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U.S. NUCLEAR REGULATORY COMMISSION

RESPONSE TO FREEDOM OF INFORMATION ACT (FOIA) REQUEST

NRC FOIA REQUEST NUMBER(S): FOIA - 88-464

RESPONSE TYPE: [X] FINAL [ ] PARTIAL

DATE: NOV 4 1988

DOCKET NUMBER(S) (if applicable):

REQUESTER: Ms. Nancy G. Chapman

PART I. – RECORDS RELEASED OR NOT LOCATED (See checked boxes)

[ ]	No agency records subject to the request have been located.
[ ]	No additional agency records subject to the request have been located.
[ ]	Agency records subject to the request that are identified in Appendix ______ are already available for public inspection and copying in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC.
X	Agency records subject to the request that are identified in Appendix A * are being made available for public inspection and copying in the NRC Public Document Room, ~~1717 H Street~~ 2120 L Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
[ ]	The nonproprietary version of the proposal(s) that you agreed to accept in a telephone conversation with a member of my staff is now being made available for public inspection and coying at the NRC Public Document Room, 1717 H Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.
[ ]	Enclosed is information on how you may obtain access to and the charges for copying records placed in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC.
X	Agency records subject to the request are enclosed. Any applicable charge for copies of the records provided and payment procedures are noted in the comments section. *
[ ]	Records subject to the request have been referred to another Federal agency(ies) for review and direct response to you.
[ ]	In view of NRC's response to this request, no further action is being taken on appeal letter dated ______

PART II.A – INFORMATION WITHHELD FROM PUBLIC DISCLOSURE

X	Certain information in the requested records is being withheld from public disclosure pursuant to the FOIA exemptions described in and for the reasons stated in Part II, sections B, C, and D. Any released portions of the documents for which only part of the record is being withheld are being made available for public inspection and copying in the NRC Public Document Room, 1717 H Street, N.W., Washington, DC, in a folder under this FOIA number and requester name.

Comments

* Copies of these records are enclosed.
You will be billed by the NRC's Division of Accounting and Finance for the amount of $53.61.

SIGNATURE, DIRECTOR, DIVISION OF RULES AND RECORDS

Donnie H. Grimsley

FREEDOM OF INFORMATION ACT RESPONSE FOIA NUMBER(S): FOIA 88-464 DATE: NOV 4 1988

PART II.B – APPLICABLE FOIA EXEMPTIONS

Records subject to the request that are described in the enclosed Appendices B are being withheld in their entirety or in part under FOIA Exemptions and for the reasons set forth below pursuant to 5 U.S.C. 552(b) and 10 CFR 9.5(a) of NRC Regulations.

	1. The withheld information is properly classified pursuant to Executive Order 12356 (EXEMPTION 1)
	2. The withheld information relates solely to the internal personnel rules and procedures of NRC. (EXEMPTION 2)
	3. The withheld information is specifically exempted from public disclosure by statute indicated: (EXEMPTION 3)
	Section 141-145 of the Atomic Energy Act which prohibits the disclosure of Restricted Data or Formerly Restricted Data (42 U.S.C. 2161-2165).
	Section 147 of the Atomic Energy Act which prohibits the disclosure of Unclassified Safeguards Information (42 U.S.C. 2167).
	4. The withheld information is a trade secret or commercial or financial information that is being withheld for the reason(s) indicated: (EXEMPTION 4)
	The information is considered to be confidential business (proprietary) information.
	The information is considered to be proprietary information pursuant to 10 CFR 2.790(d)(1).
	The information was submitted and received in confidence from a foreign source pursuant to 10 CFR 2.790(d)(2).
X	5. The withheld information consists of interagency or intraagency records that are not available through discovery during litigation. Disclosure of predecisional information would tend to inhibit the open and frank exchange of ideas essential to the deliberative process. Where records are withheld in their entirety, the facts are inextricably intertwined with the predecisional information. There also are no reasonably segregable factual portions because the release of the facts would permit an indirect inquiry into the predecisional process of the agency. (EXEMPTION 5)
	6. The withheld information is exempted from public disclosure because its disclosure would result in a clearly unwarranted invasion of personal privacy. (EXEMPTION 6)
	7. The withheld information consists of investigatory records compiled for law enforcement purposes and is being withheld for the reason(s) indicated. (EXEMPTION 7)
	Disclosure would interfere with an enforcement proceeding because it could reveal the scope, direction, and focus of enforcement efforts, and thus could possibly allow them to take action to shield potential wrongdoing or a violation of NRC requirements from investigators. (EXEMPTION 7(A))
	Disclosure would constitute an unwarranted invasion of personal privacy (EXEMPTION 7(C))
	The information consists of names of individuals and other information the disclosure of which would reveal identities of confidential sources. (EXEMPTION 7(D))

PART II.C – DENYING OFFICIALS

Pursuant to 10 CFR 9.9 and/or 9.15 of the U.S. Nuclear Regulatory Commission regulations, it has been determined that the information withheld is exempt from production or disclosure, and that its production or disclosure is contrary to the public interest. The persons responsible for the denial are those officials identified below as denying officials and the Director, Division of Rules and Records, Office of Administration, for any denials that may be appealed to the Executive Director for Operations (EDO).

DENYING OFFICIAL	TITLE/OFFICE	RECORDS DENIED	APPELLATE OFFICIAL SECRETARY	APPELLATE OFFICIAL EDO
James P. Murray	Deputy General Counsel for Hearings and Enforcement	appendix B Ex. 5		X

PART II.D – APPEAL RIGHTS

The denial by each denying official identified in Part II.C may be appealed to the Appellate Official identified in that section. Any such appeal must be in writing and must be made within 30 days of receipt of this response. Appeals must be addressed as appropriate to the Executive Director for Operations or to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and should clearly state on the envelope and in the letter that it is an "Appeal from an Initial FOIA Decision."

Re: FOIA-88-464

APPENDIX A
RECORDS MAINTAINED IN THE PDR UNDER THE ABOVE REQUEST NUMBER

1. 7/13/83 Memorandum from H. Denton and R. Minogue to W. Dircks, subject: Proposal for Reviewing NRC Requirement for Nuclear Power Plant Piping, w/enclosure. (40 pages)

2. 9/10/85 Memorandum from H. Denton to R. Minogue, subject: Schedule for Resolving and Completing Generic Issue No. 119 - Piping Review Committee Recommendations, (2 pages), w/enclosure: Prioritization Evaluation Generic Issue No. 119 - "Piping Review Committee Recommendations." (17 pages)

3. 6/9/86 Memorandum from G. Arlotto to T. Speis, subject: Recommended Actions Regarding Decoupling of Seismic and Pipe Rupture Loads. (2 pages)

4. 10/2/86 Memorandum to Distribution from G. Arlotto, subject: Termination of Proposed Revision to SRP 3.9.3. (2 pages)

Re: FOIA-88-464

APPENDIX B
RECORDS TOTALLY WITHHELD

1. 4/29/86 Note to Bob Bosnak from Bill Shields, subject: Revision of SRP 3.9.3. (1 page) Exemption 5.

2. 5/15/86 Note to Guy Arlotto from Joe Scinto, re: SRP 3.9.3. (2 pages) Exemption 5

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

JUL 1 3 1983

MEMORANDUM FOR: William J. Dircks
Executive Director for Operations

FROM: Harold Denton, Director
Office of Nuclear Reactor Regulation

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

SUBJECT: PROPOSAL FOR REVIEWING NRC REQUIREMENTS FOR NUCLEAR POWER PLANT PIPING

The proposal you requested in your May 31, 1983 memorandum is enclosed. We consulted with V. Stello, T. Murley and W. Kane in the proposal development, but staff from RES and NRR were mainly responsible for its preparation. Our consultant, S. H. Bush, provided significant inputs. Some of the key elements of the proposal are as follows:

1. We propose the establishment of an NRC Piping Review Committee made up predominently of NRC personnel from the various offices. With assistance from expert consultants, the committee will pull together all available information inside and outside the NRC and review all piping related requirements. Four task groups, one each dealing with pipe cracking, seismic design of piping, pipe breaks and other dynamic loads/load combinations for piping are proposed to be established under the NRC Piping Review Committee. We also propose to solicit views from industry and other interested parties in the pursuit of our objectives.

2. We suggest one cochairman from NRR and one cochairman from RES, R. H. Vollmer and L. C. Shao respectively, to administer the NRC Piping Review Committee. Approximately 12 individuals from NRC will staff the NRC Piping Review Committee, with representation from RES, NRR, IE, OELD and the Regions. Most of these will be assigned to one or more of the task groups. Our consultant, S. H. Bush, is our nomination for Vice Chairman. A total of 17 individuals from the NRC will be involved in the entire effort.

3. We estimate a period of approximately 12 months to complete the review and formulate recommendations. A four to five man-year effort from approximately 17 NRC staff is planned. Consultant costs are estimated at about $300,000.

CONTACT: L. C. Shao
443-5908

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4. Our primary deliverables will be recommendations, where appropriate, for revising the present requirements on nuclear power plant piping, and direction as to what work should be done to respond to issues not currently amenable to resolution. Our proposal does not offer quantitative and detailed value impact analyses for any recommendation, although qualitative statements regarding the cost and safety benefits that may accrue will be included. Value impact analysis, in our view, is warranted after our recommendations are acted on, and implementation of the recommendations are being undertaken.

We envision that regulatory actions regarding nuclear power plant piping currently under way will continue to be made on a timely basis. It is strongly urged in our proposal that no such actions be delayed or deferred pending the recommendations to be delivered as a product of our proposal.

H R Denton

Harold Denton, Director
Office of Nuclear Reactor Regulation

Robert B Minogue

Robert B. Minogue, Director
Office of Nuclear Regulatory Research

Enclosure: Proposal

PROPOSAL

FOR

REVIEW OF NRC REQUIREMENTS

FOR NUCLEAR PIPING

July 7, 1983

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PROPOSAL

TABLE OF CONTENTS

I. INTRODUCTION

A. General Background

Several of the current regulatory positions relating to piping design were developed without significant data. As relevant data became available, the conservatism of some positions became apparent. Moreover, the low probability of postulated events, such as the full flow area break, when examined using probabilistic or deterministic fracture mechanics and probabilistic risk analysis, was made evident. These conservative positions have resulted in a large number of massive pipe whip restraints, component supports and snubbers which stiffen piping systems. Since stiff piping systems generate high thermal stresses and nozzle loads and can be more adversely influenced by construction, maintenance and inspection errors, many experts believe they diminish overall safety.

Other areas covered by regulatory positions, such as seismic criteria pertaining to piping, load combination criteria, construction of floor response spectra and damping values, are in need of a reassessment which could lead to revisions that would reflect a more realistic response of piping systems to faulted or extreme accident conditions.

While experimental and analytical evidence confirms the conservatism of some positions, service experience may raise questions concerning our knowledge of the actual response of piping to failure mechanisms such as intergranular stress corrosion cracking (IGSCC) in large BWR piping. This latter condition is exacerbated by the uncertainty associated with crack detection and sizing of IGSCC in austenitic stainless steel when using conventional ultrasonic testing procedures.

An example of changing positions in the United States made on the basis of the availability of significant experimental data and analytical studies is the proposed elimination of the full flow area break as a design criterion in certain PWR primary systems. The greatly expanded data base pertaining to nuclear piping and its response to operating and accident conditions should permit an objective review of existing regulatory criteria.

B. Regulatory Issues

Four groups of regulatory issues have been identified as indicated below:

(1) Pipe cracking due to intergranular stress corrosion has occurred more extensively than previously forecasted in larger-diameter piping. Issues relate to inservice inspections (NDE), evaluating repair techniques (including replacement materials), and allowing continued operation. Pipe cracking due to vibrations in small-diameter pipes and thermal fatigue are also issues, as are actions NRC may require to reduce the potential for pipe cracking.

(2) Seismic design issues relate to pipe damping and the fact that the OBE, although not directly safety related, usually controls design because of lower allowable stress levels and lower damping values. Also considered are criteria for piping with multiple independent supports, peak broadening requirements for floor response spectra, and industry design practices.

(3) Pipe break issues relate to requiring full flow area pipe breaks, determining break locations, and replacement criteria.

(4) Certain load combinations, particularly the LOCA plus SSE load combination, represent a severe design requirement leading to massive supports on piping. No studies support a causative relation between pipe break and earthquake, and for the particular case of the primary loop of a PWR, it has been demonstrated that earthquakes are extremely unlikely to induce a full flow area pipe break.

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

II. OBJECTIVE

The objective of this proposed review is to evaluate current piping regulatory requirements for light water reactor nuclear power plant design using available domestic and foreign information in order to provide recommendations on where and how we should modify our current requirements, and to identify areas for further action. The review will not impact on existing ongoing regulatory actions prior to acceptance of the final report, nor impede the resolution of any specific piping problem. However, some preliminary recommendation* approved by NRC management may be utilized on a case-by-case basis.

III. SCOPE

The scope of this review covers all safety-related piping systems and those high energy systems which are important to safety in new and operating nuclear plants. The review will be performed on a system integrated basis considering all ongoing programs. Safety-related piping systems are defined as those piping systems needed to assure the integrity of the RCL pressure boundary, to shut down the reactor and maintain it in the shutdown condition and to mitigate the consequences of accidents. (See USNRC memo dated November 20, 1981, to Distribution from H. R. Denton, entitled "Standard Definitions for Commonly Used Safety Classification Terms.")

IV. APPROACH

Focus will be given to identifiable problems arising from present requirements and from plant operating experience. The suggested approach is to establish an NRC Piping Review Committee, gather all available information, and with the help of consultants and industry, evaluate this information for use in the regulatory process. By conducting a review of existing NRC piping requirements, a report will be prepared which will present recommendations and conclusions relating to NRC nuclear power plant piping criteria.

The organizational structure, the plan of attack, and the report contents associated with this approach are described as follows:

A. Organization

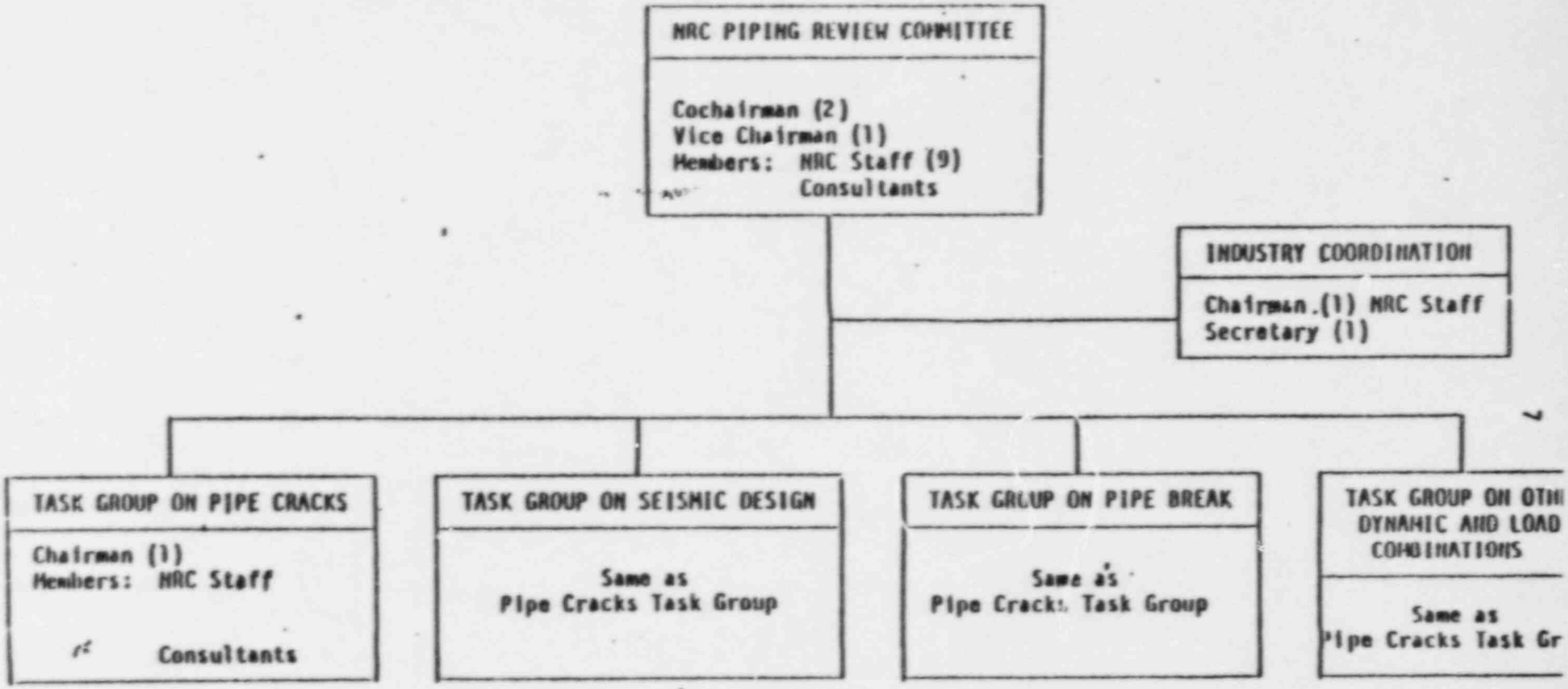
Our organization attempts to minimize impact on existing NRC programs carried out by NRC personnel, to maximize compliance with the intent of the assessment and to yield maximum credibility to the report. The following are essential to the proposed organization:

1. the substantial use of consultants to prepare position papers, review or assess submitted data, and prepare summaries for further reivew;
2. the establishment of an NRC Piping Review Committee reporting directly to the Director, NRR made up predominantly of NRC personnel from NRR, RES, I&E, AEOD, ELD, ACRS and the Regional Offices. The ELD representative will serve as a consultant, and the ACRS representative will be an observer. This Committee is expected to involve about 12 individuals. There will be four

technical task groups, each group responsible for a specific technical issue (see the organizational chart on page 7). About 17 individuals from NRC will be involved in the task groups and the NRC Piping Review Committee. Expert consultants in the fields of seismic design, piping design, systems, fracture mechanics and metallurgy will be used to critically evaluate and review the proposed recommendations and conclusions;

3. the coordination with industry and other interested parties in order to obtain information on operating experience, design, construction, maintenance and inspection problems and research results. Comments on the proposed recommendations and conclusions will also be sought from industry. There have been considerable efforts related to proposed improvement of piping design criteria and performance on the part of AIF, PVRC and EPRI.

ORGANIZATIONAL CHART
FOR
REVIEW OF NUCLEAR POWER PLANT PIPING REQUIREMENTS



B. Plan of Attack

Several steps are considered necessary to develop a definitive report. These are:

1. A clear-cut definition of the problem in each problem area should be formulated. This proposal accomplishes this in preliminary fashion. Additional effort will be required.

2. It is recognized that current NRC decisions being made on specific plants will affect some of the issues to be reviewed. Obvious examples include Generic Issues and I&E Bulletins. The status of the Generic Issues and I&E Bulletins should be reviewed and reported. NRC-funded research impacting on regulatory decisions in these problem areas will also be reviewed and reported. It is assumed that consultants will be preparing the reviews, working with appropriate personnel within NRC. This output ultimately will become a part of an appendix addressing the specific problem area.

3. Efforts by industry and other interested parties in specific problem areas should be assembled and reviewed. Since organizations such as EPRI, INPO, PVRC, AIF, A/E's, NSSS vendors and Utilities have different interests, expertise and opinions, it will be desirable for the NRC Piping Review Committee to deal with one or two focal organizations which reflect the various views of the industry and the professional societies. It is anticipated that the NRC Piping Review Committee will discuss with industry groups how this objective can be achieved. Pertinent foreign activities in research and regulatory development appropriate to the problem areas should be reviewed and assembled. The combination of (1) regulatory efforts and (2) industry and foreign programs will lead to appendices covering each problem area. Responsibility for digestion of the information and preparation of the position papers will be given to selected consultants.

4. The NRC Piping Review Committee, as an entity, would have the responsibility for the preparation of the final report for submittal to the Director, NRR. They would rely on direct assistance from their consultants to facilitate report preparation.

C. Content

The following presents an outline of the contents of the final report delineating specifically what will and will not be included.

1. There will be several appendices, each addressing a given problem. These appendices would be developed from a literature review, operating experience, and ongoing programs as well as from submittals made by interested organizations.
2. A historical section would cover the initial recognition of the problem and a chronology of regulatory and industry actions including a detailed listing of such actions appearing as Branch Technical Positions, Regulatory Guides, Regulations, Generic Issues, or in National Standards and Codes such as ASME Sections III or XI.
3. A section containing an assessment of the adequacy of available evidence for developing appropriate conclusions and recommendations. Current research by NRC, industry and foreign sources will be reviewed, and future research deemed necessary to finally resolve the problem will be cited. This section and the preceding section would deal with the technical issues.

4. The final section would consist of conclusions and recommendations with application to specific plants, including legal input where required, for each problem and subproblem where the NRC Piping Review Committee feels it has sufficient confidence to offer guidance. Explicit statements will be written for those situations where the NRC Piping Review Committee feels it is unable to draw a conclusion, and the reasons why no conclusions can be made will be identified. The implication of any recommendations made and an indication of how they affect the reliability and overall safety of piping systems will be given. Conclusions and recommendations should be directly related to relevant Branch Technical Positions, Regulatory Guides, Regulations, and Generic Issues.

5. The report will not contain the following:

   (a) a quantitative and detailed value-impact analysis for any of the NRC Piping Review Committee's recommendations or conclusions; nonetheless, qualitative statements relating to the safety and cost benefits which may accrue will be attempted.

   (b) suggested specific wording to revised Branch Technical Positions, Regulatory Guides, Regulations, or Generic Issues.

V. CURRENT STATUS

The present status of the four defined problem areas is given in terms of the technical issues, the regulatory efforts, and the research objectives and scope. This status information is given entirely from the NRC perspective; in the report developed from this proposal, however, industry and foreign views will also be given.

A. Pipe Crack

1. BWR Pipe Cracking

As a result of stress corrosion cracking found in the smaller bypass lines in BWR recirculating systems (NUREG-75/067 and the implementation document NUREG-0313), and the concern over the susceptibility of larger-diameter piping in the same system, USI A-042 was established. A technical resolution was achieved as documented in NUREG-0531. The results presented in that report formed the bases for the implementation document, NUREG-0313, Revision 1 (for comment) which was later formalized to incorporate public comments. A multi-plant action was established (MPA-B05) to implement the staff positions contained in NUREG-0313, Revision 1.

During the process of implementation, and as a result of cracking found in larger-diameter piping at Nine Mile Point in 1982, NRC issued two bulletins (IEB 82-03 and IEB 83-02) requesting that further inspections be conducted coincident with plant specific refueling outages. Also, by memorandum dated December 12, 1982, the EDO assigned NRR the lead responsibility to review and evaluate the plant specific inspection results. Subsequently, the NRR staff proposed an action plan (which was commented on by RES, DST/NRR and DL/NRR) which called for a second revision to NUREG-0313, Revision 1, to incorporate the latest developments on the technical implications for continued operation of affected plants and the long-term resolution of the situation. A draft of this revision is scheduled to be available by July 15, 1983.

Although somewhat surprised by the extensiveness of cracking, the staff maintains the position that the basic phenomenon of stress corrosion cracking in SS-304 used in BWR piping systems is the same and has not been fully unexpected for larger-diameter piping, given longer operating time. This later conclusion was clearly stated in NUREG-0313, Revision 1, which also cited the cracking experience of larger-diameter piping at a German BWR in 1978 (FRN-7590-01-M).

2. <u>PWR Pipe Cracking</u>

Several instances of cracking in PWR feedwater piping were reported in 1979. Both industry and the NRC staff have independently studied the cracking that occurred in PWR feedwater piping and have concluded that the cracking mechanism is due to thermal fatigue. Augmented inservice inspections of feedwater piping in the vicinity of steam generator nozzles appear to be prudent in order to detect any cracks that might develop before more serious problems occur.

The PWR Pipe Crack Study Group's recommendations to reduce the cracking of feedwater lines caused by thermal fatigue are contained in Table I of NUREG-0691. The short-term solution requires that plants which have experienced cracking undertake augmented inspections. If long-term remedial actions are taken and prove to be effective, the inspection requirements may be reduced to ASME Section XI requirements.

while the staff encourages consideration of remedial measures, it could be some time before their effectiveness is demonstrated. Thus, it is believed that continued augmented inservice inspections of the feedwater lines in the vicinity of steam generator nozzles for Westinghouse and Combustion Engineering facilities should be implemented and continued until it is demonstrated by experience that they are no longer needed. (Babcock and Wilcox facilities have a different feedwater system design and have not experienced feedwater line cracking.)

The NRR staff is preparing guidance for augmented inservice inspections of PWR feedwater lines in the vicinity of steam generator nozzles. This guidance will be essentially the same as that developed by I&E in 1979 and as recommended by the Pipe Crack Study Group.

Combinations of fabrication, stress, and environmental conditions have resulted in isolated instances of stress corrosion cracking of low pressure schedule 10 type 304 stainless steel piping systems in PWRs. Resolution of this issue was published in NUREG-0691. Based on operating experience, it was concluded that current ISI requirements for thin-walled piping in PWRs are adequate.

Cracking was found in the normal makeup/high pressure injection nozzles of four B&W facilities. In each case, the associated thermal sleeve had been loose or was missing. The repair consisted of hard rolling in new thermal sleeves. Whether or not any further remedial measures are necessary will depend on future inspection results.

3. General Discussion

In summary, there are several pipe cracking mechanisms that exist in nuclear reactor facilities--vibration and/or thermal fatigue, steam erosion, and stress corrosion cracking. Some, such as thermal or vibration fatigue, are directly related to plant operating conditions, and once recognized, usually can be corrected by a modification in plant operating procedures. Others, such as intergranular stress corrosion cracking (IGSCC), represent a complex interaction between material properties, fabrication procedures and reactor environment so that there is no assurance that a change in one variable will necessarily resolve the issue.

The subject of IGSCC in BWRs will be further reviewed because of its complexity, potential safety implications, and its severe impact on the plant availability. Both the NRC and the industry have devoted a great amount of resources to study the phenomenon, and to explore various remedial measures both long and short term. A great deal of information has been generated. This information needs to be reviewed, independent of day-to-day licensing activities, in order to demonstrate the efficacy of measures taken by industry. Some of the issues that need to be examined are:

- The adequacy of ultrasonic examination methods for crack detection and sizing including qualification of personnel and procedures. The development and application of improved state-of-the-art inspection techniques.
- The potential for the crack to progress completely around the pipe and for the crack front to progress uniformly, raising the possibility of a DEGB
- Whether there are bending stresses that may promote preferential crack growth in one pipe quadrant leading to leak-before-break.

- The possibility that low temperature sensitization can occur from residual elements such as phosphorus, or sulfur rather than from carbon alone. This raises the issue of possible IGSCC in extra low carbon grades of piping. The experimental evidence is not considered sufficiently definitive to resolve this issue.

The NRC research program on the subject was initiated in 1981 and is expected to develop information needed to permit an independent capability for prediction, detection and control of pipe cracking in LWR systems. The program includes:

1. Development of the means to objectively and quantitatively evaluate leak detection systems and cracks through NDE;
2. Definition of the role of stress, metallurgical variables, and environment on pipe cracking susceptibility, including the influence of plant operations on these variables.
3. Evaluation of industry proposed fixes and repair procedures.

B. Seismic Design

Before seismic loads on piping can be computed, it is usually necessary to estimate seismic ground motions, perform a soil-structure interaction analysis and calculate building response, since piping is supported, in general, by the building, and the building in turn is supported by the soil. The discussion here and the proposed review are limited only to areas that are handled by mechanical and piping engineers. Due to large uncertainties in estimating seismic input soil structure interaction effects and building response and the lack of nuclear structure failure information under seismic environments, the natural tendency in seismic design is towards conservatism. This practice and design philosophy has been implemented at each stage of the seismic design chain for piping system design. The accumulative affect of these conservatisms has made nuclear piping systems more rigid than nonnuclear piping. A more rigid piping system normally is not beneficial for routinely occurring thermal transients. This unbalanced design between thermal and seismic events introduces a question as to whether the rigid nuclear piping is more or less reliable when the risks associated with all events are considered. In recent years, increasing concern with the relation between piping reliability and stiffness has led to a joint NRC and industry effort to assess the problem. Several issues have been identified and are under assessment. They are: damping, multiply supported piping, response spectrum peak broadening, the operating basis earthquake (OBE) definition, and industry design practice.

1. Damping

Damping requirements for piping system seismic design are given in Regulatory Guide 1.61. Damping factors are used in the seismic dynamic analysis to represent energy dissipation capability of piping systems. Due to the lack of confidence in our understanding of the parameters which influence damping, lower bound damping values were used. This resulted in an excessive number of pipe supports being used in design. Two types of supports are normally used for seismic design: rigid struts and snubbers. When rigid struts are used, the piping system is stiffened. A stiff piping system may be less reliable if thermal events are taken into account. When snubbers are used in seismic design, the failure of snubbers in the lock-up mode can introduce higher thermal stress on the system. The high failure rate of snubbers has caused concern regarding piping reliability even under seismic events. The best resolution to the problem is to establish realistic damping values to be used in design. Currently, the Damping Task Group of the PVRC Technical Committee on Piping Systems is conducting an in-depth assessment to generate realistic damping values from various piping test data. It is anticipated that within one year, a recommendation from the PVRC will be available for review. RES has two research programs underway to experimentally develop and interpret damping behavior in piping. Additionally, international cooperative efforts are being pursued.

2. Multiply Supported Piping

The Standard Review Plan (SRP 3.9.3) requires a very conservative procedure in order to compute the seismic effects of piping supported between floors or between buildings. Specifically, the inertial or dynamic component of the loading is calculated using an envelop spectra of all the supports, the psuedo-static or seismic anchor movement component of loading is calculated using the most unfavorable combination of support motions, assuming that peak displacements occur simultaneously, and the dynamic and psuedostatic components are combined by the absolute sum rule (in contrast to the SRP requirement, the staff has begun to accept the SRSS rule). This position was developed at a time when the urgency to develop a position did not allow for a full and complete understanding of the behavior of multiply supported piping subject to independent seismic inputs. Since that time, RES-funded studies have been completed, or are nearing completion, which indicate that the SRP position is unreasonable and that more realistic criteria can be established. This particular technical issue will likely be resolved within a year.

3. Response Spectrum Peak Broadening

To account for uncertainties in the natural frequencies of supporting structures, Regulatory Guide 1.122 requires that the peaks of floor response spectra be broadened, typically ±15%. It has been pointed out that for piping this could have the effect of placing a few modes at the peak floor response spectrum value, since piping natural frequencies tend to be relatively closely spaced. Because peaks in the floor response specta usually are sharp and pointed (with the consequence that no more than one mode could have the peak or near peak floor response spectrum value), the regulatory guide position imposes an unnecessary conservatism on piping design. It has been recommended that a more rational way of dealing with uncertainties in the natural frequencies of supporting structures would be peak shifting rather than peak broadening. With peak shifting, the shape of the floor response spectrum is not changed, but instead, the response spectrum is shifted backwards and forwards along the frequency scale within given limits, and the most unfavorable location selected for design. This procedure has the advantage of dealing with uncertainties in supporting structure frequencies without requiring that more than one mode be at the peak value of the floor response spectrum. As the procedure is already available, and as its adequacy has already been demonstrated under research performed by RES contractors, immediate implementation of this relaxation is possible.

4. Operating Basis Earthquake (OBE) Definition

According to 10 CFR 100, Appendix A, when an earthquake exceeding the OBE occurs, shutdown of the nuclear power plant is required. Licensees must then demonstrate that no functional damage has occurred to those features necessary for continued operation without undue risk to public health and safety. Because lower allowable stresses and damping values are permitted with the OBE, many piping designs at nuclear power plants are controlled by the OBE rather than the more severe Safe Shutdown Earthquake (SSE). Loads for both the OBE and the SSE are estimated using elastic dynamic analysis procedures; however, the allowable for the SSE implies local nonlinear behavior. It has been argued that it is unreasonable to allow an earthquake which must be resisted without sustaining damage to control the sizing and proportioning of pipe supports and other elements. While the two-earthquake approach to design is commonly adopted in many countries for nuclear reactor design, there is a growing sentiment that the OBE must be set to a lower value or that allowable stresses must be increased such that piping design is not controlled by an event not directly related to safety. This specific issue, once adopted for nuclear reactor piping, is likely to have important implications for other structural features at nuclear power plants.

5. Industry Practices

There are an infinite variety of different pipe support system designs and modeling practices which can be implemented to satisfy both ASME code and NRC requirements. These different designs and modeling practices, however, are not equally reliable or equally cost effective according to results obtained from RES-funded studies. The assumptions made in modeling piping system parameters, such as those regarding support stiffness, gaps, overlap techniques and so on, can affect the final result, but may be unimportant when other design conservatisms are present. It is expected that the NRC Piping Review Committee will explore this matter. It is common industry practice to implement the first iteration in the design process which satisfies the requirements rather than optimize the pipe support system in terms of reliability or costs. Nonetheless, procedures are becoming available which can achieve these goals. It is unclear at this time how the NRC should deal with these goals and three basic issues to be decided are: (1) does the NRC have confidence in the technical foundation in the optimization algorithms? (2) should the NRC merely encourage or rather enforce utilization of the algorithms? (3) is it necessary to revise industry practice in modeling if the conservatisms in design are removed? Recommendations from the NRC Piping Review Committee are sought on both these questions.

Staff regulatory efforts and their respective status consist of the following:

1. Generic Issues A-40 and A-41 were created. The short-term effort (A-40) is to revise SRP 3.7.1-3.7.3. The long-term effort (A-41) is the Seismic Safety Margins Research Program (SSMRP). The A-40 is ongoing and will be completed in FY 1984. The SSMRP has developed a comprehensive probabilistic methodology to assess plant seismic risk and will also develop simplified methods for calculating seismic risk.

2. Generic Issue A-13 was created for snubber operability. Solutions were proposed in NUREG-0371 which consists of evaluation of industry practices associated with snubber design and qualification testing, and development of Technical Specifications and regulation changes. SRP 3.9.3, STS 3/4.7.9 for PWRs and STS 3/4.7.5 for BWRs were revised. A draft Regulatory Guide to assure a high level of snubber operability is completed.

3. Generic Issue B-25 deals with piping benchmarks. The program is providing the staff with a capability to conduct independent verification of computer codes used by licensees for his dynamic analysis of ASME Class 1, 2, and 3 nuclear piping. Generic Issue B-25 is an ongoing program which has the capability to verify time domain and frequency domain analysis, elastic and inelastic analysis, and can use either multisupport spectra input or single-enveloped spectrum input. Test verification of the codes will be completed in FY 1984.

4. Generic Issue B-51 was created for assessment of inelastic analysis techniques used in piping and support analysis. Since the ASME Code permits Level D stress limits for low probability events, and large inelastic strains may occur under such stress limits, it is important that properly qualified analysis techniques are used. Generic Issue B-51 is an ongoing issue for continuous monitoring of the ASME Code rule changes.

5. Current regulations require that components and structures shall be designed for two levels of seismic loads, namely, the OBE and the SSE. Due to different requirements in load definitions, load combinations, and stress allowables, the OBE generally controls the design. The staff has investigated the implications of having two levels of seismic design and has proposed a revision to seismic requirements in a report to the Commissioners dated April 27, 1979.

C. Pipe Break

Current requirements specify that piping and supports of the reactor coolant pressure boundary be designed to accommodate the dynamic effects of postulated pipe break events, including a LOCA induced by a double-ended rupture of the largest primary pipe. Subsequent licensing positions set by Regulatory Guide 1.46 and SRP 3.6.2 also indicated that such large breaks should be universally postulated for all high energy lines inside and outside the containment, and at several locations per pipe run. Implementing these positions has resulted in installation of numerous pipe whip restraints, snubbers, and jet shielding structures, that may not provide additional safety, and in fact, may detract from safety in some ways. For instance, piping so designed is stiffer and experiences greater stress when subject to thermal expansion, and has become more inaccessible for inservice inspection. Thus, a revision of current pipe break postulation criteria is necessary in order to achieve better balance in design for both normal and abnormal plant situations. Staff efforts and their respective status consist of the following:

1. Generic Issue A-18 was created to upgrade pipe break criteria. The task has completed the following short-term objectives:

   (a) SRP 3.6.2 was revised which provides consistent pipe break criteria for high energy lines both inside and outside the containment.

   (b) Criteria for the pipe break exclusion region at the containment penetrations were defined, and pipe rupture loads to the guard pipe were investigated.

   (c) An investigation of the deleterious effects of pipe whip restraints and snubbers to piping normal operation were conducted and reported.

Generic Issue A-18 became inactive shortly after the TMI accident due to prioritization considerations. Its long-term objectives were carried on by several research programs including investigation of pipe whip, jet impingement loads, pipe-to-pipe impact tests, piping failure mechanisms, and pipe break probability.

2. Generic Issue 40 was created to review the safety concerns associated with pipe breaks in BWR scram systems. Options acceptable to the staff consist of establishing either (a) a low probability for the event, (b) acceptable consequences for the event, or (c) availability of alternate cooling systems and initiation equipment for the event. NUREG-0803 was issued which gives guidance for resolution on a plant specific basis. GE has recently provided additional information to strengthen the generic resolution using the argument of low event probability. The new submittal is currently under staff review.

3. Generic Issue 61 was created to review the concern of postulated break at the safety/relief valve line inside the airspace of a BWR Mark I and II containment, since the steam bypassed suppression pool will rapidly pressurize the containment. NUREG-0487 was issued which required Mark II plants under construction to perform a piping fatigue analysis. Operating Mark I plants were required to conduct periodic visual inspections. The staff considers that the most affective resolution is utilizing the containment spray. New requirements regarding operational procedures and/or design modifications of containment spray systems remain to be developed.

4. The staff is planning to revise its break postulation criteria in SRP 3.6.1 and SRP 3.6.2.

Piping failures generally occur at high stress and fatigue locations, such as at the terminal ends or at locations where corrosive environment and vibrational fatigue exist. The NRC, in the development of positions for postulating pipe rupture, selected stress level and fatigue usage factor as the criteria for deciding pipe rupture. For instance, for a high energy, Class 1 piping, pipe ruptures are required to be postulated at the terminal ends and at intermediate locations where the calculated maximum stress range exceeds 2.4 S_m or the cumulative usage factor is larger than 0.1. At least two intermediate locations must be selected. If two intermediate locations cannot be determined by the above stress or usage factor criteria, the two highest stress locations are then selected. Since these locations are not fixed early in the construction process, A/Es find it very difficult to implement such criteria.

As a result of the above requirements, 300 to 400 pipe whip restraints will have to be installed for a typical PWR plant. Total costs for design, procurement and construction of those restraints are estimated in the range of 20 to 40 million dollars per unit. For a plant under construction, these requirements increase construction cost and design complexity. For an existing plant, back-fitting would be required in order to meet current standards. Also, there are several negative aspects to having pipe whip restraints. For example:

1. Access for maintenance and inservice inspection is impeded, therefore increasing radiation exposure to inspectors.

2. Higher thermal stress caused by the pipe coming in contact with the restraints is introduced.

In 1979, NRC initiated studies in both NRR and RES to assess the likelihood of having a double-ended guillotine pipe break (DEGB) in the PWR primary loop.

NRC programs in the area of piping fracture mechanics will concentrate on validating ductile fracture analysis techniques, developing a comprehensive piping materials data base, and conducting pipe tests to determine the failure modes of piping for use in developing improved pipe break criteria.

To date, the NRC has sponsored the development of experimental techniques for characterizing ductile fracture properties of piping materials along with intermediate size pipe tests to validate the use of tearing instability fracture mechanics. This work has been performed at USNRDC in Annapolis. Development of experimental techniques for materials characterization included investigation of the effects of specimen geometry and development of single-specimen computer controlled unloading compliance J-resistance curve testing. Future work in this area includes determination of environmental and loading histories on ductile fracture properties. The pipe tests conducted at USNRDC were performed using 8-inch-diameter, circumferentially-flawed, A106 grade B piping loaded in four point bending. The loading train was made compliant through the use of springs so that conditions for both stable and unstable crack extension could be created. Using J-resistance curves generated from the pipe tests, tearing instability analysis accurately predicted stable and unstable crack extension. Future pipe tests to be conducted at USNRDC will be performed on 8-inch-diameter A106 piping with circumferentially-flawed welds.

During the next 3 years, NRC's major effort in the area of piping fracture mechanics will be the Degraded Piping Program. The objective of this program is to further validate and improve the ability of ductile fracture mechanics analyses to accurately predict the loading capacity and failure mode of cracked piping and to provide experimental data which can be used to develop improved pipe break criteria. Phase I of this program began in September 1981 at Battelle Columbus Laboratories. The Phase I effort included the development of improved ductile fracture mechanics analyses, review of previous and ongoing pipe fracture programs, and development and costing of a comprehensive pipe test program. Phase II of the program, to begin in the fall of 1983, will include low energy and high energy (typical LWR pressure and temperature conditions) tests. The high energy tests, in particular, will provide necessary data for developing improved pipe break criteria to replace the currently postulated double-ended guillotine break.

Finally, beginning in fall 1983, NRC will sponsor a data base development program which will include generation of necessary fracture mechanics properties for piping materials and development of a computerized data system.

In NRR's effort, a deterministic fracture mechanics assessment based on material toughness measurements was made to evaluate PWR primary loop piping integrity. The results of the study suggested that the rupture of PWR primary loop piping is unlikely. Results from RES's probabilistic assessment of the probability of a DEGB in the PWR primary loop due to direct and indirect causes confirmed NRR's findings. This study employed a probabilistic model to assess the likelihood of a DEGB caused by indirect causes, such as structural support system failure leading to the pipe rupture. The probabilities for both direct and indirect causes are very low for the Westinghouse PWR plants studied. This assessment was conducted on austenitic stainless steel piping used for Westinghouse plants. The methodology developed in this assessment is equally capable of handling ferritic steels. An effort is currently underway in RES to assess the ferritic steel CE and B&W primary loops. We expect similar conclusions will be reached for CE and B&W plants. In the letter from H. R. Denton to M. Edelman of the Atomic Industrial Forum, dated May 2, 1983, appropriate changes to current criteria on pipe rupture requirements for the PWR primary loop are indicated. West Germany has already replaced the full flow area break requirement with a 10% break area for pipe whip and dynamic loading of reactor vessel internals. The full flow area break is still required in Germany for containment sizing, ECCS design, and major component supports.

The probabilistic methodology developed by RES is capable of handling stress corrosion failure mechanisms in BWR primary systems. An effort is currently underway in RES to assess BWR primary systems. Historically, stress corrosion cracking has occurred in BWR Reactor Coolant Loop Piping, and preliminary evaluations suggest that the BWR full flow area break assessment will be less favorable than was the PWR assessment. Current planning is to complete the BWR Reactor Coolant Loop assessment by the end of 1984. Whether there will be any recommendations to relax present requirements is difficult to predict.

Piping systems other than the primary loop are more complex due to large variations in configuration, material, loading, environment, function, and support type. Piping systems can be divided into three groups: (1) those systems for which leak-before-break can be easily established, (2) those systems for which the operating environment and piping materials suggested that leak-before-break will be very hard to be established, and (3) those systems which belong to neither one of the two groups above; that is, systems where leak-before-break is marginally established.

For the first group, it should not be very hard to develop a technical basis to assess the intermediate break requirement. The available probabilistic methodology, with support from the planned degraded piping program to provide proper failure modes for certain selected piping systems, should be able to reach some conclusions in a reasonable time. In order to eliminate the intermediate break requirement based on the leak-before-break hypothesis, it is necessary that the reliability of leak detection capability be established.

For the other two groups, the decision will be more difficult. It is anticipated that a much larger effort will be needed to better understand piping failure mechanisms and develop preventive action which can improve piping reliability.

The genesis of the leak-before-break issue goes back to a traditional staff interpretation of General Design Criterion (GDC) 4 in the context of the definition of a LOCA as including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system (Appendix A to 10 CFR 50). This interpretation has permeated through various staff guidance documents; i.e., Regulatory Guide 1.46, SRP Sections 3.6.1 and 3.6.2, and has created a situation such that later vintage plants are required to have massive pipe whip restraints and older plants were found not able to take asymmetrical LOCA loads. The later concern over the asymmetrical LOCA loads led to the establishment of USI A-2.

As an attempt to resolve the USI A-2 issue, Westinghouse submitted topical reports, on behalf of 11 licensees of 16 PWR units (one of which is a CE designed plant), using advanced fracture mechanics to demonstrate that the detection of small flaws (postulated or real) either by ISI or by leakage monitoring can be assured before the flaws can grow to critical sizes which could lead to a large break area "equivalent" in size to a double-ended guillotine LOCA. The staff has evaluated these topical reports, independently verified by research program and technical assistance program the methodology contained in these topical reports, and has concluded that a valid technical basis exists for exempting these plants from the regulation as currently interpreted. An exemption package was prepared and comments incorporated, and will be ready for NRR Director's review and submittal to CRGR before the end of June 1983.

In parallel with the above action, the Division of Engineering staff has also drafted a Commission paper for NRR management review that proposes granting similar exemptions for other PWRs not covered by Westinghouse topicals.

The third step to be undertaken is to initiate a rulemaking to seek a long-term resolution of this situation. There are some ancillary issues, such as intermediate break criteria, which would naturally be resolved as the staff revises and develops those staff guidance documents (SRP 3.6.1, 3.6.2, and 3.6.x) to support rulemaking.

The majority of technical work is complete or has reached the stage that is deemed sufficient to support these changes. The NRC Piping Review Committee will be asked to review the results of all work sponsored either by NRC or by the industry in order to confirm the statement above. However, the staff efforts including the initiation of the rulemaking will not be held in abeyance pending the outcome of this proposed review.

D. Other Dynamic Loads and Load Combinations

Piping and supports are designed to withstand a spectrum of events, and events in combinations. GDC 2 requires that earthquake events should be appropriately combined with LOCA events. Licensing position delineated in Regulatory Guide 1.48 and SRP 3.9.3 specify that components and supports should be designed to accommodate individual and combined loads due to normal operating conditions, system transients, and postulated low probability events. Loads induced by earthquake, LOCA, and system transients vary with time and their peak responses for purpose of design were assumed to occur simultaneously. This conservative approach resulted in stiff piping systems which are not only costly, but also may not be very reliable. Because of this concern, AIF has established a subcomittee on Load Combination to address this issue.

For developing a more rational treatment of loads, load combinations, and stress limits, Generic Issue B-6 was created. The staff efforts to investigate and establish licensing positions on response combination methodology were completed and reported in NUREG-0484, Revision 1. Rules to justify the use of the SRSS method were delineated based on specific acceptable nonexceedance probabilities from cumulative distribution functions. These positions were applied in licensing reviews of Mark II and Mark III plants.

SRP 3.9.3 was revised to reflect new positions on load combinations and stress limits.

In addition, stress allowables specified in the ASME Code for piping design make no distinction between whether the load is static or dynamic. The staff plans to undertake a revision to SRP 3.9.3 in order to incorporate separate stress limits for static and dynamic loads.

The issue of load combinations became more controversial in recent years. Other dynamic loads such as BWR pressure suppression pool swell loads have further complicated the issue.

These changes have raised questions concerning implementation of new regulations, increased construction costs, increased radiation exposure to maintenance crews performing inspection and maintenance actions, and reduced reliability of stiffer systems under normal operating transients.

The major NRC concern regarding these issues can be summarized in four questions:

(1) Which events are required in the piping system design basis, and which events should be considered in combination?

(2) How do we define other piping dynamic loads, for example, their magnitude, duration, and frequency characteristics?

(3) What method should be used in combining piping system responses resulting from the above loads?

(4) For each combination case, what should be the most appropriate stress allowable for design to reflect the actual capacity of the material to withstand dynamic loads?

1. Combination of Large LOCA and SSE for the PWR Primary System - Research Information Letter No. 117 identifies the following results for the PWR primary system:

   (a) Fatigue crack growth due to all transients, including earthquakes, is an extremely unlikely mechanism for inducing a large LOCA. The contribution of earthquakes to the occurrence of this unlikely event is a small percentage of the total probability.

   (b) Leak probability is several orders of magnitude higher than double-ended guillotine break probability. This supports the lead-before-break hypothesis.

   As a result of this study, the staff plans to recommend appropriate changes to current criteria on large LOCA and SSE combination for PWR primary system. In its June 14, 1983 letter to the EDO, the ACRS stated, "We find ... the decoupling of the loss of coolant accident and seismic loads to be appropriate."

2. Combination of Pipe Break and SSE for Systems Other Than The Primary Loop in PWR Plants - For systems other than the primary loop in PWR plants, the contribution of seismic loads to pipe failure is unknown. From the failure mechanism point of view, pipe rupture due to seismic load is unlikely to occur except when the piping system is constructed of corrosion susceptive materials and the system is operated under fatigue conditions in a corrosive environment. After a careful review of piping failure data and results from existing fracture mechanics research, it is possible that an improved requirement on combination of pipe break and SSE can be established for systems other than PWR primary loop piping.

3. Combination of Large LOCA and SSE for BWR Piping Systems - A great deal of intergranular stress corrosion cracking (IGSCC) has been reported in BWR piping systems. In recent years, IGSCC on larger diameter piping has been reported. IGSCC represents a complex interaction phenomena between material properties, fabrication procedures, and the operating environment. A piping system under IGSCC attack will have higher rupture probability under a seismic event. The assessment of the combination requirement of large LOCA and SSE for BWR piping systems will have to take IGSCC effects into consideration. The probabilistic fracture mechanics methodology developed by RES is capable of evaluating the seismic contribution to pipe rupture under IGSCC environments.

4. Other Dynamic Loads - Other dynamic loads in piping, such as SRV discharge loads, vibrational loads, and water hammer loads, may require an assessment to determine how they should be treated individually or in combination. Particular attention is given to the high frequency content of these loads which could be damaging to equipment mounted on piping and also to support systems embedded in concrete. The frequency content of these loads normally exceeds that associated with earthquakes. The major NRC concern with regard to dynamic loads on piping has been directed to seismic loads; however, such loads are better classified as pseudostatic, and loads generated by water hammer, water slugging, or valve opening on closure represent loads more severe than seismic in terms of rate, and magnitude. In some instances, major splits or failures of piping have occurred in secondary and tertiary piping systems, and instances have been reported of pipe supports being torn from the walls because of water hammer, without, however, failing the piping.

5. Design Limits - Current piping system design is governed by the ASME Boiler and Pressure Vessel Code. The design and service allowables stated in the ASME Code were developed for static conditions. When the structural system is subjected to short duration dynamic loads with high strain rates, the capability of structures to withstand the loading effect can be much higher than the static limits. The PVRC Technical Committee on Piping Systems has a task group on dynamic allowables that is currently assessing the available data to better define the dynamic allowables for piping design. It is recommended that this task group closely communicate with the PVRC task group in this development and start to review the PVRC data base. It is anticipated that a technical resolution may become available from the PVRC for NRC review.

VI. SCHEDULE AND DELIVERABLES

Periodic status reports will be prepared as deliverables during the course of this effort. Additionally, informal discussions will be used to keep NRC management advised as to progress being made.

A draft final report will be available in about 9 months. Completion of all tasks is estimated to occur in about 12 months from the date of authorization.

The final product will be a report reviewing the status of nuclear piping in the context of regulatory and licensing actions. The chief product will be conclusions and recommendations. They will be directly correlatable to relevant regulatory positions such as Branch Technical Positions, Regulatory Guides, I&E Bulletins, Regulations, and Generic Issues. The impact of proposed actions on existing research will be discussed. Suggestions for future research will be made when it is concluded that no position may be taken until additional information becomes available.

As indicated previously, the conclusions and recommendations will be based on the technical judgment of the participants. They will not represent regulatory positions validated by quantitative value-impact studies and written in language appropriate to regulatory documents. These actions are assumed to constitute follow-on actions presuming the conclusions and recommendations are accepted. However, qualitative statements regarding the cost and safety benefits that may accrue will be included in the report.

VII. MANPOWER AND BUDGET

The NRC manpower estimate is 4 to 5 man-years, involving about seventeen individuals. Expertise will be drawn primarily from NRR and RES, with contributions from I&E, AEOD, ELD, ACRS and the Regional Offices. Additionally, our consultant, Dr. S. H. Bush will play a predominant role in guiding this effort.

It is expected that about twenty consultants will participate in the work heretofore described. Total consultant time is forecasted at two man-years. Total consultant costs, including travel costs, should be of the order of $300,000. Consultants will be drawn from the Lawrence Livermore National Laboratory, Argonne National Laboratory, Idaho National Engineering Laboratory, Brookhaven National Laboratory, Battelle-Pacific Northwest Laboratories, and other sources, including private consultants and academia.

Additional NRC manpower will be needed after the completion of this effort to revise Regulations, Regulatory Guides, the SRR, and Branch Technical Positions.