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Protecting People and the Environment

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ORNL/TM-2011/410

A Summary of Aging Effects and Their Management in Reactor Spent Fuel Pools, Refueling Cavities, Tori, and Safety-Related Concrete Structures

Office of Nuclear Reactor Regulation

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ABSTRACT

The Oak Ridge National Laboratory conducted an independent review of operating experience at U.S. commercial nuclear power plants regarding spent fuel pool and reactor cavity leakage, boiling-water reactor Mark I containment torus corrosion and cracking, and aging degradation of safety-related concrete structures. The review was restricted to information in publicly available sources, including license renewal applications submitted by the licensees and safety evaluation reports prepared by the staff. Information compiled for spent fuel pool and reactor refueling cavity leakage focused on the cause and extent of leakage, the effects of borated water on materials, the effects on concrete and steel degradation, corrective actions, and the effects of material degradation on load-carrying capacity. The review of boiling-water reactor Mark I containment information involved identification of the causes of torus corrosion and cracking, the locations of corrosion and cracking, and engineering evaluations and acceptance criteria. The review of operating experience for age-related degradation of concrete structures concentrated on corrosion; loss of prestressing force; concrete cracking, scaling, and spalling, including freeze-thaw; loss of bond between the concrete and embedded steel reinforcement; loss of strength; and increase in porosity and permeability. The results of the review are summarized in a series of degradation occurrence tables and discussed in plant-specific case studies.

CONTENTS

Page

LIST OF FIGURES

Page

LIST OF TABLES

EXECUTIVE SUMMARY

As nuclear plants age, degradations of spent fuel pools (SFPs), reactor refueling cavities, and the torus structure of light-water reactor nuclear power plants (NPPs) are occurring at an increasing rate, primarily due to environment-related factors. During the last decade, a number of NPPs have experienced water leakage from the SFPs and reactor refueling cavities. In addition, since 2000, applications for license renewals have noted several cases of corrosion in boiling-water reactor (BWR) Mark I torus steel structures. Age-related concrete degradation has also been identified.

What is interesting about these structures is that it is often hard to assess their in situ condition because of accessibility problems. For example, the external portion of tori is partially exposed and, internally, underwater divers are used periodically to evaluate the prevailing aging effects in the pressure-suppression chamber. Similarly, a portion of the listed concrete structures are either buried or form part of other structures or buildings, or their external surfaces are invisible because they are covered with liners. Historically, the U.S. Nuclear Regulatory Commission (NRC) monitored their performance and, when appropriate, issued information notices (INs), including IN 2003-08, IN 2004-05, IN 2006-01, and IN 2011-15 to address and update NPP owners about leakage of SFPs and reactor refueling cavities and corrosion of BWR tori.

The Oak Ridge National Laboratory (ORNL), on its part, prepared NUREG/CR-6927, "Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures—A Review of Pertinent Factors," issued February 2007, that provides a discussion of concrete durability and the relationship between durability and performance, a review of the historical perspective related to concrete and longevity, a description of the basic materials that comprise reinforced concrete, information on the environmental factors that can impact the performance of NPP concrete structures, and a brief history of NPP concrete structures operating experience.

The objectives of this NUREG document (NUREG/CR-7111) are to (1) identify the cause, extent, and effect of leakage from deteriorated seals between the refueling cavity and reactor cavity areas, and/or leakage through stainless steel weld seams or through stainless steel base metal, (2) identify the cause, extent, and effect of leakage from BWR and pressurized-water reactor SFPs through the stainless steel weld seams or through the stainless steel base metal, and field activities that were performed to address the leakage, (3) identify possible causes of corrosion of the torus of different BWR Mark I plants, including locations and remedial actions, (4) provide an update on operating experience with respect to NPP reinforced concrete structures, (5) summarize mitigating actions (see Chapters 2 and 6) of the industry to manage or rectify issues associated with aging effects because of Objectives 1, 2, and 3 above, thus extending the safe operating life of NPPs, and, last but not least, (6) extend past studies and efforts (see Chapters 1 and 2 and Appendices A, B, and C) to serve as a precursor for the industry to identify degradation scenarios that potentially could dominate in the future (e.g., the impacts of historical water leakage on the structural integrity of SFPs and refueling cavity liners and associated concrete structures).

The identification of the cause, extent, effects, mitigation, and management of the resulting aging effects of leakage from SFPs and reactor refueling cavities, corrosion and cracking of the torus structure of light-water reactors, and degradation of safety-related concrete structures are considered critical subjects that impact license renewals. This summary of findings spanning from 1997 to about mid-2010 reviews past performance and presents the current state of increased watchfulness for these structures and components and the bulk materials used for the period of extended operation. It provides an impetus that could help draft additional or advanced inspection, monitoring, evaluation techniques, technology development, and aging management programs to support the identification of emerging modes of degradation and the future generation of aging-effects management for the continued safe operation of the legacy NPP fleet, thus fulfilling the NRC's mission. Such modes of degradation, for example, may surface in reinforced concrete structures where the composite material could be negatively impacted by adverse environments of borated water or where there is the possibility of alkali aggregate material reactivity. For the torus structure, the continuous undetected pitting corrosion translates into potential reduction of the capacity of the structure to resist the loading conditions to which it has been designed, including an increased sensitivity to buckling.

In brief, the chapter contents are as follows:

Chapter 1 offers the reader introductory, background material that includes relatively recent related projects that have been conducted at ORNL and regulatory historical documentation of interest. This chapter also provides a brief discussion of the topics of interest as identified in Objectives 1–4 noted above.

Chapter 2 presents background material related to aging management practices and programs. The "Maintenance Rule" and its objective are identified. The chapter also identifies several known problem areas and actions taken by the NRC to address these areas (e.g., adoption of American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, Subsections IWE and IWL; publication of NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," issued June 1995, and Regulatory Guide 1.127, Revision 1, "Inspection of Water Control Structures Associated With Nuclear Power Plants," issued March 1978; and programs that address protective coating and monitoring).

Chapter 3 links the current effort to previous degradation occurrence studies and outlines the data acquisition process. Information sources included publicly available documents related to operating experience (e.g., licensee event reports, NRC generic communications, NRC preliminary notification reports, NRC interim staff guidance, and NRC regulatory guides), NRC NUREGs, licensee inspection reports, license renewal applications, and NRC safety evaluation reports for license renewal applications. These information sources were screened, and reports that appeared to contain information related to one of the four topics of interest were binned (i.e., grouped) into the appropriate topic of interest for more detailed review.

Chapter 4 addresses the procedure utilized to further examine the information identified during the initial data acquisition process and to select reports for more detailed assessment. As a result of this process, 11, 12, 7, and 26 potential information sources were identified related to reactor refueling cavity leakage, SFP leakage, torus corrosion and cracking, and age-related concrete degradation, respectively.

Chapter 5 describes the presentation format and provides a table that summarizes the information with respect to the four topics of interest in terms of the identified NPPs (including plant type and occurrence date), structures affected, components impacted, construction materials, aging effects, aging mechanisms, environments, and types of source documents used to obtain the information. When sufficient information was available, a case study was prepared for a degradation occurrence listed in the summary table, followed by more detailed presentations in Appendices A, B, and C for the reactor refueling cavity, SFP, and torus corrosion and cracking, respectively. Seven, nine, and seven case studies were abridged for the reactor refueling cavity, SFP, and torus corrosion and cracking, respectively.

Chapter 6 includes brief descriptions of reactor refueling cavities, SFPs, and torus structures. It provides an outline of degradation occurrences and how these are addressed during the period of extended operation. The chapter also discusses the degradation sources, how they have been identified, inspections performed, corrective actions taken, and pertinent structural integrity assessments. Case studies for each of these topics of interest are then reviewed, and the results are presented here in an abbreviated form with respect to the specific areas of interest grouped by plant type. In addition, this chapter provides a review of age-related degradation of concrete structures and addresses the methodology used by the licensees to identify the concrete degradation and the approaches and procedures used to address, manage the aging effects of (see also Chapter 2), and resolve, where applicable, the identified issues.

Chapter 7 provides a summary and general conclusions derived as a result of the review of occurrences of degradation of SFPs, reactor refueling cavities, the torus structure, and safety-related concrete structures of light-water reactor NPPs. Topics for further consideration are also identified in this chapter.

Appendices A, B, and C provide case studies of leakages of SFPs and reactor cavities, and BWR Mark I containment torus corrosion and cracking for selected plants, based on the staffs' contributions to the safety evaluation reports and the Advisory Committee on Reactor Safeguards reviews during public meetings for license renewal. Although several occurrences of degradation were identified, sufficient information was not available to develop a statistically significant trend indicating that reactor refueling cavity, SFP, torus shell, or concrete structure degradation was more prevalent for a particular nuclear steam supply system supplier, engineering firm, or constructor.

Throughout the text, the selection of degradation occurrences involved identification of aging effects from publicly available documents, followed by associated aging mechanisms and environments. The grouping of degradations and how they are dealt with for the continuous safe operation of NPPs is expected to facilitate current and future efforts in license renewals. Consistency and clarity in presentation of information was maintained to the extent possible throughout the report.

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ABBREVIATIONS AND ACRONYMS*

3-D	three dimensional
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System (NRC)
AL	administrative letter (NRC)
ALARA	as low as reasonably achievable
AMP	aging management program
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASME Code	ASME Boiler and Pressure Vessel Code
B&W	Babcock & Wilcox
BFN	Browns Ferry Nuclear
BNL	Brookhaven National Laboratory
BWR	boiling-water reactor
BWR/1–6	boiling-water reactor/design phase 1, 2, 3, 4, 5, or 6
c-c	center-to-center
CE	Combustion Engineering
CFR	*Code of Federal Regulations*
CL	centerline
CLB	current licensing basis
cm	centimeter(s)
CS	carbon steel
DBA	design-basis accident
DBE	design-basis earthquake
DG	draft regulatory guide
DLR	Division of License Renewal (NRC)
DOD	Degradation Occurrence Database
DOT	Degradation Occurrence Table
dpm	drop(s) per minute
ECCS	emergency core cooling system
EN	event notification
EPRI	Electric Power Research Institute
ENS	Emergency Notification System (NRC)
FHB	fuel-handling building
GALL	generic aging lessons learned
GE	General Electric
GL	generic letter (NRC)
gpd	gallon(s) per day
gph	gallon(s) per hour
gpm	gallon(s) per minute
HPCI	high-pressure coolant injection
HVAC	heating, ventilation, and air conditioning
HX	heat exchanger
IE	inspection enforcement
IN	information notice (NRC)

* A complete list of abbreviations used by the NRC is provided in NUREG-0544, Revision 4, "NRC Collection of Abbreviations," issued July 1998.

IR	industry report
ISI	inservice inspection
LER	licensee event report (NRC)
LRA	license renewal application
LR-ISG	license renewal interim staff guidance
m	meter(s)
m^3	cubic meter(s)
m^3/s	cubic meter(s) per second
mL	milliliter(s)
mL/m	milliliter(s) per minute
mm	millimeter(s)
mPa	megapascal(s)
MK	Mark
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NPP	nuclear power plant
NPPD	Nebraska Public Power District
NRC	U.S. Nuclear Regulatory Commission
NUMARC	Nuclear Management and Resources Council
NUREG	NRC technical report designation
NUREG/CR	NUREG contractor report
OAR	owner's activity report
ORNL	Oak Ridge National Laboratory
PER	problem evaluation report
PNO	preliminary notification report (NRC)
ppm	part(s) per million
PSC	pressure-suppression chamber
PSEG	PSEG Nuclear, LLC
psi	pound(s) per square inch
PWR	pressurized-water reactor
RAI	request for additional information
RCS	reactor coolant system
RD	radioactive waste drain
RFC	reactor refueling cavity
RFO	refueling outage
RIS	regulatory issue summary (NRC)
RG	regulatory guide (NRC)
RPV	reactor pressure vessel
SCC	stress-corrosion cracking
SCSS	Sequence Coding & Search System
SER	safety evaluation report
SFP	spent fuel pool
SI	structural integrity
SIT	special inspection team
SME	subject matter expert
SNL	Sandia National Laboratories
SRP	standard review plans
SS	stainless steel
SSC	system, structure, and component
TMI	Three Mile Island
UT	ultrasonic testing

VT-1, -3	visual examination category per ASME Code
West	Westinghouse

1. INTRODUCTION

1.1 Purpose

Several of the 104 commercial nuclear power plants (NPPs) (Ref. 1) in the United States have experienced some water leakage from their spent fuel pools (SFPs) and reactor refueling cavities. In addition, since 2000, the license renewal applications (LRAs) filed by licensees have noted several cases of corrosion in the boiling-water reactor (BWR) Mark I containment torus steel structures. Concrete degradation has also been indicated in other instances. As NPPs age, additional occurrences or emerging modes of degradation of the SFPs, the reactor refueling cavities, the BWR Mark I containment tori, and concrete safety-related structures continue to be identified.

The nuclear power industry and, in particular, LRAs filed by licensees have discussed the possible causes of and various methods for mitigating or alleviating (1) the SFP leakage, (2) the reactor refueling cavity leakage, (3) the torus corrosion and cracking in BWR Mark I containments, and (4) the concrete degradation. The licensees have performed engineering evaluations as part of their current licenses or individual LRAs to justify extending operation of their specific NPP beyond the original license expiration date, initially set at 40 years of operation.

This project is an independent review of the operating experience of commercial NPPs in the United States over the past several years regarding the four phenomena noted above. The project also reviewed information provided by the licensees as part of their LRAs that pertains to these phenomena. Sources of information used in this project were restricted to those that are publicly available.

1.2 Background

1.2.1 Project History

The U.S. Nuclear Regulatory Commission (NRC), in concert with Oak Ridge National Laboratory (ORNL), has conducted several programs since 1988 addressing structures at NPPs, including the following five programs:

(1) The Structural Aging Program, conducted from 1988 until 1996, developed data and information on the aging of NPP concrete structures. The results from this program have been directly applied to LRA reviews (Ref. 2).

(2) The Inspection of Aged/Degraded Containments Program addressed factors that could potentially affect the capacity of NPP steel containments and the liners of reinforced concrete containments. This program was conducted from 1993 to 2001 (Ref. 3).

(3) The Effect of Phosphate Ion on Concrete Program investigated the potential for degradation of concrete materials in a phosphate-rich environment and was conducted from 2004 to 2006. This program also produced a primer on the durability of NPP concrete structures (Ref. 4).

(4) The Environmental Effects on Containments and Other NPP Structures Program developed information from 2004 to 2008 related to the effects of elevated temperature

on concrete materials and structures and inspection of inaccessible regions of the containment metallic pressure boundary and thick-section reinforced concrete structures (Refs. 5 and 6).

(5) The High Temperature Effects on Concrete Program, conducted from 2007 to 2010, compiled data and information on the effects of elevated temperature on concrete properties and performance. In addition, this program addressed methods for analysis of concrete structures experiencing thermal loadings (Ref. 7).

The NRC has also conducted many other studies and programs regarding safety-related structures assessment and analysis of age-related degradation of structures, NPP containment integrity, aging management techniques and determination, and other similar topics (Refs. 8 to 25). NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," issued June 1995 (Ref. 9), presents several examples of age-related degradation of structures and passive components in NPPs. NUREG/CR-6679, "Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants," issued August 2000 (Ref. 11), reiterates that the data presented in NUREG-1522 were "obtained from actual walk-downs of structures and components at six older NPPs (licensed before 1977). Occurrences of degradation were identified in intake structures/pump houses, service water piping, tendon galleries, masonry walls, anchorages, containments, and other concrete structures."

NUREG/CR-6679 further describes the aging of structures and passive components:

> Structures generally have substantial safety margins when properly designed and constructed. However, the available margins for degraded structures are not well known. In addition, age-related degradation may affect the dynamic properties, structural response, structural resistance/capacity, failure mode, and location of failure initiation. A better understanding of the effect of aging degradation on structures and passive components is needed to ensure that the current licensing basis (CLB) is maintained under all loading conditions. Structures include buildings and civil engineering features such as masonry walls, canals, embankments, underground structures, and stacks. Passive components consist of equipment that do not move or change their state to perform their intended function. Examples of passive components are tanks, cable tray systems, conduit systems, and heating, ventilation, and air conditioning (HVAC) ducts and supports.

ORNL completed a study of concrete durability (NUREG/CR-6927, "Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures—A Review of Pertinent Factors," issued February 2007) (Ref. 26), which examined the relationship between durability and performance, reviewed the historical perspective related to concrete and longevity, described the basic materials that constitute reinforced concrete, and presented information on the environmental factors that can affect the performance of NPP concrete structures. An appendix to NUREG/CR-6927 provided a brief history of the operating experience of NPP concrete structures.

In addition to this program, ORNL is currently engaged in another NRC program directly related to the potential degradation phenomena noted above: "Technical Assistance for Civil/Structural Review of License Renewal Applications." ORNL personnel are providing technical support in the area of concrete and containment aging to the project team reviewing and evaluating NPP LRAs.

1.2.2 Project-Related Documents

The Nuclear Management and Resources Council (NUMARC, the predecessor of the Nuclear Energy Institute (NEI)) submitted 11 industry reports (IRs) to the NRC for review in the 1990s (Ref. 27). Ten of the reports discussed aging issues of components and structures in NPPs, and the last report addressed a screening methodology for the proposed integrated plant assessments. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," was originally published in 1991 (Ref. 28). This Federal regulation established the procedures, criteria, and standards that would govern NPP LRAs. On March 1, 1993, the NRC staff recommended in SECY-93-049, "Implementation of 10 CFR Part 54, 'Requirements for Renewal of Operating Licenses for Nuclear Power Plants'" (Ref. 29), that the technical information in the IRs be incorporated into the proposed draft standard review plan (SRP) for license renewal. In September 1997, the NRC issued NUREG-1611, "Aging Management of Nuclear Power Plant Containments for License Renewal" (Ref. 15), which reconciled the NUMARC/NEI IRs with the newly published inservice inspection requirements for light-water-cooled plants addressing Class MC and metallic liners of Class CC components (Subsection IWE) and Class CC concrete components (Subsection IWL) contained in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) (Ref. 30). The initial SRP for the review of LRAs (NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants" (Ref. 31)) was issued in 2001, along with the NRC staff evaluation of existing aging management programs in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," issued July 2001 (Ref. 32) (also known as the "GALL Report"). The GALL Report was built on a previous report (Ref. 33) that systematically compiled information on plant aging. In September 2005, the NRC issued Regulatory Guide (RG) 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses" (Ref. 34). RG 1.188 resolved public comments on three prior draft versions of the RG (Draft Regulatory Guide (DG) 1009 (Ref. 35) in 1990, DG-1047 (Ref. 36) in 1996, and DG-1104 (Ref. 37) in 2000), NUREG-1800, and NUREG-1801 and also incorporated public comments (Ref. 38). The NRC revised and updated all three documents in September 2005, and NEI published NEI 95-10, "Industry Guidelines for Implementing the Requirements of 10 CFR Part 54—The License Renewal Rule," Revision 6 (Ref. 39), in June 2005. The NRC staff found that "Revision 6 of NEI 95-10 was acceptable for use in implementing the license renewal rule, without exceptions" (Ref. 38).

The NRC's Office of Nuclear Reactor Regulation is responsible for ensuring the public health and safety through licensing and inspection activities at all commercial nuclear power reactor facilities in the United States. The Division of License Renewal (DLR) evaluates LRAs. DLR performs its work in accordance with the requirements of 10 CFR Part 54. In addition, DLR uses the guidance provided in NUREG-1800 and NUREG-1801 and follows the precepts of RG 1.188 (Ref. 34). In December 2010, the NRC published Revision 2 of NUREG-1800 and NUREG-1801.

1.3 Program Description

The current program investigated the four phenomena listed in Section 1.1: (1) leakage from the reactor refueling cavity, (2) SFP leakage, (3) torus corrosion and cracking in BWR Mark I containments, and (4) aging degradation in concrete safety-related structures.

1.3.1 Leakage from the Reactor Refueling Cavity

This activity investigated the cause and extent of leakage from the deteriorated seals between the reactor refueling cavity and the reactor cavity areas, and leakage through the stainless steel weld seams and through the stainless steel base metal. This portion of the project also gathered information on the effect of borated water on steel (carbon, stainless shapes and liners) components, concrete steel reinforcement (i.e., rebar), and concrete.

For BWR NPPs, the primary concern is corrosion of the drywell shell and support structure. In the case of pressurized-water reactors (PWRs), borated water leakage may cause corrosion of the primary shield liner, reactor supports, and containment structure, or it may affect other structures and components on which the leaking water accumulates. This part of the project also examined and documented field activities performed by the different licensees to detect a leakage path from the reactor cavity; how they determined the extent of deterioration of the concrete, rebar, and steel; the methods they used to try to stop the leakage; and approaches aimed at evaluating the impact of leakage on the load-carrying capacity of deteriorated concrete structures.

1.3.2 Spent Fuel Pool Leakage

This activity investigated the cause and extent of leakage from BWR and PWR SFPs through the stainless steel weld seams and base metal. Information is provided on the effect of borated water on the stainless steel liner plate, rebar, and concrete. The impact of water leakage on a stainless steel liner leak detection system (e.g., telltale drains) is also scrutinized as part of the project. As described in Section 1.3.1 of this report, this part of the study also examined and documented field activities performed by the different licensees to detect a leakage path from the SFP; how they determined the extent of deterioration of the concrete, rebar, and steel; and the methods they used to stop the leakage and evaluate the load-carrying capacity of deteriorated concrete structures.

1.3.3 Torus Corrosion and Cracking in the Boiling-Water Reactor Mark I Containment

This activity addressed the possible causes of corrosion and cracking in the BWR Mark I containment torus, the so-called “inverted lightbulb” design, of the 22 BWR Mark I NPPs. This investigation included both coated and uncoated tori. The specific location where the corrosion was found is flagged and documented. This part of the report summarizes the licensee engineering evaluations performed in accordance with Subsection IWE of Section XI of the ASME Code. It also lists the acceptance criteria (when available) used by the licensees to determine if additional evaluations and monitoring are required for an extended period of operation.

1.3.4 Age-Related Degradation in Concrete Safety-Related Structures

This activity investigated occurrences involving age-related degradation of safety-related concrete structures at both PWRs and BWRs other than those associated with reactor refueling cavities and SFPs. This portion of the study also investigated the methodology used by licensees to identify the concrete degradation, the licensees' approaches, and the procedures used to resolve concrete degradation issues. The extent of the study, however, was limited primarily to events after February 2007 (see Section 3.2.5).

Results for occurrences that were not available in publicly accessible documents about aging effects, aging mechanisms, and environments were not included within the scope of this report. Also, occurrences that pertain to the following areas were not considered to be age related and therefore were not addressed in detail:

- construction defects (voids, honeycombing, delaminations) and material deficiencies (wrong material, inadequate strength)
- shrinkage cracking
- design errors
- steel liner corrosion
- liner repairs (liner replacement, steam generator replacement)
- moisture barrier degradation
- liner coating degradation
- bulging or distortion of liner plate
- liner penetration and sleeve damage
- physical damage to concrete where the degradation mechanism involves a one-time event (impact, overload)

2. AGING MANAGEMENT PRACTICES AND PROGRAMS

2.1 Aging Management Practices

Nuclear power plant owners are required to monitor the performance or condition of structures, systems, and components (SSCs) against the owner-established goals, in a manner sufficient to give reasonable assurance that such SSCs are capable of fulfilling their intended functions based on rules in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.65, as noted in "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," first issued in 1991 (Ref. 40). Often referred to as the "Maintenance Rule," 10 CFR 50.65 further requires the licensee to take appropriate corrective action when the performance or condition of an SSC does not conform to established goals. The main objective of the Maintenance Rule is to monitor the overall continuing effectiveness of maintenance programs used by the licensees of operating reactors to ensure that safety-related (and certain nonsafety-related) SSCs are capable of performing their intended functions. The rule is performance based and not prescriptive.

2.2 Known Problem Areas

The following sections describe the impact of codes and regulations and their subsections on actions taken by the U.S. Nuclear Regulatory Commission (NRC) to address known problem areas for SSCs and to establish consistency in describing degradation effects, degradation mechanisms, and environments that are important to the aging management of these structures.

2.2.1 ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWE

Inservice inspection (ISI) requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section XI, Subsection IWE, for steel containments (Class MC) and steel liners for concrete containments (Class CC) are imposed on licensees by 10 CFR 50.55a, "Codes and Standards" (Ref. 41). ASME Code Section XI, Subsection IWE, was incorporated into 10 CFR 50.55a in 1996. Before then, operating experience pertaining to the degradation of steel components of containment was gained through the inspections required by Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," and ad hoc inspections conducted by licensees and the NRC.

The following information notices (INs) or NUREG documents describe occurrences of corrosion in steel containment shells or liners of reinforced concrete containments:

- IN 86-99, "Degradation of Steel Containments," dated December 8, 1986 (Ref. 21)
- IN 88-82, "Torus Shells with Corrosion and Degraded Coatings in BWR Containments," dated October 14, 1988 (Ref. 22)
- IN 89-79, "Degraded Coatings and Corrosion of Steel Containment Vessels," dated December 1, 1989 (Ref. 23), and Supplement 1 to IN 89-79, dated June 29, 1990 (Ref. 24)

- IN 97-10, “Liner Plate Corrosion in Concrete Containments,” dated March 13, 1997 (Ref. 42)

- IN 2004-09, “Corrosion of Steel Containment and Containment Liner,” dated April 27, 2004 (Ref. 43)

- IN 2011-15, “Steel Containment Degradation and Associated License Renewal Aging Management Issues, dated August 1, 2011 (Ref. 44)

- NUREG-1522, “Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures,” issued June 1995 (Ref. 9)

Generic Letter (GL) 87-05, “Request for Additional Information Assessment of Licensee Measures To Mitigate and/or Identify Potential Degradation of Mark I Drywells,” issued in 1987 (Ref. 45), addressed the potential for corrosion of boiling-water reactor (BWR) Mark I steel drywells in the “sand pocket region.” The Section XI, Subsection IWE, aging management program considers the liner plate and containment shell corrosion and cracking concerns described in these generic communications. Implementation of the ISI requirements of Subsection IWE, in accordance with 10 CFR 50.55a, augmented to consider operating experience, and as recommended in License Renewal Interim Staff Guidance (LR-ISG) 2006-01, “Final Plant-Specific Staff Aging Management Program for Inaccessible Areas of Boiling Water Reactor Mark I Steel Containment Drywell Shell,” dated November 16, 2006 (Ref. 46), is a necessary element of aging management for steel components of steel and concrete containments through the period of extended operation.

The full scope of Subsection IWE includes steel containment shells and their integral attachments; steel liners for concrete containments and their integral attachments; containment hatches, airlocks, and moisture barriers; and pressure-retaining bolting. ASME Code Section XI, Subsection IWE, and the additional requirements specified in 10 CFR 50.55a(b)(2) constitute an existing mandated program applicable to managing the aging of steel containments, steel liners of concrete containments, and other containment components.

The primary ISI method specified in Subsection IWE is visual examination (general visual, VT-3, and VT-1). Limited volumetric examination (ultrasonic thickness measurement) and surface examination (e.g., liquid penetrant) may also be necessary in some instances to detect aging effects. Subsection IWE specifies acceptance criteria, corrective actions, and expansion of the inspection scope when degradation exceeding the acceptance criteria is found. The program attributes are augmented to incorporate aging management activities, recommended in LR-ISG-2006-01 (Ref. 46), needed to address the potential loss of material due to corrosion in the inaccessible areas of the BWR Mark I steel containments.

Degradation of threaded bolting and fasteners in closures for the reactor coolant pressure boundary has occurred as the result of boric acid corrosion, stress-corrosion cracking (SCC), and fatigue loading (Inspection Enforcement (IE) Bulletin No. 82-02, “Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants,” dated June 2, 1982 (Ref. 47), and GL 91-17, “Generic Safety Issue 29, ‘Bolting Degradation or Failure in Nuclear Power Plants,’” dated October 17, 1991 (Ref. 48)). It is to be noted that SCC has occurred in high-strength bolts used for nuclear steam supply system component supports, as described in Electric Power Research Institute (EPRI) Report NP-5769, “Degradation and Failure of Bolting in Nuclear Power Plants,” issued April 1998 (Ref. 49). The augmented ASME Code Section XI, Subsection IWE, incorporating recommendations documented in EPRI NP-5769 and EPRI TR-104213, "Bolted Joint Maintenance and Application Guide," issued December 1995 (Ref. 50), is necessary to ensure containment bolting integrity.

Inspections are augmented to require surface examination for detection of cracking such as described in IN 92-20, "Inadequate Local Leak Rate Testing," dated March 3, 1992 (Ref. 25), and to address recommendations for structural bolting delineated in NUREG-1339, "Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants," issued June 1990 (Ref. 51), and industry recommendations delineated in EPRI NP-5769; EPRI NP-5067, "Good Bolting Practices, a Reference Manual for Nuclear Power Plant Maintenance Personnel," issued in 1990 (Ref. 52); and EPRI TR-104213 (Ref. 50). The program is also augmented to require surface examination of dissimilar metal welds of vent line bellows in accordance with examination Category E-F, as specified in the 1992 edition of ASME Code Section XI, Subsection IWE. If surface examination is not possible, an appropriate test from 10 CFR Part 50, Appendix J, may be conducted for pressure boundary components.

IN 97-10 (Ref. 42) identified specific locations where concrete containments are susceptible to liner plate corrosion. Other operating experience indicates that foreign objects embedded in concrete have caused through-wall corrosion of the liner plate at a few plants with reinforced concrete containments. IN 92-20 (Ref. 25) described an occurrence of containment bellows cracking, which resulted in loss of leaktightness. More recently, IN 2006-01, "Torus Cracking in a BWR Mark I Containment," dated January 12, 2006 (Ref. 53), described through-wall cracking and its probable cause in the torus of a BWR Mark I containment. The licensee identified the cracking in the heat-affected zone at the exhaust pipe torus penetration of the high-pressure coolant injection (HPCI) turbine. The licensee concluded that the cracking was most likely initiated by cyclic loading due to condensation oscillation during HPCI operation. These condensation oscillations induced on the torus shell may have been excessive because of the lack of an HPCI turbine exhaust pipe sparger, which many licensees have installed.

Subsection IWE also requires examination of coatings that prevent corrosion, as described in NUREG-1801, Revision 2, "Generic Aging Lessons Learned (GALL) Report," issued December 2010.

2.2.2 ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL

ASME Code Section XI, Subsection IWL, was incorporated into 10 CFR 50.55a in 1996. Before then, the prestressing tendon inspections were performed in accordance with the guidance provided in Regulatory Guide (RG) 1.35, "Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containments" (Ref. 54), and RG 1.35.1 (Revision 3), "Determining Prestressing Forces for Inspection of Prestressed Concrete Containments" (Ref. 55). Operating experience pertaining to degradation of reinforced concrete in concrete containments was gained through the inspections required by Appendix J to 10 CFR Part 50 and ad hoc inspections conducted by licensees and the NRC. NUREG-1522 (Ref. 9) described occurrences of cracked, spalled, and degraded concrete for reinforced and posttensioned concrete containments. The NUREG also described cracked anchor heads for the prestressing tendons at three posttensioned concrete containments. IN 99-10, "Degradation of Prestressing Tendon Systems in Prestressed Concrete Containments," dated April 13, 1999 (Ref. 56), described occurrences of degradation in prestressing systems. The Subsection IWL program considers the degradation concerns described in these documents. Implementation of

Subsection IWL, in accordance with 10 CFR 50.55a, is a necessary element of aging management for concrete containments (Refs. 28 and 32).

2.2.3 Structures Monitoring Program

Although in many plants structures monitoring programs have been implemented only recently, plant maintenance has been ongoing since initial plant operations. NUREG-1522 (Ref. 9) documents the results of a survey sponsored in 1992 by the NRC's Office of Nuclear Reactor Regulation to obtain information on the types of distress in the concrete and steel structures and components, the type of repairs performed, and the durability of the repairs. Licensees who responded to the survey reported cracking, scaling, and leaching of concrete structures. Degradation occurrences were attributed to drying shrinkage, freeze-thaw, and abrasion. NUREG-1522 also describes the results of NRC staff inspections at six plants. The staff observed concrete degradation, corrosion of component support members and anchor bolts, cracks and other deterioration of masonry walls, and ground water leakage and seepage into underground structures. The observed and reported degradations were more severe at coastal plants than those observed in inland plants as a result of brackish and sea water. Recent license renewal applicants reported similar degradation and corrective actions taken through their structures monitoring programs. Many license renewal applicants have found it necessary to enhance their structures monitoring program to ensure that the aging effects of structures and components included in 10 CFR 54.4, "Scope" (Ref. 28), are adequately managed during the period of extended operation. There is reasonable assurance that implementation of the structures monitoring program described above will be effective in managing the aging of the in-scope structures and components (Ref. 32).

2.2.4 Regulatory Guide 1.127

RG 1.127, Revision 1, "Inspection of Water-Control Structures Associated with Nuclear Power Plants," issued March 1978 (Ref. 57), describes an acceptable basis for developing an ISI and surveillance program for dams, slopes, canals, and other raw-water control structures associated with emergency cooling water systems or flood protection of nuclear power plants. The RG 1.127 program addresses age-related deterioration, degradation caused by extreme environmental conditions, and the effects of natural phenomena that may affect water control structures. The RG 1.127 program recognizes the importance of periodic monitoring and maintenance of water control structures so that the consequences of age-related deterioration and degradation can be prevented or mitigated in a timely manner. RG 1.127 provides detailed guidance for the licensee's inspection program for water control structures, including guidance on engineering data compilation, inspection activities, technical evaluation, inspection frequency, and the content of inspection reports.

RG 1.127-based programs have detected degradation of water control structures at several nuclear power plants, and, in some cases, the degradation has required remedial action. NUREG-1522 (Ref. 9) described occurrences of and corrective actions taken for severely degraded steel and concrete components at the intake structure and pump house of coastal plants. Other degradation described in NUREG-1522 includes appreciable leakage from the spillway gates, concrete cracking, corrosion of spillway bridge beam seats of a plant dam and cooling canal, and appreciable differential settlement of the outfall structure of another. No loss of intended functions has resulted from these occurrences. Therefore, it can be concluded that the inspections implemented in accordance with the guidance in RG 1.127 have been successful in detecting significant degradation before loss of intended function occurs (Ref. 32).

2.2.5 Protective Coating Monitoring and Maintenance Program

Accounts of industry experience pertaining to coating degradation inside containment and the consequential clogging of sump strainers appear in the following:

- IN 88-82, “Torus Shells with Corrosion and Degraded Coatings In BWR Containments,” dated October 14, 1988 (Ref. 22)

- Bulletin 96-03, “Potential Plugging of Emergency Core Cooling Suction Strainers by Debris in Boiling-Water Reactors,” dated May 6, 1996 (Ref. 58)

- GL 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” dated September 13, 2004 (Ref. 59)

- GL 98-04, “Potential for Degradation of the Emergency Core Cooling System and the Containment Spray System after a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment,” dated July 14, 1998 (Ref. 60)

- ASME Code Case N-597, “Requirements for Analytical Evaluation of Pipe Wall Thinning,” dated November 18, 2003 (Ref. 61)

In July 2000, the NRC issued RG 1.54, Revision 1, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants” (Ref. 62). Monitoring and maintenance of Service Level I coatings conducted in accordance with Regulatory Position C4 are expected to be effective in managing degradation of Service Level I coatings and, consequently, to be an effective means to manage loss of material due to corrosion of carbon steel structural elements inside containment (Ref. 32).

3. DATA ACQUISITION

3.1 Introduction

A previous study, NUREG/CR-6679, “Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants,” issued August 2000 (Ref. 11), examined and assessed age-related degradation of structures and components at U.S. commercial nuclear power plants (NPPs). The purpose of the study, conducted by Brookhaven National Laboratory (BNL) under the auspices of the U.S. Nuclear Regulatory Commission’s (NRC’s) Office of Nuclear Regulatory Research, was to develop a technical basis for the validation and improvement of analytical methods to be used to address issues related to the degradation of structures and passive components. The study initially collected and evaluated NPP degradation occurrences. The study then documented the data in the Degradation Occurrence Database (DOD), which is a collection of events that occurred during operation from 1985 to 1997. This current study builds on that specific information but emphasizes operational events that have occurred since 1997. In this way, the events occurring within the industry from 1997 until about mid-2010 will be readily available. The current study complements NUREG/CR-6679 in that all occurrences concerning a nuclear power plant structure’s degradation, and those concerning the phenomena mentioned in Chapter 1 of this report, will now be accounted for from 1985 through mid-2010.

The database assembled in NUREG/CR-6679 (Ref. 11) and its associated information were developed before the publication of NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” issued July 2001 (Ref. 31), and NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” issued July 2001 (Ref. 32) (see Section 1.2.2 of this report); consequently, there is not a one-to-one correspondence between the criteria used for the BNL database and that of the current study. For example, NUREG-1800 identifies 10 aging effects and mechanisms that apply to NPP containments. NUREG-1801 identifies seven aging effects and mechanisms for structures. In comparison, NUREG/CR-6679 lists 18 categories of structures and components and 10 aging effects and mechanisms, which are not readily comparable to the items identified in NUREG-1800 and NUREG-1801. Moreover, NUREG-1801 lists 50 separate components that are included in current aging management programs at U.S. commercial NPPs. Consequently, the data and information presented in NUREG/CR-6679 are used to augment the information and data presented in this report but cannot be compared directly to the current information.

3.2 Operating Experience Data Acquisition

3.2.1 Licensee Event Reports

In NUREG-1022, Revision 2, “Event Reporting Guidelines: 10 CFR 50.72 and 50.73,” issued October 2000 (Ref. 63), the NRC modified and codified Title 10 of the *Code of Federal Regulations* (10 CFR) 50.73, “Licensee Event Report System,” effective January 1, 1984, to define and specify the types of events and conditions reportable to the NRC. NUREG-1022 (Revision 2) notes that “The purpose of the rule was to standardize the reporting requirements for all NPP licensees, to eliminate reporting events of low individual significance, and to require a more thorough documentation and analyses of reported events.” The licensees are required to report, among other things, an operation or condition prohibited by a plant’s technical specifications and a degraded or unanalyzed condition of the NPP or its principal safety barriers. The plant's technical specifications require monitoring of the containment and associated structures.

Two databases contain the 7,403 licensee event reports (LERS) filed with the NRC for licensed NPPs from 1997 through 2009, one managed by Oak Ridge National Laboratory (ORNL) and the other by Idaho National Laboratory. These two databases are both searchable and allow for retrieval of relevant LERs.

The LERs identified were from the Sequence Coding and Search System (SCSS). The SCSS is an electronic database developed to allow users to retrieve commercial nuclear plant operating experience data from LERs. Instead of providing the actual LER text, the database reduced the LER descriptive text to coded, searchable word sequences that are both computer-readable and computer-searchable. The system provides a structured format for the detailed coding of such factors as components, systems, causes, and effects. Data on component failures include the type and number of components involved, the system to which the components belong, the cause and mode of failure, and the effect of failures on plant systems. There are over 400 specific component codes and more than 100 cause-and-effect code designations.

On the other hand, LER information obtained from the Idaho National Laboratory is essentially text based. That is, the text of the LERs is searchable, and certain form data are captured individually (e.g., plant name, date of occurrence, basic unit effect), but the events are not coded into word sequences. The text search included terms such as "refueling cavity," "transfer pit," "corrosion or wastage or erosion," "refuel pool," and "liner cladding degradation." However, a simple check of the logic indicates that the same word or word combination may be identified by the search in the same document but may not have the same meaning. For example, the words "containment" and "corrosion" may appear in the same body of text, but the context of the word "containment" may have been used to describe a bottle or some other container and not be associated with the NPP containment. Similarly, the word "corrosion" may have been used to describe something other than the issues of concern for this project. These two words may then have been separated within the text and have no relationship to one another (i.e., each word is contained in a different sentence). This is an example of a false positive hit in the search. However, the search engine used to find relevant events in the text-only database had the capability to perform Boolean search logic and nesting, which reduced the large volume of text in the LERs to a more manageable size, resulting in fewer LERs to be reviewed. Creating the search logic for full-text documents was an iterative process. Often, changes were needed in potential terms to make the searches more inclusive, such that the search results would include those events of interest while also keeping the number of false positives to a minimum.

The first level of computer screening of the two databases produced 673 LERs for review. Two engineers reviewed the LERs (a primary reviewer who examined the LERs in detail and a secondary reviewer who corroborated the first reviewer's findings). If the two engineers disagreed, the event was passed on to the next phase of data acquisition, where they were stored (i.e., binning) prior to more detailed review. In this first level of review, 591 events were rejected from further consideration; thus, 82 events (or LERs) were passed on to the binning stage of the project's data acquisition phase.

3.2.2 NRC Generic Communications

The NRC employs various types of reports to communicate information regarding NPPs to the industry and the public at large. These reports offer a variety of information concerning NPP operating experience and convey or discuss NRC policy, as well as other topics of interest to the nuclear industry and the public. Many times, these NRC generic communications disclose information not readily available elsewhere (these generic communications are available at http://www.nrc.gov/reading-rm/doc-collections/gen-comm/). Therefore, these reports were screened for operational data and additional information for use in this project.

In the late 1970s and into the early 1980s, the NRC issued circulars to transmit time-sensitive information related to public health and safety; however, these were discontinued in February 1985 and were excluded from this study, as this information would have been addressed by the prior BNL study (Ref. 11). Administrative letters (ALs) are another type of NRC communication report. They inform various licensees of specific regulatory or administrative information or clarify this information. The NRC discontinued the ALs in September 1999, and only the ALs from 1997 through 1999 (19 total) were screened for this project. None of the ALs screened were found to have relevant information. The NRC also issues generic letters (GLs), which are transmittals by the NRC to request action by the licensees, distribute policy positions, or inform the industry about voluntary programs. The GLs from 1997 through 2010 were screened (21 total) for information relevant to this project. Some of these, where appropriate, were reviewed in detail to determine their relevance to the program. For example, GL 97-01, "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," dated April 1, 1997 (Ref. 64), which discusses control rod drive mechanism nozzle issues and related reactor pressure vessel penetration concerns, was reviewed in detail, identified as extraneous to this project, and then excluded from further screening.

The NRC began issuing regulatory issue summaries (RISs) as a means of transmitting information affecting nuclear licensees and the nuclear industry. Among other uses, the RISs document NRC endorsement of the resolution of issues addressed by industry-sponsored initiatives, announce staff technical positions, and discuss material that was previously found in ALs. Of the 265 RISs reviewed for relevance to this project, three (Refs. 65–67) were selected for a more detailed examination during the binning process of the data acquisition phase.

In 1971, the NRC initiated the issue of bulletins as a means of both requesting information from licensees and advising them of new or revised NRC programs that may affect operations. All of the NRC bulletins from 1997 through the present were reviewed for applicability to this project. Four bulletins (Refs. 68–71) were selected for a more thorough review during the binning process of the program.

Information notices (INs) appeared as communication tools of significant information to licensees in 1979. The INs convey significant information about safety, safeguards, or environmental issues. Since 1997, the NRC has issued 473 INs. These were evaluated for applicability, and 26 INs were subsequently reviewed in depth to determine if they contained information pertinent to the current project. Of these, 16 were included in the binning process (Refs. 42, 72–86).

3.2.3 NRC Reports

The NRC modified and codified 10 CFR 50.72, “Immediate Notification Requirements for Operating Nuclear Power Reactors”, effective January 1, 1984, to define and specify the types of events and conditions reportable to the NRC. Whereas 10 CFR 50.73 requires a 30-day written report to the NRC, 10 CFR 50.72 requires NRC notification via the Emergency Notification System (ENS) within 1 hour, 4 hours, or 8 hours, depending on the nature or severity of the event. NUREG-1022, Revision 2 (Ref. 63), describes the similarities and differences between the two regulations; the various reporting criteria in the two overlap considerably. From January 1999 to May 2010, 10,541 immediate reports were made to the ENS. A list of these is available at http://www.nrc.gov/reading-rm/doc-collections/event-status/ (Ref. 87); however, unlike the LERs, these events were not codified, nor is the list searchable. The reports are listed on a daily basis for each year, and, within each daily listing, there may be as many as 15 to 20 events. A text search of each of the reports issued prior to 1999 and each of the 10,541 reports issued since January 1999 was not feasible; therefore, a sample search was done to find events most likely to be of interest. The sample set of 994 was obtained from a combination of a simple Boolean logic text search and a random selection. The random selection followed the standard rules of search techniques and it is described below.

The set of 10 CFR 50.72 reports was assumed to be a discrete uniform distribution using a binary classification modeled by the Bernoulli distribution. A random sampling of the set of reports was performed without replacement following a hypergeometric distribution. The hypergeometric distribution satisfies several conditions. For example, the sample population is finite, the size of the population is known, the number of items with the attribute of interest is known, and each item in the sample is drawn at random. The probability that a significant event may have been “missed” and was actually in the set of events not reviewed was also estimated. In this instance, significant was defined to be a report concerning the four topics of interest to the present study. The concept of “missed” acknowledges an event of significance within the sample set with a measured degree of confidence that it was not left out of the sample set. Both Type I and Type II errors were accounted for, where a Type I error is an error in which a significant event was left out of the review that should have been included, and a Type II error was an event included in the sample set that should have been left out of the review.

In addition to conducting a generalized sampling of the set of 10 CFR 50.72 reports, another type of search focused on the included operating experience data. Via a text search engine utilizing a simple Boolean logic, events potentially reporting reactor refueling cavity leakage, spent fuel pool leakage, torus corrosion and cracking, and age-related concrete degradation were identified. Specifically, the data sources (10 CFR 50.72 reports) were obtained and stored on an ORNL computer for analysis using a desktop search engine. This search engine provided the capability for Boolean search logic and nesting, which, in turn, allowed complex text searches to be conducted. Thus, it was possible to reduce large volumes of text to more manageable numbers for events to be reviewed.

The event report database (Ref. 87) also contains preliminary notification reports (PNOs). The PNOs are brief descriptions of significant safety or safeguards events from each NRC region. The PNOs are listed by region, by year, and by number. There is no available database containing all 487 PNOs, and the total number of reported PNOs is not exact. Therefore, as with the 10 CFR 50.72 reports, this set of events was investigated using the same approach as noted above.

Also contained in the database (Ref. 87) are NRC Headquarters daily reports. These reports were typically used to communicate with the NRC Headquarters concerning management-level information, special reports, and some NRC resident inspector notifications. The NRC discontinued these reports on September 1, 2009. Like the PNOs, these reports were collected for each region, by year and number. They were not sequentially numbered, and the total number of events was approximately 1,400. Therefore, as for the PNOs, this set of reports was only sample-searched.

Two engineers (a primary reviewer who examined the reports in detail and a secondary reviewer who corroborated the first reviewer's findings, as previously discussed for a similar process) reviewed the sample set of these reports. If the two engineers disagreed, the event was passed on to the next phase of data acquisition (i.e., binning). The first-level screening resulted in a secondary review of 24 PNOs, 62 event notifications (ENs), and 5 Headquarters daily reports. This second level of review resulted in the inclusion of one PNO (Ref. 88), seven ENs (Refs. 89–95), and no Headquarters reports in the binning process of data acquisition.

In 10 CFR Part 21, "Reporting of Defects and Noncompliance" (Ref. 96), the NRC establishes the procedures and requirements for NPP license holders to report any facility, activity, or component that fails to comply with the Atomic Energy Act of 1954. Reports made in accordance with this regulation are known as "Part 21 reports." The NRC receives these reports and enters them in the Part 21 report database (Ref. 97). The titles of the 487 Part 21 reports from 1997 through 2010 were reviewed for potential inclusion in the binning process of this project. One notification, No. 2005-23, was examined and reviewed in greater detail. However, this notification was actually a 10 CFR 50.72 report (EN 41783) dated June 17, 2005, and subsequently withdrawn on August 2, 2005. The report involved the pressurizer heater elements of a pressurized-water reactor (PWR) and was eliminated from the binning process.

3.2.4 NRC Guides

The NRC has developed staff guidance documents to facilitate the review of LRAs prepared in accordance with the License Renewal Rule (Ref. 28). These guidance documents, called license renewal interim staff guidance (LR-ISG), provide a process to capture and communicate interim guidance for new insights, lessons learned, and emergent issues to create a license renewal program that progressively improves (Ref. 98). The NRC tracks and posts the various LR-ISGs (Ref. 99). The LR-ISGs were reviewed as a source of potential operating experience or lessons-learned information for the four areas of study for this project. LRA-ISG-2006-01, "Final License Renewal Interim Staff Guidance LR-ISG-2006-01: Plant-Specific Aging Management Program for Inaccessible Areas of Boiling Water Reactor Mark I Steel Containment Drywell Shell," dated November 16, 2006 (Ref. 100), was selected for the binning process for this project. The remaining LRA-ISGs were rejected from further consideration.

The NRC issues regulatory guides (RGs) "to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated accidents, and to provide guidance to applicants." NRC databases list all of the RGs (Ref. 101) and draft RGs (Ref. 102). The titles of the RGs and draft RGs were reviewed for applicability to this project. Eight RGs (Refs. 54–55, 103–108) were selected for the binning process of the data acquisition phase of this project.

3.2.5 NRC Publications

NRC publications by the staff (NUREGs) and NRC contractors (NUREG/CRs) were considered for possible operating experience information or lessons-learned data to be included as part of the data acquisition for this project. NUREG/CR-6927, "Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures—A Review of Pertinent Factors," issued February 2007 (Ref. 26), which has an extensive list of references concerning NRC publications related to concrete degradation, was considered to be a comprehensive resource. Therefore, it was decided to concentrate the current review on reports issued after this document was published (February 2007).

The NRC maintains a current list of staff and contractor reports (Refs. 109 and 110). The 81 staff reports, from NUREG-1853 through NUREG-1934, were reviewed, and only NUREG-1863, "Review of Responses to NRC Bulletin 2003-02, 'Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity,'" issued September 2006 (Ref. 18), was determined to have information relevant to this project and hence was included in the binning process. Similarly, the 75 NRC contractor reports, from NUREG/CR-6928 through NUREG/CR-7003, were reviewed. Eight of these reports were examined in more detail, and, of these eight reports, only NUREG/CR-6986, "Evaluations of Structural Failure Probabilities and Candidate Inservice Inspection Programs," issued May 2009 (Ref. 111), was considered to have relevant information for the purposes of this project and was included in the binning process.

3.3 <u>Selected Licensee Inspection Reports</u>

In 10 CFR 50.55, "Conditions of Construction Permits, Early Site Permits, Combined Licenses, and Manufacturing Licenses," the NRC establishes the terms and conditions of the license or permit (e.g., manufacturing permit, construction permit, early site permit, combined license) for each of the licensees of NPPs. For example, 10 CFR 50.55 sets forth the requirement for notifying the NRC if a substantial safety hazard exists or a defect is found in the construction of the NPP. The regulation further delineates specifications in 10 CFR 50.55a, "Codes and Standards," for systems and components of both boiling-water reactors (BWRs) and PWRs, as it states that the systems and components must "meet the requirements of the ASME Boiler and Pressure Vessel Code." Also, 10 CFR 50.55a sets forth the requirements for continuous examination, investigation, and inspection required to maintain an NPP license. Two subparts of 10 CFR 50.55a specifically govern the licensee's inspection of containments for both BWRs and PWRs. The regulations in 10 CFR 50.55a(b)(2)(viii) and 10 CFR 50.55a(b)(2)(ix) require licensees to conduct periodic inspections of their respective containment.

As a result of these Federal regulations, the licensees file regular inspection reports with the NRC in accordance with 10 CFR 50.55. These reports are called inservice inspection reports (ISIs), owner's activity reports, or refueling outage ISIs. Hereafter, for brevity, these reports will be collectively referred to as ISIs. The intent is to avoid repetition and redundancy.

The NRC Office of Nuclear Regulatory Research, Division of Engineering, investigated liner corrosion ISI reports prepared by the industry during the period from 1999 to February 2010 (Ref. 112). Reports for NPPs were examined where the containment liner came in contact with concrete. For BWRs, this included those with concrete containments (either reinforced or posttensioned) with steel liners (11 BWRs) (Ref. 14) and one with a freestanding steel primary containment (Nine Mile Point Nuclear Station, Unit 1). All PWRs were included except those with a large, dry steel cylinder primary containment, those with a reinforced concrete shield building (seven PWRs), and those with an ice condenser steel cylinder primary containment with a concrete shield building (seven PWRs). NPP ISIs examined included 12 for BWRs and 55 for PWRs. The breakdown by category was (1) 12 BWRs, (2) seven PWRs with reinforced, subatmospheric primary containment, (3) 10 PWRs with reinforced concrete dry containment, (4) two PWRs with reinforced ice condenser primary containment, and (5) 36 PWRs with posttensioned dry concrete primary containment. The resulting spreadsheet identified 268 ISIs.

The ISI listings in the resulting spreadsheet were reviewed, and a subset of potential reports to be examined in more detail was developed. Of the 268 available ISI reports, 92 were thought to contain information that would be pertinent to the issues of this project. Of these 92 ISI reports, 87 were obtained and sent on for detailed review via the binning process. Because concrete-related issues were the primary focus of the NRC study, these 87 reports would primarily be related to this report's fourth topic of investigation, concrete age-related degradation.

3.4 License Renewal Applications and Safety Evaluation Reports

3.4.1 License Renewal Applications

As indicated in Section 1.2 of this report, 10 CFR Part 54, "Requirements for Renewal of Operating Licenses for Nuclear Power Plants", governs the renewal of operating licenses for NPPs. This Federal regulation defines the NPP current licensing basis (CLB) and defines systems, structures, and components (SSCs) within the scope of the regulation. Both the CLB and SSCs are clearly defined, including what constitutes a safety-related system and which nonsafety-related systems may affect the functions of the safety-related systems. The rule further defines both the content and extent of an integrated plant assessment. In addition to 10 CFR Part 54, guidance documents (e.g., Refs. 31, 32, 34, and 39) are used to develop a license renewal application (LRA), as well as to conduct its review and acceptance.

As of August 12, 2011, the NRC listed 52 LRAs for 84 individual NPPs (Ref. 113). This includes 9 LRAs currently undergoing review and 43 LRAs that have completed the acceptance review cycle. The 43 completed LRAs represent 71 units, and the 9 LRAs currently undergoing review represent 13 units.

3.4.2 NRC Safety Evaluation Reports of License Renewal Applications

The NRC safety evaluation reports (SERs) for each of the completed LRAs were reviewed to determine if the SER contained information relating to the four areas of study for this project. The draft SERs for each of the completed LRAs were also reviewed to ascertain if any early information was contained in the draft report but not included in subsequent revisions. This meant that, in addition to the 32 SERs, an additional 54 draft SERs were reviewed. Thus, the review encompassed a total of 86 documents associated with the completed LRAs. Similarly, the SERs for the LRAs still in the review cycle were also examined.

3.5 Sorting and Binning of Data

The information sources discussed in the foregoing sections of this report are tabulated in Table 3-1. Following the initial reviews mentioned in Sections 3.1 and 3.2, two subject matter experts (SMEs) thoroughly reviewed each item identified in Table 3-1. Also, while reviewing a particular report for an event listed in Table 3-1, a reviewer may have discovered references to other accounts of the same event. In one case, in IN 2004-09, "Corrosion of Steel Containment and Containment Liner," dated April 27, 2004 (Ref. 43), more than six separate events were discussed. Each of these six events referenced other documents or records. Thus, multiple sources are noted for several events.

The SMEs placed selected items into one of four bins, with each bin corresponding to one of the areas of interest for this report. Usually, both SMEs agreed on the selection of and resulting placement into a bin; however, if the SMEs did not agree on keeping the item for the binning process, the item was kept for the binning process and a more detailed review. That is, if either SME selected an item for binning, it was sent to the next phase.

Some items that were kept for the binning process were rejected from further study or analysis after a more thorough examination by an SME. Chapter 4 of this report discusses this advanced examination of the items. Additionally, if any item selected for binning contained information pertaining to more than one area of interest, it was duplicated and retained in both bins. For example, if an LER had information regarding both concrete degradation and reactor refueling cavity leakage, then it was noted and copied to both areas. After this initial binning process, the items were collected under each area of interest to be analyzed as a group. Each item was also uniquely identified and gathered into a table listing all the items for each of the four areas.

Table 3-1 Data Acquisition

<table>
<tr><th colspan="2">Data Source</th><th>Number of Items</th><th>Potential Source of Information</th><th>1st Screen Reject</th><th>Bin</th></tr>
<tr><td colspan="6">LERs</td></tr>
<tr><td rowspan="2">Licensee Event Reports (10 CFR 50.73)</td><td>1997–2003</td><td>5,299</td><td>271</td><td rowspan="2">591</td><td rowspan="2">82</td></tr>
<tr><td>2004–2009</td><td>2,104</td><td>402</td></tr>
<tr><td colspan="6">GENERIC COMMUNICATIONS</td></tr>
<tr><td colspan="2">NRC Administrative Letters</td><td>19</td><td>0</td><td>0</td><td>0</td></tr>
<tr><td colspan="2">NRC Generic Letters</td><td>21</td><td>1</td><td>1</td><td>0</td></tr>
<tr><td colspan="2">NRC Regulatory Issue Summaries</td><td>265</td><td>3</td><td>0</td><td>3</td></tr>
<tr><td colspan="2">NRC Bulletins</td><td>13</td><td>4</td><td>0</td><td>4</td></tr>
<tr><td colspan="2">NRC Information Notices</td><td>473</td><td>26</td><td>9</td><td>17</td></tr>
<tr><td colspan="6">NRC REPORTS</td></tr>
<tr><td colspan="2">10 CFR Part 21 Reports</td><td>544</td><td>1</td><td>1</td><td>0</td></tr>
<tr><td colspan="2">NRC Preliminary Notification Reports</td><td>487</td><td rowspan="3">994 (Sample Search Set Results)</td><td>24</td><td>1</td></tr>
<tr><td colspan="2">10 CFR 50.72 Reports</td><td>10,541</td><td>62</td><td>7</td></tr>
<tr><td colspan="2">NRC Daily Reports</td><td>1,410</td><td>5</td><td>0</td></tr>
<tr><td colspan="6">NRC GUIDES</td></tr>
<tr><td colspan="2">NRC Regulatory Guides</td><td>215</td><td>8</td><td>0</td><td>8</td></tr>
<tr><td colspan="2">NRC License Renewal Interim Staff Guidance</td><td>9 + Multiple</td><td>2</td><td>1</td><td>1</td></tr>
<tr><td colspan="6">NRC PUBLICATIONS</td></tr>
<tr><td colspan="2">NRC Staff Reports (NUREGs)</td><td>81</td><td>1</td><td>0</td><td>1</td></tr>
<tr><td colspan="2">NRC Contractor Reports (NUREG/CRs)</td><td>75</td><td>8</td><td>7</td><td>1</td></tr>
<tr><td colspan="6">LICENSEE INSPECTION REPORTS</td></tr>
<tr><td colspan="2">Selected Licensee Inspection Reports</td><td>268</td><td>92</td><td>5</td><td>87</td></tr>
<tr><td colspan="6">LICENSE RENEWAL APPLICATIONS</td></tr>
<tr><td colspan="2">License Renewal Applications (LRAs)</td><td>46</td><td>46</td><td>6</td><td>40</td></tr>
<tr><td rowspan="2">NRC Safety Evaluation Reports (SERs) of LRAs</td><td>Completed LRA Review</td><td>86</td><td>86</td><td rowspan="2">24</td><td rowspan="2">83</td></tr>
<tr><td>LRA Review In Progress</td><td>21</td><td>21</td></tr>
</table>

4. DATA SORTING AND SELECTION FOR FURTHER ANALYSIS

4.1 Subject Matter Expert Sorting and Binning

The subject matter experts (SMEs) gathered the information listed for each individual event noted from the data sorting and binning process described in the previous chapter and began a systematic initial review of the 40 license renewal applications (LRAs), the 83 safety evaluation reports (SERs), and the 87 selected inservice inspection (ISI) reports that had not been previously reviewed. The initial review of these documents was conducted in a manner like the sorting and binning process discussed in Section 3.5 of this report. A second level of review was then conducted, in which the reports selected from the first level of review were further studied and scrutinized, and items containing information germane to the four areas of study for this report were separated into four bins, with each bin corresponding to a particular area of study (i.e., reactor refueling cavity (RFC) leakage, spent fuel pool (SFP) leakage, torus corrosion and cracking, and concrete degradation). A fifth area was created for events that were considered not to be applicable but were assessed as noted below to verify that they contained no pertinent information.

4.2 Events Discarded after Subject Matter Expert Sorting

As noted in the previous paragraph, events that were placed into the fifth area were reexamined, and, if they fell into one of the five classifications described below, they were excluded from further consideration

First, if the event was determined to be a case of either drywell or containment liner corrosion, which is not applicable in this study, it was discarded. Second, if the event was considered to be a construction error made during the nuclear power plant's (NPP's) original construction, it was discarded. Third, if an event was determined to be a design deficiency, it was discarded. Fourth, events described below that were determined not to be age-related concrete degradation were excluded from further study. Construction defects (voids, honeycombing, delaminations) and material deficiencies (wrong material, strength too low) were considered not to be applicable for this report. Additionally, such things as shrinkage cracking, design errors, liner repairs (liner replacement, steam generator replacement), moisture barrier degradation, liner coating degradation, bulging or distortion of liner plate, and penetration liner and sleeve damage were eliminated from further examination. As delineated in Chapter 5 and appendices, some of these items were considered for the SFPs, RFCs, tori, and safety-related concrete structures. Finally, physical damage to concrete for which the degradation mechanism involved a one-time event such as an impact was also discarded from additional study. The fifth and final category included events for which information was insufficient to permit a more comprehensive analysis. Specific examples of events that were placed into each of these categories are discussed below.

Eight events were determined to be a case of either drywell or containment liner corrosion; thus, they were not applicable for this study. The eight events occurred at (1) Nine Mile Point Nuclear Station, Unit 2, (2) and (3) Susquehanna Steam Electric Station, Units 1 and 2, (4) Donald C. Cook Nuclear Plant, Unit 1, (5) Dresden Nuclear Power Station, Unit 2, (6) Sequoyah Nuclear Plant, Unit 2, (7) Surry Power Station, Unit 2, and (8) Salem Nuclear Generating Station, Unit 2.

An event that was initially thought to be a case of torus corrosion at the Dresden Nuclear Power Station, Unit 3, was evaluated and determined not to warrant further examination because the resulting loss of material was deemed to be minor by the SMEs. Section 3.5.2.3.5 of the NRC SER (Ref. 114) of the LRA provided the following justification for the omission:

> In its discussion on operating experience, the applicant stated that examinations of the Dresden internal drywell accessible steel surfaces during RFOs revealed that the original coatings were acceptable other than exhibiting minor surface rust, paint flaking and discoloration. The applicant identified no significant degradation in the corrective action process records. The internal surfaces of the torus for each of the Dresden units were re-coated with an epoxy coating in the late 1980's during refuel outages D2R11 and D3R10. Surveillance of the coated torus internal surfaces during RFOs has resulted in local coating repairs. A review of past inspections of the torus shells indicates the majority of the problems have been attributed to blistering of coating in small areas, localized pitting, and mechanical damage. Since the application of the epoxy protective coating on the internal surfaces, torus wall thinning has not been an issue.

The above is an example of when both SMEs agreed that information was insufficient for the conduct of a comprehensive and thorough examination of this event.

4.3 Summary of Data Sorting Results for Further Analysis

Table 4-1 provides a breakdown of the results by NPP for each of the four areas of interest to the program (RFC leakage ("RFC"), SFP leakage ("SFP"), torus corrosion and cracking ("Torus"), and age-related concrete degradation ("Concrete")). Plants where one of the four areas of interest has occurred are indicated by a check mark. Multiple occurrences are indicated by more than one check mark. Table 4-2 provides a summary of the data sorting and analysis results. The next chapter of this report provides a more detailed summary and description of these results.

Table 4-1 Summary of Results by Nuclear Power Plant

NPP	RFC	SFP	Torus	Concrete
Beaver Valley 1				✓
Browns Ferry 2	✓			
Browns Ferry 3	✓			
Brunswick 1				✓
Cooper			✓	
Crystal River 3		✓		✓✓
Davis-Besse	✓	✓		✓✓
Diablo Canyon 1		✓		✓
Diablo Canyon 2		✓		✓
Duane Arnold		✓	✓	
FitzPatrick			✓	
Hope Creek	✓	✓	✓	
Indian Point 2	✓	✓		
Kewaunee	✓	✓		✓✓
Monticello				✓✓
Nine Mile Point 1			✓	✓
Nine Mile Point 2				✓
Oconee				✓
Oyster Creek	✓		✓	✓
Peach Bottom 2				✓
Peach Bottom 3				✓
Prairie Island 1	✓			
Prairie Island 2	✓			
Palo Verde 1		✓		
Pilgrim			✓	
Salem 1	✓	✓		✓
Salem 2	✓	✓		✓
Seabrook		✓		✓
Three Mile Island 1				✓
Turkey Point 3				✓
Vogtle 1				✓
Vogtle 2				✓✓

Table 4-2 Summary of Data Sorting and Analysis Results

Category	Reactor Refueling Cavity Leakage	Spent Fuel Pool Leakage	Torus Corrosion and Cracking	Age-Related Concrete Degradation
No. of Occurrences	11	12	7	26
No. of NPPs	11	12	7	21
No. of Sites	8	10	7	16
No. of Case Studies*	7	9	7	0
Appendix Location*	A	B	C	N/A

* Several case studies describing occurrences related to reactor refueling cavity leakage, SFP leakage, and torus corrosion and cracking are provided as appendices to this report.

5. PRESENTATION FORMAT AND SUMMARY OF DEGRADATION OCCURRENCES

5.1 Format for Summary Presentation of Results

Following data sorting and binning of information related to each of the four areas of interest to this program, a series of tables was generated to summarize the results obtained for each area. Each of these tables was formatted according to the following descriptions.

NPP — This stands for the nuclear power plant (NPP) name and unit number and follows the naming convention used in NUREG-1350, "2010–2011 Information Digest," Volume 23, issued August 2011 (Ref. 1). The plant name has been shortened for the sake of brevity when there will be no confusion.

Plant Type — This stands for the NPP type, boiling-water reactor (BWR) or pressurized-water reactor (PWR), in conformance with NUREG-1350, Volume 23 (Ref. 1).

Occurrence Date

This corresponds to the year of the occurrence or event. A date range may also be displayed (e.g., 2001–2005).

Structure, Component, Material, Aging Effects, and Aging Mechanism

The Degradation Occurrence Database (DOD) compiled by Brookhaven National Laboratory (BNL) in NUREG/CR-6679, "Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants," issued August 2000 (Ref. 11), and mentioned in Section 3.1 of this report was analyzed, and the findings were presented in a table called the Degradation Occurrence Table (DOT). It was constructed with 13 columns; the headers for each of the columns are defined in NUREG/CR-6679. Since the BNL study was published in 2000, much effort has gone into creating a standardized approach to aging management. The results, NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," first issued July 2001 (Ref. 31), and NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," first issued July 2001 (Ref. 32) (i.e., the standard review plan (SRP) on aging effects and mechanisms for structures and containments and the GALL Report), essentially standardized the examination of NPPs applying for a license renewal application (LRA). This current study adapted the DOT column header definitions, with adjustments to accurately reflect the types of degradation of concrete and steel structures currently known, while still maintaining consistency with NUREG-1800 and NUREG-1801. For example, "system" in the DOT became "structure" for this study, and, similarly, "sub component" became "material." Thereby, the continuity between the DOD and DOT of NUREG/CR-6679 and this study is preserved.

NUREG-1800 identifies the various aging effects and mechanisms that apply to structures and containments of NPPs as part of the SRP for LRAs for NPPs.

Structure—NUREG-1801 indicates that there are three elements of concrete containments (both reinforced and post-tensioned): concrete, steel, and prestressing systems. Steel containments have two elements: steel and concrete. BWR Mark I, II, and III containments have both steel and concrete elements.

Component—NUREG-1801 identifies in Chapter II the various components found in both steel and concrete containment structures.

Material—NUREG-1801 defines the materials of construction in Chapter IX.C.

Aging Effects—Chapter IX.E of NUREG-1801 explains the selected usage of many of the standardized aging effects due to associated aging mechanisms defined in the GALL Report. Section 3.5 of NUREG-1800 details those areas of aging management review of LRAs in support of the GALL Report.

Aging Mechanisms—Chapter IX.F of NUREG-1801 indicates that "An aging mechanism is considered to be significant when it may result in aging effects that produce a loss of functionality of a component or structure…if allowed to continue without mitigation." Chapter IX.F lists definitions of the various mechanisms associated with the GALL Report. Section 3.5 of NUREG-1800 addresses those areas of aging management review and aging mechanisms for LRAs in support of the GALL Report.

Environment Chapter IX.D of NUREG-1801 defines the standardized environments used in the GALL Report.

Source Document Type

This is a generalized compilation of the types of information sources used to determine the structure, component, material, aging effect, and aging mechanism for the degradation occurrences listed in the table. When sufficient information was provided in the source documents for a specific degradation occurrence, a case study was prepared. Section 5.3 provides more information on the case studies that have been included as appendices to this report. Specific source documents (i.e., references) used to prepare each of the case studies is provided in the appropriate appendix.

5.2 Degradation Occurrence Summary

5.2.1 Leakage from the Reactor Refueling Cavity

The sorting and binning process by the subject matter experts (SMEs) found 11 separate occurrences of leakage from the reactor refueling cavity at 11 different NPPs. These NPPs are located at eight different sites throughout the United States. Table 5-1 summarizes the

11 individual occurrences of leakage from the reactor refueling cavity. Case studies for units located at the seven of the sites are provided in Appendix A.

5.2.2 Spent Fuel Pool Leakage

The sorting and binning process by the SMEs found 12 separate occurrences of spent fuel pool (SFP) leakage at 12 different NPPs. The 12 NPPs are located at 10 different sites throughout the United States. Table 5-2 summarizes the 12 individual occurrences of SFP leakage. A detailed description of nine of the degradation occurrences listed in the table is provided in the form of case studies located in Appendix B.

5.2.3 Torus Corrosion and Cracking

The sorting and binning process by the SMEs found seven separate occurrences of torus corrosion or cracking at NPPs with a BWR Mark I containment. The seven NPPs involved are located at seven different sites throughout the United States. Two of the torus corrosion occurrences were at the same NPP but at different times. Table 5-3 summarizes the seven individual occurrences of torus corrosion and cracking. A detailed description of seven of the degradation occurrences listed in the table is provided in the form of case studies located in Appendix C.

5.2.4 Concrete Age-Related Degradation

The sorting and binning process by the SMEs found 26 separate occurrences of age-related degradation in safety-related reinforced concrete structures at U.S. NPPs. The 21 NPPs involved are located at 16 different sites throughout the United States. Several occurrences of degradation were discovered at the same NPP but at different times. Publicly available information sources did not provide sufficient information to permit the preparation of case studies discussing age-related degradation of the reinforced concrete. Table 5-4 lists the 26 individual occurrences of age-related degradation in reinforced concrete.

5.3 <u>Case Studies</u>

Each separate occurrence cited in Tables 5-1, 5-2, and 5-3 was investigated. Each investigation was extensive and included sources of publicly available information pertaining to the event. When the reference sources provided sufficient information about the occurrence, results were assembled into a case study. The individual summary descriptions are provided as appendices to this report. Appendix A contains case studies for leakage from the reactor refueling cavity, Appendix B case studies for SFP leakage, and Appendix C case studies for the torus corrosion and cracking in BWR Mark I containments.

Table 5-1 Degradation Occurrence Summary: Leakage from Reactor Refueling Cavity

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Browns Ferry 2 GE BWR MK I 1999	Reactor building	Biological shield wall and containment floor Metal containment	Concrete and steel reinforcing bars Carbon steel drywell shell	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength Steel—Loss of material	Concrete—Aggressive chemical attack Steel—General and pitting corrosion and corrosion of embedded steel	Treated water	LRA LRA SER Information notice Licensee response to NRC LRA RAIs
Browns Ferry 3 GE BWR MK I 1998	Reactor building	Biological shield wall and containment floor Metal containment	Concrete and steel reinforcing bars Carbon steel drywell shell	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength Steel—Loss of material	Concrete—Aggressive chemical attack Steel—General and pitting corrosion and corrosion of embedded steel	Treated water	LRA LRA SER Information Notice Licensee response to NRC LRA RAIs

Table 5-1 Degradation Occurrence Summary: Leakage from Reactor Refueling Cavity (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Hope Creek GE BWR/4 MK I 2009	Reactor building	Shield wall	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated water	LRA LRA SER Licensee response to NRC LRA RAIs ACRS transcript
		Metal containment	Carbon steel drywell shell	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		
Indian Point 2 West PWR 2002–2004	Containment building	Reactor refueling cavity walls	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA LRA SER Licensee response to NRC LRA RAIs
	Containment	Liner	Carbon steel	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		

Table 5-1 Degradation Occurrence Summary: Leakage from Reactor Refueling Cavity (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Kewaunee West PWR 2003	Shield building	Reactor refueling cavity walls	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength Steel—Loss of material	Concrete—Aggressive chemical attack Steel—General and pitting corrosion and corrosion of embedded steel	Treated borated water	LRA LRA SER Licensee response to NRC LRA RAIs
Oyster Creek GE BWR MK I 2005	Reactor building	Metal containment	Carbon steel drywell shell	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength Steel—Loss of material	Concrete—Aggressive chemical attack Steel—General and pitting corrosion and corrosion of embedded steel	Treated water	LRA Licensee response to NRC LRA RAIs LRA SER

Table 5-1 Degradation Occurrence Summary: Leakage from Reactor Refueling Cavity (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Prairie Island 1 West PWR 1998	Containment	Containment walls and floors	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA LRA SER
		Metal Containment	Carbon steel	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee engineering analyses Licensee response to NRC LRA RAIs
Prairie Island 2 West PWR 1998	Containment	Containment walls and floors	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA LRA SERs
		Metal Containment	Carbon steel	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee engineering analyses Licensee response to NRC LRA RAIs

Table 5-1 Degradation Occurrence Summary: Leakage from Reactor Refueling Cavity (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Salem 1 West PWR 2005	Containment building	Reactor cavity walls and floor and containment floor	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA LRA SER
		Liner	Carbon steel	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee response to NRC LRA RAIs
Salem 2 West PWR 2000	Containment building	Reactor cavity walls and floor and containment floor	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA LRA SER
		Liner	Carbon steel	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee response to NRC LRA RAIs

Table 5-1 Degradation Occurrence Summary: Leakage from Reactor Refueling Cavity (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Davis-Besse 1 B&W PWR 2000–2009	Unit 1 containment building	East/West and incore instrumentation tunnels	Concrete	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA
							LRA SER
			Steel, steel reinforcing bars	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee response to NRC LRA RAIs

Table 5-2 Degradation Occurrence Summary: Spent Fuel Pool Leakage

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Palo Verde 1 CE PWR 2005	Unit 1 fuel building	SFP	Concrete Steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength Steel—Loss of material	Concrete—Aggressive chemical attack Steel—General and pitting corrosion and corrosion of embedded steel	Treated borated water	LRA LRA SER NRC inspection report Licensee response to NRC LRA RAIs
Crystal River 3 B&W PWR 2007	Unit 3 auxiliary building	SFP floor and walls	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength Steel—Loss of material	Concrete—aggressive chemical attack Steel—General and pitting corrosion and corrosion of embedded steel	Treated borated water	LRA LRA SER Licensee response to NRC LRA RAIs

Table 5-2 Degradation Occurrence Summary: Spent Fuel Pool Leakage (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Davis-Besse 1 B&W PWR 2000–2009	Unit 1 auxiliary building	SFP floor and walls	Concrete	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA
							LRA SER
			Steel reinforcing bars	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee response to NRC LRA RAIs
Diablo Canyon 1 B&W PWR 2010	Unit 1 fuel-handling building	SFP floor and walls	Concrete	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA
							LRA SER
			Steel reinforcing bars	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee response to NRC LRA RAIs

Table 5-2 Degradation Occurrence Summary: Spent Fuel Pool Leakage (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Diablo Canyon 2 West PWR 2010	Unit 2 fuel-handling building	SFP floor and walls	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength Steel—Loss of material	Concrete—Aggressive chemical attack Steel—General and pitting corrosion and corrosion of embedded steel	Treated borated water	LRA LRA SER Licensee response to NRC LRA RAIs
Duane Arnold GE BWR/4 MK I 1994–2009	Reactor building	SFP	Concrete and steel reinforcing bars	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength Steel—Loss of material	Concrete—Aggressive chemical attack Steel—General and pitting corrosion and corrosion of embedded steel	Treated water	LRA LRA SER Licensee response to NRC LRA RAIs

Table 5-2 Degradation Occurrence Summary: Spent Fuel Pool Leakage (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Hope Creek GE BWR/4 MK I 2009	Reactor building	SFP floor and walls	Concrete	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated water	LRA LRA SER Licensee response to NRC LRA RAIs
			Steel reinforcing bars	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		
Indian Point 2 West PWR 2005	Unit 2 fuel-handling building	SFP floor and walls	Concrete	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA Task force report NRC SIT charter NRC letters NRC press releases NRC inspection report LRA SER Licensee response to NRC LRA RAIs
			Steel reinforcing bars	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		

Table 5-2 Degradation Occurrence Summary: Spent Fuel Pool Leakage (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Kewaunee West PWR 2007	Auxiliary building	Reinforced concrete walls and ceiling for drumming room	Concrete	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA LRA SER
			Steel reinforcing bars	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee response to NRC LRA RAIs
Salem 1 West PWR 2002	Unit 1 fuel-handling and auxiliary buildings	Interior concrete walls for Unit 1 fuel-handling and auxiliary buildings	Concrete	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	Information notice NRC inspection report LRA
			Steel reinforcing bars	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		LRA SER Licensee response to NRC LRA RAIs

Table 5-2 Degradation Occurrence Summary: Spent Fuel Pool Leakage (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Salem 2 West PWR 2010	Unit 1 fuel-handling and auxiliary buildings	Interior concrete walls for Unit 1 fuel-handling and auxiliary buildings	Concrete	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA
							LRA SER
			Steel reinforcing bars	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee response to NRC LRA RAIs
Seabrook Station West PWR 2011	Fuel-handling building and auxiliary buildings	Interior concrete walls for fuel-handling and auxiliary buildings	Concrete	Concrete—Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Concrete—Aggressive chemical attack	Treated borated water	LRA
							LRA SER
			Steel reinforcing bars	Steel—Loss of material	Steel—General and pitting corrosion and corrosion of embedded steel		Licensee response to NRC LRA RAIs

Table 5-3 Degradation Occurrence Summary: Torus Corrosion and Cracking

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Cooper GE BWR/4 MK I 2009	Suppression chamber (torus)	Steel elements	Carbon steel	Loss of material due to general, pitting, and crevice corrosion	Crevice corrosion	Treated water	LRA
					General corrosion		Licensee response to NRC LRA RAIs
					Pitting corrosion		LRA SER
				Cracking due to cyclic loading	Fatigue		NPPD licensee letters
Duane Arnold GE BWR/4 MK I 2009	Suppression chamber (torus)	Steel elements	Carbon steel	Loss of material due to general, pitting, and crevice corrosion	Crevice corrosion	Treated water	LRA
					General corrosion		Licensee response to NRC LRA RAIs
					Pitting corrosion		Generic letter
				Cracking due to cyclic loading	Fatigue		LRA SER

Table 5-3 Degradation Occurrence Summary: Torus Corrosion and Cracking (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
FitzPatrick GE BWR/4 MK I 2005	Suppression chamber (torus)	Steel elements	Carbon steel	Loss of material due to general, pitting, and crevice corrosion Cracking due to cyclic loading	Crevice corrosion General corrosion Pitting corrosion Fatigue	Treated water	LRA LER Preliminary notification report Event notification LRA SER Information notice
Hope Creek GE BWR/4 MK I 2004	Suppression chamber (torus)	Steel elements	Carbon steel	Loss of material due to general, pitting, and crevice corrosion Cracking due to cyclic loading	Crevice corrosion General corrosion Pitting corrosion Fatigue	Treated water	LRA LRA SER Licensee response to NRC LRA RAIs

Table 5-3 Degradation Occurrence Summary: Torus Corrosion and Cracking (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Nine Mile Point 1 GE BWR/2 MK I 2001	Suppression chamber (torus)	Steel elements	Carbon steel	Loss of material due to general, pitting, and crevice corrosion Cracking due to cyclic loading	Crevice corrosion General corrosion Pitting corrosion Fatigue	Treated water	LRA LRA SER OAR
Oyster Creek GE BWR/2 MK I 2002	Suppression chamber (torus)	Steel elements	Carbon steel	Loss of material due to general, pitting, and crevice corrosion Cracking due to cyclic loading	Crevice corrosion General corrosion Pitting corrosion Fatigue	Treated water	LRA LRA SER

Table 5-3 Degradation Occurrence Summary: Torus Corrosion and Cracking (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Pilgrim GE BWR/3 MK I 1999 and 2003	Suppression chamber (torus)	Steel elements	Carbon steel	Loss of material due to general, pitting, and crevice corrosion	Crevice corrosion	Treated water	LRA
					General corrosion		LRA SER
					Pitting corrosion		Licensee response to NRC LRA RAIs
				Cracking due to cyclic loading	Fatigue		

Table 5-4 Degradation Occurrence Summary: Concrete Age-Related Degradation

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Bellefonte 1 B&W PWR 2009	Containment	Coupling—rock anchor/tendon anchor	Steel	Loss of prestress	Hydrogen-induced SCC	Raw water	Event notification Licensee letters to NRC
Brunswick 1 GE BWR/4 MK I 2004	Service water buildings for Units 1 and 2	Interior and exterior surfaces	Concrete	Cracking and spalling	Aggressive chemical attack	Raw water	LRA LRA SER ISI report
Crystal River 3 B&W PWR 2003	Containment	Tendon gallery walls	Concrete	Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Aggressive chemical attack Leaching of calcium hydroxide	Ground water/ soil	LRA LRA SER
Crystal River 3 B&W PWR 2007	Containment	Tendons	Steel	Loss of prestress	Stress relaxation	Concrete	LRA LRA SER

Table 5-4 Degradation Occurrence Summary: Concrete Age-Related Degradation (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Davis-Besse 1 B&W PWR 2010	Unknown building and switchyard	Auxiliary feedwater pump turbine exhaust missile barrier Tower and disconnect switch concrete foundations	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Aggressive chemical attack	Raw water	LRA
Diablo Canyon 1 West PWR 2010	Intake and discharge structures	Exterior concrete surfaces	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Aggressive chemical attack	Raw water	LRA Licensee response to NRC LRA RAIs
Diablo Canyon 2 West PWR 2010	Intake and discharge structures	Exterior concrete surfaces	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Aggressive chemical attack	Raw water	LRA Licensee response to NRC LRA RAIs
Kewaunee West PWR 1997	Screen house and tunnel	Walls	Concrete	Cracking	Leaching of calcium hydroxide	Raw water	LRA LRA SER Licensee response to NRC LRA RAIs

Table 5-4 Degradation Occurrence Summary: Concrete Age-Related Degradation (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Kewaunee West PWR 2003	Not reported	Walls	Concrete	Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Aggressive chemical attack Leaching of calcium hydroxide	Ground water/soil	LRA LRA SER Licensee response to NRC LRA RAIs
Monticello GE BWR/3 MK I 1998	Not reported	Not reported	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Corrosion of embedded steel	Not reported	LRA LRA SER
Monticello GE BWR/3 MK I 2001/2002	Not reported	Not reported	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Corrosion of embedded steel	Not reported	LRA LRA SER
Nine Mile Point 1 BWR MK I 2005	Service water pipe tunnel	Walls	Concrete	Cracking	Service-induced cracking or other aging mechanisms	Ground water/soil	LRA LRA SER Licensee response to NRC LRA RAIs

Table 5-4 Degradation Occurrence Summary: Concrete Age-Related Degradation (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Nine Mile Point 2 GE BWR/5 MK II 2005	Service water pipe tunnel	Walls	Concrete	Cracking	Service-induced cracking or other aging mechanisms	Ground water/soil	LRA LRA SER Licensee response to NRC LRA RAIs
Oconee B&W PWR 1998	Containment	Concrete beneath tendon bearing plate	Concrete	Concrete cracking and spalling	Service-induced cracking or other concrete aging mechanisms	Air—indoor uncontrolled	Information notice LRA LRA SER
Oyster Creek GE BWR/2 MK I 2001	Containment	Drywell shield wall	Concrete	Cracking	Elevated temperature	Air—indoor uncontrolled	LRA LRA SER
Peach Bottom 2 GE BWR/4 MK I 2001	Emergency cooling tower and reservoir	Walls	Concrete	Cracking	Leaching of calcium hydroxide	Raw water	LRA LRA SER Licensee response to NRC LRA RAIs

Table 5-4 Degradation Occurrence Summary: Concrete Age-Related Degradation (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Peach Bottom 3 GE BWR/4 MK I 2001	Emergency cooling tower and reservoir	Walls	Concrete	Cracking	Leaching of calcium hydroxide	Raw water	LRA LRA SER Licensee response to NRC LRA RAIs
Salem 1 PWR 2010	Service water intake structure	Walls	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Aggressive chemical attack Corrosion of embedded steel	Raw water	LRA LRA SER Licensee response to NRC LRA RAIs
Salem 2 West PWR 2010	Service water intake structure	Walls	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Aggressive chemical attack Corrosion of embedded steel	Raw water	LRA LRA SER Licensee response to NRC LRA RAIs

Table 5-4 Degradation Occurrence Summary: Concrete Age-Related Degradation (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Seabrook West PWR 2006	Below-grade concrete structures	Walls and floor slabs	Concrete	Increase in porosity and permeability, cracking, loss of material (spalling, scaling), and loss of strength	Aggressive chemical attack Leaching of calcium hydroxide	Ground water/soil	LRA
TMI 1 B&W PWR 2004	Containment	Dome and ring girder	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Leaching of calcium hydroxide Corrosion	Air—outdoor	LRA LRA SER
Turkey Point 3 West PWR 2007	Containment	Buttress	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Not reported	Air—outdoor	ISI report
Vogtle 1 West PWR 1/07/1998 and 1998	Diesel generator building	Exhaust pipe concrete barriers	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Service-induced cracking or other concrete aging mechanisms	Air—indoor uncontrolled	LER LRA LRA SER

Table 5-4 Degradation Occurrence Summary: Concrete Age-Related Degradation (cont.)

NPP Plant Type Occurrence Date	Structure	Component	Material	Aging Effects	Aging Mechanism	Environment	Source Document Type
Vogtle 2 West PWR 1998	Diesel generator building	Exhaust pipe concrete barriers	Concrete and steel reinforcing bars	Cracking, loss of bond, and loss of material (spalling, scaling)	Service-induced cracking or other concrete aging mechanisms	Air—indoor uncontrolled	LER LRA LRA SER
Vogtle 2 West PWR 1999	Containment	Buttress	Concrete	Concrete cracking and spalling	Not reported	Air—outdoor	ISI report

6. REVIEW OF DEGRADATION OCCURRENCES

6.1 Introduction

Detailed descriptions of the areas related to degradation occurrences associated with the specific topics of interest to this study (i.e., reactor refueling cavity leakage, spent fuel pool (SFP) leakage, and torus corrosion and cracking) are provided in Appendices A, B, and C, respectively. The descriptions for each of these topics of interest were reviewed, and the results are summarized and presented here with respect to the specific areas of interest. Specific references pertaining to information summarized in this chapter are provided in the appropriate appendix. Finally, this chapter provides a review of age-related degradation of concrete structures. Occasionally, some information applies to more than one topic of interest and the subject matter expert has included it where most appropriate.

6.2 Leakage from the Reactor Refueling Cavity

In a pressurized-water reactor (PWR), the reactor refueling cavity is a reinforced concrete structure that extends from the reactor cavity to the outlet of the fuel transfer tube at the containment building cylindrical wall. The cavity internal surfaces are typically lined with stainless steel plate on the order of 6.35 millimeters (mm) (0.25 inches) thick. A leak chase system embedded in the walls and slabs collects leakage through the seam welds of the liner. The reactor refueling cavity is filled with borated water during refueling to permit underwater transport of fuel elements between the SFP and the reactor vessel through the reactor refueling cavity and the fuel transfer tube. The discharge end of the fuel transfer tube into the fuel transfer canal is sealed with a gasketed flange during plant operation to form the containment pressure boundary. The fuel transfer cavity can be isolated from the spent fuel storage pool and drained by using a fuel pool gate for maintenance of the fuel-handling system. The gate seal consists of short-lived inflatable seals supplied with air by the compressed air system. The cavity is drained after refueling and maintained dry during plant operation. The reactor refueling cavity pool can be on the order of 3.7 meters (m) by 8.8 m in plan by 14 m deep (12 feet by 29 feet by 46 feet). Primary age-related degradation mechanisms that can produce leakage through the stainless steel liner include fatigue, intergranular stress-corrosion cracking, and crevice corrosion. Leakage can also occur as a result of weld defects, blockage of the leakage collection system by precipitation of reaction products or foreign matter, damage to the liner, or a breakdown of the gasketed seal. Leakage primarily will occur only during refueling and could potentially result in erosion of the concrete due to the mildly acidic nature of the borated water, or in corrosion of any carbon steel component it contacts (e.g., the metallic pressure boundary or concrete steel reinforcement). Leakage that is not captured by the monitoring system can migrate through construction joints and cracks in the concrete.

For boiling-water reactor (BWR) plants, the reactor refueling cavity is located above the reactor pressure vessel and adjacent to the SFP, as illustrated in Figure 6-1. The reactor refueling cavity is constructed of reinforced concrete, with its inner surface lined with stainless steel to provide a leak-tight barrier. Potential causes of leakage from the reactor refueling cavity are cracking of the liner or welds due to fatigue resulting from periodic filling of the reactor refueling cavity and equipment pool during refueling, intergranular stress-corrosion cracking, crevice corrosion, or a breakdown of the drywell-to-cavity seal. Leakage of water initiates during refueling operations (Ref. 115). Figure 6-2 provides an example of a cross section of the drywell-to-cavity seal in a BWR Mark (MK) I and identifies one potential leakage path.

As already discussed, leakage from the reactor refueling cavity primarily occurs only during refueling operations when the cavity is flooded, at an estimated flow rate that has ranged from a drop or less per minute to on the order of 1.9 to 4.4×10^{-4} cubic meters per second (m^3/s) (3 to 7 gallons per minute). The leakage may continue a few days after draining the cavity and has been identified at the plants listed previously in Table 5-1. The leakage was primarily found by visual inspections that identified the presence of fluid in the gap between the drywell and concrete shield (BWR), mineral deposits on surfaces of structures (BWR and PWR), or at sumps (PWR).

Sufficient information was available in publicly available sources to prepare case studies for seven of the occurrences of reactor refueling cavity leakage. The case studies are provided in Appendix A. Sufficient information was not available to develop a statistically significant trend indicating that reactor refueling cavity leakage was more prevalent in BWRs or PWRs, or for a particular nuclear steam supply system supplier, engineering firm, or contractor. Table 6-1 provides a summary of results related to leakage from the reactor refueling cavity based on the information contained in Appendix A. Specific topics addressed in the table include the leakage source, primary structures potentially impacted, how leakage was identified, activities to address leakage, and assessment of structural impact.

6.3 Spent Fuel Pool Leakage

The fuel storage facility provides for receiving, storing, shielding, shipping, and handling of new and spent fuel. The fuel storage facility is an integral part of the reactor building of most BWR MK I and MK II containment plant types. For BWR MK III containment types and many PWRs, the fuel storage facility is a separate structure; however, in some plants the fuel storage facility is part of the auxiliary building (Figure 6-3). The fuel storage structure is a multistory reinforced concrete structure supported on bearing walls and/or a basemat. The SFP has reinforced concrete walls and floors for radiation shielding and a stainless steel liner that provides a leak-tight barrier. A typical SFP for a PWR can be on the order of about 12 m (40 feet) deep and 12 m (40 feet) or more in each horizontal direction. The SFP walls are constructed of reinforced concrete typically having a thickness between 0.7 and 3 m (2 and 10 feet). Cover concrete for the embedded steel reinforcement is on the order of 3.8 to 7.6 centimeters (cm) (1.5 to 3 inches). The inside surfaces of the SFP are typically lined by stainless steel plates, having a thickness of about 6 to 13 mm (0.25 to 0.5 inches), that are joined by full-penetration seam welds. Between the seams, the liner plates may also be plug welded to studs embedded in the concrete. Some plants used the liner as part of the concrete form for the fuel-handling building, and other plants “wallpapered” the liner onto the completed concrete structure after the forms were removed. At least one PWR has an unlined SFP, with the concrete sealed with an epoxy. Leakage collection systems consisting of channels embedded in the concrete at the locations of the weld seams are provided to permit monitoring and collection of any SFP leakage that might occur at the seam welds. The channels are fabricated from either carbon or stainless steel that may or may not be seal welded to the back of the liner. The channels lead to a series of telltales piped to a collection system and routed to the liquid radwaste system. SFPs for BWR plants are filled with demineralized water, whereas those for PWRs contain borated water (e.g., 2,200 to 2,400 parts per million boron, pH ~4.8). Primary age-related degradation mechanisms that can produce leakage through the stainless steel SFP liner include intergranular stress-corrosion cracking and crevice corrosion. Leakage can also occur as a result of seam or plug weld defects, blockage of the leakage collection system by precipitation of reaction products or foreign material, and damage to the liner. Leakage that is not captured by the monitoring system can migrate through construction joints and cracks in the concrete to result in release of contaminated water to the environment. Figure 6-4 provides an example cross section of a leak chase system and a postulated path for leakage occurrence. With respect to leakage that bypasses the leak collection system of PWR SFPs, there is an added concern because leakage of the borated water over an extended period of time could potentially result in erosion of the concrete due to the mildly acidic nature of the borated water, or corrosion of any carbon steel it contacts (e.g., metallic pressure boundary or concrete steel reinforcement). Often, exterior walls of the SFP are not accessible; therefore, they cannot be monitored for visible signs of degradation.

Leakage of fluid from the SFP can be a continual process, since it remains filled at all times. Leakage has been identified at the plants listed previously in Table 5-2, with reported flow rates ranging from a drop or less per minute to on the order of 4.3×10^{-6} m^3/s (100 gallons per day (gpd)). The leakage was primarily found from drainage through the telltale drains of the leak chase system, fluids emerging from cracks in reinforced concrete walls of the SFP, or the presence of mineral deposits on structures, the presence of fluid in the seismic gap between the fuel-handling building and the auxiliary building, contamination of protective shoe covers, and the presence of tritium in ground water.

Sufficient information was available in publicly available sources to prepare case studies for nine of the occurrences of SFP leakage. The case studies are provided in Appendix B. Although SFP leakage was more common for PWRs than BWRs, sufficient information was not available to develop a statistically significant trend indicating that SFP leakage was more prevalent for a particular nuclear steam supply system supplier, engineering firm, or contractor. Table 6-2 provides a summary of results related to leakage from the SFP based on the information contained in Appendix B. Specific topics addressed in the table include the leakage source, primary structures potentially impacted, how leakage was identified, activities to address leakage, and assessment of structural impact.

6.4 Torus Corrosion and Cracking

The suppression chamber (or wetwell) of a BWR MK I nuclear power plant (NPP) typically consists of 16 to 20 mitered cylindrical carbon steel (or carbon steel clad with stainless steel) segments joined together to form the shape of a torus that encircles the drywell (Figure 6-1). The torus is normally about half-full of demineralized water, which constitutes the pressure-suppression pool. The torus cross section can have a major diameter of between 29 and 34 m (95 to 112 feet), a minor diameter between 7.6 and 9.4 m (25 and 31 feet), and a shell thickness from 9.53 to 25.4 mm (0.375 to 1 inch). The suppression chamber is penetrated by access hatches as well as by vent lines that connect it to the drywell. The inside surfaces of the suppression chamber exposed to water are typically coated with an inorganic zinc primer with a modified phenolic or epoxy-based top coating. The exterior surfaces of the suppression pool of some plants have a lead-based coating. Support for the suppression chamber is provided by columns on the inside and outside diameter and by a saddle under the torus (Figure 6-5). The columns are rigidly supported at the torus shell and connected with rotating pins at the base to allow rotation from thermal expansion of the torus. The torus is anchored to and supported by the reinforced concrete foundation slab of the reactor building. Potential degradation mechanisms for the suppression chamber in areas that are uncoated, or that are not clad by stainless steel, include loss of material due to general atmospheric corrosion and localized corrosion (e.g., crevice, differential aeration, galvanic, microbiologically induced, and pitting), fatigue, and stress-corrosion cracking of stainless steel. Coating degradation can result from several types of stressors: temperature, condensation or immersion, radiation, stress-induced physical damage, and damage resulting from corrosion of the base metal.

Once the torus coating begins to degrade, the steel substrate is susceptible to moisture that can lead to corrosion. Licensees typically perform periodic visual inspections of the suppression steel shells by either draining the torus and inspecting it under dry conditions or employing divers or using cameras to inspect the submerged surfaces. Degradation of tori coatings requiring cleaning and recoating has also been identified (Ref. 22). Measured corrosion rates in some torus shells have exceeded the design corrosion rate, and plants have identified areas of the torus shell in which the thickness was at or below the minimum specified wall thickness (Ref. 22).

Sufficient information was available in publicly available sources to prepare case studies for seven of the occurrences of torus corrosion and cracking. The case studies are provided in Appendix C. Although several occurrences of torus corrosion and cracking were identified, sufficient information was not available to develop a statistically significant trend indicating that the torus corrosion and cracking was more prevalent for a particular nuclear steam supply system supplier, engineering firm, or contractor. Table 6-3 provides a summary of results related to torus corrosion and cracking based on the information contained in Appendix C. Specific topics addressed in the table include inspection methods, field observations, corrective and aging management actions, and structural integrity assessments.

6.5 Age-Related Degradation of Concrete Structures

All commercial NPPs in the United States contain concrete structures whose performance and function are necessary for protection of the safety of plant operating personnel and the general public, as well as the environment. The basic laws that regulate the design (and construction) of NPPs are contained in Title 10 of the *Code of Federal Regulations* (10 CFR), "Energy," that is clarified by documents such as regulatory guides, NUREG reports, and standard review plans.

Concrete-based structures in light-water reactor plants provide foundation, support, shielding, and containment functions. Typically the safety-related concrete structures contained in light-water reactor plants may be grouped into four general categories: primary containments, containment internal structures, secondary containments/reactor buildings, and other structures. Only information related to primary containment structures for PWR and BWR plants is summarized below. Information on other concrete structures is provided elsewhere (Ref. 2). Safety-related concrete structures are those that are relied upon to remain functional during and following design-basis events and nonsafety-related structures whose failure could prevent satisfactory accomplishment of any of the functions as defined in 10 CFR 54.4, "Scope."

Approximately 80 percent of the PWR plants that have been licensed for commercial operation in the United States use either reinforced or prestressed concrete primary containments. The concrete containments are of three different functional designs: subatmospheric (reinforced concrete), ice condenser (reinforced concrete), and large dry (reinforced and prestressed concrete). The primary differences between these containment designs relate to volume requirements, provisions for accident loadings and pressures, and containment internal structures layout. The PWR concrete containment structure generally consists of a concrete basemat foundation, vertical cylindrical walls, and dome. Leak tightness of a containment is provided by a steel liner attached to the containment inside surfaces. Exposed surfaces of the carbon steel liner are typically painted to protect against corrosion and to facilitate decontamination, should it be required. Depending on the functional design (e.g., large dry or ice condenser), the concrete containments can be on the order of 40 to 50 m in diameter and 60 to 70 m high, with wall and dome thicknesses from 0.9 to 1.4 m, and base slab thicknesses from 2.7 to 4.1 m. Figure 6-6 presents the Trojan nuclear plant cooling tower and posttensioned concrete containment prior to decommissioning and demolition. PWR plants that use a metallic primary containment (large dry and ice condenser designs) are usually contained in reinforced concrete “enclosure” or “shield” buildings that, in addition to withstanding environmental effects, provide radiation shielding and particulate collection and ensure that the free-standing metallic primary containment is protected from the natural environment.

Of the BWR plants in the United States, approximately 30 percent use a reinforced concrete primary containment. BWR containments, because of provisions for pressure suppression, typically have “normally dry” sections (drywell) and “flooded” sections (wetwell) that are interconnected via piping or vents. BWR plants that use steel primary containments have reinforced concrete structures that serve as secondary containments or reactor buildings. The secondary containment structures generally are safety-related because they provide additional radiation shielding; provide resistance to environmental and operational loadings; and house safety-related mechanical equipment, spent fuel, and the primary metal containment. Although these structures may be massive in cross section in order to meet shielding or load-bearing requirements, they generally have smaller net sections than primary containments because of reduced exposure under postulated accident loadings.

Exposure to the environment (e.g., temperature, moisture, cyclic loadings) can produce degradation of reinforced concrete structures. The rate of deterioration is dependent on the component’s structural design, materials selection, quality of construction, curing, and aggressiveness of environmental exposure. Termination of a component’s service life occurs when it no longer can meet its functional and performance requirements.

Primary mechanisms (factors) that, under unfavorable conditions, can produce premature deterioration of reinforced concrete structures include those that impact either the concrete or steel reinforcing materials (i.e., mild steel reinforcement or posttensioning systems). Degradation of the concrete can be caused by adverse performance of either its cement-paste matrix or aggregate materials under chemical or physical attack. In nearly all chemical and physical processes influencing the durability of concrete structures, the dominant factors include the transport mechanisms within the pores and cracks and the presence of water. Degradation of mild steel reinforcing materials can occur as a result of corrosion, irradiation, elevated temperature, or fatigue effects, with corrosion being the most likely form of attack. Posttensioning systems are susceptible to the same degradation mechanisms as mild steel reinforcement, plus loss of prestressing force, primarily due to tendon relaxation, and concrete creep and shrinkage. Additional information on the durability of NPP reinforced concrete structures is available (Ref. 26).

Known problem areas associated with the degradation of reinforced concrete components in NPP applications through 1986 are described in NUREG/CR-4652, “Concrete Component Aging and Its Significance Relative to Life Extension of Nuclear Power Plants,” issued September 1986 (Ref. 116). Because the operating history of these NPPs was limited (approximately 20 years or less), many of these problem areas involved the following conditions that are not directly related to a specific concrete aging mechanism or environment:

- construction defects (voids, surface honeycombing, delaminations, reinforcing bar placement errors) and material deficiencies (wrong concrete mixture, concrete strength too low, prestressing tendon wire and button head failure)

- concrete shrinkage cracking

- design errors

In 1992, the NRC received survey responses from 29 utilities concerning the types and locations of concrete distress, the types of repairs performed, and the durability of the repairs. Survey results pertaining to the concrete structures at these plants are described in NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures," issued June 1995 (Ref. 9), and summarized below:

- Locations of Deterioration—Auxiliary building and secondary containment (shield building) walls and slabs were noted as the most common locations.

- Type of Deterioration—86 percent of the plants reported cracking; 65 percent reported spalling; and over 20 percent of the respondents reported staining, honeycombing, efflorescence, and scaling.

- Causes of Deterioration—48 percent (of the respondents) reported deterioration resulting from drying shrinkage, 31 percent from freeze-thaw, and 24 percent from abrasion.

NUREG-1522 also provided results from the inspections of six plants that were licensed prior to 1977 where the following types of degradation of the reinforced concrete structures were noted:

- tendon galleries of posttensioned concrete containments

- water leakage and seepage in underground structures

- intake tunnels

NUREG/CR-6679, "Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants," issued August 2000, presented an assessment of age-related degradation of structures and passive components for U.S. NPPs (Ref. 11). Included in this assessment was a tabulation of degradation occurrences involving concrete structures, with the findings reported in NUREG-1522 (Ref. 9). A summary of degradation occurrences reported in NUREG/CR-6679 related to NPP concrete structures is presented in Figure 6-7. However, the definitions of concrete aging effects and aging mechanisms used to describe the degradation included both aging- and nonaging-related conditions, with some of the aging mechanisms for specific degradation occurrences reported as not applicable. In addition, many of the degradation occurrences reported in the above document were based on information that either is not publicly available or is published in a format that is not electronically searchable.

The identification of occurrences of degradation in the present study concentrated on the following types of concrete structures: containments, concrete service water intake and discharge structures and service water pipe tunnels, above-grade concrete structures, and below-grade concrete structures. As a minimum, selection of the occurrences involved identification of the aging effects for a damaged structure from information in publicly available documents, followed by identification of the associated aging mechanisms and environment. Occurrences that pertain to the areas identified in Section 1.3.4 were then excluded.

Application of this methodology required interpretation of text in various types of documents to develop an understanding of the relevant circumstances associated with the degradation and identification of the root cause and contributing factors of the specific event. Engineering judgment was then applied to characterize the aging effect, aging mechanism, and environment using the standardized terminology and definitions in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," first issued July 2001 (Ref. 32). Table 6-4 provides a summary of the age-related degradation occurrences that were identified in this study as well as the age-related events that were reported in NUREG/CR-6927, "Primer on Durability of Nuclear Power Plant Reinforced Concrete Structures—A Review of Pertinent Factors," issued February 2007 (Ref. 26).

Terms and definitions established by the NRC to describe aging effects, aging mechanisms, and environments applicable to concrete structures are published in NUREG-1801 (Ref. 32). These terms and definitions are used in the present study as the basis for characterizing the age-related degradation occurrences in safety-related concrete structures. Because the terms and definitions used to describe concrete aging effects and aging mechanisms in NUREG/CR-6679 (Ref. 11) were not consistent with those in NUREG-1801, results obtained in the present study will not identify all of the occurrences listed as degradation for concrete structures reported in NUREG/CR-6679.

It is to be noted that NUREG-1801 provides the technical basis and guidance to license renewal applicants for aging management practices that have been successfully used to monitor and manage aging effects in structures, thus assuring that their integrity is maintained during the period of extended operation. The programs are continuously updated by NRC staff to handle emerging structural integrity issues through license renewal interim staff guidance, newer releases/revisions of NUREG-1801, and pertinent research documented in other supplemental NRC publications and NUREGs (see Chapters 1 and 2). A license renewal applicant may reference NUREG-1801 to demonstrate that its approach is conforming to the recommended programs and their technical bases for managing aging effects in concrete structures in the period of extended operation.

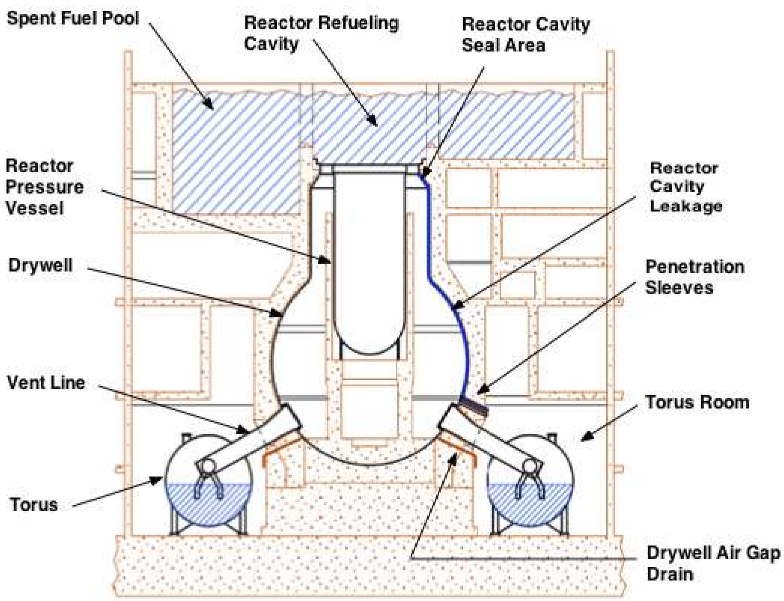

Figure 6-1 Cross section of BWR MK I showing location of reactor refueling cavity, spent fuel pool, and torus as well as one leakage path that has been observed

(Source: "Advisory Committee on Reactor Safeguards Hope Creek License Renewal Subcommittee," Official Transcript of Proceedings Nuclear Regulatory Commission, Work Order No. NRC-542, Neal R. Gross and Co., Inc., Court Reporters and Transcribers, Washington, D.C., November 3, 2010)

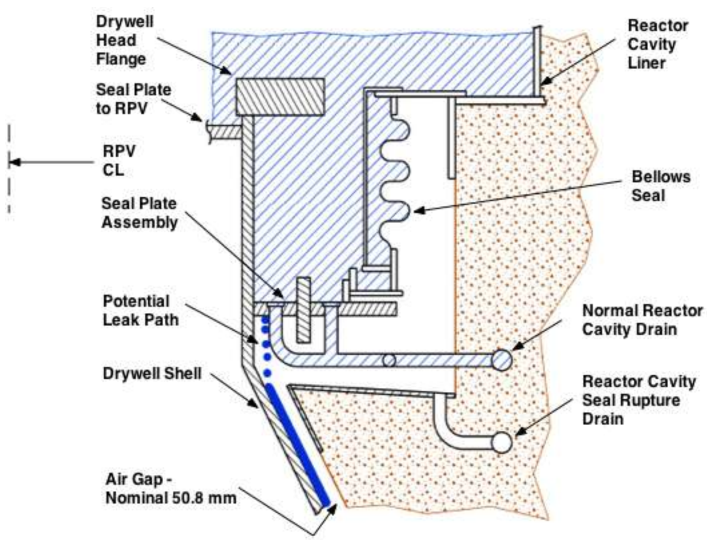

Figure 6-2 Detail of drywell—reactor cavity seal area and identification of potential leakage path

(Source: "Advisory Committee on Reactor Safeguards Hope Creek License Renewal Subcommittee," Official Transcript of Proceedings Nuclear Regulatory Commission, Work Order No. NRC-542, Neal R. Gross and Co., Inc., Court Reporters and Transcribers, Washington, D.C., November 3, 2010)

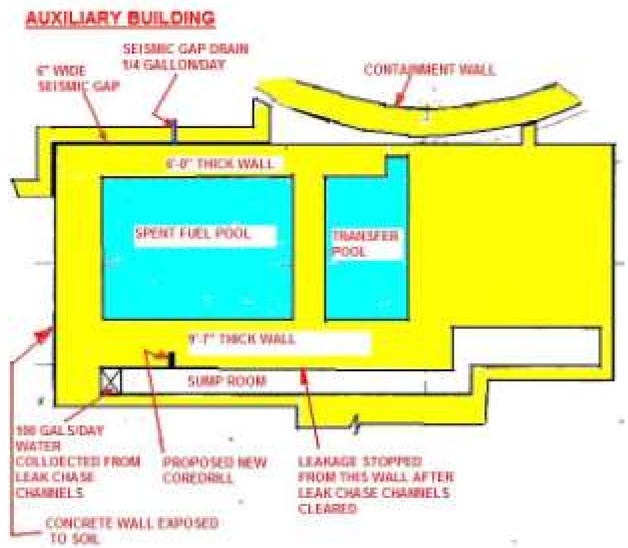

Figure 6-3 Plan view of fuel-handling building showing relative location of spent fuel pool, transfer pool, and containment building

(Source: "Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee," Official Transcript of Proceedings Nuclear Regulatory Commission, Work Order No. NRC-577, Neal R. Gross and Co., Inc., Court Reporters and Transcribers, Washington, D.C., December 1, 2010)

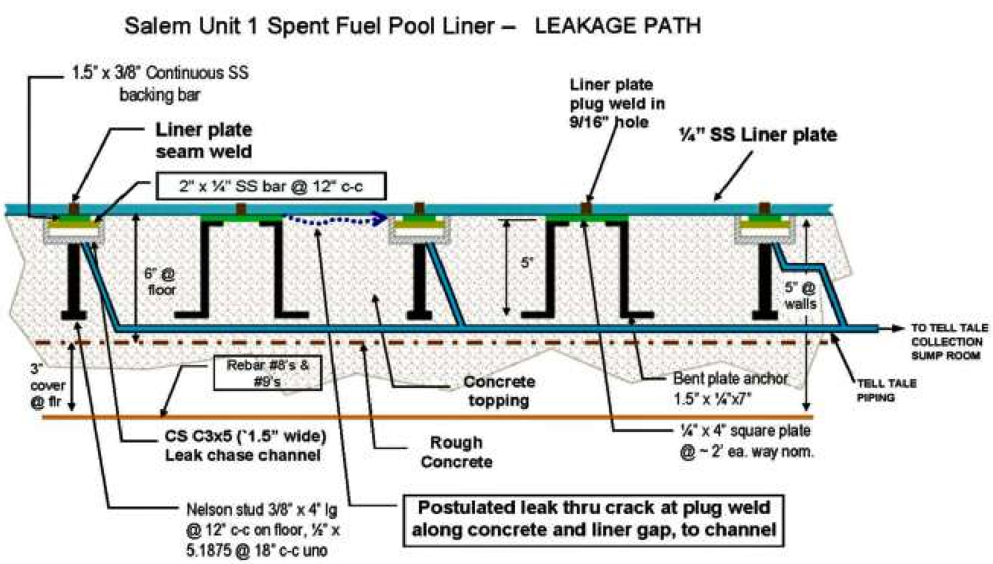

Figure 6-4 Cross section of a leak chase system for a PWR plant and a postulated path for leakage occurrence

(Source: Samuel Cuadrado de Jesús, "Advisory Committee on Reactor Safeguards (ACRS) License Renewal Full Committee Salem Nuclear Generating Station Safety Evaluation Report," U.S. Nuclear Regulatory Commission, Washington, DC, May 12, 2011)

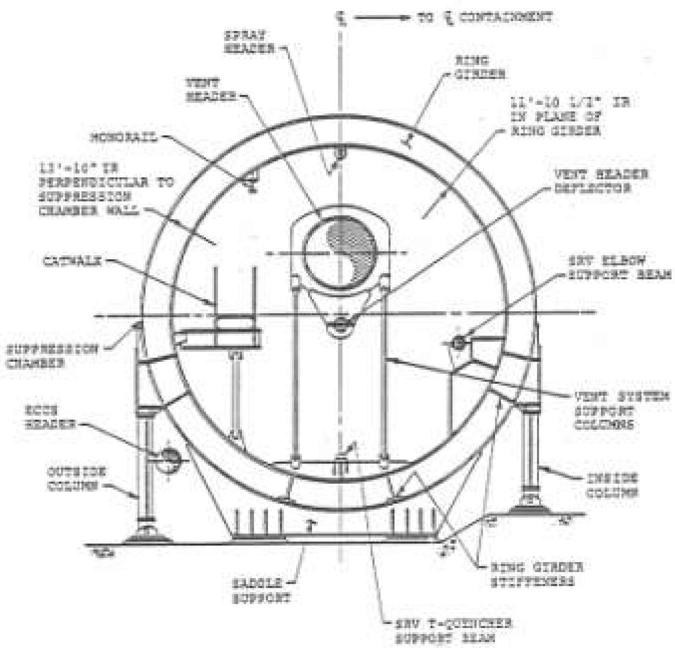

Figure 6-5 Cross section of BWR suppression chamber torus

(Source: Electric Power Research Institute, "BWR Containments License Renewal Industry Report: Revision 1," EPRI TR-103840, Palo Alto, CA, July 1994)

Figure 6-6 Trojan nuclear power plant posttensioned concrete containment

(Source: Naus, D.J. and C.B. Oland, "An Investigation of Tendon Sheathing Filler Migration Into Concrete," NUREG/CR-6598. U.S. Nuclear Regulatory Commission: Washington, DC. March 1998)

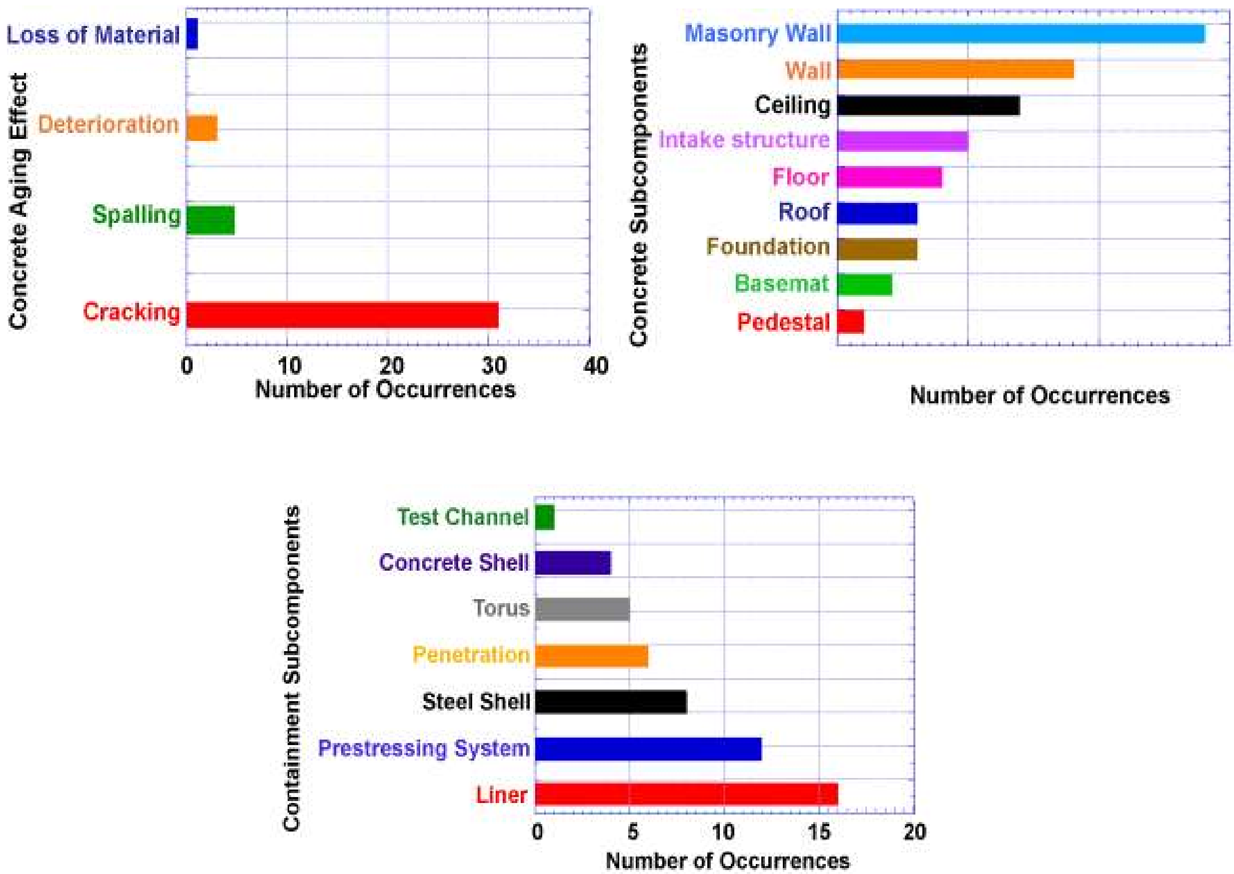

Figure 6-7 Summary of degradation occurrences reported in NUREG/CR-6679 related to nuclear power plant concrete structures, including associated metal subcomponents

(Source: Braverman, J.I., C.H. Hofmayer, R.J. Morante, S. Shteyngart, and P. Bezler. "Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants." NUREG/CR-6679. U.S. Nuclear Regulatory Commission: Washington, DC. August 2000)

**Table 6-1 Leakage from Reactor Refueling Cavity—
Summary of Information Contained in Appendix A**

Topic	Type Plant	Commentary
Leakage source	BWR	Failure of the drywell-to-refueling-cavity seal; small cracks in the liner; or cracks in either the welds of the reactor cavity seal plates, refueling bellows, or reactor cavity drain lines
	PWR	Liner seam, plug, or structural attachment weld defects; cracks in the embedment plates for reactor vessel internals stands support and the rod control cluster change fixture; or reactor cavity and fuel transfer canal liner cracks
Primary structures potentially impacted	BWR	Cylindrical portion of drywell shell and sand bed region of MK I containments; and support structures
	PWR	Reinforced concrete structures; liners of reinforced concrete containments; supports; steel containment vessels; and carbon steel structures where borated water accumulates
How leakage was identified	BWR	Water leakage from sand drains (Browns Ferry Nuclear Plant, Units 2 and 3) Water trickling from a drywell shield wall penetration that produced ponding on the torus room floor (Hope Creek Generating Station, Unit 1) Water in the gap between the drywell and concrete shield (Oyster Creek Nuclear Generating Station)
	PWR	Water leaking through liner plates that collects in a drainage trench (Indian Point Nuclear Generating Unit 2) Leaching and cracking on the outer concrete surface of the reactor refueling cavity wall (Kewaunee Power Station) Leakage in the emergency core cooling system sump and the ceiling of the regenerative heat exchanger room below the reactor refueling cavity (Prairie Island Nuclear Generating Plant, Unit 1 and Unit 2) White deposits present at several locations in containment (Salem Nuclear Generating Station, Unit 1) Liquid running down the containment liner plate and lagging under the fuel transfer canal inside containment and pooling on the concrete floor during refueling operations; leakage from the telltale drains of the leak chase system (Salem Nuclear Generating Station, Unit 2

**Table 6-1 Leakage from Reactor Refueling Cavity—
Summary of Information Contained in Appendix A (cont.)**

Topic	Type Plant	Commentary
How leakage was identified (cont.)	PWR (cont.)	Boric acid deposits over a large surface area of the containment incore instrumentation tunnel walls and undervessel area (Davis-Besse Nuclear Power Station)
Activities to address leakage	BWR	Visual inspections of the interior surface of the drywell and the interior and exterior surface of the drywell head; ultrasonic thickness measurements (Browns Ferry Nuclear Plant, Units 2 and 3) Clear the drywell air gap drain lines of blockage; verify that reactor cavity seal rupture drain lines are clear of blockage and that monitoring instrumentation is functioning properly; monitor daily leakage from the penetration sleeve, drywell air gap drain lines, and reactor cavity seal rupture drain lines; perform boroscope inspections to identify conditions that prevent water leakage from reaching the drywell lower air gap drains; observe variations in water leakage and how it is affected by water levels in the reactor cavity (Hope Creek Generating Station, Unit 1) Perform visual inspections and apply dye penetrant to the reactor refueling cavity liner to identify cracks in the stainless steel liner; apply adhesive stainless steel tape to bridge any observed large cracks, followed by the application of strippable coating; repeat the application of strippable coating to the liner prior to any future reactor refueling cavity flooding; verify that the reactor refueling cavity concrete trough drain is clear of blockage once per refueling cycle; and investigate cost-effective repair or replacement options to eliminate or reduce reactor refueling cavity leakage (Oyster Creek Nuclear Generating Station)

Table 6-1 Leakage from Reactor Refueling Cavity—
Summary of Information Contained in Appendix A (cont.)

Topic	Type Plant	Commentary
Activities to address leakage (cont.)	PWR	Apply patches (coatings) to suspect areas (ineffective); water chemistry program for reactor refueling cavity liner; visual inspections of accessible surfaces under the structures monitoring program for concrete structures (Indian Point Nuclear Generating Unit 2) Increase frequency of visual examination of concrete structures under the structures monitoring program for each refueling outage; inspect current leakage sites and containment internal structures to identify the sites of any additional leakage indications and document any new sites or changes in exiting sites in the corrective action program (Kewaunee Power Station) Initially, a spray-on sealer was applied that provided limited success due to a difficult application process and procedure inadequacies; more recently, reactor vessel internals stands and rod control cluster assembly change fixture embedment plates were repaired by replacement of existing nuts on attachment bolts with blind nuts that were seal-welded to the baseplate, application of seal weld between the baseplate and embedment plate, and examination of welds by nondestructive examination (NDE), and NDE of liner-to-floor embedment plate filet welds; perform visual inspections of the liner plate and conduct vacuum box testing of liner plate seam welds; manage aging of containment structures and containment vessels by using the structures monitoring program and the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) Section XI, Subsection IWE, program (Prairie Island Nuclear Generating Plant, Unit 1 and Unit 2) Periodically inspect telltale drains and monitor for leakage when the reactor refueling cavity is flooded, manage aging using the structures monitoring program and ASME Code Section XI, Subsection IWE, program (Salem Nuclear Generating Station, Units 1 and 2) Fill and hold activities were performed during the spring 2010 refueling outage that identified the SFP elevation at which leakage occurs; a plan is being prepared to do vacuum box testing to identify the locations of potential leakage; if testing is successful, a repair plan will be developed to mitigate leakage (Davis-Besse Nuclear Power Station)

**Table 6-1 Leakage from Reactor Refueling Cavity—
Summary of Information Contained in Appendix A (cont.)**

Topic	Type Plant	Commentary
Assessment of structural impact	BWR	Ultrasonic measurements of the drywell shell in the sand pocket region indicate no damage to integrity (Browns Ferry Nuclear Plant, Units 2 and 3) Ultrasonic examinations of the drywell shell above, around, and below the penetration area where leakage was observed and the complete circumference of the drywell shell at floor junction indicated that the thickness of the drywell shell was above the plate thickness used in design analysis; if a repair cannot be made prior to entering the period of extended operation, augmented inspections will be performed in accordance with ASME Code Section XI, Subsection IWE-1240 (Table-IWE-2500-1, Examination Category E-C), and ultrasonic thickness measurements of the drywell shell will be performed periodically to provide a corrosion rate to demonstrate that the corrosion rate is not sufficient to adversely affect the intended function of the drywell shell (Hope Creek Generating Station, Unit 1) Concrete was excavated at two locations to expose the drywell shell below the floor slab area and ultrasonic measurements were performed to characterize the vertical profile of the drywell shell in the sand bed region where water leakage was observed; core samples of drywell shell were obtained at seven locations representative of wastage to confirm the ultrasonic results; three-dimensional finite-element analyses of the as-built and degraded condition of the drywell containment vessel were performed to demonstrate that the ASME Code-allowable stresses were met (Oyster Creek Nuclear Generating Station)
	PWR	Obtain, test (i.e., compressive strength, boron and chloride concentration, pH), and examine (i.e., petrography) concrete core specimens and examine steel reinforcement removed from areas subjected to borated water; and perform chemical analyses of water leakage from the reactor refueling cavity (i.e., boron concentration, pH, iron, and calcium) (Indian Point Nuclear Generating Unit 2) Obtain, test (i.e., compressive strength), and examine (i.e., petrography) at least one concrete core specimen from a reactor refueling cavity leakage indication site if a core obtained from a location below the SFP indicates degradation (Kewaunee Power Station)

**Table 6-1 Leakage from Reactor Refueling Cavity—
Summary of Information Contained in Appendix A (cont.)**

Topic	Type Plant	Commentary
Assessment of structural impact (cont.)	PWR (cont.)	Remove concrete from the sump below reactor vessels to expose the containment vessels, followed by visual and ultrasonic inspections of the containment vessel and assessment of the removed concrete; remove a concrete sample that has been wetted by borated water leakage from the reactor refueling cavity, test for compressive strength, and conduct a petrographic examination; loss of concrete due to interaction with borated water was estimated and noted to be small relative to the overall section thickness of 1.22 to 1.52 m (4 to 5 feet); ASME Code design calculations were conducted that indicate the shell and bottom head of containment could tolerate general corrosion loss of about 12.7 mm (0.5 inches) out of 38.1 mm (1.5 inches) total thickness without risk to functionality; the impact on concrete and steel reinforcement was determined to be negligible based on results presented in the literature (Prairie Island Nuclear Generating Plant, Unit 1 and Unit 2) Based on experimental results, the depth of erosion of concrete adjacent to the reactor cavity and fuel transfer canal as a result of exposure to borated water was estimated to be on the order of 7.37 mm (0.29 inches) and therefore not sufficient to impact the concrete section or impact the steel reinforcement; perform augmented inspections under the fuel transfer canal once per containment inservice inspection period as long as leakage is observed; a concrete core obtained from the area where leakage of borated water is more extensive and more frequent will be tested and used as a leading indicator for the impact on reactor refueling cavity concrete (Salem Nuclear Generating Station, Units 1 and 2) An engineering assessment that included nondestructive testing, visual inspections, and obtaining and testing of five concrete cores from areas most affected by leakage of borated water indicated that there was no concern with the structural integrity of the concrete and embedded steel reinforcement (Davis-Besse Nuclear Power Station)

**Table 6-2 Leakage from Spent Fuel Pool—
Summary of Information Contained in Appendix B**

Topic	Type Plant	Commentary
Leakage source	BWR	Small cracks in the liner or seam welds
	PWR	Small cracks in liner, seam, or plug welds; structural attachment weld defects
Primary structures potentially impacted	BWR	Cylindrical portion of drywell shell and sand bed region of MK I containments; reinforced concrete structures; metallic pressure boundary; and support structures
	PWR	Reinforced concrete structures; metallic pressure boundary; supports; and carbon steel structures where borated water accumulates
How leakage was identified	BWR	Leakage from leak chase channels (Duane Arnold Energy Center) Leakage from the leak chase system associated with the north wall liner plate when the pool level is increased above the normal level for refueling operations (Hope Creek Generating Station, Unit 1)
	PWR	Hairline crack in the SFP concrete wall, mineral deposits at the pipe end of the leak chase channel, and ongoing slow leakage from leak chase lines (Crystal River Nuclear Generating Plant, Unit 3) Leakage from leak chase channels, periodic water leakage into the emergency core cooling system pump room, and indications of cracking and staining on the underside of the SFP and transfer pit (Davis-Besse Nuclear Power Station) Leakage from leak chase channels with no other leakage indications (e.g., cracks or boric acid deposits on structures) (Diablo Canyon Power Plant, Units 1 and 2) During excavation activities for the fuel storage building, a crack was identified in the SFP concrete wall with moisture observed along one of the cracks; 2 weeks later, a second crack appeared, leaking fluid was observed, and analysis verified that the fluid was consistent with that in the SFP (Indian Point Nuclear Generating Unit 2) White deposits were observed on the wall and ceiling of the waste drumming room below the SFP; and ongoing slow leakage from three leak chase lines (Kewaunee Power Station)

Table 6-2 Leakage from Spent Fuel Pool—
Summary of Information Contained in Appendix B (cont.)

Topic	Type Plant	Commentary
How leakage was identified (cont.)	PWR (cont.)	Low-level shoe contamination picked up in the auxiliary building mechanical penetration room; active water leakage through an exterior concrete wall of the auxiliary building; white deposits on the reinforced concrete wall of the SFP and seepage into the sump room in the fuel-handling building; the presence of borated water in the seismic gap between the fuel-handling building and auxiliary building; leakage from the telltale drains of the leak chase system; the presence of tritium in ground water adjacent to the SFP (Salem Nuclear Generating Station, Unit 1) Water seeping from SFP south and east concrete walls; the presence of white deposits appearing to be boric acid crystals; and chemical analyses of water samples indicating the SFP as the source (Palo Verde Nuclear Generating Station, Unit 1) Moisture present in the sump and leakage of 4.38×10^{-7} m^3/s (10 gpd) of water from the telltale collection pipe (Seabrook Station)
Activities to address leakage	BWR	Walkdowns of accessible areas under the SFP did not identify the presence of leakage; inspections of SFP surfaces will continue during the period of extended operation (Duane Arnold Energy Center) Visual inspections of the north concrete wall of the SFP (Hope Creek Generating Station, Unit 1)
	PWR	The leak chase outlet was cleaned, the leak chase channels were verified to be clear by snaking; implement a preventative maintenance program to periodically verify that the leak chase channels are clear and analyze samples of deposits removed from the leak chase system to check for concrete products; and maintain the SFP level at the lower end of normal water level range (Crystal River Nuclear Generating Plant, Unit 3) Six of 21 leak chase channels that had been clogged were unclogged, resulting in significant release of trapped fluid; the leak collection isolation valves were cleaned; the leak chase program will monitor borated water leakage from the SFP on a monthly basis and, when sufficient volume is available, samples will be obtained and analyzed for pH (monthly), iron (semiannually), and boron content (Davis-Besse Nuclear Power Station)

Table 6-2 Leakage from Spent Fuel Pool—
Summary of Information Contained in Appendix B (cont.)

Topic	Type Plant	Commentary
Activities to address leakage (cont.)	PWR (cont.)	Opening leak detection shutoff valves and observing any water accumulation or flow on a weekly basis; video inspections of leak chase channels to demonstrate that the channels are not blocked; when sufficient volume of leakage is available, samples are obtained and analyzed for tritium, gamma isotopic, pH, iron, and boron (Diablo Canyon Power Plant, Units 1 and 2) Leakage was attributed to defects in the liner as a result of poor workmanship during initial construction of the liner, and the defects were repaired; accessible areas of the SFP liner (about 40 percent) were inspected using robotic cameras, general visual inspection, and vacuum box testing; samples of ground water outside the SFP will be tested at a frequency of 3 months for the presence of tritium and boron (Indian Point Nuclear Generating Unit 2) Leakage indication sites are inspected monthly; portions of the auxiliary building adjacent to the SFP will be inspected annually during the period of extended operation; a multidisciplinary team will develop recommendations for inspection, testing, and repairs to remediate leakage; liner seam weld leakage detection and collection system drain lines will be inspected and repaired (if required) to ensure a clear drain path; monitor ground water for detectable levels of tritium (Kewaunee Power Station) Inspected and repaired seam welds suspected of leaking; perform videoscopic examination of telltale drains and leakage channels, clearing of the leak chase system and monitoring of telltale drains and clearing of leak chase system every 18 months; installed a drainage system and a program to remove water from the seismic gap between the fuel-handling building and auxiliary building; installed a ground water extraction (tritium reclaiming) system (Salem Nuclear Generating Station, Unit 1) A foreign object blocking a leak chase channel was removed and water was released from leak chase channels by opening valves; valves will be opened on a daily basis and water drained, recorded, and trended; drain lines were boroscoped to ensure that they were clear, with the procedure to be repeated at 2-year intervals; shallow aquifer wells were installed down gradient for sampling of water to detect the presence of radioactivity (Palo Verde Nuclear Generating Station, Unit 1)

Table 6-2 Leakage from Spent Fuel Pool—
Summary of Information Contained in Appendix B (cont.)

Topic	Type Plant	Commentary
Activities to address leakage (cont.)	PWR (cont.)	Hydro-lazing (i.e., clearing) of the SFP leak-off lines; monitoring of leak-off from telltales at monthly intervals; analyzing the leak-off collection for gamma and tritium (Seabrook Station)
Assessment of structural impact	BWR	No assessment; leakage confined to leak chase system (Duane Arnold Energy Center)
		No assessment; leakage confined to leak chase system (Hope Creek Generating Station, Unit 1)
	PWR	The only visible indication of degradation is a small hairline crack in the SFP concrete wall that is of no structural significance but will be inspected and monitored at a yearly interval (Crystal River Nuclear Generating Plant, Unit 3)
		A concrete crack on the underside of the SFP will be evaluated; concrete core bores will be obtained prior to the period of extended operation from two areas that have experienced leakage to assess the condition of the concrete and embedded steel reinforcement (Davis-Besse Nuclear Power Station)
		Engineering investigations concluded that long-term leakage was acceptable because the amount of leakage is not sufficient to affect concrete structures; cited results of EPRI TR-1019168, “Boric Acid Attack of Concrete and Reinforcing Steel in PWR Fuel Buildings,” issued June 2009 (Ref. 117); confirmed that leakage was contained within the leak chase channels (Diablo Canyon Power Plant, Units 1 and 2)
		Testing of 20 concrete cores obtained from five locations in the east wall near the location of leakage indicated that the concrete compressive strength was above the design value; steel reinforcement exposed by coring did not exhibit corrosion; windows were placed into the outer wall surface to observe the outer layer of steel reinforcement for signs of corrosion; analyses of the walls indicated that they were capable of resisting loads at least 25 percent greater than design-basis loads (Indian Point Nuclear Generating Unit 2)
		A concrete core sample will be obtained from below the SFP at the location exhibiting the greatest leakage, tested for compressive strength, and examined using petrographic methods; steel reinforcement in the core sample area will be exposed and inspected for material condition (Kewaunee Power Station)

Table 6-2 Leakage from Spent Fuel Pool—
Summary of Information Contained in Appendix B (cont.)

Topic	Type Plant	Commentary
Assessment of structural impact (cont.)	PWR (cont.)	Laboratory testing of the effect of borated water on concrete and steel reinforcement (i.e., erosion and section loss rates, respectively); detailed structural analysis of the fuel-handling building and SFP to demonstrate that the potential impact on structural margins was not significant; conducted a baseline inspection, according to American Concrete Institute (ACI) 201.1R-08, “Guide for Conducting a Visual Inspection of Concrete in Service” (Ref. 118), and ACI 349.3R-02, “Evaluation of Existing Nuclear Safety-Related Concrete Structures” (Ref. 119), of the fuel-handling building to assess its overall structural condition; nondestructive assessment (rebound hammer testing) of the relative condition of “wet” and “dry” portions of the west wall of the sump room; evaluated the impact of potential concrete erosion adjacent to the liner; perform structural inspections of the Unit 1 sump room wall every 18 months and fuel-handling building every 5 years; obtain and test concrete core samples from the east and west walls of the SFP and observe the condition of steel reinforcement exposed by concrete cores; test water drained from the telltales and seismic gap for boron, chloride, iron, and sulfate concentrations, and pH, and use established acceptance criteria to assess the potential for degradation (Salem Nuclear Generating Station, Unit 1) The SFP concrete walls were visually inspected and examined using nondestructive testing techniques (ground-penetrating radar, impulse response, and ultrasonic pulse velocity); the inspection report concluded that borated water leakage did not have an adverse effect on the SFP walls (Palo Verde Nuclear Generating Station, Unit 1) A concrete core sample will be obtained from the leakage area to evaluate the concrete and steel reinforcement (Seabrook Station)

**Table 6-3 Torus Corrosion and Cracking—
Summary of Information Contained in Appendix C**

Topic	Commentary
Inspection method	Visual inspections performed by divers every other outage to detect adverse coating conditions (e.g., flaking, peeling, blistering, discoloration, and other signs of distress), corrosion, or pitting (Cooper Nuclear Station) Visual inspections performed each refueling cycle by divers to detect discoloration, bubbling, or flaking of the coating (Duane Arnold Energy Center) Visual inspections above the waterline according to ASME Code Section XI, Subsection IWE, during refueling outages and below the waterline (when drained or water level lowered) (James A. FitzPatrick Nuclear Power Plant) Visual inspections performed by divers according to ASME Code Section XI, Subsection IWE (Hope Creek Generating Station, Unit 1) Visual inspections with examination of prior photo documentation (Nine Mile Point Nuclear Station, Unit 1) Visual inspections performed by divers every other refueling outage to detect adverse coating conditions (e.g., cracks, sags, runs, flaking, bubbles, and other defects) (Oyster Creek Nuclear Generating Station) Visual inspections under the containment inservice inspection program in line with ASME Code Section XI, Subsection IWE, program requirements (Pilgrim Nuclear Power Station)
Field observations	Large number of repairs (3,800 comprising 1.1 percent of the torus surface below waterline); excessive zinc depletion; pitting at thousands of locations, with 18 locations having pits where the thickness has been reduced greater than 10 percent of the nominal shell thickness (Cooper Nuclear Station) Coating repairs at 15,487 locations representing 5 percent of the underwater coating surface since 1995, with only one pit having degradation that exceeded the maximum allowable pit depth of 1.35 mm (0.053 inches) or 10 percent of the nominal shell thickness; no coating deficiencies requiring repair have been identified above the waterline (Duane Arnold Energy Center)

**Table 6-3 Torus Corrosion and Cracking—
Summary of Information Contained in Appendix C (cont.)**

Topic	Commentary
Field observations (cont.)	Nine pitted areas identified by visual exam when the torus was drained and cleaned in 1998 were measured to determine pit depth; subsequent ultrasonic testing performed each outage found all nine pits to be acceptable according to Code requirements; crack identified in the torus shell due to vibration fatigue from high-pressure coolant injection steam condensation oscillation loading (James A. FitzPatrick Nuclear Power Plant) Sixteen areas representing less than 0.0062 percent of the submerged portion of the torus shell were identified with metal loss up to 0.76 mm (30 mils) (Hope Creek Generating Station, Unit 1) Ultrasonic measurements indicated that the torus wall inside surface had experienced a loss of section due to corrosion (Nine Mile Point Nuclear Station, Unit 1) Blistering of coating in the shell invert and the upper shell near the waterline, with several areas including pitting damage where blisters had fractured (Oyster Creek Nuclear Generating Station) Areas of defects (e.g., depleted zinc, localized pitting corrosion, and minor surface rusting) were identified in 1999 and recoated; ultrasonic measurements in 2003 identified several measurements below nominal wall thickness (Pilgrim Nuclear Power Station)
Corrective and aging management actions	Repair coating by applying an epoxy to areas that have had localized zinc coating failures where the depth of the pits exceeds a threshold (generally 0.762 to 1.27 mm (30 to 50 mils)), with pits not requiring repair monitored at the next inspection (3 years later) for growth; remove sludge and inspect the torus in accordance with ASME Code Section XI, Subsection IWE, every refueling outage until the torus is recoated; recoat the wetted portion of the torus within 3 years after entering the period of extended operation (Cooper Nuclear Station)

Table 6-3 Torus Corrosion and Cracking—
Summary of Information Contained in Appendix C (cont.)

Topic	Commentary
Corrective and aging management actions (cont.)	Maintain photographs, inspection reports and completed checklists, records of corrective actions, and other followup information as quality assurance documents; manage the torus coating in accordance with ASME Code Section XI, Subsection IWE, requirements, the water chemistry program, and the containment inservice inspection program to minimize the potential for loss of material and cracking; recoat the torus interior surface below the waterline prior to the first refueling outage during the period of extended operation (Duane Arnold Energy Center) Manage the torus coating and monitor torus wall thickness using augmented ultrasonic thickness examinations in accordance with ASME Code Section XI, Subsection IWE, requirements; ultrasonic testing of identified pits each refueling outage to evaluate pit depth; coating discrepancies are repaired; the crack in the torus shell is repaired and the sparger assembly installed to eliminate oscillation (James A. FitzPatrick Nuclear Power Plant) Repairs to each location identified made with an epoxy coating; coatings managed by the protective coating monitoring and maintenance program; locations to be reinspected under future ASME Code Section XI, Subsection IWE, underwater inspections (Hope Creek Generating Station, Unit 1) Evaluate corrosion of the torus through inspection and analysis by determination of torus shell thickness through ultrasonic testing at 6-month intervals using a predefined grid system; determination of the corrosion rate through analysis of material coupons; and visual inspection of accessible external surfaces of the torus support structure for corrosion; protective coating monitoring and maintenance program (Nine Mile Point Nuclear Station, Unit 1) The coating is inspected every outage and repaired as required; preventative coating monitoring and maintenance program; ASME Code Section XI, Subsection IWE, program (Oyster Creek Nuclear Generating Station) Defects were recoated in 1999; ultrasonic measurements in 2003 indicated that the shell thickness was above the minimum allowable so no further action was taken; subsequent inspections conducted ultrasonic thickness measurements to monitor change in the thickness of the torus over time (Pilgrim Nuclear Power Station)

**Table 6-3 Torus Corrosion and Cracking—
Summary of Information Contained in Appendix C (cont.)**

Topic	Commentary
Structural integrity assessments	Complete an engineering analysis after each inspection to demonstrate acceptable thickness of the torus wall as impacted by pitting (Cooper Nuclear Station) The one pit exceeding 10 percent of shell thickness was dispositioned under the corrective action program as acceptable without repair in that it would not impact the structural integrity of the torus (Duane Arnold Energy Center) No analysis; the responsible design engineer determined that conditions were acceptable (James A. FitzPatrick Nuclear Power Plant) No analysis was required; the loss was less than ASME Code allowable and a corrosion allowance of 3.18 mm (125 mils) had been incorporated into the original design of the torus (Hope Creek Generating Station, Unit 1) Ultrasonic testing and analysis of corrosion rate data determined from coupons was used to demonstrate that the torus shell thickness will continue to meet minimum design requirements and that any degradation is detected before loss of intended function (Nine Mile Point Nuclear Station, Unit 1) Quantitative assessment (engineering analysis) of pits concluded that pit depths were significantly less than an established acceptance criterion (Oyster Creek Nuclear Generating Station) Ultrasonic thickness measurements will continue in areas where torus thickness was below the nominal wall thickness, with the results compared to those from previous inspection (Pilgrim Nuclear Power Station)

Table 6-4 Age-Related Degradation of Concrete

NPP Plant Type	Occurrence Description
Bellefonte Unit 1 B&W PWR	During 1975 and 1976, a series of eight rock anchor heads for Units 1 and 2 failed during construction installation after having been sealed in a highly alkaline water solution for a long period. Cause was attributed to high anchor head stresses, inclusions in steel, bending of shims and anchor plates, and unknown environmental conditions that produced SCC. Anchor heads were removed and replaced. In 2009, the reactor building containment vertical tendon V9 experienced failure of the rock anchor/tendon coupling. The failure mode was determined to be hydrogen-induced SCC. The root cause of the failure was determined to be water containing sulfides in contact with the grease surrounding the coupling in a high-stress area.
Crystal River Unit 3 B&W PWR	For several prestressing tendon surveillance inspections over the last 20 years, the lift-off forces in the hoop prestressing tendons were consistently found to be lower than the 95-percent predicted values. The cause of the low lift-off forces was attributed to high tendon wire relaxation as a result of elevated temperature effects. The licensee's responsible engineer stated that, although several tendons have demonstrated lower than expected lift-off values leading to adjacent tendons being tested, the end result in all cases thus far has met the set acceptance criteria. Leaching was identified in the tendon gallery. Concrete core samples will be taken at the inside face of the concrete up to the rebar where leaching was observed. The cores will be tested to determine if water-soluble chlorides that could lead to corrosion of the embedded steel are present. Exposed rebar will be examined for any significant corrosion.
Vogtle West PWR	Visual examination detected an area of spalling on the edge of the containment buttresses of Unit 1. An engineering evaluation report concluded that the spalling was nonstructural in nature and had no significant effect on the structural integrity of the containment. (Note: The aging mechanism and the environment for this aging effect were not reported.) On January 7, 1998, personnel were attempting to find a suspected exhaust leak where the diesel generator 1A exhaust pipe exits the roof of the diesel generator building of Unit 1. Concrete damage was discovered on the roof to the diesel generator 1A and 1B exhaust pipe concrete barriers (design-basis missile protection). Concrete had spalled off the inside of the barrier in several locations, exposing the rebar in some areas. In addition, extensive surface cracks were seen on both the interior and exterior of the barrier. The control room and the system engineer were notified of the condition. A preliminary evaluation determined that this degradation had no immediate impact on diesel generator operability. Further investigation on January 14, 1998, by a civil design engineer confirmed the degraded concrete condition previously identified. Additional inspections by plant personnel on January 21, 1998, found similar degraded barriers on the two Unit 2 diesel generators. Followup evaluations utilizing design engineering input also found no adverse impact to continued diesel generator operability. Concrete debris was removed from exhaust piping on all four diesel generators. (Note: The cause of this event was found to be the differential rate of expansion and contraction.)

Table 6-4 Age-Related Degradation of Concrete (cont.)

NPP Plant Type	Occurrence Description
Three Mile Island Unit 1 B&W PWR	Water seeping under three embedded plates on the dome has resulted in some minor leaching of the concrete. These plates extend out from a point close to the dome apex toward the general area of the vent stack on the west side of the containment. This condition was corrected by sealing the concrete to embed interface area with a caulking compound to prevent further entry of water. Reinforcing steel is exposed on the vertical face of the ring girder. Grout patches have detached from the dome surface at two locations, leaving depressions that can accumulate water. One location is along an embed on the west side of the dome. The other is on the west side of the dome close to the crane rail. These depressions were filled with epoxy grout to prevent ponding and the consequent possibility of progressive freeze-thaw damage. Cracking <0.02-cm wide (<0.008 inches) was found in the containment ring girder and around tendon bearing plates. Cracks were repaired and monitored. Concrete surface areas with previously documented damage and deterioration were reexamined. In all cases, the conditions previously recorded were found to be effectively stable. However, in several of these areas, it was determined that repair/restoration work was necessary to ensure against further deterioration. The repair and restoration work was completed in 2006. With one exception, the repaired areas consist of minor restorative work on the concrete surface or sealing against water intrusion.
Brunswick Units 1 and 2 GE BWR/4 MK I	The area surrounding the service water intake structure, adjacent to the intake canal, is subject to an aggressive environment due to high levels of chlorides and sulfates in the intake water. The service water intake structure is monitored at an increased frequency (every 2 years) because of the environment and history of degradation. Operating experience for the submerged portions of the service water intake structure is obtained from divers performing annual preventive maintenance. The only degradation observed was a minor spall of the concrete. No rebar was exposed, and an evaluation determined the damage to be cosmetic. No repairs were required. The licensee's technical staff documented the operating experience related to concrete degradation of the Units 1 and 2 service water buildings (alternate designation for the service water intake structure). The provided information covered only occurrences of degradation for accessible interior and external concrete surfaces. Degradation was attributed to the result of exposure to aggressive, raw service water. Repairs were made.
Peach Bottom Units 2 and 3 GE BWR/4 MK I	The emergency cooling tower and reservoir reinforced concrete walls are exposed to raw water and have experienced leaching of calcium hydroxide. The pH, sulfate, and chloride content of the water are significantly below the threshold limits for aggressive environment. Corrective actions were taken before loss of intended function.
Monticello GE BWR/3 MK I	The aging effects detected during the structural inspections were concrete spalling, cracking, surface deterioration and flaking, grout deterioration, corroded rebar or other steel components, and cracked welds. (Note: The aging mechanisms and the environments that caused the concrete cracking and rebar corrosion were not reported.)

Table 6-4 Age-Related Degradation of Concrete (cont.)

NPP Plant Type	Occurrence Description
Diablo Canyon Units 1 and 2 West PWR	Since 1996, the seawater intake structure was placed twice under increased watchfulness (Maintenance Rule, goal setting (a)(1) status) (Refs. 40 and 41). Each occurrence further showed the adverse impacts of a harsh saltwater environment on concrete degradation. Concrete experts and technicians inspected and documented areas of concrete degradation. Degraded conditions, including delaminations, are documented on drawings of the intake structure. These drawings are updated following each RFO and used to assess the conditions against design- and licensing-basis criteria and for trending purposes in periodically issued reports. The developed refurbishment plan embraces concrete repairs and installation of cathodic protection anodes at various locations, including the seawall and seawall refuse sump overflow opening, traveling screen forebays, circulating water conduits, pump vaults, the top deck, and the intake structure pump deck.
Salem Units 1 and 2 West PWR	In 2002, during performance of preventive maintenance walkdowns to support condition monitoring of the service water intake structure, spalling on the exterior concrete wall near watertight doors SW-1 and SW-5 was observed. The rebar was exposed as a result of the spalling, and corrosion on the rebar was noted. The condition was evaluated by design engineering and repaired in accordance with station specifications. As a followup to this condition report, a walkdown inspection of the area was performed in 2004. It was noted that the spalling condition had been repaired and no indication of additional degradation in the structure was present.
Turkey Point Units 3 and 4 West PWR	Buttress 3 was observed to have three spalls on its face with exposed rebar. These areas of spalling are located in the nonstructural concrete cover. The engineering disposition found that the concrete cover does not perform a structural function. Its purpose is to protect the rebar from exposure to the outside environment. The corrosion observed on the exposed rebar was superficial without adverse effect on the design function of the rebar. The surrounding concrete was tapped by a hammer and no hollow sounds resulted. As a corrective measure, the three subject areas were cleaned and coated with a protective coating. (Note: The aging mechanism and the environment for this aging effect were not reported.) During the 20th-year tendon surveillance, the prestresing forces of a number of tendons were found to be lower than expected. The cause was attributed to excessive wire relaxation due to sustained high temperatures around the tendons. Extensive cracking of reinforced concrete beams supporting the recirculating water pumps of the intake structures was found in 1987–1989. The cause of the cracking was corrosion of the embedded concrete steel reinforcement due to harsh environmental conditions (i.e., salt water). A corrective program was implemented to ensure structural integrity of the beams and minimize future penetration of chloride ions. Unit 4 was also affected.
Nine Mile Point Unit 1 and 2 GE BWR/2 MK I	Minor cracking of the service water pipe tunnel was observed. The service water pipe tunnel is susceptible to small wall cracks allowing leakage of ground water. The repaired areas of the service water pipe tunnel were inspected, and there was no entry of ground water in the areas repaired. Frequency of inspections following the repairs has varied from monthly (initially) to quarterly.

Table 6-4 Age-Related Degradation of Concrete (cont.)

NPP Plant Type	Occurrence Description
Kewaunee West PWR	During the 1997 periodic structure monitoring inspections of the screen house and tunnel, cracking with leaching was observed. In March 2003, during similar inspections of the screen house and tunnel, multiple concrete degradation mechanisms were observed on a wall. The noted deficiencies or aging effects were localized and include cracking, leaching, patterned cracking, and a slight surface offset. The wall surface, however, appeared dry during the inspection. Cracking with leaching was observed previously during the 1997 inspection (e.g., surface offset and localized pattern crack formations). Followup inspections in December 2004 indicated that the condition of the affected area and the overall wall were stable, with no changes observed since the last inspection. No moisture and no new cracking or leaching were observed or were apparent. In addition, no other new surface condition attributes were observed during the subsequent inspection, which suggests that the cracking is passive. The area was reexamined in April 2008 and will be included in a long-range rehabilitation plan. The structure status summary continues as “acceptable with deficiencies.” Minor leaching of calcium hydroxide in an air-outdoor environment has occurred for the main auxiliary transformer walls, the turbine building exterior wall, the shield building exterior concrete wall, and the screen house forebay exterior concrete wall. Water from rain or melting snow passes through cracks or inadequately prepared construction joints, causing leaching of calcium compound from concrete. Efflorescence, a surface phenomenon consisting of salt deposits that have been leached from concrete, was present.
Seabrook West PWR	Loss of material due to chemical attack of calcium hydroxide is considered to be an aging effect requiring management for Seabrook Station. There have been indications of leaching in below-grade concrete in Seabrook Station structures other than the containment building. Seabrook Station has scheduled specific actions (e.g., concrete testing) to determine the effects of aggressive chemical attack due to high chloride levels in the ground water. An evaluation will be performed based on the results of the testing and a determination of the concrete condition, which may lead to additional testing or increased inspection frequency. Testing of concrete may consist of concrete core samples, penetration resistance tests, petrographic analysis of the concrete core samples, and visual inspection of rebars as they are exposed during the concrete coring. Seabrook will evaluate the results of the testing and, if required, undertake additional corrective actions in accordance with the structures monitoring program corrective action program.

Table 6-4 Age-Related Degradation of Concrete (cont.)

NPP Plant Type	Occurrence Description
Farley 2 West PWR	Failure of three anchor heads on the bottom ends of vertical tendons occurred about 8 years after tensioning. Eighteen cracked anchor heads and several broken wires were also found. The cause was attributed to hydrogen stress cracking. All tendons and anchor heads from same material heat were inspected, with no additional problems noted. Twenty tendons were replaced.
Sequoyah 2 West PWR	Concrete in the outer 2.5 to 5 cm (1 to 2 inches) of the shield building was under strength as a result of exposure to freezing and thawing temperatures at an early concrete age. The degradation was determined not to impact shield building capability.
Calvert Cliffs Units 1 and 2 CE PWR	During the 20th-year surveillance of the prestressing system, low lift-off values were found for vertical tendons. Inspections of adjacent tendons in response to low lift-off values revealed that tendon wires had broken. Further examination of wires at the tops of tendons revealed additional broken tendon wires. About 30 percent of the vertical tendons in Units 1 and 2 were replaced. Engineering evaluation indicated that the cause was primarily hydrogen-induced cracking.
Summer West PWR	During the 4th tendon surveillance, the prestressing forces of a number of vertical tendons were found to be lower than expected. The cause was attributed to excessive wire relaxation due to sustained high temperatures around the tendons. The vertical tendons were retensioned.
Oconee B&W PWR	Concrete beneath the 50.8-mm (2-inch) thick anchor-bearing plate for tendon 12V6 had spalled along the outer edge; a cavity existed below the plate. Cracks in the concrete beneath the outer edge of the bearing plates were observed for a number of tendons. The environment inside the tendon gallery included water infiltration and high humidity. (Note: The aging mechanism for the cracking was not reported.)
Oyster Creek GE BWR/2 MK I	Cracking of the drywell shield wall was attributed to high temperature in the upper elevation of the containment drywell. Engineering analysis concluded that stresses were well below allowable limits, considering the existing cracked condition. Recent inspections identified no significant change in the cracked area.
Millstone Unit 3 West PWR	Accumulated white residue from a porous concrete drainage system was found in two lower drain sumps of the engineered safety features building in 1987. It was determined that the quantity of cement eroded was minor. The sumps were monitored for cement erosion, and the foundations were monitored for settlement.
San Onofre Unit 1 CE PWR	Exterior concrete walls of the intake structure and the concrete beams supporting service water pumps were cracked extensively. Cracking was due to chloride ion penetration that resulted in concrete steel reinforcement corrosion. Walls were reinforced with exterior steel plates anchored to concrete, and sacrificial zinc anodes were placed onto steel plates to protect against corrosion.
Davis-Besse Unit 1 B&W PWR	The auxiliary feedwater pump turbine exhaust missile barrier has spalled concrete and exposed rebar due to its periodic exposure to a harsh environment. The missile barrier continues to perform its design function, and the corrective action program is tracking the repair. Several tower and disconnect switch concrete foundations in the switchyard are degraded to the point that concrete has spalled off and rebar is visible. This issue was evaluated through the corrective action program. The switchyard's ground appeared to be saturated with ground water because of insufficient drainage. The corrective action program was used to evaluate this issue.

Table 6-4 Age-Related Degradation of Concrete (cont.)

NPP Plant Type	Occurrence Description
Beaver Valley Unit 1 West PWR	Cracks, water infiltration, and calcium deposits in the ceilings and walls of the service building, safeguard structure, and steam generator drain tank exist.
Robinson Unit 2	Cracking and spalling of concrete (in limited areas) in the walls and ceilings of the reactor auxiliary building, emergency diesel generator building, and intake structure exist.

7. SUMMARY AND CONCLUSIONS AND TOPICS FOR FURTHER CONSIDERATION

7.1 Summary and Conclusions

As nuclear plants age, occurrences of degradation of spent fuel pools (SFPs), reactor refueling cavities, and the torus structure of light-water reactor nuclear power plants (NPPs) are occurring at an increasing rate, primarily due to environment-related factors. Several of the 104 commercial NPPs in the United States have experienced water leakage from the SFPs and reactor refueling cavities. In addition, since the year 2000, the license renewal applications (LRAs) filed by the licensees have noted several cases of corrosion in the Mark I containment torus steel structures of boiling-water reactors (BWRs). Age-related concrete degradation has also been indicated in other instances.

The objectives of this NUREG document are to (1) identify the cause, extent, and effect of leakage from deteriorated seals between the refueling cavity and reactor cavity areas, and/or leakage through stainless steel weld seams, or through stainless steel base metal, (2) identify the cause, extent, and effect of leakage from BWR and pressurized-water reactor (PWR) SFPs through the stainless steel weld seams, or through the stainless steel base metal, and the field activities that were performed to address the leakage, (3) identify possible causes of corrosion of the torus of different BWR Mark I plants, including the locations and remedial actions, (4) provide an update on operating experience with respect to NPP reinforced concrete structures, (5) summarize mitigating actions of the industry to manage or rectify issues associated with aging effects because of Objectives 1, 2, and 3 above, thus extending the safe operating life of NPPs, and, last but not least, (6) extend past studies and efforts to serve as a precursor for the industry to identify degradation scenarios that potentially could dominate in the future (e.g., the impacts of historical water leakage on the structural integrity of SFPs and refueling cavity liners and associated concrete structures). Selection of degradation occurrences involved identification of aging effects from publicly available documents, followed by the identification of associated aging mechanisms and environments. Consistency and clarity in presentation of information was maintained to the extent possible throughout the report.

Publicly available documents related to operating experience (e.g., licensee event reports, U.S. Nuclear Regulatory Commission (NRC) generic communications, NRC preliminary notification reports, NRC interim staff guidance, and NRC regulatory guides), NRC NUREGs, licensee inspection reports, license renewal applications, and NRC safety evaluation reports of license renewal applications were reviewed and screened. Reports containing information related to one of the four topics of interest noted above were binned (i.e., grouped) into the appropriate topic of interest for more detailed review. The subject matter expert then completed a more detailed review of the potential information sources that resulted in identification of 11, 12, and 7 occurrences related to reactor refueling cavity leakage, SFP leakage, and torus corrosion and cracking, respectively. This process also resulted in identification of 26 occurrences related to age-related degradation of NPP reinforced concrete structures. Publicly available information related to the occurrences identified was then summarized.

7.1.1 Reactor Refueling Cavity Leakage

Leakage from the reactor refueling cavity primarily occurs only during refueling outages and has been identified at eight sites involving 11 units (four BWRs and seven PWRs). The primary sources of leakage for BWRs have been failure of the drywell-to-refueling cavity seal and cracks in liner plate seam welds, refueling bellows, or reactor cavity drain lines. The primary sources of leakage for PWRs have been liner plate seam or plug welds and cracks in embedment plates for reactor vessel internals stands support and the rod control cluster fixture. For BWRs, the primary concern related to leakage from the reactor refueling cavity is the potential for corrosion of the drywell shell and support structures. In the case of PWRs, leakage of borated water may cause corrosion of the metallic pressure boundary, reactor supports, and containment structure, or it may affect other structures and components on which the borated water accumulates.

Leakage has primarily been identified in BWR plants by water present in the gap between the drywell and concrete, or by water trickling from a drywell shield wall penetration. Leakage in PWR plants was primarily identified by leakage of water or boric acid deposits on walls and equipment, or their presence in sumps. Activities to address leakage from BWRs have primarily included visual inspections and ultrasonic testing of the drywell shell, application of strippable coatings or adhesive steel tape, inspection and clearing (as required) of the drainage systems, and monitoring of leakage. PWR activities to address leakage have included application of coatings to suspect areas, visual inspections of accessible surfaces of structures that could be potentially impacted, application of sealer coatings, visual inspections and nondestructive examinations (NDEs) of liner plate and seam welds, and inspection and clearing (as required) of the drainage systems.

Assessment of the structural impact of leakage for BWRs has included ultrasonic thickness measurements of the drywell shell, augmented examinations under American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) Section XI, Subsection IWE, and finite-element calculations to demonstrate that ASME Code-allowable stresses are met. Structural impact assessments for PWRs have included (1) removal of concrete cores to inspect concrete steel reinforcement, (2) strength determinations and petrographic examination of the concrete, (3) chemical analyses of water leakage, and (4) exposure of the containment vessel by concrete coring for visual examinations and ultrasonic thickness measurements of the vessel. Wall thickness measurements obtained to date have been at or above nominal, and there have not been visible signs of corrosion.

7.1.2 Spent Fuel Pool Leakage

Leakage from the SFP has been identified at 10 sites involving 12 units (2 BWRs and 10 PWRs). The primary sources of leakage for BWRs have been small cracks in the liner or seam welds. The primary sources of leakage for PWRs have been cracks in the liner plate seam or plug welds, or structural attachment weld defects. For BWRs, the primary concern related to leakage from the SFP is the potential for corrosion of the cylindrical portion of the drywell shell and sand bed region of Mark I containments, reinforced concrete structures, the metallic pressure boundary, and support structures. In the case of PWRs, leakage of borated water may erode the concrete or produce corrosion of the concrete steel reinforcement, the metallic pressure boundary, supports, and carbon steel structures where borated water can accumulate. Leakage has primarily been identified in BWR plants through leakage from the leak chase system. Leakage in PWR plants has been identified by leakage from the leak chase system, seepage associated with cracks in concrete, the presence of white deposits on structures, the presence of moisture in the seismic gap between the fuel-handling building and auxiliary building, the presence of tritium in ground water, and contamination of protective clothing. Activities to address leakage in BWRs have primarily included walkdowns of accessible areas around the SFP. PWR activities to address leakage have included inspection and clearing (as required) of the drainage systems, monitoring and analysis of leakage collected by the drainage system, visual inspections of accessible areas of the SFP liner and SFP concrete surfaces, and sampling of ground water for tritium. No assessments of the structural impact of leakage identified for BWRs has been reported because all of the plants indicated that the leakage was contained within the leak chase drainage system. Structural assessments of the PWRs have included periodic visual inspections; obtaining concrete cores to assess the condition of the concrete and the embedded concrete steel reinforcement; baseline inspections according to American Concrete Institute (ACI) 201.1R-08, “Guide for Conducting a Visual Inspection of Concrete in Service” (Ref. 118), and ACI 349.3R-02, “Evaluation of Existing Nuclear Safety-Related Concrete Structures” (Ref. 119); application of NDE techniques; and laboratory testing of the effect of borated water on concrete and concrete steel reinforcement. SFP leakage has been more common for PWRs, the leakage related to BWRs has been confined to the leakage collection system, and the borated water can potentially impact the structures it contacts. Additional information related to leakage of borated water from SFPs is provided below.

7.1.2.1 Effect of Borated Water on Materials

Borated water leakage from PWR SFPs (and reactor refueling cavities) can cause erosion of concrete as a result of borated water’s acidic nature (a pH of 4.8 to 5.2) or result in corrosion of carbon steel materials. Concrete affected by borated water exhibits softening, or erosion, due to attack of the cement paste and acid-soluble constituents in the aggregates. The attack begins at the exposed surface of the concrete, can progress inward, and will stop when the reaction is complete. The rate of degradation of concrete exposed to borated water is controlled by the diffusion of borated water into the concrete and depends on factors such as the concrete mixture proportions and constituents, pH of the fluid, time of exposure, and temperature. Formation of reaction products can slow the rate of concrete erosion because a continuous supply of calcium hydroxide is required for the process to continue. Reaction of the borated water with cementitious components in the concrete or some aggregate materials (e.g., limestone) also tends to somewhat neutralize the acidity of the solution, such that the pH of the leaking fluid can approach neutrality or become somewhat alkaline.

The concrete cover protects the carbon steel reinforcing bars from corrosion. Corrosion of the concrete steel reinforcement begins when acid attack of the cover concrete progresses to the level of the steel reinforcement. Cracks in the concrete cover and construction joints can also allow the borated water to access the steel reinforcement and initiate the corrosion process. (Cracks that completely penetrate the concrete section or construction joints can also permit the borated water to impact other structures, such as the carbon steel pressure boundary.) Corrosion of the steel reinforcement produces tensile stresses in the concrete as a result of a volume increase associated with formation of the corrosion products. If the corrosion process continues, the net section of the steel reinforcement can be reduced and there can be loss of bond between the concrete and steel reinforcement that can potentially impact the component structural integrity. Although very limited, results presented in the literature and operating experience seem to indicate that the interaction of borated water with concrete has not produced significant erosion of the concrete because borated water is a relatively “weak” acid. Corrosion of the embedded concrete steel reinforcement as a result of exposure to borated water has also been relatively minor, with no reported observations of spalled concrete.

Borated water that has migrated to the metallic pressure boundary (i.e., the liner or containment) is of most concern relative to its potential to produce corrosion resulting in loss of section or compromised leak tightness. Although results presented in the literature are very limited, ultrasonic measurements in areas of the metallic pressure boundary potentially exposed to borated water leakage have reported that the wall thickness is at or above nominal, and evidence of corrosion is not present.

7.1.2.2 Extent of Concrete and Steel Degradation

One study was identified in which laboratory testing was conducted to simulate the impact of borated water leakage on concrete. Results of the testing indicated that the projected depth of concrete degradation in the floor slab of the SFP after 70 years exposure would be on the order of 33.0 millimeters (mm) (1.3 inches) and 11.2 mm (0.44 inches) in the walls of the SFP (Ref. 120). Because these penetration depths were less than the thickness of the concrete cover to the embedded steel reinforcement, the steel reinforcement would not be impacted unless a crack in the concrete was present. It was noted that results obtained from this study were supported by examination of concrete cores removed from the Connecticut Yankee SFP. The same study also investigated the impact of borated water on the concrete steel reinforcement, with the result that degradation of the steel reinforcement was minor and that migration (or wicking) of the borated water along the interface between the concrete and embedded steel reinforcement was limited and would not cause a general loss of bond. The results of this study were then used to estimate the impact of exposure through the end of plant life. The estimated degradation was projected to reduce the available structural margin in the limiting cross section of the SFP by less than half a percentage point to 1.6 percent (i.e., a design margin ratio of 1.016).

7.1.3 Torus Corrosion and Cracking

Eight occurrences of corrosion of the BWR torus have been identified in seven units located at seven sites. The primary inspection method used to identify the presence of corrosion or degraded protective coatings has been visual, typically performed by divers. Degradation of the torus coating reduces its ability to protect the steel substrate from corrosion, and localized coating failures expose areas of the substrate to corrosion. Failed coatings can also contribute to the amount of sludge and corrosion products that collect in the suppression pool that can further increase the corrosion rate. Field observations have indicated that certain units have experienced a large number of coating repairs (e.g., in excess of 15,000 at one plant), and ultrasonic thickness measurements have generally indicated that the metal loss has not been sufficient to impact the structural integrity of the torus. Corrective or aging management actions have primarily been to repair coating defects by cleaning and application of an epoxy coating, and inspections (or augmented inspections) performed according to ASME Code Section XI, Subsection IWE, requirements. Assessments of the structural integrity of the torus have included conduct of an engineering analysis after each inspection to demonstrate acceptable thickness, and performing ultrasonic thickness measurements to establish a corrosion rate to demonstrate that the torus shell thickness will continue to meet minimum design requirements and that any degradation will be detected prior to loss of intended function.

7.1.4 Age-Related Degradation of Reinforced Concrete Structures

Primary mechanisms (factors) that, under unfavorable conditions, can produce premature deterioration of reinforced concrete structures include those that impact either the concrete or the steel reinforcing materials (i.e., mild steel reinforcement or posttensioning systems). Degradation of the concrete can be caused by adverse performance of either its cement-paste matrix or aggregate materials under chemical or physical attack. Degradation of mild steel reinforcing materials can occur as a result of corrosion, irradiation, elevated temperature, or fatigue effects, with corrosion being the most likely form of attack. Posttensioning systems are susceptible to the same degradation mechanisms as mild steel reinforcement plus loss of prestressing force, primarily due to tendon relaxation, and concrete creep and shrinkage. Identification of degradation of reinforced concrete structures in NPPs is primarily through visual inspections conducted in accordance with ASME Code Section XI, Subsection IWL, that incorporates guidance contained in ACI 201.IR-08 (Ref. 118) and ACI 349.3R-02 (Ref. 119). Posttensioning systems are evaluated for loss of prestressing force and degradation using guidance provided in ASME Code Section XI, Subsection IWL. Application of NDE methods to concrete structures has been limited, with the methods primarily used to quantify existing degradation or to investigate areas where degradation is suspected. In some cases concrete cores have been obtained to evaluate concrete strength, investigate concrete embedded steel reinforcement for corrosion, or perform petrographic examinations.

In general, the performance of NPP safety-related concrete structures has been very good. However, occurrences of concrete degradation in all likelihood will increase as the plants age, primarily due to environmental effects. Occurrences of degradation have been identified through visual inspections primarily conducted in accordance with Maintenance Rule inspection requirements, or in compliance with ASME Code Section XI, Subsection IWL, requirements. This study identified 37 occurrences of degradation (26 occurrences from the present study plus 11 occurrences from previous investigations) that have occurred at 23 sites. Examples of degradation identified include concrete cracking, concrete freezing and thawing damage, corrosion of steel reinforcement, corrosion of posttensioning tendon wires, anchor head failures due to stress-corrosion cracking or hydrogen embrittlement, leaching of tendon gallery concrete, and larger than anticipated loss of prestressing forces. Of these occurrences, concrete cracking and corrosion of embedded concrete steel reinforcement were the most common.

7.2 <u>Topics for Further Consideration</u>

With aging plants, the regulatory requirements have increased. For license renewals, NUREG-1800, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” first issued July 2001 (Ref. 31), presents a well-defined base that is consulted during staff evaluations of a licensee’s application. The technical basis of license renewals for structures centers on NUREG-1801, “Generic Aging Lessons Learned (GALL) Report,” first issued July 2001 (Ref. 32), and other regulatory documents (e.g., regulatory guides, design guides) that provide additional guidance to assess licensee programs and their elements. Topics for further consideration could include those that enhance such program elements as parameters monitored or inspected, and the detection of aging effects.

Technology is needed to develop improved in situ detection techniques for corroded torus areas and for evaluating the impact on concrete structures of aging (e.g., potential reduction in structural margins) resulting from borated water exposure or expansions due to presence of alkali silica or alkali aggregate reactions. Methodologies to assess the life expectancy of these components through modeling and simulation, supplemented where applicable with repairs and management of the detrimental aging effects, are also desired. For weldments, techniques that provide improvements in current capabilities (e.g., relative to that of vacuum box testing) for detection of aging effects, such as cracks, porosity, lack of fusion, and incomplete penetration, would be beneficial. Based on operating experience gained from current inservice coating systems, in conjunction with modeling and simulation, methods could be developed for use in evaluating the performance of newer, yet untested by the industry, coatings (or “adhesive” steel tape-based) systems to effectively stop leakage in reactor refueling cavities and SFPs, or to protect tori from further corrosion so that they can continue to perform within their design assumptions and intended functions during the period of extended operation.

Some of the topics for further consideration listed below are based on an understanding of the operating experience at NPPs as described in Appendices A, B, and C, Chapter 6, and knowledge of aging effects and aging mechanisms that can adversely affect reinforced concrete structures exposed to environments common at existing NPPs. These considerations also reflect methods for mitigating the potential consequences of aging effects on reinforced concrete structures in the next generation of NPPs. Reinforced concrete's behavior is strongly entrenched in the collective performance of the following:

- concrete, composed of chemically fused fine and coarse aggregate, cement, and admixtures

- reinforcing steel

- the "bond zone," the boundary layer between steel and concrete, that, through adhesion and mechanical interlock, contributes to the effective performance of the composite material

7.2.1 Reactor Refueling Cavity and Spent Fuel Pool Leakage

Repairs in reactor refueling cavity and SFP structures are difficult and sometimes impractical. Their leakage, however, could be stabilized or controlled through technology that could help reduce the porosity of or eliminate "pinholes" in liner weldments. Assembling operating experience related to the various types of coatings being used by licensees to stop the leakage of borated water through cracks in the reactor refueling cavity and SFP stainless steel liners is the first step to help identify coatings with good performance histories. This information would be useful in developing an improved basis for evaluating proposed plans by licensees to reduce or eliminate the reactor refueling cavity and SFP leakage through application of coating materials. Development of procedures for the repair of reactor refueling cavity and SFP leakage at seam welds (e.g., adhesive steel tape) or plug welds (e.g., composite patches) would also be beneficial with respect to reducing or eliminating leakage.

Compiling laboratory and petrographic examination results from concrete cores taken by different licensees to quantify the effects of borated water leakage on concrete and steel reinforcing bars would contribute to an improved understanding of the significance of the effect of borated water on concrete, steel reinforcing materials, and concrete-steel bond. These results also have application to the development of quantitative and predictive measures to assess the need for corrective actions, such as repairs or replacements, and for structural assessments of reinforced concrete structures. Little is understood about the effects on concrete permeability of lengthy exposures to low acidity environments. Friable and increasingly porous concrete resulting from continuous exposure to borated water flow or seepage, particularly the concrete adjacent to reinforcing steel, could potentially result in a reduction in the confinement of the reinforcing steel that could impact the composite performance of the concrete and steel. Lack of adequate adhesion and mechanical bond of steel to concrete as a result of a compromised bond zone could potentially result in a reduction in structural margins. Similarly, compromised (or loss of) clear cover to the steel reinforcement due to the exposure of concrete to borated water not only increases the vulnerability of the reinforcing steel to the environment but also increases the requirement for bond development length that may or may not be available, thus potentially reducing the overall composite structure's capacity to resist the loads to which it was originally designed. An effort, therefore, is needed to understand the aged composite's behavior (reinforced concrete) and to follow up with updated analyses accounting for the modified composite material performance, thus realistically assessing the impacted structures' design margins.

Periodic monitoring of the fluid collected by the leak chase channels and testing for items such as pH, iron and boron content, the presence of concrete constituents, and chloride concentration could provide data useful in developing criteria related to assessing the significance and potential impact of the leakage on the concrete and steel reinforcing materials and thus the structural margins. Results will also provide information to verify the source of the leakage.

As operating experience shows several occurrences of reactor refueling cavity and SFP leakage at existing plants, alternative (or improved) designs for reactor refueling cavity and SFP leakage collection systems for the next generation of NPPs would be beneficial. Alternative (or improved) designs could incorporate provisions for improved access for periodic inservice inspection of the reactor refueling cavity and SFP liners and collection system components, and the potential to repair or replace areas of the liner that experience inservice damage, including through-wall cracks and defects in the liner plate base metal and the seam and plug welds. With respect to new plants having an SFP with borated water, operating experience indicates that additional detail and improved methods concerning the examination of the plate seam welds would be beneficial, and minimization or elimination of the use of plug welds could lead to improved leak-tightness of the refueling cavities and SFPs.

7.2.2 Torus Corrosion and Cracking

Diligent monitoring of the torus structure for pitting corrosion is extremely important. Randomly developing corrosion pits introduce nonlinearities of unknown implications to the integrity of the toroidal structure. To date, because of the difficulty of the in situ measurement and monitoring of these imperfections, coupled to that associated with computational/approximation techniques to account for their influence, structural analyses for the assessment of torus structural integrity during the period of extended operation has been quite challenging. For example, the traditional modeling of corroded material next to that of pristine/uncorroded material inadvertently introduces computational inaccuracies (localized stresses due to thick/thin transitions and introduced eccentricities) that could affect the accuracy of results. Therefore, improvements of computational techniques are required to assess the impact of corrosion on the torus structural integrity.

Development of improved guidelines and more effective inspection techniques for evaluating pitting corrosion would be beneficial, as they would lead to more consistent assessments of its significance in the torus structure. These guidelines could be used to develop acceptance criteria (e.g., pit density) that complement the acceptance standards in ASME Code Section XI, Subsection IWE-3500. These complementary acceptance criteria could address specific parameters (e.g., pit depth and pit distribution) that need to be considered by the responsible individual and establish the types of engineering evaluations required to justify acceptance without repair or replacement.

Compiling operating experience related to the effectiveness of torus coating repairs in controlling or eliminating pitting corrosion would aid in identifying coating repairs with good performance histories. These data would be useful in developing an improved basis for evaluating proposed plans by licensees to repair coating damage or delay the coating repair until a later date. The results could also be used as the basis for evaluating LRAs for consistency with Aging Management Program (AMP) XI.S8, “Protective Coating Monitoring and Maintenance Program,” in NUREG-1801 (Ref. 32).

7.2.3 Concrete Age-Related Degradation

Although there is a good understanding of the behavior of posttensioned and reinforced concrete structures in the NPPs, a greater understanding of their performance and durability by taking into consideration their original design assumptions and the impact of aging will help extend their design life, ensure their structural integrity (e.g., structural margins), and reinforce the public’s trust in their continued safe operation. A thorough understanding can help reduce the time and cost of repairs, while a lack of understanding could limit the service life of a plant. Technology based on past performance or operating experience, therefore, is needed to identify and suggest repair techniques to maintain the structural integrity of older plants during the period of extended operation.

To provide an improved process for understanding concrete degradation, a consistent use of definitions and terms for describing and standardizing structures, components, materials, environments, aging effects, and aging mechanisms in NUREG-1801 when preparing reports that describe occurrences of concrete degradation would be beneficial. This would reduce possible confusion in characterizing and categorizing occurrences of reinforced concrete structure and component degradation. Consistent use of definitions and terminology, therefore, would lead to improved and more consistent evaluations of operating experience.

Other issues regarding material durability focus on material reactivity. Slower or less reactive forms of alkali-aggregate reactions may be present, resulting in alkali-aggregate reactivity occurring even though the aggregate materials may have met the American Society for Testing and Materials testing requirements with respect to potential reactivity. Development of an improved test procedure to evaluate the potential reactivity of candidate aggregate materials for use in repair activities or in construction of new NPP concrete structures would help to ensure that alkali-aggregate reactions would not occur in the future. Also, identification of procedures to evaluate the existence of or potential for future reactivity of aggregates in existing plants (i.e., future expansion) would be beneficial in formulating mitigation strategies, as well as in developing assessment criteria. An improved understanding of the resulting concrete expansion as well as the current and anticipated rates of expansion is important. The occurrence of alkali-aggregate reactivity could reduce concrete’s ductility as the reinforcing steel attempts to keep up through the bond zone with the expanding concrete. There are little understanding and few data available to perform re-analyses of reinforced concrete structures under conditions such as this (Ref. 121).

The knowledge accumulated from the above two examples on material reactivity for cement, paste, aggregates, and lengthening the life of reinforcing steel under adverse conditions would help establish databases that can be used to develop predictive tools. These tools that are based on improved understanding of age-related degradation mechanisms and the projected durability of materials would help with the assessment of the remaining life of reinforced concrete structures in NPPs.

Sacrificial anodes have been used in an attempt to prevent corrosion of carbon steel reinforcing bars embedded in concrete exposed to brackish water or seawater. They have been applied to both new construction and the repair of existing structures. A review of operating experience would provide information on the effectiveness of the sacrificial anodes in reducing the potential for corrosion of reinforced concrete structures subjected to brackish or salt water (e.g., coastal water intake structures). Development of an operating experience database would contribute to an improved basis for evaluating LRAs for consistency with AMP XI.S6, “Structure Monitoring Program,” and AMP XI.S7, “RG 1.127, Inspection of Water-Control Structures Associated with Nuclear Power Plants,” in NUREG-1801 (Ref. 32).

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APPENDIX A: LEAKAGE FROM REACTOR REFUELING CAVITY CASE STUDIES

A.1 <u>Browns Ferry Nuclear Plant, Units 2 and 3</u>

A.1.1 Introduction

On December 31, 2003, the Tennessee Valley Authority submitted a license renewal application (LRA) to the U.S. Nuclear Regulatory Commission (NRC) for its Browns Ferry Nuclear (BFN) Plant, Units 1, 2, and 3 (Ref. A.1.6.1). Results of the NRC staff's evaluation of the LRA were published in April 2006, in NUREG-1843, "Safety Evaluation Report Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3" (Ref. A.1.6.2). The NRC also published a supplement to NUREG-1843 in April 2006 (Ref. A.1.6.3).

A.1.2 Field Observations

During the Unit 2 Cycle 9 outage, a portion of the moisture barrier was replaced (Problem Evaluation Report (PER) BFPER971516) (Ref. A.1.6.4). Engineering personnel performed an examination of the exposed drywell steel containment vessel area below the moisture seal. This inspection indicated some minor pitting and localized rust but nothing indicating a challenge to nominal wall thickness. No propagation of iron oxide to the concrete surface was noted, which would be indicative of steel containment vessel corrosion below the concrete. Inspections conducted by the containment inservice inspection program during the Unit 2 Cycle 10 refueling outage and Unit 3 Cycle 9 refueling outage also identified some damaged areas of the moisture barrier (gaps, cracks, low areas or spots, or other surface irregularities) that were evaluated by engineering and replaced or repaired (PER 99-005254-000 for the Unit 2 drywell moisture seal barrier and PER 00-004163-000 for the Unit 3 drywell moisture seal barrier) (Ref. A.1.6.4).

A.1.3 Design Characteristics of Reactor Refueling Cavity

The reactor building for each unit completely encloses the reactors, the primary containment structures, and the auxiliary and emergency systems of the nuclear steam supply system. A major substructure of the reactor building is the reinforced concrete biological shield that surrounds the drywell portion of the primary containment. The reactor buildings also house features such as the spent fuel pool, steam dryer/moisture separator storage pool, reactor cavity, reactor auxiliary equipment, refueling equipment, reactor servicing equipment, and control bay. The control bay houses the main control room for plant operation and other important auxiliary systems required for plant operation. The reactor building consists of monolithic reinforced concrete floors and walls from its foundation to the refueling floor. The refueling floor, that is common for all three units, is enclosed by the steel superstructure with metal siding and a built-up roof. Blowout or pressure relief panels are installed as part of the reactor building metal siding superstructure to relieve pressure during a design-basis accident (DBA) or a design-basis earthquake (DBE).

A.1.4 Corrective Actions for Reactor Refueling Cavity Leakage

During 1987, each unit's drywell was ultrasonically tested near the sand cushion area (Ref. A.1.6.4). The results from these tests showed that the nominal thickness was maintained on each drywell. The following are the results of each unit's drywell ultrasonic testing (UT):

- Unit 1—No reading below the nominal thickness of 25.4 millimeters (mm) (1 inch) was measured, indicating that the integrity of the drywell liner plate is maintained. Periodic leakage from the sand cushion area has been observed. Corrosive species in the drainage are suspect of a higher rate of corrosion on the Unit 1 drywell liner plate than on Unit 2 and 3. However, objective evidence of serious corrosion damage was not noted.

- Unit 2—No reading below the nominal thickness of 25.4 mm (1 inch) was measured, indicating that no damage to the integrity of the drywell liner plate has occurred.

- Unit 3—No reading below the nominal thickness of 25.4 mm (1 inch) was measured, indicating that no damage to the integrity of the drywell liner plate has occurred.

During each refueling outage since the mid-1980s, visual inspections of the interior surface of the drywell and the interior and exterior surface of the drywell head and torus (suppression chamber) are performed to verify structural integrity (SI) (Ref. A.1.6.5). These inspections are performed per SI 0-SI-4.7.A.2.K, "Primary Containment Drywell Surface Visual Inspection," and BFN Technical Instruction 0-TI-417, "Inspection of Service Level I, II, III Protective Coatings." SI 0-SI-4.7.A.2.K originally included the exam requirements for the visual inspections of the protective coatings but was revised in March 2001 to remove those requirements and add the reference to BFN Technical Instruction 0-TI-417 for coating inspections. BFN Technical Instruction 0-TI-417 was written to incorporate the information for performing visual inspections of Service Level I protective coatings (DBA and non-DBA qualified). Procedure SI 0-SI-4.7.A.2.K provides for visual inspections of the following:

(1) structural components of the drywell, drywell head, torus (suppression chamber), and the exterior surfaces of the drywell head and torus (suppression chamber) (i.e., piping, connections, structural supports, penetrations, platform steel, duct supports, concrete walls, and steel shell) by visually inspecting for deterioration or structural damage, or both

(2) moisture seal barrier located on drywell elevation 167.6 meters (m) (550 feet).

(3) interior surfaces of the drywell and torus (suppression chamber) above level 0.305 m (1 foot) below the normal waterline and the exterior surface of the torus (suppression chamber) below the waterline each operating cycle for deterioration and any signs of structural damage, with particular attention to piping connections and supports, and for signs of distress or displacement

A.1.5 Structural Integrity Assessment and Test Results

Existing degradation of the drywell shells (inside and out) at the time of license renewal has not reached the minimum required thickness of 25.4 mm (1 inch) (Ref. A.1.6.2). For Unit 1, one-time confirmatory ultrasonic thickness measurements will be performed on the vertical cylindrical area immediately below the drywell flange (Ref. A.1.6.6). A commitment was made to perform these ultrasonic thickness measurements before the Unit 1 restart and before the period of extended operation for Units 2 and 3. For Units 2 and 3, the same testing will be performed in the portion of the cylindrical section of the drywell in a region where the liner plate is 19.05 mm (0.75 inches) thick. This will provide a bounding condition, since the nominal thickness of the wall in this region has the least margin.

A.1.6 References

A.1.6.1 Burzynski, M.J., Tennessee Valley Authority, letter to U.S. Nuclear Regulatory Commission, December 31, 2003.

A.1.6.2 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3." NUREG-1843. U.S. Nuclear Regulatory Commission: Washington, DC. April 2006.

A.1.6.3 — — —. "Safety Evaluation Report Related to the License Renewal of the Browns Ferry Nuclear Plant, Units 1, 2, and 3." NUREG-1843, Supplement 1. U.S. Nuclear Regulatory Commission: Washington, DC. April 2006.

A.1.6.4 Abney, T.E., Tennessee Valley Authority, letter to U.S. Nuclear Regulatory Commission, January 31, 2005.

A.1.6.5 Skaggs, M.D., Tennessee Valley Authority, letter to U.S. Nuclear Regulatory Commission, May 31, 2005.

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A.2 Hope Creek Generating Station, Unit 1

A.2.1 Introduction

On August 18, 2009, PSEG Nuclear, LLC (PSEG) submitted an LRA to the NRC for renewal of the Hope Creek Generating Station (Ref. A.2.6.1). Results of the NRC staff's evaluation of the LRA were published in June 2011, in NUREG-2102, "Safety Evaluation Report Related to the License Renewal of the Hope Creek Generating Station" (Ref. A.2.6.2). On November 3, 2010, a presentation was made to the Advisory Committee on Reactor Safeguards (Ref. A.2.6.3).

A.2.2 Field Observations

During the 2009 refuel outage, about a 6.35-mm (0.25-inch) wide trickle of water was found exiting the seal rupture drain line penetration sleeve J13 (see Figure A.2-2) from the drywell air gap region and had ponded on the torus floor. Analysis of the ponded water identified it as reactor water/refueling water. The water leakage stopped after the refueling cavity was drained at the end of the refueling outage. The suspected source of the water was a defect or tear in the refuel bellow or the liner. Because water may be trapped between the concrete and the drywell steel below penetration sleeve J13, which is located approximately 2.44 m (8 feet) above the drywell lower air gap drains, corrosion of the drywell steel containment is possible. The air gap drains were inspected in 2009, but there was no blockage or standing water found in the air gap region and the drywell shell showed no signs of corrosion. The water leakage issue was entered into the corrective action process to determine the cause of the leakage and corrective actions to prevent reoccurrence.

A.2.3 Design Characteristics of the Refueling Cavity

The reactor building includes the spent fuel storage pool liner, cask loading pit liner, reactor cavity liner, steam dryer/moisture separator storage pool liner, and spent fuel storage pool skimmer surge tank liner. The containment has no sand-pocket region. Potential leakage of water from the reactor cavity during refueling is removed by the reactor cavity seal rupture drain lines that are monitored by instrumentation designed to alarm the main control room in the event of leakage. There are also drains at the bottom of the air gap region at the junction where the shell becomes embedded in concrete.

A cross section of the reactor building and Mark I containment with the locations of the spent fuel pool, reactor cavity seal liner proximity to concrete containment, torus shell, and so forth is shown in Figure A.2-1. Further features, including the J13 penetration areas, are shown in Figure A.2-2. Cross sections of the reactor cavity seal area and the lower drywell area are detailed in Figures A.2-3, and A.2-4, respectively.

A.2.4 Corrective Actions for Refueling Cavity Leakage

Various activities were performed prior to restart from the 2009 refueling outage, and it was determined that the leakage was due to a small crack or cracks in either the welds of the reactor cavity seal plates, refueling bellows, or reactor cavity drain lines. Additional activities were planned during the 2010 outage to determine the root cause. These activities included (1) inspecting reactor cavity seal rupture drain lines for blockage and monitoring leakage daily from penetration sleeve J13, the drywell air gap drain lines, and the reactor cavity seal rupture drain lines, (2) observing variations in water leakage and characterizing how it is affected by the water levels in the reactor cavity, and (3) performing boroscope inspections below penetration sleeve J13 for conditions that prevent water leakage from reaching the drywell lower air gap drains. In addition, the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) (Ref. A.2.6.4) Section XI, Subsection IWE, program was enhanced to (1) verify that the reactor cavity seal rupture drain lines are clear from blockage and that the monitoring instrumentation is functioning properly prior to the period of extended operation and one additional time during the first 10 years of the period of extended operation, (2) investigate the source of any leakage detected by the reactor cavity seal rupture drain line instrumentation and perform an assessment of its impact on the drywell shell, (3) monitor the drains daily at the bottom of the drywell air gap for leakage when the reactor cavity is flooded, and (4) periodically monitor penetration sleeve J13 for water leakage when the reactor cavity is flooded up until corrective actions are taken to prevent leakage through J13.

A.2.5 Structural Integrity Assessment and Test Results

One-time UT thickness measurements from inside the drywell in the accessible area of the drywell shell directly below penetration sleeve J13 were scheduled to be performed in 2010. Inspection and acceptance criteria will be in accordance with ASME Code Subsections IWE-2000 and IWE-3000, respectively, and, in the event significant corrosion is detected, the condition will be entered in the corrective action program for evaluation and extent of condition determination. In addition, the area will be designated for augmented examination in accordance with ASME Code Subsection IWE-1240 requirements, as identified in Table IWE-2500-1, Examination Category E-C (Ref. A.2.6.4). UT examination of the drywell area below penetration J13 down to penetration J37 will be performed after the one-time examination in 2010 and during each inspection period (three times in 10 years) until the reactor cavity water leakage from penetration J13 is repaired.

The reactor cavity leakage will be repaired, if practical, before the period of extended operation. If repairs cannot be made prior to the period of extended operation, augmented inspections of the affected area of the drywell surface will be performed (Ref. A.2.6.5) to demonstrate that corrosion is not occurring or corrosion is progressing so slowly that age-related degradation will not jeopardize the intended function of the drywell through the period of extended operation. In addition, a corrosion rate will be developed based on past UT examinations. This rate will be used to project loss of drywell thickness through the period of extended operation, and to evaluate the results to determine if the drywell can perform its intended function during the period of extended operation with reduced thickness.

A.2.6 References

A.2.6.1 Fricker, C.J., PSEG Nuclear, LLC, letter to U.S. Nuclear Regulatory Commission, August 18, 2009.

A.2.6.2 U.S. Nuclear Regulatory Commission. “Safety Evaluation Report Related to the License Renewal of Hope Creek Generating Station.” NUREG-2102. U.S. Nuclear Regulatory Commission: Washington, DC. June 2011.

A.2.6.3 “Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Hope Creek License Renewal Subcommittee.” Work Order No. NRC-542. Neal R. Gross and Co., Inc., Court Reporters and Transcribers: Washington, DC. November 3, 2010.

A.2.6.4 American Society of Mechanical Engineers. “ASME Boiler and Pressure Vessel Code.” American Society of Mechanical Engineers: New York, NY. 2007.

A.2.6.5 Brady, B.M., U.S. Nuclear Regulatory Commission, letter to PSEG Nuclear, LLC, September 2, 2010.

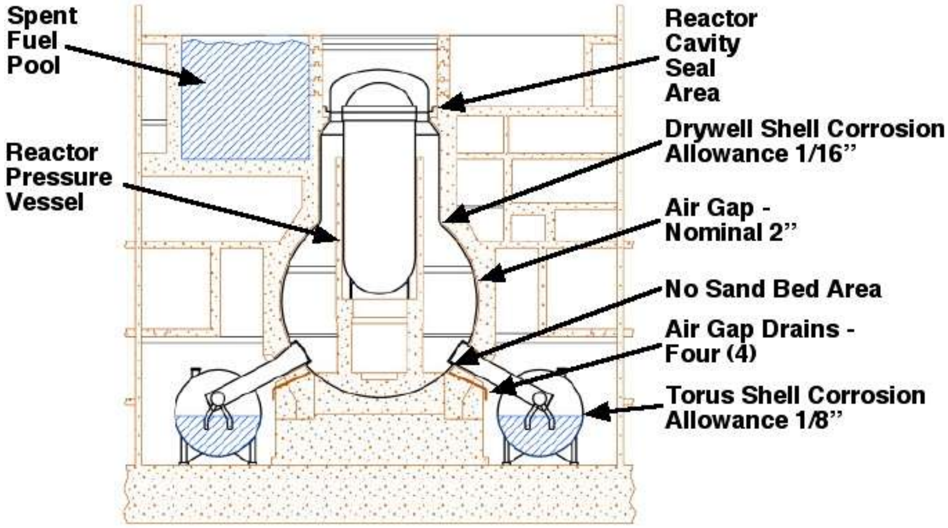

Figure A.2-1 Cross section of reactor building and Mark I containment

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Hope Creek License Renewal Subcommittee." Work Order No. NRC-542. November 3, 2010)

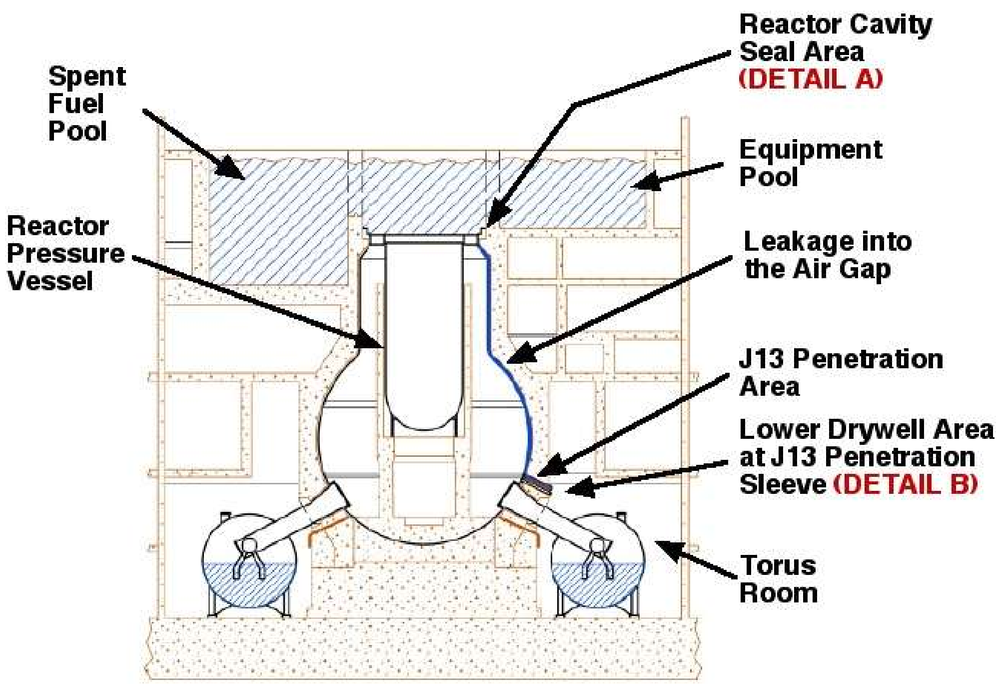

Figure A.2-2 Locations of reactor cavity seal and J13 penetration areas

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Hope Creek License Renewal Subcommittee." Work Order No. NRC-542. November 3, 2010)

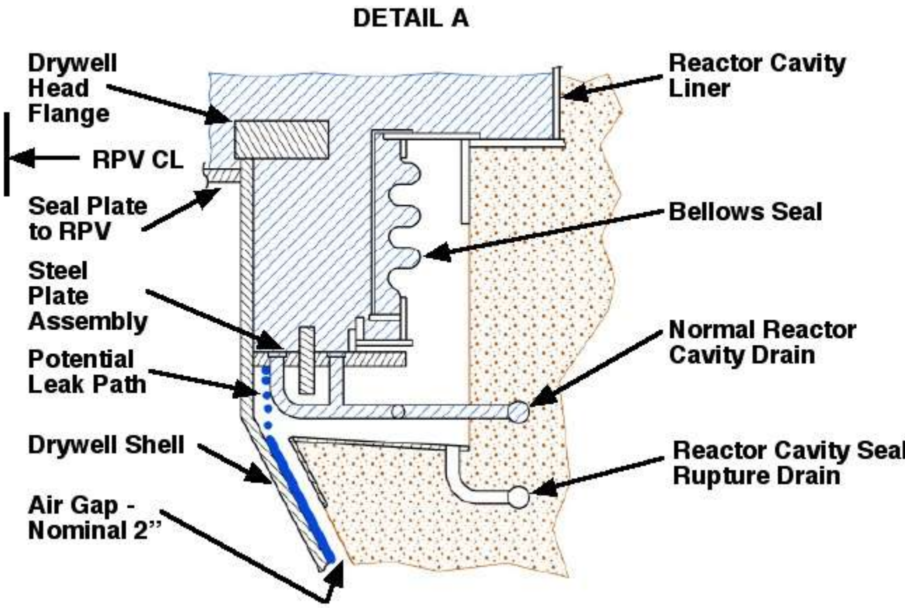

Figure A.2-3 Cross section of reactor cavity seal area

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Hope Creek License Renewal Subcommittee." Work Order No. NRC-542. November 3, 2010)

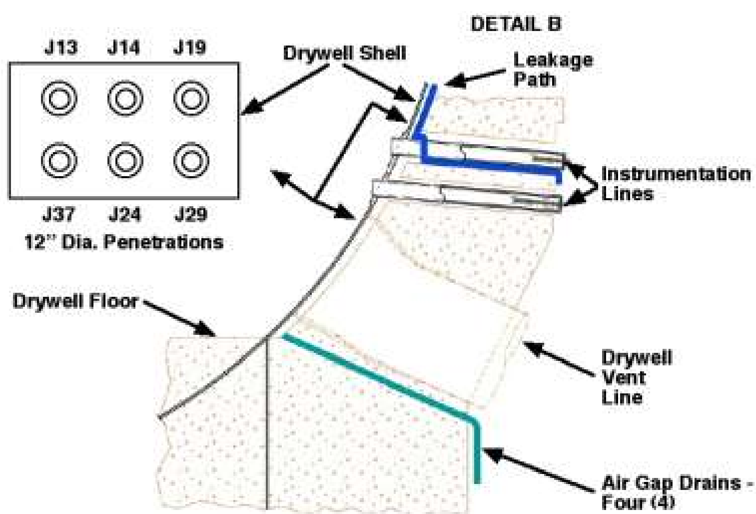

Figure A.2-4 Cross section of lower drywell area

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Hope Creek License Renewal Subcommittee." Work Order No. NRC-542. November 3, 2010)

A.3 Indian Point Nuclear Generating Unit 2

A.3.1 Introduction

By letter dated April 23, 2007 (Ref. A.3.6.1), and as supplemented by letters dated May 3 (Ref. A.3.6.2) and June 21, 2007 (Ref. A.3.6.3), Entergy Nuclear Operations, Inc., submitted an LRA to the NRC for Indian Point Nuclear Generating Units 2 and 3. The results of the NRC staff's evaluation of the LRA were published in November 2009 in NUREG-1930, "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3" (Ref. A.3.6.4).

A.3.2 Field Observations

During the first refueling outage in 1976, leakage from the refueling cavity was observed. It was attributed to the lack of leak tightness between the originally designed temporary seal of the reactor vessel flange and the reactor cavity. Once leakage was collected in the reactor cavity pit sump, it was pumped out. A plant modification followed, in which a newly designed seal resolved the problem. Leakage also occurred in the reactor vessel inlet and outlet blow-out plugs and instrumentation wireways. Leakage through these paths has been minimized by improved sealing methods.

In 1993, through visual observations, leakage from the refueling cavity was detected, and it was concluded to be coming through the stainless steel liner plates when the cavity is filled during refueling outages (normally every 24 months) (Ref. A.3.6.5). Leaks last approximately 14 days and are a mixture of reactor coolant and refueling water from the storage tank, and they begin when the water level reaches approximately 50 percent of the refueling cavity level. Total estimated flow rates are on the order of 1.8 to 4.4×10^{-4} cubic meters per second (m^3/s) (3 to 7 gallons per minute (gpm)). A small portion of the leakage from the refueling cavity enters the reactor cavity, flowing down the interior primary shield walls to a sump located in the reactor cavity, from which it is pumped to the containment sump. Leakage inside the reactor cavity has been primarily attributed to non-liner leakage associated with reactor cavity seal and nozzle inspection box cover isolation issues. No samples of the fluid flowing from the leaking areas have been analyzed for chemical composition (Ref. A.3.6.4).

Visual examination of the leakage areas has not identified any degradation of concrete.

The 1993 leakage was observed initially from three areas associated with refueling cavity construction: the liner seam, plug, and structural attachment welds on the west wall. Damage to the liner was determined to have occurred during previous refueling outages due to poor cleanliness and maintenance control. This included use of improper material and tools (a wire brush contaminated with carbon steel and containing chloride) coming in contact with stainless steel. In addition, a cut into the liner plate occurred during removal (cutting out) of temporary attachments to the liner (Ref. A.3.6.5). CeramAlloy™ patches at the 24.4- to 25.9-m (80- to 85-foot) elevation along two horizontal seam welds located on the south wall and patches along the seam weld on the north wall approximately 3.0 to 4.6 m (10 to 15 feet) above the cavity floor were identified as potential leakage sources (Ref. A.3.6.6).

A.3.3 Design Characteristics of the Refueling Cavity

The refueling cavity in the containment building is a robust structure with 1.2-m (4-foot) thick walls and low stress levels when compared to the total structural capacity (Ref. A.3.6.4).

A.3.4 Corrective Actions for Refueling Cavity Leakage

Attempts, with limited success, were made over several outages to mitigate the refueling cavity leakage. A sequential action plan with the following elements was developed (Ref. A.3.6.7) for a permanent fix to this issue:

- 2008/2009, research available technologies for leak repairs in the refueling cavity.

- Spring 2010 refueling outage—Repair area of north wall weld seams in the vicinity of the CeramAlloy™ patch and south wall along area of disbonded CeramAlloy™ patch. Analyze water leaking from the refueling cavity for boron concentration, pH, iron, and calcium.

- Spring 2012 refueling outage—Repair east wall where large CeramAlloy™ patch has disbonded and area around access ladder on northwest corner.

- Spring 2014 refueling outage—Repair areas of lower cavity where CeramAlloy™ patches have disbonded, and miscellaneous areas observed as suspect to leakage from past inspections.

- During each of the preceding outages, areas not permanently repaired will be temporarily repaired by the application of InstaCote™. Beginning with the refueling outage in Spring 2016, no InstaCote™ will be applied, to allow Entergy to determine if repairs have successfully stopped the leakage. If not, additional areas will be repaired in subsequent outages until the leakage is corrected.

The method to monitor for a degrading condition in the refueling cavity is routine visual inspection of accessible concrete surfaces under the structures monitoring program, accompanied by an inspection of concrete that has been exposed to the intermittent borated water leakage for an extended period. For aging management of the cavity steel liner, Entergy will rely on the water chemistry control—primary and secondary program.

A.3.5 Structural Integrity Assessment and Test Results

In 2008, it was noted that the leakage does not pose a threat to the SI of the refueling cavity reinforced concrete walls, which are 1.22 m (4 feet) thick, and several documented tests were cited that concluded borated water does not significantly degrade concrete properties (Ref. A.3.6.8). Furthermore, it was noted that substantial design margins for both steel and concrete exist and that the flooded condition and leakage last only 2 weeks out of a refueling cycle. Examination of a 1993 core sample removed from the Unit 2 refueling cavity wall showed that the depth of penetration of borated water was 12.7 mm (0.5 inch) into the concrete at that time. A number of attempts have been made to rectify the leakage, but to date have not been completely successful. Work will continue toward a permanent fix, but this effort will be prioritized based on safety significance and the availability of site resources (Ref. A.3.6.4).

In 2009, the licensee stated that, during the upcoming 2010 outage, a total of three core bore samples will be taken from the reinforced concrete walls that form the outer shell of the reactor refueling cavity steel liner (Ref. A.3.6.6). The locations of these core bores will be chosen based on the following:

- Locations in the vicinity of observed liner/liner patch degradation in relative proximity to the observed leak points on the concrete structure noted earlier.

- Accessibility of suspect areas based on the principle of As Low As Reasonably Achievable (ALARA) and physical interferences.

The core samples will be tested and chemically analyzed to determine the effect, if any, that past leakage has had on the concrete properties. The objectives of the physical and chemical tests of the concrete core samples are as follows:

- Determine the compressive strength of concrete.

- Determine boron and chloride concentration in concrete.

- Determine pH of concrete.

A petrographic examination will be performed on the core samples to evaluate the condition of the cementitious matrix and, to the extent possible, determine the durability of the concrete. In addition, reinforcing steel in the cored areas, when exposed, will be visually inspected to determine the extent of loss of material, if any, as a result of the borated water leakage.

It was further noted in 2009 that, if a solution to the leakage has not been achieved, Entergy will perform core bore samples and reinforcing steel inspections prior to 10 years into the period of extended operation based on the extent and location of the remaining leakage following previous repair efforts. The core samples will be tested and chemically analyzed as discussed above. Visual inspections of the reinforcing steel will follow to assess loss of material as a result of the borated water leakage.

A.3.6 References

A.3.6.1 Dacimo, F.R., Indian Point Energy Center, letter to U.S. Nuclear Regulatory Commission, April 23, 2007.

A.3.6.2 Dacimo. F.R., Indian Point Energy Center, letter to U.S. Nuclear Regulatory Commission, May 3, 2007.

A.3.6.3 Dacimo. F.R., Indian Point Energy Center, letter to U.S. Nuclear Regulatory Commission, June 21, 2007.

A.3.6.4 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3." NUREG-1930, Volumes 1 and 2. U.S. Nuclear Regulatory Commission: Washington, DC. November 2009.

A.3.6.5 Dacimo, F., Entergy Nuclear Northeast—Indian Point Energy Center, letter to U.S. Nuclear Regulatory Commission, March 24, 2008.

A.3.6.6 Dacimo, F., Entergy Nuclear Northeast—Indian Point Energy Center, letter to U.S. Nuclear Regulatory Commission, June 12, 2009.

A.3.6.7 Dacimo, F., Entergy Nuclear Northeast—Indian Point Energy Center, letter to U.S. Nuclear Regulatory Commission, November 6, 2008.

A.3.6.8 Dacimo, F., Entergy Nuclear Northeast—Indian Point Energy Center, letter to U.S. Nuclear Regulatory Commission, August 14, 2008.

A.4 <u>Kewaunee Power Station</u>

A.4.1 Introduction

On August 12, 2008, Dominion Energy Kewaunee, Inc., submitted an LRA to the NRC for its Kewaunee Power Station (Ref. A.4.6.1). Results of the NRC staff's evaluation of the LRA were published in January 2011 in NUREG-1958, "Safety Evaluation Report Related to the License Renewal of Kewaunee Power Station" (Ref. A.4.6.2).

A.4.2 Field Observations

Leaching and cracking on the outer concrete surface of the reactor refueling cavity wall (south side) was observed in April 2003 (Ref. A.4.6.1). The indications noted were localized and the overall SI of the wall appeared sound. The reactor refueling pool was flooded at the time of the observation. The hairline cracking was considered passive and did not affect the SI of the concrete wall. Based on data from earlier inspection and chemistry sampling, a small amount of borated water found its way down the wall, followed the lip of the narrow crack, and deposited boric acid crystals when it dried. The accessible wall area was cleaned. During a subsequent inspection in October 2004, there was no change in appearance from 2003, nor any indication of an active leak or the presence of moisture.

During the 2006 and 2008 refueling outages, three leakage locations were identified (Ref. A.4.6.3).

Leakage Site No. 1 is noted above. This cracked location was reinspected with the refueling pool flooded. The inspection did not find any active leakage through the crack. Based on the October 2004 inspection, it was concluded that no further action was required and that the leakage indication at Leakage Site No. 1 in 2003 was due to a small amount of borated water from a source external to the reactor refueling cavity pool.

Leakage Site No. 2, identified in October 2006, was located at the construction joint of the "A" reactor coolant system (RCS) vault. Residue was observed streaking and staining the wall and there was a small amount of moisture on the wall surface below the location of the construction joint, but there was no measurable leakage or accumulation of boric acid crystals. This leakage site was again inspected during the next refueling outage in April 2008 and wetting, or moisture, was noted with a small amount of accumulation of residue, and the presence of wall staining and streaking at the "A" RCS vault wall construction joint was also noted. This leakage site was reinspected during the 2009 refueling outage, when multiple inspections were performed at different times during the outage. The initial "as-found" inspection was followed by another inspection prior to filling the reactor refueling cavity pool, and another inspection after filling the pool, and concluded with a final inspection at the end of the outage. After the as-found inspection, the residue was removed from the leakage area. No further indication was noted until the final inspection 17 days later after the filling of the reactor refueling cavity, when a small leakage was discovered. As a result, the frequency of the inspection under the structures monitoring program was increased to document and trend the observed conditions and assess the integrity of the concrete structure.

Leakage Site No. 3, identified in March 2008, was located at the junction between the reinforced concrete biological shield wall and the base of the reactor refueling cavity. This leakage also showed accumulation of residue and streaking and staining on the wall surface. There was no quantifiable water flow from the junction and it was considered to be minor leakage. Similar to Leakage Site No. 2, Leakage Site No. 3 was further inspected multiple times during the refueling outage of 2009. Based on the long delay for leakage indication to reappear on the wall surface, potential reactor refueling cavity pool leakage at this location was also considered minor. However, the frequency of inspection has been increased under the structures monitoring program to each refueling outage in order to document and trend the observed conditions and assess the integrity of the concrete structure.

During the refueling outage in 2009, additional inspections were performed to check for the presence of other leakage and to verify that there was no moisture in contact with the reactor containment vessel. The containment basement and sump B, which is located nearest to the containment vessel, were inspected. No leakage was identified that would indicate potential moisture presence in contact with the reactor containment vessel. In addition, this inspection did not identify any additional leaks forthcoming from the reactor refueling cavity pool leakage.

A.4.3 Design Characteristics of the Refueling Cavity

The reactor containment vessel is a Class I cylindrical steel structure with a hemispherical dome roof and ellipsoidal bottom. The containment vessel is completely enclosed by the shield building. An annular space separates the shield building and the containment vessel except at the lower portion of the containment vessel that is embedded in concrete. The major concrete components are the reactor cavity shield wall, refueling pool, compartment vaults, and the floors at various elevations. The reactor cavity concrete shield wall surrounds the reactor vessel and provides biological shielding and structural support. The top of the shield wall forms the refueling cavity pool.

The reactor vessel flange is sealed to the bottom of the refueling cavity by a reactor cavity seal ring that prevents leakage of refueling water into the reactor cavity. A removable missile shield cover, constructed of a concrete slab enclosed by steel plates, is located above the reactor vessel head. The reactor cavity, refueling pool, and reactor containment vessel sump (sump A) are lined with stainless steel liner plates. All the liner plates are welded to structural shapes that are embedded and anchored in the reinforced concrete walls and floors. Sand plugs, constructed of steel boxes with a stainless steel cover, are installed to protect safety-related components located in the recessed areas of the refueling pool floor. Radiation shields at the fuel transfer tube penetration are used to protect personnel.

A.4.4 Corrective Actions for Refueling Cavity Leakage

The results of the inspections performed during the 2009 refueling outage were evaluated to provide input to determine corrective actions needed for the reactor refueling cavity pool liner leakage (Ref. A.4.6.4). A multidisciplinary team was formed to develop a remediation plan to identify and remedy reactor refueling cavity liner leakage. The plan is to be implemented during the period of extended operation (Ref. A.4.6.5). The plan includes weld examinations and identification and resealing of liner penetrations as potential leakage sites. Until the plan is implemented, the current leakage sites will continue to be inspected during each refueling outage, as well as containment internal structures with the objective of identifying any additional leakage indication sites. New leakage indications, or changes in existing leakage rates, will be documented in the corrective action program and evaluated.

A.4.5 Structural Integrity Assessment and Test Results

Based on other nuclear plant evaluations, the effect of borated water on reinforced concrete SI is considered to be minimal, and the leakages at the identified locations are very small (Ref. A.4.6.2). Consequently, it was concluded that the degradation of the reinforced concrete or the metal reactor containment vessel is negligible. To confirm this conclusion, an SI examination of the concrete slab below the spent fuel pool in the auxiliary building will be performed and used as a representative location comparable to Leakage Sites No. 2 and No. 3, because the reinforced concrete material and the environments are the same for both locations.

At the minimum, one core bore sample will be obtained near at least one of the refueling cavity leakage indication sites if the core sample obtained below the spent fuel pool related to spent fuel pool leakage indicates degradation. The core sample will be tested for compressive strength and will undergo a petrographic examination.

A.4.6 References

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A.4.6.3 Scace, S.E., Dominion Energy Kewaunee, Inc., letter to U.S. Nuclear Regulatory Commission, August 17, 2009.

A.4.6.4 Price, J.A., Dominion Energy Kewaunee, Inc., letter to U.S. Nuclear Regulatory Commission, December 28, 2009.

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A.5 Oyster Creek Nuclear Generating Station

A.5.1 Introduction

By letter dated July 22, 2005 (Ref. A.5.6.1), AmerGen submitted an LRA to the NRC for the Oyster Creek Nuclear Generating Station. Results of the NRC staff's evaluation of the LRA were published in April 2007 in NUREG-1875, Volumes 1 and 2, "Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station" (Ref. A.5.6.2).

A.5.2 Field Observations

The Oyster Creek Nuclear Generating Station first discovered water in the gap between the drywell and the concrete shield in 1980 and began investigating the cause of the water problem in 1983. Water collection varied from a few drops to 1.3×10^{-4} m^3/s (2 gpm), depending on whether the unit was in operation or was in an outage for refueling. During the spring and summer of 1986, the licensee planned work to identify and eliminate this water leakage problem (Ref. A.5.6.3).

Water that leaked from the reactor refueling cavity passed over the Firebar-D® coating that was applied to the drywell shell to allow for formation of the required seismic gap between the drywell shell and the encircling concrete shield wall. The Firebar-D® material is a magnesium oxychloride compound. Corrosion of the outside surface of the drywell shell was discovered in 1986. The leaking water that accumulated in the sand bed region corroded the embedded drywell shell.

A.5.3 Design Characteristics of the Refueling Cavity

The reactor building houses the spent fuel pool, the steam dryer/moisture separator storage pool, the new fuel storage vault, the reactor cavity, reactor auxiliary equipment, refueling equipment, and reactor servicing equipment. It also provides secondary containment when the primary containment is in service and provides primary containment during reactor refueling and maintenance operations when the primary containment system is open. The building is designed to Seismic Class I criteria and is constructed of reinforced concrete up to the refueling floor level.

The reactor cavity is a stainless-steel-lined reinforced concrete structure located inside the reactor building. It is filled with water during refueling operations. Bellows provide the seal between the reactor vessel and the drywell and between the drywell and the reactor cavity liner. Gaskets provide the seals between the drain lines and the stainless steel liner. Details of the reactor cavity seals are shown in Figures A.5-1 and A.5-2.

A.5.4 Corrective Actions for Refueling Cavity Leakage

In the mid- to late 1980s, extensive visual and nondestructive examination (NDE) inspections were conducted to determine the source of water intrusion into the seismic gap between the drywell concrete shield wall and the drywell shell, and its accumulation in the sand bed region (Ref. A.5.6.3). The inspections concluded that the refueling bellows (seals) were not the source of water leakage. The bellows were repeatedly tested using helium (external) and air (internal) without any indication of leakage. Furthermore, any minor leakage from the refueling bellows would be collected in a concrete trough below the bellows. The concrete trough is equipped with a drain line that would direct any leakage to the reactor building equipment drain tank and prevent it from entering the seismic gap (see Figures A.5-1 and A.5-2). The drain line had been checked before refueling outages to confirm that it was not blocked.

The only other seal is the gasket for the reactor cavity seal trough drain line. This gasket was replaced after the tests showed that it was leaking (see Figure A.5-2). However, the gasket leak was ruled out as the primary source of water observed in the sand bed drains because there is no clear leakage path to the seismic gap. Minor gasket leakage would be collected in the concrete trough below the gasket and would be removed by the drain line similar to leaks from the refueling bellows.

Additional visual and NDE (dye penetrant) inspections of the reactor cavity stainless steel liner identified a significant number of cracks, some of which were through-wall. Engineering analysis concluded that the cracks were most probably caused by mechanical impact or thermal fatigue and not intergranular stress-corrosion cracking. These cracks were determined to be the source of the refueling water that passes through the seismic gap. To prevent leakage through the cracks, an adhesive-type stainless steel tape was installed to bridge any observed large cracks, followed by application of a strippable coating. This repair greatly reduced the leakage and is implemented in every refueling outage when the reactor cavity is flooded.

In 1992, sand was removed from the sand bed region and loose rust cleaned from the drywell shell. An epoxy coating was then applied to the exterior surfaces of the drywell shell in the sand bed region (Ref. A.5.6.2). In addition, the concrete floor was rebuilt and reshaped with epoxy to allow drainage of any water that may leak into the region. Also, several commitments were to be implemented, as follows:

- The reactor cavity concrete trough drain will be verified to be clear from blockage once per refueling cycle. Any identified issues will be addressed via the corrective action process.

- Consistent with current practice, a strippable coating will be applied to the reactor cavity liner to prevent water intrusion into the gap between the drywell shield wall and the drywell shell during periods when the refueling cavity is flooded. This commitment applies to refueling outages prior to and during the period of extended operation

- Augmented inspections of the drywell will be performed in accordance with ASME Code (Ref. A.5.6.4) Section XI Subsection IWE that will consist of ultrasonic (UT) examinations of the upper region of the drywell and visual examinations of the protective coating on the exterior wall of the drywell shell in the sand bed region. Visual inspection of the coating will be supplemented with UT measurements from inside the drywell once before entering the period of extended operation and every 10 years thereafter.

- The sand bed region drains will be monitored on a daily basis during refueling outages and quarterly during the operating cycle

An additional commitment was made to perform an engineering study prior to the period of extended operation to investigate cost-effective replacement or repair options potentially eliminating reactor cavity liner leakage (Ref. A.5.6.2).

A.5.5 Structural Integrity Assessment and Test Results

In 1986, as part of an ongoing effort to investigate the impact of water on the outer drywell shell, concrete was excavated at two locations inside the drywell (referred to as trenches) to expose the drywell shell for ultrasonic assessment of its thickness. At an elevation of 3.12 m (10 feet, 3 inches) concrete floor slab level, 120 ultrasonic measurements were taken to characterize the vertical profile of corrosion in the sand bed region outside the shell and the minimum thickness of the shell. The minimum recorded thickness from the outside of the drywell was 15.7 mm (0.618 inches), while the minimum recorded thickness in the sand bed region from the inside was 15.3 mm (0.603 inches). These minimum recorded thicknesses are isolated local measurements and represent single-point ultrasonic measurements (Ref. A.5.6.2).

Additional inspections using ultrasonic thickness measurements were conducted during refueling outages and outages of opportunity between 1986 and 1989 to establish and characterize the extent of corrosion of the drywell shell. The initial ultrasonic measurements were not based on a sampling process but were taken in areas that correspond to locations where water leakage was observed from the sand bed region drains. The ultrasonic measurements were then expanded around the drywell perimeter and vertically to establish locations affected by corrosion. Approximately 1,000 ultrasonic thickness measurements were taken to identify the thinnest areas of the drywell shell. In addition, samples of the drywell shell were taken at seven locations, believed to be representative of general wastage, to confirm ultrasonic results. Based on the results of these inspections, elevations of 3.43 m (11 feet, 3 inches), 15.29 m (50 feet, 2 inches), and 26.64 m (87 feet, 5 inches) were identified for monitoring. The elevation of 3.43 m (11 feet, 3 inches), which corresponds to the sand bed region, showed the highest corrosion rate in 1987 (up to 0.993+/-0.086 mm per year (39.1+/-3.4 mils per year) based on 1986 and 1987 ultrasonic measurements. The high rate of corrosion in the sand bed region prompted removal of the sand as the corrective action. The high rate of corrosion in the sand bed region was attributed to galvanic corrosion of the drywell shell caused by water retained in the sand because of lack of proper drainage. The results of these measurements and subsequent analysis, which considered all design-basis loads and load combinations, confirmed that the “as found” condition of the drywell shell thickness satisfies ASME Code (Ref. A.5.6.4) Section III minimum thickness requirements.

During the Advisory Committee on Reactor Safeguards meeting on February 1, 2007, a commitment was made to perform a three-dimensional (3-D) finite-element analysis of the drywell shell prior to entering the period of extended operation (Ref. A.5.6.2). Sandia National Laboratories (SNL) developed a detailed 3-D finite-element model of the drywell containment vessel using information provided by the NRC and AmerGen. The model was used to evaluate the SI of the vessel in terms of the stress limits specified in ASME Code Section III, Division I, Subsection NE, and in terms of buckling (stability) limits specified in ASME Code Case N-284. The purpose of the SNL analysis was to examine whether the degraded drywell shell could withstand the postulated loadings without exceeding the ASME Code (Ref. A.5.6.4) requirements for stress and stability. The baseline (i.e., nondegraded) analysis was performed to isolate the effects of the degradation. The SNL analysis focused more on the relative reduction in design margin due to the corrosion than on the calculated absolute stresses or stability limits. The SNL study included stress and buckling analyses for both a representation of the containment in its degraded condition and in its original, as-built, condition. The analysis confirmed that the ASME allowable stresses are met for all three load cases examined

(Ref. A.5.6.2). The SNL analysis was followed by additional analyses prior to the period of extended operation, which confirmed the adequacy of the results (Ref. A.5.6.5).

An engineering stress and stability analysis was also conducted that demonstrated compliance with ASME Code requirements related to two scenarios: with and without sand. The analyses are documented in General Electric (GE) Reports Index Nos. 9-1, 9-2, 9-3, and 9-4 transmitted to the NRC staff between December 1990 and in 1991 (Refs. A.5.6.6 to A.5.6.9). Index Nos. 9-3 and 9-4 were revised later to correct errors identified during an internal audit and resubmitted to the NRC staff in January 1992 (Ref. A.5.6.2). In the NRC staff's safety evaluation report (SER) dated April 24, 1992, the NRC staff had made an assessment of the GE analysis for the load combination incorporating the refueling load and external pressure (Ref. A.5.6.10). The SER and attached technical evaluation report by Brookhaven National Laboratory documented the NRC staff's review of the increased capacity reduction factor due to the membrane tension and accepted the process of deriving the increased capacity reduction factor. The GE analysis assumed a uniform minimum thickness in the sand bed region of 18.7 mm (0.736 inches). The NRC staff found that the use of the increased capacity reduction factor described in the GE analysis was reasonable and consistent with ASME Code Case N-284 as well as ASME Code Section VIII, Code Case 2286 (Ref. A.5.6.2).

An additional assessment was performed to address water leakage from cracks in the stainless steel reactor cavity liner that were observed in the vicinity of the equipment pool and reactor cavity walls, indicating slight corrosion of the reinforcing bar. Based on a representative concrete core sample, it was conservatively estimated that the diameter of a typical reinforcing rebar in the localized area could be expected to be reduced by 0.051 mm (0.002 inches) per year. The walls in question were reinforced with No. 8 and No. 11 rebar. Assuming the corrosion continues for the entire 40-year life of the plant, the diameters of the No. 8 and No. 11 reinforcing bars would be reduced by 8 and 6 percent, respectively. The corrosion was localized and the reduced reinforcing bar diameters were judged to have no impact on the concrete integrity. Since the corrosion continued to be localized, it was concluded that there was no significant impact on the integrity of the reinforced concrete and that the integrity of the concrete will be maintained even if the reinforcing bar corrosion continues to the end of the period of extended operation. In addition to the assessment, the equipment pool and reactor cavity walls were visually inspected and there were no signs of water intrusion or additional indications of further deterioration.

A.5.6 References

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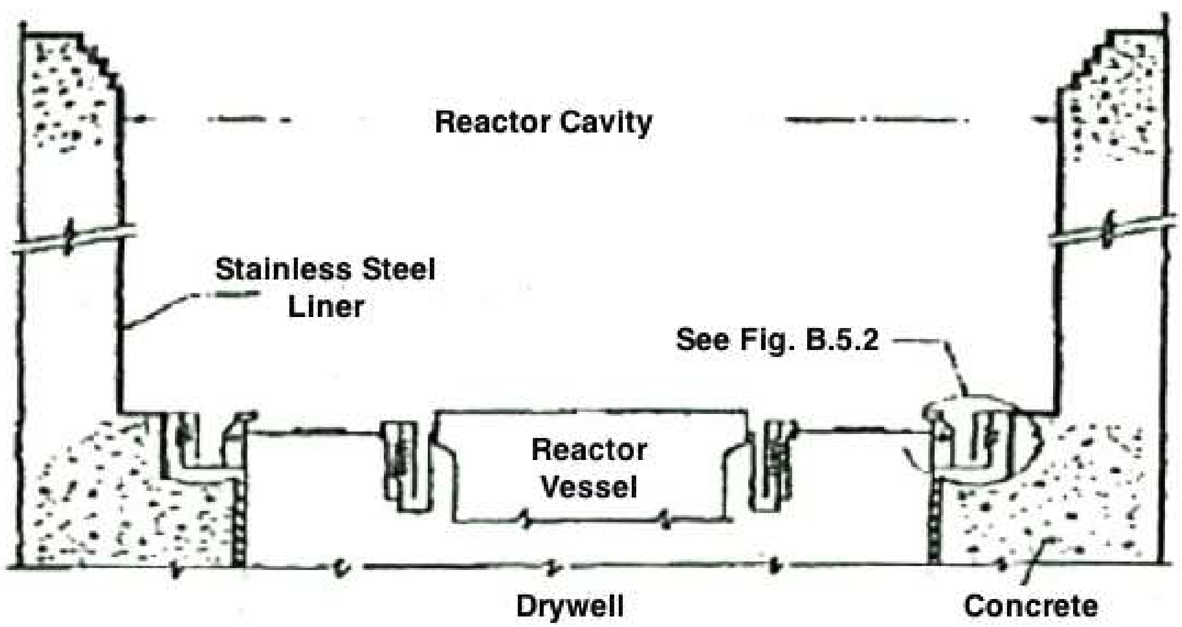

Figure A.5-1 Drywell and reactor cavity section

(Source: Gallagher, M.P., AmerGen Energy Company, LLC, letter to U.S. Nuclear Regulatory Commission, April 7, 2006.)

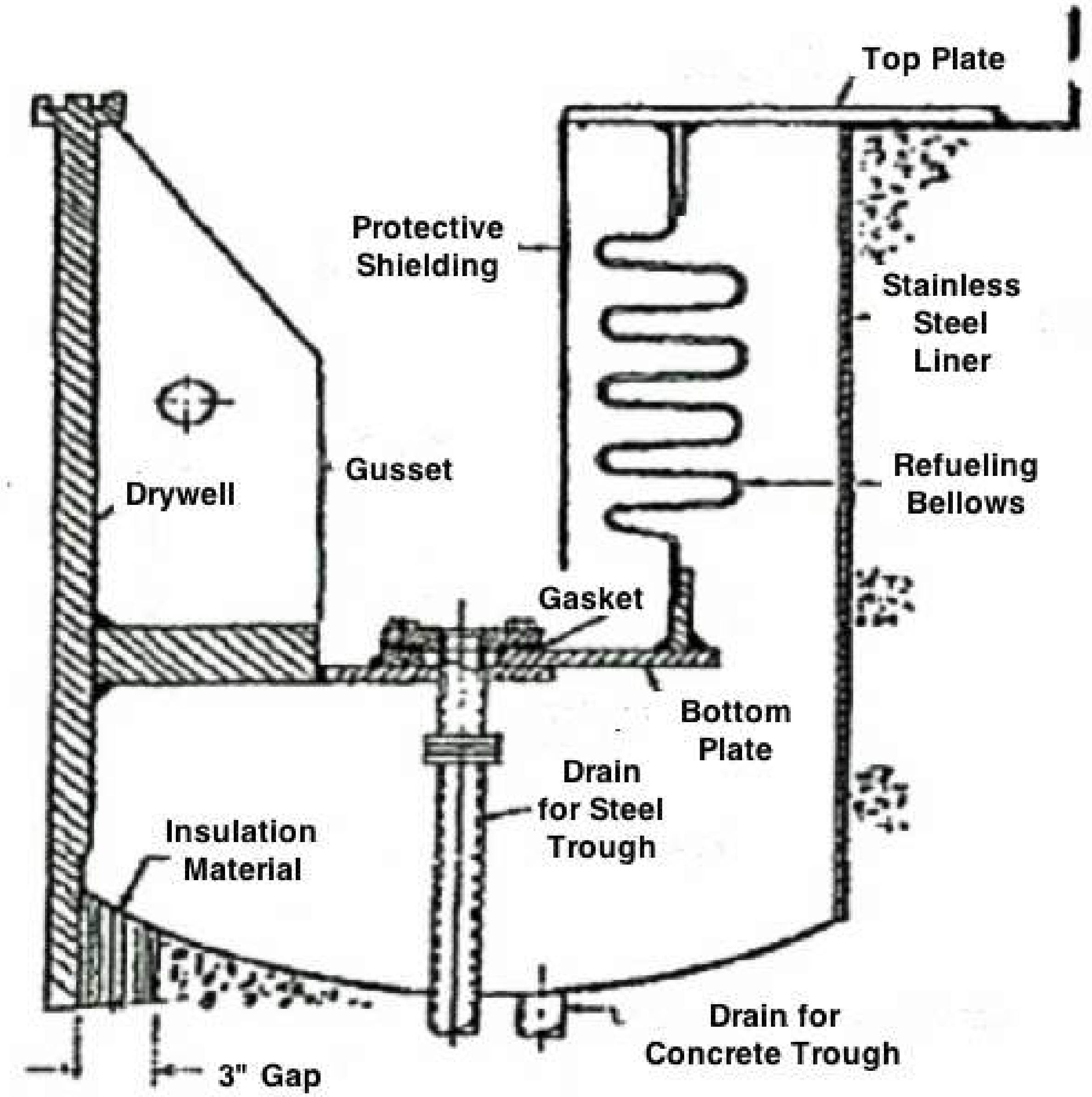

Figure A.5-2 Drywell to reactor cavity seal detail

(Source: Gallagher, M.P., AmerGen Energy Company, LLC, letter to U.S. Nuclear Regulatory Commission, April 7, 2006.)

A.6 Prairie Island Nuclear Generating Plant, Units 1 and 2

A.6.1 Introduction

On April 11, 2008, Nuclear Management Company, LLC, submitted an LRA to the NRC for Prairie Island Nuclear Generating Plant, Units 1 and 2 (Ref. A.6.6.1). The results of the NRC staff's evaluation of the LRA are provided in "Safety Evaluation Report with Open Items Related to the License Renewal of Prairie Island Nuclear Generating Plant Units 1 and 2," issued June 2009 (Ref. A.6.6.2) and NUREG-1960, "Safety Evaluation Report Related to the License Renewal of Prairie Island Nuclear Generating Plant Units 1 and 2," issued August 2011 (Ref. A.6.6.3).

A.6.2 Field Observations

Refueling cavity leakage was identified in both Units 1 and 2 (Ref. A.6.6.4). The leakage was detected by the ASME Code Section XI, Subsection IWE, program (Ref. A.6.6.5) while it was examining the Class MC pressure-retaining vessel (Refs. A.6.6.2 and A.6.6.3). Intermittent leakage indications occurred in both units since the late 1980s. The leakage indications typically began 2 to 4 days after the refueling cavity was flooded and ended approximately 3 days after the cavity was drained. A leakage rate of 1.1 to 2.2×10^{-6} m^3/s (1 to 2 gallons per hour (gph)) has been seen in the emergency core cooling system (ECCS) and regenerative heat exchanger room. The leaking fluid was chemically analyzed and determined to be similar to refueling water with a boron concentration of 2,700 parts per million (ppm), chloride concentration of 7 ppm, sulfate concentration of 0.2 ppm, and pH of 7.8. The increase in pH from the refueling cavity water to that found at the leaks was attributed to the acidity being neutralized by the carbonates and other minerals in the concrete. Water chemistry results taken at the reactor coolant drain tank floor area were similar except that the boron concentration was 5,329 ppm. This higher level was attributed to residual boron in the area from staining observed on the adjacent wall. Other potential sources of leakage, such as the reactor coolant, safety injection, and residual heat removal systems, were investigated and no other feasible source of leakage was identified.

During the Unit 1 refueling outage in 1999, leakage was detected at sump B, the sloped wall behind the reactor coolant drain tank, the ceiling above the regenerative heat exchangers, the reactor coolant pump vault, and the nuclear instrument detector at an elevation of 217.9 m (715 feet). Leakage inside the Unit 2 containment was first documented in 1998 during the refueling outage, when water was observed entering sump B from cracks in the grout around the residual heat removal suction penetration sleeves at an elevation of 211.8 m (694 feet, 10 inches). This area is grouted from the floor of the sump to the ceiling of the sump back to the containment vessel wall. The grout at sump B was removed to inspect the containment vessel wall, and no degradation was identified. Containment walkdowns also detected leakage at the basement level of containment (an elevation of 212.6 m (697 feet, 6 inches)) outside the north wall of the reactor coolant drain tank cubicle. Leakage from the reactor refueling cavity to the ECCS sump is postulated to occur as a result of flow under the refueling cavity liner through a construction joint between the floor of the transfer pit and through a wall behind the fuel transfer tube to the inner wall of the containment vessel, where it travels down and horizontally between the containment vessel and concrete to a low point of the containment vessel bottom head and then seeps through the grout into the ECCS sump. Leakage into the regenerative heat exchanger room is thought to occur from cracks in the concrete ceiling and walls. A potential leakage path and the locations of observed leakage are shown in Figures A.6-1 and A.6-2, respectively.

A.6.3 Design Characteristics of the Reactor Cavity

The auxiliary building is a multilevel, reinforced concrete and steel framed structure built on a mat foundation (Ref. A.6.6.1). Safety-related and nonsafety-related structures, systems, and components of the auxiliary building include the spent fuel pools and fuel transfer tubes. The reactor cavity, refueling cavity, and sump liners are stainless steel components.

A.6.4 Corrective Actions for Refueling Cavity Leakage

Grout was removed in 2002 from around the residual heat removal pipe sleeves in sump B to inspect the Unit 1 containment vessel steel. The action was similar to that taken in 1998 during the Unit 2 outage. Some discoloration around penetrations C30A and C30B was detected; however, no degradation of the penetrations or of the steel containment vessel was observed. Absent any degradation, no further action was taken in accordance with ASME Code Section XI, Subsection IWE.

In a letter dated December 5, 2008 (Ref. A.6.6.4), Northern States Power Company noted that the stainless steel reactor cavity liner was tested for faulty welds. Where leaks were found, repairs were made for both units. In addition to liner weld leaks, other actual and potential leak points were investigated, including the sand plug covers and bolts, neutron detector covers and bolts, fuel-lifting device bolts and baseplates, and other liner attachments. Inspections were limited to fully or partially accessible areas.

Sealing methods used to mitigate leakage have only been partially effective (Ref. A.6.6.6). A root cause evaluation of a leakage in 2009 indicated embedment plates for the reactor internals stands and the rod control cluster assembly change fixture to be the sources. Figure A.6-3 presents a cross section of the original embedment plate configuration and identifies potential leakage paths. Repair of Unit 1 embedment plates for the reactor internals stands and the rod control cluster assembly change fixture was a multistep effort. It included (1) removal of existing nuts and replacing with blind nuts, (2) seal welding of blind nuts to the baseplate, (3) application of a seal weld between the baseplate and the embedment plate, and (4) examination of welds by NDE. The repair for the embedment plate configuration is shown in Figure A.6-4. The repair eliminated the leakage source with no evidence of leakage in the ECCS sump; however, minor leakage (5.0×10^{-5} m^3/s (0.5 gph)) has been observed on the ceiling of the regenerative heat exchanger room after the refueling cavity had been flooded for 14 days. Vacuum box testing of its liner plate seam welds and NDE of the fuel transfer tube welds did not identify any leakage sources. Inspections expanded to include NDE of floor embedment fillet welds identified only one porosity indication. The rod control cluster assembly guide box wall embedment plates were identified as a potential source of leakage in Unit 1, and they were scheduled to be repaired during next refueling outage. Repairs for Unit 2 will be addressed with the same approach as those performed for Unit 1 (Ref. A.6.6.7).

A.6.5 Structural Integrity Assessment and Test Results

During the Unit 2 outage in 2008, over 150 ultrasonic thickness readings of the containment vessel from its exterior surface in the vicinity of the fuel transfer tube and above and behind the ECCS sump were obtained. All readings exceeded the two nominal vessel plate thicknesses of

38.1 mm (1.5 inches) and 88.9 mm (3.5 inches), respectively. Similar measurements for the Unit 1 containment vessel were performed in 2009. The results were the same (Ref. A.6.6.6).

During the refueling outage following embedment plate repairs, concrete will be removed from the sump below the reactor vessel to expose the containment vessel. With in situ concrete and steel reinforcement exposed, the area will then be inspected (i.e., visual and ultrasonic) and evaluated. In addition, concrete will be cored from an area wetted by borated water and tested to determine its compressive strength. Furthermore, petrographic examinations will be conducted on all removed concrete. A secondary purpose for removal of the concrete is to provide access for drainage of accumulated water. During the next two consecutive refueling outages following embedment plate repair, areas of each unit previously exhibiting leakage will be monitored to confirm that it has not reoccurred. Continued monitoring for leakage and degradation will be through periodic examinations of the containment vessels and interior surfaces performed in accordance with the ASME Code Section XI, Subsection IWE, program and the structures monitoring program (Ref. A.6.6.7). The corrective action program will address any new issues.

Assessments have also been made relative to the potential impact of the borated water on the reinforced concrete (Ref. A.6.6.8). Additional information is contained in two documents related to reactor refueling cavity leakage (Refs. A.6.6.9 and A.6.6.10). The upper-bound estimated loss of concrete behind the reactor refueling cavity liner was 7.87 mm (0.31 inches) (Ref. A.6.6.9). Because concrete walls are on the order of 1.2- to 1.5-m (4 to 5 feet) thick, the anticipated loss of load-carrying capacity would be minor. The impact of the borated water on the concrete embedded steel reinforcement was considered to be negligible because the construction joints do not have steel reinforcement across them (Ref. A.6.6.7), and the concrete cover exceeds the estimated depth of erosion. Figure A.6-5 presents an example of the condition of the concrete, steel reinforcement, containment shell, and grout after its removal in the Unit 1 ECCS sump in fall 2009.

The potential impact of concrete erosion on the liner was also evaluated (Ref. A.6.6.11). It was noted that the liner is effectively a membrane backed by 1.2- to 1.5-m (4- to 5-foot) thick concrete and that large areas of concrete washout are unlikely. However, if washout did occur behind the liner, the liner would not be expected to fail due to the ductile nature of stainless steel. In addition, visual inspections and vacuum box testing of the liner plate seam welds will be performed in the refueling cavity to look for depressions in the liner and for signs of concrete washout resulting from the reactor refueling cavity leakage.

An assessment of the containment vessel and concrete structures was performed (Ref. A.6.6.12) to determine the minimum wall thickness of potentially corroded areas of the containment vessel and the allowable concrete degradation that precludes a challenge to the functionality or SI of the containment vessels and internal structures under DBA and DBE conditions. The results indicated that both the 38.1-mm (1.5-inch) thick containment shell and the bottom head could tolerate general corrosion of approximately 12.7 mm (0.5 inch) with no significant risk to functionality. The concrete behind the transfer tube was also investigated, as it is one of the thinnest sections of concrete inside containment. Accurately determining the concrete thickness in this area was not possible using existing drawings, but based on rough scaling, the thickness could reach a minimum of about 254 mm (10.0 inches). Review of the containment internals structures drawings indicated that no significant loading of this section of concrete exists. A large concrete beam located above the transfer tube supports the dead weight of concrete above this section. Any loading from concrete above this section would result in compressive stresses only. A design-basis earthquake would result in minimal stress, as the thin section behind the transfer tube is approximately 3.05 m high by 0.30 m wide (10 feet high by I foot wide) and is tied to much heavier sections on four sides. As a result, a 30-percent margin for the section of concrete around the transfer tube is considered to be a conservative estimate consistent with other containment internal structures.

A.6.6 References

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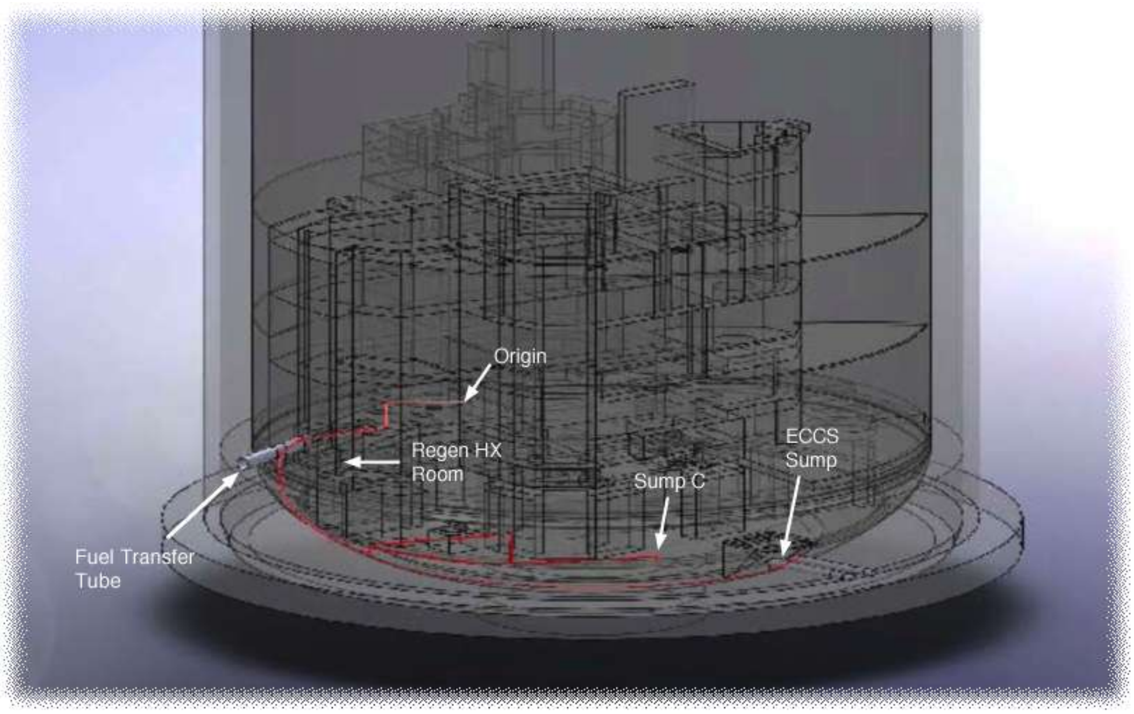

Figure A.6-1 Potential leakage paths for refueling cavity water

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee Prairie Island Nuclear Generating Station." Work Order NRC-2945. U.S. Nuclear Regulatory Commission: Washington, DC. July 7, 2009)

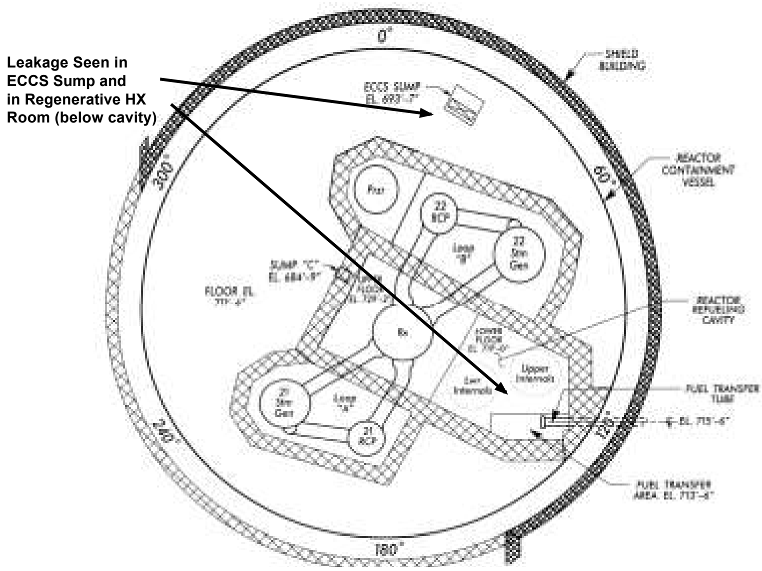

Figure A.6-2 Locations of observed leakage of refueling cavity water

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee Prairie Island Nuclear Generating Station." Work Order NRC-2945. U.S. Nuclear Regulatory Commission: Washington, DC. July 7, 2009)

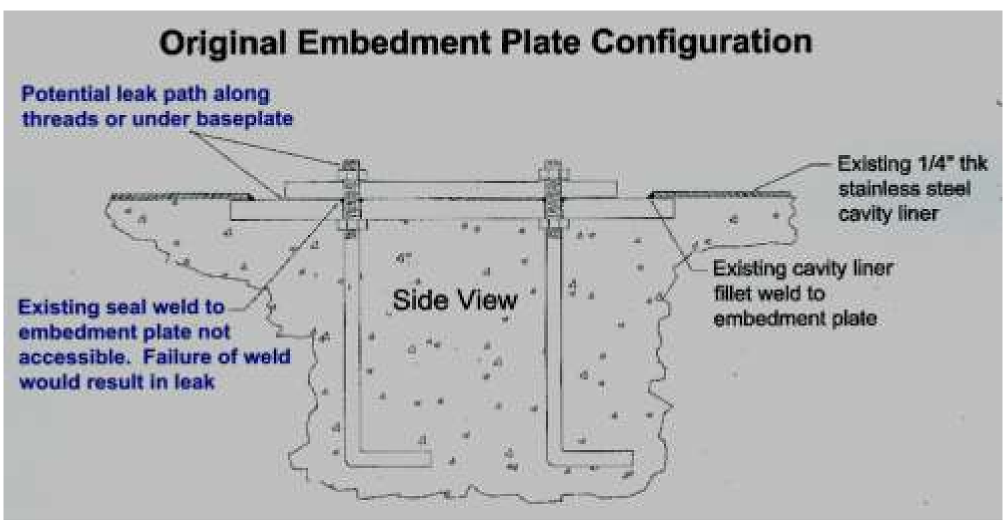

Figure A.6-3 Cross section of original embedment plate configuration showing potential leakage paths

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee Prairie Island Nuclear Generating Station." Work Order NRC-2945. U.S. Nuclear Regulatory Commission: Washington, DC. July 7, 2009)

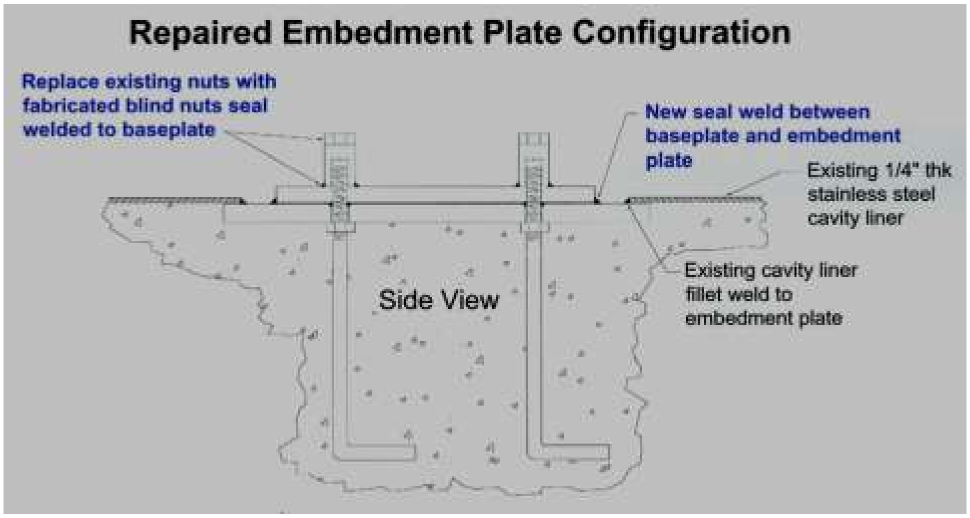

Figure A.6-4 Cross section of repair approach for embedded plates

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee Prairie Island Nuclear Generating Station." Work Order NRC-2945. U.S. Nuclear Regulatory Commission: Washington, DC. July 7, 2009).

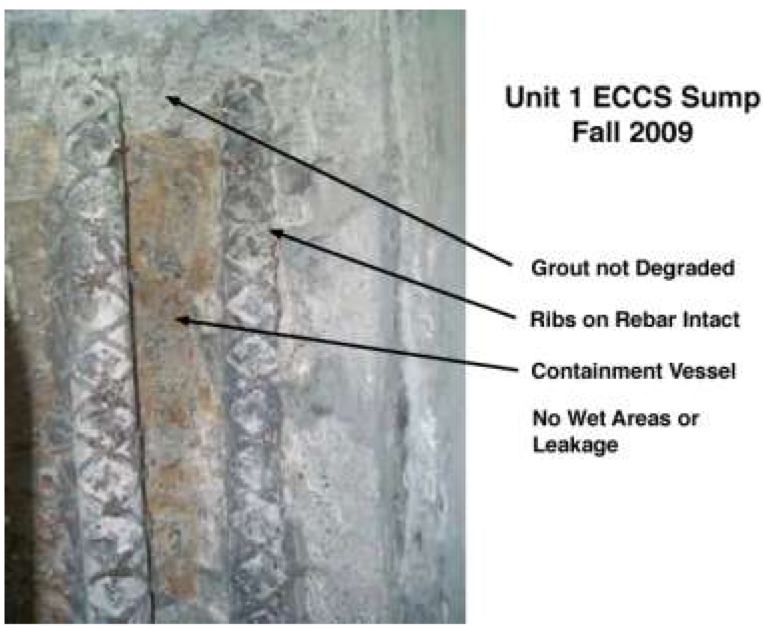

Figure A.6-5 Condition of containment shell, grout, and concrete steel reinforcement in Unit 1 ECCS after grout removal

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee Prairie Island Nuclear Generating Station." Work Order NRC-2945. U.S. Nuclear Regulatory Commission: Washington, DC. July 7, 2009)

A.7 <u>Salem Nuclear Generating Station, Units 1 and 2</u>

A.7.1 Introduction

On August 18, 2009, PSEG submitted an LRA to the NRC for Salem Nuclear Generating Station (Salem), Units 1 and 2 (Ref. A.7.6.1). The results of the NRC staff's evaluation of the LRA were published in June 2011 in NUREG-2101, "Safety Evaluation Report Related to the License Renewal of the Salem Nuclear Generating Station" (Ref. A.7.6.2).

A.7.2 Field Observations

Leakage of borated water has occurred in Salem Units 1 and 2 reactor cavities during refueling outages, but the leaks have been limited to the containment building (Ref. A.7.6.3).

During the 2005 refueling outage, white deposits were observed in several locations of the Unit 1 containment. Chemical analyses of the residue indicated the existence of boron and concrete constituents, pointing to a reactor cavity or fuel transfer canal borated water leakage. Followup walkdowns found evidence of a recent leak (water) in the N16 decay tunnel. Analysis of the water indicated that it was consistent with the reactor cavity water, which suggested blockage of the six fuel transfer canal telltale drain lines. After cleaning of the telltale lines, four of the lines produced leakage of 1 to 60 drops per minute (dpm), but the yield steadily decreased to zero during the refueling outage. No active leakage associated with the reactor cavity and fuel transfer canal liner was documented during the 2007 or 2008 refueling outages. However, during the 2008 refueling outage, another chemical analysis was performed on deposits collected from the N16 tunnel. Results again indicated that the collected residue originated from either the reactor cavity or the fuel transfer canal. During the 2010 Unit 1 refueling outage, no active leaks were observed.

For Unit 2, during multiple outages since November 2000, evidence of boric acid deposits was found on the containment liner under the fuel transfer canal. In November 2003, liquid was observed running down the containment liner plate and lagging under the fuel transfer canal and inside the containment, pooling eventually on the concrete floor. No evidence of corrosion was observed between the containment liner and the floor joint. A year and a half later, in April 2005 during a refueling outage, water was observed dripping down the wall on the Unit 2 containment liner plate. With two of the six telltales dripping and with white deposits observed in several locations on the letdown heat exchanger room walls, an inquiry was initiated. The investigation showed that there was no blockage in the two dripping telltales and that the reactor cavity was the contributor to the residue. During the 2006 refueling outage, a small amount of leakage was observed coming from a telltale in the letdown heat exchanger room. In the next refueling outage of Unit 2 in 2008, no active leakage was observed. However, during the 2009 Unit 2 refueling outage, a 60-dpm active leak was found from the telltale located above the door to the letdown heat exchanger room. Analysis of a sample of the water indicated that it was consistent with reactor cavity water.

A.7.3 Design Characteristics of the Refueling Cavity

The containment structure for Salem Generating Station, Units 1 and 2 includes the containment buildings and the containment internal structures. The containment buildings are reinforced concrete containments with a cylindrical wall, a foundation mat, and a hemispherical dome roof. Internal reinforced concrete components include the reactor cavity and primary shield walls, the fuel transfer canal, the secondary shield wall (polar crane wall), and the floor slab. The floor slab contains a shallow outer and inner trench designed to collect floor drainage and direct it to the containment building sumps. The containment building sumps and the trenches are lined with stainless steel plates and covered with stainless steel grating or perforated stainless steel plates to prevent debris from entering the sumps.

The reactor cavity is a reinforced concrete structure that houses the reactor and provides the primary shielding barrier. Internal surfaces of the cavity are lined with 6.35-mm (0.25-inch) thick stainless steel plate. The reactor cavity is filled with borated water during refueling to permit transfer of fuel elements underwater between the reactor and the spent fuel pool. The refueling floor slab is a 0.91- to 1.52-m (3- to 5-foot) thick reinforced concrete component. The slab covers the RCS compartments and provides access to the reactor cavity, the fuel transfer canal, the polar gantry crane, and other components supported on the floor. The floor slab design includes hatches with removable shield plugs to allow crane access to the reactor coolant pumps for maintenance.

A.7.4 Corrective Actions for Refueling Cavity Leakage

Chemical analyses of leakage collected during multiple refueling outages at both Units 1 and 2 identified the source of the leakage as the reactor cavity or the fuel transfer canal (Ref. A.7.6.4). Assessments for Units 1 and 2 have concluded that the potential cause of leakage is very small cracks in the reactor cavity liner or in the fuel transfer canal liner. The majority of the leakage enters the leak collection chases and drains through the telltales. Some of the telltales in the letdown heat exchanger room associated with the fuel transfer canal liner were observed to have active leaks during refueling outages. A second leakage path occurs in the vicinity of where the fuel transfer canal exits containment. The origin of the leakage is indicated as from the reactor cavity and fuel transfer canal liner, through the concrete construction joints and cracks, followed by percolation down the sides of the containment liner behind the lagging inside the containment.

The leakage only occurs when the reactor cavity and fuel transfer canal are flooded up for refueling. Active leaks have been observed sporadically only during refueling outages, with measured leakage rates less than 100 dpm. Based on the short duration of the refueling activities and the very long exposures needed to degrade reinforced concrete, it was concluded that remedial actions were not needed; however, the structures monitoring program and the ASME Code (Ref. A.7.6.5) Section XI, Section IWE, program will be used to ensure the continued integrity of the in-scope structures.

The structures monitoring program was enhanced to perform periodic inspection of the telltales associated with the reactor cavity and fuel transfer canal liner to ensure that they are free of significant blockage. When adequate leakage is collected, an analysis will be performed to determine its pH. A commitment was also made to perform augmented inspections under the fuel transfer canal where the containment liner is subjected to leakage. These inspections will be performed once per containment inservice inspection period as long as leakage is observed.

A.7.5 Structural Integrity Assessment and Test Results

An assessment of the long-term structural adequacy of the Salem Unit 1 fuel-handling building (FHB) reinforced concrete structure under potential prolonged exposure of the concrete and reinforcing steel to borated water has been conducted (Ref. A.7.6.4). The assessment made the following conclusions:

- The predicted depth of concrete degradation after 70 years of continuous exposure to borated water is 33 mm (1.3 inches).

- The degradation rate of the concrete at the concrete steel reinforcement interface is similar to the general rate of attack of concrete without rebar. Therefore, degradation of rebar at the construction joints or cracks will not spread rapidly along the rebar to impact its bond to concrete.

PSEG concluded that the findings for the FHB are directly applicable to the Unit 1 and 2 reactor cavity and fuel transfer canal reinforced concrete structures. The reactor cavity and fuel transfer canal are only filled with borated water during refueling outages, which occurs at each unit approximately 1 month out of every 18 months (about 5 percent of the operating cycle), since the Salem units perform refueling outages every 18 months. By contrast, the Unit 1 FHB assessment assumed continuous borated water exposure for 70 years, with a resulting depth of degradation of 33.0 mm (1.3 inches). Due to the shorter exposure period for the reactor cavity and fuel transfer canal concrete, the expected depth of concrete degradation will be substantially less; i.e., 7.4 mm (0.29 inches). Therefore, degradation of the reinforcing steel in the reactor cavity and fuel transfer canal reinforced concrete structure would be insignificant.

A.7.5.1 Structural Capacity

Based on experimental test results, it was concluded that degradation of the reactor cavity and fuel transfer canal reinforced concrete due to borated water leakage would be insignificant and, therefore, would have no impact on the intended function of these structures (Ref. A.7.6.4). No degradation has been detected during past inspections, and a concrete core will be taken from the spent fuel pool at a known leakage location to verify that no degradation has occurred.

A.7.6 References

A.7.6.1 Fricker, C.J., PSEG Nuclear, LLC, letter to U.S. Nuclear Regulatory Commission, August 18, 2009.

A.7.6.2 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station." NUREG-2101. U.S. Nuclear Regulatory Commission: Washington, DC. June 2011.

A.7.6.3 Joyce, T.P., PSEG Nuclear, LLC, letter to J. White, U.S. Nuclear Regulatory Commission, March 30, 2007.

A.7.6.4 Davison, P.J., PSEG Nuclear, LLC, letter to U.S. Nuclear Regulatory Commission, May 13, 2010.

A.7.6.5 American Society of Mechanical Engineers. "ASME Boiler and Pressure Vessel Code." American Society of Mechanical Engineers: New York, NY. 2007.

APPENDIX B: SPENT FUEL POOL LEAKAGE CASE STUDIES

B.1 <u>Crystal River Nuclear Generating Plant, Unit 3</u>

B.1.1 Introduction

By letter dated December 16, 2008, Florida Power Corporation submitted a license renewal application (LRA) to the U.S. Nuclear Regulatory Commission (NRC) for its Crystal River Nuclear Generating Plant, Unit 3 (Ref. B.1.6.1). In December 2010, the NRC issued a safety evaluation report (SER) with open items (Ref. B.1.6.2).

B.1.2 Field Observations

A baseline inspection of structures within the scope of the Maintenance Rule (Ref. B.1.6.3) was completed in 1997, and a hairline crack on the concrete wall of the spent fuel pool (SFP) was noted (Ref. B.1.6.1). A subsequent inspection of structures was completed in 2007 consistent with the program frequency of not exceeding 10 years. The hairline crack was reinspected and found to be dry and stable (i.e., not having increased in size since the last inspection).

In 2009, prior to a scheduled June walkdown, maintenance activities were completed to assure that the leak chase outlets were clean and not plugged. Operating experience logs in October of that year noted some of the leak chase lines as having boron accumulation at the outlet with ongoing leakage of less than 1 drop per minute, but none were plugged (Ref. B.1.6.4). Operating experience for SFP leakage indicated that the leakage was minimal at the lower end of the SFP “normal” water level range, but the leakage increased when the level was raised to the upper end of the “normal” range. Based on this experience, the SFP level is maintained at a low level to minimize liner leakage.

B.1.3 Design Characteristics of the Spent Fuel Pool

The auxiliary building is a reinforced concrete structure from an elevation of 29 meters (m) (95 feet) (an elevation of 18.6 m (61 feet) in the seawater inlet pits) to an elevation of 49.4 m (162 feet), with a sheet metal enclosed structural steel superstructure from an elevation of 49.4 m (162 feet) to an elevation of 63.7 m (209 feet). The auxiliary building partially surrounds the reactor building and contains the new fuel racks and two SFPs (Spent Fuel Pools A and B), as well as various safety-related equipment and components.

B.1.4 Corrective Actions for Spent Fuel Pool Liner Leakage

Preventative maintenance is periodically conducted (i.e., snake runs) to verify that each of the 19 leak chases is clear. Sampled deposits are analyzed for products of concrete degradation.

B.1.5 Structural Integrity Assessment and Test Results

The hairline crack in the SFP south wall will be inspected and monitored at a yearly interval.

B.1.6 References

B.1.6.1 Young, D.E., Progress Energy, letter to U.S. Nuclear Regulatory Commission, December 16, 2008.

B.1.6.2 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report with Open Items Related to the License Renewal of Crystal River Unit 3 Nuclear Generating Plant." U.S. Nuclear Regulatory Commission: Washington, DC. December 2010.

B.1.6.3 — — —. "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants." RG 1.160, Revision 2. U.S. Nuclear Regulatory Commission: Washington, DC. March 1997.

B.1.6.4 Franke, J.A., Progress Energy, letter to U.S. Nuclear Regulatory Commission, October 13, 2009.

B.2 Davis-Besse Nuclear Power Station, Unit 1

B.2.1 Introduction

On August 27, 2010, First Energy Nuclear Operating Company submitted an LRA to the NRC for its Davis-Besse Nuclear Power Station, Unit 1 (Ref. B.2.6.1).

B.2.2 Field Observations

During Cycle 13 reracking of the SFP, underwater divers used a vacuum box on the weld seams in the SFP to determine if there were any detectable leaks; none could be located. At that time, there was visible evidence of leakage in emergency core cooling system (ECCS) Pump Room No. 1, but little leakage was being seen in the leak chases. In February 2001, additional action was taken to open and verify that the 21 leak chase valves and piping were open. Six of the chases were found to be totally blocked. Upon clearing of their blockage, a significant amount of trapped fluid was drained. Leak chase channels exhibiting the largest drainage are kept constantly open, with the rest closed to reduce the likelihood of the boric acid solidifying and blocking the valves and piping.

One of the leak chase drains has consistently shown small amounts of leakage during the monthly test, as documented in the third quarter of the 2008 health report. Two other leak chase drains showed occasional leakage, but the leakage was small and the fluid was captured by the leak collection system. The corrective action program documented 140 milliliters (mL) of leakage collected during July 2008 for one zone valve. The leakage rate was calculated as 2.8 milliliters per minute (mL/min), which was higher than the trend data average of 1.0 mL/min over the previous 12 months. Based on a review of the trend data collected since 1999, occasional spikes in flow rate have occurred. The corrective action program item was designated for tracking and trending of a condition that occurs periodically in the plant. During the third quarter of 2008, the boron concentration appeared erratic in one sample, and the condition was documented in the corrective action program. Because the leak collection boron concentration is an information-only test, this condition was documented for trending purposes.

B.2.3 Design Characteristics of the Spent Fuel Pool

The auxiliary building includes the SFP, fuel transfer pit (also known as the transfer pit or fuel transfer tube pit), and cask pit walls and floors that are lined with 6.35-millimeter (mm) (0.25-inch) thick stainless steel liner plate. A watertight bulkhead gate separates the SFP from the fuel transfer pit and another separates it from the cask pit. Struts are installed on the walls between the fuel transfer pit and the SFP when the fuel transfer pit water level is below the bottom of the SFP bulkhead gate. The struts prevent the wall from becoming overstressed during a seismic event.

B.2.4 Corrective Actions for Spent Fuel Pool Liner Leakage

The leak chase monitoring program includes periodic monitoring (monthly) of the SFP, the fuel transfer pit, and the cask pit liners leak chase system. This routine task requires recording of the leakage amount collected, and the calculated leak rate. It also includes activities to cycle (i.e., open and close) the SFP, the fuel transfer pit, and the cask pit liner drain valves on a monthly basis. Each valve on the drain line capable of being cycled is opened to allow any water that has accumulated in the lines to drain into an open funnel. After a prescribed wait time, leakage is collected. The amount collected and the calculated leak rate are recorded for each of the 21 drain zones. If leakage collected from any zone drain valve is greater than 10 mL, then the sample is appropriately labeled and transported to a laboratory for boron analysis. Leak chase channel results are reviewed by the SFP system engineer. Collected leakage information and boron analysis results are recorded in the work order system. Monitoring of leakage from the leak chase system permits early determination and localization of any leakage.

For the period of extended operation, the site-specific program will be enhanced. In addition to previous monitoring for boron, samples having leakage rates of 15 mL/min will be documented in a condition report and evaluated for possible increases in monitoring frequency and other corrective actions. Furthermore, collected leakage will be analyzed monthly for its pH and semiannually for its iron content. The results will be monitored and trended to insure that there is no corrosion of the reinforcing bars in the walls or floors of the pool and pits. Furthermore, the enhanced program will annually inspect accessible outside walls and the floor (from the ceiling side) of the pool and pits, documenting indications of migrating leakage in the corrective action program. Furthermore, in 2014 and 2020, and as necessary thereafter for effective monitoring of steel reinforcement through the structures monitoring program, core bores of the affected areas will be collected for testing and evaluation of concrete strength, followed by visual inspections of the reinforcement and analysis of corrosion products. Degradations will be recorded in the corrective action program and evaluated to determine if any repairs are needed so that the SFP will continue to perform its intended functions during the period of extended operation. The core bore evaluations will also determine whether or not the SFP leakage has affected the concrete and reinforcing steel in a manner that is not bounded by the industry and Davis-Besse current operating experience (Refs. B.2.6.2 and B.2.6.3).

B.2.5 Structural Integrity Assessment and Test Results

Leakage outside the leak chase drains has been seen in several places over the years. The most extensive visible evidence of leakage was on the wall and ceiling of ECCS pump room No. 1 during the period from 2000 to 2001. This leakage has stopped and the area has been cleaned. Based on evaluations associated with this leak, there are no concerns regarding the strength or integrity of the concrete structure associated with these leaks.

B.2.6 References

B.2.6.1 Allen, B.S., FirstEnergy Operating Company, letter to U.S. Nuclear Regulatory Commission, August 27, 2010.

B.2.6.2 Byrd, K.W., First Energy Operating Company, letter to U.S. Nuclear Regulatory Commission, May 24, 2011.

B.2.6.3 Byrd, K.W., First Energy Operating Company, letter to U.S. Nuclear Regulatory Commission, August 17, 2011.

B.3 Diablo Canyon Power Plant, Units 1 and 2

B.3.1 Introduction

On November 23, 2009, Pacific Gas and Electric Company submitted an LRA to the NRC for the Diablo Canyon Power Plant, Units 1 and 2 (Ref. B.3.6.1). The SER (with open items) that documents the NRC staff's technical review of the LRA was published in January 2011 (Ref. B.3.6.2).

B.3.2 Field Observations

The Unit 1 SFP has occasional minor leakage primarily during refueling outages, and the Unit 2 SFP has persistent minor leakage that varies from 50 to 975 mL per week, with a slight increase in leakage rate during outages (Refs. B.3.6.3 and B.3.6.4), which is attributed to outage activities such as fuel handling, cask movements, and increases in water level (Ref. B.3.6.5). Based on the frequency and volume of leakage, samples are analyzed for tritium, gamma isotropic, pH, iron, and boron. With small quantities of leakage, there is not always sufficient volume to perform all of the analyses. When this occurs, the tests are performed based on the stated order. The results indicate that all concentrations are below health hazards or structural hazard levels, with concentrations below those recorded in the SFP.

B.3.3 Design Characteristics of the Spent Fuel Pool

The leak chases are located behind all liner joints (seam welds) for capturing water that potentially leaks through the liner and/or liner seams or plug welds (Ref. B.3.6.3). Any leakage through the liner is collected in the leak chases and is routed via gravity to a leakage monitoring station that has six collection points with isolation valves.

B.3.4 Corrective Actions for Spent Fuel Pool Liner Leakage

Video inspections of the Unit 1 and Unit 2 leak chase channels were conducted in 2008. No liner repairs have been performed on Unit 1 over the life of the plant. Followup video inspections were done on two of the leak chase channels for Unit 2 that were experiencing chronic minor leakage. The leak chases were reinspected in 2010. SFP liner leakage monitoring began in 1988, and the liner leak chases for both units are sampled and evaluated on a weekly basis (Ref. B.3.6.3).

B.3.5 Structural Integrity Assessment and Test Results

Evaluations to date are unable to conclusively identify the root cause of the leakage, and the structures potentially affected by the presence of borated water are the SFP concrete and structural steel (Ref. B.3.6.3). Previous engineering investigations concluded that the long-term leakage is acceptable and will have negligible effects on the concrete and reinforcing steel because the boric acid would result in slight surface scaling of the concrete having no cracks. The concrete in this state will protect the reinforcing steel from coming into contact with the boric acid.

B.3.6 References

B.3.6.1 Becker, J.R., Pacific Gas and Electric Company, letter to U.S. Nuclear Regulatory Commission, November 23, 2009.

B.3.6.2 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report with Open Items Related to the License Renewal of Diablo Canyon Nuclear Power Plant, Units 1 and 2." U.S. Nuclear Regulatory Commission: Washington, DC. January 2011.

B.3.6.3 Becker, J.R., Pacific Gas and Electric Company, letter to U.S. Nuclear Regulatory Commission, July 19, 2010.

B.3.6.4 Ferrer, N., U.S. Nuclear Regulatroy Commission, letter to J. Conway, Pacific Gas and Electric Company, June 21, 2010.

B.3.6.5 Becker, J.R., Pacific Gas and Electric Company, letter to U.S. Nuclear Regulatory Commission, September 30, 2010.

B.4 Duane Arnold Energy Center

B.4.1 Introduction

By letter dated September 30, 2008 (Ref. B.4.6.1), and supplemented by a letter dated January 23, 2009 (Ref. B.4.6.2), FPL Energy, Duane Arnold Energy Center, submitted an LRA to the NRC for the Duane Arnold Energy Center. The results of the NRC staff's evaluation of the LRA were published in November 2010 in NUREG-1955, "Safety Evaluation Report Related to the License Renewal of Duane Arnold Energy Center" (Ref. B.4.6.3).

B.4.2 Field Observations

The SFP at the Duane Arnold Energy Center has been leaking since at least 1994, and the leakage has been contained within the fuel pool liner drain system. No moisture or leakage has been found due to refueling bellows or fuel pool leakage. Inspections of the sand pocket drain lines have indicated that no moisture or leakage is present in the sand pocket area; however, in August 1985, moisture was detected in the inaccessible area on the exterior of the drywell shell in the torus room near downcomer/vent line penetration X-05C. The leakage rate was estimated at approximately 1.1×10^{-6} cubic meters per second (m^3/s) (1 gallon per hour). Chemical analysis was conducted, but it was not sufficient to confirm or disprove that the source of the leakage was the SFP; however, no other source was plausible (Refs. B.4.6.4 and B.4.6.5). It has been noted that walkdowns had been completed in accessible areas under the pool and no leaks were discovered.

B.4.3 Design Characteristics of the Spent Fuel Pool

The reactor building encloses the reactor, primary containment new and spent fuel storage pools, and other auxiliary systems associated with the nuclear steam supply system. The reactor building provides secondary containment for the reactor when in service and primary containment for the auxiliary systems and the reactor during periods when the primary containment is opened for refueling and servicing. Leakage from the SFP is channeled into one or more drainpipe lines to monitor leakage, and the drains are routed to the reactor building floor drain sump through a common trough.

B.4.4 Corrective Actions for Spent Fuel Pool Liner Leakage

In May 1990, a pinhole leak near the toe of a control rod drive line fillet weld to the drywell shell was found to be the source of moisture detected in the inaccessible area on the exterior of the drywell shell. Subsequent investigations found flaws in the southwest control rod drive penetration bundle. Ultrasonic testing (UT) of the drywell shell in the affected area did not indicate any loss of thickness due to corrosion. In addition, no leakage was identified at the other three control rod drive penetration bundles. Repairs were satisfactorily made to the southwest control rod drive bundle in 1990. Since 1990, no recurrence of control rod driveline leakage has been experienced or identified.

B.4.5 Structural Integrity Assessment and Test Results

No structural assessment has been reported.

B.4.6 References

B.4.6.1 Anderson, R.L., FPL Energy-Duane Arnold Energy Center, letter to U.S. Nuclear Regulatory Commission, September 30, 2008.

B.4.6.2 Anderson, R.L., FPL Energy-Duane Arnold Energy Center, letter to U.S. Nuclear Regulatory Commission, January 23, 2009.

B.4.6.3 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of Duane Arnold Energy Center." NUREG-1955. U.S. Nuclear Regulatory Commission: Washington, DC. November 2010.

B.4.6.4 Harris, B.K., U.S. Nuclear Regulatory Commission, letter to C. Costanzo, Florida Power & Light Company, September 14, 2009.

B.4.6.5 Costanzo, C.R., NextEra Energy Duane Arnold, LLC, letter to U.S. Nuclear Regulatory Commission, October 13, 2009.

B.5 Hope Creek Generating Station, Unit 1

B.5.1 Introduction

On August 18, 2009, PSEG Nuclear, LLC (PSEG), submitted an LRA to the NRC for renewal of the Hope Creek Generating Station (Ref. B.5.6.1). The results of the NRC staff's evaluation of the LRA were published in June 2011 in NUREG-2102, "Safety Evaluation Report Related to the License Renewal of Hope Creek Generating Station" (Ref. B.5.6.2).

B.5.2 Field Observations

Minimal leakage has been detected when the pool level is increased above the normal level and, based on subsequent inspections of the area around the pool, all of the leakage is contained within the SFP drain system.

B.5.3 Design Characteristics of the Spent Fuel Pool

The reactor building includes the spent fuel storage pool liner, cask loading pit liner, reactor cavity liner, steam dryer/moisture separator storage pool liner, and spent fuel storage pool skimmer surge tank liner. The fuel pool liner drains are connected to the area under the fuel pool liner for leak detection.

B.5.4 Corrective Actions for Spent Fuel Pool Liner Leakage

No corrective actions have been reported.

B.5.5 Structural Integrity Assessment and Test Results

The SFP liner plate leak chase system has not been included within the scope of license renewal because leak collection channels are not safety-related and are not part of the water-retaining boundary, nor are they required to maintain the structural integrity of the SFP walls (Refs. B.5.6.3 and B.5.6.4). The leak chase system is not relied upon in safety analyses or plant evaluations to perform a safety function. Therefore, the SFP liner plate leak chase system and its components do not have a license renewal intended function.

B.5.6 References

B.5.6.1 Fricker, C.J., PSEG Nuclear, LLC, letter to U.S. Nuclear Regulatory Commission, August 18, 2009.

B.5.6.2 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of Hope Creek Generating Station." NUREG-2102. U.S. Nuclear Regulatory Commission: Washington, DC. June 2011.

B.5.6.3 Ashley, D.J., U.S. Nuclear Regulatory Commission, letter to T. Joyce, PSEG Nuclear, LLC, March 31, 2010.

B.5.6.4 Davison, P.J., PSEG Nuclear, LLC, letter to U.S. Nuclear Regulatory Commission, April 22, 2010.

B.6 Indian Point Nuclear Generating Unit 2

B.6.1 Introduction

By letter dated April 23, 2007 (Ref. B.6.6.1), and as supplemented by letters dated May 3 (Ref. B.6.6.2) and June 21, 2007 (Ref. B.6.6.3), Entergy Nuclear Operations, Inc., submitted an LRA to the NRC for Indian Point Nuclear Generating (Indian Point) Units 2 and 3. The results of the NRC staff's evaluation of the LRA were published in November 2009 in NUREG-1930, "Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3" (Ref. B.6.6.4).

B.6.2 Field Observations

During a 1990 pool reracking project at Unit 2, a small hole occurred in the northeast area of the SFP stainless steel liner at about the 27.1-m (89-foot) level. The damage was discovered in 1992 when boron powder was found on the SFP east exterior concrete wall. Leakage through the hole in the pool liner was estimated at 8.6 to 12.9×10^{-7} m^3/s (20 to 30 gallons per day (gpd)), which was unnoticed due to the much larger volume of normal evaporative loss from the pool (Ref. B.6.6.5).

On August 22, 2005, additional leakage was identified during the excavation of the fuel-handling building (FHB) adjacent to the SFP south wall to install a higher capacity gantry crane, needed to load and transport fuel casks in support of the independent spent fuel storage installation project (Ref. B.6.6.6). Workers had identified cracks in the wall of the Unit 2 SFP and observed a small amount of moisture in a 0.16-mm wide by 2.13-m long (0.0156-inch wide and 7-foot long) crack on the south wall at an approximate elevation of 19.8 m (65 feet). Moreover, ground water contamination was detected in onsite monitoring wells.

During the next 2 weeks, as excavation continued, a second crack was discovered at the 18.3-m (60-foot) elevation and a temporary collection device was installed to capture leaking liquid. The cracks were visually inspected by a civil-structural engineer and the Supervisor of Civil-Mechanical Engineering. The condition of the cracks was characterized as typical of shrinkage cracking witnessed during postconstruction concrete curing. The crack weeping gradually increased following the first measurable liquid sample of 12 mL collected in September 2005. Radiological analysis of this water sample confirmed tritium and boron content consistent with the Unit 2 SFP. The analysis also revealed that the moisture contained trace amounts of cesium-134 and -137, cobalt-60, and boron. The boron concentration was about 6 to 15 times less than the concentration in the SFP, and the cesium-134 to -137 concentration indicated that the activity was about 6 to 10 years old. The crack had calcium stains emanating from it, as is expected when moisture leaches through concrete, and there was no visual evidence of steel corrosion products (rust). Due to the thickness of the SFP wall, the amount of steel reinforcement, and the lack of evidence that the small amounts of moisture and boron have caused corrosion of the reinforcing bars, it was concluded that there was reasonable assurance that the SFP wall was structurally sound and capable of performing its intended function (Ref. B.6.6.5). During the next several weeks, the cracks exhibited increased leakage to a maximum of between 1 and 2 liters per day. This rate of leakage remained stable and then declined to a minimal amount by late December 2005 (Ref. B.6.6.7). Based on an analysis, it was determined that the seepage from the crack was consistent with SFP water (Ref. B.6.6.7).

On September 20, 2005, the NRC initiated a special inspection team in accordance with a special inspection charter, to investigate the structural and radiological implications of the observed Unit 2 SFP leakage and assess the licensee's corrective measures, radiological evaluation, and investigative actions (Refs. B.6.6.8 and B.6.6.9). The NRC reviewed the conditions on site and concluded that there were no near-term safety issues. Specifically, an NRC structural specialist and a health physics inspector were sent to the Indian Point facility to assist the resident inspectors in monitoring the progress of the investigation. In early October 2005, the NRC established Web pages for information on SFP issues at Indian Point (Ref. B.6.6.10).

In January 2008, the sources of ground water contamination were noted to be the Unit 1 and 2 SFPs (Ref. B.6.6.11). While both pools contributed to the tritium contamination of ground water, the leaks from the Unit 1 SFP were determined to be the source of other contaminants, such as strontium-90, cesium-137, and nickel-63.

B.6.3 Design Characteristics of the Spent Fuel Pool

The SFP wall at Unit 2 consists of thick concrete that is heavily reinforced with its inside surface lined with 6.35-mm (0.25-inch) thick stainless steel plates that are anchored to the concrete such that there is only a small interstitial space or air gap between the liner and the concrete (Ref. B.6.6.12). Unit 2 was designed and licensed without an SFP liner leak collection system. The design provisions for the Unit 2 SFP include pool-level instrumentation with alarms in the control room and 9.5×10^{-3} m^3/s (150 gallons per minute) water makeup capacity in the event of a design-basis accident (Ref. B.6.6.12).

B.6.4 Corrective Actions for Spent Fuel Pool Liner Leakage

All fuel from the Unit 1 SFP was removed to an onsite dry storage location and the Unit 1 SFP was drained, thereby essentially eliminating the source of the ground water contamination from Unit 1 (Ref. B.6.6.8).

A one-time inspection of the accessible areas of the Unit 2 SFP was conducted beginning in 2006 (Ref. B.6.6.13). Approximately 40 percent of the liner was accessible for inspection. Inspection techniques included use of robotic cameras, general visual, and vacuum box testing. Vacuum box testing was used on areas of the liner that were deemed compromised (e.g., exhibited indications) based on the general visual and robotic camera inspections. None of the suspect areas in the SFP area failed the vacuum box test, indicating that none of the indications found were actually leaking (Ref. B.6.6.12). Identified indications were coated as a precautionary measure. Essentially 100 percent of the SFP transfer canal liner was inspected using the same techniques as used in the SFP, with the addition of UT where applicable. The inspections discovered several indications and one weld defect in the transfer canal liner. The weld defect failed the vacuum box test. The defect and the indications were repaired. The evaluation concluded that the defect and indications were the result of poor construction practices and workmanship during the initial construction activities. The combined inspections of the SFP and the SFP transfer canal were completed in 2007.

The method to be used to determine if a degraded condition exists for leakage during the period of extended operation is to continuously monitor the SFP water level and the chemistry of the ground water in the vicinity of the pool exterior walls (Refs. B.6.6.14 and B.6.6.15). The absence of leakage will affirm that no degraded conditions exist. If leakage is found, it will be evaluated under the corrective action program. If sampling indicates that ground water contains constituents associated with the SFP leakage, then an evaluation is required under the corrective action program to assess the potential for degradation and to determine appropriate corrective actions, including inspections of all accessible surfaces of the SFP liner, installation of monitoring wells in the vicinity, performance of UT examinations, concrete core bore sampling, rebar inspections, and inspections necessitating the use of remote camera technology.

B.6.5 Structural Integrity Assessment and Test Results

Actions were taken to determine the source of moisture and potential amount and extent of related soil contamination. Implementation of these actions began in September 2005 and led to the establishment of a remediation and repair plan and schedule (Ref. B.6.6.5). Specifically, the licensee is to do the following:

- Determine rebar location in relation to cracks using a rebar detection device.

- Hand-drill using a small-diameter bit several centimeters (inches) into the SFP wall in the area of the moist crack and analyze drill-bit fines for contamination. The task was completed in 2005 and showed that the collected fines appeared to be damp in the first several inches of depth and then appeared to be dry.

- Place a plastic covering over the moist cracks to attempt to capture a sufficient volume of liquid for radiochemistry analysis. The task was completed in 2005 and yielded a 12-mL sample that contained low levels of cesium-137 and cobalt-60, 1,265 parts per million (ppm) boron, and approximately 0.02 microcurie/mL tritium. These results are indicative of moisture originating from the pool.

- Sample and analyze the soil beneath the area of the crack for tritium. The task was completed in 2005 and indicated low levels of tritium near (within 0.3 m (12 inches)) the wall, but the levels decreased at 0.6 to 0.9 m (2 to 3 feet) away from the wall to nearly an undetectable level.

- Scrape material from an unaffected area of the SFP wall and test for boron content. For this effort, dry fines from drilling were used. The analysis indicated low levels of boron (less than 400 ppm).

- Determine the typical level of boron in clean concrete. Although this effort was undertaken, there is no information available.

- Determine the expected corrosion rates for steel reinforcing rods subjected to an environment containing boron.

- Gather historical documentation of SFP stainless steel liner damage and SFP sump overflows. Some liner damage information has been recovered, including Calculation CGX-00006 (structural evaluation of the Unit 2 fuel pool wall) (Ref. B.6.6.16) and Technical Report ME-3802 (Ref. B.6.6.17) (evaluation of SFP walls for Indian Point Unit 2). These are considered bounding for the current situation in terms of wall and rebar structural integrity. To date, no records on sump overflows have been recovered. Only tribal knowledge has been collected.

- Identify through a ground-penetrating radar inspection (to the extent possible) any crack depths. Ground-penetrating radar was determined to not be feasible for this task. The task was completed in 2005 by taking two 101.6-mm (4-inch) diameter cores in the area of the moist crack. One appeared to be dry, and it was presumed that this was affected by boring bit heating. Twenty-four hours later, however, the core was damp. The rebar exhibited normal surface oxidation. Visual inspections over 3 days indicate a reduction in moisture.

- Gather the radiological results of test core borings performed for dry cask storage inside fuel storage building loading bay (4), including its access road. The task, completed in 2003, indicated the existence of low-level cesium-134, cesium-137, cobalt-58, and cobalt-60 surface contamination.

- Bring in an expert structural engineer from ABS Consulting with past experience in SFP leakage. The deliverables for this task, completed in 2005, were calculations of the seepage rate.

- Contact James A. FitzPatrick Nuclear Power Plant to obtain recent operating experience of its leaking SFP liner. The task was completed in 2005 through a conference call. No additional actions were suggested; however, Indian Point found out that James A. FitzPatrick Nuclear Power Plant had an active pool liner leak earlier in the year. The leakage appeared to cease on its own, indicative of a potential pinhole forming and subsequently clogging, or the leak path (e.g., crack in concrete) closing in some manner.

- Inspect other accessible exterior areas of the SFP walls for residues/white material. Other accessible areas in addition to the south wall include the west wall and the east wall outside where the 1990–1992 leak was discovered. The east wall has no evidence of a problem subsequent to the 1990–1992 leak discoveries. The west wall in the pipe pen has some cracking and dry white streaking with no evidence of moisture. A sampling of the white material was to be completed by 2005. The west wall has some shrinkage cracking, but there is no evidence of moisture.

Underwater camera inspections of portions of the Unit 2 SFP were initiated on October 27, 2005. Three areas appearing as potential flaws in the liner were identified in the southwest corner of the pool. By December 2005, visual examination and videotaping of about 50 percent of the SFP liner surface was completed in an effort to identify locations of potential leakage. This represents all of the surfaces that can be accessed with the currently available video camera equipment because stored spent fuel in portions of the pool prevented full examination. Divers physically examined three locations of interest with a vacuum box to determine if leakage was present. In each case, however, leakage was not positively detected. Methods to apply an underwater coating to these locations to assure that these areas of interest remain leak-tight were investigated (Ref. B.6.6.18).

In August 2008, the NRC was informed that all known sources of leakage from the Unit 2 SFP have been eliminated based on completed inspections and repairs, and that a one-time inspection of the accessible 40 percent of the SFP liner above the fuel racks and 100 percent of the SFP transfer canal liner was completed in 2007 using general visual, robotic cameras and vacuum box testing techniques (Ref. B.6.6.13). It was noted that ground water outside the

Unit 2 SFP would be tested every 3 months for the presence of tritium using samples taken from adjacent monitoring wells (Ref. B.6.6.4).

In September 2005, it was concluded that, due to the thickness of the SFP walls, the amount of steel reinforcement, and the small volume of moisture and boron percolating or seeping through, there was a reasonable assurance that the SFP wall was structurally sound and capable of performing its intended function (Ref. B.6.6.5). The data collected so far appear to indicate the presence of pinhole(s) in the SFP liner. The location of and whether the leak path is active was not determined.

In February 2006, The NRC completed an inspection at Unit 2 and issued NRC Special Inspection Report No. 05000247/200501 that stated “No safety significant findings were identified” for the structural integrity of the Unit 2 SFP. The inspector examined the small hairline cracks on the south wall of the SFP in assessing the structural integrity of the Unit 2 SFP. The following documents were included in the NRC review (Ref. B.6.6.7):

- Consolidated Edison Calculation No. CGX-00006-00, “Seismic Qualification Structural Evaluation of the Unit 2 Fuel Pool Wall Considering Deteriorated Condition of Concrete Due to Pool Leak”

- United Engineers and Constructors Technical Report No. 8281, “Evaluation of Spent Fuel Pool Walls - Indian Point 2 Nuclear Power Plant”

- ABS Consulting Report 1487203-R-001, “Study of Potential Concrete Reinforcement Corrosion on the Structural Integrity of the Spent Fuel Pit.” September 2005.

In November 2008 (Ref. B.6.6.19), a design margin assessment for the Unit 2 SFP reinforced concrete east and south walls was performed. The capability of the east SFP pit wall and the south SFP pit wall to resist the design-basis loads was evaluated. The assessment considered potential concrete and reinforcement steel degradation due to observed leakage of fluids through these walls. Core boring samples were obtained and tested to evaluate concrete properties and to provide access for visual inspection of the concrete and steel reinforcement. Finite-element models for both the east and south walls were developed to determine the actual forces in the walls due to loading resulting from the design-basis earthquake, hydrostatic forces, and dead weight. Due to the symmetry of the spent fuel pit structure, results from the evaluation of these two walls are applicable to the remaining north and west walls.

The capacity of the east wall was evaluated in response to possible degradation due to an observed leak in 1992. It was determined that work in the SFP in 1990 initiated the leak by inadvertently creating a small hole in the stainless steel liner. This condition was repaired in 1992. A total of 20 core bores were taken from five locations on the east wall in the vicinity of the observed leakage to determine the condition of the concrete following exposure to borated water leakage. At each of the five locations, four individual cores of 101.6 mm (4 inches) in diameter and 381 mm (15 inches) in length were taken, resulting in a total depth of penetration into the wall of 1.52 m (60 inches). In addition, several windows in the outer surface of the wall were created to allow inspection of the outer layer of reinforcing steel. Of the 20 cores taken, all but one had compressive strengths that exceeded the design strength of 20.7 megapascals (MPa) (3,000 pounds per square inch (psi)). This one core outlier had a measured compressive strength of 16.5 MPa (2,400 psi).

The coring showed that the borated water had little or no effect on the concrete itself. Little or no corrosion was also observed in the steel reinforcement except at a location in the wall where spalling had occurred, exposing the steel to the environment. Analysis of the rust particles showed high chloride content and low boron concentration, indicating that rainwater was the primary cause of the observed corrosion. To determine the available margin in the east wall, moments were calculated using a finite-element plate model. The results of the analysis showed that the east wall was capable of resisting the applicable forces without any reinforcing steel and would incur little or no cracking as a result of the design loading. Conservatively assuming that the concrete would crack and the bending moments would be carried by the reinforcing steel, it was concluded that the load-bearing capacity of the wall is at least 31 percent greater than required.

The structural margins in the south wall due to possible concrete steel reinforcement degradation as a result of observed fluid emanating from a crack discovered in the west corner during excavation for the dry cask storage project were evaluated. The reinforcing steel in the area of the observed leak was exposed for inspection and noted to have little or no corrosion. To determine the actual forces in the south wall due to the design-basis loads, a finite-element model of the wall was also developed. Based on the resulting moments from the analysis, the margins in the south wall with respect to the ultimate moment capacity of the concrete section were determined. The available margins in the east and south walls of the SFP pit with respect to the as-designed condition ranged from a low of 25 percent at the base of the wall for the vertical steel to a high of 57 percent for the vertical steel at the crack location in the west corner of the wall. The margins for the horizontal rebar at wall mid span ranged from 43 percent to 45 percent and up to 51 percent in the vicinity of the observed crack.

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B.6.6.4 U.S. Nuclear Regulatory Commission. “Safety Evaluation Report Related to the License Renewal of Indian Point Nuclear Generating Unit Nos. 2 and 3.” NUREG-1930, Volumes 1 and 2. U.S. Nuclear Regulatory Commission: Washington, DC. November 2009.

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B.6.6.7 Blough, A.R., U.S. Nuclear Regulatory Commission, letter to F.R. Dacimo, Entergy Nuclear Operations, Inc, March 16, 2006.

B.6.6.8 U.S. Nuclear Regulatory Commission. "NRC Performing Special Inspection at Indian Point 2 Nuclear Plant; Small Amount of Leakage from Spent Fuel Pool Area under Review." *NRC News*, No. I-05-049. U.S. Nuclear Regulatory Commission: Washington, DC. September 20, 2005.

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B.6.6.10 U.S. Nuclear Regulatory Commission. "NRC Establishes Web Page for Information on Spent Fuel Pool Issues at Indian Point." *NRC News*, No. 05-136. U.S. Nuclear Regulatory Commission: Washington, DC. October 4, 2005.

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B.6.6.13 Dacimo, F., Entergy Nuclear Northeast-Indian Point Energy Center, letter to U.S. Nuclear Regulatory Commission, August 14, 2008.

B.6.6.14 Green, K., U.S. Nuclear Regulatory Commission, letter to Vice President, Operations, Entergy Nuclear Operations, Inc.-Indian Point Energy Center, April 3, 2009.

B.6.6.15 Dacimo, F., Entergy Nuclear Northeast-Indian Point Energy Center, letter to U.S. Nuclear Regulatory Commission, May 1, 2009.

B.6.6.16 G. Schwartz. "Hairline Crack in IPEC Unit 2 Spent Fuel Pool South Wall." Calculation CGX-00006, Revision 4. Indian Point Energy Center. September 15, 2005. Available at http://pbadupws.nrc.gov/docs/ML0532/ML053270506.pdf.

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B.7 Kewaunee Power Station

B.7.1 Introduction

On August 12, 2008, Dominion Energy Kewaunee, Inc. submitted an LRA to the NRC for its Kewaunee Power Station (Ref. B.7.6.1). The results of the NRC staff's evaluation of the LRA were published in January 2011 in NUREG-1958, "Safety Evaluation Report Related to the License Renewal of Kewaunee Power Station" (Ref. B.7.6.2).

B.7.2 Field Observations

As discussed in the SER, Section 3.0.3.2.18 (Ref. B.7.6.2), it was noted in a letter dated August 17, 2009 (Ref. B.7.6.3), that, after the identification of white deposits on the wall and ceiling of the waste drumming room adjacent to the SFP in December of 2007, meetings were held to discuss fuel pool makeup, housekeeping and contamination, ground water leakage concerns, and the possibility of structural degradation. The affected areas were cleaned and put under observation to find the cause of the condition and establish a corrective action plan. During monitoring, the residue occurred again in the same area after it had been cleaned, but there was no active dripping. In June 2008, monitoring and troubleshooting of the area began and included monthly visual inspections examining the change in size, shape, and color of the deposit through time-lapse photography.

After a year of monitoring the wall and ceiling of the waste drumming room, it was concluded that the residue formation remained constant. The residue formation rate was slow and, therefore, it was determined that there was no near-term concern for the integrity of the structure or potential loss of intended function. Actions would be implemented if any change in leakage trend or other signs of concrete distress were observed. Additional monitoring of the ground water to date indicates no detectable level of tritium outside the auxiliary building or in the ground water (Refs. B.7.6.4 and B.7.6.5).

B.7.3 Design Characteristics of the Spent Fuel Pool

The SFP is located at an intermediate elevation in the auxiliary building, 4.57 m (15 feet) above the basement floor. The auxiliary building is a concrete and steel multistory structure that interfaces with the shield building and turbine building. The SFP receives spent fuel from the reactor containment vessel through the fuel transfer tube. The penetration sleeve for the fuel transfer tube is embedded in the fuel transfer canal wall. The SFP, including the fuel transfer canal, is constructed of concrete with a stainless steel liner on its walls and a 2.1-m (7-foot) thick concrete base slab.

The SFP and the fuel transfer canal are divided into 10 leak detection zones, 5 for the pools, and 5 for the canal. At present, three zones, zone nos. 1, 4, and 5, are indicating leakage of approximately 17.7, 8.9, and 26.6×10^{-5} cubic meters (m^3) per day (6, 3, and 9 ounces per day), respectively, which totals 3.79×10^{-3} m^3 (one gallon) per week.

B.7.4 Corrective Actions for Spent Fuel Pool Liner Leakage

The presence of spent fuel in the storage pools makes inspection of a large part of the storage pool liner impractical due to access restrictions. A monthly leak inspection plan based on available techniques, however, was developed to identify and remediate the SFP liner leakage, including leaks of the liner pressure boundary weld seams (Refs. B.7.6.2 and B.7.6.6). Portions of the auxiliary building adjacent to the SFP will be inspected annually during the period of extended operation to identify any additional leakage indications. Any newly observed indications will be documented and entered into the corrective action program. In addition, a multidisciplinary team will be formed to develop recommendations for inspection, testing, and repairs to remediate the SFP liner leakage. The SFP liner seam weld leakage detection and collection system drain lines will be inspected and repaired, if required, to ensure a clear drain path. A routine maintenance activity will be created to continue inspection of the drain lines through the period of extended operation.

B.7.5 Structural Integrity Assessment and Test Results

Industry data related to the liner leakage of the reactor cavity and SFP at Indian Point Nuclear Generating Units 2 and 3 (Ref. B.7.6.7) and the water seepage from the refueling cavity at Prairie Island Nuclear Generating Plant, Units 1 and 2 (Ref. B.7.6.8) were referenced, indicating that, even in the presence of borated water, the conditions at the rebar remain sufficiently alkaline, resulting in negligible corrosion.

A concrete core sample will be taken prior to the end of 2011. At least one core bore sample will be taken from the waste drumming room reinforced concrete ceiling below the SFP. The core sample location and depth will be sufficient to validate the strength of the concrete and the extent of any degradation. The core sample will be tested for compressive strength and will be subjected to petrographic examination. Reinforcing steel in the core sample area will be exposed and inspected for material condition.

B.7.6 References

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B.7.6.2 U.S. Nuclear Regulatory Commission. “Safety Evaluation Report Related to the License Renewal of Kewaunee Power Station.” NUREG-1958. U.S. Nuclear Regulatory Commission: Washington, DC. January 2011.

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B.7.6.4 Price, J.A., Dominion Energy Kewaunee, Inc., letter to U.S. Nuclear Regulatory Commission, December 28, 2009.

B.7.6.5 Hernandez, S., U.S. Nuclear Regulatory Commission, letter to D.A. Heacock, Dominion Energy Kewaunee, Inc., November 20, 2009.

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B.7.6.8 Holian, B.E., U.S. Nuclear Regulatory Commission, letter to M.A. Schimmel, Northern States Power Company, Minnesota, October 16, 2009.

B.8 <u>Palo Verde Nuclear Generating Station, Unit 1</u>

B.8.1 Introduction

By letter dated December 11, 2008 (Ref. B.8.6.1), as supplemented by letter dated April 14, 2009 (Ref. B.8.6.2), Arizona Public Service Company submitted an LRA to the NRC for Palo Verde Nuclear Generating Station, Unit 1, Unit 2, and Unit 3. The results of the NRC staff's evaluation of the LRA were published in April 2011 in NUREG-1961, "Safety Evaluation Report Related to the License Renewal of Palo Verde Nuclear Generating Station, Units 1, 2, and 3" (Ref. B.8.6.3).

B.8.2 Field Observations

In July 2005, during a routine area tour, an auxiliary operator in the Unit 1 fuel building observed water seeping from the SFP south wall at the 32-m (105-foot) elevation in the cleanup pump area. There were also white deposits that looked like solidified boric acid. Upon further inspection, a second leak was discovered outside of the fuel building at the 31.7-m (104-foot) elevation of the SFP east wall. Personnel obtained samples of the water and debris outside of the Unit 1 fuel building and identified trace quantities of radioactive cobalt-60, antimony-125, and cesium-137. Samples of the leakage indicated that the source of the water was from the SFP because boron concentrations were consistent with SFP chemistry. Following the discovery of the leakage, the SFP telltale drains were opened and approximately 4.54 m^3 (1,200 gallons) of water were released from the drains.

B.8.3 Design Characteristics of the Spent Fuel Pool

The SFP is constructed of reinforced concrete with the inner surfaces lined by welded stainless steel plates (Ref. B.8.6.4). A built-in leak-detection system collects water from slight liner leaks or liner plate weld imperfections and routs it through a leak-chase system to the radioactive waste drain (RD) system, thus ensuring that leakage into the environment does not occur. The leak chase system, or telltale system, is subdivided into 10 sections, and each section has an associated RD valve. The RD valves associated with the SFP telltale system are kept closed. The telltale RD valves are opened on a daily basis and any accumulated water is drained, then the valves are reclosed. Any water is measured and recorded by operations personnel and trended by System Engineering. The water drained from the telltale lines is returned to the RD system.

B.8.4 Corrective Actions for Spent Fuel Pool Liner Leakage

All contamination that resulted from the leakage was removed from the outer SFP walls, and it was verified that no residual activity remained outside the fuel building. Potential environmental impacts from the condition were reviewed, and it was determined that no adverse effects resulted from the SFP leakage because the small amount of leakage outside the SFP building (2.37×10^{-4} m^3 (8 ounces)) could not reach the local perched or regional ground water due to their distances (21.3 and 71.4 m (70 and 300 feet), respectively) below the ground surface (Ref. B.8.6.4).

The SFP leakage that occurred in Unit 1 in July 2005 was evaluated (Ref. B.8.6.5). The cause of the water backing up in the leak chase system was due to a pressure test plug that had been lodged in the drain basin drain line since construction. The basin is the same place where the telltale RD drain valves discharge. The plug was not allowing the basin to drain, and operations personnel did not have an alternate means of draining the water out of the basin. A decision was made to stop drainage of the telltale drain valves until the issue was corrected. As a consequence, water accumulated, filling each leak chase channel. This water seeped to adjacent channels via the small gap between the concrete wall and the liner plate. Eventually, the water made its way to the exterior face of the fuel pool walls through extremely small cracks in two locations. One of these locations was on the south side of the fuel pool inside of the fuel building; therefore, the leak was contained. The other leak was on the east side of the fuel pool that is also the exterior wall of the fuel building.

Once the obstruction (plug) was removed from the drain basin drain line, a large amount of borated water was released from each RD telltale drain valve line to the RD system for processing, thus providing evidence that the leak chase lines were not obstructed. This event was entered in the corrective action program. The cause of the issue was identified, and corrective actions were taken to eliminate and prevent further leakage to the environment.

Plant personnel open the valves and record drained water on a daily basis to keep water from backing up and leaking through the concrete (Ref. B.8.6.3). In addition, drainage from the telltale system will be measured and trended, and any abnormalities will be investigated through the corrective action program. In order to ensure that the drain lines were clear, all telltale drain lines in all units were inspected via boroscope between 2008 and 2009, and subsequently on a 2-year frequency.

B.8.5 Structural Integrity Assessment and Test Results

Construction Technology Laboratories performed a nondestructive examination (NDE) of the SFP concrete walls (Refs. B.8.6.3 and B.8.6.6). The Construction Technology Laboratories inspection report concluded that the borated water leakage did not have an adverse impact on the concrete.

Leakage through the concrete has stopped completely in areas that were initially identified as showing leakage, based on the fact that there are no longer wetted areas visible (Refs. B.8.6.7 and B.8.6.8). In addition, in 2006 and 2007, shallow aquifer wells were installed down-gradient of each unit. These wells are sampled periodically and no radioactivity has been detected. No indications of leakage have been identified in Units 2 or 3.

B.8.6 References

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B.8.2.2 Mims, D.C., Arizona Public Service Company Palo Verde Nuclear Generating Station, letter to U.S. Nuclear Regulatory Commission, April 14, 2009.

B.8.6.3 U.S. Nuclear Regulatory Commission. “Safety Evaluation Report Related to the License Renewal of Palo Verde Nuclear Generating Station, Units 1, 2, and 3.” NUREG-1961. U.S. Nuclear Regulatory Commission: Washington, DC. April 2011.

B.8.6.4 Pruett, T.W., U.S. Nuclear Regulatory Commission, letter to J.M. Levine, Arizona Public Service Company, November 9, 2005.

B.8.6.5 Mims, D.C., Arizona Public Service Company Palo Verde Nuclear Generating Station, letter to U.S. Nuclear Regulatory Commission, February 19, 2010.

B.8.6.6 Regner, L.M., U.S. Nuclear Regulatory Commission, letter to R.K. Edington, Arizona Public Service Company, December 29, 2009.

B.8.6.7 Hesser, J.H., Arizona Public Service Company Palo Verde Nuclear Generating Station, letter to U.S. Nuclear Regulatory Commission, May 21, 2010.

B.8.6.8 Regner, L.M., U.S. Nuclear Regulatory Commission, letter to R.K. Edington, Arizona Public Service Company, April 28, 2010.

B.9 Salem Nuclear Generating Station, Units 1 and 2

B.9.1 Introduction

On August 18, 2009, PSEG submitted an LRA to the NRC for Salem Nuclear Generating Station (Salem), Units 1 and 2 (Ref. B.9.6.1). The results of the NRC staff's evaluation of the LRA were published in June 2011 in NUREG-2101, "Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station" (Ref. B.9.6.2).

B.9.2 Field Observations

In 1980, a small leak was discovered in the SFP telltale drains at Unit 1. Underwater inspections determined that the cause was due to leaking seam welds. After repair of the leaking seam welds, the leakage was reduced to less than 8.8×10^{-9} m^3/s (0.2 gpd) (Ref. B.9.6.3).

In September 2002, tests identified evidence of radioactive water leakage through a concrete wall located at the 24-meter (78-foot) elevation of the Unit 1 auxiliary building mechanical penetration room, a radiologically controlled area. The leak location, about 3 meters (10 feet) up a wall surface, was identified while investigating a low-level shoe contamination from the area. A task action plan was established to identify and stop sources of leakage and evaluate the possibility of undetected leakage outside building structures. Other locations were identified where radioactive water was leaking through interior walls or penetrations into both the Unit 1 auxiliary building and the Unit 1 FHB. Videoscopic inspection of the SFP telltale drains and leakage channels revealed that most of the telltale drains were blocked. As a result of the blockage, leakage through the seam welds and the plug welds accumulated in small gaps between the stainless steel liner and concrete. As the water level in the gap increased, hydrostatic pressure forced the water outside through penetrations, construction joints, and small cracks in the SFP concrete (Ref. B.9.6.1). Ultimately, the water migrated into the seismic gap between the FHB and the auxiliary building and there was evidence of seepage into the sump room in the FHB via a construction joint at the base of the pool (Ref. B.9.6.4). Figure B.9-1 shows the SFP water leakage path into the seismic gap. In 2003, the blockage was removed from the drain system. Since 2003, the leakage through the drain system has been monitored, and the volume of leakage averages 4.3×10^{-6} m^3/s (100 gpd) (Ref. B.9.6.3). Figure B.9-2 shows the current leakage paths. In 2010, evidence of a small leak was noted to be present in the Unit 2 telltale drain system. Although it was verified that the drain lines were open, the small amount of leakage will be monitored.

Relative to Unit 1 SFP leakage, it has been noted (Ref. B.9.6.2) that conditions indicate that, based on sampling of water collected from the seismic gap drain located next to the east wall, approximately 5.4×10^{-9} m^3/s (0.125 gpd) is migrating through the inaccessible east wall. There is no evidence of through-wall leakage on the accessible west wall since the telltale drains were cleared in 2003. Leakage through the south wall was not considered feasible because the wall is 11.9 m (39 feet) thick, and, based on tritium levels, leakage is not occurring through the north wall.

B.9.3 Design Characteristics of the Spent Fuel Pool

The SFP pressure boundary is fabricated from stainless steel plates welded together using a continuous backer strip to create a liner. After welding, each seam weld was vacuum tested.

To mitigate the consequences of a seam weld leak, continuous stainless steel leak chase channels were installed directly beneath each seam weld prior to concrete placement to serve as part of a telltale collection system. The channels were anchored to the concrete and positioned so the two legs of each channel straddled the seam weld and were even with the finished concrete surface. The leak chase system is designed such that any leakage collected in the channels is directed and discharged through 17 drain lines into the sump room trench outside the SFP in the FHB. The liner plates were also anchored to the concrete by stainless steel structural attachment studs embedded in the concrete. These studs extended through holes in the liner plates and were plug welded to the liner plates. This method of construction created an air gap between the concrete surface and the liner plates. In addition, the design did not specify welding of the channel legs to the liner plate. The SFP liner contains about 640 m (2,100 linear feet) of seam welds and about 1,400 plug welds. The liner creates a pressure boundary to confine the borated water that surrounds the spent fuel. The borated water in the SFP has 2,200–2,400 ppm boron and is acidic with a pH of 4.8. The bottom of the SFP liner is located at 27.4-meters (89-feet) elevation. Support for the liner plates is provided by reinforced concrete structural elements.

B.9.4 Corrective Actions for Spent Fuel Pool Liner Leakage

Leakage from the SFP is likely from small cracks in seam welds (2,100 linear feet) of adjoining liner plates or at plug welds (1,400 total) that connect the liner plates to the steel embedded in the surrounding concrete as a result of differential thermal expansion between the liner and the concrete structure (Ref. B.9.6.3). In 1995, 95 percent of all seam welds were inspected via vacuum box testing and no through-wall cracks were found, suggesting that the leakage was occurring below the sensitivity of the test (Ref. B.9.6.3). It has been estimated that the leakage rate could be the result of a single or multiple cracks having dimensions of 0.0254 mm wide by 152 mm long (0.001 inch wide and 6 inches long). The current direction is to monitor the telltale leakage, clear the leak chase system every 18 months to ensure proper drainage, and minimize the hydrostatic pressure buildup and any leakage to the environment.

B.9.5 Structural Integrity Assessment and Test Results

A structural assessment of the FHB was performed (Refs. B.9.6.3 and B.9.6.4). It included the following:

- A baseline inspection of the building was conducted consistent with American Concrete Institute (ACI) guidelines to assess the overall condition of the structure. PSEG has committed to continue monitoring of the structure in accordance with ACI 349.3R-02, "Evaluation of Existing Nuclear Safety-Related Concrete Structures" (Ref. B.9.6.5).

- An assessment of the potential reduction in structural margin due to postulated degradation of the structure over the remaining plant life was performed. This effort included testing to demonstrate the impact of boric acid on reinforced concrete and to quantify the degradation rate.

The assessment concluded the following:

- Overall, the concrete appears to be in good structural condition.

- The appearance of leaching or chemical attack and corrosion staining of undefined source on concrete surfaces does not indicate significant structural deterioration at this time.

- There were no indications of concrete surface expansion due to reinforcing steel corrosion.

To address the through-wall leakage and any possible associated degradation, the following commitments were made (Ref. B.9.6.2):

- Perform periodic structural examinations of the FHB per ACI 349.3R to ensure that the structural condition is in agreement with analysis.

- Monitor telltale leakage and inspect the leak chase system to ensure that there is no blockage.

- Test water drained from telltales and the seismic gap for boron, chloride, iron, and sulfate concentrations, and for pH. Sample readings outside the acceptance criteria, noted below as an illustrative example, will be entered into and evaluated in the corrective action program.

Chemical analysis	Acceptance Criteria		Frequency for monitoring
	SFP Telltales (West Wall)	SFP Telltales (East Wall)	
pH	6.0 < pH < 7.5	7.0 < pH < 8.5	Monthly
Chloride	≤ 500 ppm	≤ 500 ppm	Every 6 months
Sulfate	≤ 1,500 ppm	≤ 1,500 ppm	Every 6 months
Boron	Information Only	Information Only	Monthly
Iron	Information Only	Information Only	Every 6 months

- Perform one shallow core in each of the Unit 1 SFP walls (east and west) that have shown ingress of borated water through the concrete. Core samples will expose rebar, which will be examined for corrosion. East and west wall cores are to be taken by the end of 2015 and 2013, respectively.

- Perform a structural examination per ACI 349.3R every 18 months of the Unit 1 SFP wall in the sump room where previous inspections have shown ingress of borated water through the wall.

Laboratory studies were conducted to quantify the potential degradation of concrete in the FHB structure (Ref. B.9.6.3). The following conclusions are based on results of these studies:

- Borated water attacks the calcium hydroxide component of the cement paste, causing loss of bonding of the coarse and fine aggregates.

- The predicted depth of borated water intrusion in the concrete after 70 years exposure to SFP water at 37.8 degrees Centigrade (100 degrees Fahrenheit) is anticipated to be 33.0 mm (1.30 inches). This includes a two-sigma statistical uncertainty of the test data and an adjustment for temperature. Since the concrete clear cover for the walls and slab is greater than this projected degradation depth, the borated water will not reach rebar during the 70-year period.

- The wicking effect at the rebar/concrete interface was observed to be minor. That is, the degradation rate of the concrete at the rebar/concrete interface is similar to the general rate of attack of concrete without rebar. Therefore, degradation of rebar at the construction joints or cracks will not spread rapidly along the interface with the rebar, and loss of bond with the concrete will not be compromised.

The impact of borated water on the concrete embedded steel reinforcement was also addressed, with the following conclusions (Ref. B.9.6.2):

- The rebar of concern is the outer rebar, as the limiting margin cases involve compression on the poolside and tension on the outside. Accordingly, the borated water must seep through several feet of concrete before reaching the rebar. The acidic water would react with concrete along the transit path, resulting in diminishing acidity. The corrosion rate of carbon steel in de-aerated boric acid is 0.004 mm/year (0.1575 mils or 157.5 microinches/year) in a 2,400-ppm solution. However, this rate is conservative with regard to the situation in the FHB because the pH when the borated water reaches the rebar will be increased due to its reaction with the concrete.

- A reference study from Germany, published in a reputable journal, documented a carefully controlled study of corrosion of embedded rebar from flow of boric acid through a simulated crack (Ref. B.9.6.6). It showed negligible corrosion for the most aggressive conditions after a period of 2 years.

- Experience at another U.S. pressurized-water reactor showed no visible corrosion of embedded reinforcing steel from boric acid flow through a crack over several years. Rust stains on the sump room walls are minor and result in the deposition of small amounts of iron oxide.

- Borated water that leaks through the stainless steel liner will be partially deaerated as it reacts with and corrodes the carbon steel leak chase channels.

- Oxygen in the borated water that reaches the embedded concrete steel reinforcement by traveling through concrete cracks will be quickly consumed during initial oxidation reaction with the reinforcement.

- Oxygen that is consumed will not be replenished, since the water migration path to the steel reinforcement is relatively stagnant.

Corrosion of the carbon steel leak chase channels is not a concern because the channels have no impact on the structural integrity of the FHB and their sole function is to collect SFP leakage and route it to the sump via telltales.

The potential impact of erosion of the concrete in the floor of the SFP on the Unit 1 SFP liner was considered (Ref. B.9.6.2). It was noted that an assessment has determined that the liner is sufficiently ductile to accommodate the load from spent fuel racks, even if the foot of a rack was positioned over an area of degraded concrete.

The impact on structural capacity resulting from the long-term effects of exposure of concrete and steel reinforcing bars to borated water has also been evaluated (Ref. B.9.6.4). Projected degradation through the end of plant life was estimated to reduce the available margin in the limiting section by less than one-half percentage point to 1.6 percent (i.e., a design margin ratio of 1.016). Therefore, the conservative design-basis analysis of record was considered to not be invalidated by the postulated degradation.

B.9.6 References

B.9.6.1 Fricker, C.J., PSEG Nuclear, LLC, letter to U.S. Nuclear Regulatory Commission, August 18, 2009.

B.9.6.2 U.S. Nuclear Regulatory Commission. “Safety Evaluation Report Related to the License Renewal of Salem Nuclear Generating Station.” NUREG-2101. U.S. Nuclear Regulatory Commission: Washington, DC. June 2011.

B.9.6.3 Davison, P.J., PSEG Nuclear, LLC, letter to U.S. Nuclear Regulatory Commission, May 13, 2010.

B.9.6.4 Joyce, T.P., PSEG Nuclear, LLC, letter to J. White, U.S. Nuclear Regulatory Commission, March 30, 2007.

B.9.6.5 American Concrete Institute. “Evaluation of Existing Nuclear Safety-Related Concrete Structures.” ACI 349.3R-02. American Concrete Institute: Farmington Hills, MI. 2002.

B.9.6.6 Ramm, W., and M. Biscoping. “Autogenous Healing and Reinforcement Corrosion of Water-Penetrated Separation Cracks in Reinforced Concrete.” *Nuclear Engineering and Design* 179: 191–200. 1998.

B.9.6.7 “Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee.” Work Order No. NRC-577. Neal R. Gross and Co., Inc., Court Reporters and Transcribers: Washington, DC. December 1, 2010.

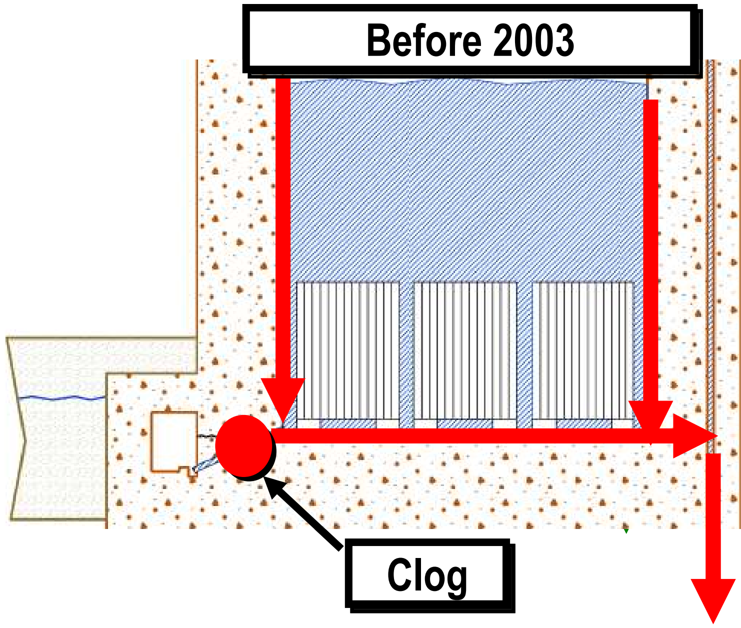

Figure B.9-1 Spent fuel pool leakage path for Salem Unit 1 before 2003

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee." Work Order No. NRC-577. Neal R. Gross and Co., Inc., Court Reporters and Transcribers: Washington, DC. December 1, 2010.)

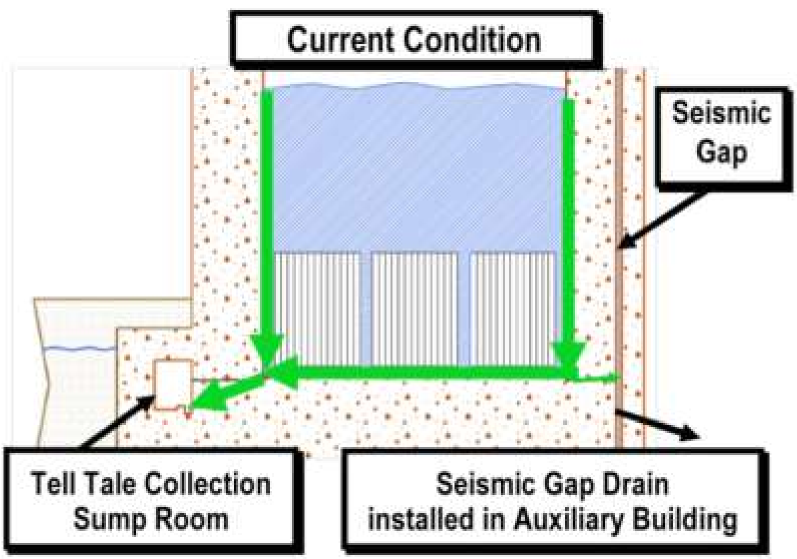

Figure B.9-2 Current spent fuel pool leakage path for Salem Unit 1

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee." Work Order No. NRC-577. Neal R. Gross and Co., Inc., Court Reporters and Transcribers: Washington, DC. December 1, 2010.)

APPENDIX C: BOILING-WATER REACTOR MARK I CONTAINMENT—TORUS CORROSION AND CRACKING CASE STUDIES

C.1 Cooper Nuclear Station

C.1.1 Introduction

By letter dated September 24, 2008 (Ref. C.1.6.1), Nebraska Public Power District submitted a license renewal application (LRA) to the U.S. Nuclear Regulatory Commission (NRC) for the Cooper Nuclear Station. The results of the NRC staff's evaluation of the LRA were published in October 2010 in NUREG-1944, "Safety Evaluation Report Related to the License Renewal of Cooper Nuclear Station" (Ref. C.1.6.2).

C.1.2 Field Observations

Since 1974, 3,800 coating repairs comprising 13.5 square meters (145 square feet), or 1.1 percent of the torus surface area below the waterline, have been made (Ref. C.1.6.2). Eighteen locations have been reported to have pits where the nominal thickness has been reduced by greater than 10 percent of the nominal shell thickness (Ref. C.1.6.3). Figure C.1-1 provides an example of a torus pit, and Figure C.1-2 provides an example of a torus pit coating repair. The locations of pit concentrations are presented in Figure C.1-3. Reference C.1.6.4 notes that no pits have been identified that require repair in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code) (Ref. C.1.6.5).

C.1.3 Design Characteristics of the Torus

The primary containment is a Mark I low-leakage pressure-suppression containment design that houses the reactor vessel, the reactor recirculating loops, and other connections of the reactor coolant system (Ref. C.1.6.1). The major components of the primary containment include a drywell, a torus (or pressure-suppression chamber (PSC)), and the connecting vent system between the drywell and the torus.

The drywell houses the reactor vessel and associated components. The drywell is a carbon steel structure surrounded by a reinforced concrete biological shield wall. Internal structures consist of a drywell fill slab, reactor pedestal, sacrificial shield wall and its lateral support, and structural steel. The reinforced concrete fill slab in the bottom of the drywell supports the reactor pedestal and other structures and components inside the drywell. A gap separates the drywell from the reactor building reinforced concrete in the area around the cylindrical portion and the spherical portion above the support transition point at the lower radius. The reinforced concrete drywell floor contains the drywell floor drain and equipment drain sumps. The reactor pedestal is a reinforced concrete cylinder supporting the reactor pressure vessel, the sacrificial shield wall, and floor framing.

One personnel access lock is provided for access to the drywell. The lock has two gasketed doors in series. A personnel access hatch is provided on the drywell head. This hatch is bolted in place. The drywell has two equipment access hatches bolted in place. The drywell top head, the two equipment hatches, the drywell and torus manways, the control rod drive removal hatch, and the stabilizer assembly inspection ports have double-gasketed closures to maintain containment leak tightness. The drywell design accommodates pressures and temperatures resulting from a breach of the reactor coolant pressure boundary up to and including an instantaneous circumferential break of the reactor recirculation piping and provides holdup for decay of radioactive material. When operating at power, the drywell is filled with nitrogen to preclude the presence of oxygen.

The torus is located below the drywell and encircles and contains treated (demineralized) water, which forms the suppression pool. The torus is a carbon-steel pressure vessel anchored to and supported by the reinforced concrete foundation slab of the reactor building.

C.1.4 Corrective Actions for Torus Corrosion and Cracking

The containment inservice inspection program and the Service Level I coating program are used to provide assurance that there is proper maintenance of the protective coatings in containment, such that they will not degrade and become a debris source that may challenge the emergency core cooling system (ECCS) (Ref. C.1.6.6). The Service Level I coating program is used to provide specific instructions for maintenance of safety-related coatings applied to concrete and steel surfaces within the drywell and torus (Ref. C.1.6.7). Although the containment inservice inspection program is not used to inspect Service Level I coating, the coatings in the purview of the program that show signs of degradation are reported for review and evaluation under the Service Level I coating program. The program specifies visual inspections during each refueling outage (RFO) of the coating surfaces for adverse coating conditions such as flaking, peeling, blistering, discoloration, and other signs of distress. Divers visually inspect coatings on the torus below the waterline. Coatings showing signs of degradation are documented in the corrective action program, reviewed, and evaluated for acceptability, repair, or replacement. The inspections are done in compliance with requirements in ASME Code Section XI, Subsection IWE.

Repairs are typically made only in areas that experience localized zinc coating failures where the pit depths exceed a threshold (generally 0.76 to 1.27 millimeters (mm) (30 to 50 mils)) (Ref. C.1.6.2). Table C.1-1 presents an example of inspection criteria (Ref. C.1.6.8). The coating is repaired locally by applying an epoxy that is intended to arrest the pitting. However, coating at pits that do not exceed the threshold is not required, but the pitting is monitored at the next inspection (3 years later) for growth. This process, however, can result in localized galvanic corrosion that can yield higher and unpredictable corrosion rates (pitting) than that of general corrosion. It has also contributed to the amount of sludge and corrosion products collecting in the suppression pool, which can further increase the corrosion rate. To address this concern, the following commitments will be implemented (Ref. C.1.6.9):

- The wetted portion of the torus will be recoated within 3 years after entering the period of extended operation (but not later than January 18, 2017).

- Sludge will be removed and the wetted portion of the torus inspected every refueling outage until the torus is recoated.

- An engineering analysis will be completed following each torus inspection that demonstrates that the projected pitting of the torus, up to the time that the torus is recoated, will not result in reduction of torus wall thickness below minimum acceptable values.

C.1.5 Structural Integrity Assessment and Test Results

An engineering evaluation determined that loss of material at the 18 pits identified above was acceptable (Ref. C.1.6.3). Supplementary volumetric examinations (ultrasonic thickness measurements) at these 18 pits located under water have not been performed. However, augmented visual testing of the wetted surfaces of the torus is performed once during each inspection period (three times in 10 years) as required by the ASME Code.

According to reference calculations, the minimum corrosion allowance available after considering the existing pits that have been identified and repaired until 2008 is 0.787 mm (31 mils). A corrosion rate of 0.066 mm per year (2.6 mils per year) has been determined based on the maximum pit growth observed over approximately 13 years (Ref. C.1.6.4). The rate was determined by comparing the results of the 2005 inspections to those identified approximately 13 years earlier. Using this rate, it was concluded that the torus is acceptable until at least 2014 (Ref. C.1.6.2).

C.1.6 References

C.1.6.1 Minahan, S.B., Nebraska Public Power District, letter to U.S. Nuclear Regulatory Commission, September 24, 2008.

C.1.6.2 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of Cooper Nuclear Station." NUREG-1944. U.S. Nuclear Regulatory Commission: Washington, DC. October 2010.

C.1.6.3 O'Grady, B.J., Nebraska Public Power District, letter to U.S. Nuclear Regulatory Commission, December 21, 2009.

C.1.6.4 Minahan, S.B., Nebraska Public Power District, letter to U.S. Nuclear Regulatory Commission, July 29, 2009.

C.1.6.5 American Society of Mechanical Engineers. "ASME Boiler and Pressure Vessel Code." American Society of Mechanical Engineers: New York, NY. 2007.

C.1.6.6 Tran, T., U.S. Nuclear Regulatory Commission, letter to S.B. Minahan, Nebraska Public Power District, September 28, 2009.

C.1.6.7 Minahan, S.B., Nebraska Public Power District, letter to U.S. Nuclear Regulatory Commission, October 22, 2009.

C.1.6.8 "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee." Work Order No. NRC-237. Neal R. Gross and Co., Inc., Court Reporters and Transcribers: Washington, DC. May 5, 2010.

C.1.6.9 O'Grady, B.J., Nebraska Public Power District, letter to U.S. Nuclear Regulatory Commission, May 4, 2010.

Table C.1-1 Example of Torus Inspection Criteria

Region Classification	Pit Type	Pit Depth	Coating Repair Required (yes/no)
Near Penetration	Shallow	0 to < 30 mil	Yes
	Deep	$\geq$ 30 mil	Yes
Near Ring Girder	Noted	< 50 mil	No
	Shallow	$\geq$ 50 mil < 90 mil	Yes
	Deep	$\geq$ 90 mil	Yes
General Shell	Noted	< 90 mil	No
	Shallow	> 90 mil < 150 mil	Yes
	Deep	> 150 mil	Yes

Source: Ref. C.1.6.7.

Figure C.1-1 Example of pitting corrosion in torus

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee." Work Order No. NRC-237. Neal R. Gross and Co., Inc., Court Reporters and Transcribers: Washington, DC. May 5, 2010.)

Figure C.1-2 Example of repair of pitting corrosion in torus

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee." Work Order No. NRC-237. Neal R. Gross and Co., Inc., Court Reporters and Transcribers: Washington, DC. May 5, 2010.)

Figure C.1-3 Example of pitting corrosion concentrations in torus

(Source: "Official Transcript of Proceedings, Advisory Committee on Reactor Safeguards Plant License Renewal Subcommittee." Work Order No. NRC-237. Neal R. Gross and Co., Inc., Court Reporters and Transcribers: Washington, DC. May 5, 2010.)

C.2 Duane Arnold Energy Center

C.2.1 Introduction

By letter dated September 30, 2008 (Ref. C.2.6.1), and supplemented by a letter dated January 23, 2009 (Ref. C.2.6.2), FPL Energy, Duane Arnold Center, submitted an LRA to the NRC for the Duane Arnold Energy Center. The results of the NRC staff's evaluation of the LRA were published in November 2010 in NUREG-1955, "Safety Evaluation Report Related to the License Renewal of Duane Arnold Energy Center" (Ref. C.2.6.3).

C.2.2 Field Observations

Numerous areas of zinc depletion of the torus coating and minor pitting and other indications were identified while conducting ASME Code (Ref. C.2.6.4) Section XI, Subsection IWE, inspections. Since 1995, over 15,000 repairs have been made to the torus coating (Ref. C.2.6.5). This represents about 5 percent of the underwater torus area. Only one pit had degradation that exceeded the maximum allowable pit depth of 1.35 mm (53 mils) (10 percent of 13.56 mm (0.534 inch), the nominal shell thickness). That pit measured 1.42 mm (56 mils) in depth and 6.35 mm (0.25 inch) in diameter. The pit was dispositioned in the corrective action program as acceptable without repair.

Inspections performed during recent outages have not identified any coating deficiencies above the waterline that have required repair (Ref. C.2.6.6).

C.2.3 Design Characteristics of the Torus

The primary containment is a Mark I containment system, employing a drywell and a separate PSC (Ref. C.2.6.1). The drywell houses the reactor vessel, the reactor recirculation loops, and branch connections of the reactor coolant system that have isolation valves at the primary containment boundary. The PSC (torus) consists of an air volume and a suppression water volume. The drywell and torus are connected through a vent system that directs flow from the drywell into the suppression water of the torus through submerged downcomers.

The torus shell was initially coated in 1973 and recoated in 1985. The normal life of the torus coating is less than 20 years (Ref. C.2.6.3).

C.2.4 Corrective Actions for Torus Corrosion and Cracking

The ASME Code Section XI, Subsection IWE inspection procedure was revised to inspect the torus coating during each outage until it is recoated.

The scope of the coatings program includes inspection of the interior and exterior surfaces of the suppression chamber (torus), vent lines, and downcomers, and the interior and accessible exterior surfaces of the drywell (Ref. C.2.6.3). Inspections are performed during each refueling cycle. Visual inspections of the suppression chamber and drywell are conducted to note any evidence of deterioration (e.g., discoloration, bubbling or flaking of the coating, corrosion, or pitting). Qualification testing and evaluation of the Service Level I coatings used for new applications or repair activities inside containment are performed in accordance with American National Standards Institute (ANSI) N101.2, "Protective Coatings (Paints) for Light-Water

Nuclear Reactor Containment Facilities" (Ref. C.2.6.7). The coating specialist also reviews inspection results to determine if updates are required to the unqualified and degraded coatings log, and evaluates whether the quantity of unqualified and degraded coatings is acceptable. Corrective actions are initiated, as appropriate, based on evaluations performed by the coating specialists.

Photographs, inspection reports and completed checklists, records of corrective actions, and other followup information are maintained as quality assurance records (Ref. C.2.6.8). These records are available for review to support aging management during the period of extended operation. The torus inspection procedure requires a review of previously performed inspection results and documentation of current results (Ref. C.2.6.3). Furthermore, the procedure specifies that it should include photographs with noted deficiencies tracked by appropriate documentation to track resolution. Examinations of the submerged portion of the suppression chamber are performed by specialty contractors, and the results and repairs are documented in the inspection report and procedure. Additionally, the initial and final inspections are videotaped, with the tapes made available for review during subsequent inspections.

C.2.5 Structural Integrity Assessment and Test Results

No structural integrity assessments have been performed. Only one pit had degradation that exceeded the maximum allowable pit depth of 1.35 mm (0.053 inches) (10 percent of the 13.56-mm (0.534-inch) nominal shell thickness). This pit measured 1.42 mm (0.056 inches) in depth and 6.35 mm (0.25 inches) in diameter. The pit was dispositioned in the corrective action program as acceptable without repair.

A commitment was made to completely recoat the torus interior surface below the waterline as well as extending it to well above any fluctuations in the water level, including the 0.61-meter (2-foot) wide splash band at water level (Ref. C.2.6.3).

C.2.6 References

C.2.6.1 Anderson, R.L., FPL Energy-Duane Arnold Energy Center, letter to U.S. Nuclear Regulatory Commission, September 30, 2008.

C.2.6.2 Anderson, R.L., FPL Energy-Duane Arnold Energy Center, letter to U.S. Nuclear Regulatory Commission, "Response to Issues Raised in Previous Status of License Renewal Application for the Duane Arnold Energy Center," January 23, 2009.

C.2.6.3 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of Duane Arnold Energy Center." NUREG-1955. U.S. Nuclear Regulatory Commission: Washington, DC. November 2010.

C.2.6.4 American Society of Mechanical Engineers. "ASME Boiler and Pressure Vessel Code." American Society of Mechanical Engineers: New York, NY. 2007.

C.2.6.5 Costanzo, C.R., NextEra Energy Duane Arnold, LLC, letter to U.S. Nuclear Regulatory Commission, December 14, 2009.

C.2.6.6 Costanzo, C.R., NextEra Energy Duane Arnold, LLC, letter to U.S. Nuclear Regulatory Commission, April 2, 2010.

C.2.6.7 American National Standards Institute, Subcommittee N101.5, American Institute of Chemical Engineers. "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities." ANSI N101.2. American Institute of Chemical Engineers: New York, NY. 1972.

C.2.6.8 Costanzo, C.R., NextEra Energy Duane Arnold, LLC, letter to U.S. Nuclear Regulatory Commission, October 13, 2009.

C.3 James A. FitzPatrick Nuclear Power Plant

C.3.1 Introduction

By letter dated July 31, 2006 (Ref. C.3.6.1), Entergy Nuclear Operations, Inc., submitted an LRA to the NRC for renewal of the operating license for the James A. FitzPatrick Nuclear Power Plant. The results of the NRC staff's evaluation of the LRA were published in April 2008 in NUREG-1905, "Safety Evaluation Report Related to the License Renewal of James A. FitzPatrick Nuclear Power Plant" (Ref. C.3.6.2).

C.3.2 Field Observations

Pitting in the wetted area of the torus shell was identified in 1998 when the torus was drained to replace the ECCS suction strainers (Ref. C.3.6.2). Further inspection of the torus identified pitting in 10 areas in 4 of the 16 torus bays. The pitting occurred at locations that had experienced some degradation of the original coating. The pitted areas have not been recoated, but they are considered as leading indicators of torus shell condition and are being monitored periodically with ultrasonic testing (UT) and visual inspection.

In June 2005, a through-wall leak in the torus shell was identified. The leak was due to a 114.3-mm (4.5-inch) crack located in the same bay as the high-pressure coolant injection (HPCI) steam exhaust discharge pipe. The root cause of the flaw was vibration fatigue from HPCI steam condensation oscillation loading. Followup torus inspections identified similar flaws in two other locations in the same bay. Corrective action included repair of the flaws and the installation of an HPCI steam exhaust sparger assembly that directs steam flow away from the torus shell. The addition of the sparger significantly reduced steam condensation oscillation loads on the torus shell.

C.3.3 Design Characteristics of the Torus

The reactor building totally encloses the primary containment, the refueling and reactor servicing areas, the new and spent fuel storage facilities, and other reactor auxiliary systems (Ref. C.3.6.1). It serves as containment during reactor refueling and maintenance operations when the primary containment is open, and as an additional barrier when the primary containment is functional. The primary containment is a Mark I pressure-suppression containment housing the reactor vessel, the reactor recirculation loops, and other branch connections of the reactor coolant system. Major components of primary containment include a drywell, a PSC, and the connecting vent system between the drywell and torus. The torus is a carbon steel pressure vessel anchored to and supported by the reinforced concrete foundation slab of the reactor building.

C.3.4 Corrective Actions for Torus Corrosion and Cracking

The interior torus suppression pool areas above and below the waterline were inspected in accordance with the ASME Code (Ref. C.3.6.3) Section XI, Subsection IWE, program during RFOs. A general visual examination was performed of the area above the waterline. Below the waterline is normally inaccessible unless the torus water level is lowered or drained for a work activity. The torus was last drained and cleaned in 1998 for the installation of the ECCS strainers. The visual examination identified nine of the most severe areas of pitting. The depths of the pits were measured at that time and a portion of these areas are monitored and measured by means of UT from the outside of the torus shell every outage. Over a 5-year period, all nine of the pitted areas examined by performing UT were found to be acceptable in accordance with the ASME Code requirements.

The through-wall leak in the torus was repaired in July 2005. The root-cause analysis determined that condensation oscillation from the HPCI turbine steam discharge provided the energy that initiated cracking. Subsequently, UT was performed at this location. In RFO 17, a visual examination was scheduled to investigate the extent of the condition. Two cracks were noted near where the HPCI discharge line had been modified with a sparger assembly designed to eliminate condensation oscillation.

C.3.5 Structural Integrity Assessment and Test Results

The torus preservation program verifies that sample locations are tracked for wall thinning. The reports are entered into a nondestructive examination database and are used for tracking to assure that adequate wall thickness is maintained. In 2004, during RFO 16, the thickness examinations were made of the nine pitted locations identified during the 1996 ASME Code Section XI, Subsection IWE examination (two at bay B, two at bay H, two at bay K, and three at bay O around the torus). Of the areas sampled, only three locations had pit depths that exceeded the design thickness.

C.3.6 References

C.3.6.1 Dietrich, P., Entergy Nuclear Operations, Inc., letter to U.S. Nuclear Regulatory Commission, July 31, 2006.

C.3.6.2 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of James A. FitzPatrick Nuclear Power Plant." NUREG-1905. U.S. Nuclear Regulatory Commission: Washington, DC. April 2008.

C.3.6.3 American Society of Mechanical Engineers. "ASME Boiler and Pressure Vessel Code." American Society of Mechanical Engineers. New York, NY. 2007.

C.4 Hope Creek Generating Station, Unit 1

C.4.1 Introduction

On August 18, 2009, PSEG Nuclear, LLC, submitted an LRA to the NRC for renewal of the Hope Creek Generating Station (Ref. C.4.6.1). The results of the NRC staff's evaluation of the LRA were published in June 2011 in NUREG-2102, "Safety Evaluation Report Related to the License Renewal of Hope Creek Nuclear Generating Station" (Ref. C.4.6.2).

C.4.2 Field Observations

In 2004, the torus shell and interior coatings were inspected by divers performing underwater ASME Code (Ref. C.4.6.3) Section XI, Subsection IWE program inspections (Ref. C.4.6.2). There were 16 areas with metal loss reported as ranging up to 0.753 mm (30 mils), and 99.99 percent of the coating was found to be smooth and tightly adhered to the base metal with no significant defects. The identified coating deficiencies were primarily small, localized areas of mechanical or impact damage. Other than minor general corrosion of the exposed surfaces, there was no damage to the base metal. The loss of material thickness of the torus at the 16 local areas did not exceed 10 percent of the nominal plate thickness of the torus shell (i.e., 25.4 mm (1 inch) nominal plate thickness). These areas were cleaned and recoated in the subsequent outage. Re-inspection of these areas will be performed during future ASME Code Section XI, Subsection IWE, program underwater inspections.

C.4.3 Design Characteristics of the Torus

The primary containment is a Mark I design and consists of a drywell, a PSC, and a vent system connecting the drywell and the PSC (Ref. C.4.6.1). The PSC, or torus, is a toroidal-shaped steel pressure vessel encircling the base of the drywell that is partially filled with demineralized water and includes internal steel framing and access hatches. The PSC is mounted on support structures that transmit loads to the reactor building foundation. Major components inside the PSC include ECCS suction strainers, the PSC (torus) spray header, the vent line header and downcomers, and T-quenchers.

C.4.4 Corrective Actions for Torus Corrosion and Cracking

Affected areas were cleaned and the underwater coating was repaired. The Service Level I Amercoat® 90 coating system is managed in accordance with GALL AMP XI.S8, "Protective Coating Monitoring and Maintenance Program" (Ref. C.4.6.4). Underwater inspections are performed under the ASME Code Section XI, Subsection IWE, program (Ref. C.4.6.3).

Although the condition of the penetration and downcomer support base metal was acceptable, the inside of a number of penetrations and 32 downcomers supports were recoated to prevent further degradation (Ref. C.4.6.1).

C.4.5 Structural Integrity Assessment and Test Results

No structural integrity assessment of the torus was performed. Evaluation of the loss of material determined that it was acceptable because the reduction in torus shell thickness had not exceeded 10 percent of the nominal plate thickness and was under the 3.175-mm (125-mils) torus corrosion allowance included in the design of the torus shell.

C.4.6 References

C.4.6.1 Fricker, C.J., PSEG Nuclear, LLC, letter to U.S. Nuclear Regulatory Commission, August 18, 2009.

C.4.6.2 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of Hope Creek Generating Station." NUREG-2102. U.S. Nuclear Regulatory Commission. Washington, DC. June 2011.

C.4.6.3 American Society of Mechanical Engineers. "ASME Boiler and Pressure Vessel Code." American Society of Mechanical Engineers. New York, NY. 2007.

C.4.6.4 U.S. Nuclear Regulatory Commission. "Protective Coating Monitoring and Maintenance Program." GALL AMP XI.S8 in "Generic Aging Lessons Learned (GALL) Report." NUREG-1801. U.S. Nuclear Regulatory Commission: Washington, DC, July 2001.

C.5 Nine Mile Point Nuclear Station, Unit 1

C.5.1 Introduction

By letter dated May 26, 2004 (Ref. C.5.6.1), Constellation Energy Group, LLC, submitted an LRA to the NRC for renewal of the operating licenses for Nine Mile Point Nuclear Station (Nine Mile Point), Units 1 and 2. Constellation Energy Group, LLC, submitted an amended LRA to the NRC on July 14, 2005 (Ref. C.5.6.2). The results of the NRC staff's evaluation of the LRA were published in September 2006 in NUREG-1900, "Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2" (Ref. C.5.6.3).

C.5.2 Field Observations

Torus wall thinning was observed in Unit 1 in the late 1980s following an extended plant shutdown. The wall thinning was attributed to the layup conditions inside the torus during the extended shutdown (Ref. C.5.6.1).

An inspection report in 2001, referenced in Ref. C.5.6.4, noted that photo documentation from previous torus entries indicated that the waterline region may potentially exhibit generalized, nonspecific corrosion, and that supplemental examinations will be performed from the outside surface of the torus by taking ultrasonic thickness measurements of the torus shell plates at the waterline region to determine the general torus shell thickness and if minimum wall thickness requirements have been violated by corrosion. Of the 360 readings taken, the lowest recorded point (pit depth) was 1.11 mm (44 mils) in bay 3-I at location B6. Since the allowable minimum wall thickness was 10.08 mm (0.397 inches) at this location, the recorded readings were found to be acceptable.

C.5.3 Design Characteristics of the Torus

The primary containment is a Mark I design that consists of a drywell, a suppression chamber in the shape of a torus, and a connecting vent system between the drywell and the suppression chamber (Ref. C.5.6.3). The torus is a freestanding carbon steel pressure vessel that consists of 20 pipe-shaped segments or bays that are mitered and welded together. The diameter of the pipe-shaped segments is 8.23 meters and the total length of the torus is 112 meters. Carbon steel plates having a nominal thickness of 11.7 mm, including a corrosion allowance of 1.6 mm, were used to fabricate the torus shell (Ref. C.5.6.5). Most areas on the outside surface of the torus are accessible for visual inspection, but the surface is coated to prevent corrosion. Structural support is provided by a series of steel columns that are welded to the torus shell and rest on a concrete floor slab. Four columns are provided in every other bay; two on the outer side and two on the inner side of the torus. The inside of the torus is partially filled with demineralized water, and all surfaces above and below the water line are not coated. The bottom surface of the torus is about 450 mm above the concrete floor. A concrete biological shield wall surrounds the torus, creating an enclosure called the torus room.

C.5.4 Corrective Actions for Torus Corrosion and Cracking

The Torus Corrosion Monitoring Program is credited for managing the aging of the Nine Mile Point Unit 1 suppression chamber (torus) (Ref. C.5.6.3). Nine Mile Point Unit 1 is required to monitor the torus wall thickness and corrosion rate in order to establish reasonable assurance that the minimum wall thickness is not reached. The effects of loss of material on the intended function(s) of the torus shell are managed in accordance with 10 CFR 54.21(c)(1)(iii).

C.5.5 Structural Integrity Assessment and Test Results

The torus corrosion-monitoring program determines by inspections, measurements, and analyses (Refs. C.5.6.2 and C.5.6.3) the (1) torus shell thickness through UT, (2) corrosion rate through material coupons, and (3) corrosion condition of the accessible external surfaces of the torus support structure through visual inspections (Refs. C.5.6.1, C.5.6.2, and C.5.6.3).

Torus wall UT measurements are obtained at approximately 6-month intervals over a predefined grid system, and corrosion sample coupons are analyzed during each RFO. Corrosion rates are determined through analysis of both data sets, with the most conservative corrosion rate for a particular torus bay used to evaluate aging of the structure. The UT results and corrosion data are trended for future reference. Visual inspection findings for the external support structure are compared to previous inspection results. Monitoring in this manner ensures that the torus shell material will not be reduced to less than the minimum required wall thickness, and that any degradation is detected before there is a loss of intended function. The torus corrosion-monitoring program is adjusted continually to account for industry experience and research. Inspection reports indicate no significant changes in the torus wall corrosion rate (Ref. C.5.6.3).

C.5.6 References

C.5.6.1 Spina, J.A., Constellation Energy, letter to U.S. Nuclear Regulatory Commission, May 26, 2004.

C.5.6.2 Spina, J.A., Constellation Energy, letter to U.S. Nuclear Regulatory Commission, July 14, 2005.

C.5.6.3 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2." NUREG-1900, Volumes 1 and 2. U.S. Nuclear Regulatory Commission: Washington, DC. September 2006.

C.5.6.4 Montgomery, B.S., Constellation Energy, letter to U.S. Nuclear Regulatory Commission, July 23, 2003.

C.5.6.5 Chicago Bridge and Iron Company *Manufacturers Data Report for Nuclear Vessels*. Form N-1, Vessel No. G-1293. Chicago Bridge and Iron Company: Greenville, PA. 1965.

C.6 Oyster Creek Nuclear Generating Station

C.6.1 Introduction

By letter dated July 22, 2005 (Ref. C.6.6.1), AmerGen Energy Company, LLC, submitted an LRA to the NRC for the Oyster Creek Nuclear Generating Station. The results of the NRC staff's evaluation of the LRA were published in April 2007 in NUREG-1875, "Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station" (Ref. C.6.6.2).

C.6.2 Field Observations

Pitting corrosion less than or equal to 1.02 mm (40 mils) was not repaired during the 1984 torus repair and recoating effort because, based on available margins, it was found to be acceptable without any size restriction and satisfied minimum uniform thickness requirements.

Inspection of the immersed coating in 2002 found blistering that was primarily in the shell invert, but also on the upper shell near the waterline (Ref. C.6.6.2). The majority of the blisters remained intact and continued to protect the base metal. However, in several areas there was pitting damage where the blisters had fractured. In addition to blistering, random blemishes that exposed the base metal were identified in the torus immersion region coating (e.g., minor mechanical damage) during the torus coating inspections. They ranged in size from 1.59 to 12.7 mm (0.0625 to 0.5 inches) in diameter. Pitting in these areas was qualitatively evaluated. Pit spreads ranged from less than 0.254 mm (10 mils) to slightly more than 1.02 mm (40 mils) in a few isolated cases. Quantitative pit depth measurements were taken in several locations in the immersion area of bay 1. Pit depths at these sites ranged from 0.23 to 1.07 mm (8 to 42 mils) and were judged to be representative of typical conditions found on the shell. Prior to the 2002 inspection, four pits greater than 1.02 mm (40 mils) were identified. The pit depths were 1.47 mm (58 mils) (one pit in 1988), 1.27 mm (50 mils) (two pits in 1991), and 1.74 mm (69 mils) (one pit in 1992). The pits were evaluated against the local pit depth acceptance criteria and found acceptable. The fractured blisters were repaired to reestablish the protective coating barrier. The following areas have been mapped for trending and analysis during future inspections: one pit of 1.07 mm (42 mils) in bay 1; one pit of 1.74 mm (69 mils) in bay 2; two pits of 1.27 mm (50 mils) in bay 6; and one pit of 1.47 mm (58 mils) in bay 10. Shell thicknesses were evaluated against code requirements and found to satisfy all design- and licensing-basis requirements.

Recent inspections indicate that the average torus shell thickness remains at 9.78 mm (0.385 inches). Based on inspections performed through 1993, it was concluded that the torus shell thickness has remained virtually unchanged following the repair and recoating efforts performed in 1984.

C.6.3 Design Characteristics of the Torus

The primary containment structure comprises the primary containment, containment penetrations, and internal structures (Ref. C.6.6.1). The primary containment is a Mark I design that consists of a drywell, a PSC, and a vent system connecting the drywell and the suppression chamber. The reactor building encloses the containment and provides secondary containment, structural support, shielding, and shelter to the containment, as well as protection for components housed within against external design-basis events.

The as-built torus wall thickness, as discussed above, is 9.78 mm (0.385 inches). The suppression chamber (torus) and vent system were originally coated with Carboline® Carbo-Zinc 11 paint.

The vent system consists of 10 circular vent lines, which form a connection between the drywell and the PSC. The lines enter the suppression chamber through penetrations provided with expansion bellows and join into a common header contained within the air space of the suppression chamber. The header discharge is through 120 downcomer pipes, which terminate below the water level in the torus. The header and the downcomer pipes are supported from the suppression chamber shell.

C.6.4 Corrective Actions for Torus Corrosion and Cracking

Inspection of the suppression chamber and vent system coatings is done by divers every other RFO and for all 20 torus bays during the period of extended operation. The coatings are monitored for cracks, sags, runs, flaking, blisters, bubbles, and other defects. The protective coating monitoring and maintenance and ASME Code (Ref. C.6.6.3) Section XI, Subsection IWE, programs are credited to manage loss of material due to corrosion for the period of extended operation.

To date, pit depths have been found to be acceptable. Only fractured blisters have been repaired. Therefore, the integrity of the torus shell has been verified to have adequate shell thickness margins to ensure that design- and licensing-basis requirements can be maintained.

C.6.5 Structural Integrity Assessment and Test Results

Acceptance criteria for pits are based on engineering analysis that uses the method of ASME Code Case N-597 (Ref. C.6.6.4) as guidance for the calculation of pit depths that will not violate the local stress requirements of either ASME Code Section III, 1977 Edition, or Section VIII, 1962 Edition (Ref. C.6.6.2). The acceptance criteria for pit depth note that isolated pits of 3.175 mm (0.125 inches) in diameter have an allowable maximum depth of 6.63 mm (0.261 inches) anywhere in the shell provided the center-to-center distance between the subject pit and neighboring isolated pits or areas of pitting corrosion is greater than 508 mm (20.0 inches). This criterion includes old pits or old areas of pitting corrosion that have been filled or recoated. Multiple pits that can be encompassed by a 63.5-mm (2.5 inches) diameter circle shall be limited to a maximum pit depth of 3.58 mm (0.141 inches) provided the center-to-center distance between the subject pitted area and neighboring isolated pits or areas of pitting corrosion is greater than 508 mm (20.0 inches). This criterion also includes old pits or old areas of pitting corrosion that have been filled or recoated.

C.6.6 References

C.6.6.1 Swenson, C.N., AmerGen Energy Company, LLC, letter to U.S. Nuclear Regulatory Commission, July 22, 2005.

C.6.6.2 U.S. Nuclear Regulatory Commission. “Safety Evaluation Report Related to the License Renewal of Oyster Creek Generating Station.” NUREG-1875, Volumes 1 and 2. U.S. Nuclear Regulatory Commission: Washington, DC. April 2007.

C.6.6.3 American Society of Mechanical Engineers. “ASME Boiler and Pressure Vessel Code.” American Society of Mechanical Engineers: New York, NY. 2007.

C.6.6.4 — — —. “Requirements for Analytical Evaluation of Pipe Wall Thinning.” ASME Code Case N-597. American Society of Mechanical Engineers: New York, NY. November 18, 2003.

C.7 Pilgrim Nuclear Power Station

C.7.1 Introduction

By letter dated January 25, 2006 (Ref. C.7.6.1), Entergy Nuclear Operations, Inc., submitted an LRA to the NRC for the Pilgrim Nuclear Power Station. The results of the NRC staff's evaluation of the LRA were published in November 2007 in NUREG-1891, "Safety Evaluation Report Related to the License Renewal of Pilgrim Nuclear Power Station" (Ref. C.7.6.2).

C.7.2 Field Observations

In 1999, the submerged regions of all 16 torus bays as well as the drywell to torus vent areas with water accumulation were inspected (Ref. C.7.6.1). The results revealed areas of defects such as depleted zinc, localized pitting corrosion, and minor surface rusting. Degraded areas were recoated to prevent further corrosion and were reexamined.

During RFO 14 in April 2003, ultrasonic thickness examination of the torus shell resulted in several measurements that were below the nominal wall thickness of 16 mm (0.629 inches) (Ref. C.7.6.1). As the measurements were all greater than the minimum allowable thickness of 14.3 mm (0.563 inches), no further actions were taken.

C.7.3 Design Characteristics of the Torus

The primary containment is a Mark I containment consisting of a drywell (which encloses the reactor vessel and the recirculation system), a PSC, and a connecting vent system (Ref. C.7.6.1).

C.7.4 Corrective Actions for Torus Corrosion

The torus is inspected under the ASME Code (Ref. C.7.6.3) Section XI, Subsection IWE, program. Degraded areas identified in 1999 were recoated. Ultrasonic measurements in April 2003 identified areas below nominal wall thickness, but the areas were all greater than the minimum allowable thickness, so no further action was taken.

C.7.5 Structural Integrity Assessment and Test Results

Containment inservice inspection examinations continue to monitor the thickness of the torus shell. In April 2003, the results of the containment inservice inspection general visual walkdown of the primary containment during RFO 14 (April 2003) were compared to those of the previous inspection (Ref. C.7.6.1). The only new indication was in the control rod drive penetration area where there was some surface corrosion, but not of significance. The control rod drive was found to be structurally acceptable. No significant corrosion was found in other areas.

C.7.6 References

C.7.6.1 Balduzzi, M.A., Entergy Nuclear Operations, Inc., letter to U.S. Nuclear Regulatory Commission, January 25, 2006.

C.7.6.2 U.S. Nuclear Regulatory Commission. "Safety Evaluation Report Related to the License Renewal of Pilgrim Nuclear Power Station." NUREG-1891. U.S. Nuclear Regulatory Commission: Washington, DC. November 2007.

C.7.6.3 American Society of Mechanical Engineers. "ASME Boiler and Pressure Vessel Code." American Society of Mechanical Engineers: New York, NY. 2007.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

Several of the 104 commercial nuclear power plants in the U.S. have experienced water leakage from the spent fuel pools and reactor refueling cavities. The leakage and its impact on structures and the corrosion in the boiling-water reactors' Mark I Containment torus steel structure have been documented in license renewal applications. Age-related concrete degradation has also been indicated in other instances. This report encapsulates publicly available information on field activities performed by different licensees to detect the leakage path from the reactor refueling cavity and spent fuel pool, determine the extent of deterioration of the concrete, rebar and steel, and the methods used to stop the leakage, and evaluate the potential impact of the leakage on the load-carrying capacity of deteriorated concrete. Possible causes of corrosion in the containment torus were also addressed for specific locations where corrosion was found, identified, and documented. The program reviewed the sources or causes of occurrences involving age-related degradation of safety-related concrete structures. Finally, topics for further consideration are provided.

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