NUREG-1222

Piping Research Program Plan

U.S. Nuclear Regulatory Commission

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Piping Research Program Plan

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ABSTRACT

This document presents the piping research program p or the Structural and Seismic Engineering Branch and the M als Engineering Branch of the Division of Engineering, Office of Nuclear Regulatory Research. The plan describes the research to be performed in the areas of piping design criteria, environmentally assisted cracking, pipe fracture, and leak detection and leak rate estimation. The piping research program addresses the regulatory issues regarding piping design and piping integrity facing the NRC today and in the foreseeable future.

The plan discusses the regulatory issues and needs for the research, the objectives, key aspects, and schedule for each research project, or group of projects focussing of a specific topic, and, finally, the integration of the research areas into the regulatory process is described. The plan presents a snap-shot of the piping research program as it exists today. However, the program plan will change as the regulatory issues and needs change. Consequently, this document will be revised on a bi-annual basis to reflect the changes in the piping research program.

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1. INTRODUCTION

The research discussed in this plan defines the central aspects of the NRC piping research program. The overall objectives of the piping research program are to provide the technical basis for rulemaking and regulatory decisions to support licensing and inspection activities, to assess the feasibility and effectiveness of safety improvements, and to increase the understanding of phenomena for which analytical methods are needed in regulatory activities.

The concept of "balance-of-design" is especially important for piping systems. Piping designs must be flexible enough to accommodate thermal expansion and other strain-controlled loads, but stiff enough to adequately withstand gravity and dynamic inertial loads. The negative aspect of overdesigning piping systems for dynamic loads has been a concern in recent years. In some cases, supports added to restrain potential dynamic movements can adversely affect overall piping system reliability. The current piping design process uses dynamic response evaluation methods that were justified largely by analytical studies using conservative assumptions about high level response and failure mechanisms. Now, "physical" data from recent dynamic failure tests and earthquake experience surveys indicate that the margins-to-failure for piping dynamic inertial loads are much greater than previously believed.

Thus improvements to the overall design of piping systems can be made by establishing more realistic inertial load criteria for piping. Initially, NRC and industry efforts concentrated mainly on reducing unnecessary conservatisms associated with piping dynamic response calculations and specification of piping input loads. Research in these areas continues, but in addition, it appears that research leading to revision of the ASME Code design criteria offers a future alternative for improving piping system design. Revisions to nozzle and support criteria, as well as piping component criteria, need to be considered.

The potential for cracking in critical LWR piping systems raises several difficult regulatory and safety issues. These include defining adequate inservice inspection (ISI) programs and techniques, determining the appropriate corrective actions for cracks detected in service, and determining the adequacy of proposed fixes for eliminating or reducing the incidence of cracking.

Addressing these regulatory issues requires the talents of diverse research disciplines, e.g., metallurgy, welding, corrosion, fracture mechanics, structural analysis, and non-destructive examination. The research program addressing pipe cracking integrates these disciplines into a coordinated approach that addresses the fundamental problem on a mechanistic level. This provides an assessment of the proposed fixes for both short and long term operation, and provides input to potential revisions to the ASME Code that would help preclude the occu[illegible] cracking.

The NRC and industry have been faced with the [illegible] to evaluate existing and postulated flaws in piping [illegible] the safe operation of the plants without imposing excessive[illegible]tive measures. Past and on-going research programs have addre[illegible] this question and significant progress has been made in resolving the dilemma. Further, recent rulemaking activities implementing the leak-before-break philosophy derived, at least in part, from enhanced confidence in flaw evaluation procedures.

Relatively recent advances in several technical disciplines have provided the tools necessary to predict the behavior of cracked piping using experimentally verified analysis procedures. Advances in the use of probabilistic analyses, supported by verified deterministic analyses, have led the NRC to accept the leak-before-break principle, with certain limitations, and to implement rule changes eliminating dynamic effects associated with the postulation of pipe breaks in specific classes of piping.

However, the criteria that must be satisfied to demonstrate leak-before-break warrant further investigation so that adequate, but not overly conservative, margins can be employed. Further, there is a significant interest in extending this limited acceptance of the leak-before-break philosophy to other aspects of plant design such as ECCS sizing and equipment qualification. Such extension would require definition of a replacement to the double-ended guillotine break as the design basis. The pipe fracture research described in this plan addresses the spectrum of flaw evaluation procedures as they pertain to further validation of the analytical methods and to defining a replacement to the double-ended guillotine break.

The following sections of this plan discuss the regulatory issues and needs that have contributed to the formulation of the research plan, describe the various research projects in the context of those issues and needs, and illustrate how the research results are expected to contribute to the regulatory process.

2. REGULATORY ISSUES AND NEEDS

2.1 Piping Design

A typical nuclear power plant will contain an estimated 10 to 50 miles of piping that has been analyzed and designed to resist earthquake and other dynamic loads. In terms of safety, cost, and regulatory effort, this large quantity of piping gives significance to each aspect of its design criteria. Unlike the design of most other nuclear plant components and structures, the design of piping for dynamic inertial loads can adversely affect thermal expansion and other strain or displacement controlled loadings. Studies such as reported in NUREG/CR-4263 (Ref. 1), which provides the bases for NRC Research Information Letter No. 142 (Ref. 2), show that over-stiffening a piping system by adding unnecessary seismic restraints can actually increase the probabilities of piping leakage and rupture. To establish an optimum level of overall reliability and safety for piping systems, there needs to be a realistic balance in design between the various dynamic loads and other loadings.

A major concern of the USNRC Piping Review Committee (PRC) was the over-design of piping systems for dynamic loads. After reviewing earthquake experience data, dynamic test data, and analytical studies, the committee concluded that the existing nuclear piping design criteria and practices were very conservative. In fact, they were too conservative in light of the negative impact these had on normal plant operation and overall reliability. The PRC summarized their concern in NUREG-1061 Volume 5 (Ref. 3):

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

The PRC made several research and standards recommendations that were intended to lead to better dynamic load design for piping systems. These recommendations plus new needs resulting from their implementation form the basis for the NRC's piping design research.

There are three broad areas where improvements can be made to the design of piping systems for dynamic loads:

- <u>Piping input loads</u> - The specification and calculation of loads transferred to the piping system supports (e.g., OBE/SSE specification, soil-structure interaction, building response analysis, load combination, etc.)

- <u>Piping response estimation</u> - Techniques and assumptions used to analyze the dynamic inertial response of the piping system itself (e.g., spectral analysis techniques, damping criteria).

- <u>ASME Boiler and Pressure Vessel Code* (Ref. 4) design criteria</u> - The allowable piping stress criteria specified in subsections NB/NC/ND-3600 of Section III of the Code. Pipe support and nozzle design criteria also need to be considered.

It should be noted that the first category includes topics outside the normal responsibilities of piping designers and reviewers (i.e., seismology and building structure design) and the following research plan does not directly cover work in this area. However, piping design needs for input loads are discussed below, along with those for response estimation and Code design criteria.

2.1.1 Piping Input Loads

Although current piping design practice places a great emphasis on seismic and BWR hydrodynamic loads, other loads can result in piping failures. In fact, studies such as reported in NUREG/CR-4263 (Ref. 1) show that while the overall reliability of piping is acceptable, normal operating loads pose a relatively larger threat to piping reliability than seismic loads. Nuclear plant experience indicates that "unanticipated loads" (such as waterhammer and flow induced vibration) can produce failures, yet these loads are generally not considered explicitly in the piping design process. To improve overall piping system reliability, it is necessary to better characterize what loads threaten piping integrity and to define the associated failure margins and uncertainties. Then, an integrated approach can be used to assess how these loads interact and establish balanced design criteria for them.

While many piping loads can be calculated directly by the piping analyst, seismic piping load calculations rely heavily on the input from other analysts. The specification of design earthquake parameters and the calculation of soil and building response involve many complexities and large uncertainties. To address these uncertainties, the seismologist and structural engineer add conservatisms in their parts of the dynamic design of a nuclear plant. In current practice,

*Will be referred to as simply the Code in the remainder of this document.

piping designers normally do not take credit for seismological and structural seismic response margins. Rather, they typically add even more conservatisms to account for the uncertainties in piping response. These compounded seismic conservatisms contribute to the current imbalance in piping load design.

The PRC's review of piping input load specification resulted in a recommendation to lessen the significance of the OBE on piping design. They stated (Ref. 3), "since designing piping systems to SSE is sufficient to ensure safety, the level of OBE should be defined as having a reasonable probability of occurrence but should be decoupled from the SSE." The specific PRC recommendation was that rulemaking be undertaken that would decouple the OBE from the SSE. Accordingly, the NRC is developing plans to revise Appendix A of 10 CFR 100 (Ref. 5) to do this. The revision may entail a broader effort than that recommended by the PRC and probably will not be completed until at least 1991. If this action does indeed go forward, it now seems that the end product will be a forward fit rule. Thus, further work would be needed to justify changes in the piping systems of already licensed plants.

Seismological considerations, soil structural interaction, and building structure response are being investigated under other research programs. The piping design research program will need to address relevant issues discovered by these other studies. For example, the Seismic Category I Structures Program has conducted shear wall tests that indicate wall stiffnesses and input load spectra frequencies may be much lower than are now considered in design. If this holds true, then the impact on current piping design must be evaluated.

2.1.2 Piping Response Estimation

The two dynamic loads most widely considered in the piping design of U.S. nuclear power plants are seismic loads and BWR hydrodynamic loads. These usually are treated as "building filtered loads", implying that the motion of the building must be considered first in the design process. After building response calculations are made and the motions at piping support locations are established, two types of piping system response are calculated. First, differential support displacements are estimated in what is called an anchor motion evaluation. This involves a relatively simple static analysis to determine differential motion between anchors. Second, dynamic analyses are used to evaluate piping inertial loads. These analyses are more complex and have caused many more licensing controversies. Regulatory concerns about piping inertial response analyses have even led to the shutdown of nuclear plants when errors were discovered.

The irony associated with the large amount of attention and effort that now goes into analyzing and evaluating piping inertial loads is that an increasing amount of earthquake experience and failure data indicate that piping inertial loads generally do not produce failures, even at load levels much greater than those used in design. In fact, when a broad view of nuclear plant safety is taken, such as in a seismic PRA or in a seismic margins review, for example, see Ref. 6, the possible failure of piping systems due to inertial loads usually is ignored. Rather, these analyses treat what are considered to be the more likely failure modes of seismic anchor motions (SAMs) and system interactions. It should be noted that the over-conservatisms in piping inertial load design, not SAM design, lead to the overuse of restraints and snubbers. Current Code design rules have higher allowable OBE stresses for SAM loads than for inertial loads, and, generally, do not require SAM piping stress evaluations for SSE levels.

In the past, it was thought that introducing more accurate and sophisticated response prediction methods was the best way to reduce unnecessary conservatisms in piping inertial load design. The PRC recommended a number of individual standards and research activities related to response estimation techniques, e.g., peak shifting, multiple spectra methods, and damping. Of these, the damping recommendations have been shown to be the most influential. Although the new damping values in Code Case N-411 (Ref. 7) have been used extensively on a case-by-case basis, and are accepted generically in Revision 24 of Regulatory Guide 1.84 (Ref. 8), their current endorsement by the NRC carries many limitations. For example, the requirements defining ground motion input spectra exclude the use of Code Case N-411 for a large percentage of plants. Also, Regulatory Guide 1.84 specifically limits the use of the new damping criteria to enveloped spectral analysis methods "pending further justification" of its use with other methods. Only in a few cases have the combined effects of piping criteria changes been evaluated. Consequently, the NRC needs a general basis for evaluating the effect of these proposed combinations.

2.1.3 ASME Code Design Criteria

After piping dynamic response calculations are made, dynamic stresses are estimated, combined with other stresses, and evaluated by the acceptance criteria given in Section III of the Code. The PRC recognized the potential benefit that could come from improving Code piping design criteria for dynamic loads. Unfortunately the data and information available at time were too limited to support specific standards recommendations in this area. The data were "limited" in the sense that while they gave indications that the margins-to-failure were much greater than assumed when the current Code rules were written, the true margins-to-failure were not well quantified. To develop improved and more realistic design criteria, the level of failure and the mechanisms that control the failure must be understood. Experience data and dynamic test data available to the PRC showed mostly "nonfailures," particularly for inertial loadings. The PRC thus did not recommend immediate changes to the Code rules, but instead recommended that, as a high priority research activity, tests be undertaken to identify seismic design margins and failure modes for typical piping systems. This has resulted in the NRC's cooperation in the joint EPRI/NRC Piping and Fitting Dynamic Reliability Program (PFDRP).

Inertial loadings from dynamic events such as earthquakes are time-varying and have limited durations and energy content. Code piping design requirements now evaluate inertial and static gravity loads by Equation 9 of Subarticles NB/NC/ND-3600. Plastic collapse of the pipe cross-section due to a maximum bending moment is assumed to be the dominant failure mode for these "primary" loads. However, an increasing volume of analysis and test results show that inertial loads cause pipe to behave differently than for static loads of the same magnitude. The dynamic margins-to-failure are greater than predicted by theoretical limit-load estimations which use linear response methods. Further, ratcheting and fatigue (i.e., cyclic effects) are the principal failure modes for inertial loading. The recently completed tests of the PFDRP show that significant changes should be made in the Code rules for inertial stresses. This would dramatically change piping system design criteria and could, in turn, reduce the number of snubbers used in nuclear power plants.

The impact of these potential changes to piping design criteria needs to be assessed along with current response calculation techniques. Studies have shown that if these suggested changes are made the current load criteria for both nozzles and supports become the limiting design consideration. Therefore, it is necessary to consider more realistic nozzle and support design criteria in conjunction with efforts to improve piping dynamic response and stress criteria.

There also is a need for confirmatory research regarding the new energy absorbing supports that now are being proposed for use in nuclear piping systems. Three types have been developed so far and each requires some piping support and piping design considerations that are outside the scope of standards currently accepted by the NRC. The use of these energy absorbing supports requires the development and evaluation of design analysis procedures, in addition to addressing reliability considerations.

The same power plants and other facilities that provided earthquake experience data for equipment in the Seismic Qualification Utilities Group (SQUG) effort, contain many piping systems. These piping systems use a variety of support configurations and include aged piping components. So far, the NRC has used piping earthquake experience data in only a qualitative sense. A more systematic study of this data could help in accepting the proposed piping design criteria. It also could be used to support simpler nuclear piping analyses. (Industrial plant piping analysis is relatively simple and yet produces piping designs with a good earthquake experience record.) To make use of this earthquake experience data, a better characterization of the types of piping configurations that do and do not perform well in earthquakes, and a better evaluation of these configurations in light of the proposed design criteria are needed.

As discussed above, the regulatory issues and needs for piping design emphasize dynamic load evaluation. It now appears that the major research and standards activities for piping and nozzle design will be completed in the FY88-89 time period. Piping support activities should be completed in FY90.

At this time, it does not seem that a great effort in piping design research is needed in FY 91 and beyond. Issues that may need to be addressed in the future include standardization and simplification of the piping design process, greater consideration of piping degradation in design, and evaluation of "nondesigned" loads, e.g., water hammer, flow induced vibration, axial loads, Level D SAMs, etc.

2.2 Environmentally-Assisted Pipe Cracking

Pipe cracking in a commercial power plant was first reported in 1965 when leaks were discovered in the Dresden Unit 1 Nuclear Reactor. In the ensuing years, a variety of leaks and cracks in light water reactors (LWR's) have occurred as a result of fatigue, corrosion, stress corrosion, or a combination of these factors. The most significant pipe cracking problem, encountered primarily in BWR's, has been intergranular stress corrosion cracking (IGSCC) in austenitic stainless steel piping at the heat affected zone of weldments. In the past ten years, this condition was responsible for over 400 pipe cracking incidents throughout the world. The resulting inspections and repairs, including in some plants replacement of entire piping systems, resulted in extensive costs and occupational radiation exposure. At one plant, the cost of replacement piping was equivalent to the initial cost of constructing the plant. Millions of dollars have been spent by the nuclear industry on research to resolve the problems.

Initially, IGSCC was considered sufficiently important by the NRC to designate it Unresolved Safety Issue A-42 (Ref. 9). The NRC staff and a series of NRC review committees have evaluated the problem and concluded that IGSCC has a low probability of causing major structural failures in piping systems if the system is properly monitored for leaks and flaws, and provided proper remedial action is taken. In this regard, NUREG-0313, Rev. 1 (Ref. 10), was issued as a resolution to USI A-42 and NUREG-0313, Rev. 2 (Ref. 11) has recently been issued as a material selection and processing guideline that sets forth the NRC staff's revised acceptable methods to control IGSCC susceptiblity of BWR piping. The research program on environmentally assisted pipe cracking provides data and support to the NRC staff and committees in these efforts.

For IGSCC to occur, three elements must exist in combination. These include: a susceptible (sensitized) material, a significant tensile stress, and an aggressive environment. The NRC has concluded that improvements in all three of these elements should be pursued to mitigate IGSCC in BWRs, even though a significant reduction in the probability of IGSCC can be accomplished by improving one or two of these elements. Proposed fixes by utilities vary from one extreme to the other and frequently include unique combinations of the elements to fit the specific plant problem.

Most of the research aimed at developing the various proposed fixes for IGSCC in BWR piping was performed by the nuclear industry and funded by EPRI and the BWR owners group. The NRC research on environmentally-assisted pipe cracking, ongoing for the past eight years, is aimed mainly at the NRC's need for an independent assessment of the proposed near-term and long-term fixes and to provide support to the NRC's licensing staff and other responsible NRC groups dealing with the problem. The scope includes defining the role of metallurgical variables, stress, and the environment on

IGSCC susceptibility, evaluating the influence of plant operations on these variables, and examining practical limits for these variables to effectively control IGSCC in LWR systems. The experimental work to date concentrates primarily on problems related to cracking in austenitic stainless steels.

Finally, the interest in plant life extension raises other questions pertaining to evaluating the "aging" of piping and piping components. One aspect of this complex topic pertains to the combined effects of cyclic loading and environment. These questions extend beyond the Class 1 piping. For example, should the cyclic loading (i.e., fatigue) of Class 2 and 3 piping and components be addressed in considering plant life extension even though fatigue is not explicitly included in the design analysis of Class 2 and 3 systems? If so, how should the effects of environment be incorporated in the analysis?

Based on work funded by the Atomic Energy Commission and by EPRI (see Section 3.2.2), it appears that the effects of LWR environments have not been addressed adequately in the Code Section III fatigue design curves. Consequently, the fatigue analyses for Class 1 piping may be questionable, particularly in the context of plant life extension. The effects of secondary side environments, various clean-up system environments, and upset environments are not expected to be less damaging than the primary LWR environments. Therefore, one must question the extent of fatigue damage in these other systems and the possible impact on plant life extension.

A starting point in answering these questions is the development of realistic fatigue life data, accounting for the effects of various environments. Other areas that warrant consideration include the evaluation of the endurance limit concept in the presence of aggressive environments, the linear damage summation used in fatigue and fatigue crack growth analyses, the effect of these environments on fatigue crack growth rates, the presence of a fatigue crack growth rate threshold, and the possible influence of aggressive environments on the threshold value.

Many of these issues and questions pertain to more than just piping. Consequently, some of the environmentally assisted cracking research projects address the topics on a broad base rather than being specific to piping.

2.3 Flaw Evaluation [illegible]

Recent changes in 1[illegible] Ref. 12), Appendix A, General Design Criterion 4 (GDC-4) e[illegible] .ing dynamic effects associated with postulated double-ended guillotine breaks in reactor coolant piping, where that piping meets rigorous acceptance criteria, have raised several technical issues that influence the implementation of the regulatory position. Specifically, the acceptance criteria, as initially defined in NUREG-1061 Volume 3 (Ref. 13), have been reviewed and revised to some extent by the NRC staff during the development of Standard Review Plan (Ref. 14) Section 3.6.3, "Leak-Before-Break Evaluation Procedures". The revision to the piping fracture acceptance criteria and the margins associated with those criteria has been based upon improved fracture analyses and experimental validation of many aspects of the analyses. However, there are still portions of the overall analysis, and the associated margins, that derive from engineering judgement and, necessarily, are restrictive.

Improvements in the fracture mechanics analyses can lead to additional improvements in the overall analysis. The greatest improvements in the fracture mechanics analyses are expected to come from quantifying the impact of various factors on the load carrying capacity of the cracked cross-section. These factors include:

- the effect of seismic/dynamic loadings,

- the effect of geometric variables (changes in section thickness, pipe to elbow welds, pipe to nozzle welds, etc.) on ovalization of the cracked cross-section when subjected to bending loads

- the degree to which variability of material properties in the vicinity of welds influences the fracture process,

- the effect of long-term thermal aging on the reduction of fracture toughness of cast stainless steels and the ability to accurately predict that effect,

- the accuracy and reliability of leak detection procedures for use outside containment, and

- the accuracy of leak rate estimation models.

The change in GDC-4 was brought about by acceptance of the "leak-before-break" philosophy for piping that meets the acceptance criteria. However, the change in the regulations pertains only to the elimination of dynamic effects from the postulated breaks. The regulations pertaining to equipment qualification, ECCS sizing, and containment design are not changed. The Commissioners noted in the response to public comment on the "limited scope" change to GDC-4, that this inconsistency in the regulations was necessary until the research had been performed to adequately define the criteria needed to warrant changes in these other areas. However, it is felt that requirements for containment design will not be changed in order to maintain defense-in-depth. Obviously, one key criterion will be the definition of a replacement to the postulated double-ended guillotine break of the large piping.

The definition of a replacement criterion is expected to be based upon credible accident scenarios rather than non-mechanistic breaks. Since pipe cracks can lead to large leaks, it is important to quantify credible leakage rates from credible pipe cracks. This information would be coupled with other credible leakage sources, such as valves, manway covers, and pump seals, to define the replacement criterion. Flaw evaluation procedures play a role in this process in the context of defining the opening area for the pipe crack, which contributes to defining the leakage rate.

The regulatory positions associated with the GDC-4 modifications are reasonably well defined. However, as noted above, there are some areas that are based on engineering judgement and that may be more restrictive than would be required if experimental validations were available. The nuclear industry has moved aggressively to take advantage of the GDC-4 modifications and, in the process, has proposed several alternatives to the criteria and margins put forth by the NRC staff.

The pipe fracture research program is designed to meet the need for experimental validation of alternative analyses, to provide the bases for relaxing margins in those analyses, and to contribute to the definition of a replacement to the double-ended break criterion.

3. PROJECT DESCRIPTIONS

3.1 Piping Design

The chief objective of the piping design research is to provide the technical bases for evaluating current design criteria and for developing and justifying changes to these criteria that should lead to designs with greater overall safety. This objective is met by better defining piping response and failure behavior due to various loadings, particularly dynamic loads, and by assessing how these loadings and their associated design criteria affect overall piping system reliability.

The scope of the piping design research discussed below is directed primarily at those research needs identified by the USNRC Piping Review Committee (PRC) in NUREG-1061 Volumes 2 (Ref. 15) and 4 (Ref. 16). These volumes deal with design criteria for seismic and other dynamic loadings. New research programs and tasks also have been included to address needs discovered in implementing the PRC recommended standards activities and in completing other piping research programs.

3.1.1 EPRI/NRC Piping and Fitting Dynamic Reliability Program (PFDRP)

This cooperative EPRI/NRC research program was initiated in 1985 with three main objectives:

- To identify failure mechanisms and failure loading levels for piping components and systems under dynamic loadings.
- To provide a data base that can be used in improving predictions of piping system response and failure due to high-level dynamic loads.
- To develop an improved and defensible set of piping design rules to be included in the Code.

This program will be completed in 1988. By that time, it will have produced an extensive set of piping dynamic failure tests showing current design rules are not only very conservative, but also are based on unlikely failure mechanisms for inertial loads -- the loads of most interest for seismic design of piping systems.

The Energy Technology Engineering Center (ETEC) has completed dynamic tests of pressurized 6-inch carbon steel and stainless steel piping systems. Very high seismic-like loads were applied before any leaks developed. Failures were due to excessive ratchetting and cracking of highly stressed components.

At ANCO, elbows, tees, reducers, support connections, nozzles, and lugs are being tested dynamically. Forty-one tests are planned. The test specimens include both carbon and stainless steel 6-inch piping components with various thicknesses. The test variables include internal pressure and both in-plane and out-of-plane loadings. The loadings generally are applied at one end of the specimen by hydraulic actuators, with weights at the other end. High-level time history inputs are repeated (usually more than twice) until the specimen ruptures. In addition, there have been two static load tests of elbows to demonstrate differences between static and dynamic failure mechanisms.

While the ANCO and ETEC tests emphasize seismic loadings, other dynamic loads also have been considered. For example, a separate set of water hammer tests are underway at ANCO, and more fundamental fatigue-ratchetting tests are being conducted at the Materials Characterization Laboratory as part of the PFDRP.

The results of the pipe component and pipe system experiments have shown surprisingly consistent general trends. Specific observations based on these results include:

1. Typical elastic piping design evaluations using the current Code are very conservative for inertial loads. Margins-to-failure of 15 to 30 are typical.

2. Dynamic failure is dependent upon cyclic effects, even at input levels of incredible earthquake size.

3. Ratchetting and wall thinning lead to the dynamic failure of pressurized piping.

4. Plastic collapse of the pipe cross-section (as assumed by Equation 9 of the Code) does not occur. Rather, it appears that dynamic load reversal prevents plastic collapse and that failure due to rachetting requires a large number of high level loading cycles.

5. Failure locations are determined by loading and geometry, with failure often occurring away from weldments.

6. "Loss-of-flow" failures did not occur. However, swelling did occur in the pressurized piping but crimping in the unpressurized piping was minimal.

7. Extensive testing at OBE and SSE levels produced no detectable permanent deformation. Even at the 5 times the SSE level, permanent deformations were very localized and small.

These observations indicate the need and justification for the Code criteria changes that are expected to result from this program. They also show the need to revise the NRC's piping "functionality" criteria given in the Standard Review Plan (Ref. 14).

General Electric currently is developing and evaluating alternative piping design rules that can be proposed as revisions to the Code's dynamic load design criteria for Classes 1, 2, and 3 piping components. Several of the foremost consultants in piping design are contributing to this effort, and both the ASME and Pressure Vessel Research Committee (PVRC) standards groups are monitoring the progress.

The NRC's contributions to the PFDRP have been funding the ETEC systems tests and 20 of the 41 ANCO component tests. Results from this program have been used to support the development of Code Case N-451 (Ref. 17), which provides alternative rules for Level B (OBE) inertial load evaluation for Class 1 piping. A similar Code Case for Class 2 and 3 piping (Code Case N-462) has also been approved by the Code committees. Both Code Cases are now being considered for NRC endorsement through Regulatory Guide 1.84.

The PVRC recently formed a task group to consider piping functionality criteria. This task group will use data from the PFDRP program in developing their recommendations for changes in the functionality criteria.

The ASME and NRC review and endorsement processes for any new piping design rules might raise technical questions that would require additional research to resolve. Although the need for such additional research is anticipated, specific efforts cannot be identified at this time.

3.1.2 Nonlinear Piping Response Prediction

The dynamic design of piping now relies chiefly on linear-elastic analysis techniques. While limited plastic deformation is allowed at Level D (SSE) design limits, it is normally assumed that the post-yield behavior is essentially elastic and that the linear-elastic analyses are appropriate. However, if higher allowable stresses are permitted for dynamic loading (such as may result from the EPRI/NRC program discussed above), it may become necessary to consider nonlinear behavior. The NRC's nonlinear piping response research has been structured to complement research efforts by EPRI and the PVRC.

Under FIN D1611, the Hanford Engineering Developing Laboratory (HEDL) studied a number of candidate nonlinear piping analysis methods. Several pretest estimates were made for the ETEC demonstration piping system fragility test (Ref. 18). Other piping test results also were used as benchmarks. "Best-estimate" methods, as well as typical Code design methods, were found to conservatively underpredict the actual failure loading levels. HEDL completed its investigation and provided recommendations for simple nonlinear response prediction methods (Ref. 19).

Under a Small Business Innovation Research contract (Contract Number NRC-04-86-129), Structural Analysis Technologies investigated simple nonlinear response analysis methods. This project reviewed related work by both HEDL and Rockwell, the latter sponsored by EPRI, and evaluated a nonlinear strain criteria for use in piping design.

These two projects have addressed the PRC's recommendation to develop "psuedolinear-elastic estimation methods." Beyond these two projects, the NRC is funding several related efforts. These include:

1. The EPRI/NRC PFDRP which will evaluate ways to account for nonlinear response.

2. A small project at the University of Akron, jointly funded by the NRC, EPRI, and the PVRC, which will evaluate simplified dynamic strain prediction methods.

3. A project at Brookhaven National Laboratory (FIN A3288) to perform blind, post-test analysis of the PWR coolant loop to be tested on the Tadotsu shaker. While failure of the test loop is not expected, pre-test predictions indicate that nonlinear piping response will occur.

4. A project at Argonne National Laboratory (FIN A2251) to perform blind, post-test analyses of the piping response measured during shaker tests of mechanical equipment (known as the SHAM Tests) conducted at the HDR facility in the Federal Republic of Germany. These tests involve hydraulic excitation of the piping system to very high dynamic load levels that are expected to produce non-linear response of the piping system.

Additionally, EPRI is funding Rockwell International to develop a simplified nonlinear piping response analysis method. A Code Case based on the Rockwell analysis method is being developed.

While there has been considerable emphasis on sharing data among these projects, there is no plan to compile and evaluate all the results. Such an effort is being considered and, if warranted, will be supported by the NRC in FY89.

3.1.3 Pipe Damping Study

The Pipe Damping Study is being conducted by the Idaho National Engineering Laboratory (INEL) under FIN A6316. This testing and data evaluation effort has supported the development and acceptance of Code Case N-411 (Ref. 7), and has provided new information on high frequency and high loading level pipe damping. Eight INEL reports have been published documenting the various tasks the NRC has sponsored (Refs. 20-27).

The PRC recommended that Regulatory Guide 1.61 (Ref. 28) be revised to endorse Code Case N-411, but this action now appears unlikely. Although the Code Case has been used extensively on a case-by-case basis, and was accepted generically in June 1986 as Revision 24 of Regulatory Guide 1.84 (Ref. 8), its endorsement by the NRC carries many limitations. Recently, the ASME Task Group on Pipe Damping began a new effort to improve pipe damping criteria and to address a wider range of application. EPRI has sponsored Bechtel to provide recommendations for revising the pipe damping criteria to the Task Group.

Revision of Regulatory Guide 1.61 will be considered once the Pipe Damping Task Group proposes their revised criteria. The NRC is funding INEL to provide consulting support to the Task Group's effort. In 1988, INEL will address technical and licensing questions on the new ASME criteria, thereby strengthening the technical basis for the criteria before they are considered by the NRC.

3.1.4 Piping Response Estimation

Brookhaven National Laboratory's (BNL's) Combinational Procedures for Piping Spectra Analysis Program (FIN A3287) was initiated in 1985 to investigate a number of new spectra methods identified by the PRC. In 1985 through 1987, this program evaluated the use of the Independent Support Motion (ISM) Method in combination with damping factors suggested by the PVRC. Information from this effort was used to support licensing staff actions regarding the ISM method. The NRC currently is establishing a new position on the ISM method and its use with Code Case N-411. A PVRC task group recently has been formed to review this topic, and will contribute to the NRC evaluation. In 1988, the BNL ISM studies (Refs. 29 and 30), related EPRI studies, and data from actual licensing analyses using multiple spectra methods will be reviewed and used to establish new guidance on the application of ISM.

As a separate task in the BNL program, several proposed improvements in the methods used to account for the effects of closely spaced modes and high frequency modes have been evaluated. The proposed methods were compared to "baseline" time history analyses and currently accepted spectra methods. The results of this task will be published in 1988.

In 1985 and 1986, Lawrence Livermore National Laboratory (LLNL) (FIN A0453) conducted a study to identify more completely the uncertainties associated with the specification of in-plant design spectra, per Regulatory Guide 1.122 (Ref. 31). Their conclusion was that a ±30% peak broadening would be more appropriate than the current ±15% specification. Further work on this has been postponed pending research conclusions from the NRC-sponsored Seismic Category I Structures Program in 1988 (see Ref. 32 for a discussion of this program). The shear wall tests conducted as part of that program indicate that actual building frequencies may be lower than currently considered, resulting in lower peak frequencies for piping input spectra. New studies to define improved piping input spectra may begin in FY89.

3.1.5 Nozzle Flexibility and Design

Studies have shown that as the piping inertial load design criteria are modified, the current design criteria for both nozzles and supports become limiting. It also is believed by many that the nozzle and support design criteria for dynamic loads may be more conservative than necessary. Therefore, work is underway to develop more realistic nozzle and support design criteria. These efforts are in conjunction with the efforts to improve piping dynamic response predictions and piping design criteria.

Since 1985, the Oak Ridge National Laboratory (FINs B0474 and B0847) has been investigating better evaluation methods for piping nozzles and branch connections subjected to dynamic loads. ORNL has examined several design methods for calculating nozzle flexibility, including the Code method and methods based on Bijlaard's and Steele's theories, and they have outlined a series of studies comparing the various analytical methods. These studies are aimed at selecting better analysis methods for use in general piping design. This initial investigation (Ref. 33) showed that a PVRC method published in Welding Research Council (WRC) Bulletin No. 297 (Ref. 34) contains deficiencies in the guidance given on flexibility factors.

An extended parameter study using the FAST2 computer program will be completed in 1988. Improved flexibility factor design guidance, in the form of analytical formulas and design graphs, and proposed revisions to the Code rules will be developed. In 1988, ORNL also will complete reviews of the Code fatigue design and primary load design criteria for piping branch connections.

Recommendations from this ORNL program already have resulted in actions by the ASME and the PVRC to improve design rules for nozzles and branch connections. Recommendations for additional changes will be made when the program ends in late FY88.

3.1.6 Support Design

As discussed above, the criteria currently used to design piping supports may need to be reevaluated in light of proposed changes to piping design criteria. It has been suggested that dynamic design margins for piping supports need not be the same as for equipment supports. A major effort by the PVRC is underway to identify ways to improve the entire design process for piping supports. Once the PVRC completes this effort, the NRC will define research programs to evaluate the recommended changes to the piping support design process. It is anticipated that the necessary research programs will be developed late in FY88.

The use of energy absorbing supports in piping design is an area where confirmatory research is needed. Currently, there are three types of these supports that have been introduced for nuclear plant use. Designing piping systems using these supports involves response estimation methods that are different from current practice. EPRI has tested systems using these supports at the HDR facility and the results are now being evaluated by the Argonne National Laboratory under FIN A2251. A research project to evaluate design procedures for piping systems using the Bechtel type of energy absorbing support is now underway at BNL under FIN A3306. This work should be completed by June 1988.

3.1.7 Cumulative Effects of Piping Criteria Changes

The evaluation and acceptance of new dynamic piping design criteria has been done largely on an item-by-item basis. Only in a few cases, such as PVRC damping combined with peak-shifting and PVRC damping combined with the ISM method, have the combined effects of piping design criteria changes been considered. Consequently, the NRC needs a basis for evaluating the effects of these proposed and potential combinations.

A "failure margins" approach could be used to look at the effect of piping design changes on overall reliability. The EPRI/NRC PFDRP (see Section 3.1.1) has helped define actual dynamic failure mechanisms and failure loading levels for new piping systems. Earthquake experience data (see Section 3.1.8) and the dynamic test results from the International Piping Integrity Research Group program (see Section 3.3.4) will supply information on the dynamic failure behavior of aged and degraded systems. The Degraded Piping Program (see Section 3.3.1) and other related efforts will supply data on piping failures under other types of loadings. Data from these programs can be used in a probilistic analysis to update the studies used to justify recent piping design changes, such as the "Stiff Versus flexible Piping Project" (Ref. 1). The NRC may fund work in this area in FY88 or FY89.

In the past, the NRC has relied on a "code margins" approach which considers design limits, rather than failure limits, in evaluating alternative procedures. However, a "response margins" approach also has been used. This approach involves establishing acceptable analytical "baseline" methods and comparing the results of these to predictions made using proposed techniques. Recently, EQE, under subcontract to LLNL (FIN A0457), completed the most comprehensive piping response margin study to date. The findings, reported in NUREG/CR-5073 (Ref. 35) show that currently accepted baseline response evaluation methods are very conservative. Further, the study found that newly proposed analytical methods produce significantly higher response predictions than the so-called "best estimate" predictions.

However, recent attempts to use response margins arguments for justifying new calculational methods have been ineffective. Partly, this is because guidance on the allowable variation and degree of exceedance of new response estimations, versus baseline estimations, have been difficult to establish for piping analyses. But the main limitation of response margin and code margin approaches is that they do not take advantage of the total conservatism in piping dynamic load design. A failure margins approach can be used to support more significant design changes. Plans for a program to assess the cumulative effects of piping criteria changes will be developed in 1988.

3.1.8 Piping Experience Data

The Seismic Qualification Utility Group (SQUG) and the work to resolve USI A-46 have been very effective in using earthquake experience data from heavy industrial facilities to establish seismic qualification acceptance criteria for several classes of nuclear power equipment and for cable trays. The power plants and other facilities that provided this experience data contain many different piping systems. These piping systems include a variety of piping support configurations and include aged piping components. Earlier studies have looked at these data but there has been little attempt to develop or justify new piping design criteria based directly on this experience. A more systematic study of these data could be of great benefit, perhaps leading to simpler piping design concepts. For example, establishing design rules to ensure ductility and other good design practices rather than requiring point-by-point dynamic stress evaluations. Industrial piping often is designed by simple pseudo-static methods and, generally is more flexible than nuclear piping systems. Failures in welded piping due solely to seismic inertial loadings have not been observed to occur even for very high (up to 0.5g peak ground acceleration) earthquake levels. The cases where failures have occurred include threaded and corroded pipe and pipe attached to unanchored tanks.

ORNL has begun a new project, under FIN B0850, to investigate how industrial plant piping earthquake experience relates to nuclear plant design and loadings. ORNL will better characterize the types of piping configurations that do and do not perform well in earthquakes, and evaluate these configurations in light of the new design criteria.

Other possible sources of useful dynamic piping experience data include information on water hammer, flow induced vibration, and pipe rupture events. Although the loading for these may be hard to define, lessons may be learned from the damage records. For instance, water hammer events often cause piping support failure but do not cause pipe rupture. Did support failure increase or decrease the probability of pipe pressure boundary failure? A study evaluating the relationship between damage and pipe design is planned for FY89.

3.1.9 Cost of Piping Dynamic Design

While the studies suggested in Section 3.1.7 are expected to demonstrate the safety benefits of changing piping dynamic load criteria, the cost impact of such changes also must be addressed. The studies reported in WRC Bulletin 300 (Ref. 36) and EPRI report NP-4843 (Ref. 37) indicate that significant savings would result from relaxing dynamic load criteria. However, the NRC requires a more detailed discussion of the cost savings than provided in these reports to complement safety studies in providing a complete cost/benefit analysis. Consequently, a small research task will be developed and implemented in mid-1988 to provide this information.

3.2 Environmentally-Assisted Cracking In LWR Piping

This plan describes environmentally-assisted cracking research that encompasses a wide range of topics. The plan addresses topics ranging from intergranular stress corrosion cracking to corrosion fatigue to erosion/corrosion damage. The research studies the various damage mechanisms, the regulations needed to minimize or preclude service cracking, and, in some cases, the viability of proposed remedies for service cracking.

These research projects have the potential for significant impact on the NRC's regulatory positions on piping. For example, the acceptance criteria for demonstrating leak-before-break include the requirement that the line not be susceptible to corrosion, e.g., stress corrosion cracking. If methods of precluding stress corrosion cracking can be developed in BWR's, then it would be much easier for those plants to satisfy the acceptance criteria. This would be a significant development from a regulatory perspective and the economic benefits to the industry could be considerable.

The nuclear industry has invested many millions of dollars in environmentally-assisted cracking research through various owners groups and the Electric Power Research Institute. The NRC research effort complements much of this industry effort although it breaks new ground in some cases.

The overall objective of the NRC's environmentally-assisted cracking research is to provide the experimental data needed to validate existing regulatory positions and to provide guidance in considering new problems and developing new regulatory positions. The scope of the research extends from examining the mechanisms that control damage processes, such as corrosion fatigue cracking and single-phase erosion/corrosion, to developing the data needed to justify changes in the applicable codes and regulations, specifically as they relate to aging of piping and piping components.

The following subsections describe the on-going and planned research projects on environmentally-assisted cracking in LWR piping systems.

3.2.1 Stress Corrosion Cracking of Piping in LWR's

A program is underway at Argonne National Laboratory (FIN A2212) with the objective of providing an independent capability for predicting stress corrosion cracking (SCC) in light water reactors and to support the NRC in evaluating remedies for pipe cracking proposed by the nuclear industry. Industry proposed remedies include procedures that produce a more favorable compressive residual stress state on the inner surface of the pipe, replacement materials that are more resistant to Intergranular Stress Corrosion Cracking (IGSCC), and changes in the reactor coolant environment that decrease the susceptibility to cracking.

The effort on SCC in piping is divided into four subtasks: (a) Long-term Aging and Analysis of In-reactor Components; (b) Evaluation of Nonenvironmental Corrective Actions for SCC; (c) Evaluation of Environmental Corrective Actions for SCC; and (d) Effect of Irradiation on SCC Susceptibility in Reactor Coolant Environments. The program seeks to evaluate potential solutions to LWR-SCC problems, both by direct experimentation (including full-scale welded pipe tests), and by developing a better basic understanding of the various phenomena.

The subtask on long-term aging and analysis of in-reactor components primarily is aimed at determining the degree of susceptibility of stainless steel pipe in BWR's to severe sensitization. Sensitization is one of the major contributing factors for IGSCC of chromium-nickel-iron alloy components in LWR's. Isothermal sensitization of austenitic stainless steels, such as Types 304 and 316, normally occurs in the temperature range of 500-800°C. At lower temperatures, sensitization is suppressed. However, it has been shown that severe sensitization can develop by long-term exposure at temperatures well below the normal sensitization range if chromium carbide nuclei are present. It is thus possible that the degree of sensitization (DOS) in LWR pipe weldments (or any other component with carbide nuclei) could increase with reactor operating time, and that the weldments may become increasingly susceptible to IGSCC.

The potential for degradation of in-reactor components due to other heating and aging treatments also is a concern. It has become increasingly important to verify results of in-service inspections, and to examine possible adverse effects of remedial treatments on cracking behavior of in-reactor components.

The primary objectives of this subtask are to: (1) examine the possibility for increased susceptibility to sensitization associated with thermo-mechanical remedial treatments [e.g., corrosion resistant cladding (CRC), heat sink welding (HSW), and induction heating stress improvement (IHSI)] and the possibility of sensitization in alternate alloys with special emphasis on Type 316 NG SS; (2) investigate the IGSCC susceptibility of materials with different thermomechanical histories, but the same nominal degree of sensitization, and (3) analyze piping with weld overlay repairs.

The degree of sensitization for aged specimens is being examined. Constant Extension Rate Tests (CERT) are being performed in oxygenated water to evaluate IGSCC susceptibility. An investigation of the dependence on thermomechanical histories of the correlation between IGSCC susceptibility and sensitization tests is in progress. IGSCC initiation and propagation tests will be performed for Type 304 SS specimens with different aging histories. Long-term aging of stainless steel including modified Type 347 SS specimens are being conducted. Analysis of in-reactor components will be performed as available and required.

The subtask on evaluation of non-environmental corrective actions for SCC is concerned with the evaluation of the proposed corrective actions for generic environmentally-assisted cracking problems, such as use of alternate materials or altered fabrication techniques. The development of corrective actions is primarily the responsibility of the vendors, utilities, and their associated organizations, such as EPRI. The objectives of this subtask are (a) the critical review and examination of the supporting data for proposed corrective actions and (b) the identification and acquisition of the additional data needed for regulatory action.

The major focus of the alternate materials studies has been on Type 316 NG SS, which is being used to replace Type 304 SS piping in several BWR's. However, modified Type 347 SS used in Germany for nuclear applications and fine-grained cast stainless steels also are being considered as replacement materials and are included in the studies. Weld overlays are of interest because of their potential for residual stress modification, and because of the inherent resistance of weld metal to SCC. Considerable emphasis is being placed on weld overlays because of their wide use as a temporary remedy and the potential benefits that could be obtained if they can be qualified for long term application.

Fracture mechanics tests will be performed on Type 316 NG SS and weld overlay specimens in simulated BWR coolant environments to determine threshold stress intensity factors required for crack propagation. The effect of transient impurity intrusions on susceptibility to SCC will be determined. The effect of heat-to-heat variability on susceptibility to cracking will be determined for Type 316 NG and modified Type 347 SS, with emphasis on the difference between "conventional" Type 316 NG SS and nuclear grade materials from Japan and Germany. The possible role of trace elements and microstructure will be examined.

The effect of surface cold work on crack initiation in Type 316 NG SS materials will be studied using crevice bent beam tests and slow strain rate tests with simulated crevices.

Additional slow strain rate tests will be performed to determine the role of nitrogen on SCC and to assess the relative SSC susceptibility of the LN and NG grades of stainless steels.

Pipe tests in impurity environments to evaluate the resistance of Type 316 NG SS to transgranular cracking under more prototypical loading conditions will be performed as will pipe tests on reference Type 304 SS pipes. After these pipes crack they will be repaired by weld overlay and returned to test. Finite element studies will be carried out to ensure that the weld overlay used on the 4 in. diameter pipes in the tests is representative of the overlays used in practice on larger diameter piping. Pipe test specimens also will be fabricated from the modified Type 347 SS.

Residual stress and metallographic studies have been performed to assess the effect of the Mechanical Stress Improvement Process

(MSIP) on reducing the susceptibility to SCC. The residual stress studies included both surface and throughwall measurements on 12-in. and 28-in. diameter weldments (Ref. 38).

The focus of the work on Types 316 NG and 347 SS is on the susceptibility to cracking in impurity environments under prototypical loading conditions and transient water chemistry. These tests include pipe tests, fracture mechanics specimen crack growth rate tests, and very long-term displacement controlled, fracture mechanics specimen crack growth rate tests to determine effects of loading history, water chemistry and crevice conditions on SCC. However, small specimen tests also will be performed to determine the effect of loading history on cracking and, hence, on extrapolating data obtained under laboratory loading histories to reactor loading histories. A significant data base has been established for crack growth under slow strain rate conditions. This will be compared with data obtained under constant load and slow cyclic loading conditions for small specimens and with the results of the pipe tests under similar loading histories. Analytical models will be developed to correlate loading history effects on SCC so that results can be translated to more complex loadings encountered in reactors.

The subtask on environmental corrective action is studying effects of water chemistries on SCC. The reactor coolant environment, under both normal and off-normal water chemistry conditions, has a profound influence on the performance and reliability of nuclear power plant components. The objective of this subtask is to evaluate proposed actions to mitigate SCC problems, primarily from the standpoint of water chemistry modifications. The possibility of defining practical limits on water chemistry during off-normal transients to mitigate environmentally enhanced cracking also will be explored. As in the other tasks, the emphasis of the program is on SCC problems with austenitic stainless steel in BWR's.

An extensive slow strain rate test program has been performed to demonstrate the strong interactions between dissolved oxygen level, the presence of impurities in the environment, and the susceptibility to cracking. This study demonstrated the extremely deleterious effects of sulfur species on sensitized stainless steels (Ref. 38).

Slow strain rate testing will be continued to explore the effect of other impurities on SCC. Since it is impossible to evaluate each impurity as extensively as was done for sulfate, it is important to develop a general picture of the role of impurities in the SCC process. Corrosion potential and conductivity provided the most convenient description of the role of water chemistry. A general picture of the regions of susceptibility in terms of these variables is now available and has been incorporated in the BWR Owners Group water chemistry guidelines (Ref. 39). Additional slow strain rate testing will be carried out to determine the effects of heat-to-heat variations and strain rate on SCC susceptibility. Fracture mechanics tests will be carried out to quantify the benefits of, and damage due to, coolant chemistries near the limits of normal and alternate hydrogen-water chemistries in BWR's.

This work should provide a reasonably complete picture of the effect of steady-state water chemistry on SCC of sensitized stainless steels. However, reactors undergo a wide variety of chemical transients. In some cases the transients may even be deliberate. Some utilities have considered the use of hydrogen water chemistry on a regular on-off cycle to minimize the higher radiation exposure due to Nitrogen-16 carry-over associated with hydrogen injection during operation. Additional work will be conducted to quantify the effect of chemistry transients on cracking susceptibility.

Irradiation-assisted stress-corrosion cracking (IASCC) of solution-annealed austenitic stainless steels in high-temperature water was first observed in fuel cladding. It is usually attributed to irradiation-induced segregation of alloying or impurity elements, although the degree of segregation of critical elements (e.g., Cr, P, S, etc.) produced by fast-neutron fluence at LWR operating temperatures has not been established. The major environmental parameter which affects IASCC is open-circuit corrosion potential. Hydrogen-water chemistry has been shown to suppress the dissolved oxygen in the recirculation piping system. However, the extent to which hydrogen-water chemistry suppresses dissolved oxygen due to radiolysis and alters the open-circuit corrosion potential of core materials, and hence, reduces SCC susceptibility of the irradiated material, has not been determined. Indeed, even in the case of conventional BWR water chemistry, the actual environment in the intense radiation field in the core is not well understood. The work in this task will concentrate on characterizing the effect of high gamma radiation fields on the corrosion potential of Type 304 SS in 289°C water containing dissolved oxygen and various impurities at low concentrations. Studies being performed under a related program on the aging and degradation of in-reactor materials will focus on the material conditions that lead to SCC susceptibility and the determination of the corrosio[illegible]al required to initiate cracking. The combined results wil[illegible] better assessment of the potential for SCC in the core re[illegible]

Tests will be performed on irradiated Type 304 SS specimens in simulated BWR water containing dissolved oxygen and hydrogen to determine whether the SCC behavior is controlled by the corrosion potential, as influenced by the local radiation field at the metal surface, or by radiolysis of the bulk water due to the overall gamma flux. The information obtained in this subtask will be used to determine the degree of improvement in SCC resistance of the steel through implementation of more stringent control of water chemistry and the potential for better in-core material performance through hydrogen additions to the feedwater of BWR's.

A catastrophic failure of a large elbow in the Surry 2 Reactor resulted from severe erosion-corrosion (EC) wall thinning in a steam generator feedwater line fitting. Subsequent in-plant inspections have shown that erosion-corrosion problems are widespread in the secondary piping systems of nuclear reactors.

Laboratory studies in Europe have shown that erosion-corrosion arises due to accelerated mass transport associated with turbulent flows. In elbows and tees, where complex flow patterns exist due to the strong secondary flows, the average mass diffusion rates can be 2-3 times higher than those occurring in fully developed flow through straight pipe. However, local mass transfer rates can be eight times greater than the nominal rate. Such locations are sites of potential wall thinning and localized failures. More recently, significant erosion-corrosion was observed in a straight pipe section considerably downstream of an elbow. Thus, it appears that even under apparently "nominal conditions" exit flows into tees, elbows, transition pieces and their combinations produce highly localized turbulent flow regions that are not well understood.

The objectives of the proposed work under this task are: (1) carry out confirmatory erosion-corrosion studies to develop (a) understanding of the effects of system geometry on erosion-corrosion, (b) mass-transfer coefficient data for piping components under flow conditions of interest in nuclear piping systems, and (c) guidelines for design to avoid high mass-transport rates; (2) assess effects of water chemistry variations (variation in pH control levels, impurities, and synergistic effects of oxygen) on expected erosion-corrosion rates; (3) assess existing models for erosion-corrosion to determine their adequacy to provide guidance for nondestructive evaluation of piping systems; and (4) assess possible detrimental effects of hydrogen-water chemistry on erosion-corrosion of carbon steel piping in BWRs.

3.2.2 Environmentally-Assisted Fatigue and Fatigue Crack Growth In LWR Materials

The Environmentally-Assisted Crack Growth in LWR Materials project is being conducted by Materials Engineering Associates as Task 2 of the Structural Integrity of L[illegible] Water Reactor Pressure Boundary Components program, contract number NRC-04-84-102. This four year project was initiated in January 1984 and is expected to be completed in September of 1988. A follow-up program is planned.

The ongoing program is multi-faceted and much of the work is focused on vessels but is applicable to piping as well. [illegible] objectives of this project are (1) to develop data needed to support the revision or use of the fatigue design curves in Section III of the Code, (2) to develop data needed to evaluate the fatigue crack growth methodology incorporated in Section XI of the Code, and (3) to develop data that will contribute to identification of the mechanism or mechanisms that control environmentally assisted cracking in LWR materials.

The fatigue life research complements research originated in the Pipe Rupture Study conducted by the General Electric Company for the Atomic Energy Commission (for example, see Ref 40). That study examined the effect of several variables, including environment, on the fatigue life of typical piping steels. The work suggested that the BWR environment had a significant and adverse effect on the fatigue life of carbon steels. Similar work was continued by General Electric under EPRI sponsorship, resulting in a significant data base for carbon steels in a BWR environment (Ref. 41). These data clearly show that revision of the fatigue design curves in Section III of the Code is warranted to account for the effects of the BWR environment.

The NRC funded fatigue life research examines the effect of a PWR environment on the fatigue life of a typical carbon steel and carbon steel weldment. The work makes use of standard "polished bar" fatigue specimens to provide a direct comparison to the existing fatigue design curves. The testing evaluates the effects of loading parameters (stress ratio, strain concentrations, and loading frequency), test temperature, and environment.

The testing also makes use of "pipe" specimens which are two lengths of carbon steel pipe joined by a girth butt weld made using welding procedures for Class 1 piping. These specimens will be pressurized, exposed to either an inert high-temperature environment or the simulated PWR environment, and subjected to uniaxial loading. The tests will be terminated once a crack penetrates the pipe wall The cycles to penetration then will be compared to the fatigue design curves, to assess the ability of the fatigue analysis and design curves to predict component life in of the PWR environment.

The crack growth research builds on the test procedures and environmental control procedures developed in earlier NRC funded research conducted at the Naval Research Laboratory (Ref. 42). The on-going research has extended those procedures to more complex testing situations, yielding data on realistic service environments, materials, loading histories, and crack geometries. Key aspects of the crack growth research include the following:

- Fatigue crack growth rate data have been generated using conventional compact specimens (CTs) to evaluate the effects of metallurgical structure, chemical composition of the steel, temperature, oxygen content of the environment, loading rate, and stress ratio. These data have been used in assessing the existing Section XI fatigue crack growth rate curves for ferritic steels and in developing fatigue crack growth rate curves for austenitic stainless steels.

- Fatigue crack growth rate experiments are being performed using surface flawed panel specimens to evaluate the effect of the PWR environment on the three dimensional flaw geometry and to assess the usefulness of the existing Section XI fatigue crack growth analysis procedures in predicting the growth of realistic cracks. The primary variables in this study include flaw geometry, material type and chemistry (A533B low sulfur vs. A533B medium sulfur vs. A106B vs. 304 stainless steel), temperature, and environment (PWR vs. air). The crack growth rate and changes in crack shape are monitored throughout each test. The Section XI methodology for predicting crack growth is evaluated by predicting the crack growth as a function of loading cycles using the CT specimen data and then comparing the predictions to the surface flawed panel test results.

- Fatigue crack growth rate data are being developed using the surface flawed panel specimen design to evaluate the effects of stainless steel cladding on the growth rate and shape changes of surface flaws, and to evaluate potential interactions between the PWR environment and the cladding that might alter the growth of a surface flaw in the ferritic steel base metal. The key variables in this study are the presence or absence of cladding, temperature, and air versus PWR environment. The results will provide a quantitative assessment of how the cladding enters into the analysis of small surface flaws in a clad component, particularly in the presence of the PWR environment.

- Fatigue crack growth rate data are being developed using CT specimens subjected to variable amplitude loadings to evaluate the potential effects of underloads and overloads on the crack growth rate behavior. The work involves experiments in an inert environment and in the PWR environment. The results will provide an assessment of the accuracy of the Section XI fatigue crack growth analysis where the crack growth increments from each loading block are combined without regard for load history effects, i.e., a linear damage summation.

The investigation into the mechanism(s) of environmentally-assisted cracking seeks to identify and describe the micromechanisms which control crack growth in LWR piping and pressure vessel materials. The effort involves a literature survey, laboratory studies, and liaisons with projects underway at other laboratories. The effort addresses material chemistry, microstructrual variables, and environment chemistry as they affect fatigue crack growth rates in piping and pressure vessel steels. The work is motivated by a need to be certain that the crack growth data base is appropriate to the structural application and not simply a testing novelty. Such certainty would allow informed decisions on topics such as replacement piping or plant aging considerations. This is a modest effort intended to provide an impartial review of the various mechanisms being proposed to explain the process(es) of environmentally-assisted cracking. The experimental effort is intended to provide a few critical experiments that will contribute to defining the controlling mechanism(s). The work supplements the extensive efforts being conducted both in the U.S. and abroad.

This work is integrated with the international efforts on environmentally-assisted cracking by the contractor's participation in the International Cyclic Crack Growth Rate (ICCGR) group. This group meets semi-annually to discuss technical advances and to foster cooperation and exchange of information among the international laboratories involved in this type of research. This interaction with the international community has provided an opportunity to present the contractor's test plan, test techniques, and results to review by other international experts in the environmentally assisted cracking technology, providing valuable guidance to the contractor and to the NRC regarding future directions for the research.

The research results are expected to affect the regulatory environment in at least two important areas. First, the fatigue life research results are expected to provide the additional evidence needed to support a broad modification to the Section III fatigue design procedure to account for the effects of environment. Such a modification to the fatigue design curves could support revision to Standard Review Plan Section 3.6.2 (Ref. 14) to alter the criteria for postulating pipe break locations based on fatigue usage factor.

The second area in which these research results are expected to affect the regulatory environment is the validation and potential revision of the Section XI fatigue crack growth analysis procedures. Definition of the mechanisms controlling the cracking will lend support to the use of the test data to predict structural behavior. Further, the broad based evaluation of the Section XI data base and analysis procedure will provide needed validation of the existing procedures.

The follow-on program to the MEA project will provide additional analytical and experimental data, primarily directed to applying past developments to reactor-typical components. This effort will address the effects of "real" defect geometries, service loadings, stress concentration gradients, environments, etc. When combined with previous research completed on the MEA project and the extensive work by others, this work will form an integrated approach to further understanding the phenomena of environmentally-assisted fatigue and fatigue crack growth. It is expected that this research will contribute to potential modifications and extensions of appropriate parts of Sections III and XI of the Code.

The fatigue design curves of Section III are based on data acquired some years ago using a variety of testing techniques, test temperatures, and, for the most part, smooth specimens. Tests were conducted in dry environments, although the actual application is often for ferritic and stainless steels which are subject to the PWR environment. The NRC needs to establish a data base which is more relevant to the corrosion-assisted fatigue situation and to address the fatigue damage concerns which apply to plant life extension issues. As part of the ongoing program, MEA conducted a series of tests to evaluate the accuracy of the stress-life (design) curves contained in Section III, including the effects of notches, environment, and temperature. MEA research produced a much more consistent, although limited, set of data on which to evaluate the accuracy of the Section III design curve, the effects of PWR primary coolant environment on fatigue life, and the applicability of stress-life design curves to fatigue damage in reactor structures. Sufficient additional data will be developed on piping materials and welds to evaluate the existing S-N code curves and, as appropriate, propose modified S-N curves for inclusion into the Section III of the Code.

The vast preponderance of the environmentally-assisted fatigue crack growth data that forms the basis for the water environment curves now incorporated in Section XI of the Code, has been developed using specimens of the compact tension type. Such specimens are characterized by relatively large crack dimensions, straight crack fronts, constant K_Is along the crack front, and a condition where the corrodent surrounds and wets all surfaces of the specimen. This is markedly different from those conditions which prevail in the case of "real" crack geometries in cylindrical bodies. For this latter condition initial crack dimensions are usually small, the crack front is not straight, the K_Is are not constant along the crack front, and the corrodent washes only the surface of the body that contains the crack. These differences between real cracks and CT specimens raise questions about the applicability of the CT data in performing fatigue crack growth assessments of defects found in real structures.

Applications-related studies conducted at MEA under the ongoing contract include extension of the existing data base of constant amplitude fatigue crack growth rates, growth of part-through cracks in both clad and unclad RPV steels, and mechanisms of environmentally-assisted cracking. However, the NRC needs a greater body of such information to assess the applicability of the crack growth rate reference curves in Section XI to actual conditions, and to develop recommendations for the correction of deficiencies in the data used by the Code, or methods of applications of the Code.

The ongoing MEA program consists of several types of experimental fatigue crack growth rate tests, calculational model development tasks, and corrosion fatigue crack growth mechanisms studies. Compact tension specimens have been used in tests at high load ratio and low delta-K to assess the accuracy of the Section XI reference curves and analysis procedures in these infrequently evaluated areas. Unclad and clad specimens with part-through cracks have been tested to ascertain the effect of two-dimensional cracks. Additional studies will evaluate in more depth "real" crack geometry effects on environmentally-assisted fatigue crack growth using part-through cracks, pressure vessel and piping steels, and more realistic test specimens such as large diameter cyclically pressurized pipe. A data base is to be established which may be applied to evaluating and, if necessary, modifying the Section XI crack growth rate reference curves.

Micromechanism models have been proposed which attempt to characterize environmentally-assisted fatigue crack growth from the viewpoints of the constitutive chemical elements of the steel, the water chemistry, the crack tip strain rate, crack tip blunting, corrosion product buildup in the crack, and crack closure phenomena. A generally validated micromechanism model is needed to allow extrapolation of existing data to materials and loading conditions other than for those evaluated empirically. Without developing such a model, experimentally derived data must be relied upon and such data is time consuming and costly to generate. In the ongoing program, MEA has conducted studies of fatigue crack growth mechanisms through a series of laboratory tests and fatigue fracture surface examinations. Development of a strain rate model incorporating two crack growth mechanisms has resulted in an analysis of data sets covering a wide range of temperatures, load ratios, and cyclic frequency test conditions. Substantiation of the calculational model through fractography and other post-test studies of plastic-zone size and corrosion product examination is being completed. The mechanism and modeling studies are being conducted to help identify potential problem areas in corrosion fatigue of PWR materials, such as particularly aggressive combinations of loads, cyclic frequencies, environments and materials. Additional studies will be conducted to enhance and validate existing mathematical models that describe mechanisms of environmentally-assisted cracking.

There is a need for a method to calculate crack extension under reactor-typical conditions, including load-time history. Most fatigue crack growth rate tests of pressure vessel and piping steels have been conducted under constant amplitude loading test conditions. Current research indicates that fatigue crack growth rates under the more reactor-typical variable amplitude loading conditions may be significantly higher than are predicted using the Section XI crack growth rate analysis procedures. The problems with the current Section XI analysis procedure stem from 1) the linear damage summation used in predicting crack growth under variable amplitude loading, and 2) the very narrow range of test variables used in developing the fatigue crack growth rate reference curves. Under the ongoing contract, MEA has conducted tests to evaluate the behavior of fatigue crack growth under variable amplitude loading in both inert and PWR primary coolant environments.

MEA evaluated fatigue crack growth behavior under service-typical load spectra and attempted to develop a mathematical model to compute crack extension under representative variable amplitude service loads. The results of this work have shown that the current Section XI crack growth analysis can be significantly in error by underpredicting crack extension for a given load history. Additional testing will be conducted to provide a stronger basis for the use of load interaction crack growth models and for possible revision of the Section XI flaw evaluation procedures so they can be used reliably in assessing suitability for service and plant aging questions.

3.3 Pipe Fracture Predictive Methodology

The objective of the pipe fracture research is to evaluate the fracture behavior of typical Light Water Reactor (LWR) piping under characteristic loading conditions to determine the ultimate load carrying capacity of cracked piping and pipe components. The effort involves both experimental and analytical projects. The parameters considered in the projects span the range of materials, crack geometries, and loading types pertinent to LWRs.

The research addresses many of the research needs identified by the Piping Review Committee in NUREG-1061 Volumes 3 (Ref. 13) and 5 (Ref. 3). Further, new efforts are being formulated to address research needs that have been defined since the Piping Review Committee completed its work.

The following subsections describe the on-going and planned pipe fracture research projects.

3.3.1 Degraded Piping Program - Phase II

The Degraded Piping Program - Phase II is a four year program being conducted by Battelle's Columbus Division under contract NRC-04-84-103. The project was initiated on March 1, 1984, and will be completed by September 30, 1988. The main objectives of the project are to conduct experiments to determine the capacity of cracked ductile piping to withstand normal, transient and accident loading conditions, and to develop and validate ductile fracture mechanics analyses for predicting the loading capacity and failure mode of cracked piping.

The project has evaluated the effects of material type (carbon steel, wrought and cast stainless steel, Inconel, and the associated weldments), pipe size, flaw geometry, and loading type (bending versus pressure versus combined bending and pressure). The experimental program includes materials obtained from nuclear power plant suppliers, cancelled nuclear power plants, and materials removed from operating nuclear power plants. The pipe sizes tested range from 4 in. diameter, Schedule 40, to 37 in. diameter with a 3.5 in. wall thickness. The test temperatures are typical of LWR operating temperatures and the loading rates are quasi-static.

The analytical efforts have addressed the full spectrum of predictive methods. The work has evaluated limit-load approaches, identifying the range of applicability for those approaches. The effort has made use of elastic-plastic three dimensional finite element analyses to evaluate the fracture behavior of cracks in both base metal and weldments. This work also contributed to assessing various J-estimation schemes (simplified ductile fracture mechanics analyses) used when the limit-load approaches are inappropriate. It has involved evaluating existing J-estimation schemes and developing modified schemes.

The results of this project already have had an impact on the regulatory process by contributing to the technical basis supporting the leak-before-break philosophy that led to modifications of General Design Criterion 4 (GDC-4) of Appendix A to 10 CFR 50. Further, the results have been used in developing the criteria that must be satisfied to take advantage of the GDC-4 changes. The results have been used in evaluating the margin of safety inherent in Section XI of the Code, Article IWB-3640, "Evaluation Procedures and Acceptance Criteria for Austenitic Piping". The results also have been used in developing flaw evaluation criteria for carbon steel piping that will be incorporated into Section XI as Article IWB-3650.

This project has been the focal point of the NRC's piping fracture research. As such, strong relationships have been established with most of the other piping fracture related projects. For example, this project has contributed materials and data to the Piping Fracture Mechanics Data Base (see Section 3.3.5). It also interacts with the Elastic-Plastic Fracture Mechanics Evaluation of LWR Alloys project (see Section 3.3.2) in the exchange of data and cross-checking of experimental procedures and results, and with the Aging of Cast Stainless Steels project (see Section 3.3.6) in providing additional materials to that project.

The project has identified areas that need further research; many of those areas have been incorporated in other on-going and planned projects. For example, this project led directly to forming the International Piping Integrity Research Group (IPIRG) (see Section 3.3.4) which is evaluating the effects of seismic and dynamic loading on the fracture behavior of cracked piping.

The Degraded Piping Program - Phase II is scheduled to be completed by the end of September 1988. The remaining research involves completing approximately 10 pipe fracture experiments, documenting those experiments, and preparing a final report on the program. The data and analyses are providing a strong basis for further research, clearly identifying areas needing additional work.

3.3.2 EPFM Evaluation of LWR Alloys

The Elastic-Plastic Fracture Mechanics (EPFM) Evaluation of LWR Alloys is a multi-year project being conducted by the David Taylor Research Center at Annapolis under Interagency Agreement RES-78-104. The project focused initially on pressure vessel integrity considerations. However, the emphasis was shifted to piping in 1980. The current tasks in the project are scheduled for completion in 1990.

The objectives of this project are to conduct experiments using small and moderate scale laboratory and pipe specimens to determine the critical parameters affecting ductile fracture toughness of nuclear grade steels and weldments, to evaluate new test methods for possible standardization, and to verify the applicability of new data analysis methods to existing specimen types. The test procedures, specimen types and data analysis procedures are examined for the purpose of developing standardized techniques that can be implemented by other laboratories to produce material property data useful in the analysis of cracked pipe. The pipe fracture experiments are conducted on small to moderate size pipe and weldments to develop data useful in validating pipe fracture analyses and to provide benchmarks for use in evaluating the material property test procedures, specimens, and data analysis procedures.

The on-going pipe fracture experiments evaluate the fracture behavior of cracked ferritic steel weldments. This work will address the effects of system compliance on the fracture instability behavior of the pipe. The tests are conducted at 550°F under four-point bending at quasi-static loading rates.

The laboratory specimen testing addresses the effects of rapid loading rates on the fracture resistance of ferritic steels. Test and analysis methods have been developed that permit characterization of the J-R curve at these loading rates. These techniques are now being employed in assessing loading rate effects.

Laboratory specimen tests also are being conducted in support of the evaluation of J_m, a modified ductile fracture toughness parameter and the potential application of J_m to specimen types other than the Compact (Tension) Specimen (CT). This effort is expected to identify the appropriate fracture toughness parameter (the traditional J or the modified J_m), define realistic limits of validity for that parameter, and determine appropriate schemes for extrapolating small specimen data to large amounts of crack growth.

The effort to standardize test procedures has contributed to the development of ASTM E813 ("J_{Ic}, A Measure of Fracture Toughness") (Ref. 43) for determining J_{Ic} (the initiation of ductile tearing) and to the development of ASTM E1152 ("Standard Test Method for Determining J-R Curves" (Ref. 44). The on-going effort is expected to lead to the standardization of the direct current potential drop technique for determining crack length during a fracture toughness test. This technique is being used by many laboratories world wide but without a standardized method, which could lead to inconsistent and, perhaps, erroneous results.

This project is integrated with the Degraded Piping Program - Phase II by cross-checking experimental procedures and test results; with the Piping Fracture Mechanics Data Base by providing data to the data base; and with the International Piping Integrity Research Group by providing a test technique for characterizing the fracture toughness of ferritic materials under dynamic loading conditions.

The results of this project have affected the regulatory env[illegible]nt in several ways and the future results are expected to contribute in similar areas. For example, earlier results on stainless steel pipe and welds contributed to the revision of the evaluation procedures for stainless steels given in the Code Section XI Article IWB-3640. The ferritic steel pipe fracture results are contributing to the development of the evaluation procedures for cracks in ferritic steel piping being incorporated into Section XI as Article IWB-3650. The contribution from the laboratory specimen testing will be proven analysis and testing procedures that can be used by other projects without further development.

3.3.3 Fracture Mechanics Analysis of Cracked LWR Piping

The Fracture Mechanics Analysis of Cracked LWR Piping project is a grant to the University of Michigan, under contract number NRC-04-87-113. The project was initiated in February 1987 and is expected to be completed in February 1989.

The objective of the project is to analytically evaluate the effects of the ovalization of a cracked pipe on the load carrying capacity of the pipe. The evaluation will address the effects of local changes in geometry (pipe to elbow welds or pipe to nozzle welds, for example) and local changes in material properties (base metal to HAZ to weld metal) on the ovalization of the cracked section.

The project will draw on data from the Degraded Piping Program - Phase II for benchmarks against which the analyses will be evaluated. Other pertinent data will be developed in the International Piping Integrity Research Group project. Those data will be incorporated as appropriate.

The results of the project will provide an improvement in the accuracy of present pipe fracture analysis schemes. Such improvements in accuracy can lead to better definition of the minimum margins necessary to assure plant safety. This eventually would be reflected in Standard Review Plan 3.6.3, "Leak-Before-Break Evaluation Procedures," which defines the acceptance criteria that must be satisfied to take advantage of the leak-before-break philosophy and associated regulatory changes.

3.3.4 International Piping Integrity Research Group (IPIRG)

The International Piping Integrity Research Group (IPIRG) project is a 3 year program being conducted by Battelle's Columbus Division under contract number NRC-04-86-106. The project was initiated on July 10, 1986. However, as discussed below, the work did not begin until February 1987. The project schedule is referenced to the date Battelle was authorized to begin work.

The IPIRG project is a commercial contract that is jointly funded by an international consortium of governments and industries. The participants have a common goal of developing a justifiable technology base for addressing seismic and dynamic loadings and, more importantly, developing an international consensus on the technology base for accepting the leak-before-break philosophy. The funding arrangement is unique in that all of the participants, governments and industrial entities, essentially are contributing to a common pool of funds used to support the research. The project schedule is linked to the execution of the participation agreements.

The objective of the IPIRG project is to develop, improve, and verify engineering methods for assessing the integrity of nuclear power plant piping containing defects. The project combines a study of the effects of material type, pipe size, and flaw geometry with a study of the effects of seismic and dynamic loading on pipe fracture behavior under typical LWR conditions of pressure and temperature. The project addresses inertial loading, dynamic displacement controlled loading, and the combination of these loadings. The pipe fracture experiments follow a progression in complexity ranging from a relatively simple cantilever configuration used for the inertial loading tests to a piping system representative of typical LWR piping system stiffness and configurations. This representative system will be used for the combined loading experiments. The project also addresses the fracture behavior of fittings (elbows, tees, etc.), the effects of local changes in stiffness on the behavior of cracks in welds (e.g., a crack in the weld between an elbow and straight pipe), and the estimation of leak rates from cracks that penetrate the pipe wall. The fracture of fittings and the effects of local changes in stiffness are being addressed experimentally in this project. The leak rate estimation aspect of the project is discussed in more detail in Section 3.4.

The IPIRG project is integrated with the Degraded Piping Program - Phase II in terms of setting many of the test parameters; with the EPFM Evaluation of LWR Alloys in terms of the dynamic loading test techniques for ferritic steel piping; and with the Piping Fracture Mechanics Data Base in terms of providing materials and material property data to the data base.

The results of this project will be used to assess the margin of safety inherent in present fracture analyses for cracked pipe subjected to seismic and dyamic loadings. Improvements in the prediction of load carrying capacity for cracked pipe eventually will be reflected in Standard Review Plan 3.6.3 which defines the acceptance criteria that must be satisfied to take advantage of the leak-before-break philosophy and associated regulatory changes.

3.3.5 Piping Fracture Mechanics Data Base

The Piping Fracture Mechanics Data Base is being prepared by Materials Engineering Associates as a task of the Structural Integrity of Light Water Reactor Pressure Boundary Components program, contract number NRC-04-84-102. This four year project was initiated in January 1984 and is expected to be completed in August

1988. However, the NRC expects to continue the data base effort and a competitive procurement is planned for mid-1988.

The objective of the data base project is to prepare a comprehensive data base of fracture toughness and related material property data for representative LWR piping materials. The effort involves collecting available data from other NRC contractors and from pertinent industry sources. When pertinent data are not available, appropriate materials are to be procured and tested. The data are to be assembled into a numeric data base that is accessible to the NRC. The numeric data base is to provide a central repository for the data and a capability to reprocess the data as new data analysis schemes are developed.

This project interacts with every pipe fracture research project where material property data are developed or where material property data are needed. The data base has been of use to other research contractors, the NRC staff, and the nuclear industry. Use by other research contractors has been to provide material property data for analytical efforts and as supporting data for comparison to other experimental efforts. Use by the NRC staff and the nuclear industry has been as a generic fracture toughness and material property data base. The staff and industry can and have used the generic data base to provide material property data when plant specific data are not available. This approach has been addressed in developing Standard Review Plan 3.6.3, "Leak-Before-Break Evaluation Procedures."

3.3.6 Aging of Cast Stainless Steels

The Long-Term Aging Embrittlement of Cast Stainless Steels in LWR Systems is a multi-year project being conducted by the Argonne National Laboratories under FIN A2243. The current scope of work is expected to be completed in 1989.

The primary objectives of this project are (1) to investigate the significance of in-service embrittlement for cast duplex stainless steels under LWR operating conditions, and (2) to evaluate possible remedies to the embrittlement problem for existing and future plants. Current assessments of long-term embrittlement are based primarily on extrapolation from high-temperature laboratory data and on Charpy impact data. Additional data on the kinetics and extent of embrittlement at LWR operating temperatures are being developed to evaluate the significance of in-service embrittlement in the context of changes in mechanical properties and fracture resistance.

The research will (1) characterize and correlate the microstructure of reactor components and laboratory-aged material with loss of fracture toughness to identify the mechanism of embrittlement, (2) determine the validity of extrapolation of experimentally observed embrittlement to long-term aging at reactor operating temperatures, (3) characterize the loss of fracture toughness in terms of fracture mechanics parameters to provide data needed to assess the safety significance of embrittlement, and (4) provide additional understanding on the effect of key compositional and metallurgical variables on the kinetics and degree of embrittlement.

The research plan calls for developing microstructural and mechanical properties data on 19 experimental heats (with compositions spanning the commercial compositions) and 6 commercial heats, as well as reactor-aged material of CF-3, -8, and -8M cast stainless steel. As a base line reference, chemical composition, hardness, ferrite content and morphology, composition of ferrite and austenite phases, and the grain structure of the various unaged materials have been determined. The materials are being aged at 290, 320, 350, 400, and 450°C for times ranging from a few hundred hours up to 50,000 hours. Specimens are tested periodically to assess the degree of embrittlement and to provide further evidence on the mechanism of embrittlement. The reactor-aged material will provide a realistic check on the laboratory aging to assure that the physical processes are consistent.

The aging data will ultimately be used to develop a predictive model of thermal aging based on chemistry, microstructure, thermomechanical treatment, service conditions, and service time, that is representative of the actual mechanism(s) controlling the embrittlement. The model then can be used confidently to predict changes in mechanical and fracture properties for use in fracture analyses.

The project is integrated with the Piping Fracture Mechanics Data Base project in that the data base project has performed some of the material property and fracture toughness testing of the cast stainless steel materials, and it is integrated with the Degraded Piping Program - Phase II in that the Degraded Piping Program has contributed materials to the aging program (both aged and unaged) and in that the aging program is aging a pipe section which will be utilized in a pipe fracture experiment.

The results of this program are expected to affect the regulatory environment in at least two ways. First, data regarding changes in the mechanical and fracture toughness properties are expected to provide guidance to the Code for possible revision of Section XI, Article IWB-3640 regarding applicability of those evaluation criteria to cast stainless steels. Second, the data will become part of the Piping Fracture Mechanics Data Base and will provide reference data on aged cast stainless steels for use in leak-before-break analyses performed by the NRC and industry, and finally, possible fixes will be recommended for restoring fracture toughness in aged components and for minimizing in-service toughness loss for new components.

3.3.7 Estimation of Large Leak Rates From Piping and Piping Components

This project is in the early planning stages. The objective of the project is to develop leak rate data essential to defining a replacement to the double-ended guillotine break (DEGB).

Public comments on the modifications of GDC-4 advocated the extension of the rule to relax pipe rupture postulation requirements for containment design, ECCS performance, and environmental qualification of electrical and mechanical equipment. The Commission response, in part, noted that there was no intent to consider near-term changes to ECCS or containment design bases. Further, the Commission was not prepared to propose new environmental design criteria for temperature, pressure, humidity or flooding. However, until a suitable replacement design basis could be established for environmental qualification, the Commission was willing to consider case-by-case relaxations in environmental qualification requirements. More recently, the Commission has moved to seek public comment on additional applications of leak-before-break technology, specifically in the areas of environmental qualification and ECCS performance requirements. This research project will contribute to defining a suitable replacement to the DEGB.

Considering a replacement to the DEGB dictates that accident scenarios other than pipe breaks be evaluated. Scenarios that might become the limiting consideration include valve bonnet failure, stuck-open relief valves, manway cover failure, or pump seal failure, in addition to large leaks from piping. Some of these scenarios have been, and are being, evaluated in other research activities. This project will provide reference information regarding credible leakage rates from pipe cracks.

The research contemplated for this project involves the development and experimental validation of a mathematical model for predicting the leak rates from large cracks in piping and piping components. The program emphasis will be placed on developing experimental data concerning crack opening areas and leak rates. These data then will be used to benchmark the leak rate analysis. The validated analysis ultimately would provide input to a "systems" analysis to evaluate equipment qualification requirements, e.g., pressure, temperature, and humidity versus time profiles.

3.4 Leak Detection and Leak Rate Estimation Models

The objectives of the leak detection and leak rate estimation modelling efforts are (1) to assess the sensitivity and reliability of the leak detection systems used in nuclear power plants both inside and outside containment, and (2) to assess the accuracy of leak rate estimation models used in leak-before-break analyses. The efforts involve both experimental and analytical work.

Current leak detection requirements focus on detection inside the containment building and are embodied in Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems" (Ref. 45). The Regulatory Guide recommends the following three methods be employed for leak detection:

- o Sump flow (mandatory),
- o Airborne-particulate radioactivity (mandatory), and
- o Condensate flow rate or airborne gaseous radioactivity.

The Regulatory Guide further recommends a sensitivity of 1 gallon per minute for equipment used to monitor identified and unidentified leakage sources. A review of the technical specifications pertaining to leak detection requirements was conducted for the Piping Review Committee and is reported in NUREG-1061 Volume 1 (Ref. 46). The report shows that of the 74 LWRs surveyed, all plants use at least one of the two required systems; 66 use sump monitors and 71 use particulate monitors. Sump monitoring is the primary method of detecting leaks. The technical specification allowed limits on unidentified leakage are 1 gallon per minute for all PWRs and 5 gallons per minute for most BWRs.

However, in four of the eight plants evaluated as part of the Integrated Plant Safety Assessment-Systematic Evaluation Program (Refs. 47-54) a one gallon per minute leak would not be detected in one hour nor did they have three leakage monitoring systems, as suggested in Regulatory Guide 1.45. Despite the shortcomings in current leak detection systems discussed in NUREG-1061 Volume 1, they were deemed adequate to detect leakage and require plant action in 1 day for through-wall cracks 4 to 10 in. long in 12 to 28-in. diameter. However, these systems may not be adequate for detecting leaks in smaller diameter piping.

As noted in Section 3.3, the NRC currently is preparing Standard Review Plan Section 3.6.3 which defines the criteria that must be satisfied to demonstrate "leak-before-break". Consequently, leak detection sensitivity and reliability are taking on a new significance. One must perform crack stability analyses using crack sizes that will produce detectable leakage when subjected to normal loads. Further, the margin of 10 on leak detection sensitivity used in the analysis reflects uncertainty in both the capabilities of leak detection systems and the leak rate estimation models. A practical consequence of the margin and leak detection sensitivities of conventional systems is that small diameter piping may be excluded from receiving the benefits of leak-before-break unless more sensitive and reliable systems are employed.

The leak-before-break acceptance criteria stipulate that leak detection systems used outside containment must be equivalent to the Regulatory Guide 1.45 systems. In meeting this requirement one must justify the sensitivity and reliability of the system. Further, simply employing systems equivalent to the Regulatory Guide 1.45 systems creates the same potential limitation for small diameter pipe outside containment as that encountered inside containment, i.e., small diameter pipe may not meet all of the acceptance criteria unless more sensitive and reliable leak detection systems are used.

Directly linked to the leak-before-break analyses for a given line is the determination of a through-wall crack length given a leak detection sensitivity and loading level. Approaches to estimating leak rates range from simply multiplying the crack opening area by a constant, to very sophisticated two-phase fluid flow models coupled with elastic-plastic fracture mechanics methods used to predict crack opening areas. Most of the data available for validating these models lie in the 1 gpm to 50 gpm range. However, applications for smaller diameter lines require predictions for leak rates much less than 1 gpm. There are questions about the validity of the models in this low flow regime, and about the possibility of particulates in the fluid "plugging" the crack flow path thereby reducing the flow rate below the detectable limit.

The leak detection and leak rate estimation model research addresses many of these problems. The nuclear industry has explored a number of leak detection systems and part of the NRC research has been devoted to independently evaluating these systems. The research also examines the viability of advanced leak detection systems, such as acoustic monitoring, to provide guidance on potential improvements over the current leak detection methods. This work was supported in the recommendations of the Piping Review Committee (Ref. 3).

The leak rate estimation modelling effort explores topics that have gained significance as the leak-before-break evaluation criteria have evolved. For example, the accuracy of the leak rate estimation model can have a significant influence on the so-called leakage size crack used in the crack stability analyses.

The following subsections describe the research projects that address leak detection and leak rate estimation modelling.

3.4.1 Leak Detection

The Leak Detection project is a multi-year project being conducted by the Argonne National Laboratories under FIN A2250. The current scope of work is expected to be completed in 1988 with recommendations incorporated into regulatory codes and standards in 1989.

The objectives of the project are (1) to assess the reliability of current leak detection systems to provide guidance to the NRC on potential deficiencies, and (2) to conduct laboratory and field tests to assess the adequacy of acoustic techniques to detect, locate, and size leaks. The effort will include tests at a reactor site using an advanced on-line acoustic monitoring system. The effort also will contribute to the experimental assessment of leak detection techniques suggested by the industry to establish their sensitivity and reliability.

This project has evaluated presently used leak detection techniques and criteria and has shown that they are not necessarily adequate to detect small leaks even when caused by large tight cracks. Further, radiation monitors are deemed inherently unreliable because of the high false alarm rate and the limitations caused by the high radiation background levels. (Ref. 55)

The assessment of acoustic monitoring techniques has made use of laboratory tests to develop correlations between leak rate and the acoustic signal, to establish minimum sensitivities, and to develop the acoustic signature characteristics of flow through IGSC cracks versus other acoustic sources (such as back flow through a check valve). The technique also has been tested successfully using simulated leaks at a reactor site that is under construction. Following these successful demonstrations, efforts have been underway to permit testing in the environment of an operating reactor, providing an assessment of the technique in a realistic environment.

The recent efforts involved broadening the data base with additional leaks and cracks at larger flow rates, finalizing data analyses and operational procedures, and defining the number of sensors required for a given coverage. The remaining effort is to conduct tests in the operating reactor environment, in conjunction with other acoustic emission monitoring work, to validate the sensitivity and reliability of leak detection techniques.

This project is integrated with the International Piping Integrity Research Group in that the IPIRG program is addressing leak rate estimation models and in that there is the potential for using acoustic monitoring techniques on one or more of the IPIRG experiments to define the point of break through as well as assessing the technique during and after a simulated seismic event.

This project is expected to affect the regulatory process by the development of a revision to Regulatory Guide 1.45 by incorporating acoustic monitoring techniques for use inside and outside the containment building, and by providing guidance on improvements in the sensitivity and reliability of other leak detection techniques. Establishing credible sensitivity and reliability values for these techniques would impact the evaluation of leak-before-break analyses in the context of establishing the leakage size crack used in the stability analyses.

3.4.2 Leak Rate Estimation Model Validation

The Leak Rate Estimation Model Validation project is being performed by Battelle's Columbus Division as part of the International Piping Integrity Research Group project, contract number NRC-04-86-106. This project was initiated on September 1, 1986, and the final report is expected by March 31, 1988.

The objective of this project is to provide a refined and experimentally validated leak rate estimation model. The project has three major activities: (1) developing a comprehensive leak rate estimation model, (2) experimentally validating the model, and (3) evaluating the potential effects of particulate plugging on leak rates. The model development and evaluation of plugging are analytical efforts while validating the model includes leak rate testing.

Developing the leak rate model involves examining various thermal-hydraulic flow models, identifying ranges of applicability for those models, and consolidating the most appropriate model(s) into a predictive scheme capable of predicting fluid flow rates over the range of interest. The leak rate modelling effort also includes examining models for predicting crack opening areas and comparison of those models to crack opening area data from the Degraded Piping Program - Phase II and other related programs. Finally, the thermal-hydraulic model and the crack opening area model will be combined into a single model capable of predicting leak rates given fluid conditions, crack size, crack geometry, pipe size, and loading conditions. Conversely, the model will be able to predict a leakage size crack given a specified leak rate, fluid conditions, pipe size, and loading conditions.

The experimental validation activity draws upon existing leak rate data. A data base of experimental leak rate data has been prepared and from that data base an experimental test matrix was developed such that if all of the experiments were conducted a comprehensive data base would exist. However, only select experiments have been conducted as part of this project. The emphasis was placed on developing data in the flow regime most pertinent to leak-before-break analyses. The testing emphasis was placed on leak rates less than two gallons per minute. Additional testing may be performed depending upon funding level of the IPIRG program and the desires of the IPIRG participants.

Three important parameters affect plugging. These are the crack opening area, the particulate size distribution, and the concentration of particulates in the fluid. The particulate plugging effort is to determine the size distribution and concentration of particulates likely to exist in nuclear power plant piping systems, and to assess the potential for these particulates to cause partial plugging of pipe cracks as a function of the crack geometry and flow conditions.

This project is integrated with the Degraded Piping Program - Phase II by using the crack opening area data generated in the Degraded Piping Program to validate the crack opening area predictive model. This project also is integrated with the Leak Detection project by providing acoustic data from the leak rate experiments in this project.

The results of this project are expected to affect the regulatory process chiefly by improving the accuracy of leakage size crack predictions used in the leak-before-break analyses, and in reducing, or at least quantifying, the uncertainty in leak rates due to the potential for particulate plugging of the crack.

4. INTEGRATION OF THE PIPING RESEARCH RESULTS WITH THE REGULATORY PROCESS

4.1 Anticipated Changes in Regulations, Codes, and Standards

4.1.1 Integration of Piping Design Research Results

The EPRI/NRC Piping and Fitting Dynamic Reliability Program (PFDRP) will provide and justify recommended changes to the piping dynamic stress rules given in Subsections NB, NC, and ND of Section III of the Code. Already the results from this program have supported the development of Code Case N-451 (Ref. 17) concerning dynamic stress allowables for Class 1 piping, a similar Code Case for Class 2 and 3 piping, and current PVRC activities intended to revise the NRC's piping functionality criteria. A comprehensive set of Code changes will be proposed when the PFDRP ends in 1988. There has been a continuing interaction with the Code committees and the NRC staff to keep all groups informed and to have feedback into the research program. It should be noted that there are now many representatives of the Section III groups and the PVRC who are directly involved with this cooperative program, or who serve as consultants.

The primary goal of the cooperative EPRI/NRC program is to provide changes to the dynamic inertial stress criteria for piping. However, as a result of these changes, other design rules may need to be revised. The results of new research to systematically evaluate piping experience data will help support these further changes. Research directed at evaluating piping support rules could lead to future revisions to Subsection NF of the Code.

Research concerning nozzle and branch connection flexibility will result in changes to Subsections NB, NC, ND, and possibly NE of Section III of the Code. As a result of ongoing ORNL research, the Code Working Group on Piping currently is considering changes to branch connection design and the Subgroup on Design is considering significant changes to nozzle and piping design rules. New guidance on calculating nozzle flexibility (replacing guidance now given in WRC Bulletins 107 (Ref. 56) and 297 (Ref. 32) is now being considered by the PVRC Subcommittee on Reinforced Openings and External Loading, and may eventually be referenced in the NRC's Standard Review Plan (SRP).

Research concerning piping response estimation will lead to changes to Standard Review Plan Section 3.9.2 and to associated regulatory guides. Regulatory Guide 1.61 (Ref. 28) and Appendix N of the Code will be revised in 1988-1989 to incorporate changes in pipe damping criteria that have evolved from previous work by the PVRC and INEL (NRC-sponsored) and current work by the ASME and Bechtel (EPRI-sponsored). NRC-sponsored work on the cumulative effect of piping criteria changes and on time-history damping will play a strong role in the staff acceptance of these new pipe damping criteria. A new NRC position on the use of the Independent Support Motion method will be developed in 1988 as a result of NRC and EPRI sponsored research. Regulatory Guide 1.92 "Combining Modal Response and Spatial Components in Seismic Response Analysis" (Ref. 57) will be updated to incorporate changes justified by NRC research on the independent support motion method, closely spaced modes, and the combination of high frequency modes. A generic NRC position on Bechtel energy absorbing supports should be established in 1988 as a result of an NRC-sponsored study. Regulatory Guide 1.122 (Ref. 29), may be revised in the future to incorporate changes resulting from research on peak-shifting techniques and research identifying the uncertainty associated with piping response calculation.

4.1.2 Integration of Environmentally-Assisted Cracking Research Results

The program is intended to assess the proposed near-term and long-term fixes for IGSSC in BWR stainless steel piping and provide support to NRR in establishing NRC positions on this and related Unresolved Safety Issues (A-1+, A-42) (Ref. 9). This work supports development of licensing criteria described in NUREG-1061 Volume 1, and and provides the data needed for modification of Regulatory Guide 1.44 (Ref. 58). Experimental results provide NRR with data and engineering information to substantiate the technical basis for licensing decisions relating to BWR pipe cracking.

This program has provided data and information on the "fixes" proposed by industry for mitigation of intergranular stress corrosion cracks including induction heating stress improvement, hydrogen water chemistry, repair by weld clad overlay, and replacement material Type 316 NG stainless steel. Further, the program will provide data to determine the extent, if at all, to which repaired cracks may continue to grow and thus threaten catastrophic failure of the pipe system especially when it is known that the weld clad overlay tends to preclude effective ultrasonic inspection to monitor any further growth. In 1984 this program contributed directly to the NRC Piping Review Committee study on IGSCC of BWR piping which was reported in NUREG-1061 Volume 1, and its implementation by NRR staff as described in SECY 83-287C (Ref. 59). The results of this project provide data and engineering for the NRR implementation document NUREG-0313, Rev. 2 (Ref. 11).

The crack growth rate data being generated is being utilized in the Code Section XI procedure to evaluate acceptability of cracks in pipes. Fatigue life research results are expected to provide the data needed to support a broad modification of the Section III fatigue design procedures to account for the effects of environment.

4.1.3 Integration of Pipe Fracture Research Results

The pipe fracture research program addresses the experimental validation of pipe fracture analysis methods, the technical justification for margins used in those analyses, and the definition of a replacement to the double-ended guillotine break criterion. The research results are anticipated to impact the regulatory environment in several ways.

The research programs that focus on pipe fracture behavior and the prediction of pipe fracture (see Sections 3.3.1 through 3.3.5) are expected to provide input to the refinement of the criteria used to justify leak-before-break and to the justification of the margins used in those analyses. Presumably, these research efforts will lead to less restrictive criteria and reduced margins in the analyses.

The results are expected to impact Standard Review Plan Section 3.6.3 by refining the criteria and margins. Finally, the results have been used in validating the flaw evaluation procedures for stainless steels as implemented in Section XI Article IWB-3640, and in developing similar criteria for evaluating flaws in ferritic steel piping. These criteria will be implemented in Section XI as Article IWB-3650.

The material property studies (Piping Fracture Mechanics Data Base and Aging of Cast Stainless Steels) are expected to provide data useful in plant specific analyses. Further, data from these programs are expected to be useful to the Section XI committee in developing flaw evaluation procedures for ferritic steel piping and in considering the applicability of flaw evaluation procedures for stainless steels to the cast stainless steels.

Finally, the program on defining a replacement to the postulated double-ended guillotine break criterion (Estimation of Large Leak Rates From Piping and Piping Components) is expected to contribute to the technical basis justifying further revision to 10 CFR 50, Appendix A, General Design Criterion 4. This revision would impact equipment qualification requirements. Consequently, changes would be expected in the pertinent Regulatory Guides and the Standard Review Plan.

4.1.4 Integration of Leak Detection and Leak Rate Estimation Research Results

The leak detection research is expected to lead to the development of regulatory guidance concerning use of acoustic monitoring techniques, and to provide guidance on the sensitivity and reliability of other leak detection techniques. This could lead to changes in the Standard Review Plan Section 3.6.3 margin on leak detection sensitivity.

The leak rate estimation model research is expected to lead to further refinements in the procedures for predicting the leakage size crack used in the leak-before-break analyses. Further, the work is expected to contribute to quantifying the uncertainty in leak due to the potential for particulate plugging of the crack.

4.2 Anticipated Schedule for Integrating the Research Results

Figure 1 depicts the major piping research program activities to be completed by fiscal year. Schedule information for each research project was discussed in Section 3.

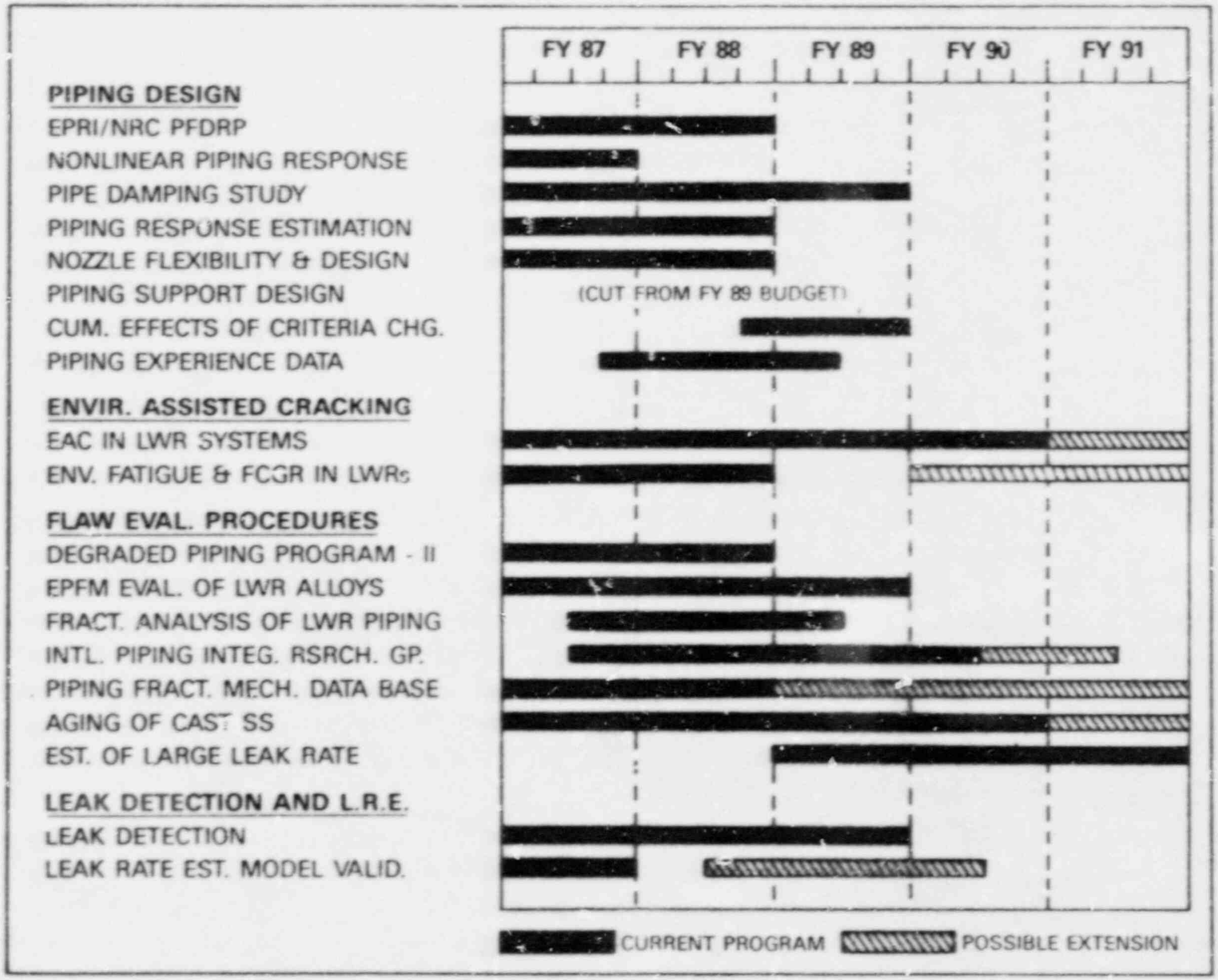

Figure 1 Piping Research Program Schedule

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13. ABSTRACT (200 words or less)

This document presents the piping research program plan for the Structural and Seismic Engineering Branch and the Materials Engineering Branch of the Division of Engineering, Office of Nuclear Regulatory Research. The plan describes the research to be performed in the areas of piping design criteria, environmentally assisted cracking, pipe fracture, and leak detection and leak rate estimation. The piping research program addresses the regulatory issues regarding piping design and piping integrity facing the NRC today and in the foreseeable future.

The plan discusses the regulatory issues and needs for the research, the objectives, key aspects, and schedule for each research project, or group of projects focussing of a specific topic, and, finally, the integration of the research areas into the regulatory process is described. The plan presents a snap-shot of the piping research program as it exists today. However, the program plan will change as the regulatory issues and needs change. Consequently, this document will be revised on a bi-annual basis to reflect the changes in the piping research program.

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