NUREG-1800

Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants

U.S. Nuclear Regulatory Commission
Office Nuclear Reactor Regulation
Washington, DC 20555-0001

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Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants

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Division of Regulatory Improvement Programs
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

UNITED STATES NUCLEAR REGULATORY COMMISSION

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Abstract

The Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) provides guidance to Nuclear Regulatory Commission staff reviewers in the Office of Nuclear Reactor Regulation. These reviewers perform safety reviews of applications to renew nuclear power plant licenses in accordance with Title 10 of the Code of Federal Regulations Part 54. The principal purposes of the SRP-LR are to ensure the quality and uniformity of staff reviewers and to present a well-defined base from which to evaluate applicant programs and activities for the period of extended operation. The SRP-LR is also intended to make information about regulatory matters widely available, to enhance communication with interested members of the public and the nuclear power industry, and to improve their understanding of the staff review process. The safety review is based primarily on the information provided by the applicant in a license renewal application. Each of the individual SRP-LR sections addresses (1) who performs the review, (2) the matters that are reviewed, (3) the basis for review, (4) the way the review is accomplished, and (5) the conclusions that are sought.

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ABBREVIATIONS

AFW	auxiliary feedwater
AMP	aging management program
AMR	aging management review
ANL	Argonne National Laboratory
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transients without scram
B&W	Babcock & Wilcox
BWR	boiling water reactor
BWRVIP	Boiling Water Reactor Vessel and Internals Project
CASS	cast austenitic stainless steel
CDF	core damage frequency
CE	Combustion Engineering
CFR	Code of Federal Regulations
CLB	current licensing basis
CRD	control rod drive
CUF	cumulative usage factor
DBA	design basis accident
DBE	design basis event
DG	Draft Regulatory Guide
DOR	Division of Operating Reactors
ECCS	emergency core cooling system
EDG	emergency diesel generator
EFPY	effective full power year
EPRI	Electric Power Research Institute
FAC	flow-accelerated corrosion
FR	Federal Register
FSAR	Final Safety Analysis Report
GALL	Generic Aging Lessons Learned
GE	General Electric
GL	generic letter
GSI	generic safety issue
HAZ	heat-affected zone
HELB	high-energy line break
HPCI	high-pressure coolant injection
HVAC	heating, ventilation, and air conditioning
I&C	instrumentation and control
IASCC	irradiation assisted stress corrosion cracking
IEEE	Institute of Electrical and Electronics Engineers
IGA	intergranular attack

ABBREVIATIONS (continued)

IGSCC	intergranular stress corrosion cracking
IN	information notice
INPO	Institute of Nuclear Power Operations
IPA	integrated plant assessment
IPE	individual plant examination
IPEEE	individual plant examination of external events
IR	insulation resistance
ISI	inservice inspection
ITG	Issues Task Group
LCD	liquid crystal display
LED	light-emitting diode
LER	licensee event report
LOCA	loss of coolant accident
LTOP	low-temperature overpressure protection
MIC	microbiologically influenced corrosion
MRV	minimum required value
NDE	nondestructive examination
NDT	nil-ductility temperature
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPS	nominal pipe size
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
NSAC	Nuclear Safety Analysis Center
NSSS	nuclear steam supply system
ODSCC	outside diameter stress corrosion cracking
OM	operation and maintenance
P&ID	piping and instrument diagrams
PLL	predicted lower limit
PRA	probabilistic risk analysis
PT	penetrant testing
P-T	pressure-temperature
PTS	pressurized thermal shock
PWR	pressurized water reactor
PWSCC	primary water stress corrosion cracking
QA	quality assurance
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RPV	reactor pressure vessel
RT	reference temperature

ABBREVIATIONS (continued)

SBO	station blackout
SCC	stress corrosion cracking
SER	safety evaluation report
SG	steam generator
S/G	standards and guides
SOC	statement of considerations
SOER	significant operating experience report
SRM	staff requirements memorandum
SRP	standard review plan
SRP-LR	standard review plan for license renewal
SS	stainless steel
SSC	systems, structures, and components
SSE	safe shutdown earthquake
TLAA	time-limited aging analysis
UFSAR	updated final safety analysis report
USI	unresolved safety issue
UT	ultrasonic testing
UV	ultraviolet

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INTRODUCTION

The Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants (SRP-LR) provides guidance to Nuclear Regulatory Commission (NRC) staff reviewers in the Office of Nuclear Reactor Regulation (NRR). These reviewers perform safety reviews of applications to renew nuclear power plant licenses in accordance with Title 10 of the *Code of Federal Regulations* (CFR) Part 54. The principal purposes of the SRP-LR are to ensure the quality and uniformity of staff reviews and to present a well-defined base from which to evaluate applicant programs and activities for the period of extended operation. The SRP-LR is also intended to make information about regulatory matters widely available, to enhance communication with interested members of the public and the nuclear power industry, and to improve their understanding of the staff review process.

The safety review is based primarily on the information provided by the applicant in a license renewal application. The NRC regulation, in 10 CFR 54.21, requires that each license renewal application shall include an integrated plant assessment (IPA), current licensing basis (CLB) changes during review of the application by NRC, an evaluation of time-limited aging analyses (TLAAs), and a final safety analysis report (FSAR) supplement.

In addition to the technical information required by 10 CFR 54.21, a license renewal application must contain general information (10 CFR 54.19), necessary technical specification changes (10 CFR 54.22), and environmental information (10 CFR 54.23). The application must be sufficiently detailed to permit the reviewers to determine (1) whether there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the CLB and (2) whether any changes made to the plant's CLB to comply with 10 CFR Part 54 are in accord with the Atomic Energy Act of 1954 and NRC regulations.

Before submitting a license renewal application, an applicant should have analyzed the plant to ensure that actions have been or will be taken to (1) manage the effects of aging during the period of extended operation (this determination should be based on the functionality of structures and components that are within the scope of license renewal and that require an aging management review), and (2) evaluate TLAAs. The license renewal application is the principal document in which the applicant provides the information needed to understand the basis upon which this assurance can be made.

10 CFR 54.21 specifies, in general terms, the technical information to be supplied in the license renewal application. Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," proposes to endorse the Nuclear Energy Institute (NEI) guidance in NEI 95-10, Rev. 3, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 — The License Renewal Rule." NEI 95-10 provides guidance on the format and content of a license renewal application. The SRP-LR sections are keyed to the RG-1.188 Standard Format document; the sections are numbered according to the section numbers in that document.

During the review of the initial license renewal applications, NRC staff and the applicants have found that most of the programs to manage aging that are credited for license renewal are existing programs. In a staff paper (SECY 99-148), "Credit for Existing Programs for License Renewal," dated June 3, 1999, the staff described options and provided a recommendation for crediting existing programs to improve the efficiency of the license renewal process. In a staff requirements memorandum (SRM) dated August 27, 1999, the NRC approved the staff recommendation and directed the staff to focus the review guidance in the SRP-LR on areas where existing programs should be augmented for license renewal. Under the terms of the SRM, the SRP-LR would reference a "Generic Aging Lessons Learned" (GALL) report, which evaluates existing programs generically, to document (1) the conditions under which existing programs are considered adequate to manage identified aging effects without change and (2) the conditions under which existing programs should be augmented for this purpose.

The GALL report (NUREG-1801) should be treated as an approved topical report. The NRC reviewers should not repeat their review of a matter described in the GALL report, but should find an application acceptable with respect to such a matter when the application references the GALL report and the evaluation of the matter in the GALL report applies to the plant. However, reviewers should ensure that the material presented in the GALL report is applicable to the specific plant involved and that the applicants have identified specific programs as described and evaluated in the GALL report if they rely on the report for license renewal.

The SRP-LR is divided into four major chapters: (1) Administrative Information; (2) Scoping and Screening Methodology for Identifying Structures and Components Subject to Aging Management Review, and Implementation Results; (3) Aging Management Review Results; and (4) Time-Limited Aging Analyses. The appendixes to the SRP-LR list branch technical positions. The SRP-LR addresses various site conditions and plant designs and provides complete procedures for all of the areas of review pertinent to each of the SRP-LR sections. For any specific application, NRC reviewers may select and emphasize particular aspects of each SRP-LR section, as appropriate for the application. In some cases, the major portion of the review of a plant program or activity may be done on a generic basis (with the owners' group of that plant type) rather than in the context of reviews of particular applications from utilities. In other cases, a plant program or activity may be sufficiently similar to that of a previous plant that a complete review of the program or activity is not needed. For these and similar reasons, reviewers need not carry out in detail all of the review steps listed in each SRP-LR section in the review of every application.

The individual SRP-LR sections address (1) who performs the review, (2) the matters that are reviewed, (3) the basis for review, (4) the way the review is accomplished, and (5) the conclusions that are sought. One of the objectives of the SRP-LR is to assign review responsibilities to the appropriate NRR branches. Each SRP-LR section identifies the branch that has the primary review responsibility for that section. In some review areas, the primary branch may require support; the branches that are assigned these secondary review responsibilities are also identified for each SRP-LR section.

Each SRP-LR section is organized into the following six subsections, generally consistent with NUREG-0800 "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," (July 1981).

1. **Areas of Review**

   This subsection describes the scope of review, that is, what is being reviewed by the branch that has primary review responsibility. It contains a description of the systems, structures, components, analyses, data, or other information that are reviewed as part of the license renewal application. It also contains a discussion of the information needed or the review expected from other branches to permit the primary review branch to complete its review.

2. **Acceptance Criteria**

This subsection contains a statement of the purpose of the review, an identification of applicable NRC requirements, and the technical basis for determining the acceptability of programs and activities within the area of review of the SRP-LR section. The technical bases consist of specific criteria, such as NRC regulatory guides, codes and standards, and branch technical positions.

Consistent with the approach described in NUREG-0800, the technical bases for some sections of the SRP-LR can be provided in branch technical positions or appendixes as they are developed and can be included in the SRP-LR.

3. **Review Procedures**

This subsection discusses the way the review is accomplished. It is generally a step-by-step procedure that the reviewer follows to provide reasonable verification that the applicable acceptance criteria have been met.

4. **Evaluation Findings**

This subsection presents the type of conclusion that is sought for the particular review area. For each section, a conclusion of this type is included in the safety evaluation report (SER), in which the reviewers publish the results of their review. The SER also contains a description of the review, including which aspects of the review were selected or emphasized; which matters were modified by the applicant, required additional information, will be resolved in the future, or remain unresolved; where the applicant's program deviates from the criteria provided in the SRP-LR; and the bases for any deviations from the SRP-LR or exemptions from the regulations.

5. **Implementation**

This subsection discusses the NRC staff's plans for using the SRP-LR section.

6. **References**

This subsection lists the references used in the review process.

The SRP-LR incorporates the staff experience in the review of the initial license renewal applications. It may be considered a part of a continuing regulatory framework development activity that documents current methods of review and provides a basis for orderly modifications of the review process in the future. The SRP-LR will be revised and updated periodically, as needed, to incorporate experience gained during future reviews, to clarify the content or correct errors, to reflect changes in relevant regulations, and to incorporate modifications approved by the NRR Director. A revision number and publication date are printed in a lower corner of each page of each SRP-LR section. Because individual sections will be revised as needed, the revision numbers and dates will not be the same for all sections. The table of contents indicates the revision numbers of the most current sections. Comments and suggestions for improvement should be sent to the Director, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Notices of errors or omissions should be sent to the same address.

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CHAPTER 1

ADMINISTRATIVE INFORMATION

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1.1 DOCKETING OF TIMELY AND SUFFICIENT RENEWAL APPLICATION

Review Responsibilities

Primary - Branch responsible for license renewal projects
Secondary - Branch responsible for environmental review and branches responsible for technical review, as appropriate

1.1.1 Areas of Review

This section addresses (1) the review of the acceptability of a license renewal application for docketing in accordance with 10 CFR 2.101 and the requirements of 10 CFR Part 54, and (2) whether a license renewal application is timely and sufficient, which allows the provisions of 10 CFR 2.109(b) to apply. Allowing this regulation, which was written to comply with the Administrative Procedures Act, to apply to the application means that the current license will not expire until the NRC makes a final determination on the license renewal application.

The review described in this section is not a detailed, in-depth review of the technical aspects of the application. The docketing and finding of a timely and sufficient renewal application does not preclude the NRC reviewers from requesting additional information as the review proceeds, nor does it predict the NRC's final determination regarding the acceptance or rejection of the renewal application. A plant's current license will not expire after the passing of the license's expiration date if the renewal application was found to be timely and sufficient. During this time, and until the renewal application has been approved by the NRC, the licensee must continue to perform its activities in accordance with the facility's CLB, including all applicable license conditions, orders, rules, and regulations.

In determining whether an application is acceptable for docketing, the following areas of the license renewal application are reviewed.

1.1.1.1 Docketing and Sufficiency of Application

The license renewal application is reviewed for acceptability for docketing as a sufficient application in accordance with 10 CFR 2.101, 10 CFR Part 51, and 10 CFR Part 54.

1.1.1.2 Timeliness of Application

The timeliness of a license renewal application is reviewed in accordance with 10 CFR 2.109(b).

1.1.2 Acceptance Criteria

1.1.2.1 Docketing/Sufficiency of Application

NRC staff determine acceptance for docketing and sufficiency on the basis of the required contents of an application, established in 10 CFR 2.101, 10 CFR 51.53(c), 54.17, 54.19, 54.21, 54.22, and 54.23. A license renewal application is sufficient if it contains the reports, analyses, and other documents required in such an application.

1.1.2.2 Timeliness of Application

In accordance with 10 CFR 2.109(b), a license renewal application is timely if it is submitted at least 5 years before the expiration of the current operating license and it is determined to be sufficient.

1.1.3 Review Procedures

A licensee may choose to submit plant-specific reports addressing portions of the license renewal rule requirements for NRC review and approval prior to submitting a renewal application. An applicant may incorporate (by reference) these reports or other information contained in previous applications for licenses or license amendments, statements, or correspondence filed with the NRC, provided that the references are clear and specific. However, the final determination of the docketing of a sufficient renewal application is made only after a formal license renewal application has been submitted to the NRC.

For each area of review, NRC staff should implement the following review procedures.

1.1.3.1 Docketing and Sufficiency of Application

Upon receipt of a tendered application for license renewal, the reviewer should determine whether the applicant has made a reasonable effort to provide the required administrative, technical, and environmental information.[1] The reviewer should use the review checklist provided in Table 1.1-1 to determine whether the application is reasonably complete and conforms to the requirements outlined in 10 CFR Part 54.

Items I.1 through I.10 in the checklist address administrative information: for the purpose of this review, the reviewer should check the "Yes" column if the required information is included in the application. Item II in the checklist addresses timeliness of the application.

Items II.1 through II.3, III, and IV in the checklist address technical information, the FSAR supplement, and technical specification changes, respectively. Chapters 2, 3, and 4 of the SRP-LR provide information regarding the technical review. Although the purpose of the docketing and sufficiency review is not to determine the technical adequacy of the application, the reviewer should determine whether the applicant has provided reasonably complete information in the application to address the renewal rule requirements. The reviewer may request assistance from appropriate technical review branches to determine whether the application provides sufficient information to address the items in the checklist so that the staff can begin their technical review. The reviewer should check the "Yes" column for a checklist item if the applicant has provided reasonably complete information in the application to address the checklist item.

Item V of the checklist addresses environmental information. The environmental review staff should review the supplement to the environmental report prepared by the applicant in accordance with the guidelines in NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants," Supplement 1, "Operating License Renewal" (Ref. 2). The

[1] NRC Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses" (Ref. 1), provides guidance on the format and content of a renewal application.

reviewer should check the "Yes" column if it is determined that the renewal application contains environmental information consistent with the requirements of 10 CFR Part 51.

The application should address each item in the checklist in order to be considered reasonably complete and sufficient. If the reviewer determines that an item in the checklist is not applicable, the reviewer should include a brief statement that the item is not applicable and provide the basis for the statement.

If information in the application for a checklist item is either not provided or not reasonably complete and no justification is provided, the reviewer should check the "No" column for that checklist item. By checking the "No" column for any checklist item, except Item VI as discussed in Subsection 1.1.3.2, the reviewer indicates that the application is not acceptable for docketing as a sufficient renewal application, unless the applicant modifies the application to provide the missing or incomplete information.

If the reviewer determines that the application is not acceptable for docketing as a sufficient application, the letter to the applicant should clearly state that (1) the application is not sufficient and is not acceptable for docketing, and (2) the current license will expire at its expiration date. The letter should also include a description of the deficiencies found in the application and offer an opportunity for the applicant to modify its application to provide the missing or incomplete information. The reviewer should review the modified application, if submitted, to determine whether it is acceptable for docketing as a sufficient application.

If the reviewer is able to answer "Yes" to the applicable items in the checklist, the application is acceptable for docketing as a sufficient renewal application. The applicant should be notified by letter that the application is accepted for docketing. Normally, the letter should be issued within 30 days of receipt of a renewal application. A notice of acceptance for docketing of the application and notice of opportunity for a hearing regarding renewal of the license will be published in the *Federal Register*.

If the application is acceptable for docketing as a sufficient application, the staff should begin their technical review. For license renewal applications, the NRC intends to maintain the docket number of the current operating license for administrative convenience.

1.1.3.2 Timeliness of Application

Upon receipt of a tendered application for license renewal, the reviewer performs a docketing and sufficiency review, as discussed in Subsection 1.1.3.1.

If the sufficient application is submitted at least 5 years before the expiration of the current operating license, the reviewer checks the "Yes" column for Item VI in the checklist. If an applicant has to modify its application, as discussed in Subsection 1.1.3.1, before the staff can find the application acceptable for docketing as a sufficient application, the modified application should be submitted at least 5 years before the expiration of the current operating license.

If the reviewer checks the "No" column in Item VI in the checklist, indicating that a sufficient renewal application has not been submitted at least 5 years before the expiration of the current operating license, the letter to the applicant should clearly state that (1) the application is not timely, (2) the provisions in 10 CFR 2.109(b) have not been satisfied, and (3) the current license will expire on the expiration date. However, if the application is otherwise determined to be acceptable for docketing, the technical review can begin.

1.1.4 Evaluation Findings

The reviewer determines whether sufficient and adequate information has been provided to satisfy the provisions outlined here. Depending on the results of this review, one of the following conclusions is included in the staff's letter to the applicant:

- The NRC staff has determined that the applicant has submitted sufficient information that is acceptable for docketing, in accordance with 10 CFR 54.19, 54.21, 54.22, 54.23, and 51.53(c). However, the staff's determination does not preclude the request for additional information as the review proceeds.

- The application is *not acceptable* for docketing as a timely and sufficient renewal application.

1.1.5 Implementation

Except in cases in which the applicant proposes an acceptable alternative method for complying with specified portions of NRC regulations, the method described herein will be used by NRC staff in their evaluation of conformance with NRC regulations.

1.1.6 References

1. NRC Regulatory Guide 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses," U.S. Nuclear Regulatory Commission, July 2001.

2. NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants," Supplement 1, "Operating License Renewal," U.S. Nuclear Regulatory Commission, October 1999.

Table 1.1-1. Acceptance Review Checklist for Docketing of Timely and Sufficient Renewal Application

				Yes	No
I.	**General Information**				
	1.	Application identifies specific unit(s) applying for license renewal		☐	☐
	2.	Filing of renewal application 10 CFR 54.17(a) is in accordance with:			
		A.	10 CFR Part 2, Subpart A; 10 CFR 2.101	☐	☐
		B.	10 CFR 50.4		
			a. Application is addressed to the Document Control Desk as specified in 10 CFR 50.4(a)	☐	☐
			b. Signed original application and 13 copies are provided to the Document Control Desk. One copy is provided to the appropriate Regional office [10 CFR 50.4(b)(3)]	☐	☐
			c. Form of the application meets the requirements of 10 CFR 50.4(c)	☐	☐
		C.	10 CFR 50.30		
			a. Application is filed in accordance with 10 CFR 50.4 [10 CFR 50.30(a)(1)]	☐	☐
			b. Application is submitted under oath or affirmation [10 CFR 50.30(b)]	☐	☐
	3.	Applicant is eligible to apply for a license and is not a foreign-owned or foreign-controlled entity [10 CFR 54.17(b)]		☐	☐
	4.	Application is not submitted earlier than 20 years before expiration of current license [10 CFR 54.17(c)]		☐	☐
	5.	Application states whether it contains applications for other kinds of licenses [10 CFR 54.17(d)]		☐	☐
	6.	Information incorporated by reference in the application is contained in other documents previously filed with the Commission, and the references are clear and specific [10 CFR 54.17(e)]		☐	☐
	7.	Restricted data or other defense information, if any, is separated from unclassified information in accordance with 10 CFR 50.33(j) [10 CFR 54.17(f)]		☐	☐
	8.	If the application contains restricted data, written agreement on the control of accessibility to such information is provided [10 CFR 54.17(g)]		☐	☐

Table 1.1-1. Acceptance Review Checklist for Docketing of Timely and Sufficient Renewal Application (continued)

		Yes	No
9.	Information specified in 10 CFR 50.33(a) through (e), (h), and (i) is provided or referenced [10 CFR 54.19(a)]:		
	A. Name of applicant	☐	☐
	B. Address of applicant	☐	☐
	C. Business description	☐	☐
	D. Citizenship and ownership details	☐	☐
	E. License information	☐	☐
	F. Construction or alteration dates	☐	☐
	G. Regulatory agencies and local publications	☐	☐
10.	Conforming changes, as needed, to the standard indemnity agreement have been submitted (10 CFR 140.92, Appendix B) to account for the proposed change in the expiration date [10 CFR 54.19(b)]	☐	☐

II. Technical Information

		Yes	No
1.	An integrated plant assessment [10 CFR 54.21(a)] is provided, and consists of:		
	A. For those SSC within the scope of license renewal [10 CFR 54.4], identification and listing of those structures and components that are subject to an aging management review (AMR) in accordance with 10 CFR 54.21(a)(1)(i) and (ii)		
	a. Description of the boundary of the system or structure considered (if applicant initially scoped at the system or structure level). Within this boundary, identification of structures and components subject to an AMR. For commodity groups, description of basis for the grouping	☐	☐
	b. Lists of structures and components subject to an AMR	☐	☐
	B. Description and justification of methods used to identify structures and components subject to an AMR [10 CFR 54.21(a)(2)]	☐	☐

Table 1.1-1. Acceptance Review Checklist for Docketing of Timely and Sufficient Renewal Application (continued)

Item	Yes	No
C. Demonstration that the effects of aging will be adequately managed for each structure and component identified, so that their intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation [10 CFR 54.21(a)(3)]		
a. Description of the intended function(s) of the structures and components	☐	☐
b. Identification of applicable aging effects based on materials, environment, operating experience, etc.	☐	☐
c. Identification and description of aging management programs	☐	☐
d. Demonstration of aging management provided	☐	☐
2. An evaluation of TLAAs is provided, and consists of:		
A. Listing of plant-specific TLAAs in accordance with the six criteria specified in 10 CFR 54.3 [10 CFR 54.21(c)(1)]	☐	☐
B. An evaluation of each identified TLAA using one of the three approaches specified in 10 CFR 54.21(c)(1)(i) to (iii)	☐	☐
3. All plant-specific exemptions granted pursuant to 10 CFR 50.12 and in effect that are based on a TLAA are listed, and evaluations justifying the continuation of these exemptions for the period of extended operation are provided [10 CFR 54.21(c)(2)]	☐	☐
A. Listing of plant-specific exemptions that are based on TLAAs as defined in 10 CFR 54.3 [10 CFR 54.21(c)(2)]	☐	☐
B. An evaluation of each identified exemption justifying the continuation of these exemptions for the period of extended operation [10 CFR 54.21(c)(2)]	☐	☐
III. An FSAR supplement [10 CFR 54.21(d)] is provided and contains the following information:		
1. Summary description of the aging management programs and activities for managing the effects of aging	☐	☐
2. Summary description of the evaluation of TLAAs	☐	☐

Table 1.1-1. Acceptance Review Checklist for Docketing of Timely and Sufficient Renewal Application (continued)

		Yes	No
IV.	**Technical Specification Changes**		
	Any technical specification changes necessary to manage the aging effects during the period of extended operation and their justifications are included in the application [10 CFR 54.22]	☐	☐
V.	**Environmental Information**		
	Application includes a supplement to the environmental report that is in accordance with the requirements of Subpart A of 10 CFR Part 51 [10 CFR 54.23]	☐	☐
VI.	**Timeliness Provision**		
	The application is sufficient and submitted at least 5 years before expiration of current license [10 CFR 2.109(b)]. If not, application can be accepted for docketing, but the timely renewal provision in 10 CFR 2.109(b) does not apply	☐	☐

CHAPTER 2

SCOPING AND SCREENING METHODOLOGY FOR IDENTIFYING STRUCTURES AND COMPONENTS SUBJECT TO AGING MANAGEMENT REVIEW AND IMPLEMENTATION RESULTS

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2.1 SCOPING AND SCREENING METHODOLOGY

Review Responsibilities

Primary - Branch responsible for quality assurance
Secondary - Branches responsible for systems, as appropriate

2.1.1 Areas of Review

This section addresses the scoping and screening methodology for license renewal. As required by 10 CFR 54.21(a)(2), the applicant, in its integrated plant assessment (IPA), is to describe and justify methods used to identify systems, structures, and components (SSCs) subject to an aging management review (AMR). The SSCs subject to AMR are those that perform an intended function, as described on 10 CFR 54.4 and meet two criteria:

1. They perform such functions without moving parts or without a change in configuration or properties, as set forth in 10 CFR 54.21(a)(1)(i), (denoted as "passive" components and structures in this SRP), and

2. They are not subject to replacement based on a qualified life or specified time period, as set forth in 10 CFR 54.21 (a)(1)(ii), (denoted as "long-lived" structures and components).

The identification of the SSCs within the scope of license renewal is called "scoping." For those SSCs within the scope of license renewal, the identification of "passive," "long-lived" structures and components that are subject to an AMR is called "screening."

To verify that the applicant has properly implemented its methodology, the staff reviews the implementation results separately, following the guidance in Sections 2.2 through 2.5.

The following areas relating to the applicant's scoping and screening methodology are reviewed.

2.1.1.1 Scoping

The methodology used by the applicant to implement the scoping requirements of 10 CFR 54.4, "Scope," is reviewed.

2.1.1.2 Screening

The methodology used by the applicant to implement the "screening" requirements of 10 CFR 54.21(a)(1) is reviewed.

2.1.2 Acceptance Criteria

The acceptance criteria for the areas of review are based on the following regulations:

- 10 CFR 54.4(a) as it relates to the identification of plant SSCs within the scope of the rule;
- 10 CFR 54.4(b) as it relates to the identification of the intended functions of plant SSCs determined to be within the scope of the rule; and
- 10 CFR 54.21(a)(1) and (a)(2) as they relate to the methods utilized by the applicant to identify plant structures and components subject to an AMR.

Specific criteria necessary to determine whether the applicant has met the relevant requirements of 10 CFR 54.4(a), 54.4(b), 54.21(a)(1), and 54.21(a)(2) are as follows.

2.1.2.1 Scoping

The scoping methodology used by the applicant should be consistent with the process described in Section 3.0, "Identify the SSCs Within the Scope of License Renewal and Their Intended Functions," of NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Rev. 3 (Ref. 1), or the justification provided by the applicant for any exceptions should be found to be acceptable by the reviewer.

2.1.2.2 Screening

The screening methodology used by the applicant should be consistent with the process described in Section 4.1, "Identification of Structures and Components Subject to an Aging Management Review and Intended Functions," of NEI 95-10, Rev. 3 (Ref. 1).

2.1.3 Review Procedures

Preparation for the review of the scoping and screening methodology employed by the applicant should include the following:

- Review of the NRC's safety evaluation report (SER) that was issued upon receipt of the operating license for the facility. This review is conducted for the purpose of familiarization with the principal design criteria for the facility and its CLB, as defined in 10 CFR 54.3(a).
- Review of Chapters 1 through 12 of the Updated Final Safety Analysis Report (UFSAR) and the facility's technical specifications for the purposes of familiarization with the facility design and the nomenclature that is applied to SSCs within the facility (including the bases for such nomenclature). During this review, the SSCs should be identified that are relied upon to remain functional during and after design basis events (DBEs), as defined in 10 CFR 50.49(b)(1)(ii), for which the facility was designed, to ensure that the functions described in 10 CFR 54.4(a)(1) are successfully accomplished. This review should also yield information regarding seismic Category I SSCs as defined in Regulatory Guide 1.29, "Seismic Design Classification" (Ref. 2). For a newer plant, this

information is typically contained in Section 3.2.1, "Seismic Classification," of the UFSAR consistent with the Standard Review Plan (NUREG-0800) (Ref. 3).

- Review of Chapter 15 (or equivalent) of the UFSAR to identify the anticipated operational occurrences and postulated accidents that are explicitly evaluated in the accident analyses for the facility. During this review, the SSCs that are relied upon to remain functional during and following design basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the functions described in 10 CFR 54.4(a)(1) should be identified.

- The set of design basis events as defined in the rule is not limited to Chapter 15 (or equivalent) of the UFSAR. Examples of design basis events that may not be described in this chapter include external events, such as floods, storms, earthquakes, tornadoes, or hurricanes, and internal events, such as a high-energy-line break. Information regarding design basis events as defined in 10 CFR 50.49(b)(1) may be found in any chapter of the facility UFSAR, the Commission's regulations, NRC orders, exemptions, or license conditions within the CLB. These sources should also be reviewed to identify systems, structures, and components that are relied upon to remain functional during and following design basis events (as defined in 10 CFR 50.49(b)(1)) to ensure the functions described in 10 CFR 54.4(a)(1).

- Review of the facility's Probabilistic Risk Analysis (PRA) Summary Report that was prepared by the licensee in response to Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f)," dated November 23, 1988 (Ref. 4). This review should yield additional information regarding the impact of the Individual Plant Examination (IPE) on the CLB for the facility. While the LR Rule is "deterministic," the NRC in the statement of considerations (SOC) accompanying Rule also states: "In license renewal, probabilistic methods may be most useful, on a plant-specific basis, in helping to assess the relative importance of structures and components that are subject to an aging management review by helping to draw attention to specific vulnerabilities (e.g., results of an IPE or IPEEE)" (60 FR 22468). For example, the reviewer should focus IPE information pertaining to plant changes or modifications that are initiated by the licensee in accordance with the requirements of 10 CFR 50.59 or 10 CFR 50.90.

- Review of the results of facility's Individual Plant Examination of External Events (IPEEE) study conducted as a follow-up to the IPE performed as a result of GL 88-20 to identify any changes or modifications made to the facility in accordance with the requirements of 10 CFR 50.59 or 10 CFR 50.90.

- Review of the applicant's docketed correspondence related to the following regulations:

  (a) 10 CFR 50.48, "Fire Protection,"

  (b) 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants,"

(c) 10 CFR 50.61, "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events,"

(d) 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram Events for Light-Water-Cooled Nuclear Power Plants," and

(e) 10 CFR 50.63, "Loss of All Alternating Current Power." [applicable to pressurized water reactor (PWR) plants].

Other staff members are reviewing the applicant's scoping and screening results separately following the guidance in Sections 2.2 through 2.5. The reviewer should keep these other staff members informed of findings that may affect their review of the applicant's scoping and screening results. The reviewer should coordinate this sharing of information through the license renewal project manager.

2.1.3.1 Scoping

Once the information delineated above has been gathered, the reviewer reviews the applicant's methodology to determine whether its depth and breadth are sufficiently comprehensive to identify the SSCs within the scope of license renewal, and the structures and components requiring an AMR. Because "[t]he CLB represents the evolving set of requirements and commitments for a specific plant that are modified as necessary over the life of a plant to ensure continuation of an adequate level of safety" (60 FR 22465, May 8, 1995), the regulations, orders, license conditions, exemptions, and technical specifications defining functional requirements for facility SSCs that make up an applicant's CLB should be considered as the initial input into the scoping process. 10 CFR 50.49 defines DBEs as conditions of normal operation, including anticipated operational occurrences, DBAs, external events, and natural phenomena for which the plant must be designed to ensure (1) the integrity of the reactor pressure boundary, (2) the capability to shut down the reactor and maintain it in safe shutdown condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to those referred to in 10 CFR 50.34(a)(1), 50.67(b)(2), or 100.11, as applicable. Therefore, to determine the safety-related SSCs that are within the scope of the rule under 10 CFR 54.4 (a)(1), the applicant must identify those SSCs that are relied upon to remain functional during and following these DBEs, consistent with the CLB of the facility. Most licensees have developed lists or database that identify systems, structures and components relied on for compliance with other regulations in a manner consistent with the CLB of their facilities. Consistent with the licensing process and regulatory criteria used to develop such lists or databases, licensees should build upon these information sources to satisfy 10 CFR Part 54 requirements.

With respect to technical specifications, the NRC states (60 FR 22467):

> *The Commission believes that there is sufficient experience with its policy on technical specifications to apply that policy generically in revising the license renewal rule consistent with the Commission's desire to credit existing regulatory programs. Therefore, the Commission concludes that the technical specification limiting conditions for operation scoping category is unwarranted and has deleted the requirement that identifies systems, structures, and components with operability requirements in technical specifications as being within the scope of the license renewal review.*

Therefore, the applicant need not consider its technical specifications and applicable limiting conditions of operation when scoping for license renewal. This is not to say that the events and functions addressed within the applicant's technical specifications can be excluded in determining the SSCs within the scope of license renewal solely on the basis of such an event's inclusion in the technical specifications. Rather, those SSCs governed by an applicant's technical specifications that are relied upon to remain functional during a DBE, as identified within the applicant's UFSAR, applicable NRC regulations, license conditions, NRC orders, and exemptions, need to be included within the scope of license renewal.

For licensee commitments, such as licensee responses to NRC bulletins, generic letters, or enforcement actions, and those documented in staff safety evaluations or license event reports, and which make up the remainder of an applicant's CLB, many of the associated SSCs need not be considered under license renewal. Generic communications, safety evaluations, and other similar documents found on the docket are not regulatory requirements, and commitments made by a licensee to address any associated safety concerns are not typically considered to be design requirements. However, any generic communication, safety evaluation, or licensee commitment that specifically identifies or describes a function associated with a system, structure, or component necessary to fulfill the requirement of a particular regulation, order, license condition, and/or exemption may need to be considered when scoping for license renewal. For example, NRC Bulletin 88-11, "Pressurizer Surge Line Thermal Stratification," states:

> *The licensing basis according to 10 CFR 50.55a for all PWRs requires that the licensee meet the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Sections III and XI and to reconcile the pipe stresses and fatigue evaluation when any significant differences are observed between measured data and the analytical results for the hypothesized conditions. Staff evaluation indicates that the thermal stratification phenomenon could occur in all PWR surge lines and may invalidate the analyses supporting the integrity of the surge line. The staff's concerns include unexpected bending and thermal striping (rapid oscillation of the thermal boundary interface along the piping inside surface) as they affect the overall integrity of the surge line for its design life (e.g., the increase of fatigue).*

Therefore, this bulletin specifically describes conditions that may affect compliance with the requirements associated with 10 CFR 50.55a and functions specifically related to this regulation that must be considered in the scoping process for license renewal.

An applicant may take an approach in scoping and screening that combines similar components from various systems. For example, containment isolation valves from various systems may be identified as a single system for purposes of license renewal.

Staff from branches responsible for systems may be requested to assist in reviewing the plant design basis and intended function(s), as necessary.

The reviewer should verify that the applicant's scoping methods document the actual information sources used (for example, those identified in Table 2.1-1).

Table 2.1-2 contains specific staff guidance on certain subjects of scoping.

2.1.3.1.1 Safety-Related

The applicant's methodology is reviewed to ensure that the safety-related SSCs are identified to satisfactorily accomplish any of the intended functions identified in 10 CFR 54.4(a)(1). The reviewer must ascertain how, and to what extent, the applicant incorporated the information in the CLB for the facility in its methodology. Specifically, the reviewer should review the application, as well as all other relevant sources of information outlined above, to identify the set of plant-specific conditions of normal operation, DBAs, external events, and natural phenomena for which the plant must be designed to ensure the following functions:

- The integrity of the reactor coolant pressure boundary;
- The capability to shut down the reactor and maintain it in a safe shutdown condition; or
- The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1), 50.67(b)(2), or 100.11, as applicable.

2.1.3.1.2 Nonsafety-Related

The applicant's methodology is reviewed to ensure that nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1) are identified as being within the scope of license renewal.

The scoping criterion under 10 CFR 54.4(a)(2), in general, is intended to identify those nonsafety-related SSCs that support safety-related functions. More specifically, this scoping criterion requires an applicant to identify all nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of the applicable functions of the SSCs identified under 10 CFR 54.4(a)(1). Section III.c(iii) of the SOC (60 FR 22467) clarifies the NRC's intent for this requirement in the following statement:

> *The inclusion of nonsafety-related systems, structures, and components whose failure could prevent other systems, structures, and components from accomplishing a safety function is intended to provide protection against safety function failure in cases where the safety-related structure or component is not itself impaired by age-related degradation but is vulnerable to failure from the failure of another structure or component that may be so impaired.*

In addition, Section III.c(iii) of the SOC provides the following guidance to assist an applicant in determining the extent to which failures must be considered when applying this scoping criterion:

> *Consideration of hypothetical failures that could result from system interdependencies that are not part of the current licensing bases and that have not been previously experienced is not required. [...] However, for some license renewal applicants, the Commission cannot exclude the possibility that hypothetical failures that <u>are part of the CLB</u> may require consideration of second-, third-, or fourth-level support systems.*

Therefore, to satisfy the scoping criterion under 10 CFR 54.4(a)(2), the applicant must identify those nonsafety-related SSCs (including certain second-, third-, or fourth-level support systems) whose failures are considered in the CLB and could prevent the satisfactory accomplishment of the safety-related function identified under 10 CFR 54.4(a)(1). In order to identify such systems, the applicant should consider those failures identified in (1) the documentation that makes up its CLB, (2) plant-specific operating experience, and (3) industry-wide operating experience that is specifically applicable to its facility. The applicant need not consider hypothetical failures that are not part of the CLB, have not been previously experienced, or are not applicable to its facility.

For example, the safety classification of a pipe at certain locations, such as valves, may change throughout its length in the plant. In these instances, the applicant should identify the safety-related portion of the pipe as being within the scope of license renewal under 10 CFR 54.4(a)(1). However, the entire pipe run, including associated piping anchors, may have been analyzed as part of the CLB to establish that it could withstand DBE loads. If this is the case, a failure in the pipe run or in the associated piping anchors could render the safety-related portion of the piping unable to perform its intended function under CLB design conditions. Therefore, the reviewer must verify that the applicant's methodology would include (1) the remaining nonsafety-related piping up to its anchors and (2) the associated piping anchors as being within the scope of license renewal under 10 CFR 54.4(a)(2).

It is important to note that the scoping criterion under 10 CFR 54.4(a)(2) specifically applies to those functions "identified in paragraphs (a)(1)(i), (ii), and (iii)" of 10 CFR 54.4 and does not apply to functions identified in 10 CFR 54.4(a)(3), as discussed below.

2.1.3.1.3 "Regulated Events"

The applicant's methodology is reviewed to ensure that SSCs relied on in safety analyses or plant evaluations to perform functions that demonstrate compliance with the requirements of the fire protection, environmental qualification, pressurized thermal shock (PTS), anticipated transients without scram (ATWS), and station blackout (SBO) regulations are identified. The reviewer should review the applicant's docketed correspondence associated with compliance of the facility with these regulations.

The scoping criteria in 10 CFR 54.4(a)(3) require an applicant to consider *"[a]ll SSC relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the [specified] Commission regulations[.]"* In addition, Section III.c(iii) (60 FR 22467) of the SOC states that the NRC intended to limit the potential for unnecessary expansion of the review for SSCs that meet the scoping criteria under 10 CFR 54.4(a)(3) and provides additional guidance that qualifies what is meant by *"those SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with the Commission regulations"* in the following statement:

> *[T]he Commission intends that this [referring to 10 CFR 54.4(a)(3)] scoping category include all SSC whose function is relied upon to demonstrate compliance with these Commission[ ] regulations. An applicant for license renewal should rely on the plant's current licensing bases, actual plant-specific experience, industry-wide operating experience, as appropriate, and existing engineering evaluations to determine those SSC that are the initial focus of license renewal review.*

Therefore, all SSCs that are relied upon in the plant's CLB (as defined in 10 CFR 54.3), plant-specific experience, industry-wide experience (as appropriate), and safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations identified under 10 CFR 54.4(a)(3), are required to be included within the scope of the rule. For example, if a nonsafety-related diesel generator is required for safe shutdown under the fire protection plan, the diesel generator and all SSCs specifically required for that generator to comply with and NRC regulations shall be included within the scope of license renewal under 10 CFR 54.4(a)(3). Such SSCs may include, but should not be limited to, the cooling water system or systems required for operability, the diesel support pedestal, and any applicable power supply cable specifically required for safe shutdown in the event of a fire.

In addition, the last sentence of the second paragraph in Section III.c(iii) of the SOC provides the following guidance for limiting the application of the scoping criteria under 10 CFR 54.4(a)(3) as it applies to the use of hypothetical failures:

> *Consideration of hypothetical failures that could result from system interdependencies, that are not part of the current licensing bases and that have not been previously experienced is not required. (60 FR 22467)*

The SOC does not provide any additional guidance relating to the use of hypothetical failures or the need to consider second-, third-, or fourth-level support systems for scoping under 10 CFR 54.4(a)(3). Therefore, in the absence of any guidance, an applicant need not consider hypothetical failures or second-, third-, or fourth-level support systems in determining the SSCs within the scope of the rule under 10 CFR 54.4(a)(3). For example, if a nonsafety-related diesel generator is relied upon only to remain functional to demonstrate compliance with the NRC SBO regulations, the applicant need not consider the following SSCs: (1) an alternate/backup cooling water system, (2) non-seismically-qualified building walls, or (3) an overhead segment of non-seismically-qualified piping (in a Seismic II/I configuration). This guidance is not intended to exclude any support system (whether identified by an applicant's CLB, or as indicated from actual plant-specific experience, industrywide experience [as applicable], safety analyses, or plant evaluations) that is specifically required for compliance with, the applicable NRC regulation. For example, if a nonsafety-related diesel generator (required to demonstrate compliance with an applicable NRC regulation) specifically requires a second cooling system to cool the diesel generator jacket water cooling system for the generator to be operable, then both cooling systems must be included within the scope of the rule under 10 CFR 54.4(a)(3).

The applicant is required to identify the SSCs whose functions are relied on to demonstrate compliance with the regulations identified in 10 CFR 54.4(a)(3) (that is, whose functions were credited in the analysis or evaluation). Mere mention of an SSC in the analysis or evaluation does not necessarily constitute support of an intended function as required by the regulation.

For environmental qualification, the reviewer verifies that the applicant has indicated that the environmental qualification equipment is that equipment already identified by the licensee under 10 CFR 50.49(b), that is, equipment relied upon in safety analyses or plant evaluations to demonstrate compliance with NRC regulations for environmental qualification (10 CFR 50.49).

The PTS regulation is applicable only to PWRs. If the renewal application is for a PWR and the applicant relies on a Regulatory Guide 1.154 (Ref. 5) analysis to satisfy 10 CFR 50.61, as described in the plant's CLB, the reviewer verifies that the applicant's methodology would include SSCs relied on in that analysis that are within the scope of license renewal.

For SBO, the reviewer verifies that the applicant's methodology would include those SSCs relied upon during the "coping duration" phase of an SBO event (Ref. 6).

2.1.3.2 Screening

Once the SSCs within the scope of license renewal have been identified, the next step is determining which structures and components are subject to an AMR (i.e., "screening") (Ref. 1).

2.1.3.2.1 "Passive"

The reviewer reviews the applicant's methodology to ensure that "passive" structures and components are identified as those that perform their intended functions without moving parts or a change in configuration or properties in accordance with 10 CFR 54.21(a)(1)(i). The description of "passive" may also be interpreted to include structures and components that do not display "a change in state." 10 CFR 54.21(a)(1)(i) provides specific examples of structures and components that do or do not meet the criterion. The reviewer verifies that the applicant's screening methodology includes consideration of the intended functions of structures and components consistent with plant CLB, as typified in Table 2.1-4 (Ref. 1).

The license renewal rule focuses on "passive" structures and components because structures and components that have passive functions generally do not have performance and condition characteristics that are as readily observable as those that perform active functions. "Passive" structures and components, for the purpose of the license renewal rule, are those that perform an intended function, as described in 10 FR 54.4, without moving parts or without a change in configuration or properties (Ref. 2). The description of "passive" may also be interpreted to include structures and components that do not display "a change of state."

Table 2.1-5 provides a list of typical structures and components identifying whether they meet 10 CFR 54.21(a)(1)(i).

10 CFR 54.21(a)(1)(i) explicitly excludes instrumentation, such as pressure transmitters, pressure indicators, and water level indicators, from an AMR. The applicant does not have to identify pressure-retaining boundaries of this instrumentation because 10 CFR 54.21(a)(1)(i) excludes this instrumentation without exception, unlike pumps and valves. Further, instrumentation is sensitive equipment and degradation of its pressure retaining boundary would be readily determinable by surveillance and testing (Ref.6). If an applicant determines that certain structures and components listed in Table 2.1-5 as meeting 10 CFR 54.21(a)(1)(i) do not meet that requirement for its plant, the reviewer reviews the applicant's basis for that determination.

2.1.3.2.2 "Long-Lived"

The applicant's methodology is reviewed to ensure that "long-lived" structures and components are identified as those that are not subject to periodic replacement based on a qualified life or specified time period. Passive structures and components that are not replaced on the basis of a qualified life or specified time period require an AMR.

Replacement programs may be based on vendor recommendations, plant experience, or any means that establishes a specific replacement frequency under a controlled program.
Section f(i)(b) of the SOC provides the following guidance for identifying "long-lived" structures and components:

> *In sum, a structure or component that is not replaced either (i) on a specified interval based upon the qualified life of the structure or component or*

(ii) periodically in accordance with a specified time period is deemed by §54.21(a)(1)(ii) of this rule to be "long-lived," and therefore subject to the §54.21(a)(3)[AMR][22478].

A qualified life does not necessarily have to be based on calendar time. A qualified life based on run time or cycles are examples of qualified life references that are not based on calendar time (Ref. 3).

Structures and components that are replaced on the basis of performance or condition are not generically excluded from an AMR. Rather, performance or condition monitoring may be evaluated later in the IPA as programs to ensure functionality during the period of extended operation. On this topic, Section f(i)(b) of the SOC provides the following guidance:

> *It is important to note, however, that the Commission has decided not to generically exclude passive structures and components that are replaced based on performance or condition from an [AMR]. Absent the specific nature of the performance or condition replacement criteria and the fact that the Commission has determined that the components with "passive" functions are not as readily monitorable as components with active functions, such generic exclusion is not appropriate. However, the Commission does not intend to preclude a license renewal applicant from providing site-specific justification in a license renewal application that a replacement program on the basis of performance or condition for a passive structure or component provides reasonable assurance that the intended function of the passive structure or component will be maintained in the period of extended operation. [60 FR 22478]*

2.1.4 Evaluation Findings

When the review of the information in the license renewal application is complete, and the reviewer has determined that it is satisfactory and in accordance with the acceptance criteria in Subsection 2.1.2, a statement of the following type should be included in the staff's safety evaluation report:

> The staff concludes that there is reasonable assurance that the applicant's methodology for identifying the systems, structures, and components within the scope of license renewal and the structures and components requiring an aging management review is consistent with the requirements of 10 CFR 54.4 and 10 CFR 54.21(a)(1).

2.1.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of NRC regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

2.1.6 References

1. NEI 95-10, Rev. 3 "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Nuclear Energy Institute, March 2001.

2. Regulatory Guide 1.29, Rev. 2, "Seismic Design Classification," September 1978.

3. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

4. Generic Letter (GL) 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities- 10 CFR 50.54(f)," dated November 23, 1988.

5. Regulatory Guide 1.154, "Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors," January 1987.

6. Letter from Dennis M. Crutchfield of NRC to Charles H. Cruse of Baltimore Gas and Electric Company, dated April 4, 1996.

7. NUREG-1723, "Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Stations, Units 1, 2, and 3," March 2000.

8. Letter to Douglas J. Walters, Nuclear Energy Institute, from Christopher I. Grimes, NRC, dated August 5, 1999.

9. Summary of December 8, 1999, Meeting with the Nuclear Energy Institute (NEI) on License Renewal Issue (LR) 98-12, "Consumables," Project No. 690, January 21, 2000.

10. Letter to William R. McCollum, Jr., Duke Energy Corporation, from Christopher I. Grimes, NRC, dated October 8,1999.

11. NEI 95-10, Rev. 0, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54 - The License Renewal Rule," Nuclear Energy Institute, March 1, 1996.

Table 2.1-1. Sample Listing of Potential Information Sources

Verified databases (databases that are subject to administrative controls to assure and maintain the integrity of the stored data or information)
Master equipment lists (including NSSS vendor listings)
Q-lists
Updated Final Safety Analysis Reports
Piping and instrument diagrams
NRC Orders, Exemptions, or License Conditions for the facility
Design-basis documents
General arrangement or structural outline drawings
Probabilistic risk assessment summary report
Maintenance rule compliance documentation
Design-basis event evaluations (including plant-specific 10 CFR 50.59 evaluation procedures)
Emergency operating procedures
Docketed correspondence
System interaction commitments
Technical specifications
Environmental qualification program documents
Regulatory compliance reports (including Safety Evaluation Reports)
Severe Accident Management Guidelines

Table 2.1-2. Specific Staff Guidance on Scoping

Issue	Guidance
Commodity groups	The applicant may also group like structures and components into commodity groups. Examples of commodity groups are pipe supports and cable trays. The basis for grouping structures and components can be determined by such characteristics as similar function, similar design, similar materials of construction, similar aging management practices, or similar environments. If the applicant uses commodity groups, the reviewer verifies that the applicant has described the basis for the groups.
Complex assemblies	Some structures and components, when combined, are considered a complex assembly (for example, diesel generator starting air skids or heating, ventilating, and air conditioning refrigerant units). For purposes of performing an AMR, it is important to clearly establish the boundaries of review. An applicant should establish the boundaries for such assemblies by identifying each structure and component that makes up the complex assembly and determining whether or not each structure and component is subject to an AMR (Ref. 1). NEI 95-10, Revision 0, Appendix C, Example 5 (Ref. 11), illustrates how the evaluation boundary for a control room chiller complex assembly might be determined. The control room chillers were purchased as skid mounted equipment. These chillers are part of the control room chilled water system. There are two (2) control room chillers. Each is a 100% capacity refrigeration unit. The functions of the control room chillers are: to provide a reliable source of chilled water at a maximum temperature of 44°F, to provide a pressure boundary for the control room chilled water system, to provide a pressure boundary for the service water system, and to provide a pressure boundary for the refrigerant. All of these functions are considered intended functions. Typically, control room chillers are considered as one functional unit; however, for purposes of evaluating the effects of aging, it is necessary to consider the individual components. Therefore, the boundary of each control room chiller is established as follows: 1. At the inlet and outlet flanges of the service water system connections on the control room chiller condenser. Connected piping is part of the service water system. 2. At the inlet and outlet flanges of the control room chilled water system piping connections on the control room chiller evaporator. Connected piping is part of the control room chilled water system. 3. For electrical power supplies, the boundary is the output terminals on the circuit breakers supplying power to the skid. This includes the cables from the circuit breaker to the skid and applies for 480 VAC and 120 VAC. 4. The interface for instrument air supplies is at the instrument air tubing connection to the pressure control regulators, temperature controllers and transmitters, and solenoid valves located on the skid. The tubing from the instrument air header to the device on the skid is part of the instrument air system. 5. The interface with the annunciator system is at the external connection of the contacts of the device on the skid (limit switch, pressure switch, level switch, etc.) that indicates the alarm condition. The cables are part of the annunciator system. Based on the boundary established, the following components would be subject to an aging management review: condenser, evaporator, economizer, chiller refrigerant piping, refrigerant expansion orifice, foundations and bolting, electrical cabinets, cables, conduit, trays and supports, valves

Table 2.1-2. Specific Staff Guidance on Scoping (continued)

Issue	Guidance
Hypothetical failures	For 10 CFR 54.4(a)(2), an applicant should consider those failures identified in (1) the documentation that makes up its CLB, (2) plant-specific operating experience, and (3) industry-wide operating experience that is specifically applicable to its facility. The applicant need not consider hypothetical failures that are not part of CLB and that have not been previously experienced. For example, an applicant should consider including (1) the portion of a fire protection system identified in the UFSAR that supplies water to the refueling floor that is relied upon in a DBA analysis as an alternate source of cooling water that can be used to mitigate the consequences from the loss of spent fuel pool cooling, (2) a nonsafety-related, non-seismically-qualified building whose intended function as described in the plant's CLB is to protect a tank that is relied upon as an alternate source of cooling water needed to mitigate the consequences of a DBE, and (3) a segment of nonsafety-related piping identified as a Seismic II/I component in the applicant's CLB (Ref. 8).
Cascading	For 10 CFR 54.4(a)(3), an applicant need not consider hypothetical failures or second-, third, or fourth-level support systems. For example, if a nonsafety-related diesel generator is only relied upon to remain functional to demonstrate compliance with the NRC's SBO regulations, an applicant may not need to consider (1) an alternate/backup cooling water system, (2) the diesel generator non-seismically-qualified building walls, or (3) an overhead segment of non-seismically-qualified piping (in a Seismic II/I configuration). An applicant may not exclude any support system (identified by its CLB, actual plant-specific experience, industry-wide experience, as applicable, or existing engineering evaluations) that is specifically required for compliance with, or operation within, applicable NRC regulation. For example, if the analysis of a nonsafety-related diesel generator (required to demonstrate compliance with an applicable NRC regulation) specifically requires a second cooling system to cool the diesel generator jacket water cooling system for the diesel to be operable, then both cooling systems must be included within the scope of the rule (Ref. 8).

Table 2.1-3. Specific Staff Guidance on Screening

Issue	Guidance
Consumables	Consumables may be divided into the following four categories for the purpose of license renewal: (a) packing, gaskets, component seals, and O-rings; (b) structural sealants; (c) oil, grease, and component filters; and (d) system filters, fire extinguishers, fire hoses, and air packs. The consumables in both categories (a) and (b) are considered as subcomponents and are not explicitly called out in the scoping and screening procedures. Rather, they are implicitly included at the component level (e.g., if a valve is identified as being in scope, a seal in that valve would also be in scope as a subcomponent of that valve). For category (a), the applicant would be able to exclude these subcomponents using a clear basis, such as the example of ASME Section III not being relied on for pressure boundary. For category (b), these subcomponents may perform functions without moving parts or a change in configuration, and they are not typically replaced. It is expected that the applicant's structural AMP will address these items with respect to an AMR program on a plant-specific basis. The consumables in category (c) are short-lived and periodically replaced, and can be excluded from an AMR on that basis. Likewise, the consumables that fall within category (d) are typically replaced based on performance or condition monitoring that identifies whether these components are at the end of their qualified lives and may be excluded, on a plant-specific basis, from AMR under 10 CFR 54.21(a)(1)(ii). The applicant should identify the standards that are relied on for the replacement as part of the methodology description (for example, NFPA standards for fire protection equipment) (Ref. 9).
Heat exchanger intended functions	Both the pressure boundary and heat transfer functions for heat exchangers should be considered because heat transfer may be a primary safety function of these components. There may be a unique aging effect associated with different materials in the heat exchanger parts that are associated with the heat transfer function and not the pressure boundary function. The staff would expect that the programs that effectively manage aging effects of the pressure boundary function can, in conjunction with the procedures for monitoring heat exchanger performance, effectively manage aging effects applicable to the heat transfer function (Ref. 10).
Multiple functions	Structures and components may have multiple functions. The intended functions as delineated in 10 CFR 54.4(b) are to be reviewed for license renewal. For example, a flow orifice that is credited in a plant's accident analysis to limit flow would have two intended functions. One intended function is pressure boundary. The other intended function is to limit flow. The reviewer verifies that the applicant has considered multiple functions in identifying structure and component intended functions.

Table 2.1-4. Typical "Passive" Structure and Component Intended Functions

Structures
Provide rated fire barrier to confine or retard a fire from spreading to or from adjacent areas of the plant
Provide shelter/protection to safety-related components
Provide structural and/or functional support to safety-related equipment
Provide flood protection barrier (internal and external flooding event)
Provide pressure boundary or essentially leaktight barrier to protect public health and safety in the event of any postulated design-basis events.
Provide spray shield or curbs for directing flow (e.g., safety injection flow to containment sump)
Provide shielding against radiation
Provide missile barrier (internally or externally generated)
Provide shielding against high-energy line breaks
Provide structural support to nonsafety-related components whose failure could prevent satisfactory accomplishment of any of the required safety-related functions
Provide pipe whip restraint
Provide path for release of filtered and unfiltered gaseous discharge
Provide source of cooling water for plant shutdown
Provide heat sink during station blackout or design-basis accidents
Components
Provide pressure-retaining boundary so that sufficient flow at adequate pressure is delivered
Provide filtration
Provide flow restriction (throttle)
Provide structural support to safety-related components
Provide electrical connections to specified sections of an electrical circuit to deliver voltage, current or signals
Provide heat transfer

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
1	Structures	Category I Structures	Yes
2	Structures	Primary Containment Structure	Yes
3	Structures	Intake Structures	Yes
4	Structures	Intake Canal	Yes
5	Structures	Other Non-Category I Structures Within the Scope of License Renewal	Yes
6	Structures	Equipment Supports and Foundations	Yes
7	Structures	Structural Bellows	Yes
8	Structures	Controlled Leakage Doors	Yes
9	Structures	Penetration Seals	Yes
10	Structures	Compressible Joints and Seals	Yes
11	Structures	Fuel Pool and Sump Liners	Yes
12	Structures	Concrete Curbs	Yes
13	Structures	Offgas Stack and Flue	Yes
14	Structures	Fire Barriers	Yes
15	Structures	Pipe Whip Restraints and Jet Impingement Shields	Yes
16	Structures	Electrical and Instrumentation and Control Penetration Assemblies	Yes
17	Structures	Instrumentation Racks, Frames, Panels, and Enclosures	Yes
18	Structures	Electrical Panels, Racks, Cabinets, and Other Enclosures	Yes
19	Structures	Cable Trays and Supports	Yes
20	Structures	Conduit	Yes
21	Structures	Tube Track	Yes
22	Structures	Reactor Vessel Internals	Yes
23	Structures	ASME Class 1 Hangers and Supports	Yes
24	Structures	Non-ASME Class 1 Hangers and Supports	Yes
25	Structures	Snubbers	No
26	Reactor Coolant Pressure Boundary Components (Note: the components of the RCPB are defined by each plant's CLB and site specific documentation	ASME Class 1 Piping	Yes
27	Reactor Coolant Pressure Boundary Components	Reactor Vessel	Yes
28	Reactor Coolant Pressure Boundary Components	Reactor Coolant Pumps	Yes (Casing)

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
29	Reactor Coolant Pressure Boundary Components	Control Rod Drives	No
30	Reactor Coolant Pressure Boundary Components	Control Rod Drive Housing	Yes
31	Reactor Coolant Pressure Boundary Components	Steam Generators	Yes
32	Reactor Coolant Pressure Boundary Components	Pressurizers	Yes
33	Non-Class I Piping Components	Underground Piping	Yes
34	Non-Class I Piping Components	Piping in Low Temperature Demineralized Water Service	Yes
35	Non-Class I Piping Components	Piping in High Temperature Single Phase Service	Yes
36	Non-Class I Piping Components	Piping in Multiple Phase Service	Yes
37	Non-Class I Piping Components	Service Water Piping	Yes
38	Non-Class I Piping Components	Low Temperature Gas Transport Piping	Yes
39	Non-Class I Piping Components	Stainless Steel Tubing	Yes
40	Non-Class I Piping Components	Instrument Tubing	Yes
41	Non-Class I Piping Components	Expansion Joints	Yes
42	Non-Class I Piping Components	Ductwork	Yes
43	Non-Class I Piping Components	Sprinklers Heads	Yes
44	Non-Class I Piping Components	Miscellaneous Appurtenances (Includes fittings, couplings, reducers, elbows, thermowells, flanges, fasteners, welded attachments, etc.)	Yes
45	Pumps	ECCS Pumps	Yes (Casing)
46	Pumps	Service Water and Fire Pumps	Yes (Casing)
47	Pumps	Lube Oil and Closed Cooling Water Pumps	Yes (Casing)
48	Pumps	Condensate Pumps	Yes (Casing)
49	Pumps	Borated Water Pumps	Yes (Casing)
50	Pumps	Emergency Service Water Pumps	Yes (Casing)
51	Pumps	Submersible Pumps	Yes (Casing)
52	Turbines	Turbine Pump Drives (excluding pumps)	Yes (Casing)
53	Turbines	Gas Turbines	Yes (Casing)

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
54	Turbines	Controls (Actuator and Overspeed Trip)	No
55	Engines	Fire Pump Diesel Engines	No
56	Emergency Diesel Generators	Emergency Diesel Generators	No
57	Heat Exchangers	Condensers	Yes
58	Heat Exchangers	HVAC Coolers	Yes
59	Heat Exchangers	Primary Water System Heat Exchangers	Yes
60	Heat Exchangers	Treated Water System Heat Exchangers	Yes
61	Heat Exchangers	Closed Cooling Water System Heat Exchangers	Yes
62	Heat Exchangers	Lubricating Oil System Heat Exchangers	Yes
63	Heat Exchangers	Raw Water System Heat Exchangers	Yes
64	Heat Exchangers	Containment Atmospheric System Heat Exchangers	Yes
65	Miscellaneous Process Components	Gland Seal Blower	No
66	Miscellaneous Process Components	Recombiners	The applicant shall identify the intended function and apply the IPA process to determine if the grouping is active or passive.
67	Miscellaneous Process Components	Flexible Connectors	Yes
68	Miscellaneous Process Components	Strainers	Yes
69	Miscellaneous Process Components	Rupture Disks	Yes
70	Miscellaneous Process Components	Steam Traps	Yes
71	Miscellaneous Process Components	Restricting Orifices	Yes
72	Miscellaneous Process Components	Air Compressor	No
73	Electrical and I&C	Alarm Unit (e.g., fire detection devices)	No
74	Electrical and I&C	Analyzers (e.g., gas analyzers, conductivity analyzers)	No
75	Electrical and I&C	Annunciators (e.g., lights, buzzers, alarms)	No
76	Electrical and I&C	Batteries	No

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
77	Electrical and I&C	Cables and Connections, Bus, electrical portions of Electrical and I&C Penetration Assemblies (e.g., electrical penetration assembly cables and connections, connectors, electrical splices, terminal blocks, power cables, control cables, instrument cables, insulated cables, communication cables, uninsulated ground conductors, transmission conductors, isolated-phase bus, nonsegregated-phase bus, segregated-phase bus, switchyard bus)	Yes
78	Electrical and I&C	Chargers, Converters, Inverters (e.g., converters-voltage/current, converters-voltage/pneumatic, battery chargers/inverters, motor-generator sets)	No
79	Electrical and I&C	Circuit Breakers (e.g., air circuit breakers, molded case circuit breakers, oil-filled circuit breakers)	No
80	Electrical and I&C	Communication Equipment (e.g., telephones, video or audio recording or playback equipment, intercoms, computer terminals, electronic messaging, radios, transmission line traps and other power-line carrier equipment)	No
81	Electrical and I&C	Electric Heaters	No Yes for a Pressure Boundary if applicable
82	Electrical and I&C	Heat Tracing	No
83	Electrical and I&C	Electrical Controls and Panel Internal Component Assemblies (may include internal devices such as, but not limited to, switches, breakers, indicating lights, etc.) (e.g., main control board, HVAC control board)	No
84	Electrical and I&C	Elements, RTDs, Sensors, Thermocouples, Transducers (e.g., conductivity elements, flow elements, temperature sensors, radiation sensors,watt transducers, thermocouples, RTDs, vibration probes, amp transducers, frequency transducers, power factor transducers, speed transducers, var. transducers, vibration transducers, voltage transducers)	No Yes for a Pressure Boundary if applicable
85	Electrical and I&C	Fuses	No

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
86	Electrical and I&C	Generators, Motors (e.g., emergency diesel generators, ECCS and emergency service water pump motors, small motors, motor-generator sets, steam turbine generators, combustion turbine generators, fan motors, pump motors, valve motors, air compressor motors)	No
87	Electrical and I&C	High-voltage Insulators (e.g., porcelain switchyard insulators, transmission line insulators)	Yes
88	Electrical and I&C	Surge Arresters (e.g., switchyard surge arresters, lightning arresters, surge suppressers, surge capacitors, protective capacitors)	No
89	Electrical and I&C	Indicators (e.g., differential pressure indicators, pressure indicators, flow indicators, level indicators, speed indicators, temperature indicators, analog indicators, digital indicators, LED bar graph indicators, LCD indicators)	No
90	Electrical and I&C	Isolators (e.g., transformer isolators, optical isolators, isolation relays, isolating transfer diodes)	No
91	Electrical and I&C	Light Bulbs (e.g., indicating lights, emergency lighting, incandescent light bulbs, fluorescent light bulbs)	No
92	Electrical and I&C	Loop Controllers (e.g., differential pressure indicating controllers, flow indicating controllers, temperature controllers, controllers, speed controllers, programmable logic controller, single loop digital controller, process controllers, manual loader, selector station, hand/auto station, auto/manual station)	No
93	Electrical and I&C	Meters (e.g., ammeters, volt meters, frequency meters, var meters, watt meters, power factor meters, watt-hour meters)	No
94	Electrical and I&C	Power Supplies	No
95	Electrical and I&C	Radiation Monitors (e.g., area radiation monitors, process radiation monitors)	No
96	Electrical and I&C	Recorders (e.g., chart recorders, digital recorders, events recorders)	No

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
97	Electrical and I&C	Regulators (e.g., voltage regulators)	No
98	Electrical and I&C	Relays (e.g., protective relays, control/logic relays, auxiliary relays)	No
99	Electrical and I&C	Signal Conditioners	No
100	Electrical and I&C	Solenoid Operators	No
101	Electrical and I&C	Solid-State Devices (e.g., transistors, circuit boards, computers)	No
102	Electrical and I&C	Switches (e.g., differential pressure indicating switches, differential pressure switches, pressure indicator switches, pressure switches, flow switches, conductivity switches, level indicating switches, temperature indicating switches, temperature switches, moisture switches, position switches, vibration switches, level switches, control switches, automatic transfer switches, manual transfer switches, manual disconnect switches, current switches, limit switches, knife switches)	No
103	Electrical and I&C	Switchgear, Load Centers, Motor Control Centers, Distribution Panel Internal Component Assemblies (may include internal devices such as, but not limited to, switches, breakers, indicating lights, etc.) (e.g., 4.16 kV switchgear, 480V load centers, 480V motor control centers, 250 VDC motor control centers, 6.9 kV switchgear units, 240/125V power distribution panels)	No
104	Electrical and I&C	Transformers (e.g., instrument transformers, load center transformers, small distribution transformers, large power transformers, isolation transformers, coupling capacitor voltage transformers)	No
105	Electrical and I&C	Transmitters (e.g., differential pressure transmitters, pressure transmitters, flow transmitters, level transmitters, radiation transmitters, static pressure transmitters)	No
106	Valves	Hydraulic Operated Valves	Yes (Bodies)
107	Valves	Explosive Valves	Yes (Bodies)
108	Valves	Manual Valves	Yes (Bodies)
109	Valves	Small Valves	Yes (Bodies)
110	Valves	Motor-Operated Valves	Yes (Bodies)
111	Valves	Air-Operated Valves	Yes (Bodies)

Table 2.1-5. Typical Structures, Components, and Commodity Groups, and 10 CFR 54.21(a)(1)(i) Determinations for Integrated Plant Assessment (continued)

Item	Category	Structure, Component, or Commodity Grouping	Structure, Component, or Commodity Group Meets 10 CFR 54.21(a)(1)(i) (Yes/No)
112	Valves	Main Steam Isolation Valves	Yes (Bodies)
113	Valves	Small Relief Valves	Yes (Bodies)
114	Valves	Check Valves	Yes (Bodies)
115	Valves	Safety Relief Valves	Yes (Bodies)
116	Valves	Dampers	No
117	Tanks	Air Accumulators	Yes
118	Tanks	Discharge Accumulators (Dampers)	Yes
119	Tanks	Boron Acid Storage Tanks	Yes
120	Tanks	Above Ground Oil Tanks	Yes
121	Tanks	Underground Oil Tanks	Yes
122	Tanks	Demineralized Water Tanks	Yes
123	Tanks	Neutron Shield Tank	Yes
124	Fans	Ventilation Fans	No
125	Fans	Other Fans	No
126	Miscellaneous	Emergency Lighting	No
127	Miscellaneous	Hose Stations	Yes

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2.2 PLANT-LEVEL SCOPING RESULTS

Review Responsibilities

Primary - Branches responsible for systems
Secondary - Branch responsible for electrical engineering

2.2.1 Areas of Review

This section addresses the plant-level scoping results for license renewal. 10 CFR 54.21(a)(1) requires the applicant to identify and list structures and components subject to an aging management review (AMR). These are "passive," "long-lived" structures and components that are within the scope of license renewal. In addition, 10 CFR 54.21(a)(2) requires the applicant to describe and justify the methods used to identify these structures and components. The staff reviews the applicant's methodology separately following the guidance in Section 2.1.

The applicant should provide a list of all the plant systems and structures, identifying those that are within the scope of license renewal. If the list exists elsewhere, such as in the UFSAR, it is acceptable to merely identify the reference. The license renewal rule does not require the identification of all plant systems and structures. However, providing such a list may make the review more efficient. On the basis of the DBEs considered in the plant's CLB, and other CLB information relating to nonsafety-related systems and structures and certain regulated events, the applicant would identify those plant-level systems and structures within the scope of license renewal, as defined in 10 CFR 54.4(a). This is "scoping" of the plant-level systems and structures for license renewal. To verify that the applicant has properly implemented its methodology, the staff focuses its review on the implementation results to confirm that there is no omission of plant-level systems and structures within the scope of license renewal.

Examples of plant systems are the reactor coolant, containment spray, standby gas treatment (BWR), emergency core cooling, open and closed cycle cooling water, compressed air, chemical and volume control (PWR), standby liquid control (BWR), main steam, feedwater, condensate, steam generator blowdown (PWR), and auxiliary feedwater systems (PWR).

Examples of plant structures are the primary containment, secondary containment (BWR), control room, auxiliary building, fuel storage building, radwaste building, and ultimate heat sink cooling tower.

Examples of components are the reactor vessel, reactor vessel internals, steam generator (PWR), and light and heavy load-handling cranes. Some applicants may have categorized such components as plant "systems" for their convenience.

After the plant-level scoping, the applicant should identify the portions of the system or structure that perform an intended function, as defined in 10 CFR 54.4(b). Then the applicant should identify those structures and components that are "passive" and "long-lived" in accordance with 10 CFR 54.21(a)(1)(i) and (ii). These "passive," "long-lived" structures and components are those that are subject to an AMR. The staff reviews these results separately following the guidance in Sections 2.3 through 2.5.

The applicant has the flexibility to determine the set of systems and structures it considers as within the scope of license renewal, provided that this set includes the systems and structures that the NRC has determined are within the scope of license renewal. Therefore, the reviewer need not review all systems and structures that the applicant has identified to be within the scope of license renewal because the applicant has the option to include more systems and components than those defined to be within the scope of license renewal by 10 CFR 54.4.

The following areas relating to the methodology implementation results for the plant-level systems and structures are reviewed.

2.2.1.1 Systems and Structures Within the Scope of License Renewal

The reviewer verifies the applicant's identification of plant-level systems and structures that are within the scope of license renewal.

2.2.2 Acceptance Criteria

The acceptance criteria for the area of review define methods for determining whether the applicant has identified the systems and structures within the scope of license renewal in accordance with NRC regulations in 10 CFR 54.4. For the applicant's implementation of its methodology to be acceptable, the staff should have reasonable assurance that there has been no omission of plant-level systems and structures within the scope of license renewal.

2.2.2.1 Systems and Structures Within the Scope of License Renewal

Systems and structures are within the scope of license renewal as delineated in 10 CFR 54.4(a) if they are

- Safety-related systems and structures that are relied upon to remain functional during and following DBEs [as defined in 10 CFR 50.49(b)(1)] to ensure the following functions:

  - The integrity of the reactor coolant pressure boundary,

  - The capability to shut down the reactor and maintain it in a safe shutdown condition, or

  - The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1), 50.67(b)(2), or 100.11, as applicable.

- Nonsafety-related systems and structures whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1) above.

- Systems and structures relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), PTS (10 CFR 50.61), ATWS (10 CFR 50.62), and SBO (10 CFR 50.63).

2.2.3 Review Procedures

The reviewer verifies the applicant's scoping results. If the reviewer requests additional information from the applicant regarding why a certain system or structure was not identified by the applicant as being within the scope of license renewal for the applicant's plant, the reviewer should provide a focused question, clearly explaining what information is needed, explaining why it is needed, and how it will allow the staff to make its safety finding. In addition, other staff members review the applicant's scoping and screening methodology separately following the guidance in Section 2.1. The reviewer should keep these other staff members informed of findings that may affect their review of the applicant's methodology. The reviewer should coordinate this sharing of information through the license renewal project manager.

For the area of review, the following review procedures are to be followed.

2.2.3.1 Systems and Structures Within the Scope of License Renewal

The reviewer determines whether the applicant has properly identified the plant-level systems and structures within the scope of license renewal by reviewing selected systems and structures that the applicant did not identify as being within the scope of license renewal to verify that they do not have any intended functions.

The reviewer should use the plant UFSAR, orders, applicable regulations, exemptions, and license conditions to determine the design basis for the SSCs (if components are identified as "systems" by the applicant). The design basis determines the intended function(s) of an SSC. Such functions determine whether the SSC is within the scope of license renewal under 54.4.

This section addresses scoping at a system or structure level. Thus, if any portion of a system or structure performs an intended function as defined in 10 CFR 54.4(b), the system or structure is within the scope of license renewal. The review of the individual portions of systems and structures that are within the scope of license renewal are addressed separately in Sections 2.3 through 2.5.

The applicant should submit a list of all plant-level systems and structures, identifying those that are within the scope of license renewal. The reviewer should sample selected systems and structures that the applicant did not identify as within the scope of license renewal to determine if they perform any intended functions. The following are examples:

- The applicant does not identify the radiation monitoring system as being within the scope of license renewal. The reviewer may review the UFSAR to verify that this particular system does not perform any intended functions at the applicant's plant.

- The applicant does not identify the polar crane as being within the scope of license renewal. The reviewer may review the UFSAR to verify that this particular structure is not "Seismic II over I," denoting a non-seismic Category I structure interacting with a Seismic Category I structure as described in Position C.2 of Regulatory Guide 1.29, "Seismic Design Classification" (Ref. 1).

- The applicant does not identify the fire protection pump house as within the scope of license renewal. The reviewer may review the plant's commitments to the fire protection regulation (10 CFR 50.48) to verify that this particular structure does not perform any intended functions at the plant.

- The applicant uses the "spaces" approach for scoping electrical equipment and elects to include all electrical equipment on site to be within the scope of license renewal except for the 525 kV switchyard and the 230 kV transmission lines. The reviewer may review the UFSAR and commitments to the SBO regulation (10 CFR 50.63) to verify that the 525 kV switchyard and the 230 kV transmission lines do not perform any intended functions at the applicant's plant.

Table 2.2-1 contains additional examples based on lessons learned from the review of the initial license renewal applications, including a discussion of the plant-specific determination of whether a system or structure is within the scope of license renewal.

The applicant may choose to group similar components and structures together in commodity groups for separate analyses. If only a portion of a system or structure has an intended function and is addressed separately in a specific commodity group, it is acceptable for an applicant to identify that system or structure as not being within the scope of license renewal. However, for completeness, the applicant should include some reference indicating that the portion of the system or structure with an intended function that is evaluated with the commodity group.

Section 2.1 contains additional guidance on the following:

- Commodity groups
- Complex assemblies
- Hypothetical failure
- Cascading

If the reviewer does not identify any omissions of systems and structures from those within the scope of license renewal, the staff would have reasonable assurance that the applicant has identified the systems and structures within the scope of license renewal.

- If the reviewer determines that the applicant has satisfied the criteria described in this review section, the staff would have reasonable assurance that the applicant has identified the systems and structures within the scope of license renewal.

2.2.4 Evaluation Findings

The reviewer verifies that the applicant has provided information sufficient to satisfy the provision of the SRP-LR and that the staff's evaluation supports conclusions of the following type, to be included in the safety evaluation report:

> The staff concludes that there is reasonable assurance that the applicant has appropriately identified the systems and structures within the scope of license renewal in accordance with 10 CFR 54.4.

2.2.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specific portions of NRC regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

2.2.6 References

1. Regulatory Guide 1.29, Rev. 2, "Seismic Design Classifications," September 1978.

Table 2.2-1. Examples of System and Structure Scoping and Basis for Disposition

Example	Disposition
Recirculation cooling water system	One function of the recirculation cooling water system is to remove decay heat from the stored fuel in the spent fuel pool. However, the fuel handling accident for the plant assumes that the spent fuel pool cooling systems, and thus the recirculation cooling water system, is not functional during or following such an event. Thus, the recirculation cooling water system is not within the scope of license renewal based on this function.
SBO diesel generator building	The plant's UFSAR indicates that certain structural components of the SBO diesel generator building for the plant are designed to preclude seismic failure and subsequent impact of the structure on the adjacent safety-related emergency diesel generator building. In addition, the UFSAR indicates that certain equipment attached to the roof of the building has been anchored to resist tornado wind loads. Thus, the SBO diesel generator building is within the scope of license renewal.

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2.3 SCOPING AND SCREENING RESULTS: MECHANICAL SYSTEMS

Review Responsibilities

Primary - Branches responsible for systems
Secondary - None

2.3.1 Areas of Review

This section addresses the mechanical systems scoping and screening results for license renewal. Typical mechanical systems consist of the following:

- Reactor coolant system (such as reactor vessel and internals, components forming part of coolant pressure boundary, coolant piping system and connected lines, and steam generators).

- Engineered safety features (such as containment spray and isolation systems, standby gas treatment system, emergency core cooling system, and fan cooler system).

- Auxiliary systems (such as new and spent fuel storage, spent fuel cooling and cleanup systems, suppression pool cleanup system, load handling system, open and closed cycle cooling water systems, ultimate heat sink, compressed air system, chemical and volume control system, standby liquid control system, coolant storage/refueling water systems, ventilation systems, diesel generator system, and fire protection system).

- Steam and power conversion system (such as turbines, main and extraction steam, feedwater, condensate, steam generator blowdown, and auxiliary feedwater).

10 CFR 54.21(a)(1) requires an applicant to identify and list structures and components subject to an aging management review (AMR). These are "passive," "long-lived" structures and components that are within the scope of license renewal. In addition, 10 CFR 54.21(a)(2) requires an applicant to describe and justify the methods used to identify these structures and components. The staff reviews the applicant's methodology separately following the guidance in Section 2.1. To verify that the applicant has properly implemented its methodology, the staff focuses its review on the implementation results. Such a focus allows the staff to confirm that there is no omission of mechanical system components that are subject to an AMR by the applicant. If the review identifies no omission, the staff has the basis to find that there is reasonable assurance that the applicant has identified the mechanical system components that are subject to an AMR.

An applicant should list all plant-level systems and structures. On the basis of the DBEs considered in the plant's CLB and other CLB information relating to nonsafety-related systems and structures and certain regulated events, the applicant should identify those plant-level systems and structures within the scope of license renewal, as defined in 10 CFR 54.4(a). This is "scoping" of the plant-level systems and structures for license renewal. The staff reviews the applicant's plant-level "scoping" results separately following the guidance in Section 2.2.

For a mechanical system that is within the scope of license renewal, the applicant should identify the portions of the system that perform an intended function, as defined in 10 CFR 54.4(b). The applicant may identify these particular portions of the system in marked-up piping and instrument diagrams (P&IDs) or other media. This is "scoping" of mechanical components in a system to identify those that are within the scope of license renewal for a system.

For these identified mechanical components, the applicant must identify those that are "passive" and "long-lived" as required by 10 CFR 54.21(a)(1)(i) and (ii). These "passive," "long-lived" mechanical components are those that are subject to an AMR. This is "screening" of mechanical components in a system to identify those that are "passive" and "long-lived."

The applicant has the flexibility to determine the set of structures and components for which an AMR is performed, provided that this set includes the structures and components for which the NRC has determined that an AMR is required. This is based on the SOC for the license renewal rule (60 FR 22478). Therefore, the reviewer need not review all components that the applicant has identified as subject to an AMR because the applicant has the option to include more components than those required to be subject to an AMR pursuant to 10 CFR 54.21(a)(1).

2.3.2 Acceptance Criteria

The acceptance criteria for the areas of review define methods for determining whether the applicant has met the requirements of NRC regulations in 10 CFR 54.21(a)(1). For the applicant's implementation of its methodology to be acceptable, the staff should have reasonable assurance that there has been no omission of mechanical system components that are subject to an AMR.

2.3.2.1 Components Within the Scope of License Renewal

Mechanical components are within the scope of license renewal as delineated in 10 CFR 54.4(a) if they are

- Safety-related SSCs that are relied upon to remain functional during and following DBEs [as defined in 10 CFR 50.49(b)(1)] to ensure the following functions:
  - The integrity of the reactor coolant pressure boundary;
  - The capability to shut down the reactor and maintain it in a safe shutdown condition; or
  - The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.
- All nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii).
- All SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), PTS (10 CFR 50.61), ATWS (10 CFR 50.62), and SBO (10 CFR 50.63).

2.3.2.2 Components Subject to an Aging Management Review

Mechanical components are subject to an AMR if they are within the scope of license renewal and perform an intended function as defined in 10 CFR 54.4(b) without moving parts or a change in configuration or properties ("passive"), and are not subject to replacement based on a qualified life or specified time period ("long-lived") [10 CFR 54.21(a)(1)(i) and (ii)].

2.3.3 Review Procedures

The reviewer verifies the applicant's scoping and screening results. If the reviewer requests additional information from the applicant regarding why a certain component was not identified by the applicant as being within the scope of license renewal or subject to an AMR for the applicant's plant, the reviewer should provide a focused question, that clearly explains what information is needed, why the information is needed, and how the information will allow the staff to make its safety finding. In addition, other staff members review the applicant's scoping and screening methodology separately following the guidance in Section 2.1. The reviewer should keep these other staff members informed of findings that may affect their review of the applicant's methodology. The reviewer should coordinate this sharing of information through the license renewal project manager.

For each area of review, the following review procedures are to be followed.

2.3.3.1 Components Within the Scope of License Renewal

In this step, the staff determines whether the applicant has properly identified the components that are within the scope of license renewal. The Rule requires applicants, to identify components that are subject to an AMR; not components that are within the scope of license renewal (WSLR). Whereas in the past LRAs have included a table of components that are WSLR, the staff does not expect that information to be submitted with future LRAs. Although that information will be available at plant sites for inspection, the reviewer must determine through sampling of P&IDs, and review of FSAR and other plant documents, what portion of the components are within scope. The reviewer must check to see if any components exist that the staff believes are within scope but are not identified by the applicant as being subject to an AMR (and request that the applicant provide justification for omitting those components that are "passive" and "long lived").

The reviewer should use the UFSAR, orders, applicable regulations, exemptions, and license conditions to determine the design basis for the SSCs. The design basis specifies the intended function(s) of the system(s). That intended function is used to determine the components within that system that are required for the system to perform its intended functions.

The reviewer should focus the review on those components that are not identified as being within the scope of license renewal, especially the license renewal boundary points and major flow paths. The reviewer should verify that the components do not have intended functions. Portions of the system identified as being within the scope of license renewal by the applicant do not have to be reviewed because the applicant has the option to include more components within the scope than the rule requires.

Further, the reviewer should select system functions described in the UFSAR that are required by 10 CFR 54.4 to verify that components having intended functions were not omitted from the scope of the rule.

For example, if a reviewer verifies that a portion of a system does not perform an intended function, is not identified as being subject to an AMR by the applicant, and is isolated from the portion of the system that is identified as being subject to an AMR by a boundary valve, the reviewer should verify that the boundary valve is subject to an AMR, or that the valve is not necessary for the within-scope portion of the system to perform its intended function. Likewise, the reviewer should identify, to the extend practical, the system functions of the piping runs and components that are identified as not being within the scope of license renewal to ensure they do not have intended functions that meet the requirements of 10 CFR 54.4.

Section 2.1 contains additional guidance on the following:

- Commodity groups
- Complex assemblies
- Hypothetical failure
- Cascading

If the reviewer does not identify any omissions of components within the scope of license renewal, the reviewer would have reasonable assurance that the applicant has identified the components within the scope of license renewal for the mechanical systems.

Table 2.3-1 provides examples of mechanical components scoping lessons learned from the review of the initial license renewal applications and the basis for their disposition.

2.3.3.2 Components Subject to an Aging Management Review

In this step, the reviewer determines whether the applicant has properly identified the components subject to an AMR from among those which are within the scope of license renewal (i.e., those identified in Subsection 2.3.3.1). The reviewer should review selected components that the applicant has identified as within the scope of license renewal but as not subject to an AMR. The reviewer should verify that the applicant has not omitted from an AMR components that perform intended functions without moving parts or without a change in configuration or properties and that are not subject to replacement on the basis of a qualified life or specified time period.

Starting with the boundary verified in Subsection 2.3.3.1, the reviewer should sample components that are within the scope of license renewal for that system, but were not identified by the applicant as subject to an AMR. Only components that are "passive" and "long-lived" are subject to an AMR. Table 2.1-5 is provided for the reviewer to assist in identifying whether certain components are "passive." The applicant should justify omitting a component from an AMR that is within the scope of license renewal at their facility and is listed as "passive" on Table 2.1-5. Although Table 2.1-5 is extensive, it may not be all inclusive. Thus, the reviewer should use other available information sources, such as prior application reviews, to determine whether a component may be subject to an AMR.

For example, an applicant has marked a boundary of a certain system that is within the scope of license renewal. The marked-up diagram shows that there are pipes, valves, and air compressors within this boundary. The applicant has identified piping and valve bodies as subject to an AMR. Because Table 2.1-5 indicates that air compressors are not subject to an AMR, the reviewer should find the applicant's determination acceptable.

Section 2.1 contains additional guidance on screening the following:

- Consumables
- Heat exchanger intended functions
- Multiple functions

If the reviewer does not identify any omissions of components from those that are subject to an AMR, the staff would then have reasonable assurance that the applicant has identified the components subject to an AMR for the mechanical systems.

Table 2.3-2 provides examples of mechanical components screening developed from lessons learned during the review of the initial license renewal applications and bases for their disposition.

If the applicant determines that a component is subject to an AMR, the applicant should also identify the component's intended function, as defined in 10 CFR 54.4. Such functions must be maintained by any necessary AMRs. Table 2.3-3 provides examples of mechanical component intended functions.

2.3.4 Evaluation Findings

The reviewer verifies that the applicant has provided information sufficient to satisfy the provisions of the SRP-LR and that the staff's evaluation supports conclusions of the following type, to be included in the safety evaluation report:

> The staff concludes that there is reasonable assurance that the applicant has appropriately identified the mechanical system components subject to an aging management review in accordance with the requirements stated in 10 CFR 54.21(a)(1).

2.3.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specific portions of NRC regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

2.3.6 References

None.

Table 2.3-1. Examples of Mechanical Components Scoping and Basis for Disposition

Example	Disposition
Piping segment that provides structural support	The safety-related/nonsafety-related boundary along a pipe run may occur at a valve location. The nonsafety-related piping segment between this valve and the next seismic anchor provides structural support in a seismic event. This piping segment is within the scope of license renewal.
Containment heating and ventilation system ductwork downstream of the fusible links providing cooling to the steam generator compartment and reactor vessel annulus	This nonsafety-related ductwork provides cooling to support the applicant's environmental qualification program. However, the failure of the cavity cooling system ductwork will not prevent the satisfactory completion of any critical safety function during and following a design basis event. Thus, this ductwork is not within the scope of license renewal.
Standpipe installed inside the fuel oil storage tank	The standpipe as described in the applicant's CLB ensures that there is sufficient fuel oil reserve for the emergency diesel generator to operate for the number of days specified in the plant technical specifications following DBEs. Therefore, this standpipe is within the scope of license renewal.
Insulation on boron injection tank	The temperature is high enough that insulation is not necessary to prevent boron precipitation. The plant technical specifications require periodic verification of the tank temperature. Thus, the insulation is not relied on to ensure the function of the emergency system and is not within the scope of license renewal.
Pressurizer spray head	The spray head is not credited for the mitigation of any accidents addressed in the UFSAR accident analyses. The function of the pressurizer spray is to reduce reactor coolant system pressure during normal operating conditions. Therefore, the spray head is not within the scope of license renewal.

Table 2.3-2. Examples of Mechanical Components Screening and Basis for Disposition

Example	Disposition
Diesel engine jacket water heat exchanger, and portions of the diesel fuel oil system and starting air system supplied by a vendor on a diesel generator skid	These are "passive," "long-lived" components having intended functions. They are subject to an AMR for license renewal even though the diesel generator is considered "active."
Fuel assemblies	The fuel assemblies are replaced at regular intervals based on the fuel cycle of the plant. They are not subject to an AMR.
Valve internals (such as disk and seat)	10 CFR 54.21(a)(1)(i) excludes valves, other than the valve body, from AMR. The statements of consideration of the license renewal rule provide the basis for excluding structures and components that perform their intended functions with moving parts or with a change in configuration or properties. Although the valve body is subject to an AMR, valve internals are not.

Table 2.3-3. Examples of Mechanical Component Intended Functions

Component	Intended Function[a]
Piping	Pressure boundary
Valve body	Pressure boundary
Pump casing	Pressure boundary
Orifice	Pressure boundary flow restriction
Heat exchanger	Pressure boundary heat transfer
Reactor vessel internals	Structural support of fuel assemblies, control rods, and incore instrumentation, to maintain core configuration and flow distribution

[a] The component intended functions are those that support the system intended functions. For example, a heat exchanger in the spent fuel cooling system has a pressure boundary intended function, but may not have a heat transfer function. Similarly, not all orifices have flow restriction as an intended function.

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2.4 SCOPING AND SCREENING RESULTS: STRUCTURES

Review Responsibilities

Primary - Branch responsible for plant systems
Secondary - None

2.4.1 Areas of Review

This section addresses the scoping and screening results of structures and structural components for license renewal. Typical structures include the following:

- The primary containment structure;
- Building structures (such as the intake structure, diesel generator building, auxiliary building, and turbine building);
- Component supports (such as cable trays, pipe hangers, elastomer vibration isolators, equipment frames and stanchions, and HVAC ducting supports);
- Nonsafety-related structures whose failure could prevent safety-related SSC from performing their intended functions (that is, seismic Category II over I structures).

Typical structural components include the following: liner plates, walls, floors, roofs, foundations, doors, beams, columns, and frames.

10 CFR 54.21(a)(1) requires the applicant to identify and list structures and components subject to an aging management review (AMR). These are "passive," "long-lived" structures and components that are within the scope of license renewal. In addition, 10 CFR 54.21(a)(2) requires an applicant to describe and justify the methods used to identify these structures and components. The staff reviews the applicant's methodology separately following the guidance in Section 2.1. To verify that the applicant has properly implemented its methodology, the staff focuses its review on the implementation results. Such a focus allows the staff to confirm that there is no omission of structures that are subject to an AMR by the applicant. If the staff's review identifies no omission, the staff has a basis to find that there is reasonable assurance that the applicant has identified the structural components that are subject to an AMR.

An applicant should list all plant-level systems and structures. On the basis of the DBEs considered in the plant's CLB and other CLB information relating to nonsafety-related systems and structures and certain regulated events, the applicant should identify those plant-level systems and structures within the scope of license renewal, as defined in 10 CFR 54.4(a). This is "scoping" of the plant-level systems and structures for license renewal. The staff reviews the applicant's plant-level "scoping" results separately following the guidance in Section 2.2.

For structures that are within the scope of license renewal, an applicant must identify the structural components that are "passive" and "long-lived" in accordance with 10 CFR 54.21(a)(1)(i) and (ii). These "passive," "long-lived" structural components are those that are subject to an AMR ("screening"). The applicant's methodology implementation results for identifying structural components subject to an AMR is the area of review.

The applicant has the flexibility to determine the set of structures and components for which an AMR is performed, provided that this set includes the structures and components for which the NRC has determined that an AMR is required. This flexibility is described in the statements of consideration for the License Renewal Rule (60 FR 22478). Therefore, the reviewer should not focus the review on structural components that the applicant has already identified as subject to an AMR because it is an applicant's option to include more structural components than those subject to an AMR, pursuant to 10 CFR 54.21(a)(1). Rather, the reviewer should focus on those structural components that are not included by the applicant as subject to an AMR to ensure that they do not perform an intended function as defined in 10 CFR 54.4(b) or are not "passive" and "long-lived."

2.4.2 Acceptance Criteria

The acceptance criteria for the areas of review define methods for determining whether the applicant has met the requirements of NRC regulations in 10 CFR 54.21(a)(1). For the applicant's implementation of its methodology to be acceptable, the staff should have reasonable assurance that there has been no omission of structural components that are subject to an AMR.

2.4.2.1 Structural Components Subject to an Aging Management Review

Structural components are within the scope of license renewal as delineated in 10 CFR 54.4(a) if they are

- Safety-related SSCs that are relied upon to remain functional during and following DBEs [as defined in 10 CFR 50.49(b)(1)] to ensure the following functions:

  - The integrity of the reactor coolant pressure boundary;

  - The capability to shut down the reactor and maintain it in a safe shutdown condition; or

  - The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2), or 10 CFR 100.11, as applicable.

- All nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii).

- All SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations for fire protection (10 CFR 50.48), environmental qualification (10 CFR 50.49), PTS (10 CFR 50.61), ATWS (10 CFR 50.62), and SBO (10 CFR 50.63).

Structural components are subject to an AMR if they are within the scope of license renewal and perform an intended function as defined in 10 CFR 54.4(b) without moving parts or a change in configuration or properties ("passive"), and are not subject to replacement based on a qualified life or specified time period ("long-lived") [10 CFR 54.21(a)(1)(i) and (ii)].

2.4.3 Review Procedures

The reviewer verifies the applicant's scoping/screening results. If the reviewer request additional information from the applicant regarding why a certain structure was not identified by the applicant as subject to an AMR for the plant, the reviewer should provide a focused question that clearly explain what information is needed, why the information is needed, and how the information will allow the staff to make its safety finding. In addition, other staff members review the applicant's scoping and screening methodology separately following the guidance in Section 2.1. The reviewer should keep these other staff members informed of findings that may affect their review of the applicant's methodology. The reviewer should coordinate this sharing of information through the license renewal project manager.

For each area of review, the following review procedures are to be followed:

2.4.3.1 Structural Components Within the Scope of License Renewal

In this step, the staff determines which structures and structural components are within the scope of license renewal. The Rule requires applicants, to identify structures that are subject to an AMR; not structures that are within the scope of license renewal (WSLR). Whereas in the past LRAs have included a table of structures that are WSLR, the staff does not expects that information to be submitted with future LRAs. Although that information will be available at plant sites for inspection, the reviewer must determine through sampling of P&IDs, and review of the FSAR and other plant documents, what portion of the components are within scope. The reviewer should check to see if any structures exist that the staff believes are within scope but are not identified by the applicant as being subject to an AMR (and request that the applicant provide justification for omitting those structures that are "passive" and "long lived").

2.4.3.2 Structural Components Subject to an Aging Management Review

In general, structural components are "passive" and "long lived." Thus, they are subject to an AMR if they are within the scope of license renewal. For each of the plant-level structures within the scope of license renewal, an applicant should identify those structural components that have intended functions. For example, the applicant may identify that its auxiliary building is within the scope of license renewal. For this auxiliary building, the applicant may identify the structural components of beams, concrete walls, blowout panels, etc., that are subject to an AMR. The applicant should justify omitting a component from an AMR that is within the scope of license renewal at their facility and is listed as "passive" on Table 2.1-5. Although Table 2.1-5 is extensive, it may not be all inclusive. Thus, the reviewer should use other available information, such as prior application reviews, to determine whether a component may be subject to an AMR.

As set forth below, the reviewer should focus on individual structure not subject to an AMR, one at a time, to confirm that the structural components that have intended functions have been identified by the applicant. In a few instances, only portions of a particular building are within the scope of license renewal. For example, a portion of a particular turbine building provides shelter for some safety-related equipment, which is an intended function, and the remainder of this particular building does not have any intended functions. In this case, the reviewer should verify that the applicant has identified the relevant particular portion of the turbine building as being within the scope of license renewal and subject to an AMR.

The reviewer should use the UFSAR, orders, applicable regulations, exemptions, and license conditions to determine the design basis for the SSCs. The design basis specifies the intended function(s) of the system(s). That intended function is used to determine the components within that system that are required for the system to perform its intended functions.

The reviewer should focus the review on those structural components that have not been identified as being within the scope of license renewal. For example, for a building within the scope of license renewal, if an applicant did not identify the building roof as subject to an AMR, the reviewer should verify that the roof has no intended functions, such as a "Seismic II over I" concern in accordance with the plant's CLB. The reviewer need not verify all structural components that have been identified as subject to an AMR by the applicant because the applicant has the option to include more structural components than the rule requires to be subject to an AMR.

Further, the reviewer should select functions described in the UFSAR to verify that structural components having intended functions were not omitted from the scope of the review. For example, if the UFSAR indicates that a dike within the fire pump house prevents a fuel oil fire from spreading to the electrically driven fire pump, the reviewer should verify that this dike has been identified as being within the scope of license renewal. Another example, if a non-safety-related structure or component is included in the plant's CLB as a part of the safe shutdown path resulting from the resolution of USI A-46, the reviewer should verify that the structure or component has been included within the scope of license renewal.

The applicant should also identify the intended functions of structural components. Table 2.1-4 provides typical "passive" structural component intended functions.

The staff has developed additional scoping/screening guidance. For example, some structural components may be grouped together as a commodity, such as pipe hangers, and some structural components are considered consumable materials, such as sealants. Additional guidance on these and others are contained in Section 2.1 for the following:

- Commodity groups
- Hypothetical failure
- Cascading
- Consumables
- Multiple functions

If the reviewer does not identify any omissions of components from those that are subject to an AMR, the staff would have reasonable assurance that the applicant has identified the components subject to an AMR for the structural systems.

Table 2.4-1 provides examples of structural components scoping/screening lessons learned from the review of initial license renewal applications and the basis for disposition.

If the applicant determines that a structural component may be subject to an AMR, the applicant should also identify the component's intended functions, as defined in 10 CFR 54.4. Such functions must be maintained by any necessary AMPs.

If the reviewer determines that the applicant has satisfied the criteria described in this review section, the staff would have reasonable assurance that the applicant has identified the components that are within the scope of license renewal and subject to an AMR.

2.4.4 Evaluation Findings

The reviewer verifies that the applicant has provided information sufficient to satisfy the provisions of the SRP-LR and that the staff's evaluation supports conclusions of the following type, to be included in the safety evaluation report:

> The staff concludes that there is reasonable assurance that the applicant has appropriately identified the structural components subject to an aging management review in accordance with the requirements stated in 10 CFR 54.21(a)(1).

2.4.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specific portions of NRC regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

2.4.6 References

None.

Table 2.4-1. Examples of Structural Components Scoping/Screening and Basis for Disposition

Example	Disposition
Roof of turbine building	An applicant indicates that degradation or loss of its turbine building roof will not result in the loss of any intended functions. The turbine building contains safety-related SSCs in the basement, which would remain sheltered and protected by several reinforced concrete floors if the turbine building roof were to degrade. Because this roof does not perform an intended function, it is not within the scope of license renewal.
Post-tensioned containment tendon gallery	The intended function of the post-tensioning system is to impose compressive forces on the concrete containment structure to resist the internal pressure resulting from a DBA with no loss of structural integrity. Although the tendon gallery is not relied on to maintain containment integrity during DBEs, operating experience indicates that water infiltration and high humidity in the tendon gallery can contribute to a significant aging effect on the vertical tendon anchorages that could potentially result in loss of the ability of the post-tensioning system to perform its intended function. However, containment inspections provide reasonable assurance that the aging effects of the tendon anchorages, including those in the gallery, will continue to perform their intended functions. Because the tendon gallery itself does not perform an intended function, it is not within the scope of license renewal.
Water-stops	Ground water leakage into the auxiliary building could occur as a result of degradation to the water-stops. This leakage may cause flooding of equipment within the scope of license renewal. (The plant's UFSAR discusses the effects of flooding.) The water-stops perform their functions without moving parts or a change in configuration, and they are not typically replaced. Thus, the water-stops are subject to an AMR. However, they need not be called out explicitly in the scoping/screening results if they are included as parts of structural components that are subject to an AMR.

2.5 SCOPING AND SCREENING RESULTS: ELECTRICAL AND INSTRUMENTATION AND CONTROLS SYSTEMS

Review Responsibilities

Primary - Branch responsible for electrical and instrumentation and controls engineering
Secondary - None

2.5.1 Areas of Review

This review plan section addresses the electrical and instrumentation and controls (I&C) scoping and screening results for license renewal. Typical electrical and I&C components that are subject to an aging management review (AMR) for license renewal include electrical cables and connections.

10 CFR 54.21(a)(1) requires an applicant to identify and list structures and components subject to an AMR. These are "passive," "long-lived" structures and components that are within the scope of license renewal. In addition, 10 CFR 54.21(a)(2) requires an applicant to describe and justify the methods used to identify these structures and components. The staff reviews the applicant's methodology separately following the guidance in Section 2.1. To verify that the applicant has properly implemented its methodology, the staff focuses its review on the implementation results. Such focus gives the staff reasonable assurance that there has been no omission of electrical and I&C components that are subject to an AMR by the applicant. If the staff's review identifies no omission, the staff has a basis to find that there is reasonable assurance that the applicant has identified the electrical and I&C components subject to an AMR.

An applicant should list all plant-level systems and structures. On the basis of the DBEs considered in the plant's CLB and other CLB information relating to nonsafety-related systems and structures and certain regulated events, the applicant would identify those plant-level systems and structures that are within the scope of license renewal, as defined in 10 CFR 54.4(a). This is "scoping" of the plant-level systems and structures for license renewal. The staff reviews the applicant's plant-level "scoping" results separately following the guidance in Section 2.2.

For an electrical and I&C system that is within the scope of license renewal, an applicant may not identify the specific electrical and I&C components that are subject to an AMR. For example, an applicant may not "tag" each specific length of cable that is "passive" and "long-lived," and performs an intended function as defined in 10 CFR 54.4(b). Instead, an applicant may use the so-called "plant spaces" approach (Ref. 1), which is explained below. The "plant spaces" approach provides efficiencies in AMR of electrical equipment located within the same plant space environment.

Under the "plant spaces" approach, an applicant would identify all "passive," "long-lived" electrical equipment within a specified plant space as subject to an AMR, regardless of whether these components perform any intended functions. For example, an applicant could identify all "passive," "long-lived" electrical equipment located within the turbine building ("plant space") to be subject to an AMR for license renewal. In the subsequent AMR, the applicant would evaluate the environment of the turbine building to determine the appropriate aging management activities for this equipment. The applicant has options to further refine this encompassing scope on an as-needed basis. For this example, if the applicant identified elevated temperatures in a particular area within the turbine building, the applicant may elect to further refine the scope in this particular area by identifying electrical equipment that is not subject to an AMR and excluding this equipment from the AMR. In this case, the excluded electrical equipment would be reported in the application as not being subject to an AMR.

10 CFR 54.21(a)(1)(i) provides many examples of electrical and I&C components that are not considered to be "passive" and are not subject to an AMR for license renewal. Therefore, the applicant is expected to identify only a few electrical and I&C components, such as electrical penetrations, cables, and connections, that are "passive" and subject to an AMR. However, the TLAA evaluation requirements in 10 CFR 54.21(c) apply to environmental qualification of electrical equipment, which is not limited to "passive" components.

An applicant has the flexibility to determine the set of electrical and I&C components for which an AMR is performed, provided that this set includes the electrical and I&C components for which the NRC has determined an AMR is required. This is based on the statements of consideration for the License Renewal Rule (60 FR 22478). Therefore, the reviewer need not review all components that the applicant has identified as subject to an AMR because the applicant has the option to include more components than those required by 10 CFR 54.21(a)(1).

2.5.2 Acceptance Criteria

The acceptance criteria for the areas of review define methods for determining whether the applicant has met the requirements of NRC regulations in 10 CFR 54.21(a)(1). For the applicant's implementation of its methodology to be acceptable, the staff should have reasonable assurance that there has been no omission of electrical and I&C system components that are subject to an AMR.

2.5.2.1 Components Within the Scope of License Renewal

Electrical and I&C components are within the scope of license renewal as delineated in 10 CFR 54.4(a) if they are

- Safety-related SSCs that are relied upon to remain functional during and following DBEs (as defined in 10 CFR 50.49(b)(1)) to ensure the following functions:
  - The integrity of the reactor coolant pressure boundary;
  - The capability to shut down the reactor and maintain it in a safe shutdown condition; or
  - The capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to the guidelines in 10 CFR 50.34(a)(1), 10 CFR 50.67(b)(2) or 10 CFR 100.11, as applicable.
- All nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii) or (iii).
- All SSCs relied on in safety analyses or plant evaluations to perform a function that demonstrates compliance with NRC regulations for fire protection (10 CFR

50.48), environmental qualification (10 CFR 50.49), PTS (10 CFR 50.61), ATWS (10 CFR 50.62), and SBO (10 CFR 50.63).

2.5.2.2 Components Subject to an Aging Management Review

Electrical and I&C components are subject to an AMR if they are within the scope of license renewal and perform an intended function as defined in 10 CFR 54.4(b) without moving parts or without a change in configuration or properties ("passive"), and are not subject to replacement based on a qualified life or specified time period ("long-lived") [10 CFR 54.21(a)(1)(i) and (ii)].

2.5.3 Review Procedures

The reviewer verifies the applicant's scoping and screening results. If the reviewer requests additional information from the applicant regarding why a certain component was not identified by the applicant as being within the scope of license renewal or subject to an AMR for the plant, the reviewer should provide a focused question, that clearly explain what information is needed, why the information is needed, and how the information will allow the staff to make its safety finding. In addition, other staff members review the applicant's scoping and screening methodology separately following the guidance in Section 2.1. The reviewer should keep these other staff members informed of findings that may affect their review of the applicant's methodology. The reviewer should coordinate this sharing of information through the license renewal project manager.

The reviewer should verify that an applicant has identified in the license renewal application the electrical and I&C components that are subject to an AMR for its plant. The review procedures are presented below and assume that the applicant has performed "scoping" and "screening" of electrical and I&C system components in that sequence. However, the applicant may elect to perform "screening" before "scoping", which is acceptable because regardless of the sequence, the end result should encompass the electrical and I&C components that are subject to an AMR.

The scope of 10 CFR 50.49 electric equipment to be included within 10 CFR 54.4(a)(3) is that "long-lived" (qualified life of 40 years or greater) equipment already identified by licensees under 10 CFR 50.49(b), which specifies certain electric equipment important to safety. Licensees may rely upon their listing of environmental qualification equipment, as required by 10 CFR 50.49(d), for purposes of satisfying 10 CFR 54.4(a)(3) with respect to equipment within the scope of 10 CFR 50.49 (60 FR 22466). However, the License Renewal Rule has a requirement (10 CFR 54.21(c)) on the evaluation of TLAAs, including environmental qualification (10 CFR 50.49). Environmental qualification equipment is not limited to "passive" equipment. The applicant may identify environmental qualification equipment separately for TLAA evaluation and not include such equipment as subject to an AMR under 10 CFR 54.21(a)(1). The environmental qualification equipment identified for TLAA evaluation would include the "passive" environmental qualification equipment subject to an AMR. The TLAA evaluation would ensure that the environmental qualification equipment would be functional for the period of extended operation. The staff reviews the applicant's environmental qualification TLAA evaluation separately following the guidance in Section 4.4.

For each area of review, the following review procedures are to be followed.

2.5.3.1 Components Within the Scope of License Renewal

In this step, the staff determines whether the applicant has properly identified the components that are within the scope of license renewal. The Rule requires applicants to identify components that are subject to an AMR; not components that are within the scope of license renewal (WSLR). Whereas, in the past, LRAs have included a table of components that are WSLR, the staff does not expects that information to be submitted with future LRAs. Although that information will be available at plant sites for inspection, the reviewer must determine through sampling of one line diagrams, and review of FSAR and other plant documents, what portion of the components are within the scope of license renewal. The reviewer must check to see if any components exist that the staff believes are within the scope but are not identified by the applicant as being subject to AMR (any request that the applicant provide justification for omitting those components that are "passive" and "long lived").

The reviewer should use the UFSAR, orders, applicable regulations, exemptions, and license conditions to determine the design basis for the SSCs. The design basis specifies the intended function(s) of the system(s). That intended function is used to determine the components within that system that are required for the system to perform its intended functions.

The applicant may use the "plant spaces" approach in scoping electrical and I&C components for license renewal. In the "plant spaces" approach, an applicant may indicate that all electrical and I&C components located within a particular plant area ("plant space"), such as the containment and auxiliary building, are within the scope of license renewal. The applicant may also indicate that all electrical and I&C components located within another plant area ("plant space"), such as the warehouse, are not within the scope of license renewal. Table 2.5-1 contains examples of this "plant spaces" approach and the corresponding review procedures.

The applicant would use the "plant spaces" approach for the subsequent AMR of the electrical and I&C components. The applicant would evaluate the environment of the "plant spaces" to determine the appropriate aging management activities for equipment located there. The applicant has the option to further refine this encompassing scope on an as-needed basis. For example, if the applicant identified elevated temperatures in a particular area within a building ("plant space"), the applicant may elect to identify only those "passive," "long-lived" electrical and I&C components that perform an intended function in this particular area as subject to an AMR. This approach of limiting the "plant spaces" is consistent with the "plant spaces" approach. In this case, the reviewer verifies that the applicant has specifically identified the electrical and I&C components that may be within the scope of license renewal in these limited "plant spaces." The reviewer should verify that the electrical and I&C components that the applicant has elected to further exclude do not indeed have any intended functions as defined in 10 CFR 54.4(b).

Section 2.1 contains additional guidance on scoping the following:

- Commodity groups
- Complex assemblies
- Scoping events
- Hypothetical failure
- Cascading

If the reviewer does not identify any omissions of components from those that are within the scope of license renewal, the staff would have reasonable assurance that the applicant has identified the components within the scope of license renewal for the electrical and I&C systems.

2.5.3.2 Component Subject to an Aging Management Review

In this step, the reviewer determines whether the applicant has properly identified the components subject to an AMR from among those which are within the scope of license renewal (i.e., those identified in Subsection 2.5.3.1). The reviewer should review selected components that the applicant has identified as being within the scope of license renewal to verify that the applicant has identified these components as being subject to an AMR if they perform intended functions without moving parts or without a change in configuration or properties and are not subject to replacement on the basis of a qualified life or specified time period. The description of "passive" may also be interpreted to include structures and components that do not display "a change in state."

Only components that are "passive" and "long-lived" are subject to an AMR. Table 2.1-5 lists many typical components and structures, and their associated intended functions, and identifies whether they are "passive." The reviewer should use Table 2.1-5 in identifying whether certain components are "passive." The reviewer should verify that electrical and I&C components identified as "passive" in Table 2.1-5 have been included by the applicant as being subject to an AMR. Although Table 2.1-5 is extensive, it may not be all inclusive. Thus, the reviewer should use other available information sources, such as prior application reviews, to determine whether a component may be subject to an AMR.

Section 2.1 contains additional guidance on screening the following:

- Consumables
- Multiple intended functions

If the reviewer does not identify any omissions of components from those that are subject to an AMR, the staff would have reasonable assurance that the applicant has identified the components subject to an AMR for the electrical and I&C systems.

2.5.4 Evaluation Findings

The reviewer verifies that the applicant has provided information sufficient to satisfy the provisions of the SRP-LR and that the staff's evaluation supports conclusions of the following type, to be included in the safety evaluation report:

> The staff concludes that there is reasonable assurance that the applicant has appropriately identified the electrical and instrumentation and controls system components subject to an aging management review in accordance with the requirements stated in 10 CFR 54.21(a)(1).

2.5.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specific portions of NRC regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

2.5.6 References

1. SAND96-0344, "Aging Management Guideline for Commercial Nuclear Power Plants-Electrical Cable and Terminations," Sandia National Laboratories, September 1996, page 6-11.

Table 2.5-1. Examples of "Plant Spaces" Approach for Electrical and I&C Scoping and Corresponding Review Procedures

Example	Review Procedures
An applicant indicates that all electrical and I&C components on site are within the scope of license renewal.	This is acceptable, and a staff review is not necessary because all electrical and I&C components are included without exception and would include those required by the rule.
An applicant indicates that all electrical and I&C components located in seven specific buildings (containment, auxiliary building, turbine building, etc.) are within the scope of license renewal.	The reviewer should review electrical systems and components in areas outside of these seven buildings ("plant spaces"). The reviewer should verify that the applicant has included any direct-buried cables in trenches between these building as within the scope of license renewal if they perform an intended function. The reviewer should also select buildings other than the seven indicated (for example, the radwaste facility) to verify that they do not contain any electrical and I&C components that perform any intended functions.
An applicant indicates that all electrical and I&C components located on site, except for the 525 kV switchyard, 230 kV transmission lines, radwaste facility, and 44 kV substation, are within the scope of license renewal.	The reviewer should select the specifically excluded "plant spaces" (that is, the 525 kV switchyard, 230 kV transmission lines, radwaste facility, and 44 kV substation) to verify that they do not contain any electrical and I&C components that perform any intended functions.

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CHAPTER 3

AGING MANAGEMENT REVIEW RESULTS

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3.1 AGING MANAGEMENT OF REACTOR VESSEL, INTERNALS, AND REACTOR COOLANT SYSTEM

Review Responsibilities

Primary - Branch responsible for materials and chemical engineering
Secondary - Branch responsible for mechanical engineering

3.1.1 Areas of Review

This review plan section addresses the aging management review (AMR) of the reactor vessel, internals, and reactor coolant system. For a recent vintage plant, the information related to the reactor vessel, internals, and reactor coolant system is contained in Chapter 5, "Reactor Coolant System and Connected Systems," of the plant's final safety analysis report (FSAR), consistent with the Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants (NUREG-0800) (Ref. 1). For older plants, the location of applicable information is plant-specific because their FSAR may have predated NUREG-0800.

The reactor vessel, internals, and reactor coolant system includes the reactor vessel and internals. Also included for BWRs are the reactor coolant recirculation system and portions of other systems connected to the pressure vessel extending to the first isolation valve outside of containment or to the first anchor point. These connected systems include residual heat removal, low-pressure core spray, high-pressure core spray, low-pressure coolant injection, high-pressure coolant injection, reactor core isolation cooling, isolation condenser, reactor coolant cleanup, feedwater, and main steam. For PWRs, the reactor coolant system includes the primary coolant loop, the pressurizer and pressurizer relief tank, and the steam generators. The connected systems for PWRs include the residual heat removal or low pressure injection system, core flood spray or safety injection tank, chemical and volume control system or high pressure injection system, and sampling system.

The staff has issued a generic aging lessons learned (GALL) report addressing aging management for license renewal (Ref. 2). The GALL report documents the staff's basis for determining whether generic existing programs are adequate to manage aging without change or generic existing programs should be augmented for license renewal. The GALL report may be referenced in a license renewal application and should be treated in the same manner as an approved topical report.

Because a license renewal applicant may or may not be able to reference the GALL report as explained below, the following areas are reviewed.

3.1.1.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

The applicant may reference the GALL report in a license renewal application to demonstrate that the programs at its facility correspond to those reviewed and approved in the report and that no further staff review is required. If the material presented in the GALL report is applicable to the applicant's facility, the staff should find the applicant's reference to the report acceptable. In making this determination, the staff should consider whether the applicant has identified specific programs described and evaluated in the GALL report. The staff, however, should not repeat its review of the substance of the matters described in the report. Rather, the staff should confirm that the applicant verifies that the approvals set forth in the GALL report for generic programs apply to the applicant's programs.

3.1.1.2 Further Evaluation of Aging Management as Recommended by the GALL Report

The GALL report provides the basis for identifying those programs that warrant further evaluation during the staff review of a license renewal application. The staff review focus should be on augmented programs for license renewal.

3.1.1.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

The GALL report provides a generic staff evaluation of certain aging management programs (AMPs). If the applicant does not rely on a particular program for license renewal, or if the applicant indicates that the generic staff evaluation of the elements of a particular program does not apply to its plant, the staff should review each such AMP to which the GALL report does not apply.

The GALL report provides a generic staff evaluation of programs for certain components and aging effects. If the applicant has identified particular components subject to aging management review (AMR) for its plant that are not addressed in the GALL report, or particular aging effects for a component that are not addressed in the GALL report, the staff should review the applicant's AMPs applicable to these particular components and aging effects.

3.1.1.4 FSAR Supplement

The FSAR supplement summarizing the programs and activities for managing the effects of aging for the period of extended operation is reviewed.

3.1.2 Acceptance Criteria

The acceptance criteria for the areas of review describe methods for determining whether the applicant has met the requirements of the NRC's regulations in 10 CFR 54.21.

3.1.2.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

Acceptable methods for managing aging of the reactor vessel, internals, and reactor coolant system are described and evaluated in Chapter IV of the GALL report (Ref. 2). In referencing this report, the applicant should indicate that the material presented is applicable to the specific plant involved and should provide the information necessary to adopt the finding of program acceptability as described and evaluated in the report. The applicant should also verify that the approvals set forth in the GALL report for generic programs apply to the applicant's programs. The applicant may reference appropriate programs as described and evaluated in the GALL report.

3.1.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

The GALL report indicates that further evaluation should be performed for the following.

3.1.2.2.1 Cumulative Fatigue Damage (BWR/PWR)

Fatigue is a time-limited aging analysis (TLAA) as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of this TLAA is addressed separately in Section 4.3.

3.1.2.2.2 Loss of Material due to Pitting and Crevice Corrosion (BWR/PWR)

1. Loss of material due to pitting and crevice corrosion could occur in the PWR steam generator shell assembly. The existing program relies on control of chemistry to mitigate corrosion and ISI to detect loss of material. The extent and schedule of the existing steam generator inspections are designed to ensure that flaws cannot attain a depth sufficient to threaten the integrity of the welds. However, according to NRC Information Notice (IN) 90-04 (Ref. 4), if general corrosion pitting of the shell exists, the program may not be sufficient to detect pitting and corrosion. The GALL report recommends augmented inspection to manage this aging effect. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

2. Loss of material due to pitting and crevice corrosion could occur in BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ASME Section XI inservice inspection (ISI). However, the existing program should be augmented to detect loss of material due to pitting or crevice corrosion. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water, and eddy current testing of tubes to ensure that the component's intended function will be maintained during the period of extended operation. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.3 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement (BWR/PWR)

1. Certain aspects of neutron irradiation embrittlement are TLAAs as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The evaluation of this TLAA is addressed separately in Section 4.2.

2. Loss of fracture toughness due to neutron irradiation embrittlement could occur in the reactor vessel. A reactor vessel materials surveillance program monitors neutron irradiation embrittlement of the reactor vessel. Reactor vessel surveillance programs are plant specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant is required to submit its proposed withdrawal schedule for approval prior to implementation. Thus, further staff evaluation is required for license renewal. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3. Loss of fracture toughness due to neutron irradiation embrittlement and void swelling could occur in Westinghouse and B&W baffle/former bolts. The GALL report recommends further

evaluation to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.4 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking (BWR/PWR)

1. Crack initiation and growth due to thermal and mechanical loading or SCC (including intergranular stress corrosion cracking [IGSCC]) could occur in small-bore reactor coolant system and connected system piping less than NPS 4. The existing program relies on ASME Section XI ISI and on control of water chemistry to mitigate SCC. The GALL report recommends that a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping be conducted to ensure that cracking has not occurred and the component intended function will be maintained during the extended period. The AMPs should be augmented by verifying that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. A one-time inspection of a sample of locations is an acceptable method to ensure that the aging effect is not occurring and the component's intended function will be maintained during the period of extended operation.

2. Crack initiation and growth due to thermal and mechanical loading or SCC (including IGSCC) could occur in BWR reactor vessel flange leak detection line and BWR jet pump sensing line. The GALL report recommends that a plant specific aging management program be evaluated to mitigate or detect crack initiation and growth due to SCC of vessel flange leak detection line. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3. Crack initiation and growth due to thermal and mechanical loading or SCC (including IGSCC) could occur in BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate SCC and on ASME Section XI inservice inspection (ISI). However, the existing program should be augmented to detect cracking due to SCC or cyclic loading. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water, and eddy current testing of tubes to ensure that the component's intended function will be maintained during the period of extended operation.

3.1.2.2.5 Crack Growth due to Cyclic Loading (PWR)

Crack growth due cyclic loading could occur in reactor vessel shell and reactor coolant system piping and fittings. Growth of intergranular separations (underclad cracks) in low-alloy or carbon steel heat affected zone under austenitic stainless steel cladding is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all the SA 508-Cl 2 forgings where the cladding was deposited with a high heat input welding process. The methodology for evaluating the underclad flaw should be consistent with the current well-established flaw evaluation procedure and criterion in the ASME Section XI Code. See the Standard Review Plan, Section 4.7, "Other Plant-Specific Time-Limited Aging Analysis," for generic guidance for meeting the requirements of 10 CFR 54.21(c).

3.1.2.2.6 Changes in Dimension due to Void Swelling (PWR)

Changes in dimension due to void swelling could occur in reactor internal components. The GALL report recommends further evaluation to ensure that this aging effect is adequately managed. The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IXB, ASME Section XI. This inspection is not sufficient to detect the effects of changes in dimension due to void swelling. GALL recommends that a plant-specific aging management program should be evaluated. The applicant provides a plant-specific AMP or participates in industry programs to investigate aging effects and determine appropriate AMP. Otherwise, the applicant provides the basis for concluding that void swelling is not an issue for the component. The applicant should either provide the basis for concluding that void swelling is not an issue for the component or provide a program to manage the effects of changes in dimension due to void swelling and the loss of ductility associated with swelling.

3.1.2.2.7 Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking (PWR)

1. Crack initiation and growth due to SCC and primary water stress corrosion cracking (PWSCC) could occur in PWR core support pads (or core guide lugs), instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains. The GALL report recommends further evaluation to ensure that these aging effects are adequately managed. The GALL report recommends that a plant-specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting crack initiation and growth due to SCC. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

2. Crack initiation and growth due to SCC could occur in PWR cast austenitic stainless steel (CASS) reactor coolant system piping and fittings and pressurizer surge line nozzle. The GALL report recommends further evaluation of piping that does not meet either the reactor water chemistry guidelines of TR-105714 or material guidelines of NUREG-0313 (Ref. 5). Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3. Crack initiation and growth due to PWSCC could occur in PWR pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni alloys. The existing program relies on ASME Section XI ISI and on control of water chemistry to mitigate PWSCC. However, the existing program should be augmented to manage the effects of SCC on the intended function of Ni-alloy components. The GALL report recommends that the applicant provide a plant-specific AMP or participate in industry programs to determine appropriate AMP for PWSCC of Inconel 182 weld. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.8 Crack Initiation and Growth due to Stress Corrosion Cracking or Irradiation-Assisted Stress Corrosion Cracking (PWR)

Crack initiation and growth due to SCC or IASCC could occur in baffle/former bolts in Westinghouse and B&W reactors. Historically the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolts at several plants have identified cracking. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project, Issues Task Group (ITG) activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant-specific basis. The GALL report recommends further evaluation to ensure that these aging effects are adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.9 Loss of Preload due to Stress Relaxation (PWR)

Loss of preload due to stress relaxation could occur in baffle/former bolts in Westinghouse and B&W reactors. Visual inspection (VT–3) should be augmented to detect relevant conditions of stress relaxation because only the heads of the baffle/former bolts are visible, and a plant-specific aging management program is thus required. The GALL report recommends a plant-specific aging management program to ensure that these aging effects are adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.10 Loss of Section Thickness due to Erosion (PWR)

Loss of section thickness due to erosion could occur in steam generator feedwater impingement plates and supports. The GALL report recommends further evaluation of a plant-specific aging management program to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.11 Crack Initiation and Growth due to PWSCC, ODSCC, or Intergranular Attack or Loss of Material due to Wastage and Pitting Corrosion or Loss of Section Thickness due to Fretting and Wear or Denting due to Corrosion of Carbon Steel Tube Support Plate (PWR)

Crack initiation and growth due to PWSCC, ODSCC, or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion or deformation due to corrosion could occur in alloy 600 components of the steam generator tubes, repair sleeves and plugs. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. The GALL report recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, should be developed to ensure that this aging effect is adequately managed.

3.1.2.2.12 Loss of Section Thickness due to Flow-accelerated Corrosion

Loss of section thickness due to flow-accelerated corrosion could occur in tube support lattice bars made of carbon steel. The GALL report recommends that a plant-specific aging management program be evaluated and, on the basis of the guidelines of NRC Generic Letter 97-06, an inspection program for steam generator internals be developed to ensure that this aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.13 Ligament Cracking due to Corrosion (PWR)

Ligament cracking due to corrosion could occur in carbon steel components in the steam generator tube support plate. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. The GALL report recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, be developed to ensure that this aging effect is adequately managed.

3.1.2.2.14 Loss of Material due to Flow-accelerated Corrosion (PWR)

Loss of material due to flow-accelerated corrosion could occur in feedwater inlet ring and supports. As noted in Combustion Engineering (CE) Information Notice (IN) 90-04 and NRC IN 91-19 and LER 50-362/90-05-01, this form of degradation has been detected only in certain CE System 80 steam generators. The GALL report recommends further evaluation to ensure that this aging effect is adequately managed. The GALL report recommends that a plant-specific aging management program be evaluated because existing programs may not be capable of mitigating or detecting loss of material due to flow-accelerated corrosion. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.2.15 Quality Assurance for Aging Management of Nonsafety-Related Components

Acceptance criteria are described in Branch Technical Position IQMB-1 (Appendix A.2 of this standard review plan).

3.1.2.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.2.4 FSAR Supplement

The summary description of the programs and activities for managing the effects of aging for the period of extended operation in the FSAR supplement should be appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the bases for determining that aging effects will be managed during the period of extended operation.

3.1.3 Review Procedures

For each area of review, the following review procedures are to be followed.

3.1.3.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

The applicant may reference the GALL report in its license renewal application, as appropriate. The staff should not repeat its review of the substance of the matters described in the report. If the applicant has provided the information necessary to adopt the finding of program acceptability as described and evaluated in the GALL report, the staff should find the applicant's reference to the report in a license renewal application acceptable. In making this determination, the reviewer verifies that the applicant has provided a brief description of the system, components, materials, and environment. The reviewer also verifies that the applicant has stated that the applicable aging effects and industry and plant-specific operating experience have been reviewed by the applicant and are evaluated in the GALL report. The reviewer verifies that the applicant has identified those aging effects for the reactor vessel, internals, and reactor coolant system components that are contained in the report as applicable to its plant. In addition, the reviewer ensures that the applicant has stated that the plant programs covered by the applicant's reference contain the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL report.

The reviewer should verify that the applicant has stated that certain of its AMPs contain the same program elements as the corresponding generic program described in the GALL report and upon which the staff relied in its evaluation. The reviewer should also verify that the applicant has stated that the GALL report is applicable to its plant with respect to these programs. The reviewer verifies that the applicant has identified the appropriate programs as described and evaluated in the GALL report. Programs evaluated in the report regarding the reactor vessel, internals, and reactor coolant system components are summarized in Table 3.1-1 of this review plan. No further staff evaluation is necessary if so recommended in the GALL report.

3.1.3.2 Further Evaluation of Aging Management as Recommended by the GALL Report

3.1.3.2.1 Cumulative Fatigue Damage (BWR/PWR)

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviews the evaluation of this TLAA separately following the guidance in Section 4.3 of this standard review plan.

3.1.3.2.2 Loss of Material due to Pitting and Crevice Corrosion (BWR/PWR)

1. The GALL report recommends further evaluation for the management of loss of material due to pitting and crevice corrosion of the PWR steam generator shell assembly. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ISI for detection. Based on NRC IN 90-04 (Ref. 4), if general corrosion pitting of the shell exists, the existing program requirements may not be sufficient to detect loss of material due to pitting and corrosion, and additional inspection procedures may be required. The reviewer verifies on a case-by-case basis that the applicant has proposed a program that will manage loss of material due to pitting and crevice corrosion by providing enhanced inspection and supplemental methods to detect loss of material and ensure that the component intended function will be maintained during the extended period.

2. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water and eddy current testing of tubes for the management of loss of material due to pitting and crevice corrosion in BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate corrosion and on ASME Section XI ISI for detection. However, the inspection requirements should be augmented to detect loss of material due to pitting and crevice corrosion, and an augmented program to include temperature and radioactivity monitoring of the shell-side water and eddy current testing of tubes is recommended to ensure that the component's intended function will be maintained during the period of extended operation. The reviewer verifies on a case-by-case basis that the applicant has proposed an augmented program that will manage loss of material due to pitting and crevice corrosion and ensure that the component intended function will be maintained during the extended period.

3.1.3.2.3 Loss of Fracture Toughness due to Neutron Irradiation Embrittlement (BWR/PWR)

1. Neutron irradiation embrittlement is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c)(1). The staff reviews the evaluation of this TLAA following the guidance in Section 4.2 of this standard review plan.

2. The GALL report recommends further evaluation of the reactor vessel materials surveillance program for the period of extended operation. Neutron embrittlement of the reactor vessel is monitored by a reactor vessel materials surveillance program. Reactor vessel surveillance programs are plant specific, depending on matters such as the composition of limiting materials, availability of surveillance capsules, and projected fluence levels. In accordance with 10 CFR Part 50, Appendix H, an applicant must submit its proposed withdrawal schedule for approval prior to implementation. Thus, further staff evaluation is required for license renewal. The reviewer verifies on a case-by-case basis that the applicant has proposed an adequate reactor vessel materials surveillance program for the period of extended operation. Specific criteria for an acceptable AMP is provided in chapter XI, Section M31 of the GALL report.

3. The GALL report recommends further evaluation for the management of loss of fracture toughness due to neutron irradiation embrittlement and void swelling of Westinghouse and B&W baffle/former bolts. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.4 Crack Initiation and Growth due to Thermal and Mechanical Loading or Stress Corrosion Cracking (BWR/PWR)

1. The GALL report recommends a plant-specific destructive examination or a nondestructive examination (NDE) that permits inspection of the inside surfaces of the piping for the management of crack initiation and growth due to thermal and mechanical loading or SCC of small-bore reactor coolant system and connected system piping (less than NPS 4). The existing program should be augmented by verifying that service-induced weld cracking is not occurring in the small-bore piping less than NPS 4, including pipe, fittings, and branch connections. See Chapter XI.M32, "One-Time Inspection" for an acceptable verification method. The GALL report recommends that the inspection include a representative sample of the system population, and, where practical and prudent, focus on the bounding or lead components most susceptible to aging due to time in service, severity of operating conditions, and lowest design margin. For small-bore piping, actual inspection locations should be based on physical accessibility, exposure levels, NDE examination techniques, and locations identified in Nuclear Regulatory Commission (NRC) Information Notice (IN) 97-46. Combinations of NDE, including visual, ultrasonic, and surface techniques, are performed by qualified personnel following procedures consistent with the ASME Code and 10 CFR 50 Appendix B. For small-bore piping less than NPS 4 in., including pipe, fittings, and branch connections, a plant-specific destructive examination or NDE that permits inspection of the inside surfaces of the piping should be conducted to ensure that cracking has not occurred. Follow-up of unacceptable inspection findings should include expansion of the inspection sample size and locations. The inspection and test techniques prescribed by the program should verify any aging effects because these techniques, used by qualified personnel, have been proven effective and consistent with staff expectations. The staff reviews to confirm that the program includes measures to verify that unacceptable

degradation is not occurring, thereby validating the effectiveness of existing programs or confirming that there is no need to manage aging-related degradation for the period of extended operation. If an applicant proposes a one-time inspection of select components and susceptible locations to ensure that corrosion is not occurring, the reviewer verifies that the proposed inspection will be performed using techniques similar to ASME Code and ASTM standards including visual, ultrasonic, and surface techniques (Refs. 6 and 7) to ensure that the component's intended function will be maintained during the period of extended operation.

2. The GALL report recommends that a plant specific aging management program be evaluated for the management of crack initiation and growth due to thermal and mechanical loading or SCC (including IGSCC) in BWR reactor vessel flange leak detection line and BWR jet pump sensing line. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3. The GALL report recommends an augmented program to include temperature and radioactivity monitoring of the shell-side water, and eddy current testing of tubes for the management of crack initiation and growth due to thermal and mechanical loading or SCC (including IGSCC) of the BWR isolation condenser components. The existing program relies on control of reactor water chemistry to mitigate SCC and on ASME Section XI inservice inspection (ISI) to detect leakage. However, the existing program should be augmented to detect cracking due to SCC or cyclic loading. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.5 Crack Growth due to Cyclic Loading (PWR)

The GALL report recommends further evaluation of programs to manage crack growth due to cyclic loading in reactor vessel shell and reactor coolant system piping and fittings. Growth of intergranular separations (underclad cracks) in low-alloy or carbon steel heat affected zone under austenitic stainless steel cladding is a time-limited aging analysis (TLAA) to be evaluated for the period of extended operation for all the SA 508-Cl 2 forgings where the cladding was deposited with a high heat input welding process. The methodology for evaluating the underclad flaw should be consistent with the current well-established flaw evaluation procedure and criterion in the ASME Section XI Code. The Standard Review Plan, Section 4.7, "Other Plant-Specific Time-Limited Aging Analysis," provides generic guidance for meeting the requirements of 10 CFR 54.21(c). The staff reviews the evaluation of this TLAA separately following the guidance in Section 4.7 of this standard review plan.

3.1.3.2.6 Changes in Dimension due to Void Swelling (PWR)

The GALL report recommends further evaluation of programs to manage changes in dimension due to void swelling for reactor internal components. Changes in dimension due to void swelling could occur in reactor internal components. The GALL report recommends further evaluation to ensure that this aging effect is adequately managed. The reactor vessel internals receive a visual inspection (VT-3) according to Category B-N-3 of Subsection IWB, ASME Section XI. This inspection is not sufficient to detect the effects of changes in dimension due to void swelling. The GALL report recommends further evaluation of a plant-specific aging management program. The applicant should provide a plant-specific AMP or participate in industry programs to investigate aging effects and determine an appropriate AMP. Otherwise, the applicant should provide the basis for concluding that void swelling is not an issue for the component. The applicant should either provide the basis for concluding that void swelling is not an issue for the component or provide a program to manage the effects of changes in dimension due to void swelling and the loss of ductility associated with swelling. The reviewer verifies on a case-by-case basis that the applicant has either proposed a program to manage changes in dimension due to void swelling in the pressure vessel internal components or provided the basis for concluding that void swelling is not an issue.

3.1.3.2.7 Crack Initiation and Growth due to Stress Corrosion Cracking or Primary Water Stress Corrosion Cracking (PWR)

1. The GALL report recommends that a plant-specific aging management program is to be evaluated to manage crack initiation and growth due to SCC and primary water stress corrosion cracking (PWSCC) in PWR core support pads (or core guide lugs, instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

2. The GALL report recommends further evaluation of programs to manage crack initiation and growth due to SCC of PWR cast austenitic stainless steel (CASS) reactor coolant system piping and fittings and pressurizer surge line nozzle. The GALL report recommends further evaluation of piping that does not meet either the reactor water chemistry guidelines of TR-105714 or material guidelines of NUREG-0313 (Ref. 5). The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3. The GALL report recommends further evaluation of programs to manage crack initiation and growth due to PWSCC of PWR pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni alloys. The existing program relies on ASME Section XI ISI to detect cracks and on control of water chemistry to mitigate PWSCC. However, the program should be augmented to manage the effects of SCC on the intended function of Ni-alloy components. The GALL report recommends the applicant provides a plant-specific AMP or participate in industry programs to determine appropriate AMP for PWSCC of Inconel 182 weld. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.8 Crack Initiation and Growth due to Stress Corrosion Cracking or Irradiation-Assisted Stress Corrosion Cracking (PWR)

The GALL report recommends further evaluation of crack initiation and growth due to SCC or IASCC in Westinghouse and B&W baffle/former bolts. Historically the VT-3 visual examinations have not identified baffle/former bolt cracking because cracking occurs at the juncture of the bolt head and shank, which is not accessible for visual inspection. However, recent UT examinations of the baffle/former bolts at several plants have identified cracking. The industry is currently addressing the issue of baffle bolt cracking in the PWR Materials Reliability Project ITG activities to determine, develop, and implement the necessary steps and plans to manage the applicable aging effects on a plant-specific basis. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.9 Loss of Preload due to Stress Relaxation (PWR)

The GALL report recommends further evaluation of loss of preload due to stress relaxation could occur in baffle/former bolts in Westinghouse and B&W reactors. Visual inspection (VT–3) should be augmented to detect relevant conditions of stress relaxation because only the heads of the baffle/former bolts are visible, and a plant-specific aging management program is thus required. The GALL report recommends a plant-specific aging management program to ensure that these aging effects are adequately managed. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.10 Loss of Section Thickness due to Erosion (PWR)

The GALL report recommends further evaluation of a plant-specific aging management program for the management of loss of section thickness due to erosion of steam generator feedwater impingement plates and supports. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.11 Crack Initiation and Growth due to PWSCC, ODSCC, or Intergranular Attack or Loss of Material due to Wastage and Pitting Corrosion or Loss of Section Thickness due to Fretting and Wear or Denting due to Corrosion of Carbon Steel Tube Support Plate (PWR)

The GALL report recommends further evaluation of (1) crack initiation and growth due to PWSCC, ODSCC, or intergranular attack (IGA); or (2) loss of material due to wastage and pitting corrosion; or (3) deformation due to corrosion in alloy 600 components of the steam generator tubes, repair sleeves, and plugs. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. The GALL report recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, be developed to ensure that this aging effect is adequately managed. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.12 Loss of Section Thickness due to Flow-accelerated Corrosion

The GALL report recommends further evaluation of loss of section thickness due to flow-accelerated corrosion of the tube support lattice bars made of carbon steel. The GALL report recommends a plant-specific aging management program be evaluated and on the basis of the guidelines of NRC Generic Letter 97-06, an inspection program for steam generator internals should be developed to ensure that this aging effect is adequately managed. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.13 Ligament Cracking due to Corrosion (PWR)

The GALL report recommends further evaluation of ligament cracking due to corrosion in carbon steel components in the steam generator tube support plate. All PWR licensees have committed voluntarily to a SG degradation management program described in NEI 97-06; these guidelines are currently under NRC staff review. The GALL report recommends that an AMP based on the recommendations of staff-approved NEI 97-06 guidelines, or other alternate regulatory basis for SG degradation management, be developed to ensure that this aging effect is adequately managed. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.14 Loss of material due to Flow-accelerated Corrosion (PWR)

The GALL report recommends that a plant-specific aging management program be evaluated to manage loss of material due to flow-accelerated corrosion in the feedwater inlet ring and supports. As noted in Combustion Engineering (CE) Information Notice (IN) 90-04 and NRC IN 91-19 and LER 50-362/90-05-01, this form of degradation has been detected only in certain CE System 80 steam generators. The staff reviews the applicant's proposed program on a case-by-case basis to ensure that an adequate program will be in place for the management of these aging effects.

3.1.3.2.15 Quality Assurance for Aging Management of Nonsafety-Related Components

The applicant's aging management programs for license renewal should contain the elements of corrective actions, the confirmation process, and administrative controls. Safety-related components are covered by 10 CFR Part 50 Appendix B, which is adequate to address these program elements. However, Appendix B does not apply to nonsafety-related components that are subject to an AMR for license renewal. Nevertheless, the applicant has the option to expand the scope of its 10 CFR Part 50 Appendix B program to include these components and address the associated program elements. If the applicant chooses this option, the reviewer verifies that the applicant has documented such a commitment in the FSAR supplement. If the applicant chooses alternative means, the branch responsible for quality assurance should be requested to review the applicant's proposal on a case-by-case basis.

3.1.3.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Review procedures are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.1.3.4 FSAR Supplement

The reviewer verifies that the applicant has provided information, equivalent to that in Table 3.1-2, in the FSAR supplement for aging management of the reactor vessel, internals, and reactor coolant system for license renewal. The reviewer also verifies that the applicant has provided information, equivalent to that in Table 3.1-2, in the FSAR supplement for Subsection 3.1.3.3, "Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report."

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license conditions, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

As noted in Table 3.1-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

3.1.4 Evaluation Findings

The reviewer verifies that the applicant has provided sufficient information to satisfy the provisions of this review plan section, and the staff's evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The staff concludes that the applicant has demonstrated that the aging effects associated with the reactor vessel, internals, and reactor coolant system will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the current licensing basis during the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the programs and activities for managing the effects of aging for the reactor vessel, internals, and reactor coolant system as reflected in the license conditions.

3.1.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

3.1.6 References

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1981.

2. NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, July 2001.

3. NEI 97-06, "Steam Generator Program Guidelines," Nuclear Energy Institute, December 1997.

4. NRC Information Notice 90-04, "Cracking of the Upper Shell-to-Transition Cone Girth Welds in Steam Generators," U.S. Nuclear Regulatory Commission, January 26, 1990.

5. NUREG-0313, Rev. 2, "Technical Report on Material Selection and Processing Guidelines for BRW Coolant Pressure Boundary Piping, U.S. Nuclear Regulatory Commission, January 1988.

6. EPRI TR-107569-V1R5, "PWR Steam Generator Examination Guidelines, Rev. 5," Electric Power Research Institute September 1997.

7. NRC Regulatory Guide 1.83, "Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes," U.S. Nuclear Regulatory Commission, June 1974.

8. NRC Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes (for Comment)," U.S. Nuclear Regulatory Commission, May 1976.

9. NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," U.S. Nuclear Regulatory Commission, August 3, 1995.

10. NRC Information Notice 90-10, "Primary Water Stress Corrosion Cracking (PWSCC) of Inconel 600," U.S. Nuclear Regulatory Commission, February 23, 1990.

11. NRC Information Notice 90-30, "Ultrasonic Inspection Techniques for Dissimilar Metal Welds," U.S. Nuclear Regulatory Commission, May 1, 1990.

12. NRC Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," May 2, 1989.

13. NSAC-202L-R2, "Recommendations for an Effective Flow-accelerated Corrosion Program," Electric Power Research Institute, April 1999.

14. NRC Information Notice 96-11, "Ingress of Demineralizer Resins Increase Potential for Stress Corrosion Cracking of Control Rod Drive Mechanism Penetrations," February 14, 1996.

15. NRC Generic Letter 97-06, "Degradation of Steam Generator Internals," U.S. Nuclear Regulatory Commission, December 30, 1997.

16. BWRVIP-29 (EPRI TR-103515), *BWR Water Chemistry Guidelines-Revision 3, Normal and Hydrogen Water Chemistry*, Electric Power Research Institute, Palo Alto, CA, February 1994.

17. EPRI NP-5769, *Degradation and Failure of Bolting in Nuclear Power Plants*, Volumes 1 and 2, Electric Power Research Institute, Palo Alto, CA, April 1988.

18. EPRI TR-105714, *PWR primary Water Chemistry Guidelines-Revision 3*, Electric Power Research Institute, Palo Alto, CA, Nov. 1995.

19. NRC Generic Letter 88-01, *NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping*, January 25, 1988.

20. NRC Generic Letter 97-01, *Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations*, April 1,1997.

21. NRC Information Notice 97-46, *Unisolable Crack in High-Pressure Injection Piping*, July 9, 1997.

22. NRC Regulatory Guide 1.99, Rev. 2, *Radiation Embrittlement of Reactor Vessel Materials,* May 1988.

23. NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, U.S. Nuclear Regulatory Commission, November 1980.

24. NUREG-1339, *Resolution of Generic Safety Issue 29: Bolting Degradation or Failure in Nuclear Power Plants*, Richard E. Johnson, U.S. Nuclear Regulatory Commission, June 1990.

25. EPRI TR-104213, *Bolted Joint Maintenance & Application Guide,* Electric Power Research Institute, Palo Alto, CA, December 1995.

26. NEI letter dated Dec. 11, 1998, Dave Modeen to Gus Lainas, "Responses to NRC Requests for Additional Information (RAIs) on GL 97-01."

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR/PWR	Reactor coolant pressure boundary components	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (See Subsection 3.1.2.2.1)
PWR	Steam generator shell assembly	Loss of material due to pitting and crevice corrosion	Inservice inspection; water chemistry	Yes, detection of aging effects is to be further evaluated (See Subsection 3.1.2.2.2.1)
BWR	Isolation condenser	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection; water chemistry	Yes, plant specific (See Subsection 3.1.2.2.2.2)
BWR/PWR	Pressure vessel ferritic materials that have a neutron fluence greater than 10^{17} n/cm^2 (E>1 MeV)	Loss of fracture toughness due to neutron irradiation embrittlement	TLAA, evaluated in accordance with Appendix G of 10 CFR 50 and RG 1.99	Yes, TLAA (See Subsection 3.1.2.2.3.1)
BWR/PWR	Reactor vessel beltline shell and welds	Loss of fracture toughness due to neutron irradiation embrittlement	Reactor vessel surveillance	Yes, plant specific (See Subsection 3.1.2.2.3.2)
PWR	Westinghouse and B&W baffle/former bolts	Loss of fracture toughness due to neutron irradiation embrittlement and void swelling	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.3.3)
BWR/PWR	Small-bore reactor coolant system and connected systems piping	Crack initiation and growth due to SCC, intergranular SCC, and thermal and mechanical loading	Inservice inspection; water chemistry; one-time inspection	Yes, parameters monitored/inspected and detection of aging effects are to be further evaluated (See Subsection 3.1.2.2.4.1)
BWR	Jet pump sensing line, and reactor vessel flange leak detection line	Crack initiation and growth due to SCC, intergranular stress corrosion cracking (IGSCC), or cyclic loading	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.4.2)
BWR	Isolation condenser	Crack initiation and growth due to stress corrosion cracking (SCC) or cyclic loading;	Inservice inspection; water chemistry	Yes, plant specific (See Subsection 3.1.2.2.4.3)
PWR	Vessel shell	Crack growth due to cyclic loading	TLAA	Yes, TLAA (See Subsection 3.1.2.2.5)

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
PWR	Reactor internals	Changes in dimension due to void swelling	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.6)
PWR	PWR core support pads, instrument tubes (bottom head penetrations), pressurizer spray heads, and nozzles for the steam generator instruments and drains	Crack initiation and growth due to SCC and/or primary water stress corrosion cracking (PWSCC)	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.7.1)
PWR	Cast austenitic stainless steel (CASS) reactor coolant system piping	Crack initiation and growth due to SCC	Plant specific	Yes, plant specific(See Subsection 3.1.2.2.7.2)
PWR	Pressurizer instrumentation penetrations and heater sheaths and sleeves made of Ni-alloys	Crack initiation and growth due to PWSCC	Inservice inspection; water chemistry	Yes, AMP for PWSCC of Inconel 182 weld is to be evaluated (See Subsection 3.1.2.2.7.3)
PWR	Westinghouse and B&W baffle former bolts	Crack initiation and growth due to SCC and IASCC	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.8)
PWR	Westinghouse and B&W baffle former bolts	Loss of preload due to stress relaxation	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.9)
PWR	Steam generator feedwater impingement plate and support	Loss of section thickness due to erosion	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.10)

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
PWR	(Alloy 600) Steam generator tubes, repair sleeves, and plugs	Crack initiation and growth due to PWSCC, outside diameter stress corrosion cracking (ODSCC), and/or intergranular attack (IGA) or loss of material due to wastage and pitting corrosion, and fretting and wear; or deformation due to corrosion at tube support plate intersections	Steam generator tubing integrity; water chemistry	Yes, effectiveness of a proposed AMP is to be evaluated (See Subsection 3.1.2.2.11)
PWR	Tube support lattice bars made of carbon steel	Loss of section thickness due to FAC	Plant specific	Yes, plant specific (See Subsection 3.1.2.2.12)
PWR	Carbon steel tube support plate	Ligament cracking due to corrosion	Plant specific	Yes, effectiveness of a proposed AMP is to be evaluated (See Subsection 3.1.2.2.13)
PWR (CE)	Steam generator feedwater inlet ring and supports	Loss of material due to flow-corrosion	Combustion engineering (CE) steam generator feedwater ring inspection	Yes, plant specific (See Subsection 3.1.2.2.14)
BWR/PWR	Reactor vessel closure studs and stud assembly	Crack initiation and growth due to SCC and/or IGSCC	Reactor head closure studs	No
BWR/PWR	CASS pump casing and valve body	Loss of fracture toughness due to thermal aging embrittlement	Inservice inspection	No
BWR/PWR	CASS piping	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No
BWR/PWR	BWR piping and fittings; steam generator components	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR/ PWR	Reactor coolant pressure boundary (RCPB) valve closure bolting, manway and holding bolting, and closure bolting in high pressure and high temperature systems	Loss of material due to wear; loss of preload due to stress relaxation; crack initiation and growth due to cyclic loading and/or SCC	Bolting integrity	No
BWR	Feedwater and control rod drive (CRD) return line nozzles	Crack initiation and growth due to cyclic loading	Feedwater nozzle; CRD return line nozzle	No
BWR	Vessel shell attachment welds	Crack initiation and growth due to SCC, IGSCC	BWR vessel ID attachment welds; water chemistry	No
BWR	Nozzle safe ends, recirculation pump casing, connected systems piping and fittings, body and bonnet of valves	Crack initiation and growth due to SCC, IGSCC	BWR stress corrosion cracking; water chemistry	No
BWR	Penetrations	Crack initiation and growth due to SCC, IGSCC, cyclic loading	BWR penetrations; water chemistry	No
BWR	Core shroud and core plate, support structure, top guide, core spray lines and spargers, jet pump assemblies, control rod drive housing, nuclear instrumentation guide tubes	Crack initiation and growth due to SCC, IGSCC, IASCC	BWR vessel internals; water chemistry	No
BWR	Core shroud and core plate access hole cover (welded and mechanical covers)	Crack initiation and growth due to SCC, IGSCC, IASCC	ASME Section XI inservice inspection; water chemistry	No
BWR	Jet pump assembly castings; orificed fuel support	Loss of fracture toughness due to thermal aging and neutron embrittlement	Thermal aging and neutron irradiation embrittlement	No

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR	Unclad top head and nozzles	Loss of material due to general, pitting, and crevice corrosion	Inservice inspection; water chemistry	No
PWR	CRD nozzle	Crack initiation and growth due to PWSCC	Ni-alloy nozzles and penetrations; water chemistry	No
PWR	Reactor vessel nozzles safe ends and CRD housing; reactor coolant system components (except CASS and bolting)	Crack initiation and growth due to cyclic loading, and/or SCC, and PWSCC	Inservice inspection; water chemistry	No
PWR	Reactor vessel internals CASS components	Loss of fracture toughness due to thermal aging, neutron irradiation embrittlement, and void swelling	Thermal aging and neutron irradiation embrittlement	No
PWR	External surfaces of carbon steel components in reactor coolant system pressure boundary	Loss of material due to boric acid corrosion	Boric acid corrosion	No
PWR	Steam generator secondary manways and handholds (CS)	Loss of material due to erosion	Inservice inspection	No
PWR	Reactor internals, reactor vessel closure studs, and core support pads	Loss of material due to wear	Inservice inspection	No
PWR	Pressurizer integral support	Crack initiation and growth due to cyclic loading	Inservice inspection	No
PWR	Upper and lower internals assembly (Westinghouse)	Loss of preload due to stress relaxation	Inservice inspection; loose part and/or neutron noise monitoring	No
PWR	Reactor vessel internals in fuel zone region (except Westinghouse and Babcock & Wilcox [B&W] baffle bolts)	Loss of fracture toughness due to neutron irradiation embrittlement, and void swelling	PWR vessel internals; water chemistry	No

Table 3.1-1. Summary of Aging Management Programs for Reactor Vessel, Internals, and Reactor Coolant System Evaluated in Chapter IV of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
PWR	Steam generator upper and lower heads; tubesheets; primary nozzles and safe ends	Crack initiation and growth due to SCC, PWSCC. IASCC	Inservice inspection; water chemistry	No
PWR	Vessel internals (except Westinghouse and B&W baffle former bolts)	Crack initiation and growth due to SCC and IASCC	PWR vessel internals; water chemistry	No
PWR	Reactor internals (B&W screws and bolts)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No
PWR	Reactor vessel closure studs and stud assembly	Loss of material due to wear	Reactor head closure studs	No
PWR	Reactor internals (Westinghouse upper and lower internal assemblies; CE bolts and tie rods)	Loss of preload due to stress relaxation	Inservice inspection; loose part monitoring	No

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System

Program	Description of Program	Implementation Schedule*
ISI	The program consists of periodic volumetric, surface, and/or visual examination of components and their supports for assessment, signs of degradation, and corrective actions. This program is in accordance with ASME Section XI, 1995 edition through the 1996 addenda.	Existing program
Water chemistry	To mitigate aging effects on component surfaces that are exposed to water as process fluid, chemistry programs are used to control water chemistry for impurities (e.g., chloride, fluoride, and sulfate) that accelerate corrosion. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits based on EPRI guidelines of TR-103515 for water chemistry in BWRs, TR-105714 for primary water chemistry in PWRs, and TR-102134 for secondary water chemistry in PWRs.	Existing program
One-time inspection	To verify the effectiveness of the water chemistry control program by determining if the aging effect is not occurring or the aging effect is progressing slowly so that the that the intended function will be maintained during the period of extended operation, a one-time inspection of small-bore piping less than NPS 4, including pipe, fittings, and branch connections, using suitable techniques at the most susceptible locations is performed. Actual inspection locations should be based on physical accessibility, exposure levels, and NDE techniques, and locations identified in NRC IN 97-46.	Inspection should be completed before the period of extended operation.
Bolting integrity	This program consists of guidelines on materials selection, strength and hardness properties, installation procedures, lubricants and sealants, corrosion considerations in the selection and installation of pressure-retaining bolting for nuclear applications, and enhanced inspection techniques. This program relies on the bolting integrity program delineated in NUREG-1339 and industry's recommendations delineated in EPRI NP-5769, with the exceptions noted in NUREG-1339 for safety-related bolting and in EPRI TR-104213 for pressure retaining bolting and structural bolting.	Existing program

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)

Program	Description of Program	Implementation Schedule*
Reactor vessel surveillance	Periodic testing of metallurgical surveillance samples is used to monitor the progress of neutron embrittlement of the reactor pressure vessel as a function of neutron fluence, in accordance with Regulatory Guide (RG) 1.99, Rev. 2.	The surveillance capsule withdrawal schedule should be revised before the period of extended operation.
Boric acid corrosion	The program consists of (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow-up inspection for adequacy. This program is implemented in response to GL 88-05.	Existing program
Thermal aging and neutron irradiation embrittlement of cast austenitic stainless steel	The program consists of (1) determination of the susceptibility of cast austenitic stainless steel components to thermal aging embrittlement, (2) accounting for the synergistic effects of thermal aging and neutron irradiation, and (3) implementing a supplemental examination program, as necessary.	Program should be implemented before the period of extended operation.
Reactor Head Closure Studs	This program includes inservice inspection ISI. For boiling water reactors (BWRs), this program also includes additional preventive actions and inspection techniques.	Existing program
Flow-accelerated corrosion	The program consists of the following: (1) conduct appropriate analysis and baseline inspection, (2) determine extent of thinning and replace/repair components, and (3) perform follow up inspections to confirm or quantify and take longer-term corrective actions. This program is in response to NRC GL 89-08.	Existing Program
Quality assurance	The 10 CFR Part 50, Appendix B program provides for corrective actions, confirmation process, and administrative controls for aging management programs for license renewal. The scope of this existing program will be expanded to include nonsafety-related structures and components that are subject to an AMR for license renewal.	Program should be implemented before the period of extended operation.
Vessel closure head penetration	The program assesses degradation of CRD mechanism nozzle and other vessel closure head penetrations, and consists of a review of the scope and schedule of inspection, including the leakage detection system, to assure detection of cracks before the loss of intended function of the penetrations. This is in response to NRC GL 97-01.	Existing program

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)

Program	Description of Program	Implementation Schedule*
BWR Control Rod Drive Return Line Nozzle	The AMP monitors the effects of cracking on the intended function of the component by detection and sizing of cracks by ISI in accordance with the NUREG-0619 and alternative recommendation of GE NE-523-A71-0594. NUREG-0619 specifies UT of the entire nozzle and penetration testing (PT) of varying portions of the blend radius and bore. GE NE-523-A71-0594 specifies UT of specific regions of the blend radius and bore. UT techniques and personnel qualification are according to the guidelines of GE NE-523-A71-0594.	Program should be implemented before the period of extended operation.
Steam generator tube integrity	This program consists of SG inspection scope, frequency, acceptance criteria for the plugging and repair of flawed tubes in accordance with the plant technical specifications that includes commitments to NEI 97-06.	Existing program
Loose part monitoring	The program consists of loose part monitoring of reactor vessel and primary coolant systems in accordance with ASME OM-S/G-1997 standards. The program addresses methods, intervals, parameters to be measured and evaluated, and records requirements.	Existing program
Neutron noise monitoring	The program consists of neutron noise monitoring for the detection of loss of axial preload at the core support barrel's upper support flange, and can detect physical displacement and motion of reactor internals in accordance with ASME OM-S/G-1997 standards. The program addresses methods, intervals, parameters to be measured and evaluated, acceptance criteria, and records requirements.	Existing program
BWR Vessel Internals	The program includes (a) inspection and flaw evaluation in conformance with the guidelines of applicable and staff-approved boiling water reactor vessel and internals project (BWRVIP) documents and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (EPRI TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components.	Existing program
Plant-specific AMP	The description should contain information associated with the basis for determining that aging effects will be managed during the period of extended operation.	Program should be implemented before the period of extended operation.

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)

Program	Description of Program	Implementation Schedule*
BWR Vessel ID Attachment Welds	The program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-48 and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (EPRI TR-103515).	
BWR Stress Corrosion Cracking	The program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) coolant pressure boundary piping made of stainless steel (SS) is delineated in NUREG-0313, Rev. 2, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 and its Supplement 1. The program includes (a) preventive measures to mitigate IGSCC and (b) inspections to monitor IGSCC and its effects.	Existing program
BWR Penetrations	The program includes (a) inspection and flaw evaluation in conformance with the guidelines of staff-approved boiling water reactor vessel and internals project (BWRVIP)-49 and BWRVIP-27 documents and (b) monitoring and control of reactor coolant water chemistry in accordance with the guidelines of BWRVIP-29 (EPRI TR-103515) to ensure the long-term integrity and safe operation of boiling water reactor (BWR) vessel internal components.	Existing program
Nickel-Alloy Nozzles and Penetrations	The program includes (a) primary water stress corrosion cracking (PWSCC) susceptibility assessment to identify susceptible components, (b) monitoring and control of reactor coolant water chemistry to mitigate PWSCC, and (c) inservice inspection ISI of reactor vessel head penetrations to monitor PWSCC and its effect on the intended function of the component. For susceptible penetrations and locations, the program includes an industry wide, integrated, long-term inspection program based on the industry responses to NRC Generic Letter (GL) 97-01.	Existing program
Thermal Aging of Cast Austenitic Stainless Steel	This program includes (a) determination of the susceptibility of cast austenitic stainless steel components to thermal aging embrittlement and (b) for potentially susceptible components aging management is accomplished through either enhanced volumetric examination or plant- or component-specific flaw tolerance evaluation.	Existing program

Table 3.1-2. FSAR Supplement for Aging Management of Reactor Vessel, Internals, and Reactor Coolant System (continued)

Program	Description of Program	Implementation Schedule*
PWR Vessel Internals	The program includes (a) augmentation of the inservice inspection (ISI) to include enhanced VT-1 examinations of non-bolted components, and other demonstrated acceptable methods for bolted components for certain susceptible or limiting components or locations, and (b) monitoring and control of reactor coolant water chemistry in accordance with the EPRI guidelines in TR-105714 to ensure the long-term integrity and safe operation of pressurized water reactor (PWR) vessel internal components.	Program should be implemented before the period of extended operation.
BWR Feedwater Nozzle	This program includes (a) enhancing inservice inspection (ISI) specified in the American Society of Mechanical Engineers (ASME) Code, Section XI, with the recommendation of General Electric (GE) NE-523-A71-0594 to perform periodic ultrasonic testing inspection of critical regions of the BWR feedwater nozzle.	Existing program

* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

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3.2 AGING MANAGEMENT OF ENGINEERED SAFETY FEATURES

Review Responsibilities

Primary - Branch responsible for materials and chemical engineering
Secondary - Branch responsible for mechanical engineering

3.2.1 Areas of Review

This review plan section addresses the aging management review (AMR) of the engineered safety features. For a recent vintage plant, the information related to the engineered safety features is contained in Chapter 6, "Engineered Safety Features," of the plant's FSAR, consistent with the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (NUREG-0800) (Ref. 1). The engineered safety features contained in this review plan section are generally consistent with those contained in NUREG-0800 except for the refueling water, control room habitability, and residual heat removal systems. For older plants, the location of applicable information is plant-specific because their FSAR may have predated NUREG-0800. The engineered safety features consist of containment spray, standby gas treatment (BWRs), containment isolation components, and emergency core cooling systems.

The staff has issued a GALL report addressing aging management for license renewal (Ref. 2). The GALL report documents the staff's basis for determining whether generic existing programs are adequate to manage aging without change, or generic existing programs should be augmented for license renewal. The GALL report may be referenced in a license renewal application, and should be treated in the same manner as an approved topical report.

Because a license renewal applicant may or may not be able to reference the GALL report as explained below, the following areas are reviewed:

3.2.1.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

The applicant may reference the GALL report in a license renewal application to demonstrate that the applicant's programs at its facility correspond to those reviewed and approved in the report, and that no further staff review is required. If the material presented in the GALL report is applicable to the applicant's facility, the staff should find the applicant's reference to the report acceptable. In making this determination, the staff should consider whether the applicant has identified specific programs described and evaluated in the GALL report. The staff, however, should not repeat its review of the substance of the matters described in the report. Rather, the staff should ensure that the applicant verifies that the approvals set forth in the GALL report for generic programs apply to the applicant's programs.

3.2.1.2 Further Evaluation of Aging Management as Recommended by the GALL Report

The GALL report provides the basis for identifying those programs that warrant further evaluation during the staff review of a license renewal application. The staff review should focus on augmented programs for license renewal.

3.2.1.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

The GALL report provides a generic staff evaluation of certain aging management programs. If the applicant does not rely on a particular program for license renewal, or if the applicant indicates that the generic staff evaluation of the elements of a particular program does not apply to its plant, the staff should review each such aging management program to which the GALL report does not apply.

The GALL report provides a generic staff evaluation of programs for certain components and aging effects. If the applicant has identified particular components subject to AMR for its plant that are not addressed in the GALL report, or particular aging effects for a component that are not addressed in the GALL report, the staff should review the applicant's aging management programs applicable to these particular components and aging effects.

3.2.1.4 FSAR Supplement

The FSAR supplement summarizing the programs and activities for managing the effects of aging for the period of extended operation is reviewed.

3.2.2 Acceptance Criteria

The acceptance criteria for the areas of review describe methods for determining whether the applicant has met the requirements of the NRC's regulations in 10 CFR 54.21.

3.2.2.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

Acceptable methods for managing aging of the engineered safety features are described and evaluated in Chapter V of the GALL report (Ref. 2). In referencing this report, the applicant should indicate that the material presented in the GALL report is applicable to the specific plant involved, and provide the information necessary to adopt the finding of program acceptability as described and evaluated in the report. The applicant should also verify that the approvals set forth in the GALL report for generic programs apply to the applicant's programs. The applicant may reference appropriate programs as described and evaluated in the GALL report.

3.2.2.2 Further Evaluation of Aging Management as Recommended by the GALL Report

The GALL report indicates that further evaluation should be performed for the following.

3.2.2.2.1 Cumulative Fatigue Damage

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The evaluation of this TLAA is addressed separately in Section 4.3 of this standard review plan.

3.2.2.2.2 Loss of Material due to General Corrosion

1. The management of loss of material due to general corrosion of pumps, valves, piping, and fittings associated with some of the BWR emergency core cooling systems [high pressure coolant injection, reactor core isolation cooling, high pressure core spray, low pressure core

spray, low pressure coolant injection (residual heat removal)] and with lines to the suppression chamber and to the drywell and suppression chamber spray system should be further evaluated. The existing aging management program relies on monitoring and control of primary water chemistry based on EPRI guidelines of TR-105714 for PWRs (Ref. 3) and BWRVIP 29 (EPRI TR-103515) for BWRs (Ref. 4) to mitigate degradation. However, control of primary water chemistry does not preclude loss of material due to general corrosion at locations of stagnant flow conditions. Therefore, verification of the effectiveness of the chemistry control program should be performed to ensure that corrosion is not occurring. The GALL report recommends further evaluation of programs to manage loss of material due to general corrosion to verify the effectiveness of the chemistry control program.). A one-time inspection of select components at susceptible locations is an acceptable method to determine whether an aging effect is not occurring or an aging effect is progressing very slowly such that the component's intended function will be maintained during the period of extended operation.

2. Loss of material due to general corrosion could occur in the containment spray (PWR) and drywell and suppression chamber spray (BWR) systems header and spray nozzle components, standby gas treatment system components (BWR), containment isolation valves and associated piping, the automatic depressurization system piping and fittings (BWR), emergency core cooling system header piping and fittings and spray nozzles (BWR), and the external surfaces of PWR and BWR carbon steel components. The GALL report recommends further evaluation on a plant specific basis to ensure that the aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RLSB-1 (Appendix A.1 of this standard review plan).

3.2.2.2.3 Local Loss of Material due to Pitting and Crevice Corrosion

1. The management of local loss of material due to pitting and crevice corrosion of pumps, valves, piping, and fittings associated with some of the BWR emergency core cooling system piping and fittings [high pressure coolant injection, reactor core isolation cooling, high pressure core spray, low pressure core spray, low pressure coolant injection (residual heat removal)] and with lines to the suppression chamber and to the drywell and suppression chamber spray system should be evaluated further. The existing aging management program relies on monitoring and control of primary water chemistry based on EPRI guidelines of TR-105714 for PWRs (Ref. 3) and BWRVIP 29 (EPRI TR-103515) for BWRs (Ref. 4) to mitigate degradation. However, control of coolant water chemistry does not preclude loss of material due to crevice and pitting corrosion at locations of stagnant flow conditions. Therefore, verification of the effectiveness of the chemistry control program should be performed to ensure that corrosion is not occurring. The GALL report recommends further evaluation of programs to manage the loss of material due to pitting and crevice corrosion to verify the effectiveness of the chemistry control program). A one-time inspection of select components at susceptible locations is an acceptable method to determine whether an aging effect is not occurring or an aging effect is progressing very slowly so that the component's intended function will be maintained during the period of extended operation.

2. Local loss of material from pitting and crevice corrosion could occur in the containment spray (PWR) components, containment isolation valves and associated piping, the buried portion of the refueling water tank external surface (PWRs), and automatic depressurization system piping and fittings (BWR). The GALL report recommends further evaluation to

ensure that the aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RSLB-1 (Appendix A.1 of this standard review plan).

3.2.2.2.4 Local Loss of Material due to Microbiologically Influenced Corrosion

Local loss of material due to microbiologically influenced corrosion (MIC) could occur in BWR and PWR containment isolation valves and associated piping in systems that are not addressed in other chapters of the GALL report. The GALL report recommends further evaluation to ensure that the aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RSLB-1 (Appendix A.1 of this standard review plan).

3.2.2.2.5 Changes in Properties due to Elastomer Degradation

Changes in properties due to elastomer degradation could occur in seals associated with the standby gas treatment system ductwork and filters. The GALL report recommends further evaluation to ensure that the aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RSLB-1 (Appendix A.1 of this standard review plan).

3.2.2.2.6 Local Loss of Material due to Erosion

Local loss of material due to erosion could occur in the high pressure safety injection pump miniflow orifice. This aging mechanism and effect will apply only to pumps that are normally used as charging pumps in the chemical and volume control systems (PWRs). The GALL report recommends further evaluation to ensure that the aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RSLB-1 (Appendix A.1 of this standard review plan).

3.2.2.2.7 Buildup of Deposits due to Corrosion

The plugging of components due to general corrosion could occur in the spray nozzles and flow orifices of the drywell and suppression chamber spray system. This aging mechanism and effect will apply since the spray nozzles and flow orifices are occasionally wetted, even though the majority of the time this system is on standby. The wetting and drying of these components can aid in the acceleration of this particular corrosion. The GALL report recommends further evaluation to ensure that the aging effect is adequately managed. Acceptance criteria are described in Branch Technical Position RSLB-1 (Appendix A.1 of this standard review plan).

3.2.2.2.8 Quality Assurance for Aging Management of Nonsafety-Related Components

Acceptance criteria are described in Branch Technical Position IQMB-1 (Appendix A.2 of this standard review plan.)

3.2.2.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Acceptance criteria are described in Branch Technical Position RSLB-1 (Appendix A.1 of this standard review plan).

3.2.2.4 FSAR Supplement

The summary description of the programs and activities for managing the effects of aging for the period of extended operation in the FSAR supplement should be appropriate such that later changes can be controlled by 10 CFR 50.59. The description should contain information associated with the bases for determining that aging effects will be managed during the period of extended operation.

3.2.3 Review Procedures

For each area of review, the following review procedures are to be followed.

3.2.3.1 Aging Management Programs Evaluated in the GALL Report that Are Relied on for License Renewal

The applicant may reference the GALL report in its license renewal application, as appropriate. The staff should not repeat its review of the substance of the matters described in the report. If the applicant has provided the information necessary to adopt the finding of program acceptability as described and evaluated in the GALL report, the staff should find the applicant's reference to the report in a license renewal application acceptable. In making this determination, the reviewer verifies that the applicant has provided a brief description of the system, components, materials, and environment. The reviewer also verifies that the applicant has stated that the applicable aging effects and industry and plant-specific operating experience have been reviewed by the applicant and are evaluated in the GALL report. The reviewer verifies that the applicant has identified those aging effects for the engineered safety features components that are contained in the report as applicable to its plant. In addition, the reviewer ensures that the applicant has stated that the plant programs covered by the applicant's reference contain the same program elements that the staff evaluated and relied upon in approving the corresponding generic program in the GALL report.

The reviewer should verify that the applicant has stated that certain of its aging management programs contain the same program elements as the corresponding generic program described in the GALL report, and upon which the staff relied in its evaluation. The reviewer should also verify that the applicant has stated that the GALL report is applicable to its plant with respect to these programs. The reviewer verifies that the applicant has identified the appropriate programs as described and evaluated in the GALL report. Programs evaluated in the report regarding the engineered safety features components are summarized in Table 3.2-1 of this review plan section. No further staff evaluation is necessary if so recommended in the GALL report.

3.2.3.2 Further Evaluation of Aging Management as Recommended by the GALL Report

3.2.3.2.1 Cumulative Fatigue Damage

Fatigue is a TLAA as defined in 10 CFR 54.3. TLAAs are required to be evaluated in accordance with 10 CFR 54.21(c). The staff reviews the evaluation of this TLAA separately, following the guidance in Section 4.3 of this standard review plan.

3.2.3.2.2 Loss of Material due to General Corrosion

1. The GALL report recommends further evaluation of programs to manage the loss of material due to general corrosion of piping and fittings associated with some of the BWR emergency

core cooling systems [high pressure coolant injection, reactor core isolation cooling, high pressure core spray, low pressure core spray, low pressure coolant injection (residual heat removal)] and with lines to the suppression chamber and to the drywell and suppression chamber spray system to verify the effectiveness of the chemistry control program. A one-time inspection of select components at susceptible locations is an acceptable method to determine whether an aging effect is not occurring or an aging effect is progressing very slowly such that the component's intended function will be maintained during the period of extended operation.

The reviewer reviews the applicant's proposed program to determine whether corrosion is not occurring or the corrosion is progressing very slowly so that the component's intended function will be maintained during the period of extended operation. If an applicant proposes a one-time inspection of select components at susceptible locations to ensure that corrosion is not occurring, the reviewer verifies that the applicant's selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. The inspection techniques may include visual, ultrasonic, and surface examination techniques. Follow-up actions are to be based on the inspection results.

2. The GALL report recommends further evaluation of programs to manage the loss of material due to general corrosion of containment spray (PWR) and drywell and suppression chamber spray (BWR) systems header and spray nozzle components, standby gas treatment system components (BWR), containment isolation valves and associated piping, the automatic depressurization system piping and fittings (BWR), emergency core cooling system header piping and fittings and spray nozzles (BWR), and the external surfaces of PWR and BWR carbon steel components. The reviewer reviews the applicant's proposed programs on a case-by-case basis to ensure that an adequate program will be in place for the management of general corrosion of these components.

3.2.3.2.3 Local Loss of Material due to Pitting and Crevice Corrosion

1. The GALL report recommends further evaluation of programs to manage the loss of material due to pitting and crevice corrosion of piping and fittings associated with some of the BWR emergency core cooling system piping and fittings [high pressure coolant injection, reactor core isolation cooling, high pressure core spray, low pressure core spray, low pressure coolant injection (residual heat removal)] and with lines to the suppression chamber and to the drywell and suppression chamber spray system to verify the effectiveness of the chemistry control program. A one-time inspection of select components at susceptible locations is an acceptable method to determine whether an aging effect is not occurring or an aging effect is progressing very slowly such that the component's intended function will be maintained during the period of extended operation.

   The reviewer reviews the applicant's proposed program to determine whether corrosion is not occurring or the corrosion is progressing very slowly so that the component's intended function will be maintained during the period of extended operation. If an applicant proposes a one-time inspection of select components at susceptible locations to ensure that corrosion is not occurring, the reviewer verifies that the applicant's selection of susceptible locations is based on severity of conditions, time of service, and lowest design margin. The inspection techniques may include visual, ultrasonic, and surface examination techniques. Follow-up actions are to be based on the inspection results.

2. The GALL report recommends further evaluation of programs to manage the local loss of material due to pitting and crevice corrosion of containment spray (PWR) components, containment isolation valves and associated piping, the outer buried surface of the refueling water tank (PWR), and the automatic depressurization system piping and fittings (BWR). The reviewer reviews the applicant's proposed programs on a case-by-case basis to ensure that an adequate program will be in place for the management of local loss of material due to pitting and crevice corrosion of these components.

3.2.3.2.4 Local Loss of Material due to Microbiologically Influenced Corrosion

The GALL report recommends further evaluation of programs to manage the local loss of material due to MIC of the BWR and PWR containment isolation valves and associated piping. The reviewer reviews the applicant's proposed programs on a case-by-case basis to ensure that an adequate program will be in place for the management of local loss of material due to MIC of the BWR and PWR containment isolation barriers.

3.2.3.2.5 Changes in Properties due to Elastomer Degradation

The GALL report recommends further evaluation of programs to manage changes in properties due to degradation of elastomer seals associated with BWR standby gas treatment system ductwork and filters. The reviewer reviews the applicant's proposed programs on a case-by-case basis to ensure that an adequate program will be in place to manage changes in properties due to degradation of elastomer seals in the standby gas treatment system.

3.2.3.2.6 Local loss of Material due to Erosion

The GALL report recommends further evaluation of programs to manage local loss of material due to erosion of the high pressure safety injection pump miniflow orifice. The reviewer reviews the applicant's proposed programs on a case-by-case basis to ensure that an adequate program will be in place to manage this aging effect.

3.2.3.2.7 Buildup of Deposits due to Corrosion

The GALL report recommends further evaluation of programs to manage the plugging of spray nozzles and spargers of the drywell and suppression chamber spray system. The reviewer reviews the applicant's proposed programs on a case-by-case basis to ensure that an adequate program will be in place to manage this aging effect.

3.2.3.2.8 Quality Assurance for Aging Management of Nonsafety-Related Components

The applicant's aging management programs for license renewal should contain the elements of corrective actions, the confirmation process, and administrative controls. Safety-related components are covered by 10 CFR Part 50 Appendix B, which is adequate to address these program elements. However, Appendix B does not apply to nonsafety-related components that are subject to an AMR for license renewal. Nevertheless, the applicant has the option to expand the scope of its 10 CFR Part 50 Appendix B program to include these components and address the associated program elements. If the applicant chooses this option, the reviewer verifies that the applicant has documented such a commitment in the FSAR supplement. If the applicant chooses alternative means, the branch responsible for quality assurance should be requested to review the applicant's proposal on a case-by-case basis.

3.2.3.3 Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report

Review procedures are described in Branch Technical Position RSLB-1 (Appendix A.1 of this standard review plan).

3.2.3.4 FSAR Supplement

The reviewer verifies that the applicant has provided information, equivalent to that in Table 3.2-2, in the FSAR supplement for aging management of the engineered safety features for license renewal. The reviewer also verifies that the applicant has provided information, equivalent to that in Table 3.2-2, in the FSAR supplement for Subsection 3.2.3.3, "Aging Management Evaluations that Are Different from or Not Addressed in the GALL Report."

The staff expects to impose a license condition on any renewed license to require the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 50.71(e)(4). As part of the license condition, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59.

As noted in Table 3.2-2, an applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.

3.2.4 Evaluation Findings

The reviewer verifies that the applicant has provided information sufficient to satisfy the provisions of this review plan section, and the staff's evaluation supports conclusions of the following type, to be included in the staff's safety evaluation report:

> The staff concludes that the applicant has demonstrated that the aging effects associated with the engineered safety features will be adequately managed so that there is reasonable assurance that these systems will perform their intended functions in accordance with the current licensing basis during the period of extended operation. The staff also concludes that the FSAR supplement contains an appropriate summary description of the programs and activities for managing the effects of aging for the engineered safety features as reflected in the license conditions.

3.2.5 Implementation

Except in those cases in which the applicant proposes an acceptable alternative method for complying with specified portions of the NRC's regulations, the method described herein will be used by the staff in its evaluation of conformance with NRC regulations.

3.2.6 References

1. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, July 1981.

2. NUREG-1801, "Generic Aging Lessons Learned (GALL)," U.S. Nuclear Regulatory Commission, July 2001.

3. EPRI TR-105714, PWR primary Water Chemistry Guidelines-Revision 3, Electric Power Research Institute, Palo Alto, CA, Nov. 1995.

4. EPRI TR-103515, BWR Water Chemistry Guidelines-Revision 1, Normal and Hydrogen Water Chemistry, Electric Power Research Institute, Palo Alto, CA, February 1994.

Table 3.2-1. Summary of Aging Management Programs for Engineered Safety Features Evaluated in Chapter V of the GALL Report

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR/PWR	Piping, fittings, and valves in emergency core cooling system	Cumulative fatigue damage	TLAA, evaluated in accordance with 10 CFR 54.21(c)	Yes, TLAA (see Subsection 3.2.2.2.1)
BWR	Piping, fittings, pumps, and valves in emergency core cooling system	Loss of material due to general corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated (see Subsection 3.2.2.2.2.1)
BWR/PWR	Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to general corrosion	Plant specific	Yes, plant specific (see Subsection 3.2.2.2.2.2)
BWR	Piping, fittings, pumps, and valves in emergency core cooling system	Loss of material due to pitting and crevice corrosion	Water chemistry and one-time inspection	Yes, detection of aging effects is to be further evaluated (see Subsection 3.2.2.2.3.1)
BWR/PWR	Components in containment spray (PWR only), standby gas treatment (BWR only), containment isolation, and emergency core cooling systems	Loss of material due to pitting and crevice corrosion	Plant specific	Yes, plant specific (see Subsection 3.2.2.2.3.2)
BWR/PWR	Containment isolation valves and associated piping	Loss of material due to microbiologically influenced corrosion	Plant specific	Yes, plant specific (see Subsection 3.2.2.2.4)
BWR	Seals in standby gas treatment system	Changes in properties due to elastomer degradation	Plant specific	Yes, plant specific (see Subsection 3.2.2.2.5)
PWR	High pressure safety injection (charging) pump miniflow orifice	Loss of material due to erosion	Plant specific	Yes, plant specific (see Subsection 3.2.2.2.6)

Table 3.2-1. Summary of Aging Management Programs for Engineered Safety Features Evaluated in Chapter V of the GALL Report (continued)

Type	Component	Aging Effect/ Mechanism	Aging Management Programs	Further Evaluation Recommended
BWR	Drywell and suppression chamber spray system nozzles and flow orifices	Plugging of nozzles and flow orifices due to general corrosion	Plant specific	Yes, plant specific (see Subsection 3.2.2.2.7)
BWR/PWR	Piping and fittings of CASS in emergency core cooling system	Loss of fracture toughness due to thermal aging embrittlement	Thermal aging embrittlement of CASS	No
BWR/PWR	Components serviced by open-cycle cooling system	Local loss of material due to corrosion and/or buildup of deposit due to biofouling	Open-cycle cooling water system	No
BWR/PWR	Components serviced by closed-cycle cooling system	Loss of material due to general, pitting, and crevice corrosion	Closed-cycle cooling water system	No
BWR	Emergency core cooling system valves and lines to and from HPCI and RCIC pump turbines	Wall thinning due to flow-accelerated corrosion	Flow-accelerated corrosion	No
PWR	Pumps, valves, piping, and fittings in containment spray and emergency core cooling systems	Crack initiation and growth due to SCC	Water chemistry	No
BWR	Pumps, valves, piping, and fittings in emergency core cooling systems	Crack initiation and growth due to SCC and IGSCC	Water chemistry and BWR stress corrosion cracking	No
PWR	Carbon steel components	Loss of material due to boric acid corrosion	Boric acid corrosion	No
BWR/PWR	Closure bolting in high pressure or high temperature systems	Loss of material due to general corrosion, loss of preload due to stress relaxation, and crack initiation and growth due to cyclic loading or SCC	Bolting integrity	No

Table 3.2-2. FSAR Supplement for Aging Management of Engineered Safety Features

Program	Description of Program	Implementation Schedule*
Bolting integrity (BWR/PWR)	This program includes periodic inspection of closure bolting for indication of potential problems including loss of reload, cracking, and loss of material. This program consists of guidelines on materials selection, strength and hardness properties, installation procedures, lubricants and sealants, corrosion considerations in the selection and installation of pressure-retaining bolting for nuclear applications, and enhanced inspection techniques. This program relies on the bolting integrity program delineated in NUREG-1339 and industry's recommendations delineated in EPRI NP-5769, with the exceptions noted in NUREG-1339 for safety-related bolting, and EPRI TR-104213 for pressure retaining bolting and structural bolting.	Existing program
Boric acid corrosion (PWR)	The program consists of (1) visual inspection of external surfaces that are potentially exposed to borated water leakage, (2) timely discovery of leak path and removal of the boric acid residues, (3) assessment of the damage, and (4) follow up inspection for adequacy. This program is implemented in response to GL 88-05.	Existing program
Closed-cycle cooling water system (BWR/PWR)	The program relies on preventive measures to minimize corrosion by maintaining inhibitors and by performing non-chemistry monitoring consisting of inspection and nondestructive evaluations based on the guidelines of EPRI-TR-107396 for closed–cycle cooling water systems.	Existing program
Flow-accelerated corrosion (FAC) (BWR/PWR)	The program consists of (1) conduct appropriate analysis and baseline inspection, (2) determine extent of thinning, and replace/repair components, and (3) perform follow-up inspections to confirm or quantify and take longer-term corrective actions. The program relies on implementation of EPRI guidelines of NSAC-202L-R2.	Existing program
One-time inspection	To verify the effectiveness of the water chemistry control program by determining if the aging effect is not occurring or the aging effect is progressing so slowly that the intended function will be maintained during the period of extended operation, a one-time inspection of pumps, valves, piping, and fittings associated with certain BWR emergency core cooling systems [high pressure coolant injection, reactor core isolation cooling, high pressure core spray, low pressure core spray, low pressure coolant injection (residual heat removal)]; and with pipe lines in a BWR plant to the suppression chamber and to the drywell and suppression chamber spray system is performed.	The inspection should be completed before the period of extended operation

Table 3.2-2. FSAR Supplement for Aging Management of Engineered Safety Features (continued)

Program	Description of Program	Implementation Schedule*
Open-cycle cooling water system (BWR/PWR)	The program includes (a) surveillance and control of biofouling, (b) tests to verify heat transfer, (c) routine inspection and maintenance program, (d) system walk down inspection, and (e) review of maintenance, operating, and training practices and procedures. The program provides assurance that the open-cycle cooling water system is in compliance with General Design Criteria and Quality Assurance to ensure that the open-cycle cooling water (or service water) system can be managed for an extended period of operation. This program is in response to NRC GL 89-13.	Existing program
Plant-specific AMP	The description should contain information associated with the basis for determining that aging effects will be managed during the period of extended operation.	Program should be implemented before the period of extended operation
Quality assurance	The 10 CFR Part 50 Appendix B program provides for corrective actions, the confirmation process, and administrative controls for aging management programs for license renewal. The scope of this existing program will be expanded to include nonsafety-related structures and components that are subject to an AMR for license renewal.	Program should be implemented before the period of extended operation
Thermal aging embrittlement of CASS AMP (BWR/PWR)	The program consists of the determination of the susceptibility of CASS piping and fittings in PWR ECCS systems including interfacing pipe lines to the chemical and volume control system and to the spent fuel pool; and in BWR ECCS systems including interfacing pipe lines to the suppression chamber and to the drywell and suppression chamber spray system in regard to thermal aging embrittlement based on the casting method, Mo content, and ferrite percentage. For potentially susceptible piping, aging management is accomplished either through enhanced volumetric examination or component-specific flaw tolerance evaluation.	Existing program
Water chemistry (BWR/PWR)	To mitigate aging effects on component surfaces that are exposed to water as a process fluid, chemistry programs are used to control water impurities (e.g., chloride, fluoride, sulfate) that accelerate corrosion. This program relies on monitoring and control of water chemistry to keep peak levels of various contaminants below the system-specific limits based on EPRI guidelines of TR-103515 for water chemistry in BWRs, and TR-105714 for primary water chemistry in PWRs.	Existing program

Table 3.2-2. FSAR Supplement for Aging Management of Engineered Safety Features (continued)

Program	Description of Program	Implementation Schedule*
BWR Stress Corrosion Cracking	The program to manage intergranular stress corrosion cracking (IGSCC) in boiling water reactor (BWR) coolant pressure boundary piping made of stainless steel (SS) is delineated, in part, in NUREG-0313, Rev. 2, and Nuclear Regulatory Commission (NRC) Generic Letter (GL) 88-01 and its Supplement 1. The program includes (a) preventive measures to mitigate IGSCC and (b) inspections to monitor IGSCC and its effects	Existing Program
* An applicant need not incorporate the implementation schedule into its FSAR. However, the reviewer should verify that the applicant has identified and committed in the license renewal application to any future aging management activities to be completed before the period of extended operation. The staff expects to impose a license condition on any renewed license to ensure that the applicant will complete these activities no later than the committed date.		