WESTINGHOUSE OWNERS GROUP
WOG
ESTABLISHED-1979

OG-01-035
May 23, 2001

WCAP-14756-A
Project Number 686

To: Document Control Desk
US Nuclear Regulatory Commission
Washington, DC 20555-0001

Attention Mr. Christopher Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Program
Office of Nuclear Reactor Regulation
One White Flint North
Mail Stop 1G 15
11555 Rockville Pike
Rockville, MD 20852-2738

Subject: Westinghouse Owners Group
Transmittal of Approved Version of WOG Aging Management Report, "Aging Management Evaluation for Pressurized Water Reactor Containment Structure," WCAP-14756-A (MUHP-6110)

References

1. NRC letter, April 13, 2001, Grimes to Newton, "Acceptance for Referencing of Generic License Renewal Program Topical Report Entitled, 'Aging Management Evaluation for the Pressurized Water Reactor Containment Structure,' WCAP-14756, Revision 0, December 196

2. OG-00-094, Request for Preparation of Final Safety Evaluation for WOG Generic Technical Reports on Aging Management, September 22, 2000, to C.L. Grimes, NRC

This letter transmits 23 copies of the approved, non-proprietary, WCAP referenced above. This report evaluates the aging of PWR Containment Structures components as described in the report to ensure that their intended functions will be maintained during an extended period of operation. This WCAP was prepared in the manner described in WOG letter OG-00-094 (Reference 2). Previous WCAPs generated in accordance with Reference 2 did not go though a revision process.

However, references to Westinghouse Electric Corporation were changed to Westinghouse Electric Company throughout the original report text in order to reflect the current name of the company.

The report transmitted herewith bears a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in this report which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.790 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room

Rec'd at NRC/DLD from NEM on 06/14/01
D043

and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

If you require further information, please contact Roger Newton, Nuclear Management Company, at (920) 755-6522 or Charlie Meyer, Westinghouse, at (412) 374-5027.

Very truly yours,

Robert H Bryan

Robert H. Bryan, Chairman
Westinghouse Owners Group

Enclosures

cc: WOG Steering Committee (1L)
WOG Primary Representatives (1L)
WOG License Renewal Working Group (1L)
C. Grimes, USNRC (1L, 23R)
S. Bloom, USNRC (1L)
R. Etling, W- ECE 5-43 (1L)
G. Bischoff, CEOG Windsor (1L)
H. A. Sepp, W- ECE 4-15 (1L)
A. P. Drake, W- ECE 5-16 (1L)

Westinghouse Non-Proprietary Class 3

WCAP-14756-A

Aging Management Evaluation for Pressurized Water Reactor Containment Structure

Westinghouse Electric Company

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-14756-A

AGING MANAGEMENT EVALUATION
FOR
PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURE

May 2001

Westinghouse:
R. F. Condrac
W. S. LaPay

Gilbert Commonwealth:
Dale Krause
Eric Blocher

Funded by:

Westinghouse Owners Group (WOG)
Life Cycle Management/License Renewal (LCM/LR) Program
and
Electric Power Research Institute

Approved: ______________________________
R. Llovet, Westinghouse Program Manager
WOG LCM/LR Program

Prepared by Westinghouse Electric Company for use by Members of the Westinghouse Owners Group. Work performed in Shop Order MUHP-6120 under direction of the WOG LCM/LR Program Core Group

WESTINGHOUSE ELECTRIC COMPANY
Energy Systems Business Unit
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355

o:\2656-A:doc:1b-043001

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

April 13, 2001

Mr. Roger A. Newton, Chairman
Westinghouse Owners Group
Wisconsin Electric Power Company
231 West Michigan
Milwaukee, Wisconsin 53201

SUBJECT: ACCEPTANCE FOR REFERENCING OF GENERIC LICENSE RENEWAL PROGRAM TOPICAL REPORT ENTITLED, "AGING MANAGEMENT EVALUATION FOR THE PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURE" WCAP-14756, REVISION 0, DECEMBER 1996

Dear Mr. Newton:

The staff of the U.S. Nuclear Regulatory Commission has reviewed the topical report entitled, "Aging Management Evaluation for the Pressurized Water Reactor Containment Structure," WCAP-14756, Revision 0 which the Westinghouse Owners Group (WOG) submitted in December 1996, as part of the Generic License Renewal Program (GLRP). The resultant final safety evaluation report (FSER) is transmitted to you as an enclosure to this letter. We prepared this FSER after reviewing the technical evaluation report (TER) developed under contract by Brookhaven National Laboratory. We concur with the findings of the TER.

In a letter dated March 1, 2001, the WOG requested the staff to revise the Renewal Applicant Action Item 28 as delineated in the draft safety evaluation of January 31, 2001, and proceed with the final safety evaluation report on WCAP-14756. Accordingly, the staff revised the FSER.

As indicated in the FSER, the staff found the topical report acceptable for GLRP members' plants to reference in a license renewal application to the extent specified and under the limitations delineated in the staff FSER and the associated topical report. The limitations include committing to the accepted aging management programs defined in the topical report, and completing the renewal applicant action items described in Section 4.0 of the FSER. An applicant referencing the topical report and meeting these limitations will provide sufficient information for the staff to make a finding that there is reasonable assurance that the applicant will adequately manage the effects of aging so that the intended functions of the containment covered by the scope of the report will be maintained consistent with the current licensing basis during the period of extended operation.

The staff does not intend to repeat its review of the matters described in the report and found acceptable in the FSER when the report appears as reference in a license renewal application, except to ensure that the material presented applies to the specified plant.

In accordance with the procedures established in NUREG-0390, "Topical Report Review Status," the staff requests that the WOG publish the accepted version of WCAP-14756 within three months after receiving this letter. In addition, the published version will incorporate this letter and the enclosed FSER between the title page and the abstract.

To identify the version of the published topical report that was accepted by the staff, the staff requests the WOG include "-A" following the topical report number (e.g., WCAP-14756-A).

Sincerely,

Christopher I. Grimes, Chief
License Renewal and Standardization Branch
Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation

Project No. 686

Enclosure: Final Safety Evaluation Report

cc w/encl: See next page

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
CONCERNING
WESTINGHOUSE OWNERS GROUP
GENERIC TECHNICAL REPORT WCAP-14756
REVISION 0, DECEMBER 1996
"AGING MANAGEMENT EVALUATION FOR THE
PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURE"

1.0 INTRODUCTION

Pursuant to Section 50.51 of Title 10 of the *Code of Federal Regulations* (10 CFR 50.51), the U.S. Nuclear Regulatory Commission (NRC) issues licenses to operate nuclear power plants for a fixed period of time not to exceed 40 years; however, the NRC may renew these licenses for a fixed period of time not to exceed 20 years beyond expiration of the current operating license. The Commission's regulations in 10 CFR Part 54, published May 8, 1995, set forth the requirements for the renewal of operating licenses for commercial nuclear power plants.

Applicants for license renewal are required by the license renewal rule to perform an integrated plant assessment (IPA). The first step of the IPA, required by 10 CFR 54.21(a)(1), is to identify and list structures and components that are subject to an aging management review; 10 CFR 54.21(a)(2) requires the applicant to describe and justify the methods used in meeting the requirements of 10 CFR 54.21(a)(1); and 10 CFR 54.21(a)(3) requires that, for each structure and component identified under 10 CFR 54.21(a)(1), the applicant demonstrate that the effects of aging will be adequately managed so that the intended function or functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. Furthermore, the applicant must provide an evaluation of time-limited aging analyses (TLAAs), as required by 10 CFR 54.21(c), including a list of TLAAs, as defined in 10 CFR 54.3.

By letter dated December 11, 1996, the Westinghouse Owners Group (WOG) submitted the generic technical report (GTR) WCAP-14756, "Aging Management Evaluation for the Pressurized Water Reactor Containment Structure," for staff review and approval. The GTR identifies aging mechanisms, and presents options for managing aging effects to ensure that the intended functions of systems, structures, and components (SSCs) are maintained. The GTR states that the aging management options are to be developed into programs by utilities requesting license renewal. The GTR provides generic technical bases for demonstrating that the aging management options will adequately manage aging effects to maintain intended functions of SSCs, in accordance with the CLB for an extended period of operation.

2.0 SUMMARY OF TOPICAL REPORT

WCAP-14756 describes three typical configurations of containment structures applicable to Westinghouse pressurized water reactors, the applicable containment boundary, associated structural components, applicable aging effects, aging management programs, and time-limited aging analyses (TLAAs). The content and organization of the GTR are described in more detail in Section 2 of the enclosed Technical Evaluation Report (TER) prepared by the Brookhaven National Laboratory.

Enclosure

3.0 EVALUATION

The staff, with technical assistance from Argonne National Laboratory and Brookhaven National Laboratory (BNL), reviewed the GTR, and issued requests for additional information (RAI). After reviewing the GTR and WOG's responses to the staff's RAI, BNL issued a technical evaluation report (TER). The staff reviewed the GTR, WOG's responses to the RAI, and the TER.

As a result of their review, BNL recommended specific renewal applicant action items (Section 4.1), open items (Section 4.2), and confirmatory items (Section 4.3). At the request of WOG, the staff has reflected all of the actions as applicant action items in the following evaluation. In addition, the WOG would append all of the responses to the RAI in the GTR; therefore, the staff has incorporated the recommended confirmatory items as part of its conclusions in this evaluation. RAI 33 requested missing publication dates for references and bibliography documents listed in Section 6.0 of the WOG GTR. The requested dates were provided in the WOG response to the RAI.

Individual license renewal applicants are expected to include a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs in the FSAR supplement submitted with a license renewal application, in accordance with 10 CFR 54.21(d). This is Applicant Action Item 2.

The NRC staff has reviewed the BNL evaluation and concurs with the findings and recommended actions, as incorporated in the following safety evaluation. The staff has combined the open items in Section 4.2 of the TER with the applicant action items in Section 4.1 of the TER. The combined list is provided in Section 4 of this DSER.

3.1 Components and Intended Functions

The scope of the GTR is intended to be applicable to all domestic commercial nuclear power plants with the Westinghouse nuclear steam supply system. In referencing the GTR, a license renewal applicant should verify the applicability of the GTR to its plant. This verification should identify plant-specific data not covered by the GTR and that will be evaluated in the license renewal application. The extent of this verification should include: (i) verification that its plant is bounded by the GTR, (ii) a commitment to implement programs described as necessary in the GTR to manage the effects of aging during the period of extended operation, and (iii) verification that the programs committed to are conducted in accordance with appropriate regulatory controls (e.g. 10 CFR Part 50, Appendix B). Further, the renewal applicant will identify any deviations from the aging management programs which this GTR describes as necessary to manage the effects of aging during the period of extended operation or to maintain the functionality of the containment structure, and deviations from other information presented in the GTR (e.g., materials of construction). The renewal applicant will evaluate any such deviations in accordance with 10 CFR 54.21(a)(3) and (c)(1) on a plant-specific basis. This is Applicant Action Item 1.

As described in Section 3.1 of the TER, the GTR describes the various intended functions of the containment structure and structural components that are within the scope of license renewal. The following functions, which are specific to containment structures and are understood to be covered by the various intended functions, should be addressed explicitly in

the license renewal application: (1) providing structural or functional support of safety-related systems, structures, and components following a design basis accident (DBA); (2) serving as an external missile barrier consistent with the design and licensing basis; and (3) providing passive heat sinks during a DBA or station blackout in addition to the spray system. This is part of Applicant Action Item 1.

The GTR describes three containment types and associated structural components that are typically applicable to Westinghouse reactors. The description of the containment and structural components includes a range of configurations and dimensions, as described in Section 3.1.2 of the TER. In an RAI response, the WOG acknowledged errors in the basemat thickness range for Types 1, 2, and 3 containments and provided a corrected list.

Because of the variety of containment configurations and applicable structural components covered in the GTR, individual plant applicants will need to provide a comprehensive list of structures and components subject to an aging management review, the methodology used to develop this list as part of their license renewal applications, and the applicability of the GTR. Any components determined by the applicant to be subject to an aging management review for license renewal but not within the scope of the GTR need to be addressed in the license renewal application. This is Applicant Action Item 3.

Because of the potential for aging effects unique to plant-specific configurations of the containment structure and structural components, the license renewal application should augment the list of structures and components with cross-section drawings for the containment structures, and detailed drawings of the sand pocket region and other plant-specific features, if applicable. This is Applicant Action Item 4.

The application should also include legible drawings of equipment and penetration details as part of the description of the containment structure components. This is Applicant Action Item 5.

For prestressed concrete containments, the license renewal applications should indicate whether the tendon access gallery is included as a containment structure component subject to an aging management review. If it is, provide the details of the aging management review and the credited aging management program. If not, provide a technical basis for its exclusion, addressing the potential for degradation of the lower vertical tendon anchors resulting from the environmental conditions in the tendon access gallery. This is Applicant Action Item 6.

The GTR identified structural connections as containment structure components that require aging management in Table 2-1. However, there is no definition or description of structural connections in GTR Section 2.0. A definition and a description of the AMP for structural connections should be provided in individual license renewal applications. This is Applicant Action Item 25.

The GTR identified embedments as a containment structure components that require aging management in Table 2-1. However, there is no definition or description of embedments in GTR Section 2. A definition and a description of the AMP for embedments are needed. This is Applicant Action Item 26.

Upon successful completion of the applicant action items described above, the staff will be able to conclude that a license renewal application referencing WCAP-14756 will have satisfied the scoping requirements in 10 CFR 54.4(a) and 10 CFR 54.21(a)(1).

3.2 Effects of Aging

As described in Section 3.2.1 of the TER, the GTR provides a listing of primary and secondary aging effects for all containment structure component and aging mechanism combinations which are evaluated in detail in the TER.

The GTR broadly describes operating experience related to the assessment of aging effects applicable to the combinations of containment and structural component configurations and environments. However, individual plants may have unique operating experience that could define additional aging effects or affect the assessment of the effectiveness of aging management programs. Therefore, individual license renewal applications should discuss plant-specific operating experience relevant to age-related degradation of containment structure components and how this experience has been considered in the aging management review. This is Applicant Action Item 7.

For concrete containments, WOG described criteria that were relied on to exclude leaching of calcium hydroxide and reaction with aggregates as significant aging mechanisms. Individual applicants should verify that the original plant design and construction specifications satisfy these criteria. If this aging effect cannot be excluded, the applicant should describe the aging management program (AMP) which is credited to manage this aging effect. This is Applicant Action Item 8.

WOG also addressed the potential for thermal embrittlement of concrete, and concluded that concrete structures that do not experience localized temperatures substantially higher than the 200°F limit in the ACI code will not lose their intended function as a result of thermal embrittlement. Therefore, individual renewal applications should describe whether local heating of containment concrete at the main steam and/or any other penetrations results in sustained concrete temperatures exceeding 200°F. If this condition exists, the applicant would be expected to provide an aging management review and describe the credited aging management program. This is Applicant Action Item 9.

Upon successful completion of the applicant action items described above, the staff will be able to conclude that a license renewal application referencing WCAP-14756 will have addressed the aging effects applicable to the containment and structural components.

3.3 Aging Management Programs

As described in Section 3.3 of the TER, WOG identified seven aging management programs (AMPs) that would be relied on by individual plants in conjunction with the inspection and testing programs in the current licensing basis that are based on the ASME Code, the ACI Code and the approved Inservice Inspection Program.

In the response to RAIs, WOG clarified and augmented the GTR on several important details of the AMP descriptions. In the response to RAI 24, WOG acknowledged the necessity of

implementing Appendices VII and VIII of the ASME Code when ultrasonic examination is used. WOG stated that when implementing an aging management program that references Subsection IWE, and it is necessary to utilize augmented ASME Section XI NDE inspection methods, the training qualifications and certification of ultrasonic examination personnel will meet Appendix VII and Appendix VIII. In responses to RAIs 6, 7, 30 and 31 WOG corrected text errors in Table 3-1. RAI 20 clarified that AMP 5.5 refers to IWE examination categories taken from the 1989 Code Edition, instead of the 1992 Code Edition; in the response, WOG acknowledged this error and corrected examination categories.

Because of the variety in the individual plant licensing bases, inservice inspection programs and design codes, individual license renewal applicants will have to identify the codes, edition and/or date of codes and standards which govern plant containment design, inspection and repair. This is Applicant Action Item 10.

The GTR describes two programs to manage freeze-thaw where that aging effect is applicable, one for concrete containments and the other for concrete shield buildings. Therefore, individual license renewal applicants will have to specify whether freeze-thaw is an applicable aging mechanism which will be managed by AMP 5.1 or AMP 5.2 depending on the affected structure. If not, the applicant will be expected to provide the technical basis for its exclusion. This is Applicant Action Item 11.

Similarly, the GTR describes two programs to manage corrosion due to agressive chemical attack, one for concrete containments and the other for concrete shield buildings and foundation mats. Therefore, individual license renewal applicants will have to specify whether aggressive chemical attack is an applicable aging mechanism which will be managed by AMP 5.3 or AMP 5.4 depending on the affected structure. If not, the applicant will be expected to provide the technical basis for its exclusion. This is Applicant Action Item 12.

The extent to which ground water corrosion is an aging effect that warrants an aging management program depends on plant-specific site characteristics, the availability of chemistry data, and seasonal variations. Therefore, individual license renewal applicants are expected to provide details of the groundwater monitoring program and discuss potential seasonal variation in ground water chemistry. This is Applicant Action Item 13.

For prestressed concrete containments, the prestressing tendons would be inspected and maintained in accordance with Section IWL requirements in Section XI of the ASME Code. There are ingredients in the grease used to pack the tendons that could cause degradation of the concrete. Therefore, individual license renewal applicants should discuss plant experience with respect to tendon grease leakage and, if applicable, how the leakage will be managed; the application should also discuss the potential effects of grease leakage on the shear load capacity of the containment structure. This is Applicant Action Item 14.

Stress corrosion cracking (SCC) is a concern for dissimilar metal welds. For the period of extended operation, the staff concludes that ASME Section XI, IWE examination Categories E-B and E-F and augmented VT-1 visual examination of bellows assemblies and dissimilar metal welds would provide adequate inspection. Therefore, the license renewal applicant needs to describe its plant-specific program to address SCC for dissimilar metal welds. The

program should include stainless steel bellows assemblies, if the material is not shielded from a corrosive environment. This is Applicant Action Item 15.

Coatings are provided on structural elements as a protective feature. While WOG concluded that there is no need for an aging management program for coating degradation, the staff concurs with the BNL conclusion that coating inspection and repair is usually an integral part of the maintenance programs for steel structures. Therefore, license renewal applicants should discuss the plant-specific coatings monitoring and maintenance program and specify whether it is credited as an AMP for containment steel elements. This is Applicant Action Item 16.

For prestressed concrete containments, the GTR indicates that AMP 5.6 will be used to manage potential degradation of the containment post-tensioning systems. License renewal applicants should specify whether post-tensioning system degradation will be managed by AMP 5.6 (Section XI, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants, Examination Category L-B, Unbonded Post-Tensioning System, 1992 Code Edition with 1992 Addenda of the ASME Code) and the additional requirements delineated in 10 CFR 50.55a(b)2(ix). If not, the applicant should provide the technical basis for its exclusion. This is Applicant Action Item 17.

Settlement of the containment foundation is a potential aging mechanism which will be managed by AMP 5.7. The WOG indicated that most of the settlement occurs within the first 5 to 6 years of plant operation. Therefore, this is a plant-specific issue. The license renewal applicant should specify whether settlement is an applicable aging mechanism that will be managed by AMP 5.7. If not, the applicant should provide the technical basis for its exclusion. This is Applicant Action Item 18.

The GTR did not address the errosion of cement in sub-foundation layers of porous concrete basemats, because WOG considered that type of construction to be limited. Therefore, license renewal applicants should identify whether erosion of the porous concrete sub-foundation layer is an applicable aging effect and, if applicable, provide an aging management review and describe the aging management program. This is Applicant Action Item 19.

The "Draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," issued in 1997 and updated in April 2000, identifies ten (10) elements (attributes) as appropriate for an acceptable AMP. The staff has concluded based on operating experience and the assessment of effective maintenance programs that these ten elements will ensure an adequate aging management program. The GTR listed only six (6) attributes to form the basis for each aging management program. The GTR predates the Draft Standard Review Plan for the review of License Renewal Applications for Nuclear Power Plants, and states in Section 4.0 that the report only presents program attributes for the AMPs, and that plant-specific details of the AMPs will be developed during the preparation of license renewal applications. Therefore, applicants for license renewal will be responsible for developing and describing the plant-specific AMPs and addressing each of the ten elements specified in the Draft Standard Review Plan. This is Applicant Action Item 20.

The GTR does not commit to inspection of inaccessible areas when there is no indication of degradation of adjacent accessible areas, except when the potential for degradation is "event driven"; i.e., some unusual event has occurred which has the potential to degrade inaccessible

areas of the containment structures. Therefore, the GTR cannot be referenced by license renewal applicants for managing aging of inaccessible areas. Individual license renewal applicants are required to describe a program for inspection of inaccessible areas or adopt a program endorsed by the staff in similar applications. This is Applicant Action Item 27.
The aging effects in concrete due to leaching of calcium hydroxide and the alkali-aggregate reaction are identified in the GTR as not requiring aging management; however, this conclusion cannot be justified on such a broad generic basis and a plant-specific evaluation should be performed to determine if these aging effects warrant an aging management program. These aging mechanisms, if applicable, can be managed by ASME Code, Section XI, Examination Category L-A. This is a required inspection program, which will continue into the period of extended operation. The GTR identifies examination category L-A of IWL to manage freeze-thaw (AMP5.1) and to manage aggressive chemical attack and steel corrosion (AMP5.3), where applicable. According to the GTR, it may not be necessary to credit examination category L-A to manage these aging mechanisms if on a plant specific basis, the applicant determines these aging mechanisms are not applicable. Consequently, in these cases, examination category L-A would not be credited to manage concrete degradation and steel corrosion for License Renewal. In general, it is intended that maximum credit be taken for existing mandated programs (e.g., examination category L-A) in the development of an applicant's aging management program. The license renewal applicants should specify that they are implementing examination category L-A as an aging management program for containment concrete or propose a suitable alternative. This is Applicant Action Item 28.

Upon successful completion of the applicant action items described above, a license renewal application referencing WCAP-14756 will have addressed the aging management program requirements such that the staff can conclude that there is reasonable assurance that applicable aging effects will be adequately managed in accordance with 10 CFR 54.21(a)(3).

3.4 Time-Limited Aging Analyses

Time-limited aging analyses (TLAAs) are defined in 10 CFR 54.3, as described in Section 3.4 of the TER. The GTR evaluated TLAAs associated with component fatigue and loss of prestress loads.

The WOG GTR indicates that the license renewal applicant may update an existing design fatigue analysis to account for the additional years of plant operation or manage the effects of the aging through aging management programs. The GTR uses AMP 5.5 for managing the effects of fatigue during the renewal license period, and basically endorses the ASME Code Section XI surveillance and testing program. For components where fatigue TLAAs exist, this option would allow the fatigue Section III cumulative usage factors (CUF) to be exceeded during the period of extended operation. The staff has not endorsed this option on a generic basis at this time. An applicant wishing to pursue this option would have to obtain staff review and approval on a case-by-case basis. For components where fatigue TLAAs do not exist (are not addressed in 10 CFR 54.21), aging effects due to fatigue can be addressed by either a Section III fatigue analysis (including the additional years for the period of extended operation) or by adequately managing these effects for the period of extended operation. This is Applicant Action Item 21.

Because of the variety of structural components covered by the GTR, applicants should specify the containment structure components which have fatigue design analyses, and provide plant-specific details of the TLAAs for prediction of cumulative fatigue usage through the period of extended operation. This is Applicant Action Item 22.

Some containment structure components may be susceptible to fatigue, but do not have a fatigue design analysis as part of the current licensing basis. The applicant should specify whether any plant-specific components do not have a fatigue TLAA, but warrant an aging management program. In addition to implementation of AMP 5.5, the staff would expect that the requirements of 10 CFR 50.55a would apply. This is Applicant Action Item 23.

WOG determined that the calculations to determine the prestress loss rate for the period of extended operation, for prestressed concrete containments, would be addressed by individual applicants. Individual applicants will be expected to provide plant-specific details of the TLAA for prediction of tendon prestress losses. This is Applicant Action Item 24.

Upon successful completion of the applicant action items described above, the staff will be able to conclude that a license renewal application referencing WCAP-14756 will have adequately addressed time-limited aging analyses in accordance with 10 CFR 54.21(c)(1).

4.0 <u>RENEWAL APPLICANT ACTION ITEMS</u>

The following renewal applicant action items are to be addressed in the plant-specific license renewal application that references the WOG GTR:

1. The license renewal applicant will (i) verify that its plant is bounded by the GTR, (ii) commit to implement programs described as necessary in the GTR to manage the effects of aging during the period of extended operation, and (iii) verify that the programs committed to are conducted in accordance with appropriate regulatory controls (e.g. 10 CFR Part 50, Appendix B). Further, the renewal applicant will identify any deviations from the aging management programs which this GTR describes as necessary to manage the effects of aging during the period of extended operation or to maintain the functionality of the containment structure, and deviations from other information presented in the GTR (e.g., materials of construction). The renewal applicant will evaluate any such deviations in accordance with 10 CFR 54.21(a)(3) and (c)(1) on a plant-specific basis.

   The following functions, which are specific to containment structures and are understood to be covered by the various intended functions, should be addressed explicitly in the license renewal application: (1) providing structural or functional support of safety-related systems, structures, and components following a design basis accident (DBA); (2) serving as an external missile barrier consistent with the design and licensing basis; and (3) providing passive heat sinks during a DBA or station blackout in addition to the spray system.

2. A summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs is to be provided in the license renewal FSAR supplement, in accordance with 10 CFR 54.21(d).

3. Individual plant applicants will need to provide a comprehensive list of structures and components subject to an aging management review and the methodology used to develop this list as part of their license renewal applications. Any components determined by the applicant to be subject to an aging management review for license renewal but not within the scope of the GTR are required to be addressed in the license renewal application.

4. Provide cross-section drawings for the containment structures, and detailed drawings of the sand pocket region and other plant-specific features, if applicable.

5. Provide legible drawings of equipment and penetration details as part of the description of the containment structure components.

6. For prestressed concrete containments, indicate whether the tendon access gallery is included as a containment structure component subject to an aging management review. If it is, provide the details of the aging management review and the credited aging management program. If not, provide a technical basis for its exclusion, addressing the potential for degradation of the lower vertical tendon anchors resulting from the environmental conditions in the tendon access gallery.

7. Discuss plant-specific operating experience relevant to age-related degradation of containment structure components and how this experience has been considered in the aging management review.

8. For concrete containments, verify that the original plant design and construction specifications satisfy the criteria which are relied upon to exclude leaching of calcium hydroxide and reaction with aggregates as significant aging mechanisms. If these mechanisms are not excluded, describe the aging management program (AMP) which is credited to manage the aging effects associated with these aging mechanisms.

9. For concrete containments, discuss whether local heating of containment concrete at the main steam and/or any other penetrations results in sustained concrete temperatures exceeding 200°F. If this condition exists, provide an aging management review and describe the credited aging management program.

10. Identify the codes, edition and/or date of codes and standards which govern plant containment design, inspection and repair.

11. Specify whether freeze-thaw is an applicable aging mechanism which will be managed by AMP 5.1 or AMP 5.2, as applicable. If not, provide the technical basis for exclusion.

12. Specify whether aggressive chemical attack is an applicable aging mechanism which will be managed by AMP 5.3 or AMP 5.4, as applicable. If not, provide the technical basis for exclusion.

13. Provide details of the groundwater monitoring program and discuss potential seasonal variation in ground water chemistry.

14. For prestressed concrete containments, discuss plant experience with respect to tendon grease leakage and, if applicable, how the leakage will be managed; also discuss the potential effects of grease leakage on the shear load capacity of the containment structure.

15. Each license renewal applicant needs to describe its plant-specific program to address the stress corrosion cracking (SCC) for dissimilar metal welds, and for stainless steel bellows assemblies, if the material is not shielded from a corrosive environment. For the period of extended operation, ASME Section XI, IWE examination Categories E-B and E-F and augmented VT-1 visual examination of bellows assemblies and dissimilar metal welds are required or a suitable alternative proposed.

16. Discuss the plant-specific coatings monitoring and maintenance program and specify whether it is credited as an AMP for containment steel elements.

17. For prestressed concrete containments, specify whether post-tensioning system degradation will be managed by AMP 5.6 (Section XI, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants, Examination Category L-B, Unbonded Post-Tensioning System, 1992 Code Edition with 1992 Addenda of the ASME Code) and the additional requirements delineated in 10 CFR 50.55a(b)2(ix). If not, provide the technical basis for exclusion.

18. Specify whether settlement of the containment foundation is an applicable aging mechanism which will be managed by AMP 5.7. If not, provide the technical basis for exclusion.

19. Identify whether erosion of the porous concrete sub-foundation layer is an applicable aging mechanism; if applicable, provide an aging management review and describe the credited aging management program.

20. The GTR listed only six (6) attributes to form the basis for each aging management program. However, the "Draft Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," dated April 21, 2000, identifies ten (10) elements (attributes) as appropriate for an acceptable AMP. The GTR predates the Draft standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants, and states in Section 4.0 that the report only presents program attributes for the AMPs, and that plant-specific details of the AMPs will be developed during the preparation of license renewal applications. Therefore, applicants for license renewal will be responsible for developing and describing the plant-specific AMPs and addressing each of the ten elements specified in the Draft Standard Review Plan.

21. The WOG GTR indicates that the license renewal applicant may update an existing design fatigue analysis to account for the additional years of plant operation or manage the effects of the aging mechanism through aging management programs. The GTR uses AMP 5.5 for managing the effects of fatigue during the renewal license period, and basically endorses the ASME Code Section XI surveillance and testing program. For components where CLB fatigue TLAAs exist, this option would allow the CLB fatigue Section III cumulative usage factors (CUF) to be exceeded during the period of

extended operation. The staff has not endorsed this option on a generic basis at this time. An applicant wishing to pursue this option would have to obtain staff review and approval on a case-by-case basis. For components where CLB fatigue TLAAs do not exist (are not addressed in 10 CFR 54.21), aging effects due to fatigue can be addressed by either a Section III fatigue analysis (including the additional years for the period of extended operation) or by adequately managing these effects for the period of extended operation.

22. Specify the containment structure components and provide plant-specific details of the TLAAs for prediction of cumulative fatigue usage through the period of extended operation.

23. Specify those containment structure components for which fatigue is an applicable aging mechanism, but no CLB fatigue analysis based on a 40 year plant life exists. In addition to implementation of AMP 5.5, the requirements of 10 CFR 50.55a should be met.

24. For prestressed concrete containments, provide plant-specific details of the TLAA for prediction of tendon prestress losses through the period of extended operation.

25. The GTR identified structural connections as containment structure components that require aging management in Table 2-1. However, there is no definition or description of structural connections in GTR Section 2.0. A definition and a description of the AMP for structural connections are needed.

26. The GTR identified embedments as containment structure components that require aging management in Table 2-1. However, there is no definition or description of embedments in GTR Section 2. A definition and a description of the AMP for embedments are needed.

27. The GTR does not commit to inspection of inaccessible areas when there is no indication of degradation of adjacent accessible areas, except when the potential for degradation is "event driven"; i.e., some unusual event has occurred which has the potential to degrade inaccessible areas of the containment structures. Therefore, the GTR cannot be referenced by license renewal applicants for managing aging of inaccessible areas. Individual license renewal applicants are required to describe a program for inspection of inaccessible areas or adopt a program endorsed by the staff in similar applications.

28. The aging effects in concrete due to leaching of calcium hydroxide and alkali aggregate reaction are identified in the GTR as not requiring aging management. This is unacceptable because plant-specific evaluation of their applicability is needed. Therefore, if these aging mechanisms (leaching of calcium hydroxide and alkali aggregate reaction) are applicable, applicants would be required to propose a plant specific aging management program. Alternatively, applicant can credit the ASME Code, Section XI, Examination Category L-A as an adequate aging management program.

5.0 CONCLUSION

The staff reviewed the generic technical report (GTR) WCAP-14756, "Aging Management Evaluation for the Pressurized Water Reactor Containment Structure," submitted by Westinghouse Owners Group and the technical evaluation report on the GTR by Brookhaven National Laboratory (BNL). On the basis of its review, the staff concludes that, upon successful completion of the applicant action items, applicants referencing the GTR in a license renewal application will satisfy the requirements of 10 CFR 54.21(a)(3) and 54.21(c)(1), and the staff will have sufficient information to conclude that there is reasonable assurance that applicable aging effects will be adequately managed for the period of extended operation and applicable time-limited aging analyses have been evaluated in accordance with 10 CFR 54.29(a)(1) and (a)(2) for containment structure components within the scope of the GTR.

TECHNICAL EVALUATION REPORT (TER)

FOR

WESTINGHOUSE OWNERS GROUP
GENERIC TECHNICAL REPORT WCAP-14756
REVISION 0, DECEMBER 1996
"AGING MANAGEMENT EVALUATION FOR PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURE"

PROJECT NO. 686

Prepared by

R. Morante
P. Bezler
J. Braverman
Energy Sciences and Technology Department
Brookhaven National Laboratory

for

NRC/NRR/DRIP/RLSB

October 2000

TABLE OF CONTENTS

1.0 INTRODUCTION

Pursuant to Section 50.51 of Title 10 of the Code of Federal Regulations (10 CFR 50.51), licenses to operate nuclear power plants are issued by the U.S. Nuclear Regulatory Commission (NRC) for a fixed period of time not to exceed 40 years; however, these licenses may be renewed by the NRC for a fixed period of time not to exceed 20 years beyond expiration of the current operating license. The Commission's regulations in 10 CFR Part 54, published May 8, 1995, set forth the requirements for the renewal of operating licenses for commercial nuclear power plants (Reference 1).

Applicants for license renewal are required by the license renewal rule to perform an integrated plant assessment (IPA). The first step of the IPA, 10 CFR 54.21(a)(1), requires the applicant to identify and list structures and components that are subject to an aging management review; 10 CFR 54.21(a)(2) requires the applicant to describe and justify the methods used in meeting the requirements of 10 CFR 54.21(a)(1); and 10 CFR 54.21(a)(3) requires that, for each structure and component identified in 10 CFR 54.21(a)(1), the applicant demonstrate that the effects of aging will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis (CLB) for the period of extended operation. Furthermore, the applicant must provide an evaluation of time-limited aging analyses (TLAAs) as required by 10 CFR 54.21(c), including a list of TLAAs, as defined in 10 CFR 54.3.

1.1 Westinghouse Owners Group Generic Technical Report

By letter dated December 20, 1996, the Westinghouse Owners Group (WOG) submitted Generic Technical Report (GTR) WCAP-14756, "Aging Management Evaluation for the Pressurized Water Reactor Containment Structure," (Reference 2), for NRC staff review and approval. The objectives of this GTR are to (1) identify and evaluate aging effects that degrade components which affect systems or structures intended functions, (2) identify and evaluate time-limited aging analyses (TLAAs), and (3) provide options, in terms of activities and program attributes, to manage these aging effects, and if necessary address TLAAs. The aging management options provided in the GTR are to be developed into programs by utilities requesting license renewal. The GTR provides generic technical bases supporting a part of a demonstration that the management options adequately manage aging effects to maintain intended functions for systems, structures, and components, consistent with the current licensing basis, for an extended period of operation.

1.2 Applicability of the GTR

The GTR is intended to be applicable generically to domestic commercial nuclear power plants with the Westinghouse nuclear steam supply system (NSSS), as listed in Table 1-1 of the GTR. Use of the GTR, as referenced by a license renewal application, should include a verification of all the bounding information against plant-specific data. This verification should identify the applicability of the GTR to the plant and determine what plant-specific data are not covered by the GTR and will be evaluated as part of the license renewal application.

2.0 SUMMARY OF THE GENERIC TECHNICAL REPORT

The GTR describes different types, parts, and materials of Westinghouse pressurized water reactor (PWR) containments and their boundaries, and identifies and evaluates age-related degradation mechanisms and the TLAAs. The GTR also describes how the age-related degradation can potentially degrade the intended functions of the PWR containment and identifies aging effects and TLAAs that require management. The GTR further provides aging management plan attributes, including their details and implementation guidance, that form the basis for programs to be developed and implemented by utilities to manage aging effects for the extended period of operation.

2.1 Intended Functions and Containment Structure Components

2.1.1 Intended Functions

The GTR indicates that the intended functions of the Pressurized Water Reactor (PWR) containment structure that are within the scope of license renewal include:

(1) ensuring the integrity of the reactor coolant pressure boundary

(2) ensuring the capability to shutdown the reactor and maintain it in a safe shutdown condition

(3) ensuring the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR 100 guidelines

(4) ensuring compliance with U.S. NRC regulations for environmental qualification (10 CFR 50.49) where intended functions (1) and (2) are included as a result of the structural support provided by the containment.

2.1.2 Containment Structure Components

Section 2 of the GTR describes the scope of the report. This includes the containment boundary, engineering and design data, identification of TLAAs, general maintenance practices, and a summary of the aging effects. The containment boundary consists of the interior and exterior surfaces of the reinforced concrete containment shell, including the basemat, or the interior surface of the steel containment and the exterior surface of the shield building, including the common basemat. The boundary includes all penetration assemblies in the containment shell, such as mechanical and electrical penetrations, the equipment and personnel hatches and the fuel transfer tube. For mechanical penetrations, the entire mechanical penetration assembly (except the process piping) and the welds to the process pipe are in the scope of the GTR. For electrical penetrations, all metallic components that are part of the containment pressure boundary (but excluding nonmetallic seal materials) are in the scope of the GTR. The Westinghouse ice condenser, if part of the plant containment system, is not included in the scope of the GTR.

Three configurations of the PWR containment are addressed in the report. These include:

(1) Steel containment with reinforced concrete shield building
  a. with ice condenser
  b. no ice condenser

(2) Steel-lined reinforced concrete
  a. reactor building is atmospheric
  b. reactor building is sub-atmospheric
  c. reactor building is sub-atmospheric with ice condenser

(3) Steel-lined reinforced concrete with post-tensioning
  a. three directional post-tensioning
  b. vertical post-tensioning only

WOG divides the PWR containment into ten containment structure components (stated as parts or subcomponents in the GTR). These are the shell and dome, structural mat, structural connections, concrete reinforcing and tendons, steel liner, embedments, electrical penetrations including connectors, penetrations for gas and fluid systems that include isolation valves, fuel transfer tube, and equipment and personnel hatches that include seals. All of these containment structure components perform intended functions in a passive manner and are long-lived. Therefore, they are within the scope of 10 CFR Part 54 and are subject to an aging management review.

A listing of age-related degradation mechanisms applicable to PWR containment structures, a summary of containment aging evaluations, and an aging effects list are provided in GTR Tables 2-16, 2-17 and 2-18, respectively.

2.2 Effects of Aging

Section 3 of the GTR discusses the age-related degradation mechanisms that affect the PWR containment and evaluates time-limited aging analyses. The report states that the following aging mechanisms require aging management programs:

- Freeze-thaw of concrete
- Aggressive chemical attack on concrete
- Corrosion in reinforcing steel
- Corrosion of steel liners, steel containment shells, and penetrations
- Degradation in containment post-tensioning systems
- Mechanical wear and/or loss of pressure retention of airlocks and hatches
- Embrittlement of gaskets
- Stress corrosion cracking of penetrations and bellows
- Coating degradation
- Fatigue of penetrations and bellows

Table 3-1 of the GTR provides a summary of PWR containment structures aging evaluations and the status of applicable agreements reached between the Nuclear Energy Institute (NEI) and NRC. The table lists the aging degradation mechanisms, the corresponding aging effects, the components affected, the criteria/program used to evaluate them, and the applicable NEI/NRC

agreements or positions. Detailed descriptions of the degradation mechanism, aging effect evaluation and aging effect management, for each aging degradation mechanism, are provided in Section 3.2 of the GTR.

2.3 Aging Management Programs

A summary listing of degradation mechanisms/effects managed by aging management programs, into the period of extended operation, is provided in Section 3.4 of the GTR. It is stated that aging mechanisms are adequately managed using seven aging management programs, AMP 5.1 through AMP 5.7, which are based on the current licensing basis (CLB), including ASME Code Section XI and ACI Code, inspection and test programs. Programs AMP 5.1 and AMP 5.2 address the aging effects due to freeze-thaw in concrete; programs AMP 5.3 and AMP 5.4 address the aging effects due to aggressive chemical attack of concrete and corrosion of the reinforcing steel; program AMP 5.5 addresses the aging effects for penetrations and liner/steel containment; program AMP 5.6 addresses the aging effects for the post-tensioning systems; and program AMP 5.7 addresses settlement of foundations. A detailed description of the aging management activities and programs is provided in Section 4 of the GTR. A listing of each program, along with the containment structure components managed, aging mechanism and aging effect, is provided in Table 4-2.

The GTR describes attributes of acceptable utility programs to manage aging effects for the extended period of operation. Table 4-1 of the GTR lists six (6) attributes intended to form the basis for aging management programs (AMPs) that will be developed and implemented by utilities during an extended period of operation. These attributes include scope of the program, surveillance techniques to be used to detect aging effects, frequency of the surveillance, acceptance criteria to determine when corrective actions are necessary, the corrective actions, and confirmation techniques. A summary description of the attributes of each aging management program, with code references to ASME Section XI, 1992 Edition with 1992 addenda (Reference 3) and ACI codes and standards, where applicable, is provided in Tables 4-9 through 4-15 of the report.

2.4 Time Limited Aging Analyses

It is stated in Section 3.3 of the GTR that prestress force losses and fatigue have the potential to be defined as TLAA effects. The TLAAs potentially applicable are:

- analytical prediction of time-dependent loss of prestress force loads in prestressing systems
- number of fatigue cycles at penetration anchors in concrete containments, and where appropriate, calculated cumulative fatigue usage factors
- number of fatigue cycles in bellows of mechanical penetrations, and where appropriate, calculated cumulative fatigue usage factors
- number of fatigue cycles of mechanical penetrations, and where appropriate, calculated cumulative fatigue usage factors

3.0 EVALUATION

BNL reviewed the GTR and the responses to the staff's RAIs to determine if WOG has demonstrated that the effects of aging of the containment structure components will be adequately managed so that the intended functions will be maintained, consistent with the CLB, for the period of extended operation.

The license renewal applicant will (i) verify that its plant is bounded by the GTR, (ii) commit to programs described as necessary in the GTR to manage the effects of aging during the period of extended operation and (iii) verify that the programs committed to are conducted in accordance with appropriate regulatory controls (e.g., 10 CFR Part 50, Appendix B). Further, the renewal applicant will identify any deviations from the aging management programs which this GTR describes as necessary to manage the effects of aging during the period of extended operation or to maintain the functionality of the containment structure, and deviations from other information presented in the GTR (e.g., materials of construction). The renewal applicant will evaluate any such deviations in accordance with 10 CFR 54.21(a)(3) and (c)(1) on a plant-specific basis. This is Renewal Applicant Action Item No. 1 (see Section 4.1).

10 CFR Part 54.21(c) requires an evaluation of TLAAs. BNL reviewed the report and the responses to the staff's RAIs submitted by the WOG to determine if the TLAAs covered by the report were evaluated for license renewal in accordance with 10 CFR 54.21(c)(1).

A summary of description of the programs and activities for managing the effects of aging and the evaluation of TLAAs is to be provided in the license renewal FSAR supplement, in accordance with 10 CFR 54.21(d). This is Renewal Applicant Action Item No. 2.

3.1 Intended Functions and Containment Structure Components

3.1.1 Intended Functions

The GTR indicates that the following PWR containment structure component intended functions are within the scope of license renewal as described in 10 CFR 54.4:

(1) Ensuring the integrity of the reactor coolant pressure boundary

(1) Ensuring the capability to shutdown the reactor and maintain it in a safe shutdown condition

(2) Ensuring the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR 100 guidelines

(3) Ensuring compliance with U.S. NRC regulations for environmental qualification (10 CFR 50.49)

BNL agrees with the above assessment, but finds that the following three additional functions, which are unique to containment structures, should be added to the intended functions identified above.

(1) Providing structural or functional support of safety-related systems, structures, and components following a design basis accident (DBA).

(2) Serving as an external missile barrier consistent with design basis commitments.

(3) Providing a heat sink during a DBA or station blackout in addition to the spray system.

The above identified functional additions are designated as Open Item No. 1 (see Section 4.2).

3.1.2 Containment Structure Components

The WOG considered three PWR containment types and their variations. These included steel containments with and without ice condensers; steel-lined reinforced concrete containments which are atmospheric, sub-atmospheric, and sub-atmospheric with ice condenser; and steel-lined reinforced concrete containments with three-directional or vertical-only post-tensioning. For the steel containment, the reinforced concrete shield building was considered part of the containment.

The WOG divided the containments into ten containment structure components:

- Shell and dome
- Structural foundation mat
- Structural connections
- Concrete reinforcing and tendons
- Steel liner
- Embedments
- Electrical penetrations including connectors
- Penetrations for gas and fluid systems that include isolation valves
- Fuel transfer tube
- Equipment and personnel hatches that include seals

The following sections discuss the containment structure components evaluated in the GTR. The staff notes, however, that the report is generically applicable to nuclear power plants with a Westinghouse NSSS and does not necessarily constitute a complete listing of the containment structure components subject to an aging management review for the WOG member plants, as required by 10 CFR 54.21(a)(1). In addition, WOG does not describe and justify the methodology for the generation of such a list, as required by 10 CFR 54.21(a)(2). Therefore, individual plant applicants will need to (1) provide a list of the plant-specific containment structure components subject to an aging management review, and (2) describe and justify the methodology used to develop this list, as part of their license renewal applications. This issue is designated as Renewal Applicant Action Item No. 3.

In RAI No. 5 (NRC letter dated February 19,1998), the staff requested WOG to provide cross-section drawings and configuration descriptions for other types of containments (i.e. Types 1a, 1b, 2a, 2b, 2c, 3a and 3b) which were not included in the GTR, and figures showing details of the sand pocket region and embedded shell region. In response, WOG indicated that the figures included in the GTR were provided to show the general type of configurations for the three types of Westinghouse PWR containments, and that the subtypes and other plant-specific features will be included in the plant license renewal application. This issue is designated as Renewal Applicant Action Item No. 4.

Since many of the sketches of equipment details presented in the GTR are not legible (e.g. Figures 2-2, 2-7, 2-10 and 2-13), the staff requested in RAI No. 32 (NRC letter dated February 19,1998) that clear and legible figures be provided. In response (letter dated May 29, 1998), WOG indicated that the sketches in the GTR provide overall general characteristics of the different configurations and types, and do not provide details to the level requested in the RAI. WOG indicated that the requested details would be provided as needed as part of the license renewal application. This issue is designated as Renewal Applicant Action Item No. 5.

3.1.2.1 Containment Shell and Dome

The configurations of the PWR containment vary from plant to plant. WOG grouped them into three basic configuration types; steel containment with reinforced concrete shield building, steel-lined reinforced concrete, and steel-lined reinforced concrete with post-tensioning. Only the general type of configurations for the three types of containment are addressed within the GTR. WOG stated that subtypes within the general types, along with plant-specific features (e.g., sand pocket regions), will be included in the plant-specific license renewal applications.

3.1.2.1.1 Steel Containment with Reinforced Concrete Shield Building

The containment function is performed by a steel containment vessel in combination with a separate reinforced concrete shield building that surrounds the steel containment. The shield building has a vertical right cylinder wall, supported on a flat circular basemat and capped with a hemispherical dome. The containment vessel is a free standing steel structure consisting of a cylindrical wall, a hemispherical dome, and a bottom liner plate enclosed in concrete. The vessel is typically anchored to the shield building foundation with anchor bolts welded around the circumference of the base. The containment and shield building are separated by an annular air space. The air space and containment, or just the airspace is maintained at sub-atmospheric pressure. The containment may be equipped with an ice condenser (vapor suppression), designated Type 1a by WOG, or may be without an ice condenser (dry suppression), designated Type 1b by WOG. The ice condenser system, if used, is located inside the containment, between the crane wall and the steel shell. The ice condenser system is excluded from the scope of the GTR.

The steel containment vessel is typically 115 feet in diameter and 171 feet in overall height. The bottom liner plate is 1/4 inch thick, anchored to the foundation slab, and covered by a reinforced concrete floor slab 2 feet in thickness. The thickness of the cylinder wall and dome may be uniform or vary with height. If uniform, the typical thicknesses are 3/4 inches and 11/16 inches, for the wall and dome respectively. If non-uniform, the cylinder wall thickness typically varies

from 1-3/8 inches at its base to ½ inch at the spring line. The dome varies from 7/16 inches at the spring line to 15/16 inches at its apex. On the exterior surface of the vessel, either circumferential and vertical stiffeners or circumferential ring girders with vertical stringers are fitted. Local shell thickening is provided at penetrations.

The typical shield building has a cylindrical wall 127 feet in diameter and 3 feet thick. The dome inner radius is 87 feet and the dome thickness ranges from 24 to 30 inches. The containment basemat slab has a diameter greater than the shield building wall and is stated to range in thickness from 6 to 9 feet. The cylinder wall reinforcing steel is applied throughout the structure in a horizontal and vertical pattern near the inner and outer surface. The vertical reinforcement is continued into the dome and extended into the basemat. Penetrations smaller than 12 inches fit within the reinforcing steel pattern. For penetrations larger than 12 inches the reinforcement is formed in hoops around the opening. The dome reinforcement is arranged in a radial and circumferential pattern with the radial bars being an extension of the wall vertical bars. The slab reinforcement consists of concentric and radial bars at the top and bottom surfaces.

Upon review of the information presented in the GTR, the staff noted that the minimum containment basemat thickness is essentially four feet for containment types 1a and 1b and can be as low as two feet for Type 3 containments. This was addressed in RAI No. 29 (NRC letter dated February 19,1998). WOG agreed with the staff's comment in its response to the RAI (WOG letter dated May 29, 1998) and indicated that appropriate corrections would be made in the GTR. This issue is designated as Confirmatory Item No. 1 (see Section 4.3).

3.1.2.1.2 Steel-Lined Reinforced Concrete Containment

The structure is a vertical cylinder capped with a hemispherical dome supported on a flat basemat. It is designated Type 2 by the WOG. It may be designed to operate at sub-atmospheric pressure to limit leakage.

The reinforced concrete cylinder is 4 to 4-1/2 feet thick, with an inside diameter of 140 feet and a length of 131 feet. The dome has an inside radius of 70 feet and a thickness of 2-1/2 to 3-1/2 feet. The basemat diameter varies from 153 to 158 feet and the thickness varies from 6 to 10 feet. Steel reinforcement is arranged horizontally and vertically, and supplemented with diagonal reinforcement, in the cylinder wall. Steel reinforcement is arranged meridionally and horizontally in the dome. Steel reinforcement is arranged in a rectangular grid at the bottom of the basemat and in a concentric and radial grid at the top of the basemat. All reinforcement is located near the concrete surfaces. The cylindrical wall vertical steel reinforcement extends into both the dome and basemat.

3.1.2.1.3 Steel-Lined Reinforced Concrete Containment with Post-Tensioning

This containment, designated Type 3, is similar to the Type 2 containment except that post-tensioning tendons are used in addition to the conventional steel reinforcement in the concrete cylinder wall and dome. The Type 3a containment uses three directional post-tensioning while Type 3b uses vertical post-tensioning only.

The cylindrical wall portion of the Type 3a containment is prestressed by a post-tensioning system composed of horizontal and vertical tendons. The horizontal tendons are either placed in three 240 degree segments using three vertical buttresses spaced 120 degrees apart, or in six 120 degree segments using six vertical buttresses spaced 60 degrees apart. The three way dome tendons are anchored at the sides of the ring girder. A continuous tendon access gallery is provided beneath the base slab for the installation and inspection of the vertical tendons.

The vertical tendons are anchored at the top of the ring girder at the dome periphery and at the bottom of the foundation slab. The horizontal tendons are anchored at two buttresses spaced either 240 degrees apart, or 120 degrees apart, for the three buttress and six buttress configurations, respectively. In either configuration the horizontal tendons pass through one intervening buttress. All tendons are continuous and are bent to curve around major penetrations.

The cylinder wall portion of a Type 3b containment is reinforced circumferentially and prestressed vertically. The prestressing system consists of a number of tendons placed at intervals around the periphery at the cylinder wall centerline. The tendons are composed of a number of high strength steel bars or steel wire. Each tendon is sheathed with a 6 inch galvanized steel pipe or a galvanized corrugated steel tube or conduit. Corrosion protection of post-tensioned steel tendons is provided by filling the sheath with microcrystalline petrolatum containing organic-based corrosion inhibitors. For one Westinghouse plant, the H. B, Robinson Unit 2, corrosion protection is provided by filling the ducts with portland cement grout.

In RAI No. 9 (NRC letter dated February 19,1998), the staff questioned whether the tendon access gallery is part of the containment structure and whether it is subject to aging management review for license renewal. If the tendon access gallery is not considered part of the containment structure, the concern is that degradation of the tendon gallery could lead to degradation of the tendon prestressing system.

In its response, WOG indicated that tendon access galleries are not generically considered to be part of the containment. Therefore, access galleries are only subject to aging management review if they are considered, on a plant-specific basis, to support the integrity of the prestressing system. WOG indicated that the above statement would be noted in the plant-specific license renewal application. The issue of plant-specific discussion of its tendon access gallery is designated as Renewal Applicant Action Item No. 6.

3.1.2.2 Foundations

PWR containment foundations typically consist of a reinforced mat that is supported on rock, soil, or a deep foundation such as piles. Pile foundations are not within the scope of the GTR and, if used, are to be addressed in the plant-specific application for licence renewal.

3.1.2.3 Structural Connections

Structural connections have been identified as a containment structure component by WOG. However, no specific description of structural connections is provided in Subsection 2.3 of the

GTR. A definition and description of structural connections needs to be added to this subsection. This issue is designated as Open Item No. 2.

3.1.2.4 Concrete Reinforcement and Tendons

Concrete reinforcement was placed to meet the requirement of the applicable codes and standards. Different code editions are used based on the vintage of the plant. The arrangement of reinforcing steel varied somewhat between containment types. For the shield building the vertical reinforcement steel continues from the cylindrical wall to the dome and extends into the foundation slab. In the wall, steel reinforcing bars are applied throughout the structure and placed in a horizontal and vertical pattern near each face. No special reinforcement is provided near penetrations less than 12 inches in diameter. For larger penetrations reinforcement steel is either terminated at the opening and supplemental steel added, or the reinforcement is continuous and is bent to curve around the opening. Dome reinforcement is arranged in a radial and circumferential pattern, where the radial bars are continued from the vertical cylinder bars. Additional reinforcement schemes may be used. The slab reinforcement pattern consists of concentric circular bars combined with radial bars, at the top and bottom face, arranged to permit uniform spacing of the vertical wall rebar that extends into the mat.

For the steel-lined reinforced concrete containment and the steel-lined reinforced concrete with post-tensioning containment the wall reinforcement consists of horizontal bars located near both faces of the wall and rows of vertical bars placed near each face, supplemented by inclined bars. The dome reinforcement consists of meridional and horizontal hoop bars. In the basemat the bottom reinforcement is a rectangular grid pattern, while the top rebar consists of concentric circular bars combined with radial bars that are arranged to permit a uniform spacing of the vertical wall bars that extend into the basemat.

A tendon may consist of a number of 1/4 or ½ inch diameter steel wires or a single round steel bar. The tendons are anchored to a bearing plate. The tendons are installed in metal sheaths that form ducts through the concrete between anchorage points. The sheaths, after venting and draining, are filled with a grease used for permanent tendon corrosion protection. The grease is a modified, refined petroleum oil based product that is pumped into the sheathing after tendon installation.

3.1.2.5 Steel Liner

The steel liner is made up of a cylinder capped with a hemispherical dome and attached at the bottom to a mat liner completely lining the concrete vessel. The wall, dome and mat liner plate thicknesses are 3/8, ½ and 1/4 inches, respectively. The liner plate is a continuously welded steel membrane supported and anchored to the inside of the containment and designed to maintain a leaktight integrity. The liner pressure boundary includes embedments, insert plates, and penetrations. Leak chase channels are installed over liner seam welds which include mat, wall, dome and penetrations. The basemat liner plate is anchored to the foundation slab using two methods. Either the liner is welded to a ring plate that is anchored in the base slab and then welded to the skirt ring of the wall liner; or the liner plate is directly anchored with embedded stiffeners and anchors. A reinforced 2 foot thick concrete slab is poured over the liner plate and may or may not be anchored through the floor liner to the basemat.

3.1.2.6 Embedments

Embedments have been identified as a containment structure component by WOG. However, no specific description of embedments is provided in Subsection 2.3 of the GTR. A definition and description of embedments needs to be added to this subsection. This issue is designated as Open Item No. 3.

3.1.2.7 Electrical Penetrations

The scope of this evaluation includes the metallic components of the typical electrical penetrations that are part of the containment pressure boundary. Seal materials such as epoxy and silicone rubber are not included in the scope of the GTR. There are generally four types of electrical cable penetrations required in the containment depending on the type of cable involved:

- Medium voltage power, 4160 V or 6.9 kV
- Low voltage power, control and instrumentation, 600 V and lower
- Thermocouple leads
- Special instrumentation coaxial and triaxial circuits

Typically, medium and low voltage electrical penetrations consist of carbon steel pipe canisters with stainless steel header plates bolted to one or both of the ends. Identical, hermetically sealed ceramic multi-pin connectors are mechanically connected to the header plate(s) for all conductors rated less than 600 volts. High voltage conductors use single-conductor, hermetically sealed ceramic bushings, also mechanically connected to both header plates. A flange on each canister is welded to the penetration sleeve. The electrical penetration assembly permits pressure and leakage tests to be performed at the shop and after installation in the containment. A tap, convenient to the exterior of the containment, is provided for pressurizing the canister. The terminations of the conductors to the connectors inside the canisters are potted to protect against moisture. A plug is provided to permit purging with dry nitrogen.

Penetration assemblies for Westinghouse containments, were manufactured and supplied by four major vendors:

- Westinghouse Electric Corporation
- Conax Corporation
- D.G. O'Brien
- Bunker Ramo

Of these four, D.G. O'Brien and Bunker Ramo are no longer in business. Westinghouse and Conax are the two vendors that currently supply penetration assemblies and provide the necessary services for penetrations supplied by D.G. O'Brien or Bunker Ramo.

During an EQ inspection, the U.S. NRC identified a deficiency in the Bunker Ramo low voltage penetration qualification testing method. The deficiency is detailed in Information Notice 88-29. The notice states that the insulation resistance (IR) measurements performed during the accident simulation testing were not frequent enough to evaluate the impact of the IR values on the accuracy of the connected instrument circuits.

For the steel containment, the nozzle that is welded into the steel containment plate and the steel header plate for the electrical terminals are included in the scope. The wiring, sealing compound, fixtures to hold the sealing compound, and seal welds, which connect the fixtures to the header plate, are not included in the scope of this evaluation.

3.1.2.8 Penetrations for Gas and Fluid Systems that Include Isolation Valves

The mechanical penetrations, or piping penetrations, are provided for fluid carrying pipes and for air purge ventilating piping. In certain steel containment designs, vacuum breaker penetrations are provided through the steel containment and into the annulus between the containment and the shield building. Various systems are used to classify the piping penetrations for design purposes. In general, the classification is based on whether thermal movement of the process line is expected. The piping penetrations are typically classified into high or moderate temperature service. A third type of penetration is a multiple pipe penetration, where more than one line passes through the penetration. The multiple pipe penetrations accommodate small-diameter tubes for sampling lines. Socket weld couplings may be welded into the penetration header plate to mate with the seamless tubing.

In steel containments, the penetration assemblies span from the inner wall of the containment shell to the outer wall of the concrete shield building. They provide containment leak tightness and support for the penetrating pipe, and also accommodate thermal movement of the pipe. The closure between the steel containment and the process pipe is provided either by a bellows assembly or by direct welding of the pipe to the steel shell. For high temperature penetrations, a bellows is used to permit relative movement between the pipe and the containment shell, while the pipe support loads are carried by the more robust concrete shield building wall.

Penetration closures for smaller lines and cold lines may be provided by direct welding of the pipe to the steel containment. In this case, a bellows assembly is provided at the concrete shield building to accommodate the differential building movements. Bellows assemblies may also be used for the closure to the steel containment for the cold and smaller lines. However, direct welding is typically provided, where feasible, because the rigid-plate-to-pipe weld provides a more reliable seal at the primary containment boundary than bellows details.

Mechanical (piping) penetrations for concrete containments may be classified as single barrier or double barrier penetrations. The single barrier penetration provides a single rigid closure between the containment liner and the process pipe. It may consist of a flued head, a standard pipe cap, or a plate with a segment of sleeve pipe. The double barrier penetration provides a closure barrier at both faces of the containment wall and is composed of a sleeve and multiple pipe caps. The penetrations in concrete containments provide a support point for the piping system. The robust cylindrical wall of concrete containments typically serves as an anchor point for hypothetical pipe rupture loadings from high energy lines such as main steam and feedwater.

For the high-temperature penetrations, design features are provided to limit the temperature in the concrete adjacent to the penetration (local area) to less than 200°F for normal operation, 350°F for short-term unusual conditions, and 650°F for jets due to postulated pipe rupture conditions. Systems to provide the necessary cooling range from active forced air or water

cooling systems within the penetration sleeve to passive systems consisting of insulation and cooling fins. In some designs, thermocouples are placed in the concrete local to the penetration to monitor temperature in the concrete.

In general, piping penetration nozzles are designed and fabricated to conform to the ASME Code requirements in effect when the plant was built. For older plants, ASME Code Section III, Class B was used. For later plants, ASME Code Section III, Division 1, Subsection NE (Class MC) was used. Class MC penetration assemblies were code-stamped with the NPT stamp for nuclear parts.

In concrete containments, penetrations and anchorages to the concrete shell are designed for forces and moments resulting from operating conditions or postulated pipe rupture. External guides, stops, or increased pipe wall thickness are provided to limit stresses on the penetration and on the adjacent liner plate.

Penetration reinforcing plates and the welds of pipe sleeves to them are shop stress-relieved as a unit in accordance with the ASME Code requirements to ensure a minimum of field welding at the penetrations. Full-penetration butt welds are used to connect the sleeve and the attachment hardware around the process piping. The closure between the sleeve and the process piping consists of flued heads, plates, or drilled pipe caps.

Spare penetrations for concrete containments generally consist of a sleeve with welded end cap closures. The arrangement and details are basically the same for steel containments except that the sleeve is welded to the steel containment plate rather than to the concrete containment liner. For some plants, spare penetrations that are used only during outages may incorporate bolted blind flanges with flexible seals for ease of removal and replacement.

The entire mechanical penetration assembly (excluding the process piping within the penetration) and the welds to the process pipe are included in the scope of the GTR. Flued heads, containment vessel nozzles, bellows assemblies and all other items that perform a containment function are in the scope of the GTR. For smaller piping and cold lines, if a bellows is not used, the closure weld between the containment steel and the process pipe is in the scope of the GTR. Piping insulation is not included, although in some cases the piping insulation also serves to maintain the temperature of the concrete adjacent to the penetration sleeve within permissible limits.

3.1.2.9 Fuel Transfer Tube

The fuel transfer tube penetrates the reactor building and links the refueling canal in the reactor building with the fuel transfer canal in the fuel handling building. For a concrete containment it consists of the transfer tube and a sleeve welded integrally into the containment liner. The closure between the transfer tube and the sleeve that is welded integrally into the containment liner typically consists of a circular plate shop-welded to the tube and a short segment of pipe to mate with the sleeve. All welds in the closure between the tube and liner are full-penetration welds except for the flange adaptor ring welded to the end of the tube. The blind flange adaptor ring is connected to the transfer tube pipe by two separate continuous partial penetration welds. For older vintage plants, the bimetallic transition weld between the stainless steel transfer tube

and the carbon steel plate or pipe segment was shop-welded for better control of quality. For later plants, with improved welding materials and techniques, the transition weld was made in either the shop or field. For double-barrier designs, a welded canopy is placed over the connecting hardware between the tube and liner.

The fuel transfer tube penetration arrangement through a steel containment is basically the same as for the concrete containment with closure being made to the steel containment vessel rather than the containment liner. In some designs, the closure to the steel containment vessel may consist of a flexible bellows assembly.

During normal operation, the fuel transfer tube penetration is closed and serves as part of the containment pressure boundary. The closure at the pressure boundary end inside containment typically consists of a blind flange with double seals. The closure on the fuel handling building end outside the containment generally consists of a gate valve supported from the end of the transfer tube.

The scope of the GTR includes all parts of the fuel transfer tube that serve as part of the containment system such as the blind flange with double seals, the closure detail between the containment liner (or steel containment) and the transfer tube, and the transfer tube itself. In addition, the gate valve at the outboard end of the transfer tube is included. The bellows assembly connections between the stainless steel canal liner and the transfer tube are not included in the scope of the GTR.

3.1.2.10 Equipment and Personnel Hatches that Include Seals

Typically, two personnel access airlocks are provided at two different floor levels of the containment for normal and emergency ingress and egress.

A typical personnel airlock through a concrete containment consists of a cylindrical barrel section with leaktight doors at each end. The airlock is supported by the concrete containment wall. Airlocks through steel containments are basically the same with the exception that they may be supported by the concrete shield building rather than the steel containment. The leaktight closure between the steel containment and the airlock barrel is by a flexible connection such as a bellows assembly.

For some containments, two different size airlocks are used. The larger airlock is typically used as the primary access and the smaller airlock for emergency access. In some containments, one of the personnel airlocks penetrates the dished cover of the equipment hatch.

The airlocks are designed to withstand all containment design conditions with either or both doors closed and locked. The doors open toward the center of the containment and are thus sealed under containment pressure. The airlock may be pressurized to demonstrate leaktightness without pressurizing the containment. Each airlock is pneumatically tested in the shop for pressure and leakage. Quick-acting equalizing valves connect the personnel lock with the interior and exterior of the containment vessel for the purpose of equalizing pressure in the two systems when entering or leaving the containment.

The two personnel airlock doors are interlocked to prevent both being opened simultaneously and to ensure that containment is always maintained by one door being completely closed before the other door can be opened. The interlocking system has the capability to be bypassed allowing the doors to be left open during plant cold shutdown. In most cases, operation of the lock is manual, without power assist.

Each airlock door is provided with flexible seals. The arrangement and type of seal varies from plant to plant. In most cases, either double gasket seals that have provisions for local leakage testing between the seals or inflatable seals are provided.

Airlocks are designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III. In most cases, the airlocks are code-stamped in accordance with the requirements for Code Class MC. For plants built prior to the ASME incorporation of Subsection NE for Class MC, the portions of the airlocks not backed by concrete typically conformed to ASME Section III for Class B requirements, but were not code-stamped.

The scope of the GTR includes each entire airlock including the equalizing valves, handwheel shaft seals, and door seals. Electrical penetrations of the airlock bulkheads are not included.

The typical equipment hatch consists of a double-gasketed flange, a bolted dished head and a barrel liner sleeve through the containment wall. Supplementary equipment such as a monorail and hoist system are usually associated with the equipment hatch to permit efficient usage of the large opening. In later-generation plants, the hatch diameter is typically sized to permit removal and replacement of a steam generator. In the older plants, the diameter is adequate to support most maintenance activities but does not permit passage of an intact steam generator. In some cases, a smaller diameter emergency access personnel airlock is incorporated into the equipment hatch head, providing a dual function design.

The hatch design and fabrication conforms to the applicable ASME Code requirements in effect for the particular vintage plant. The hatch may be ASME code-stamped for the later vintage plants but is not stamped for the early plants. Typically, the head is convex inward toward the design pressure. The space between the double gaskets on the hatch flange can be pressurized for local leakage checking. The diameter ranges from 14 to 24 feet. The smaller diameter hatches in the earlier plants were designed to accommodate replacement of the reactor vessel O-ring seal. The larger diameter hatches provided in the later plants were sized for steam generator replacement.

Equipment hatch designs are basically the same for both concrete and steel containments. The scope of the GTR includes the entire hatch including the flexible seals between the hatch barrel flange and the head flange.

3.2 Effects of Aging

Section 2.7 of the GTR discuses the potential aging mechanisms and associated aging effects applicable to PWR containments. Tables 2-16, 2-17, and 2-18 of the GTR summarize the potential aging mechanisms and aging effects considered in the "Aging Management Review,"which is documented in GTR Subsections 3.2.1 thru 3.2.40. GTR Section 3.4 "Aging

Effect Evaluation Summary" lists the containment component/aging mechanism combinations that WOG has identified as requiring an aging management program for License Renewal.

The following paragraphs provide the evaluation of the GTR Sections delineated above.

3.2.1 Potential Aging Mechanisms and Associated Aging Effects

GTR Table 2-16 lists the age related degradation mechanisms applicable to PWR containment structures correlated in accordance with the material and component affected. Mechanisms that affect concrete, reinforcing steel, prestressing system components and steel components, in both concrete and steel containments are listed.

BNL considers the aging mechanism list provided in GTR Table 2-16 to be a reasonable summary of the aging mechanisms that potentially affect PWR containment structures. However, two additional potential aging mechanisms should have been listed: erosion of porous concrete subfoundation layers, and grease leakage in concrete containments. These aging mechanisms are discussed in Section 3.3 of this TER. It is also noted that wear and seal degradation for penetrations, hatches and airlocks are not included in Table 2-16; they are, however, included in Table 2-17.

With inclusion of the items noted, the staff considers the degradation mechanism list provided in GTR Table 2-16 to be an acceptable summary of the aging mechanisms that can potentially affect PWR containment structures.
GTR Table 2-17 provides a summary of the aging evaluations performed by WOG for each containment structure component/aging mechanism combination described in subsections 3.2.1 thru 3.2.40 of the GTR. For each containment structure component, the aging mechanisms that affect it and the associated aging effects are listed. Also provided is an identification of the aging management program that will be used to manage the aging effects and references to subsections of the GTR where the aging evaluation and TLAAs, if applicable, are discussed. As noted above, erosion of porous concrete subfoundation layers and grease leakage in concrete containments are not identified. See Section 3.3 of this TER.

GTR Table 2-18 provides a numbered listing of primary aging effects which are referenced by Table 2-17and two secondary effects, identified as A and B, which are also referenced in Table 2-17. The aging evaluations are described in GTR Subsections 3.2.1 thru 3.2.40, for all containment structure component/aging mechanism combinations which are identified in Table 2-17.

The aging effects considered for concrete include cracking, scaling, spalling, increased porosity and permeability, reduction in strength and modulus of elasticity. Other effects considered include loss of protective covering and chemistry (i.e., lowering concrete pH and degrading protective oxide films) on reinforcing steel, resulting in the corrosion of the embedded steel.

The aging effects resulting from corrosion of steel components in concrete include increase in volume through rust by-products, cracking of the surrounding concrete, and reduction in cross-sectional area or thickness. Elevated temperatures result in a reduction in strength and modulus of elasticity for steel, while irradiation embrittlement results in a increase in yield strength,

decrease in the ultimate tensile ductility, and increase in the ductile-to-brittle transition temperature. A secondary effect of elevated temperature is loss of bond strength between embedded steel and concrete. Fatigue results in cracking in steel components and surrounding concrete.

The aging effects considered for corrosion in post-tensioning systems include decreases in cross-sectional area, reduction in prestress force, breakage of wires or strands, and leakage of corrosion inhibiting grease.

Aging effects associated with aging mechanisms that affect penetrations include loss of material, cracking of steel components, fatigue-induced cracking, and loss of seal or pressure retention.

The effect of foundation settlement is an increase of stress in piping or other systems interconnecting adjacent buildings, due to the additional movement between anchor points.

In RAI No. 3 (NRC letter dated February 19,1998), the staff requested the WOG to discuss the operating experience of the containment structure and its components relating to the effects of aging, including any applicable generic communications. In response, the WOG indicated that Sections 2 and 7 of the GTR provide a detailed description of the operating experience. In particular Section 2.6.3 describes survey data of maintenance and inspection history from the International Atomic Energy Agency, Section 2.6.4 provides summaries of observed degradations associated with structures within the scope of the GTR, and in Section 7 (an appendix) a summary of age-related degradation observed in concrete containments is given for plants at four sites.

WOG further stated that license renewal applicants will discuss the results of past containment examinations, which will demonstrate the operating experience at their specific plant. Therefore, the submittal of plant-specific operating experience is designated as Renewal Applicant Action Item No. 7.

3.2.2 Aging Mechanisms Requiring an Aging Management Program

Table 2-17 provides a listing of the aging mechanisms and identifies the WOG aging management program (AMP), if any, that will be used to manage each mechanism. In Table 2-17, the aging mechanisms requiring an aging management program and the applicable WOG aging management programs are

Component	Aging Mechanism	Aging Management Program
Concrete	Freeze thaw (3.2.1)	AMP-5.1 and AMP-5.2
	Aggressive chemical attack (3.2.7)	AMP-5.3 and AMP 5.4
	Fatigue at penetration anchors (3.2.9)	AMP-5.5
Reinforcing Steel	Corrosion (3.2.10)	AMP-5.3 and AMP-5.4

Component	Aging Mechanism	Aging Management Program
Liner	Corrosion (3.2.14)	AMP-5.5
	Coating degradation (3.2.15)	AMP-5.5
Post-Tensioning Systems	Corrosion (including microbial) (3.2.17)	AMP-5.6
	Concrete degradation (3.2.17)	AMP-5.6
	Prestress force losses (3.2.20)	AMP-5.6
	Stress corrosion cracking (3.2.21)	AMP-5.6
Electrical Penetrations	Bellows TGSCC (3.2.24)	AMP-5.5
Mechanical Penetrations	Bellows fatigue (3.2.25)	AMP-5.5
	Fatigue (3.2.9, 3.2.26)	AMP-5.5
	Embrittlement of gaskets (3.2.27)	AMP-5.5
	Corrosion and SCC (3.2.28)	AMP-5.5
Fuel transfer Tube Penetration	Mechanical wear (3.2.29)	AMP-5.5
	Embrittlement of gaskets (3.2.30)	AMP-5.5
	Corrosion and SCC (3.2.28, 3.2.31)	AMP-5.5
Airlocks and Hatches	Mechanical wear (3.2.32)	AMP-5.5
	Gasket embrittlement(3.2.27, 3.2.34)	AMP-5.5
	Loss of pressure retention (3.2.35)	AMP-5.5
Foundations	Settlement (3.2.37)	AMP-5.7
Free-standing Steel containment	Fatigue (3.2.39)	AMP-5.5
	Corrosion (3.2.40)	AMP-5.5

Numbers in parentheses refer to the GTR subsection number which contains the corresponding aging evaluation.

3.2.3 Aging Mechanisms Not Requiring an Aging Management Program

WOG concluded that many aging mechanisms did not require an aging management program. These items are listed in Table 2-17 and are designated NR (not required) in the column titled Aging Management Program. Also listed in the table are the GTR report subsections in which an assessment of the listed aging effect/mechanism is provided.

The aging mechanisms for which no aging management program is considered necessary, grouped by containment structure component, are listed below. Numbers in parentheses refer to the GTR subsection number which contains the corresponding aging evaluation.

Component	Aging Mechanism
Concrete	Leaching (3.2.2), alkali aggregate reaction (3.2.3), neutron irradiation embrittlement (3.2.4), interaction with aluminum (3.2.5), thermal aging embrittlement (3.2.6), bond strength reduction - direct current (3.2.8)
Reinforcing Steel	Elevated temperature effects (3.2.11), irradiation embrittlement (3.2.12), fatigue (3.2.13)
	Liner Elevated temperature effects (3.2.11), irradiation embrittlement (3.2.12), fatigue (3.2.13), fatigue at attachments and discontinuities (3.2.16)
Post-tensioning	Elevated temperature effects (3.2.18), irradiation embrittlement (3.2.19) Systems
Steel Embedments	Corrosion (3.2.22)
Electrical Penetrations	Material compatibility (3.2.23)
Airlocks and Hatches	Fatigue (3.2.16, 3.2.33), elevated temperature effects (3.2.36)
Free-standing Steel Containment	Strain aging (3.2.38)

The staff raised several questions regarding the lack of aging management programs for various aging mechanisms. In Section 3.2 of the GTR, the aging effects in concrete due to leaching of calcium hydroxide and alkali aggregate reaction are identified as not requiring aging management because their effects have been determined not to be detrimental. In RAI No. 1 (NRC letter dated February 19,1998), the staff requested the WOG to address applicable aging effects such as those described in the working draft SRP-LR and to propose appropriate aging management programs for renewal, or provide detailed justifications for excluding any applicable aging effects.

The GTR and response to RAI No. 1 state that because plant construction used specific national codes, standards, and guides such as ACI 201.2R-77 and ASTM C295 or C227, concrete aging mechanisms such as leaching of calcium hydroxide and reaction with aggregates would be precluded. Since it is not evident that all Westinghouse PWR concrete containments were designed and constructed to all of these codes, standards and guides, verification of this issue is designated as Renewal Applicant Action Item No. 8.

Also in response to RAI No. 1, WOG replied that the GTR addresses Westinghouse PWR containment structures generically. Some isolated cases may exist where a degradation mechanism and effect such as leaching of calcium hydroxide may develop. These will be plant-specific issues which will be discussed within a license renewal application. BNL acknowledges WOG's evaluation that some of the aging mechanisms have not shown detrimental effects. Nevertheless, these aging mechanisms may produce detrimental effects in the future. Therefore, current mandated inspections in accordance with 10CFR50.55a/ASME Code Section XI, Subsection IWL, Examination Category L-A should be generically credited for aging management of concrete for license renewal. This issue is designated as Open Item No. 4.

Section 3.2.6 of the GTR concluded that there is no need for the identification of aging management options for aging effects caused by thermal embrittlement in concrete. In RAI No. 10 (NRC letter dated February 19,1998), the staff requested the WOG to justify this conclusion for the main steam line penetrations through containment where temperatures could reach 500^0 F, substantially higher than the 200^0 F limit in the ACI Code. In response, the WOG indicated that local temperatures beyond 200^0 F are not permitted because of ACI 349 code limits. For hot piping penetrations, cooling coils and/or insulation are needed to maintain local temperatures within the 200^0 F code limit.

This response does not fully address the concern because (1) many plants were not designed to ACI 349 and (2) it is not clear from the response whether all hot penetrations that exceed the 200^0 F code limit at all plants have cooling coils and/or insulation that maintain local temperatures within the 200^0 F code limit. Satisfaction of the 200^0 F local limit for concrete around hot penetrations should be addressed on a case-by-case basis in the applicant's License Renewal submittal. If an applicant cannot verify this, then a plant-specific aging management program would be needed to address elevated concrete temperature effects. This issue is designated as Renewal Applicant Action Item No. 9.

3.3 Aging Management Programs

Section 4.1 of the GTR states that the aging effects for the systems, structures, and components within the scope of the GTR are managed using current licensing basis (CLB) inspection and test programs based on ASME Code Section XI, Subsections IWE and IWL, and American Concrete Institute (ACI) codes. These programs include current plant maintenance, inspection, and testing programs that follow the 1992 Code Edition and Addenda of ASME Section XI, Subsections IWE and IWL, in compliance with 10 CFR 50.55a. WOG noted that in U.S. NRC SECY-96-080, the 1992 ASME Code, including the 1992 Addenda, of Section XI, Subsections IWE and IWL, was recognized by the NRC as effective in managing the aging effects associated with containment structures.

WOG recommended that a utility incorporate into its inservice inspection programs, for the extended period of operation, aging management programs (AMPs) that are based on the 1992 Code Edition, with 1992 Addenda, of ASME Section XI, Subsections IWE and IWL. Further, WOG also recommended that the modifications specified in SECY-96-080 (introduced in SECY-93-328, the proposed rule) to address NRC concerns related to tendon examinations and inaccessible areas should also be included in the AMPs

WOG identified seven AMPs that would serve as the basis for aging management at utility plants. GTR Table 4-2 provides a listing of the AMPs and the components, aging mechanisms and aging effects they address. The programs are: AMP-5.1 and AMP-5.2 to address freeze thaw in the concrete containment and shield buildings respectively, AMP-5.3 and AMP-5.4 to address aggressive chemical attack - corrosion in the concrete containment and shield buildings respectively, AMP-5.5 to address degradation of metal components such as the liner, containment shell, penetrations and airlocks/hatches, AMP-5.6 to address degradation of Post-Tensioning systems and AMP-5.7 to address foundation settlement. Descriptions of the AMPs are presented in GTR Sections 4.1.6 thru 4.1.11, with comprehensive summaries of each plan provided in GTR Tables 4-9 thru 4-15. Each AMP is discussed below.

The staff raised a number of questions that relate to all the AMPs. In RAI No. 4 (NRC letter dated February 19,1998), the staff requested that when specific codes, standards, or related documents are used or described in the evaluation of aging management for containment, the GTR should include the full title, edition, and the year of publication. This was not done for some of the documents such as ACI publications in Tables 4-10 and Table 4-12. In response the WOG stated that sufficient information is given to permit identification of the intended document. The list of references or bibliography provides the date and title. For some cases it may not be appropriate to provide the applicable year of publication because the GTR applies to all Westinghouse PWR plants which have different licensing bases. Where it is not possible to indicate the applicable year of publication, the WOG stated that it will be necessary for the utility to identify this information in their license renewal application. This will be addressed in the plant-specific license renewal application and is designated as Renewal Applicant Action Item No. 10.

Section 4.1.4 of the GTR addresses nondestructive examination/sampling inspection technology. In RAI No. 24 (NRC letter dated February 19,1998), the staff requested WOG to address the implementation of Appendix VII and Appendix VIII of ASME Code Section XI when conducting ultrasonic examinations of containment structures. In response, WOG stated that when implementing an aging management program that references Subsection IWE, and it is necessary to utilize augmented ASME Section XI NDE inspection methods, the training qualifications and certification of ultrasonic examination personnel will meet Appendix VII and Appendix VIII. BNL finds WOG's response acceptable, however, this should be incorporated in the GTR. This issue is designated as Confirmatory Item No. 2.

In RAI Nos. 2, 22, and 23 (NRC letter dated February 19,1998), the staff requested WOG to discuss aging management for inaccessible areas of containment structures. In response to these RAIs, WOG stated that potential degradation of inaccessible areas for these aging mechanisms are addressed in Section 4.0 of the GTR. The aging management programs defined in the report meet the inaccessible area requirements based on 10 CFR 50.55a, which specifies evaluation of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation to such inaccessible areas.

WOG also stated that it recognized the working draft SRP-LR encourages the review, on a case-by-case basis, of inaccessible areas when conditions in accessible areas may not indicate degradation. WOG stated that this review would only be done if it is event-driven (e.g.,

occurrence of an accident). This is a generic concern which should be addressed in the GTR. This issue is designated as Open Item No.5.

3.3.1 AMP-5.1 & AMP-5.2 - Concrete Containment, Concrete Shield Building - Freeze-Thaw Degradation

The GTR indicates that two programs will be used to manage freeze-thaw in concrete, program AMP-5.1 for concrete containments and program AMP-5.2 for concrete shield buildings. The programs will be applied on a plant-specific basis for plants where this aging mechanism is recognized to occur. In program AMP-5.1, containment structures will be examined and maintained in accordance with Section XI, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants, Examination Category L-A, of the 1992 Code Edition with 1992 Addenda, of the ASME Code, and the guidelines for condition surveys provided in ACI 201.1R-68. In program AMP-5.2, shield building structures will be examined and maintained in accordance with the guidelines in ACI Codes and Standards including ACI-201.1R-68, ACI-207.3R-79 and ACI 224.1R-89 and ASTM Standard C823.

The two aging management programs (AMP-5.1 and 5.2) described above are acceptable to the staff to manage the aging effects resulting from freeze-thaw because they rely on appropriate, applicable codes and standards. However, because freeze-thaw has been identified as a plant-specific aging mechanism which is potentially significant only in colder geographic regions, the applicability of this aging mechanism and adherence to AMP-5.1 and AMP-5.2 needs to be specified by each license renewal applicant. Therefore, this issue is designated as Renewal Applicant Action Item No. 11.

3.3.2 AMP-5.3 & AMP-5.4 - Concrete Containment, Concrete Shield Building and Basemat - Aggressive Chemical Attack - Corrosion

The GTR indicates that two programs will be used to manage aggressive chemical attack and corrosion in below grade concrete: program AMP-5.3 for concrete containment and program AMP-5.4 for concrete shield building and foundation mat. The programs will be applied on a plant-specific basis for plants where groundwater chemistry and interior leakage provide an environment conducive to degradation by these mechanisms. In program AMP-5.3, accessible areas will be examined in accordance with Section XI, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants, Examination Category L-A, of the 1992 Code Edition with 1992 Addenda, of the ASME Code. Groundwater quality and the identification of leakage on the inside of the containment would be monitored at each refueling outage.

If aggressive chemistry is indicated in the groundwater and/or soil, inspection of the concrete in the affected zone would be performed. Sample areas of exterior surfaces that are below the groundwater table would be visually inspected. Quoting from Section 4.1.7 of the GTR," If deterioration is found at the sample area, the acceptability of inaccessible areas is evaluated in accordance with the changes to 10 CFR 50.55a, as described in SECY-96-080. Concrete containments are evaluated using the revised rule para 50.55a (b)(2)(ix)(E), while steel liners and steel containments are evaluated using the revised rule para 50.55a (b)(2)(x)(A)." Compliance with 50.55a requires that the licensee evaluate the acceptability of inaccessible areas when

conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. Groundwater management (e.g. change water chemistry or redirect groundwater as necessary) is identified as an option when repairs are not feasible.

In program AMP-5.4 the concrete surfaces of the shield building and foundation mat will be examined in accordance with the guidelines in ACI Codes and Standards including ACI 201.1R-68, ACI 207.3R-79 and ACI 224.1R-89 and ASTM Standard ASTM C823. Monitoring will be performed to identify leakage of water in the shield building and, if chemistry is questionable, the quality of groundwater. The monitoring of leakage and water chemistry will be performed at every refueling outage while the inspections will be performed every five years. Corrective actions include: changing water chemistry, redirecting groundwater or repairs in accordance with ACI guidelines, as necessary.

The aging management programs AMP-5.3 and 5.4 described above are considered acceptable to manage the aging effects resulting from aggressive chemical attack- corrosion because they rely on appropriate, applicable standards. However, because the programs will be applied on a plant-specific basis for plants where groundwater chemistry and interior leakage provide an environment conducive to degradation by these mechanisms, the applicability of this aging mechanism and adherence to AMP-5.3 and AMP-5.4 needs to be specified by each license renewal applicant. This is designated as Renewal Applicant Action Item No. 12.

In RAI No. 19 (NRC letter dated February 19,1998), the staff requested additional information regarding the ground water chemistry monitoring program. Questions were raised relating to the effect of seasonal variation on the water chemistry and the technical basis for the water sample chemistry acceptance criteria. WOG provided the basis for the acceptance criteria. The staff has identified a concern regarding the effects of seasonal variation on the water chemistry. Since such information is plant-specific, this is designated as Renewal Applicant Action Item No. 13.

Because grease leakage in prestressed concrete containments has occurred at some plants, the staff requested, in RAI No. 28 (NRC letter dated February 19,1998), a discussion on how the aging effects of grease leakage into the concrete is being managed.

WOG responded that the examination and inspection of grease leakage, its significance, and its impact on the integrity of prestressed concrete containments would follow the ASME Section XI, Subsection IWL requirements. WOG stated that if there are ingredients within the grease that would cause degradation of the concrete, the utility should consider this as part of a concrete aggressive chemical attack mechanism, and manage the effect during the license renewal period following the aging management program AMP-5.3. WOG classified the potential effect of grease on the shear load capability of the concrete structure as a plant-specific issue.

BNL finds WOG's response acceptable, except for the potential effect on shear load capacity. Since WOG has identified this as a plant-specific issue, it is designated as Renewal Applicant Action Item No. 14.

3.3.3 AMP-5.5 Liner, Steel Containment Shell, Penetrations, Coatings and Airlocks and Hatches - Embrittlement and Loss of Pressure Retention, Mechanical Wear, Fatigue, Corrosion, SCC, and TGSCC

The GTR indicates that program AMP-5.5 will be used to manage potential aging effects for steel containment shells, steel liners, coatings, penetrations, airlocks and hatches due to corrosion, SCC, TGSCC, embrittlement and loss of pressure retention, and fatigue. In AMP-5.5, the components will be examined and maintained in accordance with ASME Code, Section XI, Subsection IWE "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants", 1992 Edition with 1992 Addenda, and the requirements of Appendix J Leak Rate Testing. IWE examination categories are E-A, E-B, E-C, E-D, E-F, E-G, and E-P from the specified code edition (1992 with 1992 addenda). In response to RAI No. 11, WOG acknowledged that incorrect examination categories (corresponding to the 1989 edition) are listed in Table 4-13. This is Confirmatory Item No. 4. The program will also include the identification and monitoring of leakage inside containment. For leakage, monitoring will be performed at each refueling outage and the corrective actions include the removal of standing water and the cleanup and restoration of affected surfaces.

In RAI No. 26 (NRC letter dated February 19,1998), the staff requested a discussion pertaining to the performance of examinations specified in Examination Category E-B for pressure retaining welds and Examination Category E-F for pressure retaining dissimilar metal welds of Subsection IWE of ASME Section XI Code (Reference 3) for license renewal.

WOG indicated that pressure retaining welds (Examination Category E-B) are visually inspected using the VT-1 examination method. Examination methods for suspect areas and augmented inspections, alternate examination methods, inspection frequency, acceptance criteria, corrective actions, and confirmation follow the IWE requirements as defined in Table 4-13 of the GTR for aging management AMP-5.5. Pressure retaining dissimilar welds (Examination Category E-F) are inspected using surface examination methods as defined in Subsection IWA-2220 of Section XI. Alternate examination methods, inspection frequency, acceptance criteria, corrective actions, and confirmation follow the IWE requirements as defined in Table 4-13 of the GTR for aging management AMP-5.5. Since the performance of examination for Categories E-B and E-F follows Subsection IWE of the Code, BNL considers this acceptable.

Stress corrosion cracking (SCC) is a concern for dissimilar metal welds, and in the case of stainless steel bellows assemblies, SCC may be significant if the material is not shielded from a corrosive environment. For the period of extended operation, examination Categories E-B and E-F and augmented VT-1 visual examination of bellows assemblies and dissimilar metal welds are warranted. Each license renewal applicant needs to define its plant-specific program to address this concern. This is designated Renewal Applicant Action Item No. 15.

Coating degradation is listed in the executive summary of the GTR as an aging mechanism that requires an aging management program. Although it is not listed as a degradation mechanism in Table 2-16, it is listed as an aging mechanism affecting the liner and managed by AMP-5.5, in Table 2-17. In the list of aging effects, Table 2-18, it is listed and characterized as a secondary effect. The aging evaluation for this mechanism is provided in GTR Subsection 3.2.15.

In Section 3.2.15 of the GTR, WOG concluded that there is no need for an aging management program to address coating degradation. Instead, the ISI programs are relied on to detect the aging effects of coating degradation and to provide criteria for the acceptance of repairs and subsequent inspections. It was further concluded that the loss of coating does not directly result in an aging effect although corrosion could result from the loss of the coating. Referring to the description of AMP-5.5, the staff notes that no unique inspection or evaluation for coatings is referenced. No specific statement regarding the repair of coatings is provided, although repairs of damaged areas would follow the requirements and recommendations of the ASME IWA and IWE programs.

BNL concludes that AMP-5.5 does not include any provision to monitor and maintain the condition of protective coatings on steel elements of containment. Most, if not all, licensees have existing coatings programs which can be credited for license renewal. Monitoring and maintenance of protective coatings should be identified as an AMP for steel elements of containment. Consequently, each applicant should discuss its plant-specific coatings program and specify whether it is being credited for managing aging of containment steel elements. This is designated Renewal Applicant Action Item No. 16.

3.3.4 AMP-5.6 Containment Post-Tensioning System Degradation - SCC, Corrosion, Loss of Prestress Loading

The GTR indicates that program AMP-5.6 will be used to manage potential degradation of containment post-tensioning systems. The program will be applied on a plant-specific basis to those plants that include post-tensioning systems in their containment design. In the program the tendon systems will be examined, tested, and maintained in accordance with Section XI, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants, Examination Category L-B, Unbonded Post-Tensioning System, of the 1992 Code Edition with 1992 Addenda of the ASME Code.

In addition to the above, Section 4.1.10 of the GTR recommends that the utility inspection program also include four specific recommendations for tendon examination addressed in Regulatory Guide 1.35, Rev. 3 and an evaluation of the potential degradation of adjacent inaccessible areas. Since these requirements are also specified in 10 CFR 50.55a(b)2(ix), compliance with these requirements must be addressed as part of the aging management program in each plant-specific license renewal application. This issue is designated as License Renewal Applicant Specific Action Item No.17.

A concern was raised in RAI No. 27 (NRC letter dated February 19,1998) regarding the effect of elevated temperature on the prestressing force of tendons. Elevated temperature can develop due to sun exposure or proximity to hot penetrations. The concern was raised because the selection of a relatively few sample tendons may not include tendons that are affected by the elevated temperatures.

The staff has subsequently concluded in its resolution of this generic license renewal issue that the effects of elevated temperature on tendon prestress losses are adequately managed by the existing tendon surveillance and testing programs. Therefore, this issue is no longer significant.

3.3.5 AMP-5.7 Foundation - Settlement

The GTR indicates that program AMP-5.7 will be used to monitor the potential differential settlement of concrete foundations on soil. This program will be applied on a plant-specific basis to those plants recognized to be susceptible to settlement due to: the soil groundwater characteristic on which the plant is founded, soft soil conditions, or the use of piles in the foundation. In the program, a baseline inspection would be made to document settlement and building gap measurements and building misalignments. Thereafter, an evaluation at a frequency of at least once every five years is recommended. The inspections are to be performed by a qualified engineer and appropriate actions are to be taken to mitigate any conditions detrimental to continuous plant operation. The GTR recognizes that for the plants susceptible to settlement, programs to monitor changes in the groundwater table and to detect potentially significant settlement are included in the CLB.

In RAI No. 21 (NRC letter dated February 19,1998), the staff requested a description of how the settlement monitoring program ensures that differential settlement of the containment basemat does not exceed the design criteria for the containment structure and its base mat, including sites that may be experiencing significant changes in ground water conditions. In addition, a description of how the program satisfies the elements of an aging management program as defined in the working draft SRP-LR was requested.

In its response, WOG indicated that most of the settlement occurs within the first 5 to 6 years of operation. Only those plants with significant long-term settlement issues will be affected by this age-related degradation mechanism. The settlement monitoring program for those plants susceptible to settlement is not within the scope of the GTR. Details of the settlement monitoring program would be provided in the license renewal application. Since this will be addressed on a plant-specific basis, this issue is designated as Renewal Applicant Action Item No. 18.

In RAI No. 25, the staff indicated that if sub-foundation layers of porous concrete are used in the construction of a containment concrete basemat, then the management of aging effects due to erosion of cement should be discussed. WOG responded that this type of aging effect has not been addressed in the report because this type of construction is considered to be limited. If a plant uses porous concrete for sub-foundation layers beneath the basemat of a containment the utility would have to address this construction and aging effect in its plant-specific license renewal application. Therefore, this issue is designated as Renewal Applicant Action Item No. 19.

As stated in Section 2.3 above, the GTR listed only six (6) attributes to form the basis for each aging management program. However, the "Draft Standard Review Plan for the Review of License Renewal applications for Nuclear Power Plans", dated April 21, 2000, identifies ten (10) elements (attributes), as appropriate, for an acceptable AMP. The GTR predates the Draft standard Review Plan for the review of Licencing Renewal applications for Nuclear Power Plans, and states in Section 4.0 that the report only presents program attributes for the AMPs, and that plant-specific details of the AMPs will be developed during the preparation of license renewal applications. Therefore, applicants for license renewal will be responsible for developing and

describing the plant-specific AMPs with each AMP consisting of ten (10) elements. This issue is designated as Renewal Applicant Action Item No. 20.

3.4 Time-Limited Aging Analyses

Time-limited aging analyses (TLAAs) are defined in 10 CFR 54.3 as those licensee calculations and analyses that:

(1) Involve systems, structures, and components within the scope of license renewal, as stated in 10 CFR 54.4(a);

(2) Consider the effects of aging;

(3) Involve time-limited assumptions defined by the current operating term, for example, 40 yrs.;

(4) Were determined to be relevant in making a safety determination;

(5) Involve conclusions or provide the bases for conclusions related to the capability of the system, structure, or component to perform its intended functions, as stated in 10 CFR 54.4(b); and

(6) Contained or incorporated by reference in the current licensing basis.

10 CFR 54.21(c)(1) requires the applicant to demonstrate that:

(i) The analyses remain valid for the period of extended operation;

(ii) The analyses have been projected to the end of the period of extended operation; or

(iii) The effects of aging on the intended functions(s) will be adequately managed for the period of extended operation.

The TLAAs evaluated in the GTR for containment structures are

- Loss of prestress force loads in prestressing systems
- Fatigue of penetration anchors
- Fatigue of penetration bellows
- Fatigue of mechanical penetrations

The results of the evaluations are presented in Tables 3-3 and 3-4 of the GTR.

Loss of prestress forces in tendons require that the analyses be updated for the period of extended operation. For mechanical penetrations, WOG states that a utility may need to perform a fatigue analysis to show that an existing analysis remains valid for the period of extended

operation. As an alternate to performing analyses, WOG states that a utility may choose to adequately manage the effects of aging using the ASME Section XI surveillance and testing programs. For penetration anchors and penetration bellows, WOG states that the latter approach is applicable.

In RAI No. 12 (NRC letter dated February 19,1998), the staff requested the WOG to specify the specific structural components designed for fatigue loadings and whether fatigue is a TLAA for license renewal. The RAI response indicated that the following components were evaluated in Section 3.2 of the GTR for fatigue:

Fatigue at Penetration Anchors
Fatigue - Reinforcing Steel
Fatigue at Attachments and Discontinuities - Liner, Airlocks, and Hatches
Fatigue - Mechanical Penetration Bellows
Fatigue - Mechanical (Piping) Penetrations
Fatigue - Airlock and Hatches
Fatigue - Free Standing Steel Containment

In Section 3.3 of the GTR, fatigue is identified as a TLAA for the following three components:

Concrete Containment Penetration Anchors
Mechanical Penetration Bellows
Mechanical Penetrations associated with piping

BNL finds response to RAI No.12 acceptable. However, additional questions were raised regarding the evaluation of fatigue in RAI Nos. 15, 16, 17, and 18. After review of the responses to these RAIs, three (3) renewal applicant action items are identified:

- Two options are presented in the GTR for addressing TLAAs related to fatigue of steel components: update an existing design fatigue calculation to account for the additional years of operation, or utilize ASME Code Section XI surveillance and testing programs. Per 10 CFR 54.21(c)(1)(i) and 54.21(c)(1)(ii), the staff finds the first option acceptable. The second option (corresponding to 10 CFR 54.21(c)(1)(iii)) allows the component to be included in an ASME Code Section XI ISI program. This option would allow the CLB fatigue Section III cumulative usage factor (CUF) to be exceeded during the period of extended operation. The staff has not endorsed this option on a generic basis at this time. An applicant wishing to pursue this option would have to request staff review and approval on a case-by-case basis. For those components where no CLB fatigue TLAA exists (not addressed in 10 CFR 54.21), aging due to fatigue can be addressed by either a Section III fatigue analysis in accordance with the CLB (including the additional years for the license renewal period), or an aging management program. This issue is designated as Renewal Applicant Action Item No. 21.

- Applicants should identify plant-specific components with CLB fatigue analyses and provide the corresponding cumulative fatigue usage through the period of extended

operation. Since this is a plant-specific issue, it is designated as Renewal Applicant Action Item No. 22.

- The plant-specific components which are subject to fatigue but do not have a fatigue analysis need to be identified and the plant-specific technical basis for managing the effects of fatigue needs to be described. This issue is designated as Renewal Applicant Action Item No.23.

3.4.1 Loss of Prestress Force Loads in Prestressing Systems

The loss of prestress force loads in the prestressing system was determined by WOG to meet all six criteria for defining a TLAA. Regarding the requirements of 10 CFR 54.21(c)(1), WOG determined:

> the analyses of prestress force loss require an update for the period of extended operation,
>
> the analyses have been projected to the end of the extended period of operation, and
>
> the effects of aging on the intended functions will be adequately managed for the extended period of operation with aging management program AMP-5.6

In RAI No. 13 (NRC letter dated February 19,1998), the staff requested WOG to discuss how the prestress loss rate is determined for the additional 20 years of operation to ensure that its intended function is maintained. The staff stated that Table 4-14 of the GTR does not address calculations for the prestress loss as a consideration in the AMP. The staff considers tendon prestress evaluation as a TLAA that needs to be evaluated for license renewal in accordance with 10 CFR 54.21(c).

WOG's response indicated that a description of how the prestress loss rate is determined is not given in the GTR because it is considered a plant-specific issue that should be addressed in the applicant's license renewal submittal. Since this will be addressed on a plant-specific basis, the issue is designated as Renewal Applicant Action Item No. 24.

3.4.2 Fatigue of Penetration Anchors

The evaluation of fatigue TLAAs for penetration anchors is based on Option (iii) of 10 CFR 54.21(c)(1). The GTR proposes to manage these effects by existing Section XI surveillance and testing programs. For components where CLB fatigue TLAAs exist, this would allow the CLB Section III CUF to be exceeded during the period of extended operation. The staff has not endorsed this option on a generic basis at this time. An applicant wishing to pursue this option would have to request staff review and approval on a case-by-case basis. For components where CLB fatigue TLAAs do not exist (not addressed in 10 CFR 54.21), aging effects due to fatigue can be addressed by either a Section III fatigue analysis (including the additional years for the period of extended operation) or by adequately managing these effects for the period of extended operation. This issue is addressed by Renewal Applicant Action Item No. 21.

3.4.3 Fatigue of Penetration Bellows

The TLAA for penetration bellows is identical to the TLAA for penetration anchors. The evaluation presented in Section 3.4.2 above also applies to penetration bellows.

3.4.4 Fatigue of Mechanical Penetrations

The evaluation of fatigue TLAAs for mechanical penetrations is based on Options (i) and (ii) of 10 CFR 54.21(c)(1), which state that existing CLB fatigue analyses should remain valid for the period of extended operation, or be projected to the end of the period of extended operation. The staff finds this acceptable. The GTR also indicates in Note (4) to Table 3-4 that managing the aging effects due to fatigue for the period of extended operation is an option in lieu of performing additional analyses. This option would allow the component to be included in an ASME Code Section XI ISI program, and would permit the CLB fatigue Section III CUF to be exceeded during the period of extended operation. The staff has not endorsed this option on a generic basis at this time. An applicant wishing to pursue this option would have to request staff review and approval on a case-by-case basis. This issue is addressed by Renewal Applicant Action Item No. 21.

4.0 CONCLUSIONS

BNL has reviewed the WOG GTR (Reference 2) and the responses to the staff's RAIs submitted by WOG. On the basis of its review, BNL concludes that, upon resolution of the open and confirmatory items discussed in Sections 4.2 and 4.3 below, the WOG GTR provides an acceptable demonstration that the aging effects for containment structure components within the scope of the GTR will be adequately managed, so that there is reasonable assurance that these containment structure components will perform their intended functions in accordance with the CLB for the period of extended operation. BNL also concludes that the GTR provides an acceptable evaluation of time-limited aging analyses for the period of extended operation.

Upon successful resolution of the open and confirmatory items, any WOG member plant may reference this GTR in a license renewal application to satisfy the requirements of 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the components within the scope of this GTR will be adequately managed and to satisfy the requirements of 10 CFR 54.21(c)(1) for demonstrating that appropriate findings are made regarding evaluation of time-limited aging analyses for the period of extended operation.

BNL also concludes that (1) subject to the resolution of open and confirmatory items, (2) upon satisfaction of the renewal applicant action items set forth in Section 4.1 below, and (3) upon inclusion of the aging management programs and the TLAA evaluations contained in this GTR in the FSAR supplement, referencing this GTR in a license renewal application will provide the staff with sufficient information to make the necessary findings required by 10 CFR 54.29(a)(1) and (a)(2) for containment structure components within the scope of this GTR.

4.1 Renewal Applicant Action Items

The following renewal applicant action items are to be addressed in the plant-specific license renewal application when incorporating the WOG GTR in a renewal application:

(1) The license renewal applicant will (i) verify that its plant is bounded by the GTR, (ii) commit to programs described as necessary in the GTR to manage the effects of aging during the period of extended operation and (iii) verify that the programs committed to are conducted in accordance with appropriate regulatory controls (e.g. 10 CFR Part 50, Appendix B). Further, the renewal applicant will identify any deviations from the aging management programs which this GTR describes as necessary to manage the effects of aging during the period of extended operation or to maintain the functionality of the containment structure, and deviations from other information presented in the GTR (e.g., materials of construction). The renewal applicant will evaluate any such deviations in accordance with 10 CFR 54.21(a)(3) and (c)(1) on a plant-specific basis.

(2) A summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs is to be provided in the license renewal FSAR supplement, in accordance with 10 CFR 54.21(d).

(3) Individual plant applicants will need to provide a comprehensive list of structures and components subject to an aging management review and the methodology used to develop this list as part of their license renewal applications. Any components determined by the applicant to be subject to an aging management review for license renewal but not within the scope of the GTR are required to be addressed in the license renewal application.

(4) Provide cross-section drawings for the containment structures; and detailed drawings of the sand pocket region and other plant-specific features, if applicable.

(5) Provide legible drawings of equipment and penetration details as part of the description of the containment structure components.

(6) For prestressed concrete containments, indicate whether the tendon access gallery is included as a containment structure component, subject to an aging management review. If it is, provide the details of the aging management review and the credited aging management program. If not, provide a technical basis for exclusion, which addresses the potential for degradation of the lower vertical tendon anchors resulting from the environmental conditions in the tendon access gallery.

(7) Discuss plant-specific operating experience relevant to age-related degradation of containment structure components and how this experience has been considered in the aging management review.

(8) For concrete containments, verify that the original plant design and construction specifications satisfy the criteria which are relied upon to exclude leaching of calcium

hydroxide and reaction with aggregates as significant aging mechanisms. If this is not the case, describe the aging management program which is credited to manage the aging effects associated with these aging mechanisms.

(9) For concrete containments, discuss whether local heating of containment concrete at the Main Steam and/or any other penetrations results in sustained concrete temperatures exceeding 200°F. If this condition exists, provide an aging management review and describe the credited aging management program.

(10) Provide governing edition and/or date for codes and standards which are plant-specific.

(11) Specify whether freeze-thaw is an applicable aging mechanism which will be managed by AMP 5.1 or AMP 5.2, as applicable. If not, provide the technical basis for exclusion.

(12) Specify whether aggressive chemical attack is an applicable aging mechanism which will be managed by AMP 5.3 or AMP 5.4, as applicable. If not, provide the technical basis for exclusion.

(13) Provide details of the groundwater monitoring program and discuss potential seasonal variation in ground water chemistry.

(14) For prestressed concrete containments, discuss plant experience with respect to tendon grease leakage and, if applicable, how it is being managed; also discuss the potential effects of grease leakage on the shear load capacity of the containment structure.

(15) Each license renewal applicant needs to define its plant-specific program to address the stress corrosion cracking (SCC) problem because it is a concern for dissimilar metal welds, and in the case of stainless steel bellows assemblies, SCC may be significant if the material is not shielded from a corrosive environment. For the period of extended operation, examination Categories E-B and E-F and augmented VT-1 visual examination of bellows assemblies and dissimilar metal welds are warranted.

(16) Discuss the plant-specific coatings monitoring and maintenance program and specify whether it is credited as an AMP for containment steel elements.

(17) For prestressed concrete containments, specify whether post-tensioning system degradation will be managed by AMP 5.6 (Section XI, Subsection IWL, Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants, Examination Category L-B, Unbonded Post-Tensioning System, 1992 Code Edition with 1992 Addenda of the ASME Code) and the additional requirements delineated in 10 CFR 50.55a (b)2(ix). If not, provide the technical basis for exclusion.

(18) Specify whether settlement of the containment foundation is an applicable aging mechanism which will be managed by AMP 5.7. If not, provide the technical basis for exclusion.

(19) Identify whether erosion of porous concrete subfoundation layer is an applicable aging mechanism; if applicable, provide an aging management review and describe the credited aging management program.

(20) As stated in Section 2.3 above, the GTR listed only six (6) attributes to form the basis for each aging management program. However, the "Draft Standard Review Plan for the Review of License Renewal applications for Nuclear Power Plans", dated April 21, 2000, identifies ten (10) elements (attributes), as appropriate, for an acceptable AMP. The GTR predates the Draft standard Review Plan for the review of Licencing Renewal applications for Nuclear Power Plans, and states in Section 4.0 that the report only presents program attributes for the AMPs, and that plant-specific details of the AMPs will be developed during the preparation of license renewal applications. Therefore, applicants for license renewal will be responsible for developing and describing the plant-specific AMPs with each AMP consisting of ten (10) elements.

(21) The WOG GTR indicates that the license renewal applicant may update an existing design fatigue analysis to account for the additional years of plant operation, or manage the effects of the aging mechanism through aging management programs. The GTR uses AMP 5.5 for managing the effects of fatigue during the renewal license period, and basically endorses the ASME Code Section XI surveillance and testing program. For components where CLB fatigue TLAAs exist, this option would allow the CLB fatigue Section III CUF to be exceeded during the period of extended operation. The staff has not endorsed this option on a generic basis at this time. An applicant wishing to pursue this option would have to obtain staff review and approval on a case-by-case basis. For components where CLB fatigue TLAAs do not exist (not addressed in 10 CFR 54.21), aging effects due to fatigue can be addressed by either a Section III fatigue analysis (including the additional years for the period of extended operation) or by adequately managing these effects for the period of extended operation.

(22) Specify the containment structure components and provide plant-specific details of the TLAAs for prediction of cumulative fatigue usage through the period of extended operation.

(23) Specify those containment structure components for which fatigue is an applicable aging mechanism, but no CLB fatigue analysis based on a 40 year plant life exists. In addition to implementation of AMP 5.5, the requirements of 10 CFR 50.55a should be met.

(24) For prestressed concrete containments, provide plant-specific details of the TLAA for prediction of tendon prestress losses through the period of extended operation.

4.2 Open Items

Based on BNL's review of the WOG GTR and the WOG responses to NRC's RAIs, the following open items need to be resolved:

(1) The following intended functions, which are specific to containment structures, should be added to the discussion on intended functions of the GTR:

1. Providing structural or functional support of safety-related systems, structures, and components following a design basis accident (DBA).

2. Serving as an external missile barrier consistent with design basis commitments. The containment structure is designed for all missile forces.

3. Providing a heat sink during a DBA or station blackout in addition to the spray system.

(2) The GTR identified structural connections as a containment structure component that requires aging management in Table 2-1. However, there is no definition and description of structural connections in GTR Section 2. A definition and description for structural connections are needed.

(3) The GTR identified embedments as a containment structure component that requires aging management in Table 2-1. However, there is no definition and description of embedments in GTR Section 2. A definition and description for embedments are needed.

(4) The aging effects in concrete due to leaching of calcium hydroxide and alkali aggregate reaction are identified in the GTR as not requiring aging management. This is unacceptable because plant-specific evaluation of their applicability is needed. These aging mechanisms, if applicable, can be managed by ASME Code, Section XI, Examination Category L-A. This is a mandated CLB inspection program, which will continue into the period of extended operation.

The GTR identifies examination category L-A of IWL to manage freeze-thaw (AMP5.1) and to manage aggressive chemical attack and rebar degradation (AMP5.3), where applicable. According to the GTR, it may not be necessary to credit examination category L-A to manage these aging mechanisms, if on a plant specific basis, the applicant determines these aging mechanisms are not applicable. Consequently, in these cases, examination category L-A would not be credited to manage concrete degradation for License Renewal.

In general, it is intended that maximum credit be taken for existing mandated programs (e.g., examination category L-A) in the development of an applicant's aging management program. Implementation of examination category L-A as an aging management program for containment concrete should be specified in the GTR.

(5) The WOG GTR does not commit to inspection of inaccessible areas when there is no indication of degradation of adjacent accessible areas, except when the potential for degradation is "event driven"; i.e., some unusual event has occurred which has the potential to degrade inaccessible areas of the containment structures. This qualification was not considered acceptable. Therefore, the GTR cannot be referenced by license renewal applicants for managing aging of inaccessible areas. Plant-specific resolution of this issue will be required.

4.3 Confirmatory Items

Based on the BNL review, the following Confirmatory Items need to be addressed by WOG:

(1) RAI 29 identified errors in Section 2 of the WOG GTR concerning the range of basemat thicknesses for Type 1 and Type 3 containments. In the RAI response, the WOG acknowledged the errors and provided a corrected list of basemat thickness range for Types 1, 2, and 3 containments.

(2) RAI 24 addresses the implementation of ASME Appendix VII and Appendix VIII of Section XI, when ultrasonic examination is used. In the RAI response, the WOG acknowledged the necessity of implementing Appendices VII and VIII when ultrasonic examination is used.

(3) RAIs 6, 7, 30 and 31 identified text errors in Table 3-1. In the RAI responses, the WOG acknowledged these errors and identified the corrected text.

(4) RAI 20 identified that AMP 5.5 refers to IWE examination categories taken from the 1989 Code Edition, instead of the 1992 Code Edition. In the WOG RAI response, this error is acknowledged and the correct examination categories are identified.

(5) RAI 33 requested missing publication dates for references and bibliography documents listed in Section 6.0 of the WOG GTR. The requested dates were provided in the WOG response to the RAI.

5.0 REFERENCES

(1) 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," published in Federal Register, Vol. 60, No. 88, pp. 22461-22495, May 8, 1995.

(2) WCAP-14756, "Aging Management and Evaluation for Pressurized Water Reactor Containment Structure," Westinghouse Owner's Group, December 1996.

(3) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," the American Society of Mechanical Engineers, 1992 Edition with 1992 Addenda.

(4) "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants," Working Draft, September 1997.

(5) Letter to Document Control Desk (NRC), from T.V. Greene (WOG), dated December 11, 1996, submitting WCAP-14756, Revision 0, "Aging Management Evaluation for Pressurized Water Reactor Containment Structures."

(6) Letter to R.A. Newton (WOG), from R.K. Anand (NRC), dated February 19, 1998, transmitting "Request for Additional Information Regarding the Westinghouse Owners Group Topical Report WCAP-14756, Revision 0, December 1996."

(7) Letter to Document Control Desk (NRC), from R.A. Newton (WOG), dated March 17, 1998, "Planned Date for Completion of Responses to RAI's."

(8) Letter to R.K. Anand (NRC), from R.A. Newton (WOG), dated May 29, 1998, Response to NRC Request for Additional Information on WOG Generic Technical Report WCAP-14756, "License Renewal Evaluation: Aging Management Evaluation for Pressurized Water Reactor Containment Structures."

ACKNOWLEDGEMENTS

The authors would like to acknowledge the efforts and support of the Westinghouse Owners Group (WOG) Ad Hoc Technical Group Chairman for this report, Mr. Don Eggett of Commonwealth Edison, and Greg Robison of Duke Power.

The authors would also like to thank the other members of the WOG Life Cycle Management/ License Renewal (LCM/LR) Program Core Group, and especially the Group Chairman, Mr. Roger Newton of Wisconsin Electric, for their direction and support during the preparation of this report.

Finally, the authors would like to thank those individuals from the WOG utilities and the Sandia National Laboratory who performed the technical reviews of this report throughout its development. The information and technical comments from L. J. Branstetter and J. T. Nakos, SNL; D. V. Ramsey and M. J. Ferlisi, Duke Power; J. A. Reininger, AEP; Dan Zebrowski, RGE; Jeff Gehlhar, NSP; and A. L. Reimer, WEPCO; and Robert E. Nickell of Applied Science & Technology; as well as the assistance provided by Gilbert/Commonwealth and the Yankee Atomic, are gratefully acknowledged.

The special contributions of Charlie Meyer, Mike Canton, Frank Peduzzi, Narendra Prasad, Charlie Kim, Gene Kubansik, and Bob Funkhouser, all of Westinghouse Electric Company, Energy Systems Business Unit, are also gratefully acknowledged. Thanks are also due to Gordon Vytlacil of Westinghouse Electric Company, Energy Systems Business Unit, and Leslie McSwain and June Crevar for their coordination and for their editorial assistance.

DISCLAIMER OF RESPONSIBILITY

This report was prepared by the WOG LCM/LR Working Group as an account of work sponsored by the WOG. Neither members of the WOG, Westinghouse Electric Company, nor any person acting on their behalf:

- Makes any warranty, express or implied, with respect to the use of any information, apparatus, method, or process disclosed in this report or use thereof may not infringe privately owned rights, or
- Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this report

COPYRIGHT NOTICE

EXECUTIVE SUMMARY

This report evaluates aging of the pressurized water reactor (PWR) containment structure to ensure that intended functions will be maintained during an extended period of operation. The PWR containment structure performs the intended functions of:

- Ensuring the integrity of the reactor coolant pressure boundary (RCPB)
- Ensuring the capability to contain a shut down of the reactor and maintain it in a safe shutdown condition
- Ensuring the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR 100 guidelines
- Ensuring compliance with the U.S. NRC's regulations for environmental qualification (10 CFR 50.49)

PWR containment structures are subject to an aging management review because they maintain intended functions, are passive, and are long-lived. This aging management review identifies mechanisms that cause aging effects and presents options that manage these effects to ensure that the intended functions are maintained.

The scope of this report includes domestic, commercial nuclear power plants with Westinghouse nuclear steam supply systems (NSSSs). For the PWR containment structure, the scope is limited to maintaining the structural integrity of the containment structure and penetrations, including protection of internal systems and structures as well as the external environment. This evaluation was performed in support of the WOG LCM/LR program.

Effects from all design limits, time-limited aging analyses (TLAAs), aging, and industry issues have been evaluated. Options to manage aging effects that degrade intended functions are provided. For the PWR containment structure, the following require aging management programs:

- Freeze-thaw of concrete
- Aggressive chemical attack on concrete
- Corrosion in reinforcing steel
- Corrosion of steel liners, steel containment shells, and penetrations
- Degradation in containment post-tensioning systems
- Mechanical wear and/or loss of pressure retention of airlocks and hatches
- Embrittlement of gaskets
- Stress corrosion cracking of penetrations and bellows
- Coating degradation
- Fatigue of penetrations and bellows

Penetrations associated with high temperature may require action by the utility to perform a fatigue analysis, per TLAA requirements, to show that an existing analysis remains valid, or can be projected, to the extended period of operation.

Options to manage aging are part of current industry practice based on the 1992 Code Edition, and Addenda, of ASME Section XI, Subsections IWE and IWL. The effectiveness of these programs during an extended period of operation is justified since they are in compliance with 10 CFR 50.55a and NRC Regulatory Guides.

In conclusion, this evaluation has shown that the PWR containment structure intended functions will be maintained by these aging management options (when implemented) during an extended period of operation. In addition, the system intended functions supported by the PWR containment will also be maintained.

TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF FIGURES

ACRONYMS

AAR	Alkali aggregate reaction
ACI	American Concrete Institute
AEC	Atomic Energy Commission
AISC	American Institute of Steel Construction
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARD	Age-related degradation
ARDM	Age-related degradation mechanism
ASA	American Standards Association
ASME	American Society of Mechanical Engineers
ASR	Alkali-silica reaction
ASTM	American Society for Testing and Materials
AWS	American Welding Society
CLB	Current licensing basis
CS	Carbon steel
CTL	Construction Technology Laboratories
DBA	Design basis accident
DOE	Department of Energy
EPR	Ethylene propylene rubber
EPRI	Electrical Power Research Institute
EQ	Environmental qualification or equipment qualification
FEA	Finite element method
FSAR	Final safety analysis report
GTR	Generic Technical Report
GWL	Groundwater level
HVAC	Heating, ventilation, and air conditioning
IAEA	International Atomic Energy Agency
ID	Inner Diameter
IEEE	Institute of Electrical and Electronics Engineers
ILRT	Integrated leak rate test
IR	Insulation resistance
ISI	Inservice inspection
LCM/LR	Life Cycle Management/License Renewal
LER	Licensee event report
LOCA	Loss-of-coolant accident
LWR	Light water reactor
MIC	Microbiologically induced corrosion

ACRONYMS (Continued)

NA	Not applicable
NDE/NDT	Non-destructive examination/non-destructive testing
NEI	Nuclear Energy Institute
NIST	National Institute of Standards and Technology
NMR	Nuclear magnetic resonance
NPRDS	Nuclear plant reliability data system
NPT	Registered ASME symbol for a qualified manufacturer or fabricator of nuclear parts or components
NR	Not required
NRC	Nuclear Regulatory Commission
NSSS	Nuclear steam supply system
NUMARC	Nuclear Management and Resources Council; now part of NEI, the Nuclear Energy Institute
NUREG	Nuclear regulatory guideline
OD	Outside diameter
ORNL	Oak Ridge National Lab
PVC	Poly vinyl chloride
PWR	Pressurized water reactor
RCPB	Reactor coolant pressure boundary
RCS	Reactor coolant system
RHR	Residual heat removal
SAG	Structural aging program
SAT	Structural acceptance test
SC	Structure or component
SCC	Stress corrosion cracking
SCV	Steel containment vessel
SI	Safety injection
SS	Stainless steel
SSC	Systems, structures, and components
TGSCC	Transgranular stress corrosion cracking
TLAA	Time-limited aging analysis
UBC	Uniform building code
UFSAR	Updated final safety analysis report
USASI	United States of American Standards Institute
UTS	Ultimate tensile strength
WOG	Westinghouse Owners Group
XLPE	Cross-linked polyethylene

DEFINITIONS

Aging Management Review

Identification and evaluation of aging effects to determine which aging effects require management during an extended period of operation.

Current Licensing Basis (CLB)

The set of U.S. NRC requirements applicable to a specific plant and a licensee's written commitment for ensuring compliance with and operation within applicable U.S. NRC requirements and the plant-specific design basis (including all modifications and additions to such commitments over the life of the license) that are docketed and in effect.

Nuclear Power Plant

Nuclear power facility of a type described in 10 CFR 50.21(b) or 50.22.

Time-Limited Aging Analyses (TLAAs)

Licensee calculations and analyses that:

1. Involve systems; structures, and components within the scope of license renewal, as delineated in § 54.4(a);

2. Consider the effects of aging;

3. Involve time-limited assumptions defined by the current operating term, for example, 40 years;

4. Were determined to be relevant by the licensee in making a safety determination;

5. Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in § 54.4(b); and

6. Are contained or incorporated by reference in the CLB.

1.0 INTRODUCTION

The objectives of this report are to:

- Identify and evaluate aging effects that degrade component functions that support system or structure intended functions
- Identify and evaluate time-limited aging analyses (TLAAs)
- Provide options, in terms of activities and program attributes, to manage these aging effects, and if necessary address TLAAs

System-level intended functions will be maintained by maintaining structure or component (SC) functions that support system intended functions. Hereafter, those SC functions that support system intended functions will be referred to as SC intended functions.

Aging management options identified in this report, when implemented, will ensure that the containment structure intended functions are maintained during an extended period of operation.

This evaluation starts by identifying why the system, structure, or component (SSC) is within the scope of the license renewal rule. An SSC is within the scope of the rule if it supports an intended function. SCCs within the scope of the rule are:

1. The safety-related systems, structures, and components that are relied on to remain functional during and following design-basis events (10 CFR 50.49 (b)(1)) to ensure the following functions:
    a. The integrity of the reactor coolant pressure boundary,
    b. The capability to shut down the reactor and maintain it in a safe shutdown condition, or
    c. The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR Part 100 guidelines.
2. All nonsafety-related systems, structures, and components whose failure could prevent any of the functions identified in paragraph 1 a, b, or c above.
3. All systems, structures, and components relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the U.S. NRC's regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), pressurized thermal shock (10 CFR 50.61), anticipated transients without scram (10 CFR 50.62), and station blackout (10 CFR 50.63).

An intended function is the basis for including an SSC within the scope of license renewal as defined above.

The evaluation continues by determining if the SC is subject to an aging management review. An SC is subject to an aging management review if the SC:

- Supports or performs an intended function of a system or structure within the scope of Part 54

- Performs an intended function in a passive manner

- Is long-lived

The PWR containment structure parts or subcomponents within the scope of the rule and subject to an aging management review are identified in Section 2.0. Section 2.0 also identifies mechanisms that cause aging effects and identifies TLAAs. The aging management review (Section 3.0) describes age-related degradation mechanisms to identify resulting aging effects. Aging effects and TLAAs are then evaluated to determine degradation of intended functions. Options managing the effects of aging and TLAAs that degrade intended functions are provided in Section 4.0.

Demonstration that these programs adequately manage aging effects to maintain intended functions, consistent with the CLB, for an extended period of operation is provided partially by this report and is completed by the plant-specific LR application. This report provides the generic technical basis supporting a part of the demonstration. The technical basis explains how the programs manage aging effects and why the programs will remain effective during an extended period of operation. The aging management options provided in this evaluation are to be developed into programs by utilities applying for a renewed license. Plant-specific implementation of these programs, as documented in the LR application, completes the demonstration process.

In February 1996, Oak Ridge National Laboratory issued a report [Ref. 1] that was prepared as part of a program "to assist the U.S. NRC in their assessment of the effects of corrosion on the structural capacity and leaktight integrity of metal containment vessels and steel liners of reinforced concrete structures in nuclear power plants." The report was reviewed, and this generic technical report (GTR) is consistent with the assessment methodology given.

The aging management programs identified to address the significant aging effects associated with the containment structures have attributes that are in conformance with U.S. NRC SECY-96-080, April 16, 1996 [Ref. 2]. The attributes reference the 1992 ASME Code Section XI with the 1992 Addenda of Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants." The specified modifications and clarifications given in SECY-96-080 to amend 10 CFR 50.55a are recommended for incorporation into a utility's license renewal plan that addresses containment structures. The four modifications to the final rule of 10 CFR 50.55a are:

- Expansion of the evaluation of inaccessible areas of concrete containments to include metal containments and the liners of concrete containments

- Permission of alternative lighting and resolution requirements for remote visual inspection of the containment

- Examination of pressure-retaining welds and pressure-retaining dissimilar metal welds are optional

- An alternative sampling plan has been added

The clarification to the rule that more clearly defines the frequency of Subsection IWE general visual examination is also recommended for inclusion in the aging management programs.

1.1 APPLICABILITY

This evaluation is generically applicable to domestic commercial nuclear power plants with the Westinghouse nuclear steam supply system (NSSS), as listed in Table 1-1. Preparation of the report included establishment of boundaries by Westinghouse Electric Company as well as utility reviewer confirmation of these boundaries to a practical extent. Use of this report, as referenced by a license renewal application, should include a verification of all the bounding information against plant-specific data. This verification will identify that the report is applicable to the plant or what plant-specific data are not covered by this report and will be evaluated as part of the license renewal application.

1.2 STRUCTURE/COMPONENT SCOPE

The evaluation of the PWR containment structure includes the containment subcomponents that are addressed in the scope of this evaluation and listed in Table 1-2. The shield building is also included in the scope of the aging management evaluation reported within this report. It is noted that the Westinghouse ice condenser system is excluded. The boundaries of the containment generic technical report (GTR), include the entire mechanical penetration assembly, exclusive of the process piping within the penetration. The welds to the process pipe are included in this scope. Flued heads are included in the scope. Piping insulation is not included in the scope although in some cases the piping insulation also serves to maintain the temperature of the concrete adjacent to the penetration sleeve within permissible limits.

For the high-temperature steel containment piping mechanical penetrations, the scope of this evaluation includes all items that perform a containment function except for the process pipe. This includes the containment vessel nozzle, the bellows assembly, and the multiple flue head.

For the smaller piping and cold lines, the scope of this evaluation includes the closure weld of the steel containment to the process pipe. The process pipe is not included. Where bellows assemblies are used for smaller piping and cold lines, the included scope is the same as for the high-temperature lines described previously.

TABLE 1-1
COMMERCIAL WESTINGHOUSE NUCLEAR POWER PLANTS IN THE UNITED STATES

Plant Name	Net MWe	Commercial Operation Date
Robinson 2	683	3/71
Shearon Harris	860	5/81
Braidwood 1 & 2	1120	7/88 & 10/88
Byron 1 & 2	1105	9/85 & 8/87
Zion 1 & 2	1040	12/73 & 9/74
Haddam Neck	590	1/68
Indian Point 2	970	8/74
Indian Point 3	965	8/76
Catawba 1 & 2	1129	6/85 & 8/86
McGuire 1 & 2	1129	12/81 & 3/84
Beaver Valley 1 & 2	810 & 830	1/76 & 11/87
Turkey Point 1 & 2	666	12/72 & 9/73
South Texas Project 1 & 2	1250	8/88 & 6/89
Donald C. Cook 1 & 2	1020 & 1060	8/75 & 7/78
Seabrook	1150	7/90
Millstone 3	1146	4/86
Prairie Island 1 & 2	503 & 500	12/73 & 12/74
Diablo Canyon 1 & 2	1073 & 1087	5/85 & 3/86
Salem 1 & 2	1106	6/77 & 7/81
R.E. Ginna	470	7/70
Virgil C. Summer	885	1/84
Joseph M. Farley 1 & 2	814 & 824	12/77 & 7/81
Alvin W. Vogtle 1 & 2	1100 & 1097	6/87 & 5/89
Sequoyah 1 & 2	1148	7/81 & 6/82
Watts Bar 1 & 2	1177	1996
Comanche Peak 1 & 2	1150	8/90 & 7/93
Callaway	1125	4/85
North Anna 1 & 2	911 & 909	6/78 & 12/80
Surry 1 & 2	781	12/72 & 5/73
Point Beach 1 & 2	485	12/70 & 10/72
Kewaunee	503	6/74
Wolf Creek	1135	9/85

The nozzle that is welded into the steel containment plate and the steel header plate for the electrical terminals are included in the scope. The wiring, sealing compound, fixtures to hold the sealing compound, and seal welds, which connect the fixtures to the header plate, are not included in the scope of this evaluation.

Electrical penetrations of the containment shell are included in the scope of the report. Electrical penetrations have assemblies that consist of both metallic and nonmetallic subcomponents. The scope of this evaluation includes all metallic components of the electrical penetration that are part of the containment pressure boundary. Seal materials such as epoxy and silicone rubber are not included in this evaluation but are included in reports that address the environmental qualification of these types of materials. Nonmetallic assemblies are not included in the scope since they are replaced as needed as defined by the leak detection tests.

TABLE 1-2
SCOPE OF CONTAINMENT SUBCOMPONENTS ADDRESSED IN REPORT

Shell and dome of containment and shield building
Structure mat (foundation) of containment and shield building
Structural connections
Concrete reinforcing, tendons, etc.
Steel liner
Embedments
Electrical penetrations including connectors
Penetrations for gas and fluid systems that include isolation valves
Fuel transfer tube
Equipment and personnel hatches that include seals

The scope includes entire personnel airlocks. Airlock components such as equalizing valves, handwheel shaft seals, and door seals are included. Electrical penetrations of the airlock bulkheads are not included.

Equipment hatch designs are basically the same for both concrete and steel containments. The scope of this evaluation includes the entire hatch including the flexible seals between the hatch barrel flange and the head flange.

The scope also includes all parts of the fuel transfer tube that serve as part of the containment system such as the blind flange with double seals, the closure detail between the containment liner (or steel containment) and the transfer tube, and the transfer tube itself. In addition, the gate valve at the outboard end of the transfer tube is included. The bellows assembly

connections between the stainless steel canal liners and the transfer tube are not included in the scope.

The use of the term PWR containment within this report includes all of the scope as defined above.

2.0 IDENTIFICATION OF TIME-LIMITED AGING ANALYSES AND AGING EFFECTS

This section identifies the time-limited aging analyses (TLAAs) and aging effects related to the PWR containment. First, the PWR containment is described in general terms. This description includes the boundary of the PWR containment covered in this report. Next, the reason why the PWR containment is within the scope of the license renewal rule is provided. This reason identifies the intended functions maintained by the PWR containment. The parts of the PWR containment that are subject to an aging management review are then identified and described in detail. These detailed descriptions identify related TLAAs and age-related degradation mechanisms. Finally, aging effects resulting from age-related degradation mechanisms are identified.

2.1 GENERAL DESCRIPTION AND BOUNDARY

PWR containments for plants included in this study share commonality of function, yet differ in the details of their design. Fundamentally, PWR containments maintain structural integrity, and their structures and penetrations perform the intended functions identified in this section.

The nuclear power plant design incorporates multiple barriers between the prime radioactivities and the public to provide protection from unacceptable radiation exposure. These barriers consist of:

- The fuel element cladding, which encapsulates the fuel material and the fission products
- The reactor coolant system (RCS) boundary, which contains any leakage from the fuel elements
- A containment structure, which encloses a major portion, if not all, of the RCS

The containment building serves as the last engineered barrier to the release of radioactivity from the containment atmosphere to the environment. Therefore, the containment design and construction details have a significant effect on the safety of a nuclear power plant.

The containment building limits accident peak temperatures and pressures and contains the energy of a loss-of-coolant accident (LOCA) or pipe break accident. The reactor containments fall into one of two general function categories: dry or vapor suppression. Dry containments rely on volume and physical strength to contain the energy released by an LOCA. Vapor suppression containments use ice to condense or suppress the effects of energy contained in the steam or vapor released by an accident. Also, the lower portions of the containment act as a sump for the collection of liquids spilled during an accident, from which liquid may be redrawn for recirculation by the safety injection (SI) and residual heat removal (RHR) pumps during the recirculation phase.

The configurations and materials of the containments vary depending on the design (e.g., steel containment, concrete containment, ice condenser containment). This section addresses the

similarities and differences of the various containment designs as well as providing information related to:

- Containment function
- Containment description/configuration
- Materials of construction
- Engineering and design data
- Codes, standards, and regulatory requirements
- Industry issues and maintenance requirements

The containment boundary consists of the interior and exterior surface of the reinforced concrete containment shell, including the basemat, or the interior surface of the steel containment and the exterior surface of the shield building, including the common basemat. The shield building is considered part of the containment boundary for plants with a steel containment since it aids in the performance of intended functions. The boundary includes all penetration assemblies in the containment shell, such as mechanical and electrical penetrations, and the equipment hatch.

2.2 PARTS OR SUBCOMPONENTS SUBJECT TO AN AGING MANAGEMENT REVIEW

The PWR containment structure performs the following:

- Protect the environment, plant personnel, and equipment from unacceptable radiation exposure and release during normal operating or accident conditions and minimize the consequences of an accident

- Provide structural support for interior structures and systems, and protect them from external loadings (e.g., wind, hurricane, tornado, etc.)

- Provide for transfer of electricity, fuel, gas, and fluids through the containment barrier for use during normal operation

- Provide for transfer of personnel and equipment

- Provide for emergency access

- Provide a containment venting function

Consistent with the above, the PWR containment performs the intended functions of:

- Ensuring the integrity of the reactor coolant pressure boundary[1]
- Ensuring the capability to shut down the reactor and maintain it in a safe[1] shutdown condition
- Ensuring the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the 10 CFR 100 guidelines
- Ensuring compliance with the U.S. NRC regulations for environmental qualification (10 CFR 50.49)

The functions of the structures, penetrations, and hatches are discussed in more detail in the following subsections.

The parts or subcomponents that specifically support these intended functions are listed in Table 2-1 and described in Section 2.3. Note that all of the parts listed are subject to an aging management review.

TABLE 2-1
SUMMARY OF PARTS OR SUBCOMPONENTS REQUIRING AGING MANAGEMENT REVIEW

Part or Subcomponent	Aging Management Review Required?
Shell and dome	Yes
Structural mat (foundation)	Yes
Structural connections	Yes
Concrete reinforcing, tendons, etc.	Yes
Steel liner	Yes
Embedments	Yes
Electrical penetrations including connectors	Yes
Penetrations for gas and fluid systems that include isolation valves	Yes
Fuel transfer tube	Yes
Equipment and personnel hatches that include seals	Yes

---

[1]This intended function is included as a result of the structural support provided by containment.

The PWR containment structures, parts, or components perform intended functions in a passive manner and are long-lived; therefore, they are subject to an aging management review.

2.2.1 Structural Functions

The containment structure encloses and protects the major portion of a PWR nuclear steam supply system (NSSS) and serves as the last engineered barrier to the release of radioactivity. During the operating life of the plant, the containment structure:

- Limits the leakage rate of contaminants resulting from any LOCA and other postulated accidents
- Provides continued radiation shielding during normal plant operation and during accident conditions
- Protects the reactor vessel and other safety-related systems, equipment, and components against resulting loads from all postulated external environmental conditions and missiles

2.2.2 Penetration Functions

All penetrations through the containment pressure boundary are components of the containment system and are designed to limit leakage of radioactive materials from the containment interior to the outside environment in the event of an accident.

In addition to the containment function performed by all penetrations, each penetration performs service-related functions depending on the particular type of penetration. Penetrations may also serve as support points for systems such as piping passing through the penetration.

The following describes the specific functions of various types of containment penetrations.

2.2.2.1 Mechanical Penetrations

Mechanical penetrations provide passage for process piping to transmit liquids or gases across the containment pressure boundary. Mechanical penetrations in some plants are also designed to limit radiation streaming to areas that require personnel access.

2.2.2.2 Electrical Penetrations

Electrical penetrations provide passage for electrical and instrumentation conductors across the containment pressure boundary while maintaining a leak-tight seal.

2.2.2.3 Personnel Airlocks

Personnel airlocks provide safe and reliable passage of personnel and small equipment across the containment pressure boundary during plant operation without compromising the containment system. During plant shutdown, the interlocking system for the two airlock doors can be deactivated, as permitted by technical specification limitations, to provide more ready access to the containment.

2.2.2.4 Equipment Hatch

The equipment hatch provides the means for moving larger pieces of equipment and structures across the containment pressure boundary. The equipment hatch is typically used during plant shutdown; it most commonly consists of a single-dished hatch cover.

2.2.2.5 Fuel Transfer Tube Penetration

The fuel transfer tube links the refueling canal in the reactor building with the fuel transfer canal in the fuel handling building. The fuel transfer tube penetration serves as the underwater pathway for moving the fuel assemblies into and out of the reactor building as part of the refueling operations that occur during plant shutdown.

2.2.2.6 Spare Penetrations

During an outage, spare penetrations can readily be converted into additional permanent mechanical or electrical penetrations. The spare penetration provides a cost-effective means to provide an additional penetration without having to create a new opening through the containment by destructive means.

2.2.2.7 Residual Heat Removal Penetrations

Residual heat removal (RHR) penetrations provide passage of the RHR piping across the containment pressure boundary. The RHR system removes heat from the containment in the event of an accident.

2.3 DESCRIPTIONS

The configurations of the PWR containment vary from plant to plant even though they provide the same primary functions. The PWR containment design configurations for the commercial Westinghouse PWR plants within the United States are given in Table 2-2. The designs are grouped into basic configurations, defined as Type 1 through Type 3, with subgroup variations a through c, as applicable. The three configuration types and variations are:

1. Steel Containment with Reinforced Concrete Shield Building
    a. With ice condenser
    b. No ice condenser

2. Steel-Lined Reinforced Concrete

   a. Reactor building is atmospheric

   b. Reactor building is sub-atmospheric

   c. Reactor building is sub-atmospheric with ice condenser

3. Steel-Lined Reinforced Concrete with Post-Tensioning

   a. Three directional post-tensioning

   b. Vertical post-tensioning only

The basic configurations are described in Subsection 2.3.1 and summarized for the plants within the scope of this aging evaluation in Table 2-3.

2.3.1 Configuration Description

2.3.1.1 Steel Containment with Reinforced Concrete Shield Building - Type 1a and 1b

The containment function is performed by a steel containment vessel in combination with a separate reinforced concrete shield building that surrounds the steel containment. The shield building has a vertical right cylindrical wall capped with a shallow hemispherical dome and is supported by a flat circular basemat. The containment vessel is a low-leakage, free-standing steel structure consisting of a cylindrical wall, a hemispherical dome, and a bottom liner plate enclosed in concrete. The vessel is typically anchored to the shield building foundation via anchor bolts welded around the circumference of the cylinder base. The steel containment and the shield building are separated by an annular air space. The airspace and containment, through the vacuum relief system, or just the airspace may be maintained at sub-atmospheric pressure to prevent leakage.

TABLE 2-2
COMMERCIAL WESTINGHOUSE PLANT CONTAINMENT CONFIGURATIONS

Plant Name	Containment Type	Liner	Post-Tensioning	Ice Condenser	Reinf. Concrete Shield Building	Steel Enclosure Building
Robinson 2	Reinf. Concrete	Steel	Vertical	No	No	No
Shearon Harris	Reinf. Concrete	Steel	Not used	No	No	No
Braidwood 1 & 2	Reinf. Concrete	Steel	Three direc.	No	No	No
Byron 1 & 2	Reinf. Concrete	Steel	Three direc.	No	No	No
Zion 1 & 2	Reinf. Concrete	Steel	Three direc.	No	No	No
Haddam Neck	Reinf. Concrete	Steel	Not used	No	No	No
Indian Point 2	Reinf. Concrete	Steel	Not used	No	No	No
Indian Point 3	Reinf. Concrete	Steel	Not used	No	No	No
Catawba 1 & 2	Steel Cylinder	None	Not used	Yes	Yes	No
McGuire 1 & 2	Steel Cylinder	None	Not used	Yes	Yes	No
Beaver Valley 1 & 2	Reinf. Concrete	Steel(1)	Not used	No	No	No
Turkey Point 1 & 2	Reinf. Concrete	Steel	Three direc.	No	No	No
South Texas 1 & 2	Reinf. Concrete	Steel	Three direc.	No	No	No
Donald C. Cook 1 & 2	Reinf. Concrete	Steel	Not used	Yes	No	No
Seabrook	Reinf. Concrete	Steel	Not used	No	No	Yes
Millstone 3	Reinf. Concrete	Steel(1)	Not used	No	No	Yes
Prairie Island 1 & 2	Steel Cylinder	None	Not used	No	Yes	No
Diablo Canyon 1 & 2	Reinf. Concrete	Steel	Not used	No	No	No
Salem 1 & 2	Reinf. Concrete	Steel	Not used	No	No	No
R.E. Ginna	Reinf. Concrete	Steel	Vertical	No	No	No

TABLE 2-2 (Continued)
COMMERCIAL WESTINGHOUSE PLANT CONTAINMENT CONFIGURATIONS

Plant Name	Containment Type	Liner	Post-Tensioning	Ice Condenser	Reinf. Concrete Shield Building	Steel Enclosure Building
Virgil C. Summer	Reinf. Concrete	Steel	Three direc.	No	No	No
Joseph M Farley 1 & 2	Reinf. Concrete	Steel	Three direc.	No	No	No
Alvin W Vogtle 1 & 2	Reinf. Concrete	Steel	Three direc.	No	No	No
Sequoyah 1 & 2	Steel Cylinder	None	Not used	No	Yes	No
Watts Bar 1 & 2	Steel Cylinder	None	Not used	Yes	Yes	No
Comanche Peak 1 & 2	Reinf. Concrete	Steel	Not used	No	No	No
Callaway	Reinf. Concrete	Steel	Three direc.	No	No	No
North Anna 1 & 2	Reinf. Concrete	Steel(1)	Not used	No	No	No
Surry 1 & 2	Reinf. Concrete	Steel(1)	Not used	No	No	No
Point Beach 1 & 2	Reinf. Concrete	Steel	Three direc.	No	No	No
Kewaunee	Steel Cylinder	None	Not used	No	Yes	No
Wolf Creek	Reinf. Concrete	Steel	Three direc.	No	No	No

Notes:

(1) Reactor building is subatmospheric.

TABLE 2-3
PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURE CONFIGURATION CLASSIFICATION

Plant Name	Containment Type	Wall Thickness (ft.)(1)	Dome Thickness (ft.)(1)	Foundation Mat Thickness (ft.)
Robinson 2	3b	3.5	2.5	10.0
Shearon Harris	2a	4.5	2.5	12.0
Braidwood 1 & 2	3a	3.5	3.0	12.0
Byron 1 & 2	3a	3.5	3.0	12.0
Zion 1 & 2	3a	3.5	2.75	9.0
Haddam Neck	2a	4.5	2.5	9.0
Indian Point 2	2a	4.5	3.5	9.0
Indian Point 3	2a	4.5	3.5	9.0
Catawba 1 & 2	1a	3.0	2.25	6.0
McGuire 1 & 2	1a	3.0	2.25	6.0
Beaver Valley 1 & 2	2b	4.5	2.5	10.0
Turkey Point 1 & 2	3a	4.5	2.5	12.0
South Texas 1 & 2	3a	4.0	3.0	18.0
Donald C Cook 1 & 2	2c	3.5	2.5 to 3.5	10.0
Seabrook	2a	4.0	3.5	10.0
Millstone 3	2b	4.5	2.5	10.0
Prairie Island 1 & 2	1b	2.5	2.0	4.0
Diablo Canyon 1 & 2	2a	3.67	2.5	14.5
Salem 1 & 2	2a	4.5	3.5	16.0
R. E. Ginna	3b	3.5	2.5	2.0
Virgil C. Summer	3a	4.0	3.0	12.0
Joseph M Farley 1 & 2	3a	3.75	3.25	9.0
Alvin W Vogtle 1 & 2	3a	3.75	3.75	10.5
Sequoyah 1 & 2	1a	3.0	2.0	9.0
Watts Bar 1 & 2	1a	3.0	2.0	9.0
Comanche Peak 1 & 2	2a	4.5	2.5	12.0
Callaway	3a	4.0	3.0	10.0
North Anna 1 & 2	2b	4.5	2.5	10.0

TABLE 2-3 (Continued)
PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURE CONFIGURATION CLASSIFICATION

Plant Name	Containment Type	Wall Thickness (ft.)(1)	Dome Thickness (ft.)(1)	Foundation Mat Thickness (ft.)
Surry 1 & 2	2b	4.5	2.5	10.0
Point Beach 1 & 2	3a	3.5	2.5	8.0
Kewaunee	1b	2.5	2.0	8.0
Wolf Creek	3a	4.0	3.0	10.0

Notes:

(1) Wall and dome thickness refers to the thickness of the concrete portion of the primary containment or the shield building for the free-standing steel containment.

The containment may be of either Type 1a (vapor suppression, equipped with an ice condenser system) or Type 1b (dry suppression, without an ice condenser system). The ice condenser system resides inside the containment between the crane wall and steel shell. In vapor suppression containments the crane wall separates the ice condenser compartment from the rest of the containment compartment and provides structural support for the polar crane. Figure 2-1 provides a Type 1 containment cross-section with an ice condenser.

Steel Containment

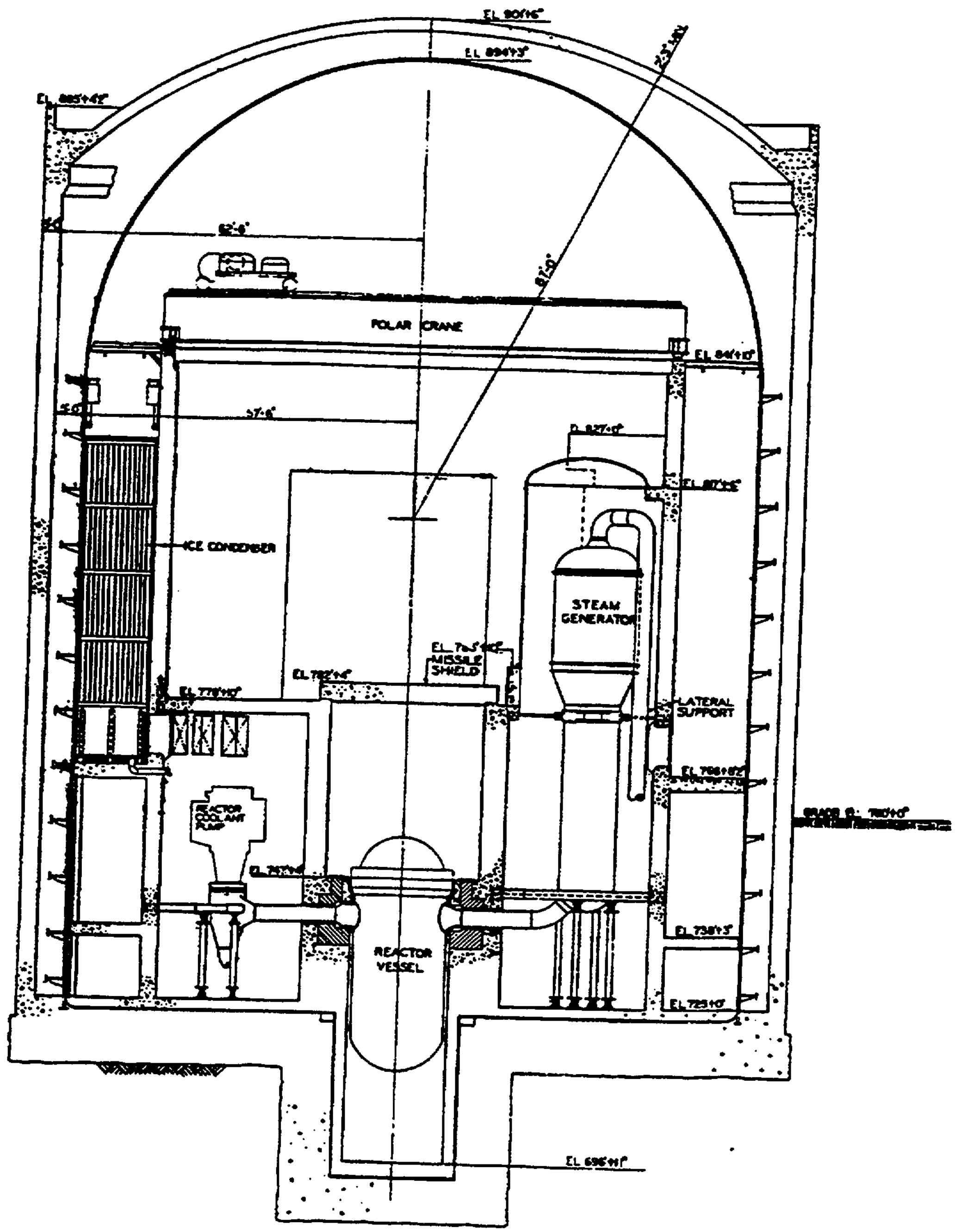
Typical dimensions for the steel containment vessel include a 115-foot diameter for the cylindrical wall, a dome radius of 57.5 feet, and overall height of 171 feet. The bottom liner plate is 1/4-inch thick, is anchored to the foundation slab, and is covered by a reinforced concrete floor slab with a nominal thickness of 2 feet that forms the containment floor. The bottom liner plate functions as a leak-tight membrane and is not designed for structural capabilities. The thickness of the dome and walls of the vessel may vary for various power plants. The thickness of a cylinder wall or dome may be uniform or vary with height. The typical uniform thickness is 3/4 inch for the wall and 11/16 inch for the dome. The non-uniform cylinder wall thickness typically varies from 1-3/8 inches at the bottom to 1/2 inch at the spring line, where the dome and wall meet. The dome varies from a 7/16-inch thickness at the spring line to 15/16 inch at the apex.

On the shell exterior the steel containment vessel has either circumferential and vertical stiffeners or circumferential ring girders with vertical stringers. These stiffeners and girders are required to maintain stresses within the allowable limits. Penetrations are provided for mechanical, electrical, and personnel egress and ingress. Local reinforcing around these penetrations is provided for strengthening.

Shield Building

The shield building is made up of a cylindrical wall with a dome and foundation slab. Typical dimensions for the cylinder walls are a 127-foot diameter and a 3-foot thickness. The dome thickness varies from plant to plant, with values ranging from 24 to 30 inches. The inner radius is consistently 87 feet. The containment basemat foundation consists of a thick, circular reinforced concrete slab with a diameter slightly greater than that for the cylindrical wall. The thickness ranges from 6 to 9 feet.

Vertical reinforcement steel continues from the cylindrical wall to the dome and extends into the foundation slab. In the wall, conventional steel reinforcing bars are applied throughout the structure and placed in a horizontal and vertical pattern near each face. No special reinforcement is provided near penetrations less than 12 inches in diameter, since these



SECTION A-A
REACTOR BUILDING AND CONTAINMENT
STRUCTURAL OUTLINE

Figure 2-1 Type 1 Containment Cross-Section

penetrations do not significantly disturb the reinforcing pattern. For penetrations larger than 12 inches, reinforcement steel is either terminated at the opening and supplemental reinforcing is added, or the reinforcement is continuous and is bent to curve around the opening. Supplemental steel has equal or greater strength than the terminated reinforcement.

Dome reinforcement is arranged in a radial and circumferential pattern, where the radial bars are continued from the vertical bars in the cylindrical wall. Additional reinforcement schemes may be employed.

The slab reinforcement pattern consists of concentric circular bars combined with radial bars, at the top and bottom face, arranged to permit uniform spacing of the vertical wall rebar that extends into the mat.

2.3.1.2 Steel-Lined Reinforced Concrete - Type 2a, 2b, and 2c

PWR containment function is provided by a steel-lined reinforced concrete structure. The structure has a vertical cylindrical wall capped with a hemispherical dome and is supported on a flat basemat that is founded in bedrock. The structure may be designed to operate under sub-atmospheric conditions to limit leakage. A typical cross-section of a Type 2 containment is shown in Figure 2-2.

The cylindrical wall thickness ranges from 4 to 4-1/2 feet, has a 140-foot inside diameter (ID), and is 131 feet from the mat to the spring line. Reinforcement consists of horizontal bars located near both the inner and outer faces of the wall and rows of vertical bars placed near each wall face supplemented by inclined steel bars.

The dome has an inside radius of 70 feet and is 2-1/2 to 3-1/2 feet thick. The internal height from the basemat to the center of the dome varies from 201 to 219 feet. The inside radius of the dome is equal to the inside radius of the cylinder so that the discontinuity due to the change in thickness at the springline is on the outside. Reinforcement consists of meridionally placed reinforcement, extending from the vertical bars of the cylindrical wall, and horizontal hoop bars, both placed in each layer of reinforcement.

The basemat foundation diameter varies from 153 to 158 feet, and the thickness ranges from 6 to 10 feet. The bottom reinforcement is a rectangular grid pattern while the top reinforcement consists of concentric circular bars combined with radial bars that are arranged to permit a uniform spacing of vertical wall rebar that extends into the mat.

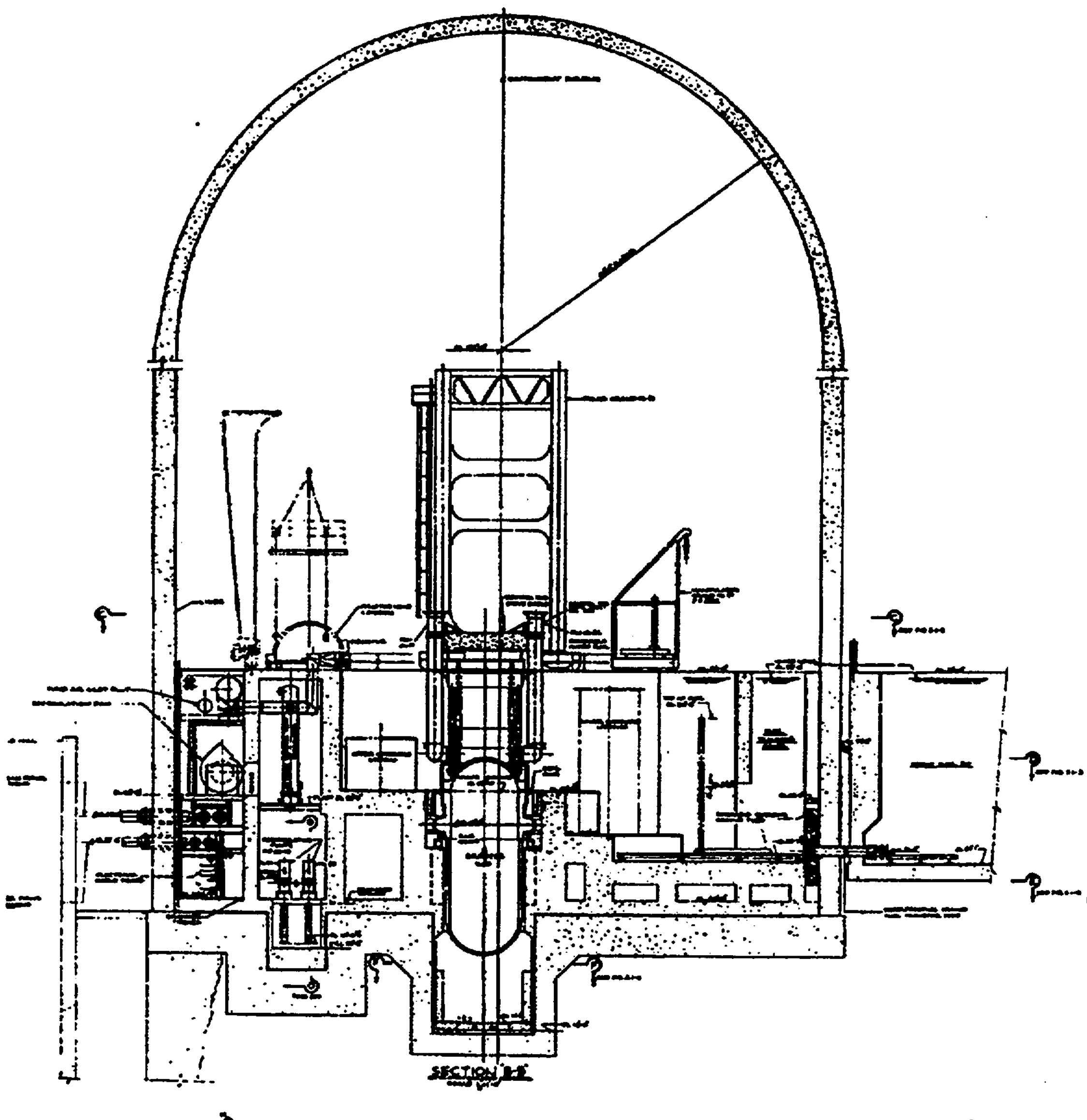


Figure 2-2 Type 2 Containment Cross-Section

Steel Liner

The steel liner is made up of a vertical cylindrical portion closed at the top by a hemispherical dome and attached at the bottom by a mat liner completely enclosing the containment with a steel liner. The wall liner plate is 3/8-inch thick, the dome liner plate is 1/2-inch thick, and the mat liner is 1/4-inch thick. The liner plate is a continuously welded steel membrane supported by and anchored to the inside of the containment at sufficient intervals so that the overall deformation of the liner is the same as the concrete structure under all loading conditions. The liner functions as a leaktight membrane under conditions encountered throughout the operating life of the plant. The liner is designed to resist all direct loads and to accommodate deformation of the concrete containment structure without compromising the leaktight integrity. The liner pressure boundary includes embedments, insert plates, and penetrations. Leak chase channels are installed over penetration to liner seams and over knuckle plate to liner seams to ensure a controlled release.

The 1/4-inch thick steel basemat liner plate is anchored to the foundation slab. Two methods are employed to anchor the liner to the basemat: either the liner is welded to a ring plate that is anchored in the base slab and then welded to the skirt ring of the wall liner; or the liner plate is directly anchored with embedded stiffeners and anchors. A reinforced 2-foot thick concrete slab (fill mat) is poured over the liner plate and may or may not be anchored through the floor liner to the basemat. The fill mat stiffens the liner against sub-atmospheric pressure and protects it from high temperatures associated with accident conditions.

2.3.1.3 Steel-Lined Reinforced Concrete, with Post-Tensioning - Types 3a and 3b

This containment type is similar to the steel-lined reinforced concrete containment except that post-tensioned tendons are used with conventional steel reinforcement in the concrete cylindrical wall and dome. Three directional post-tensioning or vertical post-tensioning is found. The concrete foundation is a conventionally reinforced mat. Figure 2-3 provides a cross-sectional view of a Type 3 containment.

Three Directional - Type 3a

The cylindrical portion of the containment concrete is prestressed by a post-tensioning system composed of horizontal and vertical tendons. The horizontal tendons are either placed in three 240-degree segments using three vertical buttresses spaced 120 degrees apart (the three-buttress configuration; see Figure 2-4), or in six 120-degree segments using six vertical buttresses spaced 60 degrees apart (the six-buttress configuration). The three-way dome tendons are anchored at the side of the ring girder. The buttresses act as a support for the anchorages. The dome has a three-way, criss-cross tendon pattern in which groups of tendons intersect at 120 degrees. A continuous access gallery is provided beneath the base slab for installation and inspection of the vertical tendons.

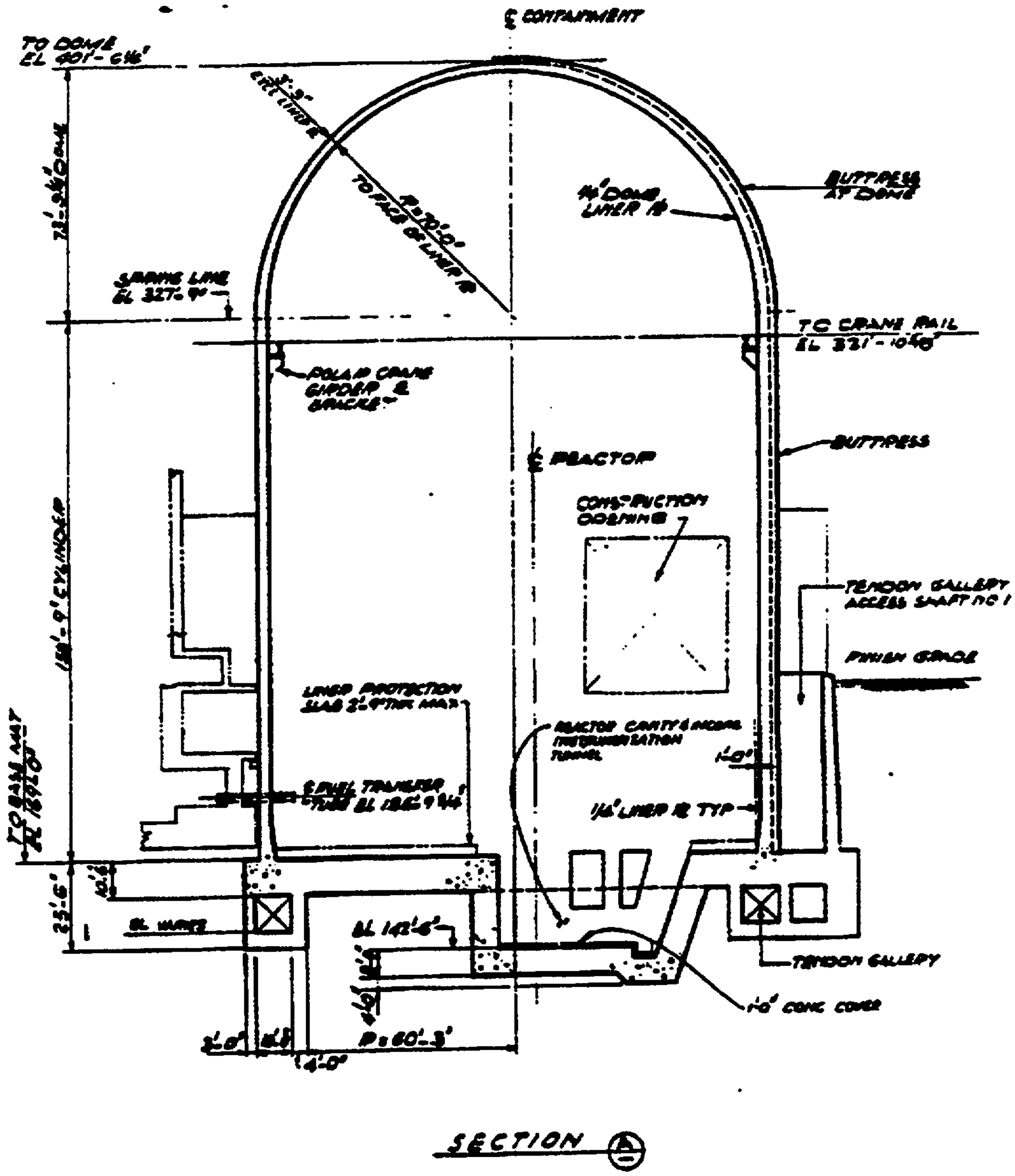


Figure 2-3 Type 3 Containment Cross-Section

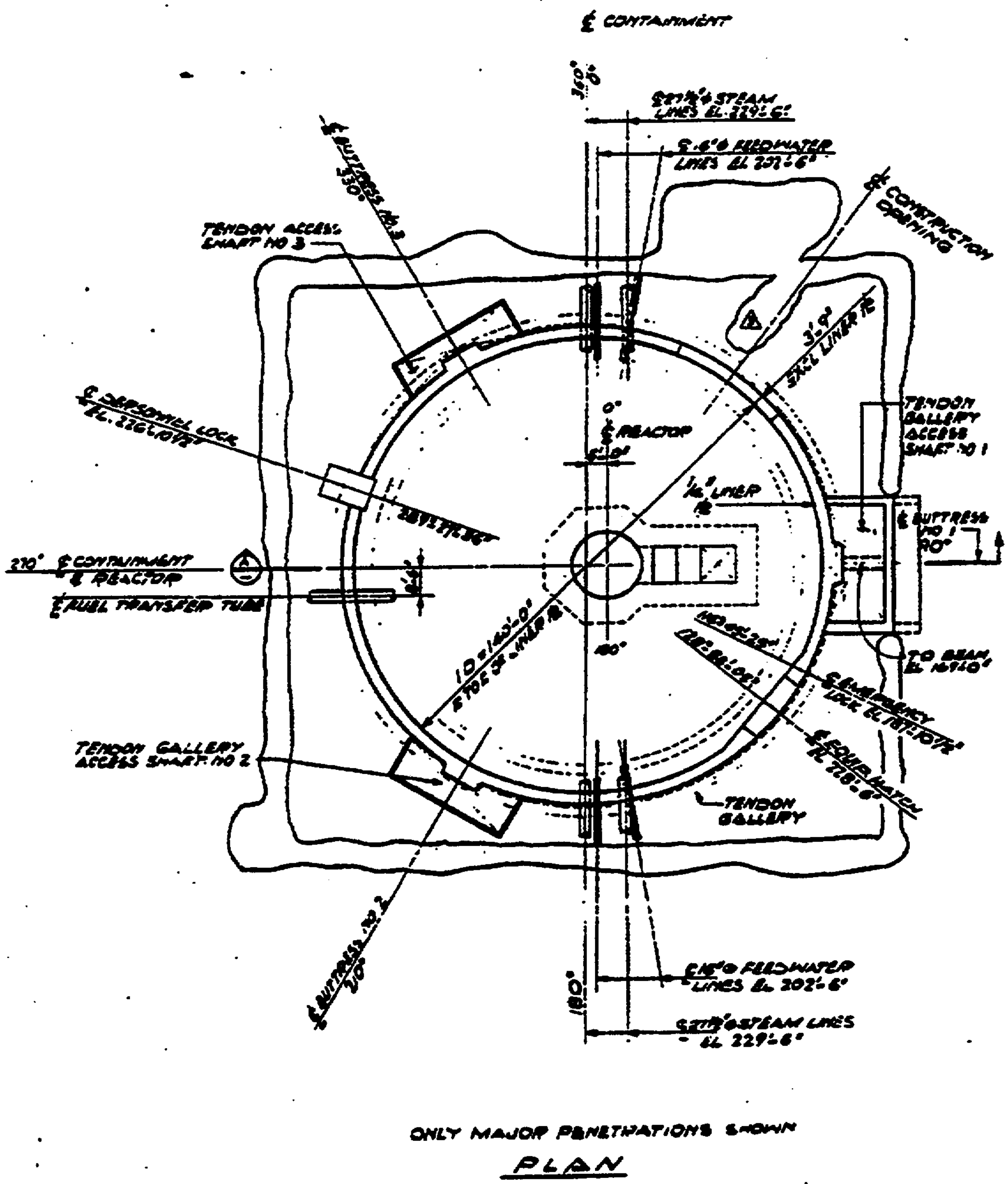


Figure 2-4 Type 3 Three-Buttress Containment

The tendons are installed in metal sheaths that form ducts through the concrete between anchorage points. Sheaths are provided with vents and drains to permit the release of trapped air during greasing and accumulated water prior to greasing. The vents and drains are sealed in the greasing operation. The sheathing filler material, grease, used for permanent corrosion protection is a modified, refined petroleum oil base product. The material is pumped into the sheathing after tendon preloading.

The vertical tendons are anchored at the top of the ring girder at the dome periphery and at the bottom of the foundation slab. The hoop tendons are anchored at only two buttresses. The three-buttress configuration has tendons spanning an arc of 240 degrees, bypassing an intermediate buttress, while the tendon for the six-buttress configuration spans an arc of 120 degrees, also bypassing an intermediate buttress. Horizontal and vertical tendons are continuous and are bent to curve around major penetrations.

Vertical - Type 3b

The cylinder walls are concrete reinforced circumferentially and post-tensioned prestressed vertically. Reinforcement includes hoop reinforcing steel and vertical tendons post-tensioned to a value sufficient to ensure there will be no tensile stresses in the concrete due to membrane forces when design load combinations are applied. The tendons are composed of a number of high-strength steel bars or steel wire.

The prestressing system for post-tensioning the containment structure for the vertical direction consists of a number of tendons placed at intervals around the periphery of the containment at the cylinder wall centerline. Each tendon is sheathed with 6-inch galvanized steel pipe or some type of galvanized corrugated steel tubing or conduit. Ducts or ends are capped or sealed at the lower end or at both ends. Corrosion protection of post-tensioned steel tendons is provided by filling the ducts or sheaths housing the tendons with portland cement grout or microcrystalline petrolatum containing organic-based corrosion inhibitors. Use of non-grouted, post-tensioned steel tendons is prevalent in U.S. nuclear plant construction with grouted tendons applied only at one commercial Westinghouse plant, H. B. Robinson Unit 2. A tendon may consist of a number of 1/4-inch or 1/2-inch diameter steel wires or a single round steel bar. The tendons are anchored to a bearing plate. Round steel bar tendons are anchored to the bearing plate using a grip nut, while a buttonhead anchorage system is applied for the wire tendons. A grip nut is a modified positive action wedge anchor with the adjustment capability of a threaded anchor. The buttonhead anchorage system positive anchorage and performs well when subjected to seismic loading. Button heads are formed on the wire by cold upsetting and bear upon a perforated steel anchor head.

2.3.1.4 Foundations

PWR containment foundations typically consist of a reinforced concrete mat that is supported on rock, soil, or on a deep foundation such as piles. Pile foundations have been used for relatively few PWR containments and should be addressed in the license renewal applications for those individual plants.

2.3.1.5 Leak Chase or Weld Channel System

The leak chase or weld channel system consists of a network of steel channels welded in place over weld seams on the steel liner or steel containment vessel. The system of channels can be pressurized to the design pressure for testing the leaktightness of the steel liner or steel containment welds, or for containment leak rate testing. A zoning system may be employed where sections of channels are separated by welded dams, permitting the tracing or location of the source of leakage. The channels may be applied on the steel liner, including mat, wall liner, dome, and penetrations and are provided with fittings that permit the repressurization of the system. Other plants may have the channels applied only on the base liner, which is covered with a 2-foot thick concrete pour forming the containment floor slab. The channels may or may not be accessible after construction, depending on the plant design; therefore, use is abandoned except for local leak tests near penetrations. Figures 2-5A and 2-5B show a cross-section of the leak chase components.

The channels are subject to the same aging degradation mechanisms as the steel liner or containment, mainly corrosion.

2.3.1.6 Penetrations

Mechanical

Typical mechanical piping penetrations through concrete containments are shown in Figures 2-6 and 2-7. Figure 2-6 shows single-barrier piping penetrations with a single closure between the process pipe and the containment liner. Single-barrier designs are typical of later-generation containment penetrations. Figure 2-7 shows the earlier double-barrier penetration designs. Double-barrier designs provide a permanent captive air space between the process pipe and sleeve for local leak testing of the penetration assembly.

Mechanical penetrations in concrete containments provide a support point for the piping system. The robust cylindrical wall of concrete containments typically serves as an anchor point for hypothetical pipe rupture loadings from high energy lines such as main steam and feedwater.

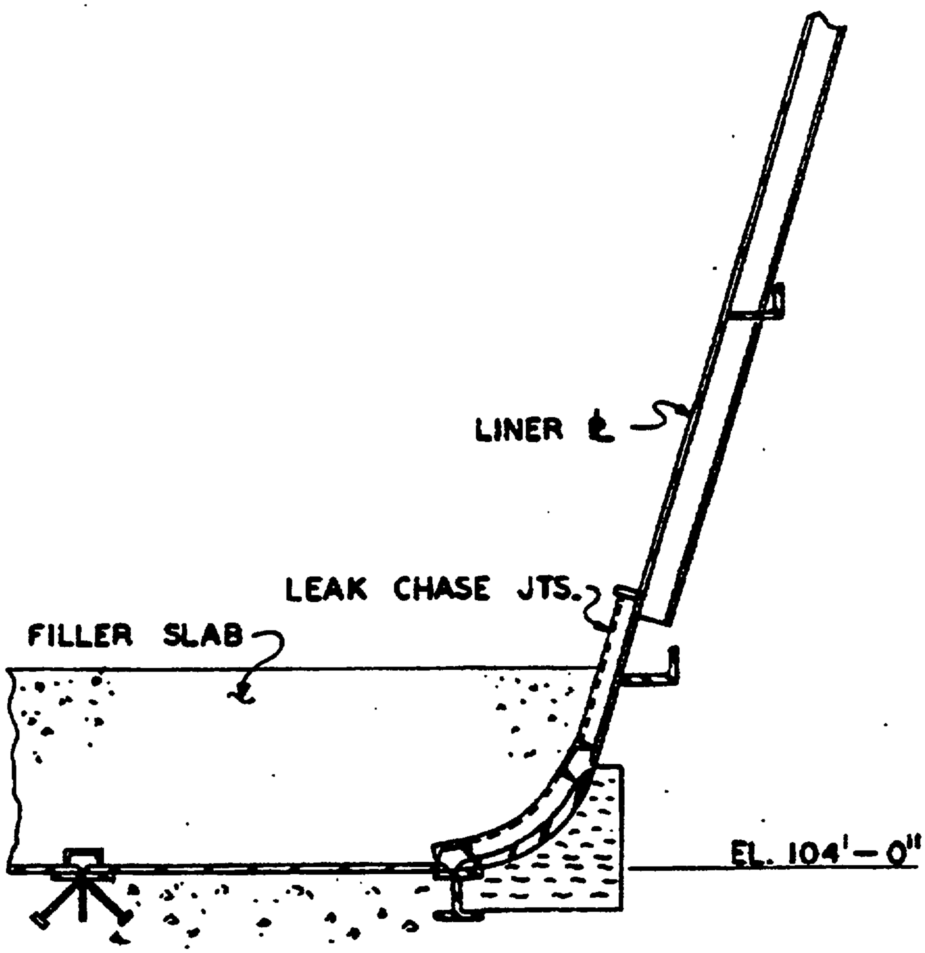


Figure 2-5A Leak Chase Component Cross-Section

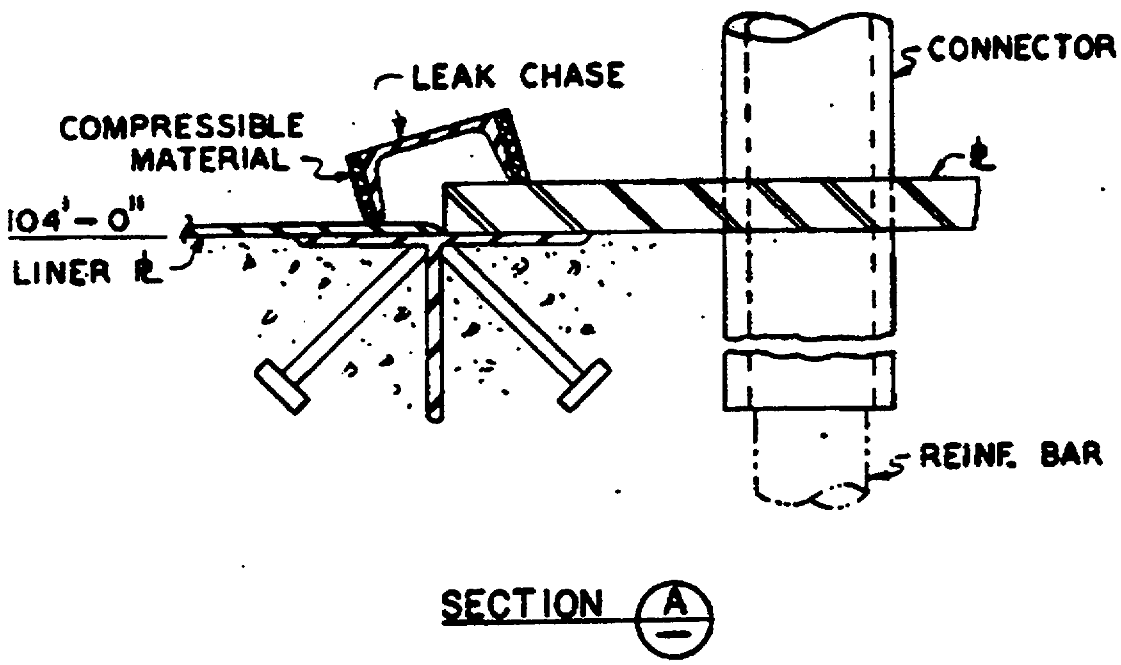
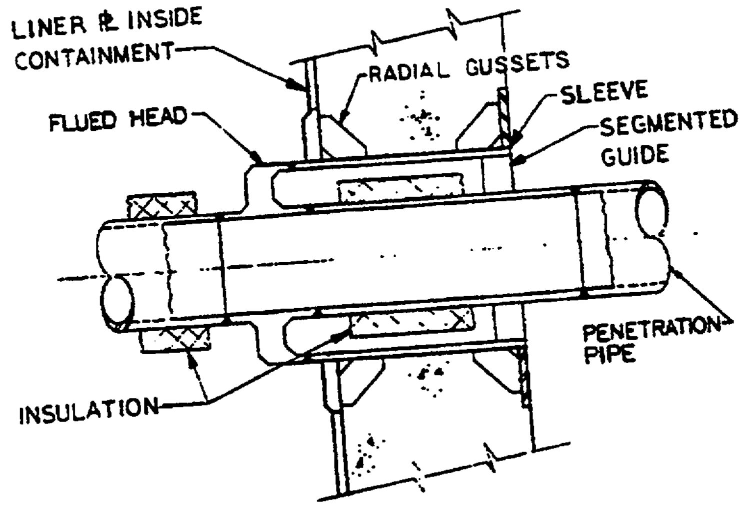
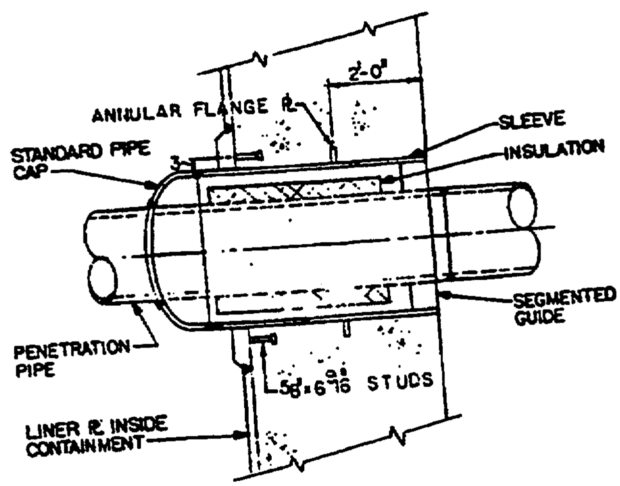


Figure 2-5B Leak Chase Component Cross-Section

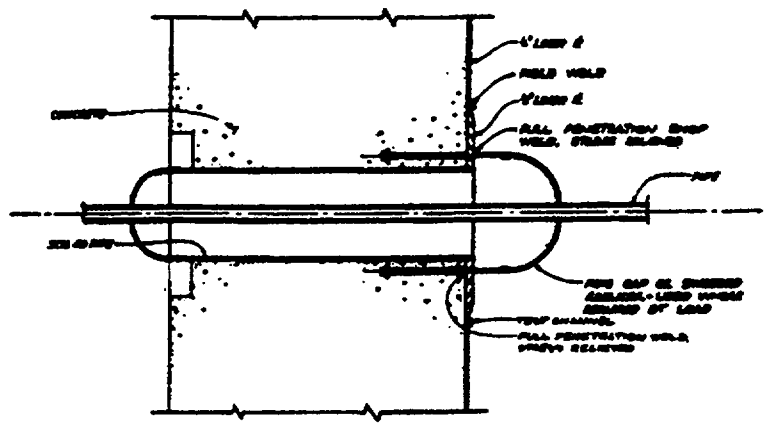


SINGLE PIPE PENETRATION
FOR HIGH ENERGY LINES

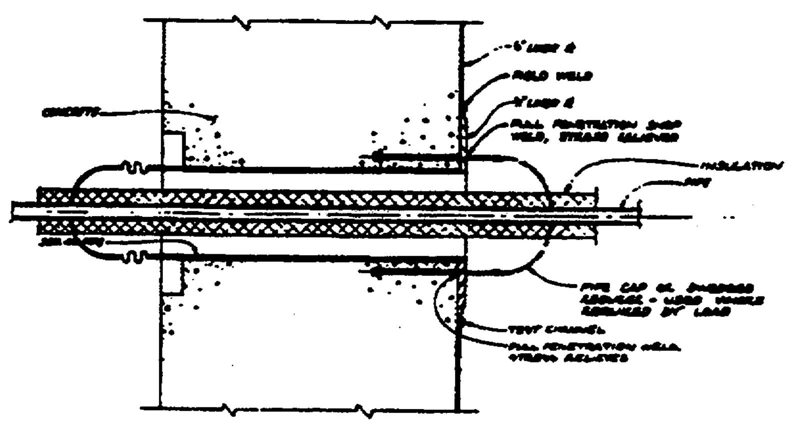


SINGLE PIPE PENETRATION
FOR MODERATE ENERGY LINES

Figure 2-6 Concrete Containment Mechanical Piping Penetrations



TYPICAL UNINSULATED PIPE PENETRATION



TYPICAL INSULATED PIPE PENETRATION

Figure 2-7 Concrete Containment Mechanical Piping Penetrations

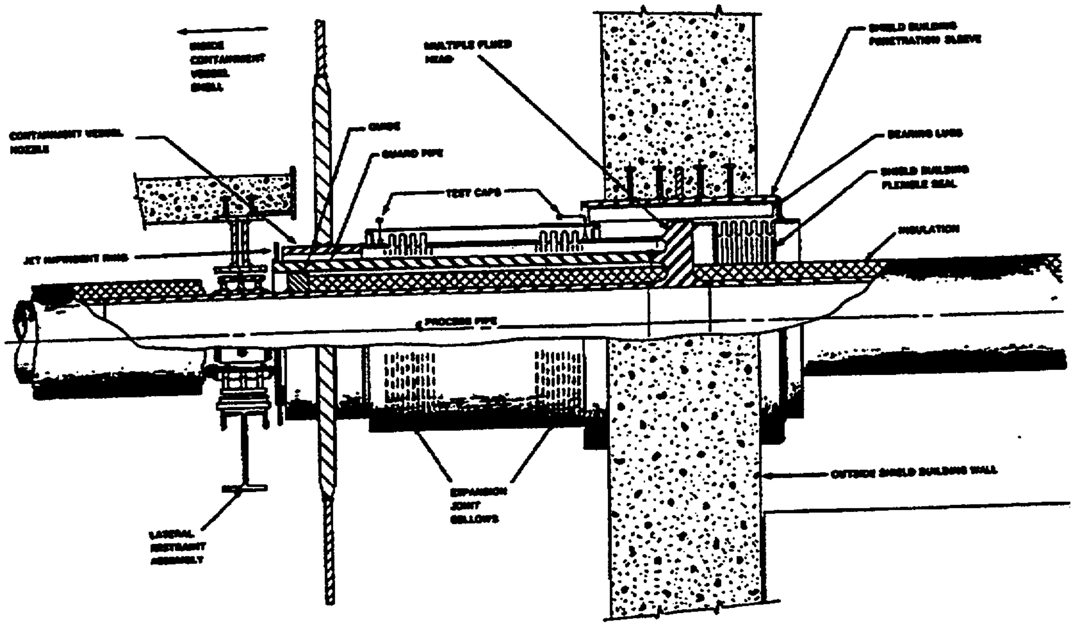
Various systems are used to classify the piping penetrations for design purposes. In general, the classification is based on whether thermal movement of the process line is expected. The piping penetrations are typically classified into high or moderate temperature service. A third type of penetration is a multiple pipe penetration, where more than one line passes through the penetration. Multiple penetrations include small-diameter tubes for sampling lines. Socket weld couplings may be welded into the penetration header plate to mate with the seamless tubing.

The mechanical penetrations, or piping penetrations, are provided for fluid carrying pipes and for air purge ventilating piping. In certain steel containment designs, vacuum breaker penetrations are provided through the steel containment and into the annulus between the containment and the shield building.

For steel containments, process lines traverse the boundary between the inner steel containment and the outer shield building by means of penetration assemblies consisting of several components, as shown in Figures 2-8, 2-9, and 2-10. Suitable details are provided to accommodate piping thermal movement. Similarly, for high temperature double-barrier penetrations, as shown in Figure 2-7 for concrete containments, one of the barriers consists of a bellows to permit unrestrained thermal growth of the pipe. The stainless steel bellows assemblies are two-ply construction and designed for a conservative number of movement cycles, such as a greater than expected number of heatup-cooldown cycles for an 80-year design life.

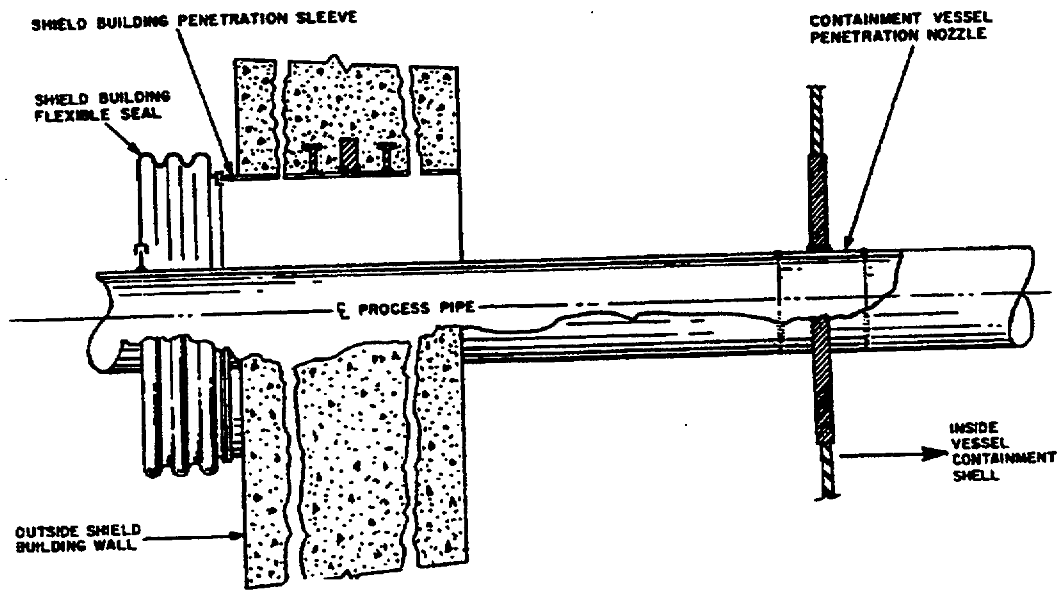
Typical mechanical piping penetrations through steel containments are shown in Figures 2-8 and 2-9. The closure between the steel containment and process pipe is provided by either a bellows assembly detail as shown in Figure 2-8 or by a direct welding of the piping to the steel containment as shown in Figure 2-9.

High-temperature penetrations such as the main steam and feedwater typically use the bellows detail because it permits relative movement between the piping and the steel containment so that relatively high support loads from large bore lines are taken by the support at the more robust concrete shield building wall rather than by the steel containment. The bellows closure provides the necessary pressure seal but does not physically support the pipe. This arrangement permits differential seismic and thermal movements between the steel containment, concrete shield building, and containment interior structures without impacting the containment function.



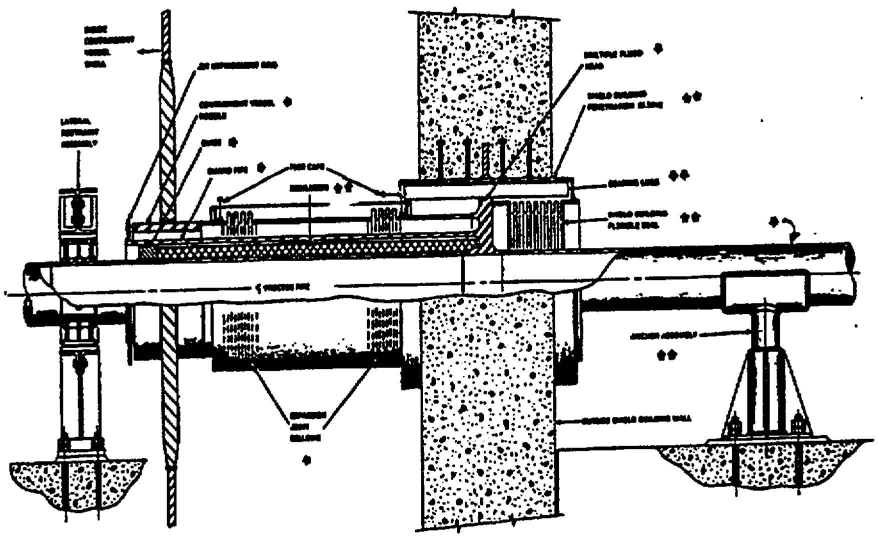
Main Steam and Feedwater Piping Penetrations Assembly

Figure 2-8 Steel Containment Mechanical Penetrations



Cold Piping Penetrations Assembly

Figure 2-9 Steel Containment Mechanical Properties



* IN SCOPE OF CONTAINMENT GTR

** NOT IN SCOPE OF CONTAINMENT GTR

Figure 2-10 Steel Containment Residual Heat Removal Penetrations

Penetration closures for smaller lines and cold lines may be provided by direct welding of the pipe to the steel containment as shown in Figure 2-9. This detail provides both pressure sealing and physical support. In this case, a bellows assembly is provided at the concrete shield building to accommodate the differential building movements. Bellows assemblies may also be used for the closure to the steel containment for the cold and smaller lines. However, the direct welding detail shown in Figure 2-9 is typically provided, where feasible, because the rigid-plate-to-pipe weld provides a more reliable seal at the primary containment boundary than bellows details.

Mechanical (piping) penetrations for concrete containments may be classified according to arrangement as follows:

- Single-Barrier Penetrations

  Single-barrier penetrations are shown in Figure 2-6. This arrangement provides a single "rigid" closure between the containment liner and the process pipe. The closure may consist of a flued head forging, a standard pipe cap, or a plate with a segment of sleeve pipe.

- Double-Barrier Penetrations

  The arrangement of double-barrier piping penetrations for concrete containments is shown in Figure 2-7. Two typical arrangements are used depending on the design temperature for the process pipe.

Electrical

An electrical penetration through a concrete containment is shown in Figure 2-11.

Electrical-type penetrations through steel containments are shown in Figures 2-12 and 2-13.

The scope of this evaluation includes the metallic components of the typical electrical penetration that are part of the containment pressure boundary. A typical cable feed-through module is shown in Figures 2-12 and 2-13. There are generally four types of electrical cable penetrations required in the containment depending on the type of cable involved:

- Medium voltage power, 4160 V or 6.9 kV
- Low voltage power, control and instrumentation, 600 V and lower
- Thermocouple leads
- Special instrumentation coaxial and triaxial circuits

Typically, medium and low voltage electrical penetrations consist of carbon steel pipe canisters with stainless steel header plates bolted to one or both of the ends. Identical, hermetically sealed ceramic multi-pin connectors are mechanically connected to the header plate(s) for all conductors rated less than 600 volts. High voltage conductors use single-conductor, hermetically sealed ceramic bushings, also mechanically connected to both header plates. A flange on each canister is welded to the penetration sleeve. The electrical penetration assembly permits pressure and leakage tests to be performed at the shop and after installation in the containment. A tap, convenient to the exterior of the containment, is provided for pressurizing the canister. The terminations of the conductors to the connectors inside the canisters are potted to protect against moisture. A plug is provided to permit purging with dry nitrogen.

Fuel Transfer Tube

Penetration of the reactor building by any means compromises the containment and requires measures to be taken to ensure closure during an accident or unexpected leaks. The fuel transfer tube penetrates the reactor building and links the refueling canal in the reactor building with the fuel transfer canal in the fuel handling building. Therefore, the design of the fuel transfer tube and its penetration of the reactor building must ensure that a breach of the building is not possible under the spectrum of accident and loading conditions.

A fuel transfer tube penetration arrangement through a concrete containment is shown in Figure 2-14. As shown in the figure, the closure between the transfer tube and the sleeve that is welded integrally into the containment liner typically consists of a circular plate shop-welded to the tube and a short segment of pipe to mate with the sleeve. All welds in the closure between the tube and liner are full-penetration welds except for the flange adaptor ring welded to the end of the tube. The blind flange adaptor ring is connected to the transfer tube pipe by two separate continuous partial penetration welds. For older vintage plants, the bimetallic transition weld between the stainless transfer tube and the carbon steel plate or pipe segment was shop-welded for better control of quality. For later plants, with improved welding materials and techniques, the transition weld was made in either the shop or field. For double-barrier designs, a welded canopy is placed over the connecting hardware between the tube and liner.

A fuel transfer tube penetration arrangement through a steel containment is shown in Figure 2-15. The closure between the tube and steel containment in most cases is basically the same as for the concrete containment: a rigid connection to a sleeve shop-welded into the steel containment. In some designs, the closure to the steel containment vessel may consist of a flexible bellows assembly.

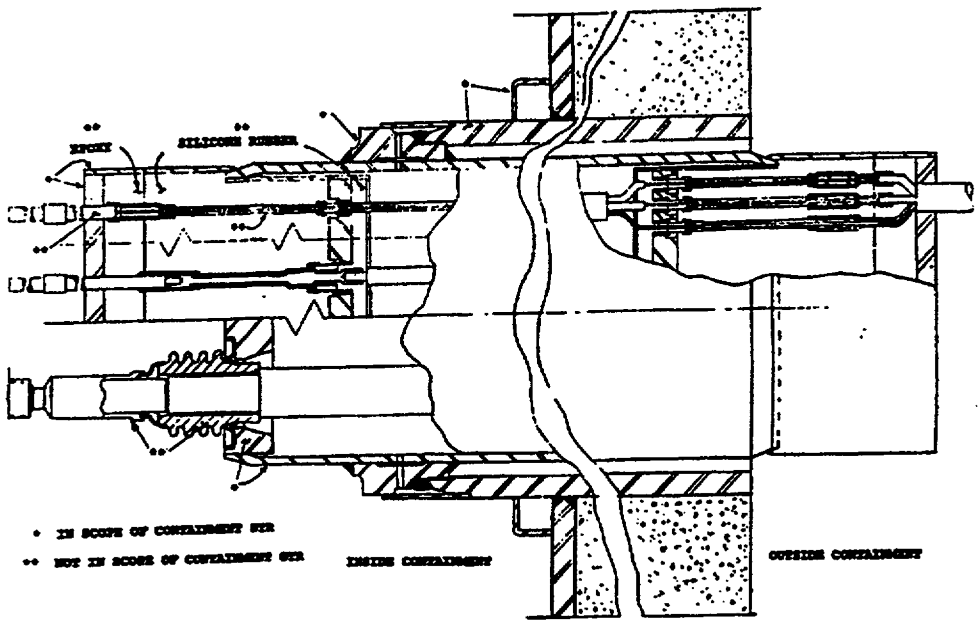


Figure 2-11 Concrete Containment Electrical Penetrations

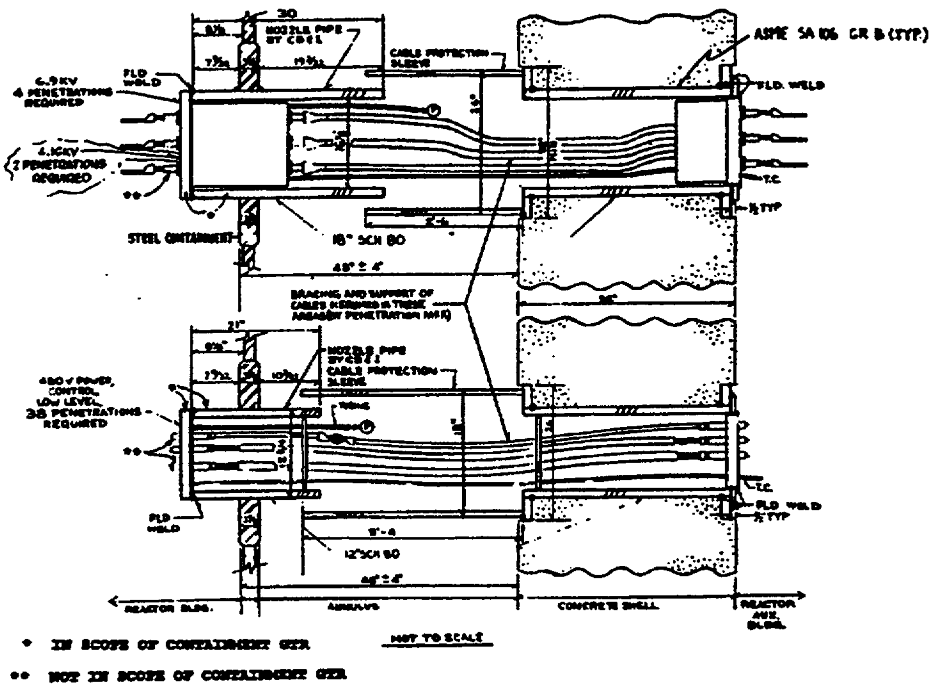


Figure 2-12 Steel Containment Electrical Penetrations

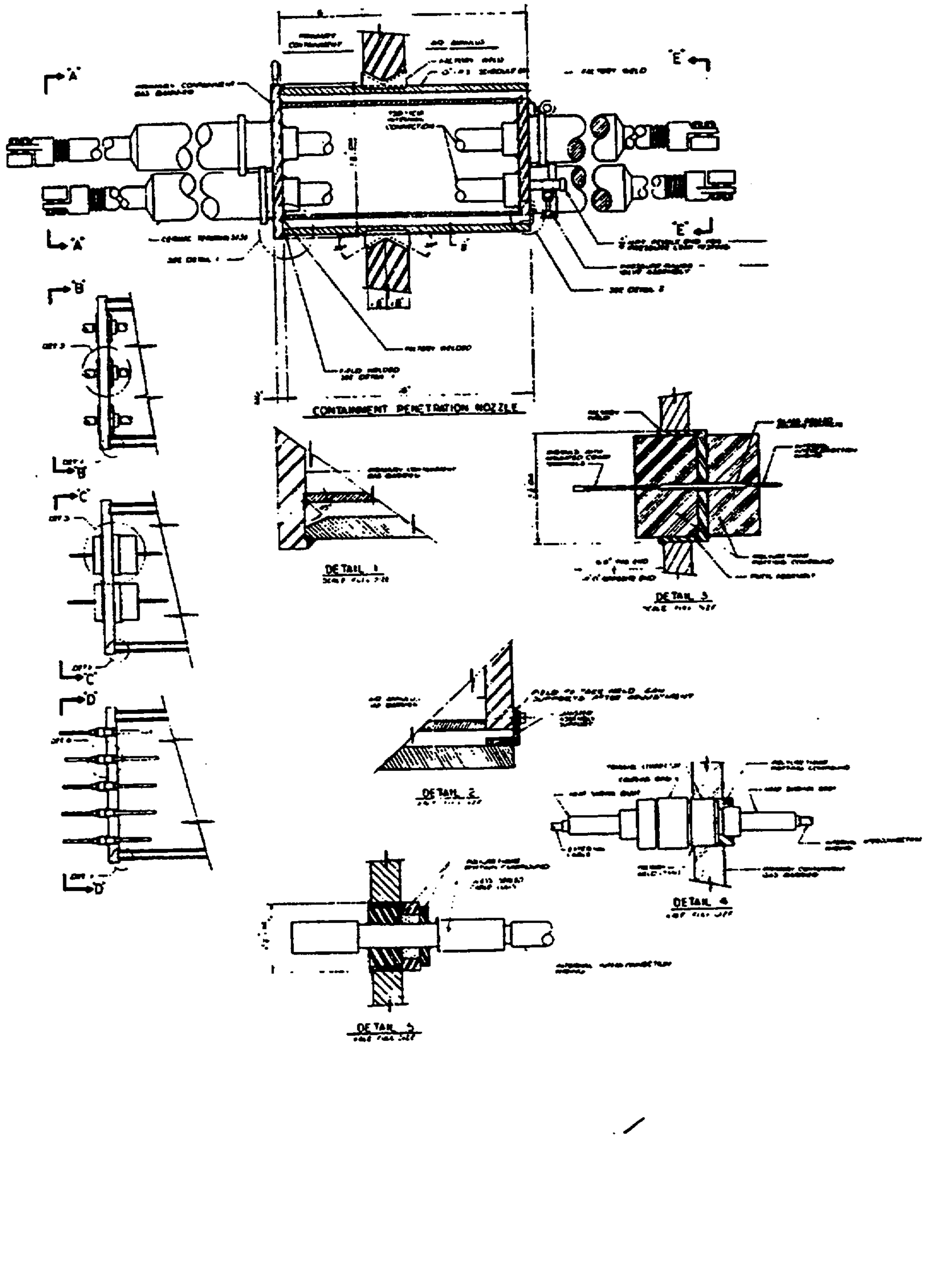


Figure 2-13 Steel Containment Electrical Penetrations

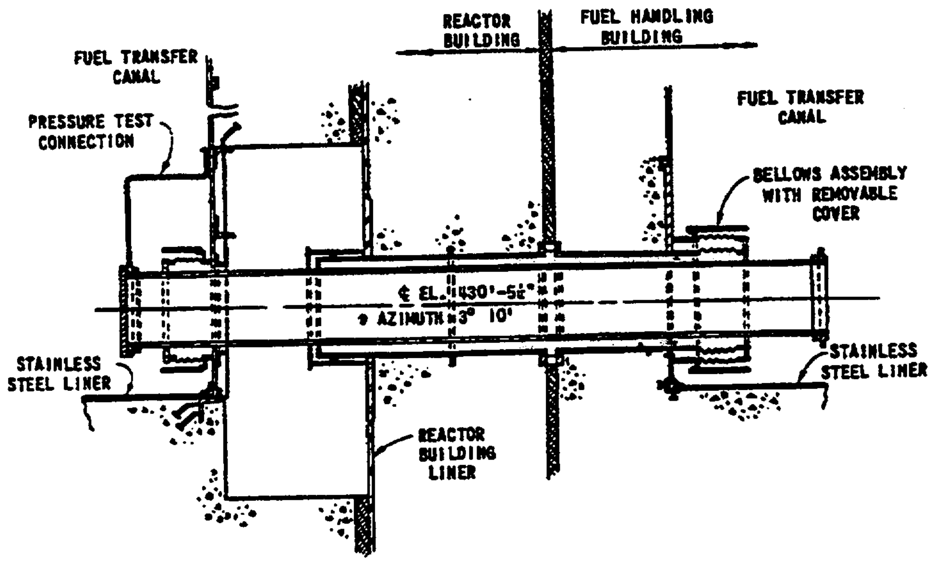


Figure 2-14 Concrete Containment Fuel Transfer Tube

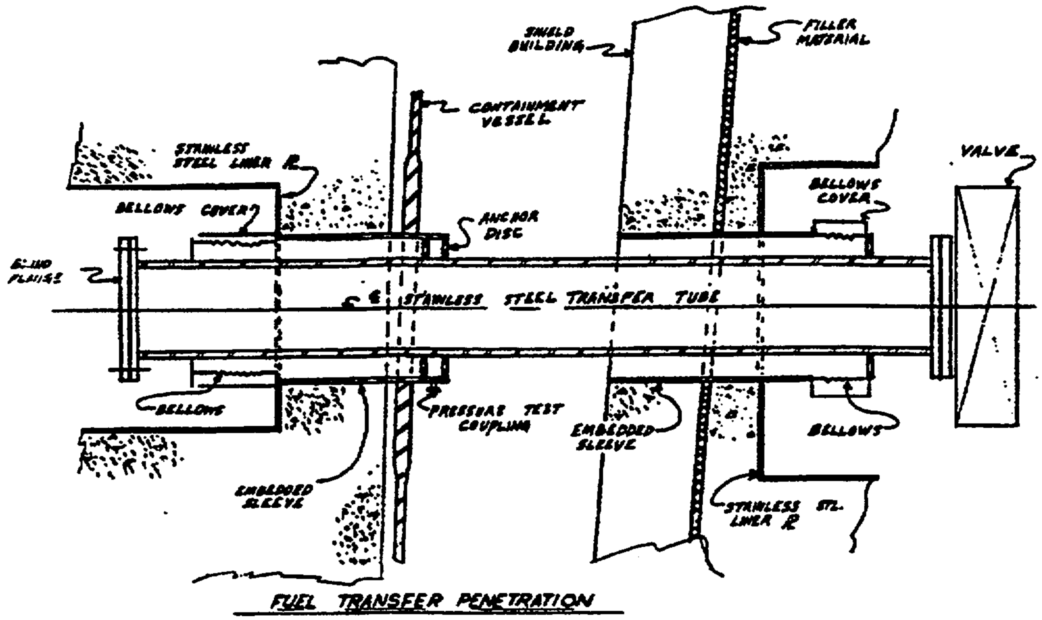


Figure 2-15 Steel Containment Fuel Transfer Tube

During normal operation, the fuel transfer tube penetration is closed and serves as part of the containment pressure boundary. The closure at the pressure boundary end inside containment typically consists of a blind flange with double seals. The closure on the fuel handling building end outside the containment generally consists of a gate valve supported from the end of the transfer tube.

The fuel transfer tube also provides support for the transfer mechanisms and the fuel assemblies against natural hazards, such as earthquakes, that are postulated to occur during transfer operations.

Residual Heat Removal

A residual heat removal (RHR) penetration is shown in Figure 2-16. The blind flange shown at the end inside containment is used for periodic local leak testing of the penetration. The blind flange is removed during normal plant operation. The containment system boundary extends out to and includes the chamber surrounding the isolation valve. The boundary is at the connection of the penetration sleeve to the chamber surrounding the isolation valve and at the weld of the RHR piping within the penetration to the isolation valve.

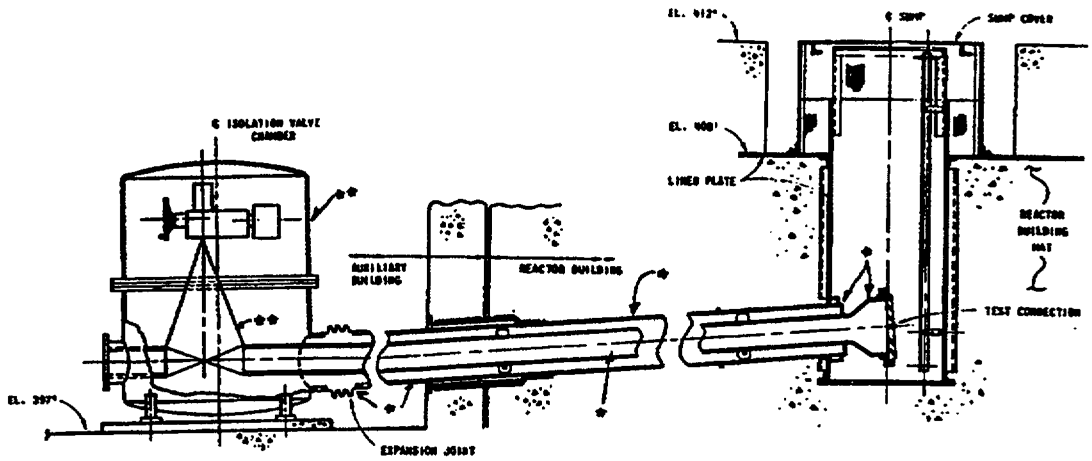
Spare Penetrations

Spare penetrations generally consist of a sleeve with welded end cap closures. A spare penetration through a concrete containment is shown in Figure 2-17. The arrangement and details are basically the same for steel containments except that the sleeve is welded to the steel containment plate rather than to the concrete containment liner. For some plants, spare penetrations that are used only during outages may incorporate bolted blind flanges with flexible seals for ease of removal and replacement.

2.3.1.7 Personnel Airlocks

Typically, two access airlocks are provided at two different floor levels of the containment for normal and emergency ingress and egress.

A typical personnel airlock through a concrete containment is shown in Figure 2-18. The airlock consists of a cylindrical barrel section with leaktight doors at each end. The airlock is supported by the concrete containment wall. Airlocks through steel containments are basically the same with the exception that they may be supported by the concrete shield building rather than the steel containment. The leaktight closure between the steel containment and the airlock barrel is by a flexible connection such as a bellows assembly.



* IN SCOPE OF CONTAINMENT GTR

** NOT IN SCOPE OF CONTAINMENT GTR

Figure 2-16 Concrete Containment Residual Heat Removal Penetrations

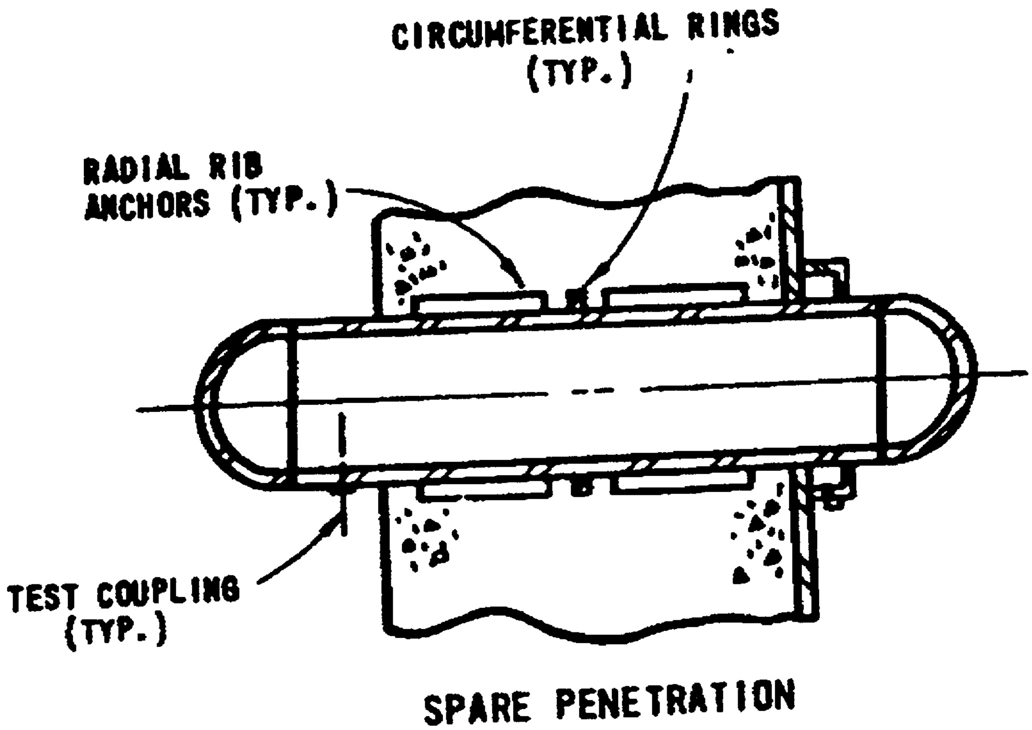


Figure 2-17 Concrete Containment Spare Penetrations

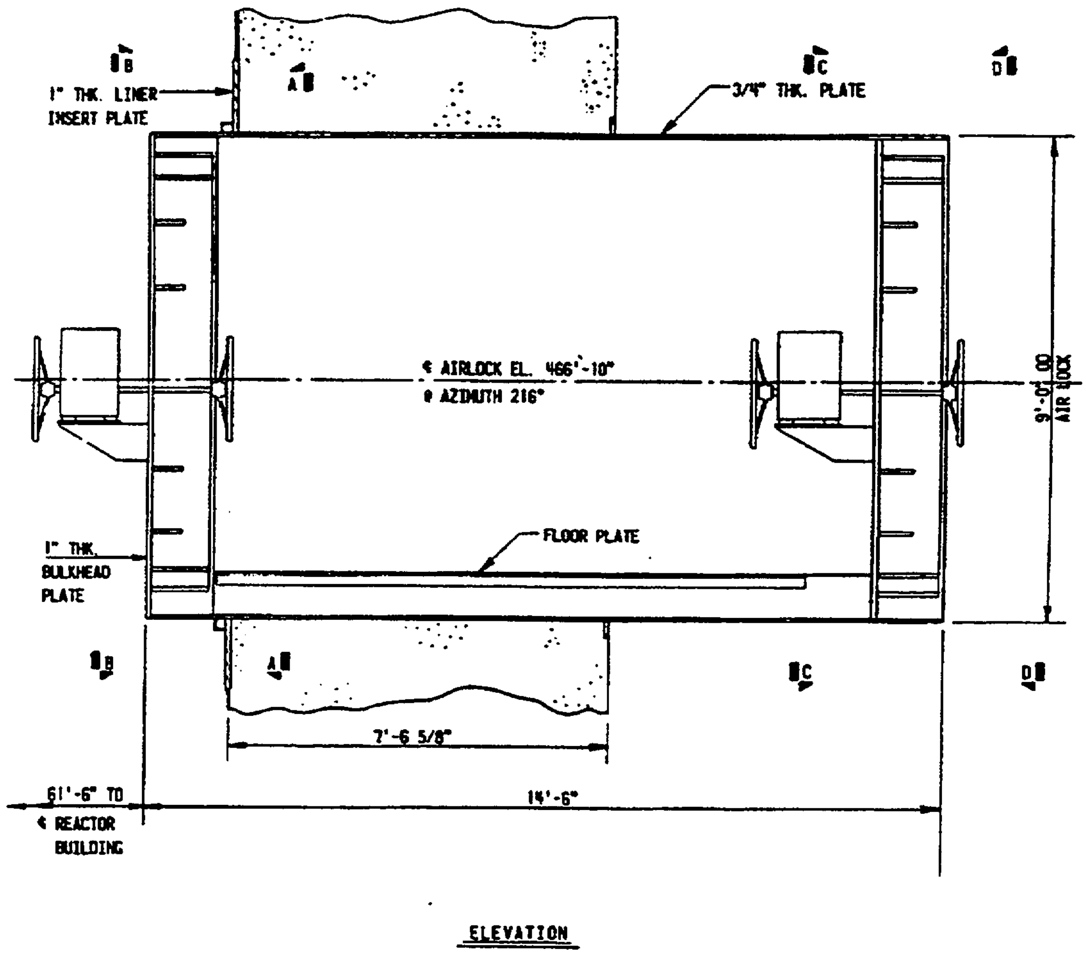


Figure 2-18 Concrete Containment Personnel Airlock

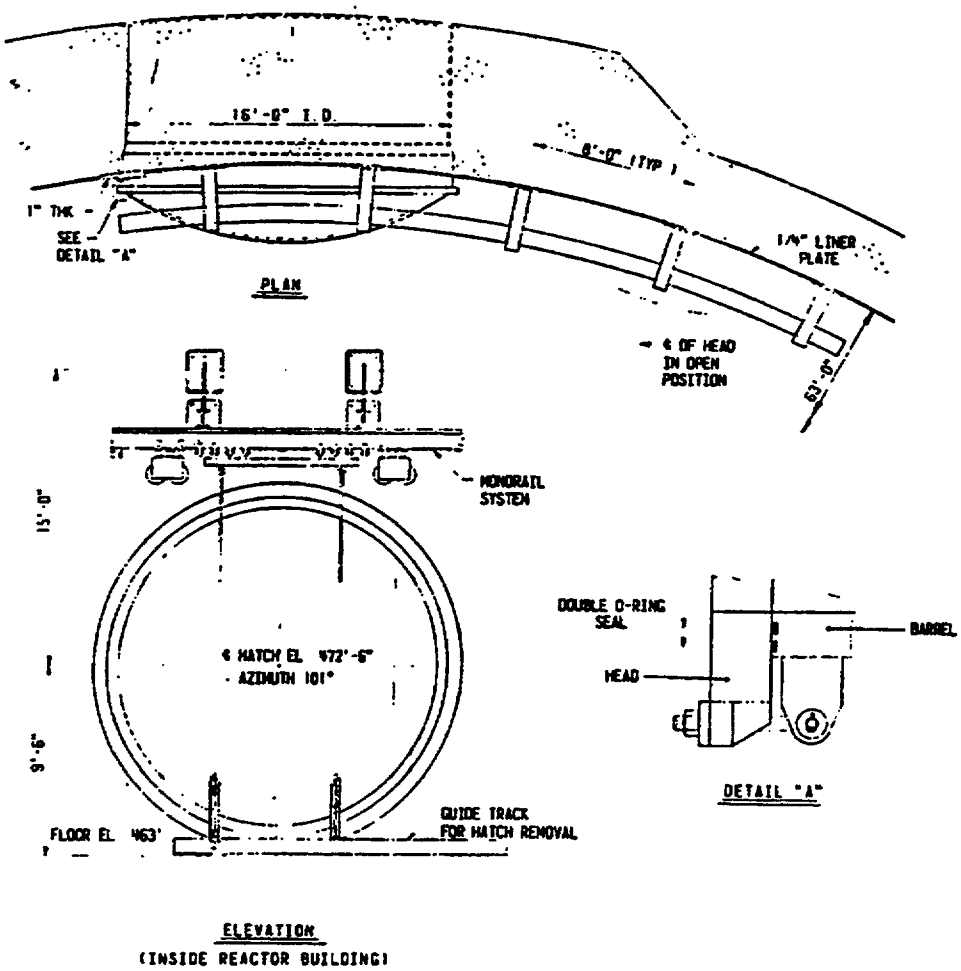


Figure 2-19 Concrete Containment Equipment Hatch

2.3.1.8 Equipment Hatch

An equipment hatch arrangement is shown in Figure 2-19. Supplementary equipment such as a monorail and hoist system are usually associated with the equipment hatch to permit efficient usage of the large opening. In later-generation plants, the hatch diameter is typically sized to permit removal and replacement of a steam generator. In the older plants, the diameter is adequate to support most maintenance activities but does not permit passage of an intact steam generator.

In some cases, a smaller diameter emergency access personnel airlock is incorporated into the equipment hatch head, providing a dual function design.

2.3.1.9 Access Airlocks

Generally, two airlocks are provided into the containment for personnel ingress and egress. For some containments, two different size airlocks are used. The larger airlock is typically used as the primary access and the smaller airlock for emergency access. In some containments, one of the personnel airlocks penetrates the dished cover of the equipment hatch.

Personnel airlocks typically consist of a double door, welded steel assembly as shown in Figure 2-18. The airlocks are designed to withstand all containment design conditions with either or both doors closed and locked. The doors open toward the center of the containment and are thus sealed under containment pressure. The airlock may be pressurized to demonstrate leaktightness without pressurizing the containment. Each airlock is pneumatically tested in the shop for pressure and leakage. Quick-acting equalizing valves connect the personnel lock with the interior and exterior of the containment vessel for the purpose of equalizing pressure in the two systems when entering or leaving the containment.

The two personnel airlock doors are interlocked to prevent both being opened simultaneously and to ensure that containment is always maintained by one door being completely closed before the other door can be opened. The interlocking system has the capability to be bypassed allowing the doors to be left open during plant cold shutdown. In most cases operation of the lock is manual, without power assist.

Each airlock door is provided with flexible seals. The arrangement and type of seals varies. In most cases, double gasket seals that have provisions for local leakage testing between the seals or inflatable seals are provided.

2.4 ENGINEERING AND DESIGN DATA

Engineering and design data important to the aging evaluation of the PWR containment structure are presented. These include: codes and standards, materials, design pressures and temperatures, and design loading.

2.4.1 Containment Structural Codes and Requirements

The applicable codes and requirements for the containment structure are a function of the containment design and vintage. Basically, three codes are employed to define design criteria, depending on the construction characteristics of the structure:

Concrete	-	American Concrete Institute (ACI)
Vessel	-	American Society of Mechanical Engineers (ASME)
Structural Steel	-	American Institute of Steel Construction (AISC)
Welding	-	American Welding Society (AWS) and ASME

Different code editions are used based on the plant vintage. Tables 2-4 to 2-6 provide a summary of the applicable codes for each of the plants.

2.4.2 Penetrations

Table 2-7 lists the codes and standards applicable to the containment penetrations including the airlocks, equipment hatch, fuel transfer tube, and mechanical and electrical penetrations.

2.4.3 Containment Structural Design Data

Various materials are used in the construction of the containment structure. They are defined in Tables 2-8 to 2-10. Materials are given for seven basic areas of the containment structure:

- Airlock/hatch
- Liner plate
- Reinforcement steel
- Structural steel
- Pre-stress or post-tension tendons
- Tendon sheathing
- Penetration pipe sleeve

TABLE 2-4
CONTAINMENT DESIGN CODES

Plant Name	ACI	ASME	Structural Code	Welding
Robinson 2	318-63	Sect III	AISC 1963	AWS D1.1-76
Shearon Harris	359 318-1971	Sect III Div 1 & 2-1975	AISC 1969	AWS D1.1-75
Braidwood 1 & 2	318-71,77,80	Sect III Div 1 & 2-72	AISC 1969	AWS D1.1
Byron 1 & 2	318-71,77,80	Sect III Div 1 & 2-72	AISC 1969	AWS D1.1
Zion 1 & 2	318-63	Sect III Div 1-65 Sect VIII	AISC 1963	ASME Sect IX
Haddam Neck	318-63	ASME Followed	AISC 1963	Not Available
Indian Point 2	318-63	Sect III Div 1	AISC 1963	ASME Sect VIII
Indian Point 3	318-63	Sect III Div 1	AISC 1963	Not Available
Catawba 1 & 2	318-71	Sect III Div 1-71	AISC 1963	Not Available
McGuire 1 & 2	318-63	Sect III Sub B-68	AISC 1969	Not Available
Beaver Valley 1 & 2	318-71	Sect III	AISC 1969	AWS D1.1-72
Turkey Point 1 & 2	318-63	Sect III Sub B Sect VIII	AISC 1963	Not Available
South Texas 1 & 2	359	Sect III Div 1	AISC 1969	AWS D1.1-75
Donald C. Cook 1 & 2	318-63	Sect III-1968	AISC 1963	ASME Sect VIII

TABLE 2-5
CONTAINMENT DESIGN CODES

Plant Name	ACI	ASME	Structural Code	Welding
Seabrook	318-71	Sect III Div 1 & 2 - 75	AISC 1963	ASME Sect II Part C
Millstone 3	318-71	Sect III Div 1 & 2 - 71	AISC 1969	AWS D1.1-72, 73, 79
Prairie Island 1 & 2	318-63	Sect III Div 1-77	AISC 1963	AWS D1.0
Diablo Canyon 1 & 2	318-63	Sect III Sect VIII-68	AISC 1969	AWS D1.0-66
Salem 1 & 2	318-63	Sect III-68	AISC 1969	Not Available
R. E. Ginna	318-63	Sect III & VIII	AISC 1963	ASME Sect IX
Virgil C. Summer	318-71	Sect III Div 1 & 2	AISC 1969	AWS D1.1-72
James M. Farley 1 & 2	318-63	Sect III-68	AISC 1969	AWS D2.0
Alvin W. Vogtle 1 & 2	318-71	Sect III Div 2 1977	AISC 1969	Not Available
Sequoyah 1 & 2	318-63	Sect III Sub B-1968	AISC 1969	AWS D1.1-72

TABLE 2-6
CONTAINMENT DESIGN CODES

Plant Name	ACI	ASME	Structural Code	Welding
Watts Bar 1 & 2	359 318-71	Sect III-Div 1	AISC 1969	AWS D1.1/2-74
Comanche Peak 1 & 2	359	Sect III Div 1 & 2	AISC 1969	AWS D12.1-61
Callaway	318-71	Sect III-74	AISC 1969	AWS D1.1-75
North Anna 1 & 2	349-80, 318-71	Sect III & VIII-69	AISC 1978	Not Available
Surry 1 & 2	318-63 Part IV-B	Sect III-68	AISC 1963	AWS D12.1
Point Beach 1 & 2	318-63	Sect III & VIII	AISC 1963	ASME IX
Kewaunee	318-63	Sect III Sub Sect B	AISC 1963	ASME Sect IX
Wolf Creek	318-71 349-80	Sect III Div. 1, Div. 2, 1979	AISC 1969	AWS Sect D1.1-75

TABLE 2-7
CONTAINMENT PENETRATIONS CODES AND STANDARDS

American Nuclear Society

- ANS 7.60 - Proposed Standard for Leakage Testing of Containment Structures (July 14, 1967)
- AEC Technical Safety Guide 7.5.1, "Reactor Containment Leakage Testing and Surveillance Requirements" (December 15, 1966)

American National Standards Institute

- ANSI N5.12 B Protective Coatings (Points) for the Nuclear Industry
- ANSI N6.2 B Safety Standard for Design, Fabrication, and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors
- ANSI N45.4 B Leakage Rate Testing of Containment Structure for Nuclear Reactors
- ANSI N101.2 B Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities
- ANSI N101.4 B Quality Assurance for Protective Coatings to Nuclear Facilities

American Society of Mechanical Engineers

- ASME Boiler and Pressure Vessel Code

  Section II - Material Specifications Parts A and C
  Section V - Non-Destructive Examination
  Section VIII - Unfired Pressure Vessels
  Section IX - Welding Qualifications
  Section III - Nuclear Vessels (Applicable to Code Class B for older plants).
  Section III - Nuclear Plant Components, Division 1 (Applicable to Class MC Components)
  Section III - Nuclear Plant Components, Division 2 (Applicable to Concrete Reactor Vessels and Containments)

TABLE 2-7 (Continued)
CONTAINMENT PENETRATIONS CODES AND STANDARDS

Code of Federal Regulations, Title 10, Part 50
• Appendix J Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors
• USASI B31.1.0, Code for Pressure Piping
Institute of Electrical and Electronics Engineers
• IEEE 317 – Standards for Electrical Penetration Assemblies in Containment Structures for Nuclear Generating Stations
• IEEE - Guide for Electrical Penetration Assemblies in Containment Structures for Stationary Nuclear Power Reactors
• IEEE - Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generation Stations (April 1971)
United States Nuclear Regulatory Commission
• Regulatory Guide 1.19 - Nondestructive Examination of Primary Containment Liner Welds
• Regulatory Guide 1.57 - Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components
• Regulatory Guide 1.63 - Electrical Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants
• Regulatory Guide 1.163 – Performance Based Containment Leak – Test Program

TABLE 2-8
CONTAINMENT MATERIALS

Plant Name	Airlock/Hatch	Liner Plate	Reinf. Steel	Structural Steel	Tendons-ASTM	Tendon Sheathing	Penetration Pipe Sleeve
Robinson 2	A516 Gr 55	A442 Gr 60	A40; A432-65	A36	A322-64; A29	A53	A333 Gr 1 A516 Gr 60
Shearon Harris	SA516 Gr 70	A516 Gr 70	A615 Gr 60	A36	Not used	Not used	A333 Gr 6 SA155 Gr 1
Braidwood 1 & 2	A516 Gr 60	A516 Gr 60	A615 Gr 60	A36	A421 TP BA	22ga A53 Gr B	A333 Gr 1 or 6 A516 Gr 60
Byron 1 & 2	A516 Gr 60	A516 Gr 60	A615 Gr 60	A36	A421 TP BA	22ga A53 Gr B	A333 Gr 1 or 6 A516 Gr 60
Zion 1 & 2	SA333 Gr 1	A442 Gr 60	A615 Gr 60	A36	A421	24ga A366	A333 Gr 1
Haddam Neck	A201 Gr A & B	A442 Gr 60	A408	A36	Not used	Not used	A333 Gr 1
Indian Point 2	A300 Cl 1 A516 Gr 60	A442 Gr 60-65	A432 Gr 60 A408	A36	Not used	Not used	A333 Gr 1 A442 Gr B
Indian Point 3	A300 Cl 1; A201 Gr B	A442 Gr 60-65	A432 Gr 60-65	A36	Not used	Not used	SA333 Cl 1 A201 Gr B
Catawba 1 & 2	SA516 Gr 60	SA516 Gr 60	A615 Gr 40 & 60-72	A36; SA306; SA588	Not used	Not used	SA333 Gr B SA516 Gr 60
McGuire 1 & 2	Sect III Sub B-68	SA516 Gr 60	A615 Gr 40 & 60	A36	Not used	Not used	Not Available
Beaver Valley 1 & 2	SA537 Gr B	SA537 Gr B	A615 Gr 40, 50, 60	A36-67	Not used	Not used	SA537 Gr B
Turkey Point 1 & 2	A516	A36; A442	A615 Gr 40 & 60	A36	A421 Tp BA	24 ga --	A333 A155 Cl 1

TABLE 2-9
CONTAINMENT MATERIALS

Plant Name	Airlock/Hatch	Liner Plate	Reinf. Steel	Structural Steel	Tendons-ASTM	Tendon Sheathing	Penetration Pipe Sleeve
South Texas 1 & 2	SA516 Gr70	SA285 Gr A SA516 Gr 60	SA615-72 Gr 60	SA36	SA421-77-BA	A529	SA333 Gr 6 SA155 Gr KCF
Donald C. Cook 1&2	A516 Gr70	A442 Gr 60	A615 Gr 40	A36-67	Not used	Not used	A333 Gr 6
Seabrook	SA516 Gr 60 & 70	SA516 Gr 60	A615 Gr 60	SA36	Not used	Not used	SA240 Type 304
Millstone 3	SA516 Gr 60	SA537 C1;1, C1.2	SA516 Gr 40	A36	Not used	Not used	SA537 Gr B SA516 Gr 60 SA333 Gr 6
Prairie Island 1 & 2	SA516 Gr 70	SA516 Gr 70	Not available	Not available	Not used	Not used	SA333 Gr 6
Diablo Canyon 1 & 2	A516 Gr 70 and A300	A516 Gr 70	A615 Gr 40 or 60	A36	Not used	Not used	A106 Gr 6 A516 Gr 70
Salem 1 & 2	A-516 Gr 60	A442-66 Gr 60	A432-65 A615 Gr 40	A36	Not used	Not used	A155 KC-70 Cl 1
R. E. Ginna	A201-61T Gr B A300-58	A442-60T Gr 60	A15-64 A408-62T	A36-63T	A421-59T Tp BA	A53R A106	A201-61T Gr B A300-58
Virgil C. Summer	SA516 Gr 70	SA516 Gr 60	A615 Gr 70-72	Ae6 SA36	A421	A53-729 Type S, Gr B	SA333 Gr 6 SA155 Gr KCF 70
Joseph M. Farley	SA516 Gr 70	A285 Gr B	A615 Gr 60	A36	A421-65 Type BA	22g corrugated tubing	SA333 Gr 6
Alvin W. Vogtle 1 & 2	SA516 Gr 70	A516 Gr 70 SA285	A615 Gr 60	A36	A416	A527SA 333 Gr 1 & 6	SA516 Gr 70 SA333 Gr 1 & 6

TABLE 2-10
CONTAINMENT MATERIALS

Plant Name	Airlock/Hatch	Liner Plate	Reinf. Steel	Structural Steel	Tendons-ASTM	Tendon Sheathing	Penetration Pipe Sleeve
Sequoyah 1 & 2	A516 Gr 60 & 70	A516 Gr 60 & 70[1]	A615-68 Gr 60	A36	Not used	Not used	A36
WATTS Bar	A36	SA516 Gr 70[1]	A615-72 Gr 60	A36	Not used	Not used	Not available
Comanche Peak 1 & 2	A516 Gr 70-74 SA537 Cl 2-74	SA537 Cl 2-74	A615-72	A36	Not used	Not used	A516 Gr 70-74
Callaway	SA155 KCF70	SA285 Gr A SA516 Gr 8-	A615-72 Gr 6-	A36	A421 Tp BA	A527	SA516 Gr 70
North Anna 1 & 2	Not available	SA537 Gr B SA516 Gr 60	A615-68 Gr 40 & 60	A36	Not used	Not used	A333 Gr 3 A156 Gr 60
Surry 1 & 2	SA442 Gr 60	SA442 Gr 60	A615 Gr 40	A36	Not used	Not used	A442 Gr 60
Point Beach 1 & 2	A516	A442 Gr 60	A432 A15	A36	A421 Tp BP	Not used	A333 Gr 1 or 6 A166 Gr KC 70 Class 1
Kewaunee	Not available	A516 Gr 70	A15 Gr 40 A408 Gr 40 A432 Gr 60	A36	Not used	Not used	SA333 Gr 6
Wolf Creek	SA155 KCF 70	SA285 Gr B SA516 Gr 70	A615 Gr 60	A36	A421 Type BA	A527 A528	SA333 Gr 6

Table 2-11 provides a summary of design temperatures and pressures for Westinghouse PWR containments. Design pressures and temperatures vary with respect to containment configuration. The largest difference is seen between vapor suppression and dry suppression containments. Vapor suppression relies on the use of an ice condenser system, and dry suppression is based on volume and physical strength. Lower design temperatures and lower design pressures are found in plants with ice condensers.

2.4.4 Access Airlock and Equipment Hatch Design Data

2.4.4.1 Access Airlocks

Personnel airlock design data are shown in Table 2-12. Airlock diameters (and door clear openings) and barrel lengths vary depending on the intended usage. Later plants tended to have larger airlocks suited for more flexible movement of equipment and materials to support operations and maintenance.

Generally, airlock floors are designed for a distributed live load such as 200 psf and a concentrated wheel load such as 1000 pounds.

Airlocks are designed and fabricated in accordance with the ASME Boiler and Pressure Vessel Code, Section III. In most cases, the airlocks are code-stamped in accordance with the requirements for Code Class MC. For the earlier plants built prior to the ASME incorporation of Subsection NE for Class MC, the portions of the airlocks not backed by concrete typically conformed to ASME Section III for Class B requirements, but were not code- stamped.

Airlock design pressure and temperature are the same as used for the general containment design. Containment design pressures and temperatures for each plant are shown in Table 2-11. The plate material that comprises the airlock pressure vessel components is carbon steel that complies with ASME material specifications, such as SA 516 for fine-grained materials with ductile material properties suitable for low temperature use. Charpy impact testing was performed on the materials at temperatures below service temperatures in accordance with the ASME Code. Minimum service metal temperatures vary from plant to plant, depending on location. Nil-ductility transition temperatures for the materials are typically 0°F, or lower.

Airlock component thicknesses are determined in accordance with the ASME Code requirements and include corrosion allowance where applicable.

TABLE 2-11
WESTINGHOUSE CONTAINMENT DESIGN PRESSURE AND TEMPERATURE
(REFERENCE FSAR, SEE NOTES (1), (2), AND (3))

Plant Name	Pressure (psig)	Temperature (°F)
Robinson 2	42	263
Shearon Harris	45	Not available
Braidwood 1 & 2	50	280
Byron 1 & 2	50	280
Zion 1 & 2	47	271
Haddam Neck	40	271
Indian Point 2	47	271
Indian Point 3	47	271
Catawba 1 & 2	15	250
McGuire 1 & 2	15	190
Beaver Valley 1 & 2	45	280
Turkey Point 1 & 2	55	283
South Texas Project 1 & 2	57	286
Donald C. Cook 1 & 2	12	196/244(1)
Seabrook	52	271
Millstone 3	45	280
Prairie Island 1 & 2	46	268
Diablo Canyon 1 & 2	47	246
Salem 1 & 2	47	271
R. E. Ginna	43	286
Virgil C. Summer	57	283
Joseph M. Farley 1 & 2	54	220
Alvin W. Vogtle 1 & 2	60	270
Sequoyah 1 & 2	10.8	220
Watts Bar 1 & 2	13.5	250
Comanche Peak 1 & 2	50	280

TABLE 2-11 (Continued)
WESTINGHOUSE CONTAINMENT DESIGN PRESSURE AND TEMPERATURE
(REFERENCE FSAR, SEE NOTES (1), (2), AND (3))

Plant Name	Pressure (psig)	Temperature (°F)
Callaway	60	320
North Anna 1 & 2	45	280
Surry 1 & 2	45	268(est)
Point Beach 1 & 2	60	286
Kewaunee	46	268
Wolf Creek	60	320

__Notes__:

(1) Ambient containment temperature is 50°F to 120°F.

(2) Ambient containment pressure is 14.7 psia (0 psig) or less.

(3) The containment relative humidity varies between 15 percent and 70 percent during normal operation. During refueling or abnormal operational conditions, the relative humidity may reach 100 percent.

These values have been used in design unless thermal analyses, or tests are performed and documented to justify the use of lower values.

TABLE 2-12
TYPICAL AIRLOCK DESIGN DATA

Type	Material	Barrel Diameter	Barrel Length	Door Size W X L
Personnel Access	SA516 Gr 60 or 70 SA537 Class 1	9 to 12 feet	12 to 15 ft.	W 3 to 5 feet L 6.5 to 8 feet
Auxiliary or Emergency Access	Same	5 feet	10 feet	2.5-foot diameter

2.4.4.2 Equipment Hatch

The hatch design and fabrication conforms to the applicable ASME Code requirements in effect for the particular vintage plant. The hatch may be ASME code-stamped for the later vintage plants but is not stamped for the early plants.

The hatch is fabricated using the same carbon steel materials as the airlocks. The hatch is furnished with a double-gasketed flange and a bolted, dished head. Typically, the head is convex inward toward the design pressure. The thickness of the head varies depending on the diameter of the opening. For example, a 16-foot diameter hatch opening on one plant requires a 1-1/4 inch thick head for a 57-psig design pressure. The barrel portion of the hatch is typically much thicker than required, based on permissible stresses. In general, the equipment hatch components are conservatively designed and include substantial design margins. The space between the double gaskets on the hatch flange can be pressurized for local leakage checking. The diameter ranges from 14 to 24 feet. The smaller diameter hatches in the earlier plants were designed to accommodate the reactor vessel O-ring seal. The larger diameter hatches provided in the later plants were sized for steam generator replacement.

As for the airlock design, the design pressure and temperature for the equipment hatch is the same as for the containment design.

2.4.5 Electrical Penetrations

Penetration assemblies for nuclear power plants, and specifically for Westinghouse containments, were manufactured and supplied by four major vendors:

- Westinghouse Electric Corporation
- Conax Corporation
- D.G. O'Brien
- Bunker Ramo

Of these four, D.G. O'Brien and Bunker Ramo are no longer in business. Westinghouse and Conax are the two vendors that currently supply penetration assemblies and provide the necessary services for penetrations supplied by D.G. O'Brien or Bunker Ramo.

The penetration assemblies supplied by the above vendors have been accepted by the industry and the U.S. NRC for use in the containments, with one exception. During an EQ inspection, the U.S. NRC identified a deficiency in the Bunker Ramo low voltage penetration qualification testing method. The deficiency is detailed in the Information Notice 88-29. The notice states that the installation resistance (IR) measurements performed during the accident simulation testing were not frequent enough to evaluate the impact of the IR values on the accuracy of the connected instrument circuits.

2.4.6 Mechanical Penetrations

For the high-temperature penetrations, design features are provided to limit the temperature in the concrete adjacent to the penetration (local area) to less than 200°F for normal operation, 350°F for short-term unusual conditions, and 650°F for jets due to postulated pipe rupture conditions. Systems to provide the necessary cooling range from active forced air or water cooling systems within the penetration sleeve to passive systems consisting of insulation and cooling fins. In some designs, thermocouples are placed in the concrete local to the penetration to monitor temperature in the concrete.

In general, piping penetration nozzles are designed and fabricated to conform to the ASME Code requirements in effect when the plant was built. For older plants, ASME Code Section III, Class B was used. For later plants, ASME Code Section III, Division 1, Subsection NE (Class MC) was used. Class MC penetration assemblies may be code-stamped with the NPT stamp for nuclear parts.

In concrete containments, penetrations and anchorages to the concrete shell are designed for forces and moments resulting from operating conditions or postulated pipe rupture. External guides, stops, or increased pipe wall thickness are provided to limit stresses on the penetration and on the adjacent liner plate.

Penetration reinforcing plates and the welds of pipe sleeves to them are shop stress-relieved as a unit in accordance with the ASME Code requirements to ensure a minimum of field welding at the penetrations. Full-penetration butt welds are used to connect the sleeve and the attachment hardware around the process piping. The closure between the sleeve and the process piping consists of flue heads, plates, or drilled pipe caps. Construction materials for mechanical penetrations are listed in Table 2-13.

2.4.7 Fuel Transfer Tube Penetration

The fuel transfer tube is typically a 20-inch outside diameter (OD) stainless steel tube supplied by Westinghouse. The length of the tube varies somewhat depending on the particular plant configuration, but is typically about 15 plus feet long. Design parameters for the tube and connections are as follows:

- Code: The tube assembly is code-stamped as a Class 2 part in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Division 1.
- Material: Tube: ASME-SA-240, Type 304 (stainless steel, 0.375-inch thick)
- Blind Flange and Flange Adaptor Ring: ASME-SA-182, Type 304 (forged stainless steel, 2 inches thick with a 27.5-inch diameter)
- Attachment Assembly:

  Ring Plate: SA240, Grade 304
  Pipe Segment: SA106, Grade B or SA516, Grade 70
  Bellows: SA240, Type 304

2.5 TIME-LIMITED AGING ANALYSES

- Time-limited aging analyses (TLAAs) are those licensee calculations that:
- Involve the effects of aging
- Involve time-limited assumptions defined by the current operating term, for example 40 years
- Involve systems, structures, and components within the scope of license renewal
- Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, or component to perform its intended functions
- Were determined to be relevant by the licensee in making a safety determination
- Are contained or incorporated by reference in the current licensing basis

TABLE 2-13
MECHANICAL PENETRATIONS
MATERIALS OF CONSTRUCTION

Penetration Component	Material
Sleeve	SA 333 Gr 1 and Gr 6 SA 155 Class 1 Gr 60 and Gr 70 SA 312 TP304 (stainless steel) SA 106 Gr B
Closure Plates	SA 516 Gr 60 and Gr 70 SA 537 Class 1
Bellows	SA 240, Type 304 (stainless steel)
Socket Weld Couplings	SA 182, Type F316
Pipe Fittings	SA 420 Gr WPL-6
Welding Material	E 6010 E 7018
Process Pipe	SA 106 Gr C

TLAAs have been identified as part of the original design process for the PWR containment and consist of the following:

- Analytical prediction of time-dependent loss of prestress force loads in prestressing systems
- For the concrete containment structure, number of fatigue cycles at penetration anchors, and where appropriate, calculated cumulative fatigue usage factors
- Bellows number of fatigue cycles in mechanical penetrations, and where appropriate, calculated usage factors
- Number of fatigue cycles of mechanical penetrations, and where appropriate, calculated usage factors

These TLAAs are evaluated in Section 3.3.

2.6 GENERAL MAINTENANCE PRACTICES

Current maintenance practice consists of periodic inspection and repair as required. Inspection and test requirements are defined in regulatory guides and Appendix J of 10 CFR 50. Section XI of the ASME Code also provides requirements for inspection. A description of required testing and inspections is provided, followed by a discussion of aging mechanisms and mitigation practices. Reference 3 provides an overview of current inspection requirements for concrete structures in nuclear power plants. Further, Reference 1 contains a summary of current maintenance and repair practices to address aging and deterioration of the structures within the scope of this GTR.

Figure 2-20 provides a flow chart summary of current inspection and repair guidelines.

2.6.1 Inspection Regulations

Current inspection requirements for containments are defined by: Appendix J to 10 CFR 50, NRC Regulatory Guide 1.35, NRC Regulatory Guide 1.90, 10 CFR 50.55a, and NRC Regulatory Guide 1.163.

Appendix J to 10 CFR 50 provides general inspection requirements for accessible exterior and interior surfaces and leak test requirements for the overall structure. Regulatory Guide 1.35 defines testing and visual inspection requirements for ungrouted post-tensioned tendons, and Regulatory Guide 1.90 provides testing and visual inspection requirements for grouted post-tensioned tendons.

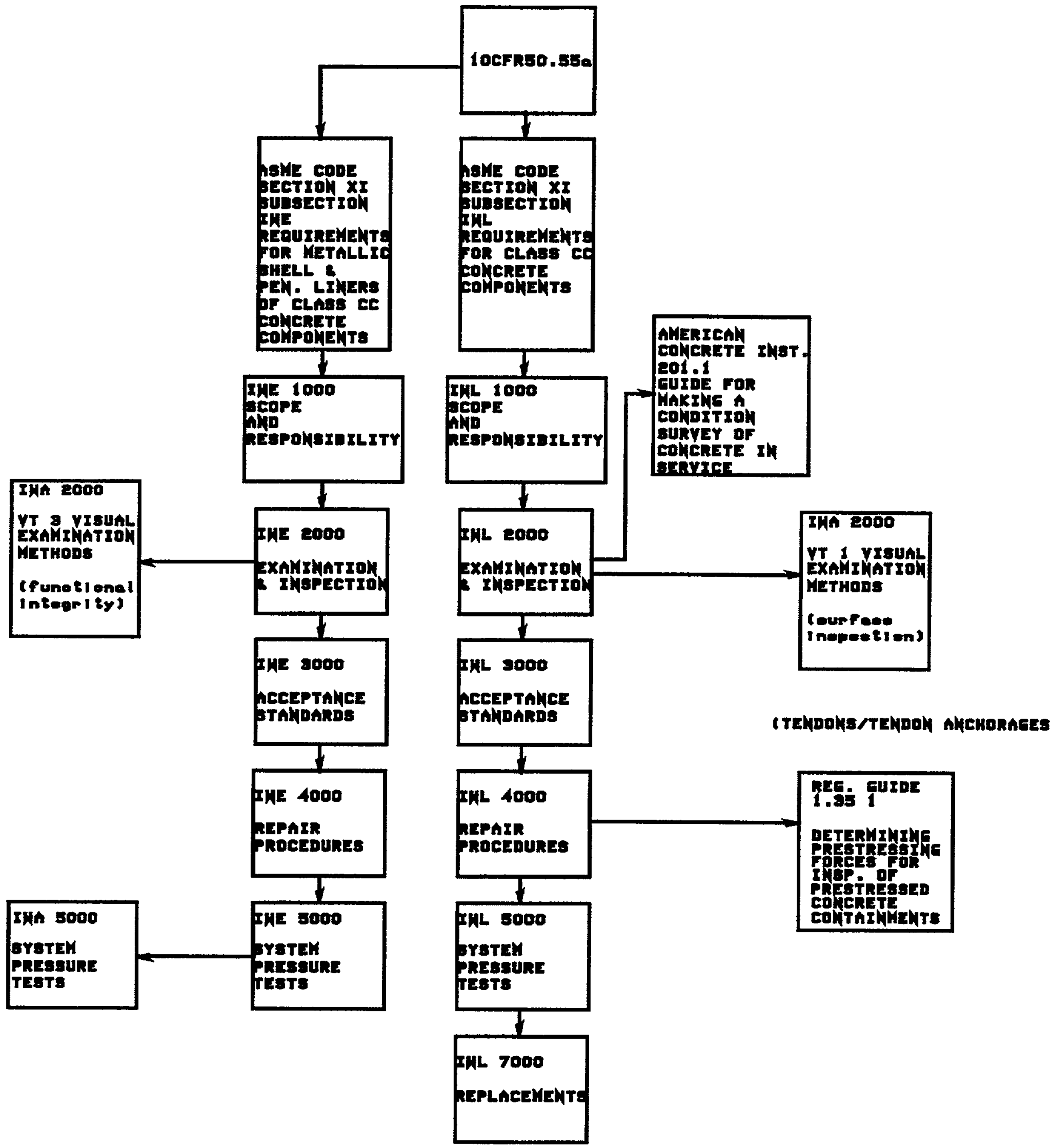


Figure 2-20 Inservice Inspection and Repair Guidelines Flow Chart

NRC SECY-96-080 issues an amendment to 10 CFR 50.55a incorporating by reference the 1992 Edition of the ASME Boiler and Pressure Vessel Code, including the 1992 Addenda. The incorporation includes Section XI, Subsection IWE, "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants," and Subsection IWL, "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants." Additional modifications to the rule also address inaccessible areas, visual inspection methodology, pressure-retaining welds, inspection sampling plans, and redundant inspections through application of both Subsections IWE and IWL. Recommendations for tendon examinations provided in Regulatory Guide 1.35, but not addressed in Subsection IWL, are also in the regulation.

Licensees are required to incorporate Subsections IWE and IWL, and the additional rule changes, into their ISI program.

Three types of tests are required by Appendix J to 10 CFR 50 that supplement the Section XI requirements and are designated as Type A, B, and C. The Type A test measures the containment integrated leakage rate after the containment has been completed and at periodic intervals. Type B tests are intended to detect local leaks and provide measurement of leakage across pressure-containing or leakage-limiting boundaries for containment penetrations. Type C tests measure containment isolation valve leakage rates. Section V.A of Appendix J requires a general visual examination of accessible interior or exterior surfaces of containment structures prior to Type A testing to detect structural degradation.

The testing frequency (until September 1995) for the Type A test, following the pre-operational test, is three times during a 10-year operating period, at equal intervals. Type B tests, except for the air lock, are conducted at each reactor shutdown but at intervals no greater than 2 years. Air lock testing is conducted prior to fuel loading, at 6-month intervals thereafter with the containment at peak internal pressure and after periods when the airlock was opened. The above testing frequencies were changed with the development of a new regulatory guide (1.163) [Ref. 4] by the U.S. NRC based on performance testing, allowing alternate testing frequencies. These changes are discussed in Reference 1 and are given below [Sections 2.5 and 2.5.2 of Ref. 1]:

> In September 1995, the U.S. NRC amended Appendix J (10 CFR 50) to provide a performance-based option for leakage rate testing as an alternative to the existing prescriptive requirements [Ref. 5]. The amendment is aimed at improving the focus of the body of regulations by eliminating prescriptive requirements that are marginal to safety and by providing licensees greater flexibility for cost-effective implementation methods for regulatory safety objectives. Now that Appendix J has been amended, either Option A – *Prescriptive Requirements* or Option B – *Performance-Based Requirements* can be chosen by a licensee to meet the requirements of Appendix J.
>
> Now that Appendix J has been amended [Ref. 5], licensees may voluntarily comply with Option B requirements rather than continue using established leakage rate test schedules. Option B allows licensees with good integrated leak rate test performance history to reduce the Type A testing frequency from three tests in 10 years to one test in

10 years. For Type B and C tests, Option B allows licensees to reduce testing frequency on a plant-specific basis based on the operating experience for each component and establish controls to ensure continued performance during the extended testing interval. The U.S. NRC position on performance-based containment leak testing is discussed in Regulatory Guide 1.163 [Ref. 4].

It is noted, and as discussed in SECY-96-080, that Appendix J, the final revision, does not change the current containment visual inspection frequency requirements. The current frequency of performing visual inspections three times in 10 years is maintained. This is consistent with Regulatory Guide 1.163, which accompanies the final revision to Appendix J, and Subsection IWE and IWL. Note, however, that a longer interval of up to 10 years is acceptable between Type A tests.

The scope of Regulatory Guide 1.35 encompasses testing of a random sample of post-tensioned tendon assemblies for ungrouted tendons. Testing includes visual examinations, preload testing, and material tests. Visual inspection of containment concrete exterior surfaces, tendon anchorage assembly hardware, bottom grease caps for vertical tendons, and concrete surrounding tendon anchorages is required. Indications for concrete are spalling, cracking, and severe scaling, while grease caps are inspected for leakage, and hardware is examined for cracking, corrosion, and deformation. Preloading is tested using a liftoff load test, while tensile tests are performed on tendon wires and strands. The chemistry of filler grease is also analyzed, and removal and replacement of grease is monitored.

Regulatory Guide 1.35.1 provides guidance on the determination of prestressing forces in the steel tendons. Its purpose is to clarify the U.S. NRC's position in Regulatory Guide 1.35 on the construction of load tolerance bands for groups of tendons to enhance the small-sample inspection program already in place per Regulatory Guide 1.35. The tendons are grouped by time-dependent characteristics:

Time-dependent load losses caused by:

- Concrete shrinkage
- Concrete creep
- Relaxation of prestressing steel

Regulatory Guide 1.35.1 enumerates the factors associated with the time-dependent losses, such as humidity and temperature, and details how to use these factors in developing tolerance bands for the tendon groups.

Initial load losses and environmental factors are also considered:

Initial Load Losses

- Slip at anchorages
- Friction between tendon and tendon duct
- Elastic shortening of tendons
- Effects due to load sequencing

Other

- Tendon failure due to corrosion or material deficiency
- Effects of temperature variation

Regulatory Guide 1.35 recommends the comparison of actual prestress force measurements with the predicted values for randomly selected tendons. Implementation of the guidelines of Regulatory Guide 1.35.1 yields a better correlation between the predicted and actual values on a group-by-group basis. Although the Regulatory Guide is mandatory only for plants with construction permits/design approval dated after July 31, 1990, its recommendations are good practice for earlier plants.

The scope of Regulatory Guide 1.90 covers testing of a random sample of post-tensioned tendon assemblies for grouted tendons. Testing includes visual examinations, preload testing for ungrouted test tendons, and prestress level or deformation under pressure monitoring for grouted tendons. Visual inspection of structural discontinuities and areas of heavy stress concentrations is recommended, supplemented by pulse velocity testing when cracks are discovered. Preloading of ungrouted test tendons is tested using a liftoff load test. Prestress monitoring for grouted tendons is accomplished through monitoring tensile strains in the wires of a tendon, or strain gauges or stress or strain meters are applied at a section of the structure.

In general, the test frequency for both Regulatory Guides 1.35 and 1.90 is 1, 3, and 5 years after the first structural integrity test, and then every 5 years thereafter.

Licensees will be required to incorporate Section XI and Subsections IWE and IWL of the 1992 Edition of the ASME Code, including the 1992 Addenda, as well as other additional modifications into their ISI program to comply with changes to 10 CFR 50.55a as described in SECY-96-080.

Subsection IWE provides guidelines for the inservice inspection, repair, and replacement of Class MC pressure-retaining components and their integral attachments as well as of metallic shell and penetration liners of Class CC pressure-retaining components and their integral attachments. Subsection IWL provides guidelines for the inservice inspection and repair of the reinforced concrete and the post-tensioning systems of Class CC components.

NRC SECY-93-328 [Ref. 6], which issued the proposed rule change to 10 CFR 50.55a, introduced additional modifications to 10 CFR 50.55a to address U.S. NRC concerns related to tendon examinations and inaccessible areas. The modifications maintained in SECY-96-080 are listed below.

- Four recommendations for tendon examination included in Regulatory Guide 1.35, Rev 3, but not addressed in IWL are included in the amended rule. Regulatory Guide 1.35 requires the following:
  - Requires that grease caps that are accessible must be visually examined to detect grease leakage or grease cap deformation
  - Requires the preparation of an Engineering Evaluation Report when consecutive surveillance indicates a trend of prestress loss to below the minimum prestress requirements
  - Requires an evaluation to be performed for instances of wire failure and slip of wires in anchorages
  - Addresses sampled sheathing filler grease and reportable conditions
- Visible evidence of concrete degradation, such as leaching and surface cracking, may be an indication of degradation in adjacent inaccessible areas. Therefore, an evaluation of the potential degradation of adjacent inaccessible areas should be performed.

Four additional modifications to 10 CFR 50.55a, incorporated through SECY-96-080 in response to public comments made on SECY-93-328, include:

- Expansion of the evaluation of inaccessible areas of concrete containments to include metal containments and the liners of concrete containments.
- Permission of alternative lighting and resolution requirements for remote visual inspection of the containment.

  The maximum direct examination distance specified in Table IWA-2210-1 may be extended and the minimum illumination requirements specified in Table IWA-2210-1 may be decreased provided that the conditions or indications for which the visual examination is performed can be detected at the chosen distance and illumination.
- Examination of pressure-retaining welds and pressure-retaining dissimilar metal welds are optional.
- An alternative sampling plan has been added.

The utility should document the following in accordance with ASME, Section XI IWA-6000, for each inaccessible area identified:

- A description of the type and estimated extent of degradation and the conditions that led to the degradation
- An evaluation of each area and the result of the evaluation
- A description of necessary corrective actions

The above requirement is identical for the evaluation of suspect inaccessible areas identified through visual inspection of concrete areas near tendon anchorage or through examination of metal containments and the liners of concrete containments.

Provisions have been added to the amended rule, through SECY-96-080, to prevent duplicate examinations required by both the periodic routine and expedited examination program requirements. Further, the utility is allowed to use recently performed examinations of the post-tensioning system to satisfy the requirements for the expedited examination of the containment post-tensioning system.

2.6.2 ASME Code Section XI Inspection

ASME Code inspection requirements are defined for both the concrete and the steel liner of the concrete containment as well as for the shell of the steel containment. The following describes current code requirements.

Concrete

Subsection IWL of ASME Code Section XI can be applied for preservice examination, inservice inspection, and inspection subsequent to repair or replacement for the reinforced concrete of the concrete containment. Areas and components that are exempt include tendon end anchorages that are inaccessible and portions of the concrete that are inaccessible where inspection is obstructed by the liner, foundation material or backfill (below grade), or adjacent structures or components.

The present inservice inspection schedule for concrete containments, concrete, and unbonded post-tensioning systems is set for 1, 3, and 5 years following the preservice structural integrity test, and every 5 years thereafter.

All surfaces and tendon and anchorage areas, including those protected by coatings, except as exempted as previously discussed, are visually examined for evidence of conditions indicative of degradation, as defined in ACI 201.1. A VT-3C visual examination is conducted for all accessible areas to determine the general structural condition through the identification of suspect areas, where evidence of deterioration is found. Evidence of degradation includes cracking, spalling, staining, wetness, and discoloration. Specifications for examination method VT-3 are employed, i.e., those for minimum illumination, maximum direct examination distance,

and maximum procedure demonstration lower case character height. VT-1C visual examinations are conducted for selected suspect areas. Examination specifications for examination method VT-1 are employed.

Repairs or replacement, where warranted, is provided and re-inspection occurs prior to acceptance of the corrective action.

Steel

Subsection IWE of ASME Code Section XI can be applied for preservice examination, inservice inspection, and inspection subsequent to repair or replacement for the Class MC pressure-retaining components and their integral attachments. Inspections are made prior to leak rate testing. Embedded or inaccessible portions of the containment vessels, parts, and appurtenances are exempt.

Parts to be inspected following ASME Section XI inservice inspection requirements are categorized into accessible surface areas, welds, pressure-retaining bolting, seals, and moisture barriers for steel liners and steel containment. Welds are examined only as part of the containment surface. Examination intervals are scheduled based on frequency and extent of examination, where the frequency is uniform but the extent varies with the examination category. Currently, the IWE Inspection Program A is defined where the first interval, following any preservice examinations, must be conducted within the first 3 years of plant life, the second within 10 years, the third within 23 years, and the fourth within 40 years. The inspection period may be extended up to 1 year so that the inspection may be conducted during plant outages. The program is required to be repeated for plant life extension, with a 10-year inspection interval. An alternative Inspection Program B is based on successive 10-year inspection intervals.

Accessible surface areas of the steel containment vessel pressure-retaining boundary, except those that are submerged or insulated, are subject to general visual examination. VT-3 visual examination is applied for areas including those that are submerged and insulated. Paint or coatings shall not be removed for visual inspection. Coated areas are examined for evidence of flaking, blistering, peeling, discoloration, and other signs of deterioration. Uncoated areas are examined for evidence of cracking, discoloration, wear, pitting, excessive corrosion, arc strikes, gouges, surface discontinuities, dents, and other signs of surface irregularities.

Containment penetration welds, as part of the surface, and pressure-retaining bolting are visually inspected using the VT-1 examination method. A 25-percent random sample of welds is selected for examination, while all bolts are examined and torque- or tension-tested. A surface examination is applied for a 50-percent random sample of dissimilar material welds. VT-3 examinations are applied for seals, gaskets, and moisture barriers.

Areas or components with identified flaws, degradation, or repairs where the area or component is found to be acceptable for continued service are subject to augmented examination, VT-1 visual or volumetric examination. These areas or components no longer require augmented

examination when flaws, degradation, or repairs are found to be unchanged after three consecutive inspections.

Leak rate testing is conducted in accordance with 10 CFR 50, Appendix J, as discussed in Subsection 2.6.1.

Tendons

Subsection IWL of ASME Code Section XI can also be applied for unbonded or nongrouted post-tensioning systems, with guidelines similar to those for Regulatory Guide 1.35.

Inservice inspection intervals of unbonded post-tensioning systems are specified in the concrete section above. The items to be inspected include the tendon, wire or strand, anchorage hardware and surrounding concrete, the corrosion protection medium, and free water. The tendon wire or strand is subjected to both visual and mechanical testing. A VT-1 visual examination should be performed on the tendon anchorheads, wedges, buttonheads, shims, and the concrete extending outward a distance of 2 feet from the edge of the bearing plate. The chemistry and volume of the corrosion protection medium and free water are monitored. The chemistry is monitored for levels of chlorides, nitrates, and sulfides as well as pH, which may contribute to a corrosive environment. Documentation includes observations of cracks in the concrete and tendon anchorage hardware along with broken strands and damaged or missing hardware. All of the above ASME Section XI guidelines are similar to those established in Regulatory Guide 1.35.

Examination of corrosion protection medium and free water is also performed with guidelines similar to those found in Regulatory Guide 1.35.

Visual Examination

Current visual examination categories are differentiated as follows:

VT-1 - Examinations are conducted at a maximum distance of 2 feet to detect surface flaws, cracks, wear, corrosion, or erosion.

VT-2 - Examinations are conducted at a maximum distance of 6 feet to detect leakage from pressure-retaining components without leakage collection systems during system pressure tests.

VT-3 - Examinations are conducted at a maximum distance of 4 feet to determine general mechanical and structural conditions such as settings, clearances, connection integrity, wear, corrosion, and erosion.

VT-1C - This examination category has the same requirements as VT-1 but applies to concrete.

VT-3C - This examination category has the same requirements as VT-3 but applies to concrete.

Remote visual examination may be applied instead of direct visual examinations following recommendations of ASME IWA.

2.6.3 IAEA Maintenance and Inspection History

The International Atomic Energy Agency (IAEA) has conducted a survey to determine current inspection, maintenance, and repair practices at domestic and foreign plants, entitled the IAEA Survey of Nuclear Power Plant Owners/Operators on the Management of Aging of Concrete Containment Buildings. The inspection and testing scope of the survey is more detailed than the requirements of ASME Section XI, querying respondents on nondestructive testing applied for when indications are discovered and for repair preparations. Survey data for maintenance and inspection history are summarized in Tables 7-1 and 7-2 of Section 7.1.

2.6.4 Observed Degradation

This section provides summaries of observed degradation associated with the structures within the scope of this report. Attention is given to degradation identified in the IAEA surveys, U.S. NRC Information Notice 89-79 nuclear plant reliability data system (NPRDS) and licensee event reports (LERs), and leak rate testing. Reference 1 contains a summary of containment pressure boundary component degradation occurrences at commercial nuclear power plants in the United States. Also, given in Reference 7 are discussions pertaining to the historical performance of electrical and mechanical penetrations. From the evaluation of the general and specific failure for penetration systems, it has been found that in general, electrical and mechanical penetrations are reliable and experience a low failure rate. It is noted that:

- Detection of electrical penetration assembly failures appears to occur mostly through leak rate testing and monitoring of pressure instrumentation.
- Approximately one-fourth of all electrical-penetration-related failures noted were attributed to improper design, maintenance, or testing.
- Degradation of mechanical penetrations has been identified primarily through leak rate and visual inspections.
- For access penetrations, most of the instances of age-related degradation are related to the degradation of gaskets and seals. In most cases, these components were refurbished or replaced. Natural aging of materials and mechanical damage has been the primary causes of degradation.

IAEA Results

The IAEA survey limited response on degradation indicates that concrete cracking is common, resulting mostly from shrinkage. Other probable causes include freeze-thaw and leak rate tests. In most instances, the cracks were not significant enough to require repairs. Remedial actions, when applied, consisted of protective recoating or polymer impregnation for renewed

insurance against moisture penetration. Table 7-3 summarizes the survey results on observed degradation.

Steel liner corrosion damage was observed by one utility at the junction of the liner with the 2-foot thick fill mat forming the floor of the containment. Degradation resulted from exposure to chemical attack and was corrected by protective recoating. Similar problems, as described in U.S. NRC Information Notice 89-79, are discussed below.

Corrosion to containment penetrations was noted for the D.C. Cook plant. This consisted of light rust corrosion to the exterior surface as a result of the loss of protective coating. The problem was corrected with the reapplication of the protective coating.

Other degradation was noted at the Cook plant consisting of voids/honey combing, staining, pop-outs, and efflorescence. Voids were repaired with dry pack, and other degradations were not significant enough to require repair.

U.S. NRC Information Notice 89-79

Significant coating damage and base metal corrosion was observed on the outer surface of the steel containment shell for the ice condenser equipped containments of McGuire 1 and 2 as well as Catawba 1 and 2. Damage was detected during visual examinations performed in conjunction with the 10 CFR 50 Appendix J integrated leak rate test. The containments, similar for both plants and units, consist of a free-standing steel shell containment enclosed by a concrete shield building. An approximately 6-foot wide annular space separates the steel containment from the shield building. Degraded areas were discovered on the inside and outside surfaces of the steel shell. The damaged exterior areas for McGuire and Catawba were found at the intersection of the shell wall with the concrete annulus floor, above and below the floor. The below-floor corrosion resulted from the lack of a sealant at the interface of the shell and annulus floor. The interior areas for McGuire Unit 1 were located at the floor level between the upper and lower containment compartments in the vicinity of the ice condenser, where cork is used to fill a 2-inch gap between the steel containment and the concrete floor. The probable cause for the exterior corrosion was attack by condensed boric acid coolant leaking from instrument line compression fittings. The interior damages most likely resulted from attack by moisture contained in the cork and originating from the ice condenser or condensation.

Corrosive damage consisted of general coating failure and surface corrosion with localized pitting. The shell thickness remained above the minimum thickness required by ASME Code, Section III, for the area of the maximum reduction in material thickness.

This problem was resolved through remedial actions that included weld repair and recoating, performance of more detailed inspections, and development of acceptance criteria for coatings and sealant material. Cork material and failed coatings were removed, coatings were reapplied, and joint sealant materials were used to prevent water and acid from entering joints.

Visual examination requirements of existing programs proved effective in identifying the problem, and timely and effective repair by the utility resolved the problem.

NPRDS and LER Data

The nuclear plant reliability data system (NPRDS) and licensee event reports (LERs) provide some information on the degradation experience of containment components. The LER data include reports of air lock seal failures, electrical penetration seal degradations, and containment isolation valve seating, packing, or seal failures. Each failure was detected during local leak rate tests or by associated visual inspection, indicating the effectiveness of current inspection programs. Other degradation experience includes breakdown of the seal between the crane wall and the containment building, as noted at one plant, and corrosion to the steel containment vessel due to standing water in the annulus area, similar to that reported in U.S. NRC Information Notice 89-79. Corrective actions or repair methods are noted along with other data for age-related containment degradation in Table 2-14.

The NPRDS data contain a large number of entries related to degradation of hatches, penetrations, and the fuel transfer canal. Only a representative sample is provided in Table 2-15. The NPRDS General Report indicates a multitude of instances where seals and mechanical components for the personnel airlock or hatch failed due to mechanical wear and aging. The seal failures were detected with local leak rate tests or associated visual inspection. Also a number of O-ring seal failures and bellows or bellows component failures or damage were noted for penetrations. Repairs were made in all instances.

Existing inspection, maintenance, and repair programs are sufficient to preclude mechanical wear from the list of significant age-related degradation mechanisms for plant life extension for the subcomponents under consideration. These subcomponents are not considered to be long-lived and passive, and damage is repaired on discovery.

Leak Rate Testing

The frequency of the Type A integrated leak rate test (ILRT) may result in unnecessary degradation to the reinforced concrete containment. During the test, the containment is subjected to an internal pressure equal to the design pressure for the structure for most plants. (A few older plants use a test pressure below the design value.) This follows the structural acceptance test (SAT), where the containment is required to withstand an internal pressure 15 percent greater than the design pressure level, before plant operation. Repeated openings may create a permanent seepage path for water, resulting in corrosive attack on reinforcing steel or chemical attack on the concrete. Significant degradation could result only from extensive exposure to water carrying aggressive chemical agents. Such exposure occurs only below grade where exposure to groundwater is possible; however, extensive cracking due to leak rate testing is not likely to occur in this location. Above-grade cracking in the concrete is accessible for inspection and can be managed by current inspection and repair programs. The U.S. NRC has amended Appendix J (10 CFR 50) to provide a performance-based option for leak rate testing as an alternative to existing prescriptive requirements, which reduces testing frequency on a plant-specific basis. As a result, leak rate testing does not constitute a significant source of age-related degradation.

TABLE 2-14
AGE-RELATED DEGRADATION DATA FROM LICENSEE EVENT REPORTS

Plant	Degradation Mechanism	Component	Discovery Method	Repair Method
D. C. Cook 2	Unknown	Divider Barrier Seal	Seal Surveillance Test(1)	Seal Replaced
Wolf Creek	Mechanical Wear	Air Lock Shaft Seal	Air Lock Leak Rate Test	Door Shaft Replaced
Catawba 1	Corrosion (Standing Water Boric Acid)	Exterior of Steel Cont. Vessel (SCV)	Appendix J General Visual Inspection	SCV Repaired & Recoated
McGuire 1	Corrosion (Environment Interaction)	Steel Cont. Vessel	Appendix J General Visual Inspection	Repaired
McGuire 1	Corrosion (Boric Acid Deposits)	Steel Cont. Vessel	Appendix J General Visual Inspection	Repaired & Recoated
Wolf Creek	Unknown	Cont. Isolation Valve	Local Leak Rate Test	Parts Replaced
Braidwood 2	Seal Degradation	Cont. Purse Valve	Local Leak Rate Test	Seals Replaced
Trojan	Normal Packing Degradation	Cont. Spray & RHR Valve	Local Leak Rate Test	Packings Tightened
North Anna 2	Seating Surface Degradation	Isolation Check Valve	Local Leak Rate Test	Cleaned & Relapped
D. C. Cook 2	Unknown	Cont. Isolation Valve	Leak Rate Testing	Repaired & Retested
D. C. Cook 2	Misalignment Disk & Valve Seat	Isolation Valve	Leak Rate Testing	Valve Replaced

TABLE 2-14 (Continued)
AGE-RELATED DEGRADATION DATA FROM LICENSEE EVENT REPORTS

Plant	Degradation Mechanism	Component	Discovery Method	Repair Method
D. C. Cook 1	Seating Surface Degradation	Isolation Valve	Leak Rate Testing	Repaired & Retested
Kewaunee 1	Residue on Interior Surfaces	Cont. Isolation Valve	Inservice Timing Test	Valves Rebuilt
Kewaunee 1	Design/Mfg. Defect	Cont. Isolation Valves	Local Leak Rate Test	Replace Gaskets & Adjustment
Conn Yankee 1	Design Deficiency	Cont. Isolation Valves	Quarterly Surv. Test	Lock in Closed Position (Short-Term Fix)
D. C. Cook 1	Seating Degradation	Pressurization System Valve	Leak Rate Test	Not Given

Notes:

(1) Seal between containment wall and crane wall.

TABLE 2-15

AGE-RELATED DEGRADATION DATA FROM THE NUCLEAR PLANT RELIABILITY DATA SYSTEM

Component	Part	Degradation Mechanism	Discovery Method	Number of Occurrences	Repair
Containment Equipment Hatch	Door Seal	Mechanical Wear	Surveillance	1	Replaced
	Hatch Seal	Mechanical Wear	Leak Rate Test	1	Replaced
	Inner Door Ring-Feeder Assembly	Mechanical Wear	Containment Exit	1	Replaced
	Shaft Pin & Mounts	Mechanical Wear	Surveillance	1	Replaced
	Wires & Fittings	Seal Shrinkage	Surveillance	1	Replaced
	O-rings	Mechanical Wear	Leak Rate Test	1	Replaced
Containment Personnel Hatch	Cam Follower/ Bearings	Mechanical Wear	Surveillance & Cont. Entry	4	Replaced & Reworked Parts
	Door Seals/Gaskets	Mechanical Wear	Surveillance, Leak Rate Test & Alarm	26	Replaced
	Inner Door Clutch	Mechanical Wear	Surveillance	1	Adjusted
	Outer Door Actuator	Mechanical Wear	Containment Exit	1	Parts Replaced
	Inner Door Mechanism	Mechanical Wear	Surveillance	2	Repaired & Adjusted
	Inner & Outer Doors Not Functional	Mechanical Wear	Containment Exit	1	Replaced
	Drive Shaft Seal	Mechanical Wear	Leak Rate Test	3	Replaced
	Outer Door Interlock	Mechanical Wear	Surveillance & Containment Exit	2	Replaced Cam Roller & Pin

TABLE 2-15 (Continued)

AGE-RELATED DEGRADATION DATA FROM THE NUCLEAR PLANT RELIABILITY DATA SYSTEM

Component	Part	Degradation Mechanism	Discovery Method	Number of Occurrences	Repair
Containment Personnel Hatch	Actuator Arm Roller Cam	Mechanical Wear	Surveillance	1	Positioned & Tightened
	Outer Door Shaft	Mechanical Wear	Surveillance	3	Replaced
	Inner Door Seals & Shafts	Mechanical Wear	Leak Rate Test	1	Replaced
	Inner Door Pall Operating Arm	Mechanical Wear	Surveillance	1	Straightened
	Door O-rings	Mechanical Wear	Surveillance, Leak Rate Test & Cont Entry/Exit	11	Replaced
	Connecting Rod	Mechanical Wear	Surveillance	1	Replaced
	Outer Door Bushings & Cams	Mechanical Wear	Leak Rate Test	1	Replaced
	Rollers & Latching Blocks	Mechanical Wear	Leak Rate Test	1	Replaced
	Alignment Bar	Mechanical Wear	Surveillance	1	Tightened
	Various Inner & Outer Door Parts	Mechanical Wear	Surveillance	1	Replaced Parts/Gear Alignment
	Outer Door Limit Switch	Mechanical Wear	Surveillance	1	Adjusted
	Inner Door Control Switch	Mechanical Wear	Containment Exit	1	Repositioned & Tightened Switch

TABLE 2-15 (Continued)

AGE-RELATED DEGRADATION DATA FROM THE NUCLEAR PLANT RELIABILITY DATA SYSTEM

Component	Part	Degradation Mechanism	Discovery Method	Number of Occurrences	Repair
Containment Personnel Hatch	Inner Door Drive Actuator	Mechanical Wear	Surveillance	1	Replaced
	Shaft Seal Assy. Bolts	Mechanical Wear	Leak Rate Test	1	Tightened
	Inner Door Handwheel	Mechanical Wear	Containment Exit	1	Adjusted & Tightened Parts
	Outer Door Shaft Coupling	Abnormal Wear	Surveillance	1	Aligned & Tightened
	Outer Door Upper Shaft Seals	Mechanical Wear	Surveillance	1	Replaced
	Outer Door Key & Keyway	Mechanical Wear	Surveillance	1	Replaced
	Seal on Upper & Lower Shaft & Door	Not Given	Surveillance	1	Replaced
	Clutch Assembly Slipping	Mechanical Wear	Containment Entry	1	Tighten Setscrews
	Inner Door Bearing	Mechanical Wear	Surveillance	1	Replaced
	Inner & Outer Door Ball Valves	Mechanical Wear	Surveillance	1	Rebuilt
Containment Auxiliary Hatch	Shaft Seals	Mechanical Wear	Surveillance	1	Replaced
	Outer Door Shaft	Mechanical Wear	Surveillance	1	Replaced
	Various Hatch Parts	Mechanical Wear	Surveillance	1	Replaced

TABLE 2-15 (Continued)
AGE-RELATED DEGRADATION DATA FROM THE NUCLEAR PLANT RELIABILITY DATA SYSTEM

Component	Part	Degradation Mechanism	Discovery Method	Number of Occurrences	Repair
Containment Auxiliary Hatch	Interlock Rod & Bearings	Mechanical Wear	Surveillance	1	Replaced
	Valve	Mechanical Wear	Surveillance	1	Freed & Lubricated
	Outer Door Handwheel Seal	Abnormal Wear/Age	Surveillance	1	Replaced
	Outer Door Seal	Mechanical Wear	Surveillance	1	Replaced
	Door Seal	Mechanical Wear	Surveillance	1	Replaced
Containment Emergency Hatch	Cam Follower	Mechanical Wear	Volume Test	1	Replaced
	Door Outer Gasket	Mechanical Wear	Leak Rate Test	1	Replaced
	Inner Diameter Seal	Mechanical Wear	Leak Rate Test	1	Replaced
	Outer Door Gaskets	Mechanical Wear	Surveillance	1	Replaced
	Inner Door Cam Followers	Mechanical Wear	Surveillance	1	Replaced
	Interlock Parts	Mechanical Wear	Surveillance	1	Replacement & Adjustment
	O-rings	Mechanical Wear	Leak Rate Test	1	Replaced

TABLE 2-15 (Continued)

AGE-RELATED DEGRADATION DATA FROM THE NUCLEAR PLANT RELIABILITY DATA SYSTEM

Component	Part	Degradation Mechanism	Discovery Method	Number of Occurrences	Repair
Lower Personnel Airlock	Door Seal	Mechanical Wear	Surveillance & Leak Rate Test	3	Replaced
	Airlock Seal	Mechanical Wear	Leak Rate Test	2	Replaced
	Outer Door Pin & Push Rod	Mechanical Wear	Containment Exit	1	Replaced
Upper Personnel Airlock	Check Valve	Mechanical Wear	Not Given	1	Replaced
	Airlock Seal	Mechanical Wear	Door Operation	1	Replaced
80-foot Airlock Door	Outer Door Gasket	Mechanical Wear	Surveillance	1	Replaced
	Outer Door Ball Valve	Mechanical Wear	Leak Rate Test	1	Rebuilt
	Miter Gears & Spacer	Mechanical Wear	Attempted Exit @ 100% Power	1	Replaced
	Outer Door Drive Chain, Check Valve & Gasket	Mechanical Wear	Attempted Entry @ 100% Power	1	Replaced
95-foot Airlock Door	Outer Door Seal	Mechanical Wear	Alarms	1	Replaced
Aux. Bldg. Airlock Door	Inner Door Seal	Mechanical Wear	Surveillance	1	Replaced
Mechanical Penetration	Seal	Unknown	Leak Rate Test	12	Resealed
Electrical Penetration	Assembly Module Seal	Mechanical Wear	Leak Rate Test	10	Replaced

TABLE 2-15 (Continued)
AGE-RELATED DEGRADATION DATA FROM THE NUCLEAR PLANT RELIABILITY DATA SYSTEM

Component	Part	Degradation Mechanism	Discovery Method	Number of Occurrences	Repair
Piping Penetration	Sleeve	Mechanical Wear	Surveillance	2	Repaired
	Bellows Support Plate	Mechanical Wear	Surveillance	1	Weld Repair
	Bellows	Mechanical Wear	Surveillance	1	Repaired
Cont. Electrical Penetration	O-rings & Modules	Aging	Leak Rate Test	2	Replaced
Cont. Penetration	Bellows	Mechanical Wear	Surveillance	1	Welded Hole
Fuel Transfer Canal	Valve	Mechanical Wear	Prior to Fuel Reload	1	Parts Repaired & Replaced

SECY-96-080

Additional information is given in SECY-96-080 pertaining to inservice inspection deterioration detection. Review of the table summarizing the occurrences of structural degradation reveals that about a third (11 of 28) of the occurrences are found for the WOG plants listed in Table 1-1. All occurrences associated with the WOG plants were detected by utility inspections.

Occurrences of corrosion in metal containments and liners of concrete containments reveal that 13 of 29 occurrences were found at one of the plants listed in Table 1-1, and 4 of the 13 occurrences, about 30 percent, were detected by NRC audits. The degradation was detected by the audit before the loss of the intended function, and repairs or evaluation were performed before the plant returned to service. The frequency of CLB inspections should be based on the time required to detect the indications of degradation and evaluate or repair them before the loss of intended function, and all accessible areas of the interior and exterior containment structures should be inspected prior to the performance of Type A tests (10 CFR 50, Appendix J). The damage would have been detected during the next scheduled utility inspection of the accessible surface areas, where the degradation would have still been discovered and repairs or evaluations made well before the loss of intended function. The audit findings do not invalidate the effectiveness of the CLB inspection programs, but enhance the effectiveness, ensuring that the inspections are thorough and meet CLB commitments. It can be concluded that the current ongoing inspection programs used by utilities as part of their current licensing basis (CLB) are effective in identifying degradation and deterioration.

2.7 AGING EFFECTS

From the industry issues and maintenance history, the mechanisms potentially causing significant aging effects to the PWR containment have been identified. Table 2-16 summarizes age-related degradation mechanisms applicable to the PWR containment components. Table 2-17 provides a summary of the aging mechanisms and aging effects associated with each subcomponent. Also provided is a reference to the section where these mechanisms and effects are evaluated in Section 3.0. Also given in this table is the reference to the subsection where time-limited aging analyses (TLAAs) are evaluated when applicable for a particular component. Table 2-18 lists the primary and secondary aging effects that are applicable to the PWR containment subcomponents.

The aging effects for aging mechanisms attacking the concrete include cracking, scaling, spalling, increased porosity, and permeability, in addition to loss of strength, both compressive and tensile, and loss in modulus of elasticity. Secondary effects include loss of protective covering and chemistry, i.e., lowering concrete pH and degrading protective oxide films on reinforcing steel, resulting in the corrosion of the embedded steel. All effects are discussed in Section 3.2.

The aging effects for corrosion of steel components include increase in volume through rust by-products, cracking of the surrounding concrete, and reduction in cross-sectional area or thickness. Elevated temperatures result in the reduction in strength and modulus of elasticity

for steel, while irradiation embrittlement results in the increase in yield strength, decrease in the ultimate tensile ductility, and increase in the ductile-to-brittle transition temperature. A secondary effect of elevated temperature is loss of bond strength between embedded steel and concrete. Fatigue results in cracking in steel components and surrounding concrete.

Corrosion in post-tensioning systems results in aging effects including decrease in cross-sectional area, reduction in prestress force, breakage of wires or strands, and leakage of corrosion inhibiting grease. The effects of SCC are the cracking of steel components and reduction in prestress force.

Aging effects associated with aging mechanisms that affect penetrations include loss of material, cracking of steel components, fatigue-induced cracking, and loss of seal or pressure retention.

The effects of settlement, when significant, are loss of support clearance in piping or other systems interconnecting adjacent buildings, inducing additional stress.

Some of the mechanisms resulting in these effects can be classified as both natural aging mechanisms and event-driven mechanisms, while other mechanisms are natural aging mechanisms. Effects of the event-driven mechanisms may result from occurrence of an event, such as leakage of piping systems, and also occur as a result of natural aging. These include aggressive chemical attack of concrete, corrosion and coating degradation of the steel liner or steel containment. All other mechanisms are considered to be natural aging mechanisms.

TABLE 2-16

AGE-RELATED DEGRADATION MECHANISMS

APPLICABLE TO PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES

Material System	Containment Component	Degradation Mechanism
Concrete	Reinforced/Prestressed Concrete Containments Concrete dome Concrete containment wall above grade Concrete containment wall below grade Concrete basemat Concrete containment wall below grade (embedded steel) Concrete basemat (embedded steel) Free-Standing Steel Containments Concrete basemat Concrete basemat (embedded steel)	Chemical Attack Leaching Alkali-aggregate reactions Sulfate attack Bases and acids (aggressive chemicals) Physical Attack Freeze/thaw cycling Thermal exposure/thermal cycling Irradiation Interaction with aluminum Microbial attack Corrosion of embedded steel Settlement
Mild Steel Reinforcement	Reinforced/Prestressed Concrete Containments Dome reinforcing steel Containment wall reinforcing steel above grade Containment wall reinforcing steel below grade Basemat reinforcing steel Free-Standing Steel Containments Basemat reinforcing steel	Corrosion Elevated temperature Irradiation Fatigue
Prestressing/Post-Tensioning	Prestressed Concrete Containments Prestressing tendons	Corrosion Elevated temperature Irradiation Fatigue Loss of prestressing force

TABLE 2-16 (Continued)

AGE-RELATED DEGRADATION MECHANISMS

APPLICABLE TO PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES

Material System	Containment Component	Degradation Mechanism
Liner/Free-Standing Steel Containment	Reinforced/Prestressed Concrete Containments Containment liner interior surface Containment liner exterior surface above grade Containment liner exterior surface below grade Basemat liner interior surface Basemat liner exterior surface Liner anchors above grade Liner anchors below grade Free-Standing Steel Containments (Cylindrical/Spherical/Elliptical Bottom) Embedded shell region Common Components Penetration sleeves Penetration bellows Personnel airlock Equipment hatches	Corrosion Elevated temperature Irradiation Fatigue Strain aging Settlement

TABLE 2-17
SUMMARY OF CONTAINMENT AGING EVALUATIONS

Material, System, or Component	Aging Mechanism	Aging Effects (See Table 2-18)	Aging Assessment - Report Subsection	Aging Management Program(2)	TLAA Evaluation (Section 3.3)
Concrete	Freeze-Thaw(7)	1,2,3,4,A	3.2.1	AMP-5.1 and AMP-5.2	No
	Leaching	2,B	3.2.2	NR	No
	Alkali Aggregate Reaction	1,5,B	3.2.3	NR	No
	Neutron Irradiation Embrittlement	1,5,7	3.2.4	NR	No
	Interaction with Aluminum	7	3.2.5	NR	No
	Concrete Thermal Aging Embrittlement	1,3,7	3.2.6	NR	No
	Aggressive Chemical Attack	1-5,7,B	3.2.7	AMP-5.3 and AMP-5.4(1)	No
	Bond Strength Reduction - Direct Current	8	3.2.8	NR	No
	Fatigue at Penetration Anchors	1,2,A,B	3.2.9	AMP-5.5	Yes
Reinforcing Steel	Corrosion	1,2,8-10,A,B	3.2.10	AMP-5.3 and AMP-5.4(1)	No
	Elevated Temperature	Not Significant	3.2.11	NR	No
	Irradiation (Embrittlement)	11	3.2.4, 3.2.12	NR	No
	Fatigue	1,8,A	3.2.13	NR	No

TABLE 2-17 (Continued)
SUMMARY OF CONTAINMENT AGING EVALUATIONS

Material System or Component	Aging Mechanism	Aging Effects (See Table 2-18)	Aging Assessment - Report Subsection	Aging Management Program[2]	TLAA Evaluation (Section 3.3)
Liner	Elevated Temperatures	Not Significant	3.2.11[3]	NR	No
	Irradiation	11	3.2.12[3]	NR	No
	Fatigue	12	3.2.13[3]	NR	No
	Corrosion	10	3.2.14	AMP-5.5	No
	Coating Degradation	A	3.2.15	AMP-5.5	No
	Fatigue at Attachments and Discontinuities	12	3.2.16	NR	No
Post-Tensioning Systems	Corrosion (Including Microbial) and Concrete Degradation	10,13-15,B	3.2.17	AMP-5.6	No
	Elevated Temperature	20	3.2.18	NR	No
	Irradiation	11	3.2.4, 3.2.19	NR	No
	Prestress Force Losses	B	3.2.20	AMP-5.6	Yes
	Stress Corrosion Cracking	13,16,B	3.2.21	AMP-5.6	No
Steel Embedments	Corrosion	1,4	3.2.22	NR	No
Electrical Penetrations	Material Compatibility	10	3.2.23	NR	No
	Bellows TGSCC	16,17	3.2.24	AMP-5.5	No

TABLE 2-17 (Continued)
SUMMARY OF CONTAINMENT AGING EVALUATIONS

Material System or Component	Aging Mechanism	Aging Effects (See Table 2-18)	Aging Assessment - Report Subsection	Aging Management Program	TLAA Evaluation (Section 3.3)
Mechanical Penetrations	Bellows Fatigue	12,17	3.2.25	AMP-5.5	Yes
	Fatigue	12,17	3.2.9, 3.2.26	AMP-5.5	Yes
	Embrittlement of Gaskets	17	3.2.27	AMP-5.5	No
	Corrosion and SCC	10,16,17	3.2.28	AMP-5.5	No
Fuel Transfer Tube Penetration[5]	Mechanical Wear	10	3.2.29	AMP-5.5	No
	Embrittlement of Gaskets	17	3.2.30	AMP-5.5	No
	Corrosion and SCC	10,16,17	3.2.28, 3.2.31	AMP-5.5	No
Airlocks and Hatches[6]	Mechanical Wear	10	3.2.32	AMP-5.5	No
	Fatigue	12,17	3.2.16, 3.2.33	NR	No
	Embrittlement of Gaskets	17	3.2.27, 3.2.34	AMP-5.5	No
	Loss of Pressure Retention	17	3.2.35	AMP-5.5	No
	Elevated Temperature	Not Significant	3.2.36	NR	No
Foundations	Settlement	1,18	3.2.37	AMP-5.7	No

TABLE 2-17 (Continued)
SUMMARY OF CONTAINMENT AGING EVALUATIONS

Material System or Component	Aging Mechanism	Aging Effects (See Table 2-18)	Aging Assessment - Report Subsection	Aging Management Program	TLAA Evaluation (Section 3.3)
Free-Standing Steel Containment	Strain Aging	19	3.2.38	NR	No
	Fatigue	12	3.2.39	AMP-5.5(4)	Yes
	Corrosion	10	3.2:40	AMP-5.5(1)	No

Notes:

1. For inaccessible below-grade structures
2. Management programs defined by:
   NR = None required
   AMP = Aging management program
3. Not significant aging effects as given in referenced sections
4. Applies only to penetration bellows, see also Subsection 3.2.25.
5. For fatigue, see mechanical penetration
6. For corrosion, see mechanical penetration
7. The freeze-thaw aging management program is applicable only as indicated in Subsection 3.2.1 and is a plant-specific issue.

TABLE 2-18
AGING EFFECT LIST

AGING EFFECTS - PRIMARY	
1.	Cracking of the concrete
2.	Increased porosity and/or permeability of the concrete
3.	Scaling of the concrete surface
4.	Spalling of the concrete surface
5.	Excessive expansion of the concrete (internal or overall)
6.	Exudations and surface deposits
7.	Decrease in tensile and compressive strength and/or modulus of elasticity
8.	Loss of bond strength between reinforcement steel and the concrete
9.	Increase in the volume of reinforcement or embedded steel resulting from the formation of rust by-products, resulting in concrete cracking
10.	Reduction in cross-sectional area or thickness, or loss of material
11.	Decrease in ultimate tensile ductility, increase in ductile-to-brittle transition temperature, increase in yield stress
12.	Fatigue-induced cracking of component
13.	Reduction in prestress force
14.	Breakage of wires or strands
15.	Leakage of corrosion inhibiting medium
16.	Cracking of steel component
17.	Loss of seal or pressure-retaining capability
18.	Added stress induced by loss of supporting system clearances
19.	Decrease in ductility
20.	Decrease in tensile strength of steel
AGING EFFECTS - SECONDARY	
A.	Corrosion resulting from loss of protective concrete cover, coating, or protective concrete chemistry
B.	Loss of strength

3.0 TIME-LIMITED AGING ANALYSES AND AGING EFFECT EVALUATIONS

This section describes each of the significant age-related degradation mechanisms that affect the PWR containment and evaluates how the effects caused by these mechanisms can potentially degrade the intended functions of the PWR containment. This section also evaluates time-limited aging analyses (Section 3.3). All aging effects and time-limited aging analyses that require management during an extended period of operation are identified.

3.1 INDUSTRY ISSUES

Table 3-1 presents a summary of the industry aging issues and the status of NEI/U.S. NRC agreements or positions relative to the significance or nonsignificance of the applicable age-related degradation mechanisms (ARDMs) or ARDM/component combination.

The causes of adverse aging effects in PWR containment structures are discussed and evaluated in Section 3.2, and aging management is discussed in Section 4.0.

The following specific issues have been identified from the list of industry aging issues as plausible and potentially significant age-related degradation mechanisms for the containment structures:

- Aggressive chemical attack (for below-grade concrete containment and basemat)
- Corrosion in embedded steel/rebar (for below-grade concrete containment and basemat)
- Corrosion of steel containment/steel liner (inaccessible)
- Corrosion in tendons
- Loss of prestress forces in post-tensioning systems

Fatigue is potentially significant, but is manageable, as discussed in Sections 3.0 and 4.0.

The following issues are not significant if the containment concrete construction meets prescribed criteria as discussed in Table 3-1 and Section 3.2 (i.e., plant-specific issues only):

- Freeze-thaw damage (only for applicable weathering conditions)
- Alkali-aggregate reaction (only for reactive aggregates where reaction is driven by water infiltration resulting from other plausible ARDMs)
- Settlement (plants founded on soft compressible soils)
- Corrosion of embedded steel/rebar (cracking of concrete must be present)

TABLE 3-1
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
General	Age-related degradation effects	Concrete & Steel Containment Components	Concrete & steel	Open Issues G-7, S-7	The ARDMs are evaluated for significance using the available research & industry data. If acceptance criteria (including a review of plant performance history to ensure that contra-dictory evidence does not exist) are satisfied, then the inspection for that mechanism/component combination is not needed.	Unresolved issue (One-time inspection) *NEI Position* Resolution of the effects of age-related degradation ARDM is based on the review/evaluation of plant-specific features, including appropriate current licensing basis (CLB) documents/ information. General baseline inspections are not warranted if the criteria used in the evaluations are validated. *U.S. NRC Position* A one-time focused inspection of containment is proposed to identify existing degradation mechanisms (if any) and to take necessary corrective actions so that the containments are able to take the challenges during the license renewal term.

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Freeze-thaw	Scaling, cracking, & spalling permeability	Concrete Containments Reinforced/Prestressed • Concrete dome • Concrete containment wall above grade • Concrete containment wall below grade • Concrete basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Concrete basemat	Concrete	Open Issue S-10	Freeze-thaw is nonsignificant for concrete containment structures located in a geographic region of negligible weathering conditions (weathering index <100 day - inch/yr);[1] and if located in severe (weathering index >500 day-inch/yr) or moderate (100-500 day-inch/yr) weathering conditions, the concrete mix design meets the air content & water-to-cement ratio requirements of ASTM C260[2] or equivalently, the ASME Sect. III. Division 2,[3] paragraph CC 2231.7.1.[3] Containment integrity monitoring program includes periodic examination of accessible concrete surfaces in accordance with the procedures of type A[7] integrated leak rate test, or in accordance with ASME Sect. XI, Subsect. IWL, or as part of the tendon surveillance program (where applicable) in accordance with Regulatory Guide 1.35. Aging effects that can impair the function of the containment are assessed and repaired as required under the programs.	*U.S. NRC Position* Freeze-thaw damage of the concrete dome area (S-10) is potentially significant.

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Leaching of Calcium Hydroxide	Increase of porosity & permeability	Concrete Containments Reinforced/Prestressed • Concrete dome • Concrete containment wall above grade • Concrete containment wall below grade • Concrete basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Concrete basemat	Concrete	Closed	Leaching of calcium hydroxide is nonsignificant for concrete containment structures not exposed to flowing water; and for structures that are exposed to flowing water but are constructed using the guidance of ACI 201.2R-77[4] to ensure dense, well-cured concrete with low permeability and controlled cracking through proper arrangement & distribution of reinforcement.	Nonsignificant
Aggressive Chemical Attack	Increase of porosity & permeability cracking & spalling	Concrete Containments Reinforced/Prestressed • Concrete containment wall below grade • Concrete basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Concrete basemat	Concrete	Closed	In cases where containment concrete is exposed to aggressive groundwater (pH <5.5 chloride >500 ppm & sulfate >1500 ppm), periodic inspection of accessible concrete surfaces as part of Type A integrated leak rate test performed under Appendix J, 10 CFR 50[7], or in accordance with ASME Sect. XI, Subsect. IWL[8], exam. category L-A & guidelines of ACI 201.1.[9] The IR found no generic programs that effectively manage the effects of aggressive chemical attack for concrete basemats of free-standing steel (flat bottom/ice condenser) and reinforced/prestressed concrete containment structures and components, and for reinforced/prestressed concrete containment walls below grade. Plant-specific options include a phased approach based on: (1) evaluation of groundwater; (2) inspection and testing of concrete; and (3) management of groundwater.	Accessible concrete surfaces are periodically examined in accordance with the procedures of Type A[7] integrated leak rate test, or in accordance with ASME Section XI, Subsect. IWL.[8] Management for the effects of aggressive chemical attack of concrete surfaces that are not periodically examined due to inaccessibility requires further plant-specific evaluation.

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Reaction with Aggregates (Alkali - Aggregate Reactions)	Expansion & cracking	Concrete Containments Reinforced/Prestressed • Concrete dome • Concrete containment wall above grade • Concrete containment wall below grade • Concrete basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Concrete basemat	Concrete	Open Issue S-12	Reactions with aggregates are nonsignificant for concrete containment structures constructed either from aggregate taken from geographic regions other than those known to yield aggregates suspected of or known to cause alkali-aggregate reactions, [4,10] or from aggregate that was investigated, tested, & subject to petrographic exam equivalent to that required by ASME Section III, Division 2, Class CC.[3] ASTM C295[11] and ASTM C227[12] which showed that the aggregate is non-reactive; or if the aggregate was examined and found potentially reactive, the provisions of ACI 201.2R-77[4] or equivalent were followed.	Unresolved issue *NEI Position* For concrete containment structures that meet the criteria, reaction with aggregates is nonsignificant ARDM. *U.S. NRC Position* Alkaline-aggregate reactions can not be ruled out. Tests involving aggregates alone are not satisfactory in predicting aggregate performance. Alkaline-aggregate reaction may occur after 25 or more years. Use of possolans & low alkali content cement may not control reactions for concretes fabricated using sand-gravel aggregates (S-12) Exposed concrete surfaces subjected to significant wetting or ponding should be visually examined for signs of AAR if potentially reactive aggregates were used. Where evidence of AAR is present, samples should be removed and petrographically examined and tested to determine the problem and impact on intended component function.

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Aggressive Chemical Attack	Increase of porosity & permeability cracking & spalling	Concrete Containments Reinforced/Prestressed • Concrete dome • Concrete containment wall above grade	Concrete	Closed	Nonsignificant	Degradation caused by aggressive chemical attack is nonsignificant for concrete containment structures not exposed to aggressive environment (pH <5.5), or to chloride or sulfate solutions beyond defined limits (>500 ppm chloride,[5] and 1500 ppm sulfate);[6] or if exposed to groundwater that exceeds the pH, chloride, or sulfate limits, the exposure is for intermittent periods only.
Reaction with Aggregates (Alkali - Aggregate Reactions)	Expansion & cracking	Concrete Containments Reinforced/Prestressed • Concrete dome • Concrete containment wall above grade • Concrete containment wall below grade • Concrete basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Concrete basemat	Concrete	Closed	Degradation from exposure to elevated temperature is nonsignificant for concrete containment structures general areas maintained at operating temperatures <150°F and local area temperatures 200°F[3,13] or for structures that operate above these limits, plant-specific justification is provided in accordance with ACI 349-85.[13]	Nonsignificant
Elevated Temperature	Loss of strength & modulus	Concrete Containments Reinforced/Prestressed • Dome reinforcing steel • Cont. wall reinforcing Steel above grade • Cont. wall reinforcing steel below grade • Basemat reinforcing steel Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Basemat reinforcing steel	CS reinforcing steel	Closed	Normal bulk operating temperatures within PWR containment structures are 120-150°F which are well below the 600°F level at which the structural integrity of rebar/concrete combination begins to be significantly affected.[14]	Nonsignificant

CS - Carbon Steel

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Elevated Temperature	Loss of strength & modulus Increase in relaxation (creep) losses	Concrete Containments Prestressed • Prestressing tendons	CS prestressing steel	Closed	PWR containment prestressing tendons subjected to temperatures less than 140°F will not experience a decrease in strength or modulus of elasticity. Periodic inservice examination, in accordance with Examination Category L-B (tendon prestressing force and elongation measurement), is an effective program for managing the potentially significant effects of elevated temperature on stress relaxation losses in prestressing tendons	Nonsignificant *U.S. NRC Position:* The U.S. NRC accepted the IR programs, primarily because of the similarity to Reg. Guide 1.35, as effectively managing tendon prestressing losses in PWR prestressed concrete containment structures.
Elevated Temperatures	Loss of strength & modulus	Concrete Containments Reinforced/Prestressed • Containment liner int. surface • Containment liner above grade exterior surface • Containment liner below grade exterior surface • Basemat liner interior surface • Basemat liner exterior surface • Liner anchors above gr. • Liner anchors below gr. Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom • Containment shell int. surface • Containment shell ext. surface • Embedded shell region • Sand pocket region	CS prestressing steel	Closed	Normal operating temperatures within PWR containment structures are 120-150°F, which are well below the 700°F level at which the yield strength and modulus of elasticity of the steel liners, free-standing steel containment shells, and associated components begin to be significantly affected.[14]	Nonsignificant

CS - Carbon Steel

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Elevated Temperatures (Continued)		Free-Standing Steel Cont. with Flat Bottom & an Ice Condenser • Dome shell interior surface • Dome shell exterior surface • Cylindrical shell int. surface • Cylindrical shell ext. surface • Embedded shell region • Basemat liner • Liner anchors Common Components • Penetration sleeves • Penetration bellows • Personnel airlock • Equipment hatches	SS, CS CS			
Irradiation of Concrete	Loss of strength & modulus	Concrete Containments Reinforced/Prestressed • Concrete dome • Concrete containment wall above grade • Concrete containment wall below grade • Concrete basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Concrete basemat	Concrete	Closed	The neutron fluence levels & maximum integrated gamma doses incurred by PWR containment concrete do not exceed the level at which measurable degradation of concrete strength properties occurs (10^{19} n/cm^2 & 10^{10} rads, respectively).[5,16]	Nonsignificant
Irradiation of Steel	Loss of fracture toughness	Concrete Containments Reinforced/Prestressed • Dome reinforcing steel • Containment wall Reinforcing steel above grade • Basemat reinforcing steel Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Basemat reinforcing steel	CS reinforcing steel	Closed	The cumulative neutron flux experienced by reinforced concrete PWR containment structures is far below the threshold level of 10^{19} n/cm^2 for degradation of reinforcing steel properties.[16]	Nonsignificant

CS - Carbon Steel, , SS - Stainless Steel

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Irradiation of Steel	Loss of fracture toughness	Concrete Containments Prestressed • Prestressing tendons	CS prestressing steel	Closed	The neutron fluence levels and maximum integrated gamma doses incurred by PWR containment tendons & corrosion inhibitors are below the threshold to incur age related degradation (<4 x 10^{16} n/cm^2 & 10^{10} rads. respectively).[13]	Nonsignificant
Irradiation of Steel	Loss of fracture toughness	Concrete Containments Reinforced/Prestressed • Containment liner int. surface • Containment liner above grade exterior surface • Containment liner below grade exterior surface • Basemat liner interior surface • Basemat liner exterior surface • Liner anchors above gr. • Liner anchors below gr. Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom • Containment shell int. surface • Containment shell ext. surface • Embedded shell region • Liner anchors	CS	Closed	The neutron fluence level incurred by PWR containment liners or free-standing steel containment shells is far below the level of 2 x 10^{17} n/cm^2 (>1 MeV), which could cause a change in mechanical or physical properties.[17]	Nonsignificant

CS - Carbon Steel

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Irradiation of Steel (Continued)		Free-Standing Steel Cont. with Flat Bottom & an Ice Condenser • Dome shell interior surface • Dome shell exterior surface • Cylindrical shell int. surface • Cylindrical shell ext. surface • Embedded shell region • Basemat liner • Liner anchors Common Components • Penetration sleeves • Penetration bellows • Personnel airlock • Equipment hatches	 SS, CS CS			
Corrosion of Embedded Steel	Cracking, spalling, loss of bond & loss of material	Concrete Containments Reinforced/Prestressed • Concrete dome • Concrete containment wall above grade • Dome reinforcing steel • Containment wall Reinforcing steel above grade	Embedded CS & reinforcing CS (rebar) in concrete	Open Issue S-42	Nonsignificant for concrete not exposed to aggressive environment, pH < 11.5 or chlorides >500 ppm;[18] or if exposed to aggressive environment concrete has relatively high strength [27.6 MPa (4 ksi)], low water-to-cement ratio (0.35-0.45), adequate air entrainment (3-6%), low permeability, and designed in accordance with ACI 318[5] or ASME Sect. III, Div. 2.[3] An inspection/ walkdown to look for signs of rebar corrosion will be performed for older plants when the above criteria is not met by review of the CLB. The phased program to evaluate below grade concrete described on sheet 3 of 18 for aggressive chemical attack will also apply to corrosion of embedded reinforcing steel in below grade components.	*NEI Position:* For concrete containment structures that meet the criteria, corrosion of embedded steel or rebar is nonsignificant ARDM. *U.S. NRC Position:* The limits on chloride content in concrete in ACI 318 do not apply to early vintage plants (S-42). A plant inspection/walkdown is necessary to inspect for signs of rebar corrosion in older plants where it cannot be established that the criteria is met based on review of the CLB.

CS - Carbon Steel, SS - Stainless Steel

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Corrosion of Embedded Steel	Cracking, spalling, loss of bond, & loss of material	Concrete Containments Reinforced/Prestressed • Concrete containment wall below grade • Concrete basemat • Containment wall reinforcing steel below grade • Basemat reinforcing steel • Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Concrete basemat • Basemat reinforcing steel	Embedded CS & reinforcing CS (rebar) in concrete	Open Issue S-42.	In cases where containment concrete is exposed to aggressive groundwater (pH <11.5 chloride, >500 ppm, & sulfate >1500 ppm) periodic inspection of accessible concrete surfaces as part of Type A integrated leak rate test performed under Appendix J, 10 CFR 50,[7] or in accordance with ASME Sect. XI, Subsect. IWL,[8] exam category L-A & guidelines of ACI 201.1.[9] An inspection/ walkdown to look for signs of rebar corrosion will be performed for older plants where the above criteria is not met by review of the CLB. The phased program to evaluate below grade concrete described on sheet 3 of 18 for aggressive chemical attack will also apply to corrosion of embedded reinforcing steel in below grade components.	*NEI Position:* Accessible concrete surfaces are periodically examined in accordance with the procedures of Type A[7] integrated leak rate test, or in accordance with ASME Sect. XI, Subsect IWL.[8] *U.S. NRC Position:* The limits on chloride content in concrete in ACI 318 does not apply to early vintage plants (S-42). A plant inspection/ walkdown is necessary to inspect for signs of rebar corrosion in older plants where it cannot be established that the criteria is met based on review of the CLB.
Corrosion of Liner and Related Components (Above Grade)	Loss of material, stress corrosion cracking, leakage	Concrete Containments Reinforced/Prestressed • Containment liner interior surface • Containment liner above grade exterior surface • Basemat liner interior surface • Liner anchors above grade Common Components • Penetration sleeves • Dissimilar metal welds • Personnel airlock • Equipment hatches	CS	G-5, G-12, S-5, S-16, S-38 to S-40, S-62	Galvanic corrosion & corrosion due to aggressive aqueous solutions will not occur if dissimilar metals are not used in construction & if aggressive groundwater (chlorides >500 ppm) is not present. SCC is not significant because PWR containment liners only experience compressive stresses due to dead load & prestress (tensile stresses and a corrosive environment are necessary for SCC).	Nonsignificant

CS - Carbon Steel

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Corrosion of Steel Containment and Related Components	Loss of material, stress corrosion cracking, leakage	Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom • Containment shell interior surface • Containment shell exterior surface Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Dome shell interior surface • Dome shell exterior surface • Cylindrical shell exterior surface mmon Components • Penetration bellows	CS SS	G-5, G-12, S-5, S-6, S-16, S-38 to S-40 Closed	Galvanic corrosion & SCC are not significant ARDMs if dissimilar metals are not used in the construction of PWR free-standing steel containment; & in the case of SS bellows assemblies for CS vent lines or pipe sleeves if the materials are protected by shields from corrosive environment.	Nonsignificant
Corrosion of Liner (Below Grade)	Loss of material	Concrete Containments Reinforced/Prestressed • Containment liner below grade exterior surface • Basemat liner exterior surface • Liner anchors below grade e-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom • Embedded shell region • Sand pocket region Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Embedded shell region • Basemat liner • Liner anchors	CS	G-5, G-8, G-8, G-11, G-15, G-16, S-24, S-25, S-36, S-37, S-54, S-64, S-66, S-69, S-72, S-73, S-74, S-77 Open	Periodic examination & monitoring of accessible areas in accordance with ASME Sect. XI, Subsect. IWE,[8] exam, categories E-D, E-F, & E-P; & areas exempt from inspection monitored to maintain required wall thickness minimums by UT performed[19] in accordance with existing standards are effective programs. Focused inspections will be performed for areas where aggressive aqueous solutions can collect (G-07, 5-70). The program to identify and evaluate corrosion of below grade or inaccessible steel structures is described in Section 6.2 of the IR report.	Unresolved Issue: *NEI Position*: ASME Sect. XI,[8] Subsect. IWE, requires visual examination of accessible surfaces prior to any Type A test to uncover evidence of structural degradation; supplementary methods for condition monitoring confirmation of minimum required wall thickness by UT methods;[19] affected areas evaluated in accordance with criteria of ASME Sect. III,[20] and repair & replacement in accordance with ASME Sect. XI,[8] Subsect. IWE-4000 & 7000.

CS - Carbon Steel SS - Stainless Steel

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Corrosion of Liner (Below Grade) (Continued)						*U.S. NRC Position:* The effects of corrosion are potentially significant for the embedded shell region and the sand pocket region of free-standing steel containments (cylindrical/ spherical/elliptical bottom) and for containment liner surfaces where aggressive aqueous solutions can collect. Focused inspections are recommended for surface locations where aggressive aqueous solutions may collect.
Corrosion of Tendons	Loss of material	Concrete Containments Prestressed • Prestressing tendons	CS	G-8, G-9, G-19, G-11, G-16, S-9, S-18, S-42, S-50, S-64 Open issue S-61	Periodic inservice examination of tendon anchorage hardware in accordance with the provisions of RG 1.35[21] or the requirements of ASME Sect. XI,[8] Subsect. IWL, including visual examination of tendon anchorage hardware, evaluation of corrosion protection medium, & identification & testing of any free water; repair & replacement; are effective programs in managing degradation by corrosion of prestressing tendons & anchor heads.	Unresolved issue: *NEI Position:* RG 1.35 & ASME Sect. XI, Subsect. IWL require testing & examination of tension & leakage of corrosion protection medium; VT-1 includes anchor head, bearing plates, wedges, buttonheads, shims, & concrete; acceptance criteria IWL-3221.2 include absence of physical damage, corrosion limits; & minimum specified material properties; IWL-2525-1 examines corrosion protection medium & any free water; repair & replacement. *U.S. NRC Position:* IWL lacks certain criteria in RG 1.35 to address long term effects of large amounts of grease leakage on concrete compressive strength and bonding with rebar (S-61).

CS - Carbon Steel

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Loss of Prestress Force	Reduction of design margin	Concrete Containments Prestressed • Prestressing tendons	CS	G-9, G-14, S-18, S-33, S-45, S-47, S-48, S-52, S-53 Closed	The program to monitor loss of prestress force consists of periodic monitoring of pre-stressing losses in accordance with tendon lift-off test of RG 1.35,[21] validation with predictions of prestressing loss; identification of reportable conditions of RG 1.35; documentation of RG 1.16[22] & plant-specific evaluation & corrective actions are effective in managing the effects of pre-stressing loss.	Resolved Issue: Inspection & load monitoring to detect progressive reduction in the levels of prestress evaluation for the license renewal term using RG 1.35,[21] & corrective action.
Fatigue	Cumulative fatigue damage	Concrete Containments Reinforced/ Prestressed • Concrete dome • Concrete containment wall above grade • Concrete containment wall below grade • Concrete basemat • Dome reinforcing steel • Containment wall reinforcing steel above grade • Containment wall reinforcing steel below grade • Basemat reinforcing steel Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom • Containment shell int. surface • Containment shell ext. surface Free-Standing Steel Cont. with Flat Bottom & an Ice Condenser • Dome shell interior surface • Dome shell exterior surface • Cylindrical shell int. surface • Cylindrical shell ext. surface • Concrete basemat • Basemat reinforcing steel Common Components • Personnel airlock • Equipment hatches	Concrete including embedded CS & reinforcing CS (rebar) in concrete CS	G-12, S-5, S-21, S-38 to S-40 Closed	Containment concrete, reinforcing steel, prestressing systems, steel liners, & free-standing steel containments are designed to have good fatigue strength properties (105 cycles) of below yield load application in accordance with ASME Sect. III, Division 2,3 & ACI 215R-7423 codes. Potential low-cycle fatigue due to localized elevated temperatures are not anticipated to be significant for these components.	Nonsignificant

CS - Carbon Steel

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Fatigue	Cumulative fatigue damage	Common Components • Penetration sleeves and pressure-retaining attachments • Penetration bellows	CS SS,CS	G-11, G-16, S-14, S-64, S-68 Open Issue G-3, G-4	Fatigue reanalysis conducted in accordance with ASME Sect. III,[20] Subsect. NB, to show that fatigue usage factors are maintained below unity throughout the license renewal term, monitoring of penetration temperatures may be required to establish the magnitude & frequency of transients; ISI in accordance with ASME Sect. XI[8] Subsect. IWE, to insure that component integrity is maintained in the presence of known or suspected fatigue damage, including a flaw are effective to manage the effects of fatigue damage accumulation or fatigue crack growth. Visual inspection of the response at hot penetration sleeve anchorage areas to be performed as part of the ASME Sect. XI, Subsect. IWE ISI.	Unresolved issue *NEI Position:* Fatigue re-analysis of penetrations in accordance with ASME Sect. III, Subsect. NB,[20] & ISI in accordance with ASME Sect. XI, subsect, IWE,[8] exam, category E-B requires visual VT-1 of containment penetration welds, including bellows seal circumferential weld. *U.S. NRC Position:* Flaw evaluations & appropriate references should be included for fatigue of bellows assemblies (G-3). Fatigue of penetration sleeve anchors can be induced by thermal cyclic loading & may not be detectable by the leak rate tests (G-4). These areas should be examined during plant inspection/ walkdown.
Concrete Interaction with Aluminum	Loss of strength	Concrete Containments Reinforced/Prestressed • Concrete dome • Concrete containment wall above grade • Concrete containment wall below grade • Concrete basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Concrete basemat	Concrete	Closed	Adverse effects of concrete interactions with aluminum would have been identified during the initial structural acceptance test prior to initial operation. If no degradation of concrete strength was noted during initial structural testing, or if aluminum piping was not used for concrete placement, then concrete interaction with aluminum is not significant.	Nonsignificant

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

Aging Related Degradation Mechanism	Aging Effects	Components	Materials	U.S. NRC Comment Number	Criteria/Program	NEI/U.S. NRC Agreement or Position
Settlement	Cracks, distortion, increase in component stress level	Concrete Containments Reinforced/Prestressed • Concrete basemat Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Concrete basemat	Concrete	Open issue S-63	Structure settlement monitoring initiated during construction phase to confirm that actual settlement is consistent with the allowances included in design basis & continued settlement monitoring during operation for sites with soft soil and/or significant changes in groundwater conditions. The program for settlement monitoring for sites susceptible to continued inelastic settlement after 20 years or more includes monitoring the clearances between the containment and adjacent structures and the effect of differential movement between the adjacent buildings on pipe systems supported by both buildings.	Unresolved issue *NEI Position:* Structure settlement monitoring during construction, & continued monitoring during operation for sites with soft soil and/or significant changes in groundwater conditions. *U.S. NRC Position:* The monitoring program needs to examine clearances between the containment adjacent buildings and the effects of differential settlement movement between adjacent buildings on piping attached to both.
Strain Aging	Loss of fracture toughness (thermal aging embrittlement)	Free-Standing Cylindrical & Spherical Steel Containment with Elliptical Bottom • Containment shell int. surface • Containment shell ext. surface • Embedded shell region • Sand pocket region Free-Standing Steel Containment with Flat Bottom & an Ice Condenser • Dome shell interior surface • Dome shell exterior surface • Cylindrical shell int. surface • Cylindrical shell ext. surface Common Components • Penetration sleeves • Penetration bellows • Personnel airlock • Equipment hatches	CS CS CS	Closed	Dynamic strain aging is nonsignificant for free standing steel containment structures that do not allow loads to exceed the elastic limit. Static strain aging is nonsignificant for free standing steel containment structures that were not cold worked; or if cold worked during the forming process, the plates were normalized or stress relieved or both after forming with minimal (<5%) subsequent cold working	Nonsignificant

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

1. ASTM C33-82, "Standard specification for Concrete Aggregates," American Society for Testing and Materials, Philadelphia, PA.
2. ASTM C260-77, "Specification for Air Entraining Admixture for Concrete, American Society for Testing and Materials, Philadelphia, PA.
3. ASME B & PV Code, "Boiler and Pressure Vessel Code of Design and Construction Practices," Section III, Division 2: "Code for Concrete Reactor Vessel and Containments," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017, 1986 Edition. Subsection CC "Concrete Containments."
4. ACI 201.2R-77, "Guide to Durable Concrete," American Concrete Institute.
5. ACI 318, "Building Code Requirements for Reinforced Concrete," American Concrete Institute.
6. "Concrete Degradation Monitoring and Evaluation," N. Prasad et al., NUREG/CP-0100, Proc. Intl. Nuclear Power Plant Aging Symposium, U.S. Nuclear Regulatory Commission, Washington DC.
7. CFR 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, US Government Printing Office, Washington, DC.
8. ASME B&PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practice," Section XI: "Rules for In-Service Inspection (ISI) of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017, 1992 Edition with 1992 Addenda.
   Subsection IWE: "Requirements for Class MC and Metallic Liners of Class CC Components of Light-Water Cooled Power Plants."
   Subsection IWL: "Requirements for Class CC Concrete Components of Light-Water Cooled Power Plants."
9. ACI 201.1R-68, "Guide for Making a Condition Survey of Concrete in Service," American Concrete Institute, Detroit, MI, Revised 1984.
10. "Petrographic Identification of Reactive Constituents in Concrete Aggregate," B. Mather, ASTM Proc. Vol. 48, American Society of Testing and Materials, Philadelphia, PA, pp. 1120-1125, 1948.
11. ASTM C295-85, "Practice for Petrographic Examination of Aggregate for Concrete," American Society of Testing and Materials, Philadelphia, PA.
12. ASTM C227-87, "Test Method for Potential Alkali Reactivity of Cement-Aggregate Combination," American Society of Testing and Materials, Philadelphia, PA.
13. ACI 349-85, "Code Requirements for Nuclear Safety Related Concrete Structures," American Concrete Institute.
14. "Resistance to High Temperatures," P. Smith, in Significance of Tests and Properties of Concrete - Making Materials, American Society for Testing and Materials, STP 169B, Chapter 25, 1978.
15. ACI Publication SP-55, "The Effects of Nuclear Radiation on the Mechanical Properties of Concrete," H. R. Hillsdorf, J. Kropp, and H. J. Koch, Dougles McHenry Intl. Symp. on Concrete and Concrete Structures, American Concrete Institute, 1978.
16. "Concrete, Cements, Mortars, and Grouts," H. E. Hungerford, et al., Engineering Compendium on Radiation Shielding - Subsection 9.1.12. Volume II. Spring-Verlag New York, Inc. NY, 1975.
17. "Neutron Irradiation Embrittlement of Reactor Pressure Vessel Steels," L. E. Steele, International Atomic Energy Agency, Vienna, Austria, 1975.
18. "Composition and Properties of Concrete," Second Edition, G. E. Troxell, H. E. Davis, and J. W. Kelly, McGraw-Hill, 1968.

TABLE 3-1 (Continued)
SUMMARY OF PRESSURIZED WATER REACTOR CONTAINMENT STRUCTURES AGING EVALUATION AND STATUS OF NEI/U.S. NRC AGREEMENTS

19. ASTM E797-81, Practice for Measuring Thickness by Manual Ultrasonic Pulse-Echo Contact Method," American Society of Testing and Materials, Philadelphia, PA, 1981.

20. ASME B&PV Code - "Boiler and Pressure Vessel Code of Design and Construction Practices," Section III: "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers (ASME), 345 East 47th Street, New York, NY 10017.
    Subsection NB: "Class 1 Components."
    Subsection NE: "Class MC Components."

21. Regulatory Guide 1.35, Revision 3, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures," U.S. Nuclear Regulatory Commission, July 11, 1990.

22. Regulatory Guide 1.16, Revision 4, "Reporting of Operating Information - Appendix A, Technical Specification," U.S. Nuclear Regulatory Commission, August 1975.

23. ACI 215 R-74, "Consideration for Design of Concrete Structures Subjected to Fatigue Loading," American Concrete Institute, 1986.

3.2 AGING MANAGEMENT REVIEW

3.2.1 Freeze-Thaw – Concrete

Mechanism Description

Repeated cycles of freezing and thawing can cause severe damage to susceptible concrete. The aging effects and indications of frost attack are pattern cracks and the eventual disintegration of the concrete surface through increased surface porosity and permeability, scaling, and spalling. Disintegration of the concrete surface can reduce the protective cover of the concrete over the reinforcing steel, eventually leading to corrosion of the reinforcing steel and other degradation mechanisms.

The cement matrix and certain coarse aggregates may be susceptible to frost attack under freezing conditions. The resistance of the cement mortar matrix to frost attack is dependent on the amount of entrained air, the spacing of the entrained air bubbles, and the permeability of the concrete to water penetration. Lack of entrained air or too large a spacing of the entrained air bubbles can result in degradation when moisture is available and the concrete lacks permeability resistance to the penetration of the moisture. Properly proportioned, manufactured, placed, finished, and cured concrete typically ensures concrete that is relatively impermeable to water penetration. Damage may occur under conditions of partial to full saturation with the critical level of saturation at about 85 percent for most concretes [Ref. 8].

The resistance of absorptive coarse aggregates to freeze-thaw damage depends primarily on the absorption characteristics (volume of fine pores) of the aggregate, the presence of moisture to saturate the aggregate, and the permeability of the hardened cement mortar matrix to the passage of water. For damage to occur by freezing of absorptive coarse aggregates, the aggregate must be saturated [Ref. 9]. Saturation can only occur when water is available from an outside source. Since concrete in containments experience seasonal drying through the exposed surface, the critical saturation of the coarse aggregates does not occur. Therefore, the coarse aggregate in air-entrained concrete is generally not damaged by freezing, even when it is absorptive aggregate.

Freezing and thawing is potentially a problem for exposed concrete only in the northern states that experience severe weather conditions.

According to Reference 10, concrete with high cement content and low water-cement ratio (0.36±), and of optional air entrainment (3 to 6 percent), also has good resistance to freezing and thawing because of its high density, which provides a high impermeability that limits the entry of water into the concrete capillary system. Typically, air entrainment is incorporated for containment concrete. The entrained air bubbles provide the relief for pressures developed by free water as it freezes and expands.

Aging Effect Evaluation

Freeze-thaw damage typically occurs on relatively flat concrete surfaces, such as pavements, where water can remain in contact with the concrete. The flat surfaces on top of the ring girder for certain containment designs may be subject to local areas of freeze-thaw damage, depending on the effectiveness of the drainage design provisions to limit the accumulation of water.

The aging effects of freeze-thaw, only if unmitigated, could potentially result in the degradation of the pressure-retaining capability of the containment and the structural capacity, which support intended functions. These include the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Freeze-thaw damage, if present, is expected to be a local condition and by itself will not affect the intended strength function of the containment to account for design loads. Freeze-thaw damage starts at the surface and is readily detected by surface inspections.

The following three conditions are necessary for significant freeze-thaw damage to containment concrete components:

1. Concrete must be exposed to water and be capable of becoming partially or fully saturated with water.

2. Concrete must be located in a geographical region where the weathering index, as defined by ASTM C33, is at least a weathering index of 100 [Ref. 10].

3. Concrete mix design must be inadequate for the exposure conditions.

If either or both conditions (1) and (2) above are not met, then the concrete components are not subject to freeze-thaw damage.

Even in the case where both conditions (1) and (2) are present, freeze-thaw damage is still not significant where the concrete mix design meets the air content and water-cement ratio requirements of ASTM C260, or the equivalent requirements in ASME Code Section III, Division 2, Paragraph CC-2231.7.1 [Ref. 11].

Aging Effect Management

This degradation mechanism is potentially significant in colder geographic regions, but only when the conditions under the aging effect evaluation are not met. This mechanism is therefore a plant-specific issue and does not require industry-wide consideration. The aging effects, cracking, spalling, scaling, etc., caused by this potentially significant degradation mechanism can be managed through an inservice examination (ISE) program, as applicable on a plant-specific basis. This ISE program will identify the indications of freeze-thaw and provide criteria

for repair and subsequent inspection. The ISE program is described in detail in Subsection 4.1.6, aging management options AMP-5.1 and AMP-5.2.

3.2.2 Leaching of Calcium Hydroxide - Concrete

Mechanism Description

Water from rain or melting snow moving through concrete at locations such as cracks, poor construction joints, and areas of inadequate consolidation can dissolve the calcium hydroxide (lime) in the concrete. The rate of leaching depends on the temperature, chemistry, and mobility of the water, and the amount of soluble constituents in the cement paste. The water must move through the concrete to cause leaching; water moving over the surface does not cause significant leaching.

Evidence of leaching is typically the white deposits left on the surface of concrete that has been subjected to cycles of wetting and drying. These white deposits are the free lime from the concrete combined with carbon dioxide from the air. When the calcium hydroxide has been leached over a period of time, the remaining by-products of the decomposition include silica and aluminum gels having no strength.

Aging Effect Evaluation

Aging effects for leaching of calcium hydroxide include increased concrete permeability and degradation of the protective concrete chemistry that protects embedded steel. Leaching of calcium hydroxide (lime) from concrete over a long period of time increases the concrete permeability, eventually causing a reduction in concrete strength. Leaching increases cement matrix porosity making the concrete more susceptible to other forms of aggressive attack. Leaching lowers the concrete pH and can degrade the protective oxide film around steel reinforcing.

Significant leaching, along with other types of degradation such as reinforcing corrosion that may follow leaching, is necessary for the load carrying function of the containment to be impaired.

The aging effects of leaching, only if unmitigated and extensive, could potentially result in the degradation of the pressure-retaining capability of the containment and the structural capacity, which support intended functions. These include the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

The following two conditions are necessary for leaching to be a significant cause of adverse aging effects:

- The structure must be exposed to water that flows through the concrete. (Water running off the surface of the concrete or stagnant water does not cause significant leaching.)

- Defects in the concrete such as cracks, voids, or low strength are necessary to permit movement of the water through the concrete.

If the concrete structure(s) is not exposed to flowing water, or if the concrete is dense and well-cured with a low permeability, leaching is not significant and requires no further evaluation. This is typically the case for containment concrete components that have been constructed in accordance with ACI recommendations such as in ACI 201.2R-77 [Ref. 9]. Therefore, leaching of calcium hydroxides is not a significant degradation mechanism for PWR containment concrete components.

Aging Effect Management

Due to the lack of detrimental aging effect caused by leaching of calcium hydroxide from the concrete, there is no need for the identification of aging management options.

3.2.3 Alkali Aggregate Reaction - Concrete

Mechanism Description

Alkali-aggregate reactions (AARs) are chemical reactions of certain aggregates in concrete that occur when the concrete is exposed to large amounts of water on a regular basis. Several different types of AARs may occur depending on the mineralogy of the aggregate. In most cases, the alkalies in cement react to a minor degree with the aggregate, increasing the bond. In the case of AARs, the reactions result in the formation of certain solid constituents, which can expand and crack the concrete when exposed to water. The three most significant types of AARs are the alkali-silicate, cement-aggregate, and the expansive carbonate reactions [Ref. 9].

Three chemical reactions occur between aggregates and alkalies:

- Alkali-silica reactions result when silica minerals from certain geographical areas react with alkaline solutions. Reactive material in the presence of potassium, sodium, and calcium hydroxides derived from the cement reacts to form an alkali-silicate complex (solids) that can expand when exposed to water that penetrates the concrete. A list of known deleterious reactive rocks is given in Table 5.2.1 of Reference 9.

- Cement-aggregate reaction can occur between alkalies in the cement and some siliceous constituents of certain sand-gravel aggregates from geographical areas in Kansas, Nebraska, and Wyoming [Ref. 9]. The damage results from moderate interior expansion caused by the AARs together with surface drying shrinkage under severe drying conditions in western areas such as Kansas and Nebraska.

- Carbonate aggregate and alkalies can produce expansion and cracking effects. This AAR occurs with the presence of certain argillaceous dolomitic limestones from some midwestern and eastern states [Ref. 9].

Laboratory test methods are available to test for and to confirm both alkali-silicate reactivity and alkali-carbonate activity [Ref. 12].

A field test method is available to identify alkali-silicate reaction [Ref. 13], but none are available for the alkali-carbonate reaction. To identify an alkali-silicate reaction, a uranyl acetate solution is applied to the surface of the concrete. The gel from the alkali-silicate reaction absorbs the uranyl ion. The gel then fluoresces yellowish-green under ultraviolet light.

AARs can result in the following aging effects [Ref. 10]:

- Excessive internal and overall expansion.
- Cracking, usually of random patterns on a large scale, occurs in unreinforced concrete. Long cracks paralleling the direction of the reinforcing steel along with random bridging cracks occurs in reinforced concrete.
- Cracks that may be large at the concrete surfaces but that extend into the concrete only a distance of 6 to 18 inches.
- Silica gelatinous exudations and whitish amorphous deposits on the surface or within the mass of the concrete, especially in voids and adjacent to some affected pieces of aggregate. The gel is not present with alkali-carbonate reactions.
- Peripheral zones of reactivity, alteration, or infiltration in the aggregate particles, particularly those particles containing opal and certain types of acid and intermediate volcanic rocks.
- Lifeless, chalky appearance of freshly fractured concrete.
- Eventual loss of strength after extensive damage.

Aging Effect Evaluation

The aging effects of AARs (only if unmitigated and extensive) could potentially result in the degradation of the pressure-retaining capability of the containment and the structural capacity, which support intended functions. These include the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

In many cases, AARs have a minor effect, resulting in low stresses. Reinforcing steel usually is effective in limiting crack width. However, AAR degradation effects, if extensive, can result in the loss of structural load-carrying capability necessary to ensure the containment function. Degradation effects such as cracking can increase the permeability to water leading to further degradation.

The time required for AAR to manifest in effects such as cracking and gel exudation varies considerably, but signs of the disruption are usually observed within 4 to 7 years. The deterioration takes the form of pattern cracking where the concrete is free to expand. After the crack pattern forms, there is further opening of the cracks as the interior concrete expands. Where the concrete is restrained in one direction, the pattern of the cracks parallels the axis of restraint [Ref. 10].

Many PWR containments are sufficiently old that AARs, if present, should be detectable by the identification of reaction products by petrographic analysis or by observance of surface cracking. However, AARs can be latent depending on the availability of water necessary for the reaction. Petrographic analysis can determine the latent possibility of AARs.

AAR is dependent on a renewable supply of moisture. Concrete components potentially subject to these reactions include the areas continuously wet or alternately wet and dry, such as below-grade basemat and shell areas in contact with groundwater and not protected by waterproofing, as well as areas of the dome and ring girder.

AAR can potentially impact the load-carrying function of the containment. However, because of the use of nonreactive aggregates or the design of an appropriate concrete mix incorporating low alkali cement and/or pozzolan when reactive aggregates were used, AAR has not been identified as a significant degradation mechanism for nuclear power plants.

Aging Effect Management

Due to a lack of detrimental aging effect caused by AAR in the concrete, there is no need for the identification of aging management options.

3.2.4 Irradiation – Concrete

Mechanism Description

Exposure of concrete to neutrons and/or gamma rays above certain levels can cause changes in the concrete properties. An aging effect resulting from prolonged exposure of concrete to irradiation is a decrease in tensile and compressive strengths, as well as the modulus of elasticity. Irradiation can also result in other effects, such as internal volume changes (expansion) and cracking. Because the degradation affects properties, it is not readily observable by physical indications.

Aging Effect Evaluation

Intended functions degraded by loss of concrete strength or modulus of elasticity are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

In Reference 12, the radiation levels are compared to threshold limits. It was found that neutron and gamma radiation effects are not significant for any PWR containment structure or component. The summary of the evaluation given in Reference 12 is provided below:

> (1) PWR containment concrete is exposed to neutron and gamma radiation fluences below the degradation threshold limits of 10^{19} n/cm^2 and 10^{10} rads, respectively; (2) PWR concrete containment reinforcing steel is exposed to neutron radiation fluences below the degradation threshold limit of 10^{19} n/cm^2; (3) PWR prestressed concrete containment prestressing tendons are exposed to neutron radiation fluences below the degradation threshold limit of 4 X 10^{16} n/cm^2; and (4) PWR concrete containment steel liners and PWR steel containment shells are exposed to neutron irradiation fluences below the degradation threshold limit of 2 X 10^{17} n/cm^2.

Aging Effect Management

Due to lack of detrimental aging effect caused by irradiation, there is no need for the identification of aging management options.

3.2.5 Interaction with Aluminum – Concrete

Mechanism Description

Concrete strength reduction can occur when concrete is placed by pumping through aluminum piping. After this phenomenon was identified around 1969, specifications were written to preclude the use of aluminum piping. The interaction between the aluminum and concrete occurs in the period immediately after concrete placement when strength development is most rapid.

The aging effect of concrete interaction with aluminum is reduced concrete strength, potentially impacting the load-carrying function of the containment. Concrete interaction with aluminum is only a consideration for plants where containment concrete was placed using aluminum piping.

Aging Effect Evaluation

Intended functions degraded by loss of concrete strength are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings. For containments where aluminum piping was used to place concrete, any significant reduction in concrete strength would have been detected during the structural acceptance testing.

Aging Effect Management

Due to lack of detrimental aging effect caused by aluminum interaction, there is no need for the identification of aging management options.

3.2.6 Concrete Thermal Aging Embrittlement

Mechanism Description

Thermal aging is the result of prolonged exposure of concrete to elevated temperatures. Aging effects include creep, loss of compressive strength and modulus of elasticity, surface scaling, and cracking. The effects of elevated temperatures on concrete properties of creep, compressive strength, and modulus of elasticity are described in Table 3-2.

Containment concrete temperatures generally do not exceed 120°F to 150°F during normal operation. This range is within the industry recognized limit of 150°F [Ref. 11] where temperature effects are not a significant degradation factor.

Local areas around hot piping penetrations may be subject to higher temperatures depending on the effectiveness of the design provisions used, such as cooling coils and/or insulation. Local temperatures around the penetrations are permitted up to 200°F by ACI 349 [Ref. 14].

TABLE 3-2
TEMPERATURE EFFECTS ON CONCRETE PROPERTIES

Property	Effect	Ref. No.
Creep	Creep depends on the stress-strength ratio and the concrete temperature over time. For specimens tested between 35-205°F, the relation of creep to stress-strength ratio in the range of 10-70% was linear. Concrete creep is typically accounted for in the general containment design at the appropriate temperature through use of an effective modulus of elasticity for analysis and in the prestress loss predictions used to determine the required initial prestress force.	1
Compressive Strength	• Compressive strength tests conducted on specimens heated for short duration to temperatures of 200 to 1600°F, in stressed and unstressed conditions, and for different aggregates indicate loss of strength as the temperature is increased. At 400°C (725°F), compressive strength can be reduced to 50% of the strength of unheated specimens.	2
	• The effects of moisture content on compressive strength for various concrete mixes exposed to temperatures up to 500°F is provided. Containment concrete is usually considered as a sealed condition. Sealed concrete has greater reduction in compressive strength than unsealed concrete when subject to elevated temperature. Depending on the aggregate, the compressive strength can vary between an increase of 10% and a decrease of 15% at 392°F (200°C).	3,4
Modulus of Elasticity	The effect of temperature on the modulus of elasticity for sealed and unsealed specimens is provided. The modulus decreases as the temperature increases; the decrease is greater for sealed specimens. Between 20 and 200°C the decrease is approximately linear. At 150°C (302°F) the modulus is about 45% of that for the sealed specimens at 20°C (68°F). Reductions in E are generally greater than those for compressive strength.	3,4

References:

References 2 to 3 are from ACI Publication SP 25 Temperature and Concrete.

(1) Nasser, K. W., "Creep of Concrete at Low Stress-Strength Ratios and Elevated Temperatures," Paper SP-25-5.

(2) Abrams, M. S., "Compressive Strength of Concrete at Temperatures to 1600°F," Paper 25-2.

(3) Lankard, D. R., D. L. Birkimer, F. F. Fondriest, and M. J. Snyder, "Effects of Moisture Content on the Structural Properties of Portland Cement Concrete Exposed to Temperatures up to 500°F," Paper SP-25-3.

(4) Kong, F. K., et al., *Handbook of Structural Concrete*, Part V Structures, McGraw-Hill.

Aging Effect Evaluation

Intended functions degraded by the aging effects of concrete thermal aging embrittlement (if sufficient detrimental effects are indicated) are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Current code requirements for permissible concrete temperatures in concrete structures that are important to the safety of the nuclear plant are included in ASME B&PV Code Section III, Division 2, CC-3440 for the containment building, and in the ACI 349 Nuclear Safety Structures Code, Appendix A - Thermal Considerations, subparagraph A.4, Concrete Temperatures, for the other Seismic Category I structures.

The concrete temperature limitations in both of these codes are the same:

- Normal operation or any other long-term period:

  The temperatures shall not exceed 150°F except for local areas such as around penetrations where the temperatures are not to exceed 200°F.

- Accident conditions or other short-term period:

  Temperatures shall not exceed 350°F at the surface except that local areas may reach 650°F from steam or water jets in event of a pipe failure.

- Higher temperatures may be permitted if tests are provided to evaluate the reduction in concrete strength and the reduction is applied to the design allowables. Evidence must be provided that the increased temperatures do not cause deterioration of the concrete either with or without load.

Thermal aging can result in reduction of concrete compressive strength, tensile strength, and modulus of elasticity. Reductions in these concrete properties greater than 10 percent begin to occur at temperatures in the range of 180°F to 200°F.

Long-term exposure to temperatures greater than 300°F can cause surface scaling and cracking. However, for the temperature ranges normally found in PWR containments, degradation due to elevated temperature is not expected.

Therefore, high-temperature concrete embrittlement is considered to be a nonsignificant degradation mechanism for the containment concrete components listed in Table 1-2.

Aging Effect Management

Due to lack of detrimental aging effect caused by thermal aging embrittlement in the concrete, there is no need for the identification of aging management options.

3.2.7 Concrete Aggressive Chemical Attack

Mechanism Description

Aggressive chemical attack can occur by exposure of the concrete to acidic, chloride-bearing, or sulfate-bearing solutions. Chemical attack alters the concrete through chemical reaction with either the cement paste or the coarse aggregate. The attack typically starts at the surface. With the presence of cracks and over a prolonged period of exposure, larger parts of the component section can be affected.

Concrete is highly alkaline (pH > 12.5) and can be degraded by acidic solutions having pH less than 5.5, such as occur in certain soils [Ref. 15].

Aging effects include cracking, spalling, loss of strength, increased porosity, permeability, disintegration, and eventual loss of load carrying capacity.

Solutions containing sulfates of potassium, sodium, and magnesium in sufficient concentration can produce significant expansive stresses within the concrete, leading to cracking, spalling, and strength loss. Sulfate attack can be severe when the concrete is saturated and is more likely when alternating saturation and drying are encountered [Ref. 12]. The expansive reaction takes place between the tricalcium aluminate phase of Portland cement and the sulfate ions to produce calcium sulfo-aluminate hydrates [Ref. 16].

Acid solution attack can cause increased porosity, permeability, and reduced concrete strength.

Sulfate attack can produce expansive stresses within the concrete that can result in cracking, spalling, and strength loss.

If concrete is exposed to frequent cycles of wetting and drying by a sulfate containing water, the sulfates will concentrate at the free surfaces that are then subject to disintegration. The disintegration may be preceded by the appearance of pattern cracking.

Aggressive chemical attack, if significant, could eventually impact the load carrying function of the containment.

Continued or frequent cyclic exposure to the following aggressive chemical environment is necessary for aggressive chemicals to cause significant concrete degradation: acidic solutions with pH < 5.5, chloride solutions with > 500 ppm [Ref. 17], and sulfate solutions with >1500 ppm [Ref. 18].

Aging Effect Evaluation

The aging effects of aggressive chemical attack (only if unmitigated and extensive) could potentially result in the degradation of the pressure-retaining capability of the containment and the structural capacity, which support intended functions. These include the protection of the

environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

For above-grade PWR containment concrete components such as the dome and wall that are not in contact with groundwater, the above conditions are not met, and aggressive chemical attack is a nonsignificant degradation mechanism.

For below-grade containment concrete components such as the foundation, the potential for degradation depends on the extent of the aggressive chemical environment in the groundwater, the level of the groundwater with respect to the structures, and the presence of a waterproofing membrane.

Concrete for containment components typically has a high cement content, a low water-cement ratio, and has been properly cured, resulting in low permeability and resistance to aggressive chemical solutions. Therefore, if below-grade concrete components are exposed only infrequently to groundwater solutions that exceed the above-chemical limits, then the degradation is not significant.

If below-grade concrete structures are exposed for extended periods to groundwater solutions that exceed the limits above then degradation is potentially significant.

Aging Effect Management

Aggressive chemical attack is a combined event-driven and natural aging degradation mechanism. The event-driven aging effects are managed through timely inspection and repair, i.e., for areas accessible for inspection. See Section 4.1 for details. The only cases where concrete aggressive chemical attack can potentially cause detrimental aging effects are those where concrete containment structures are below grade and are exposed for extended periods to groundwater solutions that exceed the defined chemical limits, i.e., subject to natural aging mechanisms. The aging effects, cracking, spalling, scaling, etc., caused by this potentially significant degradation mechanism can be managed through an enhanced inservice inspection (ISI) program. This ISI program will identify the conditions conducive to aggressive chemical attack, both internal and external to containment, mitigation actions, and subsequent inspection and repair. The degree of management is based on the indications for potential damage. The ISI program is described in detail in Subsection 4.1.7, aging management options AMP-5.3 and AMP-5.4.

3.2.8 Concrete Bond Strength Reduction - Direct Current

Mechanism Description

Test results and studies [Ref. 19], have indicated that impressed direct current can reduce the bond-strength of steel in concrete, an aging effect. The concrete at the interface can begin to soften at high current flow (up to 1000 mA/ft^2 for a 5-year period). Cathodic protection systems for reinforcing steel are typically designed to operate at about 2 mA/ft^2 (ft^2 of steel surface), much less than the levels of current used in the tests.

When the direct current is sufficiently high and is impressed over a sufficient period of time, the bond-strength may be reduced although the effect is variable.

Aging Effect Evaluation

Intended functions degraded by the aging effects of concrete bond-strength reduction (if sufficient detrimental effects are indicated) are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Cathodic protection currents will not impair the bond-strength and the capability of the concrete/reinforcing steel system to perform its intended function, since the cathodic protection system is generally designed to operate at approximately 2 mA/ft2 of steel surface. This level is well below the threshold of (limit) 1000 mA/ft^2. Therefore, cathodic protection current-related bond-strength reduction is considered to be a nonsignificant degradation mechanism for plant license renewal.

Aging Effect Management

Due to lack of detrimental aging effect caused by the reduction of bond-strength between the rebar and concrete due to direct current, there is no need for the identification of aging management options.

3.2.9 Fatigue at Penetration Anchors

Mechanism Description

The penetration sleeves in PWR concrete containments are typically anchored to the concrete shell through embedments welded to the exterior surface of the sleeve as shown in Figure 2-6. Anchorage design details vary from plant to plant. The anchorage to the concrete shell typically serves as a fixed anchor point for the piping system analyses. For some designs, thermal cyclic loads from the hot piping system cause shear forces in the concrete at the penetration sleeve anchors.

The concrete surrounding the penetration sleeve anchorage is exposed to load and temperature fluctuations and can potentially be damaged by fatigue. Fatigue damage begins as microcracking in the cement paste matrix near potential stress raisers within the concrete such as coarse aggregates or reinforcing steel. Continued cycles of load and temperature can cause the microcracks to grow and coalesce, potentially exposing the reinforcing steel to corrosion or resulting in the loss of the concrete capability to carry the loads. Therefore, the aging effect is fatigue-induced cracking.

Aging Effect Evaluation

Intended functions degraded by the aging effects of fatigue at penetration anchors (if sufficient detrimental effects are indicated) are the protection of the environment from unacceptable

release of radiation and the protection of containment interior structures and systems from external loadings.

If the number of load cycles is less than 106 and the maximum shear on the concrete alone caused by the cyclic range is less than one-half of the maximum allowable shear, then shear fatigue is not a concern [Ref. 16]. For a lower number of load cycles, the increase in the permissible stress range can be determined using Goodman diagrams in accordance with ACI 215R [Ref. 20].

Shear fatigue on the concrete alone is also not a concern if all of the shear reaction can be accounted for by reinforcing steel. The reinforcing steel has good fatigue-strength properties for 10^5 cycles of below yield load [Ref. 20].

Aging Effect Management

Thermal cycling of attached hot piping systems causes a potentially significant stress having a fatigue effect on hot penetrations without bellows for PWR concrete containments, and at penetration bellows assemblies for PWR free-standing steel containments. Cracks caused by fatigue can be managed by the ASME Code Section XI, Subsection IWE inspection and leak rate testing programs. These ISE programs will detect the presence of cracks and provide criteria for the acceptance of repairs and subsequent inspection. Aging management program AMP-5.5 describes the attributes of this inspection program (see Subsection 4.1.9).

3.2.10 Corrosion – Reinforcing Steel

Mechanism Description

Reinforcing corrosion is a principal cause of deterioration in concrete structures. The aging effects resulting from corrosion of reinforcing steel include cracking of the concrete, increased permeability due to the presence of concrete cracks, loss of bond-strength between steel and concrete, reduction in reinforcement cross-section, loss of protective cover and chemistry due to concrete cracking, and eventual loss of strength. The mechanisms that cause reinforcing corrosion are described in Reference 21. Corrosion is an electrochemical process that results in the formation of ferric oxide (rust) from the metallic iron. The corrosion products have a significantly greater volume than the original metal, resulting in tensile stresses and cracking in the surrounding concrete.

To sufficiently preclude reinforcing corrosion, concrete should:

- Be good quality
- Be dense
- Have low permeability
- Adequately cover the reinforcing

Concrete with low permeability contains less water and therefore exhibits lower electrical conductivity.

The high alkaline environment (pH > 12.5) provided by the concrete around the reinforcing bar causes a passive iron oxide film to form on the reinforcing surface that protects the reinforcing from corrosion. The protective film can, however, be destroyed following leaching of the alkaline constituents of the concrete by water or by carbonation. Carbonation involves carbon dioxide from the air reacting with calcium hydroxide in the concrete to form calcium carbonate. The loss of the calcium hydroxide lowers the concrete pH. Chloride ions, either present in the concrete or from external sources, can also destroy the passive iron oxide film protection. Inferior quality concrete that is porous and has significant microcracking will permit entry of the degradation factors resulting in corrosion of the reinforcing.

Reinforcing steel corrosion products have a significantly higher volume than the original metal. This expansion causes cracking in the concrete that leads to rust staining, spalling, and more severe cracking. Further deterioration occurs as more of the reinforcing steel is exposed to the corrosive environment. Eventually loss of bond, as well as reduction in the rebar cross-section occurs. This condition impacts the load carrying function of the containment.

Above-grade components may be exposed to aggressive environments on an intermittent basis. Below-grade surfaces in a zone of fluctuating water level can be exposed to aggressive environments on a relatively continuous basis and are a concern regarding potential reinforcing corrosion.

Aging Effect Evaluation

The aging effects of corrosion of reinforcing steel (only if unmitigated and extensive) could potentially result in the degradation of the pressure-retaining capability of the containment and the structural capacity, which support intended functions. These include the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Corrosion is insignificant for reinforced concrete components constructed in accordance with design codes that limit excessive cracking, thereby protecting the reinforcement against infiltration of aggressive environments. Corrosion is insignificant even for reinforced components where some cracking is found and for above-grade structures that are not exposed to aggressive environments on a relatively continuous basis if the following conditions are met:

- The component is not exposed to an aggressive environment (pH < 11.5 or > 500 ppm chlorides and sulfates).
- High-quality concrete having low permeability due to relatively high strength (4000 psi), low water-cement ratio (0.35 to 0.45), and air entrainment (3 to 6 percent). Aggregates are well graded enhancing low permeability.
- Concrete cover over reinforcing in accordance with the accepted design codes ACI 318 or ASME Section III, Division 2.

Aging Effect Management

This mechanism does not require an aging management program if the conditions described above are met. The only portion of the PWR containment where corrosion of reinforcing steel can potentially cause detrimental aging effects are those containment components that are below-grade in a zone of fluctuating water level exposed to aggressive environments on a relatively continuous basis. The aging effects caused by corrosion of the reinforcement steel—cracking, increased permeability, etc.—can be managed through the enhanced ISI program for the management of aggressive chemical attack. This ISI program will identify the conditions conducive to aggressive chemical attack, both internal and external to containment, mitigation actions, and subsequent inspection and repair. The degree of management is based on the indications for potential damage. The ISI program is described in detail in Subsection 4.1.8, aging management options AMP-5.3 and AMP-5.4.

3.2.11 Elevated Temperature - Reinforcing Steel

Mechanism Description

The yield-strength and modulus of elasticity of mild steel reinforcement will be reduced by about 15 percent at 700°F with further reductions as the temperature is increased. At temperatures up to 600°F, bond of the reinforcing to the concrete is not significantly affected [Ref. 22]. This mechanism has no significant aging effect.

Aging Effect Evaluation

Elevated temperature is a nonsignificant degradation mechanism for PWR containment reinforcing steel because the normal operating temperature range within PWR containments is 120°F to 150°F, well below the temperature range where degradation becomes significant. Containment temperatures can range to 300°F under accident conditions. Accident temperatures are short-term and well below the threshold for degradation of the material properties.

Aging Effect Management

Due to lack of detrimental aging effect caused by elevated temperature to the reinforcing steel in PWR containments, there is no need for the identification of aging management options.

3.2.12 Irradiation (Embrittlement) - Reinforcing Steel

Mechanism Description

Neutron irradiation above the threshold fluence level of 10^{19} neutrons/cm^2 can produce changes in the mechanical properties of carbon steels, or aging effects, including an increase in the yield-strength, decrease in the ultimate tensile ductility, and increase in the ductile-to-brittle transition temperature. This phenomenon is usually referred to as radiation-induced embrittlement and is not visibly observable.

Aging Effect Evaluation

Intended functions degraded by the aging effects of irradiation of reinforcing steel (if sufficient detrimental effects are indicated) are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Irradiation is a nonsignificant degradation mechanism for PWR containment mild steel reinforcing because the fluence level anticipated during normal operation is well below the threshold, 1019 neutrons/cm2, for degradation of the mechanical properties. See also Subsection 3.2.4.

Aging Effect Management

Due to lack of detrimental aging effect caused by irradiation of the PWR containment reinforcing steel, there is no need for the identification of aging management options.

3.2.13 Fatigue – Reinforcing Steel

Mechanism Description

Fatigue damage can result in structural materials subject to cyclic loadings. For concrete components, fatigue effects under a significant number of stress repetitions can include microcracking of the concrete that can lead to further deterioration, including exposure and corrosion of reinforcing steel. Microcracking can also result in loss of bond between the reinforcing and concrete.

Degradation of steel components due to fatigue is not detected until cracks are formed and propagate within the material. The aging effect resulting from this degradation mechanism is cracking of the concrete and reinforcement steel.

Aging Effect Evaluation

Intended functions degraded by the aging effects of fatigue of reinforcing steel (if sufficient detrimental effects are indicated) are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Containment concrete, reinforcing, and free-standing steel containment shells have good fatigue-strength properties for the low stress-level cycles (below yield). Containments designed in accordance with ASME Code [Ref. 11] and ACI 215R -74 [Ref. 23] have high fatigue-strength for 105 cycles of below-yield stress.

Fatigue is a nonsignificant degradation mechanism for PWR concrete/reinforcing systems and for free-standing steel containment components. High-cycle fatigue will not result in significant degradation due to the use of design codes that limit stresses for the PWR containment,

including those for the period of extended operation, to values much lower than the fatigue strength. This is supported by Reference 24.

Aging Effect Management

Due to lack of detrimental aging effect caused by fatigue of PWR containment reinforcing steel, there is no need for the identification of aging management options.

3.2.14 Corrosion - Liner

Mechanism Description

The primary cause of degradation to the liner is corrosion. The aging effect is loss of metal thickness, which could result in loss of pressure retention capabilities. The areas of primary concern for corrosion degradation include embedded areas, the inaccessible side, areas where the coating system is deteriorated, and the area at and below the sealant detail at the concrete metal interface. For corrosion to occur at the outside surface of the liner below grade, interconnected cracks in the concrete must be present that provide a pathway for aggressive groundwater to reach the liner. In addition, the membrane waterproofing, where provided, must be damaged or deteriorated to permit passage of groundwater.

Liners are typically coated with primer or a primer-finish coat system to provide corrosion protection. Liner corrosion can result from galvanic action of dissimilar metals, stress corrosion cracking (SCC), or electrochemical corrosion from exposure to aggressive aqueous solutions (groundwater). Local corrosion attack is the primary concern.

The three factors that must be present for SCC to occur are tensile stresses at or near yield, a corrosive environment, and a susceptible material. SCC can occur in susceptible materials such as austenitic stainless steel, including SA-240 Type 304 or 308 in a corrosive environment such as in the presence of chlorides and acidic solutions. The tensile stresses can be either externally applied or residual. Heat-affected zones at welds and creviced geometries are particularly prone to SCC. Typically, there is no visual evidence of SCC.

Galvanic corrosion can result when dissimilar metals are in contact. An electrical potential exists between the two metals that results in electrons flowing from one of the metals (the anode) to the other metal (the cathode), resulting in a loss in metal thickness at the anode but no change to the cathode.

Corrosion can occur at local areas where aggressive solutions are permitted to contact the liner at floor levels. The primary location of concern is at the seal between the liner and the concrete floor slab covering the bottom floor liner plate from corrosion.

Corrosion of the liner on the side adjacent to the concrete is precluded by the concrete, which provides protection against the ingress of moisture. The concrete also provides an alkaline environment that protects the liner.

Below-grade surfaces of the liner that could be exposed to aggressive groundwater solutions through cracks in the concrete are potential areas for liner corrosion.

Aging Effect Evaluation

The aging effects of corrosion of the liner steel, only if unmitigated and extensive, could potentially result in the degradation of the pressure-retaining capability of the containment and the structural capacity, which support intended functions. These include the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Corrosion of the inside surface of the liner will not result unless (a) the protective coating is removed or damaged or, (b) protective seals between the liner and adjacent floor are lost or damaged, for inaccessible regions. The rate of attack depends on the aggressive character of the environment. Liner corrosion causes a reduction in the plate thickness that could affect the leaktightness function of the liner. Current inspection, testing, and repair programs have been effective in protecting surfaces accessible for inspection, and inaccessible surfaces through inspection of seals.

Corrosion of the outside surface of the liner will not result unless (a) the surface is below-grade, (b) the groundwater is aggressive, (c) the membrane protecting the concrete from the aggressive groundwater fails, and (d) the concrete fails to protect the outer surface of the liner. Managing aggressive chemical attack, where it is a potentially significant degradation mechanism, is necessary to control effects of this degradation mechanism. Aggressive chemical attack is a potentially significant degradation mechanism when aggressive aqueous solutions having chloride and/or sulfate concentrations > 500 ppm or 1500 ppm, respectively, and pH < 5.5 are present.

For above-grade liner surfaces, galvanic corrosion is nonsignificant if dissimilar metals are not used in the construction.

SCC in stainless steel is significant only in the presence of high tensile stress and a corrosive environment. Liner stresses are typically compressive; therefore, SCC is considered a nonsignificant degradation mechanism for the liner.

Aging Effect Management

Corrosion of the steel liner is a combined event-driven and natural aging degradation mechanism. The event-driven aging effects are managed through timely inspection and repair, i.e., for areas accessible for inspection. See Section 4.1 for details on current practice. Natural aging effects are managed as described below.

The effects of liner corrosion, below-grade and on the outside surface that would result from aggressive chemical attack, can be managed through programs for managing the effects of aggressive chemical attack, required where aggressive chemical attack is a potentially significant degradation mechanism. The aging effects—loss of material thickness and pressure

retention capability caused by corrosion—can be managed through the enhanced ISI program for the management of aggressive chemical attack. This ISI program will identify the conditions conducive to aggressive chemical attack, both internal and external to containment, mitigation actions, and subsequent inspection and repair. The degree of management is based on the indications for potential damage. The ISI program is described in detail in Subsection 4.1.9.

The aging effects of liner corrosion resulting from the use of dissimilar metals can be managed through an ISI program. This ISI program will identify the indications of, or conditions conducive to corrosion damage and provide criteria for repair and subsequent inspection. The ISI program is described in detail in Subsection 4.1.9, aging management option AMP-5.5. This program, where coatings and seals are subject to inspection for aging effects, is also applicable for inaccessible portions of the inside surface of the liner where aggressive chemical attack is a potentially significant degradation mechanism.

3.2.15 Coating Degradation

Mechanism Description

Coating systems are provided for the carbon steel liner and associated components to protect against corrosion. Degradation or damage to liner coating systems can, under aggressive environmental conditions, lead to local corrosion attack of the liner and associated components. Aging effects include loss of protective cover that could result in degradation of the liner or steel containment. The stressors that cause coating system degradation include temperature, condensation and immersion, radiation, base metal corrosion, and physical damage [Ref. 25]. Once moisture and oxygen penetrate the coating to the base metal, local corrosion can result and spread under the coating system, eventually lifting the coating and exposing larger areas to corrosion. The thermal effects on coatings include differential expansion between the coating and the base metal that can result in cracking of the coating. Physical damage can include gouges, cracking, or pinholes.

Coating systems for liner surfaces that will be embedded or covered by concrete typically consist of either no coating or a primer coat. The primer coat provides a measure of corrosion protection during construction. Following concrete placement, the alkaline environment provided by the concrete protects the carbon steel liner and anchors from corrosion. A primer-finish coat system is typically provided for exposed liner surfaces. The finish coat is designed to provide a surface that can be readily decontaminated as well as to protect the liner from corrosion.

Aging Effect Evaluation

The aging effects of coating degradation (only if unmitigated and extensive) could potentially result in the degradation of the pressure-retaining capability of the containment and the structural capacity, which support intended functions. These include the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Coating system degradation is nonsignificant for liner and associated component surfaces that are protected by concrete. Coating degradation is a potentially significant degradation mechanism for coating systems that are exposed to the internal containment environment; however, the event-driven and natural mechanism is effectively managed by existing visual inspection and repair programs, as discussed in Section 4.1.

Aging Effect Management

There is no need for an aging management option program to address the effects of coating degradation since ISE programs will detect the evidence and aging effects of coating degradation and provide criteria for the acceptance of repairs and subsequent inspections. The loss of coating does not directly result in an aging effect. Corrosion could result from the loss of the coating. This aging effect is managed by AMP-5.5.

3.2.16 Fatigue at Attachments and Discontinuities – Liner, Airlocks, and Hatches

Mechanism Description

Fatigue damage can result in structural materials subject to cyclic loading. Fatigue of steel components can result in cracking, an aging effect, and result in further deterioration, including exposure of concrete to aggressive chemical attack.

Aging Effect Evaluation

Intended functions degraded by the aging effects of fatigue at attachment and discontinuities (if sufficient detrimental effects are indicated) are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Attachments such as polar crane brackets, attachment point details for spray piping in the dome area, and other supports have been designed to permissible stresses, less than the yield in accordance with applicable design standards such as the ASME code [Ref. 11]. Where tensile forces are carried directly across the thickness of the liner, the permissible stress is typically reduced to one-half of the normal permissible stress.

The permissible stresses and design details required by the design codes provide good fatigue strength for a high number of cycles (generally 105 and higher) of loading below yield. Therefore, fatigue is a nonsignificant degradation factor for these structures since the actual cycles are low. This same conclusion is given in Reference 24.

Aging Effect Management

Due to lack of detrimental aging effect caused by fatigue at attachments and discontinuities, there is no need for the identification of aging management options.

3.2.17 Corrosion of Metal Components and Concrete Degradation – Post-Tensioning Systems

Mechanism Description

Corrosion of post-tensioning/prestressing tendons is typically the result of localized attack, including pitting, stress corrosion, hydrogen embrittlement, or a combination of these. Aging effects include reduction in cross-sectional area, reduction in prestress force, breakage of wires or strands, and leakage of corrosion-inhibiting grease. Failure of prestressing tendons can also occur as a result of microbiologically induced corrosion (MIC). Protection of the tendons against corrosion is typically provided by filling the tendon ducts with organic corrosion inhibitors (grease).

Pitting occurs locally by an electrochemical process in the presence of halide ions (typically chlorides) and results in a reduction of the cross-section area.

Hydrogen embrittlement may occur when hydrogen atoms enter the metal lattice and significantly reduce the metal ductility, potentially resulting in brittle fracture. Exposure to hydrogen sulfide may precede the hydrogen embrittlement.

Corrosion can result in the breakage of the wires or strands reducing the prestress forces applied to the containment. In the event the condition is severe, the prestress force may be reduced to below the minimum forces required to account for the design loadings.

Organic corrosion inhibitors (grease) may leak from the duct or sheathing system of the post-tensioning system and degrade the strength of the surrounding concrete. The grease may contain chlorides, nitrates, or sulfides, or a low pH, along with a free water content, all providing an environment that could potentially attack the strength of the concrete in a manner similar to that for the aggressive chemical attack mechanism. Leakage would result from corrosion or flaws in the duct or sheathing system.

Aging Effect Evaluation

The aging effects of corrosion and concrete degradation could degrade the pressure-retaining capability of the containment and the structural capacity, which support the intended functions of protection of the environment from unacceptable release of radiation and protection of containment interior structures and systems from external loadings.

Corrosion of the prestressing system is a potentially significant means of containment degradation. Because of the important load carrying function of the prestress system and the potential susceptibility of the system to corrosion, the prestress system is inspected on a periodic basis to ensure it is functioning as designed and has not deteriorated.

Grease leakage is a potentially significant degradation mechanism, but potential effects are already monitored by CLB practice. Leakage will not result unless significant degradation of the sheathing or the duct containing the tendon occurs. Leakage could result from corrosion or

flaws in the duct or sheathing system, or from using too high of a pressure when injecting replacement grease, which could tear the sheathing joints. Flaws alone would not provide significant leakage to affect the concrete strength. Corrosion from external aggressive chemical attack, below-grade, would result in significant damage to the structure before the sheathing could corrode. Internal corrosion resulting from the grease chemistry and the presence of free water would not be likely since the chemistry and free-water content is monitored with programs the same or similar to that defined in Regulatory Guide 1.35 or ASME Section XI, Subsection IWL, as discussed in Section 4.1. In the event of a grease leak, the monitored reserve alkalinity and chemistry of the grease is not conducive to promote aggressive chemical attack.

Grease leakage is potentially significant only where:

- Grease and free water chemistry provide an environment conducive to corrosion of the sheathing

- Sufficient leakage has occurred

These are effectively managed by current testing and inspection programs and require continued management through plant life extension.

Aging Effect Management

The current maintenance programs following Section XI, Subsection IWL, of the ASME Code are sufficient to manage the aging effects of metal component corrosion and concrete degradation into the extended period of operation. These ISE and testing programs identify the conditions conducive to, or evidence of corrosion and concrete degradation, and provide criteria for the acceptance of mitigation actions, repairs, and subsequent inspections.

Aging effects resulting from externally induced corrosion of the tendon ducts or sheathing can also be managed through the Section XI, Subsection IWL, ISI program of the ASME Code. This ISI program will identify the conditions conducive to aggressive chemical attack, both internal and external to containment, mitigation actions, and subsequent inspection and repair. The degree of management is based on the indications for potential damage. The ISI program is described in Subsection 4.1.10, aging management options AMP-5.6.

3.2.18 Elevated Temperatures – Post-Tensioning Systems

Mechanism Description

Exposure of heat-treated and drawn prestressing wire to elevated temperatures can result in reduced tensile strength, an aging effect, due to permanent alterations of the internal crystalline transformations created during annealing.

Aging Effect Evaluation

Intended functions degraded by a significant reduction of tensile strength are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

The temperature range experienced by the prestressing system in PWR containments is well below the temperature where the reduction becomes significant. Studies and testing indicate that the reduction in tensile strength for prestressing steels is less than 10 percent for temperatures up to 400°F [Ref. 26]. As a result, elevated temperatures are not a significant degradation mechanism for license renewal.

Aging Effect Management

Due to the lack of a detrimental aging effect caused by elevated temperatures on the prestressing systems of PWR containments, there is no need for the identification of aging management options.

3.2.19 Irradiation – Post-Tensioning Systems

Mechanism Description

The mechanism description is similar to that for irradiation of reinforcing steel, Subsection 3.2.12. Aging effects include an increase in the yield-strength, decrease in the ultimate ductile capacity, and increase in the ductile-to-brittle transition temperature.

Aging Effect Evaluation

The aging effects of irradiation of the post-tensioning systems would degrade the structural capacity and pressure-retaining capability of the containment, which support the intended functions of protection of the environment from unacceptable release of radiation and protection of containment interior structures and systems from external loadings.

Degradation of the mechanical properties of prestressing steels by exposure to high levels of neutron irradiation is not a significant degradation factor for the prestress systems in prestressed concrete PWR containments. Studies have shown that exposure of prestressing steels to neutron fluence level of 4x1016 neutrons/cm2 has no affect on the relaxation behavior of the steel [Ref. 27].

Radiation exposure levels for prestressing tendons are below the threshold limits; therefore, irradiation is not a significant degradation factor (see Subsection 3.2.4).

Aging Effect Management

Due to lack of detrimental aging effect caused by irradiation on the prestressing systems of PWR containments, there is no need for the identification of aging management options.

3.2.20 Prestress Force Losses

Mechanism Description

Loss of prestress force is an aging effect that can be attributed to several mechanisms, some of which are age-related and others that are not. The contributors to the loss of tendon force as measured at the original lockoff include: friction; end anchorage deflection (slip); elastic shortening; wire stress relaxation; and concrete creep/shrinkage. Wire stress relaxation and concrete creep/shrinkage are time-dependent losses and, therefore age-related.

Wire stress relaxation is dependent on the relaxation characteristic of the particular steel, the initial stress level as a percent of the ultimate strength, the exposure temperature, and time. Creep and shrinkage of concrete represent volume changes in the concrete that occur over time and cause a reduction in tendon prestress force. Wire stress relaxation losses increase significantly at elevated temperatures. Exposure of prestressing wire to a temperature of 140°F for 50 years results in a 300 percent increase in loss of prestress force due to wire stress relaxation compared to the relaxation for wire at 68°F. The effect on the required prestress level depends on the relaxation characteristic of the prestressing steel used in the containment and the percent of wire relaxation assumed in the original design loss calculations.

Aging Effect Evaluation

Loss of prestress below the minimum design level degrades the pressure-retaining capability and the structural capacity of the containment, which support the intended functions of the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings. This aging effect requires management for the extended period of plant life.

Aging Effect Management

The loss of prestress force has been identified as a potential time-dependent degradation effect. This potential source of degradation is currently managed by plant surveillance and testing programs following Section XI, Subsection IWL, of the ASME Code. These ISE and testing programs monitor the loss of prestress and conditions conducive to or evidence of corrosion and concrete degradation and provide criteria for the acceptance of mitigation actions, repairs, and subsequent inspections. Note that calculation of the acceptable predicted prestress loss rate for the current license term is based on the assumption of a 40-year life. A revised predicted prestress loss rate must be calculated for the extended operation period and monitored for the plant life extension period, up to 20 years. Aging management program AMP-5.6 describes the attributes of this inspection program (see Section 4.1.10).

3.2.21 Stress Corrosion Cracking – Post-Tensioning System

Mechanism Description

Stress corrosion cracking can occur in tendon anchor heads when the material is susceptible, is under tensile stress, and is in a conducive environment. Anchorheads fabricated from high-strength, low-alloy steel bolting material may be subject to stress corrosion cracking in the presence of environmental conditions that include sulfates, ammonia, nitrates, chlorides, and fluorides.

Aging effects include cracking of steel components, loss of strength, and loss of prestress force.

Aging Effect Evaluation

The prestress system load-carrying function of the containment can be affected by stress corrosion cracking. ISI programs monitor the performance and deteriorization of the prestress system, and the management of the degradation mechanism must be extended into the period of plant life extension.

Intended functions degraded by the aging effects of stress corrosion cracking in the post-tensioning system are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Aging Effect Management

The potential aging effects due to stress corrosion cracking of the prestressing systems in PWR containments can be managed by inspection and testing programs as described in Subsection 4.1.10, aging management option AMP-5.6.

3.2.22 Corrosion – Steel Embedments

Mechanism Description

Corrosion is the primary degradation factor for miscellaneous steel embedments. The mechanisms and aging effects are the same as described for the liner (Subsection 3.2.14). Miscellaneous steel embedments consist of the following categories:

- Embedded parts of attachments for equipment and system supports located on the inside of the containment, e.g., polar crane supports, equipment hatch head removal system supports, and piping system supports such as the spray piping in the dome.

- Embedded parts of attachments located on the exterior of the containment, e.g., stack supports, crane supports on top of the ring girder, and lightning rod supports.

Aging Effect Evaluation

Intended functions degraded by the aging effects caused by the corrosion mechanism (if sufficient detrimental effects are indicated) are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings.

Corrosion of embedments located inside containment is considered to be nonsignificant because the concrete surrounding the embedment provides an alkaline environment that inhibits corrosion.

Similarly, corrosion of embedments located on the outside surface of concrete PWR containments is nonsignificant to the intended load-carrying function of the containment. The surrounding concrete protects the embedment from corrosion. Where the embedment is exposed to exterior environment conditions and the corrosion protection coating on the embedment is not maintained, corrosion may occur at the periphery of the embedment and eventually progress along the embedment below the surface of the concrete. Local rust staining and concrete spalling may occur at the embedment. Typical embedments located on the exterior of the containment are relatively small in comparison to the robust concrete containment structure. Corrosion and minor spalling at these local points are nonsignificant degradation factors and do not impact the pressure-containing and load-carrying functions of the containment.

Aging Effect Management

Due to lack of detrimental aging effect caused by corrosion of the steel embedments of the PWR containments, there is no need for the identification of aging management effects.

3.2.23 Material Compatibility - Electrical Penetrations

Mechanism Description

Material compatibility testing has been conducted to determine the effects of polymer outgassing on metal corrosion. The tests included both Type 304 stainless steel and carbon steel. The polymer sources included polysulfone (#1700 and #3500), Teflon FEP, and Kerite.

Specimens were placed in an oven and subjected to temperatures ranging from 300°F up to 600°F for durations up to 60 days. Test results indicated that the metals and alloys had insignificant amounts of corrosion when exposed to the outgasses for these short-term, high-temperature tests. The testing conditions are significantly more severe than will be experienced by the materials in service conditions over the life of the plant, including the license renewal period.

Loss of material is the relevant age-related degradation effect.

Aging Effect Evaluation

Intended functions degraded by the aging effects caused by the material compatibility mechanism (if sufficient detrimental effects are indicated) are the protection of the environment from unacceptable release of radiation and the loss of transfer of electricity to support the mitigation of the consequences of an accident.

The degradation resulting from the testing was insignificant. In the worst case, carbon steel experienced a 0.10 percent weight loss from the effects of Teflon FEP outgassing for the 60-day, 392°F average temperature corrosion test. The testing confirmed that polymer outgassing is not a significant degradation factor for the metallic electrical penetration components.

Aging Effect Management

Due to lack of detrimental aging effect due to material outgassing on the electrical penetrations of the PWR containments, there is no need for the identification of aging management effects.

3.2.24 Transgranular Stress Corrosion Cracking – Electrical Penetration Bellows

Mechanism Description

For some plants, flexible metallic bellows assemblies are incorporated into the penetration assemblies, including the electrical penetration assemblies, and are part of the containment pressure boundary. According to NUREG/CP-0120 [Ref. 28], flexible metallic bellows have demonstrated a history of common mode failures that could challenge the intended leaktightness function of the containment. The frequency of cracks resulting from transgranular stress corrosion cracking (TGSCC) significantly increases between 10-15 years after installation.

The following conditions were found to contribute to the formation of the cracks:

- Standing water
- Chemical residue from containment cleaning operations
- Original weld coatings and smoke particulates

Aging effects include the cracking of the bellows component and loss of pressure retention.

Aging Effect Evaluation

Intended functions degraded by the aging effects caused by the TGSCC mechanism (if sufficient detrimental effects are indicated) are the protection of the environment from unacceptable release of radiation and the loss of transfer of electricity to support the mitigation of the consequences of an accident.

The magnitude of the leakage from bellows assemblies varies. In a number of cases, bellows failures have been observed through visual inspections on penetrations that successfully passed Appendix J, Type B local leak-rate testing. Crack initiation and growth are the relevant age-related degradation effects.

Aging Effect Management

TGSCC of the metallic bellows assemblies, which are part of penetrations, can result in cracks resulting in the loss of leaktightness. The potential aging effects can be managed by the existing plant inspection and testing program associated with penetrations following Section XI, Subsection IWE, of the ASME Code. Penetration bellows assemblies require maintenance of their pressure-retaining function. Leaktightness pressure tests (Type A or B) are required. Visual inspection during leak rate testing is required by 10 CFR 50, Appendix J to ensure leaktight integrity. Repairs or replacement of the bellows are made per plant procedures. These ISE and testing programs detect the cracking resulting from TGSCC and provide criteria for the acceptance of repairs and subsequent inspections. Aging management program AMP-5.5 describes the attributes of this inspection program (see Subsection 4.1.9).

3.2.25 Fatigue - Mechanical Penetration Bellows

Mechanism Description

Fatigue is a potentially significant degradation mechanism for bellows that are part of the containment system boundary. Bellows assemblies are typically included as part of the containment boundary in steel containments. For older vintage concrete containments, bellows assemblies may be used for the outer containment barrier for hot penetrations in the double-barrier penetration design such as shown in Figure 2-7. Fatigue degradation to bellows caused by cycles of thermal movements between the process piping and the containment is not significant unless there is damage such as scratches or dents in the bellows. These local defects or damage can result in stress concentrations that reduce the fatigue life.

Aging effects include fatigue-induced cracking and loss of pressure retention.

Aging Effect Evaluation

The intended containment function that may be affected by bellows fatigue is the protection of the environment from unacceptable release of radiation, as the result of cracking of the bellows and the subsequent loss of pressure-retaining capability.

ISI and testing of bellows, which is current practice, has been used to effectively detect local damage. The damaged bellows can be replaced or repaired. These measures need to be extended for the period of extended plant life.

Aging Effect Management

The effects of fatigue at mechanical penetration bellows are managed following ASME Code Section XI, Subsection IWE, inspection and testing programs into the extended period of operation. These ISE and testing programs detect the cracking resulting from fatigue, and provide criteria for the acceptance of repairs, replacement, and subsequent inspections. Aging management program AMP-5.5 describes the attributes of this inspection program (see Subsection 4.1.9).

3.2.26 Fatigue – Mechanical (Piping) Penetrations

Mechanism Description

Fatigue is a progressive failure of a structural part under repeated, cyclic, or fluctuating loads. Almost all structural materials, ferrous or nonferrous, are subject to fatigue. When a structural part repeatedly experiences fluctuating stresses, damage at microscopic levels may be initiated and accumulated in the material, which eventually leads to cracking.

The degree of damage is proportional to the applied stress and number of stress repetitions. However, fatigue will not occur below the endurance limit, the threshold stress level. The endurance limit is a material property that depends on the chemistry, method of manufacturing, heat treatment, etc. The endurance limit of a material is determined from a series of tests on the applied stress (S) versus the number of cycles (N) to failure.

Aging effects include fatigue-induced cracking and loss of pressure retention capability.

Aging Effect Evaluation

Intended functions degraded by the aging effects caused by fatigue are the protection of the environment from unacceptable release of radiation and the loss of transfer of fluids to support the mitigation of the consequences of an accident.

Single-Barrier Penetrations

The flued head forging functions as part of the process pipe itself, as well as the support for the process pipe. In addition, the flued head functions as part of the containment system boundary. For Class 2 piping systems, the flued head may be designated Class 2 and analyzed as part of the piping system in accordance with Section NC rules. In this case, an allowable stress range S_A is determined for expansion stresses due to temperature cycles. This stress range limit incorporates a stress range reduction factor based on the number of anticipated equivalent full temperature cycles. The factor is 1 for 7000 or less full temperature cycles. Full temperature cycles for most hot piping such as the main steam and feedwater systems correspond to startup and shutdown. Startup-shutdown cycles are considerably less than 7000 over the life of the plant including the license renewal term. Therefore, existing analyses should be applicable for the license renewal term.

The part of the penetration between the process pipe and sleeve extension may be designed as a Class 2 pipe support in accordance with Subsection NF. Subsection NF rules for Class 2 plate and shell pipe supports do not require analysis for cyclic loads. The maximum stresses in the support (secondary stresses) are limited to 3Sm for normal and upset conditions, and an evaluation for cyclic operation is not required.

For some plants, the flued head may be designated Class MC as part of the containment system. In this case, analysis for cyclic operation in accordance with ASME Code paragraph NE-3221.5 is required. Typically, an analysis and evaluation is done in accordance with NE-3221.5(d) (NB-3222.4(d) for older plants to justify that an analysis for cyclic operation including the calculation of the cumulative usage factor is not required. Where the NE-3221.5(d) analyses used an Sa stress (from the ASME Code fatigue curves) corresponding to 106 cycles, the analyses for the current 40-year plant license are acceptable to justify the 20-year plant life extension. However, if the analyses used specified numbers of pressure, temperature, and mechanical load cycles that were less than 106, then the specifications must be reviewed for the plant life extension to determine whether the number of cycles in the CLB is conservatively adequate to envelop the additional cycles for the 20-year plant life extension. Alternatively, the NE-3221.5(d) requirements can be re-evaluated for the increased total number of anticipated cycles.

Double-Barrier Penetrations

These penetrations are qualified for the plant license renewal term for cycles of pressure, temperature, and mechanical loads similar to the single-barrier penetration arrangement described above.

Low (ambient) and moderate temperature piping double-barrier penetrations may include plate or pipe cap closures at both the inside and outside ends of the penetration. This arrangement may restrain thermal growth of the process pipe within the penetration for the operating temperature range. Where this arrangement was used and the process pipe operating temperature exceeds 150EF, an analysis to demonstrate compliance with NE-3221.5(d), "Vessels Not Requiring Analysis for Cyclic Operation," could be done if not performed under the CLB. The analysis, to address the license renewal term, should assume either 106 cycles (conservative), or the number of cycles that includes consideration of the license renewal term.

In lieu of re-analyses for cyclic operation, as described above for certain concrete containment-rigid, double-barrier penetration arrangements, an inspection program can be used to manage potentially significant fatigue for these types of penetration assemblies following requirements similar to ASME Section XI.

Aging Effect Management

The effects of fatigue are potentially detrimental to the continued function of mechanical (piping) penetrations. The potential aging management effects can be managed by verifying that fatigue damage is not possible through the review of the penetration fatigue calculations for the extended period of operation (Section 3.2). Where necessary, appropriate corrective actions

could be taken to improve fatigue life following the existing plant procedures. In lieu of re-analysis, test and inspection programs can be performed following current plant procedures that reflect ASME Section XI, Subsection IWE requirements. Following these inspection and testing programs into the extended period of operation or use of analysis is sufficient to manage such aging effects. These ISE and testing programs detect the cracking resulting from fatigue and provide criteria for the acceptance of repairs and subsequent inspections. Aging management program AMP-5.5 describes the attributes of this inspection program (see Subsection 4.1.9).

3.2.27 Embrittlement and Permanent Set of Gaskets – Mechanical Penetrations

Mechanism Description

Gaskets fabricated from flexible materials are incorporated into certain mechanical penetration details such as at the joint between a blind flange and the flanged end of a penetration sleeve. The gaskets are subject to aging degradation such as embrittlement and permanent set resulting from exposure to environmental conditions such as high temperatures and irradiation. The impact of these stressors depends on the severity of the conditions, the gasket material, and the percent compression of the gasket. The percent compression that will result in permanent set depends on the material properties and the design details.

The aging effect is loss of seal or pressure-retention capability.

Aging Effect Evaluation

The intended function degraded by the aging effect caused by embrittlement and permanent set of gaskets is the protection of the environment from unacceptable release of radiation, resulting from the loss of the pressure-retaining capability of the gasket.

Embrittlement and permanent set of gaskets are managed by current inspection and maintenance programs where the gaskets are subject to periodic Type B local leak testing in accordance with 10 CFR 50, Appendix J. Local leak rate testing programs and associated visual inspections detect the aging effect, loss of seal or pressure retention, resulting from gasket degradation. Defective gaskets are replaced and the penetration seals retested.

Aging Effect Management

Inspection and maintenance programs following ASME Section XI, Subsection IWE, into the extended period of operation are sufficient to manage degradation. Aging management program AMP-5.5 describes the attributes of the inspection program (see Subsection 4.1.9).

3.2.28 Corrosion – Mechanical Penetration

Mechanism Description

The description, effects, and significance of corrosion on penetration sleeves and other carbon steel penetration components is basically the same as described in Subsection 3.2.14 for the containment liner. Aging effects include reduction in thickness and cracking of steel components.

Aging Effect Evaluation

Intended functions degraded by the aging effects caused by mechanical penetration corrosion are the protection of the environment from unacceptable release of radiation and the loss of transfer of fluids to support the mitigation of the consequences of an accident as a result of the loss of pressure-retaining capability and/or cracking.

With the exception of the residual heat removal (RHR) penetrations that penetrate the foundation mat and may be located below groundwater level, mechanical penetrations are typically located above-grade or within buildings. Corrosion due to galvanic action or aggressive aqueous solutions is not significant if dissimilar metals were not used in the design. SCC and galvanic corrosion are potential degradation factors for dissimilar welds between carbon and stainless steel for certain penetration designs. SCC may occur in the presence of tensile stresses and a corrosive environment. Management of SCC and corrosion for the RHR penetration, if located below the groundwater level, is required for plant life extension.

Aging Effect Management

Potential corrosion (SCC and galvanic) degradation effects exist for the mechanical penetrations. The aging effects caused by corrosion and SCC of the mechanical penetrations, cracking and loss of material, can be managed by an ISI program. This ISI program will identify the conditions conducive to corrosion, repair acceptance, and subsequent inspection. The ISI program is described in detail in Subsection 4.1.9, aging management option AMP-5.5.

3.2.29 Mechanical Wear – Fuel Transfer Tube Penetration

Mechanism Description

Mechanical wear can occur for the movable or active components associated with the fuel transfer tube. This includes the removable blind flange closure, located at the inside containment end of the transfer tube that functions as part of the containment system pressure boundary. Mechanical wear is characterized by the loss of metal or local plastic deformations at metal surfaces in contact with and moved frequently against each other over an extended period of time.

The aging effect of mechanical wear is loss of material.

Aging Effect Evaluation

Figure 3-1 shows the blind flange that is typically connected to a davit arm that is used to support and move the flange clear of the tube when the tube is used to move fuel in and out of the containment. The moving parts may experience some wear over time; however, mechanical wear on those surfaces that are part of the davit mechanism does not jeopardize the containment function of the blind flange, and therefore does not degrade an intended function.

Any wear that could interfere with the capability of the blind flange to be removed and replaced would be detected during normal refueling operations and is handled by normal inspection, maintenance, and repair procedures following ASME Code Section XI, Subsection IWE.

Aging Effect Management

Any detrimental aging effect due to mechanical wear in the fuel transfer tube and gates, is managed following ASME Code Section XI, Subsection IWE, as described in AMP-5.5. Subsection 4.1.9.

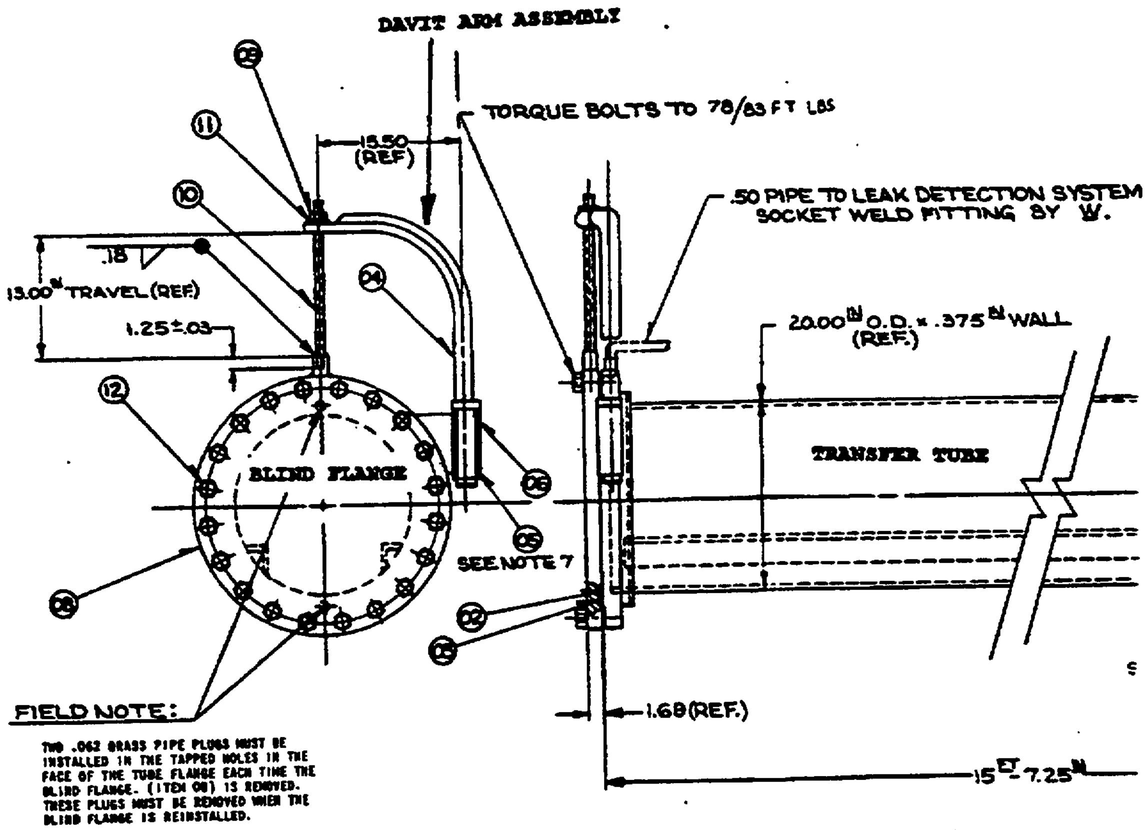


Figure 3-1 Fuel Transfer Tube Blind Flange Assembly

3.2.30 Gasket Degradation - Fuel Transfer Tube Penetration

Mechanism Description

The leaktight seal between the blind flange and the mating flange on the end of the penetration sleeve is typically achieved by a double O-ring flexible gasket similar to the gaskets for mechanical penetrations discussed in Subsection 3.2.27. Embrittlement or permanent set could result in loss of leaktightness.

Aging effects include loss of seal and pressure-retention capability.

Aging Effect Evaluation

The intended function degraded by the aging effects caused by embrittlement and permanent set of gaskets is the protection of the environment from unacceptable release of radiation, resulting from the loss of the pressure-retaining capability of the gasket.

Embrittlement and permanent set of these gaskets are potentially significant aging concerns; however, an existing program can be shown to be capable of managing the effects of embrittlement and permanent set, as discussed in Section 4.1. Elements of the existing program include Type B leak rate testing and replacement on a scheduled basis. Defective gaskets are replaced and the gasket seal is retested as required.

Aging Effect Management

Compliance with the plant inspection and testing programs into the extended period of operation, which are based on ASME Code Section XI, Subsection IWE, is sufficient to manage such effects. Local leak rate testing programs detect the aging effect, loss of seal or pressure retention, resulting from gasket degradation. Aging management program AMP-5.5 describes the attributes of the inspection program (Subsection 4.1.9).

3.2.31 Corrosion - Fuel Transfer Tube Penetration

Mechanism Description

A dissimilar or bimetallic weld is required to connect the stainless steel fuel transfer tube to the carbon steel containment penetration sleeve. The bimetallic weld is usually located at some point between the containment liner (or steel containment) and the transfer tube, not directly on the transfer tube itself. At this boundary, interface SCC is possible.

Aging effects include reduction in thickness and cracking of steel components.

The fuel transfer tube is water-filled and may be subject to corrosion.

Aging Effect Evaluation

The intended function degraded by the aging effects, caused by corrosion of the fuel transfer tube penetration, is the protection of the environment from unacceptable release of radiation, resulting from cracking or corrosive loss of thickness.

Given certain conditions that include a corrosive environment, e.g., presence of chlorides, tensile stresses in the weld from loads or residual tensile stresses from the welding, the bimetallic material is susceptible to SCC.

The environment within the fuel transfer tube can be wet and dry, increasing the possibility of corrosion from oxidation as well as galvanic action.

This mechanism requires management through the period of extended plant life.

Aging Effect Management

Potential SCC exists in the area of dissimilar welds of the fuel transfer tube and gates penetration. Further potential degradation from corrosion is possible. The aging effects caused by corrosion and SCC of the fuel transfer tube penetration—cracking and loss of material—can be managed by an ISI program. This ISI program will identify the conditions conducive to corrosion, repair acceptance, and subsequent inspection. The ISI program is described in detail in Subsection 4.1.9, aging management option AMP-5.5.

3.2.32 Mechanical Wear - Airlocks and Hatches

Mechanism Description

Airlock doors and hatch covers are typically opened or removed and replaced by manual, mechanical means. There are no control systems associated with the equipment hatch. The airlocks are typically designed with an interlock system that ensures the containment pressure boundary is maintained when the airlock is used for access during reactor operation. The interlock permits only one of the two airlock doors to be open at one time. The interlock system is typically wired to and controlled at the plant main control room. Defeating the interlock system by control room operator action does not automatically open both airlock doors. The doors are opened manually using handwheel shafts and associated gearing. Therefore, malfunction of the control system to the interlock does not necessarily result in the loss of the containment pressure-retaining function. However, mechanical wear can occur for the movable or active components of the airlocks and hatches.

The aging effect of mechanical wear is loss of material.

Aging Effect Evaluation

The intended function degraded by the aging effects caused by mechanical wear of the airlocks and hatches is the protection of the environment from unacceptable release of radiation, resulting from loss of material.

Mechanical wear that could potentially impact these functions will be detected by the local Type B leak rate testing performed as required by 10 CFR 50, Appendix J. Component inspections performed during Type A rate testing or surveillances performed in accordance with ASME Code Section XI, Subsection IWE for steel containment components ensure the component load-carrying integrity is maintained. Therefore, any degradation effects due to mechanical wear that could impact the intended functions of the airlocks or hatch are managed by preventive maintenance activities and required testing and inspection programs and are nonsignificant for license renewal.

Aging Effect Management

Current maintenance and inspection programs are sufficient to manage mechanical wear following ASME Code Section XI, Subsection IWE. Aging management program AMP-5.5 describes the attributes of the inspection program (see Subsection 4.1.9).

3.2.33 Fatigue - Airlock and Hatches

Mechanism Description

See Subsection 3.2.26.

Aging Effect Evaluation

Intended functions degraded by the aging effects caused by fatigue are the protection of the environment from unacceptable release of radiation and the loss of transfer of fluids to support the mitigation of the consequences of an accident.

The controlling load combinations for the design of these components include the containment design pressure load and the seismic design loads. Design pressure and temperature loadings produce relatively few loading cycles and therefore are not a significant fatigue consideration. Containment pressure and temperature ranges associated with the operating conditions of the plant are relatively minor in magnitude. Stresses due to these ranges are less than yield.

Fatigue is a nonsignificant aging degradation factor for the airlocks and equipment hatch. The number of cycles, including consideration of the license renewal period, is well below the cyclic fatigue capability of the airlock and hatch.

For typical Class MC airlock and hatch designs, the stress report demonstrated compliance with the requirements of ASME Code Section III, paragraph NE-3221.5(d). Where the requirements of this paragraph are met, stress cycles are not significant, and fatigue due to cyclic operation is

not a consideration. In this case, the cumulative usage factor is not calculated. The requirements of NE-3221.5(d) include:

- Atmospheric-to-operating pressure cycles
- Normal operation pressure fluctuation
- Temperature difference - startup and shutdown
- Temperature difference - normal operation
- Temperature difference - dissimilar metals
- Mechanical loads

For airlocks and hatches designed to earlier versions of the ASME Code, the requirements of paragraph NB-3222.4(d) were typically applied to demonstrate that a full fatigue analysis was not required. Paragraph NB-3222.4(d) requirements were basically the same as for NE-3221.5(d) listed above.

Aging Effect Management

Due to lack of detrimental aging effect due to fatigue of the airlocks and hatches, there is no need for the identification of aging management effects.

3.2.34 Gasket Degradation - Airlock and Hatches

Mechanism Description

The leaktight seal at the hatch cover and the airlock doors is typically achieved using flexible seals that may become embrittled over time or undergo a permanent set depending on the design detail configuration, percent compression set of the gasket, and environmental condition.

Aging effects include loss of seal and pressure-retention capability.

Aging Effect Evaluation

The intended function degraded by the aging effect caused by embrittlement and permanent set of gaskets is the protection of the environment from unacceptable release of radiation, resulting from the loss of the pressure-retaining capability of the gasket.

Similar to the discussion in Subsection 3.2.27 for the mechanical penetrations, embrittlement and permanent set of gaskets are managed by current inspection and maintenance programs for the hatch and airlocks where the gaskets are subject to Type B local leak rate testing in accordance with 10 CFR 50, Appendix J. Defective gaskets are replaced and the gasket seal is retested as required.

Aging Effect Management

Inspection and maintenance programs following ASME Code Section XI, Subsection IWE into the extended period of operation are sufficient to manage embrittlement and permanent set. Local leak rate testing programs detect the aging effect, loss of seal or pressure retention, resulting from gasket degradation. Aging management program AMP-5.5 describes the attributes of the inspection program (see Subsection 4.1.9).

3.2.35 Loss of Pressure Retention – Penetrations of Airlock Bulkheads

Mechanism Description

Each airlock bulkhead pressure boundary accommodates several types of penetrations including the handwheel shaft penetrations, penetrations for equalizing valve(s), electrical penetrations, and penetrations for pressurizing and leak-testing systems.

An equalizing valve penetration typically consists of a steel sleeve welded into the bulkhead with a pressure equalizing valve mounted on the sleeve. This arrangement is provided to enable the pressure within the airlock to equalize with the pressure in the destination volume outside of the airlock.

Loss of pressure-retention capability is an aging effect resulting from corrosion, mechanical wear, gasket embrittlement, or fatigue.

Aging Effect Evaluation

The intended function degraded by the aging effect of loss of pressure retention is the protection of the environment from unacceptable release of radiation, resulting from cracking or loss of material.

The leaktight function is verified by 10 CFR 50, Appendix J, Type B testing of the airlock. Therefore, aging of the equalizing valve penetrations is managed by current inspection and maintenance programs where the pressure boundary leaktight function of the penetrations is monitored.

Similarly, aging degradation that could adversely impact the pressure-retaining function of the electrical penetrations, the handwheel shaft seals, and the penetrations for pressurizing and leak testing systems is managed by periodic local Type B leak rate testing in accordance with 10 CFR 50, Appendix J or similar requirements.

Aging Effect Management

The potential aging effects are managed by the plant existing inspection and testing programs following 10 CFR 50, Appendix J, per ASME Code Section XI, Subsection IWE. Local leak rate testing programs detect the aging effect, loss of seal or pressure retention, resulting from

gasket degradation. Aging management program AMP-5.5 describes the attributes of the inspection program (see Subsection 4.1.9).

3.2.36 Elevated Temperature – Airlock and Hatches

Mechanism Description

The normal operating temperatures within the PWR containments are 120°F to 150°F, significantly below the level (about 700°F) where the mechanical properties of the steel such as yield-strength and modulus of elasticity begin to exhibit reduction from the original design values.

No aging effects result.

Aging Effect Evaluation

Elevated temperatures are a nonsignificant degradation factor for the airlocks and equipment hatches because the temperatures will be lower than the threshold for degradation of the material properties.

Aging Effect Management

Due to lack of detrimental aging effect caused by elevated temperature to the airlock and hatches, there is no need for the identification of aging management options.

3.2.37 Foundation Degradation

Mechanism Description

The degradation factors and their significance discussed in Subsections 3.2.2 to 3.2.8 and 3.2.10 to 3.2.13 for the concrete containment and reinforcing material systems apply to the foundation mat.

A potentially significant degradation factor for foundations is settlement. The magnitude of settlement depends on the bearing pressure exerted by the building weight and the physical properties of the supporting foundation medium.

Settlement may occur as the containment is constructed and the weight of the structure is accumulated. Settlement may continue slowly for many years after construction, but most of the settlement occurs within the first 5 or 6 years of operation. Changes in subsurface conditions such as lowering the water table can initiate additional settlement.

Aging effects include cracking and potential system overstress. Major differential settlement, if present, can be seen as concrete cracking or apparent differences in surface elevation. Differential settlement between adjacent buildings can result in the loss of support clearance in

interconnecting piping systems and other systems, including added stress and potentially overstress.

Aging Effect Evaluation

Intended functions degraded by the aging effects of significant differential settlement are the protection of the environment from unacceptable release of radiation, the result of cracking, and the loss of transfer of fluids or electricity to support the mitigation of the consequences of an accident, as a result of the overstress and failure of systems interconnecting adjacent buildings.

Settlement for most PWR containments occurs elastically as the load is introduced during construction. Settlement is monitored during construction by taking survey measurements and comparing the results with the design predictions. For most plants, settlement is minor following construction and is a nonsignificant aging mechanism.

Where the mat foundation is supported by soft soil or changes in the groundwater level following construction are significant, settlement may be a potentially significant age-related degradation mechanism. For plants with potentially significant long-term settlement, monitoring is performed during the plant life.

Because of possible changes in the site conditions over the life of the plant that could increase settlement, i.e., lowering of the groundwater table, programs to monitor changes in groundwater table and to detect potentially significant settlement are part of the CLB for susceptible plants. Compliance with the CLB is part of the license renewal commitment.

Aging Effect Management

Surveillance programs monitor differential settlement for plants susceptible to this degradation mechanism and provide acceptance criteria so that predicted design values are not exceeded. Aging management program AMP-5.7 describes the attributes of the inspection program (see Subsection 4.1.11).

3.2.38 Strain Aging - Free-Standing Steel Containment

Mechanism Description

Static strain aging can occur under certain conditions after the material has been cold-formed. At ambient temperatures, static strain aging can cause substantial property changes within 2 to 3 years after the material is cold-worked. Increased temperature will accelerate the strain age embrittlement. Dynamic strain aging occurs during plastic straining. Dynamic strain aging is not anticipated for the steel containment components because the strains associated with the service loads are below the elastic strain limit. Dynamic strain aging is not a significant degradation factor if the containment design philosophy limited stresses to the elastic range under the design loads. Static strain aging may result 2 to 3 years after the material is cold-worked. The most susceptible materials are low carbon rimmed or capped steels that are severely cold-worked during forming processes.

Decrease in ductility is the aging effect associated with strain aging.

Aging Effect Evaluation

The intended function degraded by the loss of ductility is the protection of the environment from unacceptable release of radiation and protection of containment interior structures and systems from external loadings.

Static strain aging is not a significant degradation factor if the containment plate was not severely cold-worked (less than 5 percent) during the forming process. This is typically the case for containment plate.

If severe cold-working of the steel was used in the forming process, but the plate has been normalized, stress relieved, or both following the forming with minimal subsequent cold-working, then static strain-aging is not a significant degradation factor for the steel containment plate.

Typical PWR free-standing steel containments consist of SA-516 Grade 70 or SA-212 Grade 70 plate steel, which is low-carbon steel (0.27 to 0.30 percent C). These materials are normalized, stress-relieved, or both, following the forming process. Therefore, static strain aging is a nonsignificant degradation factor for steel containments.

Aging Effect Management

Due to the lack of detrimental aging effects caused by strain aging of the free-standing steel containment, there is no need for the identification of aging management options.

3.2.39 Fatigue - Free-Standing Steel Containment

Mechanism Description

Fatigue is a progressive failure of a structural part under repeated, cyclic, or fluctuating loads. Almost all structural materials, ferrous or nonferrous, are subject to fatigue. When a structural part repeatedly experiences fluctuating stresses, damage at microscopic levels may be initiated and accumulated in the material, which eventually leads to cracking, an aging effect.

The degree of damage is proportional to the applied stress and number of stress repetitions. However, fatigue will not occur below the endurance limit, the threshold stress level.

Aging Effect Evaluation

Intended functions degraded by the aging effect caused by fatigue are the protection of the environment from unacceptable release of radiation and the loss of transfer of fluids to support the mitigation of the consequences of an accident resulting from the degradation of the pressure-retention capability and structural capacity due to cracking.

- **General Areas**

  Fatigue is a nonsignificant, age-related degradation mechanism for the general areas of the free-standing steel containment if the structural design complies with requirements of the ASME Code Section III, which has provisions to ensure a good fatigue life. The steel containment design typically has good fatigue strength for 10^5 cycles of below-yield load application when the design complies with ASME Code requirements. Low-cycle fatigue due to localized elevated temperatures is not significant for the general containment.

- **Local at Discontinuities**

  Fatigue is a nonsignificant degradation factor at the discontinuities in the steel containment geometry because design stresses are typically limited to levels that are below yield for service load conditions.

- **Local at Attachments**

  Fatigue is a nonsignificant degradation factor for attachments to the steel containment such as polar crane supports. Stresses in attachments are limited by design criteria to stresses that are less than yield for the design loads associated with the polar crane.

- **Penetration Bellows**

  Localized cyclic loading of penetration bellows assemblies results from the temperature changes in the hot piping system associated with the particular penetration. Potentially significant aging degradation of the bellows due to fatigue can result, especially at local flaws in the bellows due to fabrication defects and/or corrosion. The aging effects caused by bellows fatigue require management through the period of plant life extension. ASME Code Section XI, Subsection IWE, inspection and repair practices can be extended.

Aging Effect Management

The only potential aging effect due to fatigue is associated with penetration bellows. See Subsection 3.2.25. Aging management program AMP-5.5 describes the attributes of the inspection program (see Subsection 4.1.9).

3.2.40 Corrosion - Free-Standing Containment

Mechanism Description

Microbiologically influenced corrosion (MIC) occurs in locations where moisture is permitted to stand and stagnate in contact with the metal containment. A typical corrosion rate of carbon steel in an environment subjected to MIC is 10 to 30 mils per year [Ref. 25]. MIC can occur both in the presence of or absence of oxygen. Micro-organisms can cause corrosion by several

different mechanisms depending on the particular micro-organism. Certain micro-organisms produce acidic waste products. Other micro-organisms consume the protective coating or oxidize the metal directly. Any location where water is permitted to stand and stagnate in contact with the metal containment is a potential corrosion site. Containment low points such as slump drain lines and penetration sleeves are potential sites of MIC. Other potential sites include plant-unique locations where spills, leaks, condensate, or plugged drains permit collection and stagnation of water.

The local region where the steel containment is embedded in the supporting concrete is a potentially susceptible location for crevice corrosion. During startup and shutdown the differential thermal expansion of the steel containment versus the surrounding concrete can eventually disrupt the seal, creating a small gap that can provide access for moisture to uncoated portions of the steel containment.

Aggressive chemical attack is a potential cause of local corrosion of the steel containment interior surface. Reactor coolant water has a boric acid content of 0.2 to 0.4 percent [Ref. 25]. The containment may be exposed to that solution through leaks from piping or valves or spillage during refueling.

Galvanic corrosion can result when two dissimilar metals are in direct contact. The electrical potential between the two dissimilar metals causes a flow of electrons from the less resistant metal (carbon steel) to the more resistant metal (stainless steel), resulting in corrosion of the carbon steel. This corrosion can occur at penetrations where a bimetallic weld exists between stainless steel bellows assemblies and carbon steel sleeves. Other potential galvanic corrosion sites are plant-specific, depending on the materials in contact with the containment at the particular plant.

The aging effect associated with corrosion is loss of thickness.

Aging Effect Evaluation

Intended functions degraded by the aging effects caused by corrosion of the steel containment are the protection of the environment from unacceptable release of radiation and the protection of containment interior structures and systems from external loadings as a result of the degradation of the pressure-retaining capability and structural capacity.

For above-grade parts of the steel containment, the individual plant application for license renewal may demonstrate that the conditions necessary for corrosion caused by galvanic action, corrosion due to aggressive aqueous solutions, MIC, and SCC are not present for the free-standing steel containment. As a result, corrosion is a nonsignificant degradation factor for the free-standing steel containment above-grade.

For below-grade parts of the steel containment, corrosion is a potentially significant degradation factor on the exterior surface of the steel containment if the steel containment is exposed to aggressive groundwater solutions. Cracking of the concrete beneath the embedded portion of the steel containment shell and rupture of the continuous membrane waterproofing (if present)

is necessary for the groundwater to reach the surface of the steel containment. Exposure to aggressive groundwater can result in pitting corrosion that could compromise the leaktightness function of the steel containment [Ref. 25].

Aging Effect Management

Corrosion of the free-standing steel containment is a combined event-driven and natural aging-driven degradation mechanism. The event-driven aging effects are managed through timely inspection and repair, i.e., for areas accessible for inspection. Natural aging effects are managed as described below.

The aging effects of free-standing steel containment corrosion can be managed through an ISI program. This ISI program will identify the indications of or conditions conducive to corrosion damage and provide criteria for repair and subsequent inspection. The ISI program is described in detail in Subsection 4.1.9, aging management option AMP-5.5.

3.3 TIME-LIMITED AGING ANALYSIS EVALUATION

In the TLAA evaluation, a list of time-limited aging analyses must be defined that:

- Involve systems, structures, and components within the scope of license renewal, as delineated in §54.4(a) of 10 CFR 50
- Consider the effects of aging
- Involve time-limited assumptions defined by the current operating term, for example, 40 years
- Were determined to be relevant by the licensee in making a safety determination
- Involve conclusions or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in §54.4(b)
- Are contained or incorporated by reference in the CLB

For those defined, it must be demonstrated that:

- The analyses remain valid for the period of extended operation
- The analyses have been projected to the end of the period of extended operation
- The effects of aging on the intended function(s) will be adequately managed for the period of extended operation

Two aging effects evaluated have the potential to be defined as TLAA effects. They are: prestress force losses and fatigue. Specifically, and as summarized previously in Table 2-17, the component and effect are:

- Analytical prediction of time-dependent loss of prestress force loads in prestressing systems
- For the concrete containment structure, number of fatigue cycles at penetration anchors, and where appropriate, calculated cumulative fatigue usage factors
- Bellows number of fatigue cycles in mechanical penetrations, and where appropriate, calculated usage factors
- Number of fatigue cycles of mechanical penetrations, and where appropriate, calculated usage factors

Each of the above are evaluated as to whether they should be considered as belonging to the list of time-limited aging analyses. The evaluation is summarized in Tables 3-3 and 3-4.

As seen from this table, only mechanical penetrations associated with high temperature may require action by the utility to perform a fatigue analysis to show that an existing analysis remains valid, or be projected, to the extended period of operation. The utility may choose to adequately manage the effects of aging on the intended functions using the ASME Code Section XI surveillance and testing programs. The other component aging effects are adequately managed using this method (ASME Code Section XI surveillance and testing programs).

TABLE 3-3
TIME-LIMITED AGING ANALYSES REQUIREMENTS FOR BELONGING TO EQUIPMENT LIST

Requirements	Prestressing System Prestress Force Losses	Concrete Containment Penetration Anchors Fatigue	Mechanical Penetrations Bellows Fatigue	Mechanical Penetrations (Piping) Fatigue
Involve SSCs within scope of license renewal	Yes	Yes	Yes	Yes
Consider the effects of aging	Yes	Yes	Yes	Yes
Involve time-limited assumptions defined by the current operating term (e.g., 40 years)	Yes(1)	No(1)	No(1)	Yes(2)
Were determined to be relevant by the licensee in making a safety determination	Yes	Yes	Yes	Yes
Involve conclusions or provide the basis for conclusions related to the capability of the SSC to perform its intended functions	Yes	Yes	Yes	Yes
Are contained or incorporated by reference in the CLB	Yes	Yes	Yes	Yes

Notes:

(1) Potential source of degradation managed by plant surveillance and testing.

(2) Analysis for cyclic operation is an alternative to visual examination and testing.

TABLE 3-4
TIME-LIMITED AGING ANALYSES DEMONSTRATION REQUIREMENTS

Demonstration Requirements	Prestressing System Prestress Force Losses	Concrete Containment Penetration Anchors Fatigue	Mechanical Penetrations Bellows Fatigue	Mechanical Penetrations Fatigue
Analyses remain valid for the extended period of operation	No(5)	Not applicable (1)	Not applicable (1)	Yes(3)
Analyses have been projected to the end of the extended period of operation	Yes(5)	Not applicable (1)	Not applicable (1)	Yes(3)
Effects of aging on the intended functions will be adequately managed for the extended period of operation	Yes(2) (AMP-5.6)	Yes(2) (e.g., leak rate testing)	Yes(2) (e.g., leak rate testing)	Yes(4)

Notes:

(1) Adequacy of component related to time-dependent degradation effect not based on analysis.

(2) Effects of aging adequately managed by CLB surveillance and testing programs.

(3) A utility can perform an analysis to demonstrate adequacy of the component to perform its intended function through the period of extended operation.

(4) Effects of aging can be managed using CLB surveillance and testing programs.

(5) Analyses require update for the period of extended operation.

3.4 AGING EFFECT EVALUATION SUMMARY

The aging mechanisms for components evaluated were summarized in Table 2-17. In this table, those that are not likely to be significant were indicated as having no aging management program requirements (so indicated by NR). Those mechanisms that may result in effects of aging on the intended functions that must be adequately managed are currently using license basis inspection and test programs based on ASME Code Section XI. Aging management programs to be used for the extended operation period are based on current inspection and test activities that are defined and presented in Section 4.1. The attributes are based on plant existing maintenance, inspection, and testing programs and practices. The attributes from the programs will remain adequate to manage effects during an extended period of operation since the degradation resulting from the aging mechanisms does not increase significantly between inspection and testing periods.

Listed below are the mechanisms/effects that are managed by aging management programs into the extended operation period:

- Concrete, freeze-thaw; AMP-5.1 and AMP-5.2
- Concrete, aggressive chemical attack; AMP-5.3 and AMP-5.4
- Fatigue at penetration anchors of concrete containments; AMP-5.5
- Corrosion of reinforcing steel in inaccessible on below-grade concrete structures; AMP-5.3 and AMP-5.4
- Corrosion of containment steel liner; AMP-5.5
- Coating degradation on steel liners; AMP-5.5
- Corrosion of prestressing systems; AMP-5.6
- Prestress force loss of prestressing systems; AMP-5.6
- Stress corrosion cracking of prestressing systems; AMP-5.6
- TGSCC of electrical penetration bellows; AMP-5.5
- Bellows fatigue in mechanical penetrations and in free-standing steel containment; AMP-5.5
- Mechanical penetration fatigue associated with high-temperature piping; AMP-5.5
- Mechanical penetration embrittlement of gaskets; AMP-5.5
- Fuel transfer tube penetration embrittlement of gaskets; AMP-5.5
- Fuel transfer tube penetration mechanical wear; AMP-5.5
- Corrosion and SCC in mechanical penetrations and fuel transfer tube penetration; AMP-5.5
- Corrosion due to dissimilar weld materials in fuel transfer tube penetration; AMP-5.5
- Airlocks and hatches, mechanical wear, and embrittlement of gaskets; AMP-5.5
- Loss of pressure retention at airlocks and hatches; AMP-5.5
- Foundation settlement; AMP-5.7
- Fatigue for free-standing steel containment; AMP-5.5
- Corrosion in free-standing steel containments; AMP-5.5