Nuclear Energy Standards Coordination Collaborative

# Concrete Codes and Standards for Nuclear Power Plants:

## Recommendations for Future Development

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***Nuclear Energy Standards Coordination Collaborative***
National Institute of Standards and Technology
American National Standards Institute

# Concrete Codes and Standards for Nuclear Power Plants: Recommendations for Future Development

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# Preface

The Nuclear Energy Standards Coordination Collaborative (NESCC) is a joint initiative of the American National Standards Institute (ANSI) and the National Institute for Standards and Technology (NIST) to identify and respond to the current needs of the nuclear industry. NESCC was created in June 2009. More details on NESCC and its activities can be found on the following website:
(http://www.ansi.org/standards_activities/standards_boards_panels/nescc/overview.aspx?menuid=3 )

In December 2009, NESCC formed a task group "Concrete Codes and Standards for Nuclear Power Plants", referred to as the "Concrete Task group" (CTG) in this report. This report summarizes the recommendations for new nuclear power plants as prepared by the committee and submitted to NESCC on July 2011.

# Membership

This report was prepared by the NESCC Concrete Task Group (CTG). Membership on the CTG was open, i.e., no attempt was made to limit the membership. Efforts were made to include representatives from standards development organizations (SDO) and the construction industry who are involved in new nuclear power plant construction.

**Chair:** The Chair of the CTG was Chiara F. Ferraris, National Institute of Standards and Technology, Gaithersburg (USA)

The table below lists the organization that participated and their representatives as CTG members

| **Organization represented** (in alphabetical order) | **Name** |
|---|---|
| ACI | Matthew Senecal |
| AISC | Daniel Kaufman |
| AMEC | Corina Aldea |
| AmerenUE  Callaway Nuclear Plant | Tom Grothe |
| AREVA | Todd Oswald |
| ASME | Byran Erler |
| ASTM | Anthony Quinn & Joe Koury |
| BASF | Emmanuel Attiogbe |
| Bechtel Power | Javeed Munshi & Sanj Malushte |
| Carrasquillo Associates | Ramón L. Carrasquillo |
| Commision Nacional de Seguridad Nuclear | Pablo Ruiz Lopez |
| Consultant | Tony Fiorato |
| DOE | Tom Miller & Mathew Hutmaker |
| Dominion Virginia Power | Chuck Zalesiak |
| Duke Energy | Steven Lefler |
| EPRI | Maria Guimaraes |
| Exelon | Ronald Janowiak |
| FMC Lithium Division | David Stokes |
| ICA Fluor | Juan Carlos Santos |
| INL | Kevin Wilkins |
| J.D. Stevenson, Consulting Engineer (Member ACI-359 and ASCE-4) | John Stevenson |
| Los Alamos National Laboratory | Michael Salmon |
| NCMA | Jason Thompson |
| Purdue Univ. | Amit Varma |
| Sargent & Lundy/ Chair 359 | Arthur "Curt" Eberhardt |
| Savannah River Remediation, LLC/ Chair 349 | Ranjit Bandyopadhyay (deceased) |
| Southern Company | Partha Ghosal |
| Unistar | Jay Leininger & Olivier Mallet |
| University of Kansas | David Darwin |
| Kansas State University | J.K. Shultis |
| US-NRC | Herman Graves |
| Westinghouse | Keith Coogler |

## Reviewers (not NESCC-CTG members)

| Organization represented (in alphabetical order) | Name |
|---|---|
| Areva | Taha Al-Shawaf |
| Bechtel Power Corp. | Mukti Das |
| Ceratechnic | James Hicks |
| Clemson Univ. | Prasad Rangaraju |
| Concrete Reinforcing Steel Institute | Neal Anderson |
| CSA | Inga Hipsz |
| CTL group | Matthew D'Ambrosia |
| Euclid Chemical | William Phelan |
| Euclid Chemical Co | Philip Brandt |
| FMC Lithium Division | Claudio Manissero |
| Holcim | Al Innis |
| Hycrete | David Darwin |
| ICA Fluor | David Ortiz |
| Korea Institute of Nuclear Safety | Sang-Yun Kim |
| Nelson Stud Welding | Harry Chambers |
| NFPA | Paul May |
| NIST | Nancy McNabb |
| NRC | Jacob Philip, Madhumita Sircar |
| PCI | James Toscas |
| Politecnico di Milano | Pietro Gambarova |
| Scanscot Technology AB | Ola Jovall |
| Shaw Group | Harini Santhanam |
| Sika | Ketan Sompura |
| Structural E.I.T. | Sarah Czerniewski |
| Structural Group | Peter Emmons, Nate Sauers, Adam Brown |
| Tourney Consulting | Paul Tourney |
| US Army Corps of Engineers | Brian Green |
| Walker Restoration Consultants | Nam Shiu |
| Western Michigan Univ. | Upul Attanayake |
| Westinghouse | Michael Burke |

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# 1. Introduction

The Nuclear Energy Standards Coordination Collaborative (NESCC) is a joint initiative of the American National Standards Institute (ANSI) and the National Institute for Standards and Technology (NIST) to identify and respond to the current needs of the nuclear industry. NESCC was created in June 2009. More details on NESCC and its activities can be found on the following website:
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In December 2009, NESCC formed a task group "Concrete Codes and Standards for Nuclear Power Plants", referred to as the "Concrete Task group" (CTG) in this report. The request (Appendix A) for the formation of the task group had the following scope:

- Establish coordination and consistency of safety and non-safety related concrete requirements in nuclear power plants.
- Identify new design requirements for safety related concrete components, and develop a plan to incorporate these new requirements into codes and standards.
- Identify and review all U.S. Nuclear Regulatory Commission (NRC) Regulatory documents related to concrete for nuclear power plants. Note that this goal was too ambitious and cannot be achieved by this TG. Nevertheless, NRC is in the process to update most Regulatory Guides.

The CTG started immediately recruiting members to solicit active participation of representatives from relevant Standards Development Organizations (SDO) and concrete industry organizations involved in the construction of new nuclear power plants. At the conclusion of its work, there were 28 members representing 31 organizations including NRC and NIST.

The group met regularly by virtual meetings and face-to-face at the ACI Convention in Chicago (April 2010). Each member was asked to contribute on topics related to their expertise and to review the report. A formal vote was conducted to ensure that the concerns of all members were addressed. At least 7 ballots were conducted by e-mail. Each time a report and a ballot form were sent to members and reviewers. The comments were, as assigned by the voter, either **Primary (P)** comments identify technical issues, or **Editorial (E)** comments identify editorial issues. All CTG members were invited to comments and resolve the P comments. The Chair addressed the E comments directly. After each ballot, a new report in track changes was sent to the members to address the primary comments.

The report has three main sections to examine issues relevant to the construction of new nuclear power plants:

- Issues related to SDOs
- Issues unrelated to SDOs
- Research needs

It should be noted that this report is limited to discussion of concrete construction.

# 2. Objectives Overview

Four objectives were listed in the request for the CTG (Appendix A). A short summary is given here, along with references to appropriate sections.

1) *Review the Mattson report[1], NRC NUREG CR 5973[2], and all NRC-regulatory documents to identify all references to concrete codes and standards applied to both material and structural specifications for nuclear power plants;*

   The Mattson report and the NUREG/CR-5973 report list standards and codes referenced in NRC documents. However, they did not reflect the major gaps in the standards and codes in the nuclear industry. The objective of the NUREG/CR-5973 report was to provide a list of standards from SDOs related to nuclear facilities.

   The list of ASTM standards provided in those documents was not examined in detail, because NRC does not reference them directly in their documents. Their usage is implemented by referencing other SDO's standards[3] (codes, specifications, criteria, and guidelines). In Section 3, there is an analysis of the gaps in the concrete standards, specifications, and codes for SDO's is provided.

2) *Categorize all the concrete codes and standards that are referenced in each NRC-regulatory document.*

3) *Identify relevant concrete codes and standards missing from the NRC-regulatory documents*

   Objectives 2, and 3 were addressed simultaneously by discussing the codes and standards issued by each associated SDO (ACI, ANSI, ASCE, AISC, etc.) and regulations issued by DOE. The discussion targeted codes and standards that need to be modified. A list of issues and recommendations for each code and standard has been developed, and details are given in Section 3.

4) *Identify research needs to fill knowledge gaps in existing concrete codes and standards*

   This was addressed by drafting a list of potential areas where research might lead to updated or new standards and codes that would improve or facilitate the construction of new nuclear power plants. The results were discussed in Sections 4 and 5.

It should be noted that this report is intended to be an overall snapshot on what should be done to facilitate the construction of NPP. This Report is neither a code nor a standard, but only a set of coordinated recommendations to the SDO's involved with concrete design of nuclear power plants in hopes of "harmonizing" commonly cited concrete codes and standards. These recommendations only identify gaps, overlaps, or conflicts in existing codes and standards. In as much as there are CTG members, representing the various SDO's involved in this report, there will be ample opportunity to clarify any recommendation that is potentially

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1 "Construction Codes and Standards: Avoidance of New Nuclear Power Plant Construction Delays," Department of Energy, September 2008.

2 (NUREG/CR-5973) Codes and Standards and Other Guidance Cited in Regulatory Documents, Rev. 2, August 1994.

3 A standard, as used in the context of this report, is a document that provides a consensus-approved method or procedure to accomplish a stated objective.

unclear to a committee that is assigned to address these recommendations. It is the hope of the CTG that this document will be on the agenda of the appropriate SDO committees by the end of 2010. Individual SDO committees will need to expend the appropriate amount of time to thoroughly discuss, effectively resolve, and publish code/standard requirements that are clear, logical, and understandable.

# 3. Discussion of Standards Development Organizations (SDO) and Relevant Documents

This section discusses those SDOs that are referenced in NRC documents, or that should be referenced. A list of issues that should be resolved by the SDOs is presented and commented upon. This list can be used by the SDO to address concerns that identify potential barriers to the construction of a nuclear power plant.

To ensure that the recommendations in this report are clearly stated a uniform format was adopted. Each recommendation will be structured as much as possible in the following way:

*Title*
*a) Status today*
*b) What needs to be changed for application to a nuclear power plant?*
*c) Why does it need to be changed? Provide a rational for the change with a reference or example*

## *3.1. American Concrete Institute (ACI)*

A nuclear power plant is a facility consisting of multiple structures, the reactor building and various other structures including emergency power, control, spent fuel and nuclear waste storage and fuel handling buildings. Some of the non-safety related buildings are designed using ACI 318, and safety-related buildings use ACI 349 or 359. Because all three codes can each be implemented at one site, they should maintain a level of consistency to ensure safe, durable, and economical construction.

The American Concrete Institute (ACI) produces numerous documents related to concrete technology. ACI Committee 318 develops a code, ACI 318, that provides minimum design and construction requirements for buildings. ACI has two committees specifically related to nuclear construction: ACI Committee 349 (Concrete Nuclear Structures) and ACI Committee 359 (Concrete Components for Nuclear Reactors). ACI Committee 349 has developed “Code Requirements for Nuclear Safety-Related Concrete Structures (ACI 349-06) and Commentary.” ACI Committee 359 is a joint committee between ACI and ASME. ASME is the controlling SDO in the partnership. This joint committee has developed the “Code for Concrete Containments – Rules for Construction of Nuclear Facility Components (ACI 359-07)” which is Section III Division 2 of the *2007 ASME Boiler and Pressure Vessel Code*. ACI 349, and to a lesser extent, ACI 359 are based on modifications of ACI 318; they clearly state the circumstances where ACI 318 is not applicable.

Other ACI standards are used to design nuclear power plants and need to be maintained (see Section 3.1.2). Another issue that was identified is that ACI needs to establish training certification for technicians specifically for NPP (see section 3.1.3)

In the following subsections, specific recommendations are made related to ACI 349, 359 and 318 and other ACI documents.

### 3.1.1. Recommendations for ACI 318, 349, and 359

***Coordination needed among ACI 318, 349 and 359 documents***

It is strongly recommended that concrete design requirements and construction provisions between ACI 318, 349 and 359 should be consistent. Where construction provisions must differ due to change in safety levels, the provision, or commentary should clearly identify the change in requirement. The following list of recommendations was developed by the CTG over the course of several meetings. The scope is not to give a full developed solution but just a starting point for future implementation.

- Harmonize the minimum reinforcement for base mats.
    - a) ACI 318, 349 and 359 have different requirements that apply to base mats. ACI 318 requires a minimum reinforcement area fraction of 0.0018 that is considered excessive for the very thick base mats used in nuclear construction.
    - b) Rather than a flat number (that applies to all thicknesses), this should be based on better scientific rationale including ACI 207 for mass concrete. If strength does not control reinforcement in a mat foundation in a NPP, then minimum reinforcing requirements for temperature and shrinkage control need to be specified for serviceability. Currently, designers are invoking the slab reinforcing ratio from ACI 318 (i.e. $\rho = 0.0018$), which can produce excessive amounts of reinforcing for thicker mats.
    - c) In order to make the reinforcement more reasonable for thick mats and reduce unnecessary congestion. Serviceability reinforcement ratio minimum requirements for mats in NPP structures would be helpful and promote more consistent designs. ACI 359 and ACI 349 members made a deliberate decision to use the allowable strain criterion for the design of concrete containment structures—the most important structures among nuclear power plant structures that require leak tightness—and the ultimate strength criterion for less important structures. That deliberate decision is logical and well reasoned with respect to safety because the ultimate strength criterion does not provide information on strain values in steel reinforcement and crack width in concrete sections while the strain criterion does. However, for other concrete mat structures, where a leak-tight composite steel liner is not involved, alternate rationale for minimum reinforcement requirements would be an improvement to the design minimum reinforcement requirements currently listed in ACI 318 and ACI 349. Such alternate rationale for minimum reinforcement requirements should consider both strength and serviceability of the mat under all required load combinations. Nevertheless, the Committee agrees that further study will be required prior to recommending elimination or replacement of current ACI 359 design criteria, and the Committee does not have any intent to implement a less conservative criterion unless sufficient study/research is available to demonstrate that an adequate level of safety will be maintained. Adequate criteria must be provided to ensure that the steel lined containment will maintain the required leak tightness. There is no

intent to change or eliminate existing test requirements that ensure the leak tightness and structure integrity of the containment. Design provisions will continue to assure that strain limits are evaluated and controlled to ensure the serviceability of the containment. In short, additional study is required to ensure that all of the above concerns are adequately addresses prior to implementing any significant change to current design criteria.

- Coordinate Section 9.2 of ACI 349, evaluating other possible load combinations, with the AISC N690-06 load combinations for possible adoption in the next revision of ACI 349.

    a) This comment deals with the fact that the load factors for Severe Environmental, Extreme Environmental and Abnormal load combinations in ACI 349, AISC N690 and the Standard Review Plan in NUREG-0800 are not in agreement. AISC N690 load combinations are clearly divided into 3 categories; Normal Loads (i.e. loads which have a high probability of being present during any one year of the design life of the facility), Severe Loads (which include those environmental loads that have a $10^{-2}$/year probability of exceedence) and Extreme Loads (which have loads, both environmental and accidental, having a very low probability of exceedence equal to or less than $10^{-4}$/year).

    b) The demand side of the code (i.e. load cases, load factors, load combinations) needs to be consistent across all of these design codes for NPP's, since demand is not related to the construction material.

    c) It is a reasonable expectation for a code user to be able to read consistent demand side requirements in ACI 349 or AISC N690, and hopefully someday those load combinations will be consistent with the Standard Review Plan in NUREG-0800. Coordination of demand side requirements across ACI 349 and AISC N690 is a great starting place. See Appendix B of this CTG report. The AISC N690 suggested three load categories should be applied also to ACI-349 and include the changes contained in NRC R.G. 1.142 as shown in Table B-2.

- Inconsistencies between the three documents for seismic design

    a) ACI 318/349 have separate shear provisions for walls in Chapter 11 for non-seismic and Chapter 21 for seismic loads. The two sets of procedures give different capacities for the wall with the same reinforcement.

    *b)* Recommend that two equations (ACI 318/349 Equations (11-5) and (21-7)) and the equation for shear walls in ASCE 43 (Eq. 4-3) be revised for consistency. Recommend establishing consistency, where possible, among the load cases, load factors, and load combinations listed in ACI 349 and 359. **(***see suggestion in Appendix B.)* This recommendation will require additional research to implement and will become a long-term activity. In the short-term, CTG recommends that ACI Committee 359 identify the research needed for ASME-ACI to move to a strength design for reinforced concrete containment design, similar to the current ACI design methodology.

    *c)* Because with new plant concepts involve a common base mat, it is difficult to design the containment for one set of loading and design parameters based on service load while the rest of the CAT I structures sitting on the same base mat are designed for Strength level design per ACI 349. It creates a lot of interface issues, design issues, and unnecessary additional computation involving hundreds of load combinations.

1. Design provisions for anchorage
    a) The design provisions for anchorage to concrete are almost identical in ACI 318 and ACI 349. The design provisions for anchorage to concrete in ACI 359 are not well developed and often end up defaulting to ACI 349.
    b) Recommend that the design provisions for anchorage to concrete in ACI 318 and ACI 349 be harmonized to the extent possible. ACI 359 should conform to the provisions in ACI 349.
    c) The provisions in ACI 349 are similar to those in ACI 318. ACI 359 on the other hand does not have any provisions at all. Harmonizing design provisions for anchorage to concrete across ACI 318, ACI 349 and ACI 359, wherever possible, would promote commonality, reduce errors, thereby, simplify design and construction.
2. Headed bars
    a) ACI 318-08 in section 12.6 allows headed bars which have proven to be very successful in reducing congestion in nuclear construction in Europe.
    b) It is recommended that ACI 349 and ACI 359 consider this change as well.
    c) It would simplify construction and significantly reduce congestions around anchorage points which is a real concern for ACI 349 and ACI 359 structures. It would also allow headed bars to be used in nuclear structures. Not doing so prevents the application of this widely used technology.
3. Coordination of updating schedule ACI 318, ACI 349, and ACI 359
    a) Due to different updating schedule, the dates (versions) of referenced standards within each document differ.
    b) This means that standards for materials specifications and test methods can differ between the documents (although aimed at the same materials), and can result in different requirements for materials between say the containment structure and other safety-related structures. An effort should be made to coordinate the version of referenced standards that may be invoked on a site. This may be difficult to do in the individual codes (ACI 318, 349, and 359) because of differences in update cycles, but should be addressed in procedures for construction specifications.
    c) For example, ACI 349-06 references ASTM C150-04a for Portland cement while ACI 359-07 references ASTM C150-89.
4. Durability and construction provisions in ACI 349 and ACI 359
    a) The provisions in the two documents have similar, but not identical requirements.
    b) There is no reason to have different significant figures specified that can result in misunderstandings or disputes. An effort should be made to harmonize these unnecessary differences in requirements.

c) For example, ACI 349-06 permits air content to be reduced by 1.0 % for concrete strengths above 5000 psi. ACI 359-07 permits air content to be reduced by 1 % for all concretes. There are numerous other examples. ACI should establish a coordination committee that would facilitate harmonization and synchronization.

### *ACI 349: Concrete Nuclear Structures*

In this section, more specific recommendations are given relative to ACI 349:

5. ACI 349 seismic Category I structures

   a) Currently, ACI 349-06, Chapter 21, is written in a dependent code format, which contains more exceptions than similarities from the seismic reinforcing detailing requirements listed in ACI 318, Chapter 21. This format and approach to code writing is cumbersome for the code user.

   b) Due to the very different seismic performance requirements for concrete structures designed in accordance with ACI 318 versus that of ACI 349 (i.e. inelastic versus elastic), it is recommended that ACI 349, Chapter 21, be rewritten as an independent, standalone chapter from ACI 318, Chapter 21.

   c) Seismic reinforcing detail requirements in ACI 349, Chapter 21 should be independent in format and technical bases from those listed in ACI 318, Chapter 21

6. Concrete mixes allowed

   a) Although there is much data supporting the quality of commercially available concrete mixes, these concrete mixes are not easily approved for use in NPP's due to the different design requirements that are currently specified for concrete mixes to be used in NPP's. It is believed by the concrete specifiers that many of the commercially available concrete mixes would perform satisfactorily in NPP applications, thereby simplifying the specification requirements and reducing overall cost.

   b) Recommend that concrete design requirements be simplified as much as possible to permit commercially, locally available concrete mixes to be used. Impose only those concrete design requirements (beyond those of ACI 318) necessary to ensure the more stringent quality assurance requirements associated with nuclear safety related construction.

   c) ACI 318 and ACI 301 requirements should be sufficient for mixture designs, but provisions for thermal control and alkali-silica reactivity should be included for nuclear structures.

7. Thermal effect

   a) Experience in structural behavior of concrete building structures subjected to normal environmental temperature ranges (i.e. $120^0$F to $-20^0$F) with erection temperatures between $80^0$F to $35^0$F exhibit no damage. The stresses induced by these temperature differentials are secondary (deformation limited) in nature. Consistent with metal piping and vessel design flexibility analysis which include thermal effects analysis are generally not required for design temperatures below

150 $^0$F. Adding the secondary normal differential temperature induced stresses to primary stress resultants in concrete structures is an unnecessary and incorrect over conservatism because their failure modes are different. Failure due to secondary loads is cyclic induced fatigue or ratcheting while primary load failure is caused by plastic instability. In general, fatigue ratcheting induced failures do not occur unless there are a significant number of cycles present (> 25 equivalent full stress cycles beyond 2 times yield stress). Fatigue failure for cyclic stresses in the less than 2 times yield stress range would require 1000's of cycles. Currently there is no minimum threshold or any discussion of an acceptable minimum threshold for thermal gradients, ΔT, or mean temperature rises, Tmean, in reinforced concrete structures under ACI 349. A rational discussion of minimally acceptable thermal gradients, ΔT, and mean temperature rises, $T_{mean}$, has been provided in ACI 349.1R-07. Under these recommended provisions, $T_{mean}$ < 50 °F and ΔT < 100 °F, would be acceptable in NPP structures without further evaluation.

b) Recommend that the committee consider moving thermal effect recommendations from ACI 349.1R-07 into ACI 349, *Appendix E,* i.e. limits for acceptance without further temperature evaluation on ΔT = 100 °F. See Appendix C for further discussion on this issue and how ACI 349 has responded.

c) To clarify to both licensee and the NRC that it is OK to do this - otherwise the language is considered to be weak and raises a lot of questions needing supporting documentation. Under the recommended provisions of ACI 349.1R-07, $T_{mean}$ < 50 °F and ΔT < 100 °F, would be acceptable in NPP structures without further evaluation, simplifying the evaluation of minor thermal effects.

d) Starter mixes design in congested areas

   a) The starter mix design is not allowed in ACI 318 and ACI 359/ASME.

   b) ACI Committee 301, instead of ACI Committee 349 (section 5.10.9), should determine the starter mix design in situations of congestion. Recommend that ACI 349 remove starter mix design provision and refer to ACI 301.

   c) Specification of a starter mix is a material specification requirement, which is better aligned with the scope of ACI 301 versus ACI 349. Consolidating material specification requirements into ACI 301 prevents unnecessary duplication of material requirements and reduces the potential for conflicting material specifications in ACI 349.

e) Design Requirements for Seismic Category I

   a) No design requirements currently exist for Seismic Category I liquid containing structures.

   b) It was recommended that a Task Group from ACI Committee 349 be established to coordinate with Committee 350 members to develop such design requirements in ACI 349.

   c) Spent fuel pools are the major example of a Seismic Category I liquid containing structures in a NPP. ACI 350 provides a rational methodology for calculating impulsive and convective hydrodynamic force components due to the sloshing effect of the liquid. However, the seismic requirements that have been incorporated into ACI 350 come from ASCE 7, not US NRC Regulatory Guidance, which is required of spent fuel pools in NPP's. Therefore, it would be helpful to have seismic design requirements, consistent with US NRC Regulatory Guidance, and industry accepted hydrodynamic force calculation methodologies, incorporated into ACI 349 to form a consistent design basis for Seismic Category I liquid containing structures for NPP's.

### *ACI 359: Concrete Components for Nuclear Reactors*

In this section, more specific recommendations are given relative to ACI Committee 359:

- Containment liner erection tolerances
  - a) Containment liner erection tolerances are very generous.
  - b) ASME Section III, Division 1, NE-4221.1; Division 2, CC-4522.1.1 and non-mandatory Appendix D needs to be revisited.
  - c) Since the steel liner and reinforced concrete of NPP containments form a composite structure, their construction tolerances need to be coordinated, with consideration given to the design of both the steel liner and reinforced concrete shell.
- Design for impulse and impactive loads
  - a) ACI 359 does not provide much guidance for design for impulse and impactive loads. The relevant NRC document is NRC DG 1176, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts"
  - b) Recommend that ACI 359 consider adopting or referencing ACI 349 provisions with consideration that ACI 359 containment structures are required to be leak tight under both design basis accident and normal pressure conditions, however; ACI 349 structures are not.
  - c) Since the demand side requirements for impulsive and impactive loads in ACI 349 and ACI 359 ought to be consistent with the Standard Review Plan in NUREG-0800 (i.e. see Comment No. 2), the provisions of ACI 349, Appendix F could be adopted for ACI 359 containment structures, with the proviso that leak tight integrity be maintained. Again, it is a reasonable expectation for a code user to be able to read consistent demand side requirements in ACI 349 or ACI 359, and hopefully someday those load combinations will be consistent with the Standard Review Plan in NUREG-0800. Coordination of demand side requirements across ACI 349 and ACI 359 is a great starting place. See Appendix B of this CTG report.

### *Coordination between 349 and 359*

- Consistency between ACI 359 and 349
  - a) The analysis, load combinations and design provisions of ACI 359 are very different from those in ACI 318 or 349 because of the design philosophy of

containment structures and the fact that non-prestressed reactor containment cylinders and domes are in biaxial tension. In ACI 359 there is a strong preference for the use of working stress design, since containment structures are designed to remain elastic except for local response to impact loads to minimize cracking under design conditions that do not involve thermal accident conditions.

b) Recommend that ACI 359 consider updating Section CC 3000 of ACI 359 to be, consistent with analysis procedure, design for strength and serviceability, flexure, shear, torsion, development, prestressed design, detailing in Chapters 8 – 21 of ACI 318 and ACI 349. This information should be consistent with NRC RG 1.142, Rev 2 and R.G. 1.136. However, there are other opportunities:
    - Harmonization of some of the basic design provisions between ACI 349 and ACI 359 (ASME B&PVC Section III, Subsection CC3000…)
    - Add References in CC 3000, for example for detailing and anchorage provisions of ACI 318/349. However, some of the seismic detailing provisions of ACI 318 may not be applicable since containment structures are required to remain elastic under seismic conditions and ACI-359 design may include concrete sections in biaxial tension not typically applicable in ACI-349 structures.

c) It does not serve well to have different loads, design methods and provisions to design concrete in different structures (containment vs. other CAT I). It compounds the effort at various levels for no obvious reason or benefit. It is time to revise and rationalize the design across various Codes. Harmonizing design provisions for concrete structures, where applicable, across ACI 318, ACI 349 and ACI 359, would promote commonality, reduce errors, thereby, simplifying design and construction.

- Anchorage development and splicing reinforcement,

  a) Very different quality control requirements currently exist for anchorage development and splicing reinforcement in ACI 349 versus ACI 359, yet similar performance criteria is desired. It would seem reasonable that the quality control requirements could be applied to anchorage development and splice reinforcement in both ACI 349 and ACI 359.

  b) It is recommended that the quality control requirements be consistent between ACI 349 and 359.

  c) Consistency between quality control requirements in ACI 349 and ACI 359 will permit more uniform site procedures that can minimize errors

- Revision of Section 359- CC-3440

  a) Identical temperature limits exist in ACI 349, Appendix E and ACI 359, Section 3440 for long-term, general and local area, as well as for short-term, general and local areas (i.e. long-term, 150 °F and 200 °F and short-term, 350 °F and 650 °F).

  b) Recommend that the ACI Committee 359 consider revising Section CC-3440 to comply with proposed changes in *ACI 349, Appendix E Code and Commentary*.

  c) See ACI 349, Appendix E

### 3.1.2. Recommendations for Other ACI Documents

***Reinforcing Steel Erection Tolerances***

ACI-TAC organized a subcommittee in 2004 to eliminate tolerance differences between ACI Committee Documents, such as ACI 117, 301, 318, and 349. The subcommittee completed this task in 2008 (TAC Fall 2008 Minutes). At that time, there were no known tolerance conflicts within concrete construction as reported by ACI. It should be noted that tolerances between concrete and other materials or products have not been resolved.

***ACI 351 - Foundations for Equipment and Machinery***

a) ACI 318 provisions were developed for regular commercial/residential building type structures. Issues arise related to design philosophy, detailing and minimum reinforcement for temperature and shrinkage, shear and torsion. ACI 349, because it is based on ACI 318, has similar issues.

b) Recommend that ACI Committee 351 publish examples in a design guide to address the design of oversize sections encountered in industrial structures for example thick mats, generator pedestals, etc.

c) There is no guidance currently available on how to design oversize sections. In the absence of this info, users default to ACI 318 which is for regular sized members of buildings. Applying ACI 318 provisions to oversize members creates a whole lot of inconsistencies and results in very conservative designs. Foundations for equipment and machinery have special vibration concerns that are not adequately addressed in ACI 318 and ACI 349. It would be helpful for a designer to refer to design examples, performed in ACI 351, to address similar foundation and equipment concerns for NPP's.

***ACI 447 – Joint ASCE - Finite Element Analysis of Reinforced Concrete Structures***

Recommend that ACI Committee 447 develop guidelines on the post-processing and interpretation of finite element output. Nuclear structures are analyzed using very sophisticated and elaborate finite elements (FE) models (ANSYS, GTSTRUDL, SAP 2000, etc[3]…) involve very detailed output at the element level. No guidelines are generally available on how to process and interpret the post-processing of finite element output (i.e. averaging, smoothing, neglecting, etc…) for NPP structures. Engineers are exercising judgment, without industry guidance, to determine how output should be post-processed to best reflect structural behavior, whether it be global behavior or at the member level.

The draft ASCE 4 Standard contains rules for finite element modeling of concrete which should be considered for inclusion in a document developed by ACI Committee 447 as the appropriate ACI Committee to address this issue. ACI 447 should be able to implement guidelines that can be used on an industry wide basis.

[3] Commercial equipment, instruments, and materials mentioned in this report are identified to foster understanding. Such identification does not imply recommendation or endorsement by the NESCC, nor does it imply that the materials or equipment identified are necessarily the best available for the purpose

A comment could be made to this statement: This recommendation specifically applies to design of shearwalls which can be found in abundance in nuclear world. Unfortunately, the design of these walls is based on equations derived from testing of cantilever wall panels. But when we have a finite element model with thousands of elements and need to interpret that element level data to meet the above equation/code intent, there is a problem. It is more of an issue in nuclear world as they rely on more elaborate models with very fine mesh/element sizes which is expected in this industry. Although, this guideline would be of general use, the real beneficiary would be the nuclear industry.

***ACI 301 - Specifications for Concrete***

a) ACI 301 is generally included in the project specifications,

b) Recommend harmonizing redundant and/or conflicting design information in ACI 301, ACI 318, ACI 349, and ACI 359. ACI 301, ACI 349, and ACI 359 should review relevant requirements and make recommendations where possible to improve consistency between the documents.

c) Greater consistency and uniformity of concrete construction specifications will minimize inadvertent errors because suppliers and contractors will not have to account for as many differences in requirements.

***ACI 311 – Inspection of Concrete***

It should also be noted that ACI 311 (“Guide for Concrete Inspection Specification for Ready-Mixed Concrete Testing Services”) may not apply to nuclear applications. Recommend that a specific guide on this topic be developed specifically for nuclear power plants and other nuclear facilities.

***Heavyweight Concrete in ACI Documents***

Nuclear power plants contain very thick slabs and walls used as radiation shielding. Therefore, guides, codes and specification related to the design, proportioning, placing, heavyweight concrete should be available. The following outdated documents should be revised as soon as possible or they risk to be considered historical by ACI and archived, especially if they were not re-approved recently.

- ACI 211.1-91: Standard Practice for Selecting Proportions for Normal, Heavyweight and Mass Concrete (Reapproved 2009).
- ACI 304.1R-92: Guide for the Use of Preplaced Aggregate Concrete for Structural and Mass Concrete Applications (Reapproved 2005).
- ACI 304.3R-96: High Density Concrete: Measuring, Mixing, Transporting, and Placing

Note: Often, high density or heavyweight concrete is employed simply by using high density aggregate in ordinary mix designs.

### 3.1.3. Status of Nuclear Inspector Certification Program

Pre-2006 Program Structure—Level I and II Inspectors must meet the requirements of Appendix V, Section III, Division II of the ASME B&PVC. The Level III Inspectors are responsible for selection, training, evaluation, and qualification of all Level I and II Inspectors.

The Level III Inspectors must meet the requirements of Appendix V, including the successful completion of ACI's Level III Concrete Inspector examination. Level III Inspectors are certified by their employer. All Level III Nuclear Inspectors on file with ACI are qualified under the following structure.

- Education/Work Experience – ACI reviews applications for compliance with prerequisites.
- Concrete Technology Exam – ACI administers and grades a 4-hour exam maintained by an ad hoc committee from the concrete and nuclear construction industry and based on the ACI Manual of Concrete Practice [in 3 Parts pre-1980 (2400 pp); expanded to 5 Parts in 1980 (2900 pp); currently 7 Parts (7230 pp)].
- ASME/ACI 359 Code Exam – ACI administers and grades a 4-hour exam maintained by an ad hoc committee from the concrete and nuclear construction industry and based on the ASME/ACI 359 Boiler and Pressure Vessel Code.
- ACI issues a letter to successful candidate's employer informing that prerequisites have been met and the exams passed. If the candidate changes employer, ACI sends a letter upon request to new employer to facilitate certification by new employer. Eligibility for certification is essentially for life with no retesting required.

Post-2006 Program Structure—Changes were made in the ASME Boiler and Pressure Vessel Code 2006 Addendum, but not known if it has been used to qualify any Level III Inspectors, as this is at the discretion of the N-Stamp holder.

- Individuals who have not been employed as a Level III Inspector during the last 5 years prior to seeking certification are required to retest to reinstate their Level III certification.
- Requirements for ACI's Level III Concrete Inspector examination revised to reflect available programs. Level III candidate must be certified as ACI Concrete Construction Special Inspector (which requires certification as ACI Concrete Field Testing Technician – Grade I); approximately 1250 pages of technical information covered by these certification programs.
- Employer is now responsible for:
    - the creation, administration and grading of exams covering the ASME Code and relevant quality assurance aspects of a nuclear facility; and
    - review of applicants qualifications for education prerequisites.

Potential 2011 Program Structure—Qualifications are restructured to take advantage of existing commercial programs and their maintenance infrastructure.

- ACI Concrete Field Testing Technician – Grade I + Nuclear rider = Old ASME Level I
- ACI Concrete Construction Special Inspector (CCSI) + Nuclear rider = Old ASME Level II
- ACI Nuclear CCSI (as described above) + Concrete Quality Technical
- Manager (CQTM: under development) + Nuclear rider = Old ASME Level III

Restructuring in this manner would provide:

- More consistent/standard core qualifications
- More rapid development and launch of needed programs

- Opportunity to assess/incorporate new technology/standards into the qualification system in an orderly (psychometrically sound) fashion
- A mechanism for consistent/formal reassessment/retesting of inspection personnel at all levels
- Parameters for development of standardized training for core qualifications

Inspection at the Job Site— In addition, there are at least three categories of structures on Nuclear Power Plant sites in which concrete is used:

1. Safety-related structures such as the containment vessel governed by ASME Sec. III, Div. 2/ACI 359
2. Nuclear safety-related structures other than the containment vessel governed by ACI 349
3. Commercial structures governed by ACI 318

It is unclear how the N-Stamp holder accomplishes the inspection of these types of structures. ASME Sec. III, Div. 2 explicitly addresses the responsibilities of the various level inspectors, whereas ACI 349 and ACI 318 do not. The requirements for inspection should be addressed for all three building types and harmonized.

## *3.2. American Institute of Steel Construction (AISC)*

Modular construction consisting in building walls or slabs off site and then transporting them to the construction site can in some instances reduce construction time and cost. Steel Composite shear walls (SC) are particularly applicable for the purpose but it seems that no design requirements for such walls currently exist in AISC N360 Specification or any other published SDO document.

Some progress is being made as consistent design requirements for concrete composite compression members, as currently listed in ACI 349-06, Section 10.16 (i.e. safety related) and ACI 318-08, Section 10.13 (i.e. non-safety related), with requirements, are currently published in AISC/ANSI 360-05 Chapter I and Chapter NI of AISC N690-06. AISC Committee TC 12 is already leading an effort. A subcommittee of this committee has made significant progress in developing standards for modular construction.

This is an area in which AISC has done as the NESCC CTG recommends. Before forming the AISC sub-committee on modular composite construction, the options on how to coordinate with ACI were carefully considered.

- A first draft was provided to the subcommittee members in late September 2010.
- The final document, with comments from the AISC main committee addressed, should be ready in mid to late 2012 (after more ballot cycles with the main committee).
- Formal publication from AISC should occur in late 2012.

As an alternative to the loads and load combinations specified in ACI and AISC Standards, consider asking ASCE to expand ASCE 43-05 to address all natural hazard loads for nuclear safety-related structures in addition to seismic

## 3.3. Coordination ACI and ASME

The American Society of Mechanical Engineers (ASME) has a joint committee with ACI (ACI 359). Therefore, the mechanisms for coordination are in place. The following recommendations are given:

1. Strength Design (ACI) vs. Service/Factored Load Design (ASME) requirements
   a. ASME/ACI 359 reinforced concrete containments and their composite steel liners are designed according to Service/Factored Load (ASME) requirements. ACI 349 structures are designed according to strength design (ACI) requirements. The two (2) methodologies are very different and do not align well to make comparisons for margins of safety from one methodology to the other.
   b. Recommend an evaluation of strength design methodology for all new containment structure design. This recommendation will require additional research to implement and will become a long-term recommendation. In the short-term, CTG recommends that ACI Committee 359 identify the research needed for ASME to move to an ultimate strength design methodology for concrete reactor containment design, similar to current ACI-318 and 349 design methodology. ASME adopted ACI 318 in 1970, and it should be updated to take advantage of technical evolution in the concrete design. Concrete containments are not special structure to warrant to keep the technology and know how of 1970.
   c. The composite steel liner in NPP containments transfers internal pressures to the outer reinforced concrete containment shell. Design concerns for liner strain can be met by calculating reinforced concrete shell displacements. From that point on, there is little difference in behavior between the reinforced concrete containment shell and reinforced concrete structures in ACI 349. Evolving towards a strength design basis would promote commonality and simplify the design requirements for reinforced concrete structures in NPP's.

## 3.4. American National Standards Institute/American Nuclear Society (ANSI/ANS)

ANSI facilitates the development of American National Standards Institute (ANSI) by accrediting the procedures of standards development organizations (SDOs). The following standards referenced by NRC are still current and applicable to the next generation of nuclear power plant construction. ANSI through their respective SDO's must ensure that they are updated on a regular basis.

- ANSI/ANS N18.2: Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants
- ANSI/ANS N18.7: Administrative Controls and Quality Assurance for the Operation Phase of Nuclear Power Plants
- ANSI/ANS 56.8: Containment System Leakage Testing Requirements
- ANSI N170: Standards for Determining Design Basis Flooding at Power Reactor Sites
- ANSI/ANS 2.2: Earthquake Instrumentation Criteria for Nuclear Power Plants

- ANSI/ANS 2.3: Standard for Estimating Tornado and Extreme Wind Characteristics at Nuclear Power Sites
- ANSI/ANS 2.7 Guidelines for Assessing Capability for Surface Faulting at Nuclear Power Sites
- ANSI/ANS 2.8 Determining Design Basis Flooding at Power Reactor Sites
- ANSI/ANS 2.10 Guidelines for Retrieval, Review, Processing and Evaluation of Records Obtained from Seismic Instrumentation
- ANSI/ANS 2.11 Guidelines for Evaluating Site-Related Geotechnical Parameters at Nuclear Power Sites
- ANSI/ANS 2.12Guidelines for Combining Natural and External Man-Made Hazards at Power Reactor Sites
- ANSI/ANS 2.26 Categorization of Nuclear Facility Structures, Systems, and Components for Seismic Design
- ANSI/ANS--6.4 Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants
- ANSI/ANS--6.4.2 Specification for Radiation Shielding Materials

The most current standard for "Quality Assurance Program Requirements for Nuclear Power Plants" and other nuclear facilities would be NQA-1, ISO 9001, or DOE 10 CFR830.120, as applicable and not ANSI N45.2.

ANS should accelerate the preparation of the new standard (ANS 58.16 applicable to safety classification of nuclear safety related structures, systems and components for other than nuclear reactors) from the expected 2012.

### 3.5. American Society of Civil Engineers (ASCE)

ASCE should ensure that all its standards and codes are updated in a timely fashion. The following were targeted as especially relevant to the new nuclear power plant construction (in parenthesis the year of publication):

| | |
|---|---|
| ASCE 4-98 (2000) | Seismic Analysis of Safety-Related Nuclear Structures |
| ASCE 43-05 (2005) | Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities. |

ASCE 7-10 and 43-05 use "Seismic Design Category (SDC)" but ACI 349 does not (Chapter 21). All three standards should be reviewed and consistent terminology for safety-related structures should be developed.

### 3.6. Electric Power Research Institute (EPRI)

There are, to this date, no guidelines on concrete construction for nuclear facilities from EPRI that need to be revised or updated.

NOTE: EPRI is not a Standards Development Organization.

## 3.7. Nuclear Energy Institute, NEI

There are, to this date, no guidelines on concrete construction for nuclear facilities from NEI that need to be revised or updated.

NOTE: NEI provides guidance documents associated with NPP construction, but is not a Standards Development Organization.

## 3.8. National Fire Protection Association (NFPA)

An authoritative source on public safety, the NFPA develops, publishes, and disseminates more than 300 consensus codes and standards intended to minimize the possibility and effects of fire and other risks. The following NFPA standards as they relate to concrete should be maintained as they are relevant to the next generation of nuclear power plants.

- NFPA 253 Standard Method of Test for Critical Radiant Flux of Floor Covering Systems Using a Radiant Heat Source (also ASTM E119). A new edition of NFPA 253 was released in 2011.

- NFPA 259 Standard Test Method for Potential Heat of Building Material. A new edition of NFPA 259 will be released in 2013

- NFPA 251 Standard Methods of Tests of Fire Resistance of Building Construction and Materials. This standard is scheduled for withdraw as of June 2011, when the final approval to withdraw the document will be determined (until final decision by the committee). Presently, the committee recommends using ASTM E119, *Standard Test Methods for Fire Tests of Building Construction and Materials*, and ANSI/UL 263, *Fire Tests of Building Construction and Materials*.

- NFPA 255 Standard Method of Test of Surface Burning Characteristics of Building Materials. NFPA 255 has been withdrawn, with the last edition having been released in 2006. ASTM E84, *Standard Test Method of Surface Burning Characteristics of Building Materials*, and ANSI/UL 723, *Standard for Safety Test for Surface Burning Characteristics of Building Materials* have been determined by a review task group to be the appropriate standards for replacement of NFPA 255.

NRC should update its regulations and regulatory guidance to ensure that the most recently referenced standards are cited.

## 3.9. Coordination DOE, NRC and SDOs

The Department of Energy (DOE), the Nuclear Regulatory Commission (NRC), ACI, and ASCE need to coordinate requirements in the standards, codes and specifications used in construction[4] of nuclear facilities. The following four (4) major issues were identified:

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[4] Construction includes the sub-activities of documentation, material selection, material qualification, design, fabrication, placement, erection, examination, inspection, and testing.

- Design requirements
    a. Gaps and conflicts in design requirements exist in some of the cited DOE and NRC standards or guidelines particularly as they relate to Design Basis Environmental Loads
    b. For example, DOE uses a graded approach based on probability of environmental load exceedence in ANS 2.26 for earthquake for Seismic Design Categories SDC-4 and SDC-3 while the NRC in regulatory guide RG 1.143 defines the seismic load as one-half the SSE for Safety Class RWIIa SSC which is equivalent to SDC-3.
    c. Non uniformity of the design will create confusion.

- Design Basis Accident
    a. A number of specific Design Basis Accidents to be considered in the design of a nuclear facility are identified in U.S. NRC Standard Review Plan and Regulatory Guides, one of which is the postulated break of a high energy ($P_o$ >250 psi or $T_o$ > 200 °F) pipe loadings for which are specified in ACI 349 as Y type loads. On the other hand, with the possible exception of a heavy load drop, there are no specific Design Basis Loads identified in DOE Standards (i.e. DOE Std. 1189, Integration of Safety Into the Design Process). However, specific reference in DOE 1189 is made to DOE Std. 1020 which does reference ACI-349 for design of performance categories PC-3 and PC-4 structures as defined in DOE Std. 1021. While process piping in DOE facilities is generally not high pressure, they do exist and there does not appear to be an application of the Y loads of ACI 349 to DOE regulated structures having high pressure piping as well as any of the other accident loads identified by the U.S. NRC with the exception of "heavy load drop."
    b. It is recommended that ACI Committee 349, DOE and the NRC coordinate and incorporate the Design Basis Accident events at least for "heavy load drop," "high energy pipe break," and interior flooding of structures.
    c. The demand side requirements for NPP structures ought to be consistent with the Standard Review Plan in NUREG-0800 (i.e. see Comment No. 2). Therefore, it is reasonable to coordinate as many Design Basis Accident events for NPP structures to avoid unnecessary code requirements. Coordination of demand side requirements across ACI 349, DOE and the US NRC is a great starting place. See Appendix B of this report.

- High Density of reinforcing rebars
    a. Next generation nuclear power plant construction is expected to be a challenge from a constructability point of view due to the high density of reinforcing rebars that render consolidation of concrete arduous.
    b. Ways to mitigate this situation where standardization is necessary are: (1) use of higher grade reinforcing steel (thus reducing the amount of reinforcing rebars needed) and (2) self consolidating concrete (thus mitigating the problem with consolidation by vibration). ACI guidelines would be necessary.
    c. Improper consolidation of concrete in areas of congested reinforcement can result in honeycombing and voids. Vibration and consolidation procedures need to be carefully established as they are critical to ensure a high quality concrete.

- Review of NRC and DOE
    a. NRC and DOE need to review any new version of SDOs documents before they are accepted to be used in a nuclear power plant constructions
    b. Expedite NRC's and DOE's review of the most widely used SDO's codes and standards to identify and hopefully resolve those construction requirements that are in conflict with the NRC current technical positions. An alternative of having official representation by NRC and DOE in the SDOs (and thus getting the SDOs adopt their input) and accepting the industry-consensus standards without amendment would be a great improvement
    c. It is inefficient for SDO's to continually revise their codes and standards with new design requirements that are not endorsed by the NRC or DOE..

# 4. Issues Unrelated to SDOs

In this section, road blocks for the construction of new generation of nuclear power plants are examined. This is not considered an exhaustive list.

## 4.1. Materials

For each new nuclear power plant (NPP), approximately 100,000 $m^3$ (130,000 cubic yards) of concrete will be needed, on average[5]. This fact implies that material selection for concrete mixture designs needs to ensure conformity to current standards and codes and that a sufficient material supply is locally available. As quality control in construction is vital, it is advisable to utilize, as much as possible, commercially available concrete batch mix materials, for which adequate records are maintained and engineering experience is available. The choice of materials to enable assured concrete service life is also important, as nuclear power plants will have an expected service life, including power production and shutdown, of on the order of 60 – 75 years.

The composition of concrete has greatly changed from the 1980s when the last power plant in the U.S. was constructed. Therefore, the following issues should be considered:

- Cements are much finer, their bulk chemistry and mineral characteristics have changed, and their interaction with supplementary cementitious materials (SCM) must be taken quantitatively into account
- Supplementary cementitious material (SCM) usage should be strongly encouraged, especially for mass concrete, in order to enhance the concrete performance characteristics. SCM characterization is paramount to ensure that it provides anticipated durability and early age properties (see also section 5.3). Nuclear codes should not introduce arbitrary barriers to SCM use.
- A clear relationship between aggregate mineral morphology and ASR should be developed[6] (see Section 5.3). The problem is not new, as discussed in 1972[7]. Aggregate sources should be tested to avoid potential alkali-silica reaction (ASR), which may be exacerbated due to the temperature, moisture, and radiation conditions found in nuclear power plants. It should be noted that relying solely on a supply of truly inert aggregate to prevent the issue is not without risk, will often be uneconomical in some geographic regions, and should be supplemented with preventive measures in the binder fraction. Recent developments in evaluation of potential for ASR and specification guidance should be considered[8,9], recognizing

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5 Per F. Peterson, Haihua Zhao, and Robert Petroski, "Metal And Concrete Inputs For Several Nuclear Power Plants", Report UCBTH-05-001, February 4, 2005 http://www.nuc.berkeley.edu/pb-ahtr/papers/05-001-A_Material_input.pdf

6 Fournier B., Bérubé M-A.,, Folliard K.J., Thomas M., " ASR Benchmarking Workshop Final Report", Chicago June 2006, http://www.fhwa.dot.gov/pavement/concrete/asrbench.cfm

7 Davis H.S., "Concrete for Radiation Shielding – In Perspective", Int. Symp. "Concrete For Nuclear Reactors", ACI –SP34, Berlin 1970 pp. 3-13

8 Fournier B., Bérubé M-A.,, Folliard K.J., Thomas M , "Report on the Diagnosis, Prognosis, and Mitigation of Alkali-Silica Reaction (ASR) in Transportation Structures," FHWA, Jan. 2010; http://www.fhwa.dot.gov/pavement/pub_details.cfm?id=674

also that all national (all branches of the military as well as the FAA) airfield pavement specifications, and some state DOT specifications, have performance criteria significantly more conservative than what is currently in ASTM or AASHTO in regards to preventing deleterious manifestations of ASR, as these were found to be insufficiently conservative in reliably preventing the issue. Performance test criteria should be developed or adopted that can reliably prevent deleterious effects of ASR from occurring. Mixture designs should reduce the potential for ASR by the combination of limits on alkali content and the utilization of appropriate SCMs and/or lithium-based admixture according to performance criteria. In addition, ASR performance criteria must be applied to appropriately sized lots of materials, due to the scale of these projects and the inherent variability of the materials that will be utilized, and not simply approved for an entire project as is often the case in actual practice to date.

- Aggregates having a higher density than aggregates used in non-nuclear construction are sometimes used for radiation shielding. These aggregates need to be fully characterized to ensure that they have the desired characteristics, due to changing sources used in construction 30 years ago when many plants were constructed.
- The behavior of irradiated concrete should be further investigated. The existing data is becoming dated and newer concretes and additives should be examined.

It should be noted that SCM was used in NPPs in other countries such as Canada. For instance, the concrete of the reactor building (RB) at Point Lepreaux Generating Station (PLGS) in New Brunswick, Canada contains fly ash in the mix design. Construction of the RB began in 1975 and was completed in 1981; commercial operation started in 1983. Visual inspections conducted in 2000 and 2009, as well as detailed assessments of the dome and the ringbeam conducted in 2010 showed that although deficiencies were identified which require remedial action to ensure continued good performance of the RB, overall the structural elements investigated are in satisfactory condition. It is believed that the use of FA in the concrete of the RB at PLGS is the principal reason for its observed superior performance compared to other RBs of similar age[10]. Usage of slag would also be beneficial.

## *4.2. Implementation of New Technologies*

Technologies often used in the infrastructure construction could be easily imported for use in nuclear power plants. This approval would help constructability. Therefore, the NRC and DOE should have an accelerated process to approve new technologies. The Nuclear Energy Enabling Technology (NEET) workshop[11], "Nuclear Energy Enabling Technologies", held in Rockville on July 29, 2010 stated that concrete "The essential of civil engineering construction, concrete

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[9] "Standard Recommended Practice for Determining the Reactivity of Concrete Aggregates and Selecting Appropriate Measures for Preventing Deleterious Expansion in New Concrete Construction," AASHTO, 2010 http://www.fhwa.dot.gov/pavement/concrete/asr/hif09001/hif09001.pdf

[10] Aldea C.-M., Shenton B., DeMerchant M.M., Gendron T., *Aging management program of the reactor building concrete at Point Lepreau Generating Station*, Proceedings AMP 2010 International Workshop on Ageing Management of Nuclear Power Plants and Waste Disposal Structures, Toronto, ON, 2010, Ed. V. L'Hostis, #01

[11] NEET workshop "Nuclear Energy Enabling Technologies Workshop report", July 29, 2010, Rockville, Maryland, http://events.energetics.com/NEETWorkshop2010/reports.html

and steel, will be redeveloped using improved concrete compositions that are more uniform and more predictable physical properties". Quality control tools and advanced field tests of concrete that are currently used in the infrastructure construction should be adopted for NPP. The draft of a complete list is beyond the scope of this report.

A technology that should be beneficially adopted in nuclear facilities is the use of self consolidating concrete (SCC) in heavily reinforced walls or domes of nuclear structures. There are no references to SCC in any NRC regulatory documents.

## *4.3. Foreign Standards and Codes*

### 4.3.1. General

Although in the United States no nuclear power plant has been built in the last 30 years, plants have been built in other countries, for instance in France, Finland, China and Russia since the year 2000. Therefore, updated standards and codes should exist in other countries. These standards and codes should be examined carefully by SDOs to determine what parts could be adopted by American SDOs. A list is given here for information only.

The Companion Guide to the ASME Boiler & Pressure Vessel Code, Volume 3 has a chapter (Chapter 49) on the codes for nuclear construction in other countries. More details need to be analyzed.

International and National Codes other than ACI 318 have been examined. ISO committee TC 71, *Concrete, reinforced concrete and prestressed concrete*, has developed ISO 19338, *Performance and assessment requirements for design standards on structural concrete*. The scope of this document states: "This International Standard provides performance and assessment requirements for design standards for structural concrete. It can be used for international harmonization of design and construction requirements. This International Standard includes: a) requirements, which define the required structural concrete performance, b) criteria, which give means for expressing the requirements, and c) assessment clauses, which give acceptable methods of verifying the specific criteria."

CTG suggests that a similar document be made by TC 71 for nuclear applications.

The document also identifies international codes that are deemed to satisfy the ISO 19338. Currently the list of approved standards is:

- **A.2.1 American Concrete Institute standards**
  - *Building Standards Requirements for Structural Concrete*, ACI 318-08, 475 pp., American Concrete Institute, Farmington Hills, Michigan, 48331, USA.
  - *Analysis and Design of Reinforced Concrete Bridge Structures*, ACI 343R-95, 158 pp., American Concrete Institute, Farmington Hills, MI, 48331, USA.
- **A.2.2 European standards**
  - EN 1992-1-1, *Eurocode 2: Design of concrete structures — Part 1: General rules and rules for buildings,* CEN, Brussels.

- **A.2.3 Japanese standards**
  - *AIJ Standard for Structural Calculation of Reinforced Concrete Structures*, 1999, 412 pp., Architectural Institute of Japan, Tokyo 108-8414, Japan (in Japanese).
  - *AIJ Standard for Structural Design and Construction of Prestressed Concrete Structures*, 1998, 473 pp., Architectural Institute of Japan, Tokyo 108-8414, Japan (in Japanese).
  - *Standard Specifications for Concrete Structures*, Japan Society of Civil Engineers, Tokyo, 160-0004, Japan, 2002:
  - ˜ *Part 1. Structural Performance Verification* (Japanese version, 257 pp.; English version, 274 pp.).
  - ˜ *Part 2. Seismic Performance Verification* (Japanese version, 133 pp.; English version, 47 pp.).
  - ˜ *Part 3. Materials and Construction* (Japanese version, 380 pp.; English version, 443 pp.).
- **A.2.4 Australian standards**
  - AS 3600:2001, *Concrete Structures*, 176 pp.
- **A.2.5 Colombian standards**
  - Colombian Code — National Structural Concrete Standards; included in NSR-98, *Colombian Code for*
  - *Earthquake Resistant Design and Construction*.
- **A.2.6 Saudi Arabian standards**
  - SB 304, *Saudi Building Code: Concrete Structures*, Riyadh, Saudi Arabia, L.D. No. 1428/1200, 2007.
- **A.2.7 Brazilian standards**
  - NBR 6118, Design of Structural Concrete — Procedure, 2006, 220 pp.
- **A.2.8 Egyptian standards**
  - ECP 203, *Egyptian Code for the Design and Construction of concrete Structures, limit states design method*."

#### *4.3.2. Information for Specific Countries*

## Europe:

A list of codes related to concrete construction would be

- EN1990 Eurocode 0: Basis of structural design
- EN1991 Eurocode 1: Actions on structures
- EN1992 Eurocode 2: Design of concrete structures
- EN1993 Eurocode 3: Design of steel structures
- EN1994 Eurocode 4: Design of composite steel and concrete structures
- AFCEN (http://www.afcen.org) in France publishes extension documents for nuclear design such as the RCC-G (1988) used for civil works.

The civil design of EPR plants is performed according to **ETC-C** (**E**PR **T**echnical **C**ode for **C**ivil works). This code is a nuclear code which follows the basic requirements of the Eurocodes. It includes however some adaptations to cope with the specific requirements of the nuclear industry. The content of this code is validated by the French Safety Authorities. Initially, it was a joint development of French and German designers reviewed both by French and German Safety Authorities. Normally, this code will be published through AFCEN (http://www.afcen.com/index.php?menu=rcc_g_fr). However, it seems that it is not yet available. Only the ETC- F (Fire Protection) is on the point to be issued.

## India

The containment structure of nuclear power plants in India, Kaiga 1&2 and RAPP 3&4 were built using high performance concrete with microsilica[12,13,14]. As described in the Ref [6,7,8]: "A concrete was needed that had high tensile strength and compressive strength with good durability. The parameters were: 1) Compressive strength 60 MPa; 2) Split tensile strength 3.9 MPa; 3) Permeability (DIN 1048): 5 mm; 4) Slump: 175 ± 25 mm (for easy placing and pumping)". It seems that the mix design was developed by optimizing the material selection and by including microsilica. The codes used are the French codes RCC-G from AFCEN (http://www.afcen.org).

## Canada

The Canadian Standards Association (CSA) is responsible for developing and maintaining the standards and codes. The standards listed below are specific for concrete or concrete containment structures. For concrete reactor buildings information regarding material requirements for patching and overlay materials for repairing concrete structures, materials performance characteristics and repair specifications is available in COG (CANDU Owners' Group) publications.

- CSA A864 Guide to the Evaluation and Management of Concrete Structures Affected by Alkali-Aggregate Reaction.

This Guide has been written by Canadian experts on the diagnosis and treatment of concrete structures affected by alkali-aggregate reaction (AAR). The purpose of the Guide is to provide information on the signs and symptoms of concrete affected by AAR, how to identify AAR and distinguish it from other mechanisms of concrete deterioration, and how to assess the extent and severity of AAR in concrete. It also provides advice on the best ways of evaluating, maintaining, and treating structures affected by AAR. This Guide will be of use to people involved in such work and to the owners of structures affected by AAR.

- CSA N285.5 Periodic inspection of CANDU nuclear power plant containment components.

This Standard specifies requirements for the periodic inspection of containment system components, including containment pressure suppression systems, in CANDU nuclear power plants. The inspection requirements specified in this Standard provide assurance of the structural integrity of metallic and plastic containment system components. Note: The following are addressed by other Standards or procedures as specified in Table 1: (a) leak tightness and operability requirements; and (b) inspection and testing requirements for concrete, embedded parts, and elastomeric seals.

12 Nuclear Power Stations in India, http://www.concrete.elkem.com/dav/dbe49c59c8.pdf

13 Basu P.C., "Performance requirements of HPC for Indian NPP structures", Indian Concrete Journal, vol. 73 #9, September 1999

14 Basu P.C., "NPP containment structures: Indian experience in silica fume-based HPC", Indian Concrete Journal, vol. 75 #10, October 2001

This Standard specifies requirements for (a) inspection; (b) accessibility; (c) inspection methods and procedures; (d) personnel qualifications; (e) inspection criteria; (f) inspection program development; (g) inspection frequency; (h) evaluation of inspection results; (i) disposition of defects; (j) repairs; (k) documentation; and (l) records.

- CSA N287.1-93 (R2004) General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.

This Standard provides the general requirements used in the design, construction, testing, and commissioning of concrete containment structures for CANDU nuclear power plants designated as class containment and is directed to the owners, designers, manufacturers, fabricators, and constructors of the concrete components and parts.

Note: The requirements of the N287 Series of CSA Standards generally exceed the requirements of the National Building Code of Canada.

This Standard includes definitions that are applicable in this Standard and in the other Standards forming the N287 Series.

This Standard includes the responsibilities of the incumbent organizations and individuals with respect to the functions mentioned in Clause 1.4, together with the pertinent documentation required as a statement of quality assurance.

This Standard contains the general requirements for the design, fabrication, construction, installation, examination, testing, and commissioning of containment structures of reinforced (prestressed and non-prestressed) concrete that are installed as components of a CANDU nuclear power plant.

This Standard includes the general requirements for certain metallic and nonmetallic parts that form part of the containment boundary and that are placed in their final position within the concrete or are attached to the concrete, as in the case of containment liners, anchorage systems, and appurtenances.

- CSA N287.2-08 Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.

This Standard specifies requirements for the materials used in concrete containment structures of containment systems designated as class containment components, parts, and appurtenances in CANDU nuclear power plants.

- CSA N287.3-93 (R2004) Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.

This Standard specifies requirements for the materials used in concrete containment structures of containment systems designated as class containment components, parts, and appurtenances in CANDU nuclear power plants.

- CAN/CSA-N287.4-92 (R2008) Construction, Fabrication and Installation Requirements for Concrete Containment.

This Standard specifies the construction, fabrication, and installation requirements that apply to concrete containment structures of a containment system designated as class containment components, parts, and appurtenances for nuclear power plants.

This Standard provides personnel qualification requirements for the work performed in accordance with this Standard.

- CSA N287.5-93 (R2004) Examination and Testing Requirements for Concrete Containment Structures for CANDU.

This Standard provides the examination and testing requirements that will apply to the work of any organization participating in the construction, installation, and fabrication of parts or components of concrete containment structures, or both, that are defined as class containment.

- CSA N287.6-94 (R2004) Pre-Operational Proof and Leakage Rate Testing Requirements for Concrete Containment.

This Standard provides the requirements for pre-operational proof tests and leakage rate tests of concrete containment structures of a containment system designed as Class Containment components. Note: Other components and appurtenances of the containment system may be tested coincident with the tests in this Standard, provided the intent of this Standard is observed and the requirements of CSA Standard CAN/CSA-N285.0 are considered.

- CSA N287.7-08 In-service examination and testing requirements for concrete containment structures for CANDU nuclear power plants.

This Standard provides requirements for in-service examinations and positive pressure leakage-rate testing of concrete containment structures of a containment system that are designated as class containment components.

Periodic testing and inspection of other components and appurtenances of the containment system designed in accordance with the CSA N285 series of Standards are beyond the scope of this Standard. These are covered in CAN/CSA-N285.5.

Note: CSA N287.7-08 has code provisions for in-service examination and testing for leak tightness of the reactor building (RB), including the frequency of the RB leak rate tests.

- CSA S448.1-93 (R2005) Repair of Reinforced Concrete in Buildings.

This Standard specifies requirements for the periodic inspection of containment system components, including containment pressure suppression systems, in CANDU nuclear power plants.

The inspection requirements specified in this Standard provide assurance of the structural integrity of metallic and plastic containment system components.

*Note:* The following are addressed by other Standards or procedures as specified in Table 1: (a) leak tightness and operability requirements; and (b) inspection and testing requirements for concrete, embedded parts, and elastomeric seals.

This Standard specifies requirements for: (a) inspection; (b) accessibility; (c) inspection methods and procedures; (d) personnel qualifications; (e) inspection criteria; (f) inspection program development; (g) inspection frequency; (h) evaluation of inspection results; (i) disposition of defects; (j) repairs; (k) documentation; and (l) records.

- RD-337: Design of New Nuclear Power Plants (November 2008)

This document is available online at:
http://www.cnsc.gc.ca/eng/lawsregs/regulatorydocuments/published/html/rd337/

The scope of the document is stated on the website: " This regulatory document sets out the expectations of the Canadian Nuclear Safety Commission (CNSC) concerning the design of new water-cooled nuclear power plants (NPPs or plants). It establishes a set of comprehensive design expectations that are risk-informed and align with accepted international codes and practices."

It continues stating that: "RD-337 represents the CNSC's adoption of the principles set forth by the International Atomic Energy Agency (IAEA) in NS-R-1, Safety of Nuclear Plants: Design, and the adaptation of those principles to align with Canadian expectations. The scope of RD-337 goes beyond IAEA's NS-R-1 to address the interfaces between NPP design and other topics, such as environmental protection, radiation protection, ageing, human factors, security, safeguards, transportation, and accident and emergency response planning.  Similar to NS-R-1, RD-337 considers all licensing phases, because information from the design process feeds into the processes for reviewing an application for a Licence to Construct an NPP, and other licence applications. "

# 5. Research Needs

## 5.1. High Strength Reinforcing Steel

Issues requiring research for reinforcement with strengths of Grade 80 and higher are: (1) splice and development length design of straight bars, (2) anchorage of hook bars, (3) use of high-strength bars for seismic loading, (4) use of headed reinforcing bars to develop high strength reinforcing steel, and (5) use of mechanical splices or couplers with high strength reinforcing steel.

On item (1), adequate information is available to make decisions on how to proceed. That information demonstrates that for bars with a nominal yield strength of 60 ksi, the test/calculated ratios based on ACI 318 are on the order of 1.24. In contrast, for Grade 75 and higher, the test/calculated ratio is only 1.02. The ACI 408 equation provides a consistent value of approximately 1.24 in both cases. Two potential routes should be considered: (1) using the ACI 408 equation and (2) adding a factor to the ACI 318 equation to provide the higher development and splice lengths needed for bars with yield strengths above 75 ksi.

On item (2), no information exists on the anchorage strength of hooked bars for Grade 75 and higher. In fact, the provisions in ACI 318 are based on a small number of tests, and very little information exists on hooked bars overall.

On item (3), three areas require attention. They are (a) the spacing of stirrups and ties needed to limit buckling of Grade 80 bars in compression when they become plastic, (b) the inelastic cyclic performance of flexural members, and (c) bond slip through beam-column joints under cyclic loading.

On item (4), ACI 318 currently limits $f_y$ to 60 ksi for the design of headed bars. This limitation is based on a total lack of data for headed bars of higher strength, and, as a result, heads cannot be used to anchor Grade 75 or 80 headed bars. The formulation of design criteria for high-strength headed bars will require tests that develop bars to at least 80 ksi. Those tests should address (a) size and geometry of the head and (b) spacing and clustering effects. Confinement has been shown to have little if any impact on the capacity of headed bars with stresses at failure up to 60 ksi subjected to static loading, but has improved the performance of some configurations of headed bars under cyclic loading and may need to be considered.

On item (5), data exists on the mechanical splice performance for high-strength bars. The main task will be to consolidate that information.

ACI Committees 349 and 359 need to review issues in items (1) and (5), and address (2) the anchorage of higher-strength hooked bars, (3) the seismic performance of members reinforced with higher strengths steels, and (4) design criteria for high headed bars made with higher strengths steels.

In addition, based on review of a draft of NESCC recommendations and ongoing research, the Concrete Reinforcing Steel Institute (CRSI) is developing a research plan for obtaining the data needed to permit confident use of high-strength reinforcement in nuclear construction.[15] The CRSI Road Map addresses questions in the following areas relevant to use of high-strength reinforcement:

- Feasibility Study for Containment/Safety-Related Structure Designs
- Database of Properties
- Stress-Strain Characteristics and Ductility
- Development Length and Tension Lap Splices
- Compression Lap Splices
- Standard Hooks
- Mechanical Splices
- Bending and Straightening
- Headed Bars
- Seismic Design Requirements

The CRSI plan also recommends joint industry participation, as well as an expert panel to review the needs and progress of research, and thus could be linked to ACI 349 and ACI 359 activity in this area.

## *5.2. Concrete Radiation Shielding*

Neutrons and gamma photons incident on a concrete radiation shield can cause thermal gradients that can lead to stresses that cause cracking. Radiation and the thermal cycling of such shields are not addressed in ACI 349 and may cause concrete properties to change. Also of concern, and probably not addressed by ACI 349, is the dehydration of concrete shields caused by long term exposure to temperatures above about 90 °C. The loss of water may make concrete shields less effective in stopping neutrons. ANSI/ANS-6.4-2006 addresses the loss of free water from concrete due to heating and its effects on shielding. It recommends that aggregates have high percentage of bound water like serpentine.

No standards that address this degradation in concrete's ability to shield against neutrons are available. There was much early work that is summarized in the *Engineering Compendium on Radiation Shielding*, R.G. Jaeger Ed., Vol. II, Springer-Verlag, pp.386-408 (1975). Some information about the efficacy of concrete neutron shields with various degrees of dehydration can be found in the literature, for example:

- Chilton, A.B., Effect of Material Composition on Neutron Penetration of Concrete Shields, Report 10425, National Bureau of Standards, Washington, DC, 1971.
- Wang., X. and R.E. Faw, "Transmission of Neutrons and Secondary Gamma Rays Through Concrete Shields," *Radiation Protection Dosimetry*, 60(3), 213—222, 1995.
- Shultis, J.K. and R.E. Faw, *Radiation Shielding*, American Nuclear Society, La Grange Park, IL, 2000.

[15] *CRSI Road Map for High-Strength Reinforcing Steel in Nuclear Power Plant Construction*, Concrete Reinforcing Steel Institute, DRAFT, January 14, 2011.

The effectiveness of concrete neutron shields is easily calculated using modern radiation transport codes, *if* the degree of concrete dehydration can be estimated. Because the reactor vendor is generally responsible for designing the shielding for the "nuclear island", there is a need to determine from such vendors if or how they account for concrete dehydration and other changes in the concrete properties due to radiation when designing the shielding.

## 5.3. Durability of Concrete

Nuclear power plants would be more economical if their service life can be reliably designed for ages longer than 60 years. To achieve assured service life of the concrete, models and standards should be available or developed that can quantitatively, with known uncertainty, predict the service life of the concrete materials used for their construction. Research that both draws on the best current experimental and numerical durability and service life knowledge and carries out projects to fill in knowledge gaps needs to be performed.

External attack on thick elements common at NPP is of relative little concern over 100 years time span. Internal attack, however, can cause destruction in short time scales regardless of element thickness. Therefore, the design should consider alkali silica reaction (ASR) cracking by either characterizing the aggregates or by addition of additives (SCM for instance), internal sulfate attack, DEF (delayed ettringite formation), etc... To ensure durability and avoid internal attack, the selection of concrete constituents is an essential part. Developing better mineralogical characterization of aggregates will help to avoid ASR, which can cause important degradation decades after construction is completed. On the other hand, performance based tests could allow the use of ASR susceptible aggregates by establishing the acceptable expansion that can be reduced by suitable mixture design (addition of SCM for instance). Early age behavior, such as setting time and early strength, will affect the construction schedule. For instance, a long setting time would result in delay on demolding concrete elements. These kinds of problems can arise especially when using supplementary cementitious materials (SCM), so that research on the mineralogical characterization of SCMs and their interaction with portland cement at early ages should pay large dividends in reducing costly and time-consuming concrete construction rework when working with dense rebar arrangements.

Measurements and inspection on the construction site are essential and cannot be accomplished without proper test standards.

## 5.4. Performance-Based Design

The performance-based design of concrete[16, 17] is not yet fully implemented in non-nuclear construction. Options for implementation of the prescription to performance (P2P) initiative should be considered for nuclear power plant construction. The obstacle to full

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16 RN Tarun and BW Ramme; High strength Concrete Containing Large Quantities of Flyash, ACI Materials journal, p 111-116 (1989)

17 VK Mathur, Use of Higher Volume Fly Ash in Concrete for Building Sector, CII-CANMET –CIDA, HVFA Project, January 2005

implementation is the lack of test methods to measure desirable properties and the lack of models to predict performance after 50 or 100 years of service.

The recommendation is that performance-based design requirements be adopted, in cases where the technology is mature enough to justify such adoption. One possibility would be to allow contractors to propose other mixes as long as they can prove the quality of the final product.

## 5.5. Ultra-High Performance Concretes (UHPC)

Ultra-High Performance Concretes (UHPC), or reactive powder concrete (RPC) are relatively new cementitious based materials with low permeability and incorporating fibers to obtain a very ductile and durable material. RPC has been used in some instances in nuclear power plants, such as in France to build cross-flow cooling towers[18]. Standards and codes need to be developed to allow a wider use of these materials that possibly could reduce the rebar congestion in some components of the plant.

## 5.6. Use of Lapped Splices in Regions of Low Biaxial Tension

The use of welded or mechanical splices of reinforcement in regions of biaxial tension where tensile stresses perpendicular to the reinforcement are well below expected tensile crack stress in the concrete is both time consuming and expensive. It may be possible to show by comprehensive testing of this condition that lapped splices will reach the ultimate tensile capacity of the reinforcement being spliced. The testing would have to be very comprehensive. This is an area in which there is no data.

## 5.7. Temperature Loading on Concrete

Temperature loads on concrete are generally deformation limited in that the structure to which they are applied will deform and thereby reduce or remain constant; hence, should be considered as secondary stresses in the structure. This phenomenon is well understood in ASME steel vessel or piping Code design where no limit is placed on such secondary stresses if they have limited extreme load application and for normal service load application their stress is limited to twice yield stress independent of any other load.

Experience has shown concrete structures in normal environmental service where temperature differentials do not exceed 100 $^{o}$F do not show any loss of structural capacity. It is suggested that a number of concrete structures can be modeled and analytically subjected to a 100 $^{o}$F temperature differential to demonstrate that the twice yield stress criteria is not exceeded. This research could form the basis of eliminating a temperature design load if a less than 100 $^{o}$F temperature differential criteria is met and thereby simplifying and reducing the structural design effort for concrete structures with no reduction in concrete structure capacity.

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18 Birelli G., Dugat J., Bekaert A., “The use of RPC in Cross-Flow Cooling Towers”, Proc. of Int. Symp. on High-Performance and Reactive powders Concretes, Sherbrooke, Canada, August 1998

High Density concrete (HDC) is not used in nuclear power plant construction[19, 20], but it is used for dry storage containers[21]. The work was initiated because the cement industry in Canada has moved away from manufacturing "special use" portland cements (moderate heat, low heat and high sulfate resistance, e.g. similar to Type IV, Type II and Type V cements - ASTM C150), which were approved for this application. Blended cements were used to satisfy current and future requirements of limiting heat-generation capacity of HDC during the initial period of curing and subsequent cooling to avoid thermal cracking (and losing its shielding ability), as well as protection of embedded steel components from corrosion, which are of concern for the application. Performance-based specifications were developed; HDC uses high density aggregate and blended cements with a high volume of slag (50% slag replacement by mass, or mass of cementitious materials). According to Schindler and Folliard[22], the use of fly ash provides lower temperature rise than slag mixtures and far less than portland cement. Therefore, mixtures including fly ash and other supplementary cementitious materials should be preferred. This will improve the density, while reducing porosity as well[23].

Recommendation, slag should not be preferred as the reference shows slag contributes more heat than fly ash on an equivalent proportion basis.

## 5.8. Present Research

### 5.8.1. Westinghouse-Bechtel: Steel Composite (SC) Walls

At Purdue University, the Center for Structural Engineering and Emerging Technologies in Nuclear Power Plants (SEET-NPP Center), was established[24]. As part of Purdue University and the Center, the following activities were performed:

1) Bechtel sponsored research focusing on the effects of combined thermal and mechanical loading on SC walls. Purdue University conducted full scale experimental research of SC designs representing steam generator compartments. The experimental results were used to verify analytical models, and develop design guidelines. The design guidelines will be included in the draft Appendix on SC design in the AISC N690 Specifications. All this research is in the public domain. Papers are published in SMiRT 19[25] and 20[26].

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[19] Aldea C.-M., Cornelius B.J., Balinski J., Shenton B., Sato J., *High volume slag blended cement concrete for high density concrete at mid-range temperatures*, SP-263-7, ACI SP-263 Slag cement concrete, Ed. Aldea C.-M, pp. 95-110, 2009.

[20] Aldea C.-M., Cornelius B.J., Balinski J., Shenton B., *Performance of high volume supplementary cementing materials for high density concrete at elevated temperature*, Proceedings 12th International Congress on the Chemistry of Cement, Montreal, Canada July 8-13, 2007, eds. Beaudoin J.J., Makar J.M. and Raki L., TH8-01.3.

[21] Aldea C.-M., Shenton B., Cornelius B.J., *Performance-based specifications and sustainable development using slag cement*, ACI SP-269-4, ACI SP-269 Concrete: the sustainable material of choice, eds. Emily Lorenz, Julie Buffenbarger and Corina-Maria Aldea, pp. 37-48, 2010.

[22] Schindler and Folliard, :Influnce of Supplementary Cementing Materials on the Heat of Hydration of Concrete", Advances in Cement and Concrete IX Conference, August 2003

[23] VK Mathur, Use of Higher Volume Fly Ash in Concrete for Building Sector, CII-CANMET –CIDA, HVFA Project, January 2005.

[24] In 2011, Dr. Amit Varma was the director of this center and helped prepare this session

[25] 19th International Conference on Structural Mechanics in Reactor Technology, Toronto (Canada) August 2007

[26] 20th International Conference on Structural Mechanics in Reactor Technology, Espoo (Finland) August 2009

2) Purdue, Bechtel, and Southern Nuclear sponsored research on the effects of combined in-plane forces and out-of-plane moments on the behavior and design of SC walls. The results from this study was presented to the AISC sub-committee on modular construction (Nov. 18, 2009) and the NRC (Dec. 21, 2009) as part of Westinghouse's shield building review meeting. The design guidelines will be included in the draft Appendix on SC design in the AISC N690 Specifications. This research is in the public domain. The report is available from Purdue University. It is being finalized, and journal papers are being prepared for submission.

3) Westinghouse sponsored experimental and analytical research to verify their shield building design, which is an SC structure. This work included experimental behavior, analysis, and design of their designs:

(a) SC-to-RC anchorage connections
(b) out-of-plane flexural and shear behavior of SC modules
(c) in-plane shear behavior of SC modules
(d) composite action behavior of SC modules

The results of these studies were submitted and presented to the NRC review panel (May 5, 2010) and the ACRS (April 24, 2010). The results will be in the public domain in the future after the NRC review process is complete.

4) Purdue University, Westinghouse, and Bechtel have sponsored research on the out-of-plane shear and flexural behavior of SC modules with and without transverse reinforcement, and with and without self-consolidating concrete. This work is in progress, and in the public domain. The results will be available and published before the end of 2010.

5) Purdue University, Westinghouse, and Bechtel have also sponsored research on the behavior of SC-to-RC anchorage connections. This work is in progress and in the public domain. The results will be available by the end of the year.

6) Westinghouse is in the process of sponsoring experimental and analytical research to verify the designs of their SC modules other than the shield building, namely those inside containment, or part of the auxiliary building. This work is just beginning, and is geared towards addressing questions from the UK regulators where the AP1000 design is being reviewed.

Participants:

- Dr. Sanj Malushte is the cognizant program manager and contact from Bechtel. He is also the chair of the sub-committee developing specifications for SC design.
- Mrs. Penny Toniolo is the contact person at Westinghouse. She is project manager, and
- Mr. Keith Coogler is the technical person from Westinghouse.
- Mr. Don Moore is the contact person at Southern Nuclear.
- Dr. Bob Kennedy has been involved in an advisory role in most of the work related to Westinghouse shield building.

## 5.9. Others Research Needs

The list presented here was not developed extensively and it should be used for developing research plan of the NPP that are under approval or even beyond:

- Standards for modular construction vs. field manufacturing
- Acceptance by NRC and industry of direct integration of updated codes and standards into the design software. This would need the development of models for verification in design software
- Develop adequate code cases as needed to update construction codes
- Integrate lessons learned since last construction. Better integration of documents such as NUREG/CR- 4652 chapter on “Generic Agin Lessons Learned” in codes and standards.
- Perform a study and perhaps testing to determine the extent to which the Concrete Strength Design provisions of ACI 349 can be integrated into the ACI 359/ASME Division 2 Concrete Containment Code. This is especially relevant when applied to the area of prestressed concrete containment design. Sargent & Lundy has a Work Group on Code Modernization working on a research proposal, but this task is in a very early stage of development.

# 6. Summary

The CTG met several times to reach a consensus on the standards and codes needs for the construction of new nuclear power plants. The work concentrated on examining relevant documents of SDOs and on identifying research in progress and research needs.

In examining the SDO documents recommendations were made to improve the clarity of the documents and minimize real or perceived inconsistencies between documents. The research list is quite long and showed that much can still be done to improve cost-effective construction of safe and durable nuclear power plants. New technologies available in the commercial marketplace (bridges to buildings) should be examined and adopted on a fast track as appropriate for nuclear power plants to increase constructability and reduce costs.

A primary recommendation is that the NRC needs to implement a process to ensure that the most up to date standards and codes available are used in the Regulatory Guides and other documents.

## Appendix A: Request for CTG

**Nuclear Energy Standards Coordination Collaborative**

**Application for NESCC Task Group**

ANSI American National Standards Institute

NIST National Institute of Standards and Technology U.S. Department of Commerce

**1.) Submitter Contact Information:**

Chiara “Clarissa” Ferraris
NIST, MS 8615, 100 Bureau Dr., Gaithersburg MD 20899
clarissa@nist.gov; Ph: 301-975 6711

**2.) Task Group Name:** *Concrete Codes and Standards for Nuclear Power Plants*

**3.) Scope and Objectives of Task Group:**

Scope: Establish coordination and consistency of safety and non-safety related concrete requirements in new nuclear power plants (NPP).

Identify new design requirements for safety related concrete components, and develop a plan to incorporate these new requirements into codes and standards.

Identify and review all U.S. Nuclear Regulatory Commission (NRC) Regulatory documents related to concrete for nuclear power plants

Objectives: The process will be carried out in a series of sequential steps:

5) Review the Mattson report[27], NRC NUREG CR 5973[28], and all NRC-regulatory documents to identify all references to concrete codes and standards applied to both material and structural specifications for nuclear power plants;

6) Categorize all the concrete codes and standards that are referenced in each NRC-regulatory documents as:

[27] ”Construction Codes and Standards: Avoidance of New Nuclear Power Plant Construction Delays,” Department of Energy, September 2008.

[28] (NUREG/CR-5973) Codes and Standards and Other Guidance Cited in Regulatory Documents, Rev. 2, August 1995.

- Up-To-Date: The reference is the most relevant
- Outdated but Appropriate for Application: An updated version exists
- Reference Needs Revision: The reference is obsolete, or a different code or standard takes precedence

7) Identify relevant concrete codes and standards missing from the NRC-regulatory documents

8) Identify research needs to fill knowledge gaps in existing concrete codes and standards

4.) **Expected Results of Task Group:** Inventory of relevant standards with gaps and overlaps analysis. Recommendations for revision or new development of concrete standards, Review of current citation of concrete codes and standards in NRC regulatory documents.

**5.) Name and Contact Information of Task Group Convener:**

Chiara "Clarissa" Ferraris
NIST, MS 8615, 100 Bureau Dr., Gaithersburg MD 20899
clarissa@nist.gov
301-975 6711

**6.) Identified Participants:**

Participants should be recruited from

- relevant SDOs: e.g. ACI, ASME and AISC
- nuclear power plant industry (NSSS vendors, Engineering Design & Construction (EDC) firms, utilities)
- NRC staff (licensing and inspection)
- accredited/certifying concrete inspectors
- Other experts in specific fields as needed

7.) **Date, Time and Location of Meetings:** TBA in coordination with participants. It is anticipated, that virtual meeting and electronic communication would be utilized as much as possible. Meetings to be held coincident with scheduled relevant technical committee meetings.

# Appendix B: Load Combinations Suggestions

Current ACI 349-06 Load Combinations are contained in Section 9.2 of the Code. In Table B-1, a comparison between the Load Combinations of ACI-349 to those of AISC N690 is shown. In Table B-1, three load categories of Normal, Severe and Extreme Environmental/Abnormal of AISC N690 were used for ACI-349 as well.

In general, there is no reason for these two load and load combinations tables to be different unless there are specific load phenomena such as pre-stress loads which are unique. A suggested change to the way thermal loads, $T_o$ and $T_a$, are considered in design is shown in bold in the general notes of Table 1. Thermal loads should not be considered when the temperature differential between different operation states or start-up or construction and maximum operating temperature is less than 100 °F (56 °C).

Table B-2 shows the changes in the ACI-349 table that would be necessary to incorporate the guidelines given in the U.S. NRC's R.G. 1.142 Rev. 2. These changes are shown in bold.

In Table B-3 shows the changes to Table CC3230-1 of ACI-359 necessary to incorporate the guidance published by the U.S. NRC in Standard Review Plan 3.8.1.

**TABLE B-1 Design Basis Loads and Load Combinations for ACI-349**

| <u>**AISC N690-06**</u> | <u>**ACI-349-06**</u> |
|---|---|
| **Normal Load Combinations**<br>1. $1.4 (D + R_o) + T_o + C_{cr}$<br>2. $1.2 (D + R_o + T_o) + 1.6L + 1.4C + 0.5(L_r$ or S or R)<br>3. $1.2 (D + R_o + T_o) + 1.6 (L_r$ or S or R) + 0.8L + 1.4C | **Normal Load Combinations**<br>1. $1.4 (D + F + R_o) + T_o$<br>2. $1.2 (D + F + T_o + R_o) + 1.6 (L + H) + 1.4C_{cr} + 0.5 (L_r$ or S or R)<br>3. $1.2 (D + F + T_o + R_o) + 0.8 (L + H) + 1.4 C_{cr} + 1.6 (L_r$ or S or R) |
| **Severe Environmental Load Combinations**<br>4. $1.2 (D + R_o) + 1.6W + 0.8L + C + 0.5 (L_r$ or S or R) $+ T_o$<br>5. $1.2 (D + R_o) + 1.6E_o + 0.8L + C + 0.2 (L_r$ or S or R) $+ T_o$ | **Severe Environmental Load Combinations**<br>4. $1.2 (D + F + R_o) + 1.6 (L + H + E_o)$<br>5. $1.2 (D + F + R_o) + 1.6 (L + H + W)$ |
| **Extreme Environmental and Abnormal Load Combinations**<br>6. $D + 0.8L + C + T_o + R_o + E_S$<br>7. $D + 0.8L + T_o + R_o + W_t$<br>8. $D + 0.8L + C$ or $1.2P_a + R_a + T_a$<br>9. $D + 0.8L + (P_a + R_a + T_a) + (Y_r + Y_j + Y_m) + 0.7E_s$ | **Extreme Environmental and Abnormal Load Combinations**<br>6. $1.0 (D + F + C_{cr} + H + T_a + R_a + E_s) + 0.8L$<br>7. $1.0 (D + F + C_{cr} + H + T_a + R_a + W_t) + 0.8L$<br>8. $1.0 (D + F + C_{cr} + H + T_a + R_a) + 1.2 P_a + 0.8L$<br>9. $1.0 (D + F + C_{cr} + H + T_a + R_a + P_a + Y_r + Y_j + Y_m) + 0.8L$ |

**General Notes applicable to AISC T-12 and ACI-349:**

(1) In application of $T_o$ and $T_a$ the thermal gradient and structural restraint effects shall be considered as appropriate. **If the $T_o$ or $T_a$ temperature differential is less than 100 $^oF$, $T_o$ or $T_a$ can be taken equal to 0 $^oF$.**

**Notes for AISC T-12:**

(1) Where the structural effect of differential settlement is significant, it shall be included with the dead load.
(2) Where required, loads due to fluids with well-defined pressures shall be treated as dead loads, and loads due to lateral earth pressure, ground water pressure, or pressure of bulk materials shall be treated as live loads.
(3) If the dead load acts to stabilize the structure against the destabilizing effects of lateral force or uplift, the factor on dead load shall be 0.90 of the assigned factor and that on other gravity loads (L, $L_r$, S, C) shall be zero.
(4) If the OBE is not part of the design basis, Load Combination 5 need not be evaluated.
(5) In Load Combinations 8 and 9, the maximum values of $P_a$, $T_a$, $R_a$, $Y_j$, $Y_r$, and $Y_m$, including an appropriate dynamic load factor, shall be used unless a time-history analysis is performed to justify otherwise. In combination 9 the required strength criteria shall first be satisfied without $Y_y$, $Y_r$ and $Y_m$. In Load Combinations 7 through 9, when considering concentrated loads $Y_j$, $Y_r$, and $Y_m$ for tornado borne missiles.
(6) In addition to the abnormal loads, hydrodynamic loads resulting from a loss of coolant accident (LOCA) and/or safety relief valve actuation shall be appropriately considered for steel structure components subjected to these loads. Any fluid structure interaction associated with these hydrodynamic loads and those from the postulated seismic loads shall be taken into account.
(7) In Load Combination 6, the load "C: shall be permitted to be waived provided it can be demonstrated that the probability of $E_s$ and C occurring at the same time is less than 10E-6.

***Notes for ACI-349***

(1) In the design for normal loads, consideration shall be given to the forces due to such effects as pre-stressing, vibration, impact, construction and testing.
(2) In the determination of earthquake loads, consideration shall be given to the dynamic response characteristics of the concrete structure and its foundation and surrounding soil.
(3) The determination of impulsive and impactive loads, such as the loads associated with missile impact, whipping pipes, jet impingement, and compartment pressurization, shall be consistent with the provisions of Appendix F.
(4) Design of structures and structural members using the load factor combinations and strength reduction factors of Appendix C shall be permitted. The use of load factor combinations shown in the Table in conjunction with strength reduction factors of Appendix C shall not be permitted.
(5) Where the structural effect of differential settlement, creep, shrinkage, or expansion of shrinkage compensating concrete are significant, they shall be included with the dead D

in Eq. (4) through (9). Estimations of these effects shall be based on a realistic assessment of such effects occurring in service.

(6) Load combinations in the Table shall be evaluated with 0.9D to assess the adverse effects of reduced dead load. For any other load (for example, **L**), if the load reduces the effects of other loads, the corresponding factor for that load shall be taken as 0.9 of the assigned factor, if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the factor for that load shall be taken as zero.

(7) Where applicable, impact effects of moving loads shall be included with the crane load, $C_{cr}$.

(8) In Eq. (8) and (9), the maximum values of $P_a$, $T_a$, $R_a$, $Y_j$, $Y_r$, and $Y_m$, including an appropriate dynamic load factor, shall be used unless an appropriate time history analysis is performed to justify otherwise.

(9) Equation (7) shall be satisfied first without the tornado missile load, and Eq. (9) shall be first satisfied without $Y_j$, $Y_r$, and $Y_m$. When considering these impactive and impulsive loads, local section strength and stresses may be exceeded provided there will be no loss of intended function of any safety related systems. For additional requirements related to impulsive and impactive effects, refer to Appendix F.

(10) If resistance to other extreme environmental loads such as extreme flood is specified for the NPP, then an additional load combination shall be included with the additional extreme environmental load substituted for $W_t$ in Eq. (7).

(11) For post tensioned anchorage zone design, a load factor of 1.2 shall be applied to the maximum pre-stressing steel jacking force.

(12) In eq. (6), the crane load $C_{cr}$ may be omitted if probability analysis demonstrates that the simultaneous occurrence of an SSE (DBE) with crane usage is not credible.

(13) In eq. (6) and (9), it shall be permitted to reduce the load effects of $E_{ss}$ by 10 %, provided the exceedence probability of $E_{ss}$ is equal to or lower than (1) 1.0E-5 (median), for NPP structures, and (2) 1.0E-4 (mean) for other nuclear facilities structures.

(14) If the $T_o$ and $T_a$ temperature differential is less than 100 $^oF$, $T_o$ or $T_a$ can be taken equal to 0 $^oF$.

**TABLE B-2 Modification of ACI 349-06 Load Combinations to Incorporate Regulatory Guide 1.142 Rev. 2 Changes (Changes indicated in Bold/underline)**

**<u>ACI-349-06</u>**

**Normal Load Combinations**

1. $1.4\,(D + F + R_o) + T_o$
2. $1.2\,(D + F + T_o + R_o) + 1.6\,(L + H) + 1.4C_{cr} + 0.5\,(L_r \text{ or } S \text{ or } R)$
3. $1.2\,(D + F + T_o + R_o) + 0.8\,(L + H) + 1.4\,C_{cr} + \underline{\mathbf{1.6}}\,(L_r\ \underline{\mathbf{or}}\ S \text{ or } R)$

**Severe Environmental Load Combinations**

4. $1.2\,(D + F + \cancel{T_o +}\ R_o) + 1.6\,(L + H + E_o)$
5. $1.2\,(D + F + \cancel{T_o +}\ R_o) + 1.6\,(L + H + W)$

**Extreme Environmental and Abnormal Load Combinations**

6. $2.0\,(D + F + C_{cr} + H + \underline{\mathbf{(T_a \text{ or } 1.2\,T_o)}} + R_a + E_s) + 0.8L$
7. $1.0\,(D + F + C_{cr} + H + \underline{\mathbf{(T_a \text{ or } 1.2\,T_o)}} + R_a + W_t) + 0.8L$
8. $1.0\,(D + F + C_{cr} + H + \underline{\mathbf{(T_a \text{ or } 1.2\,T_o)}} + R_a) + \underline{\mathbf{1.4}}\,P_a + 0.8L$
9. $1.0\,(D + F + C_{cr} + H + \underline{\mathbf{(T_a \text{ or } 1.2\,T_o)}} + R_a + P_a + Y_r + Y_j + Y_m + E_s) + 0.8L$

**<u>Added Notes:</u>**

(1) In Load Combinations 7 the $W_t$ load may be used to also represent malevolent or accidental vehicle assault, aircraft impact and external explosion blast wave

(2) Additional limits on assumed ductility given in Appendix F are contained in U.S. NRC Regulatory guide 1.142 Rev. 2 dated Nov. 2001

(3) The values of $T_o$ or $T_a$ may be taken equal to zero when the temperature differentials being considered are less than 100 $^oF$.

(4) While $E_s$ load is combined with $Y_x$ loads in Eq. 9, the approach taken in AISC T-12 is to combine with a reduced Safe Shutdown Earthquake Load. This would appear to be more rational (i.e. 0.7 $E_{s)}$ than the current ACI-349 or ACI-359 approach for concrete building structures or concrete reactor containment structures which combines a design basis containment pressure load (typically taken as 1.1 to 1.2 times maximum calculated pressure load) with the Safe Shutdown Earthquake Load when the containment design pressure load would result from failure of structures, systems and components specifically designed for the safety Safe Shutdown Earthquake Load. The use of a reduced Safe Shutdown Earthquake Seismic Load with a design basis Loss of Coolant Accident Pressure Load is the design for containment design in Canada as well as Japan.

There should be an additional case for use in one-time, very well defined temporary loadings. Under those unique conditions, the use of 1.6 L is quite conservative and limiting. Unfortunately, ACI 349 (or 318) does not recognize this situation. This should be addressed.

**Table 3 Suggested Changes to Table CC-3230-1**

2007 SECTION III, DIVISION 2

TABLE CC-3230-1
LOAD COMBINATIONS AND LOAD FACTORS

| Category | $D$ | $L(1)$ | $F$ | $P_t$ | $G$ | $P_a$ | $T_t$ | $T_o$ | $T_a$ | $E_o$ | $E_{ss}$ | $W$ | $W_t$ | $R_o$ | $R_a$ | $R_r$ | $P_v$ | $H_a$ |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| **Service** | | | | | | | | | | | | | | | | | | |
| Test | 1.0 | 1.0 | 1.0 | 1.0 | ... | ... | 1.0 | ... | ... | ... | ... | ... | ... | ... | ... | ... | ... | ... |
| Construction | 1.0 | 1.0 | 1.0 | ... | ... | ... | ... | 1.0 | ... | ... | ... | 1.0 | ... | ... | ... | ... | ... | ... |
| Normal | 1.0 | 1.0 | 1.0 | ... | 1.0 | ... | ... | 1.0 | ... | ... | ... | ... | ... | 1.0 | ... | ... | 1.0 | ... |
| **Factored** | | | | | | | | | | | | | | | | | | |
| Severe environmental | 1.0 | 1.3 | 1.0 | ... | 1.0 | ... | ... | 1.0 | ... | 1.5 | ... | ... | ... | 1.0 | ... | ... | 1.0 | ... |
| | 1.0 | 1.3 | 1.0 | ... | 1.0 | ... | ... | 1.0 | ... | ... | ... | 1.5 | ... | 1.0 | ... | ... | 1.0 | ... |
| Extreme environmental | 1.0 | 1.0 | 1.0 | ... | 1.0 | ... | ... | 1.0 | ... | ... | 1.0 | ... | ... | 1.0 | ... | ... | 1.0 | ... |
| | 1.0 | 1.0 | 1.0 | ... | 1.0 | ... | ... | 1.0 | ... | ... | ... | ... | 1.0 | 1.0 | ... | ... | 1.0 | ... |
| Abnormal | 1.0 | 1.0 | 1.0 | ... | 1.0 | 1.5 | ... | ... | 1.0 | ... | ... | ... | ... | ... | 1.0 | ... | ... | ... |
| | 1.0 | 1.0 | 1.0 | ... | 1.0 | 1.0 | ... | ... | 1.0 | ... | ... | ... | ... | ... | 1.25 | ... | ... | ... |
| | 1.0 | 1.0 | 1.0 | ... | 1.25 | 1.25 | ... | ... | 1.0 | ... | ... | ... | ... | ... | 1.0 | ... | ... | ... |
| Abnormal/severe environmental | 1.0 | 1.0 | 1.0 | ... | 1.0 | 1.25 | ... | ... | 1.0 | 1.25 | ... | ... | ... | ... | 1.0 | ... | ... | ... |
| | 1.0 | 1.0 | 1.0 | ... | 1.0 | 1.25 | ... | ... | 1.0 | ... | ... | 1.25 | ... | ... | 1.0 | ... | ... | ... |
| | 1.0 | 1.0 | 1.0 | ... | 1.0 | ... | ... | 1.0 | ... | 1.0 | ... | ... | ... | ... | ... | ... | ... | 1.0 |
| | 1.0 | 1.0 | 1.0 | ... | 1.0 | ... | ... | 1.0 | ... | ... | ... | 1.0 | ... | ... | ... | ... | ... | 1.0 |
| Abnormal/extreme environmental | 1.0 | 1.0 | 1.0 | ... | 1.0 | 1.0 | ... | ... | 1.0 | ... | 1.0 | ... | ... | ... | 1.0 | 1.0 | ... | ... |

NOTE:
(1) Includes all temporary construction loading during and after construction of containment.

to the geometry of the containment shall be used. In the design of local sections, consideration shall be given to the redistribution of moments and forces in a statically indeterminate structure because of cracking of the concrete, and to the stiffening effect of buttresses or other integral portions of the containment.

*(b)* The short-term as well as the long-term foundation soil properties shall be considered. In order to ensure consideration of the critical condition, a range of values of soil constants shall be considered.

*(c)* For prestressed containments, the analytical methods selected for construction and normal category load combinations shall account for the creep characteristics and the thick section geometry that is characteristic at ring girders and buttresses.

**CC-3320 SHELLS**

*(a)* Containments are normally thin shell structures. Elastic behavior shall be the accepted basis for predicting internal forces, displacements, and stability of thin shells. Effects of reduction in shear stiffness and tensile membrane stiffness due to cracking of the concrete shall be considered in methods for predicting maximum strains and deformations of the containment. Equilibrium check of internal forces and external loads shall be made to ensure consistency of results.

*(b)* Although shell analysis may be based on membrane theory, additional consideration is required for bending and shear forces at penetrations, intersection with base mat, discontinuities, and the stresses and strains caused by temperature variations.

*(c)* The stability of the containment shall be verified, giving consideration to the possible reduction in the buckling capacity caused by large deflections, creep effects, and specified construction tolerances.

*(d)* Model tests may be used instead of the design analysis if they are conservative and represent the prototype containment.

*(e)* Model tests may be used to check the validity of assumptions involved in mathematical analysis.

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| Category | D | L(1) | F | $P_t$ | G | $P_a$ | $T_t$ | $T_o$ | $T_a$ | $E_o$ | $E_{SS}$ | W | $W_t$ | $R_o$ | $R_a$ | $R_r$ | $P_v$ | $H_a$ |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| Abnormal (2) | 1.0 | 1.0 | 1.0 | | 1.0 | 1.0 | … | … | … | … | … | … | … | … | … | … | … | … |

(2) In this load combination $P_a$ is represented by $P_{g1}$ + ($P_{g2}$ or $P_{g3}$) but not less than D+ kPa (45.0 psig) where: $\mathbf{P_{g1}}$ **=** pressure resulting from an accident that releases hydrogen generated from 100-percent metal-water reaction fuel cladding, $\mathbf{P_{g2}}$ **=** pressure resulting from uncontrolled hydrogen burning, $\mathbf{P_{g3}}$ **=** pressure resulting from post-accident inerting, assuming carbon dioxide is the inerting agent. This is per SRP 3.8.1 and NRC RG 1.136

# Appendix C: Discussion of Appendix E of ACI 349/1R-07

(Approved ACI 349 in 2011)

**Appendix E (approved but not published yet)** addressed the increase temperature in following way:
"E.4—Concrete temperatures E.4.1 The following temperature limitations are for normal operation or any other long-term period. The concrete surface temperatures shall not exceed 150 °F except for local areas, such as around penetrations, which are allowed to have increased temperatures not to exceed 200 °F. This limit is permitted to be increased to 180 °F for general surface area and 230 °F for local surface area if the tested concrete strength (e.g., measured compressive strength at 28-day or more) is equal to or greater than 115% of the specified 28-day compressive design strength.

**Commentary**: RE.4—Concrete temperatures Experience with specific types of concrete indicates that higher temperatures than those given in E.4.1 and E.4.2 may be allowed without loss of significant compressive strength. The observed effects, however, are dependent on the type of cement, aggregate, and admixture used in the concrete mix. Also, adequate test data do not exist to establish higher temperature limits for inclusion in the Code. Available data indicate that elevated temperature exposure can produce greater decrease in tensile strength and modulus of elasticity than the decrease in compressive strength. Therefore, in applications where the tensile strength and modulus of elasticity may be important, potential decreases in their values as a result of elevated temperature should be evaluated. E2

Increasing the limit to 180 °F for general surface area and 230 °F for local surface area if tested concrete strength (e.g., measured compressive strength at 28-day or more) is equal to or greater than 115% of the specified 28-day compressive design strength is conservative because the strength gain with the age is not taken into account. Nuclear power plant concrete strengths are generally significantly greater than the specified value so that the relatively small decrease in strength as a result of heating up to limits given will not significantly erode the design (safety) margins. These values are derived from code committee's interpretation of available thermal test data on concrete performance.

If a reinforced concrete structural element in one of the new generation nuclear power plants (e.g., ESBWR, AP1000, etc.) is required to maintain its functional and performance requirements at temperatures in excess of code limits, or at moderately elevated temperatures for extended periods of time, techniques for optimizing the design of structural elements to resist these exposures should be investigated (i.e., material selection and design). With respect to material selection, the performance of the concrete materials can be improved by (1) minimizing the moisture content through aggregate gradation, placement techniques, or use of extended-range water-reducing agents; (2) utilizing aggregates having good thermal stability and low thermal expansion characteristics such as lightweight or refractory materials; and (3) incorporating fibrous reinforcing materials such as short, randomly oriented steel fibers to provide increased ductility, dynamic strength, toughness, tensile strength, and improved resistance to spalling. An alternative approach would be to design the concrete mix for higher strength so that any losses in properties resulting from long-term thermal exposure will still provide adequate design (safety) margins.E2

Because the provisions of E.4.1 and E.4.2 are considered conservative, E.4.3 provides a mechanism by which project-specific acceptance criteria may be developed. Projects where higher temperature limits are needed to assure appropriate structural behavior, and/or to gain regulatory acceptance, may undertake testing programs for their specific concrete design. Testing must simulate the actual, expected operating conditions for the intended use. Such testing must include a sufficient number of specimens so that a pre-established probabilistic acceptance criterion may be satisfied. It should be noted that temperature limits given in this section relate to concrete surface—not the air temperature, pipe surface temperature, or the temperature at some internal point. In the absence of a measured concrete surface temperature, some extrapolation by a rational method (e.g., heat transfer analysis) may be needed if the specified limits are approached. If temperature limits in local areas are exceeded and visual or other non-destructive examinations reveal concrete deteriorations, appropriate corrective action should be taken before returning the plant to service.

# Appendix D: Hybrid Steel and Concrete Containment Design Standard

All Advanced BWR's being considered in future nuclear power plants use the Mark II type of containment so maybe the information presented here might not be needed. A hybrid steel and concrete containment design standard where containment design pressure membrane load is carried by a steel shell and localized missile or jet impingement loads are carried by a combination of a steel plate shell and a concrete shell acting compositely.

This type of design as shown in Figure E-1 came about in Mark III BWR type containments where initially a hybrid containment consisting of a reinforced concrete base mat supported a structure steel shell cylinder and dome. At the time of the original design of this containment system localized loads on the containment shell caused by fluid jet impingement on the containment shell in the region of the wet well were not recognized as significant load. When such loads were quantified during construction stage it was determined that the steel containment cylinder walls which were typically about 1.0 in thick were not capable of resisting such localized concentrated loads which induced bending in the steel containment shell within in code allowable bending stress limits. As a result some of these hybrid type containments had concrete poured in the regions of the wet well between the existing concrete shield wall and the steel containment shell where the existing steel shell was stud connected to the new concrete so that the concrete would act compositely with the steel shell to resist the localized wet well bending moments.

It is not clear that the hybrid-type design just described will be used in the future since current ABWR containment designs seem to be more consistent with the earlier BWR Mark II containment design as shown in Figure E-2 which for the most part used conventional reinforced concrete defined bar or prestressed designs. However, to the extent that the Mark III hybrid containment design may be used in the future, there is no design code that addresses the hybrid containment design just described.

Current composite steel plate faced on one or two faces of walls seems to be limited to a containment shield building or inside containment walls which are not designed to resist significant containment design pressure induced forces and moments.

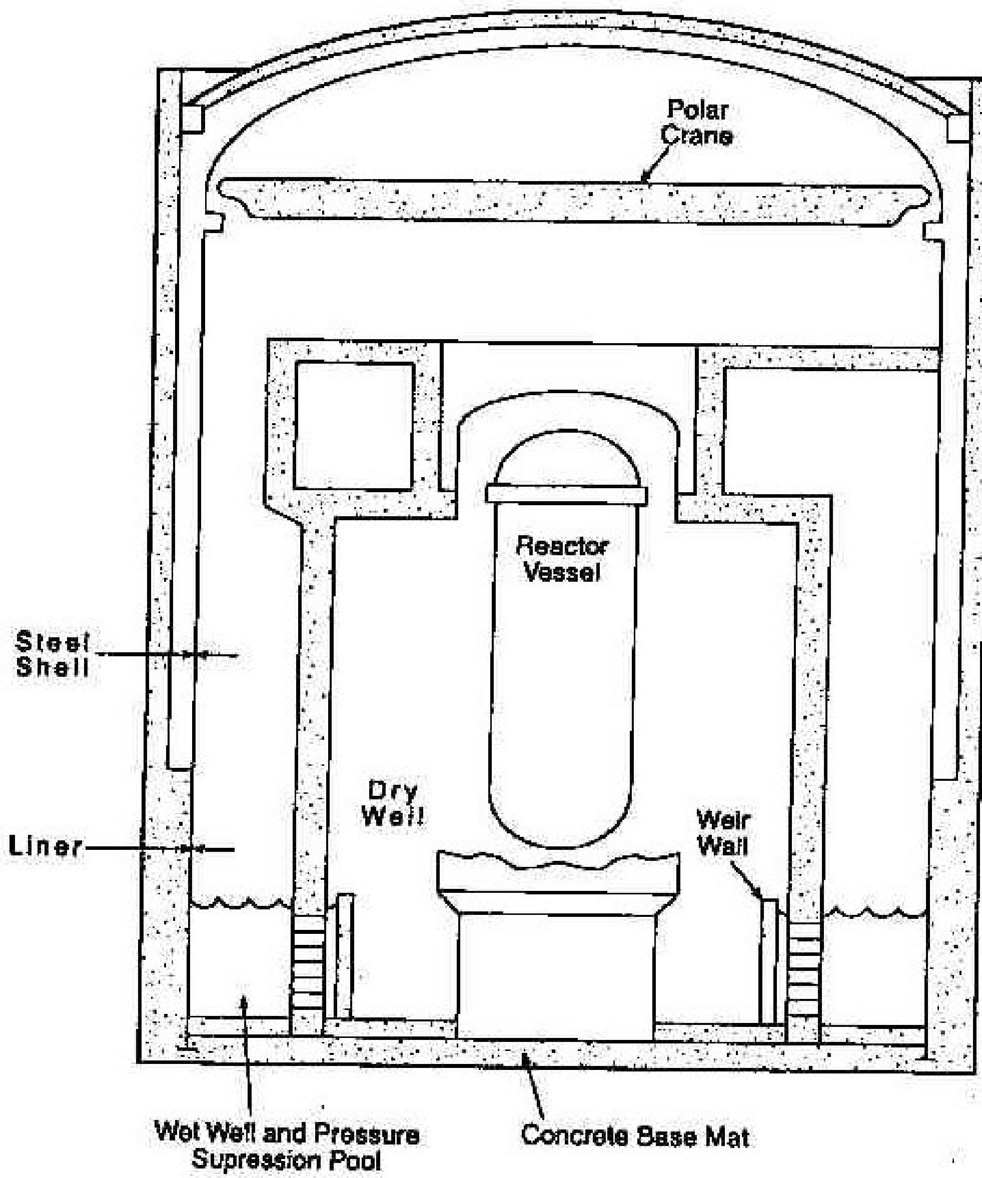


Figure D-1 BWR Mark III Hybrid Containment

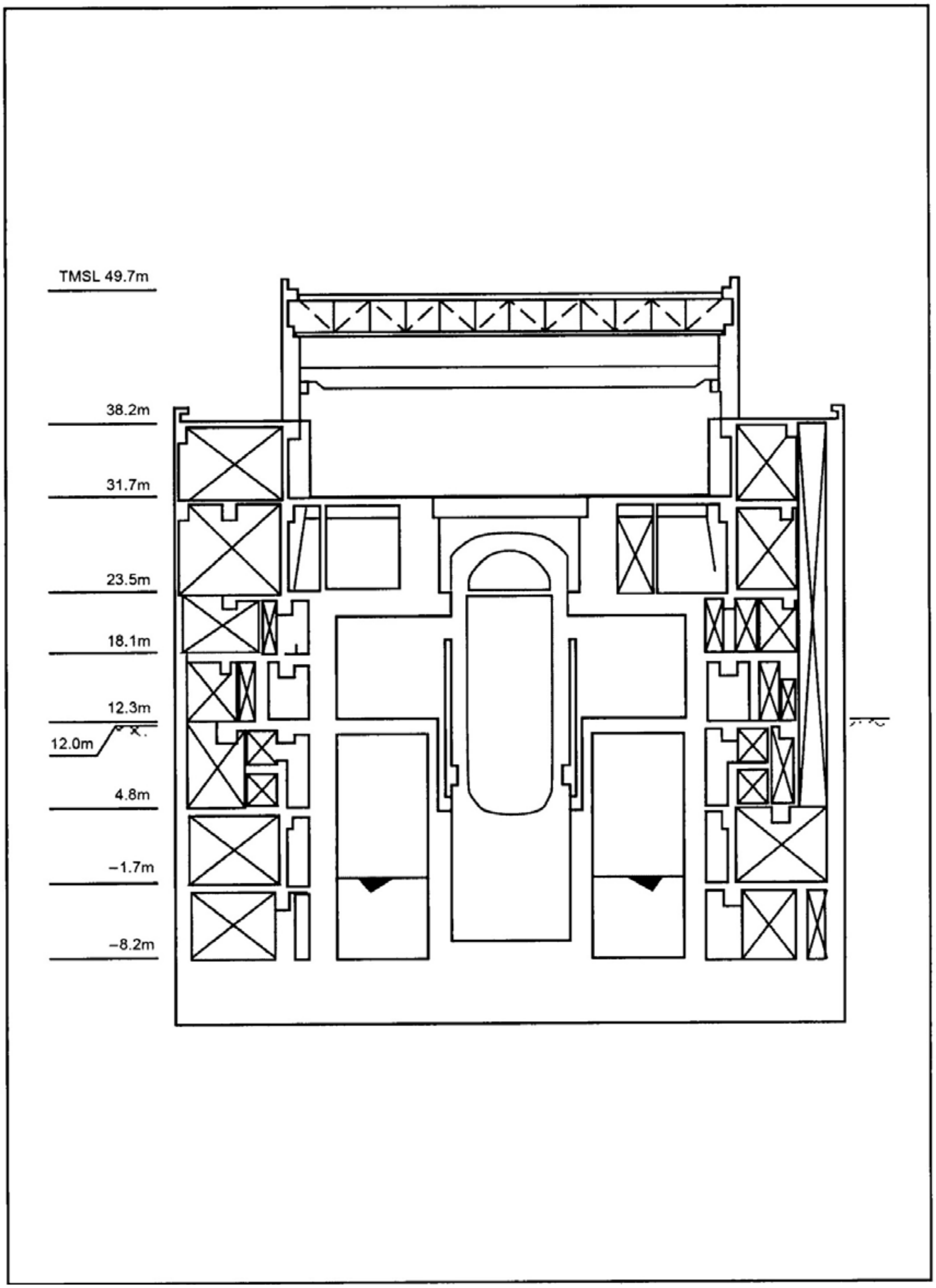


Figure D-2 BWR Mark II Concrete Containment