Perceived Margins of Safety in Nuclear Power Plant Structures under Evolving Design Codes and Loading Criteria

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Abstract

Early nuclear power plants within the United States were constructed to criteria which have since undergone considerable revision. A comparative study of the past and the current structural design bases for Seismic Category I buildings and civil engineering structures within ten older U.S. plants is described.

The work is being conducted under the aegis of the NRC's Systematic Evaluation Program. On a plant-specific, structure-by-structure basis, the study considers the changes in the analytical provisions of the principal design codes (ASME, ACI, and AISC codes) in conjunction with changes in the postulated load and loading combinations.

Baseline documents, study methodology, classification of potential impacts of criteria changes on perceived margins of safety, and preliminary findings are discussed.

1. Introduction

With the advent of nuclear power, provisions addressing facilities for nuclear applications were progressively introduced into the codes and standards to which plant buildings and structures are designed. Because of this evolutionary development, these structures, in older plants, conform to different versions of structural codes--many of which have since undergone considerable revision. Regulatory agency licensing criteria have also seen development, contributing further to the non-uniformity of the requirements to which early plants were licensed.

The United States Nuclear Regulatory Commission (NRC), through the Systematic Evaluation Program (SEP), undertook an assessment of the safety of older nuclear power plants within the United States in the light of current design criteria and, as one aspect of the SEP, supported the work reported here. The objectives, methods, progress, and findings to date of this work are discussed.

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2. Scope

The study undertakes a comparison of criteria used for the design of SEP plant Seismic Category I civil engineering structures, i.e., those structures--which because of their importance to safe shutdown, accident mitigation, or suppression of off-site radiation dose--are normally examined in NRC licensing reviews. Explicit among the criteria to be considered in the study are the structural design codes and the loads and loading combinations postulated for these structures.

The focus of the study upon the needs of the SEP effort provides a set of natural boundaries to the scope:

- Only code versions used in the design of the Seismic Category I structures in the ten selected SEP plants are studied.
- It is desirable to base SEP assessments, to the greatest extent possible, upon factors which may be quantified. Accordingly, the study is limited to sections of the applicable codes dealing with analytical design requirements. Sections of the codes governing material testing, construction procedures, quality assurance, inspection, and the like are not examined because, even though they are important to safety, it is hard to quantify the safety impact of alternative procedures.
- The overall SEP effort is large, consisting of many tasks, spanning a broad spectrum of safety-related issues; the work reported here is but one of these tasks. A number of others deal in detail with specific aspects of plant structural design. To avoid duplication of effort, interface boundaries were established. In such areas, findings of the detailed studies are incorporated into the results of the code comparison review.

3. Objective

The broad objective of the NRC's Systematic Evaluation Program is to assess the safety of older nuclear power plants in accordance with the intent of the requirements governing the licensing of current plants and to provide assurance, possibly involving backfitting, that operation of these plants conforms to the general level of safety required of modern plants.

Many aspects of plant safety must be reviewed, among them the structural integrity of safety-related plant structures. One approach would be to perform a full reanalysis of all existing structures using analytical techniques reflecting best current practice and imposing today's design criteria--a task of very large proportions. An alternative approach, the one actually adopted, is to review the design basis then in effect and to compare it to current requirements, determine the deviations from current requirements, assess the potential impact on plant safety of identified deviations, and then evaluate the impact on as-built structures.

Thus, the objective of the code comparision review is to carry out this approach on a plant-specific and structure-specific basis.

4. Margins of Safety and Review Criteria

There are several bases upon which margins of safety may be defined and discussed. The most often used is the margin of safety based on yield strength. In this conventional use, the margin of safety reflects the reserve capacity of linearly elastic structures to withstand extra loading without experiencing an incipient permanent change of shape anywhere throughout the structure. Simultaneously, it reflects the reserve load carrying capacity existing before the structure is brought to the limit for which an engineer can be certain the computations (based on elastic behavior of the metal) apply. Because the yield strengths of common structural steels are generally well below their ultimate strengths, the engineer knows that in most (but not all) cases, the structure possesses substantial reserve capacity--beyond his computed margin--to carry additional load.

There are other useful ways, however, to speak of safety margins and these (not the conventional one) are particularly relevant to the aims of the Systematic Evaluation Program. One may speak of a margin of safety with respect to code allowable limits. This margin reflects the reserve capacity of a structure to withstand extra loading while still conforming to all criteria governing its design. One may also speak (if it is made clear in advance that this is the intended meaning) of margins of safety based upon ultimate strength, i.e., the margin to structural failure. Both steel and concrete structures exhibit much higher "margins of safety" on this second basis.

These latter concepts of "margin of safety" are very significant to the SEP review. Indeed the basic review concept, at least as it relates to structural integrity, cannot be easily defined in any quantitative manner without considering both. The SEP review is predicated on the assumption that it is unrealistic to expect that plants which were built to, and were in compliance with, older codes will still conform to current criteria in all respects. The SEP review seeks to assess whether or not plants meet the "intent" of current licensing criteria; <u>not</u> to require that older plants be brought into conformance with all present requirements to the letter. Thus, it is not expected or demanded that all structures show positive margins of safety <u>based upon code allowables</u> in meeting all current requirements; but it is demanded that margins of safety <u>based upon ultimate strength</u> are not only positive, but ample; and that the design provides the general level of safety that current licensing requirements assure.

5. <u>Methodology</u>

The approach taken in the study can be conveniently divided into six areas; each is discussed in turn.

5.1 <u>Study Documents</u>

Information sources include:

- NRC's Standard Review Plan [1] staff position papers, current and past editions of Regulatory Guides [2], and relevant NRC NUREG documents.
- Current editions of ASME, AISC, and ACI structural codes [3-7] and their commentaries, together with older versions of these codes or their counterparts.
- Each plant's Final Safety Analysis Report (FSAR), sometimes with pertinent sections of the supporting stress reports, drawings of plant layout and structures, and other information solicited from licensees.
- Technical Evaluation Reports from other SEP topics and other related documents.

5.2 <u>Identification of Applicable Codes and Criteria</u>

5.2.1 <u>Current Design Base</u>

The NRC staff is responsible for reviewing all applications for permits to construct or operate nuclear power plants and for ensuring that safety criteria are met. Safety review procedures are prescribed in NUREG-0800, Rev. 1, July 1981, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" [1], the SRP.

Individual sections of the SRP address in detail each of the subject matters to be reviewed, describe the technical basis for the review, and cite the acceptance criteria that must be satisfied. For Seismic Category I structures, SRP sections 3.8.1 through 3.8.5 describe an acceptable design basis and provide a specific, operative, and documented definition of current licensing requirements for these structures.

Because it is written to guide licensing reviews over an extended period, the SEP does not identify relevant codes and standards by revisions, date, or number. The most recently published versions apply; for purposes of the code review, published versions (as of the starting date of the assignment) were selected [3-7]. Thus, although no nuclear power plant may be actually licensed in accordance with the documents which define present licensing practice in the study, plants were compared to requirements which would be appropriate for a hypothetical nuclear power plant constructed to NUREG-800 standards in effect as of January 1981.

5.2.2 SEP Plant Design Bases

Each plant FSAR stated the codes and standards which serve as that plant's original design basis. These were identified on a structure-by-structure basis, giving credit (where applicable) in cases where more recent construction or backfitting had followed more modern requirements than those used for the balance of the plant.

5.3 Code Comparison Reviews

Each older code is paired with its modern counterpart and a paragraph-by-corresponding paragraph comparison is made.

After screening out portions which concur in technical content, the review focuses on the remaining portions and seeks to account for the intent of changes and their impact upon safety margins. In many cases these are clear. But the period covered in the study spans the introduction of basic changes in design philosophy in both the ASME and ACI codes; so the intent and implications of changes are not always obvious. Such cases are set aside for special study.

Here, a variety of analytical techniques are used, depending on the situation. One general approach is to select a basic structural element (a beam, a column, a frame, a slab, or the like) and examine its design, under both the older and the current criteria. A typical structural element and a simple loading are selected, and the element is then designed to the older code requirements. Next, the load carrying capacity of this structure is reexamined using current code criteria. Finally, the load carrying capacities of the element, as shown by older criteria and as determined by the current criteria are compared.

In making these studies, an attempt is made to use structural elements, model dimensions, and load magnitudes that are representative of actual structures. For studies that were parameterized, an attempt is made to span the parametic range encountered in nuclear structures.

Although one must be cautious about claiming that results from simplified models may be totally applicable to the more complex situations occurring in real structures, it is felt that such examples provided reasonable guidance for making rational judgments concerning the impact of changed code provisions on perceived margin of safety.

5.4 Assessment of the Potential Impact of Code Changes

The code comparisons are carried out with minimal reference to structures in the field; consequently, at this stage, no attempt is made to assess the structural adequacy of any specific structure at a particular SEP plant. Whether or not potentially adverse impacts are actually realized depends upon the existing margin of safety in the as-built structure. This determination is made in a subsequent review by the licensees.

The identified code changes are scale rated according to their potential to alter perceived margins of safety in structural elements. Three categories are designated:

Scale A Change - The new criteria have the potential to substantially degrade margins of safety as perceived under the former criteria.

Scale B change - The new criteria operate to reduce margins of safety but not enough to cause engineering concern about the adequacy of any structural element.

Scale C change - The new criteria will give rise to larger margins of safety than were exhibited under the former criteria.

In assigning scale classifications, an efficient design to original criteria is assumed. That is, it is postulated that (a) the provision in question controls design and (b) the structural member to which the code provision applies was proportioned to be at (or close to) the allowable limit. The impact scale rating is assigned accordingly.

5.5 Loads and Loading Combinations

The requirements governing loads and load combinations to be considered in the design of civil engineering structures for nuclear service have also been revised since the older nuclear power plants were constructed and licensed. Current requirements for Seismic Category I structures mandate consideration of 21 load categories and require that the mutual effects of all applicable loads in these categories be considered in appropriate combinations. Depending on the structure and the design code, 9 to 14 combinations are prescribed. Not all plant structures experience all 21 load categories. Moreover, in sparsely loaded structures, some loading combinations coalesce. Structures must be considered individually.

Accordingly, two tables are prepared for each Seismic Category I structure. One lists the currently applicable load categories, the other the currently applicable load combinations. Loads and load combinations actually considered in the design are superimposed.

Where load disparities are found (either in the categories considered or in load magnitudes assumed), scale ratings are assigned on a basis that parallels the scale ratings for design code changes.

Disparities in load combinations are also ranked. However, this ranking is based on different considerations. When disparities in load combinations are evident, it does not appear to serve any engineering purpose to require reassessment of the structure for every currently required load combination. Instead, a limited number of loading cases (typically two) associated with extreme loadings are identified for licensee review.

The considerations guiding the selection of these cases are:

1. For purposes of the SEP review, it is not believed necessary to require an extensive reanalysis of structures under all load combinations currently specified.

2. SEP plants have been in full power operation for a number of years. During this time, they have experienced a wide spectrum of operating and upset conditions. There is no evidence that major Seismic Category I structures lack integrity under these operating conditions.

3. The most severe load combinations occur under emergency and accident conditions. These are also the conditions associated with the greatest consequences to public health and safety.

4. If demonstration of structural adequacy under the most severe load combinations currently specified for emergency and accident conditions is provided, a reasonable inference can be drawn that the structure is also adequate to sustain the less severe loadings associated with less severe consequences.

5.6 Licensee Review

Preliminary study findings have been assembled by plant in ten Technical Evaluation Reports and have been forwarded to the licensee concerned. Licensees have been requested to review their plant structures in order to verify the accuracy of the findings relevant to their plants, and to evaluate the actual impacts that scale A code changes and scale A loading criteria changes (considered in combination, if appropriate) may have on individual structures within them. A number of licensees have done so and have forwarded their findings to the NRC. These findings are currently under review.

6. Preliminary Findings

The approach taken in this study appears to provide a means by which the assessment of the effects of design code changes on perceived margins of safety can be made more tractable. Typically about two dozen code changes per plant have been designated scale A and therefore require review. Some of these apply to specific structural details at a limited number of locations, e.g., newly introduced controls over shear loading of coped beams, increased development lengths prescribed for rebar anchored in regions of biaxial tension, and guidance for column web stiffeners at framed joints. Others affect more widespread structural regions and are more difficult to evaluate. Such, for example, are the provisions of Appendixes B and C of the ACI code which provide rules for the design of steel embedments, and provisions to foster ductile response and energy absorption in structures subject to impulsive or impact loadings, respectively. Table I shows a typical set of changed code provisions rated Scale A. The AISC code is taken as the example, and current (1980) code paragraph numbers are shown in parentheses.

Changes in postulated loadings are also widespread in their effect on structures. These changes occur for tornado, extreme snow, pipe break, and seismic and thermal events.

Licensees are taking a variety of approaches in making the assessments. In many cases, ample margins to accommodate the identified changes can be demonstrated. One licensee has undertaken a comprehensive structural upgrade program to increase integrity against the larger loadings now postulated, with all modifications designed to current criteria. Another is adopting a systems approach and reviewing selected structures of the safe shutdown system. Others are using an already developed probabilistic risk assessment to see where modifications would provide the greatest risk reduction.

References

1. NUREG-800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Rev. 1, July 1981.
2. NRC Regulatory Guides 1.10 (1973); 1.57 (1973); 1.61 (1973); 1.115 (1977); 1.124 (1979); 1.142 (1978); and 1.143 (1979)
3. ASME Boiler and Pressure Vessel Code, Section III, Division I, Sub-sections NE and NF; Division 2 (Subtitled ACI 359), American Society of Mechanical Engineers, 1980.
4. ACI 318, "Building Code Requirements for Reinforced Concrete," American Concrete Institute, 1977.
5. ACI 349, "Code Requirements for Nuclear Safety-related Concrete Structure," American Concrete Institute, 1980.
6. ACI 531, "Building Code Requirements for Concrete Masonry Structures," 1979.
7. AISC, "Specification for Design, Fabrication and Erection of Structural Steel for Buildings," American Institute of Steel Construction, 1980.

Table I. Typical Set of Code Changes Rated Scale A (Exemplified by the AISC Code)

Composite Construction

1. Shear connectors in composite beams (1.11.4)
2. Composite beams or girders with formed steel deck (1.11.5)
3. Width of concrete flange - limitations (1.11.1)

Compression Elements

1. Width-to-thickness ratio higher than specified in 1.9.1.2 (Appendix C)
2. Members where sidesway is not prevented (1.8.3)

Tension Members

1. Load transmitted by bolts or rivets (1.14.2.2)
2. Built-up members (1.18.3)

Members Subject to Axial and Bending Stress (1.6)

Members Designed to Operate in an Inelastic Regime

Spacing of lateral bracing (2.9)

Rolled Sections and Built-up Members

Partial length cover plates (1.10.4)

Web Plate Girders

1. Subject to shear and tension stresses (1.10.7)
2. Stiffeners (1.10.10.2)

Connections

1. Beam ends with top flange coped, subject to shear (1.5.1.2.2)
2. Moment carrying or restrained member connections (1.15.5.2 through 1.15.5.4)