Paragraph

The Code does not provide stress limits for Design Conditions which cause plasticity in the piping for ferritic material. The older rules for Class 2/3 (and B31.1) limited the allowable stress for Design Conditions to S_h (2/3 S_y or 1/3 Su, whichever is lower). The most recent changes to Class 2/3 to bring Equation (8) in line with Class 1 uses an allowable of 1.5 S_h which can result in longitudinal stresses reaching the minimum yield value of the material.

Page 7, 2nd Paragraph from Bottom

Rigorous inelastic analysis is not in an early research stage. It has been performed for piping and other components for a number of years, particularly in the liquid metal field. The problem is that this approach is not economically reasonable for all LWR piping. Strain limits have been established in Code Cases for high temperature piping and in Appendix F for inelastic analysis. The basis for these, or the margin, is perhaps not well defined. However, for accident conditions, or for detail functionality, accumulated strain in the order of 5% has been used. I would agree that inelastic analysis of piping systems should not be used for design of LWR piping but we should not legislate against it for certain situations.

Page 9, 1st Paragraph

Equation (9) controls inertial earthquake moments in all cases and can be used to control anchor motions at the option of the designer. If anchor motions are not used in Equation (9), they must be considered in Equation (10) or (11). Standard practice for Equation (10) or (11) is to add 1/2 range of earthquake moments (anchor motion induced) to the range of thermal expansion moments or to use the range of earthquake anchor motion moments, whichever is greater.

Page 9, 3rd Paragraph, Section 2.2.1

I believe Appendix N talks about 10 significant earthquake cycles per event. With respect to the remainder of 2.2.1, I always have problems when an author takes one load and does an analysis with it. It is similar to applying cyclic pressure to a component to fail the component in 20 or 30 cycles. The fact is that the magnitude of pressure required is not allowed by other Code rules. In the case of this work that is not entirely true, but the fatigue evaluation for Class 1 piping will require consideration of other loads which are combined with the seismic event in accordance with the Dynamic Specification. Further, Tables 2 and 3 should include pressure effects (2500 psi and 1500 psi to reflect PWR and BWR conditions) and some estimate of weight effects (say 2000 psi). This would change the Table 2 results dramtically.

Page 15, Section 2.2.2, 3rd Paragraph

I recognize that Rodabaugh and Moore feel that the Class 1 piping fatigue evaluation is only acceptable because it compares well with the B31.1 approach. However, the rules were drafted based on NB-3200 criteria and stress determination techniques and the fact that they compare well points out (in my mind) that fatigue failure, and protection against it, is not a new phenomenon. Whether one test material specimens and develops design curves to accommodate fabrication techniques or test components and develops a design curve we end up at essentially the same point. Equation (4) on Page 15 was not the basis for Class 1 fatigue rules or the acceptance thereof. As an aside, the relative agreement between Class 1 and B31.1 speaks well for the brilliance of the authors of the B31.1 rules.

Page 17

I would support approach numbers (3) and (4) but I disagree with the allowable stress limits used in (4).

Page 18

This may be where my problems with (4) come from. Since $i = C_2K_2/2$, then I don't think we need to again divide the stresses (102 ksi and 64.4 ksi) by 2. We use a range of moment (M_B) but we multiply it by an i value which already contains the 1/2 factor and the resulting stress is an amplitude and not a range.

Page 22, Section 2.3, Last Paragraph

I am not sure that history agrees with preoperational testing resolving vibration problems. It may be too early to tell since most plants which have had vibration failures may not have been subjected to current preoperational testing requirements. However, we do know that failures occur and we should gather that information, determine causes and provide guidance to the industry for use in the design stage to preclude failure.

Page 23, Section 2.4, 1st Paragraph

This is not a true presentation of Code criteria but it's not worth worrying about. Everett's bottom line here is true, earthquake anchor displacements are not put into Equation (9) of Class 1. In the last sentence this statement is true for pressure boundary but not for the supports.

Page 25, Section 2.4.1, 1st Paragraph

This is a much better dissertation on Code rules, particularly Design vs. Level A, etc. However, as I read on, the discussion with respect to Levels C and D is out of order since the Design Specification and the FSAR spell out what events are considered in C and D and this interpretation of the Code rules does not agree with anyone else. In fact, Code interpretations have been written in this area which clearly point out that only inertial moments need to be considered for Levels C and D.

Pages 26 and 27

I have not read this in detail and I'm sure I would not agree totally with the precise wording change. However, I do object to deleting F-1430. This should not be done since there are a number of reasons why I may want to use Appendix F, particularly for inelastic analysis. If you want to restrict use of Appendix F to other than SSE, then maybe I would grudgingly agree.

Page 30 and on, Section 2.5

I don't think anyone would support strain rate effects for an earthquake event and I think this report should say that. For other dynamic loads I would agree with the last paragraph on Page 38.

Now, a general comment. I vigorously support the conclusion of the author to remove earthquake from primary stress consideration and to deal with it in a fatigue/plastic ratcheting sense. I think we need to look at assuring ratcheting protection a little more closely. It would be more presentable to me if the paper made recommendations and defended them on a plant realistic basis and did not spend a lot of time trying to outwit the Code.

Hope the above helps and I will try to review the others soon.

Very truly yours,

TELEDYNE ENGINEERING SERVICES

Don

Donald F. Landers
Senior Vice President

DFL/lh

PART II

Industry Comments on Draft Staff Report

Review & Synthesis Associates
Spencer H. Bush, P.E. • 630 Cedar / Richland, Washington 99352

June 21, 1984

Dr. John O'Brien
Mechanical/Structural Engineering Branch
Division of Engineering Technology, NRR
U. S. Nuclear Regulatory Commission
Mail Stop 1130-SS
Washington, D. C. 20555

Dear John:

COMMENTS ON DRAFT REPORT OF TASK GROUP ON OTHER DYNAMIC LOADS AND LOAD COMBINATIONS, USNRC

Enclosed are my general and specific comments concerning the subject report. One general comment has to do with its unevenness. I recognize that the sections were written by several people; however, Section 8 in particular differs markedly from the other sections.

One other suggestion pertains to Section 9. All of the foreign information is included in this section in contrast to a comparison of foreign approaches to a given area such as water hammer. It doesn't appear to impact on the recommendations and could easily become an appendix.

A technical editor whose primary function was to develop a uniform format could markedly improve the "readability" of the report.

Very truly yours,

Spencer H. Bush

Spencer H. Bush, P.E., Ph.D.
Consultant
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Enclosure

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DRAFT REPORT OF
TASK GROUP ON OTHER DYNAMIC LOADS AND LOAD COMBINATIONS

General Comments

- As noted in the cover letter, the variability of language in the text reduces readability.

- You have recommendations only. Should have conclusions to serve as bases for the recommendations?

- I could argue that some omitted recommendations have more impact than those in the Executive Summary.

- For consistency, should you pull all conclusions (?) and recommendations together into one section (a la NUREG-1061, Vol. 1)?

- Section 4, Page 2 -- The statement in the long paragraph re: "...although special attention must be directed towards maintaining the reliability of heavy component supports.." is important. If covered in SRP's, it should be cited.

- Section 4.3, p. 3, bottom ¶. The blanket statement "There is a general consensus that anticipated water hammer events should be combined with earthquakes and plant dynamic events" may be true; however, I'm in doubt as to whom makes up the consensus. Clarify.

- Section 4.3, p. 4. "...to the prevailing view...". Whose prevailing view? NRC, industry?

- Section 4.5, Item ii. I don't understand the citation of heavy component supports here. The remainder should be Klecker's.

- Section 5.2, p. 2. What is the difference between SRP and BTP? They used to be the same.

- Section 5.3, p. 4, 1st ¶. "...unanimous opinion...". Is this really the case?

- Section 5.3, p. 4, last ¶. It repeats the top of p. 3. Okay?

- Section 5.4, pp. 5-6. I have problems with format. Bullet at top of Page 5 apparently is a lead into the following two headings--or is it three? The third heading can be read as under independent inputs. If the third applies, do the next two bullets revert to major items? I assume they do, but it's confusing.

- Section 5.5, p. 6, first bullet. Where is the justification for this work--and why?

- Section 6.2, p. 2, last ¶. In passing, ASME XI permits credit for strain hardening.
- Section 6.3, p. 4. Clarifies application of >10% increase cited in the Executive Summary.
- Section 7.4, ¶ in quotes. As written, this infers you can ignore water hammer if not specified in Design Specifications.
- Section 7.4, p. 3. Items a through h are apples, oranges, bananas, etc.; e.g., a, b, d, f are one category; c, e are another, g ?, and h another. It could be written more clearly.
- Section 7.5, p. 5. Item b is h above. Why not drop h?
- Section 8 is markedly different in format and much harder to follow. It needs extensive editing, or the other sections need beefing up.
- Section 8.2, p. 2. The paragraph starting "unanticipated vibratory loads" is ambiguously phrased.
- Section 8.3, p. 3, bottom ¶. How do you test for unanticipated loads?
- Section 8.4. Types 1, 2 and 3 vibratory loads need defining.
- Appendix A. I'm confused as to why this is included.

Specific Comments

- Recommendation 4 under Executive Summary is a subset of # 3 if I believe the body of your report. (Also note algebra.)
- Recommendation 5 is ambiguously phrased. The >10% refers to σy, but can be inferred to be ε.
- Shouldn't Items 2 and 6 follow one another to highlight water hammer?
- The point isn't made as to how Item 10 differs from current practice.
- Under 3.2, Item 3, and in the body of the text, I don't come away with the significance and need for the action.
- If 3.2, Item 5, is important, shouldn't it be in 3.1?
- Section 4.4, first bullet. Isn't the long-term effects item more logically in Shou Hou's writeup?
- Section 4.4, second bullet. Either this should be handled by TGPB or it should be clarified re: sizing containment, etc. Certainly the last portion is Klecker's responsibility.

- Section 4.4, 3rd bullet. This is phrased to be optional assuming it is specified in the Design Specifications. Wouldn't it be better to recommend its inclusion in the Design Specifications?
- Section 4.4, Item iii. This will require amendment which is a major effort, yet it isn't in the Executive Summary.
- Section 5.2, p. 2. SRP or Reg. Guides.
- Section 5.3, p. 4, ¶2. Reputable evaluations
- Section 5.4, p. 6. Regulatory Guide 1.92
- Section 8.4, p. 3, last line. in/order
- Section 8.5, p. 6. transients
- Section 8.5, p. 8, Item 1, last line. progr~~r~~am
- Section 9.0, p. 4. maximum or maxima?
- Section 9.0, p. 17. Paper/and
- Section 9.0, p. 20, 6.1. sur~~r~~vey

Comments on Appendices (Other Than A After Section 9)

With regard to the appendices, I didn't spend a great deal of time on format, editing, etc. I read them for flavor and concentrated on those where I felt most comfortable. Obviously, I made no attempt to check model or mathematical validity. The following comments are more for flavor.

J. D. Stevenson. On page 6, I can't follow the logic in the bottom paragraph regarding SSE loadings on BWR recirculation pump and pump support failure. Supposedly, it was covered in UCRL-15340 but I couldn't unearth it. There appears to be an extrapolation from the lack of design of the recirculation pump for DEGB to the SSE. Perhaps it should be clarified.

R. P. Kennedy. This paper gives a good overview of the current status of dynamic load criteria as well as ongoing work at BNL, etc. My basic question is one of charter. Both this and the preceding appendix could easily apply to the Seismic Design Task Group. Is there a clear definition of scope for each Task Group? I could not get the recommendations to track the body of the report.

E. C. Rodabaugh. Page 3 makes the point that ASME III is mute regarding handling dynamic loadings such as SRV's. Isn't this a significant item, particularly if handled inelastically? I assume this is embodied in 3.2. I'm surprised 3.2 doesn't appear under 6.4.

Al Serkiz. I'm not in general agreement with the philosophy expressed in the Water Hammer Appendix; therefore, I'll not comment.

R. C. Guenzler. In essence, this appendix accepts the status quo with the possible exception of fatigue loads. The one problem I see is that any analytic solution assumes a priori that both design and fabrication of the pipe-to-valve joint is correct. Two of our more dramatic failures occurred when this was not the case.

J. D. Stevenson. Much of the meat in Section 8 is lifted directly from this appendix. I'm not in favor of being so specific as a general rule, feeling that is the responsibility of the implementing organization. Some of the changes strike me as relatively trivial; however, I'm not prepared to argue pro or con.

Commentators. I could predict from the tenor of some letters what axes were being ground. I'm afraid I consider some responses as being politically rather than technically motivated.

SHB:dp
6/21/84

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June 28, 1984

Mr. John O'Brien, Chairman
Task Group on Other Dynamic
Loads and Load Combinations
U.S. NRC Piping Review Committee
Nuclear Regulatory Commission
Washington, D.C. 20555
Mail Stop NL 5650

Dear Mr. O'Brien:

COMMENTS ON EVALUATION OF OTHER DYNAMIC LOADS AND LOAD COMBINATIONS"
(NUREG-1061, VOLUME IV, DRAFT) U.S. NRC PIPING REVIEW COMMITTEE

Please let us compliment the Task Group and your effort to reduce the postulated conservatism inherent in the dynamic analysis procedures of piping systems.

The following comments are offered for the NUREG-1061, Volume IV (Draft) prepared by the U.S. NRC Piping Review Committee:

1. Evaluation of Flawed (Degraded) Ductile Piping

   Unless physically justified in special case(s), postulation of flawed (degraded) Category I piping, as recommended by Sections 3.2.2, 4.5.iii, 7.4h, and 7.5.b is not warranted. A generic study to evaluate the responses of ductile piping with postulated flaws to the waterhammer or seismic loads will yield only trivial results. The value impact to the industry on these recommendations needs to be assessed.

2. Waterhammer

   Section 4.3 states that anticipated waterhammer events should be combined with earthquakes and plant dynamic events. We suggest that the SRSS method be mentioned as appropriate for combining these dynamic effects for the concurrent events.

Sections 4.3 and 7.4 discuss unanticipated waterhammer events in a very confusing manner. We suggest that unanticipated or accident events not be included in the design basis, but that all probable waterhammer events be clearly identified and included in the design basis.

In Section 7.4, crossing the disciplinary nature of waterhammers that identifies the exemplary major events, appears to be out of place and/or incorrect.

3. Independent Support Motion Method

Clarifications, references, and acceptance criteria are needed on the suggested "Group", "Group Responses", and "Algebraic Summations".

4. High-Frequency Response Combinations

Definition of high-frequency responses and the justification of algebraic summation are needed. Since most of the modes in a large piping system are closely spaced and governed by the present absolute summation rule, an option to allow SRSS for all high-frequency modes as proposed by BNL should be studied.

5. Nonlinear Analysis

Generally, nonlinear analysis is a time history analysis, which should not be tied to frequency as stated in 3.1.8. The concept of limit stop (gap between pipe and support) is useful in pipe rupture analysis (whip and jet impingement), but it is not practical for a nonlinear analysis of the piping system as stated in 8.4.(3).

6. Strain Rate Effects

Recommendation No. 5 in Section 3.1: Strain rate effects should not be considered for dynamic loading of piping in nuclear power plants. We suggest this recommendation be deleted.

In Section 6 we don't agree with strain rate effects being appropriate for piping in nuclear power plants. Paragraph 6.4, 2nd item - Do not add the statement to Section 3.6.2 iii 2.a of Standard Review Plant. We don't agree that up to 10 percent increase is appropriate either.

7. Additional Studies

   o To adopt the PVRC Task Force recommendation of composite ARS, i.e., use of 5 percent damping for frequencies $\leq$ 10 Hz, 2 percent damping for frequencies $>$ 20 Hz, and use linear interpolation between 10 and 20 Hz.

   o When postulated rupture of reactor coolant loop piping may be excluded from the design basis, why is this exclusion limited to short term effects only? Perhaps more investigations should be conducted to better define the need to consider this effect for containment and compartment pressurization effects.

Very truly yours,

Louis Nieh

Louis Nieh
Consulting Engineer

CC: Pat Higgins
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DR. N. W. EDWARDS, P.E.
President

June 28, 1984
NWE-84-071

Mechanical/Structural Engineering Branch
Division of Engineering Technology
Office of Nuclear Regulatory Research
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Dr. John A. O'Brien, Chairman
Task Group on Other Dynamic Loads and Load Combinations
US NRC Piping Review Committee

Subject: Review of Draft Task Group Report

Reference: March 2, 1984 Letter, N. W. Edwards to D. Landers, Providing Comments on Task Group Consultant Position Papers

Dear Dr. O'Brien:

We at NUTECH appreciated the opportunity to review the draft Task Group Report made available via your May 30, 1984 memo. Several of our engineers have reviewed the Staff recommendations, foreign information, and consultant position papers provided. Although we believe the Staff recommendations made in this draft still leave large amounts of selective conservatism in the piping design process, we also recognize that significant improvements are proposed. The Task Group draft report does suggest some "first steps" to be taken. The key point is that the process of establishing Staff position statements, aimed at achieving an improved balance in the piping design process, should begin right away with whatever material is acceptable to support some change.

The NUTECH comments provided in Reference 1 would still apply to the material in the Task Group report. We all must keep in mind that this Task Group report addresses only one segment of the factors that can influence the overall design. One of the major reasons nuclear plant piping design is in need of some "overhaul" is because there has been a tendency to focus too much attention on single technical issues or on very narrow aspects of the design process, causing a lack of consideration for the overall balance needed. It is hoped that a lesson has been learned and appropriate consideration will be given to other Task Group inputs when the Piping Review Committee compiles the single set of criteria statements for use in evaluating plant piping designs.

It is important to follow up on the work undertaken by the Piping Review committee and its task groups. Although the effort to date has been substantial, there will be additional issues which should be resolvable when considering the compensating aspects of other factors or with minimal additional study. We encourage the involvement of representatives who actually perform the design process for these programs and future programs of this type.

Thank you for allowing us this opportunity to provide comments. NUTECH would be pleased to be an active participant in this sort of activity for the other task groups, or any other related activity affecting the material-structural-mechanical aspects of nuclear plant design.

Very truly yours,

L Hibon for

N. W. Edwards

NWE/d

nutech

PSE-84-056

Westinghouse Electric Corporation

Water Reactor Divisions

Plant Engineering Division

Box 355
Pittsburgh Pennsylvania 15230

June 26, 1984

Mr. J. A. O'Brien, Chairman
U.S. Nuclear Regulatory Commission
Mail Stop NL-56150
Washington, D. C. 20555

Dear Mr. O'Brien:

SUBJECT: Comments on Draft NUREG-1061, Vol. IV - Evaluation of Other Dynamic Loads and Load Combinations

The NRC Staff recommendations in the subject draft NUREG represent a significant step forward in piping design by taking advantage of the latest available technical data and expert opinions. We are pleased to have the opportunity to provide comments on this draft NUREG. Attachment 1 provides comments on the NRC Staff recommendations for revison to present criteria and additional study. These comments represent our major concerns about the new positions. Attachment 2 provides comments on the technical papers in Appendix A of NUREG-1061. In addition, a meeting between Westinghouse and the NRC Staff has been scheduled to discuss details and definitions that would become a part of future NRC criteria.

If further clarification of our comments is needed, please contact us.

Very truly yours,

K. C. Chang, Manager
Piping Systems Engineering

Approved: for T. C. Esselman
D. A. Bartol, Manager
Engineering Mechanics

/hmb

cc: T. C. Esselman
J. J. McInerney

ATTACHMENT 1

Westinghouse comments and suggestions are provided along with major concerns on the following areas:

1. Executive Summary
2. Response Combinations
   Major Concerns
   a. There is too much emphasis on absolute sum method for ISMA which leads to excessive conservatism.
   b. There is no specific method described for calculating the high frequency mode response for the ISMA method.
   c. The method of combination of groups for dynamic and pseudo-static responses should be the same.
   d. The method of modal combination should allow for algebraic signs in the closely-spaced modes.
3. Stress Limit/Dynamic Allowables
4. Event Combinations

ATTACHMENT 1

1. EXECUTIVE SUMMARY

1. Section 3.1, Item 3:

   We believe that "the support motion method" is intended to mean the seismic spectral analysis method for structures subject to non-uniform excitation". Further, any such method has a very detailed set of requirements for phasing characteristics.

2. Section 3.1, Item 4:
   This should precisely define and specify what is meant by "present square root sum of the squares", and "any combinational sequence". Alternatively, this item can refer the reader to another reference for the precise definition and specification.

3. The title of Draft NUREG-1060 Vol. IV contains the word "other". Use another appropriate word(s) in place of "other".

4. The draft NUREG-1060 Vol. IV uses the following phrase or a phrase similar to the following phrase at many places throughout the body of its contents:
   "Multiply supported piping with "independent" inputs".
   It is not clear what the word "independent" means or why it is used. Does it mean "statistically <u>independent</u>"? It appears that the intent is to say "non-uniform" inputs.

2. Response Combinations

a. Section 5.3, Page 3:

Comment on fifth sentence:

Based on more in-depth review, it is Westinghouse strong opinion that this sentence be replaced with the contents of Technical Comment A.1 in Attachment 2.

b. Section 5.3, Page 4, Line 5:

- Insert after "leading to unconservatisms", the following:

"On the other hand, the Position Paper in Appendix A by R. P. Kennedy recommends that the combination of groups for the pseudo-static response be performed by retaining, if available,"the relative phasing of support motions" of the building structures".

c. Section 5.4, Page 5, 2nd Paragraph:

- Wording should be changed to reflect comment A.1 in Attachment 2.

d. Section 5.4, Page 5, 3rd Paragraph:

- Replace the first two sentences of this paragraph with the following:

"Group responses for pseudo-static response should be combined in the same manner as for the inertial response".

e. Section 5.5, Page 6:

- Add the following new item:

"Additional effort is needed on the proper treatment of the pseudo-static component. This component is currently considered a primary load for several components (e.g., pipe supports), even though the type of failure for this portion of the response is not well defined."

f. Appendix A of "Position Paper on Response Combination" by R. P. Kennedy, March, 1984.

- The equations in Appendix A apply to uniform spectra excitation. Acceptable method(s) for high frequency mode response calculation for the ISMA method should be added. Reference 3 in comment A.1 of Attachment 2 provides one such method.

3. Stress Limits/Dynamic Allowables

a. Section 6.5, Page 4:

- replace "none" with "Studies should be carried out in testing of standard piping components and weldments to determine inelastic response characteristics and allowable strains. The application of such allowables and the development of simplified inelastic analysis methods will provide an accurate and realistic design."

4. Event Combinations

Comments

The latitude to use probability for event combination criteria is a meaningful step to a more reasonable definiton of faulted load combinations. It is hoped that as more data is gathered and more analyses performed that the same philosophy is used and accepted on auxiliary piping systems. The type of break as well as the postulated location of the break should be studied with a coordinated philosophy based on probabilities used for both. The elimination of arbitrary intermediate breaks would be a welcome extension to the work on the elimination of the DEGB on the primary system.

The recommendations on waterhammer are reasonable, but they would be more useful, if they were more specific.

ATTACHMENT 2

Westinghouse comments and suggestions are provided on the following technical papers in NUREG-1061:

1. Position Paper on "Stress Limits/Dynamic Stress Allowables for Piping", by E. C. Rodabaugh.
2. Position Paper on "Response Combinations", by R. P. Kennedy.
3. Position Paper on "Piping System Dynamic and Thermal Stress Response Induced by Thermal-Hydraulic Transients", by R. C. Guenzler.
4. Position Paper on "Water Hammer and Other Dynamic Loads", by A. W. Serkiz.

ITEM 1 - Position Paper on "Stress Limits/Dynamic Stress Allowables for Piping", by E. C. Rodabaugh

<u>Comments/Concerns</u>

1. Other issues that should be addressed in developing a new stress limit/dynamic allowables are:
   - strain hardening
   - cyclic load failure

2. Realistic damping, such as recommended by the PVRC Sub-Committee should be allowed to be used in piping analysis. (This will probably be in another Position Paper).

Item 2 - Westinghouse Comment on "Position Paper on Response Combinations", by R. P. Kennedy.

A. Major Technical Comments

1. Section 2.1.1, Page 4, except top nine lines, and entire Page 5, except last seven lines:

This section relies very heavily on the work by Brookhaven National Laboratories (BNL). The BNL work has not been widely studied or evaluated, since Reference 29, on Page R-3, is not widely available.

A paper by Drs. Subudhi and Bezler of BNL (see Reference (1) below) studied a simple problem and proposed three methods for combination of grouped responses; namely, algebraic, square-root-sum-of-squares (SRSS) and absolute sum. This paper discussed some preliminary conclusions regarding the group combination methods and the pseudo-static component of the responses. It did not provide or discuss methodology for a more general and yet a practical situation involving various grouped responses, all of which simply cannot be subjected to just one of the above three proposed combination methods.

It should be noted that there is an extensive amount of research and development related to seismic spectral analysis for structures subject to non-uniform excitation. This research and development has been ongoing for many years in the U.S.A., as well as abroad. This is evident from the papers by Drs. K. M. Vashi and C.-W. Lin (See References 3 and 4 below). In view of this, the write-up in Section 2.1.1 is very limited because it relies on research effort of only BNL and because it does not utilize other research and development work mentioned above. This situation is not acceptable. Our recommendation is summarized below.

Briefly speaking, use the algebraic combination within a group and for two or more groups judged to be proportionally related. The SRSS combination is applicable for groups judged to be uncorrelated Absolute combination may be used only as a last resort in the absence of another more realistic combination method. Westinghouse definitions of a group are illustrated by the following examples; support response spectra from the same building with similar response spectral shape, or spectra at supports with elevations and locations in close proximity, where it is judged that building responses from the same mode dominate.

Based on the above, it is our strong opinion that changes be considered to Section 2.1.1.

References

1. Subudhi, M., and Bezler, P., "Seismic Analysis of Piping Systems Subjected to Independent Support Excitation", Pages 21 to 30 of, "Seismic Analysis of Power Plant Systems and Components", the ASME 4th National Congress on Pressure Vessel & Piping Technology, PVP-Vol. 73, Portland, Oregon, June, 1983.
2. Kennedy, R. P., "Position Paper on Response Combinations", SMA 12209-OB, Structural Mechanics Associates, Newport Beach, California, December,1983.
3. Vashi, K. M., "Seismic Spectral Analsyis for Structures Subject to Non-Uniform Excitation", ASME Paper 83-PVP-69, ASME-PVP Conference in Portland, Oregon, 1983.
4. Lin, C.-W., Loceff, F. "A New Approach to Compute Spectrum Response with Multiple Support Response Spectra Input", Nuclear Engineering and Design, 60 (1980), pp 347-352.

2. Section 2.1.5:
For combination of groupings of support displacement (seismic anchor motions) responses, apply essentially the same approach as described in various comments above. The only exception is that there is no modal combination involved.

3. Section 2.2, Page 11, Equation 4:
Change β_j' to β_j.

4. Section 3.4, Page 30:
Include another alternative to Rule 2 as follows:

"Alternatively, one may represent the combined response of all modes with frequencies equal to or greater than f^r by the full static response of the system subjected to force equal to mass times the zero period acceleration."

5. Page 31, Table 1:
Since this table reflects the interim NRC recommendations which are expected to change shortly, we have not provided any detailed comments.

6. Page 33, Table 3:
Suggest that this table be rewritten to incorporate comments (1) through (4) above.

7. Section 2.2, Page 9:
Another method of combining closely-spaced modes, which is similar to the DSC and CQC methods and is supported by Westinghouse, is described below and is proposed for inclusion in the NUREG-1061.

In order to account for the effects of any closely-spaced modes that may be present, the resultant response of interest for design purposes due to excitation by a given earthquake component is obtained by the following modified square-root-sum-of-squares (SRSS) combination of the corresponding mode-by-mode maximum responses due to the earthquake component under consideration. In equation form, the modified SRSS combination, which degenerates to the regular SRSS combination in absence of closely-spaced modes, is represented by:

$$R_i = \left[\sum_{k=1}^{N} R_{ik}^{2} + 2\sum_{j=1}^{S}\sum_{\ell=M_j}^{N_j-1}\sum_{n=\ell+1}^{N_j} R_{i\ell}R_{in}\epsilon_{\ell n}\right]^{1/2}$$

where R_i = value of combined response for ith direction excitation component

R_{ik} = response for direction i, mode k

N = total number of modes having frequencies lower than the zero-period-acceleration (ZPA) frequency f^r

S = number of groups of closely spaced modes. The groups of closely spaced modes are formed such that the difference between the frequencies of the last mode and the first mode in the group does not exceed 10 percent of the lower frequency. Groups are formed starting from the lowest frequency and working towards successively higher frequencies in such a way that no one frequency is to be in more than one group.

M_j = lowest modal number associated with group j of closely spaced modes

N_j = highest modal number associated with group j of closely spaced modes

$\epsilon_{\ell n}$ = coupling factor defined below

$$\epsilon_{\ell n} = [1 + (\frac{\omega_\ell' - \omega_n'}{\beta_\ell'\omega_\ell + \beta_n'\omega_n})^2]^{-1}$$

$$\omega_\ell' = \omega_\ell[1 - (\beta_\ell)^2]^{1/2}$$

$$\beta_\ell' = \beta_\ell + \frac{2}{\omega_\ell t_d}$$

ω_ℓ = frequency of closely-spaced modes ℓ (rad/sec)

β_ℓ = fraction of critical damping in closely-spaced mode ℓ.

t_d = duration of the earthquake (sec). This parameter is plant-specific.

B. Other Technical Comments

1. Section 1.1, Page 1, Item 2:
   The following definition is suggested for use throughout the report. "High frequency modes are those modes with frequencies equal to or greater than the frequency at which spectral accelerations begin to reduce to about the zero period acceleration (ZPA)."

2. Section 1.1, Page 1, last four lines:
   Regulatory Guide 1.92 does not differentiate between well-spaced modes, closely-spaced modes or high frequency modes. Note that the SRP and Regulatory Guide 1.92 require inclusion of all significant modes including high frequency modes.

3. Section 2.1.1, Page 3:
   Suggest that the fourth sentence be deleted since there are many reasons why the ISMA technique has recently come into vogue, including being more realistic and more technically rigorous.

4. Section 2.1.1, Page 3, fifth sentence:
   Provide clarification on how a single response spectra is selected for a group of supports. Confirm that the contributions to the response of motions at various supports within a group are algebraically combined. (See Major Technical Comment (1) in Part A for clarification).

5. Section 2.3.1, Page 13, same as Comment (1) above.

6. Section 2.3.2, Page 15:
   Clarify the meaning of the word relative and its subsequent use in describing R^P and R_i^P.

7. Page 32, Table 2:
   A. For grouping method, incorporate the following definition of a group in place of the one that is in the table:

"The group of closely-spaced modes is formed such that the difference between the frequencies of the last mode and the first mode in the group does not exceed 10% of the lower frequency. The group is formed starting from the lowest frequency and working toward successively higher frequencies in such a way that no one frequency is to be in more than one group."

B. Modify the second sentence under C_{jk} column for 10% method as follows:

"If modal frequencies ω_j and ω_k satisfy the following relation then $C_{jk} = 1.0$:

$$\omega_j - \omega_k \leq 0.1\ \omega_k \text{ and } \omega_j \geq \omega_k."$$

8. Section 3.1, Page 28, Item 3:
Clarify the meaning of "significant" on Line 10. See also comment (1) above.

9. Section 4, Page R-3:
Include references 3 and 4 from Comment (1) in Part A above.

10. Appendix A, Page A-3:
Suggest a change to last sentence as follows:
"The total response from the combined higher modes are then combined by SRSS rule with the total response from the combined lower modes."

11. Appendix A
The last paragraph of Section 2 on Page A-2 should be deleted and replaced with the following:
"If, for all DOF i, this fraction $|K_i|$ is equal to or less than 0.1, one can exclude Step 3 below and neglect the response from higher modes (with $m > M$). If, for any DOF i, this fraction $|K_i|$ exceeds

0.1, one should include the response of higher mode (with $m > M$) as described in Step 3 below.

Item 3: "Position Paper on Piping System Dynamic and Thermal Stress Response Induced by Thermal Hydraulic Transients" by R. C. Guenzler

Comments/Concerns

1. For water (or steam) hammer type events, the time-step for thermal hydraulic calculations should be equal to or less than the wave travel time across the smallest fluid volume length. For water slug discharge events, a time-step that results in stable solutions should be utilized. Comparison to test data should be made if data is available.

2. Simultaneous valve actuation cases are often investigated. It is agreed that the probability is small of other opening sequences producing significantly greater loadings.

3. Careful consideration of uncoupled valve/piping thermal hydraulic response is adequate for system evaluation.

4. From a structural analysis point of view, the time-step size should be sufficiently small to closely approximate system response to the applied hydrodynamic forces.

Suggestions

(None)

Item 4: "Position Paper on Water Hammer and Other Dynamic Loads", by A. W. Serkiz.

Comments/Concerns

1. It is agreed that efforts to reduce the incidence of unanticipated water hammers should continue.

Suggestions

1. The system designer should include any definable water hammer event in the preparation of design and operational specifications, in order to provide protection against unanticipated water hammers.

bnl

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June 1, 1984

Dr. John O'Brien
US Nuclear Regulatory Commission
5650 Nicholson Lane
Rockville, MD 20852

Dear Dr. O'Brien:

Dr. M. Subudhi and I have reviewed the preliminary copy of the "Staff Recommendations on Response Combinations" transmitted to BNL. We were pleased to see that the majority of the recommendations we advanced in NUREG/CR-3811 concerning multiply supported piping with independent seismic inputs were accepted. We feel that the revisions will provide more realistic estimates of piping behavior.

Our recommendation that group responses should be combined by the SRSS method when computing the dynamic component of response was not accepted. Instead the absolute sum method, with exceptions when the groups are phase uncorrelated, or the groups are in different buildings, is being recommended. We assume that by selecting the absolute sum method the staff has elected to assure the conservative prediction of the dynamic component of response. If that is so, we are confused with the exception concerning different buildings. In our case studies the RHR model incorporated an interface between two types of structure and the BNL model BM2 involved two building structures. For both of these cases we noted that the degree of conservatism exhibited by the dynamic response estimates were marketly reduced in the vicinity of the structure interface. In fact, for these situations only the absolute sum method could be relied upon to provide conservative response estimates. In light of this, we interpret the staff recommendation as providing leniency in just that situation where more stringency may be appropriate.

In the BNL study the degree of phase correlation between support groups was not assessed. For the two LLNL models, for which the bulk of the results were developed, the information necsssary to permit this assessment was not available. However, for supports contained within a single structure it seems reasonable to assume that the support groups exhibit phase correlated motions, at least for the dominant modes. For these situations, cases where the piping was contained within a single structure, the preditions of the dynamic component of response, by all group combination methods, exhibited increased levels of conservatism. For these situations the SRSS group combination method was clearly acceptable and the absolute group combination procedure very conservative. This finding seems again contrary to the staff recommendation which requires absolute summation for phase correlated support group motions.

As you suggested, M. Subudhi did confer with representatives from Westinghouse. Their view concerning groups in different structures or phase uncorrelated are based on statistical considerations. They have requested a copy of NUREG/CR-3811 and it is anticipated they will comment on the recommendations advanced in that report.

In closing, we are pleased that the majority of the recommendations advanced in NURG/CR-3811 have been accepted. It is our opinion that the proposed recommendation requiring absolute summation between support groups in the computation of the dynamic component of response will increase the level of conservatism associated with the component beyond that inherent in current practice (envelope spectra method). Given that, it is anticipated that applicants will continue to use the envelope spectra method to compute the dynamic component of response and will adopt the staff recommendations in all other aspects.

Sincerely yours,

Paul Bezler / lg

Paul Bezler, Group Leader
Dynamic Response Evaluation Group

jm
cc: M. Subudhi

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13. ABSTRACT (200 words or less)

This report deals with six topical areas: Event Combinations, Response Combinations, Stress Limits and Dynamic Allowables, Water Hammer Loadings, Relief Valve Opening and Closing Loads and Piping Vibration Loads. Recommendions prepared by the staff were based on consultant position papers and industry comments and treat revisions to present NRC requirements and directions for future research. Foreign information was obtained from sources in Belgium, Canada, France, Italy, Japan, Sweden and the Federal Republic of Germany. In addition, the report contains qualitative value impacts for the proposed recommendations. This report was developed over a period of approximately one year, and partially fulfills and complies with the requirements of the July 13, 1983 memorandum from the Directors of the Offices of Nuclear Reactor Regulation and Nuclear Regulatory Research to NRC's Executive Director for Operations.

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