TECHNICAL ISSUES AND U.S. NUCLEAR REGULATORY COMMISSION'S RESEARCH RELATED TO THE PRIMARY PRESSURE BOUNDARY

L.C. Shao

Division of Engineering, U.S. Nuclear Regulatory Commission, Washington, DC, USA

ABSTRACT
This paper presents an overview of regulatory and technical issues and the U.S. Nuclear Regulatory Commission's research related to safety and structural integrity of the primary pressure boundary. Primary pressure boundary issues include the reactor pressure vessels, piping, steam generators, and reliability of non-destructive examination methods for detecting and sizing flaws.

INTRODUCTION
The U.S. Nuclear Regulatory Commission (NRC) has a continuing research program aimed at assuring safe operation of light-water reactor power plants. A part of NRC's research programs is to examine structural integrity issues for passive components such as the reactor pressure vessels, primary piping, and the steam generator tubes. This research program focuses on the critical components to assure that they will perform safely under normal operating and postulated accident conditions during the plant life. Additionally, the results of this research program are being utilized in the NRC's activities for license renewal and advanced reactor design reviews.

As nuclear power plants continue to age, plant operators and regulatory agencies alike must face system integrity and safety issues in a realistic manner. We cannot ignore or try to minimize the significance of operational issues. At the same time, it is not appropriate to continue to impose overly "conservative" solutions to issues in lieu of finding the correct technical solution. Now, more than ever, an aggressive research program is needed to help find better technical solutions which maintain adequate levels of safety and at the same time eliminate unnecessary conservatism.

In the last two years, two plants in the United States have faced significant technical issues; issues that have challenged the industry and the NRC alike. In both cases, the owners have elected to permanently shutdown the plants for economic reasons. Other sources of power were available at lower costs. However, the technical issues encountered at these plants - vessel embrittlement issues and steam generator tube cracking issues - need to be addressed for other plants in the U.S. and in other countries.

To the end of finding sound technical solutions to the structural integrity and safety issues facing the nuclear industry, NRC has expanded its interactions with industrial groups such as the Nuclear Utilities Management and Resources Council (NUMARC) and the Electric Power Research Institute (EPRI) in the U.S., and with the international technical community in general. The paper describes the regulatory and technical issues being addressed, related to the primary pressure boundary, and summarizes the research activities underway to find the technical solutions to support regulatory activities.
STRUCTURAL INTEGRITY ISSUES

The structural integrity research program has been designed to assure the integrity of the primary pressure boundary components and related systems and components. Emphasis is placed on the reactor pressure vessels, piping systems, and steam generators. Non-destructive examination of these components is included in the research program because of the important role it can play in assuring component integrity. The regulatory and technical issues, and the research program for each of these areas are discussed below.

1.0 REACTOR PRESSURE VESSEL INTEGRITY ISSUES

The Reactor Pressure Vessel (RPV) is the key component in the primary pressure boundary. It houses and supports the reactor core and facilitates channeling of the coolant from the inlet piping, through the core, to the outlet piping. It is also the only element in the primary pressure boundary for which engineered safety systems cannot provide protection in case of rupture. Because of the importance of the RPV, there is a continuing effort to develop and refine the technical bases for evaluating the structural integrity of this key component, and to ensure its continued safe operation. The reactor vessel materials adjacent to the core are subject to loss of fracture toughness after years of irradiation exposure. Embrittled reactor vessels are more vulnerable to pressure and thermal loading transients.

The goal of the NRC's RPV research program is to provide validated methods for predicting pressure vessel integrity under operating and accident conditions. Achieving this goal is necessary to support regulatory actions to resolve a number of regulatory issues. Resolving these regulatory issues depends on understanding and solving the underlying technical issues. There are seven regulatory issues that involve reactor pressure vessel integrity: Pressurized Thermal Shock, Pressure-Temperature Limits, Low-Temperature Over-Pressure Protection system setpoints, requirements for vessels fabricated using materials with low Charpy upper-shelf energy, thermal annealing criteria, plant license renewal criteria, and reliability of detection and sizing of flaws. It is interesting to observe that resolution of each of these regulatory issues depends on resolving technical issues in the same four areas: embrittlement, fracture mechanics, material properties, and non-destructive examination.

Pressurized Thermal Shock (PTS) - It is a severe and relatively sudden decrease in coolant temperature along with either sustained high system pressure or subsequent re-pressurization. Transients of this type have occurred in a few Pressurized Water Reactors (PWR's), and there have been precursors for similar events at other plants. Transients of this type represent a potentially significant challenge to highly embrittled reactor pressure vessels. High thermal gradients may cause the existing cracks in the material to propagate and the pressure loading may cause through-wall cracking. Completely assessing the potential significance involves having the tools to correctly assess the level of embrittlement and its influence on the material properties, the analysis tools to fully evaluate the potential for vessel failure, knowledge of the flaw sizes that might exist in RPVs, and the non-destructive examination tools necessary to determine vessel-specific flaw distributions if this is warranted. These tools are necessary in evaluating the efficacy and cost-effectiveness of potential mitigative measures, such as heating safety injection water or thermal annealing.

Pressure-Temperature (P-T) Limit for Heatup and Cooldown - The safe operation of reactor pressure vessels depends, at least in part, on defining the pressure and temperature regimes where catastrophic failure of the vessel is not likely. When the reactor vessel is cold, it may be subjected to brittle fracture during heatup and cooldown. In the U.S., the allowable pressure-temperature limits are defined by Appendix G of Section III (or Section XI) of the ASME Boiler and Pressure Vessel Code, and by the procedures specified in
the NRC Standard Review Plan 5.2.2 (NUREG-0800). These procedures make use of a linear elastic fracture mechanics analysis, a reference flaw size -- the so-called quarter thickness flaw -- and a reference fracture toughness curve adjusted for the vessel-specific level of embrittlement. With increasing levels of embrittlement, the allowable region for operation can become prohibitively small so that start-up and shutdown operations are difficult if not impossible without violating the administrative limits. The technical issues that must be addressed here are making certain that the embrittlement estimates are appropriate without being unnecessarily conservative. Similarly, the reference flaw size was chosen as the quarter thickness flaw, and no "credit" is given for non-destructive examination of the vessel, even though we have requirements to periodically inspect the vessel, and are implementing requirements to make the inspections more effective. We now should give consideration to a vessel-specific reference flaw size, that would be determined, at least in part, by the periodic inspection program. However, if smaller flaw sizes are to be considered, fracture mechanics analysis methods must be made more rigorous, taking into consideration the effects of the stainless steel cladding, and residual stresses. Similarly, we need criteria to evaluate the usefulness of mitigative measures, such as thermal annealing.

Low-Temperature Over Pressure (LTOP) Protection - As noted, the pressure-temperature (P-T) limits are administrative limits imposed to protect against catastrophic, or non-ductile, failure of the pressure vessel. However, service experience indicated that the administrative limits were being violated too frequently. Consequently, requirements for a protection system were imposed; the so-called low-temperature over-pressure protection or LTOP system. The setpoints for this system -- the temperature below which the system must be operative and the pressure to which system pressure must be limited -- were established to assure that the administrative pressure-temperature limits were not exceeded. However, with increasing levels of embrittlement, the LTOP system setpoints were imposing a significant burden on several plants, and would have made it impossible for some plants to startup without challenging the physical protection system. This has an adverse impact on safety because of the relatively high potential for safety relief valves to stick open once lifted. Thus, imposing very conservative criteria to protect the conservative pressure-temperature limits, which were imposed to protect against non-ductile failure of the pressure vessel, were actually creating a situation that could challenge plant safety, with little improvement in protection of the reactor pressure vessel.

The U.S. industry proposed revisions to the analyses and criteria used to determine the LTOP setpoints, and the NRC's research program evaluated these criteria. While there were a few areas identified that warranted further discussion, the proposal was technically sound. However, it imposed a significantly more rigorous fracture mechanics analysis, and would have imposed more rigorous inspection requirements.

Before a final determination concerning the industry proposal was made, Section XI of the ASME Code passed a Code Case that tied the LTOP system setpoints to the existing pressure-temperature limits. This approach is consistent with current Code philosophy which allows physical pressure to exceed specific administrative limits by small amounts which are determined by pressure relieving devices. While the NRC has not endorsed this Code Case yet, the approach avoids the need for more rigorous analyses and inspections.

The evaluation of LTOP setpoints has highlighted several weaknesses and some strengths in structural integrity technology. First, the embrittlement estimates can be very conservative, and more realistic assessments are needed. Next, the use of a lower-bound reference fracture toughness curve must be questioned. If we are interested in addressing protection system set points, would it not be more reasonable, and adequately conservative to use a fracture
toughness curve that reflects the potential for crack initiation, the $K_{IC}$ curve for example? Also, the method for indexing the generic reference curve to vessel-specific embrittlement estimates appears to be introducing a significant level of conservatism. The fracture mechanics analyses proposed by the industry reflected a very detailed treatment of the vessel loadings. However, there were many aspects of these analyses that needed further discussion and validation; the effects of cladding and residual stresses for example. Finally, making use of a smaller reference flaw size would necessarily require more rigorous inspections, possibly requiring the use of advanced inspection techniques, but certainly requiring careful assessment of the performance of the inspection teams.

**Low Charpy Upper-Shelf Energy Issue** - The preceding three "regulatory" issues stem from concerns about non-ductile, or catastrophic failure of the pressure vessel. In the U.S., we also have a concern with the ductile failure of some pressure vessels that were fabricated with materials that have a relatively low resistance to ductile tearing, as characterized by low values of Charpy upper-shelf energy. Until recently, a particular class of welds -- high-copper welds fabricated using Linde 80 flux -- were believed to be the major concern. However, test data on A-302 B plate material showed remarkably low ductile tearing resistance for thick section specimens. Information provided by the U.S. industry in response to the NRC's Generic Letter 92-01 suggested that some pressure vessels in the U.S. could have a concern with low tearing resistance plate materials. However, the issue has identified the need for better embrittlement estimates, better fracture analysis methods, and a better relationship between laboratory specimen data and structural performance. The issue also has raised the question of the potential for ductile tearing to cause a fracture-mode conversion to cleavage fracture that would not be predicted from a simple linear elastic or elastic-plastic fracture mechanics analysis. This is particularly important to the PTS evaluations.

**Thermal annealing** - It is a means of recovering the loss in fracture toughness due to embrittlement induced loss of ductility. It is a potential mitigative measure where cleavage fracture is a concern. It also has the potential for mitigating concerns with ductile tearing. However, to make use of any annealing method, a better understanding of the effects of thermal annealing on the recovery of fracture toughness and re-embrittlement rates is needed. Further, careful consideration of the engineering aspects of the annealing process is needed. For example, what temperature limits should be imposed for concrete structures? Should surveillance specimens be required during the annealing process? What post-annealing inspections should be required? These and many other questions must be addressed before thermal annealing will be accepted in the U.S. Yet, this process, which has been widely used by the Russians and in Eastern Europe, offers the potential for significantly extending the useful life of the reactor pressure vessel.

**Plant License Renewal** - Nuclear power plants represent a significant financial investment for the owners. Not surprisingly, there has been much interest in extending the license period for these plants. However, each of the integrity issues discussed above has the potential for limiting the useful life of reactor pressure vessels. Thermal annealing has the potential for mitigating these life-limiting issues. There may be other issues, perhaps not as well defined as those discussed above, that could similarly limit the useful life of the reactor pressure vessels, piping and steam generators.

### 1.1 Reactor Pressure Vessel Safety Research

The reactor pressure vessel (RPV) safety research program is a broad based program that addresses each of the critical technical issues that contribute to resolution of the regulatory issues. The RPV safety research program is the NRC's largest and oldest research effort. In the 27 year history of the program, the NRC and its predecessor the AEC, have invested over $100 million
in this critical research area. Despite this significant investment, much
remains to be done. Further, the remaining issues are some of the most
challenging yet to be addressed.

Fracture Mechanics Research: The fracture mechanics research effort
includes both analytical and experimental research. This research makes use
of state-of-the-art analytical methods to find engineering solutions to
practical issues. The work has addressed the effects of cladding, located at
the inside wall of an RPV to protect the base material, on crack initiation
under pressurized thermal shock (PTS) loading, and has shown that including
the fracture toughness of the cladding in the analysis can have a significant
impact on the crack sizes that would be predicted to lead to vessel failure.
This work could have a significant impact on the way PTS evaluations are
conducted in the U.S. in two ways. First, changing the fracture analysis
methodology to include cladding fracture toughness will require a somewhat
more rigorous analysis, but it will lead to lower probability of failure
estimates; much lower in some cases. The other impact would come in determin-
ing the flaw sizes that must be detected by non-destructive examination
techniques in order to alter the flaw size distributions that would be
considered in the evaluations. While the analysis results are encouraging,
work remains to be done to provide a better basis for the material properties
for the irradiated cladding materials. This remains a significant uncertainty
in the methodology to include cladding effects, both for PTS evaluations and
for other vessel integrity evaluations that consider relatively shallow flaws.

The fracture mechanics research has shown an elevation in the fracture
toughness for the shallow flaw depths typically considered in PTS evaluations.
Undoubtedly, this elevation is due to a loss of crack-tip constraint in the
test specimens. However, a similar loss of constraint would be expected for
shallow flaws in the pressure vessel, which would lead to an increase in the
fracture toughness that would be used in the fracture evaluation and a
resulting reduction in the probability of failure estimates.

This was encouraging information. However, it was mitigated by the concern
that bi-axial loading effects could increase the effective crack-tip con-
straint, leading to lower fracture toughness. The analytical results suggest-
ed that in reality, the fracture behavior might be unaltered or, perhaps, even
worse than anticipated based on the traditional "deep crack" data. Indeed,
preliminary experimental results suggest that bi-axial loading does increase
crack tip constraint and reduces the effective fracture toughness. Additional
work is underway to more fully evaluate the situation, and to develop an
analysis methodology that correctly includes both effects.

It is clear from this work that crack-tip constraint plays a major role in
correctly predicting the actual fracture behavior. This observation has been
reinforced by our participation in the efforts of the CSNI's (Committee on
Safety of Nuclear Installations) Principal Working Group No. 3, Fracture
Assessment Group, in evaluating the large-scale pressurized thermal shock
experiments that have been conducted in several countries. It seems very
likely that the variability in the experimental results, and in the prediction
of those results, stems largely from changes in crack-tip constraint and how
they are modeled by the analysts.

The NRC has initiated a multi-year effort aimed at resolving the questions
about how to incorporate crack tip constraint in vessel fracture analyses.
This work is largely analytical but is being supplemented by focused experi-
mental efforts. The constraint effects research effort is broad based,
bringing together the best theorists and practitioners to develop a fracture
analysis methodology that is theoretically valid, but that can be implemented
in a regulatory framework. Finally, the analysis methodology will be experi-
mentally validated so that it can be used with confidence in safety analyses.
Validation, of course, has been a cornerstone of the NRC's pressure vessel research for many years. Several large-scale experiments have been conducted in the NRC's Heavy Section Steel Technology program, and analyses of experiments conducted in other countries have been conducted as part of the CSNI program. However, comparison of the analysis and experimental results has been disappointing. The constraint effects research will lead to an improved analysis method, and we are looking for an appropriate validation experiment(s). Plans for large-scale experiments in the United Kingdom may provide the needed validation. However, at this stage the search is continuing.

Embrittlement Research: The reactor pressure vessel may become very embrittled after years of exposure to neutron irradiation. The embrittled vessel is more susceptible to sudden changes in pressure and temperature across the vessel wall-thickness. The embrittlement research effort involves: analytical and experimental research to improve and validate the methods for predicting neutron fluence and energy spectrum at the pressure vessel wall; test reactor irradiations to determine the effects of key variables on embrittlement trends in a controlled environment; use of advanced techniques for determining the mechanisms that control the embrittlement; statistical methods to develop mathematical models for predicting the level of embrittlement given information concerning the neutron fluence, the material chemistry, and the operating environment; analytical and experimental research to evaluate the effects of thermal annealing on recovery of fracture toughness and on reembrittlement rates; and validation of the models using materials removed from decommissioned reactors. The overriding objective is to develop a quantitative method for predicting the effects of neutron irradiation on material properties for reactor vessel applications.

A necessary input to establishing the degree of neutron embrittlement of the reactor pressure vessel is the neutron fluence, both the peak value and the distribution of fluence in the beltline region. The neutron fluence, which cannot be measured directly, is predicted using the neutron fluence determined at surveillance locations combined with neutron transport calculations. The neutron fluence at surveillance locations is determined by a process of: (1) dosimetry measurements, (2) transport calculations to compute fluence for the capsule location, and (3) a consolidation of the measurements and calculations to reduce uncertainties of the predictions.

Dosimetry research: This has led to an improved set of differential cross-sections of nuclides which have been included in the Evaluated Data Files developed by the National Nuclear Data Center. Use of the new Data Files is expected to significantly improve the ability to predict neutron fluences in reactor vessels. Work has been initiated to process these files into structured libraries that can be applied to determining reactor vessel fluences. Also, research continues to improve the methodology and data bases for obtaining the fluences and fluence-rates at critical locations of reactor vessels. Sources of uncertainties are being identified and sensitivity of the uncertainties to fluence values are being established. New approaches to fluence calculation such as the use of ex-vessel surveillance dosimetry, are being validated. This research has led to the development of a regulatory guide on dosimetry which is scheduled for public comment during 1993.

Test reactor irradiations have been used for many years to examine the effects of irradiation on material properties. These well controlled irradiation experiments offer opportunities to examine the effects of key variables, as well as any interactions among them. The NRC research program has completed seven irradiation series, examining the effects of irradiation on fracture toughness of base metals, welds, and stainless steel cladding. Three additional irradiation series are planned. The 10th series, currently underway, examines the effects of irradiation on the ductile tearing resistance of high-copper welds fabricated using Linde 80 flux. These welds were removed from
the canceled Midland, Unit 1, reactor vessel, and represent production welds. As such, it is anticipated that the degree of scatter in the results will be quite high. This complicates interpretation of the results, but also will give an indication of the variability that can be expected in actual pressure vessels. The 8th irradiation series will examine a high-copper weld fabricated with Linde 80 flux, but fabricated to minimize the variability in chemistry. This irradiation series will offer the opportunity to examine changes in fracture toughness in a setting where the variability due to other factors, such as copper content, is minimal. Finally, the 9th irradiation series will use the same material as the 8th series, but will investigate the effects of thermal annealing on property recovery and reembrittlement. These three irradiation series will provide an experimental basis for determining material properties for welds with low ductile tearing resistance.

The test reactor results from the 8th irradiation series will be compared with surveillance program results from operating commercial power reactors that have included the Midland material in supplemental "research" capsules. This will provide an excellent opportunity to compare the differences between test reactor irradiations and power reactor surveillance results.

The number of variables that could have a significant influence on embrittlement is large and interrelated, and an empirical approach cannot completely resolve the issue. Therefore increasing emphasis has been given to study the underlying mechanisms of neutron radiation and the resulting embrittlement. There has been significant progress through the use of high resolution instruments, such as the atom probe field ion microscope (APFIM) and small angle neutron scattering (SANS). This progress has improved confidence in interpreting the test reactor results and in defining additional irradiation programs. The International Group on Radiation Damage Mechanisms (IGRDM), formed to promote cooperation on addressing these issues, has provided discussion and interaction on this subject, furthering the efforts to identify the underlying mechanisms controlling neutron irradiation embrittlement.

Annealing and embrittlement-mechanisms research: This is a relatively modest effort at NRC as compared to work done in other countries. Still, we have made valuable contributions in understanding neutron flux effects. The continuing program is emphasizing experimental evaluation of the separate and combined effects of major variables on irradiation embrittlement, microstructural and micro-mechanical characterization of irradiated pressure vessel steels, and post-irradiation annealing studies. In the theoretical and modeling areas the program is directed towards studying fracture mechanisms and mechanics, microstructure-property relations and microstructural evolution. Thermodynamics, kinematic, and microstructural models are being developed to address interactive effects on embrittlement and post-irradiation annealing. Irradiation studies include the interactions of flux, temperature, fluence and microstructure over a range of copper and nickel compositions. This research has shown that the dominant irradiation embrittlement mechanisms of RPV steels are the accelerated formation of extremely small (1 to 2 nanometer) copper-rich precipitation in steel microstructure combined with the formation of defect clusters in the matrix of the steel microstructure.

An effort to develop improved correlations of irradiation embrittlement data was initiated in 1992. This will use all embrittlement data, from U.S. and a number of other countries, to develop correlations for shift in transition temperature, and upper-shelf Charpy energy decrease as functions of controlling parameters, such as material chemistry, neutron flux and fluence, and irradiation temperature. The data from other countries will provide a broader range of chemistry and irradiation conditions than can be obtained using only the U.S. surveillance data and U.S. materials. Results from this program will be evaluated to determine if revisions to the procedures in the NRC's Regulatory Guide 1.99 on embrittlement of reactor vessel materials are necessary.
The embrittlement program has provided initial data to demonstrate the effectiveness of thermal annealing in recovering degradation in mechanical properties due to irradiation damage. The results of the annealing work are being supported by recently initiated industry efforts, and by results from research performed in Russia and exchanged under the auspices of the Joint Coordinating Committee on Civilian Nuclear Reactor Safety (JCCNRS). The combined results of these efforts provide reasonable assurance that thermal annealing is a practical method for mitigating the effects of neutron irradiation damage. Additional work is in progress to improve our ability to predict annealing recovery and re-embrittlement rates, based specifically on the 9th irradiation series discussed earlier. The NRC currently is preparing regulatory guidance on thermal annealing procedures. As part of that process some further work may be identified, but no new initiatives are planned at present.

As with the fracture mechanics research, validation is the foundation of the embrittlement research program. The NRC was involved with Germany in examining the materials removed from the Gundremmingen reactor, and we worked with the Electric Power Research Institute in a vain attempt to obtain materials from the Shippingport reactor pressure vessel. We currently are actively working with the U.S. industry to determine if materials from the Yankee Rowe pressure vessel can be obtained, and we are working with the Japan Atomic Energy Research Institute to obtain materials from the pressure vessel of the Japan Power Demonstration Reactor. We believe materials subjected to long-term service irradiation offer a unique opportunity to test our models in a realistic setting. These materials also offer opportunity for the mechanists to evaluate their physical models that cannot be duplicated in the laboratory.

Material Property Research: Most of the material property research is related directly to the embrittlement research. However, there are efforts underway to support revisions to test standards that will provide a consolidated test method for fracture toughness testing, and efforts are underway to evaluate the size criteria that are embodied in the American Society for Testing and Materials' Standard Test Method E-399, "Standard Test Method for Plane-Strain Fracture Toughness of Metallic Materials." If these size criteria can be relaxed then more of the available fracture toughness test data could be used in determining a crack initiation fracture toughness curve for pressure vessel steels. A more complete data base would permit statistical analysis of the data, and could lead to a statistically based reference curve rather than the lower-bound curve currently employed.

The material property research also is examining the ductile tearing resistance of A-302B plate material. Recall from the discussion of the low Charpy upper-shelf energy regulatory issue that recent test data had shown a very low tearing resistance for this class of plate material. The research currently underway is being conducted cooperatively with the nuclear power industry to determine if the low toughness results are representative of the actual vessel materials, or if they are unique to the plate material used in the research program, which had received special thermo-mechanical processing to produce a low Charpy upper-shelf energy. Early unpublished results suggest that the result is representative.

Future plans for the material property research effort include further evaluation of the material properties for stainless steel cladding, and the variability in those properties, and the evaluation of thermal embrittlement effects for both cladding and base metal.

2.0 PRIMARY PIPING ISSUES

Typical nuclear power plants contain a considerable length of primary piping. Service experience for this vast amount of piping has been remarkably good. However, that experience has identified aspects of the design criteria that warrant review; it has identified degradation mechanisms that, if left undetected, can lead to leaks or breaks in the piping; it has highlighted the
need for periodic inspection of the piping; and it has highlighted the need for methods to evaluate the significance of flaws found during the inspections. The NRC's piping research addresses each of these areas, although the inspection program is included as part of the larger non-destructive examination program, and is discussed separately below.

The ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Division I provides rules for the design of piping systems in nuclear power plants. In general, the design rules have been proven over the years to result in a design that affords reasonably certain protection of life and property; and provides a margin for deterioration in service so as to give a reasonably long safe period of usefulness. However, because of lack of adequate supporting data, conservatism was introduced into the initial design calculations and regulatory acceptance criteria. These requirements led to use of a large number of snubbers (with high failure rates) and other supports resulting in stiff piping systems. These stiff systems restricted thermal movement during the plant operations. Also, because of the large number of supports and requirements to provide protection against postulated breaks, plants have become congested making access difficult for inspection and maintenance. It has been contended that stiff piping systems may have negative impact on the overall plant safety as it leads to undesirable performance curing plant operations while the costs associated with the seismic designs are significant and have been rising over the period.

Because of the above concerns the NRC, industry groups and the professional societies undertook in-depth examinations of seismic requirements for piping during early eighties. The NRC formed a Piping Review Committee (PRC) in 1983 which published several recommendations in 1985. The Pressure Vessel Research Committee (PVRC) on piping systems formed various task groups starting in 1982. Based on recommendations of these two committees and other initiatives, extensive research activities were conducted in 1980's by both the NRC and industry. Several test programs were conducted in US and other countries. The EPRI/NRC sponsored Piping and Fitting Dynamic Reliability Program was one of the most significant effort leading to a better understanding of the behavior and failure of piping systems when subjected to a seismic excitation. Other research efforts included collection and evaluation of earthquake experience data, and development and benchmarking of analytical techniques.

As a result of the PRC recommendations and some of the research findings, NRC made some significant changes to the piping design criteria in 1980's. These changes dealt with both the seismic and pipe rupture criteria. Specifically, they included: (1) use of leak-before-break for some piping systems to eliminate pipe whip restraints, jet impingement shields and large bore hydraulic snubbers (environmental qualification requirements were unchanged); (2) revision of Standard Review Plan, Section 3.6.2, to eliminate dynamic and environmental effects of arbitrary intermediate pipe ruptures; (3) revision of Standard Review Plan, Section 3.6.1, to eliminate the jet impingement effects of the arbitrary one square foot break in the break exclusion zone; (4) conditional approval of ASME Code Case N-411 to permit use of higher damping values; (5) approval to use peak shifting rather than peak broadening of floor spectra; (6) approval to use independent support motion; and (7) use of higher stress limits for ensuring piping functional capability.

Recent developments in nuclear industry have necessitated changes to the design philosophy of piping systems. The NRC staff recognized the need for specific guidance to ensure that a consistent safety margin is maintained in view of the evolving design philosophy. For example, one major change to the design philosophy is the proposed life of a nuclear power plant being extended from the current 40 years to 60 years. The effects of plant aging and cyclic fatigue considerations become a more significant factor in the design life of nuclear piping systems as the period of usefulness of nuclear power plants is
increased. In addition to these changes, the NRC staff is currently revising Appendix A to 10 CFR Part 100, Seismic and Geologic Siting Criteria for Nuclear Power Plants. In the proposed revision, issued for public comments on October 20, 1992, the NRC staff has allowed the elimination of design requirements associated with the Operating Basis Earthquake removing a major source of controversy and unintentional conservatism in some cases. These changes, to a great extent, should lead to a more flexible piping systems.

However, there are several other initiatives proposed by the industry to reconcile the overall piping design requirements with clearer understanding of the likely failure modes of piping systems under earthquake loadings. There is a need to evaluate the many design criteria changes, and their overall impact on piping design and on piping integrity.

Service experience for piping has identified several degradation mechanisms that can lead to leaks or ruptures of the piping. The primary degradation mechanisms include erosion/corrosion in single-phase systems, intergranular stress corrosion in BWR's, and thermal and mechanical fatigue. The single-phase erosion/corrosion mechanism is well understood, and there are computer programs available to predict systems and locations within those systems that are most susceptible. Further, there are inspection techniques that are effective in identifying this damage form.

Intergranular stress corrosion cracking (IGSCC) in BWR piping systems is a well known issue. Research conducted by the NRC, EPRI, and many other organizations in U.S. and international community has identified the mechanisms, and produced mitigative measures. The mitigative measures that have evolved from the research program include water chemistry control, change of material, altering the residual stress distribution through welding procedures for repairs or replacements or through mechanical devices for existing piping, and the application of weld overlay repairs for welds that are cracked. The major remaining issues related to IGSCC are the predicted crack growth rates, the procedures for evaluating cracks detected in service, and the reliability and accuracy of the non-destructive examination methods and inspection teams.

Fatigue, whether it stems from vibration or from thermal cycling, is a major concern for piping. The concern is exacerbated by the deleterious effects of the water coolant, which can hasten fatigue crack initiation and which increases the rate at which fatigue cracks propagate under service loadings. Fatigue can lead to piping failures -- leaks or ruptures -- during the design life of the piping. Because it is a progressive damage mechanism, and more operation means more fatigue damage, fatigue is also a major consideration for plant license renewal evaluations. The major needs in this area are for data to permit evaluation of the effects of water environment at operating temperatures on fatigue life for typical piping materials, and for data on the effect of the water environment on fatigue crack growth rates in these materials.

IGSCC leads to cracking, which if left undetected can lead to failure of the piping systems. However, improvements in inservice inspection techniques have made it more likely that cracking will be detected before it reaches a size that must be repaired before resuming operation. This has led to increased requirements for flaw evaluation methods, not just for stainless steels but also for carbon steels. The ASME Code, Section XI, committee has implemented flaw evaluation procedures for exactly this purpose. However, there are aspects of those procedures that warrant further examination to make certain that the margin of safety is at least as much as anticipated. Pipe fracture experiments conducted thus far support the Code analyses. However, data from a broader range of materials, crack sizes, and loading rates is needed to provide final validation for the analysis methods. Also, material properties, particularly fracture toughness, used in these analyses were based on a limited data base. Work is needed to further examine the fracture toughness
of piping materials and weldments, under a variety of loading conditions, to make certain that the generic properties used by the Code are appropriate.

It was noted under the discussion of piping issues that NRC had accepted leak-before-break for piping that meet rigorous acceptance criteria. NRC has developed a methodology that is used to evaluate piping systems to determine if a leak-before-break relief can be applied. The methodology is similar to the one used by the ASME Code, with differences in leak detection sensitivity and in the amount of flaw growth allowed. Still, the need for experimental validation of the analysis methods and the need for a more comprehensive database of material properties is similar to the needs for validation of the Code's flaw evaluation procedures.

Material properties, especially fracture toughness, play a significant part in both the Code's flaw evaluation procedures and the leak-before-break evaluations. It is important that initial properties are known, as well as changes in those properties that occur during operation. Degradation of fracture toughness due to thermal aging of cast stainless steels is a well known example of a property change that occurs during service. Research conducted for the NRC and in other laboratories in several countries has led to a reasonably complete understanding of this phenomenon, and to empirical correlations that permit reasonably accurate estimates of fracture toughness. Work is continuing in the U.S. to evaluate the implications of this toughness loss to structural integrity and on inspection requirements.

If changes to design criteria are accepted, and if they can be applied to existing piping systems, then changes to existing piping systems could be implemented that could bring about cost savings primarily in terms of maintenance activities. For example, snubber reduction could be justified which would reduce maintenance and inspection costs. Also, if some pipe supports could be removed then congestion inside containment could be reduced, making maintenance activities easier, and facilitating inspection of piping systems.

Fatigue could be a life-limiting consideration for some piping and piping components, and the potential for environmental effects to exacerbate the fatigue issue could further limit the life of some fatigue critical components. Methods for evaluating the useful remaining life in piping systems are needed. These methods must be applicable to piping designed according to the older B31.1 standard as well as to the more recent ASME Code Section III criteria. Further, the role to be played by non-destructive examination must be defined, and performance requirements for both the inspection equipment and personnel must be defined.

Finally, there can be an interaction among several of these issues. For example, there is significant interest in revising the piping design criteria in such a way that the stresses imposed by earthquakes would be significantly greater than currently allowed. While there remains some debate over this proposal, there has been very little discussion of the impact of higher stresses on the flaw tolerance of piping systems. Current design stress limits result in piping systems that can tolerate very large flaws without posing a significant threat of rupture, even under accident loadings. However, as the stress limits are increased, the flaw size that could survive an accident, such as an earthquake, is reduced. If the stresses are allowed to increase sufficiently high, the concept of an allowable flaw size cannot be supported. It is important to explore the interactions between and among the various issues to make certain that changes made for one purpose do not have an adverse impact on overall piping integrity considerations.

2.1 PIPING RESEARCH

The NRC's primary piping research program has been reduced over the last few years, and we anticipate closing this research area as a major activity over the next 4-5 years. The research program emphasizes examining and potentially implementing the piping design criteria changes, evaluating the effects of the
water coolant and operating temperatures on fatigue and fatigue crack growth rates which falls under the general category of environmentally assisted cracking, and validating the pipe fracture analysis methods and materials property data bases. Each of these areas is discussed below.

Piping Design Research. The Advanced Reactor Corporation, a utility group involved in the detail design of Advanced Light Water Reactors (ALWRs), formed a Technical Core Group (TCG) in mid-1992 which will develop industry positions on the piping design. The industry position is anticipated to be submitted to the staff in 1993. The ASME Section III Subgroup on Design has also formed a special task group to provide an integrated plan for future changes to the design sections of the nuclear piping code in 1993. The NRC is planning to initiate a research program in 1993 which will review all of the proposed changes and changes which have occurred during recent years in piping design with a view to examining the cumulative impact on safety margins of these changes which are being proposed on their individual merits. The research program will focus on answering the following types of questions: (1) significant changes leading to reduction in the piping margins have been implemented, are additional changes advisable?; (2) in addition to the pressure boundary, how do these changes affect functional capability and structural integrity?; and (3) how will the proposed changes affect the attachments such as pumps and valves? One crucial question is how do these changes affect the observed failure modes such as failure of pipe due to excessive displacement of attached equipment. The NRC program will also consider the effects of potential degradations, tolerances, and requirements of ASME Section XI in reviewing the proposed changes. The ultimate aim of the program is to develop the staff integrated acceptance criteria for seismic design of the piping systems, which reflects its true behavior, while retaining adequate safety margins.

Environmentally Assisted Cracking Research. Current fatigue design for reactor structural components is based on the ASME Code Section III and its fatigue design curves. The design curves, which were developed more than 20 years ago, were obtained by adding a correction factor to the mean-data curve obtained from room-temperature tests on smooth specimens in air. The correction factor was intended to account for the differences between structural components and the test specimens and was intended to account for a variety of factors including the effect of surface finish, size, and data scatter.

Based on results obtained in earlier work in the US and Japan as well as on the results obtained in the ongoing NRC research program, it is now clear that the Code curves can significantly overestimate fatigue lives under some reactor loading and environmental conditions. Since no consensus design procedure is available, data from ongoing tests and from the literature and programs in Europe and Japan were evaluated to develop interim design curves that more adequately describe fatigue life in the high temperature aqueous environments characteristic of light water reactors.

The current data on the fatigue crack growth in pressure vessel and piping materials have been obtained almost solely in tests where the ferritic materials have been completely exposed to the simulated reactor coolant environment. In reality, these materials are clad with austenitic stainless steels and only a very small portion of the material is exposed to the reactor coolant. Tests show that crack growth rates are higher in the clad materials and cracking is easier to initiate and sustain in cladding, but the differences are relatively small. It appears that in most cases, existing predictive methods give an adequate description of crack growth rates; however, some additional testing is being performed to confirm this observation.

Pipe Fracture and Material Property Research. The NRC's pipe fracture research involves both analytical and experimental programs to provide a comprehensive, and fully validated fracture analysis capability that can be
applied to both flaw evaluations and leak-before-break analyses. The work started in early 1980’s and has continued at several laboratories since that time. The early work examined fracture behavior of circumferentially cracked small diameter (4-6 inch) stainless steel piping subjected to slowly increasing loads. The experimental work moved forward to examine fracture behavior of carbon steel pipes and the welds in both carbon and stainless steels, and larger diameter piping, finally leading to an internationally funded research program -- the First International Piping Integrity Research Group or IPIRG-I program -- that examined fracture behavior of degraded pipes under seismic loading.

The analytical work started with simple limit-load solutions for circumferentially oriented, through-wall cracks. The analytical methods have advanced to the point where state-of-the-art methods involve elastic-plastic finite element analyses, that can account for cyclic loading effects representative of seismic events. Predictions of the pipe fracture experiments have shown that the analysis methods are generally conservative.

The NRC's pipe fracture research is continuing with an analytical and experimental program to evaluate the effects of relatively short circumferential cracks in a variety of materials, emphasizing larger diameter piping, quasi-static and seismic loading rates. Additionally, the NRC has organized the Second International Piping Integrity Research Group (IPIRG-2) which is considering relatively short circumferentially oriented cracks in a representative piping system that is subjected to simulated seismic loading. The IPIRG-2 program also is examining the fracture of pipe fittings with cracks.

These two programs, which will be completed by the end of 1995, are expected to provide the final validation for the pipe fracture methods. However, further work is needed to provide comprehensive data bases for representative piping material properties. Materials are being sought from decommissioned nuclear power plants so that typical properties can be further documented. This activity is coupled with the pressure vessel integrity research effort related to decommissioned reactors. To the extent that materials from decommissioned reactors can be obtained, piping materials will be sought as well as materials from the pressure vessel. The materials of interest are welds in both carbon and stainless steels, and service aged cast stainless steels.

Finally, the pipe fracture research effort is interacting with the piping design effort to evaluate the potential impact of piping design criteria changes on the flaw tolerance of typical LWR piping systems. The activity is primarily analytical, but some experimental verification may be necessary.

Erosion-Corrosion of Piping -- As a result of the Surry, Unit 2, pipe failure in December 1986, U.S. NRC performed a review of the available data and mechanistic understanding of erosion-corrosion of ferritic material as well as its major occurrences in nuclear power plants. Included in this study was a failure analysis of the tee-elbow joint that failed in the Surry plant’s main feed-water piping. This effort contributed to the development of inspection and mitigation programs by the U.S. nuclear power plant industry, EPRI, and the NRC to control the issue of erosion-corrosion. Computer programs developed by EPRI are being used by the nuclear industry and are being evaluated for possible inclusion in the ASME Code, Section-XI.

3.0 STEAM GENERATOR TUBE INTEGRITY ISSUES

Steam generator tube integrity has been a concern for the NRC for several years. The steam generator tube integrity research project identified a need for advanced tube inspection capabilities. It also highlighted the complexity of predicting tube integrity and the need for reliable inspection techniques.

Service experience has identified a number of damage mechanisms affecting tube integrity: denting, wastage, pitting, stress corrosion cracking, fatigue, and intergranular attack. Water chemistry control, or lack of control, led to most of these issue. However, in some cases attempts to
modify the secondary water chemistry to resolve one issue simply led to other issues. Since the damage mechanisms are time dependent, the inspection techniques must be capable of evaluating tubes for a variety of damage types, which can lead to the need for specialized techniques to detect and quantify particular damage types.

The major issue for steam generator tube integrity is to provide highly reliable inspection systems -- equipment, personnel, and analysis methods -- that can detect and quantify degradation mechanisms before tube integrity is degraded to the point where leaks or ruptures could occur under either normal operation or accident conditions. There also is a need for validated methods for predicting tube failure and the resulting leak rates under normal and accident conditions. Finally, there is a need for data on the rates of damage progression -- stress corrosion crack growth rates and intergranular attack rates -- that can be used in tube integrity analyses that can be used to supplement the inspections, to help establish tube inspection intervals, and to provide a quantitative assessment of steam generator tube integrity.

3.1 STEAM GENERATOR TUBE INTEGRITY RESEARCH

The emphasis of NRC research on steam generator tube integrity has been to develop generic guidance for performance demonstration qualification of eddy current inspection systems, along with recommendations for enhanced inspection methods and sampling plans. This guidance was included in a draft revision for staff review of Regulatory Guide 1.83 which covers in-service inspection of steam generator tubing. Results of the Steam Generator Tube Integrity Research Program indicated a need to improve the reliability of steam generator tube inspections. Thus, work was initiated to develop performance demonstration requirements to ensure that eddy current inspection systems (i.e., personnel, equipment, and procedures) possess an adequate capability to identify the known forms of tube degradation that occur in steam generators. Research has been underway for several years on improved eddy current test equipment. It has resulted in a design for a 16 reflection coil array probe capable of mixing multiple signal frequencies to identify different forms of degradation, even in the presence of tube supports, copper deposits, etc. The new probe and evaluation equipment will be field tested later in 1993.

The NRC has also been developing additional information on the capabilities of eddy current inspection systems to detect and size crack-type flaws through participation in the International Program for Inspection of Steel Components (PISC). Crack-type flaws are of great interest to the NRC because they are the most frequent cause of steam generator tube failure, they are the most difficult flaw type to reliably detect and size, and they significantly decrease tube integrity. The PISC program is conducting an evaluation of the effectiveness of steam generator tube inspection techniques in which seven USA teams are participating. During 1992 several tube mockups were circulated in the US. Analysis of the data from this study will be performed in 1993 with the conclusion of the program scheduled for 1994.

A steam generator tube mockup for on-site evaluations of the quality of eddy current in-service inspections has been designed, and the structural elements were fabricated. Two of the flaw types (wastage and fatigue cracks) to be included in the mockup were also fabricated. Future research efforts will be directed to evaluation of tube burst models, leak rate models, and crack growth rates. Particular emphasis will be placed on developing and validating the technology needed to perform realistic structural integrity assessments for steam generator tubes.

4.0 RELIABILITY OF FLAW DETECTION & SIZING ISSUES

It is important that the flaws can be detected and sized reliably, so that proper mitigating actions can be taken to avoid sudden failure of the components. Non-destructive examination is a means of assuring the structural integrity of primary pressure boundary components. The needs for more
rigorous inspections for reactor pressure vessels were discussed earlier. The
overriding issues for non-destructive examination, whether for pressure
vessels or piping, relate to quantitative descriptions of the probability of
detecting a flaw as a function of flaw size, the accuracy with which flaw size
can be determined, and the qualification of inspection teams, commonly termed
performance demonstration. Beyond developing the technology needed for
improved examination performance, codification of the requirements on equip-
ment, techniques, and performance demonstration is needed.

4.1 NON-DESTRUCTIVE EXAMINATION RESEARCH

The NRC's non-destructive examination research program includes studies of
improved methods for the reliable detection and accurate sizing of flaws in
carbon steel, and in wrought and cast stainless steels for in-service inspec-
tion of piping and pressure vessels. It also includes studies of on-line
continuous monitoring techniques, using acoustic emission, for crack growth
and leak detection.

Improving the Detection and Sizing of Flaws. An improved method for
reliable detection and accurate sizing of flaws in LWR primary circuit
components is the Synthetic Aperture Focusing Technique for Ultrasonic Testing
(SAFT-UT). The SAFT-UT technology is based on physical principles of ultra-
sonic wave propagation and uses computers to process the data to produce high-
resolution, three-dimensional images of flaws to aid the inspector in locating
and sizing them. The SAFT-UT system has been used to inspect a reactor
pressure vessel, as part of the international Program for Inspection of Steel
Components (PISC) assessing the effectiveness of advanced ultrasonic technolo-
gies. A final report is being prepared showing the results of this inspec-
tion. The SAFT-UT technology has been transferred to General Electric for
integration into their next generation reactor vessel inspection system.

Inservice Inspection System Qualification. Field experience and research
including both national and international studies over the last decades have
shown that in-service inspection (ISI) was not sufficiently effective or
reliable. The NRC research indicated a need for qualification of the entire
ISI process, including personnel, equipment, and procedures. Therefore,
during 1981 the NRC staff and its contractor, Pacific Northwest Laboratory,
began development of recommended criteria and processes for a formal qualifi-
cation program. After several modifications, and meetings with the industry,
a document was finalized which provided recommendations and criteria for a
qualification program including performance demonstrations. During 1984, the
NRC staff and its contractor began working with the ASME code for approval.

Since then the ASME Code, Section XI, approved Mandatory Appendix VII on
Personnel Training and Qualification (in 1989) and Appendix VIII on Perfor-
mance Demonstration of Ultrasonic Testing (UT) ISI Systems (in 1990). The NRC
and its contractor have conducted a detailed review and evaluation of these
two appendices. During 1992, proposed revisions to upgrade them were
provided to the cognizant ASME Code groups. Further, we have been working
with the industry to review and evaluate their plans to implement these
appendices. Close coordination is maintained with the industry's Performance
Demonstration Initiative (PDI) group through the Nuclear Utility Management
and Resources Council (NUMARC) to monitor progress and critique plans.

Advanced Ultrasonic Imaging Systems. Additional Code requirements were
prepared and submitted to the ASME Section V Subcommittee to fulfill a need
for Code rules to cover the computerized UT imaging systems that are being
used by the NDE/ISI industry for examining nuclear power plant components.
These proposed Code rules, which include requirements for the proper use of
SAFT-UT, were approved for publication in the 1992 Addenda to the ASME Section
V Code on Nondestructive Examination.

Risk-Based Inspection. Improved criteria for in-service inspection planning
(what, where, when, and by what method) are being developed using risk-based
approaches. Plant-specific pilot studies featuring the Surry-1 Nuclear Power Station have shown the risk-based approach to be workable. As a result, ASME Section XI has formed a Task Group for implementation of risk-based methods into Code rules for in-service inspection of nuclear power plant components. Expected benefits include reduced inspections where such inspections are not justified by safety concerns and occupational exposure, and a redirection of inspection efforts to other components with greater safety significance.

**Equipment Interaction Matrix.** Ultrasonic inspection reliability of nuclear components is known to vary with human factors, equipment characteristics, procedures, etc. NRC research has shown that changing the equipment parameters in an ultrasonic inspection can greatly affect the results of the inspection. The ASME Code provides tolerance levels for equipment parameters. Results of the NRC research have been used by the ASME Code, Section XI, to develop and update rules for acceptable tolerance levels.

**Surface Roughness Evaluation.** Currently, there are no ASME Code requirements dealing with acceptable surface conditions for ultrasonic inspections. The NRC and EPRI cooperative research has shown how ultrasonic inspections can be affected by surface conditions, using computer models, validated by experiments, for propagation of ultrasound. The validated models will be used to develop recommendations to the ASME Code on surface roughness limits to allow reliable ultrasonic inspection.

**Coarse Grained Materials.** Improved methods are being evaluated for the reliable and effective inspection of centrifugally cast stainless steel components of pressurized water nuclear power reactors. Cracks in these components that are greater than a certain size must be detected by an effective inspection system. Two concepts are being investigated: (a) low frequency ultrasonics for greater sound penetration and less interference from grain boundaries, and (b) adaptive ultrasonics to compensate for changes in the internal structure of these coarse grained materials.

**Continuous Monitoring for Crack Growth and Leak Detection.** The NRC funded research has produced technology in support of continuous monitoring of acoustic emission (AE) to detect initiation and extension of flaws in components as they might occur during reactor operation. The technology and application methods have been validated off-reactor in several major tests. An ASTM Standard E 1139 and an ASME Code Case N-471 have been generated and approved for use to guide and regulate application. The research program and results have been summarized in NUREG/CR-5645 report. The final step in this effort has been to validate the AE technology on an operating reactor by monitoring a weld flaw indication at Philadelphia Electric Company's Limerick, Unit 1, reactor. Continuous AE monitoring at Limerick demonstrated that it can be effectively applied to an operating reactor power-plant.

Similar AE technology also provides a very sensitive coolant leak detection capability. This aspect was developed under the NRC sponsorship, with the results presented in NUREG/CR-5134, "Application of Acoustic Leak Detection Technology for the Detection and Location of Leaks in Light Water Reactors". Guidance on the use of AE for leak detection is being incorporated in a revision of Regulatory Guide 1.45 on leakage detection systems for reactor coolant pressure boundary.

Benefits from this work include increased safety by detection and evaluation of crack growth as it occurs, improved detection and location of coolant leaks as they initiate, and reduced personnel exposure to radiation by reduced need for manual inspection of reactor components.

**International Reliability Studies.** The NRC has been an active participant in the international Program for Inspection of Steel Components (PISC) which is assessing the effectiveness of technologies and procedures for in-service inspection of nuclear power plant components. The output from this program will aid regulators and Code bodies in establishing technical bases for
improving ISI requirements. The NRC has been in a proactive leadership role in developing the PISC program objectives, and has funded the Pacific Northwest Laboratory (PNL) to support the design of studies, fabricate flawed specimens, implement testing, and analyze comprehensive data bases. Specific PISC studies include addressing the influence of human factors on inspection reliability; pressure vessel inspection capability using SAFT-UT; inspection of stainless steel piping, nozzles, and dissimilar metal welds; and inspection of steam generator tubing mockups. These experimental studies are nearing completion, and important results are becoming available. The experimental portion of the PISC program is scheduled for completion at the end of 1993, and the results and analyses of data from all the studies will be released to participants over the next year or so. As these results become available, they will be used to develop and support upgraded inspection requirements.

Specific activities performed during 1992 included developing work scopes for contracts to fund UT/ISI teams from USA to participate in Wrought Stainless Steel Reliability tests. This reliability study, which includes cracked pipes removed from service, will be conducted in PNL facilities to accommodate the special needs of this work. All seven U.S. teams completed their inspections of the wrought stainless steel capability specimens, and these six large pipe specimens were returned to Europe for further inspections and/or destructive evaluations. Inspection schedules and preliminary logistics planning have been completed for the PISC-III capability studies for Cast/Wrought Stainless Steel specimens and Cast/Cast Stainless Steel specimens. When available, it is expected that the PISC-III results will provide significant impetus to the NRC efforts to upgrade selected ASME Code requirements.

Support to NRC Regulators. The research program is helping NRC staff meet their responsibilities by assisting in training the staff in the developing technologies in inservice inspection. During the past year, effort focused on developing procedures and test blocks for the detailed review and evaluation of computer-based ultrasonic inspection systems. A computer-based ultrasonic inspection system was rented and subjected to a rigorous review and evaluation, including a hands-on seminar for the NRC staff. Finally, the results of the review and evaluation were presented to the NRC Technical Advisory Group on Nondestructive Examination.

CONCLUDING REMARKS
This paper has provided an overview of the key structural integrity issues facing the nuclear industry today. These are issues for the owners of nuclear power plants and for regulatory authorities alike. Several examples have been cited where research conducted in the international community has led to the solution of difficult issues. Most of the difficult issues are common to many nuclear power plants. Cooperative research efforts, both formal and informal, provide an effective and an efficient international method for addressing these complex issues.

The paper has also stressed interaction between the regulatory issues and the research efforts. This interaction is vital since a strong research program is the foundation for an effective regulatory program. It presents an opportunity to resolve technical issues in an effective manner. We all are working to build upon that foundation to help provide a safe and reliable source of energy which will meet our present and future needs.

ACKNOWLEDGEMENT
This paper contains information about work carried out or sponsored by NRC, an Agency of the U.S. Government. Acknowledgement is made for the input provided by various staff at NRC. The paper was assembled by Dr. Shah N. Malik.