Perspectives and Insights from ACRS Review of Staff's Safety Evaluations of License Renewal Applications

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ABSTRACT

This report has been prepared for use by the NRC Advisory Committee on Reactor Safeguards (ACRS) in its ongoing review of staff's safety evaluations of license renewal applications. Reactor license renewal regulations and the role of ACRS in license renewal process have been discussed. The Committee's observations and recommendations on staff's safety evaluations of license renewal applications have been summarized to provide insights and perspectives on previous Committee's review of license renewal applications. An overview of international perspectives on materials degradation issues and aging management has also been presented.

The views expressed in this paper are solely those of the author and do not necessarily represent the views of the ACRS.

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ABBREVIATIONS

ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
AMP	Aging Management Program
ASLBP	Atomic Safety and Licensing Board Panel
ATWS	Anticipated Transient without Scram
BNL	Brookhaven National Laboratory
BWR	Boiling Water Reactor
CFR	Code of Federal regulations
CLB	Current Licensing Basis
DBA	Design Basis Accident
EDO	Executive Director for Operations
ESF	Engineered Safety Feature
FSAR	Final Safety Analysis Report
GALL	Generic Aging Lessons Learned
GEIS	Generic Environmental Impact Statement
IAEA	International Atomic Energy Agency
IPA	Integrated Plant Assessment
LOCA	Loss-of-Coolant Accident
LOSP	Loss of Offsite Power
LWR	Light Water Reactor
NEI	Nuclear Energy Institute
NEPA	National Environmental Policy Act
NRC	Nuclear Regulatory Commission
NSSS	Nuclear Steam Supply System
PRA	Probabilistic Risk assessment
PWR	Pressurized Water Reactor
QHOs	Quantitative Health Objectives
RCS	Reactor Coolant System
RPV	Reactor pressure Vessel
SBO	Station Blackout
SER	Safety Evaluation Report
SGTR	Steam Generator Tube Rupture
SNL	Sandia National Laboratories
SRM	Staff Requirements Memorandum
SRV	Safety Relief Valve
SSCs	Systems, Structures, and Components
TLAAs	Time Limited Aging Analyses

1 INTRODUCTION

In the U.S., the original plant life is established by the regulatory process. The Atomic Energy Act and NRC regulations limit the initial operating licenses of nuclear power plants to 40 years, but also permit such licenses to be renewed for up to another 20 years. The original 40-year term was selected on the basis of economic and antitrust considerations, rather than by technical limitations. However, the selection of this term may have resulted in individual plants being designed on the basis of an expected 40-year service life.

The final rule containing the NRC regulations for the license renewal safety review was published in 1995 in Part 54 of the *Code of Federal Regulations* (10 CFR Part 54) [1] known as the "license renewal rule." According to 10 CFR 54.25 each license renewal application shall be referred to the ACRS for a review and report. There are 104 reactors in the U.S. originally licensed to operate for 40 years. To date, the NRC has approved license renewal for 60 reactors. Given the current competitiveness of nuclear power generation, it is expected that most, if not all operating plants, will apply for extending their operating licenses.

The ACRS has contributed significantly to the success of the license renewal program by establishing expectations on the quality of the submittals and of the license renewal programs committed to by licensees.

This report has been prepared for use by the Committee in its ongoing review of staff's safety evaluations of license renewal applications. A number of reference materials, including License Renewal Rule (10 CFR Part 54), Frequently Asked Questions for License Renewal of Nuclear Power Reactors (NUREG-1850) [2], license renewal section of the NRC Public Website [3], and the ACRS reports on the past reviews of license renewal applications were reviewed for the preparation of this report.

The report begins with an overview of license renewal regulations and the role of ACRS in license renewal process. It then summarizes the Committee's observations and recommendations on the previous staff's safety evaluations of license renewal applications. The report also presents an overview of international perspectives on materials degradation issues and aging management.

2 REACTOR LICENSE RENEWAL REGULATIONS

The License Renewal Rule (10 CFR Part 54), first issued in 1991 and then amended in 1995, establishes the technical and procedural requirements for renewing operating licenses. The rule is based on two key principles:

1. The regulatory process, continued into the extended period of operation, is adequate to ensure that the current licensing basis of all currently operating plants provides an acceptable level of safety, with the possible exception of the detrimental effects of aging on certain systems, structures, and components, and possibly a few other issues related to safety only during the period of extended operation, and

2. Each plant's current licensing basis is required to be maintained during the renewal term

The U.S. NRC regulations (10 CFR 54.21) require that each application for a renewal license for a nuclear plant provide an evaluation that addresses the technical aspects of plant aging and describes the ways those effects will be managed over the life of the nuclear plant. This must contain the following information:

a) An Integrated Plant Assessment (IPA)
b) Current Licensing Basis (CLB) changes during NRC review of the application
c) An evaluation of Time Limited Aging Analyses (TLAAs)
d) An Final Safety Analysis Report (FSAR) supplement

An Integrated Plant Assessment identifies and lists structures and components subject to an aging management review (AMR). These include "passive" structures and components that perform their intended function without moving parts or without a change in configuration or properties. These include such components as the reactor vessel, the steam generators, piping, component supports, seismic Category I structures, etc. To be in scope, the item must also be "long-lived" to be considered during the license renewal process. Long-lived means the item is not subject to replacement based on a qualified life or specified time period.

Each year following submittal of the license renewal application and at least three months before scheduled completion of the NRC review, an amendment to the renewal application must be submitted that identifies any change to the CLB of the facility that materially affects the contents of the license renewal application, including the Final Safety Analysis Report supplement.

Time Limited Aging Analyses (TLAAs), as defined in 10 CFR 54.3, are those licensee calculations and analyses that;

1. Involve systems, structures, and components within scope of the license renewal rule, as delineated in 10 CFR 54.4(a);
2. Consider the effects of aging;
3. Involve time-limited assumptions defined by the current operating term, for example, 40 years;
4. Were determined to be relevant by the licensee in making a safety determination;

5. Involve calculations or provide the basis for conclusions related to the capability of the system, structure, and component to perform its intended functions, as delineated in 10 CRF 54.4(b); and
6. Are contained or incorporated by reference in the CLB.

For an evaluation of TLAAs, the applicant must demonstrate that: (i) the analyses remain valid for the period of extended operation; (ii) the analysis have been projected to the end of the period of extended operation; or (iii) the effects of aging on the intended function(s) will be adequately managed for the period of extended operation

A supplement to the Final Safety Analysis Report for the facility must contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation.

The U.S. NRC regulations (10 CFR 54.22) require that each application for a renewal license for a nuclear plant also include any technical specification changes or additions, with justification, necessary to manage the effects of aging during the period of extended operation.

The scope of the License Renewal Rule includes safety-related systems, structures, and components (SSCs), non safety-related SSCs whose failure could affect the performance of safety-related SSCs, and SSCs that are relied on to demonstrate compliance with the NRC's regulations for fire protection, environmental qualification, pressurized thermal shock, anticipated transients without scram, and station blackout. Since active components are already managed by many established plant programs, such as those required by the Maintenance Rule, the focus of the License Renewal Rule is on long-lived passive components that are not subjected to periodic replacement. The implementation of the rule requires that affected components be explicitly identified and subjected to an aging management review. Existing plant programs with enhancements or exceptions, or new programs, must be shown to provide adequate monitoring, inspection, and corrective action for components in scope for license renewal. Systems and components qualified for a 40-year life must be shown to be capable of continued safe operation for the period of extended operation. According to 10 CFR 54.25 each license renewal application shall be referred to the ACRS for a review and report.

In order to support consistent application of the License Renewal Rule, the NRC and the Nuclear Energy Institute (NEI) prepared several license renewal guidance documents. Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," [4] which endorses NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54- The License Renewal Rule," [5] provides guidance on the preparation of license renewal applications.

NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants," [6] provides guidance in reviewing applications. The standard review plan incorporates by reference the "Generic Aging Lessons Learned (GALL) Report," NUREG-1801, [7] which provides the technical bases for evaluating proposed aging management programs and determining which programs should be augmented for license renewal and which programs can adequately manage aging without change. The GALL Report contains a large amount of information regarding the aging and operating experience of passive components in nuclear power plants. Its technical bases are derived from operating experience and research on aging degradation of materials in all plant environments. The GALL Report is subjected to periodic updates as new information becomes available from operating experience, reviews of license renewal applications, and research on aging degradation of plant materials.

In addition to its mission of protecting public health and safety under the Atomic Energy Act, the NRC is charged with protection of the environment in the use of nuclear materials. Each license renewal applicant must include a supplement to the environmental report that contains an analysis of the plant's impact on the environment if allowed to continue operation beyond the initial license. The NRC performs plant-specific reviews of environmental impacts of operating life extension in accordance with National Environmental Policy Act (NEPA) and the requirements of 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions." The environmental protection regulations in 10 CFR Part 51 [8] were revised on December 18, 1996, to improve regulatory efficiency in environmental reviews for license renewal and codify the findings documented in the NUREG-1437, “Generic Environmental Impact Statement (GEIS) for License Renewal of Nuclear Plants.” [9]

3 THE ROLE OF ACRS IN REACTOR LICENSE RENEWAL PROCESS

The license renewal review process proceeds along two parallel paths: One is a safety review, which evaluates whether the plant can continue to operate safely during the period of extended operation. The other is an environmental review, which evaluates the interaction between the plant and the surrounding environment. Figure 1 illustrates the safety and environmental review process and the interrelationships among various review activities. According to 10 CFR 54.25, the safety evaluation aspect of each license renewal application is referred to the ACRS for a review and report. An ACRS review is essential, given the potential safety implications of extending power operation of a significant number of plants for 20 years beyond their current licensed terms.

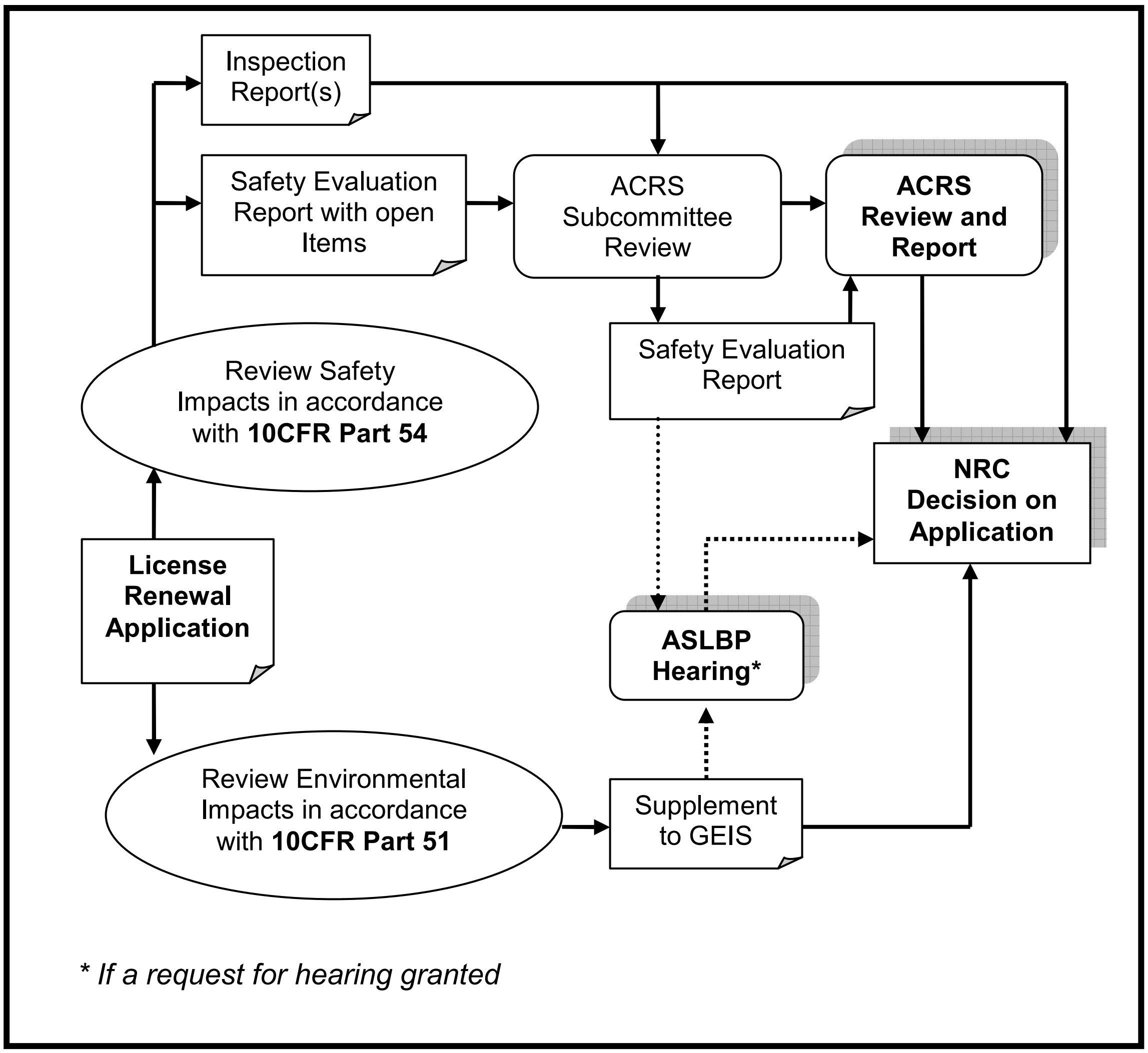

Figure 1 The License Renewal Review Process

The NRC Office of Nuclear Reactor Regulation (NRR) reviews the license renewal application and supporting documentation. The review results in a safety evaluation report. NRC also perform license renewal inspections to sample the process used by the utility to identify the structures and components requiring review and to verify that aging management is being implemented consistent with the application and the staff's safety evaluation report.

The license renewal application and the staff's safety evaluation report are reviewed by the ACRS. ACRS review begins shortly after safety evaluation report with open items is prepared by the staff. In-depth reviews are done by the ACRS License Renewal Subcommittee. With input from Subcommittee members, Subcommittee Chairman develops proposed ACRS position. Briefings by the applicant and the NRC staff are provided to both the Subcommittee and Full Committee. ACRS positions are developed after extensive deliberations by the full Committee. When the Committee has completed its review, its report is submitted to the Commission. At times, ACRS issues "interim" letters to identify issues of concern and items for which additional information, discussions, and clarifications are needed.

When the application for license renewal is submitted, there is an opportunity for individuals or groups to petition for a hearing to address specific issues related to either plant safety or environmental impacts. If granted, a hearing is held and the decision of the Atomic Safety and Licensing Board Panel (ASLBP) is presented to the Commission for its consideration in making a decision on renewing the license.

4 INSIGHTS FROM PREVIOUS ACRS REVIEW OF STAFF'S SAFETY EVALUATIONS OF LICENSE RENEWAL APPLICATIONS

The ACRS has contributed significantly to the success of the license renewal program by establishing expectations on the quality of the submittals and of the license renewal programs committed to by licensees.

The ACRS reviewed earlier drafts of the license renewal guidance documents in 2000 and 2001. In a letter dated April 13, 2001 [10], the ACRS commented that the staff should encourage applicants to include the results of the scoping process in their applications. The Committee noted that this would facilitate the review process by making license renewal applications more understandable. The staff agreed with the ACRS. The improved license renewal guidance documents, including the staff-endorsed NEI license renewal guidance document (NEI 95-10), indicate that an applicant should provide scoping information.

To date, the ACRS has completed the review of 37 license renewal applications and the associated staff's safety evaluation reports involving 64 nuclear power units. The license renewal applications reviewed by the Committee are listed in Table 1. The ACRS recommendations, observations, and comments provided during its past reviews of license renewal applications that may have generic implications are presented below.

<u>CALVERT CLIFFS NUCLEAR POWER PLANT, UNITS 1 AND 2</u>

> *Effective inspections are important to manage aging-induced degradation in order to avoid surprises. It is prudent, for example, to conduct periodic, enhanced visual inspections of reactor internals until data are available to indicate that stress corrosion cracking is not a plausible degradation mechanism in pressurized water reactors. To date, no cracking has been observed in these components at the Calvert Cliffs units.* **(May 19, 1999)**

> *The issue of thermal aging of cast stainless steels has been resolved for the Calvert Cliffs license renewal application. We believe that the resolution proposed in the application is technically satisfactory and could be used by future applicants.* **(May 19, 1999)**

> *Several of the open items such as the effects of the reactor coolant environment on fatigue life and the thermal fatigue of American Society of Mechanical Engineers (ASME) Class 1 small-bore piping may have generic implications for other applications for license renewal***. (DEC. 10, 1999)**

> *After the SER was issued, the staff identified void swelling as a potential mode of degradation for pressurized water reactor vessel internals. Baltimore Gas and Electric Company (BGE) committed to participate in the industry programs to address the significance of void swelling and to develop an inspection program if needed.* **(Dec. 10, 1999)**

Table 1 License Renewal Applications Reviewed by ACRS

Plant	Design	Subcommittee meeting Date	Full Committee meeting Date	ACRS Report Date	Remarks
Calvert Cliffs, Units 1 & 2	PWR	April 28-28, 1999 Nov. 18, 1999	May 5, 1999 Dec. 2, 1999	May 19, 1999 Dec. 10, 1999	
Oconee, Units 1, 2 & 3	PWR	June 30-July 1, 1999 Feb. 24, 2000	Sep. 1, 1999 March 2, 2000	Sep. 13, 1999 March 13, 2000	
Arkansas Nuclear One, Unit 1	PWR	Feb. 22, 2001	March 1, 2001 May 10, 2001	May 18, 2001	
Edwin I. Hatch, Units 1 & 2	BWR	March 25, 2001 Oct. 25, 2001	April 5, 2001 Nov. 8, 2001	April 16, 2001 Nov. 16, 2001	
Turkey Point, Units 3 & 4	PWR	Sep. 25, 2001 March 13, 2002	Oct. 5, 2001 April 4, 2002	April 19, 2002	The first Westinghouse-designed reactor reviewed
North Anna Units1&2 Surry, Units 1&2	PWR	July 9, 2002	Dec. 5, 2002	Dec. 18, 2002	
Peach Bottom, Units 2 & 3	BWR	OCT. 30,2002	Nov. 7, 2002 March 6, 2003	March 14, 2003	
St. Lucie, Units 1 & 2	PWR	April 9, 2003	Sep. 11, 2003	Sep. 17, 2003	
Fort Calhoun, Unit 1	PWR	June 11, 2003	Oct. 1, 2003	Oct. 9, 2003	
McGuire, Units 1&2 Catawba, Units 1&2	PWR	Oct. 8, 2002	Oct. 16, 2002 Feb. 6, 2003	Feb. 14, 2003	The first plants with ice-condenser containment reviewed

Table 1 License Renewal Applications Reviewed by ACRS (continued)

Plant	Design	Subcommittee meeting Date	Full Committee meeting Date	ACRS Report Date	Remarks
H.B. Robinson, Unit 2	PWR	Oct. 30, 2003	March 4, 2004	March 18, 2004	
V.C. Summer Unit 1	PWR	Dec. 3, 2003	March 4, 2004	March 17, 2004	
R.E. Ginna, Unit 1	PWR	Nov. 4, 2003	April 16, 2004	April 23, 2004	The oldest PWR currently in operation in the U.S.
Dresden, Units 2 &3 Quad Cities, Units 1&2	BWR	April 14, 2004	Sep. 9, 2004	Sep. 16, 2004	
Farley, Units 1 &2	PWR	Nov. 3, 2004	April 7, 2005	April 14, 2005	
Arkansas Nuclear One, Unit 2	PWR	Dec. 1, 2004	May 5, 2005	May 13, 2005	
D.C. Cook Units 1&2	PWR	Feb. 9, 2005	July 6, 2005	July 18, 2005	
Millstone Units 2&3	PWR	April 6, 2005	Sep. 8, 2005	Sep. 22, 2005	
Point Beach Units 1 &2	PWR	May 31, 2005	June 1, 2005 Nov. 1, 2005	June 9, 2005 Nov. 18, 2005	
Browns Ferry Units 1,2&3	BWR	Sept. 21, 2005 Oct. 5, 2005	Oct. 6, 2005 March 3, 2006	Oct. 19, 2005 March 23, 2006	

Table 1 License Renewal Applications Reviewed by ACRS (continued)

Plant	Design	Subcommittee meeting Date	Full Committee meeting Date	ACRS Report Date	Remarks
Brunswick, Units 1 & 2	BWR	Feb. 8, 2006	May 4, 2006	May 17, 2006	
Nine Mile Point Units 1 & 2	BWR	April 5, 2006	July 12, 2006	August 2, 2006	
Monticello	BWR	May 30, 2006	Sep. 7, 2006	Sep. 19, 2006	
Palisades	PWR	July 11, 2006	Nov. 1, 2006	Nov. 17, 2006	
Oyster Creek	BWR	Oct. 3, 2006 Jan. 18, 2007	Feb. 1, 2007	Feb. 8, 2007	
Pilgrim	BWR	April 4, 2007	Sep. 6, 2007	Sep. 26, 2007	
Vermont Yankee	BWR	June 5, 2007	Feb. 7, 2008 March 6, 2008	March 20, 2008	
James A. FitzPatrick	BWR	Sep. 5, 2007	March 6, 2008	March 20, 2008	
Wolf Creek Unit1	PWR	March 5, 2008	Sep. 4, 2008	Sep. 17, 2008	
Shearon Harris Unit 1	PWR	May 7, 2008	Oct. 2, 2008	Oct. 16, 2008	
Vogtle Units 1 and 2	PWR	Nov. 5, 2008	April 2, 2009	April 17, 2009	

Table 1 License Renewal Applications Reviewed by ACRS (continued)

Plant	Design	Subcommittee meeting Date	Full Committee meeting Date	ACRS Report Date	Remarks
National Bureau of Standards Test Reactor	Tank type, heavy water-moderated reactor	Feb. 4, 2009	April 2, 2009 June 3, 2009	June 16, 2009	
Beaver Valley Units 1 and 2	PWR	Feb. 4, 2009	July 8, 2009 Sep. 10, 2009	Sep. 16, 2009	
Indian Point Units 2 and 3	PWR	March 4, 2009	Sep. 10, 2009	Sep. 23, 2009	
Three Mile Island Unit1	PWR	April 1, 2009	Sep 10, 2009	Sep. 28, 2009	
Susquehanna Units 1 and 2	BWR	April 1, 2009	Oct. 8, 2009	Oct. 23, 2009	
Prairie Island Units 1 and 2	PWR	July 7, 2009	Dec. 3, 2009	Dec. 10, 2009	

OCONEE NUCLEAR STATION

One-time inspections for evidence of additional plausible modes of degradation for which there is no current experience will be most useful if performed late in the current licensing period. We agree with this strategy and recommend that the staff develop relevant guidance for future applicants. **(September 13, 1999)**

We believe that determination of the design-basis accidents and other accidents that define SSCs within the scope of 10 CFR Part 54 is a generic issue for older plants licensed before NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), was issued in September 1975. Additional guidance needs to be developed for this determination. **(September 13, 1999)**

Although updating the supplement to the Final Safety Analysis Report (FSAR) prior to approving the license renewal application is not required by Part 54, we believe that this should be done and recommend that a requirement for updating the supplement to the FSAR be considered in any future revision to Part 54. **(September 13, 1999)**

Active components such as fuses, which are replaced easily, should not be included in the scope of Part 54. **(September 13, 1999)**

We agree with the staff and industry that additional research and experience are needed to determine the significance of void swelling as a potential mode of degradation for pressurized water reactor internals. Because of the uncertainties, we believe that a focused inspection program as suggested by the staff is a prudent approach for this aging management issue. **(September 13, 1999)**

A number of SER open items involved reactor vessel internal components. Aging effects to be addressed included changes in dimensions due to void swelling, cracking in reactor vessel internal noncast austenitic stainless steel components, cracking of baffle-former bolts, embrittlement of cast austenitic stainless steel components, thermal embrittlement of vent valves, and reduction in fracture toughness. Duke has addressed these open items in the Oconee Reactor Vessel Internals Aging Management Program (RVIAMP). This program includes participation in industry initiatives to investigate these aging effects, inspections, and reports to be provided to the NRC on a periodic basis. A final report will be submitted by Duke to the NRC near the end of the initial license period for Unit 1. The final report will contain the test results from the Babcock & Wilcox Owners Group's RVIAMP and the recommended inspection program for Oconee. On the basis of this information, Duke will implement an aging management program for the reactor vessel internals. We find the proposed program comprehensive and adequate for resolving the reactor vessel internals open items. **(March 13, 2000)**

ARKANSAS NUCLEAR ONE, UNIT 1

The staff should determine whether modification of the current guidance in NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," is required to reflect the lessons learned from the ANO-1 application regarding aging *management of small-bore piping and medium-voltage buried cable.* **(May 18, 2001)**

EDWIN I. HATCH NUCLEAR STATION, UNITS 1 AND 2

Southern Nuclear Operating Company (SNC) incorporated by reference several Boiling Water Reactor Vessel and Internals Project (BWRVIP) topical reports into the Hatch license renewal application. We agree with the staff that the guidelines in the BWRVIP topical reports effectively support license renewal. **(April 16, 2001)**

The SER clarifies staff positions on non-safety-related seismic II-over-I piping systems, long-lived passive components of skid-mounted complex assemblies, fan housings, and damper frames. These clarifications provide significant guidance that could prevent these issues from becoming open items in future applications. They should be incorporated into the generic license renewal guidance documents. **(November 16, 2001)**

We also considered the possibility that the external coating of a tank could be damaged at some location during installation and result in localized fuel oil leakage. Such damage would be of concern during the current license term and, thus, would not be specific to the period of extended operation. The safety consequences would not be significant because the potential leakage would not cause substantial depletion of the fuel oil inventory before it would be detected. We concur with the staff's determination that loss of material of the diesel fuel oil storage tanks is not an aging effect requiring management during the period of extended operation. **(November 16, 2001)**

TURKEY POINT NUCLEAR PLANT, UNITS 3 AND 4

During our review we questioned why certain SSCs were not included in scope, and in all cases the applicant provided appropriate justification for the exclusion. Among these SSCs were the startup transformers that connect the plant to the offsite power source, which typically provides the alternate AC power source during a station blackout (SBO) event. The applicant argued that Turkey Point does not rely on restoration of offsite power to recover from an SBO event. Instead, it relies on the installed capability to cross-connect the emergency diesel generators (EDGs) from one unit to the other. During an SBO event, each of the four EDGs on site is capable of carrying all essential loads of both units. Sufficient diesel fuel is maintained on site to provide the required long-term alternate power source. During our visit to the site, the applicant used the plant simulator to demonstrate its ability to cross-connect the EDGs from the control room. This capability was used during Hurricane Andrew. On this basis, we concur with the applicant that the EDGs provide an effective alternate power source during an SBO event. Subsequently, the staff has determined, however, that components connecting the units to the offsite power source, including the startup transformers, are needed to fulfill the requirements of the SBO Rule. Therefore, they are part of the licensing basis and must be included in the scope of license renewal. The applicant has agreed to meet this requirement. **(APRIL 19, 2002)**

Unlike previous applicants, FPL has not proposed an aging management program for non-EQ medium-voltage cables that are exposed to significant moisture. The *applicant stated that these cables are designed with lead sheath to prevent failure from moisture ingress. The applicant presented information, including significant industry operating experience, [which] indicates that this type of jacket provides an impermeable barrier. Based on this information, we agree with the applicant and the staff that no aging management program is needed for non-EQ medium-voltage cables that are subjected to significant moisture.* **(APRIL 19, 2002)**

NORTH ANNA POWER STATION UNITS 1 AND 2 AND THE SURRY POWER STATION UNITS 1 AND 2

We questioned the method by which reactor coolant piping is to be inspected in light of the failure of the initial volumetric in-service inspection to detect vessel nozzle cracking at V.C. Summer. Although continued improvement in the inspection methodology is warranted, the staff considers current methods adequate to detect primary water stress corrosion cracking. This is a generic issue and we remain concerned with the effectiveness of inspection techniques. Dominion has committed to employ best industry practices as they are developed. **(December 18, 2002)**

During the discussion of time-limited aging analyses, we expressed a concern that the applicant had not submitted its evaluations of the reactor vessel margins for pressurized thermal shock and upper shelf energy. The staff had accepted the applicant's position that these values were acceptable without performing an independent evaluation. Subsequently, the staff obtained this information from the applicant and the staff performed an independent evaluation. Although in some cases the margins are small, we agree with the staff's position that margin does exist. We believe that in the future such critical parameters should be reviewed by the staff. The staff agreed to require that these data be provided with future license renewal applications. **(December 18, 2002)**

In several situations, Dominion and other applicants have committed to actions based on future technology development. In Dominion's case, two examples are (1) the method for inspecting reactor coolant piping, and (2) the method for testing of medium-voltage cables exposed to moisture. The NRC staff needs to continue to keep abreast of these developing technologies and review and approve methodologies at the appropriate time. **(December 18, 2002)**

PEACH BOTTOM ATOMIC POWER STATION UNITS 2 AND 3

The scram at Peach Bottom Unit 2 that occurred on December 21, 2002, highlighted a number of weaknesses in the current corrective action and preventive maintenance programs. We expect that ongoing corrective actions committed by the licensee will resolve these weaknesses. **(March 14, 2003)**

ST. LUCIE NUCLEAR PLANT UNITS 1 AND 2

The groundwater at the St. Lucie site is characterized by high concentrations of chlorides and sulfates that create an aggressive environment for concrete structures. The applicant has committed to enhance those elements of the St. Lucie's Systems and Structures Monitoring Program that deal with inspections of accessible and inaccessible concrete structures. This Program will be enhanced to include specific provisions consistent with industry standards and inspection guidelines for monitoring concrete structures. The monitoring plan for inaccessible concrete structures includes inferring material conditions of inaccessible structures from inspection of accessible structures exposed to groundwater and opportunistic inspections of below-grade concrete. The applicant stated that during construction, concrete of sufficient quality was used to inhibit degradation of concrete and protect the embedded reinforcing steel. No concrete degradation has been found during opportunistic inspections of inaccessible concrete structures performed in 1997 and 2002. Based on this information, we agree with the staff that the enhancements proposed by the applicant provide reasonable assurance that the integrity of concrete structures at St. Lucie will be adequately monitored during the period of extended operation. **(September 17, 2003)**

FORT CALHOUN STATION, UNIT 1

Buckling of the containment liner plate has occurred in a small localized area. The applicant has analyzed this condition and concluded that this buckling does not affect the functionality of the containment liner plate. We agree with the staff that this issue is not an unanalyzed age-related issue. **(October 9, 2003)**

MCGUIRE NUCLEAR STATION UNITS 1 AND 2 AND CATAWBA NUCLEAR STATION UNITS 1 AND 2

The McGuire and Catawba LRA includes a new aging management program, the Alloy 600 Aging Management Review. This program is intended to identify Alloy 600/690, 82/182, and 52/152 locations; to rank susceptibility to primary water stress corrosion cracking (PWSCC); and to verify that nickel-based alloy locations are adequately inspected by the In-service Inspection Program, the Control Rod Drive Mechanism and other Vessel Head Penetration (VHP) programs, the Reactor Vessel Internals Program, and the Steam Generator Integrity Program. This review will provide general oversight and management of cracking due to PWSCC. We applaud this initiative to provide comprehensive oversight of activities to manage PWSCC. Given the current challenge created by PWSCC, we encourage Duke to implement this program soon, in the current license term, rather than waiting for the end of the initial license terms of the four units. **(February 14, 2003)**

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT 2

Robinson Nuclear Plant site has aggressive ground water due to a low pH. The applicant has committed to inspect the dam spillway and the intake structures every 10-years and will also perform opportunistic inspections of inaccessible concrete structures. **(March 18, 2004)**

The pressurizer spray head is not in scope and, given its importance for cooldown, we questioned its omission. The applicant responded that the accident-basis analysis for plant operation does not include pressurizer spray so its exclusion is permissible. The applicant further stated that degradation of the nozzle would be noticed during normal operation. **(March 18, 2004)**

VIRGIL C. SUMMER NUCLEAR STATION

Since earlier ultrasonic testing failed to identify the “A” hot leg to vessel nozzle weld defect before it propagated completely through the pipe wall, we questioned the effectiveness of the applicant’s Alloy 600 AMP for managing primary water stress corrosion cracking (PWSCC) in ASME Class 1 dissimilar welds (e.g., Alloy 82/182 welds). The applicant stated that it continues to take advantage of improvements in ultrasonic testing methods and is now using the latest ultrasonic technology. Furthermore, the applicant has committed to incorporate emerging regulatory requirements and industry recommendations into its Alloy 600 program prior to the period of extended operation. We found the applicant’s commitment acceptable. **(March 17, 2004)**

R. E. GINNA NUCLEAR POWER PLANT

The applicant has indicated that 31 license renewal commitments have been incorporated into the Ginna Corrective Action Tracking System. Less than half of these commitments have been completed. The remaining activities are scheduled for completion prior to the period of extended operation. We encouraged RG&E to establish a schedule for implementing these commitments well ahead of the beginning of the license renewal period so as not to place an unreasonable demand on both the applicant and NRC resources. **(April 23, 2004)**

DRESDEN 2 & 3 AND QUAD CITIES 1 & 2 NUCLEAR POWER STATIONS

The staff should require that, prior to entering the period of extended operation, Exelon conduct an evaluation to ensure that operating experience at extended power uprate (EPU) levels is properly addressed by the aging management programs. The staff should review and approve this evaluation. **(Sept. 16, 2004)**

The steam dryers should be included in the scope of license renewal for Dresden and Quad Cities. **(Sept. 16, 2004)**

The staff should develop guidance to apply [above] Recommendations to future boiling water reactor (BWR) license renewal applications. **(Sept. 16, 2004)**

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

As in previous reviews, we questioned the adequacy of opportunistic inspections of inaccessible buried piping and tanks, in lieu of periodic inspections at a plant-specific frequency, as specified in the GALL Report. The applicant has committed to enhancing its Buried Piping and Tank Inspection Program by performing an inspection within 10 years of entering the period of extended operation unless an opportunistic inspection has occurred within this 10-year period. This program enhancement is appropriate. The staff has also included this 10-year inspection as *new generic guidance in the proposed revision to the GALL Report.* **(April 14, 2005)**

ARKANSAS NUCLEAR ONE, UNIT 2

Implementation is key to effective aging management programs. Although the applicant's Structures Monitoring-Masonry Wall Program is consistent with the GALL Report, the staff's audit of this program revealed that the initial baseline examinations were not documented properly, the first 5-year reexamination was not performed, and qualifications for personnel responsible for walkdowns were not established. The Annual Assessment Letter for ANO, Units 1 and 2, dated March 3, 2004, had already identified a substantive cross-cutting issue concerning problem identification and resolution. Based on the Annual Assessment Letter dated March 2, 2005, the weaknesses in the ANO-2 Problem Identification and Resolution Program appear to have been corrected. Maintaining an effective problem identification and resolution program is critical to the success of the aging management programs. **(May 13, 2005)**

DONALD C. COOK NUCLEAR PLANT, UNITS 1 AND 2

To be effective, the aging management programs need to be appropriately implemented .During the aging management program inspections, the staff found that walkdowns performed as part[s] of the System Walkdown Program were not conducted quarterly as stated in the license renewal application. Also, the applicant noted that it had not evaluated two coupons from the Boral Surveillance Program. This program monitors the performance of absorber materials in the spent fuel pool by periodically measuring the physical and chemical properties of coupon samples that receive a higher radiation dose than the functional boral panels. The applicant has implemented corrective actions to ensure that the commitments will not be missed in the future. **(July 18, 2005)**

MILLSTONE POWER STATION, UNITS 2 AND 3

Analyses of reactor vessel neutron embrittlement (upper shelf energy, pressurized thermal shock screening criterion, and pressure-temperature limits) performed by the applicant and independently verified by the staff demonstrate that the limiting reactor vessel beltline welds and plate materials will satisfy the acceptance criteria for the period of extended operation. Both the applicant and the staff chose to use a conservative lifetime capacity factor of 90 percent for determining neutron fluence. We agree. **(September 22, 2005)**

POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

We recognize that the license renewal rule does not include specific consideration of current operating performance. However, aspects of current performance may affect the development of license renewal programs and commitments as well as the effectiveness of the implemented programs. **(June 9, 2005)**

*An adequate CAP [corrective action program] is a key element in the successful implementation of the aging management programs critical to license renewal. A review of the events leading to the issuance of the CAL [*Confirmatory Action

Letter] *leads to the conclusion that the applicant's CAP has been in a degraded condition for a long time. The Region III staff stated that the problems are not in the design of the program but in its implementation. The inspections have also identified other weaknesses in the area of human performance. Errors in engineering calculations have been identified and are being corrected, but this work is not yet complete. These errors may have an impact on long-lived passive components.* **(June 9, 2005)**

In support of its final SER, the staff normally audits and inspects only a fraction of the license renewal programs and commitments. In the case of the PBNP [Point Beach Nuclear Plant], the staff should take additional actions to increase confidence that the requirements of the license renewal rule have been met and that there is reasonable assurance that aging degradation can be adequately managed. These actions may include, for example, an expanded inspection of license renewal commitments and a focused review of the effectiveness of the CAP before the PBNP enters the period of extended operation. **(June 9, 2005)**

The staff should expand the scope of its post-approval site inspection to verify that all license renewal programs have been implemented and commitments have been met. In addition, the staff should review the effectiveness of the PBNP corrective action program (CAP) before PBNP enters the period of extended operation. **(November 18, 2005)**

<u>BROWNS FERRY NUCLEAR PLANT UNITS 1, 2, AND 3</u>

The plant-specific operating experience for BFN [Browns Ferry Nuclear Plant] Unit 1, by itself, does not fully meet the intent of the license renewal rule. In addition, many components have been subjected to an extended period of layup that is unusual in plant experience. The SER documents in several places how the applicant plans to compensate for the lack of plant-specific operating experience. The final SER should include a cohesive discussion of the applicability of BFN Units 2 and 3 operating experience to Unit 1 and the compensating actions taken where such experience is not sufficient. **(October 19, 2005)**

The final SER should include a description of the attributes of the new Periodic Inspection Program for BFN Unit 1 components that will not be replaced before restart. **(October 19, 2005)**

If the extended power uprate (EPU) is implemented, the staff should require that TVA evaluate Units 1, 2, and 3 operating experience at the uprated power level and incorporate lessons learned into their aging management programs prior to entering the period of extended operation. **(October 19, 2005)**

The drywell refueling seals should be included within the scope of license renewal and be subjected to periodic inspections. Alternatively, as proposed by the staff, the drywell shells should be subjected to periodic volumetric inspections to detect external corrosion. **(March 23, 2006)**

If the extended power uprate (EPU) is implemented before the period of extended operation, the staff should require that TVA evaluate the operating experience of Units 1, 2, and 3 at the uprated power level and then incorporate lessons learned *into their aging management programs prior to entering the period of extended operation.* **(March 23, 2006)**

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

The staff's new two-tiered process for reviewing the scoping of balance of plant (BOP) systems was effective and improved the efficiency of the review. This process should be used by the staff in its review of future license renewal applications. **(May 17, 2006)**

The construction details of the Mark I containments used in this plant are unique. The drywell uses reinforced concrete as the load bearing structural component with an inner liner of carbon steel which serves as a leak-tight membrane. While liner integrity is important to ensure leak tightness, the structural integrity of the liner is not important in maintaining the integrity of the pressure boundary. The applicant proposes a combination of visual inspections to detect liner bulges and corrosion as well as the integrated leak rate tests as an adequate containment liner AMP. The staff has accepted this approach. We concur. **(May 17, 2006)**

NINE MILE POINT NUCLEAR STATION, UNITS 1 AND 2

The applicant's initial license renewal application was not of adequate quality. In reviewing the application, the staff generated 323 Requests for Additional Information (RAIs) and 385 audit questions. The large number of RAIs prompted the applicant to request a delay to prepare an amended license renewal application. The amended license renewal application was more complete and of higher quality. **(August 2, 2006)**

MONTICELLO NUCLEAR GENERATING PLANT

*Aging of the drywell shell of [*Monticello Nuclear Generating Plant] *MNGP will be managed through the use of the ASME Section XI, Subsection IWE Program. We agree with this approach. Even though this Program does not include ultrasonic testing, this approach was chosen by NMC and accepted by the staff because the plant has several design features that prevent water accumulation behind the shell. During each refueling outage, water leakage is monitored from the refueling seal bellows, the drywell air gap drains, and the sand pocket drains. The refueling seal is within the scope of license renewal. Ultrasonic inspections performed in the past did not identify any degradation.* **(September 19, 2006)**

PALISADES NUCLEAR POWER PLANT

The NMC [Nuclear Management Company] application for renewal of the operating license for PNP should be approved. Continued operation during the entire period of extended operation is contingent on the resolution of the issues associated with three Time-Limited Aging Analyses (TLAAs) related to reactor pressure vessel (RPV) integrity. **(November 17, 2006)**

The applicant identified the systems and components requiring TLAAs and reevaluated them for 20 additional years of operation. As required by 10 CFR Part 54, the applicant must identify any exemptions granted under 10 CFR 50.12 which *rely on a TLAA and determine if that exemption should be continued for an additional 20 years of operation. No such exemption currently exists in the PNP licensing basis. The applicant reexamined 23 TLAAs. All of these TLAAs are valid, without restriction, for 20 more years of operation, except for three TLAAs associated with reactor vessel neutron embrittlement, namely: reactor vessel upper shelf energy, reactor vessel pressurized thermal shock, and reactor vessel pressure-temperature curves. In each of these cases, PNP will exceed the acceptance limits prior to the end of the extended period of operation.* **(November 17, 2006)**

OYSTER CREEK GENERATING STATION

We concur with the staff's proposal to impose license conditions to increase the frequency of the drywell inspections and to monitor the two drywell trenches to ensure that the sources of water are identified and eliminated. **(February 8, 2007)**

The staff should add a license condition to ensure that the applicant fulfills its commitment to perform an engineering study prior to the period of extended operation to identify options to eliminate or reduce the leakage in the OCGS [Oyster Creek Generating Station] *refueling cavity liner.* **(February 8, 2007)**

The staff should add a license condition to ensure that the applicant fulfills its commitment to perform a 3-D (dimensional) finite-element analysis of the drywell shell prior to entering the period of extended operation. **(February 8, 2007)**

PILGRIM NUCLEAR POWER STATION

For the TLAAs associated with neutron embrittlement, the applicant used the Radiation Analysis Modeling Application (RAMA) fluence methodology for its reactor vessel fluence evaluations. RAMA is an NRC-approved methodology, but it has not been benchmarked for BWR-3 designs. The calculations of fluence must be benchmarked against at least one credible plant-specific surveillance capsule. The applicant has not completed its benchmarking of the RAMA code for PNPS [Pilgrim Nuclear Power Station] due to discrepancies between the fluence values obtained from the RAMA code and the dosimetry data. An alternative analysis provided by the applicant showed that substantial margin exists for the most limiting components and that the fluence used for the TLAAs was conservative. However, the staff required that the applicant provide a correctly benchmarked analysis for the period of extended operation that meets regulatory requirements. The applicant plans to remove a capsule during a future outage after precisely measuring its location and plans to perform an analysis using this capsule to complete the benchmarking. In parallel, the applicant is working with the Electric Power Research Institute (EPRI) to benchmark the code using data from another BWR-3. The staff concluded that either of these approaches could meet the regulatory requirements. The applicant has committed to complete an analysis that meets the regulatory requirements and submit it to the NRC for approval before entering the period of extended operation. The results of the completed analysis will be reviewed against the fluence values used for the TLAAs to ensure that the values used were conservative. The staff has concluded that this approach is acceptable and has proposed a license condition that would require the analysis to *be submitted to the staff on or before June 8, 2010. We concur with the staff's conclusion and the proposed license condition.* **(September 26, 2007)**

The applicant initially took exception to the GALL Report for the manner in which environmental effects were taken into account in the fatigue analyses. After further discussion with the staff, the applicant made the commitment to be consistent with the GALL Report. The staff will issue a supplement to the final SER to document this commitment. The supplement to the SER was not available for our review, but we concur with the staff's resolution of this issue as discussed at the meeting. **(September 26, 2007)**

<u>VERMONT YANKEE NUCLEAR POWER STATION</u>

The [Vermont Yankee Nuclear Power Station] VYNPS application includes a significant number of exceptions to the approaches specified in the GALL Report. We reviewed these exceptions and agree with the staff that they are acceptable. Other recent license renewal applications have exhibited a similar trend toward an increasing number of exceptions to the GALL Report. The staff agrees that future updates of the GALL Report should incorporate alternative approaches which are used by the industry and have been approved by the staff. This will reduce the number of exceptions to the GALL Report in future applications and will facilitate the staff review. **(March 20, 2008)**

The applicant stated, and the NRC inspectors confirmed, that the VYNPS drywell shell and the torus shell are in good physical condition. The VYNPS drywell design minimizes the potential for water intrusion, provides diverse methods for preventing and identifying potential water leakage into the air gap should this occur, and minimizes corrosion potential since there is no water-retaining foam or insulation in the air gap. The plant has not experienced any refueling bellows or refueling cavity leakage events. Drywell aging will be managed by Inspection Program B of the American Society of Mechanical Engineers (ASME) Code, Section XI, Subsection IWE. These inspections will be augmented with ultrasonic testing (UT) if unexpected flaws or areas of degradation are found. The torus condition meets design requirements, and no margin has been lost due to corrosion since the torus was re-coated in 1998. The torus condition will be monitored by ongoing IWE inspections of the coating and UT measurements for the next three refueling outages. **(March 20, 2008)**

VYNPS has recently completed its first year of operation at 20 percent uprated power level. The applicant stated that inspection of the steam dryers during the first outage following the uprate did not reveal fatigue indications seen elsewhere in the industry. There were indications identified as intergranular stress corrosion cracking which were dispositioned as acceptable. For this outage, flow accelerated corrosion (FAC) inspections were increased by 50% over the pre-uprate number. The applicant stated that the results of these inspections were satisfactory and consistent with the VYNPS analytical modeling for FAC. The enhanced number of inspections will continue through the next two refueling outages to confirm the ability of the VYNPS CHECWORKS model to conservatively predict FAC rates at the uprated power level. **(March 20, 2008)**

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

During our meetings with the staff and the applicant, we reviewed the physical condition of the drywell shell of the Mark 1 containment of JAFNPP and the associated AMPs. Aging of the drywell shell and torus of JAFNPP will be managed through the use of the American Society of Mechanical Engineers (ASME) Code Section XI, Subsection IWE Program. This program provides for inspection of primary containment components and the containment vacuum breaker system piping and components. The aging effects are managed by periodic sampling inspections, evaluation of inspection results, and repair of any significant degradation. Drywell monitoring includes periodic boroscopic inspections of the sand cushion area, visual inspection of the interior drywell caulk seal, and inspection of the drywell interior coating system. The JAFNPP [FitzPatrick Nuclear Power Plant] drywell design minimizes the potential for water intrusion and includes an alarm system that annunciates in the control room if leakage from the refueling cavity occurs during refueling. The applicant stated that the plant has not experienced occurrences of leakage through the refueling bellows into the area monitored by the air gap leakage detection system. The applicant stated, and the NRC inspectors confirmed, that the JAFNPP drywell shell is in good physical condition. **(March 20, 2008)**

Pitting in the wetted area of the torus shell was identified in 1998 when the torus was drained to replace the emergency core cooling system suction strainers. Further inspection of the torus identified pitting in ten areas in four 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 re-coated. They are considered as leading indicators of torus shell condition and are being monitored periodically with ultrasonic testing and visual inspection. The staff and the NRC inspection team reviewed the JAFNPP Containment inservice inspection program and concluded that [Entergy Nuclear Operations] ENO's program includes appropriate requirements for continued inspection of the torus, evaluation of observed degradation, and prediction of remaining service life. We concur with this conclusion. **(March 20, 2008)**

WOLF CREEK GENERATING STATION, UNIT 1

As a result of the staff's review of the WCGS [Wolf Creek Generating Station and other recent license renewal applications, the staff issued draft Regulatory Information Summary (RIS), 2008-xx, "Fatigue Analysis of Nuclear Power Plant Components," for public comment. This draft RIS identifies instances where a simplified fatigue analysis methodology can lead to a non-conservative result. This simplified methodology is not consistent with the methodology described in the American Society of Mechanical Engineers (ASME) Code, Section III, Subarticle NB-3200. In response to the staff's Requests for Additional Information (RAIs), the applicant performed confirmatory analyses for the hot leg surge line nozzle and charging nozzles. **(September 17, 2008)**

The results of the confirmatory analyses indicated that the calculated Cumulative Usage Factors (CUFs) for these components, based on the simplified methodology, are conservative as compared to results based on the ASME NB-

3200 methodology. However, there are two issues not currently fully analyzed, both relate to thermally induced cyclic metal fatigue. To resolve the first metal fatigue issue, the applicant has committed to update the count of thermal cycles for the early years of plant operation, during which thermal cycle counts were not collected in a systematic and rigorous manner. Regarding the second issue, the applicant has recently determined that a thermal sleeve is not present in the charging nozzle as assumed in a previously submitted analysis. The applicant is in the process of performing a reanalysis, which is consistent with the commitment documented in the final SER. Through these license renewal commitments, the applicant will perform the required fatigue analyses in a conservative manner and in sufficient time to permit thorough staff review and approval of these analyses prior to the start of the extended period of operation. **(September 17, 2008)**

<u>SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1</u>

Prior to the period of extended operation, the staff should inspect the applicant's programs for managing water intrusion into underground cable vaults and cable insulation testing. **(October 16, 2008)**

Corrosion of the containment liner at the base slab was detected in 1997. The moisture barrier was replaced in 1998, and only minor corrosion has been observed during subsequent inspections. Minor corrosion and pitting was recorded in 1993, 2000, and 2004 on exterior and interior surfaces of the containment spray and residual heat removal valve enclosures in the Auxiliary Building, which form part of the containment pressure boundary. No significant material loss has been reported. We agree with the staff's conclusion that the HNP ASME Section XI, Subsection IWE Program will adequately detect and manage the effects of containment liner corrosion. **(October 16, 2008)**

The HNP {Shearon Harris Nuclear Power Plant] current licensing basis (CLB) analyses include credit for closure of the feedwater regulating valves and bypass valves as a redundant method for main feedwater isolation during a main steamline break inside containment. According to the CLB, the feedwater regulating and bypass valves are not classified as safety-related components. The valves are located in the Turbine Building and close automatically from a main feedwater isolation signal, a loss of power signal from the reactor protection system, loss of control air, or loss of DC power. The staff raised a concern that the license renewal requirements for safety-related components, specified under 10 CFR 54.4(a)(1) should apply to these valves, due to their main feedwater isolation function. The applicant responded that Section 15.1.5 of the Standard Review Plan specifically allows credit for backup nonsafety-related components to mitigate the consequences of a main steamline break inside containment, following a single failure of an active safety-related isolation valve. As a result, the staff concludes in the final SER that the feedwater regulating and bypass valves are properly categorized as nonsafety-related components, that the requirements of 10 CFR 54.4(a)(2) apply to these valves, and that no additional SSCs need to be included within the HNP license renewal scope to ensure the isolation function of these valves. We agree with these conclusions. **(October 16, 2008)**

VOGTLE ELECTRIC GENERATING PLANT, UNITS 1 AND 2

During its site inspection, the staff observed water in manholes which contain medium voltage cables that are important to safety. We did not see any evidence of environmental qualification of these cables and associated splices for submerged operation. The staff has identified water in manholes as a generic, current operating plant issue in Information Notice 2002-12, "Submerged Safety-Related Electrical Cables," and Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients." During the current period of operation, the staff will address the issue of water in manholes through the Reactor Oversight Process. **(April 17, 2009)**

NATIONAL BUREAU OF STANDARDS TEST REACTOR

In response to the concern regarding the seismic design basis of the reactor, raised during the February 4, 2009 Subcommittee meeting, the applicant performed a seismic walk-down. A potential problem with an unreinforced block wall was noted. The applicant has committed to reanalyze this situation and take necessary corrective actions. The staff will follow up on this issue. **(June 16, 2009)**

The reactor vessel is an aluminum tank in the form of a right circular cylinder with an elliptical bottom head and a flanged top. The vessel is designed and built to comply with the ASME Boiler and Pressure Vessel Code for Unfired Pressure Vessels, with a design pressure of 50 psig. Integrity of the reactor vessel and associated components is monitored by leakage monitoring and by periodic visual inspection of selected accessible portions of the vessel. Recent visual and photographic examinations of the vessel showed a minimum of corrosion or other deterioration of the vessel. Neutron embrittlement was considered as a prospective aging mechanism. The fluence levels are such that the highest embrittlement effect is at the edges of the thimble guides closest to the core. Even in this case, the material retains significant ductility. The neutron fluence at the vessel walls is too low to cause significant embrittlement within the proposed vessel lifetime. **(June 16, 2009)**

BEAVER VALLEY POWER STATION, UNITS 1 AND 2

During its site inspection, the staff observed water in manholes that contain medium-voltage cables that are important to safety. The applicant has agreed that, although the cables may be suitable for submerged service, they are not qualified for that service. They have made commitments to demonstrate, using an acceptable methodology, that the cables will continue to perform their intended function; or will implement measures to minimize cable exposure to significant moisture; or will replace the cables with cables qualified for submerged service. **(September 16, 2009)**

The impact of containment liner corrosion on the current licensing basis of the plant is being reviewed and will be resolved under the provisions of the applicant's current 10 CFR Part 50 operating licenses. **(September 16, 2009)**

INDIAN POINT NUCLEAR GENERATING UNITS 2 AND 3

The applicant has committed to a quarterly sampling program to test for changes in tritium concentrations in groundwater in close proximity to the IP2 spent fuel pool. The applicant has installed over 40 monitoring wells, most of which are multi-level and range up to 300 feet in depth. These wells are configured with level transducers and sample ports for chemical and radiological sampling. Any significant changes in the groundwater, such as an increase in the tritium level, will be evaluated as an indication of potential leakage from the IP2 spent fuel pool. If leakage is identified in the future, it will be resolved using the applicant's corrective action program. **(September 23, 2009)**

Industry experience with leaks in buried piping and tanks has revealed the need for an inspection program. As a result of Indian Point operating experience, such as the recent IP2 condensate return line leak, Entergy amended its buried piping and tanks inspection program to include additional testing of buried components. They have committed to 51 inspections prior to entering the period of extended operation and additional periodic inspections during the period of extended operation. This inspection and monitoring program is consistent with the GALL Report and significantly exceeds the minimum number of inspections required in similar programs at other plants. **(September 23, 2009)**

THREE MILE ISLAND NUCLEAR STATION, UNIT 1

In the 1990s, corrosion of the containment liner was detected in several locations behind and just above the moisture barrier at the containment floor. The corrosion was caused by leakage of borated water and degradation of the moisture barrier seal that allowed water to collect between the moisture barrier and the inner surface of the liner. The moisture barrier was replaced in 2007. At that time, the previous licensee inspected the entire periphery of the liner to a depth of approximately four to eight inches in the opened gap and confirmed that no corrosion extended below the moisture barrier sealing surface. The applicant has verified that the containment liner currently meets all design requirements, and all I dentified locations of the corrosion have been recorded. The applicant will perform weld repairs to restore the liner to its nominal thickness for all locations where the base metal thickness is reduced by more than 10%. These repairs will be performed during the 2009 outage to replace the steam generators. We agree with the staff's conclusion that these corrective actions and continued monitoring through the ASME Section XI, Subsection IWE Program will provide adequate assurance of the liner integrity. **(September 28, 2009)**

High-density spent fuel storage racks containing Boral panels were installed at TMI-1 in 1992. Different rack designs are used in two regions of the spent fuel pool. In Region 1, the racks have a water gap between adjacent storage cells that functions as a flux trap. In Region 2, there is no water gap between adjacent cells. The Boral panels in Region 1 have thinner sheathing than the panels in Region 2. Corrosion and blistering of Boral surveillance coupons were detected in 1997 and 2008. The largest blister was approximately 1-inch in diameter, 0.058-inch deep, and was filled with water. The previous licensee performed analyses to confirm that the largest blister would not reduce the neutron absorption capacity of the Boral or *the neutron attenuation in the Region 1 water gaps, even if the blister were filled with gas. The staff concluded that the TMI-1 Water Chemistry Program and the Boral Surveillance Program will adequately manage the aging effects from Boral corrosion during the period of extended operation. The staff also stated that Interim Staff Guidance is currently being prepared to address the general topic of neutron absorbing materials in fuel storage racks. We agree that these programs will adequately manage the effects of Boral corrosion.* **(September 28, 2009)**

During the site AMP audit, the staff observed water in manholes which contain medium voltage cables that are important to safety. The staff identified water intrusion into underground cable ducts and manholes as a generic, current operating plant issue in Information Notice 2002-12, "Submerged Safety-Related Electrical Cables," and in Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients." The applicant stated that they will re-grade areas surrounding cable manholes, replace manhole lid gaskets, and refurbish cable vault French drains to minimize water intrusion. The staff will continue to address this issue through the Reactor Oversight Process during the current period of operation.
(September 28, 2009)

The license renewal application did not fully document TMI-1 plant-specific operating and maintenance experience for the five- to ten-year period that is recommended by the Nuclear Energy Institute (NEI) guideline NEI 95-10. Exelon originally referred to information in the Electric Power Research Institute (EPRI) report, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 4," as a surrogate for TMI-1 plant-specific operating and maintenance experience from November 30, 2001 through December 31, 2004. Actual plant-specific operating and maintenance records were reviewed only for the period from January 1, 2005 through November 30, 2006. In response to questions raised during our interim review of the TMI-1 license renewal application and the associated NRC staff's SER with open items, the applicant completed a full review of the TMI-1 plant-specific operating experience for the five-year period ending November 30, 2006. The staff audited this operating experience during a supplemental inspection conducted in July 2009. We agree that the augmented plant-specific operating experience review appropriately supports the license renewal application. **(September 28, 2009)**

<u>SUSQUEHANNA STEAM ELECTRIC STATION, UNITS 1 AND 2</u>

The staff's review of the applicant's operating experience revealed that inaccessible medium voltage cables in certain manholes at SSES have experienced significant exposure to water, i.e., cable in standing water for more than a few days. In addition, during a walk down, the staff found several feet of water in Manhole Numbers 2 and 16. The staff identified water in manholes as a generic, current operating plant issue in Information Notice 2002-12, "Submerged Safety-Related Electrical Cables," and Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures That Disable Accident Mitigation Systems or Cause Plant Transients." The staff will resolve the issue of water in manholes during the current period of operation through the Reactor Oversight Process, in accordance with the requirements of 10 CFR Part 50. ***(October 23, 2009)***

PPL has committed to implement a Non-EQ Inaccessible Medium-Voltage Cables Program involving two parts: first inspection (and draining, if necessary) of the applicable manholes on a periodic basis; and second, the development of a testing program to confirm that the conductor insulation on the applicable cables is not degrading. This program applies to six cables associated with the offsite power supply for SSES. These are the only inaccessible medium voltage cables at SSES that are within the scope of license renewal. These cables are exposed to significant moisture and are energized more than 25% of the time. The Non-EQ Inaccessible Medium-Voltage Cable Program is a new AMP which will require the applicant to test the cables and to evaluate plant-specific operating experience to determine an appropriate inspection frequency for the manholes.
(October 23, 2009)

PRAIRIE ISLAND NUCLEAR GENERATING PLANT (PINGP), UNITS 1 AND 2

For 20 years, both PINGP units have experienced intermittent leakage of borated water from their refueling cavities, when flooded for refueling, into and through the reinforced concrete structures within containments. The leak rate has been 1 to 2 gallons per hour, as measured by accumulation in lower levels of the containments. Earlier efforts to locate and seal the sources of this leakage were not effective, and additional measures have recently been taken to prevent further leakage. The staff established an Open Item to address three issues related to the prior, and any future, refueling cavity leakage: (1) the leaking borated water may contact the containment vessel, remain in contact with the vessel between outages, and cause degradation; (2) the leaking borated water may contact the concrete reinforcement and cause degradation; and (3) the leaking borated water may react with the concrete and cause degradation. **(December 10, 2009)**

Also, the applicant has committed to perform additional inspections and evaluations, prior to entering the period of extended operation, to ensure that no unacceptable degradation of the containment vessel, the concrete reinforcement, or the concrete has occurred as a result of the intermittent leakage from the refueling cavities. The commitments include the removal of concrete from a low point in the containment and inspection of the exposed containment vessel bottom head and reinforcing steel. Also, the applicant will remove concrete samples known to have been exposed to borated water leakage, test them for compressive strength, and perform a petro graphic examination to evaluate for degradation
(December 10, 2009).

The staff identified water in manholes as a generic, current operating plant issue in Information Notice 2002-12, "Submerged Safety-Related Electrical Cables," and Generic Letter 2007-01, "Inaccessible or Underground Power Cable Failures that Disable Accident Mitigation Systems or Cause Plant Transients." There is one manhole at PINGP that contains medium voltage cables in scope of license renewal and is subject to periodic inspection for the accumulation of water.
(December 10, 2009)

NSPM has committed to implement a Non-EQ Inaccessible Medium-Voltage Cables Program involving two parts: first inspection (and draining, if necessary) of the applicable manholes on a periodic basis; and second, the conduct of periodic testing to confirm that the condition of the conductor insulation on the applicable *cables is not degrading. This new AMP will be implemented prior to entering the period of extended operation.* **(December 10, 2009)**

5 INTERNATIONAL PERSPECTIVES ON MATERIALS DEGRADATION ISSUES AND AGING MANAGEMENT

Safe control of aging of nuclear power plants is an important concern for all plant owners and safety regulatory authorities in the world.

Contrary to U.S., there is no expiration time for the operating license in many countries (see Table 2). The periodic safety review is the principal method applied to reactors to ensure that the plant is adequately safe for a further period of operation. However, according to different countries, the operating authorization given by the regulatory authority to the plant operator is not associated with the same formal process. Formal aging management evaluation processes exist in some countries, for quite short periods (i.e., one year in Spain, two in the U.K.); in others, it appears through a requirement of ability for safety demonstration at any moment (in France and Belgium). In practice, safety aging management is implemented through the periodic safety review approach widely accepted in many countries. The periodic safety review is a safety concept mainly developed in the European countries and was introduced later in the International Atomic Energy Agency (IAEA) documents [11]. The periodic safety reviews are complementary to the routine reviews of nuclear power plant operation (including modifications to hardware and procedures, significant events, and operating experience) and special safety reviews following major events of risk significance. The frequency of review varies from country to country; typically every ten years. The periodic safety review necessitates licensees to take into account advances in technology unconstrained by licensing basis as in U.S.

The objective of these periodic safety reviews are to assess the cumulative effects of plant aging and plant modifications, operating experience, technical developments, and siting aspects. The reviews include an assessment of plant design and operation against current safety standards and practices in order to propose any eventual improvement. The reviews also examine an extension of the original design basis of the plant, in particular postulated initiating events (internal and external) not considered earlier.

The material degradation issues have been the subject of numerous studies in different countries and by several international organizations [12]. These studies have led to the establishment of various programs or projects specifically dedicated to the management of aging of SSCs.

The various aging aspects leading to slow degradation of SSCs are evaluated during periodic safety assessment. However, aspects related to more quick changes (in particular those affecting active components) are managed on a continuous basis through an appropriate maintenance and component qualification.

Table 2 Operating License Periods in Various Countries [13]

Country	License Period Approach
United States of America	Fixed Term (40 years, with 20-year renewal option)
France	Lifetime
Japan	Lifetime
United Kingdom	Lifetime
Republic of Korea	Lifetime
Germany	Lifetime
Canada	Fixed term (2-5 years)
Sweden	Lifetime
Spain	Variable (5-10 years) Case-by-case, no fixed term but moving to 10-year standard for nuclear facilities that complete periodic safety review
Belgium	Lifetime
Czech Republic	Lifetime
Switzerland	Lifetime (except for 2 plants with term licenses based on historical technical concerns)
Finland	Fixed term (10-20 years)
Hungry	Lifetime
Mexico	Fixed term (30 years)
Netherland	Lifetime

6 SUMMARY AND CONCLUSIONS

An overview of license renewal regulations and the role of ACRS in license renewal process were presented. According to 10 CFR 54.25, the safety evaluation aspect of each license renewal application is referred to the ACRS for a review and report.

To date, the ACRS has completed the review of 30 license renewal applications and the associated staff's safety evaluation reports involving 54 nuclear power units. The ACRS recommendations, observations, and comments provided during its past reviews of license renewal were presented to provide insights and perspectives on previous Committee's review of license renewal applications.

An overview of international perspectives on materials degradation issues and aging management were also presented.

7 REFERENCES

1. *Code of Federal regulations,* Title 10, part 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants.*

2. U.S. Nuclear Regulatory Commission, “Frequently Asked Questions for License Renewal of Nuclear Power Reactors,” NUREG-1850, March 2006.

3. U.S. Nuclear Regulatory Commission, Reactor License Renewal, http://www.nrc.gov/reactors/operating/licensing/renewal.html

4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.188, “Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses,” revision 1, September 2005.

5. Nuclear Energy Institute, “Industry Guideline for Implementing the Requirements of 10 CFR Part 54- The License Renewal Rule,” NEI 95-10, revision 5, January 2005.

6. U.S. Nuclear Regulatory Commission, “Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants,” NUREG-1800, Rev.1, September 2005.

7. U.S. Nuclear Regulatory Commission, “Generic Aging Lessons Learned (GALL) Report,” NUREG-1801, Rev. 1, September 2005.

8. *Code of Federal regulations,* Title 10, part 51, Environmental Protection Regulations for Domestic licensing and Related Regulatory Functions,

9. U.S. Nuclear Regulatory Commission, “Generic Environmental Impact Statement for License Renewal of Nuclear Plants,” NUREG-1437, May 1996.

10. Report dated April 13, 2001, from G. E. Apostolakis, Chairman, Advisory Committee on Reactor Safeguards, to R. A. Meserve, Chairman, U.S. Nuclear Regulatory Commission, Subject: Proposed Final License Renewal Guidance Documents.

11. International Atomic Energy Agency, “Periodic Safety Review of Nuclear Power Plants,” IAEA Safety Standards Series, Safety Guide No. NS-G-2.10, 2003.

12. European Commission, “Nuclear Safety and the Environment, Safe Management of NPP Ageing in the European Union,” EUR 19843 EN, May 2001.

13. H.P. Nourbakhsh, “An Overview of Differences in Nuclear Safety Regulatory Approaches and Requirements between United States and Other Countries,” ACRS, October, 2004.