UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 13, 1992

ORGANIZATION: Nuclear Management and Resource Council (NUMARC)

SUBJECT: SUMMARY OF MEETING WITH NUMARC - DISCUSSION OF REVISION 1, PWR CONTAINMENT STRUCTURES LICENSE RENEWAL INDUSTRY REPORT

On February 26, 1992, the NRC staff met with NUMARC in a public meeting at the Electric Power Research Institute's (EPRI's) Washington, D.C. office to discuss the review of Revision 1 of Pressurized Water Reactor (PWR) Containment Structures License Renewal Industry Report (IR). EPRI and the Department of Energy's representative, Sandia National Laboratory (SNL), joined the discussion by televideo from EPRI's Palo Alto office. Enclosure 1 lists the meeting attendees.

The meeting was prompted by the staff's comments that the revised IR does not have sufficient information to justify some conclusions and parts of the IR are not consistent with the final Part 54 license renewal rule. The staff's comments were based on a preliminary evaluation of Revision 1 of the IR. The comments are reflected in the mark-up of the IR (Enclosure 2) and were briefly discussed during the meeting. As a result, the staff noted that the IR would have to be revised before it would be readily useable by a licensee or before an NRC safety evaluation report (SER) would be appropriate.

A summary of points discussed in the meeting is provided below:

NUMARC offered an overview of their assessment of the NRC comments: (1) discussion of technical issues were not overwhelming and should be fairly easy to rectify, (2) the IR was developed on technical basis and was not intended to keep up with the progress of the Part 54 final license renewal rule or the standard review plan for license renewal (SRP-LR), and (3) the meeting would be seen as continuing the dialogue on the IRs rather than resolving technical issues on the spot. NUMARC also expressed the concern that the comment response document (CRD) was not used to evaluate the IR issues in terms of technical adequacy. It appeared to NUMARC that the mark-ups were a complete review from scratch. In clarifying the intent of the mark-ups, the staff noted that the mark-ups provided an overall perspective of the substantially revised IR. It was also noted that the staff used the revised CRD as a road map for reviewing the revised IR, but would not consider the CRD as part of the IR.

NUMARC stated that continued use of marked-up IRs should be a model for the NUMARC Phase III IR development process. The staff and NUMARC agreed to postpone detailed discussion of technical comments to a future meeting.

Am-7
NUMARC

DF03 1/1

NUMARC asked for clarification of the process of preparing a draft SER. The staff stated that NUMARC would not have an opportunity to review and comment on draft SERs before they are made available to the public. The process for IR review and approval will be the same as for any topical report. Specifically, the staff will follow existing NRC procedures and practices. In general, the process has three phases. The first includes staff review of an IR and resolution of the applicable technical issues and the preparation of a draft SER. The second includes appropriate internal NRC review, and publication of the draft SER for public and industry comment. The third includes comment resolution and further revision of the SER as appropriate, followed by internal review and publication of the final SER.

Additionally, the staff discussed the types of findings to be contained in the SERs. The findings related to the IRs involve a determination of whether or not the IR provided the basis for a conclusion in evaluating age-related degradations. The staff further commented that the revised IR was deficient in that: (1) it is not consistent with the Part 54 final rule, (2) the ASME Section XI containment inservice inspection provisions have not been endorsed by the staff; therefore, the IR should provide technical descriptions independent of these proposed ASME subsections, (3) the IR should not reference the EPRI reports which are not available to the public, and (4) the term "safety function" is not interchangeable with "required function" which, in accordance with the rule, would have much broader definition. NUMARC stated that the IRs were developed based upon the concept of safety function or safety-related function. As a result, the discussion in each IR has a scope which is significantly less than the scope required by the license renewal rule.

The staff also noted that the IR was vague in considering the second license renewal principle of "maintenance of CLB" as the acceptance criteria for evaluating age-related degradation. The discussion of this concept in the IR is ambiguous. NUMARC stated that they had difficulty making statements on CLB maintenance in the IR as they had not reviewed the CLB for any plant for the purpose of preparing the IRs. To this end, both the staff and NUMARC considered standardized language as a positive step but also recognized that the IR will need to be revised.

With respect to future schedules, the staff noted that the schedule for the final SERs would be tied to completion of the steps discussed above. The IRs should be consistent with the Part 54 final rule in addressing technical issues and the AIP items. There was no definitive estimate of the schedule for issuing the final SER, which would depend on the effort required to resolve public comments to the draft SER. NUMARC stated that they needed to work out details to estimate the amount of time required to respond to staff's actions. Further IR revisions would require approximately 90 days for the participating utilities to reach consensus on major issues.

NUMARC proposed a list of dates for future meetings with the staff to discuss IR issues and related policies. The staff agreed to look into the first available date of March 12, 1992, for a meeting to discuss in detail the PWR Containment Structures IR mark-ups. Additionally, issues concerning schedules and milestones for the IR review process were discussed. The need to review each IR and define milestones and schedules was noted. The staff stated that the entire IR process, originally scheduled to be completed in 1991, was being reevaluated in light of other priority activities. A meeting to discuss each IR and schedules will be held with NUMARC at a future date.

Original signed by:
David Tang, Mechanical Engineer
Technical Section
License Renewal Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Meeting Attendees
2. Mark-up, Revision 1 of the IR

cc: E. Griffing, NUMARC

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NAME:	LLuther	: DTang	: PTKuo	: JCraig
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DOC. NAME: TANG MEETING SUMMARY

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Enclosure 1

NRC/NUMARC MEETING
GENERAL DISCUSSIONS ON
PWR CONTAINMENT IR COMMENTS AND PHASE 3 PROCESS
FEBRUARY 26, 1992
9:00 A.M. - 1:00 P.M.
(MEETING CONDUCTED AT EPRI WASHINGTON, DC OFFICES)

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Enclosure 2

Marked-ups

PRESSURIZED WATER REACTOR
CONTAINMENT STRUCTURES
LICENSE RENEWAL INDUSTRY REPORT

NUMARC REPORT 90-01, REVISION 1

SEPTEMBER 1991

NUCLEAR MANAGEMENT AND RESOURCES COUNCIL, INC.

1776 EYE STREET, N.W.

WASHINGTON, DC 20006-3706

ACKNOWLEDGEMENTS

Preparation of this industry report was performed under the direction of the NUMARC NUPLEX Working Group, chaired by John DeVincentis, Vice President, Yankee Atomic Electric Company. Funding an development of this report was supplied by the Electric Power Research Institute (EPRI), under the project management of Jeff Byron. The early drafts of the report were prepared by Stone and Webster Engineering and the final drafts by Bechtel Power Corporation. In addition to those mentioned herein, NUMARC wishes to thank the many others from the industry who contributed to the development of this report.

NOTICE

PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES LICENSE RENEWAL INDUSTRY REPORT

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ACRONYMS & UNITS

This section defines the acronyms and units used frequently throughout this report.

ACRONYMS

ACI - American Concrete Institute
ANS - American Nuclear Society
ASME - American Society of Mechanical Engineers
ASTM - American Society for Testing and Materials
BWR - Boiling Water Reactor
CFR - Code of Federal Regulations
DBA - Design Basis Accident
FSAR - Final Safety Analysis Report
FSSC - Free-Standing Steel Containment
IEEE - Institute of Electrical and Electronics Engineers
ILRT - Integrated Leak Rate Test
IR - Industry Report
MIC - Microbiologically induced corrosion
NRC - Nuclear Regulatory Commission
PWR - Pressurized Water Reactor
RCC - Reinforced Concrete Containment
RG - Regulatory Guide
SCC - Stress Corrosion Cracking

UNITS

mA/ft^2 - Milliamps per square feet
MWe - Megawatts electrical
n/cm^2 - Neutrons per square centimeter
psi - Pounds per square inch
ppm - parts per million
rad - Gamma radiation dose unit (100 ergs/gram)

① Not an equivalent
② Misleading
③ safety function(s) is too limited. See 54.3 definition for SSC ITLR.

SECTION 1
SUMMARY

1.1 PURPOSE

The U.S. nuclear power industry, through coordination by the Nuclear Management and Resources Council (NUMARC), and sponsorship by the U.S. Department of Energy (DOE) and the Electric Power Research Institute (EPRI), has evaluated age related degradation effects for a number of major plant systems, structures and components, in the license renewal technical Industry Reports (IRs). License renewal applicants may choose to reference these IRs in support of their plant-specific license renewal applications, as an equivalent to the integrated plant assessment provisions of the license renewal rule (10 CFR Part 54). ①

1.2 SCOPE

② This IR provides the technical basis for license renewal for U.S. PWR containment structures. The scope of the report includes: (1) steel-lined reinforced concrete and (2) free-standing steel PWR containment structures. Steel-lined reinforced concrete containments may or may not be prestressed; both types are considered. Passive, spray suppression, and ice condenser (i.e., vapor suppression) types of free-standing steel containments are considered. In all cases, the scope includes those elements required to maintain the pressure boundary following a postulated design basis accident. For example, the bellows portions of free-standing steel containment penetration assemblies are included in the scope in those cases where the bellows provide a continuation of the containment pressure boundary. Containment internal structures are excluded from the scope of this IR. The scope of the report is discussed in further detail in Section 2.

1.3 METHODOLOGY

The license renewal technical evaluation consists of four parts. <u>First</u>, the component evaluation basis is established (Section 3), consisting of component descriptions, the general component design bases, and relevant component operating history. <u>Second</u>, the age-related degradation mechanisms that could affect these components are described and their potential significance to component safety function(s) performance, as defined in 10 CFR Part 100, Appendix A, paragraph (a) of section VI, is evaluated (Section 4). ③ <u>Third</u>, for combinations of age-related degradation mechanisms and components that are determined to have a potentially significant effect on component safety function, the ③

① EPs under administrative control are implemented by the facility operating procedures and reviewed by the on-site review committee. They are plant-specific

capability of effective programs to manage the potentially significant effects of age-related degradation is examined (Section 5). Fourth, for cases where generic effective programs ① cannot be shown to be capable of managing the effects of age-related degradation, aging management options for plant-specific programs are described (Section 6).

The age-related degradation mechanisms considered in this report are:

Concrete
- Freeze-thaw
- Leaching of calcium hydroxide
- Aggressive chemicals
- Reactions with aggregates
- Corrosion of embedded steel
- Elevated temperature
- Irradiation

Reinforcing Steel
- Corrosion
- Elevated temperature
- Irradiation

Prestressing System
- Corrosion
- Elevated temperature
- Irradiation
- Prestressing losses

Liner
- Corrosion
- Elevated temperature
- Irradiation

Miscellaneous Age-Related Degradation Mechanisms
- Fatigue
- Concrete interaction with aluminum
- Settlement

Free-Standing Steel Containment Degradation Mechanisms
- Strain aging
- Corrosion

These mechanisms were identified from a review/evaluation of nuclear power plant operating experience, relevant laboratory data, and related experience in other industries.

An age-related degradation mechanism is considered significant if, when allowed to continue without any additional prevention or mitigation measures, it cannot be shown that the component would maintain its safety function ② during the license renewal period. The potential for significant age-related degradation of specific components evaluated in this IR is dependent on their design features (Section 3.1), their design basis (Section 3.2), their operating history (Section 3.3), and the extent to which they are susceptible to the age-related degradation mechanisms (Section 4). If it can be shown that the PWR containment component is either not susceptible to the degradation mechanism under

② CLB maintenance

① CLB maintenance
② See ①, page 1-2

consideration, or is susceptible to such a small degree that the component safety function is not adversely affected throughout the license renewal term, then the component/degradation mechanism combination is not significant (Section 4). Otherwise, the component/degradation mechanism combination is potentially significant. If a potentially significant component/degradation mechanism combination is adequately addressed by effective programs, as justified in the report, then the issue is considered to be resolved on the basis that the degradation is managed acceptably (Section 5). Combinations of mechanisms and components for which generic effective program elements cannot be shown to manage potentially significant age-related degradation require plant-specific evaluation. ①

②

Recommendations for plant-specific aging management options for such issues are then provided (Section 6), if required.

A license renewal applicant intending to take credit for a particular conclusion of this report is responsible for the review/evaluation of plant-specific features, including appropriate current licensing basis (CLB) documents/information, to assure that the assumptions and criteria used as the basis for the determination of insignificance in Section 4 of this report are applicable to the component under consideration. Similarly, the license renewal applicant is responsible for the review/evaluation of their plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements used to manage the effects of potentially ~~significant~~ aging degradation in Section 5 of this report, ~~or their justified equivalent~~, are committed for use at their plant. Plant-specific evaluations are required for any age-related degradation/component issues for which the license renewal applicant is unable to demonstrate that the generic stipulated program elements are committed for use, or the basis for the conclusions are applicable, at their plant. ②

1.4 SPECIFIC CONCLUSIONS FOR LICENSE RENEWAL

A summary of the specific conclusions for license renewal is provided in the following subsections. These conclusions are grouped as follows, based on the method of resolution and the corresponding IR section where they are resolved:

Types of Conclusion

- Age-related degradation mechanisms that are considered to be non-significant for any component (Summarized in 1.4.1, Details in Section 4).

① "Safety functions" too limited
② Ambiguous. Any exception?

- Age-related degradation mechanisms that are considered to be non-significant for specific components (Summarized 1.4.2, Details in Section 4).

- Potentially significant age-related degradation mechanisms for specific components that are effectively managed by effective programs (Summarized in 1.4.3, Details in Section 5).

- Potentially significant age-related degradation mechanisms for specific components that are subject to plant-specific management (Summarized in 1.4.4, Details in Section 6).

This PWR Containment Structures License Renewal Industry Report evaluated thirteen age-related degradation mechanisms applicable to six classes of structural materials or components. In general, the degradation mechanisms affect different components or groups of components differently. The evaluations in Sections 4, 5, and 6 address each degradation mechanism for all PWR containment components for which it is applicable. The following summary sections are also presented in this manner.

1.4.1 Non-Significant Degradation Mechanisms

Those age-related degradation mechanisms that are not significant to the ability of PWR containment components to perform their intended safety functions throughout the license renewal term are summarized in this section. For these age-related degradation mechanisms, specific criteria and corresponding justification for these conclusions are provided in Section 4 of this report that can be used as the basis for generic resolutions of age-related degradation mechanism/component issues. License renewal applicants intending to reference these generic conclusions are responsible for a review/evaluation of plant-specific features, including appropriate CLB documents/information, to assure that there are no deviations from the assumptions and criteria used in Section 4 of this report. The following age-related degradation mechanisms are non-significant for all applicable PWR containment structures: ①
②

- FREEZE-THAW (Section 4.1.1) is a non-significant age-related degradation mechanism for any PWR concrete containment structure that is located in a geographic region subject to "negligible" weathering conditions (i.e., a weathering index of less than 100 day-inches per year). Further, for any PWR concrete containment structure located in a geographic region subject to "moderate" to "severe" weathering conditions, freeze-thaw will not cause significant degradation because the concrete mix design (low permeability, proper air entrainment) ensures adequate protection. ③

③ Misleading; list assumptions and criteria here as needed
see page 4-7

① Inconsistent ; CLB maintenance

② use consistent words

- LEACHING OF CALCIUM HYDROXIDE (Section 4.1.2) is a non-significant age-related degradation mechanism for any PWR containment structure concrete that is not exposed to flowing water. Further, where PWR containment concrete is exposed to flowing water, because containment concrete is dense, well-cured, and therefore of low permeability, consistent with the guidance provided in ACI 201.2R, degradation caused by leaching of calcium hydroxide will not be significant. ①

- REACTIONS WITH AGGREGATES (Section 4.1.4) is a non-significant age-related degradation mechanism for any PWR concrete containment structure because either (1) the aggregate used was from geographic regions other than those known to yield potentially reactive aggregate or petrographic examination has shown that the aggregate used in the containment construction is non-reactive, or (2) if the aggregate is potentially reactive, the restrictions and requirements of ACI 201.2 for concrete mix design for use with potentially reactive aggregate, ~~or a justified equivalent,~~ were employed.

① Delete it.

- ELEVATED TEMPERATURE (Section 4.1.6) is a non-significant age-related degradation mechanism for the concrete portions of any PWR containment structure, because either normal bulk and local surface temperatures are below the threshold limits of 150 and 200F, respectively, or special provisions were made to offset the effects of elevated temperature. ②

  ELEVATED TEMPERATURE (Section 4.2.2) is a non-significant age-related degradation mechanism for containment reinforcing steel because the steel experiences temperatures well below the degradation threshold limit of 600F. ②

  ELEVATED TEMPERATURE (Section 4.3.2) is a non-significant age-related degradation mechanism for prestressing tendons because PWR containment prestressing tendons are not subjected to temperatures in excess of 140F.

  ELEVATED TEMPERATURE (Section 4.4.2) is a non-significant age-related degradation mechanism for PWR containment liners and free-standing steel containment shells because the temperatures to which they are subjected are well below their degradation threshold of 700F. ②

- IRRADIATION (Sections 4.1.7 and 4.2.3) is a non-significant age-related degradation mechanism for PWR containment structure concrete and reinforcing steel because the cumulative radiation exposures are below the degradation ② threshold limits of 10^{19} neutrons/cm² and 10^{10} rads for concrete and 10^{19} neutrons/cm² for reinforcing steel for ~~both~~ the ~~current and~~ license renewal term.

  Delete

  IRRADIATION (Section 4.3.3) is also a non-significant age-related degradation mechanism for prestressing tendons because the cumulative tendon radiation exposure throughout the license renewal term will be less than 4×10^{19} neutrons/cm², which has been shown to cause negligible tendon degradation.

① Use consistent words See p. 4-55

② Structural acceptance tests will not tell the difference

③ Incorrect statement ④ Loose words; be specific

IRRADIATION (Section 4.4.3) is a non-significant age-related degradation mechanism for PWR containment liners and free-standing steel PWR containment shells because the cumulative radiation exposure will remain below the degradation threshold of 2 x 10^{17} neutrons/cm^2. ①

- CONCRETE INTERACTION WITH ALUMINUM (Section 4.5.2) is a non-significant age-related degradation mechanism for any PWR concrete containment structure for which aluminum pipelines were not used during initial concrete placement. Further, where aluminum pipelines were used, because potentially significant adverse effects would have been identified and addressed during initial structural acceptance tests, degradation caused by concrete interaction with aluminum is non-significant. ②

- STRAIN AGING (Section 4.6.1) can be either static strain aging or dynamic stain aging. Dynamic strain aging is a non-significant age-related degradation mechanism for any free-standing steel containment structure because design loads do not exceed the elastic limits of the steel. ③ Static strain aging is a non-significant age-related degradation mechanism for any free-standing containment structure because the steel is not severely cold worked during forming. ④ Further, where severe cold working was employed during forming, but normalizing or stress relieving was performed afterwards, age-related degradation caused by static strain aging will not be significant.

1.4.2 Non-Significant Component/Degradation Mechanism Combinations

In addition to the age-related degradation mechanisms that are non-significant for any PWR containment structure or component, some degradation mechanisms are non-significant for selected PWR containment components. Criteria and corresponding justification for these conclusions are provided in Section 4 of this report that can be used as the basis for generic resolutions of age-related degradation mechanism/component issues. License renewal applicants intending to reference these generic conclusions are responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, to assure that there are no deviations from the assumptions and criteria used in Section 4 of this report. These non-significant PWR containment component/degradation mechanism combinations are summarized below:

- AGGRESSIVE CHEMICALS (Section 4.1.3) will not cause significant age-related degradation for PWR concrete containment structures and components that are not exposed to environmental conditions exceeding certain defined limits (pH < 5.5, > 500 ppm chlorides, ~~and~~ "or" > 1500 ppm sulfates). Further, intermittent exposure to environmental conditions exceeding the above defined limits will not cause significant degradation of the containment concrete. The following PWR containment concrete components are not exposed to the ⑤

⑤ Loose words

① Loose words
② Unclear & inconsistent, see p. 4-17 pH<11.5
③ Incorrect statement

environmental conditions associated with potentially significant aggressive chemical degradation, and require no further evaluation:

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade

Only below grade portions of PWR containment structures exposed to sustained aggressive environmental conditions as described above are subject to potentially significant degradation. These components require further evaluation (Sections 5.1, 6.1). ①

- CORROSION OF EMBEDDED STEEL AND REINFORCING STEEL (Sections 4.1.5 and 4.2.1) is a non-significant age-related degradation mechanism for PWR containment components that are not exposed to all of the necessary conditions for corrosion to occur: an aggressive environment (aqueous solution with pH > 11.5), a pathway (significant concrete cracking), and a supply of oxygen. The following PWR containment components are not exposed to the environmental conditions associated with potentially significant corrosion of embedded steel, and require no further evaluation: ②

  and/reinforcing ?

  Reinforced and Prestressed Concrete Containments

  1. Concrete dome
  2. Concrete containment wall above grade

  Only the embedded and reinforcing steel in those below grade portions of the containment structure that could be exposed to aggressive groundwater are susceptible to potentially significant corrosion degradation. These components require further evaluation (Sections 5.1, 6.1).

- LINER CORROSION (Section 4.4.1) is a non-significant age-related degradation mechanism for the PWR containment liner components and metallic common components in the following list because these components are not subjected to the tensile stress and corrosive environment necessary for stress corrosion cracking and are not exposed to aggressive groundwater. Thus the following PWR containment components are not subject to potentially significant corrosion degradation, and require no further evaluation: ③

  Reinforced and Prestressed Concrete Containments

  1. Containment liner interior surface
  2. Containment liner above grade exterior surface
  3. Basemat liner interior surface ④
  4. Liner anchors above grade

④ What if there is groundwater intrusion?

① Misleading
② Loose words

Common Components

1. Penetration sleeves
2. Dissimilar metal welds
3. Personnel airlock
4. Equipment hatches

Only the below grade metallic components of the containment structure that could be continuously exposed to aggressive groundwater are susceptible to potentially significant corrosion degradation. These components require further evaluation (Sections 5.4, 6.2). ①

- FATIGUE (Section 4.5.1) is a non-significant age-related degradation mechanism for the following PWR containment components because the design codes and standards that were used provide a good fatigue life, and because the projected number of cycles to be experienced by the structure is less than that considered in the design. No further evaluation is required for fatigue degradation of the following PWR containment components: ②

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade
3. Concrete containment wall below grade
4. Concrete basemat
5. Dome reinforcing steel
6. Containment wall reinforcing steel above grade
7. Containment wall reinforcing steel below grade
8. Basemat reinforcing steel

Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottom

1. Containment shell interior surface
2. Containment shell exterior surface

Free-Standing Steel Containment with Flat Bottom and an Ice Condenser

1. Dome shell interior surface
2. Dome shell exterior surface
3. Cylindrical shell interior surface
4. Cylindrical shell exterior surface
5. Concrete basemat
6. Basemat reinforcing steel

① Need to address administrative control requirements. See Comment #1, page 1-2.

Common Components

1. Personnel airlock
2. Equipment hatches

Localized cyclic thermal loadings can cause potentially significant fatigue degradation of penetration sleeves without bellows and penetration bellows for free-standing steel containments, which require further evaluation (Section 5.6).

- FREE-STANDING STEEL SHELL CORROSION (Section 4.6.2) is a non-significant age-related degradation mechanism for the following free-standing PWR steel containment components because these components are not subjected to the environmental conditions under which potentially significant corrosion could occur:

Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottoms

1. Containment shell interior surface
2. Containment shell exterior surface

Free-Standing Steel Containments with Flat Bottom and an Ice-Condenser

1. Dome shell interior surface
2. Dome shell exterior surface
3. Cylindrical shell interior surface
4. Cylindrical shell exterior surface

Common Components

1. Penetration bellows

Corrosion is potentially significant for embedded free-standing steel containment components, which require further evaluation (Sections 5.4, 6.2).

1.4.3 Potentially Significant Component/Degradation Mechanism Combinations Managed by Effective Programs

Another set of component/degradation mechanism combinations is generically resolved because effective programs for maintenance, inservice inspection, surveillance, testing, and analytical assessment are capable of managing the effects of potentially significant degradation. License renewal applicants intending to take credit for an effective program are responsible for the review/evaluation of appropriate plant-specific features, including appropriate CLB documents/information to assure that the program elements required to ①

① Testing procedure for structural degradation not part of Appendix J

② Reference to IWE/IWL not proper

manage the effects of potentially significant age-related degradation are committed for use at their plant.

- CONCRETE DEGRADATION DUE TO AGGRESSIVE CHEMICALS AND CORROSION OF REBAR AND EMBEDDED STEEL (Section 5.1) The degradation of accessible concrete surfaces, such as exterior surfaces above grade, is detectable and can be evaluated, based upon periodic inspection and testing procedures detailed in Appendix J to 10CFR50, and Subsections IWE and IWL to Section XI of the ASME Code. ① ②

  However, as indicated in Section 1.4.2, accessible concrete surfaces are not subject to potentially significant degradation due to aggressive chemicals or the corrosion of rebar and embedded steel. Below-grade or otherwise inaccessible regions that are not periodically inspected require additional aging management to control these component/degradation mechanism issues (Section 6.1). ③

- CORROSION OF PRESTRESSING SYSTEM (Section 5.2) can be identified and managed by established programs for visual inspection of tendon anchor heads, as well as periodic examination of the corrosion protection medium (grease) to ensure adequacy, as detailed in ASME Section XI Subsection IWL and Regulatory Guide 1.35. Thus, corrosion degradation of the following can be managed using these effective programs: ②

  Prestressed Concrete Containments

  1. Prestressing tendons

- PRESTRESSING LOSSES (Section 5.3) Progressive reductions in the levels of prestress can be detected by inspection and load monitoring programs, and the effects of the reduction evaluated for the license renewal term using the requirements of ASME Section XI, Subsection IWL and Regulatory Guide 1.35. Thus, the following components can be managed using effective programs for detecting prestressing losses: ②

  Prestressed Concrete Containments

  1. Prestressing tendons

- CORROSION OF CONCRETE CONTAINMENT LINERS AND FREE-STANDING STEEL CONTAINMENT SHELLS (Section 5.4) can be controlled through the use of coatings that protect the ferritic steel surfaces. Detection and management of general or localized corrosion degradation in accessible areas is provided by the inspection procedures of Section XI, Subsection IWE of the ASME Code and Type A integrated leak rate tests. ③ ④ ②

③ mitigate ?

④ Must satisfy EP criteria of 54.21(a)(6)

(1) Must satisfy EP criteria of 54.21(a)(6)

However, as indicated in Section 1.4.2, accessible areas of concrete containment liners and free-standing steel containment shells are not subject to potentially significant age-related corrosion degradation. Below-grade and other regions that are not readily accessible require additional consideration (Section 6.2).

- SETTLEMENT (Section 5.5) can be identified and mitigative measures taken before settlement of the following PWR containment structures becomes significant using effective settlement monitoring techniques: (1)

  Reinforced and Prestressed Concrete Containments

  1. Concrete basemat

  Free-Standing Steel Containment with a Flat Bottom and an Ice-Condenser

  1. Concrete basemat

- FATIGUE OF HOT PENETRATIONS WITHOUT BELLOWS AND PENETRATION BELLOWS ASSEMBLIES (Section 5.6) Effective examination and fatigue evaluation procedures are available that can be used to evaluate the fatigue design basis for penetrations and bellows assemblies to determine the significance of continued cyclic loading beyond the initial license term, thereby providing a means of controlling fatigue degradation of the following components:

  Common Components

  1. Penetration sleeves
  2. Penetration bellows

1.4.4 Potentially Significant Component/Degradation Mechanism Combinations Requiring Plant-Specific Management

There is a final set of PWR containment component/degradation mechanism combinations which has been found to be potentially significant, and for which effective programs do not manage the degradation. Applicants for license renewal must perform a plant-specific evaluation for these component/degradation mechanism combinations. Recommendations for aging management options for these issues are provided in the Section 6 subsections indicated below.

- AGGRESSIVE CHEMICALS (Section 6.1) can cause potentially significant degradation of below-grade portions of concrete containments; CORROSION of rebar or embedded steel in below-grade concrete components that are in contact with the soil may also experience potentially significant degradation. Aging management options for addressing these component/degradation mechanism

(1) Change "or" to "and."
(2) Loose words.

combinations include monitoring of groundwater levels and chemical, analysis of groundwater and soil. The following components require plant-specific aging management resolution for these degradation mechanisms:

Reinforced and Prestressed Concrete Containments

1. Concrete containment wall below grade
2. Concrete basemat
3. Containment wall reinforcing steel below grade
4. Basemat reinforcing steel

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat
2. Basemat reinforcing steel

- CORROSION OF INACCESSIBLE OR BELOW GRADE REGIONS OF STEEL LINERS OR FREE-STANDING CONTAINMENT SHELLS (Section 6.2) - The corrosion of those portions of steel liner steel free-standing containment shells in inaccessible or below-grade areas have been identified as a plant-specific age-related degradation mechanism which cannot be shown to be adequately controlled for the extended license term by established procedures. A phased inspection program is described in the report as an option for managing this issue. Plant-specific evaluations are necessary for the following components: (1) (2)

managed ? (1)

Reinforced and Prestressed Concrete Containments

1. Containment liner below grade exterior surface
2. Basemat liner exterior surface
3. Liner anchors below grade

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Embedded shell region
2. Basemat liner
3. Liner anchors

1.5 CONCLUSIONS FOR PWR CONTAINMENT STRUCTURES

Age-related degradation of PWR containment structural components is limited and manageable. Inservice inspection, surveillance, testing, and analytical assessment are capable of managing the effects of potentially significant degradation for many of these components. Aging management of potentially significant degradation/component issues that are not controlled by effective programs will be developed on an as-needed, plant-specific basis. Table 1-1 provides a summary of the age-related degradation

managed ?

mechanism/component issues evaluated in this Industry Report. The subsections in which applic ble issues are resolved are indicated in the table.

TABLE 1-1

SUMMARY OF PWR CONTAINMENT STRUCTURAL COMPONENTS/AGE-RELATED DEGRADATION MECHANISMS

CONTAINMENT COMPONENTS	DEGRADATION MECHANISM												
	FREEZE-THAW	LEACHING OF CALCIUM HYDROXIDE	AGGRESSIVE CHEMICALS	REACTION WITH AGGREGATES	CORROSION OF EMBEDDED STEEL	ELEVATED TEMPERATURE	IRRADIATION	CORROSION	PRESTRESSING LOSSES	FATIGUE	CONCRETE CONTRACTION W/ ALUMINUM	SETTLEMENT	STRAIN AGING
CONCRETE CONTAINMENTS													
REINFORCED/PRESTRESSED													
• CONCRETE DOME	4.1.1	4.1.2	4.1.3	4.1.4	4.1.5	4.1.6	4.1.7	–	–	4.5.1	4.5.2	–	–
• CONCRETE CONTAINMENT WALL ABOVE GRADE	4.1.1	4.1.2	4.1.3	4.1.4	4.1.5	4.1.6	4.1.7	–	–	4.5.1	4.5.2	–	–
• CONCRETE CONTAINMENT WALL BELOW GRADE	4.1.1	4.1.2	8.1	4.1.4	8.1	4.1.6	4.1.7	–	–	4.5.1	4.5.2	–	–
• CONCRETE BASEMAT	4.1.1	4.1.2	8.1	4.1.4	8.1	4.1.6	4.1.7	–	–	4.5.1	4.5.2	5.5	–
• CONTAINMENT LINER INTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.4.1	–	–	–	–	–
• CONTAINMENT LINER ABOVE GRADE EXTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.4.1	–	–	–	–	–
• CONTAINMENT LINER BELOW GRADE EXTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	8.2	–	–	–	–	–
• BASEMAT LINER INTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.4.1	–	–	–	–	–
• BASEMAT LINER EXTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	8.2	–	–	–	–	–
• LINER ANCHORS ABOVE GRADE	–	–	–	–	–	4.4.2	4.4.3	4.4.1	–	–	–	–	–
• LINER ANCHORS BELOW GRADE	–	–	–	–	–	4.4.2	4.4.3	8.2	–	–	–	–	–
• DOME REINFORCING STEEL	–	–	–	–	–	4.2.2	4.2.3	4.2.1	–	4.5.1	–	–	–
• CONTAINMENT WALL REINFORCING STEEL ABOVE GRADE	–	–	–	–	–	4.2.2	4.2.3	4.2.1	–	4.5.1	–	–	–
• CONTAINMENT WALL REINFORCING STEEL BELOW GRADE	–	–	–	–	–	4.2.2	4.2.3	8.1	–	4.5.1	–	–	–
• BASEMAT REINFORCING STEEL	–	–	–	–	–	4.2.2	4.2.3	8.1	–	4.5.1	–	–	–
PRESTRESSED ONLY													
• PRESTRESSING TENDONS	–	–	–	–	–	4.3.2	4.3.3	5.2	5.5	–	–	–	–

LEGEND

– = NOT APPLICABLE

4.X.X = NOT A SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM (ARDM). SUBSECTION NUMBER WHERE ARDM IS EVALUATED

5.X = POTENTIALLY SIGNIFICANT ARDM CONTROLLED BY AN EFFECTIVE PROGRAM. SUBSECTION IN WHICH PROGRAM IS DESCRIBED

8.X = POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM. SUBSECTION IN WHICH AGING MANAGEMENT OPTIONS ARE PROVIDED

TABLE 1.1

SUMMARY OF PWR CONTAINMENT STRUCTURAL COMPONENTS/AGE-RELATED DEGRADATION MECHANISMS

CONTAINMENT COMPONENTS	DEGRADATION MECHANISM												
	FREEZE-THAW	LEACHING OF CALCIUM HYDROXIDE	AGGRESSIVE CHEMICALS	REACTION WITH AGGREGATES	CORROSION OF EMBEDDED STEEL	ELEVATED TEMPERATURE	IRRADIATION	CORROSION	PRESTRESSING LOSSES	FATIGUE	CONCRETE INTERACTION W/ ALUMINUM	SETTLEMENT	[illegible] AGING
FREE-STANDING CYLINDRICAL AND SPHERICAL STEEL CONTAINMENTS WITH ELLIPTICAL BOTTOM													
∘ CONTAINMENT SHELL INTERIOR SURFACE	-	-	-	-	-	4.4.2	4.4.3	[illegible]	-	4.5.1	-	-	4.8.1
∘ CONTAINMENT SHELL EXTERIOR SURFACE	-	-	-	-	-	4.4.2	4.4.3	4.8.2	-	4.5.1	-	-	4.8.1
∘ EMBEDDED SHELL REGION	-	-	-	-	-	4.4.2	4.4.3	8.2	-	-	-	-	4.8.1
∘ SAND POCKET REGION	-	-	-	-	-	4.4.2	4.4.3	8.2	-	-	-	-	4.8.1
FREE-STANDING STEEL CONTAINMENT WITH FLAT BOTTOM AND AN ICE-CONDENSER													
∘ DOME SHELL INTERIOR SURFACE	-	-	-	-	-	4.4.2	4.4.3	4.8.2	-	4.5.1	-	-	4.8.1
∘ DOME SHELL EXTERIOR SURFACE	-	-	-	-	-	4.4.2	4.4.3	4.8.2	-	4.5.1	-	-	4.8.1
∘ CYLINDRICAL SHELL INTERIOR SURFACE	-	-	-	-	-	4.4.2	4.4.3	4.8.2	-	4.5.1	-	-	4.8.1
∘ CYLINDRICAL SHELL EXTERIOR SURFACE	-	-	-	-	-	4.4.2	4.4.3	4.8.2	-	4.5.1	-	-	4.8.1
∘ EMBEDDED SHELL REGION	-	-	-	-	-	4.4.2	4.4.3	8.2	-	-	-	-	-
∘ BASEMAT LINER	-	-	-	-	-	4.4.2	4.4.3	8.2	-	-	-	-	-
∘ LINER ANCHORS	-	-	-	-	-	4.4.2	4.4.3	8.2	-	-	-	-	-
∘ CONCRETE BASEMAT	4.1.2	4.1.3	8.1	4.1.4	8.1	4.1.8	4.1.7	-	-	4.5.1	4.5.2	5.5	-
∘ BASEMAT REINFORCING STEEL	-	-	-	-	-	4.2.2	4.2.3	8.1	-	4.5.1	-	-	-
COMMON COMPONENTS													
∘ PENETRATION SLEEVES	-	-	-	-	-	4.4.2	4.4.3	4.8.1	-	5.6	-	-	4.8.1
∘ DISSIMILAR METAL WELDS	-	-	-	-	-	-	-	4.8.1	-	-	-	-	-
∘ PENETRATION BELLOWS	-	-	-	-	-	-	4.4.3	4.8.2	-	5.6	-	-	4.8.1
∘ PERSONNEL AIRLOCK	-	-	-	-	-	-	4.4.3	4.8.1	-	4.5.1	-	-	4.8.1
∘ EQUIPMENT HATCHES	-	-	-	-	-	-	4.4.3	4.8.1	-	4.5.1	-	-	4.8.1

LEGEND

\- = NOT APPLICABLE

4.X.X = NOT A SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM (ARDM), SUBSECTION NUMBER WHERE ARDM IS EVALUATED

5.X = POTENTIALLY SIGNIFICANT ARDM CONTROLLED BY AN EFFECTIVE PROGRAM, SUBSECTION IN WHICH PROGRAM IS DESCRIBED

8.X = POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM, SUBSECTION IN WHICH AGING MANAGEMENT OPTIONS ARE PROVIDED

(1) "safety functions" is too limited.

SECTION 2
INTRODUCTION

2.1 BACKGROUND

Previous studies have shown the PWR containment structure to be one of the critical elements to be considered for license renewal [1, 2, 3, 4, 5]. Further pilot plant studies on PWR containment [6, 7] demonstrated the technical feasibility of extended life for PWR containment structure components. The combination of this previous work confirmed the need for a generic assessment of PWR containment structures for license renewal.

2.2 PURPOSE

This industry report (IR) builds upon these studies in order to provide a generic technical basis for license renewal of PWR containment structure components. The generic evaluations have the objective of demonstrating the capability of systems, structures and components to continue to perform their intended safety functions throughout the license renewal term. (1)

2.3 SCOPE

This document provides guidance that can be used by license renewal applicants to address age-related degradation of PWR concrete and free-standing steel containment structures. Both post-tensioned and conventionally reinforced concrete containment designs are included. For post-tensioned systems, tendons and anchorage hardware are included. In all cases, the scope includes those elements required to maintain the pressure boundary following a postulated design basis accident. Containment internal structures are discussed separately in the IR on Class I structures, as are the concrete shield structures associated with free-standing steel containments.

The scope does not include components subject to routine replacement during normal plant maintenance activities. In addition, reactor vessel and other equipment supports are excluded from this report. Also excluded from the scope of this IR are piles, grouted tendon systems, and rock anchors since they are not used regularly enough to warrant generic consideration. Any plant-unique containment features that are not evaluated in this IR need to be addressed by the individual license renewal applicant.

The report does not consider penetrations of concrete containments as unique elements of the containment, but rather as subcomponents of the liner that provide an interface for electrical and piping systems passing through the structure. Structural components specifically addressed are listed in Table 2-1.

2.4 AGE-RELATED DEGRADATION MECHANISMS

The age-related degradation mechanisms covered in this report, and the PWR containment components they affect are listed below.

Concrete
- Freeze-thaw
- Leaching of calcium hydroxide
- Aggressive chemicals
- Reactions with aggregates
- Corrosion of embedded steel
- Elevated temperature
- Irradiation

Reinforcing Steel
- Corrosion
- Elevated temperature
- Irradiation

Prestressing System
- Corrosion
- Elevated temperature
- Irradiation
- Prestressing losses

Liner
- Corrosion
- Elevated temperature
- Irradiation

Miscellaneous Age-Related Degradation Mechanisms
- Fatigue
- Concrete interaction with aluminum
- Settlement

Free-Standing Steel Containment Degradation Mechanisms
- Strain aging
- Corrosion

These mechanisms were identified from a review of nuclear power plant operating experience, relevant laboratory data, and related experience in other industries.

2.5 EVALUATION APPROACH

The PWR containment components that are within the scope of this report are evaluated with respect to the listed age-related degradation mechanisms in a four-step process. First, the evaluation basis is defined in Section 3 of this report by describing the component design features (Section 3.1), their design basis (Section 3.2), and their operational history

Table 2-1
PWR CONTAINMENT COMPONENTS

Concrete Containment
(Reinforced or Prestressed)

- Concrete dome
- Concrete containment wall above grade
- Concrete containment wall below grade
- Concrete basemat
- Containment liner interior surface
- Containment liner above grade exterior surface
- Containment liner below grade exterior surface
- Basemat liner interior surface
- Basemat liner exterior surface
- Liner anchors above grade
- Liner anchors below grade
- Dome reinforcing steel
- Containment wall reinforcing steel above grade
- Containment wall reinforcing steel below grade
- Basemat reinforcing steel

Prestressed Concrete Containment

- Prestressing tendons and ducts

Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottom

- Containment shell interior surface
- Containment shell exterior surface
- Embedded shell region
- Sand pocket region

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

- Dome shell interior surface
- Dome shell exterior surface
- Cylindrical shell interior surface
- Cylindrical shell exterior surface
- Embedded shell region
- Basemat liner
- Liner anchors
- Concrete basemat
- Basemat reinforcing steel

Common Components

- Penetration sleeves
- Dissimilar metal welds
- Penetration bellows
- Personnel airlock
- Equipment hatches

(1) "Safety functions" is too limited
(2) See comment #1, Page 1-2

(Section 3.3). The latter includes an assessment of the generic applicability of reported service experience.

Second, the age-related degradation mechanisms are described and evaluated generically in Section 4 of the report with respect to their potential significance to the continued safety function performance of PWR containment structure components during the license renewal term. (An age-related degradation mechanism is defined to be significant for a component if when allowed to continue without any additional prevention or mitigation measures, it cannot be shown that the component would continue to maintain its intended safety functions during the license renewal term.) If it can be shown that the component is either not susceptible to the age-related degradation mechanism under consideration, or is susceptible to such a small degree that the component safety function performance is not affected throughout the license renewal term, then the component/degradation mechanism combination is not significant. Otherwise, the component/degradation mechanism combination is potentially significant. (1) (1) (1)

Third, potentially significant component/degradation mechanism combinations are generically evaluated in Section 5 of the report with respect to the capability and corresponding basis of effective programs of inspection, testing, maintenance, surveillance, and analytical assessment to manage the effects of the age-related degradation. If a potentially significant component/degradation mechanism combination is adequately addressed by effective programs, then the issue is considered to be resolved on the basis that the degradation is managed acceptably. Combinations of mechanisms and components for which effective programs cannot be shown to manage potentially significant age-related degradation require plant-specific evaluation. (2)

Finally, recommendations for plant-specific management options for significant age-related degradation are provided in Section 6 of this report, if required.

A license renewal applicant intending to take credit for a particular conclusion of this report is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, to assure that the assumptions and criteria used as the basis for the determination of insignificance in Section 4 of this report are applicable to the component under consideration. Similarly, the license renewal applicant is responsible for the review/evaluation of their plant specific features, including appropriate CLB documents/information, in order to assure that the program elements used to manage the effects of potentially significant degradation in Section 5 of this report, ~~or their justified equivalent,~~ are committed for use at their plant. Plant-specific evaluations are required for any age-related degradation/component issues for which the license renewal applicant is Del unable to demonstrate that the generic stipulated program elements are committed for use, or the bases for the conclusions are applicable, at their plant.

(1) EPRI reports not in PDR.

2.6 REFERENCES

1. C. A. Negin et al. Extended Life Operation of Light Water Reactors: Economic and Technological Review, EPRI NP-2418, Vols 1 and 2, Electric Power Research Institute, Palo Alto, CA, June, 1982.

2. I. Spiewak and R. S. Livingston, The Longevity of Nuclear Power Systems, EPRI NP-4208, Electric Power Research Institute, Palo Alto, Calif., August 1985.

3. D. J. Naus, Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants, NUREG/CR-4652, ORNL/TM-10059, Oak Ridge National Laboratory, Oak Ridge, Tenn., September 1986.

4. B. M. Morris and J. P. Vora, Nuclear Plant Aging Research (NPAR) Program Plan, NUREG-1144, Division of Engineering Technology, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C., July 1985.

5. N. Prasad and R. Orr, "Concrete Degradation Monitoring and Evaluation," pp. 84-88 in Proceedings of the International Nuclear Power Plant Aging Symposium, NUREG/CP-0100, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, D.C., August 30-September 1, 1988.

6. Virginia Power Co., Northern States Power Co., and Grove Engineering Inc., LWR Plant Life Extension, EPRI NP-5002, Electric Power Research Institute, Palo Alto, CA, January, 1987.

7. Virginia Power Co., Grove Engineering Inc., Westinghouse Electric Corp., Stone & Webster Engineering Corp., and Temple, Barker, and Sloan, Inc., PWR Pilot Plant Life Extension Study at Surrey Unit 1: Phase I, EPRI NP-5289, Electric Power Research Institute, Palo Alto, CA, July, 1987.

(1) Incorrect statement ; also see p. 4-44

SECTION 3
COMPONENT EVALUATION BASIS

This section of the report provides the age-related degradation evaluation basis for PWR containment structure components to support issuance of a renewed license. This evaluation basis consists of the design features of the components (Section 3.1), including design function, material specifications, and different approaches used by the architect engineers responsible for the designs; the codes, standards, and regulations that govern the construction and operations of the components (Section 3.2); and relevant items related to the operating history of PWR containment components (Section 3.3).

3.1 DESIGN FEATURES

3.1.1 Design Evolution

The containment consists of those elements required to maintain the integrity of the pressure boundary following a postulated design basis event.

Prior to 1965, containments for plants between 50 MWe and 400 MWe consisted of steel vessels, either (1) free-standing steel cylinders with hemispherical domes and elliptical or flat bottoms (Figures 3-1 and 3-2) [1], or (2) steel spheres (Figure 3-3) [2]. Table 3-1 identifies the plants with free-standing steel containments; the type of design, that is, cylindrical or spherical; and the type of pressure suppression.

As plant size increased to around 800 MWe, the increased metal thickness and the need for post-weld heat treatment began to influence the design of steel containments. Thus, in the mid-1960s, some designs changed to composite, steel lined, reinforced concrete containments [1]. These typically consisted of 10 ft thick base mats, 4 ft-6 in. thick cylindrical walls and 2 ft-6 in. to 3 ft-6 in. thick hemispherical domes, with pressure retaining capability provided by a steel liner of varying thickness (1/4 to 1/2 in.). (1) Concrete design strength varied between 3000 and 5000 psi with reinforcing steel having yield strengths of 40,000, 50,000 and 60,000 psi. (See Figures 3-4 and 3-5.)

Prestressed containments came into being in the late 1960s. Two of these (Ginna and Robinson 2) were prestressed in only the vertical direction. Fully prestressed containments developed in three phases; these are depicted in Figure 3-6.

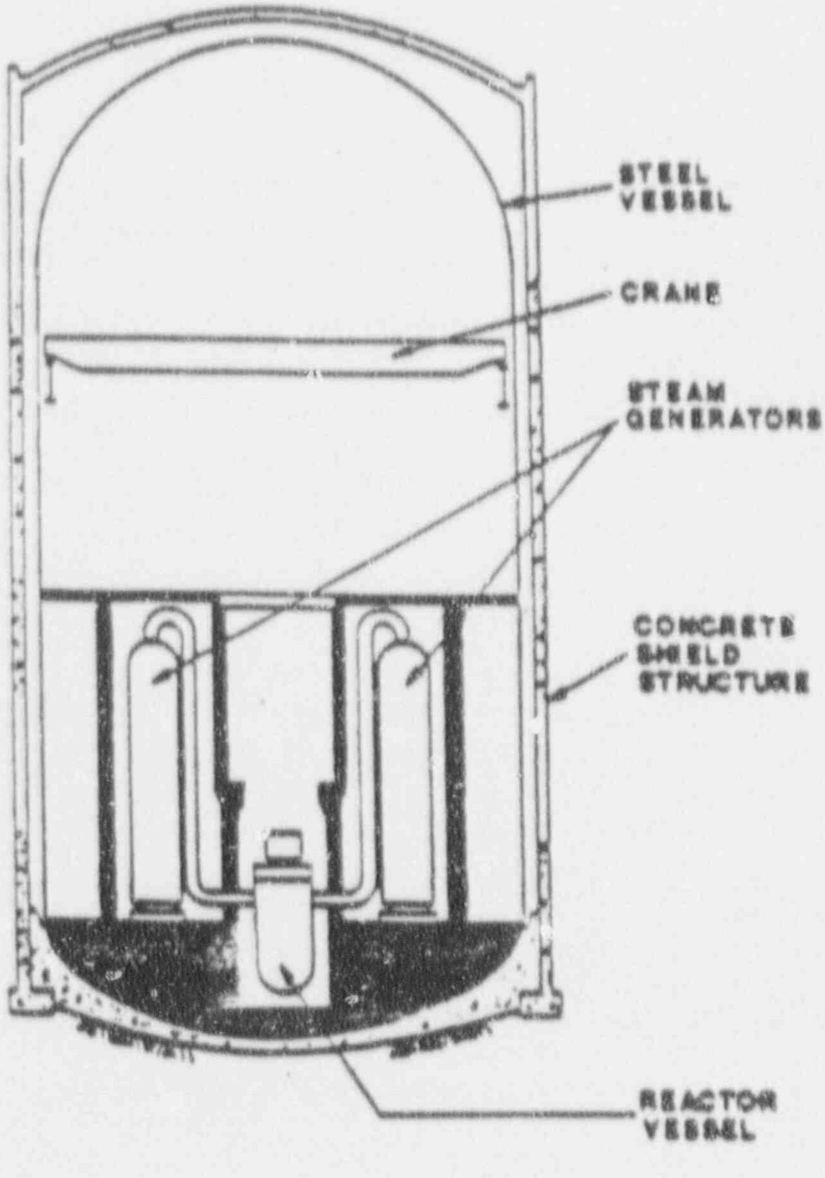


Figure 3-1. Free-Standing PWR Steel Containment with Elliptical Bottom

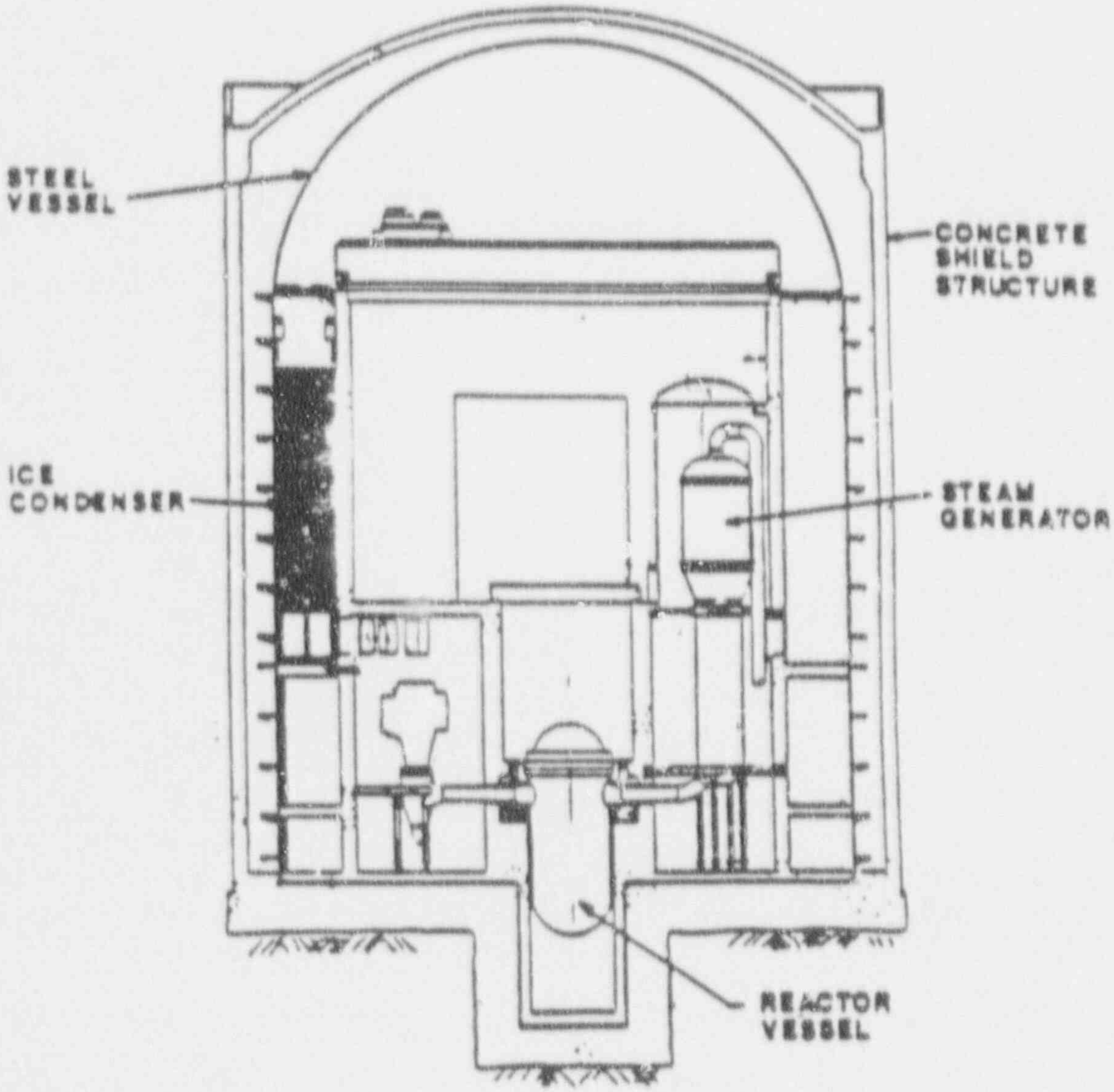


Figure 3-2. Free-Standing PWR Steel Containment with Flat Bottom and an Ice Condenser

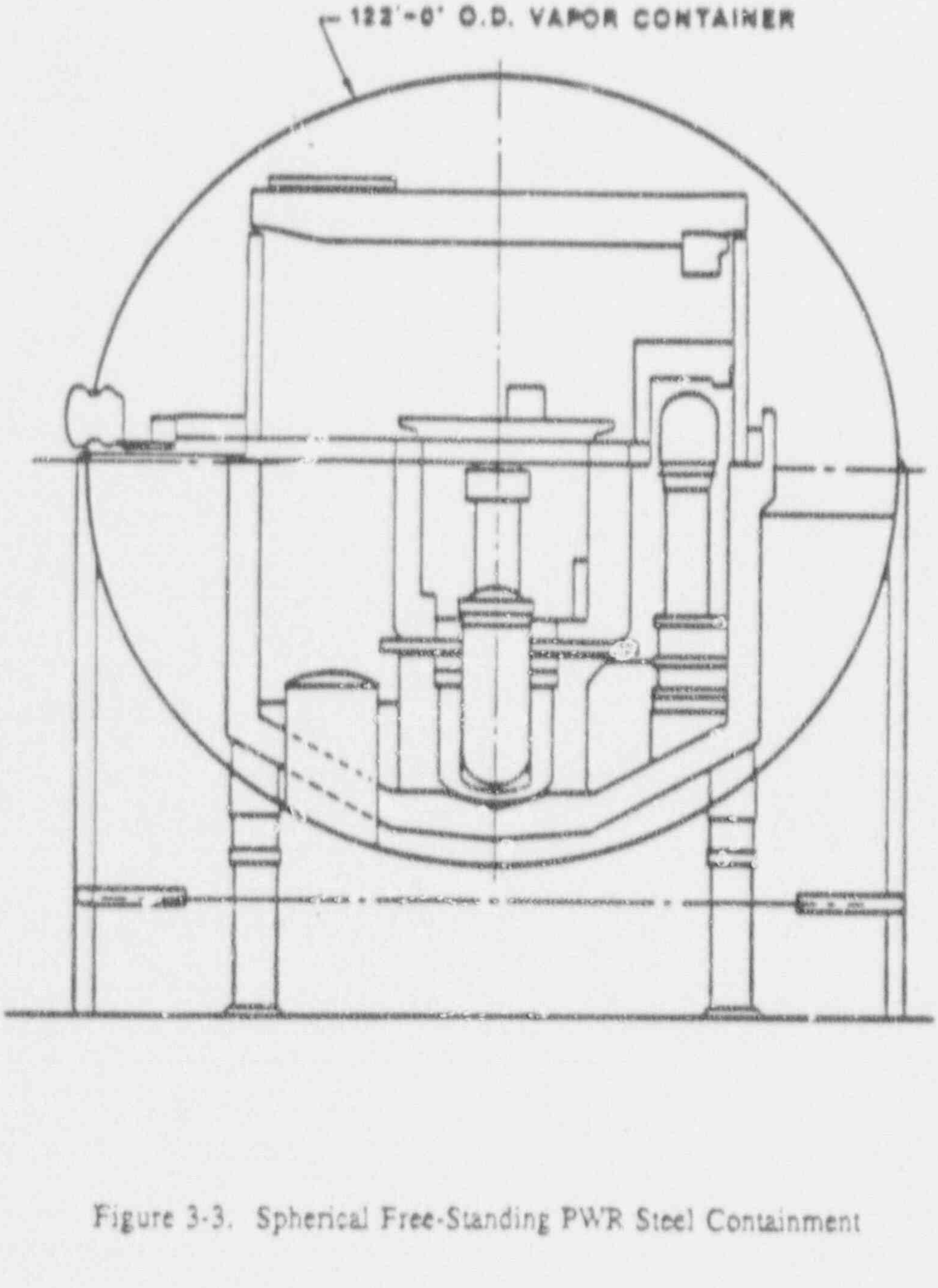


Figure 3-3. Spherical Free-Standing PWR Steel Containment

Table 3-1

FREE-STANDING STEEL CONTAINMENTS

Plant	Design	Suppression
San Onofre 1	Sphere	Spray
Yankee Rowe	Sphere	Passive
St. Lucie 1 & 2	Cylinder	Spray
Waterford 3	Cylinder	Spray
Prairie Island 1 & 2	Cylinder	Spray
Davis-Besse	Cylinder	Spray
WNP-3	Cylinder	Spray
Kewaunee	Cylinder	Spray
Sequoyah 1 & 2	Cylinder	Vapor/Spray
Watts Bar 1 & 2	Cylinder	Vapor/Spray
McGuire 1 & 2	Cylinder	Vapor/Spray
Catawba 1 & 2	Cylinder	Vapor/Spray

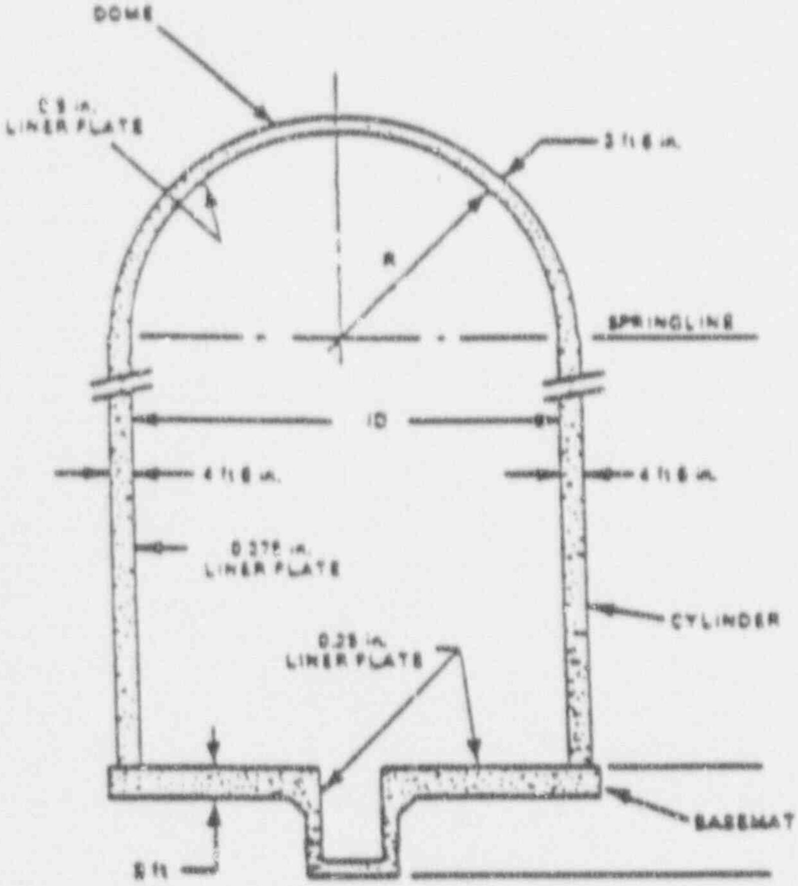


Figure 3-4. Typical Steel-Lined Reinforced Concrete Containment
(Source: See Reference 3)

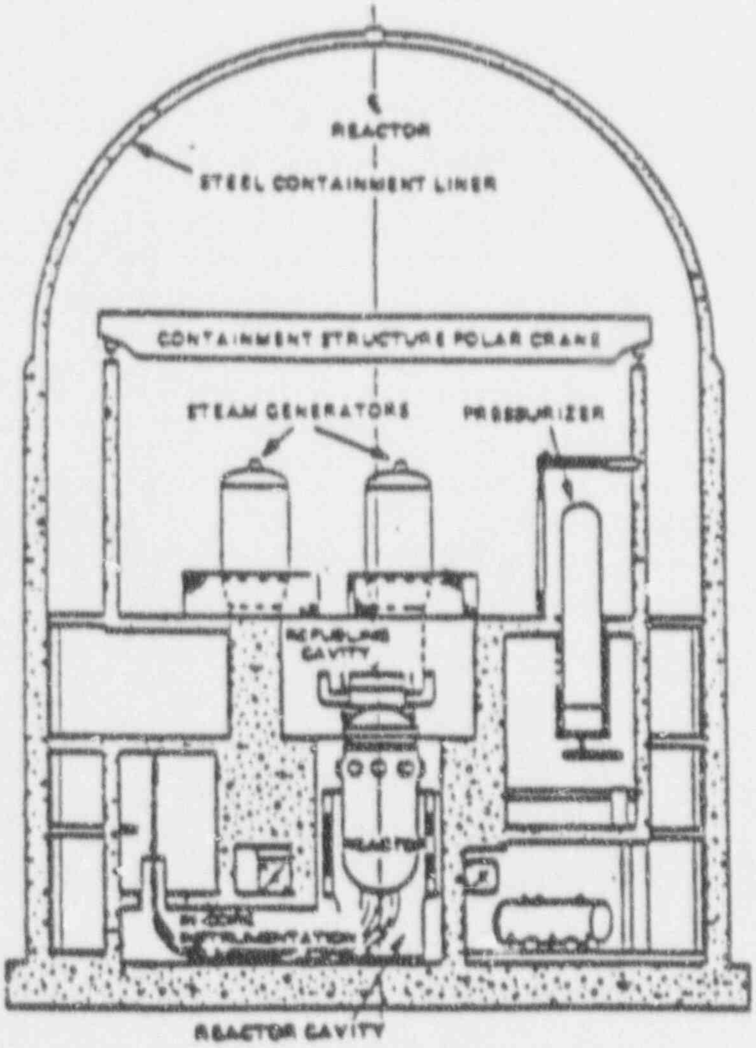


Figure 3-5. Typical Reinforced Concrete Containment
(Source: See Reference 3)

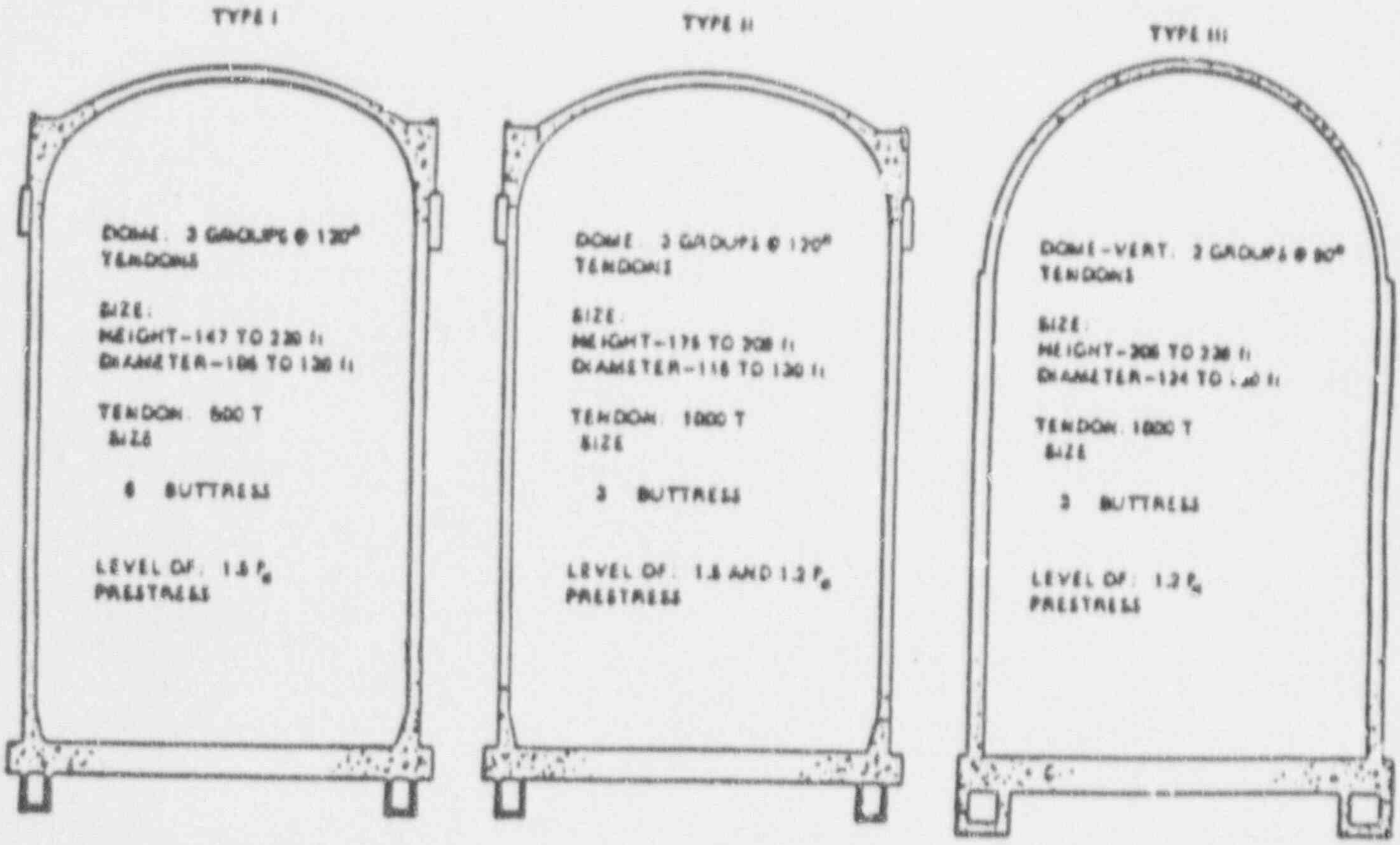


Figure 3-6. Generic Types of Prestressed Concrete Containments

The initial phase for prestressed containments consisted of 2 ft-6 in. thick ellipsoidal domes, a flat base mat generally between 8 and 10 ft thick and a 3 ft-6 in. thick cylindrical wall with a large ring girder (Figure 3-7) at the junction of the wall top and the dome [1]. Prestressing was provided vertically from the underside of the base mat extending through the cylindrical wall with anchorage at the top of the ring girder. Hoop prestressing was accomplished using a six buttress configuration with individual hoop tendons anchored between 120° segments or at every second buttress (Figure 3-8). Dome prestressing was provided by three-directional placement of tendons at 120° intervals. Tendons consisted of assemblies having 90-1/4 in. diameter wires. Reinforcing steel of 40,000 and 60,000 psi yield strength was also used, along with concrete design strengths between 4,000 and 5,500 psi.

Due to the large number of tendons involved and the normal evolutionary growth in plant size, a second phase of prestressed containment evolved. The number of buttresses was reduced from 6 to 3, with hoop tendons anchored at 240° intervals. Tendon capacity also doubled with the use of assemblies having 180-1/4 in. diameter wires or 55-1/2 in. diameter seven-wire strand.

In the third stage of development, the ellipsoidal dome was replaced by the hemispherical dome. Instead of vertical tendons being anchored at the top of the ring girder, one continuous tendon was placed up one side of the wall, across the hemispherical dome and down the opposite side. This U-shaped vertical tendon concept resulted in two groups of tendons oriented 90° apart and eliminated the ring girder. Hoop tendons remained in the 3 buttress configuration.

3.1.2 General Design Features

There are three types of PWR containment suppression systems: passive suppression, spray suppression, or vapor suppression. These containment types are designed to resist the pressure resulting from the energy released during a postulated loss of coolant accident. Passive suppression containments provide for reducing the accident pressure and temperature by convection to and conduction through the containment wall and by the heat sink of the internal reinforced concrete structures. Some passive suppression containments operate at subatmospheric pressure to reduce the resultant pressure after a postulated accident. Spray suppression containments provide a spray system that reduces the accident pressure and temperature by spraying water into the containment environment, thus providing a condensing action. Vapor suppression containments provide a passive means to reduce the accident pressure by first passing the escaped steam through

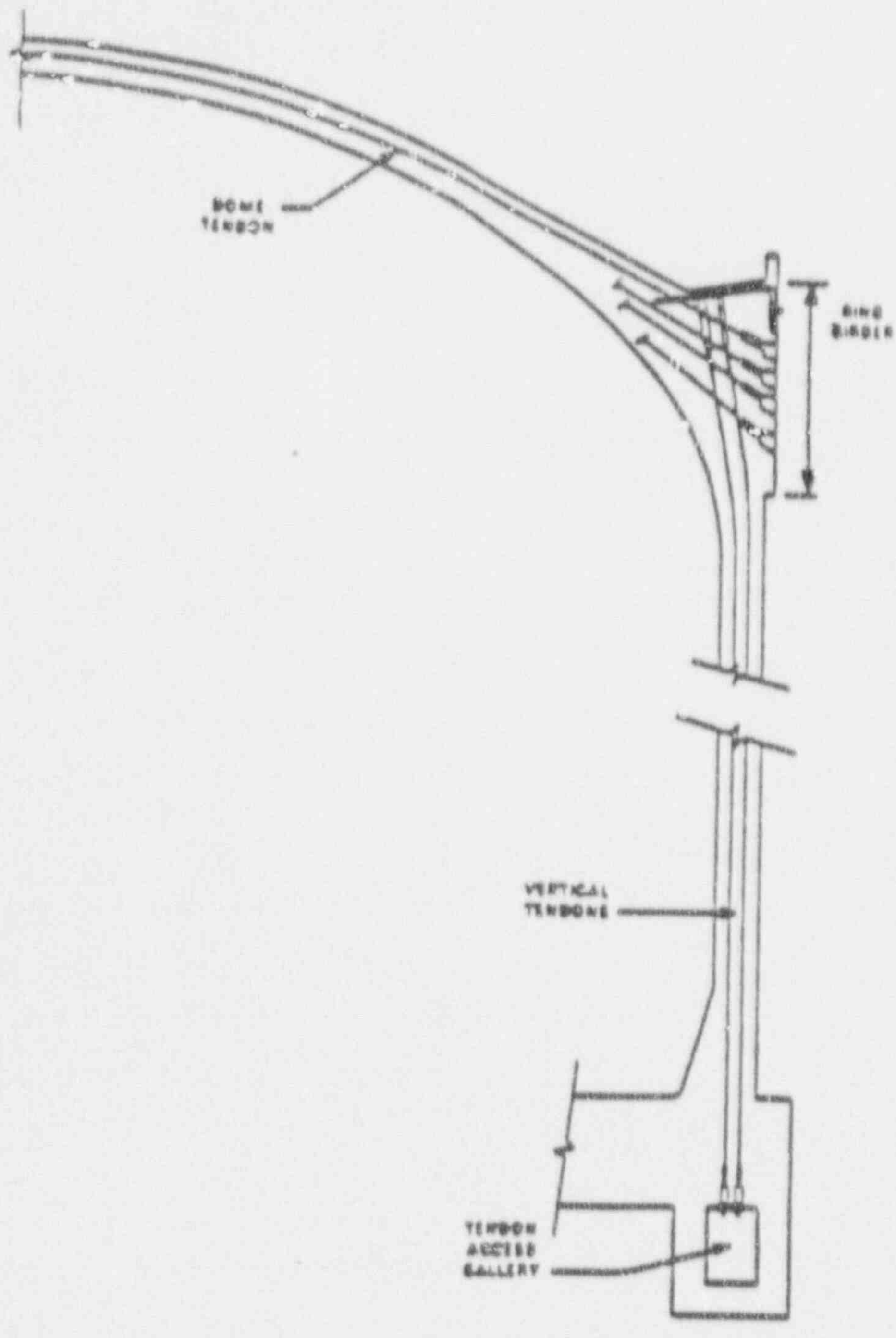
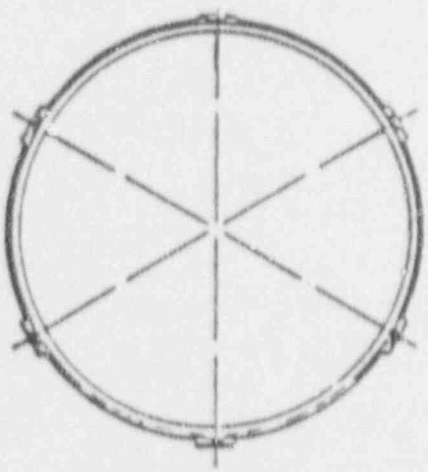
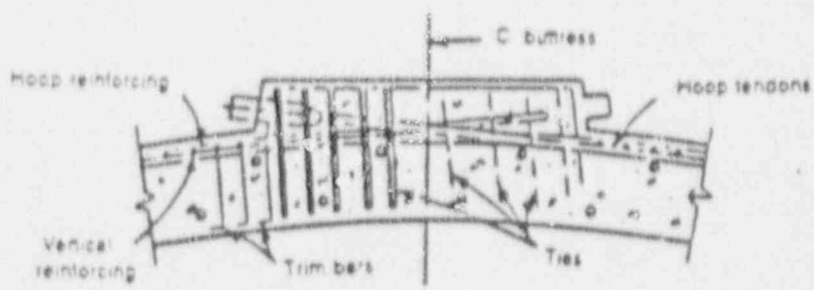


Figure 3-7. Typical PWR Prestressed Containment
With Ring-Girder Design



PLAN



TYPICAL BUTTRESS REINFORCING PLAN

Figure 3-8. Six Buttress Arrangement
(Source: See Reference 2)

(1) Incorrect statement; also see p. 4-4

ice filled compartments (Figure 3-9). Vapor suppression containments have approximately one-half of the free volume of a typical dry (passive suppression) containment.

3.1.3 Reinforced Concrete Containment Design Features

The reinforced concrete containment consists of a vertical cylinder, a hemispherical dome, and a flat base slab with bonded reinforcing steel. Pressure retaining capability is provided by a steel membrane that is anchored to the concrete with shear connectors (studs) or other rolled shapes (angles or tees). The liner plate resists structural loads, but is not considered to reduce the containment's reinforcing requirements for any of the design loads, even though it does contribute to the ultimate containment capacity.

3.1.4 Prestressed Concrete Containment Design Features

A prestressed containment is similar to a reinforced containment with vertical cylinder, flat base slab, and convex dome. The dome can be hemispherical, torispherical, or ellipsoidal. The vertical cylinder and dome are prestressed; the base slab employs bonded reinforcing steel and is incidentally prestressed only in the area where wall tendons pass through the mat (Figure 3-10). Prestressed containments also contain bonded reinforcing steel to control cracking and accommodate bending, shear, and temperature stresses. As in a reinforced concrete containment, the steel liner provides the pressure boundary.

3.1.5 Free-Standing Steel Containment Design Features

The free-standing steel containments consist of continuously welded steel plates that have the configurations shown in Figures 3-1, 3-2, and 3-3. The steel plates provide the pressure boundary and structural capability to withstand both the accident conditions and the structural loads imposed by equipment, piping systems, and supports. The cylindrical steel containments (Figures 3-1 and 3-2) are supported by reinforced concrete mats, whereas the spherical containment may be supported by steel columns attached at the mid-height of the sphere (Figure 3-3) or may be set into the ground on a concrete foundation with sand fill.

The cylindrical containments have a missile shield building which protects the free standing steel from external environmental conditions. The spherical containment may be exposed to external environmental conditions, which are included in the design considerations.

The concrete shield structure is not directly linked to the free-standing steel containment, except at piping penetration areas. The piping penetrations are provided with bellows to

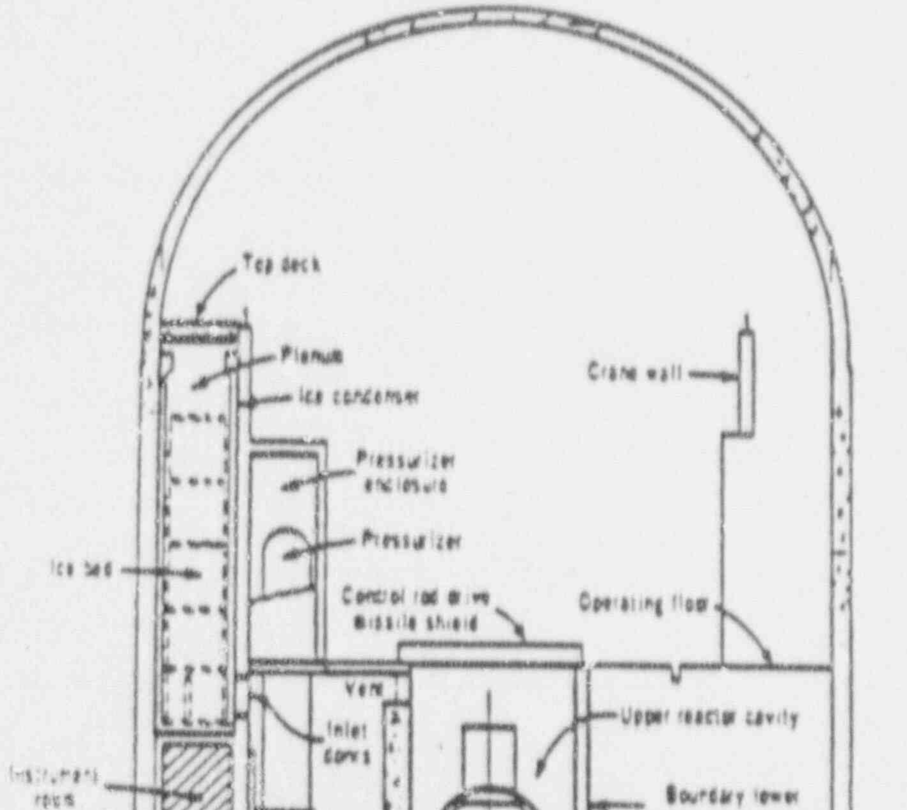


Figure 3-9. Ice Condenser Concrete Containment
(Source: See Reference 2)

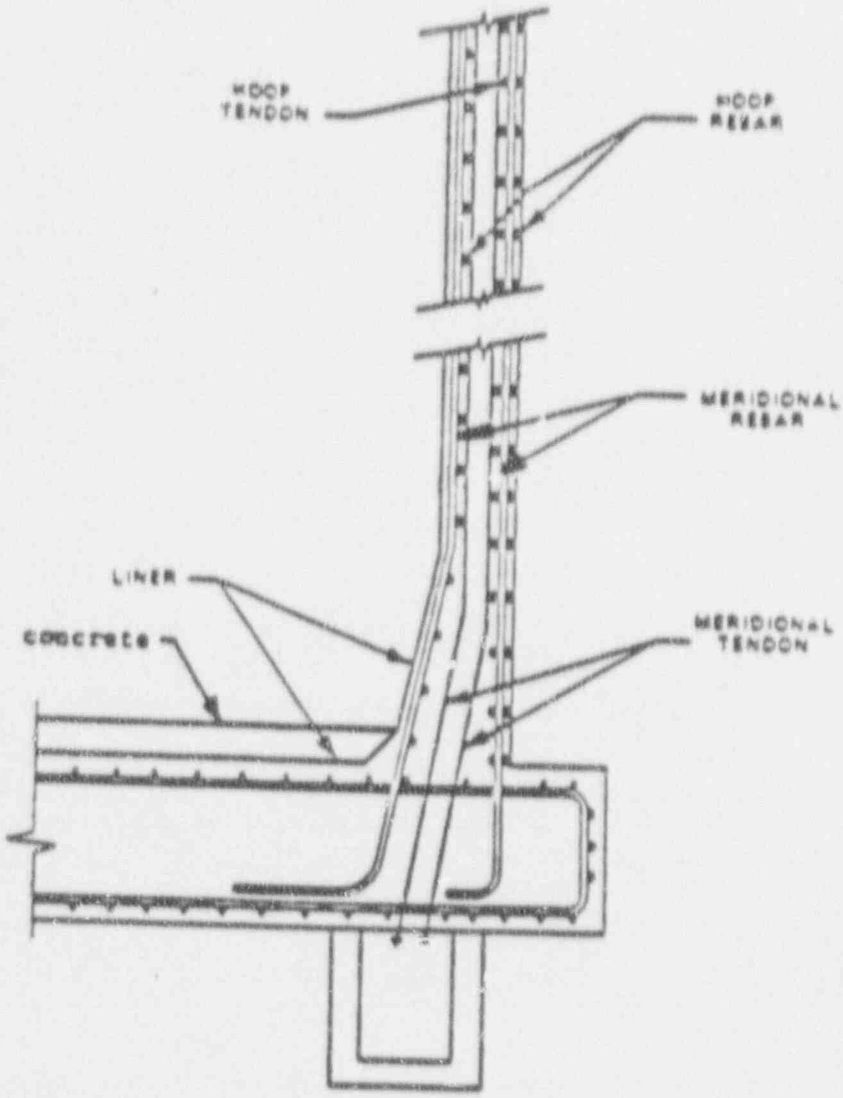


Figure 3-10. Wall Base Mat Junction of Prestressed Containment
(Source: See Reference 3)

① Incorrect statement

allow for differential displacement between the two structures. The shield structure protects the free-standing steel containment from missiles, tornados, and other air-borne hazards. As indicated in Section 2.3, the shield structure is not included in the scope of this IR.

3.2 CODES, STANDARDS, AND REGULATIONS

3.2.1 Background

Commercial nuclear power plants are designed for much more stringent loading conditions than those associated with conventional structures [2]. Normally, conventional structures consider service loads, which include construction loads and those which occur during normal operation, and severe environmental loads, such as infrequent high velocity winds or earthquake. In nuclear plant design this is taken one step further by designing for extreme environmental loads, that is loads having small frequencies of occurrence in the range of 10^{3} to 10^{4} per reactor year.

In nuclear plant design, <u>the same load behavior limits have been used for both the service and severe environmental load conditions.</u> The service load category includes those loads occurring during construction and in the course of normal operation, such as transient or test conditions. The severe environmental category covers conditions which may occur over the life of the plant, but are not anticipated to occur very often. The specific conditions which make up the service and severe environmental loads are dead, live, and construction loads; wind, snow, soil and hydrostatic pressure and buoyancy forces; loads from piping and electrical systems and equipment supports; temperature and pressure conditions during operation; and the operating basis earthquake. —①

The extreme environmental load category involves those loads which have a low probability of occurrence. These loads include the safe-shutdown earthquake, tornado wind, tsunami (for plants adjacent to coastal areas), plant-generated or tornado-borne missiles (including aircraft impact for plants close to airports), and design basis accident temperatures, pressures and jet impingement loads.

3.2.2 Codes and Standards

PWR containment structures were designed and constructed in accordance with the codes and standards effective at the date of the construction permit, or at the time the purchase order was placed with the fabricator or constructor. This resulted in a variety of different

① Provide dates
② Cannot be referenced if no dates are provided see pp. 3-24, 3-25

codes and editions of codes being applied, due in particular to the evolution of nuclear codes and standards during the 1960s and early 1970s.

①

①

The earliest PWR concrete containments were designed in accordance with existing building codes such as American Concrete Institute (ACI) Standard 318, "Building Code Requirements for Reinforced Concrete" [4]. The earliest PWR freestanding steel containment structures complied with the rules of the ASME Boiler and Pressure Vessel Code [5], Section VIII, "Unfired Pressure Vessels."

In 1965, ASME Section III, "Nuclear Vessels," became the applicable code for containments. Under Section III, vessel classification was required, with the containment defined as a Class 2 vessel initially and a Class B vessel later. In 1971, a new Section III Code was issued, titled "Nuclear Power Plant Components." Since 1971, containment structures have been classified in the ASME Code as CC (concrete containments) or MC (metal containments).

Even though the codes and standards that explicitly address nuclear power plant containment structures were developed over a number of years, materials, fabrication, and construction practices did not change dramatically, but were formalized as standards applicable to nuclear power plants. The biggest changes were made in analytical design techniques, where more sophisticated finite element shell analyses methods were adopted, and in the introduction of more rigorous quality assurance requirements.

In addition to ACI 318 and later, the ASME code, a variety of other industry standards were often used in the design of nuclear power plant containment structures. Some examples include ACI 301, "Specification for Structural Concrete for Buildings," [6] which provides general guidance for the design, selection of materials, and construction of concrete structures. Guidance for design and construction techniques to minimize cracking of concrete structures is provided in ACI 224R, "Control of Cracking in Concrete Structures" [7]. Aggregate selection is addressed in ANSI/ANS 6.4, "Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," [8] and the ASME Boiler and Pressure Vessel Code, Section III, Division 2 [9]. Selection of concrete and construction methods to minimize the effects of freeze-thaw are addressed by ASTM C260, "Specification for Air Entraining Admixture for Concrete" [10] and ACI 306, "Recommended Practice for Cold Weather Concreting" [11].

②

Standards that provide guidance for inspection and monitoring during service include ACI 201.1R, "Guide for Making a Condition Survey of Concrete in Service" [12]; ACI 224.1R, "Causes, Evaluation and Repair of Cracks in Concrete Structures" [13]; and ACI 207.3R,

(1) Reference to IWE/IWL not proper
(2) Incorrect; withdrawn.
(3) not clear - "preservice examination requirement"

"Practices for Evaluation of Concrete in Existing Massive Structures for Service Condition" [14]. Recent ASME Code activities have provided inservice inspection requirements for free-standing steel containments and concrete containment liners through Section XI, Subsection IWE, and for concrete containments, through Section XI, Subsection IWL [15]. (1) Neither of these ASME Code documents has been officially adopted by the NRC by reference in 10 CFR Part 50. However, ASME Nuclear Code Cases are available for implementation: Case N-486 referencing Subsection IWE and Case N-478 referencing Subsection IWL [16]. (2)

(2)

Plant-specific action is required to adopt the provisions of Subsection IWE or IWL, or to reference Code Cases N-486 or N-478, in the plant inservice inspection program. Both Subsections IWE and IWL, and the Code Cases, require a preservice examination of accessible containment areas, as defined in IWE-2200 and IWL-2200. Implementation of the preservice examination requirement is subject to the approval of regulatory authorities.

(1)

(3)

3.2.3 Federal Regulations

10 CFR Part 50 provides for the issuance of permits to construct and licenses to operate nuclear power plants [17]. General Design Criteria (GDC) 1 of Appendix A to 10 CFR Part 50 requires that "structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed." General Design Criteria 2 requires that "structures important to safety be designed to withstand the effects of natural phenomena (such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches) without loss of capability to perform their safety function." General Design Criteria 4 requires that "structures important to safety be able to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss of coolant accidents (LOCAs). These structures must also be appropriately protected against dynamic effects including the effects of missiles, pipe-whip, and flooding that may result from equipment failures and from events and conditions outside the nuclear power unit."

In addition to GDCs 1, 2, and 4, additional functional design requirements for containment structures are defined in GDCs 16 and 50 [17]. The underlying regulatory criterion for the containment structure is that it is the final barrier against the release of significant amounts of radioactive fission products in the event of an accident. The containment structure must be capable of withstanding, without loss of function, the pressure and temperature conditions resulting from postulated loss-of-coolant, steam line or feedwater line break

(1) Leak tightness; incorrect statement
(2) Misleading; IWE/IWL inspections are not currently reqd
(3) Reference to IWL not proper

accidents. In addition, the containment structure must also maintain functional integrity in the long term following a postulated accident.

Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," defines quality assurance requirements for the design, construction, and operation of those structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The Appendix B quality assurance requirements apply to all activities affecting the safety-related functions of those structures, systems, and components, and include the following activities: designing, purchasing, testing, operating, maintaining, repairing, refueling, and modifying.

1

Periodic testing of the containment to Appendix J of 10 CFR Part 50) is intended to provide assurance of containment pressure retaining capability at the time the test is conducted, and to provide a reasonable assurance that the integrity of the containment liner will not be impaired between subsequent test intervals. The currently required examinations (Type A testing and the required containment surface inspections, and the inservice inspection of prestressing systems, along with inspections similar to those provided by ASME Section XI, Subsection IWE and proposed Subsection IWL) provide an adequate basis for detection of significant age-related degradation. As stated in NRC Information Notice No. 89-79 [18], Appendix J requires that "... a general visual inspection of the accessible surfaces in the containment be performed before each integrated leak rate test. The purpose of this inspection is to identify any evidence of structural deterioration or other problems that may affect containment integrity or leak tightness." 2

3.2.4 Regulatory Guidance

The NRC has issued a number of Regulatory Guides which provide recommendations for addressing specific regulatory concerns related to PWR containments. One Regulatory Guide relevant to PWR containment license renewal is Regulatory Guide 1.35 [19], "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," originally issued in 1974 and revised most recently in July 1990. Guidance is provided for inservice examination and assessment of prestressed concrete containments with ungrouted tendons. Many of the recommendations in Regulatory Guide 1.35 are identical to provisions of ASME Section XI, Subsection IWL [15]. ~~A time-dependent effect~~ 3 is contained in Regulatory Guide 1.35 that must be addressed by the applicant differently for license renewal than for the original license term. In this case, prestressing losses must be predicted for the term of operation, to which actual prestressing losses must be

(1) Change to "must be"
(2) Irrelevant
(3) lacks details

compared. The prestressing loss predictions should be reviewed for consistency, and recalculated, as necessary, in order to reflect the license renewal term. (1)

3.3 OPERATING HISTORY

A key method of identifying and assessing potential age-related degradation mechanisms for containment structures is to review the operating and maintenance history. Since many of the PWR containment structures have been in operation for twenty years or more, data relevant to age-related degradation currently exists. This subsection describes the operating and maintenance history of concrete and free-standing steel PWR containment structures.

3.3.1 Historical Performance of Concrete Structures

Concrete structures have demonstrated excellent resistance to age-related degradation. Portland cement has been in use since the early 1800's. A series of tests were conducted at the University of Wisconsin beginning in 1923, for which 50-year data were reported [20]. All concrete in these tests exhibited good weathering qualities.

(2)

Naus [3] reports that three concrete reactor buildings at the Savannah River Plant were inspected after approximately 25 years of operation to determine suitability for an additional 20 to 30 years of operation. Except for some repairable cracking, all structures were found to be in satisfactory conditions. Serviceability projections beyond 45-55 years were not made.

(4) The good performance record of containment concrete is attributable to high quality design standards and construction practices, supplemented by thorough inspections during construction. A review of PWR containment performance identified only a few isolated instances of problems: over 90% of the containments fully met design, construction and quality standards. In each instance of deficiencies, the rigorous inspections conducted during construction identified the problems, all of which were corrected. The high quality of design and construction of PWR concrete containments provides assurance of good performance. (5)

(4) Loose words
(5) References?

① CLB Maintenance not addressed
② How about leak tightness?
③ Misleading ④ Loose write

3.3.2 Concrete Containment Performance Testing

Containment structures are designed with substantial margin in their capability to perform their intended function of providing a pressure boundary to release of fission products. This margin provides additional confidence that, while some age-related degradation may occur, the existing margin more than offsets the effects of this degradation. The adequacy of this design margin has been demonstrated, both by analyses to identify the ultimate pressure capability of the containment structure and by scale-model testing. (1)

(2)

The Brookhaven National Laboratory completed a study to determine the ultimate pressure capability of concrete containment structures [21]. This study demonstrated that containment structures would not fail at pressures up to a factor of 2 or 3 above design pressure. Additional conservatism may exist in the analysis depending on the actual material properties at a specific plant and a more realistic definition of what constitutes a failure in the containment pressure boundary.

The Sandia National Laboratories (SNL) completed tests on a 1/6-scale model of a conventionally reinforced containment [22]. The tests involved pressurization of the model to internal pressures significantly beyond the design pressure of full-scale containment structures. The model remained structurally intact at pressures (2) up to 145 psig, which is more than 3 times the design pressure of the containment structures.

(3)

Localized deterioration of containment concrete may have no effect on containment capability, particularly where ultimate capacity is determined by factors other than those affected by deterioration. Aging of the liner is another matter. In the SNL test, failure was caused by a 20-in. tear in the liner, while smaller tears in other areas of the liner contributed a small amount to the integrated leakage from the model. Localized liner corrosion could significantly lessen the ultimate capacity by reducing liner thickness, thus illustrating the need to closely monitor liner corrosion. (3)

3.3.3 Historical Performance of Reinforcing and Prestressing Systems

(4)

The performance history of reinforcing and prestressing systems has been quite good, with only a few instances of age-related degradation identified. Naus reports [3] surveillance information from prestressed containments and prestressed concrete reactor vessels from the United States, France, Sweden, and the United Kingdom, as follows: (5)

- In the United States, surveillance reports have concluded that the respective containments were in good condition. Little water has been found in tendon

(5) Irrelevant

(1) Loose words
(2) Irrelevant

ducts except that, in one containment, a significant amount of water was found in several ducts. Corrosion was found to be minor, demonstrating the effectiveness of corrosion inhibitors under severe conditions. Otherwise, a few instances of wire corrosion have been reported, but wire breaks did not generally result and corrosion was so minor that complete replacement was not required. It was generally concluded that corrosion occurred prior to filling the ducts with corrosion inhibitor. Incidents of incomplete duct filling have been reported along with improper tendon stressing, and although corrected, have not caused any serious difficulties. Observation of missing buttonheads have been made on some wires, but the number of non-effective wires permitted under design assumptions was not exceeded. (1) (1)

- There have been some other isolated instances of prestressing system problems at U.S. nuclear plants. At Bellefonte, failures of 8 top anchor heads of rock anchor tendons occurred prior to a two-stage grouting process. One of these tendons also had 23 of its 170 wires fail. Stress corrosion cracking was identified as the cause of these failures. At Byron, 4 anchor heads failed between 1 and 64 days after prestressing tendons in the Unit 1 containment, with failure attributed to temper embrittlement. Anchor head failures at Farley 1 (6 failures) and 2 (21 failures) occurred approximately 8 years after prestressing, with stress corrosion cracking identified as the cause. (2)

- In France, prestressed containments use grouted tendons (2) (except for 4 vertical tendons of the first unit built at a site). Through 1982, ten leakage and structural pressure tests had been performed, and leakage rates were within prescribed limits and containment response was elastic and consistent with design.

- In Sweden, 6 prestressed containments were in operation through 1982, 5 of them using ungrouted tendons. Reported inspections indicated that broken wires, missing buttonheads, or serious corrosion have not occurred. Small amounts of water have been found in a few grease caps, but tests show that the grease was in good condition, and that tensile and bending tests of wires yielded good results. Steel properties have not been affected with time and prestressing losses were generally less than expected.

- The performance of prestressing systems in the United Kingdom prestressed concrete reactor vessels has been good and no problems have been encountered with loss of tendon load. A small number of cases have been identified where buttonheads were missing. (2)

- The major percentage of [tendon] corrosion occurred during construction, although it was not considered serious enough to warrant tendon replacement. In France, performance has been satisfactory. In two instances, extensive corrosion was detected but subsequent corrosion was arrested by changing the conduit air sweeping system from periodic to continuous operation to control humidity. In the U.S. (Fort St. Vrain), data available from 1971 through 1984

(1) Should also be discussed in Section 4.3.1.1, p. 4-37
(2) Misleading (3) Irrelevant
(4) cannot be substantiated

show the performance to be good. Corroding wires and broken tendons were identified in 1984, with failures attributed to general corrosion and stress corrosion cracking resulting from acids formed during microbiological attack on the grease. An analysis which included the effects of the degraded tendons found the vessel able to withstand operating pressures. A proposal has been made to fill the ducts with a nitrogen blanket as a means to stop the corrosion and to increase the frequency of visual inspection and lift-off tests. (1)

(2) (3)

Therefore, relatively isolated observations of failures in reinforcing and prestressing systems for containment and reactor vessel structures provide confidence that such systems have inherent capability to perform their intended function over extended periods of service. (4)

3.3.4 Historical Performance of Liners

A survey of prestressed concrete pressure vessel and containment inservice inspection experience [23] included a review of 100 Licensee Event Reports. In only one instance was a Type A test failure caused by leakage other than through penetrations or valves. In this one instance, two holes had inadvertently been drilled through the liner. Consequently, past history supports that liner degradation sufficient to compromise containment integrity has not occurred.

3.3.5 Steel Containment Performance Testing

Steel containment structures also are designed with substantial margins to perform their intended function of providing a leakage barrier to the release of fission products. As with the concrete containment structures discussed in Section 3.3.2, this margin provides additional confidence that while some age-related degradation may occur, the existing margin more than offsets any degradation effects. The adequacy of this design margin has been demonstrated, both by analyses to identify the ultimate pressure capability of free standing steel containment structures and by scale-model tests. (5)

Owners of ice-condenser free standing steel containments have performed studies to determine the ultimate pressure capability of their containments as part of their response to new regulatory requirements for degraded core hydrogen control (10 CFR Part 50.44). Analyses have been completed for the Tennessee Valley Authority's Sequoyah Nuclear Power Plant and Watts Bar Nuclear Power Plant, Indiana Michigan Power's D. C. Cook Nuclear Power Plant, and Duke Power Company's Catawba Nuclear Plant and McGuire Nuclear Plant. These analyses have shown that the ultimate pressure capacity of the containment structures is significantly higher than the design pressure. (5)

(5) CLB Maintenance

(1) CLB maintenance

The Sandia National Laboratories (SNL) conducted tests on 1/32- and 1/8-scale models of free-standing steel containments [22]. These tests demonstrated that free-standing steel containments are capable of withstanding pressurization up to two times the design pressure of the full-scale containments. (1)

3.4 REFERENCES

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3. D. J. Naus, Concrete Component Aging and its Significance Relative to Life Extension of Nuclear Power Plants, NUREG/CR-4652, ORNL/TM-10059, Oak Ridge National Laboratory, Oak Ridge, Tenn., September 1986.

4. "Building Code Requirements for Reinforced Concrete," ACI 318, American Concrete Institute, Detroit, Michigan.

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7. "Control of Cracking in Concrete Structures," ACI 224R, American Concrete Institute, Detroit, Michigan.

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10. "Specification for Air Entraining Admixture for Concrete," ASTM C260, American Society for Testing and Materials, Philadelphia, Pennsylvania.

11. "Recommended Practice for Cold Weather Concreting," ACI 306, American Concrete Institute, Detroit, Michigan.

12. "Guide for Making a Condition Survey of Concrete in Service," ACI 201.1R, American Concrete Institute, Detroit, Michigan.

13. "Causes, Evaluation, and Repair of Cracks in Concrete Structures," ACI 224.1R, American Concrete Institute, Detroit, Michigan.

14. "Practices for Evaluation of Concrete in Existing Massive Structures for Service Condition," ACI 207.3R, American Concrete Institute, Detroit, Michigan.

Cannot be referenced if no dates are provided

Dates

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19. Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures", Revision 3, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C., July, 1990.

20. G. W. Washa and K. F. Wendt, "Fifty Year Properties of Concrete," Journal of the American Concrete Institute, American Concrete Institute, Detroit, Michigan, January 1975.

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22. Electric Light and Power, September 1987.

23. Technical Report - An International Survey of In-Service Inspection Experience with Prestressed Concrete Pressure Vessels and Containments for Nuclear Reactors, Federation Internationale De La Precontrainte, Wexham Springs, Slough SL3 6PL, 1982.

(1) "Safety function(s)" is too limited

SECTION 4
AGE-RELATED DEGRADATION MECHANISM ASSESSMENT

In the previous section of the report, the component evaluation basis for PWR containment components was described, consisting of component design features (Section 3.1); codes, standards and regulations governing their construction and operation (Section 3.2); and relevant operation history (Section 3.3). This section of the report describes the age-related degradation mechanisms that could affect PWR containment components, and evaluates the potential significance of the effects of these mechanisms on the continued safety function(s) performance of these components throughout the license renewal term.

(1)

The set of age-related degradation mechanisms evaluated in this section is derived from a review/evaluation of component service experience, relevant laboratory data, and related experience from other industries. The set consists of the following:

Concrete
- Freeze-thaw
- Leaching of calcium hydroxide
- Aggressive chemicals
- Reactions with aggregates
- Corrosion of embedded steel
- Elevated temperature
- Irradiation

Reinforcing Steel
- Corrosion
- Elevated temperature
- Irradiation

Prestressing System
- Corrosion
- Elevated temperature
- Irradiation
- Prestressing losses

Liner
- Corrosion
- Elevated temperature
- Irradiation

Miscellaneous Age-Related Degradation Mechanisms
- Fatigue
- Concrete interaction with Aluminum
- Settlement

Free-Standing Steel Containment Degradation Mechanisms
- Strain aging
- Corrosion

When experience has shown combining two or more of these mechanisms to be significant, any such synergistic effect has been explicitly evaluated.

① Comment #3, p.1-1
② Comment #1, p.1-2

The technical evaluation of a particular age-related degradation mechanism and its effects on the continued safety performance of a particular PWR containment component leads to one of two conclusions: (1) the effects of the mechanism are potentially significant to that component, and further evaluation is required in Section 5 relative to the capability of effective programs to manage the effects of the age-related degradation; or (2) the effects of the age-related degradation are not significant to the ability of that component to perform its intended safety function throughout the license renewal term. For the latter case, specific criteria and corresponding justification are provided in this section that can be used as the basis for generic resolution of the age-related degradation mechanism/component issue.

License renewal applicants intending to reference these generic conclusions are responsible for a review/evaluation of plant specific features, including appropriate CLB documents/information, in order to assure that there are no deviations from the assumptions and criteria used in the report. This review should compare the design basis for particular components with the representative design bases given in Sections 3.1 and 3.2. The component operating history should also be compared to the generic performance parameters described in Section 3.3. Finally, the specific assumptions and criteria used in this section should be examined to assure that they~~, or justified equivalents~~, apply to the component under consideration. ③

Delete

4.1 CONCRETE

Deterioration of hardened concrete can be caused by aggressive environmental factors such as exposure to chemicals, corrosion of embedded steel, chemical reactions with aggregates, and/or extreme environmental conditions. Other types of degradation, such as cracking, occur very early in the life of a concrete member, while the concrete is in either the plastic or hardening state. Plastic concrete, that is, that which has not achieved its final set, can exhibit cracking due to:

- excessive water loss resulting in random craze cracks,
- creep and plastic shrinkage caused by rapid drying and the subsequent volume change that occurs when concrete at the surface is restrained by the concrete below the embedded reinforcing steel,
- inadequately prepared subgrades.

Cracking of this type does not usually provide a diminished structural capability, unless such cracking leads to the entry of constituents creating an aggressive environment.

③ Sections 3.1, 3.2, and 3.2 lack specifics; cannot be implemented as asserted.

Hardened concrete may crack early in its life due to (1) volume changes caused by drying, (2) expansion and contraction due to differences in temperature, and/or (3) resistance to load. The following subsections focus on the time-related degradation mechanisms that may cause a PWR containment structure to deteriorate and thus can be described as age-related phenomena.

4.1.1 Freeze-Thaw

4.1.1.1 Mechanism Description

Repeated freezing and thawing is a mechanism known to be capable of causing severe deterioration in both the mechanical properties and physical form of concretes that are susceptible to such action. To make concrete immune to the effects of freezing and thawing, Mather [1] has summarized three factors which must be considered in the design and placement of concrete to provide immunity to freeze-thaw effects:

A. the cement paste must have an entrained air system with an appropriate void spacing factor,

B. the aggregate must be of a sufficiently high quality to resist scaling,

C. the in-place concrete must be allowed to mature sufficiently before being exposed to cyclic freezing and thawing.

Concrete used in PWR containment structures contains an air entrained admixture which meets ASTM C260-77 [2] requirements. In addition, the quantity of admixture contained in the concrete should result in an adequate percentage of air. Regarding maturity, it can be stated that the more complete the chemical reactions are in the concrete prior to the first freeze-thaw exposure, the more likely that it will be resistant to freeze-thaw action.

Figure 4-1 shows the effect of air content on durability. Optimal concrete durability is achieved by ensuring the air content percentages are within the ranges specified in ACI 301-84 [3]. The optimal air content is based on the nominal maximum size of coarse aggregate, but is generally between 3 and 6%.

Concrete lacking a suitable entrained air system is susceptible to degradation by the action of repeated freeze-thaw cycles on hardened cement paste, and on certain types of aggregates used. The cause of this phenomena is water freezing within the pores of the concrete, creating hydraulic pressure. This pressure produces either an increase in the size of the cavity, due to the volumetric increase associated with ice formation, or forces some

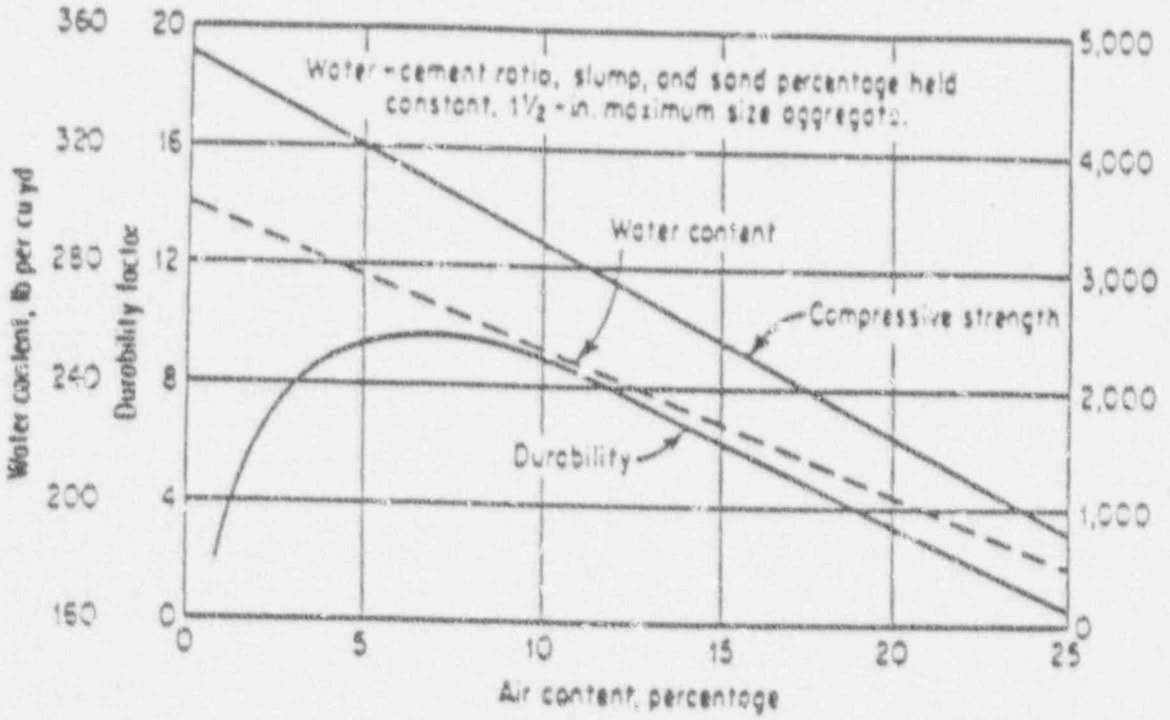

Figure 4-1
Effects of Air Content on Durability,
Compressive Strength, and Required Water Content of Concrete
(Source: See Reference 4)

water out of the cavity to surrounding areas into small voids created by entrained air bubbles. If these voids are filled with water, there will be no relief of the pressure and freeze thaw degradation may result. The same phenomena have been observed to occur in certain porous aggregates, depending upon the size and number of pores within the aggregate and on the pore size distribution and permeability [5].

Based on the above discussion, the following factors increase concrete's resistance to freeze-thaw degradation:

A. adequate air content,

B. low permeability, i.e., concrete with low water-to-cement ratio, adequate placing and curing,

C. protection of concrete from freeze-thaw until adequate strength has developed, and

D. surface coatings applied to frequently wetted-dried surfaces.

Freeze-thaw degradation is characterized by scaling, cracking, and spalling. Scaling or surface flaking occurs in the presence of moisture and is aggravated by the use of deicing salts. In extreme cases of freeze-thaw degradation, cracking, spalling, and scaling reduce the cover over reinforcing steel, reduce concrete strength, and eventually expose the reinforcing steel to accelerated corrosion and the concrete to the expansive effects of the resulting corrosion products, thereby weakening the concrete's resistance to further attack by aggressive environments.

Evaluation of the aggregate quality and required concrete maturity to prevent freeze-thaw degradation is difficult, and should be based on an evaluation of available laboratory test data coupled with the service history of the materials used in the mix design and the guidance provided in ACI-306-83, "Recommended Practice for Cold Weather Concreting" [6]. The use of concretes made with adequate entrained air and aggregates with a competent service history are expected to produce concrete with good-to-excellent resistance to freeze-thaw degradation.

Areas of the country that experience the combination of numerous freeze-thaw cycles with significant amounts of winter rainfall would be more likely to exhibit degradation than areas in milder climates. "Standard Specification for Concrete Aggregates," ASTM C33-82 [7], groups the areas of the U.S. into "severe," "moderate," and "negligible" weathering regions, depending upon the weathering index, which is the product of the average annual

① Effects should be evaluated for the extended period of operation; inconsistent; see Freeze-Thaw discussion p. 1-4

number of freezing cycle days and the average annual winter rainfall. The weathering index is in excess of 500 day-inches for the "severe" region, between 100 and 500 day-inches for the "moderate" region, and less than 100 day-inches for the "negligible" region [7].

4.1.1.2 Significance to License Renewal

Surfaces exposed to the weather that can become saturated with water and freeze are vulnerable to freeze-thaw degradation. For PWR containments, the flat or near-flat surfaces of the dome, and the ring girder of post-tensioned containments are areas which could be susceptible to freeze-thaw degradation. Those structures in "moderate" and "severe" weathering regions have the greatest potential for freeze-thaw degradation. PWR containment structures in regions exposed to "negligible" weathering are not subject to freeze-thaw degradation. In addition, because of the design and construction standards used for concrete PWR containment structures, freeze-thaw degradation of exterior walls in "moderate to severe" weathering regions will not be significant. For some PWR containment structures, the use of coatings on containment dome surfaces and the sloping surfaces of ring girders provides further protection from freeze-thaw degradation.

Concrete used for containment structures is produced using sound principles of concrete constituent material selection and mix design. Containment concrete contains an appropriate amount of entrained air (3-6%) necessary for freeze-thaw resistance, and an amount of cement that both enables the achievement of the desired concrete design strength and a water-to-cement ratio conducive to reduced permeability. On some containments, coatings have been provided on dome surfaces and the sloping surfaces of prestressed containment ring girders, which will further minimize the penetration of water. Because of these preventive measures, freeze-thaw degradation is highly unlikely to occur on a wide scale or over substantially large areas of a PWR containment.

There have been localized degradation incidents at units located in geographic regions with severe weather conditions. These incidents have served to identify the limited susceptible set and have resulted in the implementation of remedial actions and planned inspections. These incidents of freeze-thaw damage to containment domes are attributed to original construction defects rather than age-related degradation, and are plant-specific. Damage observed to date has resulted in the implementation of appropriate remedial actions thereby preventing potentially significant degradation. ①

① call the date out ; ASTM C260-77.

4.1.1.3 Summary of Freeze-Thaw

If PWR containment concrete components are not exposed to the weather in such a way that they can become saturated with water, then they are not vulnerable to freeze-thaw damage.

If PWR containment structures are located in a geographic region subject to "negligible" weathering conditions, i.e., a weathering index of less than 100 day-inches per year [7], then freeze-thaw damage is not a significant age-related degradation mechanism for the containment concrete, and requires no further evaluation.

If PWR containment structures are located in a geographic region subject to "severe" (weathering index greater than 500 day-inches per year) or "moderate" (weathering index between 100 and 500 day-inches per year) weathering conditions, but if the concrete mix design meets the air content and water-to-cement ratio requirements of ASTM C260 [2] ① (or, equivalently, the ASME Code requirements of Section III, Division 2, Paragraph CC-2231.7.1 [8]), then freeze-thaw damage is not a significant age-related degradation mechanism for the containment concrete, and requires no further evaluation.

This conclusion applies to all PWR containment concrete components:

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade
3. Concrete containment wall below grade
4. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that the weathering index or concrete mix design satisfy the above assumptions and criteria.

4.1.2 Leaching of Calcium Hydroxide

4.1.2.1 Mechanism Description

Water passing through cracks, inadequately prepared construction joints, or areas that were inadequately consolidated during placing can dissolve some calcium-containing products in the concrete. The most readily soluble of these is calcium hydroxide (lime). When most of the calcium hydroxide has been leached away, other cementitious constituents become exposed to chemical decomposition, eventually leaving behind silica and alumina gels with little or no strength [4]. Water, either from rain or melting snow, that contains small amounts of calcium ions can readily leach lime from concrete. The water's aggressiveness or ability to leach calcium hydroxide depends on its dissolved salt content and its temperature. This leaching action of the water can only occur if the water passes through the concrete. Water that merely passes over the surface will not cause significant leaching.

Leaching of calcium hydroxide is observed on concrete that is alternately wetted and dried. The white deposits that are left on the surface of the concrete are a solution of water, free lime from the concrete, and carbon dioxide that has been absorbed from the air. The leachate from the concrete is nearly colorless, until the carbon dioxide is absorbed and the material dries as a white deposit.

Leaching over long periods increases the porosity and permeability of concrete, making it more susceptible to other forms of aggressive attack, and reducing its strength. Leaching also lowers the pH of the concrete and threatens the integrity of the protective oxide film around the rebar.

Problems relating to the dissolving and leaching action of percolating water are tied directly to permeability. Resistance to leaching and efflorescence is thus related to ensuring the use of concrete with low permeability. A dense concrete with a suitable cement content that is well cured will have low absorption and be less susceptible to the calcium hydroxide dissolving action of rain or groundwater. Although many concrete containment basemats are constructed with waterproofing membranes, no credit is taken for their remaining effectiveness.

Any factor that tends to improve the compressive strength of the concrete will have a beneficial effect on water tightness. Therefore, the better the quality of the constituent materials, the less permeable the concrete. ACI 201.2R-77 [9] provides guidance to assure a dense well-cured concrete. Low water-to-cement ratio, smaller coarse aggregate, long curing periods, entrained air, and thorough consolidation all contribute to water tightness.

Figures 4-2 and 4-3 show the relationships between permeability, water-to-cement ratio, aggregate size and curing time.

4.1.2.2 Significance to License Renewal

Concrete PWR containment structures that are exposed to rainwater or groundwater may be susceptible to leaching of calcium hydroxide. Cracks and improperly prepared construction joints provide the easiest mechanisms for entry of water, and are likely areas for leaching. PWR containments are constructed of a dense, well-cured concrete with an amount of cement suitable for strength development, and achievement of a water-to-cement ratio which is characteristic of concrete having low permeability. In addition, to cause leaching, the water must be flowing, rather than just filling a crack or void. Such water flow is unlikely, even if the ground water is of chemical makeup that could leach lime from concrete. Cracks may be more numerous in reinforced concrete containments than in post-tensioned containments, but should be sufficiently tight to prevent the free flow of water necessary for dissolution of free lime to occur.

4.1.2.3 Summary of Leaching of Calcium Hydroxide

If the containment structure is not exposed to flowing water, then leaching of calcium hydroxide is not a significant age-related degradation mechanism for concrete components, and requires no further evaluation.

If the concrete containment structure is exposed to flowing water, but if the structure was constructed using a dense, well-cured concrete assuring low permeability, consistent with the guidance provided in ACI 201.2R-77 [9], then degradation caused by the leaching of calcium hydroxide is not significant, and requires no further evaluation.

This conclusion applies to all PWR containment concrete components:

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade
3. Concrete containment wall below grade
4. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

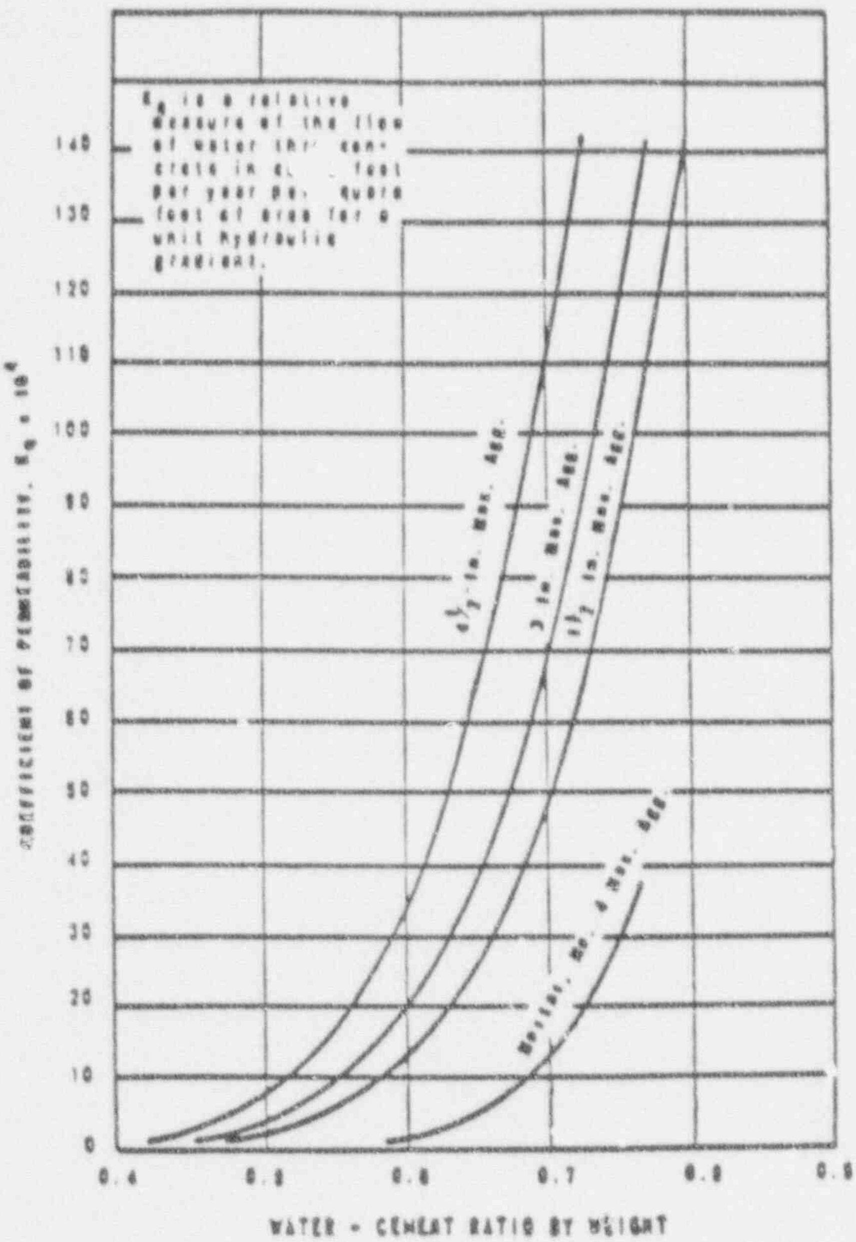

Figure 4-2. Relationship Between Coefficient of Permeability and Water-to-Cement Ratio (Source: See Reference 10)

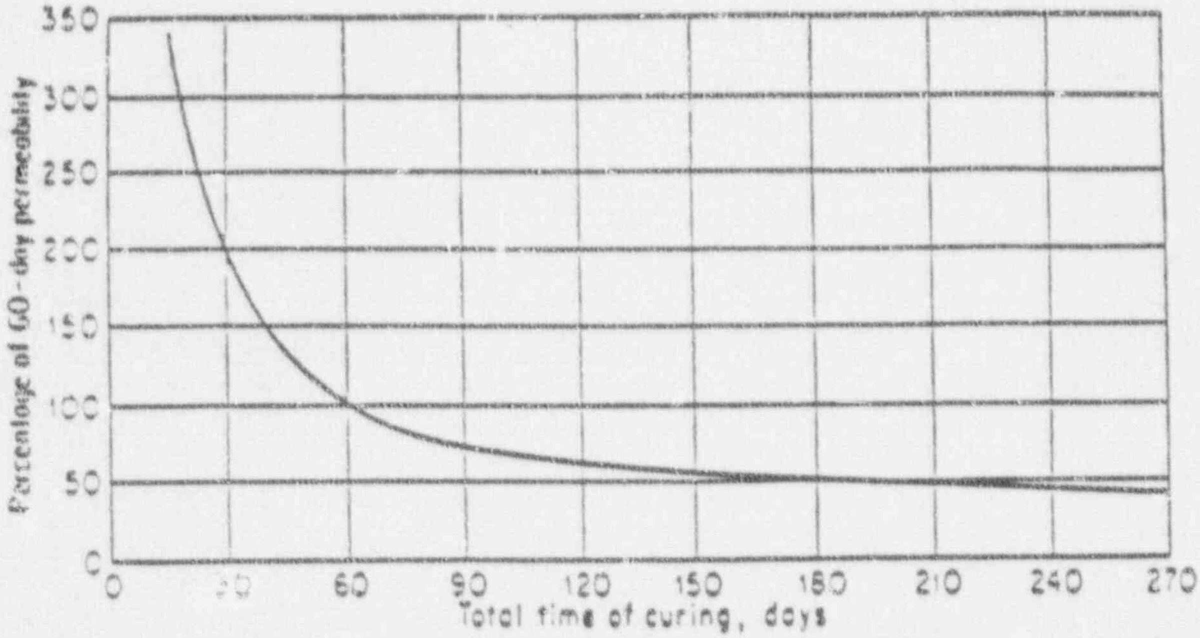

Figure 4-3
Effect of Curing Period on Permeability
(Source: See Reference 4)

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that the site conditions relating to flowing water and the concrete mix design satisfy the above assumptions and criteria.

4.1.3 Aggressive Chemicals

4.1.3.1 Mechanism Description

Concrete, being highly alkaline (pH > 12.5), is degraded by acids [5]. Portland cement concrete is not truly acid-resistant, although varying degrees of resistance can be obtained depending upon the materials used and the care taken in placing, consolidating and curing. Acid attack can increase porosity and permeability of concrete, reduce its alkaline nature at the surface of the attack, reduce strength, and render the concrete subject to further deterioration. No Portland cement concrete, regardless of its composition, will withstand exposure to highly acidic water for long periods. A dense, concrete with low permeability and low water-to-cement ratio may provide an acceptable degree of protection against mild acid attack [9].

Sulfates of potassium, sodium, and magnesium may attack concrete, depending upon the concentration present in soils and/or groundwater. Sulfate attack, generally more prevalent in the western half of the U.S. [10], can be severe when the concrete is saturated and is more likely when alternating saturation and drying conditions are encountered [4]. In addition, the exposed surfaces of containment structures located near industrial plants which contribute to the sulfur-based acid rain phenomenon could be subject to deterioration. Sulfate attack can produce significant expansive stresses within the concrete, leading to cracking, spalling, and strength loss. Once established, these conditions allow further exposure to aggressive solutions. Use of adequate cement content, low water-to-cement ratio, and thorough consolidation and curing contribute to low permeability and provide effective protection against sulfate attack. Use of the appropriate cement type (e.g., ASTM C150, Type II) and pozzolan (e.g., fly ash) also increase sulfate resistance [10].

4.1.3.2 Significance to License Renewal

Acid attack may occur where concrete is exposed to aggressive aqueous solutions. The environment inside containment does not include exposure of concrete to aggressive chemicals. Because of its properties, concrete used in PWR containment structures is not

① Loose words
② [11] ... [12] ?

significantly affected by acid rain. Thus, internal concrete and external concrete above grade are not subjected to the conditions necessary to cause potentially significant degradation due to aggressive chemicals. Therefore, only below grade portions of the containment which may be exposed to sulfate bearing soils or groundwater are susceptible to degradation due to aggressive chemicals. The potential for degradation depends upon the composition of the soil/groundwater, the level of the groundwater in relation to below grade portions of the containment, and the presence or lack of a waterproof membrane.

For resistance to chlorides and/or sulfates, a high cement content, low water-to-cement ratio, and thorough consolidation and curing contribute to low permeability and provide the best protection. Cement type also is significant in sulfate resistance (e.g., ASTM C150, Type II [11], a cement type commonly used in containment construction, possesses moderate sulfate resistance). Containments are generally constructed with concrete possessing these attributes. In addition, the soil or groundwater chemistry must be aggressive (pH < 5.5 or chemical concentrations above the threshold limits of 500 ppm chlorides [12], 1500 ppm sulfates [13]) and the exposure conditions must allow contact for chemical attack to occur. These factors collectively lessen the likelihood that chloride or sulfate attack of PWR containment structures will be significant. ①

4.1.2.3 Summary of Aggressive Chemicals

If PWR containment structures are not exposed to an aggressive chemical environment (< 5.5 pH), or to chloride/sulfate solutions beyond defined limits (> 500 ppm chlorides [12] and 1500 ppm sulfates [13]), then degradation caused by aggressive chemical attack is not significant for containment concrete, and requires no further evaluation. The following PWR containment concrete components are not exposed to aggressive groundwater: ②

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade

A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that the assumptions and environmental exposure limit criteria given above, or their justified equivalent, are met.

If PWR containment concrete components are exposed to groundwater that exceeds the pH, chloride, or sulfate limits defined above (aggressive groundwater), but if this exposure is

(1) Loose words

(1) for intermittent periods only, then degradation caused by aggressive chemicals is not significant.

If PWR containment structures are exposed to aggressive groundwater for extended periods, (1) then the degradation caused by aggressive groundwater attack is potentially significant. This is limited to the below grade concrete structural components listed below. Further evaluation of these PWR containment structures that may be exposed to aggressive groundwater for extended periods is provided in Section 5.1 of this report.

Reinforced and Prestressed Concrete Containments

1. Concrete containment wall below grade
2. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

4.1.4 Reactions with Aggregates

4.1.4.1 Mechanism Description

Chemical reactions are possible between certain aggregates and alkalies [14]. These alkalies are predominantly introduced by cement, but also may come from admixtures, salt-contaminated aggregates, and penetration by seawater or solutions of deicing salt. Three types of reactions may occur depending upon the composition of the aggregates. They are alkali-aggregate reaction, cement-aggregate reaction, and expansive alkali-carbonate reaction.

Alkali-aggregate reaction, more properly designated as alkali-silica reaction, involves aggregates which contain silica and alkaline solutions. All silica minerals have the potential to react with alkaline solutions, but the degree of reaction and ultimate degradation incurred can vary significantly. Alkali-silica reactions can cause expansion and severe cracking of concrete structures. Reactive material in the presence of potassium, sodium, and calcium oxides derived from the cement reacts to form solids which can expand upon exposure to water. Expansion due to alkali-aggregate reaction was recognized as early as 1940 by T.E. Stanlon, "Expansion of Concrete Through Reaction Between Cement and Aggregate," Proceedings ASCE V.66 Dec. 1940 [15] and was also reported in 1958 by William Leech "Chemical Reactions," Special Technical Publication No. 169, 1958, [16] American Society for Testing Materials, Philadelphia. A map and data

(1) All the containment; ACI 201.2R-77.

showing geographic areas known to yield natural aggregates suspected of or known to be capable of alkali-silica reaction are included in ACI 201.2 [9]. The map and the data were published in 1966 and as early as 1941, respectively. A 1960 report by ACI Committee 201 indicated that the rocks which may induce rapid alkali-silica reactions are found predominantly in the western half of the United States. Alkali-silica reactive rocks which are characterized by slow reaction rates were recognized as early as 1941. The reactivity of such aggregates might not be recognized until the structures were over 20 years old, even if used in combination with high alkali cement. These rocks include granite gneisses, metamorphosed subgraywackes, and some quartz and quartzite gravels.

Cement-aggregate reaction is a second type of reaction between the alkalies in cement and some siliceous constituents of the aggregates, which is complicated by environmental conditions that produce high concrete shrinkage and alkali concentrations on the surface due to drying. Sand-gravel aggregates in the Kansas, Nebraska and Wyoming areas have been subject to this type of reaction.

A third type of reaction can occur between certain carbonate aggregates and alkalies, which in some instances produces expansion and cracking. Certain limestone aggregates have been reported as reactive, these being from Ontario, Canada, and from Illinois, Indiana, Iowa, Michigan, Missouri, New York, South Dakota, Virginia and Wisconsin in the United States.

Aggregates which react with alkalies can cause expansion of varying severity, even to the extent of producing cracking of the concrete and resulting loss of strength and durability if the expansion is severe. The cracking is irregular and has been referred to as "map cracking."

Aggregates used in containment concrete are specifically investigated, tested, and petrographically examined to determine the potential for reactivity with alkalies. Generally, nonreactive aggregates have been used. However, this may not always have been possible due to the unavailability of nonreactive aggregates. In these cases, potentially reactive aggregates may have been used under the provisions of ACI 201.2R-77 [9], which include the following:

- Some percentage of the aggregate is replaced by non-reactive aggregates.
- A cement having a low alkali content is used, preferably as low as practical, but not in excess of 0.6 percent.

(1) Does reference 5 supercede Ref. 7 ?
(2) Call the editions out. ASTM C295-85, ASTM C227-8

- A pozzolan is used that has been shown to be effective in preventing excessive expansion at the prescribed quantity.
- Both low alkali cement and an effective pozzolan are used in combination.
- Total alkalies in the concrete from all sources are limited to 3.0 kg/m³ [5]. (1)

4.1.4.2 Significance to License Renewal

Moisture must be available for chemical reactions between aggregates and alkalies to occur. Consequently, areas that are either consistently wet or alternatively wet and dry are susceptible to deterioration given the presence of potentially reactive aggregates. Such areas would include unprotected portions of the containment basemat and shell in contact with groundwater, and areas of the dome and ring girder concrete.

Operating history does not indicate that structural integrity is significantly affected by alkali-aggregate reactions. Aggregates used in containment concrete are investigated, tested, and petrographically examined in accordance with ASME Section III Division 2 Class (2) CC [8], ASTM C295 [17], and ASTM C227 [18] to determine the potential for reactivity with alkalies. In most nuclear plant construction, non-reactive aggregates were used. However, where aggregate reactivity was considered a possibility, a limitation was imposed on cement alkalies throughout containment construction, and/or an effective pozzolan was used in combination with the cement.

Chemical reactions of aggregates with both fast and slow reaction rates were recognized as early as 1940. The petrographic method to identify the reactive constituents in concrete aggregates was first published in the 1948 ASTM Proceedings [19]. In 1961, Title No. 58-24 "Selection and Use of Aggregates for Concrete" was approved and published by ACI Committee 621. Subsequently, in 1962, Title No. 59-57 "Durability of Concrete in Service" was approved and published by ACI Committee 201. Both documents provide guidance in selection of aggregates and cements to avoid alkali-aggregate reactions. The requirements and guidelines addressed in the ACI Manual were strictly followed ~~for PWR containment~~ (3) design and construction. Although some highway pavements and bridges have exhibited aggregate reaction degradation, there is no evidence of degradation due to reactive aggregates in PWR concrete containment components.

4.1.4.3 Summary of Reactions with Aggregates

If the aggregate used for PWR containment construction was taken from geographic regions other than those known to yield aggregates suspected of or known to cause alkali-

(3) Need reference

① Need edition dates.

② pH of Concrete? Aqueous solution? Will this require core sampling? See p. 4-1.

aggregate reactions, or if it was investigated, tested, and subject to a petrographic examination conducted in accordance with ASME Section III, Division 2 Class CC [8], ASTM C295 [17], or ASTM C227 [18], which showed that the aggregate used in containment construction is non-reactive, then reactions with aggregates is not a significant age-related degradation mechanism for PWR containment concrete components, and no further evaluation is required.

①

If the aggregate was examined and found to be potentially reactive, but if the provisions of ACI 201.2R-77 [9] or their justified equivalent were adhered to, then reactions with aggregates is not a significant age-related degradation mechanism, and no further evaluation is required.

These conclusions apply to all PWR containment concrete components:

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade
3. Concrete containment wall below grade
4. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that the assumptions and criteria for non-reactive aggregate, and the restrictions on the use of potentially reactive aggregate are met.

4.1.5 Corrosion of Embedded Steel

4.1.5.1 Mechanism Description

Concrete's high alkalinity (pH > 12.5) provides an environment around embedded and reinforcing steel which protects it from corrosion. However, if the pH is reduced (pH < 11.5) by the intrusion of aggressive ions [e.g., chlorides > 500 ppm], corrosion can occur [4]. A reduction in pH could be caused by the leaching of alkaline products through cracks, entry of acidic materials, or carbonation. Chlorides could be present in constituent materials of the original concrete mix (i.e., cement, aggregates, admixtures and water), or ②

②

(V) I a thro a condition or a unknown.

introduced environmentally. The severity of corrosion is influenced by the properties and type of cement and aggregates, and the concrete moisture content.

Corrosion products have a volume greater than the original metal. The presence of corrosion products subjects the concrete to tensile stress, eventually causing hairline cracking, followed by rust staining, spalling, and more severe cracking. These actions will expose more reinforcing steel to a potentially corrosive environment and the concrete to further deterioration. A loss of bond between the concrete and embedded or reinforcing steel will eventually occur, along with a reduction in bar cross section. These conditions can ultimately impair structural integrity.

The degree to which concrete will provide satisfactory protection for embedded or reinforcing steel is in most instances a function of the quality of the concrete and the depth of concrete cover over the steel. The permeability of concrete is also a major factor affecting corrosion resistance. Concrete of low permeability contains less water under a given exposure and hence is more likely to have low electrical conductivity and better resistance to corrosion. Such concrete also resists absorption of salts and their penetration to the embedded or reinforcing steel and provides a barrier to oxygen which is an essential element of the corrosion process. Low water-to-cement ratios and adequate air entrainment increase resistance to water penetration and thereby provide greater resistance to corrosion [9].

4.1.5.2 Significance to License Renewal

(1)

Contianment concrete is of high quality with relatively high strength (4000 psi), low water-to-cement ratio (0.35 to 0.45), and air entrainment (3 to 6 percent). In addition, the aggregates used are well graded, which contributes significantly to low permeability. Containment structures designed in accordance with ACI 318 [12] or ASME Section III Division 2 [8] have concrete cover over embedded and reinforcing steel to provide corrosion protection. The existence of concrete cover over the embedded steel, together with the properties of good quality, well-consolidated, and properly cured concrete, prohibit significant deterioration of embedded and reinforcing steel due to corrosion. Industry standards such as ACI 318 [12] are designed specifically to minimize cracking through reliance on proper reinforcement distribution.

The primary areas where aggressive ions could be present are along the exterior surfaces of the containment shell and dome where moisture and oxygen may have access to the outermost layer of embedded or reinforcing steel. Chlorides, either from the atmospheric release of industrial/chemical plants or which exist at or near ocean sites, could gain access

(1) Loose words
(2) Concrete?

to the steel through existing cracks in the concrete. However, these above ground locations are exposed to an aggressive environment only intermittently; below grade exterior surfaces, especially in the zone of fluctuating water level, could be exposed to aggressive groundwater on a more or less continuous basis. Only those exterior concrete components that are exposed to an aggressive environment on an ongoing basis are susceptible to embedded steel corrosion. (1) (1)

4.1.5.3 Summary of Corrosion of Embedded Steel

(2) If PWR containment structure embedded steel components are not exposed to an aggressive environment (pH ≤ 11.5 or > 500 ppm chlorides with oxygen available [4]), then age-related degradation due to corrosion of embedded steel will not be significant. No further evaluation is required for the following concrete components that are not exposed to groundwater:

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that the assumptions and criteria given above are met.

If the groundwater is not aggressive as defined above, then below grade concrete components will not be subject to significant degradation due to corrosion of embedded steel.

If an aggressive environment as defined above is present but the containment structure concrete is relatively high strength (4000 psi), has a low water-to-cement ratio (0.35 to 0.45), and adequate air entrainment (3 to 6 percent), it will have low permeability, and if the containment structure was designed in accordance with ACI 318 [12] or ASME Section III Division 2 [8], the reinforcement distribution will minimize crack development and the concrete cover over embedded steel components will effectively prohibit exposure of embedded steel components to the corrosive environment, thereby preventing significant age-related degradation of the embedded steel components due to embedded steel corrosion. (3)

(3) Were there cases that containments were designed to ACI 318 but with strength < 4000 psi?

① Loose words
② The absolute strength loss can be big.

If PWR containment embedded steel is exposed to aggressive groundwater for extended ① periods, then age-related degradation due to embedded steel corrosion is potentially significant. Further evaluation of the following PWR containment concrete components and their susceptibility to embedded steel corrosion is provided in Section 5.1:

Reinforced and Prestressed Concrete Containments

1. Concrete containment wall below grade
2. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

4.1.6 Elevated Temperature

4.1.6.1 Mechanism Description

The compressive strength, tensile strength, and modulus of elasticity of concrete are reduced when it is subjected to prolonged exposure to ① elevated temperatures. Figures 4-4, 4-5, and 4-6 suggest that reductions in excess of 10% begin to occur in the range of 180 to 200F [20]. Test condition variables include:

- prevention of moisture loss;
- conditions of loading during specimen heating; and
- specifics of concrete mix proportions, specimen size, degree of curing, and length of time that specimen was heated and allowed to stabilize before load testing.

There are some generalizations [20] that can be made relative to elevated temperature effects on concrete, namely:

- ② Strength loss is minimized when concrete is heated while in a loaded condition. For containments, dead load will always be applied during heating. Thus, this factor has a positive influence on containment concrete strength.
- Lean concrete mixes (low cement content) lose less strength than rich mixes (high cement content). Concrete mixes used for PWR containments are relatively rich which has a negative influence on containment strength.

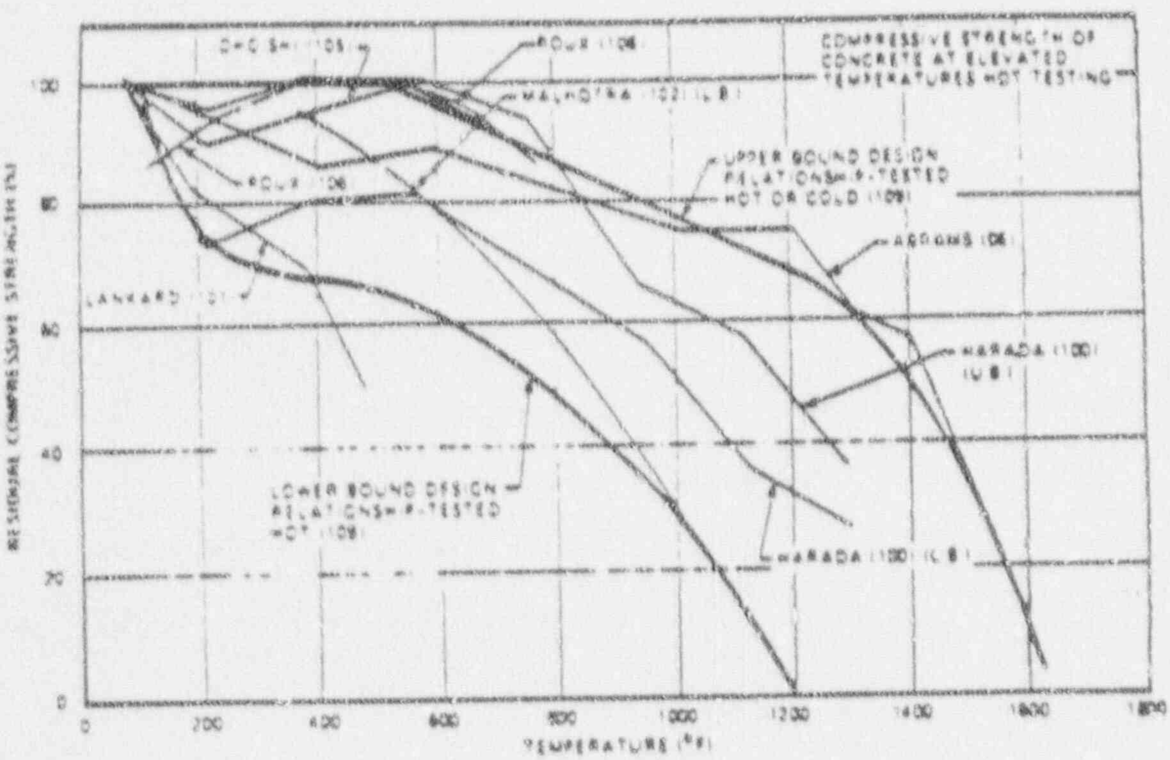

Figure 4-4
Effect of Temperature Exposure on Compressive
Strength of Concrete (Hot Testing)
(Source: Reference 20)

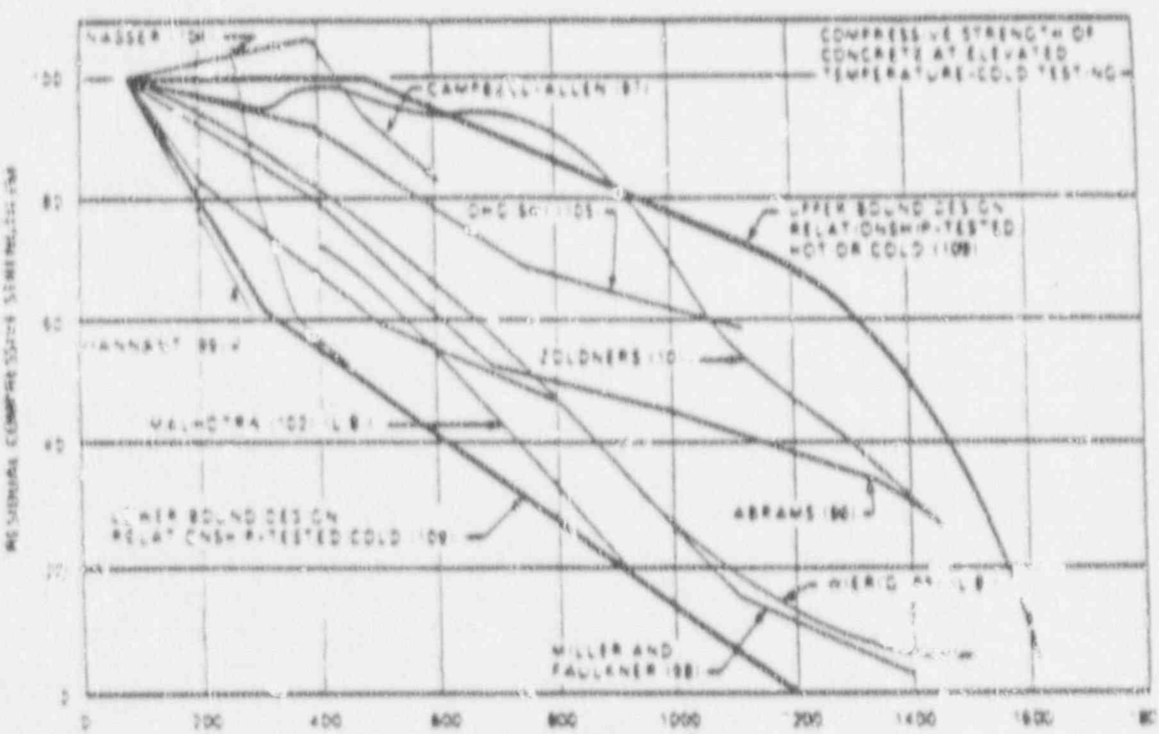

Figure 4-5
Effect of Temperature Exposure on Compressive
Strength of Concrete (Cold Testing)
(Source: Reference 20)

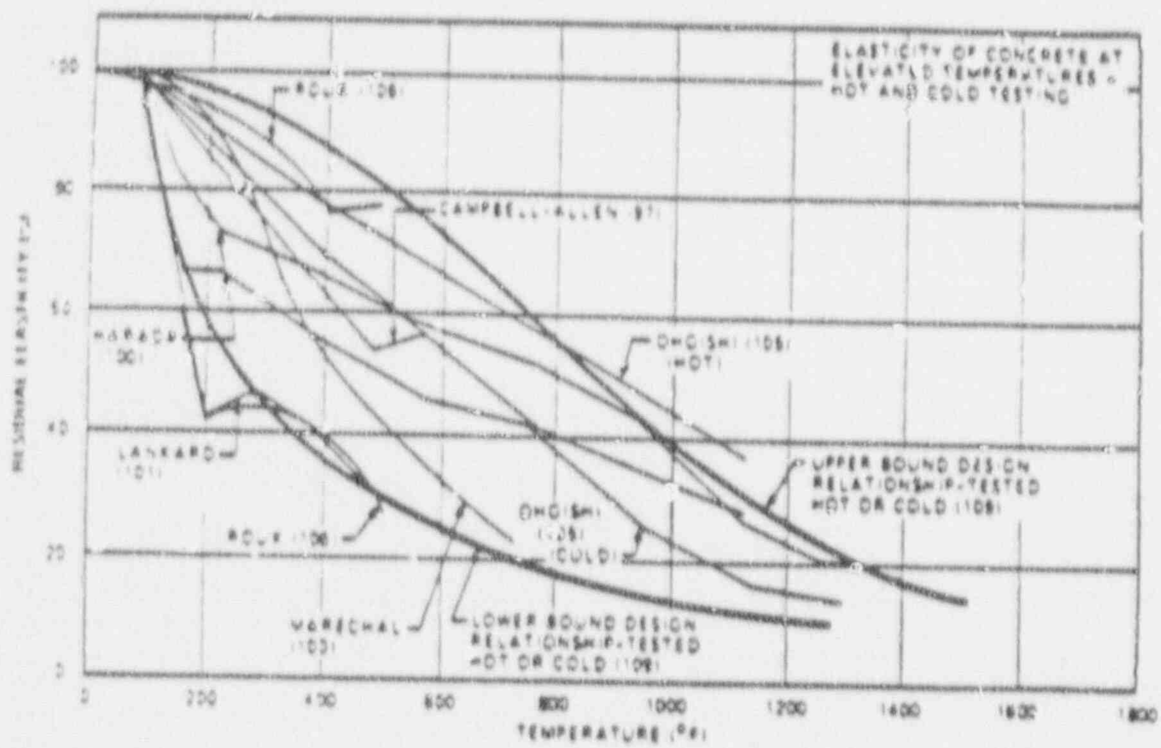

Figure 4-6
Effect of Temperature Exposure on Modulus of
Elasticity of Concrete (Hot and Cold Testing)
(Source: Reference 20)

~~Appropriately~~ ... ?

(3) Misleading; why not ACI 359?

- The reduction in modulus of elasticity is more pronounced than that of compressive strength, in the temperature range of concern for this report. ~~Containment integrity is more directly affected by strength than modulus of elasticity.~~ Thus, the reduction in modulus of elasticity is not significant. (1)

- Thermal cycling of temperature causes a greater strength loss than exposure to a single rise to the same temperature level. Containment concrete is subjected to thermal cycling. Therefore, this fact has a negative influence on the containment structure during the license renewal term.

- Finely crystalline aggregates are more durable from a thermal standpoint than coarse-grained aggregates. The grain structure of PWR containment concrete aggregate is plant-specific. Therefore, the conclusion for this factor is plant-specific.

- Concrete age affects the magnitude of strength loss due to exposure to elevated temperatures; the older the concrete, the lower the strength loss. This will have a positive influence on containment performance during the license renewal term.

As a result of long term exposure to high temperatures (> 300F), surface scaling and cracking may be exhibited. Otherwise, there is no visible physical manifestation of concrete degradation due to elevated temperatures.

4.1.6.2 Significance to License Renewal

(3)

(2)

Containment concrete generally does not experience temperatures higher than 120 to 150F during normal operation. At temperatures in this range, design standards for concrete structures (e.g., ACI 318 [11] and 349 [21]) do not require any special considerations. Section CC-3440 of the ASME Boiler and Pressure Vessel Code, Section III, Division 2 [8] indicates that as long as concrete temperatures do not exceed 150F, aging due to elevated temperature exposure will not be significant. Whereas PWR containment concrete does not experience bulk temperatures higher than 150F during normal operation, concrete near penetrations could be exposed to higher temperatures, depending upon the effectiveness of penetration design and the use of cooling coils and/or insulation. ACI 349 [21] allows local area temperatures to reach 200F before special provisions are required. If containment temperatures do not exceed these code values, elevated temperature is not a significant age-related degradation mechanism for PWR containment concrete. Where containment concrete is subject to localized temperatures in excess of 200F, provisions are included in the design to accommodate these conditions. Where such provisions have been made (e.g., the use of high temperature concrete around hot pipes that penetrate the containment wall), the elevated temperature will not cause significant age-related degradation during the license renewal term. (3)

① Use consistent words

4.1.6.3 Summary of Elevated Temperature

① If PWR containment normal bulk operating temperatures are maintained below the threshold degradation temperature of 150F [8] and local area temperatures are maintained below the limit of 200F [21], exposure of PWR containment concrete to elevated temperatures is not a significant age-related degradation mechanism, and requires no further evaluation.

35-9

Further, if the above temperature limits are exceeded, but plant-specific justification is provided in terms of containment concrete strength properties at elevated temperature or as the result of the application of other special provisions described by ACI 349 [21], ~~or their justified equivalent,~~ Delete then exposure of PWR containment concrete to elevated temperatures will not cause significant age-related degradation. This conclusion applies to all PWR containment concrete components:

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade
3. Concrete containment wall below grade
4. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that either the above temperature limit assumptions and criteria are met, or, if the temperature limits are exceeded, appropriate and justified special provisions have been made.

4.1.7 Irradiation

4.1.7.1 Mechanism Description

Concrete can undergo changes in properties if exposure to neutron and/or gamma radiation exceeds certain levels. The following conclusions relevant to containment concrete have been made in the literature [22]:

① Inconsistent; any margin? Use consistent words
② Inconsistent; missing discussion?

- Heat caused by radiation effects in the aggregates and matrix (up to 250F temperature increase considered possible) may cause a reduction in mechanical properties, loss of moisture, and volume change.

- Neutron radiation with a fluence in excess of 10^{19} neutrons/cm^2 may have a detrimental impact on Portland concrete's mechanical properties. ①

- The effect of secondary gamma irradiation, produced by neutrons at the liner, is currently unknown. ②

- Radiation effects on concrete attenuate to insignificance at distances of approximately 20 inches from the exposed face.

- Nuclear radiation seems to have little effect on the shielding properties of concrete beyond the moisture loss produced by heating.

The effects of radiation on concrete mechanical properties are illustrated in Figures 4-7 through 4-10. Radiation degradation of concrete is not readily observable.

4.1.7.2 Significance to License Renewal

Substantial shielding reduces the neutron flux and energy reaching the PWR containment shell, resulting in levels of accumulated exposure during the course of normal operation that are far below the levels necessary to cause degradation. This shielding is provided by the water inside the reactor vessel, the reactor vessel itself, the biological shield wall (concrete) and a substantial air gap. Neutron flux at the nearest containment shell location varies depending on the individual plant geometry, power level, and fuel type. However, PWR neutron fluence levels at the containment wall are typically less than 10^{17} neutrons/cm^2. These levels may be exceeded at very localized areas but would be no larger than 10^{18} neutrons/cm^2, which remains well below the threshold (10^{19} neutrons/cm^2) [22] for age-related radiation or radiation heating degradation. ① The maximum integrated gamma dose at the outside of the reactor pressure vessel corresponding to 80 years of operation is 9.3×10^9 rads which is below the 10^{10} rad dose at which measurable degradation begins (Figure 4-10). Further, the gamma radiation dose for containment concrete is also substantially mitigated due to distance and shielding, making the effects of gamma radiation on containment concrete during the license renewal term insignificant. ②

4.1.7.3 Summary of Irradiation

If the neutron fluence levels and maximum integrated gamma doses that will be incurred by the containment concrete components throughout the license renewal period do not

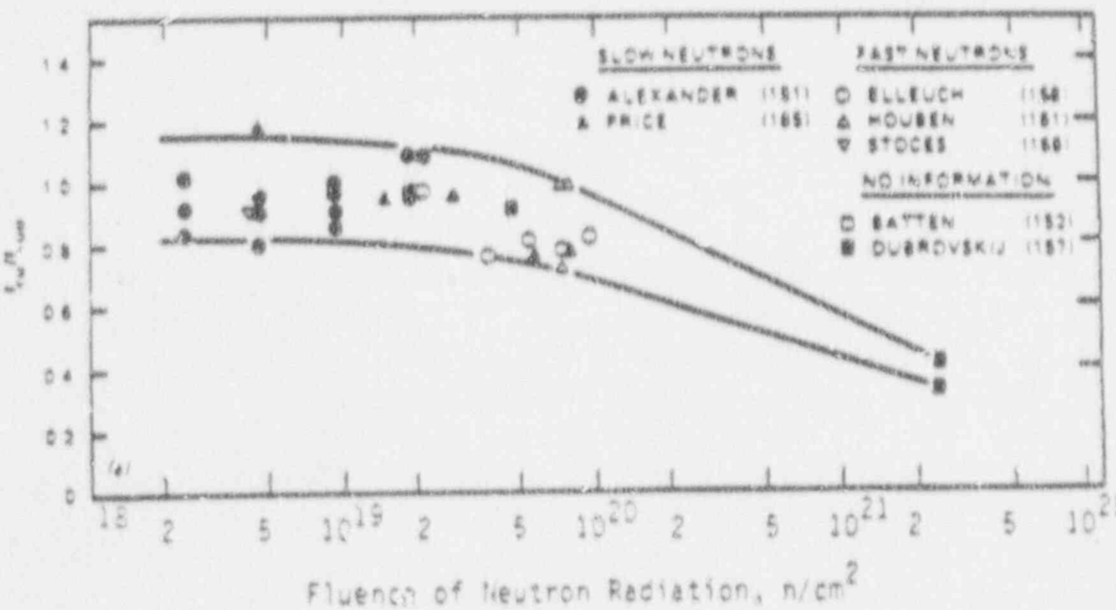

Figure 4-7
Compressive Strength of Concrete Exposed to Neutron Radiation f_{cu}, Related to Strength of Untreated Concrete f_{cuo}
(Source: See Reference 22)

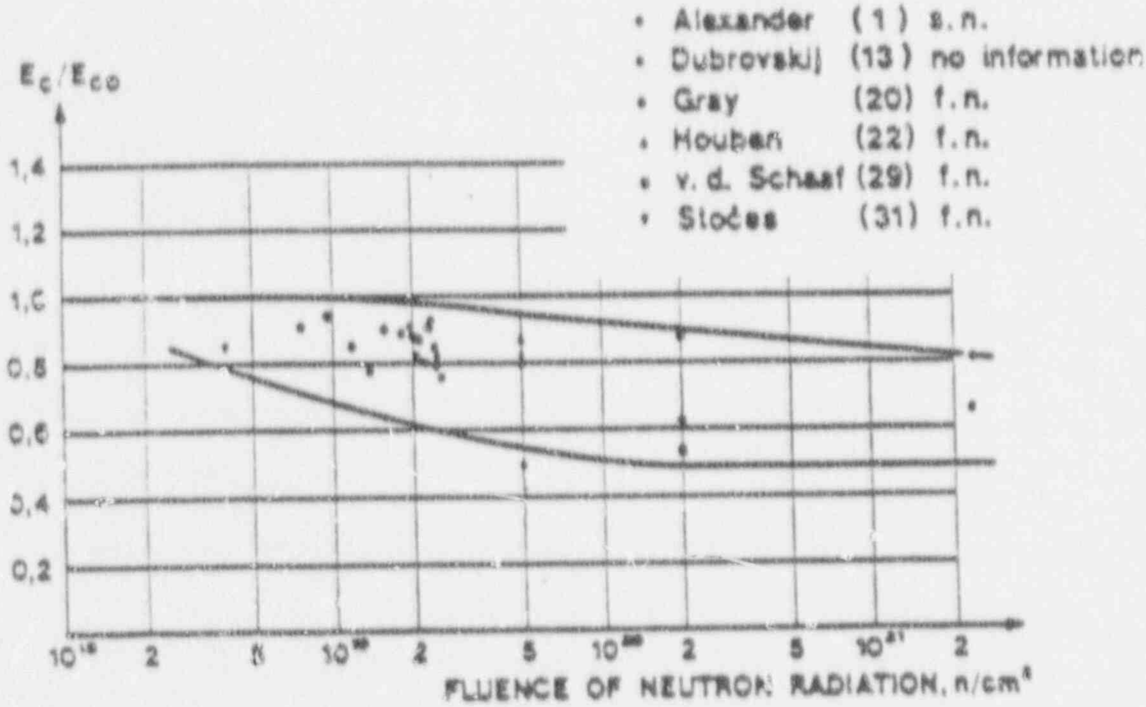

Figure 4-8
Modulus of Elasticity of Concrete After Neutron
Radiation E_c, Related to Modulus of Elasticity E_{co}
(Source: See Reference 22)

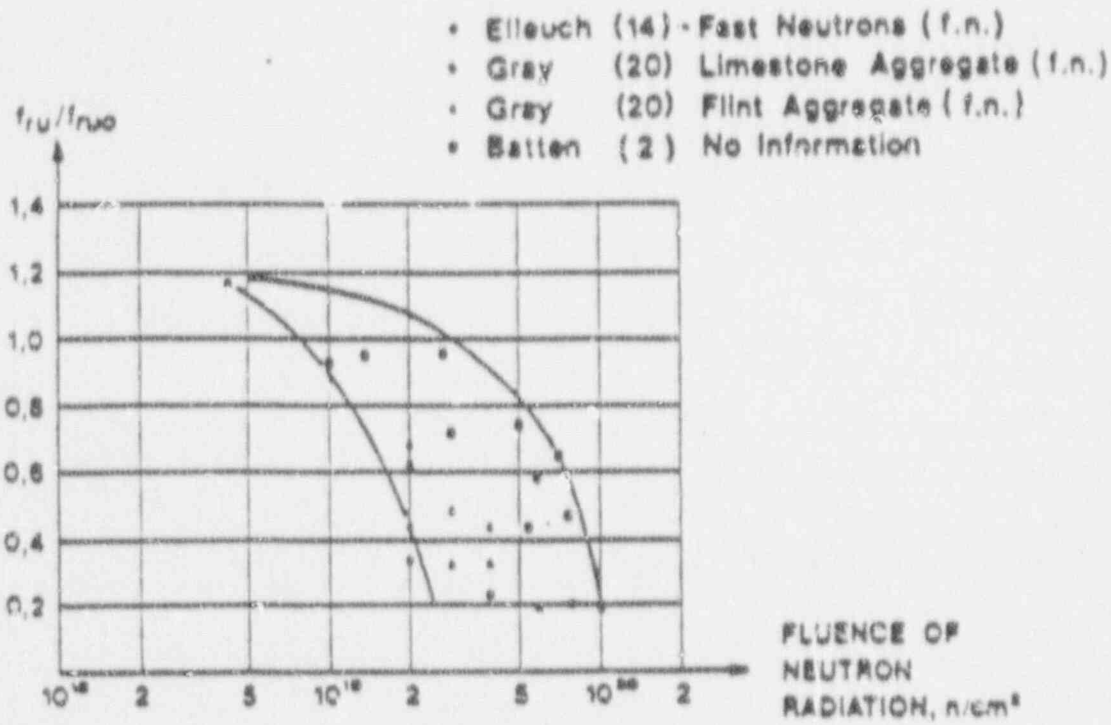

Figure 4-9
Tensile Strength of Concrete Exposed to Neutron
Radiation f_{ru}, Related to Strength of Untreated Concrete f_{ruo}
(Source: See Reference 22)

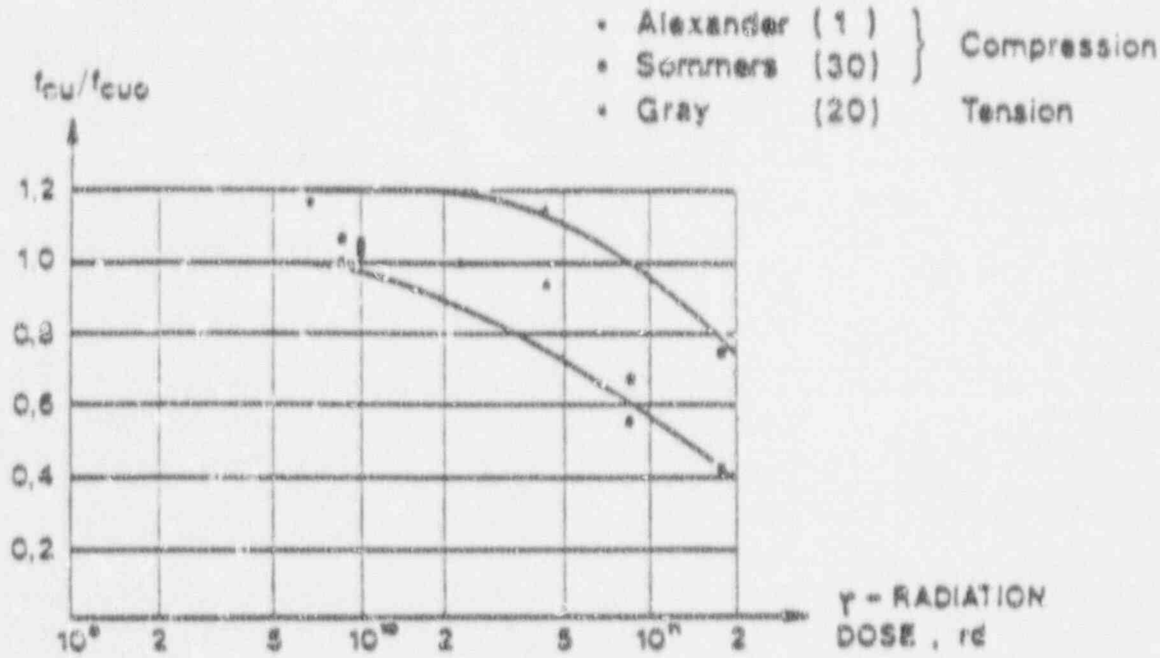

Figure 4-10
Compressive and Tensile Strength of Concrete
Exposed to Gamma - Radiation f_{cu}, Related to Strength of
Untreated Concrete f_{cuo}
(Source: See Reference 22)

(1) Use consistent words

exceed the degradation threshold (1) values of 10^{19} neutrons/cm^2 [22] and 10^{10} rads [12], respectively, then irradiation is not a significant age-related degradation mechanism for PWR containment concrete and requires no further evaluation for all PWR containment concrete components:

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade
3. Concrete containment wall below grade
4. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that these assumptions and criteria are met.

4.2 REINFORCING STEEL (REBAR)

Reinforcing steel (rebar) is designed to resist both tensile and compressive forces in concrete containment structures. The rebar is provided to supplement the concrete compressive strength as required by the design loading. The inability of concrete to resist tensile loads requires that rebar be provided to resist the tensile forces. The following subsections focus on the age-related degradation mechanisms that may affect PWR containment reinforcing steel.

4.2.1 Corrosion

4.2.1.1 Mechanism Description

Protection of reinforcing steel is dependent upon the quality of the concrete and its ability to exclude environmental factors that promote corrosion. Oxygen, moisture, and aggressive ions, most notably chlorides, must be present for electrochemical corrosion to occur. Other factors which may affect the rate of corrosion are: lack of uniformity in the concrete and steel; the pH of the pour water; carbonation of the Portland cement paste; cracks in the concrete; and galvanic effects due to contact between dissimilar metals. Design features such as mix proportions, depth of cover over reinforcing steel, crack control measures, and implementation of measures designed specifically for corrosion protection, play an important role in preventing the onset and/or controlling the rate of rebar corrosion [23]. Refer to Subsection 4.1.5 for additional discussion of corrosion.

Deterioration of concrete from hairline cracking, rust spalling, and more severe cracking may be the result of rebar corrosion. In time, structural distress may occur as a result of either the loss of bond between the steel and concrete or because of a reduction in rebar cross section. These conditions can lead to impairment of structural integrity.

As discussed in Subsection 4.1.5, the quality of concrete and continuity of concrete cover over the reinforcing steel play important roles in preventing rebar corrosion. Protection of rebar from stray electrical currents, isolation of rebar from dissimilar metals, application of coatings, and prevention of exposure to moisture are some other techniques used to prevent rebar corrosion.

4.2.1.2 Significance to License Renewal

As discussed in Subsection 4.1, PWR containment concrete is sound and durable. While some cracks could exist, the distribution of reinforcing steel in the containment structure is designed to control the width of these cracks, thereby minimizing the reinforcing steel corrosion potential. Containment structures designed in accordance with ACI 318 [11] have sufficient concrete cover over rebars to provide adequate corrosion protection. In addition, the concrete and ingredient material properties of PWR containment structures conform to ACI and ASTM standards, thereby assuring good quality, well consolidated, and properly cured concrete, reducing the potential for corrosion. Further, industry standards such as ACI 318 assure that cracking is controlled through reliance on reinforcement distribution.

[12 ?

[12] ?

The primary areas of susceptibility to rebar corrosion are on the exterior surfaces of the containment structure, where moisture, oxygen, and aggressive ions have potential access to the rebar. Chloride in groundwater in excess of 500 ppm [11] can make rebar in the groundwater fluctuation zone susceptible to corrosion. Below this zone, insufficient oxygen is present and above this zone, insufficient water is available for this mechanism to be active.

4.2.1.3 Summary of Corrosion

If reinforcing steel is not exposed to aggressive ions in solution (< 500 ppm chlorides [11]), then age-related degradation due to corrosion of the reinforcing steel will not be

[12] ?

② [12] ?

significant, and no further evaluation is required. The following PWR containment components are not exposed to aggressive ions in solution (aggressive groundwater):

Reinforced and Prestressed Concrete Containments

1. Dome reinforcing steel
2. Containment wall reinforcing steel above grade

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that this criterion is met.

If PWR containment concrete is subject to an aggressive environment on an intermittent basis, but the containment structure concrete conforms to industry standards ~~such as~~ ACI 318 [11] or its equivalent, thereby ensuring a dense, well-cured concrete with reinforcement distribution designed to control cracking, then corrosion of the reinforcing steel will not be a significant age-related degradation mechanism. ① Deli ②

If the PWR concrete containment structure is exposed to significant concentrations of aggressive ions in solution (i.e., > 500 ppm chlorides [11]) on a sustained basis and has a ready supply of oxygen, corrosion of the reinforcing steel is a potentially significant degradation mechanism and must be evaluated further. For PWR containment structures, the areas of susceptibility are limited to below grade exterior walls and the basemat. Further discussion of reinforcing steel corrosion is provided in Section 5.1 for the following components: ② ①

Reinforced and Prestressed Concrete Containments

1. Containment wall reinforcing steel below grade
2. Basemat reinforcing steel

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Basemat reinforcing steel

4.2.2 Elevated Temperature

4.2.2.1 Mechanism Description

Hot rolled reinforcing bars exhibit a reduction in yield strength and modulus of elasticity at elevated temperatures. At 700F, the reduction is limited to 15%. Above this

① Use consistent words

temperature, the reductions become more pronounced, reaching about 50% at 1100F and more than 80% at 1400F. At temperatures up to 600F, the overall structural integrity of the rebar/concrete combination will not be significantly affected [24].

Reinforcing steel exposed to high temperatures could expand, causing concrete cracking and spalling and possibly a lessening of bond if the concrete in immediate contact with it undergoes dehydration.

4.2.2.2 Significance to License Renewal

A temperature of 700F must be reached before significant reductions in yield strength and modulus of elasticity will occur. Since normal operating temperatures within PWR containments are much lower (120 to 150F), elevated temperature effects on reinforcing steel will not cause significant age-related degradation.

4.2.2.3 Summary of Elevated Temperature

If the temperatures experienced by PWR containment structures are below 600F, the threshold temperature at which the structural integrity of the rebar/concrete combination begins to be significantly affected [24], then elevated temperature is not a significant age-related degradation mechanism for the containment structure reinforcing steel and requires no further evaluation for any PWR containment concrete component:

Reinforced and Prestressed Concrete Containments

1. Dome reinforcing steel
2. Containment wall reinforcing steel above grade
3. Containment wall reinforcing steel below grade
4. Basemat reinforcing steel

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Basemat reinforcing steel

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that these assumptions and criterion are met.

① Use consistent words

4.2.3 Irradiation

4.2.3.1 Mechanism Description

Steel degradation due to neutron irradiation results from the displacement of atoms from their normal lattice positions to form both interstitials and vacancies. The effect of this mechanism is to increase the yield strength, decrease the ultimate tensile ductility, and increase the ductile to brittle transition temperature [25,26]. These defects on a macroscopic level produce what is referred to as radiation induced embrittlement, which is encountered in the design and operation of reactor pressure vessels. Currently available data indicate that these effects on the mechanical properties of the steel are measurable at 10^{18} neutrons/cm^2 (E > 1 MeV) [25].

Exposure to high neutron irradiation can lead to changes in the mechanical properties of the steel, as seen by comparing the stress-strain curves for unirradiated and irradiated mild steels in Figure 4-11 [26]. As shown, the yield strength of the steel increased from 36 ksi to 94 ksi and the ultimate strain decreased from 44% to 21%. The latter represents a reduction in ductility of rebar subjected to high radiation exposure (10^{19} neutrons/cm^2).

①

4.2.3.2 Significance to License Renewal

Radiation does not have the potential to cause significant degradation to PWR containment structure reinforcing steel during the license renewal period, since the radiation fluence and flux levels anticipated during normal operation (10^{14} neutrons/cm^2) are well below the threshold of degradation (10^{18} neutrons/cm^2). Radiation degradation of reinforcing steel is not visibly observable.

4.2.3.3 Summary of Irradiation

①

If the cumulative neutron flux experienced by reinforced concrete PWR containment structures is below the 10^{19} neutrons/cm^2 degradation threshold [26], then irradiation of PWR containment reinforcing steel requires no further evaluation. This conclusion applies to the reinforcing steel in the PWR containment concrete components listed below.

Reinforced and Prestressed Concrete Containments

1. Dome reinforcing steel
2. Containment wall reinforcing steel above grade
3. Containment wall reinforcing steel below grade
4. Basemat reinforcing steel

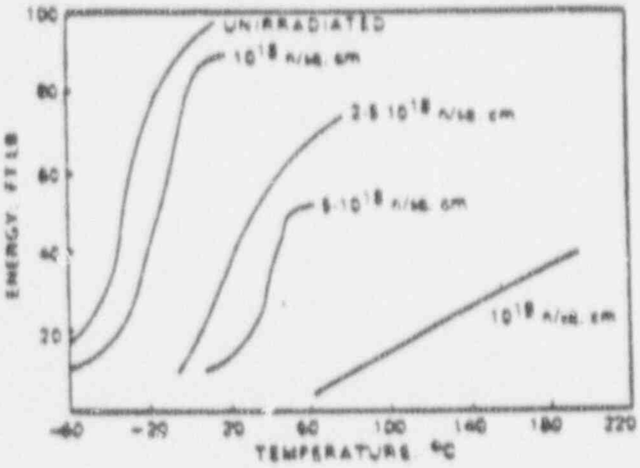

Figure 4-11. Charpy V-Notch Energy/Temperature Curves for Unirradiated and Irradiated Mild Steel
(Source: See Reference 26)

(1) Should discuss microbiologically induced corrosion. See comment #1, p. 3-22.

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Basemat reinforcing steel

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that these assumptions and criteria for cumulative reinforcing steel fluence are met.

4.3 PRESTRESSING SYSTEM

4.3.1 Corrosion (1)

4.3.1.1 Mechanism Description

When corrosion of prestressing tendons occurs, it is generally in the form of localized corrosion. Most corrosion-related failures of prestressing tendons have been attributed to pitting, stress corrosion, hydrogen embrittlement, or some combination of these [27]. Pitting is a highly localized form of corrosion. The primary parameter affecting its occurrence and rate is the environment surrounding the metal. The presence of halide ions, particularly chloride ions, is associated with pitting corrosion.

Stress corrosion results from the simultaneous presence of a conducive environment, a susceptible material, and tensile stress. The environmental factors known to contribute to stress corrosion cracking (SCC) in carbon steels are hydrogen sulfide, ammonia, nitrate solutions, and seawater. Prestressing tendon anchor heads which are constructed of a high strength, low alloy steel bolting material, are subject to SCC.

Hydrogen embrittlement (technically not a form of corrosion) occurs when hydrogen atoms, produced by corrosion or excessive cathodic protection potential, enter the metal lattice. Hydrogen produced by corrosion is not usually sufficient to result in hydrogen embrittlement of carbon steel. Cathodic polarization is the usual method by which this hydrogen is produced. The interaction between the dissolved hydrogen atoms and the metal atoms results in a loss of ductility manifested as brittle fracture.

Corrosion of prestressing wires causes cracking or a reduction in wire cross sectional area. In either case, the prestressing forces applied to the concrete are reduced. If the prestress forces are reduced below the design level, a reduction in design margin would result.

(D) Lanner se ~~~~~~

4.3.1.2 Significance to License Renewal

The occurrence of corrosion-related failures of prestressing tendons in containment structures has been limited [27]. The evolution of petroleum-based grease products for use in the tendon ducts has substantially reduced the possibility of tendon corrosion.

Prestressed concrete tendon ducts are lined with steel to prevent interaction between the concrete and the corrosion-inhibiting grease. Potential grease leakage could occur, and would be most likely at the tendon anchorage, with a small potential for leakage through the tendon ducts. Regulatory Guide 1.35 ~~type~~ inspections would identify conditions of leakage before long-term exposure occurs. Delete

Containment concrete members are several feet thick and any grease penetration into the concrete would be minor (a few inches). The greases used are petroleum-based products. According to the "Concrete Construction Handbook" [28], petroleum-based products have no effect on cured concrete. Therefore, age-related degradation due to grease leakage is not plausible. (1)

The potential does, however, exist for significant age-related degradation of tendons and anchor heads due to corrosion.

4.3.1.3 Summary of Corrosion

Corrosion of PWR containment prestressing system tendons and anchor heads is potentially significant and requires further evaluation. Effective programs for managing this degradation are discussed in Section 5.2.

4.3.2 Elevated Temperature

4.3.2.1 Mechanism Description

The effects of exposure of heat-treated and drawn prestressing wire to elevated temperatures are similar to those resulting from the annealing process. There is a loss in tensile (yield and ultimate) strength, and an increase in relaxation and creep losses. These changes in material behavior are due to alterations in the crystal structure of the metal and do not reverse upon cooling. Exposure to temperatures up to 400F reduces the tensile strength of the prestressing wire by approximately 10% [25]. Figure 4-12 illustrates the effects of elevated temperatures on the relaxation properties of prestressing wire. Exposure to a temperature of 140F for 50 years causes a 300% increase in relaxation (percent of

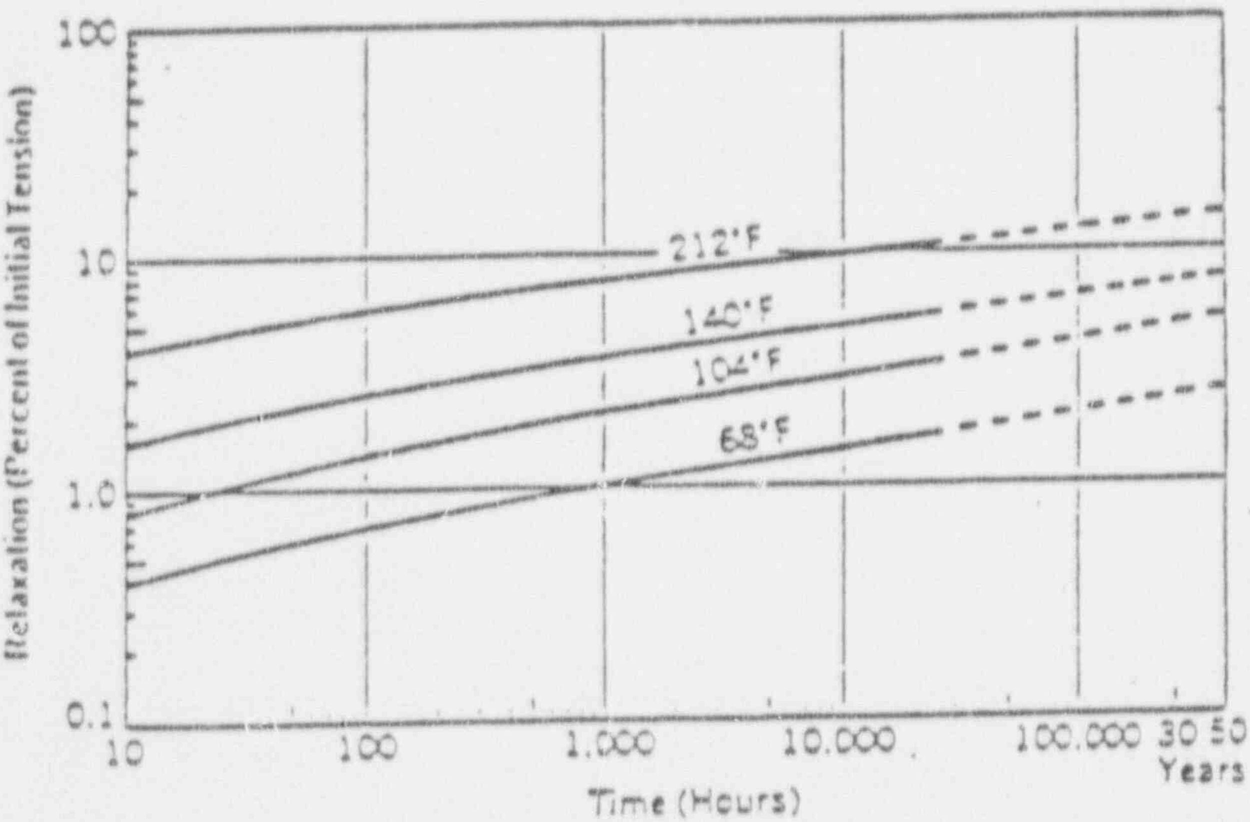

Figure 4-12
Relaxation Properties of Prestressing Wires
At Various Temperatures
(Source: See Reference 25)

(1) Cd. form

initial tension) over wire tested at room temperature (68F) for the same period. The effects of elevated temperatures on prestressing tendons are not readily observable.

4.3.2.2 Significance to License Renewal

PWR containment prestressing tendons that are subjected to temperatures less than 140°F will not experience significant loss of tensile strength. Further, the effect of elevated temperatures on the relaxation and creep properties of containment tendons is considered during design, in calculating the prestress losses. Prestress losses are evaluated in Section 4.3.4.

4.3.2.3 Summary of Elevated Temperature

If the temperatures to which PWR containment prestressing tendons are subjected are below 140F, significant degradation due to elevated temperature exposure will not occur, and no further evaluation is required. A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that these assumptions and criteria. (1)

4.3.3 Irradiation

4.3.3.1 Mechanism Description

Irradiation affects the mechanical properties of steel by dislodging atoms from the metal lattice, creating vacancies and interstitial atoms. This increases the tensile strength of the metal, but reduces the ductility. For prestressing wires and strands, high levels of radiation exposure could cause a decrease in the expected relaxation levels. Studies have shown that exposure of prestressing wire to a neutron fluence of 4×10^{16} neutrons/cm^2 has a negligible effect on the mechanical properties of the wire [21].

High levels of gamma irradiation could cause a loss of viscosity in the grease used in the tendon ducts. Tests of corrosion inhibitors specifically formulated to protect prestressing tendons indicated no changes in physical properties outside the original material specification range when irradiated to 10^{10} rads [25].

(1) Use consistant words

4.3.3.2 Significance to License Renewal

PWR containment tendons and corrosion inhibitors will not receive enough radiation exposure during normal operation to incur age-related degradation and off-normal exposures do not add to this exposure significantly. Radiation exposure levels are well below the 10^{19} neutrons/cm^2 level that is typical for the containment wall, and also well below the degradation threshold. (1)

4.3.3.3 Summary of Irradiation

If the cumulative PWR containment prestressing tendon radiation exposure is less than 4×10^{16} neutrons/cm^2, which has been shown to produce negligible degradation [21], then irradiation of tendons will not cause significant degradation during the license renewal period and requires no further evaluation. A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that these assumptions and criteria are met.

4.3.4 Prestressing Losses

4.3.4.1 Mechanism Description

After the prestressing tendons are tensioned during construction, there is a tendency for the resulting stress to reduce over time. This reduction in stress, termed prestress loss, can be caused by several factors. These are:

- stress relaxation of the prestressing wire,
- shrinkage creep, or elastic deformation of the concrete,
- anchorage seating losses,
- tendon friction, and
- reduction in wire cross section due to corrosion.

With the exception of corrosion-induced wire cross sectional loss (discussed in Subsection 4.3.1), these losses are calculated and considered in the design process.

Prestressing losses are not readily observable.

(1) Guidance to staff not firm -
(2) Loose words (3) be specific

4.3.4.2 Significance to License Renewal

Prestressing losses result in a reduction of the compressive forces applied to the concrete during initial tensioning. If the losses were to exceed those considered in the design, the result would be a reduction in the design margin. Prestressing losses are presently monitored as part of the Inservice Inspection Program, by periodic liftoff tests as described in Regulatory Guide 1.35 [29], and the ASME Code Section XI, Subsection IWL-2520 [30]. (1) These inspection and evaluation programs are described in Sections 5.2.1 and 5.3.1. If losses are greater than expected, the tendons are retensioned or replaced.

(2)

Prestress losses are calculated and considered in the design of prestressed concrete containments. The methods for calculating the losses are well established, and are conservative in the absence of unusual circumstances. However, the existing calculations consider a service life of 40 years. Therefore, it cannot be demonstrated that the losses that will occur in the extended license period will not exceed the design values.

4.3.4.3 Summary of Prestressing Losses

Prestressing losses in the tendon systems for PWR prestressed concrete containments were considered in the design and are periodically monitored as part of the Inservice Inspection Program during the initial license period. (3) These inspection and surveillance programs, which will be continued throughout the license renewal period, are discussed further in Section 5.3.

4.4 LINER

This subsection provides evaluations of age-related degradation mechanisms that are applicable to PWR concrete containment liners. The evaluations of the effects of elevated temperature and irradiation, and the conclusions drawn also apply to free-standing steel containments, as indicated in the summary paragraphs.

4.4.1 Corrosion

4.4.1.1 Mechanism Description

Liner corrosion could be either galvanic corrosion, stress corrosion cracking, or electrochemical corrosion resulting from exposure to aggressive aqueous solutions as described in Section 4.2.1.

Galvanic corrosion occurs when the electrical potential difference between dissimilar metals, placed in contact with each other, results in the flow of electrons between them. The less resistant metal becomes the anode in this couple, and is subject to corrosion, while the more resistant metal becomes the cathode and corrodes very little, if at all [31]. The rate of galvanic corrosion is a function of the potential difference between the metals, the environment in which they are located, polarization behavior of the metals, and the geometric relationship of the metals. Galvanic corrosion reduces the thickness of the anode metal.

The phenomenon of stress corrosion cracking (SCC) may result in fracture of the metal. It is defined as cracking under combined action of corrosion and tensile stresses. The stresses may be either applied (external) or residual (internal). The stress corrosion cracks themselves may be either transgranular or intergranular, depending upon the metal and the corrosive agent. As is normal in all cracking, the cracks are perpendicular to the tensile stress. Usually there is little or no obvious visual evidence of corrosion.

The three principal factors necessary to initiate stress corrosion cracking are:

- tensile stresses
- corrosive environment
- susceptible material.

The tensile stresses necessary to cause SCC must be at or near the material's yield point. This is facilitated when the material is substantially cold worked, contains residual stress from welding, or is subjected to significant applied loads. Corrosive environments which induce SCC are highly material-dependent. For austenitic stainless steels, such as SA-240 Type 304 or 308, halogens (i.e., chlorides) and acids promote SCC. With respect to material susceptibility austenitic stainless steels are prone to SCC, particularly when sensitization is present (such as in heat affected zones) and at creviced geometries.

Corrosion of the liner is possible under certain contributing conditions. This may be true, particularly for the floor liner plate beneath the interior concrete bottom floor slab. Specific conditions of plant operation, humidity, the presence of aggressive fluids, and maintenance practices are involved in determining the degree to which the floor liner plate may be susceptible to corrosion. Corrosion of the liner from the concrete side due to acid rain, salt-containing atmospheres, and groundwater is highly improbable. Cracks which may exist in the concrete will be sufficiently tight to minimize penetration of moisture, oxygen, and chlorides to the degree that could cause significant degradation.

(1) cannot be substantiated
(2) Contraditory to that of p. 3-1

Liner corrosion resulting from exposure to aggressive aqueous solutions which occurs on the side of the liner in contact with the concrete would not produce noticeable distress. Because of liner ductility, the strain produced by expansive forces generated by the creation of corrosion products would be spread over a large area with rather small, localized displacement of the liner. This displacement would be undetectable. Areas of the dome and cylindrical wall liner plate would respond in this manner.

Likewise, corrosion of the floor liner plate would not be easily detected. Corrosion that might occur either on the top or bottom (underside) of the floor liner plate would be resisted by the reinforced concrete bottom floor slab that exists in varying thicknesses in PWR designs. (1)

On exposed surfaces of the liner, protective coatings could lose the ability to adhere to the corroding steel surface and coating degradation would be apparent.

4.4.1.2 Significance to License Renewal

Since the containment liner is constructed from a series of individual steel plates welded together to form a continuous pressure boundary, both the plate material and the welds are subject to the same potential degradation mechanisms. The significance of potential degradation of the liner is considered to apply equally to the plate material and the welds.

Liner corrosion results in a reduction of liner plate thickness. Excessive reductions in thickness could compromise the pressure boundary provided by the liner.

Stress corrosion cracking is an age-related degradation mechanism that affects stainless steels. The PWR containment liner plate is not a load-bearing structural component. (2) The induced strains in the liner plate result from conformation to the concrete containment's deformation which only imposes compressive stresses under normal operating conditions (3) due to dead load and prestress load (for prestressed concrete containments). The environment inside the containment structure is dry under normal operating conditions, and therefore the liner plate is not exposed to corrosive environmental conditions. Therefore, the conditions for SCC to occur do not exist for the PWR containment liner plate, and age-related degradation due to SCC will not be significant.

[12]

The floor liner plate is susceptible to corrosion from exposure to aggressive fluids. Groundwater which has in excess of 500 ppm chlorides [11] could cause liner corrosion. The primary paths of ingress for these fluids are the construction and expansion/contraction joints in the concrete floor slab which covers the bottom floor liner plate, and

(3) Incorrect statement

(1) [illegible]
(2) Incorrect statement

the joint at the junction of this concrete slab and the cylindrical wall (Figure 4-13). Depending upon joint size, the integrity of joint filler and sealant materials, and the length of exposure and chemical makeup of the fluid, this fluid could reach the liner surface. Although some portions of the liner have the added protection afforded by leak chase channels, originally installed during construction to enable leak testing of the welded joints between plate segments, the major surface area of the liner could be exposed to corrosion degradation by this process. Depending upon the continued ability of these leak chase channels to prevent corrosion fluids from reaching the liner seam, welds, and adjacent plate material, accelerated corrosion could occur because this area is not in the high pH environment provided by concrete.

Corrosion which originates from the side of the liner in contact with concrete could initiate in locations where cracks in the concrete retain moisture. Such locations would include portions of the exterior concrete exposed to fluctuating levels of groundwater, and to a lesser degree, dome areas which are relatively flat and can retain rainwater. Prestressed containments are less likely to experience this type of liner corrosion from the concrete side because cracks are fewer and more tightly closed than those which exist in conventionally reinforced containments, such as in the dome and cylindrical wall. The condition of the basemat concrete in prestressed containments would be similar to that of conventionally reinforced containments because the basemat is not prestressed.

4.4.1.3 Summary of Corrosion

(1)

Containment liners are subject to galvanic corrosion, stress corrosion cracking (SCC), or corrosion resulting from exposure to aggressive aqueous solutions (groundwater).

Stress corrosion cracking is a phenomenon that occurs in stainless steels, but becomes significant only if tensile stress and a corrosive environment exist. Because PWR containment liners are not designed to resist mechanical loads and only experience compressive stresses due to dead load and prestress (for prestressed containment structures), age-related degradation of PWR liners from SCC will not cause significant degradation during the license renewal term, and requires no further evaluation. (2)

[12]

If aggressive groundwater (chlorides > 500 ppm) [11] is not present, then corrosion of the containment liner plate due to aggressive aqueous solutions will not occur. Thus, the following PWR containment liner components will not be subject to significant corrosion degradation during the license renewal term:

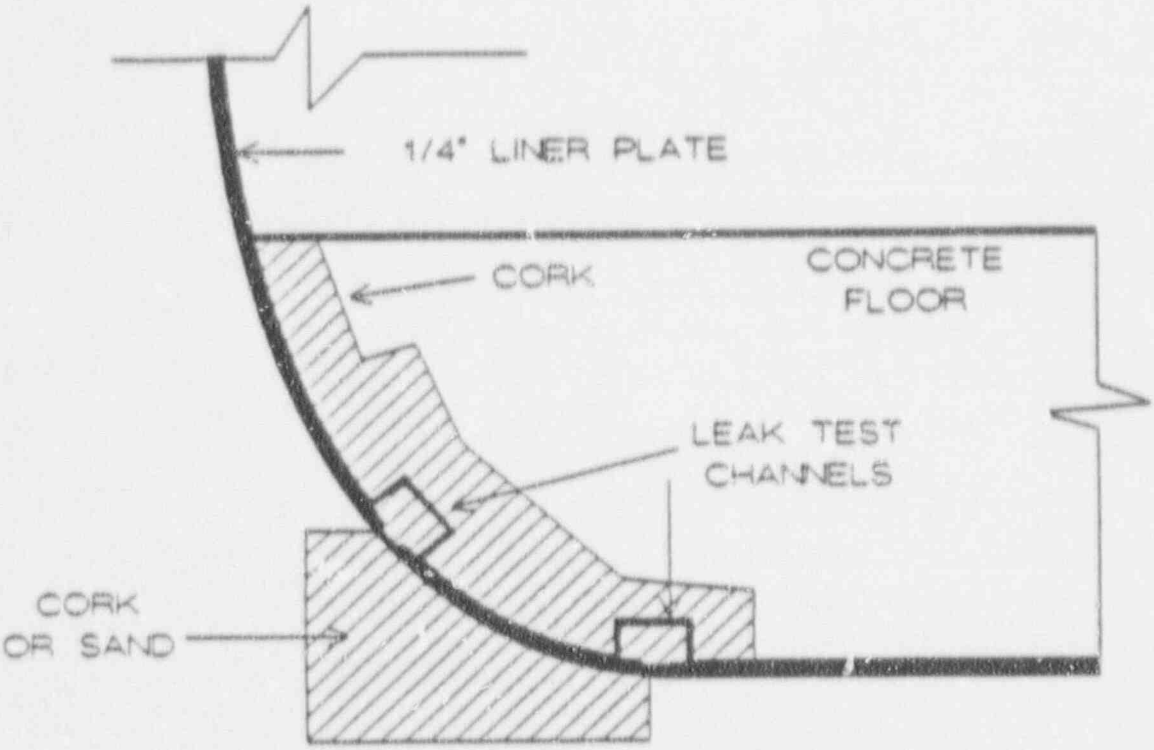

Figure 4-13
Typical Edge Detail at Wall-Base Mat Junction
(Source: See Reference 32)

Reinforced and Prestressed Concrete Containments

1. Containment liner interior surface
2. Containment liner above grade exterior surface
3. Basemat liner interior surface
4. Liner anchors above grade

Common Components

1. Penetration sleeves
2. Dissimilar metal welds
3. Personnel airlock
4. Equipment hatches

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that the above assumptions and criteria are met.

If aggressive groundwater (chlorides > 500 ppm) [11] is present, then corrosion of the liner plate is potentially significant for the PWR containment components listed below. Effective programs to manage this degradation are described in Section 5.4.

Reinforced and Prestressed Concrete Containments

1. Containment liner below grade exterior surface
2. Basemat liner exterior surface
3. Liner anchors below grade

Free-Standing Containments with Flat Bottom and an Ice-Condenser

1. Basemat liner
2. Liner anchors

4.4.2 Elevated Temperature

4.4.2.1 Mechanism Description

The effects of elevated temperatures on concrete PWR containment liners and free-standing steel containment shells are the same as those discussed in Subsection 4.2.2 for reinforcing steel with regard to the effect on yield strength and modulus of elasticity. Relatively high temperatures (700F) must be reached before even small reductions (15 percent) are reflected in these properties.

① Use consistent words

Temperatures sufficient to cause changes in tensile properties would likely cause detectable discoloration and failure of the coating system, and distortion of the liner or free-standing steel shell.

4.4.2.2 Significance to License Renewal

Normal operating temperatures inside containments (120 to 150F) are significantly below those associated with reductions in the tensile properties of the liner or free-standing steel shell. Although higher temperatures might occur locally around penetrations, unless they exceed 700F, they would not alter the properties of the steel. The stresses developed from elevated temperature exposure are expected to be sufficiently low to prevent degradation by low cycle fatigue.

There is evidence that containments have experienced localized bulging of liners near penetrations. This is believed to be the result of thermal loads. This phenomena is a normal operating occurrence that is dependent upon penetration design and the need for or use of penetration cooling. Where this has occurred, mitigative actions have been taken to ensure continued containment integrity.

4.4.2.3 Summary of Elevated Temperature

If PWR containment liner and free-standing steel shell operating temperatures are below the degradation threshold of 700F, then elevated temperature will not cause significant ① age-related degradation during the license renewal term, and no further evaluation is required. This conclusion applies to the following PWR containment components:

Reinforced and Prestressed Concrete Containments

1. Containment liner interior surface
2. Containment liner above grade exterior surface
3. Containment liner below grade exterior surface
4. Basemat liner interior surface
5. Basemat liner exterior surface
6. Liner anchors above grade
7. Liner anchors below grade

(1) Provide Specifics

Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottom

1. Containment shell interior surface
2. Containment shell exterior surface
3. Embedded shell region
4. Sand pocket region

Free-Standing Containments with Flat Bottom and an Ice-Condenser

1. Dome shell interior surface
2. Dome shell exterior surface
3. Cylindrical shell interior surface
4. Cylindrical shell exterior surface
5. Embedded shell region
6. Basemat liner
7. Liner anchors

Common Components

1. Penetration sleeves

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that these assumptions and criterion are met.

4.4.3 Irradiation

4.4.3.1 Mechanism Description

The effects of irradiation on steel liners are similar to those described for reinforcing steel in Subsection 4.2.3. Irradiation causes an increase in the brittle-to-ductile transition temperature, beginning at a neutron fluence of 2×10^{17} neutrons/cm^2 (> 1 MeV) [32] (Figure 4-14).

(1)

For the level of neutron fluence under consideration, no visible evidence of degradation is expected.

4.4.3.2 Significance to License Renewal

The fluence and flux to which the containment liner will be subjected during the license renewal term are far below the levels which could cause a change in liner physical properties.

(1)

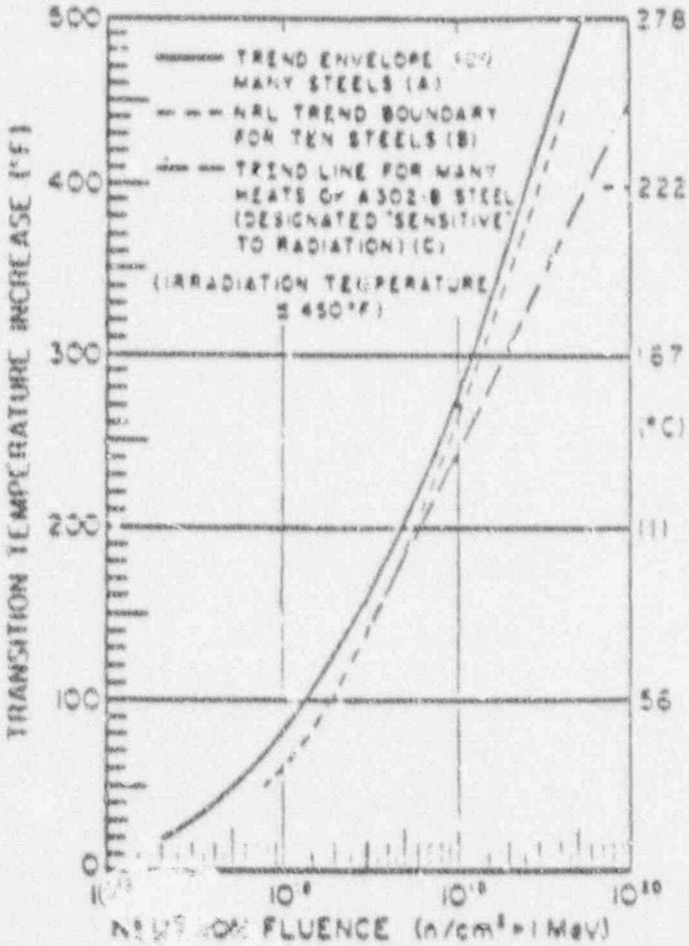

Figure 4-14
Extremes of Radiation Embrittlement Trends
(Source: See Reference 32)

(1) Use consistent words

4.4.3.3 Summary of Irradiation

If the cumulative radiation exposure that ~~will be~~ (1) experienced by concrete PWR containment liners or free-standing steel containment shells throughout the license renewal term is below the degradation threshold [2 x 10^{17} neutrons/cm^2 (> 1 MeV)], then no further evaluation of irradiation is required. This conclusion applies to the PWR containment components listed below:

Reinforced and Prestressed Concrete Containments

1. Containment liner interior surface
2. Containment liner above grade exterior surface
3. Containment liner below grade exterior surface
4. Basemat liner interior surface
5. Basemat liner exterior surface
6. Liner anchors above grade
7. Liner anchors below grade

Free-Standing Cylindrical and Spherical Containments with Elliptical Bottom

1. Containment shell interior surface
2. Containment shell exterior surface
3. Embedded shell region
4. Sand pocket region

Free-Standing Containments with Flat Bottom and an Ice-Condenser

1. Dome shell interior surface
2. Dome shell exterior surface
3. Cylindrical shell interior surface
4. Cylindrical shell exterior surface
5. Embedded shell region
6. Basemat liner
7. Liner anchors

Common Components

1. Penetration sleeves
2. Penetration bellows
3. Personnel airlock
4. Equipment hatches

(1) Loose words

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that these assumptions and criterion are met.

4.5 MISCELLANEOUS AGE-RELATED DEGRADATION MECHANISMS

Three additional age-related degradation mechanisms fall into a miscellaneous category, either because of specificity with respect to a particular containment element or because of commonality of discussion for all containment elements.

4.5.1 Fatigue

4.5.1.1 Mechanism Description

Fatigue damage can be a problem in materials subjected to cyclic loadings. Fatigue is the progressive degradation produced by the cyclic application of loadings which are less than the maximum allowable static loading. Repeated or cyclic loading can ultimately cause fatigue failure.

For concrete components, the effects of fatigue loading initiate as internal microcracking within the hardened concrete paste and at reinforcing steel boundaries. If stress repetitions are great enough, microcracks may extend to the external surface, causing possible fracture of the cover concrete. This fracture may not cause failure of the structure (as compared to brittle fracture in metals), but may promote further debilitation of the exposed reinforcing steel or internal crack propagation. The physical manifestations of fatigue damage to reinforcing steel will be undetectable on the concrete surface.

For steel components, fatigue degradation is not detectable until cracks initiate and grow to detectable size on the surface of the material.

4.5.1.2 Significance to License Renewal

Both concrete and steel PWR containment components can be subjected to cyclic loadings, and therefore are subject to fatigue degradation. Containment concrete, reinforcing ~~steel~~, (1) and free-standing steel containment shells have good fatigue strength properties for hundreds of thousands to millions of cycles of below-yield load application. Low cycle fatigue, which implies high stresses (at or above yield for steels and a high percentage of static strength for concrete) and a relatively few cycles of load application (less than 100), may be more limiting. Review of the loading that containments experience during normal

① Liner is not a load-bearing member
② CLB Maintenance
③ Loose words

operating life indicates that the periodic Type A, integrated leak rate tests are the major source of load changes.

However, the number of cycles of load are generally low (about 15 for a 40-year operating life, or about 23 for a 60-year operating life) with only low to moderate stress levels due to:

- use of load factors in design,
- limitations on allowable stresses imposed by applicable code provisions,
- contribution of liner to containment structural response, ①
- repeatability of spacing of reinforcing steel and tendons,
- actual versus minimum required material properties. ②

Containment concrete, reinforcing steel, prestressing system components, steel liners, and free-standing steel containments are designed to have good fatigue strength properties (10^5 cycles) of below yield load application in accordance with ASME [8], and ACI [33] codes. ③ ④ Localized elevated temperatures are not anticipated to be capable of developing the transient stress conditions capable of causing potentially significant low cycle fatigue, except for hot penetrations without bellows for concrete containments and penetration bellows assemblies of free-standing steel containments.

4.5.1.3 Summary of Fatigue

③

If the structural design provided a good fatigue life consistent with ASME [8] and ACI [33] codes, then fatigue will not cause significant age-related degradation of the following PWR containment components throughout the license renewal term:

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade
3. Concrete containment wall below grade
4. Concrete basemat
5. Dome reinforcing steel
6. Containment wall reinforcing steel above grade
7. Containment wall reinforcing steel below grade
8. Basemat reinforcing steel

Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottom

1. Containment shell interior surface
2. Containment shell exterior surface

④ Lacks details

(1) Editorial - Missing words.

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Dome shell interior surface
2. Dome shell exterior surface
3. Cylindrical shell interior surface
4. Cylindrical shell exterior surface
5. Concrete basemat
6. Basemat reinforcing steel

Common Components

1. Personnel airlock
2. Equipment hatches

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that the above assumptions and criteria are met.

Localized cyclic thermal loading can cause potentially significant fatigue of hot piping penetrations without bellows assemblies for concrete containments, or fatigue of bellows for free-standing steel containments. Where these conditions exist, further evaluation is required. Effective programs for are discussed in Section 5.6 for the following PWR containment components:

(1)

Common Components

1. Penetration sleeves without bellows
2. Penetration bellows for free-standing steel containments

4.5.2 Concrete Interaction with Aluminum

4.5.2.1 Mechanism Description

Concrete strength can be reduced when it is pumped through aluminum piping during placement. This phenomenon was identified around 1969, and specifications for concrete placing began to prohibit the use of aluminum. The major effects of this aluminum/concrete interaction would have occurred during the period immediately following concrete placement when a considerable percentage of strength development takes place.

The reaction between concrete and the aluminum piping inhibits the ability of the concrete to achieve its full strength potential.

(1) See Comment #2, p1-6; structural acceptance tests will not tell the difference.

4.5.2.2 Significance to License Renewal

The tendency to achieve less strength, if it occurred to a significant magnitude, would have been identifiable as an abnormality in the overall response of the containment structure during the initial structural acceptance test at 115 percent of design pressure. Any containment having concrete placed through aluminum pipelines which successfully completed its acceptance tests was not adversely affected by this placing condition. (1)

4.5.2.3 Summary of Concrete Interaction with Aluminum

If, during construction of a concrete PWR containment, aluminum pipelines were not used for concrete placement, then concrete interaction with aluminum requires no further evaluation.

The effects of concrete interaction with aluminum are exhibited soon after placement. If aluminum pipelines were used for concrete placement, the adverse effects would have been identified during the initial structural acceptance test, prior to initial operation. Accordingly, if no degradation of concrete strength was noted during the initial structural testing, then concrete interaction with aluminum is not a significant age-related degradation mechanism for the concrete containment components, and requires no further evaluation for the following:

Reinforced and Prestressed Concrete Containments

1. Concrete dome
2. Concrete containment wall above grade
3. Concrete containment wall below grade
4. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that the above assumptions and criteria are met.

4.5.3 Settlement

4.5.3.1 Mechanism Description

Structures have a tendency to settle as they are constructed and during their early life. The most pronounced settlement is evidenced in the first several months after construction. The amount of settlement depends on the physical properties of the foundation material, which may range from rock (little or no settlement likely) to compacted soil (subject to some degree of settlement) to soft soil (likely to experience some settlement). Settlement of structures may occur during the design life resulting from changes of environmental conditions, such as lowering of the water table.

Structural members of both steel and concrete buildings may be affected by differential settlement between supporting foundations within a building or between two buildings. The alignment of equipment may be offset by uneven settlement of the supporting foundation(s). Rigid components (e.g., piping systems and cable trays) bridging between buildings may similarly be affected by differential displacement at the supporting points.

Settlement is generally small and is commonly determined by survey.

4.5.3.2 Significance to License Renewal

Settlement has not been noted as a widespread age-related degradation mechanism for PWR containment structures. Settlement directly related to construction work is readily evident early in the life of the structure. This construction-related degradation, typically monitored throughout the construction program and compared with settlement design allowances, is not considered an aging mechanism. Sites with soft soil and/or sites with significant changes in underground water conditions over a long period of time may be susceptible to significant settlement. For plants where significant long-term settlement if plausible, monitoring is continued during operation.

4.5.3.3 Summary of Settlement

Settlement of PWR containment structures occurs mostly during construction and in the first several months after construction. However, because of the possibility of changes in site conditions that might affect settlement (i.e., the groundwater table), settlement is a potentially significant age-related degradation mechanism for all PWR containment types. Effective programs for identifying potentially significant settlement are described in Section 5.5.

(1) incorrect statement
(2) loose words; be specific

4.6 FREE-STANDING STEEL CONTAINMENT - DEGRADATION MECHANISMS

Those portions of free-standing steel containments that are backed by or embedded in concrete may be exposed to the same degradation mechanisms identified for the concrete, reinforcing steel, and embedded steel liners of the concrete containments. Those portions of the steel shell that are either embedded in concrete or adjacent to that embedment, as shown in Figure 4-15, are addressed in Subsections 4.1.5 and 4.4. This subsection identifies the applicable degradation mechanisms for those portions of free-standing steel containments that are not backed by or embedded in concrete, and assesses their potential to cause significant degradation of those components during the license renewal period.

4.6.1 Strain Aging

4.6.1.1 Mechanism Description

Strain aging is associated with the redistribution of carbon and nitrogen atoms in cold-worked carbon steels. These atoms migrate to the dislocations (one-dimensional defects of the crystal structure), locking them. Strain aging results in higher yield strength, higher ultimate tensile strength, lower notch toughness, and reduced ductility. As the concentration of the free carbon and nitrogen atoms is decreased, strain aging effects are reduced. Carbon-related strain aging at temperatures below 200F is negligible, due to the low solubility of carbon in this temperature range.

There are two types of strain aging: static strain aging, which occurs after the material has been deformed; and dynamic strain aging, which occurs during plastic straining. Dynamic strain aging is not expected in the carbon steel components of free-standing steel containments during their service life, since the strains associated with the design service loads are below the elastic limit of the material. Static strain aging is possible in the carbon steel plates of free-standing steel containments, which are cold formed during construction with free nitrogen present. At ambient temperatures, static strain aging can result in substantial property changes within two to three years after the material is cold worked. Static strain aging is accelerated with an increase in temperature. (1)

Strain aging degradation is not readily observable.

4.6.1.2 Significance to License Renewal

Materials most susceptible to the phenomenon of strain age embrittlement are low carbon rimmed or capped steels which are severely cold worked during forming processes [34]. (2)

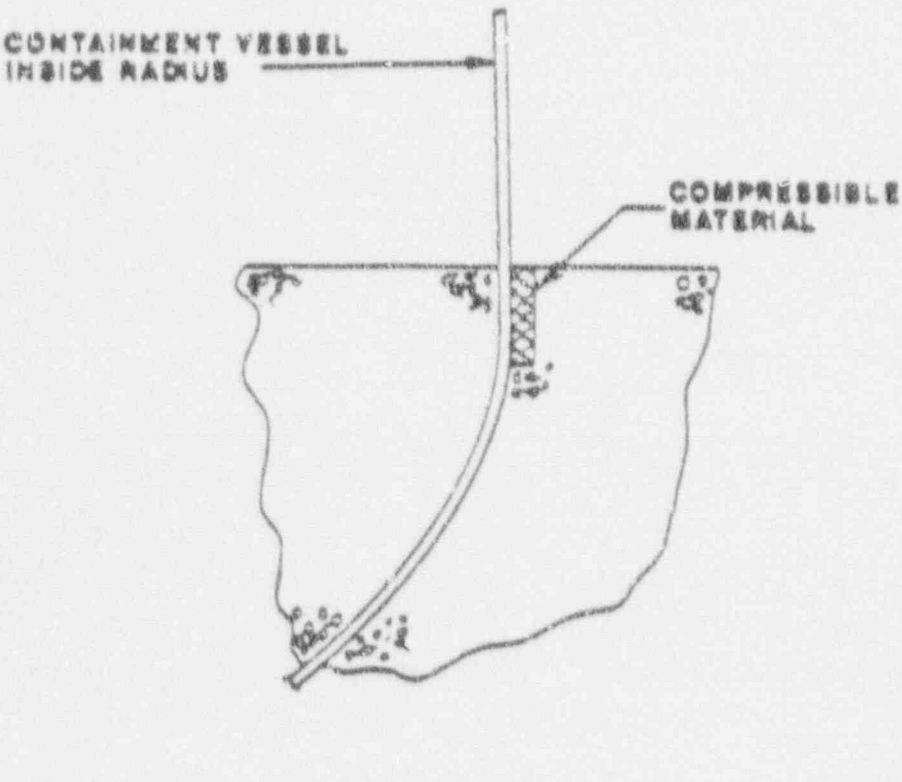

Figure 4-15
Steel Shell/Concrete Interface Zone

(1) Loose words
(2) "safety function" is too limited

The PWR free-standing steel containment is made from SA-516 Grade 70 or SA-212 Grade 70 plate steel, which is a low-carbon steel (0.27 to 0.30% C), and which has been normalized or stress relieved or both, following plate rolling. Cold working is minimal (generally less than 5%) to achieve the desired containment geometry. Strain aging requires stressing of the material to above its yield stress, and aging at temperatures above 200F (ambient temperature strain aging occurs shortly after cold working). The PWR containment has a maximum temperature during normal operation of approximately 150F, and loading conditions do not produce service stresses in the range of the material yield strength.

(1)

4.6.1.3 Summary of Strain Aging

There are two types of strain aging, static strain aging and dynamic strain aging. If the design philosophy of the containment structure does not allow loads to exceed the elastic limit of the material, then dynamic strain aging will not cause significant age-related degradation of free-standing steel containment structures, and requires no further evaluation.

For static strain aging, if the steel used in free-standing steel containment construction was not severely cold worked during the forming process, then static strain aging will not affect the continued safety function performance of free-standing steel containment structures, and requires no further evaluation.

(1)

(1) If severe cold working of the steel was used in the forming process, but the plates are normalized, or stress relieved or both after forming with minimal (< 5%) subsequent cold working, then static strain aging will not affect the continued safety function performance of free-standing steel containment structures, and requires no further evaluation. (2)

A license renewal applicant intending to take credit for this conclusion is responsible for the review/evaluation of plant-specific features, including appropriate CLB information/documents, in order to assure that the above assumptions and criteria are met for the following components:

<u>Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottom</u>

1. Containment shell interior surface
2. Containment shell exterior surface
3. Embedded shell region
4. Sand pocket region

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Dome shell interior surface
2. Dome shell exterior surface
3. Cylindrical shell interior surface
4. Cylindrical shell exterior surface

Common Components

1. Penetration sleeves
2. Penetration bellows
3. Personnel airlock
4. Equipment hatches

4.6.2 Corrosion

4.6.2.1 Mechanism Description

The types of corrosion applicable to free-standing steel containments are general corrosion, galvanic corrosion, and stress corrosion cracking. These degradation mechanisms are described in this subsection.

General corrosion can take place when steel is exposed to oxygen and moisture. General corrosion of an exposed surface results in uniform wall thinning.

Galvanic corrosion occurs when the electrical potential difference between dissimilar metals, placed in contact with each other, results in the flow of electrons between them. The less resistant metal becomes the anode, and is subject to corrosion, while the more resistant metal becomes the cathode and corrodes very little, if at all [31]. The rate of galvanic corrosion is a function of the potential difference between the metals, the environment in which they are located, polarization behavior of the metals, and the geometric relationship of the metals. Galvanic corrosion may reduce the thickness of the bellows wall or sleeve. This mechanism is a concern if moisture is present at the junction of the dissimilar materials.

The phenomenon of stress corrosion cracking (SCC) may result in fracture of the metal. It is defined as cracking under combined action of corrosion and tensile stresses. The stresses may be either applied (external) or residual (internal). The stress corrosion cracks themselves may be either transgranular or intergranular, depending upon the metal and the corrosive agent. As is normal in all cracking, the cracks are perpendicular to the tensile stress. The three principal factors necessary to initiate stress corrosion cracking are:

(1) Lacks details

- tensile stresses
- corrosive environment
- susceptible material.

The tensile stresses necessary to cause SCC must be at or near the material's yield point. This is facilitated when the material is substantially cold worked, contains residual stress from welding, or is subjected to significant applied loads. Corrosive environments which induce SCC are highly material-dependent. For austenitic stainless steels, such as SA-240 Type 304 or 308, halogens (i.e., chlorides), and acids promote SCC. With respect to material susceptibility, austenitic stainless steels are prone to SCC, particularly when sensitization is present (such as in heat affected zones) and at creviced geometries. Stress corrosion cracking is difficult to detect visually, because little or no macroscopic plastic deformation occurs as the crack propagates.

(1) Corrosion in inaccessible areas may be indicated by staining, or the collection of corrosion products, in areas adjacent to the affected area. Use of thickness measurements to identify localized corrosion will be less effective than for general corrosion, due to uncertainty in identifying affected areas.

4.6.2.2 Significance to License Renewal

The four areas unique to PWR free-standing steel containments, in terms of corrosion potential, are the outside surface of the containment (Figure 4-16), where condensation could form due to high humidity and the proximity of the ice condenser; the steel shell/concrete interface (Figure 4-14), where hydrophilic nonmetallic materials may collect moisture; the spherical shell exposed to environmental conditions (Figure 3-3); and the inside surface of expansion bellows in the piping penetration assembly (Figure 4-17), where condensation may form.

The outside surface and steel shell-to-concrete basemat interface is susceptible to corrosion promoted by the presence of moisture. However, much of this portion of the containment is readily accessible for inspection, where any deterioration can be detected, the consequences evaluated, and mitigative actions taken. Effective programs for monitoring this mechanism are described in Section 5.4.

The penetration bellows in metal containments are potential sites for galvanic corrosion. The bellows are stainless steel, while the rest of the vent lines or pipe sleeves are carbon steel. Based on their relative positions on the galvanic chart, the carbon steel would be susceptible to galvanic corrosion. The bellows, however, are protected by shields to prevent continued presence of moisture, thereby preventing the onset of galvanic corrosion.

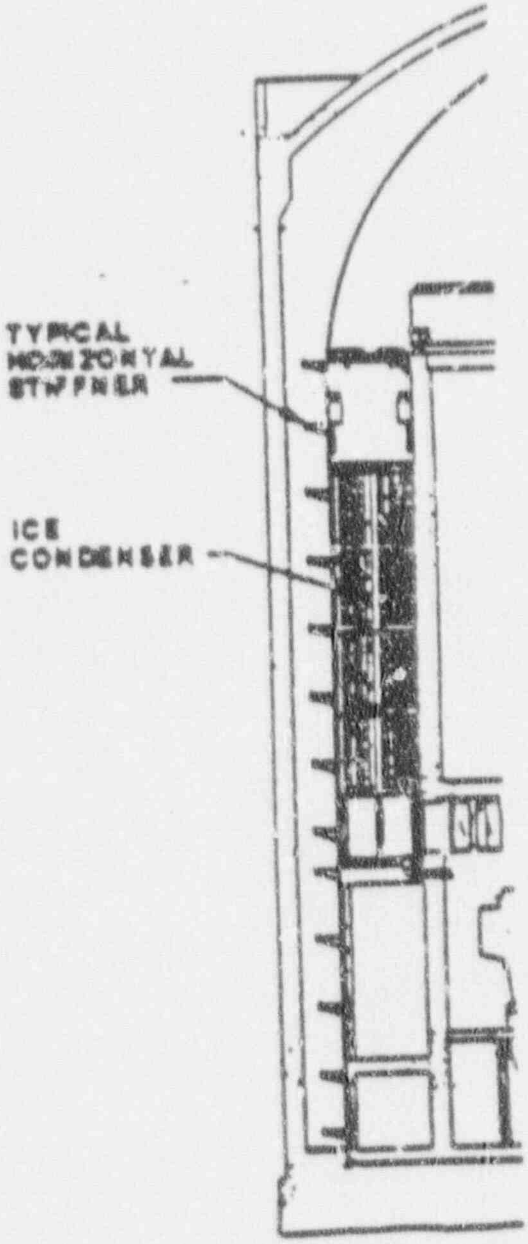

Figure 4-16
Free-Standing Steel Containment
Ice Condenser/Stiffener Area

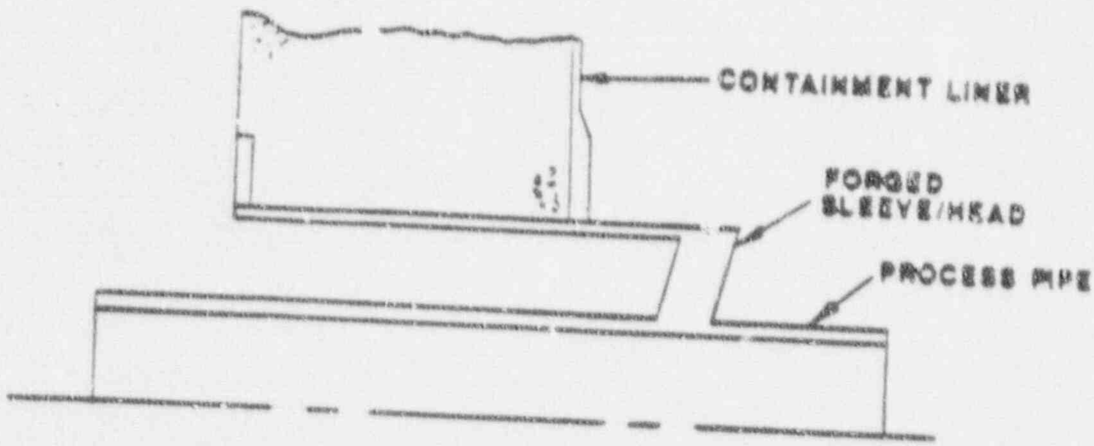

Figure 4-17(a)
Concrete Containment Hot Penetration

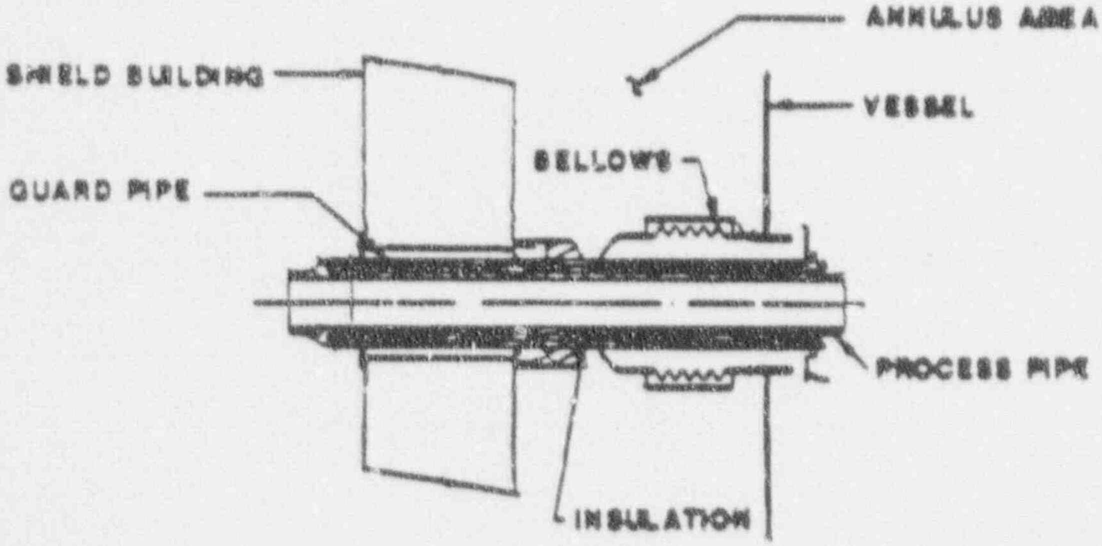

Figure 4-17(b)
Free-Standing Steel Containment
Hot Penetration

① "Safety function performance" is too limited

In addition, the carbon steel surfaces have a coating which provides additional galvanic corrosion protection.

The bellows are also susceptible to degradation by pitting and/or crevice corrosion. The expansion bellows' potential for corrosion may be exacerbated by the presence of foreign materials in the convolutions, such as grinding particles, sand blast grit, and so forth. The guard pipe shown in Figure 4-17(b) can prevent the direct visual examination of potential corrosion sites. Another complicating factor for some steel containments is the inability to visually inspect the containment shell interior near the ice condenser, such that spot ultrasonic examination may be needed to verify wall thickness.

Stress corrosion cracking degradation is only plausible for austenitic stainless steel components, which are limited to bellows assemblies in free standing steel containments. In addition to the tensile stresses and susceptible material, a corrosive environment must be present for SCC to occur. The bellows assembly is welded to a carbon steel containment penetration sleeve, center spool (where provided), and the penetration flued head. The attachment welds are dissimilar metal welds and are of a creviced geometry, thereby creating the potential for stress corrosion cracking. However, the attachment design minimizes any operational stresses from cycling or pressure testing. The environment is not corrosive with respect to chlorides or acids. Therefore, SCC will not cause significant degradation of the containment bellows during the license renewal term.

4.6.2.3 Summary of Corrosion

If dissimilar metals were not used in the construction of a PWR free-standing steel containment, then degradation by galvanic corrosion will not be significant.

①

Similarly, if austenitic stainless steels were not used, or if, as in the case of stainless steel bellows assemblies, the materials are protected from corrosive environments, then SCC will not affect the continued safety function performance of any PWR free standing steel containment component during the license renewal term, and requires no further evaluation. These conclusions apply to the following free-standing steel containment components:

Free-Standing Cylindrical and Spherical Steel Containments With Elliptical Bottoms

1. Containment shell interior surface
2. Containment shell exterior surface

(1) "Safety function" is too limited

Free-Standing Steel Containments With Flat Bottom and an Ice-Condenser

1. Dome shell interior surface
2. Dome shell exterior surface
3. Cylindrical shell interior surface
4. Cylindrical shell exterior surface

Common Components

1. Penetration bellows

A license renewal applicant intending to take credit for these conclusions is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that these assumptions and criteria are met.

Where free standing steel containment components are exposed to oxygen and moisture, general corrosion is potentially significant. Further, where dissimilar metals were used, then galvanic corrosion degradation is potentially significant. Effective programs to manage corrosion degradation of the following components are described in Section 5.4 of the report:

Free-Standing Cylindrical and Spherical Steel Containments With Elliptical Bottoms

1. Embedded shell region
2. Sand pocket region

Free-Standing Steel Containments With Flat Bottom and an Ice-Condenser

1. Embedded shell region

4.7 SUMMARY

(1)

In Subsections 4.1 through 4.6, the age-related degradation mechanisms that could affect PWR containment components were described and their potential significance to the continued safety function performance of these components throughout the license renewal term was evaluated. Specific assumptions and criteria were provided to enable the license renewal applicant to determine whether a particular age-related degradation mechanism/component combination requires further evaluation (is potentially significant), or whether a review of appropriate CLB documents/information would confirm that the degradation mechanism will not affect the component's capability to perform its intended safety function throughout the license renewal term.

(1)

Table 4-1 summarizes the results of these generic component/age-related degradation mechanism evaluations for each PWR containment type. Component/degradation mechanism combinations that were determined to be nonsignificant are identified by the Subsection in which the mechanism is evaluated. Those component/degradation mechanism combinations that were determined to require further evaluation are indicated with an "X", and are evaluated further in Section 5.

TABLE 4-1

PWR CONTAINMENT STRUCTURAL COMPONENTS/AGE-RELATED DEGRADATION MECHANISMS

CONTAINMENT COMPONENTS	DEGRADATION MECHANISM											
	FREEZE-THAW	LEACHING OF CALCIUM HYDROXIDE	AGGRESSIVE CHEMICALS	REACTION WITH AGGREGATES	CORROSION OF EMBEDDED STEEL	ELEVATED TEMPERATURE	IRRADIATION	CORROSION	PRESTRESSING LOSSES	FATIGUE	CONCRETE INTERACTION W/ ALUMINUM	SETTLEMENT	STRAIN AGING
CONCRETE CONTAINMENTS													
REINFORCED/PRESTRESSED													
o CONCRETE DOME	4.1.1	4.1.2	4.1.3	4.1.4	4.1.5	4.1.6	4.1.7	-	-	4.5.1	4.8.2	-	-
o CONCRETE CONTAINMENT WALL ABOVE GRADE	4.1.1	4.1.2	4.1.3	4.1.4	4.1.5	4.1.6	4.1.7	-	-	4.5.1	4.8.2	-	-
o CONCRETE CONTAINMENT WALL BELOW GRADE	4.1.1	4.1.2	X	4.1.4	X	4.1.6	4.1.7	-	-	4.5.1	4.8.2	-	-
o CONCRETE BASEMAT	4.1.1	4.1.2	X	4.1.4	X	4.1.6	4.1.7	-	-	4.5.1	4.8.2	X	-
o CONTAINMENT LINER INTERIOR SURFACE	-	-	-	-		4.4.2	4.4.3	4.4.1	-	-	-	-	-
o CONTAINMENT LINER ABOVE GRADE EXTERIOR SURFACE	-	-	-	-		4.4.2	4.4.3	4.4.1	-	-	-	-	-
o CONTAINMENT LINER BELOW GRADE EXTERIOR SURFACE	-	-	-	-	-	4.4.2	4.4.3	X	-	-	-	-	-
o BASEMAT LINER INTERIOR SURFACE	-	-	-	-	-	4.4.2	4.4.3	4.4.1	-	-	-	-	-
o BASEMAT LINER EXTERIOR SURFACE	-	-	-	-	-	4.4.2	4.4.3	X	-	-	-	-	-
o LINER ANCHORS ABOVE GRADE	-	-	-	-	-	4.4.2	4.4.3	4.4.1	-	-	-	-	-
o LINER ANCHORS BELOW GRADE	-	-	-	-	-	4.4.2	4.4.3	X	-	-	-	-	-
o DOME REINFORCING STEEL	-	-	-	-	-	4.2.2	4.2.3	4.2.1	-	4.5.1	-	-	-
o CONTAINMENT WALL REINFORCING STEEL ABOVE GRADE	-	-	-	-	-	4.2.2	4.2.3	4.2.1	-	4.5.1	-	-	-
o CONTAINMENT WALL REINFORCING STEEL BELOW GRADE	-	-	-	-	-	4.2.2	4.2.3	X	-	4.5.1	-	-	-
o BASEMAT REINFORCING STEEL	-	-	-	-	-	4.2.2	4.2.3	X	-	4.5.1	-	-	-
PRESTRESSED ONLY													
o PRESTRESSING TENDONS	-	-	-	-	-	4.3.2	4.3.3	X	X	-	-	-	-

LEGEND

- = NOT APPLICABLE

X.X.X = NOT A SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM (ARDM), SUBSECTION NUMBER WHERE ARDM IS EVALUATED

X = POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM

TABLE 4-1

PWR CONTAINMENT STRUCTURAL COMPONENTS/AGE-RELATED DEGRADATION MECHANISMS

CONTAINMENT COMPONENTS	DEGRADATION MECHANISM: FREEZE-THAW	LEACHING OF CALCIUM HYDROXIDE	AGGRESSIVE CHEMICALS	REACTION WITH AGGREGATES	CORROSION OF EMBEDDED STEEL	ELEVATED TEMPERATURE	IRRADIATION	CORROSION	PRESTRESSING LOSSES	FATIGUE	CONCRETE INTERACTION W/ ALUMINUM	SETTLEMENT	STRAIN AGING
FREE-STANDING CYLINDRICAL AND SPHERICAL STEEL CONTAINMENTS WITH ELLIPTICAL BOTTOM													
o CONTAINMENT SHELL INTERIOR SURFACE	-	-	-	-	-	4 4 2	4 4 3	4 8 2	-	4 5 1	-	-	4 8 1
o CONTAINMENT SHELL EXTERIOR SURFACE	-	-	-	-	-	4 4 2	4 4 3	4 8 2	-	4 5 1	-	-	4 8 1
o EMBEDDED SHELL REGION	-	-	-	-	-	4 4 2	4 4 3	X	-	-	-	-	4 8 1
o SAND POCKET REGION	-	-	-	-	-	4 4 2	4 4 3	X	-	-	-	-	4 8 1
FREE-STANDING STEEL CONTAINMENT WITH FLAT BOTTOM AND AN ICE-CONDENSER													
o DOME SHELL INTERIOR SURFACE	-	-	-	-	-	4 4 2	4 4 3	4 8 2	-	4 5 1	-	-	4 8 1
o DOME SHELL EXTERIOR SURFACE	-	-	-	-	-	4 4 2	4 4 3	4 8 2	-	4 5 1	-	-	4 8 1
o CYLINDRICAL SHELL INTERIOR SURFACE	-	-	-	-	-	4 4 2	4 4 3	4 8 2	-	4 5 1	-	-	4 8 1
o CYLINDRICAL SHELL EXTERIOR SURFACE	-	-	-	-	-	4 4 2	4 4 3	4 8 2	-	4 5 1	-	-	4 8 1
o EMBEDDED SHELL REGION	-	-	-	-	-	4 4 2	4 4 3	X	-	-	-	-	-
o BASEMAT LINER	-	-	-	-	-	4 4 2	4 4 3	X	-	-	-	-	-
o LINER ANCHORS	-	-	-	-	-	4 4 2	4 4 3	X	-	-	-	-	-
o CONCRETE BASEMAT	4 1 1	4 1 2	X	4 1 4	X	4 1 8	4 1 7	-	-	4 5 1	4 5 2	X	-
o BASEMAT REINFORCING STEEL	-	-	-	-	-	4 2 7	4 2 3	X	-	4 5 1	-	-	-
COMMON COMPONENTS													
o PENETRATION SLEEVES	-	-	-	-	-	4 4 2	4 4 3	4 4 1	-	X	-	-	4 8 1
o DISSIMILAR METAL WELDS	-	-	-	-	-	-	-	4 4 1	-	-	-	-	-
o PENETRATION BELLOWS	-	-	-	-	-	-	4 4 3	4 8 2	-	X	-	-	4 8 1
o PERSONNEL AIRLOCK	-	-	-	-	-	-	4 4 3	4 4 1	-	4 5 1	-	-	4 8 1
o EQUIPMENT HATCHES	-	-	-	-	-	-	4 4 3	4 4 1	-	4 5 1	-	-	4 8 1

LEGEND

- = NOT APPLICABLE

4 X X = NOT A SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM (ARDM); SUBSECTION NUMBER WHERE ARDM IS EVALUATED

X = POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM

① Date
② Editorial

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(1) Safety function" is too limited

SECTION 5
EFFECTIVE PROGRAMS FOR POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION

In the previous section of this report, the age-related degradation mechanisms that could affect PWR containment structures were described and their potential significance to the continued safety function performance (1) of these structures throughout the license renewal term was evaluated. Specific assumptions and criteria were provided to enable the license renewal applicant to determine whether a particular age-related degradation mechanism/component combination requires further evaluation (potentially significant), or whether a review of appropriate plant design, construction and operating history documents and records would confirm that the degradation mechanism does not affect the component's capability to perform its intended safety function (1) throughout the license renewal term.

This assessment reviewed age-related degradation mechanisms that have been observed in collective nuclear plant operating experience, relevant laboratory data, and related experience in other industries. The age-related degradation mechanisms considered were:

Concrete
- Freeze-thaw
- Leaching of calcium hydroxide
- Aggressive chemicals
- Reactions with aggregates
- Corrosion of embedded steel
- Elevated temperature
- Irradiation

Reinforcing Steel
- Corrosion
- Elevated temperature
- Irradiation

Prestressing System
- Corrosion
- Elevated temperature
- Irradiation
- Prestressing losses

Liner
- Corrosion
- Elevated temperature
- Irradiation

Miscellaneous Age-Related Degradation Mechanisms
- Fatigue
- Concrete interaction with aluminum
- Settlement

Free-Standing Steel Containment Degradation Mechanisms
- Strain aging
- Corrosion

(1) See comment #1, p. 1-2
(2) Add statements to address EP criteria
(3) CLB Maintenance

The significance of each of these age-related degradation mechanisms was evaluated for the components of each type of PWR containment structure.

In this section those combinations of age-related degradation mechanisms and components that require further evaluation are re-examined in terms of the capability of effective programs for maintenance, inservice inspection, surveillance, testing and analytical assessment to manage the effects of the potentially significant degradation. Combinations of mechanisms and components for which generic program elements effectively manage the age-related degradation are considered to be adequately addressed. (1) (1)

(2) A license renewal applicant intending to take credit for an effective program is responsible for the review/evaluation of appropriate plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements required to manage the effects of potentially significant age-related degradation, or ~~their justified equivalent,~~ continue being used at their plant. Plant-specific evaluations are required for any age-related degradation/component issues for which the license renewal applicant is unable to demonstrate that the generic stipulated program elements are committed for use, or the basis for the conclusions are applicable, at their plant. Recommendations for plant-specific aging management options for these issues are provided in Section 6 of this report, if required. Delete (1)

An age-related degradation mechanism is defined to be significant for a component if, when allowed to continue without an effective program, the capability of the component to perform its intended safety function throughout the license renewal term would be compromised. The potential significance of an age-related degradation mechanism was determined in Section 4 by examining the component design features (Section 3.1), the component design basis (Section 3.2), its operating history (Section 3.3), and its susceptibility to the degradation mechanism being considered. If it could be shown that the component is either not susceptible, or is susceptible to such a small degree that the component's safety function is maintained throughout the license renewal term, then the component/degradation mechanism combination is not significant. (3)

For PWR containment structures, the following age-related degradation mechanism/component combinations were determined in Section 4 to be potentially significant. These specific issues are examined further in this section in terms of the capability of effective programs to manage the effects of potentially significant age-related degradation.

(1)

CONCRETE DEGRADATION

Aggressive Chemicals

Reinforced and Prestressed Concrete Containments

1. Concrete containment wall below grade
2. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

Corrosion of Embedded Steel

Reinforced and Prestressed Concrete Containments

1. Concrete containment wall below grade
2. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

Corrosion of Reinforcing Steel

Reinforced and Prestressed Concrete Containments

1. Containment wall reinforcing steel below grade
2. Basemat reinforcing steel

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Basemat reinforcing steel

PRESTRESSING SYSTEM DEGRADATION (CONCRETE CONTAINMENTS)

Corrosion

1. Prestressing tendons and anchor heads

Prestressing Losses

1. Prestressing tendons

CONTAINMENT LINER/FREE-STANDING STEEL SHELL DEGRADATION

Corrosion

Reinforced and Prestressed Concrete Containments

1. Containment liner below grade exterior surface
2. Basemat liner exterior surface
3. Liner anchors below grade

Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottom

1. Embedded shell region
2. Sand pocket region

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Embedded shell region
2. Basemat liner
3. Liner anchors

MISCELLANEOUS AGE-RELATED DEGRADATION

Fatigue

Common Components

1. Penetration sleeves
2. Penetration bellows

Settlement

Reinforced and Prestressed Concrete Containments

1. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

5.1 CONCRETE DEGRADATION: AGGRESSIVE CHEMICALS AND CORROSION OF REBAR AND EMBEDDED STEEL

Aggressive chemical attack of below grade exterior PWR containment concrete by aggressive groundwater was identified in Section 4.1.3 as a source of potentially significant

(1) Lacks acceptance criteria, etc; EP Criteria requirements
(2) Reference to IWL not proper
(3) AIP not followed; Discuss the programs in their own right

age-related degradation. Under the same groundwater conditions, corrosion of embedded steel and reinforcing steel in exterior below grade concrete were also identified as potentially significant degradation mechanism/structural component combinations (Sections 4.1.5 and 4.2.1, respectively). Only below grade containment concrete exposed to fluctuating groundwater is affected. To mitigate groundwater attack, most plants utilize waterproof membranes underneath the basemat and outside the lower portions of the reactor building wall. Although this provides a measure of protection, the integrity of the membrane cannot easily be verified, and therefore, no credit is taken in these evaluations for the continued performance of the waterproof membrane.

The following concrete containment inspection and surveillance programs apply to accessible surfaces, which were determined in Section 4 not to be subject to potentially significant age-related degradation. Aging management options for inaccessible portions of concrete containments are provided in Section 6.1.

5.1.1 Inspection and Surveillance Programs

Accessible concrete surfaces of reinforced and prestressed concrete containments are subject to periodic examinations as a part of the Type A integrated leak rate tests performed under (1) Appendix J of 10 CFR 50 [1]. These same examination requirements are now codified in ASME Section XI, Subsection IWL (2) [2], as Examination Category L-A. The schedule for these visual examinations is given in IWL-2410. (3) With the exception of the initial five-year period of operation, examination is required every five years. (3) The visual examination is conducted in accordance with the provisions of IWL-2512, which refers to ACI 201.1 (4) (Guide for Making a Condition Survey of Concrete in Service) for the definition of relevant conditions that are indicative of damage or degradation. Such relevant conditions include: (1) excessive cracking of the concrete, such as that accompanying the accumulation of corrosion products; (2) spalling or related loss of concrete; (3) discoloration or staining of the concrete surface, such as that which might accompany migration of corrosion products.

Portions of the concrete that are covered by the liner, foundation material, or backfill, or that are otherwise obstructed by adjacent structures, components, parts, or appurtenances (2) are exempt from these examination requirements, in accordance with IWL-1220. All accessible surfaces, included coated areas, are subject to the examination requirements.

In addition to the general concrete surface, both Regulatory Guide 1.35 [3] and IWL-2524 (3) (3) require a visual (VT-1) examination of the concrete areas around tendon anchorage areas, extending outward a distance of two feet from the bearing plate. Relevant conditions include concrete cracking with widths greater than 0.01 inches.

(4) Need edition date

① Subject to EP criteria requirements of 54.21(a)(6)
② ASP not followed
③ Discuss administrative control requirement

5.1.2 Summary of Concrete Degradation Resulting from Aggressive Chemicals and Corrosion of Rebar and Embedded Steel

If concrete surfaces subject to aggressive chemical attack are accessible, and if they are periodically examined in accordance with the procedures that accompany Type A integrated leak rate tests, or in accordance with the requirements of ASME Section XI, Subsection IWL, Article IWL-2512, then potentially significant concrete degradation from aggressive chemical attack and corrosion of reinforcement or embedded steel is managed effectively. —(1) (2)

A license renewal applicant intending to take credit for these effective programs is responsible for the review/evaluation of plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements required to manage the effects of potentially significant aggressive chemical attack of concrete or corrosion of embedded steel or rebar, or their justified equivalent, ~~are committed for use at their plant.~~ (3)
Delete

If the concrete surfaces have not been periodically examined in accordance with these provisions and requirements, due to inaccessibility, then degradation of concrete surfaces and corrosion of reinforcement or embedded steel caused by aggressive chemical attack is significant, and further evaluation is required. Aging management options for ~~controlling~~ Mitigating the effects of aggressive chemical attack of inaccessible areas such as the below-grade, exterior PWR containment components listed below from sulfate-bearing soils or aggressive groundwater are given in Section 6.1 of this report. (4)

Reinforced and Prestressed Containments

1. Concrete containment wall below grade
2. Concrete basemat
3. Containment wall reinforcing steel below grade
4. Basemat reinforcing steel

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat
2. Basemat reinforcing steel

5.2 CORROSION OF PRESTRESSING TENDONS AND ANCHORS

Corrosion, including stress-corrosion cracking, of prestressing tendons and associated anchorage hardware was identified as a potentially significant age-related degradation mechanism for prestressed concrete containments in Section 4.3.1. Such degradation is considered during design, construction, and operation, through the use of tendon duct filler

④ Loose words; Be specific

① Reference to IWL not proper
② ASP not followed

material that protects the prestressing wire. Petroleum-based grease products help to prevent ingress of moisture that might attack the wire.

5.2.1 Inspection and Surveillance Programs

Regulatory Guide 1.35 and ASME Section XI, Subsection IWL contain provisions for managing the effects of such degradation. For example, Regulatory Position 5 from Regulatory Guide 1.35 and IWL-2523.2 require examination and testing of a previously-stressed tendon wire or strand from one tendon of each tendon group. The examinations cover the entire length of the sample to detect evidence of corrosion or other damage. In addition, tensile testing is required for these samples (one at each end of the wire and one at mid-length), in order to obtain yield strength, ultimate strength and elongation data to compare with the original properties. Acceptance criteria are listed in IWL-3221.2, including absence of physical damage, corrosion within plant-specific limits, and material properties meeting minimum specified values. ① ②

Further protection from corrosion is provided by examinations of the corrosion protection medium (e.g., grease) and any free water, as required by IWL-2525. Samples of the corrosion protection medium are removed from each end of the tendon examined, along with any free water in sufficient quantity, and tested for alkalinity, water content, aggressive ions, and pH. Limits are given in Table IWL-2525-1. ② ②

① Visual examination (VT-1) is required for the tendon anchorage hardware including bearing plates, anchor heads, wedges, buttonheads, shims, and the concrete extending outward a distance of two feet from the bearing plate. Reportable conditions include: (1) concrete cracks having widths greater than 0.01 inches; (2) corrosion, broken or protruding wires, missing buttonheads, broken strands, and cracks in tendon anchorage hardware; and (3) broken wires or strands, protruding wires or detached buttonheads following retensioning of tendons which have been detensioned. Of particular importance is any evidence of corrosion or cracks in the tendon anchorage hardware. Repair or replacement is required for conditions exceeding the acceptance criteria of IWL-3221.3. ②

Leakage or depletion of the corrosion protection medium is monitored through the provisions of Regulatory Guide 1.35 and ASME Section XI, Subsection IWL. Regulatory Guide 1.35 stipulates that "the amount of sheathing filler grease removed and replaced should be compared to assess grease leakage within the structure." Article IWL-2526 provides codification of this requirement, including documentation of any differences. ① ②

(1) Reference to IWL not proper

5.2.2 Summary of Corrosion of Prestressing Tendons and Anchor Heads

If tendon anchorage hardware has been examined in accordance with the provisions of Regulatory Guide 1.35 [3] or the requirements of the ASME Section XI, Subsection IWL, (1) specifically, visual examination of the tendon anchorage hardware, evaluation of the corrosion protection medium (e.g., grease), and identification and testing of any free water, then potentially significant degradation caused by corrosion of prestressing tendons and anchor heads is managed effectively.

A license renewal applicant intending to take credit for these effective programs is responsible for reviewing their related plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements required to manage the effects of potentially significant corrosion of prestressing tendons and anchor heads, or their justified equivalent, are in place.

Delete

5.3 PRESTRESSING LOSSES

Prestress loss was identified as a potentially significant age-related degradation mechanism for prestressed concrete containments in Section 4.3.4. Several of the factors responsible for prestress losses are time-dependent and have not been verified for the extended operating period. Therefore, the extended operating period could result in losses greater than those considered in the initial design.

5.3.1 Inspection of Prestressing Tendons

In accordance with Regulatory Guide 1.35 [3], lift-off tests of tendons are performed periodically. Sampling procedures assure that an adequate representation of tendon type will be selected, including those that are detensioned for inspection of possible damage and those used for lift-off testing. Prestressing force measurements are compared to the time-dependent prediction of prestressing loss described in Subsection 3.2.4 of this report. This prediction must be updated to reflect the proposed term of license renewal, and to provide the basis for lift-off test comparisons during the license renewal period. Reportable conditions are described in Regulatory Guide 1.35 when the measured force for the selected tendon in a group of tendons falls below 90% of the prescribed lower limit established by the prestress loss prediction. Additional testing of adjacent tendons may be necessary to establish reportable conditions when the prestressing losses range between 90% and 95% of the prescribed lower limit. Any reportable conditions identified during lift-off testing must be documented and transmitted to the NRC in accordance with the recommended reporting program of Regulatory Guide 1.16, "Reporting of Operating Information -

① AIP not followed

Appendix A, Technical Specification" [4]. The reportable conditions identified in Regulatory Guide 1.35 are similar to those outlined in Article IWL-3221. Plant-specific action may be necessary to determine the cause of pervasive losses of prestressing force exceeding the limits of Regulatory Guide 1.35 or IWL-3221, and to establish the necessary corrective actions. ① ①

5.3.2 Summary of Prestressing Losses

If prestressing losses are periodically monitored in accordance with the tendon lift-off test provisions of Regulatory Guide 1.35 or the tendon force measurement requirements of ASME Section XI, Subsection IWL, Article IWL-2522, and compared satisfactorily with predictions of prestressing loss valid for the license renewal term, then potentially significant degradation caused by prestressing loss is managed effectively. ①

A license renewal applicant intending to take credit for these effective programs is responsible for reviewing their related plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements required to manage the effects of potentially significant prestressing losses, ~~or their justified equivalent~~, are committed for use at their plant. Delete

5.4 LINER/FREE-STANDING STEEL SHELL CORROSION

Galvanic corrosion at dissimilar metal joints, stress corrosion cracking, and corrosion due to aggressive chemicals were identified as potentially significant age-related degradation mechanisms for specific areas of steel liners of concrete containments in Section 4.4.1, as well as portions of free-standing steel containments in Section 4.6.2. The areas of concern are as follows:

Reinforced and Prestressed Concrete Containments

1. Containment liner below grade exterior surface
2. Basemat liner exterior surface
3. Liner anchors below grade

Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottom

1. Embedded shell region
2. Sand pocket region

(1) AIP not followed; IWE cannot be referenced before being endorsed by the staff

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Embedded shell region
2. Basemat liner
3. Liner anchors

The following containment liner/free-standing steel shell examination and surveillance programs apply to accessible surfaces, which were determined in Section 4 not to be subject to potentially significant age-related degradation. Aging management options for inaccessible portions of containment liners and free-standing steel containment shells are provided in Section 6.2.

5.4.1 Examination and Surveillance

Accessible surfaces of free-standing steel containment shells and liner plates of concrete containments are subject to general periodic examinations in conjunction with Type A integrated leak rate tests performed under Appendix J of 10 CFR 50 [1]. These same examination requirements are now codified in ASME Section XI, Subsection IWE [2], as Examination Category E-P. Section V.A of Appendix J requires a general inservice examination of accessible interior and exterior surfaces of containment structures and components prior to any Type A test, in order to uncover any evidence of structural degradation that may affect either the containment integrity or excessive leakage. Examination Category E-P requires a visual (VT-3) examination of accessible surfaces of metal shell or liner plate prior to any Type A leak rate test. In addition, Examination Category E-D requires a visual (VT-3) examination of all seals and gaskets on airlocks, hatches, and other devices that are required to assure the satisfaction of containment leakage limits, and requires a similar examination of internal and external moisture barrier materials at concrete-to-metal interfaces intended to prevent intrusion of moisture against the pressure-retaining metal containment shell or liner. This includes caulking, flashing and other sealants. Examination Category E-F requires a visual (VT-3) examination of 50% of any dissimilar metal welds at each inspection interval, covering the weld metal and the base metal extending one wall thickness beyond the edge of the weld.

(1) (1) (1) (1)

Relevant conditions for the VT-3 examinations for coated areas include evidence of flaking, blistering, peeling, discoloration, and other signs of distress. Relevant conditions for uncoated areas include cracking, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents, and other signs of surface irregularities. The examination is to be performed without the removal of any paint or coatings. If any supplemental surface or volumetric examination is required, the paint or coating should be

① AIP not followed
② Reference to ASME inservice examination not proper. Need to justify the exemption for extended period of oper.

removed down to the base or weld metal surface, examined, and the areas repainted or recoated.

(1) IWE-1241 identifies those base metal areas that are suspect in terms of potential corrosive attack, such as those with either no or minimal corrosion allowance, or areas where there has been a loss or absence of protective coatings. Typical locations are those exposed to standing water, repeated wetting and drying, persistent leakage, and those with geometries that permit water accumulation, condensation, and microbiological attack. Such areas may include surfaces wetted during refueling, concrete-to-steel shell or liner interfaces, embedment zones, leak chase channels, drain areas, or sump liners.

(2) Inaccessible areas are exempt from these ASME Code inservice examination requirements.

5.4.2 Condition Monitoring

Supplementary examination methods for condition monitoring of free-standing steel shells or concrete containment metallic liners include the confirmation of minimum required wall thickness. Standard procedures are available, using ultrasonic (UT) measurement of suspect areas, provided that at least one surface is accessible for transmission and detection of the UT signal. "Practice for Measuring Thickness by Manual Pulse/Echo Ultrasonic Contact Method," ASTM E 797 (ASME SE-797, see ASME Code Section 5) [5] provides a (3) pulse-echo procedure for measuring the half transit time through the material, given a calibration for the sound velocity in the material.

Other monitoring techniques for free-standing steel shells or concrete containment metallic liners include observation of water leaking, dripping, or pooling in containment areas where such evidence is not normal or expected. Vapor condensation on interior or exterior surfaces, and the flow path of the condensate, also provide indications of potentially suspect areas.

5.4.3 Mitigation

Once identified, areas of free-standing steel containments or concrete containment metallic liners affected by local corrosion require a detailed technical evaluation to verify pressure boundary performance and structural integrity. Criteria for the evaluation are provided in Section III of the ASME Code [6]. Alternatively, repair or replacement of the deficient shell portion needs to be performed in accordance with the methods and procedures given in the ASME Section XI, Subsections IWE-4000 and 7000 [2]. (4)

③ call out edition date

④ Reference to IWE not proper

① AIP not followed
② what are they? call them out.
③ Loose words; be specific

5.4.4 Summary of Liner/Free-Standing Steel Shell Corrosion

If accessible areas of free-standing steel containment shells and liners of concrete containments are periodically examined and/or monitored in accordance with the requirements of ASME Section XI, Subsection IWE, Examination Categories E-D, E-F, and E-P, then potentially significant degradation caused by corrosion is managed effectively. ① If areas exempt from periodic inservice examination are monitored so as to maintain required wall thickness minimums, through a program of ultrasonic thickness measurements carried out in accordance with existing standards, then potentially significant degradation caused by corrosion is deemed to be managed effectively. ② ③ ④

A license renewal applicant intending to take credit for these effective programs is responsible for reviewing their related plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements required to manage the effects of potentially significant liner/free-standing steel shell corrosion, ~~or their justified equivalent,~~ are in place. Delete

Free-standing steel containment shell and concrete containment liner areas for which the program elements of Sections 5.4.1 and 5.4.2 are either not applicable, or are not in place, may be subject to potentially significant age-related corrosion degradation. Aging management options are given in Section 6.2 of this report for the following components:

Reinforced and Prestressed Concrete Containments

1. Containment liner below grade exterior surface
2. Basemat liner exterior surface
3. Liner anchors below grade

Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottom

1. Embedded shell region
2. Sand pocket region

Free-Standing Steel Containment with Flat Bottom and an ice-Condenser

1. Embedded shell region
2. Basemat liner
3. Liner anchors

④ EP criteria should be required

① Lacks details ; EP criteria must be satisfied
② Basemat is part of the containment

5.5 SETTLEMENT

In Section 4.5.3, it was concluded that differential settlement is a potentially significant age-related degradation mechanism for PWR containment structures. Structure settlement monitoring is initiated during the construction phase, and is only continued throughout plant operation if the plant site has soil conditions and/or groundwater conditions that indicate a likelihood of significant long-term settlement.

5.5.1 Settlement Monitoring

Structure settlement occurs within the first few years of load application, and settlement after the start of operation is generally minor. Settlement (absolute and differential) is detected through visual observation and, more accurately, with elevation survey data. A number of plants are currently monitoring settlement using widely accepted methods [7] that provide early indication of potentially significant settlement. When settlement approaches the design or acceptance criteria, re-evaluation of the containment is a standard practice. ①

5.5.2 Summary of Settlement

In Section 4.5.3 of this report it was determined that differential settlement can be a potentially significant age-related degradation mechanism for PWR containment structures. Established effective settlement monitoring methods provide adequate management of settlement for PWR containment structures, as follows:

Reinforced and Prestressed Concrete Containments

1. Concrete basemat

②

Free-Standing Steel Containment with a Flat Bottom and an Ice-Condenser

1. Concrete basemat

A license renewal applicant intending to take credit for these effective programs is responsible for reviewing their related plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements required to manage the effects of potentially significant settlement, or their justified equivalent, are committed for use at their plant.

Delete

(1) AIF not followed
(2) IWE cannot be referenced
(3) Loose words; be specific

5.6 FATIGUE OF HOT PENETRATIONS WITHOUT BELLOWS (CONCRETE CONTAINMENT) AND PENETRATION BELLOWS ASSEMBLIES (FREE-STANDING STEEL CONTAINMENT)

Fatigue damage of hot penetrations without bellows for PWR concrete containments and penetration bellows assemblies of PWR free-standing steel containments was identified as a potentially significant age-related degradation mechanism in Section 4.5.1. In both cases, the issue is thermal expansion cycling caused by startups and shutdowns. In general, these thermal expansion stresses do not cause fatigue damage, unless the localized temperatures at the penetrations are excessive.

5.6.1 Tests and Inspections

(1)

ASME Code Section XI, Subsection IWE provides examination requirements for class MC pressure retaining components and their integral attachments, and for metallic shell and penetration liners of class CC pressure retaining components and their integral attachments. Examination Category E-B requires a visual (VT-1) examination of containment penetrations welds, including any flued head and bellows seal circumferential welds joined to the penetration. The examination requirement is limited to those welds subject to cyclic loads and thermal stress during normal plant operation, e.g., hot penetrations. It should be pointed out, however, that ASME Nuclear Code Case N-198-1 [8] provides for an exemption from examination for the Class 1 & 2 piping weldments of the penetrations. This exemption does not apply to the penetration sleeve or the bellows. (1)

(2)

As an alternative to the Subsection IWE inspection, the penetration can be analyzed to assure that the fatigue usage factor is less than unity for the license renewal term in accordance with procedures of NE-3221.5 [6]. If this alternative is chosen, the thermal loading cycles that represent the fatigue design transients must be shown to envelope the actual operating transients. In general, the expected fatigue usage factor for 80 years of operation would be less than unity. Monitoring of penetration temperatures may be necessary to establish the magnitude and frequency of the operating transients.

(3)

5.6.2 Summary of Fatigue

In Section 4.5.1 of this report it was determined that fatigue is a potentially significant age-related degradation mechanism for PWR containment components that are projected to have relatively high fatigue usage factors at some point during the license renewal term. Components identified as having relatively high projected fatigue usage factors were hot

(1) Follow Branch Technical Position on fatigue
(2) Loose words

penetrations without bellows assemblies (concrete containments) and penetration bellows assemblies (free-standing steel containments).

(1) ~~Effective programs of fatigue re-analysis~~, conducted in accordance with the ASME Code Section III, Subsection NB, are able to show that the fatigue usage factors for these components can be maintained below unity throughout the license renewal term. Effective programs of inservice inspection, conducted in accordance with ASME Code Section XI, Subsection IWE, are able to ensure that component integrity is maintained throughout the license renewal term in the presence of known or suspected fatigue damage, including the continued service of a component with an otherwise rejectable flaw, as justified by an engineering evaluation. (2)

A license renewal applicant intending to take credit for these effective programs is responsible for reviewing their related plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements required to manage the effects of potentially significant fatigue damage accumulation or fatigue crack growth, or their justified equivalent, are committed for use at their plant.

Delete

5.7 SUMMARY OF SECTION 5 CONCLUSIONS

Section 5 demonstrates that several potentially significant age-related degradation mechanisms that may affect PWR containments during the license renewal term can be effectively managed through currently accepted methods.

Potentially Significant Age-Related Degradation Mechanisms
Managed by Effective Programs

Prestressing System
- Corrosion
- Prestressing losses

Miscellaneous Age-Related Degradation Mechanisms
- Settlement
- Fatigue

On the basis of these effective programs, these issues need not be evaluated further for license renewal, beyond assurance by the applicant that no plant-specific features exist which would preclude the applicant from verifying these conclusions, and that the programs elements required to manage the effects of potentially significant degradation, or their justified equivalent, are committed for use at their plant.

Delete

(1) Describe EP criteria requirements per 54.21(a)(6)

Table 5-1 summarizes the results of Sections 4 and 5 of this report. As indicated in the table by an "X", degradation of regions of the steel liner of reinforced concrete containments, the concrete, or free-standing steel containment shells that are inaccessible for inspection could not be shown to be within acceptable limits on the basis of current tests, inspections, and analytical techniques. Additional measures are necessary to confirm the continued performance of these containment components during the license renewal period. Each licensee must assure that an effective program for addressing these issues is committed for use on a plant-specific basis during the license renewal term. (1) Suggested guidelines for the resolution of the remaining age-related degradation mechanism/structural component combinations are presented in Section 6.

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TABLE 5-1

PWR CONTAINMENT STRUCTURAL COMPONENTS/AGE-RELATED DEGRADATION MECHANISMS

CONTAINMENT COMPONENTS	DEGRADATION MECHANISM												
	FREEZE-THAW	LEACHING OF CALCIUM HYDROXIDE	AGGRESSIVE CHEMICALS	REACTION WITH AGGREGATES	CORROSION OF EMBEDDED STEEL	ELEVATED TEMPERATURE	IRRADIATION	CORROSION	PRESTRESSING LOSSES	FATIGUE	CONCRETE INTERACTION W/ ALUMINUM	SETTLEMENT	STRAIN AGING
CONCRETE CONTAINMENTS													
REINFORCED/PRESTRESSED													
o CONCRETE DOME	4.1.1	4.1.2	4.1.3	4.1.4	4.1.5	4.1.6	4.1.7	–	–	4.5.1	4.5.2	–	–
o CONCRETE CONTAINMENT WALL ABOVE GRADE	4.1.1	4.1.2	4.1.3	4.1.4	4.1.5	4.1.6	4.1.7	–	–	4.5.1	4.5.2	–	–
o CONCRETE CONTAINMENT WALL BELOW GRADE	4.1.1	4.1.2	X	4.1.4	X	4.1.6	4.1.7	–	–	4.5.1	4.5.2	–	–
o CONCRETE BASEMAT	4.1.1	4.1.2	X	4.1.4	X	4.1.6	4.1.7	–	–	4.5.1	4.5.2	5.5	–
o CONTAINMENT LINER INTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.4.1	–	–	–	–	–
o CONTAINMENT LINER ABOVE GRADE EXTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.4.1	–	–	–	–	–
o CONTAINMENT LINER BELOW GRADE EXTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	X	–	–	–	–	–
o BASEMAT LINER INTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.4.1	–	–	–	–	–
o BASEMAT LINER EXTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	X	–	–	–	–	–
o LINER ANCHORS ABOVE GRADE	–	–	–	–	–	4.4.2	4.4.3	4.4.1	–	–	–	–	–
o LINER ANCHORS BELOW GRADE	–	–	–	–	–	4.4.2	4.4.3	X	–	–	–	–	–
o DOME REINFORCING STEEL	–	–	–	–	–	4.2.2	4.2.3	4.2.1	–	4.5.1	–	–	–
o CONTAINMENT WALL REINFORCING STEEL ABOVE GRADE	–	–	–	–	–	4.2.2	4.2.3	4.2.1	–	4.5.1	–	–	–
o CONTAINMENT WALL REINFORCING STEEL BELOW GRADE	–	–	–	–	–	4.2.2	4.2.3	X	–	4.5.1	–	–	–
o BASEMAT REINFORCING STEEL	–	–	–	–	–	4.2.2	4.2.3	X	–	4.5.1	–	–	–
PRESTRESSED ONLY													
o PRESTRESSING TENDONS	–	–	–	–	–	4.3.2	4.3.3	5.2	5.3	–	–	–	–

LEGEND

– = NOT APPLICABLE

4.X.X = NOT A SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM (ARDM), SUBSECTION NUMBER WHERE ARDM IS EVALUATED

5.X = POTENTIALLY SIGNIFICANT ARDM CONTROLLED BY AN EFFECTIVE PROGRAM, SUBSECTION IN WHICH PROGRAM IS DESCRIBED

X = POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM, AGING MANAGEMENT OPTIONS PROVIDED IN SECTION 6

TABLE 5-1

PWR CONTAINMENT STRUCTURAL COMPONENTS/AGE-RELATED DEGRADATION MECHANISMS

CONTAINMENT COMPONENTS	DEGRADATION MECHANISM												
	FREEZE-THAW	LEACHING OF CALCIUM HYDROXIDE	AGGRESSIVE CHEMICALS	REACTION WITH AGGREGATES	CORROSION OF EMBEDDED STEEL	ELEVATED TEMPERATURE	IRRADIATION	CORROSION	PRESTRESSING LOSSES	FATIGUE	CONCRETE INTERACTION W/ ALUMINUM	SETTLEMENT	STRAIN AGING
FREE-STANDING CYLINDRICAL AND SPHERICAL STEEL CONTAINMENTS WITH ELLIPTICAL BOTTOM													
o CONTAINMENT SHELL INTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.8.2	–	4.5.1	–	–	4.8.1
o CONTAINMENT SHELL EXTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.8.2	–	4.5.1	–	–	4.8.1
o EMBEDDED SHELL REGION	–	–	–	–	–	4.4.2	4.4.3	X	–	–	–	–	4.8.1
o SAND POCKET REGION	–	–	–	–	–	4.4.2	4.4.3	X	–	–	–	–	4.8.1
FREE-STANDING STEEL CONTAINMENT WITH FLAT BOTTOM AND AN ICE CONDENSER													
o DOME SHELL INTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.8.2	–	4.5.1	–	–	4.8.1
o DOME SHELL EXTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.8.2	–	4.5.1	–	–	4.8.1
o CYLINDRICAL SHELL INTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.8.2	–	4.5.1	–	–	4.8.1
o CYLINDRICAL SHELL EXTERIOR SURFACE	–	–	–	–	–	4.4.2	4.4.3	4.8.2	–	4.5.1	–	–	4.8.1
o EMBEDDED SHELL REGION	–	–	–	–	–	4.4.2	4.4.3	X	–	–	–	–	–
o BASEMAT LINER	–	–	–	–	–	4.4.2	4.4.3	X	–	–	–	–	–
o LINER ANCHORS	–	–	–	–	–	4.4.2	4.4.3	X	–	–	–	–	–
o CONCRETE BASEMAT	4.1.1	4.1.2	X	4.1.4	X	4.1.6	4.1.7	–	–	4.5.1	4.5.2	5.5	–
o BASEMAT REINFORCING STEEL	–	–	–	–	–	4.2.2	4.2.3	X	–	4.5.1	–	–	–
COMMON COMPONENTS													
o PENETRATION SLEEVES	–	–	–	–	–	4.4.2	4.4.3	4.4.1	–	5.6	–	–	4.8.1
o DISSIMILAR METAL WELDS	–	–	–	–	–	–	–	4.4.1	–	–	–	–	–
o PENETRATION BELLOWS	–	–	–	–	–	–	4.4.3	4.8.2	–	5.8	–	–	4.8.1
o PERSONNEL AIRLOCK	–	–	–	–	–	–	4.4.3	4.4.1	–	4.5.1	–	–	4.8.1
o EQUIPMENT HATCHES	–	–	–	–	–	–	4.4.3	4.4.1	–	4.5.1	–	–	4.8.1

LEGEND:

– = NOT APPLICABLE

4.X.X = NOT A SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM (ARDM), SUBSECTION NUMBER WHERE ARDM IS EVALUATED

5.X = POTENTIALLY SIGNIFICANT ARDM CONTROLLED BY AN EFFECTIVE PROGRAM, SUBSECTION IN WHICH PROGRAM IS DESCRIBED

X = POTENTIALLY SIGNIFICANT AGE-RELATED DEGRADATION MECHANISM, AGING MANAGEMENT OPTIONS PROVIDED IN SECTION 8

5.8 REFERENCES

1. 10CFR Part 50, Domestic Licensing of Production and Utilization Facilities, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," Office of the Federal Register National Archives and Records Administration, U.S. Government Printing Office, Washington, D.C.

2. Code for Concrete Reactor Vessels and Containments, ASME Boiler and Pressure Vessel Code, Section XI, American Society of Mechanical Engineers, New York, NY.

3. Regulatory Guide 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," Revision 3, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C., July 11, 1990.

4. Regulatory Guide 1.16, "Reporting of Operating Information - Appendix A, Technical Specification," Rev. 4, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C., August, 1975.

5. Practice for Measuring Thickness by Manual Ultrasonic Pulse-Echo Contact Method, ASTM E 797-81, American Society for Testing and Materials, Philadelphia, Penn., 1981.

6. Code for Concrete Reactor Vessels and Containments, ASME Boiler and Pressure Vessel Code, Section III, Subsection NE, Class MC Components, American Society of Mechanical Engineers, New York, NY.

7. "Classification Standards of Accuracy, and General Specification of Geodetic Control Surveys," Federal Geodetic Survey, National Oceanic and Atmospheric Administration, Washington, D.C., May 1974.

8. Code Cases, Nuclear Components, ASME Boiler and Pressure Vessel Code, American Society of Mechanical Engineers, New York, NY.

(1) "safety function" is too limited

SECTION 6
MANAGEMENT OPTIONS FOR SIGNIFICANT AGE-RELATED DEGRADATION

In the previous section of this report, effective programs of inservice inspection, testing, surveillance, and analytical assessment were evaluated with respect to their capability to manage the effects of potentially significant age-related degradation. If the elements of an effective program are determined to be capable of managing the effects of potentially significant age-related degradation for a particular component so that its intended safety function is not compromised during the license renewal term, then that combination of component and degradation mechanism was deemed to be adequately addressed. (1)

A license renewal applicant intending to take credit for the effective program is responsible for the review/evaluation of their related plant-specific features, including appropriate CLB documents/information, in order to assure that the program elements used to manage the effects of potentially significant age-related degradation, or their equivalent, are committed for use at their plant. Delete

Age-related degradation/component issues for which effective program elements cannot be shown to adequately manage the effects of potentially significant age-related degradation, or for which license renewal applicants are unable to demonstrate that the stipulated program elements are in place, require plant-specific evaluation.

For PWR containment structures, the following age-related degradation mechanism/component combinations were determined in Section 4 to be potentially significant, and were beyond the scope of the effective programs examined in Section 5. These mechanism/component combinations must be evaluated by license renewal applicants on a plant-specific basis. Options for aging management programs for these potentially significant age-related degradation mechanism/component combinations are provided in this section.

CONCRETE DEGRADATION

Aggressive Chemicals

Reinforced and Prestressed Concrete Containments

1. Concrete containment wall below grade
2. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

Corrosion of Embedded Steel

Reinforced and Prestressed Concrete Containments

1. Concrete containment wall below grade
2. Concrete basemat

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Concrete basemat

Corrosion of Reinforcing Steel

Reinforced and Prestressed Concrete Containments

1. Containment wall below grade reinforcing steel
2. Basemat reinforcing steel

Free-Standing Steel Containment with Flat Bottom and an Ice-Condenser

1. Basemat reinforcing steel

CONTAINMENT LINER/FREE-STANDING STEEL SHELL DEGRADATION

Corrosion

Reinforced and Prestressed Concrete Containments

1. Containment liner below grade exterior surface
2. Basemat liner exterior surface
3. Liner anchors below grade

Free-Standing Cylindrical and Spherical Steel Containments with Elliptical Bottom

1. Embedded shell region
2. Sand pocket region

Free-Standing Steel Containments with Flat Bottom and an Ice-Condenser

1. Embedded shell region
2. Basemat liner
3. Liner anchors

(1) Change "or" to "and"
(2) Need specifics; ASP not followed

Applicants for license renewal must develop a program to manage these remaining degradation issues during the license renewal term. This section provides options which may be used by a licensee in developing a plant-specific aging degradation management of a PWR containment structure. These options take the form of preventive and mitigative measures, or improved inspection, testing and analytical assessment procedures. These options are not to be construed as mandatory for individual plant implementation, but are to be considered acceptable alternatives.

Section 6.1 contains strategies for the management of concrete containments including below-grade or inaccessible concrete attack by fluctuating aggressive groundwater. The aging degradation management of embedded and reinforcing steel is also addressed. Section 6.2 outlines options for the management of inaccessible regions of steel liners or free-standing steel containment shells. Alternatives are described which can be used to identify corrosion of these steel components.

6.1 Management of Below Grade or Inaccessible Reinforced Concrete Structures (1)

The general management approach for PWR concrete containment structures consists of a detailed evaluation of those structural components that can be adversely affected by aggressive groundwater (pH < 5.5 or chloride or sulfate concentration greater than 500 or 1500 ppm, respectively) and corrosion of embedded steel or reinforcing steel.

Techniques for detecting and evaluating aging in concrete components are described in NUREG/CR-4652 [1] "Concrete Degradation Monitoring and Evaluation," in NUREG/CP-100 [2] discusses: (1) concrete degradation mechanisms and consequences, (2) available techniques to monitor concrete structures, and (3) proposed acceptance criteria. Inspections of concrete containments can be performed in accordance with the applicable sections of ACI 201.1R-68 [3], ACI 224.R-80 [4] and ACI 224.1R-84[5]. (2)

A phased evaluation program as outlined in the following three steps is recommended for evaluation of containment structures.

Phase I Evaluation of Groundwater

The groundwater chemistry should be evaluated in cases where the groundwater is in direct contact with the foundation and exterior walls. For most plants, groundwater quality will be available from the environmental monitoring program. Where groundwater samples are needed, they should be obtained from wells which are representative of the quality of water around the containment building.

(1) Require excavation?

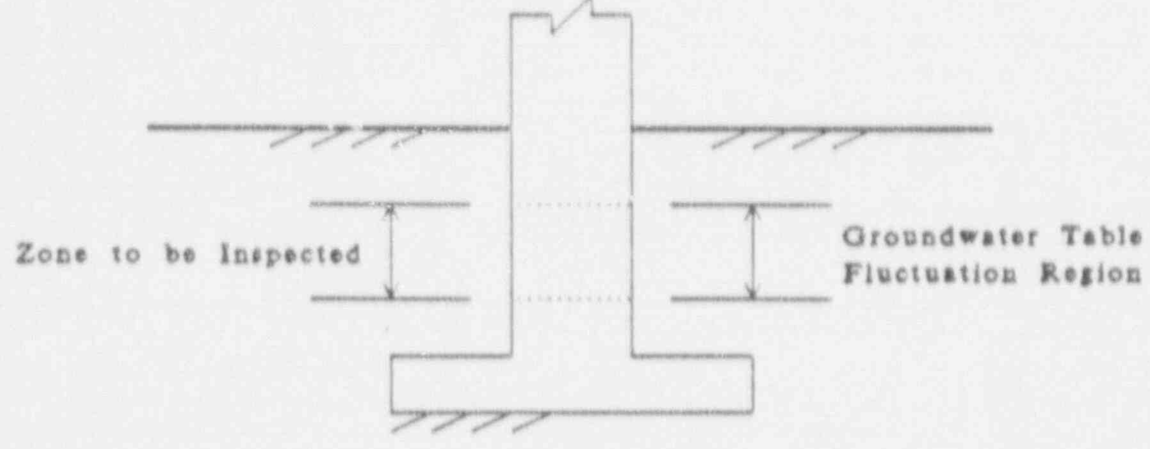
If the pH of the groundwater is > 5.5 and the chloride and sulfate concentration is less than 500 and 1500 ppm, respectively, the groundwater will not adversely affect the concrete, and no further action is required.

Phase II Inspection and Testing

Inspection and testing is performed in cases where the groundwater that is in direct contact with the containment concrete foundation and exterior walls below grade was determined in Phase I to have a pH < 5.5 or a chloride or sulfate concentration greater than 500 or 1500 ppm, respectively.

Inspection and Testing of Concrete

In cases where the chloride and/or sulfate concentrations exceed the above limits, it is suggested that the applicant visually inspect the accessible areas of exterior walls below the groundwater table and evaluate the condition of the concrete in accordance with the provisions of ACI 207.3R-79, "Practices for Evaluation of Concrete in Existing Massive Structures for Service Conditions" [6]. The critical zone of an exterior sub-grade wall to be inspected is in the zone of fluctuating groundwater table, as depicted in the following configuration: (1)



Accepted methods to obtain the necessary information and accepted methods to implement corrective actions are detailed in ACI 207.3R-79.

(1) Change "or" to "and."
(2) How could visual inspection be performed for inaccessible areas?

Assessment

Where effective repairs cannot be implemented or when it may be more economically attractive due to the extent of structural repair required, groundwater management as outlined below should be considered as an alternative.

Phase III Management of Groundwater

Management of groundwater may be required in cases where (1) the Phase I groundwater evaluation indicates a pH < 5.5 or chloride or sulfate concentrations are greater than 500 or 1500 ppm, respectively, and (2) the testing and evaluation performed in Phase II indicate significant concrete degradation or embedded steel or reinforcing steel corrosion.

The following options should be considered:

1. Lowering of groundwater table by subsurface drainage consisting of a piping system which drains down the water table using pumps.

2. Installation of a barrier system to minimize the aggressive groundwater in contact with concrete structure of the containment building.

(1)

6.2 Management of Below Grade or Inaccessible Steel Structures

The general management approach for steel liners or free standing steel containment shells consists of a detailed evaluation to identify corrosion that, if left undetected, could lead to structure deterioration.

Potentially significant degradation of structure steel components in accessible areas is bounded by effective programs as discussed in Section 5. The susceptible locations for steel corrosion which cannot be detected by routine inspections include inaccessible areas such as below grade portions of steel liners or containment shells and those portions of steel liners where accessibility is limited.

The inspection of structural steel components is performed using a phased inspection program similar to that described for concrete structures in Section 6.1.

(2) Phase I includes a visual inspection of representative portions of the steel components in susceptible locations to identify structural steel corrosion, cracking, or other visible indications of degradation. If the protective medium is sound, then no further evaluation is required. If the protective concrete and joints, the protective sealants and caulking, or

other water proofing barriers are all sound, the environment necessary for corrosion to occur will not exist. If no significant degradation is found in Phase I, Phase II and III testing and evaluation is not required.

Phase II is only implemented for structural steel components when the results of Phase I indicate significant deterioration. Phase II techniques may include radiographic testing, magnetic particle testing, and/or liquid penetrant testing in accordance with ASTM E94-77 [7], ASTM E709-80 [8], and ASTM E165-80 [9], respectively, for detection of cracking in welds, heat-affected zones, and the base metal. The extent of corrosion can be determined using impressions. Phase III testing is only implemented in cases where the Phase II inspection and testing does not provide conclusive results that enable the licensee to control the effects of the identified degradation.

Phase III of the aging degradation management program for steel liners or free-standing containment structures consists of destructive testing, including cutting samples from the structural steel for chemical analysis; metallurgical evaluation; and embrittlement, stress corrosion, and tensile testing in order to determine the extent of embrittlement, cracking, and corrosion.

Any repair that may be initiated as a result of Phase II or III inspection and testing should be performed in accordance with written procedures.

6.3 References

1. Naus, D. J., Concrete Com[illegible]ent Aging and Its Significance Relative to Life Extension of Nuclear Power Plants, NUREG CR-4652, ORNL/TM-10059, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 1986.

2. Prasad, N., et al., "Concrete Degradation Monitoring and Evaluation," Proceedings of the International Nuclear Power Plant Aging Symposium, NUREG/CP-0100, August 1988.

3. Guide for Making a Condition Survey of Concrete in Service, ACI 201.1R-68, American Concrete Institute, Detroit, Michigan, Revised 1984.

4. Control of Cracking in Concrete Structures, ACI 224.R-80, American Concrete Institute, Detroit, Michigan.

5. Causes, Evaluation and Repair of Cracks in Concrete Structures, ACI 224.1R-84, American Concrete Institute, Detroit, Michigan.

6. Practices for Evaluation of Concrete in Existing Massive Structures for Service Condition, ACI 207.3R-79, American Concrete Institute, Detroit, Michigan, Revised 1985.

7. Recommended Practice for Radiographic Testing, ASTM E94-77, American Society for Testing and Materials, Philadelphia, Penn.

8. Magnetic Particle Examination, ASTM E709-80, American Society for Testing and Materials, Philadelphia, Penn.

9. Liquid Penetrant Inspection Method, ASTM E165-80, American Society for Testing and Materials, Philadelphia, Penn.