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Description of Current and Planned Research in Structural Engineering

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G. Bagchi

Division of Reactor Safety Research
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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ABSTRACT

The Nuclear Regulatory Commission's (NRC) research needs in the safety and reliability of nuclear power plant structures are addressed by research programs under the Structural Engineering Research Branch (SRB) which represents one element of General Reactor Safety Research (GRSR) with the Division of Reactor Safety Research (RSR). The objectives, background, research needs, project description and project results for each program element are discussed. The base year for this program is 1980. In some cases, research needs may change and projects will be modified. The projects described reflect current assumptions about long-term funding levels. Should those assumptions not be realized, project scopes will have to be modified.

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DESCRIPTION OF CURRENT AND PLANNED RESEARCH
IN STRUCTURAL ENGINEERING

1. INTRODUCTION

The structural engineering research program plan is an analytical and experimental program designed to provide information necessary for the safety analyses of structures at nuclear power plants under anticipated operational, upset and postulated accident conditions. This information must include response characteristics, definition of failure modes, and failure probabilities as well as limits of deformation necessary to ensure functionality of active and passive components related to safety. The primary objective of the program is to provide the NRC licensing staff with methods, techniques and criteria for evaluating structural adequacy in terms of margins of safety and failure probabilities for structures at nuclear power plants when subjected to normal, upset and accident conditions. In addition, assurance is required of the engineering behavior of safety related systems and components to meet licensing requirements.

In the past several years, the engineering research needs related to structural behavior were addressed as a part of other research programs in water reactor safety research or as a limited part of technical assistance work. With a few exceptions, the problems in structural design, behavior and testing were not formally addressed in sufficient depth. Increased attention, however, has been drawn to the need to resolve the large number of engineering problems reported in operating plants. These problems are principally in the areas of mechanical, structural and materials engineering. To respond to these more pronounced needs under the discipline of structural engineering, the Structural Engineering Research Branch was recently established as a part of the new Office of the Assistant Director for General Reactor Safety Research. This branch will address the increased research needs to quantify margins of safety in structural design. Research will also be undertaken to better understand problems related to plant behavior during and after earthquakes and other events, and to major issues related to design and construction requirements. In addition, attention will also be directed to improving plant safety. This aspect is responsive to the congressional request to the NRC to improve the safety of LWR nuclear power plants.

The structural engineering safety research program will concentrate on the analysis, design and construction of structures related to the plant safety. The program has its major current emphasis on margins of safety in seismic design and is integrated with the mechanical engineering research program plan in this area to avoid duplication. The other programs are related to events in addition to the seismic event; they support the licensing information needs for assuring plant safety in response to plant internal as well as external forces.

An essential part of the research planning and management is to maintain cognizance of related research being conducted elsewhere. Consequently, liaison with foreign and domestic organizations, including government agencies, will continue in order to keep apprised to their research. Most important, of course, is our interaction with the offices of NRC which we serve in order to have a firm understanding of their needs.

The structural design process proceeds as follows:

(a) define and quantify the loads a structure must sustain;

(b) predict the effects these loads will have on the structure; and

(c) check to assure that performance under the load will be satisfactory.

Structural engineering research is organized into three major programs that roughly parallel the structural design process. These major programs, which are described in the sequel, are concerned with the choice of design loads, prediction of structural response and structural performance.

The structural engineering research program is designed to supply the information and tools necessary for evaluation and confirmation of safety in the safety-related structures at nuclear facilities. This research includes providing independent safety data derived by analytical methods, testing, experimentation and the correlation and interpretation of information from other sources. The programs are designed to understand mechanistic processes which lead to failure and to provide better quantified estimates of design margins for safety of existing and proposed nuclear facilities.

In addition to the three major programs, short-term projects are undertaken to satisfy immediate needs for information to support licensing activities. These projects are limited in scope and, in total for any fiscal year, consume less than ten percent of branch resources. Characteristically, a project is undertaken because of a perception on the part of the user offices that this approach would be more effective, than for them to proceed with a technical assistance contract. Expertise of Structural Engineering Research Branch members in the subject is typically the determining factor.

Recently, we have undertaken three projects. One, in support of Task Action Plan (TAP) A-1, Water Hammer Effects, required predictions of the forces associated with water hammer events. This work was performed by EG&G, Idaho. Another, in support of TAP A-38, Tornado Missile Events, involved developing force-time histories consistent with the hypothesized automobile missile. This work is being performed by Chiapetta, Welch and Associates. The last has as its objective, making available, for licensing staff use, computer codes that could be used for independent verification of submissions on the soil-structure interaction question. That work was performed at Brookhaven National Laboratory.

In the future, we anticipate that requests for research support will continue. Because of the ad hoc nature of this work, it is difficult to project topics that will be addressed. However, three areas seem to be likely possibilities. One is the specification of tests and associated acceptance criteria to assure performance of concrete expansion anchors under extreme loads. Another is the seismic performance of masonry block walls with emphasis on acceptance criteria. The last is the effectiveness of existing walls as protection against turbine missiles.

2. DESIGN LOADS PROGRAM

A crucial element in assuring the safety of nuclear plants is the choice of the level of loading for which the plants should be designed. The requirement that nuclear plant facilities be able to withstand extreme environmental and accidental loadings has placed new demands on the structural engineering profession. Some of the loadings postulated for nuclear plants had not previously been considered quantitatively in structural engineering practice. Either the occurrence of the event causing the load was considered too rare to be included in the design basis or the consequences of the rare failures to be expected were considered tolerable. Of particular interest, are loadings due to floods, hydrogen explosions, and water hammers. The present aim is to investigate existing literature and test data, perform any necessary testing, and develop procedures and methodologies for quantitatively assessing the safety of equipment and structures subjected to these loading conditions.

Another area of interest is the way in which the effects of loads should be combined for design purposes. Current practice is based mainly on the use of load factors to combine postulated accident and extreme environmental loads with normal operating conditions. There is no a priori reason to believe that the risks of exceeding design loads are consistent for the load combinations used; nor is it clear that the relatively large number of combinations are all necessary, since some may envelop others. All that can be said is that the probability of exceeding any one of the combinations is small. A more rational basis for choosing load combinations and associated load factors is desirable.

2.1 Engineering Characterization of Seismic Input

Objective

To develop recommendations for choosing design input-motion at the foundation level of nuclear power plant structures consistent with a free-field motion specified at the surface.

Background

It is well known that the peak ground acceleration in the nearfield is not correlated to the magnitude of the event. A minor fault could potentially cause the same peak ground acceleration in the epicentral region freefield as a major fault. Some differentiation must exist between the levels of seismic energy release associated with events of different magnitudes. Very few data exist on recorded accelerograms in the epicentral region. It is also felt that, due to energy dissipation at the foundation and soil interface, high frequency input motion from the freefield cannot be transmitted in the building foundation. According to prevailing views, it is safe to design nuclear power plants for less than the expected peak ground acceleration if the duration of the peak motion is sufficiently small.

The design of structures, systems, and components of nuclear power plants has been keyed to design response spectra of fixed shapes (Ref. 1) corresponding to different damping levels. The spectral ordinates were derived from a statistical analysis of elastic response spectra for a number of earthquake motions recorded at locations ranging from the 10 to 100 kilometers from the epicenters and in different soil conditions.

The choice of an appropriate acceleration value to characterize free-field motion for use in design has been the subject of a number of studies (Ref. 2, 3, 4, 5). The fact that a response spectrum does not explicitly account for duration of motion, distance from the epicenter, or earthquake magnitude has been noted (Ref. 6). There have been, however, correlations to indicate that these effects can be accounted for by adjustments of the spectral ordinates.

Of principal interest for design purposes is a spectrum which reflects the motion of the rigid mat on which critical structures at nuclear power plants are typically founded. Rocking and torsional motions are likely to be of significance. Preliminary efforts have been made to model the effects of wave passage on large foundations. However, these efforts have not included those soil-structure interaction effects which may be significant for large, deeply embedded foundations. Some work has been done in this area, but as yet, there have been no applications to nuclear power plant structures.

Research Needs

The choice of a spectrum for engineering design purposes must be based on equivalence of structural response. An additional consideration exists for nuclear power plant design. There are two design level earthquakes: Operating Basis Earthquake (OBE) and Safety Shutdown Earthquake (SSE). Response to the OBE must be essentially elastic while, for the SSE, any permanent deformations must be small and localized.

Project Description

Two tasks have been delineated:

Task I - Review the relevant literature and determine what analytical tools and studies will be necessary. Develop analytical studies, as required, to estimate response spectra and base input motions. Comparison of results with existing data, as available, will be undertaken during this task. Recommendations on a method for choosing response spectra based on translational motion will be completed as part of this task.

Task II - Recommendations on defining base motion shall be completed during this task. This shall include integration with the response spectrum methodology developed in Task I. Sensitivity studies to assess the significance of the phenomena studied on foundation level response spectra shall be included.

Project Results

Two results are contemplated. The first will improve the current state-of-the-art by providing a quantitative basis for choosing response spectra based on translational motion. Considerations here include the effects of: duration of motion, distance from the epicenter, short pulses of high acceleration, and earthquake magnitude. The second result will provide recommendations for methods to be used in selecting design response spectra or time histories to be used as motions at the foundation level. Additional considerations here include depths of embedment, soil-structure interaction, wave passage effects, and torsional and rocking motions.

2.2 Flood Effects

Objective

To assess the probability and consequences of exceedance of flood levels at nuclear power plants and to quantify the margins of flood protection available.

Background

Current methods and criteria for analysis of flood levels and for flood protection are summarized in Ref. 7 through 10. The assumption underlying current practice is that a nuclear power plant, hardened against the most severe postulated flooding conditions, is adequate to protect the public health and safety. Postulated flooding conditions are analyzed deterministically using techniques and procedures evolved from practice by other Federal agencies (primarily the U. S. Army Corps of Engineers, NOAA, FERC, and the Bureau of Reclamation). Furthermore, these techniques and procedures consider the range of causative mechanisms, including tropical storms, large- and small-scale extra tropical precipitation and wind storms, geoseismic activity and dam failures. No assessment is made of the probability of the flood conditions postulated.

Research Needs

A perspective is needed on the likelihood of failure of flood protection, the consequences of failure, the residual risks inherent in inadequate flood condition/flood protection criteria, or the degree of conservatism associated with the present methodology.

Project Description

Investigations of the use of probabilistic techniques, with emphasis on estimating recurrence rates for flood levels, will be conducted under a task devoted to hydrologic research on riverine, coastal, and the Great Lake sites. Since both flooding phenomena and protective structures differ for these locations, separate studies will be required.

All aspects of the flood phenomena will be considered in a task devoted to estimating the damage that could be caused by flooding at a plant site. These include: levels in the body of water, groundwater levels, wave activity, floating objects, influence of high water on the soil (erosion, slides), time delays and rates of innundation.

Another task will address the consequences of flood damage at a plant site in terms of the possible compromise of plant safety systems. As a measure of risk, the probability of core-melt accident will be expressed as a function of flood level.

Project Results

This project will provide a definitive basis for estimating the risk associated with both current flood protection methods and proposed alternatives.

2.3 Effects of Hydrogen Explosions

Objective

The objective of this project is to develop an understanding of the behavior and capacity of containment structures under internal explosions.

Background

Under certain conditions exothermic chemical reactions may occur in hydrogen-oxygen mixtures. In light water reactors (LWR's), hydrogen is generated from the primary coolant water, both during normal operation and during accidents. Sources of hydrogen during normal operation include aqueous corrosion of core metals, electrolysis and radiolysis of water. The sources of oxygen generation in the coolant during normal operation are in-leakage of air, and again, water electrolysis and radiolysis. During an accident that involves core heatup, hydrogen may be produced in the core by the high-temperature reaction of water with metals, namely with zirconium from the zircaloy fuel cladding and with iron from the molten steel. Large quantities of hydrogen gas may, thus, accumulate in the reactor vessel, as was actually the case in the Three Mile Island (TMI) accident of March 28, 1979. However, the amount of oxygen in the primary coolant system (preexisting oxygen plus that produced by radiolysis during the course of the accident) is such that a major hydrogen-oxygen reaction in the reactor pressure vessel is very unlikely, in spite of the conditions of high pressure, temperature and intense radiation. Should hydrogen escape from the pressure vessel due to loss-of-coolant, it may combine with the oxygen in the containment atmosphere into a flammable or even explosive mixture. In the course of certain accidents, the situation in the containment atmosphere may become more serious by the additional production of hydrogen by radiolysis of water in the reactor sump and by reactions of zinc contained in the paint on the inner containment spray solutions.

With a certain composition range, mixtures of hydrogen, air and (in general) steam will burn. Hydrogen burning occurs and propagates at slow speed, causing a quasi-static increase in pressure and temperature in the containment atmosphere. When confined, hydrogen-air-steam mixtures within a certain composition range may detonate. There is no experimental evidence that such mixtures will detonate under free-expansion conditions similar to the ones found inside a containment vessel. Nevertheless, such a detonation cannot be excluded, especially in situations involving turbulence and nuclear radiation. Hydrogen detonation, if it occurs, has the form of a shock wave propagation with supersonic speed.

Internal explosions may have catastrophic consequences, as they may cause failure of several engineered safety systems. In such an event, the containment structure is the last defense line against early release of radioactive fission-products into the atmosphere. Since hydrogen detonations and steam explosions are very unlikely, containment structures are not designed against them. However, since these are not inconceivable events, a thorough understanding of the containment behavior under the associated extreme loading conditions should be developed.

Research Needs

Although the question of containment capacity under internal explosions has been addressed in the past (Ref. 15, 16), no reliable answer has been given because of oversimplifications, both in the treatment of the dynamic load caused by the explosion and in the mechanical models of the containment structure.

Project Description

The project will proceed by exploring a sequence of tasks, some of which are interactive. The task outline is as follows:

1. The construction of numerical models for the explosion phenomenon and the generated shock wave, and the use of these models for the prediction of wall pressure as a function of explosion parameters (such as type, magnitude and location).
2. Development of an understanding of the dynamic characteristics, first, of linear elastic containment structures.
3. Assessment of the effect of containment deformability on the wall pressure induced by the shock waves.
4. Development of a capability for nonlinear dynamic analysis of containment shells, subjected to pressure loadings.
5. Identification and modeling of the uncertainties in the load and in the containment properties.
6. Using the capabilities developed in Task 4 to predict the response and to assess the integrity of the containment under postulated internal explosions. Development of an in-depth understanding of the behavior and performance of deterministic sensitivity analyses to the dominant uncertainties.

7. Construction of simple, but still accurate mechanical models to replace the sophisticated models in Task 6, and use of these models in a probabilistic analysis of containment behavior under strong internal explosions.

Project Results

This project will provide the following results:

(1) an understanding of the behavior and the limit capacity of containment structures under internal explosions;

(2) identification and quantification of the uncertainties in the phenomenon; and

(3) simple, but accurate, behavior models which can be used in a probabilistic analysis of the phenomenon.

2.4 Load Combinations for Design

Objective

The objective of this project is to provide a basis for improving the ways in which structural design load combinations are chosen.

Background

Because of their safety significance, nuclear plant structures are postulated to be subjected to simultaneous transient events in combination with various plant conditions. Current criteria evolved from basic civil engineering codes and specifications and neither ensure uniform probabilities of exceedance nor provide uniform margins of safety for the load combinations considered.

Existing methods for statistical combination of load effects have led to mixed results when compared with the constant load combination factor approach (Ref.17, 18). Depending on the relative magnitude of the individual load effects, the constant load factor approach may either under or overestimate the combined load effect. Information defining the extreme load effects for various transients are, however, limited.

Present strength design philosophy assumes inelastic behavior at failure but introduces load factors to ensure elastic response at the service levels. Nuclear plant structures are subjected to extreme transient events for which only local inelastic behavior and limited overall inelastic response are acceptable.

Research Need

A systematic evaluation should be made of the various event combinations and criteria for combining responses for various structural types with the intent of finding load factors leading to uniform margins of safety.

Project Description

Nuclear plant structures must be able to withstand both severe environmental loads and loads associated with postulated plant accidents, as well as combinations of both types with normal plant operating loads. Three tasks have been delineated, the first two of which can proceed in parallel.

Task 1 will build upon the work done in support of standard writing organizations in the United States, Canada and Europe. Those efforts have been directed at the development of a general loading criterion for building design. Not all the environmental loads postulated for nuclear plant structures are considered in the design of conventional building. However, many, such as live, dead and wind loads, are. Methods will be developed to choose load factors for the environmental loads used for design of nuclear plant structures that reflect the probability of exceeding the design load in the plant lifetime.

Task 2 will concentrate on loadings peculiar to nuclear power plants. These are mainly due to postulated transient events and accidents. Methods will be developed to permit the choice of load levels and load factors for design on a risk-consistent basis.

Task 3 will involve the development of load combination criteria within the context of probabilistic limit states design. This will permit use of the definition of structural reliability by the probability of not exceeding one of the limit states for the structure or its elements. A limit state is, simply, a condition when a structure becomes unfit for its purpose in some way. Limit states can be associated with both strength and deformation. Limit states design is, thus, a formal procedure for considering explicitly, different possible modes of structural behavior.

Project Results

The expected results of this project will include a detailed evaluation of the basis for various event combinations, and recommended criteria for combining loads consistent with the corresponding acceptance criteria.

3. STRUCTURAL RESPONSE PREDICTION PROGRAM

After agreement on the loads a structure must be able to sustain, there follows, in the structural design process, what is usually called the analysis phase. Here, analytical methods are used to predict the behavior of the structure under the design loadings. For conventional buildings, the analytical procedures in general use have been developed through years of experience. In general, the procedures have been calibrated against laboratory measurements on scaled models and, in some cases, on full-scale buildings. There is a general confidence in the structural engineering profession about the ability of those procedures to predict performance of buildings under design loadings. The degree of confidence is related to the amount of relevant experimental data. In regard to common building types for which many tests have been run and for loadings which can be easily replicated in the laboratory, there is a great deal of confidence. For example, the behavior of steel-framed buildings under gravity loads is felt to be quite predictable. Where data are sparse, as in nuclear power plant structures, or loadings difficult to simulate, as in earthquakes, there is less confidence.

In the absence of experimental results, a great deal of reliance is placed on analytical methods. For some structures, where different analytical models of the same structure have been shown to give similar results, that reliance seems well justified at least for the loadings studied. When there is not a great body of such evidence, the best that can be said is that the response predictions have been performed in a rational way and are based on experience gained for other types of structures.

While that approach has merit, there is uncertainty about the accuracy of response predictions. The objective of this program is to reduce those uncertainties.

3.1 Soil-Structure Interaction

Objective

The objective of this project is to improve the state-of-knowledge about the response of heavy, deeply embedded structures to seismic ground motions.

Background

Stiff lightweight structures move essentially as the earth moves during seismic events. Heavy stiff structures, such as nuclear power plants, significantly modify motions of the earth in the vicinity of the structure such that foundation motions, which determine seismic loads, depend to a large degree on the mass and flexibility of the plant itself. This type of interaction between soil and structure can be computed by several techniques using various assumptions and leading to results with different levels of conservatism. No real consensus exists in the seismic engineering community on the reliability of the various techniques used to model the soil-structure interaction phenomenon.

Research Need

Recent surveys (Ref. 19, 20) of the state-of-the-art in soil-structure interaction analysis have been performed. These surveys indicate that although much work has been done and that a number of calculational techniques exist, there is no great confidence in the ability of current modeling techniques to predict the effects of soil-structure interaction.

Project Description

Current activities are limited to the soil-structure interaction project of the Seismic Safety Margins Research Program (SSMRP). (For a complete discussion of this program, see (Ref. 14). The major outcome of Phase I of the SSMRP will be a ranking of soil-structure interaction prediction uncertainties and other contributors to uncertainties about the effectiveness of seismic design of nuclear power plants. Should soil-structure interaction effects prove to be a major contributor, further work will be undertaken to reduce that uncertainty.

Project Results

A definitive assessment of the significance of uncertainties about current ability to model the soil-structure interaction phenomenon, vis-a-vis other uncertainties in seismic design, will be made. This will include comparisons of the predictions made by different modeling techniques.

3.2 Building Response

Objective

The objective of this project is to reduce the uncertainty about methods used to predict the response of buildings to earthquake motions.

Background

The theory of structural response has been developed to a stage beyond that routinely exercised in the seismic analysis and design of nuclear power plants. The major reason why simplified, rather than complex structural models are used is economic; the simpler codes require less engineering effort and computer time than more complex ones. The development of a new generation of computers, advances in developing efficient numerical techniques, and a modularization of computer programs may mean that a special purpose computer program could be developed which could economically evaluate the response of much more complex structural models than is presently used. These models should provide a much more realistic estimate of the structural response than the present ones. For example, some nuclear power plant structures are highly asymmetric and it is not clear that anything but a three-dimensional model of such a structure will produce a realistic estimate of the structural respon[illegible] Such three-dimensional analyses will be quite expensive computa[illegible]lly and their cost may not justify the added accuracy.

Research Need

Current practice in predicting building response utilizes models which are known to be an approximate, but to an unknown degree. An assessment of the improvement to be gained by the use of more complex models is needed.

Research Program

Current activities are limited to the structural building response project of the SSMRP. (For a complete discussion of this program, see (Ref. 14). The major outcome of Phase I of the SSMRP will be a ranking of building response prediction uncertainties and other contributors to uncertainties about the effectiveness of seismic design of nuclear power plants. Should uncertainties about current methods used to predict building response prove to be a major contributor, further work will be undertaken to reduce that uncertainty.

Project Results

A definitive assessment of the significance of uncertainties about the current ability to predict building response to seismic motion, vis-a-vis other uncertainties in seismic design, will be made.

3.3 Dynamic Testing of Structures

Objective

This program is to provide a rational basis for decisions concerning the inclusion of dynamic testing of nuclear power plant structures as part of the licensing process.

Background

Seismic design practice, as applied to nuclear power plant structures, has undergone significant change in recent years. As a result, there is an acknowledged diversity of opinion concerning the proper methods to apply in specific cases, the range of validity of particular methods, and the relationship between current practice and experimental observation. This diversity of opinion is usually revealed in an overall commitment to conservatism whenever substantive questions of applicability are raised. To reduce unneeded conservatism and provide a rational connection between the design methods which are acceptable in the licensing process and the variety of possible methodologies, there are several questions of verification that must be answered. In particular, the connection between responses predicted by analysis and the response of the as-built plant must be clear, and the qualifications inherent in the applied models must be documented and acceptable.

It should also be noted that problems of unneeded conservatism and the validation of design methodology are not restricted to seismic analysis. In general, the treatment of most severe loading conditions, both dynamic and thermal, raise the same questions and can be addressed using methods similar to those employed in seismic analysis.

A major question that occurs in developing validated seismic design methods concerns the function of direct dynamic testing of the as-built plant. In particular, what tests are feasible, what are the limitations of the methods that will utilize the test data, and what confidence levels can be assigned to data obtained from various tests? Further, the validation of seismic analysis methods which are integrated with dynamic testing of full-scale structures is intimatley associated with the presumed condition of the plant at the time of testing. Thus, any discussion of such validation must acknowledge seismic history as a major contributing factor. While it may be possible that specific classes of analysis will not require a complete treatment of load history, it will always be necessary to acknowledge history effects by evaluating how their neglect will alter reliability and accuracy of results. In all cases, the inherent accuracy limits imposed by modeling assumptions must be considered and trade-offs in accuracy versus simplicity of analysis must be evaluated.

Research Need

The applicability of testing methods as adjunct to, or possibly substitute for, analytical methods to predict structural response or assess structural damage must be examined.

Project Description

The current program will develop a critical evaluation of alternate methods of dynamic tests - both preoperational and post-earthquake - and recommend an experimental methodology that is consistent with available analytical methods. It is anticipated that the range of methods considered will extend beyond those used in conventional design practice. These analytical methods, based primarily upon the extensive system identification literature, will also be subjected to critical evaluation to ensure that the recommended testing and analysis procedures are feasible and compatible. Verification of recommended methods will utilize existing experimental sources such as data obtained from explosive testing at the Heissdampfreaktor (HDR), preoperational test data obtained from foreign sources, and relevant nonreactor experiments appearing in the open literature. Evaluations and recommendations will be extended to cover treatment of damaged structures from the viewpoint of expected changes in baseline experiments, impact of damage estimates on requalification of plant structures, and recommended testing and analysis procedures needed to supply relevant damage estimates.

Three basic tasks have been identified. The tasks are as follows:

Task 1: <u>Evaluation and Utilization of Dynamic Tests on Existing Structures</u>

This task will involve participation in existing experimental programs and gathering of additional data from exchange programs, and the open literature. It will provide the baseline information for use in evaluation of analysis methods and in justifying recommended testing procedures. An interim assessment of the value of dynamic testing will be prepared.

Task 2: Feasibility of Preoperational Testing and Verification of Methods of Seismic Analysis

The major objectives are:

(a) development of a critical evaluation of available analysis methods and their applicability to feasible levels of preoperational testing,

(b) verification of selected methods through use of the data base developed in Task 1, and

(c) recommendation of an integrated experimental/analytical approach to preoperational seismic analysis of undamaged power plant structures.

Task 3: Assessment of Seismic and Post-Accident Damage and Requalification of Damaged Structures

In parallel with the objective of Task 2, this task will develop extensions of dynamic testing methodology and associated analytical methods which incorporate a rational measure of dynamically induced damage and which relate this damage assessment to both feasible post-earthquake and post-accident testing and to structure requalification.

Project Results

A basis for deciding on the potential of testing techniques to reduce the uncertainties in response prediction will be developed through an integrated evaluation of dynamic testing methods, the utility of analytical models which use experimental results, the verification of such methods and models as applied to both as-built, undamaged structures and those that may have sustained damage from seismic activity, or from other severe dynamic or thermal loads.

3.4 Benchmarking of Computer Codes

Objective

This project is intended to improve confidence in the ability of computer codes to predict the response of structures at nuclear power plants.

Background

Recently, questions have arisen about the adequacy of computer codes used for seismic analyses of piping systems in nuclear power plants. Computer codes are also used in determining the adequacy of Category I structures under seismic loading. The objective of this project is to improve the capability of the Nuclear Regulatory Commission (NRC) staff to assess the analyses submitted by applicants for Category I structures.

As part of the licensing process for nuclear power plants, the NRC staff reviews the integrity and functional adequacy of structures under normal and extreme loading conditions. Material submitted by applicants and reviewed by the NRC staff includes:

1. description of the structures;
2. applicable codes, standards and specifications;
3. loads and load combinations;
4. design and analysis procedures;
5. structural acceptance criteria;
6. materials, quality control and special construction techniques; and
7. testing and in-service surveillance programs.

Review of the material submitted is carried out as indicated in the Standard Review Plan (SRP). Structural engineering topics are concentrated in the SRP Chapter 3, "Design of Structures, Components, Equipment and Systems." Typically, part of the information submitted is a statement that design and analysis procedures utilize certain computer codes. In order to assure that the codes used give reliable results, applicants are asked to demonstrate that either:

(a) the computer program is recognized and widely used, with a sufficient history of successful use to justify its applicability and validity without further demonstration by the applicant, or

(b) the computer program solutions to a series of test problems with accepted results have been demonstrated to be substantially identical to those obtained by a similar program which meets the criteria of (a) above. The test problems shall be demonstrated to be similar to, or within the range of applicability for, the problems analyzed by the computer program to justify acceptance of the program, or

(c) the program solutions to a series of test problems are substantially identical to those obtained by hand calculations, or from accepted experimental tests or analytical results published in technical literature. The test problems shall be demonstrated to be similar to the problems analyzed to justify acceptance of the program.

There are shortcomings in the current NRC procedures for verification of accuracy of analyses. While a history of successful use is reassuring, it is not a guarantee. Some well-known computer codes, with the capability to perform dynamic analysis through modal superposition techniques, have been known to give erroneous results for symmetric structures. In more complex system codes, various options and subroutines may hardly ever be utilized, which implies that the history of use of codes must be qualified by examining the complexity of the code and the various options it provides. The method wherein the solutions of the proposed codes are verified against the solutions of codes available in the public domain is not foolproof either. In this method, it is necessary that the test problems be substantially similar to those that can be solved by codes in the public domain. In general, the test problems do not represent real structural and mechanical systems in nuclear reactor applications and may not test those parts of the proposed code that were developed to make the proposed code better than the publicly available codes. Finally, a comparison with classical solutions suffers from the same drawback since the proposed code was, presumably, developed to address problems for which classical solutions do not exist.

Research Need

A computer code, or set of codes, is needed for NRC staff use, to reliably predict structural response. Benchmark problems must be developed to demonstrate adequate performance over the ranges of variables typically encountered.

Project Description

Computer codes will be developed to perform calculations of the type required in safety analyses of nuclear plant structures. The most important element in the code development is verification of reliable performance of calculations required for nuclear plant structures, not just those needed for the design of conventional buildings.

Verification of design calculations involves:

(a) verification of the algorithm utilized to perform calculations, and

(b) verification of the model used to represent the behavior of safety-related structural systems.

Both of these topics are to be considered in this research project, but comparisons to test data will be limited to existing data.

Benchmark problems will have to be devised to permit an evaluation of code performance. The computer code (s) need not be new developments. Computer codes, or parts of computer codes, in the public domain are acceptable if their reliability can be demonstrated. Simplification of input requirements and enhancement of output by means of graphic displays will significantly contribute to efficiency for NRC staff use. Ease of utilization is desirable, but reliability of the results cannot be compromised and attain it.

Project Results

Computer codes will be available to permit the NRC staff to either check the reliability of codes used by applicants or perform independent safety analyses.

4. STRUCTURAL PERFORMANCE PROGRAM

After the loads that a structure must sustain have been determined, and after the responses to those loadings have been predicted, the final stage in a design is to proportion the structure so that adequate margins of safety exist. Simply put, a safety margin is the perceived difference between the actual capability of a structure and that calculated to be necessary to sustain the design loading. The purpose of the safety margins is to assure adequate performance of the structure in view of uncertainties about the actual load on the structure and its actual response. The smaller the uncertainties, the less the margin need be. The intent of the two programs previously described is to reduce uncertainty about loading and response. This program will concentrate on establishing the capacities of structures so that margins of safety can be determined.

Current practice relies on criteria put forth in codes and standards to assure that structures will perform adequately. The relevant codes and standards are:

Structures	Code/Standard
Concrete Containment	Division 2, Section III, Boiler Vessel Code of the American Society of Mechanical Engineers
Steel Containment	Division 1, Section III, Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers
Concrete Category I Structures (other than containment)	Code Requirements for Nuclear Safety-Related Concrete Structures, ACI-349-76, American Concrete Institute
Steel Category I Structures (other than containment)	Specification for the Design Fabrication and Erection of Structural Steel for Buildings, American Institute of Steel Construction

The above codes and standards were developed in a evolutionary fashion from historical practice in the design and engineering of pressure vessels, buildings and steel structures. In various instances, those codes had to be modified to reflect not only the more stringent loads in nuclear power plant structures, but also the differences between conventional structures and nuclear plant configurations. In many instances, the bases for the requirements of these codes and standards have been more engineering judgement than definitive test data. The extrapolation to cover nuclear structures has led to concerns of appropriateness of the rules. Questions concerning adequacy and/or overconservatism of certain requirements have been difficult to resolve in the absence of relevant test data.

In addition to, and interrelated with, the above determination of code and standard applicability and adequacy is the determination of the capacities and associated failure modes of containments and other Category I structures. Containment design, with its emphasis on leak tightness, has always been grounded in elastic design methods where little effort has been devoted to the predictions of ultimate capacities. Similarly, other Category I structures tend to be heavy shear-wall structures which differ substantially from traditional framed structures. Therefore, the applicability of the vast body of analytical and experimental work completed for those traditional structures is questionable for nuclear plants. In order to estimate the reserve capacity or safety margin of containment and other Category I structures, reliable estimates of the load levels associated with unsatisfactory performance (limit states) must be developed.

A major consideration in assuring adequate structural performance is the correspondence between the structure as designed and as actually built.

Quality assurance and quality control procedures are intended to provide a structure in conformance with the assumptions made during design. In reality, large structures are never built exactly as designed. Often, situations encountered during construction require modifications. This is not, in itself, harmful and may result in an improvement in structural performance. There have been, however, disastrous failures in conventional structures that were attributable to erosion of the design safety margin by construction practices. There is also a long history of inadequate performance in conventional structures due to careless workmanship. There have also been instances discovered at nuclear power plants, but none, as yet, have led to any serious consequences for public health and safety. It may be that the quality assurance and quality control procedures in nuclear plant construction, which far exceed those for conventional construction, are adequate to assure that construction deficiencies will be found and corrected before the public is exposed to any great risk. But that cannot be shown to be so. Moreover, there is a feeling in the structural engineering profession that current procedures focus too much on record keeping to the detriment of actual inspection and do not differentiate well between minor deviations of no great safety significance and those with the potential to lead to serious consequences.

4.1 Safety Margins for Containments

Objective

To develop reliable methods of predicting the capacities of containment buildings at relevant limit states.

Background

The containment structures at nuclear power plants provide a barrier to contain any fission products accidently released from within the reactor pressure boundary. The design of these structures is generally governed by the thermal and pressure loads assumed to be a part of a loss-of-coolant accident (LOCA). The structures are dimensioned to assure that deformations, and hence leakage, under this condition are limited to acceptable levels. The design procedure is based on elastic methods and, while quite reliable for the assumed loading conditions, cannot be extrapolated to give estimates of the actual capacity of a containment to sustain loads in excess of the design basis. A containment building can be expected to perform satisfactorily at pressure and temperature combinations more demanding than those for which it was designed; however, no reliable estimate of the actual loading associated with unsatisfactory performance can be made.

In addition to the generally symmetric loadings which govern its design, a containment building must be able to withstand an array of accident and extreme environmental loadings, most of which are nonsymmetric.

Among the loadings of principal interest are those due to seismic events and internal explosions. The accident at the TMI plant, a d an attendant's concern about the possibility of a hydrogen explosion, reemphasized the existing uncertainty about the ability of containments to sustain extreme loadings outside the design envelope.

Research Need

Experimental evidence is needed so that methods of predicting containment capacity can be calibrated.

Project Description

Both analytical and experimental avenues will be pursued. The experimental effort will be divided into two parts. The first will involve design of the experiment; the second will be the actual testing. Three tasks have been identified:

Task 1 - Analytical Effort

This task will emphasize the development of an analytical or semi-empirical method to predict the ultimate capacities of concrete containments (both reinforced and prestressed types) and of steel containments subject to accident and extreme environmental loadings. The main effort will be directed toward the experimental verification of such methods under the eventual completion of Task 2 of this program. Inelastic analyses of cracked concrete containment structures may be required to more accurately evaluate the deformation of the vessel, the distortion of the liner, and the general response of the structure. The study should take into account factors such as the effective shear stiffness of a cracked wall under the characteristics of reversing dynamic loads, the potential splitting effects in biaxially tensioned concrete from shear-induced dowel forces on large numbers of No. 18 reinforcing bars, and the effects of high frequency vibration on shear transfer behavior and resulting structural response. In addition, capacity and failure modes under combined loadings will be considered. This will account for any interaction effects between loads acting simultaneously and provide a better estimate of the structural capacity.

Task 2 - Experimental Design

This task will involve a study of the feasibility of testing scaled models of containments to establish failure modes and capacities. Construction, instrumentation, and testing procedures must be considered. The credibility of the results to be expected from scaled testing must be assessed, as well as how these results would be best incorporated into the analytical procedures of Task 1.

This task will be, essentially, a planning effort. It will involve design of the experimental facility needed to perform the requisite testing program. Full-scale testing is not feasible, and an appropriate choice of scale for model testing will be, perhaps, the most significant result of the first phase. Current thinking is that the 1/10 scale used in recent Polish (Ref. 21) and Canadian (Ref. 22) pressure tests may be too small. It would be desirable, particularly for concrete containments, to choose the scale so that materials representative of those used in full-scale structures can be utilized. A scale closer to 1/4 is thought to be necessary to accomplish this. A thorough assessment of the applicability of results for a range of scale factors will be necessary. Cost must also be considered. Recommendations for tests to be performed for the tests specimens and test facility will be provided.

Task 3 - Testing Effort

After the recommendations of Task 2 are evaluated, the plan for the actual testing program will be finalized. Tests to failure under symmetric and unsymmetric pressure loadings are contemplated as well as tests to failure under lateral loadings. The main purpose of the tests is to permit calibration of the predictive methods developed under Task 1. These predictive methods will be developed and checked by the scale model tests, for reinforced and prestressed concrete containments, and for steel containments of hemispherical and spherical shapes.

Project Results

This project will result in methods which can be used to verify predictions of containment behavior under design loadings and to estimate safety margins available to accommodate loads outside the design basis.

4.2 Safety Margins for Category I Structures

Objective

The objecitve of this project is to develop reliable estimates of the actual capacities of Category I structures to sustain loads in excess of those for which they were designed.

Background

The buildings, other than containments, which house safety-related equipment at nuclear power plants are often heavy, low concrete shear wall structures. Wall and roof panels range in thickness from 1 to 2 feet. These buildings are similar in appearance to industrial buildings and the code governing their design is based on the one used for those more conventional structures. However, the internal load distribution in these heavy Category I buildings differs from that encountered in framed structures. Consequently, the body of analytical and experimental evidence developed over the years for framed structures is not directly applicable.

With increasing frequency, it has become necessary to assess safety margins of Category I structures in nuclear power plants that can be subjected to a variety of severe dynamic loadings. This has often occurred in connection with changed perceptions of the seismic hazard at a plant. The loadings of interest, while of low probability, can happen during the operating lifetime of nuclear plants. Examples are (a) dynamic loadings from a possible explosion or other pressure transients in the containment buildings, (b) missile loadings from equipment failures, (c) dynamic loads from piping systems, and (d) seismic loadings. The analytical capability to address these problems has been steadily refined over recent years and the state-of-the-art is such that both failure modes and failure loads can be predicted for many simple structures for which the material behavior is well characterized. However, for many Category I structures, such as heavy concrete shear wall structures, the analyst is stymied in his efforts to confidently predict either failure modes, failure loads, or even the dynamic behavior in the nonlinear range because of a lack of available experimental data for the structural materials and elements.

Research Need

An appreciation of the actual safety margin inherent in the design of Category I buildings is necessary to estimate the risk associated with a possible failure of these buildings under accidental or extreme environmental loading.

Project Description

This project is aimed toward assessing the margin to failure for common classes of nuclear Category I structures. The main method used for this assessment will be an analytically supported and carefully planned experimental program. A detailed study of design and construction methods currently in use will be necessary. The recent American Concrete Institute publication, "Reinforced Concrete Structures in Seismic Zones," (Ref. 23) is an excellent summary of the extensive research that has been conducted on the various components of reinforced concrete structures. The possibility and desirability of accounting, in the seismic design procedure, for energy dissipation by means other than equivalent viscous damping will receive special attention.

During the course of the experimental program, a data base of information regarding specific damping values and their physical cause for the given class of Category I structures, and a data base of failure modes for this class will be built and maintained. The best analytical methods and models to make use of these data bases will be investigated and demonstrated by analytically modeling the selected experiments. The following tasks can be identified:

Task 1 - <u>Identification Survey of Category I Structures for Structural Classification Purposes</u>

A survey of the existing and planned nuclear plants will be made so that the Category I (exclusive of primary containment) structures can be identified as belonging to structural classes, and the methods used to predict ultimate strength will be examined. As a part of this task, for the concrete structures, the construction technique (methods of reinforcing, concrete [illegible]gth, etc.) used and planned for future use will be accounted.

Task 2 - <u>Selection of Representative Structures For Analysis and Testing</u>

The classes will be studied, and from each class, a structure that is judged to be most representative will be selected for analytical and experimental modeling. Here, heavy shear walls will receive special attention. For structures that interact with sensitive equipment, the possibility of including the equipment or a portion thereof in the experiment will be carefully evaluated, particularly with respect to defining loss-of-functionality. Identification of loadings that will be considered are (a) blast, (b) missile and (c) seismic loading.

Full-scale experiments on Category I structures pose obvious difficulties, especially when the purpose of the experiment is to determine capacities at failure. However, in the preparation of an appropriate experimental program plan, the possibility of large-scale experiments will not be ruled out. The utilization of subassemblies, components, and less than full-scale structures to determine margins to failure of Category I structures will be carefully considered in the preparation of the proposed test plan. The work reported by Clough (Ref. 24) is indicative to the advances recently made in this area.

Task 3 - Analytical Modeling

During the selection process of representative structures for analysis and testing, some preliminary analytical modeling will begin. The modeling will be as simple as possible so that various parameter studies can be carried out to assist experiment design. Ultimate capacity of the structures will be determined. As part of this effort, the effects of scaling will be investigated using the representative structures.

Once the initial experiments have been selected, detailed analytical modeling will begin. Ultimate capacity of the structures to be tested will be determined with the best codes and methods available. Information regarding the sensitivity of various parameters in the analytical modal will be invaluable in planning the instrumentation necessary to separate various effects and in ensuring that in the planned experiment the desired effect can be measured.

Task 4 - Testing Effort

After the structures to be tested are selected in Task 2, experiments will begin. Parallel analytical modeling will also begin as test plans are established. This analytical modeling will be chosen to make the best use of the experimental data bases. Recommendations for possible changes in design methods and philosophy will be formulated and their merit evaluated. A report will be issued summarizing the results as applied to concrete structures.

Project Results

This project will result in methods which can be used to verify predictions of Category I structures under design loadings and estimate the margins of safety available to accommodate loadings outside the design basis.

4.3 Buckling of Steel Containments

Objective

The objective of this project is to develop experimental evidence against which analytical predictions of shell buckling loads can be checked.

Background

A steel containment vessel is usually a welded steel structure composed of a cylindrical shell attached to a dome structure, and a containment base plate. Because steel containment vessels provide an important pressure barrier for nuclear reactors, their design should resist the most adverse combination of loadings to which they might conceivably be subjected.

The primary function of the containment structure is to localize the effects of a LOCA. The onset of the LOCA condition immediately subjects the containment shell to asymmetric internal dynamic pressures with very high magnitudes. Static, thermal, seismic, and other loadings are postulated to act simultaneously with the LOCA, but the LOCA is usually the most critical loading condition for both shell stress intensity and buckling criteria.

The current standard methods for determining the buckling loads of steel containment vessels that are subjected to asymmetrical dynamic pressure loads have not been verified by testing or accurate analysis.

These formulae are based on linear elastic buckling theory for unaxial stress states. The state of compressive stresses generated, due to loads such as dead loads, seismic, external pressure, etc., are biaxial and inconstant - biaxial because there are hoop and meridional stresses together, inconstant because both hoop and meridional stresses can vary along the length of the cylindrical shell. Also, the problem of dynamic buckling of the containment shell in the presence of asymmetrical loads, such as that due to seismic and safety relief valve blowdowns, has not been adequately addressed.

Numerical methods to predict shell buckling have been developed using finite element techniques. There is confusion as to which of several analysis methods should be applied to a given problem. In Reference 13, a procedure is described that uses a bifurcation analysis with an accurate linear prestress field determined from the dynamic loading. This prestress is modified by increasing the in-plane compressive stresses and shearing stresses by suitable knockdown factors as determined from experimental data on uniform states of stress. This method is recommended for testing on nuclear containment designs. Reference 13 further points out that the current ASME pressure vessel code is inconsistent in this area and can lead to a selection of the least conservative analysis. Furthermore, there are not enough experimentally determined data on either expected imperfections of shape and support conditions, or expected knockdown factors for nonuniform stress states to permit these analytical methods to be used indiscriminately. It is further pointed out (Ref. 13) that the portions of the current ASME pressure vessel code used for reduction factors are not applicable to combined loadings of stiffened structures.

Research Need

An assessment is needed of the margins of safety against buckling failure for steel containments.

Research Program

A testing program will be established to check the accuracy and applicability of methods proposed to predict buckling loads for containment shells. Coordination with on-going analytical prediction programs will be established to assure that the experimental design will be able to encompass all the variables thought to be significant.

Project Results

A comparison of analytical and experimental results for shell buckling predictions will be provided. The adequacy of the predictive methods for ranges of variables significant in containment design will be summarized.

4.4 Adequacy of Codes and Standards

Objective

To develop the information necessary to assess the adequacy of selected requirements of reinforced concrete codes.

Background

The design of concrete containments is governed by Division 2, Section III, of the Boiler and Pressure Vessel Code of the American Society of Mechanical Engineers. The design of other Category I concrete structures is governed by ACI Standard 349-76, entitled "Code Requirements for Nuclear Safety-Related Structures." Questions have arisen about the adequacy or degree of conservatism associated with certain provisions of these codes. Areas of particular concern are those related to requirements for resisting tangential and peripheral shear in reinforced concrete containments, and for assuring adequate ductility in reinforced concrete structures under dynamic loading. The code provisions in these areas are based on a consensus judgement and do not rest on specific experimental evidence.

Research Need

Experimental evidence is needed to assess the adequacy or degree of conservatism in code requirements.

Project Description

Two tasks have been identified.

Task 1 - Allowable Shear in Containments

Testing programs are currently in progress at PCA and Cornell University. These programs investigate the effects of level of biaxial tension, percentage of reinforcement, diagonal steel, sequence of loading, and reversing shear load on behavior of containment walls. In the current program, the main emphasis has been on tangential shear loading (Ref. 25, 26, 27, 28). Pilot tests to investigate punching shear have been performed (Ref. 29).

The primary objective of the test program described in this proposal is to investigate effects of additional parameters. Tests will be selected to provide data where current information is insufficient to ensure the safety of containment vessels.

The areas include:

(1) extending the present test program to include other variables; and

(2) determining the effect of biaxial tension on the peripheral or "punching" shear strength.

Tests will be conducted on large and small specimens. Large-size specimens will be used to simulate full-size containments. Small specimens will be used to evaluate effects of different variables.

Reports describing the test program, test results, and their implications will be prepared.

Task 2 - Ductility of Concrete Structures

A complete literature search and evaluation of available experimental and analytical results will be performed. Sources will be catalogued according to the type of structure or element, (frames, walls, etc.) and according to the causes of nonlinearity. Various hysteretic models will be compared and evaluated, mainly by looking at their influence on typical structures in a nuclear reactor facility.

The study of references would result in usable definitions and descriptions of nonlinear behavior to be used in computer modeling. The significance of the differences in existing modeling will be tentatively evaluated using simplified models of the types of concrete structures of interest.

The load-deformation characteristics of several types of elements will be incorporated in nonlinear dynamic analysis programs. Such programs are available for several types of applications. Some are detailed studies at the local level (for example, finite element analyses of reinforced concrete elements), while others are used for the analysis of complex structures.

This synthesis of information and programs is essential to the rational assessment of the safety of, and the design procedures used for nuclear structures, such as walls, floors, and other elements of auxiliary buildings and containments.

Project Results

Experimental evidence will provide a basis for reassessing code requirements for containment shear and ductility of concrete structures.

4.5 Effectiveness of Quality Assurance and Inspection Procedures

Objective

The objective of this project is to provide a basis for designing inspection programs during plant construction.

Background

The NRC Office of Inspection and Enforcement has been reevaluating the adequacy of its inspection program for power reactors. The intent is to determine if the effectiveness of the program can be enhanced by an expanded application of independent verification methods.

A recent study (Ref. 12) utilized risk analysis methods to identify those plant functions, systems, and components that are most important to public safety. Independent verification options were identified that would be effective in ensuring that those important plant functions, systems, and components are available when needed. That study is not directly applicable to the design of a construction inspection program, but the identification of vital plant systems and methods to verify their availability, provides a perspective for a construction inspection program.

Inadequate construction practices can threaten public safety if, under an extreme loading condition, the structure either deforms too excessively to permit proper operation of safety equipment or fails in such a way as to damage vital systems. The current practice of relying mainly on quality assurance and control procedures to assure adequate construction does have some degree of effectiveness. It is an outgrowth of the practice utilized in conventional buildings where failure rates are historically low. It has, however, two major weaknesses. First, it has never been shown that the modifications of conventional practice included in quality control and assurance procedures for nuclear power plants, are sufficient to provide the additional safety margins desired. Finally, reliance on inspection performed by outside parties compromises the independence of the NRC staff statements about the adequacy of plant construction.

Research Need

A basis is needed for designing an NRC inspection program to verify the adequacy of plant construction.

Project Description

This project will focus on several tasks. Minimum information necessary for the NRC staff to be able to state that the plant as-built, and in the case of operating plants, as existing, is structurally adequate in regard to public health and safety, must be identified. It is necessary to determine how much information can be inferred from the quality assurance and control procedures in use. The historical evidence provided by operating plants will provide a way to assess current practice. Consideration will also be given to the effectiveness of in-service inspections to assure plant safety. Emphasis will be given to the added safety benefit to be gained by additional inspection. It may be necessary to identify and develop nondestructive examination techniques that can improve NRC staff capability for assessing structural adequacy.

An assessment should be made of the contribution of nuclear quality assurance and quality control in civil/structural construction. Recommendations should be made regarding any additional procedures that may be necessary to improve safety.

Finally, specific recommendations should be made aiming at improvements in design criteria, material specifications, construction tolerances, acceptance standards, reinforcement and structural steel fabrication details.

Project Results

This project will result in recommendations for independent inspections by the NRC staff that can markedly increase the assurance of plant safety.

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The Nuclear Regulatory Commission's (NRC) research needs in the safety and reliability of nuclear power plant structures are addressed by research programs under the Structural Engineering Research Branch, (SRB) which represents one element of General Reactor Safety Research (GRSR) with the Division of Reactor Safety Research (RSR). The objectives, background, research needs, project description and project results for each program element are discussed. The base year for this program is 1980. In some cases, research needs may change and projects will be modified. The projects described reflect current assumptions about long-term funding levels. Should those assumptions not be realized, project scopes will have to be modified.

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